71-9269

QSA GLOBAL

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30 June 2006

Ms. Jill Caverly
Project Manager
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
One White Flint
Rockville, MD 20852

RE: TAC No. L23950, Supplemental Information for Current Amendment USA/9269/B(U)-96 for the Model 650L Transport Package

Dear Ms. Caverly:

The following is provided in response to your email on 28 June 2006 and our subsequent conversation on 29 June 2006.

1. Section 5 has been revised to clarify the method used perform surface correction factors for newly manufactured 650L devices.

2. Tables 5.1a through 5.1c have been revised to list the 1 meter dose rate measurements in units of mSv/hr (mrem/hr) instead of the previous values of μSv/hr (mrem/hr). This is done for consistency with the other tables in this section.

3. Further clarification was provided in our discussion regarding the response to G-1(j) and no further response is necessary at this time.

4. As discussed, Section 7.4.1.1. applies to the transportation vehicles/accessory equipment that may be used when the package is not offered to a common carrier for transport but is transported by the package user on their company vehicles. These surveys are suggested as they apply to the transportation vehicle and not the package. Package survey requirements are referenced, as required by the regulations, in Section 7.1.3.2 and 7.3.2.

5. The preliminary paragraph in Section 7 has been revised to include a specification of the version of IAEA that applies in the references throughout Sections 7 and 8 of the SAR.

6. The changes to Drawing R65006 Rev H were described in our letter dated 16 May 2006 and included only a minor change on page 1 which changed the "-85" reference to "-96". No other changes were made from Rev G of the drawing previously submitted.

NMSS01

Revision 5 to the Model 650L SAR is submitted in its entirety with the exception of Section 2.12. As these documents were unchanged in this Revision we request that Section 2.12 provided with Revision 3 of the SAR be consolidated into the enclosed copy of Revision 5 of the SAR. Changes to the text of Revision 5 of the SAR addressing items discussed in this letter are indicated by vertical lines in the right hand margin. Should you have any additional questions or wish to discuss this submission, please contact me as shown below.

Sincerely,

Lori Podolak

C. Compton — RA/QA Approval

30 Jun 06 — Date

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Enclosure:

- SAR Revision 5, Minus Section 2.12 (These sections unchanged from Revision 4 & 3 of the SAR)
- List of Affected Pages

List of Affected Pages	
Revision 1, 19 November, 1999	Replacement of Appendix A Index; Added Drawing R65006, Revision F to Appendix A.
Revision 2, 31 October, 2000	Replacement of Appendix A Index; Added Drawing R65006, Revision G to Appendix A.
Revision 3, 31 October 2005	Reformatting of the SAR to cover compliance to IAEA TS-R-1 and NRC guidance format. Addition of Se-75 as authorized nuclide.
Revision 4, 6 June 2006	Changes to Sections 1, 2, 5, 7 and 8 to address NRC RAI dated 31 May 2006
Revision 5, 29 June 2006	Changes to pages 5-1, 5-2, 5-5, 5-6 and 7-1.

Safety Analysis Report

QSA Global Inc.

Model 650L
Type B(U) - 96
Transport Package

29 June 2006

Revision 5

Contents

List of Tables

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Section 1 - GENERAL INFORMATION

1.1 Introduction

The Model 650L is designed as an industrial radiography source changer and transport package for Type B quantities of special form radioactive material. It conforms to the Type B(U)-96 criteria for packaging in accordance 10 CFR 71, 49 CFR 173, and the IAEA Regulations for the Safe Transport of Radioactive Material (TS-R-1) which were in effect at the time of sign-off of this report.

1.2 Package Description

(Reference:

- *10 CFR 71.33*
- *IAEA TS-R-1, paragraph 220 & 807)*

The Model 650L package is constructed in accordance with the drawings included in Section 1.4. The package measures approximately 13 ¼ inches (337 mm) in tall by 10 inches (254 mm) wide by 8 ¼ inches (210 mm) deep. The general package information is shown in Table 1.2a:

Table 1.2a: Model 650L Package Information

Identification	Nuclide	Form	Maximum Capacity[1]	Chemical/ Physical Form	Maximum Content Weight	Maximum Decay Heat[3]	Maximum DU Weight	Maximum Package Weight
650L	Ir-192	Special Form[2] Sources	240 Ci	Metal	< 1 gram	4.8 Watts	44 lbs (20 kg)	90 lbs (41 kg)
	Se-75	Special Form[2] Sources	300 Ci	Metal-Selenide Compound	< 1 gram	1.52 Watts		

[1] Maximum Activity for Ir-192 is defined as output Curies as required in ANSI N432 and 10 CFR 34.20 and in line with TS-R-1 and Rulemaking by the USNRC and the USDOT published in the Federal Register on 26 January 2004.
[2] Special Form is defined in 10 CFR 71, 49 CFR 173, and IAEA TS-R-1.
[3] Maximum decay heat for Ir-192 is calculated by correcting the output activity to content activity. A factor of 2.3 is used for Ir-192 to account for source capsule and self-absorption in this conversion. No corrections are made for Se-75.

1.2.1 Packaging

Except for the shield assembly, fill foam and some components of the lock assembly, all materials of construction are stainless steels. The major components of the package consist of the following:

- Inner and Outer Shells
- Depleted Uranium shield
- Locking assemblies
- Protective Lid

The following paragraphs describe the major components of the transport package.

1.2.1.1 Source Capsule and Shield Assembly: The special form capsule is shielded by a titanium "U" tube set in depleted uranium. The tube is crimped in the middle of the "U" to provide a positive stop for the source assembly and two "effective" source tubes in the shield. On some source changers, additional shielding is provided by lead positioned at various locations around the depleted uranium shield. The lead is secured in place prior to setting the shield in the shell configuration with polyurethane foam (see Section 1.2.1.2).

1.2.1.2 Inner and Outer Shells: The shield assembly is protected by two carbon steel shells. The inner shell is rectangular and the outer shell is cylindrical. The shells are positioned between two stainless steel top and bottom plates. The plates are secured with four stainless steel, 5/16-18 hex head through bolts. All steel-uranium interfaces are separated with copper shims. The void between the depleted uranium shield and the inner and outer shells is filled with rigid polyurethane foam.

1.2.1.3 Protective Lid: During transport, the locking assembly is protected by a carbon steel lid. The lid is secured to the top plate by four stainless steel, 3/8-16 hex head bolts. The lid is provided with a tamper indicating seal during shipment.

1.2.1.4 Source Locking Assembly: The package incorporates two stainless steel and brass locking assemblies which secure the source(s) inside the shield. Each lock assembly secures one source in one source tube. The locking assemblies are secured to the top plate by four ¼-20 stainless steel screws. Two of the screws on each lock assembly are safety wired for added security of the lock assembly to the top plate.

1.2.2 Containment System

(Reference:
- *10 CFR 71.33(a)(4)*
- *IAEA TS-R-1, paragraph 213 and 501(b))*

The locking assembly on the Model 650L transport packages is shown on the drawing included in Section 1.4. The radioactive material of these source assemblies is sealed in a special form source capsule. The source capsule, stop ball and connector are swaged to a flexible steel wire to form the source wire assembly.

The containment system for the Model 650L transport package is the radioactive source capsule referred to in Section 4.1 of this Safety Analysis Report. This source capsules transported in the 650L transport package are certified as special form radioactive material under 10 CFR Part 71, USDOT regulations in 49 CFR and the IAEA Regulations for the Safe Transport of Radioactive Material (TS-R-1).

1.2.3 Contents

(Reference:

- *10 CFR 71.33(b)*
- *IAEA TS-R-1, Section IV & paragraph 807(a))*

The Model 650L transport package is designed to transport special form capsules containing the isotopes listed in Table 1.2a. The maximum decay heat for Ir-192 in Table 1.2a has been adjusted to account for content activity of the source. Based on the decay energy and total content activity, Ir-192 produces the maximum decay heat when transported in this package. Actual content to output activity varies based on the capsule configuration as well as variations in isotope self-absorption. A factor of 2.3 was used for Ir-192 to convert output activity to content activity as this factor reflects the worst case variation for Ir-192 sources transported in this package. No correction was necessary for Se-75 as the activity transported is based on content activity and not output activity. The source capsules are loaded into the Model 650L device and secured according to the procedure described in Section 7.

The maximum weight of the contents for the shield containers is also listed in Table 1.2a. The content weight values are calculated based on the package capacity and the lowest specific activity of Ir-192 (200 Ci/gram) used in source production for these devices.

Note: Ir-192 of higher specific activity can be used but this would produce sources with lower total mass of the contents. Se-75 has a lower density than Ir-192 and will produce source capsules of lesser maximum weight than their Ir-192 counterparts. Values listed in the Table 1.2a are the maximum content masses.

1.2.4 Operational Features

This package does not involve complex containment systems for source securement. The sources for this package are all special form, welded capsules. The source wire assembly is held securely in the device by components of the lock assembly attached to the top plate. One of these components, the lockslide, engages the source wire and prevents it from being pulled through the top of the lock assembly.

When the Model 650L device is prepared for transport, the lock slide is locked in the secured position by a key lock, a shipping cap is installed above the source in the lock assembly, and the protective lid is secured to the top plate over the lock assemblies.

1.3 General Requirements for All Packages

1.3.1 Minimum Package Size

(Reference:

- *USNRC, 10 CFR 71.43(a)*
- *USDOT, 49 CFR 173.412(b)*
- *IAEA TS-R-1, paragraph 634)*

The Model 650L transport package is approximately 10 inches (254 mm) long, 8 ¼ inches (210 mm) wide and 13 ¼ inches (337 mm) high. Therefore, it exceeds the minimum package size requirements specified in the referenced regulations.

1.3.2 Tamper-Indicating Feature

(Reference:

- *USNRC, 10 CFR 71.43(b)*
- *USDOT, 49 CFR 173.412(a)*
- *IAEA TS-R-1, paragraph 635)*

Two bolts which secure the protective lid of the Model 650L package to the top plate, are seal wired to provide a tamper indicating seal for the package. This seal wire is not readily breakable, therefore if it is broken during transport, it serves as evidence of possible unauthorized access to the contents.

1.4 Appendix: Drawings of the Model 650L transport packages.

Section 1.4 Appendix: Drawings of the Model 650L transport packages.

FIGURE WITHHELD UNDER 10 CFR 2.390

QSA GLOBAL 40 NORTH AVE, BURLINGTON, MA 01803		DESCRIPTIVE DRAWING	
TITLE	650L SOURCE CHANGER		
SIZE A	DWG. NO. R65006		REV H
SCALE: NONE	SHEET 1 OF 4		

FIGURE WITHHELD UNDER 10 CFR 2.390

QSA GLOBAL	SIZE A	DWG. NO. R65006	REV H
40 NORTH AVE. BURLINGTON, MA 01803	SCALE: NONE	SHEET 2 OF 4	

FIGURE WITHHELD UNDER 10 CFR 2.390

	SIZE	DWG. NO.	REV
QSA GLOBAL	A	R65006	H
40 NORTH AVE, BURLINGTON, MA 01803	SCALE: NONE	SHEET 3 OF 4	

FIGURE WITHHELD UNDER 10 CFR 2.390

QSA GLOBAL	SIZE A	DWG. NO. R65006	REV H
40 NORTH AVE, BURLINGTON, MA 01803	SCALE: NONE	SHEET 4 OF 4	

Section 2 - STRUCTURAL EVALUATION

This section identifies and describes the principal structural engineering design of the packaging, components, and systems important to safety and compliance with the performance requirements of 10 CFR Part 71.

2.1 Description of Structural Design

(Reference:
- *10 CFR 71.33(a)*
- *IAEA TS-R-1, paragraph 220 & 807(b))*

2.1.1 Discussion

The Model 650L transport package is described in Section 1.2.

2.1.2 Design Criteria

The Model 650L transport package is designed to comply with the requirements for Type B(U) packaging as prescribed by 10 CFR 71 and IAEA TS-R-1. All design criteria are evaluated by a straightforward application of the appropriate section of 10 CFR 71 or IAEA TS-R-1.

2.1.3 Weight and Centers of Gravity

The transport package weighs a maximum of 90 lbs (41 kg). The center of gravity of the 650L transport packages is approximately 4 ¼ inches (64 mm) above the bottom plate along its central axis.

2.1.4 Identification of Codes and Standards for Package Design

2.1.4.1 Package Design

See Section 2.1.2 relating to design criteria of the package. No specific codes or standards were directly incorporated in the design effort of the finished assembly for the 650L transport packages. However the design was based on the Type A and Type B(U) container requirements of 49 CFR, 10 CFR 71 and IAEA regulations in effect at the time of the package component design.

2.1.4.2 Fabrication & Assembly

All container fabrication (including assembly) is controlled under the QSA Global Inc. Quality Assurance Plan approved by the USNRC and ISO. All welding under this plan adheres to AWS or ASME standards appropriate to the materials and designs fabricated. All safety critical hardware meets ASME-B18 standards. All external fabrication deemed critical to safety is either verified to equivalent in-house standards or dedicated as appropriate for use prior to release as part of this transport package.

The containers were designed and fabricated under a QSA Global Inc. (or its predecessors) QA program in the USA. Any new manufactured containers to this design will continue to be completed under the QSA Global Inc. Quality Assurance Plan approved by the USNRC and ISO.

2.1.4.3 Maintenance & Use

Maintenance and use of these transport container assemblies is described in Sections 7 and 8.

2.2 Materials

(Reference:

- *10 CFR 71.33(a)(5)*
- *IAEA TS-R-1, paragraph 220 & 807(b))*

2.2.1 Material Properties and Specifications

Table 2.2a lists the relevant mechanical properties (at ambient temperature) of the principal materials used in the Model 650L transport package. The location and use of these materials is shown on the drawings contained in Section 1.4. The reference for the table information is listed in the last column of the table.

Table 2.2a: Mechanical Properties of Principal Safety Related Transport Package Materials

Material	Tensile Strength	Yield Strength	Source
Depleted Uranium	65 kpsi	30 kpsi	Reference #2
Copper	25 kpsi	9 kpsi	Reference #3, page 224
Titanium	145 kpsi	134 kpsi	Reference 4 page 98
Lead	1.8 kpsi	8 kpsi	Reference 3 page 550
Stainless Steel	75 kpsi	30 kpsi	Reference #1, page 854
Carbon Steel	53 kpsi	36 kpsi	Reference 1, page 205
Strain-Hardened Stainless Steel Bolts	125 kpsi	100 kpsi	Reference 5, page 25

Resource references:

1. American Society for Metals. Metals Handbook, Volume 1, Tenth Edition. Ohio: Materials Park, 1990.

2. Lowenstein, Paul. *Industrial Uses of Depleted Uranium.* American Society for Metals. Metals Handbook, Volume 3, Ninth Edition.

