71-9216

SAFETY ANALYSIS REPORT

FOR

CNS MODEL 1-13G SHIPPING CONTAINER

Revision 0

December 1987

Submitted By:

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1.0 GENERAL INFORMATION

1.1 INTRODUCTION

This Safety Analysis Report describes a reusable shipping package designed to protect radioactive material from both normal conditions of transport and hypothetical accident conditions. The package is designated as the Model CNS 1-13G capable of transporting Fissile Class III, solid metal or metal oxide by-product material.

1.2 PACKAGE DESCRIPTION

1.2.1 Packaging

A. Cask and Overpack

A stainless steel encased, lead shielded cask for fissile, solid metal or metal oxides by-product material. The cask is a right circular cylinder 68 inches high by 38-1/2 inches in diameter. The cask has a cavity which is 54 inches high by 26-1/2 in diameter. The cask side wall consists of 1/2 inch thick outer steel shell, a 5 inch lead shell and a 1/2 inch thick inner steel shell. The outer base of the cask is comprised of a 1/8-inch steel shim plate and backing ring and a 1/2 inch steel plate welded to form a 5/8 inch thick base which is integrally welded to the outer steel shell of the side wall. The cask lid is a lead-filled flanged plug. The cask closure is sealed by a silicone rubber gasket. Positive closure is accomplished by six one inch diameter studs. The cask is equipped with a cavity drain line and plug.

The cask is provided with two lifting ears bolted and seal welded to the cask body 180° apart. The closure lid is provided with a single lifting lug. A double-walled steel cylinder protective overpack encloses the cask during transport and is bolted to a steel pallet with 2-inch diameter studs. The carbon steel protective overpack is equipped with two rectangular lifting loops and two tie-down ears. The cask packaging weight is approximately 25,500 pounds.

B. Liners

1. Lead (CNS Dwg. C-110-B-06402-004)

A steel encased lead shielding liner which is a right circular cylinder with a hollow center. The liner has a 26-1/4 inch outside diameter and is 59 inches high (with lid). The liner wall consists of a nominal .38 inch 304 stainless steel inner shell, a 2-1/4 inch lead shell and a nominal .38 inch outer stainless steel shell. The liner lid consists of two 2 inch thick plates. Closure is made with a 1/2 inch diameter steel rod bail with a 24 inch circular diameter. The lead liner weighs approximately 5,150 pounds.

2. Polyethylene (CNS Dwg. C-110-B-06402-004)

A neutron source shield liner which is a right circular cylinder with a hollow center cavity. The liner has a 26 inch outer diameter and is 44-3/4 inches high. The inner cavity is 2-1/4 inches in diameter by 22-1/4 inches deep. The liner is equipped with an 8 inch diameter by 11-1/2 inch stepped plug. The cylinder walls are made from nominal .12 inch thick 6061 aluminum. The top and bottom plates on the lid and the bottom of the cylinder are .25 inches thick. The top plate of the cylinder is .50 inches thick. Shielding is provided by 5% borated polyethylene filler. Two 3 inch by 3 inch by 1/2 inch ears with 1-1/2 inch diameter connections are used for lifting. The liner weighs approximately 750 pounds.

3. Auxiliary Shield (CNS Dwg. 8651-E-02)

Body. The auxiliary shield body is a solid vessel with a cylindrical shape and an internal cavity. The body external dimensions are nominally 30.5 inches in height by 24 inches in diameter. The internal cavity, formed by the lid and the body, is nominally 14 inches high with a 7.62 inch diameter. The body is of all welded-steel construction with encapsulated chemical grade lead used as the primary shielding material. The shield wetted surfaces (exterior and cavity) are epoxy painted for corrosion resistance and ease of decontamination. The steel wall thickness in the radial direction is 0.5 inches for both the inner and outer walls. The upper steel head is fabricated from two, 2 inch layers of steel plate. The shield bottom head is fabricated from 2 inch thick steel plate. The lead shield thickness varies from 5.5 to 7.18 inches to develop an equivalent lead thickness of approximately 7-3/4 inches. A stepped drain line extends from the cavity bottom to the side wall. The drain line is normally plugged. The sealing flange has eight 3/4-10 studs; two taper pins for lid positioning, and a groove to accomodate a 3/8-inch diameter rubber "O" ring. The "O" ring configuration is comparable to that used to seal other radioactive material. The body weight is approximately 4,900 lbs. A 3-point sling is used to lift the shield assembly.

Lid. The lid is fabricated from a 2 inch steel plate and encapsulated lead. It is recessed into the shield body. The stud/nut closure and "O" ring gasket in the body provide the sealed joint. Two alternative lids will be used depending on the contents. In the first lid design, provision has been made to alternatively load the shield cavity through a port in the lid. This port is formed by removal of a stepped lead plug and a threaded cap. This is an operating simplification and would only be utilized when loading small hardware components. The lid has two 1/4 inch diameter vent lines with plugged ends angled into the steel head. The second lid design, which will be employed with larger components, does not have a stepped plug. The lids are lifted with a 3-point handling sling. The weight of both shield lids is nominally 200 pounds. Note that no attachments on the exterior of the assembled lid and shield body extend more than about one inch above the lid.

1.2.2 Operational Features

No special operational considerations are required for the CNS 1-13G cask.

1.2.3 Contents of Packaging

A. Plutonium in excess of twenty (20) curies per package must be in the form of metal, metal alloy, or reactor fuel elements; and

(i) By-product material and special nuclear material as solid metal or oxides. Decay heat not to exceed 600 watts. The radioactive material shall be in the form of fuel rods, or plates, fuel assemblies, or meeting the requirements of special form radioactive material.

500 gm U-235 equivalent mass; or

(ii) Neutron sources meeting the requirements of special form radioactive material.

500 gm U-235 equivalent mass. Decay heat not to exceed 50 watts; or

(iii) Irradiated PuO_2 and UO_2 fuel rods clad in zircalloy of stainless steel. Decay heat not to exceed 600 watts. All fuel rods shall be contained within a closed 5-inch Schedule 40 pipe with a maximum useable length of 39-5/8 inches.

1,200 gm fissible material with no more than 300 gm fissile material per 5 inch Schedule 40 pipe.

(iv) Process solids, either dewatered, solid, or solidified in a secondary sealed container meeting the requirements for low specific activity radioactive material.

(v) Solid nonfissile irradiated metal hardware, reactor control rods (blades), and reactor start-up sources; and segmented boron carbide tubes (tube contents not to exceed a Type A quantity).

(vi) Radioactive (Hot Cell) waste materials immobilized with cement grout and contained in a 55-gallon (or extended 55-gallon drum) DOT Specification 17H or 17C steel drum, lid and closure. The waste material must be packaged in accordance with the Procedural Outline of the Immobilization of Cell Waste Using Cement Grout, Attachment D to this application. The cement grout must be at least 50 volume percent (estimated) of the drum contents and relatively uniformly distributed throughout the drum. At least 3/4 inch thick layer of grout must cover all radioactive waste contents. Decay heat not to exceed 100 watts, and fissile material not to exceed 500 grams U-235 equivalent mass.

B. Neutron Sources

A poylethylene liner is available for the shipment of neutron sources within the CNS Model 1-13G. An analysis of the polyethylene liner is included as Attachment A to this application.

C. Irradiated Metal Hardware

An auxiliary shielded inner container is available for shipment of irradiated metal hardware within the CNS Model 1-13G. An analysis of the auxiliary shielded container is included as Attachment B to this application.

1.3 APPENDIX

1.3.1 Chem-Nuclear Systems, Inc. "As Built" Drawings

1.3.1.1 The CNS 1-13G Cask is fabricated in accordance with the following CNSI Drawings:

- Drawing No. C-110-B-06402-001, Rev. A
- Drawing No. C-110-B-06402-002, Rev. A
- Drawing No. C-110-B-06402-003, Rev. C
- Drawing No. C-110-B-06402-004, Rev. A

1.3.1.2 A CNS 1-13G auxiliary shielded inner container and shoring plug are fabricated in accordance with CNSI Drawings No. 8651-E-02, Rev. A and 8651-C-01, Rev. B.

FIGURE WITHHELD UNDER 10 CFR 2.390

CHEM-NUCLEAR SYSTEMS, INC.	TITLE 1-13G SHIPPING CONTAINER
	C-110-B-06402-001 A

FIGURE WITHHELD UNDER 10 CFR 2.390

4-21-97

SCALE —

CHEM-NUCLEAR SYSTEMS, INC.

TITLE
I-13G
OVERPACK

C-110-B-06402-002 A

FIGURE WITHHELD UNDER 10 CFR 2.390

C	PER ECD # C-110-B-06402-003-03	CHEM-NUCLEAR SYSTEMS, INC.	TITLE 2-13G SHIPPING CASK
B	PER ECD # C-110-B-06402-003-02		
A	PER ECD # C-110-B-06402-003-01		C-110-B-06402-003 C

SCALE —

FIGURE WITHHELD UNDER 10 CFR 2.390

SCALE —

CHEM-NUCLEAR SYSTEMS, INC.

TITLE 1-13G CASK LINERS

C-110-B-06402-004 | A

FIGURE WITHHELD UNDER 10 CFR 2.390

Q.A. [illegible]	DATE 2-22-84	CHEM-NUCLEAR SYSTEMS, INC.	TITLE CASK COMPOSITE ASSY.
DRAWN M. J. AHEARN	DATE 11/7/83		
CHECKED [illegible]	11-7-83		DRAWING NUMBER 8651-C-01
APPROVED K. T. [illegible]	11-16-83		REV. B

FIGURE WITHHELD UNDER 10 CFR 2.390

CHEM-NUCLEAR SYSTEMS, INC.	
CASK AUXILIARY SHIELD	
Drawn: Mike Anderson	10-31-83
Material: SEE NOTE 6	
Scale: 1/2" = 1"	Size: E
Drawing No.: 8651-E-02	Rev.: A

2.0 STRUCTURAL EVALUATION

2.1 STRUCTURAL DESIGN

2.1.1 Discussion

The package has been designed to provide a shielded containment vessel that can withstand the loading due to the normal conditions of transport, as well as those associated with the hypothetical accident conditions.

The CNS 1-13G is designed to protect the payload from the following conditions: transport environment, 30-foot drop test, 40-inch puncture test, 1475^{o}F thermal exposure and transfer or dissipation of any internally generated heat. The design of the package satisfies these requirements.

Principal structural members of the system consist of:

- Overpack and Base

- Containment Vessel

These components are identified on the drawings in Appendix 1-3.

2.1.2 Design Criteria

The CNS Model 1-13G has been designed to be a simple, strong package that will provide maximum flexibility for multiple usage, as well as minimize exposure to operating personnel.

There are no components of the packaging or its contents which are subject to chemical or galvanic reaction; no coolant other than naturally circulating air is used during transport. A lock wire and seal of a type that must be broken if the package is opened is affixed to the package closure. If that portion of the protective jacket (overpack) which is used in the tie-down system or that portion which constitutes the principal lifting device failed in such a manner to allow the package to separate from the tie-down and/or lifting devices, the basic protective features of the protective jacket and the enclosed cask would be retained. The package (contents, cask and protective jacket) regarded as a simple beam supported at its ends along its major axis, is capable of withstanding a static load, normal to and distributed along its entire length equal to five times its fully loaded weight, without generating stress in any material of the packaging in excess of its yield strength. The packaging is adequate to retain all contents when subjected to an external pressure of 25 pounds per square inch gauge. The method for determining static loads was originally submitted by General Electric for their G.E. Model 100. This analysis has been included as Attachment C to this application.

The calculative methods employed in the design of the protective jacket are based on strain rate studies and calculations on a literature search of the effects on materials under impact conditions. The intent was to design a protective jacket that would not only satisfy the requirements of the U.S. Nuclear Regulatory Commission and the Department of Transportation prescribing the procedures and standards of packaging and shipping, and the requirements governing such packaging and shipping, but would also protect the shielded cask from significant deformation in an accident. In the event that the package was involved in an accident, a new protective jacket could be readily supplied and the shipment continued with minimal delay.

The effectiveness of the strain rate calculations and engineering intuitiveness in the design and construction of protective jackets was demonstrated with the General Electric Shielded Package - Model 100. The protective jacket design for the CNS Model 1-13G was scaled from the design of the Model 100 in accordance with cask weight and dimensions, maintaining static load safety factors greater than or equal to unity, and in accordance with the intent to protect the shielded cask from any deformation in the event of an accident.

2.2 WEIGHTS AND CENTERS OF GRAVITY

Cask Assembly

• Body	~17,100 lbs.
• Lid	~ 2,000 lbs.
	~19,100 lbs.

Lead Liner

• Body	~ 4,500 lbs.
• Lid	~ 650 lbs.
	~ 5,150 lbs.

Polyethylene Liner

• Body & Lid	~ 750 lbs.

Auxiliary Shield

• Body	~ 4,900 lbs.
• Lid	~ 200 lbs.
	~ 5,100 lbs.

Overpack Assembly

- Baseplate ~1,400 lbs.
~5,000 lbs.
~6,400 lbs.

The center of gravity of this package is located at the geometric center of the package.

2.3 MECHANICAL PROPERTIES OF MATERIALS

The CNS 1-13G Cask and protective jacket are made of stainless steel, lead and carbon steel. The structural evaluation is based upon the design and construction of a General Electric Shielded Package - Model 100 which is made of similar materials (Refer to Attachment C of this Application.).

2.4 GENERAL STANDARDS FOR ALL PACKAGES

This section demonstrates that the general standards and loading conditions for all packages are met.

2.4.1 Minimum Package Size

The shielded cask is 38-1/2 inches outer diameter by 67 inches high which exceeds the minimum package size as specified in 10CFR71.43(a).

2.4.2 Tamperproof Feature

A lock wire and seal of a type that must be broken if the package is opened is affixed to the package closure (See CNS Drawing C-110-B-06402-001).

2.4.3 Positive Closure

The cask has one lid which is bolted to the cask body with six 1.0-8UNC bolts which are torqued for positive closure.

A threaded drain plug is installed in the cask drain line and sealed with appropriate sealant applied to the threads of the plug.

2.4.4 Chemical and Galvanic Reactions

There are no components of the packaging or its contents which are subject to chemical or galvanic reaction.

2.5 LIFTING AND TIE-DOWN STANDARDS FOR ALL PACKAGES

2.5.1 Lifting Devices

The evaluation of package lifting devices is based upon the analysis of the General Electric Model 100 Series Cask. This analysis is included as Attachment C to this application.

2.5.2 Tie-Down Devices

The evaluation of package tie-down devices is based upon the analysis of the General Electric Model 100 Series Cask. This analysis is included as Attachment C to this application.

2.6 NORMAL CONDITIONS OF TRANSPORT

2.6.1 Heat

The thermal evaluation for the CNS Model 1-13G is discussed in Section 3.0.

2.6.2 Cold

Packaging components, i.e., steel shells and lead shielding, are unaffected by temperature extremes of -40°F and 130°F. Package contents, at least singly-encapsulated, or contained in specification 2R containers, or other inner containers, but not limited to special form, will not be affected by these temperature extremes.

2.6.3 Reduced External Pressure

The package will withstand an external pressure of 0.5 times standard atmospheric pressure.

2.6.4 Increased External Pressure

Evaluation of the package due to increased external pressure is based upon the analysis of the General Electric Model 100 Series Cask. This analysis is included as Attachment C to this application.

2.6.5 Vibration

Inspection of casks similar to the CNS Model 1-13G (i.e., GE Model 1600) used since 1962 reveals no evidence of damage of significance to transport safety.

2.6.6 Water Spray

Since the package exterior is constructed of steel, there is no effect on the package from the water spray test.

2.6.7 Free Drop

Since the container is constructed of steel, there is no damage to containment resulting from dropping the container through the standard drop heights after being subjected to water spray.

2.6.8 Corner Drop

This requirement is not applicable since the CNS Model 1-13G is constructed of steel and weighs more than 110 lbs.

2.6.9 Compression

Although the cask itself weighs more than 10,000 pounds and, therefore, is not required to be subjected to the compression test, an analysis of the overpack (protective jacket) has been performed on the General Electric Model 100 Series Cask and is presented in Attachment C to this application.

2.6.10 Penetration

There is no effect on containment or overall spacing from dropping a thirteen pound 1-1/4 inch diameter bar from four feet onto the most vulnerable exposed surface of the packaging.

CONCLUSIONS

As a result of the above assessment, it is concluded that under normal conditions of transport, the package complies with the following conditions:

1. There will be no release of radioactive material from the containment vessel.

2. The effectiveness of the packaging will not be substantially reduced.

3. There will be no mixture of gases or vapors in the package which could, through any credible increase in pressure or an explosion, significantly reduce the effectiveness of the package.

2.7 HYPOTHETICAL ACCIDENT CONDITIONS

The effectiveness of the strain rate calculations and engineering intuitiveness of the design and construction of protective jackets (overpacks) was demonstrated with the GE Shielded Package - Model 100. Extrapolations of the Model 100 data were used in the design and construction of the CNS Model 1-13G overpack. The increased weight and dimensions of the Model 1-13G container over the Model 100 container necessitated a protective jacket wall of 1/2 inch steel compared to a 1/4 inch wall for the Model 100.

2.7.1 Free Drop

The design and construction of the CNS Model 1-13G overpack (protective jacket) was based on an extrapolation of the proven data generated during the design and construction of the GE Model 100 and on the results of cask drop experiments by C.B. Clifford[1,2] and H.G. Clarke, Jr.[3,4] to determine the protective jacket wall thickness that would withstand the test conditions without breaching the integrity of the Model 1-13G cask.

The evaluation indicated a protective jacket wall thickness of 1/2 inch. The intent of the design for the CNS Model 1-13G is, during accident conditions, to sustain damage to the packaging not greater than the damage sustained by the GE Model 100 during its accident condition tests. It is expected that damage not exceeding that suffered by the GE Model 100 will result if the CNS Model 1-13G is subjected to the 30 foot drop test. An example calculation is enclosed in Section 2.8.1 of this application. A set of photographs demonstrating the GE Model 100 Drop Test is enclosed in Section 2.8.2.

2.7.2 Puncture

The intent of the design for the CNS Model 1-13G is to sustain less or equal damage to the packaging during accident conditions than the deformation suffered by the GE Model 100. It is expected that deformation not greater than that sustained by the GE Model 100 will be received by the CNS Model 1-13G in the event that the package is subjected to the puncture test. Refer to Section 2.8.2 for photographs of the GE Model 100 puncture test.

2.7.3 Thermal

The thermal test for the CNS Model 1-13G is presented in section 3.0.

