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		Design Agent									
1	/	Cog.Eng.: T Romano	Tom Romano	6-9-97	H1-15						
1	/	Cog. Mgr.: JG Field		6/9/97	H1-15						
1	/	QA: CR Hoover		6/9/97	H1-15						
1	/	Safety: DW McNally	DW McNally	6/9/97	H1-15						
		Env.									
1	/	JF Henderson	JHenderson	6/9/97	L1-03						
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Safety Evaluation for Packaging (Onsite) SERF Cask

T. Romano
Waste Management Federal Services, Inc., Northwest Operations
Richland, Washington 99352
U.S. Department of Energy Contract DE-AC06-96RL13200

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Abstract: This safety evaluation for packaging (SEP) documents the ability of the Special Environmental Radiometallurgy Facility (SERF) Cask to meet the requirements of WHC-CM-2-14, *Hazardous Material Packaging and Shipping*, for transfer of Type B quantities (up to highway route controlled quantities) of radioactive material within the 300 Area of the Hanford Site. This document shall be used to ensure that loading, tiedown, transport, and unloading of the SERF Cask are performed in accordance with WHC-CM-2-14. This SEP is valid until October 1, 1999. After this date, an update or upgrade to this document is required.

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LIST OF TERMS

ANSI	American National Standards Institute
ARF	airborne release fraction
ASME	American Society of Mechanical Engineers
atm	atmosphere
B&PV	Boiler and Pressure Vessel (Code)
Bq	becquerel
CFR	*Code of Federal Regulations*
Ci	curie
cm	centimeter
cm^3	cubic centimeter
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
EDE	effective dose equivalent
FFTF	Fast Flux Test Facility
ft	foot
g	gram
Gy	gray
Gy/h	gray per hour
Hz	hertz
IAEA	International Atomic Energy Agency
in.	inch
kg	kilogram
km	kilometer
km/h	kilometers per hour
kPa	kilopascal
lb	pound
m	meter
MAP	mixed activation products
MeV	megaelectronvolt
MFP	mixed fission products
mi	mile
MPa	megapascal
mph	miles per hour
mrem/h	millirem per hour
mSv/h	millisievert per hour
neutrons/s	neutrons per second
PCF	probability of release from crush failure
PIF	probability of release from impact failure
PFF	probability of release from fire failure
PNC	Power Reactor and Nuclear Fuel Development Corporation
PPF	probability of release from puncture failure
PRTR	Plutonium Recycle Test Reactor (Cask)
psi	pounds per square inch
QA	quality assurance
QL	quality level
rem/h	rem per hour
RF	respirable fraction
SEP	safety evaluation for packaging
SERF	Special Environmental Radiometallurgy Facility (Cask)
Sv	sievert

LIST OF TERMS (cont)

Sv/h	sievert per hour
TBq	terabecquerel
THI	Transportation Hazard Indicator
W	watt

SAFETY EVALUATION FOR PACKAGING (ONSITE) SERF CASK

PART A: DESCRIPTION AND OPERATIONS

1.0 INTRODUCTION

1.1 GENERAL INFORMATION

Three safety evaluations for packaging (SEP) have been prepared for the family of 327 Building casks used within the 300 Area. The family of casks consists of the Special Environmental Radiometallurgy Facility (SERF) Cask, Radioactive Waste Disposal Cask, and the Plutonium Recycle Test Reactor (PRTR) Graphite Cask.

This SEP evaluates and documents the ability of the SERF cask to meet the requirements of WHC-CM-2-14, *Hazardous Material Packaging and Shipping*, for transfer of Type B quantities (up to highway route controlled quantities) of radioactive material within the 300 Area of the Hanford Site.

This document shall be used to ensure that loading, tiedown, transport, and unloading of the package are performed in accordance with WHC-CM-2-14.

The SERF Cask is used to transport Fast Flux Test Facility (FFTF) and Power Reactor and Nuclear Fuel Development Corporation (PNC) fuel, mixed plutonium-uranium oxide fuel pins, uranium oxide fuel pins, depleted uranium, N Reactor fuel elements and scrap, enriched uranium, structural material from reactors, and cesium chloride capsules. Examination of irradiated fuel assemblies is the main mission of the 327 Facility. Shipments of material among the 323, 324, 325, 326, 327, and 3270 Buildings range from multiple complete fuel pins to metalographic samples. Specifically, the SERF Cask is used for full-length fuel pins and high-dose items because it has the largest capacity compared to other casks used within the 300 Area.

1.2 SYSTEM DESCRIPTION

The SERF Cask consists of two, lead-filled, concentric, stainless steel cylinders; an integral, lead-filled, rotating cylindrical valve located in one end; a plug; and a push rod assembly located at the other end. Two lifting trunnions are located at the center of gravity of the cask. The cask is equipped with a yoke and is horizontally oriented. It is 66.0 cm (26.0 in.) in diameter and 339.2 cm (133.6 in.) in length. The internal cavity is 257.9 cm (101.5 in.) long with a diameter of 19.4 cm (7.63 in.). There is 20.3 cm (8.00 in.) of lead between the internal and external shells. When in transit, plates are bolted over the valve and plug to seal the cavity. The empty weight of the cask is 11,793.6 kg (26,000 lb), and the gross weight is 12,020.4 kg (26,500 lb).

1.3 REVIEW AND UPDATE CYCLES

This SEP is valid until October 1, 1999. An update or upgrade to this document is required beyond that date.

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2.0 PACKAGING SYSTEM

2.1 CONFIGURATION AND DIMENSIONS

The SERF Cask is a cylindrical shipping cask with a stainless steel shell. The cask is equipped with a yoke and is horizontally oriented. The outer steel shell of the SERF cask is 1.60 cm (0.625 in.) thick and has an outer diameter of 66.0 cm (26.0 in.). The inner steel shell has an inner diameter of 19.4 cm (7.63 in.) and is 1.27 cm (0.50 in.) thick. The lead shielding between the two shells is 20.3 cm (8.00 in.).

The integral closure valve is constructed as a lead-filled cylindrical steel shell. A hole with an inside diameter of 19.4 cm (7.625 in.) that is transverse to the axis of rotation allows loading and unloading of the cask when the valve is open and provides lead shielding when the valve is closed. The integral closure valve is actuated by a hand crank and is opened or closed by rotating 90°. During transit, the integral closure valve is held closed by a stop plate.

The lead between the inner and outer shell of the cask provides the bulk of the shielding. Some additional shielding is provided by the steel shells.

The bottom plug in the cask is a lead-filled steel shell. It is bolted in the bottom end of the cask. It contains a small port to allow entry of a push rod that is required to push the bin container through the valve into a hot cell. The port is filled with an iron plug when not in use and with an iron push rod when in use.

Seal plates are provided for each end of the cask. These plates are bolted over the valve and plug to provide a seal during transit. The seal plate over the valve (stop plate seal assembly) has a projection that fits into a recess in the valve to provide a positive lock.

The lifting bail is attached to the SERF Cask to allow for lifting the cask in a horizontal position.

The crank assembly provides for remote operation of the cask valve operating mechanism. The crank assembly penetrates through the SERF Cask support.

The contents of the cask are placed in inner containers prior to loading the cask. Fuel pins are shipped in stainless steel tubes.

2.2 MATERIALS OF CONSTRUCTION

All structural components are constructed of either 304 or 304L stainless steel. Neoprene[1] gaskets are used to seal the cask. Lead is used for shielding.

2.3 WEIGHTS

The empty weight of the cask is 11,793.6 kg (26,000 lb), with a maximum gross weight of 12,020.4 kg (26,500 lb).

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[1] Neoprene is a trademark of E. I. du Pont de Nemours and Company.

2.4 CONTAINMENT BOUNDARY

The containment boundary consists of the inner container or the double-wrapped plastic. The SERF Cask retains the contents, but is not considered a containment boundary. No credit is taken for containment provided by the SERF Cask.

2.5 CAVITY SIZE

The internal cavity is 257.9 cm (101.53 in.) long with a diameter of 19.4 cm (7.625 in.).

2.6 SHIELDING

Shielding is provided by the 20.3 cm (8 in.) of lead between the internal and external stainless steel shells and an integral lead-filled rotating cylindrical valve located at one end.

2.7 LIFTING DEVICES

Two lifting trunnions are located at the center of gravity of the cask.

2.8 TIEDOWN DEVICES

The SERF Cask is not equipped with any tiedown attachment devices.

3.0 PACKAGE CONTENTS

3.1 GENERAL DESCRIPTION

Materials to be transported in the SERF Cask include irradiated FFTF and PNC fuel, mixed plutonium-uranium oxide fuel pins, uranium oxide fuel pins, depleted uranium, N Reactor fuel elements and scrap, enriched uranium, structural material from reactors, and cesium chloride capsules.

3.2 CONTENTS RESTRICTIONS

The materials specified in this section are the only materials authorized for shipment in the SERF Cask within the 300 Area of the Hanford Site. Contact dose rates on the cask shall be less than 200 mrem/h.

3.2.1 Radioactive Materials

Table A3-1 gives the radioactive contents limits for the SERF Cask.

Table A3-1. SERF Cask Contents Limits.

Material		Activity limit	
		TBq	Ci
Mixed material inventory	Fissile materials and α emitters*	-----*	-----*
	Mixed fission products	37	1000
	Mixed activation products	555	15000
Cesium capsules (2)	Cesium chloride	2960	80000
Fuel	Fuel	-----*	-----*

*Fissile/fissionable materials limited by criticality safety as shown in Part B, Section 2.1.1.

Absorbed and unabsorbed liquids are not authorized for shipment in the SERF Cask.

Only dry solid materials, as described in Part A, Section 3.1, shall be shipped in the cask. Organic materials are not authorized except plastic bags/wrapping. All materials shall be enclosed in an inner container as shown in Table A3-2.

Table A3-2. Inner Container Description.

Inner Container	Contents	Examples
Pin tubes - tubing and fittings must have a working rating of 3000 psi with an outer diameter of ½ in. or ¾ in. The tubes are welded on one end. A Swagelok* fitting is used as closure.	Fuel pins and dispersible material	FFTF, PNC, and N Reactor fuel; mixed plutonium-uranium oxide fuel pins, uranium oxide fuel pins, dispersible scrap, pieces of fuel, and fines.
1 gallon container with friction fit lid	Nondispersible solid contents**	Solid scrap, structural material.
Specification 2R per 49 CFR 178.360	Dispersible solid contents	Fuel pieces, scrap, and fines of activated fuel and materials.
Large solid items	Large solid items to big for a paint can will have fixed surface contamination and put directly into the cask.[1]	Solid structural material and activated metals.
Cesium capsules	cesium chloride	----

FFTF = Fast Flux Test Facility.
PNC = Power Reactor and Nuclear Fuel Development Corporation.

*Swagelok is a trademark of the Crawford Fitting Company.

**Surface contamination limits not to exceed 100 times the Table A4-1 limits. Verification by survey or the use of a fixative such as paint is required.

[1] 49 CFR 178, 1996, "Specifications for Packagings," *Code of Federal Regulations*, as amended.

Fissile limitations are given in Table A3-3 for shipments of fuel pins, N Reactor fuel assemblies, and scrap under dry conditions. The information for Table A3-3 was extracted from *Limits for Mixed Oxide Fuel Pins, N Reactor Fuel Elements and Scrap in the SERF Cask and the Waste Cask* (Larson 1995). Table A3-4 gives the limits for fuel pins of different compositions (^{239}Pu, PuC, PuN, PuO_2, U metal, UC, UN, and UO_2). No other contents shall be included with fuel pin and fuel element shipments.

Scrap materials shall not exceed the limits of Table A3-3 and may include pieces of fuel pins and fuel elements, activated structural materials, and laboratory samples. Organic materials are not authorized except plastic bags/wrapping.

Table A3-3. Criticality Limits in the SERF Cask for Dry (H:≤X5) Conditions.

Limits.	Maximum fuel diameter (in.)	SERF Cask	
Mixed oxide fuel pin		Maximum no. fuel pins (length of 13.5 in.)	Maximum no. fuel pins (length of 37 in.)
<25 wt% $Pu(>10)O_2$-$U(<94)O_2$	0.220	52	36
<25 wt% $Pu(>10)O_2$-$U(<36)O_2$	0.230	67	44
<16 wt% $Pu(>10)O_2$-$U(<41)O_2$	0.261	58	38
<31 wt% $Pu(>10)O_2$-$U(<0.72)O_2$	0.220	90	54
<25 wt% $Pu(>10)O_2$-$U(<68)O_2$	0.250	50	36
<5 wt% $Pu(>8)O_2$-$U(<9)O_2$	0.500	46	26
N Reactor fuel element	Maximum fuel diameter (in.)	Maximum fuel element length of 26.1	
Mark IA and IV inner elements	1.170	195	
Mark IA and IV outer elements	2.350	48	
Mark IA and IV fuel assemblies	2.350	48	
Scrap	----	250 g ^{235}U only or 150 g total fissionable*	

Pu = plutonium.
SERF = Special Environmental Radiometallurgy Facility (Cask).
U = uranium.

*No accountable amounts of ^{242m}Am, ^{243}Cm, ^{245}Cm, ^{249}Cf, or ^{251}Cf are permitted.

Source: Larson, S. L., 1995, *Limits for Mixed Oxide Fuel Pins, N Reactor Fuel Elements and Scrap in the SERF Cask and the Waste Cask* (NCS Basis Memo 95-3, Rev. 1, to M. Dec, August 31), Battelle Pacific Northwest Laboratory, Richland, Washington.

Table A3-4. Fuel Pin Limits for the SERF Cask.

Fuel type	Maximum enrichment (preirradiation composition)	Maximum fuel pin outer diameter (in.)	Maximum fuel length (in.)	Maximum fuel pin limit
Mixed U and Pu compounds (excluding ^{233}U)	100 wt% ^{235}U or ^{239}Pu	0.8	37	15
Mixed U and Pu compounds (excluding ^{233}U)	100 wt% ^{235}U or ^{239}Pu	1.3	37	8

Source: Hawkes, E. C., 1995, *Limits for Fuel Pins in the Waste Cask, the 327 Building A-Cell, and the 324 Building Shielded Materials Facility Hot Cells* (NCS Basis Memo 95-5 to M. Dec, September 6), Battelle Pacific Northwest Laboratory, Richland, Washington.

3.2.2 Nonradioactive Hazardous Materials

The authorized contents shall consist of no hazardous materials other than the radioactive materials described in Part A, Section 3.2.1.

4.0 TRANSPORT SYSTEM

4.1 TRANSPORTER

The transporter consists of a low boy or flatbed trailer and tractor. The trailer shall be rated for the weight of the loaded cask and sized such that the cask does not protrude beyond the edges of the trailer. One cask will be transported per trailer.

4.2 TIEDOWN SYSTEM

The SERF Cask shall be attached to the flatbed trailer with a system consisting of tiedown devices and/or blocking and bracing devices. The tiedown system shall meet the requirements of 49 CFR 393, Subpart I.

The SERF Cask shall be centered and placed horizontally on the bed of the trailer for shipment. The long axis of the cask is centered along the long axis of the trailer. The package is to be secured to U.S. Department of Transportation regulations (49 CFR 393.100). The cask is secured to the trailer by chains, cables, or straps, which are placed across and inboard of each support-welded plate on the cask and affixed to the trailer. Lifting holes on the cask support plates welded to the cask are not to be used for securement. All tiedown and trailer attachment points must have an aggregate working capacity of 6,350 kg (14,000 lb). The end of the cask facing the front end of the trailer must be secured against sliding with devices that have an aggregate working capacity of 8,165 kg (18,000 lb).

Alternative configurations that have been shown to meet 49 CFR 393, Subpart I, are acceptable.

4.3 SPECIAL TRANSFER REQUIREMENTS

4.3.1 Routing and Access Control

The cask shall be transported over a predetermined route and in accordance with WHC-CM-2-14.

4.3.2 Radiological Limitations

The dose rate must be less than 200 mrem/h at the surface of the cask, less than 10 mrem/h at 2 m from the trailer, and less than 2 mrem/h at any space normally occupied by personnel. Transfer of the SERF Cask above these limits is not authorized.

External contamination limits for the exterior of the SERF Cask are as shown in Table A4-1.

Table A4-1. External Container Contamination Limits.

Contaminant	Maximum permissible limits		
	Bq/cm^2	μCi/cm^2	dpm/cm^2
Beta and gamma emitters and low toxicity alpha emitters	0.4	10^{-5}	22
All other alpha-emitting radionuclides	0.04	10^{-6}	2.2

Source: 49 CFR 173.443, 1997, "Shippers--General Requirements for Shipments and Packagings, *Code of Federal Regulations*, as amended.

4.3.3 Speed Limitations

The SERF Cask shall be transported at a maximum speed of 8.1 km/h (5.0 mph).

4.3.4 Environmental Conditions

There shall be no shipments at temperatures below 0 °C (32 °F) or during periods of dense fog or adverse road conditions (such as snow or ice).

4.3.5 Frequency of Use and Mileage Limitations

A risk analysis was performed on the 327 Building casks to determine mileage limitations. The results of the evaluation determined that shipments of the 327 Building casks shall not exceed a total of 16.0 km (10.0 mi) per year for all casks combined. This mileage limitation does not apply to empty packaging shipments.

4.3.6 Emergency Response

The shipping and receiving facilities, Radiation Protection, Packaging Engineering, and Transportation Logistics shall be notified of all accidents involving radioactive material shipment that result in vehicle damage, container damage, personnel injury, or contamination spread.

5.0 ACCEPTANCE OF PACKAGING FOR USE

5.1 NEW PACKAGING

The SERF Cask was originally approved for use per the *Hazardous Materials Packaging and Shipping Manual*, MG 137, Rev. 1 (HEDL 1981). Since that time, the SERF Cask has been in use. No new casks will be manufactured; therefore, new packaging acceptance requirements will not be addressed. Requirements for reuse are given in the following section.

5.2 PACKAGING FOR REUSE

Prior to loading, the cask shall be visually inspected for physical damage and corrosion. Closing mechanisms and valves shall be checked for proper operation and closure. Visual inspections shall be documented in the facility operating procedures.

If required, the cask shall be decontaminated prior to reuse to meet the external contamination limits per Table A4-1.

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6.0 OPERATING REQUIREMENTS

6.1 GENERAL REQUIREMENTS

The following are requirements for the use of the SERF Cask. Prior to loading and shipment of the cask, specific operating procedures with appropriate Quality Assurance/Quality Control hold points shall be written by the user and approved per WHC-CM-2-14. The procedures shall implement the requirements of this section and the additional requirements found in Part A of this SEP.

For loading and unloading operations, the following general requirements shall be performed.

1. Visually inspect the SERF Cask for cracks or damage.
2. Visually inspect the lifting attachments for cracks and damage.
3. Verify that radiological contamination limits are within the allowable limits shown in Table A4-1 of this SEP.
4. Verify radiological dose rates are acceptable prior to shipment of the cask in accordance with Part A, Section 4.3.2, of this SEP.

6.2 LOADING OF CONTENTS INTO THE CASK

6.2.1 Inner Container Loading

1. Prior to loading contents, visually inspect the inner container for damage or corrosion. Ensure containers meet the requirements of Part A, Section 3.2.1.
2. Verify the closure mechanism is in good condition and operates properly.
3. For one-gallon paint cans, ensure the lid is properly deployed after contents are loaded into the can.
4. For fuel shipments, ensure that Swagelok[2] fittings are properly installed after loading fuel pins into the stainless steel tubes.

6.2.2 Preparing the Cask for Loading

1. Verify that the contents to be loaded into the cask are as authorized in Part A, Section 3.0, of this SEP and that criticality limits have not been exceeded.
2. Verify that the cask is positioned properly at the cell loading port and is ready to receive the contents. Open the cell loading port.

---

[2] Swagelok is a trademark of the Crawford Fitting Company.

6.2.3 Loading Contents into the Cask

1. Attach the push/pull rod to the rear end of the cask scoop fixture, commonly called the cask boat.

2. Release the integral closure valve locking mechanism and rotate the valve 90°.

3. Release the locking pin holding the boat and push the boat into the cell to the desired position.

4. Load the materials onto the boat and pull the boat back into the cask. Dunnage will be added as necessary to fill void space in the package. Dunnage may consist of empty cans or similar items. Secure the boat locking pin.

5. Rotate the integral closure valve 90° and secure the locking mechanism.

6. Close the cell loading port and move the cask away from the cell.

7. Perform radiological dose and contamination surveys to verify levels have not exceeded the limits authorized in Part A, Section 4.0, of this SEP. Decontaminate if contamination limits are exceeded. Do not ship if the radiological dose exceeds 200 mrem/h on the surface of the cask.

8. Following radiological survey, wrap the ends of the cask with 10-mil plastic and tape to the cask body using duct tape.

NOTE: The cesium chloride capsules have a stainless steel overpack, which meets the requirement for an inner container.

6.3 PREPARATION OF THE CASK FOR SHIPMENT

1. Verify the shipping papers have been prepared properly and the cask is properly marked and labeled per the requirements of WHC-CM-2-14.

2. Position the transport vehicle where it will be accessible to the overhead crane.

3. Verify lifting equipment is in accordance with the *Hanford Site Hoisting and Rigging Manual*, DOE-RL-92-36 (RL 1993).

4. Attach lifting equipment to the cask lifting attachment and the crane hook.

5. Lift the cask from the floor to allow a radiological survey of the cask to be performed.

6. Move the cask over the vehicle and slowly lower it into position on the transport vehicle.

7. Unhook the lifting equipment from the cask and install tiedown attachments per the requirements of Part A, Section 4.2.

8. Prior to transport, verify that the shipping documentation has been completed per WHC-CM-2-14 and signed by a trained Hazardous Material Shipper.

6.4 UNLOADING THE CASK

1. Position the transport vehicle where it will be accessible to the overhead crane.

2. Perform radiological contamination and dose surveys of the cask to verify that the limits in Part A, Section 4.0, have not been exceeded.

3. Remove the tiedown equipment from the cask and transport vehicle.

4. Attach the lifting equipment to the cask lifting attachment and the crane hook.

5. Lift the cask off of the transport vehicle and move to a designated location. Remove the plastic from the ends of the cask.

6. Perform a radiological contamination survey of the cask ends to verify the contamination limits have not been exceeded.

7. Position the cask at the cell loading port and remove the rigging equipment from the cask.

8. Open the cell loading port.

9. Attach the push/pull rod to the rear end of the cask scoop fixture, commonly called the cask boat.

10. Release the locking mechanism and rotate the integral closure valve 90°.

11. Release the locking pin holding the boat and push the boat into the cell to the desired position.

12. Unload the materials from the boat and pull the boat back into the cask. Secure the locking pin.

13. Rotate the integral closure valve 90° and secure the locking mechanism.

14. Close the cell loading port and move the cask away from the cell.

15. Perform radiological dose and contamination surveys to verify levels have not exceeded the limits authorized in Part A, Section 4.0, of this SEP. Decontaminate if contamination limits are exceeded.

6.5 EMPTY PACKAGING

To be transported as an empty radioactive container, the SERF Cask must be prepared for transport in accordance with 49 CFR 173.428.

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7.0 QUALITY ASSURANCE REQUIREMENTS

7.1 INTRODUCTION

This section describes the quality assurance (QA) requirements for operation of the SERF Cask. The packaging was fabricated in the 1960s following a quality program in effect at that time. The SERF Cask is used to perform onsite intra-area shipments at the Hanford Site. The format and requirements for the use of the SERF Cask on the Hanford Site are in accordance with WHC-CM-4-2, *Quality Assurance Manual*, and WHC-CM-2-14.

7.2 GENERAL REQUIREMENTS

These requirements apply to activities, including loading, unloading, and transportation operations, that could affect the quality of the packaging and associated hardware. The overall packaging is classified per WHC-CM-2-14 as a Transportation Hazard Indicator (THI) 1.

THI 1 packaging systems, defined in WHC-CM-2-14, represent the highest level of hazard for the contents. A packaging system assigned this level has the potential of causing a dose consequence to an individual greater than 25 rem beyond the Hanford Site boundary.

Each THI invokes a quality level (QL) designator (defined in WHC-CM-2-14) consisting of two parts: an alpha designator and a numerical designator. The alpha designator assigns the fabrication, testing, use, maintenance standards, and quality requirements for each component of the packaging system. The numeric designator following the letter is the THI number of the packaging system. Because the SERF Cask ships a Type B quantity of material, the package as a whole is assigned a QL designator of A-1.

Documentation and review requirements are based upon the QL of the package. Changes or discoveries of noncompliance for all QL A-1 components and activities shall be reviewed by the unreviewed safety question screening process to ensure the quality and safety of the change or discovery. Changes to the SEP safety bases (contents, shielding, structural, containment, criticality) will require unreviewed safety question screening regardless of QL.

7.3 ORGANIZATION

The organizational structure and the assignment of responsibility shall be such that quality is achieved and maintained by those who have been assigned responsibility for performing the work. Quality achievement is to be verified by persons or organizations not directly responsible for performing the work.

Packaging Engineering of Rust Federal Services Inc., Northwest Operations, and the onsite user are responsible for the quality of the work performed by their respective organizations and for performing the following activities:

- Follow the current requirements of this SEP, WHC-CM-4-2, and WHC-CM-2-14
- Provide instructions for implementing QA requirements.

The cognizant manager, Quality Assurance, is responsible for establishing and administering the Hanford QA program as stated in WHC-CM-4-2.

7.4 QA PLAN AND ACTIVITIES

7.4.1 Design Control

Design control is not applicable. This cask was fabricated over 30 years ago, and no design changes will be made. B&W Hanford Company is the design authority for the package.

7.4.2 Procurement and Fabrication Control

Procurement and fabrication control is not applicable. This package is over 30 years old, and no casks will be procured in the future.

7.4.3 Control of Operations/Processes

Loading/unloading procedures written by the user will be used to ensure acceptable operation of the packaging. Those loading/unloading procedures shall be consistent with this SEP. The loading/unloading procedures identify actions required by personnel to safely and properly load and unload the packaging in accordance with this SEP.

Quality Control inspection checklists are established to ensure that final inspection verifies compliance with the following items.

- The SERF Cask is properly assembled.
- All acceptance criteria (Part A, Section 5.0) are met for use of the package.
- Operational (Part A, Section 6.0) and maintenance procedures (Part A, Section 8.0) are properly completed.

7.4.4 Control of Inspection

Control of inspection and testing will be accomplished by facility procedures incorporating the requirements of Part A, Section 7.4.3.

7.4.5 Test Control

Test control is not applicable. No testing is required on this package.

7.4.6 Control of Measuring and Test Equipment

Any measuring equipment that is used shall meet the accuracy and calibration requirements as required by WHC-CM-4-2; i.e., radiation survey equipment.

7.4.7 Control of Nonconforming Items

Identification, documentation, evaluation, and disposition of nonconforming items and activities shall be accomplished per WHC-CM-4-2, regardless of the assigned QL.

7.4.8 Corrective Action

Nonconformance, or conditions adverse to quality, are evaluated as described in Part A, Section 7.4.8, and the need for corrective action is determined in accordance with WHC-CM-4-2.

7.4.9 QA Records and Document Control

Records that furnish documentary evidence of quality shall be specified, prepared, and maintained per WHC-CM-4-2. This includes all procedures, inspection reports, the SEP, and any nonconformance reports that are developed while this cask is used.

7.4.10 Audits

Internal and external independent assessments are performed in accordance with WHC-CM-4-2.

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8.0 MAINTENANCE

8.1 GENERAL REQUIREMENTS

Maintenance procedures shall be written by the user facility. Maintenance of the cask will be in accordance with the following requirements in Part A, Section 8.2.

8.2 INSPECTION AND VERIFICATION SCHEDULES

The SERF Cask is a reusable overpack and shall be inspected every two years. Inspections shall be performed using the *Radioactive Material Shipping Container User Biennial Inspection Checklist.* Weld inspections of the lifting apparatus shall be performed every five years and shall be performed using the *Radioactive Material Shipping Container NDT of Lifting Apparatus (5 Year)* (see Part A, Section 10.2, for examples of the sheets).

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9.0 REFERENCES

49 CFR 173, 1995, "Shippers--General Requirements for Shipments and Packagings," *Code of Federal Regulations*, as amended.

49 CFR 393, 1997, "Parts and Accessories Necessary for Safe Operation," *Code of Federal Regulations*, as amended.

HEDL, 1981, *Hazardous Materials Packaging and Shipping Manual*, MG 137, Rev. 1, Hanford Engineering Development Laboratory, Richland, Washington.

Larson, S. L., 1995, *Limits for Mixed Oxide Fuel Pins, N Reactor Fuel Elements and Scrap in the SERF Cask and the Waste Cask* (NCS Basis Memo 95-3, Rev. 1, to M. Dec, August 31), Battelle Pacific Northwest Laboratory, Richland, Washington.

PNL, 1993, *300 Area Radioactive Material Transport Plan*, PNL-RMTP-300-1, Pacific Northwest Laboratory, Richland, Washington.

RL, 1993, *Hanford Site Hoisting and Rigging Manual*, DOE/RL-92-36, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

WHC-CM-2-14, *Hazardous Material Packaging and Shipping*, Westinghouse Hanford Company, Richland, Washington.

WHC-CM-4-2, *Quality Assurance Manual*, Westinghouse Hanford Company, Richland, Washington.

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10.0 APPENDICES

10.1 DRAWINGS

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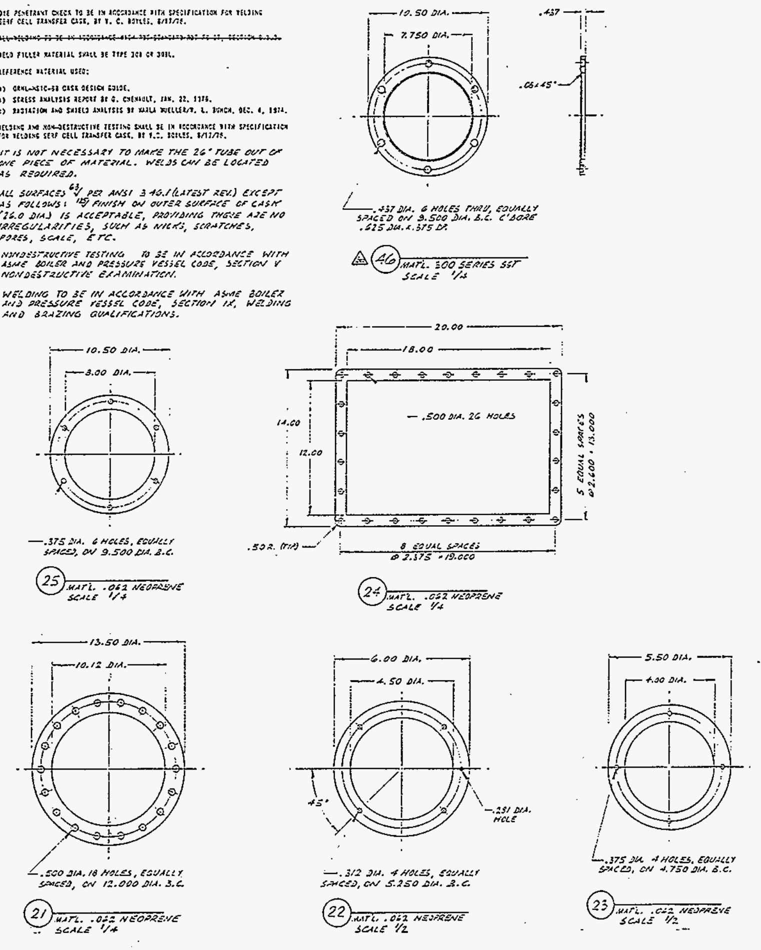
GENERAL NOTES:
(UNLESS OTHERWISE SPECIFIED)

1. TOLERANCES: .X ± .1, .XX ± .03, .XXX ± .010, ANGULAR ± 1°.
2. BREAK ALL SHARP EDGES AND REMOVE ALL BURRS.
3. ~~ALL SURFACES 63 PER ANSI B46.1 (LATEST REV.)~~
4. ALL MACHINE TOOL RADII .03 MAX.
5. ALL MATERIAL SHALL BE AS SPECIFIED, OR APPROVED EQUAL QUALITY.
6. INTERIOR TO BE FREE FROM DIRT, GREASE, ETC. BEFORE POURING LEAD. POUR LEAD SLOWLY AND CONTINUOUSLY TO PREVENT VOIDS.
7. SEQUENCE OF ASSEMBLY, MACHINING AND WELDING TO BE VENDORS RESPONSIBILITY.
8. ~~ALL EXPOSED SURFACES TO BE SMOOTH AND POLISHED.~~
9. RADIUS ALL CORNERS.
10. CALCULATED WEIGHT 25,000 LBS (APPROX).
11. DYE PENETRANT CHECK TO BE IN ACCORDANCE WITH SPECIFICATION FOR WELDING SERF CELL TRANSFER CASK, BY V. C. BOYLES, 8/17/76.
12. ~~ALL WELDING TO BE IN ACCORDANCE WITH RDT STANDARD [illegible], SECTION [illegible]~~
13. WELD FILLER MATERIAL SHALL BE TYPE 308 OR 308L.
14. REFERENCE MATERIAL USED:
    a) ORNL-NSIC-68 CASK DESIGN GUIDE.
    b) STRESS ANALYSIS REPORT BY G. CHENAULT, JAN. 22, 1976.
    c) RADIATION AND SHIELD ANALYSIS BY KARLA MUELLER/R. L. BUNCH, DEC. 4, 1974.
15. WELDING AND NON-DESTRUCTIVE TESTING SHALL BE IN ACCORDANCE WITH SPECIFICATION FOR WELDING SERF CELL TRANSFER CASK, BY V.C. BOYLES, 8/17/76.
16. IT IS NOT NECESSARY TO MAKE THE 26" TUBE OUT OF ONE PIECE OF MATERIAL. WELDS CAN BE LOCATED AS REQUIRED.
17. ALL SURFACES 63 PER ANSI B46.1 (LATEST REV.) EXCEPT AS FOLLOWS: 125 FINISH ON OUTER SURFACE OF CASK (26.0 DIA.) IS ACCEPTABLE, PROVIDING THERE ARE NO IRREGULARITIES, SUCH AS NICKS, SCRATCHES, PORES, SCALE, ETC.
18. NONDESTRUCTIVE TESTING TO BE IN ACCORDANCE WITH ASME BOILER AND PRESSURE VESSEL CODE, SECTION V NONDESTRUCTIVE EXAMINATION.
19. WELDING TO BE IN ACCORDANCE WITH ASME BOILER AND PRESSURE VESSEL CODE, SECTION IX, WELDING AND BRAZING QUALIFICATIONS.



QTY	QTY	P.N.	DESCRIPTION	MAT'L/CATALOG NO.	[illegible]
	2	41	5/8-11 UNC x 1¼ LG. EYE BOLT	STL.	
	8	42	3/8 LOCKWASHER, MEDIUM SERIES, SPLIT	SST	
	44	43	1/2 LOCKWASHER, MEDIUM SERIES, SPLIT	SST	
	4	44	1/4 LOCKWASHER, MEDIUM SERIES, SPLIT	SST	
	4	45	5/16-18UNC-2A x 1¼ LG. FLAT HEAD SOCKET CAP SCREW	SST, PER ASTM A193 TYPE B8	
	1	46	SPACER	SHEET 1	

QTY	QTY	P.N.	DESCRIPTION	MAT'L/CATALOG NO.	[illegible]
	X	1	ASSEMBLY	SHEET 2	
	1	2	CASK	SHEET 3	
		3	23,000 LBS (APPROX.) PIG LEAD, CORRODING GRADE,	ASTM B29	
	1	4	END PLUG	SHEET 7	
	1	5	AUXILIARY PUSH ROD COVER	SHEET 5	
	1	6	LOCK RETAINER	SHEET 8	
	1	7	PUSH ROD LOCK	SHEET 8	
	1	8	END COVER	SHEET 7	
	1	9	LOCKING PIN	SHEET 4	
	1	10	PIN RETAINER	SHEET 8	
	1	11	KNOB	SHEET 9	
	1	12	CRANK HANDLE	SHEET 9	
	1	13	DRUM	SHEET 10	
	1	14	BEARING RETAINER	SHEET 10	
	1	15	BEARING RETAINER	SHEET 10	
	1	16	COVER PLATE	SHEET 10	
	1	17	COVER PLATE	SHEET 10	
	1	18	SHAFT	SHEET 10	
	1	19	SHAFT	SHEET 10	
	1	20	STOP PLATE	SHEET 10	
	1	21	SEAL	SHEET 1	
	1	22	SEAL	SHEET 1	
	2	23	SEAL	SHEET 1	
	1	24	SEAL	SHEET 1	
	1	25	SEAL	SHEET 1	
	4	26	3/8-16 UNC-2A x 1¼ LG. BOLT, HEX. HD.	SST	
	18	27	1/2-13 UNC-2A x 1-3/4 LG. BOLT, HEX. HD.	SST, PER ASTM A193 TYPE B8	
	8	28	1/4-20 UNC-2A x 5/8 LG. SOCKET HEAD CAP SCREW	SST	
	25	29	1/2-13 UNC-2A x 1-1/2 LG. SOCKET HEAD CAP SCREW	SST, PER ASTM A193 TYPE B8	
	8	30	3/8-16 UNC-2A x 1" LG. FLAT HEAD SOCKET CAP SCREW	SST, PER ASTM A193 TYPE B8	
	2	31	1/4-20 UNC-2B NUT, HEX	SST	
	2	32	BEARING "FAFNIR"	7205 W	
	2	33	BEARING "TORRINGTON"	B-2416	
	2	34	RETAINING RING "TRUARC"	N5000-437	
	1	35	"O" RING, "PARKER" SIZE 2-212	NEOPRENE	
	1	35	"O" RING, "PARKER" SIZE 2-119	NEOPRENE	
	1	37	SPRING, "ASSOCIATED SPRING CORP."	C0720-057-1000	
	1	38	1/8 DIA. x 1/2 LG. DOWEL PIN	SST	
	1	39	1/8 DIA. x 7/8 LG. DOWEL PIN	SST	
	1	40	1/4 DIA. x 1¼ LG. DOWEL PIN	SST	

EXCEPT AS NOTED
QA EFFORT REQUIRED
TYPE OR CATEGORY II
ENGINEER V.C. Boyles
QA [illegible]

BY/DATE		CHECKED		APPROVED		DESCRIPTION	REV
T.T.A 7-10-80		7/80 JPZ				PC 18/25 WAS "GASKET" PC 5 WAS "PUSH ROD COVER"	5
KRW		UCB			GHV	a) QA LEVEL WAS III a) IN GENERAL NOTES, NOTES #18 & 19 ADDED	4
T.T.A 7-10-80		UCB			GHV	a) L.M. PT. #46 ADDED a) ZN. E-10, PT. #46 ADDED	3
C.H.S.	SA	UCB			GHV	c) NOTE #16 ADDED b) L.M., PN 3 WAS COMMERCIAL GRADE a) NOTE 3 DELETED, NOTE 17 ADDED.	2
T.T.A 7-16-80		UCB			GHV	d) NOTES #13, 14 & 15 ADDED. c) NOTE #12 DELETED b) NOTE #11 WAS: DYE PENETRANT CHECK TO BE IN ACCORDANCE WITH ASME BOILER AND PRESSURE VESSEL CODE SECTION 5, ARTICLES 6 AND 24. a) NOTE #3 DELETED	1

REVISIONS

DWG. NO.		DRAWING TITLE
H-3-38571		LIFTING BAIL
H-3-38575		BOAT DETAILS
END USE		CELL TRANSFERS

REFERENCE DRAWINGS

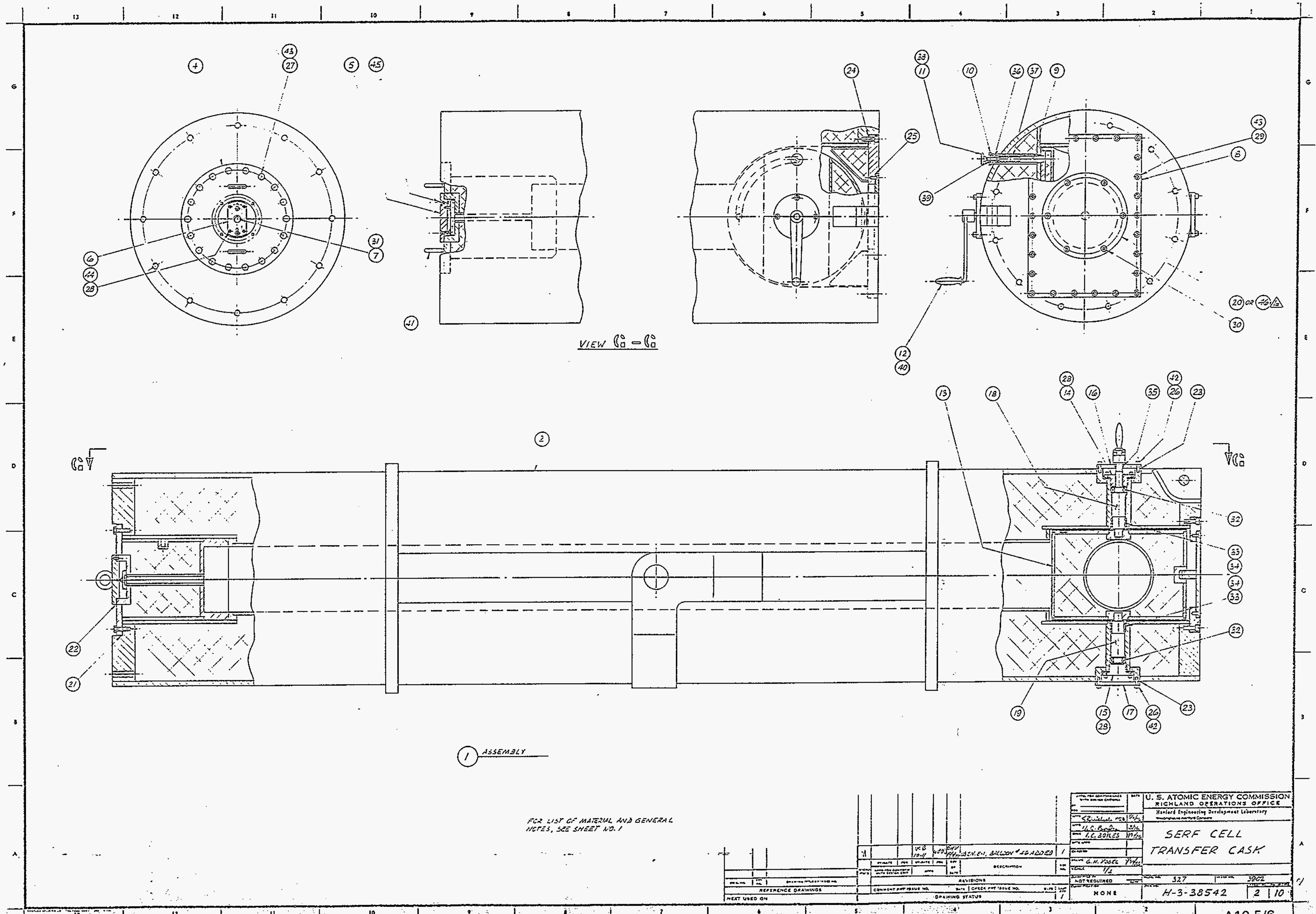
NEXT USED ON

DRAWING STATUS

U. S. ATOMIC ENERGY COMMISSION
RICHLAND OPERATIONS OFFICE
Hanford Engineering Development Laboratory

SERF CELL
TRANSFER CASK

V.C. BOYLES
DRAWN G. H. VOGEL
SCALE NOTED
NOT REQUIRED
NONE
327
5902
H-3-38542
1 / 10



FOR LIST OF MATERIAL AND GENERAL NOTES, SEE SHEET NO. 1

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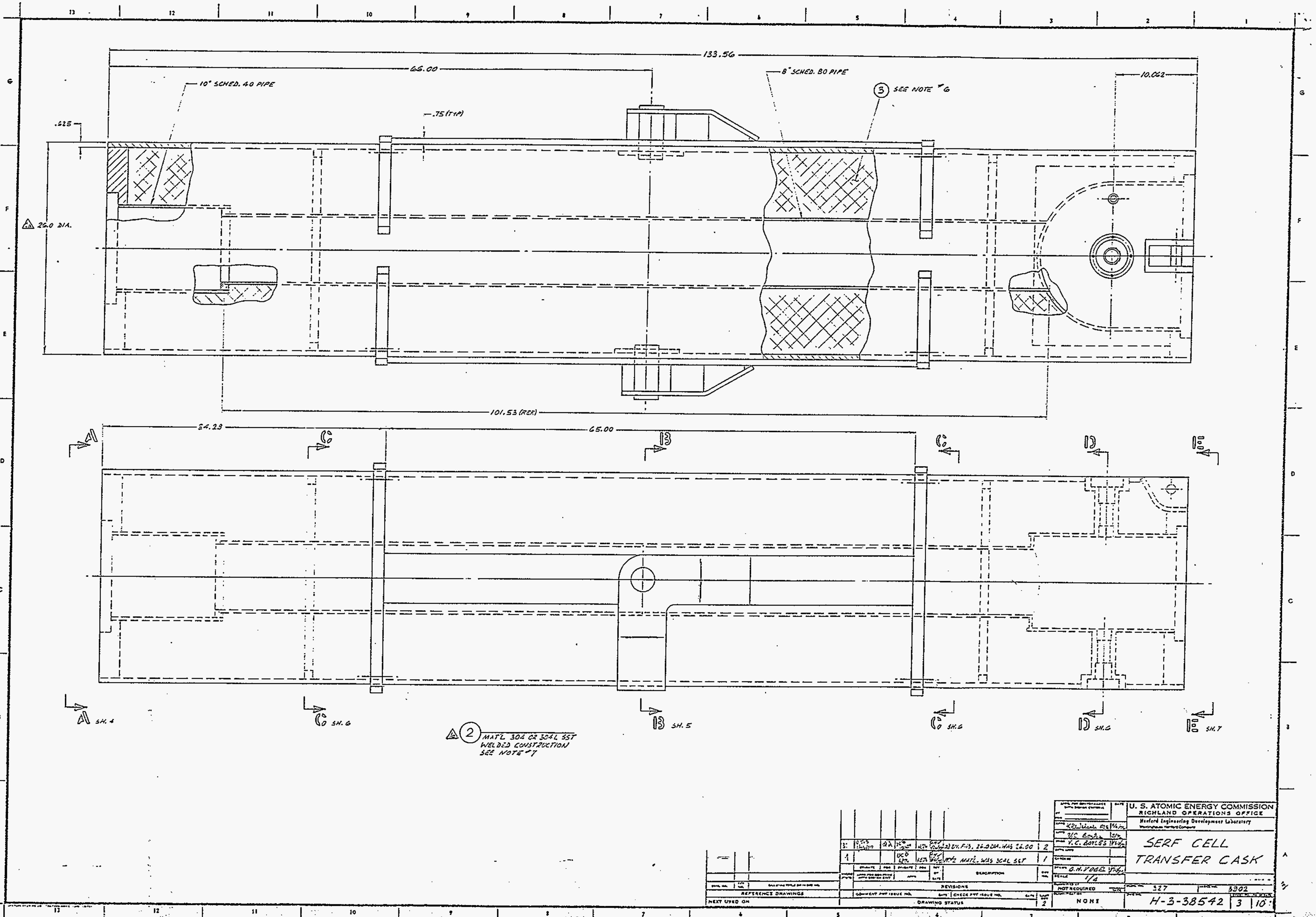
Hanford Engineering Development Laboratory

