SPEC SOURCE OF QUALITY

February 14, 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) revised 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, and appendices. We have also provided a consolidated SAR that includes all of the previously approved changes to the certificate and the changes that are part of this amendment request. Appendix C in this document contains a table with each section changed with the rationale for the change with the before and after text when practicable. We have also provided this information electronically to Mr. Pierre 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
Regulatory and Quality Manager

W:\REGLATORY\APPLICAT\SPEC-150\2011ConslApp\application, 2nd sum 021111, RAIexplanation.wpd

NMSS24

Source Production & Equipment Co., Inc.
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 Consolidated SAR and Appendix C.
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 D1.9 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 for serial number 1475 and newer". We have included Appendix D in the consolidated 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 and revised SAR.
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 serial number 1475 and up. SPEC-150's manufactured prior serial number 1475 must continue to meet the current certificate conditions, 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.	Additional normal conditions testing was performed with a suitable strength source assembly and the shielding evaluation demonstrated that the SPEC-150 meets normal conditions requirements. The 1997 hypothetical accident testing shielding evaluation demonstrates that the SPEC-150 meets hypothetical accident conditions requirements. See Appendix B for additional information demonstrating the ability of the SPEC-150 to meet external radiation standards in normal and accident conditions.
	10 CFR 71.43(f) (no loss in normal conditions),	
	10 CFR 71.47 (external radiation standards) when normally prepared for transit of less than 200 mrem/hr at the surface	
	and the transport index not exceeding 10,	
	10 CFR 71.51(a)(1) (no loss in normal conditions) and	
	(a)(2)(<1rem at 1 meter in hypothetical accident conditions) .	
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.
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.

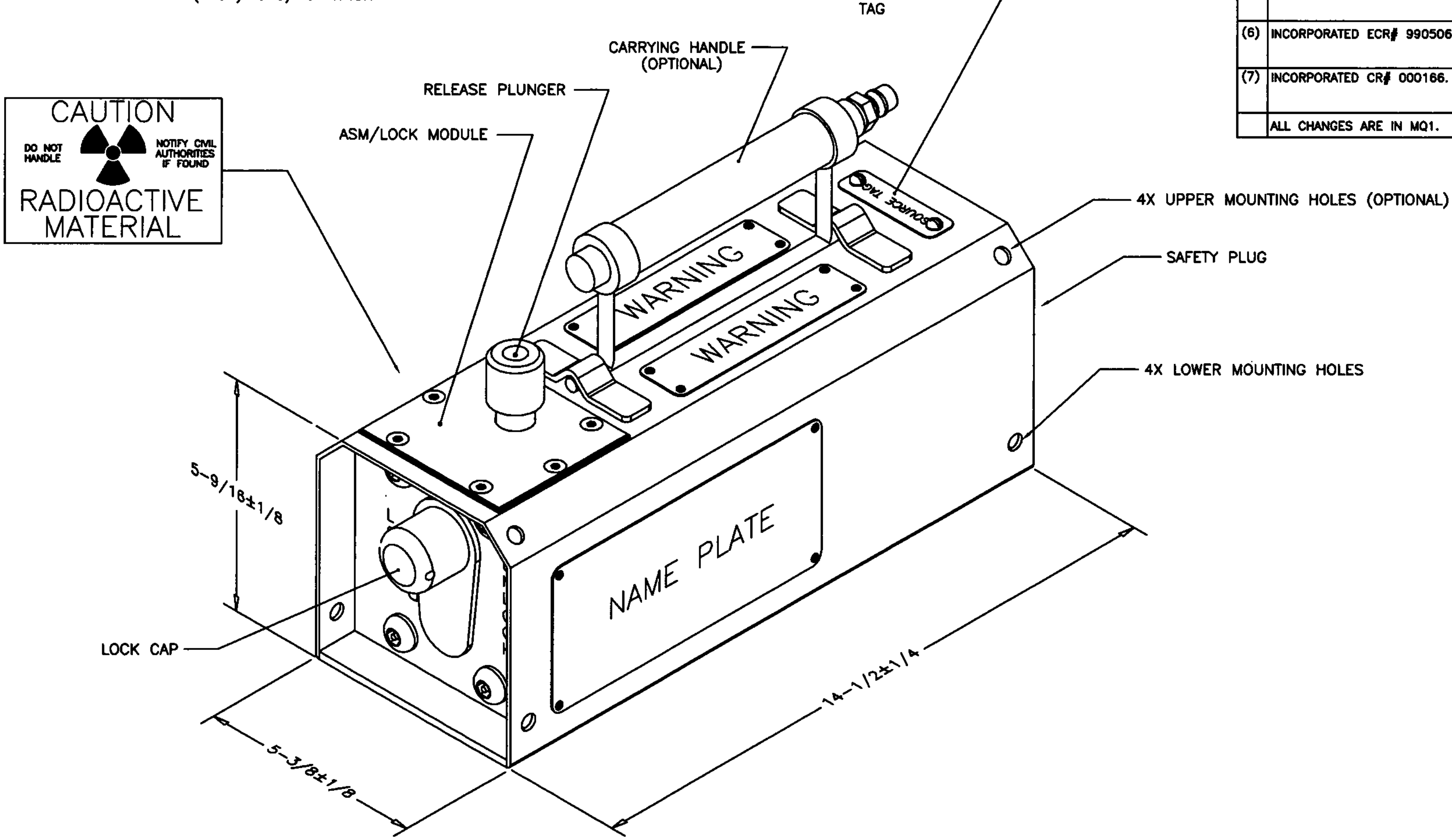
Appendix A: Revised drawings.

Drawing	Rev	Description of Changes from Current Certificate
15B000	9	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 for serial number 1475 and newer.
15B002A	8	Added fabrication and inspection notes for ITS and non-ITS welds, added material specifications for safety related components for serial number 1475 and newer.
15B008	7	Removed notes that did not specifically describe the depleted uranium shield. Added minimum weight and density for serial number 1475 and newer.
19B005	2	Added material specifications for safety related components for serial number 1475 and newer, the source assembly lock, device lock, lock module housing and lock module faceplate.
19B006	2	Added material specifications for safety related components for serial number 1475 and newer, the source assembly lock and device lock.
(190909)	(0)	(No changes needed, original drawing was not replaced and is not appended.)

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 FOR SERIAL NUMBERS 1475 AND NEWER.
   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

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



UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES TOLERANCES ARE			SOURCE PRODUCTION & EQUIPMENT CO., INC. 113 TEAL ST, ST ROSE, LA 70087
AS NOTED	APPROVALS	DATE	
	DRAWN KP	2-8-11	SPEC-150 TYPE B(U) PACKAGE ISOMETRIC VIEW
DO NOT SCALE DRAWING	CHECKED MQ1	MQ1	
TREATMENT NONE	ISSUED MQ1	MQ1	SIZE C DWG NO. 15B000 REV 9
FINISH NONE	QA CLASS N/A		SCALE: NTS 00063309 SHEET 1 of 1

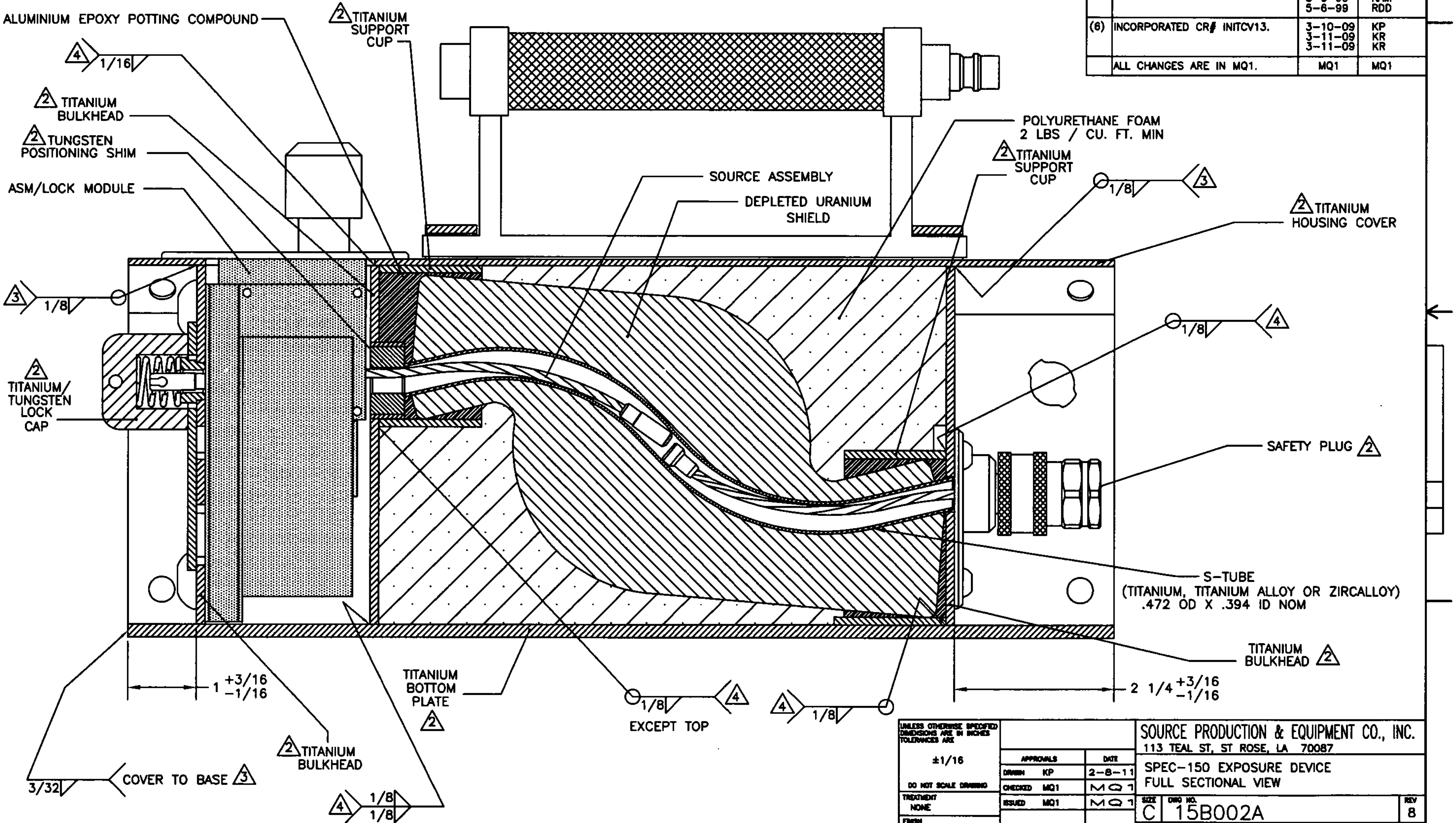
CONTROLLED COPY NO

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.
5. NOTES 2,3,4 ARE APPLICABLE TO SPEC-150 SERIAL NUMBERS 1475 AND NEWER.

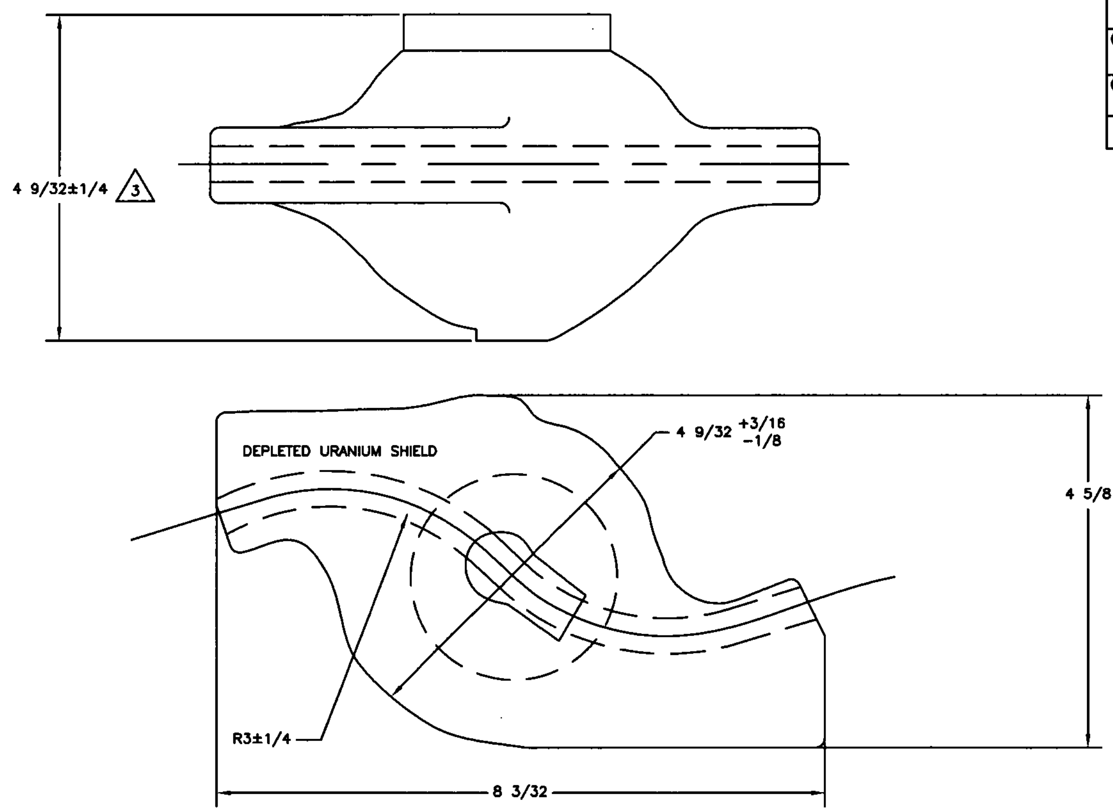
REVISIONS

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



UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES TOLERANCES ARE ±1/16 DO NOT SCALE DRAWING	APPROVALS / DATE	SOURCE PRODUCTION & EQUIPMENT CO., INC. 113 TEAL ST, ST ROSE, LA 70087
	DRAWN KP 2-8-11	SPEC-150 EXPOSURE DEVICE FULL SECTIONAL VIEW
TREATMENT NONE	CHECKED MQ1 MQ1	SIZE C DWG NO. 15B002A REV 8
FINISH NONE	ISSUED MQ1 MQ1 QA CLASS N	SCALE: NTS 00062708 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 RDD
(2)	SEE QA FILE FOLDER 15B008.	4-13-95 4-13-95 4-13-95	S. BYRD KC RDD
(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# 000188.	3-10-09 3-11-09 3-11-09	KP KR KR
	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.
4. THE MINIMUM WEIGHT AND MATERIAL DENSITY REQUIREMENTS ARE APPLICABLE TO SPEC-150 SERIAL NUMBERS 1475 AND NEWER.

UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES TOLERANCES ARE FRACTIONS ±1/8 DO NOT SCALE DRAWING	APPROVALS	DATE
	DRAWN KP	2-8-11
	CHECKED MQ1	MQ1
TREATMENT NONE	ISSUED MQ1	MQ1
FINISH NONE	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 7

SCALE: NTS | 00062607 | SHEET 1 OF 1

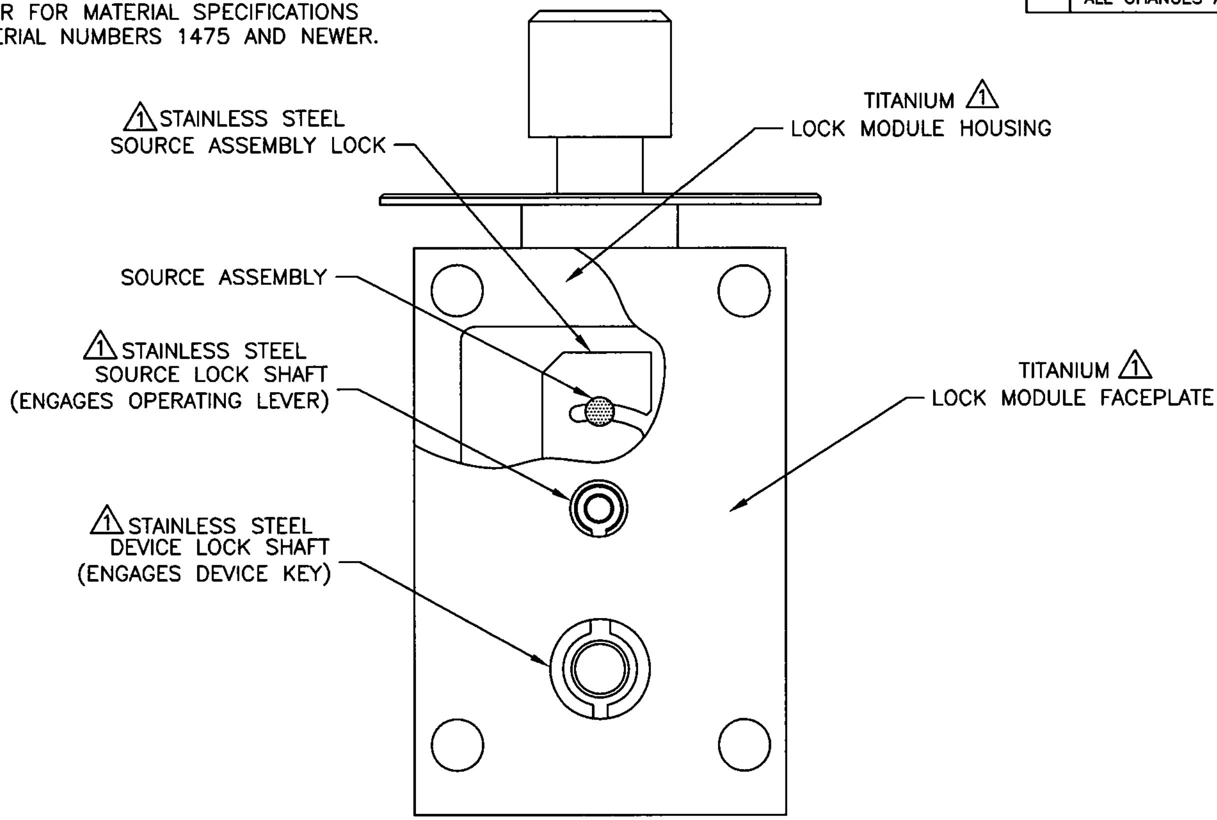
CONTROLLED COPY NO	

REVISIONS

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

NOTE:

⚠1 SEE SAR FOR MATERIAL SPECIFICATIONS FOR SERIAL NUMBERS 1475 AND NEWER.



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	2-7-11	LOCK MODULE– MODEL LM-200, SPEC
TREATMENT NONE	CHECKED MQ1	MQ1	SIZE B DWG NO. 19B005 REV 2
FINISH NONE	APPROVED MQ1	MQ1	SCALE: NTS 00106002 SHEET 1 OF 1
	QA CLASS N		

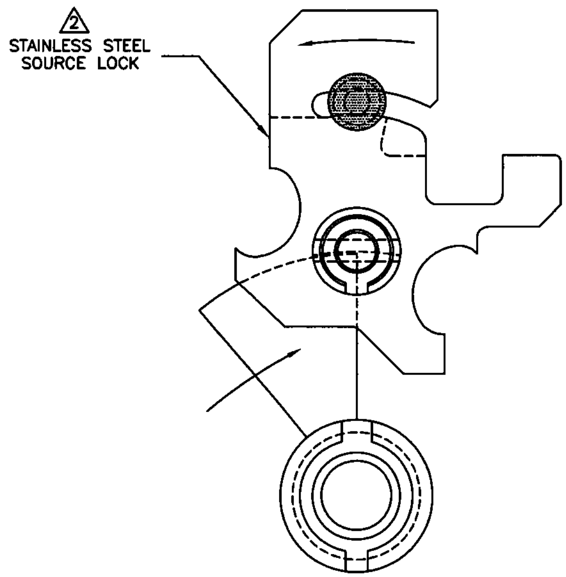
CONTROLLED COPY NO	

REVISIONS

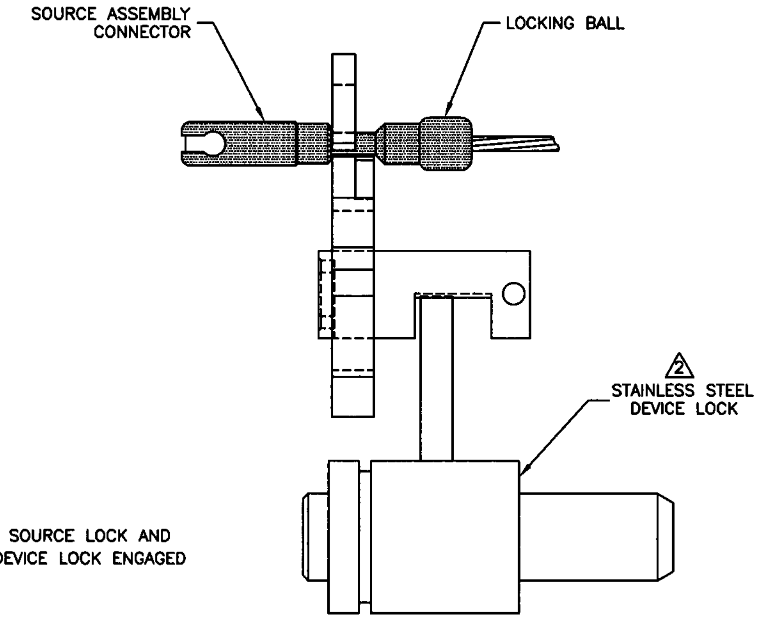
REV	DESCRIPTION	DATE	APPROVED
	ALL CHANGES ARE IN MQ1.	MQ1	MQ1

NOTES:

1. ARROWS INDICATE DIRECTION OF ROTATION TO UNLOCK.
2. (△2) SEE SAR FOR MATERIAL SPECIFICATIONS FOR SERIAL NUMBERS 1475 AND NEWER.



VIEW FROM LOCK END



SOURCE LOCK AND
DEVICE LOCK ENGAGED

SIDE VIEW

UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES TOLERANCES ARE N/A DO NOT SCALE DRAWING			SOURCE PRODUCTION & EQUIPMENT CO., INC. 113 TEAL ST, ST ROSE, LA 70087
	APPROVALS	DATE	DEVICE LOCK OPERATION (LOCKED)– MODEL LM–200, SPEC
	DRAWN KP	2–7–11	
	CHECKED MQ1	MQ1	
TREATMENT NONE	APPROVED MQ1	MQ1	SIZE C DWG NO. 19B006 REV 2
FINISH NONE	QA CLASS N		SCALE: NTS 00106102 SHEET 1 OF 1

Appendix B:
2011 Clarification of External Radiation Levels,
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
Revised 02/04/11 to include normal conditions tests performed 02/03/11

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). This report is intended to provide a full analysis of SPEC's compliance with the required dose rates for the SPEC-150. It is also the basis for the additional information in Section 5.0 of the 2011 consolidated SAR.

The following table from NUREG-1886 Joint Canada - United States Guide for Approval of Type B(U) and Fissile Material Transportation Packages contains the regulatory requirements for the maximum radiation levels in mSv/h and mrem/h. All shielding tests were performed with survey meters that measure in mR/hr. Therefore, for the purpose of this report and section 5.0 of the SAR, survey results will be converted from mR/hr by multiplying mR/hr by a factor of 0.87 when needed for comparison to regulatory requirements. This factor is applicable per the Health Physics Society per the following web page: http://www.hps.org/publicinformation/ate/q1055.html.

This summary of the shielding evaluations demonstrate that the SPEC-150 meets dose rate requirements when loaded to capacity with Ir192. The analysis in Appendix E contains 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	0.6 (62)	1.0 (103)	0.8 (81)	0.006 (0.6)	0.010 (1.0)	0.008 (0.8)
Neutron	NA	NA	NA	NA	NA	NA
Total	0.6 (62)	1.0 (103)	0.8 (81)	0.006 (0.6)	0.010 (1.0)	0.008 (0.8)
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				0.050 (5.0)	0.050 (5.0)	0.045 (4.5)
Neutron				NA	NA	NA
Total				0.050 (5.0)	0.050 (5.0)	0.045 (4.5)
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 on the SPEC-150 as prepared for transport with a 131.29 curie Ir-192 source on February 3, 2011 at the conclusion of normal conditions testing. The readings were extrapolated to 150 curies, and the readings at the surface were also corrected by a factor of 1.2 for the ½" distance from the radiation detector probe to the surface of the package. The survey results have also been converted from mR/hr in order to demonstrate that the SPEC-150 meets 10 CFR 71.47 and are as follows (these values were also included in Table 1).

Table 2: Radiation Levels as Prepared for Transport After Normal Conditions Test

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	0.6 (62)	1.0 (103)	0.8 (81)	0.006 (0.6)	0.010 (1.0)	0.008 (0.8)
Neutron	NA	NA	NA	NA	NA	NA
Total	0.6 (62)	1.0 (103)	0.8 (81)	0.006 (0.6)	0.010 (1.0)	0.008 (0.8)
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)

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

Normal conditions testing was performed on 02/03/2011 with a 131.29 curie source to demonstrate that the SPEC-150 meets 10 CFR 71.43(f) and 10 CFR 71.51(a). (This test was performed again as the original testing was performed with a low curie source.) 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 3% 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 Table 3 (all dose rates are expressed in mR/h).

Table 3: Actual Surface Radiation Levels Before and After 02/03/11 Normal Conditions Testing - mR/hr				
Surface	Pre-Drop	Post-Drop	Change	% of Change
Left	86	84	-2.0	-2%
Right	58	60	+2.0	+3%
Bottom	68	70	+2.0	+3%
Top	52	50	-2.0	-4%
Lock	82	72	-10.0	-12%
Outlet	58	58	0	0

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. The extrapolated and corrected surface radiation levels met 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 was 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 mR/h).

Table 4: Radiation Levels After Initial Hypothetical Accident Conditions Testing						
	Actual Readings:			Extrapolated and Corrected:		
Surface:	mR/h	mSv/h	mrem/h	mR/h	mSv/h	mrem/h
Top	20	0.17	17	164	1.43	143
Right	28	.024	24	229	1.99	199
Lock	26	0.23	23	213	1.85	185
Outlet	14	0.12	12	115	1.00	100
Bottom	18	0.16	16	147	1.28	128
Left	20	0.17	17	164	1.43	143

Additional 9 meter drop tests and puncture tests were performed in 1997 (see SAR 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: mR/h		Extrapolated and Corrected: mSv/h (mrem/h)					
	Surface	1m	Surface			1 Meter		
	mR/h	mR/h	mR/h	mSv/h	mrem/h	mR/h	mSv/h	mrem/h
Top	14	1.0	97	0.843	84	5.8	0.050	5.0
Right	14	0.9	97	0.843	84	5.2	0.045	4.5
Lock	11	1.0	76	0.663	66	5.8	0.050	5.0
Outlet	10	0.9	69	0.602	60	5.2	0.045	4.5
Bottom	11	1.0	76	0.663	66	5.2	0.045	4.5
Left	20	0.9	138	1.205	120	5.8	0.050	5.0

Appendix C: List of Changes in Consolidated SAR

#	Change	Text was:	Is:
1.1 1.3.2C 1.3.3B 1.3.4A 1.3.4C 1.3.4D 1.3.4E 1.3.4F 2.4.8 2.4.11 3.1 4.1.1 8.1.6 8.2.6	Accommodate addition of Selenium-75 and Ytterbium-169 to approved contents.	Text formerly referred to Ir-192 as the only radioactive contents.	Either Selenium-75 and Ytterbium-169 were added to the Iridium-192 in the section, or Iridium-192 was eliminated from the description of the radioactive material or source.
1.2	Updated IAEA standard to 1996 revision per 03/28/05 submittal.	IAEA Safety Series No. 6 It is requested that the Certificate of Compliance reflect that it is based on IAEA Safety Series No. 6, 1985 Edition (As Amended 1990).	IAEA Safety Standards Series The SPEC-150 Certificate of Compliance reflects that it is based on IAEA Safety Standards Series No. TS-R-1, 1996 Edition (Revised).
1.3.1	Gross weight is 53.5 pounds per 03/13/09 submittal.	Gross weight: Maximum 53 pounds.	Gross weight: Maximum 53.5 pounds.
1.3.2A	Correct terminology	The device consists of a . . . Multiple securing and locking mechanisms are installed at the lock end of the device and a safety plug is installed in the outlet nipple at the other end.	The package consists of a . . . Multiple securing and locking mechanisms are installed at the lock end of the package and a safety plug is installed in the outlet nipple at the other end.
	Gross weight is 53.5 pounds per 03/13/09 submittal.	The package weighs a maximum of 24 kg (53 pounds).	The package weighs a maximum of 24.3 kg (53.5 pounds).
1.3.2.B	Clarify that more than one shielding pad can be used, change materials per 03/13/09 revision.	The optional shielding pad is not used to qualify a device to meet the allowable radiation levels following the Type B package Hypothetical Accident Conditions tests. The shielding pad is a solid, round, tungsten disk with a maximum 3/4 thickness, and one pound weight.	The optional shielding pad(s) are not used to qualify a device to meet the allowable radiation levels following the Type B package Hypothetical Accident Conditions tests. The shielding pad(s) are solid tungsten, lead or DU, are a maximum of 3/4" thick, and weigh a maximum total of one pound.
1.3.2.D, E, G 2.6.1	Eliminate item numbers, item numbers are no longer listed on drawings.	See Drawing 15B002A, Item Nos. 7 and 8. . . . Drawing 15B002A, Item No. 1. See Drawing 15B002A, Item Nos. 14 and 15. See Drawing 15B002A, Item No. 12. See Drawing 15B002A, Item No. 13. . . . Drawing 15B002A, Item No. 9.	See Drawing 15B002A.
1.3.2.E	Update description of outlet nipple.	An outlet nipple, which is a commercially available male quick disconnect mechanical coupling, is screwed into the outlet panel which is affixed to the outlet end plate on the model SPEC-150.	An outlet nipple is attached to the outlet panel which is affixed to the outlet end plate on the model SPEC-150.

#	Change	Text was:	Is:
1.3.2.F	Remove material for handle, not applicable to package.	An aluminum handle attached with stainless steel rods are provided as a convenience to carry the model SPEC-150 in the field when it is being used as an industrial radiography device.	A handle is provided as a convenience to carry the model SPEC-150 in the field when it is being used as an industrial radiography device.
1.3.2.I.	Remove the CoCA number as it is not relevant, and is subject to change. Remove stainless steel from capsule description.	The primary containment vessel to prevent the release of radioactive material is the sealed source capsule, which meets the requirements of special form radioactive material in 10 CFR 71.75 pursuant to IAEA Certificate of Competent Authority Number USA/0095/S. Approximate dimensions of the stainless steel capsule is one inch long by 1/4 inches diameter.	The primary containment vessel to prevent the release of radioactive material is the sealed source capsule which meets the requirements of special form radioactive material in 10 CFR 71.75 as certified by the IAEA Competent Authority. Approximate dimensions of the capsule are one inch long by 1/4 inches diameter.
1.3.2.J	Correct the drawing number for the lock module in last sentence of first paragraph. Correct grammar in first sentence of second paragraph. Change order of sentences in third paragraph, the device lock is the primary mechanism for holding the source assembly lock in the closed position. Correct drawing numbers in 4th paragraph.	See Drawing 15B625, ASM/Lock Module Housing. The source assembly lock is held in the closed position by two spring loaded plungers located inside the lock module. It is also held in the closed position by the device lock. The device lock is a solid, stainless steel, fan-blade shaped part that is operated by the device key. See Drawings 19B005, 19B006, 19B007 and 190909 which depicts the SPEC-150 Lock Module and the Device and Source Lock Operation.	See Drawings 19B005, Lock Module and 19B006, Device Lock Operation (Locked). The source assembly lock is held in the closed position by the device lock. The device lock is a solid, stainless steel, fan-blade shaped part that is operated by the device key. The lock is also held in the closed position by two spring loaded plungers located inside the lock module. See Drawings 19B005, 19B006, and 190909 which depict the SPEC-150 Lock Module and the Device and Source Lock Operation.
1.3.4.B	Remove the CoCA number as it is not relevant, and is subject to change.	The sealed source capsule meets the requirements of special form radioactive material pursuant to 10 CFR 71.75 as demonstrated by IAEA Certificate of Competent Authority Number USA/0095/S.	The sealed source capsule meets the requirements of special form radioactive material pursuant to 10 CFR 71.75 as demonstrated by an IAEA Certificate of Competent Authority.
1.3.4.C., D.	Remove the term solid metallic wafer, it is unnecessary, and applicable specifically to Iridium. Remove stainless steel from capsule description.	Iridium-192 solid metallic wafers are encapsulated in a stainless steel cylindrical capsule measuring approximately 3/4 inches by 1/4 inches diameter which is swaged onto a flexible cable approximately 7-7/8 inches long forming a source assembly.	Iridium-192, Selenium-75 or Ytterbium-169 is encapsulated in a cylindrical capsule measuring approximately 3/4 inches by 1/4 inches diameter which is swaged onto a flexible cable approximately 7-7/8 inches long forming a source assembly.
		The density of solid metallic iridium is approximately 22.5 grams per cubic centimeter. The weight of the Iridium-192 contents is negligible.	The density of Iridium-192, Selenium-75 and Ytterbium-169 is approximately 22.5, 4.79 or 6.96 grams per cubic centimeter respectively. The weight of the radioactive contents is negligible.

#	Change	Text was:	Is:
2.0	Update welding procedure per revised drawing 15B002A.	All thermal metal joining (TMJ) of structural joints are performed in accordance with SPEC Titanium GTAW TMJ Procedure P51-1, QAM 9.6 of the quality assurance program, U. S. Nuclear Regulatory Commission Certificate of Compliance No. 0102.	All welds depicted on certificate drawings are performed in accordance with either the applicable requirements of ASME Section VIII, Division I, or in accordance with AWS D1.9 and SPEC's Quality Assurance Program, U. S. Nuclear Regulatory Commission Certificate of Compliance No. 0102. For SPEC-150's older than serial number 1475 not meeting these code requirements, all thermal metal joining (TMJ) of structural joints are performed in accordance with SPEC Titanium GTAW TMJ Procedure P51-1.
2.1.1.B.	Remove material for capsule, not relevant to package.	The stainless steel capsule provides the primary containment vessel preventing the release of radioactive material and meets the requirements of 10 CFR 71.75 for special form radioactive material.	The sealed source capsule provides the primary containment vessel preventing the release of radioactive material and meets the requirements of 10 CFR 71.75 for special form radioactive material.
2.1.2.A.	Updated IAEA standard to 1996 revision per 03/28/05 submittal.	Added sentence to paragraph.	The SPEC-150 also meets the requirements of the 1996 Edition of TS-R-1.
2.2	Gross weight is 53.5 pounds per 03/13/09 submittal.	The model SPEC-150 weighs a maximum of 53 pounds.	The model SPEC-150 weighs a maximum of 53.5 pounds.
2.3.3	Refer to new Bill of Materials table that is included in Appendix D, eliminate 440C stainless steel as it is no longer used in the automatic securing mechanism.	Titanium sheet and plate, ASTM B265-90 commercial grade 2, is used for the package shell, end plates, inner bulkhead, and the ASM/lock module. Titanium tubing, ASTM B337 commercial grade 2, is used for the lock cap and support cups. Inside the automatic securing mechanism Series 300 and 440C stainless steel is used.	Titanium sheet and plate is used for the package shell, end plates, inner bulkhead, and the ASM/lock module. Titanium is used for the lock cap and support cups. Inside the automatic securing mechanism Series 300 stainless steel is used. See Bill of Materials (next page) for specific material description effective starting with SPEC-150 serial number 1475. (Bill of Materials is inserted here.)
	Change shielding pad materials per 03/13/09 revision.	Optional tungsten shielding pads are used as needed.	Optional tungsten, lead or depleted uranium shielding pads are used as needed.
	Eliminate description of handle.	Aluminum is used for the carrying handle, but it is not a structural part of the package.	None
2.4.6	Remove "inner" from last sentence.	In fact the test specified for normal conditions of transport did not cause any significant effect on the inner model SPEC-150 package.	In fact the test specified for normal conditions of transport did not cause any significant effect on the model SPEC-150 package.
2.5.2	Add reference to Appendix 9.6 for 1997 hypothetical accident conditions testing.	The results of tests described below in Section 2.9 for hypothetical accident conditions adequately demonstrate that there would be no possibility of 27 Ci Iridium-192 escaping from the package in one week nor would there be any radiation levels exceeding one rem per hour at one meter from the external surface of the package.	The results of tests described below in Section 2.9 and in Appendix 9.6 for hypothetical accident conditions adequately demonstrate that there would be no possibility of 27 Ci Iridium-192 escaping from the package in one week nor would there be any radiation levels exceeding one rem per hour at one meter from the external surface of the package.
2.6.4	Refer to 02/03/11 Normal Conditions Tests.	Add: A used SPEC-150, serial number 331 was used for the normal condition accident tests in 2011.	

#	Change	Text was:	Is:
2.8.7	Refer to 02/03/11 Normal Conditions Tests.	A model SPEC-150 prototype package, Prototype No. 4, was dropped from a distance of 4 feet onto an essentially unyielding surface.	Both a prototype package, Prototype No. 4, and a used package, serial number 331 were dropped from a distance of 4 feet onto an essentially unyielding surface.
2.8.9	Refer to 02/03/11 Normal Conditions Tests.	A model SPEC-150 prototype was subjected to the impact of a 1-1/4 inches diameter steel cylinder weighing 13 lbs falling a distance of 40 inches.	Both a prototype package and a used package were subjected to the impact of a 1-1/4 inches diameter steel cylinder weighing 13 lbs falling a distance of 40 inches.
2.9.1	Add reference to 1997 tests.	Add: Additional free drop tests were performed in 1997 on SPEC-150 serial number 500. See Appendix 9.6 for complete test report.	
2.9.1.A.4	Clarify that all shielding tests were performed with survey meters that measure in mR/hr.	The radiation level at the top center surface of the device after the 3rd drop increased from 0.24 mSv (24 millirem) per hour to 0.8 mSv (80 millirem) per hour.	The radiation level at the top center surface of the device after the 3rd drop increased from 24 mR/hr to 80 mR/hr.
2.9.1.A.5		The radiation level at one meter from the lock end was 5.57 mSv (557 millirem) per hour extrapolated to 150 curies.	The radiation level at one meter from the lock end was 557 mR/ hr extrapolated to 150 curies.
2.9.1.A.7		The cumulative damage resulted in an increase in radiation level at the lock end of the SPEC-150 to 5.57 mSv (557 millirem) per hour at one meter extrapolated to 150 curies of Ir-192 which remains far below the maximum allowable limit of 10 mSv (1000 millirem) per hour at one meter from the surface.	The cumulative damage resulted in an increase in radiation level at the lock end of the SPEC-150 to 557 mR/hr at one meter extrapolated to 150 curies of Ir-192 which remains far below the maximum allowable limit of 10 mSv (1000 millirem) per hour at one meter from the surface.
2.9.1.B	In second paragraph, eliminate references to ANSI N432 and mR/hr terminology.	Eliminate "Shielding pads are limited to a maximum weight of one pound and a maximum thickness of 3/4"inch, and therefore are not a factor in meeting the hypothetical accident condition criteria. For example; assume that a shielding pad of maximum thickness is used to reduce the radiation level at one meter from the surface of the SPEC-150 device to meet the ANSI N432-1980 limit of 2 mR/hr. If the pad was lost due to an accident the maximum resulting radiation level at one meter would be approximately 64 mR/hr which is far below the allowable limit of 1000 mR/hr required to pass the test.".	Eliminated, ANSI N432:1980 is not relevant to Type B packages.
2.9.1.B.2	Clarify that all shielding tests were performed with survey meters that measure in mR/hr.	The radiation levels were less than 0.5 millirem per hour at 1 meter in all directions. This extrapolates to 3.3 millirem per hour at 150 curies.	The radiation levels were less than 0.5 mR/hr at 1 meter in all directions. This extrapolates to 3.3 mR/hr at 150 curies.
2.9.1.B.3		Radiation level readings at one meter were less than 3.3 mrem/hr from all six surfaces.	Radiation level readings at one meter were less than 3.3 mR/hr from all six surfaces.
2.9.2	Add reference to 1997 tests.	Add: Additional puncture tests were performed in 1997 on SPEC-150 serial number 500. See Appendix 9.6 for complete information.	

#	Change	Text was:	Is:
2.9.2.C	Add reference to 1997 tests.	Add: Additional Puncture Test information is documented in Appendix 9.6, "1997 Puncture Tests".	
2.10	Eliminate reference to Iridium-192, wafers, update IAEA reference, statement that iridium wafers could quality as special form, and USA/0095/S. Eliminate reference to Appendix 9.3, it contained the obsolete Certificate of Competent Authority No. USA/0095/S.	Iridium-192 wafers are encapsulated in a capsule which meets the requirements of special form radioactive material pursuant to 49 CFR 173.403(z), 10 CFR 71.77 and Paras 142, 502-504 IAEA Safety Series No. 6 "Regulations for the Safety Transport of Radioactive Material" (1985 Edition as amended 1990). The individual iridium wafers could qualify as special form radioactive material, if it were not for the minimum dimension requirement; but the capsule represents the primary containment vessel. The capsule meets the requirements of special form radioactive material as demonstrated by IAEA Certificate of Competent Authority No. USA/0095/S. See Appendices Section 9.3, Documents.	Radioactive material is encapsulated in a capsule which meets the requirements of special form radioactive material pursuant to 49 CFR 173.403(z), 10 CFR 71.77 and Paras 239, 602-604 IAEA Safety Series TS-R-1 "Regulations for the Safety Transport of Radioactive Material", 1996 Edition (Revised). The capsule represents the primary containment vessel for the radioactive material. The capsule meets the requirements of special form radioactive material as demonstrated by an IAEA Certificate of Competent Authority.
2.10.1	Change IAEA reference and para 502 to 602.	The sealed source capsule meets the minimum dimension requirement of 5 mm for special form radioactive material in compliance with IAEA Safety Series No. 6, para 502.	The sealed source capsule meets the minimum dimension requirement of 5 mm for special form radioactive material in compliance with IAEA Safety Series TS-R-1, para 602.
2.10.5	Eliminate reference to iridium wafers.	The capsule and the iridium wafers will withstand sustain temperatures greater than 1475° F for ten minutes without adverse effects.	The capsule and the radioactive material will withstand sustain temperatures greater than 1475° F for ten minutes without adverse effects.
2.10.6	Remove the CoCA number as it is not relevant, and is subject to change.	As a result of previously performed evaluations resulting in the issuance of IAEA Certificate of Competent Authority No. USA/0095/S and . . .	As a result of previously performed evaluations resulting in the issuance of an IAEA Certificate of Competent Authority and . . .
5.0	RAI-13: Revise first sentence to remove information that is not applicable to the package (it is applicable to the radiography device).	A shielding evaluation of the model SPEC-150 was performed in conjunction with the application as an industrial radiography device in accordance with 10 CFR 34.20 and American National Standards Institute N432-1980.	A shielding evaluation of the model SPEC-150 was performed to demonstrate compliance with NRC and IAEA requirements.

#	Change	Text was:	Is:
5.0	RAI-13: Eliminate remaining portion of first paragraph to remove information that is not applicable to the package (it is applicable to the radiography device).	A shielding evaluation of the model SPEC-150 was performed in conjunction with the application as an industrial radiography device in accordance with 10 CFR 34.20 and American National Standards Institute N432-1980. The same test, test packages and test result data are included in this application. The ANSI criteria after the normal conditions test is more stringent than the 10 CFR Part 71 criteria. The radiation level can not exceed 200 millirem per hour at the surface or 50 millirem per hour at 50 millimeters. The radiation level at one meter from the surface can not exceed 2 millirem per hour. The acceptance criteria after the 9 meter drop and puncture test is the same as the 10 CFR Part 71 criteria, which requires that the radiation level can not exceed 1000 millirem per hour at one meter. The NRC Office of Nuclear Materials Safety and Safeguards reviewed the model SPEC-150 industrial radiography device application and the Louisiana Division of Radiation Protection reviewed and approved it.	Removed.
5.0	Add reference to additional testing. Remove statement that theoretical calculations have not been used as they were used to demonstrate that the radiation levels with Se-75 and Yb-169 would be lower than those for Ir-192.	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.	Adequate shielding design for the model SPEC-150 was 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.
5.0	Clarify that all shielding tests were performed with survey meters that measure in mR/hr.	Added this statement in second paragraph: All shielding tests were performed with survey meters that measure in mR/hr. Therefore, survey results will be converted from mR/hr by multiplying mR/hr by a factor of 0.87 when needed for comparison to regulatory requirements. This factor is applicable per the Health Physics Society. See http://www.hps.org/publicinformation/ate/q1055.html.	
5.0	In last paragraph, eliminate reference to original radiation readings.	The unadjusted surface radiation readings and their locations are presented on sketches in Appendix 9.	Removed

#	Change	Text was:	Is:
5.1	In paragraph 2, replace original shielding information with reference to table from NUREG-1886 and explanation that the shielding results for Ir-192 were used.	Measurements were taken on the surface of Prototype No. 4 before the normal conditions of transport and hypothetical accident condition tests. Radiation readings were taken at points on an approximate one-inch by one-inch grid located on each of the six sides of the package. This provided 75 points on the top, 90 points each on the bottom and two sides, and 40 points each on the end plates for a total of 425 measurement points. A correction factor was applied for the diameter of the detector probe. Measurements were taken with a 137 Ci Iridium-192 source and the results extrapolated to 150 Ci Iridium-192. (Original radiation readings chart) 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 8/26/94. The survey was made before both 30-foot drops and a puncture test conducted on 8/26/94. See Appendix 9. Measurements were taken of the maximum radiation level at one meter from each of the six surfaces of Prototype No. 4 using a 137 Ci Iridium-192 source and the results were extrapolated to 150 Ci Iridium-192. (Original radiation readings chart)	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 Ir-192. Ir-192 will produce a higher external dose rate than Se-75 or Yb-169. See analysis in Appendix 9.7 for the calculated dose rates for Se-75 and Yb-169. (New shielding chart, Table 1, from Appendix B)
5.2	Replace entire section with information from Appendix B demonstrating that the SPEC-150 met normal condition radiation limits at the surface and at one meter after normal conditions testing in 2011.	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: (Original radiation readings chart)	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 on the SPEC-150 as prepared for transport with a 131.29 curie Ir-192 source on February 3, 2011 at the conclusion of normal conditions testing. (See Appendix 9.3) The readings were extrapolated to 150 curies, and the readings at the surface were also corrected to allow for the ½" 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). (Table 2 from Appendix B)

#	Change	Text was:	Is:
5.2	Replace entire section with information from Appendix B demonstrating that the SPEC-150 met normal condition radiation limits at the surface and at one meter after normal conditions testing in 2011. Replace original survey readings with those from the normal conditions testing performed 02/03/11 demonstrating that there was no significant change in radiation levels as a result of normal conditions testing.	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.	Normal conditions testing was performed on 02/03/2011 with a 131.29 curie source to demonstrate that the SPEC-150 meets 10 CFR 71.43(f) and 10 CFR 71.51(a). (This test was performed again as the original testing was performed with a low curie source.) 10 CFR 71.43(f) and 10 CFR 71.51(a) 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 3% 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 Table 3 (all dose rates are expressed in mR/h). (Table 3 from Appendix B)
5.3	Eliminate description of testing on Prototype 2, replace discussion of Prototype 4 testing with information from Appendix B.	Although not required, four successive 30 foot drop tests were conducted on Prototype No. 2 to obviate any question about selection of the most vulnerable point of impact followed by a puncture test. The damage was cumulative. Radiation levels were measured after the fourth 9 meter drop test and again after the one meter puncture test. Since the design of the SPEC-150 was revised after the tests were conducted on Prototype No. 2 this data is not required. The data on Prototype No. 2 is included in this application to supplement the data provided for Prototype No. 4 and to further demonstrate that the shield design meets the package shielding requirements. The maximum radiation levels at one meter from each of the six surfaces were extrapolated to 150 Ci Iridium-192 are tabulated below: (Original radiation readings chart)	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 in 1994, 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 mR/h).

#	Change	Text was:	Is:
5.3	Replace discussion of Prototype 4 testing with information from Appendix B.	Prototype No. 4 was subjected to a 9 meter drop test, followed by a one meter puncture test, and followed by a second 30 foot drop test on August 26, 1994. After the conclusion of these cumulative tests the maximum radiation levels measured at one meter from each of the six surfaces of the camera were extrapolated to 150 Ci Iridium-192 and the results are tabulated below: (Original radiation readings chart)	(Table 4 from Appendix B)
5.3	Replace discussion of Prototype 4 testing with information from Appendix B.	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.	Eliminated

#	Change	Text was:	Is:
5.3	Replace discussion of Prototype 4 testing with information from Appendix B regarding 1997 testing.	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. 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.	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 from Appendix B)
5.4	Update to eliminate information specific to 1994 tests, and add reference to Se-75 and Yb-169.	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.	The source assembly used in the normal condition of transport and hypothetical accident conditions radiation level measurements was a model SPEC G-60 loaded with Ir-192. Ir-192 will produce a higher external dose rate than Se-75 or Yb-169. See analysis in Appendix 9.7 for the calculated dose rates for Se-75 and Yb-169.
5.5	Add reference to models used for 1997 and 2011 testing, remove statement about theoretical calculations.	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.	Physical radiation measurements were performed on prototype packages, SPEC-150 serial number 500, and a used SPEC-150 serial number 331.

#	Change	Text was:	Is:
5.6	Eliminate references to prototype 2, it is not relevant to the shielding evaluation. Reword paragraph to cite the 1997 and 2011 tests to demonstrate compliance with normal and hypothetical accident conditions shielding requirements.	The results and evaluations conservatively showed that there was no significant increase in radiation levels for the normal condition of transport tests. The maximum radiation level of 557 mrem/hr at one meter from the surface of Prototype No. 2 package after conclusion of four drop tests, and one puncture test demonstrates the model SPEC-150 meets the one rem per hour at one meter criteria for the hypothetical accident conditions. The shielding evaluation of Prototype No. 2 is significant to the extent that the package meets the shielding criteria for a Type-B package. The design of the depleted uranium shield used in Prototype No. 2 is the same design that was tested in Prototype No. 4 and will be used for production packages. The maximum radiation levels after the conclusion of the four 30-foot drop tests and two puncture tests on the redesigned Prototype No. 4 conclusively demonstrates that the model SPEC-150 meets the shielding requirements for a Type B package. This application reflects the design used for Prototype No. 4.	Additional normal conditions tests were conducted in 2011 to conclusively demonstrate the SPEC-150 was 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. 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. Additional hypothetical accident conditions tests conducted in 1997 demonstrated that the SPEC-150 is 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.
7.1.1	Add check of fasteners, see RAI-9.		Add: Visually check the exposed fasteners and welds.
8.1.2	Change terminology from critical structural to important to safety with reference to the weld joints. Add reference to drawing 15B002A for a depiction of welds. Describe the weld joints that are visually inspected. Clarify that liquid penetrant inspections include a visual inspection.	. . . on critical structural joints. All exterior TMJ thermal metal joining structural joints are liquid penetrant tested. The method of inspection of the production packages will consist of a combination of in-process and final inspection of all structural TMJ weld joints that connect the titanium plates. Dye 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.	. . . 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.

#	Change	Text was:	Is:
8.1.2	Describe inspection codes that will be used, the same code will be used for inspections that is used for the welding. Move statement on where visual inspection is performed to the beginning of the section.	Dye penetrant inspection and inspector qualification is performed in accordance with ASTM E-165. The accept/reject criteria meets ASME Section VIII, Division 1, "Rules for Construction of Pressure Vessels" Appendix 8, Paragraph 8.3 "Evaluation of Indications" and 8.4 "Acceptance Standards." Visual inspection is performed on the TMJ 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. Visual inspection and inspector qualification is performed in accordance with ASME Section V, "Nondestructive Examination" Article 9, "Visual Examination." The accept-reject criteria meets ASME Section VIII, Division 1, "Rules for Construction of Pressure Vessels" UW-36 "Fillet Welds."	Visual and Liquid Penetrant inspections are performed in accordance with the same code as that used for welding, either the ASME Code for Boilers and Pressure Vessels, Section VIII, Division 1 or AWS D1.9. In either case, either the 2007 edition or an updated revision will be used.
8.1.5	Eliminate details of conformance to radiography device requirements. Require that every shield be surveyed prior to first use on the surface and at one meter.	A radiation profile is performed on the camera as part of the final inspection. Although 10 CFR 71.47 allows maximum radiation levels of 200 mR/hr at the package's surface and 10 mR/hr at one meter from the package's surface, the model SPEC-150 will not exceed 200 mrem/hr at the surface of the camera and 2 mrem/hr at one meter from the surface of the camera when the activity is extrapolated to 150 Ci of Iridium-192 in compliance with 10 CFR 34.20(a) which references American National Standards Institute N432-1980. Prior to shipment of the camera with a source assembly the package is surveyed to assure compliance with transportation requirements.	Every shield is surveyed prior to first use to determine that it will meet NRC shielding requirements when extrapolated to 150 Ci. The survey is performed on the surface of the SPEC-150 and at one meter. Prior to each shipment of the camera with a source assembly the package is surveyed to assure compliance with transportation requirements.
8.2.1	Add check of fasteners, see RAI-9.	Periodic 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.	Periodic 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. However, a visual check of all external fasteners and welds should be performed.
9.1	Update drawing list per Appendix A	Eliminate drawings no longer referenced on the CoC.	List drawings from Appendix A.
9.3	Eliminate special form CoCA, replace with normal conditions tests.	9.3 IAEA Certificate of Competent Authority USA/0095/S, Revision 7	2011 Normal Conditions Test Report
9.4	Added "1994" for clarification	Sketches of Drop Test Impact Orientations	1994 Sketches of Drop Test Impact Orientations
9.5		Sketches of Highest Surface Radiation Survey Data	1994 Sketches of Highest Surface Radiation Survey Data

#	Change	Text was:	Is:
9.6	Renamed report	1997 Puncture Tests	1997 30' Drop Test & Validation of Previous Puncture Tests
9.7	Added Shielding Analysis, Se-75 and Yb-169	None	9.7 Shielding Analysis, Se-75 and Yb-169

Appendix D: Materials of Construction

<table>
<tr><th colspan="3">BILL OF MATERIALS (serial numbers 1475 and newer)</th></tr>
<tr><th>Component</th><th colspan="2">Material</th></tr>
<tr><td>Bottom plate</td><td>Grade 2 Titanium</td><td rowspan="7">ASME SB-265 or ASTM B 265
ASME SB-337 or ASTM B 337
ASME SB-338 or ASTM B 338
ASME SB-348 or ASTM B 348
ASME SB-381 or ASTM B 381
ASME SB-861 or ASTM B 861
ASME SB-862 or ASTM B 862</td></tr>
<tr><td>Bulkheads</td><td>Grade 2 Titanium</td></tr>
<tr><td>Bulkhead cups</td><td>Grade 2 Titanium</td></tr>
<tr><td>Housing cover</td><td>Grade 2 Titanium</td></tr>
<tr><td>Lock module housing</td><td>Grade 2 Titanium</td></tr>
<tr><td>Lock module faceplate</td><td>Grade 2 Titanium</td></tr>
<tr><td>Lock cap</td><td>Grade 2 Titanium</td></tr>
<tr><td>Lock Cap Insert</td><td>Tungsten</td><td>ASTM B 777</td></tr>
<tr><td>Positioning shim</td><td>Tungsten</td><td>ASTM B 777</td></tr>
<tr><td>Source assembly lock</td><td rowspan="2">300 Series*
Stainless Steel

* The design of the SPEC-150 does not subject the stainless steel parts to tensile stress that approaches or exceeds the yield strength of any 300 series grade stainless steel.</td><td rowspan="2">ASME SA 213A or ASTM 213
ASME SA 240A or ASTM 240
ASME SA 249A or ASTM 249
ASME SA 269A or ASTM 269
ASME SA 270A or ASTM 270
ASME SA 271A or ASTM 271
ASME SA 276A or ASTM 276
ASME SA 312A or ASTM 312
ASME SA 336A or ASTM 336
ASME SA 358A or ASTM 358
ASME SA 376A or ASTM 376
ASME SA 409A or ASTM 409
ASME SA 473A or ASTM 473
ASME SA 479A or ASTM 479
ASME SA 480A or ASTM 480
ASME SA 484A or ASTM 484
ASME SA 554A or ASTM 554
ASME SA 581A or ASTM 581
ASME SA 582A or ASTM 582
ASME SA 666A or ASTM 666
ASME SA 688A or ASTM 688</td></tr>
<tr><td>Device lock</td></tr>
</table>

BILL OF MATERIALS (serial numbers 1475 and newer)		
Component	Material	
Safety Plug Assembly	300 Series Stainless Steel	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 quick disconnect ASME SA 213A or ASTM 213 ASME SA 240A or ASTM 240 ASME SA 249A or ASTM 249 ASME SA 269A or ASTM 269 ASME SA 270A or ASTM 270 ASME SA 271A or ASTM 271 ASME SA 276A or ASTM 276 ASME SA 312A or ASTM 312 ASME SA 336A or ASTM 336 ASME SA 358A or ASTM 358 ASME SA 376A or ASTM 376 ASME SA 409A or ASTM 409 ASME SA 473A or ASTM 473 ASME SA 479A or ASTM 479 ASME SA 480A or ASTM 480 ASME SA 484A or ASTM 484 ASME SA 554A or ASTM 554 ASME SA 581A or ASTM 581 ASME SA 582A or ASTM 582 ASME SA 666A or ASTM 666 ASME SA 688A or ASTM 688 RR-W-410 Cable Manufacturing Standard

Appendix E:
John Munro Report
Shielding of the SPEC-150 with ^{75}Selenium and ^{169}Ytterbium
as revised 02/02/2011

Report

By: John J. Munro III

Date: 02 February 2011

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)

As explained by J. H. Hubbell and S. M. Seltzer of NIST[i], a narrow beam of monoenergetic photons with an incident intensity I_0, penetrating a layer of material with mass thickness x and density ρ, emerges with intensity I given by the exponential attenuation law:

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

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

For a broad-beam of monoenergetic photons, the "Build-Up factor" must be included. The Broad Beam transmission of photons through a material can be expressed as:

$$\tau(E) = \frac{I}{I_0} = B(E,x)\,e^{-\frac{\mu}{\rho}(E)\rho x} \qquad [3]$$

where: $B(E,x)$: Exposure Build-Up Factor for photons of Energy E and shielding thickness X

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

$$\tau = \sum_i f(E)_i\, \tau(E) = \sum_i f(E)_i\, B(E,x)_i\, e^{-\frac{\mu}{\rho}(E)_i \rho x} \qquad [4]$$

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

^{75}Selenium

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

For ^{75}Selenium, we have used the exposure rate constant of 0.20 R m^2 h^{-1} Ci^{-1}, as presented in the 1970 Radiological Health Handbook,[ii] and have not used the exposure rate constant presented in the 1992 edition of The Health Physics and Radiological Health Handbook,[iii] which is merely a recitation of the data presented in ORNL/RISC-45.[iv]

To justify the use of this exposure rate constant, we offer the following explanation.

The exposure rate constant is the exposure rate at a specific distance from a given amount of a photon-emitting radionuclide. Typically, this is presented in units of Roentgens per hour (R/hr) at a distance of one (1) meter from a one (1) curie point source of that radionuclide.

This concept of the exposure rate constant provides a convenient means for determining the activity of a source based upon a measurement of the exposure rate at a specific distance from the source.

In the case of ^{75}Selenium, this method is complicated by the photon spectrum. The exposure rate constant for ^{75}Selenium is reported to be 1.03 R-m^2/h-Ci in ORNL/RISC-45 and 0.595 R-m^2/h-Ci by Shilton.[v] These values are presented for an unencapsulated non-self-absorbing point source of ^{75}Selenium.

However, approximately 67% of this exposure rate is a result of photons with energies less than 12 keV.

The typical radiography source is encapsulated in stainless steel, with a wall thickness on the order of 0.25 mm. The transmission of a 12 keV photon through 0.25 mm of steel is 2.5 x 10^{-11}. Therefore, essentially none of the less than 12 keV photons emerge from the source encapsulation. (Interestingly, in the absence of any encapsulation, the transmission of 12 keV photons through one meter of air is less than 64%.)

Consequentially, if the intrinsic exposure rate constant (which includes the <12 keV photons) were used to determine the activity of an encapsulated source based upon measurement of the exposure rate at some specified distance (in which there were virtually no <12 keV photons), the activity would be significantly underestimated (by a factor of ~3). This could lead to significant safety and regulatory issues.

To account for this anomaly, a number of investigators have determined the exposure rate constant based only on the photons with energies above 12 keV. These have been reported in the 1970 edition of the Radiological Health Handbook[vi] as 0.20 R-m^2/h-Ci, by Shilton[vii] as 0.201 R-m^2/h-Ci, and by Weeks et al.[viii] as 0.199 R-m^2/h-Ci.

Most recently, Ninkovic et al.[ix] has reported: "In the process of analysing accessible data on the air kerma rate constants and its precursors for many radionuclides used most often in practice (6–17) it was concluded that published data are in strong disagreement." They cite ORNL/RISC-45 as one of these references. They continue: "That is the reason we decided to recalculate this (sic) quantities on the basis of the latest data on gamma ray spectra and on the latest data for mass energy-transfer coefficients for air." For ^{75}Selenium, Ninkovic et al. have determined an exposure rate constant of 0.205 R-m^2/h-Ci using only photons above 20 keV.

For these reasons, we have determined that the use of the value of 0.20 R-m^2/h-Ci permits a reasonably accurate determination of the activity of an encapsulated source based on measurement of the exposure rate at a specific distance from the source. This has become the accepted value of the exposure rate constant in the industrial radiography industry and is currently used by the only other supplier of ^{75}Selenium sources in the United States.

Moreover, the NRC has endorsed this value of 0.20 R-m^2/h-Ci, taken from the 1970 edition of the Radiological health Handbook, in a several NRC Certificates of Compliance (Certificate 9296, issued 6 June 2008; Certificate 9269, issued very recently on 14 September 2010.

Therefore, as noted above, for ^{75}Selenium, we have used the exposure rate constant of 0.20 R m^2 h^{-1} Ci^{-1}, as presented in the 1970 Radiological Health Handbook.

The ^{75}Selenium spectrum consists of 20 photons with energies above 15 keV.[x] The mass attenuation coefficient for uranium for each of these photon energies was interpolated from the values of NIST.[xi]

The minimum shielding of the SPEC-150 is 48 mm of Uranium. Exposure Buildup values for each energy and a thickness of 48 mm of Uranium were interpolated from data for Uranium in ANS/ANSI-6.4.3-1991.[xii]

Using Eq. [4], the value of the transmission, $\tau(E)$, was calculated for this 48 mm thickness of uranium (ρ=18.7 g cm^{-3}). The details of these calculations are presented in the following table:

Energy (keV)	$f(E)_i$	$\frac{\mu}{\rho}(E)$	Buildup $B(E,x)$	$\tau(E)$
24.4	1.95E-04	69.1125	1.02	0.000E+00
66.1	1.95E-03	5.5755	1.80	1.578E-220
80.9	1.42E-05	3.3370	2.20	2.584E-135
96.7	6.96E-03	2.1270	2.20	1.867E-85
121.1	4.78E-02	4.3191	2.82E+12	5.770E-158
136.0	1.86E-01	3.2884	1.43E+11	1.721E-118
198.6	7.46E-03	1.3495	3.57	6.618E-55
249.4	5.96E-08	0.7898	1.99	1.932E-38
264.7	4.19E-01	0.6868	1.84	1.294E-27
279.5	1.89E-01	0.6039	1.78	9.690E-25
303.9	1.10E-02	0.4781	1.60	4.043E-21
373.5	2.59E-05	0.3392	1.90	2.942E-18
400.7	1.30E-01	0.3018	1.85	4.118E-13
418.8	1.40E-04	0.2804	1.90	3.125E-15
468.6	4.51E-06	0.2325	2.00	7.773E-15
542.4	1.99E-07	0.1823	2.10	3.276E-14
557.8	1.42E-08	0.1740	2.10	4.910E-15
572.2	5.74E-04	0.1668	2.10	3.811E-10
617.8	7.72E-05	0.1468	2.30	3.373E-10
821.6	3.09E-06	0.0972	2.22	1.115E-09
			Total Transmission, τ:	1.834E-09

The resultant calculated transmission of photons of ^{75}Selenium through 48 mm of uranium is 1.834 x 10^{-9}.

Using this value of transmission the minimum shielding of the SPEC-150, (i.e. 48 mm) in Eq. [1], the exposure rates, when containing 150 Ci of ^{75}Selenium are calculated to be:

	At Surface (0.068 m from Source)	At 1 meter from Surface (1.068 m from Source)
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.

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.

For similar reasons described above for ^{75}Selenium, we have not used the exposure rate constant of ORNL/RISC-45. Rather, we have used the value of 0.125 R-m^2/h-Ci for the ^{169}Ytterbium exposure rate constant taken from a very recent paper by Cazeca et al.[xiii] These authors characterized a high dose rate brachytherapy source of ^{169}Ytterbium with physical characteristics very similar to industrial radiography sources. Therefore, we have determined that the use of the value of 0.125 R-m^2/h-Ci permits a reasonably accurate determination of the activity of an encapsulated source based on measurement of the exposure rate at a specific distance from the source.

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 minimum shielding of the SPEC-150 is 48 mm of Uranium. Exposure Buildup values for each energy and a thickness of 48 mm of Uranium were interpolated from data for Uranium in ANS/ANSI-6.4.3-1991.[xiv]

Using Eq. [4], the value of the transmission, $\tau(E)$, was calculated for this 48 mm thickness of uranium (ρ=18.7 g cm^{-3}). The details of these calculations are presented in the following table:

Energy (keV)	$f(E)_i$	$\frac{\mu}{\rho}(E)$	Buildup $B(E,x)$	$\tau(E)$
20.8	2.16E-03	103.8415	1.02	0.000E+00
42.8	3.47E-04	16.7222	2.13	0.000E+00
45.9	1.21E-05	13.9510	2.25	0.000E+00
49.8	1.16E-01	11.3945	2.28	0.000E+00
50.6	6.50E-04	10.9245	2.28	0.000E+00
50.7	1.98E-01	10.8529	2.28	0.000E+00
50.9	6.47E-04	10.7921	2.28	0.000E+00
51.5	1.92E-05	10.4488	2.28	0.000E+00
57.3	1.95E-02	7.9838	2.25	0.000E+00
57.5	3.76E-02	7.9121	2.25	0.000E+00
59.0	1.26E-02	7.4066	2.32	5.495E-291
63.0	2.15E-03	6.2806	2.27	7.141E-248
63.1	8.51E-02	6.2529	2.27	3.410E-245

Energy (keV)	$f(E)_i$	$\frac{\mu}{\rho}(E)$	Buildup $B(E,x)$	$\tau(E)$
65.9	9.82E-06	5.6166	2.27	2.505E-224
72.0	3.63E-06	4.4799	2.22	1.855E-180
85.1	2.91E-06	2.9407	2.19	1.474E-120
93.6	5.59E-03	2.3106	2.00	9.493E-93
95.7	2.46E-06	2.1855	2.00	3.137E-91
95.9	2.46E-06	2.1769	2.00	6.824E-91
98.0	2.12E-06	2.0577	2.00	2.585E-86
101.4	1.01E-05	1.8881	2.06	5.193E-79
105.2	6.86E-06	1.7212	2.10	1.155E-72
109.8	4.82E-02	1.5452	2.15	6.035E-62
113.6	1.44E-05	1.4166	2.15	1.855E-60
114.0	1.16E-05	1.4054	2.15	4.098E-60
117.4	1.20E-04	4.6498	8.34E+12	5.519E-173
118.2	5.67E-03	4.5749	5.44E+12	1.416E-168
129.9	1.02E-03	3.6606	1.55E+11	3.160E-135
130.5	3.88E-02	3.6222	1.55E+11	3.764E-132
156.7	4.21E-05	2.3556	9.73E+07	6.107E-89
173.9	6.74E-06	1.8449	1.82E+04	1.476E-73
177.2	1.10E-01	1.7644	4.00E+03	7.314E-67
193.2	4.10E-05	1.4408	17.98	5.017E-60
198.0	2.02E-01	1.3599	3.55	6.994E-54
199.8	9.11E-05	1.3310	3.55	4.198E-56
206.0	2.01E-05	1.2384	3.13	3.327E-53
213.9	1.80E-05	1.1329	2.78	3.424E-49
226.3	1.65E-06	0.9927	2.12	7.047E-45
228.7	1.34E-06	0.9682	2.12	5.116E-44
240.3	8.10E-04	0.8617	2.12	4.405E-37
261.1	1.32E-02	0.7092	1.90	5.644E-30
291.2	3.84E-05	0.5486	1.90	2.997E-26
294.5	9.02E-06	0.5340	1.90	2.606E-26
301.7	2.13E-05	0.4839	1.58	4.609E-24
306.8	8.51E-04	0.4706	1.58	6.076E-22
307.5	2.84E-03	0.4688	1.58	2.377E-21
307.7	9.54E-02	0.4683	1.58	8.371E-20
334.0	1.81E-05	0.4087	1.75	3.711E-21
336.6	9.87E-05	0.4033	1.76	3.289E-20
356.7	1.58E-06	0.3662	1.76	1.478E-20
370.9	1.03E-04	0.3433	1.76	7.517E-18
379.3	4.90E-06	0.3307	1.76	1.113E-18
386.7	4.09E-06	0.3202	1.76	2.371E-18
452.6	2.59E-07	0.2464	1.93	1.243E-16
464.7	5.31E-08	0.2358	1.93	6.591E-17
465.7	2.81E-06	0.2350	1.93	3.753E-15
466.7	2.87E-07	0.2341	1.93	4.148E-16
475.0	2.92E-06	0.2274	1.93	7.734E-15
494.4	2.30E-05	0.2127	2.11	2.482E-13
500.4	1.40E-07	0.2085	2.11	2.197E-15
507.8	2.41E-08	0.2034	2.11	5.986E-16
515.1	6.81E-05	0.1986	2.10	2.579E-12

Energy (keV)	$f(E)_i$	$\frac{\mu}{\rho}(E)$	Buildup $B(E,x)$	$\tau(E)$
528.6	2.01E-06	0.1903	2.10	1.611E-13
546.2	2.59E-08	0.1802	2.10	5.144E-15
562.4	2.12E-06	0.1716	2.10	9.076E-13
570.9	2.00E-06	0.1674	2.10	1.255E-12
579.9	3.54E-05	0.1631	2.15	3.334E-11
600.6	2.16E-05	0.1538	2.20	4.794E-11
624.9	9.70E-05	0.1440	2.20	5.190E-10
633.3	1.38E-07	0.1408	2.15	9.583E-13
642.9	1.55E-06	0.1374	2.15	1.471E-11
663.6	4.02E-06	0.1303	2.15	7.206E-11
693.5	1.89E-07	0.1211	2.20	7.909E-12
710.4	7.54E-07	0.1163	2.20	4.842E-11
739.4	4.21E-08	0.1088	2.20	5.305E-12
760.2	1.96E-08	0.1039	2.30	4.009E-12
773.4	5.01E-06	0.1010	2.30	1.333E-09
781.6	7.25E-08	0.0992	2.35	2.313E-11
			Total Transmission, τ:	2.115E-09

The resultant calculated transmission for photons of ^{169}Ytterbium through 48 mm of uranium is 2.115E-09.

