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Civilian Radioactive Waste Management System
Management & Operating Contractor

Evaluation of Codisposal Viability for Aluminum-Clad DOE-Owned Spent Fuel: Phase I Intact Codisposal Canister

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Prepared for:

U.S. Department of Energy
Westinghouse Savannah River Co.
Savannah River Site
P.O. Box 616
Aiken, SC 29802

Prepared By:

Civilian Radioactive Waste Management System
Management & Operating Contractor
1180 Town Center Drive
Las Vegas, NV 89134

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Prepared By: Thomas W. Doering
Petter Gottlieb, Preparer

Concurrence: _____ Date: 8-15-97
Donald A. Nitti, Checker

Approved By: _____ Date: 8.14.97
Thomas W. Doering, Manager
Waste Package Design

Approved By: Hugh A. Benton Date: 8/15/97
Hugh A. Benton, Manager
Waste Package Development

Approved By: Hugh A. Benton Date: 8/15/97
for Alden M. Segrest, Manager
MGDS Development

Evaluation of Codisposal Viability for Aluminum-Clad DOE-Owned Spent Fuel: Phase I - Intact Codisposal Canister

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Evaluation of Codisposal Viability for Aluminum-Clad DOE-Owned Spent Fuel: Phase I - Intact Codisposal Canister

1. Purpose

This evaluation is prepared by the Mined Geologic Disposal System (MGDS) Waste Package Development Department (WPDD) to provide an assessment of the vialibility of disposing of aluminum (Al)-based Department of Energy-owned research reactor spent nuclear fuel (DOE-SNF) in a codisposal waste package with five canisters of vitrified high-level waste (HLW). Figure 4.1.1-1 shows the DOE-SNF codisposal canister surrounded by five HLW canisters. Analyses were performed for criticality safety, structural strength, thermal limits, and the effect of DOE-SNF on the total waste package surface dose rates. The objective was to provide sufficient detail to establish the technical viability of the Aluminum-based DOE-SNF codisposal canister option. This report focuses on the DOE-SNF canister and on how it interfaces with the waste container and repository.

Two DOE-SNF fuel types were selected by the Alternative Technology Program of the Westinghouse Savannah River Company (Ref. 8.3) as representative of the range of variations (particularly with respect to criticality) found in Al-based research reactor fuels. These two fuel types were the high-enrichment Massachusetts Institute of Technology (MIT) reactor fuel and the medium-enrichment Oak Ridge Research (ORR) reactor fuel. The MIT fuel has an initial maximum enrichment of 93.5 weight percent U-235 and the ORR fuel has an initial maximum enrichment of 20.56 weight percent U-235. Criticality calculations were performed for intact fuel contained within the codisposal canister for fully flooded conditions as typically assumed as worst case for both transport and disposal. Thermal, structural and shielding analyses were also performed for intact fuel contained within the codisposal canister for repository conditions. Also, sufficient criticality analyses of the potential degraded states of MIT and ORR fuel within an intact codisposal canister basket were performed in order to establish the quantity of stainless steel/boron alloy needed to ensure subcriticality if the fuel degrades within an intact basket.

These studies constitute Phase I of the evaluation of aluminum-clad DOE owned spent fuel. Phase II will evaluate the possibility and probability of criticality in a more severely degraded mode, in which the fissile material could be released from the codisposal canister and reconfigured (with sufficient moderator) into a critical mass within in the waste package. Phase III will evaluate the possibility and probability of criticality in a still more degraded mode, in which the released fissile material is transported out of the waste package and accumulates in the drift or host rock of the repository. Phases II and III will not involve any further thermal, structural, or shielding analysis, so the technical viability shown in this Phase I report can be regarded as final. The criticality evaluations of Phases II and III may, however, identify needs for additional criticality control measures or reductions in the fissile mass loading of codisposal canisters..

2. Quality Assurance

The Quality Assurance (A) program applies to this document. The work reported in this document is part of the preliminary waste package (WP) design that will eventually support the MGDS License Application Design phase. This analytical work, when appropriately confirmed, can impact the proper functioning of the waste package; therefore, the waste package has been identified as an MGDS Q-List item important to safety and waste isolation (pp. 4, 15, reference 8.1). The waste package is on the Q-List by direct inclusion by the Department of Energy (DOE), without conducting a QAP-2-3 *Classification of Permanent Items* evaluation. The Waste Package Development Department responsible manager has evaluated this activity in accordance with QAP-2-0, *Conduct of Activities*. The analysis activities supporting this document are subject to *Quality Assurance Requirements and Description* (QARD; reference 8.2) requirements, as determined by an evaluation performed in accordance with QAP-2-0, *Conduct of Activities* (Perform Criticality, Thermal, Structural, and Shielding Analyses, 03/14/97). As specified in NLP-3-18, *Documentation of A Controls on Drawings, Specifications, Design Analyses, and Technical Documents*, this activity is subject to QA controls.

All design inputs which are identified in this document are for the preliminary stage of the WP design process; all of these design inputs will require subsequent confirmation (or superseding inputs) as the waste package design proceeds. This document will not directly support any OCRWM construction, fabrication, or procurement activity and therefore is not required to be procedurally controlled as TBV (to be verified). In addition, the inputs associated with this document are not required to be procedurally controlled as TBV. However, use of any data from this document for input into documents supporting construction, fabrication, or procurement is required to be controlled as TBV in accordance with the appropriate procedures.

The specific activities involved with the production and review of this document have been performed according to an approved Technical Document Preparation Plan (Ref. 8.39).

3. Method

The methodology used for these analyses is similar to that used for corresponding evaluations of commercial SNF. In most cases the same computer codes are used for corresponding analyses. The same regulatory requirements are used, wherever appropriate. Although Phase I is primarily concerned with the evaluation of the waste package design with respect to criticality of intact SNF, sufficient criticality analyses of the potential degraded states of MIT and ORR fuel within an intact codisposal canister basket were performed in order to establish the quantity of stainless steel/boron alloy needed to ensure subcriticality if the fuel degrades within an intact basket.

3.1 Neutronics

The nuclear reactivity of the codisposal canister within a waste package was analyzed with the MCNP4A computer code (reference 8.10), which was also used to compute waste package surface dose rates. The gamma, neutron, and thermal source strengths for shielding and thermal analyses were obtained from the SAS2H sequence of the SCALE 4.3 code system (reference 8.11).

The reactivity of the codisposal canister was evaluated for both intact MIT and ORR reactor DOE-SNF. In addition, the progressive degradation of the Al-clad fuel, due to aqueous corrosion, was analyzed within the codisposal canister. A scenario typical of those which can lead to criticality is described in Section 3.3, below.

Some degradation of the aluminum matrix will occur prior to corrosion breach of the codisposal canister,(e.g., deformation due to creep), which is being evaluated in ongoing Savannah River Site (SRS) programs. Such degradation may be significant for storage and transportation evaluations, but it will be overshadowed by the dissolution of a major fraction of the aluminum matrix after a few hundred years of aqueous attack following the breaking of the waste package and the codisposal container.

3.2 Thermal and Structural

The structural analyses were performed using the commercially available ANSYS 5.1 finite-element code. A finite-element model of the MIT-SNF codisposal canister was developed and analyzed for the bounding loads of the waste package tip-over design basis event (DBE). The results of this analysis were plotted in terms of displacement contours to determine at what location the displacements were large enough to cause the fuel assemblies to deform. The results of the finite-element method solutions were also analyzed in terms of the maximum stress contours to determine if the codisposal canister stresses exceed the material yield or ultimate tensile strength.

The thermal analysis method employed was two dimensional (2-D) finite element analysis (FEA) using ANSYS 5.1. The analysis used the repository drift temperature (previously determined by three-dimensional drift scale FEA of the repository thermal transient) as a boundary condition, and applied the heat loads in the HLW canisters and codisposal canister to determine the internal temperature distribution at various times. The waste package thermal calculation used steady state analysis to evaluate the temperatures at several different times after emplacement in the repository.

An effective thermal conductivity for the fuel assemblies was developed from the porosity (volume fraction of gas within the fuel assembly) and the thermal conductivities of aluminum metal and helium (the effect of the uranium alloy was neglected.)

The thermal model did not include the effect of convective cooling by the helium which fills the void space in the intact codisposal waste container. A simple estimate of this convective cooling was made in reference 8.20, using a rectangular cavity to model the approximately triangular areas between HLW canisters and the codisposal canister and between the HLW canisters and the waste package inside wall. This estimate used the standard textbook formula for the convective heat transfer in terms of the Prandtl number of the flow, the Rayleigh number, and the dimensions of the cavity, and it considered the heat transfer characteristics of the entire waste package including contact and conduction within and between internal components. The results showed that convection should, at best, increase the heat transfer rate by much less 10%, which would lower the peak temperature by only a few degrees. Therefore, the smeared heat transfer model can be viewed as reasonable.

The analyses presented in this document are based on the best data available at this time. Due to apparent inconsistencies in the thermal input data, SRS is currently assessing the accuracy of this information. When an update set of the thermal values become available a revision of the calculations is recommended.

3.3 Evaluation of Degradation Processes/ Scenarios

An important part of the methodology is to determine the likely degradation modes of the aluminum based fuel, and whether any are likely to lead to criticality. The Phase I analysis presented in this report is intended to evaluate only those degradation processes which still leave the SNF degradation products in the codisposal canister and then the waste container. A review of aqueous corrosion and repository infiltration rates has indicated that the waste package barriers of some of the waste containers may be penetrated by water in 3,000 to 10,000 years following emplacement. Within a few thousand years following this penetration, the codisposal canister can be expected to be breached, exposing the DOE-SNF to aqueous attack. The MIT and ORR fuel would be expected to degrade through oxidation within a few hundred years of breach of the DOE-SNF canister. Although the possible uranium and aluminum ions released from such degradation processes are generally insoluble, they are likely to form colloids which can be distributed throughout the water within the codisposal canister. Uranium and aluminum oxides in water have been observed to form hydrates with a gel-like appearance and an effective solid density of as low as 10% of the initial density.

Since the most uniform distribution of the high enriched uranium is the most reactive configuration, this study conservatively models the distribution of the neutronically significant material as being uniformly distributed throughout the DOE-SNF canister. The Al-based fuel forms will be assumed to degrade to a mix of hydrated Al and U oxides spread throughout the available volume of water, which results in a minimum solids (colloid particles) density of 35% by volume (reference 8.15). This distribution is homogeneous at the macroscopic level and will, therefore, be modeled in MCNP as uniform at the atomic level.

4. Design Inputs

All design inputs which are identified in this document are for the preliminary stage of the design process; all of these design inputs will require subsequent confirmation (or superseding inputs) as the codisposal canister and waste package designs proceed. This document will not directly support any CRWMS construction, fabrication, or procurement activity and therefore is not required to be procedurally controlled as TBV.

4.1 Design Parameters

4.1.1 Codisposal Waste Package

The codisposal waste package consisting of 5 HLW canisters surrounding a DOE-SNF codisposal canister is shown in Figure 4.1.1-1. The dimensions are from the conceptual design sketches included in references 8.15, 8.19, and 8.20. The barrier materials are typical of those used for commercial SNF waste packages.

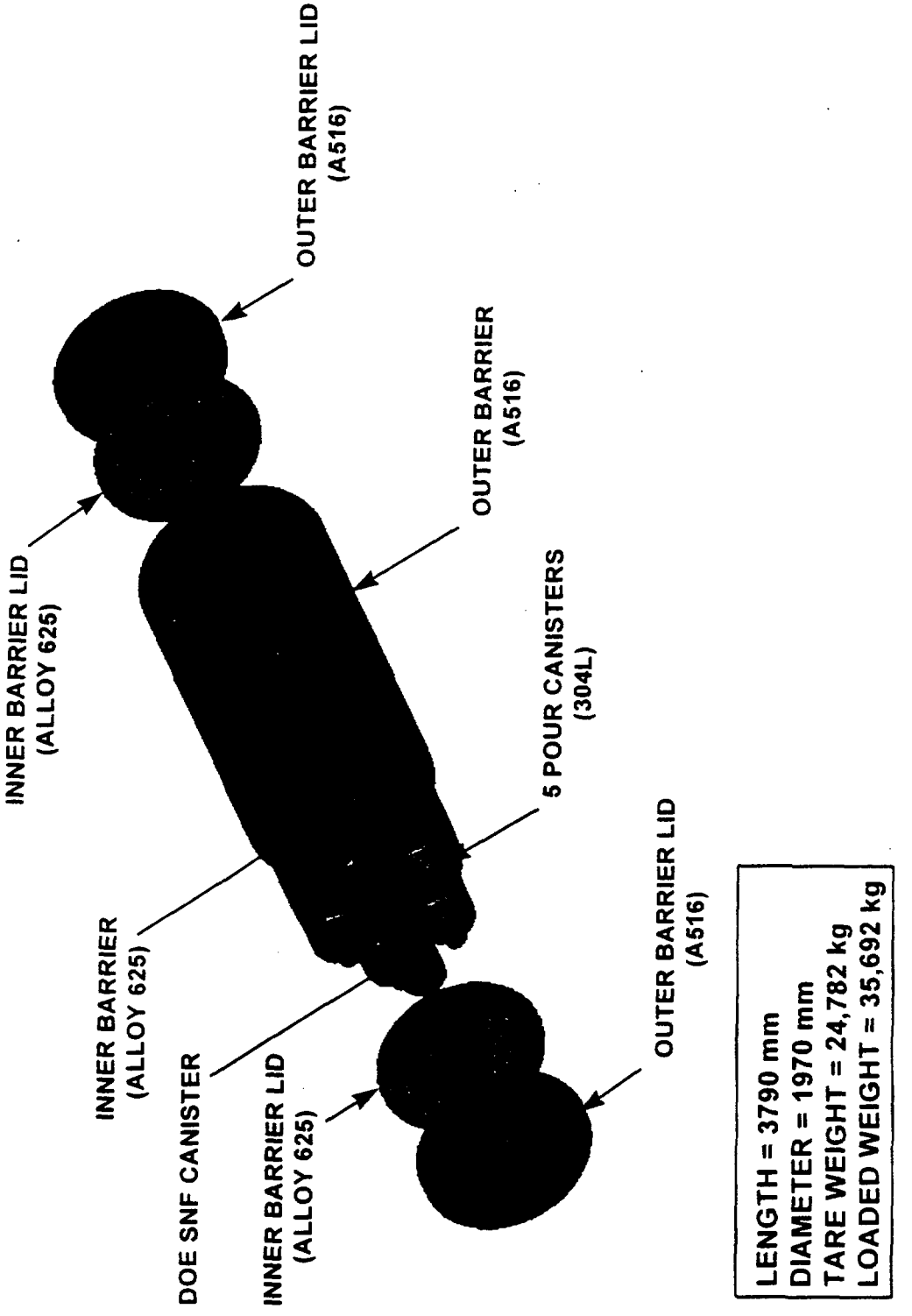

Figure 4.1.1-1 Codisposal Waste Package Assembly

4.1.2 Al-based DOE-SNF

4.1.2.1 Massachusetts Institute of Technology SNF

The characteristics of the MIT SNF were obtained from the MIT fuel Appendix A of reference 8.3. The geometry of the MIT plate/assembly were taken from drawings (R3F-3-2, R3F-1-4) provided by SRS as part of reference 8.3. The following description of the MIT assembly is supplemented by MCNP model shown in Figure 6.4.1.1-1. The MIT fuel assembly is constructed from 15 flat plates tilted at a sixty degree angle so that the resulting assembly has a rhomboidal (equilateral parallelogram with 60° acute angles) cross section, instead of the more common square or hexagon cross section. The MIT fuel length values used in these analyses are shorter than the original as-built length of the MIT assembly because the top and bottom ends of the assembly, which do not contain uranium materials, have been removed by cutting. The fuel plates consist of an aluminum cladding over an uranium/aluminum ($U\text{-}Al_x$) alloy. The maximum fuel mass for the MIT assembly is 514.25 grams of U-235 with an enrichment of 93.5 weight percent and one weight percent of U-234. The amount of aluminum present in the $U\text{-}Al_x$ alloy fuel meat is 30.5 weight percent. For the intact fuel neutronics model, the $U\text{-}Al_x$ fuel meat alloy is spread uniformly over the maximum space which the drawings indicate it could occupy, leaving a remaining void fraction of 0.6353 (Ref. 8.15), available for occupancy by water.

The conservative values of the burnup for the MIT fuel were derived from Appendix A data provided by SRS (Ref. 8.3). The maximum burnup for the MIT fuel was less than 8100 MWD/MTU. The shortest total time in reactor (including down time) to accumulate this burnup is 2517 days. The reactor power level is 9.68 MW/MTU.

Fuel Plates

The flat plates are 2.552 (+0.000, -0.002) inches wide, and 23 inches long. All 15 plates are the same and have a finned cladding surface with a total thickness of 0.080 ± 0.003 inches including a fin height of 0.010 ± 0.002 inches on both faces. The fuel alloy is 0.030 (+0.000, -0.002) inches thick, 2.177 (+0.000, -0.1875) inches wide, and 22.375 ± 0.375 inches long.

A = 0.03"
B = 0.025"

Fuel Element

The aluminum outer shroud which encloses the 15 fuel plates on 4 sides is a 2.405 inch outside dimension rhomboid with a 0.044 inch thick wall parallel with the fuel plates and a 0.188 inch thick comb plate at 60° to the fuel plates, and a nominal length (after cutting) of 23.368 inches. The fuel plates are centered within this rhomboid angled 60 degrees off the comb plate. The plates are fixed relative to each other by comb plates along two sides and the lip of the end fittings across the top and bottom. Drawing R3F-1-4 (Ref. 8.3) shows a fuel plate center-to-center spacing of 0.158", which is the spacing of the notches on the comb plates.