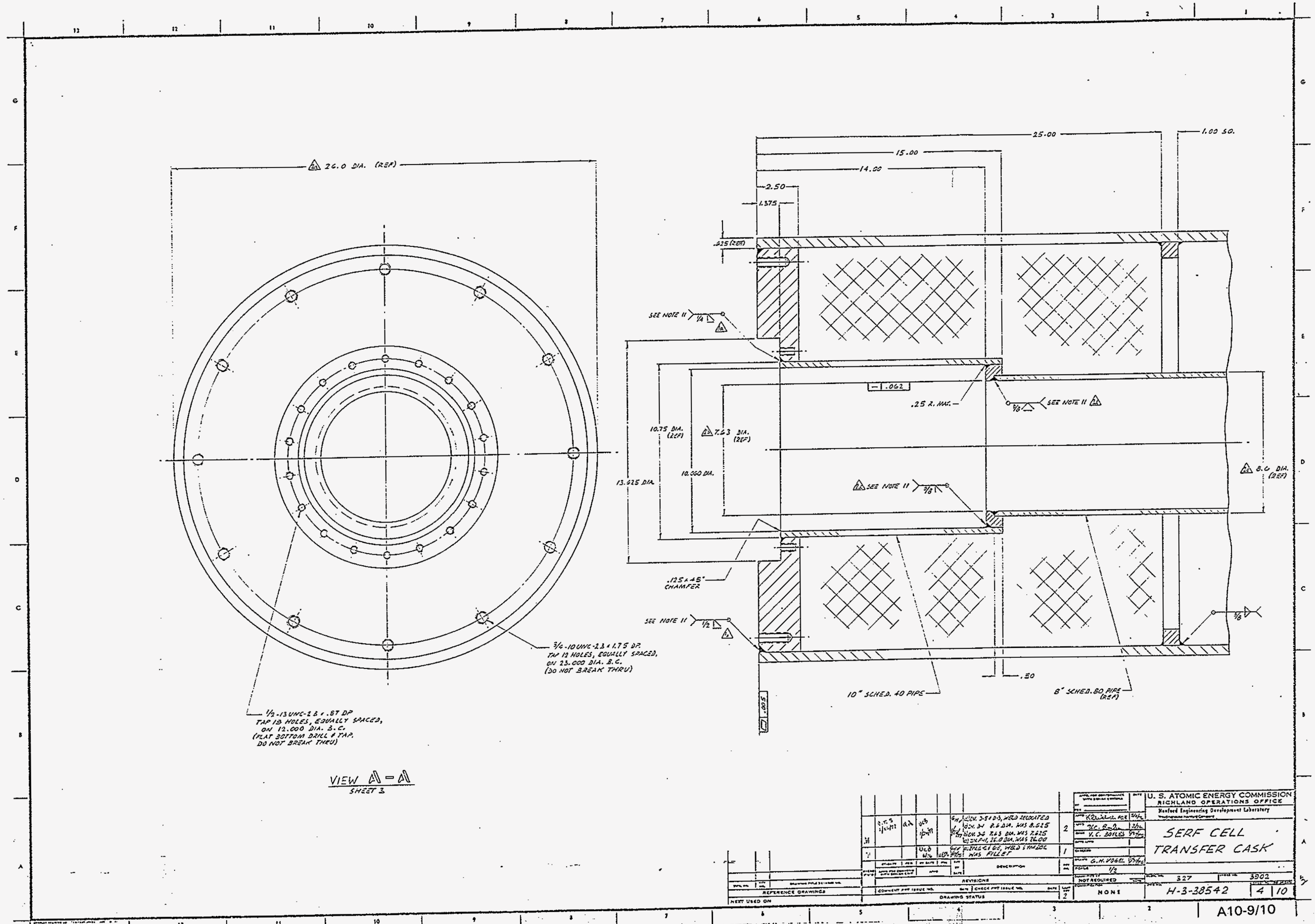
SERF CELL
TRANSFER CASK

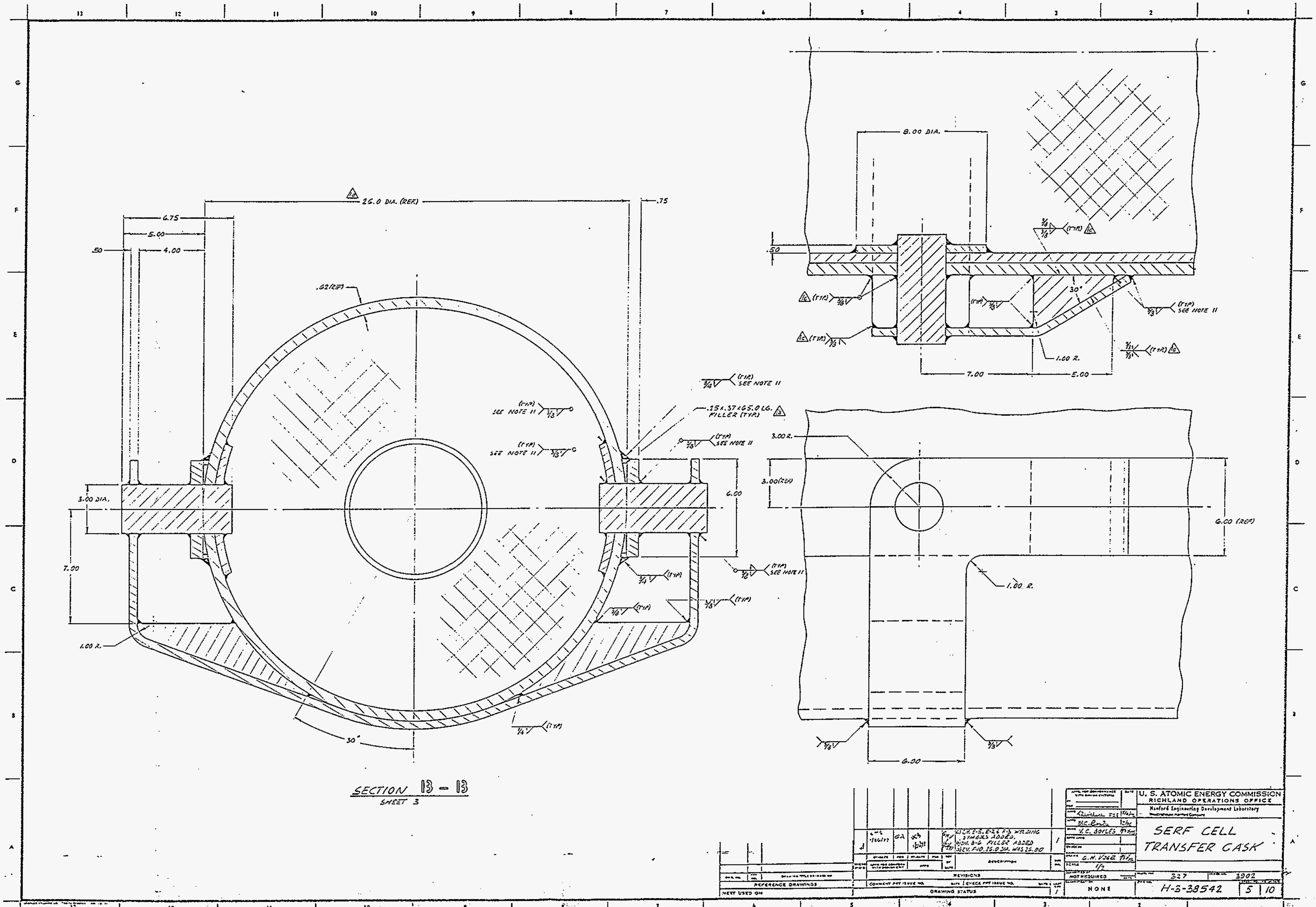
1 | ECN E-1, BALLOON # 46 ADDED

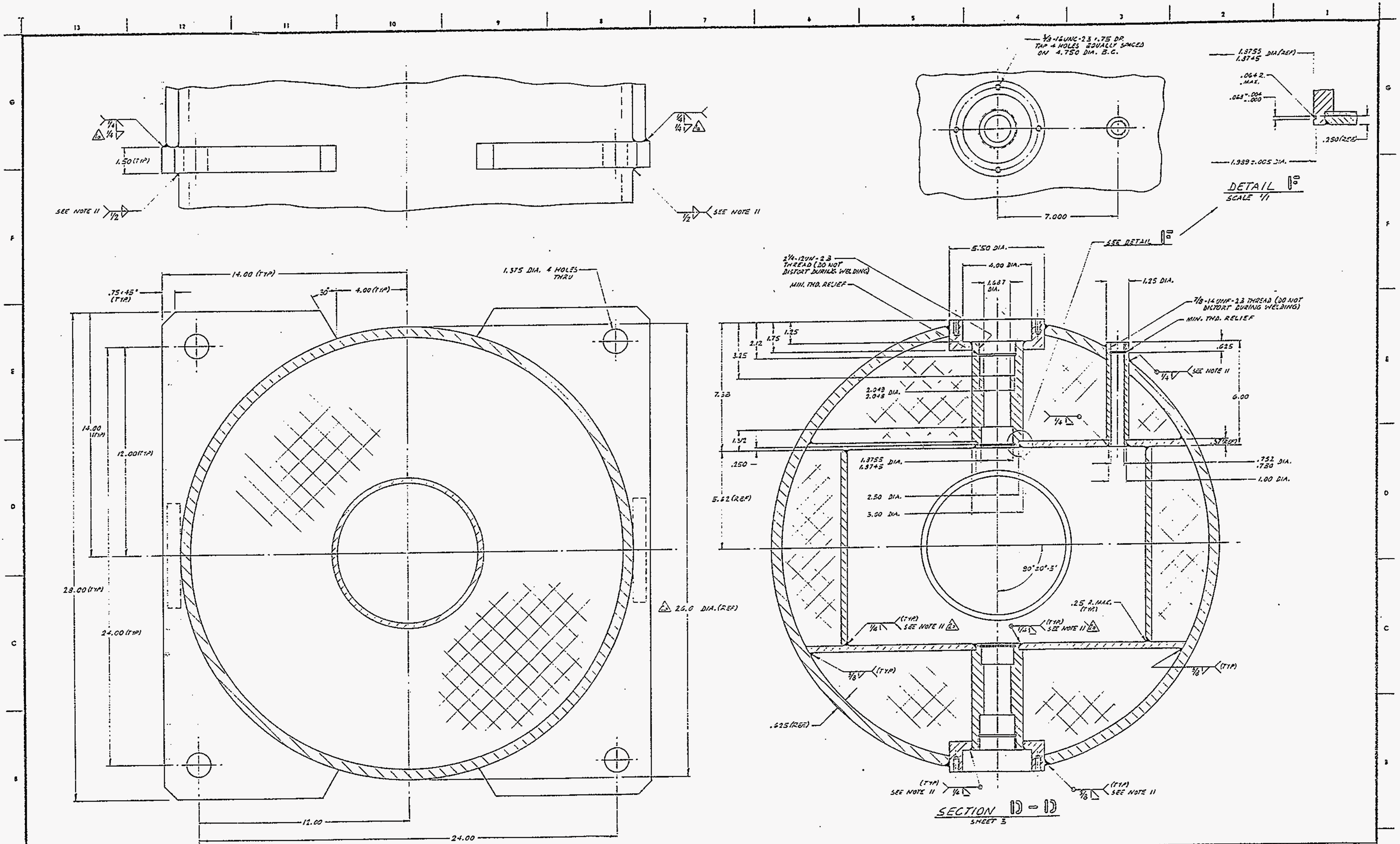
I. C. BOILES
G. H. VOSEL
1/2
NOT REQUIRED
NONE
327
3902
H-3-38542
2 | 10

REVISIONS
REFERENCE DRAWINGS
NEXT USED ON
DRAWING STATUS


133.56
66.00
10° SCHED. 40 PIPE
8" SCHED. 80 PIPE
③ SEE NOTE #6
10.062
.625
.75 (TYP)
26.0 DIA.
101.53 (REF)
24.23
65.00
A SH. 4
C SH. 6
B SH. 5
C SH. 6
D SH. 6
E SH. 7
② MAT'L 304 OR 304L SST WELDED CONSTRUCTION SEE NOTE #7
2 ECN F-13, 26.0 DIA. WAS 26.00 2
MAT'L WAS 304L SST 1
DESCRIPTION
REVISIONS
REFERENCE DRAWINGS
NEXT USED ON
DRAWING STATUS
U. S. ATOMIC ENERGY COMMISSION
RICHLAND OPERATIONS OFFICE
Hanford Engineering Development Laboratory
Westinghouse Hanford Company
V.C. BOYLES
G.H. VOGEL
SCALE 1/4
NOT REQUIRED
NONE
SERF CELL
TRANSFER CASK
327
3302
H-3-38542
3
10


26.0 DIA. (REF)
25.00
1.00 SQ.
15.00
14.00
2.50
1.375
.625 (REF)
SEE NOTE 11 1/4
SEE NOTE 11 1/2
.125 x 45° CHAMFER
13.625 DIA.
10.75 DIA. (REF)
7.63 DIA. (REF)
10.060 DIA.
.062
.25 R. MAX.
SEE NOTE 11 3/8
3/8 SEE NOTE 11
8.6 DIA. (REF)
3/8
.005
.50
10" SCHED. 40 PIPE
8" SCHED. 80 PIPE (REF)
3/4-10UNC-2B x 1.75 DP. TAP 12 HOLES, EQUALLY SPACED, ON 23.000 DIA. B.C. (DO NOT BREAK THRU)
1/2-13UNC-2B x .87 DP TAP 18 HOLES, EQUALLY SPACED, ON 12.000 DIA. B.C. (FLAT BOTTOM DRILL & TAP. DO NOT BREAK THRU)
VIEW A-A
SHEET 3
REVISIONS
DESCRIPTION
2
ZONE D-5 & D-3, WELD RELOCATED
ZONE D4 8.6 DIA. WAS 8.625
ZONE D6 7.63 DIA. WAS 7.625
ZONE F-11, 26.0 DIA. WAS 26.00
1
ZONE C-6 & C-6, WELD SYMBOL WAS FILLET
DRAWING STATUS
REFERENCE DRAWINGS
NEXT USED ON
U. S. ATOMIC ENERGY COMMISSION
RICHLAND OPERATIONS OFFICE
Hanford Engineering Development Laboratory
V. C. BOYLES
G. H. VOGEL
SCALE 1/2
NOT REQUIRED
NONE
SERF CELL
TRANSFER CASK
327
3902
H-3-38542
4
10


26.0 DIA. (REF)
.75
6.75
5.00
.50
4.00
.62(REF)
3.00 DIA.
7.00
1.00 R.
30°
6.00
(TYP) SEE NOTE 11
.15 x .37 x 65.0 LG. FILLER (TYP.)
(TYP) SEE NOTE 11
(TYP)
SECTION B – B
SHEET 3
8.00 DIA.
.50
(TYP)
30°
(TYP) SEE NOTE 11
1.00 R.
7.00
5.00
3.00 R.
3.00(REF)
6.00 (REF)
1.00 R.
6.00
ECN. E-5, E-26 F-5 WELDING SYMBOLS ADDED.
ECN. B-6 FILLER ADDED
REV. F-10. 26.0 DIA. WAS 26.00
REVISIONS
REFERENCE DRAWINGS
NEXT USED ON
DRAWING STATUS
U. S. ATOMIC ENERGY COMMISSION
RICHLAND OPERATIONS OFFICE
Hanford Engineering Development Laboratory
V.C. BOYLES
G.H. VOGEL
1/2
NOT REQUIRED
NONE
SERF CELL
TRANSFER CASK
327
3902
H-3-38542
5 10



SECTION C-C
TYP. 2 PLACES
SHEET 3

	CT.S 4/26/77	G.A.	[illegible]	[illegible]	[illegible]	[illegible] C4105 WELD RELOCATED [illegible] 26.0 DIA. WAS 26.00	2
			[illegible]	[illegible]	[illegible]	[illegible] WELD SYMBOL WAS DOUBLE GROOVE	1
						DESCRIPTION	
REVISIONS							
	COMMENT PRT ISSUE NO.					CHECK PRT ISSUE NO	2
DRAWING STATUS							

REFERENCE DRAWINGS
NEXT USED ON

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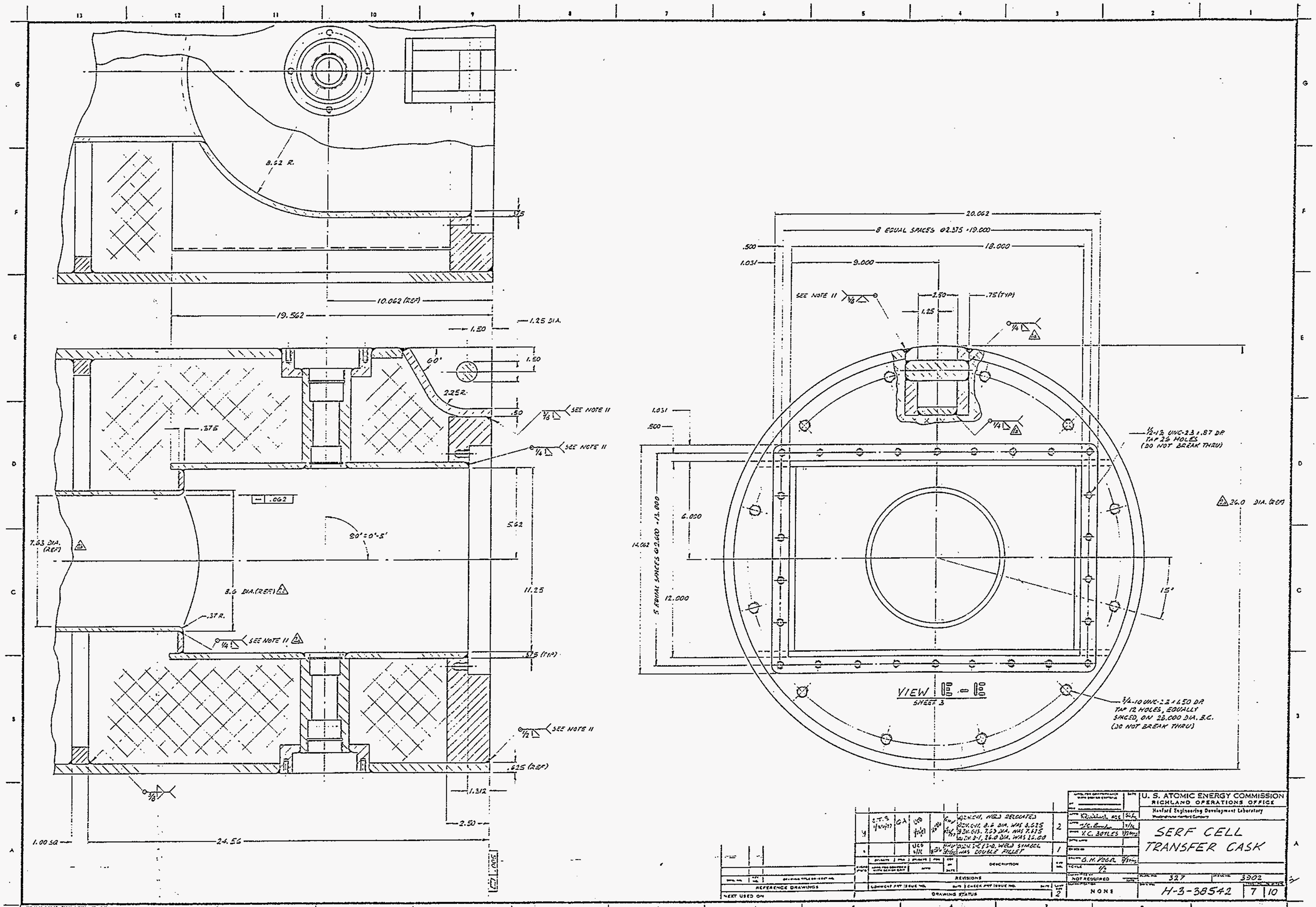
RICHLAND OPERATIONS OFFICE
Hanford Engineering Development Laboratory
Westinghouse Hanford Company

V.C. Boyd
I.C. BOYLES
G.H. VOGEL
SCALE 1/2 & NOTED
NOT REQUIRED
NONE

SERF CELL
TRANSFER CASK

327
3902
H-3-38542
6 | 10



U. S. ATOMIC ENERGY COMMISSION
RICHLAND OPERATIONS OFFICE
Hanford Engineering Development Laboratory
Westinghouse Hanford Company

SERF CELL
TRANSFER CASK

4	C.T.S. 1/21/77	G.A.	UCD 1/21/77		GHV 1/21/77	4) ZN.C-11, WELD RELOCATED ZN.C-11, 8.6 DIA. WAS 8.625 ZN.C-15, 7.63 DIA. WAS 7.625 ZN.D-1, 26.0 DIA. WAS 26.00	2
			UCD 1/11		GHV	ZN.D-5, E3-D, WELD SYMBOL WAS DOUBLE FILLET	1

REVISIONS

DESCRIPTION

REFERENCE DRAWINGS

NEXT USED ON

DRAWING STATUS

V.C. BOYLES

G. H. VOGEL

SCALE 1/2

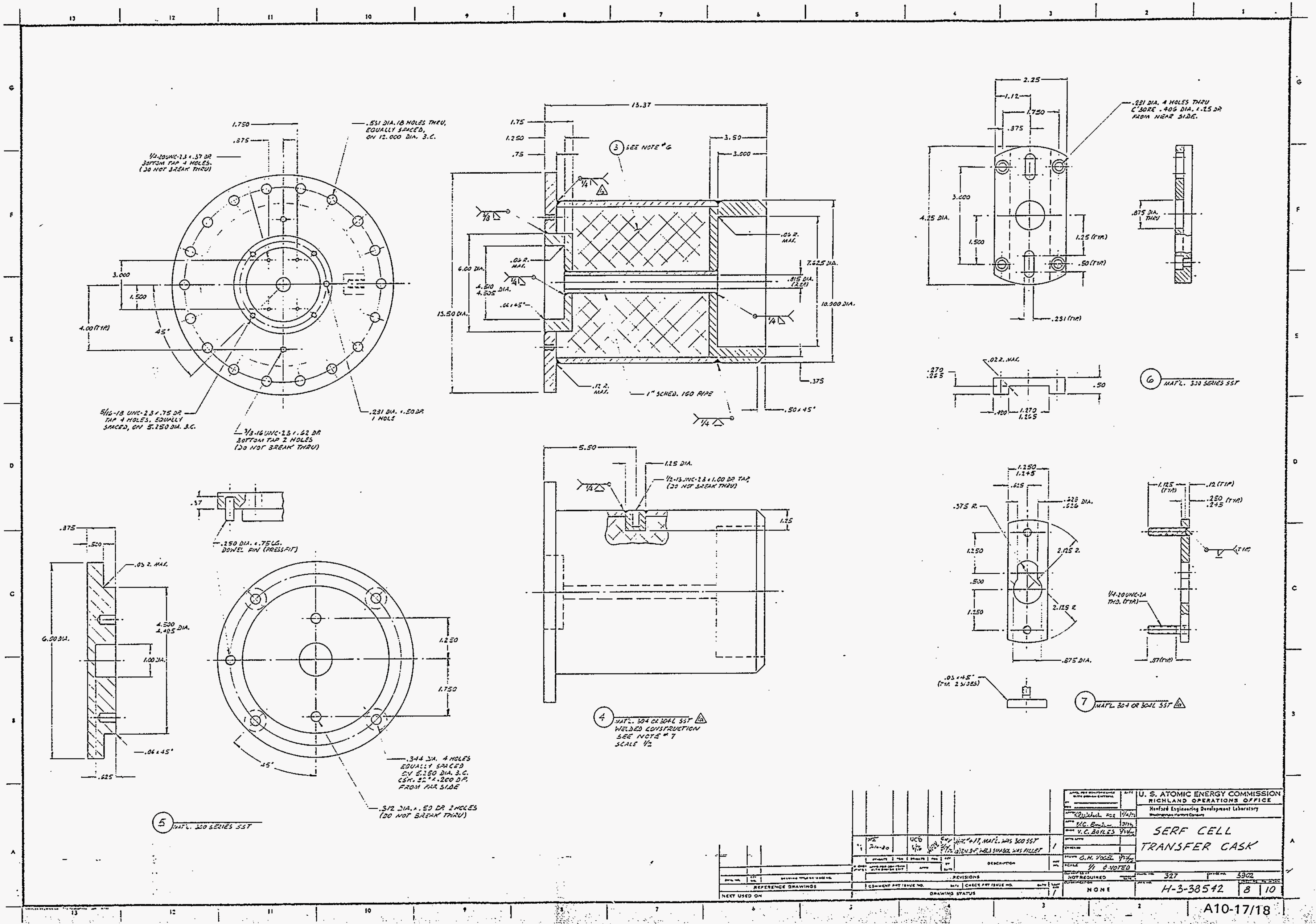
NOT REQUIRED

NONE

327

3902

H-3-38542 7 10



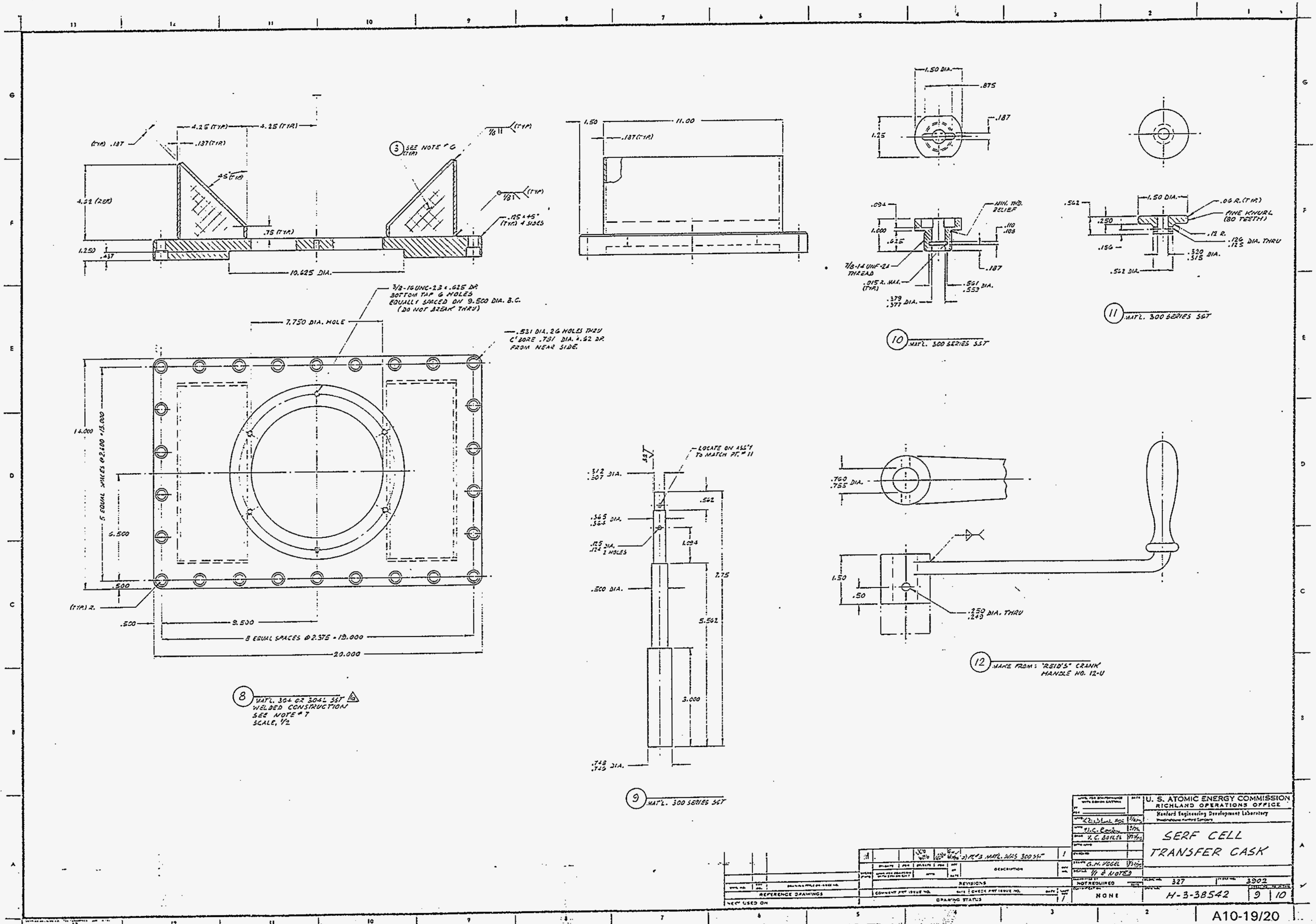
U. S. ATOMIC ENERGY COMMISSION
RICHLAND OPERATIONS OFFICE
Hanford Engineering Development Laboratory

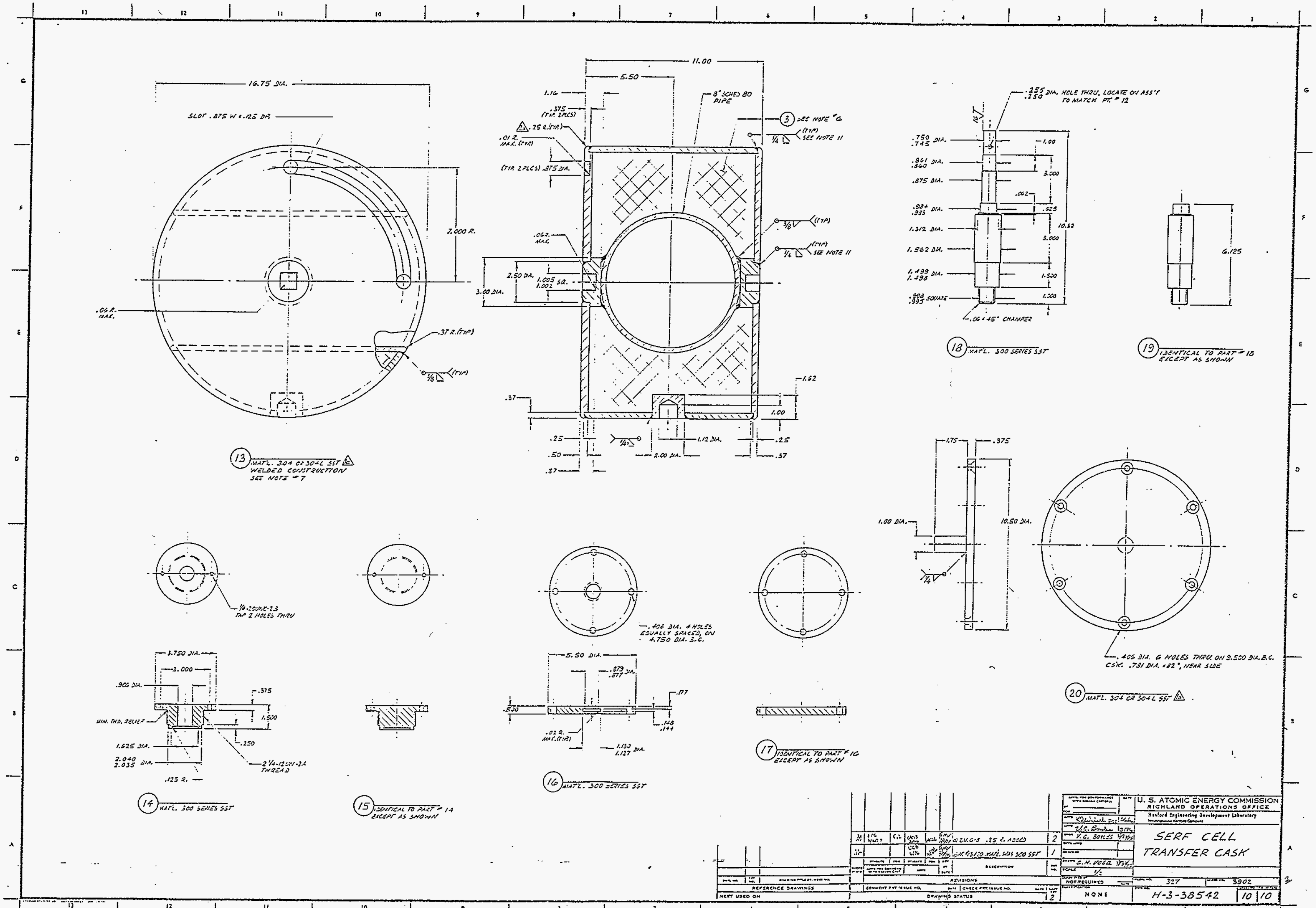
SERF CELL
TRANSFER CASK

DRAWN G.H. VOGEL
SCALE 1/1 & NOTED
V.C. BOILES
NONE
327
3902
H-3-38542 8 10

REVISIONS: 1 — ①SH. 4+17, MAT'L. WAS 300 SST ②ZV.3-F, WELD SYMBOL WAS FILLET

A10-17/18


(TYP) .137
4.25 (TYP)
4.25 (TYP)
.137 (TYP)
45° (TYP)
4.52 (REF)
.75 (TYP)
1.250
.437
10.625 DIA.
③ SEE NOTE # 6 (TYP)
1/8
1/8
(TYP)
.125 x 45° (TYP) 4 SIDES
1.50
11.00
.137 (TYP)
3/8-16UNC-2B x .625 DP. BOTTOM TAP 6 HOLES EQUALLY SPACED ON 9.500 DIA. B.C. (DO NOT BREAK THRU)
7.750 DIA. HOLE
.531 DIA. 26 HOLES THRU C'BORE .781 DIA. x .62 DP. FROM NEAR SIDE.
14.000
5 EQUAL SPACES @ 2.600 = 13.000
6.500
.500
(TYP) 2.
.500
9.500
8 EQUAL SPACES @ 2.375 = 19.000
20.000
⑧ MAT'L. 304 OR 304L SST
WELDED CONSTRUCTION
SEE NOTE # 7
SCALE, 1/2
LOCATE ON ASS'Y TO MATCH PT. # 11
.512
.507 DIA.
.562
.565
.564 DIA.
.125
.124 DIA. 2 HOLES
1.094
7.75
.500 DIA.
5.562
3.000
.748
.745 DIA.
⑨ MAT'L. 300 SERIES SST
1.50 DIA.
.875
.187
1.25
.094
1.000
.625
MIN. THD. RELIEF
.110
.108
7/8-14 UNF-2A THREAD
.015 R. MAX. (TYP)
.187
.561
.553 DIA.
.379
.377 DIA.
⑩ MAT'L. 300 SERIES SST
.562
1.50 DIA.
.06 R. (TYP)
FINE KNURL (80 TEETH)
.250
.12 R.
.126
.125 DIA. THRU
.156
.520
.515 DIA.
.562 DIA.
⑪ MAT'L. 300 SERIES SST
.760
.755 DIA.
1.50
.50
.250
.249 DIA. THRU
⑫ MAKE FROM: "REID'S" CRANK HANDLE NO. 12-U
U. S. ATOMIC ENERGY COMMISSION
RICHLAND OPERATIONS OFFICE
Hanford Engineering Development Laboratory
SERF CELL
TRANSFER CASK
MAT'L. WAS 300 SST
DESCRIPTION
REVISIONS
REFERENCE DRAWINGS
NEXT USED ON
COMMENT PRT ISSUE NO.
CHECK PRT ISSUE NO.
DRAWING STATUS
V. C. BOILES
G. H. VOGEL
1/1 & NOTED
NOT REQUIRED
NONE
327
3902
H-3-38542
9
10



2	1/5	C.L.	UCB		GHV	2 ZN. G-3 .25 R. ADDED	2
			UCB		GHV	ZN. F3 1/20 MAT'L. WAS 300 SST	1

REVISIONS

REFERENCE DRAWINGS

NEXT USED ON

DRAWING STATUS

U. S. ATOMIC ENERGY COMMISSION
RICHLAND OPERATIONS OFFICE
Hanford Engineering Development Laboratory

V.C. Boyles

DRAWN G.H. Vosea

SCALE 1/2

NOT REQUIRED

NONE

SERF CELL TRANSFER CASK

327

3902

H-3-38542

10 | 10

10.2 USER BIENNIAL INSPECTION CHECKLIST AND NONDESTRUCTIVE TESTING OF LIFTING APPARATUS

B&W Hanford
Nuclear Facilities Operations
327 Facility

RADIOACTIVE MATERIAL SHIPPING CONTAINER
USER BIENNIAL INSPECTION CHECKLIST

Container No: A-50 Inspection Date:_________ Next Inspection Due:_________

Item No.	Spec; Section or Dwg; View	Characteristic/Requirement	Reference or Test Method	Inspected By (Init.)
1.	H-3-38542 Sheet #2	Closure "drum" position marked "OPEN" and "CLOSED" on cask.	Visual Inspection	_________
2.	H-3-38542 Sheet #2	Closure "drum" handle position is correct in reference to the "OPEN" and "CLOSED" markings.	Functional Test	_________
3.	H-3-38542 Sheet #2	Closure "drum" locking device functions properly.	Functional Test	_________
4.	H-3-38542 Sheet #2	Closure "drum" will close properly with transfer "boat" positioned to rear of cask.	Functional Test	_________
5.	H-3-38542 Sheet #2	Rear locking device holds the transfer "boat" in position.	Functional Test	_________
6.	H-3-38542 Sheet #2	Verify that closure "drum" rotates freely.	Functional Test	_________
7.	JHB 327-5	Verify that lifting "bail" is installed properly & excessive wear is not evident.	Visual Inspection	_________
8.	JHB 327-5	Verify surface is free of damage and rust.	Visual Inspection	_________
9.	JHB 327-5	Verify that no weld cracking is evident.	Visual Inspection	_________
10.	JHB 327-5	Check for loose bolts and/or broken parts.	Visual Inspection	_________

Custodian Signature: ______________________

BATTELLE
Pacific Northwest Laboratories
Remote Systems Technology Department

RADIOACTIVE MATERIAL SHIPPING CONTAINER NDT OF LIFTING APPARATUS (5 YEAR)

Container No: A50 Inspection Due Date: ______ Next Inspection Due: ________

Item No.	Spec: Section or Dwg: View	Characteristic/Requirement	Test Method	Inspected By (QA)
1.	H-3-38542 Sheet #5	Welds attaching trunnions to cask (zones D-12 & D-7). Two welds on each side of cask.	N/A per NCR A10138. Use restricted.	Init:____ Date:____
2.	H-3-38542 Sheet #6	Welds attaching lifting plates to cask (zones E-12 & E-8). Two welds on each end of cask.	Dye penetrant	Init:____ Date:____
3.	RDT E 12-7T Para.4.4.1.a	Welds attaching lifting plates to lifting bail.	Magnetic particle	Init:____ Date:____
4.	RDT E 12-7T Para.4.4.1.a	Four bolts attaching lifting bail to cask lifting plates.	Magnetic particle	Init:____ Date:____

EQUIPMENT TEST INFORMATION

1. Requester verifies that the equipment to be examined is:
   - ___ Safe.
   - ___ Possible unsafe conditions. ____________________
   - ___ Radiation - Dose Rate. ____________________
2. Anticipated part temperature: ___ Ambient ___Other ______________
3. Material type to be examined: ____________________
4. Area to be inspected: ___ Spot Inspection ______________
   ___ Full Inspection (100% of area requested).
5. QA Plan: __________ Impact Level: _________ Test Procedure: __________
6. Acceptance Standard: ____________________
7. Comments: ____________________

   ____
8. QA Rep. contacted: ______________ Test Date: _______ Time: _____

Custodian Signature: ____________________

PART B: PACKAGE EVALUATION

1.0 INTRODUCTION

The Special Environmental Radiometallurgy Facility (SERF) Cask is evaluated in this part for normal transport conditions for onsite transportation. Documentation is required to demonstrate that the container will prevent loss of the contents during all normal handling and transport conditions.

1.1 EVALUATION SUMMARY AND CONCLUSIONS

As shown by the following evaluations, the SERF Cask will prevent loss of contents through all normal transport conditions. Ensuring that loss of contents will be precluded is demonstrated by evaluating the cask for normal transport conditions and risk analysis, showing that an accident resulting in loss of contents is incredible.

1.1.1 Contents

The typical contents of the SERF Cask are evaluated in Part B, Section 2.0.

1.1.2 Radiological Risk

The risk evaluation for the SERF Cask demonstrates that the SERF Cask meets the onsite transportation safety criteria. In order to satisfy those criteria, the family of 327 Building casks, which include the SERF Cask, the Radioactive Waste Disposal Cask, and the Plutonium Recycle Test Reactor (PRTR) Graphite Cask, cannot be transported more than a total for the three casks of 16.0 km (10.0 mi) a year. This does not apply to empty cask shipments.

1.1.3 Containment

Containment was evaluated based on the inner containers providing the containment function. The SERF Cask retains the contents, but the SERF Cask is not considered to be a containment boundary. The containment evaluation is presented in Part B, Section 4.0.

1.1.4 Shielding

The shielding analysis for the source term is presented in Part B, Section 5.0. The analysis demonstrates that a dose of 200 mrem/h will not be exceeded with the source term presented in Part A, Section 3.0.

1.1.5 Criticality

The criticality analysis for the SERF Cask is contained in *Limits for Mixed Oxide Fuel Pins, N Reactor Fuel Elements and Scrap in the SERF Cask and the Waste Cask* (Larson 1995). The limit for ^{235}U only is 250 g, while the limit for other fissionable materials is 150 g. Specific fuel pin and fuel element limits are given in Part A, Section 3.0.

1.1.6 Structural

The structural analysis for the SERF Cask is contained in Part B, Chapter 7.0. This analysis shows the SERF Cask and the inner containers contain the contents and maintain shielding during all normal transport conditions. Accident conditions are addressed in the risk evaluation.

1.1.7 Thermal

The thermal analysis for the SERF Cask is contained in Part B, Section 8.0. This analysis demonstrates that the cask performs acceptably during extreme weather conditions on the Hanford Site.

1.1.8 Gas Generation

The gas generation evaluation in Part B, Section 9.0, prohibits materials that may produce gas from being loaded into the SERF Cask. The contents of the SERF Cask will be dry to prevent gas generation from radiolysis.

1.1.9 Tiedown System

The package tiedown evaluation in Part B, Section 10.0, ensures that the SERF Cask will remain on the trailer under normal transport conditions.

1.2 REFERENCE

Larson, S. L., 1995, *Limits for Mixed Oxide Fuel Pins, N Reactor Fuel Elements and Scrap in the SERF Cask and the Waste Cask* (NCS Basis Memo 95-3, Rev. 1, to M. Dec, August 31), Battelle Pacific Northwest Laboratory, Richland, Washington.

2.0 CONTENTS EVALUATION

2.1 CHARACTERIZATION

Contents to be transported in the SERF Cask will consist of Fast Flux Test Facility (FFTF) and Power Reactor and Nuclear Fuel Development Corporation (PNC) fuel, mixed oxide, 20% ^{240}Pu, 0.9 wt% enriched uranium, structural material from reactors, and cesium chloride capsules. The contents shall be limited to the maximum allowable source term shown in Table B2-1.

Table B2-1. Maximum Allowable Source Term.

Source	Material	Activity limit	
		TBq	Ci
Mixed inventory	Fissile materials and α emitters*	-----*	-----*
	Mixed fission products	37	1000
	Mixed activation products	555	15000
Cesium capsules	Cesium chloride	2960	80000
Fuel	Fuel	-----*	-----*

*Fissile/fissionable materials limited by criticality safety as shown in Part B, Section 2.1.1.

Table B2-2. Decay Heat and A_2 Calculation for the SERF Cask Mixed Inventory.

Nuclide	Activity		Heat production factor (W/Ci)	Heat generation rate (W)	A_2 (Ci)	A_2s
	Bq	Ci				
^{90}Sr	3.70 E+13	1.00 E+03	1.16 E-03	1.16 E+00	2.70 E+00	3.70 E+02
^{90}Y*	3.70 E+13	1.00 E+03	5.54 E-03	5.54 E+00	0.00 E+00	0.00 E+00
^{60}Co	5.55 E+14	1.50 E+04	1.54 E-02	2.31 E+02	1.08 E+01	1.39 E+03
^{137}Cs	3.70 E+13	1.00 E+03	1.01 E-03	1.01 E+00	1.35 E+01	7.41 E+01
^{137m}Ba*	3.50 E+13	9.46 E+02	3.92 E-03	3.71 E+00	0.00 E+00	0.00 E+00
^{239}Pu	4.01 E+11	1.09 E+01	3.06 E-02	3.32 E-01	5.41 E-03	2.01 E+03
Total	7.01 E+14	1.90 E+04		2.43 E+02		3.84 E+03

* This radionuclide is a daughter as defined in 49 CFR 173.433, therefore, its activity was set to 0 for the A_2 calculations.

Table B2-3. Decay Heat and A_2 Calculation for the SERF Cask Cesium Capsule Inventory.

Nuclide	Activity		Heat production factor (W/Ci)	Heat generation rate (W)	A_2 (Ci)	A_2s
	Bq	Ci				
^{137}Cs*	2.96 E+15	8.00 E+04	1.01 E-03	8.08 E+01	1.35 E+01	5.93 E+03
^{137m}Ba**	2.80 E+15	7.568 E+04	3.92 E-03	2.97 E+02	0.00 E+00	0.00 E+00
Total	5.76 E+15	1.56 E+05		3.77 E+02		5.93 E+03

*This is a highway route control quantity

**This radionuclide is a daughter as defined in 49 CFR 173.433, therefore, its activity was set to 0 for the A_2 calculations.

As shown in Table B2-2, the number of A_2s is less than 3,000 and the total inventory is less than 1,000 TBq (27,000 Ci); therefore, the mixed radioactive material inventory is a fissile, Type-B, non-HRCQ (49 CFR 173). However, the cesium capsule inventory, as shown in Table B2-3, exceeds 1,000 Tbq (27,000 Ci); therefore, the cesium capsule inventory is a Type-B, HRCQ (49 CFR 173).

Tables B2-2 and B2-3 do not include spent nuclear fuel due to the variability of the payload. The spent nuclear fuel payload is limited by criticality safety as shown in Part B, Section 2.1.1 and thermal limits as shown in Part B, Section 2.2. A spent nuclear fuel payload may qualify as a fissile, Type-B, HRCQ.

2.1.1 Fissile Material Content

The fissile/fissionable material limits for the SERF Cask are shown in Table B2-4 for shipments of mixed oxide fuel pins, N Reactor fuel assemblies, and scrap under dry conditions. Table B2-5 contains the limits for fuel pins of different compositions (^{239}Pu, PuC, PuN, PuO_2, U metal, UC, UN, and UO_2. Note that mixtures of fissionable material are possible provided that the sum-of-fractions method shown in *Criticality Safety* (PNL 1994) is followed.

Table B2-4. Criticality Limits in the SERF Cask for Dry (H:≤X5) Conditions.

Limits	Maximum fuel diameter (in.)	Waste Cask	
Mixed oxide fuel pin		Maximum no. fuel pins (length of 13.5 in.)	Maximum no. fuel pins (length of 37 in.)
$<$25 wt% Pu($>$10)O_2-U($<$94)O_2	0.220	54	40
$<$25 wt% Pu($>$10)O_2-U($<$36)O_2	0.230	68	48
$<$16 wt% Pu($>$10)O_2-U($<$41)O_2	0.261	58	40
$<$31 wt% Pu($>$10)O_2-U($<$0.72)O_2	0.220	87	58
$<$25 wt% Pu($>$10)O_2-U($<$68)O_2	0.250	56	40
$<$5 wt% Pu($>$8)O_2-U($<$9)O_2	0.500	46	28
N Reactor fuel element	Maximum fuel diameter (in.)	Maximum fuel element length of 26.1	
Mark IA and IV inner elements	1.170	136	
Mark IA and IV outer elements	2.350	33	
Mark IA and IV fuel assemblies	2.350	33	
Scrap	----	250 g ^{235}U only or 150 g total fissionable**	

Pu = plutonium.
U = uranium.

*The fuel compositions are given in terms of the maximum weight% Pu in the total U + Pu, the minimum weight % ^{240}Pu in the total Pu, and the maximum ^{235}U enrichment in the total U.

*No accountable amounts of ^{242m}Am, ^{243}Cm, ^{245}Cm, ^{249}Cf, or ^{251}Cf are permitted.

Source: Larson, S. L., 1995, *Limits for Mixed Oxide Fuel Pins, N Reactor Fuel Elements and Scrap in the SERF Cask and the Waste Cask* (NCS Basis Memo 95-3, Rev. 1, to M. Dec, August 31), Battelle Pacific Northwest Laboratory, Richland, Washington.

Table B2-5. Fuel Pin Limits for the SERF Cask.

Fuel type	Maximum enrichment (preirradiation composition)	Maximum fuel pin outer diameter (in.)	Maximum fuel length (in.)	Maximum fuel pin limit
Mixed U and Pu compounds (excluding ^{233}U)	100 wt% ^{235}U or ^{239}Pu	0.8	37	15
Mixed U and Pu compounds (excluding ^{233}U)	100 wt% ^{235}U or ^{239}Pu	1.3	37	8

Source: Hawkes, E. C., 1995, *Limits for Fuel Pins in the Waste Cask, the 327 Building A-Cell, and the 324 Building Shielded Materials Facility Hot Cells* (NCS Basis Memo 95-5 to M. Dec, September 6), Battelle Pacific Northwest Laboratory, Richland, Washington.