Using this value of transmission for the minimum shielding of the SPEC-150, (i.e. 48 mm) in Eq. [1], the exposure rates, when containing 150 Ci of ^{169}Ytterbium are calculated to be:

	At Surface (0.068 m from Source)	At 1 meter from Surface (1.068 m from Source)
Calculated (48 mm Uranium)	0.0086 mR/hr	3.5×10^{-5} mR/hr

This result is less than 0.004% of the regulatory limit for the surface of the container and less than 0.0004% 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.

---

i 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, NISTIR 5632 http://physics.nist.gov/PhysRefData/XrayMassCoef/chap2.html accessed 02 February 2011

ii Radiological Health Handbook, rev. ed., U.S. Public Health Service, Bureau of Radiological Health, Rockville, MD, 1970.

iii Shleien B (ed), The Health Physics and Radiological Health Handbook, Revised Edition, Published by Scinta, Inc. 2421 Homestead Drive , Silver Spring Md. 20902, 1992

[iv] ORNL/RSIC-45, "Specific Gamma-Ray Dose Constants for Nuclides Important to Dosimetry and Radiological Assessment", May, 1982

[v] Shilton MG, Advanced, Second-Generation Selenium-75 Gamma Radiography Sources, presented at the 15th World Conference on Non-Destructive Testing, Rome, Italy, 15-21 October 2000 Revised and amended August 2003

[vi] Radiological Health Handbook, rev. ed., U.S. Public Health Service, Bureau of Radiological Health, Rockville, MD, 1970.

[vii] Shilton MG, Advanced, Second-Generation Selenium-75 Gamma Radiography Sources, presented at the 15th World Conference on Non-Destructive Testing, Rome, Italy, 15-21 October 2000 Revised and amended August 2003

[viii] 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.

[ix] Ninkovic MM Raicevic JJ and Adrovic F, Air Kerma Rate Constants For Gamma Emitters Used Most Often In Practice, Radiation Protection Dosimetry (2005), Vol. 115, No. 1–4, pp. 247–250

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

[xi] 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

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

[xiii] Cazeca MJ, Medich DC, and Munro JJ, Monte Carlo characterization of a new Yb-169 high dose rate source for brachytherapy application, Med Phys. 37(3):1129-1136, March 2010.

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

SOURCE PRODUCTION AND EQUIPMENT CO., INC.
113 Teal Street St. Rose. Louisiana 70087

CONSOLIDATED APPLICATION
for
NRC CERTIFICATE OF COMPLIANCE
USA/9263/B(U)

Model SPEC-150
Type B(U) Radioactive Material Package
February 14, 2011

TABLE OF CONTENTS

1. GENERAL INFORMATION

1.1 Introduction

The Source Production & Equipment Company, Inc. model SPEC-150 is an industrial radiography exposure device approved for use by the Louisiana Radiation Protection Division, and it is authorized to contain a maximum source activity of 150 Curies (5.55 TBq) (output) of Iridium-192, Selenium-75 or Ytterbium-169 as a sealed source. It is used as a package to transport the sealed Iridium-192, Selenium-75 or Ytterbium-169 source by licensed industrial radiographers as private carriers to perform nondestructive testing, and it is transported by common carriers and for export and import.

1.2 IAEA Safety Standards Series

The SPEC-150 Certificate of Compliance reflects that it is based on IAEA Safety Standards Series No. TS-R-1, 1996 Edition (Revised).

1.3 Model SPEC-150 Packaging

1.3.1 Gross weight: Maximum 53.5 pounds.

1.3.2 Description

A. The package consists of a depleted uranium shield inside a welded titanium housing measuring approximately 14.1 cm (5-9/16 inches) high, 13.6 cm (5-3/8 inches) wide, and 36.8 cm (14-1/2 inches) long. See Drawing 15B000, Isometric View and 15B002A, Full Sectional View. The depleted uranium shield includes a curved S-Tube that the source travels through when used as a radiography device. See Drawing 15B008, Depleted Uranium Shield. Depleted uranium is cast around a titanium or zircalloy S-Tube to provide a minimum of 1.8 inches of radiation shielding from the center of the shield which contains a sealed source capsule. The DU shield is coated with a primer paint by the manufacturer of the casting. The source assembly is secured in the S-tube during transportation. Multiple securing and locking mechanisms are installed at the lock end of the package and a safety plug is installed in the outlet nipple at the other end. The package weighs a maximum of 24.3 kg (53.5 pounds). The DU shield weighs a maximum of 37-1/4 pounds.

B. The S-Tube and depleted uranium shield is designed to prevent direct streaming of radiation through the S-Tube even with the lock cap and safety plug removed. Radiation levels are not uniform among depleted uranium castings. Occasionally shielding pads of tungsten, lead or DU not exceeding one pound are used to further reduce low level radiation where marginal radiation levels have been found to exist at the surface or one meter from the surface of the package in order to meet the radiation level requirements of 10 CFR 71.47. The optional shielding pad(s) are not used to qualify a device to meet the allowable radiation levels following the Type B package Hypothetical Accident Conditions tests. The shielding pad(s) are solid tungsten, lead or DU, are a maximum of 3/4" thick, and weigh a maximum total of one pound. It is attached directly to the coated surface of the depleted uranium shield with an epoxy potting compound. It is further secured in place by a polyurethane foam material with a density of two - three pounds per cubic foot that fills the interior cavity of the package and completely surrounds the DU shield. The pad is used at any accessible location on the surface of the DU shield. However, it cannot be used at any location that would require the modification of any component of the package to install. Historically, the most common location for the pad is on the hot top of the DU shield (left side of the package).

C. Iridium-192, Selenium-75 and Ytterbium-169 are neither fissionable nor a neutron emitters, therefore no materials are used as neutron absorbers or moderators.

D. The depleted uranium shield is secured in the model SPEC-150 package by two titanium cups which are filled with Devcon F epoxy potting compound. The cups are welded to the outlet end plate and the inner bulkhead plate. See Drawing 15B002A. Lateral movement of the shield toward the outlet end is limited by the outlet end plate, Drawing 15B002A. Lateral movement toward the lock end is limited by the metal Positioning Shim on the inner bulkhead plate shown on Drawing 15B002A. The inner bulkhead is in direct contact with the ASM/Lock Module which is bolted to the lock end plate. Therefore, movement of the shield toward the lock end in an accident is resisted by the combined structural support provided by the inner bulkhead and the lock end plate. The strong construction of the ASM/Lock Module prevents crushing in an accident. See Drawing 15B002A. The titanium cups limit movement of the shield in the other directions. An automatic securing mechanism (ASM)/lock module secures and also locks the source assembly ("pigtail") with the sealed source capsule inside the S-tube of the model SPEC-150 radiography exposure device to meet the requirements of 10 CFR 34.23(a) and equivalent agreement state regulations.

E. An outlet nipple is attached to the outlet panel which is affixed to the outlet end plate on the model SPEC-150. The outlet nipple provides a means of connecting a source tube when the model SPEC-150 is used as a radiography exposure device and serves no structural purpose. During shipment a source safety plug, a female quick disconnect coupling with a stainless steel cable and a stainless steel cap, is installed in the outlet nipple as a redundant mechanism to prevent the forward movement of the source assembly through the S-Tube toward the outlet end. See Drawing 15B002A.

F. A handle is provided as a convenience to carry the model SPEC-150 in the field when it is being used as an industrial radiography device. The original design of the SPEC-150 exposure device has four convenient mounting holes located at the bottom of each housing protective flange at the corners of the device which provide a sturdy means to attach security harnesses, pipeline trolleys, suspension lifts and permanent installation mounts. The currently approved design has an additional four holes located at the top corners of the housing's protective flange which can also be utilized for attaching a lifting or securing apparatus. The carrying handle and mounting holes are not structural parts of the package and serve no function during transport of the package, although the carrying handle and the eight holes have been tested and easily withstand more than ten times the weight of the package. The model SPEC-150 may also be lifted and secured from movement during transport without structural provisions for any lifting or tie down devices.

G. A titanium lock cap at one end protrudes approximately 7/8 inches beyond the flange of model SPEC-150. See Drawing 15B002A. It protects the source assembly connector from damage, which would only affect its operation as an industrial radiography exposure device, and it is not a structural part of the model SPEC-150 shipping package. On the other end the outlet nipple and source safety plug do not protrude beyond the flange of the model SPEC-150, and is not subject to damage during normal transport.

H. The model SPEC-150 is not hermetically sealed and is opened to ambient pressure, therefore a pressure relief system is not applicable.

I. The primary containment vessel to prevent the release of radioactive material is the sealed source capsule which meets the requirements of special form radioactive material in 10 CFR 71.75 as certified by the IAEA Competent Authority. Approximate dimensions of the capsule are one inch long by 1/4 inches diameter. Source assemblies ("pigtails") consist of the sealed source capsule swaged onto a flexible cable to which is swaged a locking ball and a drive cable connector. See Drawing 15B002A, Source Assembly.

J. Containment of the source assembly in the model SPEC-150 package is achieved by (1) the source assembly lock, which is located in the automatic securing mechanism/lock module assembly, prevents movement in both directions and is the primary mechanism to contain the source assembly in the model SPEC-150; (2) the diameter of the locking ball which can not pass through the smaller diameter orifice of the automatic securing mechanism module, (containment in this direction is maintained even when all locks are unlocked and the release plunger is depressed); (3) the automatic securing mechanism which engages the locking ball and provides a redundant mechanism to prevent forward movement through the S-Tube; (4) the lock cap which provides a redundant safety feature preventing the source assembly from coming out the lock end of the model SPEC-150; and (5) the safety plug which redundantly prevents the source assembly from passing through the outlet end. See Drawings 19B005, Lock Module and 19B006, Device Lock Operation (Locked).

The source assembly lock is a solid, stainless steel, irregularly shaped part with a curved slot that fits over the source assembly between the connector and locking ball. It prohibits movement of the source assembly in both directions. The source assembly lock is opened and closed by rotating the operating lever counterclockwise on the control assembly. The control assembly is a 25 foot long (minimum) mechanical piece of equipment that must be attached to the SPEC-150 to operate the source assembly lock (to perform radiography). The controls must be removed from the device to prepare the package for transport (to install the lock cap). The source assembly lock must be locked in order for the controls to be removed from the device. It is not possible to inadvertently leave the source assembly unlocked when preparing the package for transport.

The source assembly lock is held in the closed position by the device lock. The device lock is a solid, stainless steel, fan-blade shaped part that is operated by the device key. The lock is also held in the closed position by two spring loaded plungers located inside the lock module.

The device key must be inserted into the lock end plate with sufficient force to depress a large stainless spring, then rotated clockwise to unlock the source assembly lock. This action does not open the source assembly lock. The device lock must be locked in order for the key to be removed from the device, and the key must be removed in order to remove the controls. Like the source assembly lock, it is not possible to inadvertently leave the device unlocked when preparing the package for transport. See Drawings 19B005, 19B006, and 190909 which depict the SPEC-150 Lock Module and the Device and Source Lock Operation.

Failure of the locking system in an accident is virtually impossible unless the entire structure of the package is destroyed. For failure to occur the lock system must be subjected to (1) a compressive force applied to the spring-loaded device lock toward the outlet end simultaneously combined with, (2) a clockwise rotational force applied to the device lock, sequentially followed by (3) a counterclockwise rotational force applied to the spring-loaded source assembly lock, combined with (4) a temporary perpendicular, compressive force to depress the source release plunger, and then (5) a compressive force applied to the source assembly toward the outlet end with sufficient force to unlatch the spring-loaded release plunger. This compressive force cannot be applied to the source assembly at the same time that the compressive force (in the same direction) is applied to the device lock; otherwise, the source assembly connector will be forced against the slot in the source assembly lock and resist rotation. Since all of the above parts are inside the structure (with the exception of the release plunger) the forces must be generated by the momentum of the individual parts caused by the impact and spinning of the package in an accident. These forces must also be exerted without damaging the lock module. Inward crushing of the lock module housing would prohibit any reasonably foreseeable rotational forces from opening either lock. We have been unable to develop a hypothetical accident that is remotely capable of producing the combination and sequence of forces and events required to cause a failure of the lock system other than total destruction. As demonstrated by tests described in this application, total destruction requires forces that greatly exceed the performance requirements for a Type-B package.

K. Structural closures of openings are not employed to contain the radioactive material within the package.

L. There are no valves, sampling ports, coolants or mechanisms for heat transfer or dissipation.

1.3.3 Operational Features

A. The model SPEC-150 is a simple package and there are no operational considerations which are required for its use as a transport package.

B. Iridium-192, Selenium-75 or Ytterbium-169 is contained in a sealed source capsule which can not be operationally opened.

C. The source assembly is contained in the model SPEC-150 by (1) the source assembly lock, which is located in the automatic securing mechanism/lock module assembly, prevents movement in both directions and is the primary mechanism to contain the source assembly in the model SPEC-150; (2) the source assembly automatic securing mechanism (ASM) engaging the locking ball and preventing forward movement through the S-Tube; (3) the diameter of the locking ball which prevents movement of the source assembly out the lock end; (4) the redundant lock cap preventing loss of the pigtail out the lock end; and (5) the redundant source safety plug preventing loss of the pigtail through the outlet nipple end.

D. There are no valves, connections, piping, seals or similar containment mechanisms.

1.3.4 Contents of Packaging

A. The model SPEC-150 has been approved by the Louisiana Radiation Protection Division (agreement state radiation control agency) as a radiography exposure device with a maximum activity of 150 Ci of Iridium-192, Selenium-75 or Ytterbium-169 as a sealed source.

B. The sealed source capsule meets the requirements of special form radioactive material pursuant to 10 CFR 71.75 as demonstrated by an IAEA Certificate of Competent Authority.

C. Iridium-192, Selenium-75 or Ytterbium-169 is encapsulated in a cylindrical capsule measuring approximately 3/4 inches by 1/4 inches diameter which is swaged onto a flexible cable approximately 7-7/8 inches long forming a source assembly.

D. The density of Iridium-192, Selenium-75 and Ytterbium-169 is approximately 22.5, 4.79 or 6.96 grams per cubic centimeter respectively. The weight of the radioactive contents is negligible.

E. Iridium-192, Selenium-75 and Ytterbium-169 are not fissile materials, therefore moderator ratios and criticality configurations are not applicable.

F. The heat of decay for a maximum 150 Ci Iridium-192, Selenium-75 or Ytterbium-169 is infinitesimal and the void space in the sealed source capsule is negligible, therefore pressure buildup is not a factor.

2. STRUCTURAL EVALUATION

A structural evaluation of the model SPEC-150 was performed in conjunction with the application as an industrial radiography device in accordance with 10 CFR 34.20 and American National Standards Institute N432-1980. All of the information from the radiography device application that is relevant to a Type-B package is included in this application. Additional test and structural evaluation information has been added to this application. The NRC Office of Nuclear Materials Safety and Safeguards reviewed the model SPEC-150 industrial radiography device application and the Louisiana Radiation Protection Division reviewed and approved it.

All welds depicted on certificate drawings are performed in accordance with either the applicable requirements of ASME Section VIII, Division I, or in accordance with AWS D1.9 and SPEC's Quality Assurance Program, U. S. Nuclear Regulatory Commission Certificate of Compliance No. 0102. For SPEC-150's older than serial number 1475 not meeting these code requirements, all thermal metal joining (TMJ) of structural joints are performed in accordance with SPEC Titanium GTAW TMJ Procedure P51-1.

2.1 Structural Design

2.1.1 Discussion

There was no attempt nor necessity in the design of the model SPEC-150 to conduct theoretical engineering structural evaluations based on mechanical properties of materials. It is a small light weight package whose simple design was based on extensive years of previous experience with similar packages and methods of construction. The structural design was evaluated by actual physical tests in accordance with 10 CFR Part 71.

A. The principal structural components of the model SPEC-150 are (1) the depleted uranium shield which provides the necessary radiation shielding and protects the sealed source capsule; (2) the titanium inner bulkhead and outer end plate support cups which positions and affixes the depleted uranium shield; and (3) the titanium shell which firmly encases the depleted uranium shield and forms the outer package.

B. The sealed source capsule provides the primary containment vessel preventing the release of radioactive material and meets the requirements of 10 CFR 71.75 for special form radioactive material.

C. The source assembly, containing the sealed source capsule, is retained in the depleted uranium shield by (1) multiple locking and securing mechanisms, which require the sequential use of a key, application of external mating mechanisms, and two independent mechanical procedures to unlock the source assembly lock in the primary retention mechanism; (2) a locking ball on the source assembly which can not pass through the lock end of the model SPEC-150; (3) a lock cap providing secondary protection and redundant retention of the source assembly; and (4) a redundant safety plug preventing movement of the source assembly through the outlet nipple end of the model SPEC-150.

2.1.2 Design Criteria

A. The design of the model SPEC-150 was based on 10 CFR Parts 34 and 71; IAEA Safety Series No. 6, Regulation for the Safe Transport of Radioactive Material, 1985 Edition (As Amended 1990); and American National Standards Institute N432-1980. The SPEC-150 also meets the requirements of the 1996 Edition of TS-R-1.

B. Primary consideration was given to protecting the depleted uranium shield by limiting its movement under typical working conditions, normal transportation and hypothetical accident conditions. Lateral movement of the depleted uranium shield toward the ends of the device is limited by the inner bulkhead plate and outlet end plate. The titanium cups limits movement of the depleted uranium shield in other directions. The cups are filled with an epoxy potting compound to protect against ingress of moisture. By preventing movement of the depleted uranium shield within the housing the radiation levels after the hypothetical accident tests are within the established criteria. The principal area of concern is the nine meter drop test. The lock end of the shield is additionally secured against upward movement by direct contact with the top of the device housing. The outlet end of the shield is additionally secured against downward movement in an accident by direct contact with the bottom of the device housing.

C. Because the sealed source capsule qualifies as special form radioactive material, it is known that the sealed source capsule is not damaged by the thirty foot drop test nor the 1475° F thermal test. Located in the center of the depleted uranium shield within the model SPEC-150 case the sealed source capsule is adequately protected from any shear or crushing forces that could damage the capsule.

2.2 Weights and Centers of Gravity

The model SPEC-150 weighs a maximum of 53.5 pounds. The center of gravity is approximately the geometric center of the rectangular parallelepiped defined by the outlet end plate, the lock end plate and the outer dimensions of the case.

2.3 Mechanical Properties of Materials

2.3.1 Materials List

Structural materials used in the model SPEC-150 are principally titanium, stainless steel, depleted uranium. Epoxy potting compound, aluminum, bronze, rubber and foam are used in non-critical structural components.

2.3.2 All commercial grade materials are used in the construction of the model SPEC-150 and their mechanical properties are commonly established.

2.3.3 Titanium sheet and plate is used for the package shell, end plates, inner bulkhead, and the ASM/lock module. Titanium is used for the lock cap and support cups. Inside the automatic securing mechanism Series 300 stainless steel is used. See Bill of Materials (next page) for specific material description effective starting with SPEC-150 serial number 1475.

Bronze is used for some bushings. The radiation shield is a depleted uranium casting with a titanium or zircalloy tube through the shield. The depleted uranium shield has a maximum weight of 37 1/4 pounds. Optional tungsten, lead or depleted uranium shielding pads are used as needed.

Information plates are stainless steel to withstand the thermal test.

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

2.4 General Standards for All Packages

The model SPEC-150 meets the general standards for all packages in accordance with the provisions of 10 CFR Sections 71.43, 71.45 and 71.47.

2.4.1 Minimum Dimension

The smallest overall dimension of the package is nominally 5-3/8 inches plus or minus 1/8 inch, and therefore never smaller than 4 inches.

2.4.2 Tamper Seal

The sealed radioactive source may only be released from the package by unlocking the camera with a key pursuant to the requirements of 10 CFR 34.23. The camera can be unlocked only after sequential application of a mating mechanism and by two mutually independent operational mechanical procedures. Camera keys are not normally shipped in the same container as a model SPEC-150, but when a camera key is shipped in the same container with a model SPEC-150 it will be in a sealed envelope. When a model SPEC-150 is shipped in an overpack the overpack will employ a wire seal or tape that is destroyed upon removal for a security seal. The lock cap is designed for the installation of a wire tamper seal or a tape seal that is also destroyed upon removal meeting the requirements of 10 CFR 71.43(b).

2.4.3 Positive Closure

The primary containment system preventing the release of radioactive materials is the special form sealed source capsule which can only be opened destructively. In addition the sealed source assembly is retained in the depleted uranium shield by a key multiple securing mechanism, a redundant safety plug and a redundant lock cap. The camera can be unlocked only after sequential application of a mating mechanism and by two mutually independent operational mechanical procedures.

2.4.4 Chemical and Galvanic Reactions

The materials of construction are stable common metals which are known not to present chemical, galvanic or other reactions between the various metals. All the materials are inert to reaction with water, except for slow corrosion. A titanium-uranium or tungsten-uranium eutectic has not be shown to exist even at elevated temperatures (i.e. the titanium S-tube has been subjected to 1475° F). The depleted uranium shield is protected from corrosion by foam and epoxy moisture barriers.

2.4.5 Package Operational Containment

No valves or other devices are present which would allow radioactive contents to escape from the primary containment of the sealed source capsule. The source assembly is retained in the shield primarily by a key operated source assembly lock, additional securing mechanisms, a redundant safety plug, a restricting orifice through which the source assembly can not back out of the lock end, and a lock cap provide redundant positioning of the source assembly in the depleted uranium shield.

2.4.6 Normal Conditions of Transport

As described below in Section 2.8, Normal Conditions of Transport, the model SPEC-150 was subjected to the specified tests which demonstrated there would be no loss or dispersal of radioactive contents, no significant increase in external radiation levels, and no reduction in the effectiveness of the packaging. In fact the test specified for normal conditions of transport did not cause any significant effect on the model SPEC-150 package.

2.4.7 Surface Temperature

The maximum activity of 150 Ci in the model SPEC-150 has negligible heat of decay and the surface temperature of the package will be that of the ambient temperature.

2.4.8 Venting

Venting considerations are not applicable. Any pressure increase resulting from the decay of the maximum 150 Ci Iridium-192, Selenium-75 or Ytterbium-169 in the sealed source capsule will be negligible and will be adequately contained by the sealed source capsule.

2.4.9 Lifting Devices

A carrying handle is provided for use as a radiographic exposure device, and is not considered a structural part of the Type B package. A model SPEC-150 was suspended from a single point at the end of the carrying handle and loaded with approximately 500 pounds dead weight (53 pound package weight times a safety factor of at least ten) for a minimum of ten minutes. The handle supported the weight without any deformation or damage exceeding the minimum safety factor of three times the package weight in compliance with 10 CFR 71.45(a).

2.4.10 Tiedown Devices

Although the model SPEC-150 has eight tiedown holes located at the corners of the package their primary purpose is securing the camera during its use as an exposure device in the field and not for securing it during transport. The mounting holes are 3/8 inch diameter and are located approximately ½ inch from the end of each side panel and approximately 3/4 inch above the bottom and ½ inch below the top of the package. The mounting holes can withstand forces greatly in excess of ten times the mass of the package in compliance with 10 CFR 71.45(b)(1).

2.4.11 External Radiation Standards

External radiation levels for the model SPEC-150 package are shown to meet the requirements of 10 CFR 71.47. The model SPEC-150 containing no more than 150 Ci Iridium-192, Selenium-75 or Ytterbium-169 also does not exceed the U.S. Department of Transportation requirements specified in 49 CFR 173.441(a) of 200 mrem/hr at the surface of the package and 10 mrem/hr at one meter from the surface of the package. Instructions are provided in Section 7 Operating Procedures for preparing the package for shipment to meet the requirements for transport.

2.5 Standards for Type B Packaging

The model SPEC-150 meets the additional requirements for Type-B packages in accordance with the provisions of 10 CFR 71.51.

2.5.1 Normal Condition of Transport Test Criteria

The results of tests described below in Section 2.8 for normal conditions of transport adequately demonstrate that there would be no loss or dispersal of radioactive contents, no increase in external radiation levels, and no reduction in the effectiveness of the model SPEC-150 packaging.

2.5.2 Hypothetical Accident Conditions Test Criteria

The results of tests described below in Section 2.9 and in Appendix 9.6 for hypothetical accident conditions adequately demonstrate that there would be no possibility of 27 Ci Iridium-192 escaping from the package in one week nor would there be any radiation levels exceeding one rem per hour at one meter from the external surface of the package. In fact the source capsule containing the radioactive material remained intact and was not released from the package.

2.5.3 Activity Release Limitations

Containment by filter or mechanical cooling systems are not applicable, since there was no release of radioactive material. The source capsule remained intact after the tests for normal condition of transport and the hypothetical accident conditions.

2.6 Description of Test Packages

Model SPEC-150 prototype test packages, which were constructed in standard production fashion pursuant to applicable quality assurance procedures specified in NRC Certificate of Compliance No. 0102, were used for normal conditions of transport tests and hypothetical accident condition tests. For each of the hypothetical accident conditions tests a loaded prototype Iridium-192 source assembly, which was constructed in standard production fashion pursuant to applicable quality assurance programs, was contained within the test package to effectively measure the change in radiation levels after the tests.

2.6.1 Prototype No. 2 & 4

SPEC-150 prototypes No. 2 and 4 were used as test packages. The design of Prototype No. 2 is different from Prototype No. 4, and both meet Type-B requirements. As a result of the damage to No. 2 caused by the 30-foot drop tests the design of the SPEC-150 was revised. The revisions were incorporated into Prototype No. 4 and tested. The structural changes consist of the following items. (1) The design of the welded joint that attaches the control attachment boss to the lock end plate, Drawing 15B002A. The design was revised to increase the size and strength of the joint. (2) The design of the welded joints that attach the outlet end plate and the outlet end plate support cup to the bottom plate was revised to increase the size and strength of the joints, Drawing

15B002A, Item Nos. 1 and 7. With the exception of the above design changes, both test packages represent the basic design, and No. 4 completely represents the final design.

2.6.2 Design Changes of Production Packages

The only structural design change from Prototype No. 4 that will be incorporated into the final design of production packages is a reduction in height of the outlet end plate support cup by 5/16 inch to 2-13/32 inches. This will not reduce the strength of the cup nor the outlet end plate. With the exception of the optional use of a tungsten shielding pad described in Section 1.3.2(B), there are no other design features, details, sizes, dimensions, weights, weld materials, methods of fabrication of the test specimens that are different from production packages. The weight of Prototype No. 2 is 52 pounds. The weight of Prototype No. 4 is 52 pounds without the extra weight installed and is 53 pounds with the tungsten rod extra weight installed in the carrying handle (used for the 30-foot drop and puncture tests). The tungsten rod weighs 13 ounces.

2.6.3 Other Prototypes

Four prototype SPEC-150s have been fabricated. Prototypes No. 2 and 4 were used to perform tests to evaluate the package pursuant to 10 CFR Part 71 as a Type-B package. Prototype No. 1 was used to perform the portions of the ANSI N432-1980 tests pursuant to 10 CFR Part 34 as a radiography device. Prototype No. 3 has not been used for tests. Both #1 and #3 will continue to be used for additional tests that are not required pursuant to 10 CFR Parts 34 and 71 or other regulatory requirement.

2.6.4 A used SPEC-150, serial number 331 was used for the normal condition accident tests in 2011.

2.7 Drop Target Description

The drop target greatly exceeds the requirements outlined in IAEA Safety Series No. 37 "Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material (1985 Edition as amended 1990), which recommends a steel plate as the upper surface of a concrete block. It specifies that the combined mass of the steel and concrete should be at least 10 times that of the specimen to be dropped; that the block should be set on firm soil; that the steel plate should be at least 4.0 cm thick and floated onto the concrete while it is still wet; and that the plate should have protruding steel structures on it lower surface to ensure tight contact with the concrete.

The drop target at SPEC greatly exceeds the requirements specified in IAEA Safety Series No. 37 "Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material," (1985 Edition as amended 1990). The drop target consists of a solid carbon steel plate which measures 2'6" x 2'11" x 1-3/4" thick weighing 520 pounds. The thickness of the steel plate meets the minimum 4.0 cm IAEA requirement. The steel plate was wet floated onto the top surface of a flat horizontal concrete block which measures 4'6" x 4'6" x 4'6" thick weighing 13,668 pounds. The total weight of the drop target is 14,188 pounds which greatly exceeds ten times the mass of a 100 pound package. The concrete block is metal reinforced and is sunk to a depth of 4'2" into firm soil. A 40 foot tall structure was erected over the drop target and used to raise and release the test package from a minimum height of 30 feet (9.2 meters) above the top surface of the target. No damage nor separation of the steel plate from the concrete block occurred as a result of all tests.

2.8 Normal Conditions of Transport

2.8.1 Heat

The test at an ambient temperature of 70° C (158° F) in still air and isolation, was not performed because the materials and methods of construction would not be adversely affected in such a manner that there would be a loss or dispersal of the radioactive contents or a loss of shielding integrity that would result in more than a 20% increase in the radiation level at any external surface of the package.

2.8.2 Cold

The test at an ambient temperature of -40° C (-40° F) in still air and isolation, was not performed because the materials and methods of construction would not be adversely affected in such a manner that there would be a loss or dispersal of the radioactive contents or a loss of shielding integrity that would result in more than a 20% increase in the radiation level at any external surface of the package.

2.8.3 Reduced External Pressure

This test was not performed because the model SPEC-150 is open to the atmosphere and there are no materials in the package which would be affected by a pressure reduction to 3.5 psi absolute. The special form sealed source capsule will withstand reduced pressures much greater than the 3.5 psi absolute. A pressure reduction to 3.5 psi absolute would have no affect on the effectiveness of the model SPEC-150 package.

2.8.4 Increased External Pressure

The test was not performed because the model SPEC-150 is opened to the atmosphere and there are no materials in the package which would be affected by an increase of external pressure to 20 psi absolute. The special form sealed source capsule will withstand increased pressures much greater than the 20 psi absolute. An increase of external pressure to 20 psi absolute would have no affect on the effectiveness of the model SPEC-150 package.

2.8.5 Vibration

The effects of vibration on the package and materials of constructions incident to normal transportation is negligible. More than 1000 similar SPEC radiography exposures devices have been transported over a period of 20 years via all common modes of private and common transportation on water, highway and air without displaying damage or other noticeable effects. Vibration incident to normal transportation will not reduced the effectiveness of the packaging.

2.8.6 Water Spray

There are no materials of construction in the model SPEC-150 which would be affected by a water spray, and the production models do not easily permit water within the case. A water spray test was not conducted on the all metal package.

2.8.7 Free Drop

Both a prototype package, Prototype No. 4, and a used package, serial number 331 were dropped from a distance of 4 feet onto an essentially unyielding surface. (See target specifications above in Section 2.7, Drop Target Description.) Five cumulative drop tests were performed to obviate any question about selection of the most vulnerable point of impact. The points of impact were flat on the bottom plate, bottom right corner on the outlet end, bottom left corner on the lock end, top left corner on the outlet end and flat on the lock cap. There was no effect on the operation or the shielding capability of the package. The four foot free drop did not result in loss of radioactive contents from the package, increased radiation levels nor reduce the effectiveness of the package.

2.8.8 Corner Drop

This test is not applicable since the package is not constructed of wood or fiberboard.

2.8.9 Penetration

Both a prototype package and a used package were subjected to the impact of a 1-1/4 inches diameter steel cylinder weighing 13 lbs falling a distance of 40 inches. The point of impact was directly on the safety plug which is located at the outlet end of the device. See photograph 510A. The safety plug is the weakest structural point on the device that would also cause the most significant increase in radiation level if it were to break off. IAEA Safety Series No. 6 Regulations for the Safety Transport of Radioactive Material 1985 Edition (As Amended 1990) states in Paragraph 537(b) that the normal condition of transport test should not result in more than a 20% increase in surface radiation levels. The impact caused the outlet end panel to bend downward slightly. The safety plug and outlet nipple remained intact. See photographs 510B and 510C. There was no increase of radiation levels. The penetration test did not result in loss of radioactive contents from the package, increase radiation levels nor reduce the effectiveness of the model SPEC-150 package.

2.8.10 Compression

A model SPEC-150 prototype package was subjected to a compressive load of 266 lbs for a period of 24 hours. The test package was placed on a flat, horizontal surface. Cement blocks were loaded onto the top surface providing a total compressive force of 266 lbs. There were no observable effects of the compression test. The compression test did not result in loss of radioactive contents from the package, increased radiation levels, nor reduce the effectiveness of the model SPEC-150 package.

2.8.11 Test Summary

In compliance with 10 CFR Part 71.71, based upon the above tests and evaluations, it is determined that under normal conditions of transport:

A. There would be no loss or dispersal of radioactive contents.
B. There would be no significant increase in external radiation levels.
C. There would be no significant reduction in the effectiveness of the packaging.

2.9 Hypothetical Accident Conditions

2.9.1 Free Drops

A model SPEC-150 prototype package, Prototype No. 2, was subjected to four successive free drops and a second model SPEC-150 prototype package, Prototype No. 4, was subjected to four successive free drops from a distance of 9 meters (30 feet) onto the previously described drop target. Although not required under the test criteria, the multiple successive drops were made and the damage was cumulative to more than adequately demonstrate the durability of the package and to obviate any questions concerning selection of the most vulnerable points of impact and drop orientation. Additional free drop tests were performed in 1997 on SPEC-150 serial number 500. See Appendix 9.3 for complete information.

A. Prototype No. 2

1. Selection of Points of Impact

Sketches of the drop test impact orientations are located in Appendix 9. It is impossible to determine the most vulnerable point of impact by engineering evaluation alone. This fact was confirmed by the Engineering Department at Louisiana State University. Their opinion is that an engineering evaluation can easily identify the most likely points. However, there is no nationally recognized method to conclusively determine the most vulnerable point, particularly considering the resulting radiation levels. Therefore, the points of impact for Prototype No. 2 were based on an analysis of the design of the SPEC-150 and extensive past experience with testing numerous similar devices.

The selection of numerous points of impact is intended to demonstrate the exceptional structural integrity of the device and to obviate any concerns regarding the proper selection of points of impact and device orientation at the moment of impact. Finally, the points of impact and orientation for Prototype No. 2 was selected to provide damage information needed to make a final selection of point of impact on Prototype No. 4. The points of impact for Prototype No. 2 were selected to cause the maximum damage to the device and the maximum potential for shield movement within the device. The points selected were the right side, upper corner at the lock end, and two points on the outlet end of the exposure device.

2. 1st 9 Meter Drop Test

The point of impact for this test was the right side of the SPEC-150 test device. The intent was to inflict the maximum amount of force to shift the depleted uranium shield to the right side of the device and cause the source assembly to partially withdraw. The device was suspended from the left side to orient the center of gravity directly over the point of impact. See Photographs 53A, 53B and 53G. Upon impact the lock cap and the control attachment boss separated from the device. See Photograph 53C. The source connector was exposed, but not damaged. See photographs 53D and 53E. The radiation level at the lock end of the model SPEC-150 increased slightly, but not significantly, due to the secondary scattered radiation resulting from the loss of the lock cap. There was no other increase in radiation levels in any direction from the SPEC-150 package and there was no loss of radioactive content. All exterior TMJ joints remained intact and the source remained fully secured within the SPEC-150. See photograph 53D. The device remained in the locked position. Damage to the housing consisted of superficial scratches and denting to the right side of the device. See Photographs 53F and 53H. The imprint of the device on the drop target depicts the right side point of impact. See photograph 53I.

3. 2nd 9 Meter Drop Test

The point of impact for this test was located at the top right corner at the lock end of the SPEC-150 test package. The intent was to cause the maximum amount of damage to separate the ASM/Lock Module from the device thus removing the source assembly from the shield. The device was suspended from a single wire attached to the diagonally opposite corner at the outlet end of the package to position the center of gravity directly over the point of impact. See photographs 54A and 54B. The top right corner of the housing distorted inward and upward approximately one inch. One of the six ASM lid screws (on the right side closest to the lock end) sheared off. See photograph 54C and 54D. The source connector remained undamaged and in the locked position. See photograph 54E. A visual inspection of the device lock shaft verified that the device remained in the locked position. See photographs 54C and 54F. The TMJ welded seam that joins the top of the lock end plate to the housing split approximately three inches. See photograph 54F. No measurable displacement of the shield resulted. There was no increase in radiation level at any direction.

4. 3rd 9 Meter Drop Test

The point of impact of the third drop test was intended to be the right bottom corner at the outlet end. See photographs 55A and 55B. During the drop the device rotated sightly due to windy conditions. This caused the point of impact to be the right side edge (flange) at the outlet end which included the bottom right corner. See photograph 55D. The flange dented inward approximately ½ inch along the entire length of the right side. See photographs 55D and 55E. The depleted uranium shield shifted toward the outlet end of the device approximately 3/8 inch. The four screws holding the outlet end panel to the outlet end plate remained intact. See photograph 55F. The outlet end plate buckled outward and the end of the safety plug was dented causing the safety plug to be jammed in place. The device housing was bent upward slightly at the top right corner at the outlet end. See photographs 55D and 55G. The TMJ welded joint that joins the outlet end plate to the bottom plate was split along the entire length. See photograph 55F. The source remained in the locked position. The radiation level at the top center surface of the device after the 3rd drop increased from 24 mR/hr to 80 mR/hr.

5. 4th 9 Meter Drop Test

The point of impact for the forth drop test was the bottom left corner at the outlet end of the device. See photographs 56A and 56B. The lower left and upper right outlet end panel screws sheared off. See photographs 56C and 56D. The previously damaged TMJ weld seam that joins the outlet end plate to the bottom plate opened more, the end plate buckled outward more, and the depleted uranium shield shifted further toward the outlet end causing bottom of the outlet end plate to distort outward approximately an additional 7/8 inch. See photographs 56C, 56D and 56E. The bottom left corner at the outlet end dented inward approximately ½ inch. See photographs 56C and 56D. The TMJ welded joints that join the outlet end plate to the device housing split upward three inches on both left and right sides. See photographs 56C, 56D and 56E. The ASM/lock module shifted toward the outlet end of the device. The source remained locked and undamaged. See photographs 56F and 56G. The radiation level at the lock end of the device increased as a result of the dislocation of the depleted uranium shield away from the ASM/lock module. The source capsule was no longer in the fully shielded position within the S-tube. The radiation level at one meter from the lock end was 557 mR/ hr extrapolated to 150 curies. After four successive drop tests the SPEC-150 prototype No. 2 test device continued to meet the shielding requirements specified in 10 CFR 71.51 (a)(2).

6. Summary of Damage - Prototype No. 2

The significant structural damage consisted of (1) the splitting of the welded seam at the top of the lock end plate in the 2nd drop, (2) the splitting of the welded seam at the bottom of the outlet end plate and the shifting of the depleted uranium shield in the 3rd drop, and (3) the splitting of the welded seams along both sides of the outlet end plate and the additional outward distortion of the outlet end plate and shifting of the depleted uranium shield in the 4th drop. Additional significant damage consisted of the separation of the control attachment boss and lock cap in the first drop. The source assembly lock remained intact. With the exception of the loss of the lock cap, all other redundant source assembly securing mechanisms also remained intact.

7. Summary of Drop Tests - Prototype No. 2

The cumulative damage resulted in an increase in radiation level at the lock end of the SPEC-150 to 557 mR/hr at one meter extrapolated to 150 curies of Ir-192 which remains far below the maximum allowable limit of 10 mSv (1000 millirem) per hour at one meter from the surface. There was no loss of radioactive content.

Following all accidental drop tests the radiation levels were less than 1000 millirem per hour at one meter in all directions. The tests were video recorded and are available for review upon request.

The voluntary procedure to subject the test device to cumulative damage from four successive 9 meter drop tests more than adequately demonstrates that the SPEC-150 design greatly exceeds established standards for a Type B package.

B. Prototype No. 4

The following information describes the accidental drop tests using SPEC-150 Prototype No. 4. Prototype No. 2 successfully passed the accidental drop tests. Nevertheless, minor design revisions were made to Prototype No. 4 based on the damage evaluation of prototype No. 2. The control attachment boss was redesigned to prevent the lock cap from breaking off. Also, the TMJ welded joint design was revised to strengthen the outlet end plate and reduce the amount of structural damage caused by the momentum of the depleted uranium shield. It should be noted that the design revision objectives were successfully accomplished.

A tungsten weight was installed to the inside of the carrying handle to Prototype No. 4 to increase the weight of the test package. See Photographs 511A and 511B. This was done to represent a worst case test device. The design of the SPEC-150 allows for the use of a small tungsten shielding pad on an as-needed basis to correct minor imperfections inherent in the depleted uranium casting process.

The practice of adding weight to a device to create a worst case test was accepted by the NRC in the SPEC application for Type B approval of the SPEC C-1 source changer, Certificate of Compliance Number 9036. The addition of the weight does not add to the structural integrity of the device. Model G-60 source S/N AH2503 was installed in Prototype No. 4. On the date of the tests, August 25, 1994, the activity of the source was 23 curies (output).

1. Selection of Points of Impact

Sketches of the drop test impact orientations are located in Appendix 9. The selection of the most damaging point of impact on Prototype No. 4 for the 9 meter drop test was established by an evaluation of the damage to Prototype No. 2. The evaluation confirmed that the most damaging point of impact is the bottom corner at the outlet end of the device.

The upper right corner at the lock end caused the most exterior damage to the device housing on Prototype No. 2, but it did not damage the structural integrity nor reduce the shielding ability of the device. The right side point of impact caused virtually no damage to Prototype No. 2. The outlet end of SPEC-150 Prototype No. 2 is the point of impact that produced the greatest dislocation of the depleted uranium shield and hence the greatest increase in radiation level.

It is difficult to evaluate the damage caused by the two drops on the outlet end of Prototype No. 2 and determine conclusively if the greatest damage was caused by the drop on the bottom corner (4th drop) or the drop on the right edge (3rd drop). The outlet end housing flange at the corner acted as an impact limiter to a small degree. The housing absorbed some of the force of impact as the corner crushed inward. However, the corner point of impact produced the greater damage to the housing and weld joints in the immediate area compared to the drop on the right edge.

The objective was to select the point of impact that would most likely cause the depleted uranium shield to shift away from the lock mechanism (which holds the source assembly). Based on these observations we have concluded that the most damaging point of impact on the SPEC-150 is the bottom right corner at the outlet end of the device. This is the location that will produce the greatest chance for the depleted uranium shield to shift toward the outlet end and result in the highest radiation level.

In response to a request from the NRC dated February 24, 1995 to evaluate the package for a 30-foot drop test directly onto the lock, SPEC-150 Prototype No. 4 test package was subjected to two additional drop tests on February 25, 1995 directly onto the lock (and a puncture test onto the lock after the first additional 30-foot test).

2. 1st 9 Meter Drop Test - Prototype No. 4

The point of impact for this drop was the right bottom corner at the outlet end. See photographs 513A, 513B and 513C. To verify the drop height see photograph 513D.

The right bottom corner at the outlet end bent inward and upward 1 inch. See photographs 513E, 513F and 513G. The right side plate split 2 inches along the welded seam at the bottom plate. It should be noted that the weld joint did not separate, but the adjacent metal was torn. This damage was limited to the protective flange. There was no damage to any critical structural weld joints of the package. There were superficial scratches along the left side on the exposure device. See photograph 513H. The bottom left corner at the lock end was distorted slightly, apparently from secondary impact. See photograph 513I. The source and device remained in the locked position.

The radiation levels were less than 0.5 mR/hr at 1 meter in all directions. This extrapolates to 3.3 mR/hr at 150 curies.

3. 2nd 9 Meter Drop Test - Prototype No. 4

Based on a discussion with NRC and Louisiana Department of Radiation Protection personnel witnessing the tests performed on Prototype No. 4 a decision was made to conduct an additional 9 meter drop test after the puncture test. The point of impact was directly on the top of the exposure device based on the observation that this was the only surface of either prototype that had not been selected as a point of impact. The carrying handle was taped to the top left side of the exposure device to limit the device deflection as much as possible upon impact. See photographs 515A, 515B, 515I and 515J. See photograph 515C for drop height verification.

The top of the release plunger was the initial point of impact. See photograph 515D. There was a 1/4" deep imprint of the handle in the top of the device housing closest towards the outlet end. Also, the impact caused an imprint of the knurled carrying handle grip into the top of the device housing. See photographs 515E, 515F and 515M. The ASM Lid Plate was bent upward 1/4" between the release plunger and carrying handle. See photographs 515F and 515L. The ASM Lid Plate was dented inward at the location where the plunger was impacted. See photograph 515G. The lock end plate was bent outward 1/8" at the Control Attachment Boss. See photographs 515G, 515H and 515K. At the conclusion of the 2nd 9 meter drop test the device was surveyed by a member of the Louisiana Division of Radiation Protection staff. See photograph 515N. A complete radiation survey was performed by SPEC personnel. The radiation levels at the surface of the device, extrapolated to 150 curies, after the 2nd 9 meter accidental drop test were:

Top	163 mR/hr
Bottom	147 mR/hr
Left Side	163 mR/hr
Right Side	229 mR/hr

Outlet End	114 mR/hr
Lock End	212 mR/hr

Radiation level readings at one meter were less than 3.3 mR/hr from all six surfaces.

4. 3rd 9 Meter Drop Test - Prototype No. 4

See photographs G-01 through G-04 taken before the test. The circles with a dot in the center indicate the locations of the highest surface radiation levels. These are the same locations that existed at the conclusion of the 2nd 30-foot drop test and puncture test in August 1994. The test package contains the same source used for the August 1994 tests, source S/N AH2503, which had an activity of approximately 4.1 curies on the date of this test, February 27, 1995. Photographs G-05 through G-08 show the device before the test with the lock cap both installed and removed. Photograph G-05 shows the damage to the lock cap that was caused by one of five four-foot drop tests performed on Prototype No.4 on December 17, 1994. A close inspection of Photograph G-06 shows the source assembly lock fully engaged over the source assembly cable at the conclusion of all previous tests.

The test package was suspended from two points at the outlet end with the center of gravity positioned directly over the lock cap. See Photographs G-09 through G-11. Photograph G-12 shows the package positioned 30 feet above the target. Photographs H-01 through H-04 show the damage from the 1st 30-foot drop test on the lock cap. With the possible exception of the source assembly release plunger (which was the point of impact of the previous 30-foot drop test in August 1994) the lock cap is the point of impact most likely to damage the lock. The lock is located inside the lock module which is located behind the lock end plate of the package. There is no direct access to impact the lock itself. The lock cap is located slightly off of the geometrical center of the lock end of the package and was the initial point of impact. The test dented the end of the lock cap in approximately 1/4 inch. See Photograph H-02. The impact damaged the lock cap threads and prevented it from being able to be removed from the package. The end of the bottom plate was the secondary point of impact. Photograph H-03 shows the marks of the lock cap and bottom edge on the target outlined in red colored boarders. Besides superficial scratches there was no measurable additional damage to the bottom edge. See Photograph H-04.

There was no increase nor change in radiation level at any surface location. The activity of the Ir-192 source was too low to provide relevant radiation levels at one meter.

5. 4th 9 Meter Drop Test - Prototype No. 4

After conducting a puncture test onto the lock cap, Prototype No. 4 was suspended from two points at the outlet end with the center of gravity positioned directly over the lock cap again. See Photographs J-01 through J-03. Note that the target was painted yellow to help analyze the impact pattern on the target. Photograph J-04 shows the package positioned 30 feet above the target. Photographs K-01 through K-04 show the damage from the 2nd 30-foot drop test on the lock cap (4th 30-foot drop test with Prototype No. 4). The test dented the end of the right side of the lock cap in approximately 3/8 inch. See Photograph K-04. The upper right corner and right edge was the secondary point of impact. Photograph K-01 shows the marks of the lock cap and right edge on the yellow target. The corner dented inward approximately ½ inch. See Photograph K-02 and K-03.There was no separation of the TMJ welded joints that connect the end plate to the housing shell and bottom plate.

An attempt was made to remove the lock cap with the use of vice grips. The lock cap threads were too damaged and the effort to unscrew the lock cap resulted in scraping off the knurled surface as shown in Photograph K-04. The lock cap was disassembled and removed from the package. Photographs K-05 through H-08 show the cumulative damage after two 30-foot drop tests and the puncture test onto the lock after the lock cap was removed. The lock end plate was dented inward slightly at the left side of the lock cap. The source assembly was completely undamaged and remained fully secured and locked in the shielded position. A close inspection of Photograph K-05 shows that the source assembly lock remained fully engaged over the shank that connects the source assembly connector and locking ball. There was absolutely no increase nor change in radiation level at any surface location of the test package. The activity of the Ir-192 source was too low to provide relevant radiation levels at one meter.

6. Summary of Damage - Prototype No. 4

The damage includes the buckling and tearing of the protective flange at the bottom right corner after the first 9 meter drop. Other damage includes the imprint of the handle in the top of the exposure device; the knurled imprint of the handle grip into the top of the device housing; the slight buckling of the ASM lid plate; and the center of the lock end plate bending outward approximately 3/32 inch after the 2nd 9 meter drop test. The flange at the lock end of the device and the lock cap were dented in from the 3rd and 4th 30-foot drop tests. The puncture tests and the penetration test caused no significant damage to the device.

C. Accidental Drop Tests Summary

The radiation levels in all directions were less than 1% of the allowable limit of 1000 millirem per hour at one meter when extrapolated to 150 Ci Iridium-192. It should be noted that not a single critical structural TMJ weld joint separated or fractured. All components remained intact and attached to the device. After all tests a physical and visual inspection confirmed that the source remained in the secured, locked and fully shielded position, and the device remained locked with the safety plug and lock cap intact. The lock system was damaged only to the extent that the device lock could not be opened. The system remained fully intact and maintained complete security of the source assembly in the proper shielded position. The source assembly lock was undamaged and fully engaged over the source assembly which prohibited it from moving forward or backward. The device lock was undamaged and fully engaged with the source assembly lock to prevent it from being opened. The face plate of the lock housing was apparently dented inward which jammed the lock and prevented it from being opened with a key. The automatic securing mechanism (ASM) was undamaged and maintained a redundant means to prohibit forward movement of the locking ball toward the outlet end. The ASM housing was undamaged and maintained a redundant means to prohibit the locking ball from passing through the smaller diameter opening toward the lock end of the device. The lock cap and safety plug remained intact, installed and fully functional as redundant safety features to prevent loss of the source in either direction. There was no failure of any portion of the structural TMJ welded joints connecting the titanium plates except at the flanges. All housing plates (top, bottom, sides and both ends) remained fully intact and did not buckle. The depleted uranium shield and the source assembly remained completely intact.

The accidental drop tests test results of the model SPEC-150 Prototype No. 4 more than adequately demonstrates that the design exceeds established standards for a Type-B package and that it is extremely durable, safe and structurally sound.

2.9.2 Puncture

Two model SPEC-150 prototype packages were dropped from a distance of 1 meter (40 inches) onto the center of a six inch diameter by eight inch high carbon steel cylindrical bar. The bar was located in the center of the drop test target specified in Section 2.7, Drop Target Description. The sealed source which was installed within the package for the series of drop tests was used also for the series of puncture tests. Additional puncture tests were performed in 1997 on SPEC-150 serial number 500. See Appendix 9.3 for complete information.

A. Prototype No. 2

The point of impact was the center of the right side. This point was selected because the previous 9 meter drops caused the depleted uranium shield to shift toward the outlet end of the device. This point of impact would cause the greatest force to displace the depleted uranium shield further. See photographs 57A and 57B. The impact did not produce any damage, only minor superficial marks were caused. See photograph 57C. Radiation levels did not change.

B. Prototype No. 4

The same model SPEC-150 Prototype No. 4 was used for the test after the first 9 meter drop test. The impact point was selected so as to cause the maximum damage to the device and the maximum potential for shielding movement within the device. The point of impact for this drop was on the center of the right side. See photographs 514A, 514B, 514C and 514F. The impact had impressed the target onto the right side of the device. See photographs 514D and 514E. There was no indication of the depleted uranium shield shifting within the housing. The source assembly and device remained in the locked position. There was no increase in radiation levels.

In response to a request from the NRC dated February 24, 1995 a SPEC-150 test package was subjected to a one meter puncture test directly onto the lock. Prototype #4 was subjected to the puncture test after the first of two 30-foot drop tests directly onto the lock that was performed on February 25, 1995. The test package contained the same source used for the 30-foot drop test. The test package was suspended 40 inches above the target with the center of gravity positioned directly over the lock cap. See Photograph I-01 and I-02. The puncture test caused a slight additional superficial scratches at the point of impact on the end of the lock cap. See Photographs I-03 and I-04. There was absolutely no increase nor change in radiation level at any surface location.

C. Puncture Test Summary

The three separate puncture tests did not produced any significant damage. The maximum radiation level at one meter measured after the cumulative drop tests and extrapolated to 150 curies did not exceed the maximum 1000 mrem/hr at one meter criteria for the hypothetical accident tests. The model SPEC-150 meets the criteria established for the hypothetical accident puncture test. Additional Puncture Test information is documented in Appendix 9.6, "1997 Puncture Tests".

2.9.3 Thermal

Based on the thermal properties of the structural materials and previous thermal tests and experience with packages constructed of the same or similar materials, including materials which have more adverse thermal properties it is concluded that a temperature of 1475° F for thirty minutes would have no structural effect on the package. The primary containment of the radioactive material is the special form capsule which has been demonstrated by tests to retain its contents at 1475° F for ten minutes and due to the thermal properties of iridium and the capsule direct exposure to 1475° F for thirty-minutes would have no effect.

Assuming that the foam and the potting compound are completely volatilized the resulting gases will escape the package since it is not hermetically sealed. Loss of the foam and potting compound will not reduce the shield effectiveness of the package, nor lead to structural changes which would cause the loss of any radioactive material from the package.

The depleted uranium shield is contained within a titanium shell. Titanium has a lower thermal coefficient of linear expansion than stainless steel which has been previously employed on most other industrial radiography exposure device packages, therefore thermal expansion of the titanium shell will be less than the stainless steel shells which did not adversely affect the depleted uranium shield. Thus, the shielding and structural integrity of the package will be retained.

Exposure of the package to a temperature of 1475° F for a period of thirty minutes will not result in any release of radioactive material nor reduce the shielding effectiveness of the package.

2.9.4 Water Immersion

The water immersion test was not performed since no fissionable materials are involved in the package, and since there are no materials of construction which would be damaged by water and water pressure equivalent to a 50 foot depth for a period of eight hours.

2.9.5 Summary of Structural Damage

The only structural damage to Prototype No. 4 after the accident condition tests was limited to deformation and tearing of the flanges at both ends of the package and slight denting to the surface of the housing. The lock system was damaged only to the extent that the device lock could not be opened. In summary; the structural integrity, shield, and all features designed to maintain the radioactive source in the shielded position under hypothetical accident conditions remained intact and performed fully as intended.

2.10 Special Form

Radioactive material is encapsulated in a capsule which meets the requirements of special form radioactive material pursuant to 49 CFR 173.403(z), 10 CFR 71.77 and Paras 239, 602-604 IAEA Safety Series TS-R-1 "Regulations for the Safety Transport of Radioactive Material", 1996 Edition (Revised). The capsule represents the primary containment vessel for the radioactive material. The capsule meets the requirements of special form radioactive material as demonstrated by an IAEA Certificate of Competent Authority.

2.10.1 Description

The sealed source capsule in the model SPEC-150 package is approximately 3/4 inches long by 1/4 inches diameter. The sealed source capsule meets the minimum dimension requirement of 5 mm for special form radioactive material in compliance with IAEA Safety Series TS-R-1, para 602. Source assemblies ("pigtails") consist of the sealed source capsule swaged onto a flexible cable to which is swaged a locking ball and a connector.

2.10.2 Free Drop

Since the capsule is very light and ruggedly constructed it is apparent that effects of its impact onto a flat, horizontal, essentially surface would be negligible.

2.10.3 Percussion

The design and yield strength will permit the capsule to withstand impacts much greater than that which would be incurred from the specified three pound steel billet falling from a height of one meter onto the capsule while it rests on a lead sheet, maximum 25 mm thick, which is supported on a flat, smooth, essentially unyielding surface.

2.10.4 Bending

This test is not applicable since the sealed source capsule is less than 10 cm long.

2.10.5 Heating

The capsule and the radioactive material will withstand sustain temperatures greater than 1475° F for ten minutes without adverse effects.

2.10.6 Summary

As a result of previously performed evaluations resulting in the issuance of IAEA Certificate of Competent Authority and on the basis of the above summary assessment the primary containment vessel in the model SPEC-150 package, the sealed source capsule, meets or exceeds the requirements for special form radioactive material as specified in 10 CFR 71.75.

3. THERMAL EVALUATION

Due to the materials of construction of the model SPEC-150 which are known to have stable thermal properties and which will not be affected by the prescribed 1475° F heat test it was not necessary to incorporate any special thermal engineering features in the package for it to comply with the normal conditions of transport and the hypothetical accident conditions.

3.1 Discussion

The heat of decay from the maximum activity 150 Ci of the approved radioactive contents is negligible. There are no fluids in the model SPEC-150 package, it is not hermetically sealed, it is vented to the atmosphere, and there can be no pressure build up in the package. The effects of the free drop and percussion tests do not affect the thermal characteristics of the package since the individual materials of construction are not affected by a temperature of 1475° F. Aluminum, buna rubber, foam and epoxy potting compound are the only materials which will be affected by the 1475° F test temperature, but they are not critical to the safety of the packaging. Bronze had the next lowest melting point which is not lower than 1300° F. The bronze bushing melting would only prevent unlocking the device. The hypothetical accident temperature of 1475° F could only affect the temper of the springs in the automatic securing mechanism, but it would remain in the locked position. A temperature of -40° F would have no effect on the critical materials of construction since there are no moving operational parts of the package.

3.2 Summary of Thermal Properties of Materials

References: ASM International, Guide to Materials Engineering Data and Information, 1986.
Private Communication - Nuclear Metals, Incorporated.
Private Communication - Mitech Metals, Inc.

The materials of construction are as follows:

Structural Materials	Melting Temperature
Depleted Uranium	2070° F
Stainless Steel; 304, 316, 440C	2550° F
Titanium Grade 2	3000° F
Tungsten (alloy)	3000° F
Zircalloy 2	3270° F

Non-Structural Materials Assumed to melt or volatilized below 1475° F:

Aluminum
Bronze, Imperial
Epoxy
Polyurethane Foam
Rubber 70 Buna
Enamel Paint

From the above table it is readily apparent that a 1,475° F temperature would have no effect on the device.

There have been reports indicating a possibility of a iron-uranium eutectic formation at 1,340° F. Such eutectic formation has been associated with metallurgically clean surfaces and vacuum heat treatment. The depleted uranium casting in the model SPEC-150 is coated with enamel paint at the factory. Titanium, tungsten, foam and an epoxy potting compound would come in contact with the enamel paint on the shield exterior, but would not come in direct contact with the deplete uranium. A titanium-uranium or tungsten-uranium eutectic has not be shown to exist. Depleted uranium castings have employed titanium S-tubes for years without any indication of a titanium-uranium eutectic .

3.3 Technical Specification of Components

This section is not applicable. The only operating component in the model SPEC-150 package is the source assembly lock which is a one piece component made of stainless steel which is not affected by a 1475° F temperature. The model SPEC-150 is locked when the package is prepared for transport. There are no operating components during transport.

3.4 Thermal Evaluation for Normal Conditions of Transport

The radiation level shielding and containment of the source assembly within the model SPEC-150 is totally dependent on materials which are not adversely affected by temperatures in the range of -40° C (-40° F) to 70° C (158° F). Therefore, the model SPEC-150 package will not release it contents, will not present increased radiation levels, and will not incur any reduction in the effectiveness of the package.

3.5 Hypothetical Accident Thermal Evaluation

The radiation level shielding and containment of the source assembly within the model SPEC-150 is totally dependent on materials which are not adversely affected by a temperature of 1475° F. Therefore, it can be concluded that such a temperature will have no effect on the shielding effectiveness and the containment of the source assembly in the package. Thermal tests have been previously performed on similarly constructed radiography exposure devices and have demonstrated that the maximum accident thermal condition does not affect the radiation shield nor the containment of the source assembly.

The model SPEC-150 source assembly will not release its radioactive contents as a result of the hypothetical accident thermal condition due to the fact that the primary containment is a special form sealed source capsule which has demonstrated to withstand such temperatures and which is manufactured from materials that are not affected by a temperature of 1475° F.

4. CONTAINMENT

4.1 Containment Boundary

4.1.1 Containment Vessel

The sealed source capsule containing the radioactive material described in Section 2.10 represents the primary containment boundary and vessel. This capsule meets the requirements of 10 CFR 71.75 and 49 CFR 173.469 for special form radioactive material.

4.1.2 Containment Penetrations

Due to the size of the sealed source capsule and the location of the capsule within the model SPEC-150 there will be no penetrations of the primary containment vessel.

4.1.3 Seals and Welds

The sealed source capsule is fused in a thermal metal joining procedure to meet the requirements of special form radioactive material and there are no mechanical or chemical seals pertaining to the primary containment capsule.

4.1.4 Closure

The special form, sealed source capsule may only be opened destructively and there are no mechanical closure provisions.

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 was 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.

All shielding tests were performed with survey meters that measure in mR/hr. Therefore, survey results will be converted from mR/hr by multiplying mR/hr by a factor of 0.87 when needed for comparison to regulatory requirements. This factor is applicable per the Health Physics Society. See http://www.hps.org/publicinformation/ate/q1055.html.

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 that the source capsule is the in fully shielded position in accordance with 10 CFR 34.20(a) and American National Standards Institute N 432-1980 Section 5.1.2.4 which states "It shall not be possible to operate the lock unless the source assembly is in the fully shielded position." The automatic securing mechanism, device lock, lock cap and safety plug provide redundant safety systems for securing the source assembly in the shield in the proper position. The curvature of the S-Tube and the elongated shape of the depleted uranium shield prevent primary radiation and provides secondary shielding.

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 Ir-192. Ir-192 will produce a higher external dose rate than Se-75 or Yb-169. See analysis in Appendix 9.7 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	0.6 (62)	1.0 (103)	0.8 (81)	0.006 (0.6)	0.010 (1.0)	0.008 (0.8)
Neutron	NA	NA	NA	NA	NA	NA
Total	0.6 (62)	1.0 (103)	0.8 (81)	0.006 (0.6)	0.010 (1.0)	0.008 (0.8)
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				0.050 (5.0)	0.050 (5.0)	0.045 (4.5)
Neutron				NA	NA	NA
Total				0.050 (5.0)	0.050 (5.0)	0.045 (4.5)
10 CFR 71.51(a)(2) or 656(b)(ii)(i) of TS-R-1 Limit				10 (1000)	10 (1000)	10 (1000)

5.2 Normal Conditions of Transport

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 on the SPEC-150 as prepared for transport with a 131.29 curie Ir-192 source on February 3, 2011 at the conclusion of normal conditions testing. (See Appendix 9.3) The readings were extrapolated to 150 curies, and the readings at the surface were also corrected to allow for the ½" 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).

Table 2: Radiation Levels as Prepared for Transport After Normal Conditions Test						
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	0.6 (62)	1.0 (103)	0.8 (81)	0.006 (0.6)	0.010 (1.0)	0.008 (0.8)
Neutron	NA	NA	NA	NA	NA	NA
Total	0.6 (62)	1.0 (103)	0.8 (81)	0.006 (0.6)	0.010 (1.0)	0.008 (0.8)
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)

Normal conditions testing was performed on 02/03/2011 with a 131.29 curie source to demonstrate that the SPEC-150 meets 10 CFR 71.43(f) and 10 CFR 71.51(a). (This test was performed again as the original testing was performed with a low curie source.) 10 CFR 71.43(f) and 10 CFR 71.51(a) 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 3% 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 Table 3 (all dose rates are expressed in mR/h).

Table 3: Actual Surface Radiation Levels Before and After 02/03/11 Normal Conditions Testing				
Surface	Pre-Drop	Post-Drop	Change	% of Change
Left	86	84	-2.0	-2%
Right	58	60	+2.0	+3%
Bottom	68	70	+2.0	+3%
Top	52	50	-2.0	-4%
Lock	82	72	-10.0	-12%
Outlet	58	58	0	0

5.3 Hypothetical Accident Conditions

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 in 1994, 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 mR/h).

Table 4: Radiation Levels After Initial Hypothetical Accident Conditions Testing

	Actual Readings:			Extrapolated and Corrected:		
Surface:	mR/h	mSv/h	mrem/h	mR/h	mSv/h	mrem/h
Top	20	0.17	17	164	1.43	143
Right	28	.024	24	229	1.99	199
Lock	26	0.23	23	213	1.85	185
Outlet	14	0.12	12	115	1.00	100
Bottom	18	0.16	16	147	1.28	128
Left	20	0.17	17	164	1.43	143

Additional 9 meter drop tests and puncture tests were performed in 1997 (see Appendix 9.6). 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: mR/h		Extrapolated and Corrected: mSv/h (mrem/h)					
	Surface	1m	Surface			1 Meter		
	mR/h	mR/h	mR/h	mSv/h	mrem/h	mR/h	mSv/h	mrem/h
Top	14	1.0	97	0.843	84	5.8	0.050	5.0
Right	14	0.9	97	0.843	84	5.2	0.045	4.5
Lock	11	1.0	76	0.663	66	5.8	0.050	5.0
Outlet	10	0.9	69	0.602	60	5.2	0.045	4.5
Bottom	11	1.0	76	0.663	66	5.2	0.045	4.5
Left	20	0.9	138	1.205	120	5.8	0.050	5.0

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 loaded with Ir-192. Ir-192 will produce a higher external dose rate than Se-75 or Yb-169. See analysis in Appendix 9.7 for the calculated dose rates for Se-75 and Yb-169.

5.5 Model Specification

Physical radiation measurements were performed on prototype packages, SPEC-150 serial number 500, and a used SPEC-150 serial number 331.

5.6 Shielding Evaluation

Additional normal conditions tests were conducted in 2011 to conclusively demonstrate the SPEC-150 was 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. 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. Additional hypothetical accident conditions tests conducted in 1997 demonstrated that the SPEC-150 is 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.

6. CRITICALITY EVALUATION

This section is not applicable since the model SPEC-150 does not contain and is not designed to transport fissile material.

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. 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. Visually check the exposed fasteners and welds.

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.

Regulations require that the non-fixed (removable) contamination on the external surfaces of the outer package being shipped on a non-exclusive use basis not exceed 10^{-5} uCi/cm^2 (0.0001 uCi/cm^2) averaged over 300 cm^2 of any part of the surface, as required in 10 CFR Part 71.87(i). This may be determined by measuring the activity on wipes taken from representative locations and the above criteria is assumed to be met if the activity on any sample averaged over the surface area wiped does not exceed 10^{-5} uCi/cm^2 (0.4 Bq/cm^2 or 22 dpm/cm^2). If the contamination on the surface of the outer package exceeds the above amount it will not be shipped.

7.1.4 Transportation Requirements

The model SPEC-150 package will be properly marked, labeled and described on a shipping paper in accordance with U.S. Department of Transportation regulations. Placards will be offered to carriers transporting a Radioactive Yellow III labeled package. Shipping papers will be retained for one year in accordance with U.S. Department of Transportation regulation Section 5110 of the Federal Hazardous Materials Transportation Law, published October 1994.

7.1.5 Type B Quantity Consignee Notification

Prior to each shipment of a model SPEC-150 containing more than 27 Ci Iridium-192 the shipper shall notify the consignee of the dates of shipment and expected arrival.

7.2 Procedures for Receipt and Unloading the Package

7.2.1 Unloading

The consignee must establish written procedures for receiving the model SPEC-150 package in accordance with applicable NRC and agreement state regulations. Such procedures should provide for inspection, monitoring, notification and records. The model SPEC-150 package becomes an industrial radiography exposure after receipt by the licensed industrial radiographer user.

The source assembly is temporarily removed and then returned to the exposure device frequently throughout its use in accordance with the licensed user's procedures and in accordance with applicable NRC or agreement state regulations.

7.2.2 Receiving the model SPEC-150

A. Delivery, Pick Up and Acceptance from Carrier

Regulations require that the consignee must make arrangements to receive the model SPEC-150 when it is offered for delivery by the carrier; or must make arrangements to receive notification from the carrier at the time of arrival for pick up at the carriers facility.

The consignee must expeditiously pick up the model SPEC-150 upon receipt of notification from the carrier.

B. Receipt Survey and Inspection

Before the delivered package is opened and as soon as practicable after receiving the model SPEC-150, but no later than three hours after it is received at the consignee's facility during normal working hours or within three hours beginning the next work day if received after normal working hours the package must be monitored and inspected.

The outside package, as received, should be inspected for any indication of damage to the model SPEC-150, and the maximum external radiation levels at the surface of the outside package and at one meter from the surface of the outside package must be measured and recorded. Dents and abrasions to the overpack normally encountered in handling, loading and unloading are not generally considered evidence of damage to the model SPEC-150.

Since the sealed source in the model SPEC-150 is classified as special form radioactive material it is not required to monitor the external surfaces of the outside package for removable contamination.

C. Notification

If the measured maximum radiation levels at the surface of the outside package and at one meter from the surface of the outside package exceed either of the following limits:

Location	Maximum mrem/hr
Surface of Outside Package	200
One Meter from Surface of Outside Package	10

Then the consignee must immediately notify the final delivering carrier, and either the agreement state radiation control agency, if applicable, or the NRC regional office having jurisdiction over the location where the package was received. It is also recommended that the shipper be notified. Care should be exercised in performing the survey that the radiation levels are measured at the proper distances, that the survey meter is calibrated and operating properly, and that the stated accuracy of the survey meter be considered.

D. Records

Records of the receiving survey should be maintain for a period of three years which include at least: date and time package received or picked up; date and time monitored; identification of package by serial number; identification of source by serial number, isotope and activity (includes date of measurement); identification of individual performing survey; identification of survey meter by serial number; maximum radiation levels at surface of outside package and at one meter from surface of outside package; and corrective action and notification to carrier and regulatory agency, if applicable.

7.3 Preparation of an Empty Package for Transport

Test to verify that the SPEC-150 does not contain a radioactive source (authorized source, unauthorized source, modified source, or a source capsule that has been removed from the source assembly) by the following method. This test should be performed by authorized and monitored personnel who have been trained in radiation safety and equipped with a properly operating survey instrument.

First; remove the safety plug and survey the open outlet nipple. The depleted uranium shield is radioactive and will emit radiation even when no sealed source is installed in the package, but the highest radiation level should not exceed approximately 2 mR/hr. Second; remove the lock cap and visually inspect the device to verify that no source assembly connector is protruding. Third; attach the control assembly to the device and crank the drive cable forward two complete revolutions while monitoring the survey instrument for radiation hazards. An exposed source must be treated as an emergency. Fourth; crank the drive cable back, disconnect and remove the control assembly from the device, and install the safety plug and lock cap. As an option, before cranking the drive cable back, a dummy connector or a dummy source assembly may be attached to the drive cable and retracted into the device. If a dummy connector is used it will pull out of the device with the drive cable when the controls are removed. If a dummy source assembly is used it will remain in the device and must be disconnected from the control drive cable to remove the controls. Inspect the connector of the dummy source assembly to verify that it has no serial number.

