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GENERAL ATOMIC

GADR-55
REVISION 2

CONSOLIDATED DESIGN REPORT FOR FORT ST. VRAIN FUEL SHIPPING CASK

AUGUST 1980

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CONSOLIDATED DESIGN REPORT FOR FORT ST. VRAIN FJEL SHIPPING CASK

GENERAL ATOMIC PROJECT 1906
AUGUST 1980

CONSOLIDATED DESIGN REPORT FOR FORT ST. VRAIN FUEL SHIPPING CASK

Approved Robert L. Moore

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San Diego, California 92138

NOTICE

GADR-55 Revision 2 is a superseding consolidation of the following:

GADR-55, issued 17 April 1969
GADR-55 Supplement A, issued 15 November 1969
GADR-55 Supplement B, issued 22 May 1970
GADR-55 Supplement C, issued 15 June 1977

GADR-55 Revision 2 and GADR-55 Addendum 1, Revision 1 comprise the complete design report for the model FSV-1 radioactive materials shipping package.

CONSOLIDATED TABLE

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CONSOLIDATED TABLE
OF CONTENTS

INTRODUCTION

The complete design report for the model FSV-1 Spent Fuel Shipping Cask consists of the following volumes:

GADR-55, Revision 2 and

GADR-55, Addendum I, Revision 1.

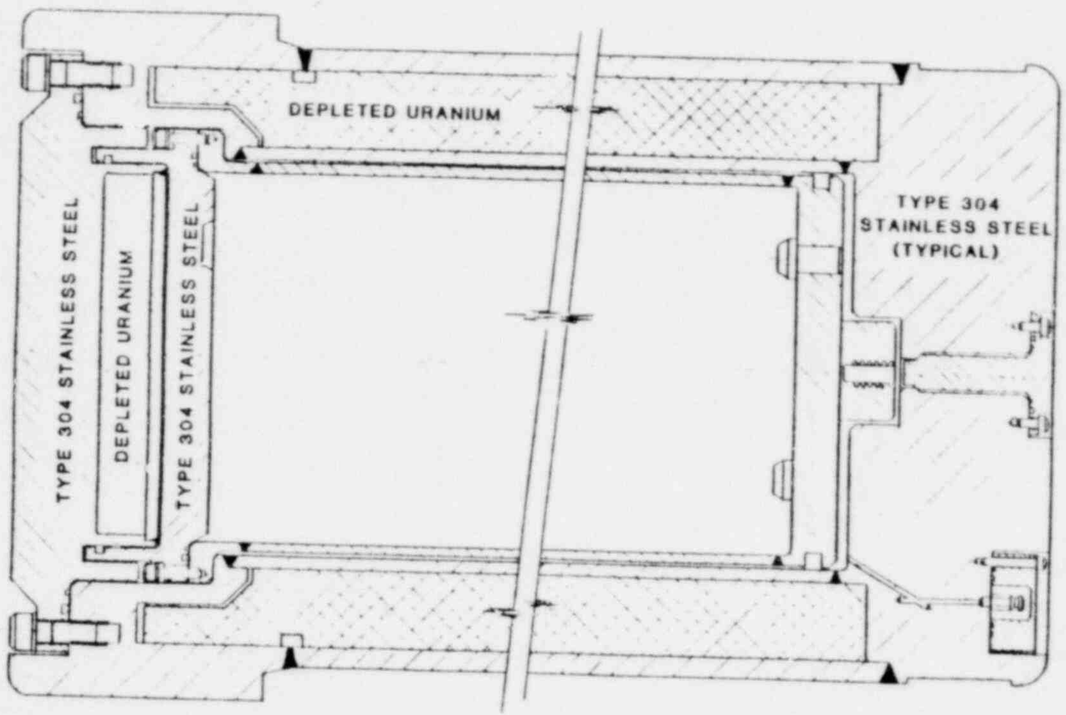
This design report is presented in separate volumes to better describe the two closure systems used with the same cask body. These closure systems are identified as follows:

NL Closure System (See Figure 1.)

Alternate Closure System (See Figure 2.)

The detailed description and the supporting analysis for the NL Closure System and the cask body is presented in GADR-55, Revision 2, "Consolidated Design Report for the Fort St. Vrain Fuel Shipping Cask". The detailed description and supporting analysis for the Alternate Closure System including the impact limiter is presented in GADR-55, Addendum 1, Revision 1, "Consolidated Design Report for the Fort St. Vrain Fuel Shipping Cask."

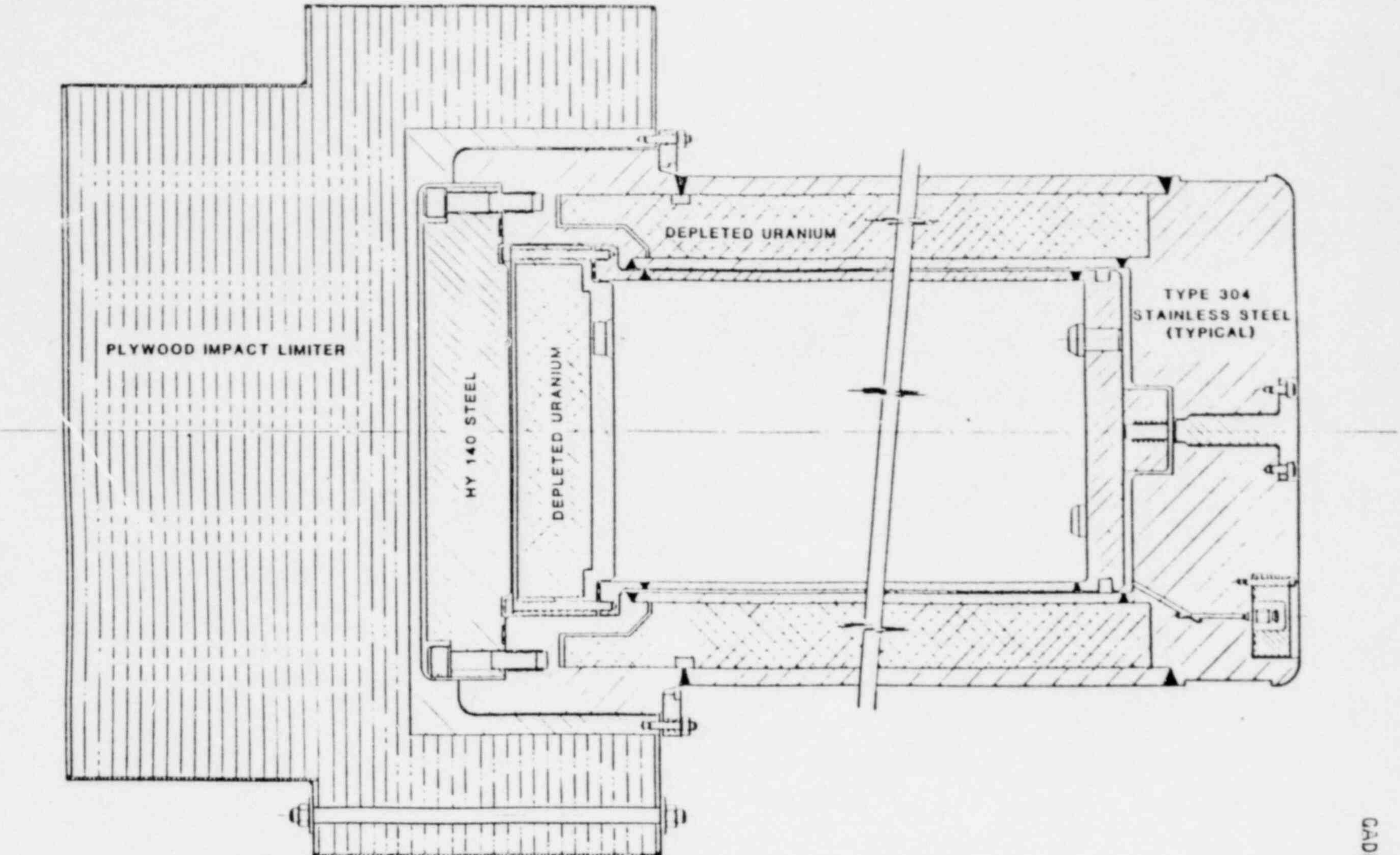
The General Atomic Company (GA) model FSV-1 Spent Fuel Shipping Cask is designed and fabricated to satisfy all of the requirements of Title 10, Code of Federal Regulations, Part 71, during the transport of spent fuel elements (see Figure 3) from the Fort St. Vrain, High Temperature Gas Reactor (HTGR). Six spent fuel elements, which are hexagonal graphite blocks containing fissile material, primarily U-233 and U-235 and fertile material, primarily thorium, are placed in the FSV-1 cask for each shipment. The FSV-1 cask consists of an inner container which provides the primary containment and a cask body which provides secondary containment, gamma shielding and structural integrity for the package. The principal materials of construction are stainless steel and depleted uranium. Silicone rubber seals are used for the inner and outer closures and the closure end of the FSV-1 cask is protected by a laminated plywood impact limiter (when the alternate closure system is used).



MODEL FSV-1 CASK WITH NL CLOSURE SYSTEM

FIGURE 1

(For details refer to GADR-55)



MODEL FSV-1 CASK WITH ALTERNATE CLOSURE SYSTEM

FIGURE 2

(For details refer to GADR -55 addendum I)

INTRODUCTION

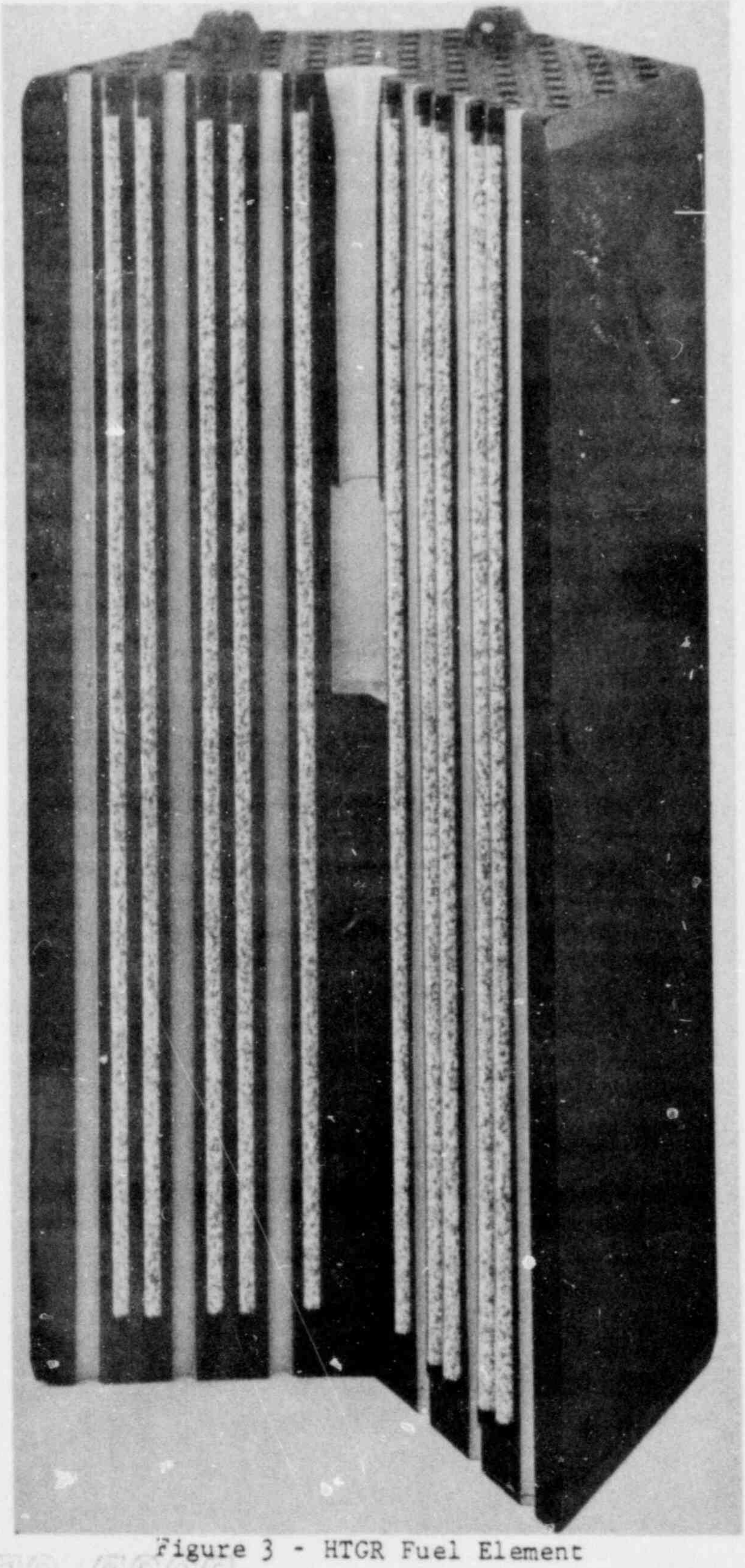


Figure 3 - HTGR Fuel Element

SECTION I

SYSTEM DESCRIPTION

SYSTEM DESCRIPTION

A. PURPOSE

The purpose of the model FSV-1 Spent Fuel Shipping Cask and the associated equipment is the transport of irradiated nuclear fuel elements and other radioactive materials.

B. NORMAL OPERATING REQUIREMENTS

The model FSV-1 cask is designed for a capacity of 6 fuel elements of the type used in the Fort St. Vrain nuclear generating station. These fuel elements are hexagonal graphite blocks that measure 14.17 inches across the flats by 31.22 inches long and weight approximately 300 pounds.

The model FSV-1 cask with its associated equipment is designed and fabricated to satisfy the requirements of 10 CFR 71 for the normal conditions of transport during the shipment of spent fuel elements and other radioactive materials.

Special interface criteria were used in the design of the FSV-1 cask and its associated equipment to assure its compatibility with the fuel element handling equipment in the Fort St. Vrain nuclear generating station. The interface with this equipment is described in the operating procedures that follow.

The FSV-1 cask is constructed of selected materials (primarily stainless steel) to allow the use of normal decontamination procedures and solutions.

C. ABNORMAL OPERATING REQUIREMENTS

The FSV-1 Spent Fuel Shipping Cask is designed to satisfy the containment requirements of the accident conditions of transport as specified in 10 CFR 71.

D. EQUIPMENT LIST

The following items of equipment will be used in the normal operation of transporting refabricated and spent fuel elements, and the transfer of same, from and to, the Fort St. Vrain fuel storage wells:

Fuel Shipping Cask

Spent Fuel Container

Extendable Truck-Trailers with Cask Supports and Tie-down Structures

Cask Lifting Apparatus

Adapter Seal Ring

Fuel Loading Port in Reactor Building Refueling Floor

Reactor Isolation Valve

Auxiliary Transfer Cask and Grapple

Temporary Shield Plug

Hot Service Facility Jib-hoist

Reactor Building 160/50 Ton Crane

Reactor Building Auxiliary 10 Ton Crane

Fuel Transfer Cask and Fuel Handling Machine Assembly

Pneumatic Right Angle Wrench

Hand Torque Wrenches (2)

E. NORMAL OPERATING PROCEDURE

At some future date the Fort St. Vrain Fuel Shipping Cask may be utilized to ship refabricated fuel elements to Fort St. Vrain as well as to ship spent fuel elements to a reprocessing facility. Beginning with the arrival of a fuel shipping cask at Fort St. Vrain the sequence of unloading refabricated fuel and loading spent fuel will be as follows:

1. Transfer of the fuel shipping cask from the truck to the fuel loading port.
   a) The tractor and trailer are backed into the reactor building to the desired depth.
   b) The brakes of the rear wheels of the trailer are locked.
   c) The extendable trailer slide locks and the rear fuel shipping cask support locks to the trailer are unfastened.
   d) The tractor is then driven in reverse to close the trailer and slide the fuel shipping cask back over the rear trailer wheels.

      (This is necessary to position the fuel shipping cask under the hatches in floors above the truck bay and to close the outside doors of the truck bay).
   e) Manually detach the personnel barrier from the truck.
   f) Using the HSF jib-hoist, remove the fuel loading port cover plate and store it on the refueling floor. Then lift the seal adapter (Shipping Cask to isolation valve) from its stored position in the loading port and place

it on the refueling floor.

g) Open the hatches in the deck immediately above the truck bay and in the refueling floor.

h) Lower the 10-ton auxiliary building crane hook thru the hatches and pick up the personnel barrier. Raise the personnel barrier and store it on the deck above the truck bay.

i) Lower the 50-ton building crane with the fuel shipping cask lifting apparatus attached to the hook thru the hatches to the desired position at the head of the fuel shipping cask.

j) Engage the cask lifting apparatus to the fuel shipping cask using the two holes provided in the head of the cask.

k) Disengage the cask head tie-down and support from the fuel shipping cask.

l) Raise the 50-ton crane hook until the fuel shipping cask is in vertical position.

m) Disengage the fuel shipping cask from the cask bottom tie-down and support.

n) Raise the fuel shipping cask to a position above the refueling floor.

o) Move the fuel shipping cask to the loading port. Rotate the crane hook to align the legs of the cask lifting apparatus to the guide plates in the loading port.

p) Lower the fuel shipping cask into the loading port.

q) Unlock the cask lifting apparatus and disengage it from the fuel shipping cask. Temporarily store the cask lifting apparatus on the refueling floor.

r) Close the hatches in the truck bay deck and the refueling floor.

2. Unloading and loading the Fuel Shipping Cask.

a) Connect the helium purge system to the connection provided in the bottom of the fuel shipping cask.

b) Pull 20-inch Hg vacuum on the fuel shipping cask cavity to check the seals. Monitor the helium tube connection for radioactivity. If radioactivity is measured, reduce to a 4-inch Hg vacuum and maintain while the cask cover is being removed. (This should prevent radioactive gas from escaping to the refueling floor upon removal of the cask cover).

c) Remove the 3 set-screw type plugs from the top of the cask cover and screw three (3) eye-bolts into place.

d) Position the 10-ton auxiliary crane with the sling-HSF (three leg wire rope sling) over the fuel shipping cask. Engage the legs of the sling to the eye-bolts in the cask cover.

e) Pick up from the refueling floor the temporary shield plug using the HSF jib-hoist, and move it to a position adjacent to the fuel loading port.

f) Unbolt the fuel shipping cask cover.

g) Raise the fuel shipping cask cover to a height which allows swinging the jib-hoist and temporary shield plug over the center of the fuel shipping cask. Lower the temporary shield plug into the lid of the spent fuel container inside the shipping cask. Disconnect the temporary shield plug from the jib-hoist hook and swing the jib-hoist away from the fuel loading port. Move the shipping cask cover to the hot service facility for inspection of bolt threads and seals.

h) Using a pneumatic right-angle wrench with a 15-inch-long socket extension, unbolt the spent fuel container lid bolts.

i) Using the HSF jib-hoist pick up the adapter seal and position it on top of the fuel shipping cask.

j) By means of the 50-ton building crane move a reactor isolation valve from storage and position it on top of the fuel loading port. Anchor and level the reactor isolation valve using the 3-inch-diameter bolts which are a part of the valve assembly.

k) Open the reactor isolation valve and lower the HSF jib-hoist hook through the valve and engage the eye-bolt in the temporary shield plug.

l) Raise the temporary shield plug out of the spent fuel container lid through the isolation valve and place the plug on the refueling floor.

m) Close the isolation valve and inflate the seal on the bottom of the isolation valve to effect a seal with the top surface of the seal adapter.

n) Using the helium purge system at the bottom of the fuel shipping cask insure that all seals are effective by pulling and holding a 20 inch of Hg. vacuum. If seals are effective, return the cask cavity pressure to ambient building pressure and shut off the helium purge system at this position.

o) Re-reeve the building crane from 50-ton capacity to 160-ton capacity.

p) Using the 160-ton capacity building crane move the auxiliary transfer cask from its storage location to on top of the reactor isolation valve on the fuel loading port. Open both the isolation valve and the valve on the bottom of auxiliary cask.

q) Lower the auxiliary transfer cask grapple into the lid of the spent fuel container. Engage the lid and remove it from the fuel shipping cask.

r) Close both the reactor isolation valve and the auxiliary transfer cask valve.

s) Move the auxiliary transfer cask to the hot service facility and lower the spent fuel container lid into the HSF for decontamination and for inspection of bolt threads and seals.

t) Using the 160-ton building crane move the fuel handling machine and the fuel transfer cask (one assembly) from a fuel storage well; with spent fuel elements in approximately one half of the fuel transfer cask, to the top of the isolation valve on top of the fuel loading port.

u) Using the helium purge system connections and controls attached to the reactor isolation valve purge the fuel shipping cask cavity.

v) Open the isolation valve and the valve in the bottom of the fuel transfer cask.

w) Using the fuel handling machine remove the refabricated fuel elements from the spent fuel container inside the fuel shipping cask and place them in the unloaded portion of the fuel transfer cask.

x) After the six (6) refabricated fuel elements have been transferred to the fuel transfer cask, use the fuel handling machine to load six (6) spent fuel elements into the spent fuel container inside the fuel shipping cask.

y) Close both the reactor isolation valve and the valve in the bottom of the fuel transfer cask.

z) Using the 160-ton building crane remove the fuel handling machine and fuel transfer cask from the fuel loading port to a fuel storage well.

3. Transfer of the Fuel Shipping Cask from the fuel loading port to the truck.

a) Move the auxiliary transfer cask with the spent fuel container lid from the hot service facility to the reactor isolation valve on the fuel loading port using the 160-ton building crane.

b) Open the reactor isolation valve and the valve in the bottom of the auxiliary transfer cask.

c) Lower the spent fuel container lid onto the container inside the fuel shipping cask.

d) Raise the auxiliary transfer cask grapple and close both the reactor isolation valve and the valve in the bottom of the auxiliary transfer cask.

e) Using the 160-ton building crane, move the auxiliary transfer cask from the fuel loading port to its storage position .

f) Pick up the temporary shield plug with the HSF jib-hoist and swing it to a position over the reactor isolation valve on top of the fuel loading port.

g) Open the reactor isolation valve and lower the inner shield plug into the spent fuel container lid. Disconnect the hoist hook from the plug and swing the HSF jib-hoist away from the fuel loading port.

h) Re-reeve the building crane from 160-ton capacity to a 50-ton capacity.

i) Using the 50-ton building crane remove the reactor isolation valve from the fuel loading port and place it in its storage position.

j) Using a pneumatic right angle wrench with a 15 inch socket extension, bolt the spent fuel container lid to the container. Final tightening of the bolts should be accomplished using a hand torque wrench seating the bolts to the specified torque of 90 ft.-lbs.

k) Using the 10-ton auxiliary building crane retrieve the fuel shipping cask cover from the hot service facility and move it to a position above the fuel shipping cask in the fuel loading port.

l) Swing the HSF jib-hoist over the fuel loading port and engage the hoist hook to the temporary shield plug.

m) Raise the temporary shield plug out of the fuel shipping cask and swing the jib-hoist away from the fuel loading port. Immediately lower the fuel shipping cask cover onto the fuel shipping cask.

n) Bolt the cask cover to the fuel shipping cask. Use a hand torque wrench to tighten the bolts to the specified torque of 1,000 ft-lbs.

o) Open the helium purge system valve at the connection to the bottom of the fuel shipping cask. Pressurize the fuel shipping cask cavity to 50 psig and close valve. Observe the pressure gage at the helium purge systems outlet. If at the end of one hour the pressure reading has remained constant at 50 psig the fuel shipping cask seals are effective.

p) Switch the helium purge system from pressure to vacuum and draw a 20-inch Hg. vacuum on the shipping cask cavity. Use a radiation monitor held next to the helium purge system tubing near the bottom of the fuel shipping cask to detect the exhausting of any radioactive gas. If no radioactivity is present, the spent fuel container seals are effective.

q) Observe the pressure gage at 20-inch Hg. vacuum for one hour. If no pressure decay occurs, all seals are effective and the fuel shipping cask loaded with spent fuel is ready to ship.

r) Return the fuel shipping cask cavity to atmospheric pressure using helium. Disconnect the helium purge system tubing from the bottom of the fuel shipping cask.

s) Secure all bolts in the fuel shipping cask cover and cask bottom by lock-wiring per MIL-STD-763A.

t) Remove the three eye-bolts from the fuel shipping cask cover and screw in the set-screw type plugs.

u) Open the hatches in the refueling floor and deck above the truck bay.

v) Pick up the fuel shipping cask lifting apparatus with the 50-ton building crane and move it to the fuel loading port.

w) Lower the fuel shipping cask lifting apparatus and engage the head of the fuel shipping cask.

x) Raise the fuel shipping cask out of the fuel loading port and move it to the hatch in the refueling floor.

y) Lower the fuel shipping cask thru the hatches until the cask bottom engages the rear cask support on the shipping cask truck trailer.

z) Continue to lower the 50-ton crane hook (which allows the fuel shipping cask to go to a near horizontal position) until the fuel shipping cask head comes to rest on the front cask support.

aa) Unlock the cask lifting apparatus and disengage it from the fuel shipping cask. Move the cask lifting apparatus to its storage position.

bb) Secure (tie down) the head of the fuel shipping cask to the front cask support.

cc) Using the 10-ton auxiliary building crane pick up the personnel barrier from the truck bay deck and place it in position over the fuel shipping cask on the truck trailer. Secure the barrier in place.

dd) Close all hatches before opening the truck bay door. Open the truck bay door and drive the tractor forward until the trailer is extended and the rear shipping cask support is moved forward to transit position. Lock both the trailer extendable slides and the rear fuel shipping cask support in place.

ee) Using a radiation monitor check the dose rate on the personnel barrier and at 6 feet from the barrier. Radiation levels must be equal to or less than the levels specified in 10-CFR-71.

ff) Drive the fuel shipping cask truck, tractor and loaded cask out of the reactor building. Close the truck bay door.

F. ABNORMAL OPERATING PROCEDURE

There are no abnormal operations anticipated for this system.

SECTION II

PHYSICAL AND FUNCTIONAL.DESCRIPTION

PHYSICAL AND FUNCTIONAL DESCRIPTION

A. GENERAL

The spent fuel container and fuel shipping cask is sized to transport six HTGR fuel elements weighing 300 pounds each, loaded in a single column. Design of all equipment is such that the gross weight of the FSV-1 cask with the tractor/trailer transport will be an absolute minimum and still satisfy all NRC and DOT regulations.

B. SPENT FUEL CONTAINER AND LID

The spent fuel container, fabricated entirely of type 304 stainless steel, is 190-11/16 inches overall in length. It has an inside diameter of 16-5/8 inches and a wall thickness of 1/2 inch. The top flange of the container is 1-1/4 inches thick with a diameter of 20-1/2 inches.

The container lid is designed to be compatible for lifting with the auxiliary transfer cask grapple. It has an overall diameter of 20-1/2 inches and is mated with the top flange of the shipping container by use of twelve special fasteners. The fasteners are 1/2 inch diameter, high strength, cadmium plated allow steel.

The assembled spent fuel container with lid will have a cavity length of 187-5/8 inches, an overall length of 195-5/16 inches, and will weigh approximately 1880 pounds.

Two "O" rings, one metallic and the other a high-temperature elastomeric polymer, will provide the primary seal for the contents of the container. A dovetail groove for the elastomer "O" ring is provided in the container lid, and the metal "O" ring is secured to the lid by stainless steel wire clips and screws.

A 3/4 inch by 3/4 inch annulus on the outside diameter of the base of the fuel container will be provided for upending the container--a secondary means of fuel removal.

Three orientation dowel pins will be welded to the inside base of the fuel container for proper positioning of the loaded fuel. In connection with this, three dowel pin holes will be drilled in the bottom of the container lid. These holes will be aligned with the dowel pins when the lid is bolted to the container. For further orientation, the container lid and top flange will be match slotted for proper positioning to a key in the shipping cask.

C. SHIPPING CASK AND COVER HEAD

The shipping cask is designed to hold the spent fuel container and will include provisions for radial orientation in the reactor building fuel loading port so that fuel elements are oriented the same as when stored in the reactor building fuel storage wells.

Basically, the cask design consists of a depleted uranium cylindrical shell encased in a stainless steel inner liner and outer shell. The inside diameter of the depleted uranium shielding will be 19 inches, with a wall thickness of 3-1/2 inches. The stainless steel inner liner will be 5/8 inch thick, and the outer shell will be 1 inch thick.

The base of the cask will be a stainless steel forging 9 inches thick and 28 inches in diameter. Two penetrations will be in the base of the shipping cask. A center-line penetration, nominally 1-1/2 inches in diameter, will serve as an opening for a ramrod which will be used to remove fuel from the shipping cask when the cask is in a hot cell and lying in a horizontal position. This penetration will also provide a drain for the shipping cask cavity for decontamination and cleaning operations. Normally, this penetration will be plugged by the base closure pin, a stainless steel weldment which will be bolted to the cask base. Metallic and an elastomeric polymer "O" rings will provide the seal for this penetration.

A 1/4-inch diameter hole extending through the cask base at a point approximately eight inches off the center is terminated on the base of the cask and is opened by use of a protected quick-disconnect nipple. This connection provides for the use of the Fort St. Vrain helium purge system for controlling the gas contents in the shipping cask cavity and for testing the effectiveness of seals.

The top section of the cask will consist of a stainless steel ring forging 31 inches in diameter and approximately 12 inches deep. This machined ring will provide the vertical support for the shipping cask when it is in the fuel loading port. The ring will also blend in with the 28-inch diameter main body of the cask, provide an annular protection and base for the closure head, provide a base for the top flange of the spent fuel container, and provide for the cask hoisting apparatus.

The cask cover is bolted to the cask with twenty-four 1-1/4-inch alloy steel cap screws. The thickness of the stainless steel head will be 4-5/16 inches. An upper shield plug of unalloyed depleted uranium 2-1/4 inches thick will be an integral part of the cask cover. A dovetailed "O" ring groove for the elastomeric polymer "O" ring and a metallic "O" ring secured to the cover by stainless steel wire clips and screws provide the seal for the cover to cask closure.

D. PHYSICAL CHARACTERISTICS

The spent fuel container and shipping cask, materials, dimensions, and weights are as follows:

1. Shielding material: unalloyed depleted uranium w/0.2% to 0.3% molybdenum. Density = 0.683 lb/in.^{3}.
2. Shells, liners, plates, etc.: type 304 stainless steel.
3. Overall shipping cask length = 208 inches.
4. Diameter of shipping cask main body = 28 inches.
5. Diameter of shipping cask top section = 31 inches.
6. Weight of fully loaded shipping cask = 46,500 pounds.

E. DRAWINGS

The following drawings depict the general assembly and functional details of the spent fuel container and shipping cask:

NL Dwg. 40065D, Rev. 1, "Spent Fuel Container & Shipping Cask. Cask Assembly - Pictorial Section"

NL Dwg. 70085F, Rev. 5, "Spent Fuel Container & Shipping Cask Assembly"

NL Dwg. 70086F, Rev. 7, "Spent Fuel Container & Shipping Cask. Top Section"

NL Dwg. 70094F, Rev. 5, "Spent Fuel Container & Shipping Cask. Bottom Section"

NL Dwg. 70296F, Rev. 2, "Spent Fuel Container & Shipping Cask Layout"

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SHEET 1 OF 1

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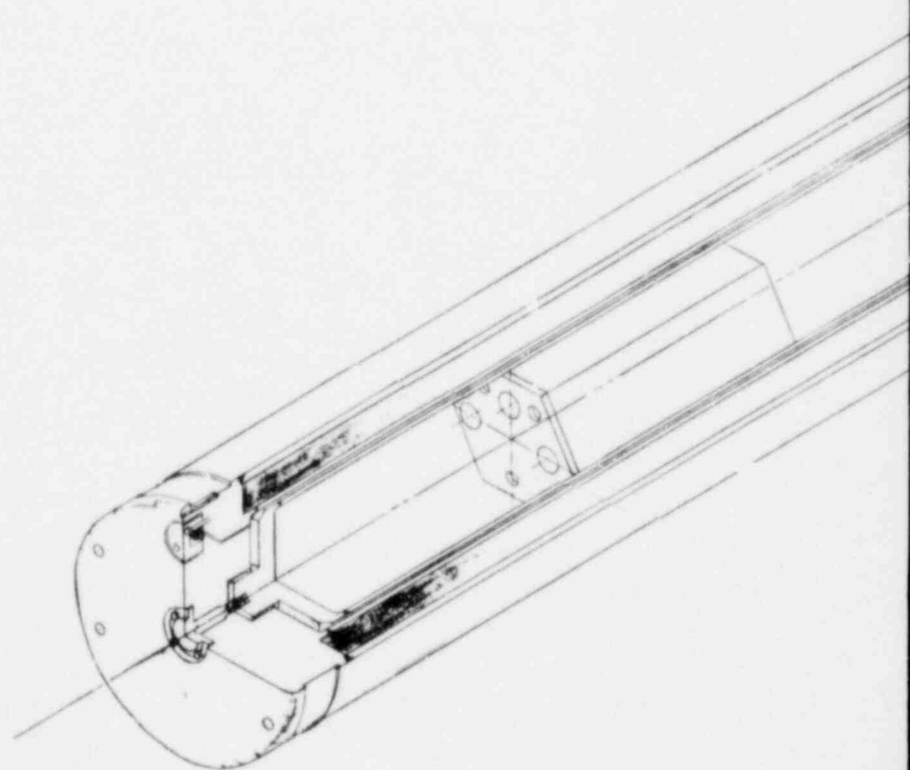

POOR ORIGINAL

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REVISIONS			
ZONE	REV	DESCRIPTION	
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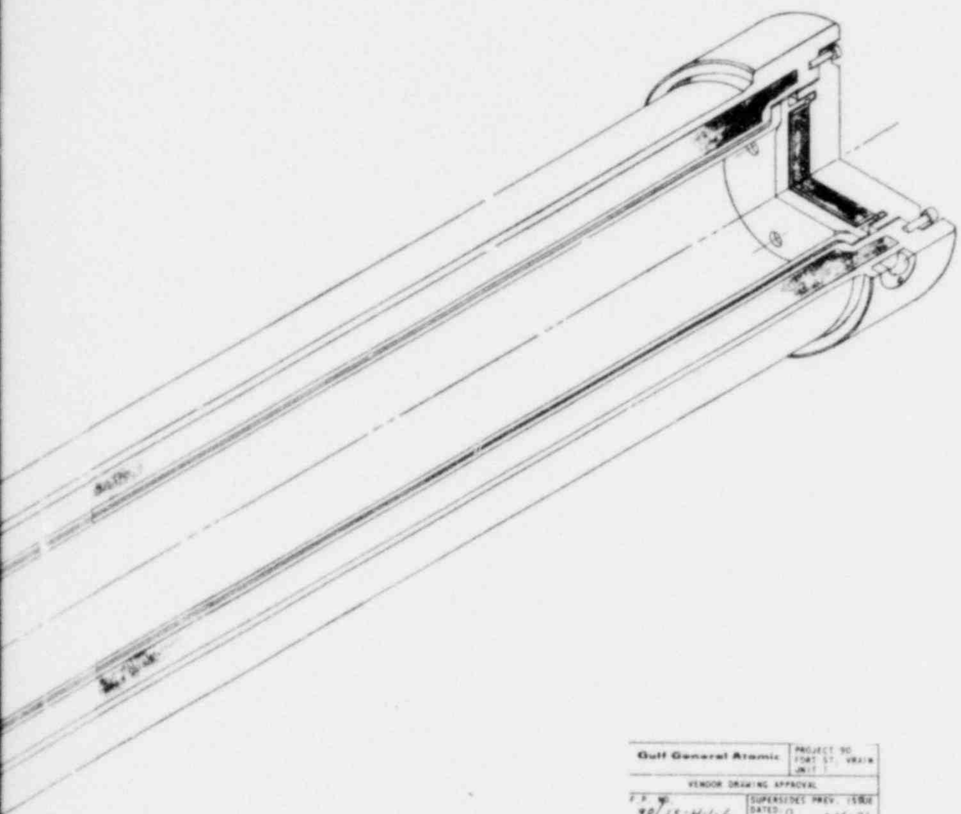

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Gulf General Atomic — PROJECT 90 FORT ST. VRAIN UNIT 1

VENDOR DRAWING APPROVAL

F.P. NO. 90/15-H-1-6 — SUPERSEDES PREV. ISSUE DATED: June 24-71

☐ 1. NOT APPROVED. REVISE AND RESUBMIT

☐ 2. APPROVED

☐ COMMENTS AS NOTED

☐ SUBMIT CERTIFIED TRANSPARENCY

☐ FINAL. NO FURTHER REQUIREMENTS

BY: DATE:

REVIEWED BY:

GGA P.S. NO. 915-H01 — EQUIP. NO. H1501-H1503

THIS ACTION DOES NOT RELIEVE THE SELLER OR CONTRACTOR OF ANY OBLIGATIONS UNDER THE CONTRACT

ITEM	QTY	NAME	MATL	SPEC	DWG Nº	DESCRIPTION

LIST OF MATERIAL

	DATE
DRAWN JJM	3-12-69
CHKD. Dec	3-12-69
ENG Decker	3-12-69
Q.C.	
PROD	

NATIONAL LEAD COMPANY
NUCLEAR DIVISION
WILMINGTON PLANT

SPENT FUEL CONTAINER & SHIPPING CASK

CASK ASSEMBLY - PICTORIAL SECTION

CODE IDENT	PROD Nº	SIZE	DRAWING Nº	REV
29932	W-5115	D	40065D	1
WT	SCALE		SHEET 1 OF 1	

POOR ORIGINAL

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SHEET 1 OF 1

H

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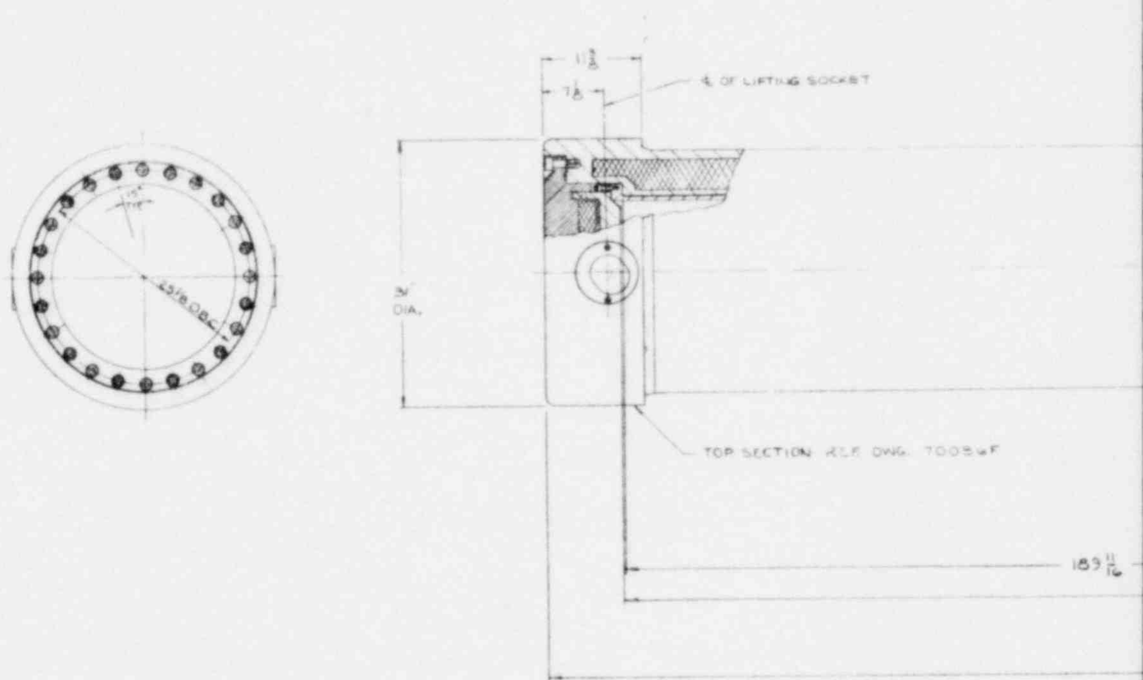

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REVISIONS

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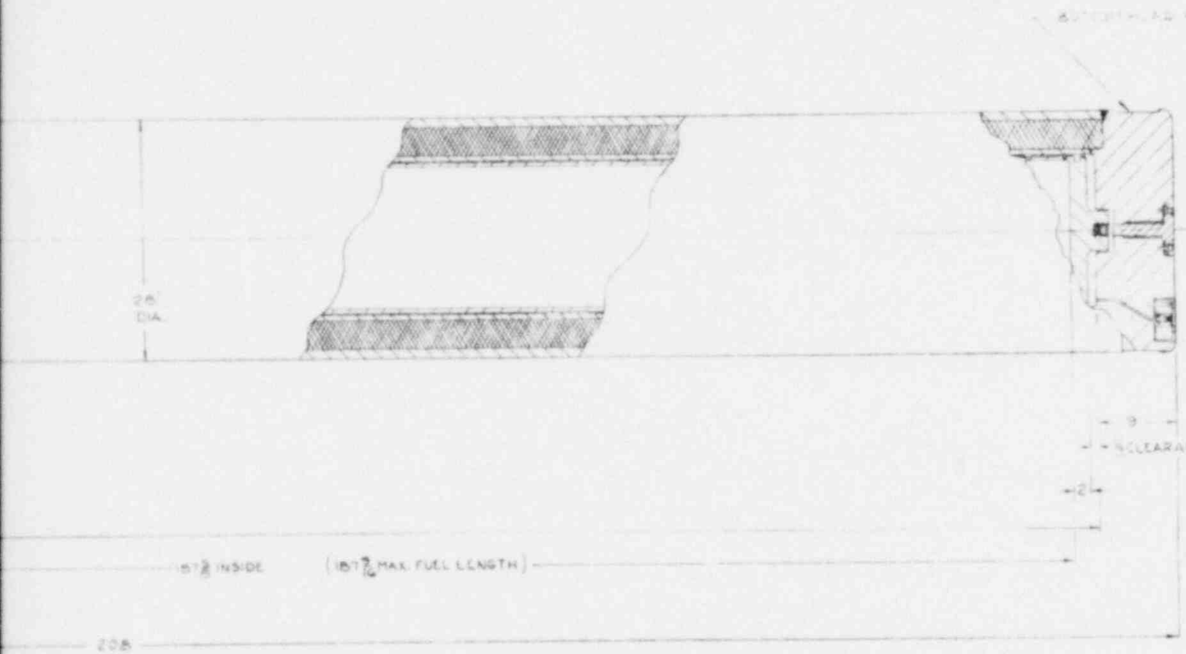