4.1.2.2 Oak Ridge Research SNF

Details of the construction of the ORR fuel element are contained in drawings M-11495-OR-001 ("19 Plate Fuel Element Assembly & Finish Machining", Ref. 8.3), M-11495-OR-003 ("Misc. Details for ORR Fuel Element", Ref. 8.3), and M-11495-OR-004 ("Fuel Plate Details" Ref. 8.3). The following description of the ORR fuel element is supplemented by the MCNP model shown in Figure 6.4.1.2-1. The element is constructed from 19 curved fuel plates which are held within two opposing aluminum comb plates. The ORR fuel length values used in these analyses are shorter than the original as-built length of the ORR assembly because the top and bottom ends of the assembly, which do not contain uranium materials, have been removed by cutting. The ORR fuel Appendix A (Ref. 8.3) contains the material information. The fuel plates consist of an aluminum cladding over an U-Si-Al fuel material. The maximum fuel mass for the ORR assembly is 347 grams of U-235 with an enrichment of 20.56 weight percent. The uranium present in the U-Si-Al alloy is 77.5 weight percent. There are 2 atoms of Si per 3 atoms of U, and Al fills out the bulk of the fuel material. As with the MIT SNF, the degraded U-Si-Al SNF can be distributed throughout the free volume; in this case, however, the volume remaining for occupancy by water is 0.4064 (Ref. 8.15).

Fuel Plates

The curved plates are manufactured as flat laminated sheets that are formed to the 5.5 inch inner radius of curvature. Seventeen of the plates are inner plates, with a thickness of 0.0494 to 0.0510 inches and a 0.0105 inch minimum aluminum cladding on both sides of a 0.020 inch nominal fuel foil, which is assumed to have a tolerance of 0.005 inches since this is the default for the drawing; these plates are 2.7955 (minimum) to 2.7985 (maximum) inches wide. Two of the plates are outer plates, with a thickness of 0.063 to 0.066 inches, with a 0.018 inch minimum cladding on both sides of a 0.020 inch nominal fuel foil these plates are 2.7925 (minimum) to 2.7955 (maximum) inches wide. The fuel foil is not as wide as the aluminum cladding (drawing M-11495-OR-003), so, for purposes of the MCNP model, an aluminum strip is used to close each side of the finished fuel plate. For the inner fuel plates, the width of the fuel foil allows a 0.126 to 0.200 inch inset from the edge of the plate on both sides. The overall length of the inner fuel plate is 24.620 to 24.630 inches and the fuel foil is centered within the plate longitudinally, with an inset at each end of 0.318 to 0.775 inches. For the outer fuel plates, the width of the fuel foil allows a 0.126 to 0.198 inch inset from the edge of the plate on both sides. The overall length of the outer fuel plate is 27.120 to 27.130 inches and the fuel foil is centered within the plate longitudinally, with an inset at each end of 1.574 to 2.011 inches. The top and bottom ends of the inner and outer fuel foils are chamfered, but this trimming of the fuel material was neglected.

Fuel Element

The aluminum comb plates enclose the 19 fuel plates on 2 sides giving a cross-section bounded by a rectangle having the approximate dimensions of 3.25 inch by 3.00 inch, and a nominal length (after cutting) of 27 1/8 inches. The fuel plates are centered within this box, and form a nearly square fuel/water region bounded by the 3.169 inch longitudinal comb plate width. The geometric arrangement is shown in Figure 6.4.1.2-1. The plates are fixed relative to each other by comb plates along two sides and by a comb strap across the top and bottom. Drawing M-11495-OR-003

("Misc. Details for ORR Fuel Element" Ref. 8.3) shows a fuel plate edge-to-edge spacing of 0.166", which is the spacing of the notches on the comb plates.

4.1.3 Structural

A two-dimensional (2-D) finite-element model of the MIT-SNF codisposal canister cross-sectional was developed in order to evaluate the effects of the tipover dynamic load on the canister structural components.

Material properties (see Assumption 4.3.9):

- For 316L stainless steel: Density = 7953 kg/m^3 (Ref. 8.30, p. 5); Poisson's ratio = 0.298 (Ref. 8.31, p. 755)(assumption 4.3.4 in the structural design analysis Ref. 8.19); Modules of elasticity = 195 GPa (Ref. 8.9, Table TM-1); Tensile Strength = 482 MPa (Ref. 8.9, Table U); Compressive Strength = 1358 MPa (Ref. 8.32, p. 34)(assumption 4.3.10 in the structural design analysis Ref. 8.19); Yield strength = 172 MPa (Ref. 8.9, Table Y-1); Elongation % in 2 in. = 40 (Ref. 8.30).
- For 304L stainless steel: Poisson's ratio = 0.29 (Ref. 8.31, p.755); Modulus of elasticity = 195 (Ref. 8.9, Table Y-1 and Table TM-1).
- For XM-19 stainless steel (oxidized by repeated heating): yield strength = 380 MPa (Ref. 8.33, p. 153), used in the alternative design discussed in Section 6.5.3.1.

Masses of 4-canister Defense High Level Waste (DHLW) waste package members (Ref. 8.8, p. II-320) (see Assumption 4.3.11) are provided below for a half-symmetry model:

- Mass of outer barrier and outer barrier lids = 5079.99 kg (10160 kg for a full-size canister)
- Mass of inner barrier, inner barrier lids, and canister guide = 1666.99 kg (3334 kg for a full-size canister)
- Mass of Savannah HLW Canister = 1000.01 kg (for 2 of the 4 total HLW canisters) (500 kg for 1 of the 4 total HLW canisters)
- Mass of vitrified waste = 3363.96 kg (for 2 of the 4 total HLW canisters) (1682 kg for 1 of the total 4 canisters)

A bounding mass value is used in the *g* load calculations for the canistered waste form:

- Total mass of the HLW canister = 2500 kg (Ref. 8.34)
- Outer diameter of HLW canister = 0.61 m (Ref. 8.7, p. 3.1-7)
- Mass of one MIT-SNF assembly = 2.8 kg (MIT Appendix A, Ref. 8.3, p. 5)

4.1.4 Thermal

Values for the thermal conductivities of stainless steel 316L, stainless steel 304L, XM-19, Alloy 625, and A 516 were obtained from Table TCD, Section II of the 1995 ASME Boiler and Pressure Vessel Code (Ref. 8.9). Since these stainless steel materials have similar thermal properties, stainless steel 316 is chosen to represent any stainless steel (316L, 304L and XM-19) used in the WP. The emissivity of 316L, 304L and XM-19 are 0.60, and the emissivities of Alloy 625 is 0.80 (p. 4-68, Ref. 8.25).

The thermal conductivity of borosilicate glass is 1.1 W/mK (Table 11.7, p. 584, Ref. 8.26), and the temperature dependent thermal conductivity of helium is taken from p. A17 of reference 8.27.

4.2 Design Criteria

The design of the engineered barrier segment (EBS) will depend on neutronic, structural, and thermal analyses of the repository waste package. Criteria that relate to the analysis of the EBS are derived from the applicable requirements and planning documents. Upper-level systems requirements are provided in the Mined Geologic Disposal System Requirements Document (MGDS-RD) (Ref. 8.4). The requirements flow down to the Engineered Barrier Design Requirements Document (EBDRD, Ref. 8.5) as specific requirements for engineered barrier segment design. The Controlled Design Assumptions Document (CDA, Ref. 8.6) provides guidance for requirements listed in the EBDRD which have unqualified or unconfirmed data with the requirement. The criteria applicable to analyses of waste package emplacement are equivalent to the applicable requirements, interface requirements, and criteria cited in the EBDRD.

The "TBD" terms identified in the available criteria in this section will not be carried to the conclusions of this document based on the rationale that the conclusions derived by this analysis are for preliminary design that will not be used as input into OCRWM documents supporting construction, fabrication, or procurement.

The following criteria are applicable to the design subject. Each criterion references the relevant EBDRD (Ref. 8.5) requirement from which it has been derived; however, it is not the intent of these analyses to show direct compliance with the referenced requirements from the EBDRD. Rather, they are used as guidelines and design goals for the preliminary design.

Structural:

4.2.1 The MIT-SNF codisposal canister will be designed so that the physical and mechanical properties of the codisposal canister will be sufficient to maintain the structural integrity of the fuel assembly against dynamic loads. This document investigates the results of a tip-over onto an essentially unyielding surface. These considerations are addressed throughout this document. [EBDRD 3.7.1.B][EBDRD 3.7.1.H]

4.2.2 The internal structure of the MIT-SNF codisposal canister will be configured to accommodate the spent fuel waste form, provide stability of the waste form, and withstand handling loads such as the tip-over event. The resistance of the canister to a tip-over event is analyzed. [EBDRD 3.7.1.3.B]

Thermal:

4.2.3 The design of waste packages shall consider the thermal effects and thermal loads. [EBDRD 3.7.1.B].

4.2.4 Limit the temperature of the high-level waste (HLW) glass to less than 400°C during storage at the producer sites and during transport to the repository. [CDA DCWP 002]

Neutronic:

4.2.5 Criticality Control

The EBDRD requirements 3.2.2.6 and 3.7.1.3.A (Ref. 8.5) both indicate that a WP criticality shall not be possible unless at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. These requirements also indicate that the design must provide for criticality safety under normal and accident conditions, and that the calculated effective multiplication factor (k_{eff}) must be sufficiently below unity to show at least a five percent margin after allowance for the bias in the method of calculation and the uncertainty in the experiments used to validate the methods of calculation.

Contrary to what ENGB wants

CDA Assumption EBDRD 3.7.1.3.A (Ref. 8.6, p. 4-32) clarifies that the above requirement is applicable to only the preclosure phase of the MGDS, in accordance with the current DOE position on postclosure criticality. This assumption also indicates that for postclosure, the probability and consequences of a criticality provide reasonable assurance that the performance objective of 10CFR60.112 is met. While the NRC has not yet endorsed any specific change for postclosure, they have indicated that they agree that one is necessary

Finally, EBDRD 3.3.1.G indicates that "The Engineered Barrier Segment design shall meet all relevant requirements imposed by 10CFR60." The NRC has recently revised several parts of 10CFR60 which relate to the identification and analysis of design basis events (Ref. 8.36) including the criticality control requirement, which was moved to 60.131(h). These changes are not reflected in the current versions of the EBDRD or the CDA. The change to the criticality requirement simply replaces the phrase "criticality safety under normal and accident conditions" with "criticality safety assuming design basis events."

This document contributes to satisfying the above requirement for preclosure by demonstrating that the intact codisposal canisters for MIT and ORR fuel will remain subcritical, assuming a five percent margin and allowing for bias and uncertainty in the method of calculation, during the WP flooding event defined in the WP Design Basis Events analysis (Ref. 8.37). The misload events discussed in that analysis are not applicable in this case, as the codisposal canisters are specifically designed for the unique physical forms of the MIT and ORR fuel, and do not take credit for burnup.

4.2.6 Shielding

EBDRD requirement 3.2.4.5 indicates that allocation of shielding requirements to the WP, if any, is TBD. The CDA has clarified this TBD in Key Assumption 031, by indicating that the WP shielding criteria should be as follows:

- WP containment barriers will provide sufficient shielding for protection of WP materials from radiation enhanced corrosion,
- Individual WPs will not provide any additional shielding for personnel protection, and,

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

This document contributes to satisfying the above criteria by demonstrating that the dose rate at the surface of the WP will not be increased by the presence of the DOE-SNF codisposal canister and will not result in significant corrosion enhancement of the outer barrier.

4.3 Design Assumptions

Based upon the rationale that the conclusions derived in this document are for preliminary design and will not be used as input into documents supporting construction, fabrication, or procurement, a TBD (to be determined) or TBV (to be verified) will not be carried to the conclusions to this document.

The assumptions used in this document are:

4.3.1 The codisposal waste package contains 16 MIT or 10 ORR DOE-SNF assemblies in the basket cross section, and assemblies are stacked four high within each position in the fuel basket for a total of 64 MIT or 40 ORR assemblies. This is the maximum number of assemblies of each type which can physically fit in the DOE-SNF canister. This assumption is used in Section 4.1 and throughout Section 6.

4.3.2 The aluminum cladding of the DOE-SNF is limited to 204°C (400°F), which was established as the limit for the storage of aluminum clad HFIR fuel, because it is related to the onset of accelerated creep (Ref. 8.40). The creep temperature limit is far below the melting temperature of aluminum (660°C). The thermal criteria indicated in Section 4.2 are assumed to apply to the thermal analysis as thermal goals for the HLW glass canister design. Although criteria 4.2.4 does not address the temperature limit for the HLW glass in the repository, it is chosen as a reference for this analysis. This assumption is used in Section 7.1.4.

4.3.3 The MIT and ORR fuel is assumed fresh (unburned) for criticality calculations. The National Spent Nuclear Fuel Program Criticality Team recently came to the consensus opinion that the benefit gained from burnup credit would not be significant enough to pursue for DOE SNF because of cost and lack of qualified data (Ref. 8.35). This assumption is used in Section 6.5.1.

4.3.4 The waste package is assumed to be fully flooded with water for criticality calculations. This is the most reactive condition and is conservative. This assumption is used in Section 6.4.1

4.3.5 The waste package is assumed to be filled with air for shielding and structural calculations. This assumption is used throughout Sections 6.4.2 and 6.4.5.

4.3.6 The waste package is assumed to be filled with helium for thermal calculations. Heat is a concern for the interior of the waste package only during the early years after emplacement, and the helium fill gas can be expected to be present during these times. This assumption is used in Section 6.4.4.2

4.3.7 It is assumed that credit can be taken for only 75% of the B-10 in any boron neutron absorber. The basis for this assumption is that the NRC typically allows credit for only 75% of the boron, unless content and uniform coverage can be verified by measurement. This assumption is used in Section 6.4.1.5.

4.3.8 The Savannah River HLW canister is assumed to be representative for HLW canisters. Reference 8.7 specifies the geometry and materials of construction. The outer diameter is 0.6095 m and the thickness is 0.009525 m. The canister inside volume is 0.736 m^3 and the glass weight is 1682 kg. The glass loading in each canister is 85% of the total volume. The basis for this assumption is that the specified reference is the best information available concerning the HLW canister design. This assumption is used throughout Section 6.

4.3.9 The tip-over accident event considered in this document includes the time period from placement of the codisposal canister into the codisposal waste package to the emplacement of the waste packages into the drift. Since the increase in the canister temperature is not anticipated to be significantly different than the room temperature in this time period, room temperature (20°C) material properties were assumed in structural analyses. This assumption is used throughout Section 6.4.3.

Perhaps this is incorrect assumption

4.3.10 The *g* load acted upon the codisposal canister by one of the HLW canisters is conservatively assumed to be transmitted through the basket assembly at a 45° angular orientation to the long parallel members (see Figure 6.3.1-1). The goal is to analyze the bounding case for the most critical stresses and deformations. This assumption is used in Section 6.4.3.2.

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4.3.11 The external force on the MIT-SNF canister was calculated using the results of a previous design analysis performed for the tip-over evaluations of the 4-canister DHLW waste package. This document was issued as a non-quality affecting document and is listed in Section 8 (Ref. 8.8). This force results in a 104*g* deceleration for the codisposal waste package. The use of "104*g* dynamic load" resulting from a 4-canister DHLW is a conservative assumption for the structural evaluations of the codisposal canister since the surface of impact is an essentially unyielding surface. The current regulation imposed by the Nuclear Regulatory Commission requires the assumption of an unyielding surface for such a design basis event. We are working with the NRC to incorporate realistic surface compliance. Ultimately, a tip over event including energy absorbing aspects of the impact surface, such as flexure, will result in lower values of *g* load on the codisposal canister. This assumption is used in Sections 4.1 and 6.4.3.

4.3.12 The minimum clearance provided between the fuel assemblies and the basket members is 1.72 mm. The basis for this assumption is the design dimensions of the basket and the

dimensions of the SNF, and their respective tolerances. This assumption is used in Sections 6.4.3.1 and 7.1.3.

4.3.13 The repository thermal loading of 83 MTU/acre is considered for the preliminary design. This value is consistent with the thermal loadings (80 - 100 MTU/acre) given in the CDA (Key 019, Ref. 8.6). Note that thermal loading is actually expressed as an area mass loading (AML) which is most representative of the integrated heat (total energy) which will be deposited in the repository by any particular source unit (e.g., waste package or assembly). Heat per unit area is highly variable, depending strongly on the age of the SNF at the time of emplacement. This assumption is used throughout Section 6.4.

4.3.14 The waste package will be emplaced in-drift in a horizontal mode. This is consistent with CDA Key 011 and Key 066, reference 8.6. This assumption is used throughout Section 6.

4.3.15 The radiation and heat sources from the MIT SNF are taken at 5 years cool time (after discharge from reactor) and at 0 years (time of pour) for the HLW canisters. The basis for the SNF cooling time is to correspond to the minimum age for acceptance of commercial SNF. The decay heat for Savannah River glass (see Section 6.7.4) is assumed to be representative of the HLW glass. This assumption is used throughout Sections 6.4.2 and 6.4.4. There no similar assumption for ORR fuel because there was no thermal analysis for this type of fuel.

4.3.16 The effects of drift backfilling will not be considered for the repository base case analysis. This assumption is consistent with CDA Key 046. This assumption is used throughout Section 6.

4.3.17 The MIT SNF assemblies are modeled with smeared properties with effective thermal conductivity. The fuel is treated as a mixture of aluminum and air. The porosity of the mixture can be calculated using the volume ratio of the void to the total assembly according to the dimensions shown in drawings R3F-1-4 and R3F-3-2, reference 8.3. The volume of the metal portion is $(2.2+0.01)\times 0.07\times L$(assembly length)$\times 15+2.38\times 0.188\times L\times 2=3.215\cdot L$ in^3; and the total volume of the assembly is $2.38\times 2.75\times L=6.545\cdot L$ in^3. Thus, the porosity of the mixture is $(6.545\cdot L-3.215\cdot L)\div 6.545\cdot L=0.509$. Using the Maxwell formula (see below) for packed beds (p. 130, Ref. 8.28), and applying the porosity of the mixture of 0.509, the thermal conductivity of aluminum 6061 of 180 W/m·K (p. 72, Ref. 8.29), and the thermal conductivity of the helium of 0.152 W/m·K (at 300 K, p. A17, Ref. 8.27), the effective thermal conductivity of the MIT fuel is obtained as 70.56 W/m·K.

$$k_e = \frac{2(k_s/k_f)^2(1-\epsilon)+(1+2\epsilon)(k_s/k_f)}{(2+\epsilon)(k_s/k_f)+1-\epsilon}\cdot k_f$$

where, k_e is the
effective thermal conductivity of the MIT fuel; k_s is the thermal conductivity of the

aluminum metal; k_f is the thermal conductivity of the air; and ε is the porosity of the MIT assembly (p. 130, Ref. 8.28). This assumption is used throughout Section 6.

