PDR 71-9027 11D

Tech /Ops

Radiation Products Division
40 North Avenue
Burlington, Massachusetts 01803
Telephone (617) 272-2000

RECEIVED
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U.S. NUC. REG. COMMISSION MAIL SECTION

29 November 1979

Mr. Charles E. MacDonald, Chief
Transportation Branch
Division of Fuel Cycle and Material Safety
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. MacDonald:

We request renewal of USNRC Certificate of Compliance No. 9027 issued for Technical Operations Models 741 and 741E Type B Packages. In accordance with your letter of 23 July 1979, eight copies of a consolidated application for this package are enclosed. In accordance with 10CFR170.31, Item 11.E, we are also enclosing a check for $150 for the renewal fee.

We are simultaneously applying to the U.S. Department of Transportation for an International Atomic Energy Agency Certificate of Competent Authority issued under the 1973 Revised Edition of IAEA Safety Series No. 6 for Type B(U) packaging.

We trust that this application satisfies your requirements for renewal of this certificate.

Sincerely,

John J. Munro III
Technical Director

1629 233

JJM/fb
Encl.

xc: R.R. Rawl, USDOT

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Renewal
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Tech Ops

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40 North Avenue
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29 November 1979

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Technical Director

1629 234

JJM/fb
Encl.

xc: R.R. Rawl, USDOT

461 C

7912200 ·6 14892

Tech/Ops

Radiation Products Division
40 North Avenue
Burlington, Massachusetts 01803
Telephone (617) 272-2000

29 November 1979

Mr. Richard R. Rawl
Health Physicist
Office of Hazardous Materials Regulation
Materials Transportation Bureau
United States Department of Transportation
Research and Special Programs Administration
Washington, DC 20590

Dear Mr. Rawl:

We request issuance of an International Atomic Energy Agency Certificate of Competent Authority for Type B(U) packaging under the 1973 Revised Edition of IAEA Safety Series No. 6 for Technical Operations Models 741 and 741E Type B Packages, USA/9027/B. We are requesting this certificate due to the serious difficulties we have encountered in transporting this package internationally with its approval based on the 1967 Edition of IAEA Safety Series No. 6.

We are enclosing two complete copies of the package description for the Models 741 and 741E. We are simultaneously applying to the U.S. Nuclear Regulatory Commission for renewal of USNRC Certificate of Compliance No. 9027 issued for this package. Eight copies of this package description have been forwarded to USNRC.

We trust that this request contains the information you require for your review. Your prompt action would be greatly appreciated.

Sincerely,

John J. Munro III
Technical Director

1629 235

JJM/fb
Encl.

xc: C.E. MacDonald, USNRC

Tech/Ops

Radiation Products Division
40 North Avenue
Burlington, Massachusetts 01803
Telephone (617) 272-2000

PACKAGE DESCRIPTION
TECHNICAL OPERATIONS
MODEL 741
USA/9027/B

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1. General Information

1.1 Introduction

The Tech/Ops Models 741 and 741E are designed for use as gamma ray projectors and shipping containers for Type B quantities of radioactive material in special form. The Model 741 differs from the Model 741E only by the addition of an electric circuit, which provides compatibility with Tech/Ops Model 657 Automatic Exposure Device. Throughout this evaluation, the Models 741 and 741E are considered interchangeable, except where specifically designated.

The Model 741 conforms to the criteria for Type B packaging in accordance with 10CFR71 and satisfies the criteria for Type B(U) packaging in accordance with IAEA Safety Series No. 6, 1973. The sources to be used in conjunction with the Model 741 are Tech/Ops sealed source assemblies Models Nos. A424-9 and A424-18. The source assemblies will contain a maximum of 240 curies of iridium-192 and 33 curies of cobalt-60 respectively, as special form.

1.2 Package Description

1.2.1 Packaging

The Model 741 is 11.25 inches (286mm) high, 19.1 inches (486mm) long, and 14 inches (365mm) wide in overall dimension. The gross weight of the package is 300 pounds (136kg). The radioactive source assembly is stored in a zircalloy or titanium "S" tube in the geometric center of the package. The weight of the uranium shield is 200 pounds (91kg). The shield is provided with a paint finish.

The shield is enclosed in a shell fabricated of $\frac{1}{4}$ inch (6.35mm) thick hot rolled steel. The shield is fixed in position within the shell by the retaining bar assemblies. The void space between the shield and the shell is filled with a castable rigid polyurethane foam. Steel-uranium interfaces are separated with 0.010 inch (0.254mm) thick copper separators.

Attached to the sides of the container are 0.625 inch (15.9mm) thick hot rolled steel side frames used for lifting the package.

Mounted at each end of the "S" tube are positioning devices. The source assembly is locked in position by means of the control cable connector and additionally secured by means of a shipping plug. A protective shipping plate ($\frac{1}{4}$ inch thick steel) is mounted over the control cable assembly.

Tamperproof seals are provided during shipment of these sources. Assembly joints which are not leak-tight provide passageways for the escape of any gas generated from decomposition of the potting foam in the event the projector is involved in a fire accident. The outer packaging is designed to avoid the collection and retention of water. The package is painted and finished to provide for easy decontamination. The radioactive material is sealed inside a source capsule, which is the containment vessel of the package.

. The Model 741 has been previously approved for use as a Type B package under USNRC Certificate of Compliance No. 9027, Rev. 1 (enclosed in Section 1.3).

1.2.2 Operational Features

The source assembly is secured in the proper position by the control cable connector and lock assembly. This assembly requires a key for operation, and thus provides positive closure. A $\frac{1}{4}$ inch (6.35mm) thick steel shipping plate is used to protect the assembly during shipment. Additionally, the source assembly is secured by means of a shipping plug inserted in the opposite end of the "S" tube. This plug is seal wired and provided with a tamperproof seal.

1.2.3 Contents of Packaging

The Model 741 is designed for a capacity of up to 33 curies of cobalt-60 as Tech/Ops Source Assembly A424-18 and up to 240 curies of iridium-192 as Tech/Ops Soure Assembly A424-9. The assemblies are in special form as prescribed in 10CFR71 and IAEA Safety Series No. 6, 1973.

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1.3 APPENDIX

- USNRC Certificate of Compliance No. 9027, Rev. 1
- Descriptive Assembly Drawing, Model 741

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Form NRC-618
(12-73)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION
CERTIFICATE OF COMPLIANCE
For Radioactive Materials Packages

1.(a) Certificate Number	1.(b) Revision No.	1.(c) Package Identification No.	1.(d) Pages No.	1.(e) Total No. Pages
9027	1	USA/9027/B( )	1	3

2. PREAMBLE

2.(a) This certifica... is issued to satisfy Sections 173.393a, 173.394, 173.395, and 173.396 of the Department of Transportation Hazardous Materials Regulations (49 CFR 170-189 and 14 CFR 103) and Sections 146-19-10a and 146-19-100 of the Department of Transportation Dangerous Cargoes Regulations (46 CFR 146-149), as amended.

2.(b) The packaging and contents described in item 5 below, meets the safety standards set forth in Subpart C of Title 10, Code of Federal Regulations, Part 71, "Packaging of Radioactive Materials for Transport and Transportation of Radioactive Material Under Certain Conditions."

2.(c) This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. This certificate is issued on the basis of a safety analysis report of the package design or application—

3.(a) Prepared by (*Name and address*):

Technical Operations, Inc.
Northwest Industrial Park
Burlington, Massachusetts 01803

3.(b) Title and identification of report or application:

Technical Operations, Inc. application dated August 15, 1974, as supplemented.

3.(c) Docket No. 71-9027

4. CONDITIONS
This certificate is conditional upon the fulfilling of the requirements of Subpart D of 10 CFR 71, as applicable, and the conditions specified in item 5 below.

5. Description of Packaging and Authorized Contents, Model Number, Fissile Class, Other Conditions, and References:

(a) Packaging

(1) Models Nos.: 741 and 741E

(2) Description

A steel encased, uranium shielded Gamma Ray Projector. Primary components consist of an outer steel shell, internal bracing, polyurethane potting material, depleted uranium shield, and a zircalloy "S" tube. The contents are securely positioned in the zircalloy "S" tube by a source cable locking device and shipping plug. Tamper-proof seals are provided on the packaging and a 1/4 inch thick steel shipping plate is bolted over the source locking mechanism for additional protection during transport. The total weight of the package is approximately 300 pounds.

POOR ORIGINAL

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5. (a) Packaging (continued)

(3) Drawings

The packaging is constructed in accordance with the following Technical Operations, Inc. Drawings Nos.:

B66001-1 thru 8	D74101, Rev. A
B66001-12, 20	B74101-1, 4
A66001-4, 5, 6, 11	A74101-2, 5, 7, 8
B65502	C74102
B65502-1	D74102-1
B65503	A74102-2, 3
B655E01	C74103
B65501-6	D74103-1
65502 Bill of Mat'ls.	C74103-2
CSK 1923	74104

(b) Contents

(1) Type and form of material

Cobalt-60 or Iridium-192 as sealed sources which meet the requirements of special form ad defined in §71.4(o) of 10 CFR Part 71.

(2) Maximum quantity of material per package

33 Curies of Cobalt-60; or
240 Curies of Iridium-192

6. The source assemblies authorized for use in this packaging are limited to Models Nos. A424-9 or A424-18 as shown in Technical Operations, Inc. Drawing No. C42400, Rev. F, Sheet 2 of 3.

7. The name plates shall be fabricated of materials capable of resisting the fire test of 10 CFR Part 71 and maintaining their legibility.

8. The package authorized by this certificate is hereby approved for use under the general license provisions os Paragraph 71.12(b) of 10 CFR Part 71.

9. Expiration date: January 31, 1980.

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REFERENCES

Technical Operations, Inc. application dated August 15, 1974.

Supplement dated: December 19, 1974.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Charles E. MacDonald

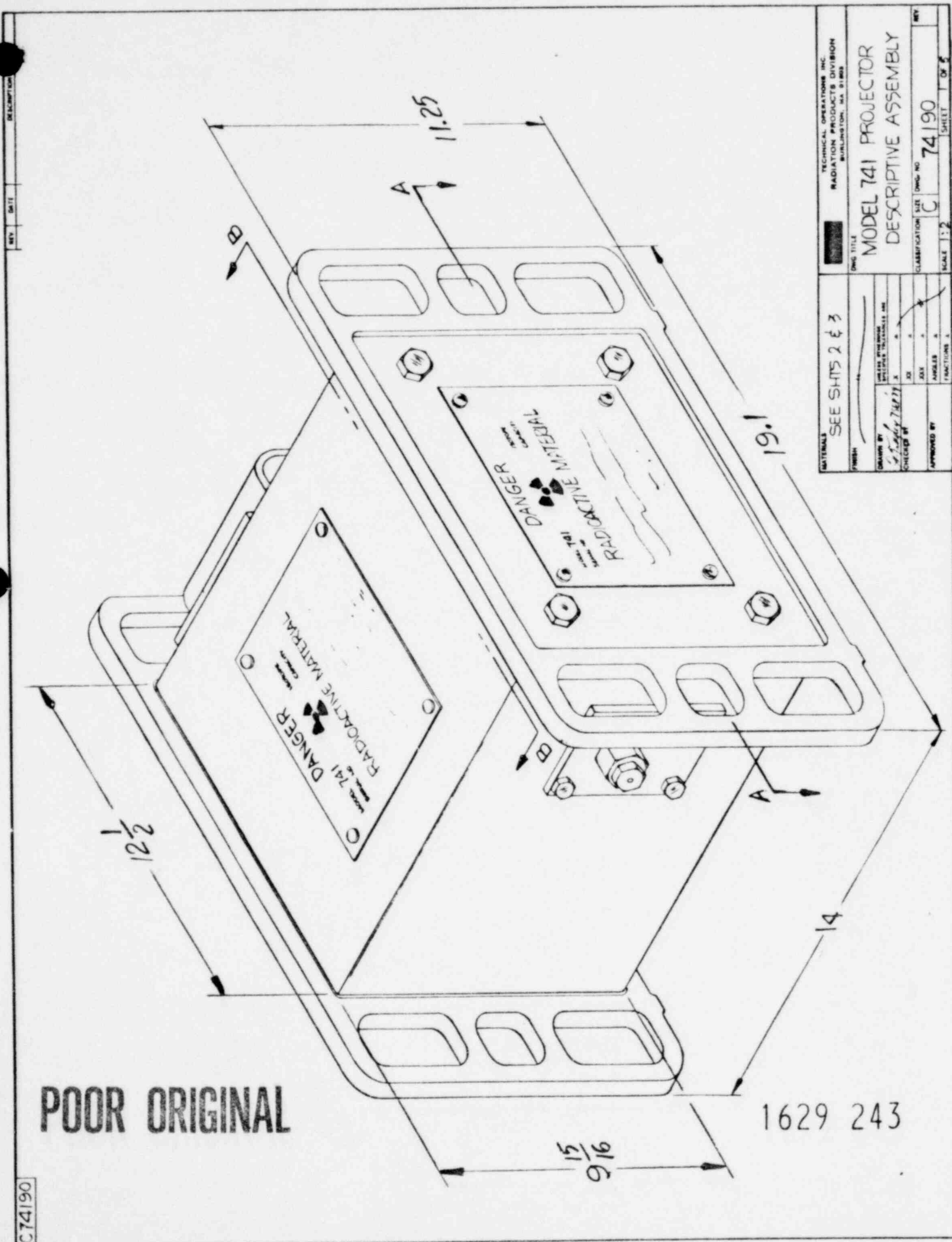
Charles E. MacDonald, Chief
Transportation Branch
Division of Fuel Cycle and
Material Safety

Date: MAR 29 1977

1629 242

REVISION 0

NOV. 29 1979



POOR ORIGINAL

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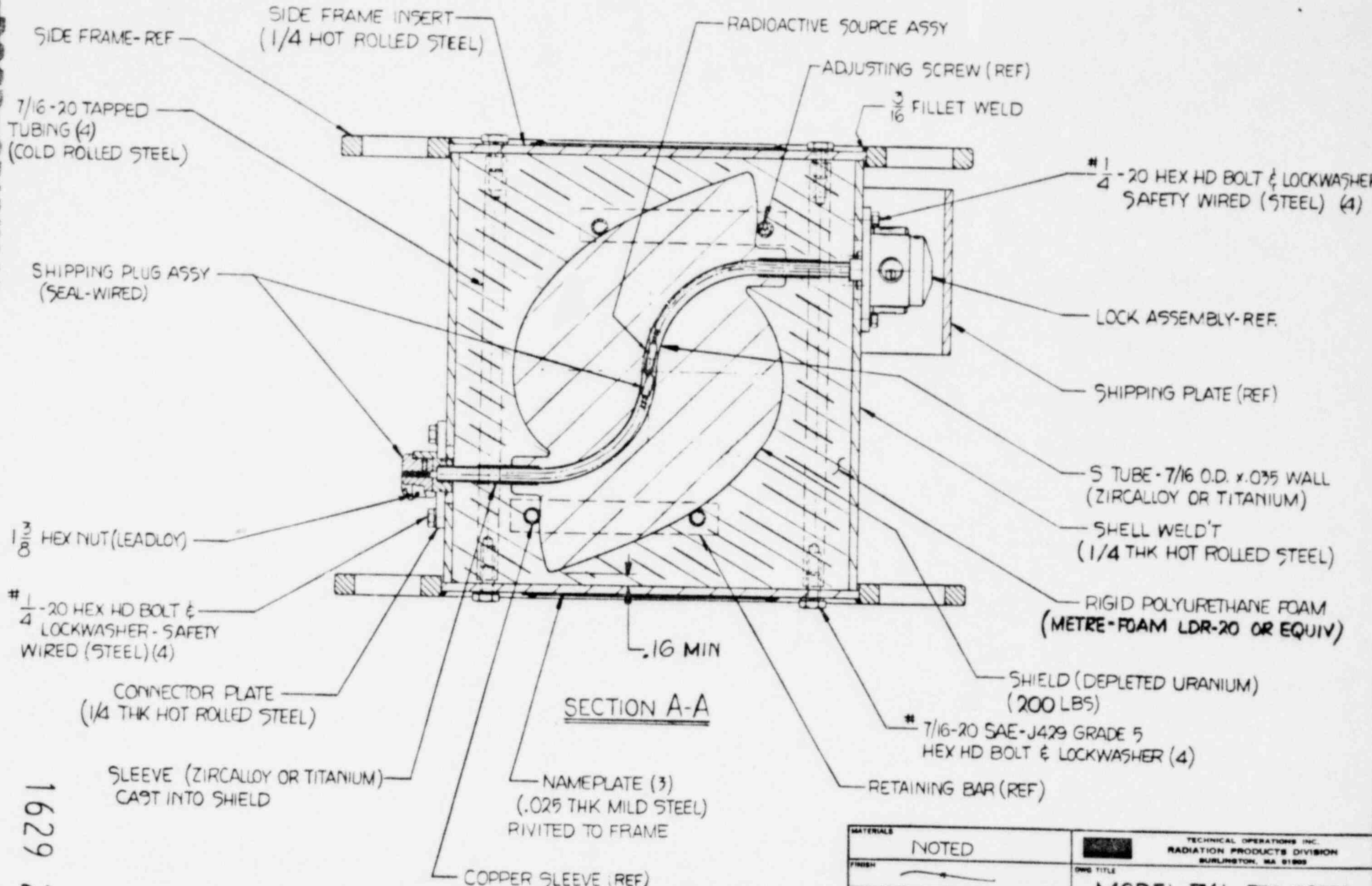
REV	DATE	DESCRIPTION



