71-9269

QSA GLOBAL

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13 January 2015

Ms. Michele Sampson, Chief
Spent Fuel Licensing Branch
U.S. Nuclear Regulatory Commission
Office of Nuclear Material Safety and Safeguards
Division of Spent Fuel Management
Mailstop 3WFN-14A44
11555 Rockville Pike
One White Flint
Rockville, MD 20852

RE: 10 CFR 71.95(a)(3) report for CoC number USA/9269/B(U)-96 and Certificate Renewal Request

Dear Ms. Sampson:

QSA Global, Inc. is making a report under 10 CFR 71.95(a)(3) concerning the Model 650L Type B package (CoC 9269). By application dated December 8, 2014, we notified your office regarding issues identified for our Model 650L package where review of the production records identified areas where the package did not fully comply with all requirements under drawing R65006 Revision J. At that time we requested, and obtained, an interim amendment to the Certificate of Compliance addressing these issues until the referenced drawing could be revised to adequately address the discrepancies identified. This 71.95 notification supplements that information by including our root cause analysis and corrective actions intended to prevent recurrence.

Included with this letter we have submitted drawing R65006 Revision K to address the issues associated with Revision J of the drawing along with some other minor drawing changes. Acceptance of Revision K to drawing R65006 should enable removal of conditions 9, 10 and 11 currently listed on Revision 7 of Certificate of Compliance USA/9269/B(U)-96. All drawing changes are described on the attached drawing change table.

Revisions to this drawing prompted some corresponding changes in the SAR. Enclosed is Revision 9 to the SAR (minus appendices 2.12.1 & 2.12.2 which remain unchanged from Revision 8), a summary table which identifies changes from Revision 8 to Revision 9 of the SAR and the list of affected pages. As part of this submission action, we are requesting renewal of the Type B approval for this package design which expires on November 30, 2015.

NMSS01

10 CFR 71.95 Root Cause Analysis and Corrective Actions to Prevent Recurrence

The 650L package has been described over time under the CoC by drawing R65006 which was originally issued at Revision A on 23 June 1995 and most recently issued at Revision J issued on 3 August 2010. Over this 15 year time frame, the level of detail expected for Type B packages increased to require greater detail for:

- Material specifications for items important to package safety/integrity
- Welding specifications for assemblies important to package safety/integrity

The material specification issue was initially addressed at Revision E to drawing R65006 (November 1999) and further revised at Revision J (August 2010). The weld requirements were added at Revision J (August 2010). These revisions placed material requirements on package components without adequately addressing pre-existing material composition since an effective date for the material compliance was not specified when the material specification was added. This omission made compliant packages at Revision H, non-compliant for components that did not incorporate a phase in date and where components in use did not fully comply with the new material specifications.

In the case of the 650L drawing revision, staff involved in the review and approval of the drawing changes implemented at Revision J of drawing R65006, failed to adequately review production drawing requirements against the new material limitations placed on the descriptive assembly drawings. This oversight caused most of the discrepancies identified in our December 8, 2014 document.

From 2009-2010, seven of the Type B package design descriptive drawings were revised in a short time span to add greater material/welding specifications requirements. This affected the Model 976 Series, 660-OP, 865, 680-OP, 741-OP, 702 and 650L designs. The 650L was the last of these drawings to be revised.

Although a work instruction existed to describe the process for issuance/revision to descriptive assembly drawings, and this procedure addressed a review of drawing changes against production drawings, this procedure was determined to be unnecessary by the Engineering Manager and made obsolete on 23 September 2008. Subsequent failure to adequately define material specifications for some of the 650L components on Revision E of drawing R65006 resulted from a combination of:

- lack of guidance documentation
- human error
- a large volume of drawing revisions processed within a small time period
- drawing revisions processed by a small group of individuals in Engineering and Regulatory

Corrective Actions taken to prevent recurrence include:

a. Reactivate, with revisions, WI-E-1303 "Descriptive Drawings" or equivalent to address generation, processing and review of descriptive assembly drawings by the Engineering department. This document should incorporate guidance contained in NUREG/CR-5502 "Engineering Drawings for 10 CFR Part 71 Package Approvals" (1998) and format/detail information requests made over the last few years for QSA Global Type B submissions.

Additionally it needs to ensure that material specifications address new, current and historic construction that may be in use on an active Type B package assembly.

b. Revise WI-R3141 "Regulatory Processing of ERF's & Regulatory Holds" to add a section specific to review and approval of descriptive assembly drawing ERFs. This should include review of changes against current and, when applicable, historic production drawings to ensure adequate coverage for material construction and specifications on descriptive drawings.

Affected staff will be trained on documents generated under actions a. and b. to ensure awareness and prevent recurrence for future drawing revisions.

Further, a review of documentation for other QSA Type B containers in active use and not already identified under a corrective action for CR 1731 will be performed to ensure those containers are compliant to Type B requirements. This will include the 880 Series, Sentry Series, 865, 680-OP and 741-OP designs. The 360 Series packages are considered not affected as they were first introduced into production under a Type B approval in 2014 and have not been in fabrication long enough to be effected by production design changes not incorporated under the descriptive drawing.

The issues identified in this letter did not contribute to any incidents or package failures related to the safe use of the Model 650L in transport. The corrective actions taken are considered sufficient to prevent recurrence of the issues identified for the Model 650L, as well as other QSA Global, Inc. Type B package approvals. Continued compliance will be verified as part of our routine Quality Assurance internal audits which include performance of Type B container processing for production staff.

Should you have any additional questions, or wish to discuss this issue or our amendment request, please contact me.

Sincerely,

Lori Podolak
Manager,
Regulatory Affairs/Quality Assurance
Ph: (781) 505-8241
Fax: (781) 359-9191
Email: Lori.Podolak@qsa-global.com

RA/QA Approval	13 JAN 2015 Date
S. G. Engineering Approval	13 JAN 2015 Date

Enclosures: Drawing R65006 Revision K
Drawing Change Table R65006 Revision J to Revision K
List of Affected Pages
SAR Revision 9 (minus Appendices 2.12.1 2.12.2)
Change Table for SAR Revision 9

cc: ATTN: Document Control Desk
Director, Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
11555 Rockville Pike
One White Flint
Rockville, MD 20852

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SEE ERF 3189	S. G. / L. D.	13 Dec 14 / 12 Jan 15	K
DESCRIPTION	APPROVALS	DATE	LTR

REVISIONS

DIMENSIONS IN INCHES

TOLERANCES:
FRACTIONS ± 1/16
.X ± 0.1
.XX ± 0.01
.XXX ± 0.005

QSA GLOBAL
40 NORTH AVE, BURLINGTON, MA 01803

DESCRIPTIVE DRAWING

TITLE 650L SOURCE CHANGER

SIZE	DWG. NO.	REV
A	R65006	K

SCALE: NONE | SHEET 1 OF 5

Security-Related Information
Figure Withheld Under 10 CFR 2.390

DIMENSIONS IN INCHES TOLERANCES: FRACTIONS ± 1/16 .X ± 0.1 .XX ± 0.01 .XXX ± 0.005	QSA GLOBAL 40 NORTH AVE, BURLINGTON, MA 01803	SIZE A	DWG. NO. R65006	REV K
		SCALE: NONE	SHEET 2 OF 5	

Security-Related Information
Figure Withheld Under 10 CFR 2.390

DIMENSIONS IN INCHES TOLERANCES: FRACTIONS ± 1/16 .X ± 0.1 .XX ± 0.01 .XXX ± 0.005	QSA GLOBAL 40 NORTH AVE, BURLINGTON, MA 01803	SIZE A	DWG. NO. R65006	REV K
		SCALE: NONE	SHEET 3 OF 5	

Security-Related Information
Figure Withheld Under 10 CFR 2.390

DIMENSIONS IN INCHES TOLERANCES: FRACTIONS ± 1/16 .X ± 0.1 .XX ± 0.01 .XXX ± 0.005	QSA GLOBAL 40 NORTH AVE, BURLINGTON, MA 01803	SIZE A	DWG. NO. R65006	REV K
		SCALE: NONE	SHEET 4 OF 5	

Security-Related Information
Figure Withheld Under 10 CFR 2.390

DIMENSIONS IN INCHES TOLERANCES: FRACTIONS ± 1/16 .X ± 0.1 .XX ± 0.01 .XXX ± 0.005	QSA GLOBAL 40 NORTH AVE, BURLINGTON, MA 01803	SIZE A	DWG. NO. R65006	REV K
		SCALE: NONE	SHEET 5 OF 5	

Summary Table of Changes to Drawing R65006 Rev J to Rev K

Change Location	Summary Change	Change Reported Pursuant to 71.95 in Document Dated 12/8/14	Impact of Change on Units Previously or Currently in Use under the Certificate	Action Taken By QSA Regarding Affected Units
Sheet 1	Added optional external paint for shipping cover description.	No	None.	None. Not applicable.
Sheet 1	Added identification of attachment rivets for package label.	No	No change to package construction or design. Information added for completeness as method of attachment not previously identified.	None. Not applicable.
Sheet 1	Added optional external surface paint and/or zinc & yellow chromate plating for outer sleeve.	Yes	No change to package construction or design. Notation change added to address existing unit construction.	Certificate amended Revision 7. Package users advised of reporting requirement per 71.95.
Sheet 1	Increase finished assembly dimension tolerances for unit to more accurately reflect tolerance for the assembled dimensions. Height tolerance increased to ± 1 inch. Width and length tolerances increased to ±1/8".	No	None. No impact on units expected. These changes are made for flexibility in future for repaired units.	None. Not applicable.
Sheet 1	Added notation to specify the 1/16" gap that can exist between the outer sleeve and the top and bottom plates on the unit.	No	None. No impact on units expected. Change made for increased accuracy for construction of assembly.	None. Not applicable.
Sheet 1	Revised weld call out for lid weldment to allow six welds at top and bottom removing visual depiction of welds on drawing.	No	None. Revision is equivalent to existing specification but with slightly more flexibility in exact placement of welds on finished assembly.	None. Not applicable.