4.3.18 The thermal analysis modeling considered conduction and radiation heat transfer. This provides conservative results for this analysis. The basis for this assumption is as follows: the helium fill gas has a very low buoyancy so thermally driven convective heat transfer will have a small or negligible contribution to the total heat transfer. Thus, the problem may be modeled with only the dominant heat transfer modes with a negligible or conservative impact upon the results (see Section 3.2). This assumption is used throughout Section 6.4.

4.3.19 It is assumed for this analysis that the waste package with disposal container will not have filler material placed inside of it. The basis for this assumption is that the consideration of degraded scenarios outside the DOE-SNF codisposal container which might require filler material is beyond the scope of this analysis. The analysis to be performed in Phase II may indicate that filler is required. This assumption is used throughout Section 6. If filler is required, the thermal and structural evaluations will be affected.

4.3.20 The DHLW-waste package surface temperatures for the 4-canister WP (time dependent), as documented in reference 8.17, will be applied as ~~the boundary~~ conditions for the detailed 2-D waste package analysis. The analysis described in reference 8.17 considers multiple WPs in the drift with different WP heat generation rates. The WP surface temperature used in this analysis is selected from the DHLW WP with 4 HLW canisters at the thermal loading of 83 MTU/acre. The thermal load on the repository is determined primarily by the commercial SNF waste packages (which have a heat rate two orders of magnitude larger than the DHLW waste packages). The DHLW waste packages alternate in position along the drift axis with the commercial SNF waste packages. In this manner, the DHLW waste package surface temperature is determined primarily by the heat generation rate of its nearest neighbor commercial SNF waste packages. The use of this surface temperature as the boundary condition may slightly under-estimate the peak internal temperatures of the codisposal WP since 5 HLW canisters are used in this design. For this evaluation, the effect is judged to be negligible. According to the temperature results listed in reference 8.17, the waste package surface temperature is very close to the drift wall temperature after 10 years. This means the surface temperature of the 4-canister DHLW WP is mostly driven by the drift wall temperature, and the decay heat at this time (much smaller than initial heat) has little effect on the surface temperature of the 4-canister DHLW WP. Since the peak internal temperatures of the MIT-SNF and the glass matrix are expected to peak after 10 years (Ref. 8.23); the peak temperature results will be reasonable values. The basis for this assumption is engineering judgement. This assumption is used throughout Section 6.4.4.

Why delete text like to 5 pack configurations

4.4 Codes and Standards

Not Applicable. Preliminary design of the waste package and codisposal canister is not controlled by codes and standards. The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Ref. 8.9) has been used only as a reference for the structural and thermal properties of materials used within the codisposal canister and waste package.

5. Use of Computer Software

5.1 Scientific and Engineering Software

The calculation of nuclear reactivity of fresh fuel configurations was performed with the MCNP4A computer code which is identified with the Computer Software Configuration Item (CSC): 30006-V.A. MCNP4A calculates k_{eff} for a variety of geometric configurations with neutron cross sections for elements and isotopes described in the Evaluated Nuclear Data File version B-V (ENDF-B/V). MCNP4A is appropriate for the fuel geometries and materials required for these analyses. The calculations using the MCNP4A software were executed on a Hewlett-Packard 9000 Series 735 workstation with the HP-UX 9.x operating system. The software qualification of the MCNP4A software, including problems related to calculation of k_{eff} for fissile systems, is summarized in the Software Qualification Report for the Monte Carlo N-Particle code (Ref. 8.10). The MCNP4A evaluations performed for this design are fully within the range of the validation for the MCNP4A software used. Access to and use of the MCNP4A software for this analysis was granted by Software Configuration Management and performed in accordance with the QAP-SI series procedures.

An allowance for calculational bias and experimental uncertainties in criticality benchmark calculations must be made per the requirements listed in Section 4.2. Forty seven criticality benchmark calculations representative for research reactor fuel were run based on reviewed experiments and MCNP models (Ref. 8.22). The sum of bias and uncertainty is less than 0.02 in k_{eff} for all cases (Ref. 8.15)

The calculation of the neutron, gamma, and thermal sources in spent MIT fuel was performed with the SAS2H code sequence (Ref. 8.11), which is a part of the SCALE 4.3 code system (CSC: 30011 V4.3). SAS2H is designed for spent fuel depletion calculations to determine spent fuel isotopic content (including radioisotopes which produce alpha particles), decay heat rates, and radiation source terms. Thus, SAS2H is appropriate for the generation of thermal and radiation sources for the calculations of this analysis. The calculations using the SAS2H software were executed on a Hewlett-Packard 9000 Series 735 workstation with the HP-UX 9.x operating system. The software qualification of the SAS2H software, including problems related to generation of isotope contents, is summarized in the Software Qualification Report for the SCALE Modular Code system (Ref. 8.11). The SAS2H evaluations performed for this design are fully within the range of the validation for the SAS2H software used. The associated 44BURNUPLIB cross section library was used for these calculations. Access to and use of the SAS2H software for this analysis was granted by Software Configuration Management and performed in accordance with the QAP-SI series procedures.

The finite element analysis computer code used for this analysis is ANSYS Version (V) 5.1 (CSC: 30003 V5.1HP) and was obtained from Software Configuration Management in accordance with QAP-SI-0 and QAP-SI-3. ANSYS is a commercially available finite element thermal and mechanical analysis code and is appropriate for the thermal analysis of waste packages, waste package emplacements, and waste package environments as utilized in this analysis. The analyses using the ANSYS software were executed on a Hewlett-Packard 9000 Series 735 workstation with the HP-UX 9.x operating system. The software qualification of the ANSYS software, including

problems of the type analyzed in this report, is summarized in the Software Qualification Report for ANSYS Version 5.1HP (Ref. 8.12). The ANSYS evaluations performed for this design are fully within the range of the validation for the ANSYS V5.1 code used. Access to and use of the code for the analysis granted and performed in accordance with the ANSYS V5.1 Life Cycle Plan (Ref. 8.13) and the QAP-SI series procedures.

5.2 Computational Support Software

The 2-D cross section model was generated with Pro/Engineer solid modeler Version 17.0. Pro/Engineer was executed on a Hewlett-Packard 9000 Series 735 workstation. Pro/Engineer Release 17.0 is not a controlled computer code and has not been qualified under the QAP-SI series of M&O procedures and will not be qualified under the M&O procedures (not required per QAP-SI-0). Pro/Engineer Version 17.0 simply provides a 2-D geometry for the use in the finite element analysis.

The data interpolation for MIT SNF heat load and computation of number densities of intact and degraded states were performed with Microsoft Excel Version 5.0. Microsoft Excel 5.0 was executed on an IBM PC compatible personal computer. Microsoft Excel Version 5.0 simply provides data manipulation for the analyses.

The presentation graphics provided in Section 6.4 was generated with the computer code Harvard Graphics Version 2.0 and is classified as computational support software. Harvard Graphics Version 2.0 was executed on an IBM PC compatible. Harvard Graphics Version 2.0 simply provides a framework to create a graphical representation of data. No calculation or modification beyond cut and paste operations with tabular ANSYS or Lotus 1-2-3 output was performed in Harvard Graphics

The AutoSketch Version 2.0 graphics package was used for the conceptual design layout of the MIT and ORR SNF codisposal baskets. AutoSketch is a simplified version of the AutoCAD software system which is appropriate for sketches.

Note: cladding, fuel + M/D degradation may lead to criticality

6. Design Analysis

6.1 Background

As part of an engineered barrier system for the containment of radio nuclides, the codisposal WP k_{eff} must not exceed 0.95 during the pre-closure phase. Further, potential degradation of the aluminum clad, U-Al_x (or U-Si-Al) metal fuel plates must not cause the reactivity of the fuel to exceed 0.95 while it is contained within the codisposal canister. Degradation of the fuel will not occur while the WP is intact; however, oxidation of the aluminum cladding and fuel alloy would occur at a much faster rate than degradation of the codisposal basket if the WP were breached. The codisposal baskets for MIT fuel and ORR are both evaluated in the intact configuration. In addition, enough degraded fuel cases have been completed to determine the amount and distribution of borated stainless steel required to be placed into the intact configuration to prevent criticality within the DOE-SNF codisposal canister.

The scenarios analyzed included:

Intact - Conceptual designs of baskets suitable for transport/storage/disposal. The intent was not to design a transport basket but rather to design a basket which would be representative of the types of transport basket which may be developed for DOE-SNF. A fully flooded condition is analyzed for both MIT and ORR fuel in their respective baskets within the Waste Package.

Degraded within codisposal canister - potential progressive degradation of fuel with all the degradation products remaining within the codisposal container.Optimum moderation was evaluated by varying the water content of the fuel alloy and surrounding moderator volume.

The progressive degradation of the fuel was evaluated in stages as follows:

1. Homogenize fuel plates and inter-plate moderator volume.
2. Homogenize entire assembly (fuel plates plus structural combs plus water).
3. Disperse homogenized material throughout basket free space.

6.2 Conceptual Design of Codisposal Canister

Conceptual designs for the baskets for MIT and ORR fuel types were prepared to serve as the basis for the criticality, shielding, structural and thermal analyses. The conceptual designs are intended to be representative of baskets which could be transported and disposed of at the repository. The analyses which were performed address the disposal of the aluminum clad fuel and do not evaluate transport; rather, design practices for spent fuel shipping casks were applied to the disposal canister. These design practices include:

- Structural load paths should be straight and continuous from one side of the basket to the other. This practice can be applied to the ORR fuel type due to its square shape but it is not possible to maintain a continuous load path for one axis of the MIT SNF basket due to the

parallelogram shape of the MIT fuel. This discontinuity in the load path requires the structural steel of the central fuel slot to carry structural loads without the benefit of a vertical support.

- Elastic-only structural analyses are required for transport baskets, but elastic-plastic analysis methodologies are permissible for storage and the elastic-plastic methodology is applied to the disposal canister. The elastic-only requirement for transport encourages the use of the thickest possible structural members throughout the basket. This practice is applied to the MIT SNF design by using thicker structural members at the outer periphery of the basket where space permits. In addition, two of the MIT assemblies in the 16-position conceptual design have been rotated to create a more space-efficient array so that these assemblies can be moved outboard and the central structural plates of the basket can be thickened. The ORR basket design consists of relatively thick structural tubes or egg-crate plates which provide structural strength.

- The use of neutron absorber materials in transport packages is limited to a 75% credit for the minimum boron content of the absorber panels in lieu of 100% inspection of the absorber panels with a neutron transmission test. A similar design practice has been established for disposal, and the criticality analyses of this report use the 75% value.

- Heat transfer paths should be uninterrupted wherever possible. This practice has been applied to the MIT SNF basket design since it is intended to be manufactured as machined components to create basket sections. The MIT SNF basket heat transfer paths should not be interrupted by gaps or manufacturing joints. The ORR conceptual basket could be fabricated from an assemblage of square tubes.

6.3 Codisposal Canister and Waste Package Arrangements

6.3.1 MIT SNF Codisposal Basket Conceptual Design

The MIT SNF codisposal basket consists of plates formed into parallelogram shaped slots in a steel disk that provide structural support for the SNF. Panels of stainless steel/boron 0.254 cm thick are attached to one side of each slot to provide neutron attenuation between the slots. Stainless steel/boron in-row separator plates, 0.213 cm thick, are provided between adjacent pairs of MIT SNF assemblies to reduce neutronic interactions between adjacent assemblies. The method of attachment of these in-row separator plates has not been evaluated in detail. Figure 6.3.3-3, illustrates how the unit basket is stacked in four layers along the axis, with the individual layers separated by between-layer separator plates of borated stainless steel 1 cm thick. The basket has void regions around the periphery of the basket to reduce the weight of the structure. Heat transfer is provided by the structural steel internal to the DOE-canister as is illustrated in Figure 6.3.1-1. The rhomboidal slots provide a 1.72 mm clearance around the MIT assembly. The inner radius of the codisposal canister is 20.465 cm.

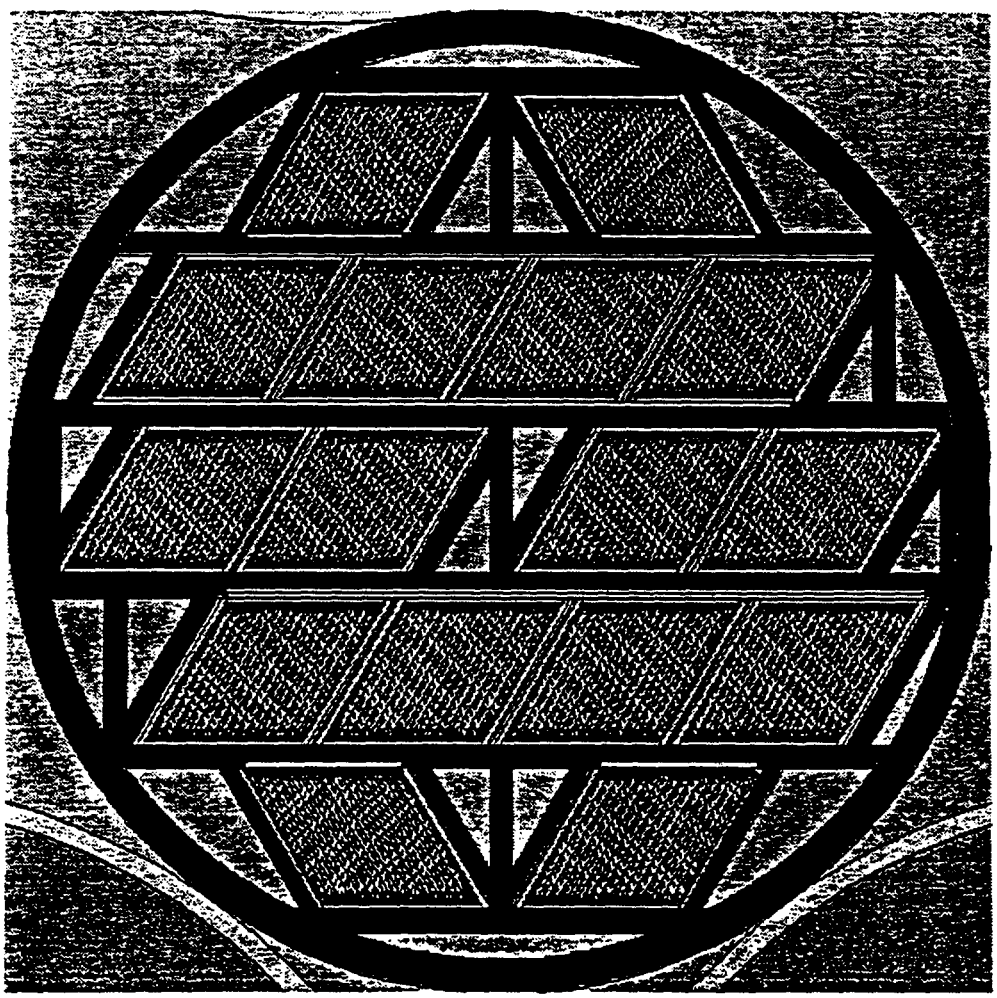


Figure 6.3.1-1 MIT Fuel Codisposal Canister Conceptual Design

6.3.2 ORR Codisposal Basket Conceptual Design

The ORR SNF canister consists of 4 layers (similar to the arrangement for the MIT SNF). Each of these layers contains a 10 assembly array, contained within ten rectangular tubes (5.0 mm wall thickness) aligned so that straight structural load paths progress from one side of the basket to the other. The tubes do not contain boron neutron absorber materials due to the moderate enrichment (20.56 weight percent U-235, initial) of the ORR fuel assemblies. A clearance of at least 2.54 mm is provided for the assembly in the basket. The cross-section layout of the ORR basket is illustrated in Figure 6.3.2-1. In this figure, the shaded segments represent structural steel. Note that the center tube of the nine-tube rectangular is offset relative to the center of the codisposal canister by 18.0 mm. This offset results from the asymmetry of the basket. The use of asymmetric baskets is the current practice among large storage and transport package designs. The inner radius of the codisposal canister is 20.465 cm.

