CHAPTER 4 – CONTAINMENT

This chapter identifies and discusses the package containment for the Normal Conditions of Transport (NCOT) and the Hypothetical Accident Conditions of Transport (HACOT).

4.1 CONTAINMENT BOUNDARY

The containment boundary is defined as the envelope surrounding the bolted closure shield plug and the cavity of F-294 container i.e., the bolted closure shield plug assembly and the inner shell of the container as per Figure 4-F1. The containment boundary envelope is bounded by ABCDEFGHIJKLA domain as per Figure 4-F1.

4.1.1 CONTAINMENT VESSEL

The containment system consists of:

1. the bolted closure shield plug assembly and
2. the inner shell assembly of F-294 container.

The upper cavity of the inner shell assembly houses the closure shield plug.

The lower cavity of the inner shell assembly houses cobalt-60 MDS Nordion sealed source C-188.

The components of the F-294 containment system are identified in Figure 4.F2.

The MDS Nordion sealed source C-188 meets the Special Form Radioactive Material requirements and is shown in Figure 4-F3.

The cobalt-60 data sheet is attached in Appendix 4.4.2

This source capsule is designed, tested and certified to

1. ANSI Standard N542 designation ANSI 77 E65646 (Ref. [3]) and
2. ANSI Standard N43.10 Bend Test (Ref. [4]) and
3. Special Form requirements of IAEA Safety Series No. 6 (Ref. [2]) which are equivalent to the Special Form requirements of 10 CFR Part 71 SS 71.75.

MDS Nordion's C-188 sealed source has been given a USNRC Sealed Source Registration Number NR-220-S-103-S. A copy of the certificate is attached in Appendix 4.4.2. Certificates of Performance to the stated standards are also attached in Appendix 4.4.2.

This C-188 source capsule design is manufactured as per MDS Nordion Technical Specifications IN/PR 0030 J1100 (Ref. [7]) and IN/TS 0146 J1100 (Ref. [8]).

4.1.2 CONTAINMENT PENETRATION

There are three (3) penetrations into the containment system.

- one (1) drainline connection to the lower cavity end cap; see Figure 4-F2
- two (2) ventline connections on the external surface of the shield plug; see Figure 4-F1.

4.1.3 SEALS AND WELDS

4.1.3.1 Seals

There are a total of four (4) seals in the containment system, see Figure 4-F4. The four seals are:

- One (1) "Neoprene" gasket between the closure plug and the inner shell assembly of F-294 container.
- Two (2) nickel gaskets on the ventline 0.5 in. dia. pipe cap connections, located on the top face of the closure plug.
- One (1) "Neoprene" gasket on the drainline located on the external surface of the container.

4.1.3.2 Welds

The welds of the containment system are identified in Figure 4-F5.

Chapter 1, Appendix 1.3.2, Sheet 2 of Drawing F629401-001 Issue 3 provides additional information on the weld size, location and inspection.

4.1.4 CLOSURE

A bolted closure (a shield plug) is a part of the containment system. See Figure 4-F6.

The bolted closure of F-294 containment system consists of:

1. a closure shield plug assembly
2. sixteen (16) fasteners (1 in. diameter, UNC. "UNBRAKO" socket head screws)
3. a female container flange which has 16 bolt holes and provides a seal face for a "Neoprene" gasket.

The male flange of the closure shield plug is fastened to the female flange of the container inner shell assembly. F-294 operating procedures are provided in Chapter 7.

For the closure shield plug, the nominal torque per bolt of 100 ft.-lb. ± 10% is specified.

For the ventline closure pipe cap, the nominal torque of 20 ft.-lb. ± 10% is specified.

For the drainline closure pipe cap, the nominal torque of 50 ft.-lb. ± 10% is specified.

4.2 REQUIREMENTS FOR NORMAL CONDITIONS OF TRANSPORT

This section uses the applicable analysis and discussion from Chapter 2 to demonstrate that the integrity of the package containment system is maintained under the normal conditions of transport as specified in 10 CFR Part 71 SS 71.71.

In normal conditions of transport: when the F-294 is subjected to the 3 ft. free drop test in the top end orientation, it is subjected to 130 g's deceleration loads (see Chapter 2, Appendix 2.10.12).

In the normal conditions of transport (loaded with 374 kCi of cobalt-60 in C-188 sealed sources), the inner shell assembly of F-294 is subjected to 16 psig internal pressure. The lower cavity wall is at 387°F and the top closure plug is at 432°F (see Chapter 3, Table 3.4-T1).

Using the above data, stress analysis of the components of the inner shell assembly and the closure plug of F-294 is carried out and presented in Appendix 4.4.3. It is concluded that the closure plug will be retained over the inner shell assembly of F-294 lead-shielded cask.

A full-scale F-294 test packaging was subjected to eight drop tests as listed below:

Test #1:	Normal Free Drop Test: top end drop orientation
Test #2:	30 ft. Free Drop: side oblique drop orientation
Test #3C:	Puncture Test: impact on the zone near lift lug fin #4
Test #4:	Puncture Test: impact cylindrical fireshield
Test #5:	Puncture Test: impact on fixed skid lower plate
Test #6:	30 ft. Free Drop Test: top end drop orientation.
Test #7:	Puncture Test: impact on the crush shield upper plate
Test #8:	Puncture Test: impact cylindrical fireshield (nameplate zone)

The drop tests were conducted on February 25, 1998 at the Chalk River Laboratory of Atomic Energy of Canada, Chalk River, Ontario, Canada. The drop test results are presented in Chapter 2, Appendix 2.10.12.

In addition, the test results as it pertains to the issue of integrity of the containment system, are recaptured and presented in Chapter 4, Appendix 4.4.7.

4.2.1 RELEASE OF RADIOACTIVE MATERIAL

Chapter 2, section 2.6 of this report discusses the performance of the F-294 package when subjected to tests simulating the normal conditions of transport. With respect to the cobalt-60 C-188 source capsule in question, these normal conditions of transport are shown to be much less severe than the tests performed on the capsule certifying it as Special Form radioactive material and to ANSI Standard N542 designation ANSI 77 E65646. Tests for normal conditions of transport allow a leakage rate of 10.8×10^{-6} Ci per hour ($A_2 \times 10^{-6}$) (Para 71.51 (a)(1) of Ref. [1]), while those for Special Form certification allow a leakage of 0.05 μCi (0.05×10^{-6} Ci) (Para 71.75 (c) (2) (vi) of Ref. [1]).

As C-188 sealed source is certified Special Form, it is therefore concluded that no radioactive material will be released from the containment system as a result of normal conditions of transport.

A full-scale F-294 test packaging was subjected to eight (8) drop tests. In the F-294 test packaging, the radioactive contents were simulated using eight (8) dummy, full-scale, C-188s in a full-scale F-313 carrier. Prior to the drop tests, the dummy C-188s were subjected to a helium leak test and they met 1×10^{-9} atm cc/sec leaktightness level. The test results are presented in Chapter 2, Appendix 2.10.12. Since the F-457 source carrier, loaded with 80 C-188 capsules, weighs only 23 lb. more than the fully loaded F-313 source carrier, test results are expected to be the same for the F-294, with the F-457 source carrier, as they are for the F-294, with the F-313 source carrier.

Relevant C-188 test results, as it pertains to the issue of the integrity of the containment system, are re-captured and presented in Chapter 4, Appendix 4.4.7.

4.2.2 PRESSURIZATION OF CONTAINMENT SYSTEM

The C-188 sources are loaded in a shielded cell at atmospheric pressure and source equilibrium temperature <u>or</u> from the underwater pool facility under a 30 ft. head of water and at pool water temperature. When the C-188 sources are in the F-294 container cavity, they reach an equilibrium temperature in 24 hours of 824°F for a 374 kCi of cobalt-60 loading. The lower cavity wall temperature is 387°F. The lower cavity is subjected to internal pressure of 16 psig, as calculated in section 4.2.2.1.