Gulf General Atomic

VENDOR DRAWING APPROVAL

ITEM	QTY	NAME	MAT'L	SPEC	DWG Nº	DESCRIPTION

LIST OF MATERIAL

	DATE
DRAWN L.M.	
CHKD	
ENG	
Q.C.	
PROD	

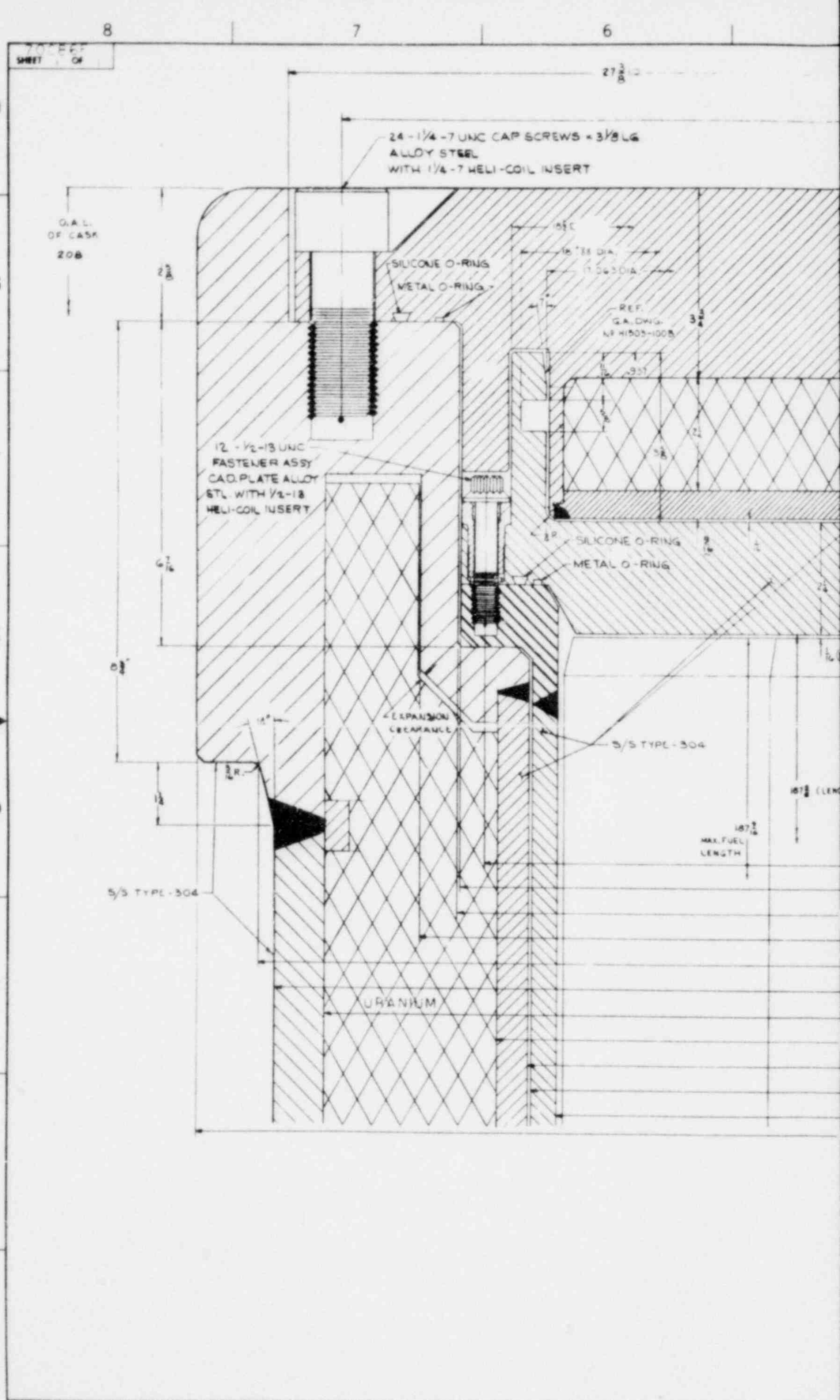
NATIONAL LEAD COMPANY
NUCLEAR DIVISION
WILMINGTON PLANT

SPENT FUEL CONTAINER & SHIPPING CASK ASSEMBLY

CODE IDENT	PROD Nº	SIZE	DRAWING Nº	REV
29932	W 5115	F	70085F	5

WT | SCALE 1/8 | SHEET 1 OF 1

POOR ORIGINAL



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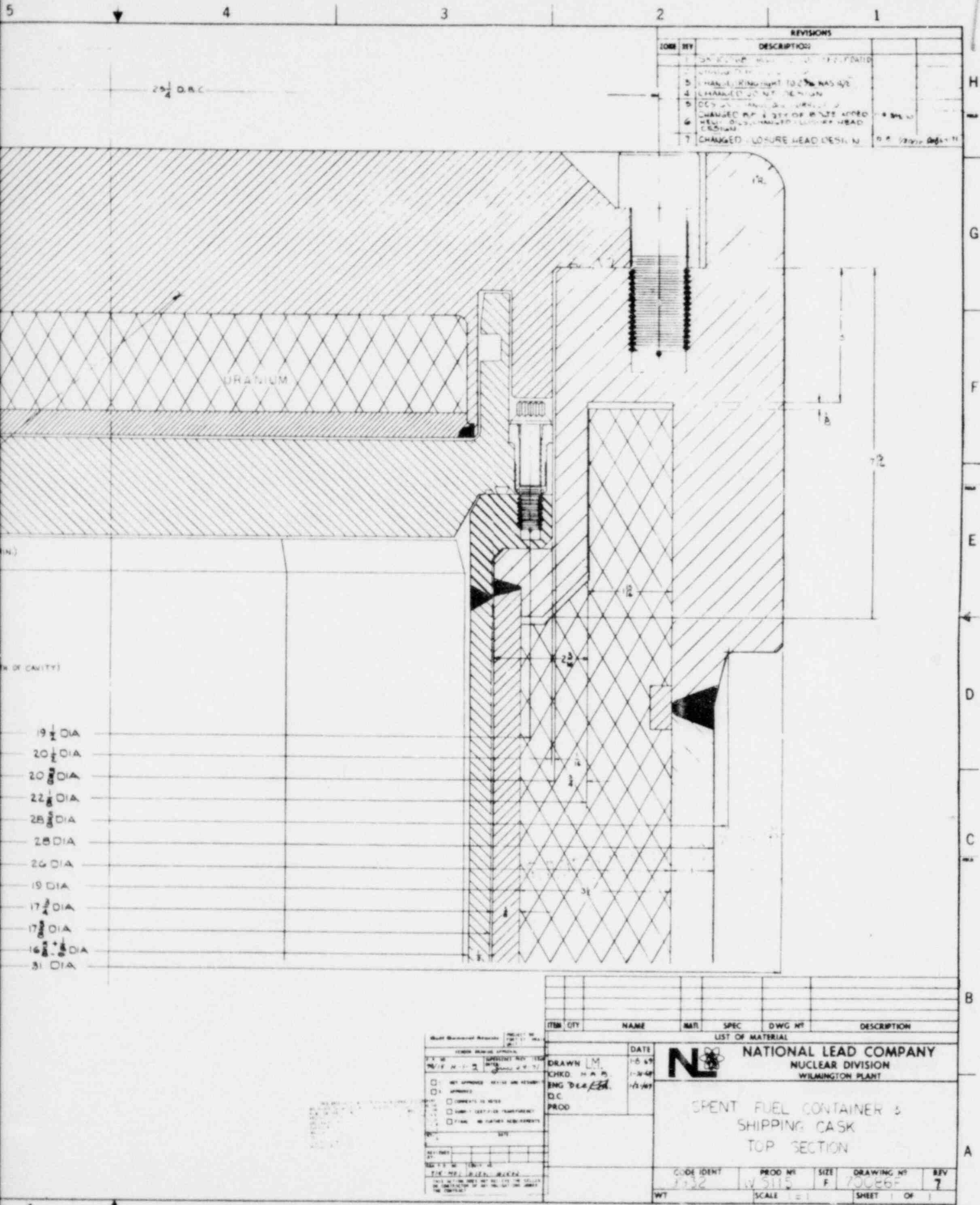

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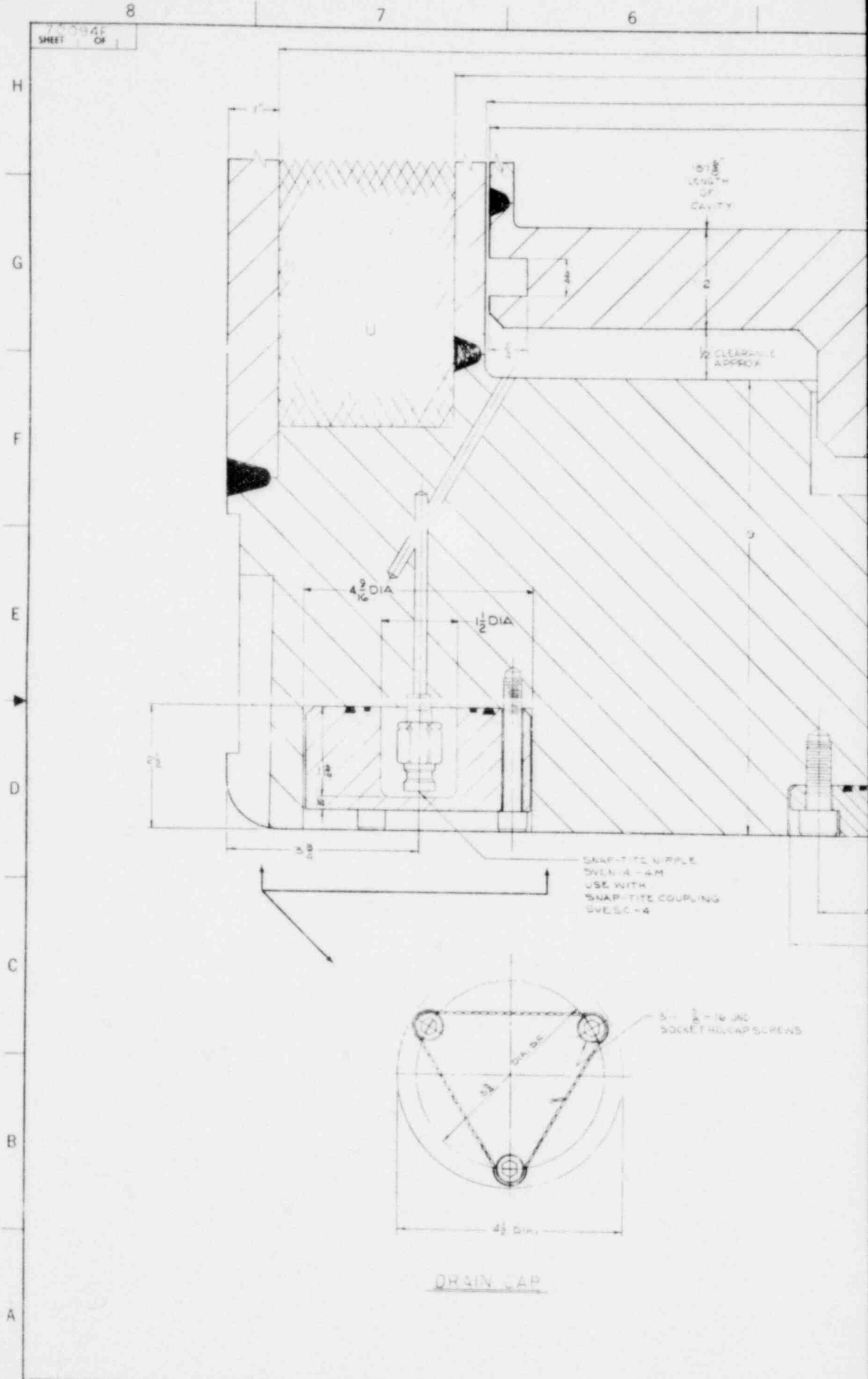

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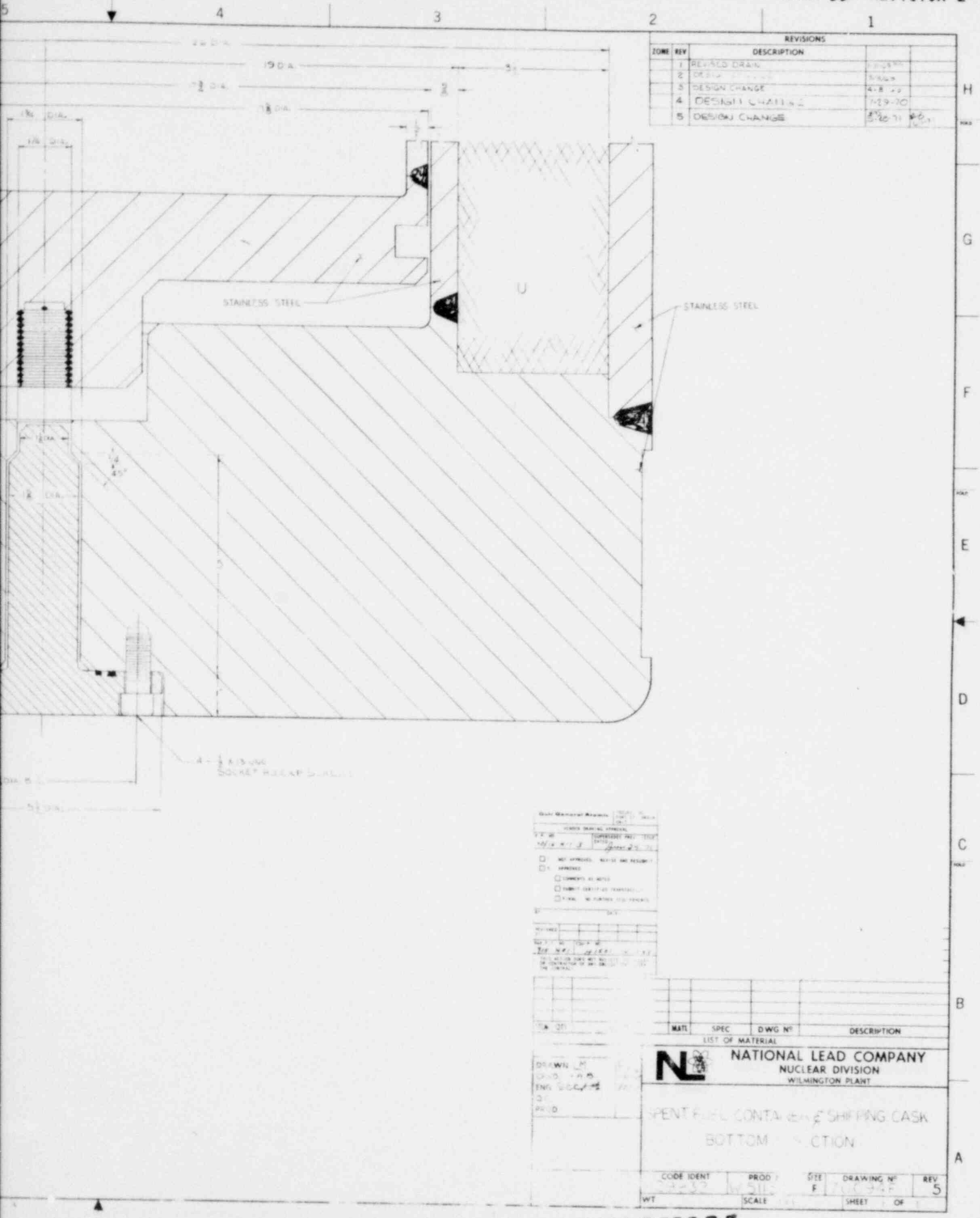

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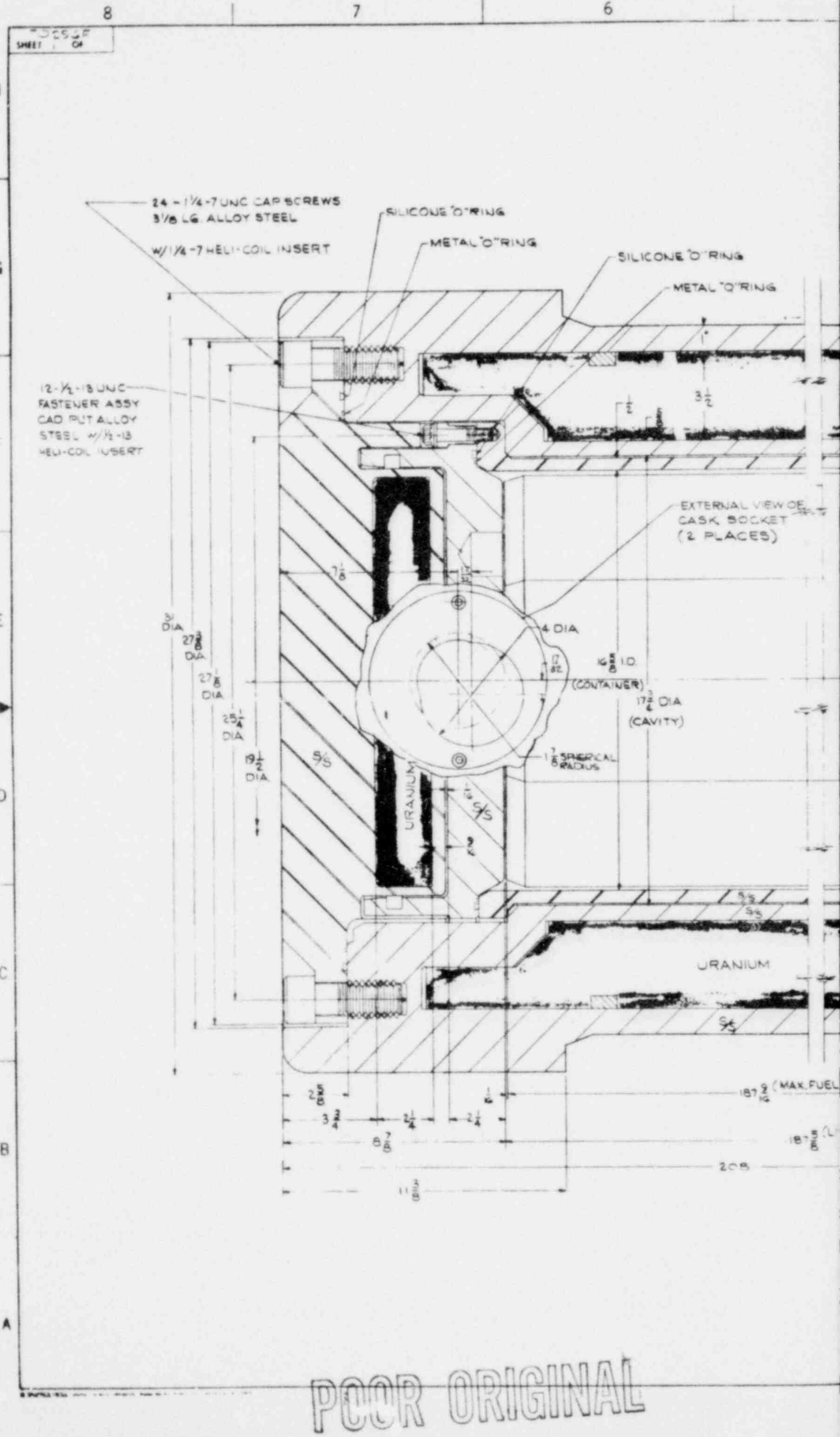

24 - 1¼-7 UNC CAP SCREWS
3⅛ LG. ALLOY STEEL
W/1¼-7 HELI-COIL INSERT
SILICONE "O" RING
METAL "O" RING
SILICONE "O" RING
METAL "O" RING
12-½-13 UNC FASTENER ASSY
CAD PLT ALLOY STEEL W/½-13
HELI-COIL INSERT
EXTERNAL VIEW OF CASK SOCKET
(2 PLACES)
4 DIA
16⅝ I.D.
(CONTAINER)
17¾ DIA
(CAVITY)
1⅞ SPHERICAL RADIUS
31 DIA
27⅜ DIA
27⅛ DIA
25¼ DIA
19½ DIA
S/S
URANIUM
URANIUM
2⅝
3¾
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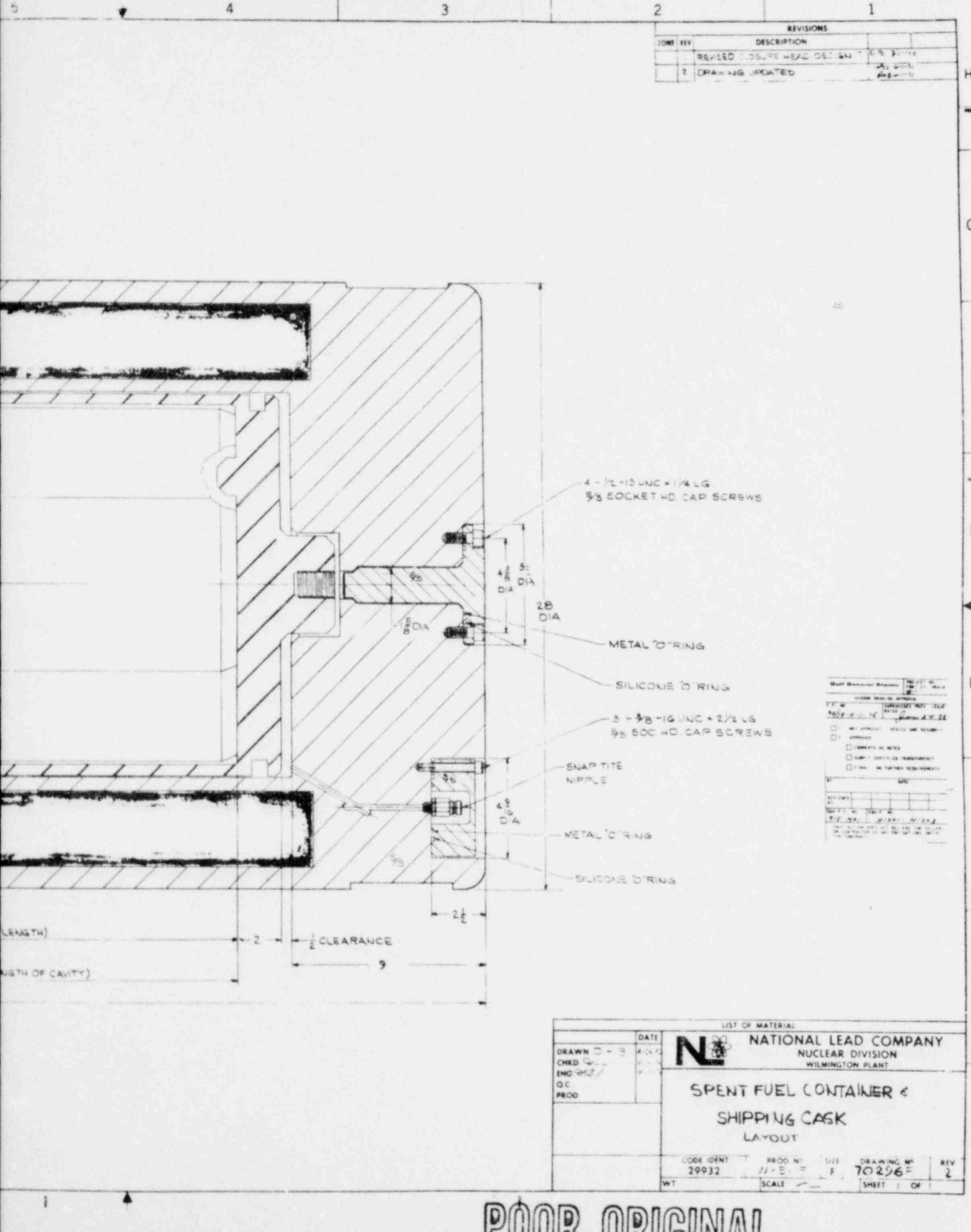
2¼
8⅞
11⅜
208
187 9/16 (MAX. FUEL



POOR ORIGINAL

SECTION III

PROPERTIES OF MATERIALS

PROPERTIES OF MATERIALS

Properties of materials used in the fabrication and assembly of the spent fuel containers and lid and the fuel shipping cask and cover are as follows:

A. "As-Cast" Unalloyed Depleted Uranium

Density

18.9 grams/cc or .683 lb./in^3

Mechanical Properties

Ultimate Tensile Strength	60,000 to 100,000 psi
Yield Strength	25,000 to 45,000 psi
Reduction in Area	10% to 40%
Elongation	8% to 15%
Modulus of Elasticity	Approximately 24 $(10)^6$ psi
Poisson's Ratio	Approximately 0.21
Shear Modulus	Approximately 12 $(10)^6$ psi
Hardness	Rockwell.B 65 to 90

Melting Point 2070° F

Thermal Expansion 6.5 $(10)^{-6}$ in/in/° F

B. Stainless Steel Pipe, Type 304 per ASTM Spec. A-351, Grade CF-8

Physical Properties

Density	0.287 lb/in.^{3}
Specific Gravity	7.94 grams/cc
Melting Range	2550° to 2650°F
Modulus of Elasticity	28×10^6 psi
Specific Heat (32° to 212°F)	0.12 Btu/lb/°F

Thermal Conductivity

At 200°F	9.4 Btu/hr/ft^2/°F/ft
At 1000°F	12.5 Btu/hr/ft^2/°F/ft

Mean Coefficient of Thermal Expansion

32° to 212°F	9.6 in./in./°F $\times 10^{-6}$
32° to 600°F	9.9 in./in./°F $\times 10^{-6}$
32° to 1000°F	10.2 in./in./°F $\times 10^{-6}$

Mechanical Properties (at 72°F)

Ultimate Tensile Strength	70,000 psi
Yield Strength	30,000 psi
Elongation	35%
Reduction of Area (approx)	60%
Hardness	R_B 88

C. Stainless Steel Forgings, Type 304 per ASTM Spec. A-182, Grade F-304

Physical Properties

Same as B, above.

Mechanical Properties

Ultimate Tensile Strength	70,000 psi
Yield Strength	30,000 psi
Elongation	40%
Reduction of Area	50%
Hardness (approx)	R_B 88

D. Uranium Welds

All uranium welding will be accomplished using single V-butt joints and inert dc tungsten arc welding. The inert gas used for shielding and trailing shields shall be of welding grade argon. The filler and base metals shall be depleted uranium. Prior to use, filler metal shall be free of oxides and other contamination.

Base metal shall be cleaned of oxides, oil, grease, and other weld contaminants. Each weld bead will be cleaned prior to the deposition of the next weld bead. Final layers of each weld will be liquid-penetrant inspected.

E. Stainless Steel Welds (Refs. 1 and 2)

All stainless steel welds will be in accordance with the "Rules for Construction of Nuclear Vessels" (Ref. 1). All of the welding procedures and welders will be qualified in accordance with "Welding Qualifications" (Ref. 2).

All stainless steel welds will be dye-penetrant inspected per MIL-STD 271D. Cracks, porosity, lack of fusion, undercutting and overlapping will not be acceptable.

F. Seals (Refs. 3 and 4)

1. All metal "O" rings shall be self-energized for use in bolted flange assemblies. These "O" rings will be made of silver plated Inconel X, tubing. Service temperature is -320° to +1200°F. The following metal "O" rings or equivalent have been selected for sealing the Fort St. Vrain spent fuel container and fuel shipping cask:

   a) Cask Closure Head Seal - United Aircraft Products, Inc. Cat. No. U-6420-22000-SEA; OD = 22.01"; ID - 21.76"; Tube Dia. = 0.125".

   b) Spent Fuel Container Lid, Seal - United Aircraft Products, Inc. Cat. No. U-6420-17430-SEA; OD = 17.44"; ID = 17.19"; Tube Dia. = 0.125".

   c) Cask Base Closure Pin Seal - United Aircraft Products Inc. Cat. No. U-6420-02813-SEA; OD = 2.81"; ID = 2.56"; Tube Dia. = 0.125".

   d) Cask Helium Connection Cap Seal - United Aircraft Products, Inc. Cat. No. U-6420-02630-SEA; OD = 2.62"; ID = 2.38"; Tube Dia. = 0.125".

2. All elastomer "O" rings shall be molded per AMS Specification 3304 of silicone rubber. Service temperature for this material is -100° to +500°F. The following silicone rubber "O" rings or equivalent have been selected for sealing the Fort St. Vrain spent fuel container and fule shipping cask:

   a) Cask Closure Head Seal - Parco No. PRP-568-392; OD = 23.375"; ID = 23.00"; Dia. = 0.187".

b) Spent Fuel Container Lid Seal - Parco No. PRP-568-388; OD = 17.375"; ID = 17.000"; Dia. = 0.187".

c) Cask Base Closure Pin Seal - Parco No. PRP-568-236; OD = 3.500"; ID = 3.25"; Dia. = 0.125".

d) Cask Helium Connection Cap Seal - Parco No. PRP-568-233; OD = 3.125"; ID = 2.875"; Dia. = 0.125".

3. All "O" rings are installed in the removable parts of the spent fuel container and fuel shipping cask. This facilitates inspection and replacement and decontamination in the Fort St. Vrain hot service facility.

G. Fasteners (Refs. 5,6,7)

1. The bolts used in the assembly of the spent fuel container are alloy steel per AMS 6322 with cadmium plating per QQ-P-416, Type 11, Class 3. These bolts are heat tested for 180KSI min, ultimate tensile strength with a hardness of R_C40-44. The 1/2-inch size used in fastening the spent fuel container lid to the container has a minimum axial tensile strength of 26,700 lbs. (Ref. 5).

2. The bolts of the spent fuel container lid are threaded into "screw-lock" inserts made of Type 18-8 stainless steel (per AMS-7245B) wire having an ultimate tensile strength of approximately 200,000 psi. These "screw-lock" inserts meet military specification for locking torque and vibration. The internal thread conforms to thread form standards issued by the Department of Commerce (Ref. 6).

3. The bolts used in the assembly of the cover head to the fuel shipping cask shall be cadmium plated per QQ-P-416 alloy steel per FF-S-86C with the following physical properties for the 1-1/4"-7 UNC size (Ref. 7):

Tensile Strength, min.	160,000 psi.
Yield Strength, min.	130,000 psi.
Heat Treatment	R_C 36-43

4. The bolts of the cover head are threaded into inserts in the shipping cask head. These inserts are to prolong the life of the threads (since they can be replaced when worn) and locking of the bolts shall be by lock-wiring per MIL-STD-763A, dated 21 July, 1967.

H. Iron - Uranium Eutectic Prevention (Ref. 8)

Investigations have shown that uranium combines with stainless steel by solid state diffusion at temperatures above 1000°F. The iron-uranium eutectic melts at 1337°F so that if the two materials are in intimate contact at this temperature a molten alloy will be formed (Ref. 8).

Other investigations have shown that uranium in contact with stainless steel will penetrate the stainless steel by solid state diffusion in 24 hours at 1400°F. At 1355°F there was no attack on the stainless steel.

Recent tests of stainless steel - uranium - stainless steel assemblies wherein the surfaces of the stainless steel next to the uranium were spray coated with a 0.005-inch-thick coating of copper showed this coating to be an effective barrier to diffusion between the stainless steel and uranium at temperatures up to 1750°F (Ref. 8). All surfaces of stainless steel in contact with the depleted uranium shielding will be coated with 0.005-inch-thick copper coating for the Fort St. Vrain Fuel Shipping Casks.

REFERENCES

1. ASME Boiler and Pressure Vessel Code Section III - Nuclear Vessels, The American Society of Mechanical Engineers, United Engineering Center, N.Y., 1965.

2. ASME Boiler and Pressure Vessel Code Section IX - Welding Qualifications, The American Society of Mechanical Engineers, United Engineering Center, N.Y., 1965.

3. "Metallic O-Rings," United Aircraft Products, Inc., Bulletin No. 596191B, Dayton, Ohio, July 15, 1959.

4. "O-Ring Design Handbook," Plastic and Rubber Products Company, Ontario, California.

5. "Bolt, External Wrenching, Self-Retained by Washer," Standard Pressed Steel Company, Jenkintown, Pa., Part Number and Specification 69241, Sheets 1, 2, and 3, March 6, 1968.

6. "Heli-Coil Screw-lock Inserts," Heli-Coil Corporation, Danbury, Connecticut, Bulletin 900.

7. Federal Specification, Screw, Cap. Socket Head, October 17, 1967.

8. Clifford. C. B., "Design and Fabrication of a Prototype Laminated Uranium Metal Shipping Cask for Large Shipment of Cobalt-60," USAEC Research and Development Report No. KY-521, Union Carbide Corporation, Nuclear Division, Paducah, Ky., April 3, 1967.

SECTION IV

WEIGHT CALCULATIONS

SECTION IV

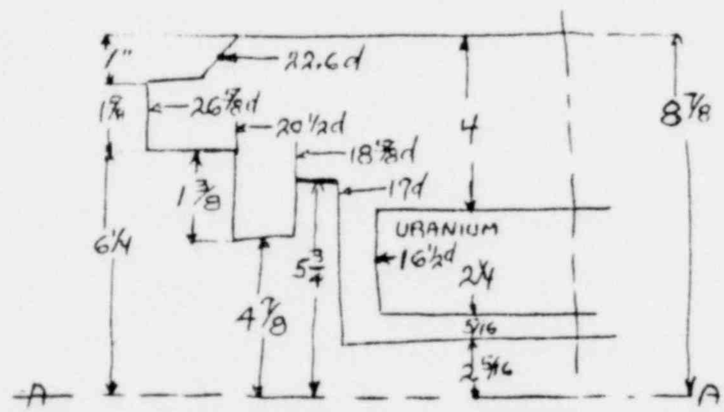
NATIONAL LEAD CO. NUCLEAR DIVISION
WILMINGTON PLANT

WEIGHT CALCULATIONS

WEIGHTS - SPENT FUEL CONTAINER AND SHIPPING CASK.

REFERENCE DWGS. 70085F REV. 3
70086F REV. 5
70094F REV. 3

A. CASK CLOSURE HEAD - WTS.



		V	y	Vy
22.6d	$400\ in^2 \times 1$	$= 400\ in^3$	$\times -8\frac{3}{8} =$	-3350
$26\frac{5}{8}d$	$560 \times 1\frac{5}{8}$	$= 910$	$\times -7\frac{1}{16} =$	-6430
$20\frac{1}{2}d$	$330 \times 1\frac{3}{8}$	$= 455$	$\times -5\frac{9}{16}$	-2530
17d	$227 \times 3\frac{7}{16}$	$= 780$	$\times -4\frac{1}{32} =$	-3140
		$+ 2545\ in^3$		$-15,450.\ in^4$
$-18\frac{7}{8}d$	$-273 \times \frac{7}{8}$	$= -239$	$\times 5\frac{5}{16}$	$+1330$
$-16\frac{1}{2}d$	$-214 \times 2\frac{1}{4}$	$= -482$	$\times 3\frac{3}{4} =$	$+1809$
		-721		$+3139$
		$+1824\ in^3$		$-12,311\ in^4$
		.29		.29
	(S.S) wt =	530 lbs	M =	-3580 in lbs.
Uranium $16\frac{1}{2}d$	$+214 \times 2\frac{1}{4}$	$= +482$	$\times -3\frac{3}{4} =$	-1809
		.683		.683
		330 lbs	M =	-1235 in lbs
	TOTAL WT. HEAD =	860 LBS	M =	-4815 in lbs.

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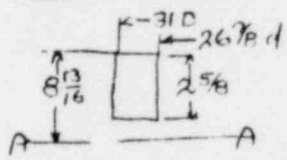
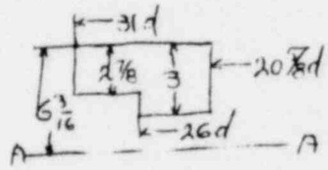
SECTION IV - WEIGHT CALCULATIONS

NATIONAL LEAD CO. NUCLEAR DIVISION
WILMINGTON PLANT

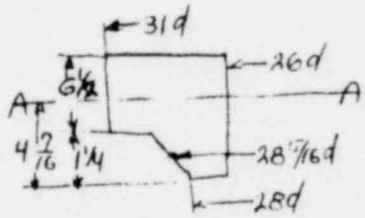
B. CASK - WTS



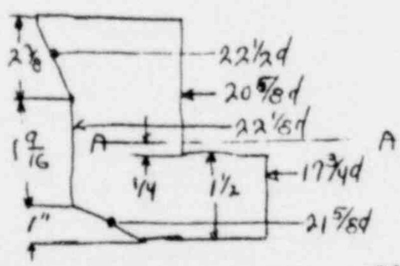
31d +752 in²
26 7/8 d −568
184 × 2 5/8 = 483 in³ × −7 1/2" = −3622
.29 .29
wt = 140 lbs M = −1050 in lbs



31d +752 in² × 2 7/8 = 2160 in³ × −4 3/4" = −10,250.
26d +530 × 1/8 = 66 × −3 1/4 = − 216.
+2226 −10,466
−20 7/8 d −335 × 3 = −1005 × −4 11/16 = + 4,850
+1221 in³ − 5616.
.29 .29
wt = 354. lbs M = −1628 in lbs



31d 752 in² × 6.5 = 4880 × −1/16 = −305.
28 5/16 d 630 × 1 1/4 = 787 × +3 13/16 = + 3000.
+5667 +2695
−26d −530 × 7 3/4 = −4100 × + 9/16 = −2300
+1567 in³ + 395.
.29 .29
wt = 455 lbs M = +114 in lbs



22 1/2 d 397 in² × 2 7/8 = 942 in³ × −2 1/4 = −2120
22 1/8 d 385 × 1 9/16 = 601 × − 9/32 = − 169
21 5/8 d 367 × 1 = 367 × + 1 1/4 = + 458
+1910 −1831
−20 5/8 d −335 × 3 7/16 = −1154 × −1 15/32 = +1670
−17 3/4 d −247 × 1 1/2 = − 370 × +1 = − 370
−1524 +1300
+ 386 − 531
.29 .29
wt = 112 lbs W = −154 in lbs

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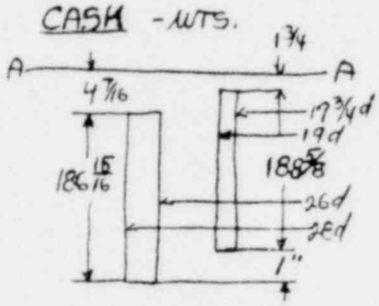
POOR ORIGINAL

SECTION IV - WEIGHT CALCULATIONS

NATIONAL LEAD CO. NUCLEAR DIVISION
WILMINGTON PLANT

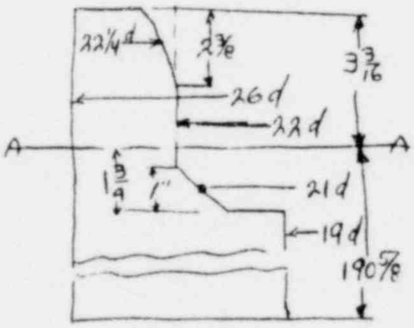
CASK - WTS.



OUTER INNER

28d 613 in²
-26d -530
83 in² × 186 15/16 = 15,500 in³ × +97 29/32 = +1,511,000.
.29 .29
(outer) wt 4500 lbs M = +439,000

19d 283.5 in²
-17 3/4 d -247
36.5 in² × 188 5/8 = 6870 in³ × +96.06 = +660,000
.29 .29
(inner) wt = 1990 lbs M = 191,200 "lbs

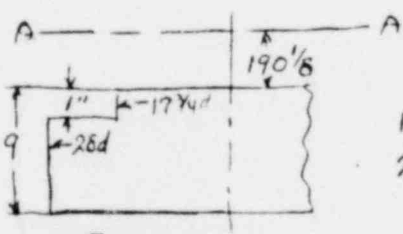


URANIUM SHIELDING

26d 530 in² × 193 13/16 = 102,750 in³ × +93 23/32 = +9,630,000.