2.7.4 Immersion-Fissile Material

The requirement of 10CFR71.73(c) is not applicable since criticality analysis assumes a flooded (wet) condition.

2.7.5 Immersion-all Packages

Annually, the cask shall be pneumatically pressurized to 15 psi and while under pressure, seals are soap bubble checked for leakage. The seals are acceptable if no visible bubbles are detected.

2.7.6 Summary of Damage

The accident tests or assessments described above demonstrated that the package is adequate to retain the product contents and that there is no change in spacing. Therefore, it is concluded that the CNS Shielded Container - Model 1-13G is adequate as packaging for the contents specified in section 1.2.3.

2.8 APPENDIX

2.8.1 Determination of Protective Jacket Wall Thickness

2.8.2 Drop Test Pictorial Review

2.8.3 References

2.8.1 Determination of Protective Jacket Wall Thickness

The Model 100 cask and protective jacket served as the model for upgrading all GE type carbon steel protective jackets. Based on the successful experimental work [1][2][3] and analytical work [3] of others, the Model 1000 cask protective jacket wall thickness was determined as follows:

Given: Model 100 cask protective jacket assembly

where length (L) = 33 in.

diameter (D) = 25-1/4 in.

weight (W) = 4450 lbs.

Prototype 1000 cask protective jacket assembly

where length (L) = 35 in.

diameter (D) = 26-3/4 in.

weight (W) = 5200 lbs.

Find: Prototype jacket wall thickness.

Solution: Basic Equations:[3]

$S_L = L_P/L_M$

$S_D = D_P/D_M$

$S_W = (W_P/W_M)^{1/3}$

$S_E = (E_P/E_M)^{1/3}$

(NOTE: All other cask jackets have been or will be analyzed with a similar procedure.)

Where

S_L = Scale factor for length

S_D = Scale factor for diameter

S_W = Scale factor for weight

S_E = Scale factor for energy

L_P = Prototype length

L_M = Model length

D_P = Prototype diameter

D_M = Model diameter

W_P = Prototype weight

W_M = Model weight

E_P = Prototype energy

E_M = Model energy

$$S_L = L_P/L_M$$
$$= 35/33$$
$$S_L = \underline{1.06}$$

$$S_D = D_P/D_M$$
$$= 26.75/25.25$$
$$S_D = \underline{1.04}$$

$$S_W = (W_P/W_M)^{1/3}$$
$$= (5200/4450)^{1/3}$$
$$S_W = \underline{1.05}$$

$$S_E = (E_P/E_M)^{1/3}$$ (Drop test height = 30 feet)

$$(5200)(30)/(4450)(30)^{1/3}$$
$$= (1.17)^{1/3}$$
$$S_E = \underline{1.05}$$

The worst case condition is to use the length scale factor to determine the required jacket wall thickness.

$S_L = T_P/T_M$

Where

T_P = Prototype wall thickness

T_M = Model wall thickness (0.25 in.)

S_L = 1.06

Rearranging the above equation yields,

$T_P = T_M \times S_L$

$= (0.25)(1.06)$

$T_P =$ <u>0.265</u> in.

Conclusion: The selected protective jacket wall thickness of 5/16 inches is more than adequate.

2.8.2 Drop Test Pictorial Review

This section consists of photographs taken prior to and following the drop test series for the GE Model 100. The Model 100 package was prepared using a Model 100 cask and the full size prototype protective jacket and base. At this point in the development of the protective jacket assembly, the jacket was attached to the base with two (2) horizontal 1-1/2 inch (3.8 cm.) diameter bolts. These bolt holes are located near the bottom of the jacket. The bolts pinned the jacket and base together in a double shear arrangement. During final design, these bolts were replaced with four (4) 1-1/2 inch (3.8 cm) bolts located in flanges welded to the bottom of the jacket.

The prototype Model 100 did not have a pallet attached at the time of testing. The addition of the flange bolts and pallet assembly would not affect the outcome of the test series.

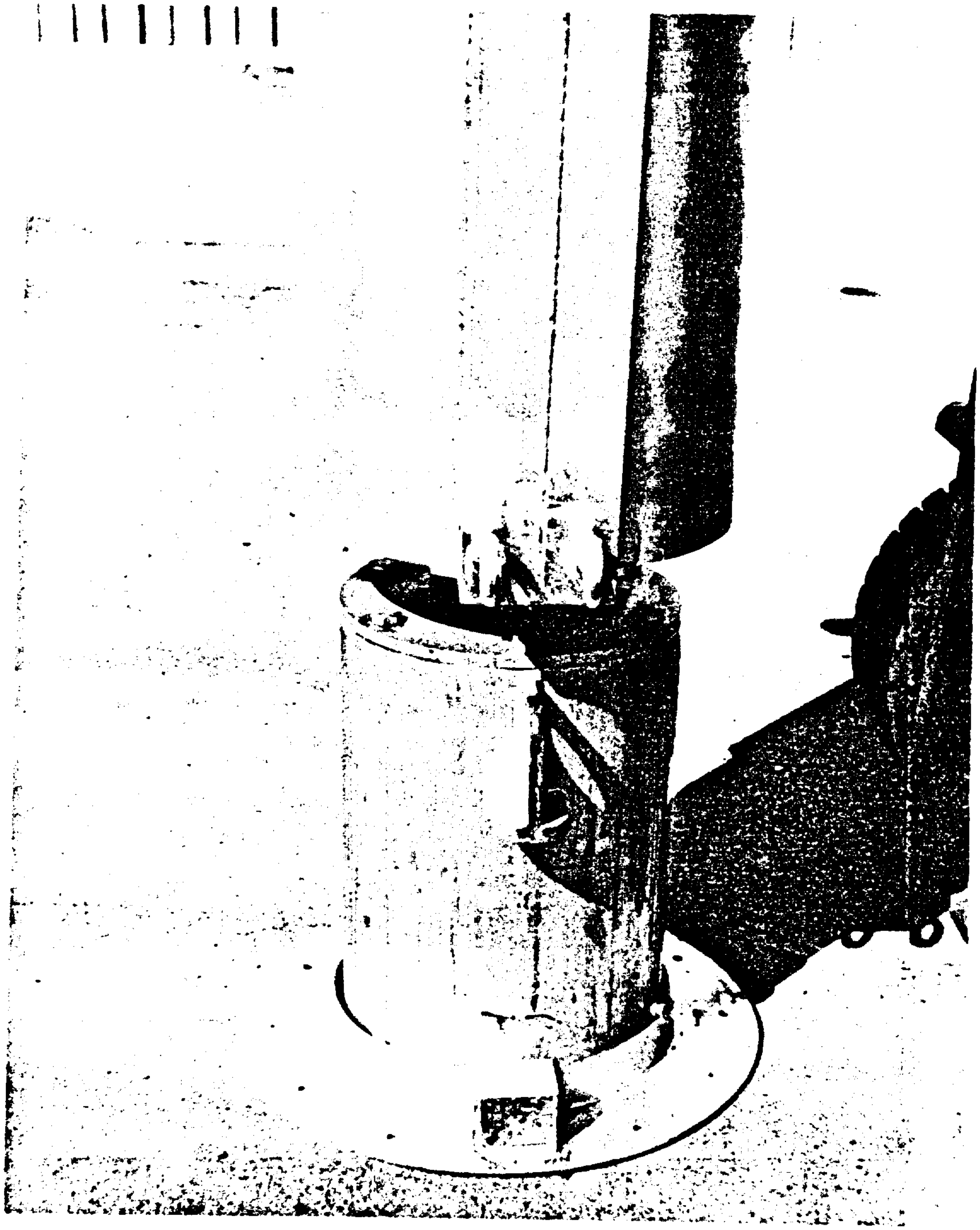


Figure 2.8.2.1 The protective jacket being lowered onto the base prior to the drop tests.

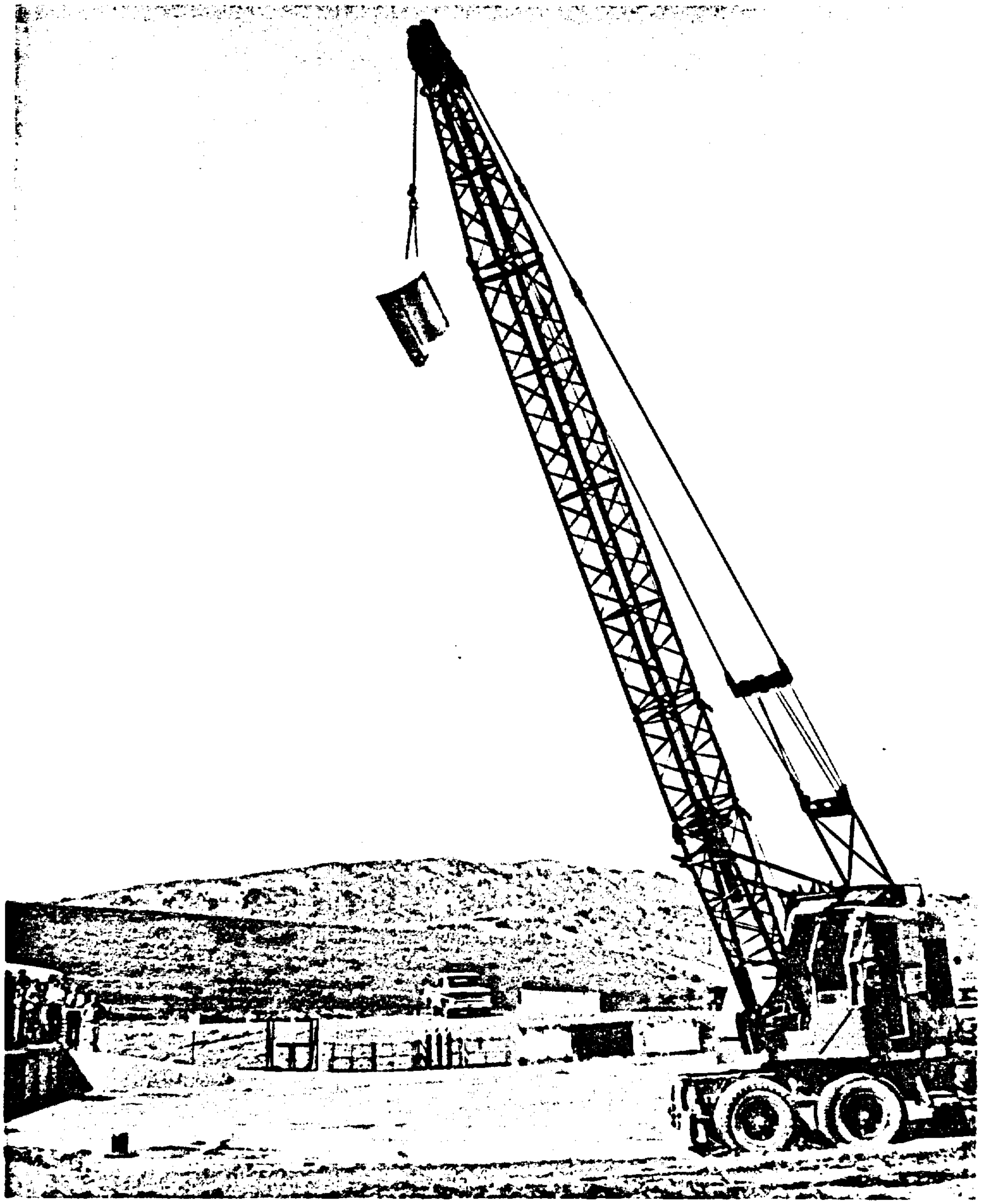


Figure 2.8.2.2 The Model 100 package is positioned above the steel surfaced drop pad ready for the 30 foot (9 m) drop test.

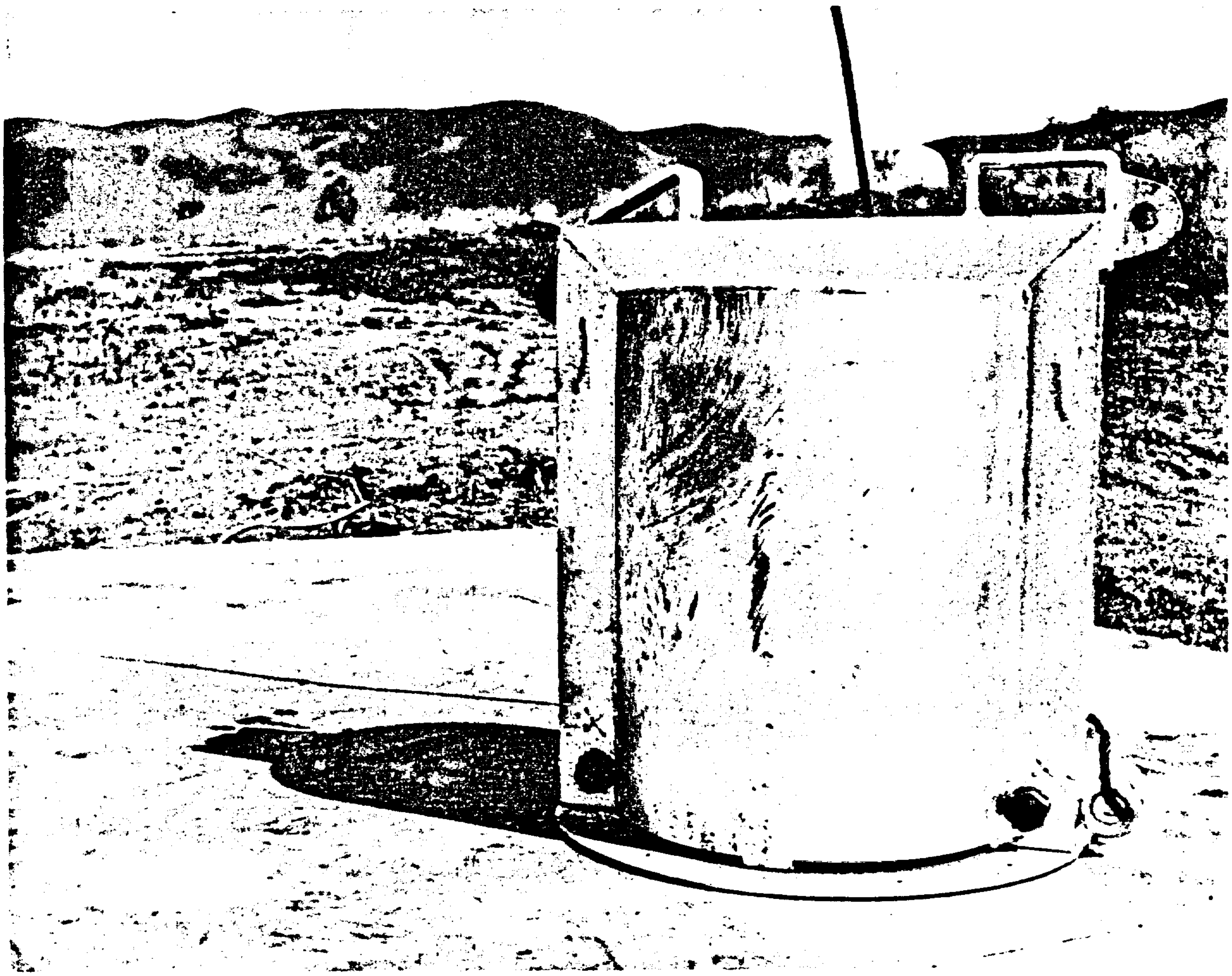


Figure 2.8.2.3 The Model 100 transportation package after the 30 foot drop test. Notice the damage to the rectangular eye on which the package landed.

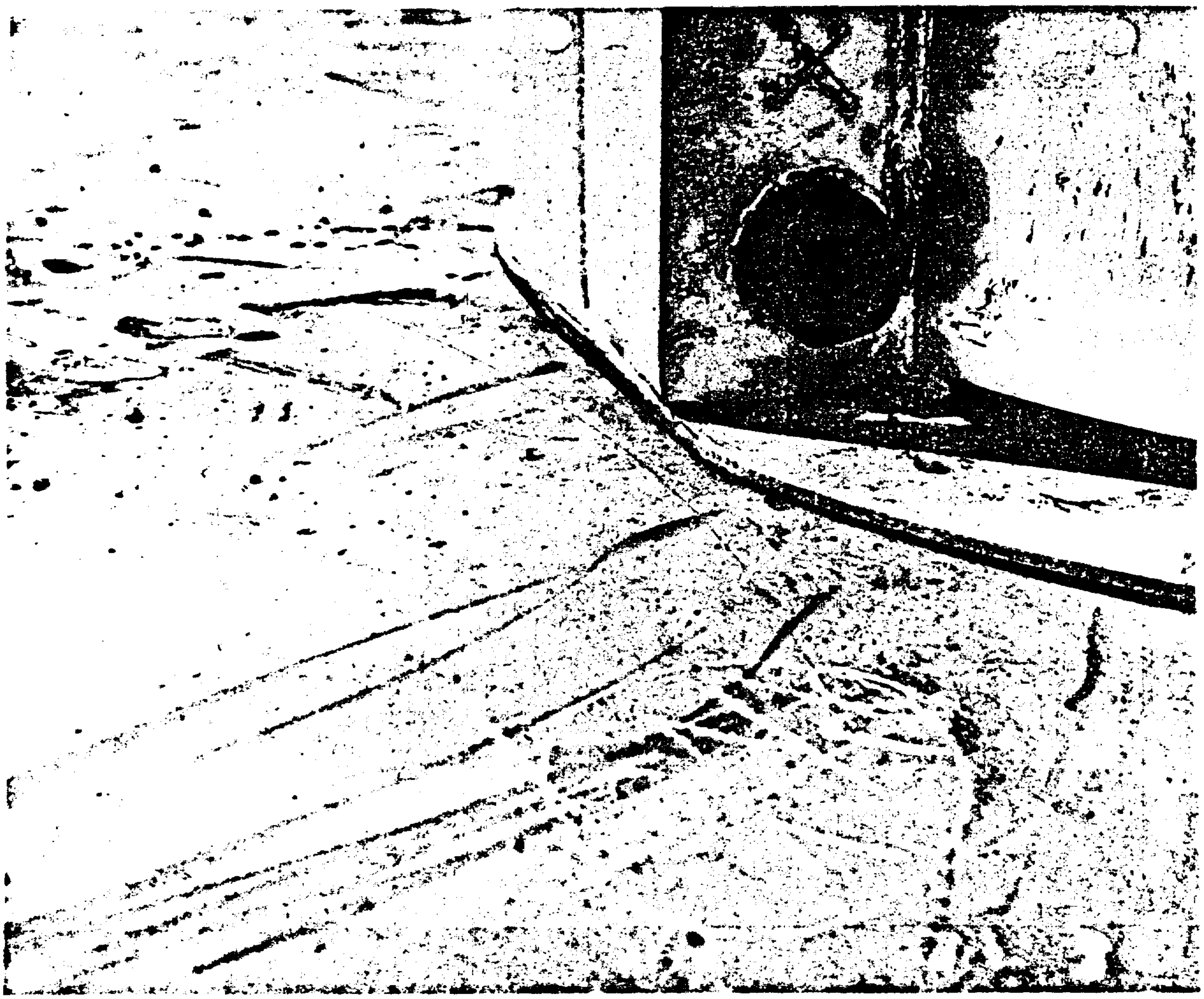


Figure 2.8.2.4 The lip of the Model 100 base after the 30 foot (9 m) drop test. The bend was caused when the package bounced after the initial impact and fell back on the base.

