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**SAFETY ANALYSIS REPORT FOR PACKAGING:
NEUTRON SHIPPING CASK, MODEL 4T**

R. T. Peterson

March 4, 1977

Prepared for U.S. Energy Research & Development
Administration under contract No. W-7405-Eng-48

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Contents

Abstract	1
1.0 Introduction	2
2.0 Package Description	3
2.1 General	3
2.2 Materials	5
2.2.1 Structural	5
2.2.2 Shielding	5
2.3 Fabrication Methods	10
2.3.1 Joining	10
2.3.2 Casting	10
2.4 Appurtenances	11
2.4.1 Coolant Channel	11
2.4.2 Pressure Relief	11
2.4.3 Lifting Devices	11
2.5 Closures	12
2.5.1 Container	12
2.5.2 Gamma Shield	12
2.5.3 Capsule	12
3.0 Contents of Packaging	14
3.1 Designated Elements	14
3.2 Radioactive Decay Power	14
3.2.1 Alpha, Beta, and Electron Capture	14
3.2.2 Gamma Emission	16
3.2.3 Spontaneous Fission	17
3.3 Neutron Emission Rate	17
3.4 Gas Evolution	18
3.4.1 Fission Gases	18
3.4.2 Helium Gas Generation	18
3.5 Criticality	19
4.0 Evaluation	21
4.1 Thermal Analysis	21
4.1.1 Normal Conditions of Transport	21
4.1.2 Hypothetical Accident Conditions	26
4.1.3 Special-Form Capsule Heating Test	28

4.2	Structural Analysis	29
4.2.1	Load Resistance	29
4.2.2	Compression	30
4.2.3	External Pressure	31
4.2.4	Internal Pressure	33
4.2.5	Impact	37
4.2.6	Lifting Devices	47
4.2.7	Structural Tiedown Devices	59
4.2.8	Truck Transport Tiedown	65
4.3	Shielding Analysis	68
4.3.1	Normal Transport Conditions	70
4.3.2	Hypothetical Accident Conditions	72
5.0	Quality Assurance Program	74
5.1	Scope	74
5.2	Responsibilities	74
5.3	Fabrication and Acceptance Inspection	74
5.4	Reuse Inspection	75
5.5	Inspection Records	75
6.0	Loading and Handling Procedure	76
7.0	Compliance With Chapter 0529	77
7.1	Part I	77
7.2	Part II	77
7.3	Part III	80
7.4	Part IV	80
	Acknowledgment	81
	References	82
Appendix A.	Chapter 0529 from Health and Safety Section of ERDA Manual	A-1
Appendix B.	Limits on Radioactive Materials	B-1
Appendix C.	List of Cask Drawings	C-1
Appendix D.	Tiedown Equipment Design Limits	D-1
Appendix E.	Sample Form for Reuse Inspection	E-1
Appendix F.	Sample Form for Loading and Handling Procedure	F-1

SAFETY ANALYSIS REPORT FOR PACKAGING: NEUTRON SHIPPING CASK, MODEL 4T

Abstract

This Safety Analysis Report for Packaging demonstrates that the neutron shipping cask can safely transport, in solid or powder form, all isotopes of uranium, plutonium, americium, curium, berkelium, californium, einsteinium, and fermium. The report describes the cask and its contents. It also evaluates transport conditions, structural parameters (e.g., load resistance, pressure and impact effects, lifting and tiedown devices), and shielding. Finally, it discusses compliance with Chapter 0529 of the Energy Research & Development Administration Manual, Safety Standards for the Packaging of Fissile and Other Radioactive Materials.

1.0 Introduction

In 1958-1959, Lawrence Livermore Laboratory (LLL) designed and built three neutron shipping casks. They were used to ship neutron-emitting radioisotopes for about 10 years and then were modified to meet additional safety requirements.

In 1974, the Atomic Energy Commission requested LLL to prepare a Safety Analysis Report for Packaging (SARP) for the casks.

In response to that request, this report fulfills the document requirements of Chapter 0529 of the Energy Research & Development Administration Manual, Safety Standards for the Packaging of Fissile and Other Radioactive Materials.^{*}

What follows demonstrates that the neutron shipping cask can safely transport, in solid or powder form, all isotopes of uranium (U), plutonium (Pu), americium (Am), curium (Cm), berkelium (Bk), californium (Cf), einsteinium (Es), and fermium (Fm).

Radioactive materials to be shipped belong in the special-form transport group, type B quantity with a 5000 Ci limit. The transport index is 10.0.

Certain limits have been placed on shipments with this cask to meet the standards of Chapter 0529. These limits appear below along with section numbers for analyses in this report:

- Maximum power from radioactive decay not to exceed 250 W (Section 4.1.1).
- Total mass of any combination of the following fissile isotopes not to exceed 15.0 g (Section 3.5):
 ^{243}Cm , ^{245}Cm , ^{247}Cm , ^{249}Cf , ^{251}Cf , ^{242}Am , ^{233}U , ^{235}U , ^{238}Pu , ^{239}Pu ,
and ^{241}Pu .
- Total mass of ^{251}Cf not to exceed 3.0 g (Section 3.5).
- Radiation dose rate at any point on the external surface of the shipping cask not to exceed 200 mrem/h (Section 4.3.1).
- Radiation dose rate at 91.4 cm from any accessible external surface of the shipping cask not to exceed 10 mrem/h (Section 4.3.1).
- Gas generation in special-form capsule not to exceed value given in Section 4.2.4.2.

Appendix B gives an example of the application of the limits.

^{*} See Appendix A for the complete text of Chapter 0529.

2.0 Package Description

For the purpose of this report, the entire assembly is designated as the "Neutron Shipping Cask." The large cylinder that embodies the neutron shielding material is the "Container." The inner vessel with the depleted-uranium shielding is the "Gamma Shield." The primary containment vessel is the "Special-Form Capsule."

2.1 GENERAL

The container is a right circular cylinder welded to a square steel pallet (Fig. 1). The opening to the cavity is closed with a 64.1-cm-(25.25-in.) diam by 30.5-cm-(12.0-in.) thick cylindrical cover. The body and cover have steel shells. A flat rubber gasket and twelve studs and nuts are used to fasten and seal the cover to the body. A composite of gypsum-spodumene is used in the container body and cover as the shielding material.

A gamma shield that is available for those radioisotopes requiring additional shielding is primarily depleted uranium, machined to fit inside a stainless steel vessel. The nominal thickness of uranium in all directions is 12 cm. The shielded cavity dimensions are 6.4-cm (2.50-in.) in diam by 13.0-cm (5.12-in.) deep. An alternate cap with a recessed hole provides a cavity depth of 21.4 cm (8.43 in.). The cap is attached to the body with eight nuts, and an o-ring can be used to seal the cavity.

The total cask mass is 3175 kg (7000 lb) with the gamma shield in use. The gamma shield and the container cover have masses of 862 kg (1900 lb) and 289 kg (638 lb), respectively. The cask is identified by a stainless steel plate welded to the outer steel shell and is stamped in the following manner:

RADIOACTIVE MATERIAL
USA/6478/BLF (ERDA-SAN)
NEUTRON SHIPPING CASK MODEL 4T
WEIGHT 7000 LBS. W/GAMMA SHIELD
WEIGHT 5100 LBS. W/O GAMMA SHIELD
LLL DRAWING NUMBER AAA-60-144963
RETURN TO:
LAWRENCE LIVERMORE LABORATORY
BLDG. 251 LIVERMORE, CALIFORNIA

Fig. 1. Neutron shipping cask.

The containment vessel for radioisotopes shipped in the cask is a special-form capsule. It is designed to be leak-tight under all normal-transport and hypothetical-accident conditions (listed in Appendix A). The capsule presently used (Fig. 2) is manufactured by Aeroquip Corporation and employs a "Conoseal" gasket. It is available in four different sizes, any of which may be used. Capsules other than those described above may be used if they meet the "special-form" requirements of Chapter 0529 (Part I, A 15 b).

When more than one capsule is to be shipped and the gamma shield is not required, a rack is used. The rack (Fig. 3) holds the capsules in a fixed array and hinders as little as possible the heat transfer processes.

2.2 MATERIALS

2.2.1 Structural

Figure 1 shows the arrangement of materials in the cask. The special-form capsule that occupies the centerline position is shown in Figure 2. Table 1 lists the structural materials and specifications.

2.2.2 Shielding

2.2.2.1 Gypsum-Spodumene

The gypsum-spodumene composite combines two parts by mass gypsum ($\text{CaSO}_4 \cdot 1/2\text{H}_2\text{O}$), two parts spodumene ($\text{LiAlSi}_2\text{O}_6$), and one part water. The ingredients are mixed for 5 to 10 min in a plaster mixer, then poured. The resulting chemical reaction ($\text{CaSO}_4 \cdot 1/2\text{H}_2\text{O} + 3/2\text{H}_2\text{O} \rightarrow \text{CaSO}_4 \cdot 2\text{H}_2\text{O}$) is exothermic; the setting time is from 10 to 30 min, depending on the size of the pour. Excess water is trapped by the setting mixture into a homogeneous composite. The resulting solid strongly resembles gypsum wall plaster. Spodumene, which acts as an aggregate in the composite, is a water insoluble mineral of 200-mesh particle size, much like sand in texture and appearance.

The Physical Test Laboratory at Lawrence Livermore Laboratory (LLL) conducted tests of the gypsum-spodumene composite. The results, as reported in Engineering Test No. 4390, are:

- Density. 1920.0 kg/m^3 (120 lb/ft^3).
- Ultimate compressive strength. 3.2 MPa (458 psi).

This was a test of four 2.5-cm-diam by 8.9-cm-long specimens conducted on a 89.0-kN (10-ton) compression-test machine with a 4.4-kN (1000-lb) load-test cell.

Part Number	A diam	B diam	C ref	D
MJN68462-1	1.345 (3.42)	1.600 (4.06)	2.98 (7.57)	2.3 (6.1)
-2	1.345 (3.42)	1.600 (4.06)	5.98 (15.19)	2.3 (6.1)
-3	1.750 (4.45)	2.000 (5.08)	7.73 (19.63)	2.3 (6.1)
MJN68462-4	0.600 (1.52)	1.000 (2.54)	8.00 (20.32)	1.4 (3.6)

All units: in. (cm)

Fig. 2. Special-form capsule with "Conoseal" gasket.

C ref	D diam	E max	Internal volume, cm ³
2.98 (7.57)	2.370 (6.02)	5.00 (12.70)	69.4
5.98 (15.19)	2.370 (6.02)	8.00 (20.32)	139.3
7.73 (19.63)	2.960 (7.52)	10.25 (26.04)	304.7
8.00 (20.32)	1.438 (3.65)	9.75 (24.77)	37.1

ts: in. (cm)

Fig. 3. Capsule rack.

Table 1. Structural materials.

Item	Material	Specification
Container shell	Steel	ASTM A 36
Pallet base	Steel	ASTM A 36
Lifting & tiedown lugs	Steel	ASTM A 36
Stud mounting sleeve annealed	Type 304 SS	QQ-S-763 b
Container cover	Steel	ASTM A 36
Gamma shield shell	Type 304 SS	
Gamma shield cap	Type 304 SS	
Gamma shield bolting	Type 304 SS	
Special form capsule body	Type 347 SS	ASTM A 276
Special form capsule cover	Type 347 SS	ASTM A 276
Special form capsule gasket	Annealed, pickled, and cold-rolled stainless steel	MIL-S-6721A, Composition Ti
Special form capsule bolting	Iron-base, high- strength stainless alloy A-286	MIL-S-8879

- Modulus of Elasticity. 8.3 GPa (1.2×10^6 psi).

This was a confined compression test of two 3.2-cm-diam by 1.3-cm-long specimens conducted on a 534.0-kN (60-ton) compression-test machine. Individual values were 8.3 GPa (1.2×10^6 psi) and 9.7 GPa (1.4×10^6 psi).

- Scratch Hardness. 83 S.

Average of five specimens as determined by a Dietert No. 63 scratch hardness tester.

The ability of the gypsum-spodumene composite to withstand freezing was demonstrated by twice chilling a specimen to 77 K (-320°F) with liquid nitrogen and warming it to room temperature. No cracking, crumbling, or other damage was evident.

Several samples of the composite were placed in 1089 K (1500°F) furnace for a 30 min period. The average mass loss was 20% (assumed to be caused by

complete loss of water). The samples remained intact without any cracking or disintegration. At 401 K (262°F), the gypsum will dehydrate, reverting to its original semihydrated form ($\text{CaSO}_4 \cdot 1/2\text{H}_2\text{O}$). At 436 K (325°F), all water is driven off.

No thermal properties tests were conducted on the composite. The values used in the calculations are based on similar materials. Table 2 lists the thermal properties found in the literature.

The values used in the calculations were:

- Thermal Conductivity. 75% of listed values for spodumene.
- Specific Heat. Constant at 921.0 J/kg · K (0.22 Btu/lb·°F)

2.2.2.2 Depleted Uranium

The properties of the depleted uranium used in the gamma shield are found in Refs. 4 and 5:

Density,	18900 kg/m ³	
Melting point, K	1405	
Specific heat	<u>K</u>	<u>J/kg·K</u>
	@200	109.1
	300	117.0
	400	124.4
	500	133.8
	600	144.7
	700	157.5
Thermal conductivity	<u>K</u>	<u>W/m · K</u>
	@200	25.1
	300	27.6
	400	28.5
	450	28.7
	600	31.4
	700	34.1
Transition temperature, K	Alpha to beta @936	
Coefficient of volumetric thermal expansion	$4.86 \times 10^{-5}/\text{K}$	

2.2.2.3 Lead

The gamma shield has a 3.14-cm-thick belt of lead surrounding the cavity which was used to encase a cooling coil (no longer used). Although the amount of lead is only a small portion of the shielding, it is in a sensitive location relative to heat transfer. A comparison of specific-heat and thermal-conductivity data for lead and uranium shows that:

Table 2. Thermal properties of shielding components.

Material	Density (kg/m ³)	Temperature (K)	Thermal conductivity (W/m · K)	Linear thermal expansion (%)	Specific heat (J/kg·K)
Spodumene ¹	2130	278	3.12	0	--
	--	444	3.58	-1.0	--
	--	611	3.97	-6.5	--
	--	778	4.26	-12.0	--
	--	889	--	-14.0 Min	--
	--	944	4.49	-13.0	--
	--	1111	4.54	0	--
Gypsum plaster ²	2114	348	2.44	--	--
Gypsum sand plaster ³	1794	294	2.38	--	921

- The specific heats are within 5% of each other.
- The thermal conductivity of lead is higher than that of uranium, up to the melting point of lead. The heat-transfer calculations, therefore, will be based on the assumption that the entire gamma shield is uranium metal. A maximum allowable temperature of 550 K will be imposed on the gamma shield to prevent melting of the lead. This limit is 50 K below the melting point of lead.

2.3 FABRICATION METHODS

2.3.1 Joining

The cask structure was fabricated by shielded-metal-arc and/or gas-tungsten-arc welding techniques. The electrode for the carbon-steel welds was typically American Welding Society (AWS) Classification E 60 (50,000 psi minimum yield point). The filler metal for the austenetic stainless steel-to-carbon steel welds was typically AWS Classification ER 309 (80,000 psi minimum tensile strength). The electrodes for the welds between the various types of austenetic stainless steel have a 80,000 psi minimum tensile strength.

2.3.2 Casting

The shielding material combines 2 parts-by-mass spodumene (ceramic grade, 200 mesh size), 2 parts-by-mass gypsum (casing type, 30 min set time) and 1

part-by-mass tap water. It is thoroughly mixed before pouring, and the container is vibrated as it receives the pour to eliminate voids.

The shielding cavity of the container has a wire-mesh screen to secure the gypsum-spodumene composite in place.

2.4 APPURTENANCES

2.4.1 Coolant Channel

Two pipes extend radially from the bottom of the container cavity to the outside shell and terminate in 1-in. pipe couplings. The ends of the pipes are welded to the inner and outer walls and are cast in place by the shielding composite. Two shielding plugs with threaded ends can be inserted into the pipes. The shielding is identical to that used in the container body.

2.4.2 Pressure Relief

The cavities containing the gypsum-spodumene shielding composite are provided with relief valves for overpressure protection. The valves have brass bodies and are set to lift at 69.0 kPa (10 psi). The valve for the container cover is mounted in a well at the top surface which affords protection from accidental damage.

Two valves, 180° apart, are used on the container body and are also recessed below the outer surface. Because of the large volume of gypsum-spodumene in the body, a gas-collection channel is built into the container below the relief valves. The passageway for gases is a triangular-shaped area formed by an 1/8-in. × 1/2-in. × 1/2-in. steel angle joined to the container outer wall. Its flow area is equivalent to a 7.6-mm (0.30-in.) i.d. tube.

2.4.3 Lifting Devices

2.4.3.1 Gamma Shield

Two stainless steel bars are welded to the shell of the gamma-shield body for lifting. They extend 8.3 cm above the top of the shell and have 3.2-cm-(1.25-in.) diam holes centered 4.1 cm from the end. The caps have a U-shaped rod with a 2.5-cm i.d. welded to the stainless steel top plate.

2.4.3.2 Container

In addition to the pallet-type base, four lugs are at the container top. They have separate holes for lifting and tiedown during shipping. The lugs

are welded to both the container outer shell and top plate. The holes in the lugs for lifting will accommodate shackles with up to a 3.2-cm-(1.25-in.) diam pin. The holes for tiedown will accommodate shackles with a maximum diameter of 3.8 cm (1.50 in.).

The cover has a 3.8-cm-(1.50-in.) diam hole in a bar in the center of the unit. The 1.9-cm-thick by 7.6-cm-wide bar extends 8.3 cm above the cover.

2.5 CLOSURES

2.5.1 Container

The container cover is fastened to the body by twelve stainless steel, slotted, hexagon-head nuts and twelve specially designed studs. (The thread size is 1-8UNC.) The studs screw into the body and have a nylon insert for thread-locking. The other end of the special stud is drilled for a cotter pin or safety wire to lock the nut in place. A flat rubber gasket between the body and cover prevents dirt from entering the container cavity.

2.5.2 Gamma Shield

The gamma-shield cap is attached to the body with eight castellated, stainless steel nuts. The body has the matching studs welded into its top plate. (The thread size is 1/2-20 UNF.) There are provisions for safety wires.

The alternate cap with the recessed hole in it (see Section 2.1) uses a different method of attachment. In place of the nuts, it has 2.5-cm- (1.0-in.) diam bolts with internal threads on the ends to accept the 1/2-20 UNF studs.

A silicone rubber o-ring is used in both cases to provide a dirt excluding seal.

2.5.3 Capsule

The special-form capsule is the containment vessel for radioactive materials. The capsule and its "Conoseal" gasket are designed for a pressure of 1034.0 kPa (150 psig) at 1089 K (1500°F). The capsule is tested at an internal pressure of 276.0 kPa (40 psi) and 1089 K (1500°F). To be acceptable, it must be leak-free when checked with a helium mass spectrometer having a sensitivity of 1×10^{-9} std cm³/s.

The metal gasket, resembling a Belleville washer, is completely enclosed by the flanges. As the cover bolts are tightened, the conical gasket is flattened and is compressed radially and axially. When the bolts have been torqued to their recommended value, the gasket material, stressed beyond its yield point, is completely confined and compressed.

3.0 Contents of Packaging

3.1 DESIGNATED ELEMENTS

The shipping cask shall transport all the isotopes of uranium, plutonium, americium, curium, berkelium, californium, einsteinium, and fermium. The material shall be in a dry solid or powder form.

3.2 RADIOACTIVE DECAY POWER

Exothermic decay reactions create the power in a radioisotope. It is the product of the decay rate and the energy associated with each decay reaction. Table 3 lists radioisotopes normally encountered when using the cask. They are produced by a (n, γ) reaction, multiple neutron capture, or decay. The following sections describe the equations and assumptions used to determine the table values. Appendix B provides an example of the use of the equations and Table 3.

3.2.1 Alpha, Beta, and Electron Capture

The decay rate $R_{(t)}$ is the product of the decay constant (λ) and the number of atoms present (N):

$$R_{(t)} = \lambda \cdot N \quad (\text{in dis/s}),$$

where

$$\lambda = \frac{0.693}{T_{1/2}} \quad (\text{in s}^{-1}),$$

$$N = \frac{B \cdot G}{A} (e^{-\lambda t}),$$

$$T_{1/2} = \text{half-life (s)},$$

$$\begin{aligned} B &= \text{Avogadro's number} \\ &= 6.023 \times 10^{23} \text{ atoms/g-mol}, \end{aligned}$$

$$A = \text{atomic weight (amu)},$$

$$G = \text{sample mass (g), and}$$

$$t = \text{elapsed time between when G is measured or calculated and sample inserted into cask (s).}$$

Table 3. Significant radioisotopes.

