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ANALYSIS OP NEUTRON SHIPPING CONTAINER 6-GS-l

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ANALYSIS OF NEUTRON SHIPPING CONTAINER 6-GS-1

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George L. Wigle, Peter S. Bringham

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Lawrence Berkeley Laboratory
University of California
Berkeley, California

February 1974

ABSTRACT

The Laboratory has built a container for shipment of Type B quantities of gamma and neutron emitting nuclides in Special Form. Shielding materials are depleted uranium and a hydrated gypsum-spodumene composite. This paper includes a description of the complete package, along with tabulated permissible quantities of specific nuclides, test data, and engineering evaluations required for compliance with AEC and DOT regulations. The DOT Special Permit is appended.

* Work done under the auspices of the U.S. Atomic Energy Commission.

ANALYSIS OF THE NEUTRON SHIPPING CONTAINER MODEL 6-GS-1

1. Scope

This analysis has been submitted for certification of a shielded shipping cask system, intended for transport of all isotopes of americium, curium, berkelium, californium, einsteinium, and fermium in solid form. The following maximum limitations will be maintained on shipments in this container:*

a) a combined maximum total heat load of 243 watts from radioactive decay heating;

b) a total neutron emission from any combination of the above elements of 1.8×10^8 neutrons per second;

c) a total net mass of 20 grams of any combination of the above elements (except for d and e below);

d) a total net mass of 15 grams of any combination of the following isotopes which are potential criticality hazards (ref: "Criticality of Transuranium Actinides - Unmoderated Systems" Bierman and Clayton; Appendix A, p. xi) ^{242m}Am, ^{243}Cm, ^{245}Cm, ^{247}Cm, ^{249}Cf, ^{251}Cf, except for e below:

e) a total net mass of .06 grams of ^{251}Cf;

f) a total integrated radiation dose of 200 mr/hr at the external surface of the shielding container;

g) a total integrated radiation dose of 10 mr/hr three feet from the shielding container.

This shipping cask system is designed for multipurpose usage. Basically it consists of two separate shields - an outer gypsum-spodumene filled neutron shield and an inner depleted uranium metal filled gamma shield. In the maximum shielding configuration the smaller gamma shield is inserted into the spodumene filled outer jacket in order to provide both gamma and neutron shielding.

The shipping container is designed to meet the requirements of the AEC Safety Standards for the "Packaging of Radioactive and Fissile Materials" Chapter 0529.

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*See Appendix A for a general discussion on the way these limitations apply to specific isotopes.

2. Description

In maximum shielding configuration, the shipping container is cylindrical in shape, 2 ft. in diameter, 2 ft. 6 in. tall, and is mounted on a steel pallet type base 2 ft. 10 in. square. The cover is cylindrical in shape, and is held in place by 15 studs with nuts. The inner cavity can accommodate several containment vessels or one gamma shield. The gamma shield is an independent vessel, and is used when the gamma shielding of the outer container alone is inadequate for the shipment. It is the same gamma shield identified as Spec. 55 (depleted uranium) included in S.P. 5872 previously issued to Lawrence Radiation Laboratory. The Spec. 55 gamma shield consists of three nestable depleted uranium cylinders, providing internal cavity sizes and shielding thicknesses as follows: (Ref.: Fig. 17)

Cavity Size I.D. X Height	Uranium Thickness (inches)			
1.58 X 3.03	1.0	---	3.0	4.0
3.87 X 5.4	---	2.0	3.0	---
8.21 X 8.5	1.0	---	---	---

The shipping container weighs 1840 lbs with 4" gamma shield, and 1230 lbs with no gamma shield. Separate containment vessels are used with each cask configuration (see items 3 and 5 below).

3. Materials of Construction

The shipping container is cylindrical in shape, and composite in construction. The materials used are listed in sequence starting from the outside of the container and moving inward:

a) 0.250 inches H.R. carbon steel plate, exterior painted with white enamel;

b) six inch thick gypsum-spodumene composite cast in place. Mix ratio (by weight) is as follows: 40% by weight of plaster of paris, 30% by weight spodumene, 30% by weight water. Cured weight is approximately 113 lbs per cubic foot.

c) 0.250 inch H.R. carbon steel plate, interior painted with white enamel;

d) 3/8 inch air space between inner (gamma shield) and outer vessel;

e) 0.125 inch type 304 stainless steel, exterior wall of inner vessel;

f) three nested cylinders of clad depleted uranium totalling 4.0 inches uranium and 0.31 inch stainless steel

g) containment vessel:

Special Form Haynes 718 high temperature alloy "Gamah" type capsule with metal gasket

4. Materials Used as Neutron Absorbers or Moderators

The 4-inch thick uranium cylinder serves as gamma shield and also as elastic scatterer for degrading neutrons with energies greater than 1 MeV. The lithium-6 in the spodumene aggregate absorbs thermal neutrons. The hydrogen combined in the gypsum serves as a neutron moderator.

5. Means of Containment

All materials shipped will be in solid form. Shipment will not exceed Type B quantity limitations as defined in 49CFR 173.389(K)(1), nor will more than exempt quantities of fissile materials be shipped.

All heavy elements will be placed in a laboratory type vial with a screw cap which is not leak tight. The vial, in turn, will be enclosed in a high temperature alloy pressure vessel with a "Gamah" sealed joint (made by Gamah Division, Stanley Aviation Corporation*). This capsule is designed with an operating pressure of 40 psig at 1500°F and a proof pressure of 80 psig at 1500°F.

Its maximum permissible leak rate is 1 X 10^{-8} standard cc's per second. Tests conducted in accordance with AEC 0529, annex 4 qualify this vessel as conforming to Special Form (see page 11).

This primary container will be located by means of metal spacers within the 6-GS-1 cavity or if additional gamma shielding is required, within the Spec. 55 uranium container. In the latter case, the Spec. 55 will also be located, if required, by spacers. In all cases, the respective containers will be centered relative to the 6-GS-1 cavity, and shims at the bottom and top will provide firm contact with the respective lids to eliminate internal shock due to vertical acceleration.

The outer cover of the shielding container and the Spec. 55 gamma shield cover are sealed with flat rubber gaskets. These seals serve to maintain cleanliness of the container; each Spec. 55 container has been tested and approved as leak-tight after 62 hours submersion under 40 inches of water.

6. Heat Dissipation

If the radioactive material is to be shipped via exclusive use carrier where a permissible 180°F surface temperature (in 130° ambient sun temperature) applies, the maximum permissible isotopic decay heat will be 243 watts. See Appendix B.

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*Reference to a company or product name does not imply approval of the product by the University of California nor the U. S. Atomic Energy Commission to the exclusion of others that may be suitable.

If the shipment is partial load via commercial carrier, the 122°F surface temperature limitation and the ambient environmental temperature then determines the permissible decay-heat loading. For example, at a typical 97°F shade temperature (LRL-Livermore) decay heating in a partial load shipment must contribute no more than 200 watts. Administrative controls, in these cases, will prescribe the source quantity per shipment under prevailing environmental conditions. See curve "Heat Load vs. Ambient Shade Temperatures" Fig. 14, Appendix B.

7. Pressure Relief System

The outer wall of the neutron-shield shipping cask, and the inner cavity are fitted with 75 psig rupture disc assemblies for the relief of pressure generated by an external fire. The disc assemblies are equipped with vacuum supports to minimize flexing in temperature and pressure extremes, and the discs themselves are plastic coated to resist normal atmosphere corrosion and retain their rupture pressure. Recommended maximum internal pressures on the cask's outer wall are 265 psig, and 456 psig on the inner wall. (Ref.: ASME Pressure Vessel Code, Sec. 8, 1968)

8. Lifting System for Cask

The shipping container is provided with four lifting lugs. One lug will safely carry more than seventeen times the weight of the container, and provides a safety factor of 3.7 over the load imposed by simultaneous 10 G, 5 G, and 2 G accelerations. The rectangular base of the container forms a pallet so that the container may be raised with a fork lift.

9. Strength of Outer Shell

Considered as a simple beam, the outer (24" O.D.) cladding will support five times the combined weight of the cask and gamma shield, and provides an additional factor of safety of 94.

10. Identification of Container

The shipping container will be identified by a stainless steel plate welded to the outer vertical steel surface. The identification plate will state the following information:

RADIOACTIVE MATERIAL
DOT SP# 6550
NEUTRON SHIPPING CASK MODEL 6-GS-1
WEIGHT 1840 LBS. W GAMMA SHIELD
WEIGHT 1230 LBS. W/O GAMMA SHIELD
LRL DWG HCD 63464
RETURN TO:
LAWRENCE RADIATION LABORATORY
BERKELEY, CALIFORNIA

11. Design Compliance with AEC 0529 Criteria

A - 1 Compliance--Radioactive material, whether in form of dry powder or metallic foil, will be placed in an appropriate container for ease of handling. This container will then be sealed inside a DOT Special Form container, which in turn, depending upon the gamma output, will be placed either inside one of the Spec. 55 stainless clad uranium containers, or mounted in locating spiders or "annular 'doughnuts'." Cavity liner and outer surface of neutron shield is painted carbon steel.

A - 2 Compliance--The neutron shield lid is secured by fifteen studs with nuts, with "tamper-proof" sealed wire running through each stud. The lid of the inner uranium container is dogged to the container body, the latches held in place by bolts in detents. Special Form containers will, of course, be sealed per DOT regulations.

A - 3a Compliance--Four lugs total. One lifting lug will support seventeen times weight of packaging. As additional safeguard, 1-1/4" thick tie down lug requires carrier's use of no smaller than 7/8" shackle, safe working load for which is 8600 lbs/shackle.

A - 3b Compliance--1/2" eyebolt safe working load is 1100 lbs. Weight of lid is 150 lbs.

A - 3c Compliance--No structural part other than lugs usable as lifting device.

A - 3d Compliance--Neutron shielding material will behave much as solid block of concrete and will remain as integral unit even though part of outer steel cladding is peeled off.

A - 4a Compliance--Lug "B" (Appendix B), which takes greatest stress, will withstand 3.7 times resultant stress from combined 10 G, 5 G, 2 G loadings.

A - 4b Not applicable--Tie downs are only part of structure which can be so used.

B Although this request does not seek certification for large quantity shipments, container 6-GS-1 does meet the requirements in this section.

B - 1 Compliance--Regarded as a simple beam, packaging will withstand 5 times its fully loaded weight, uniformly distributed, with an additional Factor of Safety of 27. (Appendix B)

B - 2 Compliance--Calculated lowest maximum permissible external stress was 146 psi for the outer steel shell (24" diameter).

C - 1 Compliance--Container 6-GS-1 is intended to be shipped for transport of isotopes of Cm, Am, Bk, Cf, Es, and Fm. Of these, certain nuclides are fissile, and exempt quantities have been calculated as limits.

^{242m}Am	.2 g	^{247}Cm	2.4 g
^{243}Cm	3.0 g	^{249}Cf	.4 g
^{245}Cm	.5 g	^{251}Cf	.06 g

Exempt quantities for the foregoing six nuclides are in the same ratio to their water-moderated <u>subcritical</u> limits as is AEC exempt quantity of 15 g ^{235}U to its water-moderated critical mass of 750 g. This ratio is 15:750 = .02.
The following nuclides shall be limited to their respective quantities:

^{237}Np	40 g	^{241}Am	115 g
^{240}Pu	150 g	^{242}Am	6 g

Exempt quantities of these latter four nuclides are based upon 1/2 ratio of exempt quantity (15 g): critical mass of those regulated Fissile Materials as defined in AEC Appendix 0529, Sec. I A.5.

C - 2 Not applicable--Liquid shipments of transuranic nuclides will not be shipped in this container.

C - 3 Not applicable

D - 1a Compliance--Calculations show that the package will withstand conditions of normal transport.

<u>Heat</u> - With administrative control of quantity of contents and/or routing of the transport relative to ambient temperatures, the external surface temperature will not exceed 180°F. (See Appendix A, and Appendix B)

<u>Cold</u> - Since there is no liquid coolant present, package will not be damaged by exposure to cold.

<u>Pressure</u> - Package will withstand several atmospheres pressure without damage (see B - 2 above).

<u>Vibration</u> - All joints in package are continuous-welded, and lid bolts are tightened to compress lid gasket. Package will sustain normal transport vibration.

<u>Water Spray</u> - All welded joints helium leak tested at fabrication. Stencilled directions on the lid request tightening torque sufficient to provide 20% compression

of lid gasket. Under these conditions, the package will not leak under a water spray.

Free Drop - A 4' drop at a position of maximum damage, (slightly off-axis, upside down, striking the lid) will at most tend to elongate two lid bolts. The package will be inspected for integrity prior to each shipment.

Penetration - Outer shell of cask, point most vulnerable to puncture, is 1/4" steel. 13 lb. test cylinder dropping 4 ft. onto this area will not damage shell or internal shielding.

Compression - Considering only outer 24" diameter shell as column, package will withstand 5 times weight of package with Factor of Safety of about 73. (Appendix B)

D - 1b Hypothetical Accident Conditions Compliance--Appendix B

Free Drop

Analysis of impact loading on lid bolts was carried out following procedure used by J. H. Evans - ORNL TM-2220.

Stress per stud on impact (of fully loaded package) is 83,000 psi; S_μ for stud material is 90,000 psi. Studs will retain the lid.

Stud retaining plate; maximum shear stress is 33,800 psi; S_μ for plate material is 58,000 psi.

Welds: Maximum stress in weld is 27,200 psi.

$S_{min.u}$ = 60,000 psi

Cavity liner: at point of maximum stress, S = 27,100 psi

$S_{min.u}$ = 60,000 psi

Lid: Max S_r = 56,000 psi; Min. ult. tensile = 58,000 psi

Puncture

We enclose a summary report of impact tests on our gypsum-spodumene mixture. Note that although test samples were unclad, an impact energy of roughly half that encountered by our full size cask under Hypothetical Accident Conditions produces only 1/4" penetration. Extending the curve indicates an expected maximum penetration of about 1/2". Neither attenuation of the primary or Compton scattered gamma is measurably altered thereby, while the neutron attenuation is decreased by about 10%. Combined dose rate at the surface from the punctured cask will then be 208. mR/hour. It can be seen from figure 2, Appendix A, that a gypsum-spodumene thickness of only 4" (a 2" penetration into the 6-inch shield) will increase the dose rate by a factor of 70%

Thermal

In our calculations we assumed:

1. Surface temperature, T_s, = 1500°F at the outer surface of gypsum-spodumene, ignoring ΔT across outer steel cladding.

2. No ΔT across inner steel cladding of gypsum.

3. No ΔT across any air gaps; there will be two to four such gaps.

4. A maximum $(T_s - T_2)$ as determining a constant heat flow rate during the entire half-hour exposure to the fire.

5. No additional heat capacity of uranium, lead, or steel inner containers.

Under these conditions, total heat flux, during the 1/2 hour fire, through outer container wall to its cavity is only 3,470 Btu. This produces a temperature rise of 13°F. Additional temperature rise at that point after removal of cask from fire will not exceed 5°F at which time internal temperatures will begin falling.

Maximum internal temperature then is

434°F	(maximum ambient)
13°F	(ΔT during fire)
5°F	(ΔT "heat lag")
452°F	final temperature

Puncture + Thermal (Fire)

With an extrapolated puncture depth of .5", and assuming complete loss of gypsum-spodumene (rather than compression), temperature rise in the impact area calculates to about 0.1°F above rise in adjoining areas, during fire test.

Water Immersion

In a fire, the lid gasket of the outer 6-GS-1 container will deteriorate. Similarly, the gasket of the inner uranium container will deteriorate if in addition to the fire, the heat input from the nuclear contents exceeds 75 watts. The innermost Special Form container container however will not leak under these conditions, since it meets all Special Form requirements (see page 10).

D - 2. 3 Compliance--Other than possible special routing and/or scheduling of shipments as determined by nuclide quantity versus surface temperatures, no special controls need be exercised by the shipper. Normal controls of external dose rate, etc., will always be applied, of course.

E - 1 a, b Compliance--see D - 1a above. Calculated internal pressure of outer cask to ambient solar heat and nuclear decay is 10.1 psig.

Cask will withstand 265 psi. Rupture disc assemblies (3) on cask have bursting pressure of 75 psi (internal).

E - 1c Compliance--No gases or vapors which are flammable or explosive will be shipped. Normal pressure rise of trapped air and vapors due to ΔT will be retained by package.

E - 1 d, e Not applicable--No coolant or cooling devices employed.

E - 2a Compliance--Quantities of fissile materials will be limited to those as tabulated in Appendix A - all amounts calculated as subcritical "as a function of the concentration---in water-reflected metal-water mixtures."

E - 2b Compliance--Geometric form or arrangement of fissile contents will not alter subcriticality due to limitations upon quantities described in Appendix A. This applies under normal or hypothetical accident conditions.

E - 2c Compliance--Seal of Special Form container will not be damaged during fire. See also E - 2a above.

E - 2d 1, 2 Compliance--See B - 2 above.

E - 2d 3 Compliance--No openings in outer container.

E - 3 Not applicable--Large quantity shipments not considered. Package does meet requirements for this section, however.

F - 1a Compliance--see Appendix B. Calculated maximum reduction in shielding due to impact is based on impact-penetration curve of "wet" gypsum-spodumene. This, plus loss due to complete dehydration, will result in total attenuation loss of 62% with a surface dose rate of 530 mR/hr.

F - 1b Seal of Special Form container will not be damaged by anticipated loss of some thermal shielding. Similarly, impact energy is almost entirely absorbed by outer cladding and shielding mixture.

F - 2 Compliance--See E - 2a, b, c above.

G - 1 Compliance--Contents of the package are limited such that any configuration within each package will be subcritical (Appendix A). Equal change in shape, volume, shielding to each package in any array of such packages due to transport damage will not be sufficient to alter the subcriticality of such an array (Appendix B).

G - 2 Compliance--Limitation of package contents is based on water reflection. II - C above.

H - 1, 2 Compliance--The contents of the package have been so limited that any number of packages would be subcritical in any arrangement when the package is undamaged, or when the package is subjected to the hypothetical accident conditions of free drop, thermal, water immersion, in that sequence.

ANNEX 4 PARAGRAPHS

Tests were conducted on the Gamah capsule #400238 to verify that it would meet Special Form requirements. Maximum permissible leakage was 1 X 10^{-8} scc/sec.

1. Free Drop

One capsule was dropped three times from a height of 30 ft. onto a steel pad on concrete. The point of contact on the capsule was controlled so that each drop would be such that maximum damage might be done to the seal. When pressurized to 100 psig helium, the leak rate was undetectable at a sensitivity of 1 X 10^{-9} scc/sec.