Summary Table of Changes to Drawing R65006 Rev J to Rev K

Change Location	Summary Change	Change Reported Pursuant to 71.95 in Document Dated 12/8/14	Impact of Change on Units Previously or Currently in Use under the Certificate	Action Taken By QSA Regarding Affected Units
Sheet 1	Added option for inner shell and outer sleeve to be stainless steel or the currently approved carbon steel configuration on sheet 1. Removed material reference to outer sleeve on sheet 2.	No	None. Change made to allow for flexibility and increased durability for future serviced units. Removal of material reference on sheet 2 eliminated redundant specification.	None. Not applicable.
Sheet 2	Added reference to titanium reinforcement sleeves on source tubes.	Yes	No change to package construction or design. Notation had previously appeared on drawing R65006 at Revision C but was inadvertently omitted when the drawing at Revision D.	Certificate amended Revision 7. Package users advised of reporting requirement per 71.95.
Sheet 2	Material description for outer sleeve deleted.	No	No change to package construction or design. Material is specified on sheets 1 and 5. Material identification on sheet 2 was duplicative and unnecessary.	None. Not applicable.
Sheet 2	Revised wording for Through Bolt description to clarify the 300 series stainless steel requirements apply to the through bolts and not the lockwasher.	No	Clarification only, no change to design.	None. Not applicable.
Sheet 2	Removed weld nut material strength table and replaced with note reference on sheet 5 to weld nut material to be austenitic grade stainless steel listed in ASTM A493.	No	No change to package construction or design. Material description change made for simplicity in purchasing.	None. Not applicable.
Sheet 2	The material legend was updated to remove information specified in another location on this drawing (e.g. sheets 1 and 5).	No	This change is administrative only, there has been no change to the package construction or design. The removed information was duplicative and unnecessary.	None. Not applicable.

Summary Table of Changes to Drawing R65006 Rev J to Rev K

Change Location	Summary Change	Change Reported Pursuant to 71.95 in Document Dated 12/8/14	Impact of Change on Units Previously or Currently in Use under the Certificate	Action Taken By QSA Regarding Affected Units
Sheet 2	Revised weld specification for weld nut from optional tack weld to specify spot weld (3 times).	No	None. New specification more accurately describes fabrication of this assembly as all weld nuts are welded in this manner.	None. Not applicable.
Sheet 3	Added description and material composition for the plunger assembly used on the lock assembly.	No	No change to package construction or design. Information added for component with limited importance to safety. This component is considered important to safety as it secures the shipping cap/quick connect fitting (a secondary protective device) to the lock assembly during transport.	None. Not applicable.
Sheet 3	Note under Section C-C regarding lock component material construction deleted and information added for applicable components for lock to identify the material as 300 Series stainless steel. This affects the lock screw, slide screw, shipping cap with spring and plunger and quick connect fitting.	No	No change to package construction or design. Change made for clarity only.	None. Not applicable.
Sheet 3	Added identification call out for the guide spring in the view showing the lock assembly open.	No	No change to package construction or design. Identification made for completeness as component was shown but not identified on drawing in the lock assembly open view.	None. Not applicable.
Sheet 3	Added identification of the spring which is part of the shipping cap assembly. Added option for shipping cap, including spring and plunger to be stainless steel.	No	No change to package construction or design. Detail added for completeness and increased flexibility only.	None. Not applicable.

Summary Table of Changes to Drawing R65006 Rev J to Rev K

Change Location	Summary Change	Change Reported Pursuant to 71.95 in Document Dated 12/8/14	Impact of Change on Units Previously or Currently in Use under the Certificate	Action Taken By QSA Regarding Affected Units
Sheet 3	Added identification of lock screw for plunger lock assembly.	No	No change to package construction or design. Detail added for clarity only.	None. Not applicable.
Sheet 3	Added applicability notation to lock screw material strength table for components as of 1 November 2010.	Yes	No change to package construction or design. The applicability date should have been incorporated at Revision J of the drawing when the table was added since components prior to that date were obtained as 18-8 stainless steel which may not have complied with the strength table requirements. The values given in the material strength table have essentially the same mechanical properties as 18-8 and other 300 Series stainless steels used to produce tamperproof button head screws. Screws to 18-8 were used on test units that demonstrated compliance to the transport requirements.	Certificate amended Revision 7. Package users advised of reporting requirement per 71.95.
Sheet 5	Revised note 2 to add additional, applicable weld code standards AWS D1.1, and D9.1. Removed code revision references.	No	This change has no significant impact on the performance of the package and more accurately specifies the welding requirements for all sub-assemblies on the package.	None. Not applicable.
Sheet 5	Note 4 revised to remove the revision reference to code A276.	No	Reference to a specific revision of this code is unnecessarily restrictive and it has been removed.	None. Not applicable.
Sheet 5	Revised note 1 to add material option of 304/304L stainless steel in addition to SAE 30304 per AMS5513	Yes	This change has no significant impact on the performance of the package. Test units used to demonstrate compliance to transport requirements were compliant to 304 stainless steel. Materially, 304, 304L and SAE 30304 per AMS 5513	Certificate amended Revision 7. Package users advised of reporting requirement per 71.95.

Summary Table of Changes to Drawing R65006 Rev J to Rev K

Change Location	Summary Change	Change Reported Pursuant to 71.95 in Document Dated 12/8/14	Impact of Change on Units Previously or Currently in Use under the Certificate	Action Taken By QSA Regarding Affected Units
			materials are all corrosion resistant stainless steels with essentially equivalent material strength and ductility. As such, the components are expected to perform identically to components tested for compliance to the normal and accident condition transport requirements.	
Sheet 5	Revised note 3 to add material option of 304 stainless steel in addition to ASTM F593 Group 1, Condition A. Also added applicability date of 18 Jan 2011 for this requirement.	Yes	Test units used to demonstrate compliance to transport requirements were compliant to 300 Series stainless steel. The material used on units prior to 18 Jan 2011 complied to 300 Series stainless steel but not confirmed compliant to ASTM F593 Group 1, Condition A. The ASTM F593 Group 1, Condition A materials have essentially the same mechanical and corrosion resistance properties as Type 304 and other 300 Series stainless steels. As such, 304 and ASTM F593 materials are expected to perform equivalently under the normal and accident condition transport requirements.	Certificate amended Revision 7. Package users advised of reporting requirement per 71.95.
Sheet 5	Deleted previous note 5.	No	Revisions to the drawing over time has made the need for this note no longer applicable. Exceptions to material requirements in the bills of material and notes attached to the drawing views for components are identified in notes listed on sheet 5. This note is not necessary.	None. Not applicable.
Sheet 5	Note 7 revised to add reference to ASTM A666 stainless steel in addition to the referenced carbon steel standard.	No	Option added to specify outer sleeve material if made from stainless steel.	None. Not applicable.

Summary Table of Changes to Drawing R65006 Rev J to Rev K

Change Location	Summary Change	Change Reported Pursuant to 71.95 in Document Dated 12/8/14	Impact of Change on Units Previously or Currently in Use under the Certificate	Action Taken By QSA Regarding Affected Units
Sheet 5	New note 8 added to specify the weld nut material as austenitic grade stainless steel listed in ASTM A493.	No	No change to package construction or design. Material description change made for simplicity in purchasing.	None. Not applicable.

Safety Analysis Report for the Model 650L Transport Package

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.
Revision 6, 14 July 2006	Changes to pages 1-1, 5-1 and 5-2.
Revision 7, April 2010	SAR updated to reflect compliance to NUREG 1886 (Final Report – March 2009).
Revision 8, August 2010	Correction of typographical errors on page 3-6 of the SAR.
Revision 9, January 2015	Changes to incorporate changes to R65006 and other minor content updates.

Revision 9 to the 650L SAR addresses issues covered in R65006 Revision K as well as other minor updates to the SAR content. These changes are listed under the SAR Section in Revision 9 where the change occurs.