Isotope	Half-life ⁶		Maximum decay power (W/g)	Maximum gas generation rate (cm ³ /g·s)	Specific neutron emission rate ⁷ (neutrons/g·s)
	Total	Spontaneous fission			
Am-241	458y	--	0.11	0.44×10^{-8}	--
Am-242m	152y	--	<0.01	--	--
Am-243	7.95×10^3 y	--	<0.01	--	--
Cm-242	162.5d	7.2×10^6 y	122.4	457×10^{-8}	1.8×10^7
Cm-243	32y	--	1.68	6.33×10^{-8}	---
Cm-244	17.6y	1.31×10^7 y	2.92	11.47×10^{-8}	1.17×10^7
Cm-245	9.3×10^3 y	--	<0.01	--	--
Cm-246	5.5×10^3 y	1.7×10^7 y	<0.01	--	9.5×10^6
Cm-247	1.6×10^7 y	--	<0.01	--	--
Cm-248	4.7×10^5 y	4.6×10^6 y	--	--	3.4×10^6
Cm-250	--	1.7×10^4 y	0.10	--	9.3×10^9
Bk-249	314d	6×10^8 y	1.24	--	3.3×10^5
Cf-249	360y	1.5×10^9 y	0.13	0.47×10^{-8}	1.1×10^5
Cf-250	13.2y	1.7×10^4 y	4.04	14.92×10^{-8}	9.3×10^9
Cf-251	800y	--	0.06	0.25×10^{-8}	--
Cf-252	2.646y	85y	39.02	72.23×10^{-8}	2.32×10^{12}
Cf-253	17.6d	--	46.48	--	--
Es-253	20.47d	6.4×10^5 y	907.9	3123×10^{-8}	2.4×10^8
Es-254	276d	7×10^5 y	68.03	238.4×10^{-8}	2.2×10^8
Es-255	38.3d	>170y	91.94	156.4×10^{-8}	$>9.1 \times 10^{11}$
Fm-254	Not used	246d	2482	75.72×10^{-8}	3.34×10^{14}
Fm-257	80d	100y	259.8	820.6×10^{-8}	1.62×10^{12}

The decay rate becomes

$$R(t) = 4.175 \times 10^{23} \left(\frac{G}{A \cdot T_{1/2}} \right) \left(e^{-0.693t/T_{1/2}} \right).$$

Table 3, assumes that t is small relative to $T_{1/2}$; hence $e^{-0.693t/T_{1/2}} \approx 1.0$. This results in the maximum decay rate. For some unlisted radio-isotopes, it may be advantageous to disregard the above assumption as it can lead to a gross overestimation of decay power.

The total energy released per decay (E) is determined by summing the energy release (Q) and intensity (F) for each decay scheme. For instance, ^{255}Es decays 8.5% of the time by α emission with a Q value of 6.8 MeV. There is also β emission that occurs 91.5% of the time with a Q value of 0.5 MeV. The total energy release per decay in this case is

$$\begin{aligned} E &= F_{\alpha} \cdot Q_{\alpha} + F_{\beta} \cdot Q_{\beta} \\ &= 0.085 \times 6.8 + 0.915 \times 0.5 \\ &= 1.036 \text{ MeV.} \end{aligned}$$

The power generation is

$$P = K \cdot R(t) \cdot E$$

where

$$K = \text{conversion factor} = 1.6022 \times 10^{-13} \frac{\text{W} \cdot \text{s}}{\text{MeV}}, \text{ and } e^{-\lambda t} = 1.$$

Thus,

$$P = 6.69 \times 10^{10} \left(\frac{G \cdot E}{A \cdot T_{1/2}} \right) \quad (\text{in W})$$

3.2.2 Gamma Emission

Gamma emission accompanies most, if not all, alpha decay. The maximum gamma decay power is calculated with the same equation as alpha decay. This equation assumes (a conservative assumption) that all the gamma rays are absorbed. In the majority of cases, the sum of the products of the gamma energy and intensity is small; thus gamma power may be neglected.

3.2.3 Spontaneous Fission

The energy release from spontaneous fission is assumed to be 200 MeV, equal to that from induced fission of ^{235}U and/or ^{239}Pu . Also, the conservative supposition is made that the 200 MeV is deposited totally within the sample. The power from spontaneous fission, where E has been replaced by 200 MeV and $T_{1/2}$ has been replaced by T_{SF} (the spontaneous fission half-life) is

$$P_{\text{SF}} = 1.34 \times 10^{13} \frac{G}{A \cdot T_{\text{SF}}} \quad (\text{in W}),$$

where T_{SF} = spontaneous fission half-life (in s), and again,

$$e^{-\lambda t} = 1.$$

3.3 NEUTRON EMISSION RATE

The neutron emission rate is required for the shielding calculations. Neutrons, which result from spontaneous fission, may be calculated by

$$\dot{\phi} = \text{spontaneous fission rate} \times \text{neutrons emitted per fission.}$$

The spontaneous fission rate per second is expressed in the same manner as the decay rate.

$$\lambda \cdot N = \left(\frac{0.693}{T_{\text{SF}}} \right) \left(\frac{6.023 \times 10^{23} G}{A} \right) \quad (\text{in fission/s}),$$

where T_{SF} = spontaneous fission half-life (in s), and

$$e^{-\lambda t} = 1.$$

The neutrons emitted per fission (ν) can be found in the literature. One available source is Ref. 7. In the absence of definite data, a conservative value is 4.0. The neutron emission rate is

$$\dot{\phi} = 4.175 \times 10^{23} \frac{G \cdot \nu}{A \cdot T_{\text{SF}}} \quad (\text{in neutrons/s}).$$

Table 3 lists the specific neutron emission rate for isotopes normally encountered in the use of the cask. Appendix B has an example of the use of the above expression and Table 3.

3.4 GAS EVOLUTION

3.4.1 Fission Gases

The radioactive sample will evolve gases as a result of spontaneous fission and alpha decay. Assuming that the spontaneous fission product distribution is the same as that of induced fission of ^{235}U , there are produced 5.18 atoms of krypton (Kr) and 21.25 atoms of xenon (Xe) per 100 atoms of ^{235}U fissioned.⁸ This data is for thermal fission of ^{235}U irradiated for 135 days with zero days cooling. The volume of gas evolved per fission is

$$V = \left(\frac{22\,400\text{ cm}^3\text{ of gas}}{\text{mol of gas}} \right) \left(\frac{\text{mol of gas}}{6.023 \times 10^{23}\text{ gas atoms}} \right) \left(\frac{0.2643\text{ gas atoms}}{\text{fission}} \right)$$
$$= 9.830 \times 10^{-21}\text{ cm}^3/\text{fission}.$$

The rate of spontaneous fission, found in Section 3.3, is

$$\lambda \cdot N = 4.175 \times 10^{23} \left(\frac{G}{A \cdot T_{\text{SF}}} \right) \quad (\text{in fission/s}).$$

The volume generation rate is therefore

$$V_{\text{SF}} = V \cdot \lambda \cdot N$$
$$= 4104 \left(\frac{G}{A \cdot T_{\text{SF}}} \right) \quad (\text{in cm}^3/\text{s}),$$

where the symbols are defined as before.

It is possible, though highly improbable, that fission gas buildup could be of consequence. Most of the radioisotopes in this report have spontaneous fission half-lives of years. It would require a rare combination of short spontaneous fission half-life, large amount of isotope, and long time in the capsule to generate a significant volume of fission gas.

3.4.2 Helium Gas Generation

Helium gas evolves from alpha decay. The quantity produced depends on the decay rate and yield of alpha particles per disintegration. Section 3.2.1 indicates the decay rate. The yield of alpha particles is the sum of the

intensities. In some cases it is 100% or 1.0, but it is frequently less.

The helium volume generation rate is

$$\begin{aligned}
 V_{\text{He}} &= V \cdot \lambda \cdot N \\
 &= \left(\frac{22\,400 \text{ cm}^3}{\text{mol of gas}} \right) \left(\frac{\text{mol of gas}}{6.023 \times 10^{23} \text{ atoms}} \right) \left(\frac{F \text{ helium atoms}}{\text{disintegration}} \right) \\
 &\quad \left(4.175 \times 10^{23} \right) \left(\frac{G}{A \cdot T_{1/2}} \right) \\
 &= 15\,527 \left(\frac{G \cdot F}{A \cdot T_{1/2}} \right) \quad (\text{in cm}^3/\text{s}),
 \end{aligned}$$

where, as before

G = mass of isotope (g),

F = intensity expressed as fraction,

A = atomic mass (amu), and

$T_{1/2}$ = half-life for alpha decay (s).

3.5 CRITICALITY

The cask can safely transport certain fissile isotopes. They are $^{242\text{m}}\text{Am}$, ^{243}Cm , ^{245}Cm , ^{247}Cm , ^{249}Cf , ^{251}Cf , ^{233}U , ^{235}U , ^{238}Pu , ^{239}Pu , and ^{241}Pu . The total combined mass of these isotopes shall not exceed 15 g. An added restriction is a mass limit of 3 g on the isotope ^{251}Cf .

Table 4 tabulates the critical masses for the isotopes.⁹ It also lists LLL limits set for the dry solid isotopes to be shipped in the cask. In no case is there a danger of attaining critical mass within LLL limits. An added margin of safety is present because the isotopes shipped will be dry whereas the listed data are for metal-water solutions.

Table 4. Critical masses of fissile isotopes.

Isotope	Calculated critical mass ^a (g)	Assigned cask limit (g)
^{242m} Am	23	
²⁴³ Cm	213	
²⁴⁵ Cm	42	
²⁴⁷ Cm	159	
²⁴⁹ Cf	32	Total mass of any combination not to exceed 15g.
²³³ U	590	
²³⁵ U	820	
²³⁸ Pu	4500 ^b	
²³⁹ Pu	530	
²⁴¹ Pu	260	
²⁵¹ Cf	10	3

^aCritical mass of solution sphere at optimum moderation, water-reflected.

^bCritical mass of metal sphere in fast, unmoderated system, steel-reflected.

4.0 Evaluation

4.1 THERMAL ANALYSIS

4.1.1 Normal Conditions of Transport

Case 1. The normal conditions of transport appear in Annex 1 of Appendix A as "Direct sunlight at an ambient temperature of 130°F (328 K) in still air." According to Ref. 10, part 173.393 (e) (2), General Packaging Requirements, "If the package is transported in a transport vehicle consigned for the sole use of the consignor, the maximum accessible external surface temperature shall be 180°F" (356 K).

Case 2. A second "normal" condition of transport is found in the same paragraph of Ref. 10: "The temperature of the accessible external surfaces of the package will not exceed 122°F (323 K) in the shade when fully loaded, assuming still air at ambient temperature." An ambient temperature of 311 K (100°F) will be assumed for this case.

Two other temperature restrictions apply to this specific cask design: a 401 K (262°F) limit on the maximum temperature of the gypsum-spodumene composite and a 550 K (530°F) limit on the lead in the gamma shield. These limits were set to prevent the gypsum from exceeding its dehydration temperature and forming a hemihydrate, and to prevent melting of the lead. The latter limit is 50 K below the melting point.

4.1.1.1 Solar Power Load

The solar power load (P_S) must be determined for case 1. It can be calculated as follows:

$$P_S = \alpha AI \quad (\text{in W}),$$

where

α = total solar absorptivity of surface,

A = area exposed to solar radiation (m^2), and

I = incident solar power intensity (W/m^2).

Solar absorptivity for the white epoxy paint used on all carbon steel surfaces is estimated from data in Ref. 11:

$$\alpha = 0.16 \quad (\text{white } \text{Y}_2\text{O}_3)$$

- = 0.18 (white ZnO)
- = 0.15 (" CaO)
- = 0.15 (" MgCO_3)
- = 0.14 (" ZrO_2)
- = 0.14 (" ThO_2)
- = 0.14 (" MgO)
- = 0.12 (" PbCO_3)

The actual type of pigment used in the paint was unavailable so α was assumed to be 0.16. The painted surface will be maintained in good condition by cleaning and repainting at appropriate intervals. The incident solar power intensity as a function of time is taken from Ref. 12. The data represent summer solar radiation striking walls and roofs at a latitude of 35°N. The maximum intensity listed occurs at this location.

The wall data are for the eight major points of the compass (east, south-east, south, etc.). Therefore, the vertical cylindrical shell of the cask was considered to be an octagon. Intensities were summed and an average uniform intensity was determined for each hour. Figures 4 and 5 plot the solar intensity on horizontal and vertical cylindrical surfaces. The curves were not smoothed to facilitate later calculations. The assumption of an average vertical intensity is valid considering the relative thermal conductivities of the steel shell and gypsum-spodumene mixture. The incident solar power load can more easily move circumferentially than radially.

4.1.1.2 Heat Losses to Environment

Heat losses to the environment are based on the cask surface temperature. The analysis includes both convection and radiation heat losses.

Convection heat transfer coefficients are determined from equations given by McAdams,² pp. 172-181:

Vertical cylinder

$$\text{Laminar: } h_c = 0.29 \left(\frac{\Delta T}{H} \right)^{0.25}$$

$$\text{Turbulent: } h_c = 0.19 (\Delta T)^{0.333}$$

Horizontal plate

$$h_c = 0.27 \left(\frac{\Delta T}{D} \right)^{0.25}$$

$$h_c = 0.22 (\Delta T)^{0.333}$$

Fig. 4. Solar intensity on horizontal cylindrical surface.

Fig. 5. Average solar intensity on vertical cylindrical surface.

where

h_c = convection heat transfer coefficient Btu/hr-ft²-°F,

ΔT = temperature difference between surface and air (°F),

H = height of cylinder (ft), and

D = diameter of cylinder top (ft).

These equations can be converted to SI* units by use of conversion factors.

*Le Systeme International d'Unités (the International System of Units).

The equations for laminar flow apply where the Rayleigh Number is between 10^4 and 10^9 . Calculation of this modulus indicates the flow is laminar except during the fire test where turbulent flow occurs.

The convection heat transfer is

$$q = h_c A (\Delta T),$$

where

$$q = \text{power (W)},$$

$$A = \text{respective area (m}^2\text{)},$$

$$\Delta T = \text{temperature difference between surface and air (K), and}$$

$$h_c = \text{W/m}^2 \cdot \text{K}.$$

The radiation heat transfer is determined from

$$q = \sigma F A (T_2^4 - T_1^4),$$

where

$$\sigma = \text{Stefan-Boltzman constant}$$

$$= 5.669 \times 10^{-8} \text{ W/m}^2 \cdot \text{K}^4,$$

$$F = \text{radiation form-factor for cask in large space}$$

$$= \epsilon_2,$$

$$\epsilon_2 = \text{emissivity of cask surface,}$$

$$\epsilon_2 = 0.96 \text{ (white epoxy paint}^{11}\text{)},$$

$$A = \text{respective area (m}^2\text{)},$$

$$T_2 = \text{cask surface temperature (K), and}$$

$$T_1 = \text{temperature of surroundings (K).}$$

4.1.1.3 Thermal Response

A heat balance for the cask was made under case 1 and 2 conditions. The unknown was the allowable radioactive decay power. Overriding restrictions were a maximum temperature of 401 K for the gypsum-spodumene composite and 550 K for the lead in the gamma shield.

The calculations were made by using the standard form of TRUMP⁽¹³⁾, a computer program for transient and steady-state temperature distributions in multidimensional system. TRUMP solves a general nonlinear parabolic partial differential equation describing flow in various kinds of potential fields, such as temperature. Steady-state and transient flow in one, two, or three dimensions are considered in geometrical configurations having simple or complex shapes and structures. Initial conditions may vary with spatial positions. Material properties, source and sink strengths, boundary conditions and other problem parameters may vary with spatial position, time, or the primary dependent variable-temperature. External sources or sinks, coupled to the system by means of specified boundary conditions, may vary with time.

The radioactive decay power was assumed to be uniformly distributed over the gamma shield inner surface. This is a conservative approach when neutron and/or gamma emitters are involved because some of this power would be deposited in the shielding.

The thermal conductivities and specific heats used throughout the calculations are those noted in section 2.2.2.1 for the gypsum-spodumene composite, and in section 2.2.2.2 for the depleted uranium. The higher thermal conductivity and lower specific heat of the steel shell were ignored. The overall effect of this assumption is to over-estimate the temperature difference; hence, it is conservative.

The container was cross-sectioned vertically and divided into 84 equal nodes about 15 cm long by 4 cm wide. The gamma shield was also sectioned into 14 nodes approximately 14 cm long by 4 cm wide, with an air gap between the inside wall of the container and the outside wall of the gamma shield. Heat transfer across this space was assumed to be by radiation only. The emissivity value used for the two stainless steel surfaces was 0.32⁽¹⁴⁾. The container bottom surface was modelled as an insulated boundary.

Case 1. A radioactive decay power of 250 W was assumed and the cask steady-state temperature distribution for an ambient temperature of 295 K was found. This temperature distribution was used as the starting point for the case 1 conditions. The solar power load as a function of time was applied as in Figs. 4 and 5. An ambient temperature as a function of time was likewise imposed. The temperature curve was determined in the following manner. The temperature data for the Nevada Test Site was examined to find the maximum recorded profile⁽¹⁵⁾. For station BJY, located near the center of the Yucca Flat area at NTS, the temperature reached a maximum of 318 K (112°F) during

the period 6/14/57 to 7/31/64. By adding 10 K (18°F) to the listed data, a day with a 130°F ambient temperature was obtained (Fig. 6).

The results of the TRUMP calculation indicate a maximum gypsum-spodumene temperature of 384 K, a maximum lead temperature of 457 K, and maximum surface temperature of 329 K. These maxima are below the allowables listed.

Case 2. The radioactive decay power 250 W was again assumed, and the ambient temperature stayed constant at 311 K. The maximum steady-state temperatures are 394 K for the gypsum-spodumene, 463 K for the lead and 320 K for the surface. All are acceptable values.

4.1.2 Hypothetical Accident Conditions

Annex 2 (in Appendix A) states the thermal conditions the cask must withstand: "Exposure to a thermal test in which the heat input to the package is not less than that which would result from exposure of the whole package to a radiation environment of 1475°F for 30 min with an emissivity coefficient of 0.9, assuming the surfaces of the package have an absorption coefficient of 0.8. The package shall not be cooled artificially until 3 hours after the test period unless it can be shown that the temperature on the inside of the package has begun to fall in less than 3 h."

The initial temperature distribution at the start of the thermal test is that which existed at the end of the case 2 conditions (Section 4.1.1). The solar power load is not applied during this test. The radiation form factor

Fig. 6. Ambient temperature profile for cask (based on data recorded at Nevada Test Site).

for the test is

$$F = \left(\frac{1}{\epsilon_2} + \frac{1}{\epsilon_1} - 1 \right)^{-1},$$

where

$$\epsilon_2 = 0.9,$$

$$\epsilon_1 = 0.8, \text{ and}$$

$$F = 0.735.$$

The temperature profiles at the horizontal centerline of the cask are shown in Fig. 7 for the thermal test. One curve shows the resultant profile after 30 min at 1075 K. The second curve indicates the profile 3 h after the end of the 1075 K exposure.

Fig. 7. Temperature profiles at hottest horizontal plane.

Examination of the curves indicates the following points:

- The thermal test has a negligible effect on the special-form capsule thermal environment.
- Dehydration of the shielding composite occurs beyond a radius of 45.5 cm during the 1075 K exposure.
- The gamma shield experiences no temperature change due to the insulating effect of the gypsum-spodumene composite and inside air gap.

4.1.3 Special Form Capsule Heating Test

Annex 4 of Appendix A states that special-form capsules must retain their contents when heated in air to a temperature of 1075 K (1475°F). They must remain at that temperature for 10 min. Accordingly, capsules assembled with helium in the cavity have been heated to 1089 K (1500°F) in an oven and allowed to soak for 15 minutes before removal. During and after cooling to room temperature, they were leak-checked with a helium mass spectrometer having a sensitivity of 1×10^{-9} std cm³/s. No leaks were detected.

The maximum temperature the capsule would experience during the hypothetical accident condition is determined below:

$$q = \sigma F A_1 (T_1^4 - T_2^4),$$

$$q = 250 \text{ W},$$

A_1 = surface area of first capsule listed in Fig. 2 (only one that will fit),

$$= 208 \text{ cm}^2,$$

A_2 = surface area of cask cavity = 339 cm^2 ,

T_2 = 463 K (from TRUMP results),

$$F = \left[\frac{1}{\epsilon_1} + \frac{A_1}{A_2} \left(\frac{1}{\epsilon_2} - 1 \right) \right]^{-1}$$

$$= \left[\frac{1}{0.32} + \frac{208}{339} \left(\frac{1}{0.32} - 1 \right) \right]^{-1} = 0.23, \text{ and}$$

$$T_1 = \left[463^4 + \frac{250}{(5.67 \times 10^{-8})(0.23)(0.0208)} \right]^{0.25}$$

$$= 992 \text{ K (1325°F)}.$$

As can be seen this temperature is 83 K below the temperature at which the capsule was actually tested. Also, this analysis ignores any heat transferred by conduction and convection from the capsule to the cask cavity.

4.2 STRUCTURAL ANALYSIS

4.2.1 Load Resistance

When regarded as a simple beam supported at its ends along any major axis, the shipping cask must be capable of withstanding a static load, normal to and uniformly distributed along its length, equal to five times the fully loaded cask weight without generating stresses in any material of the cask in excess of the yield strength of that material.