2.2 RESTRICTIONS

Contents as shown in Part B, Tables B2-1, B2-4, and B2-5, are the bounding conditions for the contents authorized in the SERF Cask. In addition, the contents shall be limited to a thermal output of 377 W (Part B, Section 8.0). All contents shall be in inner containers as described in Table B2-6.

Table B2-6. Inner Container Description.

Inner Container	Contents	Examples
Pin tubes - tubing and fittings must have a working rating of 3000 psi.	Fuel pins, and dispersible material	FFTF, PNC, and N Reactor fuel; mixed plutonium-uranium oxide fuel pins, uranium oxide fuel pins, dispersible scrap, pieces of fuel, and fines.
1 gallon container with friction fit lid.	Nondispersible solid contents[1]	Solid scrap, structural material.
Specification 2R per 49 CFR 178.360	Dispersible solid contents	Fuel pieces, scrap, and fines of activated fuel and materials.
Large solid items	Large solid items to big for a paint can will have fixed surface contamination and put directly into the cask.[1]	Solid structural material and activated metals.
Cesium capsule	Cesium chloride	

FFTF = Fast Flux Test Facility.
PNC = Power Reactor and Nuclear Fuel Development Corporation.

[1] Surface contamination of contents and container must not exceed 100 times the Table A4-1 limits. Verification by survey or the use of a fixative such as paint is required.

49 CFR 178, 1996, "Specifications for Packagings," *Code of Federal Regulations*, as amended.

2.3 SIZE AND WEIGHT

Only contents smaller than the internal cavity shall be considered for shipment in the SERF Cask. The contents weight shall not exceed 227 kg (500 lb). The total gross weight of the cask and contents shall not exceed 12,020 kg (26,500 lb).

2.4 CONCLUSIONS

The fissionable material limits described in *Limits for Mixed Oxide Fuel Pins, N Reactor Fuel Elements and Scrap in the SERF Cask and the Waste Cask* (Larson 1995) and discussed in Part B, Section 6.0, and the shielding analyses, shown in Part B, Section 5.0, demonstrate that the SERF Cask can safely ship the contents shown in Tables B2-1, B2-4, and B2-5. The established limits will preclude the possibility of a criticality and minimize radiological exposure to personnel during transport

As shown in Table B2-3 and Part B, Section 2.1, the maximum quantity of material to be shipped is a fissile, Type-B, HRCQ (49 CFR 173).

2.5 REFERENCES

49 CFR 173, 1997, "Shippers--General Requirements for Shipments and Packagings," *Code of Federal Regulations*, as amended

Larson, S. L., 1995, *Limits for Mixed Oxide Fuel Pins, N Reactor Fuel Elements and Scrap in the SERF Cask and the Waste Cask* (NCS Basis Memo 95-3, Rev. 1, to M. Dec, August 31), Battelle Pacific Northwest Laboratory, Richland, Washington.

PNL, 1994, *Criticality Safety*, PNL-MA-25, Battelle Pacific Northwest Laboratories, Richland, Washington.

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3.0 RADIOLOGICAL RISK EVALUATION

3.1 INTRODUCTION

The 327 Building casks, which include the Radioactive Waste Disposal Cask, the SERF Cask, and the PRTR Graphite Cask, are used to transport Type B, highway route controlled quantities of solid activated metals, irradiated fuel, and other solid radioactive materials among the 300 Area laboratories. Because none of the 327 Building casks are certified Type B containers, radiological risks are evaluated to determine compliance with onsite transportation safety requirements per WHC-CM-2-14. Although separate safety documentation has been issued for each cask, the radiological risk evaluation is prepared for the 327 Building casks as a composite analysis. The composite analysis was prepared due to the similarity in payload, cask type, and transport environment.

The 327 Building casks are used routinely over short distances (less than 0.40 km [0.25 mi]) within the 300 Area. The casks are transported by truck.

The assumptions for the radiological risk evaluation are summed as follows:

- Highway mode
- Approximately 0.40 km (0.25 mi) per trip
- A maximum of 24 trips per year total for the family of casks
- One cask per shipment.

For accident environments, the 327 Building casks must meet onsite transportation safety requirements as outlined in WHC-CM-2-14 and Mercado (1994). The required safety is determined by a radiological risk evaluation that uses dose consequences, risk acceptance criteria, cask failure threshold values, and Hanford Site accident frequencies. For the evaluation, accidents are categorized as resulting in impact, crush, puncture, and fire forces. Risk acceptance criteria are outlined in Part B, Section 3.2, and the dose consequence analyses results are provided in Part B, Section 3.3. Cask failure thresholds are given in Part B, Section 3.4. The analysis of accident release frequencies for associated failure thresholds is documented in Part B, Section 3.5. The accident release frequencies are compared to the risk acceptance criteria determined from the dose consequence analysis to evaluate the acceptability of the risks related to the 327 Building cask shipments.

3.1.1 Summary of Results

Based on the transport of one 327 Building cask per shipment, the dose consequence analysis resulted in a risk acceptance criterion of an annual accident release frequency of less than 10^{-7}. The release frequency and conditional probability analysis showed that the criterion is met for shipments totaling over 16.0 km (10.0 mi) per year. Therefore, 24 shipments per year of 0.40 km (0.25 mi) each easily falls within the range of the acceptable risks as required to meet onsite transportation safety per WHC-CM-2-14.

3.2 RISK ACCEPTANCE CRITERIA

Graded dose limitations for probable, credible, and incredible accident frequencies ensure safety in radioactive material packaging and transportation (Mercado 1994). The dose limitations to the offsite and onsite individual for probable, credible, and incredible accident frequencies are shown in Table B3-1.

Table B3-1. Risk Acceptance Criteria Limits.

Description	Annual frequency	Offsite dose limit* (rem)	Onsite dose limit* (rem)
Incredible	$< 10^{-7}$	None	None
Incredible	10^{-7} to $< 10^{-6}$	25	None
Credible	10^{-6} to 10^{-3}	0.5	5
Probable	10^{-3} to 1	0.01	0.2

*Total effective dose equivalent.

3.3 DOSE CONSEQUENCE ANALYSIS RESULTS

The dose consequence study for the 327 Building casks is presented in Part B, Section 4.0, of this safety evaluation for packaging. The analysis does not take credit for the package, rather it follows International Atomic Energy Agency (IAEA) guidelines and evaluates doses for a release of 100% of the material at risk. The accident results are shown in Table B3-2 for a ground-level release at the worst location with worst-case (0.5%) meteorology. The doses shown in Table B3-2 are the total committed effective dose equivalents (EDE), which are integrated over 50 years.

Table B3-2. Summary of Doses.

Exposure pathway	Offsite receptor (rem)	Onsite worker (rem)
Total effective dose equivalent	>68	>2200

When compared to the risk criteria given in Table B3-1, the potential dose to the offsite receptor requires that the 327 Building casks maintain annual accident release frequencies of less than 10^{-7} probability of occurrence per year. Therefore, the annual accident release frequency is limited to less than 10^{-7} per year.

3.4 PACKAGE FAILURE THRESHOLD ANALYSIS

Accident performance of a package is determined by the probability, given an accident, that a package is subjected to a force more severe than the package failure threshold level for that accident scenario. For the 327 Building casks, the failure thresholds are assumed to be minimal, and the package is assumed to fail if an accident occurs. The failure threshold of the 327 Building casks has been determined for puncture.

- *Impact*: No impact analysis was performed; the 327 Building casks are assumed to fail in the event of an impact.

- *Puncture*: The puncture failure threshold is based on the equivalent steel thickness of the package. The equivalent steel thickness for each of the six casks is at least 6.4 cm (2.5 in.). This evaluation is documented in Part B, Section 7.0.

- *Crush*: No crush analysis was performed; the 327 Building casks are assumed to fail in any accident involving crush.

- *Fire*: No fire failure analysis was performed, therefore the 327 Building casks are assumed to fail any accident involving a fire.

3.5 ACCIDENT RELEASE FREQUENCY ASSESSMENT

3.5.1 Approach

The accident release frequency assessment is based on the assumption that all failure modes from the different forces described as impact, puncture, crush, and fire result in the same level of consequence. The union of the package conditional release probabilities from different scenarios with similar consequences is multiplied by the frequency of truck accidents to arrive at a total annual accident release frequency.

The frequency (F) of a truck accident is the product of the annual number of trips, the number of miles per trip, and the accident rate per mile.

$$F = \frac{\text{number of trips}}{\text{year}} \times \frac{\text{miles}}{\text{trip}} \times \frac{\text{accidents}}{\text{mile}}$$

Hanford Site truck accidents have been compiled in a report using Site-specific data (Green et al. 1996), which gives the accident rate for trucks as 2.0×10^{-7} accidents per mile. For a shipment of radioactive materials that is carried out by trained truck drivers during daylight hours in good road conditions, a reduction factor of 20 can be applied to lower the rate to 1×10^{-8} (H&R 1995) accidents per mile. Appendix B of *Recommended Onsite Transportation Risk Management Methodology* (H&R 1995) summarizes statistics from the U.S. Department of Transportation (DOT) and the studies conducted by Sandia National Laboratory on accident responses of small and large packages. The report recommends reducing truck accident rates by 10 for "safe" truck drivers and another factor of two for shipment of radioactive material. These reduction factors are based on the following logic.

- *Safe truck drivers*: Hanford Site truck drivers have special training. Drivers must complete several driver's education courses, have a valid commercial driver's license with hazardous endorsement, complete specific training for highway route controlled quantities of radioactive material, and complete radiation worker and hazardous materials training. References show that drivers who participate in special safety programs reduce single-vehicle accident rates by up to a factor of 100. The H&R report (H&R 1995) recommends using an overall accident reduction factor of 10.

- *Radioactive material*: An additional factor of two is recommended based on the higher level of training required for drivers of vehicles carrying radioactive material and the higher level of caution that would be expected from drivers of cargos consisting of radioactive material.

After the frequency of accidents is calculated, it is then multiplied by the union of the conditional release probabilities determined in Part B, Section 3.5.2, to arrive at an annual accident release frequency. The annual release frequency is compared to the criteria determined from the dose consequence analysis ($<10^{-7}$).

3.5.2 Accident Release Frequency Analysis

Information for the probability of occurrence and conditional probabilities of failure is taken from *Severities of Transportation Accidents Involving Large Packages* (Dennis et al. 1978), *Severities of Transportation Accidents Volume III - Motor Carriers* (Clarke et al. 1976),[1] and H&R (1995). A simplified generic flow chart, shown in Figure B3-1, has been developed using statistics presented in Clarke et al. (1976) and Dennis et al. (1978). It visually depicts events that may occur as a result of a truck accident on the Hanford Site. Scenarios, such as immersion, that are not pertinent to the shipment of radioactive material on the Hanford Site are not included. Package failure and material release may occur from fire, impact, crush, and puncture, which for purposes of the joint probability calculations are assumed to be independent events.

The probability of an event in the flow chart, given a preceding event, is determined from the studies presented with large and small packages in Clarke et al. (1976) and Dennis et al. (1978). Thus, as can be seen in Figure B3-1, the probability of a fire only given a truck accident is 0.0110, and the probability of an accident resulting in collision or overturn is 0.8935 (Clarke et al. 1976 [p.13]). Trivial accidents are defined only in terms of the cargo and refer to those accidents that do not affect the payload (for example accidents with objects of much lesser mass).

The crush force in the flow chart represents static crush. For large packages inertial crush falls under the category of an impact force, and impact failure thresholds are accordingly evaluated for either impact or inertial crush failure, whichever is the limiting value. The conditional probability of static crush given a collision or overturn accident is found in Dennis et al. (1978 [p. II-25]) as 0.05. This means that 1 in 20 collision or overturn accidents results in static crush to the package. Use of the 0.05 value is recommended in Dennis et al. (1978) even though the study states that accident statistics indicate a lower rate would be more representative of accident conditions.

The impact environment may result in puncture or impact failure for large packages. Dennis et al. (1978) cites a value of 0.8020 for the probability of an impact or inertial crush force given an accident. Accordingly, the probability of an impact force occurring given a collision or overturn is calculated to be 0.8976 (0.8020/0.8935). In a similar manner, the conditional probability of fire given a collision or overturn is calculated from the fire frequency per accident of 1.6% (Dennis et al. 1978 [p. II-15]) and the value for the fire-only scenario of 0.0110. It is worth noting that the statistics in Dennis et al. (1978) do not discriminate between fires that affect cargo and fires that do not affect cargo. Therefore, some overconservatism may result from the assumption that all fires affect the cargo.

3.5.2.1 Conditional Release Probabilities. Conditional release probabilities for crush are either 1.0 for failure or 0 for no failure. In an accident involving crush, for example, failure occurs if the static crush failure threshold for the package is less than the weight of the truck trailer. No other static crush force will occur on the Hanford Site. For the 327 Building casks, the conditional release probability due to crush-induced failure (PCF) is 1.0 because the casks are assumed to fail in any accident involving crush forces.

The conditional probability of release from failure from fire (PFF) is determined from an H&R report (H&R 1995), which incorporates Hanford Site information for emergency response time and fire duration. The value represents the probability that the fire duration is greater than the length of time determined to be the failure point for the package. For the 327 Building casks, PFF is equal to 1.0 because the package is assumed to fail any fire.

The conditional probability of release from puncture given an impact event (PPF) represents the probability that an impact event will result in a puncture force large enough to penetrate and fail the

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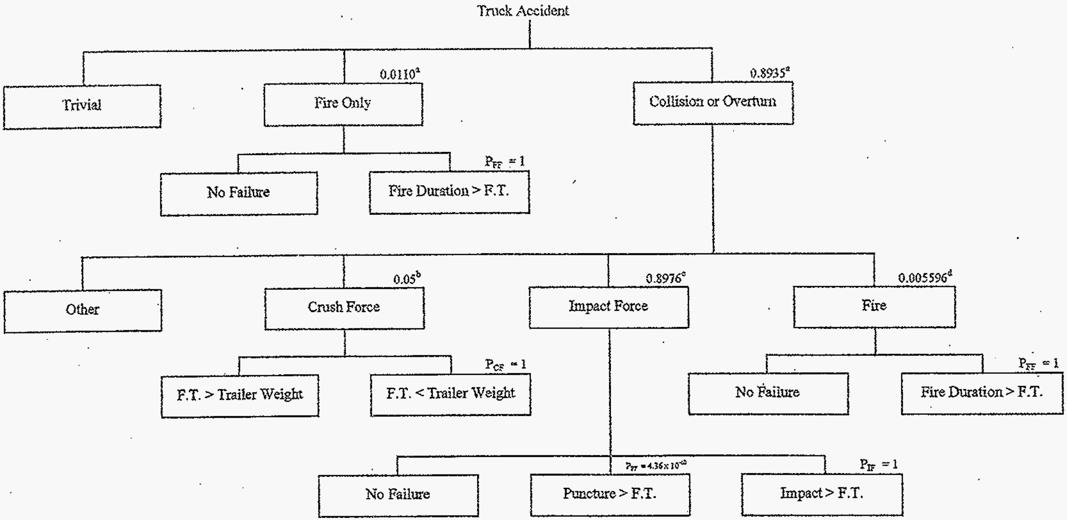
[1] Although the Dennis and Clarke reports were prepared in the 1970s, they still represent the most comprehensive and consistent reports available on accident environments encountered by radioactive materials packages.

Figure B3-1. Flow Chart for Hanford Site Large Package Truck Accidents.



F.T. = Failure Threshold
P_{FF} = Probability of Fire Failure
P_{PF} = Probability of Puncture Failure
P_{IF} = Probability of Impact Failure
P_{CF} = Probability of Crush Failure

a) Clarke, 13
b) Dennis, II-25
c) P(impact given an accident) = 0.8976 x 0.8935 = 0.8020 from Dennis, II-28 and B-8
d) Fire Frequency/accident = 1.6% (Dennis II-15); {0.0110 + 0.8935(0.005596)} = 0.016

5/20/97

fig327

equivalent steel thickness of the package. The PPF values are found in Dennis et al. (1978 [p. II-35]). The 327 Building casks have an equivalent steel thickness of 6.6 cm (2.6 in), which is rounded down to 6.4 cm (2.5 in.) for a PPF value of 4.36×10^{-10}.

The conditional probability of release from impact forces given an impact event (PIF) represents the probability that the package will be subjected to an impact resulting in a velocity change greater than that which could fail the package. As previously stated, inertial crush is included in this category. The values for the impact conditional release probabilities are found in Dennis et al. (1978 [p. II-23]). For the 327 Building casks, the PIF is conservatively assumed to be 1.0.

Table B3-3. Failure Thresholds and Conditional Release Probabilities.

Force type	Failure threshold	Conditional release probability
Crush	Fails crush	1.0
Puncture	6.4 cm (2.5 in.)	4.36×10^{-10}
Fire	Fails any fire	1.0
Impact	Fails any impact	1.0

3.5.2.2 Joint Probabilities. Conditional release probabilities and failure thresholds are shown in Table B3-3. The joint probability is calculated by taking the union of events (McCormick 1981). The equation represents the sum of the probabilities of independent events while the subtracted terms eliminate double counting arising from the overlap caused by the intersection of the events. The general equation is given as:

$$P(A_1+A_2+\ldots+A_N)=\sum_{n=1}^{N} P(A_n)-\sum_{n=1}^{N-1}\sum_{m=n+1}^{N} P(A_nA_m)+-\ldots.+(-1)^{N-1}P(A_1A_2\ldots A_N).$$

where

P(f|a) = the probability of fire given that an accident has occurred
P(fc|a) = the probability of fire and crush given that an accident has occurred

and

P(FTE f|f) = the probability that the failure threshold is exceeded by fire given that a fire has occurred

then the above equation can be expanded and written as:

P = P(f|a) P(FTE f|f) + P(c|a) P(FTE c|c) + P(I|a) P(FTE I|I) + P(p|a) P(FTE p|p) - P(fc|a) P(FTE f|f) P(FTE c|c) - P(fi|a) P(FTE f|f) P(FTE I|I) - . . .

When substituted in the above equation, the values from the flow chart in Figure B3-1 and the conditional probabilities from Table B3-3 yield a total conditional release probability of 0.978.

3.6 EVALUATION AND CONCLUSION

The total conditional release probability of 0.978 is multiplied by the frequency (F) to arrive at an annual accident release frequency. The annual accident release frequency for 24 shipments of 327 Building casks is 5.9×10^{-8}. This value is less than the 1×10^{-7} required. In fact, the 327 Building family of casks can be shipped for up to 16.0 km (10.0 mi) per year and still be less than the criterion.

3.7 REFERENCES

Clarke, R. K., J. T. Foley, W. F. Hartman, and D. W. Larson, 1976, *Severities of Transportation Accidents, Volume III - Motor Carriers,* SLA-74-0001, Sandia National Laboratories, Albuquerque, New Mexico.

Dennis, A. W., J. T. Foley, W. F. Hartman, and D. W. Larson, 1978, *Severities of Transportation Accidents Involving Large Packages*, SAND77-0001, Sandia National Laboratories, Albuquerque, New Mexico.

DOE, 1994, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities,* DOE-HDBK-3010-94, U.S. Department of Energy, Washington, D.C.

Green, J. R., B. D. Flanagan, and H. W. Harris, 1996, *Hanford Site Truck Accident Rate, 1990-1995,* WHC-SD-TP-RPT-021, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

H&R, 1995, *Recommended Onsite Transportation Risk Management Methodology*, H&R522-1, H&R Technical Associates, Inc., Oak Ridge, Tennessee.

McCormick, N. J., 1981, *Reliability and Risk Analysis: Methods and Nuclear Power Applications,* Academic Press, San Diego, California.

Mercado, J. E., 1994, *Report on Equivalent Safety for Transportation and Packaging of Radioactive Materials*, WHC-SD-TP-RPT-001, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

WHC-CM-2-14, *Hazardous Material Packaging and Shipping*, Westinghouse Hanford Company, Richland, Washington.

3.8 APPENDIX: CHECKLIST FOR REVIEW

4.0 CONTAINMENT EVALUATION

4.1 INTRODUCTION

The radioactive contents within the SERF Cask are contained inside an inner container. That inner container can be either a stainless steel pin tube 1.3 cm (0.50 in.) or 1.9 cm (0.75 in.) in diameter, a standard one-gallon paint can, or a DOT Specification 2R container per 49 CFR 178.360. The stainless steel tubes have a working rating of 20.68 MPa (3,000 psi) and have Swagelok[2] fittings on one end. Containment credit is not taken for the boundary provided by the SERF Cask itself although it does retain the inner containers.

The SERF Cask has been used for shipments within the 300 Area for approximately 20 years. Prior approval for use of the cask to ship radioactive material was given in MG 137, *Hazardous Materials Packaging and Shipping Manual,* Rev. 1 (HEDL 1981). Although the SERF Cask does not provide containment in the traditional sense, the contents are limited to dry, dispersible and nondispersible items packaged in inner containers. The cask provides shielding and, in conjunction with the inner container or plastic wrapping, prevents the release of the contents under normal transport conditions evaluated in Part B, Section 7.0.

Another consideration for the safety of the shipment is that each shipment will travel approximately 0.40 km (0.25 mi) within the 300 Area. The cask will be used for a limited time to facilitate the shutdown of the 327 Building. This SEP is valid until October 1, 1999. The risk evaluation has shown that accidents leading to a release are incredible with the annual travel limitation of 16.0 km (10.0 mi) total for the entire family of 327 Building casks.

4.2 CONTAINMENT SOURCE SPECIFICATION

The authorized radioactive contents of the SERF Cask are described in Part B, Section 2.1. For conservatism, the containment analysis assumes that the inner container consists of the double-wrapped, 10-mil plastic bags.

4.3 NORMAL TRANSFER CONDITIONS

4.3.1 Conditions To Be Evaluated

For normal transfer conditions, containment must be demonstrated during the following events.

4.3.1.1 Water Spray. The package shall be demonstrated to maintain containment through a water spray that simulates exposure to rainfall approximately 1.5 cm (0.6 in.) per hour for at least one hour.

4.3.1.2 Reduced External Pressure. The package shall be capable of withstanding a reduced external pressure of 95.2 kPa (13.8 psi) absolute.

4.3.1.3 Increased External Pressure. The package shall be capable of withstanding an increased external pressure of 102.5 kPa (14.9 psi) absolute.

4.3.1.4 Temperature. The package shall be capable of being transported over a temperature range from -33 °C to 46 °C (-27 °F to 115 °F).

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[2] Swagelok is a trademark of the Crawford Fitting Company.

4.3.1.5 Rough Transport. The package shall be evaluated to demonstrate containment subsequent to rough transport shock loads of 3.5*g* vertical to the plane of travel and 2.3 *g* in the direction of travel.

4.3.1.6 Penetration. The package shall maintain containment following the impacting force of a bar 3.18 cm (1.25 in.) in diameter with a hemispherical end weight of 6.0 kg (13.2 lb), dropped from a height of 1.0 m (3.3 ft) onto that part of the container where maximum damage is expected to occur.

4.3.1.7 Vibration. The package shall maintain containment when subjected to normal transport vibration loadings.

4.3.2 Containment Acceptance Criteria

The acceptance criteria for the SERF Cask shall be that the package retains its contents throughout normal transport conditions.

4.4 ACCIDENT CONDITIONS

4.4.1 Conditions To Be Evaluated

Accident conditions are evaluated for the SERF Cask by radiological risk and dose consequence analyses. The radiological risk evaluation is given in Part B, Section 3.0, of this SEP. The dose consequence and associated transportation hazard index are given in Part B, Section 4.6.

4.5 CONTAINMENT EVALUATION AND CONCLUSIONS

4.5.1 Normal Transport Conditions

The SERF Cask has been demonstrated to provide the structural integrity necessary to transport its payload. The containment boundary is maintained throughout all normal transport conditions.

4.5.1.1 Water Spray. The SERF Cask is a right circular cylinder constructed of steel. There are no crevices or gaps where water could be retained.

4.5.1.2 Reduced External Pressure. Part B, Section 7.3.4, demonstrates the package can withstand a reduced external pressure of 94.5 kPa (13.7 psi) absolute.

4.5.1.3 Increased External Pressure. Part B, Section 7.3.4, demonstrates the package can withstand an increased external pressure of 108 kPa (15.7 psi) absolute.

4.5.1.4 Temperature. Part B, Section 7.3.3, demonstrates the package can be transported over a temperature range from -33 °C to 46 °C (-27 °F to 115 °F).

4.5.1.5 Rough Transport. Part B, Section 7.3.8, demonstrates the package maintains containment subsequent to rough transport shock loads of 3.5*g* vertical to the plane of travel and 2.3*g* in the direction of travel.

4.5.1.6 Penetration. Part B, Section 7.3.9, demonstrates the package maintains containment during the penetration event.

4.5.1.7 Vibration. Part B, Section 7.3.5, demonstrates the package maintains containment when subjected to normal transport vibration loadings.

4.5.2 Accident Conditions

Based on the radiological risk evaluation in Part B, Section 3.0, and the dose consequence evaluation given in Part B, Section 4.6, the SERF Cask, in conjunction with the other 327 Building casks, can be transported a maximum of 16.0 km (10.0 mi) per year while still remaining within the acceptable limits for onsite and offsite receptor doses.

4.6 SUMMARY OF DOSE CONSEQUENCE RESULTS

This engineering analysis documents the dose consequence calculations used to support the Transportation Hazard Index (THI) evaluation for the 327 Building casks. Three casks are used to transfer radioactive materials among several buildings in the 300 Area. The casks used for these transfers are the Waste Cask, the SERF Cask, and the PRTR Graphite Cask. The authorized contents for each of the casks was reviewed, and the PRTR Graphite Cask was found to have the lowest allowable radioactive inventory. Dose consequence calculations for the PRTR Graphite Cask require this cask to meet THI 1 requirements. The other 327 Building Casks (Waste Cask and SERF Cask) have radioactive inventories that exceed that for the PRTR Cask; therefore, the dose consequences for the other two casks will be greater than that for the PRTR Cask. Because the PRTR Cask has to meet the highest level of requirements (i.e., those associated with a THI of 1), the other casks will also have to meet the requirements for a THI of 1, and no additional analysis is required to demonstrate this.

Table B4-1 shows the dose consequence results from each exposure pathway for the maximum authorized contents for the PRTR Graphite Cask. The table also shows the dose to each receptor, which is obtained by summing the dose contributions from each pathway. Because the offsite worker dose is greater than 25 rem, the packaging must be designed to THI 1 requirements. The criteria for a THI of 1, as stated in WHC-CM-2-14:

> "THI-1: This represents the highest level of hazard from the contents. A packaging system assigned this level transports material that has the potential of causing a dose consequence, to an individual, in excess of 25 rem at the Hanford Site boundary if fully released."

Table B4-1. Summary of Doses (rem) for the Plutonium Recycle Test Reactor Graphite Cask.

Exposure pathway	Hanford Site worker at 3 m	Public receptor*
External photon dose	2.8	NA
External dose from β-particles	5.0	NA
Inhalation and submersion from the airborne transport pathway	2.0 E+03	6.8 E+01
Total effective dose equivalent	2.0 E+03	6.8 E+01

Note: 100 rem = 1 sievert (Sv).
*This receptor is located 100 m N of the 300 Area.

4.6.1 Introduction and Overview

Three casks are used to transport radioactive materials among several buildings in the 300 Area. The casks used for these transfers are the Radioactive Waste Disposal Cask, the SERF

Cask, and the PRTR Graphite Cask. The radioactive materials most frequently transported among buildings are irradiated Fast Flux Test Facility and Power Reactor and Nuclear Fuel Development Corporation fuel, spent N Reactor fuel, mixed oxide, metal oxide, activated structural materials from reactors, and cesium chloride capsules.

An estimate of the dose consequences for various exposure pathways is necessary to determine the THI for the 327 Building casks. Part B, Section 4.6.2, discusses the general methodology used to perform the dose consequence calculations. Part B, Section 4.6.3, addresses the source term, and Part B, Sections 4.6.4 through 4.6.9, summarize the results for various exposure pathways. The analysis assumes the casks will only be transported within the 300 Area.

4.6.2 Dose Consequence Analysis Methodology

IAEA (1990) defines a standardized approach for evaluating transportation packaging requirements, called the Q-system. The Q-system methods, as outlined in IAEA (1990), have been incorporated into the document, *Report on Equivalent Safety for Transportation and Packaging of Radioactive Materials* (Mercado 1994). This document (Mercado 1994) is used to demonstrate that onsite shipments meet onsite transportation safety requirements per WHC-CM-2-14.

In the Q-system, the following five exposure pathways are considered: (1) external exposure to photons, (2) external exposure to β-particles, (3) inhalation, (4) skin contamination and ingestion, and (5) submersion in a cloud of gaseous isotopes. In special cases, such as α-particle or neutron emitters, other exposure routes are considered. In some cases a pathway will be judged to be small with respect to the others, and consideration will be minimal. Modifications to the IAEA scenarios are incorporated to more closely describe the particular conditions of the shipment. Detailed calculations for the postulated accident are performed whenever possible. However, in some cases, the IAEA guide's (IAEA 1990) worst-case rules-of-thumb are used.

The Q-system was developed as an all-encompassing generalized methodology using only the isotope as the defining variable. In this report, the specifics of the package are considered. Some of the dose pathways may be considered incredible (frequency $<10^{-6}$/yr), and although these pathways are covered in the IAEA guide, they are disregarded in the analysis.

In the IAEA system, the Q-values that are calculated are the radionuclide activities corresponding to each exposure route that causes the individual to receive the effective dose equivalent limit. The minimum Q-values define the A_2 values for the shipped materials. In the case of nondispersible materials (limited by the A_1 values), only the first two Q-values (based on exposure to external photon and external beta particles) are used. Note that for all radiation except neutrons, protons, and heavier charged particles (including α-particles), 1 gray (Gy) = 1 sievert (Sv), and 1 rad = 1 rem.

There are two receptors of interest in the Q-system. They are the Hanford Site worker and the public receptor. The Hanford Site worker is assumed to be located about 3 m from the package. The public receptor is assumed to be located at the nearest point of public access.

4.6.3 Source Term

The authorized contents for the PRTR Graphite Cask are shown in Table B4-2. The external dose due to gamma exposure was calculated assuming the mixed fission products (MFP) consist of ^{137}Cs, and the mixed activation products (MAP) consist of ^{60}Co, which produce the highest gamma dose rates. The external dose due to beta particle exposure was calculated assuming the MFP consists of ^{90}Sr/^{90}Y and the MAP consists of ^{59}Fe, which produce the highest beta dose rates. The inhalation dose was calculated assuming the MFP consists of ^{90}Sr, the MAP consists of ^{59}Fe, and the fissile material consists of 175 g (10.85 Ci) of ^{239}Pu, which produce the highest inhalation dose. The use of 175 g of ^{239}Pu bounds the 2-Ci limit for α emitters for the PRTR Graphite Cask. Note that the alpha emitters and fissile material have a negligible impact on the external gamma and beta dose rates due to the high content of ^{137}Cs, ^{90}Sr, and ^{60}Co assumed in the analyses.

Table B4-2. Plutonium Recycle Test Reactor Graphite Casks Radioactive Inventory.

Radioactive material	Cask (Ci)
Fissile material and α emitters	10.85
Mixed fission products[a]	375
Mixed activation products[b]	5

[a] Mixed fission products; e.g., ^{90}Sr, ^{137}Cs, etc.
[b] Mixed activation products; e.g., ^{60}Co, ^{54}Mn, ^{59}Fe.

4.6.4 External Dose Due to Photon (Gamma) Exposure

The IAEA scenario assumes that a person is exposed to a damaged transport package following an accident. The shielding of the package is assumed to be completely lost in the accident. This analysis will be done assuming a person remains 3 m from the source for a period of 15 minutes.

The computer code ISO-PC (Rittmann 1995) was used to calculate the dose rate 3 m from the source. The fluence-to-dose conversion factors used were the anterior-to-posterior irradiation pattern as outlined in American National Standards Institute (ANSI) standard ANSI/ANS-6.1.1-1991 (ANS 1991).

It was conservatively assumed that the MFP (375-Ci limit) consisted of ^{137}Cs and the MAP (5-Ci limit) consisted of ^{60}Co for this analysis. The other radioactive materials (α emitters, others, and fissile materials) will have a negligible contribution to the gamma dose rate compared to that from ^{137}Cs and ^{60}Co.

The PRTR Graphite Cask has an internal diameter of 7.94 cm (3.125 in.) and a length of 62 cm (24.5 in.). The source term was assumed to be homogeneously distributed throughout the cavity of the cask.

The payload for the PRTR Graphite Cask consists of mainly activated metal structural materials with a density of 7.86 g/cm^3. For this analysis, the source was taken to be iron with a density of 2.0 g/cm^3, which is about one-fourth the density of steel. The lower source density is conservative for this analysis because it results in higher dose rates due to reduced self-shielding and attenuation effects. Note that the results are not very sensitive to the selection of the source material.

The resulting dose rate from ISO-PC is 11 rem/h (0.11 Sv/h) at 3 m from the unshielded source. Therefore, the maximum total external gamma EDE for the Hanford Site worker is 2.8 rem (0.028 Sv) for a 15-minute exposure period. The ISO-PC input deck is included as Part B, Section 4.8.1.

4.6.5 External Dose Due to β-Particle Emitters

Because of the limited range of β-particles relative to that of photons, a shielding factor is used by the IAEA to account for residual shielding from material such as package debris. Except for this factor, no effort is made to account for either self-shielding or shielding from an accurate model of the damaged package. Shielding and dose rate factors are graphed in the IAEA Safety Guide No. 7

(IAEA 1990) as a function of the maximum energy of the β-particle. The IAEA beta dose rate calculation methods are based on an individual located 1 m from the unshielded source.

This analysis assumes an individual remains at a distance of 3 m from the source for a 15-minute exposure period. A factor will be applied to the dose rates calculated using the IAEA method to account for the difference between the 1-m distance assumed in developing the shielding factors and the 3-m distance in this analysis. This factor was conservatively taken to be 0.333 [(1 m/3 m)] since the dose rate falls off between $1/r^2$ and $1/r$, where r is the distance from the source. This also conservatively ignores any attenuation of the beta particles over the 3-m distance.

Table B4-3 shows the β-particle dose calculations for the inventory in Table B4-2, assuming the MFP consists of ^{90}Sr/^{90}Y and the MAP consists of ^{59}Fe. Note that ^{239}Pu and ^{235}U are not beta emitters. The total β-particle dose rate to the skin for an individual located 3 m from the source is 2.0×10^3 rem/h (2.0×10^1 Sv/h). This results in a β-particle dose of 5.0×10^2 rem (5.0 Sv) to the skin for a 15-minute exposure. Because the tissue weighting factor for the skin is 0.01 (ICRP 1991), the whole body EDE is then 5.0 rem (0.05 Sv). Note that 98.9% of the β-particle dose is due to the high-energy (2.28 MeV) β-particle emitted by ^{90}Y.

Table B4-3. β-Particle Dose Rate for Beta Emitters Contributing > 0.01% to the Total Dose.

Isotope	Activity (Ci)	Activity (Bq)	Branching ratio	E_{max} (MeV)	Dose rate factor[a]	Shielding factor[b]	Dose rate (rem/h)[c]	% Dose
^{90}Sr	3.75 E+02	1.39 E+13	1	0.54600	1.8 E-04	100	2.25 E+01	1.15
^{90}Y	3.75 E+02	1.39 E+13	0.99989	2.28390	3.1 E-04	2	1.94 E+03	98.85
Totals for beta emitters contributing > 0.01%							1.96 E+03	100.00
Totals for all beta emitters							1.96 E+03	100.00

[a] Dose rate factor in units of Gy/h or Sv/h for a 1-m Ci source from IAEA (1990).
[b] Shielding factor from IAEA (1990).
[c] Note that a factor of 0.333 is applied to the dose rates to account for a source-to-receptor distance of 3 m for this analysis, versus the 1-m distance assumed in the development of the dose rate factors from IAEA (1990).

IAEA, 1990, *Explanatory Material for the IAEA Regulations for the Safe Transport of Radioactive Material*, Safety Series No. 7, Second Edition (As Amended 1990), International Atomic Energy Agency, Vienna, Austria.

4.6.6 Inhalation and Ingestion Dose

Radioactive material may be inhaled following an accident due to resuspension or volatization of radioactive material released from the package. This section addresses the dose received by workers and the public due to exposure to airborne radioactivity during a postulated accident event.

4.6.6.1 Selection of Airborne Release Fraction. The U.S. Department of Energy (DOE) handbook, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities* (DOE 1994), was reviewed to identify applicable airborne release fractions (ARF) and respirable fractions (RF) for the authorized contents of the 327 Building casks. The PRTR Graphite Cask is only authorized to transport irradiated structural materials or encapsulated solid materials. DOE (1994) identifies a bounding ARF x RF of 1×10^{-3} for contaminated noncombustible materials not undergoing brittle fracture during shock vibration. This is a conservative airborne release fraction for the cask contents considering that the authorized contents specify that "All materials shall be enclosed in an inner container. The type of inner containment shall be sufficient to contain the item and prevent spread of removable contamination." Although the casks have not been analyzed to withstand hypothetical accident conditions, the inner container and the thick lead walls of the casks will likely prevent any catastrophic loss of the cask contents. The accident scenario envisioned for this analysis is an energetic event causing a loss of the cask closure lid, which exposes the cask contents and results in the radioactivity becoming airborne. No credit is taken for any containment provided by the inner container or the cask.

The ARF x RF of 1×10^{-3} is applied to the material at risk, which is assumed to be the entire cask radioactive inventory, to obtain the quantity of radioactive material that is made airborne for the postulated accident scenario. As mentioned in Part B, Section 4.6.3, the inhalation dose for the PRTR Graphite Cask is calculated assuming the MFP consists of ^{90}Sr, the MAP consists of ^{59}Fe, and the fissile material consists of 175 g (10.85 Ci) of ^{239}Pu, which produce the highest inhalation dose. The accident release quantities are listed in Table B4-4.

Table B4-4. Accident Airborne Release Quantities, Ci.

Nuclide	Plutonium Recycle Test Reactor Graphite Cask
^{90}Sr/^{90}Y	0.375
^{59}Fe	0.005
^{239}Pu	0.011

4.6.6.1.1 Discussion of Integrated Normalized Air Concentration Value (χ/Q'). After the radioactive material becomes airborne, it is transported downwind and inhaled by onsite workers or the public. The concentration of this material is reduced, or diluted, as it is being transported due to atmospheric mixing and turbulence. χ/Q' (s/m^3) is used to characterize the dilution of the airborne contaminants during atmospheric transport and dispersion. It is equal to the time-integrated normalized air concentration at the receptor. χ/Q' is a function of the atmospheric conditions (i.e., wind speed, stability class) and the distance to the receptor.

Bounding χ/Q' values are generated consistent with the methods described in *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,* Regulatory Guide 1.145 (NRC 1982). Because atmospheric conditions fluctuate, a bounding atmospheric condition is determined to be that condition that causes a downwind concentration of airborne contaminants that is exceeded only a small fraction of time because of weather fluctuations. Regulatory Guide 1.145 (NRC 1982) defines this fraction of exceedance as 0.5% for each sector or 5% for the overall Hanford Site. The Hanford Site is broken up into 16 sectors that represent 16 compass directions; i.e., S, SSW, SW, . . . , ESE, SE, SSE. χ/Q' values are generated for weather conditions that result in downwind concentrations exceeded only 0.5% of the time in the maximum sector or 5% of the time for the overall Site. These χ/Q' values are also referred to as 99.5% maximum sector and 95% overall Site χ/Q' values. The greater of these two values is called the bounding χ/Q' value and is used to assess the dose consequences for accident scenarios. The bounding χ/Q' value represents minimum dispersing conditions that result in maximum downwind concentrations; i.e., concentrations exceeded only a very small fraction of the time. This χ/Q' value will therefore result in very conservative estimates of accident consequences.

The χ/Q' values in this report were generated using the GXQ computer program, Version 3.1C (Hey 1993a, 1993b). The meteorological data used by GXQ are in the form of joint frequency tables. The joint frequency data are the most recent data available; they are nine-year-averaged data (1983-1991) from the Hanford Site meteorology towers located in the 300 Area. The χ/Q' values are generated using the methods described in Regulatory Guide 1.145 (NRC 1982) for a ground release with no credit taken for plume rise, plume meander, plume depletion, or any other models. This is conservative because all of these models reduce the airborne concentration at the downwind receptor locations.

Although we are interested in the dose to a Hanford Site worker at 3 m, the dose to an onsite receptor located 100 m from the release point is calculated using the worst-case χ/Q' value at 100 m. This dose is then multiplied by a factor of 30 to obtain the dose to the Hanford Site worker at 1 m in accordance with IAEA (1990). This approach is taken because the Gaussian equation, along with the parameters used to calculate the χ/Q' values, are only valid for distances of 100 m or greater. Although this analysis assumes the transport worker remains 3 m from the package, the inhalation portion of the transport worker dose is conservatively taken to be that calculated using the IAEA method for a worker located 1 m from the package.

The 327 Building casks will be transported within the 300 Area. The maximum χ/Q' value for an onsite receptor is 4.21×10^{-2} s/m^3 and occurs for an individual located 100 m N of the release point in the 300 Area.

The 300 Area is not a public exclusion area. Even though the roads may be closed during movement of the 327 Building casks, members of the public may be in the area. Therefore, it is conservatively assumed for this analysis that the public receptor is located 100 m from the release point in any compass direction. The maximum onsite and public receptor χ/Q' value will therefore be the same, i.e., the maximum public receptor χ/Q' *value is* 4.21×10^{-2} *s/m*^{3}*. The GXQ input file for the* maximum χ/Q' case is listed in Part B, Section 4.8.2. The title of the joint frequency file used by GXQ is 300 AREA - 10 M - Pasquill A - G (1983 - 1991 Average).

4.6.6.1.2 Inhalation and Submersion Dose Calculations. Because the GENII computer code Version 1.485 (Napier 1988) is the Site standard computer code for environmental release dose calculations, it was used to calculate the inhalation and submersion dose for the maximum onsite and public receptors. The airborne release quantities used in GENII are shown in Table B4-4. An example GENII input deck is listed in Part B, Section 4.8.3. The Worst Case Solubility class library was used, which is the most conservative library. The GENII libraries used were as follows:

- GENII Default Parameter Values (28-Mar-90 RAP)
- Radionuclide Library - Times<100 years (23-July-93 PDR)
- External Dose Factors for GENII in person Sv/yr per Bq/n (8-May-90)
- *Worst Case Solubilities, Yearly Dose Increments* (23-Jul-93 PDR).

The EDE from GENII for the inhalation and submersion pathways is 1.2×10^2 rem (1.2 Sv) for the maximum onsite receptor at 100 m N of the 300 Area. The inhalation dose contribution to the EDE is based on a 50-year dose commitment period. The maximum χ/Q' value from GENII was 7.4×10^{-2} s/m^3 for the maximum onsite receptor. The dose rates calculated by *GENII are proportional* to the χ/Q' values. The GXQ code calculates the 99.5% maximum sector and 95% overall Site χ/Q' values consistent with Regulatory Guide 1.145 (NRC 1982) methods, while GENII is inconsistent with Regulatory Guide 1.145 methods. As mentioned in the previous section, the maximum onsite receptor χ/Q' value from GXQ is 4.21×10^{-02} s/m^3. Therefore, the EDE for the inhalation and submersion pathways is 6.8×10^1 rem (6.8×10^{-1} Sv) for the maximum onsite receptor at 100 m using the GXQ χ/Q' value. This value was obtained by multiplying the GENII dose rate by the ratio of the GXQ χ/Q' value to the GENII χ/Q' value. Therefore the maximum public receptor dose is 6.8×10^1 rem (6.8×10^{-1} Sv).

To compensate for the fact that the onsite dose is calculated at a source-to-receptor distance of 100 m, this dose is multiplied by a factor of 30 to obtain the dose to the transport worker at 1 m in accordance with IAEA (1990). Although this analysis assumes the transport worker remains 3 m from the package, the inhalation portion of the transport worker dose is conservatively taken to be that calculated using the IAEA method for a worker located 1 m from the package. This results in an EDE of 2.0×10^3 rem (20.0 Sv) for the Hanford Site worker. Table B4-5 shows the doses for the postulated accident scenario.

Table B4-5. Inhalation and Submersion Dose (rem).

	Hanford worker (at 3 m)	Public receptor*
Effective dose equivalent	2.0 E+03	6.8 E+01

Note: 100 rem = 1 Sv.
*This receptor is located 100 m N of the 300 Area.

4.6.6.1.3 Ingestion and Ground Shine Dose. The other potential internal exposure pathway for the public receptor is the ingestion pathway. Exposure through the ingestion pathway occurs when radioactive materials that have been deposited offsite during passage of the plume are ingested either by eating crops grown in, or animals raised on, contaminated soil or through drinking contaminated water. There are DOE; DOE, Richland Operations Office; state; and federal programs in place to prevent ingestion of contaminated food in the event of an accident (RL 1994, WSDOH 1993, WS 1994, EPA 1992). The primary determinant of exposure from the ingestion pathway is the effectiveness of public health measures (i.e., interdiction) rather than the severity of the accident itself. The ingestion pathway, if it occurs, is a slow-to-develop pathway and is not considered an immediate threat to an exposed population in the same sense as airborne plume exposures.

The ground shine pathway is an additional potential external exposure pathway for the public receptor. Ground shine refers to the external dose received by a person standing on ground contaminated by radioactive materials deposited during passage of the airborne radioactive plume. Similar to the ingestion pathway, the primary determinant of exposure from the ground shine pathway is the effectiveness of public health measures (i.e., interdiction) rather than the severity of the accident itself. The ground shine pathway is a slow-to-develop pathway and is not considered an immediate threat to an exposed population in the same sense as airborne plume exposures.

Because of the large radioactive inventory contained in the casks, it is argued that in the event of an accident scenario that results in the release of a large portion of the inventory, interdictive measures (RL 1994, WSDOH 1993, WS 1994, EPA 1992) would be taken to prevent ingestion of contaminated food and exposure through the ground shine pathway. Therefore, the ingestion and ground shine pathway doses were not calculated in this report.

4.6.7 Skin Contamination and Ingestion Dose

In the IAEA guide (IAEA 1990), it is assumed that 1% of the package contents are spread over an area of 1 m^2 and handling of debris results in contamination of the hands to 10% of this level. It is further assumed that the worker is not wearing gloves but that he recognizes the possibility of contamination and washes his hands within five hours. The effective dose equivalent to the skin received by the individual is estimated from a graph provided in the IAEA guide.

The IAEA scenario for the uptake of activity due to ingestion of the material assumes that the person ingests all of the contamination from 10 cm^2 of skin over a 24-hour period. Because the dose per unit uptake via inhalation is generally the same order or larger than that via ingestion, the inhalation pathway will normally be limiting for internal contamination due to β-ray emitters. In particular, if the skin contamination dose is much larger than the inhalation dose, the ingestion pathway is not considered.

Both these pathways are ordinarily neglected when calculating the dose consequences from an onsite transportation accident. The transportation workers are trained in the appropriate response to protect themselves from experiencing unnecessary radiation exposure, including preventing skin contamination and ingestion.

4.6.8 Submersion Dose Due to Gaseous Vapor

This exposure pathway is caused by submersion in a cloud of gaseous isotopes that are not taken into the body. A rapid release of 100% of the package contents is assumed. The IAEA guide (IAEA 1990) concentrates entirely on releases within confined structures. No guidance is given for outside releases.

There are no gaseous vapors present in the cask; therefore, this exposure pathway is not applicable. Gas generation from broken fuel pins is negligible as the gas is released at the time the pin is broken.

4.6.9 Special Considerations

Alpha particle emitters are not of significance in the material considered in this report. The alpha particle emitters are of a low concentration, and their effect will be through the mechanism of inhalation that has been considered separately. Therefore, they are not addressed in this report. The quantity of radon present in the fuel is insignificant; therefore, radon is not addressed in this report.

The fuel (e.g., plutonium) contained in the casks emits neutrons through (α,n) and spontaneous fission reactions. These neutron emitters will contribute to the dose received by the Hanford Site worker, but will have a negligible impact on the public receptor. A conservative estimate of the neutron dose was made using the method described in Nelson (1996). The results indicate that the neutron dose contribution is negligible compared to the gamma dose due to the large MFP and MAP inventory. Therefore, the neutron dose was not calculated separately in this report.