POOR ORIGINAL

PAGE 1-8

1629 244

MATERIALS: NOTED	TECHNICAL OPERATIONS INC. RADIATION PRODUCTS DIVISION BURLINGTON, MA 01803
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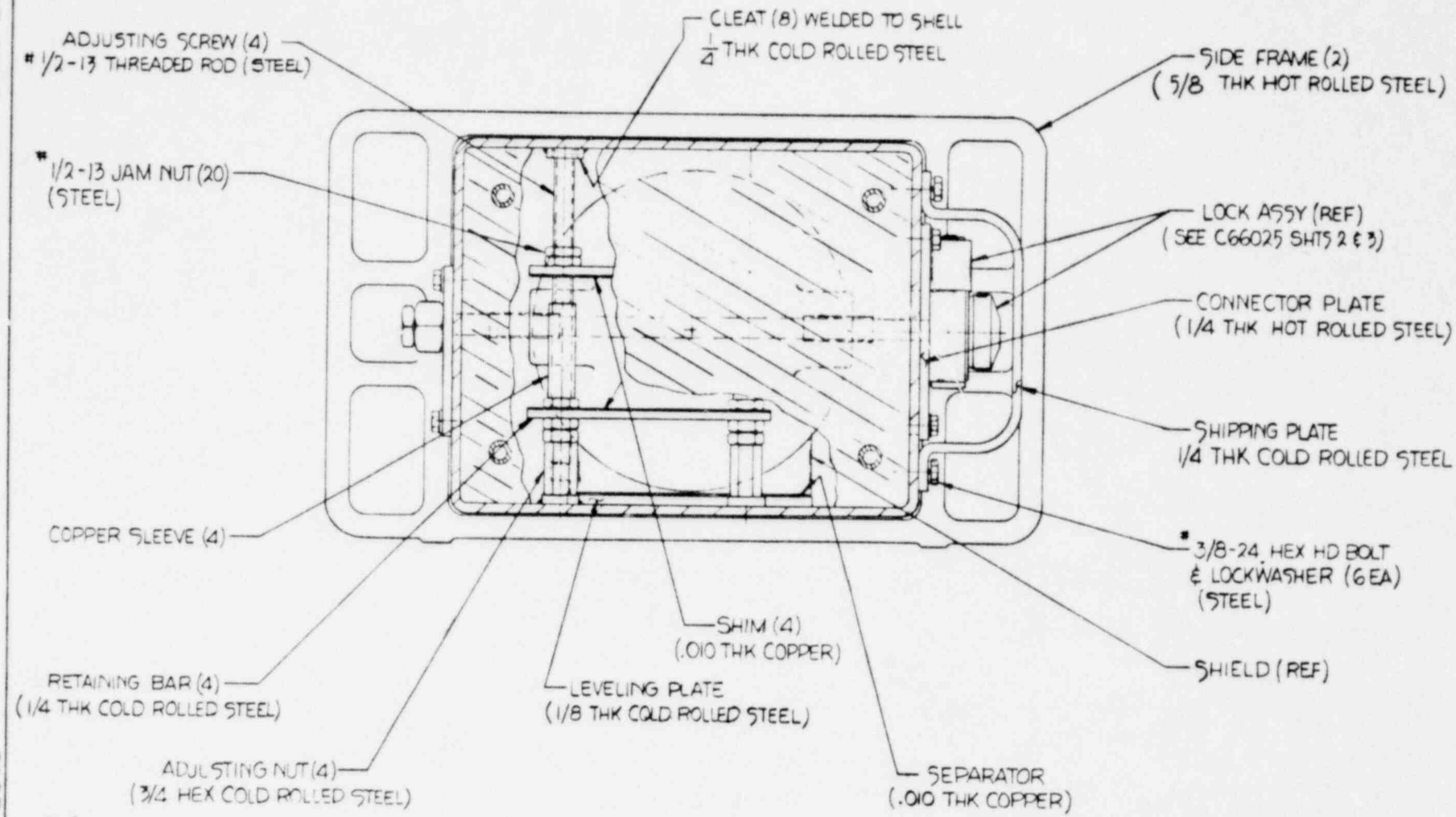
DRAWN BY: 7 Sep 79 / UNLESS OTHERWISE SPECIFIED TOLERANCES ARE: X ±, XX ±, XXX ±, ANGLES ±, FRACTIONS ±	CLASSIFICATION / SIZE: C / DWG. NO: 74190 / REV
CHECKED BY / APPROVED BY	SCALE: 1:2 / SHEET 2 OF 5

C74190

POOR ORIGINAL

REV.	DATE	DESCRIPTION



SECTION B-B

PAGE 1-9

1629 245

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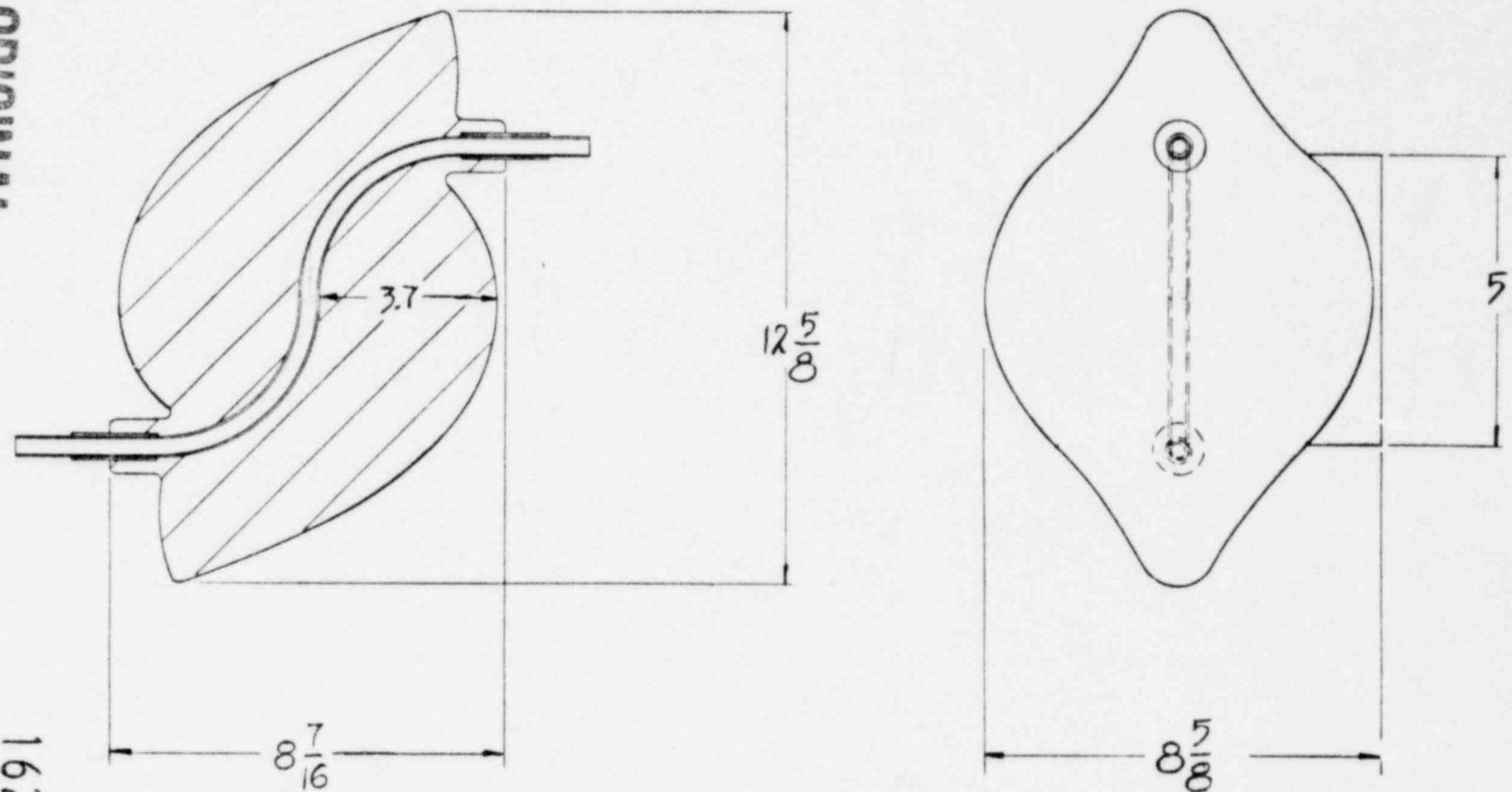
DRAWN BY: 7 SEP 77	UNLESS OTHERWISE SPECIFIED TOLERANCES ARE: .X ± / .XX ± / .XXX ± / ANGLES ± / FRACTIONS ±
CHECKED BY	CLASSIFICATION / SIZE: C / DWG. NO: 74190 / REV
APPROVED BY	SCALE 1:2 / SHEET 3 OF 5

C74190

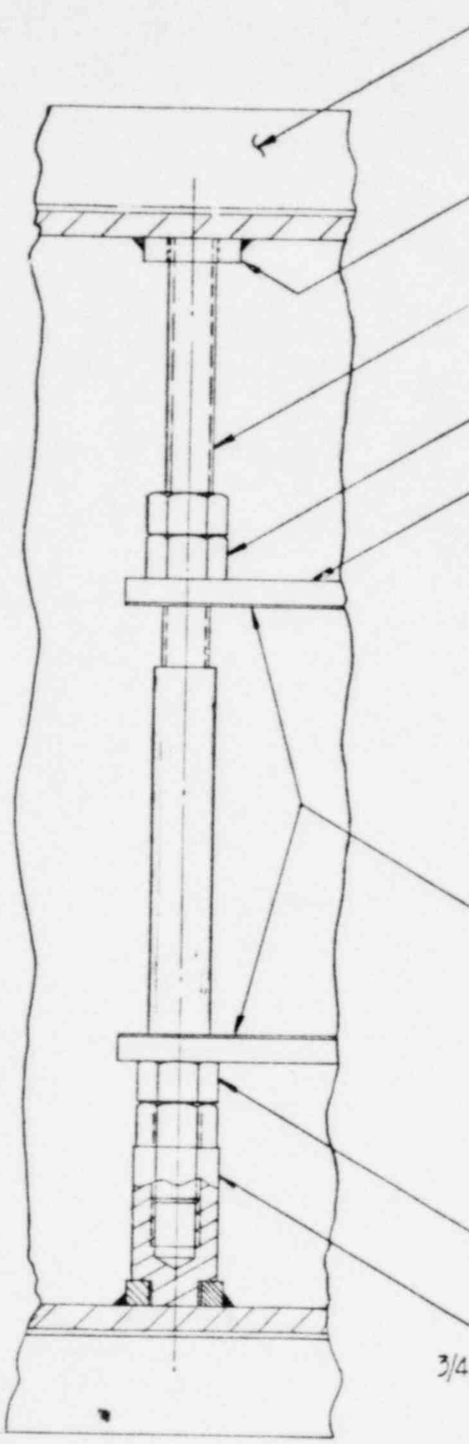


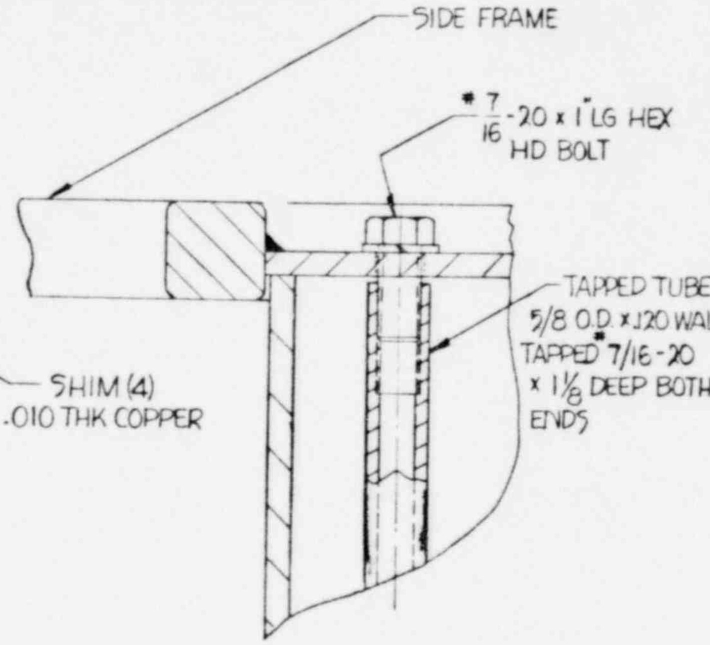
SHIELD DATA
200 LBS

MATERIALS: NOTED		TECHNICAL OPERATIONS INC. RADIATION PRODUCTS DIVISION BURLINGTON, MA 01803
FINISH		DWG TITLE: MODEL 741 PROJECTOR DESCRIPTIVE ASSEMBLY
DRAWN BY	UNLESS OTHERWISE SPECIFIED TOLERANCES ARE: .X ± .XX ± .XXX ± ANGLES ± FRACTIONS ±	CLASSIFICATION / SIZE C / DWG NO 74190
CHECKED BY		SCALE NONE / SHEET 4 OF 5
APPROVED BY		

POOR ORIGINAL

1629 246





VIEW C
SCALE 1:1

VIEW D
SCALE 1:1

1629 247

POOR ORIGINAL

TECHNICAL OPERATIONS INC.
RADIATION PRODUCTS DIVISION
BURLINGTON, MA 01803

MATERIALS: NOTED

DWG TITLE: MODEL 741 PROJECTOR
DESCRIPTIVE ASSEMBLY

SIZE: C
DWG NO: 74190
SHEET 5 OF 5

C74190

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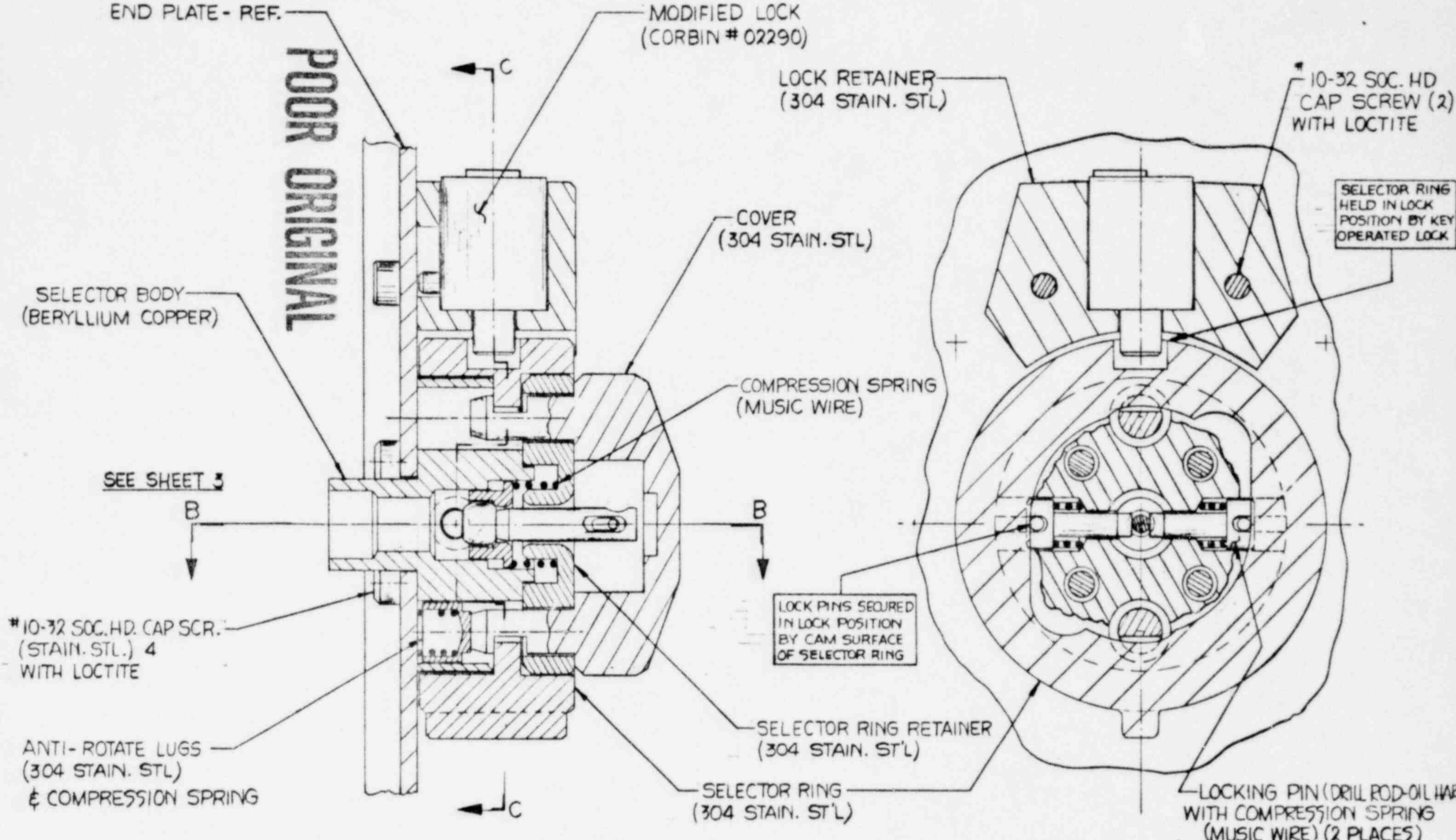
POOR ORIGINAL



