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BREEDER SPENT FUEL HANDLING PROGRAM MULTIPURPOSE CASK DESIGN BASIS DOCUMENT

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BREEDER SPENT FUEL HANDLING PROGRAM MULTIPURPOSE CASK
DESIGN BASIS DOCUMENT

A. J. Duckett
K. B. Sorenson

ABSTRACT

The Breeder Spent Fuel Handling (BSFH) Program multi-purpose cask Design Basis Document defines the performance requirements essential to the development of a legal weight truck cask to transport FFTF spent fuel from reactor to a reprocessing facility and the resultant High Level Waste (HLW) to a repository.

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CONTENTS

	<u>Page</u>
Abstract	iii
Acronyms	ix
1.0 INTRODUCTION	1
2.0 SCOPE	1
3.0 APPLICABLE DOCUMENTS	1
3.1 GENERAL	1
4.0 DESIGN	2
4.1 MATERIAL PROPERTIES	2
4.1.1 General	2
4.1.2 Containment Boundary Components	3
4.1.3 Balance of Packaging Components	3
4.2 CASK STRUCTURAL DESIGN REQUIREMENTS	4
4.2.1 General	4
4.2.2 Evaluation Methods	4
4.2.3 Normal Conditions	5
4.2.3.1 Containment Boundary Components	5
4.2.3.2 Balance of Packaging Components	6
4.2.4 Hypothetical Accident Conditions	6
4.2.4.1 Containment Boundary Components	6
4.2.4.2 Balance of Packaging Components	7
4.3 THERMAL REQUIREMENTS	7
4.3.1 Surface Temperature	7
4.3.2 Thermal Expansion	7
4.3.3 Seal Material	7
4.4 CONTAINMENT REQUIREMENTS	8
4.4.1 Normal Transport Conditions	8
4.4.2 Hypothetical Accident Conditions	8

CONTENTS (Cont'd)

	<u>Page</u>
4.5 SHIELDING REQUIREMENTS	8
4.5.1 Surface Dose Rate	8
4.5.2 Operational Dose Rate	8
5.0 PAYLOAD DEFINITION	9
5.1 PHYSICAL	9
5.2 SOURCE TERM	9
5.3 CRITICALITY	9
5.4 THERMAL	10
5.5 LEAKAGE SOURCE TERM	10
6.0 SYSTEM REQUIREMENTS	10
6.1 BASKET DEFINITION	10
6.1.1 Physical	10
6.1.1.1 Interface with Payload	10
6.1.1.2 Interface with Cask	11
6.1.2 Criticality	11
6.1.3 Thermal	11
6.1.4 Mechanical	11
6.2 CASK DEFINITION	12
6.2.1 Physical	12
6.2.2 Technical	12
6.2.2.1 Criticality	12
6.2.2.2 Thermal	13
6.2.2.3 Shielding	13
6.2.2.4 Structural	14
6.2.3 Closure	14
6.2.3.1 Bolts	15
6.2.3.2 Seals	15
6.2.3.3 Remote Handling	15

CONTENTS (Cont'd)

	<u>Page</u>
6.2.4 Fabrication	16
6.2.5 Impact Protection	16
6.2.5.1 Material	16
6.2.5.2 Energy Attenuation	16
6.2.5.3 Attachment	16
6.3 TRANSPORTER DEFINITION	17
6.3.1 Cask/Transporter Physical Interface	17
6.3.2 Weight Criteria	17
6.3.3 Cask Support Saddles and Tie-Down System	17
6.3.4 Sun Screen	18
6.3.5 Safety	18
6.3.6 Dedicated Systems	18
7.0 CASK/FACILITY INTERFACE	18
7.1 RECEIVING/SHIPPING	18
8.0 OPERATION AND MAINTENANCE	19
8.1 AVAILABILITY	19
8.2 DECONTAMINATION	19
8.3 TURN-AROUND TIME	19
8.4 HANDLING	20
8.4.1 Trunnions	20
8.4.2 Remote Handling Capabilities	20
8.5 MAINTENANCE	20
9.0 QUALITY ASSURANCE PLAN	20
10.0 REFERENCES	21
APPENDIX A REGULATIONS, CODES AND STANDARDS	A-1
APPENDIX B PHYSICAL PARAMETERS FOR FFTF FUEL ASSEMBLIES	B-1

CONTENTS (Cont'd)

	<u>Page</u>
APPENDIX C PHYSICAL PARAMETERS FOR HIGH LEVEL LIQUID WASTE (HLLW) CONTAINER	C-1
APPENDIX D GAMMA AND NEUTRON SOURCE TERMS FOR FFTF SPENT FUEL	D-1
APPENDIX E VITRIFIED WASTE LOG GAMMA RADIATION SOURCE TERMS	E-1
APPENDIX F TRUNNION SKETCH	F-1

ACRONYMS

AASHTO	-	American Association of State Highway and Transportation Officials
ANSI	-	American National Standards Institute
ASME B&PV Code	-	American Society of Mechanical Engineers Boiler and Pressure Vessel Code
ASTM	-	American Society of Testing Materials
BSFH	-	Breeder Spent Fuel Handling
CFR	-	Code of Federal Regulations
FFTF	-	Fast Flux Test Facility
HEDL	-	Hanford Engineering Development Laboratory
HLW	-	High Level Waste
LMR	-	Liquid Metal Reactor
LSPB	-	Large Scale Prototype Breeder
NRC	-	Nuclear Regulatory Commission
RH	-	Remote Handling
SNL/TTC	-	Sandia National Laboratory/Transportation Technology Center
WHC	-	Westinghouse Hanford Company

BREEDER SPENT FUEL HANDLING PROGRAM MULTIPURPOSE CASK
DESIGN BASIS DOCUMENT

1.0 INTRODUCTION

The Breeder Spent Fuel Handling (BSFH) Program objective is to design, fabricate, license and test a legal weight truck transportation system to service the back-end of the breeder cycle in the shipment of spent fuel from reactor sites to a reprocessing facility, also considering the needs of High Level Waste (HLW) resulting from the reprocessing activities. The cask to be developed should be designed to allow the incorporation of future family members, thus reducing the design and licensing efforts. Future family members may include payloads for the large scale prototype breeder fuel, process facility modification, alternate reactor concepts fuel and high level waste logs from the West Valley Demonstration Project.

2.0 SCOPE

The activities being performed within the scope of this document are sponsored by the Department of Energy, Division of Fuels and Reprocessing, Germantown, Maryland. This document identifies and defines design constraints and lists the basic performance requirements essential to the development of an engineering design for a legal weight truck packaging (cask) and transportation system to transfer spent fuel from reactors to reprocessing facilities and the resulting High Level Waste to repositories. The scope of this document includes only FFTF spent fuel and HLW vitrified glass logs from reprocessing FFTF spent fuel. Additional cask payload requirements and other pertinent information will be covered by addenda to this document, when the need arises.

3.0 APPLICABLE DOCUMENTS

3.1 GENERAL

The applicable documents (Appendix A) form a part of the Design Basis Document to the extent specified in the applicable sections of this document.