Bremsstrahlung has been included in the consideration of photon effects, and the effects of short-lived daughter products have been included in all of the calculations. Where these isotopes are significant, they are assumed to be in equilibrium with their longer-lived parent isotopes.

4.6.10 Total Dose

Table B4-1 in Part B, Section 4.6, shows the dose from each exposure pathway.

4.7 REFERENCES

49 CFR 178, 1997, "Specifications for Packagings," *Code of Federal Regulations*, as amended.

ANS, 1991, *Neutron and Gamma-Ray Fluence-to-Dose Factors,* ANSI/ANS-6.1.1-1991, American Nuclear Society, La Grange Park, Illinois.

DOE, 1994, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities,* DOE-HDBK-3010-94, U.S. Department of Energy, Washington, D.C.

DOE, 1992, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports,* DOE-STD-1027-92, U.S. Department of Energy, Washington, D.C.

DOE, 1988, *Radiation Protection for Occupational Workers,* DOE Order 5480.11, U.S. Department of Energy, Washington, D.C.

EPA, 1992, *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents,* U.S. Environmental Protection Agency, Washington, D.C.

HEDL, 1981, *Hazardous Materials Packaging and Shipping Manual*, MG 137, Rev. 1, Hanford Engineering Development Laboratory, Richland, Washington.

Hey, B. E., 1993a, *GXQ Program Users' Guide*, WHC-SD-GN-SWD-30002, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

Hey, B. E., 1993b, *GXQ Program Verification and Validation*, WHC-SD-GN-SWD-30003, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

IAEA, 1990, *Explanatory Material for the IAEA Regulations for the Safe Transport of Radioactive Material,* Safety Series No. 7, Second Edition (As Amended 1990), International Atomic Energy Agency, Vienna.

ICRP, 1991, *International Commission on Radiological Protection, Annals of the ICRP*, Publication 60, 1991, International Commission on Radiological Protection, New York, New York.

Mercado, J. E., 1994, *Report on Equivalent Safety for Transportation and Packaging of Radioactive Materials,* WHC-SD-TP-RPT-001, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

Napier, B. A., et al., December 1988, *GENII - The Hanford Environmental Radiation Dosimetry Software System*, PNL-6584, Vol. 1, UC-600, Pacific Northwest Laboratory, Richland, Washington.

Nelson, J. V., 1996, *Estimation of Neutron Dose Rates from Nuclear Waste Packages*, (internal memo 8M730-JVN-96-007 to J. R. Green, March 8), Westinghouse Hanford Company, Richland, Washington.

NRC, 1982, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*, Regulatory Guide 1.145, U.S. Nuclear Regulatory Commission, Washington, D.C.

Rittmann, P. D., 1995, *ISO-PC Version 1.98 - User's Guide,* WHC-SD-WM-UM-030, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

RL, 1994, *Emergency Implementation Procedures*, DOE-0223, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

WHC-CM-2-14, *Hazardous Material Packaging and Shipping*, Westinghouse Hanford Company, Richland, Washington.

WHC-CM-4-46, *Nonreactor Facility Safety Analysis Manual*, Westinghouse Hanford Company, Richland, Washington.

WHC, 1995, *Packaging Design Criteria, Transfer and Disposal of Long-Length Equipment, Hanford Tank Farm Complex,* WHC-SD-TP-PDC-020, Rev. 2, Westinghouse Hanford Company, Richland, Washington.

WSDOH, 1993, "Response Procedures for Radiation Emergencies," Appendix A, *Protective Action Guides*, Washington State Department of Health, Olympia, Washington.

WS, 1994, "Fixed Nuclear Facility Emergency Response Procedure," Section 10.6 - Department of Agriculture, Washington State.

4.8 APPENDICES

4.8.1 ISO-PC Input File

```
0       2   PRTR Cask Side Dose Rate - Unshielded
Cyl. Source Geom - Dose Rate at 3 m Side Surface
&Input Next= 1 , ISpec= 3 , IGeom= 7 ,  ICONC=0,  SFACT=1, DUNIT=7,
 NTheta= 30, NPsi= 20, NShld= 1 , JBuf= 1, OPTION=1,
 Slth= 62.2,
 Y= 31.1  ,
 T(1)=3.969,
 X= 303.969,
 WEIGHT(335) =   375 ,
 WEIGHT(336) =   354.75,
 WEIGHT(472) =   5,     &
1Sourc   9   2.0
End of Input
&Input Next= 6 &
```

4.8.2 GXQ Input File

```
300 Area - Sector 99.5% X/Q Values - 100 m
c GXQ Version 4.0 Input File
c   mode
   1
c
c MODE CHOICE:
c mode  = 1 then X/Q based on Hanford site specific meteorology
c mode  = 2 then X/Q based on atmospheric stability class and wind speed
c mode  = 3 then X/Q plot file is created
c
c LOGICAL CHOICES:
c   ifox  inorm icdf  ichk  isite ipop
   T     F     F     F     F     F
c ifox   = t then joint frequency used to compute frequency to exceed X/Q
c        = f then joint frequency used to compute annual average X/Q
c inorm  = t then joint frequency data is normalized (as in GENII)
c        = f then joint frequency data is un-normalized
c icdf   = t then cumulative distribution file created (CDF.OUT)
c        = f then no cumulative distribution file created
c ichk   = t then X/Q parameter print option turned on
c        = f then no parameter print
c isite  = t then X/Q based on joint frequency data for all 16 sectors
c        = f then X/Q based on joint frequency data of individual sectors
c ipop   = t then X/Q is population weighted
c        = f then no population weighting
c
c X/Q AND WIND SPEED ADJUSTMENT MODELS:
c   ipuff idep  isrc  iwind
   0     0     0     0
c DIFFUSION COEFFICIENT ADJUSTMENT MODELS:
c   iwake ipm   iflow ientr
   0     0     0     0
c EFFECTIVE RELEASE HEIGHT ADJUSTMENT MODELS:
c  (irise igrnd)iwash igrav
   0     0     0     0
c ipuff  = 1 then X/Q calculated using puff model
c        = 0 then X/Q calculated using default continuous plume model
c idep   = 1 then plume depletion model turned on (Chamberlain model)
c isrc   = 1 then X/Q multiplied by scalar
c        = 2 then X/Q adjusted by wind speed function
c iwind  = 1 then wind speed corrected for plume height
c isize  = 1 then NRC RG 1.145 building wake model turned on
c        = 2 then MACCS virtual distance building wake model turned on
c ipm    = 1 then NRC RG 1.145 plume meander model turned on
```

```
c      = 2 then 5th Power Law plume meander model turned on
c      = 3 then sector average model turned on
c iflow = 1 then sigmas adjusted for volume flow rate
c ientr = 1 then method of Pasquill used to account for entrainment
c irise = 1 then MACCS buoyant plume rise model turned on
c      = 2 then ISC2 momentum/buoyancy plume rise model turned on
c igrnd = 1 then Mills buoyant plume rise modification for ground effects
c iwash = 1 then stack downwash model turned on
c igrav = 1 then gravitational settling model turned on
c      = 0 unless specified otherwise, 0 turns model off
c
c PARAMETER INPUT:
c              reference              frequency
c  release     anemometer    mixing       to
c  height      height        height      exceed
c  hs(m)       ha(m)         hm(m)       Cx(%)
c  _________   _________   _________   _________
  0.00000E+00  1.00000E+01  1.00000E+03  5.00000E-01
c
c  initial     initial                     gravitational
c  plume       plume       release     deposition    settling
c  width       height      duration    velocity      velocity
c  Wb(m)       Hb(m)       trd(hr)     vd(m/s)       vg(m/s)
c  _________   _________   _________   _________   _________
  0.00000E+00  0.00000E+00  0.00000E+00  1.00000E-03  1.00000E-03
c
c              initial      initial                convective
c  ambient      plume        plume       release    heat release
c  temperature  temperature  flow rate   diameter     rate(1)
c  Tamb©        T0©          V0(m3/s)    d(m)         qh(w)
c  _________   _________   _________   _________   _________
  2.00000E+01  2.20000E+01  1.00000E+00  1.00000E+00  0.00000E+00
c
c (1) If zero then buoyant flux based on plume/ambient temperature difference.
c
c  X/Q         Wind
c  scaling     Speed
c  factor      Exponent
c  c(?)        a(?)
c  _________   _________
  1.00000E+00  7.80000E-01
c
c RECEPTOR DEPENDENT DATA (no line limit)
c FOR MODE     make      RECEPTOR DEPENDENT DATA
c 1 (site specific)       sector distance receptor-height
c 2 (by class & wind speed) class windspeed distance offset receptor-height
c 3 (create plot file)      class windspeed xmax imax ymax jmax xqmin power
c
c RECEPTOR PARAMETER DESCRIPTION
c sector = 0, 1, 2... (all, S, SSW, etc.)
c distance = receptor distance (m)
c receptor height = height of receptor (m)
c class = 1, 2, 3, 4, 5, 6, 7 (P-G stability class A, B, C, D, E, F, G)
c windspeed = anemometer wind speed (m/s)
c offset = offset from plume centerline (m)
c xmax = maximum distance to plot or calculate to (m)
c imax = distance intervals
c ymax = maximum offset to plot (m)
c jmax = offset intervals
c xqmin = minimum scaled X/Q to calculate
c power = exponent in power function step size
0  100  0
```

4.8.3 GENII Input File

```
######################### Program GENII Input File ############ 8 Jul 88 ####
Title: PRTR Graphite Cask - 300 Area Onsite - Inhalation & Submersion
    \SAMPL\G-AIR.AC                    Created on 01-22-1990 at 07:30
OPTIONS============================ Default
========================================
F   Near-field scenario?     (Far-field)   NEAR-FIELD: narrowly-focused
F   Population dose?         (Individual)              release, single site
T   Acute release?           (Chronic)     FAR-FIELD:  wide-scale release,
    Maximum Individual data set used                   multiple sites
                          Complete                     Complete
TRANSPORT OPTIONS============ Section  EXPOSURE PATHWAY OPTIONS===== Section
T  Air Transport               1   F  Finite plume, external       5
F  Surface Water Transport     2   T  Infinite plume, external     5
F  Biotic Transport (near-field) 3,4  F  Ground, external          5
F  Waste Form Degradation (near) 3,4  F  Recreation, external      5
                                   T  Inhalation uptake          5,6
REPORT OPTIONS========================  F  Drinking water ingestion   7,8
T  Report AEDE only                F  Aquatic foods ingestion    7,8
F  Report by radionuclide          F  Terrestrial foods ingestion 7,9
F  Report by exposure pathway         F  Animal product ingestion   7,10
F  Debug report on screen          F  Inadvertent soil ingestion

INVENTORY ##################################################################

4  Inventory input activity units: (1-pCi  2-uCi  3-mCi  4-Ci  5-Bq)
0  Surface soil source units (1- m2  2- m3  3- kg)
   Equilibrium question goes here

   --------|----Release Terms------|----------Basic Concentrations---------|
   Use when|  transport selected   |  near-field scenario, optionally      |
   --------|-----------------------|---------------------------------------|
   Release |      Surface Buried |       Surface Deep   Ground  Surface|
   Radio-  |Air    Water   Waste |Air     Soil    Soil    Water   Water |
   nuclide |/yr    /yr     /m3   |/m3     /unit   /m3     /L      /L    |
   --------|------- ------- -------|------- ------- ------- ------- -------|
   SR90    3.75E-01
   Y 90    3.75E-01
   FE59    5.00E-03
   PU239   1.09E-02
   --------|----Derived Concentrations-----|
   Use when|  measured values are known    |
   --------|-------------------------------|
   Release |Terres. Animal  Drink   Aquatic|
   Radio-  |Plant   Product Water   Food   |
   nuclide |/kg     /kg     /L      /kg    |
   --------|------- ------- ------- -------|

TIME #######################################################################

1   Intake ends after (yr)
50  Dose calc. ends after (yr)
1   Release ends after (yr)
0   No. of years of air deposition prior to the intake period
0   No. of years of irrigation water deposition prior to the intake period

FAR-FIELD SCENARIOS (IF POPULATION DOSE) ###################################

0       Definition option: 1-Use population grid in file POP.IN
0                          2-Use total entered on this line

NEAR-FIELD SCENARIOS #######################################################

        Prior to the beginning of the intake period: (yr)
0         When was the inventory disposed?  (Package degradation starts)
0         When was LOIC?  (Biotic transport starts)
0       Fraction of roots in upper soil (top 15 cm)
```

```
0        Fraction of roots in deep soil
0        Manual redistribution: deep soil/surface soil dilution factor
0        Source area for external dose modification factor (m2)

TRANSPORT #######################################################################

         ====AIR TRANSPORT=======================================SECTION
1=====
               0-Calculate PM          |0     Release type (0-3)
3        Option: 1-Use chi/Q or PM value  |F     Stack release (T/F)
               2-Select MI dist & dir   |0     Stack height (m)
               3-Specify MI dist & dir  |0     Stack flow (m3/sec)
1        Chi/Q or PM value             |0     Stack radius (m)
14        MI sector index (1=S)         |0     Effluent temp. ©
100.      MI distance from release point (m)|0     Building x-section (m2)
T        Use jf data, (T/F) else chi/Q grid|0     Building height (m)

         ====SURFACE WATER TRANSPORT================================SECTION
2=====
0        Mixing ratio model: 0-use value, 1-river, 2-lake
0        Mixing ratio, dimensionless
0        Average river flow rate for: MIXFLG=0 (m3/s), MIXFLG=1,2 (m/s),
0        Transit time to irrigation withdrawl location (hr)
        If mixing ratio model > 0:
0          Rate of effluent discharge to receiving water body (m3/s)
0          Longshore distance from release point to usage location (m)
0          Offshore distance to the water intake (m)
0          Average water depth in surface water body (m)
0          Average river width (m), MIXFLG=1 only
0          Depth of effluent discharge point to surface water (m), lake only

         ====WASTE FORM AVAILABILITY================================SECTION
3=====
0        Waste form/package half life, (yr)
0        Waste thickness, (m)
0        Depth of soil overburden, m

         ====BIOTIC TRANSPORT OF BURIED SOURCE===============SECTION 4=====
T        Consider during inventory decay/buildup period (T/F)?
T        Consider during intake period (T/F)?  | 1-Arid non agricultural
0        Pre-Intake site condition.............| 2-Humid non agricultural
                                               | 3-Agricultural

EXPOSURE ########################################################################

         ====EXTERNAL EXPOSURE===============================SECTION 5=====
        Exposure time:              |  Residential irrigation:
0          Plume (hr)               | T     Consider: (T/F)
0          Soil contamination (hr)  | 0      Source: 1-ground water
0          Swimming (hr)            |             2-surface water
0          Boating (hr)             | 0     Application rate (in/yr)
0          Shoreline activities (hr) | 0     Duration (mo/yr)
0        Shoreline type: (1-river, 2-lake, 3-ocean, 4-tidal basin)
0        Transit time for release to reach aquatic recreation (hr)
1.0       Average fraction of time submersed in acute cloud (hr/person hr)

         ====INHALATION=========================================SECTION
6=====
8766.0    Hours of exposure to contamination per year
0        0-No resus-  1-Use Mass Loading        2-Use Anspaugh model
0          pension     Mass loading factor (g/m3)  Top soil available (cm)

         ====INGESTION POPULATION==============================SECTION 7=====
0        Atmospheric production definition (select option):
0          0-Use food-weighted chi/Q, (food-sec/m3), enter value on this line
          1-Use population-weighted chi/Q
          2-Use uniform production
          3-Use chi/Q and production grids (PRODUCTION will be overridden)
```

```
0       Population ingesting aquatic foods, 0 defaults to total (person)
0       Population ingesting drinking water, 0 defaults to total (person)
F       Consider dose from food exported out of region (default=F)

        Note below: S* or Source: 0-none, 1-ground water, 2-surface water
                              3-Derived concentration entered above
        ==== AQUATIC FOODS / DRINKING WATER INGESTION=========SECTION 8====

F       Salt water? (default is fresh)

        USE      TRAN-  PROD-   -CONSUMPTION- |
        ?  FOOD  SIT    UCTION  HOLDUP  RATE  |
        T/F TYPE hr     kg/yr   da      kg/yr |    DRINKING WATER
        --- ------ ----- ------- ------ ----- |------------------------
        F  FISH    0.00  0.0E+00  0.00   0.0 | 0    Source (see above)
        F  MOLLUS  0.00  0.0E+00  0.00   0.0 | T    Treatment? T/F
        F  CRUSTA  0.00  0.0E+00  0.00   0.0 | 0    Holdup/transit(da)
        F  PLANTS  0.00  0.0E+00  0.00   0.0 | 0    Consumption (L/yr)

        ====TERRESTRIAL FOOD INGESTION=======================SECTION 9=====

        USE      GROW  --IRRIGATION--        PROD-   --CONSUMPTION--
        ?  FOOD  TIME  S RATE  TIME   YIELD  UCTION  HOLDUP  RATE
        T/F TYPE da    * in/yr mo/yr  kg/m2  kg/yr   da      kg/yr
        --- ------ ----- - ----- -----  -----  ------- ------  ------
        F  LEAF V  0.00 0   0.0   0.0    0.0   0.0E+00   0.0     0.0
        F  ROOT V  0.00 0   0.0   0.0    0.0   0.0E+00   0.0     0.0
        F  FRUIT   0.00 0   0.0   0.0    0.0   0.0E+00   0.0     0.0
        F  GRAIN   0.00 0   0.0   0.0    0.0   0.0E+00   0.0     0.0

        ====ANIMAL PRODUCTION CONSUMPTION===================SECTION 10====

        ---HUMAN---- TOTAL   DRINK  -------------STORED FEED-------------
USE      CONSUMPTION PROD-   WATER  DIET  GROW  -IRRIGATION--     STOR-
?  FOOD   RATE  HOLDUP UCTION CONTAM FRAC- TIME S RATE  TIME  YIELD AGE
T/F TYPE  kg/yr  da    kg/yr  FRACT. TION  da   * in/yr mo/yr kg/m3 da
--- ------ ------ ----- ------ ------- ---- ---- - ----- ----- ----- -----
F  BEEF    0.0   0.0   0.00   0.00  0.00  0.0 0  0.0  0.00  0.00  0.0
F  POULTR  0.0   0.0   0.00   0.00  0.00  0.0 0  0.0  0.00  0.00  0.0
F  MILK    0.0   0.0   0.00   0.00  0.00  0.0 0  0.0  0.00  0.00  0.0
F  EGG     0.0   0.0   0.00   0.00  0.00  0.0 0  0.0  0.00  0.00  0.0
                             -------------FRESH FORAGE------------
   BEEF                      0.00   0.0 0  0.0  0.00  0.00  0.0
   MILK                      0.00   0.0 0  0.0  0.00  0.00  0.0

###############################################################################
```

4.8.4 Checklist for Technical Peer Review

CHECKLIST FOR REVIEW

Document Reviewed: THI for the 327 Building Family of Casks

Scope of Review: entire document

Yes	No	NA	
[ ]	[ ]	[✓] *	Previous reviews complete and cover analysis, up to scope of this review, with no gaps.
[✓]	[ ]	[ ]	Problem completely defined.
[✓]	[ ]	[ ]	Accident scenarios developed in a clear and logical manner.
[✓]	[ ]	[ ]	Necessary assumptions explicitly stated and supported.
[✓]	[ ]	[ ]	Computer codes and data files documented.
[✓]	[ ]	[ ]	Data used in calculations explicitly stated in document.
[✓]	[ ]	[ ]	Data checked for consistency with original source information as applicable.
[✓]	[ ]	[ ]	Mathematical derivations checked including dimensional consistency of results.
[✓]	[ ]	[ ]	Models appropriate and used within range of validity or use outside range of established validity justified.
[✓]	[ ]	[ ]	Hand calculations checked for errors. Spreadsheet results should be treated exactly the same as hand calculations.
[✓]	[ ]	[ ]	Software input correct and consistent with document reviewed.
[✓]	[ ]	[ ]	Software output consistent with input and with results reported in document reviewed.
[✓]	[ ]	[ ]	Limits/criteria/guidelines applied to analysis results are appropriate and referenced. Limits/criteria/guidelines checked against references.
[✓]	[ ]	[ ]	Safety margins consistent with good engineering practices.
[✓]	[ ]	[ ]	Conclusions consistent with analytical results and applicable limits.
[✓]	[ ]	[ ]	Results and conclusions address all points required in the problem statement.
[✓]	[ ]	[ ]	Format consistent with appropriate NRC Regulatory Guide or other standards
[ ]		[✓] *	Review calculations, comments, and/or notes are attached.
[✓]	[ ]	[ ]	**Document approved.**

J. G. McFadden   J G McFadden   7/21/97

Reviewer (Printed Name and Signature)   Date

* Any calculations, comments, or notes generated as part of this review should be signed, dated and attached to this checklist. Such material should be labeled and recorded in such a manner as to be intelligible to a technically qualified third party.

4.8.5 HEDOP Review Checklist

HEDOP REVIEW CHECKLIST
for
Radiological and Nonradiological Release Calculations

Document: Transportation Hazard Index (THI) Analysis for the 327 Building Casks SEPs," May 9, 1997.

Scope of Review: Inhalation/Air Submersion Dose Calculations

YES	NO*	N/A		
[x]	[ ]	[ ]	1.	A detailed technical review and approval of the environmental transport and dose calculation portion of the analysis has been performed and documented.
[ ]	[ ]	[x]	2.	Detailed technical review(s) and approval(s) of scenario and release determinations have been performed and documented.
[ ]	[x]	[ ]	3.	HEDOP-approved code(s) were used.
[x]	[ ]	[ ]	4.	Receptor locations were selected according to HEDOP recommendations.
[x]	[ ]	[ ]	5.	All applicable environmental pathways and code options were included and are appropriate for the calculations.
[x]	[ ]	[ ]	6.	Hanford site data were used.
[x]	[ ]	[ ]	7.	Model adjustments external to the computer program were justified and performed correctly.
[x]	[ ]	[ ]	8.	The analysis is consistent with HEDOP recommendations.
[x]		[ ]	9.	Supporting notes, calculations, comments, comment resolutions, or other information is attached. (Use the "Page 1 of X" page numbering format and sign and date each added page.)
[x]	[ ]		10.	Approval is granted on behalf of the Hanford Environmental Dose Overview Panel.

* All "NO" responses must be explained and use of nonstandard methods justified.

Kathy Rhoads K Rhoads 5/30/97
HEDOP-Approved Reviewer (Printed Name and Signature) Date

COMMENTS (add additional signed and dated pages if necessary):

Item 3: GXQ used for air transport calculations; GENII results are included for comparison.

5.0 SHIELDING EVALUATION

5.1 INTRODUCTION

This shielding evaluation supports the shipment of activated materials and fuel components in the SERF Cask within the Hanford 300 Area.

5.2 DIRECT RADIATION SOURCE SPECIFICATION

The source term for the materials transported by the cask will be variable and, therefore, is described generally and evaluated on the basis of bounding conditions for gamma and neutron emissions. The source term evaluated for shielding is shown in Table B5-1. Isotopes other than those listed may be shipped in the cask, but the surface dose rate is limited as shown in Part B, Section 5.4 in all cases.

Table B5-1. Shielding Source Term.

Source	Material	Activity Limit		Comments
		Tbq	Ci	
Mixed inventory	Fissile materials and α emitters*	Not considered for shielding except for 175 g (0.4 Tbq [10.8 Ci]) of ^{239}Pu. The neutron dose rate from ^{238}Pu bounds the neutron dose rate for all other materials in this category.		
	Mixed fission products	37	1000	Considered to be all ^{137}Cs for shielding
	Mixed activation products	555	15000	Considered to be all ^{60}Co for shielding
Cesium capsules	Cesium	2960	80000	^{137}Cs
Fuel	Defined in Part B, Section 2.0*	Limits defined by criticality safety.		

*Limited by criticality safety as shown in Part B, Section 2.1.1

5.2.1 *Gamma Source*

Two gamma sources were evaluated. The first was a mixed inventory of ^{60}Co and ^{137}Cs with an equilibrium amount of the ^{137}Cs daughter, ^{137m}Ba. Although the mixed inventory may have additional gamma sources, these are bounded by the ^{60}Co and ^{137}Cs terms and were not considered separately. The second gamma source considered was for cesium capsules containing ^{137}Cs with an equilibrium amount of ^{137m}Ba.

5.2.2 Beta Source

Beta particles originating in the source do not contribute directly to the dose rate outside the casks because of the shielding provided. Although the bremsstrahlung radiation produced by the deceleration of the beta particles in the source is a potential contribution to the source, the contribution is minimal and bounded by the gamma source term discussed in Part B, Section 5.2.1. For the isotopes evaluated, however, bremsstrahlung was considered.

5.2.3 Neutron Source

The neutron source term for the mixed materials inventory was calculated using the ORIGEN2 computer code. The worst-case neutron source term was found to be from ^{239}Pu and was found to occur prior to any decay being considered. Although ^{235}U was considered, its neutron source term is negligible compared to ^{239}Pu. The source term is shown in Table B5-2 and the ORIGEN2 input file is included in Part B, Section 5.8.

Table B5-2. Neutron Source Term for 175g of ^{239}Pu

Component of source	Source strength (neutrons/s)
(α,n)	7.927 E+03
Spontaneous fission	3.967
Total	7.931 E+03

The cesium capsule inventory has no neutron emitters.

5.3 SUMMARY OF SHIELDING PROPERTIES OF MATERIALS

The shielding in the cask is provided by lead encased in a stainless steel shell. The default densities for iron and lead provided in the ISO-PC computer code were used for shielding calculations.

5.4 NORMAL TRANSPORT CONDITIONS

5.4.1 Conditions To Be Evaluated

Gamma dose rates were evaluated at the surface of each cask (with an offset of 1 cm) and at 2 m (6.6 ft) from the cask surface. Neutron dose rates were estimated at the surface of the cask.

5.4.2 Acceptance Criteria

Transportation safety specifies a maximum of 2 mSv/h (200 mrem/h) on any surface of the cask, 0.1 mSv/h (10 mrem/h) at 2 m (6.6 ft) from the cask surface, and 0.02 mSv/h (2 mrem/h) in any normally occupied space. If these limits are exceeded, material will be removed from the cask or supplemental shielding will be added.

5.4.3 Assumptions

The following assumptions were made and applied to the shielding model.

1. The air space in the closure mechanism tube was ignored. The steel in the closure mechanism tube was combined with the steel covering the outside of the mechanism.

2. The cask end opposite the closure contains a steel plunger mechanism that extends through the shielding (see drawing H-3-38542). A cylindrical source with a diameter equivalent to the plunger mechanism and a length equivalent to the cask cavity was used to determine the dose outside the plunger. This dose was added to the dose considered at the cask end, assuming a uniform steel and lead plug without the plunger. It is assumed that a steel rod (essentially a push rod) fills the hole through the plunger mechanism although this is not shown on the drawing. Such an arrangement is used on other casks of this type, such as the Long Bore Cask.

3. The mixed inventory contents were assumed to be mainly iron pieces with a nominal density of 1.5 g/cm^3. This density was arrived at by considering that one-half of the maximum inventory of 226.8 kg (500 lb) is evenly distributed throughout the entire cask volume. Note that dose rates are more sensitive to material density than material type; therefore, this is a conservative approach that reduces the effects of self-shielding in the payload.

4. The cesium capsule inventory density was considered as that of air as the cesium in the capsules occupies a small portion of the overall volume. This is a conservative approach that eliminates the effects of self-shielding in the payload.

5. The source considered is uniformly distributed throughout the cask volume.

6. The mixed activation products in the mixed inventory were conservatively assumed to be all ^{60}Co.

7. The mixed fission products in the mixed inventory were conservatively assumed to be all ^{137}Cs.

8. Bremsstrahlung was only considered for isotopes evaluated; e.g., ^{137}Cs, and not for any other isotopes that may be present, such as ^{90}Sr.

5.4.4 Shielding Model

The shielding source term considered is shown in Tables B5-1 and B5-2. The source parameters are shown in Table B5-3, and the shielding parameters are shown in Table B5-4. The data for the shielding and source parameters were taken from drawing H-3-38542.

The ISO-PC program (Rittmann 1995) was used for the gamma-ray dose rate calculations. ISO-PC uses the point-kernel integration method to compute the dose rate at a detector location. Bremsstrahlung photons are accounted for in the dose rate calculations. Fluence-to-dose conversion factors were based on an anterior-to-posterior irradiation pattern (ANS 1991).

Table B5-3. Source Parameters.

Source	Diameter		Length		Volume (cm^3)
	cm	in.	cm	in.	
Overall (entire cask interior volume)	19.368	7.625	257.886	101.530	76078.187
Plunger only (part of source directly below plunger area)	3.340	1.315	257.886	101.530	2262.726

Table B5-4. Shielding Parameters.

Detector location	Steel inner wall thickness		Lead thickness		Steel outer wall thickness		Distance to surface detector (including source thickness)	
	cm	in.	cm	in.	cm	in.	cm	in.
Side	1.270	0.500	20.320	8.000	1.588	0.625	32.861	12.938
Closure end	2.223	0.875	19.723	7.765	2.223	0.875	282.054	111.045
Plunger end*	1.270	0.500	20.625	8.120	1.905	0.750	284.861	112.150
Plunger	Diameter		Length					
	cm	in.	cm	in.				
	3.340	1.315	23.165	9.120				

*Includes cover plate and airspace on plunger end; see drawing H-3-38542, Sheet 8.

The neutron dose rate was determined using a method discussed in *Estimation of Neutron Dose Rates from Nuclear Waste Packages* (Nelson 1996). This is a very conservative method that does not take shielding or moderation into account.

5.4.5 Shielding Calculations

Table B5-5 shows the gamma dose rate estimates calculated by ISO-PC for the mixed material inventory. Table B5-6 shows the gamma dose rate for the cesium capsule inventory. For the mixed material inventory, the neutron dose rate calculations utilized data for α,n and spontaneous fission neutron production rates generated by ORIGEN. The neutron production rate information was then used in the dose rate calculation method described in *Estimation of Neutron Dose Rates from Nuclear Waste Packages* (Nelson 1996). The neutron dose rate was determined to be 0.36 mrem/h at the cask surface. The ISO-PC input file, the ORIGEN input file, and the neutron dose calculations are attached in Part B, Section 5.8.

Table B5-5. Maximum Gamma Dose Rates Around the SERF Cask for Mixed Inventory.

Detector orientation	Detector location			
	Surface (1 cm offset)		2 m	
	mSv/h	mrem/h	mSv/h	mrem/h
Side	0.45	44.95	0.06	6.18
Closure end	0.42	41.64	0.01	1.22
Plunger end plus plunger	1.58	157.79	0.04	4.18
Drivers position (from plunger end)	▨	▨	3.25 m	
			0.02	1.79

Table B5-6. Maximum Gamma Dose Rates Around the SERF Cask for Cesium Capsule.

Detector orientation	Detector location			
	Surface (1 cm offset)		2 m	
	mSv/h	mrem/h	mSv/h	mrem/h
Side	3.39 E-05	3.39 E-03	4.78 E-06	4.78 E-04
Closure end	9.67 E-5	9.67 E-03	8.21 E-06	8.21 E-04
Plunger end plus plunger	0.26	25.90	0.02	1.87

5.5 ACCIDENT CONDITIONS

A handling accident in which the source material was concentrated against the plunger end of the cask was considered for the mixed inventory only. The cesium capsules are considered to remain intact in such accident conditions.

5.5.1 Acceptance Criteria

The maximum dose rate at 1 m (3.3 ft) shall be less than 10 mSv/h (1,000 mrem/h).

5.5.2 Assumptions

The same assumptions as shown in Part B, Section 5.4.3 shall be used except that the source material will be concentrated in a cylinder that is the same diameter as the cask interior and 5.08 cm (2 in) tall. The material density will be conservatively maintained as 1.5 g/cm^3.

5.5.3 Shielding Model

The shielding model shall be the same as used in Part B, Section 5.4.4, except that the source length will 5.08 cm (2 in).

5.5.4 Shielding Calculations

Table B5-7 shows the gamma dose rate estimates for accident conditions as calculated by ISO-PC. The neutron dose rate was not calculated as it is was already shown to be insignificant when compared to the gamma dose rate, even without shielding, in Part B, Section 5.4. The ISO-PC input file is attached in Part B, Section 5.8.

Table B5-7. Maximum Accident Gamma Dose Rates Around the SERF Cask for the Mixed Inventory Payload.

Detector orientation	Detector location 1 m (3.3 ft)	
	mSv/h	mrem/h
Side	0.35	34.70
Closure end	0.08	7.88
Plunger end plus plunger	2.46	246.24

5.6 CONCLUSIONS

The gamma dose rates shown in Table B5-5 for the mixed materials inventory and B5-6 for the cesium capsule inventory are within the normal conditions of transport acceptance criteria. The criteria evaluated were a surface dose rate of less than 2 mSv/h (200 mrem/h), 0.1 mSv/h (10 mrem/h) at 2 m (6.6 ft), and 0.02 mSv/h (2 mrem/h) at the driver's position, assuming the driver is no closer than 3.25 m (10.7 ft). As shown in Table B5-7, the maximum allowed accident condition dose rates are also met for the conditions evaluated. The neutron dose rate of 0.36 mrem/h at the cask surface is inconsequential for the mixed material inventory.

It should be noted that the shielding model is very conservative, assuming a contents density of 1.5 g/cm^3 that is evenly distributed in the cask volume and assuming that the activated materials inventory is all ^{60}Co. During use, the dose rates are measured prior to shipment in accordance with facility procedures. Particular attention should be paid to the plunger end of the cask due to the reduced shielding in this area.

5.7 REFERENCES

ANS, 1991, *Neutron and Gamma-Ray Fluence-to-Dose Factors,* ANSI/ANS-6.1.1-1991, American Nuclear Society, La Grange Park, Illinois.

Nelson, J. V., 1996, *Estimation of Neutron Dose Rates from Nuclear Waste Packages* (internal memo 8M730-JVN-96-007 to J. R. Green, March 8), Westinghouse Hanford Company, Richland, Washington.

Rittmann, P. D., 1995, *ISO-PC Version 1.98 - User's Guide,* WHC-SD-WM-UM-030, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

Schmittroth, F. A., 1994, *Conversion of ORIGEN2 to Sun Workstations*, WHC-SD-NR-SWD-006, Rev. 1, Westinghouse Hanford Company, Richland, Washington.

5.8 APPENDICES

5.8.1 ISO-PC Input Files

5.8.1.1 Input File for Mixed Materials Inventory.

```
0       2   SERF Cask
 Case Side
&Input Next= 1, IPrnt=0, IGeom= 7, ICONC=0, SFACT=1,
 DUNIT=1,  NTheta= 30, NPsi= 30, NShld= 4, JBuf= 4, OPTION=0,
 Slth= 257.886,
 T(1)= 9.685,
 T(2)= 1.27,
 T(3)= 20.320,
 T(4)= 1.588,
 X= 33.863,
 Y= 128.95,
 WEIGHT(335)  =  1.0E3 ,
 WEIGHT(336)  =  946.0 ,
 WEIGHT(472)  =  1.5E4 , &
 Steel  9    1.5
 Steel  9            7.86
 Lead  14                     11.35
1Steel  9                              7.86
Dose Rate at 2 m
&Input Next=4, X = 232.863, &
 Case Closure End
&Input Next= 1, IPrnt=0, IGeom= 9, ICONC=0, SFACT=1, .
 DUNIT= 1, NTheta= 30, NPsi= 30, NShld= 4, JBuf= 4, OPTION=0,
 Slth= 9.685,
 T(1)= 257.886,
 T(2)= 2.223,
 T(3)= 19.723,
 T(4)= 2.223,
 X= 283.055,
 WEIGHT(335)  =  1.0E3 ,
 WEIGHT(336)  =  946.0 ,
 WEIGHT(472)  =  1.5E4 , &
 Steel  9    1.5
 Steel  9            7.86
 Lead  14                     11.35
1Steel  9                              7.86
Dose Rate at 2 m
&Input Next=4, X = 482.055, &
 Case Plunger End
&Input Next= 1, IPrnt=0, IGeom= 9, ICONC=0, SFACT=1,
 DUNIT= 1, NTheta= 30, NPsi= 30, NShld= 4, JBuf= 4, OPTION=0,
 Slth= 9.685,
 T(1)= 257.886,
 T(2)= 1.27,
 T(3)= 20.625,
 T(4)= 1.905,
 X= 285.861 ,
 WEIGHT(335)  =  1.0E3 ,
 WEIGHT(336)  =  946.0 ,
 WEIGHT(472)  =  1.5E4 , &
 Steel  9    1.5
 Steel  9            7.86
 Lead  14                     11.35
1Steel  9                              7.86
```

```
Dose Rate at 2 m
&Input Next=4, X = 484.861, &
Estimate Driver Position (3.25 m)
&Input Next=4, X = 609.861, &
 Case Plunger
&Input Next= 1, IPrnt=0, IGeom= 9, ICONC=0, SFACT= 2.97E-2,
 DUNIT= 1, NTheta= 30, NPsi= 30, NShld= 4, JBuf= 2, OPTION=0,
 Slth= 3.340,
 T(1)= 257.886,
 T(2)= 23.165
 T(3)= 3.175,
 T(4)= 0.635,
 X= 285.861 ,
 WEIGHT(335) =  1.0E3 ,
 WEIGHT(336) =  946.0 ,
 WEIGHT(472) =  1.5E4 , &
 Steel   9    1.5
 Steel   9         7.86
 Air     3             0.00129
1Steel   9                      7.86
Dose Rate at 2 m
&Input Next=4, X = 484.861, &
Estimate Driver Position (3.25 m)
&Input Next=4, X = 609.861, &
 Case Side Accident
&Input Next= 1, IPrnt=0, IGeom= 7, ICONC=0, SFACT=1,
 DUNIT=1,  NTheta= 30, NPsi= 30, NShld= 4, JBuf= 4, OPTION=0,
 Slth= 2.54,
 T(1)= 9.685,
 T(2)= 1.27,
 T(3)= 20.320,
 T(4)= 1.588,
 X= 33.863,
 Y= 1.27,
 WEIGHT(335) =  1.0E3 ,
 WEIGHT(336) =  946.0 ,
 WEIGHT(472) =  1.5E4 , &
 Steel   9    1.5
 Steel   9         7.86
 Lead    14                11.35
1Steel   9                      7.86
Dose Rate at 1 m
&Input Next=4, X = 132.863, &
 Case Closure End Accident
&Input Next= 1, IPrnt=0, IGeom= 9, ICONC=0, SFACT=1,
 DUNIT= 1, NTheta= 30, NPsi= 30, NShld= 5, JBuf= 5, OPTION=0,
 Slth= 9.685,
 T(1)= 2.54,
 T(2)= 255.346,
 T(3)= 2.223,
 T(4)= 19.723,
 T(5)= 2.223,
 X= 283.055,
 WEIGHT(335) =  1.0E3 ,
 WEIGHT(336) =  946.0 ,
 WEIGHT(472) =  1.5E4 , &
 Steel   9   1.5
 Air     3        0.00129
 Steel   9                 7.86
 Lead    14                      11.35
1Steel   9                             7.86
Dose Rate at 1 m
&Input Next=4, X = 382.055, &
 Case Plunger End Accident
&Input Next= 1, IPrnt=0, IGeom= 9, ICONC=0, SFACT=1,
 DUNIT= 1, NTheta= 30, NPsi= 30, NShld= 4, JBuf= 4, OPTION=0,
 Slth= 9.685,
 T(1)= 2.54,
 T(2)= 1.27,
 T(3)= 20.625,
 T(4)= 1.905,
 X= 27.34 ,
```

```
 WEIGHT(335) =  1.0E3 ,
 WEIGHT(336) =  946.0 ,
 WEIGHT(472) =  1.5E4 , &
 Steel   9     1.5
 Steel   9            7.86
 Lead   14                   11.35
1Steel   9                          7.86
Dose Rate at 1 m
&Input Next=4, X = 126.34, &
 Case Plunger Accident
&Input Next= 1, IPrnt=0, IGeom= 9, ICONC=0, SFACT= 2.97E-2,
 DUNIT= 1, NTheta= 30, NPsi= 30, NShld= 4, JBuf= 2, OPTION=0,
 Slth= 3.340,
 T(1)= 2.54,
 T(2)= 23.165
 T(3)= 3.175,
 T(4)= 0.635,
 X= 30.515,
 WEIGHT(335) =  1.0E3 ,
 WEIGHT(336) =  946.0 ,
 WEIGHT(472) =  1.5E4 , &
 Steel   9     1.5
 Steel   9            7.86
 Air     3                0.00129
1Steel   9                          7.86
Dose Rate at 1 m
&Input Next=4, X = 129.515, &
End of Input
&Input Next= 6 &
```

5.8.1.2 Input File for Cesium Capsule Inventory.

```
0        2   SERF Cask 80 kCi Cs
 Case Side
&Input Next= 1, IPrnt=0, IGeom= 7, ICONC=0, SFACT=1,
 DUNIT=1, NTheta= 30, NPsi= 30, NShld= 4, JBuf= 4, OPTION=0,
 Slth= 257.886,
 T(1)= 9.685,
 T(2)= 1.27,
 T(3)= 20.320,
 T(4)= 1.588,
 X= 33.863,
 Y= 128.95,
 WEIGHT(335) =  8.0E4 ,
 WEIGHT(336) =  75680.0 ,&
 Air     3  0.00129
 Steel   9            7.86
 Lead   14                   11.35
1Steel   9                          7.86
Dose Rate at 2 m
&Input Next=4, X = 232.863, &
 Case Closure End
&Input Next= 1, IPrnt=0, IGeom= 9, ICONC=0, SFACT=1,
 DUNIT= 1, NTheta= 30, NPsi= 30, NShld= 4, JBuf= 4, OPTION=0,
 Slth= 9.685,
 T(1)= 257.886,
 T(2)= 2.223,
 T(3)= 19.723,
 T(4)= 2.223,
 X= 283.055,
 WEIGHT(335) =  8.0E4 ,
 WEIGHT(336) =  75680.0 ,&
 Air     3  0.00129
 Steel   9            7.86
 Lead   14                   11.35
1Steel   9                          7.86
Dose Rate at 2 m
&Input Next=4, X = 482.055, &
 Case Plunger End
&Input Next= 1, IPrnt=0, IGeom= 9, ICONC=0, SFACT=1,
 DUNIT= 1, NTheta= 30, NPsi= 30, NShld= 4, JBuf= 4, OPTION=0,
```

```
 Slth= 9.685,
 T(1)= 257.886,
 T(2)= 1.27,
 T(3)= 20.625,
 T(4)= 1.905,
 X= 285.861 ,
 WEIGHT(335) =  8.0E4 ,
 WEIGHT(336) =  75680.0 ,&
 Air    3  0.00129
 Steel  9             7.86
 Lead   14                      11.35
1Steel  9                              7.86
Dose Rate at 2 m
&Input Next=4, X = 484.861, &
 Case Plunger
&Input Next= 1, IPrnt=0, IGeom= 9, ICONC=0, SFACT= 2.97E-2,
 DUNIT= 1, NTheta= 30, NPsi= 30, NShld= 4, JBuf= 2, OPTION=0,
 Slth= 3.340,
 T(1)= 257.886,
 T(2)= 23.165
 T(3)= 3.175,
 T(4)= 0.635,
 X= 285.861 ,
 WEIGHT(335) =  8.0E4 ,
 WEIGHT(336) =  75680.0 , &
 Air    3  0.00129
 Steel  9             7.86
 Air    3                   0.00129
1Steel  9                              7.86
Dose Rate at 2 m
&Input Next=4, X = 484.861, &
End of Input
&Input Next= 6 &
```

5.8.2 ORIGEN Input File

```
-1
-1
-1
TIT    PU DECAY - 175 g Pu239/ 275 g U235 - Neutron Emission Calc
BAS     PLUTONIUM DECAY IN 5 YEAR INTERVALS
LIP    0 0 0
LIB    0 1 2 3 381 382 383 9 0 0 1 1
PHO     101 102 103 10
RDA     1 METRIC TON PLUTONIUM
INP    -1 1 -1 -1 1 1
RDA     DECAY FUEL
MOV      -1 1 0 1.00
RDA
DEC     50.0          1  2  4  0
DEC      1.0          2  3  5  0
DEC     10.0          3  4  5  0
DEC     20.0          4  5  5  0
DEC     30.0          5  6  5  0
DEC     40.0          6  7  5  0
DEC     50.0          7  8  5  0
DEC     60.0          8  9  5  0
DEC     70.0          9 10  5  0
DEC     80.0         10 11  5  0
RDA
CUT     5  1.E-10  7  1.E-10  9  1.E-10  -1
OPTL    24*8
OPTF    24*8
OPTA    4*8 7 7 7 8 7 8 14*8
OUT     11 1 -1 0
STP     4
2  942390  175.000 922350 275.00   0     0     0     0
0
END
```

5.8.3 Neutron Dose Calculations

The neutron dose rate was determined using the method described in *Estimation of Neutron Dose Rates from Nuclear Waste Packages* (Nelson 1996). In this method, the total neutron source term S(T), which accounts for neutron multiplication, is determined by adding the spontaneous fission source term (S(SF)) and the α,n source term (S(α,n) and dividing by 1 minus the k_{eff}.

$$S(ST) = \frac{(S(SF) - S(\alpha, n))}{(1 - k_{eff})}$$

S(SF) and S(α,n) are determined either from Nelson (1996) or ORIGEN (see Part B, Section 5.8.2). After S(ST) is determined, it is used to determine the dose rate in the equation;

$$D(r) = \frac{0.01 \cdot S(ST)}{r^2}$$

where r is the distance from the source and D(r) is the dose in mrem/h as a function of r.

Therefore, the neutron dose for the SERF cask is estimated as follows. Using S(SF) and S(α,n) from ORIGEN and conservatively assuming a k_{eff} of 0.8 (Part B, Section 6.0), S(ST) is determined to be,

$$S(ST) = \frac{3.967 + 7.927 \ x \ 10^{3}}{(1 - 0.8)} = 3.965 \ x \ 10^{4} \ n/s.$$

Assuming r to be 33.02 cm (the approximate surface of cask as measured radially), the total neutron dose rate is estimated to be,

$$D(r) = \frac{0.01 \cdot 3.965 \ x \ 10^{4}}{33.02^{2}} = 0.36 \ mrem/h$$

5.8.4 Checklist for Technical Peer Review

CHECKLIST FOR TECHNICAL PEER REVIEW

Document: Safety Evaluation for Packaging (Onsite) SERF Cask, June 2, 1997, by John McCoy.

Scope: Shielding portion of the analysis.

Yes	No	NA	
[ ]	[ ]	[X]	Previous reviews complete and cover analysis, up to scope of this review, with no gaps.
[X]	[ ]	[ ]	Problem completely defined.
[ ]	[ ]	[X]	Accident scenarios developed in a clear and logical manner.
[X]	[ ]	[ ]	Necessary assumptions explicitly stated and supported.
[X]	[ ]	[ ]	Computer codes and data files documented.
[X]	[ ]	[ ]	Data used in calculations explicitly stated in document.
[X]	[ ]	[ ]	Data checked for consistency with original source information as applicable.
[X]	[ ]	[ ]	Mathematical derivations checked including dimensional consistency of results.
[X]	[ ]	[ ]	Models appropriate and used within range of validity or use outside range of established validity justified.
[X]	[ ]	[ ]	Hand calculations checked for errors. Spreadsheet results should be treated exactly the same as hand calculations.
[X]	[ ]	[ ]	Software input correct and consistent with document reviewed.
[X]	[ ]	[ ]	Software output consistent with input and with results reported in document reviewed.
[X]	[ ]	[ ]	Limits/criteria/guidelines applied to analysis results are appropriate and referenced. Limits/criteria/guidelines checked against references.
[X]	[ ]	[ ]	Safety margins consistent with good engineering practices.
[X]	[ ]	[ ]	Conclusions consistent with analytical results and applicable limits.
[X]	[ ]	[ ]	Results and conclusions address all points required in the problem statement.
[X]	[ ]	[ ]	Format consistent with appropriate NRC Regulatory Guide or other standards
[ ]		[X]	Review calculations, comments, and/or notes are attached.
[X]	[ ]	[ ]	Document approved.

Anthony Savino 6/ 2/97
Reviewer (Printed Name and Signature) Date

5.8.4 Checklist for Technical Peer Review

CHECKLIST FOR TECHNICAL PEER REVIEW

Document: Safety Evaluation for Packaging (Onsite) SERF Cask, June 2, 1997, by John McCoy.