Section Location	Summary Change	Comment
1.2.1.1, 2.2.2, 2.2.3 & Table 3.2.A	Added reference to tungsten.	This material option was included under drawing R65006 Revision J but had not been updated in the SAR descriptive section.
1.2.1.2	Added option for inner and outer shells to be either carbon steel or stainless steel.	Use of stainless steel will increase component performance as well as corrosion resistance with no detrimental impact on package conformance.
1.2.1.3	Added reference to optional steel washers.	This component option was included under drawing R65006 Revision J but had not been updated in the SAR descriptive section.
2.2.1 Table 2.2.A	Updated Table entries to reflect materials referenced on drawing R65006 and remove reference standards no in current use on drawing R65006.	Accuracy.
2.6.1	Revised heat output from 4.75 Watts to 4.8 Watts.	Heat output of 4.8 Watts is used in all other locations in SAR. Change made for consistency.
2.10	Revised section to reference capsule compliance to ISO Class 6 temperature requirement.	Incorporated based on assessments made in section 2.7.4.3 regarding compliance to stress calculations.
2.12.3 & 2.12.4	Referenced Revision 10 of special form certificate.	Change made based on special form certificate revision.
5.4.1 & 5.4.2	Revised section to remove details related to performance of future profiles.	No new manufacture is currently authorized under this approval and such information is no longer applicable.
7.4.2	Updated revision year for Emergency Response Guidebook.	Accuracy.

Safety Analysis Report

QSA Global, Inc.

Model 650L

Type B(U) - 96

Transport Package

January 2015

Revision 9

Contents

List of Tables

List of Figures

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 with 10 CFR 71, 49 CFR 173, IAEA Regulations for the Safe Transport of Radioactive Material No. TS-R-1 (ST-1, Revised) 1996 Edition (Revised) and Canadian Nuclear Safety Commission (CNSC) PTNS Regulations SOR/2000-208. This submission is formatted in accordance with NUREG-1886 "Joint Canada – United States Guide for Approval of Type B(U) and Fissile Material Transportation Packages" dated March 2009.

1.2 Package Description

The Model 650L package consists of a welded carbon steel shell encasing a uranium shield which houses a titanium "U" tube. The tube is crimped in the middle of the "U" to provide a positive stop for the source assembly. The package has two source locking assemblies mounted on the top cover plate that are used to secure the radioactive source in a shielded position during transport. The Model 650L package is constructed in accordance with descriptive drawing R65006 (see Appendix 1.3).The package measures approximately 13 ¼ inches (337 mm) tall by 10 inches (254 mm) wide by 8 ¼ inches (210 mm) deep. The maximum weight of the package is 90 pounds. The general package information is shown in Table 1.2.A:

Table 1.2.A: Model 650L Package Information

Package ID	Nuclide	Form	Maximum Capacity[1]	Chemical / Physical Form	Maximum Content Weight[4]	Maximum Decay Heat[3]	Maximum DU Weight	Maximum Package Weight
650L	Ir-192	Special Form[2] Sources	240 Ci	Metal	36 grams	4.8 Watts	44 lbs (20 kg)	90 lbs (41 kg)
	Se-75	Special Form[2] Sources	300 Ci	Metal-Selenide Compound	36 grams	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.
[4]Maximum content weight includes the mass of the radioactive material and the source capsule handling wire assembly for a shipment containing two source wire assemblies.

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

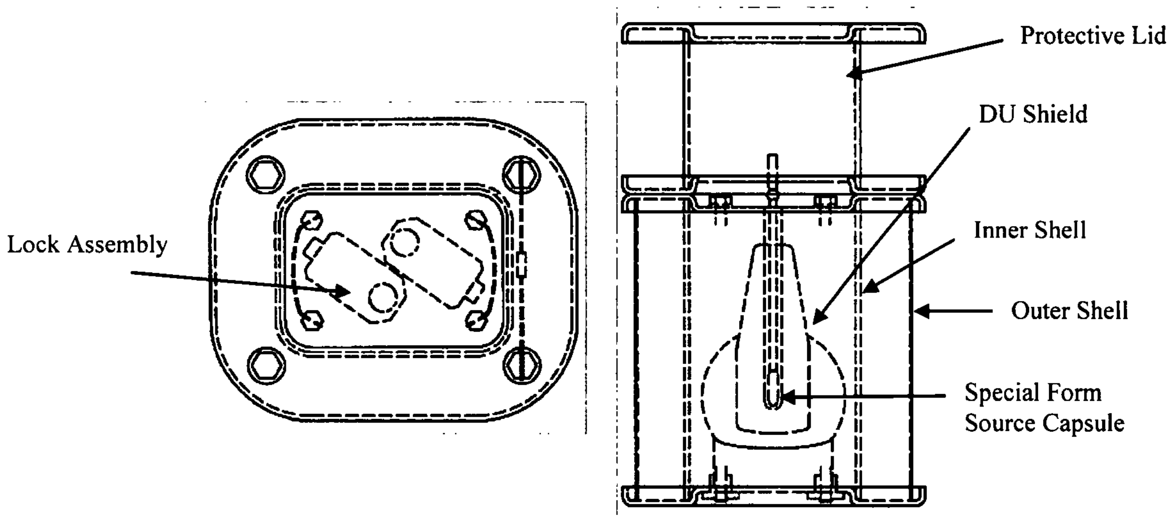


Figure 1.2.A – 650L Assembly

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 or tungsten positioned at various locations around the depleted uranium shield. The lead/tungsten 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 (or stainless 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 with optional steel washers. 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 Contents

The Model 650L transport package is designed to transport Ir-192, or Se-75, as special form capsules attached to source wire assemblies. Actual content to output activity for Ir-192 varies based on the capsule configuration as well as variations in isotope self-absorption. A factor of 2.3 was used to convert output activity to content activity for Ir-192, 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 maximum decay heat for Ir-192 is 4.8 watts, and 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. 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 package contents is 0.08 lbs (36 grams). The content weight value is based on shipment of two source assemblies. The radioactive material contained within the special form capsule along with the weight of the source wire assembly components is the value used as the maximum weight of the package contents. The radioactive material weight value is 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 1.2.A are the maximum content masses.

1.2.3 Special Requirements for Plutonium

Not applicable. This package is not used for the transportation of plutonium.

1.2.4 Operational Features

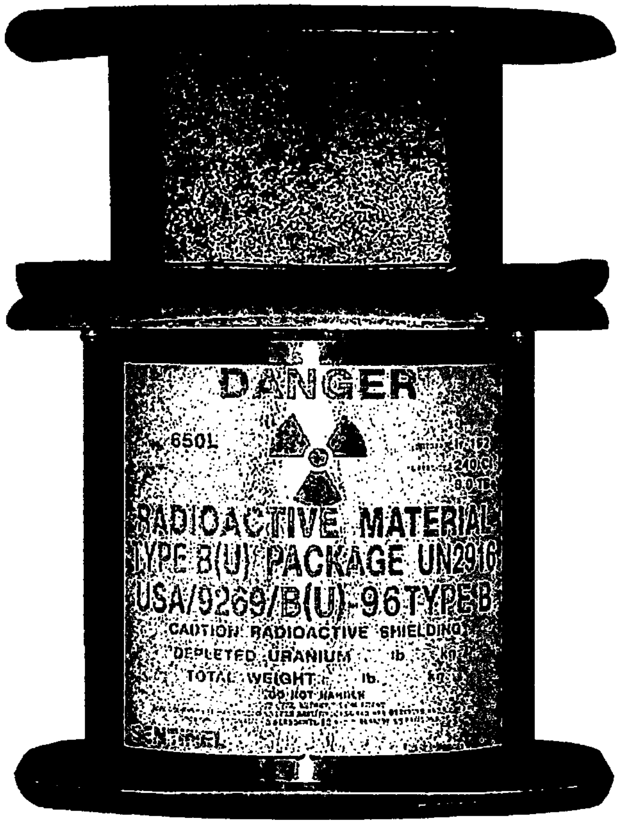
This package does not involve complex containment systems for source securement. The sources for this package are all special form, welded capsules. The capsules are attached to flexible source wire assemblies and held securely in the device by components of the lock assembly attached to the top plate. 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 Appendix

Figure 1.3.A. shows a sketch of the Model 650L package as prepared for transport. Additional drawings of the Model 650L transport package are also enclosed in this appendix .

Figure 1.3.A – Sketch of Model 650L Prepared for Transport



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 and TS-R-1.

2.1 Description of Structural Design

2.1.1 Discussion

The Model 650L transport package is described in Section 1.2, "Package Description".

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, IAEA TS-R-1 (ST-1, Revised) 1996 Edition (Revised) and CNSC PTNS SOR/2000-208. All design criteria are evaluated by a straightforward application of the appropriate section of these requirements.

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

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.

All component 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 the standards referenced on the drawings in Appendix 1.3. All hardware meets the standards referenced on the drawings in Appendix 1.3. 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.

2.2 Materials

2.2.1 Material Properties and Specifications

Table 2.2.A 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 Appendix1.3. The reference for the table information is listed in the last column of the table.

Table 2.2.A: Mechanical Properties of Principal Transport Package Materials

Material	Tensile Strength	Yield Strength	Elongation	Source
Depleted Uranium	65 kpsi	30 kpsi	12%	Reference #1
Copper	25 kpsi	9 kpsi	25%	Reference #2, page 224
Titanium	145 kpsi	134 kpsi	10%	Reference 3 page 98
Lead	1.8 kpsi	8 kpsi	30%	Reference 2 page 550
Stainless Steel (304)	75 kpsi	30 kpsi	40%	SAE 30304 per AMS5513 or ASTM A276 Condition A Hot or Cold Finished
Austenitic Stainless Steel	80 kpsi	Not specified	Not specified	ASTM A493
Cold Rolled Steel Sheet	Not Specified	20-40 ksi	30% Min	ASTM A1008/A1008M-09[1]
Hot Rolled Steel Sheet	Not Specified	30-50 ksi	25% Min	ASTM A1011/A1011M-09[1]
Strain-Hardened Stainless Steel Bolts	125 kpsi	100 kpsi	12%	ASTM A193/A193M
Stainless steel bolts	70 kpsi	30 kpsi	30%	ASTM F593 Group 1, Condition A

[1]Mechanical properties for the referenced materials listed in this standard are considered "Typical Ranges". These materials, as used in this transport package, do not provide structural support to the package containment under normal or hypothetical accident test conditions. The use of typical material properties ranges for these materials is sufficient to meet the intended performance requirements for the package.