2. Impact

A 3.6 pound, 1-inch steel bar was dropped repeatedly from a height of 40 inches on the circumference of the capsule cavity. No deformation detected as measured with a dial indicator which can be read to .0002".

3. Heating

A sealed capsule, pressurized to 40 psi helium was heated to 1450°F and in a ten minute interval climbed to 1460°F. The sensing thermocouple was attached next to the seal of the capsule. At this point the thermocouple and power feed-throughs in the stainless furnace shell showed an imminent breakdown, the furnace was shut off. Although the temperature achieved was not the prescribed 1475°F, there is no reason why an additional 15° should alter the sealing capabilities, for these reasons:

a) 15° represents less than 0.7% of the stated 1475°;

b) both materials from which the capsule assembly is fabricated fabricated (Haynes 25 for the gasket, Haynes 718 for the capsule body and lid) are high temperature, high strength alloys instead of the stainless steel usually used for Special Form capsules.

"Short Time Tensile"	at 1600°F	Type 316 Stainless	22,000 psi
" " "	at 1600°F	Haynes 718	49,000 psi
" " "	at 1650°F	Haynes 25	42,000 psi

c) The stress - temperature curve for Haynes 718 indicates a reduction in yield stress form 70 kpsi at 1460°F to 67 kpsi at 1475°F. The S-T curve for Haynes 25 indicates no reduction in yield stress from a temperature of 1460°F to 1475°F. In the case of Haynes 718, the reduction in yield stress is neither significant nor critical to the structural integrity of the capsule.

d) Because of thermal expansion, the internal pressure at peak temperature was 185 psig. Yet throughout the heat cycle from room temperature to 1460°F to cool-down, no leak was detected with the helium mass spectrometer sensitivity at 2.4 X 10^{-10} standard cc's per second.

4. Immersion

The above heat test conditions were more severe than a water immersion test. In an additional test, the capsule interior was evacuated, with an external helium atmosphere of 15 psi, and no reverse leak was detected. Immersion tests were therefore not conducted.

APPENDIX A

Limitations on Radioactive Materials

Cask to be used for neutron and gamma-emitting materials, the neutron-emitting isotopes being those of Am, Cm, Bk, Cf, Es, and Fm.

The limiting quantities on contents are imposed so as to meet the following conditions:

1. A maximum radiation of 200 mr/hr at the surface of the container and 10 mr/hr at 3 feet from the surface;

2. A maximum generation of 243 watts in the cavity of the cask;

3. An absence of potential critical mass quantities.

Table 1 lists allowable quantities for the alpha-neutron-emitting isotopes, for which the prevailing limitation is the 10 mr/hr at 3 feet restriction; the maximum allowable quantities given for each represent the emission of 1.8 X 10^8 n/sec, the emission rate of ^{252}Cf, which gives 10 mr/hr 3 feet from the surface.

Table 2 lists allowable quantities for alpha and alpha-neutron-emitting isotopes for which the prevailing limitation stems from the 243 watt restriction; they are those whose relatively short partial half-life for alpha emission results in heat from the alpha decay energy.

Table 3 gives quantities for isotopes with large thermal neutron fission cross sections, which thereby present potential criticality hazard.

Figure 1 plots allowable gamma photon curie content for 1 and 2 Mev vs. uranium liner (of thicknesses up to 4 in.) that will comply with the 10 mr/hr at 3 feet from the surface restriction.

Using the 4 inch uranium liner*, the following curies of gamma photons are allowed in the 6-GS-1 package:

MeV	Curies of Gamma Photons
1	3 X 10^4
2	69

*Spec. 55 Container described uner Item 2.

NEUTRON SHIPPING CONTAINER MODEL 6-GS-1

All isotopes listed are in Transport Group I.

Table 1: Allowable quantities of alpha-neutron-emitting isotopes for which the prevailing limitation is the 10 m4/hr at 3 feet restriction:

Isotope	Quantity	Ci
^{244}Cm	10g	0820.000
^{246}Cm	20g	0005.400
^{248}Cm	5g	0000.015
^{250}Cm	20mg	0000.0014
^{248}Cf	10mg	0015.000
^{250}Cf	20mg	0002.800
^{252}Cf	85μg	0000.048
^{254}Es	700mg	1120.000
^{255}Es	200μg	0004.200
^{257}Fm	100μg	0000.600

Table 2: Allowable quantities of alpha and alpha-neutron-emitting isotopes for which the prevailing limitation is the 243 watt heat generation: (Average alpha decay energy is approximately 6 Mev)

^{240}Cm	338 mg	6760
^{242}Cm	2.0 g	6760
^{253}Es	270 mg	6760

Table 3: Nuclides with large thermal neutron cross-sections for which the limits shall be as follows. Limiting quantities in this table are considered exempt quantities and are calculated in this manner: the ratio of exempt ^{235}U to the conservative value for critical mass in a fully moderated system is

$$\frac{15g}{750g} = .02$$

This ratio is then multiplied by the subcritical limiting mass in a moderated system as reported by Clark.[1]

---

1. H. K. Clark, Critical Masses of Fissile Transplutonium Isotopes, Abstracts from ANS Transactions, Vol. 12, No. 2, 11/69

The resulting limits are as follows:

^{242m}Am	.2g	^{247}Cm	2.4g
^{243}Cm	3. g	^{249}Cf	.4g
^{245}Cm	.5g	^{251}Cf	.06g

Table 4: Nuclides with large thermal neutron fission cross sections:

At the time of our original application for DOT approval, there existed no standards of exempt quantities for the above trans-uranium actinides. We accordingly developed the following method of calculating tentative "exempt" quantities, which method AEC approved as feasible. The ratio of exempt quantities of 15 grams ^{238}Pu to its critical mass of 4.5×10^3 grams is 3.3×10^{-3}. As an additional safety factor this ratio is halved to 1.65×10^{-3}. The new ratio is then multiplied by the computed least critical mass (steel reflected) for each of the actinides as reported by Bierman and Clayton.[2] The following masses are obtained as "exempt" quantities:

^{237}Np	40. g	^{241}Am	115. g
^{240}Pu	150 g	^{242}Am	6. g

Shipment of those isotopes whose "permissible" quantities exceed 5,000 Ci (^{249}Cf and ^{242}Cm) will be administratively controlled to 5,000 Ci or less in order to qualify as Type B quantities.

All above nuclides, Tables 1 through 4, shall be further limited in combination such that the sum, for all nuclides present, of the ratios between the total weight for each nuclide and its permissible weight does not exceed unity.

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2. S. R. Bierman, E. D. Clayton, Criticality of Transuranium Actinides, Unmoderated Systems, Abstracts from ANS Transactions, Vol. 12, No. 2., 11/69

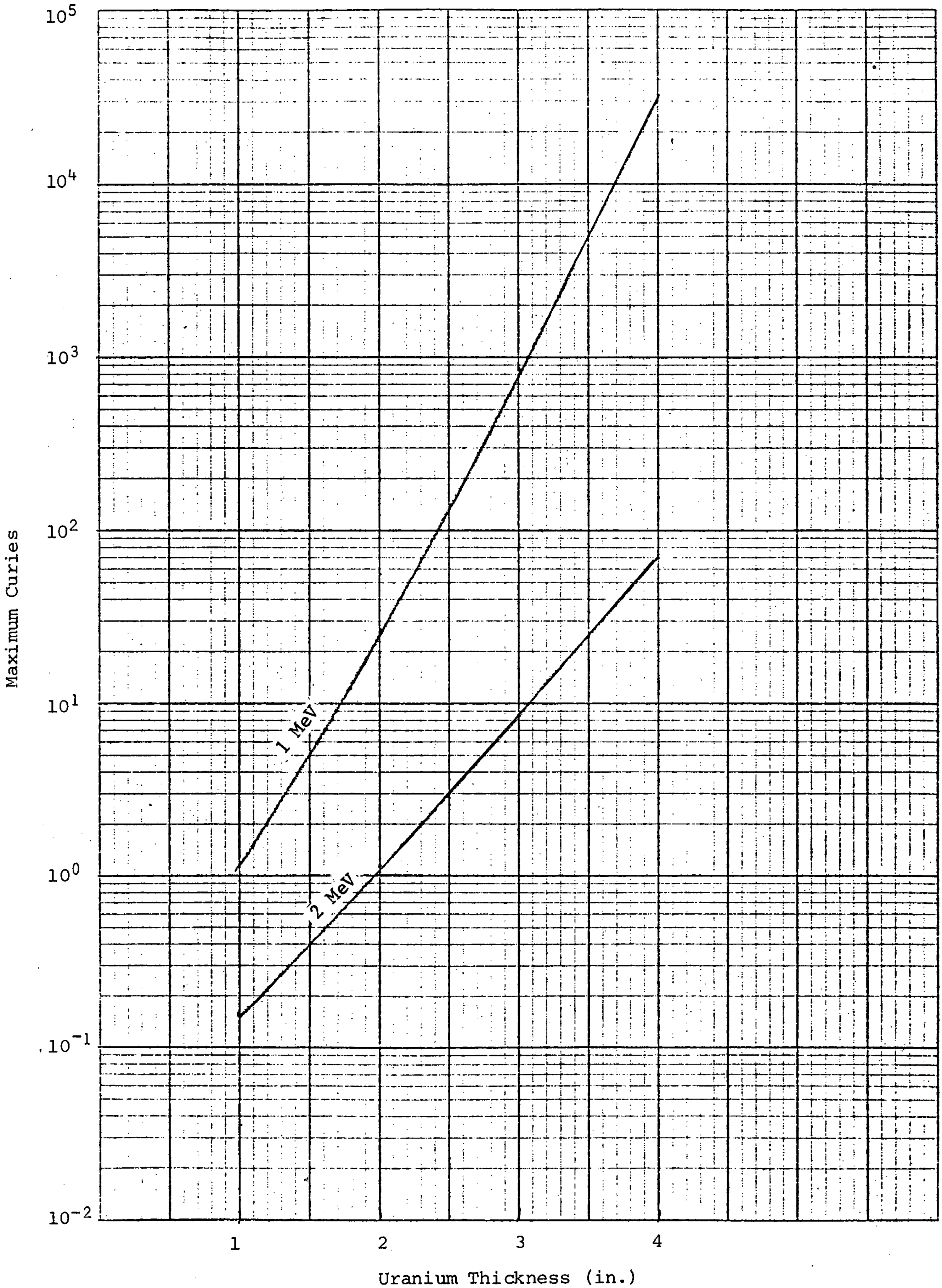


Fig. 1. Maximum gamma-emitting contents in 6-GS-1 uranium package to meet external dose rate restrictions: 200 mR/hr at surface, 10 mR/hr at 3 meters from surface.

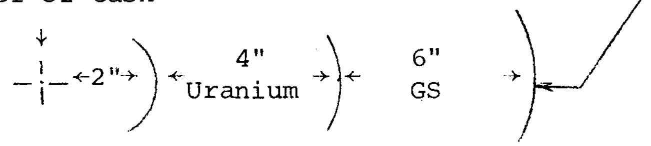
Calculation of Limiting Quantities on Contents of 6-in. GS Container

Conditions to be met:

1. Maximum radiation of 200 mrem/hr at surface and 10 mrem/hr at 3 ft from surface;
2. Maximum generation of 243 watts in cavity;
3. Absence of potential critical mass.

Condition 1: Maximum radiation of 200 mrem/hr at surface and 10 mrem/hr at 3 ft.

center of cask surface of cask



Dose rate at surface = $\frac{[4\pi r^2 D(r)]}{4\pi R^2} e^{-\Sigma_u x_u}$ [(mrem/hr)/(n/sec)] from neutrons.

where $[4\pi r^2 D(r)]$ is obtained from Fig. 2, neutron curve, = 6×10^{-2} [(mrem/hr)/(n/sec)] cm^2

r is the thickness of GS in cm = 6 in. = 15.24 cm
R is radius of cask in cm = 2 in. + 4 in. + 6 in. = 12 in. = 30.5 cm
Σ_u is fast neutron removal cross section, in cm^{-1}, for uranium = 0.0095 $cm^2/g \times 18.7\ g/cm^3 = 0.178\ cm^{-1}$
x is thickness of uranium, in cm, = 4 in. = 10.2 cm.

$$\text{Dose rate} = \frac{6 \times 10^{-2} e^{-0.178 \times 10.2}}{4\pi (30.5)^2} = 8.4 \times 10^{-7}\ [(\text{mrem/hr})(\text{n/sec})]$$
from neutrons.

Dose rate at surface from capture gamma = $\frac{[4\pi r^2 D(r)]}{4\pi R^2}$ [(mrem/hr)/(n/sec)]

where $[4\pi r^2 D(r)]$ is obtained from Fig. 2, capture gamma curve, = 3.5×10^{-4} [(mrem/hr)/(n/sec)] cm^2
r and R same as above

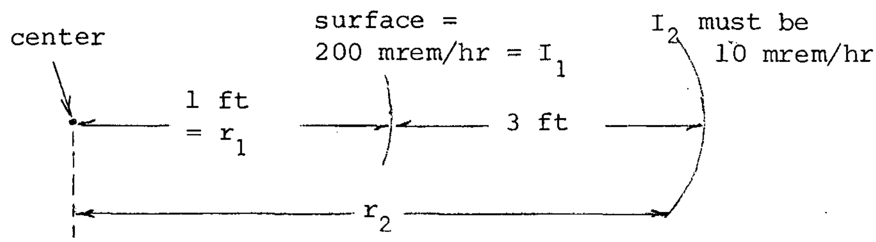
$$\text{Dose rate} = \frac{3.5 \times 10^{-4}}{4\pi (30.5)^2} = 0.3 \times 10^{-7}\ [(\text{mrem/hr})/(\text{n/sec})]$$

Total dose rate at surface = $8.4 \times 10^{-7} + 0.3 \times 10^{-7} = 8.7 \times 10^{-7}$ [(mrem/hr)/(n/sec)]

At 200 mrem/hr on surface, and 2.17×10^6 n/sec-μg of ^{252}Cf, allowable mass of ^{252}Cf, G, is

$$G = \frac{200}{8.7 \times 10^{-7} \times 2.17 \times 10^6} = 106 \mu g.$$

However, test the 10 mrem/hr at 3 ft. from surface condition:



$$I_2 = \frac{I_1 r_1^2}{r_2^2} = \frac{200 \times 1^2}{4^2} = 12.5 \text{ mrem/hr.}$$

Therefore the 10 mrem/hr condition prevails - and

allowable mass $G = \frac{10}{12.5} \times 106\ \mu g = 85\ \mu g\ {}^{252}Cf$.

So make table for various spontaneously fissioning materials to show mass that will give neutron emission equivalent to 85μg ${}^{252}Cf$. This is

$$2.17 \times 10^6 \text{ n/sec-}\mu g \times 85\ \mu g = 1.8 \times 10^8 \text{ n/sec.}$$

Now neutron/sec = $\nu\, dN/dt = \nu N \lambda$,

where ν is number of neutrons/fission
N is number of atoms in sample
λ is spontaneous fission decay constant, sec^{-1}

Using an average value of ν = 3.5 neutrons/spontaneous fission,

$$\text{n/sec} = \frac{6 \times 10^{23} \times 10^{-6}}{A} (\text{atoms}/\mu g) \times 3.5 \text{ (n/atom)} \times G\ (\mu g) \times \frac{0.693}{T_{1/2}(\text{sec})}$$

$$= \frac{1.45 \times 10^{18}\ G}{AT_{1/2}},$$

where A is atomic number of sample
$T_{1/2}$ is spontaneous fission half life in seconds
G is number of micrograms to meet condition equivalent to 85 μg ${}^{252}Cf$.

Putting this equal to 1.8×10^8 n/sec,

$$G = \frac{1.8 \times 10^8\ AT_{1/2}}{1.45 \times 10^{18}} = 1.24 \times 10^{-10}\ A\ T_{1/2}\ \mu g,$$ to meet condition equivalent to 85 μg ${}^{252}Cf$.

Example: ^{244}Cm; $T_{1/2}$ for spontaneous fission is 1.31×10^7 yr.

$$G = 1.24 \times 10^{-10} \times 244 \times 1.31 \times 10^7 \times 3.15 \times 10^7$$

$$= 1.24 \times 10^7 \ \mu g \text{ or } \sim 10 \text{ g}.$$

Table 1 shows limiting quantities of consequent spontaneously fissioning Cm, Cf, Es, and Fm isotopes, computed in the above manner--isotopes limited by Condition 1. Gamma radiation from spontaneous fission of these quantities in this cask can be shown to be negligible.

Condition 2: Maximum generation of 243 watts in cavity

Heat from alpha decay is computed as follows:

$$dE/dt = dN/dt \times E_\alpha/\text{dis.} = (N\lambda)(E/\text{dis.}),$$

where dN/dt is disintegration rate from α decay
E_α is energy per disintegration from α decay
λ is the α decay constant.

So heat rate may be expressed as P:

$$P = \frac{6\text{X}10^{23}\text{X}10^{-6}}{A} \times \frac{0.693}{T_{1/2}} \times E_\alpha(\text{Mev}) \times 1.6\text{X}10^{-6}(\text{ergs/Mev}) \times \frac{1}{10^7}(\text{watts/ergs.sec.}^{-1})$$

N: atoms/μg, where A is atomic wt.

λ, where $T_{1/2}$ is partial half life for alpha decay, in seconds

$$= \frac{6.66 \times 10^4 \ E_\alpha}{AT_{1/2}} \text{watts}/\mu g.$$

Using an average alpha decay energy of 6 MeV,

$$P = \frac{4 \times 10^5}{AT_{1/2}} = \text{watts}/\mu g.$$

At $P = 243$ watts and G the number of micrograms to give 243 watts,

$$243 = \frac{4 \times 10^5 \ G}{AT_{1/2}}$$

$$G = 60.75 \times 10^{-5} \ AT_{1/2}.$$

Example: ^{240}Cm; $T_{1/2}$ for alpha decay = 26.8 days.

$$G = 60.75 \times 10^{-5} \times 240 \times 26.8 \times 8.64 \times 10^4 = 3.38 \times 10^5 \ \mu g$$

$$= 338 \text{ mg}.$$

(Note: examining this isotope for Condition 1, limiting quantity would be:

$$G = 1.24 \times 10^{-10} \times 240 \times 7.9 \times 10^5 \times 3.15 \times 10^7 = 7.4 \times 10^5 \mu g = 740 \text{ mg},$$

vs. 355 mg above,

so the allowable quantity is dictated by Condition2.)

Table 2 shows allowable quantities of consequent transuranic isotopes limited by Condition 2.