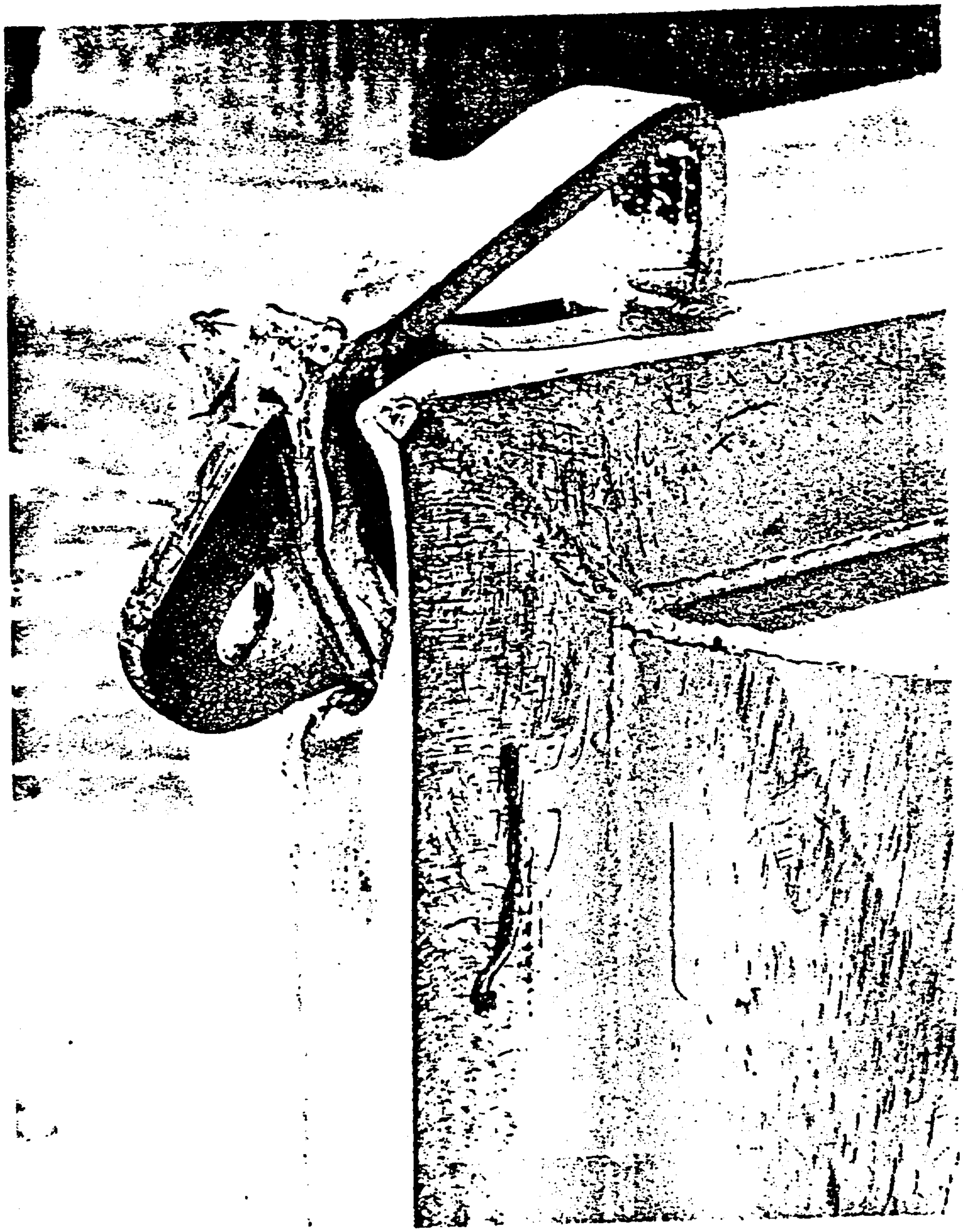


Figure 2.8.2.5 The Model 100 rectangular eye collapsed when it struck the pad during the 30 foot (9 m) drop test. Note how one leg of the ear penetrates the protective jacket, but no other deformation is apparent.

Figure 2.8.2.6 Another view of the protective jacket ear after the impact of the 30 foot (9 m) drop test.

Figure 2.8.2.7 The 6" diameter bar target for the puncture drop test is shown firmly anchored to the steel drop pad plate.

(0156W)

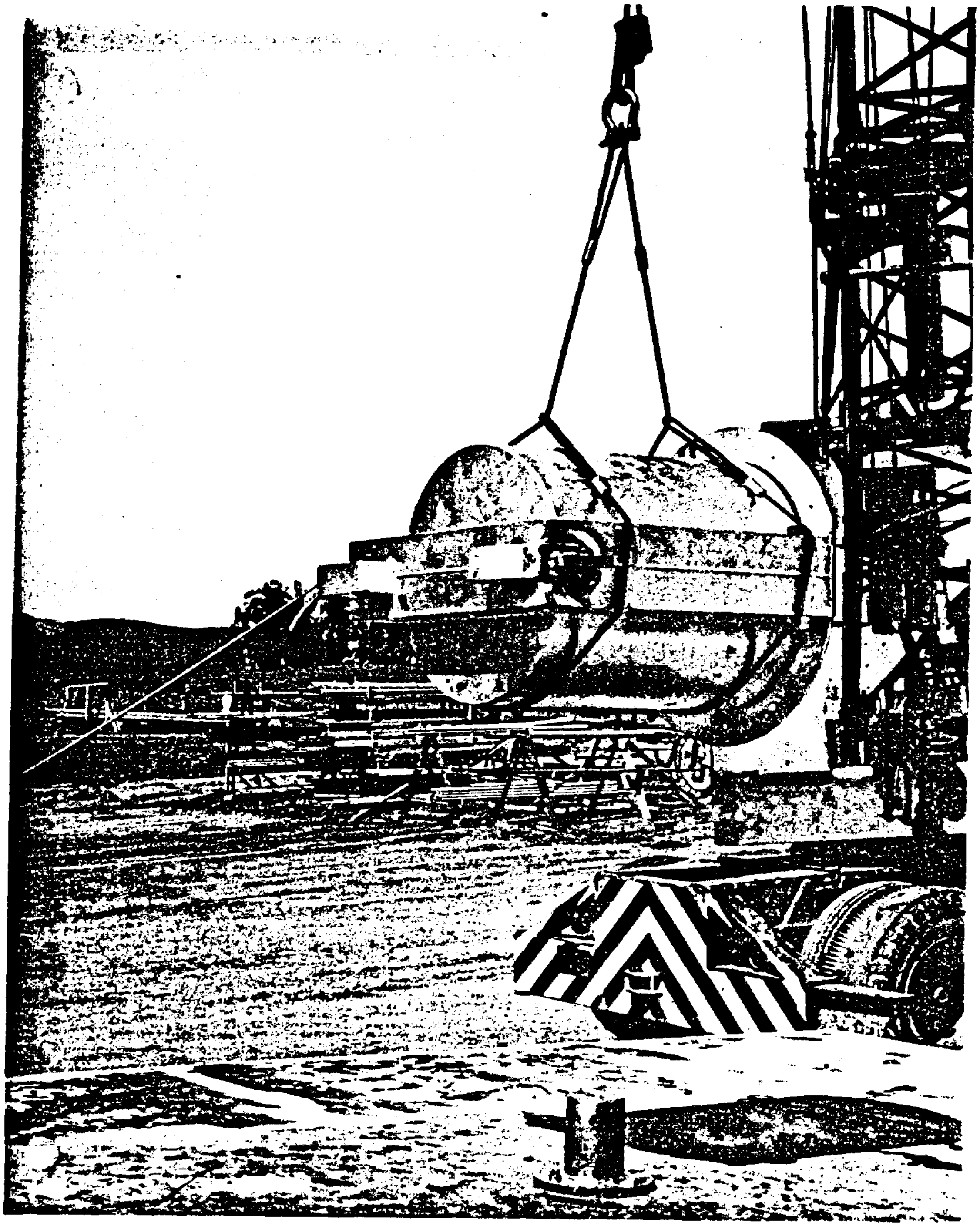


Figure 2.8.2.8 The Model 100 transportation container shown in position over the 6" diameter bar prior to the 40" (1 m) puncture drop test.

Figure 2.8.2.9 This is the damage done to the protective jacket in the 40" (1 m) puncture drop test. Notice the three distinct points of contact.

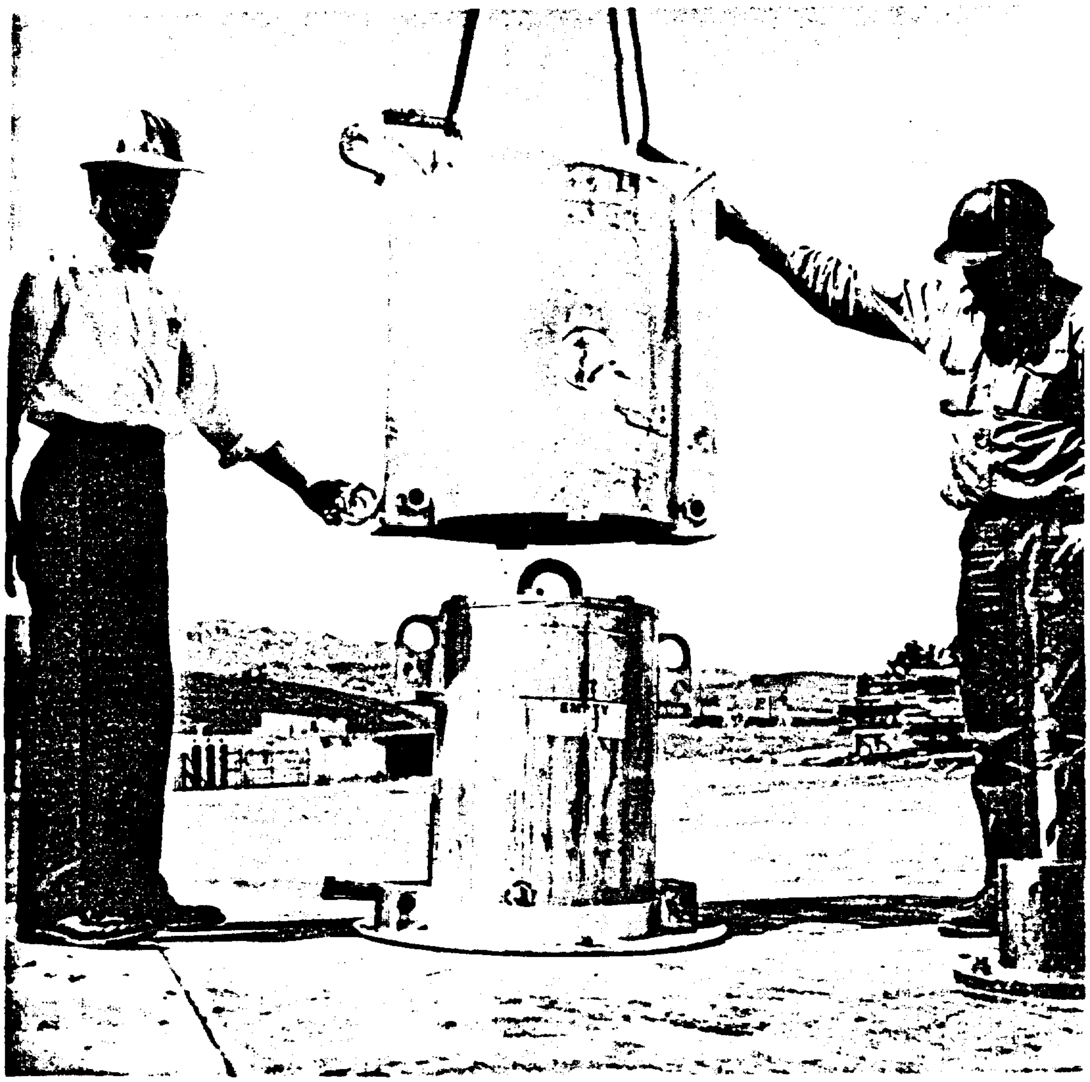


Figure 2.8.2.10 The protective jacket being removed from the base after the 30 foot (9 m) puncture drop tests. The jacket was removed easily with no binding.

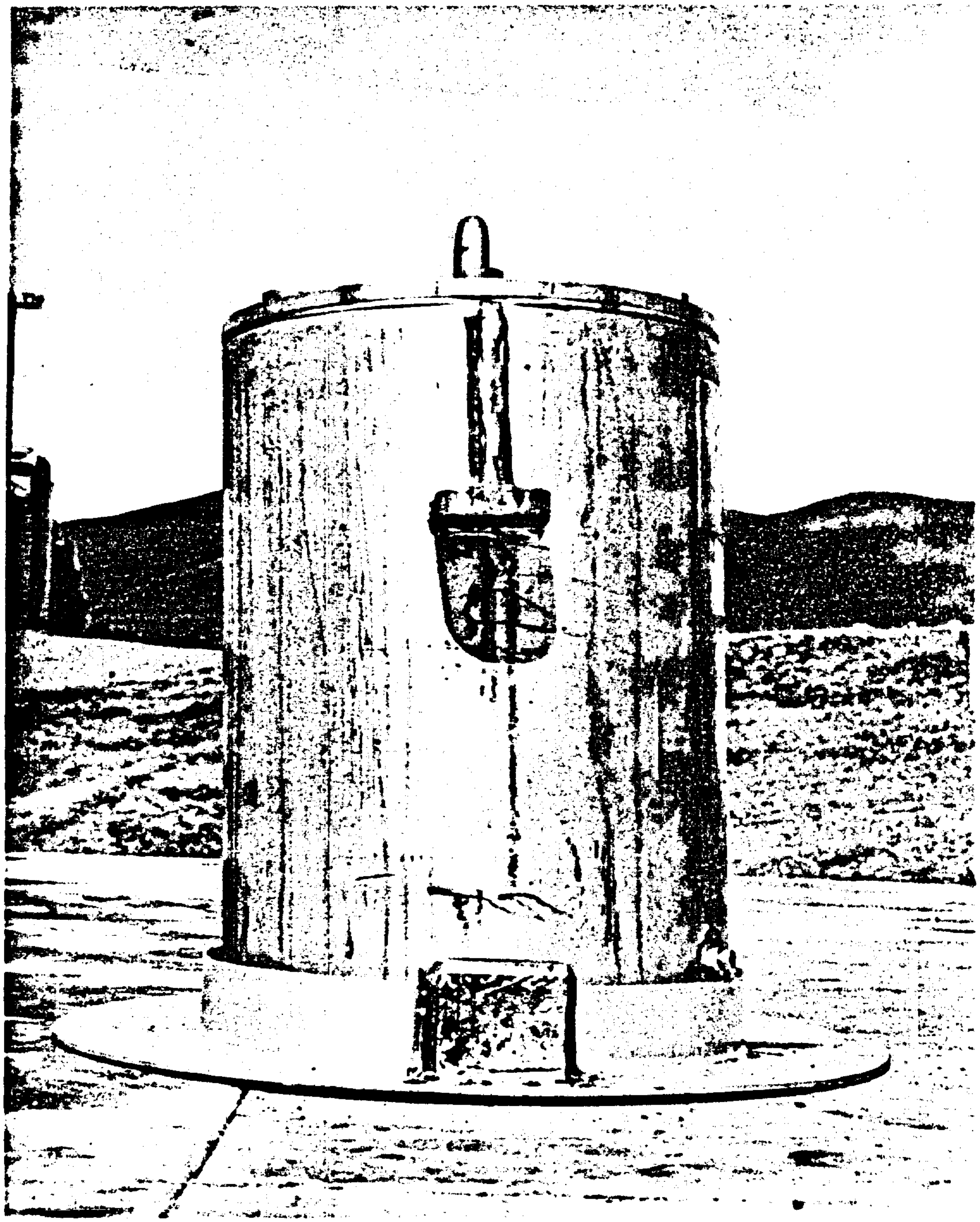


Figure 2.8.2.11 The cask "leans" because of the crushed energy absorption angles in the base.

