spec
SOURCE OF QUALITY

January 13, 2011

Mr. Pierre Saverot
NMSS, SFST- Licensing Branch
Mail Stop EBB-3D-02M
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Document Control Desk
Director, Spent Fuel Project Office
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: NRC RAI dated March 10, 2010, Certificate of Compliance No. 9263

Dear Sirs,

Attached please find Source Production and Equipment Co., Inc.'s (SPECs) response to NRC's requests for additional information (RAIs) regarding SPEC's application for revision of Certificate of Compliance No. 9263. The attachment includes specific replies to each RAI, revised drawings, revised pages for the Safety Analysis Report (SAR), and appendices. I have also enclosed two electronic copies of this document for Mr. Saverot.

Should you or any staff or reviewer have any questions regarding our application or the attached report please do not hesitate to contact me. We are in urgent need of the new certificate and would like to work with NRC to reduce the review time and cost of the review where possible.

Sincerely,

Kelley Richardt

Kelley Richardt
504-464-9471
kelley@spec150.com

W:\REGLATORY\APPLICAT\RAI, 2009, 150 300\150, RAIexplanationr7.wpd

NMSS24

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113 Teal Street St. Rose, LA 70087-9691 Phone 504/464-9471 FAX 504/467-7685 Website: www.spec150.com

ADDITIONAL INFORMATION PER NRC REQUEST FOR SPEC-150, CERTIFICATE 9263		
	RAI	SPEC's Position / Response
1	Specify which joins are structural and which are non-structural on the licensing drawings. The licensing drawings should unambiguously describe the welds on the package.	Drawing 15B000 contained notes describing the welding and inspection for "structural joints". These notes were removed from 15B000, and added to drawing 15B002A which depicts both Important to Safety (ITS) and non-ITS welds. The welding and inspection specifications for each weld are described on the drawing with arrows pointing to each weld. See Appendix A for revised drawings.
2	Clarify why internal structural joins do not undergo dye-penetrant testing.	The internal welds are not ITS welds. The internal welds are not 360 degree welds, they don't attach the bulkheads to the housing all the way around, the outer bulkhead welds do. They function as a construction aid by holding the internal components in position for the external 360 degree welds. Therefore, the primary reason to perform a visual weld inspection of the internal welds is to detect warpage, misalignments, adverse affects on the base weld material, improper fit up, and other unacceptable weld attributes. The outer 360 degree welds that attach the bulkheads to the housing are dye penetrant tested.
3	Citing a standard industrial code, specify that all thermal metal joins will be examined. State the examination methodologies and the acceptance criteria used in the weld fabrication by citing a standard industrial code.	Drawing 15B002A was revised to point to each specific weld and specify the corresponding inspection method depending on whether the weld is important to safety. Drawing notes were added as follows: - This important to safety (ITS) weld is fabricated and liquid penetrant inspected in accordance with ASME Section VIII, Division I, or, fabricated and inspected in accordance with AWS D1.9. - This weld is not ITS, and is fabricated and visually inspected in accordance with ASME Section VIII, Division I, or, fabricated and visually inspected in accordance with AWS D1.9. See Appendix A for revised drawings.
3	Ensure that Section 8.1.2 of the application is consistent with the licensing drawings.	Section 8.1.2 was revised to be consistent with the above RAI response. See Appendix C: Revised page 73.
4	Specify ASME code of construction (e.g., Section VIII, Division I) which is used as the basis for welding for thermal metal joins on the package.	Welds are performed in accordance with the applicable requirements of ASME Section VIII, Division I using procedures and welders qualified in accordance with ASME Section IX. We plan to transition both fabrication and welding to the AWS code later. See Appendix A for revised drawings.

ADDITIONAL INFORMATION PER NRC REQUEST FOR SPEC-150, CERTIFICATE 9263		
	RAI	SPEC's Position / Response
5	Clarify why the ASTM E-165 is used as the basis for the examination procedures of thermal metal joins on the package, and not Section V of the ASME code.	The statement on drawing 15B002A has been revised to state that dye penetrant weld examinations are performed in accordance with ASME Section VIII, Division I. ASME Section V, Article 24 covers Liquid Penetrant Standards, and includes SE-165 (which states that it is identical to ASTM E 165).
6	Specify a standard industrial code, e.g., ASTM or ASME, which mandates the minimum mechanical properties and level of fabrication quality for all materials used to construct components that are safety-related, with the exception of zirconium and depleted uranium alloys.	The drawings were revised by adding a note stating "See SAR for Material Specification". We have included an Appendix to Section 5.0 of the SAR listing the material specifications for materials related to package safety, with the exception of zirconium and depleted uranium alloys. The depleted uranium shields are surveyed to ensure that they shield properly. They are described in the drawings as being at least 99% pure, with a minimum density of 18.3 g/cc, and weight of 35 to 37 1/4 pounds. See Appendix A for revised drawings and Appendix D for material specifications.
7	Provide the safety classification of the polymer foam used in the package. Section 5.4.1 of NUREG/CR-6407 lists impact limiters as Category A items.	SPEC will withdraw our request to remove the QA Classification from the shield drawing. All drawing notes and the description of the s-tube have been removed from the drawing as they are not similarly classified. The polymer foam in the SPEC-150 does not function as an impact limiter. See Appendix A for revised drawings.
8	Provide dimensional information of the join near component 3 on licensing drawing 15B002A (this is on the previous, not current revision) labeled TMJ. Clarify if this join is welded or brazed.	Withdrawn, see Conversation Record (ML103210646).
9	State in Section 7.1.1 that visual inspection of exposed fasteners and welds will occur during preparation for loading.	SPEC will revise Section 7.1.1 to require that the exposed fasteners and welds be checked prior to loading. See Appendix C, revised page 39.
10	Specify a minimum weight and density for the depleted uranium shield on sheet 1 of drawing B150008. Needed to determine compliance with 10 CFR 71.31(a)(5)(iii) ("internal and external structures supporting or protecting receptacles").	Drawing 15B008 has been revised to add a minimum density of 18.3g/cc and a minimum weight of 34 pounds. As these requirements are new, they are not applicable to SPEC-150 shields manufactured prior to the issue date of USA/9263/B(U)-96, revision 8. Shields manufactured prior to that date must continue to meet the conditions of revision 7 and prior package approvals, and are inspected and surveyed to verify acceptability. See Appendices A and D.

ADDITIONAL INFORMATION PER NRC REQUEST FOR SPEC-150, CERTIFICATE 9263		
RAI		SPEC's Position / Response
11	Provide the citation for the source from which the gamma ray constants were taken.	SPEC calculated the attenuated exposure rate and a report with the technical basis is submitted with our application. See Appendix E.
12	Reword the discussion in the second paragraph of Section 5.2 (shielding before/after normal conditions tests) in the SAR (page 34) and the similar discussion on page 36, Section 5.3 (after hypothetical accident conditions) dealing with the justification for not performing measurements at one meter.	See Appendix B for additional information demonstrating the ability of the SPEC-150 to meet external radiation standards in normal and accident conditions. This report is intended to be an Appendix to section 5.0 of our application. Appendix C contains the revised SAR pages.
	10 CFR 71.43(f) (no loss in normal conditions),	This information is presented in Table 3 of Appendix B.
	10 CFR 71.47 (external radiation standards) when normally prepared for transit of less than 200 mrem/hr at the surface	This information is presented in Tables 1 and 2 of Appendix B.
	and the transport index not exceeding 10,	This information is presented in Tables 1 and 2 of Appendix B.
	10 CFR 71.51(a)(1) (no loss in normal conditions) and	This information is presented in Table 3 of Appendix B.
	(a)(2)(<1rem at 1 meter in hypothetical accident conditions) .	This information is presented in Tables 1 and 5 of Appendix B.
13	Justify the use of an 8-curie or 4-curie source for the normal conditions of transport and hypothetical accident conditions tests. The standard cited in the SAR (ANSI N432-1980 for measuring shielding efficiency) specifies activity of the source used in the test should be within a factor of 10 of the limiting content.	See Appendix B for information demonstrating the ability of the SPEC-150 to meet external radiation standards in normal and accident conditions. This report is intended to be an Appendix to section 5.0 of our application.
14	Explain / justify the difference between the "Maximum" column in the first table in Section 5.1 (package shielding) and the "Maximum Before" column in the table in Section 5.2 (pre normal conditions testing).	See Appendix B for information demonstrating the ability of the SPEC-150 to meet external radiation standards in normal and accident conditions. This report is intended to be an Appendix to section 5.0 of our application.