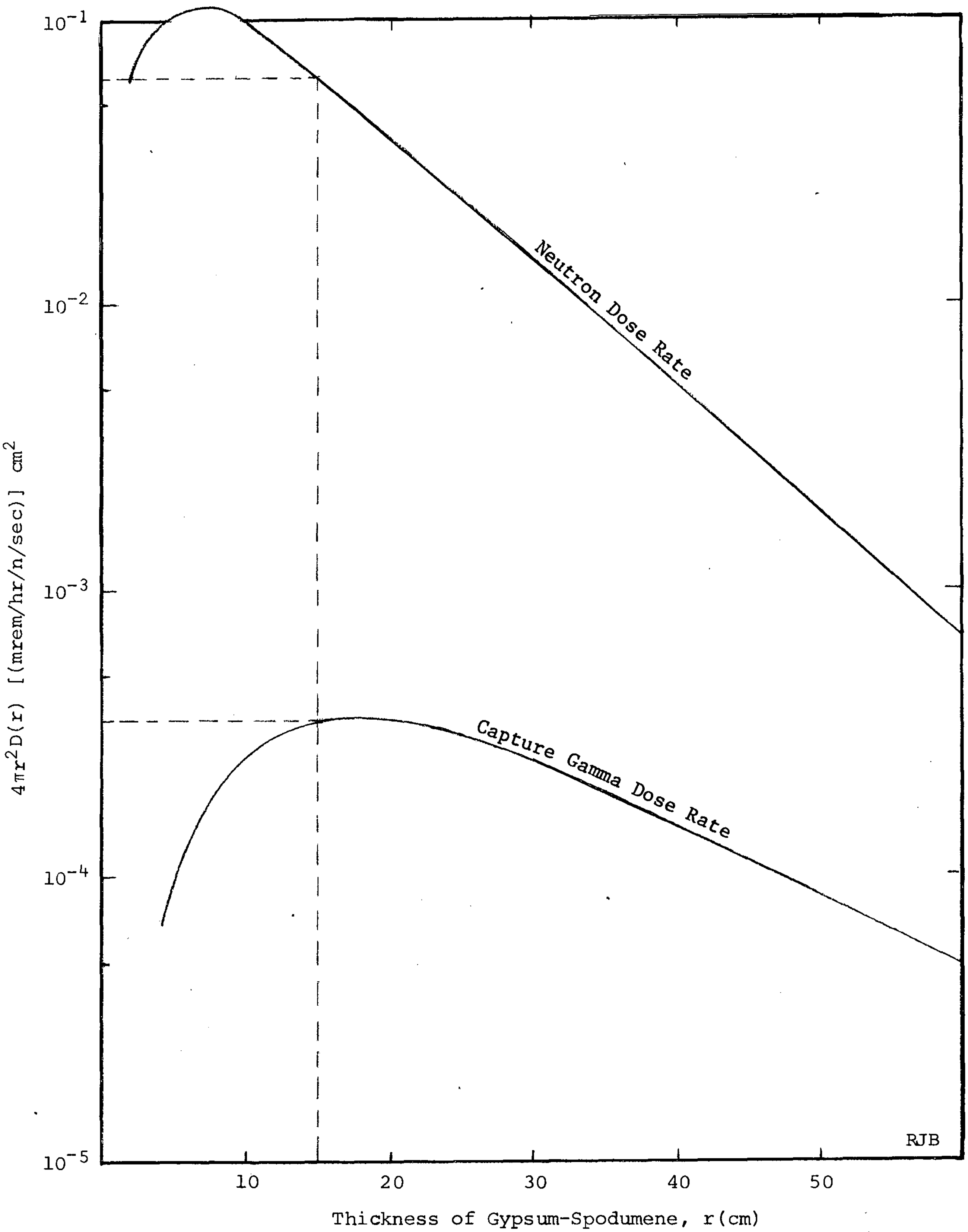


Fig. 2. Dose Rates in Gypsum-Spodumene as a function of distance from a unit point isotropic fission source (ANISN Case 9).

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ENGINEERING NOTE

FILE NO.	PAGE 21
SUBJECT: Appendix B — SHIPPING CONTAINER #6GS-1 TIE-DOWN ACCELERATION STRESSES	NAME P. Bringham, G.Wigle
	DATE October 5, 1970

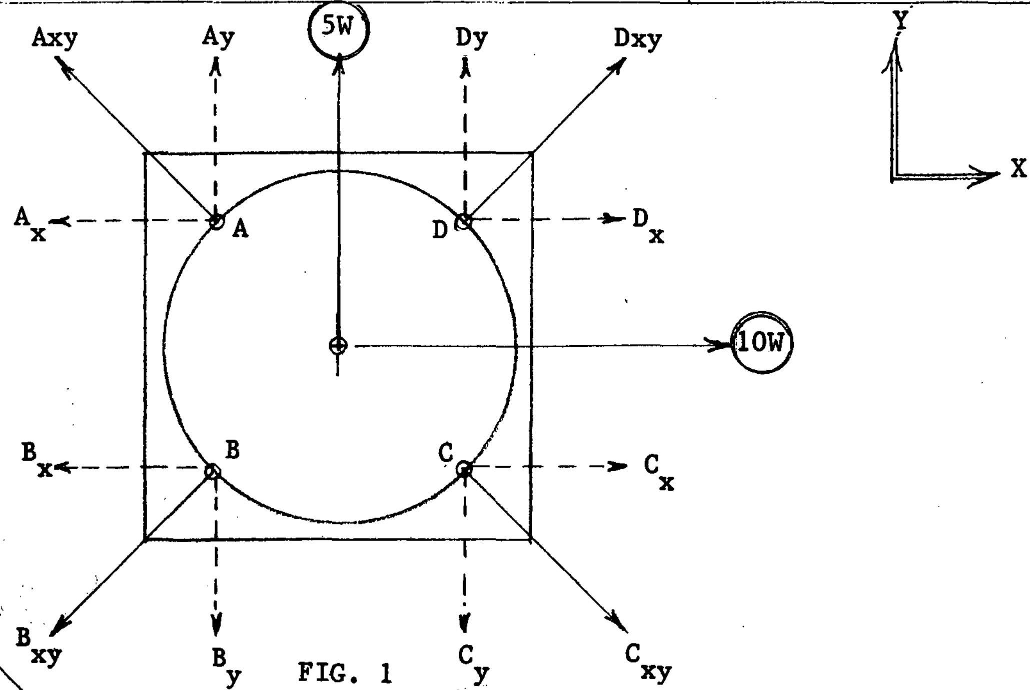


FIG. 1

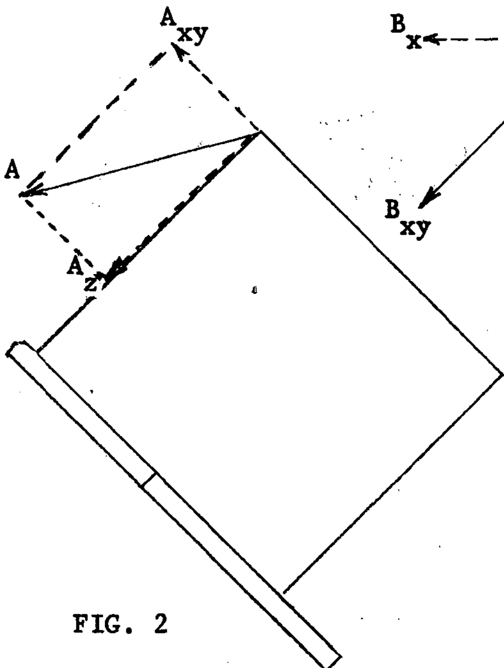


FIG. 2

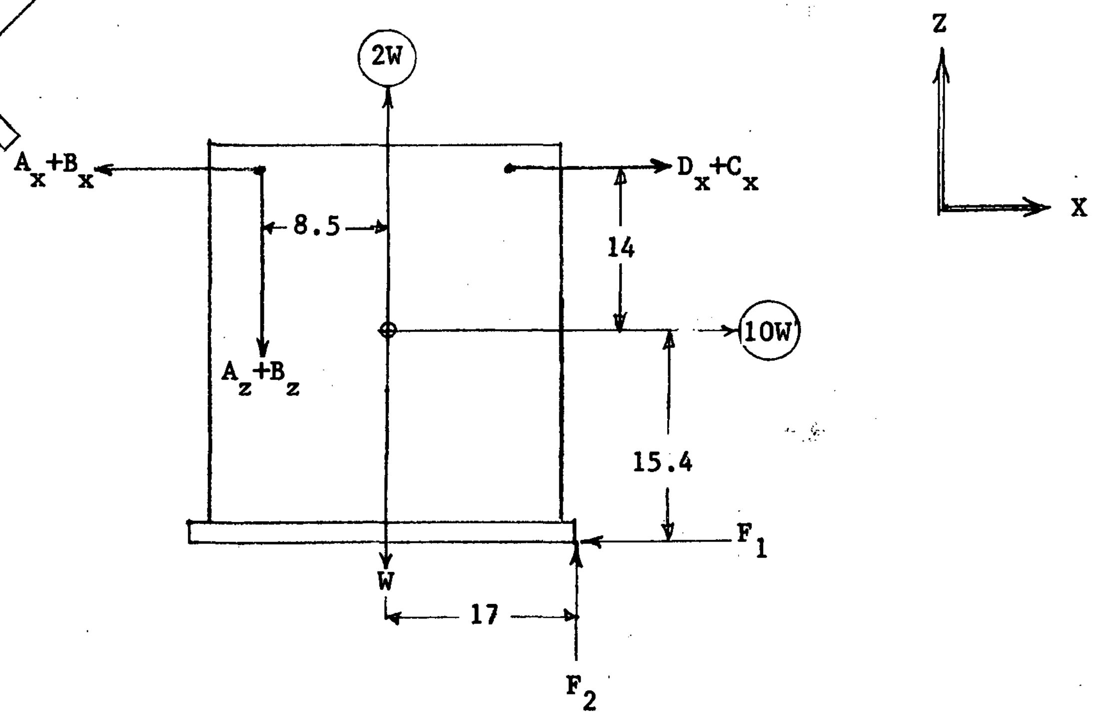


FIG. 3

I. Find reactions on A, B and F_1, F_2 due to 10ω load

II. Find reactions on B and C due to 5ω load

III. Find reactions on A, B, C and D due to 2ω load

IV. Sum reactions to find total reactions A, B, C and D due to combined 10ω, 5ω and 2ω load

----

I. Find resistive reactions A, B and F_1, F_2 due to 10ω load only. Reactions on D and C cable may be neglected.

$\Sigma F_x = 0 = -A_x - B_x - F_1 + 10\omega$

$A_x = B_x = A_y = B_y$

ΣM_y axis (Fig. 3) $= -(A_x + B_x)(14) - (A_z + B_z)(8.5) + F_1(15.4) - F_2(17) = 0$

ΣF_z (Fig. 3, neglecting 2ω) $= 0 = F_2 - (A_z + B_z) - \omega$

$A_y = A_{xy} \cos 45 = .707\ A_{xy}$

$A_{xy} = A \cos 45 = .707\ A = A_z = B_z$

$A_y = .5A$

$A + F_1 = 10\omega$ or $F_1 = 10\omega - A$

$26A - 15.4\ F_1 + 17\ F_2 = 0$

$1.41\ A - F_2 = -\omega$ or $F_2 = \omega + 1.41A$

$26A - 15.4(10\omega - A) + 17(\omega + 1.41A) = 0$

$65.4A = 137\omega$

$$\left.\begin{aligned} A &= 2.1\omega \ (= B) \\ F_2 &= 4\omega \\ F_1 &= 8.0\omega \end{aligned}\right\} \text{due to 10ω load and weight of cask only}$$

II. Find resistive reactions B, C due to 5ω load only. There are no reactions at A & D due to 5ω load.

Using similar analysis as before:

SUBJECT: Appendix B — SHIPPING CONTAINER #6GS-1, TIE-DOWN ACCELERATION STRESSES

NAME: P.Bringham,G.Wigle

DATE: October 5, 1970

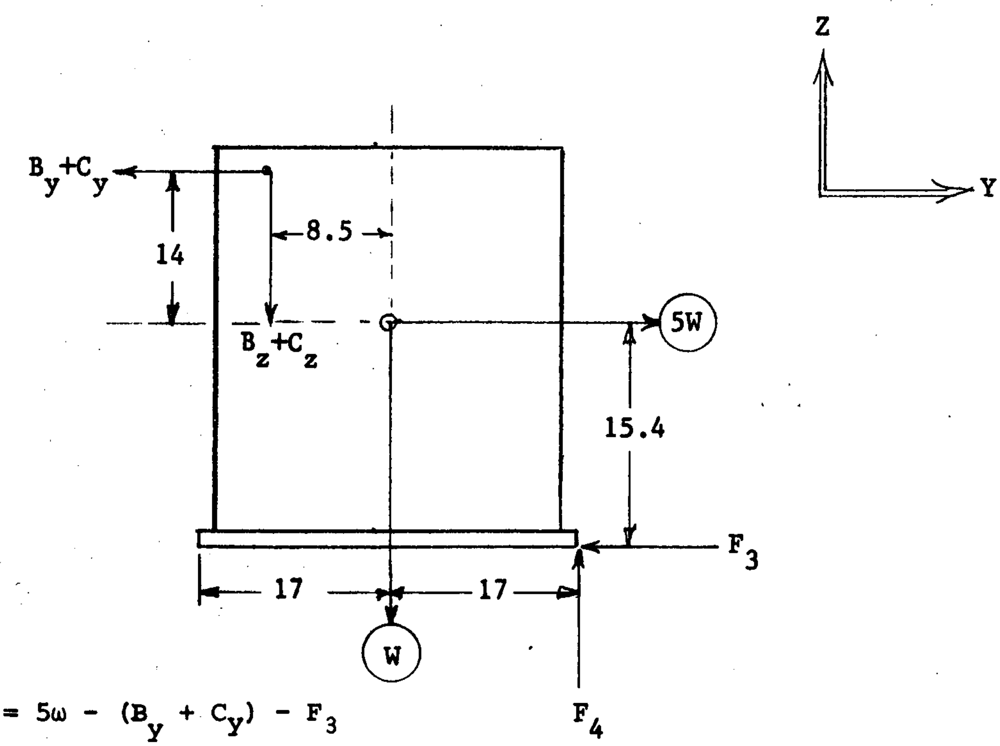


FIG. 4

$$\varepsilon F_y = 0 = 5\omega - (B_y + C_y) - F_3$$

$$B_y = C_y$$

$$\varepsilon F_2 = F_4 - (B_z + C_z) - \omega = 0$$

$$B_y + C_y = B$$

$$B_z + C_z = 1.41\ B$$

$$B_y = .707\ B_{xy}$$

$$B_{xy} = .707\ B = B_z = C_z$$

$$B_y = .5B$$

$$5\omega - B - F_3 = 0$$

$$\varepsilon M_x \text{ axis} = 0 = -14B - 8.5(1.41B) - 17\ F_4 + 15.4\ F_3 = 0$$

$$= -26B - 17(1.41B + \omega) + 15.4(5\omega - B) = 0$$

$$= 65.4B = 60\omega$$

$$\left.\begin{aligned} B &= .918\omega = C \\ F_3 &= 4.08\ \omega \\ F_4 &= 2.29\ \omega \end{aligned}\right\} \text{due to } 5\omega \text{ load only}$$

III. Resistance reactions, A, B, C, D due to 2ω load only

$$A_z = B_z = C_z = D_z$$

$$\varepsilon F_2 = A_z + B_z = C_z + D_z = 2\omega - \omega$$

$$A_z = .25\omega$$

$$A = \frac{.25\omega}{.707} = .354\omega = B = C = D$$

The 2ω load tends to relieve the loads on F_2 and F_4 by .50ω

IV. Summation of reactions A, B, C, D due to 10ω, 5ω, 2ω loads

$$A = 2.1\omega + .354\omega = 2.45\omega$$

$$B = 2.1\omega + .918\omega + .354\omega = 3.37\omega$$

$$C = .918\omega + .354\omega = 1.27\omega$$

$$D = .354\omega$$

F_2 = 3.50ω acting vertically uniformly along side CD of pallet

F_1 = 8.0ω acting horizontally uniformly along side CD of pallet

F_3 = 4.1ω acting horizontally uniformly along side AD of pallet

F_4 = 1.8ω acting vertically uniformly along side AD of pallet

V. GENERAL DATA

A. Weight, container and lid (unfilled, steel only) 610 lbs
Filled with gypsum spodumene 1230 lbs
Complete packaging (with uranium container) 1840 lbs