4.2.2.1 Maximum Internal Pressure

In the cavity of the inner assembly of F-294 container, the pressure build up is as follows:

T_1 = Ambient temperature of cavity prior to source loading = 70°F
P_1 = Pressure of the cavity prior to source loading = 14.7 psia
T_2 = Temperature of the cavity after source loading = 387°F
= Average of (C-188 temperature and cavity wall temperature.)
= (824 + 387)/2 = 605.5°F
P_2 = Pressure of the cavity after source loading = ? (unknown) psia
P_2 = P_1 x [T_2 + 460]/[T_1 + 460]
= 14.7 x [605.5 + 460]/[70 + 460]
= 14.7 x 1,066/530
= 29.6 psia
= 14.9 psig ≈ 16 psig (Design)

Therefore, the cavity of the F-294 in normal conditions of transport is at 16 psig and avg. temperature of 606°F. The lower cavity wall temperature is 387°F.

4.2.3 STRESS ANALYSIS OF CLOSURE PLUG AND INNER SHELL ASSEMBLY

Detailed analysis of the components of the containment system of F-294 during normal conditions of transport (NCOT) is carried out in Appendix 4.4.3 to demonstrate that the inner shell assembly and the closure plug can withstand internal pressure of 16 psig at 387°F and 130 g's G-load. The following is a summary of the status of each of the components of the inner shell assembly and the closure plug under appropriate driving forces and resulting in appropriate safety factors and margin of safety.

4.2.3.1 Closure Plug Bolted Joint

For the closure plug bolted joint components, the applied stresses, safety factors and margins of safety are as follows:

1. Closure plug bolted joint subject to internal pressure and G-loads:
   $W_{\text{required closure plug,NCOT}}$ = 152,600 lb.
   $W_{\text{bolt load available, NCOT}}$ = 911,000 lb.
   Safety factor = 5.97

As the Safety Factor (SF) is greater than 1, the bolted joint as specified shall be maintained during NCOT of F-294 package.

2. Bolt stress due to applied torque of 100 ft.-lb. per bolt
   Applied stress = 20,500 psi
   Allowable stress = 100,000 psi
   Safety Factor (SF) = 4.87
   Margin of Safety = SF – 1 = 4.87 – 1 = 3.87
3. Stress in the threads of the bolt hole
   Applied stress = 6,630 psi
   Allowable stress = 11,860 psi
   Safety Factor (SF) = 1.78
   Margin of Safety = SF – 1 = 1.78 – 1 = 0.78

4. Closure plug: in side drop
   4.1 Closure plug bolts
   Applied stress = 16,100 psi
   Allowable stress = 100,000 psi
   Safety Factor (SF) = 6.21
   Margin of Safety = SF – 1 = 6.21 – 1 = 5.21
   4.2 Plug cylindrical body: side impact:
   Applied compressive stress = 900 psi
   Allowable stress = 11,860 psi
   Safety Factor (SF) = 13.1
   Margin of Safety, MS = SF – 1 = 13.1 – 1= 12.1
   4.3 Plug weld WPC1 and WPC2:
   Applied shear stress = 6,530 psi
   Allowable stress = 11,860 psi
   Safety Factor (SF) = 1.81
   Margin of Safety, MS = SF – 1 = 1.81 – 1 = 0.81

4.2.3.2 Inner Shell Assembly

For inner shell assembly and components, the stresses or collapse loads, safety factors and margins of safety are given below. The safety factor (SF) is defined as: allowable stress/applied stress. For buckling, the safety factor is defined as critical load or collapse load/applied load. The allowable stress is 2/3 of yield stress of material at appropriate temperature.

Various components identified in Figure 4.F2 were subjected to internal pressure of 16 psig and stress analyzed. The details of the calculations are given in Appendix 4-4-3. Summary is given here.

1. Lower Cavity wall due to build -up of internal pressure:
   σ_{hoop} = 192 psi, SF = 56.6; MS = 55.6
2. Upper Cavity wall due to build-up of internal pressure:
   σ_{hoop} = 243 psi; SF = 44.7; MS = 43.7
3. The bending stress in the lower cavity end cap, without taking restraint of lead into account, is σ_b = 1,693 psi; SF = 11.8; MS = 10.8
4. The bending stress in the upper cavity ring flange, without taking restraint of lead into account, is s_r = 78 psi; SF = 139; MS = 138

Summary of stress analysis of inner shell assembly components under NCOT G-loads:

Various components identified in Figure 4.F2 were subjected to G-loads of 130 – 136 g's and stress analyzed. The details of the calculations are given in Appendix 4-4-3. Summary is given here.

1. Lower cavity tube buckling:
   Applied load = 73,580 lb.
   Collapse load = 247 x 10^6 lb.
   SF = 3,356; MS = 3,355

2. Lower cavity tube end cap:
   Bending stress = 28,213 psi.
   Allowable stress = 41,000 psi.
   SF = 1.45; MS = 0.45
3. Upper cavity tube buckling:
   Applied load = 174,340 lb.
   Collapse load = 1,601. x 10^6 lb.
   SF = 9,183; MS = 9,182
4. Bending of upper cavity ring flange
   Bending stress = 7,330 psi.
   Allowable stress = 11,670 psi.
   SF = 1.48; MS = 0.48
5. Container top ring flange
   Bending stress = 3,260 psi
   Allowable stress = 11,860 psi.
   SF = 3.63; MS = 2.63

As the safety factors (SF) > 1 and as the margin for safety (MS) > 0, it has been demonstrated analytically that the bolted closure joint over the inner shell assembly of F-294 shall be maintained to resist internal pressure and G-loads encountered in the F-294 containment system during the normal conditions of transport of F-294.

The purpose of the closure plug bolted joint is to provide adequate shielding but not necessarily leaktightness of the joint between the closure plug and the inner shell assembly. It has been demonstrated that the closure plug is retained when F-294 is subject to combination of forces during normal conditions of transport. Consequently the closure plug, which provides adequate shielding, shall be retained over the cavity of F-294, which houses Special Form cobalt-60 MDS Nordion C-188 sealed sources. As C-188 is certified Special Form RAM, the C-188 outer encapsulation provides the leaktightness for retention of radioactive material.

Therefore

1. as C-188 is certified Special Form RAM and provides leak tight containment AND
2. as the closure plug (the shielding) is retained over the inner shell assembly, which houses the cobalt-60 C-188 sealed sources,

F-294 shall meet the NCOT containment system requirements (10 CFR 71.51 (a) (1)).

4.2.4 COOLANT CONTAMINATION

This factor is not applicable to the F-294 since it does not contain any coolant.

4.2.5 COOLANT LOSS

Not applicable as per section 4.2.4 discussion above.

4.3 CONTAINMENT REQUIREMENTS FOR HYPOTHETICAL ACCIDENT CONDITIONS

A full-scale F-294 test packaging was subjected to eight drop tests, as listed below:

Test #1:	Normal Free Drop Test: top end drop orientation
Test #2:	30 ft. Free Drop: side oblique drop orientation
Test #3C:	Puncture Test: impact on the zone near lift lug fin #4
Test #4:	Puncture Test: impact cylindrical fireshield
Test #5:	Puncture Test: impact on fixed skid lower plate
Test #6:	30 ft. Free Drop Test: top end drop orientation.
Test #7:	Puncture Test: impact on the crush shield upper plate
Test #8:	Puncture Test: impact cylindrical fireshield (nameplate zone)

In the cavity of the F-294 test packaging, the radioactive contents were simulated using eight (8) dummy, full-scale C-188 capsules in a full-scale F-313 source carrier.

The drop tests were conducted on February 25, 1998 at the Chalk River Laboratory of Atomic Energy of Canada, Chalk River, Ontario, Canada.