In the event of conflict between documents referenced herein and the requirements of this Design Basis Document, the requirements of this document shall be considered a superseding requirement.

4.0 DESIGN

The cask shall be designed to Type B criteria (10 CFR 71) and must be capable of providing the necessary protection to the payload, when subjected to the normal conditions of transport and to the hypothetical accident conditions defined in 10 CFR 71.71 and .73.

4.1 MATERIAL PROPERTIES

4.1.1 General

Materials used in the structural fabrication of the cask shall meet the requirements of the ASME B&PV Code, Sections II and III. Any nonmetallic sealing material will be exempt from this requirement.

Structural material shall be classified as Class 1, as referenced by NRC Regulatory Guide 7.6 and as defined in ASME B&PV Code, Section III, Appendix I. Values for material properties, such as design stress intensities (S_m) and design fatigue curves, are also defined in Section III, Appendix I.

Material properties, such as non-linear stress-strain behavior and strain rate dependent yield and flow data, not available in the ASME code, may be considered in analysis provided the values are technically sound.

In accordance with 10 CFR 71.43, the packaging shall be designed using material that will inhibit chemical, galvanic or other reactions among the packaging components, or between the packaging components and the packaging contents. Minimum wall thickness shall account for any corrosion allowance.

4.1.2 Containment Boundary Components

The cask body, lids and any inner containment liners shall be designed from austenitic stainless steel. Material composition is defined in the ASME B&PV Code, Section II. Material integrally welded to the containment boundary components, such as neutron shielding covers, impact limiters, impact limiter frames or trunnions, shall be made from austenitic stainless steel.

Gamma-ray shielding shall be steel, lead or depleted uranium. Shielding shall be designed according to the limiting criteria defined in Section 4.5. Weight-to-shielding efficiency should determine the material selection for shielding.

The containment O-ring seals shall be compatible with the sealing requirement of ANSI 14.5, Section 5.0 and must be resistant to radiation and other environmental conditions. Any material type must be capable of maintaining a seal during and after hypothetical accident condition events (Section 4.2.4).

4.1.3 Balance of Packaging Components

Handling equipment, supports and tie-downs shall be designed to satisfy the equipment requirements. The cask supports and tie-downs shall be designed to withstand the forces specified in Section 6.3. Bolts shall be selected to satisfy the strength, compatibility and brittle fracture requirements of the design.

To protect against brittle fracture, ferritic steel components shall meet the requirement that two out of three Charpy V-Notch tests, per ASTM VE-23, at a temperature of -20°F shall exhibit impact energies greater than 20 ft-lb.

Neutron shielding material shall provide protection against neutron radiation, as specified 10 CFR 71.47.

The impact limiter material is described in Section 6.2.5.

4.2 CASK STRUCTURAL DESIGN REQUIREMENTS

4.2.1 General

10 CFR 71 provides the requirements for demonstration of compliance of the package with regulatory loading and environmental conditions. The NRC Regulatory Guide 7.6 describes criteria acceptable to the NRC staff for evaluation of structural design. Load combinations to be evaluated are described in NRC Regulatory Guide 7.8.

Load values to be used for design for both normal conditions and hypothetical accident conditions are defined in 10 CFR 71, Subpart F. The loading criteria is not repeated in this document. Structural loads resulting from vibration in transport shall be considered as normal loads in the cask design.

Handling equipment, such as the trunnions, shall be designed to satisfy the requirements of ANSI N14.6 - 1978, "American National Standard for Special Lifting Devices for Shipping Containers Weighting 10,000 Pounds or More for Nuclear Materials." For this product specification, the cask lift shall be considered critical. Therefore, a redundant load path lifting system shall be required.

Cask supports and tie-downs shall be designed to withstand the forces as specified in Section 6.3.3.

4.2.2 Evaluation Methods

Paragraph 71.41 of 10 CFR 71 outlines the manner in which compliance with the regulations can be shown:

71.41(a) Demonstration of Compliance

"The effects on a package of the tests specified in 71.71 (Normal Conditions of Transport) and 71.73 (Hypothetical

Accident Conditions) must be evaluated by subjecting a sample package or scale model to test, or by other methods of demonstration acceptable to the commission, as appropriate for the particular feature being considered."

For the BSFH Design, "Method of Demonstration" is interpreted as computer analysis or a combination of computer analysis and field testing. In order to demonstrate compliance, any combination of the following analytical/test techniques can be used.

- 1) Full-scale replication of the package with sufficient instrumentation to resolve the maximum stresses and deformations, and the comparisons with these measured quantities with allowable values as defined in ASME B&PV Code Section III, Appendix I.
- 2) Scale model replication, with justification of the scaling laws used, followed by a procedure equivalent to that of (1) above.
- 3) Replication, either at full-scale or sub-scale, designed to verify and qualify an analytical model. The analytical model is then used to evaluate the package to prescribed normal and hypothetical accident conditions. The results are compared to allowable values as defined in ASME B&PV Code Section III, Appendix I.

4.2.3 Normal Conditions

4.2.3.1 Containment Boundary Components

Ductile Rupture -- The general primary stress intensity, derived from the average value across the thickness of the section and produced by the normal condition mechanical loads, shall be limited to S_m as defined by the ASME B&PV Code, Section III. The local primary membrane plus primary bend stress intensities ($P_1 + P_b$) shall be limited to $1.5 S_m$. Secondary and peak stresses are excluded in determining primary membrane and bending stresses.

Fatigue -- The fatigue requirements of Subparagraph NB-3222.4 of the ASME B&PV Code, Section III, shall be met where the primary plus secondary stress intensities, excluding peak stresses, do not exceed $3 S_m$. If the primary plus secondary stress intensity exceeds $3 S_m$, the criteria of ASME B&PV Code Section III, Paragraph NB-3228, may be used in lieu of inelastic analytical methods.

Buckling -- Regulatory position C.5 from NRC Regulatory Guide 7.6 states that buckling shall not occur for BSFH containment boundary components subjected to normal transport loads. Compressive stresses shall be limited to two-thirds of the buckling stress. This is consistent with ASME B&PV Code Section III, Appendix F.

4.2.3.2 Balance of Packaging Components

The balance of packaging components shall withstand the normal conditions of transport with no substantial reduction in the effectiveness of the packaging. To protect against brittle fracture, ferritic steel components must meet the Charpy Values specified in Section 4.1.3.

4.2.4 Hypothetical Accident Conditions

4.2.4.1 Containment Boundary Components

Ductile Rupture -- The containment boundary shall meet one of the following criteria (see Section 4.2.1):

- 1) Elastic Analysis - P_m shall be less than or equal to the lesser value of $2.4 S_m$ and $0.7 S_u$. Also, $(P_1 + P_b)$ must be less than or equal to the lesser value of $3.6 S_m$ and S_u (as defined in ASME B&PV Code Section III).
- 2) Inelastic Analysis - P_m and $(P_1 + P_b)$ must be less than or equal to the greater value of $0.7 S_u$ and $S_y + 1/3 (S_u - S_y)$.