3. American Society for Metals. Metals Handbook, Volume 2, Tenth Edition. Ohio: Materials Park, 1990.

4. American Society for Materials, Metals Handbook desk Edition , Metals Park Ohio 1985

5. ASTM Standard A193, "Standard Specification for Alloy-Steel and Stainless Steel Bolting Materials for High-Temperature Service", 1992.

2.2.2 Chemical, Galvanic or Other Reactions

(Reference:
- *USNRC, 10 CFR 71.43(d)*
- *IAEA TS-R-1, paragraph 613 and 642)*

The non-safety related materials are brass and polyurethane. These materials are more susceptible to corrosion and chemical reaction than the safety materials, but pose no threat to safety or containment. The safety related materials, used in the construction of the Model 650L transport package, are depleted uranium metal, stainless steel, carbon steel, titanium, lead (in some cases) and copper. There will be no significant chemical or galvanic action between any of these components.

To prevent the possible formation of a eutectic alloy of steel and depleted uranium during the Hypothetical Accident Conditions thermal scenario, defined by 10 CFR 71.73(c)(4), copper separators are used at all steel-uranium interfaces. The steel-uranium eutectic alloy temperature is approximately 1,337°F (725°C). However, vacuum conditions and extreme cleanliness of the surfaces are necessary to produce the eutectic alloy at this low temperature. Due to the conditions in which the depleted uranium shield components are assembled and used in the shield containers, conditions sufficient to allow formation of this eutectic do not exist. With these container constructions, there will be no significant chemical or galvanic reaction between package components during normal or hypothetical accident conditions of transport.

2.2.3 Effects of Radiation on Materials

(Reference:
- *USNRC, 10 CFR 71.43(d)*
- *IAEA TS-R-1, paragraph 613)*

Lead, depleted uranium, steel and polyurethane foam have been used in this package as well as other transport packaging for decades without degradation of the package performance over time due to irradiation from the package contents.

2.3 Fabrication and Examination

(Reference:
- *10 CFR 71.33(a)(5)*
- *IAEA TS-R-1, paragraph 232, 310, 638 and 807(b))*

2.3.1 Fabrication

Package components are procured, manufactured and inspected for use under QSA Global Inc. NRC approved QA Program Number 0040.

2.3.2 Examination

Section 8 describes the acceptance testing and routine maintenance requirements for package components used on the Model 650L packages.

2.4 Lifting and Tie-Down Standards for All Packages

2.4.1 Lifting Devices

(Reference:
- *USNRC, 10 CFR 71.45(a)*
- *IAEA TS-R-1, paragraphs 502(b), 606, 607 and 608)*

The Model 650L transport package is designed to be lifted by the bottom or top plates or the top lid. The top lid is secured to the top plate by four 3/8-16, strain-hardened stainless steel hex head bolts. The bottom plate is secured to the top plate by four 5/16-18 stainless steel hex head bolts. Since the package can be lifted by either the top or bottom plate, analysis of the stress due to lifting considers the strength of the 5/16-18 through bolts, which are more limiting than the lid bolts.

Each 5/16-18 bolt has a cross-sectional area of 0.0524 in^2 (33.8 mm^2). The yield strength of the material is 30,000 psi. Thus, each bolt can support at least 1,572 lbs (713 kg) of force, or more than 17 times the package weight before failing. Therefore, the lifting device complies with the requirements of 10 CFR 71.45(a).

2.4.2 Tie-Down Devices

(Reference:
- *USNRC, 10 CFR 71.45(b) (1) (2) (3)*
- *IAEA TS-R-1, paragraph 606 and 636)*

The Model 650L has no system of tie down devices that are a structural part of the transport package. The package can be blocked and braced according to standard transportation practices.

2.5 General Considerations

(Reference:
- *10 CFR 71.41(a)*
- *IAEA TS-R-1, paragraph 807(c))*

2.5.1 Evaluation by Test

Evaluations by direct testing are documented in Test Plan 80 Report which is contained in Section 2.12.2.

2.5.2 Evaluation by Analysis

Evaluations by analysis are described in the section they apply to in this Safety Analysis Report or when applicable in Test Plan 80 Report contained in Section 2.12.2.

2.6 Normal Conditions of Transport

2.6.1 Heat

(Reference:

- *USNRC, 10 CFR 71.71(c)(1)*
- *IAEA TS-R-1, paragraph 615, 617, 618, 637, 651, 662 and 664)*

The heat sources for the Model 650L transport package is listed in Table 1.2a. Iridium-192, releases approximately 8.6 milliwatts per Curie based on assuming a decay energy of 1.46 MeV/decay. The thermal evaluation for the heat test is described in Section 3 and is based on the decay energy of Ir-192 as this is greater than Se-75 (see Table 1.2a).

2.6.1.1 Summary of Pressures and Temperatures

(Reference:

- *IAEA TS-R-1, paragraph 615 and 661)*

Table 2.6.1.a: Summary Temperatures Normal Transport

Temperature Condition	Model 650L	Comments
Insolation (38°C in full sun)	70°C (158°F)	Section 3.4.1.1.
Decay Heating (38°C in shade)	46°C (115°F)	Section 3.4.1.2

As all components are vented to ambient, no pressure will build up in the package under Normal Transport conditions that would adversely effect package performance or integrity. Evaluation of pressures for this package are contained in Section 3.4.2 and summarized in Table 3.1.4.a.

2.6.1.2 Differential Thermal Expansion

Any thermal expansion encountered during Normal Transport will be insignificant with respect to the manufacturing tolerances for the components of this package.

2.6.1.3 Stress Calculations

Stress calculations for normal transport of this package are contained in Sections 2.4.1 and 2.7.4.3. Results of these calculations demonstrate that the package meets the requirements for Normal Transport.

2.6.1.4 Comparison with Allowable Stresses

The Model 650L package was fully tested and passed under Normal Conditions of transport. It is therefore concluded that the package will satisfy the performance requirements specified by the regulations.

2.6.2 Cold

(Reference:
- *USNRC, 10 CFR 71.71 (c)(2)*
- *IAEA TS-R-1, paragraph 615, 637 and 664)*

The carbon steel components of the Model 650L transport package are susceptible to brittle fracture at low temperature. To assess the package performance under the worst case test conditions, the drop and penetration tests described in 10 CFR 71.71(c)(7) and (10) were performed with the package at the coldest temperature referenced in the regulations. This condition was most likely to produce package failure under these test conditions due to the brittle fracture nature of these package components. The transport package successfully met Type B(U)-96 Transport Tests requirements at temperatures below -40°C (-40°F), the minimum specified in the 10 CFR 71.71(c)(2), therefore it is concluded that the Model 650L transport package will withstand the normal transport cold condition.

2.6.3 Reduced External Pressure

(Reference:
- *USNRC, 10 CFR 71.71 (c)(3)*
- *IAEA TS-R-1, paragraph 643 & 619*

The Model 650L transport package is open to the atmosphere and contains no components which could create a differential pressure relative to atmospheric conditions or components within the package. The authorized contents are special form source capsules that meet a minimum ISO 2919-1999 classification of Class 3 for pressure. This classification is more limiting than the reduced external pressure requirement as it covers 25 kN/m^2 to 7MN kN/m^2. Therefore, the reduced external pressure requirements of 3.5 psi in 10 CFR, 8.7 psi (60 kPa) in 49 CFR and IAEA will not adversely affect the package containment.

Reference: ISO 2919-1999, Radiation Protection – Sealed radioactive sources - General requirements and classification.

2.6.4 Increased External Pressure

(Reference:
- *USNRC, 10 CFR 71.71(c)(4))*

The Model 650L transport package is open to the atmosphere and contain no components which could create a differential pressure relative to atmospheric conditions. The authorized contents are special form source capsules that meet a minimum ISO 2919-1999 classification of Class 3 for pressure. This classification is more limiting than the increased external pressure requirement as it covers 25 kN/m^2 to 7MN kN/m^2. Therefore, the increased external pressure requirements of 20 psi in 10 CFR 71 will not adversely affect the package containment.

2.6.5 Vibration

(Reference:
- *USNRC, 10 CFR 71.71(c)(5)*
- *IAEA TS-R-1, paragraph 612)*

The Model 650L (or its predecessor the 650) package has been in use for over two decades without failure due to vibration. The lock attachment screws and end plate through screws are tightened to a prescribed torque and safety wired to prevent unintentional release even after repeated use. It is therefore concluded that the Model 650L packages will withstand vibration normally incident to transport.

2.6.6 Water Spray

(Reference:
- *USNRC, 10 CFR 71.71(c)(6)*
- *IAEA TS-R-1, paragraph 719, 720 and 721)*

The Model 650L transport package is constructed of water-resistant materials throughout. Therefore, the water spray test would not reduce the shielding effectiveness or structural integrity of the package.

2.6.7 Free Drop

(Reference:
- *USNRC, 10 CFR 71.71(c)(7)*
- *IAEA TS-R-1, paragraph 722(a))*

The drop test pad used in the 1.2 m free drop, 9 m drop, and puncture tests consists of a monolithic concrete base 7.4 ft x 7.3 ft x 1.25 ft thick. The approximate weight of the concrete is 14,850 lbs. A 3.9 ft x 4 ft x 1 in thick steel plate was embedded in this concrete slab at the time of its construction. Before and after testing the drop pad was visually inspected for damage which could have a significant impact on package testing.

Three test specimens, as described in Test Plan 80 Report (Section 2.12.2) were subjected to the 1.2 meter (4 foot) free drop followed by the hypothetical accident 9 m drop and puncture bar drop tests. All free drop and penetration tests were conducted with the test specimen temperatures at or below -40°C (-40°F). The test specimens included standard locking assemblies. Drop orientation impact locations for the 1.2 m free drop are shown in Figures 2.6a through 2.6c. The justification for these orientations is provided in Sections 2.6.7.1 through 2.6.7.3.

The Model 650L package maintained its structural integrity and shielding effectiveness under the normal transport drop test conditions and the package complies with the requirements of this section.

2.6.7.1 Horizontal Long-Side Down Orientation

The intent of this orientation was to determine if the depleted uranium shield could move laterally through the foam during impact (which could result in source pullout from the titanium tubes), and whether brittle failure of the inner and outer shells could occur due to the low temperature testing. The long-side down orientation was selected because the long side of the package has a stiffer configuration than the short-side. This results in a shorter deceleration and higher impact load. Testing for this orientation (shown in Figure 2.6a) was performed on test specimen TP80(A).

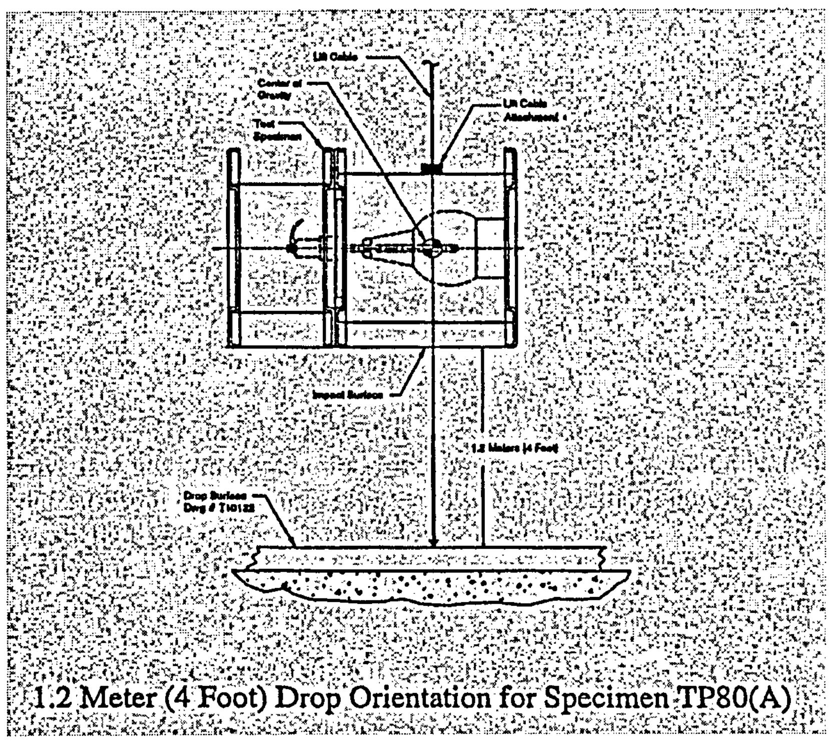


Figure 2.6a - Model 650L (TP80(A)) 1.2 m Drop Test Orientation Horizontal Long-Side Down Drop

Damage to TP80(A) was limited to impact witness marking on the bottom plate, top plate and both lid flanges. There was no significant change in the radiation profile of the test specimen after the 1.2 m (4 ft) drop test (See Section 5).

2.6.1.2 Vertical Upside Down Orientation

The intent of this test orientation was to determine if any of the following could occur and have a significant impact on the package:

2.6.1.2.1 deformation of the lid weldment,
2.6.1.2.2 crushing of the foam between the depleted uranium shield and top plate,
2.6.1.2.3 deformation (bowing upward) of the top plate due to the impact load of the depleted uranium shield applied through titanium source tubes and foam,
2.6.1.2.4 failure of the through bolts, or
2.6.1.2.5 failure of the locking assemblies could occur.

These deformations or failures could result in partial pullout of the source from its shielded position. Testing for this orientation (shown in Figure 2.6b) was performed on test specimen TP80(B).