Resource references:

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

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

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

2.2.2 Chemical, Galvanic or Other Reactions

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, tungsten (in some cases), 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

Lead, depleted uranium, tungsten, titanium, 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

2.3.1 Fabrication

Package components are procured, manufactured and inspected for use under QSA Global Inc. NRC approved QA Program Number 0040. This QA program is based on the application of guidance contained in NUREG/CR-6407 "Classification of Transportation Packing and Dry Spent Fuel Storage System Components According to Importance to Safety" (1996). Quality Class A components on the package are considered to be important to the package safety. All transport package components are evaluated and documented for compliance to the drawings provided in Appendix 1.3 prior to initial use as part of a Model 650L transport package.

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 General Requirements For All Packages

2.4.1 Minimum Package Size

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 and therefore exceeds the minimum package size requirements.

2.4.2 Tamper-Indicating Feature

The Model 650L package incorporates a seal wire attached to two bolts which secure the protective lid to the top plate. If the seal wire is broken during transport it serves as evidence of possible unauthorized access to the contents.

2.4.3 Positive Closure

These packages do not involve complex containment systems for source securement. The sources for these packages are all special form, welded capsules. The source wire assembly is held securely in the device by components of the lock assembly. One of these components, the lock slide, engages the stop ball of the source wire to prevent it from accidentally being pulled from the shielded position in the package.

When the 650L package is prepared for transport, the source assembly is secured by the lock slide, the lock plunger is depressed and the key removed, preventing source movement. A cover over the source wire connector prevents access to the source assembly. This cover is in place during transport of the package.

2.5 Lifting and Tie-Down Standards for All Packages

2.5.1 Lifting Devices

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).

2.5.2 Tie-Down Devices

The Model 650L package has no tie down devices. The package can be blocked and braced according to standard transportation practices.

2.6 Normal Conditions of Transport

2.6.1 Heat

The heat sources for the Model 650L transport package are listed in Table 2.6.A. Iridium-192 releases approximately 8.6 milliwatts per Curie based on a decay energy of 1.46 MeV/decay. Assuming all the decay energy is transformed into heat, the heat generation rate for 8.88 TBq (240 Ci) of Ir-192 (when corrected from content to output curies) would be approximately 4.8 Watts. The thermal evaluation for the heat test is described in Section 3 and is based on the decay energy of Ir-192 since it releases a greater amount of decay energy than Se-75 (see Table 1.2.A).

Assuming the entire decay heat, 4.8 Watts, is absorbed by the package, this would result in a worst case package surface temperature of 46°C (115°F) (Section 3.4.1.2). Accounting for solar heating effects (Section 3.4.1.1), the maximum temperature of the package surface was calculated to be 70°C (158°F). Since the sources in a fully loaded Model 650L package generate no more than 4.8 Watts as shown in Table 2.6.A, it can be assumed that no part of the package will be greater than 70°C (158°F) or be significantly affected by heating effects. In addition, the materials used in these packages will not be significantly affected by 70°C (158°F).

Table 2.6.A: Radionuclide Decay Energy

Radionuclide	Package Activity (Ci)	MeV/Decay	Watts/Package
Iridium-192	240	1.46	4.8
Selenium-75	300	0.86	1.52

Resource references:

Table of Isotopes, Volumes I & II, Eighth Edition. John Wiley & Sons, Inc., 1996.

2.6.1.1 Summary of Pressures and Temperatures

Table 2.6.B: 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 affect package performance or integrity. Evaluation of pressures for this package are contained in Section 3.4.2 and summarized in Table 3.1.B.

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 2.5.1 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

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

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 2 MN/m^2. Therefore, the reduced external pressure requirements of 3.5 psi in 10 CFR and 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

The Model 650L transport package is open to the atmosphere and contains 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 2 $MN/m^{2.}$ Therefore, the increased external pressure requirements of 20 psi in 10 CFR 71 will not adversely affect the package containment.

2.6.5 Vibration

The Model 650L (and 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

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

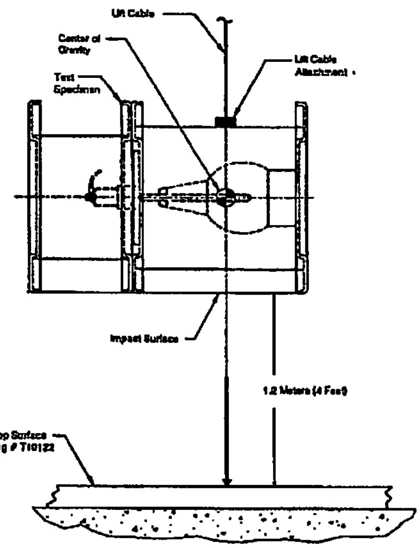
Three test specimens, as described in Test Plan 80 Report (Section 2.12) were subjected to the 1.2 meter (4 foot) free drop followed by the hypothetical accident 9 m drop and puncture bar drop tests. All free drop and penetration tests were conducted with the test specimen temperatures at or below -40°C (-40°F). The test specimens included standard locking

assemblies. Drop orientation impact locations for the 1.2 m free drop are shown in Figures 2.6.A through 2.6.C. 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 2.6.A) was performed on test specimen TP80(A).



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

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

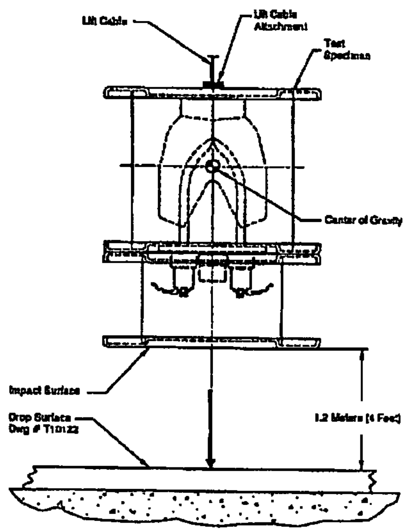
Damage to TP80(A) was limited to impact witness marking on the bottom plate, top plate and both lid flanges. There was no significant change in the radiation profile of the test specimen after the 1.2 m (4 ft) drop test (See Section 5).

2.6.7.2. 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:

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

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



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

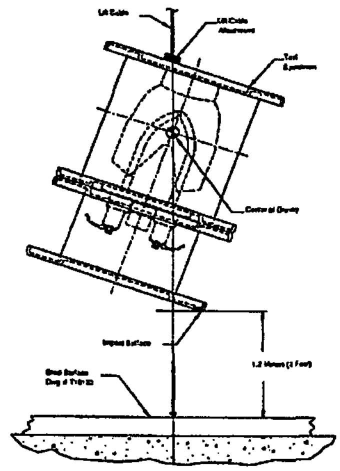
Figure 2.6.B - Model 650L (TP80(B)) 1.2 m Drop Test Orientation Vertical Upside Down Drop

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).

2.6.7.3 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 2.6.C) was performed on test specimen TP80(C).



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

Figure 2.6.C - 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

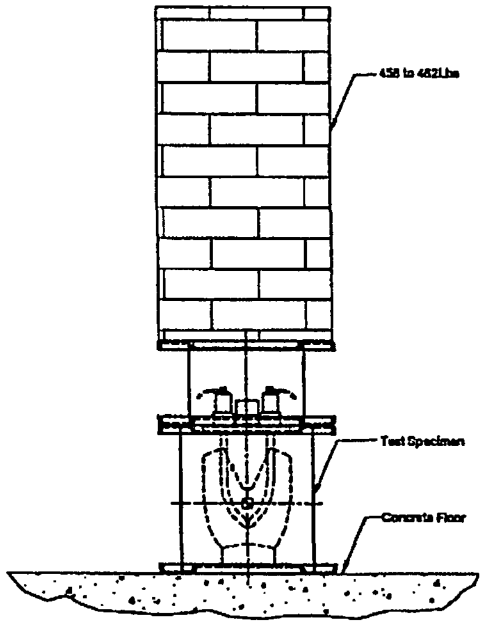
This test is not applicable, as the transport package does not transport fissile material, nor is the exterior of the transport package made from either fiberboard or wood.

2.6.9 Compression or Stacking

Test Plan 80 Report (Section 2.12) 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 2.6.D). 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.



