71-9314

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31 May 2007

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Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
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Docket No.: 71-9314 & TAC No: L24038

Dear Ms. Glenny:

The following is provided in response to your letter received via telefax on 22 May 2007 requesting additional information to support our amendment request for this package design. Items are addressed in the same order as your letter:

1-1 The requested notation change has been implemented to Tables 1.2a and 7.1.1.a. In addition Section 5.1.2 has been expanded to include additional justification regarding the acceptability of transporting more than one approved radionuclide in the container at the same time so long as the sum of ratios method is followed to limit total package contents.

1-2 Under the USDOT regulations the applicability of "Output Curies" relative to radioactive material transport only applies to Ir-192. We have clarified the notes to Tables 1.2a and 7.1.1a to indicate that Output Curies calculated based on ANSI N432-1980 applies only to Ir-192 per 49 CFR 173.435 note c. The Se-75 and Yb-169 activities are based on "Content Curies". A similar note has been clarified in Table 1.2d.

1-3a. Section 1.1 has been clarified to identify the revision of the IAEA TS-R-1 applicable throughout the remainder of the document and to clarify that all subsequent 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). Additional clarification occurs in other sections throughout the SAR where applicable.

1-3b. Tables 5.1e through 5.1l have been revised to remove the superscript 1 for the 1 meter top dose rate limit.

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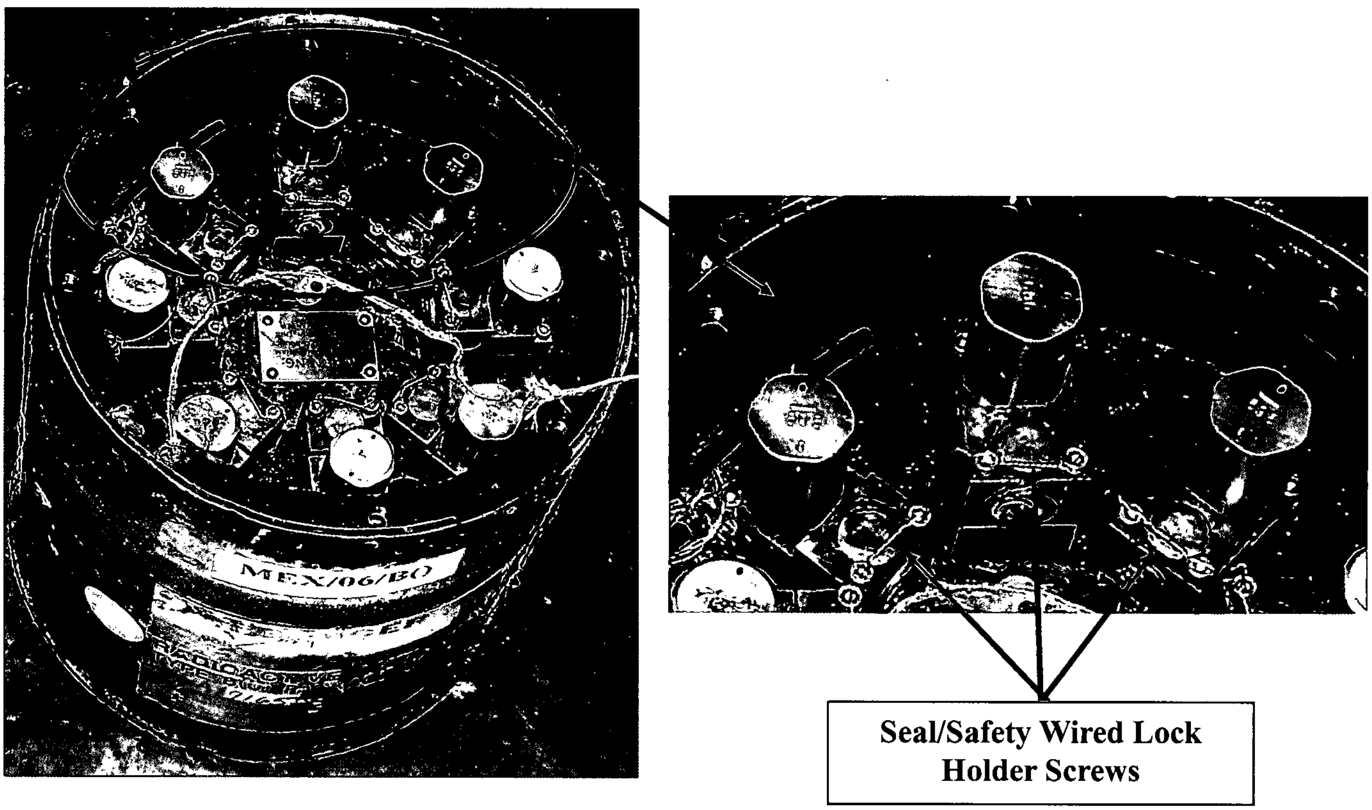
1-3c. The statements of the maximum TI in Note 2 to Table 5.1a and Note 2 on page 5-2 have been deleted as requested.

1-3d. The inconsistencies in the dose rates given in Tables 5.1c and 5.1k have been corrected.

1-3e. The sequence of steps described in section 7.1.1.2.a is correct. As shown in the following picture, the “safety” wires (or “seal” wires as noted on drawing R85590 Rev E sheet 3) are clearly visible from above the container once the lid is removed. The safety wires are not connected to the source caps or the source assemblies contained within the lock holder assembles. We have, however, revised the wording in Section 7.1.1.2.a.5 to clarify that the inspection should be performed after removal of the lid and to replace the word “safety” with “seal” to match the wording on the drawing.



1-3f. Section 7.1.1.2.d.4 has been revised to indicate a check to confirm the M10 bolts <u>are</u> fully threaded instead of <u>can be</u> fully threaded to the 3018.

1-3g. Section 7.1.1.2.f.5 has been revised to add a specific reference to “sources” in the check for foreign objects and obstructions.

1-3h. Section 7.1.2.1.c.4.i has been clarified as requested.

1-3i. The numbering of the operations steps in SAR Section 7.3.1.2 have been corrected.

In addition, it was noted during update of the SAR that some references to "AEA Technology, QSA, Inc." still occurred in the document. Places where this was referenced now indicate "QSA Global, Inc. Also, the indices were updated to ensure all tables, and major referenced sections were included in the index.

Enclosed is Revision 7 of the SAR which is complete except for Section 1.4, 2.12.1 through 2.12.3 and 5.5.1 and 5.5.2. A new appendix, Section 5.5.3 has been added and this is included. The other omitted sections remain unchanged from those currently on file with your office. Should you have any additional questions or wish to discuss this response after receipt please contact me as shown below.

Sincerely,

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C. Rayman 31 May 07
RA/QA Approval Date

31 May 07
Engineering Approval Date

Enclosures: Revised list of affected pages
SAR Rev 7 excluding Sections 1.4, 2.12.1 – 2.12.3, 5.5.1 and 5.5.2

List of Affected Pages	
Revision 0, 09 March 2004	Initial Submission of Safety Analysis Report
Revision 1 21 January 2005	Revision in Entirety in Response to RAI dated 04 August 2004 and to include the Model 976F.
Revision 2, 15 April 2005	Revision to Sections 2 and 3 addressing comments on thermal assessment raised during USNRC and AEA Technology QSA, Inc. meeting on 30 March 2005.
Revision 3, 04 May 2005	Revision to Section 1.4 drawings, Section 3.4.1.1 and Section 5.3.1.
Revision 4, 06 December 2005	Revision to Section 1.4 drawings and container weights for Models 976B through 976D in remainder of SAR. Affected pages are ii, 1-1, 1-3, 1-4, 2-1, 2-26, 2-28 and 3-19. Affected drawings from Section 1.4 are R976B, R976C, R976D, R976E, R3015, R3056, R3018 and R3078. The only change to these drawings involved increasing the maximum container weights.
Revision 5, 5 October 2006	Revision to modify shield placement in 976C container, to modify the dimensions for the 3018 shield pot and to add authorization for Se-75 and Yb-169 to all shield configurations. Affected pages are i-iv, 1-1, 1-2, 1-4, 1-9, 2-7, 3-1, 3-2, 3-3, 3-6, 3-7, 3-10, 3-11, 3-12, 3-18, 3-19, 3-20, 5-4, 5-6, 5-7, 5-8, 5-9, 7-1, 7-2 and 7-4 through 7-15. Affected drawings from Section 1.4 are R976C, R97615, R97615-1, R97615-2, R1911, R3018, R97623 and R97623A.
Revision 6, 21 February 2007	Revision in Response to RAI dated 7 Feb 07. Affected pages are 1-2, 1-5, 1-6, 1-10, 2-7, 2-19, 3-3, 3-7, 3-8, 3-9, 3-10, 3-11, 5-6, and 7-2.
Revision 7, 23 May 2007	Revision in Response to RAI received 22 May 07. Affected pages are i-v, 1-1, 1-2, 1-10, 2-1, 2-2, 2-4, 3-5, 4-1, 4-2, 5-1 thru 5-7, 5-11, 5-14, 7-1, 7-2, 7-4, 7-5, 7-7, 7-13, 7-14, 8-1 and 8-3.

Safety Analysis Report

QSA Global Inc.

Model 976 Series
Type B(U) - 96
Transport Package

23 May 2007

Revision 7

Contents

List of Tables

List of Figures

Section 1 - GENERAL INFORMATION

1.1 Introduction

The Model 976 Series are designed as transport packages and storage containers for Type B quantities of special form radioactive material. They conform to the Type B(U)-96 criteria for packaging in accordance 10 CFR 71, 49 CFR 173, and the IAEA Regulations for the Safe Transport of Radioactive Material No. TS-R-1 (ST-1, Revised) 1996 Edition (Revised) which were in effect at the time of sign-off of this report. All subsequent 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).

1.2 Package Description

(Reference:

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

The Model 976 Series packages are differentiated based on inner shield design, spacer configurations and activity capacities. The general design of the package is a steel jacketed lead and/or depleted uranium shield container housed within a cork lined, stainless steel drum. The containers are constructed in accordance with descriptive drawings in Section 1.4. Overall external dimensions for all 976 Series packages are 19 ¾" (502 mm) diameter and 21 ¼" (540 mm) tall. The package weights and isotope maximum capacities for the 976 Series are shown in Table 1 below:

Table 1.2a: Model 976 Series Package Information

Identification	Inner Shield(s)	Nuclide	Form	Maximum Capacity[1,2]	Maximum Weight
976A	855	Ir-192	Special Form Sources	1,000 Ci	136 kg (300 lb)
		Se-75		1,000 Ci	
		Yb-169		865 Ci	
976B	3015	Ir-192	Special Form Sources	350 Ci	86 kg (190 lb)
		Se-75		350 Ci	
		Yb-169		350 Ci	
976C	3056	Ir-192	Special Form Sources	1,250 Ci	86 kg (190 lb)
		Se-75		1,250 Ci	
		Yb-169		1,000 Ci	
976D	3018	Ir-192	Special Form Sources	500 Ci	86 kg (190 lb)
		Se-75		500 Ci	
		Yb-169		500 Ci	
976E	3078	Ir-192	Special Form Sources	1,000 Ci	103 kg (226 lb)
		Se-75		1,000 Ci	
		Yb-169		1,000 Ci	

Identification	Inner Shield(s)	Nuclide	Form	Maximum Capacity[1,2]	Maximum Weight
976F	1911	Ir-192	Special Form Sources	1,000 Ci	119 kg (263 lb)
		Se-75		1,000 Ci	
		Yb-169		1,000 Ci	

[1]For Iridium-192, the maximum capacity is based on the output curies which are determined by measuring the source output at 1 meter and expressing its activity in curies derived from the following: 0.48 R/hr-Ci Iridium-192 at 1 meter. (Ref: American National Standard N432-1980, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography."). For Selenium-75 and Yb-169 the maximum capacity is based on the content curies contained in the radioactive source(s).
[2] For shipments of multiple radioisotopes in a single package, the sum of the ratios of the curie quantity of each loaded isotope to the maximum allowed curie quantity of that isotope (were that isotope the only contents of the package) must be less than or equal to unity.

1.2.1 Packaging

The shield container Models 3056, 3078, 3015 and 3018 have been used in the field for over twenty (20) years without incident or problem as part of USDOT Type B endorsements of Great Britain Type B(U) approved packages. The Model 855 has been used in the field for over twenty (20) years without incident or problem as part of a USNRC and USDOT Type B approval. These containers and their associated Type B endorsement certifications are listed in Table 1.2b.

Inner Shield Container	USNRC Type B(U) Certificate	USDOT Type B(U) Endorsed Certificate	Great Britain Type B(U) Certificate	Intended Model 976 Package Designation
855	USA/9165/B(U)	USA/9165/B(U)	None	976A
3015	None	USA/0590/B(U)-85	GB/3605A/B(U)-85	976B
3018	None	USA/0592/B(U)-85	GB/3605B/B(U)-85	976D
3056	None	USA/0316/B(U)	GB/0924BZ/B(U)	976C
3078	None	USA/0250/B(U)	GB/0924BP/B(U)	976E

Table 1.2b – CROSS REFERENCE TABLE OF INNER SHIELD CONTAINER TRANSPORT APPROVAL HISTORY

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

1.2.1.1 Inner Shield: The Special Form source(s) are contained in an inner shield container. The shield containers shielding is comprised of lead, tungsten, depleted uranium or combinations of these materials. The individual shield constructions are described in Sections 1.2.1.2.1 through 1.2.1.2.5.

1.2.1.2.1 Model 855 Shield: The 855 shield container is comprised of a depleted uranium shield secured within a steel welded housing. The shield allows for the loading of up to eight individual sources within titanium J-tubes in the shield. The sources are attached to the end of a source wire assembly and prevented from movement during transport by means of lock assemblies which secure the radioactive sources at the bottom of the eight J-tubes. The

855 shield container housing is effectively a cylinder, 10 ¾ inches (273 mm) in diameter on top of a 11 ¼ inches (286 mm) diameter base. The Model 855 is approximately 11 ¾ inches (298 mm) tall (without the eyebolt height). Copper separators are installed around all exposed surfaces of the DU to prevent any steel-uranium interaction inside the shield container. The shield is further retained in place by the use of polyurethane foam which fills the voids between the shield and the inner surfaces of the 855 steel housing. For shipment, a steel cover is bolted to the top of the shield container. The cover design creates an interference with the J-tube lock mechanisms which further prevents source release during shipment. The 855 shield weighs a maximum of 225 lbs (102 kgs) and contains a maximum of 135 lbs (61 kgs) of depleted uranium.

1.2.1.2.2 Model 3056 Shield: The 3056 shield container is effectively a lead shield pot with steel bracing around the pot diameter and bottom. The 3056 is approximately 6 ¼ inches (159 mm) in diameter (not including the handle bosses), and 10.4 inches (264 mm) tall. The shield container (with the handle bosses) is 7.7 inches in diameter. The shield incorporates two lifting handles 180 degrees apart on the sides. The 3056 includes a cover which protects the source tubes and caps during shipment. Primary radiation shielding is provided by a lead pot modified to use a depleted uranium inner core shield. The inner core shield provides additional, high efficiency radiation shielding in close proximity to the source positions in transport. The shielding components are clamped together by means of a steel cradle or sheath and a flange on the upper insert. Source location and retention is provided by a fabricated insert containing ten J-tubes and the use of source tube caps. Each source is secured close to the center of the shield by means of the attached flexible source holder within the J-tube and is closed by the tube cap. The 3056 shield weighs a maximum of 114 lbs (52 kgs).

1.2.1.2.3 Model 3015 Shield: The 3015 shield container is effectively a lead pot with a steel casement enclosing the sides and top surfaces of the container. The 3015 is approximately 6 inches (155 mm) in diameter, and 10.1 inches (257 mm) tall. The shield incorporates two, side lifting handles 180 degrees apart. The 3015 includes a cover which protects the shield plug assemblies and source cavity during shipment. Primary radiation shielding is provided by a lead pot. The minimum thickness of the lead pot is approximately 1.9 inches (47 mm). The source cavity also uses a depleted uranium insert with an added tungsten insert inside of the lead shield for added dose reduction. The basic tungsten insert design has minimum side and bottom wall thickness of 0.1 inches (2.54 mm). The tungsten insert can be modified, if desired, to provide cavities for holding individual sources or to increase the wall thickness. However, the minimum insert design will comply with the specifications on the drawings included in Section 1.4. Source location and retention is provided by a shield plug assembly held in place by the shield cover lid. The 3015 shield weighs a maximum of 114 lbs (52 kgs).

1.2.1.2.4 Model 3018 Shield: The 3018 shield container is effectively a lead pot with a steel casement enclosing the sides and top surfaces of the container. The 3018 is approximately 6 inches (156 mm) in diameter, and 11 inches (279 mm) tall. The 3018 includes a cover which protects the source tube caps during shipment. Primary radiation shielding is provided by a lead pot with a depleted uranium insert. The minimum thickness of the lead pot is approximately 1.9 inches (47 mm). The main insert containing the J-tubes is comprised of lead with a bottom core which provides ½ inch (14 mm) of depleted uranium surrounding the source location. Source location and retention is provided by a fabricated insert containing four J-tubes and the use of source tube caps. Each source is secured, close to the center of the shield, by means of the attached flexible source holder within the J-tube and is closed by the tube cap. The 3018 shield weighs a maximum of 114 lbs (52 kgs).

1.2.1.2.5 Model 3078 Shield: The 3078 shield container is effectively a welded steel cylinder, 6.1 inches (155 mm) in diameter, and approximately 8.4 inches (213 mm) tall. The shield incorporates two side lifting handles 180 degrees apart. The 3078 includes a cover which protects the source cavity during shipment. Shielding is provided by a depleted uranium pot and shield plug. The minimum thickness of the depleted uranium pot is 2.3 inches (58 mm). Source location and retention is provided by a shield plug held in place by the shield cover lid. The design also allows for the use of an optional steel or aluminum can within the source cavity. This can provides negligible shielding and is intended only to facilitate source insertion and removal from the shield cavity. The 3078 shield weighs a maximum of 150 lbs (68 kgs).

1.2.1.2.6 Model 1911 Shield: The 1911 shield container is effectively a welded steel cylinder, 8 inches (203 mm) in diameter, and 8 ¾ inches (222 mm) tall (without the eyebolt or lid bolt heights). The maximum weight of the 1911 shield is 184 lbs (84 kgs). The shield lid is secured to the body by four M8 x 25 mm hex head stainless steel bolts and M8 stainless steel washers. With the shield lid secured to the body by the M8 bolts/washers, the 1911 is designed to be lifted by an M10 steel eyebolt which is threaded onto a recess in the shield lid. The eyebolt is removed after loading of the 1911 into the 976F cork lined drum and during transportation. The shield lid protects the source cavity and removable shielding during shipment.

The main shielding for the 1911 is provided by a lead shield body encased by a welded steel cylinder. The minimum thickness of the primary lead shielding pot is 2 ½ inches (64 mm). Source location and retention is provided by a shield plug and insert assembly. The shield plug, in combination with the lower shield insert, provides the inner containment for the sources inside the outer lead shield body when held in place by the shield cover lid. The design incorporates one of three insert configurations within the source cavity to allow for different source loading applications within the 1911 shield. Approval of all three shield insert configurations is for the same Ir-192 radioactive capacity of 1,000 Ci. The inner shield insert combinations are as follows:

1.2.1.2.6.a. Model 976F (Inner Container 1911) Shield Insert Configuration Option 1: Depleted Uranium Plug/Insert Configuration (P1992 and P1991)

The shield insert is approximately 6.7 lbs (3 kg) of depleted uranium encased in 1/16 inch (1.5 mm) thick stainless steel. The minimum thickness of the depleted uranium insert is 0.64 inches (16 mm). The depleted uranium insert provides a cylindrical source cavity measuring 0.92 inches (23 mm) in diameter and 1.67 inches (42 mm) deep. The design also allows for the use of an optional steel or aluminum holder within the source cavity. These holders provide negligible shielding and are intended only to consolidate and facilitate source insertion and removal from the shield cavity.

The shield plug for this configuration is approximately 5.3 lbs (2.4 kg) of depleted uranium encased in 1/16 inch (1.5 mm) thick stainless steel. The shield plug contains a minimum of 2.8 inches (71 mm) of depleted uranium directly above the shield cavity. The shield plug also incorporates a metal bail for use in lifting the plug from the container. This bail provides negligible shielding and is intended only to facilitate insertion and removal of the plug from the shield.

1.2.1.2.6.b. Model 976F (Inner Container 1911) Shield Insert Configuration Option 2: Lead Plug/Insert Configuration (L1992 and L1991)

The shield insert is approximately 2 lbs (0.9 kg) of lead. The minimum thickness of the lead insert is ½ inch (13 mm). The lead insert incorporates a brass top piece and the insert is shown in greater detail in Section 1.4.

The shield plug for this configuration is approximately 5.5 lbs (2.5 kg) of lead. The shield plug contains a minimum of 3.3 inches (85 mm) of lead directly above the shield cavity. The shield plug has a recessed cylindrical area which increases the shield cavity and measures 1.4 inches (36 mm) in diameter by 0.36 inches (9 mm) deep. The shield plug also incorporates a metal bail for use in lifting the plug from the container. This bail provides negligible shielding and is intended only to facilitate insertion and removal of the plug from the shield.

1.2.1.2.6.c. Model 976F (Inner Container 1911) Shield Insert Configuration Option 3: Tungsten Plug/Insert Configuration (T1992 and T1991)

The shield insert is approximately 4.6 lbs (2 kg) of tungsten. The minimum thickness of tungsten in the insert is 0.64 inches (16 mm). The tungsten insert provides a cylindrical source cavity measuring 0.91 inches (23 mm) in diameter and 2.7 inches (68 mm) deep. The design also allows for the use of an optional steel or aluminum holder within the source cavity. These holders provide negligible shielding and are intended only to facilitate source insertion and removal from the shield cavity.