The empty packaging contains a maximum of 37 1/4 pounds of depleted uranium and may be shipped as either labeled radioactive material package or as an excepted package, article manufactured from depleted uranium as required by applicable U.S. Department of Transportation regulations.

8. ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

8.1 Acceptance Tests (Prior to First Use)

The acceptance tests prior to first use is a combination of the in progress and final 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 same code as that used for welding, either the ASME Code for Boilers and Pressure Vessels, Section VIII, Division 1 or AWS D1.9. In either case, either the 2007 edition or an updated revision will be used.

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 the package.

8.1.3 Leak Tests

Leak tests are performed in accordance with approved procedures pursuant to the Source Production & Equipment Company, Inc. Louisiana Radioactive Material License LA-2966-L01 on the source assembly after fabrication of the source capsule, and a source assembly will be rejected if there is removable contamination in excess of 0.002 microcuries. Prior to shipment the outer surfaces of the package are monitored for removable contamination and a package will not be shipped if it exhibits more than 220 dpm/cm^2 removable contamination averaged over 300 square centimeters.

8.1.4 Component Tests

As part of the final manufacturing inspection the operation of the source assembly lock, device lock and automatic securing mechanism are tested for proper operation. The lock cap and source safety plug are tested for proper closure.

Prior to shipment with a source assembly the package is inspected to assure that the source assembly, lock cap and source safety plug are properly secured.

8.1.5 Tests for Shielding Integrity

Every shield is surveyed prior to first use to determine that it will meet NRC shielding requirements when extrapolated to 150 Ci. The survey is performed on the surface of the SPEC-150 and at one meter. Prior to each shipment of the camera with a source assembly the package is surveyed to assure compliance with transportation requirements.

8.1.6 Thermal Acceptance Tests

Thermal acceptance tests for the model SPEC-150 are not indicated since heat of decay for the maximum permissible activity Iridium-192, Selenium-75 or Ytterbium-169 source (150 Ci) is negligible.

8.2 Maintenance Program

8.2.1 Structural and Pressure Tests

Periodic 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. However, a visual check of all external fasteners and welds should be performed. Periodic 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 the package.

Quarterly inspection of the package by licensed radiography users as required by 10 CFR 34.31 is sufficient. The quarterly inspection requirements that are relevant to assure that the SPEC-150 operates properly as a Type-B package consist of a visual inspection and operational tests of the lock cap, device lock, source assembly lock, safety plug and outlet nipple. There are no quarterly maintenance requirements such as disassembly, cleaning, replacement of components, or lubrication. The inspection and maintenance procedures are described in the SPEC-150 Users Manual and should be included in the licensed radiography users' Operating Procedures required by 10 CFR Part 34.45.

8.2.2 Leak Tests

Leak test for removable contamination are required to be performed at least every six months on the sealed source pursuant to 10 CFR 34.27 or equivalent agreement state regulations. A leak test should also be performed whenever there is indication of damage to the sealed source capsule. If the tests indicate 0.005 microcurie or more of removable contamination the sealed source must be removed from use, action taken to prevent the spread of contamination, and a report filed with the applicable radiation control agency within five days. It is also recommended that Source Production & Equipment Company, Inc. be notified.

8.2.3 Subsystems Maintenance

The model SPEC-150 has no subsystems.

8.2.4 Valves, Rupture Discs, and Gaskets on Containment Vessel

Not applicable since the primary containment vessel is a small sealed source capsule.

8.2.5 Shielding

The daily and quarterly inspection program performed by the licensee pursuant to 10 CFR 34.31 and 10 CFR 34.49 or equivalent agreement state regulations, are sufficient to establish the continuing integrity of the shield.

8.2.6 Thermal

Periodic thermal tests on the model SPEC-150 is not indicated since heat of decay for the maximum permissible activity 150 curie source is negligible. There are no components which be thermally degraded by typical use and transport.

8.2.7 Miscellaneous

The daily and quarterly inspection and maintenance program required of all licensed users of the model SPEC-150 is more than sufficient to assure the continuing integrity of the package.

9. Appendices

9.1 Drawings

DRAWING		TITLE
15B000	Rev (9)	Isometric View
15B002A	Rev (8)	Full Sectional View
15B008	Rev (7)	Depleted Uranium Shield
19B005	Rev (2)	LM-200 Lock Module
19B006	Rev (2)	LM-200 Lock Operation
190909	Rev (0)	Source Lock Operation

9.2 Photographs

PHOTO	DESCRIPTION - Prototype No. 2
53A	First 9 meter drop - Set up
53B	First 9 meter drop - Set up
53G	First 9 meter drop - Set up
53C	First 9 meter drop - Lock cap separation
53D	First 9 meter drop - Source connector undamaged

53E	First 9 meter drop - Source connector undamaged
53F	First 9 meter drop - Superficial damage
53H	First 9 meter drop - Superficial damage
53I	First 9 meter drop - Imprint on target
54A	Second 9 meter drop - Drop orientation
54B	Second 9 meter drop - Drop orientation
54C	Second 9 meter drop - ASM lid screw shear
54D	Second 9 meter drop - ASM lid screw shear
54E	Second 9 meter drop - Source connector
54F	Second 9 meter drop - Remained locked
55A	Third 9 meter drop - Orientation
55B	Third 9 meter drop - Orientation
55D	Third 9 meter drop - Side Flange Dented
55E	Third 9 meter drop - Side Flange Dented
55F	Third 9 meter drop - Outlet panel
55G	Third 9 meter drop - Top of housing
56A	Fourth 9 meter drop - Orientation
56B	Fourth 9 meter drop - Orientation
56C	Fourth 9 meter drop - Outlet panel screws shear
56D	Fourth 9 meter drop - Outlet panel screws shear
56E	Fourth 9 meter drop - Outlet panel distortion
56F	Fourth 9 meter drop - Source remained locked
56G	Fourth 9 meter drop - Source remained locked
57A	Puncture test- Set up
57B	Puncture test- Set up
57C	Puncture test- No damage

PHOTO	DESCRIPTION - Prototype No. 4
510A	Penetration - Set up
510B	Penetration - Safety plug and outlet nipple intact
510C	Penetration - Safety plug and outlet nipple intact
511A	Additional weight added to handle
511B	Additional weight added to handle
513A	First 9 meter drop - Orientation
513B	First 9 meter drop - Orientation
513C	First 9 meter drop - Orientation
513D	First 9 meter drop - Height of drop
513E	First 9 meter drop - Corner damage
513F	First 9 meter drop - Corner damage
513G	First 9 meter drop - Corner damage
513H	First 9 meter drop - Superficial damage to side
513I	First 9 meter drop - Opposite corner damage
514A	Puncture test - Set up
514B	Puncture test - Set up
514C	Puncture test - Set up
514F	Puncture test - Set up
514D	Puncture test - Impression of target pin
514E	Puncture test - Impression of target pin
515A	Second 9 meter drop - Set up
515B	Second 9 meter drop - Set up
515I	Second 9 meter drop - Set up
515J	Second 9 meter drop - Set up

515C	Second 9 meter drop - Drop height
515D	Second 9 meter drop - Plunger point of impact
515E	Second 9 meter drop - Impression of handle
515F	Second 9 meter drop - Impression of handle
515M	Second 9 meter drop - Impression of handle
515L	Second 9 meter drop - ASM lid plate bent upward
515G	Second 9 meter drop - ASM lid plate dented inward
515H	Second 9 meter drop - Lock plate bent outward
515K	Second 9 meter drop - Lock plate bent outward
515N	Second 9 meter drop - Survey by LRPD staff member
F01	First 4 foot drop - Set up and orientation
F02	First 4 foot drop - Landed on bottom and rolled
F03	First 4 foot drop - Survey
F04	First 4 foot drop - Survey
F05	Second 4 foot drop - Set up and orientation
F06	Second 4 foot drop - Landed on side flange
F07	Second 4 foot drop - Survey
F08	Second 4 foot drop - Survey
F09	Third 4 foot drop - Set up and orientation
F10	Third 4 foot drop - Landed on lock end corner
F11	Third 4 foot drop - Survey
F12	Third 4 foot drop - Survey
F13	Fourth 4 foot drop - Set up and orientation
F14	Fourth 4 foot drop - Landed on outlet end corner
F15	Fourth 4 foot drop - Survey
F16	Fourth 4 foot drop - Survey
F17	Fifth 4 foot drop - Set up and orientation
F18	Fifth 4 foot drop - Landed flat on lock cap
F19	Fifth 4 foot drop - Survey
F20	Fifth 4 foot drop - Survey
G01	Before 3rd 30-foot drop - Right side, highest surface radiation spot
G02	Before 3rd 30-foot drop - Top, highest surface radiation spot
G03	Before 3rd 30-foot drop - Left side, highest surface radiation spot
G04	Before 3rd 30-foot drop - Outlet end, highest surface radiation spot
G05	Before 3rd 30-foot drop - Lock end, highest surface radiation spot
G06	Before 3rd 30-foot drop - Lock end, source assembly connector
G07	Before 3rd 30-foot drop - Lock end, source assembly connector
G08	Before 3rd 30-foot drop - Lock end, source assembly connector
G09	Before 3rd 30-foot drop - Orientation, lock cap
G10	Before 3rd 30-foot drop - Orientation, lock cap
G11	Before 3rd 30-foot drop - Orientation, lock cap
G12	Before 3rd 30-foot drop - Suspended over target
H01	After 3rd 30-foot drop - Lock end
H02	After 3rd 30-foot drop - Lock cap
H03	After 3rd 30-foot drop - Lock end and target imprint
H04	After 3rd 30-foot drop - Lock end, bottom plate
I01	Before 2nd Puncture Test - Set up
I02	Before 2nd Puncture Test - Orientation, lock cap
I03	After 2nd Puncture Test - Lock cap
I04	After 2nd Puncture Test - Lock cap
J02	Before 4th 30-foot drop - Orientation, lock cap
J03	Before 4th 30-foot drop - Orientation, lock cap
J04	Before 4th 30-foot drop - Orientation, lock cap

J01	Before 4th 30-foot drop - Orientation, lock cap
K01	After 4th 30-foot drop - Right side and target imprint
K02	After 4th 30-foot drop - Lock cap and right flange
K03	After 4th 30-foot drop - Lock cap and right flange
K04	After 4th 30-foot drop - Lock cap
K05	After 4th 30-foot drop - Source assembly lock engaged
K06	After 4th 30-foot drop - Source assembly connector
K07	After 4th 30-foot drop - Lock end
K08	After 4th 30-foot drop - Lock end

9.3 2011 Normal Conditions Test Report

9.4 1994 Sketches of Drop Test Impact Orientations

9.5 1994 Sketches of Highest Surface Radiation Survey Data

9.6 1997 30' Drop Test & Validation of Previous Puncture Tests

9.7 Shielding Analysis, Se-75 and Yb-169

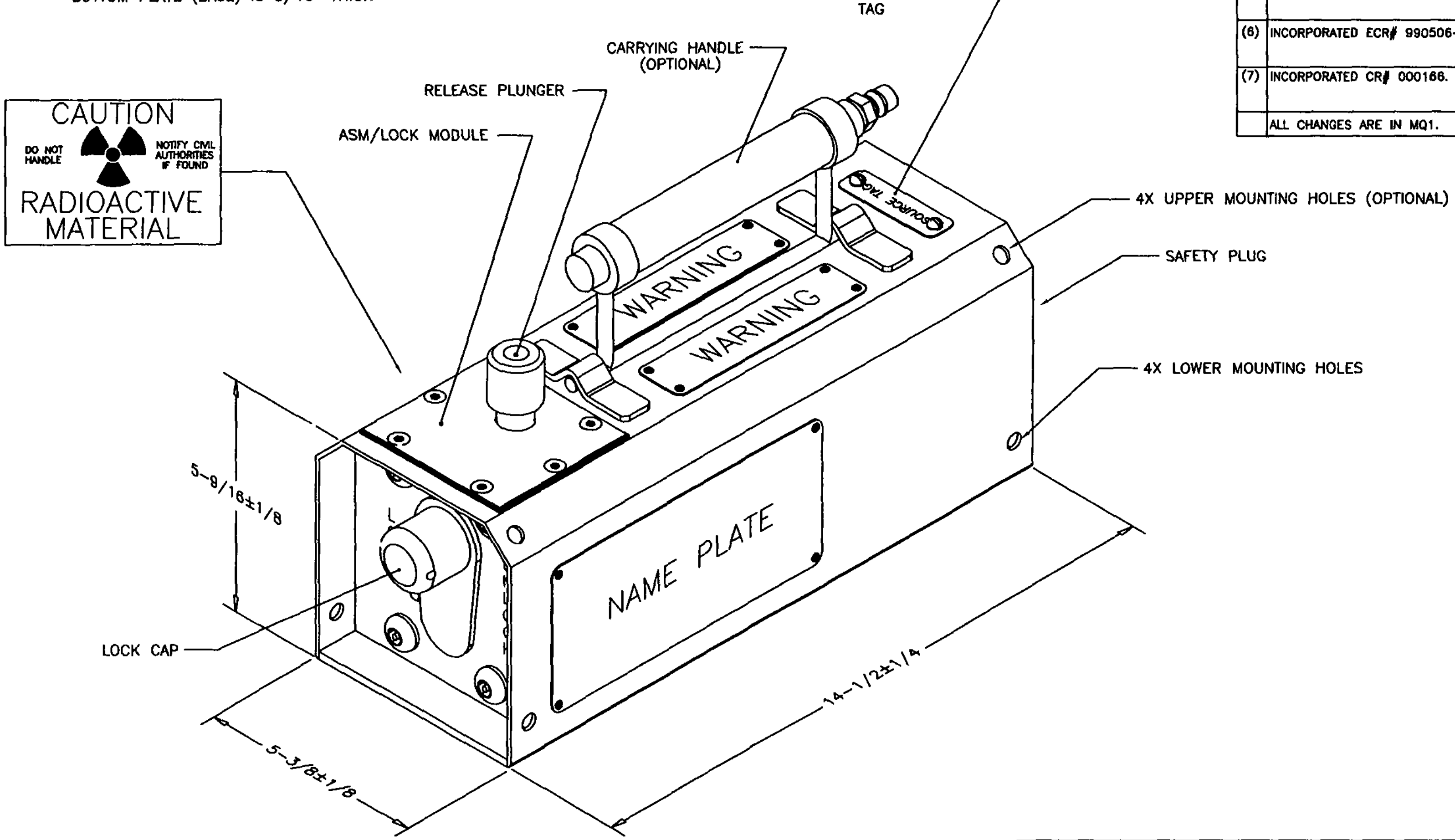
Appendix 9.1: Drawings

DRAWING		TITLE
15B000	Rev (9)	Isometric View
15B002A	Rev (8)	Full Sectional View
15B008	Rev (7)	Depleted Uranium Shield
19B005	Rev (2)	LM-200 Lock Module
19B006	Rev (2)	LM-200 Lock Operation
190909	Rev (0)	Source Lock Operation

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 FOR SERIAL NUMBERS 1475 AND NEWER.
   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



UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES TOLERANCES ARE AS NOTED DO NOT SCALE DRAWING	APPROVALS	DATE	SOURCE PRODUCTION & EQUIPMENT CO., INC. 113 TEAL ST, ST ROSE, LA 70087
	DRAWN KP	2-8-11	SPEC-150 TYPE B(U) PACKAGE ISOMETRIC VIEW
	CHECKED MQ1	MQ1	
TREATMENT NONE	ISSUED MQ1	MQ1	SIZE C · DWG NO. 15B000 · REV 9
FINISH NONE	QA CLASS N/A		SCALE: NTS · 00063309 · SHEET 1 of 1

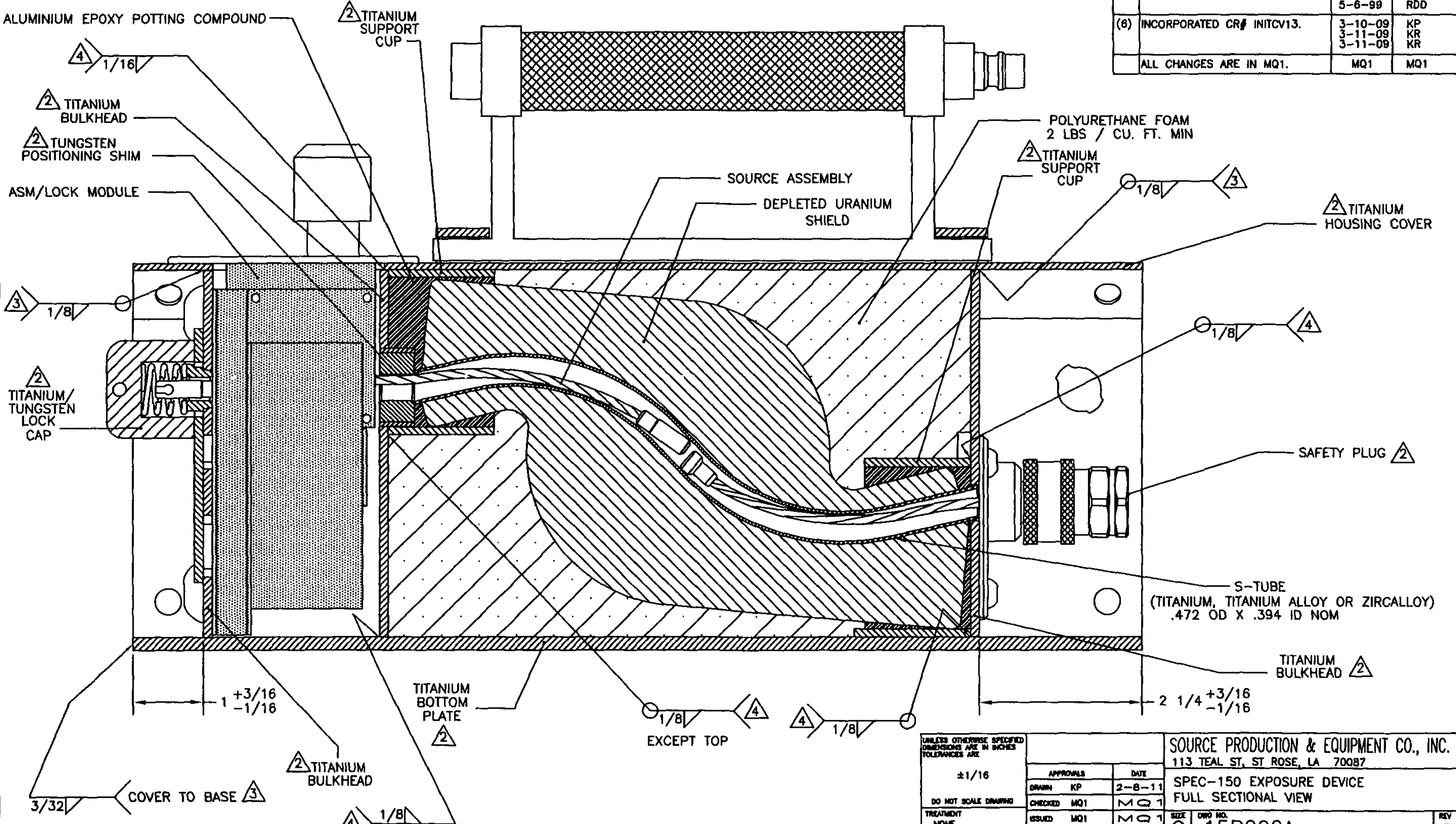
CONTROLLED COPY NO

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

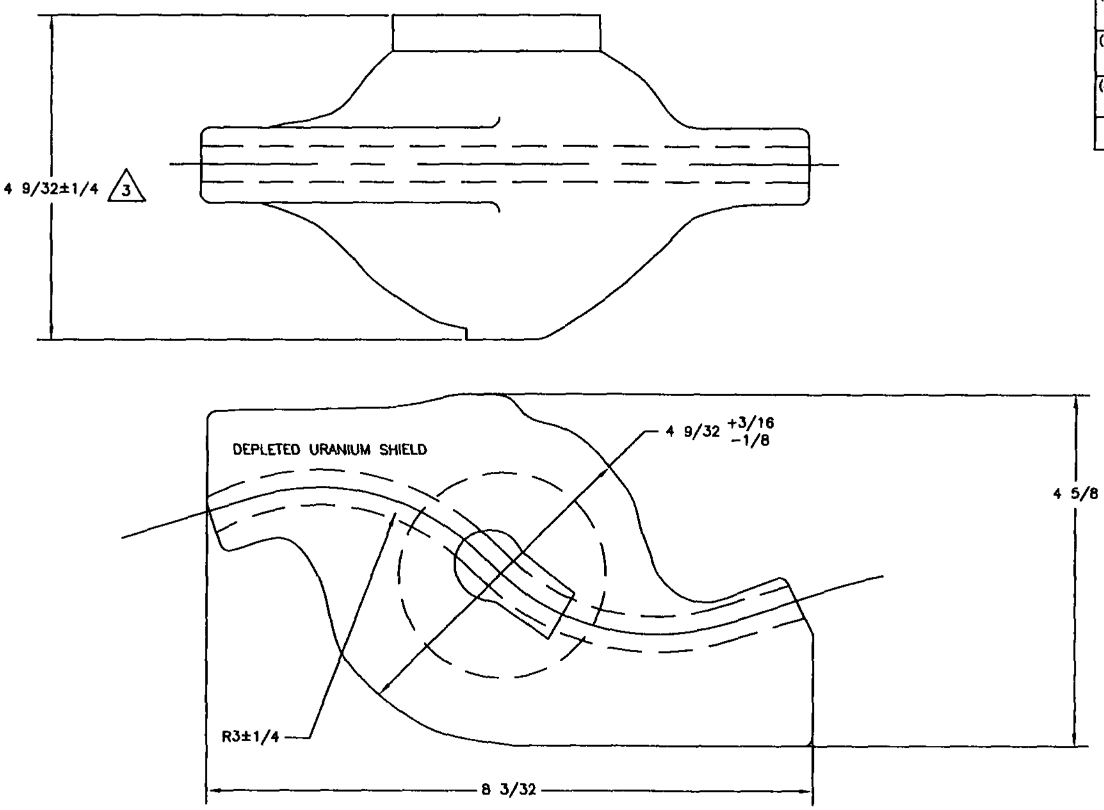
NOTES:

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.
5. NOTES 2,3,4 ARE APPLICABLE TO SPEC-150 SERIAL NUMBERS 1475 AND NEWER.



UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES TOLERANCES ARE ±1/16 DO NOT SCALE DRAWING			SOURCE PRODUCTION & EQUIPMENT CO., INC. 113 TEAL ST. ST ROSE, LA 70087
	APPROVALS	DATE	
TREATMENT NONE	DRAWN KP	2-8-11	SPEC-150 EXPOSURE DEVICE FULL SECTIONAL VIEW
	CHECKED MQ1	MQ1	
FINISH NONE	ISSUED MQ1	MQ1	SIZE C DWG NO. 15B002A REV 8
	QA CLASS N		SCALE: NTS 00062708 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 RDD
(2)	SEE QA FILE FOLDER 15B008.	4-13-95 4-13-95 4-13-95	S. BYRD KC RDD
(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# 000168.	3-10-09 3-11-09 3-11-09	KP KR KR
	ALL CHANGES ARE IN MQ1.	MQ1	MQ1



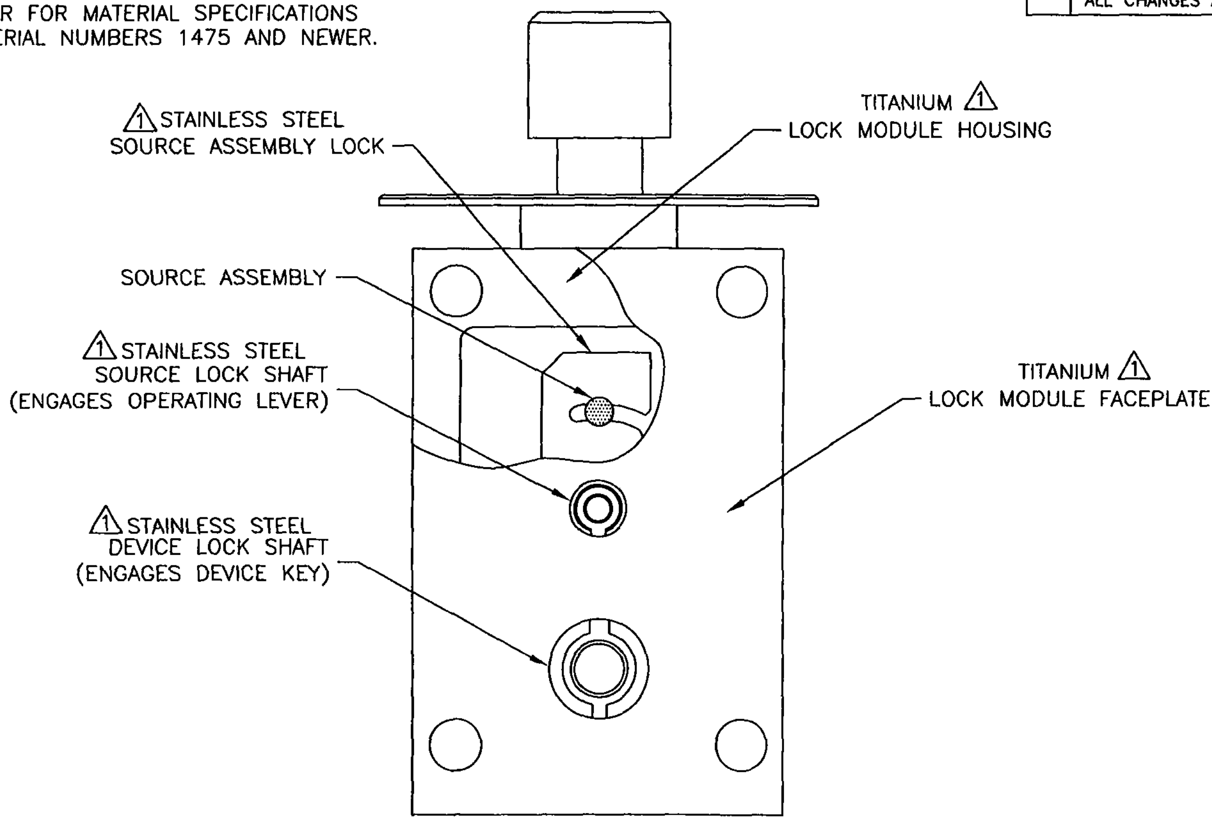
NOTES:

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.
4. THE MINIMUM WEIGHT AND MATERIAL DENSITY REQUIREMENTS ARE APPLICABLE TO SPEC-150 SERIAL NUMBERS 1475 AND NEWER.

UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES TOLERANCES ARE FRACTIONS ±1/8 DO NOT SCALE DRAWING	APPROVALS	DATE	SOURCE PRODUCTION & EQUIPMENT CO., INC. 113 TEAL ST, ST ROSE, LA 70087
	DRAWN KP	2-8-11	SPEC-150 TYPE B(U) PACKAGE DEPLETED URANIUM SHIELD
TREATMENT NONE	CHECKED MQ1	MQ1	SIZE C DWG NO. 15B008 REV 7
	ISSUED MQ1	MQ1	
FINISH NONE	QA CLASS N/A		SCALE: NTS | 00062607 | 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 FOR SERIAL NUMBERS 1475 AND NEWER.

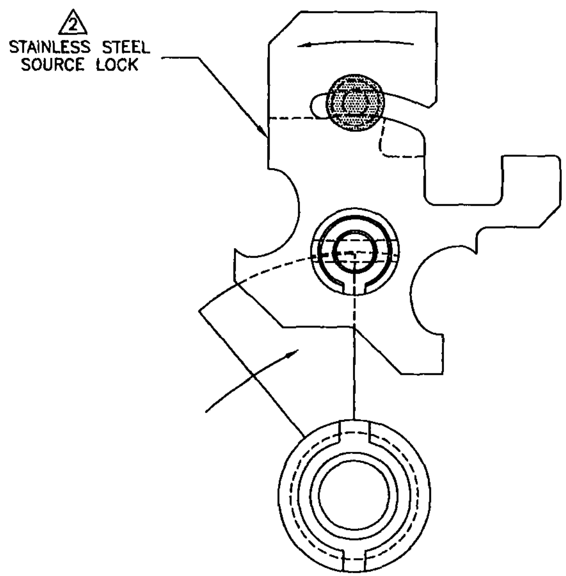


UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES TOLERANCES ARE FRACTIONS DECIMALS N/A DO NOT SCALE DRAWING	APPROVALS	DATE	SOURCE PRODUCTION & EQUIPMENT CO, INC 113 TEAL ST, ST ROSE, LA 70087
	DRAWN KP	2-7-11	LOCK MODULE- MODEL LM-200, SPEC
TREATMENT NONE	CHECKED MQ1	MQ1	
	APPROVED MQ1	MQ1	SIZE B DWG NO. 19B005 REV 2
FINISH NONE	QA CLASS N		SCALE: NTS 00106002 SHEET 1 OF 1

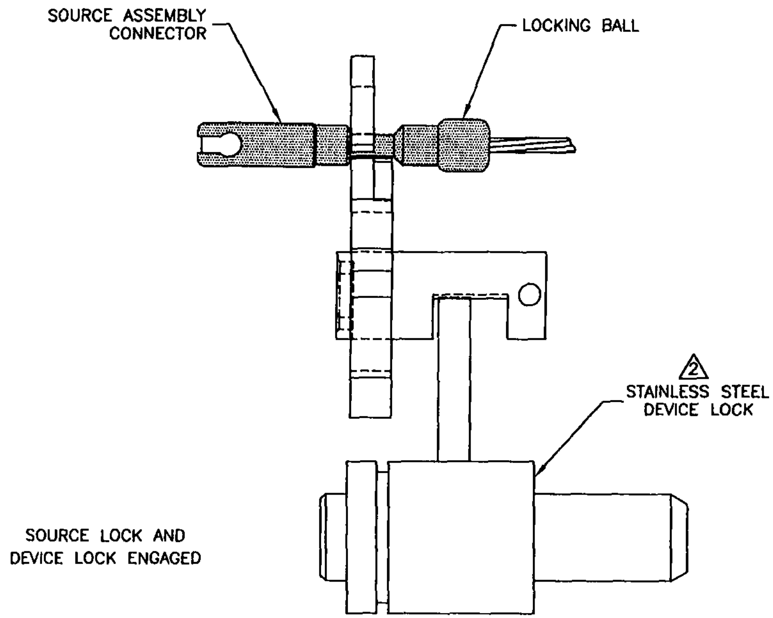
CONTROLLED COPY NO			
REVISIONS			
REV	DESCRIPTION	DATE	APPROVED
	ALL CHANGES ARE IN MQ1.	MQ1	MQ1

NOTES:

1. ARROWS INDICATE DIRECTION OF ROTATION TO UNLOCK.
2. SEE SAR FOR MATERIAL SPECIFICATIONS FOR SERIAL NUMBERS 1475 AND NEWER.



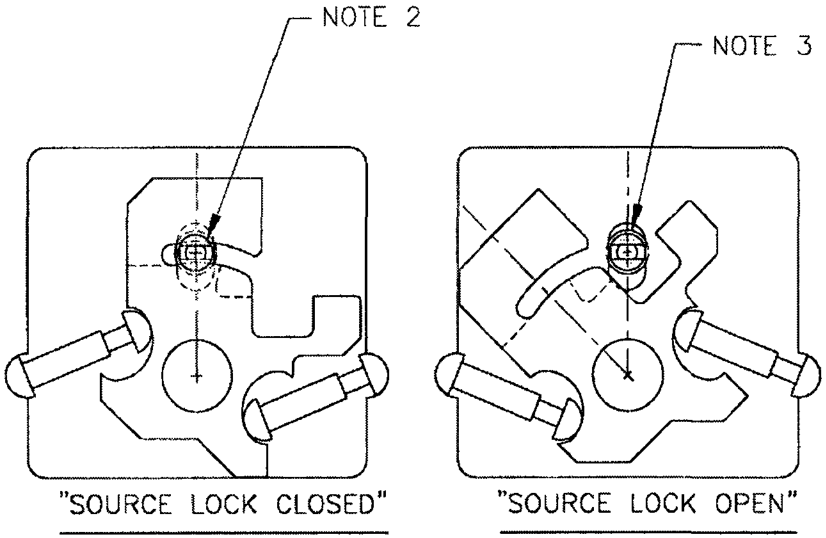
VIEW FROM LOCK END



SIDE VIEW

UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES TOLERANCES ARE N/A DO NOT SCALE DRAWING	APPROVALS	DATE	SOURCE PRODUCTION & EQUIPMENT CO., INC. 113 TEAL ST, ST ROSE, LA 70087
	DRAWN KP	2-7-11	DEVICE LOCK OPERATION (LOCKED)- MODEL LM-200, SPEC
	CHECKED MQ1	MQ1	
TREATMENT NONE	APPROVED MQ1	MQ1	SIZE C DWG NO. 19B006 REV 2
FINISH NONE	QA CLASS N		SCALE: NTS 00106102 SHEET 1 OF 1

CONTROLLED COPY NO			
REVISIONS			
REV	DESCRIPTION	DATE	APPROVED



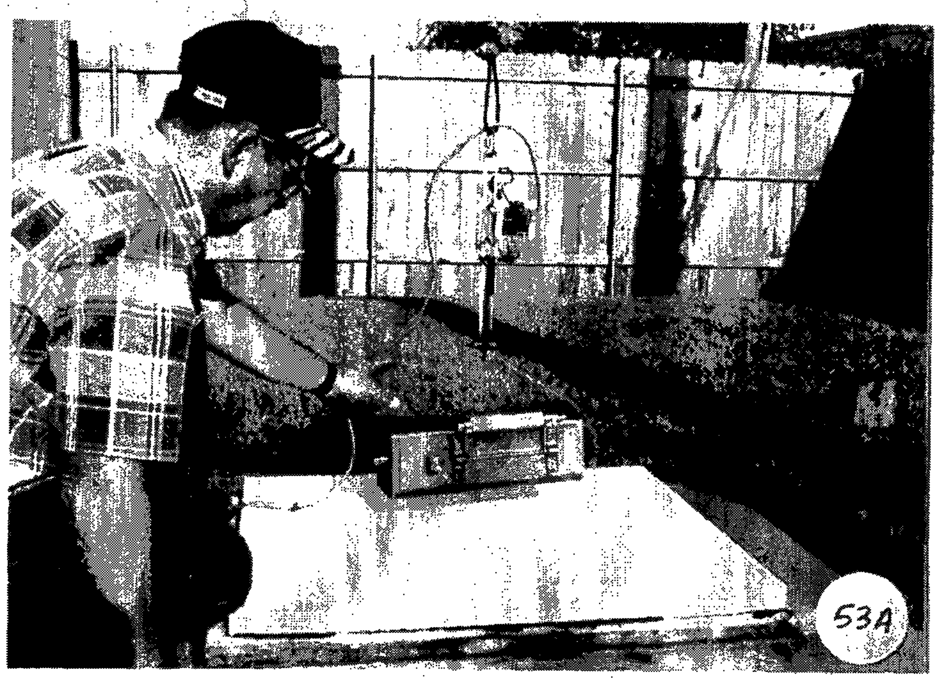
NOTE 1: VIEWS FROM THE LOCK END OF THE DEVICE.
NOTE 2: SOURCE ASSEMBLY CONNECTOR
NOTE 3: LOCKING BALL

FOR ILLUSTRATION PURPOSES ONLY

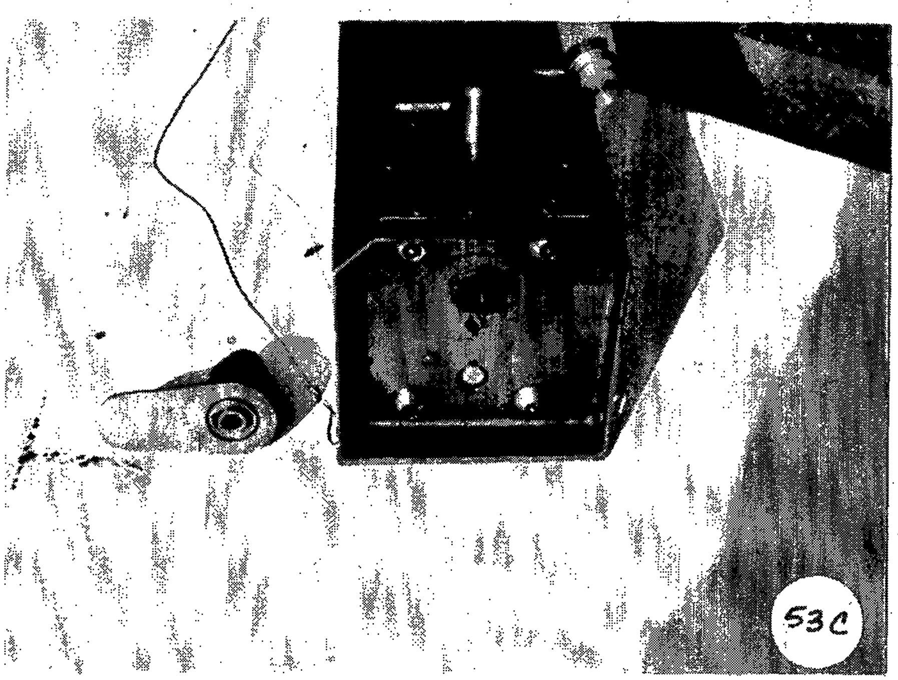
N/A	APPROVALS	DATE	SOURCE PRODUCTION & EQUIPMENT CO., INC. 113 TEAL ST, ST ROSE, LA 70087
	DRAWN RAW	7/12/99	SOURCE LOCK OPERATION, SPEC-300 USER'S MANUAL
DO NOT SCALE DRAWING	CHECKED		
TREATMENT NONE	APPROVED		SIZE C · DWG NO. 190909 · REV 0
FINISH NONE			SCALE: NTS · 00084000 · SHEET 1 OF 1