-19d -283.5 × 188 5/8 = -53,500 × +96 3/16 = -5,150,000
-21d -346 × 1 = -346. × +1 1/4 = -432
-22d -380 × 1 9/16 = -594 × -1/32 = +19
-22 1/4 d -389 × 2 3/8 = -925 × -2 = +1850
-55,365 -5,148,563
+47,385 +4,481,437
.683 .683

URANIUM wt 32,400 lbs M = +3,060,000 "#



BOTTOM

17 3/4 d 247 in² × 1 = 247 in³ × +190 5/8 = 47,100
28 d 613 × 8 4904 × +195 1/8 = +960,000
+5151 +1,007,100

wt = .29 / 1494 lbs M = .29 / 292,060 in lbs

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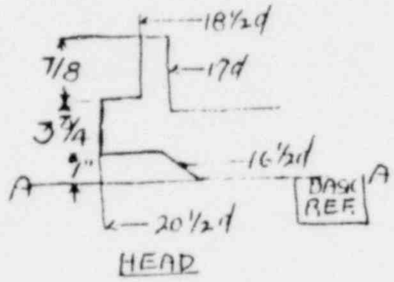
POOR ORIGINAL

SECTION IV - WEIGHT CALCULATIONS

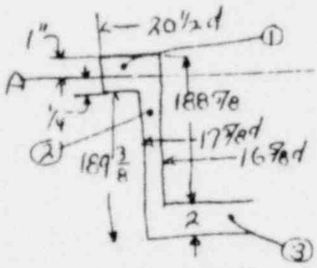
NATIONAL LEAD CO. NUCLEAR DIVISION
WILMINGTON PLANT

C. CONTAINER - WTS



HEAD

18½d 268 in² × ⅞ = 234 in³ × −5 3/16 = −1212
20½d 330 × 3¾ = 1238 × −2⅞ = −3560
16½d 213 × 1 = 213 × −½ = −107
+1685 in³ −4879
−17d −227 × 3⅞ = −766 × −3 15/16 = +3010
+919 −1869
.29 .29
wt = 266 lbs M = −542

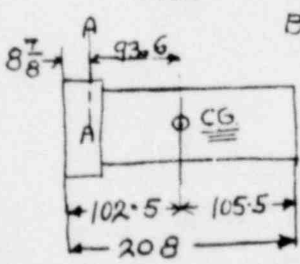


CONTAINER

wt_1 = 330 − 217 = 113 × 1¼ × .29 = 41. lbs × −⅞ = −16.
wt_2 = 244 − 217 = 27 × 187⅞ × .29 = 1470. × +93 15/16 = +138,000.
wt_3 = 244 × 2 × .29 = 142. × 188⅝ = +26,800
wt = 1653 lbs M = +164,784 in lbs

CONTENTS OF CONTAINER 300# × 6 = 1800 lbs × 93 27/32 = M = +169,000 in lbs

D. WEIGHT SUMMATION



		WT.	MOMENT
CASK	HEAD	860 LBS	−4815.
	BODY	140	−1050.
		354	−1628.
		455	+ 114.
		112	− 154.
		4500	+ 437,000
		1990	+ 191,200
		32400	+ 3,060,000
		1494	+ 292,060
CASK-TOTAL EMPTY		42,305	+ 3,474,727
	CONTAINER	266	− 542
		1653	+ 164,784
	CONTENTS	1800	+ 169,000
CASK-TOTAL-LOADED		46,024 lbs × [93.6] =	+ 4,307,969 in lbs

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POOR ORIGINAL

SECTION V

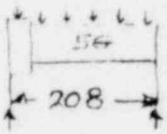
STRESS ANALYSIS

SECTION V

NATIONAL LEAD CO. NUCLEAR DIVISION
WILMINGTON PLANT

STRESS ANALYSIS

A. SIMPLE BEAM LOADING   5g - UNIFORMLY DISTRIBUTED



WT = 46024 lbs

W = 5(46024) = 230,120 lbs

$M_{max.}$ at center $= \frac{Wl}{8} = \frac{230120}{8}(208) = 6,000,000$ in lbs

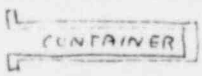
assume only outer shell in stress.

O.D. = 28"   I.D. = 26" $I = \frac{\pi}{4}(R^4 - r^4) = \frac{\pi}{4}(14^4 - 13^4) = 7750$ in^4

$Z = \frac{7750}{14} = 554$ in^3

$S_b = \frac{M}{Z} = \frac{6,000,000}{554} = 10,800$ psi   OK < 30,000 yp

B. EXTERNAL PRESSURE   25 psig   Roark XVI case 30 (Stability)



17⅝ OD - 16⅞ I.D - ½ wall - 190⅞ O.A. LGTH.

$p' = \frac{1}{4}\frac{E}{1-\nu^2}\frac{t^3}{r^3} = \frac{1}{4}\frac{30(10^6)}{(1-.09)}\frac{(1/2)^3}{(8.62)^3} = 1640$ psi   OK

C. INTERNAL PRESSURE   50 psig   Roark XIII - case 1

(containers as above)

hoop stress $S_2 = \frac{pR}{t} = \frac{50(8.62)}{1/2} = 862$ psi   OK < 30,000 yp

meridional stress $S_1 = \frac{pR}{2t} = \frac{862}{2} = 431$ psi

Radial displacement $= \frac{R}{E}(S_2 - \nu S_1)$

$= \frac{8.62}{30(10^6)}[862 - .3(431)] = \frac{6300}{30(10^6)} = .000,21$"   OK

NATIONAL LEAD CO. NUCLEAR DIVISION
WILMINGTON PLANT

D. DETERMINATION OF DYNAMIC DESIGN STRESSES.

The dynamics of the 30 foot free fall requires that a value be found for the maximum stress at impact in order to use structural analysis methods based on statics.

Literature on the subject of dynamic stresses is largely theoretical and seldom of engineering application value. The complexity can be reduced by limiting the problem to (1) compressive stresses on flat impact (2) at 44 ft/sec and (3) to steel, aluminum and uranium materials.

The most promising engineering formula is that for dynamic compression of rigid-plastic cylinders and is the one used herein. It is well authenticated by

(1) Goldsmith: Impact - eq 5.97 page 191
(2) Cristescu: Dynamic Plasticity - eq 7.8 page 55

$$\rho v_1^2 = \varepsilon_1 (P_1 - P_y) \qquad \text{Eq. D-C-1 (from authors)}$$

$\rho = \dfrac{\text{lbs/in}^3}{386}$ $\qquad v_1^2 = [44 \text{ ft/sec} \times 12]^2 = 278,784 \text{ (in/sec)}^2$

P_1 = initial, and maximum, stress corresponding to moment of impact.
ε_1 = max. strain corresponding to P_1
P_y = static compressive yield point (by .002 in/in strain method)

Values of ρ and ρv_1^2

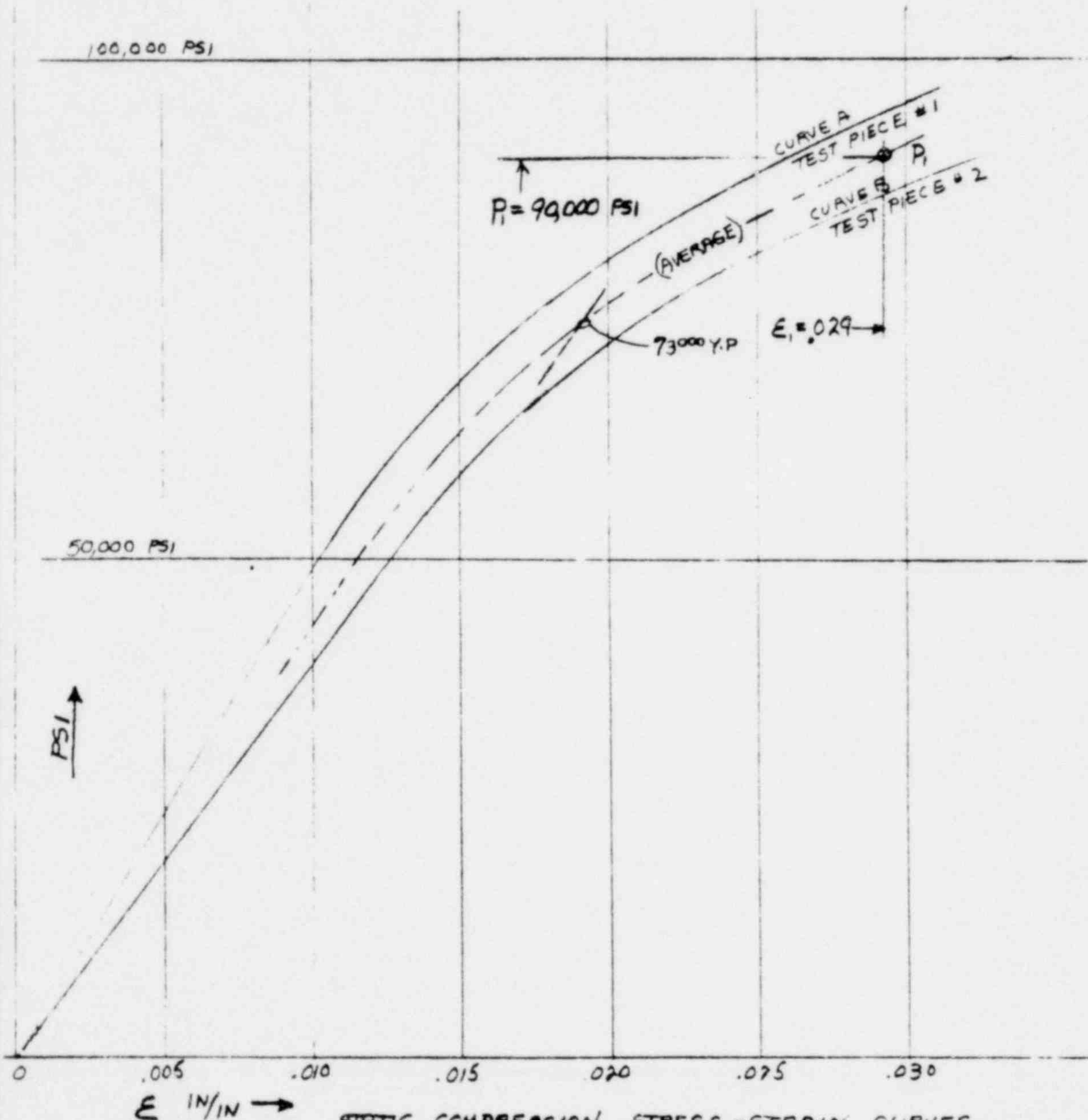
		$\rho v_1^2 =$
Aluminum	$\rho = .097/386 = .000,251$	70.
Steel (incl S.S)	$\rho = .290/386 = .000,7512$	209.
Copper	$\rho = .332/386 = .000,86$	240.
Lead	$\rho = .41/386 = .001,062$	296.
Uranium	$\rho = .683/386 = .001,769$	493.

Typical Dynamic Properties

Material	Static curve	P_y .002 offset	P_1	ε_1	$P_1 - P_y$	$\varepsilon_1 (P_1 - P_y)$	$\dfrac{P_1 - P_y}{P_y}$
URANIUM-CAST	NLC COMP.	73,000	90,000	.029	17,000	493. ✓	.233
302 S Steel	Goldsmith Fig 250 $P_1 = 40,000 + \dfrac{825,000}{e}$ $\varepsilon_1 = \dfrac{P_1 - 40,000}{828,000}$	41,350 $F_{cy} = 35000$ 304 S.S	53,800	.01675	12,450	208 ✓	.301
.24 Carbon Stl	Goldsmith Fig 273	35000	45000	.020	10,000	200 ✓	.285
Ni-Cr Steel	Goldsmith Fig 118	220,000	228,000	.024	8,000	192 ✓	.036

NATIONAL LEAD CO. NUCLEAR DIVISION

WILMINGTON PLANT



STATIC COMPRESSION STRESS-STRAIN CURVES
UNALLOYED DEPLETED URANIUM
"AS CAST" FOR SHIELDING - N.L.C. FEB. 1970
BACKUS

DATE 3/11/70 PAGE 5-3 OF

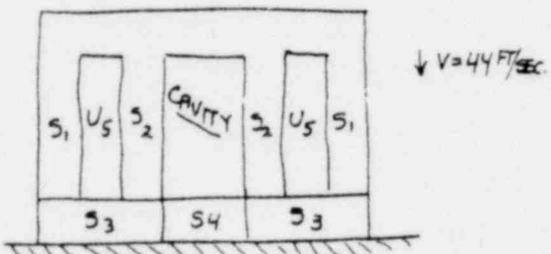
POOR ORIGINAL

NATIONAL LEAD CO. NUCLEAR DIVISION
WILMINGTON PLANT

E. FREE DROP - 30 FEET - FLAT END IMPACT.

The dynamic analysis required for the cask and its parts, in the flat drop attitude, can be based on the concept of simple cylinders which behave as solid rods impacting squarely upon a rigid surface at the velocity of the specified free fall distance of 30 feet. This velocity is 44 ft/second.

Values for stress, strain, K.E. and g loadings can be derived for the individual masses and for the cask as a whole.



MODEL FOR ANALYSIS

S_1 = Outer S.S. shell
S_2 = Inner S.S shell
U_5 = Uranium Cylinder
These are considered separate cylinders
S_1 and S_2 weights include parts of the head weights

S_3 = S.S Plate - acting as striking plate under S_1, U_5 and S_2
S_4 = part of S.S Plate stressed by impact from Container etc

	AREA	WEIGHT
S_1	86.3 in²	5125 LBS (incl. 625# head)
S_2	36.8 in²	2260 LBS (incl. 270# head)
U_5	250.9 in²	32,400 LBS.
	374. in²	39,785 LBS
S_3	378.5 in²	990 LBS.
S_4	248.5 in²	5249 LBS (incl. 266 / 1653 } container / 1800 contents)
	TOTAL	46,024 LBS.

POOR ORIGINAL

NATIONAL LEAD CO. NUCLEAR DIVISION
WILMINGTON PLANT

F. FLAT IMPACT STRESSES, STRAINS, K.E., AND G'S

Initial, and maximum, stresses developed by striking a rigid mass are calculated for Uranium and steel, derived from

$\rho v_1^2 = \varepsilon(P_1 - P_y)$ as previously given.

P_1 Uranium = 90,000 psi

P_1 ~~stainless~~ steel = 53,800 psi

For cylinders S_1, U_5 and S_2, the assumption of striking a rigid mass is conservative. Actually, they strike against an intermediate mass S_3, with some consequent reduction in stress.

S_3, on its under-surface, does strike a rigid mass, and counts S_1 U_5 and S_2 only as added load, considered as an equivalent weight of steel, added to the height of S_3 as a cylinder.

Stress differentials at the interfaces with S_3 are considered localized and quickly finding equilibrium, producing a local increase in stress intensity in S_3 to match a reduced stress in U_5

Total force developed at impacting face $F = P_1(\text{Area})$

For S_1	$F_1 = 86.3\ \text{in}^2$	× 53,800 psi	= 4,640,000 LBS
S_2	$F_2 = 36.8$	× 53,800	= 1,975,000
U_5	$F_5 = 250.9$	× 90,000	= 22,581,000

G's Developed $G = F/WT.$

S_1	4,640,000/5175 =	905 G
S_2	1,975,000/2260 =	875 G
U_5	22,581,000/32400 =	697 G
S_3	29,196,000/39,785 =	735 G For bottom of cask and closure head.

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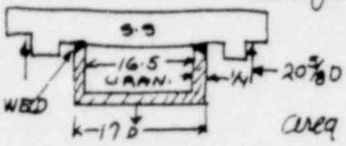
G. FLAT BOTTOM DROP.

Strength of Weld - Closure Head Flange.



Weights. Uranium block 330 lbs

S.S enclosure $\underline{30}$

360 lbs

area of weld = area of shell

$= \pi/4\,(17^2 - 16.5^2) = 12.5\ in^2$

Force in weld $= (360\ lbs) \times (735\ G) = 264{,}000$ LBS

Stress in weld $= 264{,}000/12.5 = \underline{21{,}100\ PSI}$

The full penetration weld has substantially the strength of the parent metal assume 80% of 30,000 Y.P of Stainless Steel = 24,000 PSI

$M.S = 24{,}000/21{,}100 - 1 = 11.4\%$ conservative (ultimate allowable)

Strength of Plate in bending and shear

Roark X

$r_0 = 10.75''$

$a = 12.56''$

Weight = 860 lbs - S.S. Plate and Uranium

Load $= (860) \times (735\ G) = 632{,}000$ LBS.

Stress at center of head Case 2 $t = 4''$ max.

$$\text{Max. } S_r = S_t = \frac{-3W}{2\pi m t^2}\left[m + (m+1)\log_e \frac{a}{r_0} - (m-1)\frac{r_0^2}{4a^2}\right]$$

$$= \frac{-3\,(632{,}000)}{628\,(10/3)\,4^2}\left[\frac{10}{3} + \frac{13}{3}\log_e\frac{12.56}{10.75} - \frac{7}{3}\,\frac{10.75^2}{4\times 12.56^2}\right]$$

$$= 5650\left[\frac{10}{3} + \frac{13}{3}(.203) - 3.9\right] = 5650\,(.31) = \underline{1750\ PSI}$$

OK 30,000 Y.P

Stress at edge Case 7 $t = 2\tfrac{5}{8}$ IN.

$$\text{Max. } S_r = \frac{3W}{2\pi t^2}\left(1 - \frac{r_0^2}{2a^2}\right) = \frac{3\,(632{,}000)}{628\,(2.625)^2}\left(1 - \frac{10.75^2}{2(12.56)^2}\right)$$

$$= 43{,}700\,(1 - .334) = \underline{29{,}200\ PSI}$$

OK 30,000 Y.P

Shear at edge (20.625 DIA) $t = 2\tfrac{5}{8}$ IN.

area in shear $= 2\pi r \cdot t = \pi(20.625)\,2.625$

$= 170.5\ in^2$

$$S_s = \frac{632{,}000}{170.5} = \underline{3700.\ PSI}$$

$F_{su} = .60\,(70{,}000\ PSI)$

$= 42{,}000$

$$M.S. = \frac{42000}{3700} - 1 = 10.3$$

Stresses are dynamic; strengths are static. Therefore an extra M.S. is secured.

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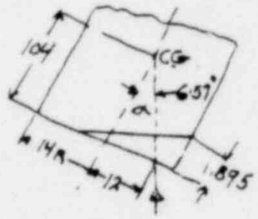
H. BOTTOM DROP IMPACT ON CORNER

In corner drop calculations the assumption is made that the ungula developed at the impact position represents the total compressive effects at a local point, rather than the actual non-linear distribution of strain thruout the whole impacting mass, both elastic and plastic. It allows calculation of max. stress values, etc., without regard for actual distribution of stress and strain at non-critical values.



The calculated ungula is a greater volume than will actually result from impact, for this reason.

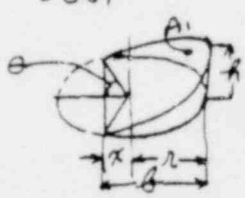
$\alpha = \text{arc tan } \frac{13}{104} = .1152$

$= 6.57°$

Vol. ungula $= \frac{h}{b} r^3 \left[\sin\theta - \frac{\sin^3\theta}{3} - \theta\cos\theta\right]$

$h/b = .1152 \qquad r^3 = 14^3 = 2744$

$\therefore \text{Vol} = (.1152)\, 2744\, [\;] = 317.\,[\;]$



assume center of impact 2" from edge, at R = 12"

Total weight impacting on S.S bottom plate = 46024 lbs.

KE = (4602' ' 60 in) = 16,600,000 IN. LBS

Mean Fl stress for S.S. approximately

$P = 53,800$ PSI

$KE/P = \frac{16,600,000}{53,800} = 308 \text{ in}^3$ = reqd. ungula vol.

$\therefore \text{Vol} = 317\,[\;] = 308. \qquad [\;] = \frac{308}{317} = .972$

Let $\theta = 100°$ Then $\left[.9848 - \frac{.9848^3}{3} - (1.745)(-.1736)\right] = .970$ ✓✓

$x = r\cos\theta = 14(.1736) = 2.43"$

$b = r + x = 14 + 2.43 = 16.43"$

$h = b(.1152) = 1.895"$

Contact area A_1 = approx. $\frac{\pi r^2}{2} + 2r x = \frac{615.75}{2} + 28(2.43)$

$A_1 = 307.9 + 68 = 375.9 \text{ in}^2$

$F_{max} = A_1 P_1 = (375.9)(53,800) = 20,200,000$ lbs.

$F/WT. = \frac{20,200,000}{46,024} = 438$ G vertically

Component along axis

$(438 \text{ G})(\cos 6.57°) = 438(.99344) = 435$ G

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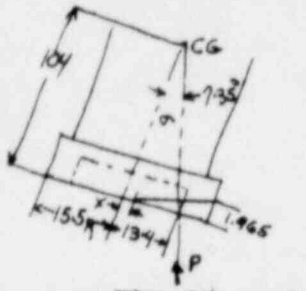
I. <u>TOP CORNER IMPACT</u> (method similar to Bottom Impact analysis)



α = arc tan $\frac{13.4}{104}$ = 7.35°

Assume center of impact at edge of closure, R = 13.4"

KE/p_1 = 308 in³ = rigid Vol. ungula

Vol = 308 in³ = $\frac{h}{b} r^3 \left[\sin\theta - \frac{\sin^3\theta}{3} - \theta\cos\theta\right]$

$\frac{h}{b} = \frac{13.4}{104} = .129 \qquad R^3 = (15.5)^3 = 3724.$

$[\;] = \frac{308}{.129(3724)} = .641$

Let θ = 89° then [ ] = .99985 − .333 − 1.5533 × .0174 = .639 ✓

x = r cos θ = 15.5 × .0174 = <u>.27"</u>

b = 15.5 − .27 = <u>15.23</u>

h = .129 b = <u>1.965</u>

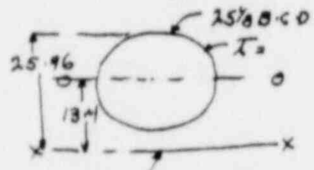
Contact area = app $\frac{\pi r^2}{2} - 2rx = \frac{754.8}{2} - 31(.27)$

A_1 = 377.4 − 8.4 = <u>369 in²</u>

$F_{max} = A_1 p_1$ = (369) 53800 = <u>19,850,000</u> LBS

F_{max}/WT = 19,850,000/46,074 = <u>430 G</u>

Component along axis = 430 × cos 7.35° = 430 × .99178 = <u>426.5 G</u>



Line of impact at P above

<u>BOLT LOADS</u>

Let A = Tensile area of 24 bolts (1¼ − 7) = 24 × .969 in² = <u>23.256 in²</u>

assume this area distributed as a thin ring with B.C. dia = 25⅛ in

Then $t = \frac{A}{2\pi r} = \frac{23.256}{\pi \times 25.125} = $ <u>.2946 in</u>

$I_{oo} = \pi R^3 t = \pi (12.5)^3 .2946$ = 1835. in⁴

$I_{xx} = I_{oo} + A(13.4)^2$ = 1835 + 4180 = 6015 in⁴

<u>Weights</u>

860 closure head
266 } Container
1653 }
1800 Contents
4579 LBS

<u>Moment about axis XX</u>

M = (4579 lb)(426.5 G) × 13.4" = <u>26,250,000"#</u>

<u>For bolt material</u>

$S_t = \frac{My}{I_{xx}} = \frac{26,250,000 \times 25.96}{6015}$ = <u>113,000</u> PSI

Bolts are heat-treated to 130,000 T.S at <u>yield point</u>

M.S. = $\frac{130,000}{113,000} - 1$ = 15%

Note that the actual <u>dynamic</u> strength is somewhat greater.
Therefore, the bolts will not yield under dynamic loading and the cask design – without fins – provides the required integrity for seals and shielding.

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J. PUNCTURE STRESS DUE TO 40" DROP

1. This cask has Uranium as shielding, in place of lead, between the inner and outer shells. Therefore, piercing of the outer shell with potential loss of shielding in the fire case cannot occur.

Instead, a shallow indent appears on the stainless steel outer shell where the edge of the 6" dia pin partly cuts into it. This is clearly shown in a drop test photograph by Union Carbide Co - Figure 19 in report KY-546 by Clifford. Actually, the pin is more damaged than the cask.

2. Energy is, however, absorbed in the three concentric shells of the cask in bending, with the possible formation of a plastic hinge.

3. The approximate distribution of moments among the shells can be made, assuming elastic deflections and an extreme fiber stress of 30,000 psi for ~~both~~ Uranium and S.S nested shells, with the same deflections, of course. Relative values follow:

34.2 %	25,300 in lbs	1" S.S outer shell
59. %	43,500 "	3½" Uranium
6.8 %	5,030 "	5/8" S.S Inner shell
100 %	73,830	

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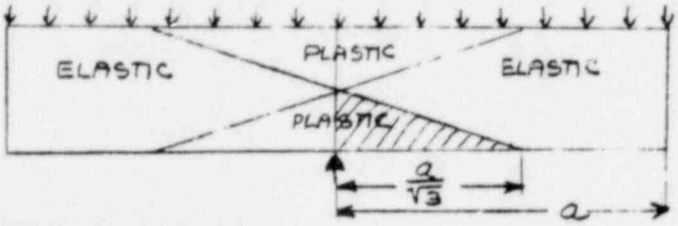
4. THE "HANDBOOK OF THE ENGINEERING SCIENCES - VOL I - THE BASIC SCIENCES" (PAGES 1302-1307) GIVES THE PLASTIC REGION DIAGRAM IN A BEAM WHEN THE MIDSECTION IS JUST COMPLETELY PLASTIC.



$a = \frac{190''}{2}$
$r_o = 13''$
$r_i = 9.5''$
$H = \frac{a}{\sqrt{3}} = 55''$

FOR THE URANIUM CYLINDER:

THE PLASTIC VOLUME = 4 × VOLUME OF SHADED UNGULA

$$= 4 \times \left(\tfrac{2}{3}\right)\left(r_o^2 - r_i^2\right) H = 4\left(\tfrac{2}{3}\right)\left(13^2 - 9.5^2\right) 55$$

$$= 11,520 \ \text{IN}^3$$

EACH PLASTIC CUBIC INCH HAS BEEN ELONGATED 2/10 % TO REACH A TERMINAL STRESS OF 30,000 PSI (Y.P FOR URANIUM)

THE ENERGY ABSORBED IS

$$U = \frac{30,000}{2}(11,520)(.002) = \underline{345,600 \ \text{IN. LBS.}}$$

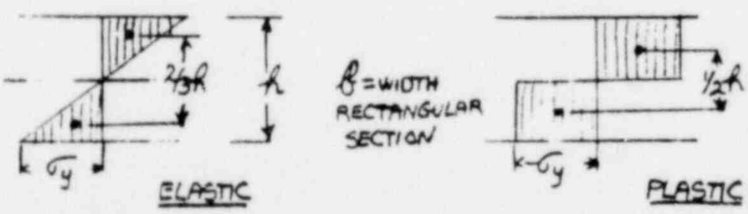
THE TOTAL ENERGY OF THE 40" FALL IS

$$U_{TOT.} = (46,500\#)\ 40'' = 1,860,000. \ \text{IN LBS}$$

THE DIFFERENCE MUST BE ABSORBED IN BENDING OF THE "PLASTIC HINGE".

$$U_B = 1,860,000 - 345,600 = \underline{1,514,400 \ \text{IN LBS}}$$

5. DETERMINATION OF PLASTIC SECTION MODULUS.



ELASTIC

PLASTIC

$$M_E = \frac{\sigma_y}{2}\ \frac{h}{2}\left(\frac{2}{3}h\right) b = \sigma_y \frac{bh^2}{6}$$

$$M_P = \sigma_y\ \frac{h}{2}\ \frac{h}{2}\ b = \sigma_y \frac{bh^2}{4}$$

$$\therefore M_P = \frac{3}{2} M_E$$

FOR THE CIRCULAR SECTION OF THE URANIUM CYLINDER

$$I = \frac{\pi}{4}(R_o^4 - R_i^4) \qquad Z_E = \frac{\pi}{4}\frac{(R_o^4 - R_i^4)}{R_o}$$

AND
$$Z_p = \frac{3}{2}\frac{\pi}{4}\frac{(R_o^4 - R_i^4)}{R_o}$$
$$= \frac{3\pi}{8}\frac{(13^4 - 9.5^4)}{13} = 1860$$

THE PLASTIC MOMENT AT THE MIDPLANE IS

$$M_p = \sigma_p Z_p$$
$$= (30,000)(1860) = 55,800,000 \text{ IN. LBS}$$

BUT $U_B = M\theta$

OR $1,514,400 = 55,800,000(\theta)$

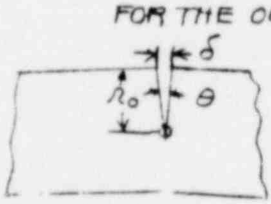
$$\theta = \frac{1,514,400}{55,800,000} = .0271 \text{ radians} \equiv \underline{1.56^\circ}$$

6. ESTIMATED ACTUAL ANGLE OF BEND AND ELONGATION

WE ARE NOW JUSTIFIED IN ASSUMING THAT THE ACTUAL CONDITIONS, SHOWING LOADINGS ON ALL THREE SHELLS, WOULD REDUCE THEIR COMMON ANGLE OF BEND TO

$$(.59)(1.56^\circ) = .925^\circ = \theta$$

FOR THE OUTER FIBERS OF THE OUTER SHELL THIS GIVES



$R_o = 14 \qquad \delta = R_o\theta = (14)(.925)(.01745) = .226''$

CONSIDER THIS TO BE THE MEASURED ELONGATION OVER A 2" GAGE LENGTH TENSION TEST PIECE.

$$\frac{.226}{2} = 11.3\% \text{ ELONGATION}$$

STAINLESS STEEL HAS 40% ELONG O.K. NO RUPTURE.

URANIUM ELONG. $= \frac{13''}{14}(11.3) = 10.5\%$ ELONG. AT $R = 13''$ - OUTER R

$= \frac{10\frac{1}{2}}{14}(11.3) = 8.5\%$ " " $R = 10\frac{1}{2}''$ - INTER R

$= \frac{9\frac{1}{2}}{14}(11.3) = 7.7\%$ " $R = 9.5$ - INNER R

THESE VALUES ARE CONSIDERED CONSERVATIVE AND SATISFACTORY

K. URANIUM WELD JOINT STUDY

1. SUCCESSFUL COMPLETION OF SIDE PUNCTURE TESTS ON A URANIUM CASTING SCALE MODEL HAS PROMPTED A REVIEW OF IMPACT CONDITION ANALYSES AND HAS ALSO ALLOWED CONSIDERATION OF A TYPE OF JOINT FOR THE URANIUM SECTIONS WHICH PERMITS A GREATLY REDUCED DEPTH OF WELDING.

THE PREVIOUS CALCULATIONS FOR SIDE WALL PUNCTURE ASSUMED THE ONSET OF PLASTIC HINGE DEFORMATION AT 30,000 PSI, AND REQUIRED THAT 83% OF THE KINETIC ENERGY OF THE 40 IN. DROP HAD TO BE ABSORBED BY THIS MECHANISM. THE NEW DROP TESTS PROVED THAT MORE THAN DOUBLE THIS HEIGHT OF DROP DID NOT PRODUCE ANY MEASURABLE PERMANENT SET. OBVIOUSLY, JUSTIFICATION OF THIS PERFORMANCE REQUIRES THAT THE CALCULATED INSTANTANEOUS PEAK BENDING STRESSES BE ALLOWED TO REACH FAR HIGHER VALUES AND STILL BE ELASTIC.

THE EFFECTS OF THIS NEW PERFORMANCE ARE NOW INVESTIGATED IN REGARD TO ACCIDENT CONDITIONS, AND INCLUDE THE BEHAVIOR OF THE REDESIGNED JOINTS UNDER SUCH LOADINGS.

2. ACCORDING TO THE TESTS, SEPARATELY REPORTED, A DROP HEIGHT OF 120 IN., WITH A REBOUND OF 31¼ IN., CAN BE CREDITED, CONSERVATIVELY, WITH KINETIC ENERGY PROPORTIONAL ONLY TO A DROP OF 120 - 31¼ = 88¾ IN. THIS MEANS THAT ENERGY 2.22 TIMES THE SPECIFIED 40 IN. DROP WAS SUCCESSFULLY ABSORBED ELASTICALLY.

3. PUNCTURE - SIDE WALL - 88¾ IN. DROP ANALYSIS.

a. THE STAINLESS STEEL OUTER SHELL, AS THE MEMBER DIRECTLY EXPOSED TO PENETRATION BY THE STEEL PISTON, IS NOT CRITICAL. THE PISTON MERELY CUTS PARTLY INTO THE WALL, DUE TO THE BACK-UP EFFECTS OF THE URANIUM CYLINDER.

b. THE URANIUM CYLINDER (AS TESTED) IS TAKEN AS A MODEL, BUT CALCULATIONS ARE NOW MADE ON A FULL SIZE CYLINDER ANALYZED AS A SEPARATE BODY IN AN ELASTIC DROP UP TO 88¾ IN.

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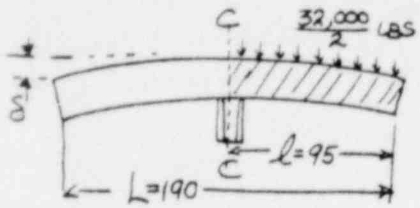

CONSIDER 1G LOAD ON CANTILEVER.

$D = 26''$ $d = 19''$

$l = 95''$ $L = 190''$

$VOL = (531-284) IN^2 (190'') = 46,930\ IN^3$ TOTAL

$WT. = 46,930 (.683) = 32,000$ LBS TOTAL

$M_C = \frac{Wl}{2} = \left(\frac{32000}{2}\right)\frac{95}{2} = 760,000$ IN.LBS

$I = .0491(26^4 - 19^4) = 16,041\ IN^4$

$Z = \frac{16041}{13} = 1235\ IN^3$

$\delta = \frac{Wl^3}{8EI} = \frac{16000\ (95)^3}{8(29)10^6(16041)} = .00369''$

$S_b = \frac{Mc}{Z} = \frac{760,000}{1235} = 615$ PSI

MARKS 5-44 GIVES $U_{CANTILEVER} = \frac{n^2}{m}\left(\frac{K}{c}\right)^2 \frac{S^2 V}{2E}$ FOR UNIFORM LOAD

$n = 2$ $m = 8$ $K =$ RAD. GYR. $= \frac{\sqrt{D^2+d^2}}{4}$ FOR TUBE SECTION

$c = D/2$

$\therefore U_{CANT.} = \frac{(2)^2}{8} \frac{4\,(D^2+d^2)}{16\ D^2} \frac{S^2 V}{2E} = \frac{D^2+d^2}{16D^2} \frac{S^2 V}{E}$ FOR ½ BEAM LGTH.

$U_B = \frac{D^2+d^2}{8D^2} \frac{S^2 V}{E}$ FOR FULL LENGTH BEAM AS ABOVE

$= \frac{26^2+19^2}{8 \times 26^2} \frac{S^2 (46,930)}{29\ (10^6)} = \frac{676+361}{8\ (676)} S^2 \frac{1615}{10^6} = \frac{310}{10^6} S^2$

NOW $S = 615$ PSI FOR 1G LOAD

$U_B = \frac{310\,(615)^2}{10^6} = \underline{117\ IN.LBS}$ NOTE: $U_B \propto S^2 \propto (G's)^2$

LINE	G'S	S PSI	δ IN.	$(G's)^2$	U_B IN.LBS.	HEIGHT DROP
1	1	615.	.00369	1	117	
2	50	30,750	.1845	2500	292,500	
3	104.5	64,300	.386	10,920	1,280,000	40
4	156.	96,000	.575	24,280	2,840,000	88.75

THE STRESS OF 96,000 PSI IS REACHED "INSTANTANEOUSLY" AND ONLY AT THE TOP MID POINT OF THE BEAM. THIS PEAK IS QUITE CONSISTENT WITH NOMINAL PROPERTIES OF UNALLOYED CAST URANIUM.

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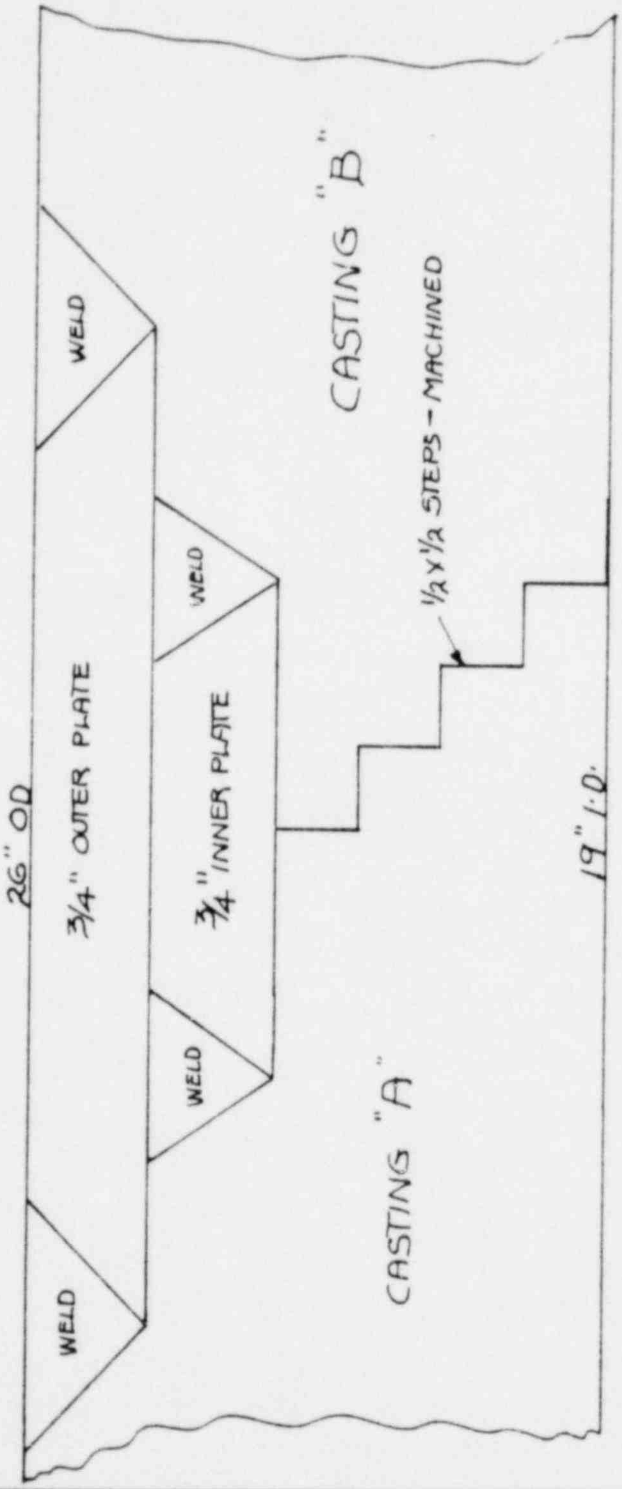

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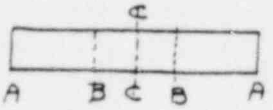
4. THE SPECIFICATION IS LIMITED TO THE VALUES OF LINE 3.



THE JOINTS ARE ACTUALLY AT POSITIONS B AND THE MOMENT AND STRESS IS REDUCED TO $\frac{4}{9}$.

$$\frac{M_B}{M_C} = \frac{(2/3\,WT)(2/3\,\ell)}{(WT)(\ell)} = 4/9 \quad \therefore S_B = \frac{4}{9} S_{℄} = \frac{4}{9}(64{,}300)$$
$$= \underline{28{,}600 \text{ PSI}}$$

SINCE THE "SOLID" CYLINDER OF THE TESTS (EQUIVALENT TO FULL DEPTH PENETRATION WELD) HAS WITHSTOOD AT ITS MIDSECTION 96,000 PSI, IT WOULD BE THEORETICALLY POSSIBLE TO REDUCE THE DEPTH OF AN OUTSIDE WELD FOR POSITION B SO THAT Z FOR THE WELD AREA IS ONLY

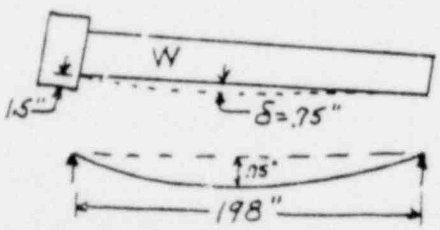
$$Z_{B2} = \frac{28{,}600}{96{,}000}\ 1235 = \underline{368}$$

TO FIND THE I.D. WHICH CORRESPONDS TO THIS Z VALUE, LET

$$Z_{B2} = 368 = \frac{.098}{26}(26^4 - d^4) = .00378\left[457{,}000 - d^4\right]$$
$$= 1725 - .00378\,d^4$$
$$d^4 = \frac{1725 - 368}{.00378} = 359{,}000$$
$$d = 24.5 \text{ IN} \qquad \therefore t = \frac{26 - 24.5}{2} = \underline{3/4}\text{" WELD RQD. } \underline{\text{MIN.}}$$

TO BE QUITE CONSERVATIVE, IT IS DESIRABLE TO DOUBLE THIS DEPTH OF WELD, USING 2 LAYERS OF 3/4 IN. PENETRATION WELDS (IN OFFSET RELATIONSHIP), PROVIDING CONTINUOUS BEAM STRENGTH IN AN OUTER ANNULUS 1½ IN. THICK OUT OF A TOTAL OF 3½ IN. WALL. THE INNER 2 IN. WOULD BE MACHINED WITH INTERLOCKING STEPS (FOUR OF ½ IN. EACH) PROVIDING CONCENTRIC SHEAR RINGS AND EFFECTIVE SHIELDING PATTERN (AS ILLUSTRATED). THIS DESIGN IS NOW INVESTIGATED IN REGARD TO THE OTHER CONDITIONS AND ATTITUDES OF IMPACT.

5. SIDE DROP - 30 FT.



DEFLECTION AT MID-LENGTH IS LIMITED TO .75" BY CONTACT WITH BASE

FIND (W·g) TO GIVE δ = .75"

$$\delta = \frac{5}{384}\frac{W\ell^3}{EI} = .75$$
$$\therefore (W \cdot g) = \frac{.75\,(384)}{5}\frac{EI}{\ell^3} = \frac{57.6\,(29)10^6\,I}{(198)^3}$$
$$\underline{W \cdot g = 215\,I}$$

LET $W = 46024$ LBS = TOTAL WEIGHT OF LOADED CASK.