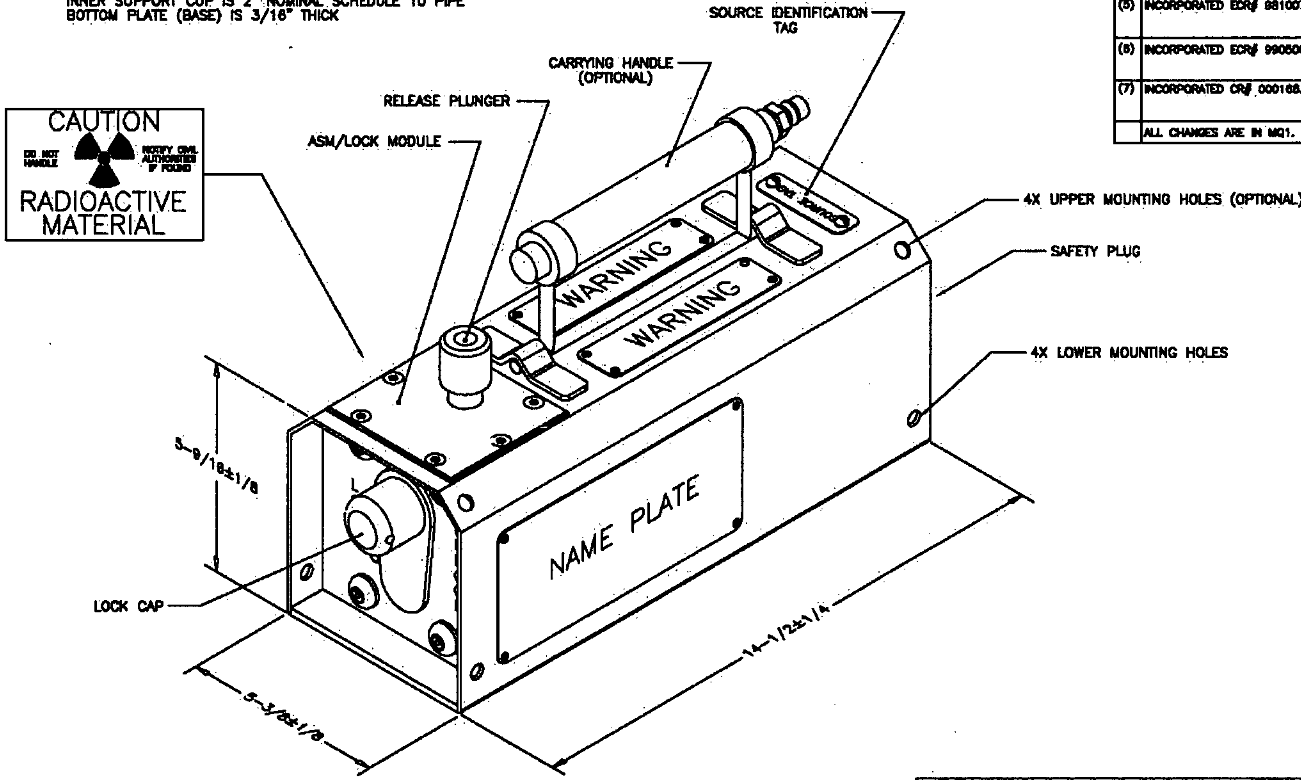
Appendix A: Revised drawings.

Drawing	Rev	Description of Changes from Current Certificate
15B000	8	Removed fabrication and inspection notes, they are now described on 15B002A. Added supplemental shielding note from 15B008 rev 5, added maximum weight and materials from 15B002A rev 6. Added note to see SAR for material specifications.
15B002A	7	Added fabrication and inspection notes for ITS and non-ITS welds, added material specifications for safety related components.
15B008	6	Removed notes that did not specifically describe the depleted uranium shield. Added minimum weight and density.
19B005	1	Added material specifications for safety related components, the source assembly lock, device lock, lock module housing and lock module faceplate.
19B006	1	Added material specifications for safety related components, the source assembly lock and device lock.
(190909)	(0)	(No changes needed, original drawing was not replaced and is not appended.)

CONTROLLED COPY NO			
REVISIONS			
REV	DESCRIPTION	DATE	APPROVED
(1)	SEE QA FILE FOLDER 15B000.	3-6-95 3-6-95 3-6-95	S. BYRD KC RDD
(2)	SEE QA FILE FOLDER 15B000.	4-5-95 4-5-95 4-5-95	S. BYRD KC RDD
(3)	SEE QA FILE FOLDER 15B000.	4-14-95 4-14-95 4-14-95	S. BYRD KC RDD
(4)	SEE QA FILE FOLDER 15B000.	9-21-95 9-21-95 9-21-95	S. BYRD KC RDD
(5)	INCORPORATED ECR# 981007-02.	10-15-98 4-20-99 4-20-99	S. BYRD RAM RDD
(6)	INCORPORATED ECR# 990506-04	5-6-99 5-6-99 5-6-99	S. BYRD RAM RDD
(7)	INCORPORATED CR# 000166.	3-10-09 3-11-09 3-11-09	KP KR KR
	ALL CHANGES ARE IN MQ1.	MQ1	MQ1

STATEMENTS OF FABRICAITON:

1. SUPPLEMENTAL SHIELDING, IF NEEDED TO MEET NORMAL CONDITION DOSE RATE LIMITS, IS ATTACHED TO THE SHIELD OR OTHER PACKAGE COMPONENTS USING ALUMINUM EPOXY POTTING COMPOUND. THE SUPPLEMENTAL SHIELDING CONSISTS OF ONE POUND OR LESS OF DEPLETED URANIUM, TUNGSTEN OR LEAD.
2. MAXIMUM WEIGHT: 53.5 LBS.
3. SEE SAR FOR MATERIAL SPECIFICATIONS
   HOUSING COVER IS 3/32" THICK
   OUTLET END SUPPORT CUP & BULKHEADS ARE 1/8" THICK
   INNER SUPPORT CUP IS 2" NOMINAL SCHEDULE 10 PIPE
   BOTTOM PLATE (BASE) IS 3/16" THICK