Appendix 9.2
Photographs

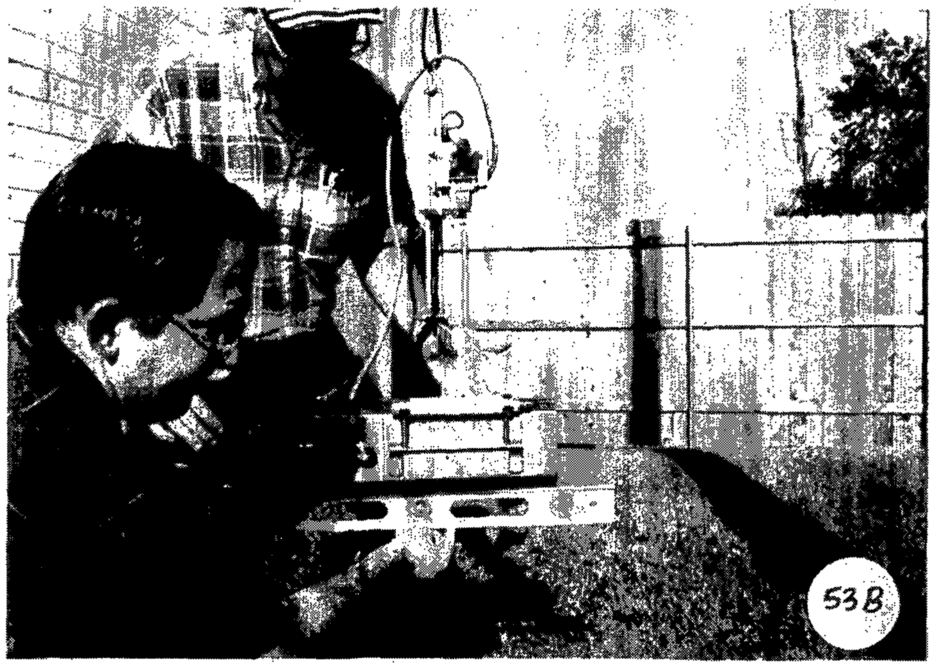
53G



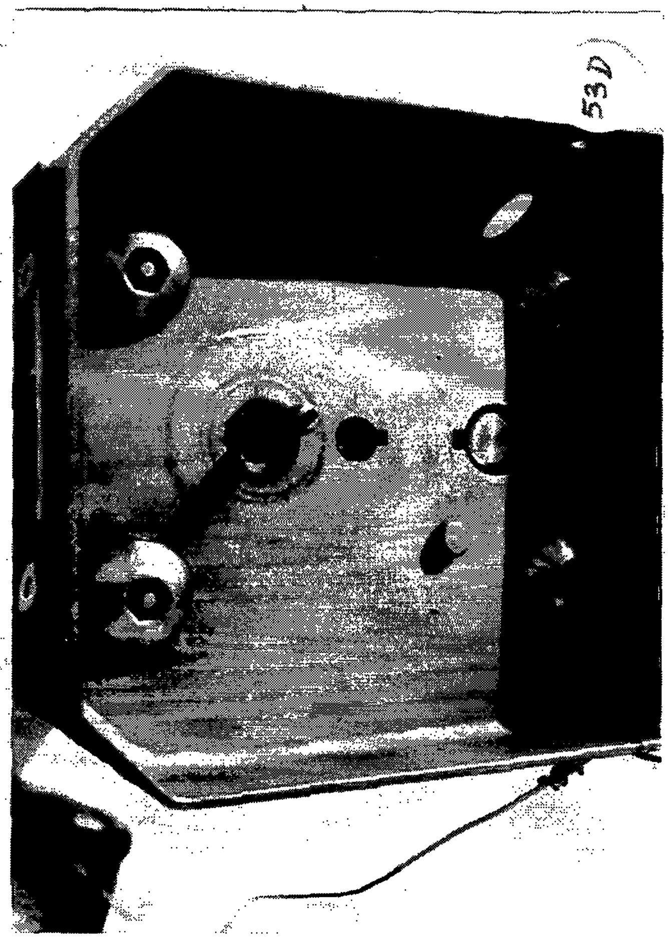
53A

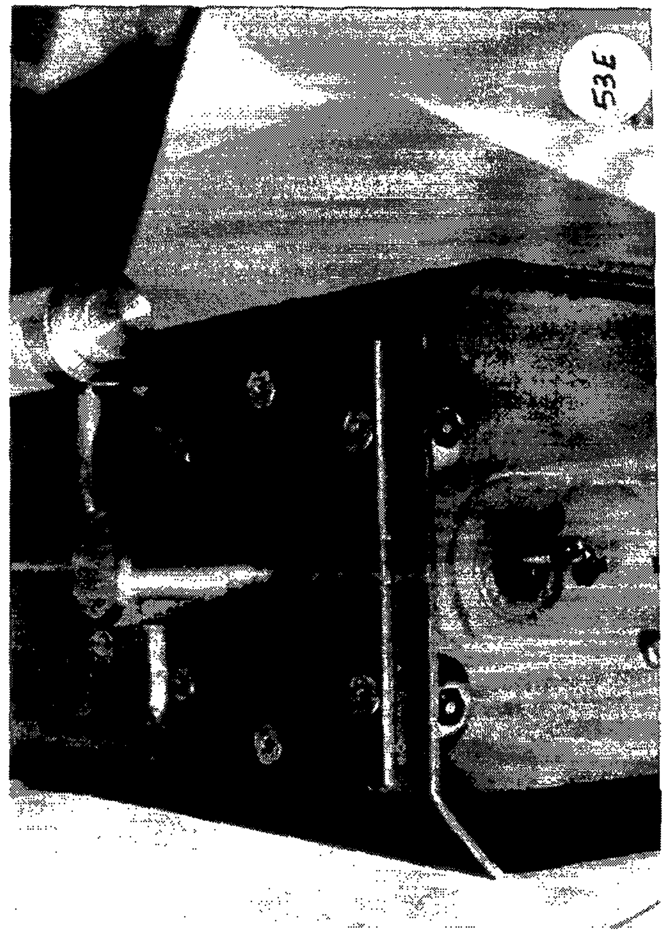


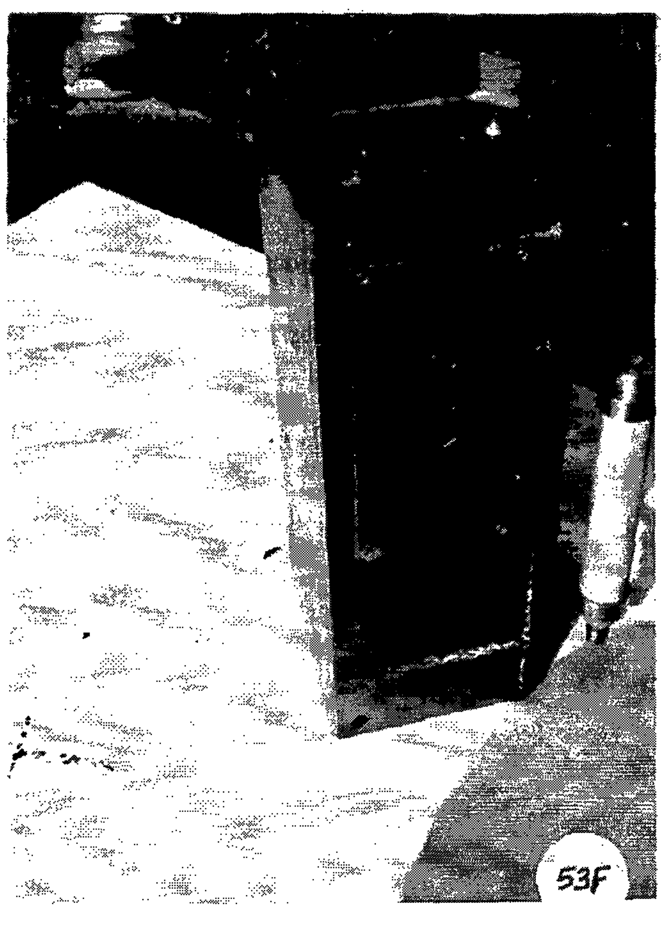
53C



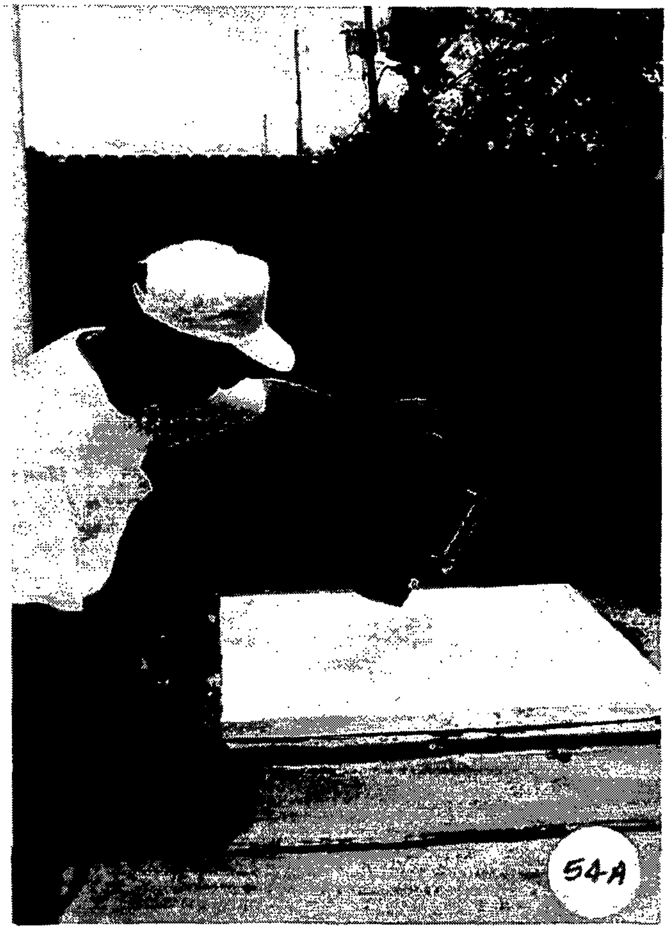
53B


53D

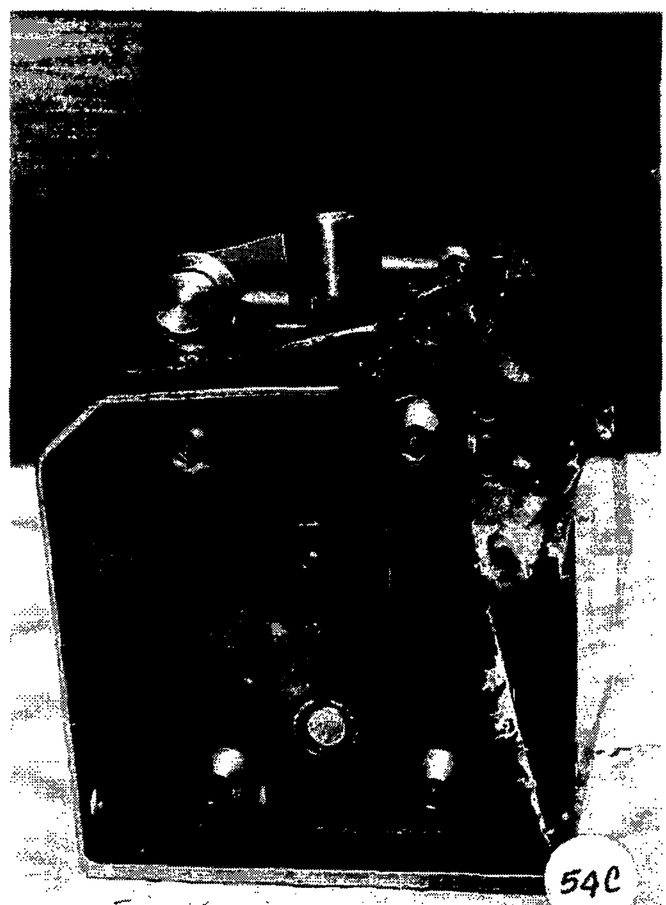

53E

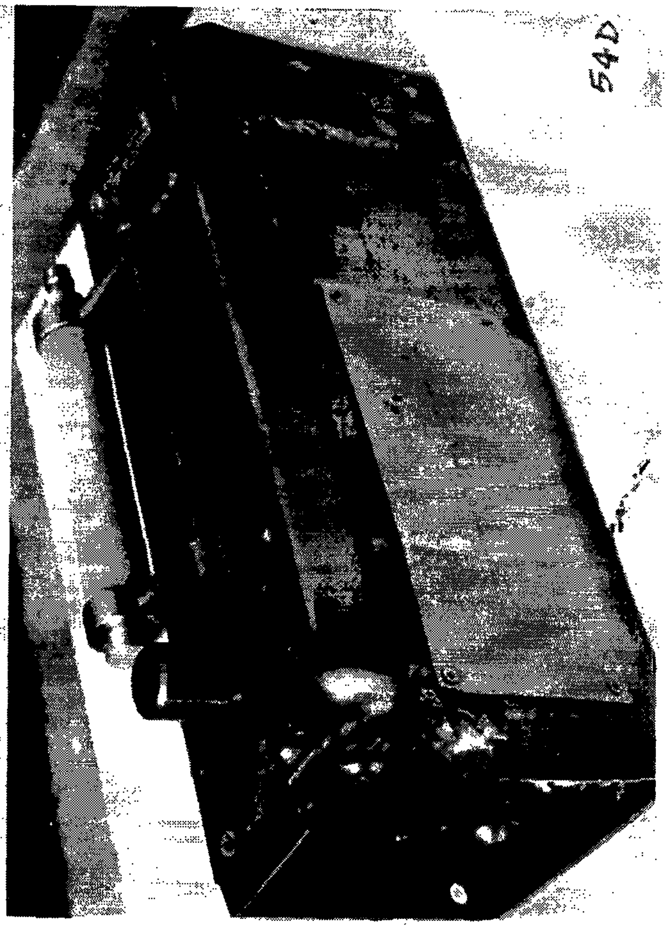

53F


53 I

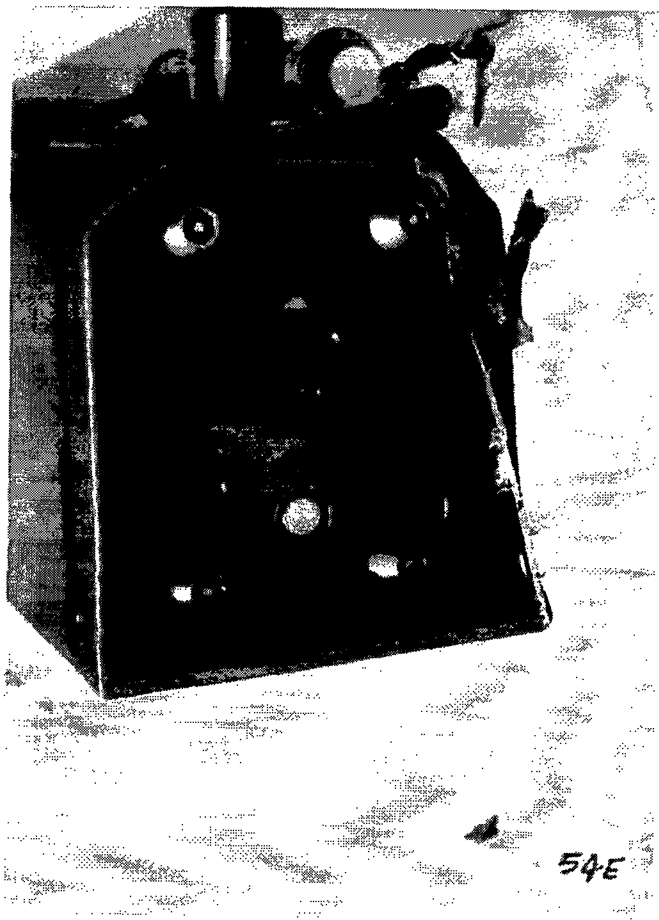

54A


54B

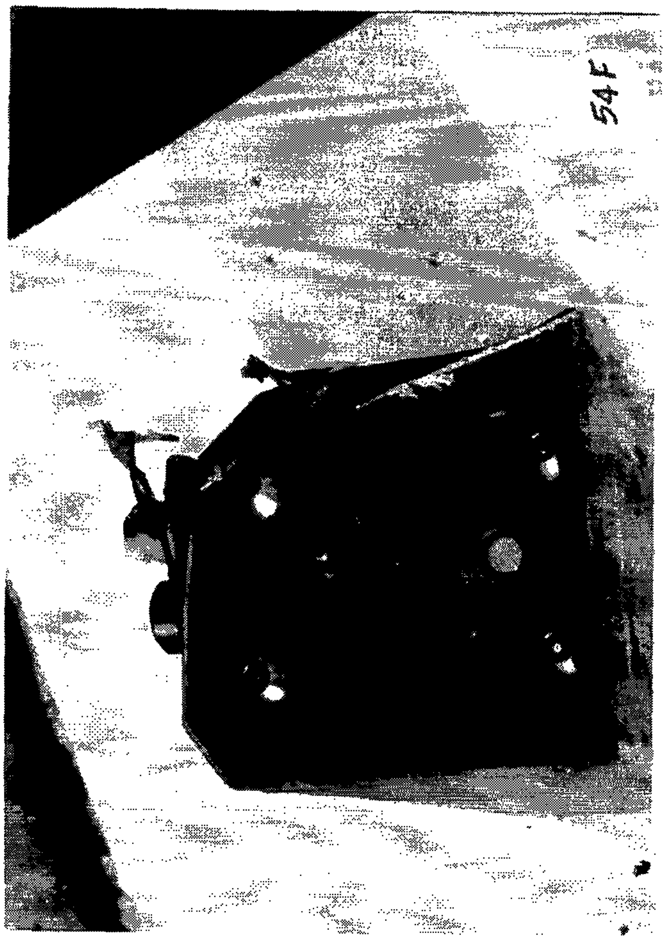

54C



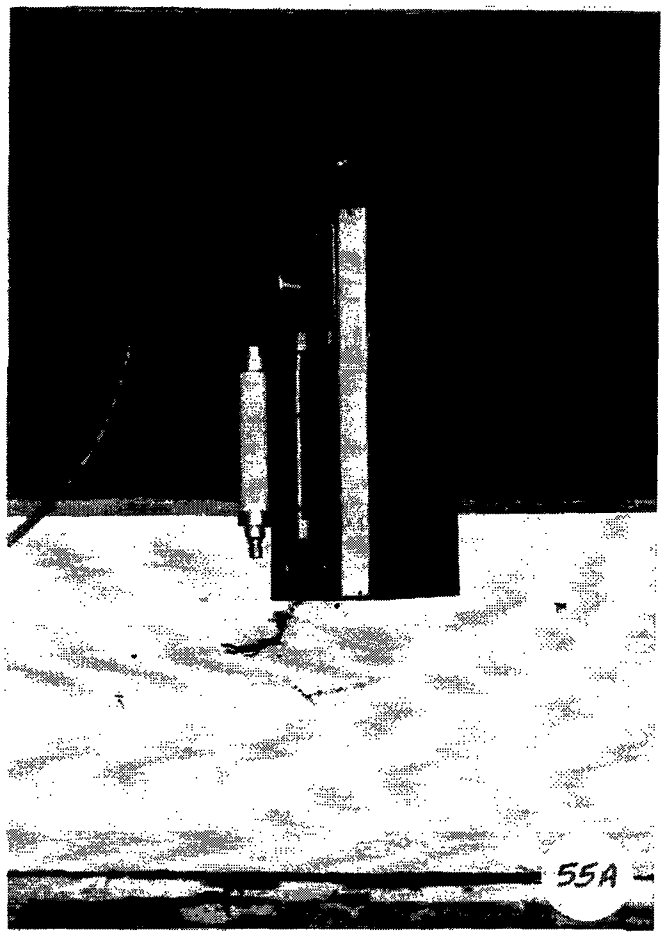
54D



54E

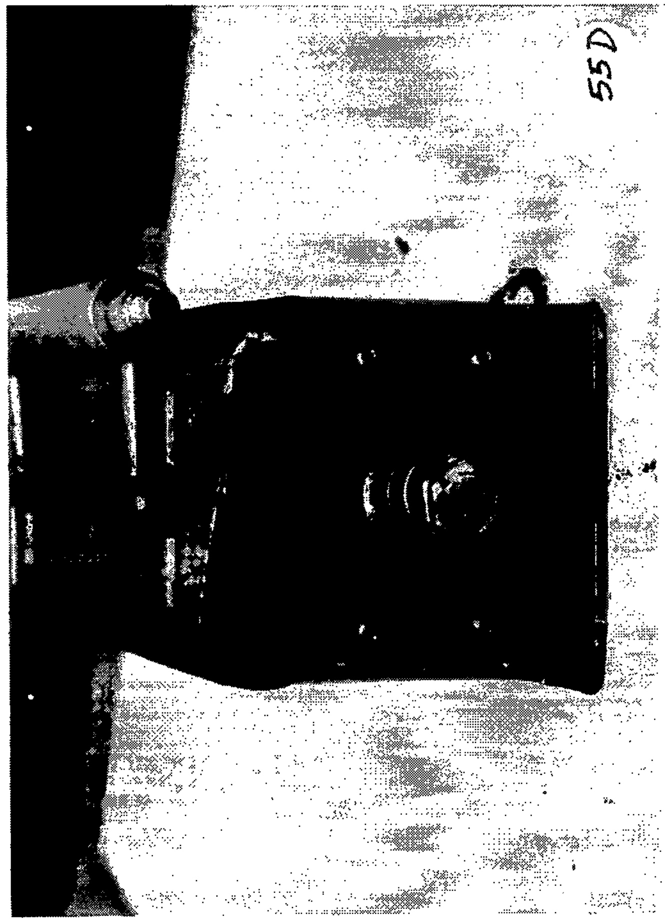


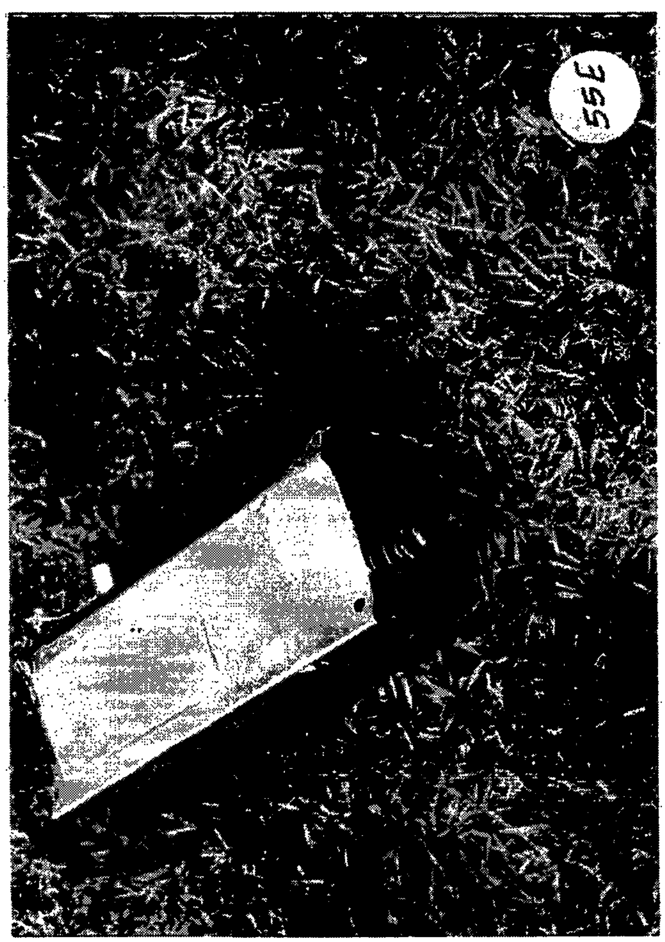
54F

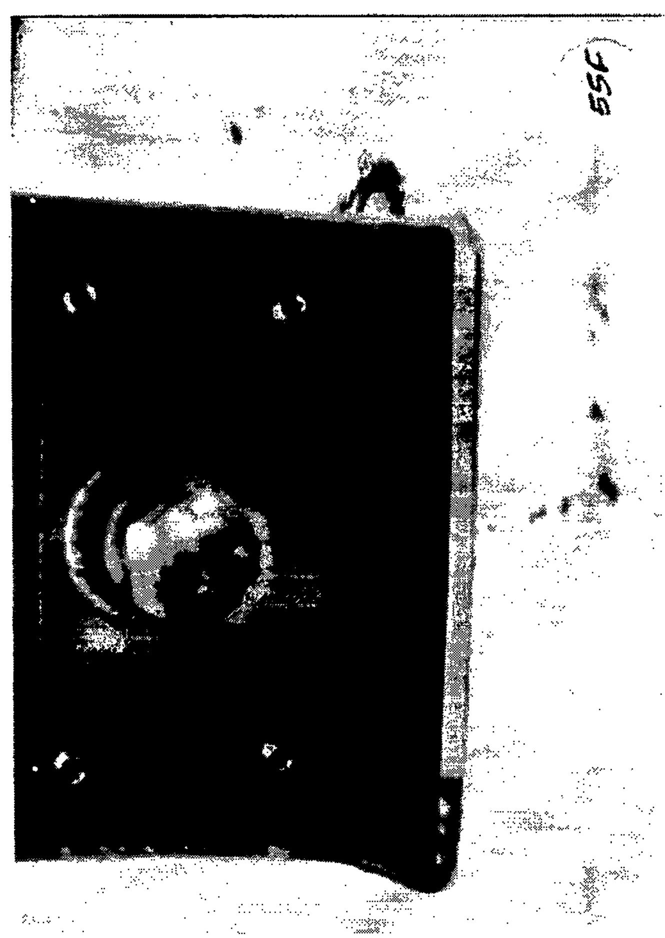


55A

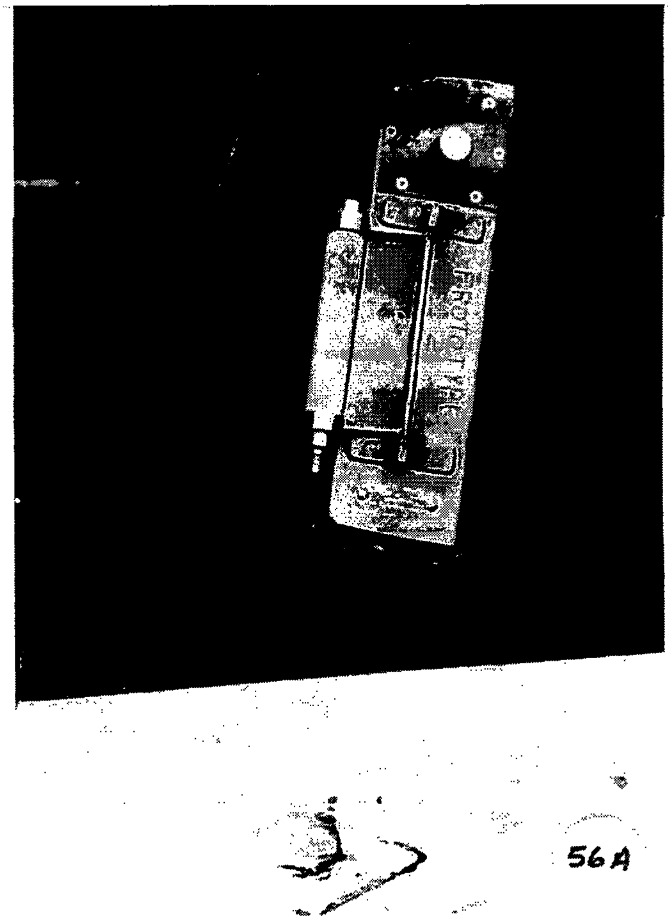

55B


55D

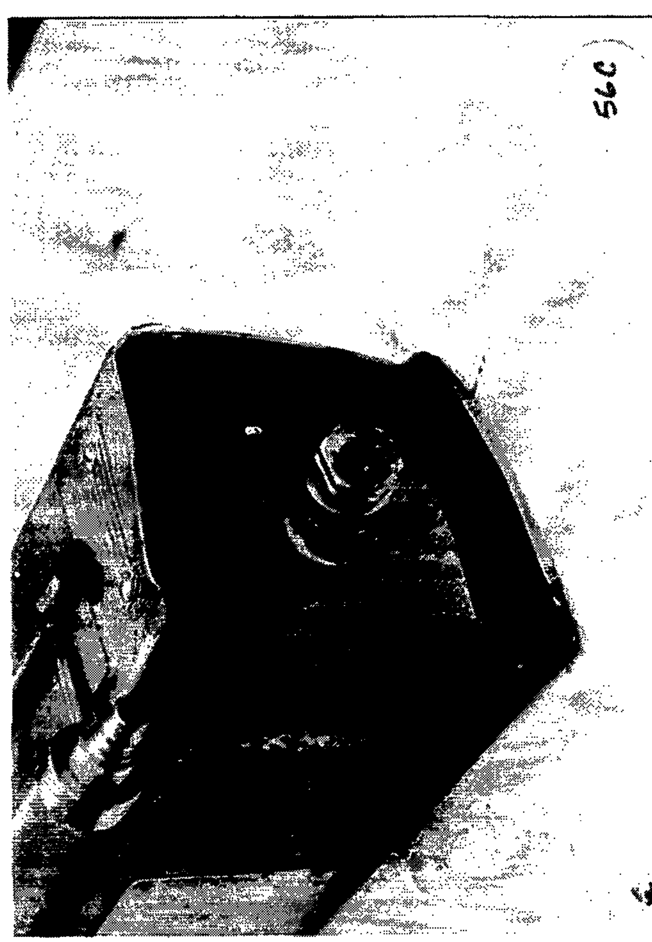

55E

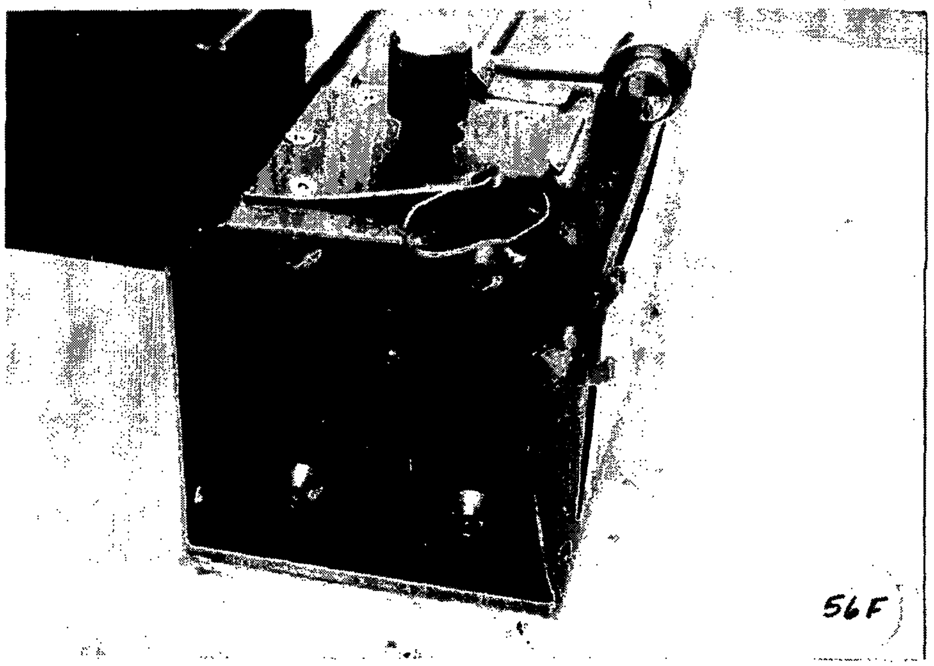

55F

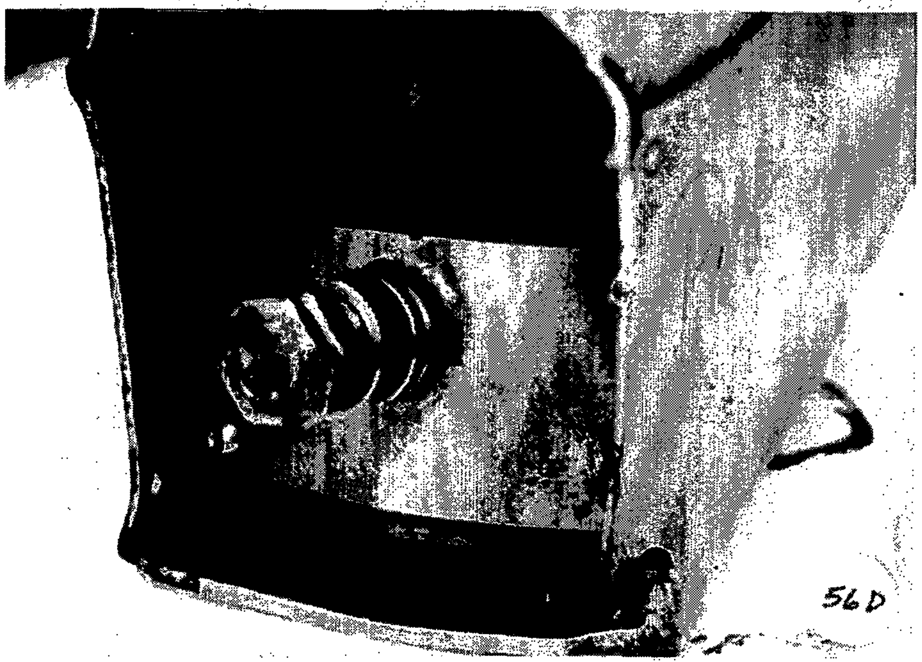

55G

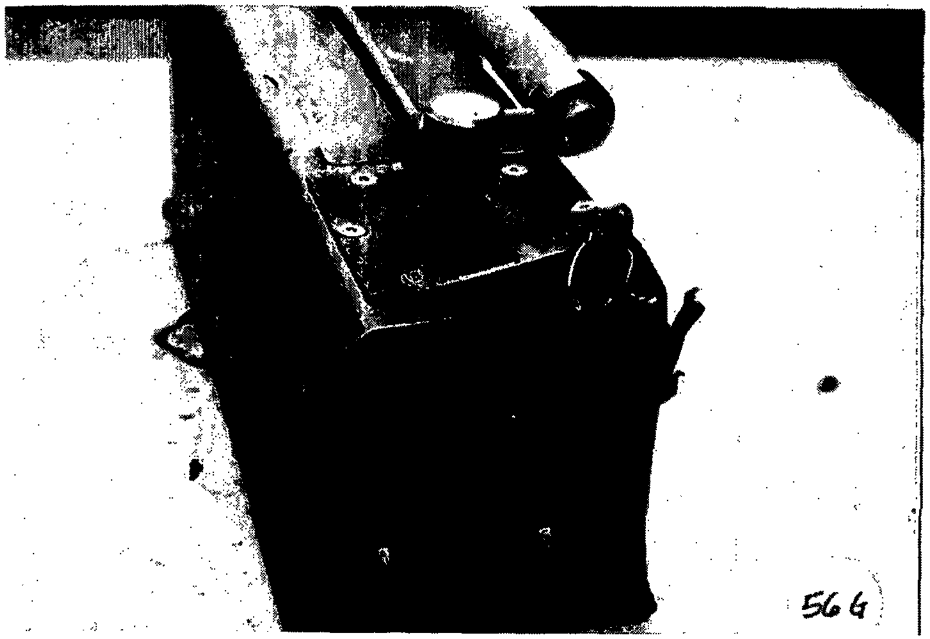

56A

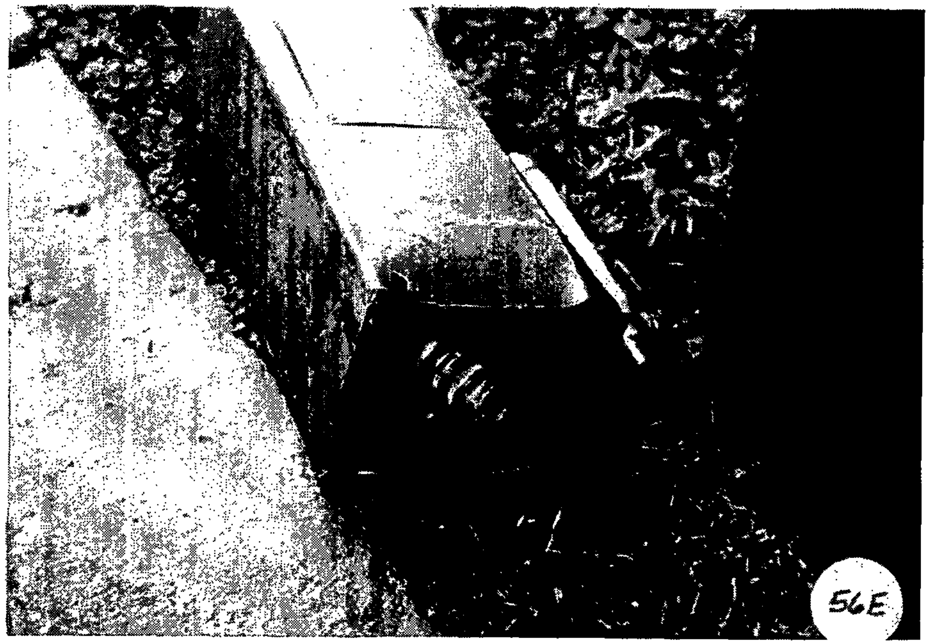

56B

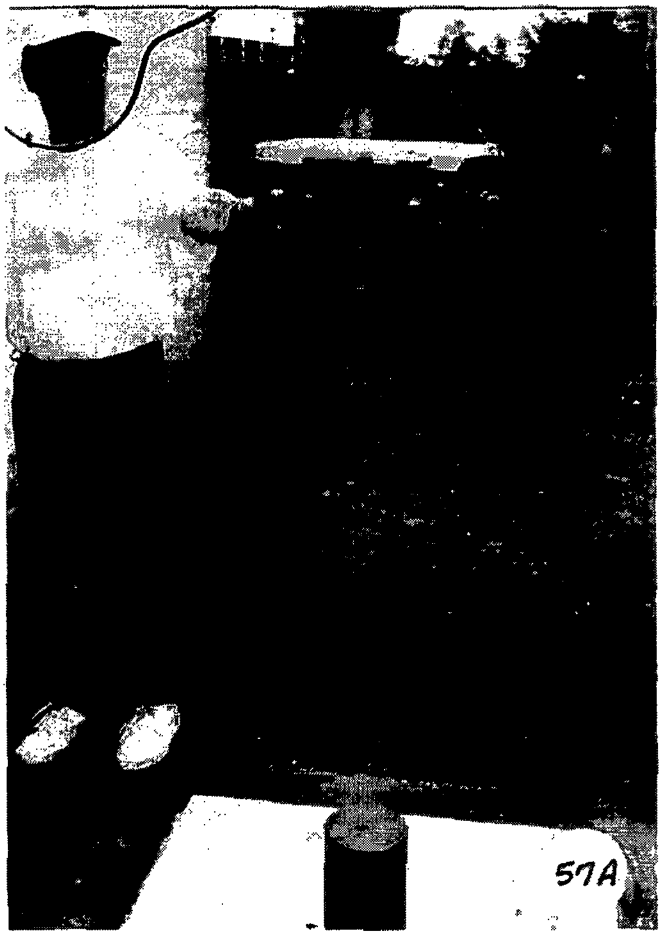

56C

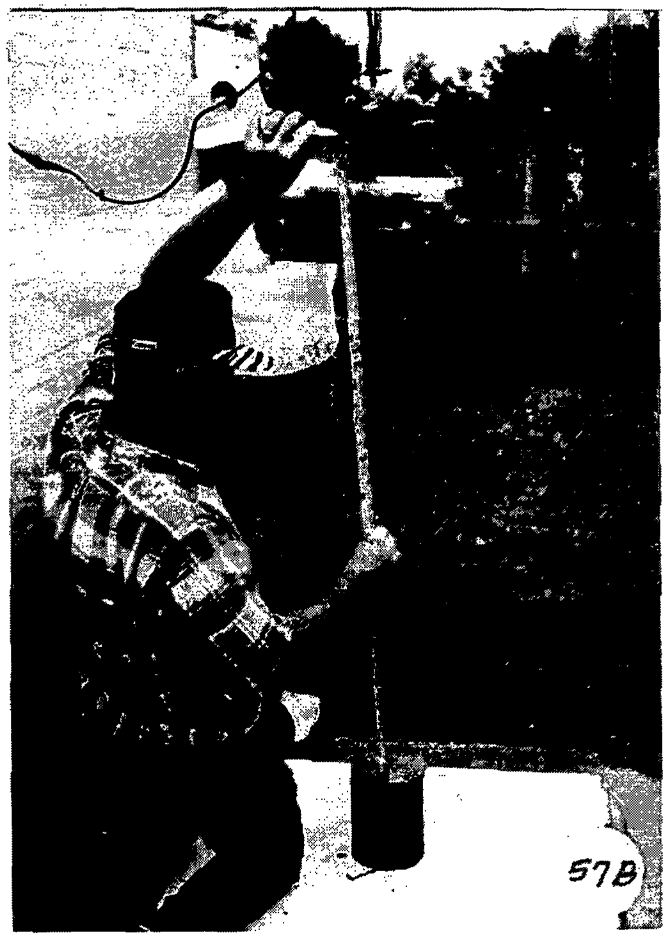

56F

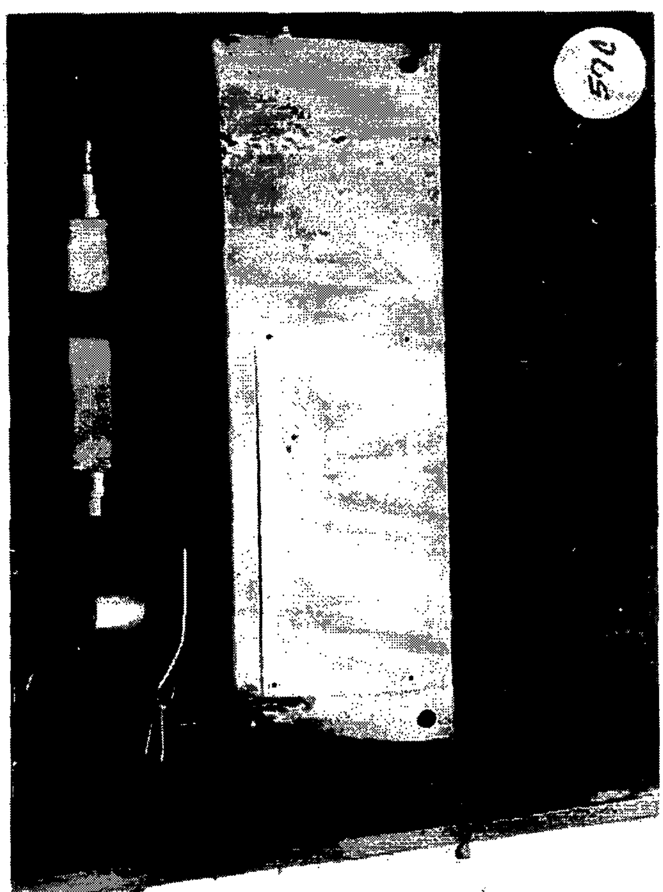

56D

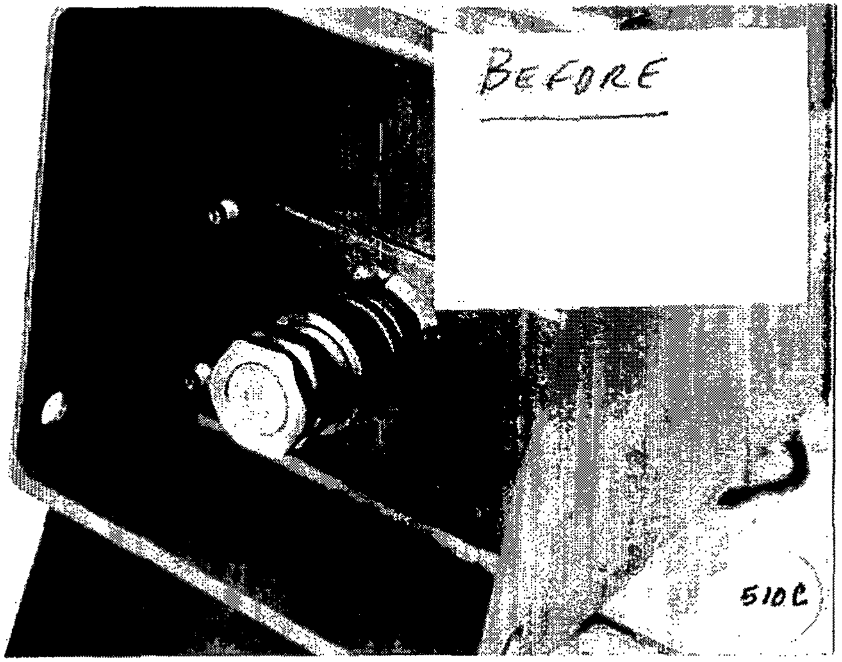

56G

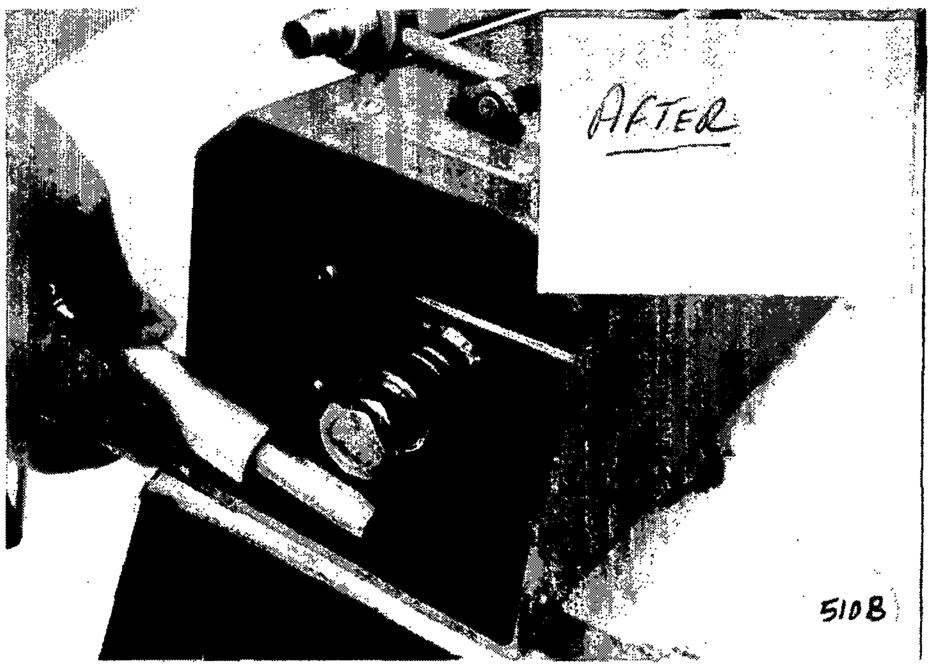

56E


57A

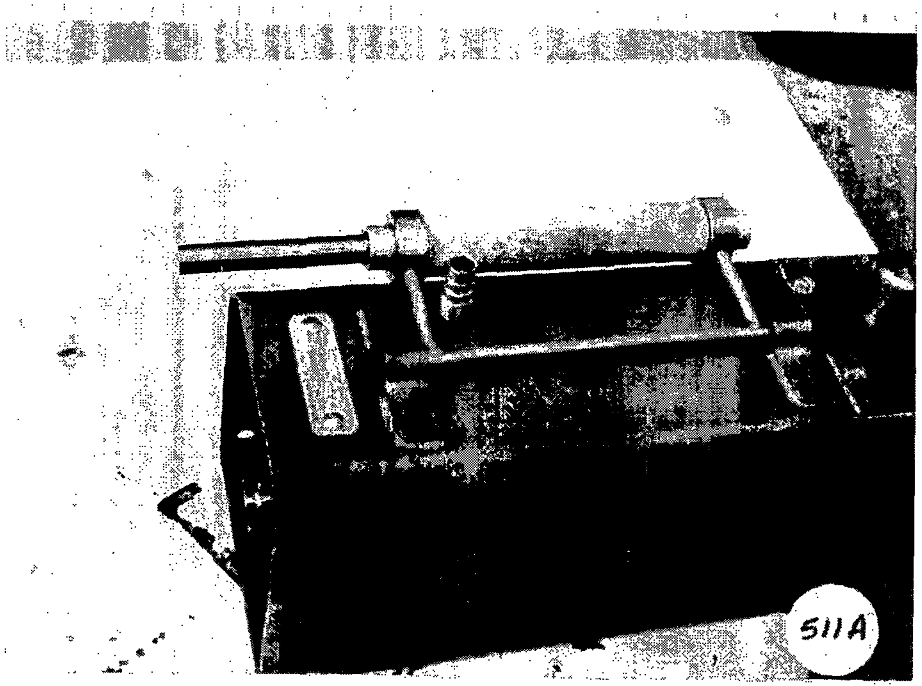

57B


57C

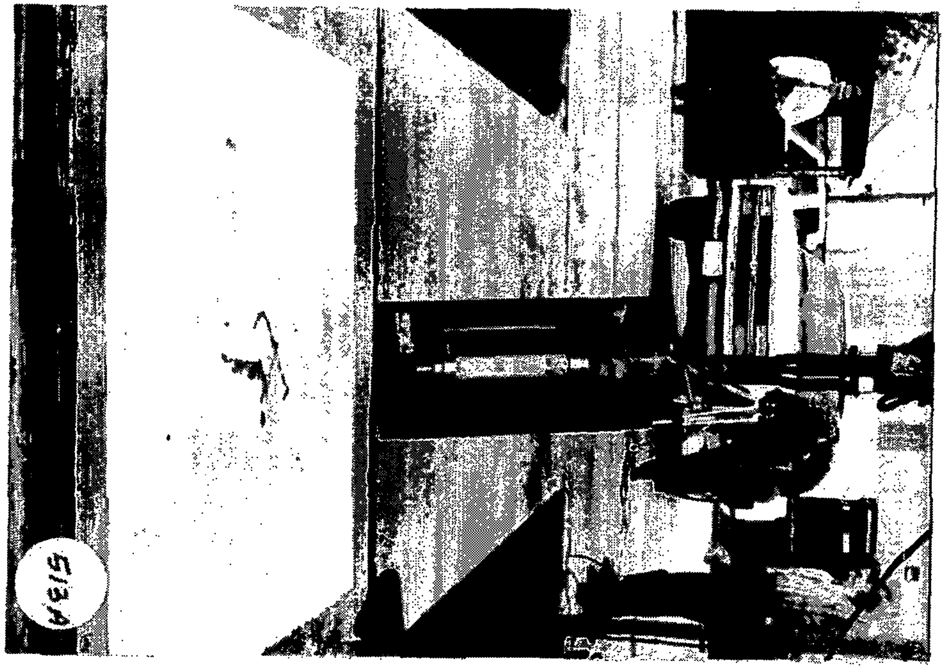

BEFORE
510C

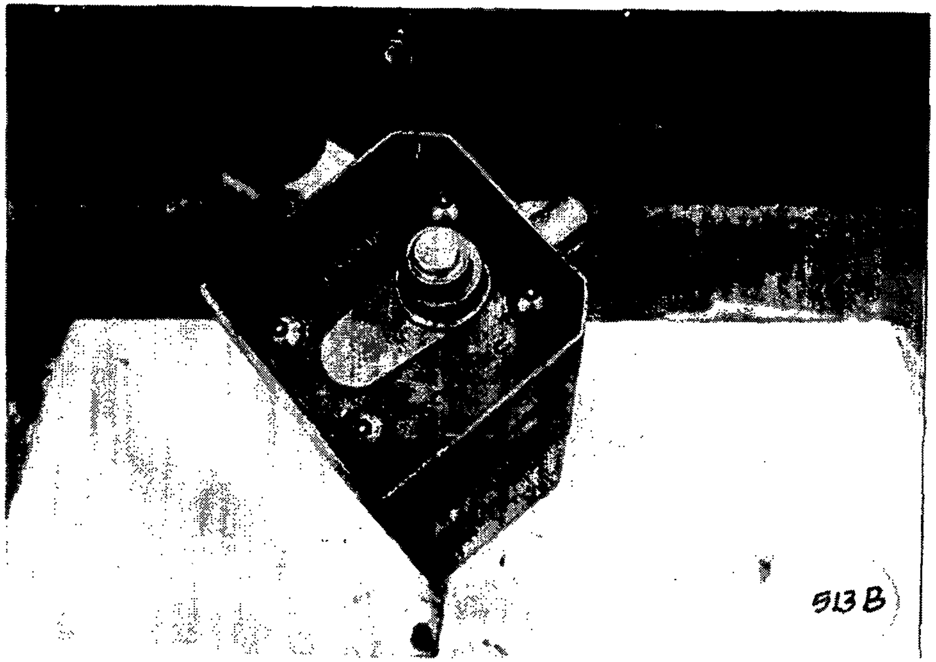

AFTER
510B


510A


511A

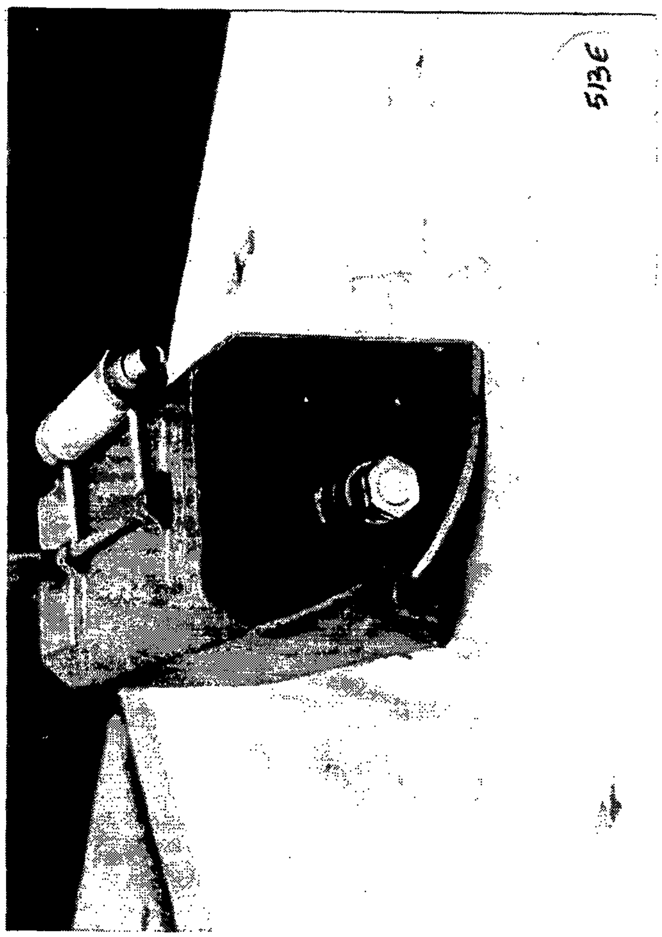

511B


513A


513B

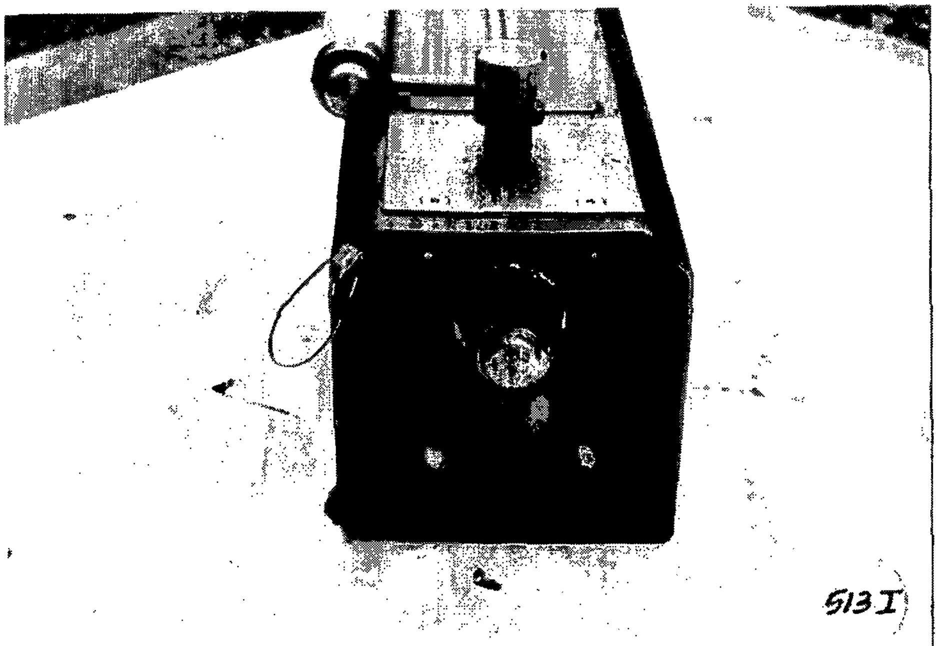
513C

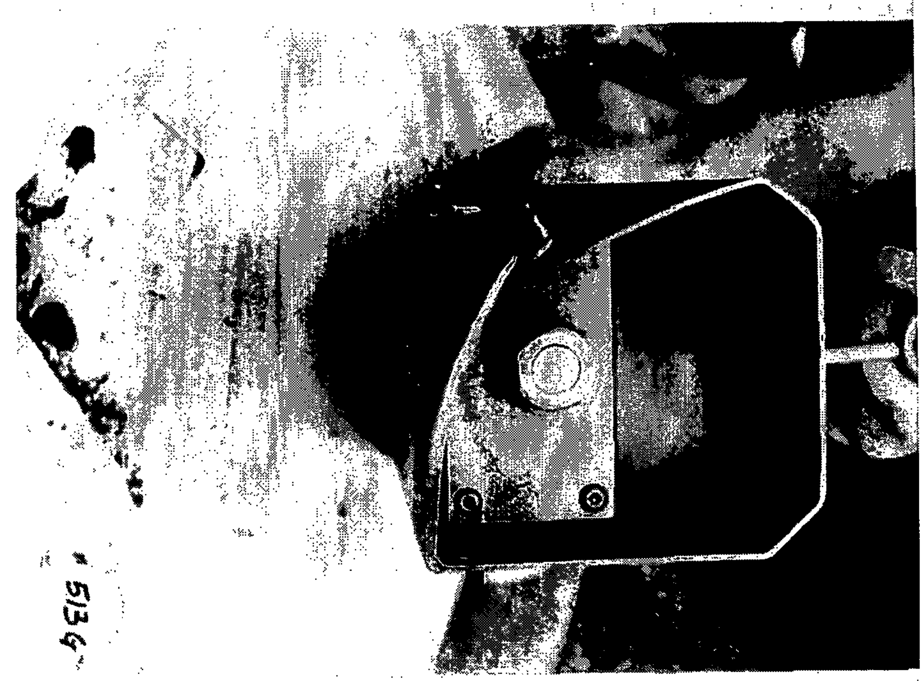
513D



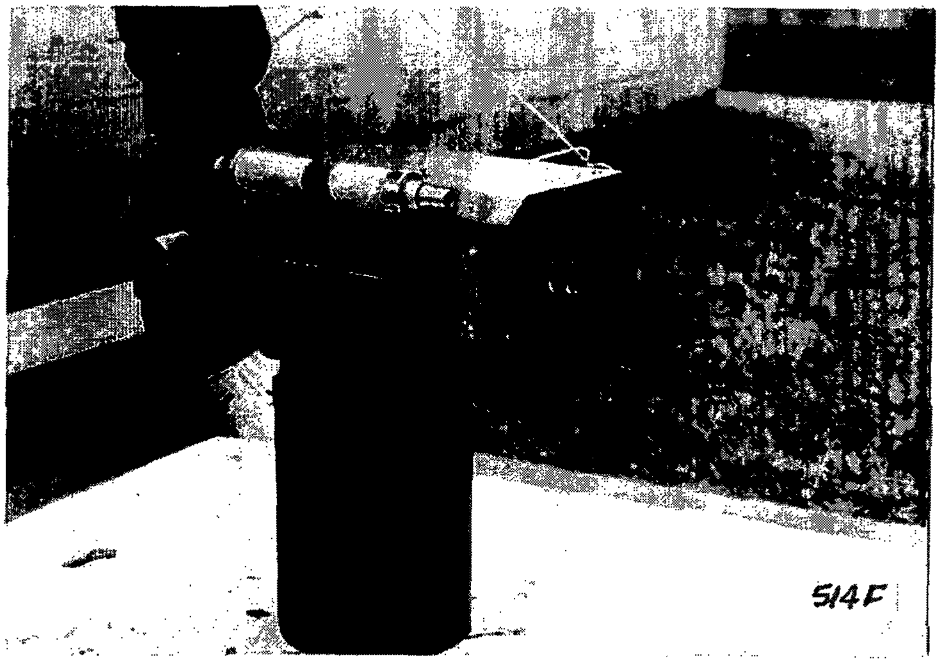
513E

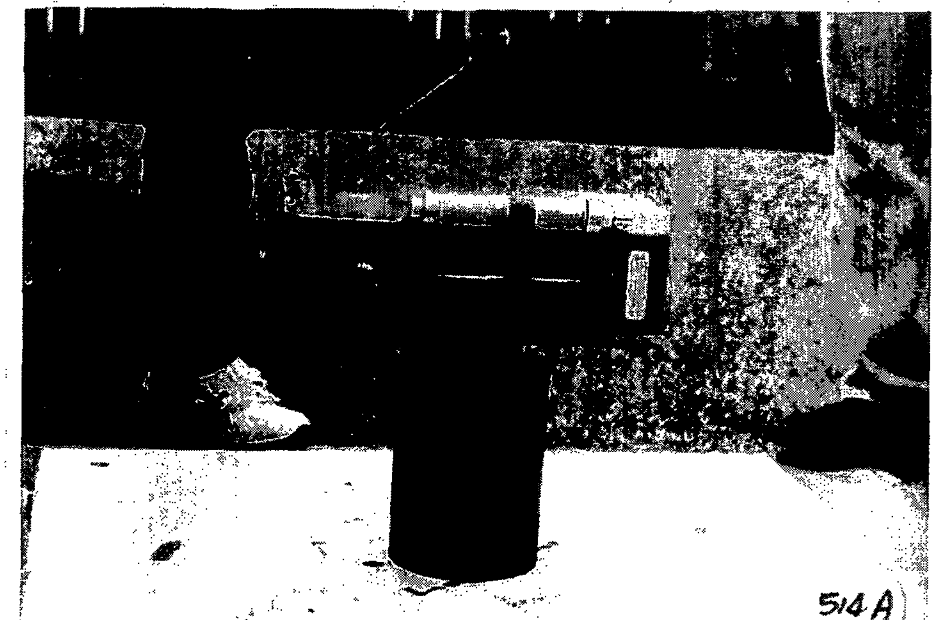
513F

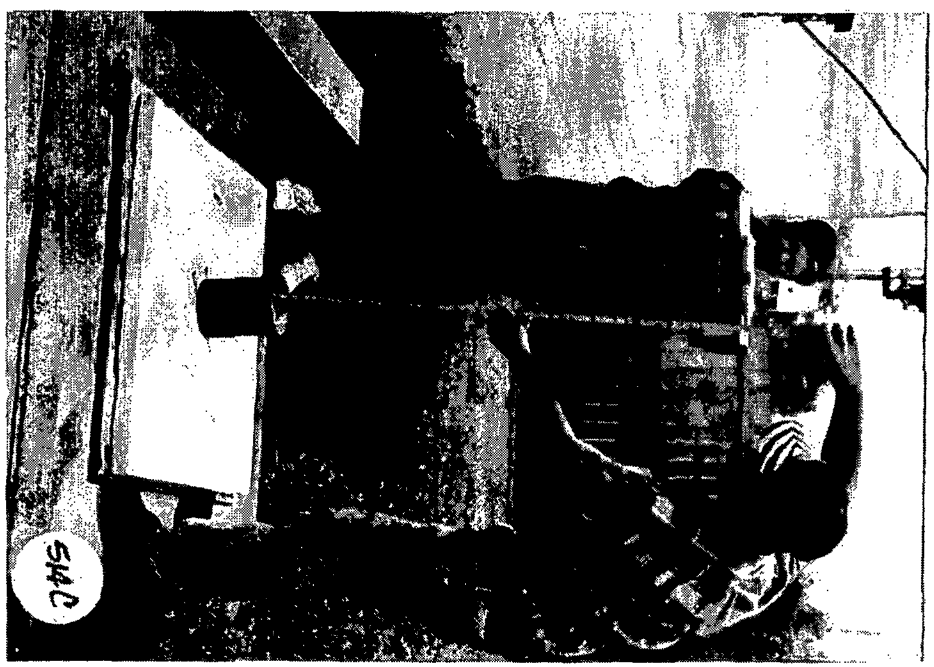

513 I


* 513 G

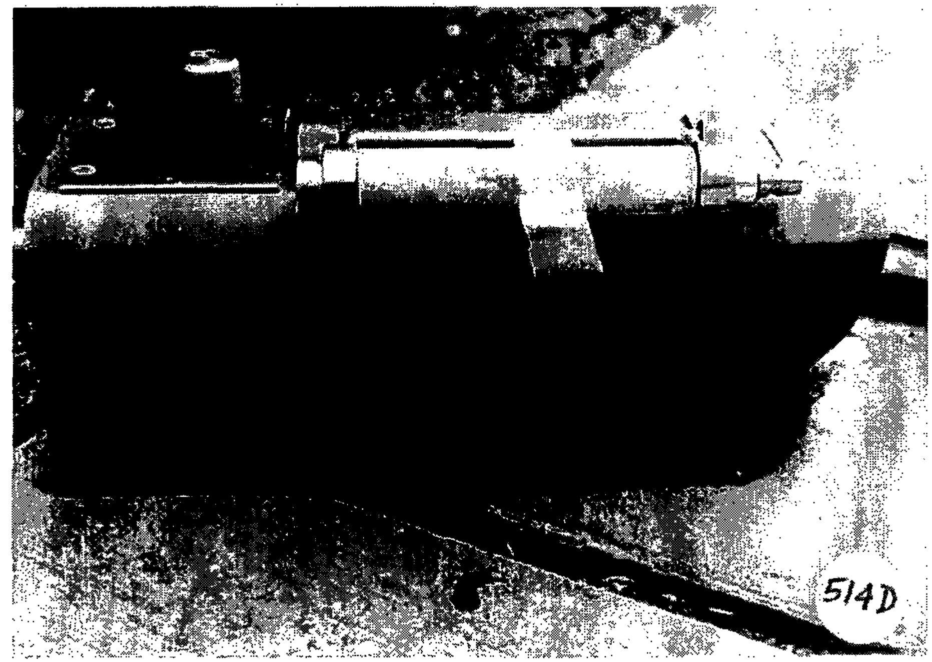

513 H

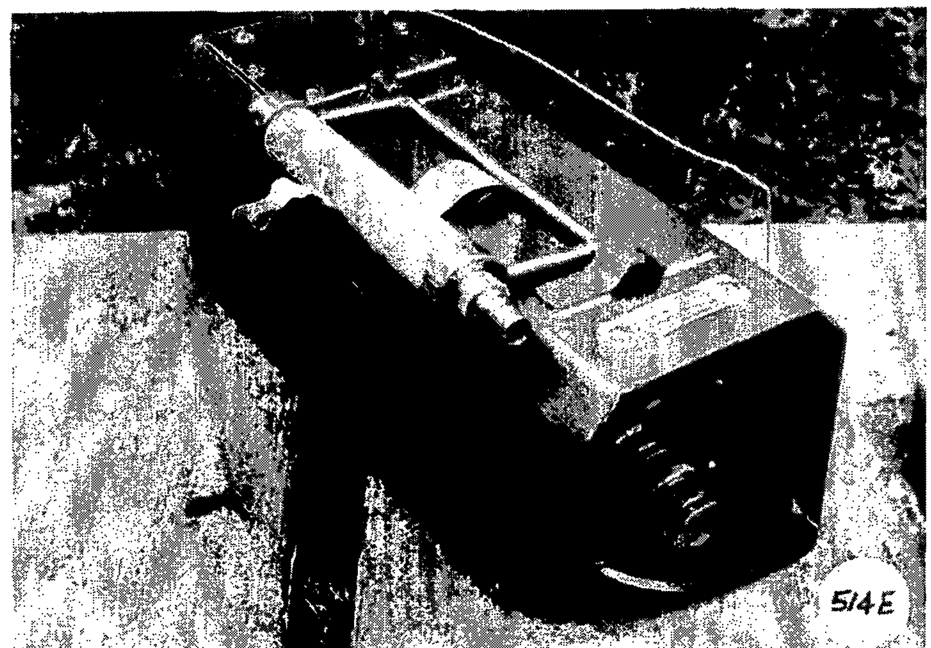

514F


514 A

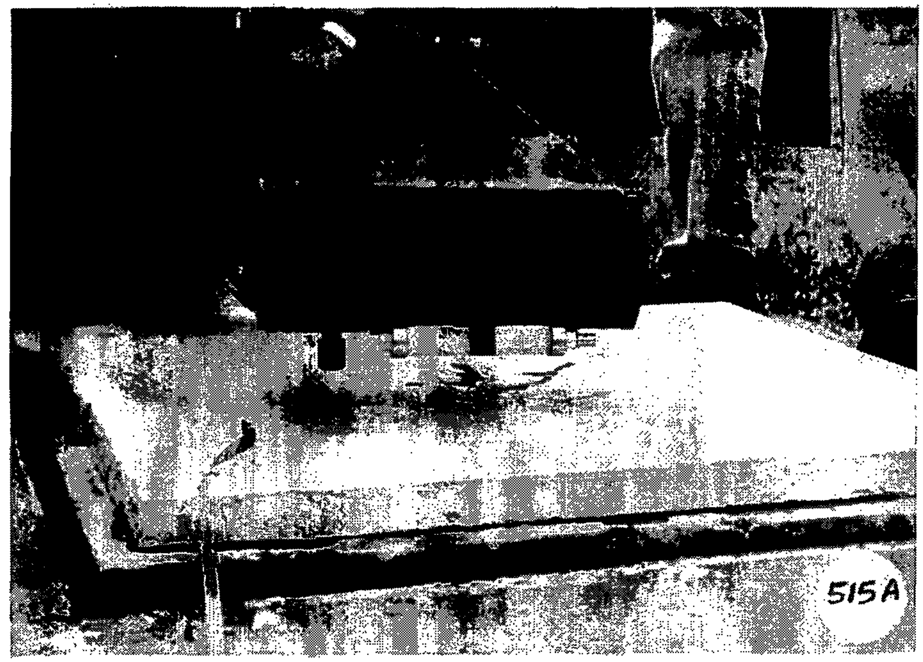

514C

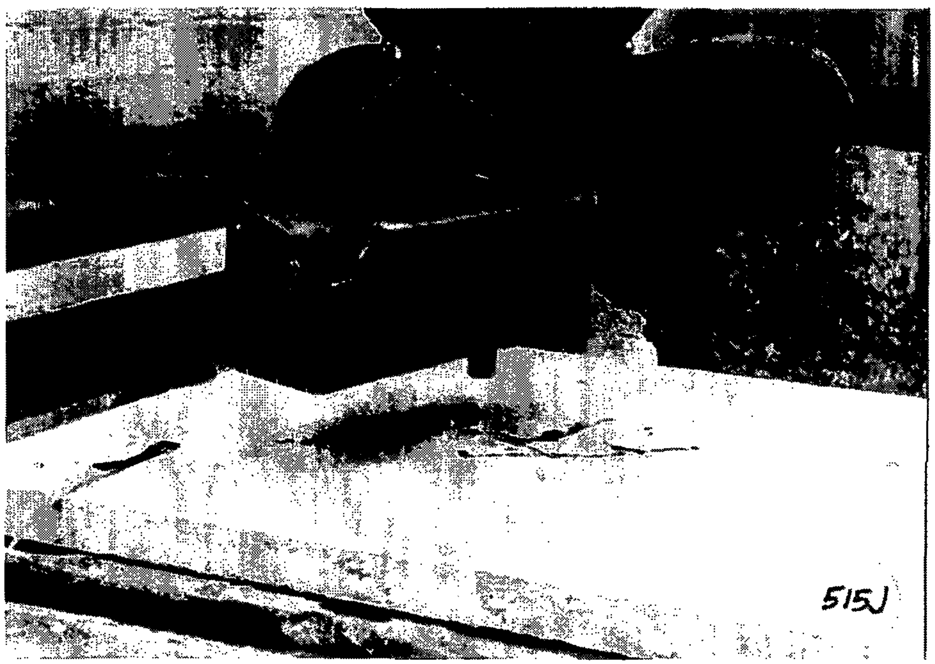

514 B

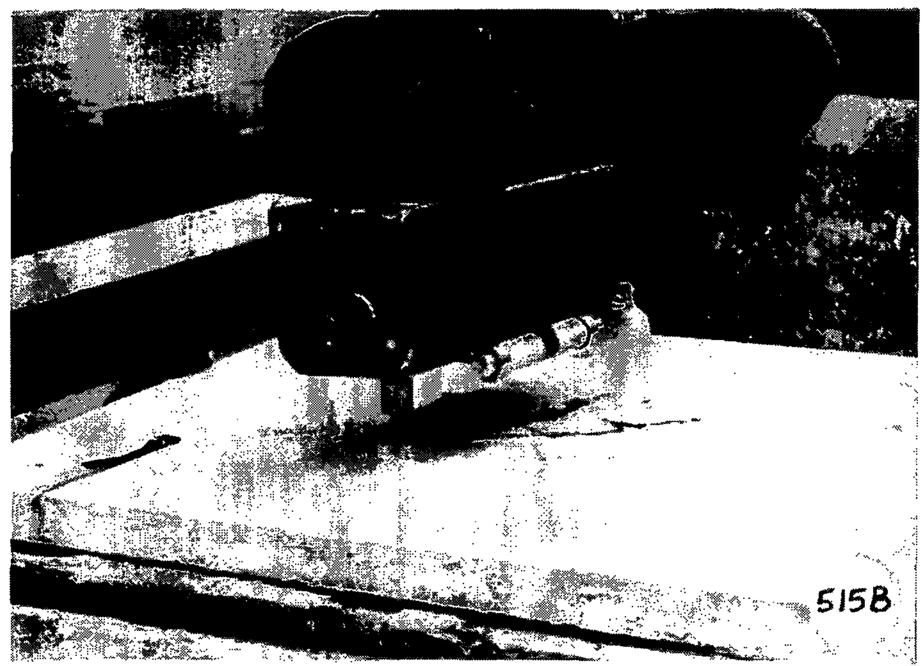

514D

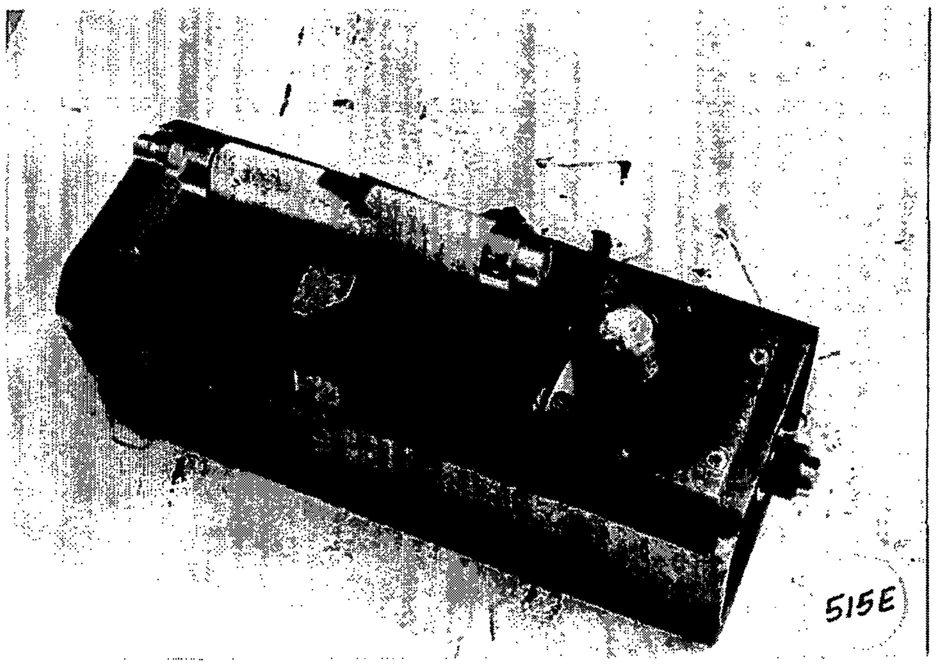

514E


515I


515A

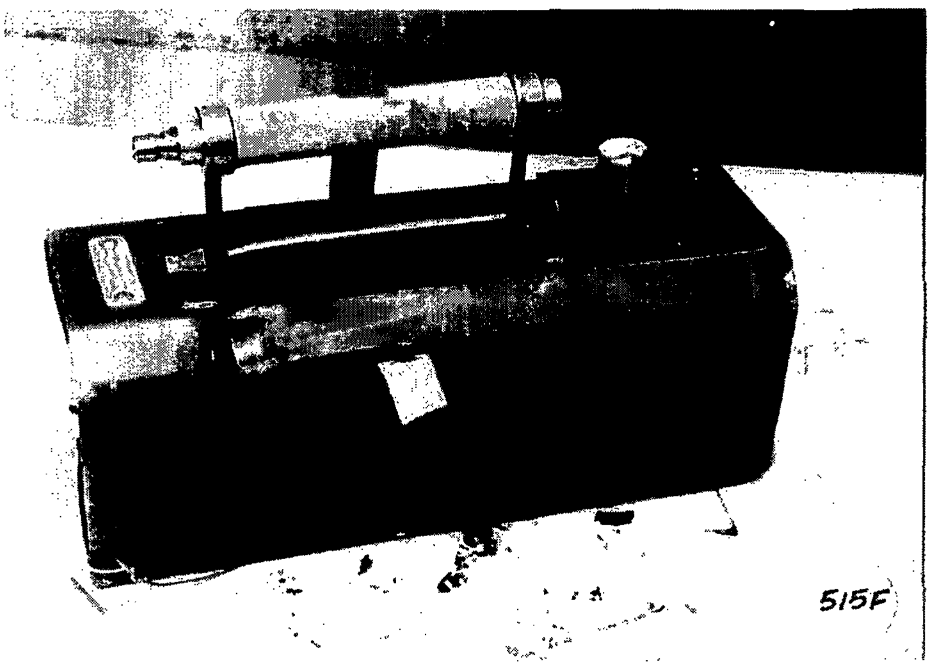

515J


515B

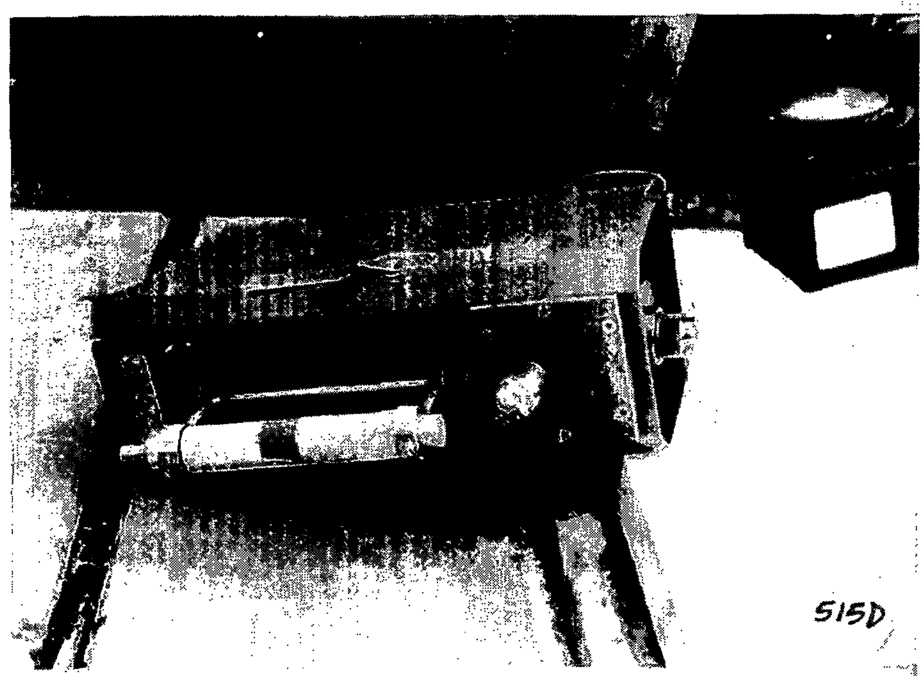


515E

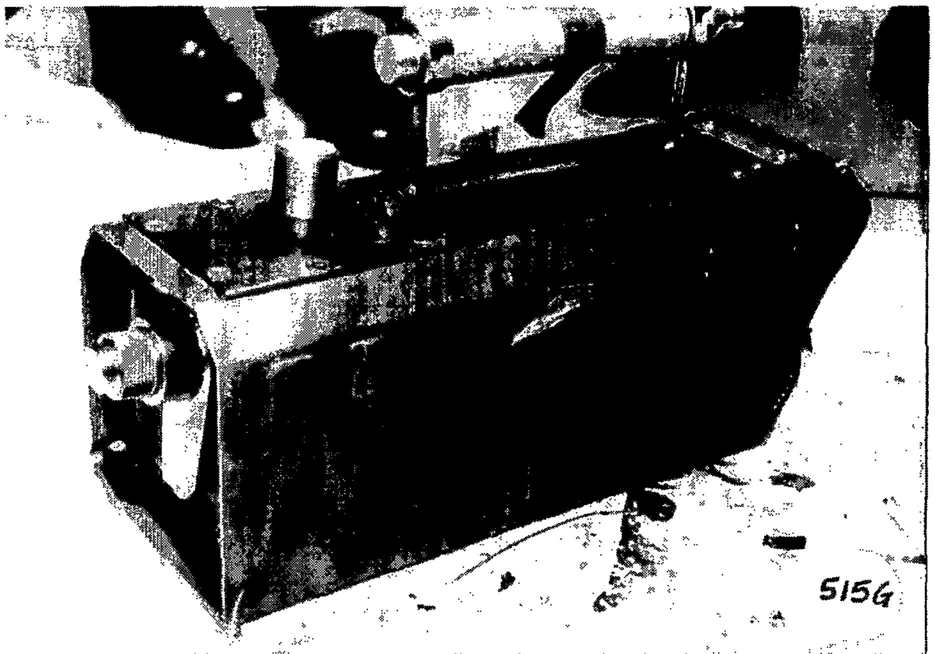
515C

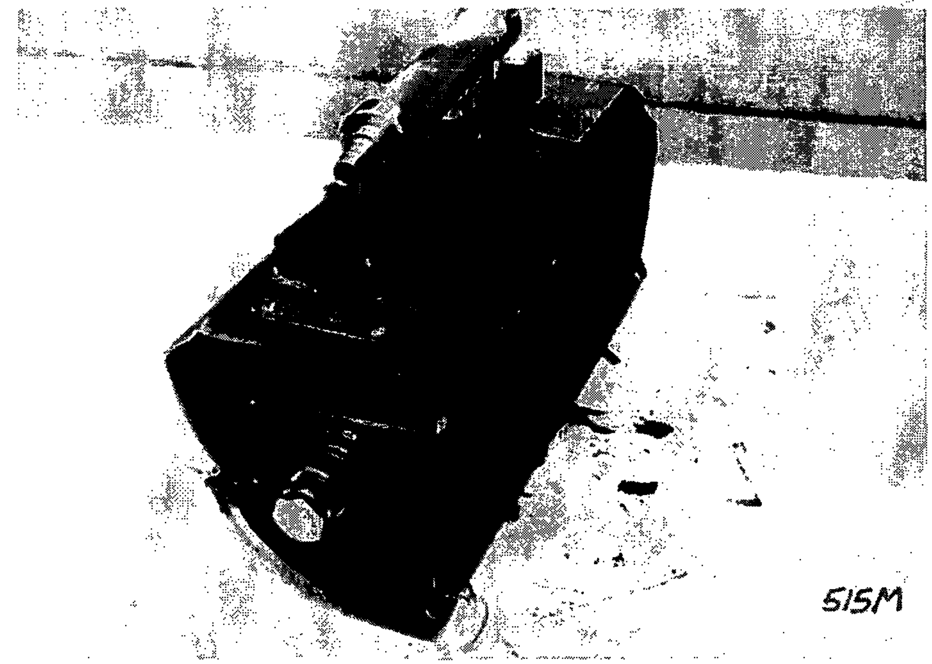


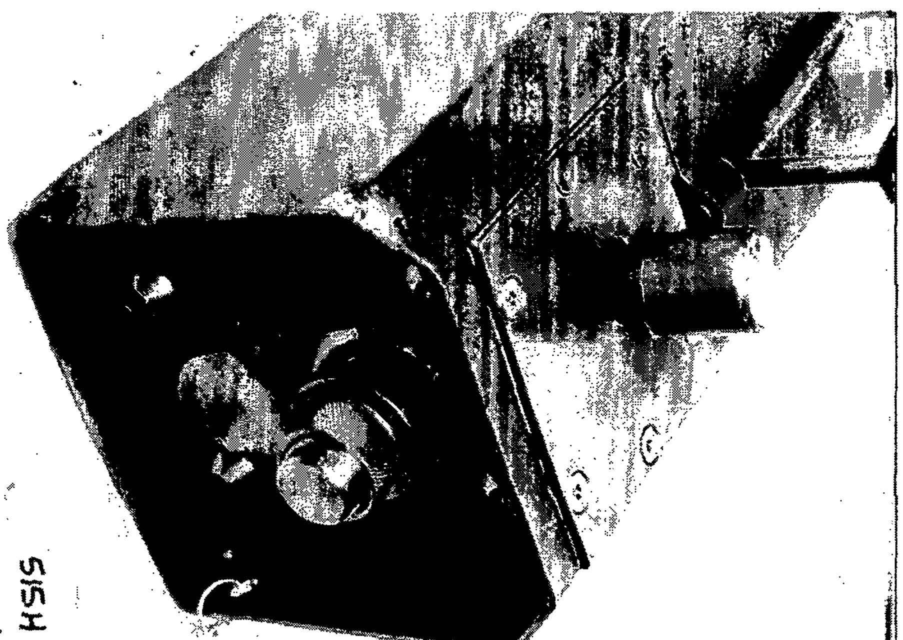
515F

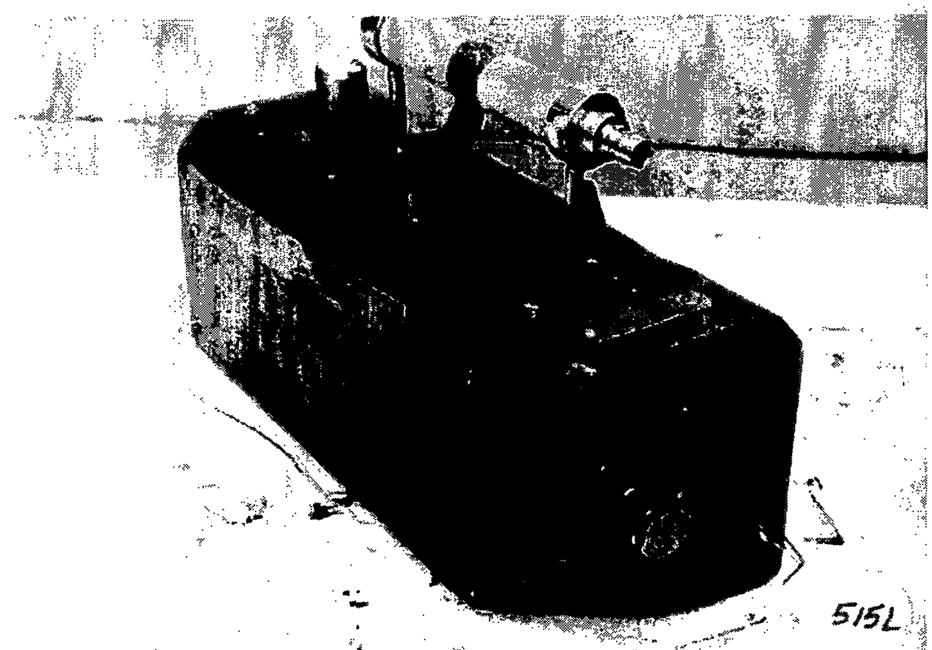


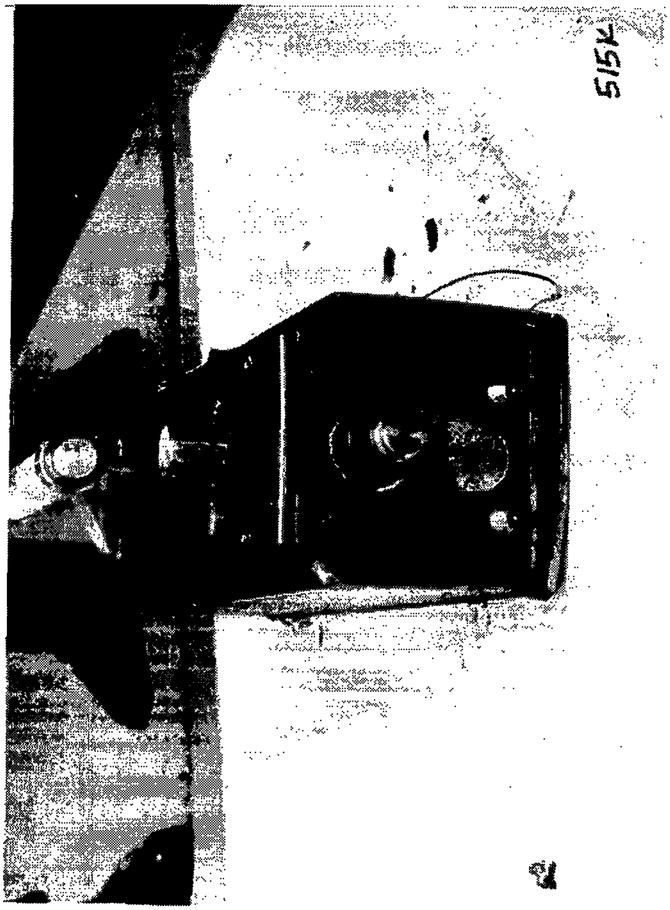
515D


515G

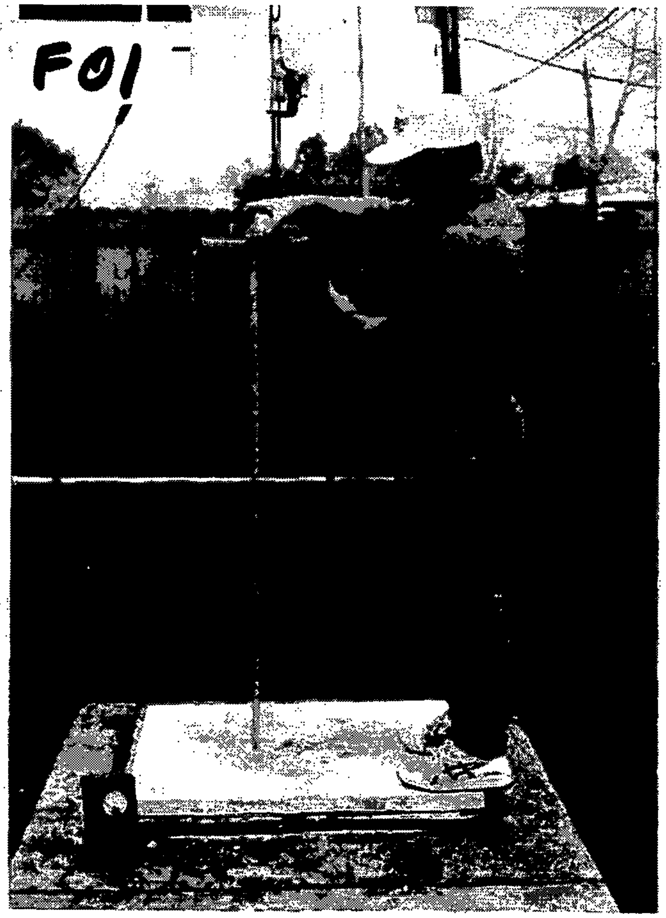

515M


515H


515L

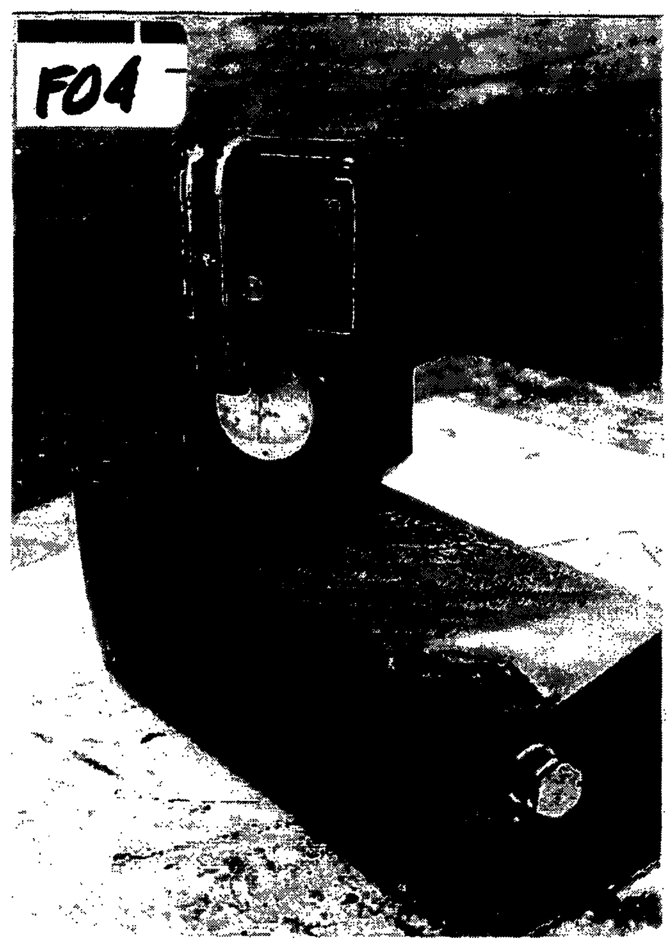

515K


515N


F01


F02

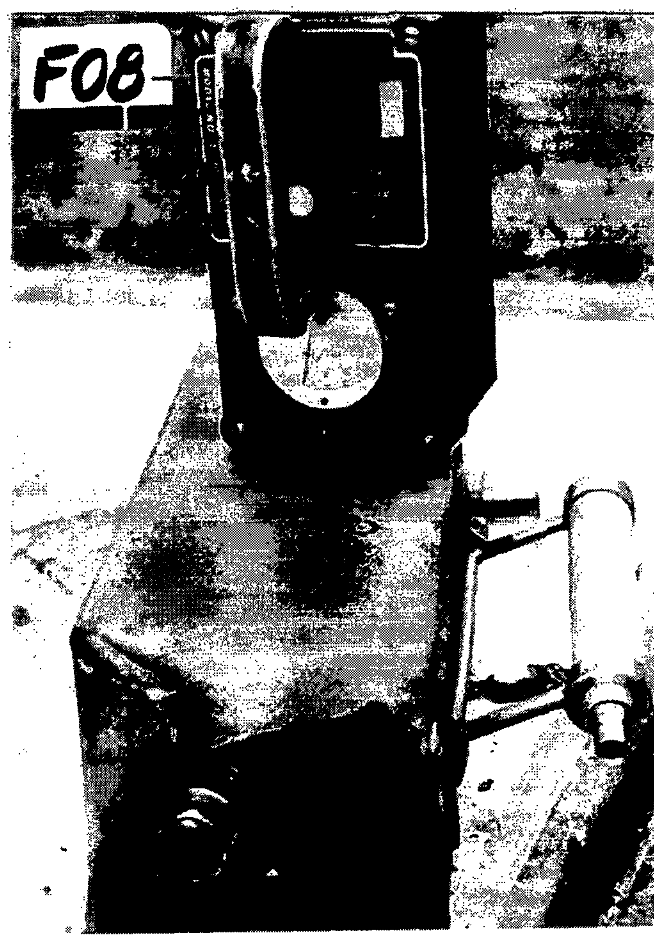

F03
FROUSSE
#4


F04


F05-

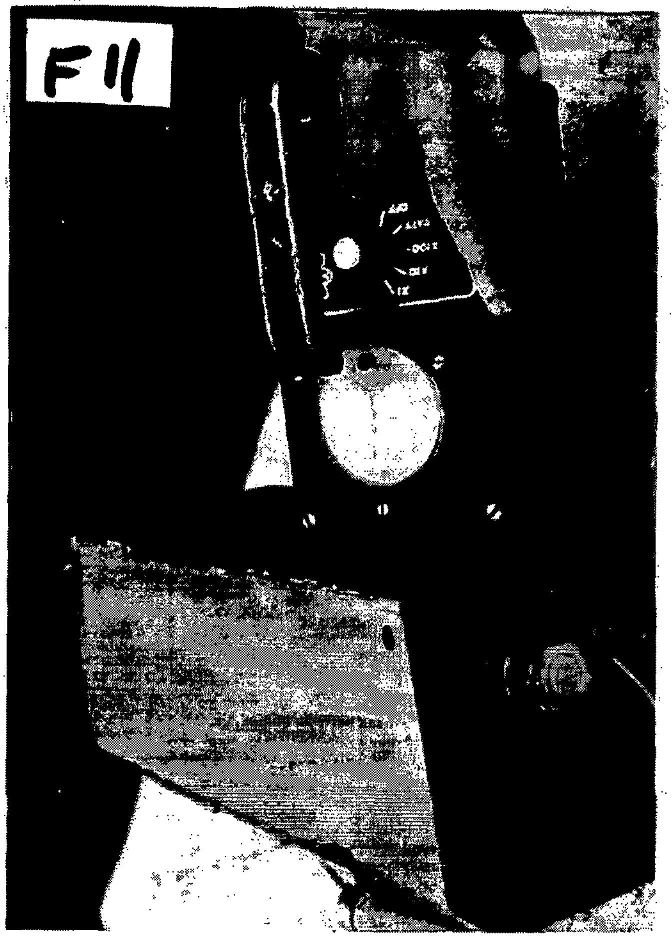

F06-


F07-


F08-


F09-

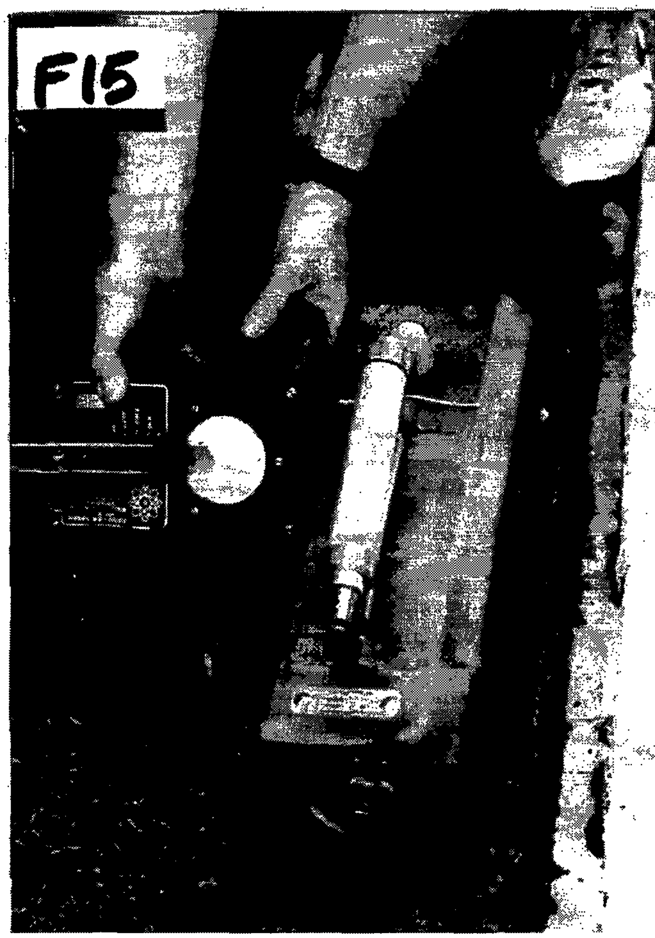

F10-


F11


F12


F13

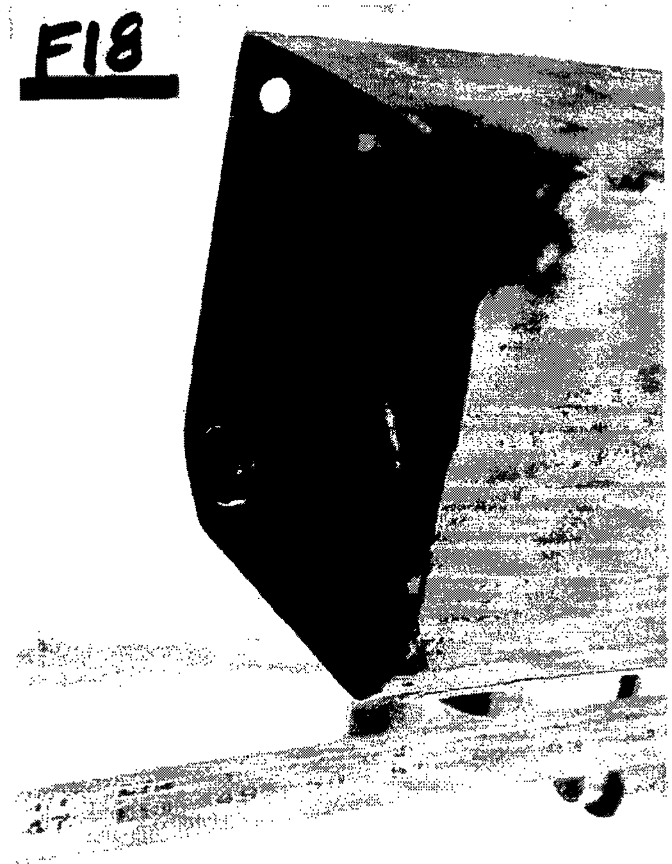

F14


F15

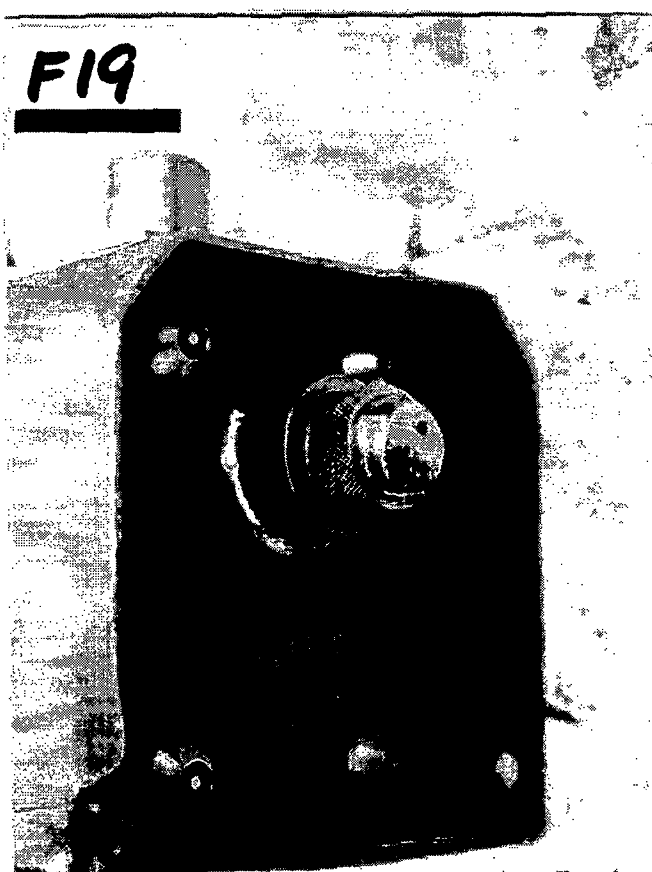

F16

F17

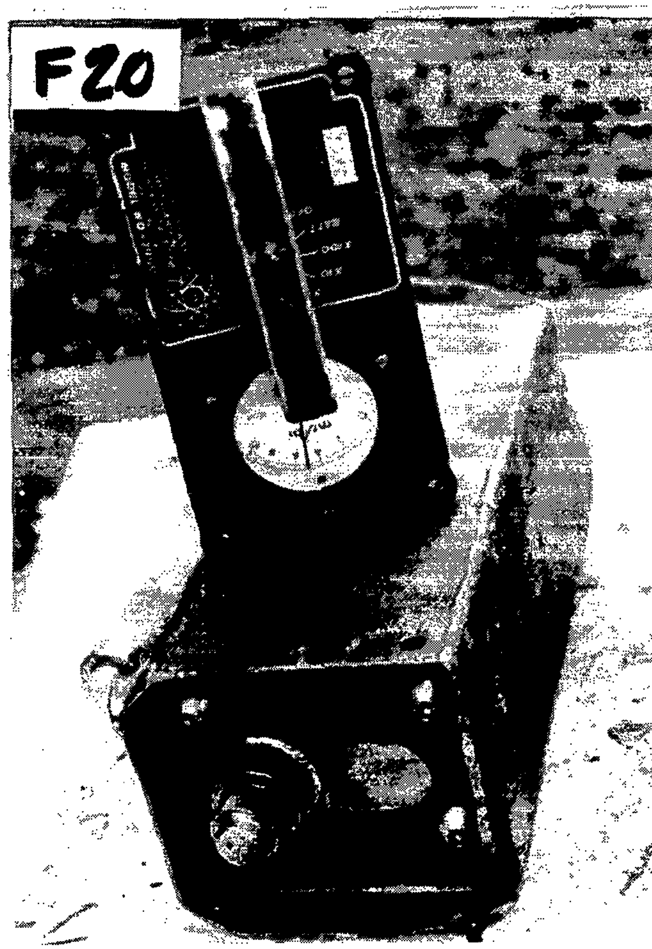
F18

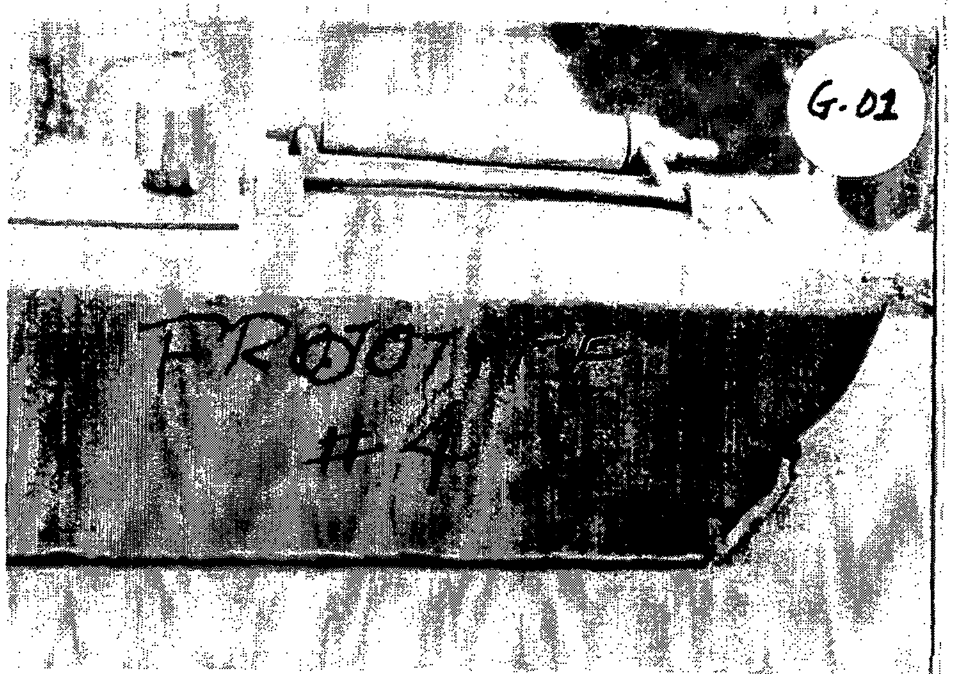


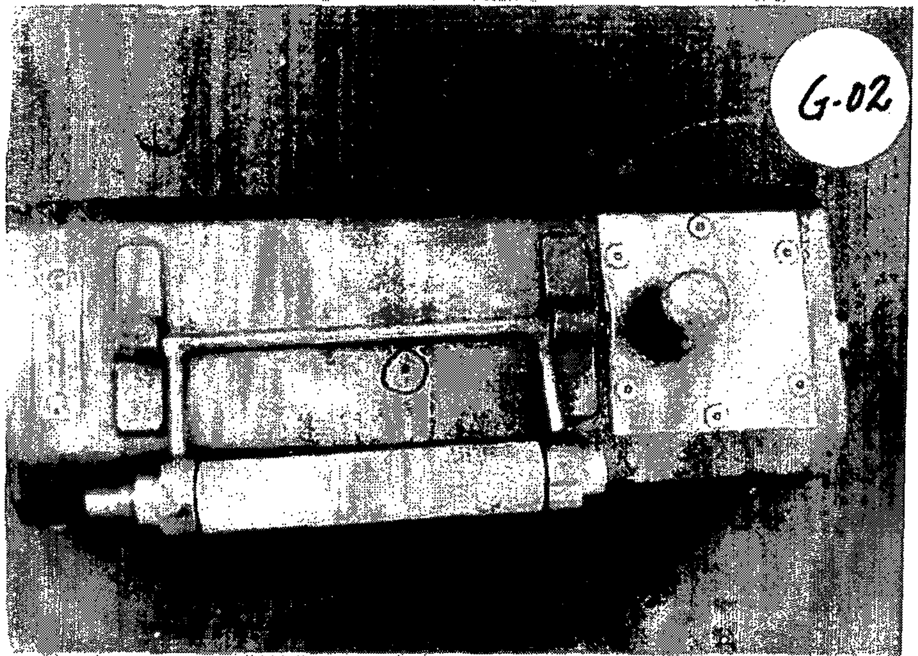
F19

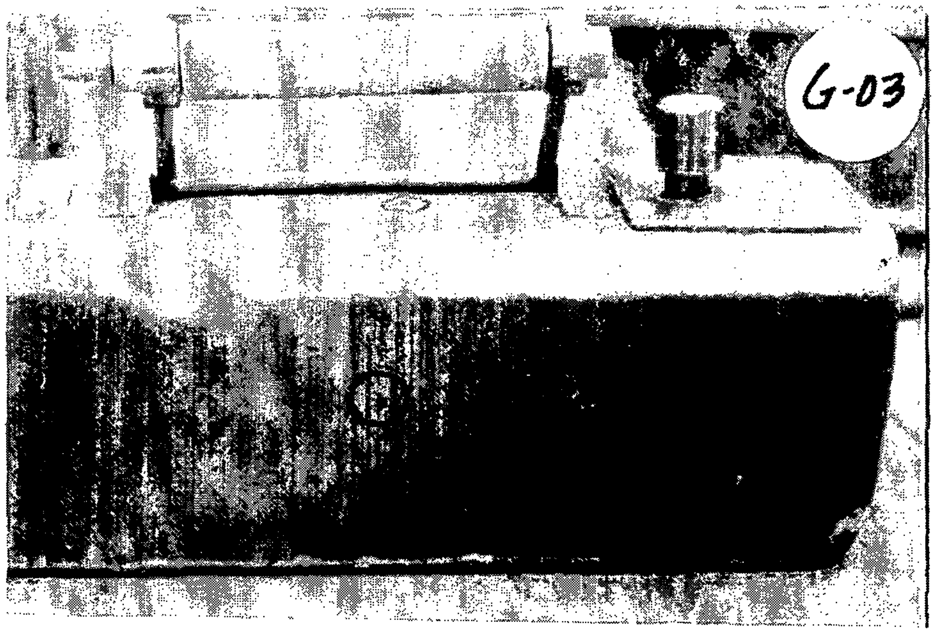


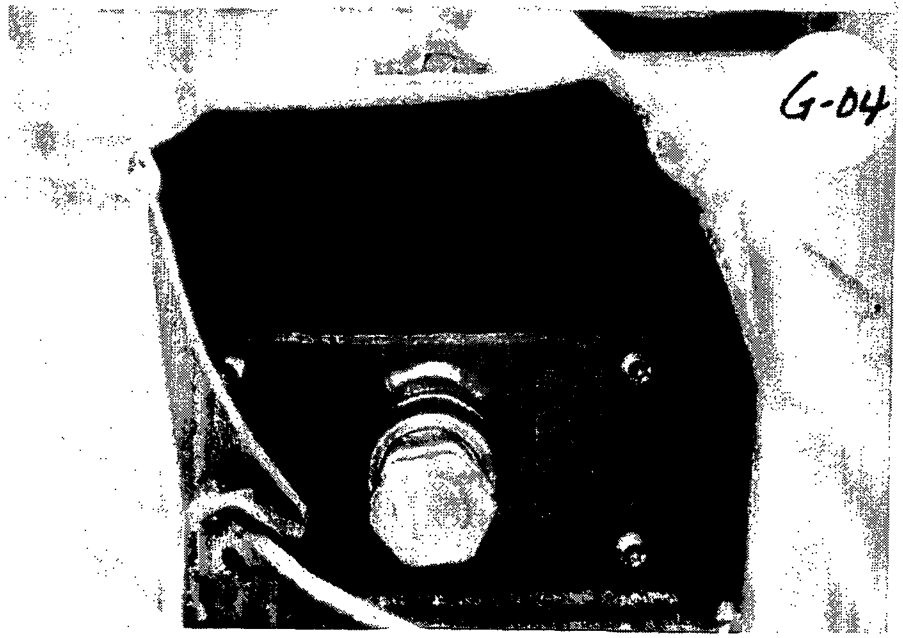
F20

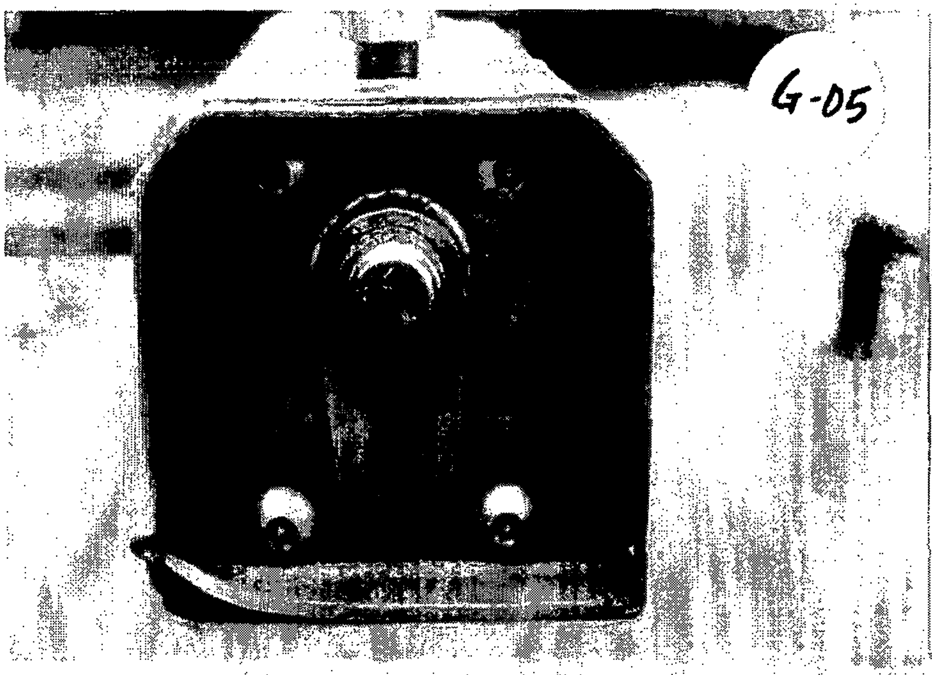


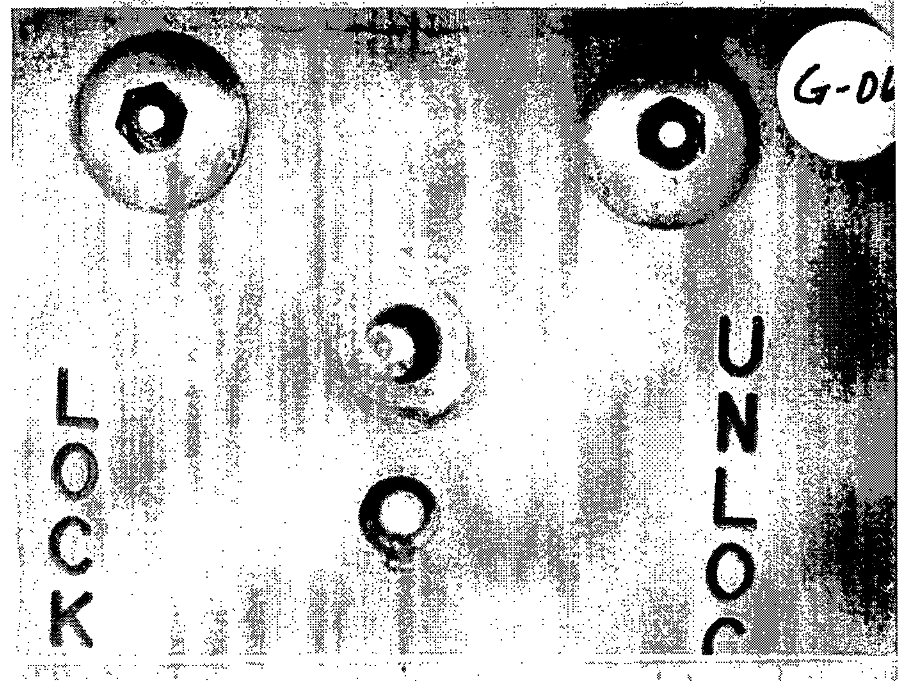

G-01
PROTOTYPE
#4

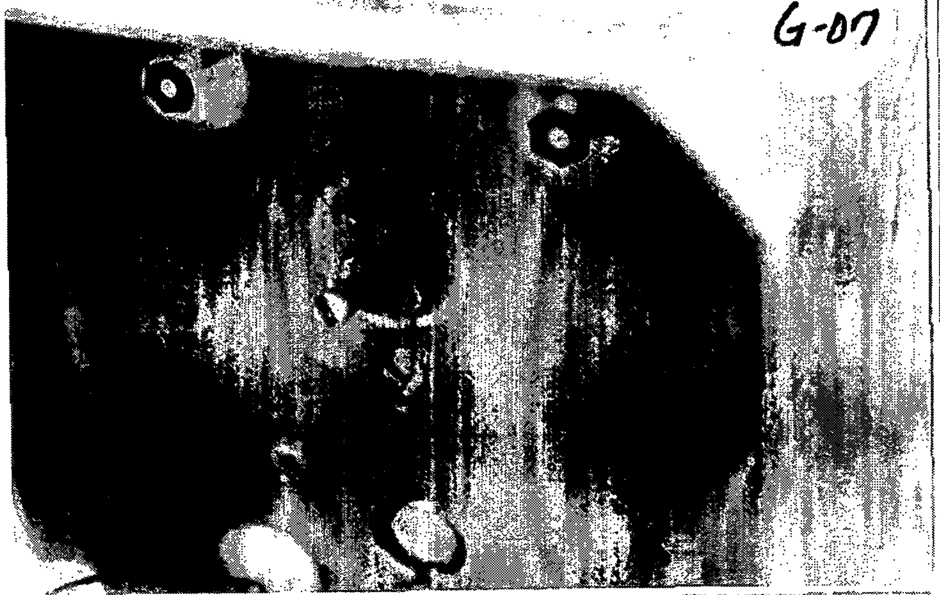

G-02

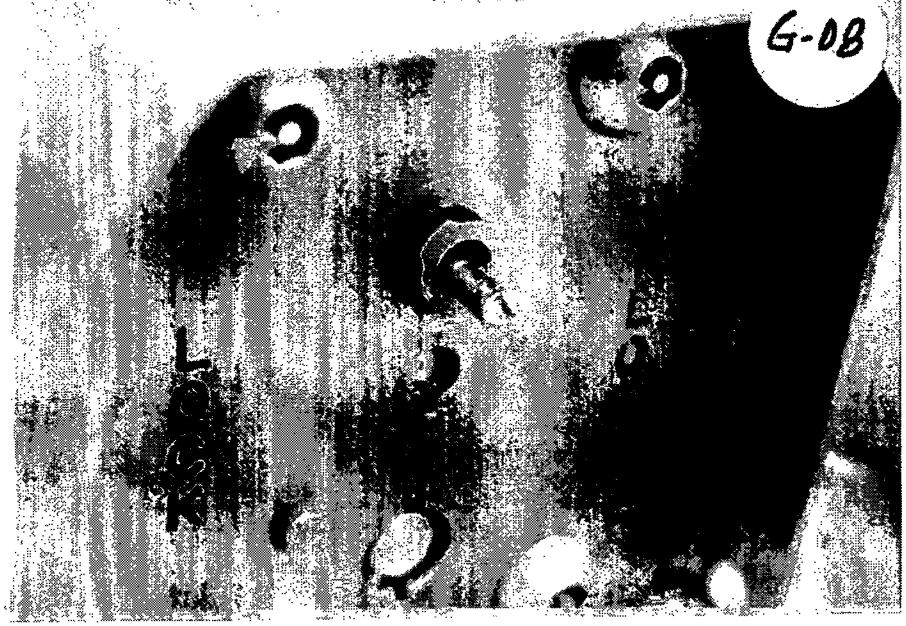

G-03


G-04


G-05


G-06
LOCK
UNLOC

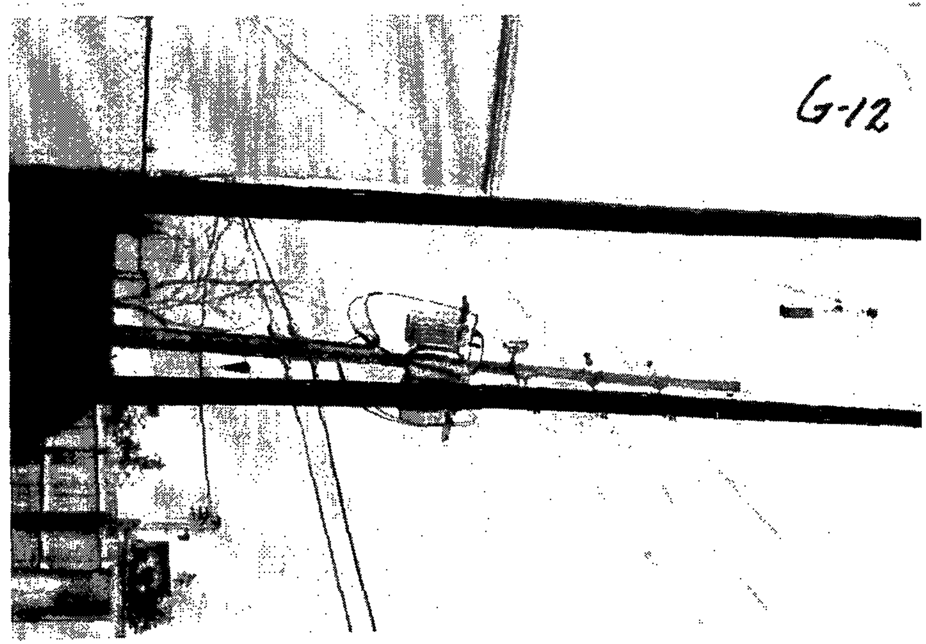

G-07

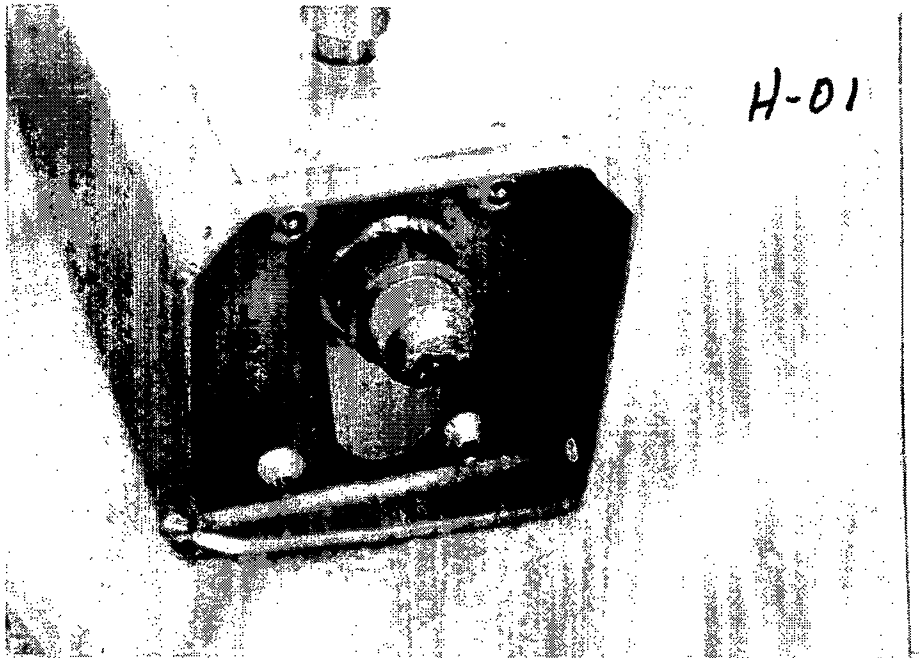
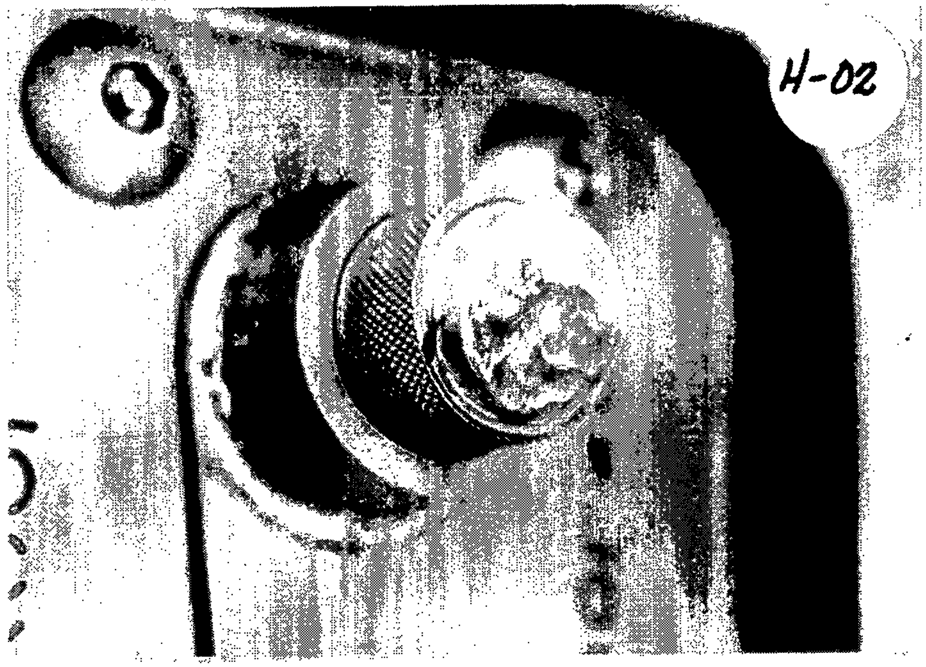

G-08
LOCK

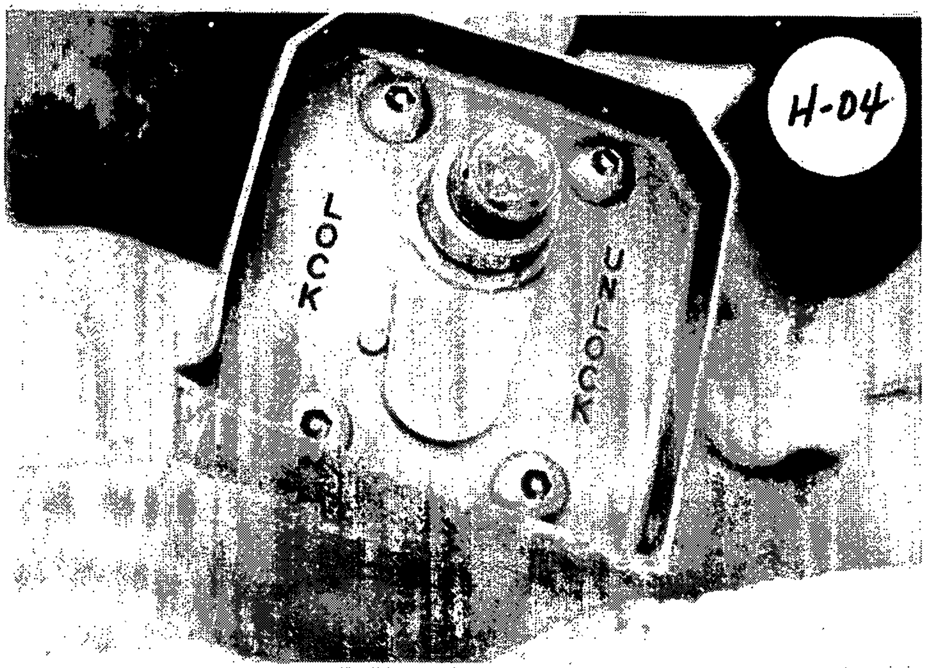

G-09

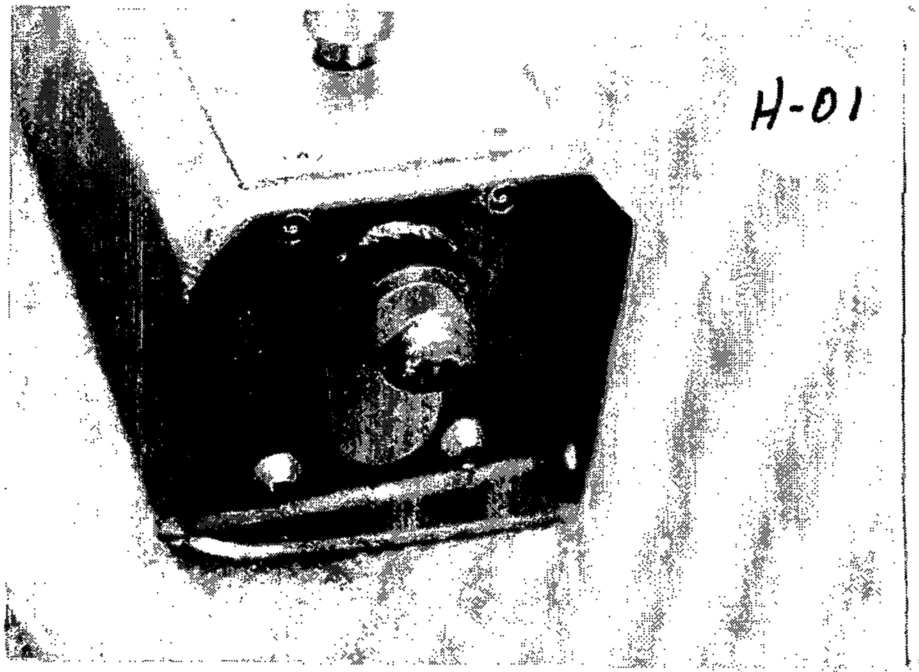

G-10

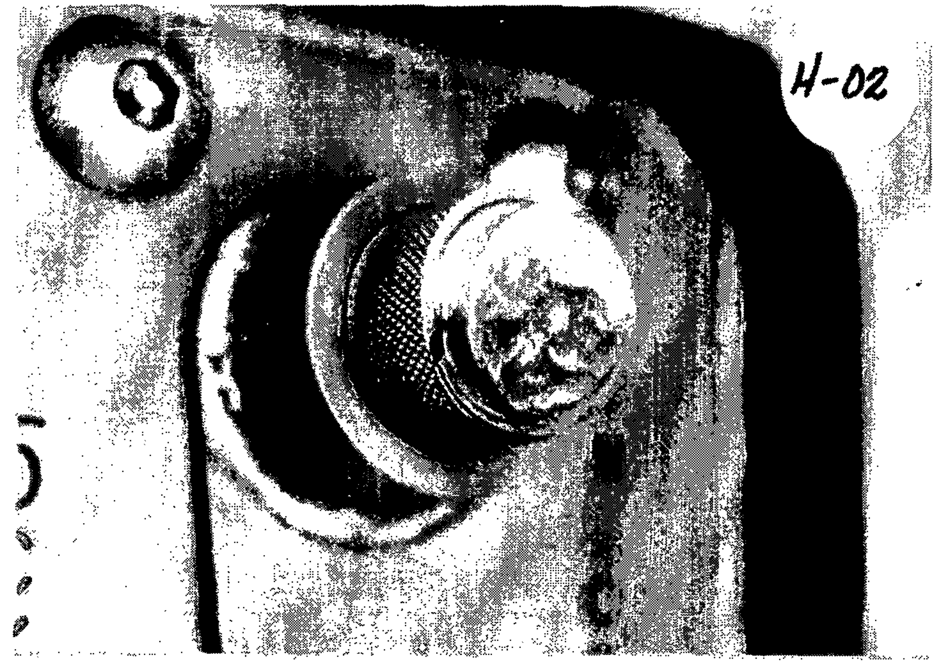

G-11


G-12

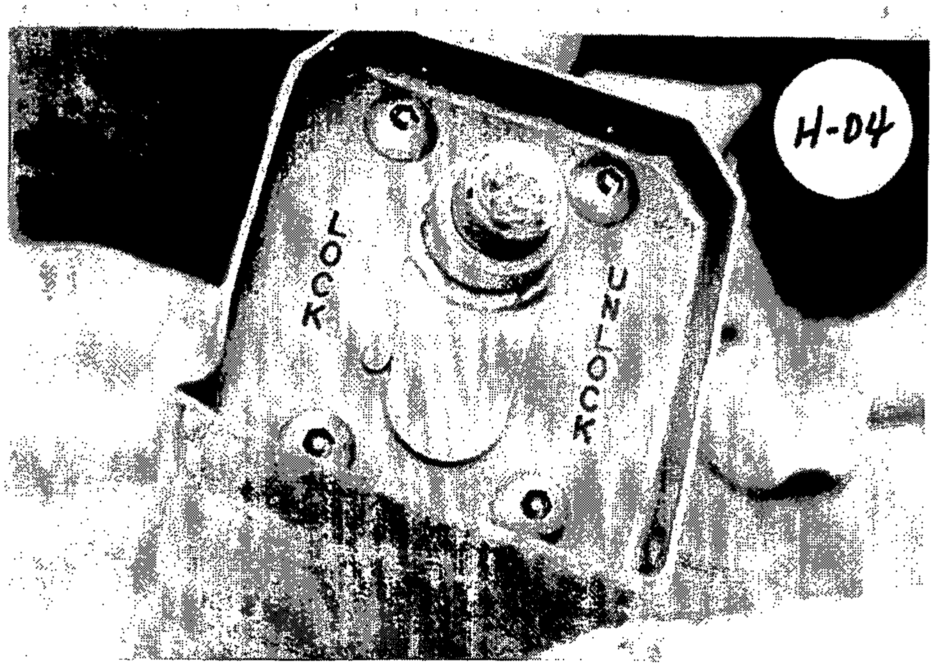

H-01

H-02

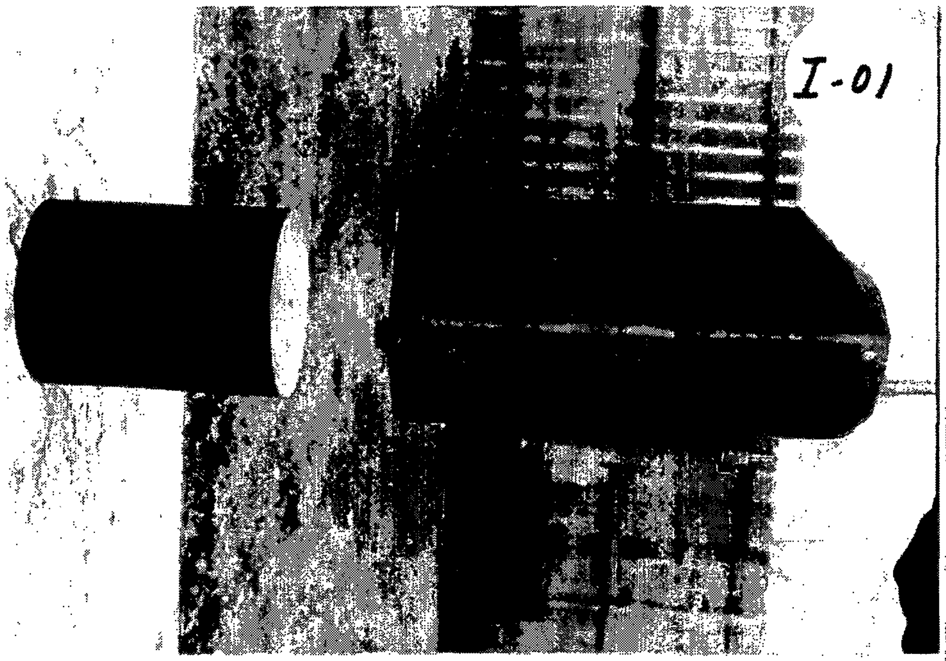
H-03

H-04
LOCK
UNLOCK