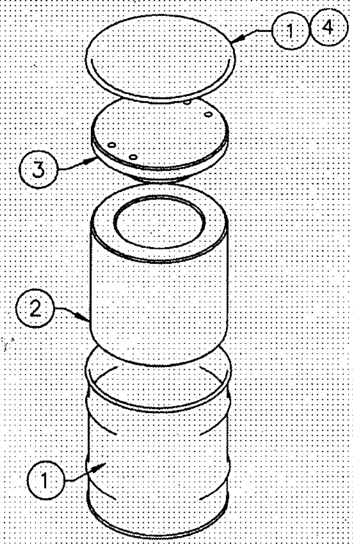
The shield plug for this configuration is approximately 7.1 lbs (3.2 kg) of tungsten. The shield plug contains a minimum of 2.1 inches (54 mm) of tungsten directly above the shield cavity. The shield plug also incorporates a metal bail for use in lifting the plug from the container. This bail provides negligible shielding and is intended only to facilitate insertion and removal of the plug from the shield.

1.2.1.2 Package Assembly Components (without shield containers): All versions of the 976 packages use a stainless steel drum with cork liner inserts to provide shield stability during transport. The drum provides structural strength to the overall package while the cork serves to limit inner shield movement during transport as well as act as a thermal insulator in case of fire. The components are described in Section 1.2.1.3.

1.2.1.3 Drum: The outer drum for this package is a 20 gallon capacity drum with 16 gauge, 0.06 inches (1.5 mm) thick stainless steel walls. The drum includes a removable lid which is secured in place using a lid closure band and four 3/8-16 x ¾ inch (19 mm) long stainless steel lid bolts. The lid bolts are inserted through four ½ inch (12.7 mm) diameter holes spaced equidistantly around the diameter of the drum. The holes are located 1 ¼ inches (32 mm) down from the top of the drum. The drum lid has four stainless steel blocks measuring 1 inch (25 mm) by 1 inch (25 mm) by ¾ inch (19 mm) tall. The steel blocks are welded on all four sides to the underside of the lid. The block welds are ¾ inch (19 mm) long on each side. The blocks are also drilled and tapped to accept a 3/8-16 bolt. The drum measures 19 ¾ inches (502 mm) in diameter and is 21 ¼ inches (540 mm) tall when assembled.

1.2.1.3.1 Drum Clamp Assembly: The drum uses a stainless steel lid closure band and M8 x 1.25, 130 mm long stainless steel hex head bolt and nut to secure the lid to the drum base. This clamp bolt is torqued to 10 ft-lbs (+2, -0 ft-lbs) prior to transport. This torque is equivalent to a 0.75-1.25 inch (19-32 mm) gap between the lid closure band sides.

1.2.1.3.2 Cork Liner Inserts: All 976 Series packages use the same basic cork liners designed for transport of the Model 855 shield configuration of the 976A package. This is comprised of a combination bottom/side liner and a top liner which fits into the bottom liner after insertion of the shield container (see Figure 1.2a).



ITEM	QTY.	DESCRIPTION
4	1	CLAMP BAND
3	1	TOP CORK INSERT
2	1	BOTTOM CORK INSERT
1	1	20 GAL. BARREL

Figure 1.2a: Exploded View of Model 976A Drum and Basic Inserts

For the Model 976B through 976F style containers, additional, secondary inserts may be used to limit movement of the shield containers during transport (see descriptive assembly drawings for details).

1.2.2 Containment System

(Reference:

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

There are two basic methods for securing the sources in the Model 976 Series shields. The methods are described in Table 1.2c. In all cases the inner shield containers are loaded into the cork inserts within the drum, the drum lid is attached by means of a lid closure band and four drum bolts securing the drum lid to the base. The lid clamp band bolt is seal wired with a tamper indicator seal.

Table 1.2c: Shield Container Containment Descriptions

Shield Identification	Containment Description
855, 3056, 3018	Source location and retention is provided by J-tubes in the shield and the use of either a locking mechanism on the source wire or source tube caps.
3015, 3078, 1911	Source location and retention is provided by a shield plug held in place by the shield cover lid. The shield cover lid is attached to the shield base by bolts.

1.2.3 Contents

(Reference:

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

The Model 976 Series transport packages are designed to transport special form capsules containing the isotopes listed in Table 1.2a. Additional information for the contents is provided in Table 1.2d. The maximum decay heat for Ir-192 in table 1.2d has been adjusted to account for content activity of the source. Actual content to output activity varies based on the capsule configuration as well as variations in isotope self-absorption. A factor of 2.3 was used to convert output activity to content activity as this factor reflects the worst case variation for Ir-192 sources transported in these packages. The source capsules are loaded into the transport package shield and secured according to the procedure for that shield container (see Section 7).

The maximum weight of the contents for the shield containers is also listed in Table 1.2d. The content weight values are based on either the actual source assembly weights (for the 855, 3018 and 3056 containers), or calculated based on the package capacity and the lowest specific activity of Ir-192 (200 Ci/gram) used in source production (for the 3078, 1911 and 3015 containers).

Note: Ir-192 of higher specific activity can be used but this would produce sources with lower total mass of the contents. Material density for Se-75 and Yb-169 are less than Ir-192, therefore the maximum content weight values listed in the Table 1.2d are the maximum content masses based on the heaviest material content which is Ir-192.

Table 1.2d: Isotope Information Permitted in the Model 976 Series Packages

Package ID	Isotope	Activity[1]	Capsule Form[2]	Chemical/Physical Form	Maximum Content Weight (grams)	Maximum Decay Heat[3]
976A	Ir-192	1,000 Ci	Special Form	Metal	176	20 Watts
	Se-75	1,000 Ci				5 Watts
	Yb-169	865 Ci				5 Watts
976B	Ir-192	350 Ci	Special Form	Metal	1.2	7 Watts
	Se-75	350 Ci				2 Watts
	Yb-169	350 Ci				2 Watts
976C	Ir-192	1,250 Ci	Special Form	Metal	220	25 Watts
	Se-75	1,250 Ci				6.4 Watts
	Yb-169	1,000 Ci				5.4 Watts
976D	Ir-192	500 Ci	Special Form	Metal	88	10 Watts
	Se-75	500 Ci				2.5 Watts
	Yb-169	500 Ci				3 Watts
976E	Ir-192	1,000 Ci	Special Form	Metal	3.3	20 Watts
	Se-75	1,000 Ci				5 Watts
	Yb-169	1,000 Ci				5.4 Watts
976F	Ir-192	1,000 Ci	Special Form	Metal	3.3	20 Watts
	Se-75	1,000 Ci				5 Watts
	Yb-169	1,000 Ci				5.4 Watts

[1] For Iridium-192, the maximum capacity is based on the output curies which are determined by measuring the source output at 1 meter and expressing its activity in curies derived from the following: 0.48 R/hr-Ci Iridium-192 at 1 meter. (Ref: American National Standard N432-1980, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography."). For Selenium-75 and Yb-169 the maximum capacity is based on the content curies contained in the radioactive source(s).

[2] Special Form is defined in 10 CFR 71, 49 CFR 173, IAEA No. TS-R-1 (ST-1, Revised) 1996 Edition (Revised).

[3] Maximum decay heat for Ir-192 is calculated by correcting the output activity to content activity. A factor of 2.3 is used to account for source capsule and self-absorption in this conversion.

1.2.4 Operational Features

These packages do not involve complex containment systems for source securement. The sources for these packages are all special form, welded capsules. The capsules may or may not be attached to flexible handling wires. Sources attached to flexible wires are held in place either by lock mechanism or source tube caps installed after the source wires are inserted into the shield J-tubes. Sources inserted into a cavity style shield container are held in place within the shield by means of a shield plug assembly and/or cover secured to the shield base. All shield containers are installed within cork liners in the 976 drum assembly and the drum lid is secured to the container by means of a bolted, seal wired lid closure band and four lid drum bolts.

1.3 General Requirements for All Packages

1.3.1 Minimum Package Size

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

The transport package exceeds the minimum size requirements since it is 19 ¾ inches (502 mm) in diameter by 21¼ inches (540 mm) tall.

1.3.2 Tamper-Indicating Feature

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

The Model 976 Series packages incorporates a steel seal wire attached to the lid closure band and lid closure band bolt. This seal wire is not readily breakable, therefore if it is broken during transport, it serves as evidence of possible unauthorized access to the contents.

1.4 **Appendix:** Drawings of the Model 976 Series transport packages.

Section 1.4 Appendix: Drawings of the Model 976 Series transport packages.

Section 2 - STRUCTURAL EVALUATION

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

2.1 Description of Structural Design

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

2.1.1 Discussion

The Model 976 Series transport packages are described in Section 1.2, "Package Description."

2.1.2 Design Criteria

The Model 976 Series transport packages are designed to comply with the requirements for Type B(U) packaging as prescribed by 10 CFR 71 and IAEA No. TS-R-1 (ST-1, Revised) 1996 Edition (Revised). All design criteria are evaluated by a straightforward application of the appropriate section of 10 CFR 71 or IAEA No. TS-R-1 (ST-1, Revised) 1996 Edition (Revised).

Some shields containers incorporated in the package were designed under previously approved QA programs, either in the USA under QSA Global, Inc. or its predecessors in the United Kingdom by competent authority under Nycomed Amersham plc. or its predecessors.

2.1.3 Weight and Centers of Gravity

The transport package weight varies from 190 lbs (86 kg) up to 300 lb (150 kg). The shipping cask weight varies from 114 lbs (52 kg) up to 225 lbs (102 kg). The center of gravity for all Model 976 Series transport packages is indicated on the drawings provided in Section 1.4.

2.1.4 Identification of Codes and Standards for Package Design

2.1.4.1 <u>Package Design</u>

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 976 Series 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 as identified in Section 1.1.

2.1.4.2 Fabrication & Assembly

All drum and cork insert 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 AWS, ASME or British Standards appropriate to the materials and designs fabricated. All safety critical hardware meets ASME-B18 standards. All external fabrication deemed critical to safety is either verified to equivalent in-house standards or dedicated as appropriate for use prior to release as part of this transport package.

Some shields containers incorporated in the package were designed under previously approved QA programs, either in the USA under QSA Global, Inc. or its predecessors in the United Kingdom by competent authority under Nycomed Amersham plc. or its predecessors. Prior to the use of these shield containers as part of the Model 976 Series transport package, they are evaluated under the QSA Global, Inc. QA program for compliance to the transport package design. Any new manufacture of shield containers to these designs will be completed under the QSA Global, Inc. Quality Assurance Plan approved by the USNRC and ISO.

2.1.4.3 Maintenance & Use

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

2.2 Materials

(Reference:

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

2.2.1 Material Properties and Specifications

Table 2.2a lists the relevant mechanical properties (at ambient temperature) of the principal materials used in the Model 976 Series transport package. The location and use of these materials is shown on the drawings contained in Section 1.4. The reference for the table information is listed in the last column of the table.

Table 2.2a: Mechanical Properties of Principal Transport Package Materials

Material	Tensile Strength	Yield Strength	Source
Depleted Uranium	65 ksi	30 ksi	Reference #2
Copper	25 ksi	9 ksi	Reference #3, p. 224
Steel (nominal)	53 ksi	36 ksi	Reference #1, p. 205
Stainless Steel	75 ksi	30 ksi	Reference #1, p. 854
Tungsten	142 ksi	109 ksi	www.matweb.com
Cork (minimum)	80 psi	NA	Reference #4, RMR214 Issue C
Lead ($^{96}Pb/^{4}Sb$)	3,990 psi	NA	www.matweb.com

Resource references:

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

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

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

4. AEA Technology plc. RMR 214 Issue 5, Raw Material Requirement, (RMR) Cork for Transport Containers (see Section 2.12.1).

2.2.2 Chemical, Galvanic or Other Reactions

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

The materials used in the construction of the Model 976 Series transport package are depleted uranium metal, steel (carbon and stainless), tungsten, lead, copper, polyurethane foam and cork. In some shield container designs, copper separators were used between steel/uranium interfaces to reduce the possible formation of a eutectic during the Hypothetical Accident Conditions thermal scenario defined by 10 CFR 71.73(c)(4). In other constructions where steel/depleted uranium interfaces exist, the steel forms a welded seal around the depleted uranium surfaces which prevents contact by air needed to create the possible formation of a eutectic alloy. In some of the shield container designs there are steel/uranium interfaces without full enclosure or copper separation. The possibility of the formation of a steel/uranium eutectic alloy at temperatures below the melting temperatures of the individual metals has been considered. 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. The cork and steel interfaces will not cause any chemical, galvanic or other adverse reactions due to contact during transport. With these container constructions, there will be no significant chemical or galvanic reaction between package components during normal or hypothetical accident conditions of transport.

2.2.3 Effects of Radiation on Materials

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

Lead, depleted uranium, tungsten, steel, polyurethane foam and cork have been used in transport packaging for decades without degradation of the package performance over time. The cork used in the drum liner inserts has been used in Type B transport packages in the United Kingdom for decades with no degradation in the material integrity over time due to irradiation from package contents.

2.3 Fabrication and Examination

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

2.3.1 Fabrication

Drum and cork inserts are procured, manufactured and inspected for use under QSA Global, Inc. NRC approved QA Program Number 0040. Existing shield containers were either originally manufactured by QSA Global, Inc. (or its predecessors) in the USA or the United Kingdom. All shield containers will be evaluated and documented for compliance to the drawings provided in Section 1.4 prior to initial use of the shield container as part of a Model 976 Series transport package.

2.3.2 Examination

Section 8 describes the acceptance testing and routine maintenance requirements for shield containers and package components used on the Model 976 Series packages.

2.4 Lifting and Tie-Down Standards for All Packages

2.4.1 Lifting Devices

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

The Model 976 Series packages are designed to be lifted by the base using a hand truck or other suitable mechanical means. For this analysis, the base is assumed to be a flat, circular plate 19 ¼ inches (502 mm) in diameter and 0.06 inches (1.5 mm) thick, supported about its outer edge. We take the supporting cylinder (the walls and bottom welded rim of the drum) to be essentially rigid for the magnitude of stresses encountered here. Any lifting would span all edges of the drum and thus allow the bottom to be supported and suspended by the edges. As such, the maximum axial stress on the base is:

$$\sigma_{max} = k\, wr^2/t^2$$

Where:

w	=	The pressure of the transport package weight (136 kg (300 lb)) distributed load over the base. = 300 lb/291 in^2 = 1.031 psi
t	=	The thickness of the base plate 0.06 inches (1.5 mm)
r	=	The radius of the base plate 9.6 inches (244 mm)
k	=	A tabulated factor for this case of flat plate. [1] = 0.75

[1] - Marks Handbook, 9th edition, pp 5-52 – 5-53, Case 2

Therefore, the stress generated in the base is 19,800 psi. With a Safety Factor of 3 applied, the maximum stress in the drum base is 59,400 psi. This is below the ultimate tensile strength of the stainless steel base which is 75,000 psi.

Calculation of the maximum deflection of the base and the bending stress at the junction of the base plate and side walls of the drum is assessed as follows. The base plate of the drum has a rolled rim that is welded to the drum wall. The inside diameter of the drum is 18 ⅛ inches (460 mm) and the drum material is 0.06 inches (1.5 mm) thick stainless steel. Conservatively the weight of the package is supported by the base plate. The maximum package weight is 300 lbs (136 kgs). The following calculation assumes the load is evenly distributed over the entire base plate. The following calculations in this Section are based on information contained in "Formulas for Stress and Strain", Fifth Edition, Raymond J. Roark and Warren C. Young, McGraw Hill Book Company, 1975.

$$D = \text{Plate Constant} = E\, t^3 / 12(1-\nu^2)$$

Where:

E	=	Modulus of elasticity of stainless steel 3 x 10^7 lb/in^2
t	=	The thickness of the base plate 0.06 inches (1.5 mm)
ν	=	Poisson ration = 0.285

This calculates to give a value for D of 588 lb-in.

The vertical deflection, y_c, is calculated as follows:

$$y_c = -q\, a^4 / 64\, D$$

Where:

a = base plate radius = 9.06 inches
q = pressure on the base plate = 300 lbs/πa^2 = 1.16 lb/in^2

Therefore, y_c = -0.208 inches

The moment at the center M_c is calculated as follows:

$$M_c = q\, a^2\, (1 + D)/16 = 7.65 \text{ in-lb/in}$$

Producing a bending stress $\sigma = 6\,(M_c/t^2)$ which equals 12,750 psi. Combining the axial and bending stresses produces a σ_c as follows:

$$\sigma_c = \{\, \sigma_{axial}{}^2 + \sigma_{bending}{}^2 \,\}^{1/2} = \{\, (19{,}800 \text{ psi})^2 + (12{,}750 \text{ psi})^2 \,\}^{1/2} = 23{,}550 \text{ psi}$$

With a safety factor of 3 applied, the maximum combined axial and bending stress on the drum base is 70,650 psi which is below the ultimate tensile strength of the stainless steel base of 75,000 psi. Therefore the package meets the requirements of 10 CFR 71.45(a).

2.4.2 Tie-Down Devices

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

The Model 976 Series packages have no tie down attachments. The package can be blocked and braced according to standard transportation practices.

2.5 General Considerations

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

2.5.1 Evaluation by Test

Evaluations by direct testing are documented in Test Plan 90 Report and Test Plan 163 Report which are contained in Sections 2.12.2 and 2.12.3 respectively.

2.5.2 Evaluation by Analysis

Evaluations by analysis are described in the section they apply to in this Safety Analysis Report or when applicable in Test Plan 90 Report and Test Plan 163 Report contained in Sections 2.12.2 and 2.12.3 respectively.

2.6 Normal Conditions of Transport

2.6.1 Heat

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

The heat source for the Model 976 Series transport packages are listed in Table 1.2d. Iridium-192, generates approximately 8.6 milliwatts per Curie based on assuming a decay energy of 1.46 MeV/decay. The thermal evaluation for the heat test is described in Section 3.

2.6.1.1 Summary of Pressures and Temperatures

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

Table 2.6.1.a: Summary Temperatures Normal Transport

Temperature Condition	Model 976	Comments
Insolation (38°C in full sun)	90.3°C (195°F)	Section 3.4.1.1.
Decay Heating (38°C in shade)	42.7°C (109°F)	Section 3.4.1.2

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

2.6.1.2 Differential Thermal Expansion

Any thermal expansion encountered during Normal Transport will be insignificant with respect to the manufacturing tolerances of the package. For example:

Expansion of the outer drum circumference is approximated by:

$$E = \pi D \alpha \Delta T$$

Where: $D =$ Diameter of the drum at top
$\alpha =$ Coefficient of Thermal expansion
$\Delta T =$ Cold temperature differential (from -40°F to 68°F)
$\Delta T =$ Hot temperature differential (from 68°F to 155°F)

Substituting we get: $E = \pi\ (19\frac{1}{4}in)(9.9\mu in/in°F)(108°F) = 0.06$ in (cold)
$E = \pi\ (19\frac{1}{4}in)(9.9\mu in/in°F)(87°F) = 0.05$ in (hot)

This translates to a diameter change of ± 0.02 inches. Manufacturing tolerance on this component is ± ¼ inch. Further, the lid closure band and lid cover will expand at approximately the same rate thus maintaining the security of the package.

Expansion of the cork circumference is approximated by:

$$E = \pi D \alpha \Delta T$$

Where: D = Diameter of the cork
α = Coefficient of Thermal expansion
ΔT = Cold temperature differential (from -40°F to 68°F)
ΔT = Hot temperature differential (from 68°F to 155°F)

Substituting we get: $E = \pi\ (18\ in)(100\ \mu in/in°F)(108°F) = 0.61$ in (cold)
$E = \pi\ (18\ in)(100\ \mu in/in°F)(87°F) = 0.49$ in (hot)

This translates to a diameter change of +0.16/-0.19 inches. Manufacturing tolerance on this component is ± ¼ inch. All other drum and cork insert components have similar tolerances. Any expansion in this temperature range will be well within these tolerances.

Tolerances between the shield inserts and the cork are even greater. As such, no interference, even with the shields will occur.

2.6.1.3 Stress Calculations

As shown in Section 2.6.1.2, thermal differentials will have no adverse effect of the interfaces between the outer drum, cork inserts and shield inserts. Mechanical loads at the maximum weight of the series (300 lbs.) are well distributed across the package bottom and are small compared to the yield strength of the steel (30,000 psi – See Table 2.2a).

Inner diameter of drum = 18 1/8 inches
Area of drum bottom = 256 in^2
Stress on drum bottom = 300 lbs/256 in^2 = 1.2 psi

Stresses will develop within the gasketed cavities in the shields. The most onerous case, Model 976A with the Model 855 shield container, is described below. As this cavity is the largest, the increased pressure will exert the most force on the cover. Further, the Model 855 is carbon steel and slightly less robust than the stainless steel components of the other shields. A perfect seal and no escaping gasses is also assumed.

Force is estimated by: $F = (\pi D^2/4)P$

Where: D = Diameter of the shield cover over the void = 10 ¾ in
P = Pressure induced by the thermal gradient = 7 psi (from Section 3.4.2.)

Therefore: $F = \pi(10\ \frac{3}{4}\ \text{in})^2/4(7\ \text{psi}) = 635\ \text{lbf}$

The cover is held by eight (8) 3/8-16 stainless steel bolts. This imparts a force of 79 lbf in each bolt. However, if all the stress is assumed to be taken by only one bolt, then the stress in that bolt equals:

$$S = Fi/A$$

Where: Fi = Force in each bolt
A = Stress area of the bolt (Machinery's Handbook, 24th Edition, by Industrial Press Inc., New York, NY for 3/8 inch bolt) = 0.0775 in^2

Solving for the bolt stress produces: $S = 635\ \text{lbf}(0.0775\ \text{in}^2) = 8{,}152\ \text{psi}$

This value is well below the tensile strength value for an ungraded stainless steel bolt (75,000 psi nominal). All other shield containers would have lower stresses as they have significantly smaller stress areas.

2.6.1.4 Comparison with Allowable Stresses

All stresses calculated in Section 2.6.1 are well below strengths for the materials of construction. Further, the Model 976 Series package was fully tested and passed under Normal Conditions of transport. It is therefore concluded that the Model 976 Series package will satisfy the performance requirements specified by the regulations.

2.6.2 Cold

(Reference:

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

The carbon steel components of the Model 976 Series transport packages are susceptible to brittle fracture at low temperature. The transport package successfully met Type B(U)-96 Normal Condition and Hypothetical Accident ConditionTransport Tests requirements at temperatures below -40°C (-40°F), the minimum specified in the regulations. This testing is described under Test Plan 90 Report and Test Plan 163 Report (Sections 2.12.2 and 2.12.3 respectively). Thus, it is concluded that the Model 976 Series transport packages will withstand the normal transport cold condition.