B. Center of gravity of outer 6GS-1 and inner uranium container is 15.4" from base (floor level).

VI. LIFTING AND TIE-DOWN LUGS

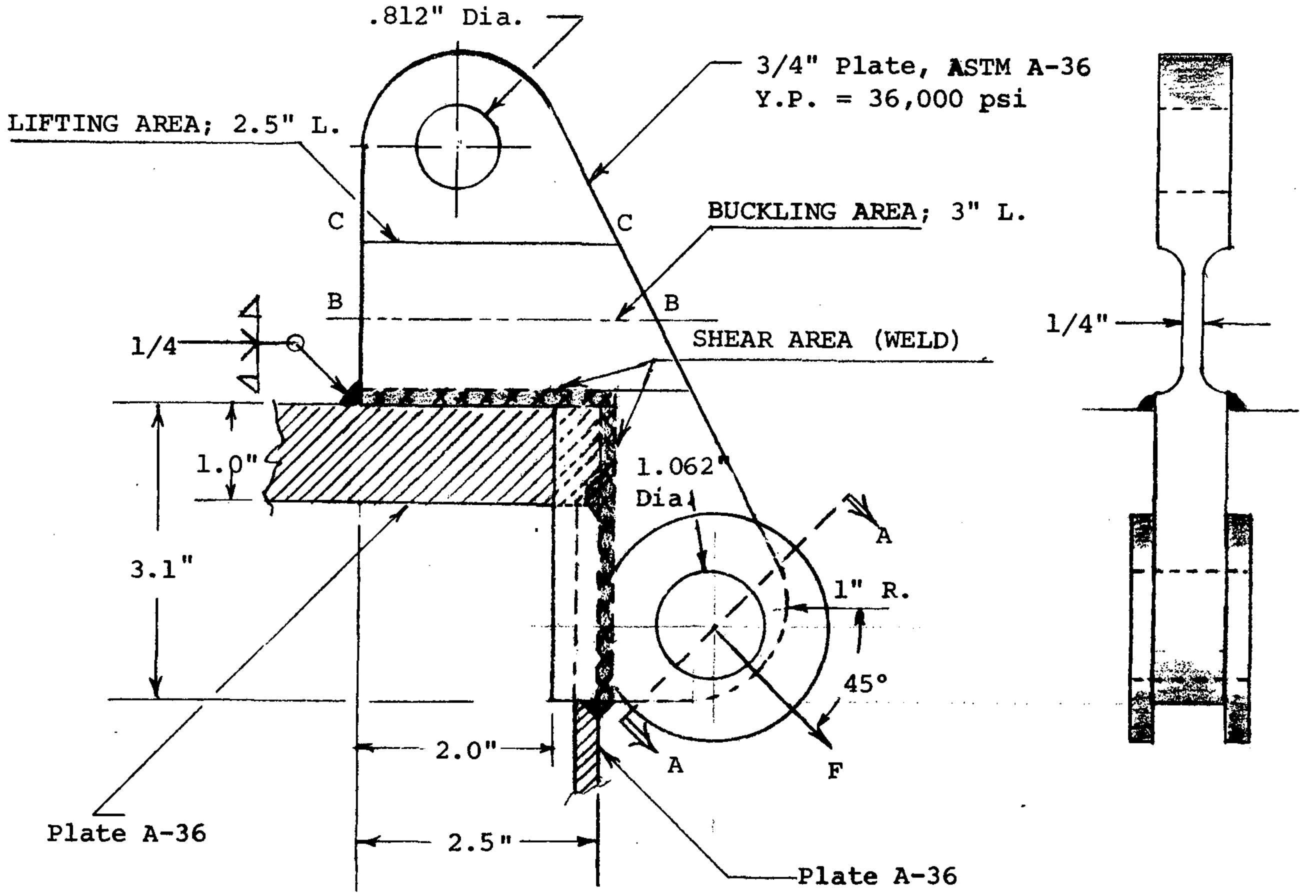


Fig. 5 a

A. As TIE-DOWN

1. TENSILE STRESS in LUG = F/A (Fig. 5b)

F = Load on lug "B" under simultaneous 10G, 5G, 2G = 3.4 (1840) = 6256 lbs

Area at A - A = 2X.25(2.37-1.062) + .75(2.37-1.062-.20) = 1.48 in^2

S_r = 4227psi; F.S. = 36,000/3970 = 8.5

SUBJECT: SHIPPING CONTAINER 6-GS-1 STRESS CALCULATIONS
Appendix B

NAME: P.Bringham/G.Wigle
DATE: Rev. 2/74

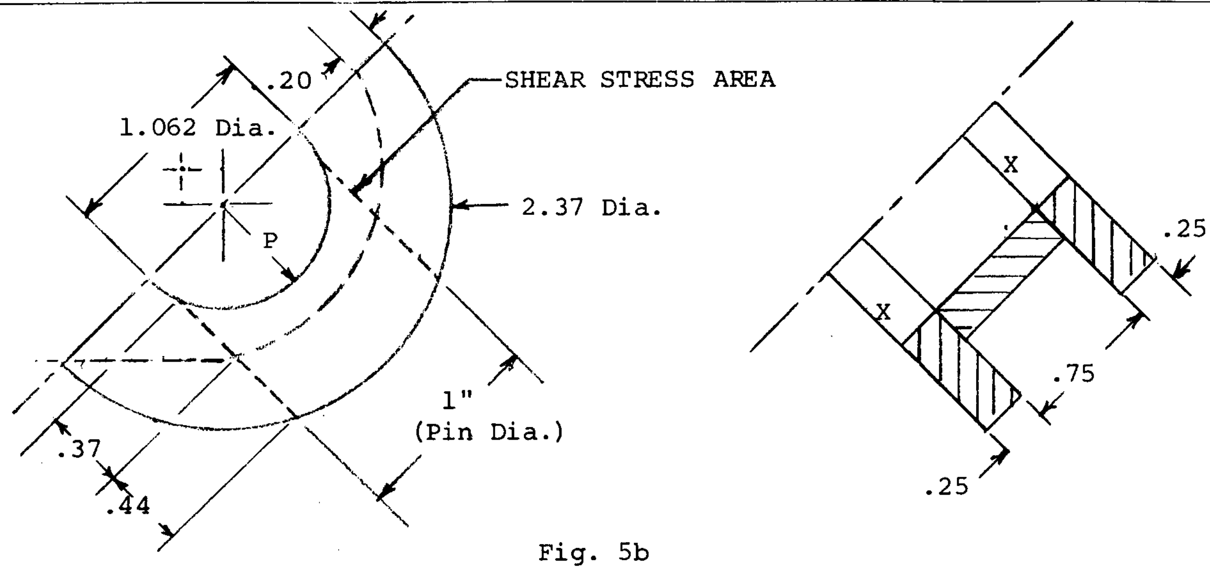


Fig. 5b

2. SHEAR STRESS at HOLE

$$S_s = \frac{P}{A} = \frac{6256}{2[(.37 \times 1.25) + (.44 \times .50)]} = 4583 \text{ psi}$$

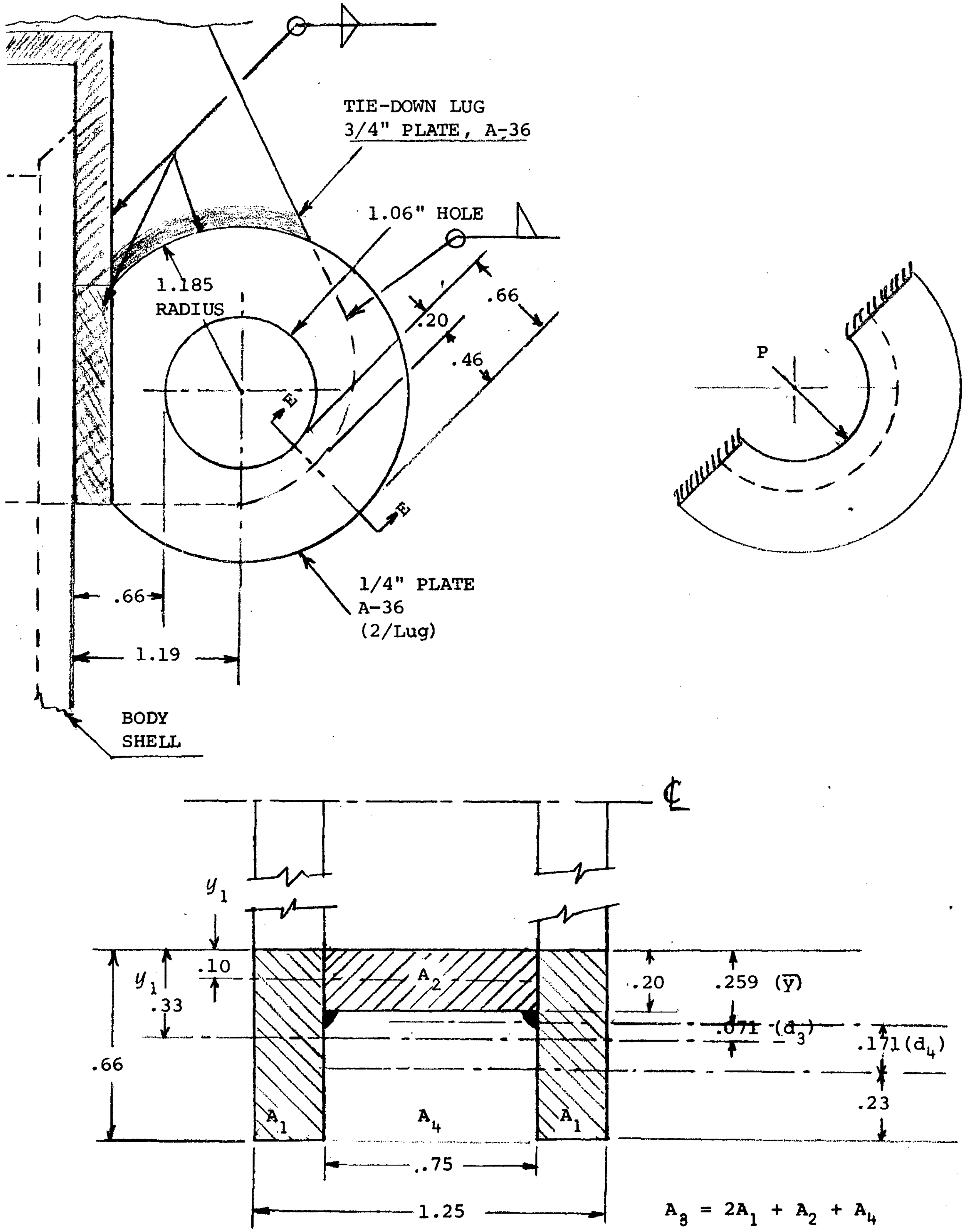
Using shear strength $= \frac{S_y}{2} = 18{,}000$ psi, F.S. = 3.9.

3. CONTACT STRESS at LUG HOLE

Since the lug hole and shackle pin are nearly equal, 1.062" and 1.00", we may assume, as with a bolt and clearance hole, that the forces involved are uniformly distributed over the projected contact area A of the pin.

The $S_c = \frac{F}{td} = \frac{6256}{1.25 \times 1.00} = 5005$ psi

$$\text{F.S.} = \frac{58{,}000}{5{,}144} = 11.6$$



VIEW E-E

Fig. 5c

SUBJECT

SHIPPING CONTAINER 6-GS-1
STRESS CALCULATIONS

Appendix B

NAME P.Bringham/G.Wigle

DATE Rev. 2/74

4. BENDING STRESS

$$\bar{y} = \frac{\Sigma Ay}{\Sigma A} = \frac{\overset{A_1}{.5 \times .66} \times \overset{y_1}{.33} + \overset{A_2}{.75 \times .20} \times \overset{y_2}{.10}}{\underset{A_1}{.5 \times .66} + \underset{A_2}{.75 \times .20}} = \frac{.109 + .015}{.33 + .150}$$

$$= \underline{.259"}$$

$$I = (I_3 + A_3\, d_3{}^2) - (I_4 + A_4\, d_4{}^2)\ ,$$

$$= (\overset{I_3}{\frac{1.25 \times .66^3}{12}} + \overset{A_3}{1.25 \times .66} \times \overset{d_3{}^2}{.071^2}) - \overset{I_4}{\frac{.75 \times .46^3}{12}} - \overset{A_4}{.75 \times .46} \times \overset{d_4{}^2}{.171^2}$$

$$= .03 + .0042 - .0061 - .0100 = \underline{.0181}$$

$$S(\text{Bending}) = \frac{M}{I/C} \times K = \frac{PLC}{8I} K$$

P = 6256 lbs. ("F" for Lug B)

L = 1.062

C = .171 +.23 = .401

K = .83 (Ref: "Strength of Materials", Singer; Harper & Bros.)

$$S = \frac{6256 \times 1.062 \times .401 \times .83}{8 \times .0181} = \frac{2270}{.1448} = 15{,}271 \text{ psi}$$

Based on min. S_u, $F.S. = \frac{58{,}000}{15{,}271} = 3.8$

SHEAR STRESS in Weldment of Reinforcing Rings

$$S_s = \frac{P}{\text{Area}} = \frac{6256}{2[7.1" \times (.707 \times .25)]} = 2{,}492 \text{ psi}$$

$$F.S. = \frac{18{,}000}{2{,}492} = 7.2$$

5. SHEAR STRESS in Weld to Cask

Shear Y.P. = 19,000 psi (AWS "Handbook")

$$S_s = \frac{F}{A} = \frac{6256}{2 \times 1/4\ (3.1)} = 4150 \text{ psi};\ F.S. = \frac{19{,}000}{4{,}036} = 4.7$$

B. As LIFTING LUG

Similarly, substituting F = Wt. of packaging = 1840 lbs.

1. TENSILE STRESS at C - C

$$Sr = \frac{F}{\text{Area at C-C}} = \frac{1840 \text{ lbs.}}{2.5" \text{ X } .25"} = 2940 \text{ psi}$$

$$F.S. = \frac{36,000}{2,940} = 12.2$$

2. SHEAR STRESS in WELD to CASK

$$= \frac{1840 \text{ lbs.}}{2 \text{ X } 1/4 \text{ X } 2.5} = 1470 \text{ psi}$$

$$F.S. = \frac{19,000}{1,470} = 12.9$$

C. IMPACT EFFECTS

The lifting lugs extend .4" above the lid so that they will experience initial impact from a vertical drop. The lugs are designed so that they sit on the top 1" thick plate rather than penetrate it (see HCD-60596).

With this arrangement, lug welds absorb none of the impact energy. Full 3/4" thick lugs would transmit much of this energy to the top plate, resting on the outer and inner shells, and thence to the entire cask. Although this might not impair the cask integrity, it is preferred that full impact energy be absorbed by the lid and cask body. Accordingly, a section of the lifting lugs is narrowed to 1/4" thickness to insure their collapse under impact, permitting the lid and hence the cask to accept almost entire impact energy.

1. Lid Capacity

$$\frac{\text{Impact Energy of Package}}{\text{Area of Lid}} = \frac{30 \text{ ft. X } 1840 \text{ lbs.}}{233 \text{ in}^2} = 2840 \text{ inch-lbs./in}^2$$

This value corresponds on the curve to an impact depth in <u>unclad</u> gypsum-spodumene of .46". Mechanically, the lid and cask body can absorb full impact energy without impairing integrity of the package. Further, as seen from the G-S shielding curve (Appendix A) loss of .46" shielding produces only about 7% loss in attenuation.

2. Capacity of Lugs

Critical load per lug at B - B = S_y A = 36,000 X (3" X .25") = 27,000

lbs. for 4 lugs; load + 108,000 lbs.

(This is a conservative value, since section B-B is greater than at the top of the narrowed area.)

Load greater than this will collapse the 4 lugs.

Impact loading on the lugs will be significantly greater than 108,000 lbs. In our "Impact Stress Analysis", p. 34, we find that a K. E. of 279,000 inch-lbs. imparted to 15 lid studs produces a loading of

S X 15 X A/stud = 81,500 X 15 X .643 = 786,067 lbs.

A K.E., then, of 680,000 inch-lbs. (entire package dropping 30 ft.) will produce a loading in the four lugs far in excess of the required minimum 108,000 lbs.

3. Capacity of Top Plate

At their buckling loading the lugs will produce a shear stress in the 1" top plate:

$$S_s = \frac{27,000}{2(2" \times 1")} = 6,750 \text{ psi}$$

Shear yield ASTM A-36 = .5 X 36,000 = 18,000 psi

Lid will not deform under shear stress from lugs.

MATERIAL OF OUTER SHELL:

Pipe -ASTM A-53-B, Y.P. = 30,000 psi

24" O.D. X .25 Wall

(Ref: Cask Designer's Guide - ORNL; NSIC-68)

$$S = \frac{MC}{I}$$

$M = 5WL/8$

W = Weight of cask = 1840 lbs

L = Length of cask = 26" - 0.9" (Top Flange) = 25.1"

C = (for cylinder) outer radius = r_2

$$I = \frac{\pi}{4} \left(r_2^{\,4} - r_1^{\,4}\right)$$

r_2 = outer radius = 12."

r_1 = inner radius = 11.75"

$$S = \frac{5WLC}{8\left[\frac{\pi}{4}\left(r_2^{\,4} - r_1^{\,4}\right)\right]}$$

$$= \frac{5 \times 1{,}840 \times 25.1 \times 12}{2\pi \left(12.^4 - 11.75^4\right)}$$

= 263 psi

$$F.S. = \frac{Y.P}{S} = 26.8$$

SUBJECT
SHIPPING CONTAINER 6GS-1
Appendix B MAXIMUM ALLOWABLE PRESSURE, INTERNAL

NAME P. Bringham, G. Wigle
DATE Rev. February 1974

I. OUTER WALL - INTERNAL PRESSURE

STEEL = 24" OD X.250 wall pipe type A53 - grade B, seamless

T.S. = 60,000 psi

Allowable T.S. (S) = 12,650 psi (UCS-27)

(Ref: ASME Pressure Vessel Code, Sec. 8, 1968)

Weld Efficiency (E) = (see below)

Wall Thickness (t) = 0.25

Wall Radius (R) = 11.75

A. Circumferential Stress (Longitudinal Joint)

E = 1.0 (UW-12)

$$P = \frac{SEt}{R+0.6t} = \frac{12,650\ (1.0)\ (0.25)}{11.75 + .15}$$

= 265 psi

B. Longitudinal Stress (Circumferential Joint Single Welded Butt)

E = 0.60

$$P = \frac{2SEt}{R - 0.4t} = \frac{2 \times 12,650\ (0.6)\ (0.25)}{11.75 - 0.10}$$

= 326 psi

II. UPPER INNER WALL - INTERNAL PRESSURE

Circumferential Stress 14" O.D.; t = .25"; R = 6.75

$$P = \frac{12,650\ (1.0)\ (0.25)}{6.90}$$

= 456 psi

Longitudinal Stress

$$P = \frac{2 \times 12,650\ (0.6)\ (0.25)}{6.65}$$

= 570 psi

SUBJECT SHIPPING CONTAINER 6GS-1

NAME P.Bringham;G.Wigle

Appendix B INTERNAL PRESSURES DUE AMBIENT SOLAR HEAT

DATE Rev. February 1974

Assume T_1 = 70 °F = 530 °R Initial loading temperature

T_2 = 434 °F = 894 °R Maximum ambient internal temperature

P_1 = 1 atmosphere = 14.7 psia

$V_1 = V_2$ = constant

$$\frac{P_1}{T_1} = \frac{P_2}{T_2}$$

$$P_2 = \frac{P_1 T_2}{T_1} = \frac{14.7 \text{ X } 894}{530}$$

= 24.8 psia = 10.1 psig

SUBJECT: SHIPPING CONTAINER 6GS-1

NAME: P.Bringham, G.Wigle

Appendix B MAXIMUM ALLOWABLE PRESSURES - EXTERNAL

DATE: Rev. February 1974

Ref: ASME Pressure Vessel Code, Sec. 8, 1968, Fig. UCS-28.2

I. OUTER SHELL: Material - A53B; Y.P. = 30,000 psi

Outside Diameter, D_O = 24"

Thickness, t = .25"

Length, L = 25.1"

$L/D_O = 1.04$; $D_O/t = 96.$

From curve, $P(D_O/t) = 14,000$ (300°F ambient)

$P = 146$ psi

II. INNER SHELL (CAVITY), UPPER: Material - A-53B; Y.P. = 30,000 psi

D_O = 14" $L/D_O = .615$

t = .25"

L = 8.6" $D_O/t = 56.$

From curve $P(D_O/t) = 17,000$ (300°F ambient

$P = 304$ psi

III. INNER SHELL (CAVITY), LOWER: Material - C1015; Y.P. = 55,000 psi

Assume Y.P. such that curve UCS - 28.2 applies, Y.P. 30 - 38 Kips.

D_O = 11.5" $L/D_O = .87$

t = .25" $D_O/t = 46.$

L = 10."