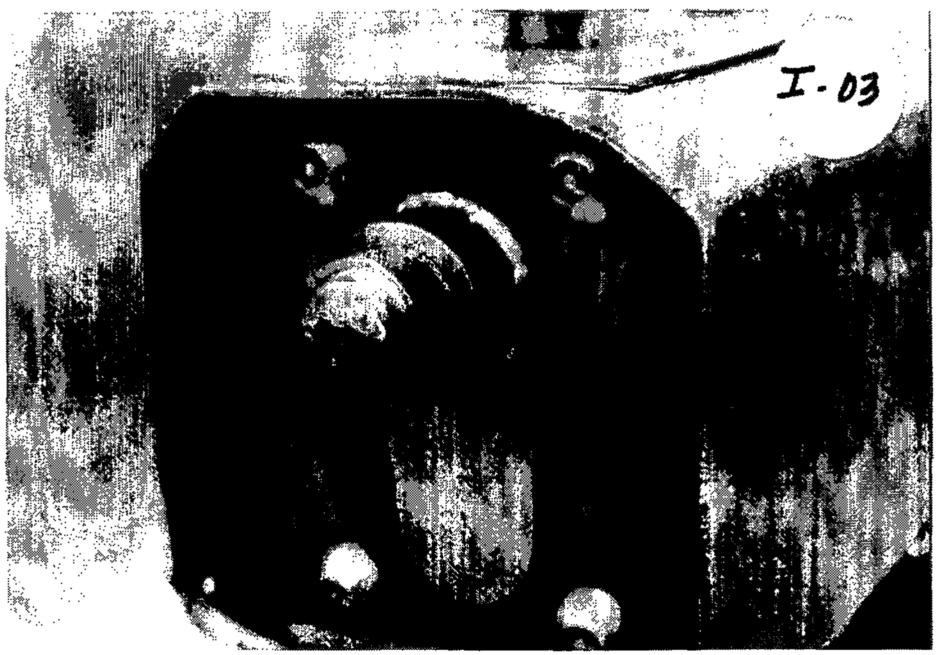
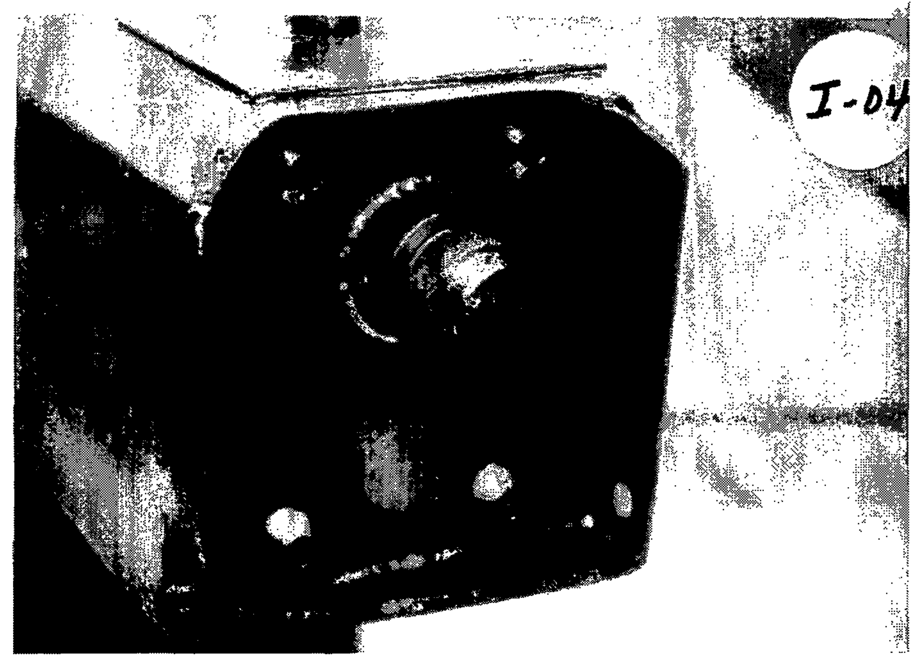

H-01

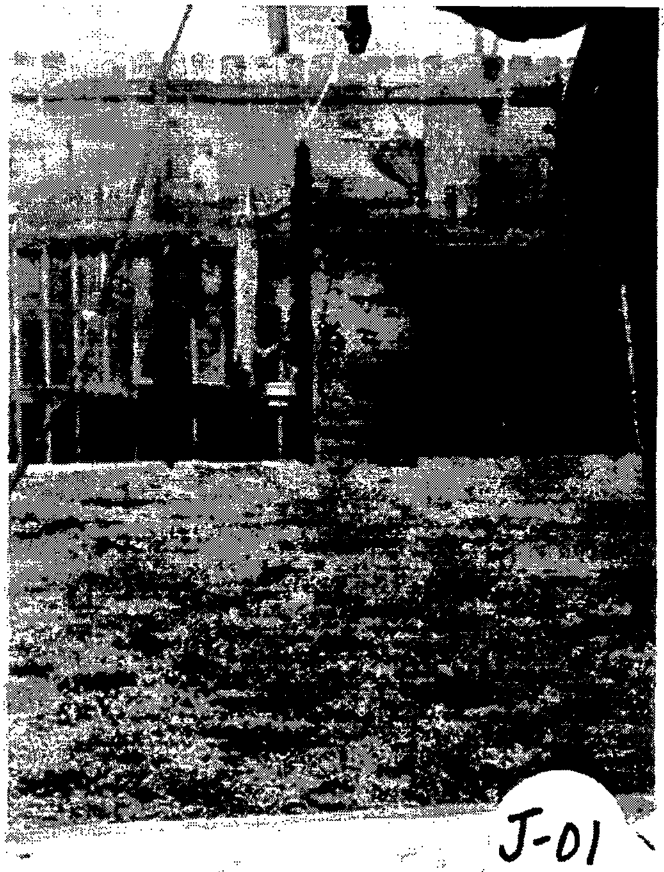

H-02

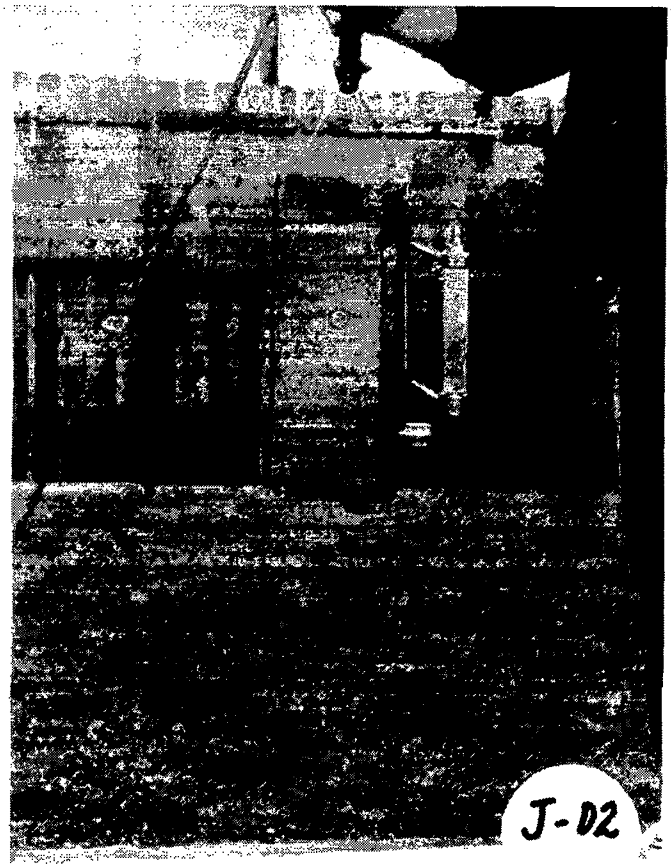

H-03

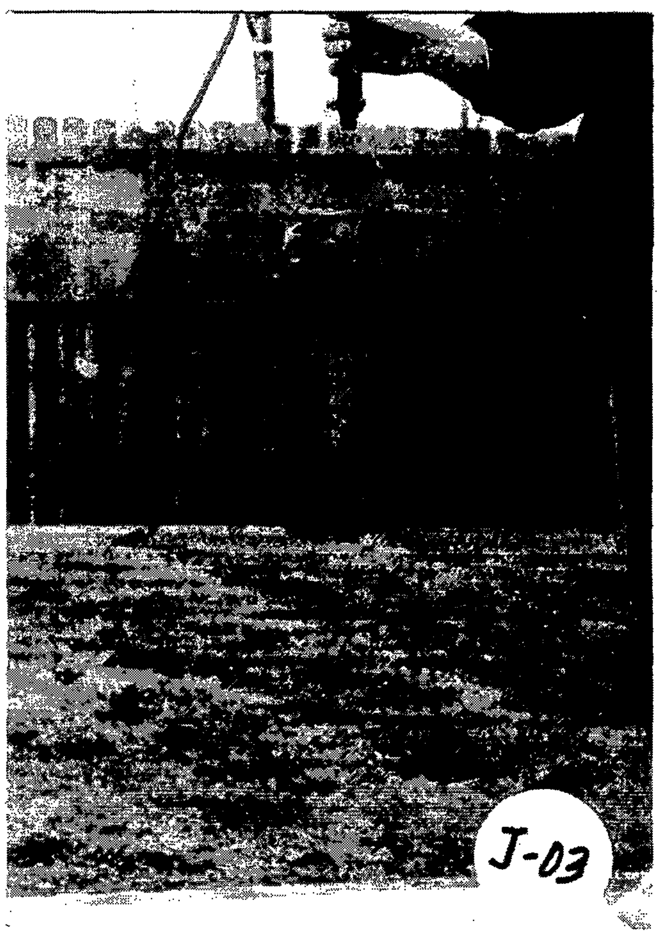

H-04
LOCK
UNLOCK

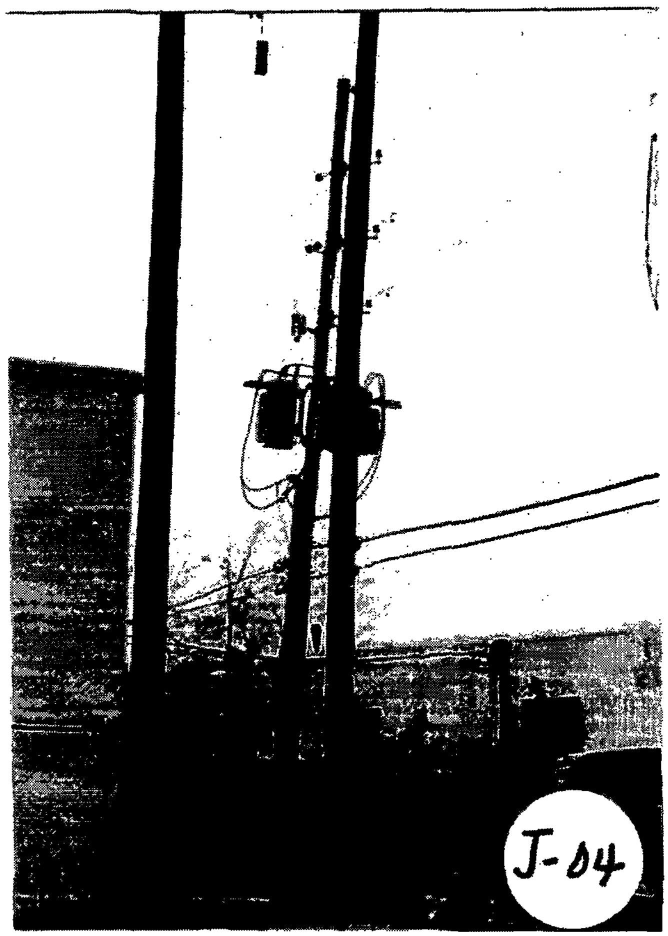

I-01

I-02

I-03

I-04

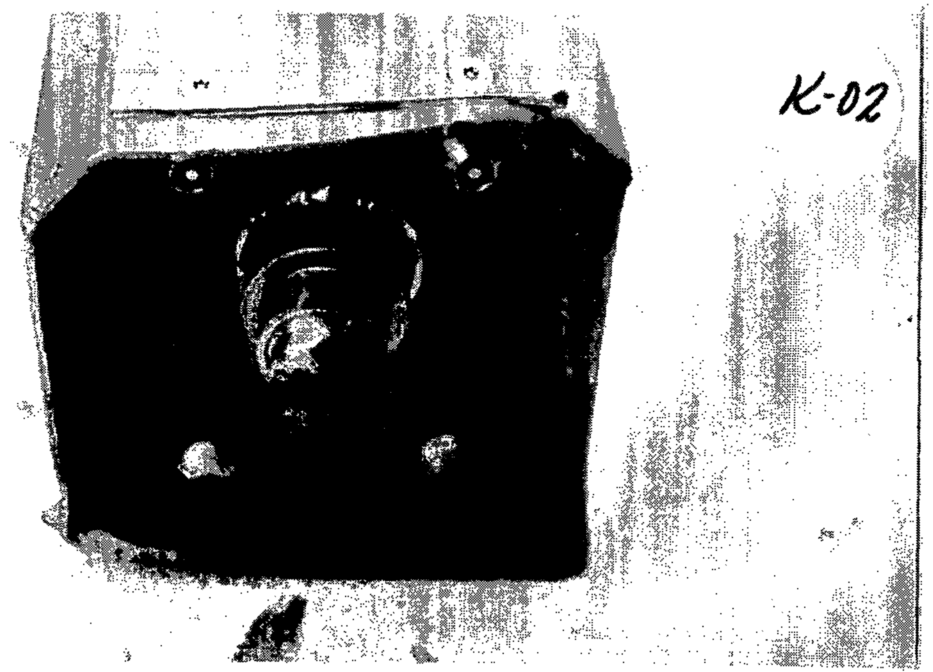

J-01

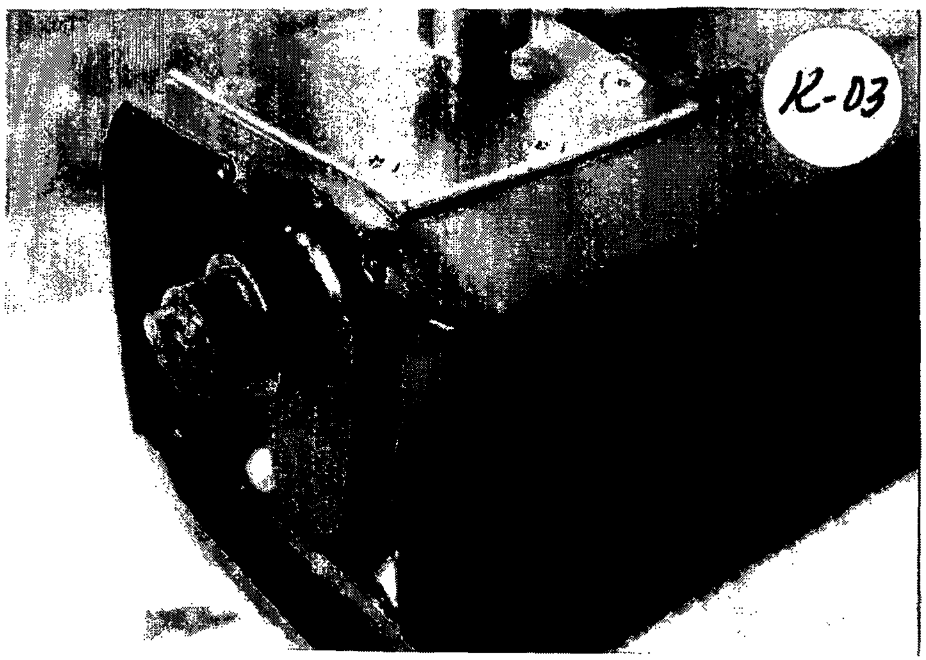

J-02

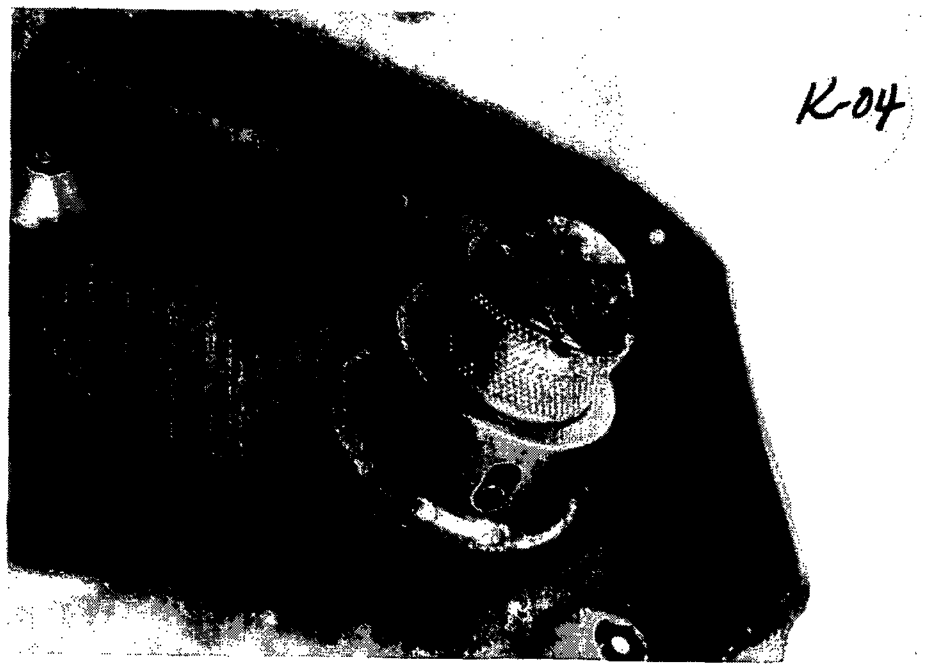

J-03

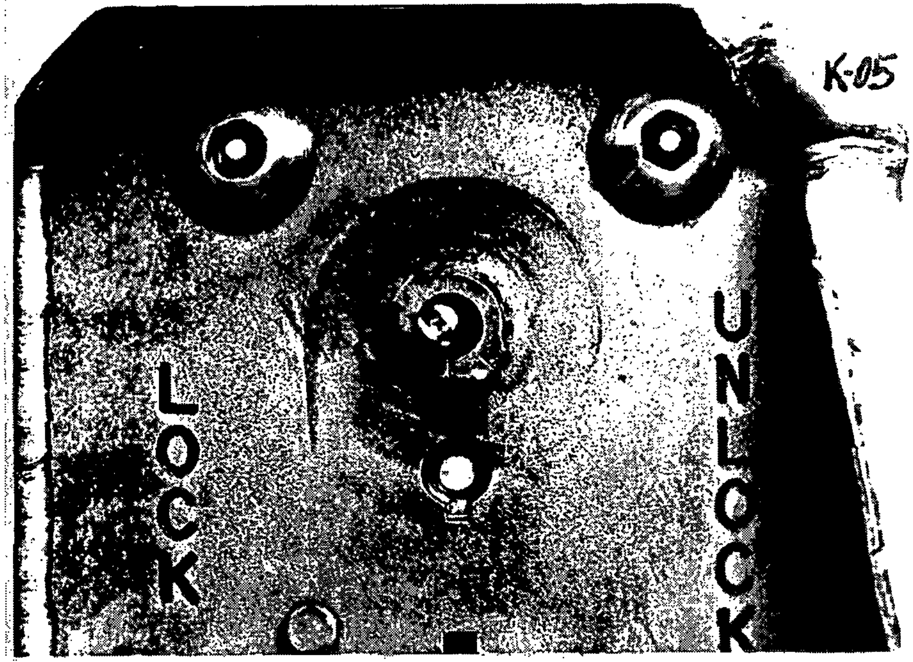

J-04

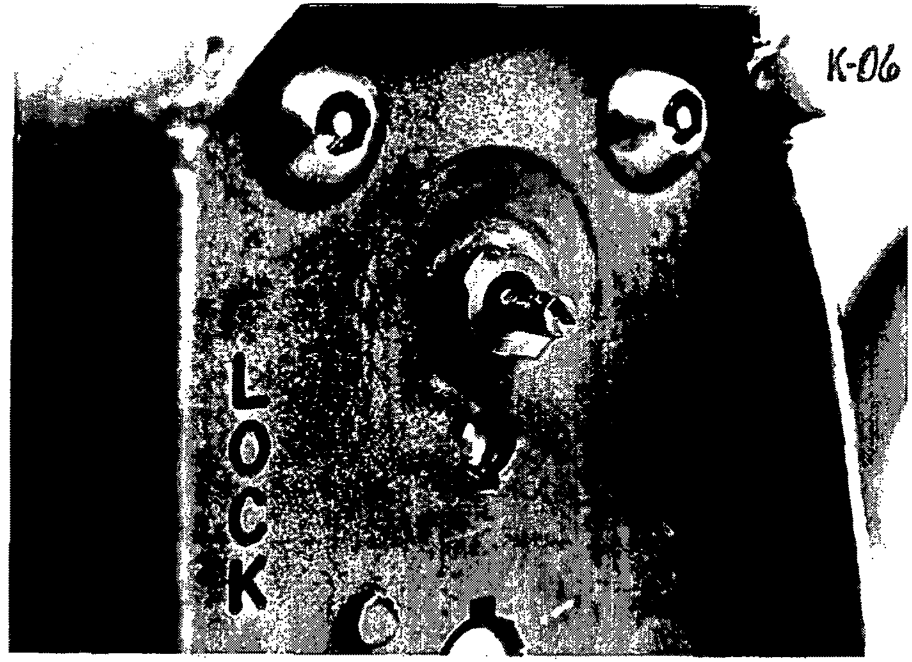

K-01

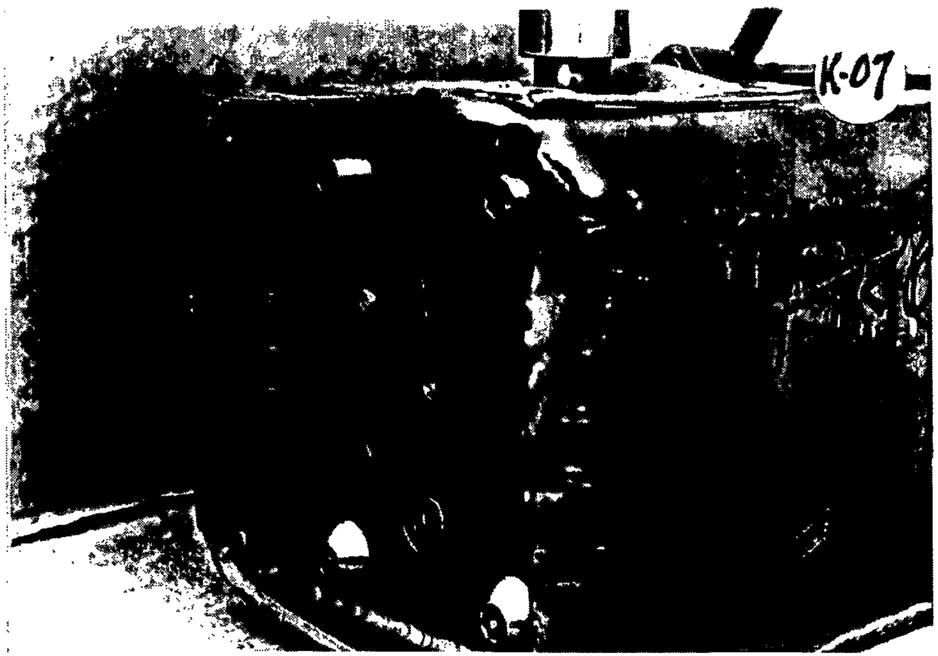

K-02

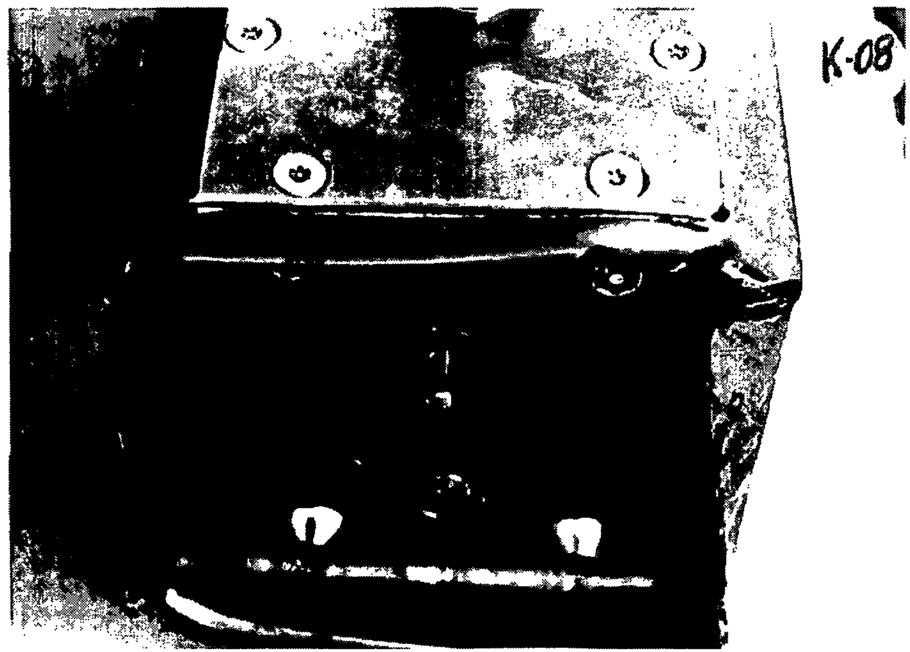

K-03


K-04


K-05
LOCK
UNLOCK


K-06
LOCK


K-07


K-08

Appendix 9.3

2011 Normal Conditions Test Report

TEST REPORT (GENERAL)
Source Production & Equipment Company, Inc.
113 Teal Street, St. Rose, LA 70087

Test Date: 02/03/11	Test Name: Normal Conditions 4' Drop & Penetration Test
Product Tested: SPEC-150, serial number 331	Component Tested:SPEC-150 Part #: 150000

Type of Test / Purpose of Test: To verify that the SPEC-150 package would meet normal conditions test requirements, and that the radiation levels at the surface and at one meter meet the normal conditions radiation limits.

Test Procedure or Reference Document (Work Instruction, ANSI standard, regulation, etc):
We selected a SPEC-150 that was no longer in use for testing, and loaded it with a 131.29 curie G-60 source. The QA Inspector surveyed the SPEC-150 on the surface and at one meter before the tests, and marked the location with the highest reading on the SPEC-150 and on the worksheets and survey reports. The penetration test was performed, followed by the drop tests. With engineering's concurrence, we selected the same points of impact as the original tests. The target for the penetration test was the safety plug. The test was performed in accordance with EG16. The 4' drop tests were performed next in accordance with EG36. The points of impact were flat on the bottom plate, bottom right corner on the outlet end, bottom left corner on the lock end, top left corner on the outlet end and flat on the lock cap. As a result of the tests there was no effect on the integrity or shielding capability of the package. The tests did not result in loss of radioactive contents from the package, significantly increased radiation levels nor reduce the effectiveness of the package.

Test Results: Photo Frame(s)______to________

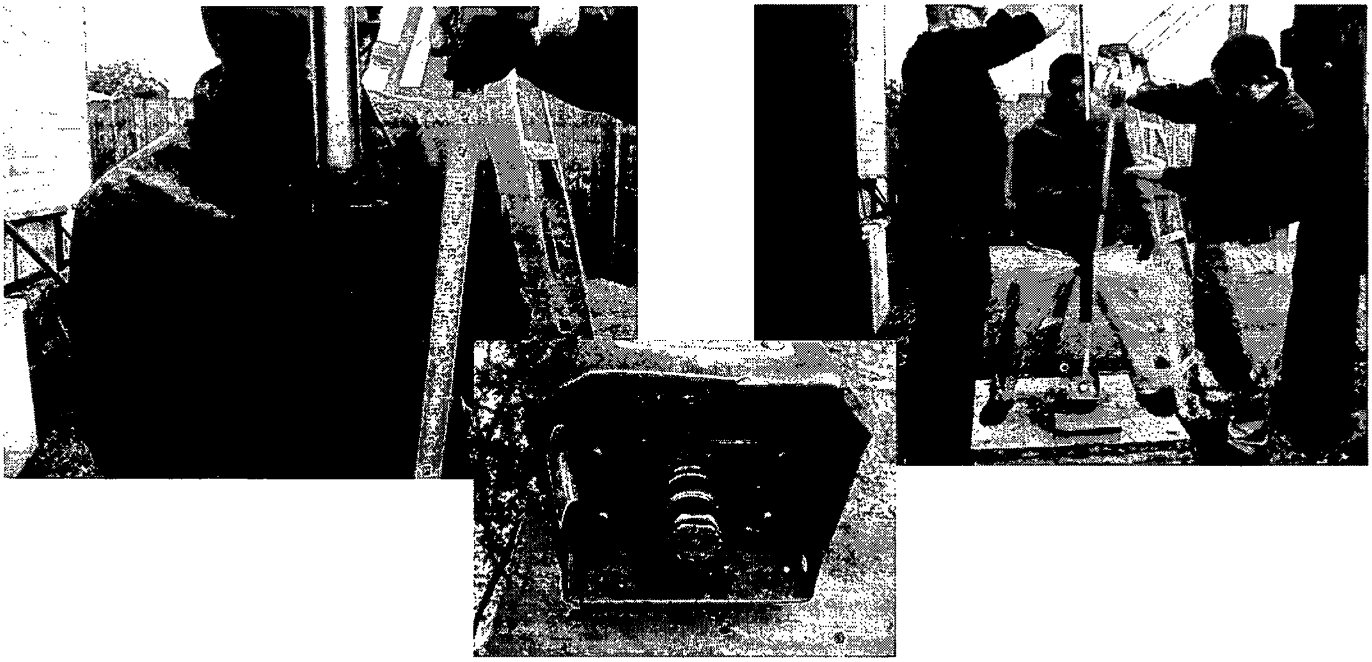
The highest percentage of change in surface readings was 3%, well below the 20% limit. The tests did not result in loss of radioactive contents from the package, significantly increased radiation levels nor reduce the effectiveness of the package. The highest corrected and extrapolated radiation level at the surface was 118 mR/hr, and 1.14 mR/hr at one meter, which is well below the regulatory limit of 200 ~~mR~~/hr and 10 ~~mR~~/hr respectively.
mRem KC mrem KC

Comments and Recommendations: See photos, next page

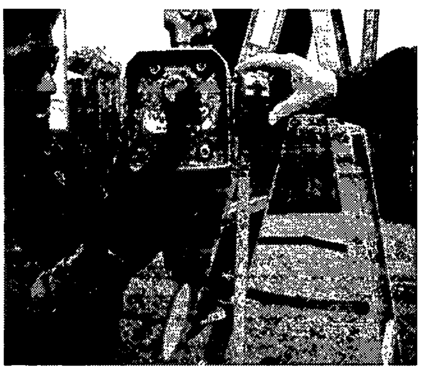
Test Performed By: Kevin Schehr Nathan Gorbach Tony Bustillo Kiet Phan Kelley Richardt KR	Witnessed By: Krissie Zambrano KZ

Review:

Evaluated, reviewed and approved by: K. Richardt (QA Mgr) on 3/4/11 date.



Penetration Test, top of pipe set at 40", package chocked in place to expose safety plug, safety plug after test.



1-flat on the bottom plate

2-bottom right corner on the outlet end

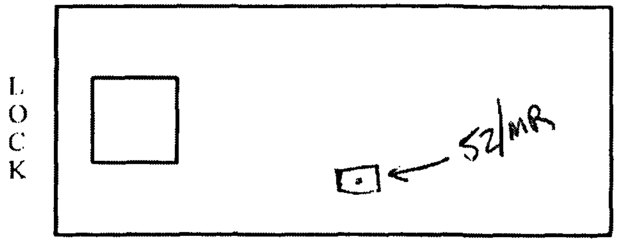
3-bottom left corner on the lock end

4-top left corner on the outlet end

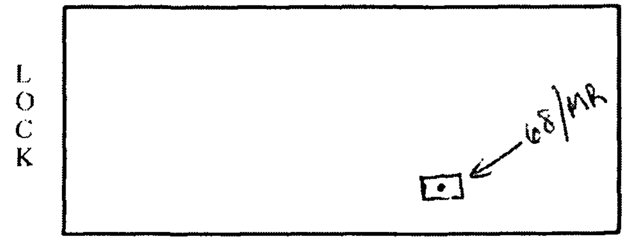
4' Drop Tests, orientation as captioned on pictures.