LOCK ASSEMBLY

SECTION C-C

PAGE 1-12

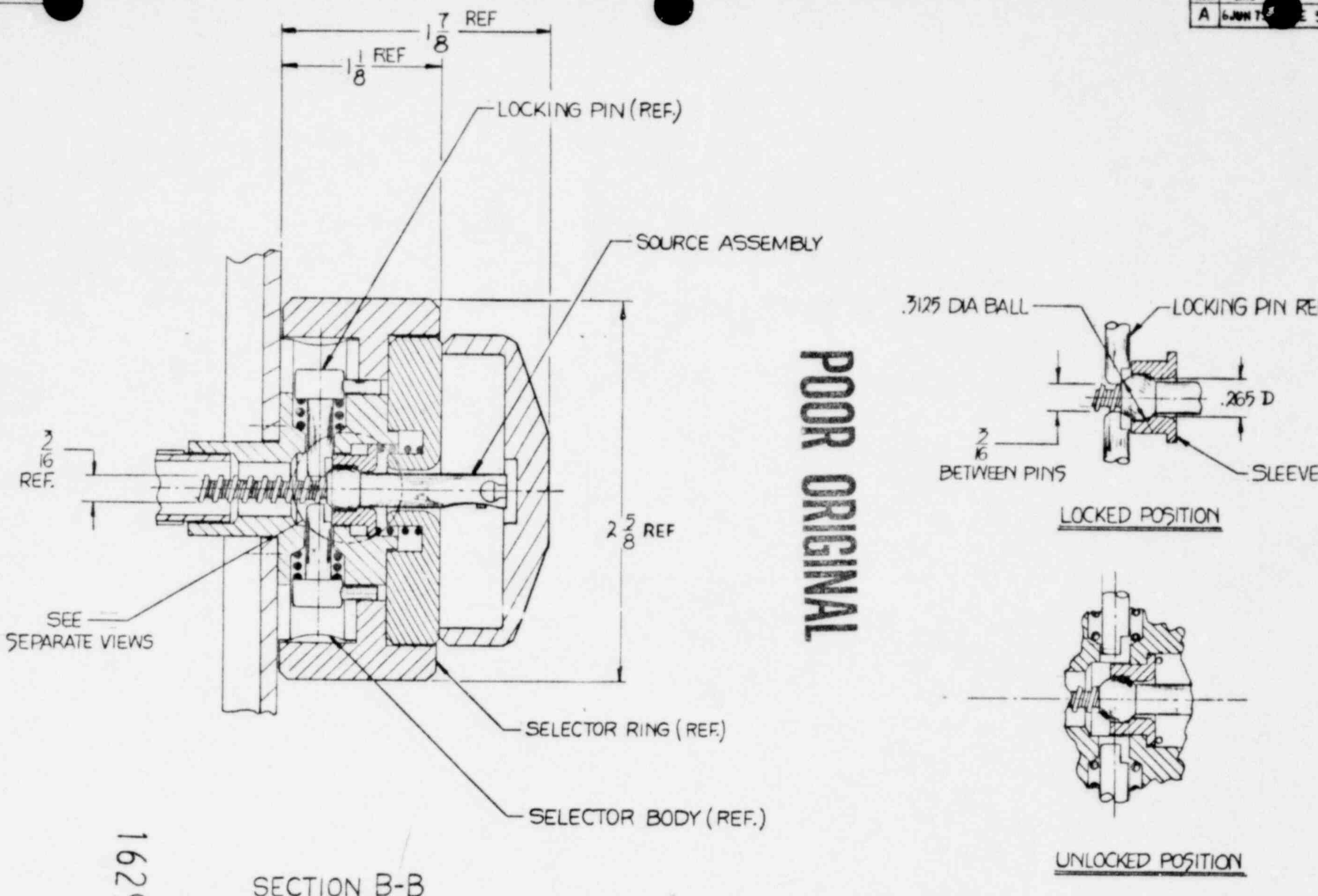
1629 248

MATERIALS: NOTED	TECHNICAL OPERATIONS INC. RADIATION PRODUCTS DIVISION BURLINGTON, MA 01803
FINISH: —	DWG TITLE: MODEL 660 GAMMA RAY PROJECTOR SHIPPING CONTAINER DESCRIPTIVE ASSEMBLY
DRAWN BY: [illegible] 6-27-78	UNLESS OTHERWISE SPECIFIED TOLERANCES ARE: .X ± / .XX ± / .XXX ± / ANGLES ± / FRACTIONS ±
CHECKED BY:	CLASSIFICATION: — SIZE: C DWG. NO: 66025
APPROVED BY:	SCALE 2:1 SHEET 2 OF 3

SPAULDING CO., INC. 21-60133

C66025

REV.	DATE	DESCRIPTION
A	6 JUN 7[illegible]	[illegible]E SHT 1



POOR ORIGINAL

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PAGE 1-13

MATERIALS: NOTED	TECHNICAL OPERATIONS INC. RADIATION PRODUCTS DIVISION BURLINGTON, MA 01803
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DRAWN BY: [signature]	UNLESS OTHERWISE SPECIFIED TOLERANCES ARE: X ±, XX ±, XXX ±, ANGLES ±, FRACTIONS ±
CHECKED BY	CLASSIFICATION; SIZE C; DWG NO 66025
APPROVED BY	SCALE 2:1; SHEET 3 OF 3

2. Structural Evaluation

2.1 Structural Design

2.1.1 Discussion

Structurally the Model 741 consists of five components: a source capsule, shield assembly, outer shell, side frames and lock assembly. The source capsule is the primary containment vessel. It meets the requirements for special form radioactive material as outlined in 10CFR71 (See Section 2.8). The shield is 200 pounds (91kg) of depleted uranium. The shield assembly fulfills two functions: It provides shielding for the radioactive material and, together with the positioning mechanisms, insures proper positioning of the source. The shield assembly is supported with retaining bars which are forced together by means of hex nuts threaded on adjusting screws. The adjusting screws and retaining bars are secured with jam nuts. The entire shield assembly is potted in a castable rigid polyurethane foam and encased in a $\frac{1}{4}$ inch (6.35mm) thick hot rolled steel shell. Steel-uranium interfaces are separated with copper. Attached to a the shell are side frames made of 0.625 inch (15.9mm) thick steel which are bolted together with 7/16-20 UNF hex head bolts. These are designed as lifting devices and impact limiters. The key operated lock assembly and control cable connector secure the source in the shielded position. A $\frac{1}{4}$ inch (6.35mm) thick steel shipping plate is installed to protect the lock from damage. Positive proof of source position is evidenced by use of a seal wired shipping plug.

2.1.2 Design Criteria

The Model 741 is designed to comply with the requirements of 10CFR71 and IAEA Safety Series No. 6, 1973. The device is simple in design, such that there are no design criteria which cannot be evaluated by straight-forward application of the appropriate section of 10CFR71 or IAEA Safety Series No. 6, 1o73.

2.2 Weights and Centers of Gravity

The Model 741 projector weighs 300 pounds (136kg). The shield assembly contains 200 pounds (91kg) of depleted uranium. The center of gravity is located approximately at the geometric center of the package.

2.3 Mechanical Properties of Materials

The Model 741 Gamma Ray Projector shell is made of hot rolled steel. This material has a yield strength of 40,000 pounds per square inch ($276 MN/m^2$). (Reference: Machinery's Handbook, 20th Edition, 1976, p. 452)

2.4 General Standards for All Packages

2.4.1 Chemical and Galvanic Reactions

The materials used in the construction of the Model 741 Gamma Ray Projector

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are uranium metal, steel, beryllium copper, bronze, copper and zircalloy or titanium. There will be no significant chemical or galvanic action between any of these components.

The possibility of the formation of the eutectic alloy of iron uranium at temperatures below the melting temperatures of the individual metals was considered. The iron uranium eutectic alloy temperature is approximately 1337°F (725°C). However, vacuum conditions and extreme cleanliness of the surfaces are necessary to produce the alloy at this low temperature. Due to the conditions under which the shields are mounted, sufficient contact for this effect does not exist.

In support of this conclusion, the following test results are presented. A thermal test of a sample of bare depleted uranium metal was performed by Nuclear Metals, Inc. The test indicated that the uranium sample oxidized such that the radial dimension was reduced by 1/32 inch. A subsequent test was performed in which a sample of bare, depleted uranium metal was placed on a steel plate and subjected to the thermal test conditions. The test showed no alloying or melting characteristics in the sample, and the degree of oxidation was the same as evidenced in the first test. A copy of the test report appears in Section 2.10.

Although the likelihood of the formation of an iron-uranium eutectic alloy is remote, copper separators are used at steel-uranium interfaces.

2.4.2 Positive Closure

The Model 741 source cannot be exposed without opening a key-operated lock. Access to the lock requires the removal of the shipping plate. Additionally, there is a shipping plug which is seal wired and provided with a tamperproof seal.

2.4.3 Lifting Devices

The Model 741 is designed to be lifted by the side frames. Each is secured by four 7/16-20 UNF SAE-J429 Grade 5 hex head bolts. These bolts are installed with 7/16 lock washers. The yield strength of these 7/16-20 UNF bolts is 10,900 pounds (48.6kN). As there is a thread engagement of 3/4 inch between the bolt and the tapped rod, the yield strength is less than the stripping strength (approximately 21,000 pounds) and thus is the limiting factor. A torque of 30 foot-pounds (41N-m) is applied to these bolts. This corresponds to a tension of approximately 4080 pounds (18.2kN). The total tensile loading on each bolt is 4310 pounds (19.2kN), due to the bolt torque and three times the weight of the package. The total torsional loading is 6100 pounds (27.1kN). Both loads are less than the yield strength of the bolts.

The weld joining the side frame to the side frame insert on the Model 741 is a 3/16 inch fillet weld. The American Welding Society "Code for Arc and Gas Welding in Building Construction" permits the stress on a fillet weld

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to be 13,600 psi ($89.6 MN/m^2$). As the shear stress on the throat of the fillet weld is the limiting factor, the allowable stress on a 3/16 inch fillet weld (throat dimension, 0.133 inch, 3.38mm) is calculated to be 1,800 pounds per linear inch (320N/mm) As the perimeter of the side frame insert is 46 inches (1.2m), the allowable load is 82,800 pounds (369kN). Hence, the allowable load on the side frame insert weld is greater than the yield strength of the bolt.

2.4.2 Tiedown Devices

The tiedown devices on the Model 741 are the side frames. As indicated in 2.4.3 above, these frames can safely support the package.

2.5 Standards for Type B and Large Quantity Packages

2.5.1 Load Resistance

Considering the package as a simple beam supported on both ends with a uniform load of 5 times the package weight evenly distributed along its length, the maximum stress can be computed from:

$$S = \frac{Fl}{8Z}$$

where:
- S: maximum stress
- F: total load (1500 pounds)
- l: Length of beam (19.1 inches)
- Z: section modulus of beam ($52.6\ in^3$)

(Reference: Machinery's Handbook, 20th ed., 1976, p. 442)

Thus, the maximum stress generated in the beam is 68 pounds per square inch ($470 kN/m^2$), which is far below the yield strength of the material, 40,000 psi ($276 MN/m^2$).

2.5.2 External Pressure

The Model 741 is open to the atmosphere; thus, there will be no differential pressure acting on it. The collapsing pressure of the source capsules can be found:

$$P = 86,670 \frac{t}{D} - 1386$$

where
- P: collapsing pressure in pounds per square inch
- t: wall thickness in inches (0.020 inch)
- D: outside diameter in inches (0.25 inch)

(Reference: Machinery's Handbook, 20th ed., 1976, p. 448)

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The collapsing pressure of the capsules is calculated to be 5550 pounds per square inch ($38.8 MN/m^2$). Therefore, the capsule can withstand an external pressure of 25psig.

2.6 Normal Conditions of Transport

2.6.1 Heat

The thermal evaluation is performed in Chapter 3 of this application. From this evaluation, it can be concluded that the Model 741 can withstand the normal heat transport conditions.

2.6.2 Cold

The metals used in the manufacture of the Model 741 can all withstand temperatures of -40°F (-40°C). The lower operating limit of the polyurethane foam is -100°F (-73°C). Thus, it is concluded that the Model 741 will withstand the normal transport cold conditions.

2.6.3 Pressure

The Model 741 is open to the atmosphere; thus, there will be no differential pressure acting on it. In Section 3.5.4, the source capsules are demonstrated to be able to withstand an external pressure reduction of 0.5 atmospheres ($50.7 kN/m^2$).

2.6.4 Vibration

The Model 741 has been in use five years. During that time there has never been a vibrational failure reported. Thus, we contend the Model 741 will not undergo a vibrational failure in transport.

2.6.5 Water Spray Test

The water spray test was not actually performed on the Model 741. We contend that the materials used in construction of the Model 741 are all highly water resistant and that exposure to water will not reduce the shielding or affect the structural integrity of the package.

2.6.6 Free Drop

The drop analysis performed in Hypothetical Accident conditions (See Section 2.7.1) is sufficient to satisfy the requirements outlined for the normal transport free drop condition in 10CFR71 and IAEA Safety Series No. 6, 1973. On this basis, we conclude that the Model 741 can withstand the free drop without impairment of the shielding or package integrity.

2.6.7 Corner Drop

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Not applicable

2.6.8 Penetration

A penetration test of the Model 741 was not actually performed. However, the similar Model 684 was subjected to the penetration test with no resultant loss of shielding or package integrity (a copy of the test report is enclosed in Section 2.10). The following analysis demonstrates that the maximum damage exhibited by the Model 741 due to the penetration test is less than that of the Model 684.

The maximum stress observed in a flat rectangular plate supported on all edges due to concentrated central loading is:

$$S = \frac{0.62F}{t^2}\left[\ln\left(\frac{L}{2r_o}\right) + 0.577\right]$$

where F : total load
t : thickness of plate (inches)
L : length of longest side (inches)
r_o: 0.325t (inches)

(Reference: Machinery's Handbook, 20th ed., 1976, p. 444)

The appropriate dimensions for the Model 741 and Model 684 are:

	Model 741	Model 684
t	0.25 inch (6.35mm)	0.1875 in. (4.76mm)
L	19.1 inch (485mm)	17 in. (432mm)
r_o	0.0812 in. (2.06mm)	0.0609 in. (1.55mm)

The calculated stress for the Model 741 is 53.0F; for the Model 684 it is 97.3F. In both cases the load F (40 inch drop of a 13 pound hemispherical billet) and the material of construction (hot rolled steel) are the same. The maximum stress, and thus the maximum damage, to the flat plate occurs in the Model 684. The shipping plate which protects the lock mechanism is the same in the two models. As the Model 684 successfully withstood the penetration condition, we conclude that the Model 741 can undergo the penetration test with no loss of structural integrity or shielding. (A copy of the test report for the Model 684 is enclosed in Section 2.10).

