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SPECIAL INSTRUCTION SHEET

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Civilian Radioactive Waste Management System
Management & Operating Contractor

Evaluation of Codisposal Viability for U-Zr/U-Mo Alloy (Enrico Fermi) DOE-Owned Fuel

TDR-EDC-NU-000002 REV 00

August 2000

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EXECUTIVE SUMMARY

INTRODUCTION

This report provides the technical information that supports the disposal, in the potential Monitored Geologic Repository (MGR) at Yucca Mountain, of the spent nuclear fuel (SNF) owned by the U.S. Department of Energy (DOE) that was removed from the Enrico Fermi Atomic Power Plant (Fermi 1). The Fermi uranium-molybdenum (U-Mo) alloy SNF is one of more than 250 forms of DOE-owned SNF. Due to the variety of the SNF, the National Spent Nuclear Fuel Program has designated nine representative fuel groups for disposal criticality analyses based on the fuel matrix composition, primary fissile isotope, and enrichment. The Fermi SNF has zirconium (Zr) cladding over a U-Mo alloy and is representative of the U-Zr and U-Mo highly enriched uranium (HEU) fuel group. Demonstration that other fuels in this group are bounded by the analysis of Enrico Fermi fuel remains to be performed in the future before acceptance of these fuel forms.

The results compiled in this report will be used to develop waste acceptance criteria. The parameters and conditions that are important to criticality control are identified herein based on the analysis needs and sensitivities of the results. Prior to acceptance of fuel from the Enrico Fermi fuel group for disposal, the criticality control items must be demonstrated to satisfy the conditions determined in this report.

The intact and degraded component criticality analyses have been performed following the disposal criticality analysis methodology, which has been documented in *Disposal Criticality Analysis Methodology Topical Report* (YMP 1998) and submitted to the U.S. Nuclear Regulatory Commission. The methodology includes analyzing the geochemical and physical processes that can breach the waste package and degrade the waste forms and other internal components. One or more addenda to the topical report will be required to establish the critical limit for DOE SNF once sufficient critical benchmarks are identified and performed.

The waste package that holds the DOE SNF canister containing the Fermi U-Mo Alloy SNF also contains five high-level radioactive waste (HLW) glass pour canisters and a carbon steel basket assembly. The HLW canisters with immobilized plutonium are not considered in the present analyses. The SNF canister is placed in a carbon-steel support basket that holds the canister in the center of the codisposal waste package (see Figure ES-1). The SRS (Savannah River Site) HLW glass canister, which comes from the Savannah River Plant (SRP) Defense Waste Processing Facility, is a cylindrical stainless steel shell with an outer diameter of approximately 610 mm (24 in.). The five HLW canisters are evenly spaced around the DOE SNF canister. The DOE SNF canister has an 18-in. outside diameter and accommodates a stack of two baskets. Twelve 4-in.-outside-diameter stainless steel (316L) pipes are welded to the base plate of the basket. A spacer at the top of the top basket restrains the pipes in place. A shipping canister made of aluminum and referred to as the -01 canister is placed in each pipe. Each -01 canister contains an aluminum canister, referred to as the -04 canister. The -04 canister contains 140 pins of the Enrico Fermi fast reactor SNF. The zirconium-clad fuel pins have been removed from the fuel assembly structure (derodded). The available space in and around the steel pipes is filled with iron (Fe) shot. The iron shot contains 3% by volume gadolinium phosphate ($GdPO_4$), i.e., 14.5 kg of $GdPO_4$ per 737.9 kg iron. Note that the space inside each pipe, but outside the -01 canister, must be filled with the same iron shot mixture.

The results in this report are based on the waste package design developed for the viability assessment. The codisposal waste packages are fabricated from materials like those that are used for the commercial SNF waste packages. The outer barrier is made of corrosion-allowance material, 100-mm-thick carbon steel. The corrosion-resistant inner barrier consists of a 20-mm-thick high-nickel alloy. The top and bottom lids are also based on the two-barrier principle and use the same materials.

This report presents the results of analyzing the 5-DHLW/DOE SNF waste package for structural, thermal, shielding, and intact- and degraded-mode criticality performance compared to the respective design criteria. Section 2.2 provides the criteria, and Section 2.3 provides the key assumptions for the various analyses.

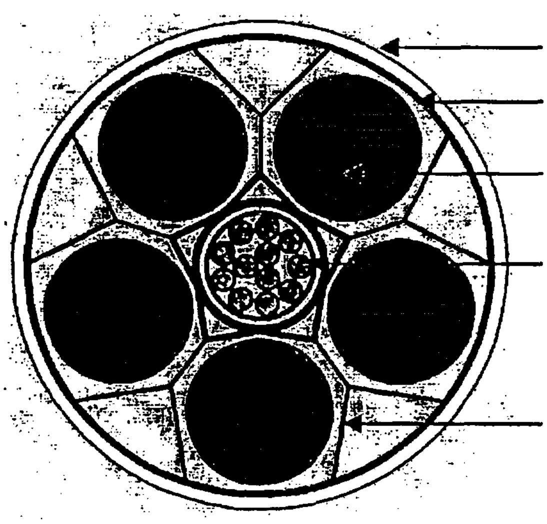

Figure ES-1. 5-DHLW/DOE Spent Nuclear Fuel Waste Package

STRUCTURAL ANALYSES

ANSYS Version 5.4–a finite-element analysis computer code–was used to perform the structural analysis of the 5-DHLW/DOE SNF (Fermi) waste package. A two-dimensional finite-element representation of the 5-DHLW/DOE SNF waste package was developed to determine the effects of loads on the container's structural components due to a tipover design-basis event (DBE) (CRWMS M&O 1997a). Calculations of maximum potential energy for each handling accident scenario (horizontal drop, vertical drop, and tipover DBEs) show that the bounding dynamic load is obtained from a tipover case in which the rotating end of the waste package experiences the highest impact load. Therefore, tipover structural evaluations are bounding for all handling accident scenarios under the constraints that the MGR surface will be designed to prevent events that exceed a 2.4-m horizontal drop and/or a 2.0-m vertical drop.

The structural response of the waste package to tipover accident loads is reported using the maximum stress values and displacements obtained from the finite-element solution to the problem. The results show that the maximum clearance gap closure inside the 4-in.-diameter pipe is 6.97 mm, whereas the available clearance between a 4-in.-diameter pipe and an Aluminum -01 canister shell is 9.49 mm. Hence, there will be no interference between the two components from a tipover DBE. The maximum stress in the Fermi 18-in.-diameter DOE-SNF canister structural components, including internals, is determined to be 265 MPa; this stress is less than the 483 MPa tensile strength of 316L SS.

THERMAL ANALYSES

The finite-element analysis computer code used for the thermal analysis of the 5-DHLW/DOE SNF (Fermi) waste package was ANSYS Version 5.4. The axial cross section at the center of the canister is represented in two-dimensional calculations. The average heat output of the Enrico Fermi fuel is assumed constant at 1.1 watts per -04 canister during the entire period of emplacement. Since the axial power peaking factor for the Fermi fuel is not known, a value of 1.25 was used for the axial power peaking factor in the analysis. Two cases for heat outputs are considered. The first one is the "nominal" case with the actual heat outputs. A more conservative case was obtained by applying a multiplier of 5.8 to the SRS glass canister heat output. The bounding case gave the maximum expected temperature within the SRS glass and is intended to provide conservative, bounding results. The thermal conductivity of the HLW glass is approximated as that of pure borosilicate glass, and the properties of density and specific heat of the HLW glass are approximated as those of Pyrex glass.

The thermal analyses show that the Fermi waste package satisfies applicable criteria under normal condition with the exception of fuel cladding temperature pending on TBD-179 (accident fire condition is not within the scope of this analysis). For the bounding case, the peak fuel temperature occurs in the DOE SNF canister and is 327.2°C, which is below the design criterion of 350°C. The result for the bounding case was a maximum temperature of 394.5 °C in the HLW, which is below the 400 °C criterion in the *Defense High Level Waste Disposal Container System Description Document* (CRWMS M&O 1999o) (see Section 2.2.2).

SHIELDING ANALYSES

The Monte Carlo radiation transport code MCNP, Version 4B2, was used to calculate average dose rates on the surfaces of the waste package. The maximum surface-dose rate occurred on the waste package outer-radial surface at the middle segment of the HLW glass canister; it was 10.95 rem/h. The maximum dose rates on the bottom and top surfaces of the waste package are about one-third and one-tenth, respectively, of the maximum dose rate on the radial surface.

While the gamma spectra of the SRS HLW glass and the Fermi SNF are similar, the total gamma intensity for each HLW glass canister is about nine times higher than that of the Fermi DOE SNF canister. Because of its low radiation intensity, its central position in the waste package, and its high-density material, the Fermi DOE SNF canister makes a very small contribution to the waste package surface dose rates. The primary gamma dose rate was approximately 2 orders of magnitude greater than the neutron dose rate.

The maximum dose rate on the outer surfaces of the waste package design described herein is 10.95 rem/h. The design criterion specifies the maximum dose rate at all external surfaces of the waste package is TBD (see Section 2.2.3).

DEGRADATION AND GEOCHEMISTRY ANALYSES

The degradation analyses follow the general methodology developed for application to all waste forms containing fissile material. This methodology evaluates potential critical configurations from the intact waste package (geometrically intact components in a breached waste package assumed to be flooded with water as a moderator) through the completely degraded waste package. The waste package design is used as the starting point for the intact configuration. Sequences of events and/or processes of component degradation are developed. Standard scenarios from the master scenario list in the *Disposal Criticality Analysis Methodology Topical Report* (YMP 1998) are refined using unique fuel characteristics. Potentially critical configurations are identified for further analysis.

The geochemistry analyses are performed using the EQ3/6 Version 7.2B geochemistry code in the solid-centered flow-through mode. A principal objective of these analyses is to assess the chemical circumstances that could lead to removal of neutron absorber material (Gd) from the waste package, while fissile materials remain behind. Such circumstances could increase the probability of a nuclear criticality occurrence within the waste package. Gadolinium is initially present as $GdPO_4$ that is combined with iron shot to produce Gd-doped iron shot, which is distributed in the space between the 4-in.-diameter stainless steel pipes and in the space between the -01 canister of Fermi fuel pins and the 4-in.-diameter pipes. Water with the composition of J-13 well water is assumed to drip in through an opening at the top of the waste package, pooling inside and eventually overflowing, allowing soluble components to be removed through continual dilution. Eighteen EQ3/6 cases have been selected and examined to identify the reasons for the chemical composition changes during the degradation of waste package materials and the flushing by J-13 well water. The results show that, even in unusual conditions, loss of Gd was insignificant, when it is present in the package as solid $GdPO_4$. The scenarios and conditions of EQ3/6 calculations are chosen to emphasize environments that could create either acid or alkaline conditions and to determine if these conditions exist long enough to cause significant Gd loss. Even with the extreme environments chosen, the differences in the results in terms of Gd loss are small in all instances.

INTACT- AND DEGRADED-MODE CRITICALITY ANALYSIS

The criticality calculations are performed using MCNP code, Version 4B2, for the internal configurations that can be created during degradation of the codisposal waste package containing Enrico Fermi SNF. The analyzed configurations covered the whole spectrum of potential critical configurations, starting with configurations of the intact flooded waste package and ending with the full degradation of the DOE SNF canister along with the HLW canisters and the waste package internal components. All analyses conservatively consider optimum conditions (within the physical limitations of the degradation scenario) for moderation, for the distribution of fissile material and degradation products, and for neutron reflection to determine the highest k_{eff} (effective neutron multiplication factor) attainable by degraded configurations.

The results from the intact and degraded component criticality calculations show that a $k_{eff}+2\sigma$ (95% confidence) of less than or equal to 0.93 is achievable for all credible configurations, if 3% by volume of gadolinium phosphate ($GdPO_4$) (or 14.5 kg per 737.9 kg of iron) is included in the iron shot that fills the available space in and around the steel pipes but outside the -01 canister.

CONCLUSIONS

In summary, the structural and thermal design criteria (with the exception of fuel cladding temperature pending on TBD-179) are met for a fully loaded DOE SNF canister containing Enrico Fermi SNF. Meeting the shielding design criteria is pending on TBD-3764. Each waste package can contain one DOE SNF canister containing up to 3360 fuel pins. An intact waste package has a k_{eff} below the critical limit of 0.93, without the addition of neutron absorbers. However, for degraded waste package configurations, a neutron absorber in the form of $GdPO_4$ in or on iron shot must be added in space between and within the 4-in.-diameter steel pipes holding the -01 canister of fuel pins inside the DOE SNF canister. With this design, there will be approximately eight DOE SNF canisters loaded with Fermi SNF, which correspond to eight codisposal waste packages.

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ACRONYMS AND ABBREVIATIONS

ASME	American Society of Mechanical Engineers
BPVC	Boiler and Pressure Vessel Code
CSCI	computer software configuration item
DBE	design basis event
DHLW	defense high-level waste
DOE	U.S. Department of Energy
DTN	data tracking number
FEA	finite-element analysis
F.M.	fissile material
HEU	highly enriched uranium
HLW	high-level radioactive waste
k_{eff}	effective neutron multiplication factor
LCE	laboratory critical experiment
MGR	Monitored Geologic Repository
PC	personal computer
PPF	power peaking factor
QARD	*Quality Assurance Requirements and Description*
SCM	Software Configuration Management
SDD	System Description Document
SNF	spent nuclear fuel
SQR	software qualification report
SRP	Savannah River Plant
SRS	Savannah River Site
SS	stainless steel
TBD	to be determined
TBV	to be verified
W.F.	waste form
WP/W.P.	waste package
2-D	two-dimensional
3-D	three-dimensional

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1. INTRODUCTION AND BACKGROUND

The Enrico Fermi Atomic Power Plant, Fermi 1, as it later became known, was built and operated to produce electricity for a group of 34 companies known as the Power Reactor Development Company. Fermi 1 used uranium as its nuclear fuel, but at the same time, this plant produced a supply of plutonium, which also could be used to fuel nuclear reactors to generate electricity. Fermi 1 was a fast-breeder nuclear reactor that used liquid sodium as its heat transfer agent, and despite its experimental nature, it was the only fast breeder reactor of commercial size in the world at that time.

Fermi 1 started operating August 23, 1963, almost 10 years after its groundbreaking. On October 5, 1966, a metal plate broke away from the reactor internals and blocked the flow of liquid sodium through a portion of the core, causing the fuel to overheat and begin melting. Damage to the reactor and fuel assemblies took about four years to repair, after which the plant operated intermittently for only a few months. The decision to decommission the plant was made on November 27, 1972.

The different aspects of the analyses reported in this document have been performed by following the methodology for disposal criticality analysis as documented in the topical report submitted to the U.S. Nuclear Regulatory Commission (YMP 1998). The methodology includes analyzing the geochemical and physical processes that degrade the waste package internals and the waste forms, after the waste package is breached. Addenda to the topical report will be required to establish the critical limit for the U.S. Department of Energy (DOE) spent nuclear fuel (SNF) once sufficient critical benchmark analyses are located and performed. In this report, a conservative and simplified bounding approach is employed to designate an interim critical limit.

The analyses performed also include structural, thermal, shielding, and intact- and degraded-mode criticality analyses. This report is in support of the work outlined in Civilian Radioactive Waste Management System (CRWMS) Management & Operating Contractor (M&O) *DOE SNF Analysis Plan for FY2000* (CRWMS 2000a).

In this technical report, there are numerous references to "codisposal container" and "waste package." Since the use of these two terms may be confusing, a definition of the terms is included here:

"(Co)disposal container" means the container barriers or shells, spacing structures and baskets, shielding integral to the container, packing contained within the container, and other absorbent materials designed to be placed internal to the container or immediately surrounding the disposal container (i.e., attached to the outer surface of the disposal container). The disposal container is designed to contain SNF and high-level radioactive waste (HLW), but exists only until the outer weld is complete and accepted. The disposal container does not include the waste form or the encasing containers or canisters (e.g., HLW pour canisters, DOE SNF codisposal canisters, multi-purpose canisters of SNF, etc.).

"Waste package" means the waste form and any containers (i.e., disposal container barriers and other canisters), spacing structures or baskets, shielding integral to the container, packing contained within the container, and other neutron absorber materials immediately surrounding and individual waste container placed internally to the container or attached to the outer surface of the disposal container. The waste package begins its existence when the outer lid weld of the disposal container is complete and accepted.

1.1 OBJECTIVE

The objective of this report is to provide sufficient detail to establish the technical viability for disposing of Fermi SNF in the potential Monitored Geologic Repository (MGR). This report sets limits and establishes values that, if and when met by a specific fuel type under the U-Zr and U-Mo (HEU) group, will be bounded by the results reported in this technical report.

Section 2, Design Information, describes the design of the codisposal container (Section 2.1, Design Parameters), with both requirements and assumptions identified (Section 2.2, Design Criteria, and Section 2.3, Assumptions). Analytical results to demonstrate the adequacy of the design and evaluate the feasibility of codisposing the Fermi U-Mo Alloy SNF in the MGR are presented in the following sections: 3, Structural Analysis; 4, Thermal Analysis; 5, Shielding Analysis; 6, Degradation and Geochemistry Analysis; and 7, Intact and Degraded Criticality Analysis. Section 8, Conclusions, provides the connections between the design criteria and analytical results to establish technical viability. References are given in Section 9.

This technical document summarizes and analyzes the results of the detailed calculations that were performed in determining the evaluation of codisposal viability of Fermi SNF. These calculation documents and the corresponding section in which their results are summarized are shown in Table 1.

Table 1. List of Supporting Documents

Discipline	Document Title	Section Summarized	Reference
Structural	*Structural Calculations for the Codisposal of Enrico Fermi Spent Nuclear Fuel in a Waste Package*	Section 3	CRWMS M&O 1999n
Thermal	*Thermal Evaluation of the Enrico Fermi Co-disposal Waste Package*	Section 4	CRWMS M&O 1999a
Shielding	*Dose Calculations for the Co-Disposal WP of HLW Canisters and Fermi U-Mo Alloy SNF*	Section 5	CRWMS M&O 1999b
Degradation and Geochemistry	*EQ6 Calculations for Chemical Degradation of Enrico Fermi Spent Nuclear Fuel Waste Packages*	Section 6	CRWMS M&O 1999m
Intact- and Degraded-Mode Criticality Analyses	*Enrico Fermi Fast Reactor Spent Nuclear Fuel Criticality Calculations: Intact Mode*	Section 7	CRWMS M&O 1999d
	Enrico Fermi Fast Reactor Spent Nuclear Fuel Criticality Calculations: Degraded Mode	Section 7	CRWMS M&O 2000e

1.2 SCOPE

This technical document, *Evaluation of Codisposal Viability for U-Zr/U-Mo Alloy (Enrico Fermi) DOE-Owned Fuel*, evaluates the performance of Fermi SNF in the 5-DHLW/DOE SNF waste package. The Fermi SNF is the representative fuel for the U-Zr and U-Mo (HEU) group. The remaining fuels in this group must be demonstrated to be bounded by the values in the Fermi SNF data report (DOE 1999).

1.3 QUALITY ASSURANCE

This technical document was prepared in accordance with AP-3.11Q, *Technical Reports*. The responsible manager for DOE Fuel Analysis has evaluated this report development activity in accordance with QAP-2-0, *Conduct of Activities*. The evaluations (CRWMS M&O 1999i and CRWMS M&O 1999l) concluded that the development of this report is subject to the DOE Office of Civilian Radioactive Waste Management *Quality Assurance Requirements and Description* (QARD) controls (DOE 2000). Though QAP-2-0, *Conduct of Activities*, has been replaced by AP-2.21Q, *Quality Determinations and Planning for Scientific, Engineering, and Regulatory Compliance Activities*, these activities remain in effect. The information provided in this report is to be used to evaluate the codisposal viability of U-Zr and U-Mo (HEU) fuel. AP-SV.1Q, *Control of the Electronic Management of Data* does not apply because there is no electronic data associated with this report.

There is no determination of importance evaluation developed in accordance with Nevada Line Procedure NLP-2-0, *Determination of Importance Evaluations*, since the report does not involve any field activity.

This technical report is based in part on unqualified data. The unqualified data is only used to determine the representative values and identify items that are important to criticality control for this fuel group by establishing the limits based on the representative fuel type (Enrico Fermi) for this group (U-Zr and U-Mo [HEU] fuel). Hence, the input values used for evaluation of codisposal viability of the U-Zr and U-Mo (Enrico Fermi) SNF do not constitute data that have to be qualified in this application. They only establish the bounds for acceptance (Assumption 2.3.6.1). Since the input values are not relied upon directly to address criticality control and waste isolation issues and since the design inputs do not affect a system characteristic that is critical for satisfactory performance, according to the governing procedure (AP-3.11Q, *Technical Reports*), the data do not need to be controlled as TBV (to be verified) (AP-3.15Q, *Managing Technical Product Input*). However, prior to acceptance of the fuel for disposal, the items that are identified as important to criticality control in Section 8.6 must be qualified by any means identified in the QARD (i.e., experiment, non-destructive test, chemical assay, qualification under a program subject to the QARD requirements).

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2. DESIGN INFORMATION

The data/technical information obtained from the following sources are considered as references: ASTM B 575-97, *Standard Specification for Low-Carbon Nickel-Molybdenum-Chromium, Low-Carbon Nickel-Chromium-Molybdenum, Low-Carbon Nickel-Chromium-Molybdenum-Copper and Low-Carbon Nickel-Chromium-Molybdenum-Tungsten Alloy Plate, Sheet, and Strip*; ASTM A 516/A 516M-90, *Standard Specification for Pressure Vessel Plates, Carbon Steel, for Moderate- and Lower-Temperature Service*; ASTM G 1-90, *Standard Practice for Preparing, Cleaning, and Evaluating Corrosion Test Specimens*; ASTM A 240/A 240M-97a, *Standard Specification for Heat-Resisting Chromium and Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Pressure Vessels*; and ASME 1995, *1995 ASME Boiler and Pressure Vessel Code*.

The number of digits in the values cited herein may be the result of a calculation or may reflect the input from another source; consequently, the number of digits should not be interpreted as an indication of accuracy. In most tables, three to four digits after the decimal place have been retained to reduce the round-off errors in subsequent calculations.

2.1 DESIGN PARAMETERS

Each of the following subsections either describes the design of the waste package or identifies the basis of the major parameters.

2.1.1 Codisposal Waste Package

The codisposal waste package contains five defense high-level waste (DHLW) canisters surrounding a DOE standardized 18-in.-diameter DOE SNF canister. The 5-DHLW/DOE SNF waste package design is based on the Viability Assessment (DOE 1998b) waste package design. The barrier materials of the waste package are identical to those used for commercial SNF waste packages. The inner barrier is composed of 20 mm of high-nickel alloy ASTM B 575 (Alloy 22) and serves as a corrosion-resistant material. The outer barrier is composed of 100 mm of carbon steel (ASTM A 516 Grade 70) and serves as a corrosion-allowance material (CRWMS M&O 1997e, pp. 56 and 72). The outside diameter of the waste package is 2,120 mm, and the length of the inside cavity is 3040 mm, which is designed to accommodate the Savannah River Plant (SRP) HLW canister (CRWMS M&O 1998b). The lids of the inner barrier are 25 mm thick; those of the outer barrier, 110 mm thick. There is a 30-mm gap between the inner and outer barrier upper lids. Each end of the waste package has a 225-mm-long skirt. Table 2 summarizes the dimensions and materials of the waste package.

Table 2. Codisposal Waste Package Dimensions and Material Specifications

Component	Material	Parameter	Dimension (mm)
Outer barrier shell	ASTM A 516 Grade 70	Thickness	100
		Outer diameter	2,120
Inner barrier shell	ASTM B 575	Thickness	20
		Inner length	3040
Top and bottom outer barrier lids	ASTM A 516 Grade 70	Thickness	110
Top and bottom inner barrier lids	ASTM B 575	Thickness	25
Gap between the upper inner and outer closure lids	Air	Thickness	30
Support pipe	ASTM A 516 Grade 70	Outer diameter	565
		Inner diameter	501.5
		Length	3030

The DOE SNF canister is placed in a 31.75-mm-thick carbon steel (ASTM A 516 Grade 70) support pipe with a nominal outer diameter of 565 mm. The support tube is connected to the inside wall of the waste package by a web-like structure of carbon-steel (ASTM A 516 Grade 70) plates that form a basket to support five long HLW canisters, as shown in Figure 1. The support tube and the plates are 3030-mm long.

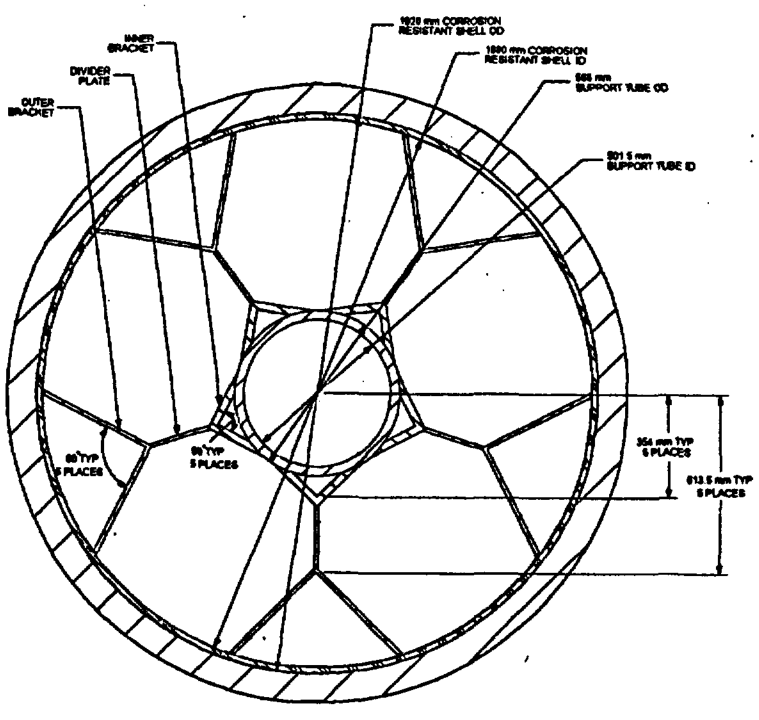

Figure 1. Codisposal Waste Package Cross Section

2.1.2 HLW Glass Pour Canisters

The SRP Defense Waste Processing Facility HLW glass canister, shown in Figure 2, is used in the Enrico Fermi waste package. It has a cylindrical stainless steel (SS) (Type 304L) shell with an outer diameter of approximately 610 mm (24.00 in.), a wall thickness of 9.525 mm, and a nominal length of 3 m (DOE 1992, p. 3.3-1). The flanged head and neck of the canister is 225.6 mm high. HLW glass occupies 85% of the volume of the canister. The glass weight is 1,682 kg, and the approximate total loaded weight of the canister is 2,182 kg (DOE 1992, pp. 3.3-6). The nominal dimensions of the canister are used for analyses. The heat generation from a single canister is 710.1watts (W) (CRWMS M&O 1997f, Attachment LII). The geometry and material specifications for HLW glass canisters are given in Table 3.

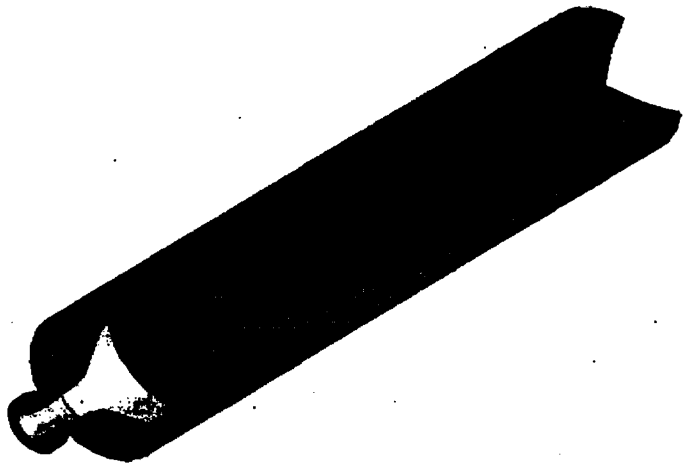

Figure 2. HLW Glass Pour Canister

Table 3. Geometry and Material Specifications for HLW Glass Canisters

Component	Material	Parameter	Value
Savannah River Site 3-m Canister	SS 304L	Outer diameter	610 mm
		Total weight of canister and glass	2,182 kg
		Fill volume of glass in canister	85%
		Wall thickness	9.525 mm
		Length	3,000 mm

SOURCE: DOE 1992

2.1.3 DOE SNF Canister

The conceptual design for the standardized 18-in.-diameter DOE SNF canister is taken from DOE 1998a, p. 7, A-ii, and A-iii. It is recognized that DOE 1998a has been revised; however, only Revision 0 was available at the time the calculations supporting this report were performed. A review of the most recent revision (Rev. 3) indicated that there is no impact on current results since neither the internal dimensions nor the materials of the canister changed. The canister is a right circular cylinder of SS (Type 316L) that contains a SS (Type 316L) basket. The basket is not a standard part of the DOE SNF canister. The basket design is modified for each specific fuel type. The basket provides structural support and acts as a guide for the fuel assembly during loading. The dimensions for the DOE SNF canister are a 457.2-mm (18.00-in.) outer diameter with a 9.525-mm (0.375-in.) wall thickness. The nominal internal length of the canister is 2575.0 mm (101.378 in.) and the nominal overall length is 2999.0 mm (118.071 in.). There is a curved carbon-steel impact plate, which varies in thickness from 15.240 mm (0.60 in.) to 50.80 mm (2.0 in.) at the top and bottom boundaries of the canister. In addition, there are 9.525-mm (0.375-in.)-thick ASME (American Society of Mechanical Engineers) flanged and dished heads and 12.70-mm (0.5 in.)-thick lifting rings at each end of the canister. The maximum loaded weight of the canister is 2,270 kg. A drawing of the canister is shown in Figure 3.