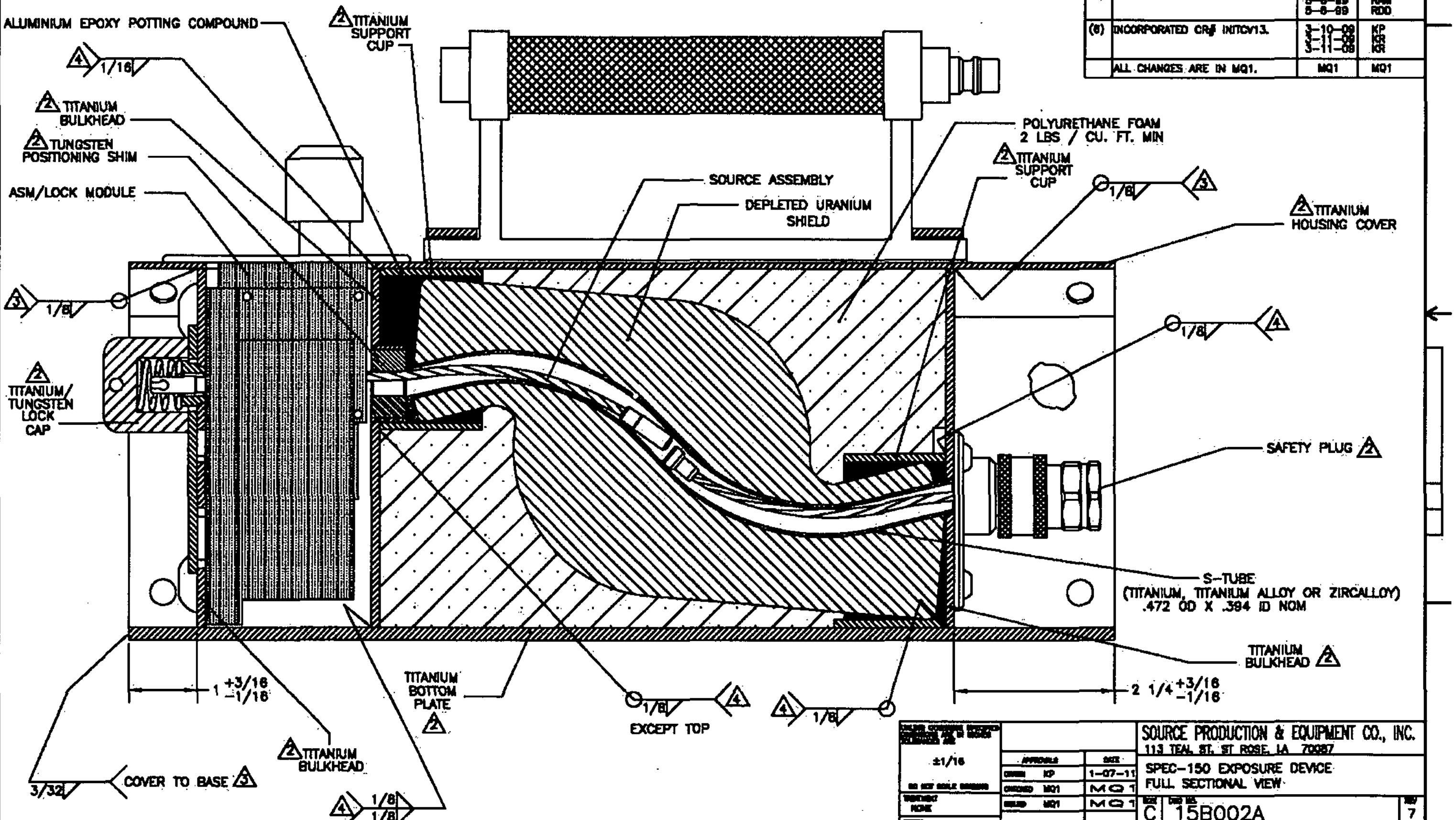


AS NOTED DO NOT SCALE DRAWING	APPROVALS	DATE	SOURCE PRODUCTION & EQUIPMENT CO., INC. 113 TEAL ST, ST ROSE, LA 70087
	DRAWN KP	1-06-11	SPEC-150 TYPE B(U) PACKAGE ISOMETRIC VIEW
	CHECKED MQ1	MQ1	
TREATMENT NONE	ISSUED MQ1	MQ1	C 15B000 REV 8
FINISH NONE	QA CLASS N/A		SCALE: NTS | 00063308 | SHEET 1 of 1

CONTROLLED COPY NO			
REVISIONS			
REV	DESCRIPTION	DATE	APPROVED
(1)	SEE QA FILE FOLDER 15B002A.	3-10-95 3-10-95 3-10-95	S. BYRD KC RDD
(2)	SEE QA FILE FOLDER 15B002A.	4-13-95 4-13-95 4-13-95	S. BYRD KC RDD
(3)	SEE QA FILE FOLDER 15B002A.	4-19-95 4-19-95 4-19-95	S. BYRD KC RDD
(4)	INCORPORATED ECR# 981007-04	10-12-98 4-20-99 4-20-99	S. BYRD RAM RDD
(5)	INCORPORATED ECR# 990506-03	5-6-99 5-6-99 5-6-99	S. BYRD RAM RDD
(6)	INCORPORATED CR# INITCV13.	3-10-09 3-11-09 3-11-09	KP KR KR
	ALL CHANGES ARE IN MQ1.	MQ1	MQ1

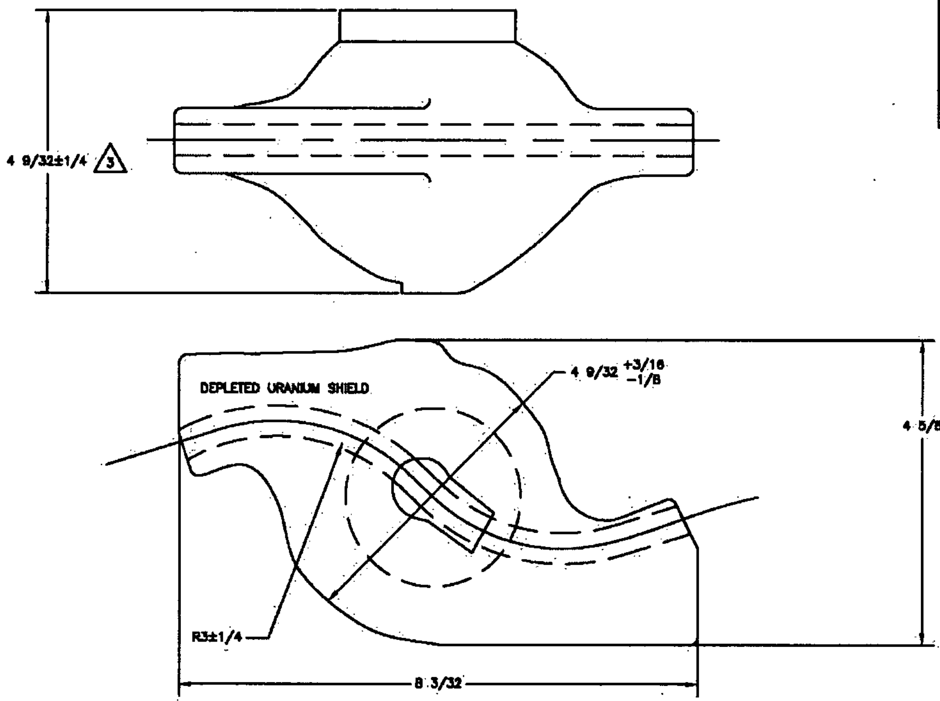
NOTES:

1. REMOVED.
2. SEE SAR FOR MATERIAL SPECIFICATIONS.
3. IMPORTANT TO SAFETY (ITS) WELDS ARE FABRICATED AND LIQUID PENETRANT INSPECTED IN ACCORDANCE WITH THE APPLICABLE REQUIREMENTS OF ASME SECTION VIII, DIVISION I, OR, FABRICATED AND INSPECTED IN ACCORDANCE WITH AWS D1.9.
4. THIS WELD IS NOT ITS, AND IS FABRICATED AND VISUALLY INSPECTED IN ACCORDANCE WITH ASME SECTION VIII, DIVISION I, OR, FABRICATED AND VISUALLY INSPECTED IN ACCORDANCE WITH AWS D1.9.



±1/16	APPROVALS	DATE	SOURCE PRODUCTION & EQUIPMENT CO., INC. 113 TEAL ST. ST ROSE, LA 70087
	DRAWN KP	1-07-11	SPEC-150 EXPOSURE DEVICE FULL SECTIONAL VIEW
DO NOT SCALE DRAWING	CHECKED MQ1	MQ1	
TREATMENT NONE	ISSUED MQ1	MQ1	SIZE C DWG NO. 15B002A REV 7
FINISH NONE	QA CLASS N		SCALE: NTS 00062707 SHEET 1 OF 1

CONTROLLED COPY NO			
REVISIONS			
REV	DESCRIPTION	DATE	APPROVED
(1)	SEE QA FILE FOLDER 15B008.	3-1-95 3-1-95 3-1-95	S. BYRD KC RDO
(2)	SEE QA FILE FOLDER 15B008.	4-13-95 4-13-95 4-13-95	S. BYRD KC RDO
(3)	INCORPORATED ECR# 981007-01	10-7-98 4/20/99 4/20/99	S. BYRD RAM RDD
(4)	INCORPORATED ECR# 990506-02	5/6/99 5/6/99 5/6/99	S. BYRD RAM RDD
(5)	INCORPORATED CR# 000198.	3-10-09 3-11-09 3-11-09	KP [illegible] [illegible]
	ALL CHANGES ARE IN MQ1.	MQ1	MQ1



NOTES:

1. WEIGHT: 34 TO 37-1/4 LBS.
2. MATERIALS OF CONSTRUCTION:
DEPLETED URANIUM, MINIMUM 99% PURE, DENSITY MINIMUM 18.3 g/cc.