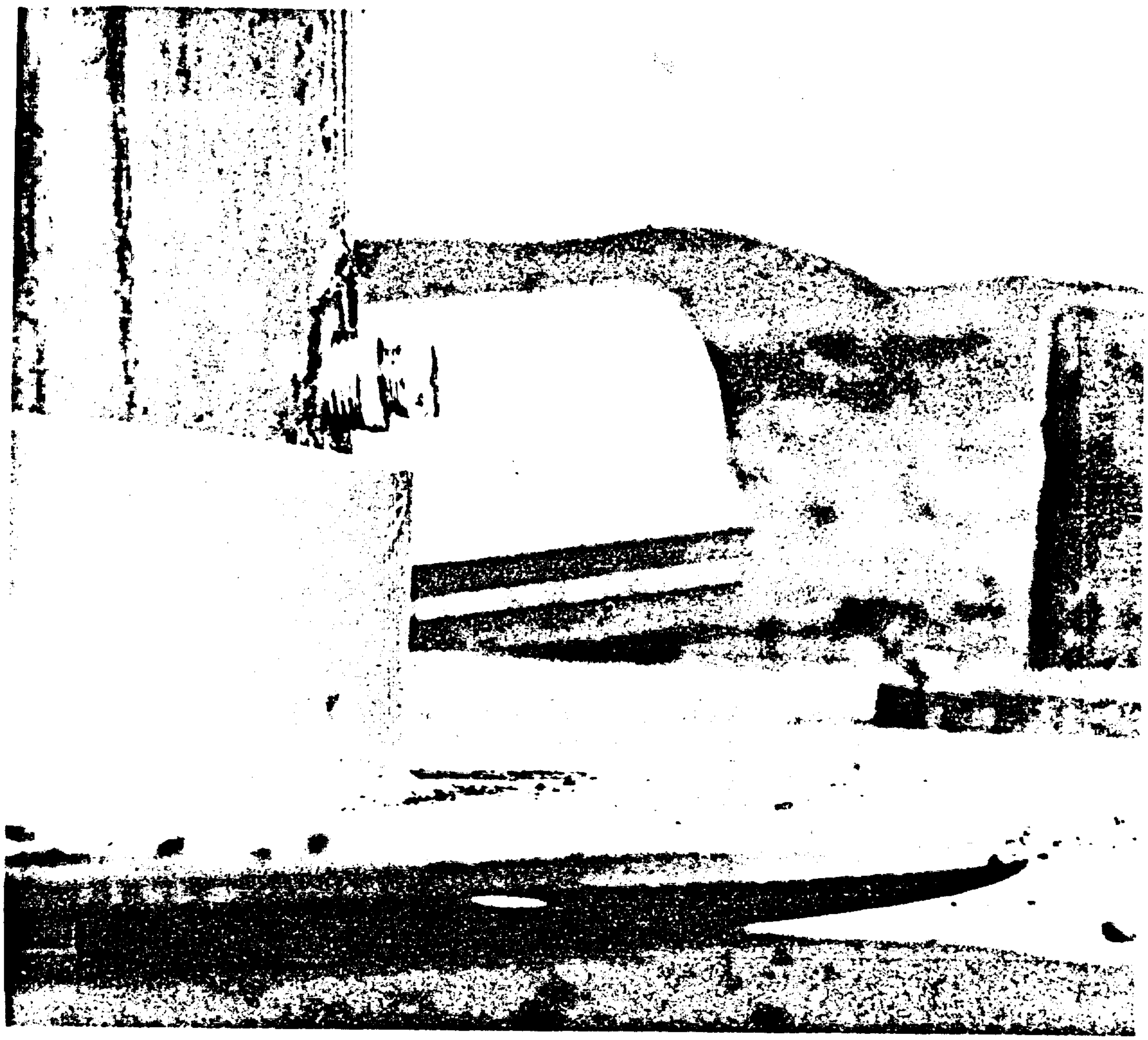


Figure 2.8.2.12 This shows the bent and partially fractured pipe plug that resulted from the drop test. The fitting is not typical. A flush hex plug is normally used.

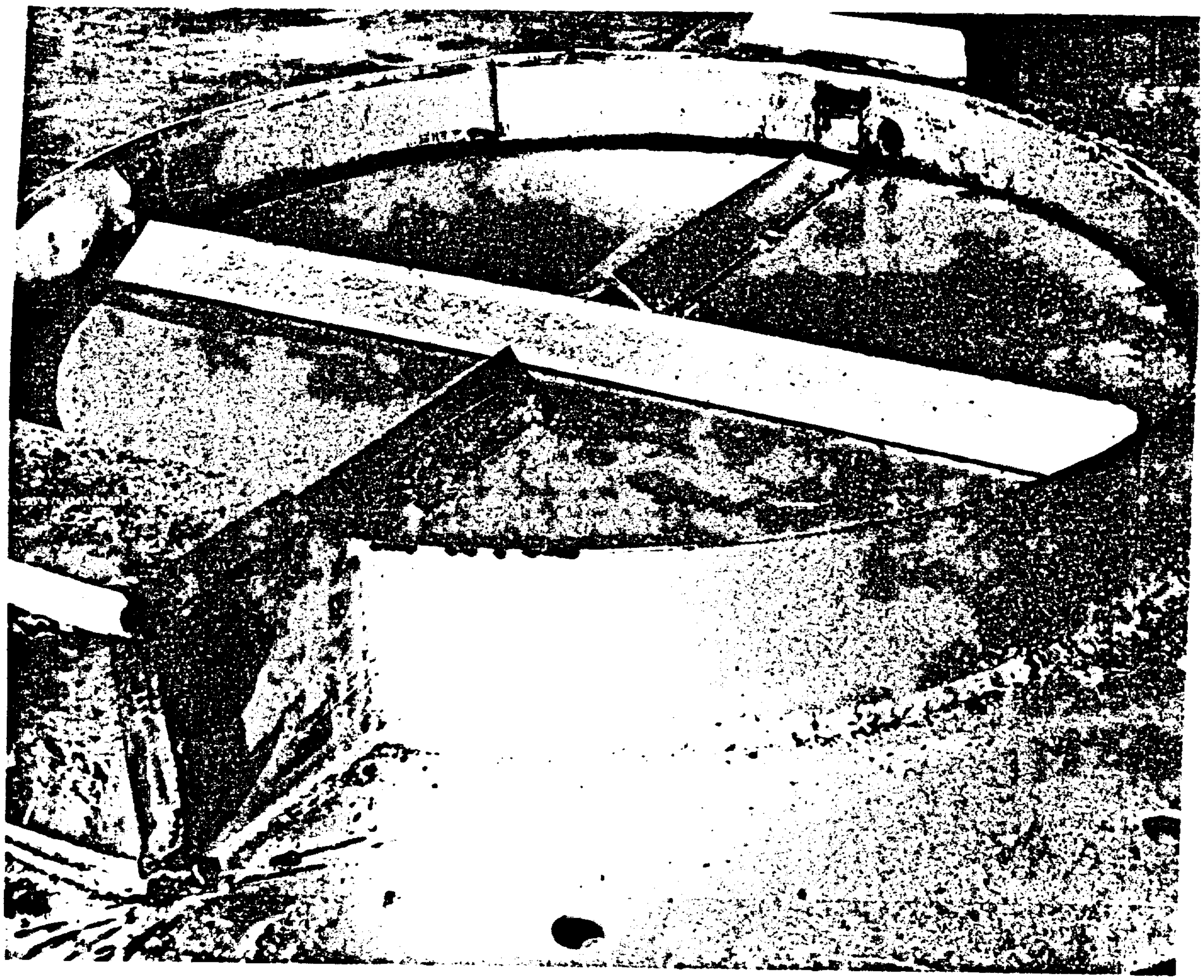


Figure 2.8.2.13 This view of the base after the drop tests shows the deformation of the energy absorbing angles.

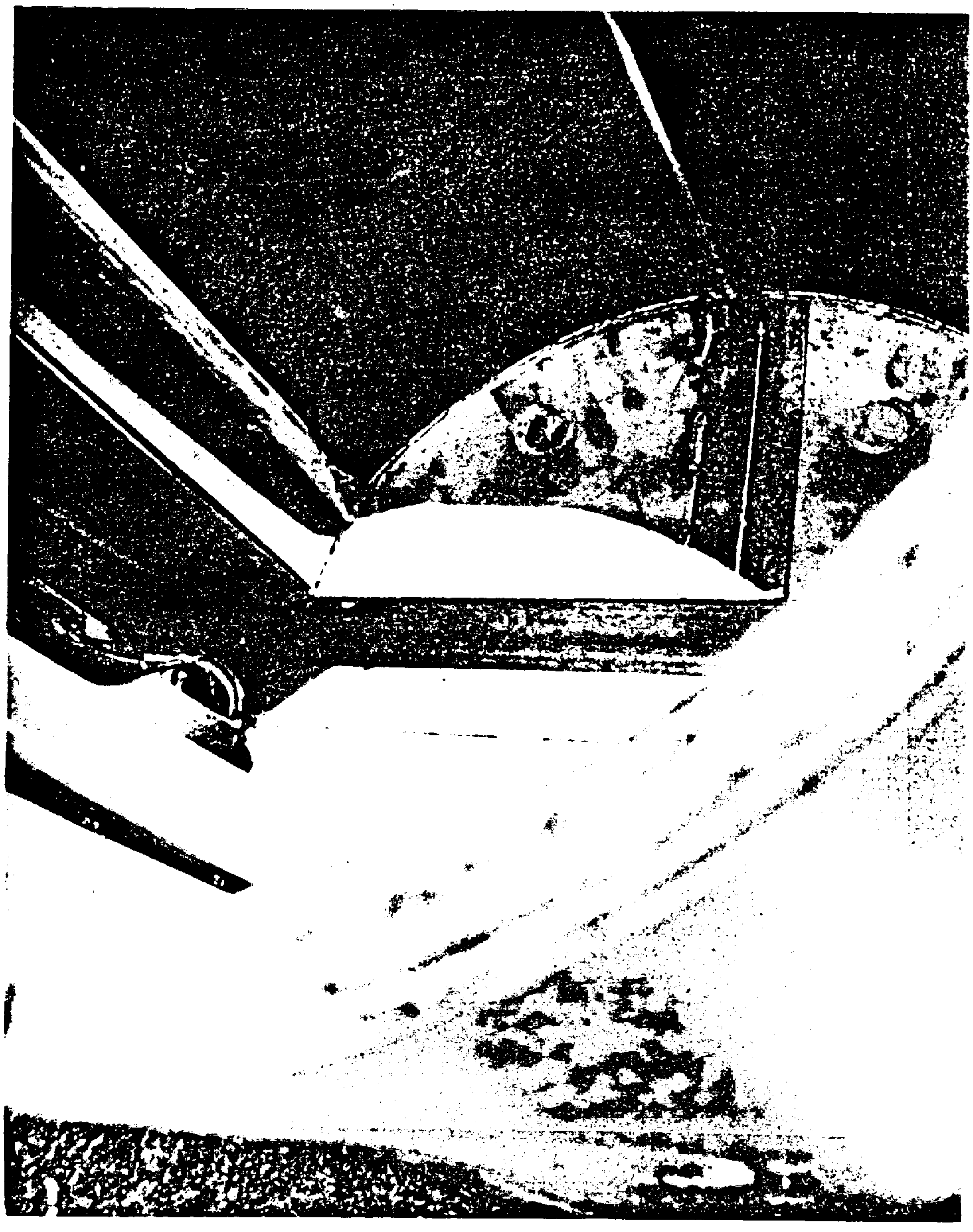


Figure 2.8.2.14 Showing the internal section of the protective jacket behind the rectangular eye which collapsed on impact. The flattened energy absorbing angle clip and the internal marking due to impact between the cask lid bolts and the inside of the jacket.

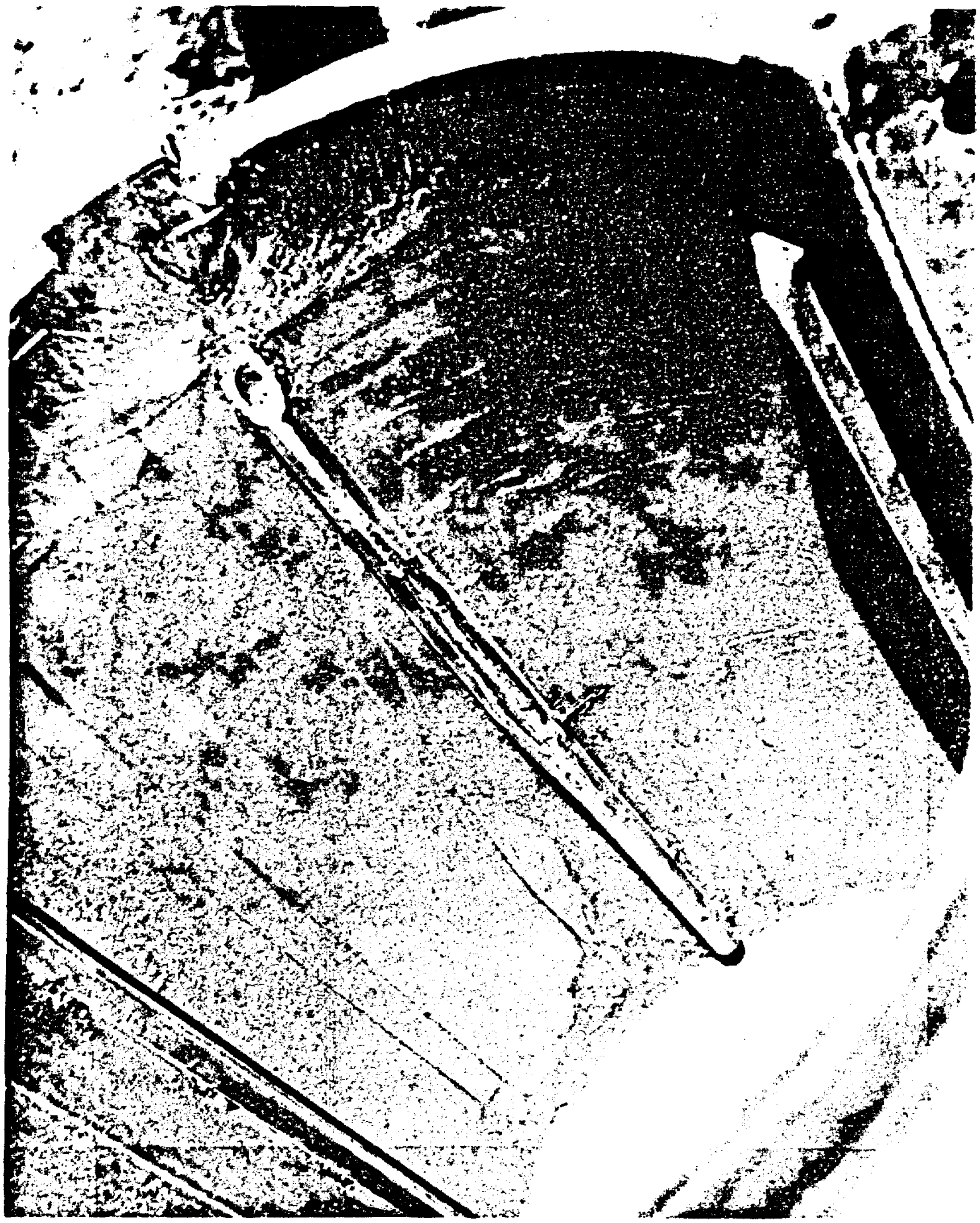


Figure 2.8.2.15 Shows the inside of the protective jacket and the partially crushed energy absorbing tube.

2.8.3 References

(1) C.B. Clifford, The Design, Fabrication and Testing of a Quarter Scale of the Demonstration Uranium Fuel Element Shipping Cask, KY-546 (June 10, 1968).

(2) C.B. Clifford, Demonstration Fuel Element Shipping Cask from Laminated Uranium - Uranium Metal-Testing Program, Proceedings of the Second International Symposium on Packaging and Transportation of Radioactive Materials, Oct. 14-18, 1968, pp. 521-556.

(3) H.G. Clark, Jr., Some Studies of Structural Response of Casks to Impact, Proceedings of the Second International Symposium on Packaging and Transportation of Radioactive Materials, Oct. 14-18, 1968, pp. 373-398.

(4) J.K. Vennard, Elementary Fluid Mechanics, Wiley and Sons, New York, 1961, pp. 256-259.

3.0 THERMAL EVALUATION

Thermal Evaluation for Normal Conditions of Transport

Packaging components, i.e., steel shells and lead shielding, are unaffected by temperature extremes of -40°F and 130°F. Package contents, at least singly-encapsulated, or contained in specification 2R containers, or other inner containers, but not limited to special form, will not be affected by these temperature extremes.

The Model 1-13G was analyzed for thermal effects with the 600 watt loading utilizing the Transient Heat Transfer-Version D (THTD) computer code using appropriate axisymmetrical, two-dimensional (r-Z geometry) models of the cask and cask with liner packages to calculate both the steady state and fire transient temperature profiles.

The THTD computer program has been used by General Electric Company extensively over the past 15 years on a variety of steady-state and transient thermal problems, including shipping casks. The THTD computer program computes transient and steady-state temperature solutions for three-dimensional heat transfer problems.

As many as 2,000 nodes may be specified with heat transferred from node-to-node (or node-to-boundary) by conduction, convection, and/or thermal radiation. Temperature solutions are obtained by iterative solution of simultaneous algebraic equations for node temperatures derived from finite-difference analysis. The use of simultaneous equations (the implicit method of formulating nodal heat balances) precludes any stability limitations on time increments and permits a direct steady-state solution at any stage of a computer run, including solutions to serve as initial conditions for a transient. Convergence of a temperature solution is recognized and controlled by tolerances on the residual heat balances which provide a measure of the "imperfection" of a solution as well as by the conventional maximum change in any nodal temperature during an iteration sweep.

Hypothetical Accident Thermal Evaluation

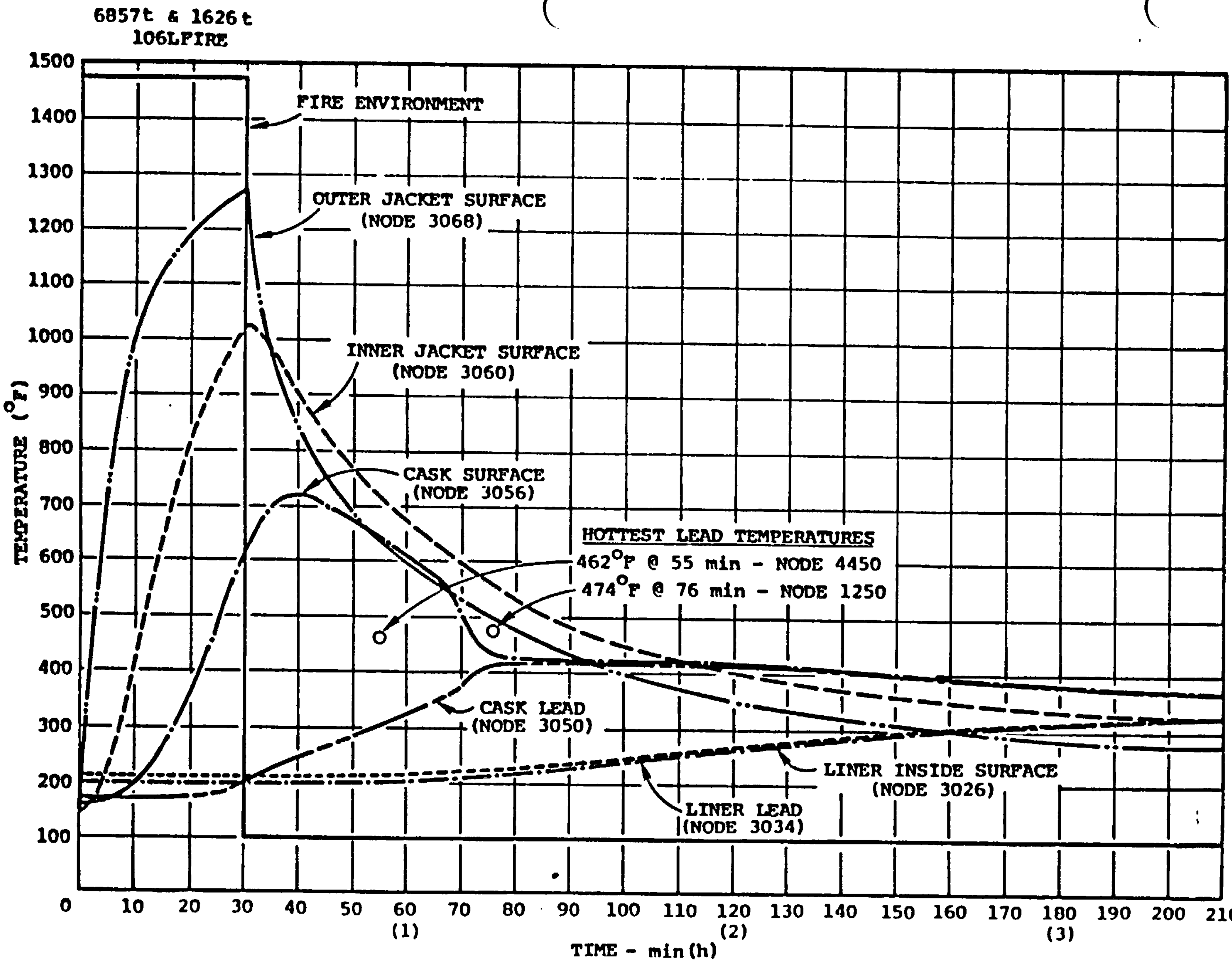
The thermal transient for the 1475°F fire was run using the THTD program described above.