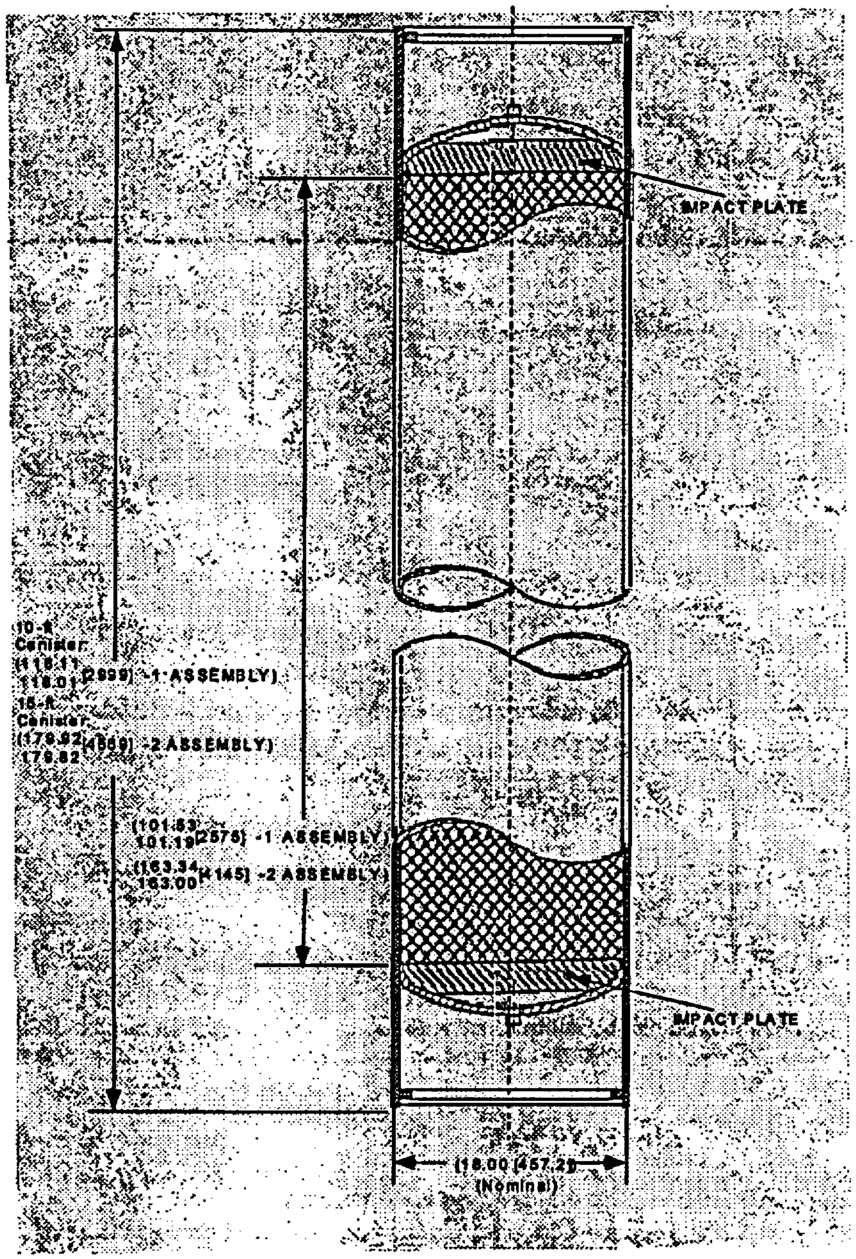

Figure 3. The Standardized 18-In.-Diameter DOE SNF Canister
(values in brackets are in mm)

Currently, all the Enrico Fermi fuel assemblies are disassembled (derodded), and the fuel pins are stored under water inside aluminum canisters. The 980 declad (cladding removed) fuel pins from seven sections of Fermi fuel are stored in the -02 aluminum canisters, but these canisters will not be shipped to the proposed repository for long-term storage without additional treatment or repackaging. The rest of the 191 (clad) fuel sections, each consisting of 140 zirconium-clad fuel pins, are stored in the -04 aluminum canisters. The -04 canisters were placed inside the -01 aluminum canisters that are to be shipped to the repository, i.e., -01 canisters contain the -04 canisters, see Figures 4, 5 (DOE 1999), 6, and 7 (CRWMS M&O 1999d) below.

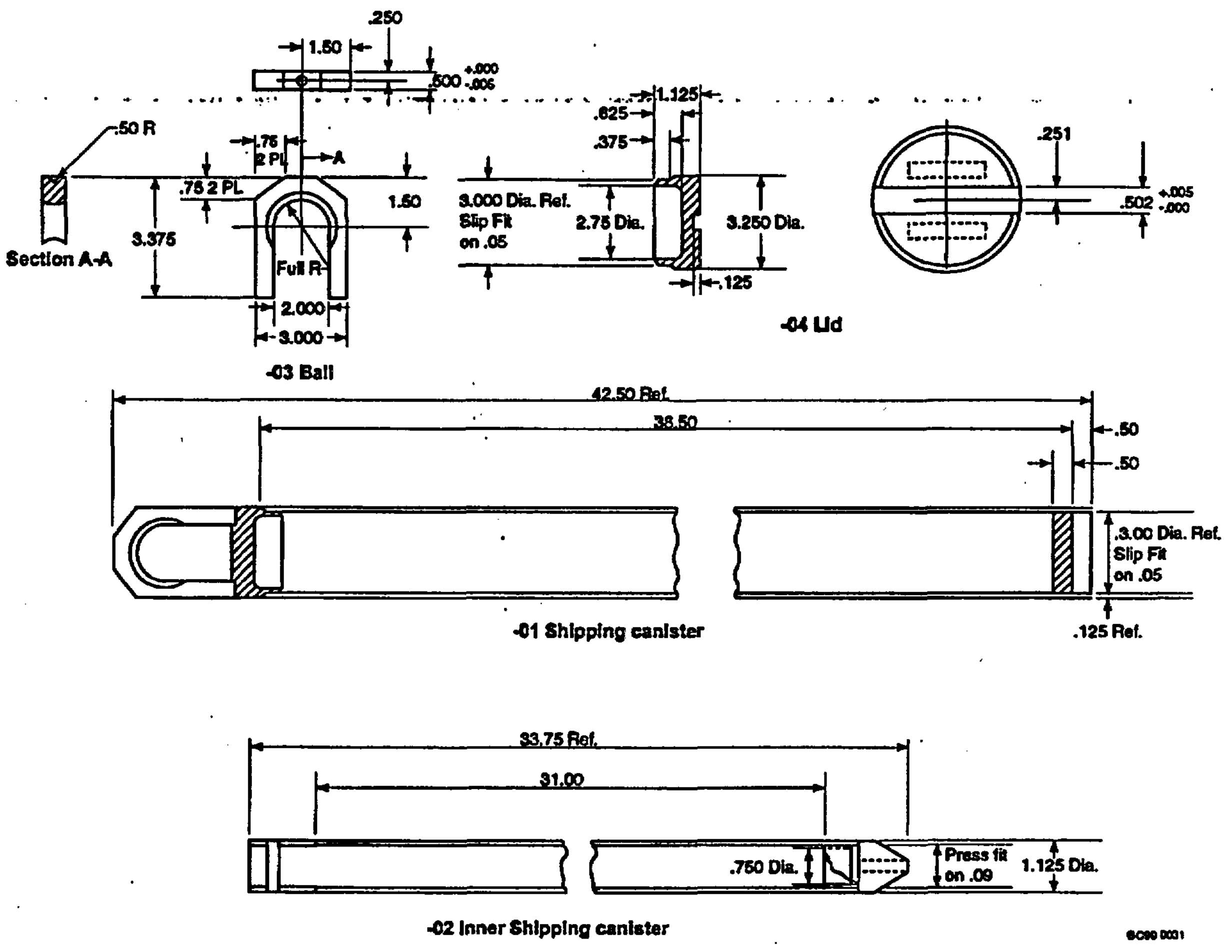

Figure 4. -01 and -02 Shipping Canisters (dimensions in inches)

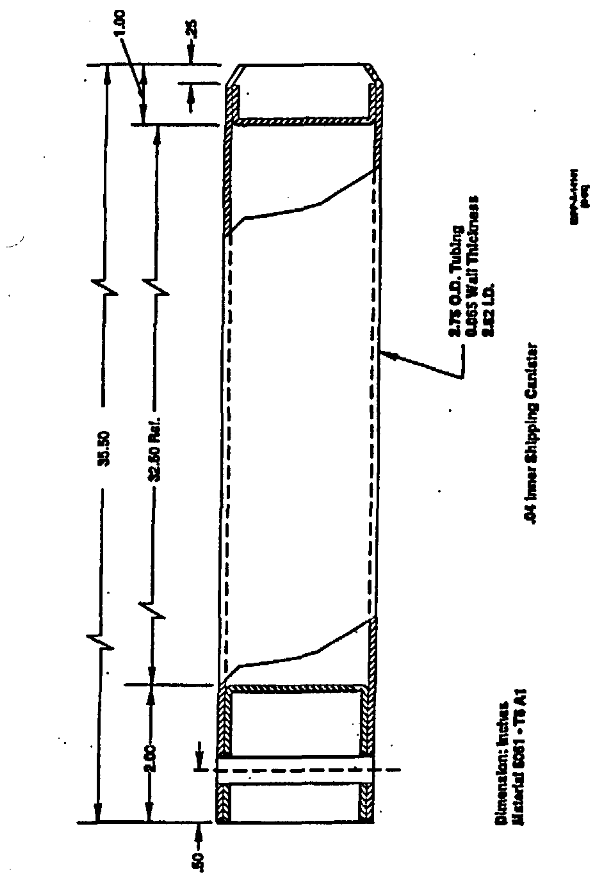

Figure 5. -04 Canister (dimensions in inches)

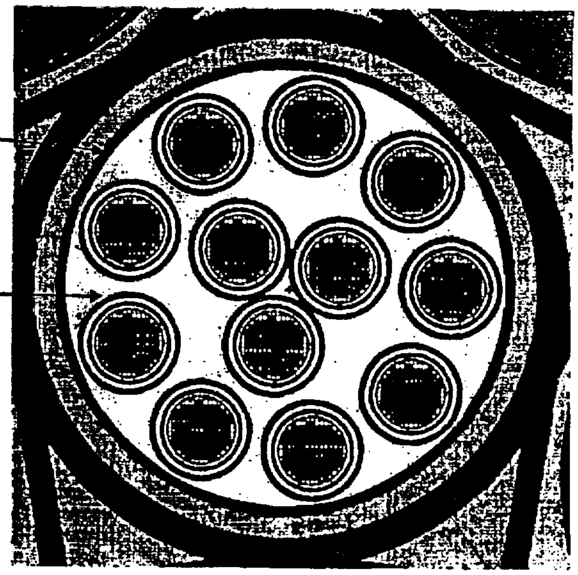

Figure 6. Represented Cross Section of DOE Enrico Fermi SNF Canister

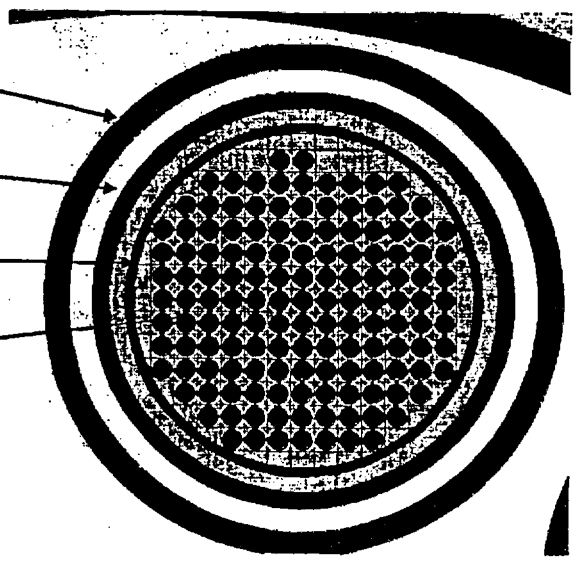

Figure 7. Enrico Fermi Fuel Pins in 4-in.-Diameter Pipe

The 191 de-rodded fuel sections (26,740 clad fuel pins) are the focus of the evaluation in this document to support repository acceptance. The pertinent design dimensions of the -01 and -04 canisters are shown in Figures 4 and 5. The bare fuel pins with the swaged, pointed ends as shown in Figure 8 are to be qualified for shipment to the repository. In each -04 canister, 140 pins were loose-packed without any supporting/spacing mechanism (Figure 7 [the grid lines in Figure 7 are part of the analysis model and do not represent any types of matrix or structured packing]).

For codisposal with the waste package, the Fermi SNF is to be stored inside the DOE SNF canister, as described above. Figure 6 is a cross-sectional view of the initial, proposed storage configuration. As shown in this figure and in Appendix A, the proposed arrangement is that nine 4-in.-diameter steel pipes that are located near the inside wall, and three steel pipes are located in a cluster at the center of the DOE SNF canister. The 12 steel pipes are welded to a base plate to form a basket. There are also dividers welded between the pipes and a lifting rod in the basket structure. Two such baskets are stacked axially as shown in Appendix A. The space between all the 4-in. steel pipes is filled with iron shot (CRWMS M&O 1996) to exclude water (prevent neutron moderation) and provide support for the steel pipes. The iron shot is also a mechanism for introducing the neutron absorber, such as gadolinium phosphate ($GdPO_4$), either in the iron shot or in the form of a mixture.

The dimensions of the -01 and -04 canisters are summarized below and are taken from DOE 1999, p. 11.

Table 4. Dimensions of the -01 and -04 Canisters

	-04 Inner Canister (inches)	-01 Shipping Canister (inches)
Overall length	35.5	42.5
Outside diameter	2.75	3.25
Inside diameter	2.62	3.0
Wall thickness	0.065	0.125
Inside length	32.5	37.5 (approximately)

NOTE: There is no filler gas inside the aluminum storage canisters. It is expected that air will fill the void.

2.1.4 DOE Enrico Fermi SNF

The Fermi SNF fuel pin is made of a solid uranium-molybdenum alloy, 3.7592 mm (0.148 in.) in diameter, and is bonded metallurgically to a zirconium tube, i.e., no gaps. The zirconium cladding has a thickness of only 5 mils (5 thousandth of an inch). The fuel material is 84.6 weight percent (wt%) uranium, that has been enriched to a nominal 25.69 percent U-235; 10.22 weight percent molybdenum; and the balance is zirconium cladding and impurities. Each fuel pin weighs 159 grams. Thus, a fuel section of 140 pins weighs about 23 kg, of which 4.816 kg is U-235. The fuel pins were originally fabricated in lengths of 12 feet or greater and were later cut into 774.70-mm (30.5-in.) lengths. The ends of the pins were swaged to a point. Figure 8 provides the general views and dimensions of the fuel pin design.

Zirconium end caps were placed on the end of each pin and secured in place by cold swaging. These Zr end caps were removed before storing the fuel in existing cans. The total length of the fuel pin after cold swaging (including the pointed swaged ends but without the free end caps) is 781.9390 mm (30.785 in.) as shown in Figure 8. The cold swaging process provides mechanical seal between the cladding on the pointed ends and the end cap to protect the U-Mo alloy.

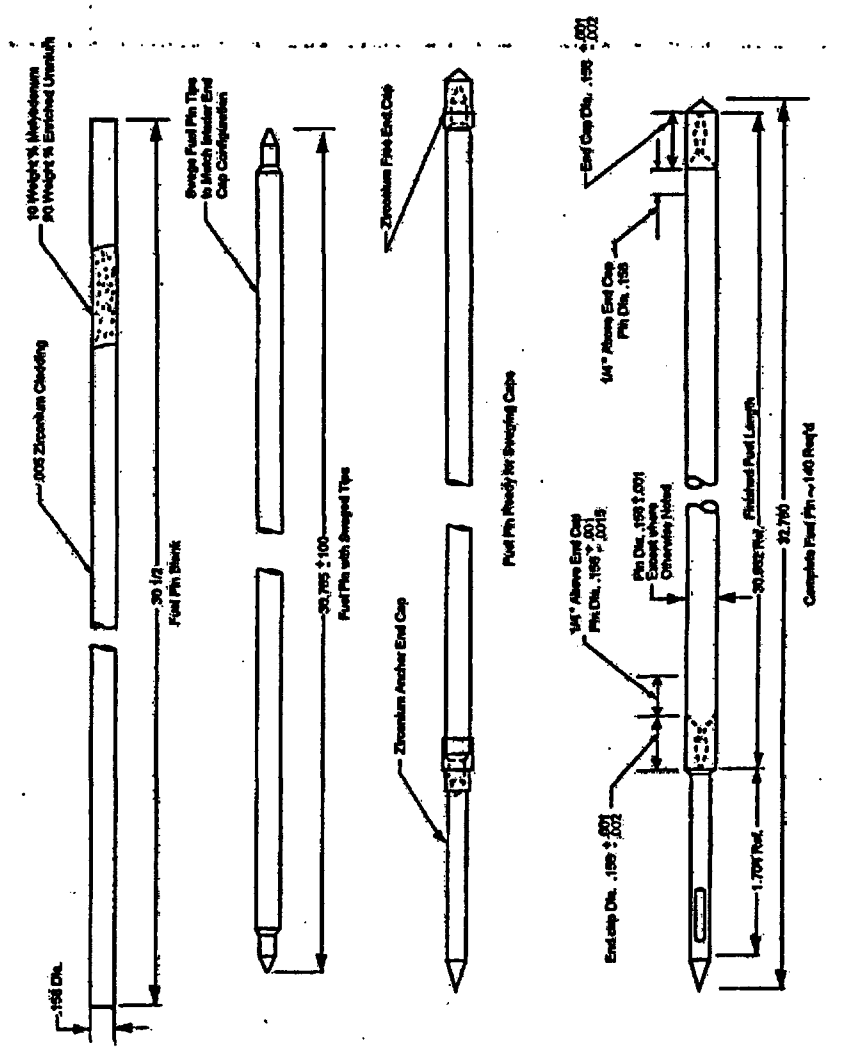

Figure 8. Fermi Fuel Pin Details (dimensions in inches)

2.1.5 Structural

A two-dimensional (2-D) finite-element representation of the 5-DHLW DOE SNF waste package was developed to determine the effects of loads on the structural components due to a tipover design basis event (DBE). Calculations of maximum potential energy for each waste package handling accident scenario (horizontal drop, vertical drop, and tipover DBE) showed that the bounding dynamic load is obtained from a tipover case in which the rotating end of the waste package experiences the highest impact load. Therefore, for the 18-in. diameter DOE SNF canister and its structural components, the structural evaluations presented in this document are bounding for all handling-accident scenarios. The finite-element representation was developed using the dimensions provided in DOE 1999.

2.1.6 Thermal

Table 5 lists the heat output of the codisposal waste-package components. The waste package contains five Savannah River Site (SRS) HLW glass canisters, with a heat output history given in CRWMS M&O 1997f (Attachment LII), and one codisposal canister containing twenty-four Enrico Fermi SNF sections (i.e., 3360 fuel pins) in two stacks (see sketch in Appendix A of this report) with a maximum heat output history given in Appendix B of DOE 1999. The heat output of the Enrico Fermi fuel is equal to 1.1 watts per section of 140 fuel pins. Since the axial power peaking factor for the Fermi fuel is not known, a bounding value of 1.25 was used for the axial power peaking factor in the analysis.

Table 5. Heat Output of Each WP Heat Source

Time After Emplace-ment (yr)	Savannah River Site HLW (W/-04 canister)	Enrico Fermi SNF (W/-04 canister)	WP Total Heat Output (W)
0	710.1	1.1	3576.9
0.1	699.2	1.1	3522.4
0.2	689.0	1.1	3471.4
0.3	679.6	1.1	3424.4
0.4	670.8	1.1	3380.4
0.5	662.6	1.1	3339.4
0.6	654.9	1.1	3300.9
0.7	647.7	1.1	3264.9
0.8	641.0	1.1	3231.4
0.9	634.8	1.1	3200.4
1	628.9	1.1	3170.9
2	585.9	1.1	2955.9
3	559.9	1.1	2825.9
4	541.5	1.1	2733.9
5	526.7	1.1	2659.9
6	513.7	1.1	2594.9

Table 5. Heat Output of Each WP Heat Source (Continued)

Time After Emplace-ment (yr)	Savannah River Site HLW (W/-04 canister)	Enrico Fermi SNF (W/-04 canister)	WP Total Heat Output (W)
7	501.7	1.1	2534.9
8	490.4	1.1	2478.4
9	479.4	1.1	2423.4
10	468.8	1.1	2370.4
20	376.0	1.1	1906.4
30	302.5	1.1	1538.9
40	244.2	1.1	1247.4
50	198.0	1.1	1016.4
60	161.2	1.1	832.4
70	131.9	1.1	685.9
80	108.5	1.1	568.9
90	89.82	1.1	475.5
100	74.84	1.1	400.6
200	18.33	1.1	118.1
300	8.15	1.1	67.2
400	4.55	1.1	49.2
500	2.79	1.1	40.4
600	1.87	1.1	35.8
700	1.37	1.1	33.3
800	1.11	1.1	32.0
900	0.96	1.1	31.2
1000	0.88	1.1	30.8

The thermal conductivity of the HLW glass is conservatively approximated as that of pure borosilicate glass, while the properties of density and specific heat are conservatively approximated as those of Pyrex glass. As with the other waste package components, only the axial cross section at the center of the canister is represented in the 2-D calculations. The values of thermal conductivity, specific heat, and density for borosilicate glass are 1.1 W/m/K, 835.0 J/kg/K, and 2,225.0 kg/m^3, respectively. The thermal conductivity is reported in CRWMS M&O 1995a, p. 13, as the mid-range value for a temperature range of 100 °C to 500 °C. The density and specific heat of Pyrex glass at 27 °C (300 K) (CRWMS M&O 1995a, p. 13) are used for the HLW glass (CRWMS M&O 1999a).

2.1.7 Shielding Source Term

The energy-dependent intensities for gamma and neutron radiation of each SRS HLW glass canister (CRWMS M&O 1997c, Attachment X, p. 1 and Attachment IX, p.1, respectively) are provided in Table 6.

Table 6. Gamma and Neutron Sources of SRS HLW Glass for the 10-Foot Canister

Gamma Source		Neutron Source	
Upper Energy Boundary (MeV)	Intensity (photons/s)	Upper Energy Boundary (MeV)	Intensity (neutrons/s)
0.05	1.3213E+15	0.10	1.970E+05
0.10	3.9581E+14	0.40	1.893E+06
0.20	3.0959E+14	0.90	6.337E+06
0.30	8.7394E+13	1.40	6.919E+06
0.40	6.3931E+13	1.85	6.123E+06
0.60	8.8265E+13	3.00	2.614E+07
0.80	1.3478E+15	6.43	3.416E+07
1.00	2.1344E+13	20.00	3.069E+05
1.33	2.9649E+13		
1.66	6.4161E+12		
2.00	5.1377E+11		
2.50	2.9370E+12		
3.00	2.0440E+10		
4.00	2.2835E+09		
5.00	5.2534E+05		
6.50	2.1058E+05		
8.00	4.1263E+04		
10.00	8.7544E+03		
Total	3.6750E+15	Total	8.208E+07

The maximum exposure of any standard Fermi fuel assembly is less than 3,130 MWd/mtU. For a typical Fermi fuel assembly with a burnup rate of 3130 MWd/mtU and 9,404-day decay time, the gamma spectrum are given in Table 7 and the total neutron source is 1.042E+03 (CRWMS M&O 1999b, p. 15).

Table 7. Gamma Sources for the Fermi U-Mo Alloy SNF Assembly at 9,404-Day Cooling Time

Upper Energy Boundary, (MeV)	Gamma Intensity, (photons/s)
0.02	4.984E+12
0.03	1.037E+12
0.05	9.112E+11

Table 7. Gamma Sources for the Fermi U-Mo Alloy SNF Assembly at 9,404-Day Cooling Time (Continued)

Upper Energy Boundary, (MeV)	Gamma Intensity, (photons/s)
0.07	9.651E+11
0.10	5.861E+11
0.15	3.796E+11
0.30	5.008E+11
0.45	2.192E+11
0.70	3.824E+12
1.00	3.578E+10
1.50	4.807E+10
2.00	9.240E+08
2.50	2.921E+05
3.00	1.414E+04
4.00	6.003E+01
6.00	1.414E+01
8.00	1.516E+00
14.00	1.665E-01
Total	1.349E+13

2.1.8 Material Compositions

The chemical compositions of the materials used in the analyses are given in Tables 8 through 12.

Table 8. Chemical Composition of ASTM B 575 (Alloy 22)

Element	Composition (wt%)	Value Used (wt%)
Carbon (C)	0.015 (max)	0.015
Manganese (Mn)	0.50 (max)	0.50
Silicon (Si)	0.08 (max)	0.08
Chromium (Cr)	20.0 - 22.5	21.25
Molybdenum (Mo)	12.5 - 14.5	13.5
Cobalt (Co)	2.50 (max)	2.50
Tungsten (Tn)	2.5 - 3.5	3.0
Vanadium (Va)	0.35 (max)	0.35
Iron (Fe)	2.0 - 6.0	4.0
Phosphorus (P)	0.02 (max)	0.02
Sulfur (S)	0.02 (max)	0.02
Nickel (Ni)	Remainder	54.63
Density = 8.69 g/cm^3		

SOURCE: ASTM B 575-97, p. 2.

Table 9. Chemical Composition of ASTM A 516 Grade 70 Carbon Steel

Element	Composition (wt%)	Value Used (wt%)
Carbon (C)	0.30 (max)	0.30
Manganese (Mn)	0.85-1.20	1.025
Phosphorus (P)	0.035 (max)	0.035
Sulfur (S)	0.035 (max)	0.035
Silicon (Si)	0.15-0.40	0.275
Iron (Fe)	Balance	98.33
Density[a] = 7.832 g/cm^3		

SOURCE: ASTM A 516/A 516M-90, Table 1.
NOTE: [a] Density of this material is given as 7.860 g/cm^3 in ASTM G 1-90, p. 7.

Table 10. Chemical Composition of Stainless Steel Type 304L

Element	Composition (wt%)	Value Used (wt%)
Carbon (C)	0.03 (max)	0.03
Manganese (Mn)	2.00 (max)	2.00
Phosphorus (P)	0.045 (max)	0.045
Sulfur (S)	0.03 (max)	0.03
Silicon (Si)	0.75 (max)	0.75
Chromium (Cr)	18.00 - 20.00	19.00
Nickel (Ni)	8.00 - 12.00	10.00
Nitrogen (N)	0.10	0.10
Iron (Fe)	Balance	68.045
Density[a] = 7.94 g/cm^3		

SOURCE: ASTM A 240/A 240M-97a, p. 2.
NOTE: [a] Density of this material is given as 7.94 g/cm^3 in ASTM G 1-90, p. 7.

Table 11. Chemical Composition of Stainless Steel Type 316L

Element	Composition (wt%)	Value Used (wt%)
Carbon (C)	0.03 (max)	0.03
Manganese (Mn)	2.00 (max)	2.00
Phosphorus (P)	0.045 (max)	0.045
Sulfur (S)	0.03 (max)	0.03
Silicon (Si)	0.75 (max)	0.75
Chromium (Cr)	16.00 - 18.00	17.00
Nickel (Ni)	10.00 - 14.00	12.00
Molybdenum (Mo)	2.00 - 3.00	2.50
Nitrogen (N)	0.10 (max)	0.10
Iron (Fe)	Balance	65.545
Density[a] = 7.98 g/cm^3		

SOURCE: ASTM A 240/A 240M-97a, p. 2.
NOTE: [a] Density of this material is given as 7.98 g/cm^3 in ASTM G 1-90, p. 7.

2.1.9 Degradation and Geochemistry

This section identifies the degradation rate of the principal alloys, the chemical composition of J-13 well water, and the drip rate of J-13 well water into a waste package. These rates are used in Section 6, Degradation and Geochemistry Analysis (CRWMS M&O 1999m).

2.1.9.1 Physical and Chemical Characteristics of Fermi Waste Package

Tables 8 through 11 provide a summary of the compositions of the principal alloys used in the degradation and geochemistry calculations. Table 12 gives the composition of the glass used in the calculations. The actual glass composition used in the glass pour containers may vary significantly from these values, since the sources of the glass and melting processes are not currently fixed. The silica and the alkali contents of the glass have perhaps the most significant bearing on EQ3/6 calculations. The amount of silica in the glass strongly controls the amount of clay that forms in the waste package, and the silica activity controls the presence of insoluble uranium phases, such as soddyite ($(UO_2)_2SiO_4 \cdot 2H_2O$). The alkali – sodium (Na), lithium (Li), and potassium (K) – content can cause pH to rise in the early stages of the EQ3/6 run, as glass degrades. The silica and alkali contents shown in Table 12 are typical for proposed DOE glasses (CRWMS M&O 1999m).