The drop test results are presented in Chapter 2, Appendix 2.10.12.

The dummy C-188s were helium leak tested prior to the drop tests and after the drop tests.

The cavity of F-294 was air pressure tested and helium leak tested prior to the drop tests and after the drop tests.

The test results, as it pertains to the issue of the integrity of the F-294 containment system, are re-captured and presented in Chapter 4, Appendix 4.4.7.

In the drop tests, when the F-294 is subjected to eight drop tests, the deceleration loads are as per Table 4.4-T1. The location of accelerometers are as per Figure 4.4-F7.

In the hypothetical accident conditions of transport of F-294 loaded with 360 kCi of cobalt-60 in C-188 sealed sources, the inner shell assembly of F-294 is subjected to 20 psig internal pressure. The lower cavity wall is at 500°F and the top closure plug is ≈ 531°F (see Chapter 3, Section 3.5).

Using the above data (deceleration G-loads, internal pressure & temperatures in the F-294 cavity), the stress analysis of the components of the inner shell assembly and the closure plug of F-294 is carried out and presented in Appendix 4.4.6. This stress analysis is supplemental to the actual drop tests carried out on the full-scale F-294 test packaging, as the F-294 test packaging containment system was leak tested prior to and after the eight (8) drop tests. (For details see Appendix 4.4.7). Results are expected to be similar for the F-294, with the F-457 source carrier, as for the F-294 with the F-313 source carrier because the F-457 source carrier, loaded with 80 C-188 capsules, weighs only 23 lb. more than the fully loaded F-313 source carrier. Also, the maximum activity is the same regardless of the configuration, therefore the total heat generated within the container will remain the same.

Table 4.4-T1
Maximum Absolute Decelerations for F-294 Transport Packaging (g's)

Test #	1	2	3	4	5	6	7	8
Accelerometer Location G1	116	136[2]	LOS	20	46	132[3]	60	22
Accelerometer Location G2	113	LOS	LOS	LOS	LOS	LOS	LOS	LOS
Accelerometer Location G3	130	66	LOS	26	58	118	32	14
Accelerometer Location G4	277[1]	73	23	35	0	0	50	15

Notes:

1. The very high G4 value for Test No. 1 is not valid. G4 is mounted on the bottom of container fixed skid, a thin plate supported around the perimeter of the skid acting as a diaphragm. As per Figure 6 of A-16485-TN-1 (designated Ref [10]), page 9, the maximum level attained by G4 does not occur until after the initial impact.

   The crush shield does not significantly deform during this test. The rigidity of the F-294 package in this orientation and drop speed is a possible cause for the high deceleration value observed.
2. Test No. 2 is the first 30 ft. drop. G1 is very near the impact point for Test No. 2. It is observed to measure the highest deceleration value. G3 and G4 are located further away from the impact point, and are subject to "pivoting" effect on impact. The crush shield fins on the impact target are greatly deformed, helping to reduce the maximum deceleration value.
3. Test No. 6 is the second 30 ft. drop. The maximum value attained by G1 for Test No. 6 is similar to that maximum attained for the first 30 ft. drop, Test No. 2. Again, the crush shield fins on the impact target are greatly deformed, helping to reduce the maximum deceleration value.
4. LOS = Loss of signal.

4.3.1 PRESSURIZATION OF CONTAINMENT SYSTEM

The C-188 sources are loaded in a shielded cell at atmospheric pressure and source equilibrium temperature or from the underwater pool facility under a 30 ft. head of water and at pool water temperature. When the C-188 sources are in the F-294 container cavity, they reach an equilibrium temperature in 24 hours of 824°F for a 374 kCi loading.

The cavity wall temperature is 387°F. The cavity is subjected to internal pressure of 16 psig per section 4.2.2.1.

These temperatures are expected to be similar for the F-294/F-313 and the F-294/F-457 package configurations since the maximum activity and the total heat generated are the same for both configurations.

4.3.1.1 Maximum Internal Pressure

In the cavity of the inner assembly of F-294 container, the pressure build up after F-294 is subject to 0.5 hour thermal test is as follows:

T_1 = Average Temperature of F-294 Cavity in NCOT = 606°F
P_1 = Pressure of the cavity in NCOT = 29.6 psia
T_2 = Average Temperature of the cavity after fire test = 728°F
P_2 = Pressure of the cavity after fire test = ? (unknown) psia

$$
\begin{aligned}
P_2 &= P_1 \times [T_2 + 460]/[T_1 + 460] \\
&= 29.6 \times [728 + 460]/[606 + 460] \\
&= 29.6 \times 1{,}188/1{,}066 \\
&= 33.0 \text{ psia} \\
&= 18.3 \text{ psig.} \\
&= 20 \text{ psig (design).}
\end{aligned}
$$

Therefore, the cavity of the F-294 in hypothetical accident conditions of transport (HACOT) is subjected to an internal pressure of 20 psig and an average environment temperature of 721°F. The cavity wall temperature is 500°F. These temperatures are expected to be similar for the F-294/F-313 and the F-294/F-457 package configurations since the maximum activity and the total heat generated are the same for both configurations.

4.3.2 STRESS ANALYSIS OF CLOSURE PLUG AND INNER SHELL ASSEMBLY

Detailed analysis of the components of the containment system of F-294 during HACOT is carried out in Appendix 4.4.6 to demonstrate that the inner shell assembly and the closure plug can withstand internal pressure of 20 psig at 500°F and 132 g's G-load in top drop orientation and 136 g's in side oblique drop orientation of F-294.

The following is a summary of the status of each of the components of the inner shell assembly and the closure plug under appropriate driving forces and resulting in appropriate safety factors. In general when the Safety Factor (SF) > 1 and the Margin of Safety (MS) > 0, it is demonstrated that the F-294 component design is adequate during Hypothetical Accident Conditions of Transport (HACOT) of F-294 package.

4.3.2.1 Closure Plug Bolted Joint

For the closure plug bolted joint, the stresses, safety factors and margins of safety are:

1. Closure plug bolted joint subject to internal pressure and G-loads:

   $W_{\text{required closure plug, HACOT}}$ = 153,200 lb.
   $W_{\text{bolt load available, HACOT}}$ = 1,586,000 lb.
   Safety factor (SF) = 10.35
   Margin of Safety = SF − 1 = 10.35 − 1 = 9.35

   As the margin of safety (MF) is greater than zero, the bolted joint as specified shall be maintained during HACOT of F-294 package.

2. Bolt Stress due to applied torque + g-load (132 g's)

   Applied stress = 31,060 psi
   Allowable stress = 180,000 psi
   Safety Factor (SF) = 5.79
   Margin of Safety = SF − 1 = 5.79 − 1 = 4.79

3. Stress in the threads of the bolt hole

   Applied stress = 6,650 psi
   Allowable stress = 70,000 psi
   Safety factor (SF) = 10.52
   Margin of Safety = SF − 1 = 10.52 − 1 = 9.52

4. Closure plug: in side drop

 4.1 Closure plug bolts

 Applied stress = 16,500 psi
 Allowable stress = 180,000 psi
 Safety Factor (SF) = 10.9
 Margin of Safety = SF – 1 = 10.9 – 1 = 9.9

 4.2 Plug cylindrical body: side impact:

 Applied compressive stress = 940 psi
 Allowable stress = 70,000 psi
 Safety Factor (SF) = 74.4
 Margin of Safety, MS = SF – 1 = 74.4 – 1 = 73.4

 4.3 Plug weld WPC1

 Applied shear stress = 13,600 psi
 Allowable stress = 70,000 psi
 Safety Factor (SF) = 5.14
 Margin of Safety, MS = SF – 1 = 5.14 – 1 = 4.14

4.3.2.2 Inner Shell Assembly

For inner shell assembly and components, the stresses or collapse loads, safety factors and margins of safety are given below. The safety factor is defined as: allowable stress/applied stress. The allowable stress is ultimate tensile stress of material at appropriate temperature.