Scope: Shielding portion of the analysis.

Yes	No	NA	
[ ]	[ ]	[X]	Previous reviews complete and cover analysis, up to scope of this review, with no gaps.
[X]	[ ]	[ ]	Problem completely defined.
[ ]	[ ]	[X]	Accident scenarios developed in a clear and logical manner.
[X]	[ ]	[ ]	Necessary assumptions explicitly stated and supported.
[X]	[ ]	[ ]	Computer codes and data files documented.
[X]	[ ]	[ ]	Data used in calculations explicitly stated in document.
[X]	[ ]	[ ]	Data checked for consistency with original source information as applicable.
[X]	[ ]	[ ]	Mathematical derivations checked including dimensional consistency of results.
[X]	[ ]	[ ]	Models appropriate and used within range of validity or use outside range of established validity justified.
[X]	[ ]	[ ]	Hand calculations checked for errors. Spreadsheet results should be treated exactly the same as hand calculations.
[X]	[ ]	[ ]	Software input correct and consistent with document reviewed.
[X]	[ ]	[ ]	Software output consistent with input and with results reported in document reviewed.
[X]	[ ]	[ ]	Limits/criteria/guidelines applied to analysis results are appropriate and referenced. Limits/criteria/guidelines checked against references.
[X]	[ ]	[ ]	Safety margins consistent with good engineering practices.
[X]	[ ]	[ ]	Conclusions consistent with analytical results and applicable limits.
[X]	[ ]	[ ]	Results and conclusions address all points required in the problem statement.
[X]	[ ]	[ ]	Format consistent with appropriate NRC Regulatory Guide or other standards
[ ]		[X]	Review calculations, comments, and/or notes are attached.
[X]	[ ]	[ ]	**Document approved.**

Anthony Savino 6/ 2/97

Reviewer (Printed Name and Signature) Date

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6.0 CRITICALITY EVALUATION

Limits for Mixed Oxide Fuel Pins, N Reactor Fuel Elements and Scrap in the SERF Cask and the Waste Cask (see Section 6.1) is a criticality safety analysis performed to establish the nuclear safety limits for the transportation and storage of mixed plutonium-uranium oxide fuel pins, uranium oxide fuel pins, and N Reactor fuel elements in combination with or separate from fissionable material scrap of various compositions in the SERF Cask. The analysis includes modeling a number of accident scenarios to ensure subcriticality even in the event of a loss of contingency error. The analysis considers dry conditions and water flooding of the cask. Because water flooding of the cask is considered only as an accident condition, the limits used in this SEP are for dry conditions. Table A3-3 shows the criticality limits for dry (H:≤X5) conditions in the SERF Cask, which are extracted from the analysis.

6.1 APPENDIX: CRITICALITY SAFETY ANALYSIS

Battelle
Pacific Northwest Laboratories

Project Number ________________

Internal Distribution

VC Asmund
LC Davenport w/o
AL Doherty
BM Durst w/o
DL Haggard
EC Hawkes
SB Johnston
DE Knowlton
SD Landsman
AW Prichard
JM Seay
RE Thornhill
Library
Record Copy
File/LB

Date August 31, 1995
To M Dec
From SL Larson SLLarson
Subject NCS Basis Memo 95-3 Rev 1, Limits for Mixed Oxide Fuel Pins, N Reactor Fuel Elements and Scrap in the SERF Cask and the Waste Cask

Reference 1:	Memo, RA Libby to LC Davenport, "NCS Memo 88-5: Criticality Safety Basis for SERF Cask," April 29, 1988.
Reference 2:	Code of Federal Regulations 10 Part 71.55 paragraph b, 1995.
Reference 3:	PNL Laboratory Safety, "Criticality Safety," PNL-MA-25, January 1994.
Reference 4:	Oak Ridge National Laboratory, SCALE 4.2 Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, v1-3, CCC-545, November 1993.
Reference 5:	W Marshall, et al., "Criticality Safety Criteria," ANS Transactions, 35, 278-279, 1980.
Reference 6:	LE Hansen, ED Clayton, RC Lloyd, SR Bierman and RD Johnson, "Critical Parameters of Plutonium Solutions," Parts I and II, *Nuclear Applications*, 6 371-390, 1969.
Reference 7:	BM Durst, SR Bierman and ED Clayton, "Handbook of Critical Experiments Benchmarks," PNL-2700, March 1978.
Reference 8:	SR Bierman, BM Durst and ED Clayton, "Critical Experiments with Subcritical Clusters of 2.35 wt% and 4.31 wt% ^{235}U Enriched UO$_2$ Rods in Water with Uranium or Lead Reflecting Walls," NUREG/CR-0796, v2, 1979.
Reference 9:	RC Lloyd, SR Bierman, ED Clayton, and BM Durst, "Criticality Studies of a Neutron Multiplier Lattice," BNWL-2031, April 1976.
Reference 10:	BM Durst, SR Bierman, ED Clayton and JF Mincey, "Critical Experiments with Solid Neutron Absorbers and Water-Moderated Fast Test Reactor Fuel Pins," *Nuclear Technology*, 48, 129-149, mid-April 1980.

Reference 11: AF Kupinski and H Toffer, "Use of the HAMMER System for Evaluating Light-Water Moderated, Critical Assemblies," DUN-7286, October 1, 1970.

Reference 12: "Nuclear Physics Research Quarterly Report," HW-60220, 55-57, April 20, 1959.

Reference 13: CL Brown and LE Hansen, "Material Buckling Experiments with 2.1 wt% ^{235}U Enriched Uranium Tubes in Light Water," BNWL-SA-1090, April 27, 1967.

Reference 14: Memo, SL Larson to LC Davenport, "NCS Basis Memo 95-3, Limits for Mixed Oxide Fuel Pins, N Reactor Fuel Elements and Scrap in the SERF Cask and the Waste Cask," July 18, 1995.

This basis memo establishes the nuclear safety limits for the storage and transportation of mixed plutonium-uranium oxide fuel pins, uranium oxide fuel pins, N Reactor fuel elements and scrap in the SERF Cask and the Waste Cask. This memo validates the limits against the requirements of DOE Order 5480.24 issued 8/12/92 and ANSI/ANS-8.1-1983 [Reaffirmed 1988].

Revision Notice

This basis memo replaces Reference 14 in its entirety. The first change made in this revision corrected the N Reactor fuel diameter given in Tables 24 and 25 to reflect the actual diameter modeled. The second change adds a paragraph below Table 22 which indicates that the ^{239}Pu scrap limit is applicable to all other fissionable materials with a larger minimum critical mass.

Review of Methods Used Previously

The minimum critical mass (MCM) and minimum critical number (MCN) of fuel scrap and pins, respectively, were calculated for use in the SERF Cask based on a variety of criteria for various fissionable materials as summarized in Reference 1. The reference assumed water reflection around the fissionable material as the basis for the limits. The previous calculations did not account for the lead reflection of the walls of the cask. None of the results were compared to experimental benchmarks. This memo supersedes Reference 1 in its entirety. Waste Cask was not previously analyzed.

Validation of the Technical Bases per ANSI/ANS-8.1-1983 [1988] Requirements

An analysis was conducted of the original technical bases per the requirements of ANSI/ANS-8.1-1983 [Reaffirmed 1988]. The results are as follows:

(1) Describe the method with sufficient detail, clarity, and lack of ambiguity to

allow independent duplication of results.

The casks were modeled using the KENO-Va code to determine the maximum subcritical number of pins. Limits were determined for mixed oxide fuel pins, uranium oxide fuel pins, N Reactor fuel elements, and scrap in both the SERF Cask and the Waste Cask (also known as the 327 #30 cask) under both dry and wet conditions. Current regulations require that flooding of the cask be considered a normal event for offsite shipments.[2] Onsite shipments, on the other hand, may be allowed to consider water flooding as an accident scenario. Also, because regulations change, limits for both flooded and dry conditions are given in this memo. The determination of which limits to use is outside the scope of this memo and must be determined by the appropriate parties involved with the shipment.

The mixed oxide pins were modeled as 13.5" and 37" in length while the N Reactor elements were modeled as 26.1" in length. To be conservative, the mixed oxide pins were modeled at the most reactive composition allowed within the bounds of the fuel type. Uranium oxide fuels of the various enrichments specified without any plutonium had to be considered as the most reactive fuel type because the fuel compositions cited only a maximum wt% plutonium and not a minimum. (The fuel compositions are given in terms of the maximum wt% Pu in the total U + Pu, the minimum wt% ^{240}Pu in the total Pu, and the maximum ^{235}U enrichment in the total U.) In fact, the 94 wt% enriched UO_2 fuel is more reactive than the 25 wt% $Pu(10)O_2$-$U(94)O_2$ fuel. For the lower ^{235}U enrichments however, the mixed oxide fuel is more reactive. The minimum ^{240}Pu concentration specified in the fuel type was used throughout as ^{240}Pu acts as a poison. The cladding was not modeled to provide conservatism because steel and Zircaloy cladding also act as neutron poisons. A search was conducted with XSDRN to find the most reactive position of the pins in the casks by varying the distance from the pins to the lead in a water filled cask. As XSDRN is a two-dimensional code, the pins were centered in the cask and the $k_{infinity}$ calculated. The results, as shown in Table 1, indicate a tight fitting lead reflector is the most reactive. In the three-dimensional KENO runs, the pins do not always completely fill the cask. Therefore, the the cask was modeled with the bore in a horizontal position and the pins were modeled as laying on the bottom of the cask for maximum lead reflection within the constraints of the problem.

Table 1 Search for Optimum Pin Position with Respect to Lead Walls

Distance between Fuel Pins and Lead Walls	$k_{infinity}$
0.001 cm	1.353
0.1 cm	1.347
0.2 cm	1.339

Table 1 Search for Optimum Pin Position with Respect to Lead Walls

Distance between Fuel Pins and Lead Walls	$k_{infinity}$
0.3 cm	1.332
0.4 cm	1.324
0.5 cm	1.317
0.6 cm	1.308
0.7 cm	1.300
0.8 cm	1.292
0.9 cm	1.283
1.0 cm	1.275
1.1 cm	1.266
1.2 cm	1.257
1.3 cm	1.249
1.4 cm	1.240

The casks were also modeled conservatively. Table 2 compares the reported dimensions of the casks to the dimensions modeled in the KENO runs.

Table 2 Cask Dimensions

Dimension	SERF Cask		Waste Cask	
	As Reported	As Modeled	As Reported	As Modeled
Lead Thickness	8"	9"	5.625"	6.64"
Interior Diameter	8"	7.6"-8.4"	8.7"	8.3"-9.1"
Interior Length	108.75"	108.75"	64.25"	64.25"
Exterior Length	133.5"	134.6"	80"	81.1"

The lead thickness and exterior length were increased in the KENO model from the reported values because thicker lead provides more reflection. The interior diameter of the casks were varied over a range of ± 0.4" with respect to the reported value. The change in the diameter did not consistently increase or decrease the reactivity of the pins as shown in the Tables 3-6 and Tables 11-12. The amount of reflection from the lead is dependent on the distance from the lead to the pins (as indicated in Table 1, tight fitting lead is most reactive). The series of KENO calculations were performed for each fuel type to determine the optimum pitch and cask inner diameter for a subcritical configuration as given in the following sections. The optimum case was then rerun with 5000 neutrons per generation to decrease the statistical uncertainty of the result as shown in latter tables. These KENO search calculations, however, tracked only 300 neutrons in each generation to decrease the computer time needed for each run. All runs modeled 120 generations with 20 generations skipped in determining the uncertainty of the calculation.

(1.1) Water Moderated Mixed Oxide Fuel Pin Results

Water Moderated 13.5" Mixed Oxide Fuel Pin Results in the SERF Cask

Table 3 tabulates the results of the search for the optimum pitch and cask inner diameter for the water moderated 13.5" mixed oxide fuel pins in the SERF Cask. The pitch and cask diameter were varied until the optimum case was bounded to ensure the optimum had been found. The optimum case for the minimum subcritical number of pins is shown in bold type in the table.

As stated earlier and shown in Table 3, the 25 wt% $Pu(10)O_2$-$U(94)O_2$ pins were modeled with the maximum amount of ^{239}Pu present, i.e., 25 wt% $Pu(10)O_2$-$U(94)O_2$) and with the minimum amount of Pu present as $U(94)O_2$. Because of the high ^{235}U enrichment, the fuel without the Pu was most reactive with water moderation. The other fuel types were modeled only with the maximum amount of Pu present as this scenario is more reactive due to the lower ^{235}U enrichment.

Table 3 Optimization of Parameters for Water Moderated 13.5" Mixed Oxide Fuel Pins in the SERF Cask

Fuel Type[1]	Cask Diameter (in)	Pitch (in)	Number of Pins	Keno Keff	Keno Uncer-tainty	K95/95
<25 wt% Pu(>10)02-U(<94)02 Pins with a Maximum Fuel Diameter of 0.220 in						
	U(94)02 Pins					
Base Case	**8.0**	**0.91**	**52**	**0.94708**	**0.00457**	**0.979**
Increase Pitch	8.0	0.94	52	0.94428	0.00413	0.976
Decrease Pitch	8.0	0.87	52	0.94684	0.00453	0.979
Increase Cask Diameter	8.4	0.91	52	0.93944	0.00429	0.971
Increase Cask Diameter	8.4	0.94	52	0.94136	0.00452	0.973
Increase Cask Diameter	8.4	0.98	52	0.94193	0.00395	0.974
Increase Cask Diameter	8.4	1.02	52	0.94134	0.00447	0.973

Table 3 Optimization of Parameters for Water Moderated 13.5" Mixed Oxide Fuel Pins in the SERF Cask

Fuel Type[1]	Cask Diameter (in)	Pitch (in)	Number of Pins	Keno Keff	Keno Uncertainty	K95/95
Decrease Cask Diameter	7.6	0.91	52	0.94214	0.00487	0.974
	25 wt% Pu(10)02-U(94)02 Pins					
Optimum Case	8.0	0.91	52	0.92263	0.00429	0.954
<25 wt% Pu(>10)02-U(<36)02 Pins with a Maximum Fuel Diameter of 0.230 in						
Base Case	8.0	0.87	67	0.94796	0.00458	0.980
Increase Pitch	8.0	0.91	61	0.92464	0.00474	0.957
Decrease Pitch	8.0	0.83	67	0.94788	0.00438	0.980
Increase Cask Diameter	8.4	0.87	67	0.94837	0.00416	0.980
Increase Cask Diameter	8.4	0.91	67	0.93743	0.00441	0.969
Decrease Cask Diameter	7.6	0.87	60	0.91898	0.00427	0.951
Decrease Cask Diameter	7.6	0.79	67	0.93547	0.00468	0.967
<16 wt% Pu(>10)02-U(<41)02 Pins with a Maximum Fuel Diameter of 0.261 in						
Base Case	8.0	0.87	58	0.95380	0.00450	0.986
Increase Pitch	8.0	0.91	58	0.95149	0.00475	0.983
Increase Pitch	8.0	0.94	56	0.94466	0.00440	0.977
Decrease Pitch	8.0	0.83	58	0.93932	0.00442	0.971
Increase Cask Diameter	8.4	0.87	58	0.94506	0.00466	0.977
Increase Cask Diameter	8.4	0.91	58	0.95369	0.00437	0.986
Increase Cask Diameter	8.4	0.94	58	0.95551	0.00446	0.987
Increase Cask Diameter	8.4	0.98	56	0.94046	0.00446	0.972
Decrease Cask Diameter	7.6	0.87	58	0.94543	0.00435	0.977
<31 wt% Pu(>10)02-U(<0.72)02 Pins with a Maximum Fuel Diameter of 0.220 in						

Table 3 Optimization of Parameters for Water Moderated 13.5" Mixed Oxide Fuel Pins in the SERF Cask

Fuel Type[1]	Cask Diameter (in)	Pitch (in)	Number of Pins	Keno Keff	Keno Uncer-tainty	K95/95
Base Case	8.0	0.75	89	0.94448	0.00466	0.976
Increase Pitch	8.0	0.79	81	0.92897	0.00481	0.961
Decrease Pitch	8.0	0.71	90	0.93780	0.00498	0.970
Increase Cask Diameter	8.4	0.75	90	0.95465	0.00535	0.987
Increase Cask Diameter	8.4	0.79	89	0.95357	0.00466	0.985
Decrease Cask Diameter	7.6	0.75	81	0.92736	0.00501	0.959
Decrease Cask Diameter	7.6	0.71	90	0.93231	0.00434	0.964
<25 wt% Pu(>10)02-U(<68)02 Pins with a Maximum Fuel Diameter of 0.250 in						
Base Case	8.0	0.91	50	0.93027	0.00440	0.962
Increase Pitch	8.0	0.94	50	0.92081	0.00430	0.953
Decrease Pitch	8.0	0.87	50	0.91850	0.00451	0.950
Increase Cask Diameter	8.4	0.91	50	0.91982	0.00438	0.952
Increase Cask Diameter	8.4	0.94	50	0.92734	0.00447	0.959
Increase Cask Diameter	8.4	0.98	50	0.93383	0.00402	0.966
Increase Cask Diameter	8.4	1.02	50	0.92867	0.00487	0.961
Decrease Cask Diameter	7.6	0.91	50	0.92718	0.00453	0.959
<5 wt% Pu(>8)02-U(<9)02 Pins with a Maximum Fuel Diameter of 0.500 in						
Base Case	8.0	1.02	46	0.92790	0.00464	0.960
Increase Pitch	8.0	1.06	44	0.93154	0.00461	0.963
Decrease Pitch	8.0	0.98	46	0.92271	0.00509	0.955
Increase Cask Diameter	8.4	1.02	46	0.93468	0.00477	0.967
Increase Cask Diameter	8.4	1.06	46	0.95099	0.00465	0.983
Increase Cask Diameter	8.4	1.10	45	0.94320	0.00409	0.975

Table 3 Optimization of Parameters for Water Moderated 13.5" Mixed Oxide Fuel Pins in the SERF Cask

Fuel Type[1]	Cask Diameter (in)	Pitch (in)	Number of Pins	Keno Keff	Keno Uncer-tainty	K95/95
Decrease Cask Diameter	7.6	1.02	43	0.90600	0.00488	0.938
Decrease Cask Diameter	7.6	0.98	46	0.91276	0.00456	0.945
[1]The compositions show the maximum wt% Pu in U + Pu, the minimum wt% ^{240}Pu in Pu and the maximum wt % ^{235}U in U.						
Benchmark Statistics						
Bias = -0.00047						
Standard Deviation = 0.01406						
Variance = 0.00020						

Water Moderated 37" Mixed Oxide Fuel Pin Results in the SERF Cask

Table 4 tabulates the results of the search for the optimum pitch and cask inner diameter for the water moderated 37" mixed oxide fuel pins in the SERF Cask. The pitch and cask diameter were varied until the optimum case was bounded to ensure the optimum had been found. The optimum case for the minimum subcritical number of pins is shown in bold type in the table.

Table 4 Optimization of Parameters for Water Moderated 37" Mixed Oxide Fuel Pins in the SERF Cask

Fuel Type[1]	Cask Diameter (in)	Pitch (in)	Number of Pins	Keno Keff	Keno Uncer-tainty	K95/95
<25 wt% Pu(>10)O2-U(<94)O2 Pins with a Maximum Fuel Diameter of 0.220 in						
	U(94)O2 Pins					
Base Case	8.0	0.98	36	0.93256	0.00439	0.964
Increase Pitch	8.0	1.02	36	0.92674	0.00433	0.959
Decrease Pitch	8.0	0.94	36	0.92345	0.00434	0.955
Increase Cask Diameter	8.4	0.98	36	0.92418	0.00365	0.956
Decrease Cask Diameter	7.6	0.98	36	0.93410	0.00432	0.966
Decrease Cask Diameter	**7.6**	**1.02**	**36**	**0.94714**	**0.00408**	**0.979**
Decrease Cask Diameter	7.6	1.06	36	0.93830	0.00426	0.970

Table 4 Optimization of Parameters for Water Moderated 37" Mixed Oxide Fuel Pins in the SERF Cask

Fuel Type[1]	Cask Diameter (in)	Pitch (in)	Number of Pins	Keno Keff	Keno Uncertainty	K95/95
	25 wt% Pu(>10)02-U(<94)02 Pins					
Optimum Case	7.6	1.02	36	0.91741	0.00418	0.949
<25 wt% Pu(>10)02-U(<36)02 Pins with a Maximum Fuel Diameter of 0.230 in						
Base Case	8.0	0.98	44	0.94335	0.00424	0.975
Increase Pitch	8.0	1.02	44	0.93970	0.00374	0.971
Decrease Pitch	8.0	0.94	44	0.93612	0.00428	0.968
Increase Cask Diameter	8.4	0.98	44	0.92564	0.00397	0.957
Decrease Cask Diameter	**7.6**	**0.98**	**44**	**0.95197**	**0.00410**	**0.984**
Decrease Cask Diameter	7.6	1.02	43	0.93229	0.00350	0.964
<16 wt% Pu(>10)02-U(<41)02 Pins with a Maximum Fuel Diameter of 0.261 in						
Base Case	8.0	1.02	38	0.93765	0.00449	0.970
Second Base Case	8.0	1.06	38	0.93785	0.00425	0.970
Increase Pitch	8.0	1.10	38	0.93057	0.00386	0.962
Decrease Pitch	8.0	0.98	38	0.92793	0.00423	0.960
Increase Cask Diameter	8.4	1.02	38	0.93126	0.00397	0.963
Increase Cask Diameter	8.4	1.06	38	0.93626	0.00448	0.968
Decrease Cask Diameter	7.6	1.02	38	0.93973	0.00396	0.972
Decrease Cask Diameter	**7.6**	**1.06**	**38**	**0.94332**	**0.00430**	**0.975**
Decrease Cask Diameter	7.6	1.10	37	0.93915	0.00367	0.971
<31 wt% Pu(>10)02-U(<0.72)02 Pins with a Maximum Fuel Diameter of 0.220 in						
Base Case	8.0	0.91	54	0.94917	0.00416	0.981
Increase Pitch	8.0	0.94	54	0.94102	0.00442	0.973

Table 4 Optimization of Parameters for Water Moderated 37" Mixed Oxide Fuel Pins in the SERF Cask

Fuel Type[2]	Cask Diameter (in)	Pitch (in)	Number of Pins	Keno Keff	Keno Uncertainty	K95/95
Decrease Pitch	8.0	0.87	54	0.94238	0.00424	0.974
Increase Cask Diameter	8.4	0.91	54	0.93741	0.00365	0.969
Decrease Cask Diameter	7.6	0.91	54	0.94683	0.00427	0.979
<25 wt% Pu(>10)O2-U(<68)O2 Pins with a Maximum Fuel Diameter of 0.250 in						
Base Case	8.0	1.02	36	0.93468	0.00448	0.967
Increase Pitch	8.0	1.06	36	0.93310	0.00437	0.965
Decrease Pitch	8.0	0.98	36	0.92599	0.00500	0.958
Increase Cask Diameter	8.4	1.02	36	0.92707	0.00409	0.959
Decrease Cask Diameter	7.6	1.02	36	0.93832	0.00416	0.970
Decrease Cask Diameter	7.6	1.06	36	0.93789	0.00448	0.970
Decrease Cask Diameter	7.6	1.10	36	0.94236	0.00356	0.974
Decrease Cask Diameter	7.6	1.14	34	0.92023	0.00403	0.952
<5 wt% Pu(>8)O2-U(<9)O2 Pins with a Maximum Fuel Diameter of 0.500 in						
Base Case	8.0	1.30	26	0.94123	0.00439	0.973
Increase Pitch	8.0	1.34	26	0.93588	0.00405	0.968
Decrease Pitch	8.0	1.26	26	0.92981	0.00410	0.962
Increase Cask Diameter	8.4	1.30	26	0.92845	0.00375	0.960
Decrease Cask Diameter	7.6	1.30	26	0.92689	0.00391	0.959

[2] The compositions show the maximum wt% Pu in U + Pu, the minimum wt% ^{240}Pu in Pu and the maximum wt % ^{235}U in U.

Benchmark Statistics

Bias = -0.00047

Standard Deviation = 0.01405

Variance = 0.00020

Water Moderated 13.5" Mixed Oxide Fuel Pin Results in the Waste Cask

Table 5 tabulates the results of the search for the optimum pitch and cask inner diameter for the water moderated 13.5" mixed oxide fuel pins in the Waste Cask. The pitch and cask diameter were varied until the optimum case was bounded to ensure the optimum had been found. The optimum case for the *minimum* subcritical number of pins is shown in bold type in the table.

Table 5 Optimization of Parameters for Water Moderated 13.5" Mixed Oxide Fuel Pins in the Waste Cask

Fuel Type[3]	Cask Diameter (in)	Pitch (in)	Number of Pins	Keno Keff	Keno Uncertainty	K95/95
<25 wt% Pu(>10)O2-U(<94)O2 Pins with a Maximum Fuel Diameter of 0.220 in						
	U(94)O2 Pins					
Base Case	8.7	0.94	54	0.94683	0.00492	0.979
Second Base Case	8.7	0.98	54	0.94655	0.00437	0.978
Increase Pitch	8.7	1.02	54	0.94131	0.00447	0.973
Decrease Pitch	8.7	0.91	54	0.94517	0.00453	0.977
Increase Cask Diameter	9.1	0.94	54	0.94287	0.00498	0.975
Increase Cask Diameter	9.1	0.98	54	0.93667	0.00434	0.969
Increase Cask Diameter	9.1	1.02	54	0.94407	0.00530	0.976
Decrease Cask Diameter	**8.3**	**0.94**	**54**	**0.95242**	**0.00478**	**0.984**
Decrease Cask Diameter	8.3	0.98	54	0.94019	0.00481	0.972
	25 wt% Pu(10)O2-U(94)O2 Pins					
Optimum Case	8.3	0.94	54	0.92292	0.00453	0.955
<25 wt% Pu(>10)O2-U(<36)O2 Pins with a Maximum Fuel Diameter of 0.230 in						
Base Case	8.7	0.91	68	0.94146	0.00363	0.973
Increase Pitch	8.7	0.94	66	0.92825	0.00483	0.960
Decrease Pitch	8.7	0.87	68	0.93328	0.00430	0.965
Increase Cask Diameter	9.1	0.91	68	0.93197	0.00462	0.964

Table 5 Optimization of Parameters for Water Moderated 13.5" Mixed Oxide Fuel Pins in the Waste Cask

Fuel Type[3]	Cask Diameter (in)	Pitch (in)	Number of Pins	Keno Keff	Keno Uncer-tainty	K95/95
Increase Cask Diameter	9.1	0.94	68	0.94467	0.00434	0.977
Increase Cask Diameter	9.1	0.98	67	0.92752	0.00429	0.970
Decrease Cask Diameter	8.3	0.91	66	0.93639	0.00401	0.968
Decrease Cask Diameter	8.3	0.87	68	0.93855	0.00394	0.970
<16 wt% Pu(>10)02-U(<41)02 Pins with a Maximum Fuel Diameter of 0.261 in						
Base Case	8.7	0.98	58	0.95057	0.00426	0.982
Increase Pitch	8.7	1.02	56	0.92775	0.00493	0.960
Decrease Pitch	8.7	0.94	58	0.93504	0.00399	0.967
Increase Cask Diameter	9.1	0.98	58	0.94766	0.00505	0.980
Increase Cask Diameter	9.1	1.02	58	0.93269	0.00404	0.965
Decrease Cask Diameter	8.3	0.98	56	0.92989	0.00437	0.962
Decrease Cask Diameter	8.3	0.94	58	0.93477	0.00401	0.967
<31 wt% Pu(>10)02-U(<0.72)02 Pins with a Maximum Fuel Diameter of 0.220 in						
Base Case	8.7	0.83	87	0.95296	0.00466	0.985
Increase Pitch	8.7	0.87	79	0.92957	0.00428	0.961
Decrease Pitch	8.7	0.79	87	0.93996	0.00420	0.972
Increase Cask Diameter	9.1	0.83	87	0.95217	0.00481	0.984
Increase Cask Diameter	9.1	0.87	86	0.94611	0.00429	0.978
Decrease Cask Diameter	8.3	0.83	79	0.92739	0.00495	0.959
Decrease Cask Diameter	8.3	0.79	87	0.94053	0.00478	0.973
<25 wt% Pu(>10)02-U(<68)02 Pins with a Maximum Fuel Diameter of 0.250 in						

Table 5 Optimization of Parameters for Water Moderated 13.5" Mixed Oxide Fuel Pins in the Waste Cask

Fuel Type[3]	Cask Diameter (in)	Pitch (in)	Number of Pins	Keno Keff	Keno Uncertainty	K95/95
Base Case	8.7	0.94	56	0.94988	0.00446	0.982
Increase Pitch	8.7	0.98	56	0.94256	0.00462	0.974
Decrease Pitch	8.7	0.91	56	0.94499	0.00420	0.977
Increase Cask Diameter	9.1	0.94	56	0.94372	0.00456	0.976
Increase Cask Diameter	9.1	0.98	56	0.94556	0.00483	0.978
Increase Cask Diameter	9.1	1.02	56	0.94140	0.00404	0.973
Decrease Cask Diameter	8.3	0.94	56	0.94346	0.00393	0.975
<5 wt% Pu(>8)02-U(<9)02 Pins with a Maximum Fuel Diameter of 0.500 in						
Base Case	8.7	1.10	46	0.94932	0.00476	0.981
Increase Pitch	8.7	1.14	45	0.94462	0.00420	0.976
Decrease Pitch	8.7	1.06	46	0.93913	0.00460	0.971
Increase Cask Diameter	9.1	1.10	46	0.94685	0.00507	0.979
Increase Cask Diameter	9.1	1.14	46	0.94670	0.00455	0.979
Decrease Cask Diameter	8.3	1.10	44	0.92992	0.00417	0.962
Decrease Cask Diameter	8.3	1.06	46	0.93269	0.00514	0.965
[3]The compositions show the maximum wt% Pu in U + Pu, the minimum wt% ^{240}Pu in Pu and the maximum wt % ^{235}U in U.						
Benchmarks Statistics						
Bias = -0.00047						
Standard Deviation = 0.01406						
Variance = 0.00020						

Water Moderated 37" Mixed Oxide Fuel Pin Results in the Waste Cask

Table 6 tabulates the results of the search for the optimum pitch and cask inner diameter for the water moderated 37" mixed oxide fuel pins in the Waste Cask. The

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pitch and cask diameter were varied until the optimum case was bounded to ensure the optimum had been found. The optimum case for the minimum subcritical number of pins is shown in bold type in the table.

Table 6 Optimization of Parameters for Water Moderated 37" Mixed Oxide Fuel Pins in the Waste Cask

Fuel Type[4]	Cask Diameter (in)	Pitch (in)	Number of Pins	Keno Keff	Keno Uncertainty	K95/95
<25 wt% Pu(>10)02-U(<94)02 Pins with a Maximum Fuel Diameter of 0.220 in						
	U(94)02 Pins					
Base Case	8.7	0.98	40	0.94278	0.00407	0.975
	8.7	1.02	40	0.94134	0.00449	0.973
Increase Pitch	8.7	1.06	40	0.93597	0.00410	0.968
Decrease Pitch	8.7	0.94	40	0.93634	0.00438	0.968
Increase Cask Diameter	9.1	0.98	40	0.93402	0.00463	0.966
Increase Cask Diameter	9.1	1.02	40	0.93975	0.00455	0.972
Decrease Cask Diameter	8.3	0.98	40	0.94633	0.00420	0.978
Decrease Cask Diameter	**8.3**	**1.02**	**40**	**0.95136**	**0.00404**	**0.983**
Decrease Cask Diameter	8.3	1.06	40	0.94988	0.00424	0.982
	25 wt% Pu(10)02-U(94)02 Pins					
Optimum Case	8.3	1.02	40	0.92739	0.00441	0.959
<25 wt% Pu(>10)02-U(<36)02 Pins with a Maximum Fuel Diameter of 0.230 in						
Base Case	**8.7**	**1.02**	**48**	**0.95393**	**0.00430**	**0.986**
Increase Pitch	8.7	1.06	48	0.94581	0.00375	0.977
Decrease Pitch	8.7	0.98	48	0.94795	0.00399	0.980
Increase Cask Diameter	9.1	1.02	48	0.94439	0.00367	0.976
Decrease Cask Diameter	8.3	1.02	48	0.94952	0.00392	0.981
<16 wt% Pu(>10)02-U(<41)02 Pins with a Maximum Fuel Diameter of 0.261 in						

Table 6 Optimization of Parameters for Water Moderated 37" Mixed Oxide Fuel Pins in the Waste Cask

Fuel Type[a]	Cask Diameter (in)	Pitch (in)	Number of Pins	Keno Keff	Keno Uncer- tainty	K95/95
Base Case	8.7	1.10	40	0.93350	0.00390	0.965
Second Base Case	8.7	1.14	40	0.93274	0.00383	0.964
Increase Pitch	8.7	1.18	40	0.91997	0.00377	0.952
Decrease Pitch	8.7	1.05	40	0.93194	0.00394	0.964
Increase Cask Diameter	9.1	1.10	40	0.92689	0.00401	0.959
Increase Cask Diameter	9.1	1.14	40	0.91568	0.00378	0.947
Decrease Cask Diameter	**8.3**	**1.10**	**40**	**0.93892**	**0.00470**	**0.971**
Decrease Cask Diameter	8.3	1.14	40	0.92174	0.00426	0.954
<31 wt% Pu(>10)02-U(<0.72)02 Pins with a Maximum Fuel Diameter of 0.220 in						
Base Case	8.7	0.94	58	0.94547	0.00400	0.977
	8.7	0.98	58	0.94508	0.00440	0.977
Increase Pitch	8.7	1.02	56	0.92642	0.00440	0.958
Decrease Pitch	8.7	0.91	58	0.94075	0.00407	0.973
Increase Cask Diameter	9.1	0.94	58	0.93989	0.00423	0.972
	9.1	0.98	58	0.94312	0.00427	0.975
Decrease Cask Diameter	**8.3**	**0.94**	**58**	**0.95440**	**0.00362**	**0.986**
	8.3	0.98	56	0.94176	0.00413	0.974
<25 wt% Pu(>10)02-U(<68)02 Pins with a Maximum Fuel Diameter of 0.250 in						
Base Case	8.7	1.02	40	0.94901	0.00426	0.981
Increase Pitch	8.7	1.06	40	0.94333	0.00425	0.975
Decrease Pitch	8.7	0.98	40	0.94339	0.00406	0.975
Increase Cask Diameter	9.1	1.02	40	0.92271	0.00375	0.954

Table 6 Optimization of Parameters for Water Moderated 37" Mixed Oxide Fuel Pins in the Waste Cask

Fuel Type[a]	Cask Diameter (in)	Pitch (in)	Number of Pins	Keno Keff	Keno Uncertainty	K95/95
Decrease Cask Diameter	8.3	1.02	40	0.95404	0.00433	0.986
Decrease Cask Diameter	8.3	1.06	40	0.95242	0.00438	0.984
<5 wt% Pu(>8)O2-U(<9)O2 Pins with a Maximum Fuel Diameter of 0.500 in						
Base Case	8.7	1.34	28	0.94192	0.00379	0.974
Second Base Case	8.7	1.38	28	0.94315	0.00375	0.975
Increase Pitch	8.7	1.42	28	0.93779	0.00384	0.969
Decrease Pitch	8.7	1.30	28	0.93893	0.00412	0.971
Increase Cask Diameter	9.1	1.34	28	0.92536	0.00434	0.957
Increase Cask Diameter	9.1	1.38	28	0.93309	0.00386	0.965
Decrease Cask Diameter	8.3	1.34	28	0.94721	0.00389	0.979
Decrease Cask Diameter	8.3	1.38	28	0.93812	0.00378	0.970
[a]The compositions show the maximum wt% Pu in U + Pu, the minimum wt% ^{240}Pu in Pu and the maximum wt % ^{235}U in U.						
Benchmarks Statistics						
Bias = -0.00047						
Standard Deviation = 0.01406						
Variance = 0.00020						

Final Results for 13.5" and 37" Mixed Oxide Fuel Pins in the SERF Cask

As stated earlier, the scoping calculations shown in Tables 3 and 4 determined the optimum parameters for each fuel type using the KENO code tracking 300 neutrons per generation. The optimum case as shown in boldface in these tables was rerun using KENO tracking 5000 neutrons per generation to reduce the statistical uncertainty in the calculation. The $K_{95/95}$ for the minimum subcritical number of pins which *would geometrically fit into the SERF Cask for each pin type is given in Table 7.* The pin batch limit is set at 50% of this subcritical number of pins to ensure subcriticality in the case of a double batch. The SERF Cask was also modeled containing one batch of each type of fuel to ensure $K_{95/95}$ is less than 0.95 under normal operating conditions. These results are shown in Table 8.

Table 7 Water Moderated Mixed Oxide Fuel Pins in the SERF Cask
Minimum Subcritical Pin Configurations

Fuel Type[5]	Fuel Length (in)	Maximum Diameter (in)	Pitch (in)	Number of Pins	Keno Keff	Keno Uncertainty	K95/95
<25 wt% Pu(>10)02-U(<94)02	13.5	0.220	0.91	52	0.95126	0.00091	0.987
<25 wt% Pu(>10)02-U(<36)02	13.5	0.230	0.87	67	0.94954	0.00109	0.986
<16 wt% Pu(>10)02-U(<41)02	13.5	0.261	0.94	58	0.95151	0.00110	0.988
<31 wt% Pu(>10)02-U(<0.72)02	13.5	0.220	0.75	90	0.95233	0.00111	0.988
<25 wt% Pu(>10)02-U(<68)02	13.5	0.250	0.98	50	0.92712	0.00113	0.963
<5 wt% Pu(>8)02-U(<9)02	13.5	0.500	1.06	46	0.94584	0.00115	0.982
<25 wt% Pu(>10)02-U(<94)02	37	0.220	1.02	36	0.94083	0.00111	0.977
<25 wt% Pu(>10)02-U(<36)02	37	0.230	0.98	44	0.94523	0.00102	0.981
<16 wt% Pu(>10)02-U(<41)02	37	0.261	1.06	38	0.94267	0.00104	0.979
<31 wt% Pu(>10)02-U(<0.72)02	37	0.220	0.91	54	0.94483	0.00095	0.981
<25 wt% Pu(>10)02-U(<68)02	37	0.250	1.10	36	0.93686	0.00094	0.973
<5 wt% Pu(>8)02-U(<9)02	37	0.500	1.30	26	0.93379	0.00107	0.970

[5] The compositions show the maximum wt% Pu in U + Pu, the minimum wt% Pu-240 in Pu, and the maximum wt% U-235 in U.

Benchmark Statistics

Bias = 0.00101

Standard Deviation = 0.01548

Variance = 0.00024

Table 8 Water Moderated Mixed Oxide Fuel Pins in the SERF Cask Reactivity at Pin Limit

Fuel Type[6]	Fuel Length (in)	Maximum Diameter (in)	Pitch (in)	Number of Pins	Keno Keff	Keno Uncer-tainty	K95/95
<25 wt% Pu(>10)02-U(<94)02	13.5	0.220	0.91	26	0.72667	0.00096	0.763
<25 wt% Pu(>10)02-U(<36)02	13.5	0.230	0.87	33	0.74074	0.00114	0.777
<16 wt% Pu(>10)02-U(<41)02	13.5	0.261	0.94	29	0.74910	0.00103	0.785
<31 wt% Pu(>10)02-U(<0.72)02	13.5	0.220	0.75	45	0.75494	0.00113	0.791
<25 wt% Pu(>10)02-U(<68)02	13.5	0.250	0.98	25	0.72362	0.00091	0.760
<5 wt% Pu(>8)02-U(<9)02	13.5	0.500	1.06	23	0.75282	0.00110	0.789
<25 wt% Pu(>10)02-U(<94)02	37	0.220	1.02	18	0.69521	0.00102	0.731
<25 wt% Pu(>10)02-U(<36)02	37	0.230	0.98	22	0.70753	0.00088	0.744
<16 wt% Pu(>10)02-U(<41)02	37	0.261	1.06	19	0.71070	0.00091	0.747
<31 wt% Pu(>10)02-U(<0.72)02	37	0.220	0.91	27	0.71479	0.00104	0.751
<25 wt% Pu(>10)02-U(<68)02	37	0.250	1.10	18	0.70296	0.00093	0.739
<5 wt% Pu(>8)02-U(<9)02	37	0.500	1.30	13	0.70875	0.00098	0.745

[6] The compositions show the maximum wt% Pu in U + Pu, the minimum wt% ^{240}Pu in Pu, and the maximum wt% ^{235}U in U

Benchmark Statistics

Bias = 0.00101

Standard Deviation = 0.01548

Variance = 0.00024

Final Results for 13.5" and 37" Mixed Oxide Fuel Pins in the Waste Cask

As stated earlier, the scoping calculations shown in Tables 5 and 6 determined the optimum parameters for each fuel type in the Waste Cask using the KENO code tracking 300 neutrons per generation. The optimum case as shown in boldface in these tables was rerun using KENO tracking 5000 neutrons per generation to reduce the statistical uncertainty in the calculation. The $K_{95/95}$ for the minimum subcritical number of pins which would geometrically fit into the Waste Cask for each pin type is given in Table 9. The pin batch limit is set at 50% of this subcritical number of pins to ensure subcriticality in the case of a double batch.

The Waste Cask was also modeled containing one batch of each type of fuel to ensure $K_{95/95}$ is less than 0.95 under normal operating conditions. These results are shown in Table 10.

Table 9 Water Moderated Mixed Oxide Fuel Pins in the Waste Cask Minimum Subcritical Pin Configurations

Fuel Type[7]	Fuel Length (in)	Maximum Diameter (in)	Pitch (in)	Number of Pins	Keno Keff	Keno Uncertainty	K95/95
<25 wt% Pu(>10)O2-U(<94)O2	13.5	0.220	0.94	54	0.94832	0.00108	0.984
<25 wt% Pu(>10)O2-U(<36)O2	13.5	0.230	0.94	68	0.94222	0.00097	0.978
<16 wt% Pu(>10)O2-U(<41)O2	13.5	0.261	0.98	58	0.94191	0.00088	0.978
<31 wt% Pu(>10)O2-U(<0.72)O2	13.5	0.220	0.83	87	0.94462	0.00114	0.981
<25 wt% Pu(>10)O2-U(<68)O2	13.5	0.250	0.94	56	0.94751	0.00094	0.984
<5 wt% Pu(>8)O2-U(<9)O2	13.5	0.500	1.10	46	0.94372	0.00116	0.980
<25 wt% Pu(>10)O2-U(<94)O2	37	0.220	1.02	40	0.95161	0.00104	0.988
<25 wt% Pu(>10)O2-U(<36)O2	37	0.230	1.02	48	0.94473	0.00102	0.981
<16 wt% Pu(>10)O2-U(<41)O2	37	0.261	1.10	40	0.93622	0.00095	0.972
<31 wt% Pu(>10)O2-U(<0.72)O2	37	0.220	0.94	58	0.95143	0.00102	0.988
<25 wt% Pu(>10)O2-U(<68)O2	37	0.250	1.02	40	0.94864	0.00092	0.985
<5 wt% Pu(>8)O2-U(<9)O2	37	0.500	1.34	28	0.94348	0.00098	0.980

[7] The compositions show the maximum wt% Pu in U + Pu, the minimum wt% ^{240}Pu in Pu, and the maximum wt% ^{235}U in U.

Benchmark Statistics

Bias = 0.00101

Standard Deviation = 0.01548

Variance = 0.00024

Table 10 Water Moderated Mixed Oxide Fuel Pins in the Waste Cask Reactivity at Pin Limit

Fuel Type[a]	Fuel Length (in)	Maximum Diameter (in)	Pitch (in)	Number of Pins	Keno Keff	Keno Uncertainty	K95/95
<25 wt% Pu(>10)O_2-U(<94)O_2	13.5	0.220	0.94	27	0.73031	0.00115	0.766
<25 wt% Pu(>10)O_2-U(<36)O_2	13.5	0.230	0.94	34	0.74868	0.00095	0.785
<16 wt% Pu(>10)O_2-U(<41)O_2	13.5	0.261	0.98	29	0.74026	0.00090	0.776
<31 wt% Pu(>10)O_2-U(<0.72)O_2	13.5	0.220	0.83	43	0.74975	0.00098	0.786
<25 wt% Pu(>10)O_2-U(<68)O_2	13.5	0.250	0.94	28	0.74135	0.00093	0.777
<5 wt% Pu(>8)O_2-U(<9)O_2	13.5	0.500	1.10	23	0.75034	0.00111	0.786
<25 wt% Pu(>10)O_2-U(<94)O_2	37	0.220	1.02	20	0.71390	0.00099	0.750
<25 wt% Pu(>10)O_2-U(<36)O_2	37	0.230	1.02	24	0.72303	0.00104	0.759
<16 wt% Pu(>10)O_2-U(<41)O_2	37	0.261	1.10	20	0.71209	0.00096	0.748
<31 wt% Pu(>10)O_2-U(<0.72)O_2	37	0.220	0.94	29	0.72795	0.00102	0.764
<25 wt% Pu(>10)O_2-U(<68)O_2	37	0.250	1.02	20	0.71929	0.00103	0.755
<5 wt% Pu(>8)O_2-U(<9)O_2	37	0.500	1.34	14	0.71857	0.00094	0.755

[a] The compositions show the maximum wt% Pu in U + Pu, the minimum wt% ^{240}Pu in Pu, and the maximum wt% ^{235}U in U

Benchmark Statistics
Bias = 0.00101
Standard Deviation = 0.01548
Variance = 0.00024

(1.2) Water Moderated N Reactor Fuel Element Results

The most reactive number and configuration of N Reactor inner fuel pins, outer fuel pins and fuel assemblies were found. First, a search was conducted on the optimum pitch and cask inner diameter for each element and cask type. These results shown in the following tables indicate the N Reactor fuel elements will not reach criticality in either of the casks without the addition of other fissionable material.

Water Moderated 26.1" N Reactor Fuel Element Results in the SERF Cask

Table 11 tabulates the results of the search for the optimum pitch and cask inner diameter for the water moderated 26.1" N Reactor fuel elements in the SERF Cask. The pitch and cask diameter were varied until the optimum case was bounded to ensure the optimum had been found. The optimum case is shown in bold type in the table.

Table 11 Optimization of Parameters for Water Moderated 26.1" N Reactor Fuel Elements in the SERF Cask

Fuel Type	Cask Diameter (in)	Pitch (in)	Number of Pins	Keno Keff	Keno Uncertainty	K95/95
Mark IA Fuel Assemblies with a Maximum Outer Diameter of 2.35 in						
Base Case	8.0	2.68	28	0.73486	0.00364	0.767
Increase Pitch	8.0	2.87	24	0.72193	0.00298	0.754
Decrease Pitch	8.0	2.48	32	0.70974	0.00329	0.741
Increase Cask Diameter	8.4	2.68	28	0.74651	0.00307	0.778
Increase Cask Diameter	8.4	2.80	28	0.74705	0.00329	0.779
Increase Cask Diameter	8.4	2.83	24	0.72689	0.00359	0.759
Decrease Cask Diameter	7.6	2.68	24	0.70146	0.00345	0.733
Decrease Cask Diameter	7.6	2.52	28	0.69377	0.00321	0.725
Tight Hexagonal Pitch	8.4	2.35	48	0.75621	0.00299	0.788
Increase Pitch	8.4	2.75	33	0.76179	0.00313	0.793
Increase Pitch	**8.4**	**3.14**	**25**	**0.80051**	**0.00336**	**0.832**
Increase Pitch	8.4	3.54	20	0.73461	0.00301	0.766
Mark IA Inner Fuel Elements with a Maximum Outer Diameter of 1.17 in						
Base Case	8.0	1.61	76	0.69085	0.00314	0.723
Increase Pitch	8.0	1.65	72	0.68867	0.00301	0.720
Decrease Pitch	8.0	1.57	80	0.68718	0.00322	0.719
Increase Cask Diameter	8.4	1.61	80	0.70430	0.00310	0.736

Table 11 Optimization of Parameters for Water Moderated 26.1" N Reactor Fuel Elements in the SERF Cask

Fuel Type	Cask Diameter (in)	Pitch (in)	Number of Pins	Keno Keff	Keno Uncer-tainty	K95/95
Increase Cask Diameter	8.4	1.65	80	0.70638	0.00304	0.738
Increase Cask Diameter	8.4	1.69	76	0.71187	0.00259	0.744
Increase Cask Diameter	8.4	1.73	72	0.70231	0.00288	0.734
Decrease Cask Diameter	7.6	1.61	68	0.66748	0.00322	0.699
Decrease Cask Diameter	7.6	1.50	80	0.66884	0.00278	0.700
Tight Packed Pitch	8.4	1.17	195	0.56646	0.00294	0.598
Mark IA Outer Fuel Elements with a Maximum Outer Diameter of 2.35 in						
Base Case	8.0	2.36	36	0.73456	0.00294	0.766
Increase Pitch	8.0	2.40	32	0.73301	0.00291	0.765
Decrease Pitch	8.0	2.32	36	0.72267	0.00286	0.754
Increase Cask Diameter	8.4	2.36	36	0.73602	0.00323	0.768
Increase Cask Diameter	8.4	2.40	36	0.75140	0.00302	0.783
Increase Cask Diameter	8.4	2.44	36	0.75568	0.00294	0.787
Increase Cask Diameter	8.4	2.48	32	0.74874	0.00327	0.780
Decrease Cask Diameter	7.6	2.36	32	0.71104	0.00348	0.743
Decrease Cask Diameter	7.6	2.24	36	0.70182	0.00276	0.733
Tight Packed Pitch	8.4	2.35	48	0.80439	0.00330	0.836
Increase Pitch	8.4	2.75	33	0.81739	0.00317	0.849
Increase Pitch	8.4	3.14	25	0.77543	0.00287	0.807
Benchmark Statistics						
Bias = -0.00047						
Standard Deviation = 0.01406						
Variance = 0.00020						

Water Moderated 26.1" N Reactor Fuel Element Results in the Waste Cask

Table 12 tabulates the results of the search for the optimum pitch and cask inner diameter for the water moderated 26.1" N Reactor fuel elements in the Waste Cask. The pitch and cask diameter were varied until the optimum case was bounded to ensure the optimum had been found. The optimum case is shown in bold type in the table.