Table 12. Chemical Composition of SRS HLW Glass

Element /Isotope	Normalized Weight Percent	Gram-Atoms	Atom Fraction	Element/ Isotope	Normalized Weight Percent	Gram-Atoms	Atom Fraction
O	44.76964	2.798E+0	5.726E-1	Ni	0.734904	1.252E-2	2.562E-3
U-234	0.000328	1.401E-6	2.867E-7	Pb	0.060961	2.942E-4	6.021E-5
U-235	0.004351	1.851E-5	3.789E-6	Si	21.88782	7.793E-1	1.595E-1
U-236	0.001042	4.412E-6	9.029E-7	Th-232	0.185591	7.997E-4	1.636E-4
U-238	1.866591	7.841E-3	1.605E-3	Ti	0.596761	1.246E-2	2.551E-3
U[a]		7.864E-3	1.609E-3	Zn	0.064636	9.885E-4	2.023E-4
Pu-238	0.005182	2.177E-5	4.455E-6	B-10	0.591758	5.910E-2	1.209E-2
Pu-239	0.012412	5.192E-5	1.063E-5	B-11	2.61692	2.379E-1	4.868E-2
Pu-240	0.002277	9.487E-6	1.941E-6	B[a]		2.970E-1	6.077E-2
Pu-241	0.000969	4.018E-6	8.223E-7	Li-6	0.095955	1.595E-2	3.264E-3
PU-242	0.000192	7.919E-7	1.621E-7	Li-7	1.380358	1.967E-1	4.026E-2
Pu[a]		5.271E-5	1.079E-5	Li[a]		2.127E-1	4.353E-2
Cs-133	0.040948	3.081E-4	6.305E-5	F	0.031852	1.677E-3	3.431E-4
Cs-135	0.005162	3.826E-5	7.830E-6	Cu	0.15264	2.402E-3	4.916E-4
Cs[a]		3.464E-4	7.088E-5	Fe	7.390665	1.323E-1	2.708E-2
Ba-137	0.112669	8.230E-4	1.684E-4	K	2.988689	7.644E-2	1.564E-2
Al	2.331821	8.642E-2	1.769E-2	Mg	0.824754	3.393E-2	6.944E-3

Table 12. Chemical Composition of SRS HLW Glass (Continued)

Element	Normalized Weight Percent	Gram-Atoms	Atom Fraction	Element	Normalized Weight Percent	Gram-Atoms	Atom Fraction
S	0.129454	4.037E-3	8.262E-4	Mn	1.55765	2.835E-2	5.802E-3
Ca	0.661884	1.651E-2	3.380E-3	Na	8.628352	3.753E-1	7.680E-2
P	0.014059	4.539E-4	9.289E-5	Cl	0.115909	3.269E-3	6.691E-4
Cr	0.082567	1.588E-3	3.250E-4	Ag	0.050282	4.661E-4	9.539E-5
				Total	100	4.887E+0	1.000E+0

NOTE: [a] HLW glass elements with more than one isotope were combined.

Table 13 (CRWMS M&O 1999m) provides the reasonable maxima and averages for the steel degradation rates. For a comparable specific surface area, the carbon steel is expected to degrade much more rapidly than the stainless steels (316L and 304L). In addition, the stainless steels contain significant amounts of chromium (Cr) and molybdenum (Mo), and under the assumption of complete oxidation, should produce more acid, per volume, than the carbon steel.

Table 13. Steel Degradation Rates

	A 516 Carbon Steel	304L SS	316L SS
Molecular Weight (g/mole)	55.055	54.664	55.363
Density (g/cm^3)	7.85	7.94	7.98
Average Rate (μm/year)	35	0.1	0.1
High Rate (μm/year)	1.0E+02	1.0	1.0

DTN: SN9911T0811199.003

Table 14 (CRWMS M&O 1999m) provides the chemical properties for two types of aluminum alloy canisters used for Fermi SNF (the properties in Table 14 do not affect the criticality results presented in this report, therefore the information contained in Table 14 does not need to be processed into TDMS per Section 5.2e)1) of AP-3.11Q/Rev. 1/ICN 1).

Table 14. Aluminum Canisters Degradation Rates

	Aluminum, Type 3003	Aluminum, Type 6061
Mol. Wt. (g/mole)	27.29523	27.20351
Density (g/cm^3)	2.71	2.71
Average Rate (moles/cm^2·s)	1.1886E-10	1.1886E-10

Rates for glass degradation in Table 15 were taken from CRWMS M&O 1995b (Figure 6.2–5) and converted to units appropriate for running EQ6. The high rate corresponds approximately to pH 9 at 70 °C, and the average rate to pH 8 at 25 °C.

Table 15. HLW Glass Degradation Rates

Molecular Weight of SRS HLW Glass (g/mol)	20.4641
Average Rate (g/m^2·day)	1.00E-4
High Rate (g/m^2·day)	3.00 E-2

DTN: SN9911T0811199.003

Table 16 (CRWMS M&O 1999m) summarizes the characteristics of the Fermi fuel. Because no ~~fission product inventory is available, the calculations use the composition of fresh fuel~~. Use of fresh fuel is conservative, since most fission products have significant neutron absorption cross sections, and the unirradiated fuel has a higher fissile content than partially spent fuel (note that the properties of the spent fuel do not affect the criticality results presented in this report because the fissile material was retained in the criticality calculations, therefore the information contained in Table 16 does not need to be processed into TDMS per Section 5.2e)1) of AP-3.11Q, Rev. 1, ICN 1).

Table 16. Fermi Fuel Compositions and Degradation Rates

Average Molecular Weight of U-Mo Alloy (g/mol)	206.6290
Density of Fuel Pins (g/cm^3)	17.424
Average Fuel Degradation Rate (μm/yr)[a]	3.000E-01
Average Fuel Degradation Rate (mol/cm^2·s)	7.947E-14

NOTE: [a] The unit in the reference (CRWMS M&O 1999m) is incorrectly used.

The web (consisting of divider plates, inner and outer brackets) of the waste package basket (see Appendix B) are composed of A 516 carbon steel and serve two purposes: they center and hold the DOE SNF canister in place and separate the glass pour containers and prevent them from transmitting excessive loads to the SNF canister in the event of a fall or tip over event. At the center of the webs is a thick (3.175 cm) cylindrical support tube, also fabricated of A 516. In a breach scenario, the webs are exposed to water and corrode before the rest of the package; they are expected to degrade within a few hundred to a few thousand years. The oxidation of steel forms hematite (Fe_2O_3), which decreases the void space in the package by ~13%, or it forms goethite (FeOOH), which can decrease the void space by ~22% (CRWMS M&O 1999m). The differences are due to the lower density of goethite compared to hematite (CRWMS M&O 1999m, Table 5-6). Thus, the void space can be significantly reduced after the breach of the package due to the formation of corrosion products from corrosion of the webs.

The DOE SNF canister fits inside the central support tube of the waste package basket. The canister is composed primarily of 316L stainless steel, with two internal, thick impact plates of carbon steel (approximated as A 516 in the calculations). To improve mechanical strength of the DOE SNF canister and reduce the probability of effective water moderation, iron shot is included for preferential corrosion by water entering the breached SNF canister. The resulting primary corrosion products will be either hematite or goethite, both of which have lower densities than the original iron shot. The larger volume of these products decreases the void volume within the DOE SNF canister that can fill with water. In addition, $GdPO_4$ will be added to the iron shot to decrease chances of internal criticality. The iron shot will be distributed among, and inside, the 4-in.-diameter pipes containing the -01 canisters with Fermi fuel pins.

2.1.9.2 Chemical Composition of J-13 Well Water

The geochemistry calculations reported in this document have used the J-13 well water composition, which is shown in Table 17, for water dripping into the waste package. Since this water composition was determined from a well drilled into the saturated zone beneath the planned repository location, there is some question of the compositional deviations to be expected for water dripping into the repository drift, which is in the unsaturated zone. Several alternative versions of the J-13 well water composition have been proposed and used in other geochemistry calculations.

The rationale for the use of J-13 water composition is that it is representative of the groundwater entering the drift because it was collected from the same stratigraphic unit that the repository will occupy. The fact that the J-13 samples were taken below the water table while the repository location is above the water table is not of concern for the following reasons. The groundwater composition is controlled largely by transport through the host rock, over pathways of hundreds of meters. The host rock composition is similar in both the unsaturated and the saturated areas of this unit and is not expected to change substantially over time. Any groundwater chemistry alteration by the initial thermal perturbation from the waste package heat will have died out well before the initial waste package breach, which is now estimated to be 50,000 years (CRWMS M&O 2000b, p. 5-2).

Silicate complexes with Gd were not included in this calculation because the available literature indicates that, if they exist, they are too unstable to permit measurement of their equilibrium constants. For example, the Gd silicates are not included in the classic study of rare-earth chemistry in the planned Swedish repository environment: *A Selected Thermodynamic Database for REE to be Used in HLNW Performance Assessment Exercises* (Spahiu and Bruno 1995, p. ii). This compilation is also suitable for use by the YMP because the chemical composition of granite (which is the host rock for the Swedish repository) is virtually identical to that of Yucca Mountain rhyolitic tuff. This document specifically states "In most natural waters, the carbonate complexes are accepted as the dominant soluble species of the rare earths" (Spahiu and Bruno 1995, p. 7). Additionally, some aspects of this issue have been more extensively discussed in a previous study of Gd loss from a waste package (CRWMS M&O 1997g, Section 5.3.2). The following two paragraphs address the sensitivity of geochemistry results to potential variations in the composition of the dripping water.

Table 17. Composition of J-13 Well Water

Component	mg/l
Na^+	45.8
K^+	5.04
Ca^{++}	13.0
Mg^{++}	2.01
NO_3^-	8.78
Cl^-	7.14
F	2.18
SO_4^{2-}	18.4
Si	28.5
PO_4^{3-}	0.12
Alkalinity (mg/HCO_3^-)	128.9
pH = 7.41	

DTN: LL980711104242.054

Two major factors control how the J-13 well water chemistry might affect EQ6 calculations. The first factor is the presumed CO_2 pressure of equilibration, which is closely coupled to the pH of the J-13 well water; and the second is the content of dissolved species, which may react with package materials and fuel, and thus affect solubilities. An example of the second factor is the amount of available dissolved silica, which can precipitate uranium as insoluble minerals like soddyite and uranophane.

In other analyses of codisposal packages, order of magnitude variations in CO_2 pressure have not had significant effects on the calculated Gd loss (CRWMS M&O 1998j, Table 5.3-1). In codisposal packages, the chemistry of the package water is influenced, overwhelmingly, by the degradation of glass and other package materials. The alkali and alkaline earth content of the glass completely swamped the native J-13 well water composition in the bulk of the EQ6 scenarios run for the geochemical calculations (CRWMS M&O 1999m). The combination of steel and glass degradation drove the pH from ~3 to ~10, far greater than the range that exists in native J-13 well water (CRWMS M&O 1998p, Figures 5-2 through 5-20). The silica content of the glass is enormously greater than the amount of silica that can be contributed from J-13 water, even with long periods of flushing at relatively high rates. The calculations in CRWMS M&O 1998p showed that in cases of significant uranium solubility, the dominant aqueous species were carbonate and phosphate complexes. The phosphate was supplied overwhelmingly from the $GdPO_4$ criticality control material and the glass, and the high aqueous carbonate concentration (resulted from glass dissolution and the fixed CO_2 partial pressure).

2.1.9.3 Drip Rate of J-13 Well Water into a Waste Package

The rates at which water drips onto a waste package and flows through it are represented as being equal. The drip rate is taken from a correlation between percolation rate and drip rate (CRWMS M&O 1998i, Tables 2.3-55 and 2.3-56, Figures 2.3-112 and 2.3-114). Specifically, percolation rates of 40 mm/yr and 8 mm/yr correlate with drip rates onto the waste package of 0.15 m^3/yr and 0.015 m^3/yr, respectively. The choice of these particular percolation and drip rates is discussed in detail in CRWMS M&O 1998m, p. 19.

For the present study, the range of allowed drip rates was extended to include an upper value of 0.5 m^3/yr and a lower value of 0.0015 m^3/yr. The upper value corresponds to the 95 percentile upper limit for a percolation rate of 40 mm/yr, and the lower value is simply 0.1 times the mean value for the percolation rate of 8 mm/yr (CRWMS M&O 1998i, pp. 2-110 through 2-113). These extreme values were used, because prior studies (CRWMS M&O 1998j, Table 5.3-1) suggested that when waste forms are codisposed with glass, the greatest chance of Gd removal occurs when: (1) initial high drip rates cause leaching of the glass and removal of alkali from the waste package due to overflow, and (2) subsequent low drip rates which allow acid to build from the degradation of stainless steel.

A correlation of percolation rate versus drip rate prepared for the TSPA for site recommendation (CRWMS M&O 2000d, Figure 3) indicates that the drip rates used in the present study correspond to percolation rates ranging from approximately 5 to 100 mm/yr.

2.2 FUNCTIONS AND DESIGN CRITERIA

The design criteria are based on the *Defense High Level Waste Disposal Container System Description Document* (CRWMS M&O 1999o), hereafter referred to as the SDD. In this subsection, the key waste package design criteria from the SDD are identified for the following areas: structural, thermal, shielding, criticality within a breached but otherwise intact waste package, degradation and geochemistry, and criticality of a degraded waste package and waste form. SDD paragraph numbers are identified below as SDD X.X.X.X.

2.2.1 Structural

2.2.1.1 "The disposal container/waste package shall prevent the breach of the waste form canister during normal handling operations."

(SDD 1.2.1.8)

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

(SDD 1.2.2.1.3) (TBV-245)

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

(SDD 1.2.2.1.4) (TBV-245)

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

(SDD 1.2.2.1.6) (TBV-245)

Calculations of maximum potential energy for each handling accident scenario (horizontal drop, vertical drop, and tipover DBEs) showed that the bounding dynamic load is obtained from a tipover case in which the rotating top end of the waste package experiences the highest impact load with maximum velocity of 6.9 m/sec (CRWMS M&O 1999n, p. 15). The maximum velocities of the waste package for 2.4 m horizontal and 2.0 m vertical drops are approximately 6.86 m/sec ($v = \sqrt{2gh}$, where g is the gravitational acceleration, and h is height) and 6.26 m/sec, respectively. Therefore, tipover structural evaluations are bounding for all handling accident scenarios considered in the SDD. Section 3.3 addresses these requirements with respect to breach of the DOE SNF canister. This analysis assumes that MGR surface design will prevent events that exceed the bounding assumptions made in deriving the conclusions in this report.

The tipover DBE may only take place during a waste package transfer operation from vertical to horizontal position (just after waste package closure) or horizontal to vertical position (upon retrieval). Section 3, Structural Analysis, demonstrates that the waste package will not breach under such a handling-accident scenario.

2.2.2 Thermal

2.2.2.1 "The waste package shall maintain the temperature of HLW glass below 400 degrees C (752 degrees F) (TBV-092) under normal conditions, and below 460 degrees C (860 degrees F) (TBV-245) for short-term exposure to fire, as specified by Criterion 1.2.2.1.11."

(SDD 1.2.1.6) (TBV-092)(TBD-245)

2.2.2.2 "The waste package shall maintain zircaloy cladding of DOE SNF to less than 350 degrees C (662 degrees F) (TBV-241) under normal conditions, and below 570 degrees C (1,058 degrees F) (TBV-245) for short-term exposure to fire, as specified by Criterion 1.2.2.1.11. The temperature of other types of DOE fuel cladding shall be limited to (TBD-179)."

(SDD 1.2.1.7) (TBV-241)(TBV-245)(TBD-179)

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

(SDD 1.2.4.4)

2.2.3 Shielding

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

(SDD 1.2.4.3)(TBD-3764)

2.2.4 Degradation and Geochemistry

There are no degradation and geochemistry criteria in the SDD to address.

2.2.5 Intact and Degraded Criticality

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

(SDD 1.2.2.1.12)

As stated in Section 8.5, the results from the intact waste-package criticality analysis show that the requirement of k_{eff} plus bias and uncertainty be less than or equal to 0.95 is satisfied.

2.3 ASSUMPTIONS

In the course of performing the calculations summarized in this document, assumptions were made regarding the analyses of the waste package for structural, thermal, shielding, intact waste package criticality, degradation and geochemistry, and the criticality of a degraded waste package. The list of major assumptions that are essential to this technical document are provided below.

2.3.1 Structural

2.3.1.1 The containment barriers are assumed to have solid connections at the adjacent surfaces. The basis for this assumption is that the inner and outer barriers are either shrunk fit or the inner barrier is weld clad onto the outer barrier inner surface (CRWMS M&O 1997b). For each one of these fabrication processes, it is reasonable to assume solid contact between the barriers. This assumption is used in Section 3.

2.3.1.2 The target surface is conservatively assumed to be essentially unyielding by using a large elastic modulus for the target surface compared to the waste package. The basis for this assumption is that a bounding set of results is required in terms of stresses and displacements and the use of an essentially unyielding surface results in slightly higher stresses in the waste package. This assumption is used in Section 3.

2.3.2 Thermal

2.3.2.1 For Enrico Fermi fuel, an axial power peaking factor (PPF) of 1.25 is assumed. This is a conservative value given in CRWMS M&O 1997f, p. 29, for pressurized water reactor fuel. The HLW glass canisters are assumed to have an axial PPF of 1.00 (CRWMS M&O 1997f, p. 53). This assumption is used throughout Section 4.

2.3.2.2 Representing only conduction and radiant heat transfer inside the waste package is assumed to provide conservative results (higher temperatures) for this calculation. The fill gas inside the waste package allows a natural convective heat transfer to occur; however, since only a few, small enclosed basket cavities exist and the temperature gradient in the enclosure is not significant, the helium circulation is considered insignificant. Thus, the problem may be represented with only the dominant heat-transfer modes, with a negligible or conservative impact upon the results. This assumption is used throughout Section 4.

2.3.3 Shielding

2.3.3.1 The contents of a Fermi DOE SNF canister are assumed to be homogenized inside the cavity of canister. This model is conservative, because the homogenization process essentially moves the radiation source closer to the outer surfaces of the waste package, allowing more particles to reach the outer surface, and decreases the self-shielding effect of the fuel (Parks et al., p. 85). This assumption is used to obtain the results provided in Section 5.

2.3.3.2 An assumed axial PPF of 1.25 was used for the SNF fuel source to bound the axial source distribution. This value is based on the predicted heat profile shown in CRWMS M&O 1997f, p. 29 for PWR fuel. This assumption was used to obtain the results provided in Section 5.

2.3.3.3 It is assumed that the dose rates due to secondary gamma rays are negligible. The basis for this assumption is that the neutron source intensities are about 10 and 8 orders of magnitude smaller (see Tables 6 and 7) than the gamma source intensity for the Fermi U-Mo alloy fuel and HLW, respectively. Therefore, no coupled neutron-photon calculation is performed. This assumption is used to obtain the results provided in Section 5.

2.3.4 Degradation and Geochemistry

2.3.4.1 It is assumed that precipitated solids remain in place and are not mechanically eroded or entrained as colloids in the advected water. This assumption conservatively maximizes the size of potential deposits of fissile material inside the waste package. This assumption is used throughout Section 6.

2.3.4.2 It is assumed that sufficient decay heat is retained within the breached waste package over times of interest to cause convective circulation and mixing of the water inside the package. The analysis that serves as the basis for this assumption is discussed in CRWMS M&O 1999m. This assumption is used throughout Section 6.

2.3.4.3 It is assumed that water may circulate freely enough in the partially degraded waste package that all degraded solid products may react with one another in the aqueous solution. By facilitating contact of any acid that may result from the corrosion of steel with neutron absorbers in spent fuel, the code conservatively enhances potential preferential loss of neutron absorbers from the waste package. This assumption is used in Section 6.

2.3.4.4 It is assumed that the inner corrosion resistant material of the waste package will react so slowly with the infiltrating water (and water ponded in the waste package) that it will have a negligible effect on the solution's chemistry. The bases for this assumption consist of the facts that the corrosion resistant material is fabricated of Alloy 22, which corrodes very slowly compared (1) to other materials in the waste package and (2) to the rate at which soluble corrosion products will be flushed from the package. This assumption is used in Section 6.

2.3.4.5 For the purposes of calculating the disposition of degradation products (particularly the principal fissile element ^{235}U and the principal neutron absorber Gd) it is assumed that the thermodynamic database used for EQ3/6 calculations is correct and sufficiently complete, with respect to the chemical reactions that could significantly effect the waste package chemistry. The basis for this assumption is that previous results have been shown to be fairly insensitive to uncertainties in the thermodynamic constants of the relevant reactions (CRWMS M&O 1999c, Section 5.3.1).

2.3.5 Intact and Degraded Criticality

2.3.5.1 For the purposes of k_{eff} calculations, it is assumed that the impurities in the undegraded fuel matrix (B, C, Cr, Fe, Ni, N, O, Zr, Cu, and others) (DOE 1999, p. 9) are replaced with molybdenum (Mo). The basis for this assumption is that the replacement makes the calculations more conservative, as the majority of the elements present in the impurities have higher thermal neutron absorption cross sections than Mo. This assumption does not impact the geochemistry calculations that consider the solubility of all of these elements individually. This assumption is used in Section 7.

2.3.5.2 It is assumed that the 316 SS dividers and basket lifting pins may be ignored in the MCNP model and iron shot fills their place. The assumption results in more conservative (higher) estimates of k_{eff} value because water can fill the void space within the particles of iron shot. This assumption was used in Section 7.

2.3.5.3 For the purposes of k_{eff} calculations, it is assumed that iron shot (Fe) and the aluminum present in the DOE SNF canister degrade (oxidize) and produce FeOOH (goethite) and AlOOH (diaspore), respectively. This assumption does not impact the geochemistry calculations, which considered cases with the formation of goethite and hematite separately (with hematite suppression as the mechanism for permitting the slightly less stable goethite to precipitate). The principal case with goethite instead of hematite was case 15 in CRWMS M&O 1999m, and showed little difference, from the hematite cases, in Gd or U loss. The assumption of diaspore formation, instead of gibbsite can have little effect on the criticality calculations because the additional hydrogen in the gibbsite would be much less than that in the water assumed to be present for all the criticality calculations. The basis for this assumption is that the choice of FeOOH, as the oxidation product of Fe, over Fe_2O_3 (hematite) makes the calculations more conservative, as the hydrogen present in FeOOH acts as neutron moderator. This assumption is used in Section 7.

2.3.5.4 Beginning of life pre-irradiation fuel compositions were used for all calculations because it is conservative to assume fresh fuel as it is more neutronically reactive than spent fuel. The fuel density is determined by using the fuel mass and the volume of the fuel. These assumptions are used throughout Section 7.

2.3.5.5 For the degraded configurations with intact fuel pins surrounded by degraded waste package (WP) internals, the fuel pins are assumed to be stacked at the bottom of the WP in regular array rather than randomly. The basis for this assumption is that it is conservative since it allows the moderation to be optimal. This assumption is used in Section 7.

2.3.6 General

2.3.6.1 The technical information related to spent nuclear fuel (DOE 1999), DOE SNF canister (DOE 1998a), glass pour canister (DOE 1992), and source term of SRS HLW glass (CRWMS M&O 1997c) is only used to determine the bounding values and identify items that are important to criticality control for this fuel group by establishing the limits based on the representative fuel type (Enrico Fermi) for this group (U-Zr and U-Mo [HEU] fuel). The technical information used establishes the bounds for acceptance. The rational for this assumption is that it was designated by the DOE SNF grouping in support of criticality and related calculations. The burden is placed on the custodian of the SNF to demonstrate before acceptance of SNF by the CRWMS that SNF characteristics identified as important to criticality control or other analyses herein are not exceeded. This assumption is used in Sections 2 through 7.

2.4 BIAS AND UNCERTAINTY IN CRITICALITY CALCULATIONS

The purpose of this section is to document the MCNP (CRWMS M&O 1998e) (identified as Computer Software Configuration Item [CSCI] 30033 V4B2LV) evaluations of Laboratory Critical Experiments (LCE) performed as part of the Disposal Criticality Analysis Methodology program. Only LCEs relevant to Enrico Fermi fuel are studied. LCEs' results listed in this section are given in CRWMS M&O (1999j) for the thermal compound (heterogeneous) HEU systems and in CRWMS M&O (1999k) for the thermal solution uranium systems. The objective of this analysis is to quantify the MCNP Version 4B2 code system's ability to accurately calculate the effective neutron multiplication factor (k_{eff}) for various configurations. MCNP is set to use continuous-energy cross sections processed from the evaluated nuclear data files ENDF/B-V (Briesmeister 1997, App. G). These cross-section libraries are part of the MCNP code system that has been obtained from Software Configuration Management (SCM) in accordance with appropriate procedures. Each of the critical core configurations is simulated, and the results reported from the MCNP calculations are the combined average values of k_{eff} from the three estimates (collision, absorption, and track length) and the standard deviation (σ) of these results listed in the final generation summary in the MCNP output. When MCNP underpredicts the experimental k_{eff}, the experimental uncertainty is added to the uncertainty at 95% confidence from the MCNP calculation to obtain the bias. This bias along with the 5% margin (see Section 2.2.5) is used to determine the interim critical limit for all MCNP calculations of the waste package with the Enrico Fermi DOE SNF canister.

2.4.1 Benchmarks Related to Intact Waste Package Configurations

Several critical experiments with highly enriched fuel pins are relevant to the Enrico Fermi fuel with respect to intact criticality analyses: HEU-COMP-THERM-003, HEU-COMP-THERM-005, HEU-COMP-THERM-006, and HEU-COMP-THERM-007 (NEA 1998).

A series of critical experiments with water-moderated hexagonally-pitched lattices of highly enriched fuel pins of cross-shaped cross section was performed over several years in the Russian Research Center "Kurchatov Institute." The 22 experiments in this category that are analyzed in this report consist of the following:

1. Fifteen two-zone lattice critical experiments corresponding to different combinations of inner and peripheral zones of cross-shaped fuel pins at two pitches. For detailed descriptions of these experimental configurations see p. 2, and pp. 7 through 14 of NEA (1998), HEU-COMP-THERM-003 (HCT-003).

2. One critical configuration of hexagonal pitched clusters of lattices of fuel pins with copper (Cu) pins. Detailed experimental configuration descriptions are available on pp. 2 through 8 of NEA (1998), HEU-COMP-THERM-005 (HCT-005).

3. Three critical configurations with uniform hexagonal lattices with pitch values of 5.6, 10.0, and 21.13 mm. Detailed experimental configuration descriptions are available on pp. 2, 5, and 6 of NEA (1998), HEU-COMP-THERM-006 (HCT-006).

4. Three critical configurations with double hexagonal lattices of fuel pins and zirconium (Zr) hydride pins. Detailed experimental configuration descriptions are available on pp. 2 through 8 of NEA (1998), HEU-COMP-THERM-007 (HCT-007).

The pitch, number of pins, and number of fuel pins were the parameters that were varied. The maximum bias for this set of calculations is 0.019 (CRWMS M&O 1999j, pp. 16 through 19, and 76).

2.4.2 Benchmarks Related to Degraded Waste Package Configurations

Critical experiments with HEU (approximately 90 wt% enrichment) nitrate solution are described in detail in NEA (1998) (HEU-SOL-THERM-001, HEU-SOL-THERM-002, HEU-SOL-THERM-003). The concentration of fissile element in the solution, enrichment, reflector type and thickness, tank diameter, and solution height were among the parameters that were varied. The maximum bias for this set of experiments is 0.016 (CRWMS M&O 1997d, pp. 26 through 32; CRWMS M&O 1999k, pp. 14 through 18).

2.4.3 Critical Limit

The worst-case bias, calculated from the MCNP simulations of the experiments described in Sections 2.4.1 and 2.4.2, is 0.02. This bias includes the bias in the method of calculation and the uncertainty in the experiments. Based on this bias, the interim critical limit is determined to be 0.93 after allowance for a five percent margin, for the bias in the method of calculation and the uncertainty in the experiments used to validate the method of calculation. This interim critical limit will be used until an addendum to the topical report is prepared to establish the final critical limit.

3. STRUCTURAL ANALYSIS

3.1 USE OF COMPUTER SOFTWARE

The finite-element analysis (FEA) computer code used for this evaluation is ANSYS Version (V) 5.4, which is identified as CSCI 30040 V5.4 (CRWMS M&O 1998c) and was obtained from SCM in accordance with appropriate procedures. ANSYS V5.4 is a commercially available FEA code and is appropriate for structural analysis of waste packages as performed in this analysis. The software qualification of the ANSYS V5.4 software, including problems of the type analyzed in this report, is summarized in the Software Qualification Report (SQR) for ANSYS V5.4 (CRWMS M&O 1998c). ANSYS V5.4 is also referred to herein as ANSYS.

3.2 DESIGN ANALYSIS

Finite-element structural analyses for the components of the 5-DHLW/DOE SNF waste package are summarized in this section. A detailed description of the finite-element representations, the method of solution, and the results are provided in CRWMS M&O 1999n. The results of these analyses are compared to the design criteria in the 1995 American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC), Section III, Subsection NB (ASME 1995), so that conclusions can be drawn regarding the structural performance of the 5-DHLW/DOE SNF waste package design.

The design approach for determining the adequacy of a structural component is based on the stress limits given in the 1995 ASME BPVC sections cited above. The ultimate tensile strength (S_u) and the design stress intensity (S_m) of the materials in Table 18 are used to present the design criteria from appropriate sections of the 1995 ASME BPVC.