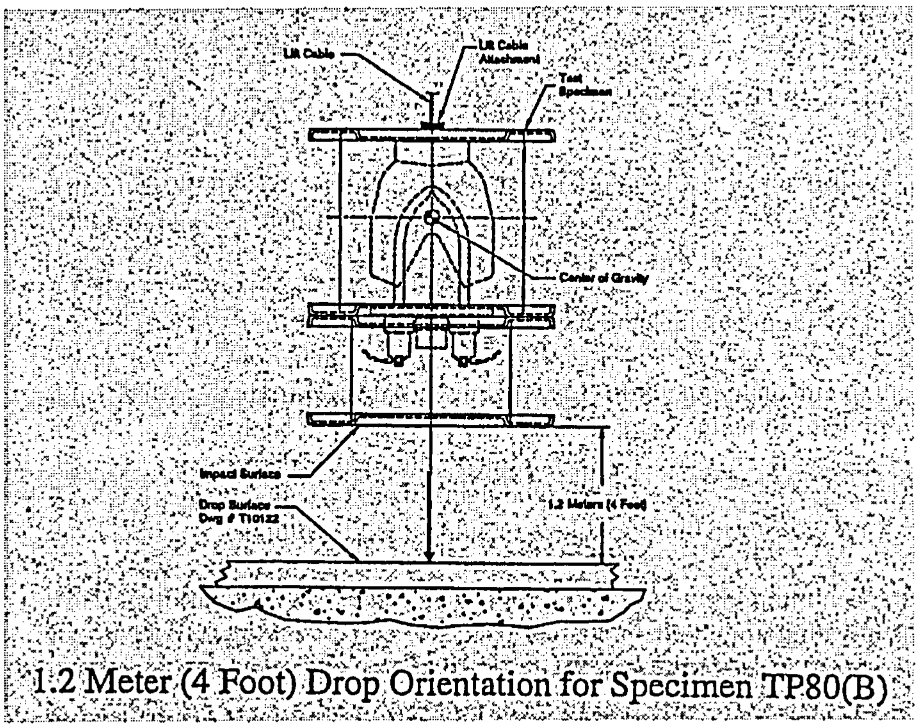
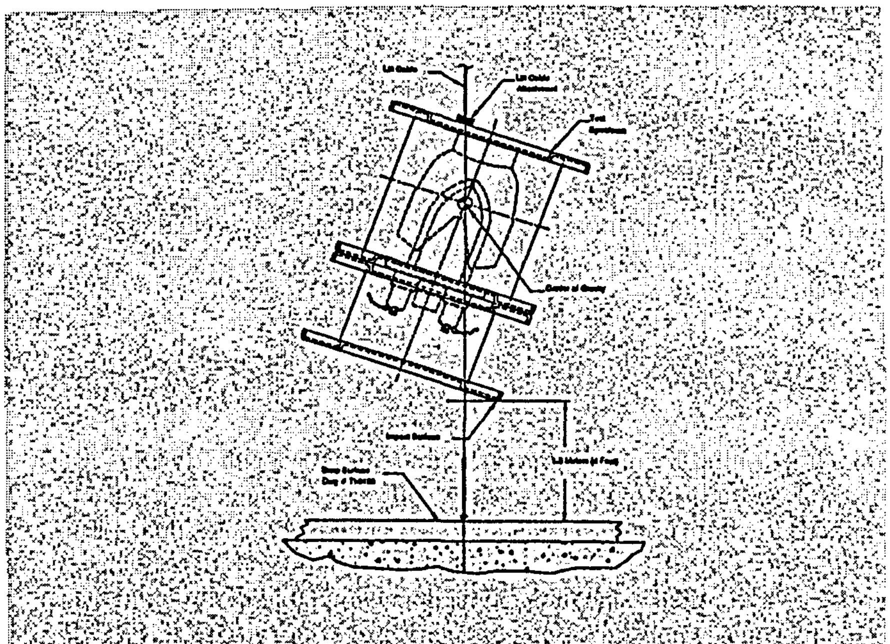


Figure 2.6b - Model 650L (TP80(B)) 1.2 m Drop Test Orientation Vertical Upside Down Drop

Damage to TP80(B) was limited to impact witness marking on the top of the lid. There was no significant change in the radiation profile of the test specimen after the 1.2 m (4 ft) drop test (See Section 5)

2.6.1.3 Vertical Top Corner Drop Orientation

The intent of this test orientation was to determine if failure of the lid or lid closure bolts could occur which could expose the locking assembly to damage during subsequent Hypothetical Accident Testing. Failure of the locking assembly could result in source pullout. Additionally, this orientation would load the through bolts in tension and could cause them to fail. Testing for this orientation (shown in Figure 2.6b) was performed on test specimen TP80(C).



1.2 Meter (4 Foot) Drop Orientation for Specimen TP80(C)

Figure 2.6c - Model 650L (TP80(C)) 1.2 m Drop Test Orientation Vertical Top Corner Down Drop

Damage to TP80(C) was limited to a 2 inch (50.8 mm) long crack in the top of the protective lid and the flange corner was bent in the drop. There was no significant change in the radiation profile of the test specimen after the 1.2 m (4 ft) drop test (See Section 5)

2.6.8 Corner Drop

(Reference:

- *USNRC, 10 CFR 71.71(c)(8)*
- *IAEA TS-R-1, paragraph 722(b))*

This test is not applicable, as the transport package does not transport fissile material, nor is the exterior of the transport package made from either fiberboard or wood.

2.6.9 Compression

(Reference:

- *USNRC, 10 CFR 71.71(c)(9)*
- *IAEA TS-R-1, paragraph 723)*

Test Plan 80 Report (Section 2.12.2) documents that the three test specimens (TP80(A), TP80(B) and TP80(C)) were subjected to compressive loads of 462 lbs (210 kg), 458 lb (208 kg) and 459 lb (208 kg) respectively, for a period of 24 hours (See Figure 2.6d). These loads exceed five times the maximum transport package weight of 90 lbs (41 kg). These loads are also greater than 13 kPa (2 lb/in^2) multiplied by the vertically projected area of the transport package.

Following the test, no damage to the specimens was observed. Radiation profiles performed at the conclusion of the test showed no significant increase in radiation levels. The Model 650L package maintained its structural integrity and shielding effectiveness and demonstrated that the packages comply with the requirements of this section.

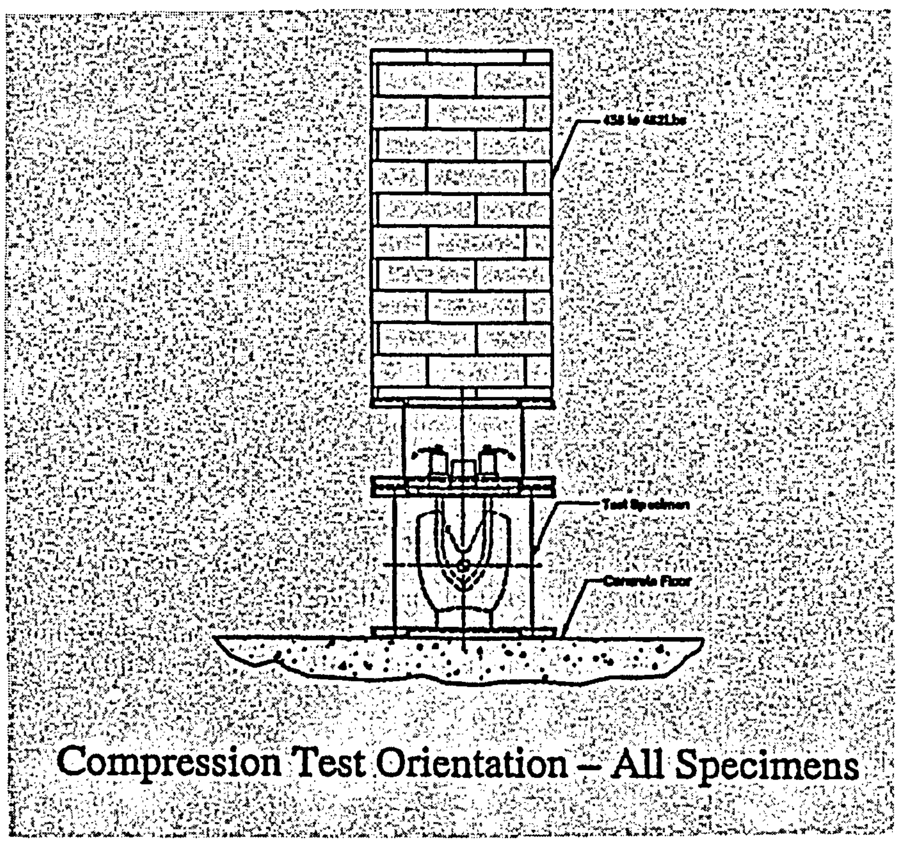


Figure 2.6d - Model 650L Compression Test Orientation

2.6.10 Penetration

(Reference:

- *USNRC, 10 CFR 71.71(c)(10)*
- *IAEA TS-R-1, paragraph 724)*

Test Plan 80 Report (Section 2.12.2) documents that the three test specimens (TP80(A), TP80(B) and TP80(C)) were subjected to the penetration test. Each specimen was impacted on the long side of the package with the intention of damaging the outer shell (See Figure 2.6e). The penetration bar impacted as intended and caused no significant damage to the specimens. In each case there was a small indentation at the point of impact.

Radiation profiles performed after testing showed no significant increase in radiation levels. The Model 650L package maintained its structural integrity and shielding effectiveness and demonstrated that the packages comply with the requirements of this section.

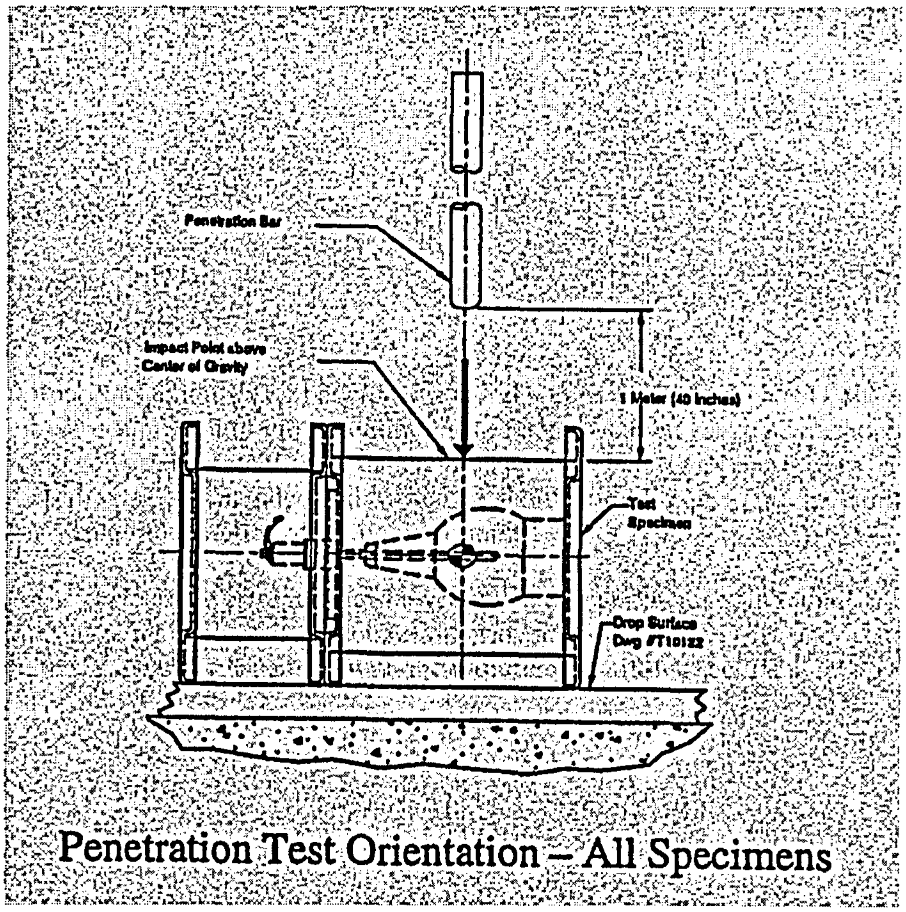


Figure 2.6e - Model 650L Penetration Test Orientation

2.7 Hypothetical Accident Conditions of Transport

(Reference:
- *USNRC, 10 CFR 71.73*
- *IAEA TS-R-1, paragraph 726)*

Sections 2.7.1 through 2.7.5 summarize evaluations and testing for the hypothetical accident conditions of transport tests. Section 2.7.6 summarizes the results of this testing.

Three (3) test specimens were used to conduct the hypothetical accident tests. Testing was performed after the test specimens had undergone the testing in Section 2.6 for Normal Conditions of transport. Detailed description of this testing is contained in Test Plan 80 Report (Section 2.12.2).

2.7.1 Free Drop

(Reference:
- *USNRC, 10 CFR 71.73(c)(1)*
- *IAEA TS-R-1, paragraph 727(a))*

Justification for all test unit drop orientations are included in Test Plan 80 Report (Section 2.12.2). All tests were conducted with the test specimen temperatures at or below -40°C (-40°F). The test specimens included standard locking assemblies.

2.7.1.1 End Drop

This orientation was used for Test Specimen TP80(B) and the orientation is shown in Figure 2.7a. The test specimen impacted flat on the top of the lid. One of the lid rivnuts cracked open, but the lid bolt remained in place. There was no other damage to the lid or lid bolts/rivnuts.

The top plate deflected up, resulting in source displacements of about 1/32 inch (0.8 mm) and 1/16 inch (1.6 mm). The carbon steel outer shell unzipped along the spot weld line and opened up about ½ inch (12 mm). The carbon steel inner shell fractured (brittle fracture) in the middle of the short side and opened up a crack about ½ inches (13 mm) wide and 3 inches (76 mm) long. The crack started at the top (under the top plate). At the end of this opening, the crack turned and continued behind the foam that fills the volume between the inner and outer shells. The foam cracked and several small pieces came out.

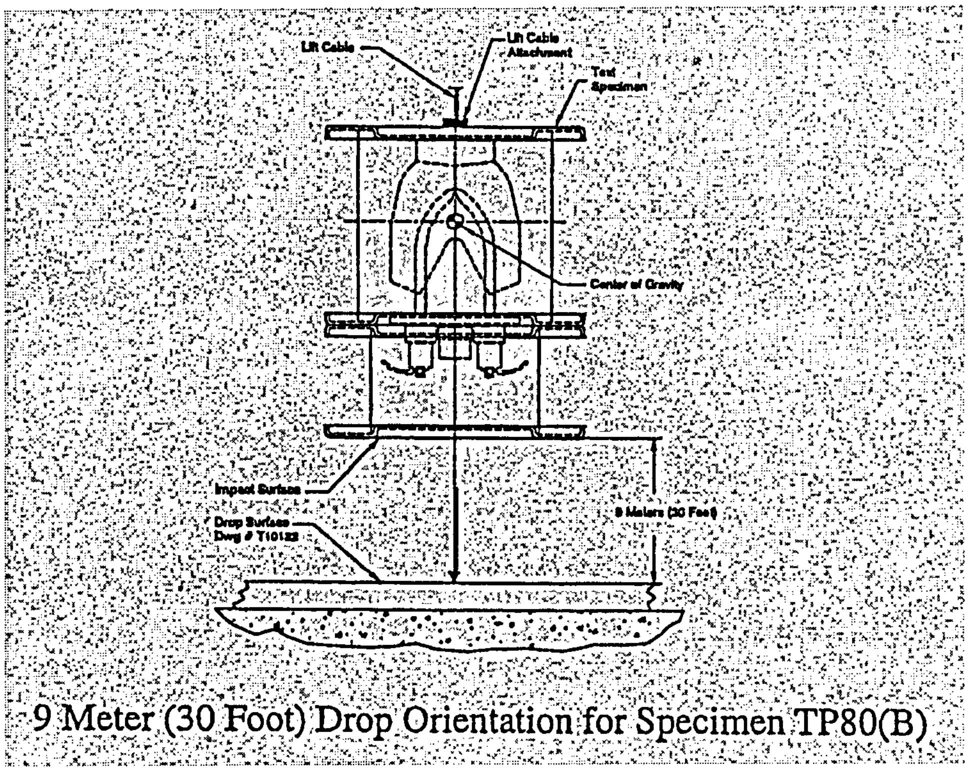


Figure 2.7a - Model 650L (TP80(B)) 9 m Drop Test Orientation – End Drop

2.7.1.2 Side Drop

This orientation was used for Test Specimen TP80(A) and the orientation is shown in Figure 2.7b. The test specimen rotated slightly so that the long edge of the bottom plate struck the ground first. The long edge of the bottom plate deformed, but no cracking was observed. The outer shell deformed at the interface with the long edge of the bottom plate. There were impact witness marks on the long edge of the top plate and the long edge of the bottom lid flange. There was a small deformation of the lid top flange.