Buckling -- For BSFH containment boundary components subjected to hypothetical accident loads, compressive stresses shall be limited to two-thirds of the critical buckling stress.

4.2.4.2 Balance of Packaging Components

The balance of the BSFH cask components shall withstand each hypothetical accident condition to the degree required to continue providing protection during the subsequent sequence of hypothetical accident conditions.

4.3 THERMAL REQUIREMENTS

4.3.1 Surface Temperature

In accordance with the requirements in 10 CFR 71.43, the accessible external surfaces of the shipping cask shall not exceed 82°C (180°F) assuming still air with an ambient temperature of 38°C (100°F) under normal transport conditions.

4.3.2 Thermal Expansion

All components shall be designed to accommodate differential thermal expansion.

4.3.3 Seal Material

The selected seal materials shall be compatible with operating temperatures during both normal and hypothetical accident conditions.

The seal materials must maintain their integrity and not exceed their maximum permissible leakage rates (Reference ANSI 14.5 Paragraph 5.3) during normal and hypothetical thermal loading conditions as defined in 10 CFR 71.71 and 71.73.

4.4 CONTAINMENT REQUIREMENTS

4.4.1 Normal Transport Conditions

Under normal transport conditions, as defined in 10 CFR 71.51, the maximum loss or dispersal of radioactive contents is $10^{-6} A_2$ per hour. A_2 values for single radionuclides are defined in 10 CFR 71.4 and Appendix A of 10 CFR 71, for a mixture of radionuclides, the A_2 value will be calculated per Paragraph II of Appendix A. ANSI N14.5 shall be used as guidance for leakage assessment and testing.

4.4.2 Hypothetical Accident Conditions

Under hypothetical accident conditions, there should be no escape of Krypton-85 exceeding 10,000 Ci/week and no escape of other radioactive materials exceeding a total amount of A_2 in one week. A_2 shall be defined the same as in Section 4.4.1.

4.5 SHIELDING REQUIREMENTS

4.5.1 Surface Dose Rate

The design of the shielding for the BSFH cask shall satisfy the requirements of 10 CFR 71.47 for a dose rate of 200 mrem/hr or less for readily accessible surfaces and a rate of 10 mrem/hr or less at 2 meters (6.6 ft) from any accessible external surface of the vehicle or cask. In addition, the radiation level in any normally occupied position of the transport vehicle shall be 2 mrem/hr or less.

4.5.2 Operational Dose Rate

The shielding shall be designed to minimize radiation exposure to operating personnel. Consideration shall be given to the location of shielding to minimize the dose rate where contact operations may be required.

5.0 PAYLOAD DEFINITION

The payloads to be addressed include the spent fuel from FFTF and the HLW from reprocessing FFTF spent fuel. Spent fuel definitions are in Appendices B through E of this report.

5.1 PHYSICAL

The physical dimensions of each payload assembly are given in Appendixes B and C. These data provide the general mechanical detail information on the assembly, such as pin diameter, active length of fuel, quantity of fuel etc.

5.2 SOURCE TERM

The average photon and neutron source per assembly, decay heat and radiation dose for FFTF spent fuel and HLW log are given in Appendixes D and E, respectively.

5.3 CRITICALITY

Federal Regulations (10 CFR 71.55) require every shipment of fissile material to remain subcritical at all times during normal transport, including loading, unloading and under hypothetical accident conditions leading to the most reactive credible configuration. The criticality of a system is discussed in terms of an effective multiplication factor, $K_{\text{effective}}$, which is defined as the ratio of the neutron production rate to the neutron loss rate of the system. The cask fuel system must remain subcritical, so that the K_{eff} is less than 1.0.

The limit for K_{eff} is 0.95 for normal and hypothetical accident conditions, including corrections for uncertainties. The criticality analysis should be made for the case where the fuel is in the most reactive condition in which it will be transported. This would normally be based on fresh (green) fuel. The criticality of each payload configuration must be determined and the

design of the inner basket shall contain the appropriate materials, if required, to ensure that the K_{eff} is less than 0.95. Neutron source terms are supplied in Appendix E.

5.4 THERMAL

The decay heat for each FFTF spent fuel assembly and HLW log are given in Appendixes D and E, respectively. The maximum temperature limit for FFTF spent fuel cladding is 1100°F. The maximum centerline temperature of the HLW vitrified log shall not exceed 932°F.

5.5 LEAKAGE SOURCE TERMS

The leakage source terms is given in Sections 4.4.1 and 4.4.2.

6.0 SYSTEM REQUIREMENTS

6.1 BASKET DEFINITION

Baskets may be designed to accommodate one or multiple payload types. All basket designs, however, must be compatible with the cask as defined in this section. Design goals, other than specific criteria used below, shall be ease of payload assembly loading/unloading, ease of basket insertion/removal from the cask, and ease of decontamination.

6.1.1 Physical

6.1.1.1 Interface with Payload

The baskets shall be designed to accommodate two payloads: FFTF spent fuel and HLW logs. Physical dimensions of each payload assembly are listed in Appendixes B and C of this document. The assemblies per cask for each payload are:

FFTF	Goal - seven assemblies/cask, maximum decay heat of 1.2 kW/assembly Minimum Acceptable - five assemblies/cask, decay heat of 1.0 kW/assembly
HLW Logs	One assembly/cask, decay heat of 7.6 kW

Each fuel assembly shall be held within the basket to prevent vibration between the basket and the fuel assembly. The fuel assembly shall be longitudinally supported. The basket shall allow for 1 in. of bow in each fuel assembly.

6.1.1.2 Interface with Cask

The BSFH cask inner cavity dimensions shall be established during the design phase. The basket shall be designed to fit into the cask cavity to minimize gaps between the basket and cask interface. However, operational requirements must be met. Differential thermal expansion between the basket and cask must be taken into consideration in determining clearances.

6.1.2 Criticality

The basket must be designed to prevent criticality occurring because of interaction between fuel assemblies. Neutron absorption can be accommodated by inserts or can be incorporated by fabrication of the complete basket from a neutron-absorbing alloy, such as boron/aluminum.

6.1.3 Thermal

The basket functions as a mechanism to transfer heat generated from the payload contents to the inner wall of the cask body via radiation and conduction. Materials shall be selected to accommodate this transfer. Differential thermal expansion of dissimilar metals shall be considered in the design.

6.1.4 Mechanical

The basket is to be designed as a removable assembly to allow the cask to accommodate multiple payloads.

For the payload assembly housing within the basket, re-entrant corners are to be radiused and outside corners shall be chamfered for decontamination and payload assembly installation purposes.

The basket can be either monolithic or an assembly of discrete components. Design considerations shall include ease of fabrication, assembly, and installation. Gaps in discrete components should be eliminated so as not to weaken the thermal and structural properties of the basket, to prevent gamma-ray streaming and to aid in decontamination.