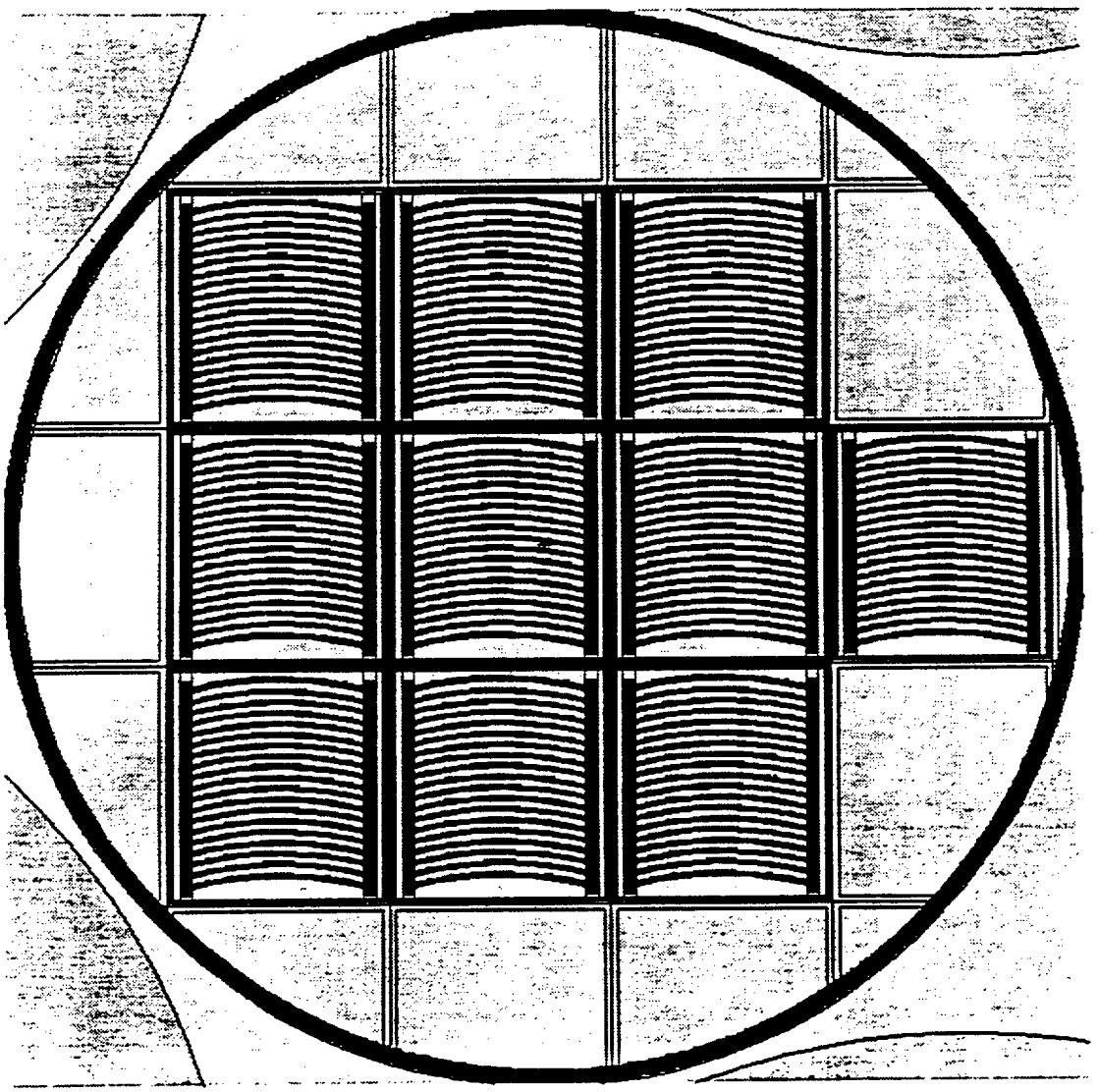


Figure 6.3.2-1. ORR Codisposal Basket Conceptual Design

6.3.3 Codisposal Waste Package Conceptual Design

A radial cross-sectional view of the waste package loaded with the 5 HLW and one DOE SNF codisposal canister is shown in Figure 6.3.3-1. For the final MCNP calculations the spacing of the canisters was modified to represent a probable configuration in the time frame that the waste packages would be penetrated and filled with water - shifted to the bottom and supported on the walls of the waste package and/or on other canisters. A radial cross-sectional view of this configuration is shown in Figure 6.3.3-2. An axial cross-sectional view of this shifted arrangement is shown in Figure 6.3.3-3. Note that the plane of this radial cross-section corresponds to a horizontal line through the center of the codisposal canister in Figure 6.3.3-2.

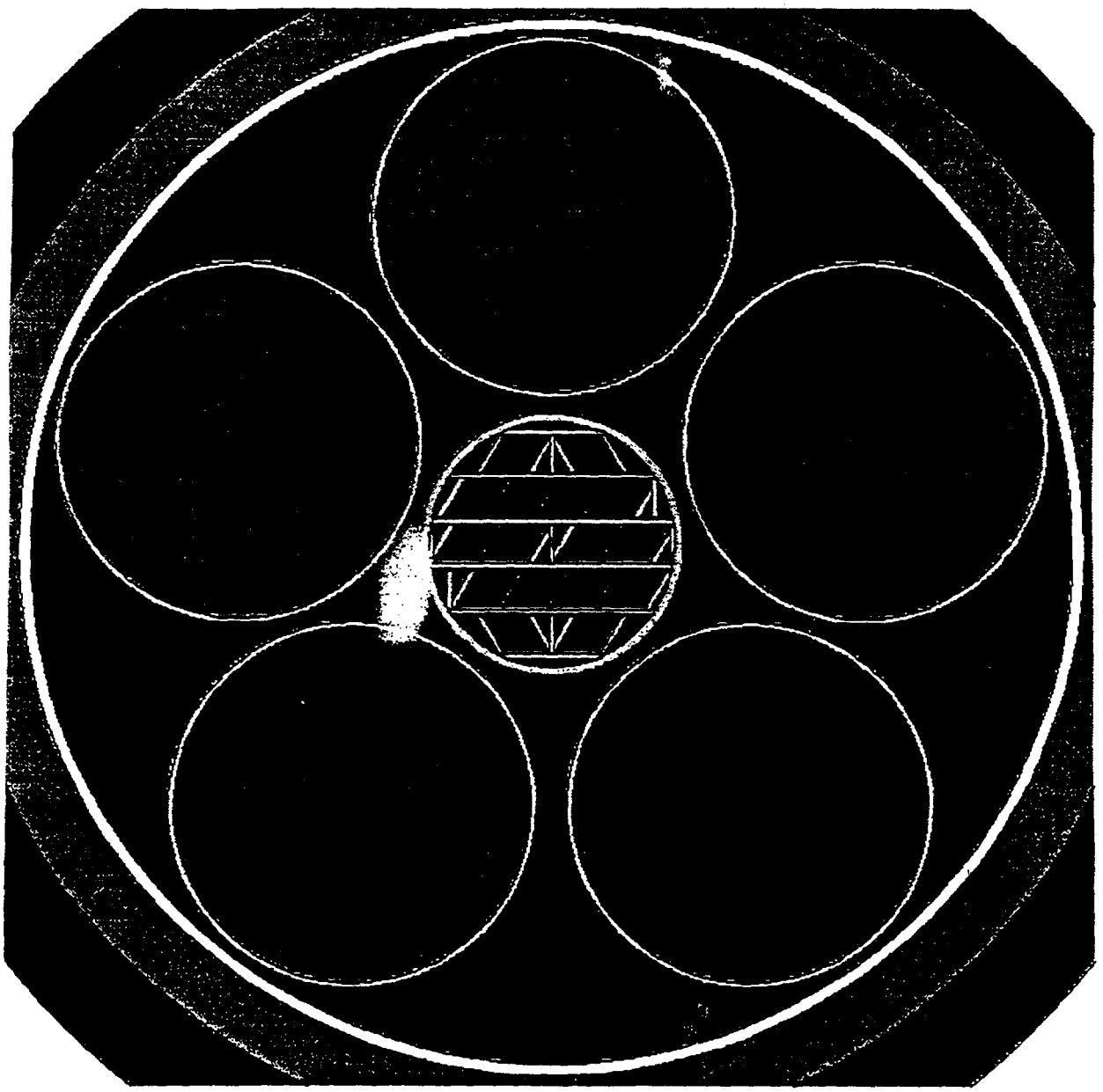


Figure 6.3.3-1. Radial Cross-Sectional View of the Codisposal Waste Package Loaded Configuration with MIT SNF Canister

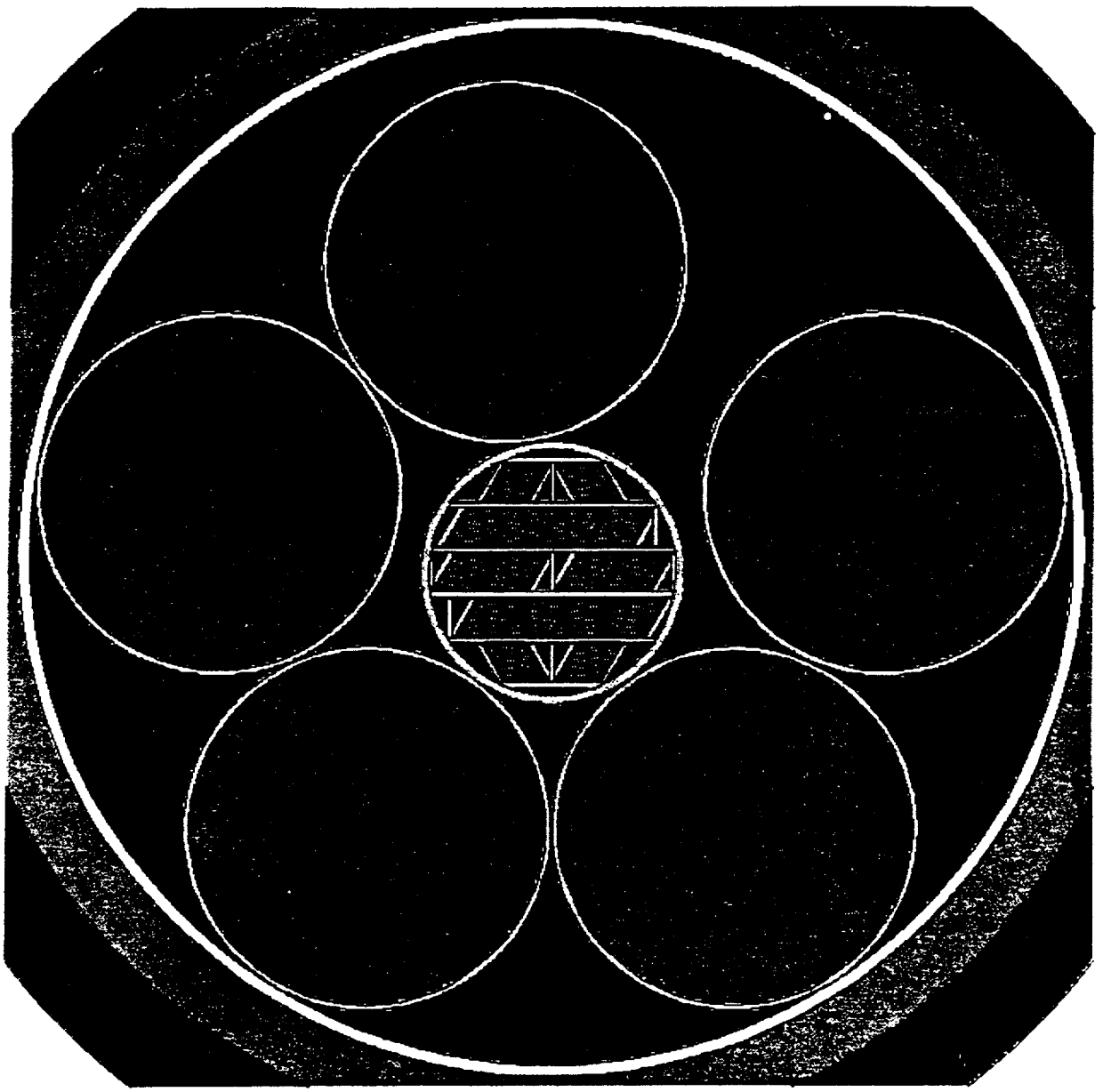


Figure 6.3.3-2. Radial Cross-Sectional View of the Codisposal Waste Package Probable Degraded Configuration with MIT SNF Canister

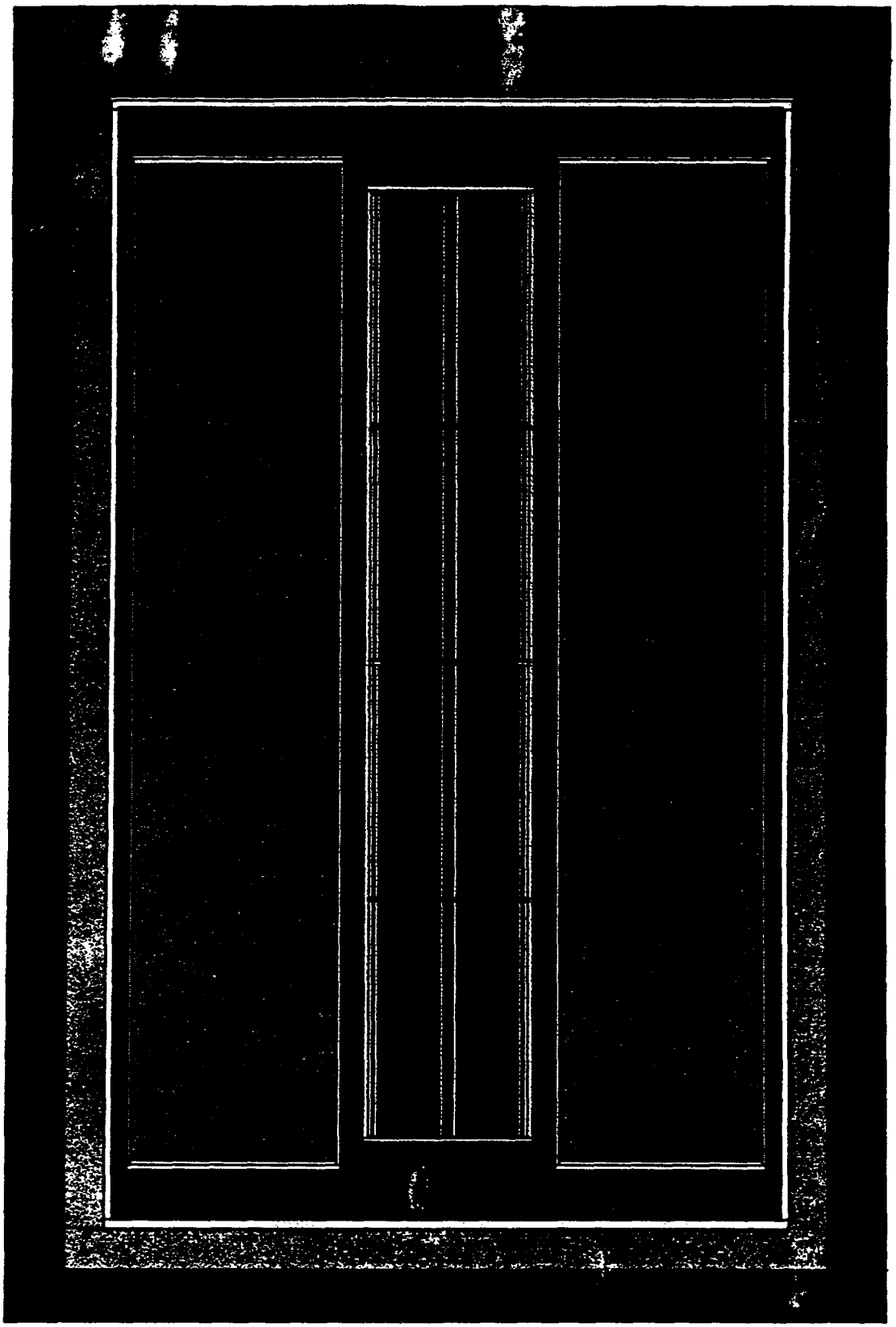


Figure 6.3.3-3. Axial Cross-Sectional View of the Waste Package Probable Degraded Configuration with MIT SNF Canister

6.4 Methods and Input Parameters

6.4.1 Criticality

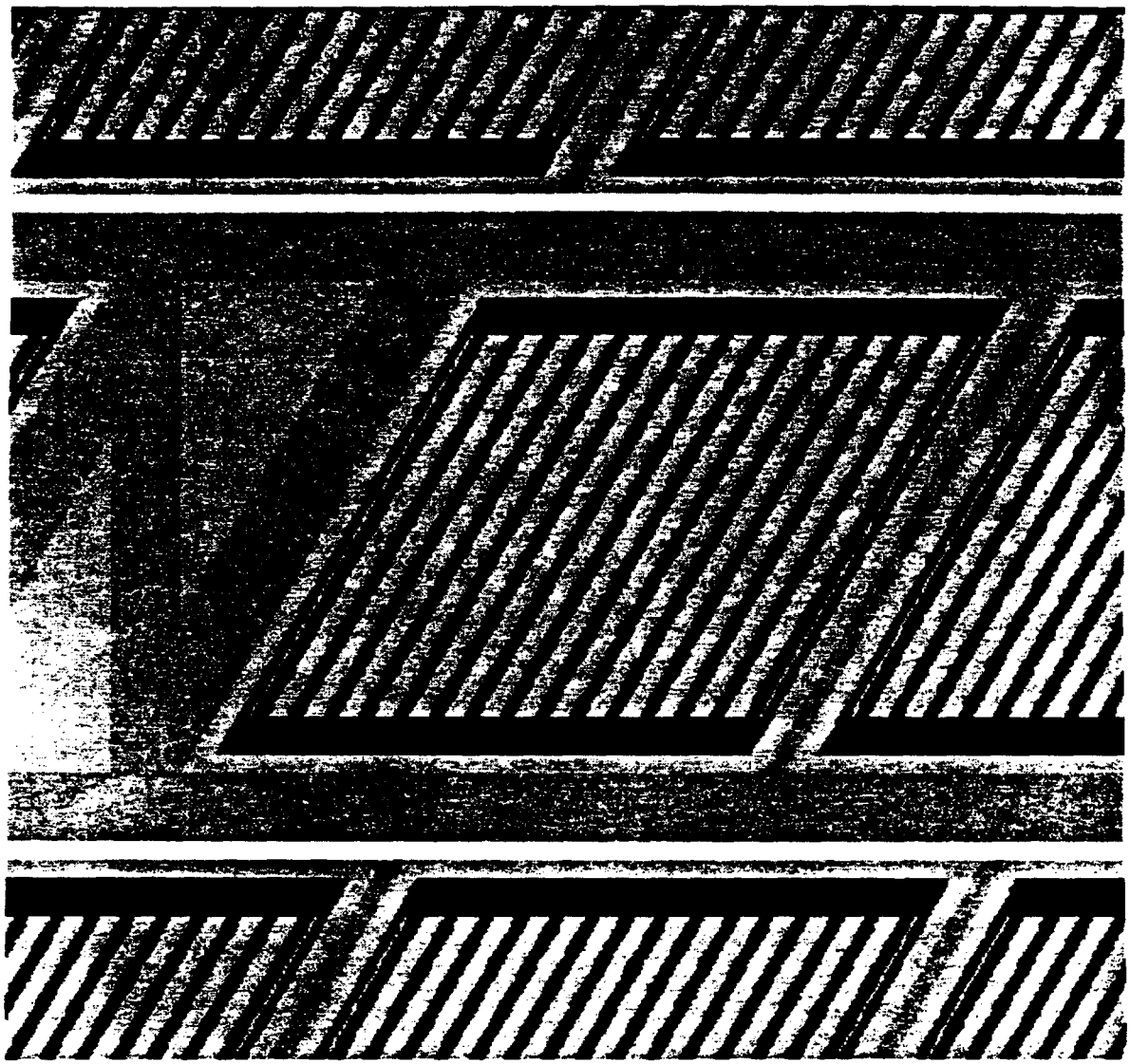
Criticality analyses of the MIT and ORR fuel types requires construction of MCNP4A geometry and material models. The development of the geometry models is summarized below. This analysis is documented with computer program output in reference 8.15. The materials models are straightforward because the structural materials of the waste package are ASME code materials and are hence well-defined, as are the water moderator (Assumption 4.3.4) and stainless steel/boron alloy.