Under fire conditions, the hottest lead temperatures are experienced at the "corners" due to the heat input from both the side surface and the top or bottom surfaces. A major objective of this accident evaluation is to determine if any shielding lead melted as a result of the event. The THTD computations show a maximum lead temperature of 479°F for the cask without liner case. The calculated maximum lead temperature for the cask with liner case is 474°F. These maximum lead temperatures are well below the 621°F melting point of lead. Graphs showing the fire transient are included in the Appendix to this section.

SECTION 3.0 APPENDIX

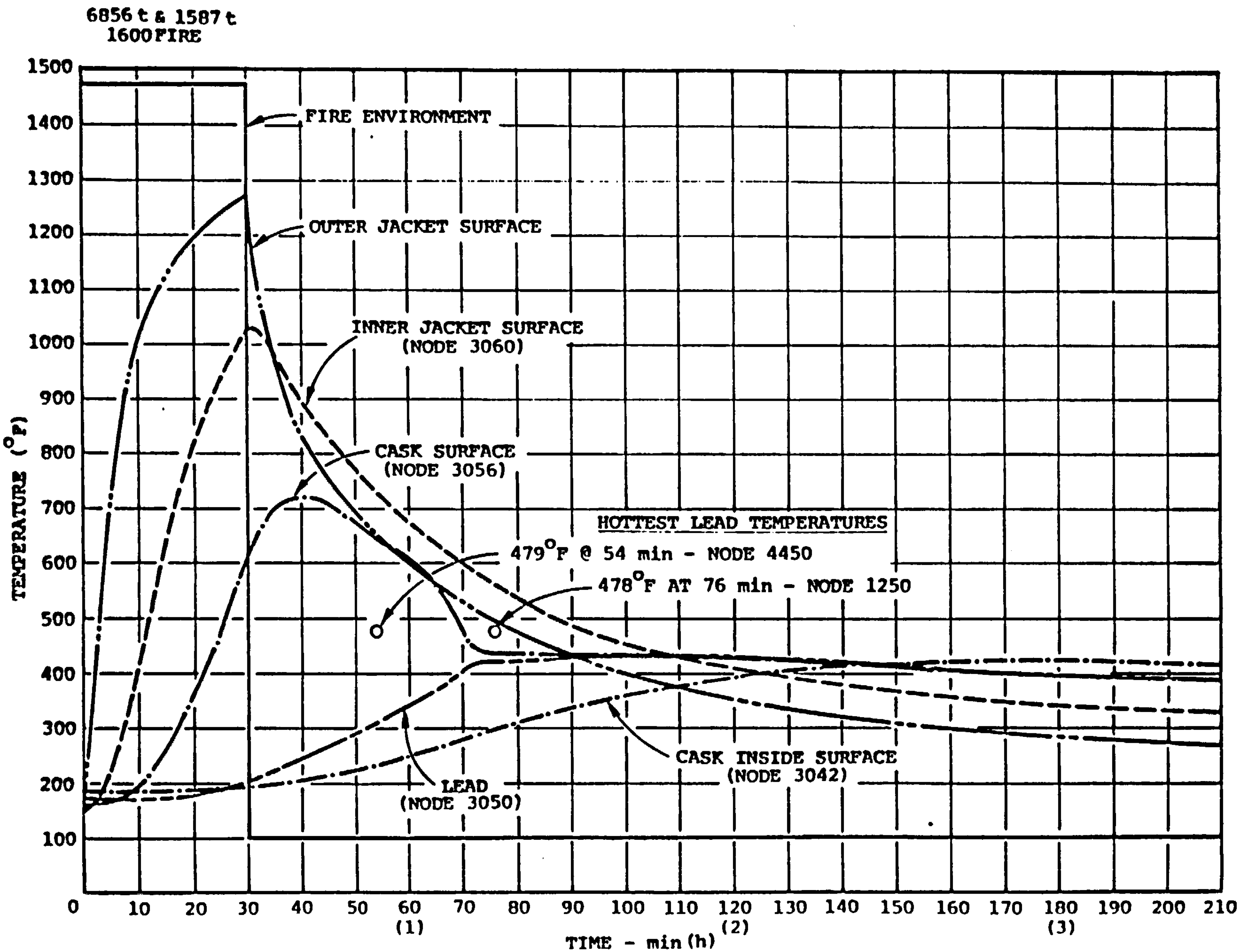
Thermal Transient Curves - THTD

(0156W)



MODEL 1-13G CASK WITH LINER
FIRE TRANSIENT TEMPERATURE
(160L FIRE)

Rev. 0



MODEL 1-13G CASK FIRE TRANSIENT TEMPERATURE (1600 FIRE)

(0156W)

Rev. 0

4.0 CONTAINMENT

This chapter describes the containment configuration and test requirements for the CNSI 1-13G cask. Both normal conditions of transport and hypothetical accident conditions are discussed.

4.1 CONTAINMENT BOUNDARY

4.1.1 Containment Vessel

The package containment vessel is defined as the inner shell of the shielded transport cask together with the associated lid, gasket seals, and lid closure bolts. The inner shell of the cask, or containment vessel, consists of a right circular cylinder of 26.5 inches inner diameter and 54.0 inches inside height. The inner shell is fabricated of a 1/2-inch thick stainless steel plate. Full penetration welds are used to attach the base plate and sealing flange to the cylindrical shell. The containment vessel is surrounded by poured lead at a typical thickness of five (5) inches on the side and six (6) inches of lead on the top and bottom. The lead is encapsulated by an outer shell formed from 0.5-inch thick stainless steel plate. The outer shell is attached to an 0.6-inch thick stainless steel base plate with a full penetration weld. The cylindrical closure lid is bolted to the cask body with six one-inch stainless steel bolts.

4.1.2 Containment Penetrations

The containment has one penetration, a 1/2-inch outer diameter by 0.065-inch wall stainless steel tube gravity drain line from the center of the cavity bottom to the side of the outer shell near the cask bottom. It is closed with a 1/2 NPT socket head stainless steel, brass, or equivalent pipe plug capable of maintaining its seal at temperatures up to at least 620°F for dry contents or a 200°F fuseable core plug for neutron shipments.

4.1.3 Seals and Welds

A minimum 3/16-inch thick flat silicone and rubber gasket or a molded rubber seal bonded to an aluminum back-up plate or equivalent can be used as a seal between the cask body and lid.

Each cask is inspected and radiographed prior to first use to ascertain that there are no cracks, pinholes, uncontrolled voids or other defects which could significantly reduce the effectiveness of the packaging.

4.1.4 Closure

The cask is closed by a lead-filled, flanged lid fitted with a lifting lug and a silicone rubber gasket. The lid is sealed with six (6) 1-inch hex head bolts equally spaced 60° apart on a 35-1/4" diameter bolt circle. The bolts are torqued to 220 ± 22 ft.lbs. (160 ± 16 ft. lbs. if lubricated) using a star pattern.

The drain in the base of the cask is plugged with a 1/2-inch square-headed pipe plug.

4.2 REQUIREMENTS FOR NORMAL CONDITIONS OF TRANSPORT

4.2.1 Containment of Radioactive Material

The cask was analyzed to show that design specifications met or exceeded 10CFR71 Regulations and DOT 49CFR171-178 Regulations. The tests or assessments set forth in accordance with ss 71.71, Normal Conditions of Transport, provide the assurance that the products are contained in the shielded package, CNSI 1-13G, during transport and there is no reduction in the effectiveness of the package which meets the requirements set forth in ss 71.51.

4.2.2 Pressurization of Containment Vessel

There is no internal pressure for the dry cavity.

4.2.3 Containment Criterion

The cask shall be pneumatically pressurized to 15 psi and, while under pressure, seals are soap bubble checked for leakage. Acceptance criteria - no visible bubbles.

4.3 CONTAINMENT REQUIREMENTS FOR HYPOTHETICAL ACCIDENT CONDITIONS

A complete description of the evaluation and the means to immobilize and contain solidified waste in the GE Model 1600, which is similar to the CNS Model 1-13G, is provided in Attachment D. As a result of the evaluation, it was determined that under accident conditions, no more than 0.018 curies of mixed fission products (MFP) or 0.01 curies of Co-60 would be released from the cask.

Section 2.7 demonstrates that the cask will maintain its containment capability throughout the hypothetical accident conditions. The cask is designed, fabricated, and leak tested to preclude a release of radioactive material in excess of the limits prescribed in 10CFR71.51.

For verifiable containment criterion, the cask is pneumatically pressurized to 15 psi and, while under pressure, seals are soap bubble checked. The leakage acceptance criteria is no visible bubbles.

4.4 SPECIAL REQUIREMENTS

Refer to Section 4.3 and Attachment D to this application.

5.0 SHIELDING EVALUATION

5.1 DISCUSSION AND RESULTS

The 1-13G cask consists of a lead and steel containment vessel which provides the necessary shielding for the various radioactive materials to be shipped within the package. (Refer to Section 1.2.3 for packaging contents.) Tests and analyses performed under Sections 2.0 and 3.0 have demonstrated the ability of the containment vessel to maintain its shielding integrity under normal conditions of transport. Prior to each shipment, radiation readings will be taken based on individual loadings to assure compliance with applicable regulations.

The 1-13G cask will be operated such that the contents in the cask will not create a dose rate exceeding 200 mR/hr on the package surface, or 10 mR/hr at two meters from the package surface. The package shielding must be sufficient to satisfy the dose rate limit of 10CFR71.51(a)(2) which states that any shielding loss resulting from the hypothetical accident will not increase the external dose rate to more than 1000 mR/hr at one meter from the external surface of the cask.

5.2 SHIELDING EVALUATION

Shielding evaluations for the polyethylene liner and the auxiliary shield are presented as Attachment A and Attachment B, respectively to this application.

6.0 CRITICALITY EVALUATION

This section provides the criticality analysis for Fissile Class III shipments in the CNS 1-13G in the form of irradiated PuO_2 and UO_2 fuel rods clad in Zircalloy or stainless steel.

6.1 ANALYSIS OF IRRADIATED PuO2 AND UO2 FUEL ROD (1200 GRAM FISSILE) LOADING

The fissile material is contained in standard waste liners constructed of 5 inch schedule 40 pipe with a maximum inside length of 39-5/16 inches, no more than four such liners are shipped at one time, each liner contains no more than 300 grams fissile material and the cask is provided with a positioning lattice such that the geometry shown in Figure 6.1 is maintained. The purpose of the positioning lattice is to improve the criticality characteristics of the cask. However, as noted in the criticality analysis, the liner is not necessary to maintain the cask in a subcritical condition (i.e., the close proximity cases).

The waste liners are closed with either a bronze or brass screw top with a 1/2 inch O-ring gasket. The gasket material may be either buna-N rubber or neoprene.

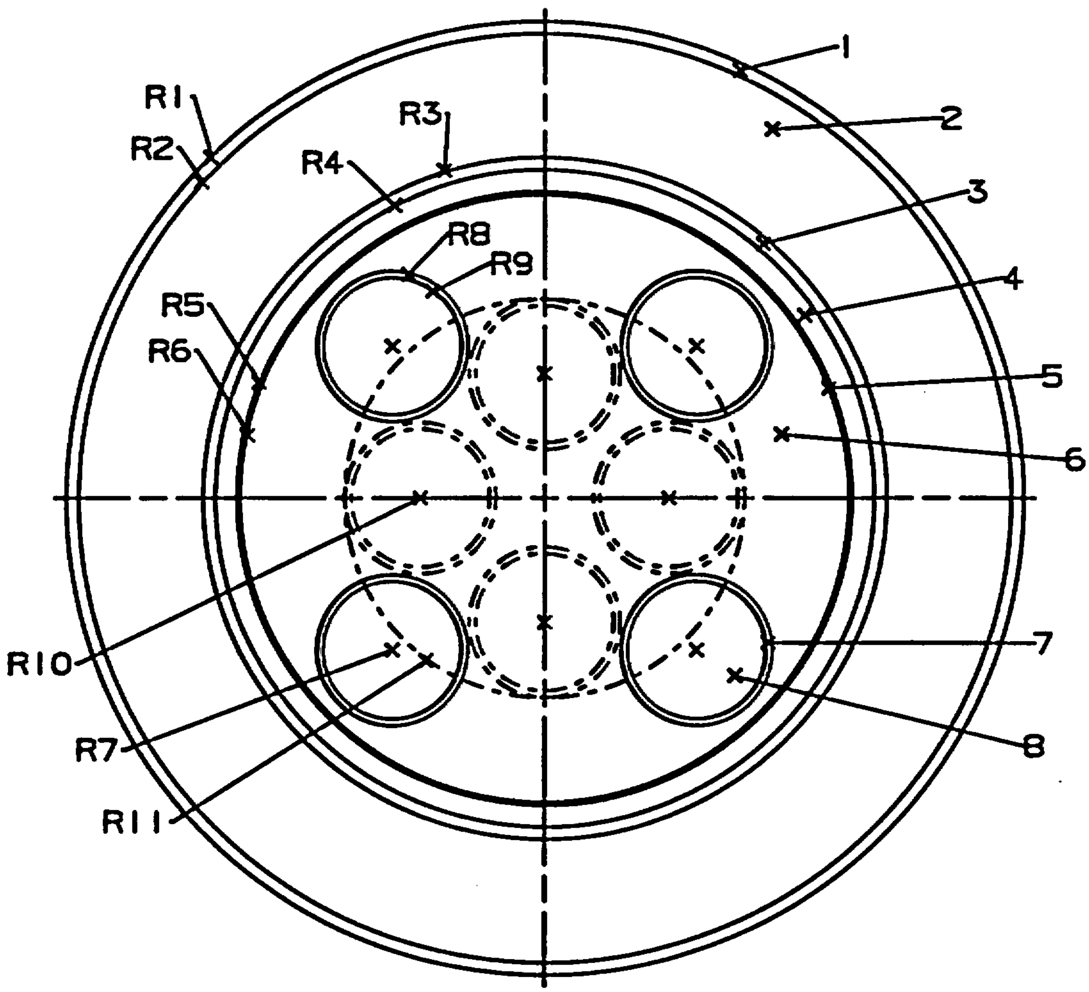


FIGURE 6.1

CNS MODEL 1-13G 1200 GRAM FISSILE LOADING GEOMETRY

The criticality analysis was performed using the computer code ANISN[1], a discrete ordinates, one-dimensional transport theory code. The problem was solved in two parts using geometries as shown on Figure 6.1. The dimensions and material regions for this figure are given in Tables 6.1 and 6.2. The normal transport configuration is shown on Figure 6.1 as solid lines. The dotted lines represent one of the accident cases considered.

From this analysis, we conclude that the 1-13G cask is critically safe for the shipment of four standard waste liners each containing 300 gm fissile (Pu^{239}, U^{233}, or U^{235}) for a total cask limit of 1200 grams fissile. This limit is safe with no restriction as to fissile type or composition.

Four cases were considered in this analysis, the design shipping geometry, the design geometry flooded, and the cases when the four liners are in close proximity. These geometries are shown in Figure 6.1. The results of the criticality calculations are as follows:

1.	Design Geometry	$k_{eff} = 0.720$
2.	Design Geometry - Flooded	$k_{eff} = 0.713$
3.	Close Proximity Geometry	$k_{eff} = 0.959$
4.	Close Proximity Geometry - Flooded	$k_{eff} = 0.976$

The case of two casks adjacent to each other was considered by calculating the infinite multiplication (k_{∞}) and the migration area (M^2) from properties determined by the ANISN calculation for the cask. The effective multiplication of the two cask system was then determined from the buckling relationship. This value was found as:

$$k_{eff} = 0.773$$

Table 6.1 CNS Model 1-13G Cask Dimensions

Radius	R-cm
R1	48.895
R2	47.625
R3	34.925
R4	33.655
R5	31.4325
R6	31.115
R7	Dimension determined by liner cell calculation*
R8	7.7203
R9	7.065
R10	Dimension determined by liner cell calculation*
R11	17.96

*$R = 1.414\ [(R_0)^2]^{1/2}$ where R_0 = liner cell radius volume.

Table 6.2 Material Regions

Region	Material	Identification
1	Stainless	Outer Liner
2	Lead	Shield
3	Stainless	Inner Liner
4	Void	Void
5	Stainless	Liner Bucket
6	Fuel Cell	Homogenized Fuel, Liner, Void
7	Aluminum	Liner
8	Fissile Material	Fuel Material

Table 6.3 Atom Densities

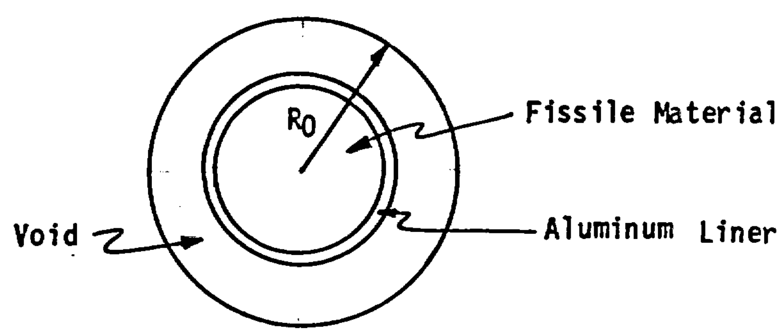
Calculation	Material	Isotopes (Atom/b-cm)	Atom Densities
Liner Cell	Liner	Aluminum	6.023×10^{-2}
	Fissile	Plutonium-239	5.796×10^{-5}
	Hydrogen	6.673×10^{-2}	
	Oxygen	3.337×10^{-2}	
Cask	Stainless	Iron	6.01×10^{-2}
	Chromium	1.72×10^{-2}	
	Nickel	8.81×10^{-3}	
	Shield	Lead	3.31×10^{-2}

The first part of the problem consisted of an infinite cylinder cell calculation for the individual fuel liners. This calculation was performed to obtain cell-weighted cross-sections to be used in subsequent calculations. Each cell consisted of the fissile material/moderator combinations contained within the waste liner, the waste liner itself, and the void surrounding each liner. The waste liner dimensions are as shown in Table 6.1. Each liner was assumed to contain 300 grams of plutonium-239, with the remainder of the liner filled with water. For the actual waste liner volume, the resulting Pu density was $0.023 \frac{gm\ Pu}{cm^2}$.