LET $I = \Sigma I$ FOR OUTER SHELL, INNER SHELL, AND 1½ IN URANIUM (WELDED)

THE REST OF THE 3½ IN. TOTAL THICKNESS OF URANIUM CONTRIBUTES MASS AND NOT STIFFNESS. THIS IS VERY CONSERVATIVE, SINCE THE MIDDLE CYLINDER SPANS THE REGION OF MAX. STRESS AND DEFLECTION. ALL THREE CYLS. HAVE THE SAME DEFLECTIONS

I_1 OUTER SHELL $= \frac{\pi}{4}(14^4 - 13^4) = 7750 \text{ IN}^4$ $\quad \Sigma I/c = \frac{17980}{14} = 1284$

I_2 INNER SHELL $= \frac{\pi}{4}(9.5^4 - 8.875^4) = 1530$ $\quad = \frac{17980}{9.5} = 1895$

I_3 URANIUM $= \frac{\pi}{4}(13^4 - 11.5^4) = \underline{8700}$ $\quad = \frac{17980}{13} = 1385$

$\Sigma I = 17,980 \text{ IN}^4$

NOW $(W \cdot G) = 215\,(\Sigma I)$

$= 215\,(17,980) = 3,880,000$ LBS

$G = \frac{3,880,000}{46,024} = \underline{83.8 \text{ G's}}$

$M = \frac{(W \cdot G)\,l}{8} = \frac{3,880,000}{8}(198) = 95,900,000$ IN. LBS.

AND S_1 OUTER SHELL $= \frac{95,900,000}{1284} = \underline{74,700 \text{ PSI}}$

S_2 INNER SHELL $= \frac{95,900,000}{1895} = \underline{50,600 \text{ PSI}}$

S_3 URANIUM $= \frac{95,900,000}{1385} = \underline{69,200 \text{ PSI}}$ $\quad < 96,000$ PSI (OF LINE 3)

ALL THESE STRESSES ARE MOMENTARY PEAKS AND LOCAL.

S_3 FOR URANIUM IS LESS THAN THAT SUSTAINED IN THE TEST.

THE SIDE DROP TEST IS CONSIDERED SATISFACTORY, WITH NO PERMANENT BEAM DEFORMATIONS — ONLY LOCAL CRUSHING DEFORMATIONS.

6 END IMPACTS THE WELD METAL IS CONSIDERED EQUIVALENT TO THE BASE METAL.

LOADS APPLIED AXIALLY WOULD FIRST PASS THRU WELD METAL AREAS, AND THEN WOULD DEVELOPE LOAD ABILITY AT THE STEP INTERFACES. SINCE THREE CASTINGS ARE USED COAXIALLY AND PARTLY WELDED, EACH OF THE TWO JOINTS HAS A MAX. LOAD = 2/3 THE MAX. LOAD FOR THE EXTREME END SURFACE AT TOP OR BOTTOM OF THE CASK.

THEREFORE, IMPACT LOADINGS OF ALL TYPES ON THE PARTIAL-DEPTH WELDED JOINTS OF THIS URANIUM SHIELDED CASK ARE CONSIDERED SUCCESSFULLY SUSTAINED WITHOUT FAILURE PER SPECIFICATIONS.

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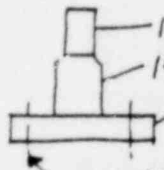
NATIONAL LEAD CO. NUCLEAR DIVISION
WILMINGTON PLANT

2. IMPACT - BOTTOM CORNER - DRAIN CAP



$\pi/4\,(1.125)^2 \times 3 = 3.0\ in^3$

$\pi/4\,(1.625)^2 \times 5 = 10.4$

$\pi/4\,(5.5)^2 \times 1 = \underline{23.8}$

$37.2\ in^3 \times .284 = \underline{10.57\ LBS}$

Component of deceleration along axis of cask in corner drop is

435 g's (see page 5.58)

Impact force on Drain Cap is

$F = (10.57)\,435 = \underline{4600\ LBS}$

Each Cap Screw is rated, at yield point stress, as

$\underline{20,400\ LBS}$ SPS High strength (Commercial) @ $\underline{130,000\ PSI\ YP}$

$\underline{4,700\ LBS}$ Stainless Steel @ 30,000 PSI YP

Minimum strength is secured with 4 Stainless steel cap screws

$4 \times 4700 = \underline{18,800\ LBS}$

$M.S = \frac{18,800}{4600} - 1 = \underline{3.08}$ OK.

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M. VIBRATION OF CASK IN TRANSIT

This cask is designed for transport by trailer only.

Therefore, the only concern is that the fundamental frequency of vibration as a simply supported beam, loaded by its own weight, be appreciably higher than the repeatable impulse frequencies of the trailer itself.

The cask is considered to have a total I equal to the sum of the individual I values of the two shells and the Uranium cylinder.

$I_{outer\ shell} = \pi/4\,(14^4 - 13^4) = 7{,}750\ in^4$

$I_{inner\ shell} = \pi/4\,(9.5^4 - 8.875^4) = 1{,}530$

$I_{Uranium} = \pi/4\,(13^4 - 9.5^4) = \underline{16{,}041}$

$I_{total} = 25{,}321\ in^4$

Total weight of cask and contents = $\underline{46{,}500\ lbs}$

$l = 208"$

$$\text{Frequency } \underline{CPS} = \frac{3.55}{\sqrt{\frac{5}{384}\frac{W l^3}{E I}}} = \frac{3.55}{\sqrt{\frac{5}{384}\frac{(46{,}500)(208)^3}{29\times10^6(25{,}321)}}}$$

$= \underline{41.2\ CPS}$ OK > trailer frequencies

Fundamental frequencies developed in trailers are generally in the range of 4 to 16 C.P.S. (See "Shock and Vibration Handbook - Vol 3 - Sect 45)

These lower frequencies are apparent from the very dimensions of the trailer, the greater concentration of loading, and the smaller I values for the trailer main beam sections.

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SECTION VI

DIFFERENTIAL THERMAL EXPANSIONS

SECTION VI

NATIONAL LEAD CO. NUCLEAR DIVISION
WILMINGTON PLANT

DIFFERENTIAL THERMAL EXPANSIONS

A. CLEARANCES

The cask and its container assemblies are stainless steel and uranium constructions. No lead is present and thus there are no problems associated with voids of this kind, which vary greatly in volume with changes in temperatures and also shift in position within the cask. The uranium is monolithic and jacketed by the steel. The dimensional proportions of the cask shielding cylinder of uranium require that there be minimal clearances for machining and assembly purposes. These clearances are a substantial part of the differential expansions which developed in several of the cases examined.

B. COEFFICIENTS OF EXPANSION

Coefficients of expansion used ($\times 10^{-6} \times {}^{\circ}F$) 6.5 uranium. This value is obtained from records of our Albany plant and refer specifically to as-cast .2% molybdenum uranium - unalloyed composition.

Stainless Steel values are from Section III, table N-426 of the Nuclear Code, as follows:

A = instantaneous values at given temperature
B = mean coefficient (from 70°F to indicated temperature)

A	B	Temp. °F
9.11	9.11	70
9.73	9.47	300
10.43	9.82	600
10.90	10.05	800

C. TEMPERATURE DISTRIBUTION

Temperature distribution through the cask under hypothetical accident and fire conditions has been obtained from memorandum III, a part of the specification, dated 18 April 1968, and titled "Heat Transfer Calculations for PSC Fuel Shipping Cask". Heat generation rates were chosen in each case to give the maximum differential temperatures.

D. CASES INVESTIGATED

The following cases are investigated relative to axial and to radial differential expansions and contractions for the two uranium bodies contained within the stainless steel structure.

Case 1 - 30 minutes after start of fire (1101 BTU/hr fuel rate)
Case 2 - 10 hours after start of fire (2322 BTU/hr fuel rate)
Case 3 - start up - cask 70° - container 240°
Case 4 - Immersion in water 70° - container and inner shell 240°
Case 5 - Low temp. - 40° whole cask - No container.
Case 6 - Low temp. -40° cask - Container and inner shell 240°

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SECTION VI - DIFFERENTIAL THERMAL EXPANSIONS

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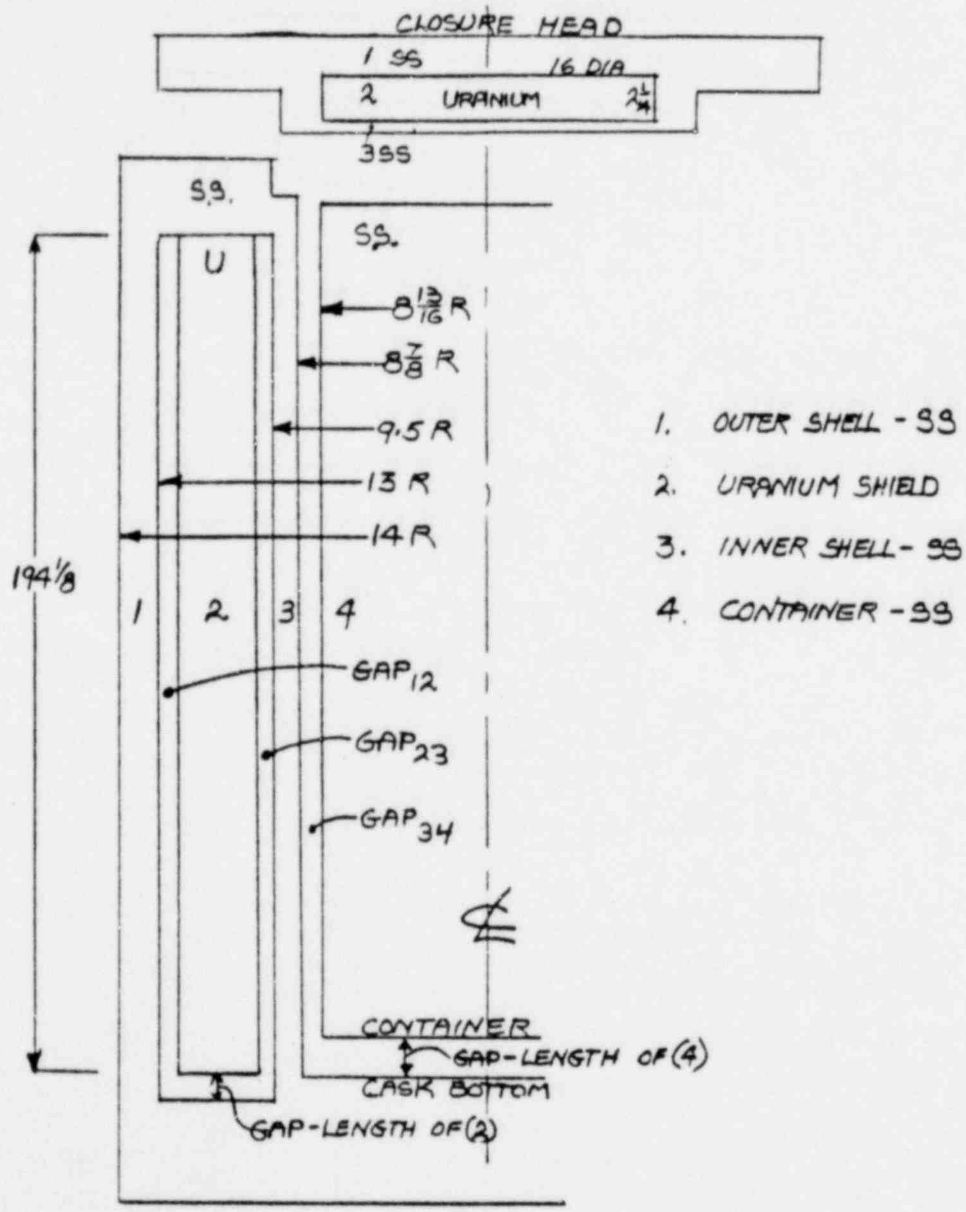

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SECTION VI - DIFFERENTIAL THERMAL EXPANSIONS

NATIONAL LEAD CO. NUCLEAR DIVISION
WILMINGTON PLANT

E. Analysis of Uranium Shielding in Cask

In the calculations + is a clearance or gap
- is an interference (based on original O gap)

The various negative (-) dr or dl values thus indicate the minimum initial clearances at 70°F required to prevent interference and stressed conditions. These requirements are reflected in the drawings.

The maximum values required for such clearances for the cask itself are:

Gap 12 = -.0322 from case 2
Gap 23 = -.015 from case 2
Gap 34 = -.0367 from case 3
Length container = -.309 from case 6

INTERFERENCES - TO BE PREVENTED BY SUITABLE MFG. CLEARANCES.

Case 1 $T_1 = 1120^\circ$ $T_3 = 430^\circ$ $T_0 = 70^\circ$ $T_4 = 370^\circ$

Gap 12 Assume $T_2 = T_3$ for max. diff.

$dr = (13")(T_1 - T_0)(10.05)10^{-6} - (13")(T_2 - T_0)(6.5)10^{-6}$

$= .137 - .0304 = +.1066"$ SS>U GAP

Gap 23 Assume $T_2 = T_3$ for min. clearance

$dr = (9.5")(T_2 - T_0)(6.5)10^{-6} - (9.5")(T_3 - T_0)(9.6)10^{-6}$

$= .0222 - .0328 = -.0106$ U<SS INTERFERANCE

Gap 34 Assume $T_4 = 370^\circ$ from 2322 BTU/hr. fuel rate

$dr = (8\text{-}7/8")(T_3 - T_0)(9.6)10^{-6} - (8\text{-}13/16")(T_4 - T_0)(9.5)10^{-6}$

$= .0306 - .0251 = +.0055$ GAP

Length of U. Assume $T_2 = T_3$ for max. differential

$dl = (194\tfrac{1}{8})(T_1 - T_0)(10.05)10^{-6} - (194\tfrac{1}{8})(T_3 - T_0)(6.5)10^{-6}$

$= 2.05 - .455 = 1.595"$ expansion SS>U GAP

CONTAINER SHOWS GAP.

SECTION VI - DIFFERENTIAL THERMAL EXPANSIONS

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WILMINGTON PLANT

Case 2 $T_1 = 220^\circ$ $T_3 = 340^\circ$ $T_0 = 70^\circ$ $T_4 = 370^\circ$

Gap 12 Assume $T_2 = T_3$ for min. clearance

$dr = (13")\,(T_1 - T_0)\,(9.4)\,10^{-6} - (13)\,(T_2 - T_0)\,(6.5)\,10^{-6}$

$= .0183 - .0228 = -.0045$ U>SS INTERFERENCE

Gap 23 Assume $T_2 = T_1$ for min. clearance

$dr = (9.5)\,(T_2 - T_0)\,(6.5)10^{-6} - (9.5)\,(T_3 - T_0)\,(9.47)\,10^{-6}$

$= .00925 - .0243 = -.015$ U<SS INTERFERENCE

Gap 34 Assume $T_4 = 370^\circ$

$dr = (8\text{-}7/8)\,(T_3 - T_0)\,(8.47)10^{-6} - (8\text{-}13/16)\,(T_4 - T_0)\,(9.47)10^{-6}$

$= .0227 - .025 = -.0023$ INTERFERENCE

Case 3 Cask at original dimen. $T_4 = 240^\circ$ $T_0 = 70^\circ$ $T_3 = 70^\circ$

Gap 34 $dr = 0 - (8\text{-}13/16)\,(T_4 - T_0)\,(9.7)10^{-6}$

$= -.0367"$ container increase - INTERFERENCE

Length Container

$dl = (187\text{-}5/8)\,(T_4 - T_0)\,(9.7)10^{-6}$

$= -.309"$ container increase INTERFERENCE

Case 4 $T_1 = 70^\circ$ T_2 assumed $= T_3 = 450^\circ$ $T_0 = 70^\circ$ $T_4 = 240^\circ$

Gap 12 $= 0 - (13)\,(T_2 - T_0)\,(6.5)\,10^{-6}$

$= 0 - .0322 = -.0322$ SS<U INTERFERENCE

Gap 23 $= 0$

Gap 34 = negligible

Length Container - in time same as case 3

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SECTION VI - DIFFERENTIAL THERMAL EXPANSIONS

NATIONAL LEAD CO. NUCLEAR DIVISION
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Case 5 $T_1 = T_2 = T_3 = -40^\circ \quad T_0 = +70$

Gap 12 $dr = (13)(T_1 - T_0)(9.11)10^{-6} - (13)(T_1 - T_0)(6.5)10^{-6}$

$= -.013 + .0093 = -.0037$ SS<U INTERFERENCE

Gap 23 $dr = (9.5)(T_2 - T_0)(6.5)10^{-6} - (9.5)(T_3 - T_0)(9.11)10^{-6}$

$= -.0068 + .0093 = +.0027$ gap

Case 6 $T_1 = T_2 = -40^\circ \quad T_3 = T_4 = 240^\circ \quad T_0 = 70^\circ$

Gap 23 $dr = (9.5)(T_2 - T_0)(6.5)10^{-6} - 9.5(T_3 - T_0)(9.11)10^{-6}$

$= -.0068 - .0147 = -.0215$ INTERFERENCE

F. Analysis of Uranium Disc in S.S. Closure Head

Case 1 Assume same gradients as given for cask.

$T_1 = 1120^\circ \quad T_2 = 430^\circ \quad T_0 = 70^\circ$

$\Delta t_s = 1050^\circ$ for S.S. space expansion

$\Delta t_u = 430 - 70 = 360^\circ$ for U expansion from original machined dimension

Length $dl = (16")[\Delta t_s 10.05 - \Delta t_u 6.5]10^{-6}$

$= 16[1050(10.05) - 360(6.5)]10^{-6}$

$= 16(11000 - 2340)10^{-6} = 0.1384$ GAP

Thickness - also negligible

$dT = \frac{2^1/_4}{16}(0.1384) = .0195$ GAP

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Case 2 $T_1 = 220^\circ$ $T_3 = 340^\circ$ $T_0 = 70^\circ$

$\Delta t_s = 150^\circ$

$\Delta t_u = 340-70 = 270^\circ$

Length $dl = 16 [\Delta t_s (9.47) - \Delta t_u (6.5)] 10^{-6}$

$= 16 [150 (9.47) - 270 (6.5)] 10^{-6}$

$= 16 (1420 - 1750) 10^{-6}$

$= 5280 (10^{-6}) = -.00528''$

Thickness $dw = .00528 \times \frac{2\text{-}1/4}{16} = -.00075$ INTERFERENCE

Case 3 no diff. exp.

Case 4 no diff. exp.

Case 5 Temp. drop $= -40 - (+70) = -110^\circ$

Length $dl = 16 (-110) (9.11-6.5) 10^{-6}$

$= -4600 (10^{-6}) = -.004,6''$ INTERFERENCE

Case 6 no diff. exp.

Summary - all cases For disc. max. interference is only .00528", assuming metal to metal fit at 70°.

Manufacturing clearances would be greater than this for the uranium disc in the closure head.

SECTION VII

SHIELDING ANALYSIS

SHIELDING ANALYSIS

A. SUMMARY

Dose rates have been calculated for several locations on and off the surface of the FSV-1 spent fuel shipping cask. The truck and trailer are used solely to transport one shipping cask.

B. GEOMETRY AND SOURCES

The cask is designed to hold six 14.22-inch hexagon by 31.22-inch long fuel blocks in a single column. The cask is made of depleted uranium encased in steel plate. Table VII-1 summarizes the important data used in the shielding design. The total activity of isotopes of importance to shielding design is approximately 58 (+4) curies. The location and values for the calculated dose rates are shown in Fig. VII-1.

Sources for fuel that had decayed for 100 days were used as input for the calculations and are listed in Table VII-2. Isotopes emiting low energy gammas were not considered. The sources represent the highest fission product inventory a fuel block will have accumulated after scheduled burnup in the equilibrium core.

C. RESULTS

The dose rates and locations of the dose points are shown in Fig. VII-1. The limit of accuracy in the calculations is $\pm 50\%$. Shielding thickness should not be increased to reflect this limit of accuracy. If dose rates exceed the calculated values, the storage time of the blocks may be increased to as long as 200 days.

It should be noted that the cask is designed to handle fuel from an equilibrium core. Some fuel blocks that are unloaded during the approach to equilibrium may have experienced higher peaking factors during operation and will therefore contain a greater amount of fission products compared with elements from an equilibrium core. The difference will be less than a factor of two. These elements will be loaded such that they occupy the bottom and top positions in the cask, having elements that have experienced a peaking factor of less than 1.0 in between them. This arrangement provides more shielding for the "hotter" elements due to the geometry of the loaded shipping cask.

TABLE VII - 1

DATA FOR PSC-SPENT FUEL SHIPPING CASK AS ASSUMED FOR RADIATION ANALYSIS

Inside radius of steel liner (21.11 cm)	8 5/16 in.
Thickness of fuel container plus inner liner (2.857 cm)	1 1/8 in.
Thickness of Uranium (8.89 cm)	3 1/2 in.
Density of Uranium	18.9 gm/cc
Thickness of outer shell liner (2.54 cm)	1.0 in.
Height of fuel column (476.0 cm)	15.6 ft.
Average density of fuel (assumed to be carbon only)	1.54 gm/cc
Shield thickness on cask bottom (27.94 cm)	11.0 in. Fe.
Shield thickness on cask top	
Steel (next to fuel) (6.19 cm)	2 7/16 in.
Uranium (5.72 cm)	2 1/4 in.
Steel (top of cask) (7.30 cm)	2 7/8 in.

TABLE VII - 2

SOURCE DATA FOR ONE FUEL BLOCK

Isotope	Curies
Y-91	1.19 (+4)
Zr-95	1.36 (+4)
Nb-95m	2.7 (+2)
Nb-95	1.36 (+4)
Ru-103	2.7 (+3)
Rh-106	1.60 (+3)
Ba-137m	2.08 (+3)
Ba-140	1.86 (+2)
La-140	1.86 (+2)
Ce-144	2.52 (+4)
Pr-144	2.52 (+4)

SECTION VII - SHIELDING ANALYSIS

D. CALCULATIONS

The GAC PATH Code was used to calculate dose rates that were then adjusted to account for buildup factors in laminated shields. The PATH Code is described in Attachment VII-1. Isotopes and their activities shown in Table VII-2 were used as input data for the shielding calculations. The activity was assumed to be uniformly distributed in each fuel block.

In order to calculate a dose rate for a laminated shield composed of uranium and steel, the dose rate is calculated using buildup factors for steel and then calculated using buildup factors for uranium. The actual dose rate results from adjusting these dose rates by an interpolation technique for buildup factors in a laminated shield.

The following example illustrates the method used for all dose rates computed. The information and method used was developed by B. A. Engholm of GAC.

1. Calculation for Dose Rate at Top of Cask

Assume predominant energy of gamma spectrum is 2 Mev (a conservative assumption).

2-7/16" Fe = 2.07 mfp (Fe) mfp = mean free path (gamma ray)

2-1/4" U = 5.18 mfp (U)

2-7/8" Fe = 2.44 mfp (Fe)

Assume 2.07 mfp (Fe) + 5.18 mfp (U) = 7.25 mfp (U)

TOTAL mfp's = 7.25 mfp (U) + 2.44 mfp (Fe) = 9.69 mfp

Br = Buildup
Br-1 for 5 mfp (U) + 2.44 mfp (Fe) [7.44 mfp total] = 5.7
Br-1 for 10 mfp (U) + 2.44 mfp (Fe) [12.44 mfp total] = 9.0

INTERPOLATING -

$$\text{Br-1 for 9.69 mfp} = 5.7 + \frac{9.69 - 7.44}{12.44 - 7.44} \quad (9.0 - 5.7)$$

Br-1 = 7.2

Dose rate at Top of Cask using PATH and Iron Buildup Factors = 179.5 mr/hr

$$\text{Actual dose rate} = 179.5 \text{ mr/hr} \left(\frac{5.7}{7.2}\right) = 142 \text{ mr/hr}$$

Dose rate = 142 mr/hr

The above technique was used in computing all dose rates.

LOCATION	DOSE RATE (mr/hr)
1	2
2	142
3	166
4	66
5	303
6	8
7	6
8	7
9	52

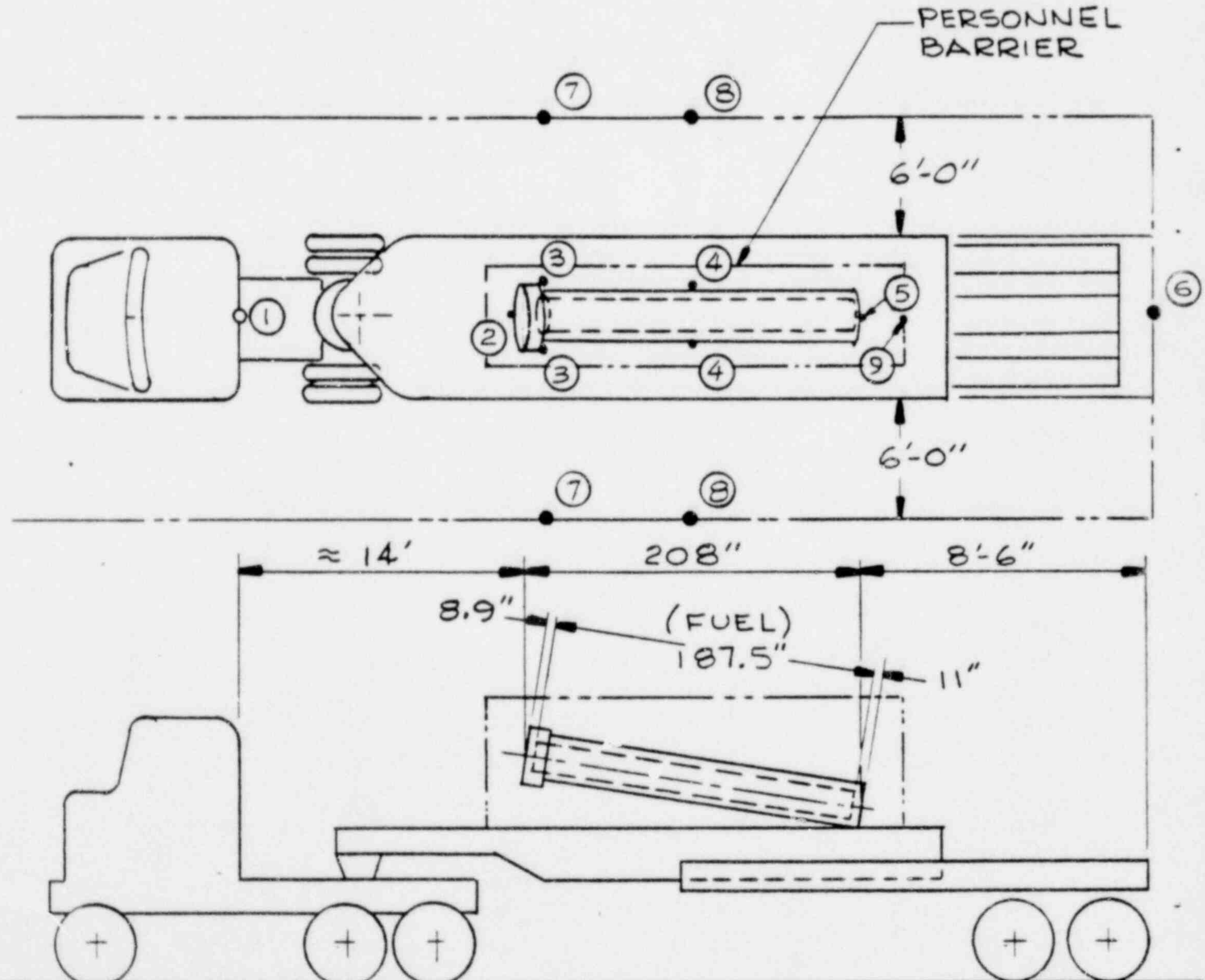

Fig. VII-1 Radiation Levels During Transit of FSV-1 Fuel Shipping Cask

E. GAMMA BUILDUP IN TWO-LAYER CONFIGURATIONS OF DIFFERENT SHIELDING MATERIALS

Frequently it is desirable to use two-and three-layer configurations of iron/concrete; iron/carbon; iron/lead/iron; or iron/depleted uranium/iron (as in the Fort St. Vrain fuel shipping cask) for gamma shielding. Therefore, sets of curves for these materials at two source energies (1 Mev and 2 Mev) have been derived. These curves have been used in HTGR shielding calculations and for computing radiation levels for the Fort St. Vrain fuel shipping cask.

F. DERIVATION OF CURVES

Dose buildup curves were derived for solid iron, solid lead, and solid uranium first plotted for point isotropic sources of 1 Mev or 2 Mev energy, using data directly from Ref. VII-2. In unpublished work, Chilton has shown that if a log-log grid is utilized with B_r-1 plotted against mean free paths (mfp), the curves will be very nearly straight lines. This method has been adopted in the attached figures, and the curves extrapolated to 40 mfp.

Next, the trends displayed in the two-layer iron/concrete configurations in Figure 21 and 22 of Ref. VII-1 were carefully studied. One set of curves would be applicable to a situation involving a point source like Co^{60} enclosed in a small -ID spherical cask. The other set of curves is appropriate to large source geometries and shields with relatively large radii or curvature, up to and including plane sources and slab shields. The latter set of curves is more generally useful for HTGR shield design. Furthermore, the trends established in these curves agree quite well with the results of two other studies (Refs. VII-3 and VII-4).

The method for drawing the curves for two-layer configurations of iron/lead and iron/uranium was pretty much artistic rather than scientific. Essentially, the envelopes between the curves for solid iron and solid lead, or solid iron and uranium, were subdivided into the two-layer curves using trends and proportions from the iron/concrete curves of Ref. VII-1.

G. EXAMPLES

In practical cases, it will generally be necessary to convert thickness to mfp. The conversions are:

MATERIAL	CONVERSION FACTOR (mfp/inch)	
	1 Mev	2 Mev
Iron	1.19	0.85
Lead	1.97	1.32
Uranium	3.6	2.3

It is important to remember that the curves are plotted for B_r-1 *versus* mfp, instead of B_r *versus* mfp. Don't forget to add 1.0 to the final buildup result.

Example 1 Shield consisting of 3 inches uranium followed by 2 inches of steel. Source energy = 2 Mev.

$$\left.\begin{array}{l} \text{mfp U} = 3 \times 2.3 = 6.9 \\ \text{mfp Fe} = 2 \times 0.85 = 1.7 \\ \text{Read from Fig. VII-5.} \end{array}\right\} \quad \text{total mfp} = 8.6$$

B_r-1 for 5 mfp U followed by 1.7 mfp Fe (i.e., 6.7 mfp total) = 4.6

B_r-1 for 10 mfp U followed by 1.7 mfp Fe (i.e., 11.7 mfp total) = 7.6

Either linear or logarithmic interpolation between 6.7 and 11.7 mfp can be used.. Linear interpolation gives:

$$B_r\text{-1 for 8.6 mfp} = 4.6 + \frac{8.6 - 6.7}{11.7 - 6.7} (7.6 - 4.6) = 5.74$$

Logarithmic interpolation yields:

$$B_r\text{-1 for 8.6 mfp} = \text{antiln} \left[\ln 4.6 + \frac{\ln\frac{8.6}{6.7}}{\ln\frac{11.7}{6.7}} \ln \frac{7.6}{4.6} \right] = 5.75$$

Therefore, the required buildup factor is $B_r = 5.75 + 1.0 = \underline{\underline{6.75}}$

Example 2 Shield consisting of 10 mfp lead followed by 3 mfp steel. Source energy = 1.6 Mev.

Read from Figure VII-2.

B_r-1 for 10 mfp Pb followed by 3 mfp Fe, for 1 Mev source energy = 15.0

Read from Figure VII-3.

B_r-1 for 10 mfp Pb followed by 3 mfp Fe, for 2 Mev source energy = 11.

Using linear interpolation between source energies:

$$B_r\text{-1 for 1.6 Mev} = 15.0 - \frac{1.6 - 1.0}{2.0 - 1.0} (15.0 - 11.2) = 12.7$$

Logarithmic interpolation gives:

$$B_r\text{-1 for 1.6 Mev} = \text{antiln} \left[\ln 15.0 - \frac{\ln 1.6}{\ln 2.0} \ln \frac{15.0}{11.2} \right] = 12.3$$

The latter is believed to be more accurate. Hence, $B_r = 12.3 = 1.0 + \underline{\underline{13.3}}$

Example 3 Shield consisting of 2.5 mfp steel followed by 5 mfp lead followed by 2.5 mfp steel. Source energy = 1.0 Mev.

Three-layer cases like this can be treated, with only small errors, as two-layer configurations, combining the first two materials into a single region. Thus,

2.5 mfp steel + 5 mfp lead $\longrightarrow$ 7.5 mfp lead

Reading Figure VII-2.

B_r-1 for 5 mfp Pb followed by 2.5 mfp Fe = 8.2

B_r-1 for 10 mfp Pb followed by 2.5 mfp Fe = 13.3

Interpolation gives:

B_r-1 for 7.5 mfp Pb followed by 2.5 mfp Fe = 10.4

Therefore,

$B_r = 10.4 + 1.0 = \underline{\underline{11.4}}$

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VII-1. B. A. Engholm, "Gamma Buildup in Heterogeneous Media," Gulf General Atomic Incorporated Report GA-8741, Sept. 16, 1968.

VII-2. H. Goldstein and J. W. Wilkins, Jr., "Calculations of the Penetration of Gamma Rays", NYO-3075, June 30, 1954.

VII-3. S. Miyasaka and A. Tsuruo, "Dose Buildup Factors of Multilayer Slabs for a Point Isotropic Source," J. Nucl. Sci. Tech. 3, 9, 393, Sept. 1966.

VII-4. W. Futtermenger Et Al., "To the Calculation of Gamma Ray Buildup Factors in Multilayered Shields," AERE-R-5773, Vol. 2, 1968.

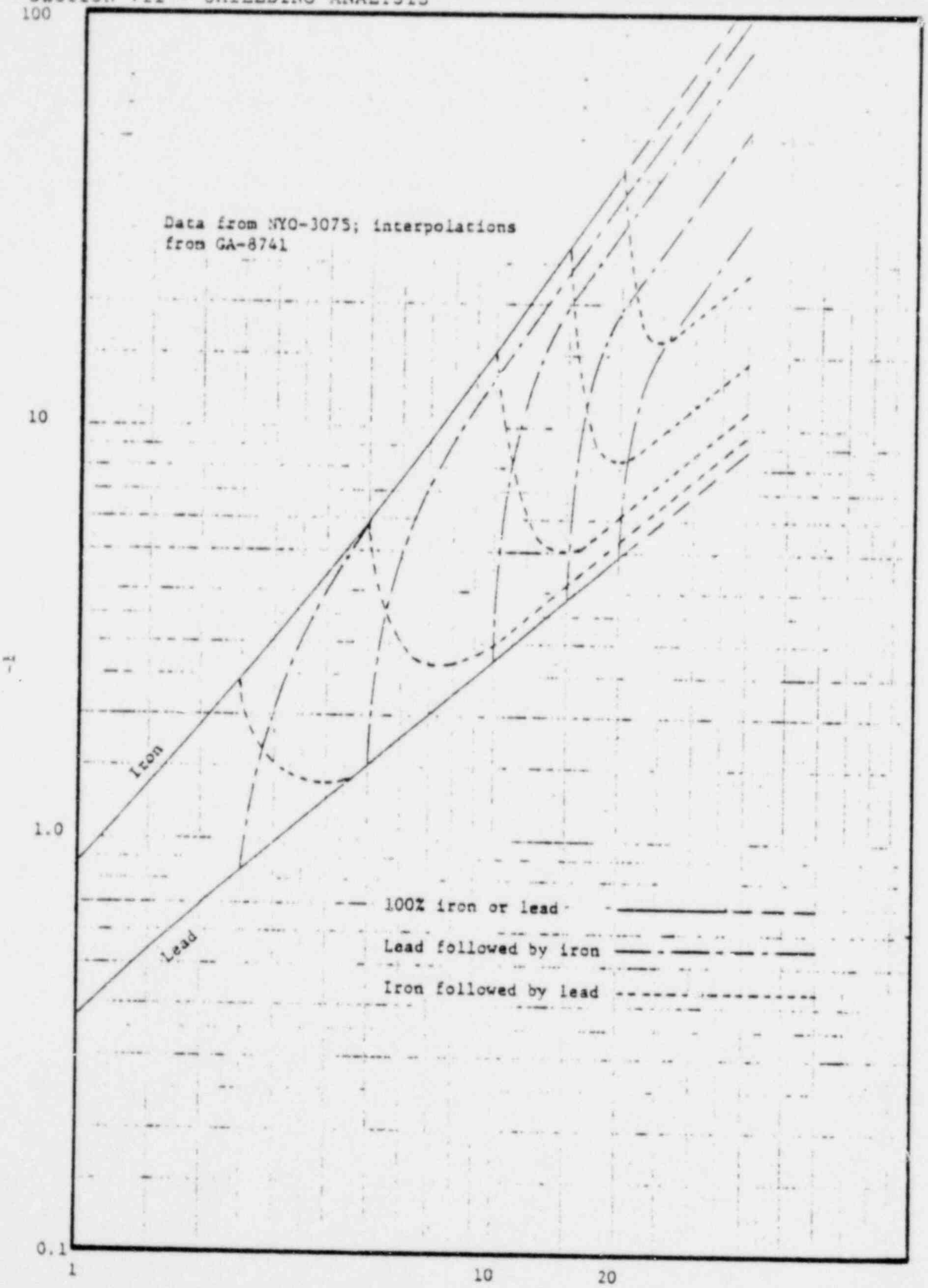

Fig. VII-2. Estimated Gamma Dose Buildup Factors in Iron-Lead Configurations (1-Mev Point Isotropic Source)

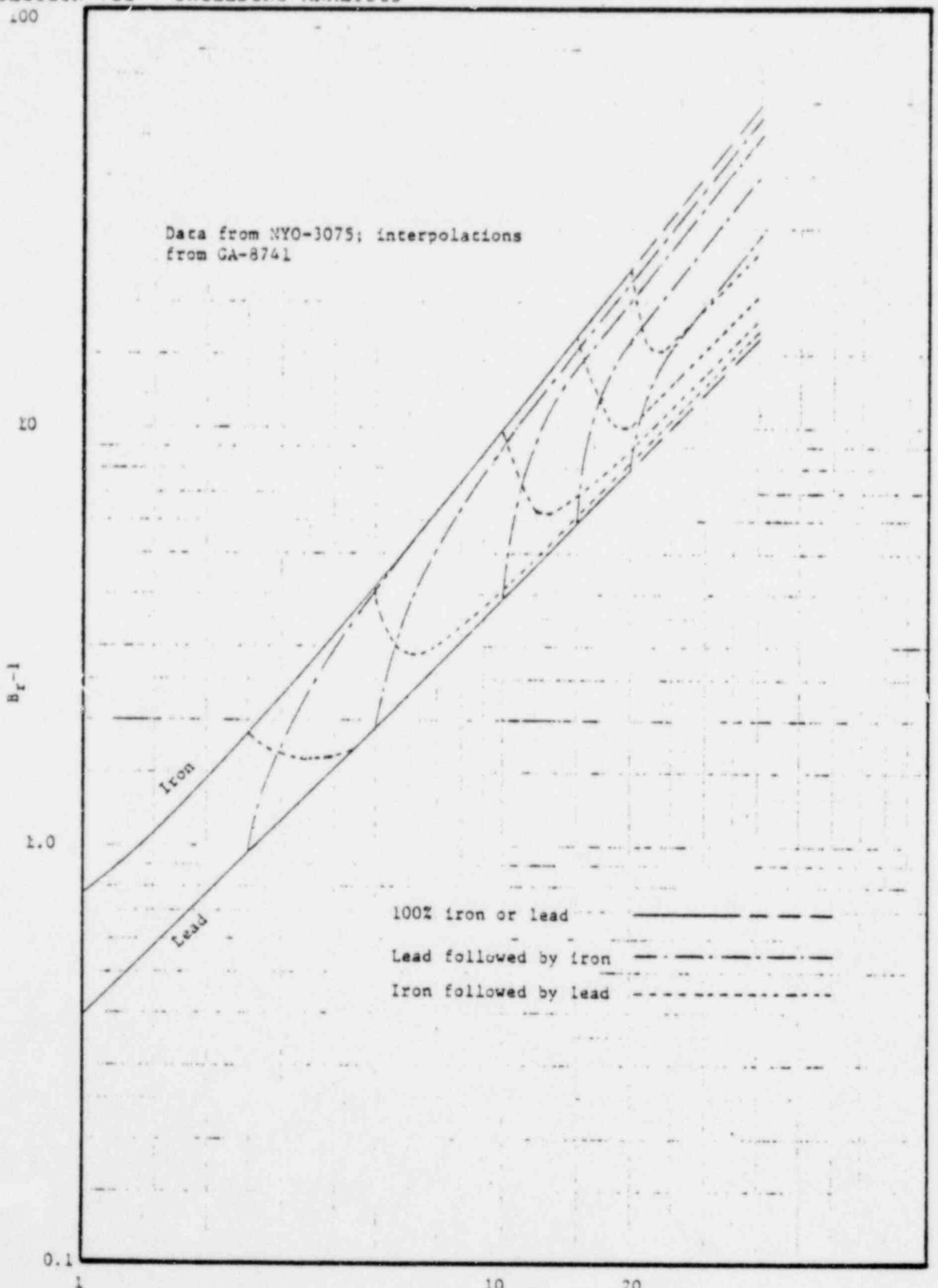

Fig. VII-3. Estimated Gamma Dose Buildup Factors in Iron-Lead Configurations (2-Mev Point Isotropic Source)

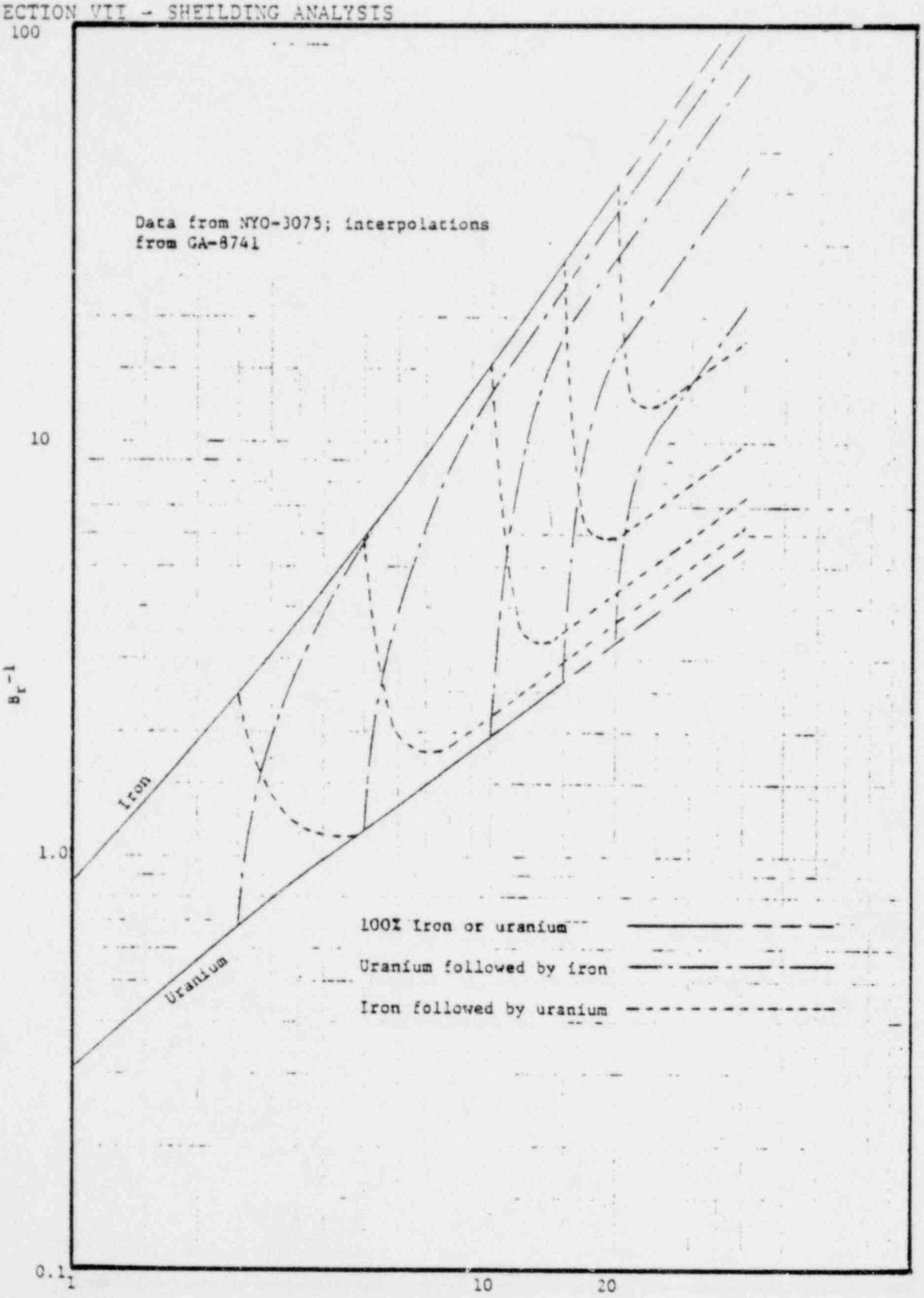

Fig. VII-4. Estimated Gamma Dose Buildup Factors in Iron-Uranium Configurations (1-Mev Point Isotropic Source)

ATTACHMENT
VII-1.