The MCNP4A model is created by selecting the "worst case" dimensions from the range of values for each dimension. The procedure is to maximize the fuel volume and moderator volume by applying the minimum thicknesses of the aluminum cladding components and the maximum widths and lengths of the fuel plates.

6.4.1.1 MIT Fuel Geometry

Explicit geometric models of the MIT fuel assembly were constructed. The fuel alloy and aluminum cladding were modeled as separate layers in close contact. The actual design spacing of the fuel plates within the assembly was used. The assemblies are shortened by removing the end fittings, and the resulting shorter length was modeled to permit the fuel zones to minimize their separation in the axial direction to maximize k_{eff}. The resulting MCNP4A model is shown in Figure 6.4.1.1-1.

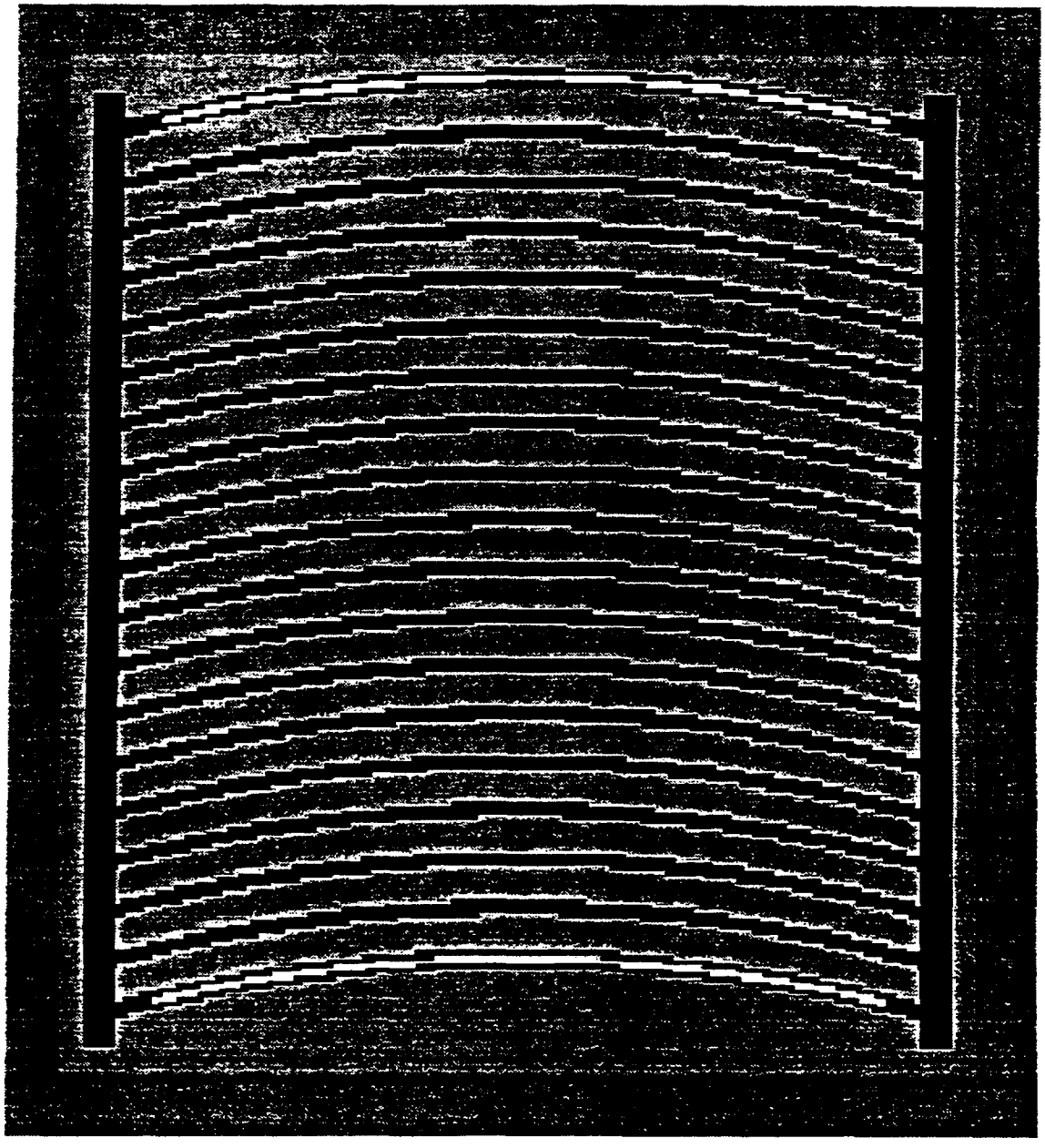
Figure 6.4.1.1-1. MCNP Model of MIT Fuel Assemblies in Basket.



6.4.1.2 ORR Fuel Element Geometry

The individual curved plates of the ORR fuel assembly were individually modeled, including the slightly different fuel alloy U-235 content of the plates at either end of the nineteen plate array. The aluminum cladding and the fuel alloy were modeled individually as separate layers in close contact. The aluminum side plates of the fuel assembly were also modeled explicitly. A picture of the resulting MCNP4A geometry is show below, in Figure 6.4.1.2-1.

Figure 6.4.1.2-1 MCNP Model of ORR Fuel Assembly



6.4.1.3 MIT SNF Codisposal Basket Geometry

The MIT SNF assemblies of nearly rhomboidal cross section are placed in a basket of 5 rows (which are actually parallelogram shaped slots to accommodate adjacent rhomboidal assemblies), the inner three rows holding 4 assemblies each and the outer two rows with 2 assemblies each. This structure is supported by stainless steel. The criticality control is provided by four sets of stainless steel/boron plates: 1) two plates between the three inner rows of assemblies, 2) two shorter plates on the inside of the two outer rows, 3) short plates between the assemblies in each row, and 4) three disk shaped separator plates between the four axial layers of assemblies. The cross section of this arrangement is shown in Figure 6.3.1-1.

6.4.1.4 ORR Codisposal Basket Geometry

The ORR conceptual basket design consists of ten rectangular tubes aligned so that straight structural load paths progress from one side of the basket to the other. The tube design does not require boron neutron absorber materials due to the moderate enrichment (20 weight percent U-235, initial) in the ORR fuel assemblies. The only criticality control material required is the set of 3 stainless steel/boron between-layer separator plates are used to isolate four axial layers of ORR assemblies, as was done in the MIT SNF codisposal basket design. This ensures that adequate neutron absorption is provided as the fuel degrades while still contained in the codisposal canister. The thicknesses of the axial separators are similar to the MIT SNF design. The resulting MCNP4A model is essentially the same as shown in Figure 6.3.2-1.

6.4.1.5 Codisposal Basket Neutron Absorber Materials

Initially, neutron absorbers used in the criticality analysis of both the MIT and ORR codisposal basket conceptual designs were based on stainless steel/boron alloy SS316B2A (0.6 wt% B). The normal practice is to derate down to 75% of the actual minimum boron content per current design practice for waste packages (Assumption 4.3.7). This practice is in accord with current NRC practice for transportation packages when 100 percent inspection of the neutron absorber panels has not been performed. As a result, an alloy with 0.80 wt% B (which is 0.6 wt%/0.75) would be required to be used in fabrication of the codisposal canisters. The required loading is provided by the next grade which contains 0.87 wt% B (SS316B3A). The final design calculations are performed with SS316B2A alloy composition which approximates 75% of the nominal B-10 loading of SS316B3A.

6.4.1.6 Waste Package

For the initial calculations, a simplified model of the waste package was constructed with the codisposal canister centered, and five HLW canisters (stainless steel canister walls omitted) arrayed about the codisposal canister. The waste package structural wall was modeled in the radial direction as a single thick layer of Alloy 825 (neutronically equivalent to actual Alloy 625 inner barrier material); however, the ends of the waste package were modeled as water reflectors since details outside the DOE-SNF canister separated by more than 15 cm of water will have very little effect on the canister reactivity. A cross-sectional view of the orientation of the canisters and waste package barrier for the MIT fuel is shown in Figure 6.4.1.6-1.

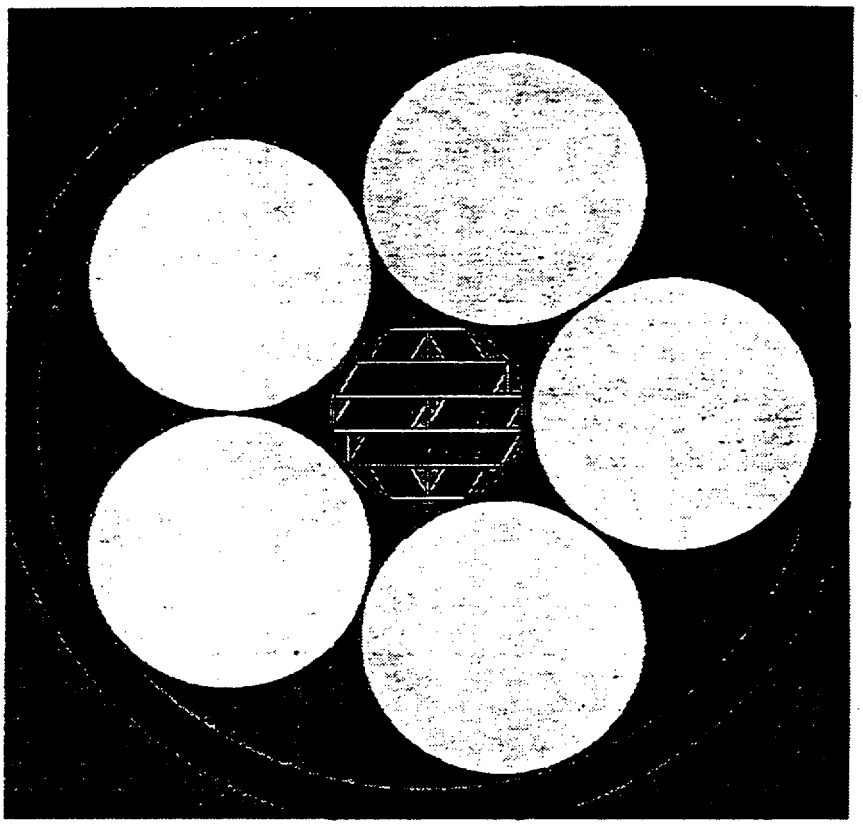


Figure 6.4.1.6-1. Simplified model of codisposal waste package for initial calculations

A detailed MCNP model of the waste package and canisters which included all components was constructed for the final calculations and reflects the geometries described in Section 6.3.3 and includes structural design modifications described in Section 6.5.3.1.

6.4.2 Shielding

6.4.2.1 Thermal and Shielding Source Term Generation

A model using the SAS2H Sequence of SCALE4.3 (Ref. 8.11) was developed based on the burnup and decay data provided by SRS for MIT fuel. There was no thermal analysis performed for the ORR fuel. This analysis is documented with computer program output in reference 8.15. For the SAS2H calculation the maximum burnup was to 8100 MWD/MTU and the time in reactor was 2500 days, as indicated in Section 4.1. The power level is 9.68 MW/MTU. The exposure time is calculated as 8100 MWD/MTU by 9.68 MW/MTU which equals 836.8 days. The down time is then calculated by dividing 2500 days - 836.8 days = 1663.2 days. Actual operation would have been up and down on a day-to-day basis. For the SAS2H calculation the exposure time was divided into quarters with one-third the down time between each exposure step. This will provide a conservative estimate of the source term and decay heat. The exposure time and decay time used in each of the steps is thus 209.2 days and 554.4 days, respectively. The resulting gamma and neutron sources for the MIT spent fuel are provided in Tables 6.4.2-1 and 6.4.2-2, respectively. HLW glass sources were also obtained from SAS2H runs (Ref. 8.38) and are listed in Tables 6.4.2-1 and 6.4.2-2.

Table 6.4.2-1 Photon Sources for MIT Fuel and HLW Canisters

Upper Energy Boundary of Group	MIT Fuel Source (per MTU)		HLW Source (per Canister)	
MeV	photons/sec	Fraction of Source	photons/sec	Fraction of Source
5.00e-2	5.69e+14	3.45e-01	1.3215e+15	3.60e-01
1.00e-1	1.69e+14	1.03e-01	3.9581e+14	1.08e-01
2.00e-1	1.22e+14	7.39e-02	3.0959e+14	8.42e-02
3.00e-1	3.58e+13	2.17e-02	8.7394e+13	2.38e-02
4.00e-1	2.62e+13	1.58e-02	6.3931e+13	1.74e-02
6.00e-1	2.61e+13	1.58e-02	8.8265e+13	2.40e-02
8.00e-1	6.94e+14	4.20e-01	1.3478e+15	3.67e-01
1.00	4.21e+12	2.55e-03	2.1344e+13	5.81e-03
1.33	2.71e+12	1.64e-03	2.9649e+13	8.07e-03
1.66	8.64e+11	5.23e-04	6.4161e+12	1.75e-03
2.00	1.49e+11	9.01e-05	5.1377e+11	1.40e-04
2.50	7.55e+11	4.57e-04	2.9370e+12	7.99e-04
3.00	4.46e+09	2.70e-06	2.0440e+10	5.56e-06
4.00	4.84e+08	2.93e-07	2.2835e+09	6.21e-07
5.00	1.69e+02	1.03e-13	5.2534e+05	1.43e-10
6.50	5.57e+01	3.37e-14	2.1058e+05	5.73e-11
8.00	8.76e+00	5.31e-15	4.1263e+04	1.12e-11
10.00	1.55e+00	9.37e-16	8.7544e+03	2.38e-12
TOTAL	1.65e+15	1.00e+00	3.6750e+15	1.00e+00

Table 6.4.2-2 Neutron Sources for MIT Fuel and HLW Canisters

Upper Energy Boundary of Group	MIT Fuel Source (per MTU)		HLW Source (per Canister)	
MeV	neutrons/sec	Fraction of Source	neutrons/sec	Fraction of Source
4.00e-1	1.64e+02	4.89e-03	2.087e+06	2.54e-02
9.00e-1	9.96e+02	2.91e-02	6.34e+06	7.72e-02
1.40	2.93e+03	8.56e-02	6.92e+06	8.43e-02
1.85	5.02e+03	1.47e-01	6.12e+06	7.45e-02
3.00	1.84e+04	5.39e-01	2.61e+07	3.18e-01
6.43	6.64e+03	1.94e-01	3.42e+07	4.17e-01
20.00	1.90e+01	5.55e-04	3.07e+05	3.74e-03
TOTAL	3.42e+04	1.00e+00	8.21e+07	1.00e+00

The heat load for an MIT assembly was also calculated by the SAS2H results in a separate ORIGEN-S case for a variety of decay times. The heat load for Savannah River glass canister was taken from reference 8.23. The heat generation per MIT assembly at various cool times is provided in Table 6.4.2-3 along with that for HLW glass.

Table 6.4.2-3 Heat Load of MIT SNF Assembly and Savannah River HLW Canister

Cooling Time (yrs)	Emplacement Time (yrs)	MIT SNF Heat (Watts)	Savannah River HLW (Watts)
5	0	0.164	526.7
7	2	0.145	501.7
9	4	0.135	479.4
20	15	0.102	376.0
40	35	0.0637	244.2
60	55	0.0397	161.2
80	75	0.0250	108.5
100	95	0.0159	74.86

6.4.2.2 Shielding Model

The gamma and neutron sources were inserted into the MCNP4A models which employed the geometry and isotopic material descriptions used for the criticality safety calculations for the MIT codisposal canister within a codisposal waste package. The basket and 64 MITfuel assemblies were homoginized to fill the codisposal. A radial cross-sectional view of this model is shown in Figure 6.4.2.2-1. For the gamma dose cases, only the gamma rays with energies greater than 0.4 Mev were specified because lower energies will not contribute significantly to the dose rate outside the waste package (Ref. 5.6) and the remaining group contributions (previously listed in Section 6.4.2.1) were renormalized. Because of the extremely low neutron source strength for the MIT SNF, no was set up for this source. Tallies were set up in the model to determine the dose rates at various points including segments on the waste package outer surface for all energies.