The volume fraction for the water was determined by assuming a theoretical plutonium density of 11.46 gm/cm^3 (the most likely density of fissile waste material to be shipped). The atom densities used in the cell calculations are shown in Table 6.3.

For the normal shipping configuration, the cell size was determined by dividing the total volume of the cask cavity into four equal volumes and calculating an equivalent cell radius. This radius represented the outer dimension of the ANISN problem with void between this and the actual outer diameter of the liner. For normal shipping configuration, this dimension was: R_O = 15.55 cm. For the case where the liners were assumed in close proximity, as shown by the dotted circles in Figure 6.1, this dimension was taken as: R_O = 8.98 cm. The geometry used in the liner cell calculation is shown in Figure 6.2.

Figure 6.2 Liner Cell Geometry



The problem was set up for ANISN using Los Alamos 16-group cross-section sets [2, 3, 4] with P_1 scattering. The problem was calculated for an infinite cylinder with a "white" boundary condition on the outer diameter. The output from the calculation included flux and cell volume weighted 16-group macroscopic cross-sections.

The second part of the problem consisted of evaluating the criticality safety of the entire cask. This was also done using ANISN where the neutron cross-sections obtained from the liner cell calculation were used for the material region that included the liners. For the normal transport configuration, this region included the entire volume within the stainless liner bucket. The dimensions for the "close proximity" case, the dotted areas in Figure 6.1 were determined by considering the total volume for four liner cells and calculating an equivalent radius for that volume. The region between this volume and the stainless steel liner bucket was assumed to be void. One additional off-normal case was considered, the normal transport geometry-flooded. This was done by substituting water for void in the cell calculation to obtain cross-sections for the cask calculation.

6.2 APPENDIX

6.2.1 References

1. Engle, W.W., "A User's Manual for ANISN", K-1693, Union Carbide, Oak Ridge, Tenn., March 30, 1967.

2. Hansen, G.E. and Roach, W.H., "Six and Sixteen Group Cross-Sections for Fast and Intermediate Critical Assemblies", LAMS-2543, Los Alamos Scientific Lab, Los Alamos, New Mexico, November, 1961.

3. Connolly, L.D., et al.., "Los Alamos Group Averaged Cross-Sections," LAMS-2941, Los Alamos Scientific Lab., Los Alamos, New Mexico, July, 1963.

4. Personal Communication, Smith, D.R. to Walker, E.E., Los Alamos Scientific Lab, Los Alamos, New Mexico, February 26, 1971.

7.0 OPERATING PROCEDURES

This section provides generic instruction for loading and unloading the CNS 1-13G cask. Detailed procedures developed, reviewed, and approved following requirements of the CNSI Q.A. program are issued to authorized users.

7.1 PROCEDURES FOR LOADING PACKAGE

CAUTION: TREAT THE INSIDE SURFACES OF THE OVERPACK AND THE LINER OR DRUM IN THE CASK AS CONTAMINATED.

7.1.1 Remove the Overpack.

7.1.1.1 Remove the wire security seals.

7.1.1.2 Remove the ten (10) 2-inch bolts from the overpack.

7.1.1.3 Remove the two (2) 1-1/2 inch bolts from the side of the overpack.

7.1.1.4 Remove the tie-down cables/chains.

7.1.1.5 Attach crane to overpack and remove.

NOTE: THIS CASK MAY BE LOADED UNDERWATER. IF THIS IS THE CASE, FOLLOW THE PROCEDURE FOR WET LOADING, SECTION 7.1.3.

7.1.2 Procedure for Dry Loading.

CAUTION: DO NOT LIFT THE CASK BY THE LIFTING LUG ON THE CASK LID.

7.1.2.1 Remove cask from trailer, if necessary, by attaching crane to cask lifting lugs.

7.1.2.2 Remove the cask lid.

7.1.2.2.1 Remove the six (6) 1-inch bolts from the cask lid.

7.1.2.2.2 Attach crane to lid lifting lug and remove lid.

7.1.2.2.3 Visually inspect the lid gasket for any cracks or tears. Replace gasket if it is damaged. Inspect and clean the gasket seating surfaces.

7.1.2.2.4 Visually inspect the cask cavity to verify integrity. Remove any liquids or foreign material.

7.1.2.2.5 Visually inspect the baseplate to cask shell weld for any indications or other defects apparent to the naked eye. This weld should be indication free.

7.1.2.3 Using crane and suitable rigging, load the liner or drum into the cask.

NOTE: CLEAN THE LINER OR DRUM BEFORE REPLACING IT IN THE CASK, TREATING DEBRIS REMOVED AS CONTAMINATED.

7.1.2.4 Replace the cask lid.

7.1.2.4.1 Attach crane to lid and position on the cask using the alignment pins.

7.1.2.4.2 Visually inspect the cask lid closure bolts for signs of cracking or other visual signs of defects. Replace any defective materials.

NOTE: TIGHTEN ALL BOLTS HAND-TIGHT BEFORE STARTING THE TORQUE SEQUENCE.

7.1.2.4.3 Replace the six (6) 1-inch bolts on the cask lid. Torque bolts to 220 ± 22 ft.-lbs. (160 ± 16 ft.-lbs., if lubricated), using a star pattern.

7.1.3 <u>Procedure for Wet Loading</u>

CAUTION: <u>DO NOT</u> LIFT THE CASK BY THE LIFTING LUG ON THE CASK LID.

7.1.3.1 Remove the cask from the trailer by attaching crane to cask lifting lugs.

7.1.3.2 Remove the cask lid.

7.1.3.2.1 Remove the six (6) 1-inch bolts from the cask lid.

7.1.3.2.2 Attach crane to lifting lug and remove lid.

7.1.3.2.3 Visually inspect the cask cavity to verify integrity. Remove any liquids or foreign materials.

7.1.3.2.4 Visually inspect the baseplate to cask shell weld for any indications or other defects apparent to the naked eye. This weld should be indication free.

7.1.3.3 Prepare to lower cask into pool.

NOTE: TREAT ANY LIQUID FROM THE CASK DRAIN AS CONTAMINATED.

7.1.3.3.1 Remove the cask drain plug.

7.1.3.3.2 Visually inspect the drain plug for signs of cracking or other visual signs of defects. Replace if defective.

7.1.3.3.3 Ensure proper operation of cask drain line. Water should flow freely from the cask drain line. If water does not flow freely, inspect drain line for obstruction.

7.1.3.3.4 Attach crane to cask lifting lugs and lower into pool.

7.1.3.3.5 If necessary, detach crane from cask and load liner or drum into cask. DO NOT damage the gasket seating surfaces, the side of the cask, or inner walls.

7.1.3.4 Replace cask lid on cask.

NOTE: LID MAY BE REPLACED AFTER CASK IS REMOVED FROM POOL.

7.1.3.4.1 Visually inspect the lid gasket for any cracks or tears. Replace gasket if it is damaged. Inspect and clean the gasket seating surfaces.

7.1.3.4.2 Attach crane to cask lid and position on cask using the alignment pins.

7.1.3.4.3 Disconnect crane from lid.

7.1.3.5 Remove cask from pool.

7.1.3.5.1 Lift cask out of pool and allow water to drain into pool.

7.1.3.5.2 After cask is completely drained, place sealant on drain plug and reinstall into drain line.

NOTE: TIGHTEN ALL THE BOLTS HAND-TIGHT BEFORE STARTING THE TORQUE SEQUENCE.

7.1.3.5.3 Replace the six (6) 1-inch bolts on the cask lid. Torque bolts to 220 ± 22 ft.-lbs. (160 ± 16 ft.-lbs., if lubricated), using a star pattern.

7.1.3.5.4 Decontaminate all external surfaces of the cask.

7.1.3.6 Replace the cask on trailer.

7.1.3.6.1 Attach crane to cask lifting lugs and place on cask baseplate in the same configuration as received.

7.1.4 Replace the overpack.

7.1.4.1 Attach crane to overpack and place overpack into proper position over cask.

7.1.4.2 Visually inspect all overpack bolts for signs of cracking or other visual signs of defects. Replace any defective materials.

7.1.4.3 Replace the ten (10) 2-inch bolts in the baseplate. Torque bolts to 400 ± 40 ft.-lbs. (300 ± 30 ft.-lbs., if lubricated).

7.1.4.4 Replace the two (2) 1-1/2 inch bolts on side of the overpack. Torque bolts to 300 ± 30 ft.-lbs. (200 ± 20 ft.-lbs., if lubricated).

7.1.4.5 Place seal wires through the 1-1/2 inch bolts on side of cask overpack.

7.1.4.6 Replace chains/cables on overpack.

7.1.5 Before the cask leaves the facility, confirm:

7.1.5.1 That any external lifting lugs are properly covered.

7.1.5.2 That trailer placarding and cask labeling meet D.O.T. specifications (CFR Title 49, Part 172).

7.1.5.3 That exterior radiation levels do not exceed 200 mR/hr on contact, 10 mR/hr at 2 meters, and 2 mR/hr in tractor cab, in accordance with 49 CFR 173.441.

7.1.5.4 That outer package is sealed with anti-tamper seals.

7.1.5.5 That the cask drain plug is sealed and securely installed.

7.2 PROCEDURES FOR UNLOADING PACKAGE

NOTE: UPON RECEIPT OF CASK, PERFORM SURVEY FOR DIRECT RADIATION AND REMOVABLE CONTAMINATION USING APPROVED PROCEDURES TO ASSURE COMPLIANCE WITH APPLICABLE REQUIREMENTS OF 10 CFR 20.205.

7.2.1 Remove the overpack.

7.2.1.1 Remove the wire security seals.

7.2.1.2 Remove the ten (10) 2-inch bolts from the overpack.

7.2.1.3 Remove the two (2) 1-1/2 inch bolts from the side of the overpack.

7.2.1.4 Remove the tie-down cables/chains.

7.2.1.5 Attach crane to overpack and remove overpack.

7.2.2 Remove the cask lid.

CAUTION: TREAT THE UNDERSIDE OF THE LID, SURFACES OF THE CASK, ANY BOLTS OR SEALS REMOVED AS CONTAMINATED.

7.2.2.1 Remove the six (6) 1-inch bolts from the cask lid.

7.2.2.2 Attach crane to lid lifting lug and remove lid.

7.2.3 The health physics technician shall conduct a radiation and contamination survey to determine offloading precautions.

7.2.4 If directed by the health physics technician, vacate all personnel from the immediate area except the crane operator and rigger. The rigger shall stand in clear view of the crane operator.

7.2.5 Remove contents of cask.

7.2.5.1 Attach appropriate lifting device to the liner or drum.

7.2.5.2 Lift the liner or drum straight up out of the cask and allow any liquid to drip off into the cask.

7.2.5.3 Place the liner or drum in position for disposal or future handling.

7.2.6 The health physics technician shall survey the interior of the cask for radiation and contamination levels. Decontaminate if acceptable levels (as per site operations) are exceeded.

CAUTION: TREAT ANY LIQUID IN THE CASK OR USED IN THE DECONTAMINATION PROCESS AS CONTAMINATED.

7.2.7 Visually inspect the inside of the cask for damage or for liquid accumulation. Contact Health Physics for instructions to remove any liquids or foreign material from cask. If the inside surfaces of the cask are damaged, remove the cask from service.

7.2.8 Place new liner or drum into cask.

NOTE: CLEAN THE LINER OR DRUM BEFORE PLACING IT INTO CASK. TREAT DEBRIS REMOVED AS CONTAMINATED.

7.2.8.1 Using appropriate lifting gear and crane, carefully lower the liner or drum into cask cavity. DO NOT damage the gasket seating surfaces, the side of the cask, or inner walls.

7.2.9 Replace cask lid.

7.2.9.1 Visually inspect the lid gasket for any cracks or tears. Replace gasket if it is damaged. Inspect and clean the gasket seating surfaces.

7.2.9.2 Visually inspect the cask lid closure bolts for signs of cracking or other visual signs of defects. Replace any defective materials.

7.2.9.3 Attach crane to lid and position on the cask using the alignment pins.

NOTE: TIGHTEN ALL BOLTS HAND-TIGHT BEFORE STARTING TORQUE SEQUENCE.

7.2.9.4 Replace the six (6) 1-inch bolts on the cask lid. Torque bolts to 220 ± 22 ft.-lbs. (160 ± 16 ft.-lbs., if lubricated), using a star pattern.

7.2.9.5 Visually inspect the baseplate to shell weld for any indications or other defects apparent to the naked eye. This weld should be indication free.

7.2.10 Replace the overpack.

7.2.10.1 Attach crane to overpack and place overpack onto baseplate using the alignment pin.

7.2.10.2 Visually inspect all overpack bolts for signs of cracking or other visual signs of defects. Replace any defective materials.

7.2.10.3 Replace the ten (10) 2-inch bolts in the baseplate. Torque bolts to 400 ± 40 ft.-lbs. (300 ± 30 ft.-lbs., if lubricated).

7.2.10.4 Replace the two (2) 1-1/2 inch bolts on side of the overpack. Torque bolts to 300 ± 30 ft.-lbs. (200 ± 20 ft.-lbs., if lubricated).

7.2.10.5 Place seal wires through the 1-1/2 inch bolts on side of overpack.

7.2.10.6 Replace chains/cables on the overpack.

7.2.11 Before the cask leaves the facility, the following shall be confirmed:

7.2.11.1 That any external lifting lugs are properly covered.

7.2.11.2 That trailer placarding and cask labeling meet D.O.T. specifications (CFR Title 49, Part 172).

7.2.11.3 That exterior radiation and contamination levels conform to requirements as established in site radiation and contamination release procedures and D.O.T. requirements.

7.2.11.4 That the outer package is sealed with anti-tamper seals.

7.2.11.5 Check the cavity drain line to determine that the drain plug is properly installed using a pipe thread sealant.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

On December 21, 1978, Chem-Nuclear Systems, Inc. filed with the NRC a description of its Quality Assurance Program in accordance with 10 CFR 71.51(a). The Chem-Nuclear Systems, Inc. Quality Assurance Program was initially approved by the NRC on October 26, 1979, and Approval Certificate No. 0231 was issued. The Chem-Nuclear Systems, Inc. Quality Assurance Program was subsequently reapproved on September 6, 1983, and January 23, 1985. Under this certificate, the continued use of the program is authorized.

The Licensed CNS 1-13G Casks owned by Chem-Nuclear Systems, Inc. which are certified under the provisions of 10 CFR 71 and built after January 1979 are designed, fabricated, assembled, tested, modified, maintained and repaired in accordance with a Nuclear Regulatory Commission approved Quality Assurance Program (Docket #71-0231).

8.1 ACCEPTANCE TESTS

8.1.1 Visual Inspection

Visual inspection and dimensional verification of the entire cask and its accessories shall be performed to verify compliance with the requirements of appropriate drawings, specifications, applicable codes and other pertinent data indicating qualitative and quantitative acceptable criteria. All visual inspection shall be carried out either directly or remotely as described hereafter. The following shall be inspected: surface condition, dimension, finishes, shape, locations, details, size of holes, cleanliness, etc.

8.1.2 Structural Test

Lifting Lugs Load Test shall also be performed. The Load Test of all lifting lugs shall be performed in accordance with a procedure approved by Chem-Nuclear Systems, Inc. Each lug shall be load tested to one and a half times the calculated load capacity of the lug and held ten minutes as a minimum. After the load test, all welding on the lugs will be examined by dye penetrant testing using the procedure approved by Chem-Nuclear Systems, Inc. and performed by qualified personnel. Dye Penetrant Test shall meet the requirements of ASME Code Section III, Division I, Subsection NB, Article NB-5000 and Section V, Article 6. Test Reports shall be documented and included in the Quality Assurance Records of the cask.

8.1.3 Leak Tests

Seal Integrity shall be performed prior to first use of the cask. A Leak Test shall be applied to the cask to assure leak tightness of the seals. The cask shall be pneumatically pressurized to 15 psi and while under pressure, seals are soap bubble checked for leakage acceptance criteria - no visible bubbles. Test Report shall be documented and included in the Quality Assurance Records of the cask.

8.1.4 Component Tests

The Licensed CNS 1-13 G Casks owned by Chem-Nuclear Systems, Inc. which are certified under the provisions of 10 CFR 71 and built after January 1979, are fabricated, assembled and tested in accordance with a Nuclear Regulatory Commission approved Quality Assurance Program (Docket #71-0231).

8.1.4.1 Miscellaneous

The baseplate to the cask shell weld shall be liquid penetrant inspected in accordance with ASME Code Section III, Division I, Subsection NB, Article NB-5000 and Section V, Article 6, for the root pass and final weld. Test Reports shall be documented and included in the Quality Assurance Records of the cask.

8.1.5 Tests for Shielding Integrity

Test for Shielding Integrity shall be performed after the lead pouring operations. A Gamma Scan shall be applied to verify lead thickness, shielding capacity and to determine existence of any voids or impurities in the poured lead. The Gamma Scan procedure shall specify an acceptance criteria for verification that the nominal lead thickness is not less than five inches. The Gamma Scan must show no greater than ten percent loss of shielding at any point based on a four inch grid spacing. The Gamma Scan shall be performed by qualified personnel in accordance with a procedure approved by Chem-Nuclear Systems, Inc. Results of the Gamma Scan shall be documented and included in the Quality Assurance Records of the cask.

8.2 MAINTENANCE PROGRAM

The Chem-Nuclear Systems, Inc. maintenance requirements for the CNS 1-13G Cask are articulated in this section. These requirements reflect the specific operating conditions, limitations and regulatory requirements.