2.6.3 Reduced External Pressure

(Reference:

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

Other than some of the shield containers, the Model 976 Series transport packages are open to the atmosphere and contains no components which could create a differential pressure relative to atmospheric conditions or components within the package. From Section 2.6.1.3, the maximum force generated within the cavities of the shields due to temperature gradients is 635 lbf. at 7 psi internal. If the 8.7 psi were then superimposed on this pressure, the stress, if taken through a single bolt, would be 18,300 psi, still significantly below the bolt's yield strength. Therefore, the reduced external pressure requirements of 3.5 psi in 10 CFR, 8.7 psi (60 kPa) in 49 CFR and IAEA will not adversely affect the package containment.

2.6.4 Increased External Pressure

(Reference:

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

Other than some of the shield containers, the Model 976 Series transport packages are open to the atmosphere and contain no components which could create a differential pressure relative to atmospheric conditions.

The shield container that would be the most affected is the Model 855. This shield had a large plate above the gasketed cavity. If we use the same logic case of a uniformly loaded plate supported on all sides (as in Section 2.4.1) we find:

$$\sigma_{max} = k\, wr^2/t^2$$

Where:

w	=	The pressure (20 psi).
t	=	The thickness of the top plate (5/16 inches)
r	=	The radius of the top plate (5 3/8 inches)

k = A factor for this case of flat plate (0.75). [1]

[1] - Marks Handbook, 9th edition, pp 5-52 – 5-53

Therefore, the stress generated in the top is 4,200 psi. This is below the yield strength of the steel top (36,000 psi). All other shield containers have either thicker covers or smaller areas. Therefore, the increased external pressure requirements of 20 psi in 10 CFR 71 will not adversely affect the package containment.

2.6.5 Vibration

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

Analysis of the 976A container will be used to justify the remaining 976 configurations as this assembly is the heaviest. The package is delivered for transport as an assembled unit. Each package is considered to be self contained as far as vibration protection is concerned. Consequently, the analysis for normal vibration during shipment in a standard or self-secured trailer is considered for an individual package.

Having the packaging mass, insulation thickness, bearing area, insulation stiffness and vibration characteristics of the trailer (self-secure trailer used in this analysis), the mean square response of the container and its contents can be determined.

The power spectral densities (PSD) for the safe-secure trailer are shown at various frequencies (Figure C1). Vertical PSD dominates the horizontal PSD. (See Appendix C of Ref. 1, listed at the end of this section.).

The 855 in the 976A package is supported by 2 inches (50.8 mm) of cork. The shield lower bearing area is 90.76 in^2 based on a 10 ¾ inch diameter. The cork density of the package is 17 lbs/ft^3 (272 kg/m^3). The Young's modulus (E) of this cork (by interpolation) is 34.8 kg/cm^2 or 495 psi (See page 4 of Ref. 2 listed at the end of this section).

The spring rate of the cork in the vertical direction can be calculated as follows (See Appendix C of Ref. 1):

$$K = \frac{AE}{L} = \frac{(90.76\ in^2)(495\ psi)}{2\ in} = 22{,}463\ \ lb/in$$

Where:
A = lower bearing area = 90.76 in^2
E = Young's Modulus for cork = 495 psi
L = support thickness of cork = 2 inches

The natural frequency of the shield and insulation spring-mass system can be calculated as follows:

$$f = \left(\frac{1}{2\Pi}\right)\left(\frac{K\,g}{W}\right)^{1/2}$$

Where: K = cork vertical spring rate = 22,463 lb/in = 269,556 lb/ft
g = gravitational acceleration = 32.2 ft/sec^2
W = net package weight = 267 lbs

The net package weight is obtained by subtracting the outer drum weight of 33 lbs from the package weight of 300 lbs, producing a net package weight of 267 lbs.

From this the natural frequency of the spring-mass system is calculated as:

$$f = \left(\frac{1}{2\Pi}\right)\left(\frac{(269{,}556\ in/ft)\,(32.2\ ft/\sec^2)}{267\ lbs}\right)^{1/2} = 28.7\ Hz$$

Conservatively estimated damping as 10% of critical, an amplification factor (Q) can be calculated:

$$Q = \left[\left(1 - r^2\right)^2 + (2rd)^2\right]^{-1/2}$$

Where: d = damping coefficient = 0.1
r = ω/ω_r, where worst case $\omega = \omega_r$, and r = 1

This produces an amplification factor of Q = 5.

The package is in resonance at 28.7 Hz and vibration above this frequency will be mechanically filtrated (isolated). Neglecting any amplification at resonance and using Fig. C1 from Reference 1 for the PSD at this frequency, the broadband, mean square g-level of input excitation on the packaging being transported in a safe-secure trailer can be calculated. The mean square acceleration response of an oscillator (in this case the package) can be defined as:

$$\text{Mean Square} = \frac{\pi\, f\, Q\,(PSD)}{2}$$

Then the Mean Square at 28.7 Hz is $\frac{\pi\,(28.7\ Hz)(5)(0.00112)g^2}{2} = 0.252\ g^2$

Therefore, the Root Mean Square at 28.7 Hz is 0.502 g.

Maximum vibration is three to four times the root mean square, or about 1.51 g to 2.01 g. This magnitude of vibration will have little effect on the package as the package has demonstrated it can withstand greater forces under the 9 m Hypothetical Accident drop testing. (This testing is described in Test Plan 90 Report and Test Plan 163 Report Sections 2.12.2 and 2.12.3 respectively).

The largest vibration envisioned for the transport package would be due to a truck wheel or tire out of balance due to manufacture, mounting, under-inflation or damage. A truck tire with a 42 inch or 3.5 foot diameter wheel has a circumference of 11 feet. When traveling between 0 and 70 MPH, the wheels will rotate between 0 and 9.3 revolutions per second, producing vibrations between 0 and 9.3 Hz. All of these frequencies are below the package resonance of 28.7 Hz. For comparison 30 inch train wheels travelling at 0 to 70 MPH create frequencies of 0 to 14.4 Hz. Still below the packages resonance.

Other components of truck vibration due to engine or driveshaft imbalances have frequencies up to and above the resonant frequency of the package. Normal amplitudes would be very small, however, especially on large trucks where these vibrations are separated from the cargo by the trailer hitch. If the amplitude of these vibrations were large, the truck would most likely break down. Airplane turbine vibrations are very high frequency, above the primary resonant frequency of the package, and very low in amplitude. (For reference 1 MPH = 1 mile/hour * 5280 feet/mile * 1 hour/3600 sec = 1.47 feet/sec).

Further, the inner shield designs used in the Model 976 Series transport packaging have been used in Type B shipments on trucks, trains and planes for decades without vibration induced failure of any shield fasteners. Shield fasteners on these containers were functionally inspected prior to use to check thread condition and engagement and general fastener condition (e.g., not bent or damaged). The same or similar fasteners on the shield containers and the outer drum assembly have been used for many transport shipments under these inspection conditions without failure of any kind in use (See Section 1.2.1 for prior transport approval references).

The steel drum and lid closure band are established designs used routinely in transport which can reasonably be expected to withstand the vibration normally incident to transport. The Model 976 Series packages also incorporate an additional level of drum securement by means of four drum lid bolts that further secures the package containment during transport. The cork used in the drum liner inserts has also been used in Type B transport packages in the United Kingdom for decades again with no degradation in the material integrity over time due to vibration during shipment. It is therefore concluded that the Model 976 Series packages will withstand vibration normally incident to transport.

References:

1. Safety Guide 100. "The Guide for Packaging and Offsite Transportation of Nuclear Components, Special Assemblies, and Radioactive Materials Associated with the Nuclear Explosives and Weapons Safety Program." Martin Marietta Energy Systems, Inc., Oak Ridge, TN 37830, November 7, 1994.

2. Antonio P.O. Carvalho, Ph.D, "Cork as a lightweight partition material...", Civil Engineering Department, College of Engineering, University of Porto, Portugal, 1997.

2.6.6 Water Spray

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

The Model 976 Series transport packages are constructed of water-resistant materials throughout. Therefore, the water spray test would not reduce the shielding effectiveness or structural integrity of the package.

2.6.7 Free Drop

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

The drop test pad used in the 1.2 m free drop, 9 m drop, and puncture tests consists of a monolithic concrete base 10 ft x 10 ft x 4 ft thick.(Reference drawing T10261). The approximate weight of the concrete is 88,000 lbs. A 4 ft x 4 ft x 1 in thick steel plate is anchored to this concrete slab by 4 anchor bolts embedded in high strength grout. This drop pad has been utilized in 9 m free drop tests of packages weighing up to 1,000 lbs. Before and after testing the drop pad is visually inspected for damage which could have a significant impact on package testing.

Test specimen TP90A was subjected to the 1.2 meter (4 foot) free drop as described in Test Plan 90 Report (Section 2.12.2). The orientation of the 1.2 meter (4 foot) free drop was selected because of its potential to cause significant deformation of the closure bolt assembly in an effort to open the drum. The specimen was dropped at approximately a 45° angle with the closure bolt down. The test specimen temperature was less than -40°C (-40°F). Photographs of the drop orientation are provided in the Test Plan 90 Report (Section 2.12.2).

The test specimen impacted the test pad as intended. Very little damage to the drum was noted. The bottom of the drum was scuffed and slightly bowed out. Upon disassembly, the cork liner had fractured and separated at the base. The Model 855 was undamaged.

Profile results of the Model 855 unit without the drum assembly were within regulatory limits for normal transport (see Section 5.1). The 855 is the heaviest container used in the Model 976 Series packages. Damage to test unit TP90A was determined to be more severe then would be expected for any other inner shield configuration in the package series, therefore representative of all packages in the series.

As described in Test Plan 90 Report, test specimens TP90B and TP90G were dropped at an angle onto the lid and drum edge causing them to buckle inward. The heavier test specimen (TP90B containing an 855 shield container, with a total package weight of 276 lbs) was dropped at a 45° angle. The deformation in TP90B was about 2 ¼ inches deep. The lid and drum rim buckled inward, folded together and an air gap was created. The cork was cracked by this buckling, but the shield container inside was undamaged.

As described in Test Plan 163 Report (Section 2.12.3), the Model 976 Series drum assembly was redesigned to incorporate four (4) bolt blocks welded to the underside of the lid and four (4) lid closure bolts which are inserted through clearance holes in the drum sides and secure into these bolt blocks on the lid. This modification provides a secondary safety mechanism should the lid clamp band fail under the Hypothetical Accident test conditions and provides added bracing the inside of the drum which reduces buckling deformation.

In described in Test Plan 163 Report (Section 2.12.3), all the test specimens were dropped at angles (either 17.5° or 45°) onto the lid and drum edge. The buckling observed was less than seen for the test specimens under Test Plan 90 Report (Section 2.12.2). Test specimen TP163(A) containing an 855 shield with a total test specimen weight of 298 lbs was dropped at a 45° angle. The deformation in this test specimen was 1 ¾ inches deep. The lid and drum rim were flattened together but no air gap was created. The interior cork was cracked by the drum deformation but held in place by the drum structure and the inner shield container was undamaged.

As assessed in this Section, and under Test Plan 163 Report (Section 2.12.3), the modifications to the drum securement (use of four lid closure bolts) will not adversely impact the testing results performed under Test Plan 90 Report. Therefore the Model 976 Series packages as described in the drawings contained in Section 1.4 will comply with the requirements of this section.

2.6.8 Corner Drop

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

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

2.6.9 Compression

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

The Test Plan 90 Report (Section 2.12.2) documents that the Model 976 Series transport package maintained its structural integrity and shielding effectiveness under the Normal Conditions of Transport compression test. The TP90A test specimen was subjected to a compressive load of 1,465 lbs (662 kg) for a period of 24 hours. Assessment of the package to five times the maximum package weight of 300 lbs (136 kg) is contained in Test Plan 163 Report (Section 2.12.3). This assessment demonstrates that the Model 976 Series packages will comply with the requirements of this section even at the increase package weight based on that the physical testing performed in Test Plan 90 Report. The actual compressive weight of 1,465 lbs (662 kg) and the maximum calculated compressive weight of 1,500 lbs (678 kg) are greater than 13 kPa (2 lb/in^2) multiplied by the vertically projected area of the transport package. Following the compression test in Test Plan 90 Report, no damage to the specimen was observed.

2.6.10 Penetration

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

Test specimen TP90A was subjected to a penetration test, as described in Test Plan 90 Report (Section 2.12.2). The penetration bar impacted as intended. The bar bent the closure bolt slightly and left a slight impression on the threads. No other damage was noted. There was no loss of structural integrity or reduction of shielding efficiency as a result of the impact. As assessed under Test Plan 163 Report (Section 2.12.3) the modifications to the drum securement (use of four lid closure bolts) will have no impact on the results of the testing performed under Test Plan 90 Report and therefore the Model 976 Series packages as described in the drawings contained in Section 1.4 will comply with the requirements of this section.

2.7 Hypothetical Accident Conditions of Transport

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

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

Three (3) test specimens were used to conduct the hypothetical accident tests. Each test specimen consisted of a separate drum and set of cork inserts. Two (2) Model 855 shield containers were used in two (2) separate Free Drop and Puncture tests. The third test configuration was conducted with a Model 1911 shield container. Detailed description of this testing is contained in Test Plan 163 Report (Section 2.12.3).

2.7.1 Free Drop

(Reference:

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

Justification for all test unit drop orientations are included in Test Plan 163 Report (Section 2.12.3)

2.7.1.1 End Drop

This orientation was used for some of the test samples under Test Plan 90 Report (Section 2.12.2). Results of this testing produced less damage to the package than was seen in the other drop orientations, therefore additional End Drop testing was not performed for test specimens under Test Plan 163 Report (2.12.3).

2.7.1.2 Side Drop

The side drop was not performed. In a side drop, most of the energy generated at impact is used in deforming the outer package and is not transmitted into the shield. A side drop would deform the outer drum resulting in a very slow deceleration, thus limiting the energy generated at impact and transmitted to the shield.

2.7.1.3 Corner Drop

Not Applicable. The 976 Series package is a drum which does not have corners.

2.7.1.4 Oblique Drops

This orientation was used for all of the test samples. Test samples were evaluated at two orientations, 45° impacts and 17.5° impacts. See Test Plan 163 Report (Section 2.12.3) for justification of these drop orientations.

2.7.1.5 Summary of Results

(Reference:

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

See Table 2.7.8.1 for test unit results summary.

2.7.2 Crush

(Reference:

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

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

2.7.3 Puncture

(Reference:

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

The puncture bar is a 6 inch diameter x 12 inch long, mild steel solid bar attached to a 12 inch x 12 inch x ½ inch thick mild steel base. The bar is attached to the base with a ¼ inch circumferential fillet weld (Reference drawing T10119). The puncture is attached to the drop test pad steel plate by four ½"-13 x ¾" long stainless steel bolts.

Justification for all test unit puncture orientations are included in Test Plan 163 Report (Section 2.12.3).

2.7.4 Thermal

(Reference:

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

Because no damage occurred during the Hypothetical Accident Conditions of Transport Tests that could result in oxidation of the DU shield or melting of the lead or tungsten shielding, thermal testing was not performed on the 976 Series test specimens. See Test Plan 163 Report (Section 2.12.3) for a more detailed justification and assessment of compliance for the Model 976 Series packages to these requirements.

2.7.4.1 Summary of Pressures and Temperatures

(Reference:

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

Table 2.7.4.1.a: Summary Table of Temperatures

Surface Temperature Condition	Model 976 Series Packages	Cork Inserts	Shield Containers
Fire Test During	90.3°C to 800°C[3] (195°F to 1,472°F)	109.4°C to 800° C[4] (229°F to 1,472°F)	109.4°C to 159° C[2] (229°F to 318°F)
Post-Fire (Maximum Temperature)	800°C[1] (1,472°F)	800° C[1] (1,472°F)	159°C[2] (318°F)

1 – From actual testing of similar packages. Reference Section 2.12.3 (See section 5.5 of Test Plan 163 Report)
2 – Maximum temperature based on thermal increase of 50°C seen in actual package testing (See section 5.5 of Test Plan 163 Report – Section 2.12.3).
3 – Maximum initial temperature of the package assumed to be bounded by the external surface reading of the Model 976 package in full sun (insolation).
4 – Maximum initial temperature of the cork inserts assumed to be bounded by the calculated external surface temperature of the shield container for the package.

All outer drum components are vented to atmosphere. As such, no pressure will build up in the units under Hypothetical Accident conditions. However, some of the shields do have small gasketed cavities. As noted below, none will develop sufficient internal pressure to detrimentally effect the device.

Table 2.7.4.1.b: Summary Table of Maximum Pressures

Package Configuration	Void Volume in^3	Fire Conditions 800°C (1,472°F) Pressure Developed	Comments
976A	285	14 psig[1,2]	¼" steel cover retained with (8) 3/8" bolts
976B	12	14 psig[1,2]	1/10" steel cover retained with (2) M10 bolts
976C	0	0 psig[2,3]	J-Tubes without sealed cover
976D	0	0 psig[2,3]	J-Tubes without sealed cover
976E	5	14 psig[1,2]	6/10" steel cover retained with (4) M8 bolts
976F	21	0 psig[2,3]	4 mm steel cover retained by (4) M8 bolts

1 – Pressure at 171°FC (350°F). After which the gasket will burn and allow release of any pressure.
2 – Initial temperature taken to be -40°C as a worst case scenario.
3 – No gasket to seal void, pressure equal to ambient.

2.7.4.2 Differential Thermal Expansion

Actual testing on similar packages has shown that any differential thermal expansion has no detrimental effect on the packages ability to pass the thermal testing portion of the Hypothetical Accident Conditions.

For the Model 976 package in whole, under the Nycomed Amersham plc. Test 1835, two (2) damaged drums were fire tested (See Test Plan 163 Report Appendix F as contained in Section 2.12.3 of this document). These drums are very similar in design to the Model 976). The tested drum measured 32.5 cm in diameter by 40.5 cm tall with minimum cork thickness on the bottom of 4 cm, on the top of 4.5 cm and on the sides of 5 cm. In contrast the Model 976 package measures 50 cm in diameter by 54 cm tall. The Model 976 Series packages have a minimum cork thickness, which is based on the Model 976A configuration containing the least cork material, of 5 cm on the bottom, 12.7 cm on the top and 8.3 cm on the sides. Neither of the packages tested under Test 1835 opened, burst or were otherwise compromised. Both test units easily passed.

It can be drawn from these actual testing results that thermal expansion will not have a significant effect on the Model 976 Series packages.

Expansion of the package circumference is approximated by:

$$E = \pi D \alpha \Delta T$$

Where: D = Outer Diameter of the drum at the top = 19 ¼”
α = Material Coefficient of Thermal Expansion
ΔT = Fire temperature differential (from 68°F to 1,475°F)

Substituting gives: $E = \pi(19\ \tfrac{1}{4}\text{''})(9.9\ \mu\text{in/in°F})(1{,}404\text{°F}) = 0.84$ in drum
$E = \pi(18\text{''})(100\ \mu\text{in/in°F})(1{,}404\text{°F}) = 7.9$ in cork

This translates to a diameter increase of 0.26 inches for the drum and 2.5 inches for the cork. Since the cork modulus of elasticity is more than 5,000 times less than stainless steel (0.032 Pa versus 210 GPa, from Mechanics of Materials, Fall 1999), the drum will keep the cork compressed within its volume. This was shown experimentally in Nycomed Amersham plc. Test 1835 (See Section 2.12.3 Test Plan 163 Report Appendix F). The lid closure band and lid will expand at approximately the same rate as the drum, thus maintaining the security of the package.

2.7.4.3 Stress Calculations

As was shown in Section 2.7.4.2, thermal differentials will have no detrimental effect on the interfaces between the outer drum, cork inserts and shield containers.

Stresses may develop within the gasketed cavities in the shields. The most onerous case, Model 976A with the Model 855 shield container, is described below. As this cavity is the largest, the increased pressure will exert the most force on the cover. Further, the Model 855 is carbon steel and slightly less robust than the stainless steel components of the other shields. Assuming a perfect seal and no escaping gasses, then no pressure will exist after ~350°C since the gasket will burn away and allow release of any pressure.

The force on the cover bolts is estimated by:

$$F = (\pi D^2/4)P$$

Where: D = Diameter of the shield cover over the void = 10 ¾ in
P = Pressure induced by the thermal gradient = 14 psig
(From Table 2.7.4.1.b)

Therefore: $F = \pi(10\ \tfrac{3}{4}\text{ in})^2/4(14\text{ psig}) = 1{,}470\text{ lbf}$

The cover is held by eight (8) 3/8-16 stainless steel bolts. This imparts a force of 158 lbf in each bolt. However, if all the stress is assumed to be taken by only two bolts, then the stress in those bolts equals:

$$S = F/A\#$$

Where: F = Force in each bolt
A = Stress area of the bolt = 0.0775 in^2
\# = Number of bolts = 2

Solving for the bolt stress produces:

$$S = 1{,}470\text{ lbf}/0.0775\text{ in}^2(2) = 9{,}483\text{ psi}$$

This value is well below the tensile strength value for an un-graded stainless steel bolt (75,000 psi nominal). Although the strength of stainless steel would decrease at this temperature, it would still be much greater than the induced stress. All other shield containers would have lower stresses as they have significantly smaller areas.

2.7.4.4 Comparison of Allowable Stresses

All stresses calculated in Section 2.7.4 are well below strengths for the materials of construction. Further, the Model 976 Series package was fully tested and passed under Normal and Hypothetical Accident Conditions of transport. It is therefore concluded that the Model 976 Series package will satisfy the performance requirements specified by the regulations.

2.7.5 Immersion - Fissile Material

(Reference:

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

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

2.7.6 Immersion - All Packages

(Reference:

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

Other than some of the shield containers, the Model 976 Series transport packages are open to the atmosphere and contain no other components that would create a differential pressure under immersion. All materials are impervious to water and would not be affected.

Some of the shield containers have cavities with neoprene gaskets. If the neoprene gaskets remain intact, the packages would be subjected to an increased external pressure of 21.7 psig (10 CFR) and 290 psi (IAEA). The shields will withstand this pressure without loss of structural integrity.