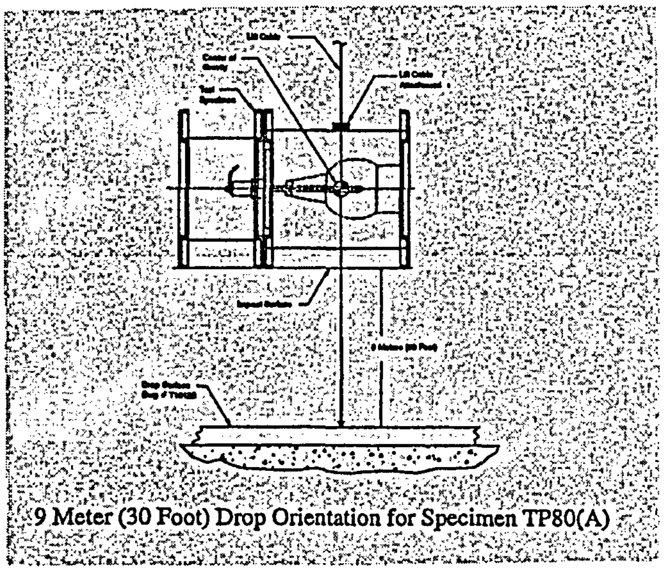


Figure 2.7b - Model 650L (TP80(A)) 9 m Drop Test Orientation – Side Drop

2.7.1.3 Corner Drop

This orientation was used for Test Specimen TP80(C) and the orientation is shown in Figure 2.7c. The specimen impacted on the top corner of the lid. The crack in the top flange of the lid, which initiated during the 1.2 meter (4 ft) drop test, increased and the top surface of the lid deflected inside the lid about ½ inch (13 mm) along one edge. The lock assemblies were not impacted during the drop. The column section of the lid and the bottom flange of the lid remained intact. There was no damage to the lid bolts or rivnuts.

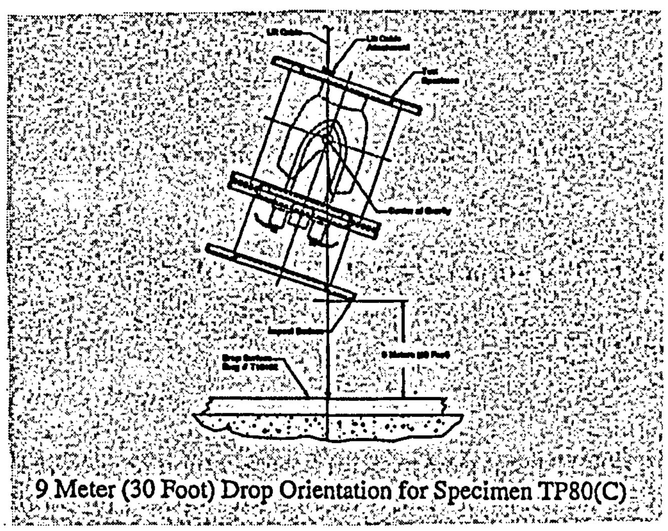


Figure 2.7c - Model 650L (TP80(C)) 9 m Drop Test Orientation – Corner Drop

2.7.1.4 Oblique Drops

The oblique drop was not performed. In an oblique drop, the energy generated at impact would be distributed across the initial and secondary impact surfaces (two upper and one lower flange). This will produce less force on impact at the initial impact location and the force from the secondary impact will cause deformation of the flanges without contributing to damage which could result in container failure.

Unlike the End, Side and Corner drops described in Sections 2.7.1.1 through 2.7.1.3, an oblique drop is less likely to cause a container failure by the mechanisms identified in Test Plan 80 Report (Section 2.12.2). These included displacement of the sources relative to the shield within the container shells and breach of the container shells sufficient to allow oxidation of the depleted uranium shield during the thermal test.

2.7.1.5 Summary of Results

(Reference:
- *USNRC, 10 CFR 71.73(a) and (b)*
- *IAEA TS-R-1, paragraph 726)*

See Table 2.7.8.1 for additional test unit results summary. In all cases, radiation profiles performed at the conclusion of all testing showed no significant increase in radiation levels for the test units and demonstrated that the 650L packages comply with the requirements of this section.

2.7.2 Crush

(Reference:
- *USNRC, 10 CFR 71.73(c)(2)*
- *IAEA TS-R-1, paragraph 727(c))*

Not applicable. This package is not used for the Type B transport of normal form radioactive material.

2.7.3 Puncture

(Reference:
- *USNRC, 10 CFR 71.73(c)(3)*
- *IAEA TS-R-1, paragraph 727(b))*

The puncture bar is a 6 inch diameter x 12 inch long, mild steel solid bar attached to a 12 inch x 12 inch x ½ inch thick mild steel base. The bar is attached to the base with a ¼ inch circumferential fillet weld. The puncture bar is attached to the drop test pad steel plate by four stainless steel bolts. Justification for all test unit puncture orientations are included in Test Plan Report 80 (Section 2.12.2) and results are summarized in the Sections 2.7.3.1 through 2.7.3.4. All tests were conducted with the test specimen temperatures at or below -40°C (-40°F). The test specimens included standard locking assemblies.

2.7.3.1 Test Specimen TP80(A)

Test Specimen TP80(A) impacted the puncture bar so that it impacted the horizontal long side of the package (see Figure 2.7d). There was a small dent on the long side of the outer shell just above the bottom plate and there were witness marks on the top plate.

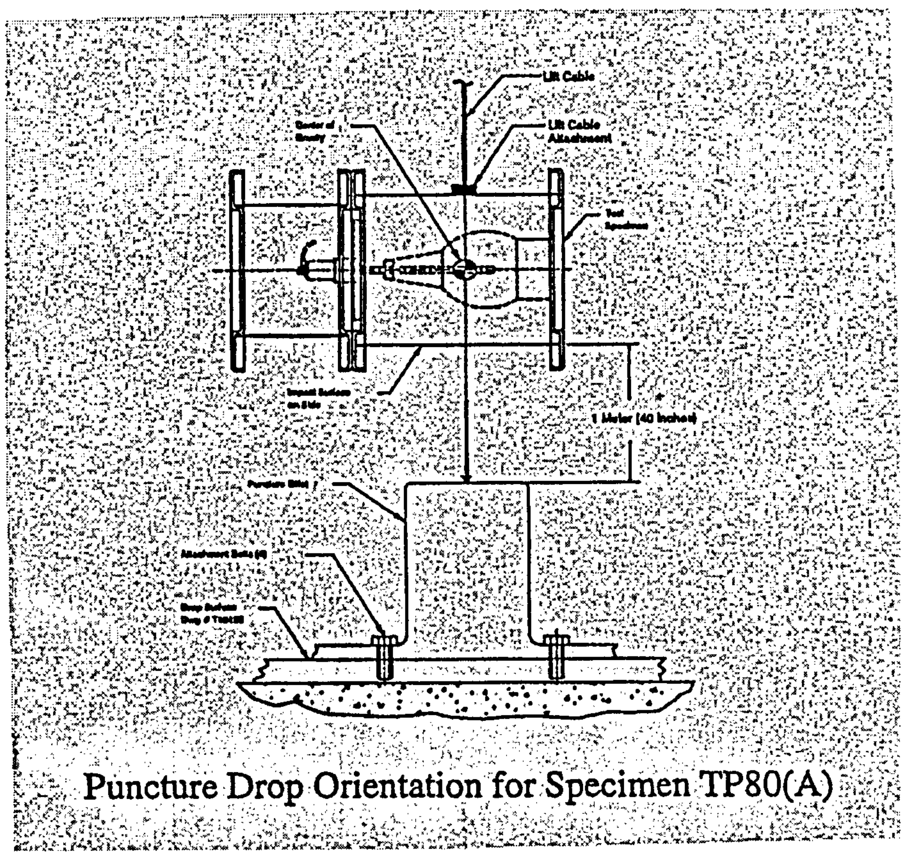


Figure 2.7d - Model 650L (TP80(A)) Puncture Test Orientation Horizontal Long-Side of the Package

2.7.3.2 Test Specimen TP80(B)

Test Specimen TP80(B) was dropped horizontally so that the edge of the puncture bar impacted the axial crack. This orientation was selected to increase the damage from the 9 meter (30 ft) drop test and to try to further open the axial crack.

The test specimen impacted directly on the crack. There were small indentations on both sides of the crack where the puncture bar impacted, but no further opening of the crack was observed.

2.7.3.3 Test Specimen TP80(C)

Test Specimen TP80(C) was dropped on the puncture bar in two orientations. The first orientation dropped the package so that the edge of the puncture bar impacted the crack in the lid. This orientation was selected to increase the damage from the 9 meter (30 ft) drop test with the specific intention of increasing the lid crack opening and trying to impact the lock assemblies.

On impact the edge of the puncture bar hit the top plate of the lid within the column of the lid increasing damage to the lid slightly. The lock assemblies were not impacted and remained protected by the lid. The lid bolts and rivnuts remained intact.

The second drop orientation for this package dropped the specimen so that the edge of the puncture bar impacted the underside of the top plate corner at a lid bolt rivnut. The intent of this orientation was to pry up the top plate and load the through bolts in tension. This orientation was also intended to damage the lid bolt connection. See Figure 2.7e.

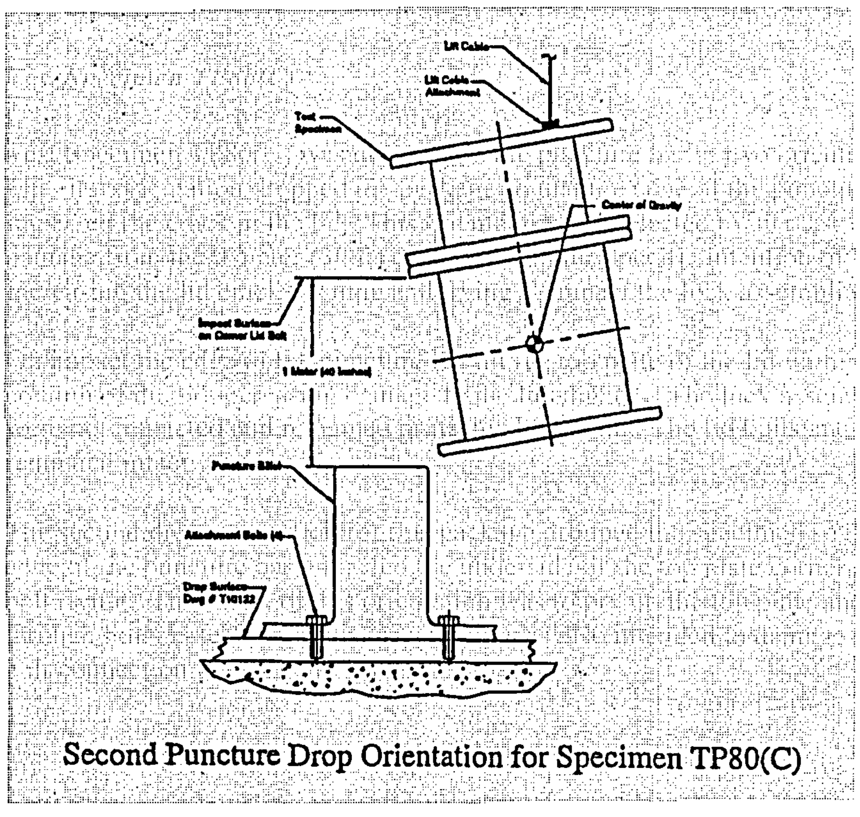


Figure 2.7e - Model 650L (TP80(C)) 2nd Puncture Test Orientation Underside of Top Endplate

In the drop, the specimen impacted on the under side of the top plate corner. There was a small deformation of the top plate edge at the impact point. The lid bolts/rivnuts were not damaged. No gaps were created at the top plate/shell interface and the through bolts remained secure.

2.7.3.5 Summary of Results

(Reference:

- *USNRC, 10 CFR 71.73(a) and (b)*
- *IAEA TS-R-1, paragraph 726)*

See Table 2.7.8.1 for additional test unit results summary. A more detailed summary is given in Test Plan 80 Report (Section 2.12.2). In all cases, radiation profiles performed at the conclusion of the puncture testing showed no significant increase in radiation levels for the test units and demonstrated that the 650L packages comply with the requirements of this section.

2.7.4 Thermal

(Reference:

- *USNRC, 10 CFR 71.73(c)(4)*
- *IAEA TS-R-1, paragraph 651 through 655, and 728)*

The thermal test was performed for test specimen TP80(B). The damage produced during the drop and puncture tests had allowed maximum source displacement in the shield as well as the potential for shield oxidation during the thermal test. The thermal test was not considered necessary for the other test specimens since their results would be less severe than TP80(B).

For TP80(B), lead shims placed under the depleted uranium shield were expected to melt during thermal testing. This would allow the shield to drop down towards the bottom plate and away from the top plate. Since the lock assemblies remained securely attached to the top plate this would allow the source assemblies to be raised above the fully shielded position in the shield.

The crack in the inner shell and the opening in the outer shell provided a path for air to reach the depleted uranium shield during thermal testing. Therefore oxidation of the shield was possible.

To obtain the largest possible displacement of the shield during thermal testing, the test specimen was placed on a jig to raise the side face of the unit to an angle (53° above horizontal) that positioned the center of gravity of the shield over the bottom plate inside edge (See Figure 2.7f). The side with the crack was facing down.