2.6.9 Compression

The gross weight of the Model 741 is 300 pounds (136kg). The maximum cross sectional area of the package is 268 square inches ($0.173m^2$). Thus, 2 pounds per square inch times the cross sectional area (600 pounds, 273kg) is less than five times the package weight (1500 pounds, 682kg). For this analysis, the load will be taken to be 1500 pounds.

The maximum stress generated in a flat rectangular steel plate with all edges fixed and a load distributed uniformly over the surface of the plate

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can be computed from:

$$S = \frac{0.5F}{t^2\left[\frac{1}{w} + 0.623\left(\frac{1}{w}\right)^5\right]}$$

where: S: maximum stress
F: total load (1500 pounds)
t: thickness of plate (0.25 inches)
w: width of plate (14 inches)
l: length of plate (19.1 inches)

(Reference: Machinery's Handbook, 20th ed., 1976, p. 444, Eq. 13)

From this relationship, the maximum stress generated in the plate is 2780 pounds per square inch ($19.2 MN/m^2$). This figure is greatly below the yield strength of the material, 40,000 pounds per square inch ($276 MN/m^2$). Thus, it can be concluded that compression will not adversely affect the package.

2.7 Hypothetical Accident Conditions

2.7.1 Free Drop

The Model 741 was not actually submitted to the 30 foot drop test. However, the Model 655 was submitted to the drop test (the test report appears in Section 2.10). The Model 741 has approximately the same weight and is constructed from the same materials as the Model 655:

	Model 655	Model 741
Length	19-3/4 inches (502mm)	19-1/8 inches (486mm)
Width	11 inches (279mm)	14 inches (356mm)
Height	10-1/4 inches (260mm)	11-1/4 inches (286mm)
Weight of Shield	280 lbs.	200 lbs.
Gross Weight of Container	385 lbs.	300 lbs.
Side Frame Material	3/4 inch thick (19.1mm) ductile iron	5/8 inch thick (15.9mm) hot rolled steel
Shell Material	$\frac{1}{4}$ inch thick (19.1mm) steel	$\frac{1}{4}$ inch thick (15.9mm) hot rolled steel

Based on the satisfactory performance of the Model 655, we conclude that the Model 741 will undergo no loss of shielding or structural integrity as a result of the 30 foot drop test.

2.7.2 Puncture

The Model 741 was not submitted to the puncture test of 10CFR71. However, the similar Model 655 (see Section 2.7.1) was submitted to the puncture test. There was no resultant damage to the container nor reduction in shielding. (A copy of the test report appears in Section 2.10). The

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shipping plate used in the Model 741 is the same as that used in the Model 676. The Model 676 puncture test report (included in Section 2.10) shows that the shipping plate withstood the puncture test. On this basis, we conclude that the Model 741 can successfully withstand the penetration condition of 10CFR71.

2.7.3 Thermal

The thermal analysis is presented in Section 3.5. There it is shown that the melting points of the materials, except the potting compound, used in the construction of Model 741 are all greater than 1475 F (800 C). Also indicated is the previous acceptability of this design (NRC Certificate of Compliance No. 9027, Rev. 1) using this evaluation.

Thus, it is concluded that the Model 741 satisfactorily meets the requirements for the hypothetical accident-thermal evaluation as set forth in 10CFR71.

2.7.4 Water Immersion

Not applicable

2.7.5 Summary of Damage

The tests designed to induce mechanical stress (drop, puncture) caused minor deformation, but no reduction in the safety features of the package. The thermal test resulted in no reduction of the safety of the package. It can be concluded that the hypothetical accident conditions have no adverse effect on the shielding effectiveness and structural integrity of the package.

2.8 Special Form

The Model 741 Gamma Ray Projector is designed for use with Tech/Ops Source Assemblies A424-9 and A424-18. These source assemblies have been certified as special form radioactive material (IAEA Certificate of Competent Authority Nos. USA/0065/S and USA/0154/S.) We contend that these certificates are sufficient evidence that the requirements for special form radioactive materials, as established in IAEA Safety Series No. 6, 1973, are satisfied.

2.9 Fuel Rods

Not applicable.

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2.10 APPENDIX

- Nuclear Metals, Inc., Test Report: Iron Uranium Alloying
- Test Report: Penetration Test, Model 684
- Test Report: Drop and Puncture Tests, Model 655
- Test Report: Puncture Test, Model 676
- Descriptive Assembly Drawings, Source Assemblies
- IAEA Certificates of Competent Authority Nos. USA/0165/S, USA/0154/S

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NUCLEAR METALS, INC.

2229 MAIN STREET
CONCORD, MASSACHUSETTS 01742
TELEPHONE 617 369-5410

28 January 1974

Technical Operations, Inc.
Radiation Products Division
South Avenue
Burlington, Massachusetts 01803

Attention: Mr. J. Lima

Gentlemen:

In response to a request by Joe Lima of Tech Ops, a simulated fire test was performed on samples of bare depleted uranium in contact with mild steel, the object being to determine what, if any, alloying or melting would occur under these conditions.

TEST DATA:

A 3/4-inch diameter x 5/8-inch long bare depleted uranium specimen was set on a 1-inch diameter x 1/8-inch thick mild steel plate, placed in a thin wall ceramic crucible. A mild steel cover plate was used on top of the crucible to act as a partial air seal. The crucible was loaded in a preheated 1450°F resistance heated furnace, held for 35 minutes, then removed and allowed to air cool under a ventilated hood.

RESULTS:

No reaction was evidenced between the two metals. Both separated readily and showed no alloying or melting characteristics.

Oxidation of the uranium was about the same degree as that reported to Joe Lima on an earlier experiment.

The test was performed by NMI on 25 January 1974.

1629 258

Very truly yours,

John G. Powers

John G. Powers
Project Engineer

REVISION 0

NOV 2 9 1979

TEST REPORT

RADIATION PRODUCTS DIVISION

BY: John J. Munro III

DATE: 5 September 1979

SUBJECT: Model 684 Penetration Test

On 5 September 1979, a penetration test was performed on a Technical Operations Model 684 Shipping Container in accordance with 10CFR71 Appendix A.8 and IAEA Safety Series No. 6, 1973, paragraphs 714a and 714b.

The hemispherical end of a vertical steel cylinder 1.25 inch in diameter weighing 14 pounds was dropped from the height of 40 inches onto the geometric center of the bottom surface of the Model 684. There was no deformation and no damage which would affect the shielding or structural integrity of the package.

A second test was conducted using the same cylinder. It was dropped from the height of 40 inches onto the shipping plate. There was no deformation and no damage which would affect the shielding or structural integrity of the package.

Documentry photographs are enclosed.

Performed by

John J. Munro III

Witnessed by

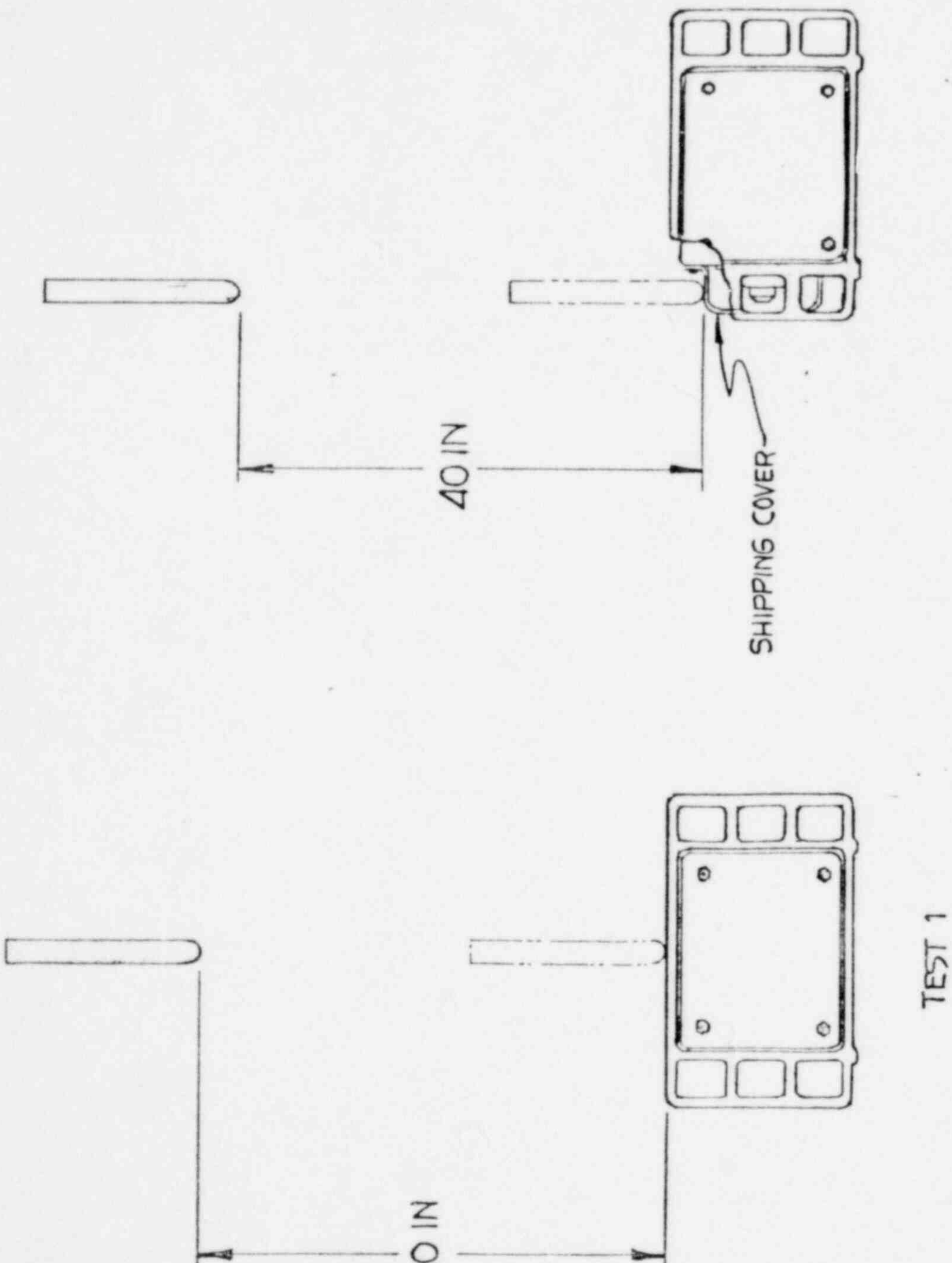
Angelo Kiklis

1629 259

REVISION O

NOV 2 9 1979

POOR ORIGINAL

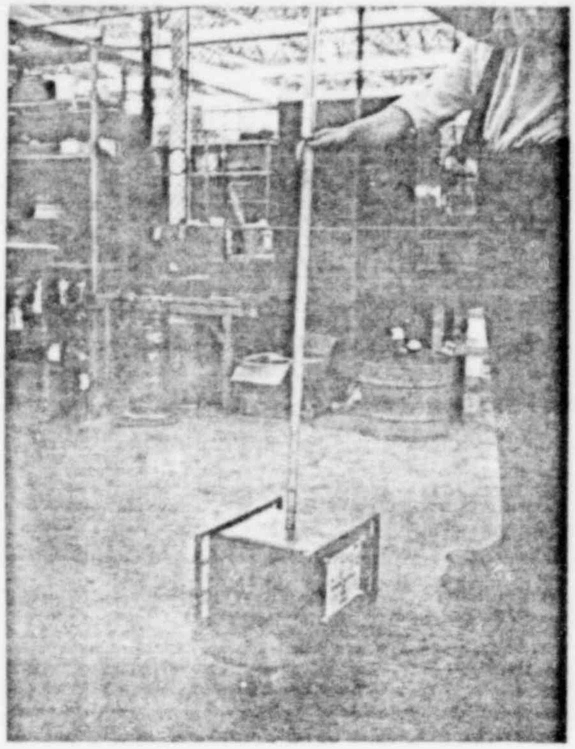


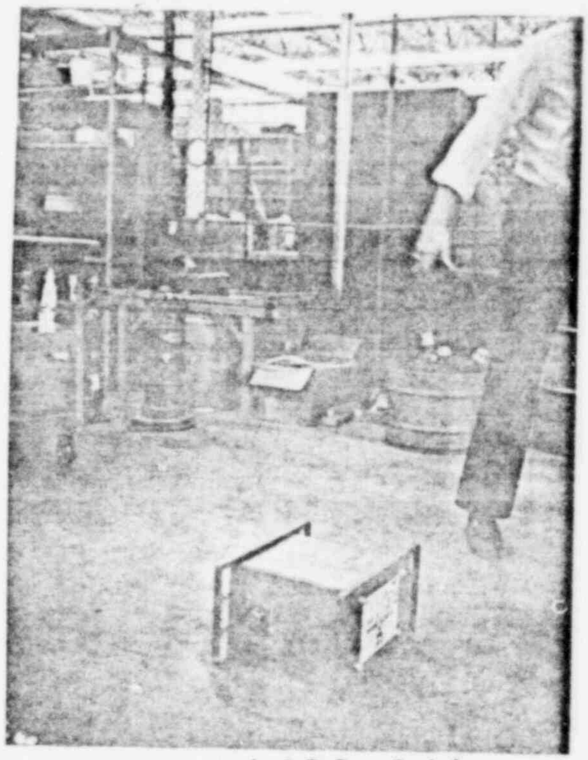
PENETRATION TEST
MODEL 684

1629 260

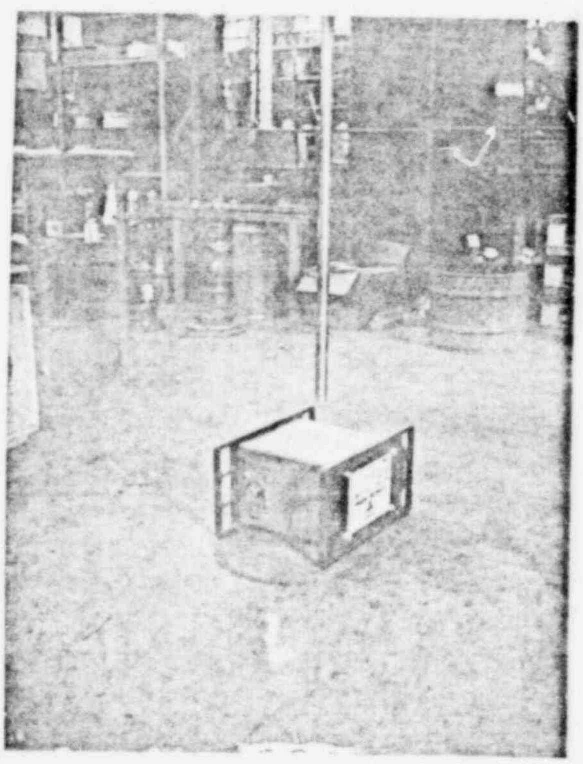
REVISION O
NOV. 2 9 1979

POOR ORIGINAL





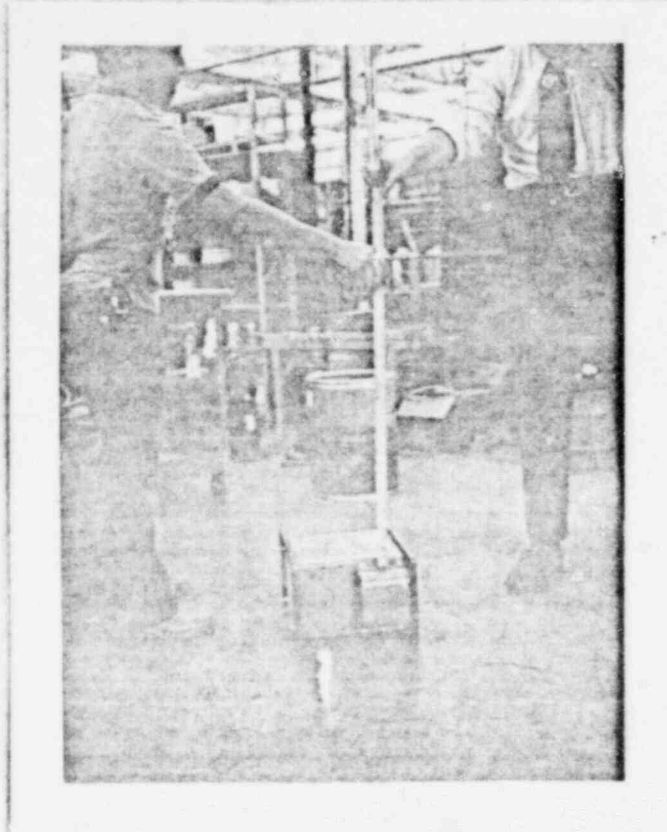
1629 261



REVISION O

NOV 2 9 1979

POOR ORIGINAL



1629 262

REVISION O

NOV. 2 9 1979

TEST REPORT

DESCRIPTION: Model 655 - 30' Drop DATE March 18, 1970

The first drop test produced no damage. The second drop test broke the corner of the side plate off. Two tie-rod bolts were sheared off one side and one sheared off on the other side. D. U. Shield remained in place. The source Tube was straight and the front nut turned freely.