If a gasket fails, the cylindrical special form source (primary containment) will be vulnerable to collapse due to the required assumed pressure increases of 21.7 psig and 290 psi for the respective regulatory references. The source capsules are fabricated from Type 304 or 310 stainless steel. This analysis bounds any special form source capsule with a maximum inside radius of 0.120 inch (3.05 mm) and a minimum wall thickness based on the weld penetration of 0.009 inch (0.23 mm). From Reference 1, the external collapsing pressure for a thin walled cylinder is:

$$P_{collapse} = (t / R)(\sigma_y / (1 + (4\sigma_y / E)(R / t)^2))$$

Where:

t	=	0.009 in (Weld Thickness)
R	=	0.120 in (Inside Radius)
σ_y	=	30,000 psi (Yield Strength) (Reference 1)
E	=	28,000 kpsi (Young's Modulus) (Reference 2)

From this relationship, the minimum collapsing pressure of the source capsule is 1,277 psi, which exceeds the required external pressure increases of 21.7 psig and 290 psi for the respective regulatory references.

Resource references:

1. Young, Warren C. Roark's Formulas for Stress & Strain, Sixth Edition. McGraw-Hill: New York, 1989, p. 634.

2. Hibbeler, R.C. Mechanics of Materials. 2nd Edition, 1991.

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

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

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

2.7.8 Summary of Damage

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

Table 2.7.8.1 summarizes the results of the Normal Conditions of Transport and Hypothetical Accident testing performed on the Model 976 Series transport packages.

Table 2.7.8.1: Summary of Damages During Test Plan 163 and Applicable Portions of Test Plan 90

Specimen	Test Performed	Test Results
TP90A 855 sn 9	Compression test	No damage
	1 meter (40 inch) penetration bar closure bolt assembly	Bolt on the retaining ring was bent with witness marks on threads. No other damage.
	1.2 meter (4 foot) closure bolt assembly down. ~45° angle.	• Bottom of drum scuffed and slightly bowed out.
	Post-Drop Inspection	• Cork liner cracked and separated at the base but still held form inside the drum. • Model 855 undamaged. • Unit profiled after Type B testing. Surface and 1 meter dose rates remained within limits of 200 mR/hr at the surface and 10 mR/hr at one meter after both normal transport and hypothetical accident condition testing. (See Section 2.12.2)
TP163(A) 855 sn 8	9 meter (30 foot) drop, lid closure band assembly down. 45° angle	• Lid closure band bolt assembly was crushed. • Lid closure band bolt broken. • Drum lid and top of the drum were creased together and folded under the bolt assembly. • Side of the drum slightly flattened and the bottom ring weld dented. • Drum lid closure bolts intact.
	1 meter (40 inch) puncture, opposite side of closure bolt, 45° angle	• Popped lid closure band off drum. Lid still secured by the 4 lid closure bolts. • Some minor denting on the side of the lid and drum.
	Post-Drop Inspection	• Model 855 undamaged. • Cork cracked but still held form inside drum. • 1 meter dose rates remained within limit of 1 R/hr (See Section 2.12.3 and Section 5.1.2)
TP163(B) 855 sn 9	9 meter (30 foot) drop, lid closure band bolt assembly up. 17.5° angle	• Lid closure band was flattened. • Drum lid and top of the drum were creased together and folded under the lid closure band. • Side of the drum slightly flattened and the bottom ring weld dented. • Lid closure band bolt as well as drum lid closure bolts intact.

Specimen	Test Performed	Test Results
	1 meter (40 inch) puncture, opposite side of lid from lid closure band bolt, 17.5° angle	• Unit dropped twice to achieve desired impact point. • Two witness marks on the side of the drum opposite the bolt, near the lid closure band. • Witness mark on the lid closure band opposite bolt. • Lid closure band bolt as well as drum lid closure bolts intact.
	Post-Drop Inspection	• Model 855 undamaged. • Cork cracked but still held form inside drum. • 1 meter dose rates remained within limit of 1 R/hr (See Section 2.12.3 and Section 5.1.2).
TP163(C) 1911 sn 13	9 meter (30 foot) drop, lid closure band bolt assembly up. 17.5° angle	• Lid closure band was flattened. • Drum lid and top of drum were creased together and under the lid closure band. • Side of the drum slightly flattened and the bottom ring weld dented. • Lid closure band bolt as well as drum lid closure bolts intact.
	1 meter (40 inch) puncture, same impact point as 9 m drop, 17.5° angle	• Dent from 9 meter impact increased. • Slight dent in side of drum. • Lid closure band bolt as well as drum lid closure bolts intact.
	Post-Drop Inspection	• Model 1911 undamaged. • Cork cracked but still held form inside drum. • 1 meter dose rates remained within limit of 1 R/hr (See Section 2.12.3 and Section 5.1.2).

Based on these results and assessments for the remaining shield containers addressed in Test Plan 163 Report (see Section 2.12.3), it is concluded that the Model 976 Series transport packages maintain structural integrity and shielding effectiveness during Hypothetical Accident Conditions and Normal Conditions of Transport.

2.8 Accident Conditions for Air Transport of Plutonium

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

2.9 Accident Conditions for Fissile Material Packages for Air Transport

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

2.10 Special Form

(Reference:

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

The Model 976 Series transport packages are designed for use with a special form source capsules with a maximum inside radius 0.12 inches (3.05 mm) and a minimum wall thickness based on the weld penetration of 0.009 inches (0.23 mm). The source capsule must be qualified as Special Form radioactive material.

2.11 Fuel Rods

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

2.12 Appendix

2.12.1 AEA Technology plc. RMR 214 Issue 5, Raw Material Requirement, (RMR) Cork for Transport Containers

2.12.2 Test Plan 90 Report Revision 2 dated April 2005 (minus Appendix B-D).

2.12.3 Test Plan 163 Report Revision 1 dated April 2005 (minus Appendix C).

Section 2.12.1 Appendix: AEA Technology plc RMR 214 Issue 5, Raw Material Requirement, (RMR) Cork for Transport Containers

Section 2.12.2 Appendix: Test Plan 90 Report Revision 2 dated April 2005 (minus Appendix B-D).

Section 2.12.3 Appendix: Test Plan 163 Report Revision 1 dated April 2005 (minus Appendix C).

Section 3 - THERMAL EVALUATION

3.1 Description of Thermal Design

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

The Model 976 Series transport packages are completely passive thermal devices having no mechanical cooling system or relief valves. All cooling of the transport package is through free convection and radiation. The maximum output activity for this package is 1,250 Ci of Ir-192. Accounting for source absorption, this equals a maximum content activity of 2,875 Ci of Ir-192. The corresponding decay heat generation rate for the content activity is approximately 25 Watts (See Table 1.2d).

3.1.1 Design Features

The Model 976 is a series of transport packages based on a previously designed, tested and approved Type B(U) transport package. The package design was approved in the United Kingdom under Type B certificate GB/3605B/B(U)-85 and was revalidated by the USDOT for Type B(U) import and export under certificate USA/0592/B(U)-85. the Model 976 Series packages are described in Section 1. The United Kingdom package design was thermal tested and the results of that testing are contained in Appendix F of Test Plan 163 Report in Section 2.12.3). The thermal tested drum measured 32.5 cm in diameter by 40.5 cm tall. In comparison, the Model 976 packages measure 50 cm in diameter by 54 cm tall. Features uniquely relevant to thermal performance are detailed below.

3.1.1.1 Cork Inserts

The inserts serve as a thermal insulator during the fire test (Hypothetical Accident). Although the cork chars during the test, due to the composition of this cork, the rate of charring is relatively slow. During an actual thermal test, the maximum depth of charring was 1 inch (25 mm) (See Appendix F of Test Plan 163 Report in Section 2.12.3).

The drum configuration which was thermal tested as described in Appendix F of Test Plan 163 Report (Section 2.12.3) had a minimum cork thickness on the bottom of 4 cm, on the top of 4.5 cm and on the sides of 5 cm. In contrast the Model 976 packages have minimum cork thickness, based on the Model 976A configuration which contains the least cork material, of 5 cm on the bottom, 12.7 cm on the top and 8.3 cm on the sides.

In all dimensions, the cork present in the Model 976 packages and surrounding the inner shield containers is greater than the cork that was present in the container design that was thermal tested as described in Appendix F of Test Plan 163 Report in Section 2.12.3.

3.1.1.2 Thin Walled Stainless Steel Drum, Lid, Lid closure band and Lid Closure Bolts

The thin walls of the drum components exhibit almost no thermal gradient. During a fire test, the entire drum will very quickly heat to a uniform temperature, eliminating stresses induced by thermal differentials within the material. Further, the drum will move and flex easily, thus relieving any thermal expansion stress without rupture.

3.1.1.3 Un-gasketed Lid

Upon charring of the cork, gasses evolve. This drum design does not use a gasket under the lip of the lid. This permits these gasses to escape and not significantly increase the pressure within the package.

3.1.1.4 Shield Containers

All shield containers are retained within the drum/cork overpack assembly, thus limiting their temperature to well below the shield container material melting points. Those shield containers using depleted uranium shielding have the depleted uranium fully enclosed in a welded steel structure. This construction prevents oxidation by severely limiting oxygen from reaching the depleted uranium shield.

3.1.2 Content's Decay Heat

From Table 1.2d, a maximum of 25 Watts of energy is available to be absorbed by the package.

3.1.3 Summary Tables of Temperatures

Table 3.1.3.a: Summary Table of Temperatures

Surface Temperature Condition	Model 976 Series Packages	Cork Insets	Shield Container Models	Comments
Insolation (38°C in full sun)	90.3°C (195°F)	90.3°C to 109.4°C (195°F to 229°F)[3]	109.4°C (229°F)	Sections 3.4.1.1. & 3.5.2.4
Decay Heating (38°C in shade)	42.7°C (109°F)	42.7°C to 109.4°C (109°F to 229°F)[3]	109.4°C (229°F)	Sections 3.4.1.2 & 3.5.2.4
Fire Test During	90.3°C to 800°C[4] (195°F to 1,472°F)	109.4°C to 800°C[5] (229°F to 1,472°F)	109.4°C to 159°C[2] (229°F to 318°F)	Section 3.5.2.4
Post-Fire (Maximum Temperature)	800°C[1] (1,472°F)	800°C[1] (1,472°F)	159°C[2] (318°F)	

1 – From actual testing of similar packages. Reference Section 2.12.2 and Section 2.12.3.

2 – Maximum temperature based on thermal increase of 50°C seen in actual package testing (See section 5.5 of Test Plan 163 Report – Section 2.12.3).

3 – Temperature of Cork Inserts assumed to be bounded by the external surface reading of the Model 976 package and the exterior surface temperature calculated for the Shield Container within the package.

4 – Maximum initial temperature of the package assumed to be bounded by the external surface reading of the Model 976 package in full sun (insolation).

5 Maximum initial temperature of the cork inserts assumed to be bounded by the calculated external surface temperature of the shield container for the package.

3.1.4 Summary Tables of Maximum Pressures

All outer drum components are vented to atmosphere. As such, no pressure will build up in the units under either Normal or Hypothetical Accident conditions. However, some of the shields do have small gasketed cavities. As noted below, none will develop sufficient internal pressure to detrimentally effect the device.

Table 3.1.4.a: 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
976A	285	8 psi[2]	14 psig[1,2]	¼" welded steel sides for cavity. Cover retained with (8) 3/8" bolts
976B	12	8 psi[2]	14 psig[1,2]	¼" welded stainless steel sides for cavity. Cover retained with (2) M10 bolts
976C	0	0 psig[2]	0 psig[2,3]	J-Tubes without sealed cover
976D	0	0 psig[2]	0 psig[2,3]	J-Tubes without sealed cover
976E	5	8 psi[2]	14 psig[1,2]	¼" welded stainless steel sides for cavity. Cover retained with (4) M8 bolts
976F	21	0 psig[2,3]	0 psig[2,3]	4 mm steel cover retained by (4) M8 bolts

1 – Pressure at 177°C (350°F). After which the gasket will burn and allow release of any pressure.

2 – Initial temperature taken to be -40°C as a worst case scenario.

3 – No gasket to seal void, pressure equal to ambient.

3.2 Material Properties and Component Specifications

3.2.1 Material Properties

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

Table 3.2a: Thermal Properties of Principal Transport Package Materials

Material	Density (lb/in^3)	Melting/Combustion Temperature	Thermal Expansion	Source
Depleted Uranium	0.68	1,130°C (2,066°F)	8µin/in°F	Reference #1, p. 6-11 and Reference #2
Copper	0.32	1,082°C (1,980°F)	9.2µin/in°F	Reference #1, p. 6-7 and 6-11
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
Tungsten	0.70	3,370°C (6,098°F)	2.4µin/in°F	Reference #1, p. 6-51
Cork	0.01	~230°C (~450°F)	NA	Reference #3
Lead (4% Sb)[1]	0.40	300°C (572°F)	15.4 µin/in°F	Reference #4, p. 11-420

[1]Note: 4% Sb Lead is used in subsequent thermal calculations in this Section as its melting point is lower than pure Lead which melts at 327°C (622°F) – ref: www.matweb.com for "Lead, Pb".

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. Amersham International plc RMR 214 Issue C.

4. Smithells, Colin J., *Smithells Metals Reference Book,* Seventh Edition, Butterworth-Heinemann Ltd, Oxford., 1992

3.2.2 Component Specifications

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

3.3 General Considerations

3.3.1 Evaluation by Analysis

Evaluations by analysis are described in the section they apply to in this Safety Analysis Report or when applicable in Test Plan 163 Report contained in Section 2.12.3.

3.3.2 Evaluation by Test

Evaluations by direct testing are documented in Test Plan 163 Report which is contained in Section 2.12.3.

3.3.3 Margins of Safety

Margins of safety are discussed in each section as appropriate. All testing and analysis resulted in no loss of source containment or securement in the transport packages. Though this demonstrates package compliance, it is difficult to quantify the margin related to these results. All physical testing used multiple specimens, with demonstrated results well within the regulatory requirements. Based on the results of the physical testing and the related analyses, we estimate the margin of safety for the Model 976 Series packages as high.

3.4 Thermal Evaluation for Normal Conditions of Transport

3.4.1 Heat and Cold

3.4.1.1 Insolation and Decay Heat
(Reference:

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

This analysis determines the maximum surface temperature produced by solar heating of the Model 976 Series transport package loaded at maximum activity in accordance with 10 CFR 71.71(c)(1) and IAEA No. TS-R-1 (ST-1, Revised) 1996 Edition (Revised). 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 assure conservatism, the following assumptions are made:

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

b. The transport package is assumed to undergo free convective heat transfer from the top and sides, as airflow to the bottom of the package will most likely be blocked by the ground and/or a pallet.

c. To maximize the temperature of the stainless steel drum surface temperature, the inside transport package faces are considered perfectly insulated so there is no conduction into the transport package. In use, the inside transport package will act as a heat sink during daylight hours and a heat source during the night, but this will be ignored for this calculation.

d. The transport package is approximated as a right cylinder with dimensions, 19 ¾ inches (0.52 m) in diameter (conservatively using the maximum lid diameter – the drum diameter is only 18 3/16 inches (0.46 m) except where it has stiffening ribs) and 21¼ inches (0.54m) high.

e. The surfaces of the transport package are assumed to be solid. The faces are considered to be sufficiently thin so that no temperature gradients exist in the faces.

f. The worst case decay heat load (25 Watts) is added to the solar heat input load.

g. The emissivity coefficient of the stainless steel transport package is assumed to be 0.3, while the absorptivity coefficient is assumed to be 0.8.

The maximum surface temperature is computed using the steady state heat balance relationship; heat input (Q_{in}) equals heat output (Q_{out}).

$$Q_{in} = Q_{out}$$

Heat Input:

The solar heat input is the combined solar heating of the top horizontal surface and the vertical side surface. The insolation data, provided in 10 CFR 71.71(c)(1), is found in Table 3.4.1a.

Table 3.4.1.a: Insolation Data

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

Top surface heat input: $Q_{IT} = 800\ W/m^2 \times 0.212\ m^2 = 170\ W$

Side surface heat input: $Q_{IS} = 400\ W/m^2 \times 0.880\ m^2 = 352\ W$

Decay heat input: $Q_{DT} = 25\ W$

Absorptivity coefficient: $A_c = 0.8$

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

Total heat input: $Q_{IN} = A_c\,(Q_{IT} + Q_{IS}) + Q_{DT} = 443\ W$

Heat Output:

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

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

Where:

B	=	$5.67 \times 10^{-8}\ W/m^2\ K^4$ (Stefan-Boltzmann Constant)
E	=	0.3 (Emissivity of rough stainless steel @ 300°K)
A_{TS}	=	$1.048\ m^2$ (top and side surface area)
T_W	=	The maximum surface temperature of the package (°C)
T_A	=	38°C (ambient temperature, per 10 CFR 71.71(c)(1))

Therefore:

$$Q_R = 1.78 \times 10^{-8}\{(T_W + 273)^4 - (311)^4\} = 1.78 \times 10^{-8}\ (T_W + 273)^4 - 166.76 \quad \text{(Equation 1)}$$

Top surface convection: $Q_T = H_T\ A_T\ (T_W - T_A)$ (Equation 2)

Where:

A_T = $0.1976\ m^2$ (the top surface area)
H_T = The free convection coefficient for a flat horizontal surface

For a heated plate facing up, the free convection coefficient for laminar flows is (Reference: Fundamentals of Heat and Mass Transfer, F. P. Incropera, 4th Edition, 1996, Ch. 9).

$H_T = 0.54\ [(g\ \beta\ (T_W - T_A)\ L^3) / (\nu\ \alpha)]^{1/4}\ (K / L)$

Where:

g = $9.8\ m/s^2$
β = $0.003215\ (1/(T_A + 273))$
L = 0.13 m (Area / Perimeter)
ν = $18.9 \times 10^{-6}\ m^2/s$
α = $26.9 \times 10^{-6}\ m^2/s$
K = 28.52×10^{-3} W/mK

Therefore:

$Q_T = 0.450\ (T_W - 38)^{1.25}$ (Equation 3)

Side surface convection: $Q_S = H_S\ A_S\ (T_W - T_A)$ (Equation 4)

Where:

A_S = $0.8506\ m^2$ (the total surface area of sides)
H_S = The free convection coefficient for a flat vertical surface

As stated in "Heat Transfer", Fourth Edition by Alan J. Chapman (1984), in the case of a vertical cylinder, calculations for a vertical plate may be applied to the case of the vertical cylinder as long as the circumferential curvature of the cylinder is not great, basically that:

$$\frac{D}{L} > \frac{35}{Gr_L^{1/4}}$$

Where:

D = Diameter of cylinder = 0.50 m
L = Characteristic Length = Height cylinder = 0.54 m
Gr_L = Grashof number for the plate length which is equal to:

$$Gr_L = \frac{L^3\ g\ \beta\ \Delta t}{\nu}$$

Where:

$g = 9.8\ m/s^2$
$\beta = 0.003215\ (1/(T_A + 273))$
$\Delta t = (t_s - t_f) = 40°C\ (40°K)$
$\nu^2 = 18.90 \times 10^{-6}$ @ t_m from Chapman Appendix Table A.6.

For the Model 976 drum assemblies, this calculates to the following:

$$\frac{D}{L} > \frac{35}{Gr_L^{1/4}}$$
$$0.93 > 0.22$$

Therefore in the case of the Model 976 drum assemblies, vertical plane calculations are acceptable for use in convection calculations for a vertical cylinder as used in the following calculations.

For a vertical plate, the free convection coefficient for laminar flows is (Reference: Fundamentals of Heat and Mass Transfer, F. P. Incropera, 4th Edition, 1996, Ch. 9).

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

Where:

L = 0.201 m (Area / Perimeter)

Therefore:

$$Q_S = 0.082\ (T_W - 38) + 1.65\ (T_W - 38)^{1.25} \qquad \text{(Equation 5)}$$

Total heat output: $Q_{OUT} = Q_R + Q_T + Q_S$

Total heat input: $Q_{IN} = Q_R + Q_T + Q_S + Q_{DT} = 443$ W

Substituting for Q_R from Equation 1, Q_T from Equation 3, and Q_S from Equation 5:

$$443 \text{ Watts} = 1.78 \times 10^{-8}\ (T_W + 273)^4 + 0.082\ (T_W - 38) + 2.101\ (T_W - 38)^{1.25} - 166.76$$

Iteration of this relationship yields a maximum wall temperature (T_W) of 90.3°C (195°F). This temperature would constitute the most onerous Normal Transport thermal condition. Based on the package materials of construction, this temperature will not be sufficient to adversely affect the package containment or shielding integrity. As such the package complies with the requirements of this section.

3.4.1.2 Still Air (shaded) Decay Heating

(Reference:

- *USNRC, 10 CFR 71.43(g)*
- *IAEA TS-R-1, paragraphs 617)*

This analysis calculates the maximum surface temperature of the Model 976 Series Transport package in the shade (i.e., no insolation effects), assuming an ambient temperature of 38°C (100°F), per 10 CFR 71.43(g).