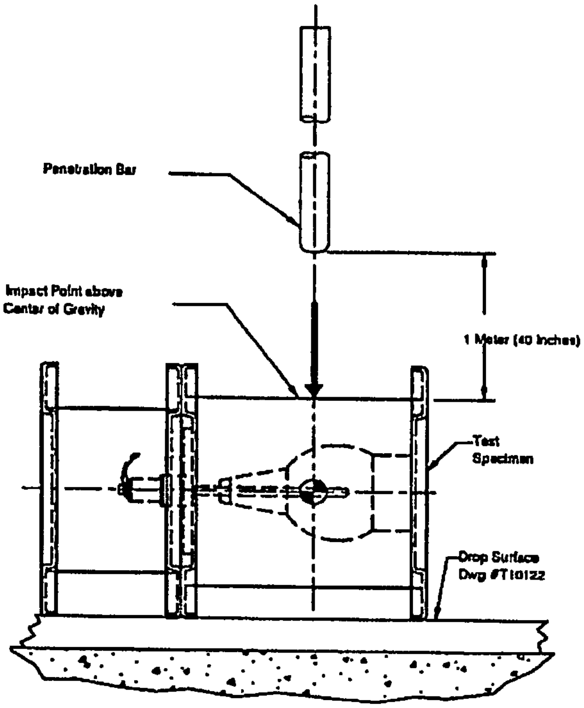
Compression Test Orientation – All Specimens

Figure 2.6.D - Model 650L Compression Test Orientation

2.6.10 Penetration

Test Plan 80 Report (Section 2.12) 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 2.6.E). 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.



Penetration Test Orientation – All Specimens

Figure 2.6.E - Model 650L Penetration Test Orientation

2.7 Hypothetical Accident Conditions of Transport

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).

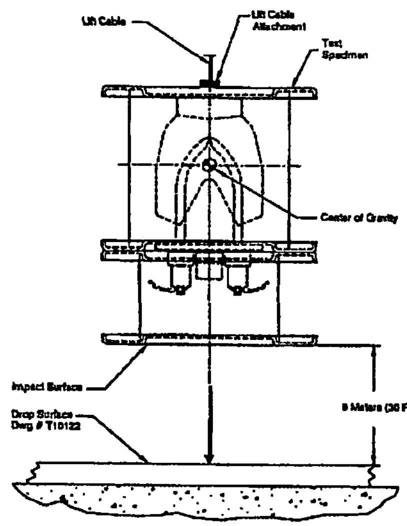
2.7.1 Free Drop

Justification for all test unit drop orientations are included in Test Plan 80 Report (Section 2.12). 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 2.7.A. 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 inch (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.

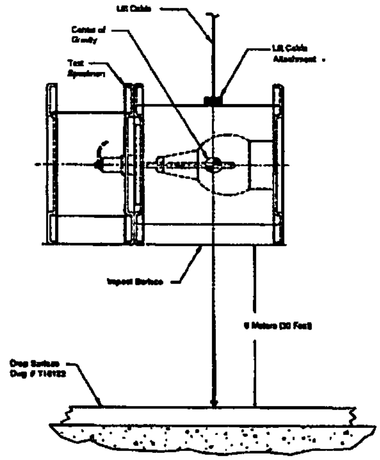


9 Meter (30 Foot) Drop Orientation for Specimen TP80(B)

Figure 2.7.A - 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 2.7.B. 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.

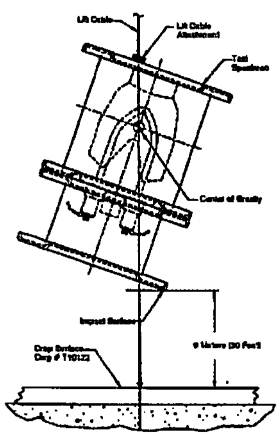


9 Meter (30 Foot) Drop Orientation for Specimen TP80(A)

Figure 2.7.B - 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 2.7.C. 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.



9 Meter (30 Foot) Drop Orientation for Specimen TP80(C)

Figure 2.7.C - 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). 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

See Table 2.7.A 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

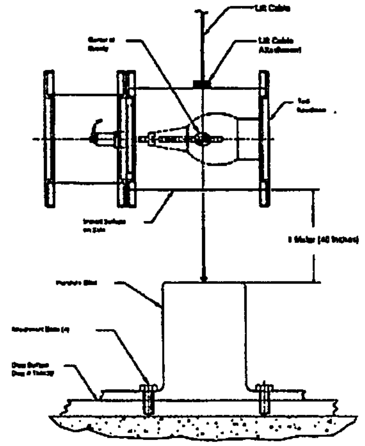
Not applicable. This package is not used for the Type B transport of normal form radioactive material.

2.7.3 Puncture

Justifications for all test unit puncture orientations are included in Test Plan Report 80 (Section 2.12) 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 2.7.D). 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.



Puncture Drop Orientation for Specimen TP80(A)

Figure 2.7.D - 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.

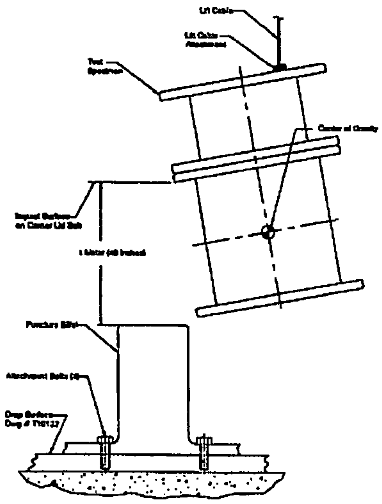
2.7.3.3 Test Specimen TP80(C)

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 2.7.E.

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.



Second Puncture Drop Orientation for Specimen TP80(C)

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

2.7.3.4 Summary of Results

See Table 2.7.A for additional test unit results summary. A more detailed summary is given in Test Plan 80 Report (Section 2.12). 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

See Section 3.5 for a discussion of the thermal test performed on the 650L package.

2.7.4.1 Summary of Pressures and Temperatures

These containers are vented to atmosphere. As such, no pressure will build up in the units under Hypothetical Accident conditions. See Tables 3.1.A and 3.1.B for summaries of temperature and maximum pressure related to the Model 650L package.

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) 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

This analysis demonstrates that the pressure inside the source capsules used in conjunction with the model 650L container, when subjected to the Hypothetical Accident Conditions of Transport thermal test, does not exceed the pressure which corresponds to the minimum yield strength at the thermal test temperature.

Under the Hypothetical Accident Conditions, it is assumed that the capsule could reach a temperature of 800°C (1,475°F). The special form capsules transported in the Model 650L package meet an ANSI N43.6 Class 6 temperature rating. For a classification of 6, the capsule has been subjected to an 800°C temperature for one hour, and subsequently thermally shocked from 800°C to 20°C with no resulting failure of weld or capsule. Therefore, any special form capsule with a minimum ANSI N43.6 Temperature Classification of 6 will not fail during Hypothetical Accident Conditions.

Reference: ANSI/HPS N43.6-1997 Sealed Radioactive Sources – Classification.

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

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

2.7.6 Immersion – All Packages

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)

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

2.7.8 Summary of Damage

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

Table 2.7.A: 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 2.6.E)	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)
	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 2.6.E)	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)
	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.

Test Specimen	Test	Actual Impact Point	Damage Observed at Test Site
	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 2.7.F	• 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.
	Penetration Bar	Side of Container Shell (See Figure 2.6.E)	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)
	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 2.7.E)	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), 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 or Packages with Large Quantities of Radioactivity

Not applicable. This package is not used for transport of plutonium or normal form radioactive material. This package is also not used for transport of special form material in quantities ≥ 3,000 A_1.

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

The Model 650L transport package is designed for use with a variety of special form source capsules which meet the ISO 2919-1999 requirements for Class 3 pressure testing and Class 6 for temperature testing. The source capsule must be special form, meet the ISO 2919-1999 Class 3 pressure criteria and meet the ISO 2919-1999 Class 6 temperature criteria for transport in the Model 650L.

Typical special form sources, transported as Type B quantities of radioactive material, in this container include the Models 875 Series and the X540/1. These special form capsules are assembled into flexible source wire assemblies (typical Models include A424-9 and A424-25W) which allow for storage of the radioactive material in the shield position of the package.

Details of the Model A424-9 source wire assembly can be found under USA SS&D registration MA-1059-S-104-S and the Ir-192 capsule under USDOT Special Form Certificates USA/0335/S-96.

Details of the Model A424-25W source wire assembly can be found under USA SS&D registration MA-1059-S-126-S and the Se-75 capsule under USDOT Special Form Certificates USA/0335/S-96 and USA/0502/S-96.

Based on performance testing, any source capsule that has been tested and achieved special form classification from a Competent Authority, has achieved an ANSI/ISO Pressure Classification rating of 3 and has also achieved an ANSI/ISO Temperature Classification of 6 can be safely transported in the Model 650L package so long as it is attached to a source wire assembly that is secured by the lock mechanisms on the top plate and locates the source capsule in the shielded position within the package. Therefore any source assembly/capsule combination meeting these criteria should be approved for transport without requirement of amendment to the Type B(U) certification.

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).

2.12.3 USDOT Special Form Certificate USA/0335/S-96 Rev 10

2.12.4 USDOT Special Form Certificate USA/0502/S-96 Rev 8

2.12.1 Test Plan 80 Rev 1 (March 1999)

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

2.12.3 USDOT Special Form Certificate USA/0335/S-96 Rev 10

U.S. Department
of Transportation

Pipeline and
Hazardous Materials
Safety Administration

East Building, PHH-23
1200 New Jersey Avenue SE
Washington, D.C. 20590

IAEA CERTIFICATE OF COMPETENT AUTHORITY
SPECIAL FORM RADIOACTIVE MATERIALS
CERTIFICATE NUMBER USA/0335/S-96, REVISION 10

This certifies that the sources described have been demonstrated to meet the regulatory requirements for special form radioactive material as prescribed in the regulations of the International Atomic Energy Agency[1] and United States of America[2] for the transport of radioactive material.