From Curve $P(D_O/t) = 14,000$ (500°F ambient)

$P = 304$ psi

The allowable stress for assumed material already has F.S. = 10 supplied by the curve. Allowable stress for actual material is proportionate to yield points, so Factor of Safety is therefore

$$10 \times \frac{55,000}{38,000} = 14.5$$

PRESSURE, INTERNAL

During the half-hour fire, the outer layers of the gypsum-spodumene will lose their added water content in the form of steam. Steam thus generated will vent through either of two rupture discs, and the shielding section will revert to atmospheric pressure. Similarly destruction of the package outer lid gasket will permit venting of cavity to air. If the seal is not destroyed, the cavity still has rupture disc assembly to permit escape of pressurized air.

All of our heat transfer studies are on the extremely conservative side in that we have not allowed for two very important factors.

1. Heat of vaporization of water in the gypsum-spodumene mixture amounts to 2,300 - 11,000 Btu, depending on ambient equilibrium temperatures, whereas heat flow to the cavity during the half-hour fire is only 3,470 Btu (see pg. 50). (There will still be moisture present after the fire.)

2. Thermal conductivity of completely dehydrated gypsum is only about a third that of "wet" gypsum. We have used the latter, higher value throughout. Upon exposure to fire, then, internal temperatures will be considerably less than our tabulated values, depending upon the degree of dehydration.

DOSE RATES

For simplicity, if we assume the worst case, complete dehydration of the shielding mixture (i.e. gypsum becomes $CaSO_4 \cdot 1/2\ H_2O$) the neutron shielding will drop by a factor of 2 - 2.2 (see Fig. 2, Appendix A). Consequent neutron dose rate will increase to about 450 mR/hr at the package surface, well below the permissible 1000 mR/hr at 1 meter from the surface. This is a 55% loss of attenuation; i.e., the attenuation is 45% of what it was.

Coupled with the decrease in attenuation to 82% due to decreased shielding on impact (see p. 39) the anticipated attenuation will be .45 X .82 = .38 = 38% of the original shield, or a loss of 62% attenuation resulting in a surface dose of

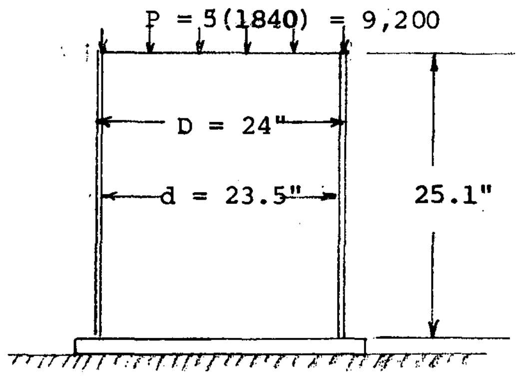
$$\frac{200}{.38} = 530 \text{ mR/hr.}$$

Gamma shielding will be maintained by appropriate internal uranium containers.

COLUMN LOADING (AEC 0529, Annex 1.)

Required: 5 times weight, end loading, vertical position
con sider external shell only as supporting load.

Fig. 7



$$K = \sqrt{\frac{24^2 + 23.5^2}{4}} = \sqrt{282.} = 16.7"$$

$$\frac{L}{K} \text{ (slenderness ratio) } = \frac{25.1"}{16.7} = 1.50$$

$$\frac{Q}{A} = \text{Allowable Stress} = \frac{\left[1 - \frac{(L/K)^2}{2C_c^2}\right]}{m} S_y$$, continuous column, welded ends*

For S_y = 36,000 psi, C_c = 126.1 = Allowable Load

m = Safety Factor

$$\frac{Q}{A} = \frac{1 - \frac{1.50^2}{2(126.1)^2}}{m} \text{ X } 36,000 \cong \frac{1}{m} \text{ X } 36,000 = \frac{36,000}{m}$$

Test Conditions: $\frac{Q}{A} = \frac{9,200 \text{ lbs}}{18.65 \text{ in}^2} = 493$ psi

So, Safety Factor $m = \frac{36,000}{493} = 73$

---

* Formula From: Roark "Formulas for Stress and Strain" 4th Ed. 1965

Impact on Corner of Cask

I. STUDS

The impact of the top corner of the cask from a drop height of 30 feet tends to accelerate the inner container which weighs 610 lbs and the lid of the primary container weighing 135 lbs independently of the primary container at time of impact. It is assumed that an "unyielding" body (uranium container) is allowed to drop onto an "unyielding" surface. The lid studs therefore would be forced to store the majority of the strain energy, and the quantity of energy that must be accounted for is the product of the drop height and the weight of the lid and cask contents.

$$U = h\ (W_L + W_i) = 268{,}000 \text{ in-lb}$$

where

$h = 360$ in
$W_L = 135$ lb
$W_i = 610$ lb

The cask lid is secured with 15 1-inch diameter studs. These studs have an unengaged length of 2 inches. The studs are made from 4140 series steel that has the idealized stress-strain curve shown in Fig. 1. The equation of the idealized curve is of the slope intercept form.[1]

$$S = M\varepsilon + S_{yp}$$

$$= \frac{27{,}000}{.27}\varepsilon + 63000$$

$$= 100{,}000\ \varepsilon + 63{,}000$$

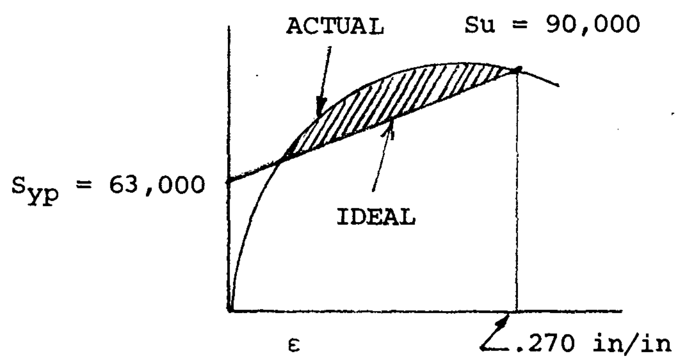


Fig. 8

[1]The stress-strain curve and analytical procedures from ORNL TM-2220 "Analysis of the ORNL Shipping Cask D-38." J. H. Evans

The total energy the 15 studs must absorb

$$U = \Sigma\int F\Delta = 15\,[F_{yp}\Delta + (F_s - F_{yp})\,\frac{\Delta}{2}\,]$$

$$\begin{aligned}
F_{yp} &= S_{yp}A;\ A = .643\ \text{in}^2 \\
F_u &= S_u A \\
F_s &= SA = (M\varepsilon + S_{yp})A \\
\Delta &= \varepsilon\ell \\
U &= wh
\end{aligned}$$

$$U = 15\,[S_{yp}A\varepsilon_s\ell + (M\varepsilon_s + S_{yp} - S_{yp})\,\frac{A\varepsilon_s\ell}{2}\,]$$

$$7.5A\varepsilon_s\ell\ (M\varepsilon_s + 2S_{yp}) - U = 0$$

$A = .651\ \text{in}^2$ (minor area of 1-14 stud)

$\ell =$ 2 in. unengaged or effective length with the thread machined off so the diameter for the two inch length equals the minor diameter of the thread. By necking the stud in this manner, the stress concentrations at the root of the threads are removed and the total strain energy absorption capabilities are increased approximately 25%.

$S_{yp} = 63{,}000$ PSI

$U = 268{,}000$ in-lb

$$7.5 \times .651 \times 2 \times \varepsilon_s\ (100{,}000\ \varepsilon_s + 126{,}000) - 268{,}000 = 0$$

$$\varepsilon_s^2 + 1.26\varepsilon_s - .27 = 0$$

$$\varepsilon_s = -\,\frac{1.26 \pm \sqrt{1.26^2 - 4(-.27)}}{2}$$

Disregarding the negative root by the quadratic formula

$\varepsilon = .185$, and the studs elongate .37"

The stress on each stud is therefore

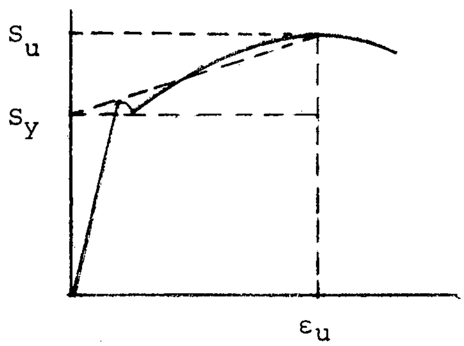
$$S = 100{,}000\ (.185) + 63{,}000 = 81{,}500\ \text{PSI}$$

(NOTE: In calculations of this type, the area between the Ideal and Actual curves is not included, so the calculated "S" is a conservative value.)

II. STUD RETAINING PLATE

The stud retaining plate (item 1 of HCD 60596) and all welds must be able to withstand stud stresses due to corner impact. An initial analysis shows that the plate is going to act inelastically; therefore, the standard elastic analysis would be invalid. The analyses I thru V herein, to be on the conservative side, will assume the full loading from a vertical axial impact being transferred through the lid onto the stud-retention plate.

The 268,000 in-lb of energy will be absorbed by localized deformations, generation of heat, and will act primarily within the plastic range of the material. It will be assumed that all of the energy of impact will be absorbed within the plastic range. The area of the stress-strain diagram (static) represents the energy absorbed per unit volume.



Material: A-36, HR
S_u = 58,000 psi (min.)
S_y = 36,000 psi (min.)
ε = 23% in 2"

U = Modulus of toughness, or energy absorbed/unit volume

$$= S_y \varepsilon_u + \left(\frac{S_u - S_y}{2}\right) \varepsilon_u = \left(\frac{S_y + S_u}{2}\right) \varepsilon_u$$

$$= 10{,}300 \frac{\text{in-lbs}}{\text{in}^3}$$

Minimum energy absorbing capability in the plastic range

E_{min} = UV, where V = Volume of plate material

$$V = \frac{\pi(24^2 - 14^2)}{4} \times 1 - \left[4\left(\frac{\pi \times 3^2}{4}\right) + 15\left(\frac{\pi \times 1^2}{4}\right)\right]$$

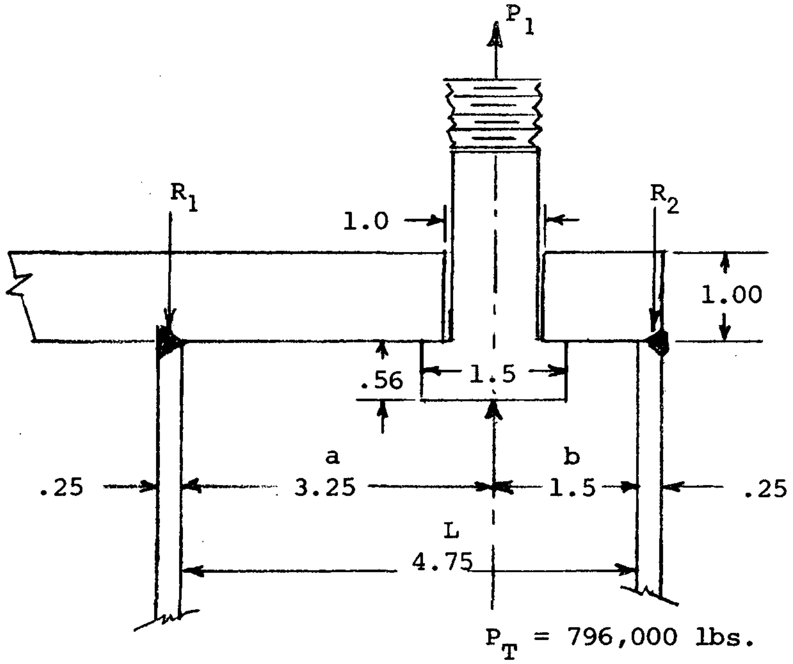
$$= 260 \text{ in}^3$$

$$E_{min} = 10{,}300 \times 260 = 2{,}580{,}000 \text{ in-lb}$$

The retaining plate therefore has a theoretical capability of absorbing 9.6 times the energy of impact if the plate does not break away from the main body of the cask. It has been previously reported and empirical data indicate that ferrous metals can sustain much higher yield and ultimate stresses over static values depending upon impact velocities.* This would, therefore, yield a greater area of the stress-strain diagram.

III. WELDS

Based on AWS standards, full-penetration weld strength is the same as for the base metal (item 7, HCD 61076) which is ASTM A53B steel, minimum ultimate strength = 58,000 psi.



P_T = 81,500 X .643 X 15
= 786,100 lbs.

P_1 = 53,000 lbs.

Figure 9

$$R_2 = \frac{Pa^2}{L^3}(L + 2b) \quad \text{(Roark)}$$

$$= \frac{7.86 \times 10^5 \ (3.25^2)(4.75 + 3)}{4.75^3} = 600{,}000 \text{ lbs.}$$

R_1 = 186 ,000 lbs.

Outer Weld

$$A_O = \frac{\pi(24^2 - 23.5^2)}{4} = 18.8 \text{ in}^2$$

$$S_O = \frac{600{,}000}{18.8} = 31{,}915 \text{ psi}$$

*NACA Technical Note #868

Inner Weld

$$A_I = \frac{\pi(14^2 - 13.5^2)}{4} \doteq 11 \text{ in}^2$$

$$S_I = \frac{186,000}{11} = 16,900 \text{ psi}$$

Welds will hold.

IV. STUD SHEAR-VERTICAL

$$S_s = \frac{P_1}{A_s} = \frac{53,000}{\pi(1)(.56)} = 30,000 \text{ psi}$$

Minimum yield, 4140 steel = 63,000 psi

V. RETAINING PLATE SHEAR

$$S_p = \frac{P_1}{A_p} = \frac{53,000}{\pi(1.5)(1)} = 11,200 \text{ psi}$$

VI. STUD SHEAR-HORIZONTAL

The cask lid assembly has been designed so that it is impossible for the lid to shear the studs.
The narrowed section of a stud is 0.905" diameter, and it's clearance hole in the lid is 1.250", so that the lid must move $\frac{1.250 - .905}{2} = .172"$ from the common center-line in order to even contact the stud.

On the other hand, maximum clearance between the lower clad shielding section of the lid, and the cask's cavity opening is $\frac{13.5 - 13.2}{2} = .150"$

The lid, then, is restrained from sliding far enough to contact the studs.

Impact on Bottom Face of Cask

The impact on the bottom face of the cask is of concern mainly from the possibility of losing shield thickness by compression of the gypsum-spodumene should the 1/4" X 6" steel bars supporting the bottom of the inner cavity collapse. The pallet directly beneath is less rigid than these bars and, therefore, is not of significant concern. The impact of the cask when dropped from a height of 30 feet tends to accelerate the inner container at time of impact. In order to minimize compression of the shielding, we will assume that items 2 and 3 (HCD 61076) are forced to store the majority of the strain energy or drop energy, U.

$U = hw = 220{,}000$ in-lb

$h = 360$ in

$w = 610$ lbs

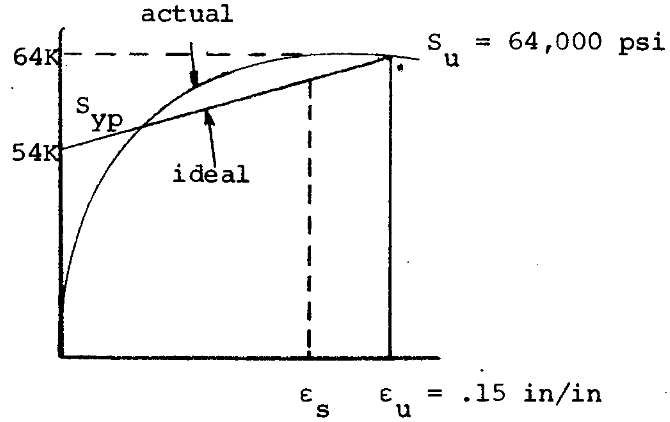
The 1/4" X 6" bars are cold-rolled steel, SAE 1018

Yield stress = 54,000 psi

Ultimate stress (minimum) = 64,000 psi

% elongation = 15

Stress-strain curves for various impact velocities* show that the percent elongation for a given impact relocity is in most cases greater than the percent elongation under static loading. We will use the static value for percent elongation which will give a conservative maximum stress value.



If the stress exceeds the proportional limit, the relations for tension and compression remain essentially the same. In case of compression the member must be short or constrained against lateral buckling. Based on this assumption, the same analogy will be used for the supporting member (item 2, 3) as was used for the lid retaining bolts.

*Donald S. Clark, The Influence of Impact Velocity on the Tensile Characteristics of Some Aircraft Metals and Alloys, NACA Technical Note No. 868, 10/42

The equation of the idealized curve is of the slope intercept form

$$S = M\varepsilon + S_{yp}$$

$$= \frac{10{,}000}{.15}\varepsilon + 54{,}000$$

$$= 66{,}700\ \varepsilon + 54{,}000$$

The total energy the steel bars must absorb:

$$U = \Sigma\int F\Delta = F_{yp}\Delta + (F_s - F_{yp})\frac{\Delta}{2}$$

$$F_{yp} = S_{yp}A$$

$$F_s = SA = (M\varepsilon + S_{yp})A$$

$$\Delta = \varepsilon_s\ell$$

$$U = wh = 220{,}000 \text{ in-lb}$$

$$A = 5.63 \text{ in}^2$$

$$\ell = 6''$$

$$S_{yp} = 54{,}000 \text{ psi}$$

$$U = S_{yp}A\varepsilon_s\ell + (M\varepsilon_s + S_{yp} - S_{yp})\frac{A\varepsilon_s\ell}{2}$$

$$= A\varepsilon_s\ell\left(S_{yp} + \frac{M\varepsilon_s}{2}\right) = 220{,}000 \text{ in-lb}$$

$$0 = \frac{AM\varepsilon_s^2\ell}{2} + A\varepsilon_s\ell S_{yp} - 220{,}000$$

$$\doteq \frac{5.63 \times 66{,}700 \times 6\varepsilon_s^2}{2} + 5.63 \times 6 \times 54{,}000\ \varepsilon_s - 220{,}000 \text{ in-lb}$$

$$\varepsilon_s^2 + 1.62\varepsilon_s - .195 = 0$$

$$\varepsilon_s = .110$$

The compressive stress in the bars is, therefore

$S = 66{,}700\ (.11) + 54{,}000 = 61{,}337$ psi
Average deformation $\Delta = \varepsilon\ell = .11 \times 6 = .66$ inches
Reduction in shielding by .66", or 1.7 cm, decreases the attenuation by about 18%, i.e. the attenuation is now 82% of what it was, with a consequent maximum neutron dose rate at the bottom surface of 240 mR/hr.
Gamma is almost unaffected.