6.2 CASK DEFINITION

The cask should be designed for remote handling and ease of decontamination. It should have the capability of transport on a legal weight truck. The design should allow commonality for the addition of other family members. The shielding must be weight and cost effective. The cask shall be designed with double containment and have the structural capability to withstand the hypothetical accident scenario. The cask design will accommodate dry loading/unloading.

6.2.1 Physical

The cask and associated hardware, including the transporter, shall not weigh more than 80,000 lb. The cask length will be dependent on the payloads to be contained by the cask. The inner shell material must be resistant to caustic corrosion induced by sodium-wetted core assemblies. The inner shell and outer surface should also be easily decontaminated. A design goal for the impact limiters would be an integral system with the cask body.

6.2.2 Technical

6.2.2.1 Criticality

The cask and its internals must be designed to assure that fissile materials cannot reach a critical configuration either under normal operation or under

a hypothetical accident scenario. Nuclear criticality safety will be evaluated for each type of fuel arranged in its most reactive condition.

6.2.2.2 Thermal

Design of the shipping cask must address heat transfer from the payloads to the atmosphere surrounding the cask. The method(s) of heat transfer considered in the cask design phase are dictated by the requirements of maintaining fuel pin cladding temperature and HLW canister centerline temperature within the prescribed limits (Reference Section 5.4 of this report).

The material selected for the cask containment and shielding (gamma-ray and neutron) must be designed to accommodate any differential thermal expansion. The design should allow thermal protection for the seals from internal payload temperatures and from the heat generated by external fires in the hypothetical accident, as defined in 10 CFR 71.73 (see Section 4.3.1 of this report).

To improve the heat transfer, considerations shall be given to treating the cask and basket surfaces to increase the emissivity. The surface contact and thermal conductance between material interfaces should be reviewed for maximum efficiency. Also, it may be necessary to develop techniques (such as heat pipes) to conduct the heat axially from the fuel region and disperse it along the length of the fuel basket and corresponding portions of the cask.

6.2.2.3 Shielding

The design of the shielding for the multipurpose/multiassembly shipping cask shall satisfy the requirements of 10 CFR 71.47 (see Section 4.5.1). The shielding design should take into consideration shaped shielding to minimize the cask weight. The shaped shielding concept can also be utilized in the inner basket to allow for flexibility in the cask utilization.

The material selected for the gamma-ray shielding is limited to steel, lead and depleted uranium. NRC has become more critical of the use of lead as

shielding. There is concern that the lead might melt, leaving the cask without adequate shielding. The hypothetical accident fire test, 10 CFR 71.73, requires the cask and shielding to survive for not less than 30 min of a heat flux not less than that of a radiation environment of 800°C (1475°F). Therefore, steel and depleted uranium provide the most logical choices. The depleted uranium provides the most efficient weight-to-diameter ratio. However, fabrication costs for the depleted uranium are greater than steel. The shielding thickness will be controlled by the most restrictive source terms listed in Section 5.2.

6.2.2.4 Structural

The structural design for the multipurpose/multiassembly shipping cask shall satisfy the requirements of 10 CFR 71, Subpart E and F, as outlined in Section 4.2. The cask inner and outer surfaces must be designed for ease of decontamination. All welds should be ground smooth and there should be no cracks or crevices to capture contaminant particles.

The cavity size of 25-3/4 in. ID was selected in the preconceptual design and is compatible with the FFTF spent fuel assemblies and the HLW log from reprocessing LMR spent fuel. Maintaining a common diameter for the various family members should improve the time required for NRC review and licensing. The cavity length will vary with the family member. It is anticipated that the family membership could consist of five casks with the same diameter, but of different lengths.

6.2.3 Closure

The BSFH Multipurpose/Multiassembly Cask shall be designed to provide a double containment boundary. The inner shield plug can be made of steel or a combination of steel and depleted uranium. A gas sampling device must be included for the cask cavity.

6.2.3.1 Bolts

Bolt sizes, quantity and material shall be selected to satisfy the strength, material compatibility and brittle fracture requirements of the design, as stated in Section 4.2.3.2. The bolt sizing for the inner shield plug and the outer cover shall be capable of withstanding the forces imposed by the hypothetical accident scenario (Reference 10 CFR 71.73). The selection of bolting material and thread design should consider the potential for galling. Standardization of bolt sizes should be considered to reduce tool inventory, etc. The bolts shall be designed for remote application (see Section 6.2.3.3).

6.2.3.2 Seals

The containment seals may be metallic, elastomer and/or a combination fabrication. The material must be capable of maintaining a seal during and after all hypothetical accidents, as well as all normal operating temperatures and pressures (Reference ANSI 14.5). The closure system must be designed to retain the seals. However, the seals must be easily replaceable (if required) at facilities not equipped to perform scheduled maintenance. The seal/closure design must include a system to verify the integrity of the seals.

6.2.3.3 Remote Handling

The cask design shall include features that allow the cask and contents to be loaded/unloaded with remote equipment. The design shall include:⁽¹⁾

- Details that allow remote alignment of component
- Simple trunnions (see Appendix F)
- Captive spring-loaded bolts for closure plate, shield plug, tie-downs, lifting fixtures, sunshields and impact limitors (if attached)
- Basket
- Remote gas sampling and leak testing system

6.2.4 Fabrication

The cask shall be designed to allow fabrication of the cask by standard manufacturing practices. All structural welds shall be defined as full penetration and all welded gaps and crevices shall be ground smooth on the inner and outer surface of the cask for ease of decontamination. The design requirements must be followed during fabrication to insure a good thermal bond between layers of the cask wall.

6.2.5 Impact Protection

The impact protection system limits the forces introduced to the cask and its contents during the hypothetical accident conditions. The impact system shall protect the cask from forces applied to both the exterior and interior surfaces.

6.2.5.1 Material

The material selection for the external and internal impact limitors should provide the necessary protection for the cask and payload.

6.2.5.2 Energy Attenuation

The impact protection system performs its primary function through the energy absorbing mechanism of inelastic deformation. This deformation takes the form of crushing or collapse of the materials to absorb energy. The energy attenuation must be capable of withstanding the hypothetical accident scenario, as defined in 10 CFR 71 Subpart F. It is particularly important to protect the cask closure sealing surface from damage and/or deformation.

6.2.5.3 Attachment

The attachment of the impact limitors can be accomplished by bolt-on assemblies or as an integral assembly as discussed in Paragraph 6.2.1. Both

concepts would require the attachment to withstand the forces induced by the hypothetical accident scenario.

6.3 TRANSPORTER DEFINITION

6.3.1 Cask/Transporter Physical Interface

Physical dimensions of the tractor-trailer rig shall conform to the criteria specified by the American Association of State Highway and Transportation Officials (AASHTO) regulations. The tractor-trailer rig plus contents (cask, sun screen, etc.) shall conform to all AASHTO regulations. Maximum height above the ground for the transporter plus contents shall be 13 ft - 0 in. The cask shall be shipped in the horizontal position.