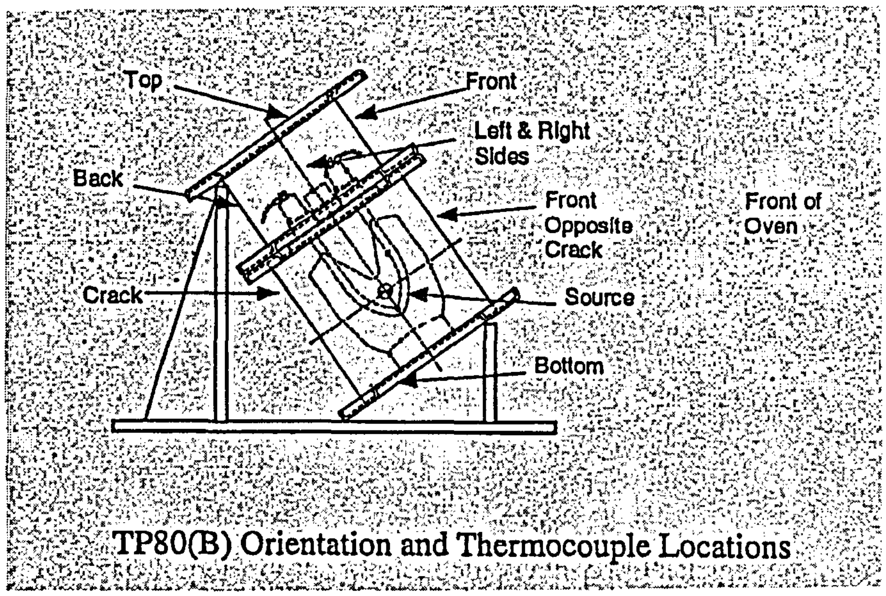


Figure 2.7f - Model 650L (TP80(B)) Thermal Test Orientation

During thermal testing, the test period of 30 minutes was started after all specimen thermocouples measured at least 810°C (1,490°F). To allow sufficient air for combustion of the specimen's polyurethane foam, the door of the oven was held open by 1 inch (25 mm) thick insulating strips placed on each side of the furnace door. This created a 1 inch (25 mm) wide by 36 inch (914 mm) long opening at the top and bottom of the oven door (total opening of 72 in^2 (465 mm^2)).

At the end of the 30 minute test interval, the specimen was removed from the furnace and allowed to self extinguish and cool down. Post-test visual inspections showed that the crack width did not change (but a cracked piece of the inner shell had dropped out of position). The polyurethane foam had burned off and some minor oxidation of the shield had occurred as evidenced by a small amount of depleted uranium oxide below the cracked shell.

Post-test radiographs showed that the shield had shifted down as expected. This resulted in some pullout of the source tubes from the top plate (less than 0.5 inches (13 mm)). The radiation profile of the device performed following the thermal test showed that the highest observed radiation level, 28 mR/hr at one meter was well below the allowable level of 1,000 mR/hr at one meter. Therefore, the 650L satisfies the thermal test requirements of 10 CFR 71.73(c)(4) and IAEA TS-R-1.

2.7.4.1 Summary of Pressures and Temperatures

(Reference:

- *IAEA TS-R-1, paragraph 502(d))*

Table 2.7a: Summary Table of Temperatures

Surface Temperature Condition	Model 650L Package
During Fire Test (Maximum Temperature)	996°C (1,825°F)
Post-Fire (Maximum Temperature)	884°C (1,623°F)

The Model 650L container is vented to atmosphere. As such, no pressure will build up in the package under Hypothetical Accident conditions.

Table 2.7b: Summary Table of Maximum Pressures

Package Configuration	Void Volume (in^3)	Fire Conditions 800°C (1,472°F) Pressure Developed
650L	0	0 psig

2.7.4.2 Differential Thermal Expansion

Physical testing has shown that any differential thermal expansion in the package has no detrimental effect on its ability to pass the thermal testing portion of the Hypothetical Accident Conditions. Design clearances between fitted components in the 650L are sufficient to allow for thermal expansion at the test temperature. It can be drawn from the actual testing results in Test Plan 80 Report (Section 2.12.2) that thermal expansion does not have a significant effect on the Model 650L package.

2.7.4.3 Stress Calculations

As was noted in Section 2.7.4.2, thermal differentials will have no detrimental effect on the interfaces between the steel shells, shield, endplates, lock assemblies or protective lid. The Model 650L transport package is open to the atmosphere and contains no components which could create a differential pressure relative to atmospheric conditions.

2.7.4.4 Comparison of Allowable Stresses

The Model 650L package was fully tested and passed under Normal and Hypothetical Accident Conditions of transport. It is therefore concluded that the Model 650L package will satisfy the performance requirements specified by the regulations.

2.7.5 Immersion - Fissile Material

(Reference:

- *USNRC, 10 CFR 71.73 (c)(5)*
- *IAEA TS-R-1, paragraphs 731-733)*

Not applicable. This package is not used for transport of Type B quantities of fissile material.

2.7.6 Immersion - All Packages

(Reference:

- *USNRC, 10 CFR 71.73 (c)(6)*
- *IAEA TS-R-1, paragraph 701 and 729)*

The Model 650L transport package is open to the atmosphere and contains no other components that would create a differential pressure under immersion. All materials are impervious to water and would not be affected.

The primary containment system in the model 650L package is a special form source, which meets the ISO 2919-1999 requirements for Class 3 pressure testing. Therefore the 650L could withstand the immersion test as Class 3 is in excess of the required 150 kPa (21.7 lb ft/in2).

2.7.7 Deep Water Immersion Test (for Type B Packages Containing More than 10^5 A_2)

(Reference:

- *USNRC, 10 CFR 71.61*
- *IAEA TS-R-1, paragraph 657, 658 and 730)*

Not applicable. This package does not transport normal form radioactive material in quantities exceeding $10^5 A_2$.

2.7.8 Summary of Damage

(Reference:

- *USNRC, 10 CFR 71.73(a) and (b)*
- *IAEA TS-R-1, paragraph 701, 702, 716 and 726)*

Table 2.7c summarizes the results of the Normal Conditions of Transport and Hypothetical Accident testing performed on the Model 650L transport packages.

Table 2.7c: Summary of Damages During Test Plan 80

Test Specimen	Test	Actual Impact Point	Damage Observed at Test Site
TP80(A) 80 lb (36.3 kg)	Compression	Weight applied to top of protective lid	No damage.
	Penetration Bar	Side of Container Shell (See Figure 2.6e)	Impact Mark. No other visible damage.
	4-foot free drop	Horizontal Long Side of Package	• Impact mark on edge of plates • Small Change in radiation profile (See Section 5 and Test Plan 80 Report Section 2.12.2)

Test Specimen	Test	Actual Impact Point	Damage Observed at Test Site
	30-foot free drop	Horizontal Long Side of Package	Bent bottom plate flange inward.
	Puncture drop	Horizontal Long Side of Package	Shallow dent on outer shell at impact point.
	Post Test Inspection	NA	• Protective Lid remained securely in place. • Locks were undamaged, sources secured. • No significant change in source positions. • Small change in radiation profile.
TP80(B) 83.6 lb (37.9 kg)	Compression	Weight applied to top of protective lid	No damage.
	Penetration Bar	Side of Container Shell (See Figure 2.6e)	Impact Mark. No other visible damage.
	4-foot free drop	Vertical Upside Down on Protective Lid	• Impact mark on top of protective lid. • Small change in radiation profile (See Section 5 and Test Plan 80 Report Section 2.12.2)
	30-foot free drop	Vertical Upside Down on Protective Lid	• Outer shell split open from top to bottom. • Inner shell cracked creating a 3 inch (76.2 mm) long by ½ inch (12.7 mm) wide opening. • Small upward deflection of top plate.
	Puncture drop	Crack in Shell Produced by 9 m drop	Bent shell inward slightly in area of crack.
	Post Drop Test Inspection	NA	• Protective Lid remained securely in place. • Locks were undamaged, sources secured. • Top plate deflection at center about 0.16 inch (4.1 mm). • No damage to through bolts. • No significant change in source position. • Outer and inner shells cracked; opening 3 inch (76.2 mm) long by ½ inch (12.7 mm) wide opening.
	Thermal	See Figure 2.6f	• Shield moved down as expected. • Polyurethane foam burned off exposing the shield. • Some oxidation of the shield near the shell crack occurred. • Shield self-extinguished after removal from oven. • Source pullout less than ½ inch (12.7 mm). • Maximum radiation level at one meter from the package was 28 mR/hr.
TP80(C) 89 lb (40.4 kg)	Compression	Weight applied to top of protective lid	No damage.

Test Specimen	Test	Actual Impact Point	Damage Observed at Test Site
	Penetration Bar	Side of Container Shell (See Figure 2.6e)	Impact Mark. No other visible damage.
	4-foot free drop	Top Corner Down on Protective Lid	• Bent corner of lid and cracked top plate of lid (brittle fracture). • Small change in radiation profile (See Section 5 and Test Plan 80 Report Section 2.12.2)
	30-foot free drop	Top Corner Down on Protective Lid	• Increased lid top plate crack length in vicinity of impact point. • Lock assemblies still protected by the lid.
	Puncture drop #1	Vertical Upside Down on Protective Lid	Broke inside of lid top plate (locks still protected.
	Puncture drop #2	Underside of Top Endplate (See Figure 2.7e)	Top plate deformed slightly.
	Post Test Inspection	NA	• Protective Lid remained securely in place. • Locks were undamaged, sources secured. • No significant change in source positions. • Small change in radiation profile.

Based on the results and assessments for the test specimens addressed in Test Plan 80 Report (see Section 2.12.2), it is concluded that the Model 650L transport packages maintain structural integrity and shielding effectiveness during Hypothetical Accident Conditions and Normal Conditions of Transport.

2.8 Accident Conditions for Air Transport of Plutonium

Not applicable. This package is not used for transport of plutonium.

2.9 Accident Conditions for Fissile Material Packages for Air Transport

Not Applicable. This package is not used for transport of Type B quantities of fissile material.

2.10 Special Form

(Reference:

- *USNRC, 10 CFR 71.75*
- *IAEA TS-R-1, paragraphs 602-604)*

The Model 650L transport package is designed for use with a special form source capsules which meets the ISO 2919-1999 requirements for Class 3 pressure testing. The source capsule must be special form and meet the ISO 2919-1999 Class 3 pressure criteria for transport in the Model 650L.

2.11 Fuel Rods

Not applicable. This package is not used for transport of fuel rods.

2.12 Appendix

2.12.1 Test Plan 80 Rev 1 (March 1999).

2.12.2 Test Plan 80 Report Minus Manufacturing Records (June 1999).

Section 2.12.1 Appendix: Test Plan 80 Rev 1 (March 1999)

Section 2.12.2 Appendix: **Test Plan 80 Report Minus Manufacturing Records (Jun 1999).**

Section 3 - THERMAL EVALUATION

3.1 Description of Thermal Design

(Reference:
- *USNRC, 10 CFR 71.33(a)(5)(v) and 71.33(b)(7)*
- *IAEA TS-R-1, paragraphs 651(b) and 655)*

The Model 650L transport package is a completely passive thermal device having no mechanical cooling system or relief valves. Cooling of the package is through free convection and radiation. There are no specific cooling or insulating design features. Pressure relief of the container is not necessary during the thermal test as the construction is not air tight and will allow venting to the atmosphere.

The maximum activity for this package is 240 Ci of Ir-192 or 300 Ci of Se-75. Accounting for self absorption in the source, this equals a maximum content activity of 552 Ci of Ir-192. The corresponding decay heat generation rate for the content activity is approximately 4.8 Watts (See Table 1.2a). The thermal evaluations are based on the decay energy of Ir-192 as this is greater than Se-75.

3.1.1 Design Features

The Model 650L transport package is described in Section 1. The containers use depleted uranium shielding. The depleted uranium is fully enclosed in the steel structure and endplates which are attached by screws. This construction prevents oxidation by severely limiting oxygen from reaching the depleted uranium shield.

3.1.2 Content's Decay Heat

From Table 1.2a, a maximum of 4.8 Watts of decay energy is available to be absorbed by the package.

3.1.3 Summary Tables of Temperatures

Table 3.1a: Summary Table of Temperatures

Surface Temperature Condition	Model 650L Packages	Comments
Insolation (38°C in full sun)	70°C (158°F)	Section 3.4.1.1
Decay Heating (38°C in shade)	46°C (115°F)	Section 3.4.1.2
Fire Test During	996°C (1,825°F)	
Post-Fire (Maximum Temperature)	884°C (1,623°F)	

3.1.4 Summary Tables of Maximum Pressures

All Model 650L containers are vented to atmosphere. As such, no pressure will build up in the units under either Normal or Hypothetical Accident conditions.

Table 3.1b: Summary Table of Maximum Pressures

Package Configuration	Void Volume IN^3	Normal Conditions 88°C (190°F) Pressure Developed	Fire Conditions 800°C (1,472°F) Pressure Developed	Comments
650L	0	0 psig	0 psig	

3.2 Material Properties and Component Specifications

3.2.1 Material Properties

Table 3.2a lists the relevant thermal properties of the important materials in the transport package. The sources referred to in the last column are listed below the table.

Table 3.2a: Thermal Properties of Principal Transport Package Materials

Material	Density (lb/in^3)	Melting/Combustion Temperature	Thermal Expansion	Source
Depleted Uranium	0.68	1,130°C (2,066°F)	8μin/in°F	Reference #1, p. 6-11 and Reference #2
Copper	0.32	1,082°C (1,980°F)	9.2μin/in°F	Reference #1, p. 6-7 and 6-11
Lead	0.41	327°C (620°F)	29.3μin/in°F	Reference #1, p. 6-7 and 6-11
Carbon Steel (nominal)	0.28	1,510°C (2,750°F)	6.3μin/in°F	Reference #1, p. 6-7 and 6-11
Stainless Steel-Type 304	0.29	1,427°C (2,600°F)	9.9μin/in°F	Reference #1, p. 6-11
Titanium	0.18	1,704°C (3,100°F)	5.2μin/in°F	Reference #1, p. 6-11
Polyurethan Foam	8 lb/ft^3	--	150μin/in°C	Reference #1, p. 6-199

Resource references:

1. Eugene A. Avallone and Theodore Baumeister III, *Mark's Standard Handbook for Mechanical Engineers, Tenth Edition,* New York: McGraw-Hill, 1996.

2. Lowenstein, Paul. *Industrial Uses of Depleted Uranium.* American Society for Metals. Metals Handbook, Volume 3, Ninth Edition.

3.2.2 Component Specifications

All components are specified and described on the drawings included in the Section 1.4.

3.3 General Considerations

3.3.1 Evaluation by Analysis

Evaluations by analysis are described in the section they apply to in this Safety Analysis Report or when applicable in the Test Plans contained in Section 2.12.

3.3.2 Evaluation by Test

Evaluations by direct testing are documented in the Test Plans contained in Section 2.12.

3.3.3 Margins of Safety

Margins of safety are discussed in each section as appropriate. All testing and analysis resulted in no loss of source containment or securement in the transport package. Though this demonstrates package compliance, it is difficult to quantify the margin related to these results. All physical testing used multiple specimens, with demonstrated results well within the regulatory requirements. Based on the results of the physical testing and the related analyses, we estimate the margin of safety for the Model 650L package as high.