3 THE TOLERANCE IS TO ALLOW FOR VARIATIONS IN THE HOT TOP.

FRACTIONS ±1/8
DO NOT SCALE DRAWING
NONE
NONE

APPROVALS		DATE
DRAWN	KP	1-06-11
CHECKED	MQ1	MQ1
ISSUED	MQ1	MQ1
QA CLASS	N/A	

SOURCE PRODUCTION & EQUIPMENT CO., INC.
113 TEAL ST. ST ROSE, LA 70087

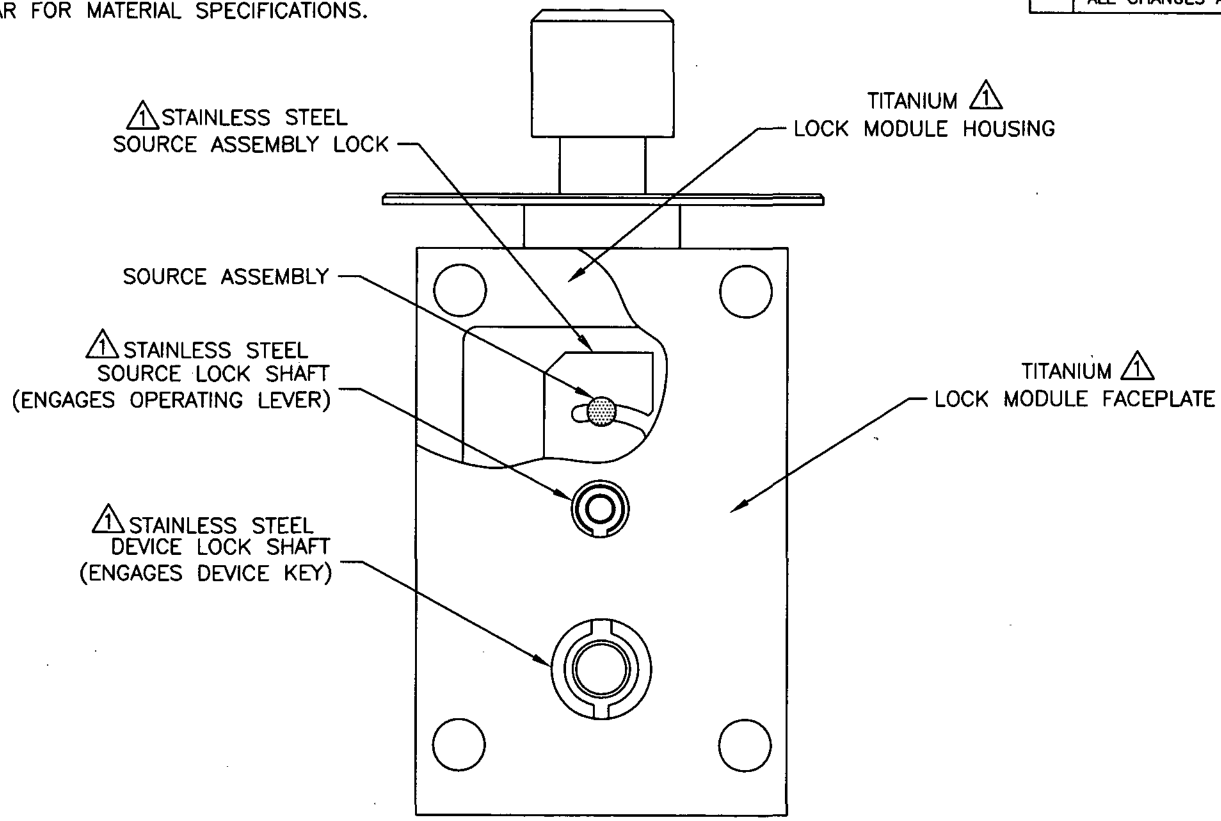
SPEC-150 TYPE B(U) PACKAGE
DEPLETED URANIUM SHIELD

SIZE C DWG NO. 15B008 REV 6.

SCALE: NTS 00062806 SHEET 1 OF 1

CONTROLLED COPY NO			
REVISIONS			
REV	DESCRIPTION	DATE	APPROVED
	ALL CHANGES ARE IN MQ1.	MQ1	MQ1

NOTE:
⚠1 SEE SAR FOR MATERIAL SPECIFICATIONS.



UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES TOLERANCES ARE FRACTIONS DECIMALS N/A DO NOT SCALE DRAWING	APPROVALS	DATE	SOURCE PRODUCTION & EQUIPMENT CO, INC 113 TEAL ST, ST ROSE, LA 70087
	DRAWN KP	1-10-11	LOCK MODULE- MODEL LM-200, SPEC
	CHECKED MQ1	MQ1	
TREATMENT NONE	APPROVED MQ1	MQ1	SIZE B DWG NO. 19B005 REV 1
FINISH NONE	QA CLASS N		SCALE: NTS | 00106001 | SHEET 1 OF 1

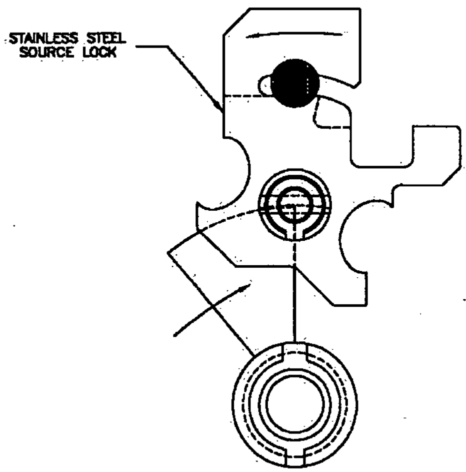
CONTROLLED COPY NO	

REVISIONS

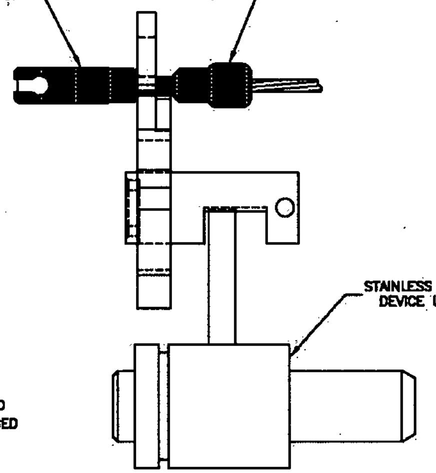
REV	DESCRIPTION	DATE	APPROVED
	ALL CHANGES ARE IN MQ1.	MQ1	MQ1

NOTES:

1. ARROWS INDICATE DIRECTION OF ROTATION TO UNLOCK.
2. (triangle) SEE SAR FOR MATERIAL SPECIFICATIONS.



VIEW FROM LOCK END



SOURCE LOCK AND DEVICE LOCK ENGAGED

SIDE VIEW

N/A	APPROVALS	DATE	SOURCE PRODUCTION & EQUIPMENT CO., INC. 113 TEAL ST. ST ROSE, LA 70087
DO NOT SCALE DRAWING	DRAWN KP	1-06-11	DEVICE LOCK OPERATION (LOCKED)- MODEL LM-200, SPEC
TREATMENT NONE	CHECKED MQ1	MQ1	SIZE C / DWG NO. 19B006 / REV 1
FINISH NONE	APPROVED MQ1	MQ1	
	QA CLASS N		SCALE: NTS / 00106101 / SHEET 1 OF 1

Appendix B:
2011 Clarification of External Radiation Levels,
to be appended to section 5.0 of the SPEC-150 SAR
in response to NRC RAI's 12, 13, 14

Source Production and Equipment Co., Inc. (SPEC)
2011 Clarification of External Radiation Levels
Report on background for answers to RAI's 12, 13, 14
SPEC-150 Package, CoC No. 9263
Prepared by K.Richardt, 11/29/10

In RAI's 12, 13 and 14, NRC requested additional information in order to verify SPEC's compliance with 10 CFR 71.43(f), 10 CFR 71.47 and 10 CFR 71.51(a)(1) and (a)(2). Several portions of the SAR will need to be revised to clarify our compliance with these requirements. Specific revisions required are listed at the end of this report. This report is intended to provide a full analysis of SPEC's compliance with the required dose rates for the SPEC-150.