6.3.2 Weight Criteria

The maximum weight of the transporter plus contents shall be 80,000 pounds. Maximum axle weight shall be as specified in the AASHTO regulations. The cask saddles shall be located on the trailer in such a manner as to distribute the cask weight to the trailer axles in proportion to their weight limitations.

6.3.3 Cask Support Saddles and Tie-Down System

The cask-to-transporter tiedown system shall be designed such that under normal conditions, the cask can be released from the transporter by a standard remote system and the cask removed from the transporter within a 15-minute time frame, using remote systems and one (1) operator.⁽¹⁾ The support saddles should be designed so that the cask could be off-loaded horizontally or rotated and off-loaded vertically.

The tie-down system and cask supports shall be designed to resist loads based on the forces given as follows:

Longitudinal	10 times the cask weight plus contents
Lateral	5 times the cask weight plus contents
Vertical	2 times the cask weight plus contents

The design loads will be applied to the center of gravity of the cask.

6.3.4 Sun Screen

Based on cask design skin temperatures, a sun screen may be required (Reference Section 4.3.1). The sun screen shall be designed to maximize the ease of the cask handling chores around the transporter.

6.3.5 Safety

Radiation levels around the transporter are defined in Section 4.5.1 and in 10 CFR 71.47. Non-fixed (removable) radioactive contamination on the external surfaces of the cask shall not exceed limits prescribed in 49 CFR 173.443. The cask surfaces will be resistant to contamination leaching.

6.3.6 Dedicated Systems

The tractor-trailer rig plus associated ancillary components shall be regarded as a dedicated system for exclusive use in radioactive cask transportation.

7.0 CASK/FACILITY INTERFACE

7.1 RECEIVING/SHIPPING

The cask/facility interface assumptions to be used for the cask design, are as follows:

- Cask loaded weight not to exceed 25 tons

- The maximum diameter not to exceed 6 feet
- The maximum length not to exceed 18 feet
- Cask and associated lifting fixtures shall be designed for minimum hook height of 30 feet
- Redundant lifting trunnions provided for in-plant handling
- The inner closure plug will be less than 4 1/2 tons and have a diameter less than 36 inches
- No special maintenance shall be required at loading/unloading station; however, seal replacement shall be possible
- The allowable radiation levels external to the cask are provided in Section 6.3.5
- The accessible external surface temperature of this cask shall not exceed 180°F (Section 4.3.1)
- The cask must provide a gas sampling port and method of seal verification

8.0 OPERATION AND MAINTENANCE

8.1 AVAILABILITY

The cask shall be designed to have an availability of 300 days per year. Cask verification testing, maintenance, etc., shall account for no more than 65 days per year.

8.2 DECONTAMINATION

Cask surfaces, exterior and interior, and basket surfaces shall be designed for ease of decontamination. Surfaces shall be smooth without pockets and crevices to facilitate rinsing, drying operations, and to eliminate leaching.

8.3 TURN-AROUND TIME

Cask turn-around time shall be defined as the time required to:

- Off-load the cask from the transporter
- Remove fuel assemblies
- Decontaminate cask and basket
- Prepare cask for future shipments

The design goal for cask turn-around time shall be 24 hours.⁽¹⁾

8.4 HANDLING

8.4.1 Trunnions

The trunnions shall be bolted to the cask (Reference Appendix F) and designed to lift three times the weight of the cask plus contents per 10 CFR 71.45(a). They shall also meet the requirements in Sections 4.2.1 and 6.3.3 of this report.

8.4.2 Remote Handling (RH) Capabilities

The cask loading/unloading operation will be designed for remote operation (see Section 6.2.3.3). Remote operation will provide personnel exposure protection during large numbers of cask throughput and allow the facility to automate the operation.⁽¹⁾

8.5 MAINTENANCE

A maintenance plan shall be developed in the design phase of this project. This plan shall include in-service inspection, scheduled maintenance, and corrective maintenance requirements.

9.0 QUALITY ASSURANCE PLAN

BSFH design and procurement shall conform to the Quality Assurance requirements specified in NRC Regulatory Guide 7.10, Annex 1, and 10 CFR 71, Subpart H.

A Quality Assurance Plan will be prepared for the project at the earliest possible time during the design phase. The QA plan shall establish authority and responsibility for quality assurance activities within the project.

Quality assurance activities will be written for all phases of program development. These phases include: design, SARP, fabrication, testing, and operation.

The Quality Assurance Plan shall specify procedures for reporting and correcting significant deficiencies, defects, and nonconformances. Requirements for reporting significant deficiencies will be extended to suppliers in procurement documents.

Quality Assurance activities shall be applied with a graded approach. The level of QA effort shall be consistent with an activities' importance to safety and performance.

10.0 REFERENCES

1. HEDL-7376, TTC-0449, Rev. 1, "Conceptual Design Report for a Remotely Operated Cask Handling System."

A P P E N D I X A

REGULATIONS, CODES AND STANDARDS

APPENDIX A

REGULATIONS, CODES AND STANDARDS

A.1 DEPARTMENT OF ENERGY (DOE)

- A.1.1 DOE 1540.2, "Administrative Procedures Hazardous Material Packaging for Transport," draft dated November 15, 1984.
- A.1.2 DOE 5480.3, "Safety Requirements for the Packaging and Transportation of Hazardous Material, Hazardous Substances and Hazardous Wastes," dated July 9, 1985.

A.2 NUCLEAR REGULATORY COMMISSION (NRC)

- A.2.1 10 CFR 71, "Packaging and Transportation of Radioactive Material," August 31, 1983.
- A.2.2 Regulatory Guide 7.4, "Leakage Tests on Packages for Shipment of Radioactive Materials," June 1975.
- A.2.3 Regulatory Guide 7.6, "Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels," March 1978.
- A.2.4 Regulatory Guide 7.8, "Load Combinations for Structural Analysis of Shipping Casks," May 1977.
- A.2.5 Regulatory Guide 7.9, "Standard Format and Content of Part 71 Applications for Approval of Packaging of Type B, Large Quantity and Fissile Radioactive Material," Revision 1, January 1980.
- A.2.6 Regulatory Guide 7.10, "Establishing Quality Assurance Program for Packaging Used in the Transport of Radioactive Materials," January 1983.

A.3 AMERICAN NATIONAL STANDARDS INSTITUTE (ANSI)

- A.3.1 ANSI N14.2, "Tie-Downs for Transport of Fissile and Radioactive Material Containers Greater than One Ton Truck Transport," (draft of document under review).
- A.3.2 ANSI B30, "Lifting Devices."