The cask is shown in Fig. 1. The container outer shell is American Society for Testing Materials (ASTM) A 36 carbon steel with a minimum specified strength of 248 MPa (36,000 psi). The maximum weight of the cask is 31.14 kN. Stresses in the container resulting from the uniform load are determined, as recommended by Shappert,⁽¹⁶⁾ from the following equation:

$$S = \frac{M}{Z},$$

where

S = stress (Pa),

M = maximum bending moment of cask = $\frac{5 WL}{8}$,

Z = section modulus of cask = $\pi R_o^2 t$ (for a large diameter, thin-walled cylinder),

W = weight of cask = 31140 N,

L = length of cask = 123.2 cm,

R_o = outside radius of cask = 52.0 cm, and

t = thickness of cask wall = 0.64 cm.

The computed maximum bending moment is

$$M = \frac{(5)(31140)(1.232)}{8} = 24000 \text{ N} \cdot \text{m}.$$

The computed section modulus is

$$Z = \pi R_o^2 t = \pi (0.52)^2 (0.0064) = 5.4 \times 10^{-3} \text{ m}^3.$$

The maximum bending stress is then

$$S_{\max} = \frac{24,000}{5.4 \times 10^{-3}} = 4.44 \text{ MPa.}$$

Since this stress value is less than 2% of the material yield stress, the cask satisfies the load-resistance requirements.

4.2.2 Compression

This test requires a compressive load equal to either 5 times the weight of the cask or 13.8 kpa (2 psi) multiplied by the maximum horizontal cross section of the cask, whichever is greater. The load must be applied during a period of 24 h, uniformly against the top and bottom of the cask in the position in which the cask would normally be transported.

The calculations are based on a load of 155.7 kN (35,000 lb) which is five times the maximum gross weight since this is the higher loading. The longitudinal compressive stress is calculated by dividing the load by the cross-sectional area of the container:

$$S = \frac{L}{\pi D t} = \frac{155.7}{\pi (104.1)(0.64)} = 7.50 \text{ MPa.}$$

This stress value is only slightly greater than 3% of the shell yield stress.

The critical buckling stress (S_{cr}) of the cylindrical shell, when subjected to uniform axial compression, is calculated¹⁷ to determine the ultimate capability of the cask:

$$S_{cr} = \frac{Eh}{R\sqrt{3(1-\mu^2)}},$$

where

E = modulus of elasticity = 207 GPa,

h = cask thickness = 0.64 cm,

R = radius of cask = 52 cm, and

μ = Poisson's ratio = 0.3.

Thus, the critical buckling stress is

$$S_{cr} = \frac{207 \times 10^6 (0.64)}{52\sqrt{3(1-0.3^2)}} = 1.53 \text{ GPa.}$$

This value for the ultimate capability is 200 times greater than the longitudinal compressive stress in the cask calculated above. Thus, placing a 155.7 kN load on the top of the cask would have no effect.

4.2.3 External Pressure

The containment vessel shall be capable of withstanding an external pressure of 172 kPa (25 psi) with no loss of contents.

The container outer shell is supported everywhere by the internal shielding material. However, by ignoring this support and assuming an empty shell, the maximum bending stresses in the container cover and the bottom-end plate are determined. The model for the bottom assumes the plate acts like two semicircular plates with all edges fixed. This assumption is based on the stiffening effect of the I beam to plate welds and plate-to-shell welds. The container cover is assumed to have fixed edges by virtue of the bolting.

$$S_{\text{bottom}} = \frac{0.42PR^2}{h^2} \quad (\text{Ref. 17}),$$

$$S_{\text{cover}} = \frac{0.75PR^2}{h^2} \quad (\text{Ref. 17}),$$

where

S = maximum bending stress (Pa),

R_{BOT} = radius of plate = 52 cm,

R_{cover} = radius of bolt circle = 36.5 cm,

h_{bot} = thickness of plate = 1.27 cm,

h_{cover} = thickness of cover = 2.54 cm, and

P = pressure = 172 kPa.

Thus,

$$S_{\text{bottom}} = \frac{0.42(172)(52)^2}{(1.27)^2} = 121 \text{ MPa},$$

$$S_{\text{cover}} = \frac{0.75(172)(36.5)^2}{(2.54)^2} = 27 \text{ MPa}.$$

The maximum bending stress in the cover is 11% of the minimum yield strength, and the bottom plate maximum stress is 49% of the yield strength for ASTM A36 carbon steel (248 MPa).

The maximum membrane stress in the cask body is the hoop stress expressed as:

$$S_{\max} = \frac{PR}{T},$$

where

$S_{\max}$ = maximum hoop stress (Pa),

P = pressure = 172 kPa,

R = radius of body = 52 cm, and

T = body wall thickness = 0.64 cm.

Therefore,

$$S_{\max} = \frac{172(52)}{0.64} = 14 \text{ MPa.}$$

This value is only 6% of the yield stress.

The third consideration is the buckling strength. The buckling strength of the container body can be determined by using the procedures specified in the ASME Pressure Vessel Code, Section VIII, (18) which provides a value for the critical pressure.

The ASME pressure vessel code states that the allowable external pressure is given by the expression

$$P_{\text{allowable}} = \frac{B}{D/t},$$

where

$P_{\text{allowable}}$ = the allowable pressure load of the vessel (psi),

D = diameter of vessel = 41.0 in.,

t = thickness of vessel = 0.25 in., and

B = constant depending on the ratios D/t and L/D (where
 L = length of vessel), = 5250.

For the cask shell, the allowable pressure load is

$$P_{\text{allowable}} = \frac{5250}{41/0.25} = 32 \text{ psi} = 220 \text{ kPa.}$$

The allowable external pressure is greater than the required evaluation pressure of 172 kPa. Note again that this ignores the fact that the shell is supported internally by the shielding composite.

4.2.4 Internal Pressure

4.2.4.1 Cask

The space inside the walls of the container and cover bodies is filled with the gypsum-spodumene shielding composite. If a fire or other source of heat is assumed, it is possible to raise the temperature of the shielding composite to the point that it decomposes. The normal process would be the release of water of hydration from the gypsum beginning at 401 K (262°F). This released water would appear as steam in the shielding space and cause a pressure increase. To preclude the pressure from rising to the point where rupture could occur, the container and cover are equipped with relief valves that lift at 69 kPa (10 psi). Hence, the shielding volumes will be able to vent as the temperature rises above the saturation temperature of 388 K (239°F).

The internal pressure capability is evaluated below. The analysis ignores the devices that prevent pressure buildup higher than 69 kPa from occurring. The components analyzed are

- Container outer cylindrical shell.
- Container top plate.
- Container inner cylindrical shell.

In the analysis, the top plate is considered a circular plate with a circular hole. Both edges are supported and fixed and the pressure acts on the entire surface. The weld at the shell-head junction justifies the outer-edge boundary condition. The second boundary condition is based on the fact that the inner edge of the top plate is joined to the bottom-end plate by the inner cylindrical shell.

Stresses in the bottom plate were calculated for the external pressure case (Section 4.2.3), and the equation is merely re-solved for the allowable pressure. The maximum allowable pressure for the container body is based on the smallest value calculated:

$$P = \frac{St_o}{R_o} = \text{outer cylindrical shell},^{17}$$

$$P = \frac{Sh^2}{\beta R_o^2} = \text{top plate},^{17}$$

$$P = \frac{St_1}{R_i} = \text{inner cylindrical shell, }^{17} \text{ and}$$

$$P = \frac{Sh^2}{0.42Ro^2} = \text{bottom plate, }^{17}$$

where

S = yield stress = 248 MPa,

β = coefficient based on ratio of o.d. to i.d. = 0.078,

P = pressure (Pa),

R_o = outer radius = 52.0 cm,

R_i = inner radius = 20.3 cm,

t_o = outer shell thickness = 6.4 mm,

h = plate thickness = 12.7 mm, and

t_i = inner shell thickness = 6.4 mm.

Thus,

$$P_{\text{out cyl}} = \frac{(248)(6.4)}{520} = 3.1 \text{ MPa,}$$

$$P_{\text{top}} = \frac{(248)(12.7)^2}{(0.078)(520)^2} = 1.9 \text{ MPa,}$$

$$P_{\text{inner cyl}} = \frac{(248)(6.4)}{203} = 7.8 \text{ MPa, and}$$

$$P_{\text{bottom}} = \frac{248(12.7)^2}{0.42(520)^2} = 352 \text{ kPa.}$$

From the above values, the maximum allowable pressure based on the yield stress is 352 kPa (51 psi) in the space occupied by the shielding composite.

The maximum allowable pressure in the container cavity is determined by comparing the cover equation (Section 4.2.3) with the inner cylinder shell result.

$$P_{\text{cover}} = \frac{Sh^2}{0.75R^2} = \frac{248(1.27)^2}{0.75(36.5)^2} = 0.4 \text{ MPa,}$$

$$P_{\text{inner cyl}} = 7.8 \text{ MPa.}$$

Since the maximum pressure for the cover is less than the inner cylinder shell value, it controls.

4.2.4.2 Special-Form Capsule

The special-form capsule, described in Section 2.5.3, is designed by the manufacturer for a pressure of 1.03 MPa at 1089 K. According to Ref. 18, the maximum allowable tensile stress at this temperature is 4.83 MPa. At the maximum temperature (992 K) that the capsule experiences (see Section 4.1.3), the maximum allowable tensile stress is 12.76 MPa. The corresponding pressure rating at 992 K is

$$P_1 = P_D \left(\frac{S}{S_0} \right) = 1.03 \left(\frac{12.76}{4.83} \right) = 2.7 \text{ MPa (395 psi).}$$

If we assume the gas in the capsule follows Charles' Law, the gas pressure at standard conditions is

$$P_2 = P_1 \left(\frac{T_2}{T_1} \right),$$

where

$$P_1 = 2.7 \text{ MPa,}$$

$$T_1 = 992 \text{ K, and}$$

$$T_2 = 293 \text{ K.}$$

Thus,

$$P_2 = 2.7 \left(\frac{293}{992} \right) = 0.80 \text{ MPa.}$$

The primary cause of the pressure is fission gas and helium gas resulting from alpha decay. The remainder of this section is devoted to an examination of the parameters involved in the generation of this gas.

4.2.4.2.1 Maximum Allowable Gas Generation - The capsule volumes are listed in Fig. 2. The free volume available for the generated gas to occupy is

$$V_F = V_C - V_S,$$

where

$$V_F = \text{free volume (cm}^3\text{),}$$

$$V_C = \text{capsule volume from Fig. 2 (cm}^3\text{), and}$$

$$V_S = \text{sample volume (cm}^3\text{).}$$

The volume of generated gas that raises the capsule pressure from atmospheric (101.3 kPa) to 0.80 MPa is found from

$$V_G = V_F \left(\frac{P_2 - P_0}{P_0} \right) = \left(\frac{0.80 - 0.101}{0.101} \right) V_F = 6.92 V_F \text{ cm}^3.$$

The fission-gas and helium-gas generation rates in Sections 3.4.1 and 3.4.2 are used to calculate the generated gas volumes:

$$V_{SF} = 4104 \left(\frac{G}{A \cdot T_{SF}} \right) \text{ cm}^3/\text{s, and}$$

$$V_{He} = 15527 \left(\frac{G \cdot F}{A \cdot T_{1/2}} \right) \text{ cm}^3/\text{s}.$$

Also, it may be noted that V_{SF} and V_{He} multiplied by the time the capsule is sealed is equal to V_G , or

$$V_G = \emptyset \left(V_{He} + V_{SF} \right),$$

where $\emptyset$ = amount of time sealed (s).

Combining the above equations yields the expression below. When the left and right hand sides are equal, the capsule pressure will be at the maximum allowable for the hypothetical accident conditions. If the left hand side is larger the capsule pressure is below the maximum allowable.

$$\frac{6.92 (V_C - V_S)}{\emptyset} \geq G \left[\frac{4104}{A \cdot T_{SF}} + \frac{15527 F}{A \cdot T_{1/2}} \right].$$

The parameters again are

A = atomic mass (amu),

G = sample weight (g),

V_S = sample volume (cm^3),

$\emptyset$ = amount of time that capsule is sealed (s),

F = number of alpha particles released per disintegration,

$T_{1/2}$ = half life for alpha decay (s),

T_{SF} = spontaneous fission half life (s), and

V_C = capsule volume from Fig. 2 (cm^3).

The second term on the right hand side of the expression is the maximum gas-generation rate for the isotopes involved. Table 3 lists the specific gas-generation rates for isotopes that would normally be encountered in the use of the cask. An example of the use of the expression and table is given in Appendix B.

4.2.5 Impact

4.2.5.1 Free Drop

4.2.5.1.1 Cask - A free drop through a distance of 9.14 m (30 ft) onto a flat, essentially unyielding horizontal surface, striking the surface in a position for which maximum damage is expected, will not reduce the effectiveness of the packaging. The validity of this statement is demonstrated by analyses for end-drop, side-drop and angle-drop geometries.

The energy to be absorbed in the free drop is

$$E = Wh_g,$$

where

$$g = \text{acceleration} = 9.8 \text{ m/s}^2,$$

$$W = \text{maximum cask mass} = 3175 \text{ kg, and}$$

$$h = \text{drop distance} = 9.14 \text{ m,}$$

Thus,

$$E = 3175 \times 9.14 \times 9.8 = 284.6 \text{ kN} \cdot \text{m}.$$

4.2.5.1.1.1 End Drop — Examination of Fig. 1 indicates that a cask dropped on its top is damaged worse than one dropped on its bottom, where the pallet base tends to deform and minimize damage to the cask itself. A cask dropped on its top would probably drive the container and container-cover lifting lugs into the shielding composite. However, the lugs could not damage the cask cavity or special-form capsule.

Energy is absorbed as the cask's outer shell and shielding composite are deformed. This deformation is considered in three cases: cases 1 and 2, using the same method but employing different models; and case 3, using a computer code. In cases 1 and 2, the cylindrical shell of the cask is assumed to function only to support the shielding composite in the radial direction. The decrease in cask height is calculated from test results obtained from Ref. 19. The text is below and the curve is shown in Fig. 8.

"The tests consisted of guided free-fall drops of steel cylinders of known weights and known end-face areas from measured heights to impact on the test surfaces (gypsum-spodumene) normal to the fall line. The impact, or fall, energies (in inch-pounds) per end-face areas of the steel bobs (in square inches) were then plotted vs the actual measured depths of penetration into the gypsum-spodumene.

"Tests with steel impacting on lead were conducted similarly.

"Unsifted gypsum-spodumene was felt to be non-homogeneous for the small face areas tested to be representative of a drop on a 6-inch diameter pedestal. Accordingly, 100 mesh gypsum was used for most tests.

"The gypsum-spodumene mixture is 40% (by weight) gypsum, 30% spodumene, 30% water, approximately 0.3% retarder. Curing time was 18 hours when the tests were made. Gypsum-spodumene specimens were bare; the cylinder impacted directly onto the test surface — no cladding intervening.

"It will be seen (Figure 8) that the plot on log-log scales is not linear. Rather the line exhibits positive curvature, stating in effect that the deeper the impact (and holding impact area constant) the more energy is absorbed per given interval of impact depth. This variability can be examined by determining the energy absorbed by each .1" interval throughout a full .9" deep impact crater, and the results from such differential analysis are shown at the right hand of the graph. These are the values used in calculating the impact depth of the package when dropped on its

Fig. 8. Impact tests on gypsum-spodumene. The right hand scale accounts for variable compressibility. Section 4.2.5.1.1 explains the scale in further detail.

side. The positive curvature, or variable compressibility, is interpreted as a continuous compression of the gypsum-spodumene mixture throughout the full impact depth until fracture occurs. (Or instead, cratering occurs, since at no time was fracturing observed.) Cratering occurs in only the top layers of mixture, and then only because the material was exposed rather than clad."

Case 1. In case 1, the entire cask top-surface area is used to absorb the impact, and the total cask mass is involved:

$$\frac{E}{A} = \frac{284.6}{0.785(1.041)^2} = 334 \text{ kN/m (1910 in.} \cdot \text{lb/in.}^2\text{)}.$$

The depth of penetration (compression) from Fig. 8 is 0.33 in. (0.84 cm).

Case 2. Examination of Fig. 1 and comparison of the masses of the cask and the gamma shield indicate the gamma shield would act as a ram during an end drop. The analysis for this case assumes that the gamma shield impacts on a cylinder of gypsum-spodumene composite the same diameter as the gamma shield. This cylinder must absorb all the deformation.

The mass of the gamma shield is 862 kg while the mass of the container cover in a 37-cm-diam circle is 95.4 kg:

$$E = W \cdot h \cdot g = (862 + 95.4)(9.14)(9.8) = 85.8 \text{ kN} \cdot \text{m}.$$

$$\begin{aligned} \frac{E}{A} &= \frac{85.8}{0.785 \times 0.37^2} \\ &= 805 \text{ kN/m (4600 in.} \cdot \text{lb/in.}^2\text{)}. \end{aligned}$$

The compression as determined from Fig. 8 is 0.65 in. (1.65 cm).

Case 3. A second approach to the deformation produced by an end drop was the use of the LLL computer code DRIVE⁽²⁰⁾. DRIVE is an elastic-plastic large-deformation computer code for calculating the collapse of a spherical shell enclosed in an accelerating rigid cavity. The code treats the large, dynamic, axisymmetrical elastic-plastic response of a single shell of arbitrary initial shape. The shell is enclosed in an elastic cavity with an initial gap between the shell and the cavity.

Loading can be applied by motion of the cavity and/or by the time and spatially varying pressure on the inner or outer surface. The cask was modeled as a 52-cm radius by 123-cm long by 6.4-mm-thick cylindrical steel

shell. The shell is filled with an elastic compressible material having a bulk modulus of 6.9 GPa. The total mass of the filler material is 2950 kg, and it is assumed to act hydrostatically on the steel shell. The cask is dropped on a steel plate with a modulus of elasticity of 207 GPa. The plate is assumed to remain elastic at all times to ensure a conservative result. The results indicate the upper-end plate will bulge outwardly a maximum of 10 cm at its center while the bottom-end plate will be dimpled inwardly a maximum of 5 cm at its center. The shell diameter will remain relatively unchanged. These results indicate no loss of shielding thickness between an object at the cask center and any other surface.

The maximum strain occurs at the corners and is 0.06 cm/cm. This denotes some yielding but no failure.

4.2.5.1.1.2 Side Drop - The side-drop analysis is performed two different ways. The first method is that used in case 1 of the end drop.

Case 1. The depth of penetration, or the extent of flattening, of the cylindrical surface is determined in incremental steps. As the depth increases, the energy absorption per-unit-area increases. Therefore, by first selecting a depth, the matching energy absorption per-unit-area can be found in Fig. 8. This value is then multiplied by the respective area to determine the energy absorption for that depth. The procedure is repeated until the sum energy absorption reaches E. The area is calculated from

$$A = 2L (2hR - h^2)^{1/2},$$

where

A = area of deformation (cm^2),

L = length of cask = 123.2 cm,

h = depth of penetration (cm),

R = cask radius = 52.05 cm, and

E = 284.6 kN · m (2.52×10^6 in. lb).

Table 5 summarizes the results of the calculation. The total penetration (actually compression) is seen to be about 0.8 in. (2.03 cm). This depth is only 6.4% of the shield thickness.

Case 2. This analysis employs the method used in the Cask Designers Guide.¹⁶ The deformation or shell flattening is calculated from

$$d \approx R(1 - \cos\theta),$$

Table 5. Deformation caused by ride drop.

Depth of compression, h (in.)	Area of deformation, A ₂ (in.)	Energy absorption per unit area, $\frac{\Delta E}{A}$ (in. - lb/in. ²)	Energy absorption in depth interval, ΔE (in. - lb)	Sum of energy absorption ΣE , (in. - lb)
0.1	196	420	82,300	82,300
0.2	277	580	160,700	243,000
0.3	339	700	237,300	480,300
0.4	391	750	293,200	773,500
0.5	436	830	361,900	1,135,400
0.6	478	920	439,800	1,575,200
0.7	515	1,000	515,000	2,090,200
0.8	550	1,080	594,000	2,684,200

where

d = deformation (in.),

R = outer radius = 20.5 in., and

θ = half angle of flat developed due to impact (degrees).

θ is a function of the nondimensional Resistance and Energy parameters which are, respectively,

$$\frac{R \cdot S_{GS}}{t_s \cdot S_s} + \frac{2Rt_e}{Lt_s} \text{ and } \frac{WH}{Rt_s LS_s},$$

where

W = cask weight = 7000 lb,

H = drop height = 30 × 12 = 360 in.,

R = outer radius = 20.5 in.,

t_s = outer shell thickness = 0.25 in.,

t_e = thickness of steel end plate = 0.50 in.,

L = length of shell = 48.5 in.,

S_s = dynamic flow stress in steel = 33,000 psi, and

S_{GS} = dynamic flow stress in gypsum-spodumene = 460 psi (assumed equal to ultimate compressive strength).