Table 18. Containment Structure Allowable Stress-Limit Criteria

Category	Containment Structure Allowable Stresses	
	Normal Conditions (ASME 1995, Division 1, Subsection NB, Articles NB-3221.1 and NB-3221.3)	Accident Conditions (Plastic Analysis,ASME 1995, Division 1, Appendix F, Article F-1341.2)
Primary membrane stress intensity	S_m	0.7 S_u
Primary membrane and bending stress intensity	1.5 S_m	0.9 S_u

This analysis is within the bounds of the structural design criteria in Section 2.2.1; however, it does not consider incredible DBEs (e.g., crane two-block events).

3.3 CALCULATIONS AND RESULTS

3.3.1 Description of the Finite-Element Representation

A two-dimensional (2-D) transient dynamic calculation has been performed for the structural analysis of the Enrico Fermi SNF canister within the 5-DHLW (defense high-level waste) DOE SNF waste package. This analysis, performed with ANSYS V5.4, considers a bounding dynamic load from a tip-over design basis event on the cross section of the waste package at its rotating end. Since the potential energy of the rotating end is larger than the potential energy of any other DBEs, the tipover DBE is considered both bounding and appropriately conservative for structural design purposes. The finite-element representation of the fuel assemblies includes the minimum level of detail needed to represent the mass and mechanical/physical properties of the fuel assemblies and to meet the computational requirements of ANSYS V5.4.

The 2-D finite-element representation is developed using the dimensions provided in Appendices A and B. A one-half-symmetry finite-element representation is developed for the waste package (see Figure 9). The finite-element representation includes the outer and inner barriers, basket assembly, support pipe, uppermost HLW pour canister, DOE SNF canister shell, 4-in. pipes and the dividers (Appendices A and B). The finite-element representation also includes masses of the other four pour canisters, the aluminum canisters and the fuel. The barriers are assumed to have solid connections at the adjacent surfaces (Assumption 2.3.1.1) and are constrained in a direction perpendicular to the symmetry plane. The one HLW pour canister that is located above the DOE SNF canister is created using 2-D elements. The remaining pour canisters are included in the representation as point mass elements at the points of contact of the pour canisters with the inner barrier and the divider plates. These locations are approximately at the mid-point of each component or segment at which the pour canisters would be in contact. This approach is a realistic way to simulate the effect of each pour canister in contact with the waste-package internals. This approach reduces the computer execution time needed for this analysis. The finite-element representation is used to determine the maximum closure of the clearance space between components inside the 4-in. pipes and the aluminum canister to determine whether there is contact between the pipes and the aluminum canister during the DBE.

First, the impact velocity of the outer surface of the inner lid is calculated for a waste package tipover DBE. Then, this velocity is conservatively used in the 2-D finite-element analysis. Since the 2-D representation does not model the lids, the calculations will indicate that the waste package components undergo more deflection and stress than would actually occur if the lids were included. The target surface is conservatively assumed to be essentially unyielding because the elastic modulus used for the target surface is large compared to the elastic modulus used for the waste package (Assumption 2.3.1.2). The target surface is constrained at the bottom to prevent its horizontal and vertical motion. Contact elements are defined between the top pour canister and the inner brackets, and between the outer barrier and the target surface. Initial configuration of the finite-element representation includes a negligibly small gap for each contact element. This approach allows enough time and displacement for the waste package and its internals to ramp up to the specified initial velocity before the impact. With this initial velocity, the simulation is then continued throughout the impact until the waste package begins to rebound; at that time, the stress peaks, and the maximum displacements are obtained.

The vitrified HLW glass material properties are represented by ambient temperature properties of general borosilicate glass. This document does not specifically report any results for the individual HLW glass canisters.

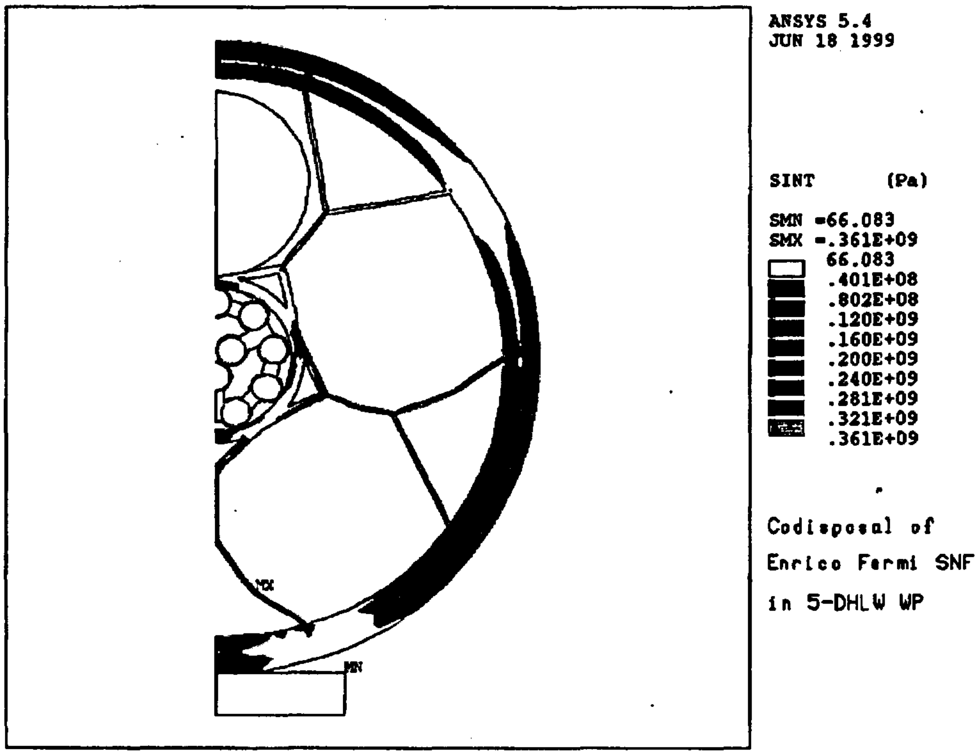

Figure 9. Stresses in the 5-DHLW/DOE SNF Waste Package

3.3.2 Results

The structural response of the waste package to tipover accident loads is reported using maximum stress values and displacements obtained from the finite-element solution to the problem. The results show that the maximum cavity closure inside the 4-in. pipe is 6.97 mm (CRWMS M&O 1999n). Available space between an individual 4-in. pipe and an aluminum -01 canister shell is 9.49 mm (CRWMS M&O 1999n). Hence, there is no interference between the two components from a tipover DBE. The maximum stress in the DOE SNF canister structural components and internals is 265 MPa (CRWMS M&O 1999n), which is less than the 483 MPa tensile strength of 316L SS (CRWMS M&O 1999n).

In performing the structure analysis, the maximum DOE canister mass of 2270 kg was utilized. The mass of the iron shot was not specifically considered in the structural analysis but was implicitly included in the maximum mass.

3.4 SUMMARY

The results given in Section 3.3 show that there is sufficient clearance between the inner diameter of the support pipe and the outer diameter of the DOE SNF canister in the case of a tipover DBE. Hence, there will be no interference between the two components, and the DOE SNF canister can be removed from the support pipe if necessary to set it inside another waste package. Additionally, there will be no breach of the DOE SNF canister.

4. THERMAL ANALYSIS

4.1 USE OF COMPUTER SOFTWARE

The FEA computer code used to analyze the (Fermi U-Mo alloy) SNF canister within the 5-DHLW DOE SNF waste package is ANSYS Version (V) 5.4. This code is identified as CSCI 30040 V5.4 and is obtained from SCM in accordance with appropriate procedures. ANSYS is a commercially available finite-element thermal- and mechanical-analysis code. ANSYS V5.4 software is qualified as documented in the software qualification report for ANSYS V5.4 (CRWMS M&O 1998c). ANSYS V5.4 is referred to herein as ANSYS.

4.2 THERMAL DESIGN ANALYSIS

A detailed description of the finite-element representation, the method of solution, and the temperature history results at specified node locations (designated by numbers 1-70) within the finite-element representation are given in CRWMS M&O 1999a Section 6. Figures 10 and 11 give the designated node locations and numbers on each component of the finite-element representation. Details of the internals of the Fermi DOE SNF canister is shown in Appendix A of this report.

Two cases were considered. The first one is the "nominal" case with the actual heat outputs. The second one is a "bounding" case, which was obtained by applying a multiplier on the Savannah River HLW canister heat output. This multiplier, equal to 5.8, leads to the temperature that is close to the maximum acceptable temperature of the HLW (400°C), in compliance with the SDD (Section 2.2.2).

Two sizes of iron shot were considered, Grades S230 and S330. The S330 thermal conductivity was used, which is most conservative since it is the lower of the two. The average of S230 and S330 densities was used in the calculations. A vibrated or settled condition, which may be expected to occur while transporting the WP, is also assumed to obtain the iron shot density value.

The WP is assumed to be evacuated, then filled with helium gas. Representing only conduction and radiation heat transfer inside the WP provides conservative temperature results for the calculation. The fill gas placed inside the WP will allow a natural convective heat transfer to exist; however, since only a few small enclosed basket cavities exist and the temperature gradient in the enclosure is very small, the heat transferred by helium circulation is insignificant.

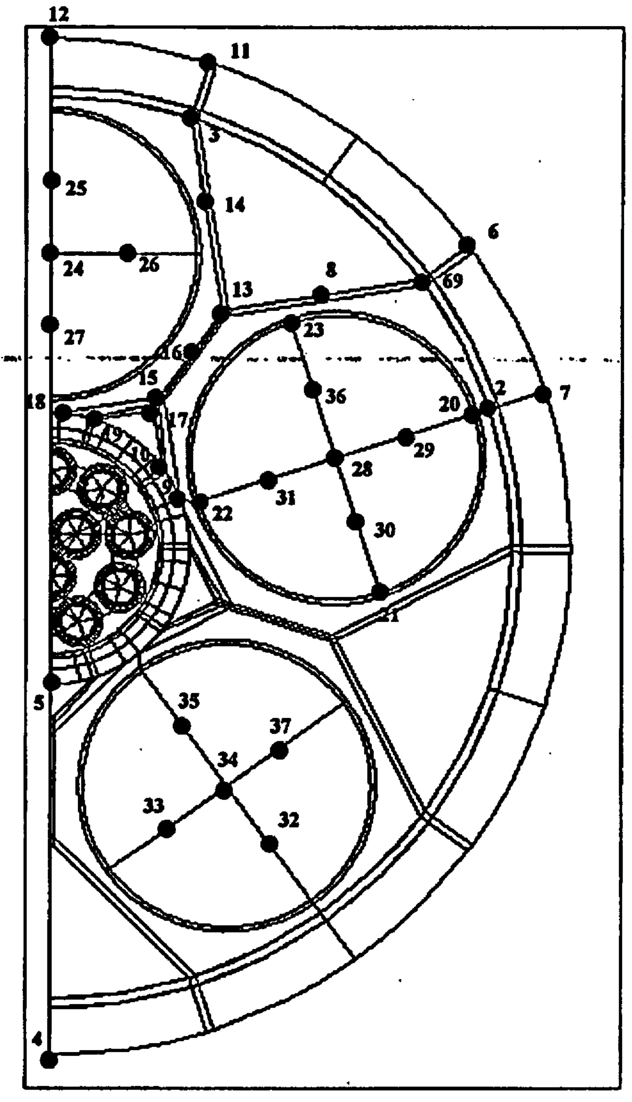

Figure 10. Node Locations and Numbers of the Finite-Element Representation of Waste Package with Fermi SNF Canister and the HLW Canisters

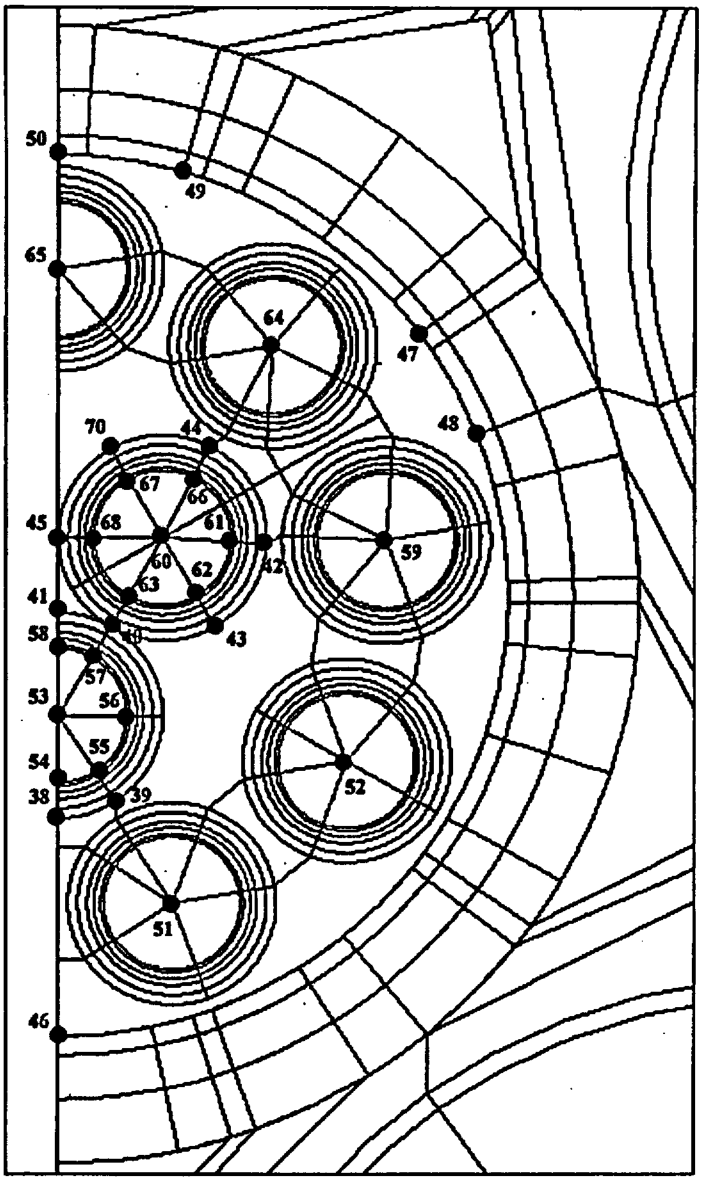

Figure 11. Node Locations and Numbers of the Finite-Element Representation of Fermi SNF Canister

4.3 CALCULATIONS AND RESULTS

The results provided in this section were derived from the ANSYS V5.4 calculations. Table 19 lists the physical location of the most important nodes shown in Figures 10 and 11.

Table 19. Physical Locations of Nodes of Interest

Node Number	Physical Location
12	WP outer surface top
35	HLW glass (maximum temperature)
53	4-in.-diameter pipe center with fuel

Figure 12 shows the peak surface temperature of the waste package and the peak fuel temperatures calculated for each case ("nominal" case and bounding case). (The temperature distribution in the waste package at the time of peak fuel temperature can be found in CRWMS M&O 1999a, Tables 6-1 through 6-10.) Table 20 summarizes the Enrico Fermi peak fuel temperatures and time of occurrence for each case.

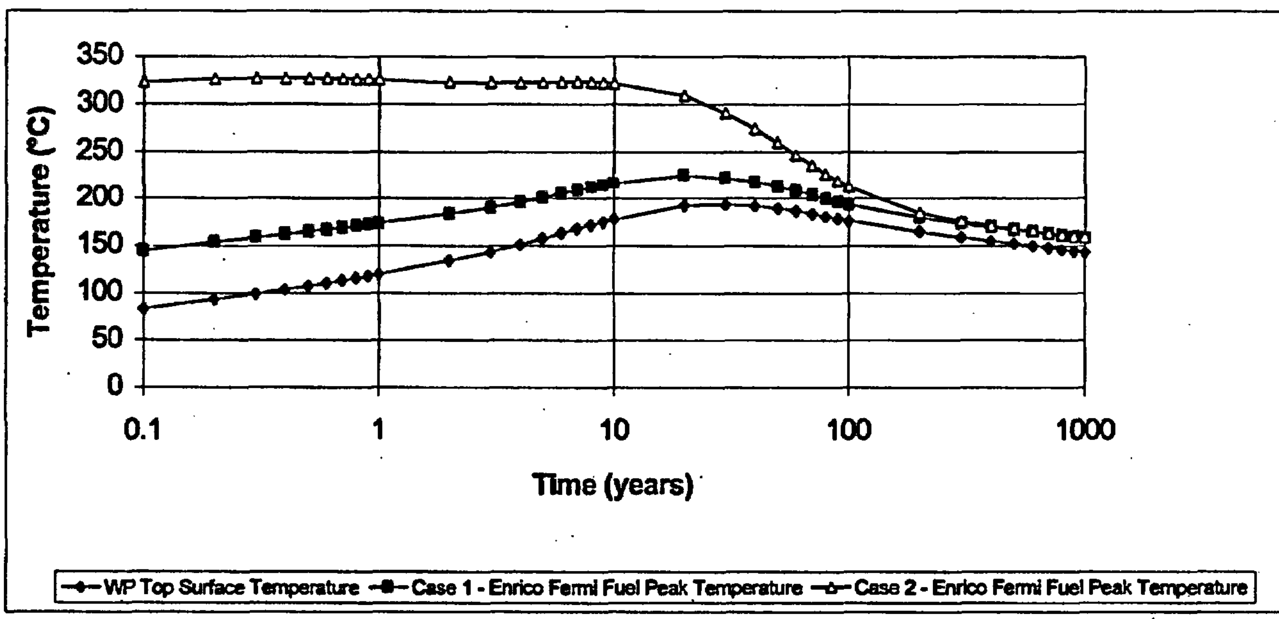

Figure 12. Temperature History for Enrico Fermi Codisposal Waste Package (case 1: nominal, case 2: bounding)

Table 20. Enrico Fermi Peak Fuel Temperatures and Time of Occurrence

Case	Peak Fuel Temperature (°C)	Time of Occurrence (yr)
1 - Nominal	225.0	20
2 - Bounding	327.2	0.4

4.4 SUMMARY

The results indicate that the maximum fuel and HLW glass temperatures for the bounding case are 327.2 °C (Node 53) and 394.5 °C (Node 35), respectively, which are below the SDD criteria discussed in Section 2.2.2.

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5. SHIELDING ANALYSIS

5.1 USE OF COMPUTER SOFTWARE

The Monte Carlo particle transport code, MCNP, Version 4B2, is used to calculate average dose rates on the surface of waste package. This code, which is identified as CSCI 30033 V4B2LV, was previously obtained from the SCM in accordance with appropriate procedures. The qualification of the MCNP software, including problems of the type analyzed in this report, is summarized in the SQR for the MCNP Version 4B2 (CRWMS M&O 1998e). The dose rate calculations on, and near, the surface of a waste package containing HLW and Fermi fuel are fully within the range of the validation performed for the MCNP computer code.

5.2 SHIELDING DESIGN ANALYSIS

The Monte Carlo method for solving the integral transport equation, which is implemented in the MCNP computer program, is used to calculate radiation dose rate for the waste package. MCNP uses continuous-energy neutron and photon cross-sections processed from the evaluated-nuclear-data files, ENDF (Briesmeister 1997, pp. 2-17 through 2-22 and App. G). The flux averaged over a surface is tallied, and the flux-to-dose rate conversion factors (Briesmeister 1997, App. H) are applied to obtain surface-dose rates for gamma and neutron radiation.

5.3 CALCULATIONS AND RESULTS

CRWMS M&O 1999b gives the details of the calculations and the results. The geometric model used in MCNP calculations is shown in Figure 13. Previous dose-rate calculations for the waste package containing only SRS HLW glass canisters show that the angular dose rate over waste package radial surfaces is uniform (CRWMS M&O 1998n, p. 33). Therefore, only axial variation of the dose rate on the waste-package radial surfaces and the radial variation of the dose rate on the waste-package top and bottom axial surfaces are studied. Figure 14 shows the surfaces and segments that are used in the dose-rate calculations. The radial surface, between the bottom and top planes of HLW glass, are equally divided into five segments, each of which is 43.2253-cm high. The first radial segment (segment 1), 87.8735-cm high, corresponds to the empty portion of the HLW canister, which is between the top of the waste package cavity and the top of the HLW glass. The waste package top and bottom axial surfaces are divided into two radial segments of 0-30 cm and 30-106 cm.

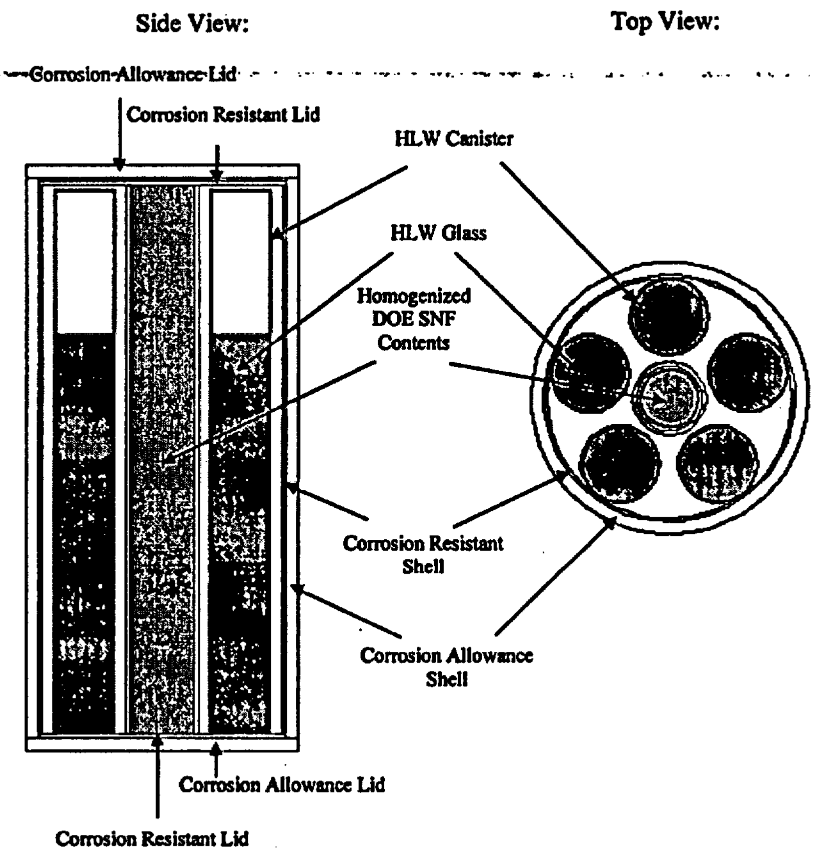

Figure 13. Vertical and Horizontal Cross Sections of MCNP Geometry Representation

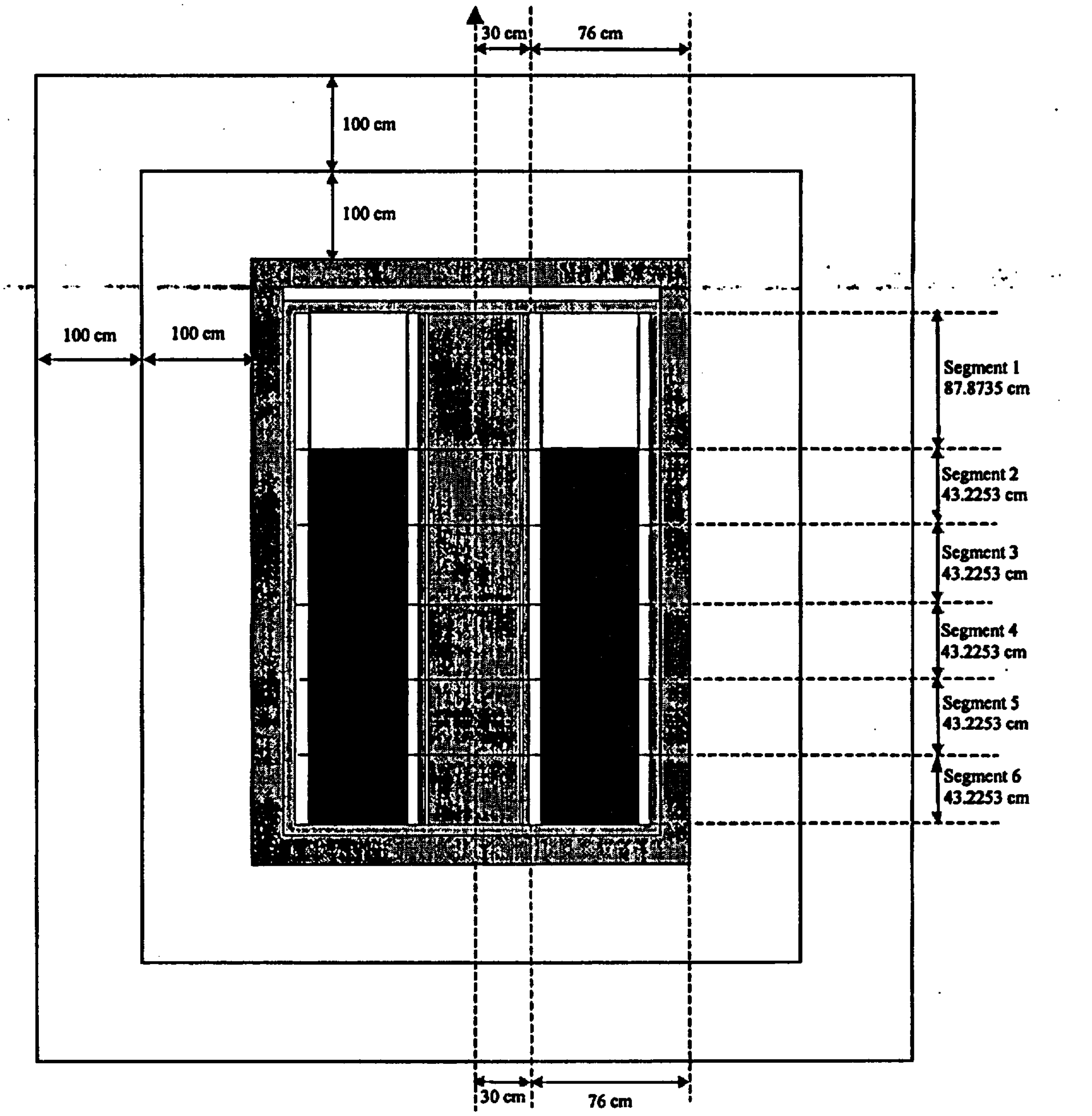

Figure 14. Surfaces and Segments Used for Dose-Rate Calculations

Tables 21 through 23 are lists of the radial and axial dose rates on the outer surface of the waste package containing the five SRS HLW glass canister and the Fermi DOE SNF canisters. The dose rates in rem/h and rad/h are practically the same due to the insignificant contribution of the neutron dose rate to the total dose rate (CRWMS M&O 1999b, pp. 21-24, and Attachment V).

Table 21. Radial Dose Rates Averaged over WP Outer-Radial Surface

Axial Location	Gamma Dose Rate (rem/h)	Neutron Dose Rate (rem/h)	Total Dose Rate (rem/h)
Segment 1	1.688	0.029	1.717
Segment 2	10.005	0.067	10.072
Segment 3	10.747	0.083	10.830
Segment 4	10.866	0.085	10.951
Segment 5	10.769	0.083	10.852
Segment 6	9.499	0.067	9.566

NOTE: The dose rates listed in this table are the upper limits of the 95 percent confidence intervals of the Monte Carlo dose rate calculations.

Table 22. Axial Dose Rates Averaged over a 30-cm Radius Surface

Axial Location	Gamma Dose Rate (rem/h)	Neutron Dose Rate (rem/h)	Total Dose Rate (rem/h)
Bottom surface of WP	0.986	0.061	1.047
Top surface of WP	0.477	0.020	0.497

NOTE: The dose rates listed in this table are the upper limits of the 95 percent confidence intervals of the Monte Carlo dose rate calculations.

Table 23. Axial Dose Rates Averaged over the Circular Segment Outside the 30-cm Radius

Axial Location	Gamma Dose Rate (rem/h)	Neutron Dose Rate (rem/h)	Total Dose Rate (rem/h)
Bottom surface of WP	3.153	0.052	3.205
Top surface of WP	1.121	0.020	1.141

NOTE: The dose rates listed in this table are the upper limits of the 95 percent confidence intervals of the Monte Carlo dose rate calculations.

5.4 SUMMARY

The maximum dose rate on the external surfaces of the waste package is 10.95 rem/h. It occurs on the outer radial surface at Segment 4 of the waste package. The dose rates on the bottom and top surfaces of the waste package are about one-third and about one-tenth, respectively, of the maximum dose rate on the outer radial surface. The design criterion specifies the maximum dose rate at all external surfaces of the waste package is TBD (TBD-3764, Section 2.2.3).

6. DEGRADATION AND GEOCHEMISTRY ANALYSIS

6.1 USE OF COMPUTER SOFTWARE

The EQ3/6 geochemistry software package originated in the mid-1970s at Northwestern University (Wolery 1992). Since 1978, Lawrence Livermore National Laboratory has been responsible for maintaining the EQ3/6 software, most recently under the sponsorship of the Civilian Radioactive Waste Management (CRWM) Program of the DOE. The EQ3/6 software package contains several computer codes, databases, and example problems. The major components of the EQ3/6 package include the following: EQ3NR (the most utilized code) – a speciation-solubility code that is also required for initializing any runs for EQ6; EQ6 – a reaction path code that calculates water/rock interaction or fluid mixing; EQPT – a data-file preprocessor; EQLIB – a supporting software library; and several supporting thermodynamic data files. The software implements algorithms describing thermodynamic equilibrium, thermodynamic disequilibrium, and reaction kinetics. The supporting data files contain both standard-state and activity-coefficient-related data.