5-flat on the lock cap

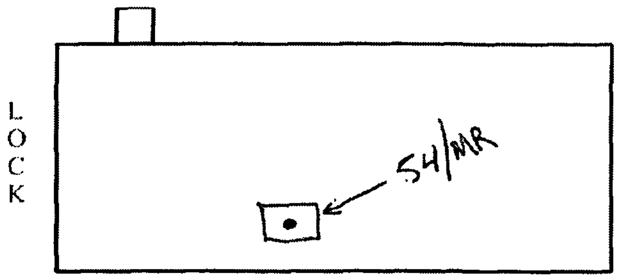
Normal Conditions Accident Testing, 02/03/11, SPEC-150 sn 331, G-60 Source sn 331									
131.29	Curies								
1.14250895	REF								
mR/hr	Before		After						
	Actual	Actual	Actual	Actual	Change		Percent of Change		
	Surface	One Meter	Surface	One Meter	Surface	One Meter	Surface	One Meter	
Left	86	1	84	1	-2	0	-2%	0%	
Right	58	0.8	60	0.8	2	0	3%	0%	
Bottom	68	0.8	70	0.8	2	0	3%	0%	
Top	52	0.6	50	0.6	-2	0	-4%	0%	
Lock	82	1	72	0.6	-10	-0.4	-12%	-40%	
Outlet	58	0.6	58	0.6	0	0	0%	0%	
mR/hr	Actual		Extrapolated		Distance Corrected				
	Surface	One Meter	Surface	One Meter	Surface	DCF 1.2			
Left	86	1	98	1.14	118				
Right	58	0.8	66	0.91	80				
Bottom	68	0.8	78	0.91	93				
Top	52	0.6	59	0.69	71				
Lock	82	1	94	1.14	112				
Outlet	58	0.6	66	0.69	80				
Above results converted from mR/hr to mrem/hr by multiplying by 0.87, to mSv/h by dividing by 100									
	mSv/h		mrem/h						
	Surface	One Meter	Surface	One Meter					
Left	103	1.0	1.0	0.010					
Right	69	0.8	0.7	0.008					
Bottom	81	0.8	0.8	0.008					
Top	62	0.6	0.6	0.006					
Lock	98	1.0	1.0	0.010					
Outlet	69	0.6	0.7	0.006					



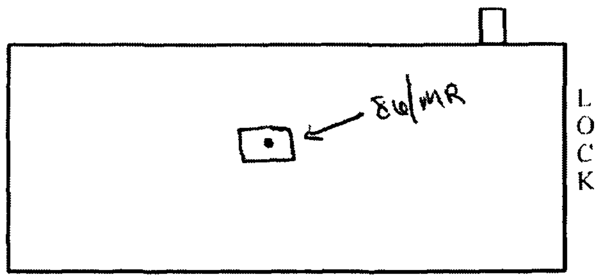
Top Side



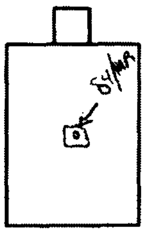
Bottom



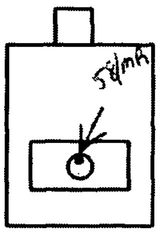
Right Side



Left Side



Lock End



Outlet End

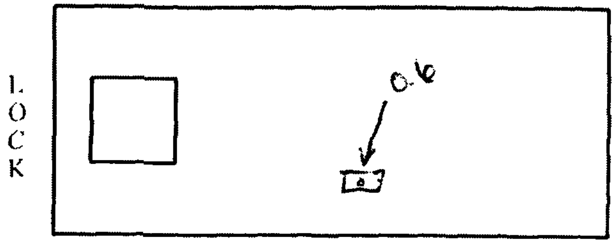
SPEC - 150 EXPOSURE DEVICE

SERIAL # 331

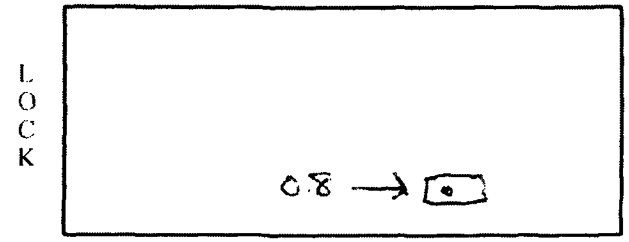
SOURCE sn: SA3112 CURIES: 131.29

ACTUAL READINGS. ☑ Surface ☐ One meter

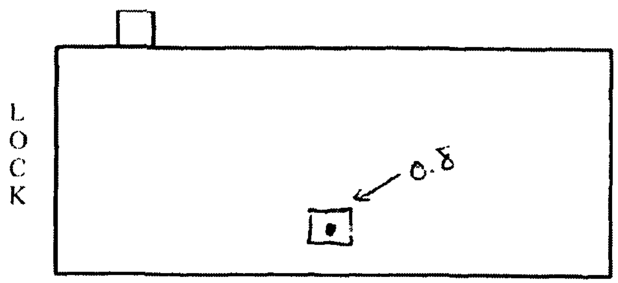
DATE: 3/5/11 Before Normal Cond Test



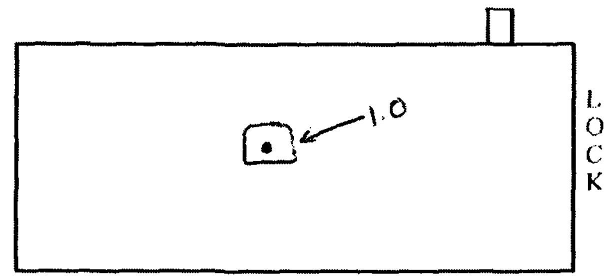
Top Side



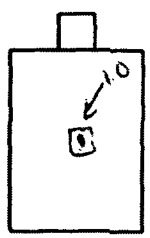
Bottom



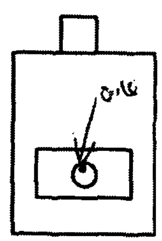
Right Side



Left Side



Lock End



Outlet End

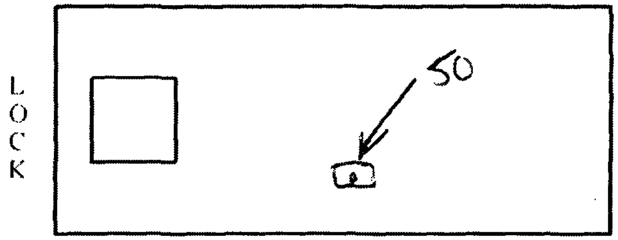
SPEC - 150 EXPOSURE DEVICE

SERIAL # 331

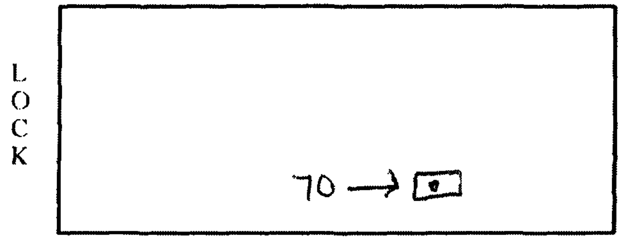
SOURCE sn: SA3112 CURIES: 131.29

ACTUAL READINGS, ☐ Surface ☒ One meter

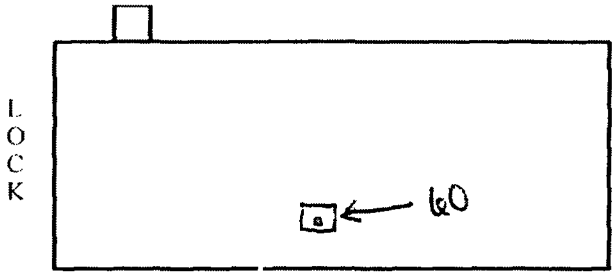
DATE: 3/3/11 Before Test



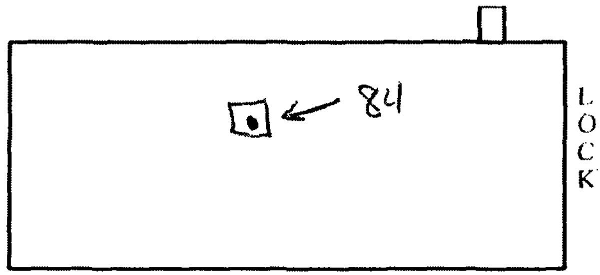
Top Side



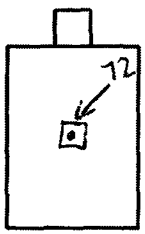
Bottom



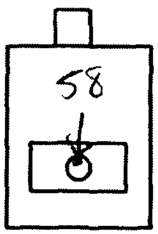
Right Side



Left Side



Lock End



Outlet End

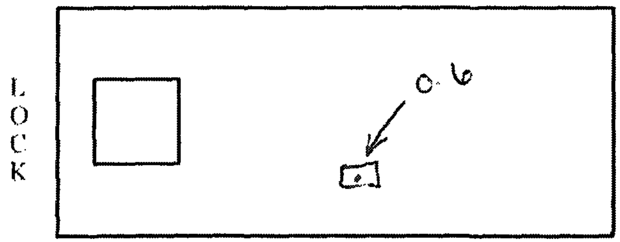
SPEC - 150 EXPOSURE DEVICE

SERIAL # 331

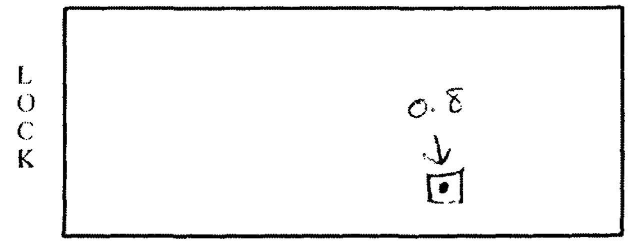
SOURCE sn: SA3112 CURIES: 131.29

ACTUAL READINGS, ☑ Surface ☐ One meter

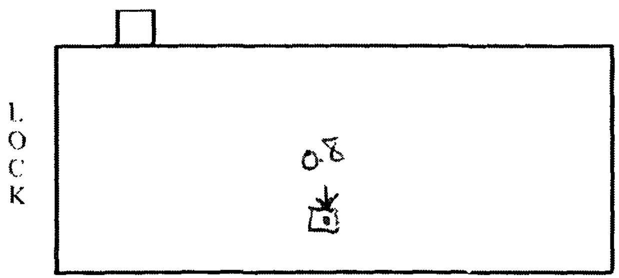
DATE: 3/3/11 Post Normal Cond. Test



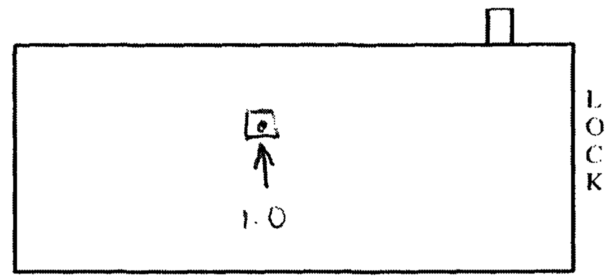
Top Side



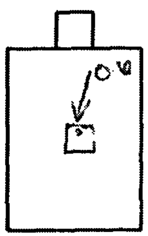
Bottom



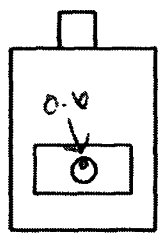
Right Side



Left Side



Lock End



Outlet End

SPEC - 150 EXPOSURE DEVICE

SERIAL # 331

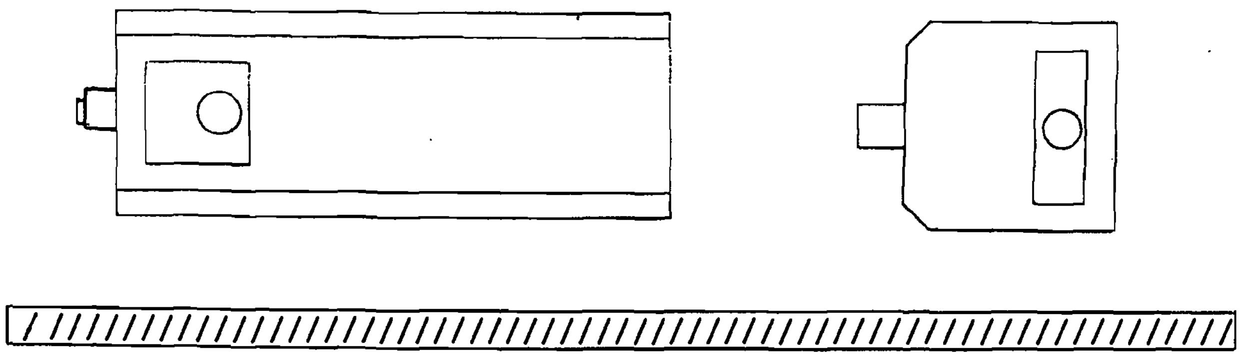
SOURCE sn: SA3112 CURIES: 131.29

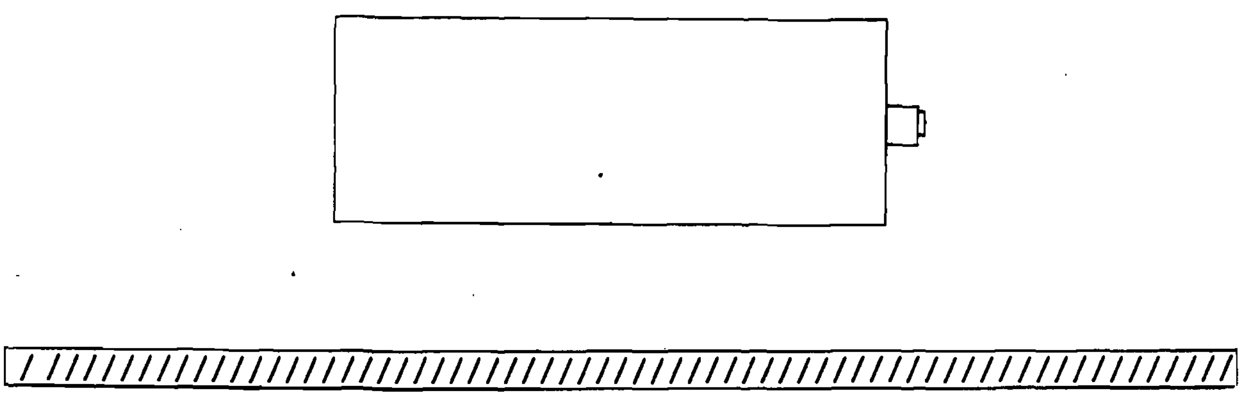
ACTUAL READINGS, ☐ Surface ☒ One meter

DATE: 2/3/11 After Test

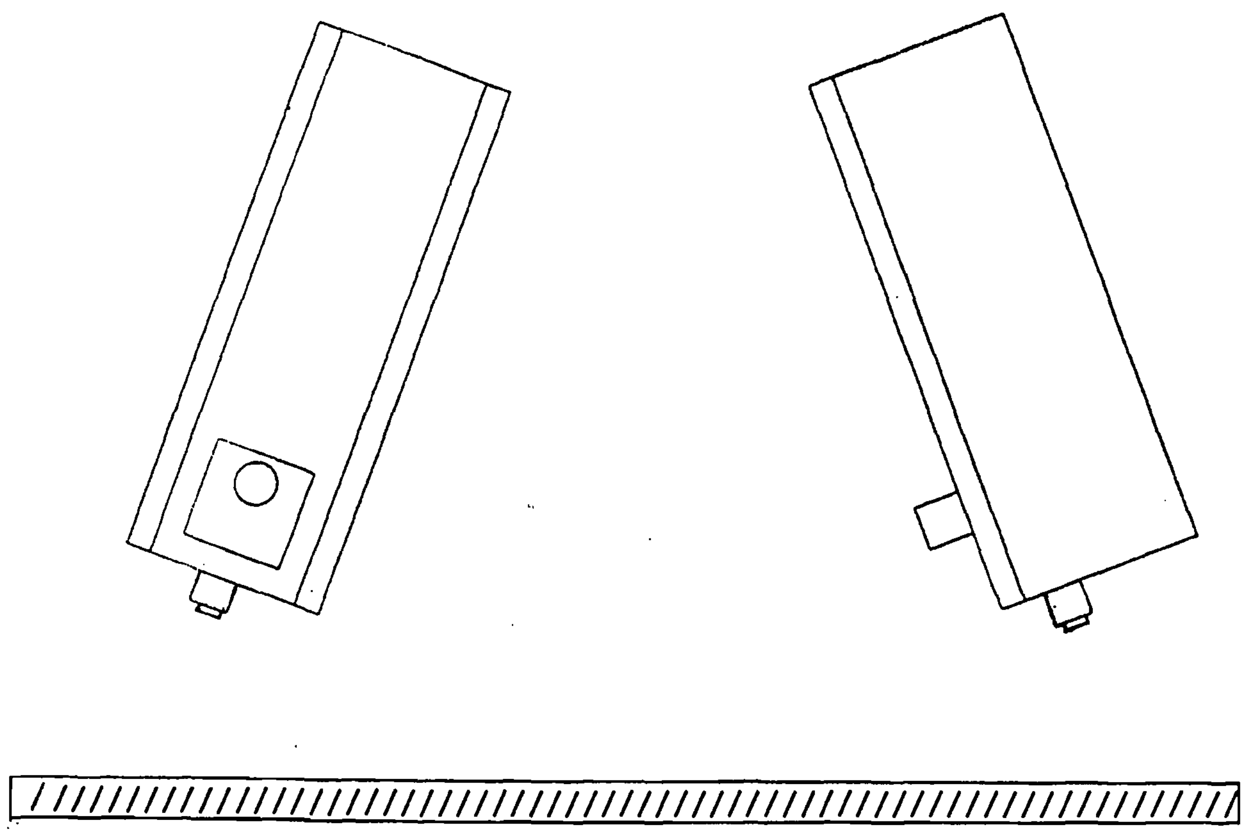
Appendix 9.4
1994 Sketches of Drop Test Impact Orientations

PROTOTYPE #2
1ST 9 METER DROP
POINT OF IMPACT;
DIRECTLY ON RIGHT SIDE



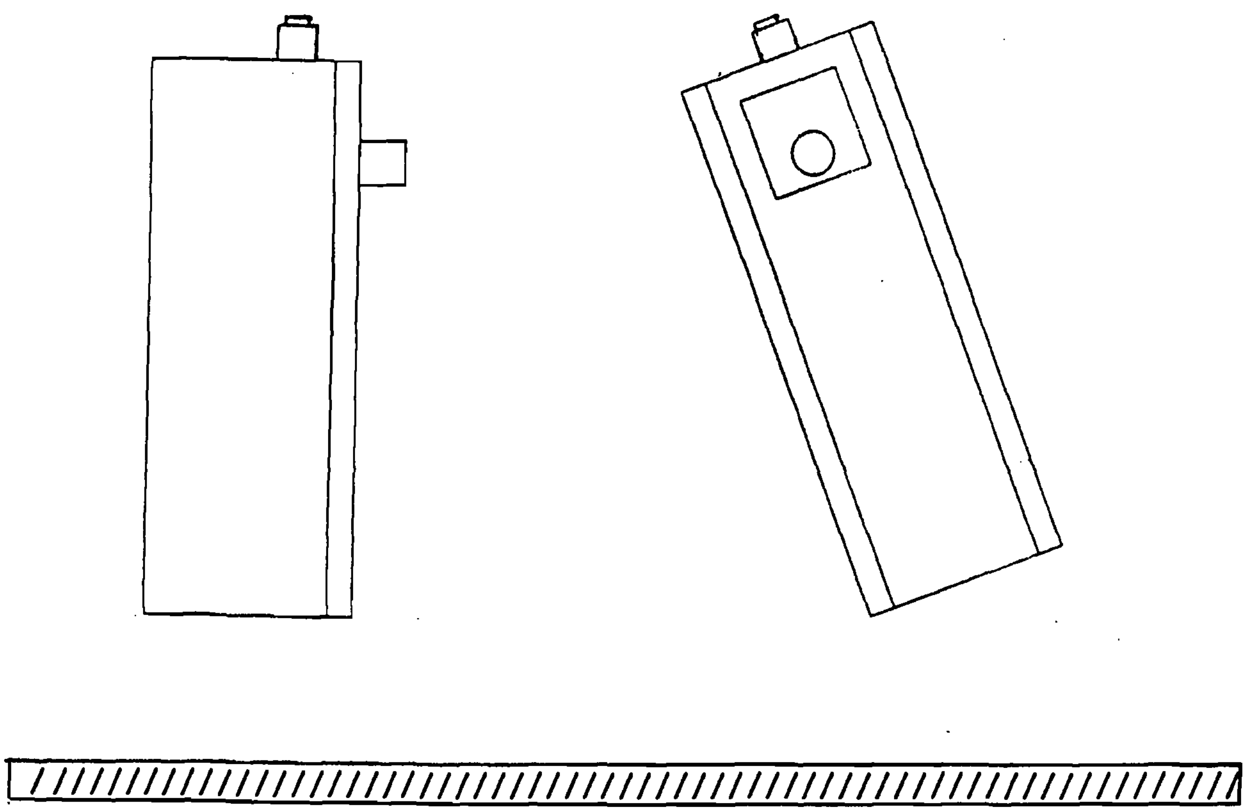


PROTOTYPE #2
2ND 9 METER DROP
POINT OF IMPACT;
TOP/RIGHT CORNER AT THE LOCK END

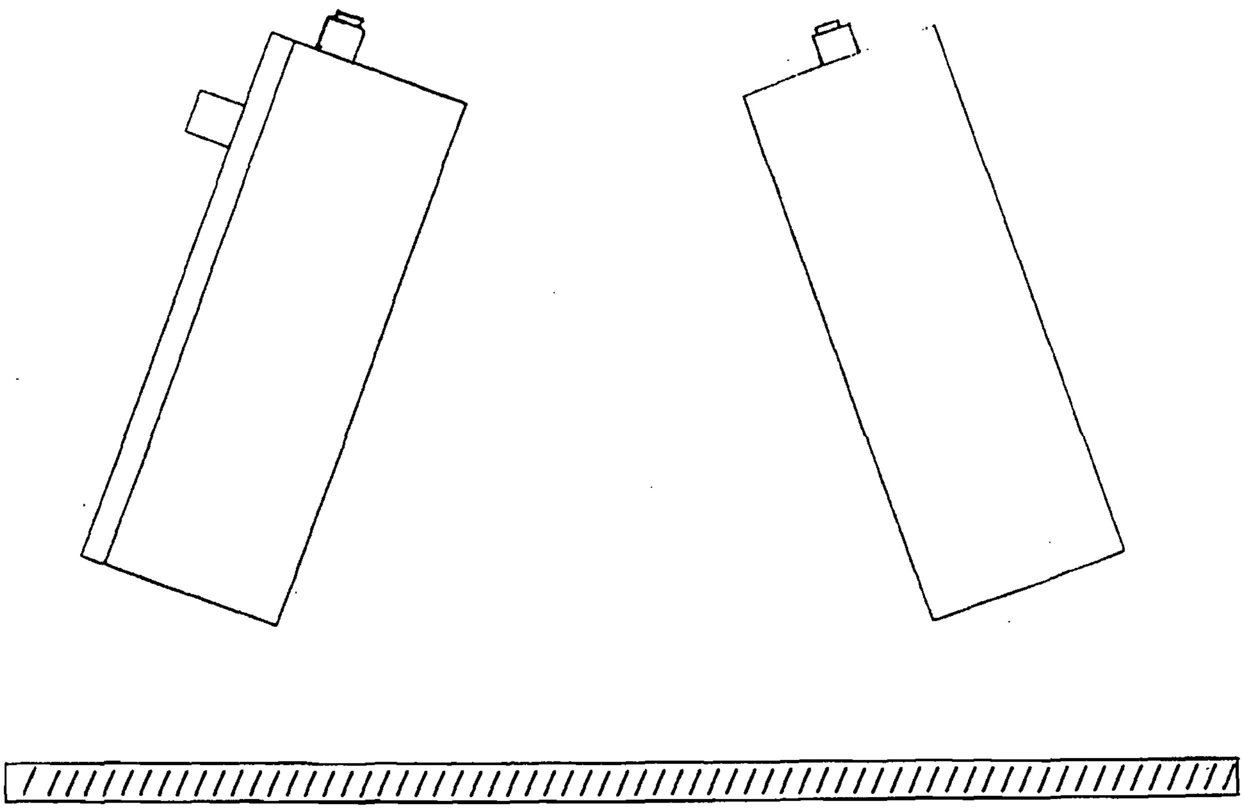


PROTOTYPE #2
3RD 9 METER DROP
POINT OF IMPACT;
RIGHT SIDE EDGE AT THE OUTLET END

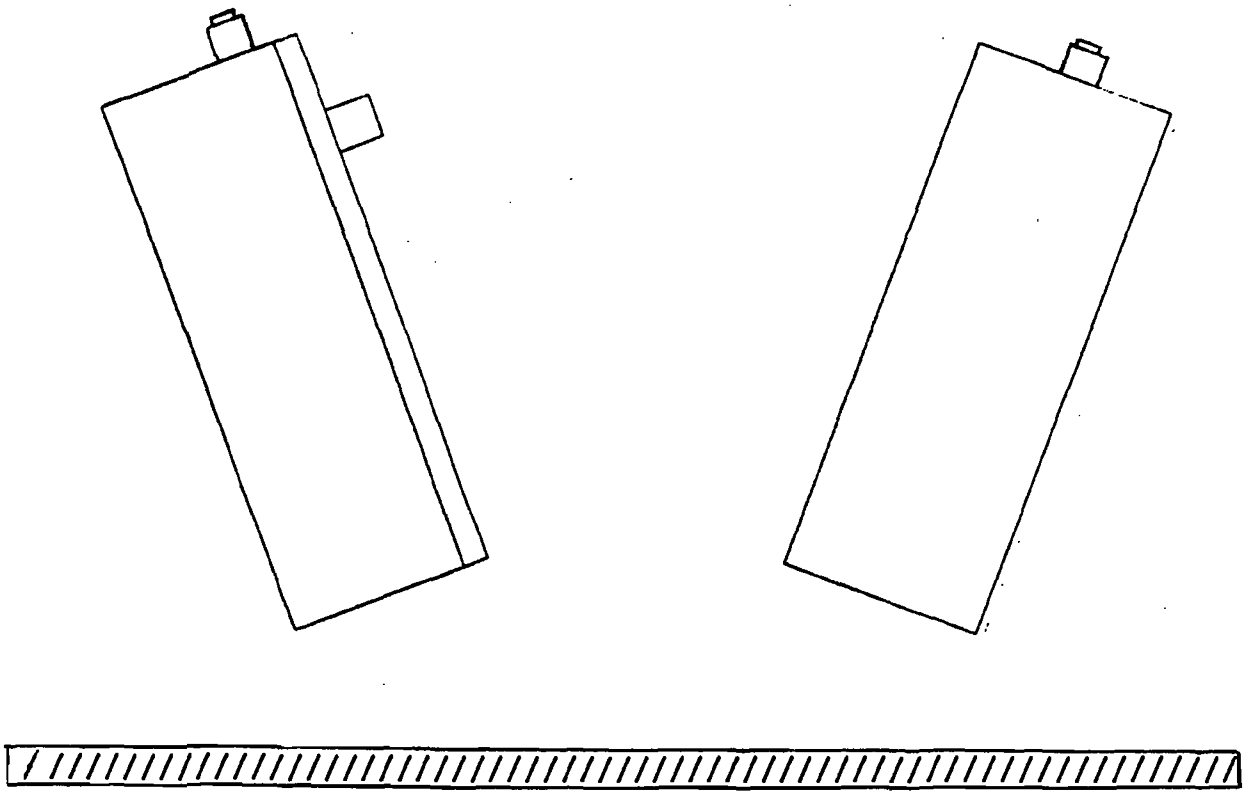
(NOTE: THE INTENDED POINT OF IMPACT WAS THE RIGHT/BOTTOM CORNER AT THE OUTLET END. DUE TO WIND CONDITIONS THE DEVICE SLIGHTLY ROTATED DURING THE DROP AND CAUSED THE IMPACT POINT TO BE THE RIGHT SIDE EDGE AT THE OUTLET END.)



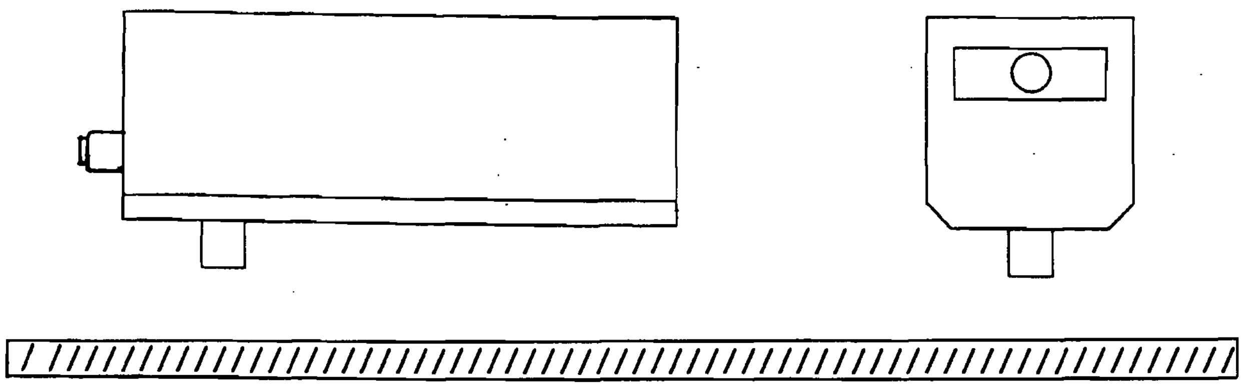
PROTOTYPE #2
4TH 9 METER DROP
POINT OF IMPACT;
BOTTOM/LEFT CORNER AT THE OUTLET END

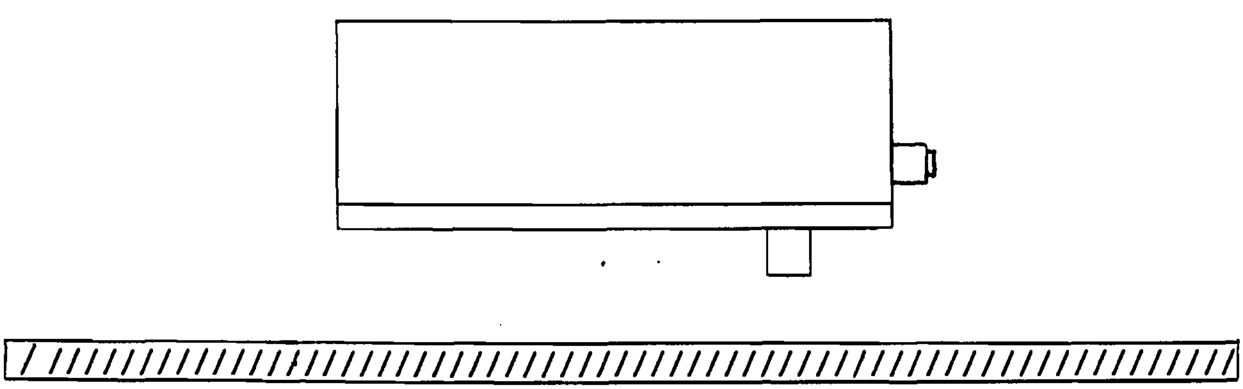


PROTOTYPE #4
1ST 9 METER DROP
POINT OF IMPACT;
RIGHT/BOTTOM CORNER AT OUTLET END

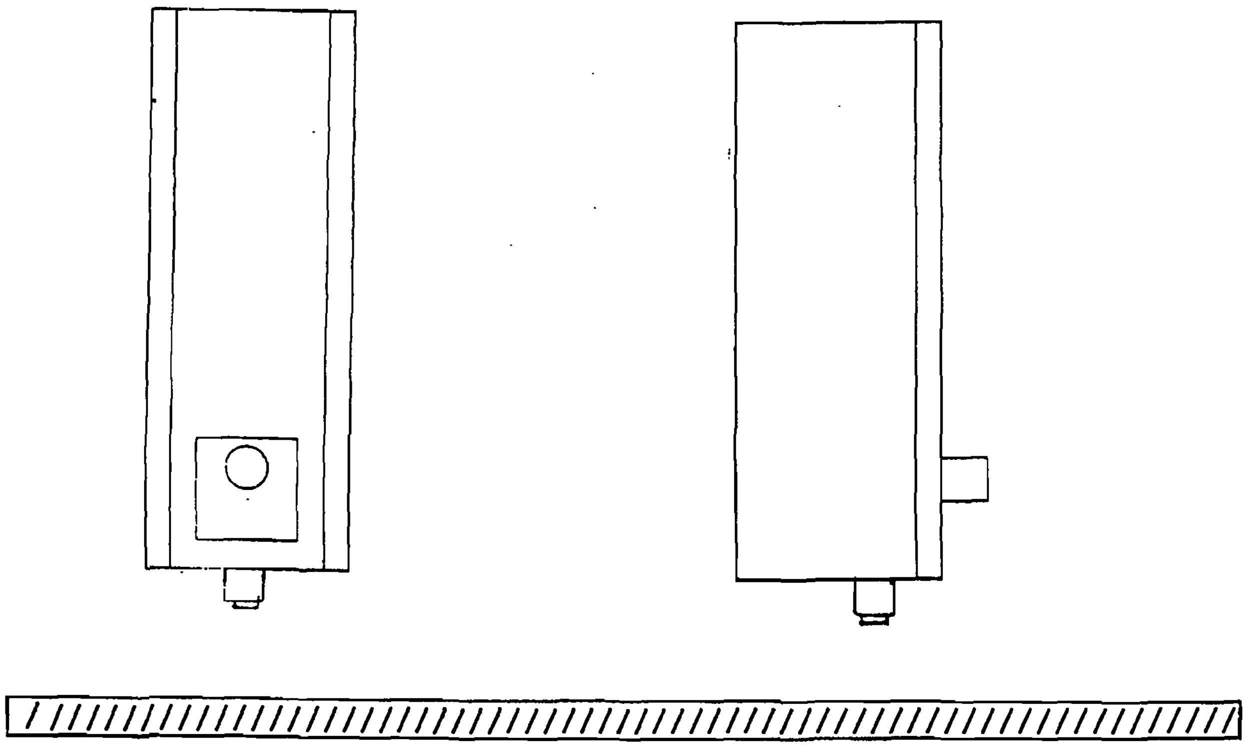


PROTOTYPE #4
2ND 9 METER DROP
POINT OF IMPACT;
DIRECTLY ON TOP

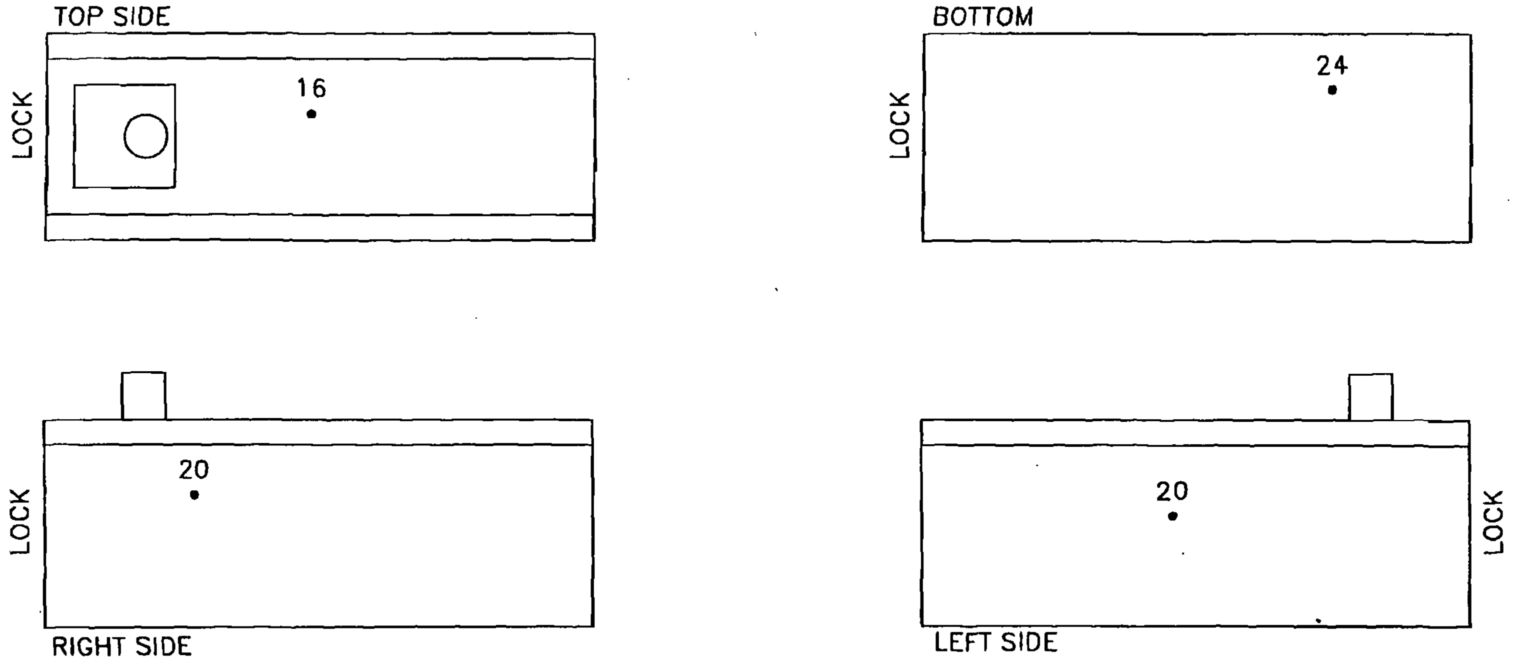


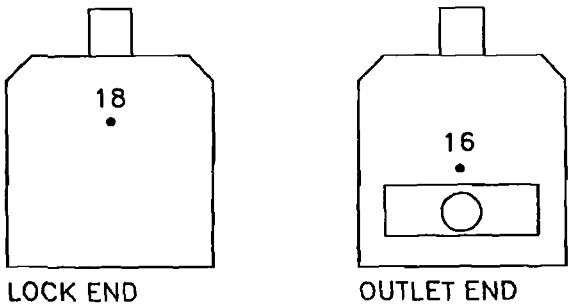


PROTOTYPE #4
3RD AND 4TH 9 METER DROPS
POINT OF IMPACT;
DIRECTLY ON LOCK END (LOCK CAP, SOURCE LOCK AND DEVICE LOCK)



Appendix 9.5
1994 Sketches of Highest Surface Radiation Survey Data





SPEC-150 EXPOSURE DEVICE

PROTOTYPE #4 RADIATION PROFILE: PRE-DROP

SURVEY DATE 8-26-94

SURVEY METER MODEL GS1000A S/N 1771 CAL. DATE 8-26-94

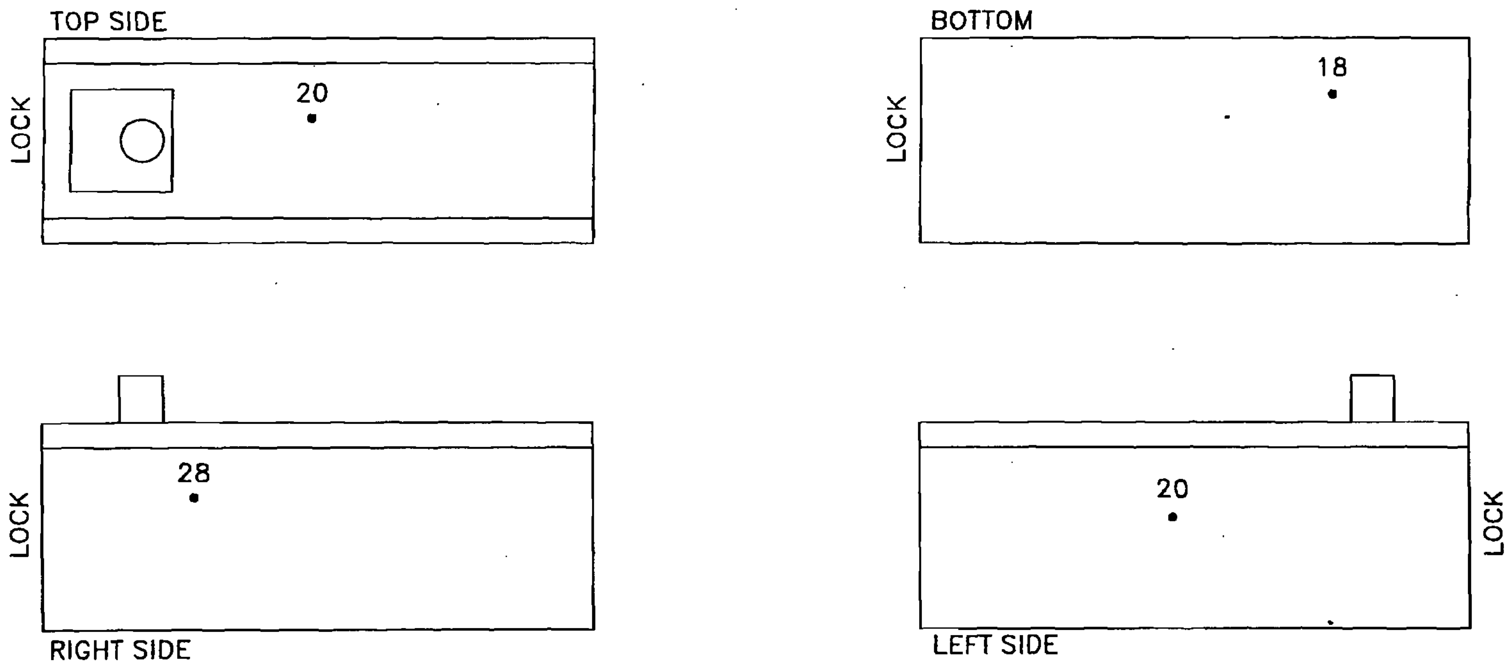
CURIES 23 SOURCE MODEL G-60 SOURCE S/N AH2503

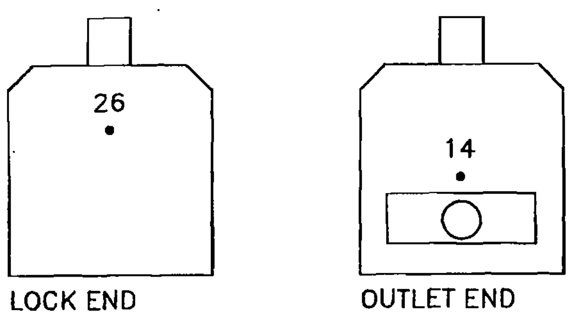
ALL READINGS @ SURFACE

READINGS ARE NOT EXTRAPOLATED OR ADJUSTED.

READINGS ARE RECORDED IN mR/hr.

FILE: SRP894A





SPEC-150 EXPOSURE DEVICE
PROTOTYPE #4 RADIATION PROFILE: POST-DROP

SURVEY DATE 8-30-94 (2 EA. 9 METERS; 1 EA. PUNCTURE TEST)

SURVEY METER MODEL GS1000A S/N 1771 CAL. DATE 8-30-94

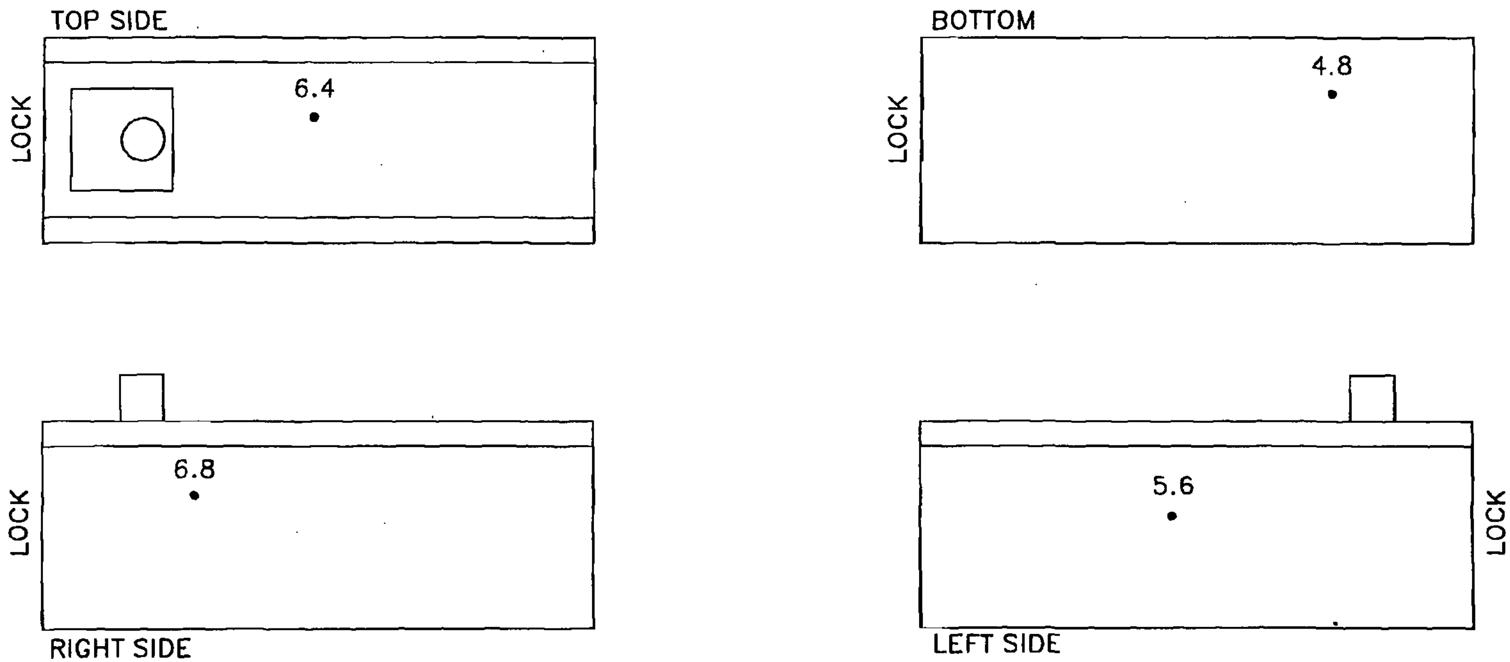
CURIES 22 SOURCE MODEL G-60 SOURCE S/N AH2503

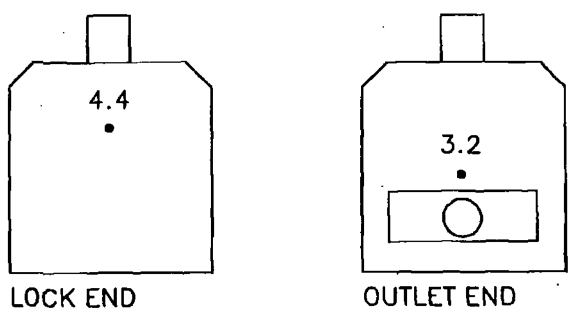
ALL READINGS @ SURFACE

READINGS ARE NOT EXTRAPOLATED OR ADJUSTED.

READINGS ARE RECORDED IN mR/hr.

FILE: SRP894B





SPEC–150 EXPOSURE DEVICE

PROTOTYPE #4 RADIATION PROFILE: PRE–DROP (4 FOOT)

SURVEY DATE 12–17–94

SURVEY METER MODEL GS1000A S/N 1771 CAL. DATE 12–5–94

CALIBRATION VERIFICATION· 12–17–94

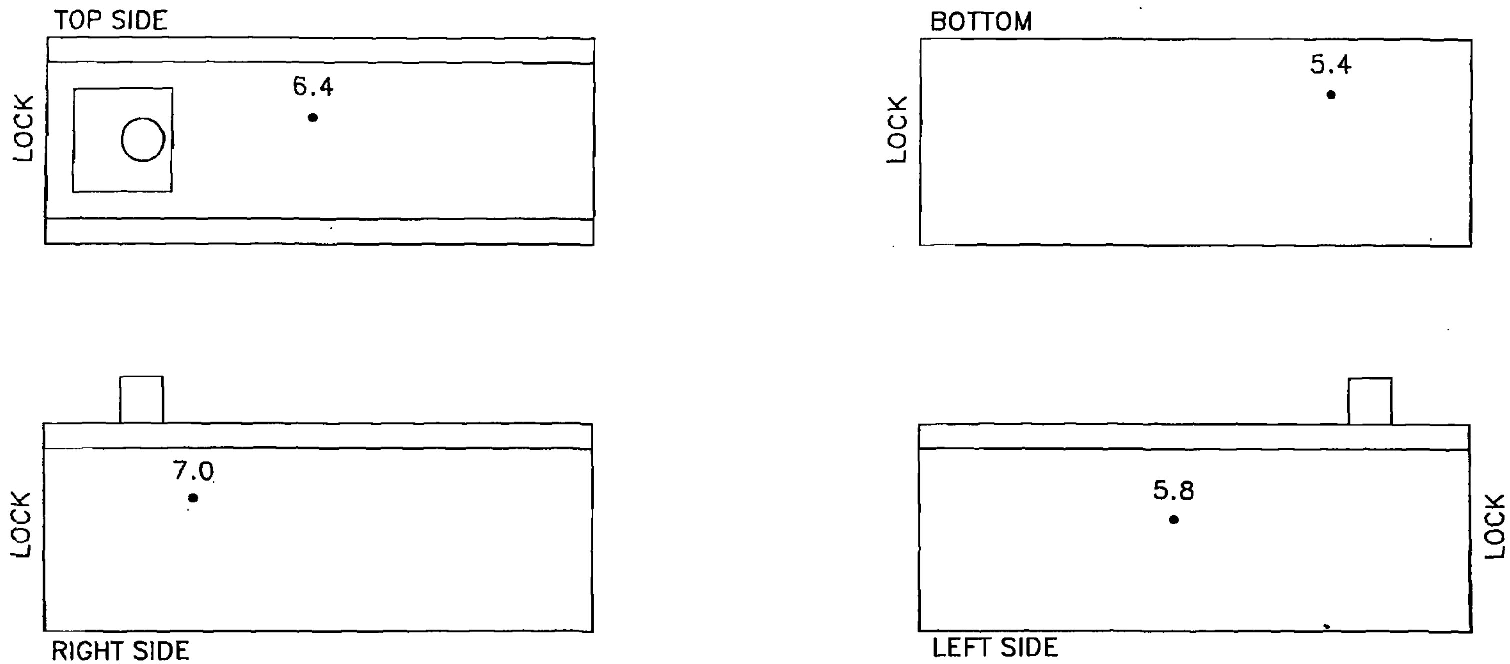
CURIES 8 SOURCE MODEL G–60 SOURCE S/N AH2503

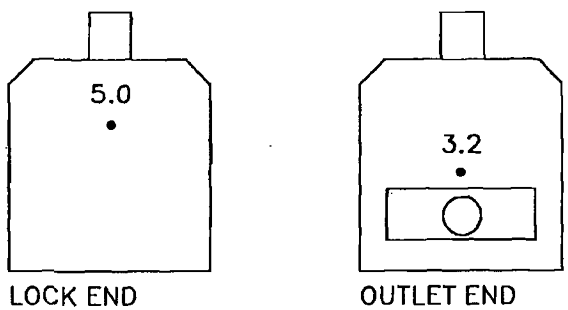
ALL READINGS @ SURFACE

READINGS ARE NOT EXTRAPOLATED OR ADJUSTED.

READINGS ARE RECORDED IN mR/hr.

FILE: SRP1294A





SPEC-150 EXPOSURE DEVICE

PROTOTYPE #4 RADIATION PROFILE: POST-DROP

(AFTER 5 EA. 4 FOOT DROPS)

SURVEY DATE 12-17-94

SURVEY METER MODEL GS1000A S/N 1771 CAL. DATE 12-5-94

CALIBRATION VERIFIED 12-17-94

CURIES 8 SOURCE MODEL G-60 SOURCE S/N AH2503

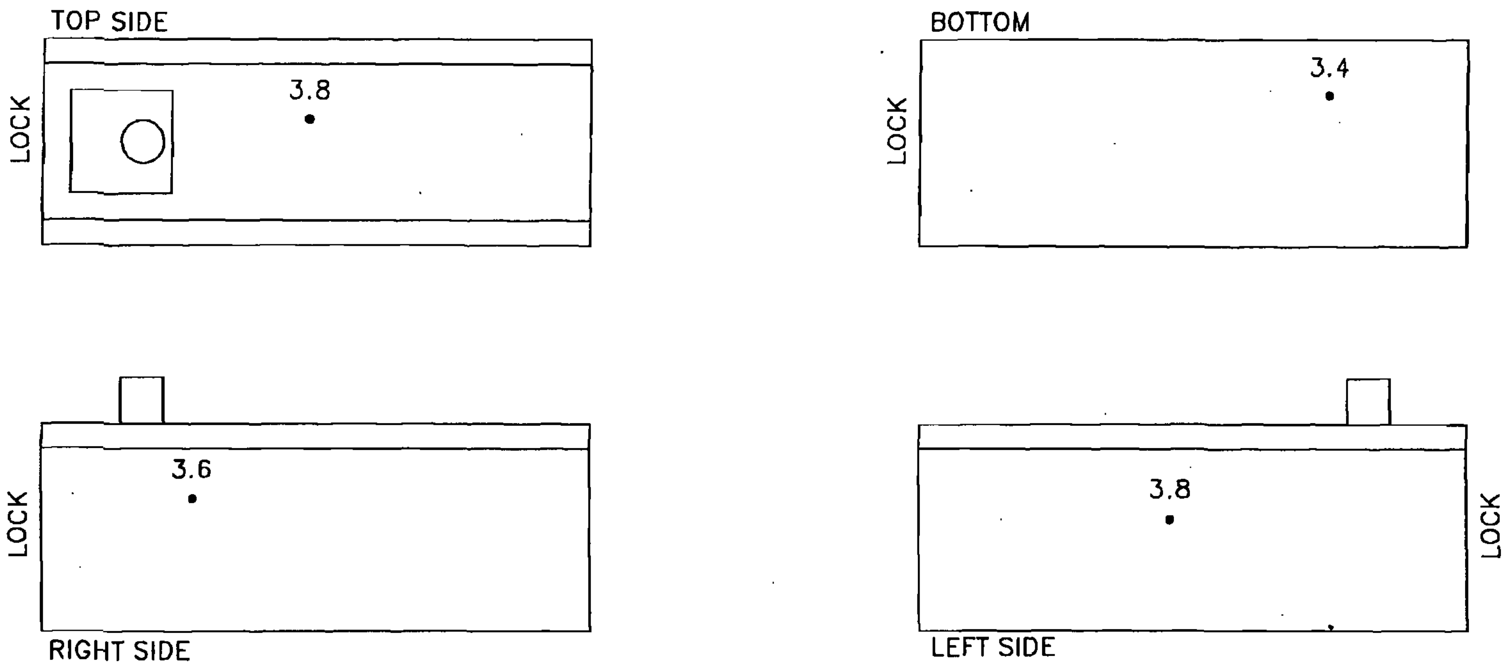
ALL READINGS @ SURFACE

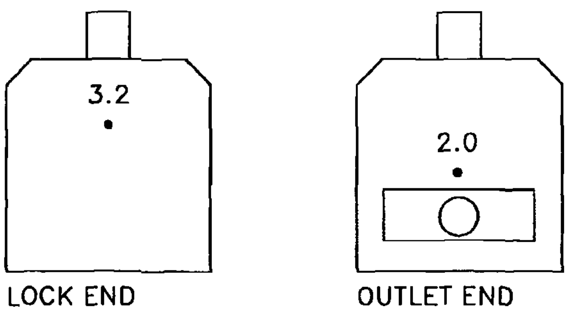
READINGS ARE NOT EXTRAPOLATED OR ADJUSTED.

READINGS ARE RECORDED IN mR/hr.

NOTE: THE DEVICE WAS SURVEYED AFTER EACH 4 FOOT DROP. THE READINGS SHOWN ARE THE HIGHEST READINGS (NOT AVERAGE) DETECTED AFTER EACH OF THE 5 DROPS.

FILE: SRP1294B





SPEC-150 EXPOSURE DEVICE

PROTOTYPE #4 RADIATION PROFILE: PRE-DROP

(2EA. @ 9 METERS; 1 EA. PUNCTURE TEST)

SURVEY DATE 2-25-95

SURVEY METER MODEL GS1000A S/N 1771 CAL. DATE 2-24-95

CALIBRATION VERIFIED 2-25-95

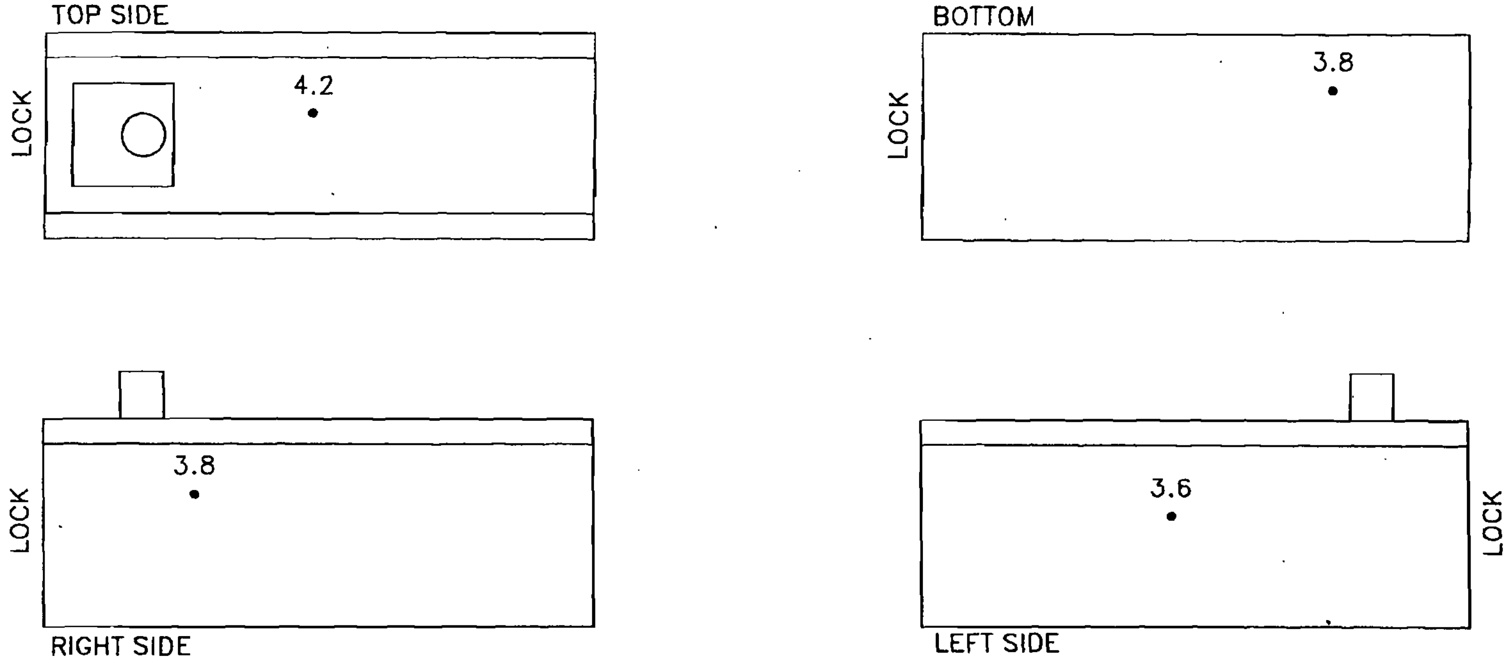
CURIES 4 SOURCE MODEL G-60 SOURCE S/N AH2503

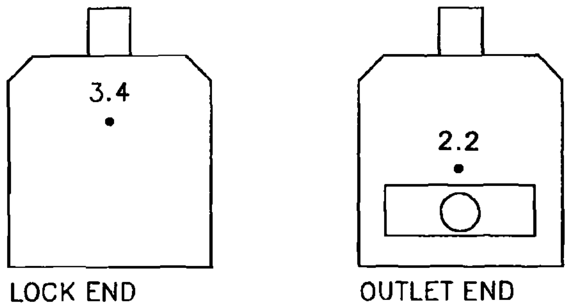
ALL READINGS @ SURFACE

READINGS ARE NOT EXTRAPOLATED OR ADJUSTED.

READINGS ARE RECORDED IN mR/hr.

FILE: SRP295A





SPEC-150 EXPOSURE DEVICE
PROTOTYPE #4 RADIATION PROFILE: POST-DROP
(2 EA. @ 9 METERS; 1 EA. PUNCTURE TEST)

SURVEY DATE 2-25-95

SURVEY METER MODEL GS1000A S/N 1771 CAL. DATE 2-24-95

CALIBRATION VERIFIED 2-25-95

CURIES 4 SOURCE MODEL G-60 SOURCE S/N AH2503

ALL READINGS @ SURFACE
READINGS ARE NOT EXTRAPOLATED OR ADJUSTED.
READINGS ARE RECORDED IN mR/hr.

FILE: SRP295B

Appendix 9.6
1997 30' Drop Test & Validation of Previous Puncture Tests

SOURCE PRODUCTION AND EQUIPMENT COMPANY, INC.
113 Teal Street, St. Rose, Louisiana 70087

1997 Hypothetical Accident Conditions Testing

Model SPEC-150 Type B(U) Package
Docket Number 71-9263

SOURCE PRODUCTION AND EQUIPMENT COMPANY, INC.
1997 Hypothetical Accident Conditions Testing
Model SPEC-150 Type B(U) Package
Docket Number 71-9263

Table of Contents Page

SOURCE PRODUCTION AND EQUIPMENT COMPANY, INC.
1997 Hypothetical Accident Conditions Testing
Model SPEC-150 Type B(U) Package
Docket Number 71-9263

1. GENERAL INFORMATION

1.1 Introduction and Background.

On June 12, 1997 the NRC informed SPEC that one of the hypothetical accident condition tests, the puncture tests, that was performed to qualify the SPEC-150 as a Type B(U) package might not be valid. This concern was based on the puncture pin mounting information SPEC voluntarily provided to the NRC the previous week. An NRC 10 CFR Part 71 inspection was conducted the week of June 16th, 1997. The inspection confirmed that the six (6) inch diameter steel pin used for the one (1) meter puncture test was not rigidly mounted to the test target pad. The previous puncture tests used a pin that was mounted on the test target pad but the pin had not been mounted rigidly to prohibit toppling and vertical movement (i.e., bolted or welded). A review of a video tape of one of the numerous puncture tests conducted proved that the pin did not topple but that it did move laterally a few inches during the test. The NRC issued a Confirmatory Action Letter dated June 24, 1997 which describes SPEC's commitment to retest the SPEC-150 to verify the validity of the previous puncture tests.

The drop test and puncture tests were conducted on June 26, 1997 in accordance with SPEC Test Plan, Rev (1) dated June 24, 1997 which were witnessed by Mr. Cass R. Chappell and Mr. Andrew Gaunt from the NRC, and Mr. Sami Aouad and Ms. Ann Troxler from the Louisiana Division of Radiation Protection. The puncture test must be conducted in sequence following the 9 meter free drop test. Two additional puncture tests were conducted for the SPEC-150 that were not included in the Test Plan. Those tests are described in the test report. The tests were conducted using the same orientations that were chosen for the previous puncture tests as an extra means to verify the validity of the previous tests. The damage from the additional puncture tests was extremely slight, insignificant in terms of structural and shielding, and virtually identical to the damage from the previous tests. The tests successfully verified the validity of all the previously performed puncture tests.

1.2 Test Report Format.

This report provides the test information that is specified in the Test Plan, Rev (1) dated June 24, 1997 which includes data from tests of other packages (SPEC 2-T and SPEC C-1). That data is not relevant for purposes of this report. The report includes additional information that was not called for in the Test Plan. An edited copy of the Test Plan, which includes some of the test data, is located in the Appendix 4. 2. The Test Plan document with original signatures is maintained as a QA record.

1.3 Radioactive Contents of Test Package.

A production source was installed in the test package to perform the accident condition tests. This is the most direct and reliable means to evaluate the displacement of the source relative to the shield and to evaluate the integrity of the shield after the tests. Experience has shown that for packages similar to radiography devices and source changers it is usually impossible to remove and replace a dummy source assembly with a live source after the hypothetical accident condition tests due to the structural damage, particularly for radiography cameras. In many cases the package must be partially dismantled. We believe it is not possible to reliably position the live source in exactly the same location as the dummy source used for the test. The adequacy of the shielding design for the package was verified by actual measurements of radiation profiles of the test sample before and after the tests, and the readings are extrapolated to the maximum authorized activity for the package.

1.4 Survey Method.

The surveys of the test packagings were performed in accordance with SPEC Survey Procedure 7.04, Rev (3) (See Appendix 4. 3). No distance correction factor from the package surface to the detector was applied to surface radiation levels because the surface readings are not required to determine if a packaging meets the shielding requirements following the hypothetical accident condition free drop and puncture tests. Background radiation was not factored out of survey readings because it does not have significant impact on the actual readings at 1 meter for the purposes of this test, which is to verify that no radiation level exceeds 1 R/hr at 1 meter when extrapolated to the maximum authorized activity of the packaging. The background radiation levels at the location in the facility where the surveys were conducted ranged from approximately 0.2 mR/hr to 0.4 mR/hr depending on the quantity of packages being prepared for transport in the adjacent shipping area of the shop.

The purpose for recording surface radiation readings before and after the tests is twofold. First, it provides additional data that either supports or refutes the structural evaluation of a package. Secondly, it is used to locate the spot on each side where the radiation level is highest from which to take the reading at 1 meter. It should be noted that even for a package with flat surfaces at 1 meter away the detector might not be located perpendicular to the highest surface reading. The highest reading at 1 meter will be located on a direct line formed by the source capsule and the highest reading spot on the surface. Unless the highest surface reading is located adjacent to the center of the DU shield, the beam of highest radiation will not be perpendicular to the package at 1 meter. Using a survey method that requires perpendicular positioning of the detector from the highest surface reading will not produce the highest actual readings at 1 meter. SPEC's survey method assures that the highest readings at 1 meter are taken.

1.5 Drop Target Description.

The drop target at SPEC greatly exceeds the requirements specified in IAEA Safety Series No. 37 "Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material," 1985 Edition as amended 1990. The drop target consists of a solid carbon steel plate which measures 35-1/4 inches x 30-1/4 inches x 1-3/4 inches thick weighing 528 pounds (See Drawing Nos. 50890-1, Rev (1) in Appendix 4. 1). The thickness of the steel plate meets the minimum 4.0 cm IAEA requirement. The steel plate was wet floated onto a 1 inch thick layer of high strength grout to keep the plate level and prevent air pockets from forming while curing. Both conditions are more common when floating directly onto wet cement. The grout is a commercial product with a minimum compressive strength of 7,250 psi which is approximately twice the compressive strength of concrete. The grout firmly attaches the steel plate to the top surface of a flat horizontal concrete block which measures 46 inches x 46 inches x 56 inches thick weighing approximately 9900 pounds. The total weight of the drop target is approximately 10,500 pounds which greatly exceeds ten times the mass of a 100 pound package. The concrete block is sunk to a depth of 52 inches into firm soil. (A detailed inspection of the drop target prior to the tests identified variations from the target description provided in previous applications.) A sturdy 40 foot tall structure was erected over the drop target and used to raise and release the test package from a minimum height of 30 feet (9.2 meters) above the top surface of the target. No damage nor separation of the steel plate, grout and the concrete block has occurred from any previous tests.

1.6 Puncture Pin Description.

The puncture pin is a 6 inch diameter x approximately 8-1/4 inch long solid mild steel rod. It is rigidly mounted to the center of the drop target pad (See Drawing No. 990001, Rev (0) in Appendix 4. 1). This is essentially the same design that was used for the previous puncture tests. However, for the previous tests the pin was not mounted in a manner to assure that it would not topple over or move laterally. The pin used for these tests is welded to a steel holder which was bolted to the steel drop target pad. The pin is inspected after each puncture test to verify that it did not topple over nor move laterally.

The length of the puncture pin was selected to be a minimum of 8 inches, in accordance with 10 CFR 71.73(c)(2), and a maximum of 8.3 inches. This is based on an analysis of each package design and previous test experience to determine which pin length will cause maximum damage to the package. The deformation resulting from previous 30 foot free drop and puncture tests has been relatively minor; far less than eight inches. The package contain no deep recesses or pockets that could be accessed better by a bar longer than 8 inches. These package will not be housed in a drum or overpack during testing. It is clear that there is no length greater than 8 inches that will cause more damage to the package than the required 8 inch pin length. Based on these factors the length of the puncture test bar will be 8 inches min, 8.3 inches max.

1.7 Selection of Test Package Orientation.

The orientation of a package must be selected to strike the package surface in a position for which maximum damage is expected. The term "maximum damage" is not defined in 10 CFR 71.73, but 10 CFR 71.51 (a) (2) specifies that as a result of testing, the radiation dose rate will not exceed one Rem/hr at 1 meter from the external surface of the package. For purposes of orientation analysis, maximum damage will be considered as the condition that provides maximum movement, or chance of movement, of the sealed radioactive source away from the fully shielded position within the depleted uranium (DU) shield. This is the condition most likely to result in increased radiation levels outside of the package. The history of testing packages has provided extensive comparative information regarding the orientations that have caused the most damage in previous tests. It should be noted that there has never been an instance in which an accident condition test has fractured, deformed or otherwise reduced the shielding capability of the DU shield, even at temperatures near minus (-) 100 degrees Fahrenheit. Therefore, orientations to promote shield and source separation is considered far more likely to cause the maximum damage. The orientation selection and supporting rational was recorded prior to the start of the tests and are included in this report. After the 9 meter free drop test the orientation for the puncture test was re-evaluated based on the damage from the 9 meter free drop. See sketches of orientations in Appendix 4. 4.

2. STRUCTURAL EVALUATION

2.1 Structural Design

2.1.1 Description of Test Packaging.

The test package is a Model SPEC-150, s/n 500, which is a production package that was constructed in standard production fashion pursuant to applicable quality assurance procedures specified in NRC Certificate of Compliance No. 0102. Design, fabrication and inspection records were verified to confirm that the test packaging meets the approved drawings. This was done to address one of the findings from the 10 CFR Part 71 inspection conducted the week of June 16th, 1997. For each of the tests an actual production model G-60 Iridium-192 source assembly, which was constructed in standard production fashion pursuant to applicable quality assurance programs, was contained within the test package. The test packaging specimen meets the design specified in Certificate of Compliance No. 9263 (See Package Drawings, Appendix 4. 5). It represents the current production model design which includes all of the design revisions that are referenced in Certificate of Compliance No. 9263, Rev (1) that were not included in the design of previously tested prototypes. Specifically, SPEC-150, s/n 500 includes the additional four (4) holes in the top corners of the package and the revised lock cap. See Drawings 15B000, Rev (4), 15B001-3, Rev (0), 15B002A, Rev (3), and 15B008, Rev (2).

2.2 Hypothetical Accident Condition - 9 Meter Free Drop Test.

2.2.1 Discussion.

The test package was chilled in dry ice to a temperature below minus (-) 80 degree Fahrenheit (See Photograph A1). This was done to address any potential concerns regarding the range of ambient temperatures from minus (-) 20 to plus (+) 100 degrees Fahrenheit to be considered for the tests. Chilling was not performed for any of the previous free drop tests for the SPEC-150.

2.2.2 Selection of Orientation.

The point of impact for SPEC-150, s/n 500, was based on an analysis of the design and extensive past experience testing numerous prototype SPEC-150 and SPEC 2-T packages. The orientation and point of impact was selected to cause the maximum damage to the package based on the maximum potential for shield movement. Essentially, the SPEC-150 is a depleted Uranium shield weighing approximately 35 lb. enclosed in a rectangular GTAW welded Titanium enclosure. The shield is retained in the enclosure by tabs or "ears" cast integrally with the shield. One of these tabs is attached to the outlet end plate. The other end tab is attached to the intermediate bulkhead within the enclosure toward the lock end. The mechanism securing the radioactive source assembly in the center of the depleted Uranium shield is attached to the lock end of the enclosure. An impact causing the shield to shift toward the outlet end of the camera, while preserving the integrity of the securing mechanism, would result in the source capsule being displaced from the fully shielded position within the depleted Uranium shield. As defined above, this would result in maximum damage.

Dropping the package with the impact flat on the outlet end of the package would be expected to maximize the chances of significant relative displacement of the shield, except that the safety plug extends almost to the end of the package. This would limit the deformation of the end plate to approximately 0.125", resulting in less than maximum damage. Therefore, this orientation was not selected.

Dropping the package with the impact point at the outlet end bottom edge or at one of the outlet end bottom corners allows for a larger potential displacement of the shield, especially if the center of gravity of the package is located approximately over the impact point to reduce package rotation at impact. Additionally, this orientation allows the reaction from the shield to bear directly on the end plate near the weld joint. Prototype packages have been dropped repeatedly and demonstrate greatest physical damage when dropped on one of the outlet end corners. Since the package housing is symmetrical in this axis, either corner may be chosen. Based on these factors the package was oriented so as to impact with the center of gravity over the outlet end bottom right corner, as viewed from the lock end of the package (See Photograph A2)(See Orientation Sketch, Appendix 4. 4).

2.2.3 Drop Test Description

A model SPEC-150 production package, S/N 500, was subjected to a free drop from a distance of 9 meters (30 feet) measured from the bottom of the package to the top of the previously described drop target. The point of impact was the bottom right corner at the outlet end of the package as planned (See Photograph A3). The package retained the planned orientation throughout the free fall and initial impact.

2.2.4 Damage Assessment.

Outlet End:	The protective flange located at the end of the bottom panel buckled upward 7/8 inch (See Photograph A4). The flange located at the end of the right side panel buckled inward 3/4 inch. The safety plug is installed in the outlet nipple which is attached to the outlet end panel. The top edge of outlet panel is bowed outward 3/32 inches at the center and is approximately 2-½ inches long (See Photograph A5). The safety plug is jammed in the outlet nipple and cannot be manually removed (See Photograph A6). Due to the deformation of the protective flanges the safety plug now protrudes beyond plane of protective flanges 1/8 inch (See Photograph A7).
Left Side:	The left side buckling extends 1-7/8 inches toward the outlet panel.
Right Side:	The right side buckling extends back 1-1/4 inches in toward outlet panel. No other damage except at the flange at the outlet end.
Top:	No damage, including the release plunger and carrying handle (See Photograph A8).
Bottom:	No damage except flange on outlet end. The bottom plate dented upward along entire width along the outlet flange end.
Lock End:	The upper right corner dented inward 1/4 inch, no other damage. The lock cap is intact and operates properly.

2.2.5 Accidental Drop Test Summary

The most significant damage is that the DU shield shifted toward the outlet end approximately 3/32 inch as expected. Not a single TMJ weld joint separated or fractured, not even at the point of impact. All housing plates (top, bottom, sides and both ends) remained fully intact. All components remained intact and attached to the package. The source remained in the locked position and was not displaced. The source assembly lock, the device lock, and the automatic securing mechanism (ASM) were completely undamaged. The lock cap and safety plug remained intact, installed and fully functional as redundant safety features to prevent loss of the source in both directions. In fact, an operational check of the lock and securing systems proved that the device remained fully functional and could be operated as a radiography device.

2.3. Hypothetical Accident Condition - Puncture Test.

2.3.1 Selection of Orientation.

Relative to the size of the package being tested, the puncture test pin is very large. It is unlikely that the puncture test pin will penetrate the package. Past puncture tests have not penetrated the exterior of the package at all or caused significant damage to this or other SPEC packages. Based on these factors an initial anticipated puncture test orientation was selected so preparations could be made in advance of the actual puncture test.

The most likely scenario for maximum damage and elevating radiation levels as a result of this test would involve breaking the outlet nipple off of the outlet end plate. This component is relatively fragile when compared with the rest of the package and could possibly be broken off by an impact against the edge of the pin. There is no guidance to require or prohibit selecting the edge of the pin rather than the center. If the outlet nipple breaks off the safety plug could come out, causing elevated radiation levels at the outlet end. Based on these factors the package will be prepared to be oriented so as to impact the safety plug with the edge of the puncture test pin with the center of gravity essentially over the safety plug. The orientation will be re-evaluated during the damage assessment after the free drop test.

The above puncture test orientation was reconsidered as part of the 30-foot free drop damage assessment. The damage to the SPEC-150 resulting from the 30-foot free drop test is described in detail above. The basic theory of inducing the largest increase in radiation levels remains unchanged. Breaking off the outlet nipple remains the intended goal. In order to accomplish this, the puncture test orientation for this package remained unchanged from the original location. The package was oriented to impact the safety plug against the edge of the puncture test pin with the center of gravity essentially over the safety plug. (See Photograph A10)

2.3.2 Puncture Test Description.
The package was dropped from a height of 40 inches measured from the point of impact to the top of the puncture pin (See Photograph A11). The interior temperature of the package, measured at the center of the S-tube of the DU shield, was -30 degrees Farenheight when the test package was removed from the chiller immediately before the test (See Photograph A9). The point of impact was the safety plug at the outlet end of the device as planned (See Photographs A12 & A13).