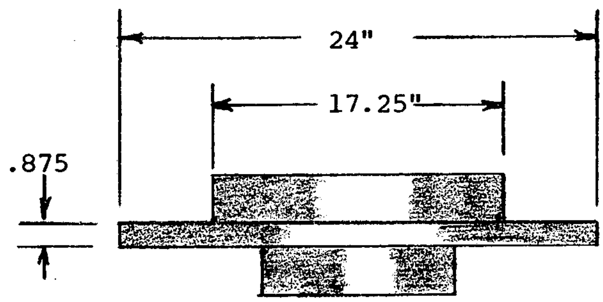


Figure 11

Following the same procedure as used with the Stud Retaining Plate (page 36), and the lid flange of the same material as the Retaining Plate, we can calculate the energy absorption capability of the lid flange only, not considering additional absorption of the upper shielding disc.

Volume of Lid Plate (to edge of upper disc) =

$$\left\{\frac{\pi}{4}[24^2 - 17.25^2 - (1.25^2 \times 15)] - (1.25 \times 3.3 \times 4)\right\} \times .875$$

O.D. I.D. Holes Slots Thickness

$$= 160 \text{ in}^3$$

$$E_{min} = U_r V = 10{,}800 \frac{\text{in-lbs}}{\text{in}^3} \times 160 = 1{,}730{,}000 \text{ in-lbs.}$$

So absorption capability of lid in plastic range is 6.4 times impact load (268,000 in-lbs).

U_r is the modulus of toughness, represented by the total area of a stress-strain diagram, and is the energy absorbed per unit volume.

$$U_r = \frac{S_y + S_u}{2}\varepsilon_u = \frac{36{,}000 + 58{,}000}{2} \times .23 = 10{,}800 \frac{\text{in-lb}}{\text{in}^3}$$

Ref: "Strength of Materials", Singer. (Harper & Brothers)

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 too 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 (Fig. 12) 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 .8" deep impact crater, and the results from such differential analysis are shown at the right hand of the graph. The extreme right hand column is, of course, the total energy absorbed by total impact depths of from .1" to .8". The left column represents the difference between adjacent values of the right column, that is $\Delta E/in^2$ for successive .1" intervals. These are the values used in calculating the impact depth of the package when dropped on its side.

The positive curvature, or "variable compressibility" is interpreted as a <u>continuous</u> 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.

Appendix B

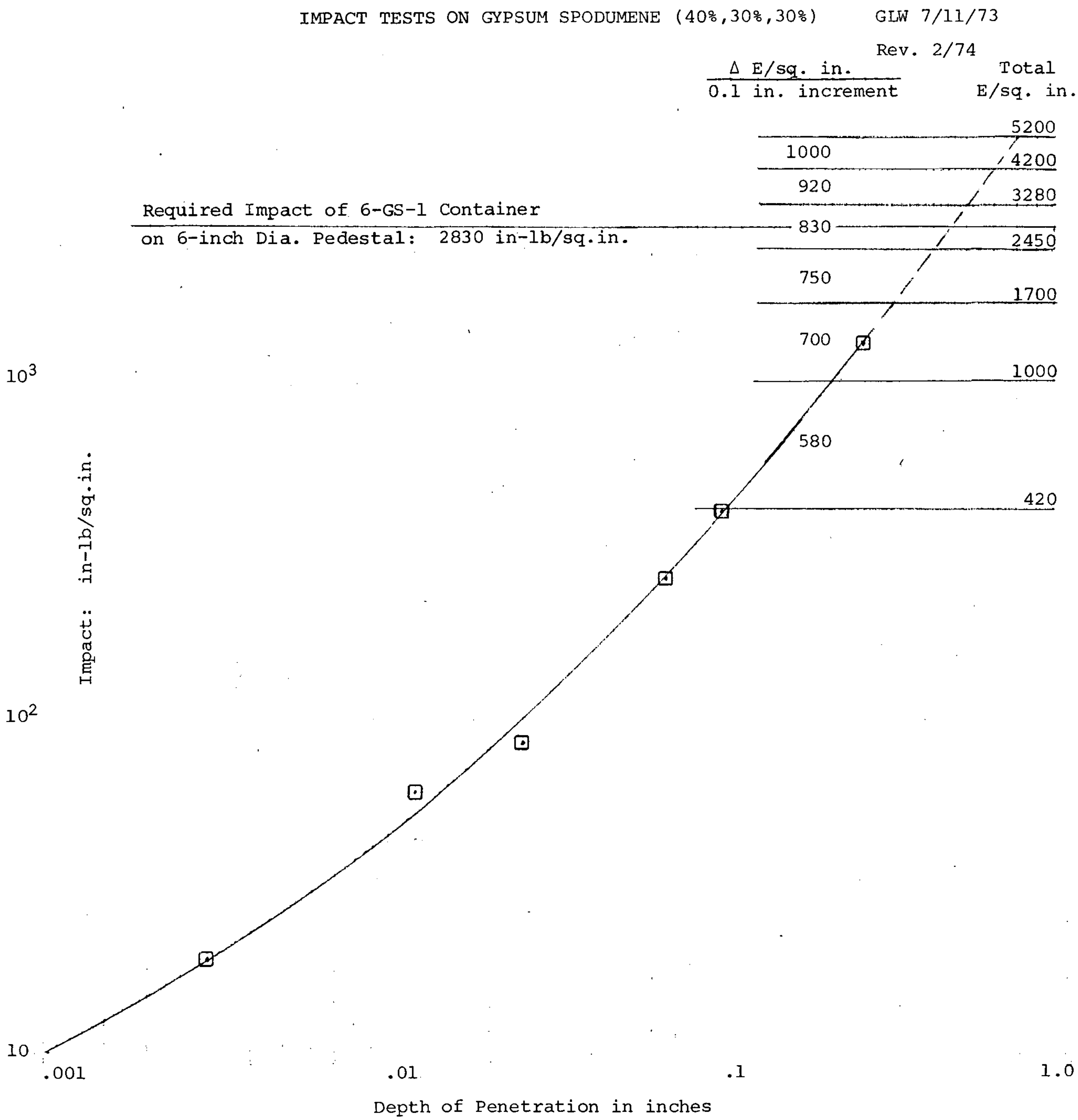


Fig. 12. Impact tests on gypsum-spodumene (40%, 30%, 30%).

Calculate depth of impact of gypsum-spodumene from 30-foot drop of entire cask landing longitudinally (on its side).

Assume impact depth represents total loss of that thickness of shielding, instead of compaction.

Ignore energy absorption effect of 1/4" steel cladding, except only as a means of containment of compacted shielding.

Ignore fixed-end effects of top and bottom cask plates, but use full cask height of 26" (including thickness of top plate).

E ≐ Energy to be absorbed = drop height X weight of cask

= 360" X 1,840 lbs = 6.8×10^5 inch-lbs.

We can approximate the depth of penetration, or the extent of flattening, of the cylindrical surface by calculating in incremental steps the changing area of the flattened portion as the depth, and chord, increases. Then by selecting from the impact curve the energy per unit area required for that particular depth, and multiplying this by the incremental area we obtain the energy in inch-pounds absorbed by the increment compaction.

Finally, by adding the resulting incremental energies, we continue until the total integrated energy equals the total kinetic energy, 6.8×10^5 inch-lbs.

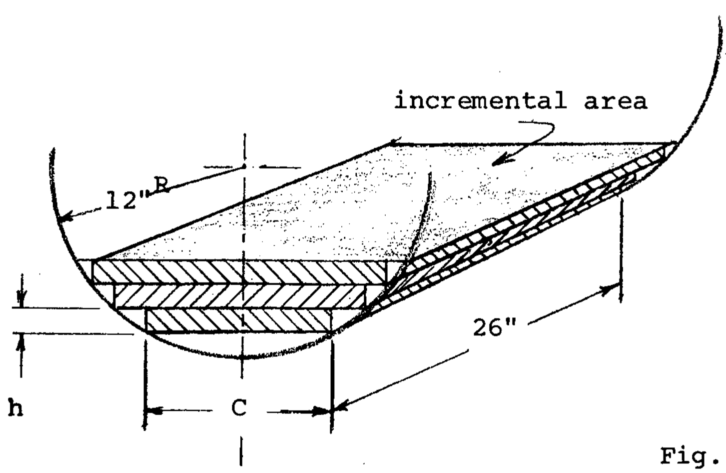


Fig. 13

If we take h in 0.1" increments we have the following table:

h in.	C in.	A (=26C) in^2	$\Delta E/in^2$ from curve	$(\Delta E/in^2) \times A$ (X 10^3)	Σ E (X 10^3)
.1	2.87	74.5	420.	31.	
.2	4.37	114.	580.	66.	97.
.3	5.40	140.	700.	98.	195.
.4	6.23	162.	750.	121.	316.
.5	6.84	178.	830.	148.	464.
.6	7.40	192.	920.	176.	640.
.7	8.03	209.	1000.	209.	849.
.63	7.68	200.	100.	20.	662.

So the total compression depth along the side of the cask will be 0.63 inches. This is a very conservative approximation. Taken in smaller increments, the total compression depth will be considerably less, since the edges of the shielding increments will then be increasingly effective in absorbing the kinetic energy.

Surface Temperature of Container - In Sun

Conditions: Still air at 130°F and maximum sunshine
References: 1. Brown and Marco "Introduction to Heat Transfer," 3rd Ed. 1958
2. Shappert "Cask Designers Guide" ORNL - NSIC - 68

The heat input to the container surface from solar radiation and source decay is equal to the heat loss to the air.

At maximum surface exposure to the sun, the cask cylinder is tipped 45° from normal to the sun. Cylinder dimensions = 2.17' X 2' diameter.

Q_s = Solar Input = AeI (1.)

A = Exposed Area
= .707[(2.17 X 2) + (3.14 X 1)] = 5.27 ft^2

e = surface absorptivity for solar radiation

= .25 (white enamel)

I = Average incident solar radiation on a normal surface (Ref. 2., p. 143)

= 144 Btu/hr - ft^2

Q_s = (5.27)(.25)(144) = 190 Btu/hr

Assuming the total heat loss to air is by convection,

Heat loss to air:

$Q_c = CR(a)^{1/3} (T_s - T_a)^{4/3} A$ (Ref. 1., Art. 8) (2.)

Where C = Coefficient for horizontal cylinder (Ref. 1.)

= .45

R = Thermal conductivity of air

= .0177 Btu/hr - ft - °F

a = Thermal property of air ($=g\beta\rho^2 C_p/\mu K$)

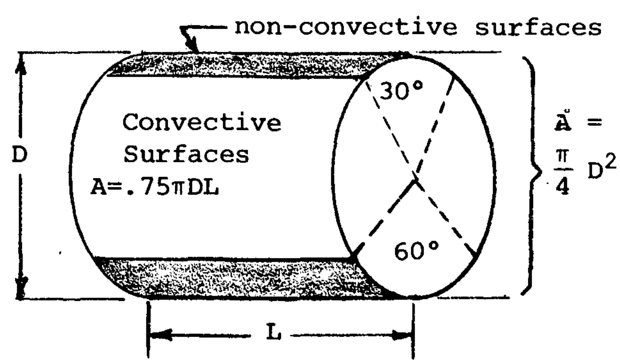
= $0.68 \times 10^6/ft^3$ - °F; $a^{1/3} = 87.9$

T_s = Surface Temperature of Container

T_a = Air Temperature = 130°F

A = Convective Area
$= .75\pi(2)(2.17) + \frac{\pi 2^2}{4}$

= 13.4 ft^2



Let Q_n = heat output of nuclear material, Btu/hr

Then $Q_c = Q_s + Q_n = 190 + Q_n$

Substituting for Q_c, and with Q_n as a variable, solve for values of T_s

$$Q_n + 190 = (.45)(.0177)(87.9)(T_s - 130)^{4/3} \quad (13.4)$$

$$T_s = \left(\frac{190 + Q_n}{9.38}\right)^{.75} + 130$$

Values of T_s vs. Q_n are charted and plotted in Fig. 14.
Use the left ordinate and upper abcissa system for surface temperatures in direct sun.

Temperatures Inside of Container

We shall limit the temperature of the inside surface of the gypsum-spodumene, T_2 to somewhat less than the saturation temperature at 75 psig, the pressure at which the pressure relief rupture discs release.

The saturation temperature is 320°F (Keenan "Steam Tables")

We therefore limit T_2 to 310°F maximum.

In so doing, we limit the nuclear output Q_n, which in turn determines a new value of T_s, other than 180°F used in the "heat loss to air" equation. The new T_s is substituted in the following equation to provide a second Q_n, and so on. By this method of double back-filling, we arrive at a value of Q_n and its consequent T_s that satisfy both equations.

$$Q = [2\pi K_g(T_2 - T_s)]/(D_2^{-1} - D_o^{-1}) \qquad (3.)$$

K_g = Thermal conductivity of Gypsum-Spodumene

= 0.28 (a conservative value, for fully hydrated mix)

T_2 = Inside temperature of Gypsum-Spodumene

= 310°F maximum

T_s = Outside surface temperature, obtained from equation #2

D_2 = Diameter of sphere of equivalent inside area = 1.26 ft.

D_o = Diameter of sphere of equivalent outside area

= 2.06 ft ($\pi D_o^2 = 13.4$)

$Q = Q_n$ to be substituted into equation #2

$Q_n = [2\pi(0.28)(310-163.5)]/(1.26^{-1} - 2.06^{-1})$

= 829 Btu/hr (243 watts)

The maximum permissible nuclear heat output is then 243 watts with a consequent maximum surface temperature of 163.5°F under conditions of still air, full sunlight, and air temperature of 130°F.

Temperature of Gamma Shield

Since the gamma shield is constructed from depleted uranium, good thermal conductors, the temperature throughout should be essentially uniform. The rate of heat transfer across the gap separating the inner and outer vessel is:

$$Q = \sigma F_a F_e A (T_1^4 - T_2^4)$$

Where σ = Stephan Boltzmann Constant

$= 0.173 \times 10^{-8}$ Btu/hr - ft^2 - °R^4

F_a = Configuration Factor

= 1.0

A = Surface area of side and one end of inner vessel, (uranium) (Items 1 + 2 + 3, Fig. 17)

= 3.4 ft^2 (10.56" O.D. X 13.25 H.)

T_1 = Temperature of inner vessel

T_2 = Temperature of outer vessel (gypsum-spodumene)

= 310°F (770°R)

Q = Heat Transfer Rate

= 829 Btu/hr

F_e = Emissivity factor

$= [1/e_1 + 1/e_2 - 1]^{-1}$

Where: e_1 = emissivity of inner vessel

= 0.50 (partially weathered stainless sheet)

e_2 = emissivity of outer vessel

= 0.95 (white enamel)

$F_e = [1/0.50 + 1/0.95 - 1]^{-1} = (2 + 1.05 - 1)^{-1}$

= 0.49

Heat Transfer Rate:

$Q = (0.173 \times 10^{-8})(1.0)(0.49)(3.4)[T_1^4 - (770)^4]$

= 829 Btu/hr

T_1 = 894 °R

= 434 °F

Surface Temperature of Container in Shade

Conditions: Still air at varying temperatures; maximum surface temperature (T_s) = 122°F.

Letting $Q_s = 0$, then $Q_c = Q_n$

T_a = Temperature of air, °F.

$$T_a = 122 - \left(\frac{Q_n}{CR(a)^{1/3} A}\right)^{.75} = 122 - \left(\frac{Q_n}{(.45)(.0177)(87.9)(13.4)}\right)^{.75}$$

$$= 122 - \left(\frac{Q_n}{9.38}\right)^{.75}$$

With Q_n as variable, obtain corresponding values of T_a.

Results are shown in Fig. 14; use right-hand ordinate and lower abcissa system for T_a in the shade.

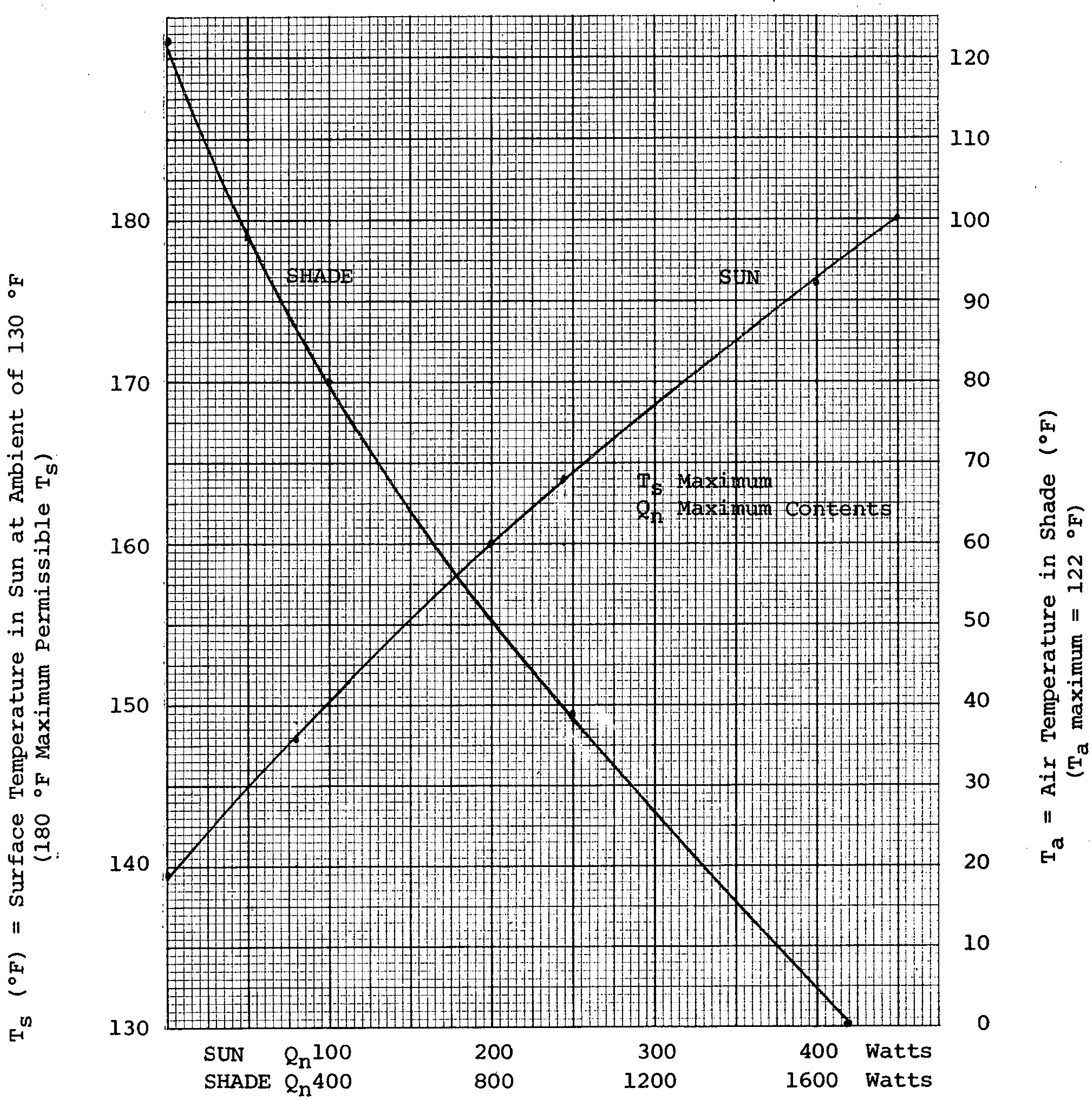
SUBJECT

SHIPPING CAPACITY vs. AMBIENT TEMPERATURE

Appendix B

NAME P.Bringham/G.Wigle

DATE Rev. February 1974



	Q_n Watts		0	80	200	244	400	450	1000	1637
	Q_n Btu/hr		0	273	682	833	1364	1534	3410	5584
SUN	T_s	°F	139	148	160	164	176	180	---	---
SUN	T_2	°F	---	---	---	310	---	---	---	---
SUN	T_1	°F	---	---	---	434	---	---	---	---
SHADE	T_a	°F	122	109	97	93	80	76	39	0

Figure 14

I. HEAT FLUX TO INNER CAVITY OF GYPSUM-SPODUMENE

$$Q = 2\pi\ K_g\ (T_S - T_1)/D_2^{-1} - D_O^{-1}$$

K_g = .28 Btu/hr - ft - °F

T_S = Temperature of outer surface gypsum-spodumene, assume 1500 °F

T_1 = Temperature of inner surface gypsum-spodumene, assume 434 °F Due to nuclear & solar heat, p. 47

D_O = Outside diameter of equivalent sphere = 2.06 ft

D_2 = Inside diameter of equivalent sphere = 1.26 ft

$$Q = 6.28 \times .28(1500 - 434)/1.26^{-1} - 2.06^{-1}$$

$$= 1867/.31$$

$$= 6023 \text{ Btu/hr}$$

II. TEMPERATURE OF INNER CAVITY OF GYPSUM-SPODUMENE AT END OF 30 MIN. FIRE

The influx of 6,023 Btu/hr., or 3,012 Btu for 30 minute duration of fire, will produce an averate ΔT at the interior.

$$\Delta T = \frac{\dot{Q}}{W_s\ C_s + W_g\ C_g}$$

W_s = weight of steel = 610 lbs. (including base)

W_g = weight of gypsum-spodumene = 620 lbs.