EQ6 calculates the irreversible reactions that occur between an aqueous solution and a set of solid, liquid, or gaseous reactants. The code can calculate fluid mixing and the consequences of changes in temperature. This code was operated in the transient mode, rather than the equilibrium mode, in order to more accurately model the degradation processes.

In this study, EQ3/6 is used to provide:

- A general overview of the nature of chemical reactions to be expected
- The degradation products likely to result from corrosion of the waste forms and canisters
- An indication of the minerals, and their amounts, likely to precipitate within the waste package.

The EQ3/6 calculations reported in this document used Version 7.2B of the code, which is appropriate for the application, and were executed on Pentium series personal computer (PCs). The EQ3/6 package has been verified by its present custodian, Lawrence Livermore National Laboratory. The source codes were obtained from SCM in accordance with the Office of Civilian Radioactive Waste Management procedure AP-SI.1Q, *Software Management*. The code was installed on Pentium PCs according to a Management and Operating Contractor-approved Installation and Test procedure (CRWMS M&O 1998h). The EQ3/6 Version 7.2B is qualified as documented in the SQR for EQ3/6 V7.2B (CRWMS M&O 1998q). EQ3/6 V7.2B is referred to herein as EQ6.

6.2 DESIGN ANALYSIS

6.2.1 Systematic Investigation of Degradation Scenarios and Configurations

Degradation scenarios comprise a combination of features, events, and processes that result in degraded configurations to be evaluated for criticality. A configuration is defined by a set of parameters characterizing the amount and physical arrangement, at a specific location, of the materials that can significantly affect criticality (e.g., fissile materials, neutron-absorbing materials, reflecting materials, and moderators). The variety of possible configurations is best understood by grouping them into classes. A configuration class is a set of similar configurations whose composition and geometry is defined by specific parameters that distinguish one class from another. Within a configuration class the values of configuration parameters may vary over a given range.

A master scenario list and set of configuration classes relating to internal criticality is given in the *Disposal Criticality Analysis Methodology Topical Report* (YMP 1998, pp. 3-1 through 3-13) and also shown in Figures 15 and 16. This list was developed by a process that involved workshops and peer review. The comprehensive evaluation of disposal criticality for any waste form must include variations of the standard scenarios and configurations to ensure that no credible degradation scenario is neglected. All of the scenarios that can lead to criticality begin with the breaching of the waste package, followed by entry of the water, which eventually leads to degradation of the SNF and/or other internal components of the waste package. This degradation may permit neutron absorber material to be mobilized (made soluble) and either be flushed out of the waste package or separated from the fissile material, thereby increasing the probability of criticality.

The standard scenarios for internal criticality divide into two groups:

1. When the waste package is breached only on the top, water flowing into the waste package collects and fills the waste package. This ponding provides water for moderation to potentially increase the probability of criticality. Further, after a few hundred years of steady dripping, the water can overflow through the hole on the top of the waste package and flush out any dissolved degradation products.

2. When the waste package breach occurs on the bottom as well as the top, the water can flow through the waste package. This group of scenarios allows the soluble degradation products to be removed more quickly, but does not directly provide water for moderation. Criticality is possible, however, if the waste package fills with corrosion products that can add water of hydration and/or plug any holes in the bottom of the waste package while fissile material is retained and absorbers are removed or separated. Silica released by the degrading HLW glass may form clay with enough water of hydration to support criticality.

The standard scenarios for the first group shown in Figure 15, which have the waste package breached only at the top, are designated IP-1, -2, and -3 (IP stands for internal to the package) according to whether the waste form degrades before the other waste package internal components, at approximately the same time (but not necessarily at the same rate), or later than the waste package internal components. The standard scenarios for the second group shown in Figure 16, which have the waste package breached at both the top and the bottom, are designated IP-4, -5, or -6 based on the same criteria. The internal criticality configurations resulting from these scenarios fall into six configuration classes described below (YMP 1998, pp. 3-10 through 3-12):

1. Basket is degraded but waste form is relatively intact and sits on the bottom of the waste package (or the DOE SNF canister), surrounded by, and/or beneath, the basket corrosion products (see Figure 17). This configuration class is reached from scenario IP-3.

2. Both basket and waste form are degraded (see Figure 18). The composition of the corrosion product is a mixture of fissile material and iron oxides, and may contain clay. It is more complex than for configuration class 1, and is determined by geochemical calculations as described in Section 6.3. This configuration class is most directly reached from standard scenario IP-2, in which all the waste package components degrade at the same time. However, after many tens of thousands of years the scenarios IP-1 and IP-3, in which the waste form degrades before or after the other components, also lead to this configuration.

3. Fissile material is moved some distance from the neutron absorber, but both remain in the waste package (see Figure 19). This configuration class can be reached from IP-1.

4. Fissile material accumulates at the bottom of the waste package, together with moderator provided by water trapped in clay (see Figure 20). The clay composition is determined by geochemical calculation, as described in Section 6.3. This configuration class can be reached by any of the scenarios, although IP-2 and IP-5 lead to this configuration by the most direct path; the only requirement is that there be a large amount of glass in the waste package (as in the codisposal waste package) to form the clay.

5. Fissile material is incorporated into the clay, similar to configuration class 4, but with the fissile material not at the bottom of the waste package (see Figure 21). Generally the mixture is spread throughout most of the waste package volume, but could vary in composition so that the fissile material is confined to one or more layers within the clay. Generally, the variations of this configuration are less reactive than for configuration class 4, therefore, they are grouped together, rather than separated according to where the fissile layer occurs or whether the mixture is entirely homogeneous. This configuration class can be reached by either standard scenario IP-1 or IP-4.

6. Fissile material is degraded and spread into a more reactive configuration but not necessarily moved away from the neutron absorber, as in configuration class 3 (see Figure 22). This configuration class can be reached by scenario IP-1.

The configuration classes 1, 2, 4, and 5 require that most of the neutron absorber be removed from the waste package. However, in configuration classes 3 and 6, the fissile material is simply moved away from the absorber or into a more reactive geometry.

Note that most of these configurations or configuration pairs (Figures 17 through 22) look quite different even though both pair members belong to the same configuration class. This apparent dissimilarity arises from the configuration class definition strategy, which classifies critical configurations according to the geometry and composition of the materials, irrespective of the container (either the DOE SNF canister, or the entire waste package).

In Sections 6.2.1.1 through 6.2.1.6, the scenarios and the resulting configuration classes that are applicable to the 5-DHLW/DOE Spent Fuel-Long waste package with Enrico Fermi fuel in the DOE SNF canister are discussed. The naming convention used for the standard scenarios in Sections 6.2.1.1 through 6.2.1.6 is slightly different from the convention used in the topical report (YMP 1998), which is shown in Figures 15 and 16. The naming convention used in these sections contains refinements to the configurations described in the topical report based on CRWMS M&O 1999g.

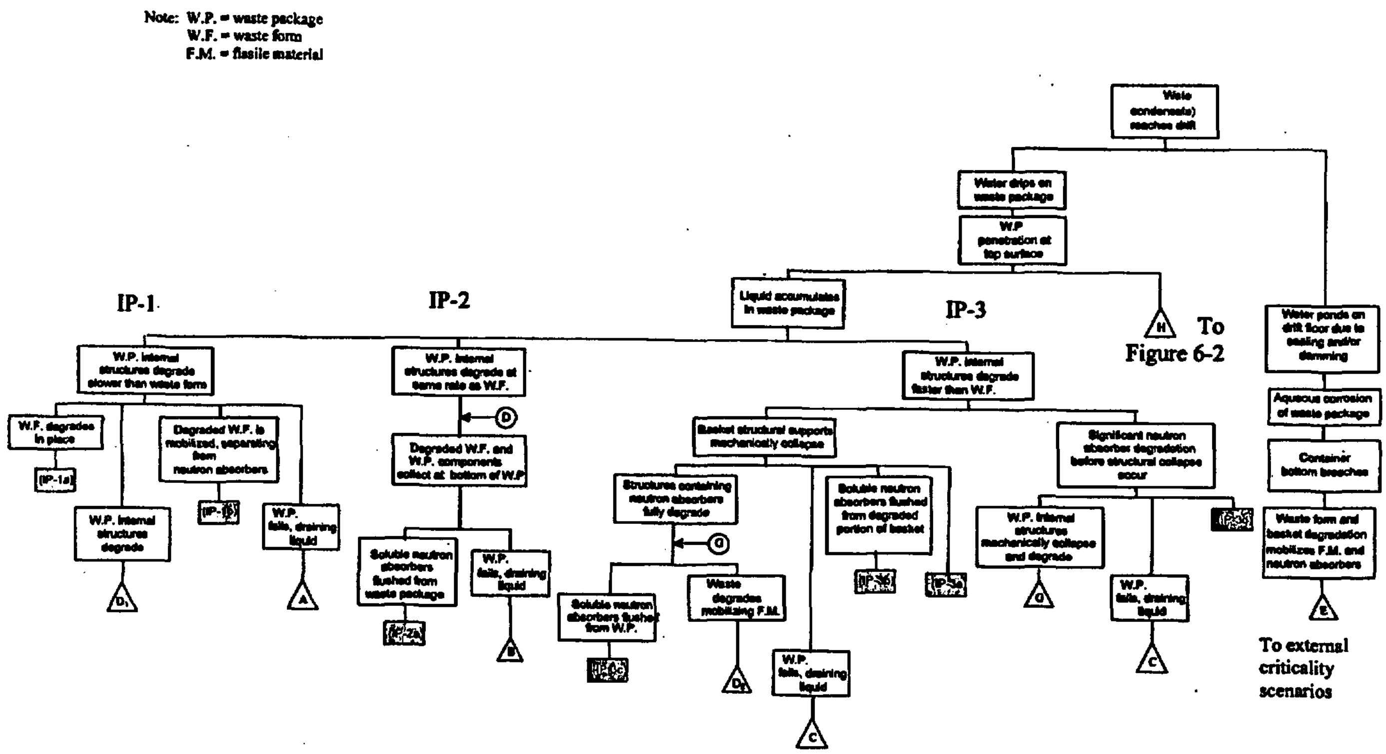

Figure 15. Internal Criticality Master Scenarios, Part 1 (YMP 1998)

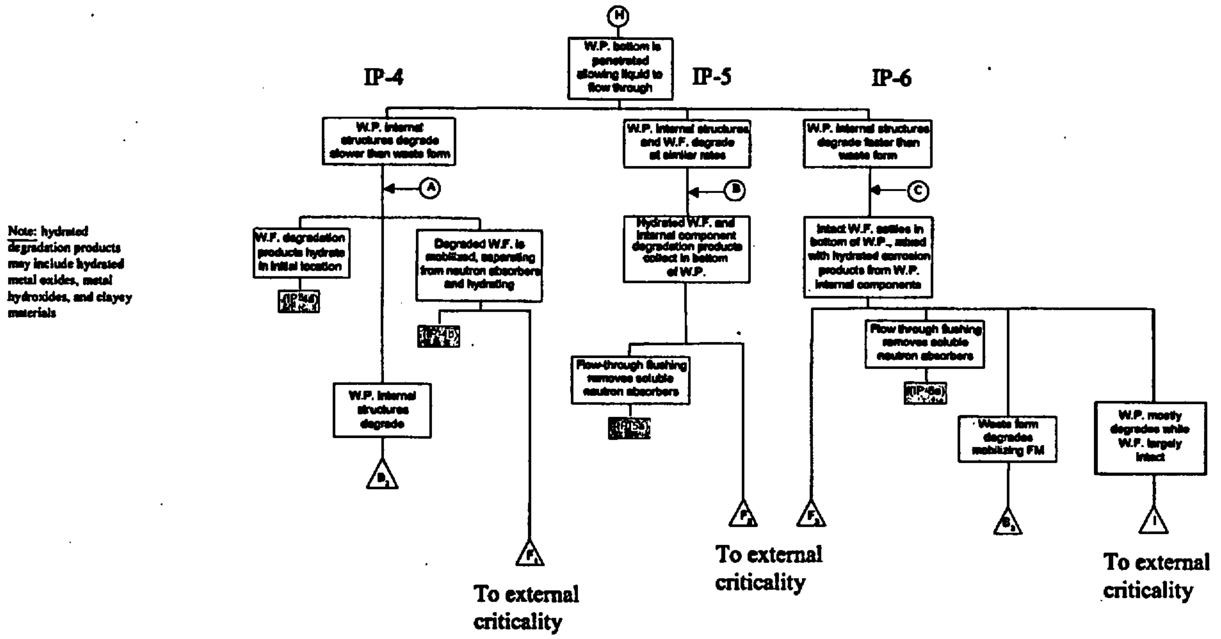

Figure 16. Internal Criticality Master Scenarios, Part 2 (YMP 1998)

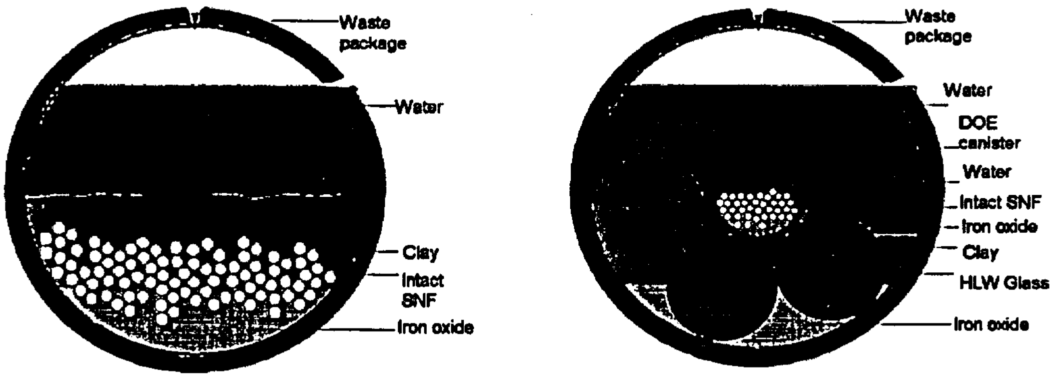

Figure 17. Examples of Degraded Configurations from Class 1

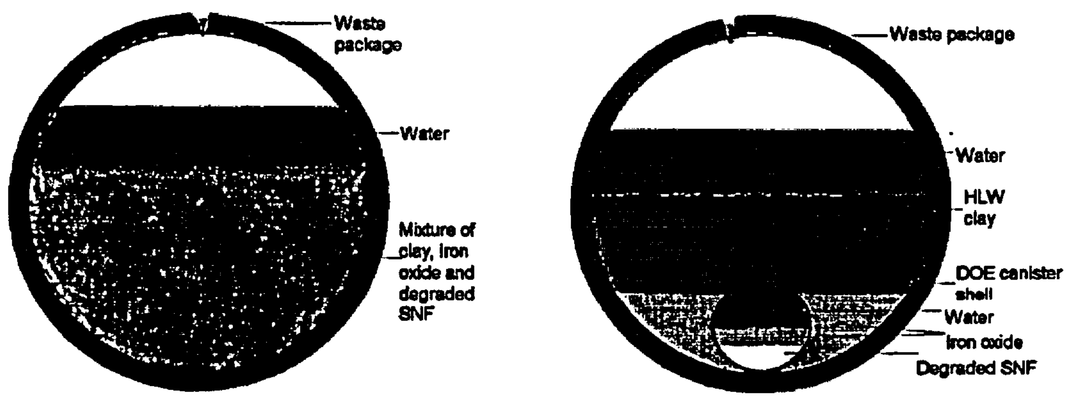

Figure 18. Examples of Degraded Configurations from Class 2

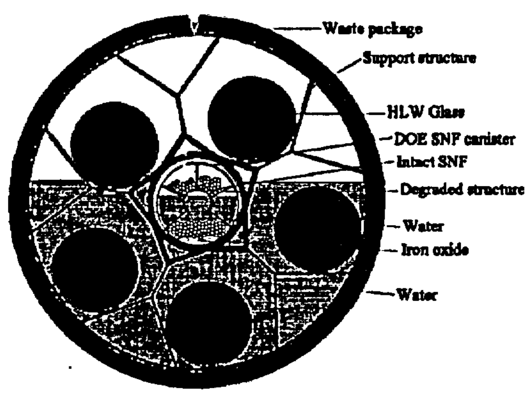

Figure 19. Example of Degraded Configuration from Class 3

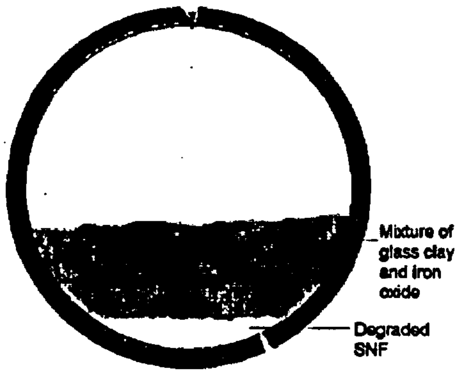

Figure 20. Example of Degraded Configuration from Class 4

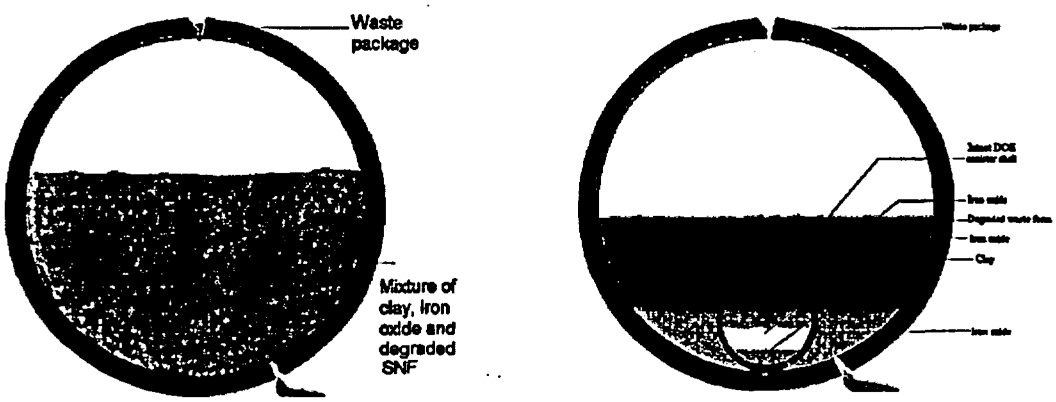

Figure 21. Examples of Degraded Configurations from Class 5

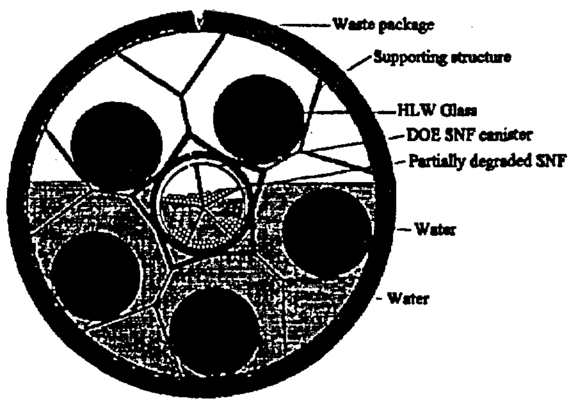

Figure 22. Example of Degraded Configuration from Class 6

The report titled *Generic Degradation Scenario and Configuration Analysis for the DOE Codisposal Waste Package* (CRWMS M&O 1999g) serves as the basis for the specific degraded waste package criticality analysis to be performed for any type of DOE spent nuclear fuel that will be codisposed with the HLW in a codisposal waste package. Starting from these guidelines, a set of degradation scenarios and resultant configurations has been developed for the codisposal waste package containing Enrico Fermi SNF. The following brief description focuses on the correspondence between both different classes of configurations and their refinements. This approach allows a systematic treatment of the degraded internal criticality analysis, taking into account all possible configurations with potential for internal criticality.

The characteristics of both the Enrico Fermi SNF and the DOE SNF canister are conservatively taken into account in the present analysis. The analysis provides the basis for evaluating the required amount of neutron absorber ($GdPO_4$) that has to be distributed inside the DOE canister to keep the system's effective multiplication factor (k_{eff}) below the critical limit. This approach considers bounding arrangements and compositions of the configurations resulting from the internal degradation of the waste package. Parametric studies are subsequently performed for identifying the most reactive configurations. The ability to specify the most reactive credible configuration allows evaluation of the neutron absorber concentration that would bring k_{eff} below the critical limit. Supplementary calculations are performed to verify the effectiveness of the neutron absorber under different moderation regimes.

6.2.1.1 Most Probable Scenario for Enrico Fermi SNF

The parameters that need to be considered to develop the most probable degradation scenario/configuration for the Enrico Fermi SNF are: the materials of the components associated with the waste package; the DOE SNF canister and the SNF; and thickness of the materials and the associated corrosion rates. The sequence of degradation can be developed, and the most probable degradation scenario/configuration can be identified by using these parameters, which are discussed below.

Corrosion Rates–The material corrosion rates are presented in Section 2.1.9 of this report. Stainless steels Type 304L and Type 316L degrade at approximately the same rate. The A 516 carbon steel degrades faster than either Type 304L or Type 316L. The composition of the iron shot is very similar to A 516 carbon steel (CRWMS M&O 1996); thus the corrosion rate of the iron shot would be the same as A 516.

Most Probable Degradation Path–Based on the material corrosion rates and the material thickness given in Table 24 below, the most probable degradation path for the waste package, the DOE SNF canister, and the Enrico Fermi SNF follows the following sequence:

1. Waste package is penetrated and flooded internally. The waste package basket (outer and inner brackets and support pipe) degrades first, because of the high corrosion rate for A 516 carbon steel.

2. HLW glass canister's stainless steel shell and glass begin to degrade. After this, there are two degradation paths:

2a. DOE SNF canister stays intact. Intact DOE SNF canister and intact SNF assembly fall on top of degraded products near the bottom of the waste package.

2b. DOE SNF canister starts to degrade.

3. DOE SNF canister is penetrated and flooded.

4. Components internal to the DOE canister are in contact with water. These components include the SNF basket structure (4-in. pipes, lifting rod, dividers, base plates), -01 and -04 aluminum canisters, and iron shot.

5. The aluminum canisters start to degrade at a rate faster than all other components.

6. Iron shot starts to degrade.

7. Degraded aluminum product mixes with other degraded steel materials locally.

8. SNF is exposed to water. After this, there are two paths:

8a. All SNF stays intact. Iron shot degrades in place and mixes with other degraded products. As a result, SNF and neutron absorber stay in place (Section 6.2.1.3). The initial void space present inside DOE SNF canister is insufficient to allow complete degradation of the internal constituents. Thus, partial degradation would prevent water flow and further degradation until canister walls are degraded to allow mixing inside the WP.

8b. All SNF degrades. The degraded SNF mixes with other degraded products and settles at the bottom of the DOE canister.

9. After sequence 8 above, there are two paths:

9a. DOE SNF canister degrades, SNF stays intact. Intact SNF falls and scatters on top of other degraded products near the bottom of the waste package. There could be some separation between the fissile material and neutron absorber (Section 6.2.1.4).

9b. DOE SNF canister and SNF degrade. Degraded SNF mixes with other degraded products and settle near the bottom of the waste package. There could be some separation between the fissile material and neutron absorber (Section 6.2.1.5).

10. Given a very long period of time, it is postulated that everything will degrade. This corresponds to the degradation scenario group IP-2 (Section 6.2.1.6) (YMP 1998). To bound the potential degraded cases, degradation of the SNF and other degradation products are assumed to mix to some degree and pile up near the bottom

of waste package. Even though there is no mechanism to cause uniform mixing of all the degradation products inside the waste package, it is considered to bound the configurations.

Table 24. Materials and Thicknesses

Components	Material	Thickness (mm)
Waste package outer bracket	A 516 carbon steel	12.7
Waste package inner bracket	A 516 carbon steel	25.4
Waste package support pipe	A 516 carbon steel	31.75
HLW glass shell	304L stainless steel	9.5
HLW glass	Glass	N/A
DOE SNF canister	316L stainless steel	9.5
DOE SNF canister basket:		
Lifting rod	316L stainless steel	25.4
4-in. pipe	316L stainless steel	4.8
Dividers A and B	316L stainless steel	9.5
Base plate	316L stainless steel	9.5
-01 shipping canister	Aluminum	3.175
-04 inner canister	Aluminum	1.651

SOURCES: CRWMS M&O 1999m and DOE 1999.

Most Probable Degradation Scenario/Configuration–Based on *Generic Degradation Scenario and Configuration Analysis for DOE Codisposal Waste Package* (CRWMS M&O 1999g), the above degradation sequences match with the degradation scenario/configurations of IP-3-A to IP-3-C (equivalent to IP-2). The details of these degradation scenario/configurations are discussed in Sections 7.4.2, 7.5.2, and 7.5.3. The most probable degradation configuration is the one with the degradation of all components inside the waste package and the DOE SNF canister. The SNF pins may stay intact but likely will degrade. The degradation scenario of IP-1, i.e., SNF degrades faster than all of the other materials, is not probable because the SNF corrosion rate is much lower than 304L or 316L SS.

The configurations described in Sections 6.2.1.3 through 6.2.1.6 are the most likely configurations, whereas the configuration discussed in Sections 6.2.1.2 is not likely.

6.2.1.2 Total Degradation of the SNF Inside Non-Degraded DOE Canister

A typical configuration for this class (Class 6 in CRWMS M&O 1999g) is a configuration characterized by a homogeneous mixture of goethite (FeOOH)/diaspore (AlOOH)/degraded fuel inside the intact 4-in.-diameter pipes. The 4-in. pipes are in their initial locations with the iron shot remaining in place. This configuration is very unlikely due to the high corrosion resistance of the zirconium cladding. The results of the criticality calculations for this configuration are given in Section 7.4.3.

6.2.1.3 Intact Fuel Pins in Partially Degraded DOE Canister

These configurations comprise the intact fuel pins distributed inside the DOE canister at various stages of degradation of the internal supporting structure. They represent refinements of the configuration Class 1 that is derived from the standard scenario group IP-3, and are described in CRWMS M&O 1999g under the configuration group IP-3-A. Also, the cases with the partially degraded support structure inside the DOE canister are refinements of the configuration Class 3 resulting from standard scenario group IP-1. For Fermi SNF, the 4-in.-diameter steel pipes, which are welded to a base plate to maintain the spacing, represent the fuel supporting structure inside the DOE SNF canister.

Different stages of degradation of the supporting structure have been considered. First, rearrangements of the 4-in. pipes inside the DOE canister are investigated (see criticality results in Section 7.4.1). Finally, a bounding configuration of an array with the intact fuel pins inside a DOE SNF canister filled with wet goethite (FeOOH) and diaspore (AlOOH) is analyzed at various pitches, to identify the most reactive configuration. The rest of the waste package (outside the DOE SNF canister) is considered filled with a wet clayey material obtained from the degradation of the HLW glass and the supporting structure. The results of the criticality calculations for this configuration are given in Section 7.4.2.

6.2.1.4 Degraded WP Internal Components with Non-Degraded Fuel Pins

These configurations result from the subsequent degradation stage of the configuration discussed in Section 6.2.1.3 and represent a refinement of the standard configuration Class 1 in CRWMS M&O 1999g (configuration group IP-3-B). A bounding arrangement is also selected for this analysis. A hypothetical arrangement of the intact fuel pins in a regular array is placed at the bottom of the waste package filled with clayey material and water. The fuel pins are settled or piled up to form a stack of fuel pins. The results of the criticality calculations for this configuration are given in Section 7.5.1.

6.2.1.5 Degraded DOE Canister Internals and Fuel Pins

This category includes the waste package configurations that are refinements of the configuration Class 2. They are obtained via any of the standard scenario groups from IP-1 to IP-3. The refinements are described in CRWMS M&O 1999g as variations of the configuration refinement IP-1-C. The DOE SNF canister outer shell still keeps the degraded mixture with the fissile material from being dispersed in the volume of the HLW clayey material in the waste package. The refinements include different locations of the DOE SNF canister within the homogeneous wet clayey material. The results of the criticality calculations for this configuration are given in Section 7.4.4.

6.2.1.6 Degraded DOE Canister and WP Internals

In this case, the whole content of the waste package is considered degraded and settled at the bottom of the waste package. The standard configuration class from YMP 1998 is Class 2, but the refinements include a large number of possible configurations (see IP-1-C, IP-2, and IP-3-C from CRWMS M&O 1999g). The approach adopted for analyzing these configurations includes first a screening of the various bounding arrangements of the degraded materials. In the subsequent steps, the actual composition of the degraded mixture is taken into account. This approach covers, in a systematic way, the spectrum of possible configurations from this class. The results of these cases are presented in detail in Section 7.5.2.

The possible final stage of the WP internal degradation is evaluated in Section 7.5.3 as a configuration comprising a homogeneous mixture of clay and water. The composition of the clay is given by geochemistry calculations.

6.2.2 Basic Design Approach for Geochemistry Analysis

The method used for this analysis involves eight steps as described below:

1. Use the basic EQ3/6 capability to trace the progress of reactions as the chemistry evolves, including estimating the concentrations of material remaining in solution as well as the composition of precipitated solids. EQ3 is used to determine a starting fluid composition for EQ6 calculations; it does not simulate reaction progress.

2. Evaluate available data on the range of dissolution rates for the materials involved, to be used as material/species input for each time step.

3. Use the "solid-centered flow-through" mode in EQ6. In this mode, an increment of aqueous "feed" solution is added continuously to the waste-package system, and a like volume of the existing solution is removed. This mode simulates a continuously stirred tank reactor.

4. Determine the fissile concentrations of fissile materials in solution as a function of time (from the output of EQ6-simulated reaction times up to $6 \cdot 10^5$ years).