The following table from NUREG-1886 Joint Canada - United States Guide for Approval of Type B(U) and Fissile Material Transportation Packages is presented as a summary of the shielding evaluations that demonstrate that the SPEC-150 meets dose rate requirements when loaded to capacity with Ir192. See analysis for the calculated dose rates for Se-75 and Yb-169.

Table 1						
Summary Table of Maximum Radiation Levels	Package Surface mSv/h (mrem/h)			1 Meter from Package Surface mSv/h (mrem/h)		
	Top	Side	Bottom	Top	Side	Bottom
Normal Conditions of Transport						
Gamma	1.1 (110)	1.4 (144)	1.1 (110)	0.011 (1.1)	0.016 (1.6)	0.011 (1.1)
Neutron	NA	NA	NA	NA	NA	NA
Total	1.1 (110)	1.4 (144)	1.1 (110)	0.011 (1.1)	0.016 (1.6)	0.011 (1.1)
10 CFR 71.47 (a) or Paragraphs 530 and 531 of TS-R-1 Limit	2 (200)	2 (200)	2 (200)	0.1 (10)*	0.1 (10)*	0.1 (10)*
Hypothetical Accident Conditions						
Gamma				0.06 (6)	0.06 (6)	0.05 (5)
Neutron				NA	NA	NA
Total				0.06 (6)	0.06 (6)	0.05 (5)
10 CFR 71.51(a)(2) or 656(b)(ii)(i) of TS-R-1 Limit				10 (1000)	10 (1000)	10 (1000)

10 CFR 71.47

10 CFR 71.47, External radiation standards for all packages, states that each package of radioactive materials offered for transportation must be designed and prepared for shipment so that under conditions normally incident to transportation the radiation level does not exceed 200 mrem/hr at any point on the external surface of the package, and the transport index does not exceed 10.

SPEC performed a detailed survey covering 425 points on the SPEC-150 as prepared for transport with a 137 curie Ir-192 source. The readings were extrapolated to 150 curies, and the readings at the surface were also corrected to allow for the distance from the radiation detector probe to the surface of the package. The survey results demonstrate that the SPEC-150 meets 10 CFR 71.47 and are as follows (these values were also included in Table 1). They are included in the current SAR in Section 5.1, package shielding, pages 33 and 34.

Table 2: Radiation Levels as Prepared for Transport

Summary Table of Maximum Radiation Levels	Package Surface mSv/h (mrem/h)			1 Meter from Package Surface mSv/h (mrem/h)		
	Top	Side	Bottom	Top	Side	Bottom
Normal Conditions of Transport						
Gamma	1.1 (110)	1.4 (144)	1.1 (116)	0.011 (1.1)	0.016 (1.6)	0.011(1.1)
Neutron	NA	NA	NA	NA	NA	NA
Total	1.1 (110)	1.4 (144)	1.1 (116)	0.011 (1.1)	0.016 (1.6)	0.011(1.1)
10 CFR 71.47 (a) or Paragraphs 530 and 531 of TS-R-1 Limit	2 (200)	2 (200)	2 (200)	0.1 (10)	0.1 (10)	0.1 (10)

In addition, through testing, SPEC demonstrated that radiation levels would increase by less than 20% by testing prototypes in normal conditions. If the radiation levels measured in the detailed survey above were increased by 20%, the external radiation levels would still meet the requirements of 10 CFR 71.47.

10 CFR 71.43(f) and 10 CFR 71.51(a)

Normal conditions testing was performed with an 8 curie source to demonstrate that the SPEC-150 meets 10 CFR 71.43(f) and 10 CFR 71.51(a). These regulations require that a package be designed, constructed and prepared for shipment so that under the normal conditions tests specified in 10 CFR 71.71 there would be no loss or dispersal of radioactive contents, no significant increase in external surface radiation levels, and no substantial reduction in the effectiveness of the package.

The SPEC-150 was tested in accordance with 10 CFR 71.71. The radiation levels were measured before and after the tests. There was no loss or dispersal of contents, the increase of 14% is below IATA's 20% definition of "significant increase", and there was no substantial reduction in the effectiveness of the package. Actual surface readings are provided in the current SAR in Appendix 9.5, Survey Data, files SRP1294A and SRP1294B, (all dose rates are expressed in mrem/h) and in Table 3.

Table 3: Radiation Levels Before and After Normal Conditions Testing

Surface	Pre-Drop	Post-Drop	Change	% of Change
Top	6.4	6.4	0.0	0.00%
Right	6.8	7.0	+0.2	2.94%

Table 3: Radiation Levels Before and After Normal Conditions Testing				
Surface	Pre-Drop	Post-Drop	Change	% of Change
Lock	4.4	5.0	+0.6	13.64%
Outlet	3.2	3.2	0.0	0.00%
Bottom	4.8	5.4	+0.6	12.50%
Left	5.6	5.8	+0.2	3.57%

10 CFR 71.51(a)(2)

10 CFR 71.51(a)(2) requires that a Type B package be designed, constructed, and prepared for shipment so that in hypothetical accident conditions there would be no escape of radioactive material exceeding a A_2 in one week, and no external radiation dose rate exceeding 10 mSv/h (1 rem/h) at 1 m (40 in) from the external surface of the package.

For the initial testing, SPEC-150 prototype 4 was loaded with 22 curies of Ir-192 and two 9 meter drop tests and 1 puncture test were performed. There was no escape of any radioactive material. The radiation levels were measured on the surface after testing and extrapolated to 150 curies and distance corrected. These extrapolated and corrected surface radiation levels meet the requirements in 10 CFR 71.51(a)(2) for radiation levels at one meter. Therefore, the dose rate at one meter would be a fraction of the 10 mSv/h (1 rem/h) limit and were not measured after the test. Actual surface readings may be found in the current SAR in Appendix 9.5, Survey Data, file SRP894B (with dose rates expressed in mrem/h).

Table 4: Radiation Levels After Initial Hypothetical Accident Conditions Testing		
Surface:	Actual Readings: mSv/h (mrem/h)	Extrapolated and Corrected: mSv/h (mrem/h)
Top	0.20 (20)	1.64 (164)
Right	0.28 (28)	2.29 (229)
Lock	0.26 (26)	2.13 (213)
Outlet	0.14 (14)	1.15 (115)
Bottom	0.18 (18)	1.47 (147)
Left	0.20 (20)	1.64 (164)

Additional 9 meter drop tests and puncture tests were performed in 1997 (see Appendix9.6). The tests were performed using SPEC-150 serial number 500 with a 26 curie source. Radiation readings were taken at the surface and one meter from the surface. The extrapolated and corrected surface readings meet the dose rate requirement at one meter. The external radiation levels are presented in Table 5. These values were used for the hypothetical accident conditions section of Table 1.