C_{ps} = specific heat capacity steel = .125 Btu/lb. - °F

C_{pg} = specific heat capacity gypsum = .259 Btu/lb. - °F

$$\Delta T = \frac{3012}{610(.125) + 620(.259)}$$

$$= 12.7\ °F \sim 13°$$

$T_1 + \Delta T = 434 + 13 = 447$ at the inner cavity wall. This does not allow for additional ΔT's across the two .25" steel cladding thickness, and it assumes initial (maximum) heat flow rate to be constant throughout the 30 minute period. No further thermal calculations were made to allow for heat capacity of uranium internal containers.

To find effect of some loss of shielding thickness from impact on thermal insulation of container. The heat input through the impacted area will be greater than that through an equal area of the non-impacted. This ΔQ, added to Q for the entire cask will represent the Q_I for the impacted cask.

Impact area 6" diameter = .20 ft^2

Impact depth = .5" (see p. 42 in Appendix B)

Remaining shield thickness (minimum) = 4.75 - .5 = 4.25" = .35'

K_g = Thermal conductivity gypsum-spodumene = .28 Btu/Hr - ft - °F

Q = Heat input to impacted area

$$= \frac{K_g A(T_O - T_I)}{.35 \text{ ft}} = \frac{.28 \times .20\ (1500 - 447)}{.35}$$

T_O = 1500 °F = Temperature of outer surface

T_I = 447 °F = Temperature of inner surface after fire

= 168 Btu/hr

Q_2 Heat input to equal area non-impacted cask (4.75" or .395' shielding)

$$= \frac{.35'}{.395'} \times 168 = 149 \text{ Btu/hr}$$

ΔQ = 168 - 149 = 19 Btu/hr or 9.5Btu/30 minutes

ΔT in interior cavity = $\frac{(3012 + 9.5)}{3012} \times 13$ °F $\doteq$ 13.04 °F temperature rise at impacted area, compared to 13 °F rise at non-impacted area

Therefore, loss of .5 inch of shielding due to impact on upright rod has negligible effect on temperature in cavity during 1/2 hour fire exposure.

An internal temperature of 447 °F has been calculated for the end of the 30-minute exposure to fire. Thermal lag, however, will produce an additional internal temperature after removal of the cask from the fire. We can very closely estimate this ΔT in the following manner.

We have empirical data taken from an actual 30-minute fire test of this Laboratory's wooden overpack DOT SP5872. (Tests were performed at Underwriter's Laboratory, Chicago.) The temperature curve, included here as fig. 18, shows the thermocouple #2 reading taken behind 3.5" wood, with a temperature rise, after fire shutdown, of 30 °F. Interior cavities in the wood, and the gypsum-spodumene shielded casks are the same, since both are designed to hold the same uranium Spec. 55 containers.

By comparing heat transfer characteristics of the two casks, we can calculate the anticipated ΔT thermal lag of the present gypsum-spodumene interior.

First, find Q_W = Heat input to interior of <u>wood</u> cask.

K_W = 0.1 Btu/hr - ft^2 - °F/ft

ΔT_W = 1475 - 70 °F = 1405°F; (1475 °F = 800C, T_S = 1475, T_1 = 70)

A_W: Diameter = 18.5" = 1.54 ft
Height = 28.5" = 2.38 ft

= 11.1 ft^2

Δx = 3.5" = .28 ft

$$Q_W = \frac{(0.1)(1405)(11.1)}{.28} = 5570 \text{ Btu/hr to inside of cask}$$

For simplicity, and to provide a conservative answer, we use Q_W as the rate, at the time of furnace shutdown, of heat input to the center until T_1 reached maximum 7 minutes later: Δt = 7 minutes

So Btu necessary to increase inner temperature of the wood cask after furnace shutoff = 5570 X 7/60 = <u>650</u> Btu for a 30 °F temperature rise.

Now, heat inputs to the two casks are proportional to their respective $\frac{KA}{\Delta x}$. Again assuming a <u>constant</u> heat flow inward,

K_G = .28 Btu/hr - ft - °F

A_G = 13.4 ft^2

Δx_G = 6" = 0.5 ft

$$\frac{KA}{\Delta x_G} = \frac{.28 \times 13.4}{.5} = 7.5 \text{ for gypsum-spodumene}$$

$$\frac{KA}{\Delta x_W} = 3.97 \text{ for wood}$$

So heat input to G-S interior = $\frac{7.50}{3.97}$ X 650 Btu = 1220 Btu.

But the temperature rise, ΔT, after furnace shutdown

$$= \frac{\text{Btu}}{W_s C_{ps} + W_G C_{pG}}$$

Using the previous values (p. 50)

$$\Delta T = \frac{1220}{610(.125) + 620(.259)} = \frac{1220}{237} = 5.1\ °F$$

The maximum temperature, then, at the interior of the gypsum-spodumene cask will be

434° (maximum ambient, nuclear + solar)

13° (ΔT during fire)

5° (ΔT "heat lag" after fire shutdown)

452°F maximum

I. Gas Evolution

Cask will be used to ship alpha-emitting nuclides. Maximum allowable of ^{244}Cm is 10 grams, which produces .0965 atmos. cc He/day. Assuming 100 days for a shipment (including storage, etc.) total He evolved is 9.65 cc at a pressure of 29.4 psi at 70 °F.

II. Heat

Maximum final temperature after fire = 452 °F = 912 °R

III. Assuming a minimum free volume of 10 cc in the Special Form Container, temperature of 452 °F produces an internal pressure of

$$\frac{29.4 \text{ psia}}{530 \text{ °R}} = \frac{P_2}{912 \text{ °R}}$$

P_2 = 50.5 psia or 35.8 psig

Temperatures and pressures are well within safe working limits of primary container.