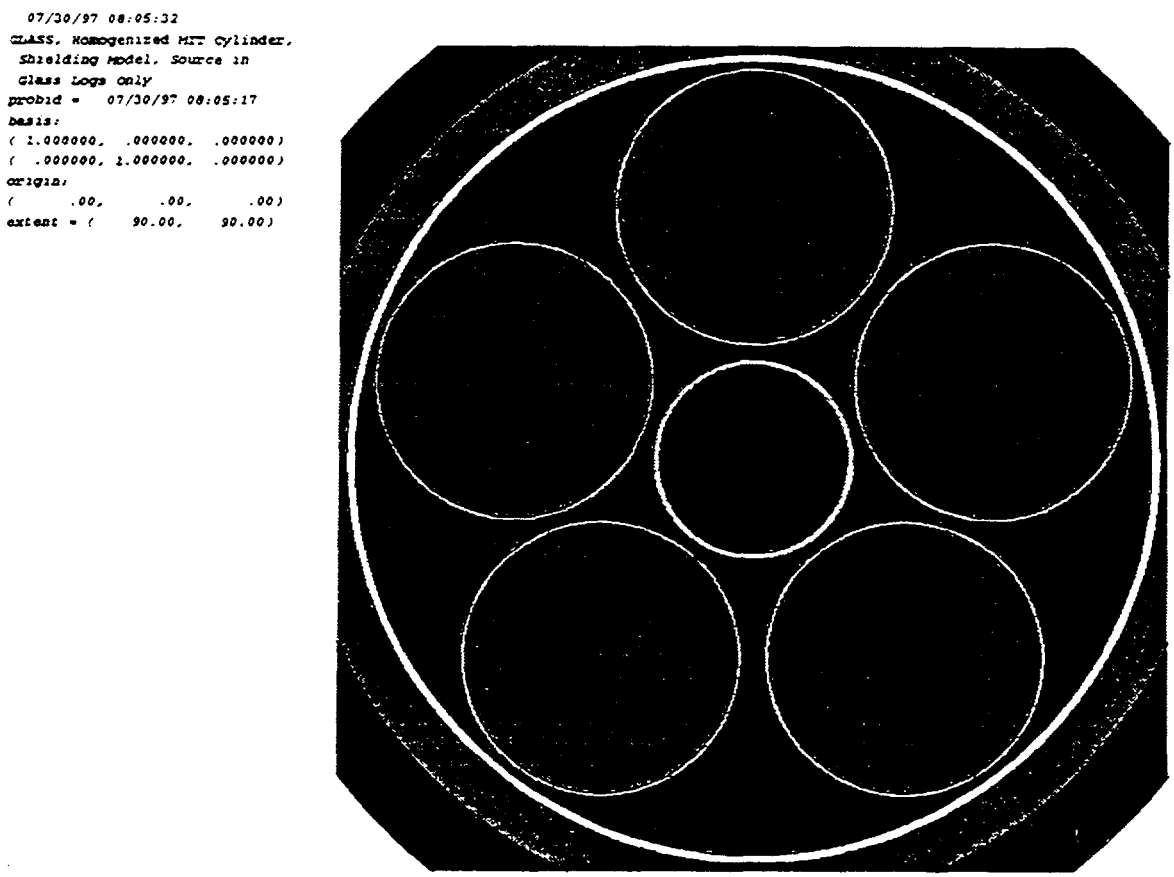


Figure 7.6.2-1 Radial Cross-Sectional View of the Waste Package Shielding Model

6.4.3 Structural

6.4.3.1 Basic Design Approach

The MIT-SNF codisposal canister is analyzed for dynamic impact loads due to a waste package tip-over event. The codisposal SNF canister is centered in the waste package surrounded by 5 HLW canisters. One of the possible recovery operations for this design basis event is to be able to retrieve the fuel so that it can be placed into another canister in case of such an accident. Hence, the basic design criteria for this tip-over event is to keep the fuel assemblies undeformed inside the codisposal canisters. This is accomplished by processing the finite-element results in terms of the canister displacements and comparing those with the available clearance (Assumption 4.3.12) between the fuel assemblies and the basket structural members. If the results show that the clearance between the fuel assemblies and the basket structural members is not completely closed, it can be concluded that the fuel assemblies will not be loaded by the basket structure. However, in case of complete closure of the clearance, the design will be deemed unacceptable because the fuel assemblies may be deformed.

A second part of the structural analysis will also evaluate the stress distribution within the codisposal canister wall. The equivalent stresses (von Mises stresses) will be compared with the material yield strength in order to determine the locations of permanent deformation in the structure.

6.4.3.2 Finite-Element Model Description

A two-dimensional (2-D) finite-element model of the codisposal canister shell and basket structure has been developed in order to perform a waste package tip-over analysis. The shell is connected to the basket members at the adjacent surfaces.

The MIT-SNF canister is analyzed for a tip-over event which could occur after it is placed into the codisposal waste package. The MIT-SNF canister basket structure is more resistant to impact loads in the horizontal and vertical orientations of the basket than it is in a 45° orientation. In order to represent the most critical load conditions on the MIT-SNF canister, the 45° orientation of the MIT-SNF canister is selected. The *g* load acted upon the canister by one of the HLW canisters is conservatively assumed to be transmitted through the basket in this angular orientation (Assumption 4.3.10). The MIT-SNF canister is supported in two places at the bottom which are the points of contact with two HLW canisters. The other two HLW canisters have no effect on the MIT-SNF canister since they will be supported by the two HLW canisters laying below the MIT-SNF canister. The external force on the MIT-SNF canister was calculated using the results of a previous design analysis performed for the tip-over evaluations of the 4-canister DHLW waste package. The resulting g load is 104 *g* (Assumption 4.3.11). The external loads on the MIT-SNF codisposal canister due to impact of the glass canister above is simulated by applying nodal forces from the 104g impact at the point of contact. A second load applied to the model is the reaction force from the 104g impact on the bottom surfaces of the MIT-SNF canister that are in contact with the two HLW canisters. The finite-element model includes two reaction constraints which were placed 36° apart on each side of the plane of impact. This angle is based on one HLW canister loading the codisposal canister, which in turn loads two HLW canisters below the

codisposal canister. This design analysis did not take structural credit for the fuel assemblies; however, the fuel assembly weights were added to the weight of the basket structure for conservatism. The calculations are given in reference 8.19.

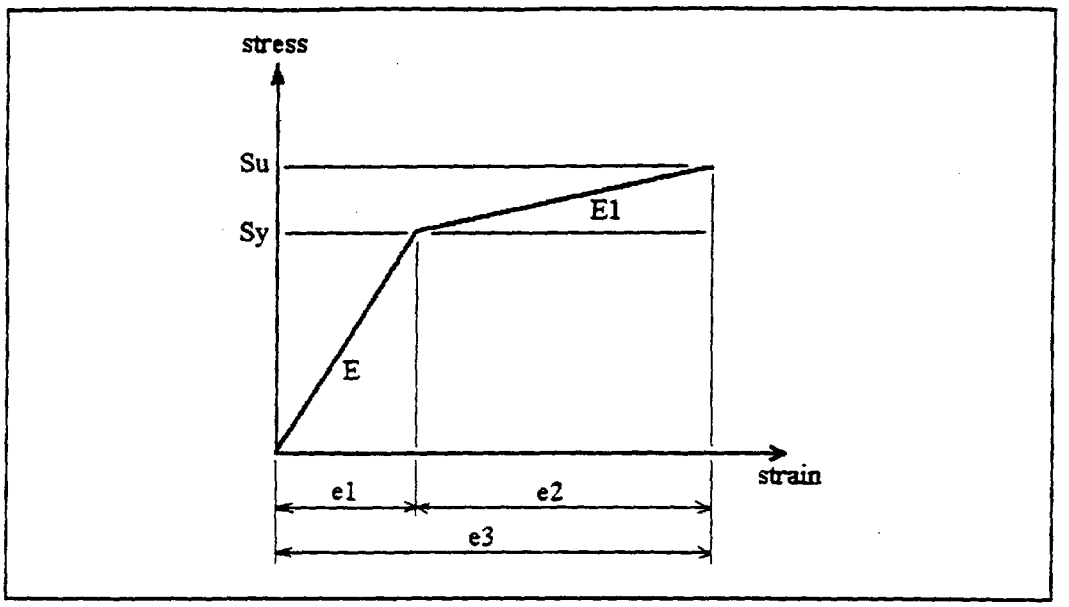
6.4.3.3 Size of Indentation between the HLW Canister and the Codisposal Canister

The interfaces between the HLW canister and the codisposal canister are line-contact with the as-built dimensions since both objects are cylinders. During an impact, three HLW canisters and the codisposal canister interact as the uppermost HLW canister presses down upon the codisposal canister, and the codisposal canister presses down upon the lower two HLW canisters. The other two HLW canisters are alongside the codisposal canister and do not structurally load it in the impact. To aid the ANSYS code in obtaining a converged numerical solution, a hand calculation was performed (Ref. 8.19) to obtain the area of the "flat spot" or indentation between each HLW canister and the codisposal canister. The width of indentation on the codisposal canister where the external forces are distributed is 2.4 mm; this width covers approximately three nodes in the structural model.

6.4.3.4 Material Property Calculations

The results of this impact simulation include elastic and plastic deformations in the codisposal container. When the materials enter the plastic range, the slope of the stress-strain curve continuously changes. Thus, a linear simplification for this curve is used to incorporate plasticity into the model. A standard approach commonly used in engineering is to connect the yield point to the ultimate tensile strength point of the material with a straight line. The stress/strain curve below illustrates the procedure and the parameters used in the calculations.

Figure 6.4.3.4 Bilinear Stress-Strain Curve



Bilinear Stress-Strain Curve

where, S_y = Yield strength of the material
S_u = Ultimate tensile strength
e_1, e_2, e_3 = strain magnitudes
E = Elastic modulus (slope of the line in the elastic region)
E_1 = Tangent modules (slope of the line in the plastic region)

The slope, E_1 is determined by :
$e_1 = S_y / E$ and $e_2 = e_3 - e_1$ where e_3 = elongation specified for material
Hence, $E_1 = (S_u - S_y) / e_2 = (0.482 - 0.172) / (0.4 - (0.172 / 195))$ (see Section 4.1)
E_1 = 0.776 GPa (calculated for 316L stainless steel)

6.4.4 Thermal

6.4.4.1 Thermal Background

As part of an engineered barrier system for the containment of radio nuclides, the codisposal WP must be shown to comply with all regulations and requirements that govern the conditions of the emplaced SNF and the near-field rock at the repository horizon. Temperatures in the WP and near-field host rock are key to radio nuclide containment, as they directly affect the oxidation rates of the metal barriers, the structural integrity of the metal HLW canisters and the glass matrix, and the ability of the rock to impede migration of radio nuclides.

Maximum allowable temperatures are based on material performance criteria and are specified as design goals for the WP/EBS design. For the glass waste form, the MGDS program has recommended a 400°C temperature limit for the glass matrix waste form as documented in reference 8.6 (CDA DCWP 002, p. 8-4). This thermal goal is to assure that the glass transition temperature is not exceeded. To limit the predicted thermal and thermo-mechanical response of the host rock and surrounding strata, maximum temperatures of 200°C for the emplacement drift wall have been specified (Ref. 8.6).

The configuration for the thermal evaluation consists of emplacing the DHLW waste packages, containing the codisposal canister, between commercial SNF waste packages, which determine the thermal environment. The method for thermal evaluation of the waste package containing the codisposal canister involves a two-model approach to determine the time-dependent WP thermal behavior. As presented in reference 8.17, a three-dimensional (3-D) transient finite element model of the WP emplacement provides the 4-canister DHLW WP surface temperature history that was used as a boundary condition in the two dimensional codisposal WP model utilized in this report. The model yields conservative peak glass matrix temperatures, MIT fuel peak temperatures, peak codisposal canister surface temperatures and peak HLW canister surface temperatures.

6.4.4.2 Thermal Model

A two-dimensional model of the codisposal waste package was developed using a uniform axial heat load. The effective thermal conductivity of the MIT SNF (Assumption 4.3.17) was used to represent the volume occupied by the spent fuel, and the thermal conductivities of glass and stainless steel 316L were used for the other major waste package contents. The helium fill gas (Assumption 4.3.6) acted as a conductor of heat only, and convection was not modeled (as explained in Section 3.2). The HLW canisters and codisposal canister are modeled as "floating" in the codisposal waste package such that HLW canisters, codisposal canister and WP inner shell do not touch each other; thus there are no conduction paths via the canister walls.

The heat loads for the HLW glass and the heat loads for the MIT fuel assembly areas were applied in the codisposal canister and HLW canisters. The heat loads were decreased as a function of time to account for radioactive decay. The heat loads were applied volumetrically throughout the fuel assembly region. The boundary condition for the 2-D model was the WP surface temperature which was determined in reference 8.17.

Since the repository rock temperatures change slowly with time (driven primarily by the commercial SNF), and the HLW glass and MIT fuel heat generation rates decrease with time, the steady state problem was solved at several different times from emplacement out to 100 years. Thus, both bounding environments and bounding heat loads were considered.

6.5 Results

6.5.1 Criticality

The criticality calculations indicated below as "initial" are based on the simplified waste package model described in Section 6.4.1.6 and the preliminary codisposal canister design. The structural analysis results presented in Section 6.5.3.1 indicate that the material and thickness of the DOE-SNF canister should be changed to XM-19 and 1.5 cm, respectively in order to fit within the space available. The calculations indicated below as "final" include modifications to the DOE-SNF canister as required in the structural analysis and include all details of the HLW canisters and waste package in the models. The effect of varying the horizontal orientation of HLW canisters with the DOE-SNF canister from the loaded configuration to the probable orientation at the time the waste package and canisters would be penetrated and filled with water is also investigated. The k_{eff} values listed in the tables below are equal to the calculated value from MCNP4A plus two sigma plus the 0.02 bias allowance defined in Section 5 ($k_{eff}=k_{calc}+2*sigma+0.02$).

6.5.1.1 MIT SNF Criticality

Initial Intact

Results obtained from the analysis in reference 8.15 for the MIT fuel in the intact configuration within the flooded codisposal canister are provided in Table 6.5.1.1-1. The intact configuration was evaluated for the effects of water moderator intrusion into the fuel matrix by varying the density of H_2O) in the maximum potential void volume within the fuel from zero to 100 percent (one gram per cubic centimeter) alloy. These calculations showed that the maximum reactivity is reached when the fuel alloy is waterlogged to the maximum extent. Note that the in-row separator plates between assemblies shown in Figure 6.3.1-1 are unborated for these cases. Stainless steel boron (SS316B2A) is used in the between-slot plates and in the between-layer (axial) separator plates.

Table 6.5.1.1-1 Intact MIT SNF Codisposal Canister Criticality Results

Case Name	Percent H2O* in Fuel Alloy	k-calculated	sigma	k_{eff}
MITA	0	0.81181	0.00116	0.83413
MITD	25	0.83265	0.00138	0.85541
MITC	50	0.84897	0.00147	0.87191
MITE	75	0.86581	0.00150	0.88881
MITF	95	0.87857	0.00151	0.90159
MITB	100	0.88019	0.00138	0.90295

* Percentage of a maximum of 63.53 volume percent water in fuel matrix voids.

Initial Degraded State Within Canister

The criticality calculations for the degraded states of the MIT SNF are documented in reference 8.15 and summarized in Table 6.5.1.1-2. The degraded states of the MIT fuel, within the codisposal canister that are evaluated herein, are described in Section 6.1. MCNP calculations evaluated the reactivity of the MIT fuel as it degrades by modeling the fuel material and moderator within the codisposal basket components in successive stages. Stainless steel boron (SS316B2A) is used in the between-slot plates and in the between-layer (axial) separator plates for all cases in Table 6.5.1.1-2. The first set of calculations, cases MITH through MITK1, show that the reactivity of the fuel is excessive if stainless steel alone is used to separate adjacent assemblies within a basket slot (in-row separator plates). The second set of calculations, cases MITL through MITO, evaluate the fuel and codisposal basket with in-row separator plates fabricated from stainless steel/boron alloy SS316B2A. In all of these cases, k_{eff} remains below the 0.95 limit.

Table 6.5.1.1-2 Degraded MIT SNF Codisposal Canister Criticality Calculations

Case Name	Divider Plates Between Assembly	Degraded Fuel Configuration	k-calculated	sigma	k_{eff}
MITH	Stainless	Plate Array with Comb Teeth in Assembly. Envelope	0.92513	0.00170	0.94853
MITI	Stainless	Plate Array Homogenized	0.95879	0.00119	0.98117
MITJ	Stainless	Entire Assembly Homogenized (including Side Plates)	0.95779	0.00133	0.98045
MITK	Stainless	Entire Cell Homogenized	0.99362	0.00128	1.01618
MITK1	Stainless	High Boron (1.6 wt% B) in Between-Slot and Between-Layer Plates	0.95003	0.00153	0.97309
MITL	SS316B2A	Plate Array with Comb Teeth in Assembly Envelope	0.85351	0.00158	0.87667
MITM	SS316B2A	Plate Array Homogenized	0.88749	0.00130	0.91009
MITN	SS316B2A	Entire Assembly (including Side Plates)	0.88015	0.00154	0.90323
MITO	SS316B2A	Entire Cell Homogenized	0.91901	0.00149	0.94199

← OVER

Final MIT Criticality Calculations

The simplified waste package model used in the initial calculations was modified to include 15 mm XM-19 for the DOE-SNF canister material and to include all components and dimensions for the waste package and HLW canisters (including stainless steel canister walls). The stainless steel/boron in all separator plates was changed to SS316B3A with a 75% of the nominal B-10 loading. A radial cross-sectional view of the model for the loaded configuration is shown in Figure 6.3.3-1. The MIT configuration with 0.213 cm thick stainless steel/boron in-row separator plates and homogenized fuel cells, previously identified as most reactive (MITO), was rerun with this model and configuration (MITOZ1). The result is shown in Table 6.5.1.1-3. Its k_{eff} is below the limit of 0.95. The spacing of the canisters was modified to represent a probable horizontal configuration in the time frame that the waste packages would be penetrated and filled with water and with the canisters shifted to the bottom and supported on the walls of the waste package and/or other canisters. A radial cross-sectional view of this configuration is shown in Figure 6.3.3-2, and an axial cross-sectional view is shown in Figure 6.3.3-3. The result for this probable configuration (MITOZ3) is shown in Table 6.5.1.1-3. The previous cases with homogenized fuel did not extend the homogenization in the axial direction beyond the length of the fuel assembly (radial only). An additional case in which the MIT assemblies were homogenized into the entire volume of the basket cells (MITOZ3A) was run corresponding to case MITOZ3 with the result shown in Table 6.5.1.1-3. The result for this configuration still falls below the 0.95 limit on k_{eff}. The MIT canister configuration with intact waterlogged fuel (MITB) (note that MITB did not include the in-row borated separator plates between assemblies) was rerun in this model (MITBZ3) with the result shown in Table 6.5.1.1-3. Note that the homogenization of the fuel into the cell which represents a possible degradation configuration (MOTOZ cases) is much more reactive than intact fuel.