Table 5: Radiation Levels After Additional Hypothetical Accident Conditions Testing				
Surface:	Actual Readings: mSv/h (mrem/h)		Extrapolated and Corrected: mSv/h (mrem/h)	
	Surface	1m	Surface	1m
Top	0.14 (14)	0.01 (1.0)	0.82 (82)	0.06 (6)
Right	0.14 (14)	0.09 (0.9)	0.82 (82)	0.05 (5)
Lock	0.11 (11)	0.01 (1.0)	0.64 (64)	0.06 (6)
Outlet	0.10 (10)	0.09 (0.9)	0.58 (58)	0.05 (5)
Bottom	0.11 (11)	0.09 (0.9)	0.64 (64)	0.05(5)
Left	0.20 (20)	0.01 (1.0)	1.16 (116)	0.06 (6)

SAR Revisions:

1. Section 5, pages 31 and 32: Revise the first sentence to read "A shielding evaluation of the model SPEC-150 was performed to demonstrate compliance with NRC and IAEA requirements". Eliminate the remainder of the paragraph as it contains details of testing requirements for industrial radiography equipment. Combine paragraphs one and two. (RAI-13)
2. Section 5.2, page 34: Eliminate the last sentence in the last paragraph, one meter readings are not needed to measure any increase in radiation levels as a result of normal conditions testing. Surface radiation levels were used to make that determination. (RAI-12)
3. Section 5.3, page 36 (A&B). Add Table below with actual surface radiation levels from Appendix 9.5, survey data, file SRP295B with dose rates expressed in mrem/h. Change last sentence from "Readings at one meter were not made because the surface readings alone verify that the package meets the radiation level requirements at one meter and because the readings at one meter would be too low to be statistically relevant." to "The radiation levels at the surface meet the requirements in 10 CFR 71.51(a)(2) for radiation levels at one meter. Therefore, the dose rate at one meter would be a fraction of the 10 mSv/h (1 rem/h) limit and actual one meter readings were not taken." (RAI-12)

Actual, Extrapolated and Corrected Surface Dose Rate, Hypothetical Accident Conditions Testing		
Surface:	Actual Readings: mSv/h (mrem/h)	Extrapolated and Corrected: mSv/h (mrem/h)
Top	0.042 (4.2)	1.89 (189)
Right	0.038 (3.8)	1.71 (171)
Lock	0.034 (3.4)	1.53 (153)
Outlet	0.022 (2.2)	0.99 (99)
Bottom	0.038 (3.8)	1.71 (171)
Left	0.036 (3.6)	1.62 (162)

Appendix C: Revised SAR Pages

Pages	Section	Description of Change
31 and 32	5.0	Revised the first sentence to read "A shielding evaluation of the model SPEC-150 was performed to demonstrate compliance with NRC and IAEA requirements". Eliminated the remainder of the paragraph as it contains details of testing requirements for industrial radiography equipment. Combined paragraphs one and two. (RAI-13)
34	5.2	Eliminated the last sentence in the last paragraph, one meter readings are not needed to measure any increase in radiation levels as a result of normal conditions testing. Surface radiation levels were used to make that determination. (RAI-12)
36A and 36B	5.3	Added Table actual surface radiation levels from Appendix 9.5, survey data, file SRP295B with dose rates expressed in mrem/h. Change last sentence from "Readings at one meter were not made because the surface readings alone verify that the package meets the radiation level requirements at one meter and because the readings at one meter would be too low to be statistically relevant." to "The radiation levels at the surface meet the requirements in 10 CFR 71.51(a)(2) for radiation levels at one meter. Therefore, the dose rate at one meter would be a fraction of the 10 mSv/h (1 rem/h) limit and actual one meter readings were not taken." (RAI-12)
39	7.1.1	Added requirement to visually check the exposed fasteners and welds.
73	8.1.2	Revised acceptance testing section to conform to revised drawing requirements for welds that are Important to Safety (ITS) and non-ITS.

4.2 Requirements for Normal Conditions of Transport

4.2.1 Release of Radioactive Material

Based on the results of the evaluations for normal conditions of transport performed in Section 2.8 above, there was no release of radioactive material from the primary containment vessel.

4.2.2 Pressurization of Containment Vessel

The is negligible gas contained within the minute void of the sealed source capsule, therefore any pressurization due to temperature or reduced pressure at flight altitudes would not effect the integrity of the sealed source capsule.

4.2.3 Coolant Contamination

No coolants are used in the package.

4.2.4 Coolant Loss

No coolants are used in the package.

4.3 Containment Requirement for the Hypothetical Accident Conditions

4.3.1 Fission Gas Products

No fissionable radioactive material is used in the model SPEC-150 package.

4.3.2 Releases of Contents

Based on the results of the Type B performance tests described in Section 2.9 the special form, sealed source capsule was not affected in any manner. Therefore, there can be no release of radioactive material from the primary containment vessel due to the conditions specified in the hypothetical accident conditions.

5. SHIELDING EVALUATION

A shielding evaluation of the model SPEC-150 was performed to demonstrate compliance with NRC and IAEA requirements. Adequate shielding design for the model SPEC-150 is established by actual measurements of radiation profiles from randomly selected prototypes, and by actual measurements of resulting radiation levels after the numerous tests performed for normal conditions of transport and hypothetical accident conditions on two test packages. Theoretical calculations have not been used.

For surface radiation levels a correction factor was applied to adjust for the distance from the Center of the detector to the surface of the package. The correction factor was based on NRC Draft Regulatory Guide and Value/Impact Statement, dated December 1979, titled "Measurements of Radiation Levels on Surfaces of Packages of Radioactive Materials." Table 1 of Appendix A was used to calculate the correction factor. This was used instead of the significantly lower correction factor that would be calculated by the use of Table 2. The assumption in Table 2 that the inverse linear expression should be used instead of the inverse square law is not accurate for the package and detector size used. The correction factor was calculated based on the smallest linear dimension of the package and was applied to the radiation readings taken at all surface locations, including both ends of the package. Lower correction factors were not calculated as the linear dimension of the package (distance to source) increased. Therefore, the corrected surface readings presented in this application are based on the highest correction factor and represent the most conservative interpretation of the regulatory guide.

The shortest linear dimension is of the SPEC-150 is 5-3/8 inches and the longest linear dimension is 14-1/2 inches. The GM tube detector is an LND model 714 which has an effective diameter of 0.190 inch and an actual exterior diameter of 0.250 inch. The GM tube detector is installed in a probe that positions the surface of the package 5/16 inch from the surface of the detector. A margin of safety was added to the calculated correction factor to adjust for inherent instrument inaccuracies. A final correction factor of 1.2 was adopted and was applied to surface radiation readings measured during the shielding evaluation tests. The unadjusted surface radiation readings and their locations are presented on sketches in Appendix 9.

5.1 Package Shielding

A depleted uranium casting weighing approximately thirty-seven pounds is used for the principal shielding material. A titanium or zircalloy S-Tube permits the source assembly to pass through the depleted uranium shield for use as an industrial radiography exposure device. When the model SPEC-150 is used as a transport package, the sealed source capsule is positioned in the center of the depleted uranium shield primarily by the lock mechanism which positions the source assembly in the device. The source assembly lock must be locked in order to prepare the package for shipment. The lock cannot be locked unless the source assembly is positioned such

One Meter from Surface	150 Ci Iridium-192
Package Surface	Maximum mrem/hr
Top	1.1
Bottom	1.1
Left Side	0.9
Right Side	1.1
Lock End	1.6
Outlet End	0.9
Combined	1.6

5.2 Normal Conditions of Transport

Radiation surveys were performed after each of the normal conditions of transport tests which were performed; free drop, penetration and compression. Radiation levels were measured at a sufficient number of locations to determine if there were any significant changes compared to the radiation levels prior to the tests. No changes in radiation levels were measured after each of the penetration and compression tests. The five 4 foot free drop tests were performed on Prototype No. 4 after the combined hypothetical accident condition tests. The maximum surface radiation levels on each of the six surfaces were measured after each drop. The results were extrapolated to 150 Ci Iridium are tabulated below:

Surface	Maximum mrem/hr		150 Ci Iridium-192			
	Before	1st	2nd	3rd	4th	5th
Top	144	135	144	131	140	126
Bottom	108	113	108	117	117	122
Right Side	153	149	158	153	153	117
Left Side	126	131	122	122	113	131
Outlet End	72	72	63	63	72	68
Lock End	99	95	108	104	104	113

The maximum change in surface radiation levels above was 14% which is less than the 20% increase in surface radiation criteria specified in IAEA Safety Series No. 6 Regulations for the Safe Transport of Radioactive Material 1985 Edition (As Amended 1990). The highest unadjusted and unextrapolated surface radiation readings and their locations are shown on the radiation profile sketch of the survey dated 12/17/94. See Appendix 9. The activity of the Ir-192 source was eight curies. The highest radiation level was located at the right side of the package and measured 7.0 mR/hr at the surface.