The puncture test (41" drop on to a 6" dia. steel billet) left a 1/16" deep x 1/8" wide x 1" long gouge on the botton.

CONCLUSION:

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BY Richard Evans WITNESSED BY Fred Hauser

REVISION C

NOV 2 9 1979

FROM LEFT, MODELS 670, 655 AND 672 GAMMA RAY PROJECTORS
AFTER CONCLUSION OF 30 FOOT DROP TEST AND PUNCTURE TEST

POOR ORIGINAL

1629 264

REVISION O

NOV. 2 9 1979

TEST REPORT

DESCRIPTION: DATE 9 December 1974

Puncture Test of Model 676 Container Connector

A Model 676 Gamma Ray Projector with Shipping Plate installed was dropped from a height of 40 inches onto a six inch diameter, eight inch high steel Billet as shown in Figure 1 a. The Container impacted on the Shipping Plate as shown in Figure 1 b.

CONCLUSION:

No damage to the container, shipping plate or control cable connector resulted. There was no reduction of shielding effectiveness nor loss of Radioactive Material.

BY Don Brasseur WITNESSED BY John J. Munro 111

1629 265

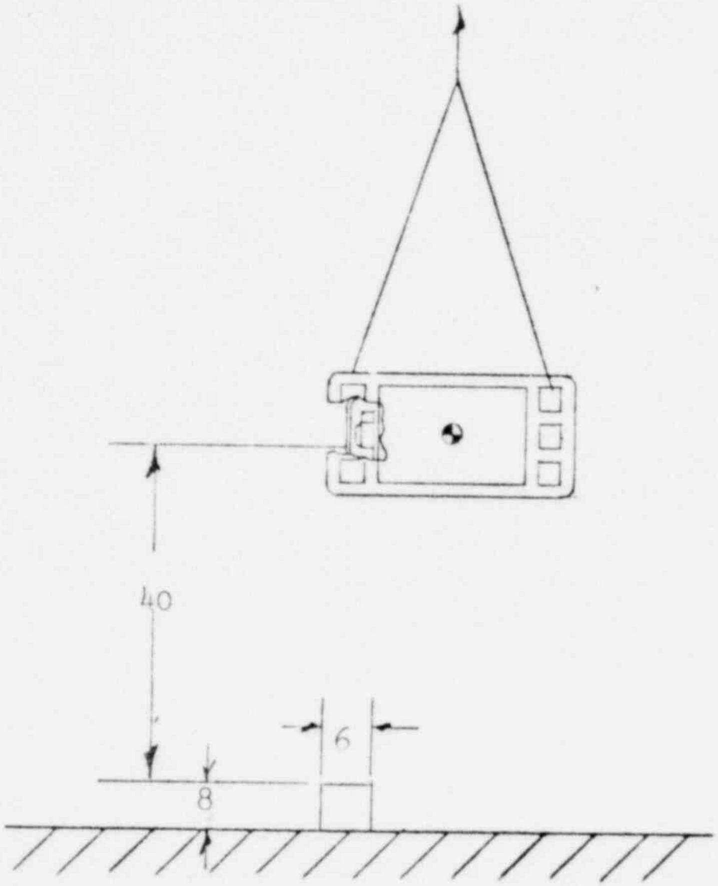
REVISION O

NOV. 2 9 1975

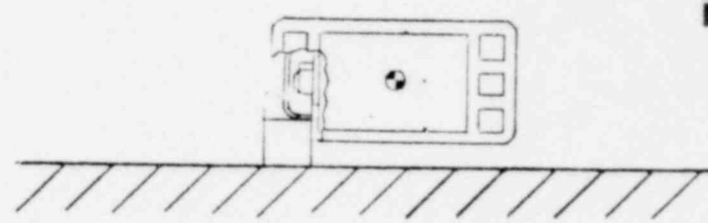
PUNCTURE TEST

CONTROL CABLE CONNECTOR ASSEMBLY

Model 676 Gamma Ray Projector
(weight 545 pounds) dropped from
a height of 40 inches onto a steel
cylinder (6in dia x 8 in high)



1a



1 b

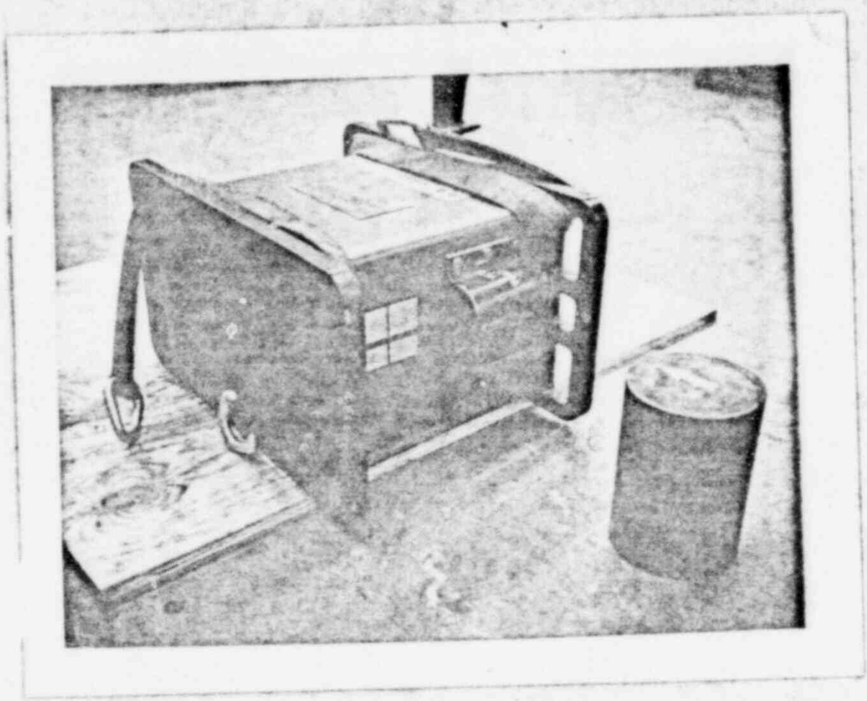
POOR ORIGINAL

1629 266

REVISION O

NOV 29 1972

POOR ORIGINAL



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Model 676 at Conclusion of Puncture Test

REVISION O

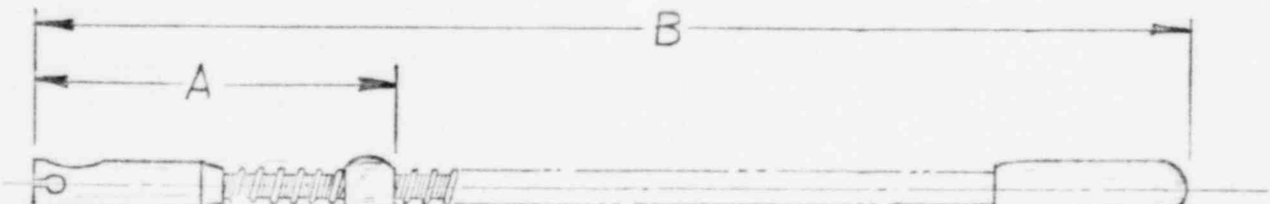
NOV. 2 9 1979

A60060

POOR ORIGINAL

REV.	DATE	DESCRIPTION

MODEL	CAPACITY (CURIES)	CAPSULE STYLE	DIM A (INCHES)	DIM B (INCHES)
A424-2	22	60011, 60001	N.A.	9 11/16
A424-3	22	60011, 60001	N.A.	11 13/16
A424-4	55	60011, 60000	N.A	20
A424-5	6	60011, 60001	N.A.	7 7/16
A424-7	165	60012, 60002	N.A.	17 1/4
A424-8	110	60011, 60000	N.A.	11 15/16
A424-10	6	60011, 60004	1.225	7 3/4
A424-11	55	60011, 60004	1.225	9 9/16
A424-12	110	60011, 60004	1.225	10 7/8
A424-13	330	60012, 60002	1.225	10 15/16
A424-14	110	60011, 60004	1.225	10 5/8
A424-15	11	60011, 60004	1.225	8 15/16
A424-16	55	60011, 60000	2.373	13 1/2
A424-17	55	60011, 60000	2.373	13 1/2
A424-18	33	60011, 60000	1.225	10 1/4
A424-19	0.11	60001, 60004	1.225	7 3/16
A453-1	110	60011, 60000	N.A.	11 15/16
A453-2	165	60012, 60002	N.A.	17 1/4
A453-5	550	60012, 60002	N.A.	17 1/4
A453-6	1100	60013, 60003	N.A.	17 1/4
A453-7	110	60011, 60000	N.A	11 15/16
A453-8	55	60011, 60000	N.A.	11 15/16
A453-9	55	60011, 60000	2.373	13 1/2
A453-10	55	60011, 60000	2.373	13 1/2



1629 268

MATERIALS

FINISH

DRAWN BY D. Tinsley 24 Jan '79

CHECKED BY

APPROVED BY

UNLESS OTHERWISE SPECIFIED TOLERANCES ARE

.X ±

.XX ±

.XXX ±

ANGLES ±

FRACTIONS ±

TECHNICAL OPERATIONS INC.
RADIATION PRODUCTS DIVISION
BURLINGTON, MA 01803

DWG TITLE

^{60}COBALT SOURCE REFERENCE

CLASSIFICATION | SIZE A | DWG. NO. 60060 | REV.