3.4 Thermal Evaluation for Normal Conditions of Transport

3.4.1 Heat and Cold

3.4.1.1 Insolation and Decay Heat
(Reference:

- *USNRC, 10 CFR 71.71(c)(1)*
- *IAEA TS-R-1, paragraph 651)*

This analysis determines the maximum surface temperature produced by solar heating of the Model 650L transport package loaded at maximum activity with the contents that produce the highest energy input in accordance with 10 CFR 71.71(c)(1) and IAEA TS-R-1. This will be compared to the Normal Transport test conditions temperature range to determine which is the most onerous for thermal stress considerations.

The model consists of taking a steady state heat balance over the surface of the transport package. The following design analysis calculates the steady state surface temperature of a cylindrical package subjected to insolation and self-heat. The analysis is based on recognized heat transfer theory and specifically, that the total heat input due to the self-heat of the radioactive contents and the insolation energy absorbed must balance the heat loss due to convection and emitted radiation from the package surface. In order to assure conservatism, the following assumptions are made:

a. The transport package is assumed to undergo free convective heat transfer from the top and sides.

b. The inside package faces are considered perfectly insulated so there is no conduction into the package. The faces are considered to be sufficiently thin so that no temperature gradients exist in the faces.

c. The lid of the package is modeled as a rectangular solid, 10 inches (254 mm) long, 8 ¼ inches (210 mm) wide and 5 inches (127 mm) high. The outer shell of the package is modeled as a cylinder, 7 7/16 inches (189 mm) in diameter and 8 ¼ inches (210 mm) long.

d. The decay heat load (4.8 Watts) is added to the solar heat input load.

e. The emissivity coefficient of the steel package is assumed to be 0.8.

f. The absorptivity coefficient of the steel package is conservatively assumed to be 1.0.

The following heat calculations are based on the steady-state equilibrium relationship between the heat gained by the package and the heat lost.

Heat Input, Q_{IN} = Heat Output, Q_{OUT} in the steady-state.
Q_{IN} = Solar Heat Input + Decay Heat
Q_{OUT} = Heat loss by Radiation and Convection

The solar heat input is the combined solar heating of the top horizontal surface (flat), side vertical surface of the lid (flat) and the side vertical surface of the outer shell (curved). The insolation data, provided in 10 CFR 71.71(c)(1), is found in Table 3.4a.

Table 3.4a: Insolation Data

Surface	Insolation for a 12 hour period (g-cal/cm^2 or W/m^2)
Horizontal base	None
Other horizontal flat surfaces	800
Non-horizontal flat surfaces	200
Curved surfaces	400

Practically all solid materials are opaque to thermal radiation (even glass is only transparent to a fairly narrow range of wavelengths), and thermal radiation is in fact either reflected or absorbed within a very shallow depth of matter. Thus for solids it is possible to neglect transmissivity and write:

$$\text{reflectivity, } \rho + \text{absorptivity, } \alpha = 1 \text{ where } \rho = 0 \text{ and } \alpha = 1$$

i.e., the sum of the radiation reflected and absorbed by the material is equal to the total incident energy. Since the reflected energy does not contribute to the heat energy contained within the system, or package, it is not necessary to consider it in the analysis. However, the absorptivity of the material is the fraction of the total incident energy entering the system, which in this case is the heat input due to insolation.

Heat input due to insolation falling on top surface, Q_{IT}

$$Q_{IT} = 800 \text{ W/m}^2 \times 0.053 \text{ m}^2 = 42.4 \text{ W}$$

Heat input due to insolation on lid side surfaces (assumed rectangular box), Q_{ILS}

$$Q_{ILS} = 200 \text{ W/m}^2 \times 0.118 \text{ m}^2 = 23.6 \text{ W}$$

Heat input due to insolation on outer shell surface (assumed cylinder), Q_{IOS}

$$Q_{IOS} = 400 \text{ W/m}^2 \times 0.124 \text{ m}^2 = 49.6 \text{ W}$$

Decay Heat Input: $Q_{DT} = 4.8$ W

The total heat input is the sum of the solar heat input multiplied by the absorptive constant α for the material plus the decay heat input.

$$Q_{IN} = \alpha(Q_{IT} + Q_{ILS} + Q_{IOS}) + Q_{DT} = 120.4 \text{ W}$$

The total heat output is the sum of the radiation and convection heat transfer (Reference: Fundamentals of Heat and Mass Transfer, F.P. Incropera, 4th Edition, 1996, p. 9-10).

Radiation Heat Transfer: $Q_R = B\,E\,A_{TS}\,\{(T_W + 273)^4 - (T_A + 273)^4\}$

Where:

$B = 5.67 \times 10^{-8}$ W/m^2 K^4 (Stefan-Boltzmann Constant)
$E = 0.8$ (emissivity)
$A_{TS} = 0.295$ m^2 (surface area of the lid top, lid sides and outer shell)
T_W = The maximum surface temperature of the package
$T_A = 38°C$ (ambient temperature)

Therefore, $Q_R = 1.34 \times 10^{-8}\,\{(T_W + 273)^4 - 9.36 \times 10^9\}$

Lid top surface convection $= Q_{LT} = H_{LT}A_{LT}(T_W - T_A)$

Where $A_{LT} = 0.053\ m^2$ (lid top surface area) and H_{LT} = the free convection coefficient for a flat horizontal surface. For a heated plate facing up, the free convection coefficient for laminar flow is:

$$H_{LT} = 0.54[(g\ \beta\ (T_W - T_A)\ L^3\ ) / (\nu\ \alpha\ )]^{0.25}\ (K / L)$$

(Reference: Fundamentals of Heat and Mass Transfer, F.P. Incropera, 4th Edition, 1996, Ch. 9)

Where:

$g = 9.8\ m/s^2$
$\beta = 0.00303$ ($1/T_{avg}$ assuming $T_{avg} = 330$ K)
$L = 0.0572$ m (Area/Perimeter)
$\nu = 18.9 \times 10^{-6}\ m^2/s$
$\alpha = 26.9 \times 10^{-6}\ m^2/s$
$K = 28.52 \times 10^{-3}$ W/m K

Therefore, $H_{LT} = 2.75\ (T_W - 38)^{0.25}$

Substituting this into the convection equation for the lid top surface produces:

$$Q_{LT} = 0.146\ (T_W - 38)^{1.25}$$

Lid side surface convection = $Q_{LS} = H_{LS}A_{LS}(T_W - T_A)$

Where $A_{LS} = 0.118\ m^2$ (total surface area of lid sides) and H_{LS} = the free convection coefficient for a vertical surface. For a heated plate, the free convection coefficient for laminar flow is:

$$H_{LS} = [0.68 + 0.67[(g\ \beta\ (T_W - T_A)\ L^3\ ) / (\nu\ \alpha\ )]^{0.25}\ ] / \{1 + (0.492\ \alpha/\nu)^{9/16}\}^{4/9}]\ (K / L)$$

(Reference: Fundamentals of Heat and Mass Transfer, F.P. Incropera, 4th Edition, 1996, Ch. 9)

Where L = 0.056 m (area/perimeter). Therefore:

$$H_{LS} = 0.346 + 2.64\ (T_W - 38) + 0.312\ (T_W - 38)^{1.25}$$

Outer shell surface convection, $Q_{OS} = H_S A_S\ (T_W - T_A)$ where $A_{OS} = 0.124\ m^2$ (total surface area of the outer shell) and H_{OS} = the free convection coefficient for a vertical surface.

For a vertical plate the free convection coefficient for laminar flow is:

$$H_{OS} = [0.68 + 0.67[(g\ \beta\ (T_W - T_A)\ L^3\ ) / (\nu\ \alpha\ )]^{0.25}\ ] / \{1 + (0.492\ \alpha/\nu)^{9/16}\}^{4/9}]\ (K / L)$$

(Reference: Fundamentals of Heat and Mass Transfer, F.P. Incropera, 4th Edition, 1996, Ch. 9)

Where L = 0.077 m (area/perimeter). Therefore:

$$H_{OS} = 0.252 + 2.44\,(T_W - 38)^{0.25}$$

Solving for $Q_{OS} = 0.031\,(T_W - 38) + 0.303\,(T_W - 38)^{1.25}$

Total Heat Output: $Q_{OUT} = Q_R + Q_{LT} + Q_{LS} + Q_{OS}$

Total Heat Input: $Q_{IN} = \alpha(Q_{IT} + Q_{ILS} + Q_{IOS}) + Q_{DT} = 120.4$ W

Setting $Q_{OUT} = Q_{IN}$ and substituting produces:

$$120.4\ \text{W} = 1.34 \times 10^{-8}\{(T_W + 273)^4 - 9.36 \times 10^9\} + 0.761\,(T_W - 38)^{1.25} + 0.072\,(T_W - 38)$$

Iteration of this relationship yields a maximum wall temperature (T_W) of 70°C (158°F). This temperature will not adversely affect the package during normal transport since the melting temperatures of all safety critical components are well above this temperature. Additionally, charring of the polyurethane foam will not begin to occur at such low temperatures. Therefore the package satisfies the requirements of 10 CFR 71.71(c)(1) and IAEA TS-R-1, paragraph 651.

3.4.1.2 <u>Still Air (shaded) Decay Heating</u>
(Reference:
- *USNRC, 10 CFR 71.43(g)*
- *IAEA TS-R-1, paragraphs 617)*

This analysis calculates the maximum surface temperature of the Model 650L Transport package in the shade (i.e., no insolation effects), assuming an ambient temperature of 38° C (100°F), per 10 CFR 71.43(g).

The same assumptions from Section 3.4.1.1 are used. To assure conservatism, the following additional assumptions are made:

a. The entire decay heat (4.8 W) is deposited in the exterior surfaces of the package.

b. The interior of the package is perfectly insulated and heat transfer occurs only from the exterior surface to the environment.

c. 100% of the total heat is conservatively assumed to be deposited in sides of the package lid.

d. The only heat transfer mechanism is free convection.

Using these assumptions, the maximum wall temperature T_W is found using:

$$T_W = (q/hA) + T_A$$

Where:

$q = 4.8$ W (heat deposited per unit time on the package surface)
$h = 5$ W/m^2 K (free convection heat transfer coefficient for air)
(Reference Fundamentals of Heat and Mass Transfer, F.P. Incropera, 4th Edition, 1996)
$A = 0.118$ m^2 (surface area of the lid sides)
$T_A = 311$°K (ambient air temperature of 38°C)

Solving for T_W produces a maximum wall temperature (T_W) of 46°C (115°F), which is less than the maximum 50°C (122°F) allowed by 10 CFR 71.43(g).

3.4.1.3 Cold Effected Materials

The carbon steel components of the Model 650L are susceptible to brittle fracture at low temperatures. However, the package successfully met the Normal and Hypothetical accident transport requirements at temperatures below -40°C (-40°F) therefore the package complies with the requirements of this section.

3.4.2 Maximum Normal Operating Pressure

All 650L components are vented to the atmosphere. As such, pressure will not build up in the packages during Normal Transport conditions. Containers will exhibit a pressure differential of 0 psi as they are vented to the atmosphere with no means for creating a pressure differential. No other contributing gas sources are present.

3.4.3 Maximum Thermal Stresses

The temperature and pressure variations described in Sections 3.4.1 and 3.4.2 will not adversely affect the transport package during normal transport since the melting temperatures of all safety critical components are well above these temperatures and the package will experience no pressures to cause package failure. It its therefore concluded that the Model 650L transport packages will maintain their structural integrity and shielding effectiveness under the normal transport thermal stress conditions.

3.5 Thermal Evaluation Under Hypothetical Accident Conditions

3.5.1 Initial Conditions

The thermal test performed and described under Test Plan 80 and Test Plan 80 Report Sections 2.12.1 and 2.12.2 for detailed description of the test specimen initial conditions.

3.5.2 Fire Test Conditions

Sections 2.12.2 for detailed description of the fire test conditions.

3.5.3 Maximum Temperatures and Pressure

Sections 2.12.2 for detailed description of the temperature variations measured in the test specimen during the thermal test. Since the 650L is vented to the atmosphere, no pressures were generated in the package during the thermal test.

3.5.4 Maximum Thermal Stresses

Sections 2.12.2 for a description of the damage induced in the test specimen due to thermal stresses generated in the package during the thermal test.

3.5.5 Accident Conditions for Fissile Material Packages for Air Transport

Not Applicable. This package is not used for transport of Type B quantities of fissile material.

3.6 Appendix

Not Applicable.

Section 4 – CONTAINMENT

4.1 Description of the Containment System

(Reference:

- *USNRC, 10 CFR 71.33(a)(4)*
- *IAEA TS-R-1, paragraph 501(a), 501(b), 639 through 643 and 645)*

4.1.1 Containment Boundary

The containment system consists of the Model 650L transport packages and the radioactive source capsule(s). The source capsule(s) shall be qualified as Special Form radioactive material under 49 CFR 173 and IAEA TS-R-1.

4.1.2 Special Requirements for Plutonium

Not applicable. This package is not used for transport of Type B quantities of Plutonium.

4.2 General Considerations

4.2.1 Type A Fissile Packages

Not applicable. This package is not used for transport of Type A quantities of fissile material.

4.2.2 Type B Packages

(Reference:

- *USNRC, 10 CFR 71.51*
- *IAEA TS-R-1, paragraphs 646 & 656)*

As demonstrated in the Test Plan 80 Report (Section 2.12.2) and supported by assessments when applicable, performance of the normal conditions of transport testing caused no loss or dispersal of radioactive contents, no significant increase in surface radiation levels and no substantial reduction in the effectiveness of the package. The Model 650L packages therefore meets the requirements of this section.

4.3 Containment Under Normal Conditions of Transport (Type B Packages)

(Reference:

- *USNRC, 10 CFR 71.51(a)(1)*
- *IAEA TS-R-1, paragraphs 656(a))*

As demonstrated in the Test Plan 80 Report (Section 2.12.2) and supported by assessments when applicable, performance of the normal conditions of transport testing caused no breach of the source capsules contained in the package. Since the source capsules are the primary containment of the radioactive contents and no release from the source capsules occurred, the Model 650L packages meet the requirements of this section.

4.4 Containment Under Hypothetical Accident Conditions (Type B Packages)

(Reference:

- *USNRC, 10 CFR 71.51(a)(2)*
- *IAEA TS-R-1, paragraphs 656(b))*

As demonstrated in the Test Plan 80 Report (Section 2.12.2) and supported by assessments when applicable, performance of the hypothetical accident conditions of transport testing, the radiation level at one meter from the surface of the package did not exceed 1 R/hr. The Model 650L packages therefore meet the requirements of this section.