5. Calculate the amount of fissile material released from the waste package as a function of time (which thereby reduces the chance of criticality within the waste package).

6. Determine the concentrations of neutron absorber material, such as gadolinium (Gd), in solution as a function of time (from the output of EQ6 over time up to $6 \cdot 10^5$ years).

7. Calculate the amount of neutron absorber material retained within the waste package as a function of time.

8. Calculate the composition and amounts of solids (precipitated minerals or corrosion products and unreacted package materials).

6.3 CALCULATIONS AND RESULTS

The calculations begin using selected representative values from known ranges for composition, amounts, and reaction rates of the various components of the waste package. Surface areas are calculated based on the initial package geometry. The input to EQ6 consists of the composition of J-13 well water, together with a rate of influx to the waste package (Section 2.1.9.3). In some cases, the degradation of the waste package is divided into stages (e.g., degradation of HLW glass before breach of the SNF canister and exposure of the fuel and its basket material to the water). The EQ6 outputs include the compositions and amounts of solid products and the solution composition. Summary of the results are presented in Section 6.3.1 below. The calculation process is described in more detail in CRWMS M&O 1999m.

6.3.1 Results of EQ6 Runs

Table 25 summarizes the conditions used for the EQ6 runs and the total percentage of Gd and U remaining at the end of runs. If fissile material remains behind in the waste package while the Gd and other neutron absorbers are flushed from the system, an internal criticality could be possible. A solubility scoping calculation revealed that total concentration of dissolved Gd phosphate complexes as a function of pH and total dissolved phosphate never exceed concentrations much greater than 10^{-9} molal for pH values between 4 and 9, and thus would not result in significant Gd loss from the system (CRWMS M&O 1998p). Two basic types of degradation scenarios are simulated and are described below.

Cases 1-9 are single stage scenarios involving simultaneous exposure of the fuel and the package materials to groundwater. These cases are designed to maximize exposure of the Gd-doped iron shot to high pH, and to stress the enhanced solubility of $GdPO_4$ under alkaline conditions. Considering that fuel pins are contained within zirconium cladding, for a conservative approach, it was assumed that cladding is fully breached immediately after contact with water.

The single-stage cases produced insignificant Gd loss; the total loss was $\leq 2.3\%$ in $\geq 2.5 \times 10^5$ years. Furthermore, when the HLW glass was allowed to degrade rapidly, the alkaline conditions produced high U loss (Table 25, Cases 3 through 9), reducing the chances of internal criticality. Some of these "alkaline" cases actually produce short-lived, very low pH (~3) when glass corrosion rates were set to low values and steel corrosion rates are set to high values. These low-pH values may not be realistic, since the glass corrosion model does not allow a feedback between pH and corrosion rate (which would tend to increase pH).

Cases 10 through 18 were the multiple-stage cases to test the effect of exposing the Gd and U to long-lived acidic conditions (pH ~5 to 6). The first-stage of EQ6 simulations achieved the highest alkalinity. In the first stage, it is assumed that the DOE SNF canister is intact, and only the HLW glass and its container, the A 516 outer web structure, and the outside surface of the DOE SNF canister are allowed to interact with the water dripping into the package. With a sufficiently high drip rate, the alkaline components of the glass are removed during this stage. In the second stage, the $GdPO_4$ and iron shot, fuel, and other components within the DOE SNF canister are exposed to J-13 well water at a much lower drip rate, allowing pH to drop. When hematite formation is suppressed (in favor of goethite), somewhat lower pH is achieved. Cases 10 through 18 resulted in no significant loss of Gd, but a few percent loss of U (Table 25). In all cases, the predicted major corrosion products are: a Fe-rich smectite clay (nontronite); hematite or goethite; pyrolusite; rutile; and Ni_2SiO_4 or $NiFe_2O_4$. The smectite and Fe oxide typically comprise over 90% of the corrosion product volume. The Gd enters into rhabdophane (hydrated $GdPO_4$) as the iron shot corrodes, and the dominant U solid is soddyite ($(UO_2)_2(SiO_4)\cdot 2H_2O$). The detail of each run is explained in CRWMS M&O 1999m.

In most EQ6 runs, minerals that generally do not form at low temperature are suppressed; such mineral phases include, but are not limited to, muscovite and mica, which are thermodynamically stable, but kinetically inhibited relative to clays. A complete list of such minerals is contained in each EQ6-input file. In some cases, hematite (the thermodynamically more stable phase of iron oxide) is suppressed to cause formation of goethite; both of these phases are actually present in rust.

Table 25. Summary of Geochemistry Results

Case	% Loss at End of Run		Rates				Fe Oxide
	U	Gd	Steel	Glass	Fuel	J-13[a]	
1	0.28	0.06	1	1	1	4	Hematite
2	13.97	0.47	1	1	1	3	Hematite
3	92.57	0.50	1	1	1	1	Hematite
4	91.92	0.00	1	2	1	1	Hematite
5	100.00	0.00	2	2	1	1	Hematite
6	92.60	2.30	2	1	1	1	Hematite
7	100.00	0.01	1	2	1	2	Hematite
8	61.00	0.03	2	2	1	2	Hematite
9	16.87	1.51	2	1	1	2	Hematite
10	0.04	0.03	2/1	2/0	0/1	4/2	Hematite
11	0.43	0.03	2/1	2/0	0/1	4/2	Hematite
12	4.15	0.03	2/1	2/0	0/1	4/2	Hematite
13	0.00	0.00	2/2	2/0	0/1	4/2	Hematite
14	0.00	0.00	2/2	2/0	0/1	4/2	Goethite
15	0.07	0.04	2/2	2/0	0/1	4/1	Hematite
16	0.09	0.07	1/1	2/0	0/1	3/1	Hematite
17	0.26	0.35	1/1	1/0	0/1	3/2	Hematite
18	0.47	0.65	2/1	0/1	0/1	4/2	Hematite

NOTES: [a] J-13 well water.

Rates encoding–
Steels: 1=average rate; 2=high rate (CRWMS M&O 1999m, Table 5-1)
Glass: 0=no glass present; 1=average rate; 2=high rate (CRWMS M&O 1999m, Table 5-3)
Fuel: 0=no fuel present; 1= average rate (CRWMS M&O 1999m, Table 5-4)
J-13: 1=0.0015 m^3/year; 2=0.015 m^3/year; 3=0.15 m^3/year; 4=0.5 m^3/year (CRWMS M&O 1999m, Section 5.1.1.3)
Cases 10 through 18 are multi-stage; rates are given in format: first stage/second stage.

The greatest Gd losses are in the EQ6 runs that maximize exposure of Gd to the high pH caused by the degradation of the HLW glass. Nonetheless, the maximum Gd loss is never greater than 2.3% over 100,000 years for any of the scenarios. Furthermore, some of the cases that show some Gd loss also show large losses of U.

6.4 SUMMARY

A principal objective of these calculations is to assess the chemical circumstances that could lead to removal of neutron absorbers (Gd) from the waste package, while fissile materials (U) remain behind. Such circumstances could increase the probability of a nuclear criticality occurrence within the waste package. Gadolinium is assumed to be present as $GdPO_4$ that is combined with iron shot, which is distributed within and among the 4-in.-diameter stainless steel pipes containing the fuel pins in -01 canisters. Water with the composition of J-13 well water is assumed to drip in through an opening at the top of the waste package, pooling inside and eventually overflowing, allowing removal of soluble components through continual dilution. This calculation selected 18 EQ6 cases and examined the results to identify the reasons for the chemical changes during degradation of waste package materials and flushing by J-13 well water. It appeared that, even in unusual conditions, loss of Gd was insignificant, when the element is present in the package as solid $GdPO_4$. The scenarios and conditions of EQ6 cases are chosen to emphasize the conditions that could create either acid or alkaline environments and to determine if these conditions are of sufficient duration to induce Gd loss. In all cases, the differences in the results were all very small.

These geochemistry results are valid within the scope of the criticality calculations for which they are intended, particularly as specified in Assumption 2.3.4.5. These results are not intended for input into TSPA, or to serve as a model for additional geochemistry calculations involving other configurations of waste package degradation. However, it should be noted that the results are consistent with the waste package geochemistry analysis model report (CRWMS M&O 2000c), which used the EQ3/6 geochemistry code in the same manner as was used for the present analysis, and with similar material parameters. The principal difference is that the present analysis focuses on the loss of Gd, while CRWMS M&O 2000c does not. In this regard, the validation in CRWMS M&O 2000c can be viewed as supporting this document.

7. INTACT- AND DEGRADED-MODE CRITICALITY ANALYSIS

7.1 USE OF COMPUTER SOFTWARE

The Monte Carlo computer code MCNP Version 4B2 (Briesmeister 1997) is used to estimate the effective neutron-multiplication factor (k_{eff}) of the codisposal waste package. This code, identified as CSCI 30033 V4B2LV (CRWMS M&O 1998e), was previously obtained from SCM in accordance with appropriate procedures and is qualified as documented in the SQR for the MCNP, Version 4B2 (CRWMS M&O 1998e).

7.2 DESIGN ANALYSIS

The MCNP Version 4B2 is used to estimate the k_{eff} values for various geometrical configurations of the Enrico Fermi SNF in the 5-DHLW/DOE Spent Fuel waste package. The k_{eff} results represent the average combined collision, absorption, and track-length estimator from the MCNP calculations. The standard deviation (σ) represents the standard deviation of k_{eff} related to the average combined collision, absorption, and track-length estimate due to the Monte Carlo calculation statistics. The calculations are performed using continuous energy cross-section libraries that are part of the qualified MCNP code system (CSCI 30033 V4B2LV). All calculations are performed with fresh-fuel isotopics (Assumption 2.3.5.4).

CRWMS M&O 1999d describes the Monte Carlo representations, the method of solution, and the results for nuclear criticality evaluations that were performed for intact and partially degraded "modes" of the DOE SNF canister contained in the waste package. The intact mode is defined as that mode in which no component of the DOE SNF canister is dislocated due to degradation of structural members within the canister (see Figure 23). These intact cases are described in Section 7.3.1. The partially degraded mode is treating the configurations obtained as a result of partial degradation of the DOE canister internal supporting structure (see Figures 24 and 25). These partially degraded cases are described in Section 7.4.1. The criticality analysis for the configurations obtained in the subsequent stages of internal degradation of the DOE canister components and WP internal constituents (degraded mode) is presented in CRWMS M&O 2000e and summarized in Section 7.4 and 7.5.

The MCNP results are presented in the following section in order to demonstrate that all foreseeable intact and degraded configurations inside the codisposal WP (see Section 6) have been investigated and the values of k_{eff} are below the interim criticality limit of 0.93. The minimum necessary amount of neutron absorber to fulfill the above requirement on k_{eff} for all investigated configurations is 14.5 kg of $GdPO_4$, which represents 3 vol.% of the initial volume of iron shot-$GdPO_4$ mixture placed within the DOE canister. Each of the configurations presented in Section 6 are addressed, but many are bound by results in subsequent configurations, and are not, therefore, fully parametrized.

The approach followed in the criticality calculations is focused on determining the minimum amount of neutron absorber required for a given class of configurations by identifying the limiting case in a screening analysis. For the intact and partially degraded configurations, the limiting case was obtained for a settled configuration of pipes inside the DOE canister (Fig. 25) No neutron absorber is required for the intact configurations comprising intact fuel assemblies (k_{eff} is below 0.93 without addition of $GdPO_4$). These configurations are bound by the results for degraded fuel assemblies homogenized within the 4-inch pipes which require 9.6 kg of $GdPO_4$. The overall limiting case for the degraded configurations (requiring 14.5 kg of $GdPO_4$), was an improbable case where the intact fuel pins are stacked in an array placed in the homogenized degradation products of the DOE canister.

7.3 CALCULATIONS AND RESULTS–PART I: INTACT-MODE CRITICALITY ANALYSIS

7.3.1 Intact Mode

This section presents the results of the intact-mode criticality analysis. Although the components (fuel pins, cladding, supporting pipes, and canisters) are considered structurally intact, water intrusion into the components is assumed in order to determine the highest k_{eff} resulting from optimum moderation. The contents of the waste package outside the DOE SNF canister are considered intact in all cases considered in this section.

For the intact mode, the contents of the DOE SNF canister are in an "as-welded/loaded position and condition," as depicted in Figure 23 with a typical fuel pin arrangement. The void space outside and inside the pipes, but outside the -01 canisters, is filled with iron (Fe) shot containing gadolinium phosphate ($GdPO_4$). The $GdPO_4$ is 1 percent by volume (1 vol.%) of the Fe-$GdPO_4$ mixture (approximately 4.84 kg of $GdPO_4$ in 753.1 kg of Fe).

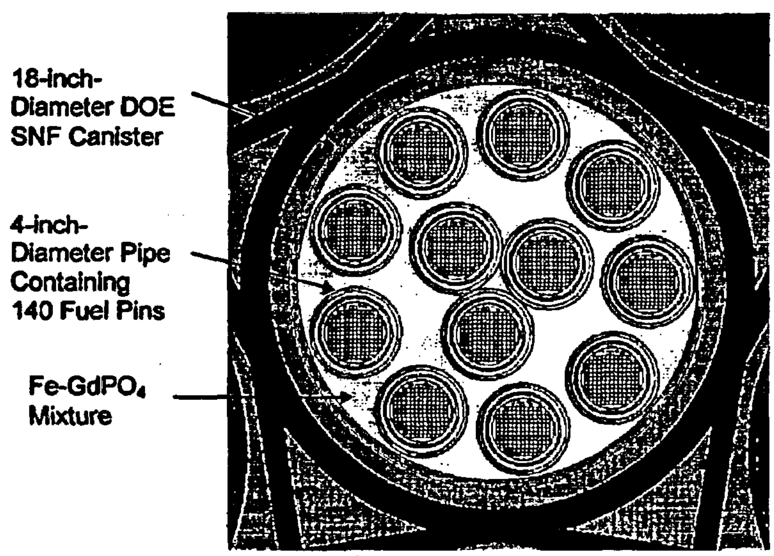

Figure 23. Cross-Sectional View of the WP Showing the Contents of the DOE SNF Canister for the Intact-Mode Analysis

The iron shot is used for moderator (water) exclusion, and the gadolinium phosphate is used as an insoluble neutron absorber. However, since the waste package is to be emplaced horizontally in the MGR, the -01 and -04 aluminum canisters inside each pipe are considered to be settled inside the pipe and the -01 canister, respectively (Figure 24 illustrates this settled configuration of the aluminum canisters).

The configurations investigated for intact-mode analysis and the results obtained are given in Table 26 (CRWMS M&O 1999d, Section 6.1). The variations of the base intact configuration included a configuration similar to the one shown in Figure 23, but with the -01 and -04 aluminum canisters contained in the three "central" pipes shifted toward the center of the DOE SNF canister. Both concentric and settled arrangements of the aluminum canisters are studied.

Water density is also varied for the base case to evaluate its impact on the k_{eff} value. The density variation simulates wetting of the shot to find optimum moderation. The rest of the waste package is considered flooded in all cases. Water fills all void space inside the waste package. Cases with no filler and no $GdPO_4$ are also studied to determine how the k_{eff} of the intact mode is affected by these conditions. It should be noted that while some cases do not seem to be realistic (physically possible), they are considered to obtain more conservative (higher) estimates for k_{eff}.

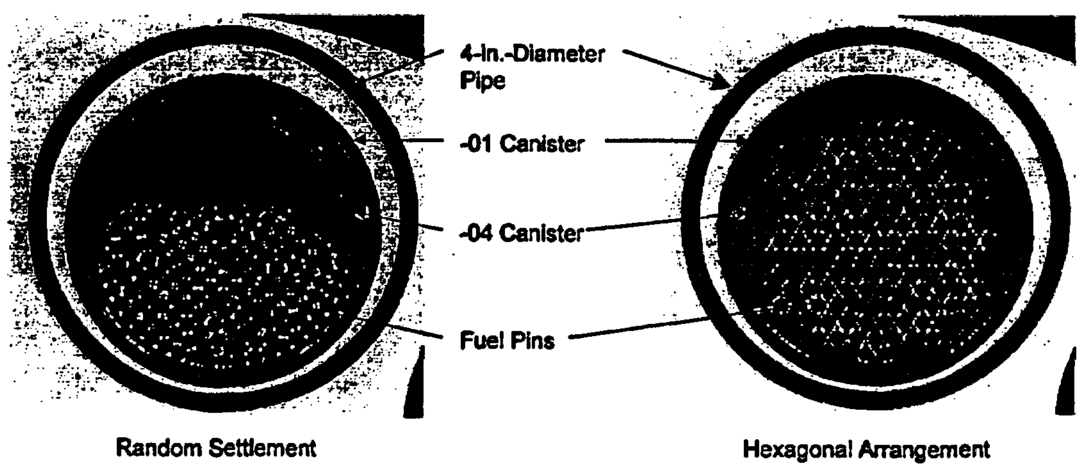

Figure 24. Different Arrangement of Fuel Pins Inside 4-in.-diameter Pipes

To examine the impact of fuel pin arrangement on the k_{eff} value, different arrangements (hexagonal, square, random-see Figures 23 and 24) of the fuel pins inside the -04 aluminum canisters are considered. Fuel pin pitch variations and differential flooding of components is investigated for the more reactive configurations discussed in Section 7.4.1.

Table 26. Results of the Intact-Mode Criticality Analysis

Case Description	Filler (Iron Shot)	Neutron Absorber (1 vol.% of the $Fe\text{-}GdPO_4$ mix)	Water Density (g/cm^3)	$k_{eff}\pm\sigma$
Base intact case. Each pipe is concentric with its -01 and -04 canisters. The fuel pins are in a square lattice arrangement	Yes	Yes	1.0	0.7271±0.0012
Similar to above case but the Al canisters inside the three "central" pipes are shifted toward the center of the WP.	Yes	Yes	1.0	0.7310±0.0011
Settled case. Similar to base case, but -01 and -04 are settled in each pipe.	Yes	Yes	1.0	0.7251±0.0011
Like above, but water density is lowered.	Yes	Yes	0.9	0.7193±0.0012
Like above, but water density is lowered.	Yes	Yes	0.8	0.7131±0.0012
Like above, but water density is lowered.	Yes	Yes	0.5	0.6809±0.0011
Like above, but no water in the waste package.	Yes	Yes	0.0	0.4107±0.0006
Similar to settled case, but the fuel pins are hexagonal even spaced.	Yes	Yes	1.0	0.7273±0.0011
Similar to settled case, but the fuel pins are randomly settled.	Yes	Yes	1.0	0.6484±0.0011
Similar to settled, but no neutron absorber is mixed with the iron shot.	Yes	No	1.0	0.8054±0.0012
Similar to settled, but with no filler (iron shot).	No	No	1.0	0.8327±0.0013

As can be seen from the above results, the variations of the key-waste package parameters from the base intact mode configuration cause a relatively small increase in the k_{eff}. The highest $k_{eff}+2\sigma$ of 0.8353 is obtained in a hypothetical configuration with no filler and no neutron absorber. Thus, the $k_{eff}+2\sigma$ for the system is well within the interim critical limit of 0.93. The decrease of the water density decreased the k_{eff} of the system; this shows that the nominal system is not over-moderated.

Occurrence of design basis events, including those with the potential for flooding the disposal container prior to disposal container sealing, is considered and bounded by the analysis results presented in Table 26 for many different intact configurations.

It should also be noted that the results from intact cases are bound by those for the degraded cases described in Section 7.4.3 (intact pipes with homogenized degraded products inside each pipe). The detailed analysis presented in this section was performed to gain more information regarding the neutronic characteristics of the system.

7.4 CALCULATIONS AND RESULTS–PART II: SCENARIOS WITH FISSILE MATERIAL RETAINED IN THE DOE SNF CANISTER

7.4.1 Partially Degraded DOE SNF Canister

The partially degraded mode refers to the cases where the 24 pipes contained in the 18-in. diameter DOE SNF canister no longer remain in their original (welded) arrangement and settle into a possibly more reactive configuration. This mode has been analyzed in detail in CRWMS M&O 1999d. The waste-package contents outside the DOE SNF canister are considered intact in all cases considered in this section. The degradation configurations and their refinements belong to the standard configuration Class 1 that is obtained via standard group scenario IP-3 (YMP 1998, p. 3-8).

Moreover, since the waste package is to be emplaced horizontally in an MGR, the -01 and -04 aluminum canisters are considered to be settled inside the steel pipes and the -01 aluminum canisters, respectively. Alternative pipe arrangements are evaluated to identify their impact on k_{eff} value. Figures 25 and 26 show the pipe arrangements considered for this mode. In the calculations for this mode, the Fe shot and the -01 and -04 aluminum canisters are assumed to be intact (non-corroded).

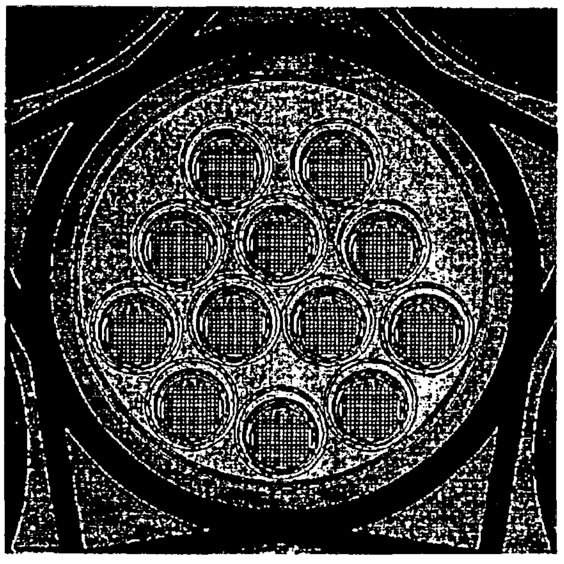

Figure 25. WP Showing a Partially Degraded, More Compact Arrangement of Pipes Inside the DOE SNF Canister

In addition to the different arrangements of the pipes mentioned above, the following conditions are also considered to evaluate the sensitivity of the partially degraded-mode criticality calculations to these conditions:

- Presence of Fe shot and $GdPO_4$
- Presence of Fe shot but absence of $GdPO_4$
- No Fe-$GdPO_4$ mixture (water fills all void space available).

The results listed in Table 27 (CRWMS M&O 1999d, Section 6.2) show that 4-in.-diameter pipe rearrangement has a major effect on k_{eff}. By comparing the base case in Table 27 with the base intact case from Table 26, k_{eff} is seen to increase from 0.7271 to 0.8014. This is due to the redistribution of the filler and neutron absorber to the periphery of the DOE SNF canister as the pipes are brought closer together. The $k_{eff}+2\sigma$ for these conditions and configurations does not exceed the interim critical limit of 0.93, even without $GdPO_4$ and iron shot. With the Fe-$GdPO_4$ mixture there is a very significant margin to the interim criticality limit.

Table 27. Results of the Partially Degraded Mode (pipe rearrangement)

Case Description	Filler (iron shot)	Neutron Absorber (1 vol.% of the Fe-$GdPO_4$ mix)	Water Density (g/cm^3)	$k_{eff}\pm\sigma$
Pipes settle to form a more compact arrangement (Figure 25).	Yes	Yes	1.0	0.8014±0.0010
Similar to above case, but with no $GdPO_4$.	Yes	No	1.0	0.8756±0.0039
Similar to above case, but with no iron shot and no $GdPO_4$.	No	No	1.0	0.9039±0.0033
Hypothetical hexagonal close-packed arrangement of pipes (Figure 26).	Yes	Yes	1.0	0.8299±0.0037
Similar to above case, but with no $GdPO_4$.	Yes	No	1.0	0.8993±0.0034

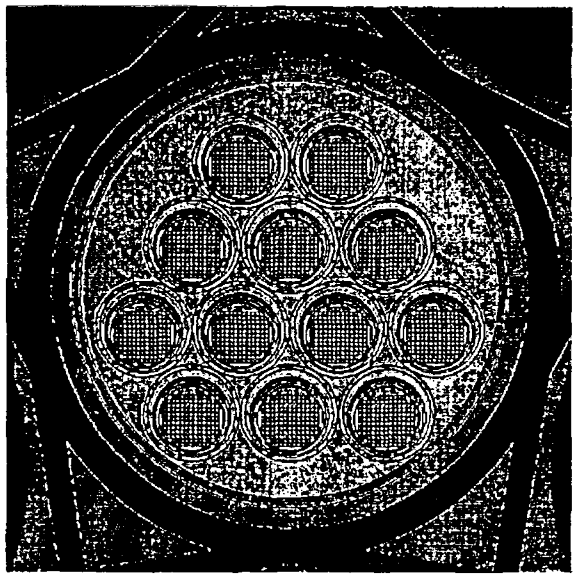

Figure 26. Hexagonal Close-Packed Arrangement of the Pipes

A separate set of calculations investigates the effect of fuel pin spacing inside 4-in.-diameter pipes and the effect of partial flooding. This analysis is performed in the partially degraded mode with more compact pipe configuration, as shown in Figure 25, and no oxidation of Fe or Al.

To study the impact of partial flooding on the k_{eff} value, it is assumed that water fills the void space in -01 and -04 aluminum canisters but no water exist in the remaining void space in the DOE SNF canister. This is not credible but is investigated because it minimizes the effectiveness of Gd as a neutron absorber.

The fuel pin spacing is increased from 0.06 cm to 0.48 cm (edge-to-edge distance) to study the impact on the k_{eff} value when there is more space (moderator) between adjacent fuel pins (e.g., less than 140 fuel pins within each -04 aluminum canister).

Table 28 summarizes the results of the cases run to study the impact of partial flooding and fuel pin spacing on the k_{eff} value (CRWMS M&O 1999d, Section 6.3). The impact of each of the factors on the k_{eff} value was studied individually as well as in combination.

Table 28. Results of the Partially Degraded Mode Criticality Analysis

Case Description	Distance Between Adjacent Pin Surfaces (cm)	$k_{eff}\pm\sigma$
Base case; more compact pipe configuration; full flooding, with 140 pins.	0.06	0.8014±0.0010
Like the above base case, but partial flooding; water in -01 and -04 canisters only, with 140 pins.	0.06	0.8632±0.0037
Like the base case, but with 130 pins.	0.12	0.8372±0.0037
Like the base case, but with 101 pins.	0.18	0.8263±0.0035
Like the base case, but with 89 pins.	0.24	0.8284±0.0041
Like the base case, but with 45 pins.	0.48	0.7723±0.0035
Like the base case, but partial flooding; water in -01 and -04 canisters only; 101 fuel pins.	0.18	0.8991±0.0030

The combined effects of partial flooding and reduction in the number of pins resulted in a maximum $k_{eff}+2\sigma$ for the partially degraded mode cases studied that does not exceed the interim critical limit of 0.93. As already mentioned, the results for all cases with intact pins are bound by those for the cases with homogenized fuel pins inside 4-inch pipes that are described in Section 7.4.3.

7.4.2 Totally Degraded Internal Structures of DOE SNF Canister with Intact Fuel Pins

These configurations comprise the intact fuel pins distributed inside the DOE SNF canister at various stages of degradation of the internal supporting structure. They represent refinements of the configuration Class 1 that results from the standard scenario IP-3, and they are described in CRWMS M&O 1999g under the configuration group IP-3-A, p.31. Configurations with fuel pins completely separated from the neutron absorber are not possible with the present design because the space between the pipes is filled with a uniform mixture of iron shot and neutron absorber.

The standard scenario group IP-3 applied to the DOE SNF canister results in the degradation of the internal supporting structure before the SNF degrades. This scenario continues the normal degradation process of the internal structure of the DOE SNF canister described in Section 6.2.1. For analyzing this class of configurations, a bounding approach is followed, as described in CRWMS M&O 2000e. It assumes that the intact fuel pins are arranged in a hypothetical square lattice with constant pitch (defined as the distance between the centers of two adjacent pins). These are incredible cases that bound all possible internal configurations of the intact pins distributed inside DOE SNF canister. The initial void space present inside DOE SNF canister is insufficient to allow complete degradation of the internal constituents. Thus, partial degradation would prevent water flow and further degradation until canister walls are degraded to allow mixing inside the WP.

Various shapes of the array of pins inside DOE SNF canister are considered. The DOE SNF canister shell is assumed to be breached but structurally intact, and the degradation products resulting from degradation of the canister internals (mainly FeOOH [goethite] and AlOOH [diaspore]) are distributed among fuel pins. The influence of the pitch variation for a hypothetical array of intact pins placed in a mass of hydrated degradation products on k_{eff} of the system is investigated. Once the most reactive configuration is found, a separate analysis is performed to determine the minimum necessary amount of neutron absorber to bring k_{eff} below the critical limit of 0.93.

Preliminary calculations (CRWMS M&O 2000e, Section 6.1.1) identified that the most conservative arrangement of pins inside DOE SNF canister is obtained for a cylindrical array of fuel pins placed in a mixture of goethite and diaspore. The layout of the configuration is presented in Figure 27. This bounding arrangement can not proceed from the credible scenarios and defies gravity effects.