Thus,

$$\frac{WH}{Rt_s LS_s} = \frac{7000 \times 360}{20.5 \times 0.25 \times 48.5 \times 33,000} = 0.307,$$

and

$$\begin{aligned} \frac{R \cdot S_{GS}}{t_s S_s} + \frac{2Rt_e}{Lt_s} &= \frac{20.5 \times 460}{0.25 \times 33,000} + \frac{2 \times 20.5 \times 0.50}{48.5 \times 0.25} \\ &= 1.143 + 1.691 = 2.834. \end{aligned}$$

From the nomograph (Fig. 2.21 of Ref. 16) relating the half angle, Resistance, and Energy parameters: $\theta = 30.5^\circ$, and

$$d = R (1 - \cos\theta) = 2.84 \text{ in (7.2 cm).}$$

4.2.5.1.1.3 Corner Drop - The analyses for the end and side drops indicate that loss of shielding thickness is not the danger in a cask drop. The danger is a failure that would allow the special-form capsule to escape from the cask, which is most probable after a top-corner drop. If the cover bolts failed, the gamma shield containing the capsule, or the rack with several capsules, could be released. The impact of the top corner of the cask on unyielding surface would cause the container cover and contents to impact against the cover.

As a worst case, the drop angle must be very close to vertical to obtain maximum impact. Larger angles would cause the cask corner to hit first and decelerate the package. Also, some of the impact energy of the cover and contents would be directed against the inner surface of the container body and not the cover.

To illustrate the worst possible condition, it was assumed that the cask fell upside down over a cavity so that the 12 cover studs and sleeves would

absorb all of the kinetic energy of the moving mass. The energy-per-stud to be absorbed is

$$E = Whg/N = \frac{1154 \times 9.14 \times 9.8}{12} = 8.62 \text{ kN} \cdot \text{m}.$$

The capability of the stud and sleeve to absorb this energy was tested by dropping two test units that duplicated the geometry and conditions for an end drop. A 96 kg mass was dropped vertically onto a rigid steel plate so that it loaded the stud in tension and the sleeve in compression. All of the kinetic energy from the moving 96-kg mass was absorbed by the stud-mounting sleeve and the stud. The stud and sleeve are shown in Fig. 9.

In both cases the stud-mounting sleeve was deformed into a bellows shape in the thin wall section, and the length of the sleeve was shortened approximately 5.08 cm. The length of each stud was increased by 7%. Graphic integration of Fig. 10, the stress-strain curve for Type 304 stainless steel, indicates that at 7% elongation, the area under the curve is 24.8 MPa (3600 psi). At the ultimate strength (55% strain), the area is 308.2 MPa (44,700 psi).

All units: in. (cm)

Fig. 9. Stud mounting detail.

Fig. 10. Stress-strain curve: Type 304 SS (based on LLL test).

This comparison shows that, theoretically, the energy absorbed in the 9.14-m-drop test is one twelfth of that required to produce failure in the stud:

$$\left(\frac{308.2}{24.82} = 12.4 \right).$$

4.2.5.1.2 Special-Form Capsule - The special-form capsule was subjected to a free-fall drop of 9.14 m. Filled with helium at atmospheric pressure, it was dropped onto a rigid steel plate. The point of contact was controlled so that the seal area was subjected to the most damaging impact. After the tests, the capsule was checked for leak tightness, with a helium mass spectrometer having a sensitivity of 1×10^{-9} standard cm^3/s . No leakage was found.

4.2.5.2 Penetration

A rod with a diameter of 3.18 cm and mass of 5.9 kg was dropped from a height of 1.02 m onto a 6.4-mm-thick steel plate simulating the cask shell. The plate showed no evidence of damage.

4.2.5.3 Puncture

The conditions of the hypothetical accident related to puncture are stated below:

The capsule is dropped 40 in. in such a position to sustain maximum damage when it strikes the top end of a vertical, cylindrical mild steel bar mounted on an essentially unyielding horizontal surface. The bar shall be 6 in. in diameter, with the top horizontal and its edge rounded to a radius of not more than 0.25 in. It shall be long enough to cause maximum damage to the package but shall not be less than 8 in. long. The long axis of the bar shall be perpendicular to the unyielding horizontal surface.

An experimental investigation was made of the puncture resistance of lead shielded steel-jacketed shipping casks and the results reports in Ref. 21. For the hypothetical accident conditions listed above, the report recommends the use of the following equation to predict incipient puncture:

$$t = \left(\frac{W}{S} \right)^{0.71},$$

where

t = minimum shell thickness to prevent incipient puncture (in.),

w = cask weight = 7000 lb,

S = ultimate tensile strength of shell = 58,000 psi (ASTM-A36), and

$$t = \left(\frac{7000}{58000} \right)^{0.71} = 0.223 \text{ in. (5.7 mm).}$$

The actual shell thickness is 6.4 mm so puncture is improbable.

4.2.5.4 Percussion

A 2.54-cm-diam steel rod with a mass of 1.36 kg was dropped 1.02 m onto the wall of the special-form capsule. The capsule tested had a maximum outside diameter and minimum wall thickness (3.00 in. o.d. × 0.125 in. wall). No damage was observed.

4.2.6 Lifting Devices

The system of lifting devices shall be capable of supporting three times the weight of the package without generating stress in any material in excess of its yield strength.

4.2.6.1 Container Body

The container body has two sets of lifting devices: the fork-lift openings at the base and the four lifting lugs at the top.

4.2.6.1.1 Base Analysis — The analysis was made on the assumption that the total weight was uniformly distributed over a rectangular area the width of the fork-lift channels and a length equal to the diameter of the cask. (See Fig. 11.)

All units: cm

Fig. 11. Container base.

The short ends of the base are assumed fixed because they are welded to the container shell while the long ends are simply supported. The stiffening effect of the I-beams and the 6.4-mm-thick steel plate at the bottom of the pallet are ignored.

The maximum stress on the container bottom plate is

$$S = \beta \frac{Wb^2}{t^2} \text{ (Ref. 17),}$$

where

$$S = \text{stress} = \text{Pa},$$

$$W = \text{load per unit area} = \frac{3175 \times 9.8 \times 3}{\frac{\pi}{4} \times (104.1)^2} = 109.7 \text{ kPa},$$

$$b = \text{width} = 53.34 \text{ cm},$$

$$t = \text{thickness} = 1.27 \text{ cm, and}$$

$$\beta = 0.7146.$$

Thus,

$$S = 0.7146 \frac{109.7 \times (0.5334)^2}{(0.0127)^2} = 138 \text{ MPa}.$$

The yield stress for the ASTM-A36 steel base is 248 MPa, or

$$\frac{138}{248} \approx 56\% \text{ of yield stress.}$$

4.2.6.1.2 Lifting-Lug Analysis (Case 1) - A model of the lug is shown in Fig. 12. The lug has a hole on the top for lifting and one on the side for tiedown. A lifting analysis is made for both holes. The lug actually is one piece and protrudes through the container top and shell. It is completely welded into both of these pieces, but the shell weld is ignored in the analysis. The vertical load-per-lug is

$$P_V = \frac{3W}{4} = \frac{3 \times 3175 \times 9.8}{4} = 23.34 \text{ kN}.$$

The load can be resolved into a component along the axis of the weld, a component perpendicular to the centroid of the weld, and a moment about the weld centroid. The load angle of 60° results from using 91-cm-(36-in.) long slings.

Fig. 12. Model of container lifting lug (case 1).

The horizontal load is

$$P_H = \frac{P_V}{\tan 60^\circ} = 13.48 \text{ kN.}$$

The counter clockwise moment due to the horizontal load is

$$M_H = P_H \times d_H = 13.48 \times 4.55 = 61.3 \text{ kN} \cdot \text{cm.}$$

The clockwise moment due to the vertical load is

$$M_V = 23.34 \times 3.1 = 72.4 \text{ kN} \cdot \text{cm, and}$$

$$\Sigma M = M_V - M_H = 11.1 \text{ kN} \cdot \text{cm.}$$

The lug-to-body weld experiences a shear stress and two normal stresses. The shear stress in the weld due to the horizontal load is

$$S_S = \frac{0.707P_H}{f(L + h)}, \quad (\text{Ref. 22})$$

where

f = fillet size = 0.95 cm,

L = length of weld on one side = 11.8 cm,

h = lug thickness = 1.9 cm, and

$P_H = 13.48$ kN.

Thus,

$$S_S = \frac{0.707 \times 13.48}{0.95(13.7)} = 7.32 \text{ MPa.}$$

The normal stress in the weld due to the vertical load is

$$S_n = \frac{0.707 P_v}{f(L + h)} = \frac{0.707 \times 23.34}{0.95(13.7)} = 12.68 \text{ MPa.}$$

The normal stress in the weld due to the bending moment is

$$\begin{aligned} S_n &= \frac{4.24 M}{f[L^2 + 3h(L + f)]} && (\text{Ref. 22}) \\ &= \frac{4.24 \times 11.1}{0.95[11.8^2 + 3 \times 1.0 \times 12.8]} = 2.34 \text{ MPa.} \end{aligned}$$

The maximum shear stress in the weld is

$$\begin{aligned} S_{MAX} &= \left[\left(\frac{S_n}{2} \right)^2 + S_S^2 \right]^{1/2} \\ &= \left[\frac{15.02^2}{2} + 7.32^2 \right]^{1/2} = 10.5 \text{ MPa.} \end{aligned}$$

By using the maximum shearing-stress theory (Treska's Yield Criteria), the weld will not fail if S_{MAX} is less than half the yield strength. The AWS E60 weld-metal yield strength is 344.8 MPa, and the factor of safety is about 16.

The lifting-lug holes are sized for steel shackles with a 2.86-cm-diam pin. The safe working load is 49.82 kN for drop, forged-pin type shackles of this size (Appendix D, Table D-4). The actual shackle load is

$$P = (P_H^2 + P_V^2)^{0.5} = (13.48^2 + 23.34^2)^{0.5} = 26.95 \text{ kN.}$$

The smallest shackle that can be used safely is one with a 2.54-cm-diam pin, which has a 38.25 kN working load. By assuming this size shackle is used inadvertently, three other stresses in the lug plate are determined. These have common simplifying assumptions:

- The bearing stress is uniformly distributed over an area equal to the pin diameter-times-the lug thickness.
- The shear stress is uniformly distributed over an area equal to two times the pin diameter-times-the distance from the hole to the edge (measured on the centerline),
- The tensile stress in the lug is based on the net cross-sectional area at the hole location.

The bearing stress calculation follows:

$$S_b = \frac{P}{dh},$$

where

$$d = \text{pin diameter} = 2.54 \text{ cm.}$$

Thus,

$$S_b = \frac{26.95}{2.54 \times 1.9} = 55.8 \text{ MPa.}$$

The shear stress is

$$S_s = \frac{P}{2he},$$

where

$$e = \text{distance to edge} = 3.81 - 1.59 = 2.22.$$

Thus,

$$S_s = \frac{26.95}{2 \times 1.9 \times 2.22} = 31.9 \text{ MPa.}$$

The tensile stress is

$$S_t = \frac{P}{A},$$

where

$$A \approx (L-D)h \approx (7.6 - 3.18)(1.9) \approx 8.5 \text{ cm}^2.$$

thus,

$$S_t = \frac{26.95}{8.5} = 31.7 \text{ MPa.}$$

Bearing strength is 160% of the yield strength⁽¹⁸⁾ or 397 MPa. Shear strength is 60% of the yield strength⁽¹⁸⁾ or 149 MPa while the yield strength is 248 MPa. The computed stresses do not approach any of these values.

4.2.6.1.3 Tiedown Lug Problem (Case 2) - The second case examined involves tiedown lugs used improperly as lifting lugs (Fig. 13). In this case, the shackle's inside depth is insufficient (the shackle contacts the lug at the point shown in the figure). The load angle of 53.5° is based on the use of a 91-cm-(36-in.) long sling.

The resolution of forces is done in the same manner as the previous case:

$$P_v = 23.34 \text{ kN,}$$

$$P_H = \frac{P_v}{\tan 53.5^\circ} = 17.27 \text{ kN,}$$

$$P = \frac{P_v}{\sin 53.5^\circ} = 29.04 \text{ kN,}$$

$$P_1 = \frac{P_v}{\sin 78^\circ},$$

$$P_1 = 23.86 \text{ kN, and}$$

$$P_2 = \frac{P_v}{\tan 78^\circ} = 4.96 \text{ kN.}$$

The summation of moments about the centroid of the weld area gives

$$M = 23.34 \times 4.0 - 4.96 \times 1.2 + 17.27 \times 6.2 = 194.5 \text{ kN} \cdot \text{cm.}$$

Fig. 13. Model of tiedown lug used improperly as lifting lug (case 2).

The shear stress is

$$S_s = \frac{0.707P_v}{f(L+h)} = \frac{0.707 \times 23.34}{0.64(16.3 + 1.9)} = 14.16 \text{ MPa.}$$

The two normal stresses are

$$S_{nh} = \frac{0.707(P_2 + P_H)}{f(L+h)} = \frac{0.707(4.96 + 17.27)}{0.64(18.2)}$$

$$= 13.49 \text{ MPa (Compression),}$$

$$S_{nb} = \frac{4.24M}{f[L^2 + 3h(L+f)]} = \frac{4.24 \times 194.5}{0.64[16.3^2 + 3 \times 1.9(18.2)]}$$

$$= 34.88 \text{ MPa (Tension).}$$

The maximum shear stress is

$$S_{MAX} = \left[\left(\frac{34.88 - 13.49}{2} \right)^2 + 14.16^2 \right]^{1/2} = 17.7 \text{ MPa.}$$

This stress is far below the yield strength. However, the shackle-tensile load increases from 26.95 kN to 29.04 kN but is still less than 38.25 kN, the shackle working load. Comparison of the geometries of the two cases shows the lug is also safe in bearing, shear and tension.

4.2.6.2 Container Cover

The container cover top plate is shown in Fig. 14. It is possible to lift the entire cask by means of the center lifting lug shown in Fig. 15. The stresses in the lug under this condition are discussed below.

4.2.6.2.1 Tensile Stress - The tensile stress at the net cross section is

$$S_t = \frac{3 \times 3175 \times 9.8}{(7.62 - 3.81)(1.90)} = 128.9 \text{ MPa.}$$

The material yield strength is 248 MPa.

Fig. 14. Container cover top plate.

Fig. 15. Container cover lifting lug.

4.2.6.2.2 Shear Stress - The shear stress at the base of the lug is

$$S_s = \frac{P}{A} = \frac{3 \times 3175 \times 9.8}{2 \times 1.90 \times 3.81} = 64.5 \text{ MPa.}$$

The shear stress at the hole for a 2.54-cm-diam pin is

$$S_s = \frac{3 \times 3175 \times 9.8}{2 \times 1.9 \times 1.9} = 129.3 \text{ MPa.}$$

The material shear strength is 149 MPa.

4.2.6.2.3 Bearing Stress - The smallest shackle that can safely be used (38.25 kN working load) has a 2.54-cm-diam pin. The bearing stress is

$$S_b = \frac{P}{dh} = \frac{3 \times 3175 \times 9.8}{2.54 \times 1.9} = 193.4 \text{ MPa.}$$

The material bearing strength is 397 MPa.

4.2.6.2.4 Maximum Plate Stress - Figure 14 shows the flat plate of the cover with a load applied to the lifting lug. The maximum plate stress according to Ref. 17 occurs at the radius b:

$$S = \frac{3P}{2\pi t^2} \frac{2a^2(m+1) \ln \frac{a}{b} + a^2(m-1) - b^2(m-1)}{a^2(m+1) + b^2(m-1)},$$

where

S = stress = Pa,

P = applied load = $3 \times 3175 \times 9.8 = 93.3 \text{ kN}$,

t = plate thickness = 2.54 cm,

m = reciprocal of Poisson's ratio = $\frac{1}{0.26} = 3.85$,

a = outer radius = 36.5 cm, and

b = inner radius = 19.0 cm.

Thus,

$$S = \frac{3 \times 93.3}{2 \times \pi \times (2.54)^2}$$

$$\frac{2(36.5)^2(3.85+1) \ln \frac{36.5}{19.0} + (36.5)^2(3.85-1) - (19.0)^2(3.85-1)}{(36.5)^2(3.85+1) + (19.0)^2(3.85-1)}$$

= 103 MPa (41% of yield strength).

4.2.6.2.5 Cross-Sectional Tensile Stress — The tensile stress on the smallest cross-sectional area of each bolt attaching the cover to the body is

$$S_t = \frac{P}{NA} = \frac{3 \times 3175 \times 9.8}{12(0.785 \times 2.04^2)} = 23.8 \text{ MPa (11% of yield strength).}$$

4.2.6.3 Gamma Shield

The depleted uranium is machined to fit inside a stainless steel shell, which has two plates welded to it for lifting purposes (Fig. 16). The load-per-plate is

Fig. 16. Gamma shield shell.

$$P = \frac{3mg}{2} = \frac{3 \times 862 \times 9.8}{2} = 12.7 \text{ kN.}$$

The shear stress in the weld is

$$S_s = \frac{0.707P}{fl},$$

where

f = fillet size = 0.64 cm,

L = length = 22.9 cm, and

P = load per plate = 12.7 kN.

Thus,

$$S_s = \frac{0.707 \times 12.7}{0.64 \times 22.9} = 6.1 \text{ MPa.}$$

The shear strength for the stainless steel weld metal is 60% of the yield strength or 124 MPa. The lifting-lug holes are sized for steel shackles with a 2.86-cm-diam pin. However, the smallest shackle that can safely lift the gamma shield has a 1.59-cm-diam pin.

The bearing stress is

$$S_b = \frac{P}{dh}, \text{ and}$$

$$S_b = \frac{12.7}{1.59 \times 0.95} = 84.1 \text{ MPa.}$$

The shear stress is

$$S_s = \frac{P}{2he},$$

where

e = distance to edge = 4.1 - 1.59 = 2.51 cm.

Thus,

$$S_s = \frac{12.7}{2 \times 0.95 \times 2.51} = 26.6 \text{ MPa.}$$

The tensile stress is

$$S_t = \frac{P}{A},$$

where

$$A = (L-D)h = (8.26 - 3.18)0.95 = 4.83.$$

Thus,

$$S_t = \frac{12.7}{4.83} = 26.3 \text{ MPa.}$$

Bearing strength is 160% of the yield strength⁽¹⁹⁾ or 331 MPa. Shear strength is 60% of the yield strength⁽¹⁹⁾ or 124 MPa while yield strength is 207 MPa. The computed stresses do not approach any of these values.

The heavier gamma-shield cap is provided with a U-bolt of 1.27-cm-diam 304 ss rod welded to the cap-top plate. The cap has a mass of 125 kg. The tensile stress in the weld is

$$S_t = \frac{1.414P}{\pi Df},$$

where

$$P = \text{load} = \frac{3 \times 125 \times 9.8}{2} = 1838 \text{ N,}$$

$$D = \text{diameter} = 1.27 \text{ cm, and}$$

$$f = \text{fillet size} = 0.64 \text{ cm.}$$

Thus,

$$S_t = \frac{1.414 \times 1838}{\pi \times 1.27 \times .64} = 10.2 \text{ MPa.}$$

This stress is 5% of the yield strength of 207 MPa.

The maximum stress in the U-bolt is compressive and is located at the bend inside diameter.⁽²³⁾

$$S_c = \frac{4.2P}{A} = \frac{4.2 \times 3 \times 125 \times 9.8}{0.785 \times 1.27^2} = 122 \text{ MPa (59% of yield strength).}$$

4.2.7 Structural Tiedown Devices

Without generating stress in any material in excess of their yield strength, the tiedown devices, which are a structural part of the packaging, shall withstand the following:

- A static force applied to the center of gravity of the package having a vertical component of two times the weight of the package.

- A horizontal component, in the direction the vehicle travels, ten times the weight of the package.
- A horizontal component, in the transverse direction, five times the weight of the package.

4.2.7.1 Vertical Load

The model is shown in Fig. 17. Summing forces in a vertical direction,

$$\Sigma F_Y = 0,$$

$$0 = 4 F_Y - 2W + W, \text{ and}$$

$$F_Y = 0.25W.$$

4.2.7.2 Horizontal Loads

The model is shown in Fig. 18. Tension members secure the cask to the transporting vehicle. Chocks are assumed to reinforce the frictional force at the junction of the base and the bed of the vehicle while initial tension in the tiedowns is neglected. With the package assumed to be shipped on the centerline of a 2.44-m (8-ft)-wide truck bed (LLI's typical practice), the angle between the tiedowns and the bed is 46.5°.

Fig. 17. Vertical load on cask.

Fig. 18. Horizontal loads on cask.

The vehicle is assumed to accelerate to the left (Fig. 18) and the package tends to rotate clockwise about the right corner of the base. However, the assumption of a completely rigid base is somewhat unrealistic and would result in an underestimate of the actual tiedown tension. To be conservative, it will be assumed that the edge of the cask shell is the center of rotation.