The same assumptions from Section 3.4.1.1 are used:

Using these assumptions, the maximum wall temperature (T_W) is found using the following steady state heat balance:

$$Q_D = Q_R + Q_T + Q_S \qquad \text{(Equation 6)}$$

Where:

Q_D	=	25 Watts (decay heat deposited on the surface)
Q_R	=	Heat radiated from surface of package
Q_T	=	Heat convected from top of package
Q_S	=	Heat convected from side of package

From Section 3.4.1.1,

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

Where:

B	=	5.67×10^{-8} W/m^2 K^4 (Stefan-Boltzmann Constant)
E	=	0.3 (Emissivity of rough stainless steel @ 300°K)
A_{TS}	=	1.048 m^2 (top and side surface area)
T_W	=	The maximum surface temperature of the package (°C)
T_A	=	38°C (ambient temperature, per 10 CFR 71.43(g))

Therefore:

$Q_R = 1.78 \times 10^{-8}\{(T_W + 273)^4 - (311)^4\} = 1.78 \times 10^{-8}(T_W + 273)^4 - 166.76$ (Equation 7)

Also from Section 3.4.1.1,

$Q_T = 0.54\ [(g\ \beta\ (T_W - T_A)\ L^3) / (\nu\ \alpha)]^{1/4}\ (K / L)\ A_T\ (T_W - T_A)$

Where:

g	=	9.8 m/s^2
β	=	0.003215 $(1/(T_A + 273))$
L	=	0.13 m (Area / Perimeter)
ν	=	18.9 x 10^{-6} m^2/s
α	=	26.9 x 10^{-6} m^2/s
K	=	28.52 x 10^{-3} W/mK
A_T	=	0.1976 m^2 (the top surface area)

Therefore:

$Q_T = 0.4497\ (T_W - 38)^{1.25}$ (Equation 8)

Also from Section 3.4.1.1,

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

Where:

L	=	0.201 m (Area / Perimeter)
A_S	=	0.8506 m^2 (the total surface area of sides)

Therefore:

$Q_S = 0.082\ (T_W - 38) + 1.651\ (T_W - 38)^{1.25}$ (Equation 9)

Substituting Equations 7, 8, and 9 into Equation 6:

$$25 \text{ Watts} = 1.78 \times 10^{-8}\ (T_W + 273)^4 + 2.101\ (T_W - 38)^{1.25} + 0.082\ (T_W - 38) - 166.76$$

Iteration of this relationship yields a maximum wall temperature (T_W) of 42.7°C (109°F), which is less than the maximum 50°C (122°F) allowed by 10 CFR 71.43(g).

3.4.1.3 Cold Effected Materials

The carbon steel components of the Model 976 Series (internal shields on the 976A and 976C) are most affected by the low Normal Transport temperature (-40°C). During testing, shock induced stresses could cause the steel to fail in brittle fracture. As such, all shock inducing testing (i.e. drops, punctures and penetrations) was carried out at the lower temperatures. Outer drums and cork inserts absorbed the majority of the energy and the carbon steel was not damaged during testing.

Cork used for the drum inserts also exhibits some brittle tendencies at lower temperatures. Again, all cork inserts were kept at or below (-40°C). The inserts exhibited cracking to varying degrees, but provided adequate protection for all of the specimens tested.

All materials exhibit some contraction due to lower temperatures. However in this limited temperature range, the Model 976 was not adversely effected as all specimens passed the Normal and Hypothetical Accident drop, puncture and compression testing.

3.4.2 Maximum Normal Operating Pressure

All outer drum components are vented to the atmosphere. As such, pressure will not build up in the packages during Normal Transport conditions. This condition is not time dependent once steady state is achieved. However, some of the shields do have small gasketed cavities. The Model 976A has the largest cavity and the largest surface area of cover to have this pressure act on. If the cavities are sealed under the lowest temperatures (-40°C) and then allowed to heat up to the highest (88°C from Section 3.4.1.1) a small pressure differential will be created.

Using the Ideal Gas Law and equating for two standard scenarios we get:

$$P_1/T_1 = P_2/T_2 \qquad \text{(Equation 13)}$$

Where: P_1 = Ambient pressure at sealing = 14.7 psi
T_1 = -40°C (233°K)
P_2 = Pressure at temperature (88°C) (361°K)
T_2 = 88°C (361°K)

Substituting into Equation 13 we get a pressure of:

$$P_2 = 22.8 \text{ psi}$$

Which is a pressure differential of 8 psi. All other gasketed cavities will exhibit this pressure. Containers with no gasket sealed cavities will exhibit a pressure differential of 0 psi as they are vented to the atmosphere with no means for creating a pressure differential. No other contributing gas sources are present.

3.4.3 Maximum Thermal Stresses

The temperature and pressure variations described in Sections 3.4.1 and 3.4.2 will not adversely affect the transport package during normal transport since the melting temperatures of all safety critical components are well above these temperatures and the pressures calculated are insufficient to cause package failure. It its therefore concluded that the Model 976 transport package will maintain its structural integrity and shielding effectiveness under the normal transport thermal stress conditions.

3.5 Thermal Evaluation Under Hypothetical Accident Conditions

3.5.1 Initial Conditions

Frequently it is difficult to determine which damaged container would be the worst case. As such multiple containers were tested. This was also the case for the fire tests performed on the Models 3605B and Model 650L (See Section 2.12.3 for Test Plan 163 Report and reference section 5.4 for an assessment of the Model 976 Series containers based on the testing performed on the Models 3605B and 650L).

Since the Model 976 Series container is symmetrical along its axis, orientation would have little effect on the thermal response of the package. Additionally, all thermal gradient calculations assume a starting temperature of –40°C, as this is the worst case scenario for gas pressure build up.

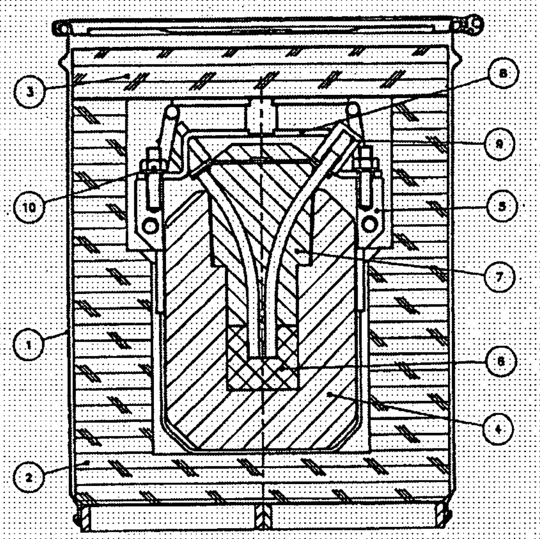
3.5.2 Fire Test Condition Assessment

The response of the package, in its various configurations, to the thermal test of 10 CFR 71.73(c)(4) is assessed from previous satisfactory thermal tests performed on the Model 3605B and the Model 650L (See Section 2.12.3 for Test Plan 163 Report and reference section 5.4).

Damage to the test units under Test Plan 163 Report (Section 2.12.3) was external. All test specimens retained closure between the lid and drum base and no air gaps were created that could allow charring of the cork greater than was observed in the thermal tests performed on the Model 3605B (Test 1835 located in Appendix F of Test Plan 163 Report in Section 2.12.3). No damage was induced in any of the inner shield containers and there was no cracking of any welds in the inner shield containers after the Hypothetical Accident drop testing.

3.5.2.1 General Considerations

Thermal testing was performed for a similar, but smaller, drum design the Model 3605B (See Figure 3.5.2a). The tested drum measured 32.5 cm in diameter by 40.5 cm tall with minimum cork thickness on the bottom of 4 cm, on the top of 4.5 cm and on the sides of 5 cm. In contrast the Model 976 package measures 50 cm in diameter by 54 cm tall. The Model 976 Series packages have a minimum cork thickness, which is based on the Model 976A configuration containing the least cork material, of 5 cm on the bottom, 12.7 cm on the top and 8.3 cm on the sides.



Item	Description	Material	No. off
1	DRUM	STEEL	1
2	PACKING INSERT	CORK	1
3	PACKING SPACER	CORK	1
4	LEAD POT	LEAD	1
5	POT CRADLE	CARBON STEEL	1
6	CORE SHIELD	TUNGSTEN ALLOY DEPLETED URANIUM	1
7	SHIELDING INSERT	LEAD	1
8	RETAINING STRAP	CARBON STEEL	1
9	TUBE CAP	BRASS	4
10	M10 COMBI NUT	STAINLESS STEEL	2

FIGURE 3.5.2a – TEST SPECIMEN CONFIGURATION FOR TEST NUMBER 1835

Test Number 1835 (see Appendix F of Test Plan 163 Report in Section 2.12.3) documents testing of a Model 3018 inner shield container (lead shielded device) inside of a cork lined steel drum assembly. The cork used in these test units was purchased to the same specification as the cork used for the test specimens under Test Plan 163 Report, however, the overall cork thickness is greater in the Model 976 style packages than was used in the specimens tested under Test Number 1835.

Testing included 9 m drop tests and puncture tests in similar orientations as were performed for the test specimens under Test Plan 163 Report. The test specimens under Test Number 1835 were tested at ambient temperature and were not cooled to -40°C prior to the 9 m and puncture drop tests as were the drop test units under Test Plan 163 Report.

As was seen with the Model 976 style test specimens, the test units described under Test Number 1835 also experienced drum deformation but no loss of the lid from the drum base. Though cracking of the cork was not specifically referenced on the sides of the cork liners in the test units from Test Number 1835, cracking of the bottom cork inserts was noted. The bottom of the cork cavity was cracked around the circumference and across diagonals in line with the drum reinforcement bars.

Thermal testing of the specimens under Test Number 1835 placed the specimens into a furnace maintained at an ambient temperature between 800°C - 820°C for a period of 30 minutes. The test specimens did not contain any radioactive contents during the thermal testing performed under Test Number 1835. The test specimens were allowed to cool for at least 18 hours before disassembly and evaluation.

Upon evaluation it was found that the top cork inserts exhibited slight charring (e.g. 22-25 mm depth) with a 26 mm thickness of the insert remaining intact. In both cases the inner lead shield container was undamage, exhibiting only the presence of a resin condensate and soot on the lead pot exterior. For both test specimens, the maximum temperature recorded by temperature strips on the exterior surface of the lead pot was 82°C. This temperature rise was less than ¼ of that necessary to reach the melting point of the lead pot (300°C). Therefore no melting or slumping of the lead shielding occurred.

Upon inspection of the test specimens under Test Plan 163 it was observed that TP163(C) exhibited the largest cork cracking on the side inserts. This test unit contained jagged cracks up to ¼" in width in the sides of the cork inserts (see Figures 28 and 29 in Test Plan 163 Report contained in Section 2.12.3). The presence of these cracks introduces the possibility of a different result in the thermal test if performed. The three thermal transport mechanisms are conduction, convection and radiation. Each will be addressed in the following assessment.

3.5.2.2 Conduction Contribution During Fire Test

The shield containers used in the Model 976 drum assembly are of two general types: (1) those which use depleted uranium for their primary shielding (e.g., 855 and 3078), and (2) those that use lead for their primary shielding with supplemental materials as part of the inner assembly shielding design (e.g., 3015, 3018, 3056 and 1911). All shield container exteriors are a steel weldment which does not melt below 1,427 °C. The melting point of depleted uranium is 1,130°C.

A calculation of the worst case, steady state conduction through the cork insulation that could be created in TP163(C) is as follows:

$$Q_X = \frac{k\,A(T_1 - T_2)}{L}$$

(Reference Fundamentals of Heat and Mass Transfer, 5th Edition, by Incropera and Dewitt, page 5.)

Where:
Q_X = the heat transfer rate in Watts
k = coefficient of thermal conductivity, 0.0314 W/m K for air (at 370°K); 0.039 W/ m K for cork (Reference: Fundamentals of Heat and Mass Transfer, 5th Edition, by Incropera and DeWitt, Appendix A)
A = cross sectional area of material (~2 cracks, ¼" wide = 0.00635 m, each 0.5 m long) = 0.00635 m^2
T_1 = Drum Wall Temperature (assumed to be thermal test temperature = 800°C or 1,027°K)
T_2 = Shield Container Initial External Temperature (assumed to be = 97°C or 370°K from Section 3.5.2.4)
L = minimum thickness of the outer and inner cork liners (located on the sides) = 0.08m + 0.04m = 0.12 m

For the thermal test, regardless of the cork condition, A, T_2, T_1 and L will be the same. The only difference will be the variation in the coefficient of thermal conductivity between air and cork. As indicated above, the coefficient for air is less than the coefficient for cork, therefore the heat transfer rate in air through the crack will be less than is experienced through the cork.

Calculation of the maximum conduction through the solid cork is based on the maximum outer area divided by the minimum cork thickness. The drum outer surface is 0.52 m in diameter and 0.54 m high for a maximum cross sectional area of 1.31 m^2. The minimum cork thickness at the bottom of the drum is 0.08 m. Therefore the maximum heat transfer rate from conduction is 420 Watts.

3.5.2.3 Convection Contribution During Fire Test

There is a limited air gap between the cork and the inner surface of the drum. As such, movement of the air around the cork inside the drum to produce convection heating will be insignificant when compared to the conductive heat transferred directing from the drum to the cork. If the crack in the cork is approximated as a solid air volume between the inner drum surface and the outer shield container, then a worst case approximation of the conductive heat transfer can be made in this limited air volume. Similar to steady state conduction, the under steady state conditions, the local heat transfer rate can be calculated as follows:

$$q_X = (T_1 - T_2) \int h \, dA_s$$

(Reference Fundamentals of Heat and Mass Transfer, 5th Edition, by Incropera and Dewitt, page 327.)

Where:
q_x = the heat transfer rate in Watts
h = convection coefficient for air = 10 Watts/m^2 °K (Reference IAEA TS-G-1.1 (ST-2))
dA_s = cross sectional area of material (~1 cracks, ¼" wide = 0.00635 m, each 0.5 m long)
T_1 = Drum Wall Temperature (assumed to be thermal test temperature = 800°C or 1,027°K)
T_2 = Shield Container Initial External Temperature (assumed to be = 97°C or 370°K from Section 3.5.2.4)

Estimating the air volume between the inner drum wall and outer shield surface as a vertical cylinder produces the following equation.

$$q_X = h \, 2 \, \pi \, r \, l \, (T_1 - T_2)$$

Where:

r = the radius of the cylinder (crack) = 0.0032 m
l = length of the crack = 0.305 m

Solving for q_x produces a worst case heat transfer rate from convection of 43 Watts along the inner surface of the cork crack.

3.5.2.4 <u>Radiant Heat Contribution During the Fire Test</u>

The jagged path of the crack through the cork prevents any radiation from the drum wall directly contacting the shield container. Without direct contact, the radiative heat transfer to the shield container surface will be insignificant in comparison to the other heat transfer contributions.

Assuming heat contribution to the inner shield from the radioactive contents produces a worst case transfer rate as follows:

The Specific heat output of Ir-192 is 8.6 mW/Ci assuming a decay energy of

1.46 MeV/decay and that 100% of the radioactive decay is transferred to thermal energy. The maximum source content for the package is 1,250 Ci. The maximum decay heat for Ir-192 in table 1.2d has been adjusted to account for content activity of the source. Actual content to output activity varies based on the capsule configuration as well as variations in isotope self-absorption. A factor of 2.3 was used to convert output activity to content activity as this factor reflects the worst case variation for Ir-192 sources transported in these packages. Therefore the total content activity would be 2,875 Ci and the total heat output assuming a conservative 100% decay during the thermal test is:

8.6 mW/Ci x 2,875 Ci = 24,725 mW = 24.73 Watts

Even assuming complete decay of the Ir-192 during the thermal test produces a heat transfer rate that is insignificant when compared with heat transfer from conduction and convection. Calculation for the Model 1911 inner shield (smallest shield diameter/wall thickness and largest source term) based on a thermal analysis from "Fundamentals of Heat and Mass Transfer", F.P. Incropera, 5th Edition, 2002 is as follows:

$$Q_{in} = Q_{radiated} = Q_{decay} = 24.73\ watts$$

Where:

$Q_{radiated}$ = heat radiated = B E A_{ts} [$(T_w + 273)^4 - (T_m + 273)^4$]

(This equation assumes no conduction or convection from all surfaces and radiative heat losses from the top and side surfaces only).

A_{ts} = Area of the top and sides = 0.174 m^2 based on:

$$A_{ts} = \left(\left[\frac{diameter}{2}\right]^2 \pi\right) + (diameter)(height)\pi$$

T_a = ambient temperature = 20°C
T_w = shield maximum equilibrium temperature
T_m = shield median temperature = $(T_a + T_w)/2$
B = Stefan Boltzmann Constant = 5.670×10^{-8}
E = emissivity for rough stainless steel surface between 300 and 400°K = 0.3

Iteration for T_w balancing the heat in to the heat radiated produces a value of 109.4°C (229°F) for the maximum temperature at the surface of the inner shield prior to the start of the thermal test.

3.5.2.5 Thermal Contribution Summary During the Fire Test

To raise the temperature of the shield containers with lead to the melting point of lead would require a significant amount of energy. The specific heat of lead, $C_p = 0.15$ kJ/kg-°K. From this relation, calculation of the required heat transfer rate is as follows:

$$Q_{Input} = C_p M (T_2 - T_1)$$

Where
Q_{Input} = Minimum heat input to melt the lightest lead container (Model 3015)
C_p = Specific Heat of Lead
M = Mass of the shield container = 114 lbs or 52 kgs for Model 3015 (Lightest shield)
T_2 = Melting temperature of lead = 573°K (300°C, see Table 3.2a)
T_1 = Ambient shield temperature = 370°K (See Section 3.5.2.4)

Therefore the required heat transfer rate to cause lead melting in the shield is 1,569 kJ or 1.569 x 10^6 Watts/sec. To achieve this in the 30 minutes (1,800 sec) of the thermal test requires a heat input of 872 Watts. Even when combining all the worst case thermal contribution factors, the required heat input in the most vulnerable area along the cork crack is less than 60% of the actual heat input that would melt the lead shield and will therefore be insufficient to degrade the lead shielding.

3.5.2.6 Additional Factors for Consideration During the Fire Test

In the case of the depleted uranium shield containers, there was no breach or weld cracking of the shield container which would allow oxygen to reach the inner depleted uranium shield. Without the presence of a continuing source of oxygen, these shields will remain intact during the thermal test. As seen in testing performed on the Model 650L (Reference USNRC CoC USA/9269/B(U)-85, Test Plan 80 Report Revision 1) thermal testing of this device where cracking to allow air to the shield had occurred resulted in production of only a small amount of depleted uranium oxide. With an air path and air circulation during the thermal test, the radiation dose rate at one meter from this unit increases by approximately 10% remaining less than 3% of the regulatory limit.

Without sufficient oxygen provided to the interior of the depleted uranium shield containers (e.g., welds intact) there will be no appreciable oxidation of the depleted uranium shield inside the steel container housings, and the

800°C temperature is well below the melting point of depleted uranium (1,130°C) therefore the shield will retain its original shape throughout the thermal test.

3.5.2.7 Thermal Assessment Summary

The thermal test will not adversely effect the structural integrity of the shield containers. The Model 855 and Model 1911 containers were physically undamaged after the 9 m and puncture drop testing. The other shield containers (e.g., Models 3015, 3018, 3056 and 3078) are lighter than the Models 855 and 1911 and would therefore be expected to sustain less damage in the drop configurations than was seen for these package assemblies (See Section 5.4 of Test Plan 163 Report in Section 2.12.3).

For shield containers incorporating lead, the exterior shield temperature will not exceed 109.4°C. The testing performed under Test Number 1835 took drum assemblies at ambient temperature prior to subjecting them to the thermal test condition. In actual practice the 976 package assemblies would have been thermally tested immediately after the puncture test and would still have been at a temperature below 0°C introducing a further temperature difference to be overcome before the shield container would be susceptible to a melting temperature.

For the Model 976 Series packages, performance of the thermal test would not produce a condition sufficient to reduce the shielding efficiency or containment efficiency of the shield containers within the 976 drum assembly. In addition, the temperature increase in the shield container surfaces will be well below the melting temperature of the lead which will preclude any shielding configuration change or lead slumping in the shield containers. By assessment, the Model 976 Series package designs would therefore meet the thermal test requirements.

3.5.3 Maximum Temperatures and Pressure

All outer drum components are vented to the atmosphere. As such, pressure will not build up in the packages during Normal Transport conditions. This condition is not time dependent once steady state is achieved. However, some of the shields do have small gasketed cavities. The Model 976A has the largest cavity and the largest surface area of cover to have this pressure act on. If the cavities are sealed under the lowest temperatures (-40°C) and then allowed to heat up until the gasket is burned away allowing release of any pressure build up at a temperature of 177°C a small pressure differential will be created.

Using the Ideal Gas Law and equating for two standard scenarios we get:

$$P_1/T_1 = P_2/T_2 \qquad \text{(Equation 13)}$$

Where: P_1 = Ambient pressure at sealing = 14.7 psi
T_1 = -40°C
P_2 = Pressure at temperature (177°C)
T_2 = 177°C

Substituting into Equation 13 we get a pressure of:

$$P_2 = 28 \text{ psi}$$

Which is a pressure differential of 14 psi. All other gasketed cavities will exhibit this pressure. Containers with no gasket sealed cavities 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. See Section 2.7.4.3.

3.5.4 Accident Conditions for Fissile Material Packages for Air Transport

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

3.6 Appendix

Not Applicable.

Section 4 – CONTAINMENT

4.1 Description of the Containment System

(Reference:

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

4.1.1 Containment Boundary

The containment system consists of the Model 976 Series transport packages and the radioactive source capsule(s). The source capsule(s) shall be qualified as Special Form radioactive material under 49 CFR 173 and IAEA No. TS-R-1 (ST-1, Revised) 1996 Edition (Revised).

4.1.2 Special Requirements for Plutonium

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

4.2 General Considerations

4.2.1 Type A Fissile Packages

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

4.2.2 Type B Packages

(Reference:

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

As demonstrated in Test Plan 90 Report and assessed under Test Plan 163 Report (Sections 2.12.2 and 2.12.3 respectively), 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 976 Series packages therefore meets the requirements of this section.

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

(Reference:

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

As demonstrated in Test Plan 90 Report and assessed under Test Plan 163 Report (Sections 2.12.2 and 2.12.3 respectively), performance of the normal conditions of transport testing caused no breach of the source capsules contained in the package. Since the source capsules are the primary containment of the radioactive contents and no release from the source capsules occurred, the Model 976 Series packages meet the requirements of this section.

4.4 Containment Under Hypothetical Accident Conditions (Type B Packages)

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

As demonstrated in Test Plan 90 Report and Test Plan 163 Report (Sections 2.12.2 and 2.12.3 respectively), after 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 976 Series packages therefore meet the requirements of this section.

4.5 Leakage Rate Tests for Type B Packages

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

The primary containment for the radioactive material in the Model 976 Series transport packages are the radioactive source capsules. All source capsules authorized for Type B transport in the Model 976 Series packages are certified as special form radioactive material under 10 CFR Part 71, 49 CFR Part 173 IAEA No. TS-R-1 (ST-1, Revised) 1996 Edition (Revised). After manufacture and again once every six months thereafter prior to transport, the source capsule is leak tested in accordance with ISO9978:1992(E) (or more recent editions) to ensure that containment of the source does not allow release of more than 0.005 μCi of radioactive material. These fabrication and periodic tests ensure that contamination release from the package does not exceed the regulatory limits.