COMPUTER CODE ABSTRACTS

The PATH Gamma-Shielding Code

1. Name of Code: PATH
2. Computer for Which Program is Designed: IBM-7044 and Univac-1108
   Programming Language: FORTRAN IV
3. Nature of Problem Solved: General Atomic has had a long-standing need for a highly flexible gamma-shielding code which could treat gamma radiation from reactors, from radioactive components, and from complex piping systems. Further requirements were simplicity of input, short running times, and concise, well-organized output format. Other codes investigated[1-3] were found to be not completely applicable to shield design at General Atomic. To fulfill these requirements, we have written PATH, a FORTRAN IV gamma-shielding code.
4. Method of Solution: The PATH code is based upon exponential attenuation from source point to dose point, adjusted by appropriate buildup factors. The heart of the code is the geometry routine, which calculates the path length in each region by a direct method rather than the time-consuming "stepping" method and significantly reduces overall running times.

   The source-shield configuration is described as a set of possibly overlapping regions. Associated with each region is a serial number, a composition, and, in some cases, a number of subregions. The subregions of a region are entirely contained within that region, but do not usually fill it; and, in fact, the composition associated with a region is the composition that fills the space not occupied by any of its subregions. Regions are available in various shapes with any axial orientation: prisms, cylinders, spheres, and frustra of cones. While all regions are nominally convex, their shapes may actually be very complex because the space in a subregion is subtracted from the region to which it belongs.

   Source-region types—point, line, disk, polygon, shell, and volume in any spatial orientation—are meshed by the code according to input instructions. A cylindrical source, for instance, can be meshed into radial and angular divisions or into rectilinear divisions. Alternatively, individual source points can be entered at any location in the configuration. The source term can be described in two ways: as source strengths (MeV/sec or photons/sec) at given energy levels, and/or as isotopes with associated activities (curies). A Table of the decay spectra of 70 isotopes is stored in the program; a subroutine will calculate the growth and decay of isotopes as a function of time.

   Input has been designed to require a minimum of data, all of which is floating point or alphabetic. Dimensions can be given in centimeters or in feet and inches. For each composition the materials are named along with density and volume fraction of each. A Table in the program contains 17-energy-group mass-attenuation coefficients for 27 basic materials. Buildup factors are computed from sets of polynomial coefficients of the type originated by Capo[4]; these are also stored in the program. All these Tables can be expanded to accommodate special source spectra, new materials, or revised data. A shield-region perturbation routine facilitates parametric studies of dose rate as a function of shield thickness or composition.
5. Restrictions:

Storage Restrictions	IBM-7044	Univac-1108
a) Number of shield regions	≤ 50	≤ 100
b) Number of source regions	≤ 90	≤ 100
c) Number of source points	≤ 1200	≤ 3000
d) Number of source energy groups	≤ 40	≤ 100
e) Number of compositions	≤ 50	≤ 50
f) Number of dose points	Unlimited	Unlimited

6. Typical Running Time: A typical problem involved a radioactive piping layout consisting of 79 pipe runs, 24 valves, and 90 fittings; IBM-7044 execution time was 8.4 mh per receiver point, while Univac-1108 execution time was 2.6 mh.
7. Present Status: In use.
8. References:

[1]E. CZAPEK et al., "Gamma Shield Design for Primary Coolant Sources Using the IBM Type 650 Computer," AECU-3778, Div. of Tech. Info. Exten., USAEC (September, 1965).

[2]J. T. MARTIN et al., "Shielding Computer Programs 14-0 and 14-1, Reactor Shield Analysis," XDC-59-2-16, General Electric Co. (January 23, 1959).

[3]D. M. PETERSON, "Shield Penetration Programs C-17 and L-63," NARF-61-39T, Convair, Ft. Worth (December 29, 1961).

[4]M. A. CAPO, "Polynomial Approximation of Gamma Ray Buildup Factors for a Point Isotropic Source," APEX-510, General Electric Co. (November, 1958).

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POOR ORIGINAL

ATTACHMENT VII-1.

COMPUTER CODE ABSTRACTS

The PATH Gamma-Shielding Code

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   Source-region types—point, line, disk, polygon, shell, and volume in any spatial orientation—are meshed by the code according to input instructions. A cylindrical source, for instance, can be meshed into radial and angular divisions or into rectilinear divisions. Alternatively, individual source points can be entered at any location in the configuration. The source term can be described in two ways: as source strengths (MeV/sec or photons/sec) at given energy levels, and/or as isotopes with associated activities (curies). A Table of the decay spectra of 70 isotopes is stored in the program; a subroutine will calculate the growth and decay of isotopes as a function of time.

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Storage Restrictions	IBM-7044	Univac-1108
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d) Number of source energy groups	≤ 40	≤ 100
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SECTION VIII

HEAT TRANSFER ANALYSIS

THERMAL ANALYSIS

A. INTRODUCTION

The fuel shipping cask is designed to safely contain the irradiated fuel elements under a variety of ambient and accident conditions. This part of the report summarizes the thermal analyses which were performed to insure adequate cask protection during several extreme conditions.

B. CRITERIA FOR THE ANALYSIS

The following three conditions were used to define the ambient conditions for the analyses:

1. Maximum temperature day. - This condition assumes an ambient temperature of 130°F around all parts of the cask. Solar heating has an effective intensity of 96 Btu/hr-ft^2, excluding the effect of the personnel shield. Fuel heat generation simulating 100 and 200 days from reactor shutdown were used.
2. Minimum temperature day. - This condition is for a -40°F ambient condition with no solar heating. One hundred day fuel heating was used for comparison with Case 1.
3. Fire accident. - The fire accident assumes a 1475°F fire completely surrounding the cask. A high surface h of 300 Btu/hr-ft^2-°F was set to simulate the worst possible condition of hot, blowing gasses. This condition lasts for 30 minutes, after which the cask is returned to Condition 1, the 130°F day.

The first and third conditions are obtained from the AEC regulations 10-CFR-71, Condition (1) from Appendix A, Item 1, and Condition (3) from Appendix B Item 3.

C. SUMMARY OF RESULTS

Steady-state and transient thermal analyses were conducted on thermal models of each end of the cask, the models extending far enough down the length of the cask to eliminate any end effects. Since no unexpected results were obtained from the analysis of the relatively simple lower end, only the temperatures of the upper end are summarized in Table VIII-1.

It is noteworthy that the surface of the cask is below 212°F, although high enough to cause discomfort to personnel in contact with it. The solar heating is a significant contributor to the total cask heat and will actually be considerably less if the aluminum roof and mesh sides are utilized. It is estimated that the maximum cask surface temperature will drop negligibly due to the effect of the roof and screen, which will cut the solar heating by approximately 2/3 and the free convection h by 1/2.

TABLE VIII-1
SUMMARY OF RESULTS

		Temperatures (°F)					
		130° Ambient			-40° Ambient		Fire Acci Maximum Temps.
	Location	100 Day Heat		200 Day Heat	100 Day Heat		
No.	Gap Tolerances	Max Gaps	Min Gaps	Max Gaps	Max Gap	Min Gap	Min Gap
1	Cask Seal	141	137	133	-14	-15	1220
2	Container Seal	155	146	137	3	0	655
3	Fuel Surface - Max	278	266	200	153	143	284
4	Fuel Centerline - Max	284	272	203	160	149	291
5	Container Wall - Max	174	159	147	28	14	502
6	Inner Shell - Max	161	149	141	13	2	715
7	Shielding - Center	152	146	137	3	0	829
8	Outer Shell - Max	146	144	134	-4	-3	1397

The cask surface temperature can be assumed to be the same as the outer shell temperature (No. 8).

Figures VIII-5 and VIII-8 are the calculated temperature matrices from the computer output. Locations 1 - 8 are identified on Fig. VIII-5.

D. METHOD OF ANALYSIS

Basic temperatures were calculated by a numerical finite-difference method. A digital computer code, RAT*, was used to perform the actual calculations.

---

*RAT is an acronym for "radial-axial temperatures."

The cask system was modeled in a form suitable for input to this code. The regions modeled are shown on Figs. VIII-2 and VIII-3.

RAT is a digital computer code that is applicable to calculating transient temperatures in a two-dimensional network of points. It may also be used to obtain steady-state solutions by extending a transient calculation to the point where time dependence of results becomes negligible.

The network is specified by establishing a grid system, locating individual materials within that grid system, and identifying the applicable thermal parameters that define those materials. The grid system must be a regular one specified by two sets of grid lines parallel to the coordinate axes in one of the following three systems: orthogonal (X-Y), cylindrical (R-Z), or circular (R-Θ).

The materials are located by subdividing the grid system into blocks, or regions of adjacent points. Each block is defined by its four bounding grid lines and the material that it contains. The material is given in terms of a material numbering system. Parameters that define each material are the applicable thermal properties and the volumetric heat generation rate.

Blocks of materials may be separated by narrow gaps that contain stagnant gases. The gases are located in terms of a gas numbering system and are defined by their individual thermal conductivities. Heat transfer across these gaps is by one-dimensional conduction and radiation.

Boundary conditions at external boundaries are specified either by a sink temperature and unit surface conductance or by the thermal parameters of a flowing coolant. These parameters are the coolant properties and flow conditions.

The thermal parameters for the materials, gases, and coolants may be given in functional form. Many of the calculation variables are available for use in these functions.

E. ANALYSIS

1. Ambient Conditions

a. Maximum Temperature Day

The maximum temperature day is defined as having an air temperature of 130°F with full sun. Further, it was assumed that the cask and trailer are in still air with no convective cooling other than free convection. Using a correlation for free convection around a horizontal cylinder, the following function was used to calculate the surface heat transfer coefficient (Ref. 1):

$$h = .221\ (S_t - T_a)^{.25}$$

where: S_t = the cask surface temperature

T_a = the ambient temperature.

Thermal radiation from the cask to its surroundings was also calculated. It was assumed that the emissivity of the cask was 0.6 and that the absorbtivity of the ambient surroundings was 1.0. Thus, using standard correlations:

$$Fe = \varepsilon_1 = 0.6$$

Solar heating was imposed on the thermal model in the form of surface heat generation. Since the model is a two-dimensional, radial-axial one, there can be no circumferential variation. Thus the heat generation was imposed around the entire circumference of the cask. The net effect is for a higher than actual total heat input with conservatively high internal temperatures.

The net solar heating is 96 Btu/hr-ft^2. Past experience with shipping cask analysis has shown that the thermal response of a cask is very slow and that it is not necessary to calculate with a time dependent solar heating function. Thus, steady-state temperatures were calculated.

b. Minimum Temperature Day

The minimum temperature day was defined as an ambient condition of -40°F with no solar heating. The same free convection heat transfer coefficient as was used for the 130°F day was incorporated, as was the ability of the cask to radiate heat to its surroundings. If it were to be assumed that the truck was moving and that a relatively high air velocity existed on the cask surface, the cask surface-to-air ΔT would be nil and the cask internal temperatures would drop accordingly.

c. Fire Accident

The fire accident is defined as a surrounding ambient condition of 1475°F. As the conditions of the fire are undefined, an overall surface heat transfer coefficient (h) simulating a strong convection condition was assumed. Reasonable values of h_c for hot, blowing gasses are in the range of 100 - 300 Btu/hr-ft^2-°F. The equivalent surface heat transfer coefficient for thermal radiation (h_r) is approximately 30 Btu/hr-ft^2-°F at a median cask surface temperature. This assumes, again, that the emissivity of the fire is 1.0. In total, an h of 300 was concluded to be a maximum reasonable value for the fire accident.

2. Cask Dimensions

Manufacturing tolerances allow a rather significant variation in the radial gaps. Although it is very unlikely that the parts would ever be manufactured and assembled such that one cask had either all maximum or minimum gaps, the analyses were made using both extreme cases to illustrate the maximum possible range of temperatures that may be encountered. These gaps are illustrated on Fig. 1. The fire accident (Case 3) was calculated using only the minimum gaps in order to illustrate the worst condition for the inner part of the cask.

The significant radial dimensions shown on Fig. VIII-1 were used with appropriate coefficients of thermal expansion. Thus, the correct gaps were recalculated as the cask changed in temperature. The net effect of this is to decrease the gap sizes under normal conditions when the inner shells are hotter than the outer shells. In the fire accident, however, the hotter outer shells will expand away from the inner shells, increasing the resistance of the gap and retarding the heat flow into the inner part of the assembly.

3. Spent Fuel Heat Generation

The fuel blocks being shipped in this cask have been irradiated, and the fissionable fuel has been partially consumed. Due to residual isotope activity, there is a continuing "after-heat" which decays with time, depending on the isotope half-life. This activity is absorbed by the fuel and cask components and is realized in the form of heat. This heat generation is predictable and has been used in the calculations.

It is assumed that the spent fuel is loaded into the shipping cask no sooner than 100 days after the reactor is shut down. Thus the maximum heat generation that the cask need be designed for is obtained from fuel loaded 100 days after the reactor shutdown.

The fuel element after-heat was calculated at GGA in analyses prior to this one. The calculated heat generation recommended for thermal analysis purposes is 2322 Btu/hr per fuel block at 100 days and 1101 Btu/hr at 200 days. Of these quantities, 88% is realized within the fuel block and the remaining 12% is generated within the first inch of the surrounding shielding.

The decay heat generation is summarized in Table VIII-2.

TABLE VIII-2
CALCULATED AFTER-HEAT GENERATION
OF FUEL BLOCK (INCL. Pa^{233})
Btu/hr

Decay Time, days	Betas	Gammas	Total
100	1353.	969.	2322.
150	967.	540.	1507.
200	767	334.	1101.

4. Thermal Properties of the Materials

Table VIII-3 lists the thermal properties of the materials which constitute the cask as they were used in the analysis.

TABLE VIII-3
THERMAL PROPERTIES OF MATERIALS

Helium

$K = 1.29 \times 10^{-3} * T_{°R}^{.674}$ Btu/hr-ft-°F

$C_p = 1.242$ Btu/lb-°F

Air

$K = 0.0146 + 1.695 \times 10^{-5} * T_{°F}$ Btu/hr-ft-°F

$C_p = 0.25$ Btu/lb-°F

Stainless Steel (Type 304)

$K = 29.1 - 0.0059 * T_{°F}$

$C_p = 55.$ Btu/ft^3-°F

$\epsilon = 0.8$

$\alpha = 9.5 \times 10^{-6}$ in./in.-°F

Depleted Uranium

$K = 14.8$ Btu/hr-ft-°F

$C_p = 38.$ Btu/ft^3-°F

$\epsilon = 0.5$

$\alpha = 9.6 \times 10^{-6}$ in./in.-°F

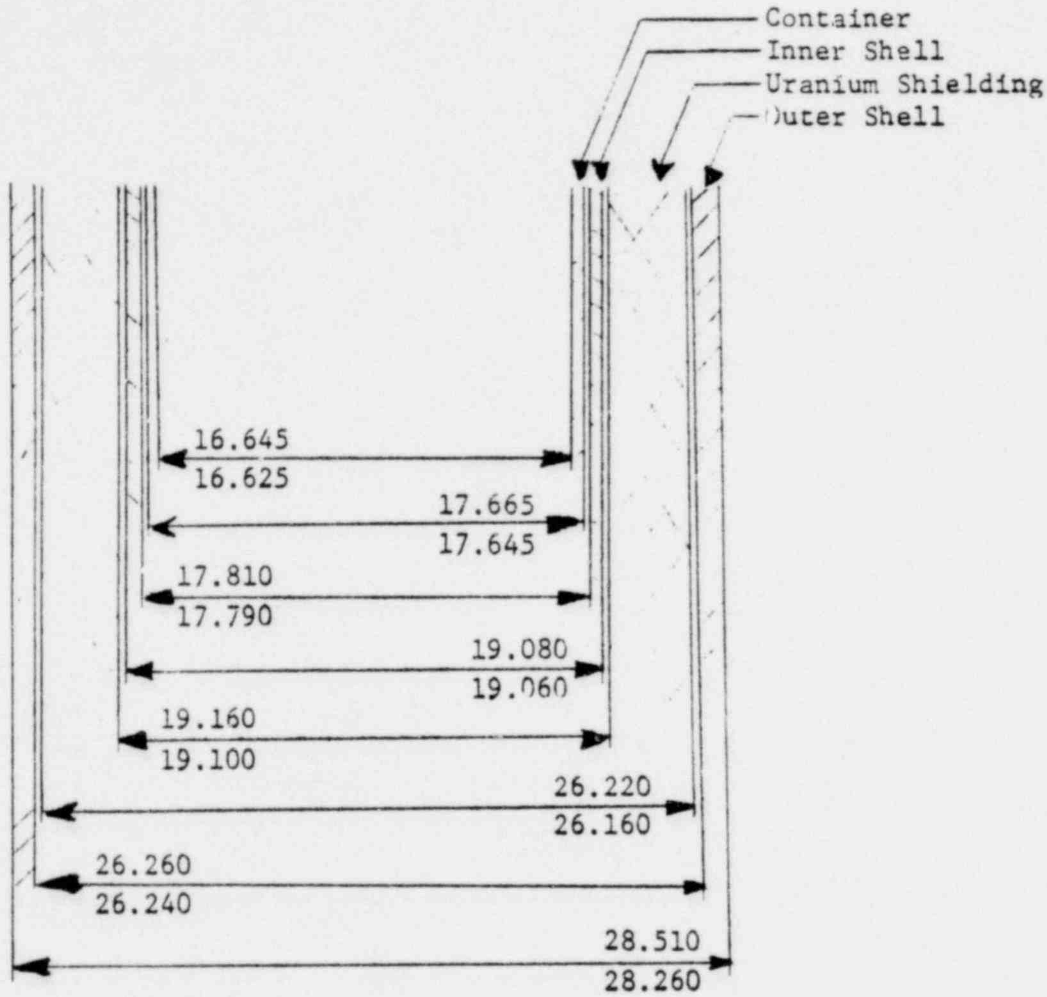
TABLE VIII-3
(Continued)

Spent Fuel Block

$K = 10.0$ Btu/hr-ft-°F

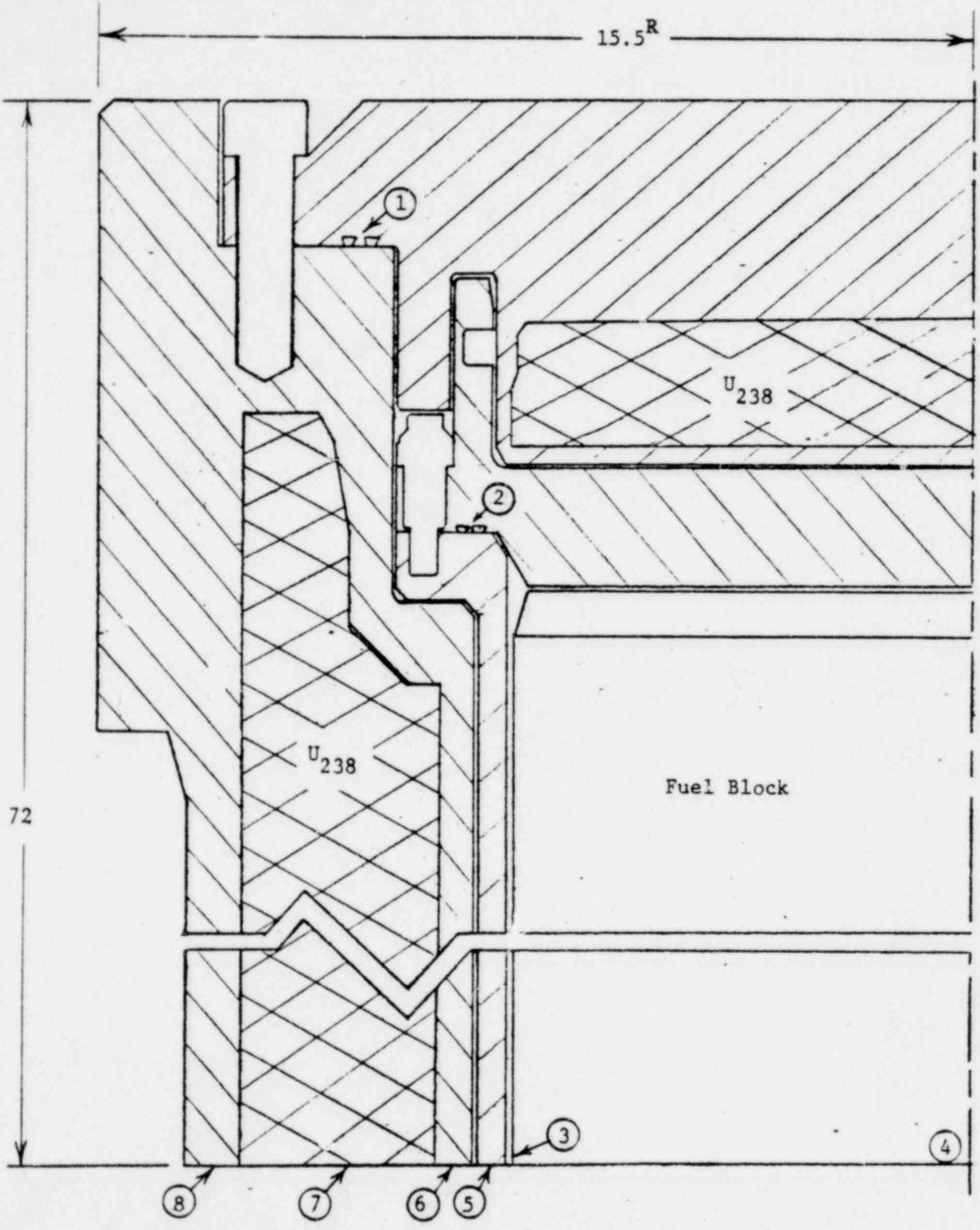
$C_p = 32.$ Btu/ft^3-hr

$\varepsilon = 0.8$



	Max	Min
Container Wall	0.520	0.500
Gap	0.0825	0.0625
Inner Shell Wall	0.645	0.625
Gap	0.050	0.010
Shielding Thickness	3.560	3.500
Gap	0.050	0.010
Outer Shell Wall	1.135	1.000

Fig. VIII-1. PSC Fuel Shipping Cask Diameters and Tolerances at Major Shells (all dimensions in inches)



Temperature locations noted in Table VIII-1 and Fig. VIII-5

Fig. VIII-2. Shipping Cask Model - Upper End

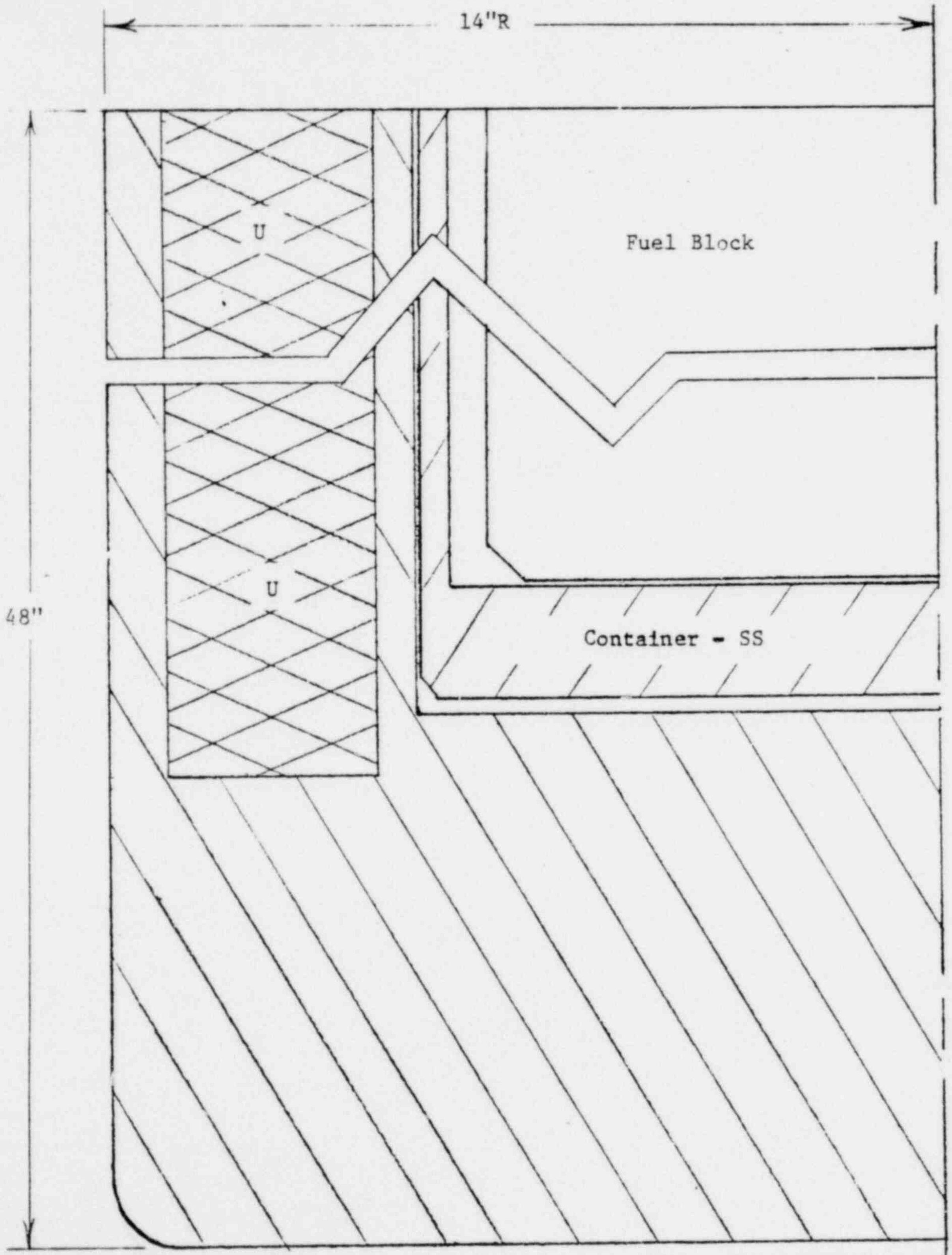


Fig. VIII-3. Shipping Cask Model - Lower End

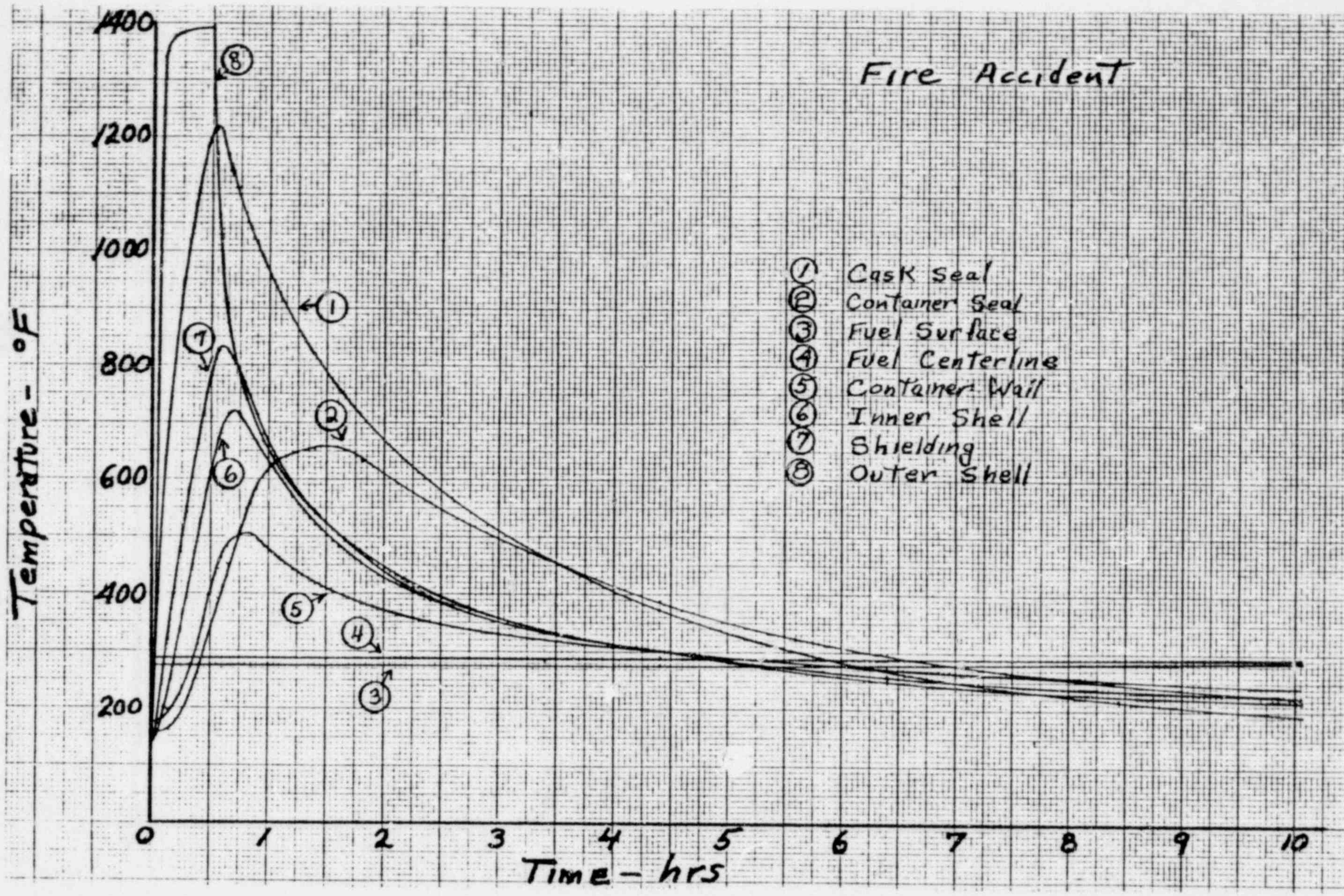


Fig. VIII-4. Fire Accident

130 °F day, 100 Day heat

Maximum Gaps - Reference Case

TEMPERATURE GRID
THE RADIAL DIRECTION IS HORIZONTAL
THE AXIAL DIRECTION IS VERTICAL
THE TEMPERATURES ARE IN DEGREES FAHRENHEIT

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25
1	0	130	130	130	130	130	130	130	130	130	130	130	130	130	130	130	130	130	130	130	130	130	130	130	130
2	0	139	139	139	139	139	138	138	138	138	138	138	138	138	138	138	138	138	139	138	138	137	137	137	137
3	0	140	140	139	139	139	139	139	139	139	139	139	139	139	138	138	138	138	138	138	138	138	137	137	137
4	0	140	140	140	140	139	139	139	139	139	139	139	139	139	139	139	138	138	138	138	138	138	138	137	137
5	0	141	141	140	140	140	140	140	140	140	139	139	139	139	139	139	138	138	139	138	138	138	138	138	138
6	0	141	141	141	141	141	141	141	141	141	141	141	141	141	140	140	138	138	138	138	138	138	138	138	138
7	0	143	142	142	142	142	142	142	142	142	141	141	142	142	140	140	139	139	139	139	138	138	138	138	138
8	0	143	143	143	143	143	143	143	143	143	142	142	142	142	140	140	139	139	139	139	139	138	138	138	139
9	0	145	145	144	144	144	144	144	144	144	143	143	143	143	141	141	139	139	139	139	139	139	138	138	138
10	0	146	146	145	145	145	145	145	145	145	144	144	144	144	141	141	140	140	140	140	139	139	138	138	138
11	0	147	146	146	146	146	146	146	146	146	145	145	145	145	144	143	140	140	140	140	140	139	138	138	138
12	0	148	148	147	147	147	147	147	147	147	146	146	146	146	146	146	142	142	142	140	140	140	139	139	139
13	0	149	149	148	148	148	147	147	147	147	147	147	147	147	147	147	143	143	143	140	140	140	139	139	139
14	0	150	150	149	149	148	148	148	148	148	148	148	148	147	147	147	144	144	143	141	141	140	139	139	139
15	0	151	151	150	150	149	149	149	149	148	148	148	148	148	148	148	144	144	141	141	141	141	140	139	139
16	0	152	152	151	151	150	150	150	145	149	149	148	148	148	148	148	145	145	142	142	141	141	140	140	140
17	0	231	230	229	227	226	205	175	156	150	149	149	149	149	149	148	146	146	142	142	142	141	140	140	140
18	0	232	231	230	228	227	209	180	158	150	150	149	147	147	147	147	146	146	142	142	142	142	140	140	140
19	0	234	233	231	230	229	213	187	170	158	151	149	147	147	147	147	146	146	143	143	142	142	141	140	140
20	0	235	234	233	231	230	215	189	172	159	153	150	147	147	147	147	147	146	143	143	143	142	141	141	141
21	0	236	235	234	232	231	216	190	173	160	153	150	147	147	147	147	147	144	143	143	143	143	141	141	141
22	0	238	238	236	234	233	218	192	174	161	155	151	148	148	148	147	144	144	144	144	143	143	141	141	141
23	0	242	241	240	238	237	222	195	177	164	157	153	149	149	149	145	145	145	144	144	144	144	141	141	141
24	0	247	246	245	243	242	226	198	180	166	159	155	150	150	150	146	146	145	145	145	145	145	142	142	142
25	0	252	252	250	248	247	231	202	183	169	161	157	152	152	152	147	146	146	146	146	146	145	142	142	142
26	0	258	257	255	253	252	235	206	186	171	164	158	153	153	153	148	147	147	147	147	146	146	142	142	142
27	0	262	261	260	258	257	239	209	188	173	165	160	154	154	154	148	148	148	147	147	147	147	142	142	142
28	0	265	264	263	261	260	242	211	190	174	166	161	155	155	155	149	148	148	148	148	147	147	142	142	142
29	0	270	269	267	265	264	246	214	192	176	168	162	156	156	156	149	149	149	148	148	148	147	143	142	142
30	0	275	274	272	270	268	249	216	194	178	169	163	157	157	157	150	149	149	149	149	148	148	143	143	143
31	0	278	277	275	273	271	252	218	196	179	170	164	158	158	157	150	150	150	149	149	149	148	143	143	143
32	0	280	279	277	275	273	254	220	197	180	171	164	158	158	158	151	150	150	150	149	149	148	143	143	143
33	0	281	280	278	276	275	255	221	197	180	171	165	158	158	158	151	150	150	150	150	149	149	143	143	143
34	0	282	281	279	276	275	255	221	197	180	171	165	159	158	158	151	150	150	150	150	149	149	143	143	143
35	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

U238 · Fuel Block · Container · Inner Shell · U238 · Outer Shell

1 2 3 4 5 6 7 8

Fig. VIII-5. PSC Fuel Shipping Cask Thermal Analysis

130 °F Day   100 Day Heat   Minimum Gaps

TEMPERATURE GRID
THE RADIAL DIRECTION IS HORIZONTAL
THE AXIAL DIRECTION IS VERTICAL
THE TEMPERATURES ARE IN DEGREES FAHRENHEIT

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25
1	0	130	130	130	130	130	130	130	130	130	130	130	130	130	130	130	130	130	130	130	130	130	130	130	130
2	0	138	138	138	138	138	138	138	138	138	137	137	137	137	137	137	137	137	137	137	137	137	136	136	136
3	0	139	139	138	138	138	138	138	138	138	138	138	138	138	138	137	137	137	137	137	137	137	137	137	137
4	0	139	139	139	139	139	138	138	138	138	138	138	138	138	138	138	138	137	137	137	137	137	137	137	137
5	0	140	140	139	139	139	139	139	139	139	139	138	138	138	138	138	138	137	137	137	137	137	137	137	137
6	0	140	140	140	140	140	140	140	140	140	140	140	140	140	139	139	138	138	138	138	137	137	137	137	137
7	0	141	141	141	141	141	141	141	141	141	140	140	140	140	139	139	138	138	138	138	138	137	137	137	137
8	0	142	142	142	142	142	141	141	141	141	141	141	141	141	139	139	138	138	138	138	138	138	137	137	137
9	0	143	143	143	143	143	143	143	143	143	142	142	142	142	140	140	138	138	139	138	138	138	138	138	138
10	0	144	144	144	144	144	144	144	144	144	143	143	143	143	140	140	139	139	139	139	138	138	138	138	138
11	0	145	145	145	144	144	144	144	144	144	144	144	144	144	143	141	139	139	139	139	139	138	138	138	138
12	0	146	146	146	145	145	145	145	145	145	144	144	144	144	144	144	140	140	140	139	139	139	138	138	138
13	0	147	147	147	146	146	146	146	146	146	145	145	145	145	145	145	141	141	141	139	139	139	138	138	138
14	0	148	148	148	147	147	146	146	146	146	146	146	146	146	146	145	142	142	142	140	140	139	139	139	139
15	0	149	149	148	148	148	147	147	147	146	146	146	146	146	146	146	143	143	140	140	140	140	139	139	139
16	0	150	150	149	149	149	148	148	146	146	146	146	146	146	146	146	143	143	141	141	140	140	140	139	139
17	0	226	225	224	222	221	201	172	153	147	147	147	147	147	146	146	144	144	141	141	141	141	140	140	140
18	0	227	227	225	224	222	206	176	155	148	148	146	145	145	145	145	144	144	141	141	141	141	140	140	140
19	0	229	228	227	225	224	209	183	166	155	148	147	145	145	145	145	144	144	142	141	141	141	140	140	140
20	0	230	230	228	226	225	211	185	168	156	149	147	145	145	145	145	144	144	142	142	142	141	141	140	140
21	0	231	231	229	228	227	212	186	169	156	150	147	145	145	145	145	145	142	142	142	142	141	141	141	141
22	0	233	233	231	229	229	213	187	170	157	150	148	145	145	145	145	143	143	142	142	142	142	141	141	141
23	0	237	236	234	233	232	216	190	172	159	152	148	145	145	145	144	144	143	143	143	143	142	142	142	142
24	0	241	241	239	237	236	220	192	174	160	153	149	146	146	146	144	144	144	144	144	143	143	142	142	142
25	0	247	246	244	242	241	226	196	176	162	154	150	147	147	146	145	145	145	144	144	144	144	143	142	142
26	0	251	250	249	247	246	229	198	178	163	155	151	147	147	147	146	145	145	145	145	144	144	143	143	143
27	0	255	254	253	251	250	232	201	180	164	156	152	148	148	147	146	145	145	145	145	145	144	143	143	143
28	0	258	257	256	254	252	234	202	181	165	157	152	148	148	148	146	146	145	145	145	145	144	143	143	143
29	0	262	261	259	257	256	237	204	182	166	157	153	148	148	148	146	146	146	145	145	145	144	143	143	143
30	0	266	265	263	261	260	241	207	184	167	158	153	149	149	149	147	146	146	146	146	145	145	143	143	143
31	0	269	268	266	264	263	243	208	185	168	159	154	149	149	149	147	147	146	146	146	146	145	144	143	143
32	0	271	270	268	266	264	244	209	186	168	159	154	149	149	149	147	147	146	146	146	146	145	144	143	143
33	0	272	271	269	267	265	245	210	186	169	159	154	149	149	149	147	147	147	146	146	146	145	144	144	143
34	0	272	271	269	267	266	246	210	186	169	159	154	150	149	149	147	147	147	146	146	146	145	144	144	144
35	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