4.2.7.2.1 Longitudinal Direction - Tension forces at C and D are negligible and are ignored. An analysis of the forces is

$$F_{BX} = F_{BXZ} \sin 45^\circ,$$

$$F_{BY} = F_{BXZ} \tan 46.5^\circ,$$

$$F_{AX} = F_{BX},$$

$$F_{BZ} = F_{AZ},$$

$$F_{BX} = F_{BZ},$$

$$\Sigma F_X = 0, \text{ and}$$

$$0 = 2 F_{BX} + \mu(W + 2F_{BY}) + R_X - 10W,$$

where

$$\begin{aligned} \mu &= \text{coefficient of sliding friction of steel on wet wood} \\ &= 0.16 \text{ (Ref. 23),} \end{aligned}$$

$$\Sigma F_Y = 0,$$

$$0 = 2 F_{BY} - R_Y,$$

$$\Sigma M_O = 0, \text{ and}$$

$$0 = 71.8(10W) - 122.2(2F_{BX}) - 92.1 (2F_{BY}).$$

Solution:

$$F_{BZ} = 1.38W = F_{AZ},$$

$$F_{BX} = 1.38W = F_{AX},$$

$$F_{BY} = 2.06W = F_{AY},$$

$$R_X = 6.41W,$$

$$R_Y = 4.12W.$$

4.2.7.2.2 Transverse Direction -- The model for transverse loads can be obtained from the previous case by considering the load applied 90° to the 10W case.

$$F_{BZ} = F_{CZ} = 0.69W,$$

$$F_{BX} = F_{CX} = 0.69W,$$

$$F_{BY} = F_{CY} = 1.03W,$$

$$R_X = 3.21W, \text{ and}$$

$$R_Y = 2.06W.$$

4.2.7.3 Tiedown Stress Calculations

Table 6 shows the sum of the forces for various points and loads. The most heavily loaded point is B. The stress calculations for the lug are similar to those done in Section 4.2.6.1.2.

4.2.7.3.1 Weld Stress -- The weld stresses due to the vertical force are

$$S_s = \frac{0.707F_{BY}}{f(L+h)},$$

where

$$F_{BY} = 3.34W,$$

Table 6. Summation of tiedown loads.

Location	Direction			
	X	Y	Z	XZ
A	1.38W	2.31W	1.38W	1.95W
B	2.07W	3.34W	2.07W	2.93W
C	0.69W	1.28W	0.69W	0.98W
D	0	0.25W	0	0

thus,

$$S_s = \frac{0.707 \times 3.34 \times 3175 \times 9.8}{0.64(16.3 + 1.9)} = 63.1 \text{ MPa};$$

$$S_{nv} = \frac{4.24 \text{ M}}{f[L^2 + 3h(L + f)]},$$

where

$$M = 3.34 \times 3175 \times 9.8 \times 5.8,$$

thus,

$$S_{nv} = 110.2 \text{ MPa}.$$

The horizontal force at the pinhole location is resolved into a tensile force at the centroid of the weld area and a moment. The moment arm is the distance from the weld centroid-to-hole location (or 4.4 cm).

$$S_{nh} = \frac{4.24 \text{ M}}{f[L^2 + 3h(L + f)]},$$

where

$$M = 2.93 \times 3175 \times 9.8 \times 4.4.$$

Thus,

$$S_{nh} = 73.4 \text{ MPa}.$$

The normal stress due to a force at the weld centroid is

$$S_n = \frac{0.707F_{BXZ}}{f(L + h)} = \frac{0.707 \times 2.93 \times 3175 \times 9.8}{0.64(10.77 + 1.59)} = 55.4 \text{ MPa}.$$

The maximum weld stress is determined from the sum of the normal stresses and the shear stress:

$$\Sigma S = 239.0 \text{ MPa}.$$

The maximum shear stress:

$$S_{MAX} = \left[\left(\frac{239}{2} \right)^2 + 63.1^2 \right]^{1/2} = 135.1 \text{ MPa}.$$

Since this value is less than the failure criterion (one-half yield strength = 172.4 MPa), the welds will not fail.

The stresses in the lug material are as follows.

4.2.7.3.2 Bearing Stress -

$$S_b = \frac{F_{BXYZ}}{dh},$$

where

$$F_{BXYZ} = 3175 \times 9.8 \times (3.34^2 + 2.93^2)^{1/2} = 138.2 \text{ kN.}$$

Thus,

$$S_b = \frac{138.2}{3.49 \times 1.9} = 208.3 \text{ MPa.}$$

Bearing strength is 397.0 MPa.

4.2.7.3.3 Shear Stress -

$$S_s = \frac{F_{BXYZ}}{2 h e},$$

where

$$e = 5.08 - 1.9 = 3.18.$$

Thus,

$$S_s = \frac{138.2}{2 \times 1.9 \times 3.18} = 114.4 \text{ MPa.}$$

Shear strength is 149 MPa.

4.2.7.3.4 Tensile Stress -

$$S_t = \frac{F_{BXYZ}}{A} = \frac{138.2}{7.35 \times 1.9} = 99.0 \text{ MPa.}$$

Yield strength is 248 MPa.

4.2.8 Truck Transport Tiedown

The criteria used for the tiedown system, other than the cask structural parts, are found in Ref. 24, Section 4.2:

"Strength: All parts of the tiedown system, including attachment of the package to the cradle or base, and attachments to the vehicle,

shall be so designed and fabricated that static stresses would not exceed 75% of the yield strength if the truck bed or trailer bed were subjected to a sustained acceleration of 2 g forward or backward, 1 g sidewise, 3 g vertically up to 2 g vertically down. Because of the gravity field, the net separating force between container and vehicle for downward acceleration is one times the container weight. For bolts, lashing rings, and tiedown lugs, the calculated stress under such conditions shall not exceed 75% of the yield strength. Required bolt torque shall be specified and shall correspond to an initial stress of no more than 75% of yield. For wire rope, chain, turnbuckles, and shackles, the load shall not exceed that given in Tables 1, 2, 3 and 4.

Stress analysis may include frictional forces and shall take account of the relative stiffness or elasticity of tiedowns which can affect the load distribution among the members."

The referenced tables can be found as Tables D-1 to D-4 in Appendix D of this report.

4.2.8.1 Longitudinal Acceleration of 2 g

The vehicle is assumed to accelerate to the left (Fig. 18). No credit is taken for chocking at the cask base.

$$F_{BX} = F_{BXZ} \sin 45^\circ,$$

$$F_{BY} = F_{BXZ} \tan 46.5^\circ,$$

$$F_{AX} = F_{BX},$$

$$F_{BZ} = F_{AZ},$$

$$F_{BX} = F_{BZ},$$

$$\Sigma F_X = 0, \text{ and}$$

$$0 = 2 F_{BX} + \mu(W + 2F_{BY}) \approx 2W.$$

As before, $\mu = 0.16$, and

$$F_{BX} = F_{BY} \frac{\sin 45^\circ}{\tan 46.5^\circ}.$$

The total tiedown tension is

$$F = \left(F_{BX}^2 + F_{BY}^2 + F_{BZ}^2 \right)^{0.5}.$$

The solution is

$$F_{BZ} = 0.743W,$$

$$F_{BY} = 1.107W,$$

$$F_{BX} = 0.743W, \text{ and}$$

$$F = 1.526W.$$

4.2.8.2 Lateral Acceleration of 1 g

$$\Sigma F_X = 0,$$

$$0 = 2 F_{BX} + \mu (W + 2F_{BY}) - W.$$

Solution:

$$F_{BZ} = 0.339W,$$

$$F_{BY} = 0.505W,$$

$$F_{BX} = 0.339W, \text{ and}$$

$$F = 0.696W.$$

4.2.8.3 Vertical Acceleration of 1 g

$$\Sigma F_Y = 0,$$

$$0 = W - 4F_{BY},$$

$$F_{BY} = 0.25W, \text{ and}$$

$$F = \frac{F_{BY}}{\sin 46.5^\circ} = 0.345W.$$

The design force is the maximum value from above.

$$\text{Tiedown tension} = 1.526W,$$

$$= 1.526 \times 3175 \times 9.8 = 47.5 \text{ kN (10,675 lb)}.$$

In Tables D-1 to D-3 of Appendix D, we see that the tiedowns may be 0.50-in. chain or 0.88-in.-diam wire rope, with at least 1.125-in.-diam pin shackles.

4.3 SHIELDING ANALYSIS

The shielding properties of the spodumene-gypsum composite were determined from computer calculations using the ANISN shielding code⁽²⁵⁾. The calculations, based on the elemental composition of the shielding material, give the combined dose rate due to neutron and capture gamma exposure per source neutron. Figure 19 shows the results for cases of total wall thicknesses of 7.0 and 30.0 cm. Also included is a curve for the shielding material compacted to 125% normal density. The shapes of the curves moderately depend on the total thickness of shielding in each case. This explains the anomaly of the higher density shielding resulting in a higher dose rate at some thicknesses.

Fig. 19. Dose attenuation for gypsum-spodumene shield.

The use of the gamma shield with neutron-emitting isotopes enhances the shielding effect of the gypsum-spodumene composite. The enhancement is caused by inelastic scattering of fission-energy neutrons by the uranium before the neutrons enter the composite. The inelastic scattering softens the fission-neutron energy spectrum and facilitates moderation and capture in the gypsum-spodumene composite. Because this augmentation is not included in the data from which Fig. 19 was plotted, the results therein are conservative.

The gamma shield is an excellent gamma-ray attenuator. Credit is taken for the attenuation produced by the depleted uranium in Fig. 20. The data points were calculated by assuming the radioisotope to be a point source located in the center of the gamma-shield cavity. The gamma-attenuating effect of the gypsum-spodumene composite is ignored, which again is a conservative approach. The supporting calculations for the gamma shield are found in Section 4.3.1.

Fig. 20. Dose rate per Curie as a function of Gamma ray energy at surface of cask with Gamma shield installed.

4.3.1 Normal Transport Conditions

The calculation for the dose rate under normal transport conditions assumes the source is on the vertical centerline of the cask. Since some of the transported isotopes will be weak gamma emitters, the analysis will be made in two separate cases. Ignored is the shielding provided by the special-form capsule, the cask inner and outer walls, and any self-shielding.

4.3.1.1 Gamma Shield Not Installed

The outer container is used without the gamma shield for a neutron-emitting isotope.

$$D = S_n \times K \times \frac{1}{4\pi R^2},$$

where

D = dose rate (mrem/h),

S_n = source strength (neutrons/s),

K = value of ordinate for case in question from Fig. 19

$[(\text{mrem/h})/(\text{n/s})]\text{cm}^2$, and

R = radius (cm).

The two dose-rate limits are 200 mrem/h at the cask surface and 10 mrem/h at 91.4 cm from cask surface. The maximum allowable source strength (S) for the first limit is

$$S_n = \frac{(4\pi R^2)(D)}{K} = \frac{(4\pi \times 52.07^2)(200)}{3.95 \times 10^{-3}} = 1.7 \times 10^9 \text{ n/s.}$$

For the second limit of 10 mrem/h,

$$S_n = \frac{(4\pi \times 143.47^2)(10)}{3.95 \times 10^{-3}} = 6.5 \times 10^8 \text{ n/s.}$$

The calculated limit for the source strength is the smaller of the above two values or

$$S_n \leq 6.5 \times 10^8 \text{ n/s.}$$

An indication of the amount of radioisotope involved can be obtained from the data in Section 3.3. and Table 3. For example, 1.0 g of ^{244}Cm produces

1.17×10^7 n/s. By ignoring other considerations for the moment, the maximum amount of ^{244}Cm that could be shipped is

$$G = \frac{S_n}{\phi} = \frac{6.5 \times 10^8}{1.17 \times 10^7} = 56 \text{ g.}$$

Disregarded in the above calculations is the effect of gamma emission by the radioisotope. It would result in an additional dose rate that must be included in the analysis. If the isotope in question is a strong gamma emitter, the situation is covered below, where the gamma shield is used.

4.3.1.2 Gamma Shield Installed

The outer container and gamma shield are used with a neutron- and gamma-emitting isotope.

$$D = \frac{BS\psi\text{Ce}^{-\mu x}}{4\pi R^2},$$

D = gamma dose rate (mrem/h),

B = dose build up factor, Berger Form,

S = source strength (Ci),

μ = linear absorption coefficient (cm^{-1}),

x = thickness of uranium = 11.14 cm,

R = radius of cask = 52.07 cm,

C = conversion factor = 3.7×10^{10} dis/s-Ci, and

ψ = dose conversion factor $\frac{\text{mrem}}{\text{h}} / \frac{\text{photons}}{\text{cm}^2 - \text{s}}$.

The results of the calculation for energies from 1 MeV to 4 MeV are stated in Table 7 and plotted in Fig. 20. The dose rate for any given amount of radioisotope would be the sum due to neutron and gamma exposure. Inspection of the decay schemes in Ref. 6 would indicate whether the primary gamma source need be considered.

An example of the use of Table 3 for estimating the maximum amount of radioisotope that can be shipped is given in Appendix B.

Table 7. Gamma shield data.

Gamma energy, E (MeV)	Linear absorption coefficient, μ (cm ⁻¹)	Dose build-up factor, ²⁶ B	Conversion factor, ψ $\left(\frac{\text{mrem}}{\text{h}} / \frac{\text{photons}}{\text{cm}^{-2} - \text{s}} \right)$	Dose rate/Ci, D/S $\left(\frac{\text{mrem}}{\text{h} - \text{Ci}} \right)$
1.0	1.457	3.6	1.93×10^{-3}	0.00067
1.5	1.051	3.8	2.65×10^{-3}	0.090
2.0	0.903	4.0	3.25×10^{-3}	0.604
3.0	0.813	4.1	4.3×10^{-3}	2.225
4.0	0.819	4.2	5.3×10^{-3}	2.646

4.3.2 Hypothetical Accident Conditions

4.3.2.1 Effect of Impact

As shown in section 4.2.5.1.1.2, the worst damage to the shielding during a drop is a reduction in thickness of 7.2 cm. Examination of Fig. 19 indicates the K factor would increase from 3.95×10^{-3} at 30 cm, to 8.6×10^{-3} at 22.8 cm. Note that this loss of shielding only affects the neutron shielding; the gamma shield is unaffected.

All other conditions being equal, the dose rate will have increased to

$$D = \frac{8.6 \times 10^{-3}}{3.95 \times 10^{-3}} D_o = 2.2 D_o ,$$

where D_o is the original existing dose rate.

If the 91.4-cm dose rate was at the maximum value of 10 mrem/h before the hypothetical accident, its new value would be 22 mrem/h. The cask surface dose rate in the area of the impact would be

$$D = \frac{S}{4\pi R^2} \times K = \frac{(6.5 \times 10^8)(8.6 \times 10^{-3})}{4\pi(44.87)^2} = 221 \text{ mrem/h.}$$

4.3.2.2 Effect of Thermal Test

Section 4.1.2 and Fig. 7 denote the effect of a 30-min exposure to a temperature of 1075 K and a subsequent 3-h cool-down. The total thickness

of shielding undergoing dehydration would be 7.0 cm. If we assume the effectiveness of the dehydrated shielding composite to be nil, the ordinate for a 23.0-cm-thick shield is found (Fig. 19) to be

$$K = 8.5 \times 10^{-3} \text{ [(mrem/h)/(n/s)] cm}^2.$$

This value is slightly less than the worst impact case; therefore, the dose rate increase will be less for the hypothetical thermal test than the impact effect. The thermal-accident dose rates are conservatively high. The dehydrated shielding composite would still have a density of 80% of its original value (Section 2.2.2.1). The 20% loss would not significantly affect its gamma-attenuation ability. The loss of water of hydration would primarily affect the ability of the shield composite to slow down neutrons to increase their probability of capture. The large proportion of low-atomic-mass elements remaining would still be somewhat effective in moderating the fission neutrons.

5.0 Quality Assurance Program

5.1 SCOPE

This section sets forth the program for quality assurance. It covers the methods of control and verification to be used during the design, procurement, fabrication, operation, maintenance, and modification phases.

5.2 RESPONSIBILITIES

The LLL Mechanical Engineering Department shall design and analyze the cask; prepare the drawings, specifications, and Safety Analysis Report; and establish inspection and acceptance criteria. Verification and audit of work shall be done within the department.

The fabricator shall assure that all specifications are met and perform and document the results of the inspection requirements specified. He shall obtain and provide the necessary material certifications and welding qualifications.

The LLL Materials Management Group shall implement the reuse inspection program, perform reuse inspections, and retain appropriate files to document such inspections.

Cask users shall provide the necessary information pertinent to the materials to be shipped. They shall also perform the appropriate packaging procedures.

5.3 FABRICATION AND ACCEPTANCE INSPECTION

The cask was completed before the existence of requirements calling for detailed written fabrication and inspection procedures. Section 2.2 and Table 1 list the materials used and their specifications.

Section 2.3 notes the fabrication methods. The cask was inspected and leak-checked prior to filling with the gypsum-spodumene, which was witnessed by LLL personnel. The cask also received a final inspection before painting.

There are no plans to fabricate any additional casks of this design. However, if there is a need for additional casks, fabrication and inspection procedures will be prepared and submitted for approval.

5.4 REUSE INSPECTION

The cask shall be examined before each use according to the reuse inspection checklist in Appendix E. Major defects that are found must be corrected prior to cask reuse. A major defect is one that judgment and experience indicate is likely to result in hazardous or unsafe conditions. It could lead to failure or reduce materially the usability of the cask.

5.5 INSPECTION RECORDS

The LLL Materials Management Group shall maintain the files documenting the receiving acceptance, reuse inspection (Appendix E), and operating check lists (Section 6.0).

6.0 Loading and Handling Procedure

The checklist in Appendix F shall be completed by the shipper and carrier whenever the cask is to be used.

7.0 Compliance With Chapter 0529

These paragraphs discuss LLL's complete compliance with the pertinent sections of Chapter 0529.

7.1 PART I

15.b. The special-form capsule is required to meet certain minimum size requirements. Examination of Fig. 2 indicates fulfillment of requirements.

15.b.1. Tests required by Annex 4 of Chapter 0529 (Appendix A) are to ascertain that the capsule will retain its contents.

- Free Drop. See Section 4.2.5.1.2.
- Percussion. See Section 4.2.5.4.
- Heating. See Section 4.1.3.
- Immersion. Immersion for 24 h in water at room temperature with a pH6 to pH8 will have no effect on a type 347 stainless steel capsule. There is no chemical, galvanic, or other reaction. Water will not enter the capsule.

7.2 PART II

A.1. Radioactive materials will be in a dry solid form. They will be placed in metal containers with which they do not react chemically and/or galvanically. There are no reactions between the packaging components of steel and gypsum-spodumene.

A.2. The special-form capsule cover, container cover and gamma shield cap are all attached by several bolts. The bolts are torqued to values that prevent them from inadvertently coming loose. In addition, the cask cover bolts are seal wired.

A.3.a. Section 4.2.6.1 analyzes the lifting lugs and pallet base.

A.3.b. Section 4.2.6.2 analyzes the container cover.

A.3.c. Not applicable (no such part).

A.3.d. The lug position is such that it cannot penetrate the cask cavity upon failure and impair the containment. The solid shielding composite is held in place by numerous hooks, a wire mesh screen, and the steel shell.

- A.4.a. See Section 4.2.7 of this LLL report.
- A.4.b. No structural part other than tiedown lug could be reasonably used.
- A.4.c. Same comments as A.3.d.
- B.1. See Section 4.2.1 of this LLL report.
- B.2. See Section 4.2.3 of this LLL report.
- C.1. See Section 3.5 of this LLL report. The limits set by LLL on the amount to be shipped prevent criticality even for a water-reflected metal/water mixture.
- C.2. Not applicable. Liquids shall not be shipped.
- C.3. Not applicable.
- D.1.a. See E discussion.
- D.1.b. See F discussion.
- D.2. No controls by the shipper are anticipated other than those required for tiedown.
- D.3. Not applicable. No deviations from normal conditions of transport or hypothetical accident conditions are required.
- E.1.a,b,c. The effects of the normal conditions of transport (specified in Annex 1 of Appendix A) on the package follow:
- Heat. The maximum surface temperature and maximum internal temperature of the package are 329 K and 457 K when exposed to direct sunlight and an ambient peak temperature of 328 K in still air. Section 4.1.1 has the analysis. These temperatures will have no effect on the package, its contents, or the seals.
 - Cold. The ambient temperature of 233 K will have no effect on the package, its contents, or the seals. Differences in thermal contraction and/or expansion will not damage the package.
 - Pressure. Atmospheric pressure of 0.5 times standard atmospheric pressure will not cause the release of radioactive gases from special-form capsule or structural damage. The analysis in Section 4.2.4 indicates the cask can withstand better than 3.5 atm of pressure differential. The special-form capsule is routinely leak-tested at a pressure differential of 1 atm. It was also tested at an internal pressure of 276 kPa at 1089 K.

- Vibration. Normal transport vibration will not effect the package. The lid fasteners are safety wired and adequately torqued to prevent loosening.
- Water Spray. The package will not experience any detrimental effects from a 30 min water spray on all exposed surfaces.
- Free Drop. A free drop of 1.22 m will not substantially reduce the package effectiveness. Section 4.2.5.1.1 analyzes a fall of 9.14 m.
- Corner Drop. Not required.
- Penetration. Test indicated no evidence of damage.
- Compression. Analysis in Section 4.2.2 shows the cask easily meets compression requirements.