- A.3.3 ANSI N14.5, "Leakage Tests on Packages for Shipment of Radioactive Material," 1977.
- A.3.4 ANSI N14.6, "Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds or More for Nuclear Materials," (draft of document under review).
- A.3.5 ANSI N14.8, "Fabricating, Testing and Inspection of Shielded Shipping Casks for Irradiated Reactor Fuel Elements," (new document to be written).
- A.3.6 ANSI N14.10.1, "Administrative Guide for Packaging and Transporting Radioactive Materials."
- A.4 AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME) BOILER AND PRESSURE VESSEL CODE
 - A.4.1 Section II, Material Specification
 - A.4.2 Section III, with Appendices - Nuclear Power Plant Components
 - A.4.3 Section IX, Welding and Brazing (Repair)
 - A.4.4 Section XI Rules for In-Service Inspection of Nuclear Power Plant Components
- A.5 OTHER
 - A.5.1 49 CFR 173, Department of Transportation, "Shippers General Requirements for Shipments and Packaging," dated May 1984.
 - A.5.2 ANSI/ASME NQA-1-1979, "Quality Assurance Program Requirements for Nuclear Power Plants."
 - A.5.3 International Atomic Energy Agency (IAEA) Safety Series No. 6, "Regulations for the Safe Transport of Radioactive Material," 1973 Revised Edition.
 - A.5.4 American Association of State Highway and Transportation Offices (AASHTO).
 - A.5.5 American Society of Testing Material (ASTM) VE-23.

A P P E N D I X B

PHYSICAL PARAMETERS FOR FFTF FUEL ASSEMBLIES

TABLE B.1
PHYSICAL PARAMETERS

	<u>FFTF</u>
<u>Fuel Pin</u>	
Active Fuel Length	36.0 in.
Clad Material	316 SST
Clad Thickness	0.38 mm
Pin Dimensions	
- Diameter	5.80 mm
- Length	2.38 m
<u>Driver Fuel Assembly</u>	
Shape (Diameter)	4.72 in. (hex)
Length	144.0 in.
Weight	500.0 lb
No. of Pins/Assembly	217
Fuel Pin Array Pitch	7.3 mm
Specific Power (MW/MTHM)	161
Heavy Metal Weight	33.0 kg
Initial Enrichment $\frac{P_u}{P_u + U}$ %	Range 22.4 to 30.9
End-of-Burnup Enrichment $\frac{P_u}{P_u + U}$ %	Range 21.4 to 29.1
Axial Burnup Profile	See Page B-3

FIGURE B.1. Axial Flux and Fission Rate Distributions, Rods Fully Withdrawn.

A P P E N D I X C

PHYSICAL PARAMETERS FOR HIGH LEVEL LIQUID WASTE (HLLW) CONTAINER

TABLE C.1
HIGH LEVEL LIQUID WASTE (HLLW) CONTAINER

Parameter	Size 12-in.
Canister ID (in.)	12.25
Waste Oxide to Vitrify (kg/campaign)	211
HLLW to Process (L/campaign)	1281
Allowable Heat Density in Glass (W/L)	41
Canister Length (ft)	10
Canister Fill Depth (ft)	8
Glass Volume (L/canister)	185.4
Total Heat Loading in Canister (kW)	7.6
Glass Density (g/ml)	2.8
Glass Produced (kg/canister)	519
Number of Canisters/Campaign	5.0
Waste Loading in Glass (wt%)	8.1
Glass Production Rate Required (kg/hr)*	3.6
Feed Concentration (g oxide/L)	325
Average Feed Rate Required (L/hr)*	11.2

*For 30-day campaign, matching reprocessing rate.

A P P E N D I X D

GAMMA AND NEUTRON SOURCE TERMS FOR FFTF SPENT FUEL

TABLE D.1

GAMMA SOURCE TERMS FOR FFTF FUEL ASSEMBLY
AFTER 150,000 MWd/MTHM BURNUP - 918 EFPD

Group No.	Mean Energy (MeV)	Source Terms (photons/s-assembly)			
		1 Year Decay	2 Years Decay	5 Years Decay	10 Years Decay
1	0.150	2.212 + 14	1.083 + 14	2.734 + 13	1.108 + 13
2	0.250	1.339 + 13	5.927 + 12	1.399 + 12	7.353 + 11
3	0.350	4.766 + 11	3.861 + 11	2.298 + 11	1.166 + 11
4	0.475	5.443 + 14	2.766 + 14	4.531 + 13	4.888 + 12
5	0.650	1.206 + 15	8.867 + 14	5.608 + 14	4.894 + 14
6	0.825	6.501 + 14	2.367 + 14	8.421 + 13	1.701 + 13
7	1.000	5.429 + 13	3.033 + 13	8.973 + 12	3.141 + 12
8	1.225	1.657 + 13	1.284 + 13	7.618 + 12	4.278 + 12
9	1.475	1.972 + 13	1.173 + 13	3.375 + 12	7.902 + 11
10	1.700	2.377 + 12	1.155 + 12	1.475 + 11	4.789 + 09
11	1.900	1.829 + 08	3.340 + 06	6.507 + 05	6.507 + 05
12	2.100	1.194 + 13	4.896 + 12	3.382 + 11	3.934 + 09
13	2.300	2.502 + 08	1.286 + 05	1.737 - 05	6.285 - 22
14	2.500	9.495 + 05	2.494 - 03	0.000	0.000
15	2.700	0.000	0.000	0.000	0.000
16	3.000	2.460 + 04	6.462 - 05	0.000	0.000
Dose Rate* (mR/hr)		1.09 x 10 ⁴	5.43 x 10 ³	1.80 x 10 ³	8.03 x 10 ²

*At a distance of 1 ft from a 7 in. thick steel wall cask.

TABLE D.2

INNER DRIVER FFTF FUEL

Group	Energy Range	α^*	Neutron Source (n/s/Subassembly)					
			0 Years	1 Year	2 Years	5 Years	7.5 Years	10 Years
1	10 MeV 3.7 MeV	0.106561	7.58 + 6	2.05 + 6	8.61 + 5	5.03 + 5	4.65 + 5	4.34 + 5
2	1.4 MeV	0.607230	4.32 + 7	1.67 + 7	4.90 + 6	2.87 + 6	2.65 + 6	2.47 + 6
3	0.50 MeV	0.206418	1.47 + 7	3.96 + 6	1.67 + 6	9.74 + 5	9.01 + 5	8.40 + 5
4	0.30 MeV	0.039251	2.79 + 6	7.54 + 5	3.17 + 5	1.85 + 5	1.71 + 5	1.60 + 5
5	0.11 MeV	0.030741	2.19 + 6	5.90 + 5	2.48 + 5	1.45 + 5	1.34 + 5	1.25 + 5
6	40.87 KeV	0.007524	5.35 + 5	1.45 + 5	6.08 + 4	3.55 + 4	3.28 + 4	3.06 + 4
7	15.03 KeV	0.001691	1.20 + 5	3.25 + 4	1.37 + 4	7.98 + 3	7.38 + 3	6.88 + 3
8	5.53 KeV	0.000402	2.86 + 4	7.69 + 3	3.25 + 3	1.90 + 3	1.75 + 3	1.64 + 3
9	2.03 KeV	0.000092	6.55 + 3	1.77 + 3	7.43 + 2	4.34 + 2	4.02 + 2	3.74 + 2
10	0.75 KeV	0.000022	1.57 + 3	4.22 + 2	1.78 + 2	1.04 + 2	9.60 + 1	8.95 + 1
11	0.27 KeV	0.000005	3.56 + 2	9.60 + 1	4.04 + 1	3.26 + 1	2.18 + 1	2.03 + 1
12	0.09 eV	0.000001	7.12 + 1	1.92 + 1	8.08	4.72	4.36	4.07

* α = Source spectrum based on 60% spontaneous fission and 40% (α , n) reactions (normalized to 1).