Fig. VIII-6. PSC Fuel Shipping Cask Thermal Analysis

-40 °F Day 100 Day Heat Maximum Gaps

TEMPERATURE GRID
THE RADIAL DIRECTION IS HORIZONTAL
THE AXIAL DIRECTION IS VERTICAL
THE TEMPERATURES ARE IN DEGREES FAHRENHEIT

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25
1	0	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39
2	0	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14	-15	-15	-15
3	0	[illegible]	-13	-13	-13	-13	-13	-13	-13	-13	-13	-13	-13	-13	-13	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14
4	0	[illegible]	-13	-13	-13	-13	-13	-13	-13	-13	-13	-13	-13	-13	-13	-13	-13	-13	-13	-13	-13	-13	-14	-14	-14
5	0	[illegible]	-13	-13	-13	-12	-12	-12	-12	-12	-12	-12	-12	-12	-12	-12	-13	-13	-13	-13	-13	-13	-13	-13	-13
6	0	[illegible]	-12	-12	-12	-12	-12	-12	-12	-12	-10	-10	-10	-10	-11	-11	-12	-12	-12	-12	-12	-13	-13	-13	-13
7	0	-12	-12	-12	-11	-11	-11	-10	-10	-10	-10	-10	-10	-10	-11	-11	-12	-12	-12	-12	-12	-12	-12	-12	-12
8	0	-12	-12	-12	-11	-10	-10	-10	-9	-9	-5	-9	-9	-9	-10	-10	-11	-11	-11	-12	-12	-12	-12	-12	-12
9	0	[illegible]	-6	-8	-7	-7	-6	-6	-5	-5	-5	-5	-5	-5	-9	-10	-11	-11	-11	-11	-11	-11	-12	-12	-12
10	0	-3	-3	-3	-3	-3	-3	-3	-3	-3	-3	-3	-3	-3	-9	-9	-9	-10	-10	-10	-10	-11	-11	-11	-11
11	0	[illegible]	-0	-0	-0	-1	-1	-1	-1	-1	-1	-1	-1	-1	-3	-5	-8	-9	-9	-9	-9	-10	-11	-11	-11
12	0	5	5	4	3	2	1	1	1	0	0	0	0	0	0	1	-6	-6	-7	-8	-9	-9	-11	-11	-11
13	0	[illegible]	[illegible]	7	5	4	4	3	3	3	3	2	2	2	2	2	-4	-5	-5	-8	-8	-8	-10	-10	-10
14	0	[illegible]	8	7	6	5	4	4	4	4	3	3	3	3	3	2	-3	-3	-3	-7	-7	-7	-10	-10	-10
15	0	9	8	8	7	6	6	6	6	4	4	4	4	3	3	3	-2	-2	-6	-6	-6	-6	-9	-9	-9
16	0	[illegible]	9	8	7	7	7	7	4	4	4	4	4	4	4	4	-1	-1	-5	-5	-5	-6	-9	-9	-9
17	0	104	103	101	99	97	49	6	5	6	5	5	5	4	4	4	-0	-0	-4	-5	-5	-5	-8	-8	-9
18	0	[illegible]	104	102	100	98	56	6	6	6	6	3	1	1	1	0	0	-0	-4	-4	-4	-5	-8	-8	-8
19	0	107	106	104	102	100	73	44	25	13	7	4	1	1	1	0	0	0	-4	-4	-4	-4	-8	-8	-8
20	0	[illegible]	107	106	103	102	83	51	30	16	6	4	1	1	1	0	0	0	-3	-4	-4	-4	-8	-8	-8
21	0	[illegible]	108	107	105	104	85	53	32	17	9	5	1	1	1	1	0	-3	-3	-3	-3	-4	-7	-7	-7
22	0	111	111	109	107	106	88	55	34	18	10	6	1	1	1	1	-2	-3	-3	-3	-3	-3	-7	-7	-7
23	0	115	114	113	111	110	91	58	36	20	12	7	2	2	2	-1	-2	-2	-2	-2	-2	-3	-6	-6	-6
24	0	[illegible]	120	118	116	115	96	62	40	23	15	9	4	4	4	-1	-1	-1	-1	-1	-2	-2	-5	-5	-5
25	0	[illegible]	126	124	122	121	101	66	43	26	17	11	6	6	5	-0	-0	-0	-0	-0	-1	-1	-5	-5	-5
26	0	132	131	130	128	127	106	70	46	29	19	13	7	7	7	0	0	0	-0	-0	-0	-0	-5	-5	-5
27	0	137	136	135	133	132	111	74	49	31	21	14	8	8	8	1	1	0	0	0	0	-0	-5	-5	-5
28	0	[illegible]	140	138	136	135	114	76	50	32	22	15	9	9	9	1	1	1	1	0	0	0	-5	-5	-5
29	0	[illegible]	145	143	141	140	118	79	53	34	24	17	10	10	10	2	2	1	1	1	1	0	-5	-5	-5
30	0	151	150	149	146	145	123	83	56	36	25	18	11	11	11	3	2	2	2	2	1	1	-4	-4	-5
31	0	155	154	152	150	149	126	85	57	37	27	19	12	12	12	3	3	3	2	2	2	1	-4	-4	-4
32	0	158	157	155	153	151	128	87	59	38	27	20	12	12	12	4	3	3	3	2	2	2	-4	-4	-4
33	0	159	158	156	154	153	129	88	59	39	[illegible]	20	13	13	13	4	3	3	3	3	2	2	-4	-4	-4
34	0	[illegible]	159	157	155	154	130	88	60	39	[illegible]	20	13	13	13	4	3	3	3	3	2	2	-4	-4	-4
35	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

Fig. VIII-7. PSC Fuel Shipping Cask Thermal Analysis

−40 °F day − 100 Day heat Minimum Gaps

TEMPERATURE GRID
THE RADIAL DIRECTION IS HORIZONTAL
THE AXIAL DIRECTION IS VERTICAL
THE TEMPERATURES ARE IN DEGREES FAHRENHEIT

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25
1	0	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39
2	0	-15	-15	-15	-15	-15	-15	-15	-15	-15	-15	-15	-15	-15	-15	-15	-15	-15	-15	-15	-15	-16	-16	-16	-16
3	0	-15	-15	-15	-15	-15	-15	-15	-15	-15	-15	-15	-15	-15	-15	-15	-15	-15	-15	-15	-15	-15	-15	-15	-15
4	0	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14	-15	-15	-15	-15	-15	-15
5	0	-14	-14	-14	-14	-14	-14	-16	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14	-14
6	0	[illegible]	[illegible]	[illegible]	-13	-13	[illegible]	-13	-13	-13	-12	-12	-12	-12	-13	-13	-13	-14	-14	-14	-14	-14	-14	-14	-14
7	0	-14	-14	-13	-13	-12	-12	-12	-12	-12	-11	-11	-11	-11	-12	-12	-13	-13	-13	-13	-13	-13	-13	-14	-14
8	0	-14	-13	-13	-12	-12	-11	-11	-11	-11	-10	-10	-10	-10	-12	-12	-13	-13	-13	-13	-13	-13	-13	-13	-13
9	0	-10	-10	-10	-9	-9	[illegible]	-8	-7	-7	-7	-7	-7	-7	-11	-11	-12	-12	-12	-12	-12	-12	-13	-13	-13
10	0	-5	-5	[illegible]	-6	-5	[illegible]	-5	-5	-5	-5	-5	-5	-5	-11	-11	-11	-11	-11	-11	-12	-12	-12	-12	-12
11	0	[illegible]	-2	-2	-3	-3	-3	-4	-4	-4	-4	-4	-4	-4	-5	-7	-10	-10	-10	-10	-11	-11	-12	-12	-12
12	0	[illegible]	2	1	[illegible]	-0	-0	-1	-1	-1	-2	-2	-2	-1	-1	-1	-8	-8	-8	-10	-10	-11	-11	-11	-11
13	0	5	4	3	2	1	1	0	0	0	0	-0	-0	-0	-0	-0	-7	-7	-7	-10	-10	-10	-11	-11	-11
14	0	5	5	4	3	2	1	1	1	1	0	0	0	0	0	-0	-6	-6	-6	-9	-9	-9	-10	-10	-10
15	0	6	5	4	3	3	3	2	2	0	0	0	0	0	0	0	-5	-5	-8	-8	-8	-9	-9	-10	-10
16	0	[illegible]	6	5	4	4	[illegible]	[illegible]	1	1	1	1	1	1	0	0	-4	-4	-7	-7	-8	-8	-9	-9	-9
17	0	96	97	95	93	91	44	2	2	2	1	1	1	1	1	0	-3	-3	-7	-7	-7	-7	-8	-8	-8
18	0	99	96	96	94	92	51	2	2	2	2	0	-2	-2	-2	-2	-3	-3	-6	-7	-7	-7	-8	-8	-8
19	0	101	[illegible]	[illegible]	[illegible]	94	73	39	21	9	3	0	-2	-2	-2	-2	-2	-3	-6	-6	-6	-7	-8	-8	-8
20	0	102	101	99	97	96	77	45	25	11	3	0	-2	-2	-2	-2	-2	-2	-6	-6	-6	-6	-7	-7	-7
21	0	103	102	100	93	97	75	48	27	12	4	0	-2	-2	-2	-2	-2	-5	-5	-5	-6	-6	-7	-7	-7
22	0	105	104	103	101	100	[illegible]	49	28	13	4	1	-2	-2	-2	-2	-5	-5	-5	-5	-5	-6	-6	-7	-7
23	0	108	103	105	104	105	[illegible]	52	30	14	5	2	-1	-1	-2	-3	-4	-4	-4	-4	-4	-5	-6	-6	-6
24	0	114	113	111	109	108	[illegible]	55	32	16	7	2	-1	-1	-1	-2	-3	-3	-3	-3	-3	-4	-5	-5	-5
25	0	119	116	117	115	114	97	58	35	17	8	4	-0	-0	-0	-2	-2	-2	-2	-3	-3	-3	-4	-4	-4
26	0	124	124	122	120	119	98	61	37	19	9	4	0	0	-0	-1	-2	-2	-2	-2	-2	-3	-4	-4	-4
27	0	129	126	125	125	123	102	64	39	20	10	5	0	0	0	-1	-1	-1	-2	-2	-2	-3	-4	-4	-4
28	0	132	131	130	125	126	105	66	40	21	11	6	1	0	0	-1	-1	-1	-1	-2	-2	-2	-4	-4	-4
29	0	137	136	134	132	131	102	69	42	22	12	6	1	1	1	-0	-1	-1	-1	-1	-2	-2	-4	-4	-4
30	0	142	141	139	137	136	112	71	44	24	13	7	1	1	1	-0	-0	-0	-1	-1	-1	-2	-3	-4	-4
31	0	145	144	142	140	139	115	73	45	24	13	8	2	2	2	0	-0	-0	-0	-1	-1	-2	-3	-3	-3
32	0	147	146	144	142	141	117	75	46	25	14	8	2	2	2	0	-0	-0	-0	-0	-1	-1	-3	-3	-3
33	0	149	148	146	144	142	118	75	46	25	14	8	2	2	2	0	-0	-0	-0	-0	-1	-1	-3	-3	-3
34	0	149	146	146	144	143	119	76	47	26	14	8	2	2	2	0	0	-0	-0	-0	-0	-1	-3	-3	-3
35	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0

Fig. VIII-8. PSC Fuel Shiping Cask Thermal Analysis

RAT-S RESULTS.

RESULTS

COOLANT TEMPERATURES

	INLET	OUTLET	FLOW (LB/HR)
INNER RADIAL	0	0	0
OUTER RADIAL	130	130	*****
UPPER AXIAL	0	0	0
LOWER AXIAL	0	0	0

THE CURRENT TIME IS 2.0086 HR. OR 120.5167 MIN. OR ********* SEC.

Lower End
130 °F Day
Maximum Gaps

TEMPERATURE GRID
THE RADIAL DIRECTION IS HORIZONTAL
THE AXIAL DIRECTION IS VERTICAL
THE TEMPERATURES ARE IN DEGREES FAHRENH

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	1
1	0	130	130	130	130	130	130	130	130	130	130	0			
2	0	144	144	144	143	143	143	143	143	142	142	130			
3	0	144	144	144	144	143	143	143	143	142	142	130			
4	0	145	145	144	144	144	144	144	143	142	142	130			
5	0	145	145	145	145	145	145	144	143	142	142	130			
6	0	146	146	146	146	146	146	146	145	142	142	130			
7	0	172	171	170	168	167	166	150	146	142	142	130			
8	0	240	239	238	236	198	164	153	147	143	142	130			
9	0	249	248	246	245	203	164	155	148	143	143	130			
10	0	259	258	256	254	209	166	156	149	144	144	130			
11	0	267	266	264	262	214	169	158	151	146	145	130			
12	0	272	271	270	267	217	171	160	152	147	146	130			
13	0	276	275	273	270	219	172	161	153	147	147	130			
14	0	277	276	274	272	220	172	161	153	148	147	130			
15	0	0	0	0	0	0	0	0	0	0	0	0			

Fig. VIII-9. PSC Fuel Shipping Cask Thermal Analysis

RAT-S RESULTS.

RESULTS

COOLANT TEMPERATURES

	INLET	OUTLET	FLOW (LB/HR)
INNER RADIAL	0	0	0
OUTER RADIAL	130	130	*****
UPPER AXIAL	0	0	0
LOWER AXIAL	0	0	0

THE CURRENT TIME IS 2.0067 HR. OR 120.4000 MIN. OR ********* SEC.

TEMPERATURE GRID
THE RADIAL DIRECTION IS HORIZONTAL
THE AXIAL DIRECTION IS VERTICAL
THE TEMPERATURES ARE IN DEGREES FAHREN

Lower end
130°F Day
Minimum Gaps

	1	2	3	4	5	6	7	8	9	10	11	12	13	14
1	0	130	130	130	130	130	130	130	130	130	130	0		
2	0	142	142	142	142	142	141	141	141	140	140	130		
3	0	142	142	142	142	142	142	142	141	141	140	130		
4	0	143	143	143	142	142	142	142	141	141	141	130		
5	0	143	143	143	143	143	143	142	142	141	141	130		
6	0	144	144	144	143	143	144	144	143	142	141	130		
7	0	166	165	164	162	160	159	146	144	142	142	130		
8	0	233	232	231	229	191	157	147	144	143	142	130		
9	0	241	241	239	237	195	156	148	146	144	143	130		
10	0	251	250	249	247	200	157	150	147	145	145	130		
11	0	259	258	256	254	205	159	151	148	146	146	130		
12	0	264	263	262	259	208	161	152	149	147	147	130		
13	0	267	266	265	262	210	162	153	150	148	148	130		
14	0	269	268	266	264	211	162	153	150	148	148	130		
15	0	0	0	0	0	0	0	0	0	0	0	0		

Fig. VIII-10. PSC Fuel Shipping Cask Thermal Analysis

[illegible] RESULTS.

RESULTS

COOLANT TEMPERATURES

	INLET	OUTLET	FLOW (LB/HR)
INNER RADIAL	0	1	
OUTER RADIAL	-39	-39	*****
UPPER AXIAL	0	0	
LOWER AXIAL	0	0	1

THE CURRENT TIME IS 2.[illegible]067 HR. OR 12[illegible].4000 MIN. OR ********* SEC. 2

Lower End
-40°F Day
Maximum Gaps

TEMPERATURE GRID
THE RADIAL DIRECTION IS HORIZONTAL
THE AXIAL DIRECTION IS VERTICAL
THE TEMPERATURES ARE IN DEGREES FAHRENHE

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	0	-39	-39	-39	-39	-39	-39	-39	-39	-39	-39	0			
2	0	-1	-1	-1	-2	-2	-2	-2	-2	-3	-3	-39			
3	0	-1	-1	-1	-1	-1	-2	-2	-2	-3	-3	-39			
4	0	-0	-0	-0	-1	-1	-1	-1	-2	-2	-3	-39			
5	0	0	-0	-0	-0	-0	-0	-0	-1	-2	-2	-39			
6	0	0	0	0	0	0	1	1	1	-2	-2	-39			
7	0	33	32	31	29	27	26	6	1	-2	-2	-39			
8	0	119	[illegible]	117	115	67	24	9	3	-2	-2	-39			
9	0	[illegible]	127	[illegible]	124	73	24	12	4	-1	-1	-39			
10	0	[illegible]	[illegible]	[illegible]	134	79	27	14	6	[illegible]	0	-39			
11	0	147	[illegible]	[illegible]	[illegible]	84	29	17	[illegible]	[illegible]	2	-39			
12	0	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	32	19	10	3	3	-39			
13	0	157	[illegible]	[illegible]	152	[illegible]	33	[illegible]	11	4	4	-39			
14	0	159	[illegible]	[illegible]	[illegible]	92	34	[illegible]	11	5	4	-39			
15	0	0	0	0	0	0	0	0	0	0	0	0			

Fig. VIII-11. PSC Fuel Shipping Cask Thermal Analysis

F. RAT - PROGRAM ABSTRACTS

1. RAT is a digital computer program for calculating transient and steady-state temperature distributions in two dimensions. The configurations of the bodies must be described by block boundaries and grid lines in either a rectangular, a cylindrical, or a polar coordinate system. Material properties may differ among blocks. Coolant streams are accommodated at external boundaries only.

Finite-difference heat transport equations, which may be nonlinear, are formulated for each cell defined by the grid lines and the heat paths between adjacent cells. The system of these equations is solved by an alternating direction method that has been found to be stable and efficient for most practical problems.

Some useful features of RAT are:

a. Direct FORTRAN IV input of material and coolant properties in functional form

b. Simple geometrical input

c. Thermal radiation across internal gaps

d. Anisotropic (bi-directional) material properties are permitted.

One small subroutine is written directly in machine language. Therefore, special attention is required in converting the code for use on different machines.

SECTION VIII - HEAT TRANSFER ANALYSIS

2. RAT calculates transient and steady-state temperatures in two-dimensional problems by the finite difference method. The system to be analyzed may be bounded by flowing coolants and may contain internal gaps. There may be radiation across these gaps. Material and coolant thermal parameters may be functions of many different calculation variables, such as time and local temperature.

Restrictions: (1) The grid lines system must be orthogonal in the rectangular, cylindrical, or polar coordinate system. (2) All radiation is treated one-dimensionally. (3) The size of a problem is governed by the following maximum values:

Radial Points	Axial Points	Radial Block Boundaries	Axial Block Boundaries	Blocks	Materials
35	35	23	23	110	15

Size:	Date:	Author:	Custodian:
47K	10/64	M. Troost	J. F. Petersen

Method: Overall effective values, which may include convection and radiation effects, are determined for the conductances between calculation points. The transient heat balance equations are then solved for the temperatures at the points using the method of Ref. 2.

Remarks: There are four versions of RAT. These versions differ from one another only in their dimensions. The program described here is the standard version. The other three versions have dimensions which give the following maximum values governing problem size:

Version	Radial Points	Axial Points	Radial Block Boundaries	Axial Block Boundaries	Blocks	Materials
A	25	50	17	28	110	15
B	27	40	25	32	110	15
C*	11	136	6	36	110	15

*This version has limited storage for material and coolant property functions.

G. EXAMPLE APPLICATION OF THE RAT CODE

The computer program RAT was developed at GGA and is relatively unknown outside of the company. In order to illustrate the validity of this program as a thermal analysis tool, a text book transient problem was used as a standard of comparison. The text and problem are noted in Ref. 3. A summary of the results obtained by duplicating the problem with RAT is given on Fig. VIII-12 along with the text results. It is noted that the computed results correlate closely with the text data.

The thermal model for the study is shown on Fig. VIII-13 with the dimensional input data and samples of the computed results at selected times on the subsequent figures.

REFERENCES

1. Aerospace Applied Thermodynamics Manual, SAE.
2. Peaceman, D. W., and Rachford, H. H., "The Numerical Solution of Parabolic and Elliptic Differential Equations," I. Soc. Indust. Appl. Math. 3 (1), 28 (1955).
3. Schneider, P. J. Conduction Heat Transfer, Addison-Wesley Company, Inc., Reading Mass., p. 236.

GGA DOCUMENTATION

1. Petersen, J. F., "Conversion of RAT, TAC, and RAT3D to Fortran V," ARD: 12:69, March 10, 1969.
2. Ludwig, D. L., "A User's Manual for the RAT Heat Transfer Code," Gulf General Atomic Informal Report GAMD-8360, November 9, 1967.

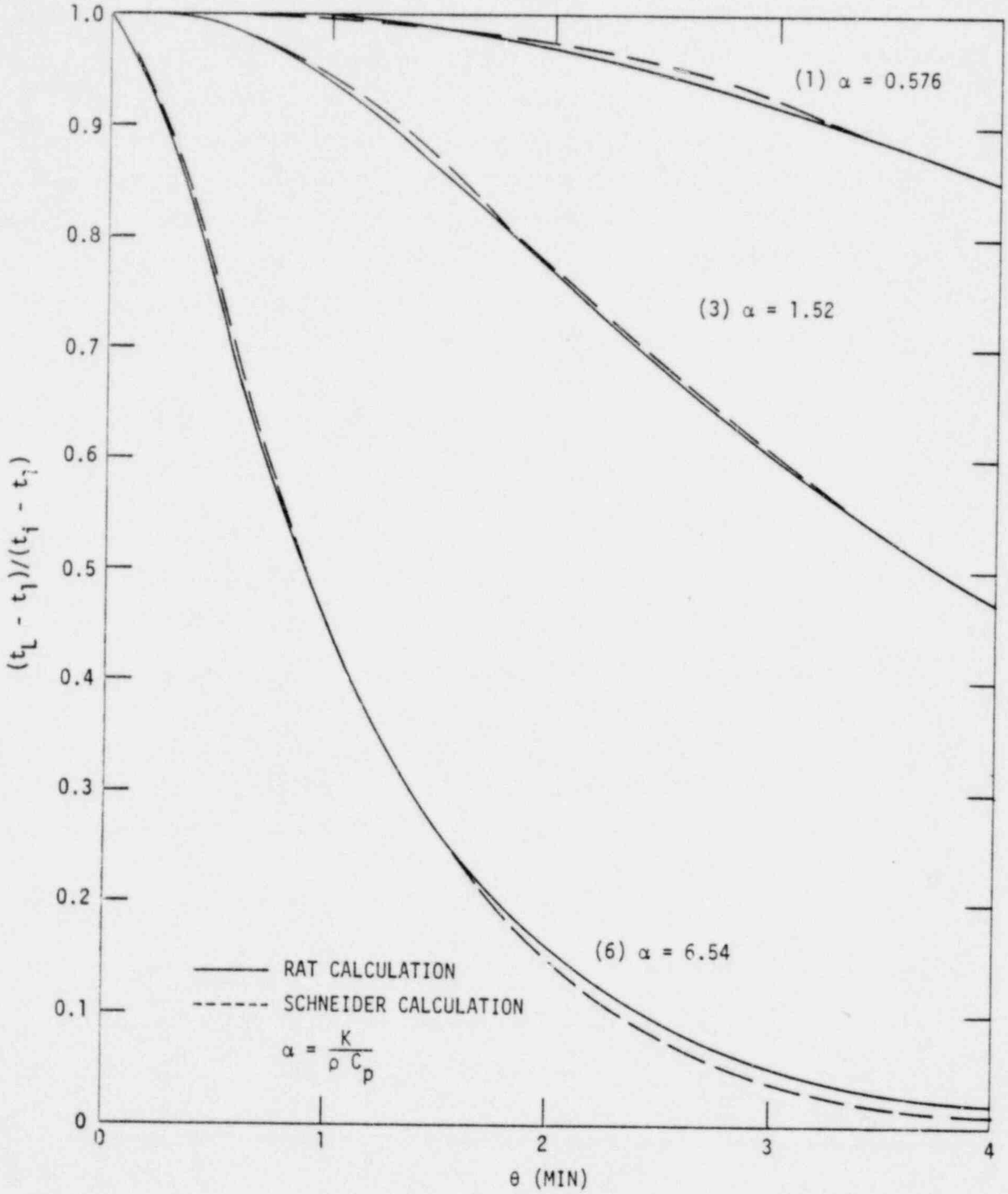


Fig. VIII-12. Results-RAT and text comparison

IMAGE EVALUATION
TEST TARGET (MT-3)

6"

MICROCOPY RESOLUTION TEST CHART

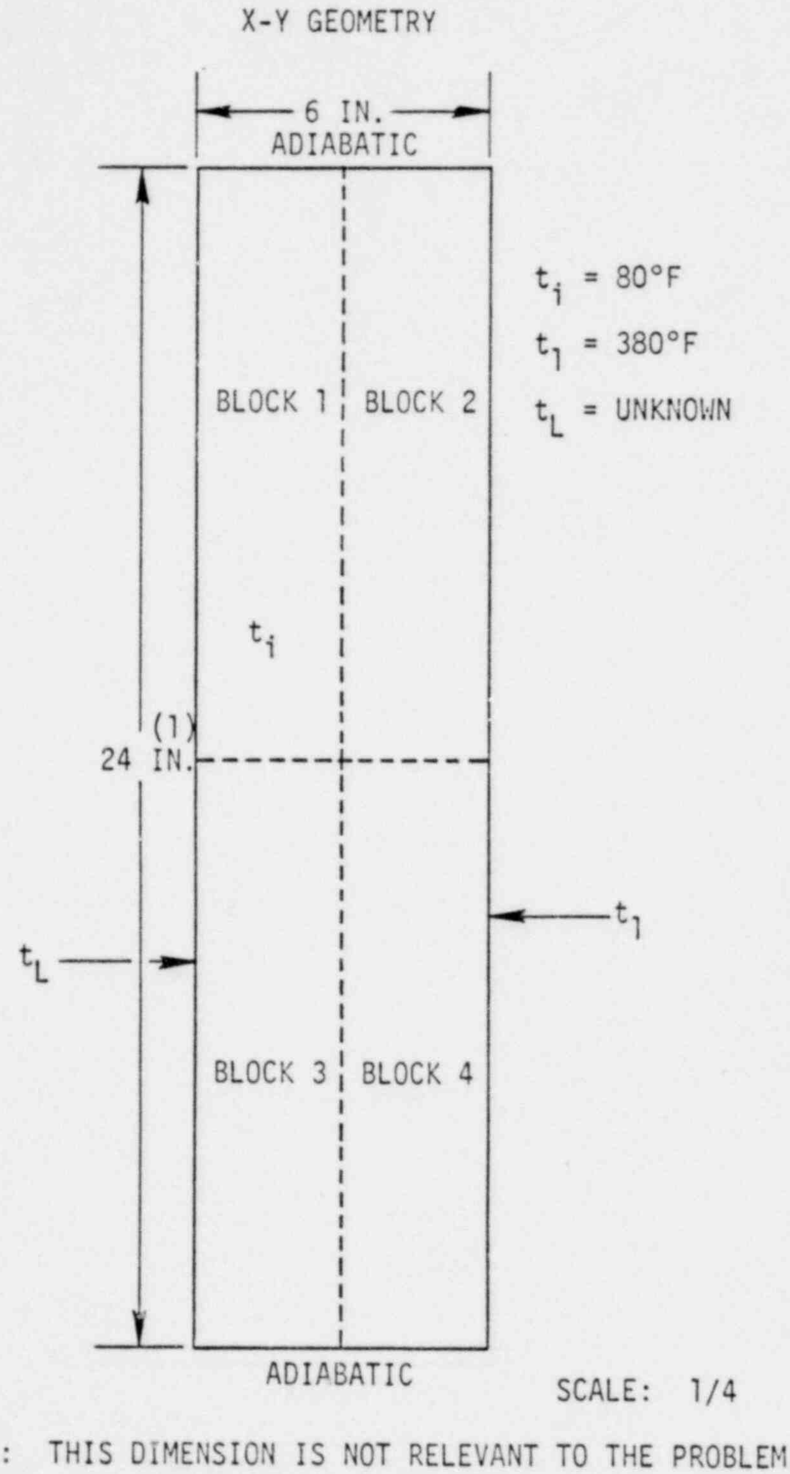


Fig. VIII-13. Thermal model - RAT test case transient 0 to 4 min

```
            RAT RESULTS                                                          30 SEP 69  PAGE      6

  RAT  -    A TWO-DIMENSIONAL TRANSIENT HEAT TRANSPORT CODE, R-A, X-Y OR R-TH GEOMETRY, GENERAL ATOMIC
CASE 1 FOR NICKEL - DIFFUSIVITY = 0.576
     CHECK TEMP. RISE AT INSULATED REAR FACE OF A PLANE WALL 6 IN.
     THICK VERSUS TIME

PRINT OF THE INPUT
```

PROPERTIES OF THE BLOCKS

BLOCK NUMBER	LOW RADIAL BOUNDARY	HIGH RADIAL BOUNDARY	LOW AXIAL BOUNDARY	HIGH AXIAL BOUNDARY	SOLID MATERIAL NUMBER	RADIAL GAP THICKNESS	RADIAL GAP MATERIAL	AXIAL GAP THICKNESS	AXIAL GAP MATERIAL
1	.000 IN.	3.000 IN.	.000 IN.	12.000 IN.	1	-.0000 IN.	-0	-.0000 IN.	-0
2	3.000 IN.	6.000 IN.	.000 IN.	12.000 IN.	1	-.0000 IN.	-0	-.0000 IN.	-0
3	.000 IN.	3.000 IN.	12.000 IN.	24.000 IN.	1	-.0000 IN.	-0	-.0000 IN.	-0
4	3.000 IN.	6.000 IN.	12.000 IN.	24.000 IN.	1	-.0000 IN.	-0	-.0000 IN.	-0

Note: It is a program requirement that there be at least two blocks in each direction

HAT RESULTS

RESULTS

COOLANT TEMPERATURES	INLET	OUTLET	FLOW (LB/HR)
INNER RADIAL	-459	80	0
OUTER RADIAL	0	0	0
UPPER AXIAL	0	0	0
LOWER AXIAL	0	0	0

THE CURRENT TIME IS .0000 HR. OR .0017 MIN. OR .10000 SEC.

TEMPERATURE GRID
THE RADIAL DIRECTION IS HORIZONTAL
THE AXIAL DIRECTION IS VERTICAL
THE TEMPERATURES ARE IN DEGREES FAHRENHEIT

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	0	0	0	0	0	0	0	0							
2	80	80	80	80	80	80	81	380							
3	80	80	80	80	80	80	81	380							
4	80	80	80	80	80	80	81	380							
5	80	80	80	80	80	80	81	380							
6	80	80	80	80	80	80	81	380							
7 t_1	80	80	80	80	80	80	81	380 t_1							
8	80	80	80	80	80	80	81	380							
9	80	80	80	80	80	80	81	380							
10	80	80	80	80	80	80	81	380							
11	80	80	80	80	80	80	81	380							
12	80	80	80	80	80	80	81	380							
13	80	80	80	80	80	80	81	380							
14	0	0	0	0	0	0	0	0							

(1) CASE FOR NICKEL
$\alpha = 0.576$

HAT RESULTS

RESULTS

COOLANT TEMPERATURES

	INLET	OUTLET	FLOW (LB/HR)
INNER RADIAL	-459	81	0
OUTER RADIAL	0	0	0
UPPER AXIAL	0	0	0
LOWER AXIAL	0	0	0

THE CURRENT TIME IS .0200 HR. OR 1.2000 MIN. OR 71.99999 SEC.

TEMPERATURE GRID
THE RADIAL DIRECTION IS HORIZONTAL
THE AXIAL DIRECTION IS VERTICAL
THE TEMPERATURES ARE IN DEGREES FAHRENHEIT

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	0	0	0	0	0	0	0	0							
2	81	81	85	96	128	196	310	380							
3	81	81	85	96	128	196	310	380							
4	81	81	85	96	128	196	310	380							
5	81	81	85	96	128	196	310	380							
6	81	81	85	96	128	196	310	380							
7	81	81	85	96	128	196	310	380							
8	81	81	85	96	128	196	310	380							
9	81	81	85	96	128	196	310	380							
10	81	81	85	96	128	196	310	380							
11	81	81	85	96	128	196	310	380							
12	81	81	85	96	128	196	310	380							
13	81	81	85	96	128	196	310	380							
14	0	0	0	0	0	0	0	0							

CASE FOR NICKEL
$\alpha = 0.576$

HAT RESULTS

RESULTS

COOLANT TEMPERATURES

	INLET	OUTLET	FLOW (LB/HR)
INNER RADIAL	-459	91	0
OUTER RADIAL	0	0	0
UPPER AXIAL	0	0	0
LOWER AXIAL	0	0	0

THE CURRENT TIME IS .0367 HR. OR 2.2000 MIN. OR 131.99998 SEC.

TEMPERATURE GRID
THE RADIAL DIRECTION IS HORIZONTAL
THE AXIAL DIRECTION IS VERTICAL
THE TEMPERATURES ARE IN DEGREES FAHRENHEIT

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	0	0	0	0	0	0	0	0							
2	91	91	101	125	170	239	330	380							
3	91	91	101	125	170	239	330	380							
4	91	91	101	125	170	239	330	380							
5	91	91	101	125	170	239	330	380							
6	91	91	101	125	170	239	330	380							
7	91	91	101	125	170	239	330	380							
8	91	91	101	125	170	239	330	380							
9	91	91	101	125	170	239	330	380							
10	91	91	101	125	170	239	330	380							
11	91	91	101	125	170	239	330	380							
12	91	91	101	125	170	239	330	380							
13	91	91	101	125	170	239	330	380							
14	0	0	0	0	0	0	0	0							

CASE FOR NICKEL
$\alpha = 0.576$

RAT RESULTS

RESULTS

COOLANT TEMPERATURES

	INLET	OUTLET	FLOW (LB/HR)
INNER RADIAL	-459	108	0
OUTER RADIAL	0	0	0
UPPER AXIAL	0	0	0
LOWER AXIAL	0	0	0

THE CURRENT TIME IS .0533 HR. OR 3.2000 MIN. OR 191.99998 SEC.

TEMPERATURE GRID
THE RADIAL DIRECTION IS HORIZONTAL
THE AXIAL DIRECTION IS VERTICAL
THE TEMPERATURES ARE IN DEGREES FAHRENHEIT

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	0	0	0	0	0	0	0	0							
2	108	108	122	151	198	262	339	380							
3	108	108	122	151	198	262	339	380							
4	108	108	122	151	198	262	339	380							
5	108	108	122	151	198	262	339	380							
6	108	108	122	151	198	262	339	380							
7	108	108	122	151	198	262	339	380							
8	108	108	122	151	198	262	339	380							
9	108	108	122	151	198	262	339	380							
10	108	108	122	151	198	262	339	380							
11	108	108	122	151	198	262	339	380							
12	108	108	122	151	198	262	339	380							
13	108	108	122	151	198	262	339	380							
14	0	0	0	0	0	0	0	0							

CASE FOR NICKEL
$\alpha = 0.576$

HAT RESULTS

RESULTS

COOLANT TEMPERATURES

	INLET	OUTLET	FLOW (LB/HR)
INNER RADIAL	-459	124	0
OUTER RADIAL	0	0	0
UPPER AXIAL	0	0	0
LOWER AXIAL	0	0	0

THE CURRENT TIME IS .0667 HR. OR 4.0000 MIN. OR 239.99997 SEC.

TEMPERATURE GRID
THE RADIAL DIRECTION IS HORIZONTAL
THE AXIAL DIRECTION IS VERTICAL
THE TEMPERATURES ARE IN DEGREES FAHRENHEIT

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	0	0	0	0	0	0	0	0							
2	124	124	139	169	215	274	343	380							
3	124	124	139	169	215	274	343	380							
4	124	124	139	169	215	274	343	380							
5	124	124	139	169	215	274	343	380							
6	124	124	139	169	215	274	343	380							
7	124	124	139	169	215	274	343	380							
8	124	124	139	169	215	274	343	380							
9	124	124	139	169	215	274	343	380							
10	124	124	139	169	215	274	343	380							
11	124	124	139	169	215	274	343	380							
12	124	124	139	169	215	274	343	380							
13	124	124	139	169	215	274	343	380							
14	0	0	0	0	0	0	0	0							

CASE FOR NICKEL
$\alpha = 0.576$

RAT RESULTS

RESULTS

COOLANT TEMPERATURES

	INLET	OUTLET	FLOW (LB/HR)
INNER RADIAL	-459	0	0
OUTER RADIAL	0	0	0
UPPER AXIAL	0	0	0
LOWER AXIAL	0	0	0

THE CURRENT TIME IS .0000 HR. OR .0017 MIN. OR .10000 SEC.

TEMPERATURE GRID
THE RADIAL DIRECTION IS HORIZONTAL
THE AXIAL DIRECTION IS VERTICAL
THE TEMPERATURES ARE IN DEGREES FAHRENHEIT

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	0	0	0	0	0	0	0	0							
2	80	80	80	80	80	80	83	380							
3	80	80	80	80	80	80	83	380							
4	80	80	80	80	80	80	83	380							
5	80	80	80	80	80	80	83	380							
6	80	80	80	80	80	80	83	380							
7	80	80	80	80	80	80	83	380							
8	80	80	80	80	80	80	83	380							
9	80	80	80	80	80	80	83	380							
10	80	80	80	80	80	80	83	380							
11	80	80	80	80	80	80	83	380							
12	80	80	80	80	80	80	83	380							
13	80	80	80	80	80	80	83	380							
14	0	0	0	0	0	0	0	0							

t_L t_1

(2) CASE FOR TIN
$\alpha = 1.52$

HAT RESULTS

RESULTS

COOLANT TEMPERATURES

	INLET	OUTLET	FLOW (LB/HR)
INNER RADIAL	-459	107	0
OUTER RADIAL	0	0	0
UPPER AXIAL	0	0	0
LOWER AXIAL	0	0	0

THE CURRENT TIME IS .0200 HR. OR 1.2000 MIN. OR 71.99999 SEC.

TEMPERATURE GRID
THE RADIAL DIRECTION IS HORIZONTAL
THE AXIAL DIRECTION IS VERTICAL
THE TEMPERATURES ARE IN DEGREES FAHRENHEIT

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	0	0	0	0	0	0	0	0							
2	107	107	121	150	197	261	338	380							
3	107	107	121	150	197	261	338	380							
4	107	107	121	150	197	261	338	380							
5	107	107	121	150	197	261	338	380							
6	107	107	121	150	197	261	338	380							
7	107	107	121	150	197	261	338	380							
8	107	107	121	150	197	261	338	380							
9	107	107	121	150	197	261	338	380							
10	107	107	121	150	197	261	338	380							
11	107	107	121	150	197	261	338	380							
12	107	107	121	150	197	261	338	380							
13	107	107	121	150	197	261	338	380							
14	0	0	0	0	0	0	0	0							

CASE FOR TIN
$\alpha = 1.52$

RESULTS

COOLANT TEMPERATURES

	INLET	OUTLET	FLOW (LB/HR)
INNER RADIAL	-459	160	0
OUTER RADIAL	0	0	0
UPPER AXIAL	[illegible]	[illegible]	0
LOWER AXIAL	0	0	0

THE CURRENT TIME IS .0367 HR. OR 2.2000 MIN. OR 131.99998 SEC.

TEMPERATURE GRID
THE RADIAL DIRECTION IS HORIZONTAL
THE AXIAL DIRECTION IS VERTICAL
THE TEMPERATURES ARE IN DEGREES FAHRENHEIT

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	0	0	0	0	0	0	0	0							
2	160	160	175	203	243	293	350	380							
3	160	160	175	203	243	293	350	380							
4	160	160	175	203	243	293	35	380							
5	160	160	175	203	243	293	350	380							
6	160	160	175	203	243	293	350	380							
7	160	160	175	203	243	293	350	380							
8	160	160	175	203	243	293	350	380							
9	160	160	175	203	243	293	350	380							
10	160	160	175	203	243	293	350	380							
11	160	160	175	203	243	293	350	380							
12	160	160	175	203	243	293	350	380							
13	160	160	175	203	243	293	350	380							
14	0	0	0	0	0	0	0	0							

CASE FOR TIN
$\alpha = 1.52$

RAT RESULTS

RESULTS

COOLANT TEMPERATURES

	INLET	OUTLET	FLOW (LB/HR)
INNER RADIAL	-459	208	0
OUTER RADIAL	0	0	0
UPPER AXIAL	0	0	0
LOWER AXIAL	0	0	0

THE CURRENT TIME IS .0533 HR. OR 3.2000 MIN. OR 191.99998 SEC.

TEMPERATURE GRID
THE RADIAL DIRECTION IS HORIZONTAL
THE AXIAL DIRECTION IS VERTICAL
THE TEMPERATURES ARE IN DEGREES FAHRENHEIT

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	0	0	0	0	0	0	0	0							
2	208	208	219	242	274	313	357	380							
3	208	208	219	242	274	313	357	380							
4	208	208	219	242	274	313	357	380							
5	208	208	219	242	274	313	357	380							
6	208	208	219	242	274	313	357	380							
7	208	208	219	242	274	313	357	380							
8	208	208	219	242	274	313	357	380							
9	208	208	219	242	274	313	357	380							
10	208	208	219	242	274	313	357	380							
11	208	208	219	242	274	313	357	380							
12	208	208	219	242	274	313	357	380							
13	208	208	219	242	274	313	357	380							
14	0	0	0	0	0	0	0	0							

CASE FOR TIN
$\alpha = 1.52$

HAT RESULTS

RESULTS

COOLANT TEMPERATURES

	INLET	OUTLET	FLOW (LB/HR)
INNER RADIAL	-459	239	0
OUTER RADIAL	0	0	0
UPPER AXIAL	0	0	0
LOWER AXIAL	0	0	0

THE CURRENT TIME IS .0667 HR. OR .0000 MIN. OR 239.99997 SEC.