Table 12 Optimization of Parameters for Water Moderated 26.1" N Reactor Fuel Elements in the Waste Cask

Fuel Type	Cask Diameter (in)	Pitch (in)	Number of Pins	Keno Keff	Keno Uncer-tainty	K95/95
Mark IA Fuel Assemblies with a Maximum Outer Diameter of 2.35 in						
Base Case	8.7	2.72	16	0.74406	0.00340	0.776
Increase Pitch	8.7	2.91	14	0.74009	0.00313	0.772
Decrease Pitch	8.7	2.68	16	0.74005	0.00325	0.772
Increase Cask Diameter	9.1	2.72	16	0.75509	0.00297	0.787
Increase Cask Diameter	9.1	2.83	16	0.76251	0.00291	0.794
Increase Cask Diameter	9.1	2.91	14	0.74674	0.00303	0.778
Decrease Cask Diameter	8.3	2.72	14	0.71742	0.00313	0.749
Decrease Cask Diameter	8.3	2.60	16	0.71930	0.00306	0.751
Tight Packed Pitch	9.1	2.35	33	0.77212	0.00296	0.804
Increase Pitch	**9.1**	**2.75**	**23**	**0.84185**	**0.00358**	**0.874**
Increase Pitch	9.1	3.14	17	0.81285	0.00338	0.844
Mark IA Inner Fuel Elements with a Maximum Outer Diameter of 1.17 in						
Base Case	8.7	1.69	40	0.69884	0.00290	0.730
Increase Pitch	8.7	1.73	38	0.68940	0.00282	0.721
Decrease Pitch	8.7	1.65	42	0.69632	0.00296	0.728
Increase Cask Diameter	9.1	1.69	42	0.70934	0.00287	0.741
Increase Cask Diameter	**9.1**	**1.73**	**42**	**0.71045**	**0.00288**	**0.742**
Increase Cask Diameter	9.1	1.77	40	0.70304	0.00250	0.735
Decrease Cask Diameter	8.3	1.69	36	0.67782	0.00323	0.709
Decrease Cask Diameter	8.3	1.57	42	0.68384	0.00313	0.716

Table 12 Optimization of Parameters for Water Moderated 26.1" N Reactor Fuel Elements in the Waste Cask

Fuel Type	Cask Diameter (in)	Pitch (in)	Number of Pins	Keno Keff	Keno Uncertainty	K95/95
Tight Packed Pitch	9.1	1.17	136	0.57859	0.00285	0.610
Mark IA Outer Fuel Elements with a Maximum Outer Diameter of 2.35 in						
Base Case	8.7	2.56	18	0.75677	0.00289	0.788
Increase Pitch	8.7	2.72	16	0.74814	0.00314	0.780
Decrease Pitch	8.7	2.52	18	0.75093	0.00360	0.783
Increase Cask Diameter	9.1	2.56	18	0.75463	0.00292	0.786
Increase Cask Diameter	9.1	2.68	18	0.76203	0.00276	0.794
Increase Cask Diameter	9.1	2.83	16	0.76123	0.00312	0.793
Increase Cask Diameter	9.1	3.03	14	0.73130	0.00297	0.763
Decrease Cask Diameter	8.3	2.56	16	0.73656	0.00334	0.768
Decrease Cask Diameter	8.3	2.44	18	0.73153	0.00334	0.763
Tight Packed Pitch	9.1	2.35	33	0.81869	0.00304	0.850
Increase Pitch	9.1	2.75	23	0.83287	0.00328	0.864
Increase Pitch	9.1	3.14	17	0.78940	0.00276	0.821
Benchmark Statistics						
Bias = -0.00047						
Standard Deviation = 0.01406						
Variance = 0.00020						

The most reactive configuration found above for each cask type was also modeled in KENO tracking 5000 neutrons per generation. These results are given in Table 13.

Table 13 N Reactor Water Moderated Fuel Element Limits in the SERF and the Waste Cask

Fuel Type	Cask Diameter (in)	Pitch (in)[9]	Number of Pins	Keno Keff	Keno Uncertainty	K95/95
SERF Cask						
Mark IA Inner Fuel Elements	8.4	1.69 Sq	76	0.70692	0.00072	0.743
Mark IA Outer Fuel Elements	8.4	2.75 Hex	33	0.81821	0.00082	0.854
Mark IA Fuel Assemblies	8.4	3.14 Hex	25	0.80117	0.00073	0.837
Waste Cask						
Mark IA Inner Fuel Elements	9.1	1.73 Sq	42	0.71159	0.00061	0.748
Mark IA Outer Fuel Elements	9.1	2.75 Hex	23	0.83400	0.00081	0.870
Mark IA Fuel Assemblies	9.1	2.75 Hex	23	0.83968	0.00090	0.876

[9] A square pitch arrangement is designated by a Sq and a hexagonal, tight-packed pitch arrangement is designated by Hex.

Benchmark Statistics

Bias = 0.00101

Standard Deviation = 0.01548

Variance = 0.00024

The preceding results indicate that N Reactor fuel elements will remain subcritical by themselves in either cask. However, criticality can occur when mixed oxide or other types or pins are also in the cask. Therefore, a finite batch limit is needed such that a fraction of a batch of N Reactor elements can be added to a fraction of a batch of another type of pins.

To determine this limit, first N Reactor fuel elements were added to a single batch of mixed oxide fuel to ensure subcriticality of a double batch. Two compositions of mixed oxide pins were used as shown in the following tables. The fuel composition that was most reactive at the pin limit was modeled because of its high reactivity. The fuel type that occupied the smallest volume under optimum conditions was also modeled as this type left a maximum amount of space for N Reactor elements in the cask. Each case was run with the mixed oxide pins at the optimum pitch found above and at a slightly reduced pitch. The pitch was reduced until the value of $K_{95/95}$ was statistically reduced indicating that a further reduction in pitch and hence the addition of more N Reactor elements would be less reactive. The N Reactor elements were modeled with a tight packed (hexagonal) pitch and totally filled the cask around the mixed oxide pins. The inner cask diameter was increased to 8.4" for the SERF Cask and 9.1" for the Waste Cask to increase the number of N Reactor elements. Water moderation was modeled throughout. The results are shown in Tables 14 and 15.

Final Results for N Reactor Fuel Elements + Mixed Oxide Pins in the SERF Cask

As shown in Table 14, the addition of the maximum number of N Reactor fuel elements geometrically possible to a single batch of mixed oxide fuel pins in the SERF Cask does not cause criticality. The maximum value of $K_{95/95}$ is less than 0.91 leaving a 90 mk margin of subcriticality.

Table 14 N Reactor Fuel Elements plus Mixed Oxide Fuel Pins in the SERF Cask

Number and Type of Fuel Pins in Cask(a)	FBR Pitch	Keno Keff	Keno Uncer-tainty	K95/95
Add Inner Fuel Elements to Pins with Most Reactive Limit				
45 13.5" <31 wt% Pu(>10)02-U(<0.72)02 Pins + 166 Mark 1A Inner Elements	0.75	0.84895	0.00101	0.885
45 13.5" <31 wt% Pu(>10)02-U(<0.72)02 Pins + 168 Mark 1A Inner Elements	0.71	0.83592	0.00099	0.872
Add Inner Fuel Elements to Pins with Smallest Volume at Limit				
26 13.5" <25 wt% Pu(>10)02-U(<94)02 Pins + 169 Mark 1A Inner Elements	0.91	0.85036	0.00097	0.886
26 13.5" <25 wt% Pu(>10)02-U(<94)02 Pins + 171 Mark 1A Inner Elements	0.87	0.84479	0.00102	0.881
Add Outer Fuel Elements to Pins with Most Reactive Limit				
45 13.5" <31 wt% Pu(>10)02-U(<0.72)02 Pins + 43 Mark 1A Outer Elements	0.75	0.89597	0.00105	0.932
45 13.5" <31 wt% Pu(>10)02-U(<0.72)02 Pins + 43 Mark 1A Outer Elements	0.71	0.89335	0.00109	0.929
Add Outer Fuel Elements to Pins with Smallest Volume at Limit				
26 13.5" <25 wt% Pu(>10)02-U(<94)02 Pins + 38 Mark 1A Outer Elements	0.91	0.90249	0.00106	0.939
26 13.5" <25 wt% Pu(>10)02-U(<94)02 Pins + 39 Mark 1A Outer Elements	0.87	0.90140	0.00129	0.938
26 13.5" <25 wt% Pu(>10)02-U(<94)02 Pins + 39 Mark 1A Outer Elements	0.83	0.90092	0.00113	0.937
26 13.5" <25 wt% Pu(>10)02-U(<94)02 Pins + 40 Mark 1A Outer Elements	0.79	0.89896	0.00116	0.935
Add Fuel Assemblies to Pins with Most Reactive Limit				
45 13.5" <31 wt% Pu(>10)02-U(<0.72)02 Pins + 43 Mark 1A Fuel Assemblies	0.75	0.85137	0.00090	0.887
45 13.5" <31 wt% Pu(>10)02-U(<0.72)02 Pins + 43 Mark 1A Fuel Assemblies	0.71	0.84212	0.00099	0.878

Table 14 N Reactor Fuel Elements plus Mixed Oxide Fuel Pins in the SERF Cask

Number and Type of Fuel Pins in Cask[10]	FBR Pitch	Keno Keff	Keno Uncertainty	K95/95
Add Fuel Assemblies to Pins with Smallest Volume at Limit				
26 13.5" <25 wt% Pu(>10)O2-U(<94)O2 Pins + 38 Mark 1A Fuel Assemblies	0.91	0.85603	0.00103	0.892
26 13.5" <25 wt% Pu(>10)O2-U(<94)O2 Pins + 39 Mark 1A Fuel Assemblies	0.87	0.85039	0.00090	0.886
[10] The mixed oxide fuel compositions show the maximum wt% Pu in U + Pu, the minimum wt% ^{240}Pu in Pu, and the maximum wt% ^{235}U in U.				
Benchmark Statistics				
Bias = 0.00101				
Standard Deviation = 0.01548				
Variance = 0.00024				

Final Results for N Reactor Fuel Elements + Mixed Oxide Pins in the Waste Cask

As shown in Table 15, the addition of the maximum number of N Reactor fuel elements geometrically possible to a single batch of mixed oxide fuel pins in the Waste Cask does not cause criticality. The maximum value of $K_{95/95}$ is less than 0.93 leaving a 70 mk margin of subcriticality.

Table 15 N Reactor Fuel Elements + Mixed Oxide Pins in the Waste Cask

Number and Type of Fuel Pins in Cask[11]	FBR Pitch	Keno Keff	Keno Uncertainty	K95/95
Add Inner Fuel Elements to Pins with Most Reactive Limit				
23 13.5" <5 wt% Pu(>8)O2-U(<9)O2 Pins + 86 Mark 1A Inner Elements	1.10	0.84382	0.00104	0.880
23 13.5" <5 wt% Pu(>8)O2-U(<9)O2 Pins + 88 Mark 1A Inner Elements	1.06	0.83094	0.00099	0.867
Add Inner Fuel Elements to Pins with Smallest Volume at Limit				
27 13.5" <25 wt% Pu(>10)O2-U(<94)O2 Pins + 90 Mark 1A Inner Elements	0.94	0.86157	0.00123	0.898
27 13.5" <25 wt% Pu(>10)O2-U(<94)O2 Pins + 91 Mark 1A Inner Elements	0.91	0.85671	0.00095	0.893
Add Outer Fuel Elements to Pins with Most Reactive Limit				
23 13.5" <5 wt% Pu(>8)O2-U(<9)O2 Pins + 21 Mark 1A Outer Elements	1.10	0.90230	0.00091	0.938
23 13.5" <5 wt% Pu(>8)O2-U(<9)O2 Pins + 21 Mark 1A Outer Elements	1.06	0.89944	0.00094	0.936
Add Outer Fuel Elements to Pins with Smallest Volume at Limit				

Table 15 N Reactor Fuel Elements + Mixed Oxide Pins in the Waste Cask

Number and Type of Fuel Pins in Cask[11]	FBR Pitch	Keno Keff	Keno Uncer-tainty	K95/95
27 13.5" <25 wt% Pu(>10)02-U(<94)02 Pins + 22 Mark 1A Outer Elements	0.94	0.91439	0.00092	0.950
27 13.5" <25 wt% Pu(>10)02-U(<94)02 Pins + 22 Mark 1A Outer Elements	0.91	0.91624	0.00093	0.952
27 13.5" <25 wt% Pu(>10)02-U(<94)02 Pins + 22 Mark 1A Outer Elements	0.87	0.91632	0.00094	0.952
27 13.5" <25 wt% Pu(>10)02-U(<94)02 Pins + 23 Mark 1A Outer Elements	0.83	0.91359	0.00090	0.950
Add Fuel Assemblies to Pins with Most Reactive Limit				
23 13.5" <5 wt% Pu(>8)02-U(<9)02 Pins + 21 Mark 1A Fuel Assemblies	1.10	0.84827	0.00101	0.884
23 13.5" <5 wt% Pu(>8)02-U(<9)02 Pins + 21 Mark 1A Fuel Assemblies	1.06	0.83899	0.00099	0.875
Add Fuel Assemblies to Pins with Smallest Volume at Limit				
27 13.5" <25 wt% Pu(>10)02-U(<94)02 Pins + 22 Mark 1A Fuel Assemblies	0.94	0.86739	0.00112	0.903
27 13.5" <25 wt% Pu(>10)02-U(<94)02 Pins + 22 Mark 1A Fuel Assemblies	0.91	0.86064	0.00099	0.897

[11] The mixed oxide fuel compositions show the maximum wt% Pu in U + Pu, the minimum wt% ^{240}Pu in Pu, and the maximum wt% ^{235}U in U.

Benchmark Statistics
Bias = 0.00101
Standard Deviation = 0.01548
Variance = 0.00024

The subcriticality of the above cases indicates that the introduction of N Reactor elements into either the SERF Cask or the Waste Cask will not cause criticality. This conclusion is supported further in Sections 1.4 and 1.5 that study unmoderated N Reactor fuel elements and the addition of N Reactor elements to a single batch of scrap, respectively.

(1.3) Unmoderated Mixed Oxide Fuel Pin Results

The fuel pins were also modeled without water flooding in the cask. These unmoderated cases assumed a H:X atom ratio of 5 and a tight packed pitch. In a tight packed geometry, the fuel volume is quite small. Various moderation conditions all resulting in a H:X ratio $\leq$ 5 were studied to determine the most reactive case. (A H:X ratio of 5 was chosen because this is the standard definition of unmoderated material used at PNL.[3]) As shown in Table 16, the case where the volume between the pins was filled with water at a H:X ratio of 5 and the remainder of the cask was voided was most reactive.

Table 16 Determination of Most Reactive Unmoderated ($H:X\leq5$) Pin Configuration in the Cask

Case[12]	KENO keff	KENO Uncertainty	K95/95
Cask completely void (H:X=0)	0.36006	0.00235	0.392
Entire cask uniformly filled with water at H:X=5	0.38437	0.00257	0.416
Only pins moderated with water at H:X=5	0.60183	0.00377	0.633
Cask filled with water around the pins at H:X=5	0.38094	0.00224	0.412

[12] Each case modeled 104 <25 wt% Pu(>10)O2-U(<36)O2 pins with a diameter of 0.230" and a length of 13.5" in the SERF Cask. The water moderation inside the cask was modeled as noted.

Benchmark Statistics

Bias = -0.00047

Standard Deviation = 0.001406

Variance = 0.00020

The fuel pin limit is set at the minimum subcritical number of pins given in Section 1.1 such that an accident that introduces water into the cask will not cause criticality. The KENO runs shown below modeled a double batch of pins to ensure a loss of mass contingency will also remain subcritical. The decreased cask diameter was used and the pins were modeled on the bottom of the cask as this configuration brings the lead closest to the pins. The pins were modeled in a tight packed, hexagonal pitch. The single batch cases were not modeled because they are less reactive than the double batch cases that are very subcritical.

The results are given in Table 17 and 18. These cases tracked 5000 neutrons per generation in KENO. The results are very low (i.e., k_{eff} ranges from 0.38 to 0.84). The KENO code is not benchmarked or validated for these low reactivities because little experimental data exists in this regime. Thus the uncertainty of the calculation is high and a determination of a $K_{95/95}$ value is not attempted. However, such low results do indicate the cases are substantially subcritical.

Final Results for Unmoderated 13.5" and 37" Mixed Oxide Fuel Pins in the SERF Cask SERF Cask

Table 17 tabulates the results of the unmoderated 13.5" and 37" mixed oxide fuel pins in the SERF Cask.

Table 17 Unmoderated (H:X≤5) Mixed Oxide Fuel Pins in the SERF Cask - Reactivity of a Double Batch

Fuel Type[13]	Fuel Length (in)	Maximum Diameter (in)	Number of Pins	Keno Keff	Keno Uncertainty
<25 wt% Pu(>10)02-U(<94)02					
U(94)02	13.5	0.220	104	0.83416	0.00130
25 wt% Pu(10)02-U(94)02	13.5	0.220	104	0.82605	0.00121
<25 wt% Pu(>10)02-U(<36)02	13.5	0.230	134	0.62287	0.00099
<16 wt% Pu(>10)02-U(<41)02	13.5	0.261	116	0.64396	0.00101
<31 wt% Pu(>10)02-U(<0.72)02	13.5	0.220	180	0.47331	0.00074
<25 wt% Pu(>10)02-U(<68)02	13.5	0.250	100	0.77726	0.00113
<5 wt% Pu(>8)02-U(<9)02	13.5	0.500	92	0.44981	0.00069
<25 wt% Pu(>10)02-U(<94)02					
U(94)02	37	0.220	72	0.75563	0.00116
25 wt% Pu(10)02-U(94)02	37	0.220	72	0.74890	0.00107
<25 wt% Pu(>10)02-U(<36)02	37	0.230	88	0.54931	0.00089
<16 wt% Pu(>10)02-U(<41)02	37	0.261	76	0.57001	0.00087
<31 wt% Pu(>10)02-U(<0.72)02	37	0.220	108	0.39783	0.00072
<25 wt% Pu(>10)02-U(<68)02	37	0.250	72	0.72122	0.00095
<5 wt% Pu(>8)02-U(<9)02	37	0.500	52	0.38090	0.00074

[13] The compositions show the maximum wt% Pu in U + Pu, the minimum wt% ^{240}Pu in Pu, and the maximum wt% ^{235}U in U.

Final Results for Unmoderated 13.5" and 37" Mixed Oxide Fuel Pins in the Waste Cask

Table 18 tabulates the results of the unmoderated 13.5" and 37" mixed oxide fuel pins in the Waste Cask.

Table 18 Unmoderated (H:X≤5) Mixed Oxide Fuel Pins in the Waste Cask - Reactivity of a Double Batch

Fuel Type[14]	Fuel Length (in)	Maximum Diameter (in)	Number of Pins	Keno Keff	Keno Uncertainty
<25 wt% Pu(>10)02-U(<94)02					
U(94)02	13.5	0.220	108	0.83440	0.00134
25 wt% Pu(10)02-U(94)02	13.5	0.220	108	0.82585	0.00102
<25 wt% Pu(>10)02-U(<36)02	13.5	0.230	136	0.61345	0.00104
<16 wt% Pu(>10)02-U(<41)02	13.5	0.261	116	0.63064	0.00090
<31 wt% Pu(>10)02-U(<0.72)02	13.5	0.220	174	0.45461	0.00075
<25 wt% Pu(>10)02-U(<68)02	13.5	0.250	112	0.80669	0.00110
<5 wt% Pu(>8)02-U(<9)02	13.5	0.500	92	0.43901	0.00072
<25 wt% Pu(>10)02-U(<94)02					
U(94)02	37	0.220	80	0.77629	0.00131
25 wt% Pu(10)02-U(94)02	37	0.220	80	0.77290	0.00095
<25 wt% Pu(>10)02-U(<36)02	37	0.230	96	0.56448	0.00093
<16 wt% Pu(>10)02-U(<41)02	37	0.261	80	0.57042	0.00080
<31 wt% Pu(>10)02-U(<0.72)02	37	0.220	116	0.40717	0.00077
<25 wt% Pu(>10)02-U(<68)02	37	0.250	80	0.74424	0.00115
<5 wt% Pu(>8)02-U(<9)02	37	0.500	56	0.39037	0.00071

[14] The compositions show the maximum wt% Pu in U + Pu, the minimum wt% ^{240}Pu in Pu, and the maximum wt% ^{235}U in U.

(1.4) Final Results for Unmoderated N Reactor Fuel Elements

The N Reactor fuel elements were also modeled without moderation. The elements were arranged in a tight packed (hexagonal) pitch filling the cask. Thus the number of elements modeled equaled the maximum number of elements that could geometrically fit into the cask. A H:X ratio of 5 was modeled uniformly throughout the cask as this is the standard maximum ratio for unmoderated fissile material used at PNL. Five thousand neutrons were modeled per KENO generation. The results are shown in Table 19.

Again, the determination of a $K_{95/95}$ value is not valid as the maximum k_{eff} is less than 0.34 and the code is not benchmarked at these low values.

Table 19 N Reactor Dry (H:X≤5) Fuel Element Limits in the SERF and the Waste Cask

Fuel Type[15]	Cask Diameter (in)	Number of Pins	Keno Keff	Keno Uncertainty
SERF Cask				
Mark IA Inner Fuel Elements	8.4	195	0.32284	0.00054
Mark IA Outer Fuel Elements	8.4	48	0.25172	0.00048
Mark IA Fuel Assemblies	8.4	48	0.31537	0.00049
Waste Cask				
Mark IA Inner Fuel Elements	8.4	136	0.33712	0.00050
Mark IA Outer Fuel Elements	8.4	33	0.27394	0.00046
Mark IA Fuel Assemblies	8.4	33	0.32306	0.00050
[15] All cases model N Reactor Mark 1A fuel elements in a tight packed (hexagonal) pitch filling the cask with a H:X ratio of 5 throughout.				
Benchmark Statistics				
Bias = 0.00101				
Standard Deviation = 0.01548				
Variance = 0.00024				

(1.5) Scrap Limits

A scrap limit was also determined for each of the casks. An aqueous 100 wt% ^{239}Pu at 32 g/l was modeled as this is the most reactive scrap solution possible. The solution was modeled as a cylinder with a radius equal to that of the interior of the cask. The length of the cylinder was calculated such that the cylinder contained the required number of grams fissile at the desired concentration. The results shown in Table 20 indicate the minimum subcritical mass is 330 g Pu or 550 g ^{235}U.

Table 20 Scrap Limits in the SERF and the Waste Casks

Case	Keno Keff	Keno Uncertainty	K95/95
SERF Cask			
330 g Pu239 Cylinder at 32 g/l	0.94908	0.00115	0.985
165 g Pu239 Cylinder at 32 g/l	0.78422	0.00116	0.820
550 g U235 Cylinder at 50 g/l	0.94096	0.00122	0.977
550 g U235 Cylinder at 60 g/l	0.95067	0.00114	0.987
550 g U235 Cylinder at 70 g/l	0.94726	0.00114	0.983
275 g U235 Cylinder at 60 g/l	0.77029	0.00117	0.806
Waste Cask			
330 g Pu239 Cylinder at 32 g/l	0.95087	0.00124	0.987
165 g Pu239 Cylinder at 32 g/l	0.76573	0.00111	0.802
550 g U235 Cylinder at 50 g/l	0.94600	0.00106	0.982
550 g U235 Cylinder at 60 g/l	0.94995	0.00124	0.986
550 g U235 Cylinder at 70 g/l	0.94546	0.00110	0.982
275 g U235 Cylinder at 60 g/l	0.74815	0.00107	0.784
Benchmark Statistics			
Bias = 0.00101			
Standard Deviation = 0.01548			
Variance = 0.00024			

To verify the scrap batch limit, scrap was added to a single batch of mixed oxide fuel pins. The most reactive mixed oxide pins at the pin limit in each cask were modeled for this scenario. The scrap was modeled as a cylinder in the center of the cask surrounded by mixed oxide pins. The length of the cylinder was set equal to that of the pins (i.e., 13.5"). The radius of the cylinder was calculated such that the cylinder contained the required grams fissile at the optimum concentration found above. The mixed oxide pins were modeled with full water moderation at the optimum pitch found in Section 1.1. The cask diameter was set equal to the outer diameter of the pins such that the pins were tightly reflected by the lead. The results shown in Table 21 indicate the scrap batch limits must be reduced below 50% of the subcritical mass to ensure the subcriticality in the event of another type of double batch error. A loss of mass contingency error resulting in the combination of one batch of scrap and one batch of mixed oxide pins is only subcritical if the scrap batch limit is set at 150 g Pu or 250 g ^{235}U.

Table 21 Double Batch containing Mixed Oxide Fuel Pins plus Fissionable Scrap in the SERF and the Waste Cask

Number and Type of Fuel Pins in Cask[16]	FBR Pitch (in)	Scrap	Keno Keff	Keno Uncertainty	K95/95
SERF Cask					
45 <31 wt% Pu(>10)02-U(<0.72)02 Pins	0.75	150 g Pu	0.94317	0.00110	0.979
45 <31 wt% Pu(>10)02-U(<0.72)02 Pins	0.71	150 g Pu	0.93724	0.00115	0.973
45 <31 wt% Pu(>10)02-U(<0.72)02 Pins	0.75	250 g U235	0.94845	0.00113	0.985
45 <31 wt% Pu(>10)02-U(<0.72)02 Pins	0.71	250 g U235	0.94658	0.00125	0.983
Waste Cask					
23 <5 wt% Pu(>8)02-U(<9)02 Pins	1.82	150 g Pu	0.93999	0.00120	0.976
23 <5 wt% Pu(>8)02-U(<9)02 Pins	1.75	150 g Pu	0.93849	0.00120	0.975
23 <5 wt% Pu(>8)02-U(<9)02 Pins	1.82	250 g U235	0.94224	0.00123	0.978
23 <5 wt% Pu(>8)02-U(<9)02 Pins	1.75	250 g U235	0.93825	0.00115	0.974

[16] All cases model a 13.5" long cylinder of Pu at 32 g/l surrounded by mixed oxide fuel pins at the given pitch. The cask inner diameter is varied to tightly reflect the pins.

Benchmark Statistics

Bias = 0.00101

Standard Deviation = 0.01548

Variance = 0.00024

A single batch of scrap was also added to a N Reactor fuel elements. The geometry was modeled similar to the mixed oxide cases with one exception. The N Reactor elements were modeled with a tight packed (hexagonal) pitch in order to model the maximum number of elements which will geometrically fit into the cask. The length of the cylinder of scrap was again set equal to the length of one fuel element (i.e., 26.1") but the entire cask was filled with N Reactor fuel. Water moderation was included around the elements. The results shown in Table 22 indicate the reactivity of the combined double batch will not exceed 0.90 including allowances for code bias and error.

Table 22 N Reactor Fuel Elements plus Fissionable Scrap in the SERF and the Waste Cask

Number and Type of Fuel Pins in Cask[17]	Number of Pins	Scrap	Keno Keff	Keno Uncertainty	K95/95
SERF Cask					
Mark IA Inner Fuel Elements	185	150 g Pu	0.78172	0.00085	0.818
Mark IA Outer Fuel Elements	45	150 g Pu	0.88101	0.00103	0.917
Mark IA Fuel Assemblies	45	150 g Pu	0.86981	0.00110	0.906
Mark IA Inner Fuel Elements	187	250 g U235	0.76838	0.00094	0.804
Mark IA Outer Fuel Elements	46	250 g U235	0.88002	0.00091	0.916
Mark IA Fuel Assemblies	46	250 g U235	0.86561	0.00087	0.902
Waste Cask					
Mark IA Inner Fuel Elements	126	150 g Pu	0.78821	0.00091	0.824
Mark IA Outer Fuel Elements	31	150 g Pu	0.89869	0.00098	0.935
Mark IA Fuel Assemblies	31	150 g Pu	0.88431	0.00110	0.920
Mark IA Inner Fuel Elements	127	250 g U235	0.77272	0.00097	0.809
Mark IA Outer Fuel Elements	31	250 g U235	0.89684	0.00094	0.933
Mark IA Fuel Assemblies	31	250 g U235	0.87556	0.00094	0.912

[17] All cases model a 26.1" long cylinder of aqueous solution of Pu at 32 g/l or U235 at 60 g/l surrounded by tight packed N Reactor fuel elements to the maximum inner cask diameter. N Reactor elements also fill the length of the cask beyond the cylinder.

Benchmark Statistics
Bias = 0.00101
Standard Deviation = 0.01548
Variance = 0.00024

Therefore the scrap limits in the SERF and Waste Casks are 250 g ^{235}U if ^{235}U is the only fissionable material present or 150 g total fissionable material. However, no accountable amounts of ^{242m}Am, ^{243}Cm, ^{245}Cm, ^{249}Cf, or ^{251}Cf are permitted. The scrap limit found for ^{239}Pu is applicable to all other fissionable materials with a minimum critical mass greater than that of ^{239}Pu, i.e. all less reactive materials, as given in Table 6.5 of Reference 3. None of the restricted materials are currently accounted for at PNL; therefore violation of this limit is not credible.

(2) **State computer programs used, the options, the recipes for choosing mesh points where applicable, the cross section sets, and any numerical parameters necessary to describe the input.**

The SCALE 4.2 system of codes[4] was used to model the casks. The codes were executed on either a HP 755 with HP-UNIX version 9.03 or a HP 735 running HP-UNIX version 9.05. Code quality assurance documentation verifies the two operating systems give identical results. The NITAWL and XSDRN codes were used to generate cell-weighted cross-sections and KENO was used to model the cask geometry and predict k_{eff}. The standard 27-group ENDF/B-IV cross-sections library was used throughout. For calculations used to search for the optimum pitch and cask diameter, KENO tracked 300 neutrons per generation. For the final calculations that set the pin limit, 5000 neutrons were tracked per generation. All calculations ran 120 generations with the first 20 generations skipped when determining the standard deviation of the run. Increasing the neutrons tracked decreases the statistical uncertainty associated with the calculated result but also significantly increases the time to complete the run. Thus, using the above scheme, the searches could be completed quickly while decreasing the uncertainty in the final result. All input files containing mesh point determinations and atom densities are included in Appendix A.

(3) **Identify experimental data and list parameters derived therefrom for use in the validation of the model.**

The experimental benchmarks modeled with the SCALE 4.2 codes are shown in Table 23. The results defined the code statistics used in the determination of the $K_{95/95}$ value for each run. Each benchmark case was modeled with both 300 and 5000 neutrons tracked per KENO generation to create a separate statistical basis for the scoping and final calculations. The same scheme was employed in generating cross-sections sets and calculating k_{eff} as was used to model the worst case geometries. All parameters required to model these benchmarks are described in the input files given in Appendix B and the references indicated in Table 23.

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Table 23 Experimental Benchmarks Modeled with KENO

		5000 neutrons/generation		300 neutrons/generation	
Case	Reference	KENO keff	KENO Uncertainty	KENO keff	KENO Uncertainty
Pu Spheres					
Full water reflected	6	1.01407	0.00111	1.01309	0.00450
Thin steel spherical shell + full water reflection	7	1.01064	0.00112	1.01191	0.00432
Thin steel spherical shell without water reflection	7	0.99318	0.00127	0.99314	0.00514
Thin steel + 4" concrete spherical shell	7	1.02249	0.00105	1.01923	0.00496
4.31 wt% Enriched UO2 Pins					
Square array with lead wall at 0 cm	8	1.00355	0.00118	0.99955	0.00404
Square array with lead wall at 2 cm	8	1.00307	0.00108	1.00538	0.00463
Square array with lead wall at 3 cm	8	0.99605	0.00106	1.00600	0.00441
Square array with lead wall at 5 cm	8	0.98712	0.00105	0.99285	0.00458
Square array without lead wall	8	0.97791	0.00113	0.98147	0.00369
93.2 wt% Enriched U-AL Pins					
Hexagonal arrangement with central void	9	0.99565	0.00127	0.99507	0.00556
Hexagonal arrangement with central H_2O	9	1.02992	0.00114	1.03083	0.00551
Mixed Oxide Pins					
Pitch = 0.38"	10	1.01884	0.00105	1.00719	0.00427
Pitch = 0.49"	10	1.01247	0.00108	1.01636	0.00525
Pitch = 0.61"	10	1.00892	0.00108	1.01253	0.00437
Pitch = 0.76"	10	1.01142	0.00094	1.00786	0.00438
Pitch = 0.98"	10	1.00855	0.00084	1.00413	0.00354
0.947 wt% Enriched U Metal Annular Elements					
Pitch=1.78"	11	0.98506	0.00073	0.98638	0.00299
Pitch=2.0"	11	0.98401	0.00069	0.98625	0.00311
1.25 wt% Enriched U Metal Annular Elements					
Pitch=1.85"	12	0.98173	0.00071	0.98303	0.00326

Table 23 Experimental Benchmarks Modeled with KENO

		5000 neutrons/generation		300 neutrons/generation	
Case	Reference	KENO keff	KENO Uncertainty	KENO keff	KENO Uncertainty
Pitch=2.0"	12	0.98682	0.00074	0.99319	0.00318
Pitch=2.1"	12	0.99105	0.00079	0.99844	0.00301
Pitch=2.2"	12	0.99120	0.00070	1.00012	0.00316
Pitch=2.4"	12	0.99360	0.00063	0.99494	0.00270
2.1 wt% Enriched U Metal Annular Elements					
Pitch=2.44"	13	0.95920	0.00080	0.95988	0.00398
Pitch=2.8"	13	0.99262	0.00091	1.00174	0.00354
Pitch=3.1"	13	1.00690	0.00096	1.00705	0.00360
Pitch=3.4"	13	1.00679	0.00084	1.00497	0.00369

(4) State the area(s) of applicability.

The fuel pin limits as given in the Conclusions section are applicable to the storage and transportation of the specified fuel type in the form of pins in the SERF and the Waste Casks respectively. The mixed oxide fuel pin composition is given as the maximum wt% plutonium in the uranium + plutonium, the minimum wt% ^{240}Pu in the plutonium, and the maximum enrichment of ^{235}U in the uranium. The N Reactor fuel element limits are applicable for Mark 1A and Mark IV. The maximum outer diameter and fuel pin length are also given. Broken lengths of pins placed in an outer sleeve are considered one pin provided the other size criteria are met.

The general limits given in PNL-MA-25[3] for fissile material are not valid in the casks. Instead, a scrap limit that applies specifically to the casks has been determined as given in the Conclusions section. The ^{235}U mass limit can be used only for ^{235}U. A more general limit that is applicable to all types of fissionable material, except ^{242m}Am, ^{242}Cm, ^{245}Cm, ^{249}Cf, or ^{251}Cf, is also given in the Conclusions.

(5) State the bias and the prescribed margin of subcriticality over the area(s) of applicability. State the basis for the margin.

Calculational results indicate that the overall code bias based on benchmark calculations tracking 5000 neutrons per generation is 1 mk. The mean of all benchmarks is 0.999 ± 0.015 and thus, the standard deviation is 15 mk. The overall code bias for the same benchmarks tracking 300 neutrons per generation is -0.5 mk with a standard deviation of 14 mk. If these values are applied to the loss of contingency scenarios in a methodology similar to that employed by Marshall (Reference 5), k_{eff} will not exceed 0.99 including experimental error and biases. The margin of subcriticality is therefore greater than 10 mk and assumes optimum conditions and a loss of contingency error. The margin of subcriticality was calculated with a 95% confidence.

Discussion of Loss of Contingencies

A loss of mass contingency error would be caused by either exceeding the pin number limit, the scrap mass limit, or both limits. All of these scenarios have been examined to ensure double contingency safety is met with the limits found previously.

The fuel pin limits are set at 50% of a subcritical number of pins to ensure a double batch error will not result in a criticality. All double batch scenarios in each cask have a k_{eff} of less than 0.99 including allowances for code bias and experimental error as shown in the previous tables.

A violation of the scrap mass limit alone would also remain subcritical. The maximum allowed mass of scrap is 250 g ^{235}U or 150 g fissile for scrap of unlimited composition. This mass limit is less than 50% of the subcritical mass of 550 g ^{235}U or 330 g fissile of unlimited concentration. Thus a double batched scenario would clearly be subcritical.

An accident exceeding both the scrap and mixed oxide pin limits was also studied. A calculation was performed to ensure subcriticality of a single batch of mixed oxide fuel pins combined with a single batch of scrap. The scrap mass limit was reduced from one half of a subcritical mass to a mass that remained subcritical in a double batch scenario. As shown in Section 1.5, the $K_{95/95}$ of this scenario does not exceed 0.99.

Any combination of N Reactor fuel elements, scrap, mixed oxide fuel pins and water moderation will not result in criticality in either SERF or the Waste Cask. To prove this, the pins were modeled at optimum pitch with water moderation and at a tight packed pitch without water. The maximum number of N Reactor fuel elements that would physically fit in the cask were also added to a single batch of mixed oxide pins, a single batch of Pu and a single batch of ^{235}U scrap. The maximum $K_{95/95}$ for all of these cases was less than 0.91 including allowances for code bias and error. Thus the minimum margin of subcriticality is 90 mk even with the loss of one contingency. Therefore, the limit of N Reactor fuel elements is set at the maximum number of elements that can be placed in each of the casks in a tight packed pitch.

The proven subcriticality of a double batch of each combination of fissionable material to be placed in the casks indicates that the sum of the fractions method of combining different types of material into one batch is valid.

Conclusions

The purpose of this memo was to validate the original technical bases for the subcriticality of storage and handling of mixed oxide pins, UO_2 fuel pins, or N Reactor fuel elements in combination with or separate from fissionable material scrap of various compositions in SERF Cask and the Waste Cask. The analysis included modeling a number of accident scenarios to ensure subcriticality even in the event of a loss of contingency error. In conclusion, new batch limits for the number of fuel pins and amount of scrap in SERF Cask and the Waste Cask have been determined. (The general limits given in PNL-MA-25 are not valid in these casks.) If water flooding of the cask must be considered under normal operation, the limits are given in Table 24. If water flooding of the cask is considered only as an accident condition, the limits given in Table 25 are applicable. For both cases, the limits are applicable only for fuel pins of the specified diameter, length and composition and for scrap of the specified composition in SERF and the Waste Casks.

Table 24 Criticality Limits in the SERF Cask and the Waste Cask with Water Moderation

		SERF Cask		Waste Cask	
Mixed Oxide Fuel Pin Limits	Maximum Fuel Diameter (in)	Maximum Fuel Pin Length of 13.5"	Maximum Fuel Pin Length of 37"	Maximum Fuel Pin Length of 13.5"	Maximum Fuel Pin Length of 37"
<25 wt% Pu(>10)O2-U(<94)O2	0.220	26	18	27	20
<25 wt% Pu(>10)O2-U(<36)O2	0.230	33	22	34	24
<16 wt% Pu(>10)O2-U(<41)O2	0.261	29	19	29	20
<31 wt% Pu(>10)O2-U(<0.72)O2	0.220	45	27	43	29
<25 wt% Pu(>10)O2-U(<68)O2	0.250	25	18	28	20
<5 wt% Pu(>8)O2-U(<9)O2	0.500	23	13	23	14
N Reactor Fuel Element Limits	Maximum Fuel Diameter (in)	Maximum Fuel Element Length of 26.1"		Maximum Fuel Element Length of 26.1"	
Mark IA and IV Inner Elements	1.170	195		136	
Mark IA and IV Outer Elements	2.350	48		33	
Mark IA and IV Fuel Assemblies	2.350	48		33	
Scrap Limit		250 g ^{235}U only or 150 g total fissionable[18]		250 g ^{235}U only or 150 g total fissionable[18]	

[18] No accountable amounts of ^{242m}Am, ^{243}Cm, ^{245}Cm, ^{249}Cf, or ^{251}Cf are permitted.

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Table 25 Criticality Limits in the SERF Cask and the Waste Cask for Dry (H:X=5) Conditions

		SERF Cask		Waste Cask	
Mixed Oxide Fuel Pin Limits	Maximum Fuel Diameter (in)	Maximum Fuel Pin Length of 13.5"	Maximum Fuel Pin Length of 37"	Maximum Fuel Pin Length of 13.5"	Maximum Fuel Pin Length of 37"
<25 wt% Pu(>10)O2-U(<94)O2	0.220	52	36	54	40
<25 wt% Pu(>10)O2-U(<36)O2	0.230	67	44	68	48
<16 wt% Pu(>10)O2-U(<41)O2	0.261	58	38	58	40
<31 wt% Pu(>10)O2-U(<0.72)O2	0.220	90	54	87	58
<25 wt% Pu(>10)O2-U(<68)O2	0.250	50	36	56	40
<5 wt% Pu(>8)O2-U(<9)O2	0.500	46	26	46	28
N Reactor Fuel Element Limits	Maximum Fuel Diameter (in)	Maximum Fuel Element Length of 26.1"		Maximum Fuel Element Length of 26.1"	
Mark IA and IV Inner Elements	1.170	195		136	
Mark IA and IV Outer Elements	2.350	48		33	
Mark IA and IV Fuel Assemblies	2.350	48		33	
Scrap Limit		250 g ^{235}U only or 150 g total fissionable[19]		250 g ^{235}U only or 150 g total fissionable[19]	

[19] No accountable amounts of ^{242m}Am, ^{243}Cm, ^{245}Cm, ^{249}Cf, or ^{251}Cf are permitted.

mass U ~0.947 enr
16 kg U/elem.

Assume d.L. = 2 mil for MOX ← ignore clad.
cc.
Mass = $\left[\frac{\pi}{4}(.22" \times 2.54)^2 (37" \times 2.54) \times 36 \text{ pins}\right] \times 10\ g/cc$
= 9.127E7 g

Concurrence: 31 Aug 95
Senior Specialist
Criticality Safety Analysis

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7.0 STRUCTURAL EVALUATION

7.1 INTRODUCTION

The SERF Cask is an onsite, intra-area packaging for transferring Type B radioactive material within the 300 Area of the Hanford Site. This section of the document defines and evaluates the normal transport condition structural requirements for intra-area transport of this package. Structural performance of the package is evaluated only for normal transport conditions; accident conditions are evaluated in the risk and dose consequence section of this document.

7.2 STRUCTURAL EVALUATION OF PACKAGE

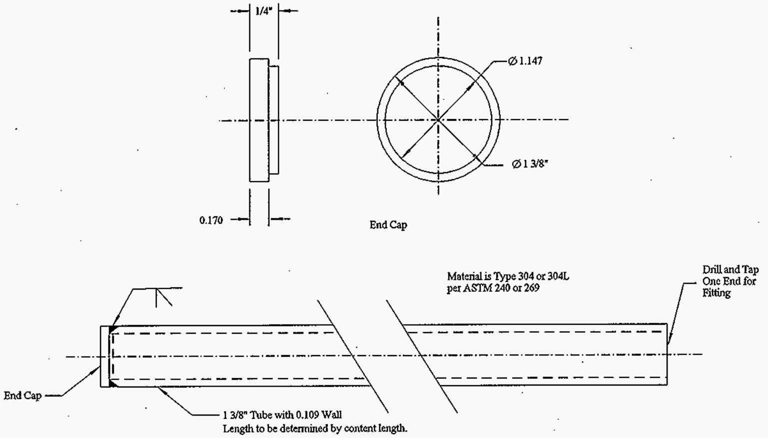
7.2.1 Package Structural Description

The SERF Cask is fundamentally a right circular cylinder of lead and stainless steel composite construction. The lead is sandwiched between outer and inner stainless steel tubular shells. At each end of the cask is a thick circular plate that is welded to both inner and outer shell, which encapsulates the lead. At the top, sealing and closure of the inner cavity is provided by a bolted blind flange with an attached shield plug of composite lead-stainless steel construction. The cask is equipped with a manually actuated integral closure value which provides closure at the bottom end. A secondary stainless steel gasketed blind flange plate is bolted on to provide sealing of the cask during transport. A handling yoke is provided on the cask and welded to the outer shell. In addition support plates are welded to the outer housing of the cask to provide support during transport and lifting attachment anchors for handling.

The cask is constructed of 304 or 304L stainless steel, which is welded by welders and weld procedures qualified to Section IX of the ASME BP&V Code (ASME 1977a) in use at the time of construction. Nondestructive examinations of the welds were performed in accordance with Section V of the ASME BP&V Code (ASME 1977b). The lead shielding meeting the requirement of ASTM B29 (ASTM 1976) was poured into the shielding cavity after construction of the cask.

Fuel, fuel scraps and filings, and dispersible materials transported in this package are contained by a duel systems of sealed tubes. The contents are first sealed in a canister assembly made up of a 304 or 304L stainless steel tube, which has a welded bottom at one end and a threaded fitting on the other. This canister tube is then inserted into a sealed overpack. The overpack (shown in Figure B7-1) is a 3.49 cm (1⅜ in.) diameter tube constructed from 304 or 304L stainless steel tubing conforming to the requirements of ASTM A240 or A269. The overpack can be fitted with one of two bottom end closures. At the bottom end the overpack can be fitted either with a welded flat stepped plate which is socket welded to the tube as shown in Figure B7-1 with a full penetration bevel weld. The bottom end plate has a center area thickness of 0.64 cm (¼ in.) with a minimum joint landing thickness of 0.43 cm (0.170 in.). The bottom end plate is welded to the tube As a alternative a threaded fitting can be fitted to the tube to provide the bottom end closure. The top end closure is provided by a threaded fitting. The threaded fitting must extend into the tube a minimum of 0.64 cm (¼ in.). As shown in Part B Section 7.5.2, the overpack has a rated internal pressure capacity of 20,684 MPa (3,000 psi).

Figure B7-1. SERF Canister Overpack.



7.2.2 Chemical and Galvanic Reactions

At the operating temperatures, no chemical or galvanic interaction of the materials will occur. The lead is encapsulated in the stainless steel consequently no are no agents to generate chemical or galvanic reactions. In addition stainless steel forms a natural oxide layers which provides protection from most corrosive agents (i.e., water) under the normal operating temperatures.

7.2.3 Sizes of Package and Cavity

The outside dimensions of the cask are 66 cm (26 in.) in diameter by 339.1 cm (133.5 in.) long. Dimensions of the inner cavity are 20 cm (8 in.) in diameter by 276.23 cm (108.75 in.) long.

7.2.4 Weight

The SERF Cask empty weight is 11,979 kg (26,300 lb) and has a maximum gross weight of 12,201 kg (26,900 lb). For calculational conservatism the empty weight of the cask is assumed as 12,247 kg (27,000 lb) with a gross weight of 12,564 kg (27,700 lb).