Various components identified in Figure 4.F2 were subjected to internal pressure of 20 psig and stress analyzed. The details of the calculations are given in Appendix 4-4-6. Summary is given here.

1. Lower Cavity wall due to build-up of internal pressure:

 σ_{hoop} = 240 psi, SF = 250, MS = 249

2. Upper Cavity wall due to build -up of internal pressure:

 σ_{hoop} = 304 psi. SF = 197; MS = 196

3. The bending stress in the lower cavity end cap, without taking restraint of lead into account, is:

 σ_b = 940 psi; SF = 74.4; MS = 73.4

4. The bending stress in the upper cavity ring flange, without taking restraint of lead into account, is:

 s_r = 98 psi; SF = 615; MS = 614

Summary of stress analysis of inner shell assembly components under HACOT g-loads:

Various components identified in Figure 4.F2 were subjected to G-loads of 132 g's and stress analyzed. The details of the calculations are given in Appendix 4.4.6. Summary is given here.

1. Lower cavity tube buckling:
   Applied load = 74,710 lb.
   Collapse load = 247 x 10^6 lb.
   SF = 3,306; MS = 3,305
2. Lower cavity tube end cap:
   Bending stress = 28,636 psi.
   Allowable stress = 70,000 psi.
   SF = 2.4; MS =1.4
3. Upper cavity tube buckling:
   Applied load = 200,100 lb.
   Collapse load = 1,601 x 10^6 lb.
   SF = 8,000; MS = 7,999
4. Bending of upper cavity ring flange
   Bending stress = 8,345 psi.
   Allowable stress = 60,000 psi.
   SF = 7.18; MS = 6.18
5. Container top ring flange
   Bending stress = 3,710 psi
   Allowable stress = 60,000 psi.
   SF = 161; MS = 160

As the safety factors > 1 and as the margin for safety is > 0, the bolted closure joint over the inner shell assembly of F-294 shall be maintained to resist internal pressure and G-loads during the hypothetical accident conditions of transport of F-294.

The purpose of the closure plug bolted joint is to provide adequate shielding but not necessary leaktightness of the joint between the closure plug and the inner shell assembly. It has been demonstrated that the closure plug is retained when F-294 is subject to combination of forces during hypothetical accident conditions of transport. Consequently the closure plug, which provides adequate shielding, shall be retained over the cavity of F-294, which houses Special Form cobalt-60 MDS Nordion C-188 sealed sources. As C-188 is certified Special Form RAM, the C-188 outer encapsulation provides the leaktightness for retention of radioactive material.

Therefore

1. as C-188 is certified Special Form RAM and provides leak tight containment AND
2. as the closure plug is retained over the inner shell assembly, which houses the cobalt-60 C-188 sealed sources,

F-294 shall meet the hypothetical accident conditions of transport (HACOT) containment system requirements (10 CFR 71.51 (a) (2)).

4.3.3 IMPACT OF THE PUNCTURE TEST

A single full-scale F-294 test packaging was subjected to eight (8) drop tests. Out of eight (8) drop tests, five (5) drop tests were 40-in drop onto a puncture pin. The drop orientations were all around the F-294 test packaging, e.g.,

Test #3C:	Puncture Test: impact on the zone near lift lug fin #4 (top corner)
Test #4:	Puncture Test: impact cylindrical fireshield (side, mid-height)
Test #5:	Puncture Test: impact on fixed skid lower plate (bottom)
Test #7:	Puncture Test: impact on the crush shield upper plate (top)
Test #8:	Puncture Test: impact cylindrical fireshield (nameplate zone) (side, mid-height)

The details of the drop test results are given in Chapter 2, Appendix 2.10.12. The test results, as it pertains to the issue of the integrity of F-294 containment system, are re-captured and presented in Chapter 4, Appendix 4.4.7.

As a result of F-294 test packaging having subjected to five (5) puncture pin drop tests, there was no significant effect on the closure plug of the F-294 nor the C-188 source capsule. Therefore, it is concluded that no breach of the F-294 containment system will occur as a result of this test.

4.3.4 IMPACT OF THE WATER IMMERSION TEST

The water immersion test discussed in Chapter 2, Section 2.7.4 will have no significant effect on the C-188 source capsules since they are designed and certified to withstand a minimum external pressure of 10000 psia (Appendix 4.4.2) which is significantly greater than the pressure increase of 21 psi resulting from this test.

In addition, in Chapter 2, Appendix 2.10.5, it is demonstrated that the container outer shell assembly (OD = 36 in.; wall = 0.5 in thick) of F-294 is capable of withstanding 664 psig external pressure, without taking credit of the cooling fins welded to the container outer shell. As the OD of the inner shell assembly (15.8 OD x 0.5 in. thick wall min.) is less than the OD of the container outer shell assembly (36 OD x 0.5 in wall), the container inner shell assembly is capable of withstanding external pressures greater than 664 psig. 664 psig capability is greater than the required test pressure of 21 psi.

The hypothetical accident conditions of transport, thermal evaluation is discussed in Chapter 3, Section 3.5 of this report. The maximum C-188 source capsule temperature predicted, using steady state normal thermal test data and transient 0.5 hour fire test simulation data, is 940°F for the 360 kCi case, which is below the testing temperature of 1472°F(800°C) which is required for Special Form to which the C-188 source capsule is certified (Ref. [3] and Appendix 4.4.2). The maximum source temperature will be higher for the F-294/F-457 package configuration since the maximum source temperature at steady state is higher than in the F-294/F-313 configuration and can be approximated at 955°F for the 360 kCi case. This is below the testing temperature of 1472°F (800°C) to which the C-188 capsule was tested.

Due to the rise in temperature, there is a pressure build-up in the C-188. In Appendix 4.4.5, calculations are presented to evaluate the stresses in the C-188 due to pressure build-up. The following is the summary of findings:

Due to internal pressure of 27 psig in the C-188 during HACOT of F-294,

1. the hoop stress in the tube away from joint = 192 psi
2. the hoop stress in the tube at the joint = 288 psi
3. the bending stress in the end cap = 5 psi.

Based on yield stress of 15,000 psi for ss316L at 955°F, the C-188 has a Factor of Safety of 52 and Margin of Safety of 51.

In summary, it is concluded that there will be no breach of containment system as a result of any of the tests representing hypothetical accident conditions of transport (HACOT).

4.3.5 FISSION GAS PRODUCTS

This requirement is not applicable since the F-294 does not contain any fissile material.

4.3.6 RELEASE OF CONTENTS

As cobalt-60 MDS Nordion C-188 sealed source meets the Special Form Radioactive Material requirements, it is concluded that there will be no breach of containment when C-188's are confined to the cavity of F-294 and therefore there will be no release of radioactive material. Therefore, the additional requirements of this section are not applicable.

4.4 APPENDICES

The following appendices are included here to support the information for this chapter.