1. Source Identification - QSA Global, Inc. Model 875 Series.

2. Source Description - Cylindrical single or double encapsulations with the outer capsule made of Type 304L stainless steel and tungsten inert gas or laser welded. Approximate outer dimensions are 6.35 mm (0.25 in.) in diameter and either 19.05 mm (0.75 in.) or 24.2 mm (0.954 in.) in length. Inner capsules, when present, are made of stainless steel or titanium. Construction of the outer capsule shall be in accordance with attached QSA Global, Inc. Drawing No. R875 OUTER, Rev. C. Construction of any inner capsule shall be in accordance with attached QSA Global, Inc. Drawing No. R875 INNER, Rev. C or QSA Global, Inc. Drawing No. R87527-40, Rev. A.

3. Radioactive Contents - No more than either 14.8 TBq (400 Ci) of Iridium-192 as a solid metal; 8.14 TBq (220 Ci) of Cobalt-60 as a solid metal; 5.56 TBq (150 Ci) of Selenium-75 as an encapsulated solid metal; 1.11 TBq (30 Ci) of Cesium-137 as encapsulated $CsCl_2$; 1.85 TBq (50 Ci) of Thulium-170 as Tm_2O_3; or 7.4 TBq (200 Ci) of Ytterbium-169 as Yb_2O_3.

4. Quality Assurance - Records of Quality Assurance activities required by Paragraph 310 of the IAEA regulations[1] shall be maintained and made available to the authorized officials for at least three years after the last shipment authorized by this certificate. Consignors in the United States exporting shipments under this certificate shall satisfy the applicable requirements of Subpart H of 10 CFR 71.

5, Expiration Date - This certificate expires June 30, 2017.

This certificate is issued in accordance with paragraph 804 of the IAEA Regulations and Section 173.476 of Title 49 of the Code of Federal Regulations, in response to the June 01, 2012 petition by QSA Global, Inc., Burlington, MA, and in consideration of other information on file in this Office.

Certified by:

Dr. Magdy El-Sibaie
Associate Administrator for
Hazardous Materials Safety

JUN 26 2012
(DATE)

Revision 10 - Issued to extend the expiration date.

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[1] "Regulations for the Safe Transport of Radioactive Material, 1996 Edition (Revised), No. TS-R-1 (ST-1, Revised)," published by the International Atomic Energy Agency (IAEA), Vienna, Austria.

[2] Title 49, Code of Federal Regulations, Parts 100 - 199, United States of America.

Security-Related Information
Figure Withheld Under 10 CFR 2.390

DIMENSIONS IN INCHES TOLERANCES: FRACTIONS ± 1/16 .X ± 0.1 .XX ± 0.01 .XXX ± 0.005	QSA GLOBAL 40 NORTH AVE, BURLINGTON, MA 01803	SIZE A	DWG. NO. R65006	REV K
		SCALE: NONE	SHEET 5 OF 5	

Security-Related Information
Figure Withheld Under 10 CFR 2.390

APPROVALS	DATE
	7-24-07
	24Aug07

UNLESS OTHERWISE SPECIFIED
DIMENSIONS IN INCHES
TOLERANCES:
FRACTIONS ± 1/8
X.X ± 0.12
X.XX ± 0.06
X.XXX ± 0.020

QSA GLOBAL
40 NORTH AVE. BURLINGTON, MA 01803

DESCRIPTIVE DRAWING

TITLE 875 SERIES SSDR OUTER CAPSULE

SIZE	DWG. NO.	REV
A	R875 OUTER	C
	SCALE: NONE SHEET 1 OF 1	

ERF # 1739

Security-Related Information
Figure Withheld Under 10 CFR 2.390

APPROVALS	DATE
R. [illegible]	25 Jul 07
L. [illegible]	25 Jul 07

UNLESS OTHERWISE SPECIFIED
DIMENSIONS IN INCHES
TOLERANCES:
FRACTIONS ± 1/8
X.X ± 0.12
X.XX ± 0.06
X.XXX ± 0.020

QSA GLOBAL
40 NORTH AVE, BURLINGTON, MA 01803

DESCRIPTIVE DRAWING

TITLE 875 SERIES INNER CAPSULE

SIZE A | DWG. NO. R875 INNER | REV C

SCALE: NONE | SHEET 1 OF 1

ERF # 1739

2.12.4 USDOT Special Form Certificate USA/0502/S-96 Rev 8

U.S. Department
of Transportation

**Pipeline and
Hazardous Materials
Safety Administration**

**East Building, PHH-23
1200 New Jersey Avenue Southeast
Washington, D.C. 20590**

**IAEA CERTIFICATE OF COMPETENT AUTHORITY
SPECIAL FORM RADIOACTIVE MATERIALS
CERTIFICATE USA/0502/S-96, REVISION 8**

This certifies that the sources described have been demonstrated to meet the regulatory requirements for special form radioactive material as prescribed in the regulations of the International Atomic Energy Agency[1] and the United States of America[2] for the transport of radioactive material.

1. Source Identification - QSA Global, Inc. Model Nos. X54 (Manufactured before January 1, 1998), X540 (Manufactured on or after February 17, 1981), and X540/1 (Manufactured on or after September 27, 2000).

2. Source Description - Tungsten inert gas or laser seal welded cylindrical single or double encapsulations. The outer encapsulation is made of titanium or stainless steel and the inner encapsulation, if used, is made of titanium, stainless steel, or aluminum. Approximate exterior dimensions are 5.15 mm (0.2 in.) maximum diameter and 15.15 mm (0.6 in.) in length (Model X54); and 5.16 mm (0.2 in.) in diameter and 7.65 mm (0.3 in.) in length (Models X540 and X540/1). Construction shall be in accordance with attached Amersham Drawing No. A10639, Issue C (Model X54) or QSA Global Inc. Drawing No. R87527, Rev. G (Models X540 and X540/1).

3. Radioactive Contents - No more than 17.0 TBq (459.5 Ci) of Cobalt-60 (Model X54); or no more than either 20.0 TBq (540.5 Ci) of Cobalt-60, 17.0 TBq (459.5 Ci) of Iridium-192, or 5.56 TBq (150.3 Ci) of Selenium-75 (Models X540 and X540/1). The Co-60, Ir-192, and Se-75 are in the form of a metal.

4. Quality Assurance - Records of Quality Assurance activities required by Paragraph 310 of the IAEA regulations[1] shall be maintained and made available to the authorized officials for at least three years after the last shipment authorized by this certificate. Consignors in the United States exporting shipments under this certificate shall satisfy the applicable requirements of Subpart H of 10 CFR 71.

5. Expiration Date - This certificate expires on July 31, 2017.

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[1] "Regulations for the Safe Transport of Radioactive Material, 1996 Edition (Revised), No. TS-R-1 (ST-1, Revised)," published by the International Atomic Energy Agency(IAEA), Vienna, Austria.

[2] Title 49, Code of Federal Regulations, Parts 100-199, United States of America.

CERTIFICATE USA/0502/S-96, REVISION 8

This certificate is issued in accordance with paragraph 804 of the IAEA Regulations and Section 173.476 of Title 49 of the Code of Federal Regulations, in response to the July 06, 2012 petition by QSA Global, Inc., Burlington, MA, and in consideration of other information on file in this Office.

Certified By:

for Dr. Magdy El-Sibaie
Associate Administrator for Hazardous Materials Safety

Jul 10 2012
(DATE)

Revision 8 - Issued to extend the expiration date.

Security-Related Information
Figure Withheld Under 10 CFR 2.390

SCALE	ISSUE	MOD No.	DATE	DRAWN	CHECKED	APPROVED	QA APPROVED
10:1	C	MS1211	4.1.85	M.A.			

Amersham
The Health Science Group

TITLE ASSEMBLY OF CAPSULE X54

TOLERANCES ——— UNLESS OTHERWISE STATED

MATERIAL ———

SURFACE TEXTURE ——— UNLESS OTHERWISE STATED

FINISH ——— REMOVE ALL BURRS

GENERAL NOTES

THIRD ANGLE PROJECTION

MODIFICATIONS INDICATED BY ISSUE IN ◇

THIS DRAWING CONFORMS TO BS308.

ALL DIMENSIONS IN MILLIMETRES UNLESS OTHERWISE STATED.

DO NOT SCALE

APPROVAL
THIS DRAWING IS NOT TO BE USED FOR ANY PURPOSE UNLESS SIGNED AS APPROVED

USED ON ———

SHT. SIZE A3

DRG NO. A10639

SHT 1 OF SHTS 1

Security-Related Information
Figure Withheld Under 10 CFR 2.390

APPROVALS	DATE
D. Prince	30 May 06
L. Roberts	30 May 06

UNLESS OTHERWISE SPECIFIED
DIMENSIONS IN INCHES
TOLERANCES:
FRACTIONS ± 1/8
X.X ± 0.12
X.XX ± 0.06
X.XXX ± 0.020

QSA GLOBAL
40 NORTH AVE, BURLINGTON, MA 01803

DESCRIPTIVE DRAWING

TITLE X540 CAPSULE SERIES

SIZE	DWG. NO.	REV
A	R87527	G

SCALE: NONE SHEET 1 OF 1

ERF #	1402

Section 3 - THERMAL EVALUATION

3.1 Description of Thermal Design

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 (8.88 TBq) of Ir-192 or 300 Ci (11.1 TBq) of Se-75. Accounting for self absorption in the source, this equals a maximum content activity of 552 Ci (20.4 TBq) of Ir-192. The corresponding decay heat generation rate for the content activity is approximately 4.8 Watts (See Table 1.2.A). The thermal evaluations are based on the decay energy of Ir-192 as this is greater than the decay energy of 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 Decay Heat of Contents

From Table 1.2.A, 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 3.1.A: 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
Maximum Fire Test Temperature	996°C (1,825°F)	See Test Plan Report TP80
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.