8.2.1 Structural Tests

A. All structural members and welds shall have been checked prior to initial use of the cask. Inspection of structural members and welds (except as specified in 8.2.1.D and 8.2.1.E are not required during routine use unless the cask has been involved in an accident or has been lifted improperly or in an overload condition. In those cases, inspection must be performed as follows:

Drop or Accident - All accessible structural members, welds, shall be visually inspected. In addition, all accessible welds must be dye penetrant tested. The Dye Penetrant Test shall be performed by qualified personnel using a Chem-Nuclear Systems, Inc. approved procedure. Dye Penetrant Test shall meet the requirements of ASME Code, Section III, Division I, Subsection NB, Article NB-5000 and Section V, Article 6. The Gamma Scan must be repeated and evaluated to the initial acceptance criteria. Test Reports shall be documented and included in the Quality Assurance Records of the cask.

Improper or Overload Lift - All welds on the primary or secondary lid which were used during the time of improper or overload lift shall be load tested and dye penetrant tested. Load and Dye Penetrant Test shall be performed in the same manner as delineated in the above paragraphs.

B. Whenever the cask requires total repainting and is sandblasted, all structural members and welds shall be visually inspected for any indications. Suspect members and welds shall be dye penetrant tested. Dye Penetrant Test shall be performed in the same manner as delineated in the above paragraphs.

C. Weld repairs, if any, shall be performed by qualified personnel using Chem-Nuclear Systems, Inc. approved procedures. Welding shall meet the requirements of ASME Code Section IX and/or AWS D1.1.

D. The baseplates to cask shell weld shall be visually inspected prior to each use in accordance with ASME Code Section III, Division I, Subsection NB, Article NB-5000 and Section V, Article 9.

E. The baseplate to cask shell weld shall be dye penetrant tested annually in accordance with ASME Code Section III, Division I, Subsection NB, Article NB-5000 and Section V, Article 6.

8.2.2 Leak Tests

Seal Integrity shall be performed on an annual basis. A Leak Test shall be applied to the cask to assure leak tightness of the seals. The cask shall be pneumatically pressurized to 15 psi and while under pressure, seals are soap bubble checked for leakage acceptance criteria - no visible bubbles. Test Report shall be documented and included in the Quality Assurance Records of the cask.

8.2.3 Gaskets on Containment Vessel

A. All gaskets shall be inspected for proper installation prior to each cask loading.

B. All gaskets shall be replaced once a year as a minimum regardless of condition.

C. Gaskets which are damaged must be replaced or repaired. Damages may include cuts, nicks, chips, indentations, or any other defects apparent to the naked eye which would affect sealing functions.

D. Any painted surfaces in contact with the gaskets shall be maintained in good condition. Any paint surface defect shall be properly retouched or repaired per paragraph 8.2.4.1.B.

8.2.4 Miscellaneous

8.2.4.1 Painted Surfaces

A. Painted surfaces shall be cleaned by steam or pressurized hot water using standard commercial equipment, chemical solutions, and procedures. There are no special precautions required in this cleaning operation.

B. Chipped or scratched surfaces shall be retouched or repainted using Chem-Nuclear Systems, Inc. approved paints.

C. Alignment stripes shall be repainted when they are chipped, peeled off, faced, or not legible. Only localized surface preparation (sanding and cleaning) is required prior to repainting of alignment stripes.

8.2.4.2 Fasteners

A. All fasteners shall be inspected for damage after each use. Fasteners shall be replaced if the following conditions exist:

- Deformed or stripped threads.
- Cracked or deformed hexes on bolt heads or nuts.
- Elongated or scored grip length area on bolts or studs.
- Severe rusting or corrosion pitting.

B. Fasteners shall be inspected for cleanliness and presence of lubricant in the threads prior to use. Any fastener found dirty shall be cleaned and relubricated.

ATTACHMENT A

POLYETHYLENE LINER ANALYSIS

POLYETHYLENE LINER ANALYSIS

The polyethylene used to fabricate the neutron source shield for the Model 1-13G package is a 5% boron-polyethylene composite. It is rated by the manufacturer for exposures up to 180^{0}F. This is a conservative value and the literature contains many references using this value as the lower limit of resistance to heat.

Above this temperature, the polyethylene base stock begins losing its structural rigidity, but actual thermal degradation or depolymerization of the melted polymer does not occur until a temperature of about 550^{0}F.[(1)] At temperatures above 680^{0}F this process occurs rapidly with volatilization of constituents.

Thermal analysis of the Model 1-13G package indicates that the cavity temperature does not exceed 400^{0}F under accident conditions. The polyethylene can therefore be expected to melt, but it is still substantially below the temperature of depolymerization where volatile constituents would be released. As a further precaution the Model 1-13G is shipped with a fuseable core plug which melts at about 200^{0}F whenever a polyethylene liner is used for supplemental neutron shielding.

Shielding analyses of the poly liner indicate that dose rates will not exceed the regulatory limit of 1 R/hr following accident conditions if the polyethylene is no longer effective.

[(1)] E. M. Fettes, Ed., Chemical Reactions of Polymers, Interscience Pub., Volume XIX.

A summary of these calculations is shown below:

MEASUREMENT LOCATION AND CONDITIONS	TRANSPORTATION LIMIT (mRem/hr)	CALCULATED TOTAL DOSE RATE FROM TYPICAL Cf-252 SOURCE IN 1-13G CASK (mRem/hr)	
Water filled cask, source centered in cavity:			Water information is for comparison only.
At side of fire shield -	200	67	
3 ft. from side of fire shield -	10	10*	
Borated polyethylene liner, source centered in cavity:			
At side of fire shield -	200	67	
3 ft. from side of fire shield -	10	10*	
Accident condition, poly shield ineffective, 3 ft. from side of fire shield:			
Source centered in cask cavity -	1000	303	
Source at side of cavity -	1000	505	

Package transport index is the limiting parameter

ATTACHMENT B

AUXILIARY SHIELDED CONTAINER ANALYSIS

TABLE OF CONTENTS

LIST OF TABLES

1.0 EQUIPMENT DESCRIPTION

1.1 GENERAL

The auxiliary shield specified in this report will provide a sealed, shielded container for handling sealed sources and operations involving both hot cell and pool loading of nonfissile, irradiated hardware components for loading and shipment within the CNS Model 1-13G, Type B, package. The steel and lead shielding is provided to minimize the radiation exposure to operators during the loading sequences prior to placement within the 1-13G package. The shielding also supplements the shielding in the cask body. The smaller size and weight of the auxiliary shield, as compared to the package body, greatly simplifies handling operations-- particularly in confined spaces. The auxiliary shield is designed to permit disposal at the burial ground. The auxiliary shield also functions to maintain a positional confinement of the radioactive material within a specified region of the package. This is required since certain of the radioactive materials have dimensions that are considerably smaller than the package internal cavity dimensions.

1.2 PACKAGING

The packaging consists of the shipping cask, described in Section 1.2.1 of the SAR.

1.3 CONTENTS

The package contents consists of the irradiated metal hardware which is contained within the auxiliary shielded container along with suitable shoring. The shipping configuration will be as shown in CNS Drawing 8651-C-01 (see Section 1.3 of the SAR).

1.3.1 Irradiated Metal Hardware.

Various types of irradiated components will be shipped in this configuration. The contents will be irradiated metal hardware (non-fissile), type B quantities. The minimum dimension of any individual component is nominally 3/16 inch. The radioactive content of the material will not exceed 39,000 Ci of Cobalt-60 or equivalent radioactive content.

1.3.2 Auxiliary Shield Container.

1.3.2.1 General. The auxiliary shield is a sealed container, constructed in accordance with the configuration shown in CNS Drawing 8651-E-02 (see Section 1.3 of the SAR). The auxiliary shield is of welded steel-lead composite construction. The bolted and gasketed joint of the closure lid is recessed within the shield body. Vent and drain provisions are provided to assure that the shield inner cavity can be drained after pool loading. The contents will always be shipped dry.

Auxiliary shields (or liners) have previously been approved for use with the 1-13G package. A comparison of the lead liner and the auxiliary shield is presented in Table One. The key differences are the full cavity height of the approved liner (51.25 inches), and a larger internal cavity than the auxiliary shield. The auxiliary shield is nominally 30.5 inches tall and will utilize wooden shoring (see CNS Drawing 8651-C-01) in the axial void formed between the shield and the package body. The weights of both vessels are approximately the same--nominally 5100 lbs.

1.3.2.2 Body. The auxiliary shield body is a solid vessel with a cylindrical shape and an internal cavity. The body external dimensions are nominally 30.5 inches in height by 24 inches in diameter. The internal cavity formed by the lid and the body is nominally 14 inches high with a 7.62 inch diameter. The body is of all welded-steel construction with encapsulated chemical grade lead used as the primary shielding material. The shield wetted surfaces (exterior & cavity) are epoxy painted for corrosion resistance and ease of decontamination. The steel wall thickness in the radial direction is 0.5 inches for both the inner and outer walls. The upper steel head is fabricated from two, 2-inch layers of steel plate. The shield bottom head is fabricated form 2-inch thick steel plate. The lead shield thickness varies from 5.5 to 7.18 inches to develop an equivalent lead thickness of approximately 7-3/4 inches. A stepped drain line extends from the cavity bottom to the side wall. The drain line is normally plugged. The sealing flange has eight 3/4-10 studs; two taper pins for lid positioning, and a groove to accommodate a 3/8-inch diameter rubber "O" ring. The "O" ring configuration is comparable to that used to seal other radioactive material. The body weight is approximately 4,900 lbs. A 3-point sling is used to lift the shield assembly.

1.3.2.3 Lid. The lid is fabricated from a 2-inch steel plate and encapsulated lead. It is recessed into the shield body. The stud/nut closure and "O" ring gasket in the body provide the sealed joint. Two alternative lids will be used depending on the contents. In the first lid design, provision has been made to alternatively load the shield cavity through a port in the lid. This port is formed by removal of a stepped lead plug and threaded cap. This is an operating simplification and would only be utilized when loading small hardware components. The lid has two 1/4-inch diameter vent lines with plugged ends angled into the steel head. The second lid design, which will be employed with larger components, does not have a stepped plug. The lids are lifted with a 3-point handling sling. The weight of both shield lids is nominally 200 lbs. Note that no attachments on the exterior of the assembled lid and shield body extend more than about one inch above the lid.

TABLE ONE

COMPARISON OF LEAD LINER AND AUXILIARY SHIELD

	Lead Liner	Auxiliary Shield
Construction	Welded Steel and Lead	Same
External Dimensions (HD), inches	53.25 x 26.25	30.5 x 24.0
Internal Cavity Dimensions (HD), inches	47.75 x 21.75	14 x 7.62
Internal Cavity Volume, ft^3	10.25	0.38
Steel Wall Thickness, inches	0.38	0.50
Encapsulated Lead Thickness, inches	1.50-2.50	5.5-7.18
Calculated Shield Weight, lb.	5150	5100
Bolted/Gasket Lid Joint . . .	No	Yes
Drain	Yes	Yes

2.0 STRUCTURAL ANALYSIS

2.1 General

The auxiliary shield and shoring are positioned within a licensed type B package. The weight of the loaded shield is within the range established in previously approved shields. The auxiliary shield is of steel-welded construction and contains the radioactive material within a sealed cavity. The bolted/gasketed joint forming the seal is comparable to that used for most nuclear shipping packages. The following analyses show that the shield confines the radioactive contents within the shield inner cavity, even during accident conditions. The bolted joint is adequate to maintain the lid on the auxiliary shield and prevent a loss of contents, even during a hypothetical drop of the package. Following the drop, any loss of shielding due to package deformation would be minimal. Also, the auxiliary shield is physically confined within the package during accident conditions.

2.2 30 FT. DROP

2.2.1 Corner Drop

A most conservative assumption is to examine the deformation to the auxiliary shield assuming that the shield experiences the 30 ft. drop independent of the package itself. The limiting drop configuration is the corner drop (over the shield center of gravity). This is analyzed as follows:

W = Maximum (Loaded) Shield WT = 5200 lb.

H = Drop Height = 360 in.

* σ_s = Dynamic flow pressure of the steel (A516-70) = 50,000 psi (min); 75,000 psi (max)

* NOTE: A range of values is needed to determine both the maximum crush depth and the maximum "G" loading on the lid bolts.

Define: V = Volume of crushed steel, in^3.
X = Crush depth, in.

Equating strain energy to potential energy,

$$\sigma_s V = W(H + X)$$

or

$$V = W(H + X) / \sigma_s \qquad \text{(Eq. 1)}$$

NOTE: $V = \phi(X)$

Also: Total force F, which is a function of the strain energy of steel deforming, is:

$$F(X) = \sigma_s \frac{dV}{dX}\ ; \quad F(X) = WG$$

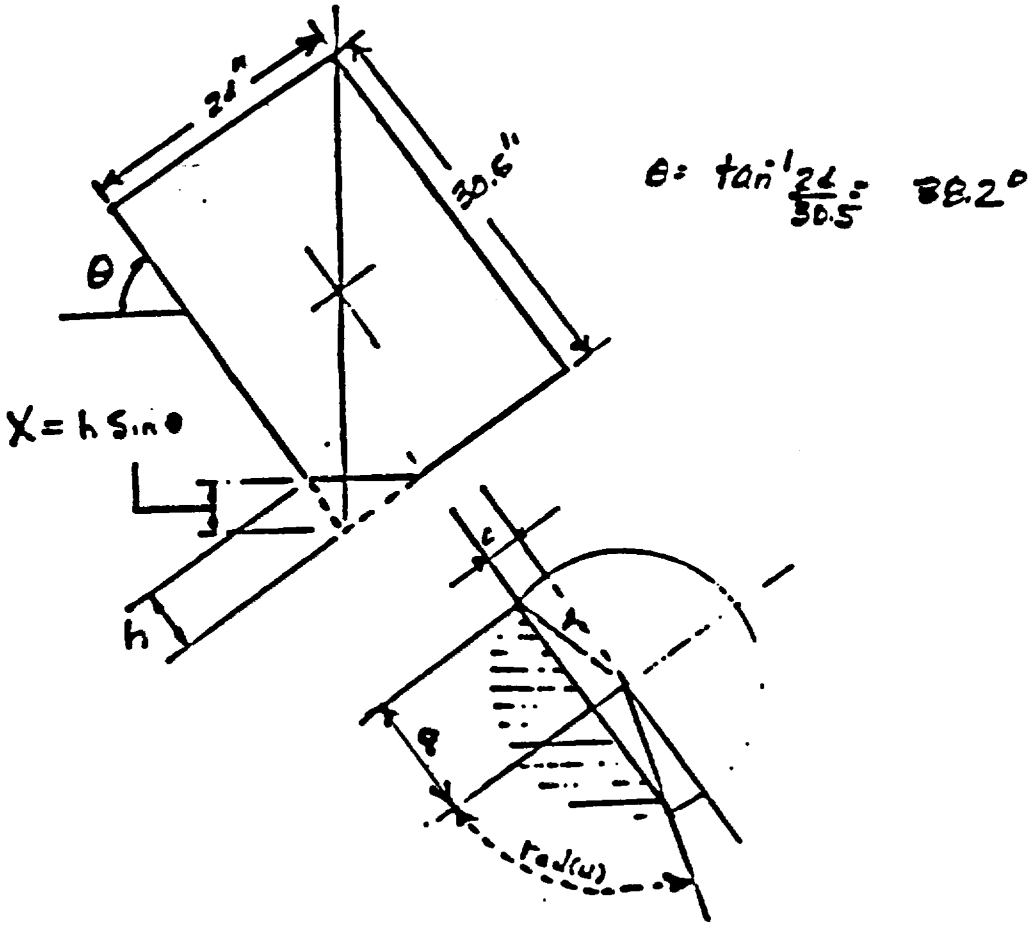
Hence:

$$G = \sigma_s \left(\frac{dV}{dX}\right) / W \qquad \text{(Eq. 2)}$$

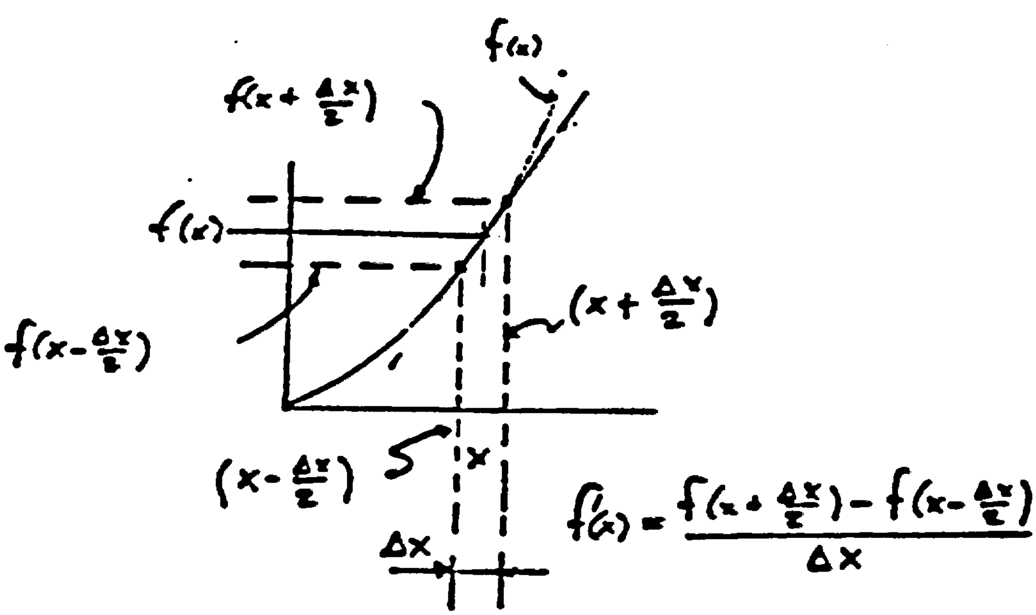
Where: $\frac{dV}{dX}$ = cross sectional area at the crush depth

The volume of crushed material is defined by the "Ungula" of a cylinder (Ref. 2).

$$V = \frac{h[a(r^{3}\frac{a^{2}}{3}) - r^{2}c \cdot rad(u)]}{(r - c)} \qquad \text{(Eq. 3)}$$



A numerical approximation as shown below was used to evaluate Eq. 3:



The solutions are as follows:

Case A--Maximum crush depth and volume of material deformed:

$V = 38.16 \text{ in}^3 \qquad X = 2.015 \text{ in.}$

$\frac{dV}{dX} = A = 46.85 \text{ in}^2$

G LOADING (PER EQ. 2)

$= \frac{50{,}000 \text{ PSI } (46.85 \text{ in}^2)}{5{,}350 \text{ LB.}} = 438 \text{ G}$

Case B--Maximum G loading on shield:

$V = 25.44 \text{ in.}^3$
$X = 1.71 \text{ in.}^2$

$\frac{dV}{dX} = A = 36.87 \text{ in.}^2$

$\text{G Loading} = \frac{75{,}000 \text{ psi } (36.87 \text{ in}^2)}{5350 \text{ lb}} = 517 \text{ G}$

From Case A: Note that the maximum deformation (2.01 in.) does not impinge on bolt circle. Conservatively, the maximum impact loading is 517 G.