The highest unadjusted and unextrapolated surface radiation readings and their locations on Prototype No. 4 are shown on the radiation profile sketch of the survey dated 8/30/94. See Appendix 9. The activity of the Ir-192 source was 22 curies. The highest radiation level was located at the right side of the package and measured 28 mR/hr at the surface. Adjusted and extrapolated to 150 curies the reading is 229 mR/hr which is far below the allowable limit of 1,000 mR/hr at one meter.

Prototype No. 4 was subjected to five four-foot drop tests December 17, 1994. A survey was made after all five tests. The highest unadjusted and unextrapolated surface radiation readings and their locations are shown on the radiation profile sketch of the survey dated 12/17/94. See Appendix 9. The activity of the Ir-192 source was eight curies. The highest radiation level was located at the right side of the package and measured 7 mR/hr at the surface. Adjusted and extrapolated to 150 curies the reading is 158 mR/hr which is far below the allowable limit of 1,000 mR/hr at one meter. The readings are assumed to be less accurate than the previous readings made on August 30, 1994 because the activity of the Ir-192 source is only eight curies.

Prototype No. 4 was subjected to a third 30-foot drop test, followed by a one meter puncture test, and a fourth 30 foot drop test on February 25, 1995. The highest unadjusted and unextrapolated surface radiation readings and their locations on Prototype No. 4 are shown on the radiation profile sketch of the survey dated 2/25/95. A survey was made after all three tests. See Appendix 9. The activity of the Ir-192 source was four curies. The highest radiation level was located at the top of the package and measured 4.2 mR/hr at the surface. Adjusted and extrapolated to 150 curies the reading is 171 mR/hr which is far below the allowable limit of 1,000 mR/hr at one meter. The readings are assumed to be less accurate than the previous readings made on August 30, 1994 and December 17, 1994 because the activity of the Ir-192 source is only four curies. The radiation levels at the surface meet the requirements in 10 CFR 71.51(a)(2) for radiation levels at one meter. Therefore, the dose rate at one meter would be a fraction of the 10 mSv/h (1 rem/h) limit and actual readings were not taken.

Actual, Extrapolated and Corrected Surface Dose Rate, Hypothetical Accident Conditions Testing		
Surface:	Actual Readings: mSv/h (mrem/h)	Extrapolated and Corrected: mSv/h (mrem/h)
Top	0.042 (4.2)	1.89 (189)
Right	0.038 (3.8)	1.71 (171)
Lock	0.034 (3.4)	1.53 (153)
Outlet	0.022 (2.2)	0.99 (99)
Bottom	0.038 (3.8)	1.71 (171)
Left	0.036 (3.6)	1.62 (162)

5.4 Source Specification

The source assembly used in the normal condition of transport and hypothetical accident conditions radiation level measurements was a model SPEC G-60 with an original activity of 137 Ci. The source was corrected for decay to each day that the tests were performed and the presented results extrapolated to an activity of 150 Ci.

5.5 Model Specification

Physical radiation measurements were performed on prototype packages and radiation surveys were performed on the prototype test packages after the tests for normal conditions of transport and hypothetical accident conditions. Theoretical calculations or scale models were not used.

7. OPERATING PROCEDURES

7.1 Procedures for Preparing and Loading the Package

Training of personnel who prepare, offer and transport hazardous material shipments, including the model SPEC-150, for transport is required pursuant to 49 CFR 172.700, and Section 10 of the Louisiana Radiation Regulations.

The source assembly is loaded into the model SPEC-150 at the SPEC facilities under the provisions of Louisiana Radioactive Material License LA-2966-L01 in accordance with procedures and radiation protection standards established under that license and in compliance with 10 CFR 71.87(f) and 10 CFR 20.1906.

The following instructions provided meet the requirements of 10 CFR 71.85, 71.87, 71.89 and 71.91.

7.1.1 General Package Inspection

Visually inspect the model SPEC-150 to determine if it is in unimpaired condition for shipment. The model SPEC-150 should be inspected to determine that it is not damaged, that the lock operates properly, that the source assembly (pigtail) is securely locked in the package, and that the safety plug and lock cap are securely positioned. Visually check the exposed fasteners and welds. Verify that the package identification plate is present and legible, which identifies the package as a model SPEC-150 and displays the Certificate of Compliance identification number.

7.1.2 Packaging

Verify that the package is proper for the contents to be shipped.

Verify that the source assembly is properly secured and locked in the model SPEC-150. The source safety plug and the lock cap must be firmly attached.

Measure the maximum surface radiation level and the maximum radiation level at one meter from the surface of the package. The maximum surface radiation level must not exceed 200 mrem/hr. The maximum radiation level at one meter from the surface of the package must not exceed 10 mrem/hr.

If the lock key is to be shipped in the same container with the camera, then seal the lock key in an envelope which will be destroyed when opened.

7.1.3 Outer Package Surface Contamination

Packages may not be shipped on a non-exclusive use basis with outer surface contamination levels exceeding the values below, and it is the shipper's responsibility to ensure that the following conditions are met.

package construction inspection tests pursuant to the quality assurance program under NRC Certificate of Compliance No. 0102 and inspection prior to shipment to a customer. In most instances when the package is shipped to a customer it contains a radioactive source assembly.

8.1.1 Visual Inspection

Each packaging is visually inspected as part of the quality assurance final package inspection after construction, which includes quality of workmanship, adherence to production specifications and drawings, presence of attached identification plates and warnings, and presence of components, such as source safety plug and lock cap. The final inspection must ensure that the package conforms to the drawings specified in the Certificate of Compliance.

Each source assembly is visually inspected after fabrication.

Prior to shipment the package is again visually inspected to assure that the source assembly is properly contained in the packaging, and the shipment is properly marked and labeled for shipment.

8.1.2 Structural and Pressure Tests

Although structural acceptance tests on the model SPEC-150 are not indicated because of the rugged design and durable materials of construction any structural failure would be apparent, a liquid penetrant test is performed during fabrication on important to safety weld joints. See drawing 15B002A which depicts specific inspection method used for specific SPEC-150 welds. Visual inspection is performed on the welded joints that connect the inner bulkhead plate to the bottom plate, the inner bulkhead support cup to the inner bulkhead, and the outlet end plate support cup to the outlet end plate and bottom plate. Both visual and liquid penetrant inspection is performed on the joints that comprise the basic structure of the package which consists of the housing cover (shell), bottom plate and both end plates.

Visual and Liquid Penetrant inspections are performed in accordance with the ASME Code for Boilers and Pressure Vessels, Section VIII, Division 1.