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

4.6 Appendix

Not Applicable.

Section 5 - SHIELDING EVALUATION

5.1 Description of Shielding Design

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

5.1.1 Design Features

The principal shielding in the Model 976 Series transport packages are depleted uranium, tungsten or lead shield assemblies used in the shield containers. Dimensional information for the individual shield containers is contained in the shield drawings included in Section 1.4. Table 3.2a lists the material densities of the packaging.

5.1.2 Summary Table of Maximum Radiation Levels

Table 5.1a includes radiation profile data obtained from the 976 Series package that was tested to the Normal Conditions of Transport under Test Plan 90 (see Section 2.12.2). Note that radiation survey results from this package were obtained after the package had also been subjected to the Hypothetical Accident Condition testing.

Table 5.1a: Model 976A with 855 sn 9 - TP90A
Summary Table of External Radiation Levels Extrapolated to Capacity of 1,000 Ci Ir-192 (Non-Exclusive Use) After Normal and Hypothetical Accident Transport Condition Testing Under Test Plan 90 Report
(Ref: Profile Sheet "855 Device Profile (Without 976 drum/cork overpack) Used for Post Test Results under Test Plan 90 Report and Pre-Test Results under Test Plan Report 163". Copy of profile sheet located in Appendix D of Test Plan 90 Report (Section 2.12.2) and Appendix D of Test Plan Report 163 (Section 2.12.3))

Normal Conditions of Transport	Package Surface mSv per hour (mrem per hour)			1 Meter from Package Surface mSv per hour (mrem per hour)		
Radiation	Top	Side	Bottom	Top	Side	Bottom
Gamma	1.41 (141)	0.38 (38)	0.75 (75)	0.019 (1.9)	0.003 (0.3)	0.008 (0.8)
Neutron	NA	NA	NA	NA	NA	NA
Total	1.41 (141)	0.38 (38)	0.75 (75)	0.019 (1.9)	0.003 (0.3)	0.008 (0.8)
10 CFR 71.47(a) Limit	2 (200)	2 (200)	2 (200)	0.1 (10)[1]	0.1 (10)[1]	0.1 (10)[1]

[1]Transport Index may not exceed 10. The Transport Index is equivalent to the 1 meter reading in mRem per hour (i.e., 5 mRem per hour at 1 meter = a Transport Index of 5.0).
[2]All packages accepted and released for shipment under this Model designation will have a Transport Index less than or equal to 10.

NOTE: Survey results in Test Plan 90 Report both before and after hypothetical accident conditions were obtained from the Model 855 outside of the drum and cork components. This produced dose rates which would be higher than the Model 855 if it had been placed inside the test drum and cork components. Values after hypothetical accident conditions are measured 1 meter from the surface of the Model 855.

Tables 5.1b through 5.1.d includes radiation profile data obtained from the 976 Series packages that were tested to the Hypothetical Conditions of Transport under Test Plan 163 (see Section 2.12.3). Notes 1 and 2 apply to Tables 5.1b through 5.1.l. Note 3 applies to Tables 5.1.b and 5.1.c only.

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

Note 2: All packages accepted and released for shipment under this Model designation will have a Transport Index less than or equal to 10.

Note 3: Survey results in Test Plan 163 Report both before and after hypothetical accident conditions were obtained from the Model 855 outside of the drum and cork components. This produced dose rates which would be higher than the Model 855 if it had been placed inside the test drum and cork components. Values after hypothetical accident conditions are measured 1 meter from the surface of the Model 855.

Table 5.1b: Model 976A with 855 sn 8 - TP163(A)
Summary Table of External Radiation Levels Extrapolated to Capacity of 1,000 Ci Ir-192 (Non-Exclusive Use) After Hypothetical Accident Transport Condition Testing Under Test Plan 163 Report

(Ref: Profile Sheet "855 Device Profile (Without 976 drum/cork overpack) Used for Post Test Results under Test Plan Report 163 (SN 8)". Copy of profile sheet located in Appendix D of Test Plan 163 Report (Section 2.12.3))

Hypothetical Accident Conditions	1 Meter from Package Surface mSv per hour (mrem per hour)		
Radiation	Top	Side	Bottom
Gamma	0.021 (2.1)	0.006 (0.6)	0.029 (2.9)
Neutron	NA	NA	NA
Total	0.021 (2.1)	0.006 (0.6)	0.029 (2.9)
10 CFR 71.51(a)(2) Limit	10 (1000)	10 (1000)	10 (1000)

Table 5.1c: Model 976A with 855 sn 9 – TP163(B)
Summary Table of External Radiation Levels Extrapolated to Capacity of 1,000 Ci Ir-192 (Non-Exclusive Use) After Hypothetical Accident Transport Condition Testing Under Test Plan 163 Report

(Ref: Profile Sheet "855 Device Profile (Without 976 drum/cork overpack) Used for Post Test Results under Test Plan Report 163 (SN 9)". Copy of profile sheet located in Appendix D of Test Plan 163 Report (Section 2.12.3))

Hypothetical Accident Conditions	1 Meter from Package Surface mSv per hour (mrem per hour)		
Radiation	Top	Side	Bottom
Gamma	0.028 (2.8)	0.006 (0.6)	0.010 (1.0)
Neutron	NA	NA	NA
Total	0.028 (2.8)	0.006 (0.6)	0.010 (1.0)
10 CFR 71.51(a)(2) Limit	10 (1000)	10 (1000)	10 (1000)

Table 5.1d: Model 976F with 1911 sn 013 – TP163(C)
Summary Table of External Radiation Levels Extrapolated to Capacity of 1,000 Ci Ir-192 (Non-Exclusive Use) After Hypothetical Accident Transport Condition Testing Under Test Plan 163 Report

(Ref: Profile Sheet "976F (1911 w/Depleted Uranium Insert Device Capacity Profile with 976 drum/cork overpack) Performed After Testing under Test Plan 163 – Test Specimen TP163C". Copy of profile sheet located in Appendix D of Test Plan 163 Report (Section 2.12.3))

Hypothetical Accident Conditions	1 Meter from Package Surface mSv per hour (mrem per hour)		
Radiation	Top	Side	Bottom
Gamma	0.006 (0.6)	0.015 (1.5)	0.006 (0.6)
Neutron	NA	NA	NA
Total	0.006 (0.6)	0.015 (1.5)	0.006 (0.6)
10 CFR 71.51(a)(2) Limit	10 (1000)	10 (1000)	10 (1000)

Tables 5.1e through 5.11 include radiation profile data used to demonstrate that all shield and package configurations will meet the external radiation level requirements for non-exclusive use transport when loaded to capacity for that package configuration.

Table 5.1e: Model 976A with 855 sn 9 – Summary Table of External Radiation Levels Extrapolated to Capacity of 1,000 Ci Ir-192 (Non-Exclusive Use)

(Ref: Profile Sheet "976A (855 Device Profile With 976 drum/cork overpack)". Copy of profile sheet located in Appendix D of Test Plan 163 Report (Section 2.12.3))

	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	0.29 (29)	0.10 (10)	0.22 (22)	0.011 (1.1)	0.003 (0.3)	0.005 (0.5)
Neutron	NA	NA	NA	NA	NA	NA
Total	0.29 (29)	0.10 (10)	0.22 (22)	0.011 (1.1)	0.003 (0.3)	0.005 (0.5)
10 CFR 71.47(a) Limit	2 (200)	2 (200)	2 (200)	0.1 (10)	0.1 (10)	0.1 (10)

Table 5.1f: Model 976B with 3015 sn P500/2128 – Summary Table of External Radiation Levels Extrapolated to Capacity of 350 Ci Ir-192 (Non-Exclusive Use)

(Ref: Profile Sheet "976B (3015 Device Capacity Profile With 976 drum/cork overpack)". Copy of profile sheet located in Appendix D of Test Plan 163 Report (Section 2.12.3))

	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	0.11 (11)	1.86 (186)	0.60 (60)	0.005 (0.5)	0.067 (6.7)	0.014 (1.4)
Neutron	NA	NA	NA	NA	NA	NA
Total	0.11 (11)	1.86 (186)	0.60 (60)	0.005 (0.5)	0.067 (6.7)	0.014 (1.4)
10 CFR 71.47(a) Limit	2 (200)	2 (200)	2 (200)	0.1 (10)	0.1 (10)	0.1 (10)

Table 5.1g: Model 976C with 3056 sn P0745-060 – Summary Table of External Radiation Levels Extrapolated to Capacity of 1,250 Ci Ir-192 (Non-Exclusive Use)

(Ref: Profile Sheet "976C Modified Insert Configuration – Performed 11 Jan 06. Copy of profile sheet located in Section 5.5.2)

	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	0.78 (78)	1.46 (146)	1.69 (169)	0.046 (4.6)	0.053 (5.3)	0.055 (5.5)
Neutron	NA	NA	NA	NA	NA	NA
Total	0.78 (78)	1.46 (146)	1.69 (169)	0.046 (4.6)	0.053 (5.3)	0.055 (5.5)
10 CFR 71.47(a) Limit	2 (200)	2 (200)	2 (200)	0.1 (10)	0.1 (10)	0.1 (10)

**Table 5.1h: Model 976D with 3018 sn P500/2057 –
Summary Table of External Radiation Levels Extrapolated to Capacity of
500 Ci Ir-192 (Non-Exclusive Use)**

(Ref: Profile Sheet "976D (3018 Device Capacity Profile With 976 drum/cork overpack)". Copy of profile sheet located in Appendix D of Test Plan 163 Report (Section 2.12.3))

	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	0.45 (45)	1.51 (151)	1.91 (191)	0.015 (1.5)	0.036 (3.6)	0.036 (3.6)
Neutron	NA	NA	NA	NA	NA	NA
Total	0.45 (45)	1.51 (151)	1.91 (191)	0.015 (1.5)	0.036 (3.6)	0.036 (3.6)
10 CFR 71.47(a) Limit	2 (200)	2 (200)	2 (200)	0.1 (10)	0.1 (10)	0.1 (10)

**Table 5.1i: Model 976E with 3078 sn 3078.04 –
Summary Table of External Radiation Levels Extrapolated to Capacity of
1,000 Ci Ir-192 (Non-Exclusive Use)**

(Ref: Profile Sheet "976E (3078 Device Capacity Profile With 976 drum/cork overpack)". Copy of profile sheet located in Appendix D of Test Plan 163 Report (Section 2.12.3))

	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	0.63 (63)	0.015 (15)	0.018 (18)	0.064 (6.4)	0.005 (0.5)	0.005 (0.5)
Neutron	NA	NA	NA	NA	NA	NA
Total	0.63 (63)	0.015 (15)	0.018 (18)	0.064 (6.4)	0.005 (0.5)	0.005 (0.5)
10 CFR 71.47(a) Limit	2 (200)	2 (200)	2 (200)	0.1 (10)	0.1 (10)	0.1 (10)

**Table 5.1j: Model 976F with 1911 sn 013 with Depleted Uranium Insert
Summary Table of External Radiation Levels Extrapolated to Capacity of 1,000 Ci Ir-192
(Non-Exclusive Use)**

(Ref: Profile Sheet "976F (1911 w/Depleted Uranium Insert Device Capacity Profile With 976 drum/cork overpack) Performed Before Testing under Test Plan 163 – Test Specimen TP163C". Copy of profile sheet located in Appendix D of Test Plan 163 Report (Section 2.12.3))

	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	0.11 (11)	0.049 (49)	0.018 (18)	0.007 (0.7)	0.011 (1.1)	0.007 (0.7)
Neutron	NA	NA	NA	NA	NA	NA
Total	0.11 (11)	0.049 (49)	0.018 (18)	0.007 (0.7)	0.011 (1.1)	0.007 (0.7)
10 CFR 71.47(a) Limit	2 (200)	2 (200)	2 (200)	0.1 (10)	0.1 (10)	0.1 (10)

Table 5.1k: Model 976F with 1911 sn 013 with Tungsten Insert – Summary Table of External Radiation Levels Extrapolated to Capacity of 1,000 Ci Ir-192 (Non-Exclusive Use)

(Ref: Profile Sheet "976F (1911 w/Tungsten Insert Device Capacity Profile With 976 drum/cork overpack)". Copy of profile sheet located in Appendix D of Test Plan 163 Report (Section 2.12.3))

	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	0.63 (63)	0.062 (6.2)	0.042 (4.2)	0.045 (4.5)	0.023 (2.3)	0.013 (1.3)
Neutron	NA	NA	NA	NA	NA	NA
Total	0.63 (63)	0.062 (6.2)	0.042 (4.2)	0.045 (4.5)	0.023 (2.3)	0.013 (1.3)
10 CFR 71.47(a) Limit	2 (200)	2 (200)	2 (200)	0.1 (10)	0.1 (10)	0.1 (10)

Table 5.1l: Model 976F with 1911 sn 013 with Lead Insert – Summary Table of External Radiation Levels Extrapolated to Capacity of 1,000 Ci Ir-192 (Non-Exclusive Use)

(Ref: Profile Sheet "976F (1911 w/Lead Insert Device Capacity Profile With 976 drum/cork overpack)". Copy of profile sheet located in Appendix D of Test Plan 163 Report (Section 2.12.3))

	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	0.73 (73)	1.27 (127)	1.12 (112)	0.034 (3.4)	0.052 (5.2)	0.034 (3.4)
Neutron	NA	NA	NA	NA	NA	NA
Total	0.73 (73)	1.27 (127)	1.12 (112)	0.034 (3.4)	0.052 (5.2)	0.034 (3.4)
10 CFR 71.47(a) Limit	2 (200)	2 (200)	2 (200)	0.1 (10)	0.1 (10)	0.1 (10)

Tables 5.1a through 5.1l include radiation profile data used to demonstrate that the Model 976 style package configurations will meet the external radiation level requirements for non-exclusive use transport when loaded to capacity for Ir-192.

The photon energies and quantities from both Se-75 and Yb-169 are less than Ir-192. The materials providing effective shielding for contents in the 976 style packages are lead, depleted uranium and steel. A evaluation of transmission factors for Se-75 and Yb-169 when compared to Ir-192 in these materials is shown in Table 5.1m

Table 5.1m: Radionuclude Transmission Shielding Assessment

Nuclide	Calculation Activity (Ci)	Γ (R m^2/hr Ci)	Transmission Ratio Steel Relative to Ir-192	Transmission Ratio Lead Relative to Ir-192	Transmission Ratio Depleted Uranium Relative to Ir-192
Ir-192	1	0.48	1	1	1
Se-75	1	0.203	0.422	0.088	0.019
Yb-169	1	0.125	0.160	0.006	5.5E-4

Transmission exposure rates were determined using Microshield V5 (See Section 5.5.3). Transmission values were calculated through a unit material thickness of 1 cm for each of the shielding materials and were calculated for a unit activity of 1 Ci of each radionuclide. The shielding material densities used in the Microshield calculations were as follows: steel (7.86 g/cm^3), lead (11.34 g/cm^3) and depleted uranium (18.7 g/cm^3).

The relative shielding reduction per shielding material for each radionuclide confirms that for a unit quantity of 1 Ci of Ir-192, the dose rate from an equivalent unit quantity of 1 Ci of either Se-75 or Yb-169 will allow a significantly lower transmission of radiation through the material for both Se-75 and Yb-169 when compared to Ir-192. In addition, the relative radiation rate in air for Se-75 and Yb-169 are over ½ that which would be expected from the same activity of Ir-192.

Since each shielding material will effectively shield Se-75 and Yb-169 to a greater extent than it will shield Ir-192 and since the maximum activity of Se-75 or Yb-169 does not exceed the maximum rated capacity for the package when loaded with Ir-192 for any 976 package, the package radiation dose rates for each Model 976 design can be bounded by results performed to demonstrate the package shielding is effective to shield the maximum capacity of Ir-192 authorized in the package design.

Also, since all shielding materials will effectively shield Se-75 and Yb-169 to a greater extent than they will shield Ir-192, any combination of sources containing either Ir-192, Se-75 or Yb-169 will also not exceed the maximum package surface and 1 meter radiation dose limits specified in the regulations if the contents of a package loaded with multiple isotopes meets the following condition:

$$\sum \frac{q_i}{Q_{CoC,i}} \leq 1,$$

Where q_i is the amount of radioisotope i loaded in the Model 976 Series package and

$Q_{CoC,i}$ is the maximum amount of radioisotope i allowed in the Model 976 Series package as specified in the Certificate of Compliance (CoC).

Since the content activity for both Se-75 and Yb-169 are equal to or less than the content activity of Ir-192 in all cases, by the preceeding assessment the Model 976 package configurations will also meet the external radiation level requirements for non-exclusive use when loaded to capacity for Se-75 or Yb-169 or when loaded with more than one different radionuclide so long as the sum of the activity ratios as described above is ≤ 1.

5.2 Source Specification

5.2.1 Gamma Source

(Reference:

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

The gamma sources allowed for transport in the Model 976 Series transport package specified in Sections 1.2.3 and 2.10.

5.2.2 Neutron Source

Not Applicable. The Model 976 Series transport packages are not used for the transportation of neutron emitting sources.

5.3 Shielding Model

5.3.1 Configuration of Source and Shielding

A shielding model was not used as the primary justification for these packages. Shielding justification was based on direct measurement. However, a estimate of the calculated shielding efficiency for the Model 976C package configuration was performed using Microshield V5.05 (see Section 5.5.1). Comparison of the estimated shielding results for the bottom of this package configuration against the measured radiation dose were in good agreement (calculated dose rate of 141 mR/hr versus measured dose rate of 169 mR/hr). See Table 5.3.1a

Table 5.3.1a: Microshield Comparison Calculations for the Model 976C Package

Case #	DU Insert	Lead on Bottom of 3056	Steel strap of 3056 (This Gap is Assumed to be air for worst case Assessment)	Inner Cork Insert Bottom	Outer Cork Insert Bottom	Drum Bottom	Microshield Calculated Surface Dose Rate
1	Density = 18 g/cc Thickness = 1 in	Density = 11.34 g/cc Thickness = 1.4 in	Density = 0.00122 g/cc Thickness = 0.1 in	Not Included in Calculation. Dose Calculated on Surface of 3056	Not Included in Calculation. Dose Calculated on Surface of 3056	Not Included in Calculation. Dose Calculated on Surface of 3056	1,614 mR/hr
2	Density = 18 g/cc Thickness = 1 in	Density = 11.34 g/cc Thickness = 1.4 in	Density = 0.00122 g/cc Thickness = 0.1 in	Assumed to be Air Density = 0.00122 g/cc Thickness = 1.25 in	Assumed to be Air Density = 0.00122 g/cc Thickness = 2 in	Steel Density = 7.82 g/cc Thickness = 0.06 in	141 mR/hr

The surface reading at the bottom for the Model 976C package from Table 5.1.g is 169 mR/hr. If the distance and shielding provided by the drum and cork inserts are not considered in the calculations, a significantly higher dose rate is obtained using Microshield. However, Microshield produces a calculated dose rate very close to the value listed in Table 5.1g if the shielding model incorporates the actual distance from the Model 3056 shield to the drum surface, and if it accounts for the presence of the cork. As shown in test Case 2 in Table 5.3.1a, a surface dose rate of 141 mR/hr is calculated for a package which assumes the dose rate is measured on the surface of the drum and the cork has a density equivalent to air.

5.3.2 Material Properties

Not Applicable. A shielding model was not used in the justification for these packages. 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 163 Report (see Section 2.12.3) for results of radiation surveys of the 976 Series transport packages. Note that no physical testing of the package configurations using the Models 3015, 3018, or 3078 were performed under either Test Plan 90 Report or Test Plan 163 Report (Ref: Sections 2.12.2 and 2.12.3 respectively). In addition, there was no physical testing of the package configuration using the Model 3056 under Test Plan 163 Report (Ref: Section 2.12.3). Radiation survey information for the package configurations containing these inner shield containers in Test Plan 163 Report (Ref: Section 2.12.3) were performed to document shielding capacity of these package configurations. Assessment of these package configurations under the normal and hypothetical accident condition testing is provided in Test Plan 163 Report Sections 5.4 and 5.5 (See Section 2.12.3).

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) and the surface measurements were also adjusted to correct for off-set of the survey meter probe from the true surface of the package.

Activity correction factors (CF_A) were obtained by using the following relationship:

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

$$For\,Example,\ if\ A_P = 834\,Ci\ and\ A_C = 1{,}000\,Ci,\ then$$

$$CF_A = \frac{1{,}000\,Ci}{834\,Ci} = 1.2\ .$$

Therefore all original surface and 1 meter profile measurements would be multiplied by a factor of 1.2 for a package profiled using 834 Ci and a package capacity of 1,000 Ci.

Radiation measurements at the surface of the container were also adjusted to compensate for the off-set of the survey meter probe from the true surface of the package.

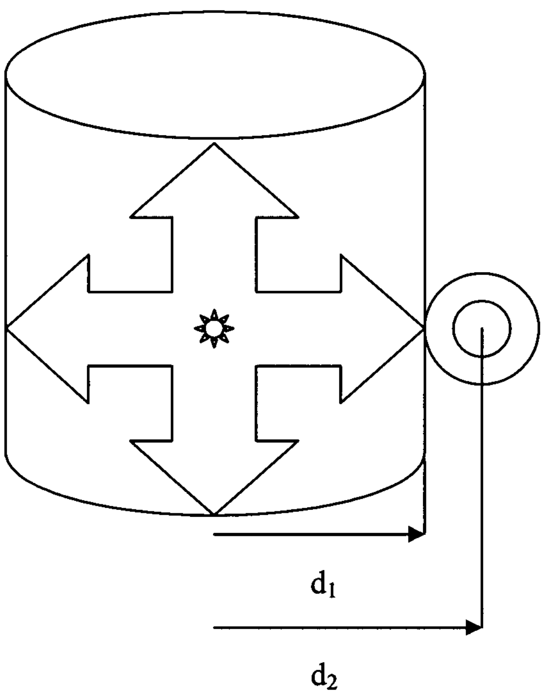
Surface correction factors (SCF) were obtained by using the following relationship:

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

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

$$SCF = \frac{9.5\,inches}{9\,inches} = 1.06$$

Therefore in the example shown, all original surface profile measurements located along the side of the drum shown in Figure 5.a would also be multiplied by a factor of 1.06 to account for surface correction of the detector to the drum. Different SCF's would be calculated for the any dimension of the container where the minimum distance from the center of the activity to the center of the radiation probe is different.