TABLE D.3
INNER DRIVER FFTF FUEL

Isotopic Activities for Various Time Periods After Fuel Discharge

Isotope	Curies/Subassembly					
	0 Years	1 Year	2 Years	5 Years	7.5 Years	10 Years
^{240}Pu	$1.815 + 2$	$1.815 + 2$	$1.815 + 2$	$1.814 + 2$	$1.814 + 2$	$1.813 + 2$
^{241}Am	91.10	90.96	90.81	90.38	90.01	89.65
^{239}Pu	$2.672 + 2$	$2.672 + 2$	$2.672 + 2$	$2.672 + 2$	$2.672 + 2$	$2.671 + 2$
^{242}Pu	$7.88 - 2$	$7.88 - 2$	$7.88 - 2$	$7.88 - 2$	$7.88 - 2$	$7.88 - 2$
^{242}Cm	$9.185 + 3$	$1.947 + 3$	$4.125 + 2$	3.925	$8.115 - 2$	$1.678 - 3$
^{244}Cm	30.18	29.05	27.96	24.94	22.67	20.61
^{246}Cm	$2.281 - 4$	$2.281 - 4$	$2.280 - 4$	$2.279 - 4$	$2.279 - 4$	$2.278 - 4$
	n/s/Subassembly					
	0 Years	1 Year	2 Years	5 Years	7.5 Years	10 Years
Neutron Source	$7.12 + 7$	$1.92 + 7$	$8.08 + 6$	$4.72 + 6$	$4.37 + 6$	$4.07 + 6$

Shelf life, 8.5 years.

Nine cycles (102 EFPD), intervening outages of 48 days.

TABLE D.4

FFTF FUEL ASSEMBLY, FISSION PRODUCT DECAY HEAT (KW)

Decay Time (Years)	50 MWD/MTHM			100,000 MWD/MTHM			150,000 MWD/MTHM			200,000 MWD/MTHM		
	3 Cycles			6 Cycles			9 Cycles			12 Cycles		
	B	Y	TOT	B	Y	TOT	B	Y	TOT	B	Y	TOT
0	147.0	145.0	292.0	149.0	146.0	295.0	150.0	148.0	298.0	151.0	149.0	300.0
0.5	0.998	0.491	1.49	1.38	0.595	1.98	1.56	0.690	2.25	1.66	0.783	2.45
0.75	0.774	0.245	1.02	1.09	0.335	1.43	1.24	0.419	1.66	1.33	0.504	1.84
1.0	0.624	0.145	0.769	0.891	0.225	1.12	1.02	0.303	1.32	1.09	0.381	1.47
1.5	0.424	0.080	0.504	0.612	0.147	0.759	0.707	0.213	0.920	0.762	0.281	1.04
2.0	0.295	0.059	0.354	0.430	0.115	0.545	0.500	0.174	0.674	0.544	0.233	0.777
3.0	0.148	0.041	0.189	0.222	0.083	0.305	0.265	0.129	0.394	0.295	0.176	0.471
4.0	0.080	0.031	0.111	0.125	0.065	0.190	0.155	0.102	0.257	0.178	0.140	0.318
5.0	0.047	0.026	0.073	0.078	0.054	0.132	0.101	0.085	0.186	0.122	0.117	0.238
10.0	0.016	0.017	0.033	0.032	0.034	0.066	0.046	0.053	0.099	0.061	0.070	0.131

One Cycle

5.5 MW - 100 days

0.0 MW - 40 days

A P P E N D I X E

VITRIFIED WASTE LOG GAMMA RADIATION SOURCE TERMS

TABLE E.1

VITRIFIED WASTE LOG GAMMA RADIATION SOURCE TERMS
(Based on Fuel Burnup of 150,000 MWd/MTHM)

Group No.	Mean Energy (MeV)	Source Terms (photons/s-metric tonne heavy metal)				
		1 Year	2 Years	5 Years	7.5 Years	10 Years
1	0.150	7.807 + 15	3.420 + 15	5.054 + 14	2.265 + 14	1.146 + 14
2	0.250	4.652 + 14	1.943 + 14	2.919 + 13	1.673 + 13	1.397 + 13
3	0.350	1.488 + 13	1.180 + 13	6.094 + 12	3.695 + 12	2.366 + 12
4	0.475	2.083 + 15	1.208 + 15	5.449 + 14	2.844 + 14	1.489 + 14
5	0.650	1.823 + 16	2.564 + 16	1.744 + 16	1.445 + 16	1.277 + 16
6	0.825	2.156 + 16	8.113 + 15	2.890 + 15	1.261 + 15	5.614 + 14
7	1.000	4.672 + 14	2.602 + 14	1.046 + 14	7.486 + 13	6.031 + 13
8	1.225	3.241 + 14	2.560 + 14	1.471 + 14	1.069 + 14	8.506 + 13
9	1.475	6.461 + 14	3.684 + 14	1.014 + 14	4.437 + 13	2.176 + 13
10	1.700	5.435 + 12	8.225 + 10	0.000	0.000	0.000
11	1.900	1.127 + 10	1.705 + 08	0.000	0.000	0.000
12	2.100	4.644 + 14	1.901 + 14	1.310 + 13	1.413 + 12	1.521 + 11
13	2.300	8.943 + 10	5.323 + 07	0.000	0.000	0.000
Decay						
Heat (kW)		32.0	12.0	3.8	3.1	2.8
Dose Rate*						
(mR/hr)		2.13×10^5	1.10×10^5	3.80×10^4	2.36×10^4	1.76×10^4

*This is the dose rate at 1 ft from the surface of a 7 in. steel shield surrounding the vitrified HLLW log.

TABLE E.2

CHARACTERISTICS OF BRET HLLW FEED TO VITRIFICATION SYSTEM
FROM REPROCESSING ON MTIHM of FFTF FUEL(a,b,c)