Pressure tests are not indicated because there is no possibility of a pressure build up which would affect the structure of the containment or the integrity of

Appendix D: Materials of Construction

Grade 2 Titanium Components Standards			300 Series Stainless Steel Standards	
ASME	ASTM		ASME	ASTM
SB-265	B 265		SA 213	A 213
SB-337	B 337		SA 240	A 240
SB-338	B 338		SA 249	A 249
SB-348	B 348		SA 269	A 269
SB-381	B 381		SA 270	A 270
SB-861	B 861		SA 271	A 271
SB-862	B 862		SA 276	A 276
			SA 312	A 312
			SA 336	A 336
			SA 358	A 358
			SA 376	A 376
			SA 409	A 409
Tungsten Components Standard:			SA 473	A 473
ASTM B 777			SA 479	A 479
			SA 480	A 480
			SA 484	A 484
			SA 554	A 554
Safety Plug			SA 581	A 581
Cable: Manufactured to RR-W-410			SA 582	A 582
Tip: Stainless steel per 300 series standards listed in right columns			SA 666	A 666
			SA 688	A 688

Quick Disconnect Coupling Specifications (component of safety plug)
Snap - Tite brand stainless steel quick disconnect, catalog number SPHC6-6F
Hofmann brand stainless steel quick disconnect, catalog number HHS3-S6
Dixon or Perfecting brand stainless steel quick disconnect, catalog number 3VF3-SS-E
Hydrason brand stainless steel quick disconnect, catalog number 0026012120
SPEC manufactured stainless steel quick disconnect

Appendix E:
John Munro Report
Shielding of the SPEC-150 with ^{75}Selenium and ^{169}Ytterbium

Report

By: John J. Munro III

Date: 31 December 2010

Subject: **Shielding of the SPEC-150 with ^{75}Selenium and ^{169}Ytterbium**

We present the following report to demonstrate the shielding efficiency of the SPEC-150 Radiographic Exposure Device/Type B(U) Container, when containing up to 150 Ci of either ^{75}Selenium or ^{169}Ytterbium, satisfies the regulatory requirements.

The SPEC-150 locates the sources approximately 68 mm (2.68 inches) from the surface of the container. The uranium thickness in the least-shielded direction is 48 mm (1.91 in).

The radiation exposure rate on the surface of the container is calculated from:

The exposure rate at a point in space is expressed as:

$$I = \frac{\Gamma A}{r^2}\tau \qquad [1]$$

where:
- Γ: Specific Exposure Rate Constant
- A: Activity of the Source (150 Ci)
- r: Distance from the source to the point of interest:
  (*i.e.* Surface of the Container: 0.068 m;
  One meter from the Surface: 1.068 m)

The Broad Beam transmission of photons through a material can be expressed as:

$$\tau_E = \frac{I}{I_0} = B(E,x)e^{-\frac{\mu}{\rho}(E)\rho x} \qquad [2]$$

where:
- τ_E: Transmission Factor
- I: Exposure Rate in the presence of shielding
- I_0: Exposure Rate in the absence of shielding
- $B(E)$: Exposure Build-Up Factor
- $\mu/\rho(E)$: Mass Absorption Coefficient
- ρ: Density of the shielding material
- x: Thickness of the shielding material along the beam path
- E: Energy of the photon

The values of $B(E)$ and $\mu/\rho(E)$ are photon energy-dependent. The resultant transmission for all of the photons emitted by the radionuclide, τ, is therefore calculated from the relationship:

$$\tau = \sum_i f(E)_i B(E,x)_i e^{-\frac{\mu}{\rho}(E)_i \rho x} \qquad [3]$$

where:
- $f(E)_i$: Fraction of exposure rate due to photons of energy E
- i: Represents the i^{th} photon

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^{75}Selenium

We applied this method to the transmission of ^{75}Selenium photons through uranium.

The ^{75}Selenium exposure rate constant was taken as 200 mR m^2 h^{-1} Ci^{-1}, as presented in the 1970 Radiological Health Handbook, presented by Weeks et al.[i], and referenced in USNRC-issued Certificates of Compliance for Radioactive Material Packages USA/9296/B(U)-96 and USA/9269/B(U)-96.

The ^{75}Selenium spectrum consists of 20 photons with energies above 15 keV.[ii] The mass attenuation coefficient for uranium for each of these photon energies was interpolated from the values presented by NIST.[iii] Exposure Buildup values were interpolated from data for Uranium presented in ANS/ANSI-6.4.3-1991.[iv]

The resultant calculated transmission for photons of ^{75}Selenium through 48 mm of uranium is.

Using Eq. 1 and the minimum shielding of the SPEC-150, (i.e. 48 mm):

	At Surface	At 1 meter from Surface
Calculated (48 mm Uranium)	0.012 mR/hr	4.8×10^{-5} mR/hr

This result is less than 0.006% of the regulatory limit for the surface of the container and less than 0.0005% of the regulatory limit at one meter from the surface.

We also repeated this calculation using extreme assumptions:

- the mass attenuation coefficients 20% less than the values used above
- the exposure buildup factors were 2X the values used above
- the Specific Gamma Ray Exposure Constant (Γ) was 5X the value used above

Under these extreme assumptions, the resultant exposure rates would be:

Under Extreme Assumptions	At Surface	At 1 meter from Surface
Calculated (48 mm Uranium)	1.2 mR/hr	0.005 mR/hr

This result is less than 0.6% of the regulatory limit for the surface of the container and less than 0.05% of the regulatory limit at one meter from the surface.

Clearly, the SPEC-150 shielding is capable of containing 150 Ci of ^{75}Selenium and maintaining the radiation levels surrounding the package within the regulatory limits.

^{169}Ytterbium

We applied this same method to the transmission of ^{169}Ytterbium photons through uranium.

The ^{169}Ytterbium exposure rate constant was taken as 180 mR m^2 h^{-1} Ci^{-1}, as presented by Mason et al.[v]

The ^{169}Ytterbium spectrum consists of 78 photons with energies above 15 keV.[2] The mass attenuation coefficients for uranium for each photon energy were interpolated from the values presented by NIST.[3] Exposure Buildup values were interpolated from data for Uranium presented in ANS/ANSI-6.4.3-1991.[4]

The resultant calculated transmission for photons of ^{75}Selenium through 48 mm of uranium is.

Using Eq. 1 and the minimum shielding of the SPEC-150, (i.e. 48 mm):

	At Surface	At 1 meter from Surface
Calculated (48 mm Uranium)	0.012 mR/hr	5.0×10^{-5} mR/hr

This result is less than 0.006% of the regulatory limit for the surface of the container and less than 0.0005% of the regulatory limit at one meter from the surface.

We also repeated this calculation using extreme assumptions:

- the mass attenuation coefficients 20% less than the values used above
- the exposure buildup factors were 2X the values used above
- the Specific Gamma Ray Exposure Constant (Γ) was 5X the value used above

Under these extreme assumptions, the resultant exposure rates would be:

Under Extreme Assumptions	At Surface	At 1 meter from Surface
Calculated (48 mm Uranium)	1.1 mR/hr	0.004 mR/hr

This result is 0.53% of the regulatory limit for the surface of the container and 0.04% of the regulatory limit at one meter from the surface.

Clearly, the SPEC-150 shielding is capable of containing 150 Ci of ^{169}Ytterbium and maintaining the radiation levels surrounding the package within the regulatory limits.

Conclusion

The shielding afforded by the SPEC-150 is sufficient to contain up to 150 Ci of either ^{75}Selenium or ^{169}Ytterbium. The SPEC 150 adequately provides sufficient shielding to meet the regulatory requirements. Even under the extreme assumptions that

- the mass attenuation coefficients are 20% less than the values used in our original calculation
- the exposure buildup factors were 2X the values used in our original calculation
- the Specific Gamma Ray Exposure Constant (Γ) was 5X the value used in our original calculation

the resultant exposure rates would be well less than 1% of the regulatory limit.

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[i] Weeks KJ, Schulz RJ, Selenium-75: a potential source for use in high-activity brachytherapy irradiators, Med Phys. 1986 Sep-Oct;13(5):728-31.

[ii] http://www.nndc.bnl.gov/nudat2/indx_dec.jsp

[iii] Hubbell JH and Seltzer SM, Tables of X-Ray Mass Attenuation Coefficients and Mass Energy-Absorption Coefficients from 1 keV to 20 MeV for Elements Z = 1 to 92 and 48 Additional Substances of Dosimetric Interest, http://www.physics.nist.gov/PhysRefData/XrayMassCoef/cover.html

[iv] Trubey DK et al., Gamma-Ray Attenuation Coefficients and Buildup Factors for Engineering Materials, ANSI/ANS-6.4.3-1991

[v] Mason DL, Battista JJ, Barnett RB, Porter AT, Ytterbium-169: calculated physical properties of a new radiation source for brachytherapy, Med Phys. 1992 May-Jun;19(3):695-703.