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

FIGURE 5.a. SAMPLE SURFACE CORRECTION FACTOR DISTANCE CRITERIA

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

5.4.3 Flux-to-Dose-Rate Conversion

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

5.4.4 External Radiation Levels

Radiation surveys for all 976 Series configurations showed maximum surface and 1 meter radiation levels from the transport packages within regulatory limits. Radiation surveys of 976 Series transport packages after undergoing normal and accident condition transport testing were also well within the regulatory limits.

5.5 Appendix

5.5.1 Microshield V5.05 Calculations for the Model 976C with 3056,
1,250 Curies of Ir-192, Profile of Bottom, Case 1 and Case 2

5.5.2 Profile Sheet "976C Modified Insert Configuration – Performed 11 Jan 06.

5.5.3 Microshield V5.05 Transmission for Various Nuclides and Materials

Section 5.5.1 Appendix: Microshield V5.05 Calculations for the Model 976C with 3056, 1,250 Curies of Ir-192, Profile of Bottom, Case 1 and Case 2

Section 5.5.2 Appendix: **Profile Sheet "976C Modified Insert Configuration – Performed 11 Jan 06.**

Section 5.5.3 Appendix: Microshield V5.05 Transmission for Various Nuclides and Materials

MicroShield v5.05 (5.05-00161)
AEA Technology QSA, Inc.

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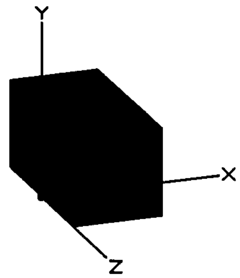
File Ref: ____________
Date: ____________
By: ____________
Checked: ____________

Case Title: Ir-192 Transmission
Description: 1 Ci Ir-192 thru 1 cm Depleted Uranium
Geometry: 1 - Point



Dose Points

	X	Y	Z
# 1	1 cm	0 cm	0 cm
	0.4 in	0.0 in	0.0 in

Shields

Shield Name	Dimension	Material	Density
Shield 1	1.0 cm	Uranium	18.7
Air Gap		Air	0.00122

Source Input
Grouping Method : Standard Indices
Number of Groups : 25
Lower Energy Cutoff : 0.015
Photons < 0.015 : Included
Library : ICRP-38

Nuclide	curies	becquerels
Ir-192	1.0000e+000	3.7000e+010

Buildup
The material reference is : Shield 1

Results

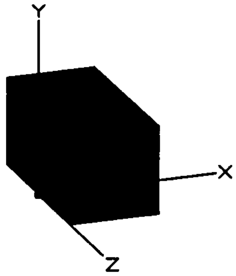
Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.015	2.032e+09	0.000e+00	2.124e-20	0.000e+00	1.822e-21
0.06	3.958e+09	3.778e-46	1.810e-19	7.504e-49	3.596e-22
0.08	1.093e+09	1.037e-18	1.214e-18	1.641e-21	1.921e-21
0.1	4.265e+07	3.514e-09	4.343e-09	5.376e-12	6.645e-12
0.15	6.682e+07	6.100e-15	4.901e-06	1.004e-17	8.070e-09
0.2	1.405e+09	2.292e-03	4.116e-03	4.045e-06	7.264e-06
0.3	5.248e+10	1.382e+05	1.878e+05	2.621e+02	3.562e+02
0.4	5.900e+08	1.128e+05	1.570e+05	2.199e+02	3.060e+02
0.5	1.900e+10	2.360e+07	3.350e+07	4.632e+04	6.575e+04
0.6	6.689e+09	2.317e+07	3.291e+07	4.522e+04	6.425e+04
0.8	1.137e+08	1.189e+06	1.694e+06	2.261e+03	3.222e+03
1.0	2.115e+07	4.087e+05	5.760e+05	7.533e+02	1.062e+03
1.5	4.598e+05	1.986e+04	2.647e+04	3.341e+01	4.454e+01
TOTALS:	8.749e+10	4.863e+07	6.905e+07	9.507e+04	1.350e+05

MicroShield v5.05 (5.05-00161)
AEA Technology QSA, Inc.

Page : 1
DOS File: SEDU.MS5
Run Date: May 23, 2007
Run Time: 9:10:49 AM
Duration: 00:00:00

File Ref: ____________
Date: ____________
By: ____________
Checked: ____________

Case Title: Se-75 Transmission
Description: 1 Ci Se-75 thru 1 cm Depleted Uranium
Geometry: 1 - Point



Dose Points

	X	Y	Z
# 1	1 cm	0 cm	0 cm
	0.4 in	0.0 in	0.0 in

Shields

Shield Name	Dimension	Material	Density
Shield 1	1.0 cm	Uranium	18.7
Air Gap		Air	0.00122

Source Input
Grouping Method : Standard Indices
Number of Groups : 25
Lower Energy Cutoff : 0.015
Photons < 0.015 : Included
Library : ICRP-38

Nuclide	curies	becquerels
Se-75	1.0000e+000	3.7000e+010

Buildup
The material reference is : Shield 1

Results

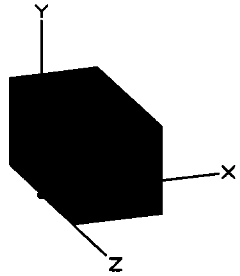
Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.015	2.093e+10	0.000e+00	2.188e-19	0.000e+00	1.876e-20
0.02	9.451e+06	0.000e+00	1.317e-22	0.000e+00	4.562e-24
0.06	4.088e+08	3.902e-47	1.870e-20	7.750e-50	3.714e-23
0.08	3.077e+06	2.919e-21	3.417e-21	4.618e-24	5.407e-24
0.1	7.846e+09	6.465e-07	7.991e-07	9.891e-10	1.223e-09
0.15	2.242e+10	2.046e-12	1.644e-03	3.370e-15	2.708e-06
0.2	5.567e+08	9.084e-04	1.631e-03	1.603e-06	2.879e-06
0.3	3.178e+10	8.367e+04	1.137e+05	1.587e+02	2.157e+02
0.4	4.203e+09	8.039e+05	1.119e+06	1.566e+03	2.180e+03
0.5	2.086e+05	2.590e+02	3.677e+02	5.084e-01	7.218e-01
0.6	1.560e+07	5.405e+04	7.678e+04	1.055e+02	1.499e+02
0.8	4.835e+04	5.054e+02	7.202e+02	9.612e-01	1.370e+00
TOTALS:	8.817e+10	9.424e+05	1.310e+06	1.832e+03	2.547e+03

MicroShield v5.05 (5.05-00161)
AEA Technology QSA, Inc.

Page : 1
DOS File: YBDU.MS5
Run Date: May 23, 2007
Run Time: 9:09:58 AM
Duration: 00:00:00

File Ref: ____
Date: ____
By: ____
Checked: ____

Case Title: Yb-169 Transmission
Description: 1 Ci Yb-169 thru 1 cm Depleted Uranium
Geometry: 1 - Point



Dose Points

	X	Y	Z
# 1	1 cm	0 cm	0 cm
	0.4 in	0.0 in	0.0 in

Shields

Shield Name	Dimension	Material	Density
Shield 1	1.0 cm	Uranium	18.7
Air Gap		Air	0.00122

Source Input
Grouping Method : Standard Indices
Number of Groups : 25
Lower Energy Cutoff : 0.015
Photons < 0.015 : Included
Library : ICRP-38

Nuclide	curies	becquerels
Yb-169	1.0000e+000	3.7000e+010

Buildup
The material reference is : Shield 1

Results

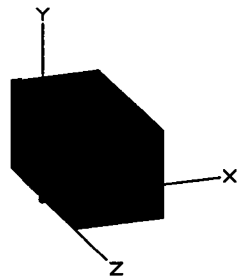
Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.015	1.678e+10	0.000e+00	1.754e-19	0.000e+00	1.504e-20
0.02	8.767e+07	0.000e+00	1.222e-21	0.000e+00	4.232e-23
0.05	5.399e+10	1.662e-77	1.979e-18	4.429e-80	5.271e-21
0.06	2.927e+10	2.794e-45	1.339e-18	5.549e-48	2.659e-21
0.1	8.105e+09	6.678e-07	8.255e-07	1.022e-09	1.263e-09
0.15	4.254e+09	3.883e-13	3.120e-04	6.394e-16	5.137e-07
0.2	2.159e+10	3.523e-02	6.325e-02	6.217e-05	1.116e-04
0.3	4.280e+09	1.127e+04	1.532e+04	2.137e+01	2.905e+01
0.4	6.509e+05	1.245e+02	1.733e+02	2.426e-01	3.376e-01
0.5	2.326e+06	2.888e+03	4.100e+03	5.670e+00	8.049e+00
0.6	3.617e+06	1.253e+04	1.780e+04	2.445e+01	3.474e+01
0.8	9.564e+04	9.996e+02	1.424e+03	1.901e+00	2.709e+00
TOTALS:	1.384e+11	2.781e+04	3.881e+04	5.364e+01	7.488e+01

MicroShield v5.05 (5.05-00161)
AEA Technology QSA, Inc.

Page : 1
DOS File: IRSTL.MS5
Run Date: May 23, 2007
Run Time: 9:08:17 AM
Duration: 00:00:00

File Ref: ___________
Date: ___________
By: ___________
Checked: ___________

Case Title: Ir-192 Transmission
Description: 1 Ci Ir-192 thru 1 cm Steel
Geometry: 1 - Point



Dose Points

	X	Y	Z
# 1	1 cm	0 cm	0 cm
	0.4 in	0.0 in	0.0 in

Shields

Shield Name	Dimension	Material	Density
Shield 1	1.0 cm	Iron	7.86
Air Gap		Air	0.00122

Source Input
Grouping Method : Standard Indices
Number of Groups : 25
Lower Energy Cutoff : 0.015
Photons < 0.015 : Included
Library : ICRP-38

Nuclide	curies	becquerels
Ir-192	1.0000e+000	3.7000e+010

Buildup
The material reference is : Shield 1

Results

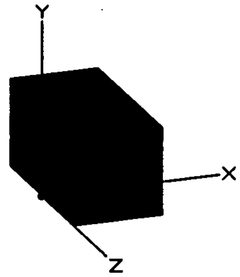
Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.015	2.032e+09	6.701e-187	2.145e-20	5.748e-188	1.840e-21
0.06	3.958e+09	2.604e+03	3.656e+03	5.173e+00	7.261e+00
0.08	1.093e+09	9.157e+04	1.460e+05	1.449e+02	2.310e+02
0.1	4.265e+07	2.299e+04	3.974e+04	3.517e+01	6.079e+01
0.15	6.682e+07	1.894e+05	3.602e+05	3.119e+02	5.932e+02
0.2	1.405e+09	7.539e+06	1.454e+07	1.331e+04	2.566e+04
0.3	5.248e+10	5.428e+08	9.909e+08	1.030e+06	1.880e+06
0.4	5.900e+08	9.112e+06	1.564e+07	1.776e+04	3.047e+04
0.5	1.900e+10	3.943e+08	6.401e+08	7.740e+05	1.256e+06
0.6	6.689e+09	1.756e+08	2.722e+08	3.428e+05	5.313e+05
0.8	1.137e+08	4.294e+06	6.222e+06	8.168e+03	1.183e+04
1.0	2.115e+07	1.054e+06	1.454e+06	1.942e+03	2.680e+03
1.5	4.598e+05	3.745e+04	4.764e+04	6.301e+01	8.015e+01
TOTALS:	8.749e+10	1.135e+09	1.942e+09	2.188e+06	3.739e+06

MicroShield v5.05 (5.05-00161)
AEA Technology QSA, Inc.

Page : 1
DOS File: SESTL.MS5
Run Date: May 23, 2007
Run Time: 9:06:32 AM
Duration: 00:00:00

File Ret: ____________
Date: ____________
By: ____________
Checked: ____________

Case Title: Se-75 Transmission
Description: 1 Ci Se-75 thru 1 cm Steel
Geometry: 1 - Point



Dose Points

	X	Y	Z
# 1	1 cm	0 cm	0 cm
	0.4 in	0.0 in	0.0 in

Shields

Shield Name	Dimension	Material	Density
Shield 1	1.0 cm	Iron	7.86
Air Gap		Air	0.00122

Source Input
Grouping Method : Standard Indices
Number of Groups : 25
Lower Energy Cutoff : 0.015
Photons < 0.015 : Included
Library : ICRP-38

Nuclide	curies	becquerels
Se-75	1.0000e+000	3.7000e+010

Buildup
The material reference is : Shield 1

Results

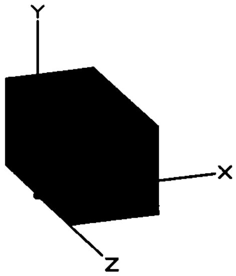
Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.015	2.093e+10	6.903e-186	2.210e-19	5.921e-187	1.895e-20
0.02	9.451e+06	1.223e-82	1.568e-22	4.236e-84	5.432e-24
0.06	4.088e+08	2.690e+02	3.776e+02	5.343e-01	7.499e-01
0.08	3.077e+06	2.577e+02	4.108e+02	4.078e-01	6.500e-01
0.1	7.846e+09	4.230e+06	7.311e+06	6.471e+03	1.119e+04
0.15	2.242e+10	6.355e+07	1.208e+08	1.047e+05	1.990e+05
0.2	5.567e+08	2.988e+06	5.762e+06	5.274e+03	1.017e+04
0.3	3.178e+10	3.288e+08	6.002e+08	6.237e+05	1.138e+06
0.4	4.203e+09	6.491e+07	1.114e+08	1.265e+05	2.171e+05
0.5	2.086e+05	4.328e+03	7.026e+03	8.496e+00	1.379e+01
0.6	1.560e+07	4.096e+05	6.350e+05	7.995e+02	1.239e+03
0.8	4.835e+04	1.826e+03	2.645e+03	3.473e+00	5.032e+00
TOTALS:	8.817e+10	4.649e+08	8.461e+08	8.673e+05	1.577e+06

MicroShield v5.05 (5.05-00161)
AEA Technology QSA, Inc.

Page : 1
DOS File: YBSTL.MS5
Run Date: May 23, 2007
Run Time: 9:07:19 AM
Duration: 00:00:00

File Ref: ____________
Date: ____________
By: ____________
Checked: ____________

Case Title: Yb-169 Transmission
Description: 1 Ci Yb-169 thru 1 cm Steel
Geometry: 1 - Point



Dose Points

	X	Y	Z
# 1	1 cm	0 cm	0 cm
	0.4 in	0.0 in	0.0 in

Shields

Shield Name	Dimension	Material	Density
Shield 1	1.0 cm	Iron	7.86
Air Gap		Air	0.00122

Source Input
Grouping Method : Standard Indices
Number of Groups : 25
Lower Energy Cutoff : 0.015
Photons < 0.015 : Included
Library : ICRP-38

Nuclide	curies	becquerels
Yb-169	1.0000e+000	3.7000e+010

Buildup
The material reference is : Shield 1

Results

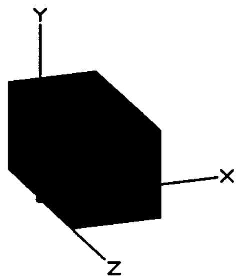
Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.015	1.678e+10	5.534e-186	1.771e-19	4.747e-187	1.519e-20
0.02	8.767e+07	1.135e-81	1.455e-21	3.930e-83	5.039e-23
0.05	5.399e+10	1.008e+02	1.307e+02	2.684e-01	3.482e-01
0.06	2.927e+10	1.926e+04	2.703e+04	3.825e+01	5.370e+01
0.1	8.105e+09	4.369e+06	7.552e+06	6.684e+03	1.155e+04
0.15	4.254e+09	1.206e+07	2.293e+07	1.986e+04	3.776e+04
0.2	2.159e+10	1.159e+08	2.235e+08	2.045e+05	3.944e+05
0.3	4.280e+09	4.428e+07	8.082e+07	8.399e+04	1.533e+05
0.4	6.509e+05	1.005e+04	1.725e+04	1.959e+01	3.362e+01
0.5	2.326e+06	4.826e+04	7.835e+04	9.474e+01	1.538e+02
0.6	3.617e+06	9.495e+04	1.472e+05	1.853e+02	2.873e+02
0.8	9.564e+04	3.612e+03	5.233e+03	6.870e+00	9.953e+00
TOTALS:	1.384e+11	1.768e+08	3.350e+08	3.154e+05	5.975e+05

MicroShield v5.05 (5.05-00161)
AEA Technology QSA, Inc.

Page : 1
DOS File: YBTRANS.MS5
Run Date: May 23, 2007
Run Time: 9:12:41 AM
Duration: 00:00:00

File Ref: ______
Date: ______
By: ______
Checked: ______

Case Title: Yb-169 Transmission
Description: 1 Ci Yb-169 thru 1 cm Lead
Geometry: 1 - Point



Dose Points

	X	Y	Z
# 1	1 cm	0 cm	0 cm
	0.4 in	0.0 in	0.0 in

Shields

Shield Name	Dimension	Material	Density
Shield 1	1.0 cm	Lead	11.34
Air Gap		Air	0.00122

Source Input
Grouping Method : Standard Indices
Number of Groups : 25
Lower Energy Cutoff : 0.015
Photons < 0.015 : Included
Library : ICRP-38

Nuclide	curies	becquerels
Yb-169	1.0000e+000	3.7000e+010

Buildup
The material reference is : Shield 1

Results

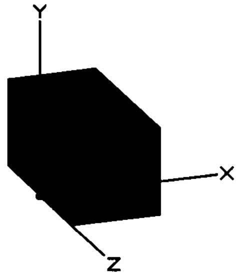
Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.015	1.678e+10	0.000e+00	1.754e-19	0.000e+00	1.504e-20
0.02	8.767e+07	0.000e+00	1.222e-21	0.000e+00	4.232e-23
0.05	5.399e+10	5.751e-29	2.050e-18	1.532e-31	5.462e-21
0.06	2.927e+10	4.710e-15	5.340e-15	9.356e-18	1.061e-17
0.1	8.105e+09	2.733e-19	2.294e-02	4.182e-22	3.509e-05
0.15	4.254e+09	1.758e-02	5.322e-02	2.894e-05	8.764e-05
0.2	2.159e+10	7.779e+03	1.052e+04	1.373e+01	1.857e+01
0.3	4.280e+09	1.418e+06	1.851e+06	2.690e+03	3.512e+03
0.4	6.509e+05	1.765e+03	2.367e+03	3.438e+00	4.612e+00
0.5	2.326e+06	1.662e+04	2.250e+04	3.263e+01	4.417e+01
0.6	3.617e+06	4.541e+04	6.118e+04	8.863e+01	1.194e+02
0.8	9.564e+04	2.330e+03	3.094e+03	4.431e+00	5.885e+00
TOTALS:	1.384e+11	1.492e+06	1.951e+06	2.833e+03	3.704e+03

MicroShield v5.05 (5.05-00161)
AEA Technology QSA, Inc.

Page : 1
DOS File: IRTRANS.MS5
Run Date: May 23, 2007
Run Time: 9:11:22 AM
Duration: 00:00:00

File Ref: ____________
Date: ____________
By: ____________
Checked: ____________

Case Title: Ir-192 Transmission
Description: 1 Ci Ir-192 thru 1 cm Lead
Geometry: 1 - Point



Dose Points

	X	Y	Z
# 1	1 cm	0 cm	0 cm
	0.4 in	0.0 in	0.0 in

Shields

Shield Name	Dimension	Material	Density
Shield 1	1.0 cm	Lead	11.34
Air Gap		Air	0.00122

Source Input
Grouping Method : Standard Indices
Number of Groups : 25
Lower Energy Cutoff : 0.015
Photons < 0.015 : Included
Library : ICRP-38

Nuclide	curies	becquerels
Ir-192	1.0000e+000	3.7000e+010

Buildup
The material reference is : Shield 1

Results

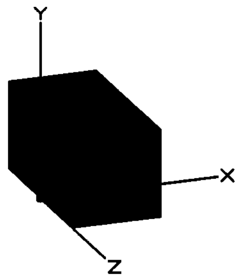
Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.015	2.032e+09	0.000e+00	2.124e-20	0.000e+00	1.822e-21
0.06	3.958e+09	6.370e-16	7.222e-16	1.265e-18	1.434e-18
0.08	1.093e+09	2.128e-04	2.564e-04	3.367e-07	4.057e-07
0.1	4.265e+07	1.438e-21	1.207e-04	2.200e-24	1.847e-07
0.15	6.682e+07	2.761e-04	8.360e-04	4.546e-07	1.377e-06
0.2	1.405e+09	5.061e+02	6.846e+02	8.933e-01	1.208e+00
0.3	5.248e+10	1.739e+07	2.270e+07	3.298e+04	4.305e+04
0.4	5.900e+08	1.600e+06	2.146e+06	3.117e+03	4.181e+03
0.5	1.900e+10	1.358e+08	1.838e+08	2.666e+05	3.608e+05
0.6	6.689e+09	8.398e+07	1.132e+08	1.639e+05	2.209e+05
0.8	1.137e+08	2.770e+06	3.679e+06	5.268e+03	6.997e+03
1.0	2.115e+07	7.745e+05	1.006e+06	1.428e+03	1.854e+03
1.5	4.598e+05	3.077e+04	3.763e+04	5.177e+01	6.330e+01
TOTALS:	8.749e+10	2.423e+08	3.265e+08	4.733e+05	6.378e+05

MicroShield v5.05 (5.05-00161)
AEA Technology QSA, Inc.