E.1.d. Not applicable (no coolant).

E.1.e. Not applicable (no cooling system employed).

E.2.a. Quantities of fissile material are limited to those described in Section 3.5 and Table 4.

E.2.b. The geometric form of the package is not substantially altered under normal conditions of transport.

E.2.c. The special-form capsule, when assembled, is leak-tight.

E.2.d. Nuclear safety of the package is not geometry-dependent.

E.3. The special-form capsule will not vent under any of the following conditions specified in Annex 1 of Appendix A.

- Heat. Capsule operating temperature range is 6 K to 1089 K.
- Cold. See above comment.
- Pressure. See comments under E.1.a,b,c of this report.
- Vibration. Capsule fasteners are adequately torqued to prevent loosening.
- Water Spray. No effect on capsule.
- Free Drop. Capsule was subjected to Annex 4 (Appendix A) test (Section 4.2.5.4).
- Penetration. Capsule was subjected to Annex 4 (Appendix A) percussion test (Section 4.2.5.4).
- Compression. No effect on capsule.

F.1.a. The reduction of shielding when subjected to Annex 2 (Appendix A) conditions is not sufficient to increase the radiation dose rate to more than 1000 mrem/h at 91.4 cm from the package.

- Free Drop. See Section 4.2.5.1.1.
- Puncture. See Section 4.2.5.3.
- Thermal. See Section 4.3.2.
- Water Immersion. No effect on shielding.

F.1.b. No radioactive material will be released from package. Same comments in F.1.a of this report apply.

F.2. See Sections E.2.a and E.2.d of this report.

H. Preceding comments in C, E, and F indicate it is impossible to achieve criticality under any of the specified conditions.

7.3 PART III

Section 5.0 describes the Quality Assurance Program.

7.4 PART IV

Section 6.0 describes the Operating Procedures.

Acknowledgment

This report is based upon work performed earlier by C. L. Hanson, M. S. Coops and J. D. Quigg, and published as UCRL-51195, "Analysis of the Neutron Shipping Container, Model 4T" (April, 1972).

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APPENDIX A
CHAPTER 0529 FROM HEALTH AND SAFETY SECTION
OF ERDA MANUAL

**U.S. ATOMIC ENERGY COMMISSION
AEC MANUAL**

Volume: 0000 General Administration
Part : 0500 Health and Safety

AEC 0529-01
OS

**Chapter 0529 SAFETY STANDARDS FOR THE PACKAGING OF FISSILE
AND OTHER RADIOACTIVE MATERIALS**

0529-01 POLICY

Fissile and other radioactive material shall be packaged and prepared for shipment in a manner that provides assurance of protection of the public health and safety during the transportation of such materials.

0529-02 OBJECTIVE

To establish safety standards for the packaging of fissile and other radioactive materials for shipment by AEC or by contractors not subject to 10 CFR 71.

0529-03 RESPONSIBILITIES AND AUTHORITIES

031 The Director, Division of Operational Safety:

- a. determines the need for, and develops new and revised safety standards to be applied in the preparation of fissile and other radioactive materials for transportation.
- b. provides a central point of coordination with the Director of Regulation for developing and revising health and safety codes pertaining to safety in the transportation of fissile and other radioactive materials which are intended for use in the AEC programs or by other federal agencies.
- c. conducts periodic appraisals to determine the adequacy of the implementation of this chapter.
- d. renders interpretations of this chapter.

032 The Director, Division of Waste Management and Transportation, in addition to the responsibilities and authorities assigned in 033, below:

- a. administers the program for design review and issuance of AEC Certificates of Compliance as provided in 056, below, AECM 5201, and 49 CFR 172.39-173.396.
- b. assists field offices in expediting essential shipments consigned to or by AEC and in securing waivers or exemptions from Federal Transportation Regulations.
- c. prepares guidance criteria and procedures for application of package testing and quality assurance standards.
- d. coordinates the total input for the General

Manager in the development and revision of transportation regulations.

033 Heads of Divisions and Offices, Headquarters, excluding Regulatory, provide guidance, instructions, standards, and criteria as described in Chapter 0101, consistent with this chapter, to assure the safe packaging of fissile and other radioactive materials, including:

- a. directing cognizant managers of field offices to require modifications of equipment procedures or practices.
- b. imposing additional requirements for packaging standards.
- c. curtailing or suspending the use of specific packages when necessary.

034 Managers of Field Offices, and the Director, Division of Naval Reactors, consistent with guidance, instructions, standards, and criteria issued pursuant to 033, above:

- a. grant AEC approval for packages which meet the standards contained in appendix 0529 and 10 CFR 71.31-71.40, and which are to be used for the transportation of fissile or other radioactive materials in greater than Type A quantities.
- b. grant AEC approval for shipments made under the National Security Exemption provided to the AEC and DOD under the Transportation of Explosives Act (18 U.S.C. 832c) and in accordance with the requirements of pertinent AEC manual chapters. Packages for such shipments must meet the policy stated in section 01.
- c. grant such alternatives to the requirements set forth in appendix 0529 as will provide equivalent protection to life or property and to the common defense and security; and within 30 days after granting an alternative, provide the Director, Division of Operational Safety, a detailed report of the reasons for granting it. The granting of such alternatives is in no way to be construed as the granting of exemptions or exceptions from or to the Department of Transportation or other regulatory agency requirements.
- d. conduct annual appraisals of contractor operations to assure compliance with the

Approved: June 14, 1973

requirements of this chapter.

- e. require that their contractors carry out, as a minimum, quality assurance programs described in this chapter, and as provided for in appendix 5201.

(NOTE: Contractors shall not be permitted to exercise any of the above authorities.)

035 The Manager, Albuquerque Operations Office, in addition to the responsibilities and authorities assigned in 032, above, shall establish safety standards for packaging and transportation of nuclear weapons and their components in accordance with 055, below.

0529-04 DEFINITION

041 AEC Contractor—for the purposes of this chapter, means a prime contractor or subcontractor of the Atomic Energy Commission who is exempt from the requirements of 10 CFR 71.

0529-05 BASIC REQUIREMENTS

051 Applicability. The provisions set forth in this chapter and its appendix apply to the Headquarters, field offices, and AEC contractors.

052 Coverage. This chapter and its appendix cover policies and procedures for the preparation of fissile and other radioactive materials for shipment outside the boundaries of AEC-controlled sites by for-hire or private carriers, or on public vehicles or aircraft.

053 Federal Regulations. When offered to the carrier, each shipment of radioactive materials shall be in compliance with the applicable safety regulations of the Department of Transportation (DOT) or the U.S. Postal Service, depending on the mode of transportation.

054 Other Regulations

- a. **International Atomic Energy Agency (IAEA) Regulations.** Each shipment of fissile and other radioactive materials consigned to a foreign country must meet the requirements set forth in IAEA Safety Series No. 6, "Regulations for the Safe Transport of Radioactive Materials." Specifically, "Requirements for Packaging and for Delivery of Packages to Transport," must be met to be in compliance with this chapter.
- b. **International Air Transport Association (IATA) Restricted Article Regulations.** Each

shipment of fissile and other radioactive materials consigned to a foreign country must meet the requirements set forth in IATA Restricted Articles Regulations when shipped via commercial aircraft.

- c. **U.S. Air Force AFM-71-4, Packaging and Handling of Dangerous Materials for Transport by Military Aircraft.** Each shipment of fissile and other radioactive materials must meet the requirements set forth in AFM-71-4 when shipped via USAF aircraft.

055 Package Standards for Radioactive Materials in Amounts Greater Than Type A Quantities

- a. Packages of radioactive materials shall be prepared for shipment and transported in accordance with the provisions of this chapter. DOT specification containers for Type B and fissile materials are considered to meet the standards of this chapter, and no specific AEC Certificates of Compliance are required for their use.
- b. Nuclear weapons and their components shall be packaged and transported in accordance with the standards in this chapter or with other standards which provide a degree of safety at least equivalent to that provided by the AEC and DOT regulations. Standards will be developed and documented under 035, above.
- c. Packages shipped under the National Security Exemption of 18 U.S.C. 832.c must be in compliance with the standards in this chapter and must also comply with the provisions of other pertinent AEC manual chapters.
- d. All other packages for fissile and other radioactive materials in amounts greater than Type A quantities shall be designed, constructed, and used in accordance with the standards contained in the attached appendix, and in 10 CFR 71.31-71.40. Materials described in 71.6 are exempt from this requirement.
- e. A quality assurance program must be established and implemented to assure that packages for radioactive materials are fabricated, maintained, and used in accordance with the regulations and approved design features. The program must meet the requirements in appendix 5201.

056 AEC Certificates of Compliance for Packages of Radioactive Materials in Excess of Type A

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Quantities. Upon determination that a package design does in fact meet the requirements of this chapter and in its appendix and the AEC Standards in 10 CFR 71.31-71.40, an AEC Certificate of Compliance will be issued by the AEC to the contractor. Procedures for obtaining and issuing these certificates are set forth in AECM 5201. No certificate is necessary for shipments meeting the requirements of 10 CFR 71.6 or 71.7.

057 AEC as Consignor. Where an AEC office serves as the actual consignor, rather than a contractor, appropriate internal procedures shall be established by the responsible field office manager to assure compliance with the standards contained in this chapter.

058 Waivers and Exemptions. Packages which do not meet the standards in the DOT Hazardous Materials Regulations, and which do not qualify for shipment under the National Security Exemption, may be shipped only under the provisions of a waiver or exemption issued by the DOT, or on public vehicles or aircraft if approved under the provisions of AEC 0529-034c. Applications for a DOT Special Permit for waiver or exemption shall be prepared in accordance with 49 CFR 170.13 and shall be forwarded to the DOT for issuance of a special permit.

059 Existing Packagings. Existing packagings for radioactive materials must meet the standards of this chapter. However, Type B packagings designed and constructed prior to February 15, 1969, which could be subject to loss of shielding resulting from subjecting the packaging to the puncture test followed by the thermal test (49 CFR 173.398(c)(2)), are also approved for continued use. The packaging design must be covered by a DOT Special Permit providing for administrative and operational controls as may be necessary to compensate for the deficiencies in package integrity and to provide equivalent safety in transportation.

0529-06 REFERENCES

- a. AEC Regulation, Title 10, Code of Federal Regulations, Part 71, "Packaging of Radioactive Material for Transport."
- b. DOT Regulations
 1. Title 49, CFR Parts 170-189, and Title

14, CFR Part 103, "Hazardous Materials Regulations."

2. Title 46, CFR Part 146, "Transportation or Storage of Explosives or other Dangerous Articles or Substances, and Combustible Liquids on Board Vessels."
- c. U.S. Postal Service Regulation, Title 39, CFR Parts 124 and 125, "Nonmailable Matter" and "Matter Mailable Under Special Rules."
- d. International Atomic Energy Agency (IAEA) Safety Series No. 6, "Regulations for the Safe Transport of Radioactive Materials," 1967.
- e. AECM 2401, "Physical Protection of Classified Matter and Information."
- f. AECM 2405, "Physical Protection of Unclassified Special Nuclear Material."
- g. AECM 0230, "Records Disposition."
- h. AECM 0530, "Nuclear Criticality Safety."
- i. AECM 0560, "Program to Prevent Accidental or Unauthorized Nuclear Explosive Detonations."
- j. AECM 5201, "Transportation and Traffic Management" (to be reissued under the title "Transportation of Property").
- k. AEC Directory of Radioactive and Fissile Materials Shipping Containers, 1969.
- l. International Air Transport Association (IATA) Restricted Article Regulations.
- m. U.S. Air Force AFM-4, "Packaging and Handling of Dangerous Materials for Transport by Military Aircraft."
- n. Transportation of Explosives Act (18 U.S.C. 832.c).

0529-07 NATIONAL EMERGENCY APPLICATION

During a national emergency, as defined in AECM 0601-04, the provisions of this chapter and appendix will continue in effect.

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PART I

DEFINITIONS AND EXEMPTIONS

A. DEFINITIONS (as used in this appendix)

1. **Carrier** means any person engaged in the transportation of passengers or property, as common, contract, or private carrier, or freight forwarder, as those terms are used in the Interstate Commerce Act, as amended, or the U.S. Postal Service.
2. **Close Reflection by Water** means immediate contact by water of sufficient thickness to reflect a maximum number of neutrons.
3. **Containment Vessel** means the receptacle on which principal reliance is placed to retain the radioactive material during transport.
4. **Fissile Classification** means classification of a package or shipment of fissile materials according to the controls needed to provide nuclear criticality safety during transportation as follows:
 - a. **Fissile Class I.** Packages which may be transported in unlimited numbers and in any arrangement and which require no nuclear criticality safety controls during transportation. For purposes of nuclear criticality safety control, a transport index is not assigned to Fissile Class I packages. However, the external radiation levels may require a transport index number.
 - b. **Fissile Class II.** Packages which may be transported together in any arrangement but in numbers which do not exceed a transport index of 50. For purposes of nuclear criticality safety control, individual packages may have a transport index of not less than 0.1 and not more than 10. However, the external radiation levels may require a higher transport index number but not to exceed 10. Such shipments require no nuclear criticality safety control by the shipper during transportation.
 - c. **Fissile Class III.** Shipments of packages which do not meet the requirements of Fissile Classes I or II and which are controlled in transportation by special arrangements between the shipper and the carrier to provide nuclear criticality safety.
5. **Fissile Materials** means uranium-233, uranium-235, plutonium-238, plutonium-239, and plutonium-241.
6. **Large Quantity** means a quantity of radioactive material, the aggregate radioactivity of which exceeds that specified in the following table for a transport group as defined in 16., below:

Radionuclide Identification	I	II	III	IV	V	VI-VII	Special Form
Radioactivity	20 Curies	20 Curies	200 Curies	200 Curies	5,000 Curies	50,000 Curies	5,000 Curies
7. **Low Specific Activity Material** means any of the following:
 - a. Uranium or thorium ores and physical or chemical concentrates of those ores.
 - b. Unirradiated natural or depleted uranium or unirradiated natural thorium.
 - c. Tritium oxide in aqueous solutions, provided the concentration does not exceed 5.0 millicuries per milliliter.
 - d. Material in which the activity is essentially uniformly distributed and in which the estimated average concentration per gram of contents does not exceed:
 - (1) 0.0001 millicuries of Group I radionuclides; or
 - (2) 0.005 millicuries of Group II radionuclides; or
 - (3) 0.3 millicuries of Group III or IV radionuclides.

NOTE: This may include, but is not

- limited to, materials of low radioactivity concentration, such as building rubble, metal, wood, and fabric scrap, glassware, paper and cardboard, solid or liquid plant waste, sludge, and ashes.
- e. Nonradioactive objects externally contaminated with radioactive material, provided that the radioactive material is not readily dispersible and the surface contamination, when averaged over an area of one square meter, does not exceed 0.0001 millicuries (220,000 disintegrations per minute), per square centimeter of Group I radionuclides or 0.001 millicuries (2,200,000 disintegrations per minute) per square centimeter of other radionuclides.
8. **Maximum Normal Operating Pressure** means the maximum gauge pressure which is expected to develop in the containment vessel under the normal conditions of transport specified in annex 1, below, considered individually.
9. **Moderator** means a material used to reduce by scattering collisions, and without appreciable capture, the kinetic energy of neutrons.
10. **Optimum Interspersed Hydrogenous Moderation** means the occurrence of hydrogenous material between containment vessels to such an extent that the maximum nuclear reactivity results.
11. **Package** means packaging and its radioactive contents.
12. **Packaging** means one or more receptacles and wrappers and their contents, excluding fissile material and other radioactive material, but including absorbent material, spacing structures, thermal insulation, radiation shielding, devices for cooling and for absorbing mechanical shock, external fittings, neutron moderators, nonfissile neutron absorbers, and other supplementary equipment.
13. **Primary Coolant** means a gas, liquid, or solid, or combination of them, in contact with the radioactive material, or if the material is in special form, in contact with its capsule, and used to remove decay heat.
14. **Sample Package** means a package which is fabricated, packed, and closed to fairly represent the proposed package as it would be presented for transport, simulating the material to be transported, as to weight and physical and chemical form.
15. **Special Form** means any of the following physical forms of radioactive material of any transport group.
- The material is in solid form having no dimension less than 0.5 millimeter or at least one dimension greater than 5 millimeters; does not melt, sublime, or ignite in air at a temperature of 1000°F; will not shatter or crumble if subjected to the percussion test described in annex 4, below; and will not be dissolved or converted into a dispersible form in amounts greater than 0.005 percent by weight if immersed in water at 68°F or placed in air at 86°F for one week.
 - The material is securely contained in a capsule having no dimension less than 0.5 millimeter or at least one dimension greater than 5 millimeters which (1) will retain its contents if subjected to the tests prescribed in annex 4, (2) is constructed of materials which do not melt, sublime, or ignite in air at 1475°F, and (3) will not be dissolved or converted into a dispersible form in amounts greater than 0.005 percent by weight if immersed for one week in water at 68°F or in air at 86°F.
16. **Transport Group** means any one of the seven groups in which radionuclides in normal form are classified, according to their toxicity and their relative potential hazard in transport (see annex 3).
- Any radionuclide not specifically listed in one of the groups in annex 3 shall be assigned to one of the groups in accordance with the following table:

Radioactive Half-Life			
Radionuclide	0 to 1000 days	1000 days to 10 ⁹ years	Over 10 ⁶ Years
Atomic Number 1-81	Group III	Group II	Group III
Atomic number 82 and over	Group I	Group I	Group III

- b. For mixtures of radionuclides the following shall apply:

- (1) If the identity and respective activity of each radionuclide are known, the permissible activity of each radionuclide shall be such that the sum of the ratios between the total activity for each group to the permissible activity for each group will not be greater than unity.
- (2) If the groups of the radionuclides are known, but the amount in each group cannot be reasonably determined, the mixture shall be assigned to the most restrictive group present.
- (3) If the identity of all or some of the radionuclides cannot be reasonably determined, each of those unidentified radionuclides shall be considered as belonging to the most restrictive group which cannot be positively excluded.
- (4) Mixtures consisting of a single radioactive decay chain where the radionuclides are in the naturally occurring proportions shall be considered as consisting of a single radionuclide. The group and activity shall be that of the first member present in the chain. Exception: If a radionuclide "x" has a half-life longer than that of the first member and an activity greater than that of any other member, including the first, at any time during transportation, the transport group of the nuclide "x" and the activity of the mixture shall be the maximum activity of that nuclide "x" during transportation.

17. **Transport Index** means the number placed on a package to designate the degree of control to be exercised by the carrier during transportation. The transport index to be assigned to a package of radioactive material shall be determined by either a. or b., below, whichever is larger. The number expressing the transport index shall be rounded up to the next highest tenth; e.g., 1.01 becomes 1.1.
- a. The highest radiation dose rate in millirem per hour at three feet from any accessible external surface of the package.

- b. The transport index for each Fissile Class II package is calculated by dividing the number 50 by the number of such Fissile Class II packages which may be transported together as determined under the limitations of part II.J.1. The calculated number shall be rounded up to the first decimal place.

18. **Type A Quantity and Type B Quantity** mean a quantity of radioactive material, the aggregate radioactivity of which does not exceed that specified in the following table:

Transport Groups (See I.A.16.)	Type A Quantity (in curies)	Type B Quantity (in curies)
I	0.001	20
II	0.05	20
III	3	200
IV	20	200
V	20	5,000
VI and VII	1,000	50,000
Special Forum	20	5,000

B. EXEMPTIONS

A shipper is exempt from all requirements of this appendix to the extent that he delivers to a carrier for transport packages each containing not more than a Type A quantity of radioactive material, as defined in A.18, above, which may include one of the following:

1. Not more than 15 grams of fissile material.
2. Thorium or uranium containing not more than 0.72 percent by weight of fissile material.
3. Uranium compounds, other than metal, (e.g., UF_6 , UF_4 , or uranium oxide in bulk form, not pelleted or fabricated into shapes) or aqueous solutions of uranium, in which the total amount of uranium-233 and plutonium present does not exceed 1.0 percent by weight of the uranium-235 content, and the total fissile content does not exceed 1.00 percent by weight of the total uranium content.
4. Homogeneous hydrogenous solutions or mixtures containing not more than:
 - a. 500 grams of any fissile material, provided the atomic ratio of hydrogen to fissile material is greater than 7600.
 - b. 800 grams of uranium-235, provided that the atomic ratio of hydrogen to fissile material greater than 5,200, and the content of other fissile material is

not more than 1 percent by weight of the total uranium-235 content.

- c. 500 grams of uranium-233 and uranium-235, provided that the atomic ratio of hydrogen to fissile material is greater than 5,200, and the content of plutonium is not more than 1 percent

by weight of the total uranium-233 and uranium-235 content.

5. Less than 350 grams of fissile material, provided that there is not more than 5 grams of fissile material in any cubic foot within the package.