TEMPERATURE GRID
THE RADIAL DIRECTION IS HORIZONTAL
THE AXIAL DIRECTION IS VERTICAL
THE TEMPERATURES ARE IN DEGREES FAHRENHEIT

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	0	0	0	0	0	0	0	0							
2	239	239	248	267	293	325	361	380							
3	239	239	248	267	293	325	361	380							
4	239	239	248	267	293	325	361	380							
5	239	239	248	267	293	325	361	380							
6	239	239	248	267	293	325	361	380							
7	239	239	248	267	293	325	361	380							
8	239	239	248	267	293	325	361	380							
9	239	239	248	267	293	325	361	380							
10	239	239	248	267	293	325	361	380							
11	239	239	248	267	293	325	361	380							
12	239	239	248	267	293	325	361	380							
13	239	239	248	267	293	325	361	380							
14	0	0	0	0	0	0	0	0							

CASE FOR TIN
$\alpha = 1.52$

POOR ORIGINAL

RAT RESULTS

INITIAL TEMPERATURE DISTRIB

THE CURRENT TIME IS -.0000 HR. OR .0000 MIN. OR .00000 SEC.

TEMPERATURE GRID
THE RADIAL DIRECTION IS HORIZONTAL
THE AXIAL DIRECTION IS VERTICAL
THE TEMPERATURES ARE IN DEGREES FAHRENHEI

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	0	0	0	0	0	0	0	0							
2	80	80	80	80	80	80	80	380							
3	80	80	80	80	80	80	80	380							
4	80	80	80	80	80	80	80	380							
5	80	80	80	80	80	80	80	380							
6	80	80	80	80	80	80	80	380							
7	80	80	80	80	80	80	80	380 t_1							
8	t_L 80	80	80	80	80	80	80	380							
9	80	80	80	80	80	80	80	380							
10	80	80	80	80	80	80	80	380							
11	80	80	80	80	80	80	80	380							
12	80	80	80	80	80	80	80	380							
13	80	80	80	80	80	80	80	380							
14	0	0	0	0	0	0	0	0							

(3) CASE FOR SILVER
$\alpha = 6.54$

RAT RESULTS

RESULTS

COOLANT TEMPERATURES

	INLET	OUTLET	FLOW (LB/HR)
INNER RADIAL	-459	155	0
OUTER RADIAL	0	0	0
UPPER AXIAL	0	0	0
LOWER AXIAL	0	0	0

THE CURRENT TIME IS .0083 HR. OR .5000 MIN. OR 30.00000 SEC.

TEMPERATURE GRID
THE RADIAL DIRECTION IS HORIZONTAL
THE AXIAL DIRECTION IS VERTICAL
THE TEMPERATURES ARE IN DEGREES FAHRENHEIT

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	0	0	0	0	0	0	0	0							
2	155	155	170	198	238	288	345	380							
3	155	155	170	198	238	288	345	380							
4	155	155	170	198	238	288	345	380							
5	155	155	170	198	238	288	345	380							
6	155	155	170	198	238	288	345	380							
7	155	155	170	198	238	288	345	380							
8	155	155	170	198	238	288	345	380							
9	155	155	170	198	238	288	345	380							
10	155	155	170	198	238	288	345	380							
11	155	155	170	198	238	288	345	380							
12	155	155	170	198	238	288	345	380							
13	155	155	170	198	238	288	345	380							
14	0	0	0	0	0	0	0	0							

CASE FOR SILVER
$\alpha = 6.54$

RAT RESULTS

RESULTS

COOLANT TEMPERATURES

	INLET	OUTLET	FLOW (LB/HR)
INNER RADIAL	-459	342	0
OUTER RADIAL	0	0	0
UPPER AXIAL	0	0	0
LOWER AXIAL	0	0	0

THE CURRENT TIME IS .0367 HR. OR 2.2000 MIN. OR 131.99998 SEC.

TEMPERATURE GRID
THE RADIAL DIRECTION IS HORIZONTAL
THE AXIAL DIRECTION IS VERTICAL
THE TEMPERATURES ARE IN DEGREES FAHRENHEIT

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	0	0	0	0	0	0	0	0							
2	342	342	344	349	356	365	374	380							
3	342	342	344	349	356	365	374	380							
4	342	342	344	349	356	365	374	380							
5	342	342	344	349	356	365	374	380							
6	342	342	344	349	356	365	374	380							
7	342	342	344	349	356	365	374	380							
8	342	342	344	349	356	365	374	380							
9	342	342	344	349	356	365	374	380							
10	342	342	344	349	356	365	374	380							
11	342	342	344	349	356	365	374	380							
12	342	342	344	349	356	365	374	380							
13	342	342	344	349	356	365	374	380							
14	0	0	0	0	0	0	0	0							

CASE FOR SILVER
$\alpha = 6.54$

RAT RESULTS

RESULTS

COOLANT TEMPERATURES

	INLET	OUTLET	FLOW (LB/HR)
INNER RADIAL	-459	367	0
OUTER RADIAL	0	0	0
UPPER AXIAL	0	0	0
LOWER AXIAL	0	0	0

THE CURRENT TIME IS .0533 HR. OR 3.2000 MIN. OR 191.99998 SEC.

TEMPERATURE GRID
THE RADIAL DIRECTION IS HORIZONTAL
THE AXIAL DIRECTION IS VERTICAL
THE TEMPERATURES ARE IN DEGREES FAHRENHEIT

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	0	0	0	0	0	0	0	0							
2	367	367	367	369	372	375	378	380							
3	367	367	367	369	372	375	378	380							
4	367	367	367	369	372	375	378	380							
5	367	367	367	369	372	375	378	380							
6	367	367	367	369	372	375	378	380							
7	367	367	367	369	372	375	378	380							
8	367	367	367	369	372	375	378	380							
9	367	367	367	369	372	375	378	380							
10	367	367	367	369	372	375	378	380							
11	367	367	367	369	372	375	378	380							
12	367	367	367	369	372	375	378	380							
13	367	367	367	369	372	375	378	380							
14	0	0	0	0	0	0	0	0							

CASE FOR SILVER
$\alpha = 6.54$

RAT RESULTS

RESULTS

COOLANT TEMPERATURES

	INLET	OUTLET	FLOW (LB/HR)
INNER RADIAL	-459	374	0
OUTER RADIAL	0	0	0
UPPER AXIAL	0	0	0
LOWER AXIAL	0	0	0

THE CURRENT TIME IS .0667 HR. OR 4.0000 MIN. OR 239.99997 SEC.

TEMPERATURE GRID
THE RADIAL DIRECTION IS HORIZONTAL
THE AXIAL DIRECTION IS VERTICAL
THE TEMPERATURES ARE IN DEGREES FAHRENHEIT

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	0	0	0	0	0	0	0	0							
2	374	374	375	375	376	378	379	380							
3	374	374	375	375	376	378	379	380							
4	374	374	375	375	376	378	379	380							
5	374	374	375	375	376	378	379	380							
6	374	374	375	375	376	378	379	380							
7	374	374	375	375	376	378	379	380							
8	374	374	375	375	376	378	379	380							
9	374	374	375	375	376	378	379	380							
10	374	374	375	375	376	378	379	380							
11	374	374	375	375	376	378	379	380							
12	374	374	375	375	376	378	379	380							
13	374	374	375	375	376	378	379	380							
14	0	0	0	0	0	0	0	0							

CASE FOR SILVER
$\alpha = 6.54$

CONDUCTION HEAT TRANSFER

by

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READING, MASSACHUSETTS

10–5 Infinite plate. Consider the heating or cooling of a large plate of uniform thickness $L = 2\delta_1$. The temperature distribution through the plate, $t(x,\theta)$, is initially ($\theta = 0$) some arbitrary function of x as $t(x,0) = t_i(x)$, whereupon both face surfaces $x = 0$ and L are suddenly changed to and maintained at a uniform temperature t_1 for all $\theta > 0$.

The solution for the temperature history $t(x,\theta)$ must satisfy the characteristic partial-differential equation of Fourier, (1–8), as

$$\frac{\partial^2 T}{\partial x^2} = \frac{1}{\alpha}\frac{\partial T}{\partial \theta}, \tag{10–6}$$

and the initial and boundary conditions

$$\begin{aligned} T &= T_i(x) && \text{at } \theta = 0; && 0 \le x \le L, \\ T &= 0 && \text{at } x = 0; && \theta > 0, \\ T &= 0 && \text{at } x = L && \theta > 0, \end{aligned}$$

where, for convenience, we let $T = t - t_1$ so that $T_i(x) = t_i(x) - t_1$. Integrating (10–6) by the separation of variables method (Article 7–8) leads to product solutions of the form

$$T_\lambda = e^{-\lambda^2\alpha\theta}(C_1 \cos \lambda x + C_2 \sin \lambda x),$$

if the separation constant is chosen as $-\lambda^2$. In these solutions $C_1 = 0$ if T is to vanish at $x = 0$ for all $\theta > 0$. If T is also to vanish at $x = L$ for all $\theta > 0$, then $\sin \lambda L = 0$, so that $\lambda = n\pi/L$. This requires that the eigenvalues be integral as $n = 1, 2, 3, \cdots$. The solution, which now takes the form

$$T = \sum_{n=1}^{\infty} C_n e^{-(n\pi/L)^2\alpha\theta} \sin \frac{n\pi}{L} x,$$

is to finally satisfy the initial condition that

$$T_i(x) = \sum_{n=1}^{\infty} C_n \sin \frac{n\pi}{L} x.$$

This result is recognized as a Fourier sine-series expansion of the arbitrary function $T_i(x)$, for which the constant amplitudes C_n are given by

$$C_n = \frac{2}{L}\int_0^L T_i(x) \sin \frac{n\pi}{L} x dx.$$

The complete solution is therefore

$$T = \frac{2}{L}\sum_{n=1}^{\infty} e^{-(n\pi/2)^2\Theta} \sin \frac{n\pi}{L} x \int_0^L T_i(x) \sin \frac{n\pi}{L} x dx, \tag{10–7}$$

where Θ is the Fourier modulus in (10–2) with $\delta_1 = L/2$. The solution (10–7) is also that for an insulated rod of length L with end temperatures maintained at t_1, the rod heating or cooling from an initial temperature state $t_i(x)$.

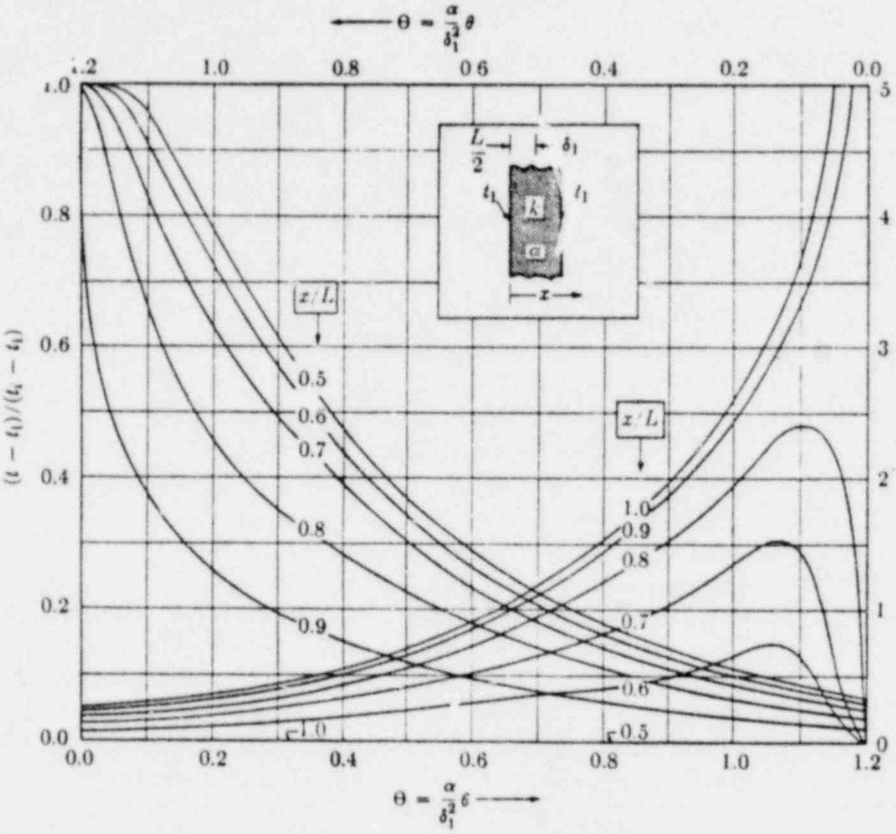


Fig. 10–2. Temperature history and instantaneous heat rate as a function of the Fourier modulus Θ for an infinite plate with negligible surface resistance.

Consider the special case represented by a uniform initial temperature $t_i(x) = t_i$. This is a practical case where $T_i(x) = t_i - t_1$, and for which (10–7) reappears in the particular form

$$\frac{t - t_1}{t_i - t_1} = \frac{4}{\pi}\sum_{n=1}^{\infty}\frac{1}{n}e^{-(n\pi/2)^2\Theta}\sin\frac{n\pi}{L}x; \quad n = 1, 3, 5, \cdots. \qquad (10\text{–}8)$$

Then the instantaneous rate at which heat is conducted across any plane of area A in the wall is $q = -kA\partial t/\partial x$, or

$$q(x,\theta) = 4\left(\frac{kA}{L}\right)(t_1 - t_i)\sum_{n=1}^{\infty}e^{-(n\pi/2)^2\Theta}\cos\frac{n\pi}{L}x; \quad n = 1, 3, 5, \cdots. \qquad (10\text{–}9)$$

Notice that the heat flow is initially infinite at the two surfaces. The temperature history (10–8) and instantaneous heat rate (10–9) are shown in Fig. 10–2 for various stations in the plate.

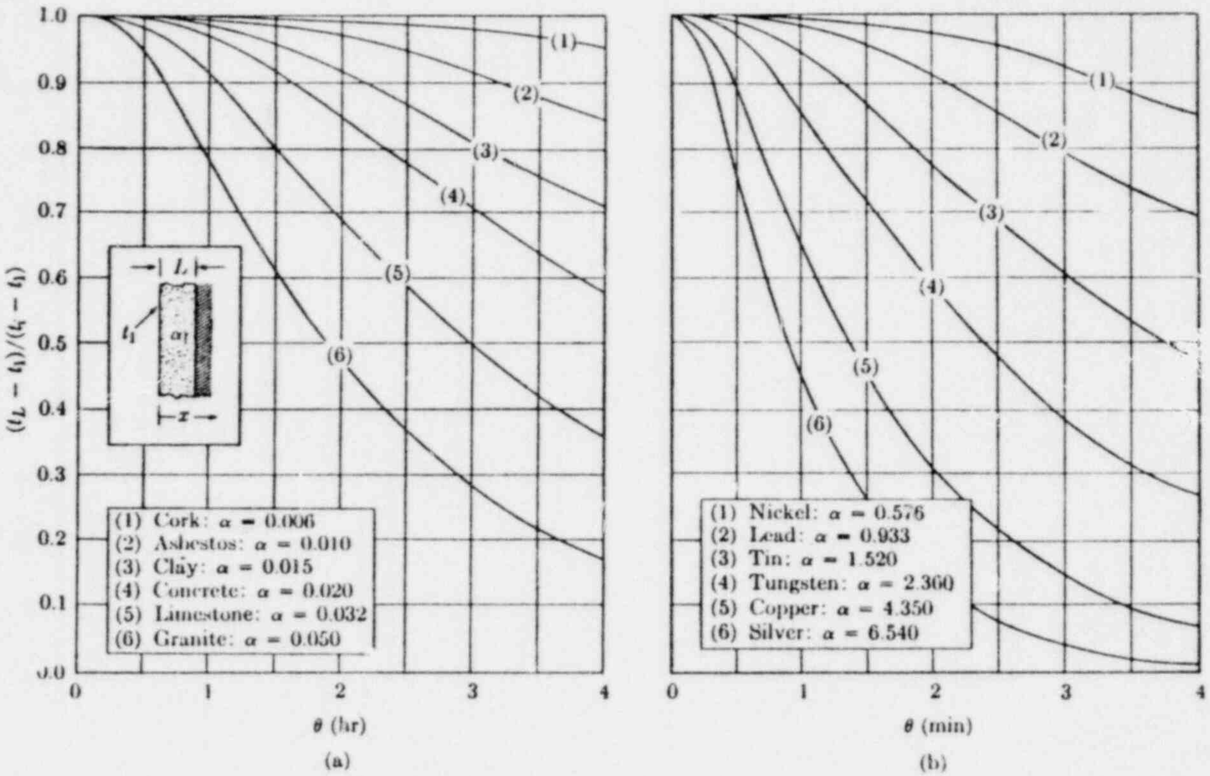


Fig. 10-3. Temperature rise at the insulated rear face of a plane wall 6" thick, for various building materials (a) and pure metals (b).

Integration of (10-9) with respect to θ from $\theta = 0$ to θ gives for the cumulative heat rate in the plate

$$Q(x,\theta) = \frac{4}{\pi^2}\left(\frac{kAL}{\alpha}\right)(t_1 - t_i)\sum_{n=1}^{\infty}\frac{1}{n^2}[1 - e^{-(n\pi/2)^2\Theta}]\cos\frac{n\pi}{L}x;$$

$$n = 1, 3, 5, \cdots. \quad (10\text{-}10)$$

The series (10-8) has been accurately computed by Olson and Schultz (1) for the mid-plane temperature history at $x = L/2$ as given by

$$\frac{t(L/2,\theta) - t_1}{t_i - t_1} = \frac{4}{\pi} = [e^{-(1/4)\pi^2\Theta} - \tfrac{1}{3}e^{-(9/4)\pi^2\Theta} + \tfrac{1}{5}e^{-(25/4)\pi^2\Theta} - \cdots] = P(\Theta). \quad (10\text{-}11)$$

Values of $P(\Theta)$ for the plate are listed in Table A-8 of the Appendix, and later on we shall show how this particular series can be combined with an analogous series for the cylinder and semi-infinite solid to obtain solutions for a variety of other cases of practical interest.

EXAMPLE 10-2. As an example in the use of the plate series $P(\Theta)$, suppose that a large mass of combustible material is piled up against the 6″-thick wall of a large room. If a flash fire suddenly raises and maintains the temperature of the outside wall surface at t_1, how long will it take for the inside surface to reach the ignition temperature of the combustible material?

Solution. If the dimensions of the room are large compared with the wall thickness, then its walls can be approximated by an infinite plate 6″ thick and at a uniform temperature t_i preceding its exposure to fire at the face surface $x = 0$. Suppose further that the combustible material is of low thermal conductivity; the face surface at $x = L$ is then considered to be adiabatic.

A solution to the problem of the infinite plate with one insulated face is also encompassed in (10-8), for if we consider a plate of double thickness $2L$ with each face at t_1, then its mid plane (around which the temperature is symmetrical) can be taken as the adiabatic face of the original plate. The temperature history at this adiabatic face is given by (10-11).

Values of $P(\Theta)$ with $L = 1$ are taken from Table A-8 and plotted in Fig. 10-3 for walls of various building materials in (a), and for large pure-metal walls in (b). Note that the units of time in (a) and (b) are hours and minutes respectively.

From these results we see that the walls of higher diffusivity have shorter allowable heating times, the duration being comparatively short for the metal walls. For example, if the ignition temperature of the stored material is 140°F, and the fire raises the outside surface temperature of the walls (initially at 80°F) to 380°F, then it would take a 6″ clay wall over three hours to reach $(t_L - t_1)/(t_i - t_1) = (140 - 380)/(80 - 380) = 0.8$, as compared with just three minutes for a 6″ lead wall.

SECTION IX

CRITICALITY ANALYSIS

SECTION-IX- CRITICALITY ANALYSIS

A. Description and Quantity of Fissile Material

The fissile material is contained in fuel elements which are hexagonal in cross section with dimensions 14.2" across flats by 31.2" high (Fig. IX-1). Each fuel element contains coolant and fuel channels which are drilled from the top face of the element. Fuel holes are drilled to within about .3" of the bottom face and are closed at the top by a .5" cemented graphite plug. The fuel channels occupy alternating positions in a triangular array within the element structure, are .5" in diameter and contain the active fuel.

The element structure consists of a conventional needle coke graphite. The fuel itself is in the form of carbide particles coated with layers of pyrolytic carbon and silicon carbide. The fuel bed contains a homogeneous mixture of two types of particles, called fissile and fertile. Fresh fissile particles contain both thorium and 93.5% enriched uranium, while fresh fertile particles contain only thorium. The important parameters of fresh particles are:

Parameter	Fissile	Fertile
Th/U	4.25	All Th
Particle Composition	$(Th/U)C_2$	$Th\ C_2$
Average Fuel Particle Diameter, μm	200	450
Average Total Coating Thickness, μm	130	140

Irradiated fuel elements contain, besides fission products, thorium, U-233, U-235, other uranium isotopes and a small quantity of plutonium. In the fertile particles, the fissile material is essentially U-233, while the fissile particles contain the residual U-235 and bred U-233.

The effective fissile material enrichment (U-235/U+Th) in fresh fuel for the initial core and reload segments varies between 2% and 12% due to radial and axial fuel zoning requirements. The most reactive fresh fuel element to be considered contains about 1.4 Kg of 93.5% enriched uranium and 11.3 Kg of

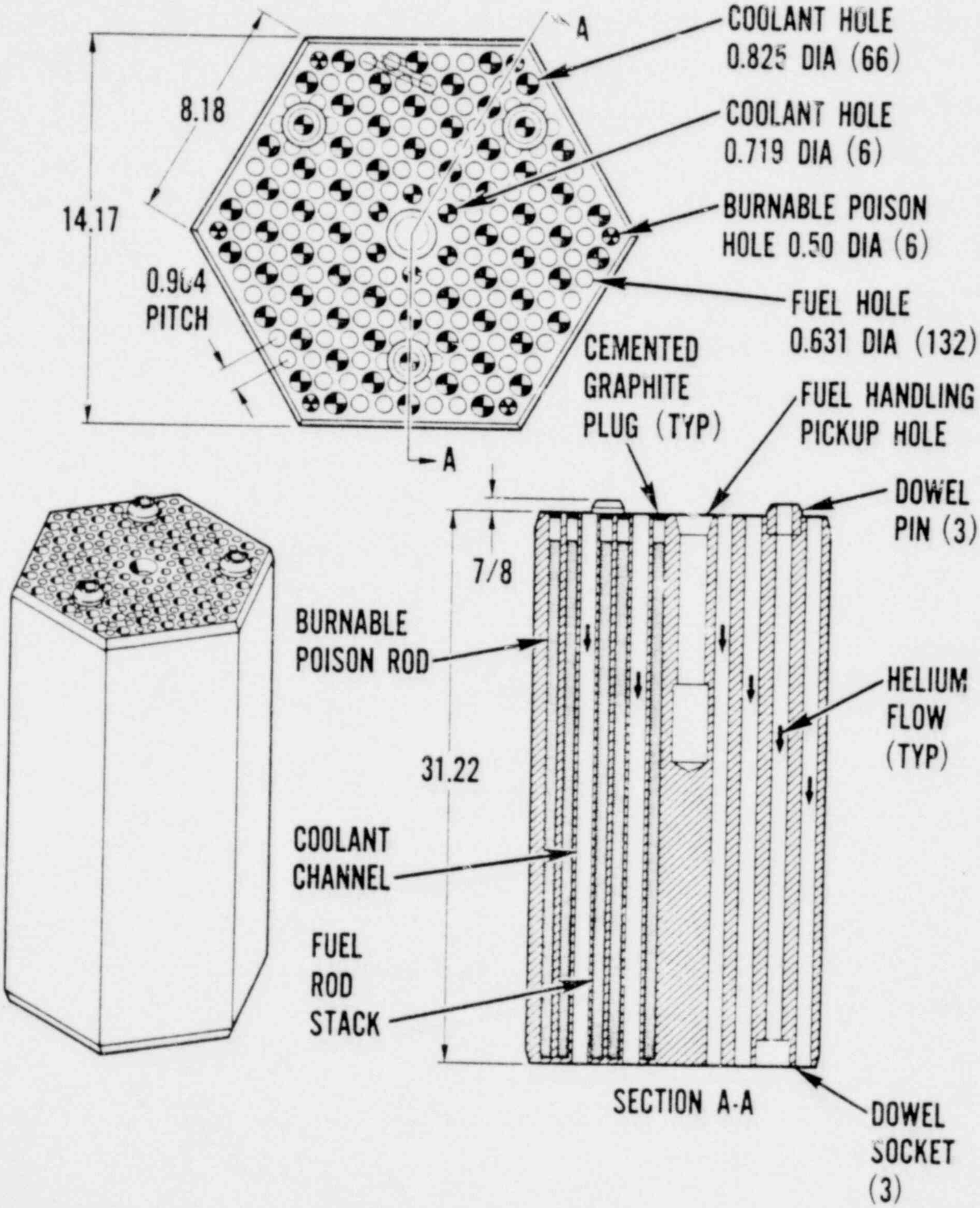


Figure IX-1. Fuel Element

thorium. Any irradiated elements will contain a smaller amount of fissile material, since the conversion ratio of the reactor is less than unity.

B. Assumptions for Criticality Evaluation

The following assumptions were used for the criticality evaluation:

1. The fuel elements contain the most reactive fresh fuel composition anticipated for fuel shipment, i.e., about 1.4 Kg of 93.5% enriched uranium and 11.3 Kg of thorium per element resulting in a maximum of 8.4 Kg of uranium and 68 Kg of thorium per container. The total amounts of all materials present in a fuel element as well as the homogenized atom densities (over a fuel element) are summarized in Table IX-1.
2. The presence of burnable poison or other neutron absorbing material, other than U-235, U-238, thorium, silicon and graphite, is neglected.
3. The fuel is at room temperature.
4. All fission products are neglected.

These assumptions are all conservative. In general spent fuel elements will contain considerably less fissile material. Also, all other fresh fuel element types are less reactive due to their lower uranium contents and/or higher thorium content.

C. Analytical Models

1. Geometry

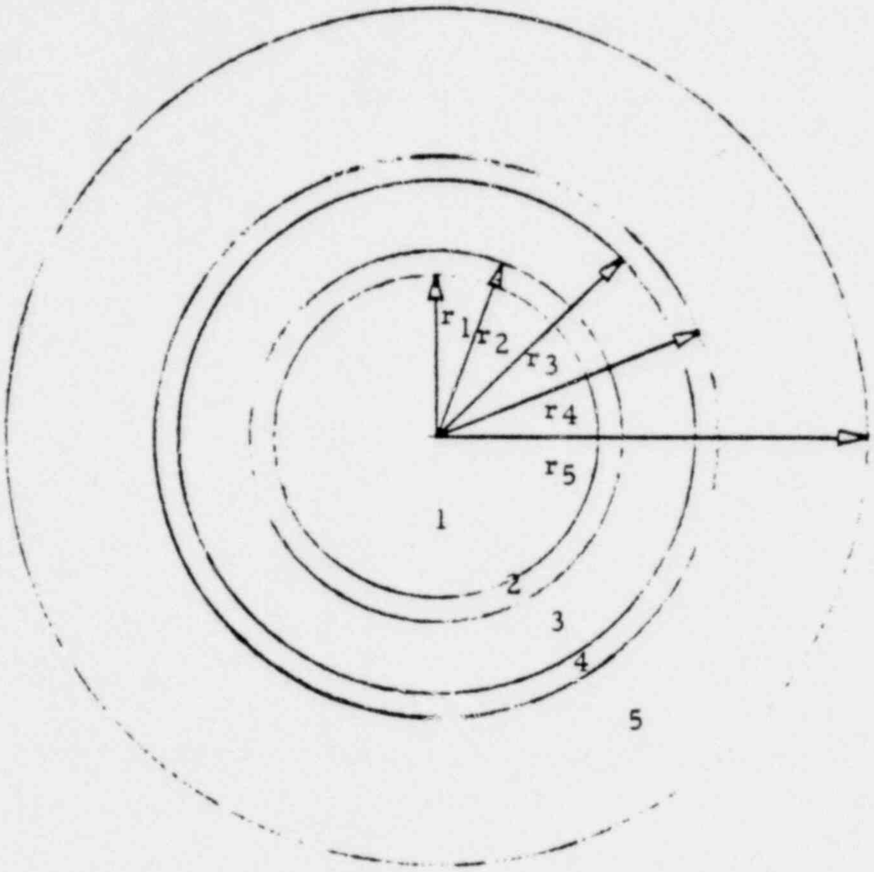
Two geometrical models were used to evaulate the criticality situation for the shipping cask. The one-dimensional model is shown in Fig. IX-2, and assumes an infinitely long cylinder. This model is adequate as long as it can be assumed that the fuel is well contained within the fuel elements.

The two-dimensional geometry model, shown in Fig. IX-3, was used for the maximum criticality situation which includes block breakage, internal flooding and an accumulation of fuel particles at the bottom of the cask.

TABLE IX-1

Most Reactive Fuel Element Material Contents

	Total Amount (Kg)	Homogenized Atom Density (atoms/b-cm)
Th-232	11.25	$3.28 \cdot 10^{-4}$
U-235	1.31	$3.76 \cdot 10^{-5}$
U-238	.078	$2.22 \cdot 10^{-6}$
Silicon	4.63	$1.12 \cdot 10^{-3}$
Carbon	111.3	$6.27 \cdot 10^{-2}$

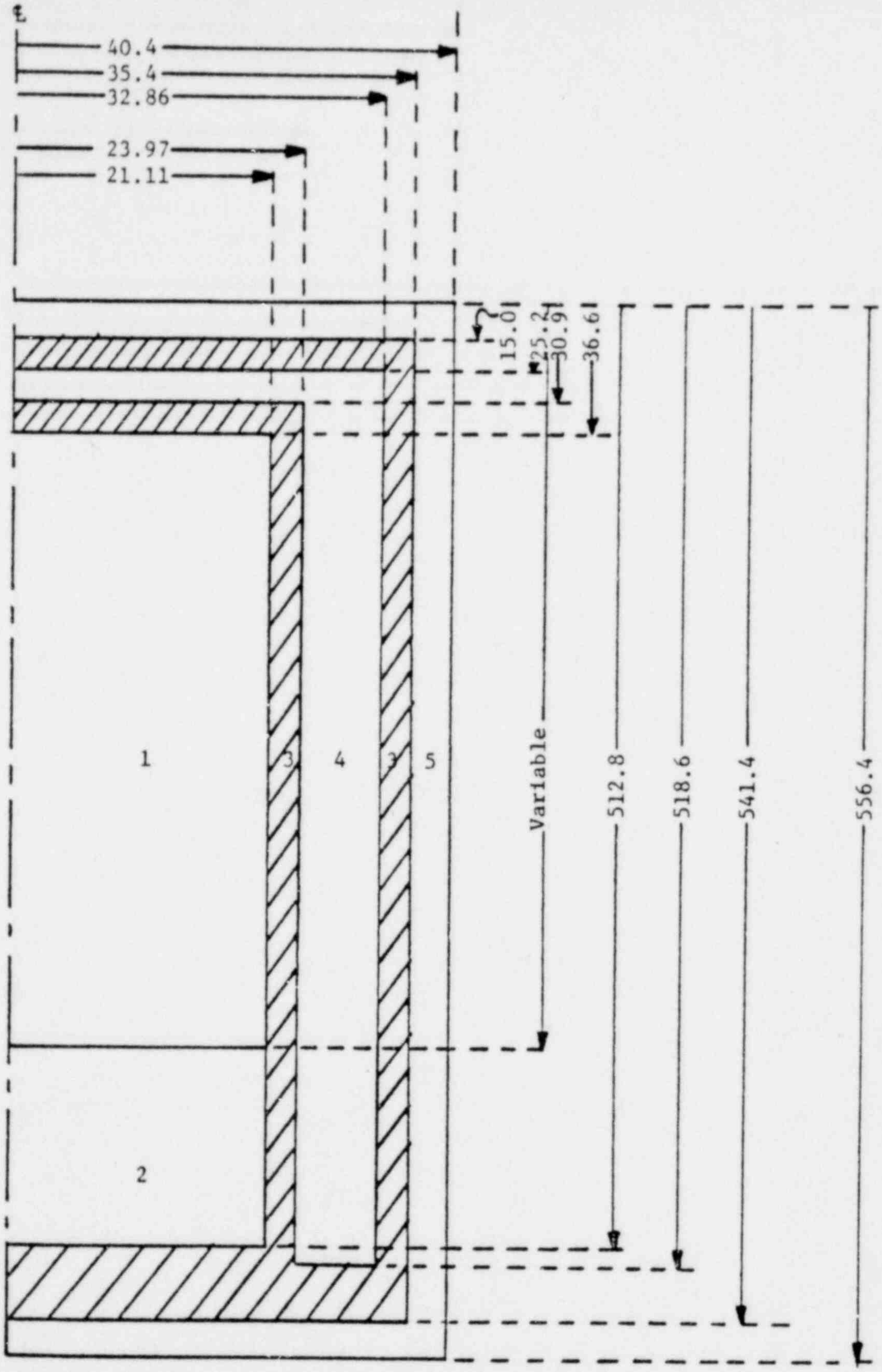


Region	Region Outer Radius	Region Contents
1	r_1 = 21.11 cm	Fuel element (+ water)
2	r_2 = 23.97 cm	Iron
3	r_3 = 32.86 cm	Uranium-235 & uranium-238
4	r_4 = 35 40 cm	Iron
5	r_5 = 55.40 cm	Void or water

Fig. IX-2. Fort St. Vr‹ ιn - Spent Fuel Shipping Cask, One-Dimensional Model

POOR ORIGINAL

All distances in cm



Region 1 : Fuel
Region 2 : Fuel
Region 3 : Iron
Region 4 : Uranium
Region 5 : Void or water

Fig. IX-3. Two-Dimensional Model

POOR ORIGINAL

2. Cross Sections

Cross sections were calculated with the GGC-4 code, a description of which is attached. (See Section I, Code Abstracts.) The calculational methods and the basic nuclear cross-section data are well established and in use for the HTGR nuclear design.

3. Computer Codes

The DTF-IV transport code was used for one-dimensional and the GAMBLE-5 code was used for the two-dimensioned calculations. Abstracts of both codes are attached. (See Section I, Code Abstracts.)

SECTION XI - CRITICALITY ANALYSIS

D. Description and Quantity of Radioactive Gasses

Fission product gasses do not contribute to pressure buildup in the cask during an accidental fire.

The fuel temperature during the accident was stated to be 290°F. At this temperature there will be essentially no fission product release. However, even if the total cask fuel inventory of volatile fission products were released, there might still be no significant pressure buildup. After 60 days of decay there will be about 1×10^6 curies and about 1500 gm moles of stable noble gas and iodine in the total core, and in a single column (the cask capacity) the amount would be about 6500 curies and 9.7 gm moles assuming a power peaking factor of 1.6 during operation. Assuming that the average half life of remaining volatile fission products is 3×10^5 sec the amount of gas associated with this activity would be

$$6500 \text{ curies} * 3.7 \times 10^{10} \frac{\text{dis atom}}{\text{sec-curie}} * \frac{\text{gm mole}}{6.023(+23)\text{atoms}} * \frac{3 \times 10^5 \text{sec}}{.693}$$

$$= 1.73 \times 10^{-4} \text{ gm moles}$$

which is negligible compared to the stable gasses which would occupy a volume at STP of

$$\frac{9.7}{454 \frac{\text{gm moles}}{\text{lb mole}}} \text{ gm moles} * \frac{385 \text{ ft}^3}{\text{lb mole}} = 8.2 \text{ ft}^3 \text{ STP}$$

SECTION IX - CRITICALITY ANALYSIS

E. Volatile Fission Gas in the Fort St. Vrain Fuel Shipping Cask

The maximum fission product activity that will be contained in the Fort St. Vrain fuel shipping cask at any time would be that associated with six maximum power fuel blocks (not necessarily from the same column in the reactor) having the full 6-year burnup. Total fission product per fuel element is given in Table IX-2.

Assuming an average power peaking factor of 1.35 over the 6 years in the reactor and 100 days of decay since shutdown, the volatile (noble gases and iodine) inventory in the fuel elements would amount to about 4,700 curies of noble gas (principally Kr-85) and 23 curies of iodine (principally I-131). In addition, about 9 gm moles of stable noble gas and iodine would be present. The total amount of volatile fission product gases would occupy a volume of about 7.5 ft^3 STP.

This fission product activity is triply contained, first by the fuel particles and their impervious TRISO coatings, secondly by the fuel element containers, and third, by the shipping cask itself. If an accident is postulated that breaches both the cask and container, no fission product gases will escape as long as the fuel particles themselves remain intact. Fission products can be annealed from fuel particles, but not in significant quantity at temperatures below 1000°F. Thus, a breached cask and cans, exposed to a fire in which the fuel elements reach a maximum temperature of 300°F, will not result in significant activity release.

If a breached cask with failed cans is assumed to be immersed in water, then the potential exists for hydrolysis of the carbide kernels of fuel particles with previously failed coatings. With unrestricted exposure to

SECTION IX - CRITICALITY ANALYSIS

Table IX-2

FUEL ELEMENT FISSION PRODUCT INVENTORY

Equilibrium Core, 6-year Operation, 100 Days Shutdown

Isotope	Activity (Curies)
Kr^{85}	786.
Sr^{89}	8600.
Sr^{90}	2300.
Y^{90}	2300.
Y^{91}	11900.
Zr^{95}	13600.
Nb^{95m}	270.
Nb^{95}	13600.
Ru^{103}	2700.
Rh^{103m}	2700.
Ru^{106}	1600.
Te^{127m}	270.
Te^{127}	270.
Te^{129m}	390.
I^{131}	24.
Te^{129}	390.
Cs^{137}	2250.
Ba^{137m}	2080.
Ba^{140}	186.
La^{140}	186.
Ce^{141}	5100.
Pr^{143}	260.
Ce^{144}	25200.
Pr^{144}	25200.
Pm^{147}	9500.
Sm^{151}	470.
Pa^{233}	34000.

moisture, essentially complete hydrolysis of all failed fuel would be expected with a corresponding release of the noble gas activity, i.e., 100% release for 100% hydrolysis. No iodine has been observed to be released by this mechanism. If the upper limit "design" fuel particle failed fraction of 10% is assumed, along with complete hydrolysis of this fraction, then the noble gas release would amount to about 470 curies of Kr-85. The potential dose from this activity is dependent on the atmospheric conditions and the distance between the source and receptor which combine to make up the effective atmospheric dilution factor. Conservatively assuming a dilution factor of 0.01 sec/m^3 and that the receptor is immersed in a semi-infinite cloud of released activity, then the received dose would be only about 2.5 millirem according to

$$\text{Dose (rem)} = 0.247 \times \text{curies released} \times \text{gamma energy (mev/dis.)} \times \text{dilution factor} \left(\frac{\text{sec}}{\text{m}^3}\right)$$

$$= 0.247 \times 470 \times 0.0021 \times 0.01 = 2.45 \times 10^{-3} \text{ rem}$$

The total amount of volatile activity in the cask (4,723 curies) and the maximum potential amount releasable (470 curies of Kr-85) are well below the permissible limits of 1000 curies.*

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* Per Title 10-CFR-Part 71, par. 71.36 (2) (ii)

F. Safety During Normal Transport

During normal transport not more than 6 fuel elements will be stacked end to end within the shipping container. No hydrogeneous or other moderating material besides the structural and coating graphite will be present.

The fuel elements occupy 81.4% of the inner section of the shipping cask. The contents of the elements were homogenized over the inner section of the cask and used in one-dimensional transport calculations. Material concentrations for the different regions consistent with Fig. IX-2 are given in Table IX-3.

The calculated multiplication constant for this condition, assuming an infinitely long container, is .40 and no criticality hazard is to be expected under any circumstances.

Table IX-3

Material Concentrations

Region	Material	Concentration (atoms/b-cm)
1	Th-232	$2.67 \cdot 10^{-4}$
	U-235	$3.06 \cdot 10^{-5}$
	U-238	$1.81 \cdot 10^{-6}$
	Si	$9.12 \cdot 10^{-4}$
	C	$5.10 \cdot 10^{-2}$
2	Fe	$8.50 \cdot 10^{-2}$
3	U-238	$4.8 \cdot 10^{-2}$
	U-235	$9.6 \cdot 10^{-5}$
4	Fe	$8.5 \cdot 10^{-2}$
5	Void	-

G. Safety During Accident Conditions

G.1 Unflooded Fuel

For the postulated accident conditions some breakage of the fuel elements has to be expected with possible accumulation of fuel particles in a corner of the cask. Detailed drop test data for irradiated fuel elements, however, are not available and some assumptions have to be made concerning the amount of particles released from the fuel elements. Assuming an upper limit of 20% particle loss and accumulation of these particles in coolant holes and the space between container wall and fuel elements at the bottom of the container, the multiplication constant is calculated to be less than .41, with the cask completely immersed in water.

In the analysis it was assumed that the fuel particles fall out of the fuel channels into the coolant channels and spaces between element and container wall and form a homogeneous mixture with the remaining fuel element graphite. At 10% particle loss the lowest 24 cm of the cask would be filled with fuel particles and fuel elements. The remaining elements would have a correspondingly reduced fuel loading. At 20% particle loss, the lowest 48 cm would be filled with particles. Material concentrations for the different fuel regions are shown in Table IX-4 consistent with the geometrical model of Fig. IX-3.