Page : 1
DOS File: SETRANS.MS5
Run Date: May 23, 2007
Run Time: 9:12:08 AM
Duration: 00:00:00

File Ref: ____________
Date: ____________
By: ____________
Checked: ____________

Case Title: Se-75 Transmission
Description: 1 Ci Se-75 thru 1 cm Lead
Geometry: 1 - Point



Dose Points

	X	Y	Z
# 1	1 cm	0 cm	0 cm
	0.4 in	0.0 in	0.0 in

Shields

Shield Name	Dimension	Material	Density
Shield 1	1.0 cm	Lead	11.34
Air Gap		Air	0.00122

Source Input
Grouping Method : Standard Indices
Number of Groups : 25
Lower Energy Cutoff : 0.015
Photons < 0.015 : Included
Library : ICRP-38

Nuclide	curies	becquerels
Se-75	1.0000e+000	3.7000e+010

Buildup
The material reference is : Shield 1

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.015	2.093e+10	0.000e+00	2.188e-19	0.000e+00	1.876e-20
0.02	9.451e+06	0.000e+00	1.317e-22	0.000e+00	4.562e-24
0.06	4.088e+08	6.578e-17	7.458e-17	1.307e-19	1.481e-19
0.08	3.077e+06	5.989e-07	7.216e-07	9.477e-10	1.142e-09
0.1	7.846e+09	2.646e-19	2.221e-02	4.048e-22	3.397e-05
0.15	2.242e+10	9.263e-02	2.805e-01	1.525e-04	4.619e-04
0.2	5.567e+08	2.006e+02	2.713e+02	3.540e-01	4.789e-01
0.3	3.178e+10	1.053e+07	1.375e+07	1.997e+04	2.608e+04
0.4	4.203e+09	1.139e+07	1.528e+07	2.220e+04	2.978e+04
0.5	2.086e+05	1.491e+03	2.018e+03	2.926e+00	3.961e+00
0.6	1.560e+07	1.959e+05	2.639e+05	3.824e+02	5.152e+02
0.8	4.835e+04	1.178e+03	1.564e+03	2.240e+00	2.975e+00
TOTALS:	8.817e+10	2.212e+07	2.930e+07	4.256e+04	5.638e+04

Section 6 - CRITICALITY EVALUATION

All parts of this section are not applicable. The Model 976 Series transport packages are not used for shipment of Type B quantities of fissile material.

Section 7 – Package Operations

Operation of the Model 976 Series transport packages must be in accordance with the operating instructions supplied with the transport package, per 10 CFR 71.87 and 71.89. All subsequent 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).

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

7.1 Package Loading

7.1.1 Preparation for Loading

The Model 976 Series packages must be loaded and closed in accordance with the following written procedures. Shipment of Type B quantities of radioactive material are authorized for sources specified in Section 7.1.1.1. Maintenance and inspection of the Model 976 Series packaging is in accordance with the requirements specified in Section 7.1.1.2.

7.1.1.1 Authorized Package Contents

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

Table 7.1.1a: Model 976 Series Package Information

Identification	Inner Shield(s)	Nuclide	Form	Maximum Capacity[1,2]	Maximum Weight
976A	855	Ir-192	Special Form Sources	1,000 Ci	136 kg (300 lb)
		Se-75		1,000 Ci	
		Yb-169		865 Ci	
976B	3015	Ir-192	Special Form Sources	350 Ci	86 kg (190 lb)
		Se-75		350 Ci	
		Yb-169		350 Ci	
976C	3056	Ir-192	Special Form Sources	1,250 Ci	86 kg (190 lb)
		Se-75		1,250 Ci	
		Yb-169		1,000 Ci	
976D	3018	Ir-192	Special Form Sources	500 Ci	86 kg (190 lb)
		Se-75		500 Ci	
		Yb-169		500 Ci	

Identification	Inner Shield(s)	Nuclide	Form	Maximum Capacity[1,2]	Maximum Weight
976E	3078	Ir-192	Special Form Sources	1,000 Ci	103 kg (226 lb)
		Se-75		1,000 Ci	
		Yb-169		1,000 Ci	
976F	1911	Ir-192	Special Form Sources	1,000 Ci	119 kg (263 lb)
		Se-75		1,000 Ci	
		Yb-169		1,000 Ci	

[1]For Iridium-192, the maximum capacity is based on the output curies which are determined by measuring the source output at 1 meter and expressing its activity in curies derived from the following: 0.48 R/hr-Ci Iridium-192 at 1 meter. (Ref: American National Standard N432-1980, "Radiological Safety for the Design and Construction of Apparatus for Gamma Radiography."). For Selenium-75 and Yb-169 the maximum capacity is based on the content curies contained in the radioactive source(s).
[2] For shipments of multiple radioisotopes in a single package, the sum of the ratios of the curie quantity of each loaded isotope to the maximum allowed curie quantity of that isotope (were that isotope the only contents of the package) must be less than or equal to unity.

7.1.1.2 Packaging Maintenance and Inspection Prior to Loading

7.1.1.2.a Instructions for the 855 Shield Container

1. Ensure all markings are legible.

2. Inspect the container for signs of significant degradation. Ensure all welds are intact, the container is free of heavy rust and cracks/damage to the steel housing which breaches the container.

3. Ensure the lid eyebolt is straight and undamaged, and that the seal gasket is present and intact.

4. Ensure that the 3/8 inch cover bolts engage in the container threaded holes and secure the lid to the container body.

5. Remove the cover and inspect the lock holder assemblies and ensure these are securely attached to the container body and that the seal wires on the lock holder mechanisms are present and intact. Ensure the lock plungers operate from the lock to the open positions using the lock plunger key.

6. Insert the source check gauge into the source tubes and ensure the gauge can be fully inserted, without obstruction, to the mark on the source check gauge for all source tubes.

7. If the container fails any of the inspections in steps 7.1.1.2.a.1-6, remove the container from use until it can be brought into

compliance with the Type B certificate.

7.1.1.2.b Instructions for the 3056 Shield Container

1. Ensure all markings are legible.

2. Inspect the container for signs of significant degradation. Ensure all welds are intact, the container is free of heavy rust and cracks/damage to the steel bracing. Examine the lead and ensure the shield pot has not been damaged in a way that significantly reduces the container shielding.

3. Ensure the source tube caps can be fully threaded onto the source tubes.

4. Ensure that the M12 x 1.75 mm retaining nut can be fully threaded onto the container threaded rods and secure the lid to the container without gaps. Ensure that the M10 x 20 mm socket head screws can be full threaded onto the container.

5. Insert the source check gauge into the source tubes and ensure the gauge can be fully inserted, without obstruction, to the mark on the source check gauge for all source tubes.

6. If the container fails any of the inspections in steps 7.1.1.2.b.1-5, remove the container from use until it can be brought into compliance with the Type B certificate.

7.1.1.2.c Instructions for the 3015 Shield Container

1. Ensure all markings are legible.

2. Inspect the container for signs of significant degradation. Ensure all welds are intact, the container is free of heavy rust and cracks/damage to the steel housing which breaches the container. Examine the lead accessible from the bottom of the container and ensure the shield pot has not been damaged in a way that significantly reduces the container shielding.

3. Ensure the shield inserts, shield plug and lid can be assembled onto the container and secured with the lid hardware.

4. Ensure that the M10 nuts and washers can be fully threaded into the container bolts and secure the shield lid to the container without gaps.

5. Ensure that the shield inserts are empty and free of debris or sources.

6. If the container fails any of the inspections in steps 7.1.1.2.c.1-5, remove the container from use until it can be brought into compliance with the Type B certificate.

7.1.1.2.d Instructions for the 3018 Shield Container

1. Ensure all markings are legible.

2. Inspect the container for signs of significant degradation. Ensure all welds are intact, the container is free of heavy rust and cracks/damage to the steel housing which breaches the container. Examine the lead accessible from the bottom of the container and ensure the shield pot has not been damaged in a way that significantly reduces the container shielding.

3. Ensure the source tube caps can be fully threaded onto the source tubes.

4. Ensure that the M8 x 20 mm long socket head cap screws can be fully threaded into the container and secure the shield cover plate to the container without gaps. Ensure the 0.125" steel shim (spacer) is in place under each end of the retaining strap and that the M10 bolts are full threaded onto the container.

5. Insert the source check gauge into the source tubes and ensure the gauge can be fully inserted, without obstruction, to the mark on the source check gauge for all source tubes.

6. If the container fails any of the inspections in steps 7.1.1.2.d.1-5, remove the container from use until it can be brought into compliance with the Type B certificate.

7.1.1.2.e Instructions for the 3078 Shield Container

1. Ensure all markings are legible.

2. Inspect the container for signs of significant degradation. Ensure all welds are intact, the container is free of heavy rust and cracks/damage to the steel housing which breaches the container.

3. Ensure the shield plug and lid can be assembled onto the container and secured with the lid hardware.

4. Ensure that the M8 x 0.8 in long hex cap screws can be fully threaded into the container and secure the shield lid to the container without gaps.

5. Ensure that the shield cavity is empty and free of debris or sources.

6. If the container fails any of the inspections in steps 7.1.1.2.e.1-5, remove the container from use until it can be brought into compliance with the Type B certificate.

7.1.1.2.f Instructions for the 1911 Shield Container

1. Ensure all markings are legible.

2. Inspect the container for signs of significant degradation. Ensure all welds are intact, the container is free of heavy rust and cracks/damage to the steel housing which breaches the container.

3. Ensure the lid eyebolt is undamaged and that it threads freely into the shield container lid.

4. Ensure that the M8 cover bolts engage in the container threaded holes and secure the lid to the container body.

5. Inspect the shield insert plug and ensure that it inserts freely over the lower shield insert and allows attachment of the shield container lid after insertion. Inspect the lower shield insert and ensure that the insert cavity is free of foreign objects, sources and obstruction.

6. If the container fails any of the inspections in steps 7.1.1.2.f.1-5, remove the container from use until it can be brought into compliance with the Type B certificate.

7.1.1.2.g Instructions for the Drum and Cork Inserts

1. Ensure all markings are legible.

2. Inspect the drum, lid and lid closure band for signs of significant degradation. Ensure all welds are intact, the container is free of heavy rust and cracks/damage to the steel.

3. Ensure the cork inserts are intact and no damage that would allow significant shield container movement during transport.

4. Ensure that the M8 x 130 mm long bolt can be fully threaded into the lid closure band nut. Ensure the lid closure band can fit around the lid and drum when these components are assembled.

5. Ensure that the four 3/8-16 x ¾ inch long lid closure bolts can be inserted through the drum body sides into the lid closure blocks and that the bolts fully thread into the lid closure blocks.

6. If the container fails any of the inspections in steps 7.1.1.2.g.1-5, 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. None of the shield configurations for the Model 976 Series packages are approved for wet loading.***

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

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

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

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

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

All necessary safety precautions and regulations must be observed to ensure safe transfer of the radioactive material.

2. Model 855 Shield Container

   i. Using remote handling techniques, load the source assemblies so that they are fully inserted into the source tubes with the active end of the source assembly inserted first. Once loaded ensure the plunger lock(s) are depressed and the key removed.

   ii. Secure the shield cover to the container using eight 3/8-16 by ¾ inch long hex head screws. Tighten the screws so that no gap exists between the screw heads, lid or container.

3. Model 3056 Shield Container

   i. Using remote handling techniques, load the source assemblies so that they are fully inserted into the source tubes with the active end of the source assembly inserted first. Attach the tube caps over the source assemblies by fully threading the caps onto the source tubes. Tighten the tube caps by light finger pressure only.

   ii. Secure the shield cover lid to the container using the M12 x 1.75 mm retaining nut. Tighten the retaining nut so that no gap exists between the retaining nut, lid or container.

4. Model 3015 Shield Container

   i. Using remote handling techniques, load the source(s) into the tungsten capsule holder within the DU shield insert contained in the shield cavity. (Contact QSA Global, Inc. for tungsten shield inserts which comply with the Type B transport package requirements for use with specific special form sources.)

   ii. Insert the shield plug fully into the shield cavity so that the bottom of the shield plug rests on the step in the shield cavity.

iii. Secure the shield lid to the container using the M10 nuts and washers to the container bolts. Tighten the nuts so that no gap exists between the nut, washer, lid or container.

5. Model 3018 Shield Container

   i. Using remote handling techniques, load the source assemblies so that they are fully inserted into the source tubes with the active end of the source assembly inserted first. Attach the tube caps over the source assemblies by fully threading the caps onto the source tubes. Tighten the tube caps by light finger pressure only.

   ii. Secure the shield cover plate to the container using the M8 x 20 mm long socket head cap screw. Tighten the socket head cap screw so that no gap exists between the screw, lid or container.

6. Model 3078 Shield Container

   i. Using remote handling techniques, load the source(s) within the shield cavity.

   ii. Insert the shield plug fully into the shield cavity so that the bottom of the shield plug rests on the step in the shield cavity.

   iii. Secure the shield lid to the container using the four M8 x 0.8 in long hex head lid screws. Tighten the hex head screws so that no gap exists between the screw, lid or container.

7. Model 1911 Shield Container

   i. Identify the shield insert configuration required for shielding of the sources to be loaded into the container (contact the manufacturer if unsure on the appropriate shield insert configuration for use with the special form sources to be transported) and confirm that the shield container is configured with the proper shield insert (e.g., depleted uranium, tungsten or lead).

ii. Using remote handling techniques, load the source(s) within the shield cavity.

iii. Insert the shield plug fully into the shield cavity so that the bottom of the shield plug rests on the top of the shield insert.

iv. Secure the shield lid to the container using the four M8 x 25 mm long hex head bolts. Tighten the hex head bolts so that no gap exists between the bolt, lid or container.

7.1.2.1.d Load the shield into the 976 drum and applicable bottom cork inserts. See drawings for the applicable cork inserts for the specific shield container to be transported in the 976 drum.

7.1.2.1.e Insert the applicable top cork inserts into the 976 drum. See drawings for the applicable cork inserts for the specific shield container to be transported in the 976 drum.

7.1.2.1.f Place the drum lid onto the drum. Insert the four 3/8-16 x ¾ inch long lid closure bolts through the drum body sides into the lid closure blocks. Tighten the lid closure bolts so that no gap exists between the bolt and the drum. Secure the lid to the drum using the drum lid closure band. Torque the drum bolt to 10 ft-lbs (+2, -0 ft-lbs) prior to transport or assure there is a 0.75-1.25 inch (19-32 mm) gap between the lid closure band sides.

7.1.3 Preparation for Transport

(Reference:

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

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

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

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

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

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

7.2 Package Unloading

7.2.1 Receipt of Package from Carrier

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

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

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

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

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

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

7.2.1.2.d Inspect the outer container for physical damage or leaking. If the package is damaged or leaking or it is suspected that the package may have leaked or been damaged, restrict access to the package. As soon as possible, contact the

Radiation Safety Office to perform a full assessment of the package condition and take necessary follow-up actions.

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

7.2.2 Removal of Contents

7.2.2.1 Unload the Model 976 in accordance with the applicable licensing provisions for the user's facility related to radioactive material handling.

7.2.2.2 Remove the inner shield container from the Model 976 drum and cork inserts.

7.2.2.3 Place the shield container in a shielded cell/enclosure capable of holding the maximum isotope capacity of this container. (Note: Transfer of source wire assemblies for the Model 855, 3018 and 3056 style shields may be performed outside of shielded cells/enclosures using radiographic controls and guide tubes if authorized under the approval of a licensing authority.)

7.2.2.4 Individual Container Instructions for Unloading

7.2.2.4.a Model 855 Shield Container

1. Remove the eight 3/8-16 by ¾ inch long hex head cover screws and remove the cover lid from the container.

2. Use remote handling techniques to unload each source tube and transfer the source to an alternate, shielded storage location. Prior to source transfer, unlock the plunger lock on the lock block assembly for the source tube to be unloaded.

3. Repeat Step 7.2.2.4.a.2 for all sources in the shield container.

7.2.2.4.b Model 3056 Shield Container

1. Remove the M12 x 1.75 mm retaining nut from the container lid. Remove the M10 screws.

2. Use remote handling techniques to unload each source tube and transfer the source to an alternate, shielded storage location. Prior to source transfer, remove the source tube cap for the source tube to be unloaded.

3. Repeat Step 7.2.2.4.b.2 for all sources in the shield container.

7.2.2.4.c Model 3015 Shield Container

1. Remove the M10 nuts and washers from the container bolts and remove the container lid.

2. Use remote handling techniques to remove the shield plug from the container cavity. Remotely unload each source and transfer the source to an alternate, shielded storage location.

7.2.2.4.d Model 3018 Shield Container

1. Remove the M8 x 20 mm socket head cap screw from the container lid. Remove the container lid.

2. Use remote handling techniques to unload each source tube and transfer the source to an alternate, shielded storage location. Prior to source transfer, remove the source tube cap for the source tube to be unloaded.

3. Repeat Step 7.2.2.4.d.2 for all sources in the shield container.

7.2.2.4.e Model 3078 Shield Container

1. Remove the M8 x 0.8 in hex head lid screws and remove the container lid.

2. Use remoté handling techniques to remove the shield plug from the container cavity. Remotely unload each source and transfer the source to an alternate, shielded storage location.

7.2.2.4.f Model 1911 Shield Container

1. Install the M10 eyebolt into the lid. Remove the M8 x 25 mm hex head lid screws and remove the container lid.

2. Use remote handling techniques to remove the shield plug from the container cavity. Remotely unload each source and transfer the source to an alternate, shielded storage location.

7.3 Preparation of Empty Package for Transport

(Reference:

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

In the following instructions, an *empty* transport package refers to a Model 976 Series 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:

7.3.1.1 For Models 3015, 3078 and 1911

7.3.1.1.a Place the shield container in a shielded cell/enclosure capable of holding the maximum isotope capacity of this container. Remove the cover. Remove any shield plugs used in that shield container and visually inspect the container for any source capsules.

7.3.1.1.b Use remote manipulators, mirrors, and radiation monitors if necessary, inspect the container to verify that it is empty.

7.3.1.1.c Once the shield cavity is determined to be empty, place all shield plugs/inserts back into the container and install the cover.

7.3.1.2 For Models 855, 3018 and 3056

7.3.1.2.a Remove all source tube caps and visually inspect the source tubes for the presence of a source assembly.

7.3.1.2.b For the Model 855, unlock the lock plungers. For all models, insert the source check gauge into each source tube. If the source check gauge cannot be inserted past the indicator mark on the gauge, the source tube may not be empty. Contact QSA Global, Inc. for further assistance after securing the source tube.

7.3.1.2.c Once the source tubes are determined to be empty, place all shield caps back onto the source tubes. For the Model 855, engage the plunger lock assemblies. Install the shield container covers on all model containers..

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

7.3.3 Assure that the levels of removable radioactive contamination on the inside surface of the shield container does not exceed 400 Bq/cm^2 (when averaged over 300 cm^2).

7.3.4 When it is confirmed that the Model 976 Series transport package is empty, prepare the transport package for shipment. Survey the assembled package to ensure the external surface radiation level does not exceed 5 μSv/h.

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

7.4 Other Operations

7.4.1 Package Transportation By Consignor

(Reference:

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

Persons transporting the Model 976 Series transport package in their own conveyances should comply with the following:

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

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

7.4.2 Emergency Response

(Reference:

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

In the event of a transport emergency or accident involving this package, follow the guidance contained in "2004 Emergency Response Guidebook: A Guidebook for First Responders During the Initial Phase of a Dangerous Goods/Hazardous Materials Incident", or equivalent guidance documentation.

Reference: "2004 Emergency Response Guidebook: A Guidebook for First Responders During the Initial Phase of a Dangerous Goods/Hazardous Materials Incident"

7.5 Appendix

Not Applicable.

Section 8 - ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

8.1 Acceptance Test

8.1.1 Visual Inspections and Measurements

8.1.1.1 New Fabrication

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

    i. The transport package was assembled properly to the applicable drawing.

    ii. Evaluate each shield container for shielding integrity when used in the applicable Model 976 Series assembly to ensure the transport dose rate requirements are met when the container is loaded to capacity.

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

    iv. 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.

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

8.1.1.2 Dedication of Pre-existing Shield Components

a. Evaluate all 855, 3056, 3015, 3018, 3078 and 1911 shield containers prior to first use in a Model 976 Series package to assure the following:

    i. Shield containers comply with the applicable drawing. (This evaluation will be performed to assess all components not requiring disassembly of the shield container such as external construction, overall dimensions, hardware compliance, weld integrity, etc. All items critical to safety are specified under the QSA Global, Inc. USNRC approved QA Program.)

ii. Evaluate each shield container for shielding integrity when used in the applicable Model 976 Series assembly to ensure the transport dose rate requirements are met when the container is loaded to capacity.

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

8.1.2 Weld Examinations

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

8.1.3 Structural and Pressure Tests

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

Prior to first use as part of a 976 Series 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 decrease 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 microcuries. The source capsules will also be subjected to leak tests under ISO9978:1992(E) (or more recent editions). The source capsules are not used if they fail any of these tests.

8.1.5 Component and Material Tests

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

8.1.6 Shielding Tests

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

8.1.7 Thermal Tests

Not applicable. The source content of the Model 976 Series packages has minimal effect on the package surface temperature and therefore no additional testing is necessary to evaluate thermal properties of the packaging.

8.1.8 Miscellaneous Tests

Not applicable.

8.2 Maintenance Program

8.2.1 Structural and Pressure Tests

Not applicable. Material certification 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 976 Series packaging system is not designed to require increased or decrease operating pressures to maintain containment during transport, therefore pressure tests of package components prior to individual shipment is not required.

8.2.2 Leakage Tests

As described in Section 8.1.4, "Leakage Tests," the radioactive source assembly is leak-tested at manufacture. In addition, the sources are leak tested in accordance with that Section at least once every six months thereafter if being transported to ensure that removable contamination is less than 0.005 microcuries. Also a contamination wipe is performed of the source J-tubes of the Model 855, 3056 and 3018 shields whenever the shield is returned to the manufacturer (typically the shield is shipped to a customer with new sources and 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 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 976 Series 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 – IAEA No. TS-R-1 (ST-1, Revised) 1996 Edition (Revised) Requirements not Otherwise Addressed – Section VI

9.1 General Package Design Requirements

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

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

The exterior surface of the 976 Series packages is comprised of a stainless steel drum. The drum incorporates a clamp ring but does not include handles which could protrude from the side surfaces. The materials and fabrication of the external drum can be easily decontaminated if necessary.

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

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

The exterior surface of the 976 Series packages is comprised of a stainless steel drum. The materials and fabrication of the external drum are water resistant and prevent, as far as practicable, the collection and retention of water.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

9.3 Requirements for Type B(U) Packages

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

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

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

9.4 Appendix

Not Applicable.