Appendix 4.4.1	List of References for Chapter 4
Appendix 4.4.2	C-188 Certificates and Other Supporting Documentation
Appendix 4.4.3	Stress Analysis of the Containment System during Normal Conditions of Transport of F-294
Appendix 4.4.4	C-188 Structural Integrity under Normal Conditions of Transport Tests
Appendix 4.4.5	C-188 Structural Integrity under Hypothetical Accident Conditions of Transport Tests
Appendix 4.4.6	Stress Analysis of the Containment System Subject to Hypothetical Accident Conditions of Transport of F-294.
Appendix 4.4.7	F-294 Prototype Container Testing: Drop Test Data Relevant to the Containment System

Figure 4-F1
Containment System of F-294

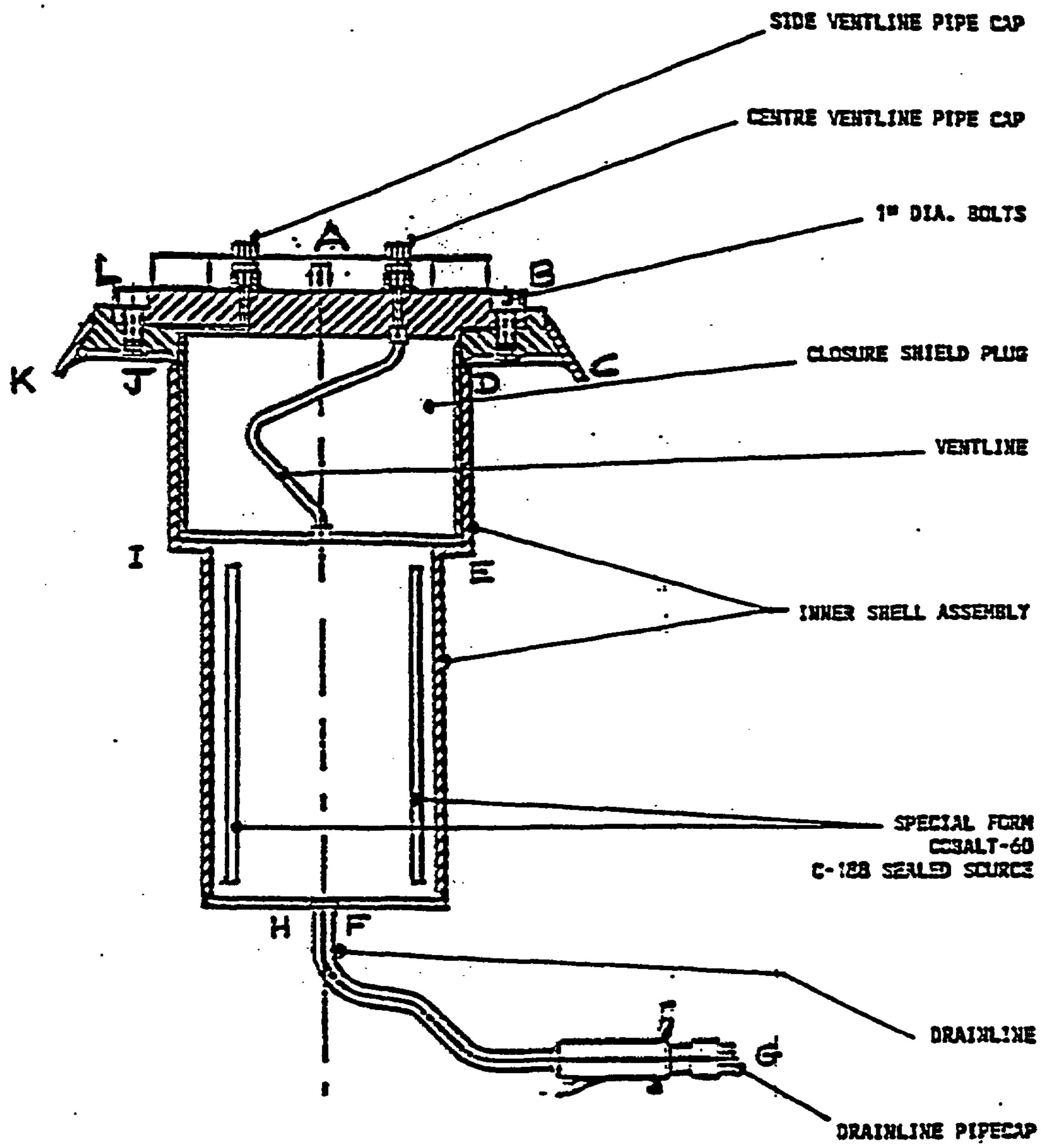


Figure 4-F2
Components of the F-294 Containment System

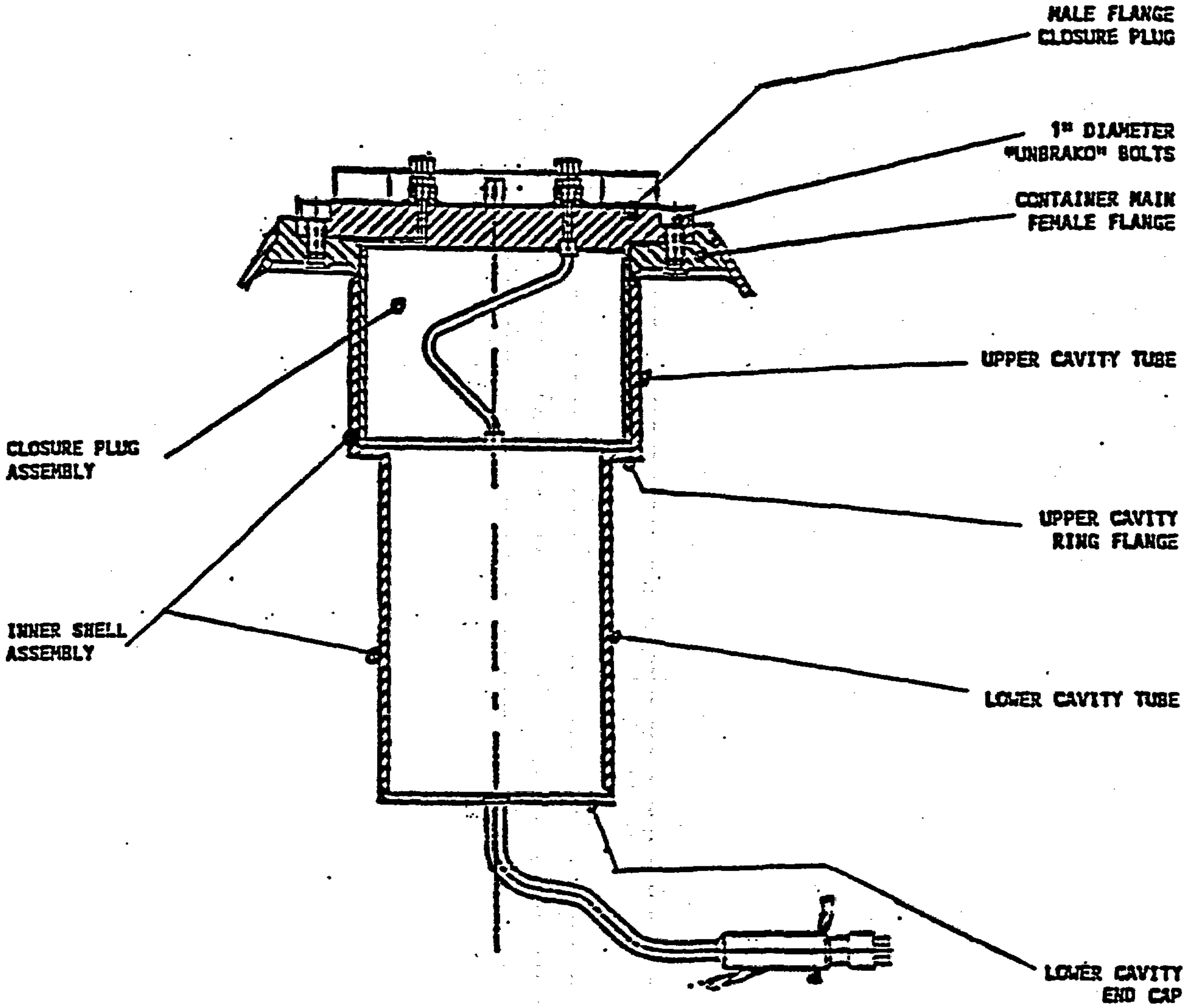


Figure 4-F3
Cobalt-60 MDS Nordion C-188 Sealed Source

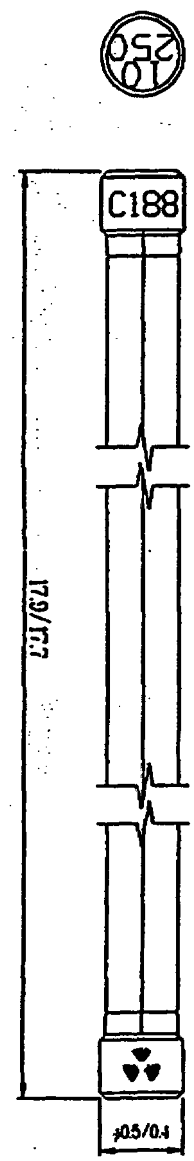


Figure 4-F4
Seals in the Containment System