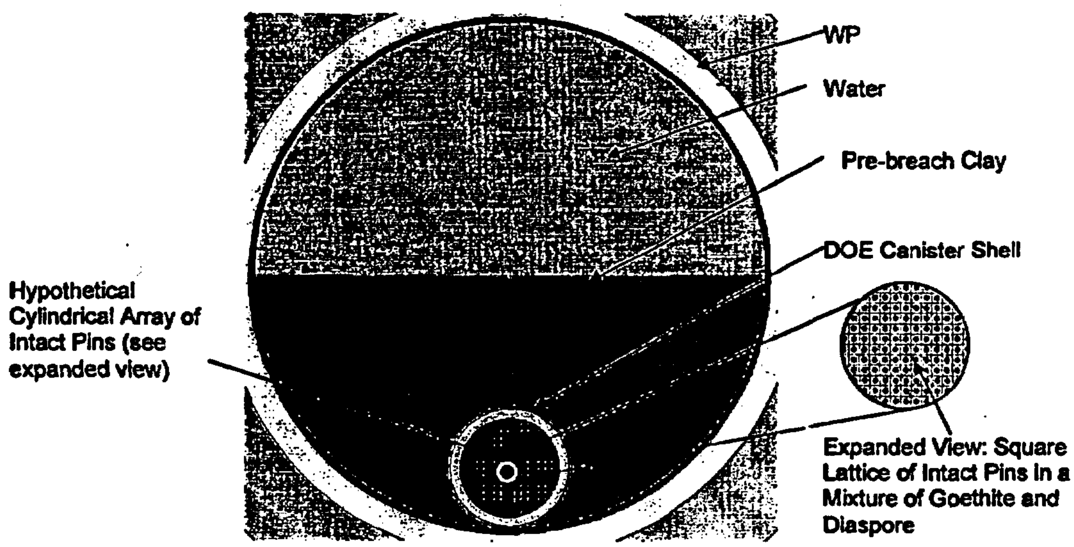

Figure 27. Transverse Cross-Sectional View of the Degraded WP Configuration with Intact Fuel Pins Dispersed within the DOE Canister Shell

Since the volume available inside the DOE SNF canister is not sufficient to accommodate all degradation products, the mixture among fuel pins can be composed from non-degraded materials, hydrated degradation products and water. As the limiting case, a uniform mixture of hydrated products is considered to fill the available space inside DOE canister, conservatively neglecting from the analysis the non-degraded materials (approximately 20% of the initial iron mass present in the intact configuration). A range of atom densities for hydrogen was investigated considering different mixtures of hydrated degradation products and/or water (CRWMS M&O 2000e, Section 6.1.1). The controlling factors are then the H/X ratio and the mass of Gd. Inclusion of the hydrated products of aluminum (diaspore) in the region of the fuel resulted in slightly higher values of k_{eff} (due to replacement of iron) and was further considered for conservatism.

As can be seen in Figure 27, the DOE SNF canister is situated at the bottom of the WP and is surrounded by dry pre-breach clay (maximum reflection). Pre-breach clay is defined as the degraded product from the steel and HLW glass inside the waste package prior to the breaching of the DOE canister, which contains the SNF and other internal supporting components. As shown in CRWMS M&O 2000e, Section 6.1.1, this arrangement is conservative when compared with results obtained for different positions of the DOE canister in the WP and for the presence of wet clay. The pre-breach clay composition is determined by multistage EQ6 calculations (Section 6.3) that assume that the DOE SNF canister is not breached during the first stage of degradation.

The most reactive case ($k_{eff}+2\sigma$ = 1.2063) is obtained when the array of pins fills the cross-sectional area of the DOE SNF canister. The fuel pin pitch (distance between centers of two adjacent pins) corresponding to this configuration is 0.94132 cm. An incremental amount of Gd is added to the FeOOH-AlOOH mixture of this representative case to determine the amount of neutron absorber necessary to decrease the k_{eff} of the system below the acceptance criterion. A quantity of 4 kg Gd (6.4 kg of $GdPO_4$) per canister is sufficient to bring $k_{eff}+2\sigma$ of the system to 0.9065. A subsequent analysis of the system indicated that the configuration with neutron absorber is over-moderated at a pitch of 0.94132 cm. The maximum reactivity of the lattice containing neutron absorber is reached for a pitch of 0.80132 cm. For this configuration, 5 kg of Gd (8 kg $GdPO_4$) per canister are needed to produce a $k_{eff}+2\sigma$ of 0.9245. It should be noted that while increased pitch promotes a greater k_{eff}, there is no identified mechanism that promotes the increased pitch(es).

In all above cases, the waste package is water-reflected. A case was run to demonstrate that the environment outside the waste package, whether tuff, water, or a mixture, has no significant impact on the configuration's k_{eff}. The k_{eff} of the waste package with reflected boundary conditions (0.9263±0.0011) is statistically identical to the k_{eff} of the water-reflected waste package (0.9227±0.0009). The results show that neutron reflection outside the waste package is not an important factor for this degraded configuration.

7.4.3 Totally Degraded Fuel Pins Inside Non-Degraded DOE Canister

The application of the standard scenario group IP-1 (CRWMS M&O 1999g) to Enrico Fermi SNF in the flooded DOE SNF canister can result in a set of distinct degraded configurations. If the fuel degradation takes place in the initial location (within the supporting structure), the configurations belong to Class 6 (CRWMS M&O 1999g). Since the design of the DOE SNF canister includes a simple supporting structure for the fuel pins (4-in.-diameter pipes), the degradation of the fuel pins takes place in the 4-in.-diameter pipes within the DOE SNF canister. This class of configurations can also be regarded as a bounding case for all intact configurations and was expanded in order to cover all possible DOE spent nuclear fuels that belong to this group.

Due to the highly corrosion-resistant properties of the cladding (Zr), the degradation of the SNF before the supporting structure is very unlikely. This configuration is analyzed in order to include the situation when the fuel cladding has been mechanically removed or damaged before or during emplacement. At the end of the spectrum of degraded configurations in this class are the cases with the fuel completely degraded within the pipes, forming a mixture with the degradation products and water. The configuration for a more compact arrangement of settled pipes is shown in Figure 28.

As a starting point for the criticality calculations in this class, a very conservative configuration with the 4-in.-diameter pipes containing only UO_2 (from degraded fuel), water and diaspore were investigated. All other constituents inside 4-in.-diameter pipes, including iron and Gd were conservatively neglected. Since the range of H/X ratios covers a very broad range, the presence of diaspore does not play a significant role for these configurations. The rest of the WP components were considered intact, including iron shot and $GdPO_4$ in the DOE SNF canister.
A direct comparison with an intact configuration analyzed in Section 7.3.1 can be made for the cases with 1 vol.% of $GdPO_4$ in the iron shot-$GdPO_4$ mixture (4.84 kg $GdPO_4$). The $k_{eff}+2\sigma$ values for these degraded configurations range from 0.8440 to 0.9506 (depending on the volume fraction of water mixed with the degraded products). The $k_{eff}+2\sigma$ value for the similar arrangement with intact fuel is 0.7295. Note that the degraded cases are more reactive and bound the values obtained with intact fuel.

Different amount of $GdPO_4$ was added to the iron shot-neutron absorber mixture to reduce k_{eff}. With 9.6 kg of $GdPO_4$ the $k_{eff}+2\sigma$ for the system is below 0.93 for all investigated H/X ratios with the exception of the configuration that has the full lengths of 4-in.-diameter pipes filled with a homogeneous mixture ($k_{eff}+2\sigma$ =0.9308). This result shows the importance of having some neutron absorber and iron shot distributed in each pipe. A separate set of cases investigated the effect of considering the presence of the filler material in each 4-in.-diameter pipe (filling the space between -01 canister and the 4-in.-diameter pipe in the Enrico-Fermi design). The goethite in excess of the available volume in each pipe was conservatively neglected (approximately 20%). The results show a significant decrease in k_{eff} and also a high effectiveness of the Gd. With 6 kg of Gd (9.6 kg of $GdPO_4$) uniformly distributed in the initial iron shot-$GdPO_4$ mixture the highest $k_{eff}+2\sigma$ drops to 0.8090. Settling the pipes as shown in Figure 28 increases $k_{eff}+2\sigma$ to 0.8843. Removing the water from the rest of the WP produces an increase in $k_{eff}+2\sigma$ of approximately 4.5% that does not exceed the interim criticality limit of 0.93. Reflection of neutrons from the materials outside WP (water, silica) has no impact on k_{eff} for this class of configurations.

As shown in (CRWMS M&O 2000e, Section 6.1.2), increasing the fuel mixture column length results in a decrease in k_{eff}. A higher $k_{eff}+2\sigma$ was obtained for a shorter column (64.83 cm) that contains no goethite and no absorber mixed with degraded fuel. The increase in $k_{eff}+2\sigma$ from the base case (length of fuel section of 77.47 cm, 9.6 kg of $GdPO_4$) to the shorter column is from 0.8090 to 0.8591.

As will be shown in the next sections, the limiting cases for the fully degraded configurations require at least 14.5 kg of $GdPO_4$. Repeating some of the above cases with this amount of neutron absorber, $k_{eff}+2\sigma$ decreased (to 0.7883 for the base case), further increasing the margin to interim criticality limit of 0.93.

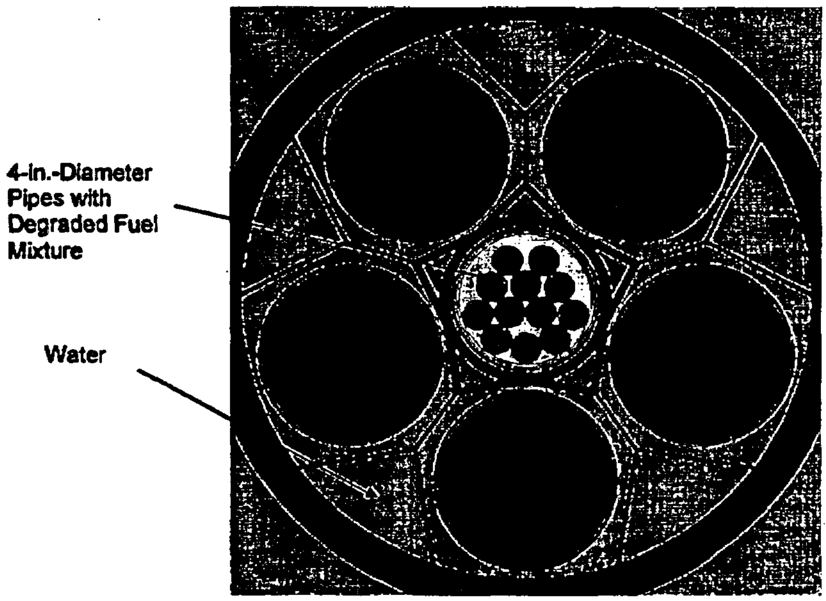

Figure 28. Cross-Sectional View of the Degraded WP Configuration with Degraded Fuel Mixture in 4-in.-Diameter Pipes

7.4.4 Degraded WP and DOE Canister Internal Structures with Intact DOE SNF Canister Shell

The configurations analyzed in this section are refinements of the configuration Class 2 (CRWMS M&O 1999g) and can be obtained via any of the standard scenarios. The configurations include an intact (but breached) DOE SNF canister outer shell that keeps the mixture with degraded fissile material from being dispersed in the HLW clayey material. All other internal structures inside WP are considered fully degraded.

As discussed in the intact mode criticality calculations (CRWMS M&O 1999d), when the iron shot degrades the void volume between the particles is filled by the expanding iron oxide, thereby displacing water. In the partially degraded cases evaluated, there is not enough space available in the canister to accommodate the entire amount of degradation products from all the filler material. This is why in the present fully degraded analysis, the space inside the DOE SNF canister is generally considered filled with hydrated degradation products (mainly goethite and diaspore) and no free water. The excess goethite (approximately 20% by volume of the initial iron shot) is conservatively neglected from the analysis due to the lack of space within the canister.

Various compositions and densities of the degraded mixture inside DOE SNF canister have been evaluated. The results (CRWMS M&O 2000e, Section 6.1.3) show that a mixture containing UO_2, diaspore, and goethite having a length equal with the initial footprint length of the fuel and completely filling the DOE SNF canister area is the most reactive. As shown in a separate set of calculations performed without diaspore, including the diaspore is conservative because aluminum has a lower absorbtion cross section than iron. If diaspore is not included, there is still enough goethite to fill the space. As mentioned above, the excess goethite is conservatively neglected in the present calculations.

The DOE SNF canister shell was evaluated in three locations. One location is at the bottom of the waste package, such that the DOE SNF canister is fully reflected by the clayey material from degraded components (see Figure 29). Another location is half submerged into the clayey material. The third location is sitting on top of the clayey material. The most reactive configuration is the one with the DOE SNF canister shell placed at the bottom of the WP, as shown in Figure 29. Note that the assumed water reflector on the outside of the waste package is a conservative assumption albeit a very small effect on the k_{eff}, as explained in Section 7.4.2.

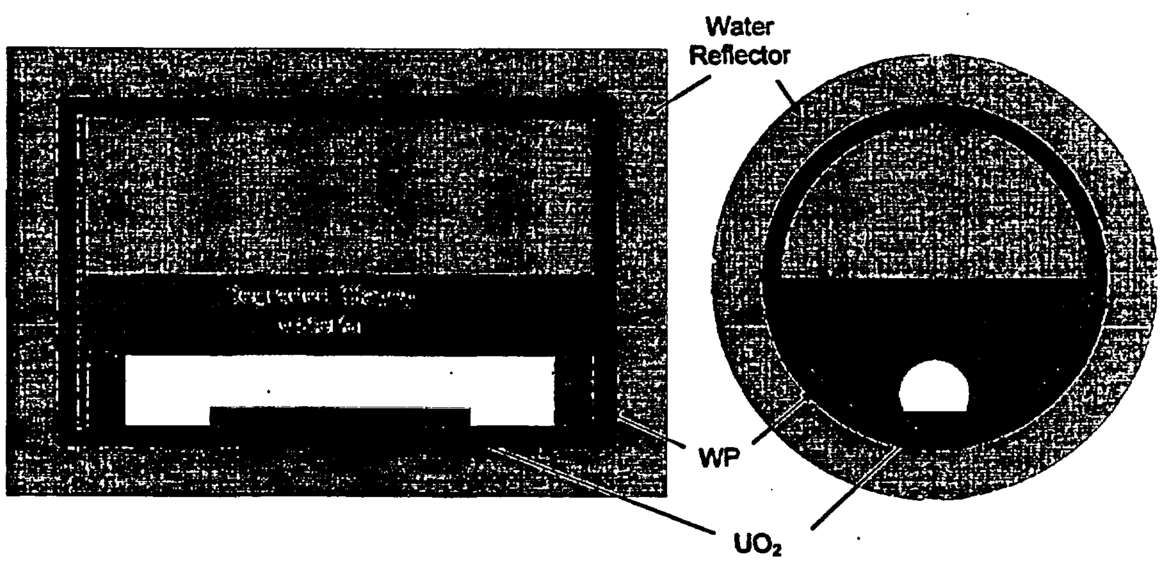

Figure 29. Cross-Sectional Views of the Breached Intact DOE SNF Canister in Clayey Material within the Waste Package

The influence of adding an incremental amount of Gd to the content of DOE SNF canister is subsequently investigated. The results (CRWMS M&O 2000e, Section 6.1.3) show that at least approximately 9 kg of Gd (14.5 kg gadolinium phosphate) must be uniformly distributed within the SNF canister to keep k_{eff} of the system below 0.93 for all investigated configurations. This is the configuration class that contains the overall limiting case for criticality calculations. The results obtained for the configurations with intact pins still contained in the DOE canister (Section 7.4.1) are bounded by the results obtained in this section.

In a separate set of calculations, the addition of a small volume fraction of water to the mixture inside DOE SNF canister (and consequent removal of similar fractions of hydrated products) produces a small decrease in k_{eff}. The highest value of $k_{eff}+2\sigma$ (0.9053) for the system containing the neutron absorber is obtained for a full reflection at the WP outer boundary. Increasing the length of the column containing degraded fuel results in a decrease of the k_{eff}.

7.5 CALCULATIONS AND RESULTS–PART III: SCENARIOS WITH FISSILE MATERIAL DISTRIBUTED IN WASTE PACKAGE

7.5.1 Degraded Waste Package Internal Structures with Intact Fuel Pins

This group of configurations, characterized by intact fuel pins immersed in the clayey material resulting from the degradation of the HLW glass and other internal components of the waste package and the DOE SNF canister, represents a refinement of the configuration Class 1. It can be reached by applying the standard scenario group IP-3 to both the DOE SNF canister and waste package. The configurations are considered likely due to the high corrosion resistance of the fuel cladding, zirconium.

In order to perform the criticality calculations for this case, a bounding approach similar to the one presented in Section 7.4.2 has been adopted. The intact fuel pins are dispersed in the clayey material in a regular square lattice with a constant pitch. Two distinct groups of configurations are analyzed. The first one assumes that the clay in the waste package is homogeneously mixed with the degradation products from DOE canister internals and shell. The main constituents of the DOE SNF degradation products considered are goethite and diaspore. Inclusion of the diaspore is conservative since it essentially acts to disperse the more absorbing goethite.

The second group of configurations assumes that the degradation products from DOE SNF canister and the clay are not mixed, with the heaviest components (goethite) settled at the bottom of the WP. The results of the criticality analysis are summarized in this section (CRWMS M&O 2000e, Section 6.2.2). The arrangements considered in analyses are presented in Figures 29 and 30.

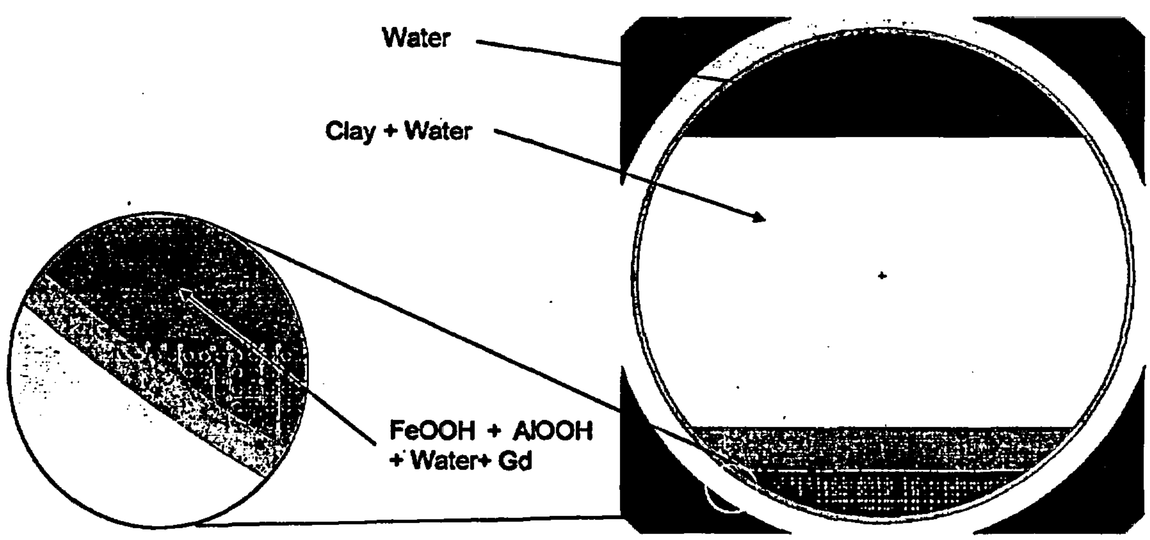

Figure 30. Intact Enrico Fermi Pins Surrounded by Degraded DOE SNF Canister Components at the Bottom of the Waste Package

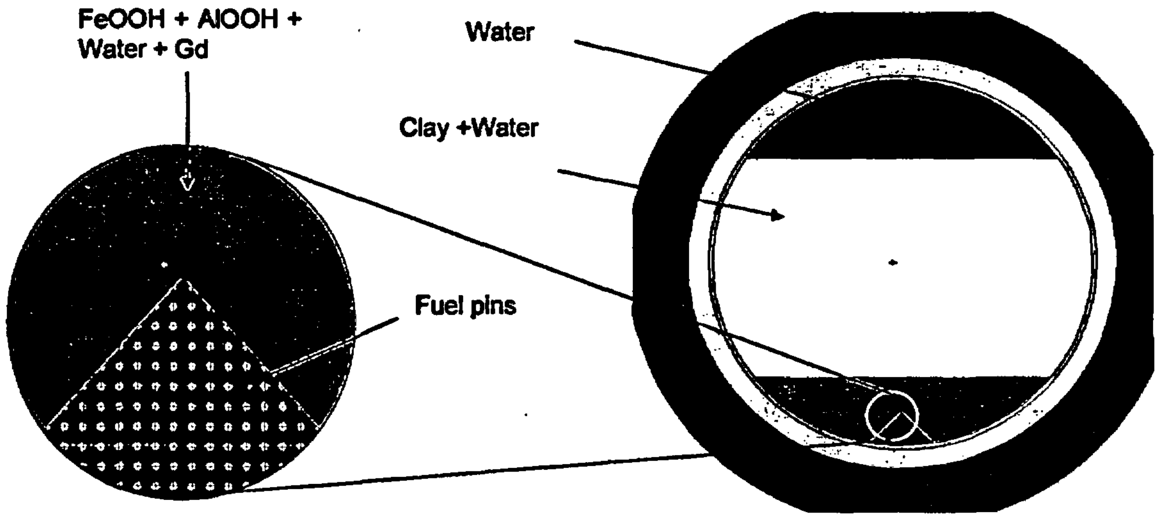

Figure 31. Intact Enrico Fermi Pins Stacked at the Bottom of the Waste Package

These configurations result as a subsequent stage of degradation of the configuration presented in Figure 27. The degraded iron and aluminum will form a layer at the bottom of the waste package with a layer of clayey material from the HLW glass above (since the density of the metal corrosion products is higher than the density of the clay). The fuel pins can be settled at the bottom of the waste package as shown in Figure 30 or can be stacked as shown in Figure 31. The pitch between fuel pins was varied, but the height of the fuel pin arrangement was not higher than the initial DOE SNF canister diameter. Fuel pins are immersed in a uniform mixture of the degradation products of the DOE SNF canister (FeOOH, AlOOH, and $GdPO_4$). The volume percent of water in this layer is varied from 28.6% to 50%. The clay layer above has a 37.5 vol.% of water.

The lattice pitch is varied between 0.4 to 1.4 cm keeping the geometrical restraints with respect to the height of the pile of pins. The maximum value of $k_{eff}+2\sigma$ for the configuration presented in Figure 30 (settled lattice of fuel pins) is 1.2599 for a fuel pin pitch of 1.15 cm and a water content of 50 vol.% in the bottom layer. A quantity of 3 kg of Gd (4.84 kg of $GdPO_4$) uniformly dispersed in the goethite (FeOOH) layer is sufficient to bring $k_{eff}+2\sigma$ of the system to 0.8505. Addition of neutron absorber changes the optimum pitch to 0.901 cm. For the configuration with this pitch, a minimum quantity of 3.5 kg Gd (5.6 kg $GdPO_4$) must be uniformly distributed in the mixture to bring $k_{eff}+2\sigma$ to 0.9205, which is below the critical limit of 0.93.

Similar calculations performed for the configuration depicted in Figure 31 (pile of fuel pins) produce the largest k_{eff} for a pitch of 0.901 cm. In order to reduce the effective neutron multiplication factor of the system ($k_{eff}+2\sigma$) to a value of 0.8958, 9 kg of Gd (14.5 kg of $GdPO_4$) must be uniformly dispersed in the goethite layer. A reflective boundary at the outer WP surface added to this configuration produces $k_{eff}+2\sigma$ of 0.9216, pointing to a more significant influence of the reflector outside the WP outer barrier for this class of configurations because the fuel pins are so close to the reflective boundary.

7.5.2 Degraded DOE SNF Canister Mixture Settled at the Bottom of the WP

After the complete degradation of all the WP internal constituents, including the DOE SNF canister, the resultant configurations can include the degradation products as layers or complex mixtures settled within the WP. These configurations belong to Class 2 (CRWMS M&O 1999g). It comprises a large number of refinements and variations, and it can be reached by any of the standard top-breach scenarios (IP-1, IP-2, or IP-3). A bounding approach was also adopted for this analysis, investigating various possible combinations of the fissile material with different degradation constituents. A typical geometry for this group of configurations is included in Figure 32. The fissile material mixture is settled at the bottom of the waste package. A layer of goethite that results from the degradation of the DOE SNF canister is placed on top of the fissile mixture. The rest of the WP is filled with a layer of clay and water. The configuration can directly result from subsequent degradation of the configurations presented in Section 7.4.2 or 7.5.1.

The Enrico Fermi fuel pins are fully degraded to uranium oxide, and the rest of the fuel pins' constituents are conservatively neglected. The diaspore resulting from the degradation of the aluminum cans is mixed with UO_2. The small amount of diaspore serve in this case to more optimally disperse the UO_2 and is conservatively included in the calculations since it displaces an equivalent amount of more absorbing goethite in all configurations with mixed layers. The fuel mixture length is equal to the initial footprint of the fuel pins. The analysis focuses on varying the fractions of the degraded constituents in the mixture. The effect of water addition in all layers is also evaluated in order to identify the optimum-moderated system. The densities of the layers are calculated in order to assure a correct physical representation with the higher densities layers placed at the bottom.

The first set of cases investigates the effect of adding various water volume fractions to the existing layers. The results indicate (CRWMS M&O 2000e, Section 6.2.3) that as the volume fraction of water in the layers exterior to the fissile layer increases, the k_{eff} decreases.

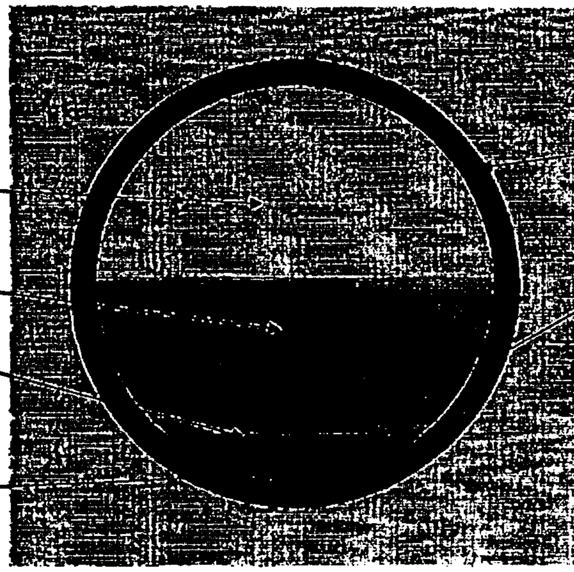

Figure 32. Fully Degraded DOE SNF Canister Settled at the Bottom of Waste Package

A similar type of parametric analysis investigates the effect of varying the volume fraction of goethite that is mixed with the fuel mixture. An optimum-moderated configuration with all layers dried is found for a goethite volume fraction in the fuel mixture of 0.56. Using this optimum mixture as a base case, a new parametric study was done to study the effect of adding water to the layers. A volume fraction of 0.37 of water in the fuel layers results in the largest k_{eff} for this configuration with no neutron absorber. The length of the fuel slurry was increased (194.4 cm) to accommodate the total volume of the mixture. All values of k_{eff} for these configurations without neutron absorber are well above 1.00. Reversing the order of the layers (dried clay at the bottom) and varying the volume fraction of water in the fuel slurry plus goethite to 0.5 produces values of k_{eff} below 0.91.

The largest k_{eff} is obtained for a configuration with the fuel slurry mixed with all available goethite and a water volume fraction of 0.44. The water volume fraction was varied in the analysis between 0 and 0.5. In order to accommodate the volume of the mixture, the fuel slurry length was increased to the full length of the waste package (304 cm). The maximum $k_{eff}+2\sigma$ for this configuration is 1.2702.

A minimum quantity of 0.935 kg of Gd (1.5 kg $GdPO_4$) uniformly dispersed in the fuel mixture is sufficient to bring the above configuration below the interim critical limit of 0.93. In order to check the effectiveness of the neutron absorber, parametric studies were done varying fuel slurry length and water volume fraction. For values of the water volume fraction between 0.3 and 0.55, the largest $k_{eff}+2\sigma$ is 0.8972. Reducing the fuel slurry length increases the k_{eff} of the system. The calculations show that for a length of 80 cm (with constant mixture volume), $k_{eff}+2\sigma$ increases to 0.9579. Adding a reflective outer boundary increases k_{eff} less than 1%. A quantity of 3.5 kg Gd (5.6 kg $GdPO_4$) distributed uniformly in the degraded fuel mixture decreases the most reactive case to 0.6368. For these fully degraded configurations, the neutron absorber proves to be extremely effective.

7.5.3 Totally Degraded DOE SNF Canister and WP Internal Structure

The mixture of clayey material obtained after total degradation of the WP internal constituents and uniformly distributed fissile material is analyzed with varying fractions of water. As expected, the results are well below the interim critical limit of 0.93. For water volume fractions between 0 and 0.3, $k_{eff}+2\sigma$ varies between 0.4 and 0.363. The clay composition includes the remaining Gd from an initial loading of 3 kg (4.84 kg $GdPO_4$) (CRWMS M&O 1999m).

7.6 SUMMARY

The results of three-dimensional (3-D) Monte Carlo criticality calculations for all anticipated intact and degraded configurations show that the requirement of $k_{eff}+2\sigma$ less than or equal to the interim critical limit of 0.93 is satisfied for the Enrico Fermi codisposal WP with at least 3.0% by volume of gadolinium phosphate (14.5 kg $GdPO_4$) uniformly distributed in the initial iron shot-$GdPO_4$ filler.