Any pressure generated within the special form source is significantly below that which would be generated during the Hypothetical Accident Conditions thermal test, which is shown in Section 2.7.4.3 to result in no loss of structural integrity or containment.

Table 3.1.B: 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 3.2.A 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 3.2.A: 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
Polyurethane Foam	8 lb/ft^3	--	150μin/in°C	Reference #1, p. 6-199
Tungsten	0.58 min	3,426°C (6,200°F)	2.5 μin/in °F	Reference #1, p. 6-11

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 Appendix 1.3.

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.

3.4 Thermal Evaluation for Normal Conditions of Transport

3.4.1 Heat and Cold

3.4.1.1 Insolation and Decay Heat

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. In order to ensure conservatism, the following assumptions are made:

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

- 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.

- 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.

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

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

- 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.

In the steady state Heat Input = Heat Output, $Q_{IN} = Q_{OUT}$

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 3.4.A.

Table 3.4.A: 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:

reflectivity, ρ + absorptivity, $\alpha = 1$ where $\rho = 0$ 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.

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

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

Radiation Heat Transfer (Q_R):

$$Q_R = B\ E\ A_{TS}\ \{(T_W + 273)^4 - (T_A + 273)^4\}$$

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, $Q_R = 1.34 \times 10^{-8}\ \{(T_W + 273)^4 - 9.35 \times 10^9\}$

Lid top surface convection (Q_{LT}):

$$Q_{LT} = H_{LT}A_{LT}(T_W - T_A)$$

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
$\beta = 0.00322$ ($1/T_A$ assuming $T_A = 311$ 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

$$H_{LT} = 2.79\ (T_W - 38)^{0.25}$$

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

$$Q_{LT} = 0.148\ (T_W - 38)^{1.25}$$

Lid side surface convection (Q_{LS}):

$$Q_{LS} = H_{LS}A_{LS}(T_W - T_A)$$

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:

$$H_{LS} = 0.346 + 2.67\ (T_W - 38)^{0.25}$$

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

$$Q_{LS} = 0.041\ (T_W - 38) + 0.315\ (T_W - 38)^{1.25}$$

Outer shell surface convection (Q_{OS}):

$$Q_{OS} = H_{OS}A_{OS}\ (T_W - T_A)$$

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:

$$H_{OS} = 0.194 + 2.47\ (T_W - 38)^{0.25}$$

Substituting and solving for Q_{OS} produces

$$Q_{OS} = 0.024(T_W - 38) + 0.306\ (T_W - 38)^{1.25}$$

As stated above, $Q_{IN} = \alpha(Q_{IT} + Q_{ILS} + Q_{IOS}) + Q_{DT} = 120.4$ W, and $Q_{OUT} = Q_R + Q_{LT} + Q_{LS} + Q_{OS}$. Setting $Q_{IN} = Q_{OUT}$ and substituting produces:

$$120.4\ W = [1.34 \times 10^{-8}\{(T_W + 273)^4 - 9.35 \times 10^9\}] + [0.148\ (T_W - 38)^{1.25}] + [0.041\ (T_W - 38) + 0.315\ (T_W - 38)^{1.25}] + [0.024\ (T_W - 38) + 0.306\ (T_W - 38)^{1.25}]$$

Which when reduced produces:

$$120.4\ W = 1.34 \times 10^{-8}(T_W + 273)^4 - 125.29 + 0.769\ (T_W - 38)^{1.25} + 0.065\ (T_W - 38)$$

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.

3.4.1.2 Still Air (shaded) Decay Heating

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).

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

- The entire decay heat (4.8 W) is deposited in the exterior surfaces of the package.
- The interior of the package is perfectly insulated and heat transfer occurs only from the exterior surface to the environment.
- 100% of the total heat is conservatively assumed to be deposited in the sides of the package lid.
- 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)
$h = 5\ \mathrm{W/m^2\ K}$ (free convection heat transfer coefficient for air)
$A = 0.118\ \mathrm{m^2}$ (surface area of the lid sides)
$T_A = 311$°K (ambient air temperature of 38°C)

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

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 Affected 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 tests at temperatures below -40°C (-40°F), therefore the package complies with the requirements of this section.

3.4.2 Temperatures Resulting in Maximum Thermal Stresses

The temperature and pressure variations described in Sections 3.4.1 and 3.4.3 will not adversely affect the transport package during normal transport. The melting temperatures of all safety critical components are well above these temperatures and there will be no pressures generated in the package to cause package failure. It is therefore concluded that the Model 650L transport package will maintain its structural integrity and shielding effectiveness under the normal transport thermal stress conditions.

3.4.3 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.

3.5 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 and 2.12 for detailed description of the test specimen initial conditions.

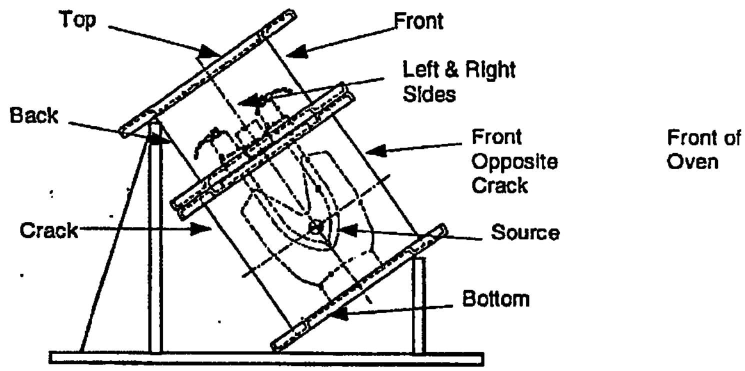
3.5.2 Fire Test Conditions

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 3.5.A). The side with the crack was facing down.



TP80(B) Orientation and Thermocouple Locations

Figure 3.5.A - 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).

3.5.3 Maximum Temperatures and Pressure

Sections 2.12 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 Temperatures Resulting in Maximum Thermal Stresses

Sections 2.12 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 Fuel/Cladding Temperatures for Spent Nuclear Fuel

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

3.5.6 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

The containment system for the Model 650L transport package are the radioactive source capsules. The source capsules transported in the 650L transport package are certified as special form radioactive material under 49 CFR 173 and IAEA TS-R-1. The special form source capsule, stop ball and connector are swaged to a flexible steel wire to form the source wire assembly.

The source wire assembly is maintained within the shielded configuration of the package by components of the lock assembly after being inserted into the shield tube. The lockslide component engages the source wire and prevents it from being pulled through the top of the lock assembly.

4.1.1 Special Requirements for Damaged Spent Nuclear Fuel

Not applicable. This package is not used for the transport of spent nuclear fuel.

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

As demonstrated in the Test Plan 80 Report (Section 2.12) 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 package, therefore, meets the requirements of this section.

4.3 Containment Under Hypothetical Accident Condition

As demonstrated in the Test Plan 80 Report (Section 2.12) 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 package, therefore, meets the requirements of this section.

4.4 Leakage Rate Tests for Type B Packages

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 (185 Bq) 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.

4.5 Appendix

Not Applicable.

Section 5 - SHIELDING EVALUATION

5.1 Description of Shielding Design

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 Appendix 1.3.

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).

Table 5.1.A: Model 650L Test Unit TP80(A) Summary Table of External Radiation Levels Extrapolated to Capacity of 240 Ci (8.88 TBq) Ir-192 (Non-Exclusive Use)

Normal Conditions of Transport[2]	Package Surface mSv/hr (mrem/hr)			1 Meter from Package Surface mSv/hr (mrem/hr)		
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) or Paragraphs 530 and 531 of TS-R-1 Limit	2 (200)	2 (200)	2 (200)	0.1 (10)[1]	0.1 (10)[1]	0.1 (10)[1]
Hypothetical Accident Conditions[2]						
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 71.51(a)(2) or Paragraph 656(b)(ii)(I) of TS-R-1 Limit				10 (1,000)	10 (1,000)	10 (1,000)

[1]Transport Index may not exceed 10.