2.3.3 Damage Assessment.
The bottom edge of hex section at the outer end of the of the safety plug (quick disconnect) fitting dented inward approximately 1/16 inch (See Photograph A14). The outlet nipple remained attached to the device. The quick disconnect fitting "collar" on the safety plug was jammed open. The safety plug was jammed on the outlet nipple.

2.3.4 Unplanned Additional Tests.
Two unplanned additional puncture tests were conducted to further validate the puncture tests that were previous conducted for the SPEC-150. The orientations of the previous puncture tests were selected to be duplicated for the two unplanned additional tests. The same test packaging, SPEC-150 s/n 500, was used for the tests.

Puncture Test #2 - Puncture Test Description & Damage Assessment.
The point of impact was the right side of the package (See Photographs A15 & A17). The package was dropped from a height of 40 inches measured from the bottom of the package to the top of the puncture pin (See Photograph A16). The impact caused a faint impression of the circumference of the puncture pin on the side of the package. The nameplate received superficial scratches. The package right side housing was dented inward approximately 1/16 inch deep (See Photograph A18). The dent is approximately 1-3/4 inch long. The damage was very slight and virtually identical to the damage caused by the previous puncture test with the same orientation.

Puncture Test #3 - Puncture Test Description & Damage Assessment.
The package was dropped from a height of 40 inches measured from the bottom of the package to the top of the puncture pin (See Photograph A20). The point of impact was the end of the lock cap at the lock end of the package (See Photograph A19, A21 & A22). Damage was limited to superficial marks on the surface of the lock cap (See Photographs A23 & A24). The lock cap remained attached to the package and fully functional. The upper right and left flanges were dented inward slightly caused by secondary impact when the package struck the drop target pad. The damage was very slight and virtually identical to the damage caused by the previous puncture test on the lock cap.

2.3.5 Puncture Test Damage Summary
The three separate puncture tests did not produced any significant physical damage.

2.4 Summary of Structural Damage and Evaluation.

The only structural damage caused by the accident condition tests was minor deformation of the protective flange at the outlet end and the slight outward bowing of the outlet end panel. The structural system, DU shield, and all features designed to maintain the radioactive source in the shielded position under hypothetical accident conditions remained intact and performed fully as designed. No design revisions are needed. The SPEC-150 remained safe and structurally sound after the drop test and puncture tests which demonstrates that the design meets the accident condition structural requirements for a Type-B package by a large margin.

3. SHIELDING EVALUATION

3.1 Package Shielding Discussion.
The shield that was fabricated into the test package is the same shield design that is referenced in Certificate of Compliance No. 9263.

3.2 Shielding Evaluation.

Although a pretest survey is not required for the accident condition tests, a pretest survey was made at the surface and 1 meter from the SPEC-150. The pre-test and post-test surface reference readings in Table 2 were made to supplement the structural evaluation. The survey data is presented below. See Sketches of survey locations in Appendix 4. 7.

TABLE 1 Shielding Evaluation SPEC-150

SPEC-150 S/N 500 - Highest Radiation Readings in mR/hr - (Model G-60 source, 26 Curies - Ir-192 on 6/25/97)

Point	Location	Pre-Test Surface Reading 6/25/97	Post-Test Surface Readings 6/26/97	Pre-Test 1 Meter Readings 6/25/97	Post-Test 1 Meter Readings 6/26/97	Post-Test 1 Meter Readings Extrapolated to 150 Curies (background included) EF = 5.822
A	Right Side	14	14	0.7	0.9	5.2
B	Left Side	20	20	0.7	1.0	5.8
C	Top	16	14	0.5	1.0	5.8
D	Bottom	12	11	0.6	0.9	5.2
E	Lock End	12	11	0.7	1.0	5.8
F	Outlet End	10	10	0.4	0.9	5.2

TABLE 2 Shielding Evaluation SPEC-150

SPEC-150 S/N 500 - Surface Reference Radiation Readings in mR/hr - (Model G-60 source, 25.76 Curies - Ir-192 on 6/26/97)

Point	Location	Pre-Test Surface Reading 6/25/97	Post-Test Surface Readings 6/26/97
G	Left Side @ Outlet End	3	3.8
H	Left Side @ Lock End	5	4.4
I	Bottom @ Outlet End	2	2.8
J	Bottom @ Lock End	2	1.8
K	Right Side @ Lock End	4	4.6
L	Right Side @ Outlet End	3	3.4
M	Top @ Lock End	3	2.4
N	Top @ Outlet End	4	4.2

3.3 Shielding Evaluation Summary.

The survey data for the SPEC-150 conservatively shows that the packaging meets the accident condition limit by a large margin. The highest extrapolated reading at 1 meter is less than 1% of the allowable limit of 1 R/hr at 1 meter. The surface reference readings in Table 2 supports the conclusions derived from the structural evaluation of the packaging. The design of the SPEC-150 meets the shielding requirements of a Type B(U) packaging.

4. APPENDICES

Appendix 4.1 **Drop Target and Puncture Pin Drawings**

Target: Drawing 50890-1 Rev (1)
Pin: Drawing 990001, Rev (0)

Appendix 4.2 **Test Procedure**

Test Plan, Rev (1), June 24, 1997

Appendix 4.3 **Survey procedure 7.04, Rev (3)**

Appendix 4.4 **Sketches of Orientations - Free Drop and Puncture**

SPEC-150, Free Drop
SPEC-150, 1st Puncture
SPEC-150, 2nd Puncture (unplanned)
SPEC-150, 3rd Puncture (unplanned)

Appendix 4.5 **SPEC-150 Package Drawings**

15B000, Rev (4)
15B001-3, Rev (0)
15B002A, Rev (3)
15B008, Rev (2)

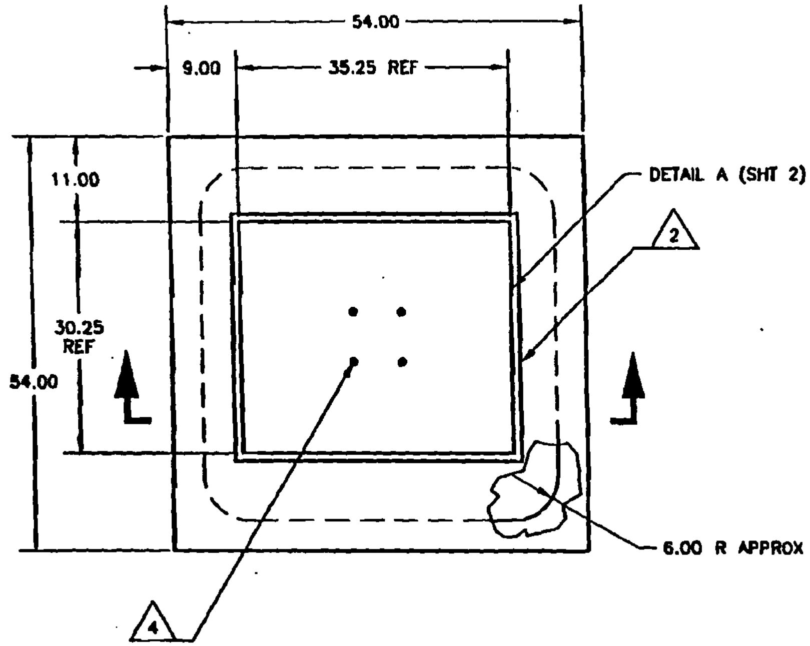
Appendix 4.6 **Photographs**

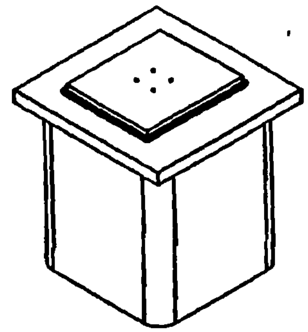
(See List)

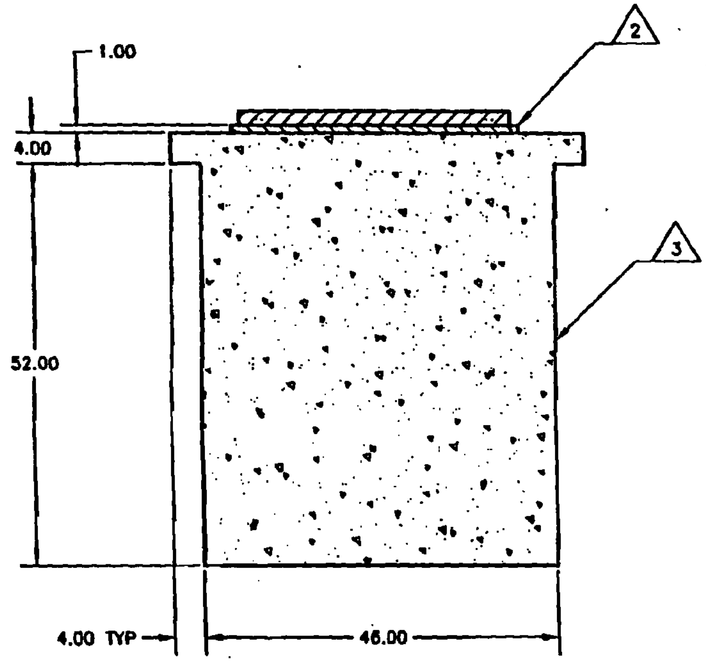
Appendix 4.7 Sketches of SPEC-150 Survey Locations.

Appendix 4.1
Drop Target and Puncture Pin Drawings

REVISIONS				
ZONE	REV	DESCRIPTION	DATE	APPROVED
	1	COMPLETLY REVISED & REDRAWN TO AS BUILT CONFIGURATION.	6/18/97 6/23/97	J FRYER P. WEBER







NOTES:

1. 30.25 x 35.25 x 1.75 CARBON STEEL PLATE
1866.0 CU IN (528 LBS)

2. GARON PRODUCTS INC. 'TIGERGROUT 10602' (7350 PSI MIN). GROUT BED TO EXTEND APPROX .75" BEYOND PLATE ALL AROUND. GROUT BED TO BE APPROX 1.00" THICK. PLATE MUST BE SET HORIZONTAL. MIX GROUT PER MANF. INSTRUCTIONS. APPROX .70 CU FT (78 LBS)

3. 3000 PSI CONCRETE APPROX 67.0 CU FT (9900 LBS)

4. PUNCTURE TEST TARGET MOUNTING HOLES. PLUG HOLES W/ CARBON STEEL DISK (DETAIL B SHT 2) WHEN PUNCTURE TEST TARGET IS NOT ATTACHED.

5. APPROX TOTAL WEIGHT 10506 LBS

UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES TOLERANCES ARE
.XX±.250
.XXX±.010
DO NOT SCALE DRAWING
TREATMENT NA
FINISH NA

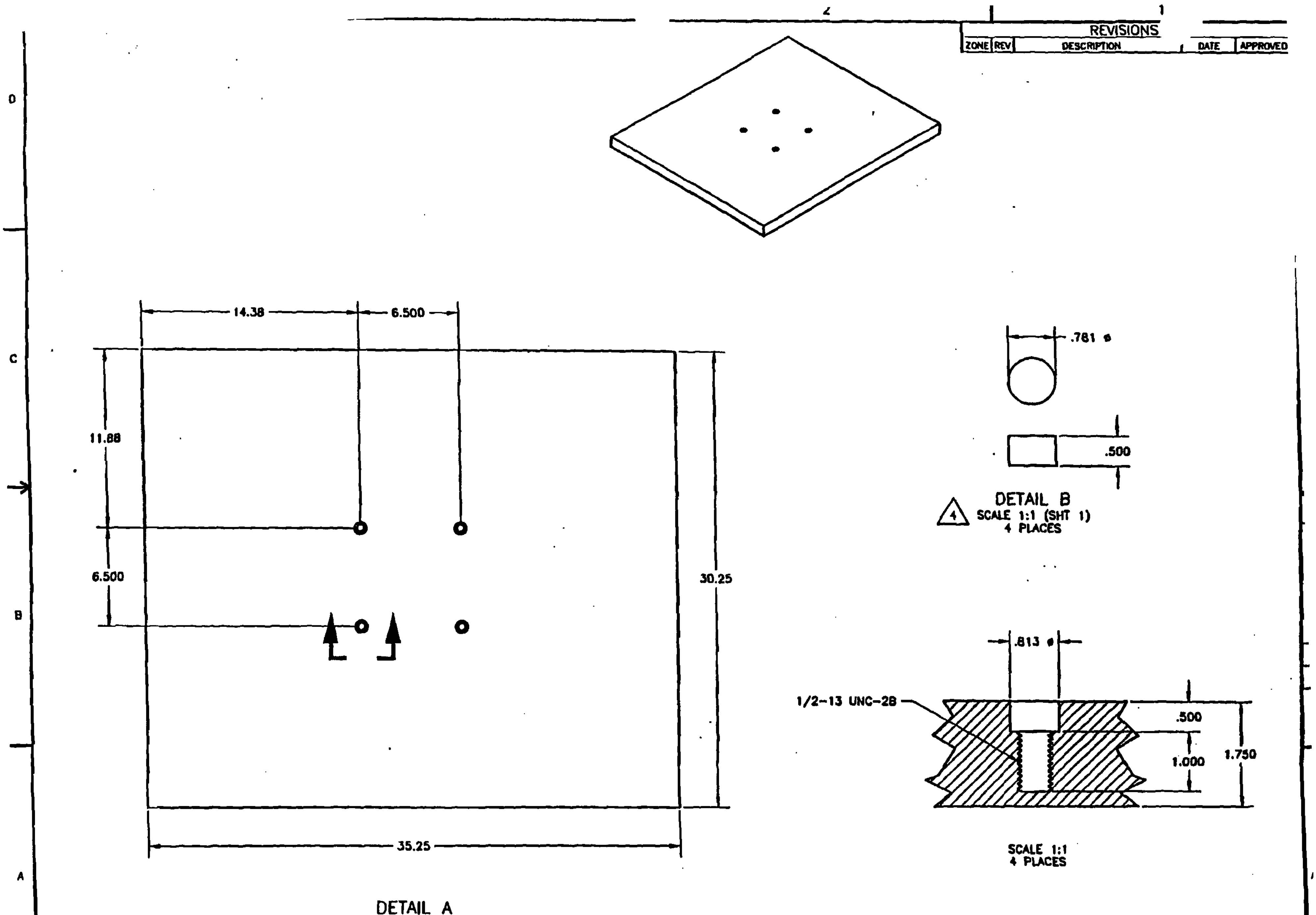
APPROVALS	DATE
DRAWN JAF	6/18/97
CHECKED PW	6/23/97
APPROVED PW	6/23/97
QA CLASS NA	

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

DROP TEST TARGET

SIZE C DWG NO. 50890-1 REV 1

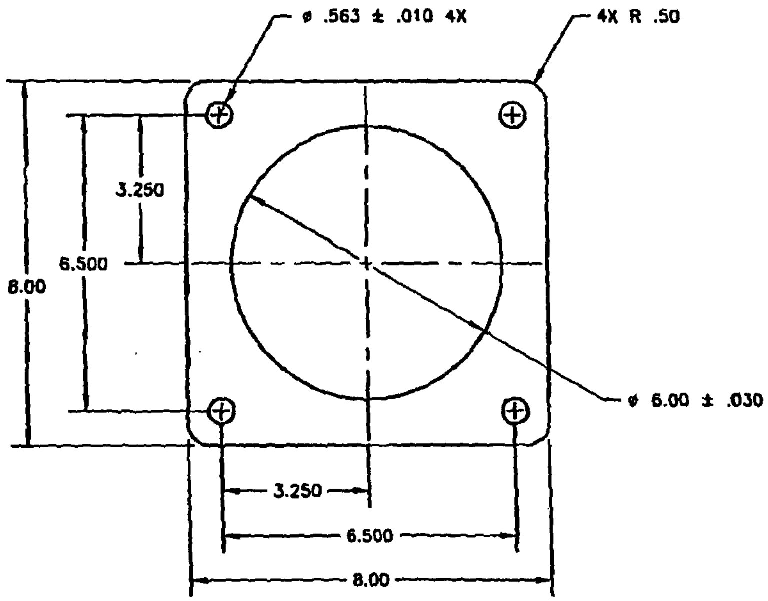
SCALE: 1"=1' 00000242 SHEET 1 of 2

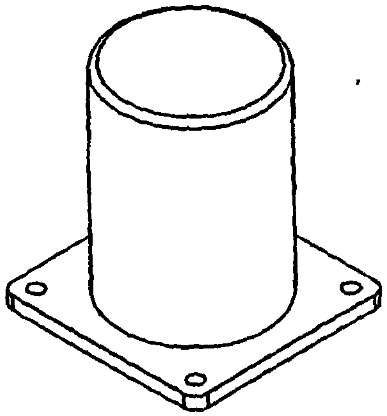


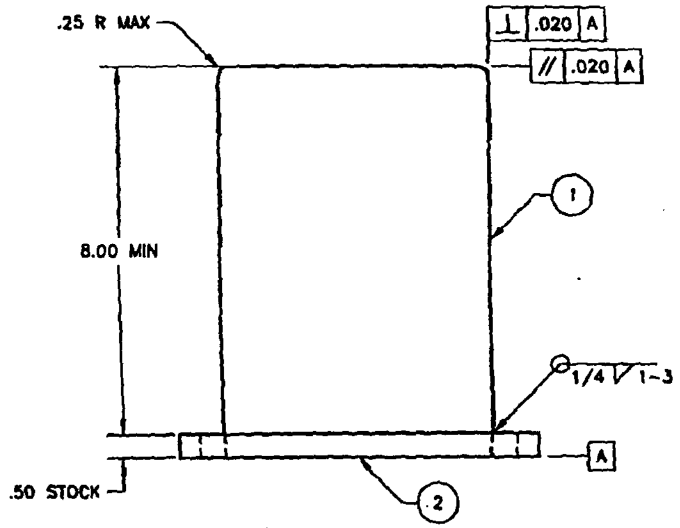
SOURCE PRODUCTION & EQUIPMENT CO INC	SIZE C	DWG NO 50890-1	REV 1
	SCALE: 1"=1"	00000243	SHEET 2 of 2

ZONE	REV	DESCRIPTION	DATE	APPROVED

REVISIONS







.45	SF	QPL		.50 PLATE	CARBON STEEL	2
.67	FT	QPL		6.00 DIA BAR	CARBON STEEL	1
QTY	U/M	MANUF	PART OR IDENTIFYING NUMBER	NOMENCLATURE OR DESCRIPTION	MATERIAL SPECIFICATION	ITEM NO.

PARTS LIST

UNLESS OTHERWISE SPECIFIED DIMENSIONS ARE IN INCHES TOLERANCES ARE .XX±.125 .XXX±.020

DO NOT SCALE DRAWING

TREATMENT NA

FINISH NA

APPROVALS	DATE
DRAWN JAF	6/18/97
CHECKED PW	6-25-97
APPROVED PW	6-25-97
QA CLASS NA	

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

PUNCTURE TEST TARGET

SIZE C | DWG NO. 990001 | REV 0

SCALE: 1/2 | 00000244 | SHEET 1 of 1

Appendix 4.2
Test Procedure

Source Production and Equipment Co., Inc.

10 CFR 71.73 Hypothetical Accident Conditions Tests

Test Data Revision (0)

June 29, 1997

1.0 Test Purpose:

To verify past Hypothetical Accident Conditions Puncture Tests on the Models SPEC-150, SPEC-2T and C-1 Type B (U) Shipping Packages.

2.0 Scope: Type B (U) Packages to be Tested:

2.1 Model SPEC-150, C.O.C. #USA/9263/B(U)

2.2. SPEC Model 2-T, C.O.C. #USA/9056/B(U)

2.3. SPEC Model C-1 and Overpack, C.O.C. #USA/9036/B(U)

2.1.1 Test Sequence:

All three of the 30' drop tests will be performed first, followed by the three Puncture Tests. The order is as follows:

30 ft Drop Test:

2.1.1.1 SPEC-150

2.1.1.2 SPEC 2-T

2.1.1.3 SPEC C-1

Puncture Test:

2.1.1.4 SPEC 2-T

2.1.4.5 SPEC C-1

2.1.4.6 SPEC-150

3.0 References:

3.1 10 CFR 71.73; Hypothetical Accident Conditions; Section (c) Tests. Tests for Hypothetical Accident Conditions must be conducted as follows:

(1) Free Drop

(2) Puncture

3.2 10 CFR 71.51; Additional requirements for Type B packages; Section (a)(2)

3.3 IAEA Safety Series No.6, Mechanical Test, Sections (a) and (b)

4.0 Precautions/Limitations:

4.1 Ensure that emergency procedures, equipment and response are in place.

4.2 Less than 20 curie sources will be used for the tests.

Note: It was necessary to increase the curie activity due to availability of sources to 30 curies maximum single source strength.

4.3 Safety glasses must be worn by all personnel in the test area.

4.4 All personnel must be monitored.

5.0 Test and Recording Equipment Required:

5.1 Drop test tower with test targets

5.2 Release mechanism

5.3 Tape measures (50' and 6' minimum)

5.4 Sufficient extension cords for required electrical usage

5.5 Stopwatch

5.6 Freezer with dry ice

5.7 Thermometers (ambient and freezer)

5.8 Lifting wires

5.9 Video cameras

5.10 Photo cameras

6.0 Testing Procedures:

6.1 SPEC-150 9 Meter Pre-test

6.1.1 Install the source into the device.

6.1.2 Record Device and Sealed Source Data: Take photos.

SPEC-150 Serial Number: 500 Weight: 52 lbs.

Source Serial Number: DA2410 Model Number: G-60 Activity/Date: 26 curies 6/25/97

6.1.3 Record radiation levels at the surface of the device and at 1 meter from the device surface in accordance with procedure 7.04 Rev (3).

Radiation levels at the surface of the device:

Top: 16 mR/hr Bottom: 12 mR/hr Left Side: 20 mR/hr Right Side: 14 mR/hr

Outlet End: 10 mR/hr Lock End: 12 mR/hr

Radiation levels at 1 meter from the surface of the device:

Top: .5 mR/hr Bottom: .6 mR/hr Left Side: .7 mR/hr Right Side: .7 mR/hr

Outlet End: .4 mR/hr Lock End: .7 mR/hr

Note: Readings are highest radiation levels at each side for both the surface and 1 meter.

6.1.4 Determine orientation of sample and provide written justification. (See Justification of Package Orientation for 30-foot Drop and Puncture Tests). Attach the drop wire to the device. Verify the orientation at ground level (bottom/right corner at outlet end).

6.1.5 Chill the device:

Put dry ice on bottom (floor) of the freezer.

Place device on top of ice.

Install thermometer inside the device.

Place ice around (in contact with) the device.

Close the freezer and record time.

Date/Time placed in freezer: 6/25/97 9:30 pm (Note: at 9:25 pm, the device temperature was 76.4 F)

Verified by: ______ Mike Frizell

6.1.6 Verify emergency procedure preparations.

6.2 SPEC-150 9 Meter Drop Test

6.2.1 Record ambient temperature and conditions:

Temperature: 77 F Conditions: Partly cloudy; No wind

Verified by: ______ Mike Frizell Date/Time: 6/26/97 10:10 am

Note: The test was initially set up at 9:06 am. At 9:20 am it began to rain and the test was postponed. The temperature and conditions of the initial set up was: 88 F Partly cloudy; No wind.

6.2.2 Post surveillance personnel.

6.2.3 Start the video.

6.2.4 Remove the frozen SPEC-150 from the freezer. Take photos.

Record device temperature: -85.6 F Record date/time: 6/26/97 10:12 am

Verified by: ______ Mike Frizell Date: 6/26/97

Note: The test was initially set up at 9:06 am. The device was taken from the freezer at 9:14 am. At 9:20 am it began to rain. Test was postponed and the device was returned to the freezer. The initial temperature of the device (before postponement of the test) was -103 F.

6.2.5 Record time elapsed from the removal of the device from the freezer to the time of impact.

6.2.6 Attach the device (drop wire) to the release mechanism.

6.2.7 Verify the orientation at ground level (bottom/right corner at outlet end). Take photos.

Orientation verified by: ______ Mike Frizell Date: 6/26/97

6.2.8 Lift the device to 30 feet (minimum).

6.2.9 Verify the height from the top of the target (steel plate surface) to the lowest point on the device. Take photos.

Height verified by: 30' 4" ______ Mike Frizell Date: 6/26/97

6.2.10 Drop the device.

6.2.11 Record time elapsed from Step 6.2.5 (above)

Time elapsed: 5 minutes 18 seconds Verified by: ______ Pete Weber

6.2.12 Perform the safety survey.

6.2.13 Perform preliminary Part 71 survey.

6.3 SPEC-150 9 Meter Post Test

6.3.1 Record the damage.

There was no damage to the drop test target. Verified by: ______ Mike Frizell

See Post Test Damage Assessment Report, Form OA 11.4, Rev (0) for damage assessment of the SPEC-150 device.

6.3.2 Weigh the device after the drop test.

Weight: 52 lbs. Verified by: Mike Frizell Mike Frizell Date: 6/26/97

6.3.3 Assess the damage to re-evaluate the orientation for the Puncture Test.

Damage Assessed by: Pete Weber Pete Weber Date: 6/26/97

Determine the orientation for the Puncture Test with rational based on damage of the 30' drop test.

Orientation: (See Justification of Package Orientation for 30-foot Drop and Puncture Tests)

Concurrence by: D. Dicharry Donny Dicharry Date: 6/26/97

6.3.4 Attach the drop wire to the device so that when the device is suspended (hanging) the orientation for the Puncture Test will be on the safety plug at the outlet end of the device.

6.3.5 Verify the orientation at ground level.

6.3.6 Install thermometer inside the device and return the device to the freezer.

Time/Date placed in freezer: 11:00 am 6/26/97 Verified by: Mike Frizell Mike Frizell

Note: The device was placed back in the freezer (after the 30' drop) begin its re-chilling. It was removed from the freezer at 11:50 for damage assessment and returned at 12:23 pm. It was then removed again at 1:00 pm for "rigging" the orientation harness and returned again at 1:15 pm.

6.4 SPEC-2T 9 Meter Pre-test

6.4.1 Install the source into the device.

6.4.2 Record Device and Sealed Source Data:

SPEC-2T Serial Number: 1152 Weight: 53.5 lbs. Take photos.

Source Serial Number: DF2501 Model Number G-3 Activity/Date: 17 curies 6/25/97

6.4.3 Record radiation levels at the surface of the device and at 1 meter in accordance with procedure 7.04 Rev (3).

Radiation levels at the surface of the device:

Top: 12 mR/hr Bottom: 22 mR/hr Left Side: 26 mR/hr Right Side: 18 mR/hr

Outlet End: 10 mR/hr Lock End: 12 mR/hr

Radiation levels at 1 meter from the surface of the device:

Top: .5 mR/hr Bottom: .7 mR/hr Left Side: .6 mR/hr Right Side: .4 mR/hr

Outlet End: .8 mR/hr Lock End: .6 mR/hr

Note: Readings are highest radiation levels at each side for both the surface and 1 meter.

6.4.4 Determine orientation of sample and provide written justification. (See Justification of Package Orientation for 30-foot Drop and Puncture Tests). Attach the drop wire to the device. Verify the orientation at ground level (bottom/right corner at the outlet end).

6.5 SPEC-2T 9 Meter Drop Test

6.5.1 Record ambient temperature and conditions:

Temperature: 78 F Conditions: Cloudy and clearing.

Verified by: Mike Frizell Mike Frizell Date/Time: 6/26/97 10:25 am

6.5.2 Post surveillance personnel.

6.5.3 Ensure that the video is running.

6.5.4 Attach the device (drop wire) to the release mechanism.

6.5.5 Verify the orientation at ground level (bottom/right corner at the outlet end). Take photos.

Orientation verified by: ______ Mike Frizell Date: 6/26/97

6.5.6 Lift the device to 30 feet (minimum).

6.5.7 Verify the height from the top of the target (steel plate surface) to the lowest point on the device. Take photos.

Height verified by: 30' 3" ______ Mike Frizell Date: 6/26/97

6.5.8 Drop the device.

6.5.9 Perform the safety survey.

6.5.10 Perform preliminary Part 71 survey.

6.6 SPEC-2T 9 Meter Post Test

6.6.1 Record the damage.

There was no damage to the drop test target. Verified by: ______ Mike Frizell

See Post Test Damage Assessment Report, Form QA 11.4, Rev (0) for damage assessment of the SPEC-2T device.

6.6.2 Weigh the device after the drop test.

Weight: 53.5 lbs. Verified by: ______ Mike Frizell Date: 6/26/97

6.6.3 Assess the damage to determine orientation for the puncture test.

Damage Assessed by: ______ Pete Weber Date: 6/26/97

Determine the orientation for the Puncture Test with rational based on damage of the 30' drop test.

Orientation: (See Justification of Package Orientation for 30-foot Drop and Puncture Tests).

Concurrence by: ______ Donny Dicharry Date: 6/26/97

6.6.4 Go to SPEC-2T Puncture Pre-test to begin preparations.

6.7 SPEC C-1 9 Meter Pre-Test

6.7.1 Install the sources into the device.

6.7.2 Record Device and Sealed Sources Data: Take photos.

SPEC-C-1 Serial Number: 283 Weight: C-1 69 lbs. Total with drum: 89 lbs.

Source Serial Number: CL1002 Model Number: T-5 Activity/Date: 19 curies 6/25/97

Source Serial Number: DA0202 Model Number: G-40T Activity/Date: 22 curies 6/25/97

6.7.3 Put the device into the drum.

6.7.4 Install the lid, ring and bolt.

6.7.5 Weigh the total package (C-1 and drum) and take photos.

6.7.6 Total Package Weight: 89 lbs. Verified by: ______ Tommy Ruiz Date: 6/26/97

Note: Scale s/n 2688; Calibrated on 3/26/97; Next due on 9/26/97.

6.7.7 Record radiation levels at the surface of the package (C-1 and drum) and at 1 meter in accordance with procedure 7.04 Rev (3).

Radiation levels at drum surface:

Top_7 mR/hr   Bottom_4 mR/hr   Side: Quadrant A_10 mR/hr   Quadrant B_7.2 mR/hr

Quadrant C_6 mR/hr   Quadrant D_7.8 mR/hr

Radiation levels at 1 meter from the surface of the drum:

Top_.4 mR/hr   Bottom_.2 mR/hr   Side: Quadrant A_.6 mR/hr   Quadrant B_.4 mR/hr

Quadrant C_.6 mR/hr   Quadrant D_.6 mR/hr

Note: Readings are highest radiation levels at each side for both the surface and 1 meter.

6.7.8 Determine orientation of sample and provide written justification. . (See Justification of Package Orientation for 30-foot Drop and Puncture Tests). Attach the drop wire to the device. Verify the orientation at ground level (flat on top of drum).

6.8 SPEC C-1 9 Meter Drop Test

6.8.1 Record ambient temperature and conditions:

Temperature:_80 F   Conditions:_Partly cloudy and clearing. No wind.

Verified by:_Mike Frizell_ Mike Frizell   Date/Time:_6/26/97 10:42 am

6.8.2 Post surveillance personnel

6.8.3 Ensure that the video is running.

6.8.4 Attach the package (drop wire) to the release mechanism.

6.8.5 Verify the orientation at ground level (flat on top of drum). Take photos.

Orientation verified by:_Mike Frizell_ Mike Frizell   Date:_6/26/97

6.8.6 Lift the package to 30 feet (minimum).

6.8.7 Verify the height from the top of the target (steel plate surface) to the lowest point on the device. Take photos.

Height verified by:_30' 7" Mike Frizell_ Mike Frizell   Date:_6/26/97

6.8.8 Drop the package.

6.8.9 Perform the safety survey.

6.8.10 Perform the preliminary Part 71 survey.

6.9 SPEC C-1 9 Meter Post Test

6.9.1 Record the damage.

There was no damage to the drop test target.   Verified by:_Mike Frizell_ Mike Frizell

See Post Test Damage Assessment Report, Form QA 11.4, Rev (0) for damage assessment of the package (C-1 and drum).

6.9.2 Weigh the package after the drop test.

Weight:_88.5 lbs.   Verified by:_Mike Frizell_ Mike Frizell   Date:_6/26/97

6.9.3 Assess the damage to re-evaluate the orientation for the puncture test.

Damage Assessed by:_Pete Weber_ Pete Weber   Date:_6/26/97

Determine the orientation for the Puncture Test with rational based on damage of the 30' drop test.

Orientation: (See Justification of Package Orientation for 30-foot Drop and Puncture Tests). (Impact on plunger knobs).

Concurrence by:_D Dicharry_ Donny Dicharry   Date:_6/26/97

6.9.4 Go to SPEC C-1 Puncture Pre-test to begin preparations.

6.10 SPEC-2T Puncture Pre-test

6.10.1 Attach the drop wire to the package and confirm the orientation at ground level.

Verified by: Not needed. This is only the pre-test step. Verification of orientation is required at the drop test step. Date: 6/26/97

6.10.2 Install Puncture Test pin to the steel test pad.

6.10.3 Verify that the pin is rigidly mounted to prevent lateral movement or tipping of the pin caused by the device dropping on the pin.

Verified by: Mike Frizell Mike Frizell Date: 6/26/97 11:35 am

6.11 SPEC-2T Puncture Test

6.11.1 Record ambient temperature and conditions:

Temperature: 86 F Conditions: Cloudy

Verified by: Mike Frizell Mike Frizell Date/Time: 2:00 pm (est.) 6/26/97

6.11.2 Ensure that the video is running.

6.11.3 Attach the device (drop wire) to the release mechanism.

6.11.4 Verify the orientation at pin level (impact on safety plug). Take photos.

Orientation verified by: Mike Frizell Mike Frizell Date: 6/26/97

6.11.5 Lift the device to 1 meter (minimum).

6.11.6 Verify height from the top surface of the pin to the lowest point on the device. Take photos.

Height verified by: 40.5" Mike Frizell Mike Frizell Date: 6/26/97

6.11.7 Drop the device.

6.11.8 Perform the safety survey.

6.11.9 Perform preliminary Part 71 survey.

6.11.10 Perform the wipe test.

Wipe CPM: 47 Background: 52 CPM uci. < .0002 uci.

Wipe test performed by: Steve Punch Steve Punch Date: 6/26/97

6.12 SPEC 2-T Puncture Post Test

6.12.1 Record the damage.

See Post Test Damage Assessment Report, Form OA 11.4, Rev (0) for damage assessment of the device.

Verified by: Mike Frizell Mike Frizell

6.12.2 Weigh the device after the Puncture Test.

Weight (Puncture Test #1 53 lbs. Verified by: Mike Frizell Mike Frizell Date: 6/26/97

6.12.3 Test performed by: Kenneth D. Lamm Jr. Dane Pierce Joseph A. Tyler

6.12.4 Test Assessment: Describe damage, weight , dose rate and all other pertinent descriptions ~~and information~~ formation.

Comments: See Post Test Damage Assessment Report, Form OA 11.4, Rev (0) for damage assessment of the device.

6.12.5 Assessment by: ~~signature~~ Kenny Carrington Date: 6/26/97

6.12.6 Test Approval:

President _______________ Date: 6/26/97

QA Manager _______________ Date: 6/26/97

6.13 SPEC C-1 Puncture Pre-test

6.13.1 Attach the drop wire to the device (C-1 without drum) and confirm the orientation at ground level. (Note: The C-1 container must remain placed inside the overpack (drum) for the Puncture Test).

This was revised after the 30' drop damage assessment. The C-1 was Puncture Tested as a stand alone package without the overpack drum. The point of impact determined was on the impact on plunger knobs.

Note: The information in 6.14U is relative to the Unplanned Test of the SPEC C-1 where setup procedures for the test were established. 6.14U was not included in the initial test procedure. The "U" designates "Unplanned"

6.14U UNPLANNED TEST SPEC C-1 Puncture Test

Description: **C-1 serial number 88; Stand alone; No drum overpack**

Weight: 67 lbs. (plus added 5 3/4 lb. lead weight; Total 72 3/4 lbs.

Point of Impact: On the Plunger Knobs

6.14u.1 Record ambient temperature and conditions:

Temperature: n/a Conditions: n/a

Verified by: n/a Mike Frizell Date/Time: n/a

6.14u.2 Inspect pin and drop target. Verify that the drop target and pin have not moved as a result of the previous test.

(Note: There was no previous Puncture Test in which to verify that the target and pin did not move as a result of a previous test. The pin was inspected prior to the Puncture Test of C-1 #88. This purpose of this test (puncture of #88) was to determine the setup and procedure for the test (since this point of impact had never been selected or performed in previous tests). This pre-determination was prompted by the damage assessment of the 30' drop test. The pin was then re-inspected prior to the following Puncture Test for C-1 serial number 283.

Verified by: n/a Mike Frizell Date: n/a

6.14u.3 Ensure that the video is running.

n/a. Unplanned test.

6.14u.4 Attach the package (drop wire) to the release mechanism.

6.14u.5 Verify the orientation at pin level (impact on plunger knobs). Take photos.

Orientation verified by: _______________ Mike Frizell Date: 6/26/97

6.14u.6 Lift the package to 1 meter (minimum).

6.14u.7 Verify height from the top surface of the pin to the lowest point on the package. Take photos.

Height verified by: 40.250" [signature] Mike Frizell Date: 6/26/97

Note: Measurement was taken from the lowest part of the package, not from the plunger knobs.

6.14u.8 Drop the package.

6.14u.9 Perform the safety survey.

N/A. Dummy sources were installed in the C-1 #88

6.14u.10 Perform the preliminary part 71 survey.

N/A. Dummy sources were installed in the C-1 #88

6.14u.11 Perform the wipe test.

Wipe CPM: n/a Background: n/a uci. n/a

Leak test performed by: [signature] N/A Steve Punch Date: n/a

KC STEVE PUNCH INADVERTENTLY SIGNED THE BLANK. KC 8/18/97

6.14 SPEC-C-1 Puncture Test (Planned Test C-1 #283)

6.14.1 Record ambient temperature and conditions:

Temperature: 86 F Conditions: Cloudy; Winds at approximately 5 mph

Verified by: [signature] Mike Frizell Date/Time: 1:45 pm (est.) 6/26/97

6.14.2 Inspect pin and drop target. Verify that the drop target and pin have not moved as a result of the previous test.

Verified by: [signature] Mike Frizell Date: 6/26/97 1:35 pm

6.14.3 Ensure that the video is running.

6.14.4 Attach the package (drop wire) to the release mechanism.

6.14.5 Verify the orientation at pin level (impact on plunger knobs). Take photos.

Orientation verified by: [signature] Mike Frizell Date: 6/26/97

6.14.6 Lift the package to 1 meter (minimum).

6.14.7 Verify height from the top surface of the pin to the lowest point on the package. Take photos.

Height verified by: 40.250" [signature] Mike Frizell Date: 6/26/97

Note: Measurement was taken from the lowest part of the package, not from the plunger knobs.

6.14.8 Drop the package.

6.14.9 Perform the safety survey.

6.14.10 Perform the preliminary part 71 survey.

6.14.11 Perform the wipe test.

Wipe CPM: 48 Background: 52 CPM uci. < .0002 uci.

Leak test performed by: [signature] Steve Punch Date: 6/26/97

6.15 SPEC C-1 Puncture Post Test

6.15.1 Record the damage.

See Post Test Damage Assessment Report, Form QA 11.4, Rev (0) for damage assessment of the device (C-1, stand alone; No drum).

Verified by: [signature] Mike Frizell

6.15.2 Weigh the device after the Puncture Test.

Weight: 68.5 lbs (C-1 container only; No drum). Verified by: [signature] Mike Frizell Date: 6/26/97

6.15.3 Test performed by: [signatures]

6.15.4 Test Assessment: Describe damage, weight, dose rate and all other pertinent descriptions and information.

Comments: See Post Test Damage Assessment Report, Form QA 11.4, Rev (0) for damage assessment of the device (C-1, stand alone; No drum)

6.15.5 Assessment by: [signature] Kenny Carrington Date: 6/26/97

6.15.6 Test Approval:

President [signature] Date: 6/26/97

QA Manager [signature] Date: 6/26/97

6.16 SPEC-150 Puncture Pre-test

6.16.1 None. All Pre-test arrangements were performed at the 9 meter post test to allow the device to be re-installed into the freezer.

6.17 SPEC 150 Puncture Test Number One (Planned Test)

6.17.1 Record ambient temperature and conditions:

Temperature: 82 F Conditions: Cloudy; No wind.

Verified by: [signature] Mike Frizell Date/Time: 2:30 pm (est.) 6/26/97

6.17.2 Inspect pin and drop target. Verify that the drop target and pin have not moved as a result of the previous test.

Verified by: [signature] Mike Frizell Date: 6/26/97

6.17.3 Ensure that video is running.

6.17.4 Remove the device from the freezer.

Record device temperature: -32.8 F Record date/time: 6/26/97 2:12 pm

Verified by: [signature] Mike Frizell Date: 6/26/97

6.17.5 Record time elapsed from the removal of the device from the freezer to the time of impact.

6.17.6 Attach the device (drop wire) to the release mechanism.

6.17.7 Verify the orientation at pin level (impact on safety plug). Take photos.

Orientation verified by: [signature] Mike Frizell Date: 6/26/97

6.17.8 Lift the device to 1 meter (minimum).

6.18.9 Verify height from the top surface of the pin to the lowest point on the device. Take photos.

Height verified by: 40.250" [signature] Mike Frizell Date: 6/26/97

6.17.10 Drop the device.

6.17.11 Record elapsed time from step 6.17.5 (above)

Time elapsed: 11 minutes 11 seconds Verified by: [signature] Pete Weber

6.17.12 Perform the safety survey.

6.17.13 Perform the preliminary Part 71 survey.

6.17.14 Perform the wipe test.

Wipe CPM: 52 Background: 52 CPM uci. <.0002 uci.

Leak test performed by: Steve Punch Steve Punch Date: 6/26/97

UNPLANNED TEST SPEC 150 Puncture Test Number Two

6.17.15 Inspect pin and drop target. Verify that the drop target and pin have not moved as a result of the previous test.

Verified by: Mike Frizell Mike Frizell Date: 6/26/97

6.17.16 Ensure that video is running.

6.17.17 Attach the device (drop wire) to the release mechanism.

6.17.18 Verify the orientation at pin level (right side). Take photos.

Orientation verified by: Mike Frizell Mike Frizell Date: 6/26/97

6.17.19 Lift the device to 1 meter (minimum).

6.17.20 Verify height from the top surface of the pin to the lowest point on the device. Take photos.

Height verified by: 40.250" Mike Frizell Mike Frizell Date: 6/26/97

6.17.21 Drop the device.

6.17.22 Perform the safety survey.

6.17.23 Weigh the device after the drop.

Weight: 52 lbs. Verified by: Mike Frizell Mike Frizell Date: 6/26/97

6.17.24 Perform the wipe test.

Wipe CPM: 64 Background: 52 CPM uci. <.0002 uci.

Leak test performed by: Steve Punch Steve Punch Date: 6/26/97

UNPLANNED TEST SPEC 150 Puncture Test Number Three

6.17.25 Inspect pin and drop target. Verify that the drop target and pin have not moved as a result of the previous test.

Verified by: Mike Frizell Mike Frizell Date: 6/26/97

6.17.26 Ensure that video is running.

6.17.27 Attach the device (drop wire) to the release mechanism.

6.17.28 Verify the orientation at pin level (directly on lock cap). Take photos.

Orientation verified by: Mike Frizell Mike Frizell Date: 6/26/97

6.17.29 Lift the device to 1 meter (minimum).

6.17.30 Verify height from the top surface of the pin to the lowest point on the device. Take photos.

Height verified by: 40.250" Mike Frizell Mike Frizell Date: 6/26/97

6.17.31 Drop the device.

6.17.32 Perform the safety survey.

6.17.33 Weigh the device after the second Puncture Test drop.

Puncture #2 Weight: 52 lbs. Verified by: Mike Frizell Mike Frizell Date: 6/26/97

6.17.34 Perform the wipe test.

Wipe CPM: 52 Background: 52 CPM uci. <.0002 uci.

Leak test performed by: Steve Punch Steve Punch Date: 6/26/97

6.18 SPEC-150 Puncture Post Test

6.18.1 Record the damage.

See Post Test Damage Assessment Report, Form OA 11.4, Rev (0) for damage assessment of the device.

Verified by: Mike Frizell ______________________ Mike Frizell

6.18.2 Weigh the device after the third Puncture Test drop.

Puncture #3 Weight: 52 lbs. Verified by: Mike Frizell ____ Mike Frizell Date: 6/26/97

Scale serial number: 2697

6.18.3 Test performed by: Kenneth N Carrington Dave Picone Joseph A Myer

6.18.4 Test Assessment: Describe damage, weight, dose rate and all other pertinent descriptions and information.

Comments: See Post Test Damage Assessment Report, Form OA 11.4, Rev (0) for damage assessment of the device.

6.18.5 Assessment by: Kenneth N Carrington ____ Kenny Carrington Date: 6/27/97

6.18.6 Test Approval:

President ______________________ Date: 6/27/97

QA Manager Mike Frizell ______________________ Date: 6/27/97

6.19 Package Test Certification

6.19.1 This is to certify that the preparations and tests for both the 30' Drop Test and the Puncture Test were performed in accordance with this procedure.

SPEC-150:	President		Date 6/27/97
	QA Manager	Mike Frizell	Date 6/27/97
SPEC-2T:	President		Date 6/27/97
	QA Manager	Mike Frizell	Date 6/27/97
SPEC C-1:	President		Date 6/27/97
	QA Manager	Mike Frizell	Date 6/27/97

6.20 Prepare Test Report

H:\PROJECTS\RETEST\PUNCDATA.WPD

Appendix 4.3
Survey Procedure 7.04

PROCEDURE

7.04 TESTING SURVEY PROCEDURE

Prepared By: Joe Fryer

APPROVAL

Kenneth N Carrington Jr 6/25/97
Kenny Carrington, Test Coordinator

Pete Weber 6/25/97
Pete Weber, General Manager

Stephen Punch 6-25-97
Steve Punch, Asst. Radiation Safety Officer

C. M. Frizell 6/25/97
Mike Frizell, Quality Assurance Manager

Revision: (3) 06/25/97

Rev No.	Revision	Revised By/Date	Checked By/Date
0	Initial Procedure	JF 06/23/97	KC 06/25/97

QUALITY ASSURANCE PROGRAM
PROCEDURE, INSTRUCTION AND SPECIAL PROCESS
DOCUMENT CHANGE EVALUATION RECORD

Document Name: TESTING SURVEY PROCEDURE

QA Document 7.04 Prepared By: J. FRYER

Description of Change Requested: INITIAL RELEASE.

Reason for Change Requested: CREATED FOR PACKAGE TESTING

Review Conducted By K. CARRINGTON / M. FRIZELL

Date: 6 25 97

Does this change conflict with the QA Program?

Yes______No ✓ QA Manager [signature] Date ______

Document Review Recommendation: [✓] Approve [ ] Reject [ ] Amend

Comments:______

Change Authorization Signature:______Date:______

Document Revision Number:______ DOcument Effective Date:______

Docuemnt Approval Signature:______ Date:______

Distribution: Form QA 6.1 Revised:______ Date:______

Dept:______ Initials:______ Date:______

Dept:______ Initials:______ Date:______

Dept:______ Initials:______ Date:______

NOTE: Initial to verify receipt, indoctrination and understanding of the revised document. As applicable, the document must remain available for use at the appropiate work station(s).

Distribution Complete:______ Date:______

PROCEDURE
7.04 TESTING SURVEY PROCEDURE
Revision (3)

1.0 Purpose:

To define the radiation survey of packages and safety procedures for use during testing of Type B packages for Normal Conditions of Transport (10CFR71.71) and Hypothetical Accident Conditions (10CFR71.73) tests.

2.0 Scope

This procedure applies to all potentially destructive tests performed during the testing of existing Type B packages or prototype packages for Type B status. These tests may include:

2.1 10CFR71.71(c)(7) Free Drop (1.2 m)
2.2 10CFR71.71(c)(8) Corner Drop (wood or fiberboard packages)
2.3 10CFR71.71(c)(9) Compression
2.4 10CFR71.71(c)(10) Penetration
2.5 10CFR71.73(c)(1) Free drop (30 m)
2.6 10CFR71.73(c)(2) Puncture

3.0 References

3.1 SPEC Procedure 6.09 Radiation Emergency Procedure
3.2 SPEC Procedure 6.10 Survey Meter Calibration
3.3 10 CFR Part 71 sections .51, .71, .73

4.0 Definitions/Acronyms

4.1 CFR Code of Federal Regulations
4.2 RSO Radiation Safety Officer

5.0 Requirements

5.1 Equipment

5.1.1 Calibrated and properly operating survey meter (with remote probe)
5.1.2 One meter stick, QA controlled
5.1.5 Permanent marker
5.1.6 Safety Glasses

5.2 Documentation

5.2.1 QA 12.1.1 Survey Instrument Calibration Certificate
5.2.2 Test Package Radiation Survey Report (Attachment #1)
5.2.3 Procedure 6.09; Radiation Emergency Response

6.0 Safety

6.1 Potential Hazards

6.1.1 High Radiation

Since these tests are potentially destructive and designed to verify package integrity,

6.1.1.1 The area shall be considered as a high radiation area after each test until otherwise demonstrated.
6.1.1.2 All non-essential personnel shall be removed from the area prior to executing each test.
6.1.1.3 All essential personnel must be monitored with a dosimeter and a TLD or film badge.
6.1.1.4 Procedure #6.09, Radiation Emergency Procedure, with all requirements (i.e. handling equipment, survey meters, response, etc) must be in effect.
6.1.1.5 The RSO or his assistant or designate will be responsible for implementation of the procedure, if necessary. All individuals with responsibility in the emergency response will be familiar with the procedure commensurate with their involvement in the action.

6.1.2 Flying Debris

6.1.2.1 All personnel must stand clear during impacts because of the potential of flying debris.
6.1.2.2 Safety glasses must be worn by all personnel in the test area during the drop tests.

7.0 Procedure

7.1 Pretest:

7.1.1 Record the following data on the Test Package Survey Report in the appropriate sections:

Date
Test Performed

Package Description
Package Model & Serial Number(s)
Radionuclide
Source Model & Serial Number
Source Activity
Survey Meter Mfg, Model & Serial Number(s)
Time test was executed (for calculation purposes if needed)

7.1.2 Locate the highest radiation level at the surface for each side of the test package using the sealed source assembly that will be installed for the test. Make a written note if the probe was flush against the surface of the device or if it was necessary to position the probe (detector) away from the surface due to the configuration of the detector holder.

7.1.3 Mark each surface of the package at the EXACT location of the highest reading. Trace the outline of the probe on the package in order to relocate its exact location after the test is performed.

7.1.4 For cylindrical packages, mark each quadrant at the location of the highest reading. Trace the outline of the probe on the package in order to relocate its exact location after the test is performed

7.1.5 Assign each reference mark a unique alphabetical designation.

7.1.6 Record the highest radiation level found for each reference mark in the "Initial" column for Surface Survey on the Test Package Radiation Survey Report Form (Attachment# 1). The levels recorded will be the actual (uncorrected) radiation readings.

7.1.7 Once the highest radiation level is located (and recorded) at the surface of the package, determine the highest radiation level at one meter extending outward from THAT SURFACE POINT by projecting the radiation "beam" from the point on the surface to the sealed source inside the package. (The one meter stick will facilitate easier survey method).

7.1.8 Record the one meter readings in the "Initial" column for One Meter Survey on the Test Package Radiation Survey Report Form (Attachment# 1). The levels recorded will be the actual (uncorrected) radiation readings.

7.1.9 For recessed areas of the packages (i.e. SPEC-2T and SPEC-150 outlet and lock ends), record surface and one meter readings using the end of the flanges as the surface. Mark the end plate (inside the flange) as the reference point.

7.1.10 In addition to the highest radiation level measured at the surface of the package, select two random points, uniformly spaced, and measure the radiation level at these randomly selected points. Mark and record their locations and levels as described in section 7.1.3, 7.1.5 7.1.6. (Note: it is not intended to locate any particular range of levels, only what the level is. This will be used after the drop test to determine if any changes in shield or source location has occurred).

7.1.11 Photograph each surface of the marked up package.

7.2 Post test:

7.2.1 Designate one individual of the test team to be responsible for performing the safety survey after each drop test. The individual shall be trained in the operation of survey meters and survey procedures. After each destructive test is performed, only the survey meter operator may approach the test package until the safety survey of the test site is completed.

7.2.2 Immediately (or within seconds) after the test package impacts the target, the survey meter operator is to begin a safety survey of the test site. No other personnel are to advance toward the test package until the area safety survey is complete. After a preliminary determination of the site radiation level, the survey meter operator may request assistance (if necessary) in rotating the package from its resting position in order to survey the surface in contact with the ground. The assistant shall leave the immediate area after the package has been rotated. The survey meter operator will then re-survey the test site. If the radiation levels in the test site area are below 100 mR/hr up to a distance of 1 meter from the package the area is determined to be "all clear".

7.2.4 If the radiation level exceeds 100 mR/hr within 32' of the package, a radiation emergency response shall be implemented and controlled by the RSO. All tests will be discontinued until all radiation levels are deemed safe by the RSO. (Note: an unshielded 20 curie source at 32' provides a dose rate of 100 mR/hr).

7.2.5 When the "all clear" signal is given the package may be moved for damage assessments, photographs, etc. and final survey profiles.

7.2.6 The survey meter operator will resurvey the package at each of the reference marks identified in steps 7.1.3 and 7.1.4 and record in the "Final" column on the Surface Survey section on the Test Package Radiation Survey Report Form (Attachment# 1). . The survey meter must be held on each point in the same orientation as the initial survey. Enter the actual (uncorrected) radiation levels found for each reference mark on the survey report for the surface locations.

7.2.7 Record the highest radiation levels of each side of the package at one meter and record in the "Final" column on the One Meter Survey section on the Test Package Radiation Survey Report Form (Attachment# 1). Readings are the actual (uncorrected) radiation levels.

8.0 Documentation

8.1 Test Package Radiation Survey Report (Attachment #1)

H:\PROJECTS\RETEST\TEST-1.WPD

SOURCE PRODUCTION & EQUIPMENT **TEST PACKAGE SURVEY REPORT**

TEST PERFORMED: ______________________________ DATE: ____________

PACKAGE: ______________________________ MODEL# ____________ SERIAL # ____________

SOURCE: ____________ ACTIVITY: ____________ MODEL# ____________ SERIAL # ____________

SURVEY METER(1): ______________________________ MODEL# ____________ SERIAL # ____________

SURVEY METER(2): ______________________________ MODEL# ____________ SERIAL # ____________

METER(1) CALIBRATION(Q/A): ____________ METER(2) CALIBRATION(Q/A): ____________

SURVEY METER OPERATOR: ______________________________ TRAINING VERIFIED(Q/A): ____________

HIGH RADIATION AREA: ________ (FT RADIUS FROM TARGET) VERIFICATION OF ABOVE(Q/A): ____________

TIME OF EXECUTION: ________ TIME AREA SURVEY COMPLETE: ________

NOTE: READINGS ARE ACTUAL (UNCORRECTED) RADIATION LEVELS.

SURFACE SURVEY				1 METER SURVEY			
LOCATION	INITIAL(MR/HR)	FINAL(MR/HR)	CHANGE	LOCATION	INITIAL(MR/HR)	FINAL(MR/HR))	CHANGE
A				A			
B				B			
C				C			
D				D			
E				E			
F				F			
G				G			
H				H			
I				I			
J				J			
K				K			
L				L			
M				M			

Appendix 4.4

Sketches of Orientations - Free Drop and Puncture

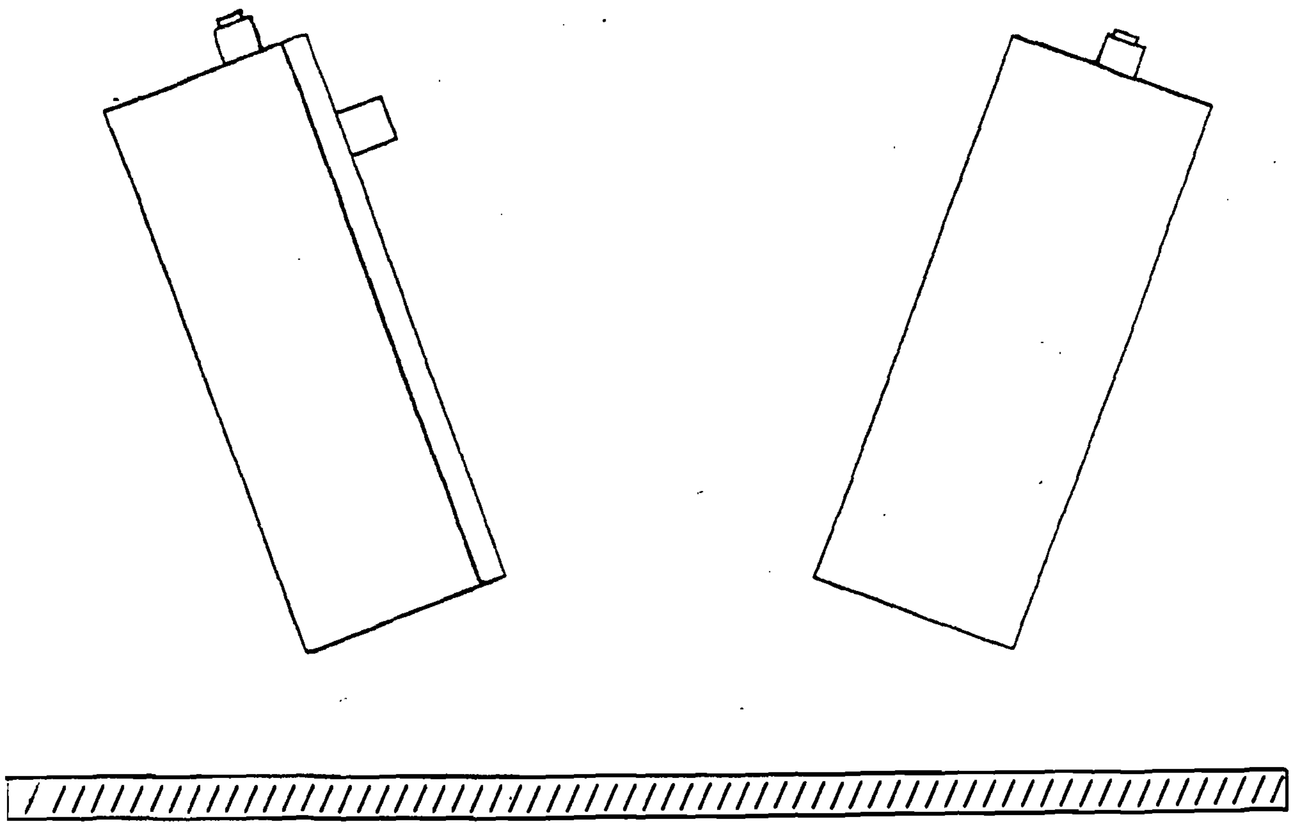
SPEC-150, Free Drop

SPEC-150, 1st Puncture

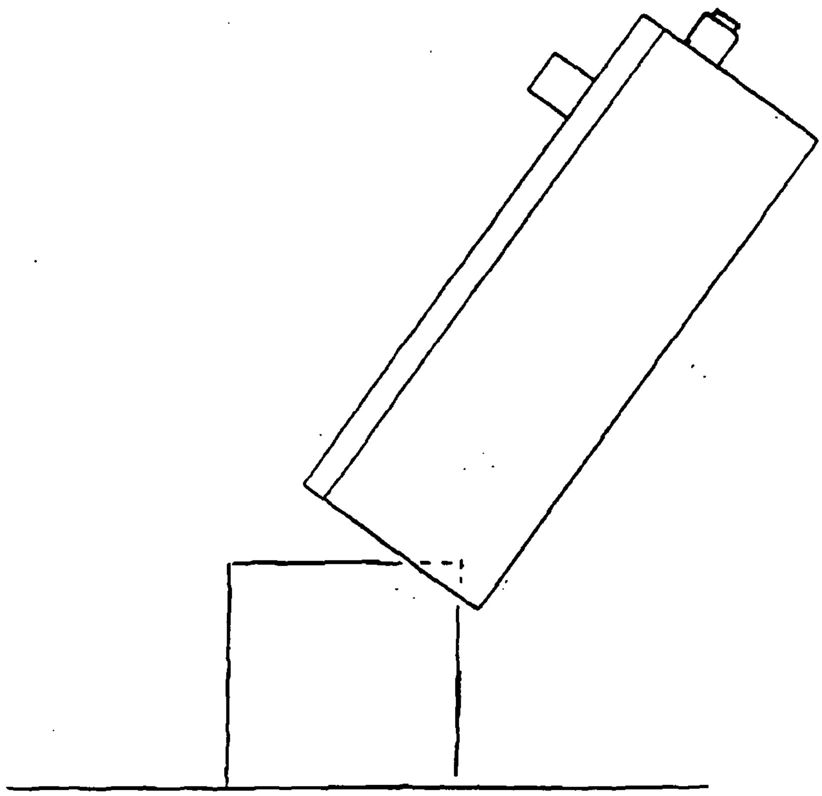
SPEC-150, 2nd Puncture (unplanned)

SPEC-150, 3rd Puncture (unplanned)

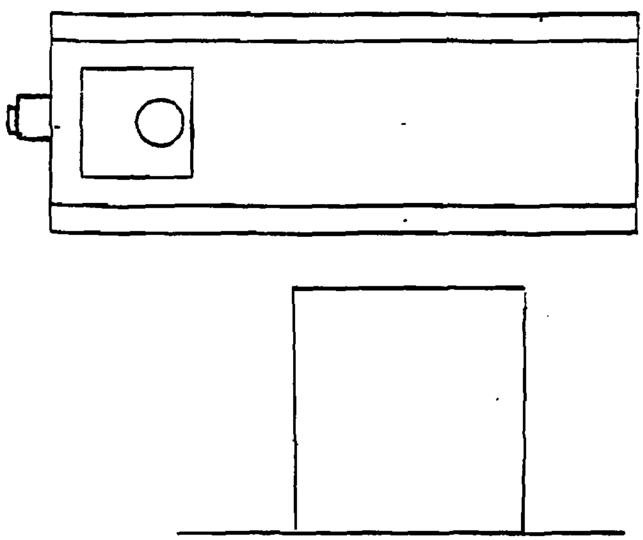
SPEC-150 Serial Number 500
30' Drop Test 6/26/97
Point of Impact: Right/Bottom Corner at Outlet End

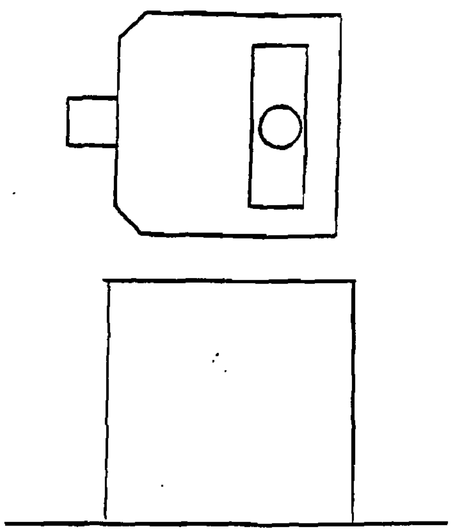


SPEC 150 Serial Number 500
Puncture Test 6/26/97
Point of Impact: Safety Plug

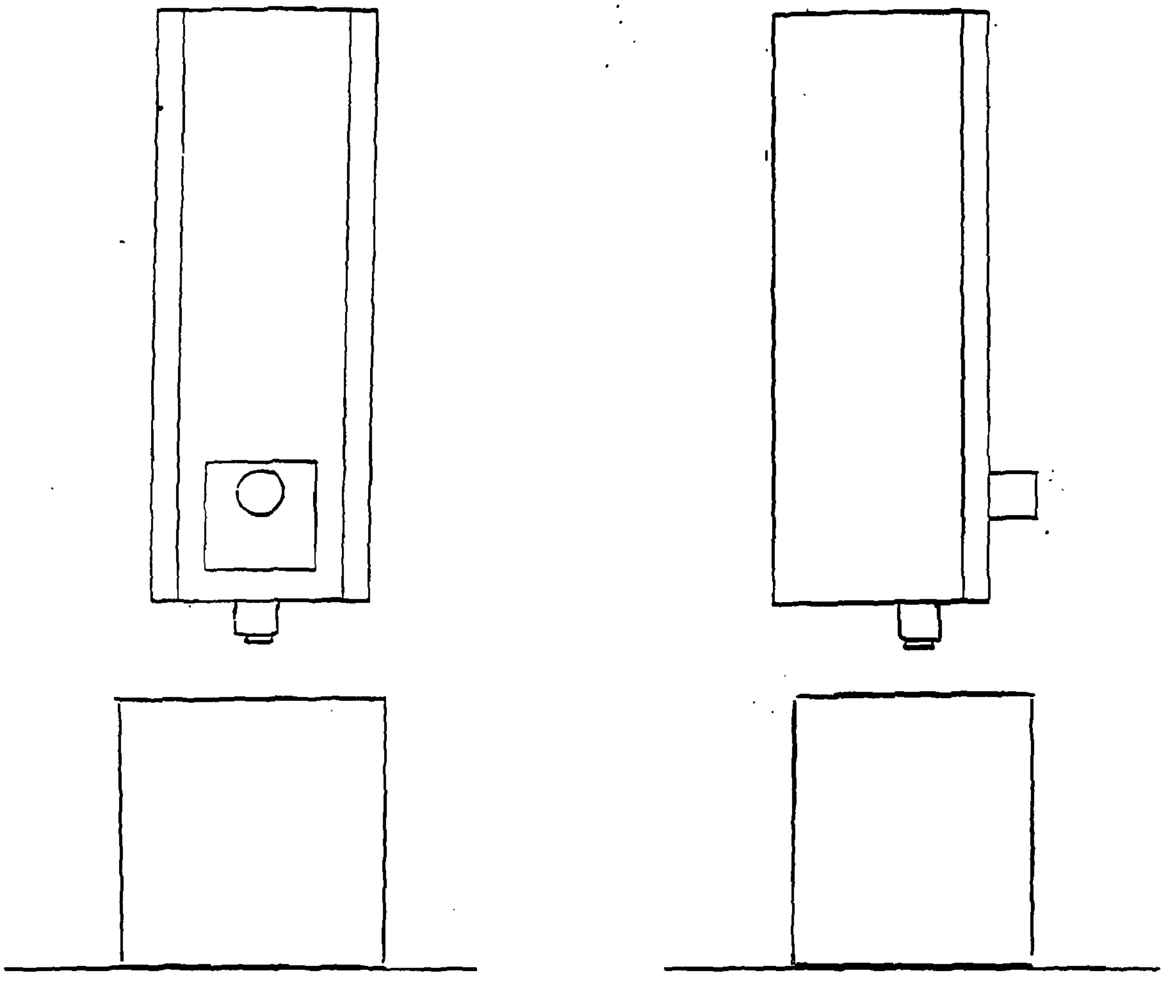


SPEC 150 Serial Number 500
Puncture Test 6/26/97
Point of Impact: Right Side





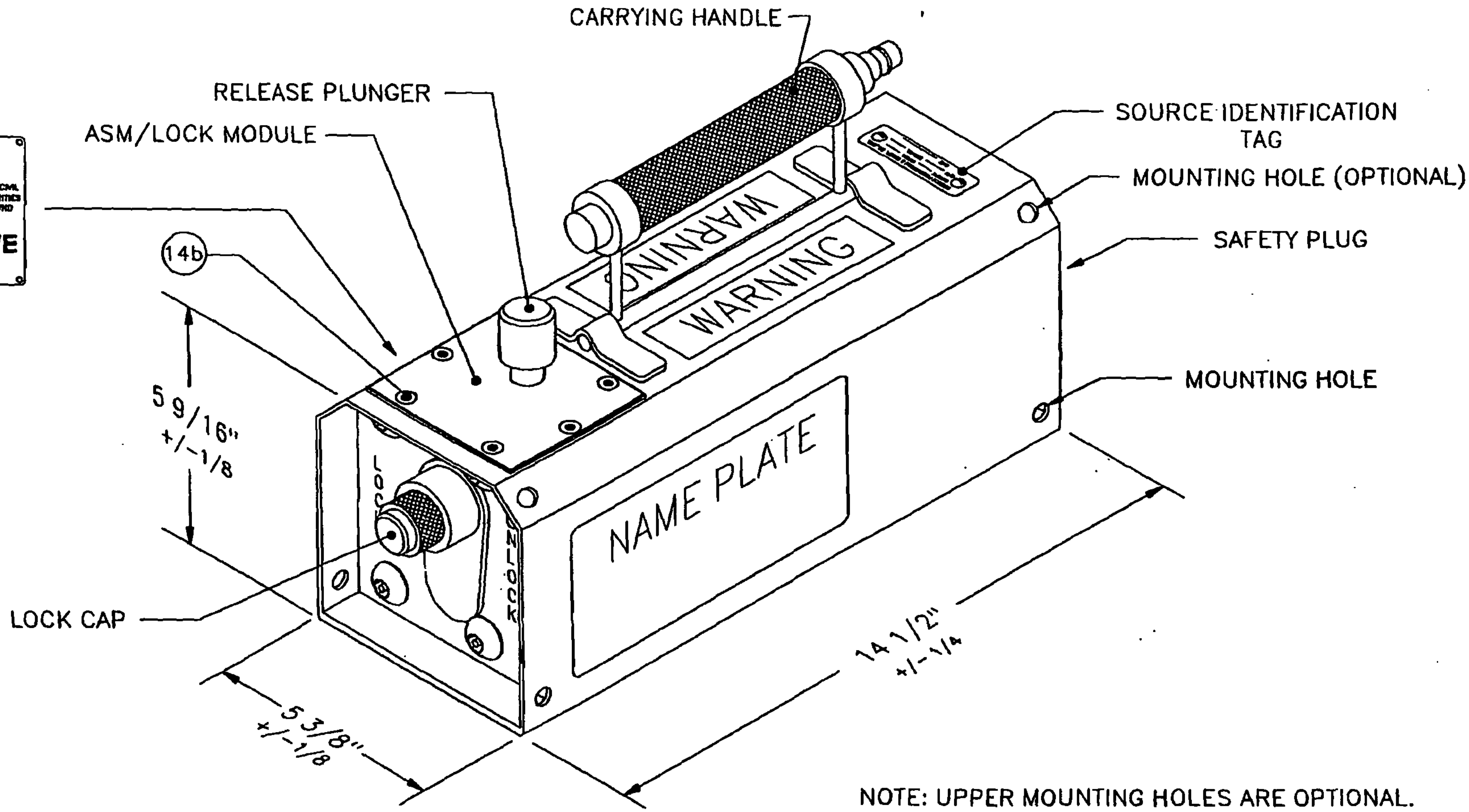
SPEC 150 Serial Number 500
Puncture Test 6/26/97
Point of Impact: Directly on Lock Cap



Appendix 4.5

SPEC-150 Package Drawings

SPEC-150: 15B000, Rev (4)
15B001-3, Rev (0)
15B002A, Rev (3)
15B008, Rev (2)



NOTE: UPPER MOUNTING HOLES ARE OPTIONAL.