Most cases analyzed require only a fraction of the indicated insoluble neutron absorber in order to be below the interim criticality limit. The representative intact configurations that were investigated do not require neutron absorber. The limiting case for the configurations with the fuel inside DOE SNF canister was obtained for a homogeneous mixture of fuel and hydrated products inside DOE SNF canister, which require 14.5 kg of $GdPO_4$ uniformly distributed in the canister volume. The overall limiting case was obtained for an extremely conservative configuration comprising a pile of fuel pins stacked at the bottom of the waste package. This configuration required the same amount of insoluble neutron absorber (14.5 kg of $GdPO_4$) to be distributed in the degraded mixture surrounding the pins.

As expected, the results from analyzing the configurations with the fuel degraded in the 4-in.-diameter pipes bound the intact and partially degraded cases with intact fuel in pipes. Also, the results for the configurations with fully degraded DOE SNF internal constituents bound the partially degraded cases. On the other hand, it can be noticed that considering the presence of the aluminum cans (or the aluminum degradation products), which are specific to Enrico Fermi SNF, resulted in higher k_{eff} values for the limiting cases (conservative assumption) than the cases that neglected the cans degradation products.

8. CONCLUSIONS

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All of the analyses presented in this report are based on the Viability Assessment (VA) (DOE 1998b) design of the 5-DHLW/DOE SNF waste package. An impact evaluation of the results presented in this report should be performed for future waste package designs.

8.1 STRUCTURAL ANALYSIS

The results from the 2-D FEA calculations given in Section 3.3 show that there is sufficient clearance between the inner diameter of the basket support pipe and the outer diameter of the DOE SNF canister for the DOE SNF canister to be removed from the waste package after a tipover DBE, which is the limiting DBE within the criteria specified in the SDD (CRWMS M&O 1999o).

The maximum deformations in each component of the waste package are acceptable. The outer barrier is directly exposed to a dynamic impact with an essentially unyielding surface. Therefore, local plastic deformations are unavoidable on the outer surface. Similarly, the basket support structure receives the direct impacts of pour canisters, which results in limited permanent deformations of the basket plates.

The results given in Section 3.3.2 show that there would be no interference between individual 4-in.-diameter steel pipes and aluminum -01 canisters within the DOE SNF canister. Thus, the waste package can be unloaded after a tipover DBE.

The ultimate tensile strength of 316L SS and the maximum stress in the 18-in.-diameter DOE SNF canister are also given in Section 3. A comparison of these values reveals that all stresses are below the material's ultimate strength. Therefore, it is concluded that the performance of the 5-DHLW/DOE SNF waste package internal design is structurally acceptable when exposed to a tipover event, as long as the DOE SNF canister loaded mass limit of 2270 kg is not exceeded.

8.2 THERMAL ANALYSIS

The HLW glass dominates the thermal heat output of the waste package. The maximum temperatures are also shown in Table 29. The HLW glass and Fermi fuel temperatures are below the design limits.

Table 29. Fermi Codisposal WP Thermal Results and Governing Criteria

WP Metric	SDD Criterion	Fermi Codisposal WP Value
Maximum WP heat output (W)	< 11,800	3,577[a]/ 20,619[b]
Maximum HLW temperature (°C)	< 400	222[a]/395[b]
Maximum DOE SNF temperature in codisposal waste package (WP) (°C)	< 350	225[a]/327[b]

NOTES: [a] Nominal case value is equal to 5 times the heat output of one HLW glass canister and 24 times the heat output of one Fermi fuel -01 canister.
[b] Bounding case with 5.8 times the heat output from the five HLW glass canister.

Two of the three SDD criteria for thermal calculations, SDD 1.2.1.6 and SDD 1.2.1.7 cited in Section 2.2.2, contain TBV-092 and TBV-241 respectively. SDD 1.2.1.7 also has a cladding temperature limit for other than zircaloy clad fuel that is unknown at this time and carries TBD-179. TBV-092 restricts the HLW glass temperature to less than 400°C, whereas TBV-241 restricts the temperature of zircaloy clad fuel to less than 350 °C. The Fermi fuel is zirconium-clad fuel and therefore must meet the unknown limit of TBD-179. The results of this analysis are compared to the zircaloy clad limit to demonstrate compliance with the known limit pending determination of TBD-179. As shown in CRWMS M&O 1999a, the results of the analyses indicate that the zircaloy cladding temperature criterion under nominal conditions is met by more than 125 °C, and the heat output limit in SDD 1.2.4.4 is met by more than 8000 watts.

8.3 SHIELDING ANALYSIS

The maximum surface-dose rate on the outer radial surface of the waste package is 10.951 rem/h at the middle segment of the SRS glass canisters (segment 4 in Figure 14). The maximum dose rates on the bottom and top surfaces of the waste package are about one-third and one-tenth, respectively, of the maximum dose rate on the radial surface.

While the gamma spectra of the SRS HLW glass and the Fermi SNF fuel are similar, the total gamma intensity for each SRS HLW glass canister is about nine times higher than that of the Fermi DOE SNF canister. Because of the lower radiation intensity from the SNF canister in the central position in the waste package, and because of the high-density of the fuel material, the Fermi DOE SNF canister makes a very small contribution to the waste-package surface-dose rates. The primary gamma dose rate dominates over neutron dose rate by approximately a factor of 50 to 100.

The SDD criterion for shielding calculations (SDD 1.2.4.3), cited in Section 2.2.3, has TBD-3764 for the maximum external surface dose rate. The results of this analysis provide the dose rate at all external surfaces of a loaded and sealed waste package, but since no acceptable dose rate has been specified, the acceptability of the external dose rate must await the resolution of TBD-3764. However, the dose rates presented herein are expected to be much lower than any likely limit. Therefore, the TBD-3764 will be carried through the conclusions in this section.

8.4 GEOCHEMISTRY ANALYSIS

The geochemistry analyses that evaluate potential critical configurations from intact through degraded follow the general methodology developed for application to all waste forms containing fissile material. Sequences of the events and/or processes of component degradation are developed. Standard scenarios listed in the topical report are refined using the unique fuel characteristics of the Fermi SNF. Potentially critical configurations were identified and analyzed.

The cases in which the HLW glass is allowed to degrade rapidly produce the alkaline conditions. These single-stage EQ6 runs produce the highest gadolinium loss (≤ 2.3% in ≥ 100,000 years) but also produce high uranium loss (up to 100%), which reduces the chances of criticality internal to the waste package.

The cases in which the gadolinium and uranium are exposed to long-lived acidic conditions (pH~5 to 6) show that 0.65% or less of the gadolinium is lost and less than 4.15% of the uranium is lost.

8.5 INTACT- AND DEGRADED-MODE CRITICALITY ANALYSIS

The criticality analyses considered all aspects of intact and degraded configuration of the codisposal waste package containing Enrico Fermi SNF, including optimum moderation condition, rearrangements of the fuel pins and fissile material, and neutron absorber distribution. The results of 3-D Monte Carlo calculations from both the intact and the degraded component criticality analyses show that the interim critical limit requirement of $k_{eff}+2\sigma$ be less than or equal to 0.93 is satisfied for the proposed design, but required more neutron absorber than initially expected. The amount of neutron absorber (gadolinium phosphate) required to satisfy the above criterion is 14.5 kg of $GdPO_4$ in or on the initial iron shot-$GdPO_4$ filler, which must be placed in and around the support pipes containing the -01 fuel canisters. With this design, there will be approximately eight DOE SNF canisters with Enrico Fermi SNF, which correspond to eight waste packages.

A number of parametric analyses were run to address or bound the configuration classes discussed in Section 6.2.1. These parametric analyses identified conditions of optimum moderation, optimum spacing between fuel pins, optimum fissile concentration, and minimum neutron absorber requirements. The results from the degraded criticality analyses show that the most reactive configurations are the configurations with fully degraded components inside DOE SNF canister and the configurations with intact fuel pins dispersed in the WP. These configurations result in $k_{eff}+2\sigma$ less than or equal to 0.93 with at least 14.5 kg $GdPO_4$ distributed in the initial iron shot-$GdPO_4$ filler. An attempt to eliminate the enrichment of the fuel as a limiting parameter in the design necessitated unreasonably high amounts of the gadolinium neutron absorber.

Much lower amounts of the gadolinium neutron absorber are necessary to keep $k_{eff}+2\sigma$ below 0.93 for the configurations with fully degraded DOE SNF canister and WP internals. This finding assures that the margin for criticality is increasing at longer disposal times, as long as the neutron absorber remains dispersed within the WP. Due to the basket design and the iron shot-$GdPO_4$ filler, the codisposal waste package containing a DOE SNF canister of spent Enrico Fermi fuel will not form critical configurations for any credible degradation scenarios.

The SDD criterion for criticality calculations (SDD 1.2.2.1.12), cited in Section 2.2.5, has TBV-245 relating to values of the interim critical limit. Intact and degraded component criticality calculations include variations on moderators and moderator densities, which encompass flooding the waste package. Occurrence of design basis events, including those with the potential for flooding the disposal container prior to disposal container sealing, is considered and analyzed using very conservative assumptions for many different intact configurations. All these configurations were below the interim critical limit of 0.93; therefore, the TBD-245 is non-critical and is not carried through the conclusions in this section.

8.6 ITEMS IMPORTANT TO CRITICALITY CONTROL

As part of the criticality licensing strategy, items that are important to criticality control will be identified during evaluation of the representative fuel types designated by the National Spent Nuclear Fuel Program. As a result of the analyses performed for the evaluation of the codisposal viability of U-Zr and U-Mo (HEU) DOE-owned fuel, several items are identified as important to criticality control. The DOE SNF canister shell is naturally an item that is important to criticality control since it confines the fissile elements to a specific geometry and location within the waste package. The internal structure that was designed for the DOE SNF canister containing the Enrico Fermi fuel is also an important criticality control item since it confines the fissile elements to a specific geometry and location within the DOE SNF canister. The use and distribution of the iron shot-$GdPO_4$ filler with at least 14.5 kg $GdPO_4$ is also important to criticality control.

Based on the conclusions derived in Section 7.6 of the degraded cases, the specified amount of gadolinium phosphate neutron absorber will have to be added with the iron shot used to fill the available space between all the 4-in.-diameter steel pipes in the DOE SNF canister basket assembly, and also fill the void space available around the -01 canisters inside each of the 4-in.-diameter pipes. Therefore, the amount of gadolinium phosphate absorber material that will be placed in the supporting basket assembly inside the DOE canister is also an item important to criticality control.

All calculations are based on a maximum of 4.817 kg of U-235 per -01 canister (one pipe). The analyses are based on the fuel pin type that has the highest uranium enrichment (enriched in U-235). The degraded configurations of the Enrico Fermi SNF bound the other types of U-Zr and U-Mo (HEU) DOE-owned SNF, as long as the limits on mass of uranium and its enrichment, and the linear density, are not exceeded.

Hence, the total mass of fissile element (U-235) should not exceed the mass used in deriving the conclusions of this report, which is 115.6 kg of U-235 per DOE SNF canister. The maximum enrichment is 25.69% in U-235. The linear density of the U-235 should not exceed 62 g/cm in each of the 24 pipes. This value is calculated by dividing the total fuel mass by the number of 4-in.-diameter pipes and by the active length of the Enrico Fermi fuel pin.

9. REFERENCES

9.1 DOCUMENTS CITED

Briesmeister, J.F., ed. 1997. *MCNP-A General Monte Carlo N-Particle Transport Code.* LA-12625-M, Version 4B. Los Alamos, New Mexico: Los Alamos National Laboratory. ACC: MOL.19980624.0328.

CRWMS M&O 1995a. *Thermal Evaluation of the Conceptual DHLW Disposal Container.* BBAC00000-01717-0200-00002 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19960626.0161.

CRWMS M&O 1995b. *Total System Performance Assessment - 1995: An Evaluation of the Potential Yucca Mountain Repository.* B00000000-01717-2200-00136 REV 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19960724.0188.

CRWMS M&O 1996. *Waste Package Filler Material Testing Report.* BBA000000-01717-2500-00008 REV 02. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19970121.0004.

CRWMS M&O 1997a. *Waste Package Design Basis Events.* BBA000000-01717-0200-00037 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19971006.0075.

CRWMS M&O 1997b. *Waste Package Fabrication Process Report.* BBA000000-01717-2500-00010 REV 02. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19971218.0275.

CRWMS M&O 1997c. *DHLW Canister Source Terms for Waste Package Design.* BBA000000-01717-0200-00025 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19970711.0019.

CRWMS M&O 1997d. *MCNP Evaluation of Laboratory Critical Experiments: Homogeneous Mixture Criticals.* BBA000000-01717-0200-00045 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19971230.0134.

CRWMS M&O 1997e. *Waste Package Materials Selection Analysis.* BBA000000-01717-0200-00020 REV 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980324.0242.

CRWMS M&O 1997f. *Preliminary Design Basis for WP Thermal Analysis.* BBAA00000-01717-0200-00019 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980203.0529.

CRWMS M&O 1997g. *Degraded Mode Criticality Analysis of Immobilized Plutonium Waste Forms in a Geologic Repository.* Predecisional Document. A00000000-01717-5705-00014 REV 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980422.0911.

CRWMS M&O 1998a. Not Used.

CRWMS M&O 1998b. *5-High Level Waste DOE Spent Fuel Waste Package Structural Calculations.* BBA000000-01717-0210-00021 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19981006.0187.

CRWMS M&O 1998c. *Software Qualification Report for ANSYS V5.4, A Finite Element Code.* CSCI: 30040 V5.4. DI: 30040-2003, Rev. 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980609.0847.

CRWMS M&O 1998d. Not Used.

CRWMS M&O 1998e. *Software Qualification Report for MCNP Version 4B2, A General Monte Carlo N-Particle Transport Code.* CSCI: 30033 V4B2LV. DI: 30033-2003, Rev. 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980622.0637.

CRWMS M&O 1998f. Not Used.

CRWMS M&O 1998g. Not Used.

CRWMS M&O 1998h. *EQ3/6 Software Installation and Testing Report for Pentium Based Personal Computers (PCs).* CSCI: LLYMP9602100. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980813.0191.

CRWMS M&O 1998i. "Unsaturated Zone Hydrology Model." Chapter 2 of *Total System Performance Assessment-Viability Assessment (TSPA-VA) Analyses Technical Basis Document.* B00000000-01717-4301-00002 REV 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19981008.0002.

CRWMS M&O 1998j. *EQ6 Calculations for Chemical Degradation of Pu-Ceramic Waste Packages.* BBA000000-01717-0210-00018 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980918.0004.

CRWMS M&O 1998k. Not Used.

CRWMS M&O 1998l. Not Used.

CRWMS M&O 1998m. *EQ6 Calculations for Chemical Degradation of PWR LEU and PWR MOX Spent Fuel Waste Packages.* BBA000000-01717-0210-00009 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19980701.0483.

CRWMS M&O 1998n. *Dose Calculations for the Co-Disposal WP of HLW Canisters and the Fast Flux Test Facility (FFTF) Fuel.* BBA000000-01717-0210-00019 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990129.0075.

CRWMS M&O 1998o. Not Used.

CRWMS M&O 1998p. *EQ6 Calculations for Chemical Degradation of Fast Flux Test Facility (FFTF) Waste Packages.* BBA000000-01717-0210-00028 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19981229.0081.

CRWMS M&O 1998q. *Software Qualification Report (SQR) Addendum to Existing LLNL Document UCRL-MA-110662 PT IV: Implementation of a Solid-Centered Flow-Through Mode for EQ6 Version 7.2B.* CSCI: UCRL-MA-110662 V 7.2b. SCR: LSCR198. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990920.0169.

CRWMS M&O 1999a. *Thermal Evaluation of the Enrico Fermi Co-disposal Waste Package.* BBAA00000-01717-0210-00014 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990325.0216.

CRWMS M&O 1999b. *Dose Calculations for the Co-Disposal WP of HLW Canisters and Fermi U-Mo Alloy SNF.* BBAC00000-01717-0210-00009 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990421.0152.

CRWMS M&O 1999c. *EQ6 Calculation for Chemical Degradation of Pu-Ceramic Waste Packages: Effects of Updated Materials Composition and Rates.* CAL-EDC-MD-000003 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990928.0235.

CRWMS M&O 1999d. *Enrico Fermi Fast Reactor Spent Nuclear Fuel Criticality Calculations: Intact Mode.* BBA000000-01717-0210-00037 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990125.0079.

CRWMS M&O 1999e. Not Used.

CRWMS M&O 1999f. Not Used.

CRWMS M&O 1999g. *Generic Degradation Scenario and Configuration Analysis for DOE Codisposal Waste Package.* BBA000000-01717-0200-00071 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19991118.0180.

CRWMS M&O 1999h. Not Used.

CRWMS M&O 1999i. *FY99 Criticality DOE SNF, 2101 9076 M3.* Activity Evaluation, March 5, 1999. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990330.0477.

CRWMS M&O 1999j. *LCE for Research Reactor Benchmark Calculations.* B00000000-01717-0210-00034 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990329.0394.

CRWMS M&O 1999k. *Laboratory Critical Experiment Reactivity Calculations.* B00000000-01717-0210-00018 REV 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990526.0294.

CRWMS M&O 1999l. *DOE Spent Nuclear Fuel (SNF) - 2101 2310 M1-M8.* Activity Evaluation, October 1, 1999. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19991001.0154.

CRWMS M&O 1999m. *EQ6 Calculations for Chemical Degradation of Enrico Fermi Spent Nuclear Fuel Waste Packages.* BBA000000-01717-0210-00029 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990702.0030.

CRWMS M&O 1999n. *Structural Calculations for the Codisposal of Enrico Fermi Spent Nuclear Fuel in a Waste Package.* BBA000000-01717-0210-00030 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19990505.0462.

CRWMS M&O 1999o. *Defense High Level Waste Disposal Container System Description Document.* SDD-DDC-SE-000001 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.19991217.0510.

CRWMS M&O 2000a. *DOE SNF Analysis Plan for FY 2000.* Development Plan TDP-EDC-MD-000003 REV 01. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000510.0169.

CRWMS M&O 2000b. *Waste Package Degradation Process Model Report.* TDR-WIS-MD-000002 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000328.0322.

CRWMS M&O 2000c. *Summary of In-Package Chemistry for Waste Forms.* ANL-EBS-MD-000050 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000217.0217.

CRWMS M&O 2000d. *Abstraction of Drift Seepage.* ANL-NBS-MD-000005 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000322.0671.

CRWMS M&O 2000e. *Enrico Fermi Fast Reactor Spent Nuclear Fuel Criticality Calculations: Degraded Mode.* CAL-EDC-NU-000001 REV 00. Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000802.0002.

DOE (U.S. Department of Energy) 1992. *Characteristics of Potential Repository Wastes.* DOE/RW-0184-R1. Volume 1. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: HQO.19920827.0001.

DOE (U.S. Department of Energy) 1998a. *Design Specification.* Volume 1 of *Preliminary Design Specification for Department of Energy Standardized Spent Nuclear Fuel Canisters.* DOE/SNF/REP-011, Rev. 0. Washington, D.C.: U.S. Department of Energy. TIC: 240539.

DOE (U.S. Department of Energy) 1998b. *Preliminary Design Concept for the Repository and Waste Package.* Volume 2 of *Viability Assessment of a Repository at Yucca Mountain.* DOE/RW-0508. Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.19981007.0029.

DOE (U.S. Department of Energy) 1999. *Fermi (U-Mo) Fuel Characteristics for Disposal Criticality Analysis.* DOE/SNF/REP-035, Rev. 0. Washington, D.C.: U.S. Department of Energy. TIC: 242461.

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

Nuclear Energy Agency 1998. *International Handbook of Evaluated Criticality Safety Benchmark Experiments.* NEA/NSC/DOC(95)03. Paris, France: Nuclear Energy Agency. TIC: 243013.

Parks, C.V.; Broadhead, B.L.; Hermann, O.W.; Tang, J.S.; Cramer, S.N.; Gauthey, J.C.; Kirk, B.L.; and Roussin, R.W. 1988. *Assessment of Shielding Analysis Methods, Codes, and Data for Spent Fuel Transport/Storage Applications.* ORNL/CSD/TM-246. Oak Ridge, Tennessee: Oak Ridge National Laboratory. ACC: NN1.19880928.0023.

Spahiu, K. and Bruno, J. 1995. *A Selected Thermodynamic Database for REE to be Used in HLNW Performance Assessment Exercises.* SKB Technical Report 95-35. Stockholm, Sweden: Swedish Nuclear Fuel and Waste Management Company. TIC: 225493.

Wolery, T.J. 1992. *EQ3/6, A Software Package for Geochemical Modeling of Aqueous Systems. Package Overview and Installation Guide (Version 7.0).* UCRL-MA-110662 PT I. Livermore, California: Lawrence Livermore National Laboratory. TIC: 205087.

YMP (Yucca Mountain Site Characterization Project) 1998. *Disposal Criticality Analysis Methodology Topical Report.* YMP/TR-004Q, Rev. 0. Las Vegas, Nevada: Yucca Mountain Site Characterization Office. ACC: MOL.19990210.0236.

9.2 CODES, STANDARDS, REGULATIONS, AND PROCEDURES

AP-2.21Q, Rev. 0, ICN 0. *Quality Determinations and Planning for Scientific, Engineering, and Regulatory Compliance Activities.* Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.20000802.0003.

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

AP-3.15Q, Rev. 1, ICN 2. *Managing Technical Product Inputs.* Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.20000713.0363.

AP-SI.1Q, Rev. 2, ICN 4. *Software Management.* Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.20000223.0508.

AP-SV.1Q, Rev. 0, ICN 1. *Control of the Electronic Management of Data.* Washington, D.C.: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. ACC: MOL.20000512.0068.

ASME (American Society of Mechanical Engineers) 1995. *1995 ASME Boiler and Pressure Vessel Code.* New York, New York: American Society of Mechanical Engineers. TIC: 245287.

ASTM A 240/A 240M-97a. 1997. *Standard Specification for Heat-Resisting Chromium and Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Pressure Vessels.* West Conshohocken, Pennsylvania: American Society for Testing and Materials. TIC: 239431.

ASTM A 516/A 516M - 90. 1991. *Standard Specification for Pressure Vessel Plates, Carbon Steel, for Moderate- and Lower-Temperature Service.* Philadelphia, Pennsylvania: American Society for Testing and Materials. TIC: 240032.

ASTM B 575-97. 1998. *Standard Specification for Low-Carbon Nickel-Molybdenum-Chromium, Low-Carbon Nickel-Chromium-Molybdenum, Low-Carbon Nickel-Chromium-Molybdenum-Copper and Low-Carbon Nickel-Chromium-Molybdenum-Tungsten Alloy Plate, Sheet, and Strip.* West Conshohocken, Pennsylvania: American Society for Testing and Materials. TIC: 241816.

ASTM G 1-90 (Reapproved 1999). 1990. *Standard Practice for Preparing, Cleaning, and Evaluating Corrosion Test Specimens.* West Conshohocken, Pennsylvania: American Society for Testing and Materials. TIC: 238771.

NLP-2-0, Rev. 5, ICN 1. *Determination of Importance Evaluations.* Las Vegas, Nevada: CRWMS M&O. ACC: MOL.20000713.0360.

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

9.3 SOURCE DATA

LL980711104242.054. Report of the Committee to Review the Use of J-13 Well Water in Nevada Nuclear Waste Storage Investigations. Submittal date: 08/05/1998.

SN9911T0811199.003. Calculations of Physical and Chemical Properties of Fast Flux Test Facility (FFTF) Waste Package. Submittal date: 11/15/99.

APPENDIX A

FERMI DOE SNF CANISTER

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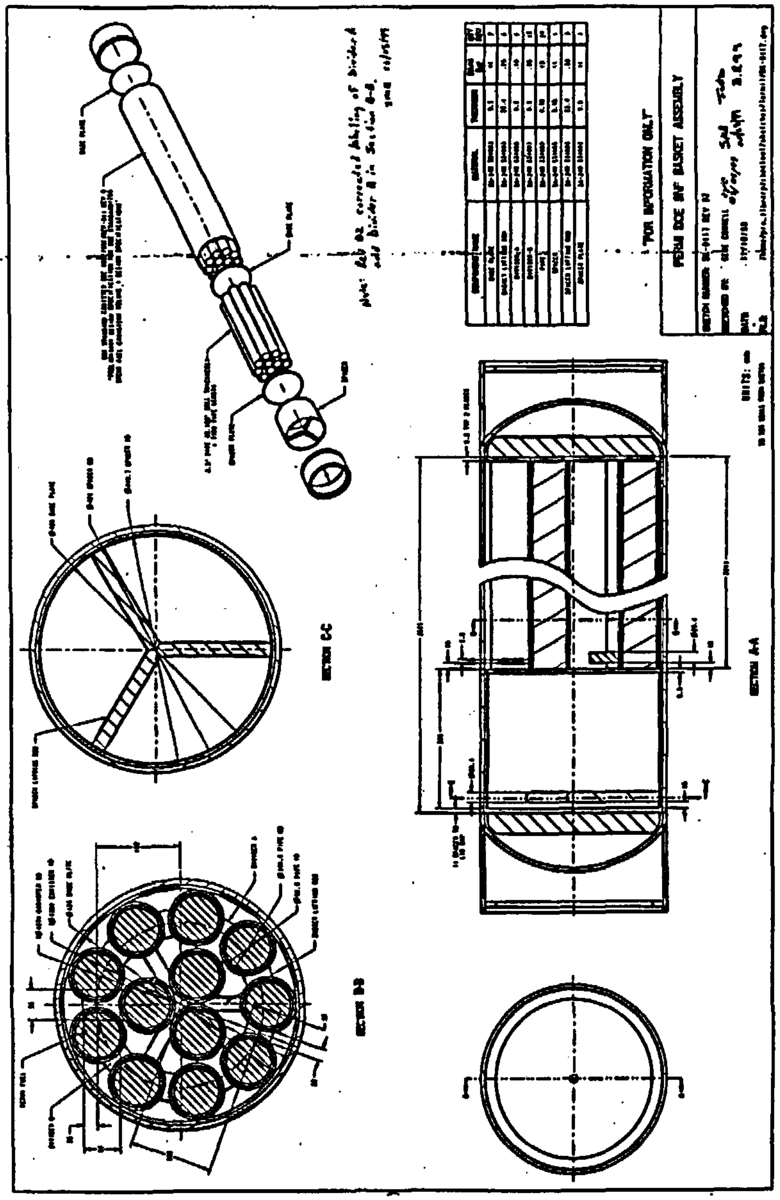

Figure A-1. Fermi DOE SNF Canister

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APPENDIX B

5-DHLW/DOE SPENT FUEL DISPOSAL CONTAINER

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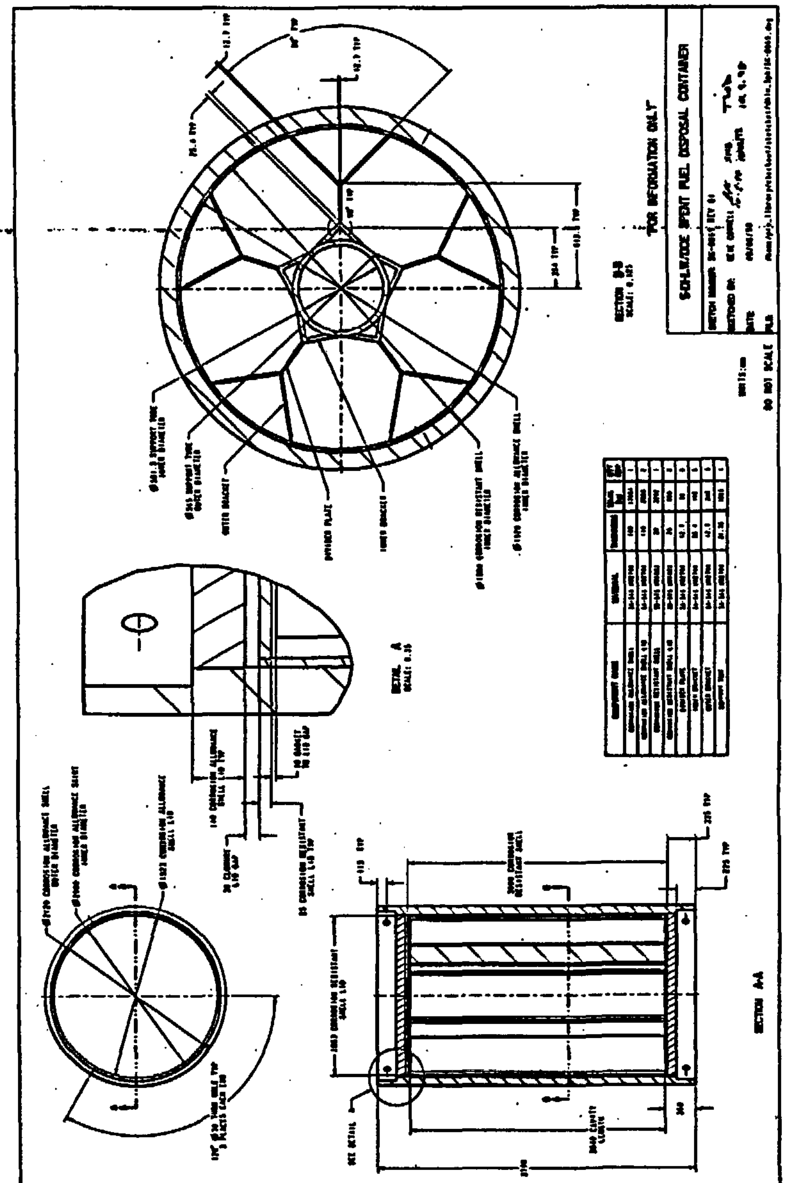

Figure B-1. 5-DHLW/DOE Spent Fuel Disposal Container

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