Constituent	Mass (g)	Activity (Ci) at		Thermal Power (W) at	
		180d(d)	1 year(d)	180d(d)	1 year(d)
Na	3.1 E 3				
Gd	2.6 E 4				
TBP	2.3 E 2				
Na DBP	5.1 E 2				
Cr	1.5 E 3	1.7 E 0	1.7 E -2	3.7 E -4	3.7 E -6
Fe	5.7 E 3	2.5 E 1	2.1 E 1	4.1 E -3	9.1 E -4
Ni	1.7 E 3	1.3 E 0	1.3 E 0	5.1 E -4	5.1 E -4
Se	6.5 E 1	6.2 E -1	6.2 E -1	1.6 E -4	1.6 E -4
Rb	3.7 E 2	1.7 E 1	1.8 E -2	7.9 E -2	8.1 E -5
Sr	8.3 E 2	2.4 E 5	8.0 E 4	6.8 E 2	1.2 E 2
Y	4.4 E 2	3.9 E 5	1.0 E 5	1.5 E 3	5.0 E 2
Zr	4.5 E 3	7.1 E 5	1.0 E 4	3.6 E 3	4.8 E 2
Nb	5.3 E 0	1.4 E 6	2.1 E 5	6.7 E 3	1.0 E 2
Mo	3.9 E 3	~0	~0	~0	~0
Tc	1.0 E 3	1.7 E 1	1.7 E -1	8.7 E -3	8.7 E -3
Ru	3.2 E 3	1.2 E 6	7.4 E 5	6.5 E 2	6.6 E 1
Rh	8.6 E 2	1.1 E 6	6.7 E 5	9.1 E 5	6.4 E 3
Pd	2.8 E 3	3.2 E -1	3.2 E -1	1.9 E -5	1.9 E -5
Ag	2.6 E 2	3.3 E 3	1.9 E 3	5.4 E 1	3.2 E 1
Cd	2.9 E 2	8.5 E 2	2.0 E 2	2.8 E 0	4.2 E -1
Sn	2.8 E 2	8.8 E 3	3.4 E 3	2.6 E 1	9.7 E 0
Sb	1.1 E 2	5.3 E 4	4.6 E 4	1.7 E 2	1.5 E 2
Te	9.9 E 2	6.5 E 4	2.4 E 4	7.3 E 1	2.2 E 1
I	6.4 E -1	9.5 E -4	8.2 E -5	3.0 E -6	3.8 E -8
Cs	6.9 E 3	2.4 E 5	2.3 E 5	8.2 E 2	7.2 E 2
Ba	2.3 E 3	1.7 E 5	1.7 E 5	6.9 E 2	6.8 E 2
La-Dy	1.6 E 4	4.0 E 6	2.6 E 6	1.4 E 4	8.6 E 3
U	3.5 E 3	3.9 E -2	3.8 E -2	1.1 E -4	1.1 E -4
Np	2.6 E 2	8.3 E 0	8.3 E 0	3.2 E -2	3.2 E -2
Pu	1.2 E 3	1.7 E 3	1.6 E 3	4.0 E 0	4.1 E 0
Am	2.1 E 3	7.9 E 3	8.2 E 3	2.3 E 2	2.3 E 2
Cm	2.6 E 1	1.7 E 5	7.6 E 4	6.1 E 3	2.8 E 3
Total metals	9.1 E 4	9.7 E 6	5.0 E 6	4.4 E 4	2.2 E 4

HNO₃: 2 M

Nitrite associated with metals: 2.0 E 3 moles

Volume: 667 L (176 gal) Density: ~1.3 g/ml

(a)Based on FFTF fuel irradiated to 60,000 MWd/MTIHM at 150 MW/MTIHM.

(b)HLLW is a blend of HAW, solvent wash waste, and LLW concentrator bottoms.

(c)The following percentages of spent fuel constituents are present in the HLLW: 1% of the clad and hardware fed to dissolver, 72% Mo, 71% Tc, 57% Ru, 52% Rh, 80% Pd, 62% Ag, 93% Te, 0.1% halogens, 0% noble gases, 99.95% other f.p.'s, 0.5% U and Pu, 99.75% Np, 99.95% Am and Cm.

(d)Time of fuel reprocessing.

TABLE E.3

ISOTOPIC ACTIVITIES IN HLLW OPERATED TO 150,000 MWd/MTHM

Isotope	Half-Life T1/2	Activity (curies)				
		1 Year	2 Years	5 Years	7.5 Years	10 Years
⁹¹ Y	58.8 d	3.80 + 4	513.0	1.26 - 3	--	--
⁹⁵ Zr	65.5 d	1.12 + 5	2350.0	2.16 - 2	--	--
⁹⁵ Nb*	90 hr	1460.0	47.0	4.40 - 4	--	--
⁹⁵ Nb	35 d	2.44 + 5	2780.0	2.48 - 2	--	--
¹⁰³ Ru	39.5 d	1.47 + 4	24.2	--	--	--
¹¹⁰ Ag*	255 d	7030.0	2600.0	133.0	11.1	0.925
¹¹⁰ Ag	25 s	7030.0	2600.0	133.0	11.1	0.925
¹¹³ Cd*	13.6 yr	885.0	841.0	722.0	635.0	559.0
¹¹⁹ Sn*	250 d	448.0	163.0	7.8	0.62	0.05
¹²³ Sn	125 d	4430.0	585.0	1.34	8.50 - 3	--
¹²³ Te*	117 d	31.7	3.64	5.50 - 3	--	--
¹²⁴ Sb	60.4 d	302.0	4.57	--	--	--
¹²⁵ Sb	2.71 yr	9.74 + 4	7.54 + 4	3.50 + 4	1.85 + 4	9750.0
¹²⁵ Te*	58 d	1.51 + 4	4810.0	2180.0	1150.0	606.0
¹²⁷ Te*	109 d	9980.0	978.0	0.921	2.80 - 3	--
¹²⁷ Te	9.4 hr	9780.0	978.0	0.921	2.80 - 3	--
¹²⁹ Te*	34.1 d	2520.0	1.5	--	--	--
¹²⁹ Te	68.7 min	2520.0	1.5	--	--	--
¹³⁴ Cs	2.1 yr	3.27 + 5	2.33 + 5	8.46 + 4	3.63 + 4	1.56 + 4
¹³⁷ Cs	30 yr	4.33 + 5	4.23 + 5	3.95 + 5	3.73 + 5	3.52 + 5
¹³⁷ Ba*	2.6 min	4.33 + 5	4.23 + 5	3.95 + 5	3.73 + 5	3.52 + 5
¹⁴¹ Ce	32.5 d	2320.0	0.96	--	--	--
¹⁴⁴ Ce	284 d	1.72 + 6	7.05 + 5	4.86 + 4	5240.0	564.0
¹⁴⁴ Pr	17.3 min	1.72 + 6	7.05 + 5	4.86 + 4	5240.0	564.0
¹⁴⁸ Pm*	41.8 d	555.0	1.3	--	--	--
¹⁴⁸ Pm	5.4 d	38.3	--	--	--	--
¹⁵² Eu	12.7 yr	114.0	108.0	91.6	80.0	69.8
¹⁵⁴ Eu	16 yr	7720.0	7390.0	6490.0	5830.0	5230.0
¹⁵⁵ Eu	1.81 yr	1.83 + 4	1.25 + 4	3960.0	1520.0	583.0
²⁴¹ Am	458 yr	5930.0	5920.0	5890.0	5870.0	5850.0
²⁴² Cm	163 d	3.56 + 5	7.50 + 4	700.0	14.2	0.29

Represents parent - daughter relationships (e.g., ¹³⁷Cs, ¹³⁷Ba).

A P P E N D I X F

TRUNNION SKETCH

TYPICAL UPPER TRUNNION

TYPICAL LOWER TRUNNION

HEDL-7554
September 17, 1985

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