2.2.2 Bottom and Side Drops

In order to estimate the maximum deformation of the cask due to any possible drop configuration of the 1-13G cask, an evaluation was made of both 30 foot bottom end and side drops. In both cases, the auxiliary shield was assumed to be totally independent of the package (this is very conservative since the auxiliary shield is contained within the cask). The method of analysis is as per Ref. 2. The calculations were used to determine the maximum possible decrease of lead shielding effectiveness.

2.2.2.1 Bottom Drop.

$$H = \frac{RWH}{\pi (R^2 - r^2)(t_s \sigma_s + k \sigma_{Pb})}$$

(From Ref. 2, pg. 63.)

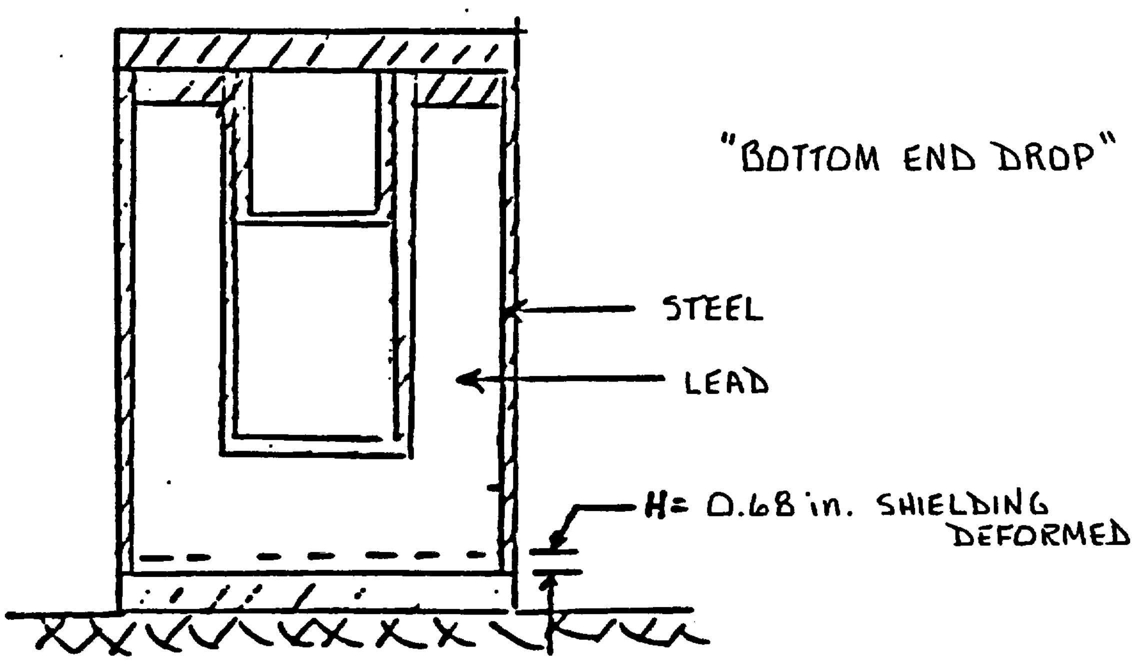
σ_s = 70,000 psi

σ_{Pb} = 5,000 psi

$$H = \frac{(11.5\,in)(5350\,lb)(360\,in)}{(3.14)(11.5\,in^2 - 4.32\,in^2)\left[(.5)(70{,}000\,psi) + (11.5)(5000\,psi)\right]}$$

H = 0.68 in.

This loss in shielding is in the upper region of the cask. Due to the recessed lid design, this will have no significant effect on shielding effectiveness.



2.2.2.2 Side Drop. This analysis is as per Section 2.7.2 of Ref. 2, evaluating the non-dimensional parameters used in the Figure 2.21 nomograph.

(1) $\frac{R}{t_s} \cdot \frac{\sigma_{Pb}}{\sigma_s} + 2\left(\frac{R}{L}\right)\left(\frac{t_e}{t_s}\right)$

$= \left(\frac{12 \text{ in.}}{.5 \text{ in.}}\right)\left(\frac{5,000 \text{ psi}}{70,000 \text{ psi}}\right) + 2\left(\frac{12 \text{ in.}}{30.5 \text{ in.}}\right)\left(\frac{3 \text{ in.}}{.5 \text{ in.}}\right)$

NOTE: $\bar{t}_e = \frac{2 \text{ in.} + 4 \text{ in.}}{2} = 3 \text{ in.}$

$1.71 + 4.72 = 6.43$

(2) $\frac{WH}{Rt_s \, L\sigma_s}$

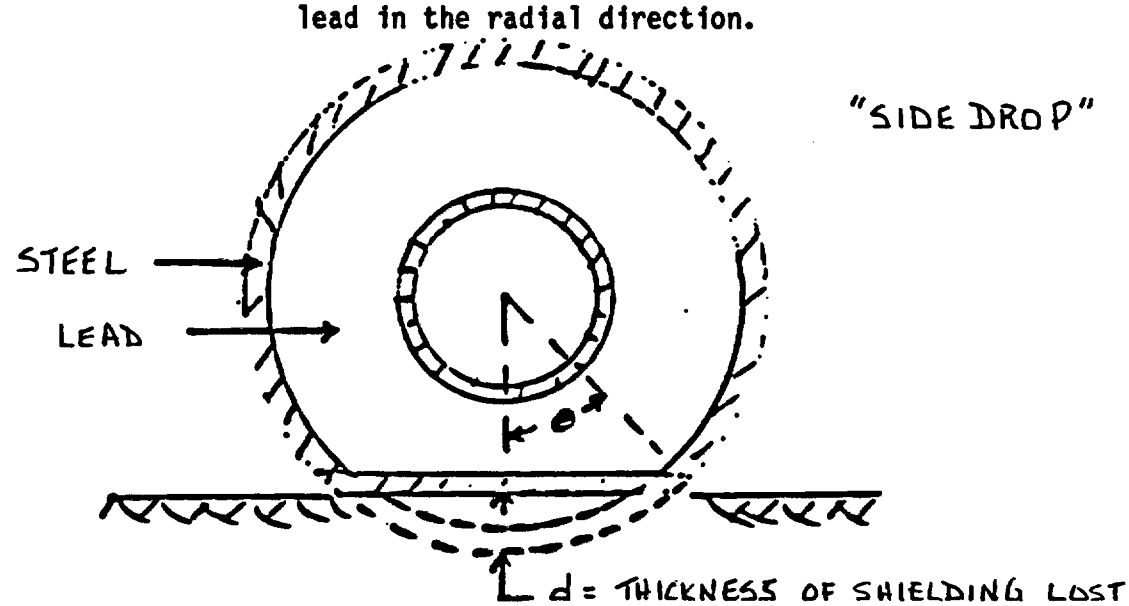
$= \frac{(5350 \#)(360 \text{ in.})}{(12 \text{ in.})(.5 \text{ in.})(30.5 \text{ in.})(70,000 \text{ psi})} = .146$

From plot: 2.21 ⇒ $\theta = 17.5^\circ$

$d = R(1 - \cos\theta) = 12(1 - \cos 17.5^\circ)$

$= 12(1 - .954) = .552$ in.

The radial deformation of only 0.552 in. (see sketch) will be of only minimal significance compared to the total shield thickness, which is comprised of 1-inch of steel plus 7.18 inches of lead in the radial direction.



2.3 CLOSURE ANALYSIS

Based on the previous analysis the structural adequacy of the bolts (studs) can be shown:

Number:	8
Size:	3/4"-10 UNC
	Cross-sectional area -- .334 in^2
Type:	A-193 GR 8 or A-325 GR 7
Ultimate:	110,000 psi

Bolt Load + g(W) x sin θ

$\theta = 38.20^0$, $\sin 38.20^0 = .62$

W = Lid Load = Wt. Lid + Wt. Contents

(NOTE: There is no internal pressure for this dry cavity.)

W = 300 lb. (max)--Assume Max Contents Wt. = 250 lb.

G max = 522 (From 2.2, previous.)

Bolt Load - 522 x 450 lb. x .62 = 145,640 lb.

$$\sigma_{Bolt} = \frac{\text{Bolt Load}}{\text{Area}} = \frac{145{,}640 \text{ lb.}}{(8)(.334 \text{ in}^2)} = 54{,}500 \text{ psi}$$

$$\text{M. S.} = \frac{110{,}000}{58{,}380} - 1 = 1.02$$

Hence, the bolted lid will remain sealed during any drop of the package. Note that this analysis is highly conservative, since any drop will be cushioned by the package.

2.3 PUNCTURE

The auxiliary shield is positioned within the cask with adequate shoring to prevent anything but minor movement of the shield. There are no sharp-edged components within the cask. The interior of the package cavity is steel, as is the exterior of the auxiliary shield. There are no radial protuberances to cause puncture in either the shield or the liner. In the axial direction, no protuberance on the auxiliary shield extends more than about one inch above the upper surface. The shoring will further serve to isolate the auxiliary shield and the cask inner liner in the axial direction. Hence, there is no possibility of loss of shielding in either the package or the auxiliary shield due to puncture.

3.0 CONTAINMENT AND SHIELDING ANALYSIS

3.1 SHIELD CONTENTS

A variety of irradiated hardware can be carried within the 13.5" x 7.62" D sealed cavity of the auxiliary shield. The following limits are imposed:

- Nonfissile
- Metallic form
- Min. dimension--3/16 inch
- Max. dimension--13.5 inch
- Total contents weight--250 lbs. (Max.)
- Cavity is shipped dry.

The structural analysis in Section 2.0 verifies that the hardware will remain within the auxiliary shield cavity -- even during the hypothetical drop accident. Also any loss of shielding due to lead deformation is less than 0.7 inches. Since the cask is functioning as the primary containment, a determination of the maximum curie content will not exceed the permissible dose after an accident or which will not exceed the permissible thermal limits is needed. The calculations are shown in Section 3.2.

3.2 SHIELDING ANALYSIS

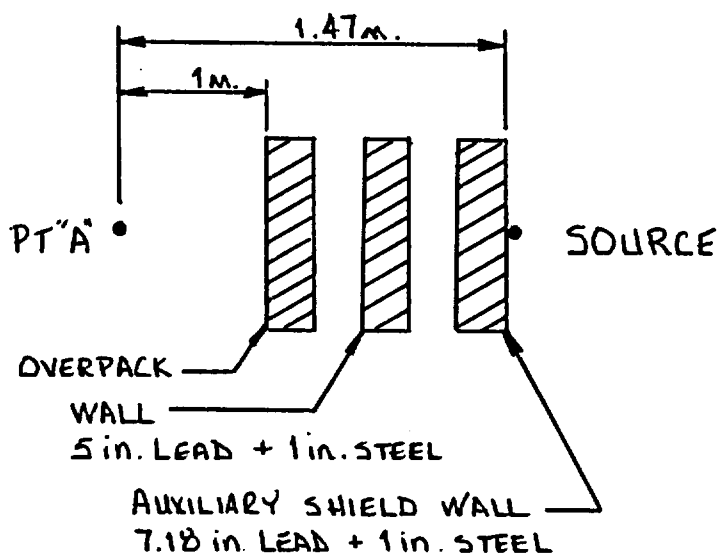
The basic criteria for carrying hardware in the package are as follows:

- Maximum package thermal limit: 600 watts.

- Dominant isotope in hardware: Co-60.

- Source is positioned at the inner wall of the auxiliary shield.

- Accident condition: Dose should not exceed 1 R/hr at one meter from the package surface.



The analysis will be based on postulating a point source configuration (which is most conservative).

To ascertain the contents Curie limit--the 600 watts thermal limit will determine the maximum Curie content to be transported:

For Cobalt-60 (2.5 MeV/disintegration released),

1 watt = 65 Ci of Co-60.

Hence, Max. Ci = 65 Ci/Watt x 600 Watts = 39,000 Ci of Co-60.

From Ref. 3, pg. 131.

For no shielding, 1 Ci of Co-60: 1.32 R/hr at 1 meter.

At a reference point 1 m from the package exterior,

Dose = $[(1)^2/(1.47)^2](1.32 \text{ R/hr}) = 0.610$ R/hr for 1 Ci,
or 39,000 Ci x 0.610 R/hr

= 23,820 R/hr, with no shielding

In the 1-13G in the radial direction, there is:
5 in. (12.7 cm) of lead
1 in. of steel.

From pp. 148, 149, Ref. 3, the attenuation factors are:

$$\beta\big|_{Pb=12.7\text{ cm.}} = 0.0012$$

$$\beta\big|_{STEEL=1\text{ in.}} = 0.55$$

Hence, solely utilizing the shield reduces the source strength at Point A to:

23,820 (.0012)(.55) = 15.8 R/hr.

Obviously, even after an accident the source will be
(1) confined within the auxiliary shield; and
(2) the auxiliary shield steel (1-2in.) and at least 5 inches of lead will be available for shielding.

How much lead and steel is required in the auxiliary shield to reduce dose to 1 R/hr after an accident?

By trial and error,

B/1" steel = 0.55
B/1.75" Pb(4.5 cm) = 0.09

15.8 R/hr (0.55)(0.09) = 0.79 R/hr which is less than 1 R/hr

Hence, only 1.75 inches of the 7-3/16 in. of lead shielding available (approx. 25%) is required to reduce dose to less than 1 R/hr.

As noted in Section 2.2, even under the most conservative assumptions, there will always be in excess of 6-1/2 inches of lead (equivalent) in the auxiliary shield available for shielding. Hence, the accident condition can be met.

4.0 THERMAL ANALYSIS

An evaluation was made of the thermal effect of the addition of the auxiliary shield to the cask shipment. The cask external temperature during normal conditions of transport is evaluated at steady-state and is dependent solely on the defined ambient conditions and the internal heat load. This does not change with the addition of the auxiliary shield.

The 30-minute fire accident is a transient analysis and will change due to the addition of the auxiliary shield. Examination was made by direct comparison to the SAR thermal analysis for the 1-13G cask (Section 3.0 of Ref. 1). In this analysis the fire accident was evaluated for the 600-watt internal heat load, considering two separate cases. A two-dimensional model was used. In the first case there was no liner in the package. In the second case the added effect of the liner was evaluated (see Table Two of this report). The liner case resulted in lower package lead and inner wall temperatures, due to the thermal "mass" of the liner. Table Two contrasts several key temperatures for these two cases. Note that the maximum temperature of the liner contents is less than 325°F. By direct comparison of the auxiliary shield and approved liner (again see Table Two), it can be assumed that the auxiliary shield (with equal mass, but thicker walls) will result in an equal-or-lower temperature than with the 1-13G without liner, which was acceptable.

TABLE TWO

COMPARISON OF FIRE TEST TEMPERATURES

Case	1-13G without Liner	1-13G with Liner
Max. 1-13G package lead temp., ^{o}F . .	479*	474*
Max. 1-13G package or liner inner wall temp., ^{o}F	425 (Package)	325 (Liner)
Max. 1-13G liner lead temp., ^{o}F	N.A.	325*
Heat load of contents, watts	600	600

(Ref. Section 3.0 (Thermal Test) and Section 3.0 Appendix of the SAR.

*Note: Lead melting temperature is 621^{o}F.

5.0 SUMMARY OF OPERATING PROCEDURES

5.1 GENERAL

The purpose of the auxiliary shield is to permit the separate handling of small irradiated components which are to be shipped in the cask. The smaller size and weight of the shield is a distinct operational advantage. The auxiliary shield container is loaded directly in the cavity of the package and all of the package operating procedures are maintained. However, special loading and unloading procedures are required for the auxiliary shield and these are summarized herein.

The auxiliary shield may be loaded with irradiated hardware either in a hot cell (dry) or in a pool (wet). The cask drain will be used during pool operations to assure that the contents are shipped dry. The shield may be loaded by <u>either</u> removing the closure lid <u>or</u> through the loading port (depending on which lid configuration is utilized). The port is an operational convenience to be used when handling smaller components.

The shielding of the auxiliary shield is sized to minimize operator radiation exposure during loading and unloading operations. Unloading consists of remotely removing the irradiated hardware and may include disposal of the auxiliary shield.

5.2 SUMMARY LOADING PROCEDURE

(1) Preliminary operations include:
  (a) removal of the closure lid or loading plug;
  (b) examination of "O" ring gasket or sealing surface;
  (c) removal of drain plug (for pool operations).

(2) Position the auxiliary shield in the loading area. In pool operations, the shield lifting sling will be employed. For hot cell operations, a wheeled dolly would be used. For certain pool operations the auxiliary cask may remain within the cask cavity.

(3) Load the hardware component(s) into the auxiliary shield cavity using remote handling tools.

(4) Replace the closure lid (or loading port plug) using the lid sling or remote handling tool.

(5) Remove the auxiliary shield from the pool (or hot cell).

(6) Drain the auxiliary shield cavity through the drain port (pool operations).

(7) Plug drain port and vent lines.

(8) Torque the closure lid nuts to 100 ft.-lbs. (or torque the threaded cap on the loading port).

(9) Wipe and dry the exterior of the assembled auxiliary shield.

(10) Load the auxiliary shield into the package cavity.

(11) Position shoring plug in the void region of the package, above the shield.

The remainder of the procedure is identical to current 1-13G package loading procedures.

5.3 SUMMARY UNLOADING PROCEDURES

This procedure is essentially identical to current 1-13G cask operating procedures, with the exception of:

(1) The auxiliary shield is removed from the package employing the lifting sling.

(2) Dispose of either the auxiliary shield or a S.C.-DHEC approved disposal container within the shield cavity. In the latter case, the shield lid will have to be remotely removed at the burial site.

6.0 GAMMA SCAN--ACCEPTANCE CRITERIA

A gamma scan test shall be performed on the auxiliary cask body and lid. The intent is to verify the quality of the lead utilized for gamma shielding.

The test procedure shall be in accordance with the provisions of the current revision of the CNSI gamma test.

The acceptance criteria are as follows:

(1) All gamma scanning will be performed on a 4" square grid system.

(2) The loss of shielding effectiveness measured shall not exceed 10% of the normal lead thickness in the direction measured.

7.0 REFERENCES

1. Consolidated Safety Analysis Report for the CNS Model 1-13G Package, dated November, 1987.

2. ORNL-NSIC-68, Cask Designer's Guide, Shappert, L., et al, February, 1970.

3. Radiological Health Handbook, dated January, 1970.