STATEMENTS OF FABRICATION:

1. ALL THERMAL METAL JOINING, "TMJ", STRUCTURAL JOINTS ARE JOINED CONTINUOUSLY ALONG THE ENTIRE LENGTH OF EACH JOINT.
2. ALL THERMAL METAL JOINING, "TMJ", OF STRUCTURAL JOINTS ARE PERFORMED IN ACCORDANCE WITH ASME SECTION IX.
3. EXTERIOR STRUCTURAL "TMJ" JOINTS ARE INSPECTED IN ACCORDANCE WITH ASTM E-165. THE ACCEPT/REJECT CRITERIA IS IN ACCORDANCE WITH ASME SECTION VIII, DIVISION 1.
4. INTERIOR STRUCTURAL "TMJ" JOINTS ARE VISUALLY INSPECTED IN ACCORDANCE WITH ASME SECTION V, ARTICLE 9.

TOLERANCES (EXCEPT AS NOTED)	REVISIONS #	DATE	DWG BY	APP'D BY
DECIMAL +/-	1	3-6-95	SB	RDD
	2	4-5-95	SB	RDD
FRACTIONAL +/- NOTED	3	4-14-95	SB	RDD
ANGULAR +/-	4	9-21-95	SB	RDD
	5			

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

SPEC-150 TYPE B(U) PACKAGE
ISOMETRIC VIEW

DRAWN BY	SCALE	MATERIAL
S. BYRD	1/4" = 1"	N/A
CHK'D BY	DATE 12-18-94	DRAWING NO. 15B000 REV.(4)
Q.A. CLASS. N/A	APPV'D BY R.D.DICHARRY	

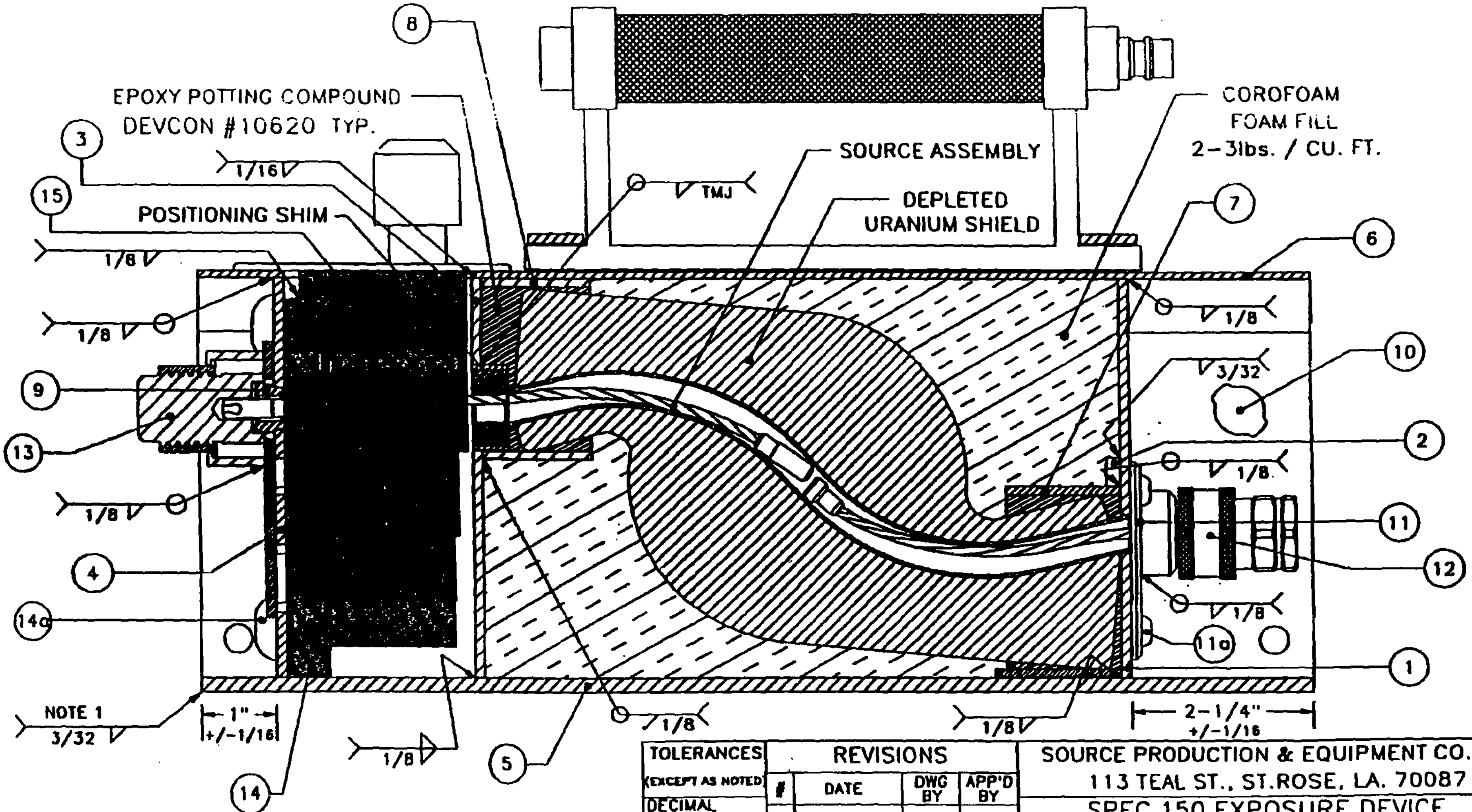
DESCRIPTION	MATERIAL	GRADE	ASTM#	DIMENSIONS
1. OUTLET END PLATE	TITANIUM	GRADE 2	B265	5.200" WIDE X 5.275" HIGH X 1/8" THICKNESS
2. OUTLET END DOUBLER PLATE	TITANIUM	GRADE 2	B265	3/4" X 2-5/8" HIGH X 3/16" THICKNESS
3. INNER BULKHEAD PLATE	TITANIUM	GRADE 2	B265	5.200" WIDE X 5.275" HIGH X 1/8" THICKNESS
4. HOUSING LOCK END PLATE	TITANIUM	GRADE 2	B265	5.200" WIDE X 5.275" HIGH X 1/8" THICKNESS
5. BOTTOM PLATE	TITANIUM	GRADE 2	B265	5.200" WIDE X 14-1/2" LONG X 3/16" THICKNESS
6. HOUSING COVER	TITANIUM	GRADE 2	B265	U-SHAPED, 5-15/32" HIGH X 5-3/8" WIDE X 14-1/2" LONG X 3/32" THICKNESS
7. OUTLET END PLATE SUPPORT CUP	TITANIUM	GRADE 2	B265	U-SHAPED, 2-17/32" HIGH X 2-1/16" WIDE X 1-1/2" LONG X 1/8" THICKNESS
8. INNER BULKHEAD SUPPORT CUP	TITANIUM	GRADE 2	B337	2" NOMINAL SCHEDULE 10 X 1-1/2" LONG WELDED OR SEAMLESS PIPE
9. CONTROL ATTACHMENT BOSS	TITANIUM	GRADE 2	B348	5/8" DIAMETER (STRUCTURE DIAMETER)
10. OUTLET END FLANGE ATTACHMENT BOSS	TITANIUM	GRADE 2	B348	5/8" DIAMETER (STRUCTURE DIAMETER)
11. OUTLET PANEL ASSEMBLY				
11a. BOLTS	STAINLESS STEEL	18-8	——	4 EACH, 1/4-20UNC-2A X 1/2" LONG
OUTLET PANEL	TITANIUM	GRADE 2	B265	4-3/4" WIDE X 2-5/8" HIGH X 1/8" THICKNESS
BOSS	TITANIUM	GRADE 2	B348	1-1/8" DIAMETER X 15/32" LONG
OUTLET NIPPLE	STAINLESS STEEL	316	——	SNAP-TITE #SPHN-6M
12. SAFETY PLUG ASSEMBLY				OVERALL LENGTH 6-9/16"
QUICK DISCONNECT	STAINLESS STEEL	316	——	SNAP-TITE #SPHN-6F
BALL AND SHANK	STAINLESS STEEL	303SE OR 304	——	5/16" DIAMETER X 1/2" LONG
CABLE	STAINLESS STEEL	304	——	7 X 7 AIRCRAFT CABLE, 1/8" DIAMETER X 5-1/16" LONG
STEM	STAINLESS STEEL	316 OR 316L	——	5/16" DIAMETER X 2-1/8" LONG
13. LOCK CAP ASSEMBLY				
LOCK CAP PLATE	TITANIUM	GRADE 2	B265	TEAR-DROP SHAPED, 3-5/8" HIGH X 1/8" THICKNESS
COLLAR	TITANIUM	GRADE 2	B348	1-1/2" O.D. X 1-17/64" I.D. X 49/64" LONG
SLEEVE	TITANIUM	GRADE 2	B348	1-17/64" O.D. X 59/64" I.D. X 5/8" LONG
NUT	TITANIUM	GRADE 2	B348	1-1/8" O.D. X 3/4" LONG W/1-8UNC-2B THREAD (FEMALE)
BOLT	TITANIUM	GRADE 2	B348	29/32" DIAMETER X 1-43/64" LONG W/1-8UNC-2A THREAD (MALE)
14. LOCK MODULE HOUSING	TITANIUM	GRADE 2	B265	3-1/4" WIDE X 4-31/32" HIGH X 1/2" THICKNESS
14a. BOLTS	STAINLESS STEEL	18-8	——	4 EACH, 1/2-20UNF-2A X 9/16" LONG
14b. SCREWS	STAINLESS STEEL	18-8	——	6 EACH, 12-24 X 1/2" LONG
15. AUTOMATIC SECURING MECHANISM (ASM)	TITANIUM	GRADE 2	B265	2-3/8" HIGH X 2-11/64" LONG X 1-1/4" THICKNESS

TOLERANCES (EXCEPT AS NOTED)	REVISIONS #	DATE	DWG BY	APP'D BY
DECIMAL +/- N/A	1			
	2			
FRACTIONAL +/- N/A	3			
ANGULAR +/- N/A	4			
	5			

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

SPEC-150 TYPE B(U) PACKAGE
MATERIALS LIST

DRAWN BY	SCALE	MATERIAL
S. BYRD	1 TO 1	NOTED
CHK'D BY KC 4/14/95	DATE 4-13-95	DRAWING NO. 15B001-3
Q.A. CLASS. N/A	APP'D BY Dechaury	REV.(0)



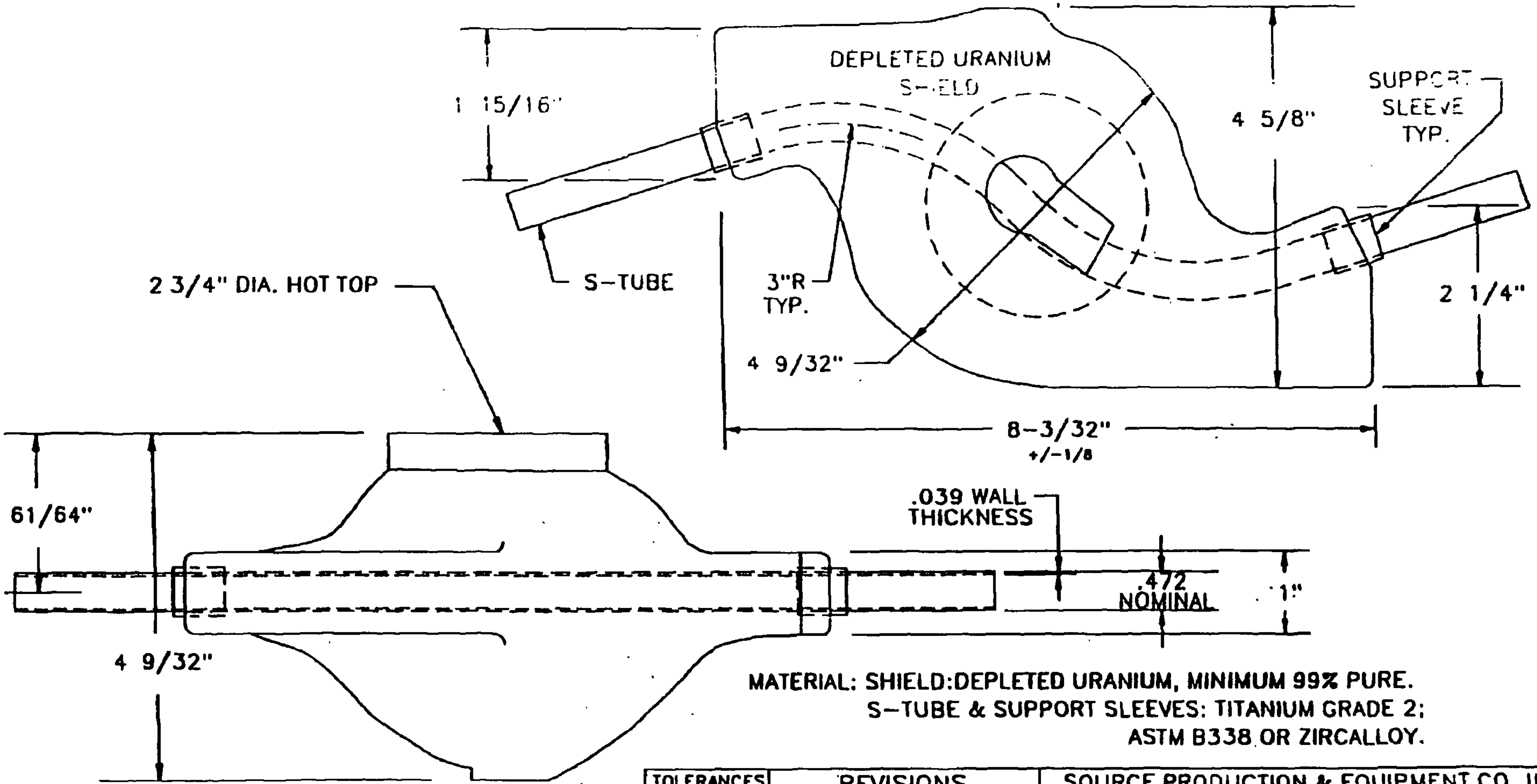
NOTE 1: WELD OF HOUSING TO BOTTOM PLATE NOT SHOWN.
NOTE 2: WEIGHT RANGE: 52 - 53lbs.

TOLERANCES (EXCEPT AS NOTED)	REVISIONS			
	#	DATE	DWG BY	APP'D BY
DECIMAL	1	3-10-95	SB	RDD
+/- N/A	2	4-13-95	SB	RDD
FRACTIONAL +/- N/A	3	4-19-95	SB	[illegible]
ANGULAR	4			
+/- N/A	5			

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

SPEC 150 EXPOSURE DEVICE
FULL SECTIONAL VIEW

DRAWN BY S.BYRD	SCALE 1/2" = 1	MATERIAL N/A
CHK'D BY [illegible]	DATE 3-1-95	DRAWING NO. 15B002A
Q.A. CLASS. N	APP'D BY R.D.DICHARRY	REV.(3)



MATERIAL: SHIELD:DEPLETED URANIUM, MINIMUM 99% PURE.
S-TUBE & SUPPORT SLEEVES: TITANIUM GRADE 2; ASTM B338 OR ZIRCALLOY.

NOTE: COATING INFORMATION:
RUSTOLEUM HIGH PERFORMANCE RED PRIMER #5269
WEIGHT RANGE: 36-1/4 TO 37-1/4lbs.

TOLERANCES (EXCEPT AS NOTED)
DECIMAL +/-
FRACTIONAL +/- N/A
ANGULAR +/-

REVISIONS

#	DATE	DWG BY	APP'D BY
1	3-1-95	SB	RDD
2	4-13-95	SB	[illegible]
3			
4			
5			

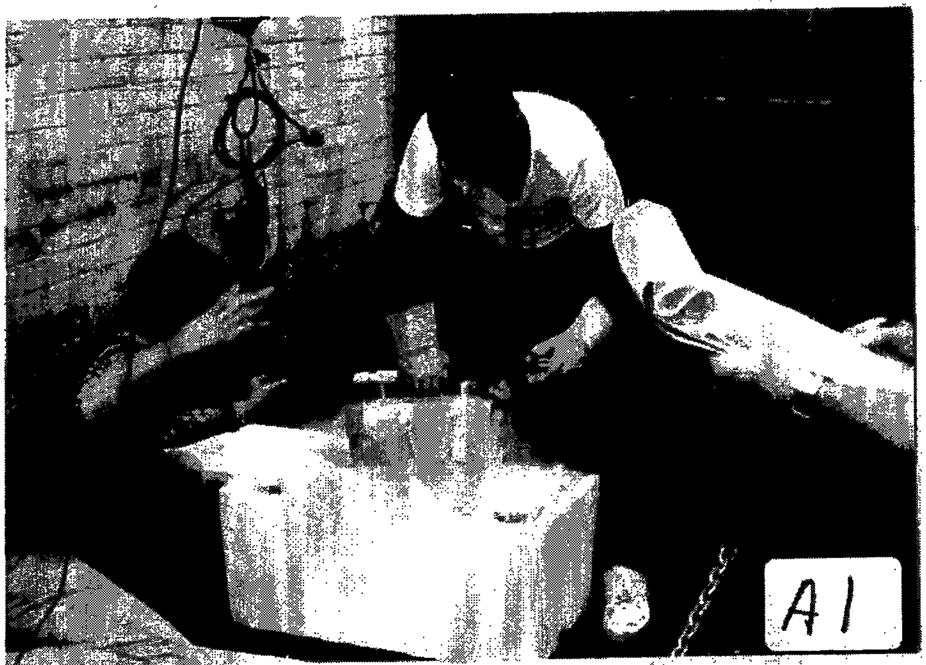
SOURCE PRODUCTION & EQUIPMENT CO.,INC
113 TEAL ST., ST.ROSE. LA. 70087

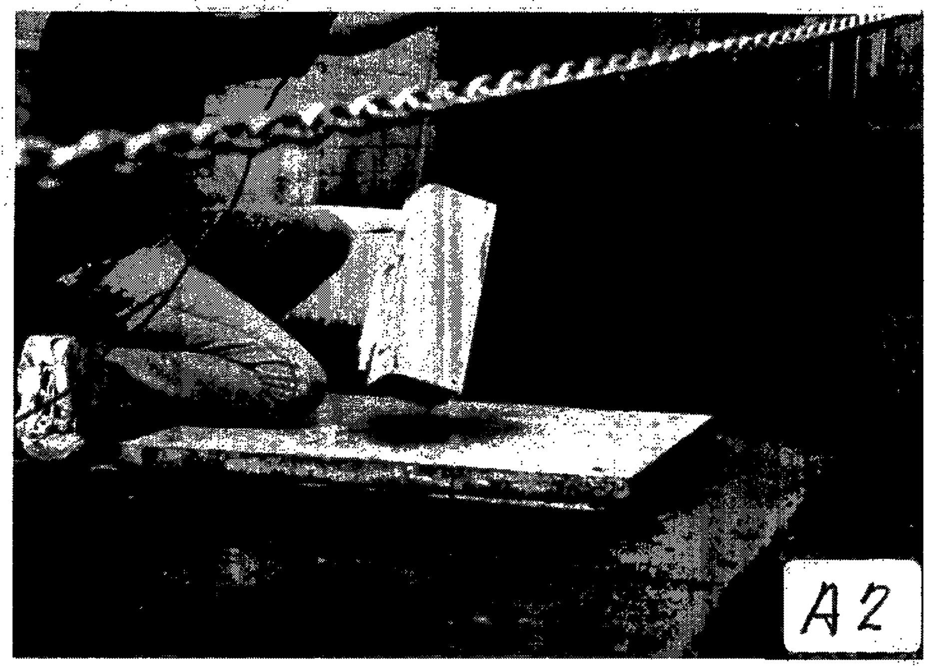
SPEC-150 TYPE B(U) PACKAGE
DEPLETED URANIUM SHIELD

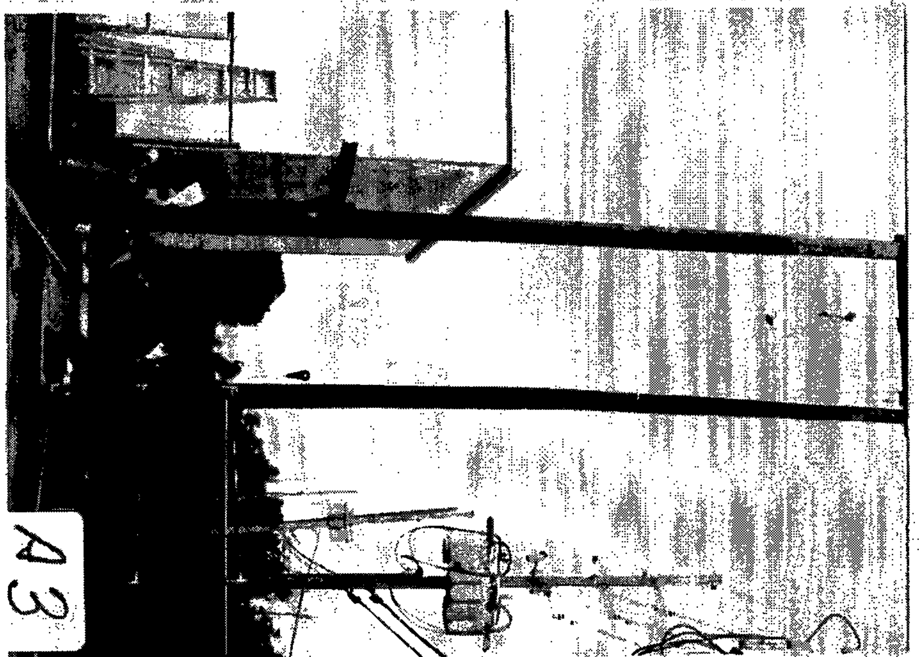
DRAWN BY S. BYRD	SCALE 1/2" = 1"	MATERIAL NOTED
CHK'D BY RC 4/14/95	DATE 12-18-94	DRAWING NO. 15B008 REV.(2)
Q.A. CLASS. Q-A	APP'D BY R.D.DICHARRY	

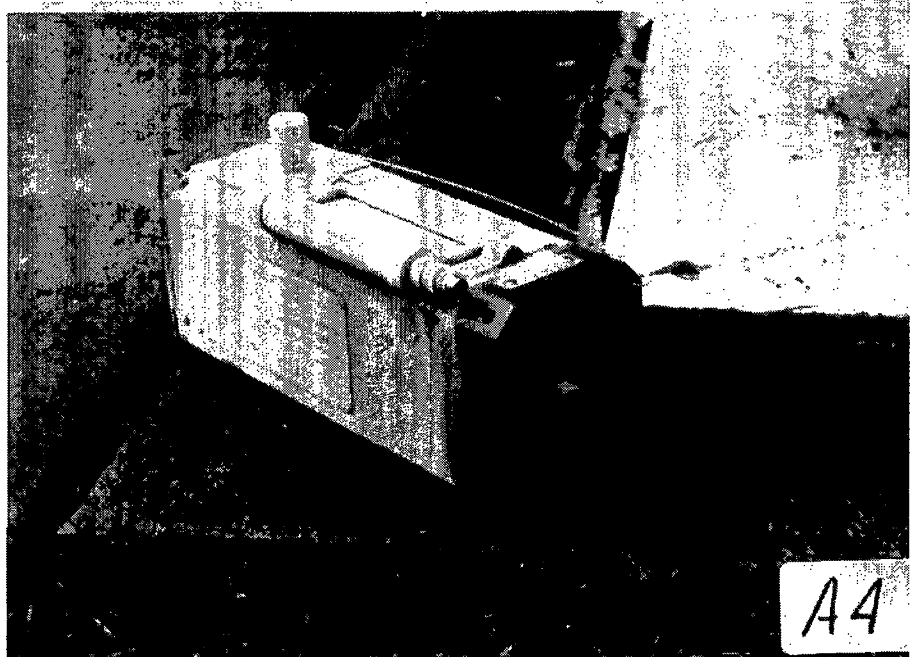
Appendix 4.6 SPEC-150 Tests Photos

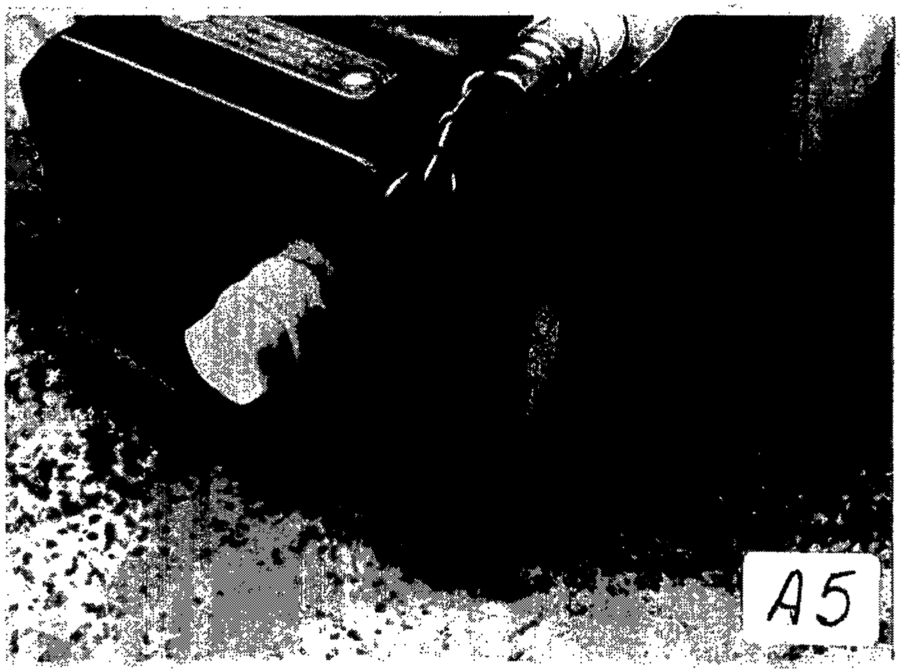
Photo	Description
A1	Before 30-foot drop, Chill
A2	Before 30-foot drop, Orientation
A3	Before 30-foot drop, Height
A4	After 30-foot drop, Damage
A5	After 30-foot drop, Damage
A6	After 30-foot drop, Damage
A7	After 30-foot drop, Damage
A8	After 30-foot drop, Damage
A9	Before 1st Puncture Test -Temperature
A10	Before 1st Puncture Test - Orientation, Safety Plug
A11	Before 1st Puncture Test - Height Check
A12	1st Puncture Test - Instant of Impact
A13	After Puncture Test - Damage, Puncture Pin
A14	After Puncture Test - Safety Plug
A15	Before 2nd Puncture Test - Orientation, Right Side
A16	Before 2nd Puncture Test - Height Check
A17	2nd Puncture Test - Instant of Impact
A18	After 2nd Puncture Test - Damage, Right Side
A19	Before 3rd Puncture Test - Orientation, Lock Cap
A20	Before 3rd Puncture Test - Height Check
A21	3rd Puncture Test - Instant of Impact
A22	After 3rd Puncture Test - Damage, Puncture Pin
A23	After 3rd Puncture Test - Damage, Lock Cap & Flange
A24	After 3rd Puncture Test - Damage, Lock Cap & Flange


A1


A2

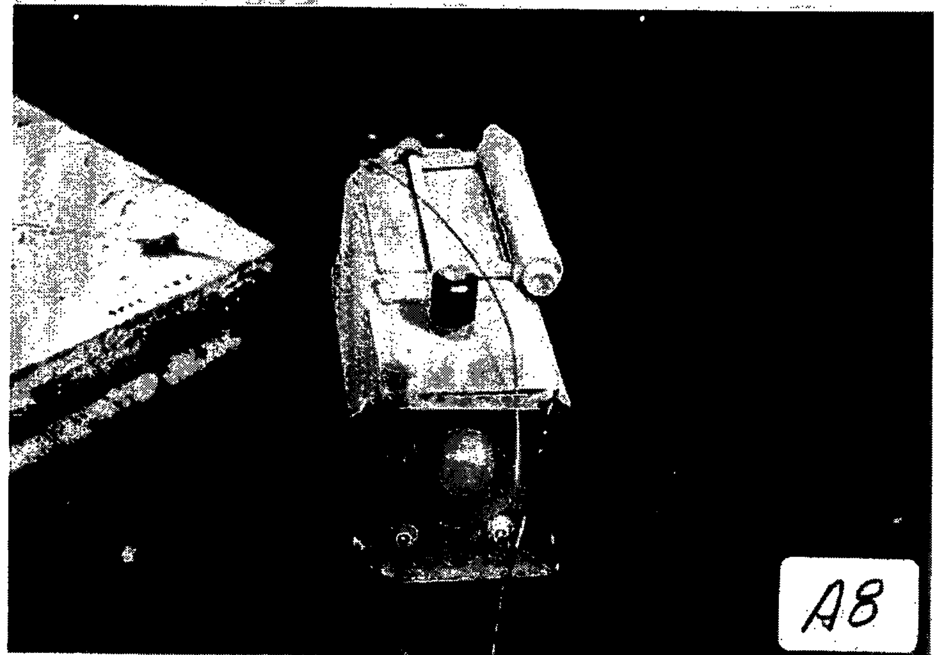

A3

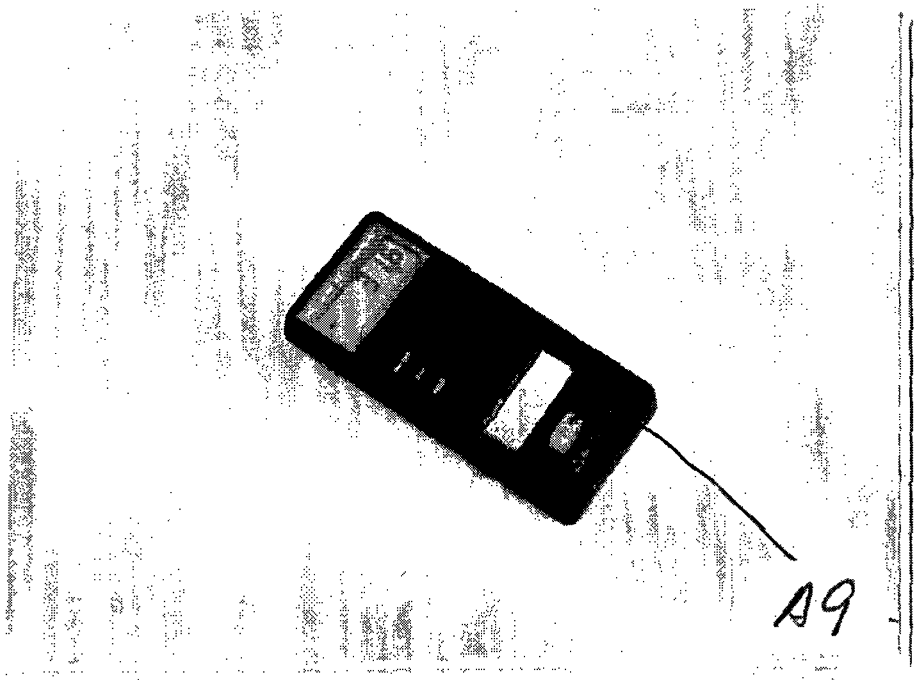

A4


A5

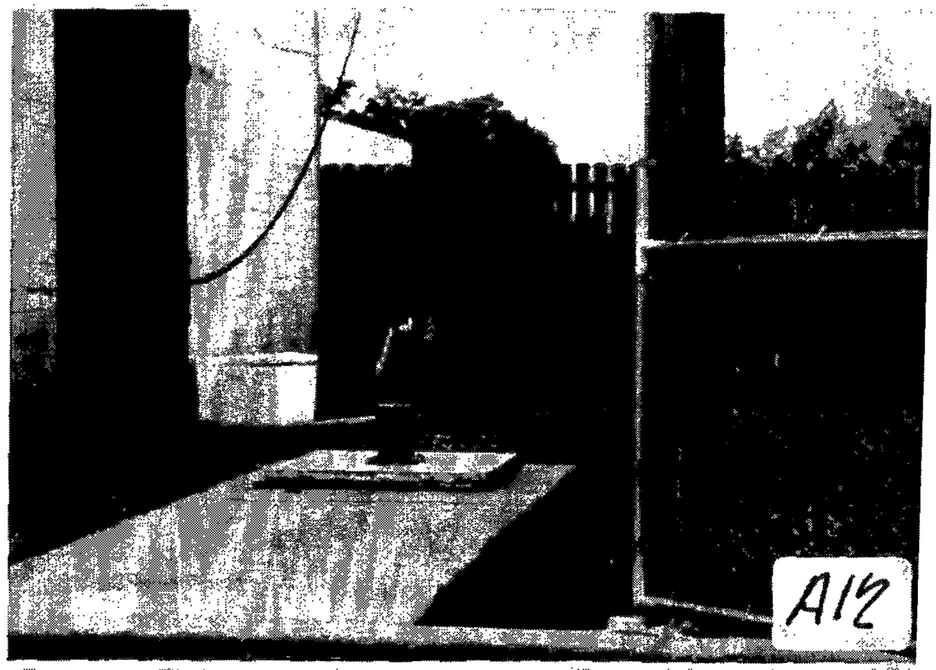

A6


A7

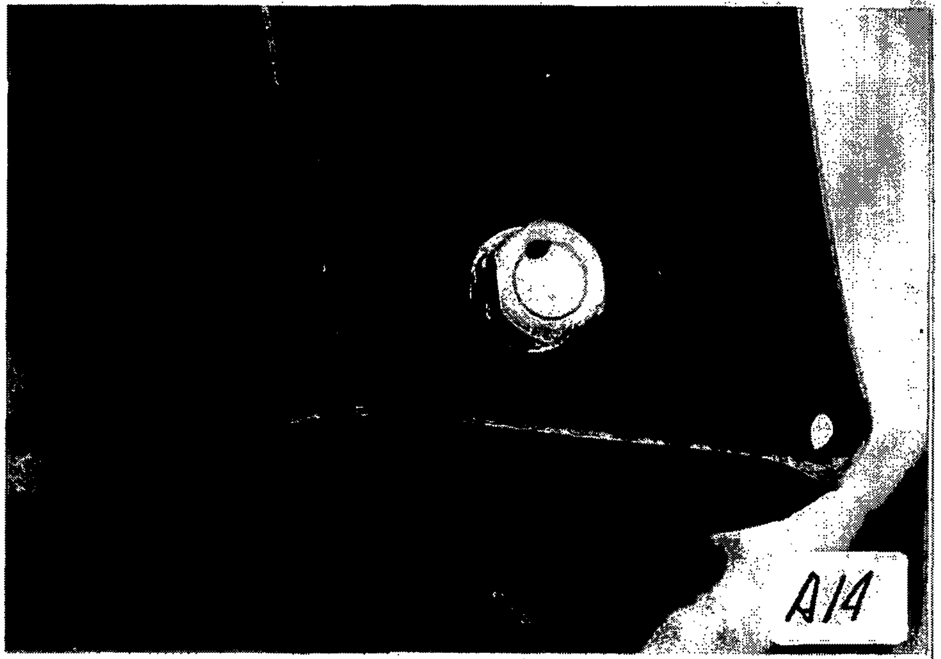

A8

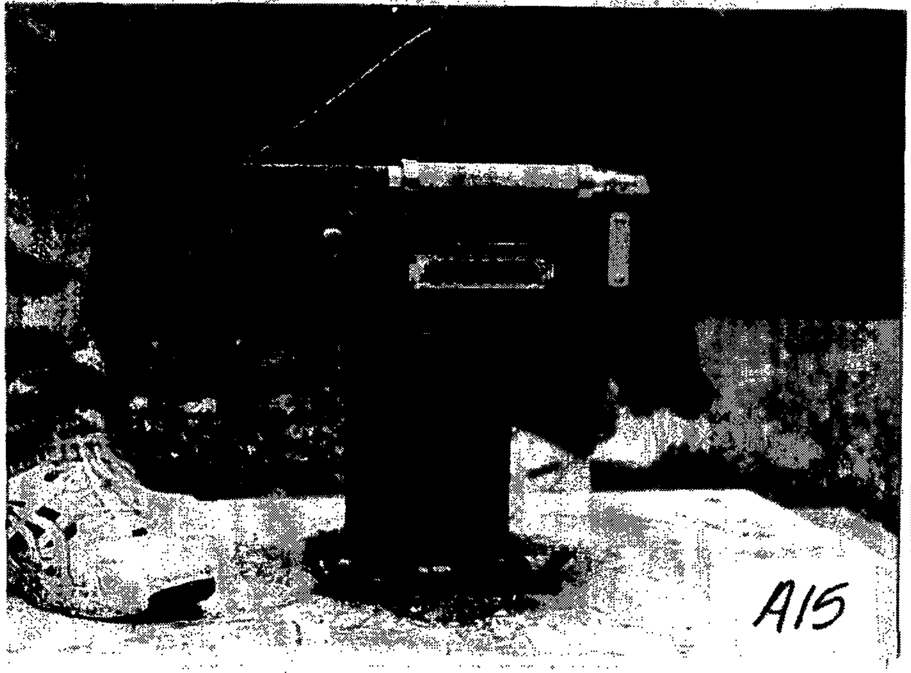

A9

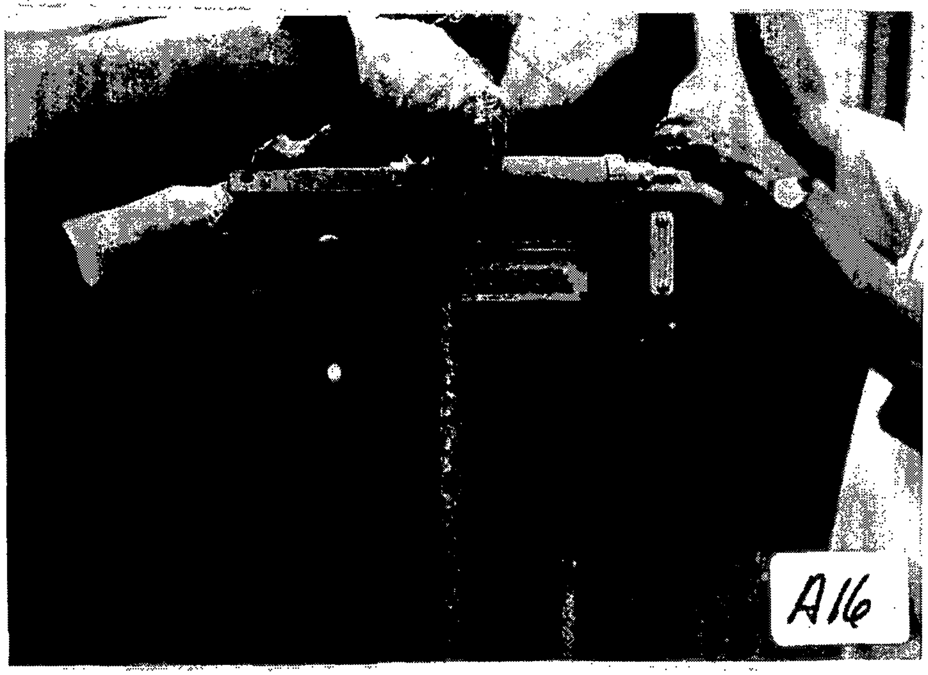
A10

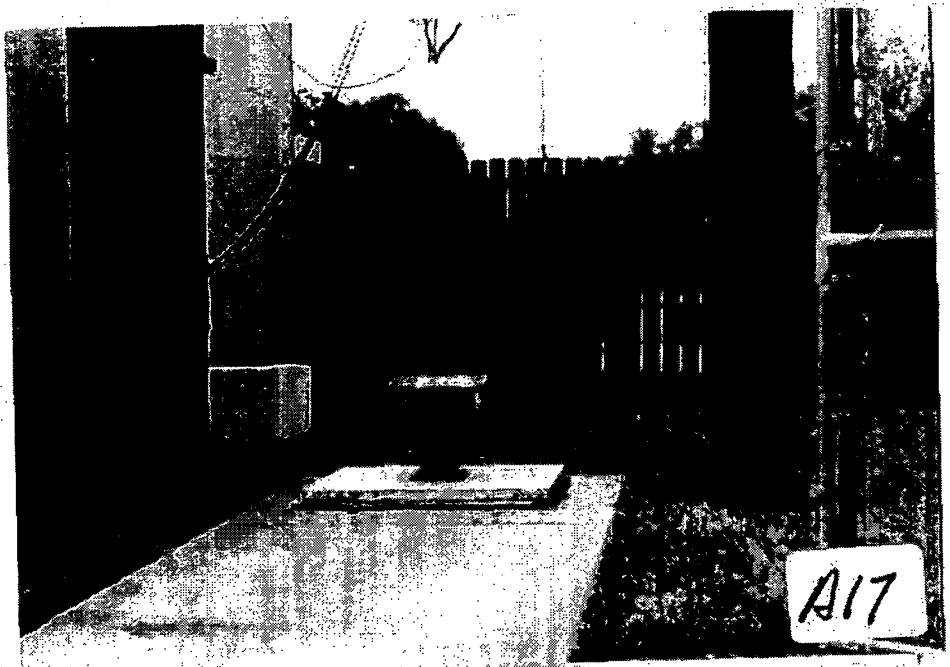
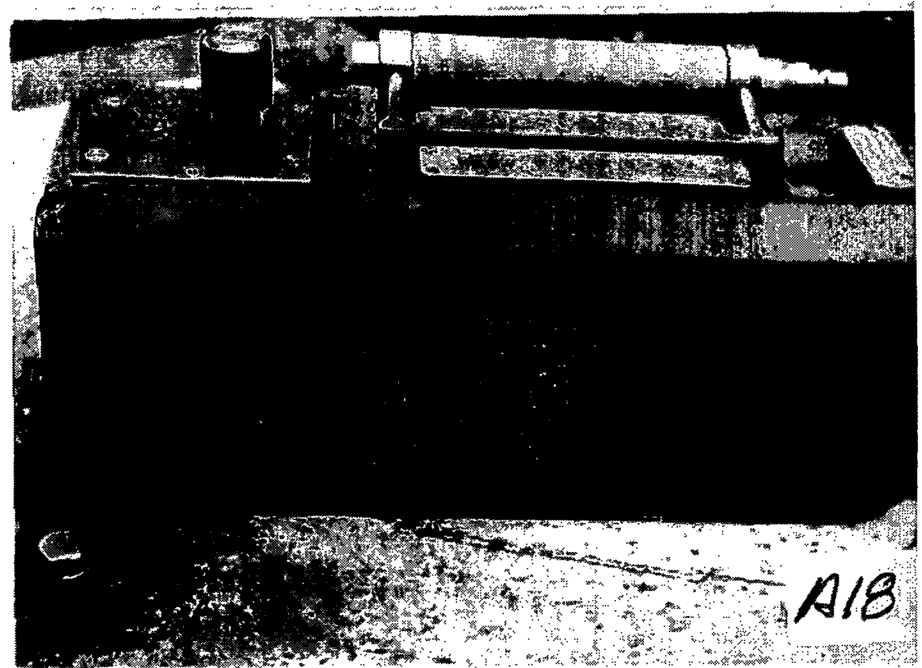
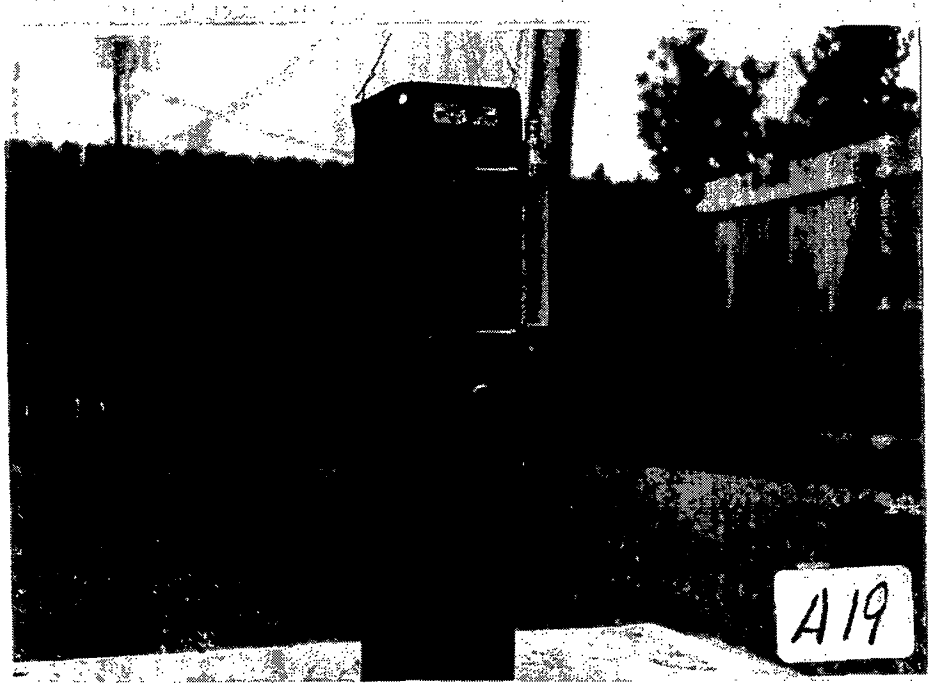
A11

A12


← OD IMPACT
A13

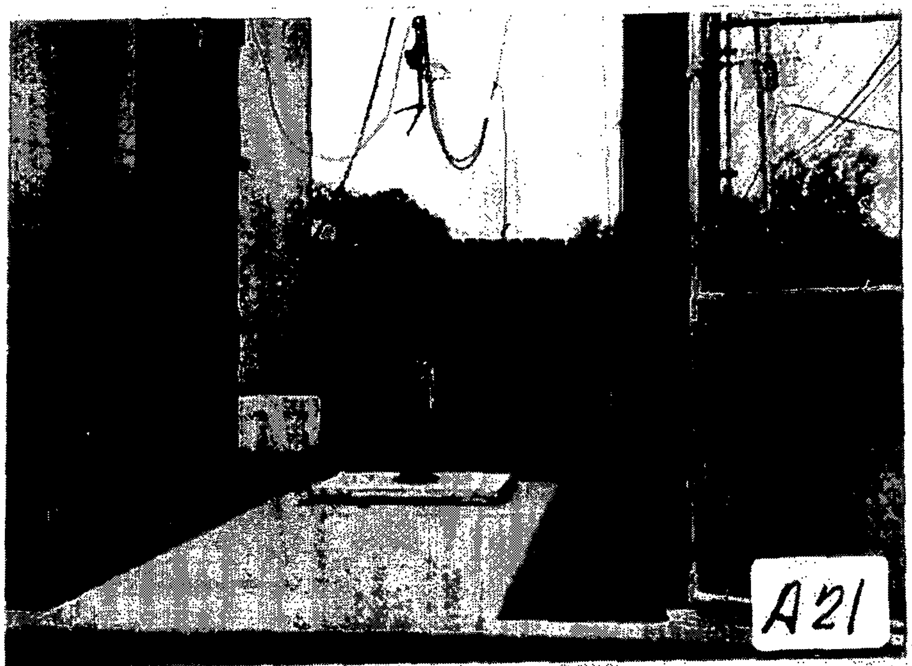

A14

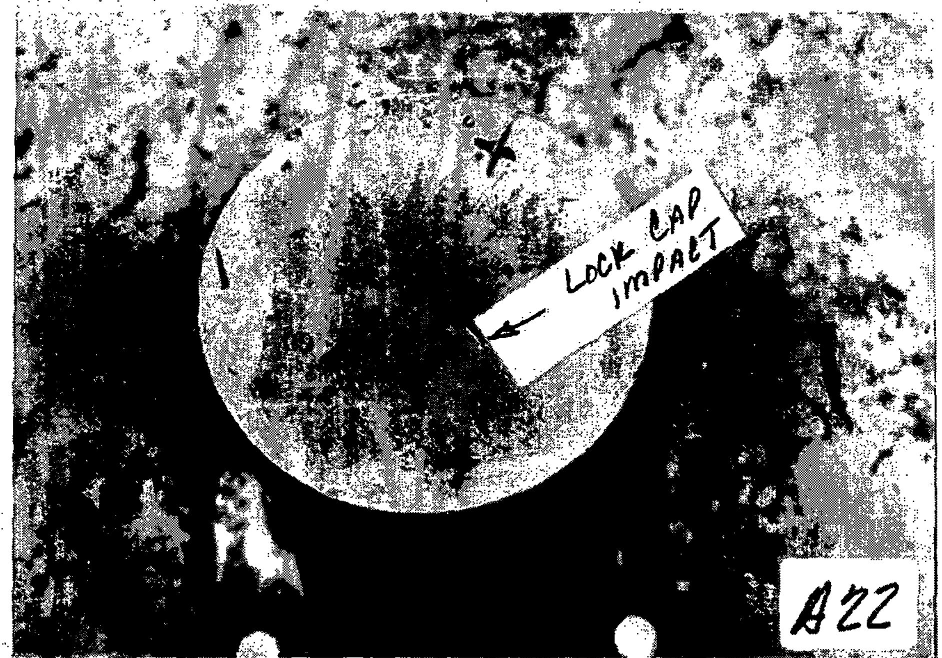

A15

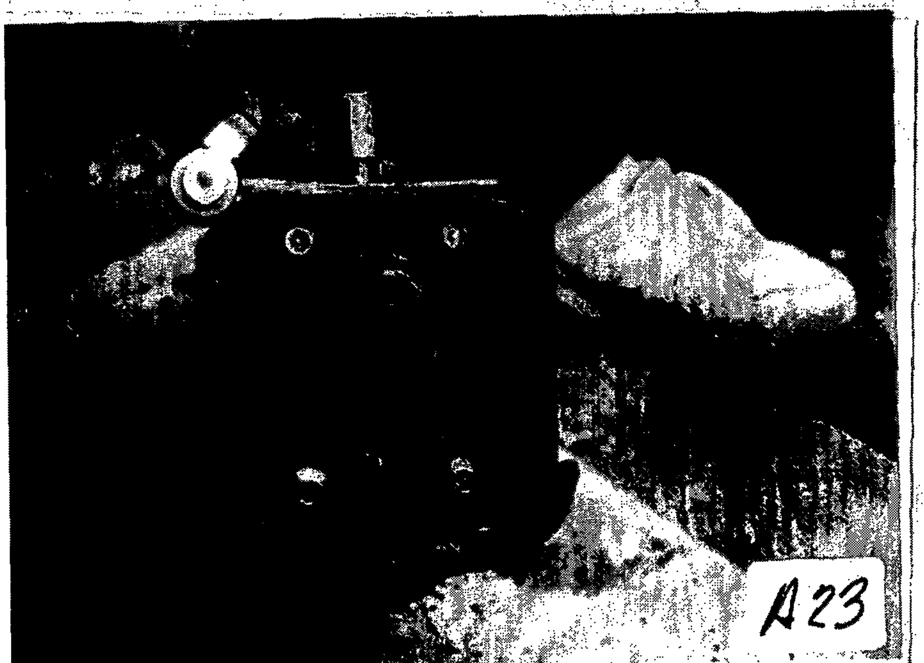

A16


A17

A18

A19

A20


A21

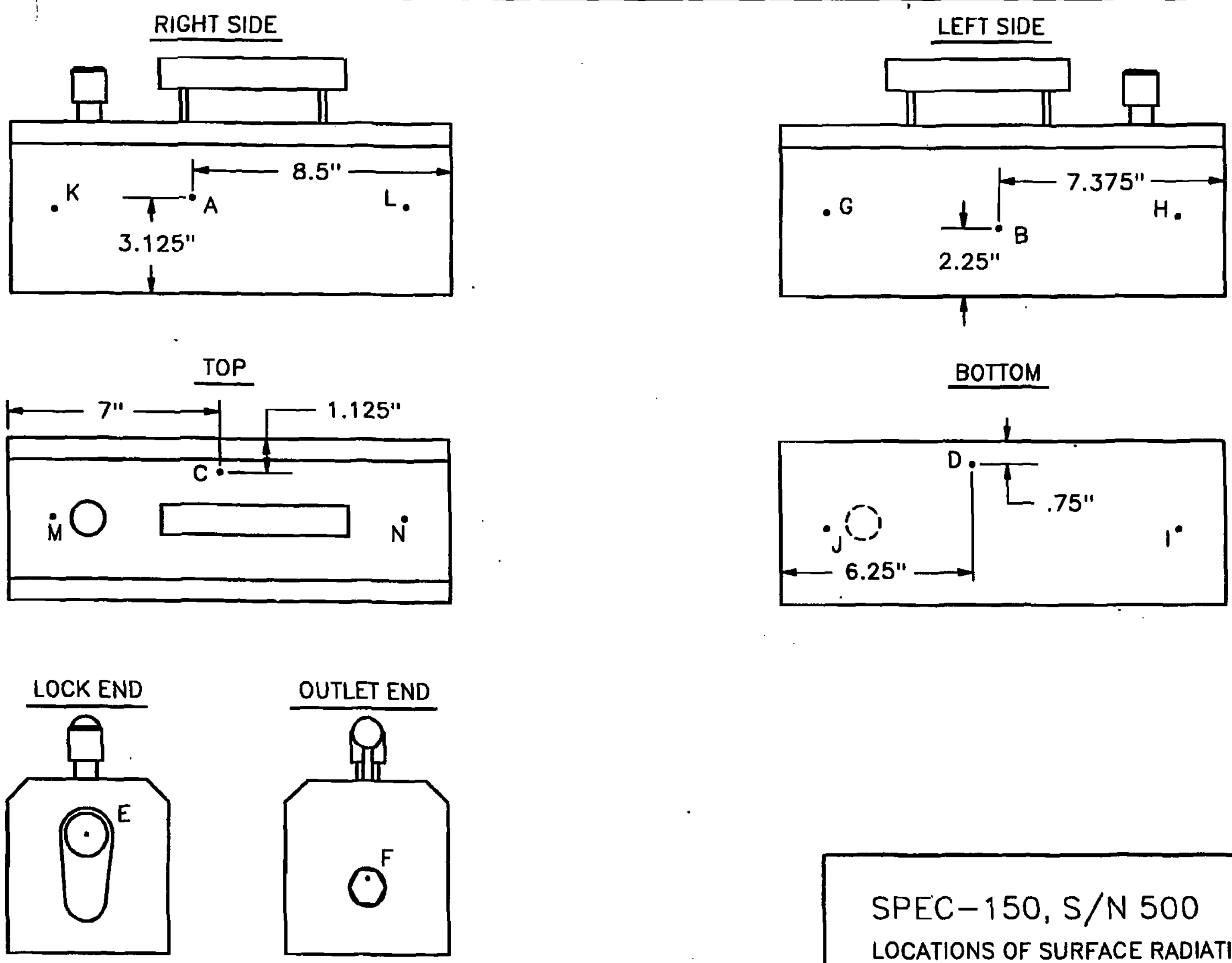

LOCK CAP
IMPACT
A22


A23


A24

Appendix 4.7

Sketches of SPEC-150 Survey Locations.



SPEC-150, S/N 500

LOCATIONS OF SURFACE RADIATION READINGS.
LOCATIONS ARE MARKED ON PACKAGE SURFACE.

FILENAME: H:\DC\SPEC\63097DT1 S.BYRD 6-30-97 SCALE: 3/16

Appendix 9.7
Shielding Analysis, Se-75 and Yb

Report

By: John J. Munro III

Date: 02 February 2011

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)

As explained by J. H. Hubbell and S. M. Seltzer of NIST[i], a narrow beam of monoenergetic photons with an incident intensity I_0, penetrating a layer of material with mass thickness x and density ρ, emerges with intensity I given by the exponential attenuation law:

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

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

For a broad-beam of monoenergetic photons, the "Build-Up factor" must be included. The Broad Beam transmission of photons through a material can be expressed as:

$$\tau(E) = \frac{I}{I_0} = B(E,x)\, e^{-\frac{\mu}{\rho}(E)\rho x} \qquad [3]$$

where: $B(E,x)$: Exposure Build-Up Factor for photons of Energy E and shielding thickness X

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

$$\tau = \sum_i f(E)_i\, \tau(E) = \sum_i f(E)_i\, B(E,x)_i\, e^{-\frac{\mu}{\rho}(E)_i \rho x} \qquad [4]$$

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

^{75}Selenium

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

For ^{75}Selenium, we have used the exposure rate constant of 0.20 R m^2 h^{-1} Ci^{-1}, as presented in the 1970 Radiological Health Handbook,[ii] and have not used the exposure rate constant presented in the 1992 edition of The Health Physics and Radiological Health Handbook,[iii] which is merely a recitation of the data presented in ORNL/RISC-45.[iv]

To justify the use of this exposure rate constant, we offer the following explanation.

The exposure rate constant is the exposure rate at a specific distance from a given amount of a photon-emitting radionuclide. Typically, this is presented in units of Roentgens per hour (R/hr) at a distance of one (1) meter from a one (1) curie point source of that radionuclide.

This concept of the exposure rate constant provides a convenient means for determining the activity of a source based upon a measurement of the exposure rate at a specific distance from the source.

In the case of ^{75}Selenium, this method is complicated by the photon spectrum. The exposure rate constant for ^{75}Selenium is reported to be 1.03 R-m^2/h-Ci in ORNL/RISC-45 and 0.595 R-m^2/h-Ci by Shilton.[v] These values are presented for an unencapsulated non-self-absorbing point source of ^{75}Selenium.

However, approximately 67% of this exposure rate is a result of photons with energies less than 12 keV.

The typical radiography source is encapsulated in stainless steel, with a wall thickness on the order of 0.25 mm. The transmission of a 12 keV photon through 0.25 mm of steel is 2.5 x 10^{-11}. Therefore, essentially none of the less than 12 keV photons emerge from the source encapsulation. (Interestingly, in the absence of any encapsulation, the transmission of 12 keV photons through one meter of air is less than 64%.)

Consequentially, if the intrinsic exposure rate constant (which includes the <12 keV photons) were used to determine the activity of an encapsulated source based upon measurement of the exposure rate at some specified distance (in which there were virtually no <12 keV photons), the activity would be significantly underestimated (by a factor of ~3). This could lead to significant safety and regulatory issues.

To account for this anomaly, a number of investigators have determined the exposure rate constant based only on the photons with energies above 12 keV. These have been reported in the 1970 edition of the Radiological Health Handbook[vi] as 0.20 R-m^2/h-Ci, by Shilton[vii] as 0.201 R-m^2/h-Ci, and by Weeks et al.[viii] as 0.199 R-m^2/h-Ci.

Most recently, Ninkovic et al.[ix] has reported: "In the process of analysing accessible data on the air kerma rate constants and its precursors for many radionuclides used most often in practice (6–17) it was concluded that published data are in strong disagreement." They cite ORNL/RISC-45 as one of these references. They continue: "That is the reason we decided to recalculate this (sic) quantities on the basis of the latest data on gamma ray spectra and on the latest data for mass energy-transfer coefficients for air." For ^{75}Selenium, Ninkovic et al. have determined an exposure rate constant of 0.205 R-m^2/h-Ci using only photons above 20 keV.

For these reasons, we have determined that the use of the value of 0.20 R-m^2/h-Ci permits a reasonably accurate determination of the activity of an encapsulated source based on measurement of the exposure rate at a specific distance from the source. This has become the accepted value of the exposure rate constant in the industrial radiography industry and is currently used by the only other supplier of ^{75}Selenium sources in the United States.

Moreover, the NRC has endorsed this value of 0.20 R-m^2/h-Ci, taken from the 1970 edition of the Radiological health Handbook, in a several NRC Certificates of Compliance (Certificate 9296, issued 6 June 2008; Certificate 9269, issued very recently on 14 September 2010.

Therefore, as noted above, for ^{75}Selenium, we have used the exposure rate constant of 0.20 R m^2 h^{-1} Ci^{-1}, as presented in the 1970 Radiological Health Handbook.

The ^{75}Selenium spectrum consists of 20 photons with energies above 15 keV.[x] The mass attenuation coefficient for uranium for each of these photon energies was interpolated from the values of NIST.[xi]

The minimum shielding of the SPEC-150 is 48 mm of Uranium. Exposure Buildup values for each energy and a thickness of 48 mm of Uranium were interpolated from data for Uranium in ANS/ANSI-6.4.3-1991.[xii]

Using Eq. [4], the value of the transmission, $\tau(E)$, was calculated for this 48 mm thickness of uranium (ρ=18.7 g cm^{-3}). The details of these calculations are presented in the following table:

Energy (keV)	$f(E)_i$	$\frac{\mu}{\rho}(E)$	Buildup $B(E,x)$	$\tau(E)$
24.4	1.95E-04	69.1125	1.02	0.000E+00
66.1	1.95E-03	5.5755	1.80	1.578E-220
80.9	1.42E-05	3.3370	2.20	2.584E-135
96.7	6.96E-03	2.1270	2.20	1.867E-85
121.1	4.78E-02	4.3191	2.82E+12	5.770E-158
136.0	1.86E-01	3.2884	1.43E+11	1.721E-118
198.6	7.46E-03	1.3495	3.57	6.618E-55
249.4	5.96E-08	0.7898	1.99	1.932E-38
264.7	4.19E-01	0.6868	1.84	1.294E-27
279.5	1.89E-01	0.6039	1.78	9.690E-25
303.9	1.10E-02	0.4781	1.60	4.043E-21
373.5	2.59E-05	0.3392	1.90	2.942E-18
400.7	1.30E-01	0.3018	1.85	4.118E-13
418.8	1.40E-04	0.2804	1.90	3.125E-15
468.6	4.51E-06	0.2325	2.00	7.773E-15
542.4	1.99E-07	0.1823	2.10	3.276E-14
557.8	1.42E-08	0.1740	2.10	4.910E-15
572.2	5.74E-04	0.1668	2.10	3.811E-10
617.8	7.72E-05	0.1468	2.30	3.373E-10
821.6	3.09E-06	0.0972	2.22	1.115E-09
			Total Transmission, τ:	1.834E-09

The resultant calculated transmission of photons of ^{75}Selenium through 48 mm of uranium is 1.834 x 10^{-9}.

Using this value of transmission the minimum shielding of the SPEC-150, (i.e. 48 mm) in Eq. [1], the exposure rates, when containing 150 Ci of ^{75}Selenium are calculated to be:

	At Surface (0.068 m from Source)	At 1 meter from Surface (1.068 m from Source)
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.

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.

For similar reasons described above for ^{75}Selenium, we have not used the exposure rate constant of ORNL/RISC-45. Rather, we have used the value of 0.125 R-m^2/h-Ci for the ^{169}Ytterbium exposure rate constant taken from a very recent paper by Cazeca et al.[xiii] These authors characterized a high dose rate brachytherapy source of ^{169}Ytterbium with physical characteristics very similar to industrial radiography sources. Therefore, we have determined that the use of the value of 0.125 R-m^2/h-Ci permits a reasonably accurate determination of the activity of an encapsulated source based on measurement of the exposure rate at a specific distance from the source.

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 minimum shielding of the SPEC-150 is 48 mm of Uranium. Exposure Buildup values for each energy and a thickness of 48 mm of Uranium were interpolated from data for Uranium in ANS/ANSI-6.4.3-1991.[xiv]

Using Eq. [4], the value of the transmission, $\tau(E)$, was calculated for this 48 mm thickness of uranium (ρ=18.7 g cm^{-3}). The details of these calculations are presented in the following table:

Energy (keV)	$f(E)_i$	$\frac{\mu}{\rho}(E)$	Buildup $B(E,x)$	$\tau(E)$
20.8	2.16E-03	103.8415	1.02	0.000E+00
42.8	3.47E-04	16.7222	2.13	0.000E+00
45.9	1.21E-05	13.9510	2.25	0.000E+00
49.8	1.16E-01	11.3945	2.28	0.000E+00
50.6	6.50E-04	10.9245	2.28	0.000E+00
50.7	1.98E-01	10.8529	2.28	0.000E+00
50.9	6.47E-04	10.7921	2.28	0.000E+00
51.5	1.92E-05	10.4488	2.28	0.000E+00
57.3	1.95E-02	7.9838	2.25	0.000E+00
57.5	3.76E-02	7.9121	2.25	0.000E+00
59.0	1.26E-02	7.4066	2.32	5.495E-291
63.0	2.15E-03	6.2806	2.27	7.141E-248
63.1	8.51E-02	6.2529	2.27	3.410E-245

Energy (keV)	$f(E)_i$	$\frac{\mu}{\rho}(E)$	Buildup $B(E,x)$	$\tau(E)$
65.9	9.82E-06	5.6166	2.27	2.505E-224
72.0	3.63E-06	4.4799	2.22	1.855E-180
85.1	2.91E-06	2.9407	2.19	1.474E-120
93.6	5.59E-03	2.3106	2.00	9.493E-93
95.7	2.46E-06	2.1855	2.00	3.137E-91
95.9	2.46E-06	2.1769	2.00	6.824E-91
98.0	2.12E-06	2.0577	2.00	2.585E-86
101.4	1.01E-05	1.8881	2.06	5.193E-79
105.2	6.86E-06	1.7212	2.10	1.155E-72
109.8	4.82E-02	1.5452	2.15	6.035E-62
113.6	1.44E-05	1.4166	2.15	1.855E-60
114.0	1.16E-05	1.4054	2.15	4.098E-60
117.4	1.20E-04	4.6498	8.34E+12	5.519E-173
118.2	5.67E-03	4.5749	5.44E+12	1.416E-168
129.9	1.02E-03	3.6606	1.55E+11	3.160E-135
130.5	3.88E-02	3.6222	1.55E+11	3.764E-132
156.7	4.21E-05	2.3556	9.73E+07	6.107E-89
173.9	6.74E-06	1.8449	1.82E+04	1.476E-73
177.2	1.10E-01	1.7644	4.00E+03	7.314E-67
193.2	4.10E-05	1.4408	17.98	5.017E-60
198.0	2.02E-01	1.3599	3.55	6.994E-54
199.8	9.11E-05	1.3310	3.55	4.198E-56
206.0	2.01E-05	1.2384	3.13	3.327E-53
213.9	1.80E-05	1.1329	2.78	3.424E-49
226.3	1.65E-06	0.9927	2.12	7.047E-45
228.7	1.34E-06	0.9682	2.12	5.116E-44
240.3	8.10E-04	0.8617	2.12	4.405E-37
261.1	1.32E-02	0.7092	1.90	5.644E-30
291.2	3.84E-05	0.5486	1.90	2.997E-26
294.5	9.02E-06	0.5340	1.90	2.606E-26
301.7	2.13E-05	0.4839	1.58	4.609E-24
306.8	8.51E-04	0.4706	1.58	6.076E-22
307.5	2.84E-03	0.4688	1.58	2.377E-21
307.7	9.54E-02	0.4683	1.58	8.371E-20
334.0	1.81E-05	0.4087	1.75	3.711E-21
336.6	9.87E-05	0.4033	1.76	3.289E-20
356.7	1.58E-06	0.3662	1.76	1.478E-20
370.9	1.03E-04	0.3433	1.76	7.517E-18
379.3	4.90E-06	0.3307	1.76	1.113E-18
386.7	4.09E-06	0.3202	1.76	2.371E-18
452.6	2.59E-07	0.2464	1.93	1.243E-16
464.7	5.31E-08	0.2358	1.93	6.591E-17
465.7	2.81E-06	0.2350	1.93	3.753E-15
466.7	2.87E-07	0.2341	1.93	4.148E-16
475.0	2.92E-06	0.2274	1.93	7.734E-15
494.4	2.30E-05	0.2127	2.11	2.482E-13
500.4	1.40E-07	0.2085	2.11	2.197E-15
507.8	2.41E-08	0.2034	2.11	5.986E-16
515.1	6.81E-05	0.1986	2.10	2.579E-12

Energy (keV)	$f(E)_i$	$\frac{\mu}{\rho}(E)$	Buildup $B(E,x)$	$\tau(E)$
528.6	2.01E-06	0.1903	2.10	1.611E-13
546.2	2.59E-08	0.1802	2.10	5.144E-15
562.4	2.12E-06	0.1716	2.10	9.076E-13
570.9	2.00E-06	0.1674	2.10	1.255E-12
579.9	3.54E-05	0.1631	2.15	3.334E-11
600.6	2.16E-05	0.1538	2.20	4.794E-11
624.9	9.70E-05	0.1440	2.20	5.190E-10
633.3	1.38E-07	0.1408	2.15	9.583E-13
642.9	1.55E-06	0.1374	2.15	1.471E-11
663.6	4.02E-06	0.1303	2.15	7.206E-11
693.5	1.89E-07	0.1211	2.20	7.909E-12
710.4	7.54E-07	0.1163	2.20	4.842E-11
739.4	4.21E-08	0.1088	2.20	5.305E-12
760.2	1.96E-08	0.1039	2.30	4.009E-12
773.4	5.01E-06	0.1010	2.30	1.333E-09
781.6	7.25E-08	0.0992	2.35	2.313E-11
			Total Transmission, τ:	2.115E-09

The resultant calculated transmission for photons of ^{169}Ytterbium through 48 mm of uranium is 2.115E-09.

Using this value of transmission for the minimum shielding of the SPEC-150, (i.e. 48 mm) in Eq. [1], the exposure rates, when containing 150 Ci of ^{169}Ytterbium are calculated to be:

	At Surface (0.068 m from Source)	At 1 meter from Surface (1.068 m from Source)
Calculated (48 mm Uranium)	0.0086 mR/hr	3.5×10^{-5} mR/hr

This result is less than 0.004% of the regulatory limit for the surface of the container and less than 0.0004% 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.

---

[i] 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, NISTIR 5632 http://physics.nist.gov/PhysRefData/XrayMassCoef/chap2.html accessed 02 February 2011

[ii] Radiological Health Handbook, rev. ed., U.S. Public Health Service, Bureau of Radiological Health, Rockville, MD, 1970.

[iii] Shleien B (ed), The Health Physics and Radiological Health Handbook, Revised Edition, Published by Scinta, Inc. 2421 Homestead Drive , Silver Spring Md. 20902, 1992

[iv] ORNL/RSIC-45, "Specific Gamma-Ray Dose Constants for Nuclides Important to Dosimetry and Radiological Assessment", May, 1982

[v] Shilton MG, Advanced, Second-Generation Selenium-75 Gamma Radiography Sources, presented at the 15th World Conference on Non-Destructive Testing, Rome, Italy, 15-21 October 2000 Revised and amended August 2003

[vi] Radiological Health Handbook, rev. ed., U.S. Public Health Service, Bureau of Radiological Health, Rockville, MD, 1970.

[vii] Shilton MG, Advanced, Second-Generation Selenium-75 Gamma Radiography Sources, presented at the 15th World Conference on Non-Destructive Testing, Rome, Italy, 15-21 October 2000 Revised and amended August 2003

[viii] 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.

[ix] Ninkovic MM Raicevic JJ and Adrovic F, Air Kerma Rate Constants For Gamma Emitters Used Most Often In Practice, Radiation Protection Dosimetry (2005), Vol. 115, No. 1–4, pp. 247–250

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

[xi] 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

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

[xiii] Cazeca MJ, Medich DC, and Munro JJ, Monte Carlo characterization of a new Yb-169 high dose rate source for brachytherapy application, Med Phys. 37(3):1129-1136, March 2010.

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