4.5 Leakage Rate Tests for Type B Packages

(Reference:

- *USNRC, 10 CFR 71.51*
- *IAEA TS-R-1, paragraphs 656(a))*

The primary containment for the radioactive material in the Model 650L transport package is the radioactive source capsules. All source capsules authorized for Type B transport in the Model 650L package are certified as special form radioactive material under 10 CFR Part 71, 49 CFR Part 173 and IAEA TS-R-1. After manufacture and again once every six months thereafter prior to transport, the source capsule is leak tested in accordance with ISO9978:1992(E) (or more recent editions) to ensure that containment of the source does not allow release of more than 0.005 μCi of radioactive material. These fabrication and periodic tests ensure that contamination release from the package does not exceed the regulatory limits.

Reference : ISO9978:1992(E) – Radiation Protection – Sealed Radioactive Sources – Leakage Test Methods.

4.6 Appendix

Not Applicable.

Section 5 - SHIELDING EVALUATION

5.1 Description of Shielding Design

(Reference:

- *USNRC, 10 CFR 71.31*
- *IAEA TS-R-1, paragraph 701 and 702)*

5.1.1 Design Features

The principal shielding in the Model 650L transport package is the depleted uranium shield assembly. In some cases additional supplemental lead shielding is added to the shield assembly as described in the drawings included in Section 1.4.

5.1.2 Summary Table of Maximum Radiation Levels

The tables in this Section include radiation profile data obtained from the 650L packages that were tested to the Normal and Hypothetical Accident Conditions of Transport under Test Plan 80 Report (see Section 2.12.2). The following notes apply to all tables in this section:

Note 1: Transport Index may not exceed 10. The Transport Index is equivalent to the 1 meter reading in mRem per hour (i.e., 5 mRem per hour at 1 meter = a Transport Index of 5.0).

Note 2: The maximum Transport Index based on the mrem per hour readings at one meter from the surface of this package was 0.8. All packages accepted and released for shipment under this Model designation will have a Transport Index less than or equal to 10.

Table 5.1a: Model 650L Test Unit TP80(A) After Normal Transport Testing
Summary Table of External Radiation Levels Extrapolated to Capacity of 240 Ci Ir-192
(Non-Exclusive Use)

Normal Conditions of Transport	Package Surface mSv per hour (mrem per hour)			1 Meter from Package Surface mSv per hour (mrem per hour)		
Radiation	Top	Side	Bottom	Top	Side	Bottom
Gamma	0.94 (94)	0.89 (89)	0.94 (94)	0.024 (2.4)	0.008 (0.8)	0.007 (0.7)
Neutron	NA	NA	NA	NA	NA	NA
Total	0.94 (94)	0.89 (89)	0.94 (94)	0.024 (2.4)	0.008 (0.8)	0.007 (0.7)
10 CFR 71.47(a) Limit	2 (200)	2 (200)	2 (200)	100 (10)	100 (10)	100 (10)

Table 5.1b: Model 650L Test Unit TP80(B) After Normal Transport Testing
Summary Table of External Radiation Levels Extrapolated to Capacity of 240 Ci Ir-192
(Non-Exclusive Use)

Normal Conditions of Transport	Package Surface mSv per hour (mrem per hour)			1 Meter from Package Surface mSv per hour (mrem per hour)		
Radiation	Top	Side	Bottom	Top	Side	Bottom
Gamma	0.71 (71)	0.83 (83)	0.83 (83)	0.002 (2.0)	0.008 (0.8)	0.007 (0.7)
Neutron	NA	NA	NA	NA	NA	NA
Total	0.71 (71)	0.83 (83)	0.83 (83)	0.002 (2.0)	0.008 (0.8)	0.007 (0.7)
10 CFR 71.47(a) Limit	2 (200)	2 (200)	2 (200)	100 (10)	100 (10)	100 (10)

Table 5.1c: Model 650L Test Unit TP80(C) After Normal Transport Testing
Summary Table of External Radiation Levels Extrapolated to Capacity of 240 Ci Ir-192
(Non-Exclusive Use)

Normal Conditions of Transport	Package Surface mSv per hour (mrem per hour)			1 Meter from Package Surface mSv per hour (mrem per hour)		
Radiation	Top	Side	Bottom	Top	Side	Bottom
Gamma	0.57 (57)	1.06 (106)	0.59 (59)	0.002 (2.0)	0.008 (0.8)	0.005 (0.5)
Neutron	NA	NA	NA	NA	NA	NA
Total	0.57 (57)	1.06 (106)	0.59 (59)	0.002 (2.0)	0.008 (0.8)	0.005 (0.5)
10 CFR 71.47(a) Limit	2 (200)	2 (200)	2 (200)	100 (10)	100 (10)	100 (10)

Table 5.1d: Model 650L Test Unit TP80(A) After Hypothetical Accident Transport Testing
(9 meter Drop Test and Puncture Bar Test)
Summary Table of External Radiation Levels Extrapolated to Capacity of 240 Ci Ir-192
(Non-Exclusive Use)

Hypothetical Accident Conditions of Transport	1 Meter from Package Surface mSv per hour (mrem per hour)		
Radiation	Top	Side	Bottom
Gamma	0.027 (2.7)	0.010 (1.0)	0.006 (0.6)
Neutron	NA	NA	NA
Total	0.027 (2.7)	0.010 (1.0)	0.006 (0.6)
10 CFR 71.47(a)(2) Limit	10 (1,000)	10 (1,000)	10 (1,000)

Table 5.1e: Model 650L Test Unit TP80(B) After Hypothetical Accident Transport Testing (9 meter Drop Test, Puncture Bar Test and Thermal Test) Summary Table of External Radiation Levels Extrapolated to Capacity of 240 Ci Ir-192 (Non-Exclusive Use)

Hypothetical Accident Conditions of Transport	1 Meter from Package Surface mSv per hour (mrem per hour)		
Radiation	Top	Side	Bottom
Gamma	0.28 (28)	0.079 (7.9)	0.011 (1.1)
Neutron	NA	NA	NA
Total	0.28 (28)	0.079 (7.9)	0.011 (1.1)
10 CFR 71.47(a)(2) Limit	10 (1,000)	10 (1,000)	10 (1,000)

Table 5.1f: Model 650L Test Unit TP80(C) After Hypothetical Accident Transport Testing (9 meter Drop Test and Puncture Bar Test) Summary Table of External Radiation Levels Extrapolated to Capacity of 240 Ci Ir-192 (Non-Exclusive Use)

Hypothetical Accident Conditions of Transport	1 Meter from Package Surface mSv per hour (mrem per hour)		
Radiation	Top	Side	Bottom
Gamma	0.022 (2.2)	0.01 (1.0)	0.005 (0.5)
Neutron	NA	NA	NA
Total	0.022 (2.2)	0.01 (1.0)	0.005 (0.5)
10 CFR 71.47(a)(2) Limit	10 (1,000)	10 (1,000)	10 (1,000)

Table 5.1g includes radiation profile data used to demonstrate that the Model 650L package configurations will meet the external radiation level requirements for non-exclusive use transport when loaded to capacity for Se-75.

Table 5.1g: Model 650L sn 274 Se-75 Profile Results Summary Table of External Radiation Levels Extrapolated to Capacity of 300 Ci Se-75 (Non-Exclusive Use)

	Package Surface mSv per hour (mrem per hour)			1 Meter from Package Surface mSv per hour (mrem per hour)		
Radiation	Top[1]	Side	Bottom	Top[1]	Side	Bottom
Gamma	0.19 (19)	0.12 (12)	0.12 (12)	0.005 (0.5)	0.005 (0.5)	0.002 (0.2)
Neutron	NA	NA	NA	NA	NA	NA
Total	0.19 (19)	0.12 (12)	0.12 (12)	0.005 (0.5)	0.005 (0.5)	0.002 (0.2)
10 CFR 71.47(a) Limit	2 (200)	2 (200)	2 (200)	0.1 (10)	0.1 (10)	0.1 (10)

[1]Profile results from the top of the 650L were taken without the protective cover installed on the package. Actual surface and one meter readings from the top of the package will be lower than noted in the table.

5.2 Source Specification

5.2.1 Gamma Source

(Reference:

- *USNRC, 10 CFR 71.33(b)(1) & (3))*
- *IAEA TS-R-1, Section IV & paragraph 807(a))*

The gamma sources allowed for transport in the Model 650L transport package are specified in Sections 1.2.3 and 2.10.

5.2.2 Neutron Source

Not Applicable. The Model 650L transport package is not used for the transportation of neutron emitting sources.

5.3 Shielding Model

5.3.1 Configuration of Source and Shielding

A shielding model was not used to justify acceptance of this package. Shielding justification was based on direct measurement.

5.3.2 Material Properties

Not Applicable. A shielding model was not used in the justification of this package. Shielding justification was based on direct measurement.

5.4 Shielding Evaluation

5.4.1 Methods

Shielding justification was based on direct measurement. All packages are profiled prior to final acceptance and shipment. This profile takes into account the maximum capacity of the package. Any package not meeting the required dose rates is rejected.

5.4.2 Input and Output Data

Radiation measurements included in this Section were adjusted to the maximum activity capacity for the package (e.g., activity correction factor). Activity correction factors (CF_A) were obtained by using the following relationship:

$$CF_A = \frac{Maximum Package Activity Capacity\ (A_C)}{Actual\ Profile Activity\ (A_P)}$$

$$For\ Example,\ if\ A_P = 235Ci\ and\ A_C = 240Ci,\ then$$

$$CF_A = \frac{240Ci}{235Ci} = 1.02$$

Therefore all original surface and 1 meter profile measurements would be multiplied by a factor of 1.02 for a package profiled using 235 Ci and a package capacity of 240 Ci.

Radiation measurements at the surface of the container will be adjusted to compensate for the off-set of the survey meter probe from the true surface of the package for newly manufactured packages.

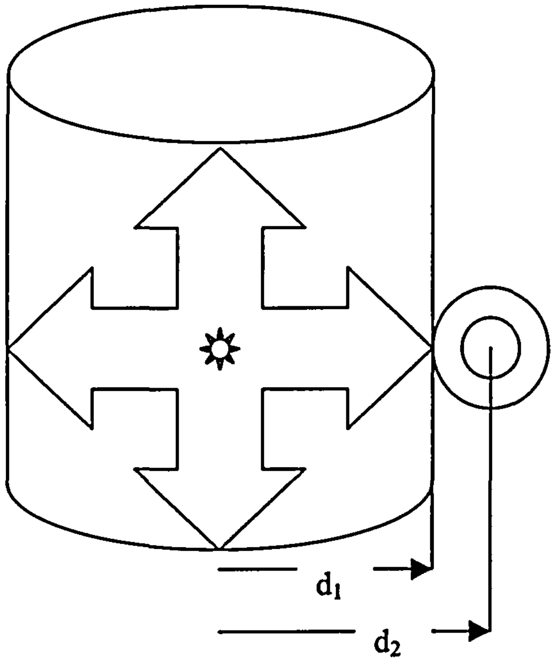
Note: The addition of correcting for the meter probe position relative to the surface of the package was an NRC requirement first implemented in the early 1980's when Amerhsam (QSA Global's predecessor) purchased Automation Industries. At that time NRC required the implementation of this correction factor for newly manufactured equipment, however, no backfit was required or performed for Type B packages manufactured prior to that date. The current population of 650L devices in use were manufactured prior to the date when surface correction factors were required. Any 650L devices manufactured under the current Type B transport approval would include surface correction factors (SCF) obtained by using the following relationship:

$$SCF = \sqrt{\frac{d_2^2}{d_1^2}}\ where\ d_1\ and\ d_2\ are\ \text{determined}\ as\ shown\ in\ Figure\ 5.a.$$

$$For\ Example,\ if\ d_1 = 9\,inches\ and\ d_2 = 9.5\,inches,\ then$$

$$SCF = \sqrt{\frac{(9.5\,inches)^2}{(9\,inches)^2}} = 1.06$$

Therefore in the example shown, all original surface profile measurements located along the side of the device shown in Figure 5.a would also be multiplied by a factor to account for surface correction of the detector to the device. Different SCF's would be calculated for the any dimension of the container where the minimum distance from the center of the activity to the center of the radiation probe is different.



d_1 =	distance from activity center to surface of container.
d_2 =	distance from activity center to surface of container plus radius of the survey meter probe.

FIGURE 5.a. SAMPLE SURFACE CORRECTION FACTOR DISTANCE CRITERIA

The radiation profile data showed no increase in radiation dose after testing beyond normal measurement variations. All test specimens met the regulatory requirements.

5.4.3 Flux-to-Dose-Rate Conversion

Not Applicable. Flux rates were not used to convert to dose rates in any shielding evaluations.

5.4.4 External Radiation Levels

Radiation surveys for the Model 650L showed maximum surface and 1 meter radiation levels from the transport packages within regulatory limits. Radiation surveys of the Model 650L transport packages after undergoing normal and accident condition transport testing were also well within the regulatory limits.

5.5 Appendix

Not Applicable.

Section 6 - CRITICALITY EVALUATION

All parts of this section are not applicable. The Model 650L transport package is not used for shipment of Type B quantities of fissile material.

Section 7 – Package Operations

Operation of the Model 650L transport package must be in accordance with the operating instructions supplied with the transport package, per 10 CFR 71.87 and 71.89. References to IAEA conform to the Type B(U)-96 criteria for packaging in accordance IAEA Regulations for the Safe Transport of Radioactive Material 1996 Edition (Revised) No. TS-R-1 (ST-1, Revised).

(Reference:

- *USNRC, 10 CFR 71.87 and 71.89*
- *IAEA TS-R-1, paragraph 501(a), 502(e) and 503)*

7.1 Package Loading

7.1.1 Preparation for Loading

The Model 650L packages must be loaded and closed in accordance with the following written procedures. Shipment of Type B quantities of radioactive material are authorized for sources specified in Section 7.1.1.1. Maintenance and inspection of the Model 650L packaging is in accordance with the requirements specified in Section 7.1.1.2.

7.1.1.1 Authorized Package Contents

(Reference:

- *USNRC, 10 CFR 71.87(a)*
- *IAEA TS-R-1, paragraph 502(f))*

Table 7.1a: Model 650L Package Information

Nuclide	Form	Maximum Capacity[1]	Maximum DU Weight	Maximum Weight
Ir-192	Special Form Sources	240 Ci	44 lbs (20 kg)	90 lbs (41 kg)
Se-75	Special Form Sources	300 Ci		

7.1.1.2 Packaging Maintenance and Inspection Prior to Loading

7.1.1.2.a Ensure all markings are legible and labels are securely fastened to the container.

7.1.1.2.b Inspect the container for signs of significant degradation. Ensure that the housing integrity is secure and does not have any significant dents, cracks of any type or rust.

[1] Maximum Capacity Activity for Ir-192 is defined as output Curies as required in ANSI N432 and 10 CFR 34.20 and in line with TS-R-1 and Rulemaking by the USNRC and the USDOT published in the Federal Register on 26 January 2004 .