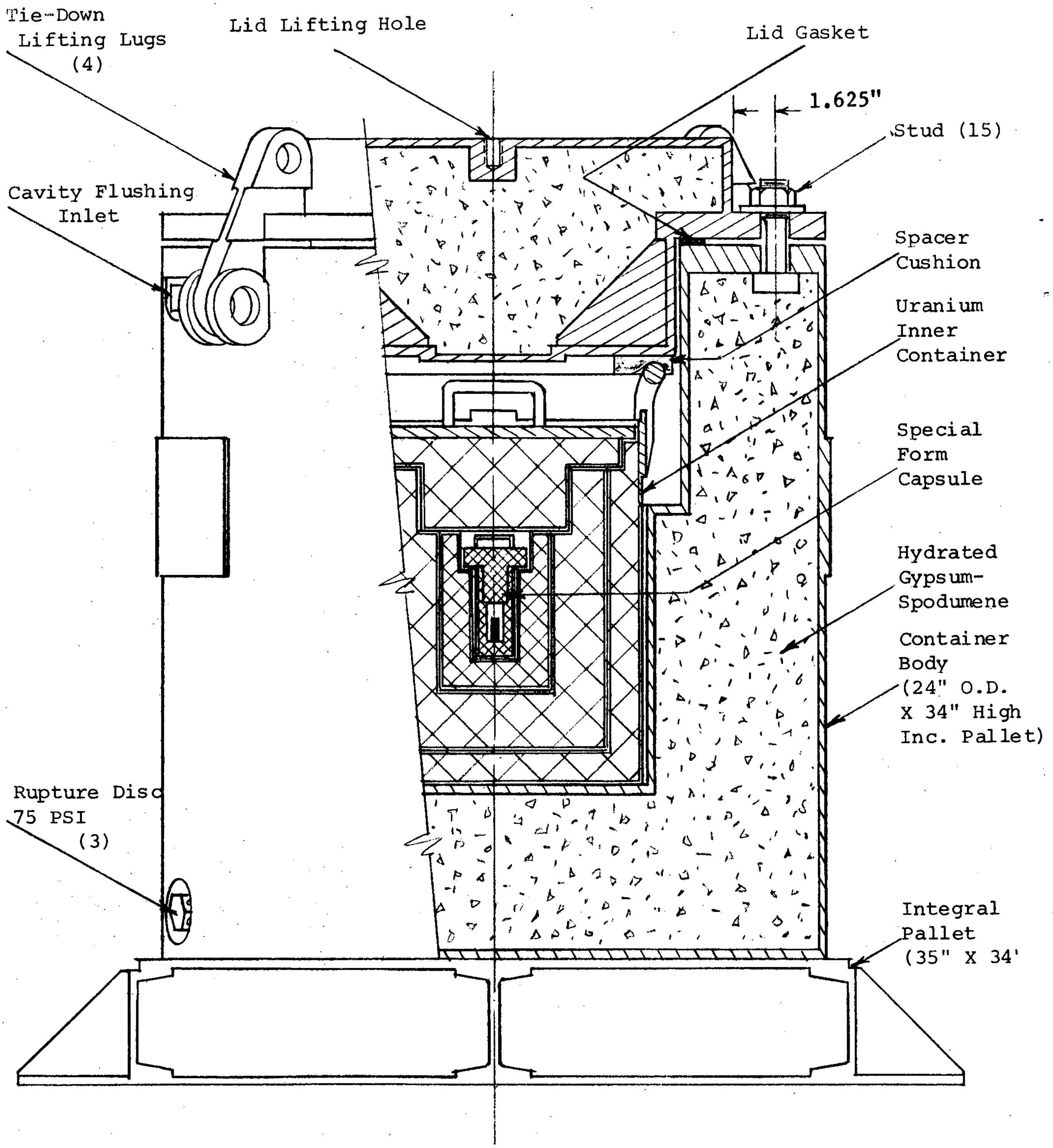


Fig. 15 Fully Assembled Model 6GS-1 Shipping Container

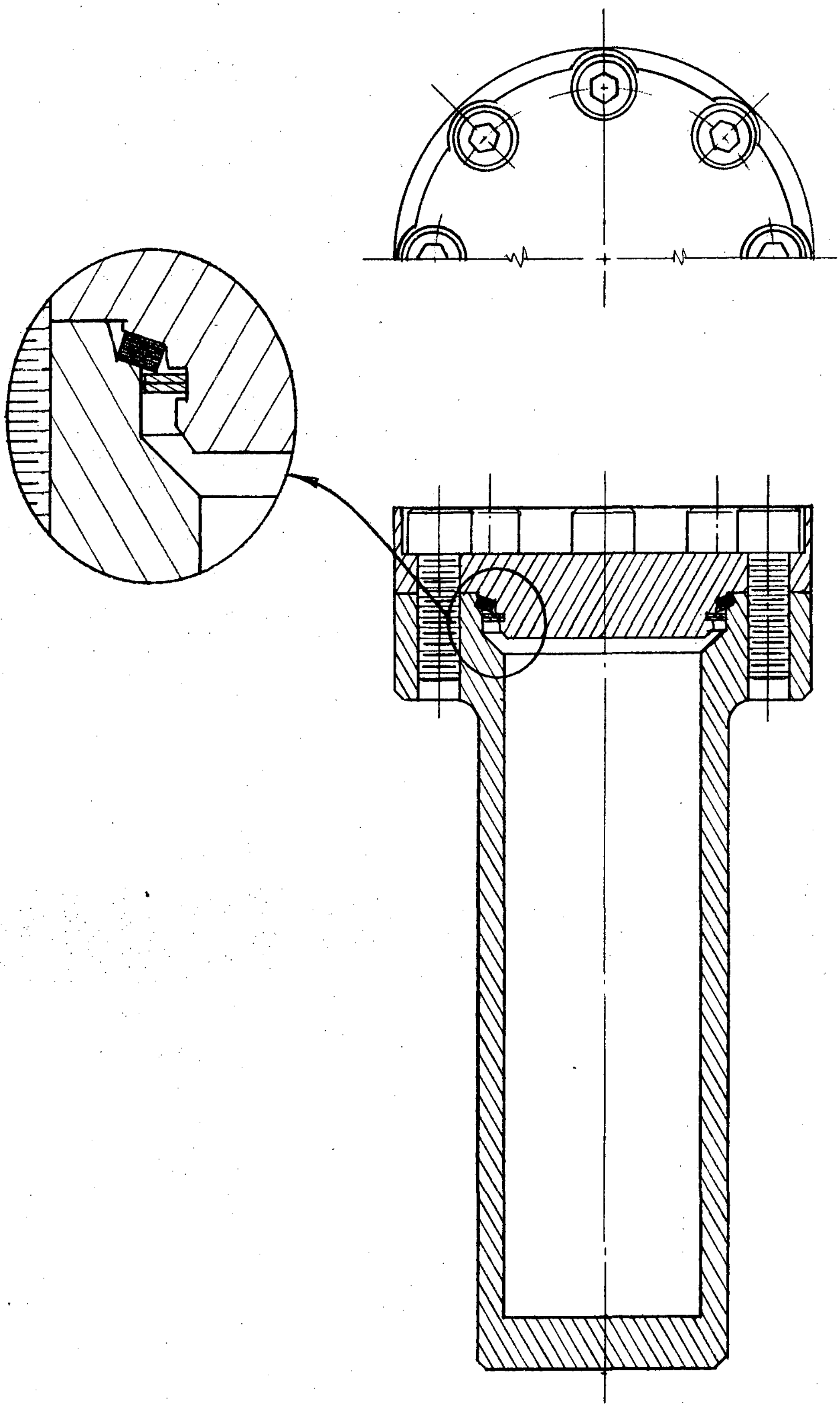


Fig. 16 "SPECIAL FORM" Capsule

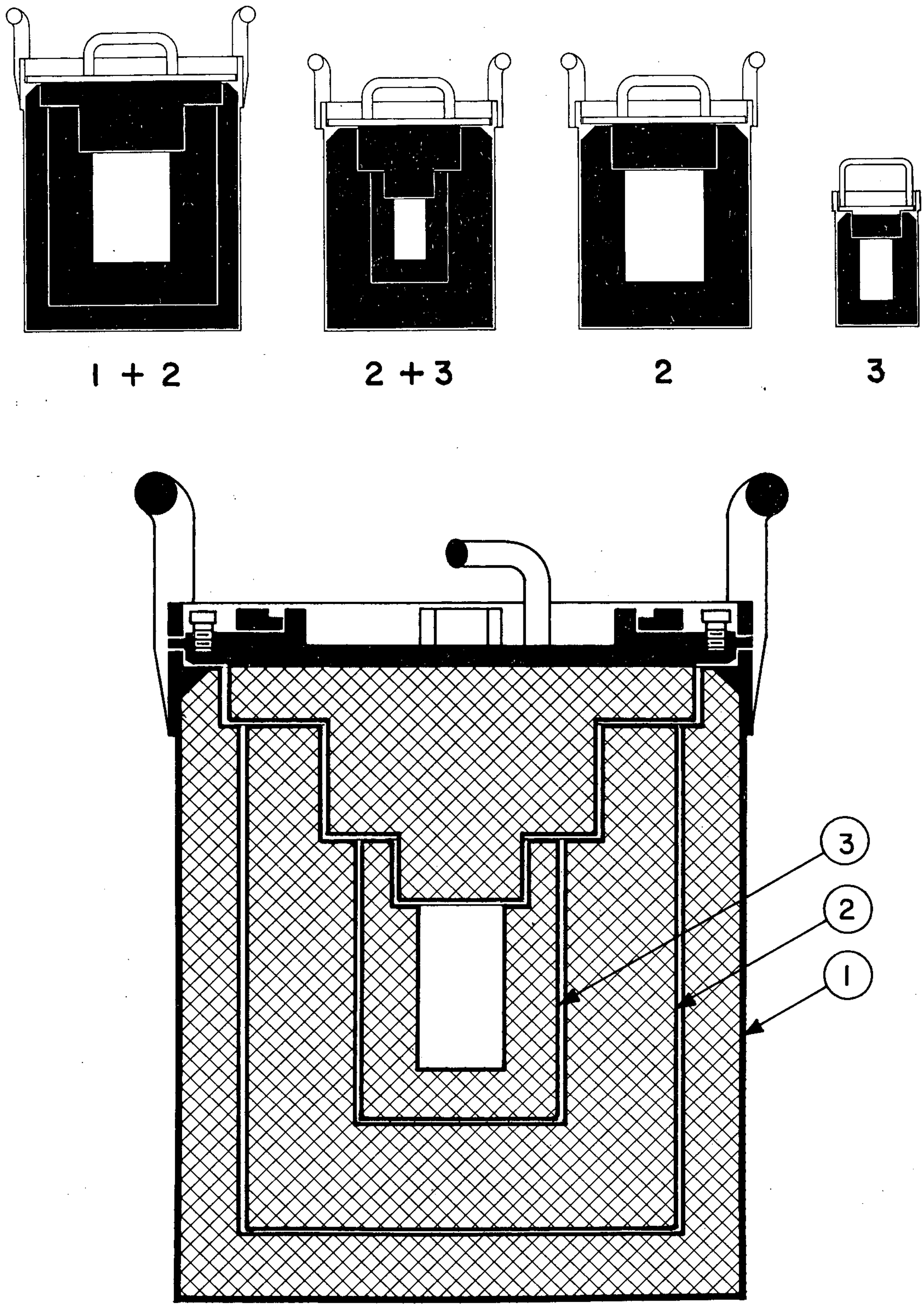


Fig. 17 Nested Uranium Shielded Container Assemblies

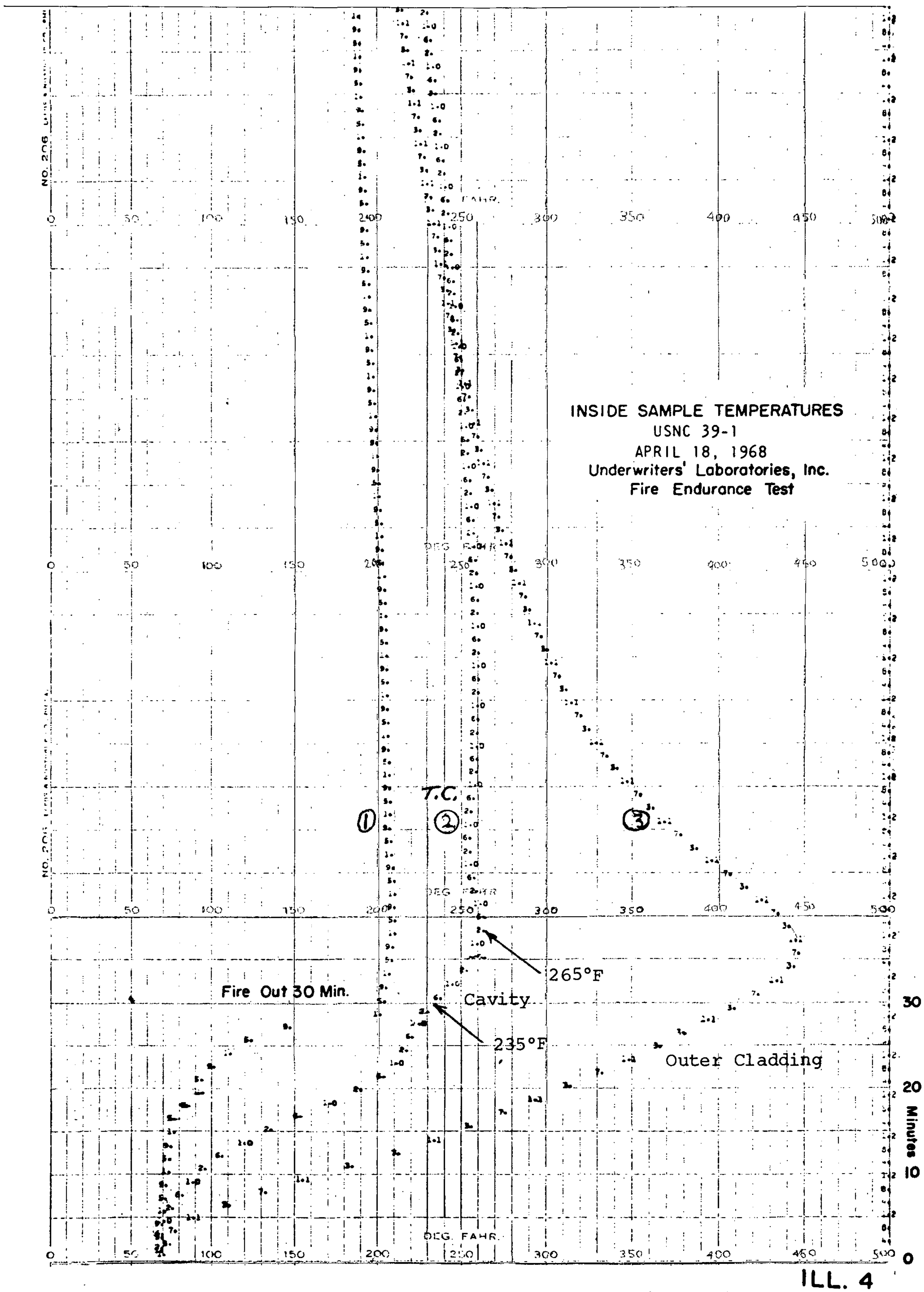


Fig. 18 Thermocouple Temperatures During Fire Test of Wood Overpack

Appendix C.

U.S. ATOMIC ENERGY COMMISSION AEC MANUAL SELECTIONS FROM CHAPTER 0529

Safety Standards for the Packaging of Radioactive and Fissile Materials

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

   a. If there is a system of lifting devices which is a structural part of the package, the system shall be capable of supporting 3 times the weight of the loaded 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 3 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 package 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 package shall be so designed that failure of the device under excessive load would not impair the containment or shielding properties of the package.

4. Tie-down Devices

   a. If there is a system of tie-down devices which is a structural part of the package, the system shall be capable of withstanding, without generating stress in any material of the package in excess of its yield strength, a static force applied to the center of gravity of the package having a vertical component of 2 times the weight of the package with its contents, a horizontal component along the direction in which the vehicle travels of 10 times the weight of the package with its contents, and a horizontal component in the transverse direction of 5 times the weight of the package with its contents.

   b. If there is a structural part of the package 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 tie-down device which is a structural part of the package shall be so designed that failure of the device under excessive load would not impair the ability of the package to meet other requirements of this section A.

B. Structural Standards for Large Quantity Packaging

Packaging used to ship a large quantity of radioactive material, as defined in I, A.6., 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 ends 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 guage.

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 the 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; and

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, and

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 a large quantity of radioactive material, as defined in I, A.6., above, shall be so designed and constructed and its contents so limited that under the normal conditions of transport specified in annex 1, 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 increase 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; and

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 so designed and constructed and its contents so limited that under normal conditions of transport, specified in annex 1, below, 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.1., I.1.a., or J.1., 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; and

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 on which nuclear safety is assessed;

(2) reduction by more than 5 percent in the effective spacing on which nuclear safety is assessed, between the center of the containment vessel and the outer surface of the packaging; or

(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 a large quantity of radioactive material as defined in I., A. 6., above, shall be so designed and constructed and its contents so limited that under the normal conditions of transport specified in annex 1, below, 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 a large quantity of radioactive material, as defined in I, A.6., above, or the shipment of fissile material when the package will contain more than .001 curie of Group I radionuclides, .05 curie of Group II radionuclides, 3 curies of Group III radionuclides, 20 curies of Group IV or Group V radionuclides, or radionuclides in special form, or 1000 curies of Group VI or Group VII radionuclides shall be so designed and constructed and its contents so limited that if subjected to the hypothetical accident conditions specified in annex 2, below, as the Free Drop, Puncture, Thermal, and Water Immersion conditions, in the sequence listed 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 3 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 I, A.7., above, and is transported on a motor vehicle, railroad car, aircraft, inland water craft, 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 hypothetical accident conditions specified in annex 2, below, as the Free Drop, Puncture, Thermal, and Water Immersion conditions, in the sequence listed 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; and

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., 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; and

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; and

2. two hundred and fifty such packages would be subcritical in any arrangement, if each package were subjected to the hypothetical accident conditions specified in annex 2, below, as the Free Drop, Thermal, and Water Immersion conditions, in the sequence listed 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; and

b. twice that number of such packages would be subcritical in any arrangement if each package were subjected to the hypothetical accident conditions specified in annex 2, below, as the Free Drop, Thermal, and Water Immersion conditions, in the sequence listed in annex 2, with close reflection by water on all sides of the array 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. 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 of 1., 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 Fissile Class III shipment shall be so designed and constructed and its contents so limited, and the number of packages in a Fissile Class III shipment shall be 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; and

2. the shipment would be subcritical if each package were subjected to the hypothetical accident conditions specified in annex 2, below, as the Free Drop, Thermal, and Water Immersion conditions, in the sequence listed in annex 2, with close reflection by water on all sides of the array and with the packages in the 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.

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 1 1/2 and 2 1/2 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.

Free Fall Distance

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

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 1 1/4 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.

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 4

TESTS FOR SPECIAL FORM MATERIAL

1. Free Drop - A free drop through a distance of 30 feet into 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 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 D

Quality Assurance Program

Design and Procurement Responsibility

I. Container Materials Procurement

With the exception of gypsum-spodumene and the lid bolts, all materials used in the outer cask are stock items at Lawrence Berkeley Laboratory. As such, they must meet Laboratory Engineering Standard References (ESR) specific for each material. ESR specifications for all metals used are equivalent to either SAE or ASTM standards. Lid bolt material was received certified as required.

Similarly, materials for the Special Form capsule, including the bolts, were certified as specified.

II. Fabrication

A. Outer Cask: Fabricated at the Laboratory, welds are full penetration. Inspected throughout fabrication for workmanship and dimensional tolerance. Shielding of complete package checked with radioactive contents.

B. Uranium containers (Spec. 55): Uranium cast and machined at Oak Ridge, no voids present. Clad in stainless steel at Lawrence Berkeley Laboratory, inspected and passed for workmanship and dimensional tolerance.

C. Special Form Capsule: Designed by Gamah Division of Stanley Aviation Corp., materials are certified as per specification. Finished capsule assemblies are inspected for workmanship and dimensional tolerance by this Laboratory.

USER'S RESPONSIBILITY

III. Container Inspection, prior to each shipment.

A. Special Form Capsule

1. Sealing surfaces, body and lid: scratches, pits, cleanliness ____
2. Edges of mating faces of flanges: dents, protrusions ____
3. Internal screw threads: thread condition, cleanliness ____
4. Sealing ring: scratches, dents, cleanliness ____
5. Screws (8): thread condition, cleanliness ____

Send any items of questionable condition along with a complete assembly to Engineering Group for examination and leak check.

B. Uranium Container

NOTE - Uranium inserts, outer shell, and lid are not interchangeable. Use only "A" items for "A" assembly, etc.

1. Outer steel shell: general condition, lug integrity ____
2. Uranium cladding: integrity, especially internal swipe check for contamination ____
3. Sealing surfaces: inserts and lid, scratches, cleanliness, flatness ____
4. Rubber gasket: cleanliness, cracks, pits, etc. ____
5. Lid latches: dents, protrusions ____

C. 6-GS-1 Cask

1. Internal cavity, cleanliness, dryness, exhaust port unobstructed ____
2. Sealing surfaces, lid and body: scratches, dents, cleanliness ____
3. Sealing gasket: cleanliness, imperfections ____
4. Lid studs: not bent, threads clean and in good condition ____
5. General: integrity of lifting lugs, pallett, body and lid ____

IV. Loading Procedure

A. Special Form Capsule

1. Assemble the capsule lid: use a new metal seal and secure it with a retaining ring.

2. Place elastomer spacer disc in bottom of capsule.

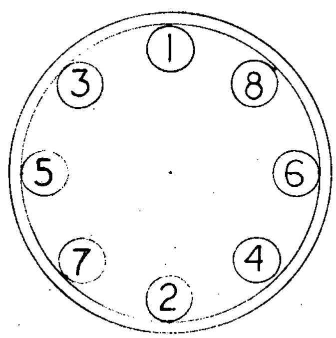
3. Place vial in capsule.

4. Place spacer disc on top of vial. (Top spacer may be temporarily cemented to bottom of lid to facilitate loading and unloading.)

5. Set the base of the capsule in the two locating pins of the wrench fixture.

6. Place the lid on the capsule and screw in all 8 lid bolts "finger tight".

7. Torque the bolts, 1/2 turn at a time, in the order shown. Torque to 75 inch-pounds.



8. Take swipe samples of outside of capsules. There must be no detectable activity.

B. Loading Uranium Cask

1. Make sure there are appropriate spacers in the cask cavity.

2. Place the capsule in the spacers, bottom end down.

3. With the elastomer gasket in place on the sealing surface of the cask, gently lower th lid onto the gasket, with the latches next to their lugs.

4. Tap the four latches into their lugs as far as they'll go. Secure by tightening each latch with its bolt; this provides additional sealing pressure.

c. Loading 6-GS-1 Cask

1. If the largest size (size#1) uranium container is not used, provide appropraite spacers in the cavity of the outer cask.

2. Set the uranium container in the spacers and place the appropraite shim on top.

3. Place the lid on the outer cask, carefully, to avoid damaging stud threads.

4. Tighten down the lid nuts in a sequence similar to the capsule (IIA7). When to lid is fully sealed, its outer edge is about]/8" - 3/16" above the cask edge.

5. Thread a safety seal wire throught the studs, and clamp it with a lead seal.

6. Remove the eyebolt from the cask lid.

V. Transporation

A. Lifting

The cask may be lifted with a fork lift, with the forks through the built-in pallet, or by means of the lifting lugs.

1. If the latter, use minimum <u>3/8</u>" wire rope (one rope per shackle) and a 1/2" shackle through each of the four lugs.

2. Center the cask with respect to the sides of the truck bed, and the pallet parallel to the truck sides.

B. Tie-down

As tie-downs, use either (1) 3/4" wire rope, and 7/8" shackles if they are used, in the lower hole of each tie-down lug; or (2) tie-down chain of a load strength of at least 6430 pounds.

C. Placards

Attach appropriate placards to the truck.

APPENDIX E

DEPARTMENT OF TRANSPORTATION
Hazardous Materials Regulations Board
Washington, D.C. 20590

SPECIAL PERMIT NO. 6550

This special permit is issued pursuant to 49 CFR 170.15 of the Department of Transportation (DOT) Hazardous Materials Regulations, as amended, and on the basis of the June 30, 1969, petition by the University of California, Lawrence Radiation Laboratory, Berkely, California, as amended in the supplements submitted to the USAEC, Division of Materials Licensing, dated January 19, 1971, February 25, 1971, May 2, 1971, July 2, 1971 and August 17, 1971.

Standard special permit requirements and conditions relating to package markings, preparation of shipping papers, shipping experience reports, etc., are published in 49 CFR 171.6. These requirements are part of this special permit.

1. Shipments of Type B quantities of radioactive materials, n.o.s., are hereby authorized in the packaging as described in this special permit and as further prescribed herein. This packaging, when constructed and assembled as prescribed herein, with the contents as authorized herein meets the standards prescribed in the DOT Regulations, Sections 173.394(b)(3) and 173.398(c).

2. Each shipper, under this permit, other than the petitioner named above, shall register his identity with this Board prior to his first shipment, and shall have a copy of this permit in his possession before making any shipment.

3. The packaging authorized by this permit consists of the Model 6-GS-1 Neutron Shipping Container system. It consists of two separate shields, an outer gypsum-spodumene neutron shield and an inner depleted uranium gamma shield. Shipments are in either of two configurations--either as the outer gypsum-spodumene filled cask alone with special internal mounting racks for the containment vessel within the cask cavity; or with the uranium-shielded cask within the outer cask. Contents must be loaded within an innermost containment vessel which meets the requirements of "special form" (§173.389(g)). The overall dimensions of the package, which is a cylindrical shape, are 2' diameter by 25" high, mounted on a 34" square structural steel pallet base. The inner cavity of the outer shield is nominally 11" diameter by 15" deep, with a bolted top plug type closure. The inner gamma shield consists of three nestable stainless steel uranium metal cylinders (DOT Spec. 55) providing a range of 1.58"-8.21" diameter by 3.03"-8.71" high inner cavity sizes. The outer wall of the outer shield is equipped with 75 psig rupture discs. The inner gamma shield

Continuation of SP 6550

must be positioned and supported within the outer cask cavity by means of wooden supports, as required. Spacers must be provided as required to prevent movement of contents. The package is further described on the following drawings (Lawrence Radiation Laboratory):

HCD 55322-19A	HCD 60071	8J8573D
HCD 55322-11	HCD 60081	8J8583A
HCD 56952	HCD 60091	8J8684D
HCD 58291	HCD 60596C	13J6212
HCD 60061	HCD 63464	

The weight of the package is either 2020 pounds (with inner gamma shield) or 1380 pounds (without inner gamma shield).

4. The contents of each package authorized by this permit consist of not more than 5000 curies of radioactive material containing not more than 75 watts thermal decay energy, meeting the requirements for "special form," and further limited to not more than 15 grams of fissile radioactive material, except that the maximum total mass of any one of the following isotopes may not exceed:

Am-242m	0.2 grams	Cf-251	0.06 grams
Cm-243	3.0 grams	Np-237	40 grams
Cm-245	0.5 grams	Pu-240	150 grams
Cm-247	2.4 grams	Am-241	115 grams
Cf-249	0.4 grams	Am-242	6.0 grams

Combinations of fissile isotopes or those as limited above are authorized, provided that the sum, for all isotopes present, of the ratio between the weight of each isotope, or combination, as appropriate to the permissible weight of that isotope (or combination) does not exceed unity.

5. The outside of each package must be plainly and durably marked "DOT SP 6550" and "TYPE B", in connection with and in addition to the other markings and labels prescribed by the DOT Regulations.

6. Each package must have its gross weight plainly and durably marked on the outside of the package.

7. This permit authorizes shipments only by cargo-only aircraft, motor vehicle and rail. For shipments by air, a copy of this permit, kept current, must be carried aboard any aircraft transporting radioactive material under these terms.

8. Prior to each shipment authorized by this permit, the shipper shall notify the consignee of the dates of shipment and expected arrival.

9. Any incident involving loss of contents of the package must be reported to this Board at the earliest feasible moment.

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10. The permit does not relieve the shipper or carrier from compliance with any requirement of the DOT Regulations, except as specifically provided for herein.

11. This permit expires on October 31, 1973.

Issued at Washington, D.C.:

for W.R. Fiste
For the Administrator
Federal Highway Administration

0 OCT 1971
(DATE)

for Mac E. Rogers
For the Administrator
Federal Railroad Administration

13 October, 1971
(DATE)

Ellis C. Langford
For the Administrator
Federal Aviation Administration

14 OCT 1971
(DATE)

Address all inquiries to: Secretary, Hazardous Materials Regulations Board, U.S. Department of Transportation, Washington, D.C. 20590. Attention: Special Permits.

Dist: a, c, d, e, h, i

ACKNOWLEDGEMENTS

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Eileen Doyle for the explicit artwork.