7.2.5 Tamper-Indicating Devices

Due to the weight of the cask closures and the intra-area shipment of this cask, no tamper-indicating devices are provided.

7.2.6 Positive Closure

Positive closure of the cask is provided by the bolted blind end flanges. The top-end blind flange is secured with 16, ½ in. bolts. At the bottom-end the flange is secured with 6, ⅜ in. bolts.

7.2.7 Lifting and Tiedown Devices

For lifting and handling, a yoke is provided on the cask along with 3.49 cm (1⅜ in.) lifting attachment holes on the support plates. The four lifting attachment holes are provided on the two support plates welded to the cask body for horizontal handling of the cask. As shown in the Part B, Section 7.5.1, the yoke and the lifting attachments meet the requirements of the *Hanford Site Hoisting and Rigging Manual* (RL 1993).

As shown in Part B, Section 10.0, the lifting attachment holes are also used as tiedown securement points. Since these attachment holes have been shown to meet the strength requirements of the *Hanford Site Hoisting and Rigging Manual* (RL 1993), they will meet the 49 CFR 393 securement requirements. This is based on the lower load requirements for tiedowns and the configuration of the tiedowns. As configured lateral and vertical loads are uni-directional and by symmetry the support plates are loaded in the same plane as for lifting. In the longitudinal direction the loading normal loading of the support plate is counteracted by the opposite set of tiedowns. Consequently, the loadings on the attachment holes is less severe for tiedown securement than for lifting.

7.3 NORMAL TRANSPORT CONDITIONS

7.3.1 Conditions to be Evaluated

Onsite structural performance of the package is assessed for Hanford Site normal conditions in this section. The onsite conditions evaluated for are hot and cold temperature extremes, reduced and increased external pressure, vibration, water spray, compression, inertial loading, and penetration. The package structural response with solar insolation is evaluated for the onsite hot ambient temperature extreme of 46 °C (115 °F) (Fadeff 1992) and without for the cold ambient temperature extreme of -33 °C (-27 °F) (Fadeff 1992). Reduced and increased external pressure structural response is evaluated for a Hanford Site maximum barometric pressure range of 0.94 atm (13.81 psi) to 1.01 atm (14.85 psi). Vibrational loading response of the package is evaluated for the parameters established in ANSI N14.23 (ANSI 1980). In the case of water spray, package response is evaluated for in leakage of water at ambient temperatures and pressures. The package structural response to compression is evaluated for compressive loads resulting from anticipated stacking onto the package. Since there are no in-transit load transfers, structural response of the package to inertial loads is evaluated for rough transport of the package based on ANSI N14.23 shock loading parameters. Penetration structural response is idealized as a loading from a 3.2 cm (1.25 in.) diameter steel rod with a rounded end weighing 6 kg (13 lb) dropping onto the package from a height of 1 m (40 in.). These loads are to be applied independently and non-sequentially.

In addition, for dispersible materials sealed in the canister tube and overpack are evaluated for containing the contents when pressure to an internal pressure of 20,684 MPa (3,000 psi). Consequently, the overpack is capable of sustaining an energy loading of 14 kJ (10,332 ft-lb) of energy. This is the energy equivalent of a 6 m (18 ft) drop of the maximum payload weight onto a hard unyielding surface.

7.3.2 Acceptance Criteria

The criteria for acceptable performance of the package are based on all major critical components of the package remain structurally functional, the contents remain contained, and only superficial damage of non-critical components is incurred during normal transport conditions. To meet the criteria for normal transport conditions, the analytical tests are to assume, as a worst case, the package is intermittently subjected to the above loading conditions during normal transport operations. Performances of the package in meeting these criteria are demonstrated by either positive margins of safety based on material yield strength or package loadings which are within acceptable limits of the component/materials used in the package.

For containing dispersible materials, the criteria for acceptable performance is based on the canister overpacks being capable of sustaining a 20,864 MPa (3,000 psi) internal pressure per the requirements of ASME Section III, Subsection NE (ASME 1992b). Performance of the canister overpack to sustain this pressure is demonstrate by the positive margins of safety based on an ASME Section III code evaluation of the overpack.

7.3.3 Hot and Cold Evaluation

Based on the thermal evaluation from Part B, Section 8.0 for a worst case heat source of 377 W from two cesium chloride capsules with solar insolation under Hanford site conditions, the maximum internal component temperature of the cask is 73 °C (163 °F). Considering the materials of construction (lead and stainless steel), no material degradation or appreciable reduction in yield strength will occur. Under these conditions the external temperature of the cask is 71 °C (159 °F), which below the rated 149 °C (300 °F) upper service temperature limit of the neoprene seals (Parker 1991). The increase in internal pressure from the rise in internal temperature is only approximately 0.3 atm (4 psi) (Part B, Section 8.0). Due to the robustness of the cask, this increase in temperature would not result in any significant loading on the cask. Consequently, structural performance of the package is not affected by the Hanford Site hot temperature extreme.

Evaluation of cold temperature package performance shows that since the primary structural material of construction is stainless steel, no extreme cold weather shipping restrictions are required. Austenitic stainless steel is not susceptible to low temperature brittle fracture. The low service temperature limit for the neoprene seals is -54 °C (-65 °F). Consequently, with an onsite low extreme cold temperature of -33 °C (-27 °F), no degradation cask performance will occur.

7.3.4 Reduced and Increased External Pressure

For Hanford Site conditions, the largest differential pressure of 0.34 atm (5 psi) is due to reduced external pressure, assuming an increased internal pressure due to an internal temperature rise. Based on the cask construction and wall thickness, neither reduced nor increased external pressure would result in any significant loading of the cask.

7.3.5 Vibration

Vibration of the package is not a concern, since the shipment occurs only twelve times a year for distance of less than 1.61 km (1 mi). Based on a speed of 24 kph (15 mph) for a distance of 1.61 km (1 mi), 12 times a year, at a loading frequency of 2 Hz (ANSI 1980) this equates to approximately 6,000 cycles per year. Relative to the loading on the materials and the material fatigue strengths, vibrational loading on this cask is not significant.

7.3.6 Water Spray

Since the package is sealed at both ends with Neoprene[3] gaskets and the cask is of all welded construction any in-leakage of water into the cask cavity is not a concern. Also during normal transport conditions, it is demonstrated that no inelastic deformation of critical cask components occurs. Consequently, under water spray conditions, in-leakage of water into the cask cavity will not occur during transport.

7.3.7 Compression

The package is shipped as an elusive use shipment, and stacking on the package is prohibited. *Consequently, package compression is not a concern.*

7.3.8 Inertial Loading

During normal transport of this package no in-transit load transfers are involved. Consequently, normal condition inertial loads would arise from rough transport shock loads of 3.5*g* vertical to the plane of travel and 2.3*g* in the direction of travel (ANSI 1980).

In the case of the rough transport shock loads of the conveyance are evaluated as intermittent in nature and applied as a single pulse to the package. Since the duration of the shock load is of such long duration (greater than 3 times the natural frequency of the system), it is applied as a quasi-static load and the package is evaluated by classical linear elastic methods. To ensure component and material loading are within the elastic range, the allowable stresses are established on the basis of the *maximum shear stress* theory of failure. The weld allowable loading is established on base material yield strength with a joint efficiency reduction factor based on ASME Section VIII (ASME 1992a).

Evaluation of rough transport loads on the package presented in Part B, Section 7.5, show that the package remains fully functional, maintains structural integrity, and maintains the contents. In the evaluation, package performance is analyzed for both *vertical and longitudinal loading. In both cases* the evaluation shows the induced stresses on the package are below the allowable stresses.

The canister overpack pressure capacity of 20,864 MPa (3,000 psi) is used to demonstrate the inertial loading performance of the canister. This is based on a sustained energy load comparison of internal energy pressure and energy from a free fall. Assuming a conservative worst case gross canister overpack weight of 272 kg (600 lb), the energy of equivalent of a 20,864 MPa (3,000 psi) pressure load is a 6 m (18 ft) free fall onto a hard unyielding surface. Based on this it is demonstrated that the canister will contain any dispersible contents within the cask under normal transport conditions considering the transimitted inertial loads of a cask shielded with lead. The lead acts to dampen and not amplify any shock loads on the payload applied to the cask.

7.3.9 Penetration

The evaluation presented in Part B, Section 7.5.1, shows the exposed surfaces of the package cannot be penetrated by a 6 kg (13 lb) object dropped from 1 m (40 in.). Results of the evaluation show that only superficial marring of the exposed stainless steel surfaces resulting from dropping of the object.

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[3] Neoprene is a trademark of E. I. du Pont de Nemours and Company.

7.3.10 Conclusions

The results of these evaluations show the package is acceptable for transport on the Hanford Site under normal transport conditions. Also results show that for dispersible materials the canister overpack will sustain pressure loads of up to 20,864 MPa (3,000 psi).

7.4 REFERENCES

ANSI, 1980, Draft *American National Standard Design Basis for Resistance to Shock and Vibration of Radioactive Material Packages Greater than One Ton in Truck Transport*, ANSI N14.23, American National Standard Institute, New York, New York.

ASME, 1995a, *Boiler and Pressure Vessel Code*, Section IX, American Society of Mechanical Engineers, New York, New York.

ASME, 1995b, *Boiler and Pressure Vessel Code*, Section V, American Society of Mechanical Engineers, New York, New York.

ASME, 1992a, *Boiler and Pressure Vessel Code*, Section VIII, Division 1, American Society of Mechanical Engineers, New York, New York.

ASME, 1992b, *Boiler and Pressure Vessel Code*, Section III, Subsection NE, American Society of Mechanical Engineers, New York, New York.

ASTM, 1976, *Annual Book of ASTM Standard*, American Society of Testing and Materials, Philadelphia, Pennsylvania.

Parker, 1991, *Parker O-Ring Handbook*, Parker Hannifin Corporation, Cleveland, Ohio.

RL, 1993, *Hanford Site Hoisting and Rigging Manual*, DOE-RL-92-36, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

7.5 APPENDICES

7.5.1 Structural Evaluation and Puncture Threshold

ENGINEERING SAFETY EVALUATION

Subject: SERF Cask NTC Structural Evaluation & Puncture Threshold Page: 1 of 11
Originator: S. S. Shiraga Date: 05/19/97
Checker: R. J. Smith Date: 05/25/97

I. Objective:

The objective of this evaluation is to evaluate the SERF Cask performance relative to the Normal Transport Conditions outlined within Section 7 of this Safety Evaluation Packaging (SEP). Also this evaluation determines adequacy of the lifting systems and, the equivalent steel thickness of the cask for determination of the puncture failure threshold.

II. References:

ANSI, 1992, ANSI N14.23, *Draft American National Standard Design Basis for Resistance to Shock and Vibration of Radioactive Material Packages Greater than One Ton in Truck Transport*, American National Standard Institute, New York, New York.

ASME, 1995, *Boiler and Pressure Vessel Code*, Section II, Part D, American Society of Mechanical Engineers, New York, New York.

Roark, R. J., 1965, *Formulas for Stress and Strain*, Fourth Edition, McGraw-Hill Book Company, New York, New York.

Blodgett, O. W., 1966, *Design of Welded Structures*, The James F. Lincoln Arc Welding Foundation, Cleveland, Ohio.

Roark, R. J., Young, W. C., 1983, *Formulas for Stress and Strain*, Fifth Edition, McGraw-Hill Book Company, New York, New York.

ASME, 1992, *Boiler and Pressure Vessel Code*, Section VIII, Division 1, American Society of Mechanical Engineers, New York, New York.

Rinehart, J. S., and Pearson, J., 1954, *Behavior of Metals Under Impulsive Loads*, American Society of Metals, Cleveland, Ohio.

ANSI, 1993, ANSI N14.6, *American National Standard for Radioactive Materials-Special Lifting Devices for Shipping Containers Weighing 10 000 Pounds (4500 kg) or More*, American National Standard Institute, New York, New York.

ORNL, 1970, ORNL-NISC-68, *Cask Designer's Guide*, Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1970.

III. Results and Conclusions:

This evaluation shows that the SERF Cask will meet normal transport conditions (NTC) and lifting loads as specified in Section 7 of this SEP. The evaluation shows the inertial loadings on the cask from rough transport are well below the material yield. Consequently, rough transport will not result in any permanent deformation of the package. The results demonstrate that the cask will not sustain any damage during NTC and remain fully functional.

Also, within this evaluation, the equivalent steel thickness of the package is determined. Based on empirical data (Rinehart, 1954) the equivalent steel thickness of the cask body is 4.6 inches.

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IV. Evaluation:

Normal Transport Conditions Evaluation of SERF Cask:

Determination of inertial loading for NTC:

During normal transport of this package no in-transit load transfers are involved. Consequently, normal condition inertial loads arise from rough transport shock loads. Rough transport loads are derived from ANSI N14.23 (ANSI, 1980). Rough transport shock loads for this package are defined as a vertical 3.5 g and longitudinal (in direction of travel) 2.3 g shock load to the package. Lateral load of 1.6 g is neglected, as it is bounded by the vertical and longitudinal loads. Assume the shock is a single pulse applied to the package as a quasi-static load. Acceptance criteria is the package remain fully functional after the event, i. e., no plastic deformation of components.

Empty weight of package: $W_p := 26000\,lbf$ Maximum weight of payload: $W_{pay} := 500\,lbf$

Gross weight of package: $W_t := W_p + W_{pay}$

Inertial loadings: Vertical: $g_v := 3.5$ Longitudinal: $g_l := 2.3$

Material parameters (ASME, 1995):

Assume cask shells are constructed of 304L stainless steel (SA 240, Class 2) at 250 °F :

Yield strength: $s_{ysst} := \left(\frac{21.3 + 19.1}{2}\right)\cdot ksi$ $s_{ysst} = 20\cdot ksi$ Poisson's Ratio: $v_{sst} := 0.31$

Elastic modulus: $E_{sst} := \left(\frac{27.6 + 27}{2}\right)\cdot 10^6\cdot psi$ $E_{sst} = 2.73\cdot 10^7\cdot psi$

Allowable stress for NTC on sst components based on ASME criteria: $s_{asst} := \frac{2}{3}\cdot s_{ysst}$

Assumed cask bolts are constructed of SA193 Type B8 (Class 2) at 250 °F:

Yield strength: $s_{yb} := \left(\frac{27.5 + 25.6}{2}\right)\cdot ksi$ $s_{yb} = 27\cdot ksi$

Allowable stress for NTC bolts based on ASME criteria: $s_{ab} := \frac{2}{3}\cdot s_{yb}$ $s_{ab} = 18\cdot ksi$

Assume lead properties at 250 °F (ORNL, 1970):

Yield strength: $s_{ypb} := 8\cdot ksi$ Poisson's ratio: $v_{pb} := 0.4$

Elastic modulus: $E_{pb} = 2\cdot 10^6\cdot psi$

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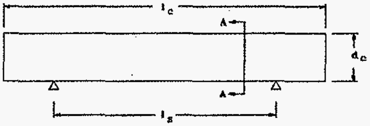
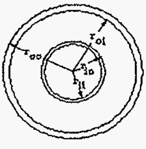
Geometric parameters of package:

Idealize the package as a composite beam with of concentric cylinders of stainless steel and lead.





Section A–A

Length of cask: $l_c := 133.56 \cdot in$

Diameter of cask: $d_c := 26 \cdot in$

Length between supports: $l_s := 66.5 \cdot in$

Outer shell wall thickness: $t_{os} := 0.625 \cdot in$

Outer shell outside radius: $r_{oo} := \frac{d_c}{2}$

Outer shell inside radius: $r_{oi} := r_{oo} - t_{os}$

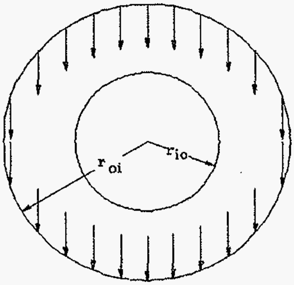
Inner shell outside radius: $r_{io} := \frac{8.6 \cdot in}{2}$

Inner shell inside radius: $r_{ii} := \frac{7.63 \cdot in}{2}$

Inner shell wall thickness: $t_{is} := r_{io} - r_{ii}$ $t_{is} = 0.49 \cdot in$

End plate thickness: $t_{ep} := 2.5 \cdot in$ End plate weld leg: $w_{leg} := 0.5 \cdot in$

Determine deflection of lead, assuming the lead is unbonded to the stainless steel walls:



Density of lead: $\rho_{pb} := 710 \cdot \frac{lbf}{ft^3}$

Idealize as ring supporting it's own weight. Use Roark, 1965, case 18, page 176.

Weight per linear inch of lead:

$$w_{pb} := \rho_{pb} \cdot \left[\pi \cdot \left(r_{oi}^2 - r_{io}^2\right)\right] \qquad w_{pb} = 174 \cdot \frac{lbf}{in}$$

Moment of inertia of the section:

$$I_{pb} := \pi \cdot \frac{r_{oi}^4 - r_{io}^4}{4} \qquad I_{pb} = 18151 \cdot in^4$$

Nominal radius: $r_{pb} := \frac{r_{oi} + r_{io}}{2}$

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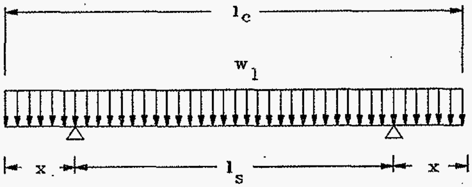
Diametrial compression of lead: $\Delta_y := -\frac{g_v \cdot w_{pb} \cdot r_{pb}^4}{E_{pb} \cdot I_{pb}} \cdot (0.4674)$ $\Delta_y = -3.78 \cdot 10^{-5} \cdot \text{in}$

This is negligible, consequently no significant load on inner shell.

Determine Loading on Cask Body due to Bending:

Since cask is shipped in the horizontal, setting on two support plates, idealize as simply supported beam with uniform loading which over hangs each end (Blodgett, 1966).



Assumed uniform load: $w_l := \frac{g_v \cdot W_t}{l_c}$ $w_l = 694 \cdot \frac{\text{lbf}}{\text{in}}$ Over hang distance: $x := \frac{l_c - l_s}{2}$ $x = 33.53 \cdot \text{in}$

Reaction loads at supports: $F_r := w_l \cdot \left(x + \frac{l_s}{2} \right)$ $F_r = 46375 \cdot \text{lbf}$

Maximum moment at center: $M_c := \frac{w_l}{8} \cdot \left(l_s^2 - 4 \cdot x^2 \right)$ $M_c = -6493 \cdot \text{lbf} \cdot \text{in}$

Moment at supports: $M_s := \frac{w_l \cdot x^2}{2}$ $M_s = 390368 \cdot \text{lbf} \cdot \text{in}$

Maximum load and moment at supports.

Determine composite moment of inertia:

Moment of inertia about center (neutral axis) of cross section:

Outer shell: $I_{os} := \pi \cdot \left(\frac{r_{oo}^4 - r_{oi}^4}{4} \right)$ Inner shell: $I_{is} := \pi \cdot \left(\frac{r_{io}^4 - r_{ii}^4}{4} \right)$

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Composite moment of inertia: $EI := E_{sst} \cdot I_{os} + E_{pb} \cdot I_{pb} + E_{sst} \cdot I_{is}$

Determine composite area:

Outer shell: $A_{os} := \pi \cdot \left(r_{oo}^2 - r_{oi}^2\right)$ Inner shell: $A_{is} := \pi \cdot \left(r_{io}^2 - r_{ii}^2\right)$

Lead shielding: $A_{pb} := \pi \cdot \left(r_{oi}^2 - r_{io}^2\right)$

Composite area factor: $AE := A_{os} \cdot E_{sst} + A_{pb} \cdot E_{pb} + A_{is} \cdot E_{sst}$

Bending stress in outer shell at maximum moment: $\sigma_{bos} := \frac{M_s \cdot E_{sst} \cdot r_{oo}}{EI}$ $\sigma_{bos} = 0.93 \cdot ksi$

Bending stress in inner shell at maximum moment: $\sigma_{bis} := \frac{M_s \cdot E_{sst} \cdot r_{io}}{EI}$ $\sigma_{bis} = 0.31 \cdot ksi$

Bending stress in lead shielding at maximum moment: $\sigma_{bpb} := \frac{M_s \cdot E_{pb} \cdot r_{oi}}{EI}$ $\sigma_{bpb} = 0.07 \cdot ksi$

Maximum shear stress in outer shell: $\tau_{sos} := \frac{F_r \cdot E_{sst}}{AE}$ $\tau_{sos} = 0.5 \cdot ksi$

Maximum shear stress in inner shell: $\tau_{sis} := \frac{F_r \cdot E_{sst}}{AE}$ $\tau_{sis} = 0.5 \cdot ksi$

Maximum shear stress in shielding: $\tau_{pb} := \frac{F_r \cdot E_{pb}}{AE}$ $\tau_{pb} = 0.04 \cdot ksi$

Since stresses are not significant, package is demonstrated to meet acceptance criteria for vertical loading.

Determine worst case end load:

Assume as a worst case entire weight of cask is inertially loaded onto the end plate, weakest component is the weld. Also assume the plate thickness is uniform and that of the thinnest section.

Idealize as a plate with a center hole, clamped and fixed at outer and inner edges, loaded with a uniformly distributed load (Roark, 1983, Table 24, 2h).

For determination of moments redefine: $r_{io} := \frac{10.75 in}{2}$

Inertial load on end plate assume uniformly distributed: $w_{pl} := \frac{g_r \cdot W_t}{\pi \cdot \left(r_{oi}^2 - r_{io}^2\right)}$ $w_{pl} = 156 \cdot psi$

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Plate factors:

$$C2 := \frac{1}{4}\cdot\left[1-\left(\frac{r_{io}}{r_{oi}}\right)^2\cdot\left(1+2\cdot\ln\left(\frac{r_{oi}}{r_{io}}\right)\right)\right] \qquad C3 := \frac{r_{io}}{4\cdot r_{oi}}\cdot\left[\left[\left(\frac{r_{io}}{r_{oi}}\right)^2+1\right]\cdot\ln\left(\frac{r_{oi}}{r_{io}}\right)+\left(\frac{r_{io}}{r_{oi}}\right)^2-1\right]$$

$$C5 := \frac{1}{2}\cdot\left[1-\left(\frac{r_{io}}{r_{oi}}\right)^2\right] \qquad C6 := \frac{r_{io}}{4\cdot r_{oi}}\cdot\left[\left(\frac{r_{io}}{r_{oi}}\right)^2-1+2\cdot\ln\left(\frac{r_{oi}}{r_{io}}\right)\right]$$

$$C8 := \frac{1}{2}\cdot\left[1-\nu_{sst}+\left(1-\nu_{sst}\right)\cdot\left(\frac{r_{io}}{r_{oi}}\right)^2\right] \qquad C9 := \left(\frac{r_{io}}{r_{oi}}\right)\cdot\left[\frac{1+\nu_{sst}}{2}\cdot\ln\left(\frac{r_{oi}}{r_{io}}\right)+\left(\frac{1-\nu_{sst}}{4}\right)\cdot\left[1-\left(\frac{r_{io}}{r_{oi}}\right)^2\right]\right]$$

$$L11 := \frac{1}{64}\cdot\left[1+4\cdot\left(\frac{r_{io}}{r_{oi}}\right)^2-5\cdot\left(\frac{r_{io}}{r_{oi}}\right)^4-4\cdot\left(\frac{r_{io}}{r_{oi}}\right)^2\cdot\left[2+\left(\frac{r_{io}}{r_{oi}}\right)^2\right]\cdot\ln\left(\frac{r_{oi}}{r_{io}}\right)\right] \qquad L14 := \frac{1}{16}\cdot\left[1-\left(\frac{r_{io}}{r_{oi}}\right)^4-4\cdot\left(\frac{r_{io}}{r_{oi}}\right)^2\cdot\ln\left(\frac{r_{oi}}{r_{io}}\right)\right]$$

$$L17 := \frac{1}{4}\cdot\left[1-\left(\frac{1-\nu_{sst}}{4}\right)\cdot\left[1-\left(\frac{r_{io}}{r_{oi}}\right)^4\right]-\left(\frac{r_{io}}{r_{oi}}\right)^2\cdot\left[1+\left(1+\nu_{sst}\right)\cdot\ln\left(\frac{r_{oi}}{r_{io}}\right)\right]\right]$$

Axial load on inner weld: $f_{iw} := w_{pl}\cdot r_{oi}\cdot\left(\frac{C2\cdot L14-C5\cdot L11}{C2\cdot C6-C3\cdot C5}\right)$ $f_{iw} = 713\cdot\frac{lbf}{in}$

Axial load on outer weld: $f_{ow} := f_{iw}\cdot\left(\frac{r_{io}}{r_{oi}}\right)-\frac{w_{pl}}{2\cdot r_{oi}}\cdot\left(r_{oi}^2-r_{io}^2\right)$ $f_{ow} = -474\cdot\frac{lbf}{in}$

Moment on inner weld: $M_{iw} := -w_{pl}\cdot r_{oi}^2\cdot\left(\frac{C3\cdot L14-C6\cdot L11}{C2\cdot C6-C3\cdot C5}\right)$ $M_{iw} = -788\cdot lbf$

Moment on outer weld: $M_{ow} := M_{iw}\cdot C8+f_{iw}\cdot r_{oi}\cdot C9-w_{pl}\cdot r_{oi}^2\cdot L17$ $M_{ow} = -562\cdot lbf$

Axial load from moment on inner weld: $f_{biw} := \frac{\left|M_{iw}\right|}{r_{io}}$ $f_{biw} = 147\cdot\frac{lbf}{in}$

Axial load from moment on outer weld: $f_{bow} := \frac{M_{ow}}{r_{oi}}$ $f_{bow} = -45\cdot\frac{lbf}{in}$

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Total load on inner weld: $f_{tiw} := f_{iw} + f_{biw}$ $\quad f_{tiw} = 860 \cdot \frac{lbf}{in}$

Total load on outer weld: $f_{tow} := f_{ow} + f_{bow}$ $\quad f_{tow} = -520 \cdot \frac{lbf}{in}$

Determine allowable on welds:

Assuming base material strengths and ASME joint efficiency factors:

Leg length of inner weld: $w_{in} := 0.25 \cdot in$ $\quad$ Leg length of outer weld: $w_{out} := 0.5 \cdot in$

ASME joint efficiency (ASME, 1992) factor assuming no inspection: $eff := 0.60$

Allowable on inner weld: $f_{iall} := 0.707 \cdot s_{ysst} \cdot w_{in} \cdot eff$ $\quad f_{iall} = 2142 \cdot \frac{lbf}{in}$

Allowable on outer weld: $f_{oall} := 0.707 \cdot s_{ysst} \cdot w_{out} \cdot eff$ $\quad f_{oall} = 4284 \cdot \frac{lbf}{in}$

Margin of safety on inner weld: $MS_{iw} := \frac{f_{iall}}{f_{tiw}} - 1$ $\quad MS_{iw} = 1.49$ ☜

Margin of safety on outer weld: $MS_{ow} := \frac{f_{oall}}{\left|f_{tow}\right|} - 1$ $\quad MS_{ow} = 7.25$ ☜

All margins of safety are positive, therefore OK.

Determine end loading of top end closure bolts:

Assume payload is restrained in cavity by dunnage and bears against the top and bottom closures. Worst case loading would occur at the top closure, since it has the fewest number of bolts. Loading results from shield plug and payload.

Weight of shield plug, ignore inner plumbing:

Plate diameter: $d_{spl} := 13.5 \cdot in$ $\quad$ Plate thickness: $t_{pl} := 0.75 \cdot in$

Shell OD: $od_{sh} := 10 \cdot in$ $\quad$ Wall thickness: $t_{sh} := 0.375 \cdot in$ $\quad$ Length: $l_{sh} := 13.37 \cdot in - t_{pl}$

Inner shell OD: $od_{ish} := od_{sh} - 2 \cdot t_{sh}$ $\quad$ Inner shell length: $l_{ish} := 3 \cdot in$ $\quad$ Inner plate thickness: $t_{ipl} := 0.5 \cdot in$

ID of inner shell: $id_{ish} := 7.625 \cdot in$

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Lead length: $l_{pb} := l_{sh} - 3.5\cdot in$ Stainless steel density: $\rho_{sst} := 0.29\frac{lbf}{in^3}$

Weight of stainless steel:

$$w_{sst} := \rho_{sst}\cdot\left[\pi\cdot\frac{d_{spl}^2}{4}\cdot t_{pl} + \pi\cdot\left(\frac{od_{sh}^2 - od_{ish}^2}{4}\right)\cdot l_{sh} + \pi\cdot\frac{od_{ish}^2}{4}\cdot t_{ipl} + \pi\cdot\frac{od_{ish}^2 - id_{ish}^2}{4}\cdot l_{ish}\right]$$

$w_{sst} = 101\cdot lbf$

Weight of lead: $w_{pb} := \rho_{pb}\cdot\frac{od_{ish}^2}{4}\cdot l_{pb}$ $w_{pb} = 80\cdot lbf$

Total plug weight: $w_{plug} := w_{sst} + w_{pb}$ $w_{plug} = 181\cdot lbf$

Assume bolts are uniformly loaded:

Nominal diameter of bolts: $d_b := 0.5\cdot in$ Lead: $l_b := \frac{13}{in}$

Tensile stress area: $A_s := 0.7854\left(d_b - \frac{0.9743}{l_b}\right)^2$ Number of bolts: $n_{bolt} := 18$

Preload on bolts:

Assume preload torque for bolts is: $T_{pre} := 240\cdot lbf\cdot in$ $T_{pre} = 20\cdot ft\cdot lbf$

Assume nut friction factor: $\mu_n := 0.2$

Preload force per bolt: $F_{pre} := \frac{T_{pre}}{d_b\cdot\mu_n}$ $F_{pre} = 2400\cdot lbf$

Total load on top closure bolts: $F_{cl} := n_{bolt}\cdot F_{pre} - g_v\cdot\left(w_{plug} + W_{pay}\right)$ $F_{cl} = 40816\cdot lbf$ ☜

Tensile stress on bolts: $\sigma_{ten} := \frac{F_{cl}}{n_{bolt}\cdot A_s}$ $\sigma_{ten} = 16\cdot ksi$ ☜

Margin of safety: $MS_{bolt} := \frac{s_{ab}}{\sigma_{ten}} - 1$ $MS_{bolt} = 0.11$ ☜

Therefore OK, since margin of safety is positive.

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Lifting Evaluation:

Determine if the cask lifting yoke meets the 3 to 1 requirements of ANSI N14.6, 1993:

Idealize lifting yoke as a short beam uniformly loaded and clamped and fixed at each end:

Length of yoke: $l_y := (4.00 - 0.75)\cdot in$ Diameter of yoke: $d_y := 3.00\,in$

Load on yoke: $F_{load} := \frac{W_t}{2}$ Dynamic load factor: $DLF := 1.25$

Cross sectional area: $A_y := \pi\cdot\frac{d_y^{\,2}}{4}$ Moment of inertia: $I_y := \pi\cdot\frac{d_y^{\,4}}{64}$

Maximum bending stress at end: $\sigma_{yb} := \frac{1}{12}\cdot\left(\frac{DLF\cdot F_{load}\cdot l_y}{I_y}\cdot\frac{d_y}{2}\right)$ $\sigma_{yb} = 1.69\cdot ksi$ ☜

Maximum shear stress: $\tau_{max} := \frac{4}{3}\cdot\frac{DLF\cdot F_{load}}{A_y}$ $\tau_{max} = 3.12\cdot ksi$ ☜

Lifting of cask is accomplished by lifting at the lifting points of the outrigger support plates :

Assume a straight vertical lift with spreader bar:

Assumed hole diameter: $d_h = 1.375\,in$ Plate thickness: $t_{pl} := 1.5\cdot in$

Number of outrigger support plates for lifting: $n_{pl} := 2$

Number of lift points: $n_{lift} := 4$ Assumed distance to edge: $d_{edge} := 2\cdot in - \frac{d_h}{2}$

Load on each lift point: $F_{load} := \frac{W_t}{n_{lift}}$ Dynamic load factor: $DLF := 1.25$

Shear tearout of lift point: $\tau_{to} := \frac{DLF\cdot F_{load}}{2\cdot\left(t_{pl}\cdot d_{edge}\right)}$ $\tau_{to} = 2.1\cdot ksi$ ☜

Bearing stress: $\sigma_{bear} := \frac{DLF\cdot F_{load}}{d_h\cdot t_{pl}}$ $\sigma_{bear} = 4.02\cdot ksi$ ☜

Assume each plate is welded for 30° of circumference, and welded on both sides:

Angular length: $\alpha_w := 30\cdot deg$ $\alpha_w = 0.52\cdot rad$

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Length of weld: $l_{ws} := r_{oo} \cdot \alpha_w$ $\qquad l_{ws} = 6.81 \cdot in$

Load on welds: $f_{wl} := \dfrac{DLF \cdot F_{load}}{l_{ws}}$ $\qquad f_{wl} = 1217 \cdot \dfrac{lbf}{in}$ ☜

Loads are well under yield strength of material.

NTC Penetration of Package:

Mass of projectile: $m_p := 6 \cdot kg$ Height of drop: $h_{dr} := 1 \cdot m$ *Hemispherical end diameter:* $d_{hemi} := 3.2 \cdot cm$

Evaluate package penetration by empirical methods (Rhinehart and Pearson, 1954):

Velocity of projectile: $v_o := \sqrt{2 \cdot g \cdot h_{dr}}$ $\qquad v_o = 14.5 \cdot \dfrac{ft}{sec}$

Assuming the test rod is a hard unyielding object, at this velocity (< 10,000 ft/sec), the force that acts on the projectile is proportional to the cross sectional area and is essentially constant during penetration.

Cross sectional area of projectile: $A_p := \pi \cdot \dfrac{d_{hemi}^2}{4}$ $\qquad A_p = 1.25 \cdot in^2$

Volume of material displaced per unit of energy constant (Rhinehart and Pearson, 1954, page 202, Table 12-1):

For steel: $K_s := 0.26 \, 10^{-4} \cdot \dfrac{in^3}{ft \cdot lbf}$

Depth of penetration into steel: $s_{sp} := \dfrac{1}{2} \cdot m_p \cdot v_o^2 \cdot \dfrac{K_s}{A_p}$ $\qquad s_{sp} = 0.001 \cdot in$ ☜

The penetration depth is not significant, no penetration into the package.

<u>SERF Cask Puncture Failure Thresholds, Equivalent Steel Thickness:</u>

Equivalent Thickness of lead to steel (Rinehart, 1954): $f_{equ} := 2.3$

OD of Outer Shell: $od_{o6} := 26 \cdot in$ $\qquad$ Outer Shell Wall Thickness: $t_{ow6} := 0.625 in$

ID of Outer Shell: $id_{o6} := od_{o6} - 2 \cdot t_{ow6}$ $\qquad id_{o6} = 24.75 \cdot in$

OD of Inner Shell: $od_{i6} := 8.625 in$ $\qquad$ ID of Inner Shell: $id_{i6} := 7.625 in$

Inner Shell Wall Thickness: $t_{iw6} := \dfrac{od_{i6} - id_{i6}}{2}$ $\qquad t_{iw6} = 0.5 \cdot in$

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Lead Thickness: $t_{pb6} := \frac{id_{o6} - od_{i6}}{2}$ $\quad t_{pb6} = 8.06 \cdot in$

Total Equivalent thickness of steel: $t_{eq6} := t_{ow6} + t_{iw6} + \frac{t_{pb6}}{f_{equ}}$ $\quad t_{eq6} = 4.6 \cdot in$ ☜

7.5.2 Pressure Evaluation

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Preparer: S. S. Shiraga Date 09/10/97
Checker: P. C. Ferrell Date 09/11/97
Section Chief: S. S. Shiraga Date 09/11/97

1.0 OBJECTIVE

The objective of this evaluation is to verify the 20,684 MPa (3,000 psi) pressure capacity of the SERF canister overpack.

2.0 REFERENCES

ASME, 1995, *Boiler and Pressure Vessel Code*, Section II, Part D, American Society of Mechanical Engineers, New York, New York.

ASME, 1992, *Boiler and Pressure Vessel Code*, Section III, Appendices, American Society of Mechanical Engineers, New York, New York.

ASME, 1992, *Boiler and Pressure Vessel Code*, Section III, Subsection NE, American Society of Mechanical Engineers, New York, New York.

3.0 ASSUMPTIONS, RESULTS, AND CONCLUSIONS

In this evaluation the overpack is assumed to be designed as shown in the first figure of the evaluation. It is assumed to be constructed of 304 stainless steel manufactured to ASTM A240 or A 269 requirements. For this classical linear elastic analysis, the overpack is idealized as a cylindrical tube with flat ends of unequal thickness. The overpack can be provided with a welded flat bottom end plate and a threaded fitting on the other end or a threaded fitting can be provided on each end. The overpack is evaluated per the requirements of ASME Section III, Subsection NE.

The results of this evaluation outline the minimum requirements for the overpack. The overpack must be a 1 ⅜ in. 304 stainless steel tube with a minimum wall thickness of 0.109 in. If the overpack is provided with a welded flat bottom end plate, the plate must be welded to the tube via a socket joint, with a minimum weld thickness being the same as the wall of the tube. The plate must have a center thickness of ¼ in. and the minimum thickness of the joint landing must 0.170 in. The threaded end fitting must have either NPT or UNF threads and must extend into the tube a minimum of ¼ in.

This overpack provides the 3,000 psi pressure boundary mandated in the SARP, for encapsulating the dispersible material and fuel scrap canisters.

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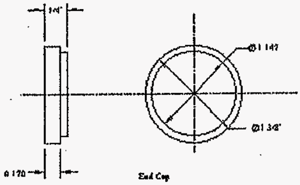
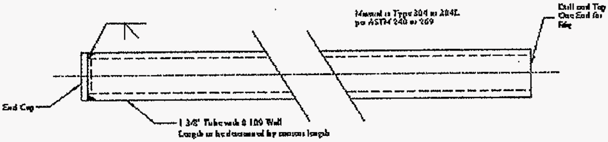
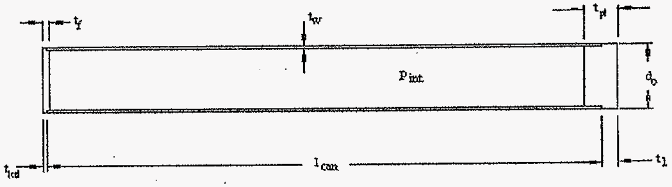
4.0 EVALUATION

SERF Canister Overpack Pressure Capacity Determination





Evaluate Canister Overpack for internal pressure to ASME Section III, Appendix, Article A-2000, A-5000, and A-6000. Idealize as a cylindrical vessel with flat ends.



Maximum internal pressure: $p_{int} := 3000\,psi$ Length of canister: $l_{can} := 37 \cdot in$

Outside diameter: $d_o := 1.375\,in$ Wall thickness: $t_w := 0.109\,in$

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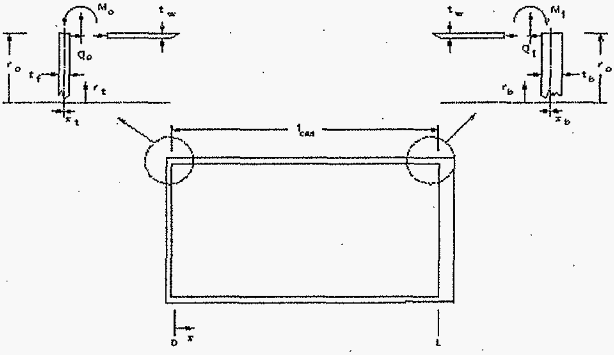

Material Properties from ASME Section II, Part D assumed at 200°F.

Assume material equivalent to SA-240 and evaluate to Section III, Subsection NE:

Elastic modulus (same for lid, vessel, bottom forging and bolts): $E_{sst} := 27.6 \cdot 10^{6} \cdot psi$

Mean coefficient of thermal expansion (same for all): $\alpha_{sst} := 8.79 \cdot 10^{-6} \cdot \frac{in}{in}$

Assume Poisson's Ratio: $\nu_{sst} := 0.29$

Allowable stress intensity: $s_m := 18.8 \cdot ksi$

Subsection NE allowable: $s_{ne} := 1.1 \cdot s_m$ $s_{ne} = 20.7 \cdot ksi$

Geometric parameters:

Outside radius of shell: $r_o := \frac{d_o}{2}$ $r_o = 0.69 \cdot in$ Thickness of step end: $t_{led} := 0.170 in$

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Thickness of at center plate: $t_f := 0.250\,in$ Assumed plug fitting thickness: $t_{pl} := 0.5 \cdot in$

Assumed plug fitting thickness at step end: $t_1 := 0.25\,in$

Inside radius of shell: $r_i := r_o - t_w$ $r_i = 0.58 \cdot in$ mean radius: $r_m := \dfrac{r_o + r_i}{2}$

Shell factors: $$\beta := \left[\frac{3 \cdot \left(1 - \nu_{sst}^{2}\right)}{\left(r_i + \dfrac{t_w}{2}\right)^{2} \cdot t_w^{2}} \right]^{\frac{1}{4}} \qquad \beta = 4.9 \cdot in^{-1} \qquad D_s := \frac{E_{sst} \cdot t_w^{3}}{12 \cdot \left(1 - \nu_{sst}^{2}\right)} \qquad D_s = 3252.07 \cdot lbf \cdot in$$

Length factor: $\dfrac{3}{\beta} = 0.61 \cdot in$ and $l_{can} = 37 \cdot in$ Therefore can be considered a long cylinder.

Ratio of outside radius to intermediate radius: $Z_s := \dfrac{r_o}{r_m}$ $Z_s = 1.09$

Ratio of outside radius to inside radius: $Y_s := \dfrac{r_o}{r_i}$ $Y_s = 1.19$

Shell influence coefficients:

$$B_{11} := \frac{\sinh\left(2 \cdot \beta \cdot l_{can}\right) - \sin\left(2 \cdot \beta \cdot l_{can}\right)}{2 \cdot \left(\sinh\left(\beta \cdot l_{can}\right)^{2} - \sin\left(\beta \cdot l_{can}\right)^{2}\right)} \qquad B_{12} := \frac{\cosh\left(2 \cdot \beta \cdot l_{can}\right) - \cos\left(2 \cdot \beta \cdot l_{can}\right)}{2 \cdot \left(\sinh\left(\beta \cdot l_{can}\right)^{2} - \sin\left(\beta \cdot l_{can}\right)^{2}\right)}$$

$$B_{22} := \frac{\sinh\left(2 \cdot \beta \cdot l_{can}\right) + \sin\left(2 \cdot \beta \cdot l_{can}\right)}{2 \cdot \left(\sinh\left(\beta \cdot l_{can}\right)^{2} - \sin\left(\beta \cdot l_{can}\right)^{2}\right)} \qquad G_{11} := \frac{\cosh\left(\beta \cdot l_{can}\right) \cdot \sin\left(\beta \cdot l_{can}\right) - \sinh\left(\beta \cdot l_{can}\right) \cdot \cos\left(\beta \cdot l_{can}\right)}{\sinh\left(\beta \cdot l_{can}\right)^{2} - \sin\left(\beta \cdot l_{can}\right)^{2}}$$

$$G_{12} := \frac{-2 \cdot \sinh\left(\beta \cdot l_{can}\right) \cdot \sin\left(\beta \cdot l_{can}\right)}{\sinh\left(\beta \cdot l_{can}\right)^{2} - \sin\left(\beta \cdot l_{can}\right)^{2}} \qquad G_{22} := \frac{-2 \cdot \left(\cosh\left(\beta \cdot l_{can}\right) \cdot \sin\left(\beta \cdot l_{can}\right) + \sinh\left(\beta \cdot l_{can}\right) \cdot \cos\left(\beta \cdot l_{can}\right)\right)}{\sinh\left(\beta \cdot l_{can}\right)^{2} - \sin\left(\beta \cdot l_{can}\right)^{2}}$$

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Lid flat plate geometric constants:

Form factor: $f_l := \frac{t_{led}}{r_o}$ $f_l = 0.25$ Poisson's ratio: $\nu_{sst} = 0.29$

$$F_{l1} := \frac{3\cdot(1-\nu_{sst})\cdot\left(2-f_l^2\right)\cdot(1-f_l)^2\cdot\left[8-f_l\cdot(4-f_l)\cdot(1-\nu_{sst})\right]}{16\cdot(2-f_l)}$$

$$F_{l2} := \frac{3}{8}\cdot(1-f_l)^2\cdot\left[(1-\nu_{sst})\cdot\left(2-f_l^2\right)+4\cdot(1+\nu_{sst})\cdot\left(1+2\cdot\ln\left(\frac{2-f_l}{2-2\cdot f_l}\right)\right)\right]$$

$$F_{l3} := \frac{3}{8}\cdot(1-\nu_{sst})\cdot(2-f_l)\cdot\left[8-f_l\cdot(4-f_l)\cdot(1-\nu_{sst})\right] \qquad F_{l4} := \frac{1}{8}\cdot\left[8-f_l\cdot(4-f_l)\cdot(1-\nu_{sst})\right]$$

Bottom flange geometric constants:

Form factor: $f_b := \frac{t_1}{r_o}$ $f_b = 0.36$ Poisson's ratio: $\nu_{sst} = 0.29$

$$F_{b1} := \frac{3\cdot(1-\nu_{sst})\cdot\left(2-f_b^2\right)\cdot(1-f_b)^2\cdot\left[8-f_b\cdot(4-f_b)\cdot(1-\nu_{sst})\right]}{16\cdot(2-f_b)}$$

$$F_{b2} := \frac{3}{8}\cdot(1-f_b)^2\cdot\left[(1-\nu_{sst})\cdot\left(2-f_b^2\right)+4\cdot(1+\nu_{sst})\cdot\left(1+2\cdot\ln\left(\frac{2-f_b}{2-2\cdot f_b}\right)\right)\right]$$

$$F_{b3} := \frac{3}{8}\cdot(1-\nu_{sst})\cdot(2-f_b)\cdot\left[8-f_b\cdot(4-f_b)\cdot(1-\nu_{sst})\right] \qquad F_{b4} := \frac{1}{8}\cdot\left[8-f_b\cdot(4-f_b)\cdot(1-\nu_{sst})\right]$$

Principal stresses in shell:

Tangential: $\sigma_t := p_{int}\cdot\frac{1+Z_s^2}{Y_s^2-1}$ $\sigma_t = 15.86\cdot\text{ksi}$ Longitudinal: $\sigma_l := \frac{p_{int}}{Y_s^2-1}$ $\sigma_l = 7.28\cdot\text{ksi}$

Radial: $\sigma_r := p_{int}\cdot\frac{1-Z_s^2}{Y_s^2-1}$ $\sigma_r = -1.31\cdot\text{ksi}$

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General primary membrane stress intensity: $\sigma_{gpsi} := \frac{p_{int} \cdot r_i}{t_w} + \frac{p_{int}}{2}$ $\sigma_{gpsi} = 17 \cdot ksi$

Maxmum value of primary plus secondary stress intensity:

$$\sigma_{mps} := \frac{2 \cdot p_{int} \cdot Y_s^2}{Y_s^2 - 1} \qquad \sigma_{mps} = 20.6 \cdot ksi$$

Margin of Safety primary membrane: $MS_{pm1} := \frac{s_{ne}}{\sigma_{gpsi}} - 1$ $MS_{pm1} = 0.19$ ☜

Margin of Safety primary plus secondary stress intensity: $MS_{ps1} := \frac{3 \cdot s_{ne}}{\sigma_{mps}} - 1$ $MS_{ps1} = 2.02$ ☜

Displacements at joint location O lid section due to pressure:

Rotational displacement: $\theta_{li} := \frac{F_{l1}}{E_{sst} \cdot \left(\frac{t_{led}}{r_o}\right)^3} \cdot p_{int}$ Radial displacement: $\delta_{li} := -\frac{t_{led}}{2} \cdot \theta_{li}$

Displacement at joint location L bottom section due to pressure:

Rotational displacement: $\theta_b := -\frac{F_{b1}}{E_{sst} \cdot \left(\frac{t_l}{r_o}\right)^3} \cdot p_{int}$ Radial displacement: $\delta_b := -\frac{t_l}{2} \cdot \theta_b$

At either end of shell, displacement of midsurface due to pressure:

Radial displacement: $\delta_{sh} := p_{int} \cdot \left[\frac{r_i^2}{E_{sst} \cdot \left(r_o^2 - r_i^2\right) \cdot r_m}\right] \cdot \left[r_m^2 \cdot \left(1 - 2 \cdot \nu_{sst}\right) + r_o^2 \cdot \left(1 + \nu_{sst}\right)\right]$

Rotational displacement: $\theta_{sh} := 0$

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Total displacements at junction O:

Radial displacement:

$$\frac{B_{11}}{2\cdot\beta^3\cdot D_s}\cdot Q_O+\frac{B_{12}}{\beta^2\cdot D_s}\cdot M_O+\frac{G_{11}}{2\cdot\beta^3\cdot D_s}\cdot Q_L+\frac{G_{12}}{2\cdot\beta^2\cdot D_s}\cdot M_L+\delta_{sh}=\frac{-2\cdot F_{13}}{3\cdot E_{sst}\cdot\left(\frac{t_{led}}{r_o}\right)}\cdot Q_O+\frac{F_{13}}{E_{sst}\cdot r_o\cdot\left(\frac{t_{led}}{r_o}\right)^2}\cdot M_O+\delta_{li}$$

Rotational displacement:

$$-\left[\frac{B_{12}}{2\cdot\beta^2\cdot D_s}\cdot Q_O+\frac{B_{22}}{2\cdot\beta\cdot D_s}\cdot M_O+\frac{G_{12}}{2\cdot\beta^2\cdot D_s}\cdot Q_L+\frac{G_{22}}{2\cdot\beta\cdot D_s}\cdot M_L\right]+\theta_{sh}=\frac{F_{13}}{E_{sst}\cdot r_o\cdot\left(\frac{t_{led}}{r_o}\right)^2}\cdot Q_O-\frac{2\cdot F_{13}}{E_{sst}\cdot r_o^{\,2}\cdot\left(\frac{t_{led}}{r_o}\right)^3}\cdot M_O+\theta_{li}$$

Total displacements at junction L:

Radial displacement:

$$\frac{G_{11}}{2\cdot\beta^3\cdot D_s}\cdot Q_O+\frac{G_{12}}{2\cdot\beta^2\cdot D_s}\cdot M_O+\frac{B_{11}}{2\cdot\beta^3\cdot D_s}\cdot Q_L+\frac{B_{12}}{2\cdot\beta^2\cdot D_s}\cdot M_L+\delta_{sh}=\frac{-2\cdot F_{b3}}{3\cdot E_{sst}\cdot\left(\frac{t_l}{r_o}\right)}\cdot Q_L+\frac{F_{b3}}{E_{sst}\cdot r_o\cdot\left(\frac{t_l}{r_o}\right)^2}\cdot M_L+\delta_b$$

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Rotational displacement:

$$\frac{G_{12}}{2\cdot\beta^2\cdot D_s}\cdot Q_O+\frac{G_{22}}{2\cdot\beta\cdot D_s}\cdot M_O\cdots +\frac{B_{12}}{2\cdot\beta^2\cdot D_s}\cdot Q_L+\frac{B_{22}}{\beta\cdot D_s}\cdot M_L+\theta_{sh} = \frac{F_{b3}}{E_{sst}\cdot r_o\cdot\left(\frac{t_l}{r_o}\right)^2}\cdot Q_L-\frac{2\cdot F_{b3}}{E_{sst}\cdot r_o^{\,2}\cdot\left(\frac{t_l}{r_o}\right)^3}\cdot M_L+\theta_b$$

Use Mathcad solve block function to solve for four unknowns with four equations (Mathcad, 1994).

$$\begin{bmatrix} Q_O \\ M_O \\ Q_L \\ M_L \end{bmatrix} := \mathrm{Find}\left(Q_O, M_O, Q_L, M_L\right)$$

Radial shear forces: $Q_O = -468\cdot\frac{\mathrm{lbf}}{\mathrm{in}}$ $Q_L = -115\cdot\frac{\mathrm{lbf}}{\mathrm{in}}$

Longitudinal bending moments: $M_O = 7\cdot\mathrm{lbf}$ $M_L = -5\cdot\mathrm{lbf}$

For other axial locations, the combined effects of loading at the two edges may be evaluated by applying the equations to the loading at each edge, seperately and superimposing the results.