7.1.1.2.c Ensure all bolts are present and secured. Assure safety wires are present and intact as noted on the drawings referenced in the Type B certificate.

7.1.1.2.d Assure the locking assemblies allow free movement of the lock slide when performing an operational test and that the plunger lock engages and is functional. Assure the shipping caps installs and secures over the source tube on the lock assemblies.

7.1.1.2.e Assure threaded holes used to secure the protective lid to the container body do not have damaged threads and engage the 3/8-16 x 7/8 inch long shipping cover bolts.

7.1.1.2.f If the container fails any of the inspections in steps 7.1.1.2.a-e, remove the container from use until it can be brought into compliance with the Type B certificate.

7.1.2 Loading of Contents

NOTE: ***These loading operations apply to "dry" loading only. The Model 650L package is NOT approved for wet loading.***

7.1.2.1 Prior to transportation, ensure the package and its contents meet the following requirements:

7.1.2.1.a The contents are authorized for use in the package.

7.1.2.1.b The package condition has been inspected in accordance with Section 7.1.1.2.

7.1.2.1.c Ensure that the source(s) are secured into place in the storage position(s) in accordance with the following requirements. Compliance with the following requirements ensures that the source(s) are securely locked in position before shipment.

1. Removal and installation of radioactive material contained within the shield container must be performed in a shielded cell/enclosure capable of holding the maximum isotope capacity of the container, or by using remote transfer operations for wire mounted sources. Container loading can only be performed by persons specifically authorized under an NRC or Agreement State license (or as otherwise authorized by an International Regulatory Authority). All necessary safety precautions and regulations must be observed to ensure safe transfer of the radioactive material.

2. Using remote handling techniques, load the source assembly so that it is fully inserted into the source tube with the inactive end of the source assembly protruding from the top of the source tube. Once loaded, push the lock slide to the "locked" position, depress the plunger lock and remove the key. Install the shipping cap over the source on the lock assembly. Repeat this step for the second source tube if transporting more than one source in the container.

7.1.3 Preparation for Transport

(Reference:

- *10 CFR 71.87*
- *IAEA TS-R-1, applicable paragraphs of Section V)*

7.1.3.1 Ensure that all conditions of the certificate of compliance are met.

7.1.3.2 Perform a contamination wipe of the outside surface of the package and ensure removable contamination does not exceed 0.0001 µCi when averaged over a wipe area of 300 cm^2.

Note: **If an overpack is used for shipment of the Model 650L, surveys and contamination wipes must first be performed on the external surface of the 650L and then on the surface of the overpack.**

7.1.3.3 Survey all exterior surfaces of the package to assure that the radiation level does not exceed 200 mR/hr at the surface. Measure the radiation level at one meter from all exterior surfaces to assure that the radiation level is less than 10 mR/hr.

7.1.3.4 Ship the container according to the procedure for transporting radioactive material as established in 49 CFR 171-178.

NOTE: The US Department of Transportation, in 49 CFR 173.22(c), requires each shipper of Type B quantities of radioactive material to provide prior notification to the consignee of the dates of shipment and expected arrival.

7.2 Package Unloading

7.2.1 Receipt of Package from Carrier

7.2.1.1 The consignee of a transport package of radioactive material must make arrangements to receive the transport package when it is delivered. If the transport package is to be picked up at the carrier's terminal, 10 CFR 20.1906 requires that this be done expeditiously upon notification of its arrival.

7.2.1.2 Upon receipt of a transport package of radioactive material:
(Reference:

- *IAEA TS-R-1, paragraph 510 and 511)*

7.2.1.2.a Survey the transport package with a survey meter as soon as possible, preferably at the time of pick-up and no more than three hours after it was received during normal working hours. Radiation levels should not exceed 200 mR/hr at the surface of the transport package, nor 10 mR/hr at a distance of 1 meter from the surface.

7.2.1.2.b Record the actual radiation levels on the receiving report.

7.2.1.2.c If the radiation levels exceed these limits, secure the container in a Restricted Area and notify the appropriate personnel in accordance with 10 CFR 20 or applicable Agreement State regulations.

7.2.1.2.d Inspect the outer container for physical damage or leaking. If the package is damaged or leaking or it is suspected that the package may have leaked or been damaged, restrict access to the package. As soon as possible, contact the Radiation Safety Office to perform a full assessment of the package condition and take necessary follow-up actions.

7.2.1.2.e Record the radioisotope, activity, model number, and serial number of the source and the transport package model number and serial number.

7.2.2 Removal of Contents

7.2.2.1 Unload the package must be in accordance with the instructions supplied with the package per 10 CFR 71.89.

7.2.2.2 Unloading of the package must also be in accordance with applicable licensing provisions for the user's facility related to radioactive material handling.

7.3 Preparation of Empty Package for Transport

(Reference:

- *IAEA TS-R-1, paragraph 520)*

In the following instructions, an *empty* transport package refers to a Model 650L transport package without an active source contained within the shielded container. To ship an empty transport package:

7.3.1. Perform the following procedure to confirm that there are no unauthorized sources within the container. Use only the gauge provided with the source changer. Do not use any other tool or gauge for another device.:

7.3.1.1. Remove the authorized source assembly from the package be in accordance with the instructions supplied with the package per 10 CFR 71.89.

7.3.1.2. After removing the source insert the depth gauge attached to the container into the empty tube(s) of the package. Read the gauge at the top of the outlet fitting.

7.3.1.3. The gauge should bottom out in the empty source tube(s) and indicate a safe condition. The red line should be flush with the top of the outlet fitting. Verify that each empty tube indicates a safe condition.

7.3.1.4. If the gauge indicates an unsafe condition (redline is above the outlet fitting) they may be an obstruction in the tube. Remove the gauge slowly while observing the survey meter. If the radiation levels increase as the gauge is being removed keep the gauge within the source tube, secure the container and contact QSA Global Inc. for further instructions.

7.3.1.5. If radiation levels remain normal as the gauge is being removed, completely remove the gauge, secure the container and contact QSA Global Inc. for further instructions.

7.3.2 Assure that the levels of removable radioactive contamination on the outside surface of the transport package does not exceed 4 Bq/cm^2 (when averaged over 300 cm^2).

7.3.3 When it is confirmed that the Model 650L transport package is empty, prepare the transport package for shipment. Survey the assembled package to ensure the external surface radiation level does not exceed 5 μSv/h.

7.3.4 Ship the container according to the procedure for transporting radioactive material as established in 49 CFR 171-178.

7.4 Other Operations

7.4.1 Package Transportation By Consignor

(Reference:

- *IAEA TS-R-1, paragraph 508, 512 through 514)*

Persons transporting the Model 650L transport package in their own conveyances should comply with the following:

7.4.1.1 For a conveyance and equipment used regularly for radioactive material transport, check to determine the level of contamination that may be present on these items. This contamination check is suggested if the package shows signs of damage upon receipt or during transport, or if a leak test on the special form source transported in the package exceeds the allowable limit of 185 Bq.

7.4.1.2 If contamination above 4 Bq/cm^2 (220 dpm/cm^2), when averaged over 300 cm^2, is detected on any part of a conveyance or equipment used regularly for radioactive material transport, or if a radiation level exceeding 5 µSv/h (0.5 mrem/hr) is detected on any conveyance or equipment surface, then remove the affected item from use until decontaminated or decayed to meets these limits.

7.4.2 Emergency Response

(Reference:

- *IAEA TS-R-1, paragraph 308 and 309)*

In the event of a transport emergency or accident involving this package, follow the guidance contained in "2000 Emergency Response Guidebook: A Guidebook for First Responders During the Initial Phase of a Dangerous Goods/Hazardous Materials Incident", or equivalent guidance documentation.

Reference: "2000 Emergency Response Guidebook: A Guidebook for First Responders During the Initial Phase of a Dangerous Goods/Hazardous Materials Incident"

7.5 Appendix

Not Applicable.

Section 8 - ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

8.1 Acceptance Test

8.1.1 Visual Inspections and Measurements

8.1.1.1 Visually inspect each transport package component to be shipped to assure the following:

8.1.1.1.a The transport package was assembled properly to the applicable drawing.

8.1.1.1.b All fasteners as required by the applicable drawings are properly installed and secured.

8.1.1.1.c The relevant labels are attached, contain the required information, and are marked in accordance with 10 CFR 20.1904, 10 CFR 40.13(c)(6)(i), 10 CFR 34, and 10 CFR 71 or equivalent Agreement State regulations.

8.1.1.2 Evaluate each Model 650L for shielding to ensure the transport dose rate requirements are met when the container is loaded to capacity.

8.1.1.3 Visual inspections and measurements will be performed in accordance with QSA Global Inc.'s USNRC approved Quality Assurance Program No. 0040.

8.1.2 Weld Examinations

Weld examinations will be performed in accordance with the applicable drawings requirements and in accordance with QSA Global Inc.'s USNRC approved Quality Assurance Program No. 0040.

8.1.3 Structural and Pressure Tests

(Reference:

- *10 CFR 71.85(a) and (b))*
- *IAEA TS-R-1, paragraph 501(a))*

Prior to first use as part of a 650L transport package, container structural conformance will be evaluated in accordance with the applicable drawings requirements and in accordance with QSA Global Inc.'s USNRC approved Quality Assurance Program No. 0040. The containment system is not designed to require increased or decreased operating pressures to maintain containment during transport, therefore pressure tests of package components prior to first use is not required.

When applicable, subject the swage coupling between the source capsule and the cable to a static tensile test with a load of 100 lbs (45 kgs). The source assembly will not be used if it fails this test.

8.1.4 Leakage Tests

The source capsules (primary containment) are wipe tested for leakage of radioactive contamination upon initial manufacture. The removable contamination must be less than 0.005 microcuries. The source capsules will also be subjected to leak tests under ISO9978:1992(E) (or more recent editions). The source capsules are not used if they fail any of these tests.

8.1.5 Component and Material Tests

Component and material compliance is achieved in accordance with the requirements in QSA Global Inc.'s USNRC approved Quality Assurance Program No. 0040.

8.1.6 Shielding Tests

The radiation levels at the surface of the transport package and at 1 meter from the surface are evaluated prior to first transport. These radiation levels, when extrapolated to the rated capacity of the transport package, must not exceed 200 mR/hr at the surface, nor 10 mR/hr at 1 meter from the surface of the transport package. Failure of this test will prevent use of the transport package as a Type B(U) package.

8.1.7 Thermal Tests

Not applicable. The source content of the Model 650L packages has minimal effect on the package surface temperature and therefore no additional testing is necessary to evaluate thermal properties of the packaging.

8.1.8 Miscellaneous Tests

Not applicable.

8.2 Maintenance Program

8.2.1 Structural and Pressure Tests

Not applicable. Material certification, or equivalent dedication process, is obtained for Safety Class A components used in the transport package prior to their initial use. Based on the construction of the design, no additional structural testing during the life of the package is necessary if the container shows no signs of defect when prepared for shipment in accordance with the requirements of Section 7 of the SAR. The 650L packaging system is not designed to require increased or decreased operating pressures to maintain containment during transport, therefore pressure tests of package components prior to individual shipment is not required.

8.2.2 Leakage Tests

As described in Section 8.1.4, "Leakage Tests," the radioactive source assembly is leak-tested at manufacture. In addition, the sources are leak tested in accordance with that Section at least once every six months thereafter if being transported to ensure that removable contamination is less than 0.005 microcuries. Also a contamination wipe is performed of the shield source tubes whenever the shield is returned to the manufacturer (typically the shield is shipped to a customer with new sources and may be returned directly to the manufacturer with decayed sources for disposition).

8.2.3 Component and Material Tests

The transport package is inspected for tightness of fasteners, proper seal wires, and general condition prior to each use as described in Section 7 and Section 8.1.1 of this SAR. No additional component or material testing is required prior to shipment.

8.2.4 Thermal Tests

Not applicable. The source content of the Model 650L packages has minimal effect on the package surface temperature and therefore no additional testing is necessary to evaluate thermal properties of the packaging prior to shipment.

8.2.5 Miscellaneous Tests

Prior to each use, a radiation survey of the package is made to assure that the radiation levels do not exceed 200 mR/hr at the surface, nor 10 mR/hr at one meter from the surface. Inspections and tests designed for secondary users of this transport package under the general license provisions of 10 CFR 71.17(b) are provided in Section 7.

8.3 Appendix

Not applicable.

Section 9 – IAEA TS-R-1 1996 Edition (Revised) Requirements not Otherwise Addressed – Section VI

9.1 General Package Design Requirements

9.1.1 *(Reference: IAEA TS-R-1, paragraph 609)*

As far as practicable, the packaging shall be so designed and finished that the external surfaces are free from protruding features and can be easily decontaminated.

The exterior surface of the 650L package is comprised of steel. The materials and fabrication of the package provides an external surface which is free from protruding features not necessary for use of the package and it can be easily decontaminated if necessary.

9.1.2 *(Reference: IAEA TS-R-1, paragraph 610)*

As far as practicable, the outer layer of the package shall be so designed as to prevent the collection and the retention of water.

The exterior surface of the 650L package is comprised of steel. The materials and fabrication of the package are water resistant and prevent, as far as practicable, the collection and retention of water.

9.1.3 *(Reference: IAEA TS-R-1, paragraph 611)*

Any features added to the package at the time of transport which are not part of the package shall not reduce its safety.

There are no added features to the package other than transport labels, markings, etc. These items are standard in package shipment and will not reduce the package safety due to their presence.

9.1.4 *(Reference: IAEA TS-R-1, paragraph 614)*

All valves through which the radioactive contents could otherwise escape shall be protected against unauthorized operation.

Not applicable. This package does not incorporate the use of valves.

9.1.5 *(Reference: IAEA TS-R-1, paragraph 616)*

For radioactive material having other dangerous properties the package design shall take into account those properties; see paras 109 and 507.

Not applicable. The contents of this package do not have any other dangerous properties other than its radioactivity.

9.2 Requirements for Type A Packages (required by TS-R-1 paragraph 650)

9.2.1 *(Reference: IAEA TS-R-1, paragraph 644)*

All valves, other than pressure relief valves, shall be provided with an enclosure to retain any leakage from the valve.

Not applicable. This package does not incorporate the use of valves.

9.2.2 *(Reference: IAEA TS-R-1, paragraph 647)*

The design of a package intended for liquid radioactive material shall make provision for ullage to accommodate variations in the temperature of the contents, dynamic effects and filling dynamics.

Not applicable. This package is not used for the transport of liquids.

9.3 Requirements for Type B(U) Packages

9.3.1 *(Reference: IAEA TS-R-1, paragraph 659)*

A package shall not include a pressure relief system from the containment system which would allow the release of radioactive material to the environment under the conditions of the tests specified in paras 719-724 and 726-729.

Not applicable. This package does not incorporate a pressure relief system.

9.4 Appendix

Not Applicable.