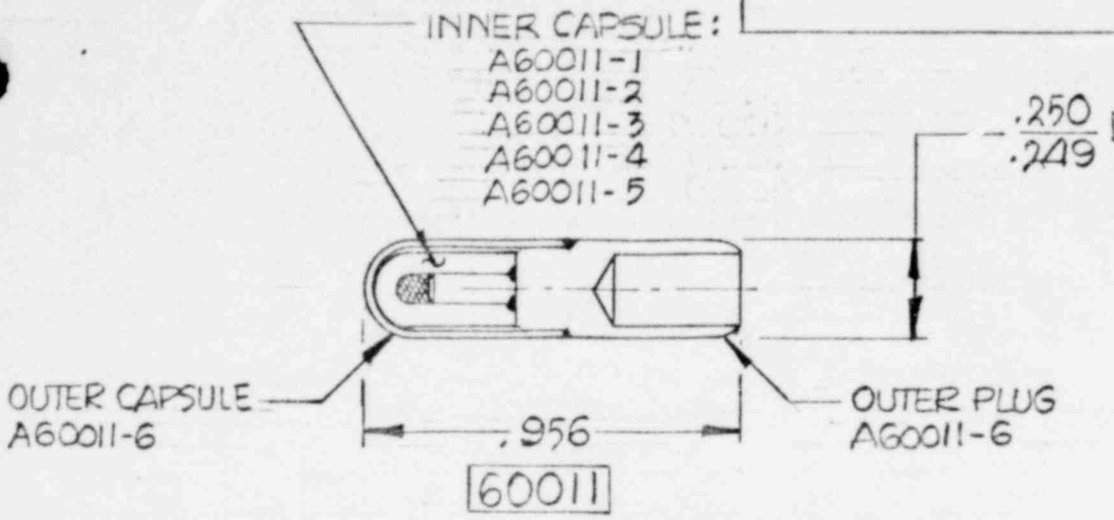
SCALE | SHEET 1 OF 3

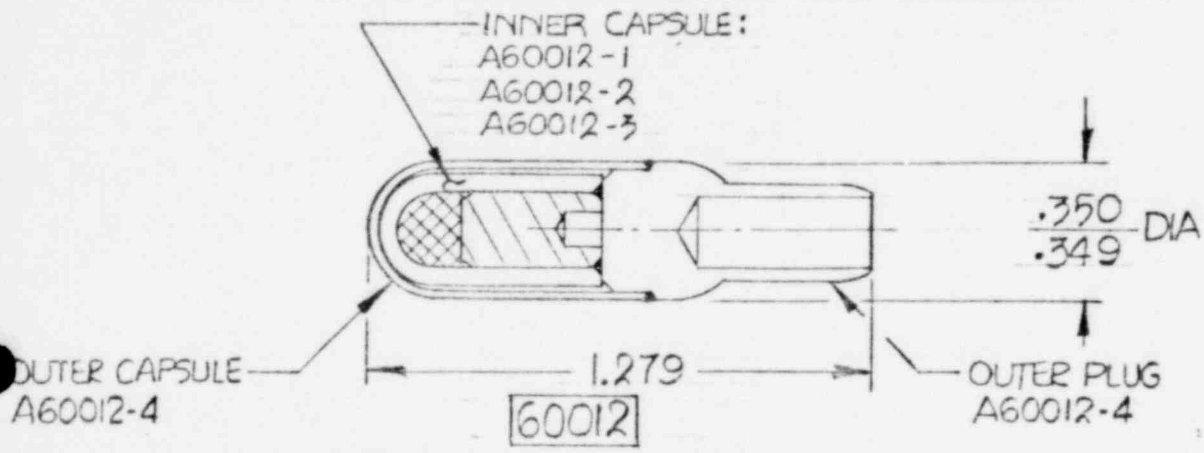
SPAULDING CO., INC. 21-60169

A60060

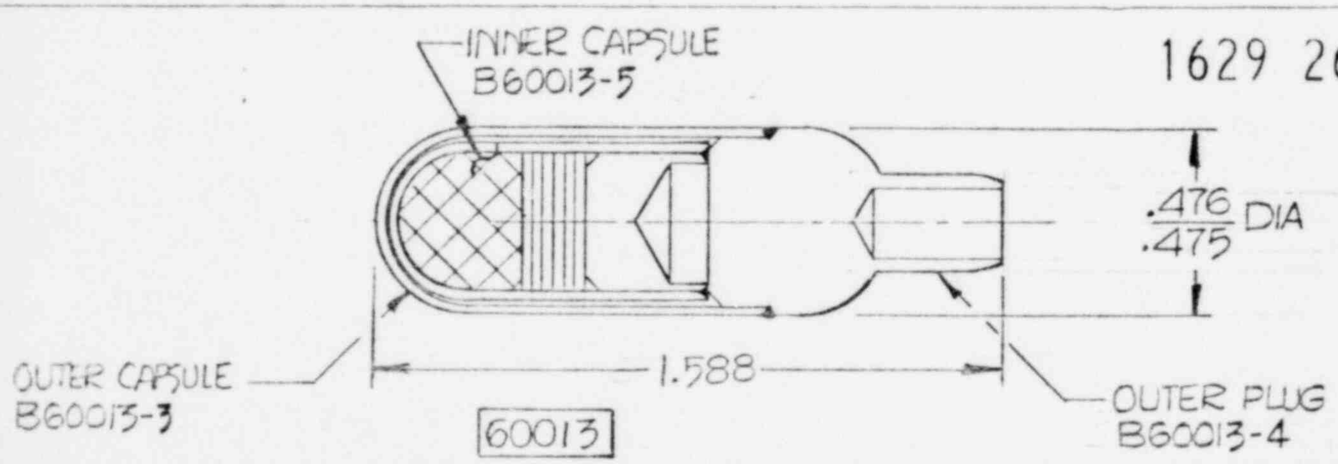
POOR ORIGINAL

REV.	DATE	DESCRIPTION





1629 269

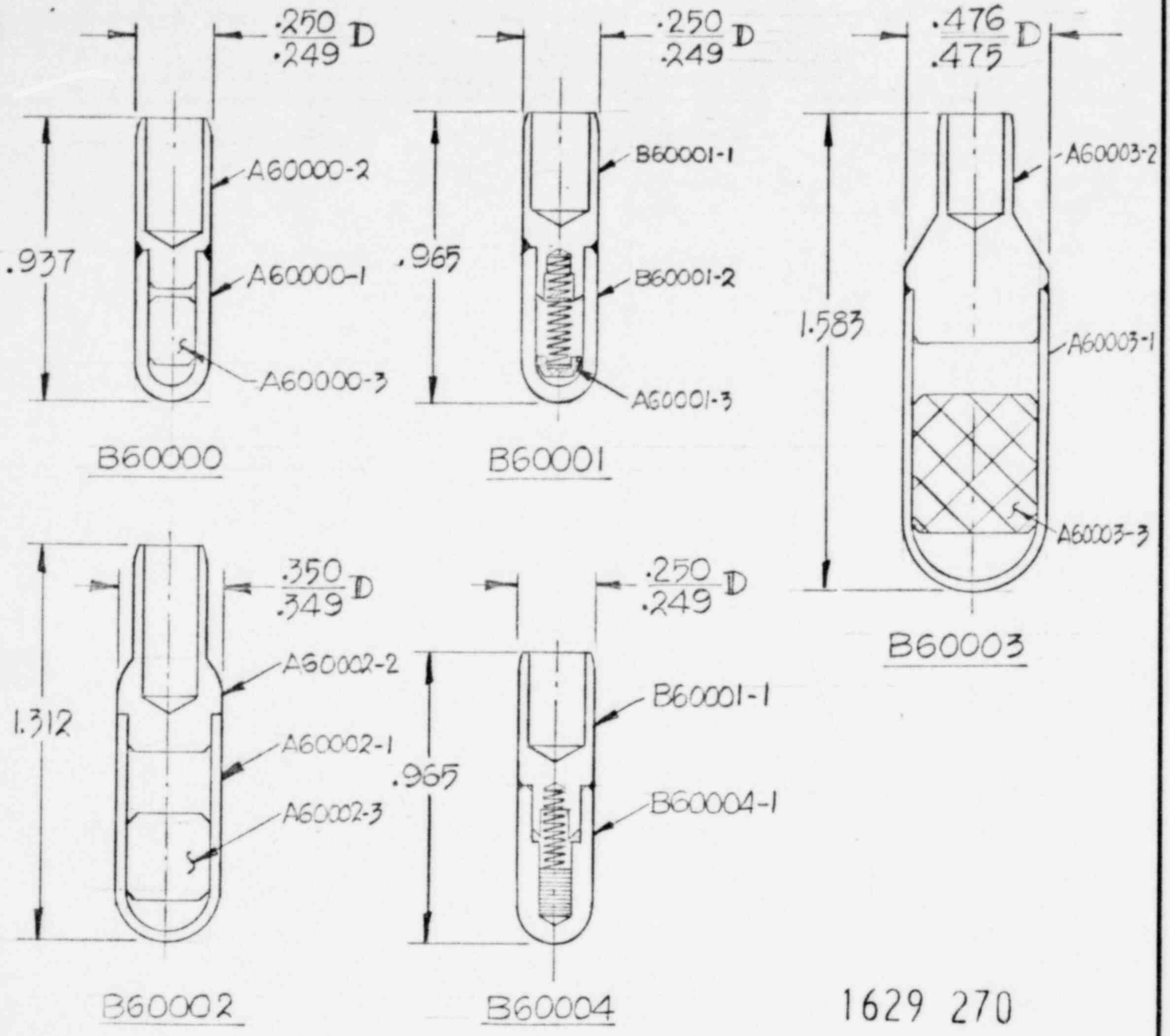


MATERIALS	TECHNICAL OPERATIONS INC. RADIATION PRODUCTS DIVISION BURLINGTON, MA 01803
FINISH	DWG TITLE COBALT ^{60} SOURCE REFERENCE (DOUBLE ENCAPSULATION)
DRAWN BY J. Kinsley 25 JUL 79	UNLESS OTHERWISE SPECIFIED TOLERANCES ARE: .X ± / .XX ± / .XXX ± / ANGLES ± / FRACTIONS ±
CHECKED BY	CLASSIFICATION / SIZE A / DWG. NO. 60060 / REV.
APPROVED BY	SCALE 2:1 / SHEET 2 OF 3

A60060

POOR ORIGINAL

REV.	DATE	DESCRIPTION



1629 270

MATERIALS		TECHNICAL OPERATIONS INC. RADIATION PRODUCTS DIVISION BURLINGTON, MA 01803
FINISH		DWG TITLE COBALT 60 SOURCE REFERENCE (SINGLE ENCAPSULATION)
DRAWN BY	UNLESS OTHERWISE SPECIFIED TOLERANCES ARE	
	.X ±	
CHECKED BY	.XX ±	
	.XXX ±	CLASSIFICATION · SIZE A · DWG. NO. 60060 · REV.
APPROVED BY	ANGLES ±	
	FRACTIONS ±	SCALE 2:1 · SHEET 3 OF 3

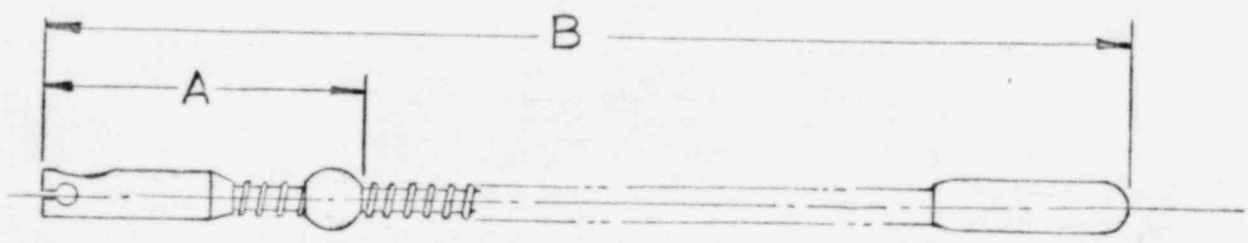
SPAULDING CO., INC. 21-60169

A60050

REV.	DATE	DESCRIPTION
A	12-18-78	SEE SHEET 2 FGT

POOR ORIGINAL

MODEL	CAPACITY (CURIES)	CAPSULE STYLE	DIM A	DIM B
A424-1	120	B60001 OR B60004	2.373	7 7/16
A424-9	120	B60001 OR B60004	1.225	7 3/16
A81401	120	B60001 OR B60004	1.875	7 3/16
A68309	120	C68310	NOT APPLICABLE; ATTACHED DIRECTLY TO DRIVE CABLE	
B69701	120	B60001 OR B60004	2.537	7 11/16
A424-20	240	B60001 OR B60004	1.225	8 15/16
A58101	240	B60006	NOT APPLICABLE NOT WIRE MOUNTED	
A424-6	120	B60001 OR B60004	N.A.	10 11/16



1629 271

MATERIALS

FINISH

DRAWN BY 6 OCT 78

CHECKED BY

APPROVED BY

UNLESS OTHERWISE SPECIFIED TOLERANCES ARE

.X	±
.XX	±
.XXX	±
ANGLES	±
FRACTIONS	±

tech/ops

TECHNICAL OPERATIONS INC.
RADIATION PRODUCTS DIVISION
BURLINGTON, MA 01803

DWG TITLE

192 IRIDIUM SOURCE REFERENCE

CLASSIFICATION | SIZE A | DWG. NO. 60050 | REV. A

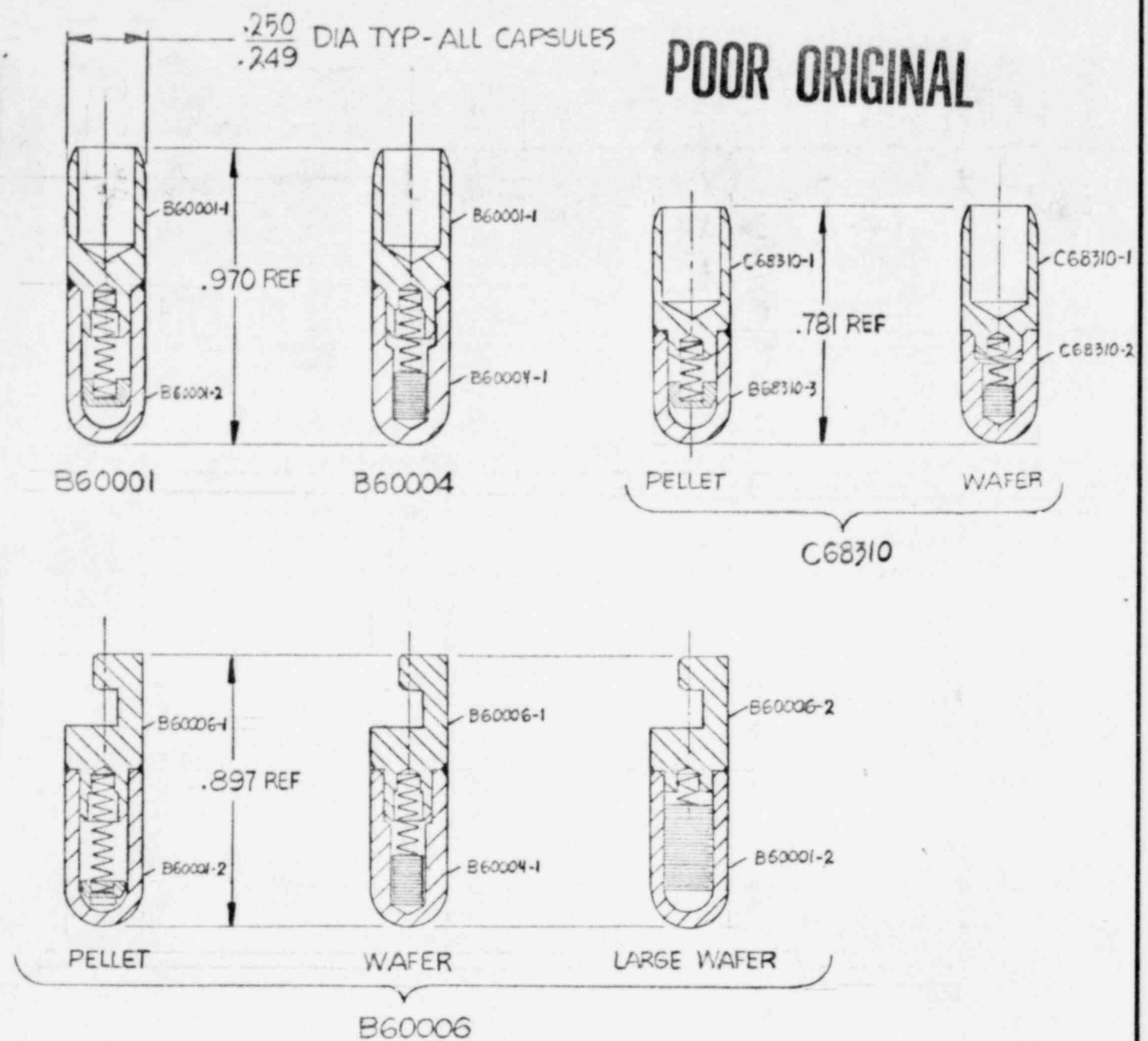
SCALE 1:1 | SHEET 1 OF 2

SPAULDING CO., INC. 21-60169

A60050

REV.	DATE	DESCRIPTION
A	12-18-78	CALLOUT B68310-3 WAS C68310-3 & B60001-2 WAS B66001-2 (ERRORS)

POOR ORIGINAL



1629 272

MATERIALS	TECHNICAL OPERATIONS INC. RADIATION PRODUCTS DIVISION BURLINGTON, MA 01803
FINISH	DWG TITLE: 192 IRIDIUM SOURCE REFERENCE
DRAWN BY [signature] 6 OCT 78	UNLESS OTHERWISE SPECIFIED TOLERANCES ARE: .X ± / .XX ± / .XXX ± / ANGLES ± / FRACTIONS ±
CHECKED BY	CLASSIFICATION · SIZE A · DWG. NO. 60050 · REV. A
APPROVED BY	SCALE 2:1 · SHEET 2 OF 2

SPAULDING CO., INC. 21-60169

REFER TO:

IAEA CERTIFICATE OF COMPETENT AUTHORITY

Special Form Radioactive Material Encapsulation

Certificate Number USA/0165/S
(Revision 0)

This certifies that the encapsulated sources, as described, when loaded with the authorized radioactive contents, have been demonstrated to meet the regulatory requirements for special form radioactive material as prescribed in IAEA[1] and USA[2] Regulations for the transport of radioactive materials.

I. Source Description and Radioactive Contents - The sources described by this certificate consist of the following Technical Operations, Inc., models which are welded capsules constructed of either 304 or 304L stainless steel to the listed capsule designs (see Appendix A) and which contain not more than the listed quantities of Cobalt-60 in metallic form:

Model	Capsule Style	Activity (Curies)
A424-2	60011, 60001	22
A424-3	60011, 60001	22
A424-4	60011, 60000	55
A424-5	60011, 60001	6
A424-7	60012, 60002	165
A424-8	60011, 60000	110
A424-10	60011, 60004	6
A424-11	60011, 60004	55
A424-12	60011, 60004	110
A424-13	60012, 60002	330
A424-14	60011, 60004	110
A424-15	60011, 60004	11
A424-16	60011, 60000	55
A424-17	60011, 60000	55
A424-18	60011, 60000	33
A424-19	60001, 60004	0.11
A453-1	60011, 60000	110
A453-2	60012, 60002	165
A453-5	60012, 60002	550
A453-6	60013, 60003	1100
A453-7	60011, 60000	110
A453-8	60011, 60000	55
A453-9	60011, 60000	55
A453-10	60011, 60000	55

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II. This certificate, unless renewed, expires on September 30, 1982.

This certificate is issued in accordance with paragraph 803 of the IAEA Regulations and in response to the July 26, 1979, petition by Technical Operations, Inc., Burlington, Massachusetts, and in consideration of the associated information therein.

Certified by:

R R Rawl

September 17, 1979

R. R. Rawl (Date)
Designated U.S. Competent Authority for the
International Transportation of Radioactive Materials
Office of Hazardous Materials Regulation
Materials Transportation Bureau
U.S. Department of Transportation

---

[1] "Safety Series No. 6, Regulations for the Safe Transport of Radioactive Materials, 1973 Revised Edition" published by the International Atomic Energy Agency (IAEA), Vienna, Austria.

[2] Title 49, Code of Federal Regulations, Part 170-178, USA.

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REVISION O

NOV. 2 S 1979

DEPARTMENT OF TRANSPORTATION
RESEARCH AND SPECIAL PROGRAMS ADMINISTRATION
WASHINGTON, D.C. 20590

REFER TO

IAEA CERTIFICATE OF COMPETENT AUTHORITY

Special Form Radioactive Material Encapsulation

Certificate Number USA/0154/S

This certifies that the encapsulated sources, as described, when loaded with the authorized radioactive contents, have been demonstrated to meet the regulatory requirements for special form radioactive material as prescribed in IAEA[1] and USA[2] regulations for the transport of radioactive materials.

I. Source Description - The sources described by this certificate are identified as the Technical Operations, Inc., Models which are described and constructed as follows:

Model No.	Capsule Style	Approximate Size (in inches, diameter x length)
A424-1	B60001 or B60004	.25 x .97
A424-6	B60001 or B60004	.25 x .97
A424-9	B60001 or B60004	.25 x .97
A424-20	B60001 or B60004	.25 x .97
A58101	B60006 Pellet, Wafer or Large Wafer	.25 x .90
A68309	C68310 Pellet or Wafer	.25 x .78
A81401	B60001 or B60004	.25 x .97
B69701	B60001 or B60004	.25 x .97

All capsules are constructed of either 304 or 304L stainless steel and conform with the following design drawings:

Capsule Style	Drawing Number
B60001	B60001 - 1 Rev. H and - 2 Rev. F
B60004	B60001 - 1 Rev. H and B60004 - 1 Rev. D
B60006 Pellet	B60006 - 1 Rev. H and B60001 - 2 Rev. F
B60006 Wafer	B60006 - 1 Rev. H and B60004 - 1 Rev. D
B60006 Large Wafer	B60006 - 2 and B60001 - 2 Rev. F
C68310 Pellet	C68310 Rev. B and B68310-3
C68310 Wafer	C68310 Rev. B

II. Radioactive Contents - The authorized radioactive contents of these sources consist of not more than the following amounts of Iridium-192 as solid metal:

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Model No.	Contents (Curies)
A424-1	120
A424-6	120
A424-9	120
A424-20	240
A58101	240
A68309	120
A81401	120
B69701	120

III. This certificate, unless renewed, expires December 31, 1981.

This certificate is issued in accordance with paragraph 803 of the IAEA Regulations[1], and in response to the November 3, 1978, petition by Technical Operations, Inc., Burlington, Massachusetts, and in consideration of the associated information therein.