[2]Table results are extrapolated to the device capacity and incorporate surface correction factors

Table 5.1.B: Model 650L Test Unit TP80(B) Summary Table of External Radiation Levels Extrapolated to Capacity of 8.88 TBq (240 Ci) Ir-192 (Non-Exclusive Use)

Normal Conditions of Transport[2]	Package Surface mSv/hr (mrem/hr)			1 Meter from Package Surface mSv/hr (mrem/hr)		
Radiation	Top	Side	Bottom	Top	Side	Bottom
Gamma	0.71 (71)	0.83 (83)	0.83 (83)	0.02 (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)	0.02 (2.0)	0.008 (0.8)	0.007 (0.7)
10 CFR 71.47(a) or Paragraphs 530 and 531 of TS-R-1 Limit	2 (200)	2 (200)	2 (200)	0.1 (10)[1]	0.1 (10)[1]	0.1 (10)[1]
Hypothetical Accident Conditions of Transport[2]						
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 71.51(a)(2) or Paragraph 656(b)(ii)(I) of TS-R-1 Limit				10 (1,000)	10 (1,000)	10 (1,000)

[1]Transport Index may not exceed 10.
[2]Table results are extrapolated to the device capacity and incorporate surface correction factors

Table 5.1.C: Model 650L Test Unit TP80(C) Summary Table of External Radiation Levels Extrapolated to Capacity of 8.88 TBq (240 Ci) Ir-192 (Non-Exclusive Use)

Normal Conditions of Transport[2]	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	0.59 (59)	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	0.59 (59)	1.06 (106)	0.59 (59)	0.002 (2.0)	0.008 (0.8)	0.005 (0.5)
10 CFR 71.47(a) or Paragraphs 530 and 531 of TS-R-1 Limit	2 (200)	2 (200)	2 (200)	0.1 (10)[1]	0.1 (10)[1]	0.1 (10)[1]
Hypothetical Accident Conditions of Transport[2]						
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 71.51(a)(2) or Paragraph 656(b)(ii)(I) of TS-R-1 Limit				10 (1,000)	10 (1,000)	10 (1,000)

[1]Transport Index may not exceed 10.
[2]Table results are extrapolated to the device capacity and incorporate surface correction factors.

Table 5.1.D 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 5.1.D: Model 650L s/n 274 Se-75 Profile Results Summary Table of External Radiation Levels Extrapolated to Capacity of 11.1 TBq (300 Ci) Se-75 (Non-Exclusive Use)

	Package Surface mSv/hr (mrem/hr)			1 Meter from Package Surface mSv/hr (mrem/hr)		
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) or Paragraphs 530 and 531 of TS-R-1 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

The gamma sources allowed for transport in the Model 650L transport package are specified in Sections 1.2.2 and 2.10.

5.2.1 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

Not applicable. 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. See Test Plan Report 80 (Section 2.12) for results of radiation surveys of the 650L test specimens. The test specimens were profiled before testing, and after the hypothetical accident testing. These results are shown in Tables 5.1.A through and 5.1.D. All radiation profile data are within regulatory acceptance limits. All newly manufactured packages were profiled prior to final acceptance and all packages are surveyed prior to shipment. This final acceptance 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:

$$CF_A = \frac{Maximum\,Package\,Activity\,Capacity\ (A_C)}{Actual\ \mathrm{Pr}\,ofile\,Activity\ (A_P)}$$

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

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

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 was 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 package surface 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.

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. All paragraph references to IAEA TS-R-1 apply to IAEA Regulations for the Safe Transport of Radioactive Material No. TS-R-1 (ST-1, Revised) 1996 Edition (Revised).

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. Shipments 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

The Model 650L transport packages are designed to transport 240 Ci (8.88 TBq) of Ir-192 or 300 Ci (11.1 TBq) of Se-75 as special form capsules attached to source wire assemblies.

The Model 650L transport packages are designed for use with a special form source capsule as approved under a Competent Authority (e.g. U.S. Department of Transportation) special form certification. Details of encapsulation as well as chemical and physical form of the radioactive material will comply with specifications approved under U.S. Department of Transportation special form certifications. Sources transported in this package must also meet a minimum ANSI N43.6-1997 Temperature Classification of 6 and Pressure Classification of 3.

7.1.1.2 Packaging Maintenance and Inspection Prior to Loading

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

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.

c. Ensure all lid cover bolts and fasteners (hardware) required for assembly of the package and as specified on the drawings referenced on the Type B transport certificate are present, intact, and fit for use. Without disassembly or removal from the device, examine the visible external surfaces of the top plate bolts and lock assembly screws for any signs of fatigue cracking.

   The bolts/fasteners must be replaced if they are no longer fit for use (e.g., threads stripped, unable to fully thread, signs of cracking, etc). Ensure any replacement hardware meets all applicable specifications listed on the drawings referenced on the Type B transport certificate.

d. Ensure the locking assemblies allow free movement of the lock slide when performing an operational test and that the plunger lock engages and is functional. Ensure the shipping caps install and secure over the source tubes on the lock assemblies.

e. Ensure 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.

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 Ensure the contents are authorized for use in the package.

7.1.2.2 Ensure the package condition has been inspected in accordance with Section 7.1.1.

7.1.2.3 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.

a. 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.

b. 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 the source is loaded, push the lock slide to the "locked" position, depress the plunger lock and remove the key.

c. Install the shipping cap over the source on the lock assembly.

d. Repeat 7.1.2.3.b and 7.1.2.3.c for the second source tube if transporting more than one source in the container.

e. Secure the shipping cover to the container top plate using the hardware specified on the descriptive assembly drawing (see the drawings referenced on the Type B transport certificate). Tighten the screws so that no gap exists between the screw heads, cover or container top plate.

7.1.3 Preparation for Transport

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 4 Bq/cm^2 (0.0001 $\mu Ci/cm^2$) 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 ensure that the radiation level does not exceed 2 mSv/hr (200 mR/hr) at the surface. Measure the radiation level at one meter from all exterior surfaces to ensure that the radiation level is less than 0.1 mSv/hr (10 mR/hr).

7.1.3.4 Ship the container according to the procedure for transporting radioactive material as established in 10 CFR 71.5.

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

7.2.2.7 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.

7.2.2.8 Upon receipt of a transport package of radioactive material:

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 2 mSv/hr (200 mR/hr) at the surface of the transport package, nor 0.1 mSv/hr (10 mR/hr) at a distance of 1 meter from the surface.

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

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.

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.

e. Record the radioisotope, activity, model number, and serial number of the source(s) and the transport package model number and serial number.

7.2.2 Removal of Contents

7.2.2.1 Unloading of 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

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 Unload the container in accordance with Section 7.2.2.

7.3.1.2 After removing the source(s) 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) there 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 Ensure that the levels of removable radioactive contamination on the outside surface of the transport package do not exceed 4 Bq/cm^2 (0.0001 $\mu Ci/cm^2$) 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/hr (0.5 mR/hr).

7.3.4 Ship the container according to the procedure for transporting radioactive material as established in 10 CFR 71.5.

7.4 Other Operations

7.4.1 Package Transportation by Consignor

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(0.005 μCi).

7.4.1.2 If contamination above 4 Bq/cm^2(0.0001 $\mu Ci/cm^2$), 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/hr (0.5 mR/hr) is detected on any conveyance or equipment surface, then remove the affected item from use until decontaminated or decayed to meet these limits.

7.4.2 Emergency Response

In the event of a transport emergency or accident involving this package, follow the guidance contained in "2014 Emergency Response Guidebook: A Guidebook for First Responders During the Initial Phase of a Dangerous Goods/Hazardous Materials Incident", or equivalent guidance documentation.

Reference: "2014 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

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

8.1.1.1 The transport package was assembled properly to the applicable drawings referenced on the Type B transport certificate.

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 All fasteners as required by the applicable drawings on the Type B transport certificate are properly installed and secured.

8.1.1.4 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.

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

Prior to first use 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.

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 μCi (185 Bq). 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.

The lock assembly of the device is tested to ensure that the security of the radioactive source will be maintained. Failure of this test prevents use of the device until the lock assembly is corrected and re-tested.

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. This survey is performed in a low background area and involves a slow scan survey of the entire surface area as well as one meter from the surface of the device. This survey is used to identify any significant void volumes or shield porosity which could prevent the finished device from complying with the dose limits in 10 CFR 71.47.

These radiation levels, when extrapolated to the rated capacity of the transport package, must not exceed 2 mSv/hr (200 mR/hr) at the surface, nor 0.1 mSv/hr (10 mR/hr) at 1 meter from the surface of the transport package. Failure of this test will prevent use of the device. In addition, the surface and 1 meter radiation levels are measured prior to every shipment. If the reading exceeds 2 mSv/hr at the surface or 0.1 mSv/hr at one meter, the package is not shipped.

Failure of the radiation profile tests for any Model 650L container indicates the potential of significant shielding porosity and causes the rejection of the affected Model 650L package. Rejected packages which do not comply with the construction requirements on the applicable drawings referenced on the Type B certificate, or that do not comply with the radiation profile requirements will not be distributed as approved Type B(U) packages.

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

Upon initial manufacture of the source assembly, and prior to first shipment of the source assembly, subject the swage coupling between the source capsule and cable to a static tensile test with a load of 100 lbs (445 N). Failure of this test will prevent use of the source in the Type B(U) transport package.

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 μCi (185 Bq). Additionally, a contamination wipe of the shield source tubes is performed 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

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 – QUALITY ASSURANCE

9.1 U.S. Quality Assurance Program Requirements

All component fabrication (including assembly) is controlled under the QSA Global, Inc. Quality Assurance program approved by the USNRC (approval number 0040) and ISO 9001.

9.2 Canada Quality Assurance Program Requirements

Not applicable. This package is originally submitted for certification in the United States and complies with the criteria in Section 9.1.