Table 6.5.1.1-3 MIT Mk2 Fuel in Codisposal Canister - Final Calculations

Case Name	Divider Plates Between Asbls	Fuel Configuration	k-calculated	sigma	k_{eff}
MITOZ1	SS316B3A (75%)	Entire Cell Homogenized, WP Loaded Configuration	0.91123	0.00156	0.93435
MITOZ3	SS316B3A (75%)	Entire Cell Homogenized, Probable WP Degraded Configuration	0.91602	0.00148	0.93898
MITOZ3A	SS316B3A (75%)	Entire Cell Homogenized to Fill Axial Space Between Separator Plates, Probable WP Degraded Configuration	0.92635	0.00149	0.94933
MITBZ3	SS316B3A (75%)	Intact Waterlogged Fuel, Probable WP Degraded Configuration	0.81013	0.00147	0.83307

6.5.1.2 ORR SNF Criticality

Initial Intact

The criticality calculations tabulated below from reference 8.15 show that, due to the lower initial enrichment (only 20.56%) the ORR fuel remains subcritical regardless of the amount of water that intrudes into the fuel matrix. This is in spite of the lack of boron neutron absorber material within the basket structure in the radial direction. (Axial separators of stainless steel/boron were provided similar to those incorporated into the MIT SNF codisposal basket.)

Table 6.5.1.2-1 Intact ORR Codisposal Canister Criticality Calculations

Case Name	Percent H2O* in Fuel Alloy	k-calculated	sigma	k_{eff}
ORR10E	0	0.84474	0.00147	0.86768
ORR10G	25	0.85567	0.00150	0.87867
ORR10H	50	0.85998	0.00154	0.88306
ORR10I	75	0.87018	0.00158	0.89334
ORR10J	95	0.87422	0.00146	0.89714
ORR10F	100	0.87446	0.00139	0.89724

* Percentage of maximum of 40.64 volume percent water in fuel matrix voids

Initial Degraded State Within Canister

The calculations for the degraded ORR fuel, contained within the codisposal canister, for the various degradation stages described in Section 6.1, are presented below in Table 6.5.1.2-2. These calculations evaluate the reactivity of the ORR fuel as it degrades by modeling the fuel material and moderator with the codisposal basket components in successive stages. The first set of calculations, cases ORRHASBL and ORRHSAB1, show that the reactivity of the fuel is excessive if the four layers of assemblies are stacked within each basket tube directly on top of one another. The second set of calculations, cases ORR1 and ORR2, evaluate the fuel and codisposal basket with between-layer separator plates fabricated from stainless steel/boron alloy SS316B2A. This analysis demonstrates the need for neutron-absorbing materials in the ORR between-layer separator plates.

Table 6.5.1.2-2 Degraded ORR Codisposal Canister Criticality Calculations

No Boron		k-calculated	sigma	k_{eff}
ORRHASBL	Homogenized Assembly	0.92887	0.00149	0.95185
ORRHSAB1	Homogenized Water Gap	0.94404	0.00148	0.96700
With Between-Layer Borated Steel Separator Plates				
ORR1	Homogenized Assembly	0.86127	0.00142	0.88411
ORR2	Homogenized Water Gap	0.88901	0.00140	0.91181

Final ORR Criticality Calculations

The detailed waste package model described in the in Section 6.5.1.1 for the probable degraded configuration was used with the ORR SNF canister composed of 1.5 cm thick XM-19. The stainless steel/boron in the between-row separator plates was changed to SS316B3A of its nonimal 75% B-10 loading. A radial cross-sectional view of the model for the probable degraded configuration is shown in Figure 6.5.1.2-1, and an axial cross-sectional view is shown in Figure 6.5.1.2-2. The ORR SNF canister configuration with homogenized fuel cells, previously identified as most reactive in the initial calculations (ORR2), was rerun with this model and configuration (ORROZ3F). The stainless steel structural members outside the basket are included in this model. The result is shown in Table 6.5.1.2-3, and the k_{eff} is below the limit of 0.95. The previous cases with homogenized fuel did not extend the homogenization in the axial direction beyond the length of the fuel assembly (radial only). An additional case in which the ORR assemblies were homogenized into the entire volume of the basket cells (ORROZ3A) was run with the result shown in Table 6.5.1.2-3. The result for this configuration still falls below the 0.95 limit on k_{eff}. The ORR SNF canister configuration with intact waterlogged fuel (ORR10F) was rerun in this new model (ORROZ3F) with the result shown in Table 6.5.1.2-3. Note that the homogenization of the fuel into the cell which represents a possible degradation configuration (ORROZ cases) is much more reactive than intact fuel.

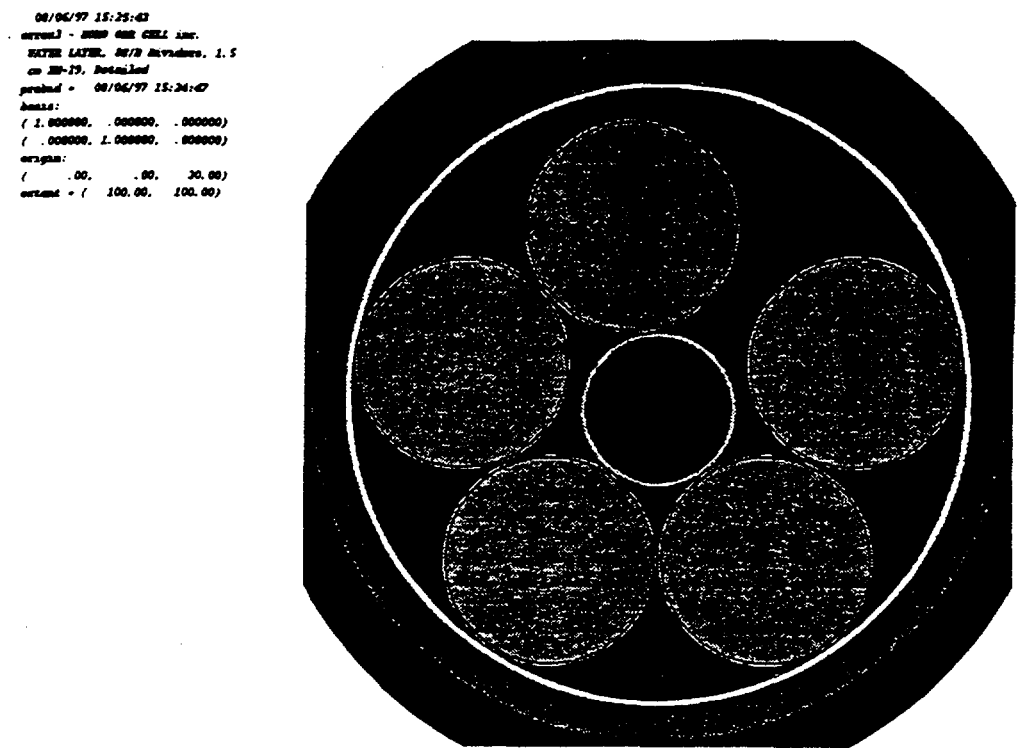


Figure 6.5.1.2-1 Radial Cross-Sectional View of the Waste Package Probable Degraded Configuration - ORR SNF Canister

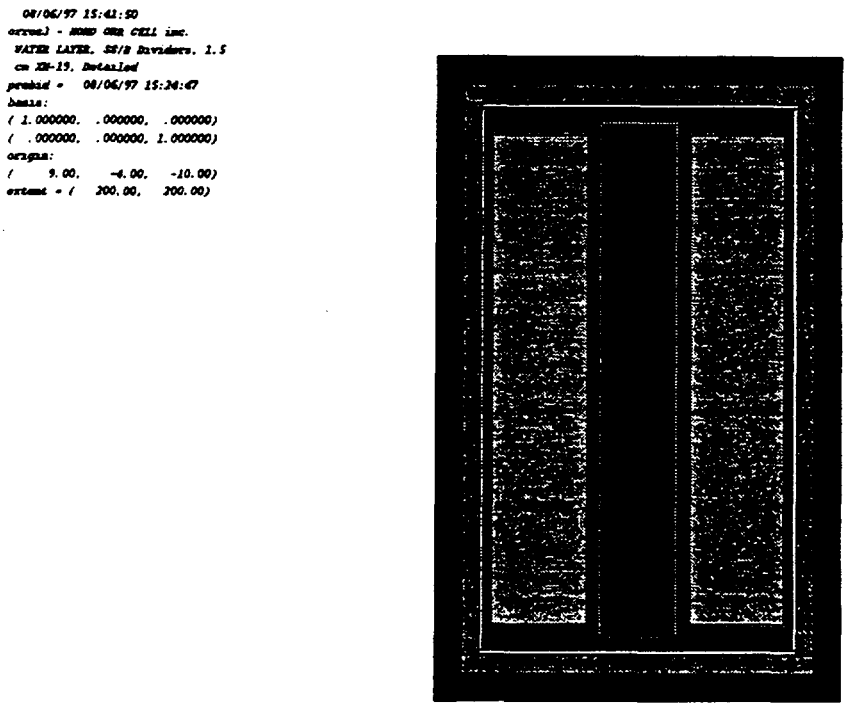


Figure 6.5.1.2-2 Axial Cross-Sectional View of the Waste Package Probable Degraded Configuration - ORR SNF Canister

Table 6.5.1.2-3 Final ORR Codisposal Canister Criticality Calculations

Case Name	Divider Plates Between Asbls	Fuel Configuration	k-calculated	Sigma	k_{eff}
ORROZ3F	SS316B3A (75%)	Entire Cell Homogenized, Probable WP Degraded Configuration	0.88043	0.00133	0.90309
ORROZ3A	SS316B3A (75%)	Entire Cell Homogenized to Fill Axial Space Between Separator Plates, Probable WP Degraded Configuration	0.91441	0.00136	0.93713
ORR10FZ	SS316B3A (75%)	Intact Waterlogged Fuel, Probable WP Degraded Configuration	0.86583	0.00126	0.88835

6.5.2 Shielding Results

A comparison of the neutron and gamma sources for the MIT SNF and HLW canisters presented in Section 6.4.2, indicates that the neutron source is insignificant to the total surface dose of the codisposal waste package considering that the total neutron source is at least 7 orders of magnitude lower than the photon source. The photon sources were normalized to the total in the waste package as indicated in Table 6.5.2-1. The MIT SNF photon source was normalized to the mass of 64 assemblies which are present in the DOE-SNF canister; the HLW canister photon source was normalized to 5 canisters which reflects the total source in the waste package. Note that the MIT fuel source is over 2 orders of magnitude lower than that for the HLW canisters; for the energy groups above 4 MeV, the MIT fuel source is over 5 orders of magnitude lower. Given this much lower source and the fact that the DOE-SNF canister will reside in the center of the waste package with the waste package walls shielded by the bulk of the HLW canisters, the effect of the DOE-SNF canister on the total surface dose is considered insignificant. The overwhelming contribution to the waste package surface dose will be the HLW canisters.

Table 6.5.2-1 Normalized Photon Sources for MIT Fuel and HLW Canisters

Upper Energy Boundary of Group	MIT Fuel Source	HLW Source
MeV	photons/sec/Codisposal Canister	photons/sec/WP (5 HLW Canisters)
5.00e-2	2.00e+13	6.61e+15
1.00e-1	5.97e+12	1.98e+15
2.00e-1	4.30e+12	1.55e+15

Upper Energy Boundary of Group	MIT Fuel Source	HLW Source
3.00e-1	1.26e+12	4.37e+14
4.00e-1	9.21e+11	3.20e+14
6.00e-1	9.18e+11	4.41e+14
8.00e-1	2.44e+13	6.74e+15
1.00	1.48e+11	1.07e+14
1.33	9.54e+10	1.48e+14
1:66	3.04e+10	3.21e+13
2.00	5.23e+09	2.57e+12
2.50	2.66e+10	1.47e+13
3.00	1.57e+08	1.02e+11
4.00	1.70e+07	1.14e+10
5.00	5.96e+00	2.63e+06
6.50	1.96e+00	1.05e+06
8.00	3.08e-01	2.06e+05
10.00	5.44e-02	4.38e+04
TOTAL	5.81e+13	1.84e+16

The three MCNP shielding cases run for the DHLW gamma source, the MIT gamma source, and the DHLW neutron source provide the following results. Te dose is reported as a value ± its relative error (1σ).

The DHLW gamma source case (MITSLD1) has a radial centerline dose rate of 9.3967 ± 0.0831 rem/hr. The dose rate out the bottom of the waste package tallied over the outer barrier lid is 1.8450 ± 0.0854.

The MIT gamma source case (MITSLD2) has an average radial centerline dose rate of 5.4821E-3 ± 0.2311 rem/hr. The peak radial centerline dose rate was calculated to be 3.6733E-2 ± 0.2515 at a location on the waste package surface unshielded by a HLW canister. The dose rate out the top of the waste package tallied over the outer barrier lid is 5.0199E-3 ± 0.0829.

The DHLW neutron dose case (MITSLD3) has a radial centerline neutron dose rate of 7.3501E-2 ± 0.0034 rem/hr and a gamma (N,gamma) dose rate of 1.7627E-4 ± 0.0133 rem/hr. The dose

rate on the bottom of the waste package tallied over the outer barrier lid is 3.5364E-2 ± 0.0019 rem/hr for neutrons and 7.6486E-5 ± 0.0083 rem/hr for gammas.

Inspection of the gamma shielding results shows that the MIT fuel in the codisposal canister contributes very little to the dose rate on the surface of the codisposal waste package. The neutron dose contribution from neutrons for either waste form is also insignificant. The dose rates on the exterior of the Codisposal waste package with the MIT codisposal canister is within acceptable limits for disposal.

6.5.3 Structural Stress and Displacement Analysis Results

A comparison of the equivalent stresses with the material yield and compressive strengths shows that the codisposal canister will experience permanent deformation in some localized regions (Ref. 8.19).

The displacement results are evaluated by comparing the maximum closure between the basket structural members of the fuel cells with the total available clearance between the fuel assemblies and the basket members. The maximum displacements are obtained in the region of the codisposal canister basket assembly where the structural loads are carried without the benefit of a vertical support. The results show that the maximum fuel cell gap closure is less than the minimum clearance provided between the fuel assemblies and the basket members (Ref. 8.19). Therefore, there will be enough gap for the fuel assemblies to rest in the basket cells without any deformation inflicted by the basket members.

A thickness of 20 mm was determined to be sufficient for the 316L stainless steel codisposal canister shell to prevent fuel assemblies from being deformed (Ref. 8.19).

6.5.3.1 Calculations for an Alternate Design

An alternate design is evaluated to reduce the outside diameter (O.D.) of the codisposal canister to provide additional clearance within the waste package. Since the interior volume of the codisposal canister is fixed by the SNF capacity, the only way to reduce the O.D. of the codisposal canister is to reduce the thickness. The alternate design differs from the original design only in terms of the type of material used for the canister shell; XM-19 stainless steel is chosen because it is stronger than 316L stainless steel in order to decrease the thickness of the codisposal canister shell. The basket member material remained as 316L stainless steel. Since the resulting equivalent stresses were not significantly over the yield strength, a correlation was developed to predict the required minimum thickness of the codisposal canister shell if XM-19 stainless steel is used.

A codisposal canister shell thickness of 15 mm of XM-19 stainless steel provides equivalent strength as the previously analyzed 316L stainless steel shell (Ref. 8.19).

6.5.4 Thermal Results

The detailed results of the thermal analysis are tabulated in reference 8.20, and summarized in figure 6.5.4-1 which shows the peak temperature at different locations in the WP. The plot indicates that the temperature variation in the codisposal canister is very small due to its low heat generation rate and lower thermal resistance (see Section 6.4.4.1). Peak temperatures inside the MIT fuel and glass waste form occur 20 years after the time of emplacement as the drift wall approaches its peak temperature (Ref. 8.17). At the time of emplacement, the heat loads for glass and MIT-SNF are at their highest, but the drift and WP surfaces are still cool. By the time the drift wall temperatures reach their peak values, 40 years after emplacement, the heat load has decayed so that WP internal temperature gradients are lower. For the MIT-SNF codisposal canister, the peak internal temperature will reach 179°C in 20 years; then it will slowly cool to 152°C over the following 80 years. During the same 80 year cooling period, the temperature gradient across the WP will decrease from approximately 35°C to 8°C. For the glass waste form, peak internal temperatures will reach 182°C, 20 years after emplacement.

The temperature profile, shown in Figure 6.5.4-1, confirms that the HLW canisters reject most of their heat to the inner wall of the waste package, not to the DOE canister in the center of the waste package. The HLW canisters (SRS pour canister) heat output is two orders of magnitude greater than the DOE aluminum based waste form. Therefore, the temperature profile shown in Figure 6.5.4-1 peaks in the HLW canisters. If the heat output of the DOE aluminum based material should happen to be higher, the peak temperature would shift to the center of the waste package, inside the DOE canister.