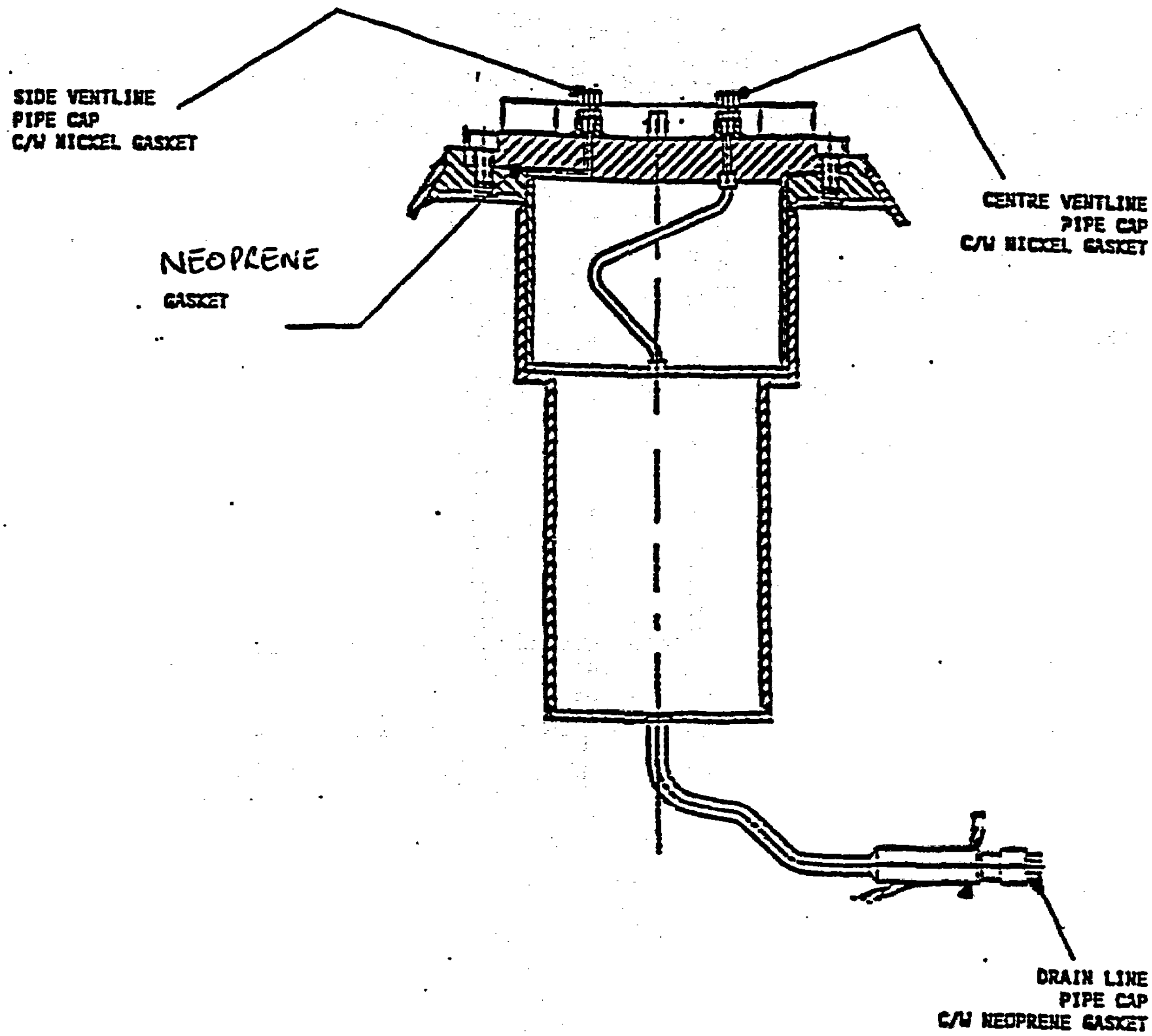


Figure 4-F5
Welds of the Containment System

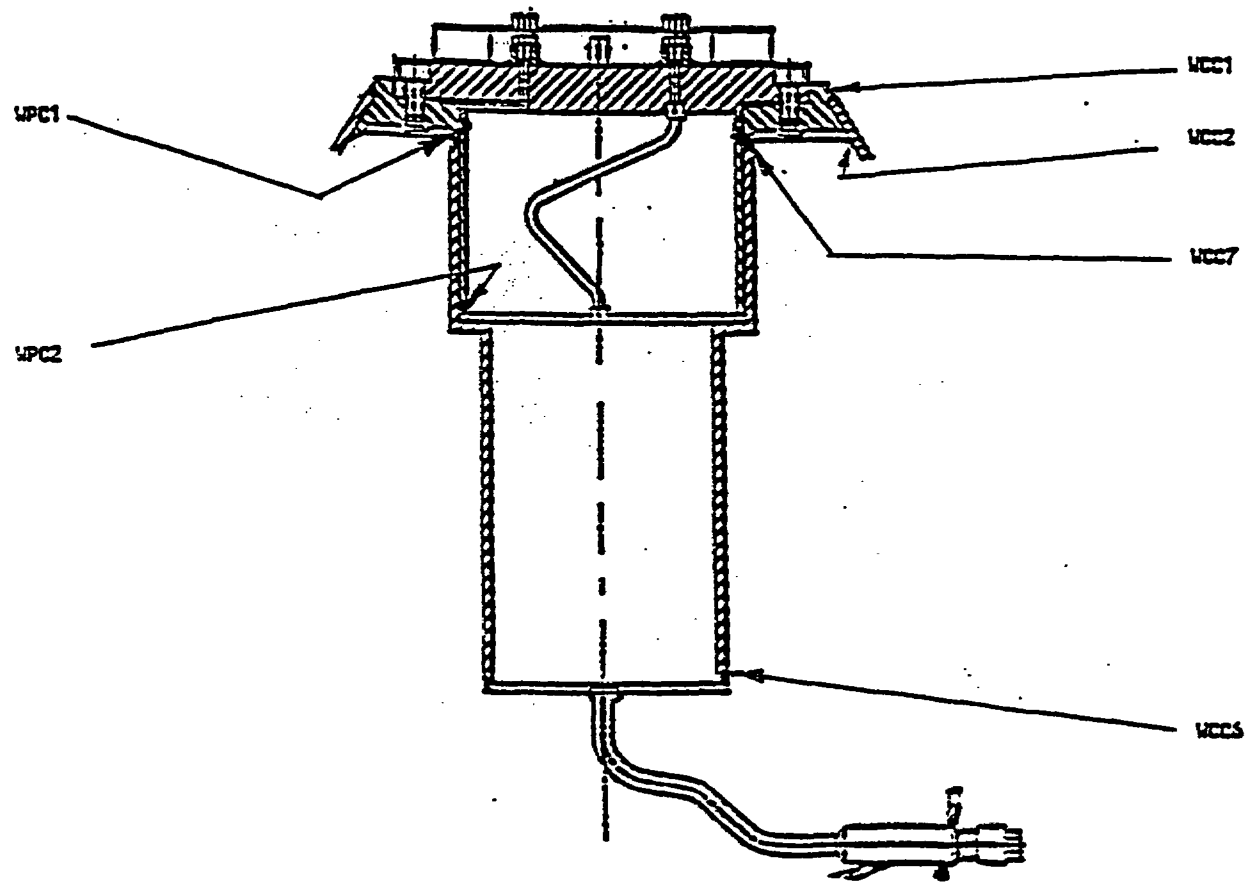


Figure 4-F6
Bolted Closure of F-294 Containment System

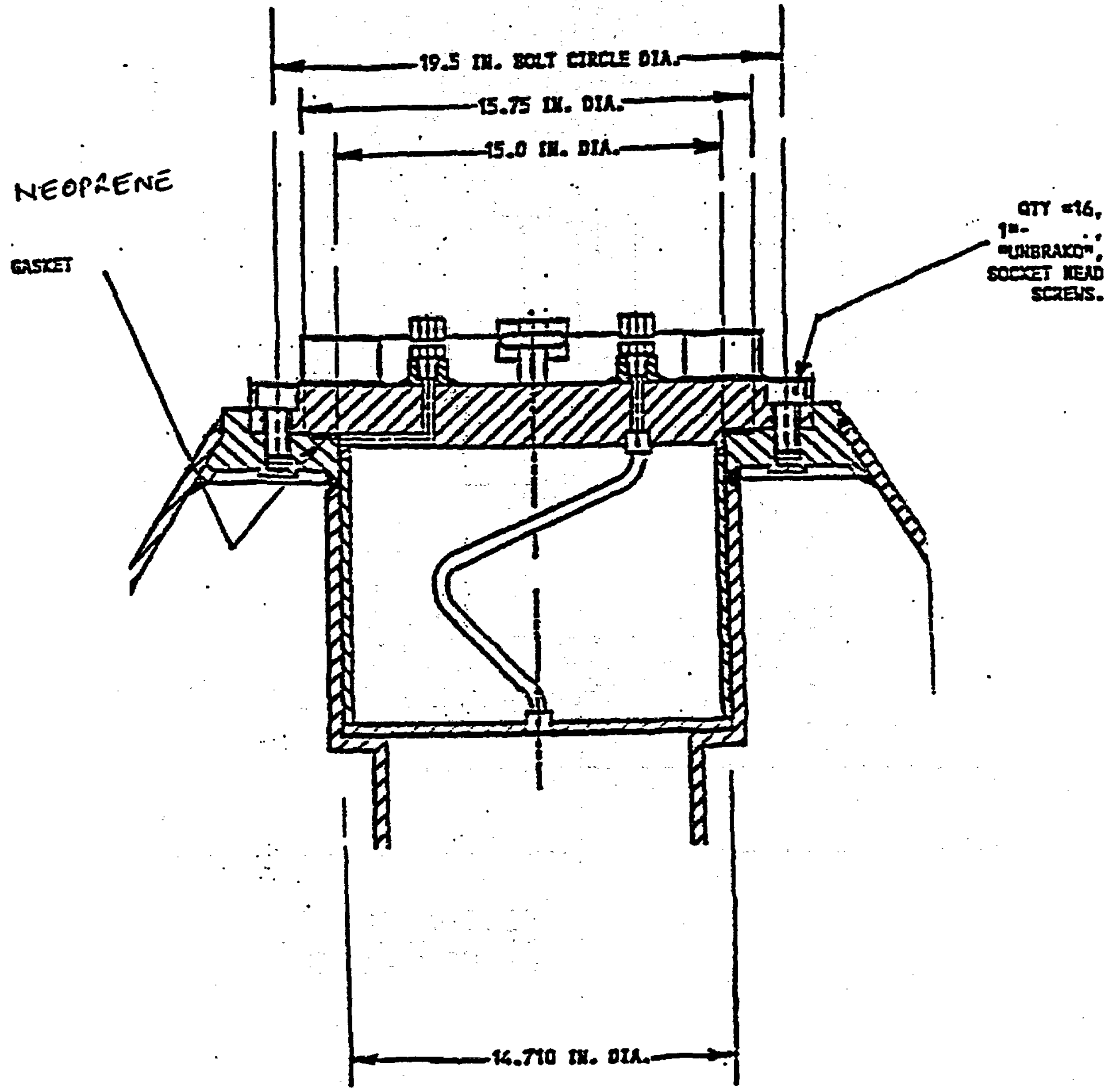
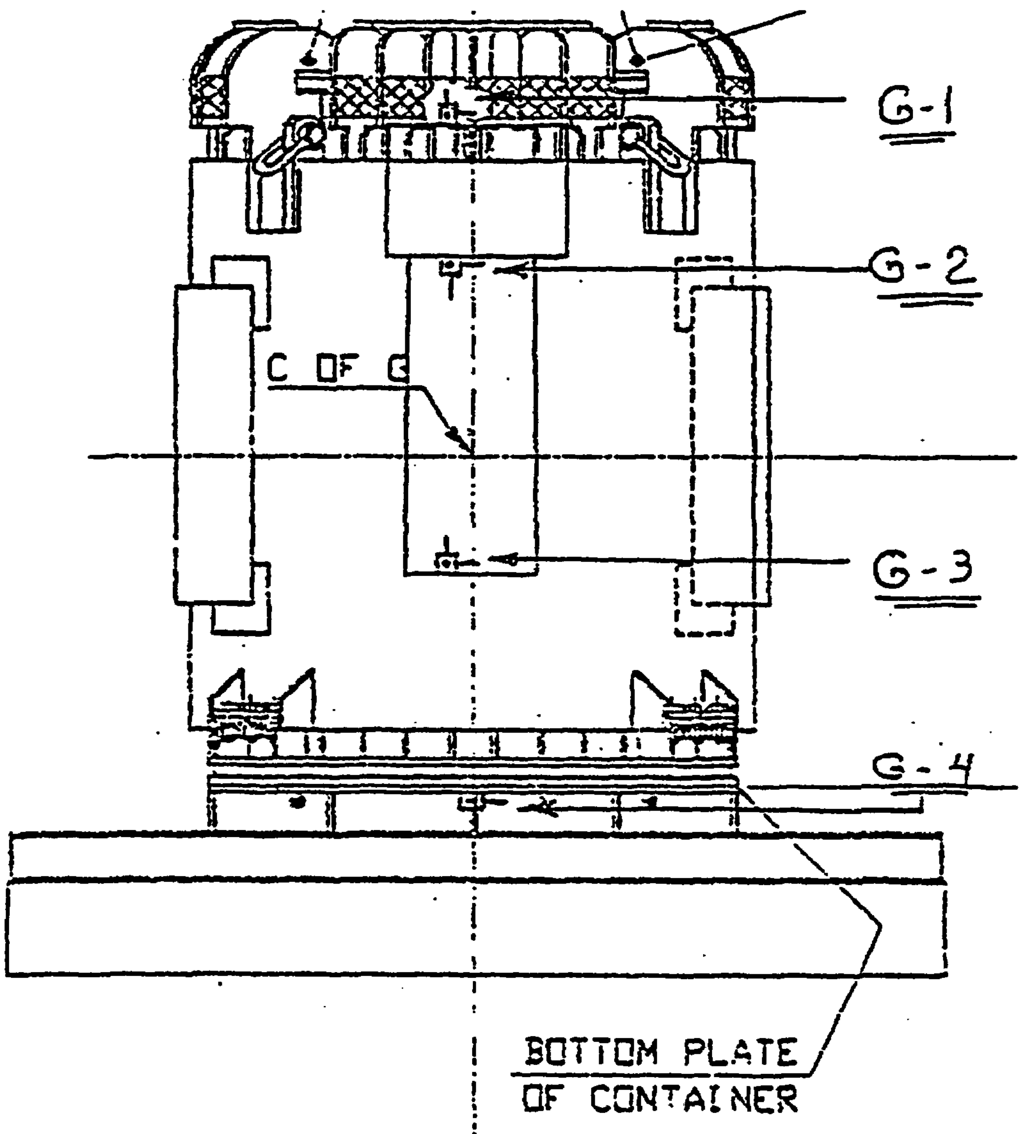


Figure 4-F7
Location of Accelerometers on F-294 Test Packaging



APPENDIX 4.4.1
LIST OF REFERENCES FOR CHAPTER 4

[1] 10 CFR, Chapter 1, Part 71 - *Packaging and Transportation of Radioactive Material*, 1-1-91 Edition.