Certified by:

R R Rawl December 15, 1978

R. R. Rawl, Health Physicist (Date)
U. S. Department of Transportation
Office of Hazardous Materials Regulation
Washington, D. C. 20590.

[1]"Safety Series No. 6, Regulations for the Safe Transport of Radioactive Materials, 1973 Revised Edition", published by the International Atomic Energy Agency (IAEA), Vienna, Austria.

[2]Title 49, Code of Federal Regulations, Part 170-178, USA.

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NOV. 29 1975

3. Thermal Evaluation

3.1 Discussion

The Model 741 Gamma Ray Projector is a completely passive thermal device and has no mechanical cooling systems or relief valves. All cooling of the package is through free convection and radiation. The maximum heat source is the 240Ci iridium-192 source. The corresponding decay heat is 2.5 watts (see Section 3.4.1).

3.2 Summary of Thermal Properties of Materials

The melting points of the metals used in the construction of the Model 741 are:

Depleted Uranium Metal	2070°F (1133°C)
Carbon Steel	2453°F (1345°C)
Copper	1940°F (1060°C)
Bronze	1840°F (1005°C)

(Reference: Machinery's Handbook, 20th ed., 1976)

Titanium	3300°F (1820°C)
Beryllium Copper	1600°F (870°C)
Zircalloy	3350°F (1845°C)

(Reference: Metals Handbook, 1961)

The rigid polyurethane foam has a minimum operating range of -100°F to 200°F (-73°C to 93°C). It will decompose at the fire test temperature of 1475°F (800°C). Decomposition will result in gaseous byproducts which will burn in air.

3.3 Technical Specifications of Components

Not applicable

3.4 Normal Conditions of Transport

3.4.1 Thermal Model

The maximum heat source in the Model 741 results from the decay of 240Ci of iridium-192. Iridium-192 decays by electron capture and beta emission. The decay energy for both processes is approximately 1.45MeV. (Reference: Radiological Health Handbook, p.403). Thus:

$$1.45\text{MeV} \times 3.7 \times 10^{10} \frac{\text{disint}}{\text{s-Ci}} \times \frac{1.6 \times 10^{-13}\text{J}}{\text{MeV}} \times 240\text{Ci} = 2.06 \text{ watts}$$

The decay heat source is conservatively taken as 2.5 watts.

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To qualify as a Type B(U) package the requirements of IAEA Safety Series No. 6, 1973, paragraphs 231 and 232 must be satisfied. The calculational model used to demonstrate compliance with these regulations is described in detail in Section 3.6, along with the results of the analysis. Essentially, it is assumed that one-fourth of the entire decay heat load is deposited uniformly in each of six sides. The smallest of the sides is assumed to reach the maximum surface temperature. Heat transfer from the side is restricted only to convective heat transfer from the upper face of the plate.

To meet the additional requirements of paragraph 240 of the IAEA regulations, a separate analysis was performed. To do this a heat balance was set up over the surface of the package, using the insolation data in Table III of the IAEA regulations. The decay heat source was considered negligible. The outer shell was assumed to be insulated from the interior of the package. Heat transfer from the package was taken to occur by radiation, and over specific surface areas by free convection. A detailed description of the model is given in the analysis in Section 3.6.

3.4.2 Maximum Temperatures

An examination of the melting points of the materials used in construction of the Model 741 show that the maximum temperatures encountered under normal conditions of transport engender no loss of structural integrity or loss of shielding of the package. The specific Type B(U) analyses (Section 3.6) show the package temperature to be below 40°C (104°F) in the shade and below 65°C (149°F) when insolated.

3.4.3 Minimum Temperatures

The minimum normal operating temperature of the Model 741 is -40°C (-40°F).

This temperature will have no adverse effect on the package.

3.4.4 Maximum Internal Pressure

Normal operating conditions generate negligible internal pressures. Any pressure generated is significantly below that of the hypothetical accident pressure, which is shown to result in no loss of shielding or containment.

3.4.5 Maximum Thermal Stresses

The maximum temperatures that occur during normal transport are low enough to insure that thermal gradients will cause no significant thermal stresses.

3.4.6 Evaluation of Package Performance for Normal Conditions of Transport

The thermal conditions of normal transport are obviously insignificant from a functional point of view for the Model 741. Also, the applicable conditions of IAEA Regulations for Type B(U) packages have been shown to be satisfied by the Model 741.

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3.5 Hypothetical Accident Thermal Evaluation

3.5.1 Thermal Model

The Model 741, including the source assembly, is assumed to reach the fire test temperature of 800 C (1475 F). At this temperature the polyurethane potting compound will have decomposed and the resulting gases will have escaped the package through the assembly joints which are not leak-tight.

3.5.2 Package Conditions and Environment

The Model 741 is considered to have undergone no significant damage during the free drop and puncture tests; thus, the package in this analysis is assumed to be free from functional damage.

3.5.3 Package Temperatures

As indicated in 3.5.1, the package reaches a maximum of 800°C (1475°F) throughout. An examination of the melting points of the materials used in the construction of the Model 741 (except the potting compound, as noted) indicates that there will be no damage to the package as a result of this temperature. The possibility of the formation of the iron-uranium eutectic alloy was addressed in Section 2.4.1, where it was concluded that the formation of the alloy was unlikely.

3.5.4 Maximum Internal Pressures

The Model 741 packaging is open to the atmosphere, insuring that there will be no pressure buildup within the package. In Section 3.6 there is an analysis of the source capsules under the fire test conditions. It is shown that the maximum internal gas pressure at this temperature is 54.7 psi (0.377MN/m^2).

The critical location for failure of the capsule is the weld. An internal pressure of 54.7 psi (0.37MN/m^2) will generate a maximum stress of 287 psi (1.98MN/m^2) in the weld. At a temperature of 870°C (1600°F) the yield strength of Type 304 or 304L stainless steel is 10,000 psi (69.0MN/m^2).

Thus, at 800°C (1475°F), the maximum stress in the capsule would only be 3% of the yield strength at that point.

3.5.5 Maximum Thermal Stresses

There are no significant thermal stresses generated during the thermal test.

3.5.6 Evaluation of Package Performance

The Model 741 will undergo no loss of structural integrity or shielding when subjected to the conditions of the hypothetical thermal accident. The pressures and temperatures generated have been demonstrated to be within acceptable limits.

3.6 APPENDIX

- Model 741 Thermal Analysis: IAEA Safety Series No. 6, 1973, paragraphs 231, 232
- Model 741 Thermal Analysis: IAEA Safety Series No. 6, 1973, paragraph 240
- Thermal Analysis, 0.25 inch O.D Capsules

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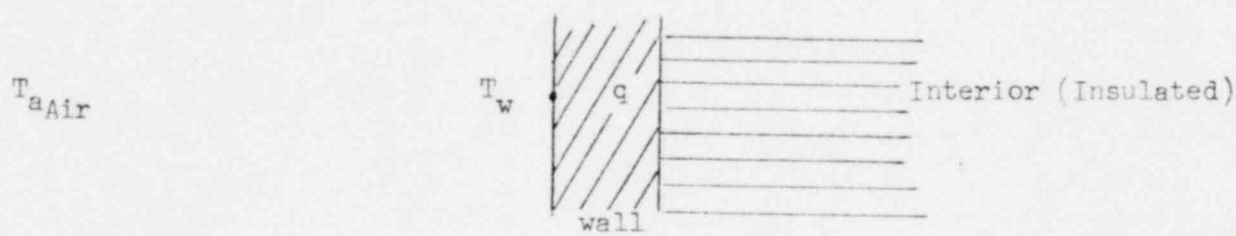
Model 741 - Thermal Analysis

Type B(U), Paragraphs 231, 232, IAEA Safety Series No. 6, 1973

This analysis is performed to demonstrate that the Model 741 Gamma Ray Projector meets the specific Type B(U) thermal requirements of paragraphs 231 and 232 of IAEA Safety Series No. 6, 1973, i.e., that the maximum surface temperature does not exceed 50°C in the shade assuming 38°C ambient temperature.

To assure conservatism, it is assumed that: (1) the entire decay heat (2.5 watts) is deposited in the exterior faces of the Model 741, (2) the interior of the Model 741 is perfectly insulated, providing heat transfer from the wall only to the atmosphere. The rectangular shape of the container means that each face eclipses a different amount of the solid angle through which the radiation (and thus decay heat) is distributed. To (conservatively) simplify, it is assumed that each of the six exterior faces receives $\frac{1}{4}$ of the total source (0.63 watts) uniformly distributed over the face.

Considering the smallest face as undergoing one-dimensional convective heat transfer:



$$T_w = \frac{q}{hA} + T_a \quad \text{where:}$$

T_w: temperature at the wall outer surface

q: (decay) heat source (0.63 watts)

A: surface area of the smallest face (0.10 m^2)

h: free convective heat transfer coefficient for air

5 watts/meter^2 - °C (Reference: Heat Transfer, J. P. Holman, 4 th Edition, p. 13)

Thus, the maximum temperature at the wall T_w is 40°C (104°F) under normal conditions of transport. This satisfies the requirements of the aforementioned regulations.

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Model 741 - Thermal Analysis

Type B(U), Paragraph 240, IAEA Safety Series No. 6, 1973

This analysis is performed to demonstrate that the Model 741 Gamma Ray Projector meets the specific Type B(U) thermal requirements of paragraph 240, IAEA Safety Series No. 6, 1973. This paragraph requires that the maximum surface temperature of a Type B(U) package not exceed 82°C (180°F) under normal conditions of transport, given insolation as outlined in Table III of the regulations and an ambient temperature of 38°C (100°F).

The calculational model used consists of taking a steady state heat balance over the surface of the package. To facilitate calculations, certain simplifying assumptions are made. These are outlined below:

Insolation

800 cal/cm^2-12hr (775 W/m^2) for the top surface, 200 cal/cm^2-12hr (194 W/m^2) for the sides and side frames, none for the base as outlined in Table III of IAEA Safety Series No. 6.

The package is finished with russett enamel. The solar absorptivity of this enamel is 0.81 (Reference: Thermal Radiation Properties Survey, G. G. Gubareff et. al., 2nd ed., 1960, p. 260). A conservative figure of 0.90 was used as the package absorptivity.

Decay Heat Load

The decay heat load (maximum 2.5 watts) is assumed negligible.

Package Orientation

The package rests on the side frames, i.e., in the normal transport orientation.

Heat Transfer Mechanisms

The Model 741 is assumed to undergo free convection and to radiate to the environment. The inside faces are considered to be insulated, so there is no conduction into the package. Further, the sides are taken to be thin enough so there are no temperature gradients present.

Radiation: The package is assumed to radiate from the outer shell only, i.e., a cube 10.1 inches (257mm) x 12.6 inches (320mm) x 13 inches (330mm).
This assumption provides for conservatism by not considering any radiative heat loss through the side frames.

Convection, top: The upper surface of the outer shell is taken to undergo free convection. To provide conservatism, the upper surfaces of the side frames are considered not to undergo convection. The heat transfer coefficient of a horizontal flat plate is given by:

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$$h_t = 1.32\left(\frac{\Delta T}{L}\right)^{0.25}$$

(Reference: Heat Transfer, J. P. Holman, 4th ed., 1976, p. 253)

where L is the average of the lengths of the sides, 0.325m.

Thus:

$$h_t = 1.74(\Delta T)^{0.25}$$

Convection, sides: The vertical components of the outer shell are considered to exhibit free convective heat transfer. For conservatism, the side frames are taken to insulated. Effectively, the vertical convective heat transfer area is that of a vertical plate 0.257m high x 2(0.320m + 0.330m) long, convecting on one side only.

The heat transfer coefficient for a vertical flat plate is:

$$h_s = 1.42\left(\frac{\Delta T}{L}\right)^{0.25}$$

(Reference: Heat Transfer, J. P. Holman, 4th ed., 1976, p. 253)

where L is the height of the plate, 0.257m.

Thus:

$$h_s = 1.99\ (\Delta T)^{0.25}$$

Taking a heat balance over the package surface:

heat in = heat out (rad. + con. top + conv. sides)

$$q_{in} = q_{rad} + q_{ct} + q_{cs}$$

$$q_{in} = 0.90\ (775\ \frac{W}{m^2}\ \times\ A_t\ +\ 194\ \frac{W}{m^2}\ \times\ A_s\ )$$

$$= 0.90\ (775 \times 0.11 + 194 \times 0.48) = 161 \text{ watts}$$

$$q_{rad} = \epsilon \sigma A_r\ (T_w^4 - T_a^4)$$

$$= (0.8)\ (5.669 \times 10^{-8}\ \frac{W}{m^2 - {}^{\circ}k^4}\ )\ (0.55m^2) \left[T_w^{\ 4} - (311^{\circ}k)^4\right]$$

$$q_{ct} = h_t A_t\ (\Delta T) \quad \text{where } \Delta T = T_w - T_a$$

$$= \left[1.74 \times (\Delta T)^{0.25}\right] (0.106m^2)\ (\Delta T)$$

$$q_{cs} = h_s A_s'\ (\ T)$$

$$= \left[1.99\ (\Delta T)^{0.25}\right] (0.48m^2)\ (\Delta T)$$

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Iteration yields a wall temperature T_w of $65^{o}C$ ($149^{o}F$). Thus, the Model 741 satisfies the requirements of paragraph 240, IAEA Safety Series No. 6, 1973.

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0.25 Inch O.D. Source Capsules - Thermal Analysis

Hypothetical Fire Conditions

This analysis is intended to demonstrate that Tech/Ops source capsules which are of 0.25 inch (6.35mm) diameter, seal welded to a minimum penetration of 0.020 inch (0.51mm), made of Type 304 or 304L stainless steel, and licensed as special form containers under IAEA Safety Series No. 6, 1973, also meet the requirements of paragraph 238, IAEA Safety Series No. 6, 1973, i.e., containment under specified thermal test conditions.

The actual containment vessel for the radioactive material is the welded source capsule. These capsules are all 0.25 inches (6.35mm) in diameter and less than 1 inch (25.4mm) in length.

The internal volume of the source capsules contains only iridium-192 or cobalt-60 metal (as a solid) and air. It is assumed at the time of loading that the entrapped air in the capsule is at standard temperature and pressure (20°C, 0.101 Meganewtons per square meter). We contend that this is a conservative assumption because, during the welding process the internal air is heated, causing some of the air mass to escape before the capsule is sealed. When the welded capsule returns to ambient temperature, the internal pressure is somewhat reduced.

As described in Tech/Ops standard source encapsulation procedure, the minimum weld penetration is 0.020 inch (0.51mm). Under conditions of internal pressure, the critical location for failure is this weld. Since the capsule has an outside diameter of 0.25 inch (6.35mm), this weld has a cross-sectional area of 0.014 square inches ($9.30mm^2$).

Under conditions of paragraph 238 of IAEA Safety Series, No. 6, it is assumed that the capsule could reach a temperature of 1475°F (800°C). Using the ideal gas law and requiring the air to occupy a constant volume:

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

P_1 = initial pressure($0.101 MN/m^2$)

T_1 = initial temperature (293°k)

T_2 = final temperature (1093°k)

The internal gas pressure could reach $0.377 MN/m^2$. It is assumed that the capsule can be treated as a thin-walled, cylindrical pressure vessel.

The maximum longitudinal tensile stress can be calculated by writing a longitudinal force balance through the weld:

$$\text{stress x area}_s - \text{pressure x area}_p = 0$$

$$S_l \pi \frac{(D_o^2 - D_i^2)}{4} - P \pi \frac{D_i^2}{4} = 0$$

where S_l = longitudinal stress

D_o = outer diameter (6.35cm)

D_i = inner diameter (5.35cm)

P = pressure ($0.377 MN/m^2$).

Thus, the longitudinal stress is $0.922 MN/m^2$

The hoop stress can be found in a similar fashion. Taking a longitudinal cross-section and summing forces:

$$\text{hoop stress x area}_s - \text{pressure x area}_p = 0$$

$$2S_h Lt - pD_i L = 0$$

where S_h = hoop stress

L = length of cylinder

t = thickness of weld (0.51m m)

Thus, the hoop stress is $1.98 MN/m^2$

At a temperature of 1600°F (870°C) the yield strength of type 304 stainless steel is 10,000 psi ($69.0 MN/m^2$) Thus, the pressure induced stresses are less than 3% of the yield strength at 800°C.