The following multiplication constants were calculated with the two-dimensional model:

No particle loss	$k_{eff} = .37$
10% particle loss	$k_{eff} = .37$
20% particle loss	$k_{eff} = .41$

The first result agrees well with the data from the one-dimensional transport calculation. The other results show that partial particle loss from the fuel elements and subsequent particle accumulation at the bottom of the cask do not significantly increase the multiplication constant. Even if the whole cask

Table IX-4

Concentrations For Most Reactive Condition (Unflooded)

Region	Material	Concentration (atoms/b-cm) 10% Particle Loss	20% Particle Loss
1	Th-232	$2.4 \cdot 10^{-4}$	$2.14 \cdot 10^{-4}$
	U-235	$2.75 \cdot 10^{-5}$	$2.45 \cdot 10^{-5}$
	U-238	$1.63 \cdot 10^{-6}$	$1.45 \cdot 10^{-6}$
	Si	$8.21 \cdot 10^{-4}$	$7.30 \cdot 10^{-4}$
	C	$5.01 \cdot 10^{-2}$	$4.91 \cdot 10^{-2}$
2	Th-232	$7.41 \cdot 10^{-4}$	
	U-235	$8.49 \cdot 10^{-5}$	
	U-238	$5.02 \cdot 10^{-6}$	
	Si	$2.53 \cdot 10^{-3}$	
	C	$6.92 \cdot 10^{-2}$	
3	Fe	.085	
4	U-238	.0478	
	U-235	.000096	
5	H	.0668	
	O	.0334	

Height of Particle Accumulation: 10% loss 24 cm

20% loss 48 cm

were filled with particles only, the overall multiplication constant of the immersed cask would be less than .55. The presence of water at the outside of the cask has no significant effect on the criticality of the system. The iron and uranium shield acts as a sink for thermal neutrons. Fast neutrons escaping from the fuel region are moderated in the water and absorbed before they can return to the fuel.

G.2 Flooded Fuel

In order to obtain an upper limit for the multiplication constant of the shipping cask under flooded conditions the following assumptions were made in addition to those stated in Section B:

1. All fuel particles leave the fuel elements (100% particle loss).
2. The fuel particles accumulate in the fuel holes, coolant holes and void spaces between fuel elements and container wall.
3. The graphite structure of the fuel elements stays intact allowing the highest concentration of fissile material in the available void space.
4. All void space between the fuel particles and in the unfueled section of the cask is filled with water.

In particular, the following cases were considered:

Case 1: All fuel particles accumulate at the bottom of the cask, filling the available void space. The particle packing fraction is .65. The overall height of the fueled section is 171 cm.

Cases 2, 3, 4: The fuel particles float in water. A homogeneous mixture of water and particles occupies the available void space up to a height of 214 cm, 257 cm and 343 cm, respectively.

Case 5: The fuel particles float in water. A homogeneous mixture of water and particles fills all available void space in the shipping cask.

Material concentrations for these 5 cases consistent with Fig. IX-3 are given in Table IX-5. The calculated multiplication constants for these cases are as

Table IX-5

Concentrations for Most Reactive Flooded Conditions

(Calculational Model of Fig. IX-3)

Region	Material	Concentrations (atoms/b-cm)				
		Case I	Case 2	Case 3	Case 4	Case 5
1	C	4.09-2	4.09-2	4.09-2	4.09-2	-
	H	3.47-2	3.47-2	3.47-2	3.47-2	-
	[illegible]	1.74-2	1.74-2	1.74-2	1.74-2	-
2	Th-232	7.41-4	5.93-4	4.94-4	3.71-4	2.67-4
	U-235	8.49-5	6.79-5	5.66-5	4.25-5	3.06-5
	U-238	5.02-6	4.02-6	3.35-6	2.51-6	1.81-6
	Si	2.53-3	2.03-3	1.69-3	1.27-3	9.12-4
	C	6.92-2	6.35-2	5.98-2	5.50-2	5.11-2
	H	1.21-2	1.66-2	1.96-2	2.34-2	2.65-2
	O	6.05-3	8.30-3	9.80-3	1.17-2	1.33-2

Other regions as in Table IX-4.

follows:

Fuel Containing Section

Case	Height (cm)	H/U-235	k_{eff}
1	171	143	.84
2	214	244	.89
3	257	346	.89
4	343	551	.85
5	476	866	.77

Fig. IX-4 shows these results graphically. The most reactive situation, a multiplication constant of about .9, is obtained if the total fuel contents, in a mixture of water and particles, occupies the available void spaces of about half the cask (235 cm). From these results it is concluded that no critical arrangement can occur even if all the fuel leaves the fuel elements and the cask is completely flooded.

H. Accuracy of Calculations

The calculational methods used for the criticality analysis are essentially the same as those used for HTGR design and the analysis of the HTGR critical facility. Comparisons of experimental and calculational results are well documented in the Final Hazards Report for the Fort St. Vrain Reactor (Section 3.5.7). These results indicate that the accuracy of the methods used is in the order $\pm .02\Delta k$.

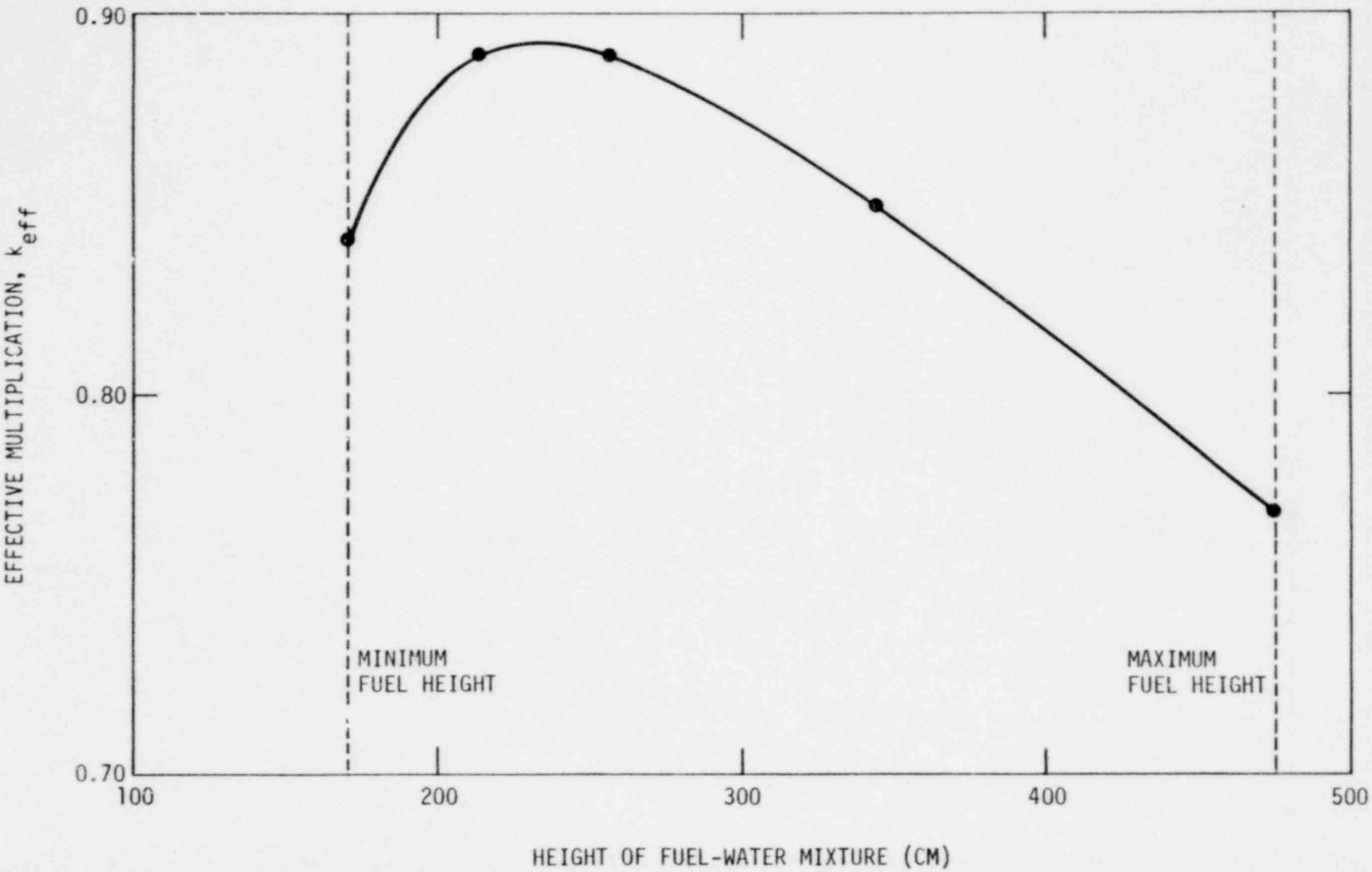

Figure IX - 4 Internal Water Flooding of Shipping Cask. Cask Loadings: 8.4 Kg of 93.5% Enriched Uranium, 68 Kg of Th-232

I. CODE ABSTRACTS

GGC4 2/6/69

1. NAME OR DESIGNATION OF PROGRAM - GGC4

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED AND OTHERS UPON WHICH IT IS OPERABLE - UNIVAC 1108

3. NATURE OF PHYSICAL PROBLEM SOLVED - THE GGC4 PROGRAM SOLVES THE MULTIGROUP SPECTRUM EQUATIONS WITH SPATIAL DEPENDENCE REPRESENTED BY A SINGLE POSITIVE INPUT BUCKLING. BROAD GROUP CROSS SECTIONS (SHIELDED OR UNSHIELDED) ARE PREPARED FOR DIFFUSION AND TRANSPORT CODES BY AVERAGING WITH THE CALCULATED SPECTRA OVER INPUT-DESIGNATED ENERGY LIMITS. THE CODE IS DIVIDED INTO THREE MAIN PARTS. A FAST (GAM) SECTION WHICH COVERS THE ENERGY RANGE FROM 14.9 MEV TO 0.414 EV, A THERMAL (GATHER) SECTION WHICH COVERS THE ENERGY RANGE FROM 0.001 TO 2.38 EV, AND A COMBINING (COMBO) SECTION WHICH COMBINES FAST AND THERMAL CROSS SECTIONS INTO SINGLE SETS. BASIC NUCLEAR DATA FOR THE FAST SECTION WHICH CONSISTS OF FINE GROUP-AVERAGED CROSS SECTIONS AND RESONANCE PARAMETERS IS READ FROM A DATA TAPE. THE FINE GROUP ABSORPTION AND FISSION CROSS SECTIONS MAY BE ADJUSTED BY PERFORMING A RESONANCE INTEGRAL CALCULATION. UTILIZING A FISSION SOURCE AND AN INPUT BUCKLING, THE CODE SOLVES THE P1, B1, B2, OR B3 APPROXIMATION TO OBTAIN THE ENERGY-DEPENDENT FAST SPECTRUM. TWO OR SIX SPATIAL MOMENTS OF THE SPECTRUM (DUE TO A PLANE SOURCE) MAY ALSO BE EVALUATED. INSTEAD OF PERFORMING A SPECTRUM CALCULATION, THE USER MAY ENTER THE LEGENDRE COMPONENTS OF THE ANGULAR FLUX DIRECTLY. FOR AS MANY INPUT-DESIGNATED BROAD GROUP STRUCTURES AS DESIRED, THE CODE CALCULATES AND SAVES (FOR THE COMBINING SECTION) SPECTRUM WEIGHTED AVERAGES OF MICROSCOPIC AND MACROSCOPIC CROSS SECTIONS AND TRANSFER ARRAYS. SLOWING DOWN SOURCES ARE CALCULATED AND SAVED FOR USE IN THE LOWER ENERGY RANGE. GIVEN BASIC NUCLEAR DATA, THE THERMAL SECTION OF GGC4 DETERMINES A THERMAL SPECTRUM BY EITHER READING IT AS INPUT, BY CALCULATING A MAXWELLIAN SPECTRUM FOR A GIVEN TEMPERATURE, OR BY AN ITERATIVE SOLUTION OF THE P0, B0, P1, OR B1 EQUATIONS FOR AN INPUT BUCKLING. TIME MOMENTS OF THE TIME AND ENERGY-DEPENDENT DIFFUSION EQUATIONS ARE CALCULATED (AS AN OPTION) USING THE INPUT BUCKLING TO REPRESENT LEAKAGE. BROAD GROUP CROSS SECTIONS ARE PREPARED BY AVERAGING FINE GROUP CROSS SECTIONS OVER THE CALCULATED SPECTRA. BROAD GROUP STRUCTURES ARE READ AS INPUT. THE COMBINING SECTION OF GGC4 TAKES THE BROAD GROUP-AVERAGED CROSS SECTIONS FROM THE FAST AND THERMAL PORTIONS OF GGC4 AND FORMS MULTIGROUP CROSS SECTION TABLES. THESE TABLES ARE PREPARED IN STANDARD FORMATS FOR TRANSPORT OR DIFFUSION THEORY CALCULATIONS. IN ADDITION, IT IS POSSIBLE TO USE THE COMBINING SECTION TO PRODUCE MIXTURES NOT USED IN THE SPECTRUM CALCULATION OR TO COMBINE

THE RESULTS OF DIFFERENT FAST AND THERMAL SECTION CALCULATIONS AND SO ON. THESE OPTIONS ARE DESCRIBED IN REFERENCE 2.

4. METHOD OF SOLUTION - IN THE FAST SECTION EITHER THE P1 OR THE B1, B2, OR B3 APPROXIMATION IS MADE TO THE TRANSPORT EQUATION USING THE POSITIVE, ENERGY-INDEPENDENT BUCKLING. IN EACH APPROXIMATION LEGENDRE MOMENTS OF THE ANGULAR FLUX ARE COMPUTED BY DIRECT NUMERICAL INTEGRATION OF THE SLOWING DOWN EQUATIONS. IN THE RESONANCE CALCULATIONS, DOPPLER BROADENED (AT AN INPUT TEMPERATURE) ABSORPTION AND SCATTERING CROSS SECTIONS ARE USED. THE RESONANCE TREATMENT ALLOWS UP TO TWO ADMIXED MODERATORS IN AN ABSORBER LUMP IMBEDDED IN A SURROUNDING MODERATOR. THE ABSORBER IN THE LUMP IS TREATED BY USING EITHER THE NARROW RESONANCE APPROXIMATION, THE NARROW RESONANCE INFINITE MASS APPROXIMATION, OR A SOLUTION OF THE SLOWING DOWN INTEGRAL EQUATIONS TO DETERMINE THE COLLISION DENSITY THROUGH THE RESONANCE. THE ADMIXED MODERATORS ARE TREATED BY USING EITHER AN ASYMPTOTIC FORM OF, OR AN INTEGRAL EQUATION SOLUTION FOR, THE COLLISION DENSITY. IN THE RESONANCE CALCULATION EITHER STANDARD GEOMETRY COLLISION PROBABILITIES ARE USED OR TABLES OF COLLISION PROBABILITIES ARE ENTERED. DANCOFF CORRECTIONS CAN ALSO BE MADE. IN THE REGION OF UNRESOLVED RESONANCES, RESONANCE ABSORPTION IS CALCULATED BY USING PORTER-THOMAS DISTRIBUTIONS, BUT ONLY S-WAVE NEUTRONS ARE CONSIDERED. IN THE THERMAL SECTION EITHER THE B0, B1, P0, OR P1 APPROXIMATION TO THE TRANSPORT EQUATION IS MADE, AND IN ALL OPTIONS LEGENDRE MOMENTS OF THE ANGULAR FLUX ARE COMPUTED. A TRAPEZOIDAL ENERGY INTEGRATION MESH IS USED, AND THE RESULTING EQUATIONS ARE SOLVED ITERATIVELY BY USING A SOURCE-NORMALIZED, OVER-RELAXED, GAUSSIAN TECHNIQUE. AVERAGES OVER BROAD GROUPS ARE PERFORMED BY SIMPLE NUMERICAL INTEGRATION. THE RESULTS OBTAINED IN THE FAST AND THERMAL SECTIONS ARE STORED ON SPECIAL TAPES. THESE TAPES MAY CONTAIN RESULTS FOR A NUMBER OF PROBLEMS, EACH PROBLEM INCLUDING FINE GROUP CROSS SECTION DATA FOR A NUMBER OF NUCLIDES. IF THE PROBLEM NUMBER IS SPECIFIED ON THESE TAPES, AND A DESIRED LIST OF NUCLIDES IS GIVEN, THE COMBINING CODE WILL PUNCH MICROSCOPIC CROSS SECTIONS FOR THE REQUESTED LIST OF NUCLIDES. THE PROGRAM ALSO TREATS MIXTURES. GIVEN THE ATOMIC DENSITIES OF THE NUCLIDES IN A MIXTURE, THE CODE WILL PUNCH MACROSCOPIC CROSS SECTIONS.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - MAXIMA OF -
   - 99 FAST GROUPS
   - 101 THERMAL FINE GROUPS
   - 99 FAST BROAD GROUPS
   - 50 THERMAL BROAD GROUPS
   - 50 BROAD GROUPS IN THE COMBINING SECTION
   - 250 RESONANCES PER NUCLIDE
   - 2 MODERATORS ADMIXED WITH A RESONANCE ABSORBER
   - 305 ENTRIES IN THE ESCAPE PROBABILITY TABLE FOR CYLINDRICAL GEOMETRIES

505 ENTRIES IN THE ESCAPE PROBABILITY TABLE FOR SLAB GEOMETRIES A SINGLE AND POSITIVE VALUE FOR THE BUCKLING (B2) MUST BE SUPPLIED.

6. TYPICAL RUNNING TIME - A B1 CALCULATION IN THE FAST SECTION FOR 3 NUCLIDES AND 6 BROAD GROUPS TAKES APPROXIMATELY 4 MINUTES ON THE UNIVAC-1108 IF A RESONANCE CALCULATION (1/2 MINUTE) IS PERFORMED FOR ONE NUCLIDE. THE THERMAL CALCULATION FOR 3 BROAD GROUPS REQUIRES APPROXIMATELY 2 MINUTES, WHICH INCLUDES ABOUT 7 SECONDS FOR THE ITERATIVE PROCEDURE. TO PUNCH STANDARD DIFFUSION AND STANDARD TRANSPORT CROSS SECTIONS FOR THIS PROBLEM REQUIRES 2 SECONDS.

7. UNUSUAL FEATURES OF THE PROGRAM - THERE IS AN OPTION IN GGC4 WHICH MAKES IT POSSIBLE TO SHORTEN THE PUNCHING PROCESS FOR LARGE TWO-DIMENSIONAL TRANSFER ARRAYS. THIS CAN BE DONE BY SPECIFYING A MAXIMUM NUMBER OF DESIRED UPSCATTERING AND DOWNSCATTERING TERMS.

8. RELATED AND AUXILIARY PROGRAMS - GGC4 IS A REVISION OF THE EARLIER PROGRAM, GGC3. TO PREPARE, HANDLE, AND UPDATE THE BASIC CROSS SECTION TAPES WHICH ARE USED AS INPUT FOR GGC4, THE FOLLOWING CODES ARE UTILIZED - MAKE, MST, PRINT, MIXER, WTFG, MGT3, SPRINT, COMBIN, AND DOP.

9. STATUS - PRODUCTION

10. REFERENCES - J. ADIR AND K. D. LATHROP, THEORY OF METHODS USED IN GGC-3 MULTIGROUP CROSS SECTION CODE, GA-7156, JULY 1967.
J. ADIR, S. S. CLARK, R. FROEHLICH, AND L. J. TODT, USERS AND PROGRAMMERS MANUAL FOR GGC-3 MULTIGROUP CROSS SECTION CODE, PARTS 1 AND 2, GA-7157, JULY 1967.
M. K. DRAKE, DESCRIPTION OF AUXILIARY CODES USED IN THE PREPARATION OF DATA FOR THE GGC-3 CODE, GA-7158, AUGUST 1967.
J. ADIR, GGC4 INPUT INSTRUCTIONS, GA MEMORANDUM, JUNE 20, 1968.
BCDCON, GA NOTE.

11. MACHINE REQUIREMENTS - 64K MEMORY WITH 11 TAPE UNITS (SOME OF WHICH MAY BE DRUM AREAS).

12. PROGRAMMING LANGUAGE USED - FORTRAN IV

13. OPERATING SYSTEM OR MONITOR UNDER WHICH PROGRAM IS EXECUTED - UNIVAC EXEC II, GAX29.

14. ANY OTHER PROGRAMMING OR OPERATING INFORMATION OR RESTRICTIONS - THERE IS NO RESTRICTION ON THE NUMBER OF PROBLEMS THAT CAN BE RUN CONSECUTIVELY IN EACH SECTION, NOR IS THERE A RESTRICTION ON THE NUMBER OF NUCLIDES PER PROBLEM. WITHOUT USING THE NEW OPTIONS, GGC4 CAN ALSO BE RUN WITH THE GGC3 INPUT INSTRUCTIONS.

15. AUTHOR OF THE ABSTRACT - J. ADIR

1DF 9/69

1. NAME OF PROGRAM - 1DF

2. COMPUTER FOR WHICH CODE IS DESIGNED - UNIVAC-1108, IBM-7030, CDC-6600, IBM-7094.

3. NATURE OF PHYSICAL PROBLEM SOLVED - THE LINEAR TIME-INDEPENDENT BOLTZMANN EQUATION FOR PARTICLE TRANSPORT IS SOLVED FOR THE ENERGY, SPACE AND ANGULAR DEPENDENCE OF THE PARTICLE DISTRIBUTION IN 1-D SLABS, CYLINDERS, AND SPHERES. INDEPENDENT SOURCE OR EIGENVALUE (MULTIPLICATION, TIME ABSORPTION, ELEMENT CONCENTRATION, ZONE THICKNESS, OR SYSTEM DIMENSION) PROBLEMS ARE SOLVED SUBJECT TO VACUUM, REFLECTIVE, PERIODIC, OR WHITE BOUNDARY CONDITIONS. A COMPLETE ENERGY TRANSFER SCATTERING MATRIX IS ALLOWED FOR EACH LEGENDRE COMPONENT OF SCATTERING. SOLUTIONS TO THE ADJOINT TRANSPORT EQUATION ARE ALSO OBTAINED.

4. METHOD OF SOLUTION - ENERGY DEPENDENCE IS TREATED BY THE MULTIGROUP APPROXIMATION AND ANGULAR DEPENDENCE BY A GENERAL DISCRETE ORDINATES APPROXIMATION. ANISOTROPIC SCATTERING IS APPROXIMATED BY A TRUNCATED SPHERICAL HARMONICS EXPANSION OF THE SCATTERING KERNEL. WITHIN-GROUP SCATTERING AND UPSCATTERING ITERATION PROCESSES ARE ACCELERATED BY SYSTEM-WIDE RENORMALIZATION PROCEDURES. CHEBYSHEV ACCELERATION IS AUTOMATICALLY APPLIED TO ACCELERATE INNER ITERATION CONVERGENCE. AT THE OPTION OF THE USER, CHEBYSHEV ACCELERATION FACTORS CAN BE ENTERED AS INPUT. APPROXIMATIONS AND ITERATIVE CYCLES HAVE BEEN DESCRIBED IN DETAIL BY LATHROP (REF. 1, BELOW). 1DF AND DTF-IV ARE ESSENTIALLY THE SAME.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM - THE VARIABLE DIMENSIONING CAPABILITY OF FORTRAN IV HAS BEEN UTILIZED SO THAT ANY COMBINATION OF NUMBER OF GROUPS, NUMBER OF SPATIAL INTERVALS, SIZE OF ANGULAR QUADRATURE, ETC., CAN BE USED THAT WILL FIT WITHIN THE TOTAL CORE STORAGE AVAILABLE TO A USER.

6. TYPICAL RUNNING TIME - A FEW MINUTES.

7. UNUSUAL FEATURES OF THE PROGRAM - ANISOTROPIC DISTRIBUTED SOURCES MAY BE USED AND THE INCOMING ANGULAR FLUX AT THE RIGHT BOUNDARY MAY BE SPECIFIED.

8. RELATED AND AUXILIARY PROGRAM - GTF, 2DF, AND TWOTRAN.

9. STATUS - PRODUCTION

10. REFERENCES - LATHROP, K.D., 'DTF-IV, A FORTRAN-IV PROGRAM FOR SOLVING THE MULTIGROUP TRANSPORT EQUATION WITH ANISOTROPIC SCATTERING, 'USAEC REPORT LA-3373, LOS ALAMOS SCIENTIFIC LABORATORY, 1965.

11. MACHINE REQUIREMENTS - NO SPECIAL REQUIREMENTS AND NO AUXILIARY STORAGE REQUIRED.

12. PROGRAMMING LANGUAGE - FORTRAN IV

13. OPERATING SYSTEM - UNIVAC EXEC II

14. OTHER PROGRAMMING INFORMATION - ALL STORAGE REQUIREMENTS ARE COMPUTED IN THE MAIN PROGRAM. BY PERFORMING THE ADDITIONS INDICATED BY THE ALGORITHM, THE PRECISE AMOUNT OF STORAGE REQUIRED FOR A PROBLEM CAN BE DETERMINED.

15. AUTHOR OF THE ABSTRACT - TAKEN FROM THE BOOK 'NUCLEAR DESIGN METHODS IN USE AT GENERAL ATOMIC.' MODIFIED SLIGHTLY 9-1-69.

GAMBLE5

1. NAME OF CODE - GAMBLE-5, A PROGRAM FOR THE SOLUTION OF THE MULTIGROUP NEUTRON DIFFUSION EQUATIONS IN TWO DIMENSIONS WITH ARBITRARY GROUP SCATTERING.

2. COMPUTER FOR WHICH PROGRAM IS DESIGNED - UNIVAC 1108

3. NATURE OF PHYSICAL PROBLEM SOLVED - THE HOMOGENEOUS TWO-DIMENSIONAL MULTIGROUP DIFFUSION THEORY EQUATIONS WITH ARBITRARY GROUP-TO-GROUP SCATTERING AND ARBITRARY FISSION TRANSFER ARE SOLVED FOR HETEROGENEOUS ASSEMBLIES IN X-Y AND R-Z GEOMETRY. HOMOGENEOUS LOGARITHMIC BOUNDARY CONDITIONS ARE USED AT THE OUTER SURFACE OF THE ASSEMBLY AND AT THE SURFACE OF NONDIFFUSION REGIONS. THE RESULTS INCLUDE THE GROUP- AND POINT-DEPENDENT NEUTRON FLUXES, THE POWER DISTRIBUTION, THE NEUTRON MULTIPLICATION FACTOR (K EFFECTIVE), AND A DETAILED NEUTRON BALANCE.

4. METHOD OF SOLUTION - THE MULTIGROUP DIFFUSION THEORY EQUATIONS ARE APPROXIMATED BY FIVE-POINT DIFFERENCE EQUATIONS FOR AN ARBITRARY NONUNIFORM MESH GRID. THE SYSTEM OF DIFFERENCE EQUATIONS IS SOLVED BY AN EXTENSION OF THE POWER METHOD TO FIND THE EIGENVECTOR (NEUTRON FLUX) AND THE EIGENVALUE (K EFFECTIVE). SUCCESSIVE LINE OVERRELAXATION IS APPLIED IN A SPECIAL FORM (EXPOENTIAL OVERRELAXATION) THAT GUARANTEES THE NON-NEGATIVITY OF THE NEUTRON FLUX. COARSE MESH REBALANCING IS USED TO IMPROVE THE PREASYMPTOTIC CONVERGENCE BEHAVIOR. A VARIATION OF AITKENS' METHOD IS USED TO IMPROVE THE ASYMPTOTIC CONVERGENCE BEHAVIOR, ASSUMING ONLY ONE ERROR MODE.

5. RESTRICTIONS ON THE COMPLEXITY OF THE PROBLEM -
   MAXIMUM NUMBER OF ENERGY GROUPS 10
   MAXIMUM NUMBER OF SPACE MESHPOINTS 20,000
   MAXIMUM NUMBER OF DIFFERENT MATERIAL REGIONS 255

6. TYPICAL RUNNING TIME - A SEVEN-GROUP PROBLEM (THREE FAST GROUPS AND FOUR THERMAL GROUPS) IN (R,Z) GEOMETRY WITH 2842 SPACE MESH POINTS TOOK 82 ITERATIONS ASSUMING A TIGHT CONVERGENCE CRITERIA (MAXIMUM RELATIVE FLUX CHANGE LESS THAN 0.00007). THE TOTAL RUNNING TIME (INCLUDING EXTENSIVE OUTPUT) ON THE UNIVAC 1108 WAS 12 MIN.

7. UNUSUAL FEATURES OF THE PROGRAM -
   A) THE COARSE MESH REBALANCING SCHEME MAKES POSSIBLE THE SUCCESSFUL SOLUTION OF DIFFICULT PROBLEMS FOR WHICH CERTAIN GROUP-MESH POINTS ARE BOTH STRONGLY AND WEAKLY COUPLED TO SOME OF THEIR NEIGHBORS (E.G., HIGHLY NONUNIFORM MESH SPACINGS OR MATERIAL PROPERTIES, AIR GAPS, CELL PROBLEMS WITH WEAK GROUP COUPLING, ETC.).
   B) SIMULTANEOUS PERFORMANCE OF COMPUTATION AND DATA TRANSFER

POOR ORIGINAL

WITH VIRTUALLY NO DELAY CAUSED BY THE USE OF DRUM STORAGE. D) ABILITY TO DO EFFICIENT RESTARTS FOR LONGER RUNNING PROBLEMS AND THE ABILITY TO ACCEPT A FLUX GUESS ON TAPE FROM A SIMILAR PROBLEM.

8. RELATED AND AUXILIARY PROGRAMS - GAMBLE-5 IS A MAJOR REVISION OF THE GAMBLE-4 CODE. SOME OF THE ESSENTIALS OF THE ITERATIVE TECHNIQUE USED HAVE BEEN ADOPTED FROM EXTERMINATOR.

9. STATUS - THE PROGRAM HAS BEEN IN PRODUCTION USE SINCE AUGUST 1967 AND MAY BE OBTAINED BY DOMESTIC USERS FROM THE ARGONNE CODE CENTER.

10. REFERENCES -

1) J.P. DORSEY AND R. FROEHLICH, 'GAMBLE-5, A PROGRAM FOR THE SOLUTION OF THE MULTIGROUP NEUTRON DIFFUSION EQUATIONS IN TWO DIMENSIONS WITH ARBITRARY GROUP SCATTERING FOR THE UNIVAC 1108 COMPUTER,' GA-8188, GULF GENERAL ATOMIC INC. (1967)
2) R. FROEHLICH, 'A THEORETICAL FOUNDATION FOR COARSE MESH VARIATION TECHNIQUES,' PROC. INTERN. CONF. RES. REACTOR UTILIZATION AND REACTOR MATH. MEXICO, D.F., 1, 219(1967).
3) J.P. DORSEY, 'GAMBLE-4, A PROGRAM FOR THE SOLUTION OF THE MULTIGROUP NEUTRON DIFFUSION EQUATIONS IN TWO-DIMENSIONS WITH ARBITRARY GROUP SCATTERING FOR THE IBM 7044 FORTRAN-IV SYSTEM,' GA-6540, GULF GENERAL ATOMIC INC. (1965).
4) T.B. FOWLER, M. TOBIAS, AND D. VONDY, 'EXTERMINATER, A MULTIGROUP CODE FOR SOLVING NEUTRON DIFFUSION EQUATIONS IN ONE AND TWO DIMENSIONS,' ORNL-TM-842, OAK RIDGE NATIONAL LABORATORY (1965).

11. MACHINE REQUIREMENTS - 65,536 WORDS OF CORE STORAGE, 3 TAPE UNITS ON 1 DATA CHANNEL, 1,572,864 WORDS OF FH-880 DRUM STORAGE FROM 1 DATA CHANNEL, AND A PERIPHERAL PRINTER.

12. PROGRAMMING LANGUAGE USED - FORTRAN IV, BUT FOR SCRATCH DATA HANDLING USE IS MADE OF UNIVAC 1108 ASSEMBLY LANGUAGE.

13. OPERATING SYSTEM - EXECII, GAX 23.

14. OTHER PROGRAMMING INFORMATION -

15. AUTHOR OF THE ABSTRACT - J.P. DORSEY AND R. FROEHLICH.
GULF GENERAL ATOMIC INCORPORATED
P.O. BOX 608
SAN DIEGO, CALIFORNIA 92112

SECTION X

MANUFACTURING AND QUALITY CONTROL PROCEDURES AND FABRICATION PLAN

NATIONAL LEAD CO. NUCLEAR DIVISION

WILMINGTON PLANT

MANUFACTURING AND QUALITY CONTROL PROCEDURES AND FABRICATION PLAN

A. MANUFACTURING AND QUALITY CONTROL PROCEDURES

All steel materials for the Spent Fuel Container and Shipping Cask will be purchased to the requirements of the ASME Code, 1968 Edition, Section III, Class B. Chemical and/or physical test reports will be provided for all materials. Welding will be performed by welders qualified to Section IX, ASME Code. Welding procedures will also be qualified in accordance with Section IX of the Code.

Detailed manufacturing procedures will be prepared outlining step by step sequences for all fabrication, machining, assembly operations, and quality control. Inspection system will be in accordance with Appendix IX of the ASME Code. All three point decimal dimensions and critical fractional dimensions will be specified in the manufacturing procedure to be recorded. All out of tolerance conditions will be recorded and customer approval obtained before acceptance. Mandatory hold and inspection points will be incorporated into the Manufacturing/QC Procedures in order that the GGA/QA Representative and the Authorized Code Inspector may witness tests and inspections being performed in accordance with ASME Code requirements.

Nondestructive testing techniques and acceptance standards will be in accordance with ASME Code Section III and will be performed by personnel qualified to ASME Code. Welds to be inspected and stages of inspection will be as shown on design drawings, and in accordance with the ASME Code.

Uranium castings will be provided by NLC Albany. Inspection of Uranium castings will be made by the Gamma Probing Method.

Process for welding uranium sections will meet design criteria for shielding value and structural strength. Nondestructive testing of uranium welds will be for purpose of detecting linear defects and will be by penetrant method.

Inspection operations, inspection records, radiographs and nondestructive test records will be audited by the GGA/QA Representative.

DATE ________ PAGE ____ OF ____

B. SHIPPING CASK FABRICATION PLAN

The top and bottom stainless steel forgings will be premachined to the extent necessary to allow fit-up with the inner liner and outer shell.

Both the inner liner and outer shell will be machined, one-piece, cast steel pipes.

The depleted uranium shielding will initially be five cast depleted uranium cylinders approximately 38 inches long by 26 inches in diameter and 3-1/2 inch wall thickness. Each cylindrical section will be inspected for integrity by gamma probing.

Basically, the fabrication plan will be as follows·

1. The final machined inner liner will be welded to the premachined bottom forging.

2. The surfaces of this weldment to be in contact with the depleted uranium will be copper spray coated.

3. The final machined depleted uranium shielding cylindrical shells will be installed individually over the inner liner. The shielding sections will be welded to obtain a one-piece shield. Joints of the assembled shield will then be inspected for integrity by gamma probing.

4. The outer shell, machined to acquire tolerances necessary for assembly over the depleted uranium shield, will be installed and welded to the bottom forging.

5. The premachined top ring forging will then be installed and welded to the inner liner and outer shell.

6. All final machining to the bottom and top forgings will then be completed.

C. CLOSURE HEAD FABRICATION PLAN

The top and bottom forgings will be premachined.

The cylinder is welded to the top forging. All surfaces that will be in contact with the depleted uranium will be copper sprayed.

The uranium shield will be inspected for integrity by gamma probing.

The uranium shield is installed in the cylinder top forging sub-assembly.

The bottom plate is welded to the assembly.

The assembly is then completely machined on all surfaces including drilling.

D. SPENT FUEL CONTAINER FABRICATION PLAN

The cylinder with top flange will be purchased completely welded and machined in one piece.

The bottom forging is welded to the cylinder and the assembly is machined to the OD.

Final machining of the top forging is then completed including transfer drilling from the lid.

SECTION XI

PRELIMINARY LOW TEMPERATURE URANIUM DROP TESTS

SECTION XI

NATIONAL LEAD CO. NUCLEAR DIVISION
WILMINGTON PLANT

PRELIMINARY LOW TEMPERATURE URANIUM DROP TESTS

A. SUMMARY

In an effort to obtain a preliminary indication of the ductility properties of uranium at sub-zero temperatures, a series of drop tests was scheduled and conducted at National Lead Company's Albany branch. mples of unalloyed depleted were tested along with various samples of low alloys to serve as a basis for comparison.

Various diameters of as cast depleted uranium round bar were used; however, each round bar was cut to length such that the length to diameter ratio was 8 to 1 - this being approximately the same as the L/D ratio of the depleted uranium in the Fort St. Vrain Shipping Cask.

Most drops were conducted from a height of 29'3" onto essentially unyielding surfaces of either steel or concrete. Two samples, also from a height of 29'3", were dropped or a sharp edged fulcrum. Temperature at the time of drop of all samples ranged between -55°F to -60°F.

Results of these preliminary drop tests indicated that the unalloyed depleted uranium exhibited good ductility properties at low temperatures. Furthermore, it wasn't until the third 29'3" drop of a test specimen that ductility failure occurred. A 1-3/16" diameter by 9-1/2" long unalloyed depleted uranium round bar was dropped on a concrete impact surface with no visible failure resulting. The same test specimen was then dropped from the same height of 29'3" and at the same temperature of -60°F on a steel plate. This time a slight bend in the bar was noted. On the third test, the specimen was dropped on the sharp edge of a steel angle. Along with a greater bend in the bar, it was noted that a small crack developed opposite the side of impact.

As stated above, this series of tests is considered preliminary even though favorable results were obtained. More precise testing procedures such as charpy impact tests will be required to obtain accurate quantitative data.

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SECTION XI - PRELIMINARY LOW TEMPERATURE URANIUM DROP TESTS

NATIONAL LEAD CO. NUCLEAR DIVISION
WILMINGTON PLANT

B. RESULTS

TEST NO.	SPECIMEN DROP NO.	MATERIAL (As-Cast)	DIAMETER (Inches)	ATTITUDE AT IMPACT	IMPACTING SURFACE	TEST RESULTS
1	1	U-2% Mo	1-1/4	Horizontal	Concrete	No failure
2	1	Unalloyed	1-3/16	Horizontal	Concrete	No failure
3	1	Unalloyed	1-3/16	Horizontal	Concrete	No failure
4	1	U-1% Mo, 1% Nb	.6	Horizontal	Concrete	No failure
5	1	U-1% Mo, 1% W	.6	45° Corner Drop	Concrete	No failure
6	1	U-1% Nb, 1% W	.6	45° Corner Drop	Concrete	No failure
7	1	U-1% Ta, 1% W	.6	Horizontal	Concrete	No failure
8	1	Unalloyed	.6	Horizontal	Concrete	No failure
9	1	Unalloyed	.6	Horizontal	Concrete	No failure
10	2	Unalloyed	1-3/16	Horizontal	Steel Plate	Slight Bend
11	2	Unalloyed	1-3/16	Horizontal	Steel Plate	Slight Bend
12	2	U-2% Mo	1-1/4	Horizontal	Steel Plate	No Failure
13	2	U-1% Mo, 1% W	.6	Horizontal	Steel Plate	No Failure
14	2	Unalloyed	.6	Horizontal	Steel Plate	Slight Bend
15	3	Unalloyed	1-3/16	Horizontal	90° Corner	Bent & Crk'd.
16	3	Unalloyed	1-3/16	Slight Angle Corner Drop	90° Corner	Bent

Test Conditions

Height of drop	29'3"
Temperature of specimens	-55°F to -60°F
Material Condition	As Cast
Material Configuration	Round Bars, L/D = 8:1

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C. RESULTS OF 1/8 SCALE URANIUM LOW TEMPERATURE IMPACT TESTS

Purpose of Tests

Experimental knowledge is required of the effects of impact loads as used in casks with uranium shielding and having relatively large L/D ratios.

The experimental test specimen was subjected to several puncture tests to determine the amount of deformation and to observe the surface condition of the uranium in the impacted areas.

Material Used

An as cast .2% - .3% molybdenum-uranium cylindrical shell, 1/8 the size of the actual shielding in the shipping cask, was used for this series of tests.

Comparative dimensions and weights:

	1/8 Scale Cylindrical Shell	Full Size Uranium Shielding
Outside Diameter	3-1/4"	26"
Inside Diameter	2-3/8"	19"
Length	24-3/8"	194"
Weight	64 lbs.	32,800 lbs.

Test Procedures

Tests were conducted using the indoor drop test facilities of NLC, Wilmington' branch.

An iron-constantan thermocouple was attached to the outer surface of the uranium cylindrical shell. The thermocouple extension leads were connected to a calibrated pyrometer indicator.

The test specimen was then submerged in a solution of acetone and dry-ice until it reached a temperature of approximately -60° C. With the thermocouple still attached, the uranium was connected to a quick release mechanism which in turn was attached to an overhead crane. The entire assembly was moved over the impact area which consisted of a 3/4" wide carbon steel fulcrum 4" deep and 12" long welded to a 12" x 12" x 3/4" carbon steel base plate. This weldment was resting on a steel anvil pad supported by a concrete foundation. With the use of a scaled line and plumb bob, the test specimen was raised to a height of 40" over the fulcrum. Its long axis was positioned level and perpendicular to the long axis of the fulcrum so that the C.G. of the shell would impact against the fulcrum.

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When the pyrometer indicator measured a surface temperature of -40°, the solenoid on the release mechanism was actuated, pulled a release pin and allowed the test specimen to free fall on the fulcrum.

This same test was conducted at 60", 80", 100" and 120". Measurements were taken after each test of the outside diameter, inside diameter, length and angle of bend.

Results

Results of all free fall fulcrum tests proved negative. The test specimen experienced no dimensional changes or deformations.

The 120" drop test had a rebound after impact of 31-1/4" above the fulcrum-the test specimen remained level during the rebound. Although no damage occurred to the uranium, the 3/4" fulcrum received an indentation approximately 1/4" deep.