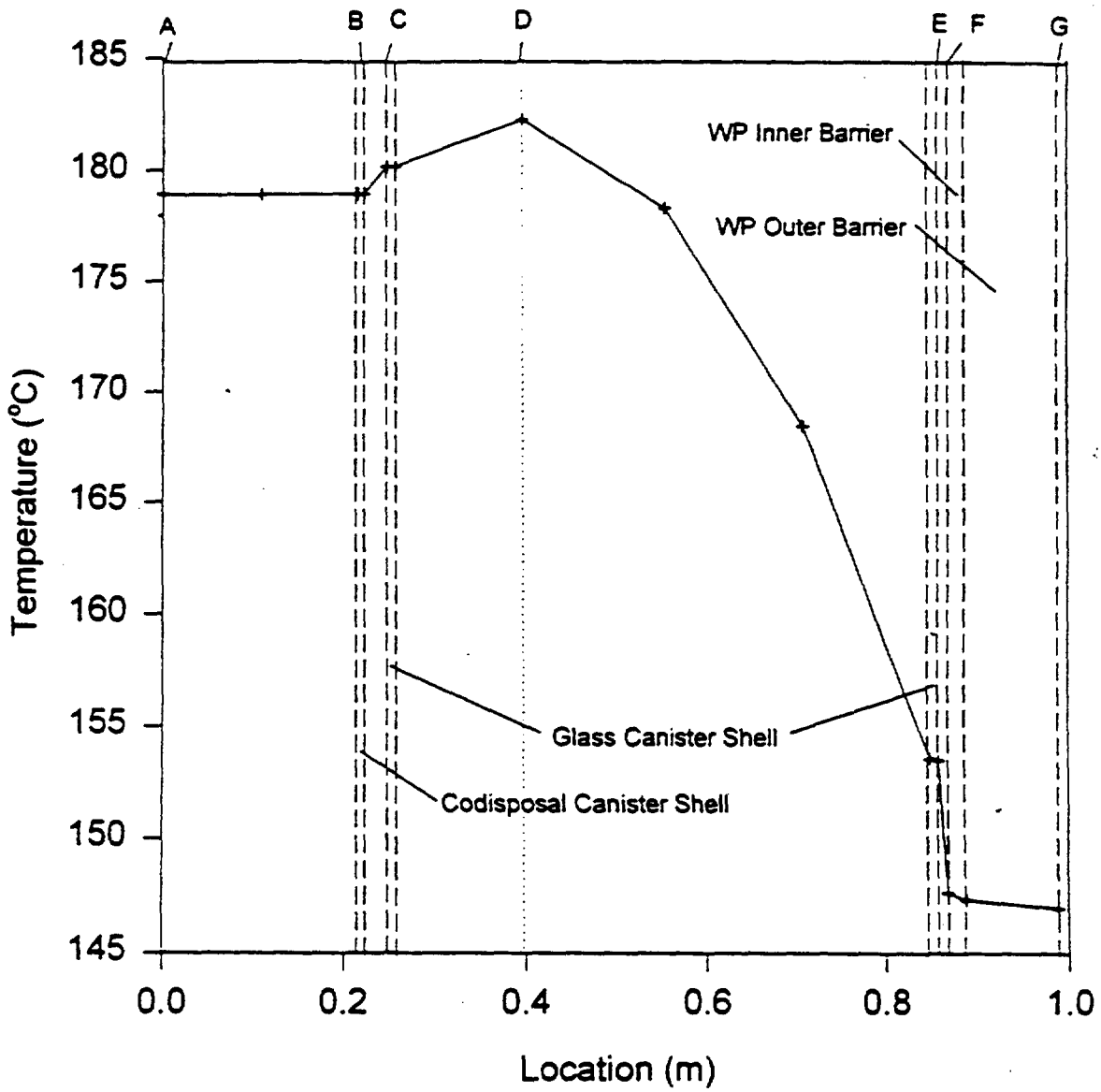


Figure 6.5.4-1. Peak Temperature Profile Across the WP (20 years after emplacement)

7. Conclusions

As identified in Sections 2.0 and 4.0, this analysis is based on unqualified/unconfirmed input data, thus the use of any data from this report for input into CRWMS documents supporting construction, fabrication, or procurement is required to be controlled as TBV in accordance with the appropriate procedures.

7.1 Analysis Results

The results of the analyses for criticality safety, shielding dose rates, structural strength, and thermal limits show that the DOE-SNF codisposal canister containing MIT or ORR fuel can meet the current requirements for repository disposal while the basket of the codisposal canister is intact and the degraded fuel remains within the codisposal canister.

7.1.1 MIT and ORR SNF Criticality

The criticality analyses performed show that the highly enriched MIT fuel can be disposed of within a codisposal canister in the codisposal waste package, provided that certain criticality control measures are implemented (e.g., borated stainless steel in-row separator plates between adjacent assemblies). Similarly, the moderately enriched ORR fuel can also be disposed of within the codisposal waste package. Evaluations of the neutronic behavior of the degraded fuel materials outside the codisposal canister (but within the waste package) will be performed as part of Phase II analyses.

7.1.2 MIT SNF Shielding

The source term comparison performed for the MIT spent fuel and the HLW canisters show that the waste package surface dose rates would not be affected by the MIT spent fuel. The analyses show that the gamma radiation dose rate contribution from the SNF in the codisposal canister and the neutron radiation dose rate contributions from both the codisposal and HLW canisters are not significant relative to the much more intense gamma source from the HLW canisters.

With regards to addressing the shielding requirement in Section 4.2.6 on increased corrosion due to radiolysis, reference 8.24 (Vol. III, p. 8-4) indicates that for iron based materials in an air/steam environment, a 100 R/hr dose rate results in a 5 times increase in corrosion rate at 250°C, and no increase in corrosion rate at 150°C. Since the WP surface dose rates are much less than 100 R/hr, and the thermal analysis (Ref. 8.20, p. 26) indicates that the codisposal WP peak surface temperature is 153°C, and occurs at 40 years following emplacement (see Table 7-1, below), it is concluded that there will be no increase in corrosion due to radiolysis. Thus the dose rates on the exterior of the codisposal waste package with the MIT SNF codisposal canister is within acceptable limits for disposal.

7.1.3 Structural

The equivalent stresses were compared to the material yield strength of the MIT-SNF codisposal canister. The stresses were higher than the yield strength in some localized regions of the shell and basket structure. However, all stresses were significantly below the ultimate tensile strength of the material. Therefore, localized permanent deformations are anticipated as a result of the dynamic loads considered in this document. However, the basic requirement is to keep the waste form from being deformed due to large displacements in the basket assembly. Hence, as long as this requirement is met in the preliminary design of the codisposal canister, small localized plastic deformations in the basket structure are not of concern.

A detailed analysis of the resulting displacements showed that the maximum deflections in the codisposal canister basket structure are smaller than the clearance available (Assumption 4.3.12) between the fuel assemblies and the basket structure. Therefore, there will be no deformation imposed on the fuel assemblies by the basket members based on the conceptual design of the MIT-SNF codisposal canister.

An alternative design was evaluated (in reference 8.19), and showed that the minimum shell thickness can be reduced by using a material with higher strength (XM-19). The recommendations for both conceptual codisposal canister designs are made in Section 7.2.

7.1.4 Thermal

Table 7-1 summarizes the peak temperatures and the time of occurrence in the WP. Peak temperatures were calculated for the glass matrix, HLW canister shell, MIT-SNF, MIT-SNF codisposal canister shell, and WP barrier.

Table 7.1.4-1 Temperature Results Summary

Peak Glass Matrix		Peak HLW Canister Outer Surface		Peak MIT SNF		Peak Codisposal Canister Outer Surface		Peak WP Outer Surface	
°C	yrs	°C	yrs	°C	yrs	°C	yrs	°C	yrs
182.3	20	180.2	20	178.9.0	20	179.0	20	153*	40*

* Note that the outer surface peak differs from the waste package surface temperature shown in Figure 6.5.4-1, because this peak occurs at 40 years following emplacement, while the temperatures shown in Figure 6.5.4-1 are at 20 years.

As indicted above, the peak glass matrix temperatures remain below 400°C, and the temperatures for the materials used in the codisposal WP are such that melting or rapid mechanical failure would not occur. The temperature for MIT SNF is also below the thermal goal of 204°C (Assumption 4.3.2). Therefore, the conceptual codisposal canister design analyzed in this document can be loaded in the codisposal waste package.

The results of the thermal evaluations indicate that the codisposal canister in the codisposal WP conceptual design can satisfy the thermal limitations (i.e., goals) during normal expected conditions for disposal in the MGDS and therefore will likely meet the MGDS requirements for repository disposal.

7.1.5 Summary

The analyses presented in this report show that the codisposal waste package can maintain criticality safety with the conceptual codisposal canister designs for the MIT and ORR spent fuel types. In addition, the MIT SNF will not have a significant effect on the surface dose rate. The structural strength of the conceptual design for the MIT fuel codisposal canister basket is adequate to prevent the fuel from being damaged in a waste package tip-over accident, although some localized plastic deformations might occur within the basket structure at high stress areas. The use of high strength type XM-19 stainless steel would provide adequate strength for the codisposal canister shell, which is not thick enough if 15 mm of Type 316L stainless steel is used to withstand the impact deceleration of 104 *g*. Another alternative would be to evolve the tip-over analysis to replace the unyielding surface methodology with a methodology which uses a physical representation of the expected impact surface. Such methodologies could reduce the g load substantially and eliminate the need for XM-19 stainless steel.

7.2 Recommendations

The following recommendations are applicable as long as the input parameters are as specified in Section 4.1:

- The MIT and ORR spent fuels can be safely disposed of in the codisposal canister, provided appropriate criticality control measures are implemented. The manufacturability of baskets should be evaluated to simplify to reduce the complexity of the designs.

- Two different stainless steels, 316L and XM-19, are deemed acceptable for the codisposal canister shell and both should be considered in the future codisposal canister designs. In general, stabilized, or austenitic stainless steels are compatible with repository disposal.

- Structural evaluations of the MIT-SNF codisposal canister designs presented in this document show that the dimensions and material properties listed in Section 4.3 are acceptable. It should be noted that these dimensions are minimum requirements for the design. If the material thicknesses are increased from the dimensions provided in the sketch (see Ref. 8.19), then the resulting stresses and displacements will be smaller; therefore, such designs will also be structurally acceptable.

It should be noted that if Phase II or Phase III evaluations are found to require additional design features (e.g.,waste package filler material), the thermal and structural evaluations may need to be reviewed.

7.3 Guidance for the Codisposal Canister Design

The results of this study should provide guidance to the designer and fabricator of the codisposal canisters for the MIT and ORR SNF, and for other DOE aluminum clad fuels having characteristics within the same envelope. The following paragraphs phrase these results to be most directly applicable to such guidance.

Thermal: Given the calculated heat load of 0.16 W/assembly, the thermal output of the fuel has no significant effect on the waste package and no additional requirements are identified.

Structural: The outer diameter of the DOE-SNF codisposal canister must be equal or less than 44 cm in order to fit in the central space of the codisposal waste package.

The codisposal canister shell should have a 15 mm wall thickness of XM-19 stainless steel or 20 mm wall thickness for 316L, with a 380 MPa yield strength to protect the fuel assemblies from being deformed under the dynamic load of 104g.

The DOE-SNF codisposal canister must be able to withstand a tip-over accident modeled by a dynamic impact simulation in which three HLW canisters and the codisposal canister interact as the uppermost HLW canister presses down upon the codisposal canister with a dynamic load of 104 *g*, and the codisposal canister presses down upon the lower two HLW canisters.

Criticality: The k_{eff} must be less than 0.95 after allowance for bias and uncertainty (ANSI/ANS-8.17) for an intact basket with both intact and degraded (homogenized) fuel within the basket assuming optimum moderator conditions and assuming only 75% credit for the neutron absorber composition.

A dispersed neutron absorber, contained in a corrosion resistant matrix (i.e., high nickel, stabilized stainless steel, austenitic stainless steel) from which the absorber is not removed (leached) at a rate faster than the fuel matrix degrades, must be utilized in the basket.

Shielding: Given that the source strengths calculated for the SNF in the codisposal canister was less than 1/100 of that for the HLW canisters (for every energy group of both gamma and neutron radiation), the radiation doses from the Al-based DOE-SNF codisposal canister have no significant effect on the total dose from the waste package and no additional requirements are identified.

7.4 Future Plans

The degradation of aluminum clad fuel can result in the redistribution of uranium materials from the original location within the codisposal canister to areas between the HLW canisters, within the waste package. The potential effects of fuel relocation within the codisposal waste package will be evaluated in Phase II.

8. References

8.1 *Yucca Mountain Site Characterization Project Q-List*, YMP/90-55Q, REV 4, U.S. Department of Energy (DOE) Office of Civilian Radioactive Waste Management (OCRWM).

8.2 *Quality Assurance Requirements and Description*, DOE/RW-0333P, REV 7 U.S. DOE OCRWM.

8.3 *Data Package from Savannah River Criticality Analysis of MIT & ORR SNF*, Including Appendix A data for MIT & ORR fuel, drawings R3F-3-2, R3F-1-4, M-11495-OR-001, M-11495-OR-003, and M-11495-OR-004, Records Batch # MOY-970605-02.

8.4 *Repository Design Requirements Document*, DOE/RW-0404P, Document Identifier (DI) Number: B00000000-00811-1708-00002 REV 02, U.S. DOE OCRWM.

8.5 *Engineered Barrier Design Requirements Document*, YMP/CM-0024, REV 0, ICN 1, Yucca Mountain Site Characterization Project.

8.6 *Controlled Design Assumptions Document*, DI Number: B0000000-01717-4600-00032 REV 04, Civilian Radioactive Waste Management System Management and Operating Contractor (CRWMS M&O). Note that REV 04 ICN 01 was not referenced as it was not used in the supporting QAP 3-9 analyses for this document.

8.7 *Characteristics of Potential Repository Wastes*, DOE/RW-0184-R1; Volume 1, U.S. DOE OCRWM.

8.8 *Finite-Element Analysis of Slap Down of Defense High-Level Waste Package,* DI Number: BBAC00000-01717-0200-00007 REV 00, CRWMS M&O.

8.9 *1995 ASME Boiler and Pressure Vessel Code*, Section II, American Society of Mechanical Engineers, 1995.

8.10 *Software Qualification Report for MCNP4A*, DI Number: 30006-2003 REV 02, CRWMS M&O.

8.11 *Software Qualification Report for The SCALE Modular Code System Version 4.3*, DI Number: 30011-2002 REV 01, CRWMS M&O.

8.12 *Software Qualification Report for ANSYS Version 5.1HP* (CSCI: 30003 V5.1HP), DI Number: 30003-2006 REV 01, CRWMS M&O.

8.13 *Life Cycle Plan for ANSYS Version 5.1HP* (CSCI: 30003 V5.1HP), DI Number: 30003-2007 REV 01, CRWMS M&O.

8.14 *BW-2901 Transportation Package*, US NRC Certificate of Compliance 71-9251.

8.15 *Neutronic Evaluation of the Codisposal Canister in the Five-Pack DHLW Waste Package*, DI Number: BBA000000-01717-0200-00052 REV 00, CRWMS M&O.

8.16 *Roark's Formulas for Stress & Strain,* Sixth Edition, Warren C. Young, McGraw-Hill Book Company, New York, NY, 1989.

8.17 *Thermal Evaluation in Support of Thermal Loading System Study*, R. H. Bahney III, Interoffice Correspondence LV.WP.RHB.08/96-201, CRWMS M&O, August 15, 1996.

8.18 *ANSI/ANS-6.1.1-1977, "American National Standard Neutron and Gamma-Ray Flux-to-Dose Rate Factors*, American Nuclear Society, LaGrange Park, IL, 1977.

8.19 *Structural Evaluation of MIT SNF Codisposal Canister*, DI Number: BBA000000-01717-0200-00051 REV 00, CRWMS M&O.

8.20 *Thermal Evaluation of the Codisposal Canister in the 5-Pack DHLW Waste Package*, DI Number: BBAA00000-01717-0200-00021 REV 01, CRWMS M&O.

8.21 Reserved.

8.22 *International Handbook of Evaluated Criticality Safety Benchmark Experiments*, NEA/NSC/DOC(95)03/I, Volume II.b, Nuclear Energy Agency, Organization for Economic Co-operation and Development, November 4, 1996 update.

8.23 *Thermal Evaluation of the Conceptual DHLW Disposal Container*, DI Number: BBAA00000-01717-0200-00003 REV 00, CRWMS M&O.

8.24 *Mined Geologic Disposal System Advanced Conceptual Design Report*, DI Number: B00000000-01717-5705-00027 REV 00, CRWMS M&O.

8.25 *Marks' Standard Handbook for Mechanical Engineers*, 9th Edition, E. A. Avallone, T. Baumeister, McGraw-Hill Book Co., 1987.

8.26 *Nuclear Chemical Engineering*, M. Benedict, T. H. Pigford, and H.W. Levi, McGraw-Hill Book Company, New York, 1981.

8.27 *Introduction to Heat Transfer*, Second Edition, F. P. Incropera and D. P. DeWitt, John Wiley & Sons Inc., 1990.

8.28 *Principles of Heat Transfer in Porous Media*, Second Edition, M. Kaviany, Springer-Verlag New York, Inc., 1995.

8.29 *Materials Engineering, Materials Selector 1991*, A Penton Publication, December 1990.

8.30 *Structural Alloys Handbook,* Vol. 2, C. Y. Ho, ed., Cindas/Purdue University, West Lafayette, IN, 1992.

8.31 *Properties and Selection: Stainless Steels, Tool Materials and Special Purpose Metals,* Metals Handbook, Ninth Edition, Vol. 3, American Society for Metals, Metals Park, OH, 1980.

8.32 *Materials Engineering Materials Selector 1991,* Clifford F. Lewis, ed., Penton Publication, Cleveland, OH, December 1990.

8.33 *Standard Specification for Heat-Resisting Chromium-Nickel Stainless Steel Plate, Sheet, and Strip for Pressure Vessels*, ASTM Designation: A 240 - 91a, American Society for Testing and Materials.

8.34 *Waste Acceptance Product Specifications for Vitrified High-Level Waste Forms*, EM-WAPS Rev. 02, U. S. Department of Energy Office of Environmental Management, December 1996.

8.35 *Report of the U.S. Department of Energy Environmental Management Criticality Safety Workshop, February 24-26, 1997, National Spent Nuclear Fuel Program*, INEL/EXT-97-00543, Idaho National Engineering Laboratory/U. S. Department of Energy Office of Environmental Management, May 1997.

8.36 *10CFR60; Disposal of High-Level Radioactive Wastes in Geologic repositories; Design Basis Events; Final Rule,* U.S. Nuclear Regulatory Commission, Federal Register, volume 61, Number 234, pp. 64257-64270, December 4, 1996.

8.37 *Waste Package Design Basis Events*, DI Number BBA000000-01717-0200-00037 REV 00, CRWMS M&O.

8.38 *DHLW Canister Source Terms for Waste Package Design*, DI Number: BBA000000-01717-0200-00025 REV 00, CRWMS M&O.

8.39 *Technical Document Preparation Plan For The Evaluation of Codisposal Viability for DOE-Owned Research Reactor Spent Fuel Reports*, DI Number: BBA000000-01717-4600-00010 REV 01, CRWMS M&O.

8.40 *Martin Marietta Energy System Specification SM519-93*, April 7, 1993.