Principal stresses due to bending at location χ=0 in. (Discontinuity): $x_o := 0\cdot\mathrm{in}$

Shell factors: $\beta_o := \left[\frac{3\cdot\left(1-\nu_{sst}^{\,2}\right)}{\left(r_i+\frac{t_w}{2}\right)^2\cdot t_w^{\,2}}\right]^{\frac{1}{4}}$ $\beta_o = 4.9\cdot\mathrm{in}^{-1}$ $D_{os} := \frac{E_{sst}\cdot t_w^{\,3}}{12\cdot\left(1-\nu_{sst}^{\,2}\right)}$ $D_{os} = 3252\cdot\mathrm{lbf}\cdot\mathrm{in}$

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Influence coefficients at location:

$f_1 := e^{\beta_o \cdot x_o} \cdot \cos(\beta_o \cdot x_o)$ $\quad f_1 = 1$ $\quad f_2 := e^{-\beta_o \cdot x_o} \cdot (\cos(\beta_o \cdot x_o) - \sin(\beta_o \cdot x_o))$ $\quad f_2 = 1$

$f_3 := e^{-\beta_o \cdot x_o} \cdot (\cos(\beta_o \cdot x_o) + \sin(\beta_o \cdot x_o))$ $\quad f_3 = 1$ $\quad f_4 := e^{-\beta_o \cdot x_o} \cdot \sin(\beta_o \cdot x_o)$ $\quad f_4 = 0$

Radial displacement: $\delta_{xo1} := \frac{Q_O}{2 \cdot \beta_o^3 \cdot D_{os}} \cdot f_1 + \frac{M_O}{2 \cdot \beta_o^2 \cdot D_{os}} \cdot f_2$ $\quad \delta_{xo1} = -5.66 \cdot 10^{-4} \cdot in$

Rotational displacement: $\theta_{xo1} := \left[\frac{Q_O}{2 \cdot \beta_o^3 \cdot D_{os}} \cdot f_3 - 2 \cdot \left(\frac{M_O}{2 \cdot \beta_o^2 \cdot D_{os}} \right) \cdot f_1 \right] \cdot \beta_o$ $\quad \theta_{xo1} = 2.55 \cdot 10^{-3}$

Longitudinal bendin moment: $M_{xo1} := \left(\frac{Q_O}{2 \cdot \beta_o^3 \cdot D_{os}} \cdot f_4 + \frac{M_O}{2 \cdot \beta_o^2 \cdot D_{os}} \cdot f_3 \right) \cdot 2 \cdot \beta_o^2 \cdot D_{os}$ $\quad M_{xo1} = 7 \cdot lbf$

Radial shear force: $Q_{xo1} := \left(\frac{Q_O}{2 \cdot \beta_o^3 \cdot D_{os}} \cdot f_2 - 2 \cdot \frac{M_O}{2 \cdot \beta_o^2 \cdot D_{os}} \cdot f_4 \right) \cdot 2 \cdot \beta_o^3 \cdot D_{os}$ $\quad Q_{xo1} = -468 \cdot \frac{lbf}{in}$

Tangential: $\sigma_{tbo} := \frac{E_{sst} \cdot \delta_{xo1}}{r_i + \frac{t_w}{2}} - \frac{6 \cdot \nu_{sst} \cdot M_{xo1}}{t_w^2}$ $\quad \sigma_{tbo} = -25.7 \cdot ksi$

Longitudinal: $\sigma_{lbo} = \frac{6 \cdot M_{xo1}}{t_w^2}$ $\quad \sigma_{lbo} = 3.6 \cdot ksi$

Radial: $\sigma_{rbo} := 0 \cdot ksi$

Radial Shear: $\tau_{rso} := \frac{|Q_{xo1}|}{t_w}$ $\quad \tau_{rso} = 4.3 \cdot ksi$

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Since at discontinuity these are secondary stresses, maximum secondary stress intensity:

$$s_{o1} := \sigma_{lbo} - \sigma_{tbo} \qquad s_{o1} = 29.3 \cdot ksi$$

Maximum total stress intensity at joint:

$$si_o := \sigma_{mps} + s_{o1} + \tau_{rso} \qquad si_o = 54.1 \cdot ksi$$

Margin of Safety shear stress: $MS_{ss1} := \dfrac{0.6 \cdot s_{ne}}{\tau_{rso}} - 1$ $\quad MS_{ss1} = 1.89$ ☜

Margin of Safety primary plus secondary stress intensity: $MS_{ps2} := \dfrac{3 \cdot s_{ne}}{si_o} - 1$ $\quad MS_{ps2} = 0.15$ ☜

Principal stresses due to bending at location $x = l_c$ (Discontinuity):

Location: $x_l := 0 \cdot in$

Shell factors:

$$\beta_l := \left[\frac{3 \cdot \left(1 - \nu_{sst}^{\,2}\right)}{\left(r_i + \frac{t_w}{2}\right)^2 \cdot t_w^{\,2}} \right]^{\frac{1}{4}} \qquad \beta_l = 4.9 \cdot in^{-1} \qquad D_{ls} := \frac{E_{sst} \cdot t_w^{\,3}}{12 \cdot \left(1 - \nu_{sst}^{\,2}\right)} \qquad D_{ls} = 3252 \cdot lbf \cdot in$$

Influence coefficients at location:

$$f_{1a} := e^{-\beta_l \cdot x_l} \cdot \cos\left(\beta_l \cdot x_l\right) \qquad f_{1a} = 1 \qquad f_{2a} := e^{-\beta_l \cdot x_l} \cdot \left(\cos\left(\beta_l \cdot x_l\right) - \sin\left(\beta_l \cdot x_l\right)\right) \qquad f_{2a} = 1$$

$$f_{3a} := e^{-\beta_l \cdot x_l} \cdot \left(\cos\left(\beta_l \cdot x_l\right) + \sin\left(\beta_l \cdot x_l\right)\right) \qquad f_{3a} = 1 \qquad f_{4a} := e^{-\beta_l \cdot x_l} \cdot \sin\left(\beta_l \cdot x_l\right) \qquad f_{4a} = 0$$

Radial displacement:

$$\delta_{xl} := \frac{Q_L}{2 \cdot \beta_l^{\,3} \cdot D_{ls}} \cdot f_{1a} + \frac{M_L}{2 \cdot \beta_l^{\,2} \cdot D_{ls}} \cdot f_{2a} \qquad \delta_{xl} = -1.84 \cdot 10^{-4} \cdot in$$

Rotational displacement:

$$\theta_{xl} := \left[-\frac{Q_L}{2 \cdot \beta_l^{\,3} \cdot D_{ls}} \cdot f_{3a} - 2 \cdot \left(\frac{M_L}{2 \cdot \beta_l^{\,2} \cdot D_{ls}} \right) \cdot f_{1a} \right] \cdot \beta_l \qquad \theta_{xl} = 0.001$$

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Longitudinal bendin moment: $M_{xl} := \left(\frac{Q_L}{2 \cdot \beta_l^3 \cdot D_{ls}} \cdot f_{4a} + \frac{M_L}{2 \cdot \beta_l^2 \cdot D_{ls}} \cdot f_{3a} \right) \cdot 2 \cdot \beta_l^2 \cdot D_{ls}$ $M_{xl} = -5 \cdot \mathrm{lbf}$

Radial shear force: $Q_{xl} := \left(\frac{Q_L}{2 \cdot \beta_l^3 \cdot D_{ls}} \cdot f_{2a} - 2 \cdot \frac{M_L}{2 \cdot \beta_l^2 \cdot D_{ls}} \cdot f_{4a} \right) \cdot 2 \cdot \beta_l^3 \cdot D_{ls}$ $Q_{xl} = -115 \cdot \frac{\mathrm{lbf}}{\mathrm{in}}$

Tangential: $\sigma_{tbL} := \frac{E_{sst} \cdot \delta_{xl}}{r_i + \frac{t_w}{2}} + \frac{6 \cdot \nu_{sst} \cdot M_{xl}}{t_w^2}$ $\sigma_{tbL} = -9 \cdot \mathrm{ksi}$

Longitudinal: $\sigma_{lbL} := \frac{6 \cdot M_{xl}}{t_w^2}$ $\sigma_{lbL} = -2.7 \cdot \mathrm{ksi}$

Radial: $\sigma_{rbL} := 0 \cdot \mathrm{ksi}$

Radial Shear: $\tau_{rsL} := \frac{|Q_{xl}|}{t_w}$ $\tau_{rsL} = 1.1 \cdot \mathrm{ksi}$

Since at discontinuity these are secondary stresses, maximum secondary stress intensity:

$s_l := \sigma_{tbL} - \sigma_{tbL}$ $s_l = 9 \cdot \mathrm{ksi}$

Maximum total stress intensity at bottom end joint:

$si_l := \sigma_{mps} + s_l + \tau_{rsL}$ $si_l = 30 \cdot \mathrm{ksi}$

Margin of Safety shear stress: $MS_{ss2} := \frac{0.6 \cdot s_{ne}}{\tau_{rsL}} - 1$ $MS_{ss2} = 10.76$ ☜

Margin of Safety primary plus secondary stress intensity: $MS_{ps3} := \frac{3 \cdot s_{ne}}{si_l} - 1$ $MS_{ps3} = 1.04$ ☜

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Determine loading at edge of end closure by methods outlined in Article A-5000, Appendices.

Location: $x_t := \frac{t_f}{2}$ Form factor: $f_f := \frac{t_f}{r_o}$

Radial location of highest impact load: $r_t := 0 \cdot in$

Geometric Constants:

$$F_2 := \frac{3}{8} \cdot \left(1 - f_f\right)^2 \cdot \left[\left(1 - \nu_{sst}\right) \cdot \left(2 - f_f^2\right) + 4 \cdot \left(1 + \nu_{sst}\right) \cdot \left(1 + 2 \cdot \ln\left(\frac{2 - f_f}{2 - 2 \cdot f_f}\right)\right)\right] \qquad F_2 = 1.38$$

$$F_4 := \frac{1}{8} \cdot \left[8 - f_f \left(4 - f_f\right) \cdot \left(1 - \nu_{sst}\right)\right] \qquad F_4 = 0.88$$

Radial and Tangential stress at center due to pressure:

$$\sigma_{rc} := \frac{p_{int} \cdot x_t}{t_f \left(\frac{t_f}{r_o}\right)^2} \cdot \left[F_2 - \frac{3 \cdot \left(3 + \nu_{sst}\right) \cdot r_t}{4 \cdot r_o}\right] \qquad \sigma_{rc} = 16 \cdot ksi$$

Radial and Tangential stress at center due to radial force and moment:

$$\sigma_{rt} := \frac{F_4}{t_f} \cdot \left(1 - \frac{6 \cdot x_t}{t_f}\right) \cdot Q_L - \frac{12 \cdot F_4 \cdot x_t}{t_f^3} \cdot M_L \qquad \sigma_{rt} = 1.3 \cdot ksi$$

Maximum stress intensity:

$$s_{bec} := \sigma_{rc} + \left|\sigma_{rt}\right| \qquad s_{bec} = 16.9 \cdot ksi$$

Margin of safety on primary membrane stress intensity: $MS_{pm2} := \frac{s_{ne}}{\sigma_{rc}} - 1$ $MS_{pm2} = 0.32$ ☜

Margin of safety on primary plus secondary stress intensity: $MS_{ps4} := \frac{3 \cdot s_{ne}}{s_{bec}} - 1$ $MS_{ps4} = 2.67$ ☜

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8.0 THERMAL EVALUATION

8.1 INTRODUCTION

The SERF Cask is an onsite, intra-area packaging for transferring Type B radioactive material within the 300 Area of the Hanford Site. This section of the document defines and evaluates the normal transport condition structural requirements for intra-area transport of this package. Thermal performance of the package is evaluated only for normal transport conditions; accident conditions are evaluated in the risk and dose consequence sections of this document.

8.2 THERMAL EVALUATION OF PACKAGE

8.2.1 Package Description

The SERF Cask is fundamentally a right circular cylinder of lead and stainless steel composite construction. The lead is sandwiched between outer and inner stainless steel tubular shells. At each end of the cask is a thick circular plate that is welded to both inner and outer shells, which encapsulates the lead. At the top, sealing and closure of the inner cavity are provided by a bolted blind flange with an attached shield plug of composite lead-stainless steel construction. The cask is equipped with a manually actuated integral closure value, which provides closure at the bottom end. A secondary, stainless steel, gasketed, blind flange plate is bolted on to provide sealing of the cask during transport. A handling yoke is provided on the cask and welded to the outer shell. In addition, support plates are welded to the outer housing of the cask to provide support during transport and lifting attachment anchors for handling.

The cask is constructed of 304 or 304L stainless steel, which is welded by welders and weld procedures qualified to Section IX of the ASME B&PV Code (ASME 1977a) in use at the time of construction. Nondestructive examinations of the welds were performed in accordance with Section V of the ASME B&PV Code (ASME 1977b). The lead shielding meeting the requirement of ASTM B29 (ASTM 1976) was poured into the shielding cavity after construction of the cask.

8.3 NORMAL TRANSPORT CONDITIONS THERMAL EVALUATION

8.3.1 Conditions To Be Evaluated

Thermal performance of the package is assessed for Hanford Site normal transport conditions in this section. The package is evaluated for the worst-case Hanford Site thermal loading condition of a still-air ambient temperature of 46 °C (115 °F [Fadeff 1992]) with decay heat sources with and without solar insolation. As a worst case the thermal performance of the package is evaluated with two cesium chloride capsules producing a total heat load of 377 W.

8.3.2 Acceptance Criteria

The criterion for acceptable performance of the package is the accessible surface of the package in still air at 46 °C (115 °F) and in the shade is not to exceed 85 °C (185 °F). This is based on this package being transported as an exclusive-use shipment. Also the maximum temperature of the cesium chloride capsules, which are assumed as the worst case thermal payload, must be maintained below 800 °C (1472 °F).

8.3.3 Thermal Evaluation and Conclusions

For this evaluation the worst-case decay heat source is assumed to generate a total heat load of 377 W. This heat load is based on two cesium chloride capsules loaded in the boat in the package. Since the capsules are loaded in the boat, the capsules can be assumed to be radially centered in the cask. For conservatism no consideration is given the heat conduction capability of the boat. Within this evaluation the maximum temperature of the payload is also determined.

Results of this evaluation show the cask component temperatures under solar insolation are bounded by the temperature of 73 °C (163 °F) for the structural evaluation. The results also show the cask meets the exterior surface temperature requirement for normal transport conditions of 85 °C (185 °F) in the shade for exclusive-use shipments. The results of the evaluation show the maximum temperature of the cesium chloride capsules under normal transport conditions do not exceed the 800 °C (1472 °F) requirements of the capsule. Also as configured in the cask, the high heat load of the cask does not melt the lead shielding under normal transport conditions. This is demonstrated by the comparison of the maximum internal cask component temperature of 73°C (163 °F) with the 327 °C (621 °F) melting point of lead.

8.4 REFERENCES

ASME, 1995a, *Boiler and Pressure Vessel Code*, Section IX, American Society of Mechanical Engineers, New York, New York.

ASME, 1995b, *Boiler and Pressure Vessel Code*, Section V, American Society of Mechanical Engineers, New York, New York.

ASTM, 1976, *Annual Book of ASTM Standard*, American Society of Testing, Philadelphia, Pennsylvania.

Fadeff, J. G., 1992, *Environmental Conditions for On-site Hazardous Materials Packages*, WHC-SD-TP-RPT-004, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

8.5 APPENDICES

8.5.1 Thermal Evaluation

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1.0 OBJECTIVE

The objective of this evaluation is to determine the cask component temperature under solar insolation for the structural evaluation and the exterior temperature in the shade as specified in Section 8 of this Safety Evaluation for Packaging (SEP). A secondary objective of this evaluation is estimate the maximum surface temperature of the payload.

2.0 REFERENCES

Irwin, J. J., 1995, WHC-SD-TP-RPT-005, Rev. 1, *Thermal Analysis Methods for Safety Analysis Reports for Packaging*, Westinghouse Hanford Company, Richland, Wash.

ORNL, 1970, *Cask Designer's Guide*, ORNL-NISC-68, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Jakob, M., Hawkins, G. A., 1957, *Elements of Heat Transfer*, John Wiley & Sons, Inc., New York, New York.

MathCad Plus 5, 1994, *User's Guide*, Math Soft Inc., Cambridge, Massachusetts.

3.0 ASSUMPTIONS, RESULTS, AND CONCLUSIONS

The basic assumption for this evaluation is that the cask is loaded with two CsCl capsules for a worst case total heat load of 377 watts. The ambient outside temperature is 115 °F, which is the maximum Hanford Site temperature. For worst case heat loading, it is assumed the capsules are mounted in the handling boat and located approximately in the geometric radial center of the cask and spaced apart in the cask, with no dunnage.

Results of this evaluation show the exterior temperature of the cask in the shade is 133 °F. Worst case internal temperature of the cask with full solar insolation is 163 °F. The estimated maximum temperature of the payload is 879 °F, based on the worst case internal temperature.

Based on the results of this evaluation, it is demonstrated that the cask will meet the NTC exterior temperature requirements of 185 °F in the shade for exclusive use shipments. Also the worst internal cask component temperatures are well below the melting point of lead

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which is approximately 621 °F. The conservatively estimated maximum temperature of the payload is below the capsule test temperature of 1472 °F, as specified for the capsule special form tests. Consequently, it is demonstrated that the lead within the cask will not melt and that the capsule will maintain its special form parameters.

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4.0 EVALUATION

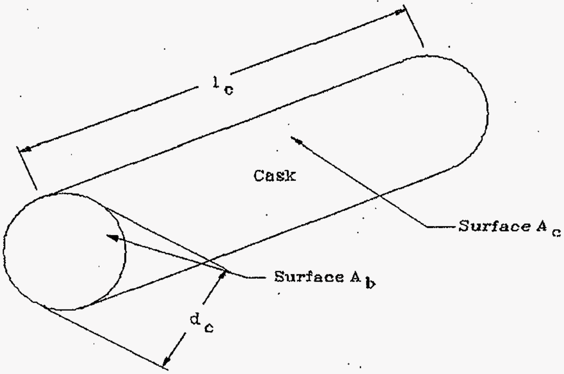
Normal Transport Conditions (NTC) Thermal Evaluation:

Determine temperature of outer shell with and without solar insolation:

Evaluate as steady state heat transfer for a horizontal cylinder with flat plate ends (Irwin, 1995):

Free convection coefficient for a horizontal cylinder: $k_{hc} := 0.27 \cdot \frac{BTU}{hr \cdot ft^2}$

Free convection coefficient for a vertical plate: $k_{vp} := 0.29 \cdot \frac{BTU}{hr \cdot ft^2}$



Length of cylinder: $l_c := 133.56 \cdot in$ Diameter of plate: $d_c := 26 \cdot in$

Surface area of cylinder: $A_c := \pi \cdot l_c \cdot d_c$ Surface area of plate: $A_b := \frac{\pi}{4} \cdot d_c^2$

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Convection coefficients: $hd_1 := \dfrac{k_{hc} \cdot A_c \cdot R^{\frac{1}{4}}}{\left(\dfrac{l_c}{ft}\right)^{\frac{1}{4}}}$ $\quad hd_2 := \dfrac{2 \cdot k_{vp} \cdot A_b \cdot R^{\frac{1}{4}}}{\left(\dfrac{d_c}{ft}\right)^{\frac{1}{4}}}$

Radient heating constant:

Stefan-Boltzman's natural constant: $\sigma_{sb} := 0.171410^{-8} \cdot \dfrac{BTU}{hr \cdot ft^2 \cdot R^4}$

Emissivity of stainless steel (Irwin 1995, page 29): $\varepsilon_s := 0.6$

Radiation coefficients $K_1 := \sigma_{sb} \cdot \varepsilon_s \cdot A_c$ $\quad K_2 := \sigma_{sb} \cdot \varepsilon_s \cdot 2 \cdot A_b$

Solar heat loading (Irwin, 1995), hourly average loading based on a 12 hr period:

Curved surfaces: $Q_{s1} := 314 \cdot \dfrac{watt}{m^2}$ $\quad Q_{s1} = 99.54 \cdot \dfrac{BTU}{hr \cdot ft^2}$

Non-vertical surfaces, flat surfaces: $Q_s := \dfrac{Q_{s1}}{2}$ $\quad Q_s = 49.77 \cdot \dfrac{BTU}{hr \cdot ft^2}$

Internal heat load: $q_{int} := 377 \cdot watt$ $\quad q_{int} = 1286 \cdot \dfrac{BTU}{hr}$

Assumed solar absorptivity (Irwin, 1995, page A-25): $\alpha_{sol} := 0.52$

Solar heat load: $q_{sol} := \alpha_{sol} \cdot \left(Q_s \cdot A_b + Q_{s1} \cdot \dfrac{A_c}{2}\right)$ $\quad q_{sol} = 2056 \cdot \dfrac{BTU}{hr}$

Total heat load: $q_{tot} := q_{sol} + q_{int}$ $\quad q_{tot} = 3342 \cdot \dfrac{BTU}{hr}$

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Outside ambient temperature is 115 °F and in Rankine: $T_o := (115+459.7)\cdot R$

Using conservation of energy: $q_{in} - q_{out} = 0$

Then by substitution: $q_{tot} - q_{rad} - q_{con} = 0$ or

$$q_{tot} - K_1\cdot\left(T_f^4 - T_o^4\right) - K_2\cdot\left(T_f^4 - T_o^4\right) - hd_1\cdot\left(T_f - T_o\right)^{\frac{1}{4}} - hd_2\cdot\left(T_f\ \ T_o\right)^{\frac{1}{4}} = 0$$

Solve for T_{f1} which is the temperature at the surface, using MathCad roots of equation solution:

$$T_{f1} := root\left[q_{tot} - K_1\cdot\left(T_f^4 - T_o^4\right) - K_2\cdot\left(T_f^4 - T_o^4\right)\ \ hd_1\cdot\left(T_f\ T_o\right)^{\frac{1}{4}} - hd_2\cdot\left(T_f - T_o\right)^{\frac{1}{4}}, T_f\right]$$

External surface temperature in sun: $T_{f1} = 619\cdot R$

Temperature in °F: $T_{ff} := \left|\frac{T_{f1} - 459.7R}{R}\right|$ $T_{ff} = 159$ ☜

Temperature in Shade:

Total shaded heat load: $q_{stot} := q_{int}$

Solve for T_{f2} which is the temperature at the surface, using MathCad roots of equation solution:

$$T_{f2} := root\left[q_{stot} - K_1\cdot\left(T_f^4 - T_o^4\right) - K_2\cdot\left(T_f^4 - T_o^4\right)\ \ hd_1\cdot\left(T_f\ T_o\right)^{\frac{1}{4}} - hd_2\cdot\left(T_f - T_o\right)^{\frac{1}{4}}, T_f\right]$$

External surface temperature in shade: $T_{f2} = 592\cdot R$

Temperature in °F: $T_{ff2} := \left|\frac{T_{f2} - 459.7R}{R}\right|$ $T_{ff2} = 133$ ☜

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With this simplified model determine temperature of inner shell with solar insolation:

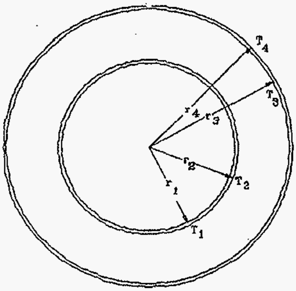
Assume as one-dimensional heat transfer and internal heat source is against inside shell wall, with no gap.

$$q_{tot} = \frac{2 \cdot \pi \cdot l_c \cdot \left(T_1 - T_4\right)}{\frac{\ln\left(\frac{r_2}{r_1}\right)}{k_{sst}} + \frac{\ln\left(\frac{r_3}{r_2}\right)}{k_{pb}} + \frac{\ln\left(\frac{r_4}{r_3}\right)}{k_{sst}}}$$

Thermal conduction properties of materials:

Conductivity (Irwin, 1995):

Lead at 212°F: $k_{pb} := 0.000447 \frac{BTU}{sec \cdot in}$ 304L stainless steel at 200°F: $k_{sst} := 0.000215 \frac{BTU}{sec \cdot in}$



Inner shell inside radius: $r_1 := \frac{7.63 \cdot in}{2}$

Inner shell outside radius: $r_2 := \frac{8.6 \cdot in}{2}$

Outer shell inside radius: $r_3 := \frac{26}{2} \cdot in - 0.675 \cdot in$

Outer shell outside radius: $r_4 := \frac{26 \cdot in}{2}$

Outer shell temperature, for full solar insolation:

$T_4 := T_{ff}$

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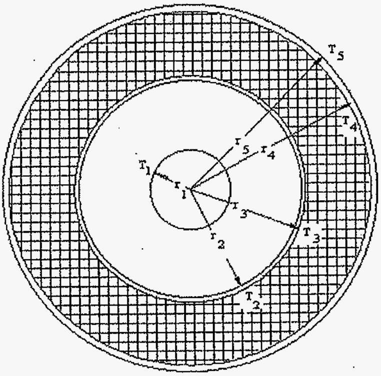
Solve heat transfer equation for inner temperature:

$$T_1 := \frac{q_{tot}\cdot\left[\frac{\ln\left(\frac{r_2}{r_1}\right)}{k_{sst}} + \frac{\ln\left(\frac{r_3}{r_2}\right)}{k_{pb}} + \frac{\ln\left(\frac{r_4}{r_3}\right)}{k_{sst}}\right]}{2\cdot\pi\cdot l_c} + T_4$$

Inside temperature of inner shell (°F): $T_1 = 163$ ☜

Determine maximum temperature of payload:



Assume capsule is in center of package, since in boat.

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Cesium capsule dimensions:

Diameter of capsule: $D_{cap} := 2.625\,in$ Radius: $r_1 := \frac{D_{cap}}{2}$

External length of capsule: $l_{cap} := 20.775\,in$

Inner shell inside radius of cask: $r_2 := \frac{7.63 \cdot in}{2}$ Inner shell outside radius of cask: $r_3 := \frac{8.6 \cdot in}{2}$

Outer shell inside radius of cask: $r_4 := \frac{26}{2} \cdot in - 0.675\,in$ Outer shell outside radius of cask: $r_4 := \frac{26 \cdot in}{2}$

Stefan-Boltzmann's constant in British Technical Units (Jakob 1957, page 28): $\sigma_{bu} := 0.1714 \cdot 10^{-8} \cdot \frac{BTU}{hr \cdot ft^2}$

Assume average air temperature in cask is 500 °F, (Irwin 1995, page B-17):

Density of air: $\rho_{air} := 0.0412 \cdot \frac{lb}{ft^3}$ Thermal conductivity of air: $k_{air} := 0.0231 \cdot \frac{BTU}{hr \cdot ft}$

Dynamic viscosity of air: $\mu_{air} := 1.89 \cdot 10^{-5} \cdot \frac{lb}{ft \cdot sec}$ Prantel Number: $Pr := 0.683$

Isobaric Compressibility: $\beta_{air} := 1.04 \cdot 10^{-3}$

Gap distance between outside of capsule and inside of cask: $\delta := r_2 - r_1$ $\delta = 2.5 \cdot in$

Radiative area of capsule: $A_r := 2 \cdot \left(\pi \cdot D_{cap} \cdot l_{cap} + 2 \cdot \pi \cdot \frac{D_{cap}^{\ 2}}{4} \right)$ $A_r = 364.3 \cdot in^2$

Assuming capsules are widely seperated and a capsule only radiates to half the cask, inside radiated area of cask:

$$A_c := (2 \cdot \pi \cdot r_2 \cdot l_c) \qquad A_c = 3201.5 \cdot in^2$$

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Gray body view factor (Irwin 1995, page 28): $\Phi_{12} := \dfrac{1}{\left(\dfrac{1}{\varepsilon_s}-1\right)+\dfrac{A_r}{A_c}\cdot\left(\dfrac{1}{\varepsilon_s}-1\right)+1}$ $\Phi_{12} = 0.57$

Radiative resistance equation across gap: $R_{rad} = \dfrac{1}{\Phi_{12}\cdot\varepsilon_s\cdot\sigma_{bu}\cdot\left(T_{1g}+T_{2g}\right)\cdot\left(T_{1g}^{2}+T_{2g}^{2}\right)\cdot A_r}$

Convective resistance across gap, assuming still air:

Gap convection constant based on dimensionless numbers defined in (Irwin 1995, page 27):

$$\mathrm{const}_1 := \frac{\ln\left(\dfrac{r_2}{r_1}\right)\cdot\dfrac{\rho_{air}^{2}}{\mu_{air}^{2}}\cdot g\cdot\delta^3\cdot\beta_{air}\cdot Pr}{\delta^3\cdot\left[\left(2\cdot r_1\right)^{-\frac{3}{5}}+\left(2\cdot r_2\right)^{-\frac{3}{5}}\right]^5} \qquad \mathrm{const}_1 = 146$$

Determine effective conductivity across gap, assuming Rayleigh number of cylinder less than 10^7:

Temperature at inside of cylinder in °F: $T_{2g} := T_1$

Equation for Rayleigh number of cylinder: $Ra_{cyl} = \mathrm{const}_1\cdot\left(T_{1g}-T_{2g}\right)$

Effective thermal conductivity: $k_{eff} = 0.386\left(\dfrac{Pr}{0.861+Pr}\right)^{\frac{1}{4}}\cdot\left[\mathrm{const}_1\cdot\left(T_{1g}-T_{2g}\right)\right]^{\frac{1}{4}}\cdot k_{air}$

Convective resistance across gap: $R_{conv} = \dfrac{1}{\left[\dfrac{2\cdot\pi\cdot k_{eff}\cdot l_c}{\ln\left(\dfrac{r_2}{r_1}\right)}\right]}$

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The amount of heat from the capsules are given up to the cask by convection and radiation:

$$q_{int} = \frac{T_{1g} - T_{2g}}{R_{rad} + R_{conv}}$$

Substituting in the variables:

$$q_{int} = \frac{T_{1g} - T_{2g}}{\left[\dfrac{1}{\Phi_{12} \cdot \varepsilon_s \cdot \sigma_{bu} \cdot (T_{1g} + T_{2g}) \cdot \left(T_{1g}^2 + T_{2g}^2\right) \cdot A_r} \ldots + \dfrac{1}{\left[\dfrac{0.386 \left(\dfrac{Pr}{0.861 + Pr}\right)^{\frac{1}{4}} \cdot \left[const_1 \cdot (T_{1g} - T_{2g})\right]^{\frac{1}{4}} \cdot k_{air} \cdot 2 \cdot \pi \cdot l_c}{\ln\left(\dfrac{r_2}{r_1}\right)}\right]}\right]}$$

Solve for T_{1g} which the temperature at the surface using MathCad roots of equation solution:

$$T_{1g} := \mathrm{root}\left[q_{int} - \frac{T_{1g} - T_{2g}}{\dfrac{1}{\Phi_{12} \cdot \varepsilon_s \cdot \sigma_{bu} \cdot (T_{1g} + T_{2g}) \cdot \left(T_{1g}^2 + T_{2g}^2\right) \cdot A_c} \ldots + \dfrac{1}{\left[\dfrac{0.386 \left(\dfrac{Pr}{0.861 + Pr}\right)^{\frac{1}{4}} \cdot \left[const_1 \cdot (T_{1g} - T_{2g})\right]^{\frac{1}{4}} \cdot k_{air} \cdot 2 \cdot \pi \cdot l_c}{\ln\left(\left(\dfrac{r_2}{r_1}\right)\right)}\right]}}, T_{1g}\right]$$

Temperature at surface of capsule in °F: $T_{1g} = 627$

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Temperature at surface of capsule in °C: $T_{1gc} = 330$

Maximum temperature of payload (Irwin 1995, page 30):

$$T_{max} := \frac{q_{int} \cdot r_1^{\,2}}{4 \cdot \left[0.386 \left(\frac{Pr}{0.861 - Pr} \right)^{\frac{1}{4}} \cdot \left[const_1 \cdot \left(T_{1g} - T_{2g} \right) \right]^{\frac{1}{4}} \cdot k_{air} \right] \cdot \left(\pi \cdot r_1^{\,2} \cdot l_{cap} \cdot 2 \right)} + T_{1g}$$

Maximum temperature of payload in°F: $T_{max} = 879$

Maximum temperature of payload in°C: $T_{maxc} = 470$

8.5.2 Thermal Output

```
User Input File: serf.in              Date/Time of execution: 09/04/97 11:00:17.81

Summary of Nuclide Information For the Source Term

                                Specific           Heat Prod  Heat Gener                        Sum of     f(I)
          Activity  Activity  Activity  Quantity   Factor      Rate                   A2      Fraction  ---------
 Nuclide    Bq         Ci       Ci/g       g        W/Ci         W        %Cont       Ci        A2        A2(I)
--------- --------- --------- --------- --------- ---------- ---------- --------- --------- --------- ---------
 SR90     3.70E+13  1.00E+03  1.40E+02  7.14E+00   1.16E-03   1.16E+00  4.58E-01  2.70E+00  3.70E+02  2.24E-02

+Y 90     3.70E+13  1.00E+03  5.40E+05  1.85E-03   5.54E-03   5.54E+00  2.19E+00  0.00E+00  0.00E+00  0.00E+00

 CO60     5.55E+14  1.50E+04  1.10E+03  1.36E+01   1.54E-02   2.31E+02  9.13E+01  1.08E+01  1.39E+03  8.42E-02

***PU239  1.85E+13  5.00E+02  6.20E-02  8.06E+03   3.06E-02   1.53E+01  6.05E+00  5.41E-03  9.24E+04  5.60E+00

          --------- ---------                               ----------                      --------- ---------
   Total = 6.48E+14  1.75E+04              Tot Heat Rate  =  2.53E+02 W        Sum of A2 = 9.42E+04
   Total = 6.11E+14  1.65E+04 (excluding daughters & nuclides with A2=0)       Sum of f(I)/A2(I)  =  5.71E+00

   A2 for Mixture of Normal Form [1/sum(f(I)/A2(I))]  =  1.75E-01
   The mixture contains  9.42E+04 A2s/g using a source term weight of 1.00E+00 g
   Normal Form : Highway Route Controlled Quantity since A2 =  9.42E+04 which exceeds 3000 * A2 for normal form

 + This radionuclide is a daughter as defined in 49 CFR 173.433, therefore, its activity was set to 0 for
   all A1/A2 calculations

 * Fissile radionuclide as defined in 49 CFR 173.403a
   Total Fissile  =  8.06E+03 - This exceeds the 15 g criteria for fissile excepted in 49 CFR 173.453a.
                                Note that other criteria in 49 CFR 173.453 may qualify this source term as
                                fissile excepted.

 ** Total TRU Activity =  5.00E+11 nCi
    TRU Activity Concentration  =  5.00E+11 nCi/g > 100 nCi/g using a source term weight of 1.00E+00 g
    Requires 5.00E+09 grams of waste matrix to be < 100 nCi/g
    [Note: TRU defined in EP-0063 as waste contaminated alpha emitters with Z > 92 and T-1/2 > 20 years and
           concentrations > 100 nCi/g of waste matrix at the time of assay.  In addition to TRU radionuclides,
           radium sources and U-233 in concentrations > 100 nCi/g are managed as TRU waste.]

 *** Indicates a fissile and TRU radionuclide
```

```
User Input File: cscl.in              Date/Time of execution: 09/04/97 10:39:30.72

Summary of Nuclide Information For the Source Term

                                   Specific            Heat Prod  Heat Gener                       Sum of     f(I)
             Activity   Activity   Activity   Quantity   Factor     Rate                 A2       Fraction  --------
  Nuclide       Bq         Ci        Ci/g        g        W/Ci        W       %Cont       Ci         A2       A2(I)
 ---------- ---------  ---------  ---------  ---------  ---------  ---------- --------- ---------  --------- ---------
  CS137     2.96E+15   8.00E+04   8.70E+01   9.20E+02   1.01E-03   8.08E+01   2.14E+01  1.35E+01   5.93E+03  7.41E-02

 +BA137M    2.80E+15   7.57E+04   0.00E+00   0.00E+00   3.92E-03   2.97E+02   7.86E+01  0.00E+00   0.00E+00  0.00E+00

            ---------  ---------                                   ---------                       --------  --------
   Total = 5.76E+15   1.56E+05              Tot Heat Rate  =  3.77E+02 W        Sum of A2 = 5.93E+03
   Total = 2.96E+15   8.00E+04  (excluding daughters & nuclides with A2=0)        Sum of f(I)/A2(I)  =   7.41E-02

   A2 for Mixture of Normal Form [1/sum(f(I)/A2(I))]  =  1.35E+01
   The mixture contains  5.93E+03 A2s/g using a source term weight of 1.00E+00 g
   Highway Route Controlled Quantity since Activity =  1.56E+05 Ci which exceeds 27000 Ci for
   normal or special form
   Normal Form : Highway Route Controlled Quantity since A2 =  5.93E+03 which exceeds 3000 * A2 for normal form

 + This radionuclide is a daughter as defined in 49 CFR 173.433, therefore, its activity was set to 0 for
   all A1/A2 calculations
```

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9.0 PRESSURE AND GAS GENERATION EVALUATION

Only dry contents are authorized for shipment in the SERF Cask; therefore, there are no pressure and gas generation concerns. If fuel pins are not intact, any gas present will be released during handling. No additional gas will be generated.

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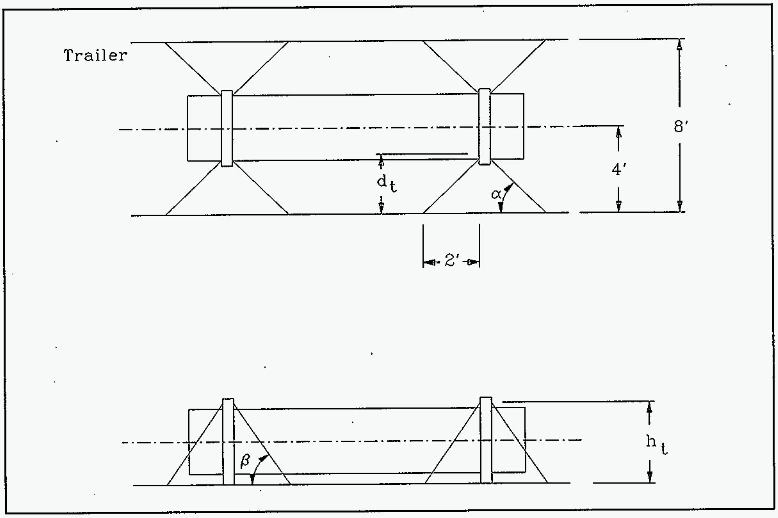
10.0 PACKAGE TIEDOWN SYSTEM EVALUATION

10.1 SYSTEM DESIGN

The SERF Cask is assumed to be centered and placed horizontally on the bed of the trailer for shipment. The long axis of the cask is centered along the long axis of the trailer. The package is to be secured in accordance with DOT regulations (49 CFR 393.100). The cask is to be secured to the trailer by eight chains or cable which are attached to the cask lift points at one end and affixed to the trailer at the other. For adequate securement of this package, the tiedowns must be configured as shown in Figure B10-1 of this section and the tiedowns have an aggregate working capacity of greater than ½ the weight of the cask.

Figure B10-1. SERF Cask Tiedown Configuration.



10.2 ATTACHMENTS AND RATINGS

Lifting holes on the cask support plates welded to the cask are to be used for securement points. Each tiedown and trailer attachment must have a minimum working capacity of 3,175 kg (7,000 lb).

10.3 REFERENCE

49 CFR 393.100, 1997, "Protection Against Shifting or Falling Cargo," *Code of Federal Regulations*, as amended.

10.4 APPENDIX: TIEDOWN EVALUATION

ENGINEERING SAFETY EVALUATION

Subject: 327 SERF Cask Tiedown Evaluation Page 1 of 3
Preparer: S. S. Shiraga Date 07/22/97
Checker: R. J. Smith Date 08/04/97
Section Chief: S. S. Shiraga Date 08/04/97

1.0 OBJECTIVE

The objective of this evaluation is to determine the capacity and configuration of the tiedown system for the 327 SERF Casks. The tiedowns are specified to the requirements of 49 CFR 393.100.

2.0 REFERENCES

49 CFR 393.100, 1995, "Protection Against Falling Cargo," Subpart I, *Code of Federal Regulations*, as amended.

3.0 ASSUMPTIONS, RESULTS, AND CONCLUSIONS

As defined in this document, shipment of the 327 SERF Cask within the 300 Area is authorized under a risk based assessment. Consequently, the tiedown system must be an engineered system to ensure that the package remains on the conveyance during all normal non-accident conditions. The system is specified based on the requirements of 49 CFR 393.100. No chocking is assumed.

As shown in the evaluation, the cask is assumed to be centered on a standard flat bed trailer. The cask is tied down at the cask lift brackets. The system consists of eight tiedowns, with a set of two tiedowns at each of the four cask lift points. Each tiedown is to be independently attached to the lift point and drawn tight. The minimum specified working strength of the tiedowns and attachments is 7,000 lb. Each tiedown shall have a minimum horizontal distance from the cask attachment to the attachment to the trailer of 24 in.

ENGINEERING SAFETY EVALUATION

Subject: 327 SERF Cask Tiedown Evaluation — Page 2 of 3
Preparer: S. S. Shiraga — Date 07/22/97
Checker: R. J. Smith — Date 08/04/97
Section Chief: S. S. Shiraga — Date 08/04/97

4.0 EVALUATION

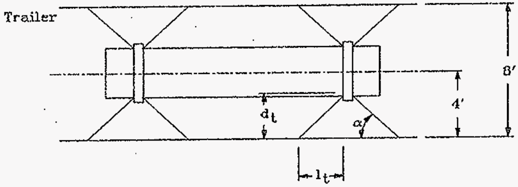
SERF Cask Tiedown Evaluation:

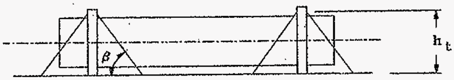
OD of cask: $od_{cask} := 26 \cdot in$ Length of cask: $l_{cask} := 133.56 in$

Gross weight of cask: $w_{cask} := 26500 lbf$ DOT inertial load factor: $g_{dot} := 0.5$

Tiedown height from deck: $h_t := 26 \cdot in$ Length to trailer attachment: $l_t := 24 \cdot in$

Load on tiedowns: $F_{ch} := g_{dot} \cdot w_{cask}$ $F_{ch} = 13250 \cdot lbf$





ENGINEERING SAFETY EVALUATION

Subject: 327 SERF Cask Tiedown Evaluation Page 3 of 3
Preparer: S. S. Shiraga Date 07/22/97
Checker: R. J. Smith RJS Date 08/04/97
Section Chief: S. S. Shiraga Date 08/04/97

Tiedown from edge of trailer: $d_t := 4\cdot ft - 12\cdot in$

Angle off horizontal: $\alpha := \operatorname{atan}\left(\frac{d_t}{l_t}\right)$ $\alpha = 56.3\cdot deg$

Angle off vertical: $\beta := \operatorname{atan}\left(\frac{h_t\cdot\cos(\alpha)}{l_t}\right)$ $\beta = 31\cdot deg$

Determine tiedowns:

Tension in lateral direction, four tiedowns acting: $T_{lat} := \frac{F_{ch}}{4\cdot(\sin(\alpha)\cdot\cos(\beta))}$ $T_{lat} = 4645\cdot lbf$

Tension in longitudinal direction, four tiedowns acting: $T_{long} := \frac{F_{ch}}{4\cdot(\cos(\alpha)\cdot\cos(\beta))}$ $T_{long} = 6967\cdot lbf$

Tension in vertical tiedowns, 8 tiedowns acting: $T_{ver} := \frac{F_{ch}}{8\cdot\sin(\beta)}$ $T_{ver} = 3216\cdot lbf$

Based on above, minimum working load of all tiedowns is specified as 7,000 lb.

DISTRIBUTION SHEET

To Distribution	From Packaging Engineering	Page 1 of 1
		Date Sept. 29, 1997
Project Title/Work Order Safety Evaluation for Packaging (Onsite) SERF Cask (HNF-SD-TP-SEP-058, Rev. 0)		EDT No. 621095
		ECN No. N/A

Name	MSIN	Text With All Attach.	Text Only	Attach./ Appendix Only	EDT/ECN Only
D. W. Claussen	S7-55	X			
R. L. Clawson	H1-14	X			
J. G. Field	H1-15	X			
C. R. Hoover	H1-15	X			
S. B. Johnston	L1-03	X			
HNF-SD-TP-SEP-058 File	H1-15	X			
P97-046	H1-15				X
Central Files	B1-07	X			