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4. Containment

4.1 Containment Boundary

4.1.1 Containment Vessel

The containment system for the Model 741 Gamma Ray Projector is either Tech/Ops Model A424-9 or Model A424-18 Source Assembly. These source assemblies are currently certified (IAEA Certificates of Competent Authority Nos. USA/0165/S and USA/0154/S) as special form containment for radioactive materials.

The actual containment vessel is the welded source capsule, either styles 60004, 60011 or 60000. The capsules are made of Type 304 or 304L stainless steel. They are seal welded with a minimum weld penetration of 0.020 inch (0.51mm). The capsules are rounded cylinders 0.25 inches (6.35mm) in diameter and less than 1 inch (25.4mm) in length. Appropriate descriptive drawings are enclosed in Section 2.10.

4.1.2 Containment Penetrations

There are no penetrations of containment. The source capsule is seal welded to provide conformity to special form requirements.

4.1.3 Seals and Welds

The containment vessel is tungsten inert gas welded. This is done in accordance with Tech/Ops standard source encapsulation procedure (see section 7.4). The minimum weld penetration is 0.020 inches (0.51mm). This has proved acceptable for licensing this vessel as special form.

4.1.4 Closure

Not Applicable

4.2 Requirements for Normal Conditions of Transport

4.2.1 Release of Radioactive Material

The source assemblies used all meet the requirements of special form radioactive material as delineated in IAEA Safety Series No. 6, 1973 and 10CFR71. Thus, there will be no release of radioactive materials under conditions of normal transport.

4.2.2 Pressurization of Containment Vessel

The source assemblies used all meet the requirements of special form radioactive material. Pressure buildup due to the conditions of the hypothetical thermal accident has been shown to create stresses well below the structural limits of the capsule (see Section 3.5). Thus, the containment vessel will withstand the pressure variations of normal transport.

4.2.3 Coolant Contamination

Not Applicable

4.2.4 Coolant Loss

Not Applicable

4.3 Containment Requirements for the Hypothetical Accident Condition

4.3.1 Fission Gas Products

Not applicable

4.3.2 Release of Contents

The hypothetical accident conditions as outlined in 10CFR71, Appendix B, 1., 2., and 3. have been shown (Sections 2.7.1, 2.7.2 and 3.5 respectively) to result in no loss of package containment.

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5. Shielding Evaluation

5.1 Discussion and Results

The Model 741 is shielded with 200 pounds (91kg) of depleted uranium. The uranium metal is cast around the zircalloy or titanium "S" tube which holds the source. The storage position for the source is at the inflection in the "S" tube.

Radiation profiles of the Model 741 containing 33Ci of cobalt-60 and 240 Ci of iridium-192 (see Section 5.5) were made. The results are presented in Table 5.1. From this data, and from previous acceptability (NRC Certificate of Compliance No. 9029, Rev. 2) it is concluded that the Model 741 complies with the regulatory standards in 10CFR71 and IAEA Safety Series No. 6, 1973.

TABLE 5.1

SUMMARY OF MAXIMUM DOSE RATES
(mR/hr)

	Contact			At 1 Meter		
	Side	Top	Bottom	Side	Top	Bottom
Gamma	140	90	135	2.0	2.0	2.0
Neutron	Not Applicable			Not Applicable		
Total	140	90	135	2.0	2.0	2.0

Hypothetical accident conditions will result in essentially no change in the above readings.

5.2 Source Specification

5.2.1 Gamma Source

The gamma sources used are encapsulated cobalt-60 in quantities of up to 33 curies, or iridium-192 up to 240 curies.

5.2.2 Neutron Source

Not Applicable

5.3 Model Specifications

Not Applicable

5.4 Shielding Evaluation

The shielding evaluation was performed on the Model 741 containing 33 curies of cobalt-60 and 240 curies of iridium-192. The radiation profile is included in Section 5.5. Extrapolation of this data to the maximum capacity of the

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package (Section 5.1) clearly indicates that the Model 741 conforms to regulatory radiation limits. As the hypothetical a ident evaluation (Section 2.7) revealed no change in the shielding arrangement, it is concluded that shielding after the hypothetical accident is essentially unchanged. Therefore, the radiation profile indicates the package will be within acceptable limits.

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5.5 APPENDIX

- Model 741: Radiation Profile

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RADIATION PROFILE

Model 741 Serial Number 186

Containing 33.0 Curies of ^{60}Cobalt

(AN/PDR-27(J))

Location	At Contact	At 1 Meter
Top	90	1.0
Bottom	135	2.0
Front	140	2.0
Rear	120	2.0
Left	115	2.0
Right	110	1.5

Model 741 Serial Number 150

Containing 240 Curies of ^{192}Iridium

(AN/PDR - 27(J))

Top	1.6	Less than 1
Bottom	1.4	Less than 1
Front	2.8	Less than 1
Rear	2.3	Less than 1
Left	1.6	Less than 1
Right	1.9	Less than 1

NOTES: All intensities are expressed in units of milliroentgens per hour.

Intensities given are the maximum intensities on the measured side.

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6. Criticality Evaluation

Not Applicable

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7. Operating Procedures

7.1 Procedures for Loading the Package

Section 7.4 describes the procedure for fabricating the special form source encapsulation. Section 7.4 also contains the procedure for loading this source assembly into the package and preparing the package for transport.

7.2 Procedures for Unloading the Package

Section 7.4 contains the procedure for unloading the source assembly from the package.

7.3 Preparation of an Empty Package for Transport

Section 7.4 describes the procedure for preparing an empty package for transport.

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7.4 APPENDIX

- Encapsulation of Sealed Sources
- Technical Operations Model 741: Procedures of Loading - Unloading the Package

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RADIATION SAFETY MANUAL
Part II In Plant Operations
Section 2

ENCAPSULATION OF SEALED SOURCES

A. Personnel Requirements

Only an individual qualified as a Senior Radiological Technician shall perform the operations associated with the encapsulation of ^{192}Iridium. There must be a second qualified Radiological Technician available in the building when these operations are being performed.

B. General Requirements

The ^{192}Iridium loading cell shall be used for the encapsulation of solid metallic ^{192}Iridium and the packaging of sealed sources such as ^{170}Thulium, ^{137}Cesium and ^{169}Ytterbium. Solid metallic ^{60}Cobalt not exceeding one curie may be handled in this cell also.

The maximum amount of ^{192}Iridium to be handled in this cell at any one time shall not exceed 1000 curies. The maximum amount of ^{137}Cs to be handled in this cell at any one time shall not exceed 100 curies.

This cell is designed to be operated at less than atmospheric pressure. The exhaust blower provided shall not be turned off except when the cell is in a decontaminated condition.

Sources shall not be stored in this cell overnight or when cell is unattended. Unencapsulated material shall be returned to the transfer containers and encapsulated sources transferred to approved source containers.

When any of the "through-the-wall" tools such as the welding fixture or transfer pigs are removed, the openings are to be closed with the plugs provided. These tools shall be decontaminated whenever they are removed from the hot cell.

C. Preparatory Procedure

1. Check welding fixture, capsule drawer and manipulator fingers from cell and survey for contamination. If contamination in excess of 0.001 μCi of removable contamination is found, these items must be decontaminated.

2. If the welding fixture or the electrodes have been changed, perform the encapsulation procedure omitting the insertion of any activity. Examine this dummy capsule by sectioning thru weld. Weld penetration must be not less than 0.020 inch.

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If weld is sound and penetration is at least 0.020 inch, the preparation of active capsules may proceed. If not, the condition responsible for an unacceptable weld must be corrected and the preparatory procedure repeated.

3. Check pressure differential across first absolute filter, as measured by the manometer on the left side of the hot cell. This is about ½ inch of water for a new filter. When this pressure differential rises to about 2 inches of water, the filter must be changed.

D. Encapsulation Procedure

1. Prior to use, assemble and visually inspect the two capsule components to determine if weld zone exhibits any misalignment and/or separation. Defective capsules shall be rejected.

2. Degrease capsule components in the Ultrasonic Bath, using isopropyl alcohol as degreasing agent, for a period of 10 minutes. Dry the capsule components at 100°C for a minimum of twenty minutes.

3. Insert capsule components into hot cell with the posting bar.

4. Place capsule in weld positioning device.

5. Move drawer of source transfer container into hot cell.

6. Place proper amount of activity in capsule . Disposable funnel must be used with pellets and a brass rivet with wafers to prevent contamination of weld zone.

7. Remove unused radioactive material from the hot cell by withdrawing the drawer of the source transfer container from the cell.

8. Remove funnel or rivet.

9. Assemble capsule components.

10. Weld adhering to the following conditions:

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   a. Electrode spacing .021" to .024" centered on joint ±.002"; use jig for this purpose.

   b. Preflow argon, flush 10 seconds.

   c. Start 15 amps.

   d. Weld 15 amps.

   e. Slope 15 amps.

   f. Post flow 15 seconds

11. Visually inspect the weld. An acceptable weld must be continuous without cratering, cracks or evidence of blow out. If the weld is defective, the capsule must be cleaned and rewelded to acceptable conditions or disposed of as radioactive waste.

12. Check the capsule in height gauge to be sure that the weld is at the center of the capsule.

13. Wipe exterior of capsule with flannel patch wetted with EDTA solution or equivalent.

14. Count the patch with the scaler counting system. Patch must show no more than .005 μCi of contamination. If the patch shows more than .005 μCi, the capsule must be cleaned and rewiped. If the rewipe patch still shows more than 0.005 μCi of contamination, steps 8 through 11 must be repeated.

15. Vacuum bubble test the capsule. Place the welded capsule in a glass vial containing isopropyl alcohol. Apply a vacuum of 15 in Hg(Gauge). Any visual detection of bubbles will indicate a leaking source. If the source is determined to be leaking, place the source in a dry vacuum vial and boil off the residual alcohol. Reweld the capsule.

16. Transfer the capsule to the swaging fixture. Insert the wire and connector assembly and swage. Hydraulic pressure should not be less than 1250 nor more than 1500 pounds.

17. Apply the tensile test to assembly between the capsule and connector by applying proof load of 75 lbs. Extension under the load shall not exceed 0.1 inch. If the extension exceeds 0.1 inch, the source must be disposed of as radioactive waste.

18. Position the source in the exit port of hot cell. Withdraw all personnel to the control area. Use remote control to insert source in the ion chamber and position the source for maximum response. Record the meter reading. Compute the activity in curies and fill out a temporary source tag.

19. Using remote control, eject the source from cell into source changer through the tube gauze wipe test fixture. Monitor before reentering the hot cell area to be sure that the source is in the source changer. Remove the tube gauze and count with scaler counting system. This assay must show no more than 0.005 μCi. If contamination is in excess of this level, the source is leaking and shall be rejected.

20. Complete a Source Loading Log (Figure II.2.1) for the operation.

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Technical Operations
Model 741

Procedures for Loading - Unloading the Package

Wear personnel monitoring devices during all source changing procedures. Monitor all operations with a calibrated, operable survey meter.

> Note: All the precautions used when making radiographic exposures must be followed.

1. Survey the projector to ensure that the source is in the proper position.

2. Locate the projector and source changer in a restricted area. Locate the devices so as to avoid sharp bends in the guide tube or control housing.

   The control cable housing bend radius should not be less than 36 inches (0.914m), and the guide tube bend radius should not be less than 20 inches (0.508m).

3. Set the source changer for operation.

4. Attach one end of a guide tube fitting to the fitting above the empty chamber in the source changer and the other end to the projector.

5. Attach the control cable to the projector:

   a. Unlock the projector with the key provided and turn the connector selector ring from the LOCK position to the CONNECT position. When the ring is in the CONNECT position, the storage cover will disengage from the projector.

   b. Slide the control cable collar back and open the jaws of the swivel connector, exposing the male portion of the connector. Engage the male and female portions of the swivel connector by depressing the spring loaded locking pin toward the projector with the thumbnail. Release the locking pin and test that the connection has been made.

   c. Close the jaws of the control cable connector over the swivel type connector.

   d. Slide the control cable collar ov-r the connector jaws. Hold the control cable collar flush against the projector connector and rotate the selector ring from the CONNECT position to the OPERATE position.

6. Crank the source into source changer.

   a. Survey this operation with a survey meter to be sure the source

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has been transferred from projector to changer.

b. With a survey meter verify radiation level does not exceed 200 mR/hr at the surface of the changer.

7. Disconnect the control cable from the source assembly. Disconnect the guide tube from the source changer. Secure the source in the source changer.

8. IF THE PROJECTOR IS TO REMAIN EMPTY:

a. Fully retract the control cable. Disengage the control cable from the projector and lock the projector.

b. Attach the identification plate of the source to the source changer.

c. Affix a green "empty" tag to projector.

d. Perform a wipe test of the projector to assure that the radiation observed is less than 0.001 microcuries per 100 square centimeters.

e. Survey the projector to assure that the radiation levels do not exceed 200mR/hr at the surface nor 10mR/hr at three feet from the surface.

f. Mark the projector: Radioactive - "LSA". Affix the proper shipping labels to the package.

g. Complete the proper shipping papers as specified in Tech/Ops Radiation Safety Manual II.6.3E(4), (5), (6).

9. IF THE PROJECTOR IS TO BE RELOADED: connect the source changer end of the guide tube to the fitting above the new source in the source changer.

10. Crank source to full retraction within the projector.

a. Survey this operation with a survey meter to be sure the source has been transferred into the projector.

b. With a survey meter verify radiation level does not exceed 200mr/hr at the surface of the projector.

11. Disconnect the control cable and lock the projector.

12. Disconnect the source guide tube from the projector and source changer.

13. Affix the identification plate of the new source to the projector and attach the identification plate of the old source to the source changer.

14. Prepare for shipment:

a. Again survey projector to insure that the radiation level does not exceed 200mr/hr at the surface of the projector.

b. Survey the radiation level at a distance of three feet from the surface of the projector. This radiation level should not exceed 10mr/hr. The highest radiation level measured at three feet from the container is used to determine the Transport Index in accordance with 49CFR173.389(h).

c. Affix the proper shipping labels.

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8. Acceptance Tests and Maintenance Program

8.1 Acceptance Tests

8.1.1 Visual Acceptance

The package is visually examined to assure that the appropriate fasteners are seal wired properly and that the package is properly marked.

The seal weld of the radioactive source capsule is visually imspected for proper closure.

8.1.2 Structural and Pressure Tests

The swage coupling between the source capsule and cable is subjected to a static tensile test with a load of seventy-five pounds. Failure of this test will prevent the source assembly from being used.

8.1.3 Leak Tests

The radioactive source capsule (the primary containment) is wipe tested for leakage of radioactive contamination. The source capsule is subjected to a vacuum bubble leak test. The capsule is then subjected to a second wipe test for leakage of radioactive contamination. These tests are described in Section 7.4. Failure of any of these tests will prevent use of this source assembly.

8.1.4 Component Tests

The lock assembly of the package is tested to assure that security of the source will be maintained. Failure of this test will prevent use of the package until the lock assembly is corrected and retested.

8.1.5 Tests for Shielding Integrity

The radiation levels at the surface of the package and at three feet from the surface are measured using a small detector survey instrument (e.g., AN/PDR-27). These radiation levels, when extrapolated to the rated capacity of the package, must not exceed 200 milliroentgens per hour at the surface nor ten milliroentgens per hour three feet from the surface of the package. Failure of this test will prevent use of the package.

8.1.6 Thermal Acceptance Tests

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Not Applicable

8.2 Maintenance Program

8.2.1 Structural and Pressure Tests

Not Applicable

8.2.2 Leak Tests

As described in Section 8.1.3, the radioactive source assembly is leak tested at manufacture. Additionally, the source assembly is wipe tested for leakage of radioactive contamination every six months.

8.2.3 Subsystem Maintenance

The lock assembly is tested as described in Section 3.1.4, prior to each use of the package. Additionally, the package is inspected for tightness of fasteners, proper seal wires and general condition prior to each use.

8.2.4 Valves, Rupture Discs and Gaskets

Not Applicable

8.2.5 Shielding

Prior to each use, a radiation survey of the package is made to assure that the radiation levels do not exceed 200 milliroentgens per hour at the surface nor ten milliroentgens per hour at three feet from the surface.

8.2.6 Thermal

Not applicable

8.2.7 Miscellaneous

Inspections and tests designed for secondary users of this package under the general license provisions of 10CFR71.12(b) are included in Section 7.4.

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