PART II
PACKAGE STANDARDS

A. GENERAL STANDARDS FOR ALL
PACKAGING

1. Packaging shall be of such materials and construction that there will be no significant chemical, galvanic, or other reaction among the packaging components or between the packaging components and the package contents.
2. Packaging shall be equipped with a positive closure which will prevent inadvertent opening.
3. Lifting devices for packagings.
 - a. If there is a system of lifting devices which is a structural part of the packaging, the system shall be capable of supporting three times the weight of the package without generating stress in any material of the packaging in excess of its yield strength.
 - b. If there is a system of lifting devices which is a structural part only of the lid, the system shall be capable of supporting three times the weight of the lid and any attachments without generating stress in any material of the lid in excess of its yield strength.
 - c. If there is a structural part of the packaging which could be employed to lift the package and which does not comply with a., above, the part shall be securely covered or locked during transport in such a manner as to prevent its use for that purpose.
 - d. Each lifting device which is a structural part of the packaging shall be so designed that failure of the device under excessive load would not impair the containment or shielding properties of the packaging.
4. Tiedown devices for packagings
 - a. If there is a system of tiedown devices which is a structural part of the packaging, the system shall be capable of withstanding, without generating stress in any material of the packaging in excess of its yield strength the following: (1) a static force applied to the center of gravity of the package having a vertical component of two times the weight of the package, (2) a horizontal component along the

direction in which the vehicle travels of 10 times the weight of the package, and (3) a horizontal component in the traverse direction of five times the weight of the package.

- b. If there is a structural part of the packaging which could be employed to tie the package down and which does not comply with a., above, the part shall be securely covered or locked during transport in such a manner as to prevent its use for that purpose.
 - c. Each tiedown device which is a structural part of the packaging shall be so designed that failure of the device under excessive load would not impair the ability of the packaging to meet other requirements of this section A.
5. Determination of transport indexes for packagings see part I, A.17. of this appendix.

B. STRUCTURAL STANDARDS FOR TYPE B
AND LARGE QUANTITY PACKAGING

Packaging used to ship a Type B or a large quantity of radioactive material, as defined in part I, A.6 and 18. above, shall be designed and constructed in compliance with the structural standards of this section. Standards different from those specified in this section may be approved by the manager or other designated official if the controls proposed to be exercised by the shipper are demonstrated to be adequate to assure the safety of the shipment.

1. **Load Resistance.** Regarded as a simple beam supported at its end along any major axis, packaging shall be capable of withstanding a static load, normal to and uniformly distributed along its length, equal to 5 times its fully loaded weight, without generating stress in any material of the packaging in excess of its yield strength.
2. **External Pressure.** Packaging shall be adequate to assure that the containment vessel will suffer no loss of contents if subjected to an external pressure of 25 pounds per square-inch gauge.

C. CRITICALITY STANDARDS FOR FISSILE
MATERIAL PACKAGES

1. A package used for the transport of fissile material shall be so designed and constructed

and its contents so limited that it would be subcritical if it is assumed that water leaks into the containment vessel; and

- a. water moderation of the contents occurs to the most reactive credible extent consistent with the chemical and physical form of its contents; and
 - b. the containment vessel is fully reflected on all sides by water.
2. A package used for the transport of fissile material shall be so designed and constructed and its contents so limited that it would be subcritical if it is assumed that any contents of the package which are liquid during normal transport leak out of the containment vessel, and that the fissile material is then:
- a. in the most reactive credible configuration consistent with the chemical and physical form of the material.
 - b. moderated by water outside of the containment vessel to the most reactive credible extent.
 - c. fully reflected on all sides by water.
3. The manager or other designated official may approve exceptions to the requirements of this section where the containment vessel incorporates special design features which would preclude leakage of liquids in spite of any single packaging error, and appropriate measures are taken before each shipment to verify the leak tightness of each containment vessel.

D. EVALUATION OF A SINGLE PACKAGE

1. The effect of the transport environment on the safety of any single package of radioactive material shall be evaluated as follows:
 - a. The ability of a package to withstand conditions likely to occur in normal transport shall be assessed by subjecting a sample package or scale model by test or other assessment to the normal conditions of transport as specified in E., below.
 - b. The effect on a package of conditions likely to occur in an accident shall be assessed by subjecting a sample package or scale model, by test or other assessment, to the hypothetical accident conditions, as specified in F., below.
2. Taking into account controls to be exercised by the shipper, the manager or other

designated official may permit the shipment to be evaluated together with or without the transporting vehicle for the purpose of one or more tests.

3. Normal conditions of transport and hypothetical accident conditions different from those specified in E. and F., below, may be approved by the manager or other designated official if the controls proposed to be exercised by the shipper are demonstrated to be adequate to assure the safety of the shipment.

E. STANDARDS FOR NORMAL CONDITIONS OF TRANSPORT FOR A SINGLE PACKAGE

1. A package used for the shipment of fissile material or more than Type A quantity of radioactive material as defined in part I, A.6 and 18, above, shall be so designed and constructed, and its contents so limited that under the normal conditions of transport specified in annex I below:
 - a. there will be no release of radioactive materials from the containment vessel.
 - b. the effectiveness of the packaging will not be substantially reduced.
 - c. there will be no mixture of gases or vapors in the package which could, through any credible increases of pressure or an explosion, significantly reduce the effectiveness of the package.
 - d. radioactive contamination of the liquid or gaseous primary coolant will not exceed 10^{-7} curies of activity of Group I radionuclides per milliliter, 5×10^{-6} curies of activity of Group II radionuclides per milliliter; 3×10^{-4} curies of activity of Group III and Group IV radionuclides per milliliter.
 - e. there will be no loss of coolant or loss of operation of any mechanical cooling device.
2. A package used for the shipment of fissile material shall be designed and constructed, and its contents so limited, that under normal conditions of transport specified in annex I, considered individually:
 - a. the package will be subcritical.
 - b. the geometric form of the package contents would not be substantially altered.
 - c. there will be no leakage of water into the containment vessel. This requirement need not be met if, in the evaluation of undamaged packages

- under H.I., I.I.a., or J.I., below, it has been assumed that moderation is present to such an extent as to cause maximum reactivity consistent with the chemical and physical form of the material.
- d. there will be no substantial reduction in the effectiveness of the packaging, including:
- (1) reduction by more than 5 percent in the total effective volume of the packaging of which nuclear safety is assessed.
 - (2) reduction by more than 5 percent to the effective spacing on which nuclear safety is assessed between the center of the containment vessel and the outer surface of the packaging.
 - (3) occurrence of any aperture in the outer surface of the packaging large enough to permit the entry of a 4-inch cube.
3. A package used for the shipment of more than Type A quantity of radioactive material as defined in part I.A.6. and 18., above, shall be so designed and constructed, and its contents so limited, that under the normal conditions of transport specified in annex 1, considered individually, the containment vessel would not be vented directly to the atmosphere.

F. STANDARDS FOR HYPOTHETICAL ACCIDENT CONDITIONS FOR A SINGLE PACKAGE

1. A package used for the shipment of more than Type A quantity of radioactive material (see part I.A.6. and 18., above) shall be so designed and constructed, and its contents so limited, that if subjected to the sequence of the hypothetical accident conditions specified in annex 2, it will meet the following conditions:
- a. The reduction of shielding would not be sufficient to increase the external radiation dose rate to more than 1000 millirems per hour at three feet from the external surface of the package.
 - b. No radioactive material would be released from the package except for gases and contaminated coolant containing total radioactivity exceeding neither:
(1) 0.1 percent of the total

radioactivity of the package contents; nor

- (2) 0.01 curie of Group I radionuclides, 0.5 curie of Group II radionuclides, 10 curies of Group III radionuclides, 10 curies of Group IV radionuclides, and 1000 curies of inert gases irrespective of transport group.

A package need not satisfy the requirements of this paragraph if it contains only low specific activity materials, as defined in part I.A.7, above, and is transported on a motor vehicle, railroad car, aircraft, inland watercraft, or hold or deck of a seagoing vessel assigned for the sole use of the shipper.

2. A package, used for the shipment of fissile material shall be so designed and constructed, and its contents so limited, that if subjected to the sequence of the hypothetical accident conditions specified in annex 2, the package would be subcritical. In determining whether this standard is satisfied, it shall be assumed that:
- a. the fissile material is in the most reactive credible configuration consistent with the damaged condition of the package and the chemical and physical form of the contents.
 - b. water moderation occurs to the most reactive credible extent consistent with the damaged condition of the package and the chemical and physical form of the contents.
 - c. there is reflection by water on all sides and as close as is consistent with the damaged condition of the package.

G. EVALUATION OF AN ARRAY OF PACKAGES OF FISSILE MATERIAL

1. The effect of the transport environment on the nuclear criticality safety of an array of packages of fissile material shall be evaluated by subjecting a sample package or a scale model, by test or other assessment, to the hypothetical accident conditions specified in H.I., I., or J., below, for the proposed fissile class, and by assuming that each package in the array is damaged to the same extent as the sample package or scale model. In the case of a Fissile Class III shipment, the manager or other designated official may, taking into account controls to be exercised

by the shipper, permit the shipment to be evaluated as a whole rather than as individual packages, and either with or without the transporting vehicle, for the purpose of one or more tests.

2. In determining whether the standards of H.2., I.1.b., and J.2., below, are satisfied, it shall be assumed that:

- a. the fissile material is in the most reactive credible configuration consistent with the damaged condition of the package, the chemical and physical form of the contents, and controls exercised over the number of packages to be transported together.
- b. water moderation occurs to the most reactive credible extent consistent with the damaged condition of the package and the chemical and physical form of the contents.

H. SPECIFIC STANDARDS FOR A FISSILE CLASS I PACKAGE

A Fissile Class I package shall be so designed and constructed and its contents so limited that:

1. any number of such undamaged packages would be subcritical in any arrangement, and with optimum interspersed hydrogenous moderation unless there is a greater amount of interspersed moderation in the packaging, in which case that greater amount may be considered.
2. two hundred and fifty such packages would be subcritical in any arrangement, if each package were subjected to the sequence of the hypothetical accident conditions specified in annex 2, with close reflection by water on all sides of the array and with optimum interspersed moderation unless there is a greater amount of interspersed moderation in packaging, in which case that greater amount may be considered. The condition of the package shall be assumed to be as described in G., above.

I. SPECIFIC STANDARDS FOR A FISSILE CLASS II PACKAGE

1. A Fissile Class II package shall be so designed and constructed and its contents so limited, and the number of such packages which may be transported together so limited, that:

- a. five times that number of such undamaged packages would be subcritical in any arrangement if closely reflected by water.
- b. twice that number of such packages would be subcritical in any arrangement if each package were subjected to the sequence of hypothetical accident conditions specified in annex 2, with close reflection by water on all sides of the array and the optimum interspersed hydrogenous moderation, unless there is a greater amount of interspersed moderation in the packaging in which case the greater amount may be considered. The condition of the package shall be assumed to be as described in G., above.

2. The transport index for each Fissile Class II package is calculated by dividing the number 50 by the number of such Fissile Class II packages which may be transported together as determined under the limitations in I., above. The calculated number shall be rounded up to the first decimal place.

J. SPECIFIC STANDARDS FOR A FISSILE CLASS III SHIPMENT

A package for a Fissile Class III shipment shall be so designed and constructed and its contents and number of packages so limited, that:

1. the undamaged shipment would be subcritical with an identical shipment in contact with it and with the two shipments closely reflected on all sides by water.
2. the shipment would be subcritical if each package were subjected to the hypothetical accident conditions specified in the sequence specified in annex 2, with close reflection by water on all sides of the array and with the packages in most reactive arrangement and with the most reactive degree of interspersed hydrogenous moderation which would be credible considering the controls to be exercised over the shipment. The condition of the package shall be assumed to be as described in G., above. Hypothetical accident conditions different from those specified in this subparagraph may be approved by the manager or other designated official if the controls proposed to be exercised by the shipper are demonstrated to be adequate to assure the safety of the shipment.

PART III

QUALITY ASSURANCE PROCEDURES FOR THE FABRICATION,
ASSEMBLY, AND TESTING OF OFFSITE SHIPPING CONTAINERS

A. ESTABLISHMENT AND MAINTENANCE OF
PROCEDURES

1. Each field office shall require its contractors to establish and maintain a quality assurance program to:
 - a. assure that the requisite standards of quality are met in the fabrication, assembly, and testing of each package.
 - b. assure that packages in use continue to meet the requisite standards of quality.

B. ELEMENTS OF A QUALITY ASSURANCE
PROGRAM

1. The program shall consist of a formal system of procedural and organizational arrangement which:
 - a. require that specific responsibilities be assigned to designated units (including those of the vendor, the fabricator, and the contractor) for assuring specified quality at all stages of construction.
 - b. designate codes, standards, and specifications for materials, equipment, methods of fabrication, testing, and performance.
 - c. provide for quality control of materials, equipment, and services in instances where these have not already been established by existing standards and

specifications.

- d. provide, as required by AECM 0504, for at least an annual audit of the AEC contractors' programs to assess their effectiveness.
- e. provide that quality assurance records are maintained in an auditable file during the service life of the container.
- f. provide for a method of determining that packagings procured for use from other sources, including AEC contractors and subcontractors or from licensees meet the requirements of AECM 0529.
- g. establish acceptance criteria in terms of measurable characteristics and the effects of appropriate tests prescribed in annexes 1, 2, and 4 and as required in part III.C.
- h. provide for a program of routine maintenance inspection and, where necessary, retesting to assure that reusable containers continue to meet the applicable design standards.
- i. provide for required training, testing, and certification of manufacturing and inspection personnel involved in special processes, such as welding and nondestructive examination, and for the required certification of equipment and procedures used in the performance of special processes.

PART IV

OPERATING PROCEDURES

A. ESTABLISHMENT AND MAINTENANCE OF PROCEDURES

1. The shipper shall establish and maintain:
 - a. operating procedures adequate to assure that the determinations and controls required by this appendix are accomplished.
 - b. regular and periodic inspection procedures adequate to assure that the shipper follows the procedures required by a., above.

B. ASSUMPTIONS AS TO UNKNOWN PROPERTIES

When the isotopic abundance, mass, concentration, degree of irradiation, degree of moderation, or other pertinent property of fissile material in any package is not known, the shipper shall package the fissile material as if the unknown properties have such credible values as will cause the maximum nuclear reactivity.

C. PRELIMINARY DETERMINATIONS

1. Prior to the first use of any packaging for the shipment of more than a Type A quantity of radioactive material or fissile materials, such packaging shall be inspected to ascertain that there are no cracks, pinholes, uncontrolled voids, or other defects which could significantly reduce its effectiveness.
2. Prior to the first use of any packaging for the shipment of more than a Type A quantity of radioactive or fissile materials, where the maximum normal operating pressure will exceed 5 pounds per square-inch gauge, the containment vessel shall be tested to assure that it will not leak at an internal pressure 50 percent higher than the maximum normal operating pressure.
3. Packaging shall be conspicuously and durably marked with its model number. Prior to applying the model number, an inspection shall be made to determine that the packaging has been fabricated in accordance with the approved design.

D. ROUTINE DETERMINATIONS

Prior to each use of a package for shipment of radioactive or fissile material, the shipper shall ascertain that the package with its contents satisfies the applicable requirements of part II including determinations that:

1. the packaging has not been significantly damaged.
2. any moderators and nonfissile neutron absorbers, if required, are as authorized.
3. the closure of the package and any sealing gaskets are present and are free from defects.
4. any valve through which primary coolant can flow is protected against tampering.
5. the internal gauge pressure of the package will not exceed, during the anticipated period of transport, the maximum normal operating pressure.
6. contamination of the primary coolant will not exceed, during the anticipated period of transport, the limits in part II.E.1.d.

E. RECORDS

This shipper shall maintain for a period prescribed in appendix 0230, "Records Disposition," a record of each shipment of fissile material and each shipment of amounts of radioactive material greater than Type A quantities as defined in part I.A.6. and 18., in single packages, showing where applicable:

1. identification of the packaging by model number.
2. details of any significant defects in the packaging, with the means employed to repair the defects and prevent their recurrence.
3. volume and identification of coolant.
4. type and quantity of material in each package, and the total quantity in each shipment.
5. for each item of irradiated fissile material:
 - a. identification by model number.
 - b. irradiation and decay history to the extent appropriate to demonstrate that its nuclear and thermal characteristics comply with approved conditions.
 - c. any abnormal or unusual condition relevant to radiation safety.
6. date of the shipment.
7. for Fissile Class III, any special controls exercised.
8. name and address of the transferee.

9. address to which the shipment was made,
10. results of the determinations required by C. and D., above.

**F. DOCUMENTATION OF TECHNICAL BACKUP
SUPPORT FOR SPECIFICATION, CERTIFIED,
AND SPECIAL PERMIT PACKAGINGS**

Packagings that have been certified by the AEC as meeting DOT regulations and packagings for which specifications have been published by the DOT or a special permit has been issued by the DOT may be used by any shipper having authority to ship radioactive or fissile materials. Therefore, it is essential that technical information and limits pertinent to the construction and use of these packagings be available to all potential users. Accordingly, steps will be taken to implement the following requirements:

1. Field office managers shall require contractors under their jurisdiction to prepare a bound distributable document for each new specification, certified, or special permit, packaging designed, developed, and fabrication by him for offsite shipment of fissile and other radioactive materials. Such a document shall also be required for existing packagings for which the DOT has issued special permits except in those instances of packagings of a highly specialized design and used solely by the originator. Should these specialized packagings be adopted for more general utilization, an appropriate technical document must then be prepared. It shall be the responsibility of the originator or first

user to prepare the document for an existing packaging if it is to be used by other AEC field offices and contractors. Obsolete packagings no longer in use and containers used for onsite movement of materials are not subject to these documentation requirements unless they are reactivated, altered, or requested for use in offsite shipments. In such instances the packager or parties requiring reactivation and/or alterations shall prepare or have prepared the appropriate document.

2. Each document shall provide, as a minimum, the following information:
 - a. a complete physical and technical description of the package,
 - b. a safety analysis report including considerations for meeting the requirements for packaging and transport safety, nuclear criticality safety, and radiological safety. Pertinent documents in existence as of the date of this revision are acceptable,
 - c. design and development information, including pertinent data, analytical methods, and the results of the prescribed tests,
 - d. tables, graphs, drawings, pictures, and technical references as required to give a clear treatment of the subject.
3. Each document shall be prepared and submitted to the Technical Information Center in accordance with appendix 3201, part III.B.2, for reproduction and distribution based upon need.

ANNEX 1

NORMAL CONDITIONS OF TRANSPORT

1. **Heat**—Direct sunlight at an ambient temperature of 130°F in still air.
2. **Cold**—An ambient temperature of -40°F in still air and shade.
3. **Pressure**—Atmospheric pressure of 0.5 times standard atmospheric pressure.
4. **Vibration**—Vibration normally incident to transport.
5. **Water Spray**—A water spray sufficiently heavy to keep the entire exposed surface of the package except the bottom continuously wet during a period of 30 minutes.
6. **Free Drop**—Between 1½ and 2½ hours after the conclusion of the water spray test, a free drop through the distance specified below onto a flat essentially unyielding horizontal surface, striking the surface in a position for which maximum damage is expected.
7. **Corner Drop**—A free drop onto each corner of the package in succession or in the case of a cylindrical package, onto each quarter of each rim, from a height of 1 foot onto a flat essentially unyielding horizontal surface. This test applies only to packages which are constructed primarily of wood or fiberboard, and do not exceed 110 pounds gross weight, and to all Fissile Class II packagings.
8. **Penetration**—Impact of the hemispherical end of a vertical steel cylinder 1½ inches in diameter and weighing 13 pounds, dropped from a height of 40 inches onto the exposed surface of the package which is expected to be most vulnerable to puncture.
9. **Compression**—For packages not exceeding 10,000 pounds in weight, a compressive load equal to either 5 times the weight of the package or 2 pounds per square inch multiplied by the maximum horizontal cross section of the package, whichever is greater. The load shall be applied during a period of 24 hours, uniformly against the top and bottom of the package in the position in which the package would normally be transported.

Free Fall Distance

<u>Package Weight (pounds)</u>	<u>Distance (feet)</u>
Less than 10,000	4
10,000 to 20,000	3
20,000 to 30,000	2
More than 30,000	1

ANNEX 2

HYPOTHETICAL ACCIDENT CONDITIONS

1. **Free Drop**—A free drop through a distance of 30 feet onto a flat essentially unyielding horizontal surface, striking the surface in a position for which maximum damage is expected.
2. **Puncture**—A free drop through a distance of 40 inches striking, in a position maximum damage is expected, the top end of a vertical cylindrical mild steel bar mounted on an essentially unyielding horizontal surface. The bar shall be 6 inches in diameter, with the top horizontal and its edge rounded to a radius of not more than one-quarter inch, and of such a length as to cause maximum damage to the package, but not less than 8 inches long. The long axis of the bar shall be perpendicular to the unyielding horizontal surface.
3. **Thermal**—Exposure to a thermal test in which the heat input to the package is not less than that which would result from exposure of the whole package to a radiation environment of 1475°F for 30 minutes with an emissivity coefficient of 0.9, assuming the surfaces of the package have an absorption coefficient of 0.8. The package shall not be cooled artificially until 3 hours after the test period unless it can be shown that the temperature on the inside of the package has begun to fall in less than 3 hours.
4. **Water Immersion (fissile material packages only)**—Immersion in water to the extent that all portions of the package to be tested are under at least 3 feet of water for a period of not less than 8 hours.

ANNEX 3

TRANSPORT GROUPING OF RADIOISOTOPES

Element ¹	Radioisotopes ²	Group	Element ¹	Radioisotopes ²	Group
Actinium (89)	Ac 227	I	Copper (29)	Cu 64	IV
	Ac 228	I	Curium (96)	Cm 242	I
Americium (95)	Am 241	I		Cm 243	I
	Am 243	I		Cm 244	I
Antimony (51)	Sb 122	IV		Cm 245	I
	Sb 124	III		Cm 246	I
	Sb 125	III	Dysprosium (66)	Dy 154	III
Argon (18)	Ar 37	VI		Dy 165	IV
	Ar 41	II		Dy 166	IV
	Ar 41 ³ (uncompressed)	V	Erbium (68)	Er 169	IV
Arsenic (33)	As 73	IV		Er 171	IV
	As 74	IV	Europium (63)	Eu 150	III
	As 76	IV		Eu 152 m	IV
	As 77	IV		Eu 152	III
Astatine (85)	At 211	III		Eu 154	II
Barium (56)	Ba 131	IV		Eu 155	IV
	Ba 133	II	Fluorine (9)	F 18	IV
	Ba 140	III	Gadolinium (64)	Gd 153	IV
Berkelium (97)	Bk 249	I		Gd 159	IV
Beryllium (4)	Be 7	IV	Gallium (31)	Ga 67	III
Bismuth (83)	Bi 206	IV		Ga 72	IV
	Bi 207	III	Germanium (32)	Ge 71	IV
	Bi 210	II	Gold (79)	Au 193	III
	Bi 212	III		Au 194	III
Bromine (35)	Br 82	IV		Au 195	III
Cadmium (48)	Cd 109	IV		Au 196	IV
	Cd 115 m	III		Au 198	IV
	Cd 115	IV		Au 199	IV
Calcium (20)	Ca 45	IV	Hafnium (72)	Hf 181	IV
	Ca 47	IV	Holmium (67)	Ho 166	IV
Californium (98)	Cf 249	I	Hydrogen (1)	H 3 (see tritium)	
	Cf 250	I	Indium (49)	In 113 m	IV
	Cf 252	I		In 114 m	III
Carbon (6)	C 14	IV		In 115 m	IV
Cerium (58)	Ce 141	IV		In 115	IV
	Ce 143	IV	Iodine (53)	I 124	III
	Ce 144	III		I 125	III
Cesium (55)	Cs 131	IV		I 126	III
	Cs 134 m	III		I 129	III
	Cs 134	III		I 131	III
	Cs 135	IV		I 132	IV
	Cs 136	IV		I 133	III
	Cs 137	III		I 134	IV
Chlorine (17)	Cl 36	III		I 135	IV
	Cl 38	IV	Iridium (77)	Ir 190	IV
Chromium (24)	Cr 51	IV		Ir 192	III
Cobalt (27)	Co 56	III		Ir 194	IV
	Co 57	IV	Iron (26)	Fe 55	IV
	Co 58 m	IV		Fe 59	IV
	Co 58	IV	Krypton (36)	Kr 85 m	III
	Co 60	III		Kr 85 ³ (uncompressed)	V

Element ¹	Radionuclides ²	Group	Element ¹	Radionuclides ²	Group
	Kr 85	III	Protactinium (91)	Pa 230	I
	Kr 85 ³ (uncompressed)	VI		Pa 231	I
	Kr 87	II		Pa 233	II
	Kr 87 ³ (uncompressed)	V	Radium (88)	Ra 223	II
Lanthanum (57)	La 140	IV		Ra 224	II
Lead (82)	Pb 203	IV		Ra 226	I
	Pb 210	II		Ra 228	I
	Pb 212	II	Radon (86)	Rn 220	IV
Lutetium (71)	Lu 172	III		Rn 222	II
	Lu 177	IV	Rhenium (75)	Re 183	IV
Magnesium (12)	Mg 28	III		Re 186	IV
Manganese (25)	Mn 52	IV		Re 187	IV
	Mn 54	IV		Re 188	IV
	Mn 56	IV		Re Natural	IV
Mercury (80)	Hg 197 m	IV	Rhodium (45)	Rh 103 m	IV
	Hg 197	IV		Rh 105	IV
	Hg 203	IV	Rubidium (37)	Rb 86	IV
Mixed Fission Products	MFP	II		Rb 87	IV
Molybdenum (42)	Mo 99	IV		Rb Natural	IV
Neodymium (60)	Nd 147	IV	Ruthenium (44)	Ru 97	IV
	Nd 149	IV		Ru 103	IV
Neptunium (93)	Np 237	I		Ru 105	IV
	Np 239	I		Ru 106	III
Nickel (28)	Ni 56	III	Samarium (62)	Sm 145	III
	Ni 59	IV		Sm 147	III
	Ni 63	IV		Sm 151	IV
	Ni 65	IV		Sm 153	IV
Niobium (41)	Nb 93 m	IV	Scandium (21)	Sc 46	III
	Nb 95	IV		Sc 47	IV
	Nb 97	IV		Sc 48	IV
Osmium (76)	Os 185	IV	Selenium (34)	Se 75	IV
	Os 191 m	IV	Silicon (14)	Si 31	IV
	Os 191	IV	Silver (47)	Ag 105	IV
	Os 193	IV		Ag 110 m	III
Palladium (46)	Pd 103	IV		Ag 111	IV
	Pd 109	IV	Sodium (11)	Na 22	III
Phosphorus (15)	P 32	IV		Na 24	IV
Platinum (78)	Pt 191	IV	Strontium (38)	Sr 85 m	IV
	Pt 193	IV		Sr 85	IV
	Pt 193 m	IV		Sr 89	III
	Pt 197 m	IV		Sr 90	II
	Pt 197	IV		Sr 91	III
Plutonium (94)	Pu 238 (F)	I		Sr 92	IV
	Pu 239 (F)	I	Sulphur (16)	S 35	IV
	Pu 240	I	Tantalum (73)	Ta 182	III
	Pu 241 (F)	I	Technetium (43)	Tc 96 m	IV
	Pu 242	I		Tc 96	IV
Polonium (84)	Po 210	I		Tc 97 m	IV
Potassium (19)	K 42	IV		Tc 97	IV
	K 43	III		Tc 99 m	IV
Praseodymium (59)	Pr 142	IV		Tc 99	IV
	Pr 143	IV	Tellurium (52)	Te 125 m	IV
Promethium (61)	Pm 147	IV		Te 127 m	IV
	Pm 149	IV		Te 127	IV

**SAFETY STANDARDS FOR THE PACKAGING OF
FISSILE AND OTHER RADIOACTIVE MATERIAL**

AEC Appendix 0529
Annex 3

Element ¹	Radionuclides ²	Group	Element ¹	Radionuclides ²	Group
	Te 129 m	III	Uranium (92)	U 230	II
	Te 129	IV		U 232	I
	Te 131 m	III		U 233 (F)	II
	Te 132	IV		U 234	II
Terbium (65)	Tb 160	III		U 235 (F)	III
Thallium (81)	Tl 200	IV		U 236	II
	Tl 201	IV		U 238	III
	Tl 202	IV		U Natural	III
	Tl 204	III		U Enriched (F)	III
Thorium (90)	Th 227	II		U Depleted	III
	Th 228	I	Vandium (23)	V 48	IV
	Th 230	I		V 49	III
	Th 231	I	Xenon (54)	Xe 125	III
	Th 232	III		Xe 131 m	III
	Th 234	II		Xe 131 ³ m (uncompressed)	V
	Th Natural	III		Xe 133	III
Thulium (69)	Tm 168	III		Xe 133 ³ (uncompressed)	VI
	Tm 170	III		Xe 135	II
	Tm 171	IV		Xe 135 ³ (uncompressed)	V
Tin (50)	Sn 113	IV	Ytterbium (70)	Yb 175	IV
	Sn 117	III	Yttrium (39)	Y 88	III
	Sn 121	III		Y 90	IV
	Sn 125	IV		Y 91 m	III
Tritium (1)	H 3	IV		Y 91	III
	H 3 (as gas, as luminous paint or absorbed on solid material)	VII		Y 92	IV
				Y 93	IV
			Zinc (30)	Zn 65	IV
				Zn 69 m	IV
Tungsten (74)	W 181	IV		Zn 69	IV
	W 185	IV	Zirconium (40)	Zr 93	IV
	W 187	IV		Zr 95	III
				Zr 97	IV

¹ Atomic Number shown in parentheses.

² Atomic weight shown after the radionuclide symbol.

³ Uncompressed means at a pressure not exceeding one atmosphere.

m Metastable state.

(F) Fissile Material.

ANNEX 4

TESTS FOR SPECIAL FORM MATERIAL

1. **Free Drop**—A free drop through a distance of 30 feet onto a flat essentially unyielding horizontal surface, striking the surface in such a position as to suffer maximum damage.
2. **Percussion**—Impact of the flat circular end of a 1-inch diameter steel rod weighing 3 pounds, dropped through a distance of 40 inches. The capsule or material shall be placed on a sheet of lead, of hardness number 3.5 to 4.5 on the Vickers scale, and not more than 1-inch thick, supported by a smooth essentially unyielding surface.
3. **Heating**—Heating in air to a temperature of 1475°F and remaining at that temperature for a period of 10 minutes.
4. **Immersion**—Immersion for 24 hours in water at room temperature. The water shall be at pH 6–pH 8, with a maximum conductivity of 10 micromhos per centimeter.

APPENDIX B
LIMITS ON RADIOACTIVE MATERIALS

Appendix B. Limits on Radioactive Materials

This appendix demonstrates how the limits listed in Section 1.0 apply to specific shipments of radioisotopes and spontaneously fissioning materials. For example, assume that reactor-produced ^{244}Cm made by long term irradiation of plutonium is to be shipped. Also assume that the basic fission products have been removed from the material and only the curium fraction and several related impurities remain. A typical analysis of such material might show the curium mass fraction to contain 80% ^{244}Cm , 2% ^{242}Cm , 1/2% ^{245}Cm , 10% ^{246}Cm , 1/2% ^{247}Cm , 5% ^{241}Am , and 2% ^{243}Am . The estimated length of time the capsule will be sealed is 60 d (5.18×10^6 s). The capsule size, tentatively chosen from Fig. 2, is 139.3 cm^3 . The sample density is 10.0 g/cm^3 .

To determine the maximum sample mass that can be shipped, one must look at each of the limits.

DECAY POWER

The specific decay power can be determined by multiplying the mass fraction by the maximum decay power (from Table 3). Table B-1 has the result.

The maximum shipment mass is

$$\begin{aligned} G &= \frac{\text{Max power}}{\text{Table B-1 total}} \\ &= \frac{250 \text{ W}}{4.8 \text{ W/g}} \\ &= 52.1 \text{ g} . \end{aligned}$$

CRITICALITY

Table B-1 lists the mass ratio of isotopes with potential criticality hazards to the total shipment mass. The maximum shipment mass is

$$\begin{aligned} G &= \frac{\text{Total allowable mass from Section 1.2 or 1.3}}{\text{Table B-1 total}} \\ &= \frac{15\text{g}}{0.01} \\ &= 1500 \text{ g} . \end{aligned}$$

Table B-1. Sample calculations for radioactive materials.

Isotope	Mass fraction	Power ^a (W/g)	Gas generation rate ^a (cm ³ /s·g)	Neutron Emission rate ^a (neutrons/s·g)	Mass ratio for critical isotopes
Am-241	0.050	0.006	0.022×10^{-8}	—	—
Am-243	0.020	0.001	—	—	—
Cm-242	0.020	2.448	9.140×10^{-8}	3.6×10^5	—
Cm-244	0.800	2.336	9.176×10^{-8}	93.6×10^5	—
Cm-245	0.005	0.001	—	—	0.005
Cm-246	0.100	0.001	—	9.5×10^5	—
Cm-247	0.005	0.001	—	—	0.005
TOTAL	1.000	4.8	18.3×10^{-8}	10.7×10^6	0.010

^aBased on 1 g of the mixed sample.

RADIATION DOSE RATE

All the isotopes in the shipment primarily emit soft gamma rays (less than 100 keV) that can be ignored. The dose rate in this case will be the result of spontaneous neutron emission. Using the limit of 6.5×10^8 neutrons/s from Section 4.3.1, one gets a maximum shipment mass of

$$\begin{aligned}
 G &= \frac{\text{Max neutron emission rate}}{\text{Table B-1 total}} \\
 &= \frac{6.5 \times 10^8 \text{ neutrons/s}}{10.7 \times 10^6 \text{ neutrons/s}} \\
 &= 60.8 \text{ g} .
 \end{aligned}$$

GAS GENERATION

Table 3 lists the gas generation rate. Section 4.2.4.2 contains the expression for the maximum gas generation rate.

$$\frac{6.92(V_C - V_S)}{\emptyset} \geq G \text{ (total from Table B-1).}$$

In this example, the sample density is known but not its volume. However, V_S equals G divided by its density. Substituting the known values,

$$\frac{6.92\left(139.3 - \frac{G}{10}\right)}{5.18 \times 10^6} \geq G (18.3 \times 10^{-8})$$

$$G = 588 \text{ g.}$$

The calculations indicate that the 250-W decay power is the limiting factor in this example. The maximum sample mass that can be shipped is 52.1 g.

APPENDIX C. LIST OF CASK DRAWINGS

List of Cask Drawings - No. 1

<u>Drawing Title</u>	<u>LLL Number</u>
Neutron shipping cask, model 4T	AAA 60-414963
Container body	AAA 60-414694
Container cover	AAA 60-414965
Stud	AAA 70-100076
Plug	AAA 68-120276
Blocking guides	AAA 60-414966
Gamma shield	
Radiation shipping cask	AAA 59-414690
Container	AAA 59-414676
Case bottom	AAA 59-414679
Drain outlet	AAA 59-414814
Boss	AAA 59-414682
Shielding plug	AAA 59-414680
Case shielding D-38	AAA 59-414686
Outer shell	AAA 59-414916
Inner shell	AAA 59-414917
Base tube	AAA 59-414688
Coil	AAA 59-414918
Cap	AAA 59-414677
Cover	AAA 59-414685
Cap tube	AAA 59-414683
Cap shielding D-38	AAA 59-414678
End plate	AAA 59-414684
Alternate cap	AAA 68-118940
Stud	AAA 59-414689
Special-form capsule	
Capsule-bolted "Conoseal" joint (Aeroquip proprietary)	AAA 69-108511

APPENDIX D
TIEDOWN EQUIPMENT DESIGN LIMITS

Table D-1. Maximum design load for improved plow steel wire rope for package tiedown.^a

Wire rope diam (in.)	Max design load for improved plow steel (lb)	Wt (lb/ft)	Approx radius of wire rope thimble (in.)
3/8	2,400	0.23	7/8
7/16	3,200	0.31	1
1/2	4,200	0.40	1-3/16
9/16	5,400	0.51	1-7/32
5/8	6,600	0.63	1-3/8
3/4	9,600	0.90	1-5/8
7/8	12,800	1.20	1-13/16
1	16,800	1.60	2

^aThe recommended maximum design load is based on data in CFR (Code of Federal Regulations) 29, Chapter XVII, Section 1926.251(c)(1), Table H-3 (7/1/73).

Table D-2. Welded link-chain working load limit.^a

Nominal size (in.)	Alloy steel chain ASTM Spec A 391 (lb)	Yield load of load link (lb)
1/4	3,250	—
3/8	6,600	7,300
1/2	11,250	12,500
5/8	16,500	18,200
3/4	23,000	25,300
7/8	28,750	31,700
1	38,750	42,700

^aThe working load limit is based on CFR 29, Chapter XVII, Section 1926.251(b)(4), Table H-1 (7/1/73).

Table D-3. Working load limits for turnbuckles.^{a,b}

Wire rope size (in.)	Jaw or eye size (in.)	Load ^c (lb)
3/8	1/2	2,440
7/16	5/8	3,300
1/2	3/4	4,280
9/16	3/4	5,400
5/8	7/8	6,680
3/4	1	9,520
7/8	1-1/4	12,880
1	1-1/4	16,720
	1-1/2	21,000
	1-3/4	31,080
	2	36,800

^aDerived from Rigger's Handbook, Broderick & Bascom Rope Co. and information from Crosby-Laughlin Co.

^bWhere turnbuckles are used, a jumper shall be installed of equivalent-rated load capacity between the points connected by the turnbuckle. The jumper (e.g., wire rope or chain) shall be of suitable size and configuration to make sure the tiedown assumes the load if the turnbuckle fails. The jumper may be an integral part of the turnbuckle.

^cSafety factor = 5.

Table D-4. Working load limits for shackles.^a

Shank (in.)	Load (lb)	Pin (in.)	Inside width (in.)	Inside length (in.)
1/2	2,800	5/8	1/2	1-7/8
5/8	4,400	3/4	5/8	2-5/16
3/4	6,400	7/8	3/4	2-5/8
7/8	8,600	1	7/8	3-3/16
1	11,200	1-1/8	1	3-1/2
1-1/8	13,400	1-1/4	1-1/8	3-15/16
1-1/4	16,400	1-3/8	1-1/4	4-5/8
1-3/8	20,000	1-1/2	1-3/8	4-15/16
1-1/2	23,800	1-5/8	1-1/2	5-1/2
1-3/4	32,400	2	1-3/4	6-5/8
2	42,400	2-1/4	2	7-3/4

^aThe working load limit is based on CFR 29, Chapter XVII, Section 1926.251 (f)(1), Table H-19 (7/1/73).

APPENDIX E. SAMPLE FORM FOR REUSE INSPECTION

USA/6478/BLF (ERDA-SAN)

	Date
	Acceptable
	Needs Work
	Initials

**APPENDIX F. SAMPLE FORM FOR LOADING AND HANDLING
PROCEDURES**

LOADING AND HANDLING PROCEDURE

USA/6478/BLF (ERDA-SAN)

Date	Initials

1. Establish that the limitations in Section 1.0 will not be exceeded. Record the values for the following items. See Appendix E for sample calculations.
 - 1.1 Maximum decay power _____ (Not to exceed 250 W)
 - 1.2 Total mass of fissile isotopes _____
 - 1.3 Total mass of ²⁵¹Cf _____
 - 1.4 Radiation dose rate at surface _____
 - 1.5 Radiation dose rate at 3 ft (91.4 cm) _____
 - 1.6 Length of time capsule can be sealed without exceeding gas generation limit _____
2. Check last Reuse Inspection check list for acceptability.
3. Special-Form Capsule loading
 - 3.1 Verify that cavity is clean and dry.
 - 3.2 Place radioisotope package in capsule.
 - 3.3 Install new metal gasket in body sealing recess with gasket cone apex upwards.
 - 3.4 Install cover.
 - 3.5 Coat cap screws with graphite powder and insert in body.
 - 3.6 Tighten the screws evenly and in steps to the following torque:

<u>Size</u>	<u>Torque</u>
0.19 to 32 UNJF	65 to 70 in.-lb
0.25 to 28 UNJF	95 to 100 in.-lb
4. Gamma Shield
 - 4.1 Verify that cavity and drain tube are clean and dry.
 - 4.2 Coat body stud threads with molybdenum disulphide.
 - 4.3 Clean gamma-shield and cap-seal surfaces.
 - 4.4 Install O-ring gasket in groove.
 - 4.5 Place special-form capsule in gamma shield.
 - 4.6 Place cap on gamma shield.
 - 4.7 Torque the nuts evenly and in steps to 450 to 470 in.-lb.

Date	Initials

4.8 Optional: Install safety wire through the fastener holes and secure loads.

5. Container Loading

5.1 Verify that container cavity is clean and dry.

5.2 Place gamma shield or capsule rack and special-form capsule in container.

5.3 Coat body stud threads with molybdenum disulphide (Lubriband M).

5.4 Clean body, and cover seal surfaces.

5.5 Install flat gasket on body.

5.6 Place cover on body.

5.7 Install nuts and torque evenly and in steps to 270 to 280 in.-lb.

5.8 Install safety wire through the studs and nuts, and secure ends with lead seal.

6. Cask Transportation

6.1 The cask may be handled by a forklift using the pallet base or by using the lifting lugs.

6.2 Four drop-forged steel shackles shall be used with the lifting lugs. The pin diameter shall be 1.00 in. or greater.

6.3 The lifting slings shall be adequate for the load and at least 3 ft. long.

7. Cask Tiedown

7.1 The cask shall be placed on the truck bed longitudinal centerline and tied down to the bed frame, not the wood decking (Fig. F-1).

7.2 The cask tiedown system shall use drop-forged steel shackles with a pin diameter of 1.125 in.

7.3 The four tension members securing the cask to the truck bed shall be 0.875 in. diam, 6-by-19-improved-plow-steel wire rope or 0.50 in. steel chain per ASTM A391.

Fig. F-1. Model of cask tied down on truck.