[2] IAEA Safety Standard, Safety Series No. 6, *Regulations for the Safe Transport of Radioactive Material*, 1985 Edition (as amended 1990).

[3] *NBS Handbook 126*, American National Standard N542; Sealed Radioactive Sources, Classification, 1977, US Department of Commerce/National Bureau of Standard.

[4] ANSI N43.10, Safe Design and Use of Panoramic, Wet Source Storage Gamma Irradiators (Category IV).

[5] Canadian Institute of Steel Construction Handbook 1967.

[6] Fraser, D.E., AECL-Commercial Products Report No. CPSR-367, "A Review of the Effects of Elevated Temperatures on Type 316L Stainless Steel for ^{60}Co Encapsulation".

[7] Nordion Document, IN/PR 0030 J1100, Technical Specification for the C-188 Sealed Source - Part I Inactive Components and Assembly.

[8] Nordion Document, IN/TS 0146 J1100, Technical Specification for the C-188 Sealed Source - Part II Active Components and Assembly.

[9] Roark, 4th Edition, *Formulas for Stresses and Strains*.

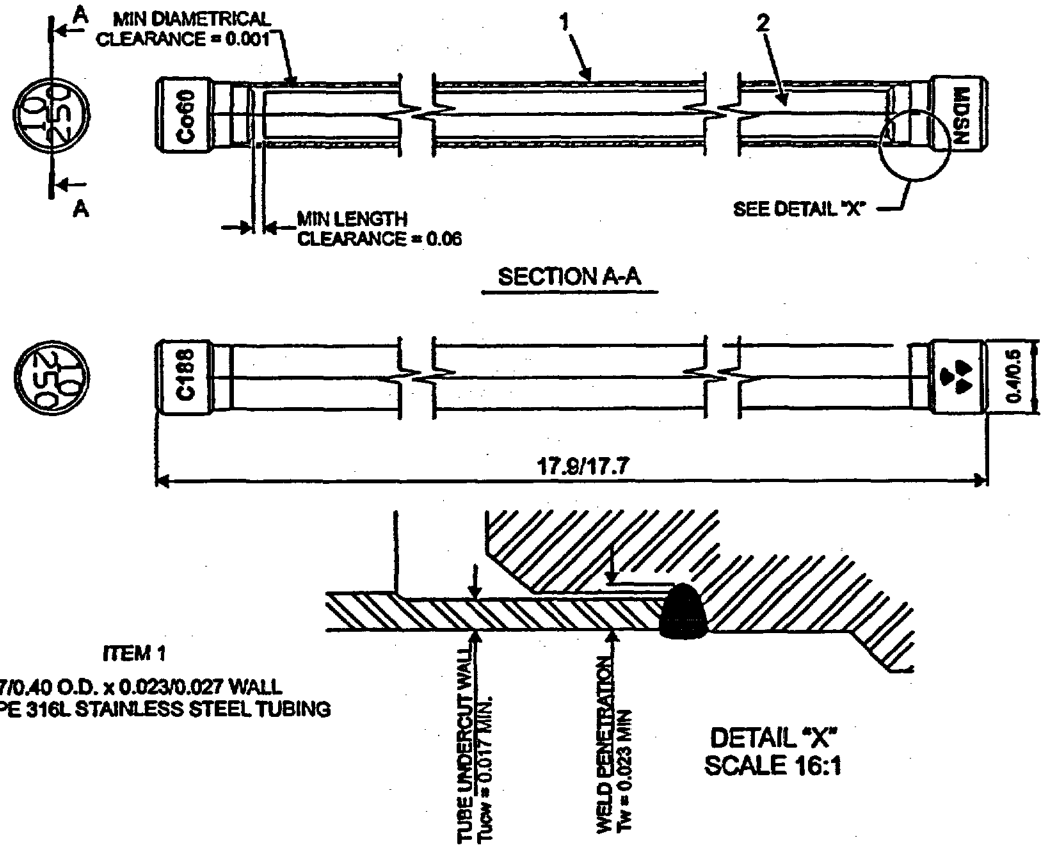
[10] Tromp, J., A-16485-TN-1: Deceleration Measurement during Drop Test of a F-294 Packaging, CRL, AECL, Chalk River, Ontario, May 1998.

[11] Birchall, R., A-16485-TN-2: Test Report: MDS Nordion F-294 Transport Packaging Testing, CRL, AECL, Chalk River, Ontario, May 1998.

APPENDIX 4.4.2
C-188 CERTIFICATES

This Appendix contains the following information and data.

1. **Cobalt-60 data sheet**
2. **8 1/2 x 11 illustration specification sheet of C-188 sealed source**
3. **MDS Nordion's ANSI Certificate No. 95 for C-188**
4. **MDS Nordion's Special Form Radioactive Material Test Summary No. 16 for C-188**
5. **Special Form RAM Certificate CDN/0010/S-96 (Rev. 6).**
6. **NR-0220-S-103-S: USNRC Source Registration for C-188**



ITEM 1
0.37/0.40 O.D. x 0.023/0.027 WALL
TYPE 316L STAINLESS STEEL TUBING

DETAIL "X"
SCALE 16:1

ITEM 2				
C-188 TYPE	INNER CAPSULE MODEL No.	# OF INNER CAPSULES	CAPSULE MATERIAL	RADIOACTIVE CONTENTS
1	C-177/C-177	2	SS 316L	PELLETS OR SLUGS
2	AC-191/AC-191	2	SS 316L	PELLETS OR SLUGS
3	AC-195/AC-195	2	ZIRCALOY 2	PELLETS OR SLUGS
4	C-246	2	SS 316L	PELLETS OR SLUGS
5	AC-339/AC-339	2	ZIRCALOY 2/4	PELLETS OR SLUGS
6	AC-345/C-348	2	ZIRCALOY 2/4 / SS 316L	SLUGS
7	C-177/AC-191	2	SS 316L / SS 316L	PELLETS OR SLUGS
8	C-177/AC-195	2	SS 316L / ZIRCALOY 2	PELLETS OR SLUGS
9	C-177/AC-339	2	SS 316L / ZIRCALOY 2/4	PELLETS OR SLUGS
10	AC-191/AC-195	2	SS 316L / ZIRCALOY 2	PELLETS OR SLUGS
11	AC-191/AC-339	2	SS 316L / ZIRCALOY 2/4	PELLETS OR SLUGS
12	AC-195/AC-339	2	ZIRCALOY 2/ZIRCALOY 2/4	PELLETS OR SLUGS
13	OTHER INNER(S)	1 or 2	SS 316L OR ZIRCALOY 2/4	SLUGS, PELLETS OR WAFERS

NOTE: ALL DIMENSIONS IN INCHES

MDS Nordion
447 March Road, P.O. Box 13500
Kanata, Ontario, Canada, K2K 1X8
Tel: (613) 592-2790 · Fax. (613) 592-6937

TITLE
C-188 Sealed Source USNRC Registration

REF. IN/SS 1381 C188	REVISED FEB 00	DCN A-1688-D03-A	
DATE Feb 98	No. C-188(U)		ISSUE 3
DRAWN / CHECKED / APPROVED	SHEET 1 OF 1		

MDS Nordion

CERTIFICATE

SEALED SOURCE
CLASSIFICATION DESIGNATION AND PERFORMANCE

Sealed sources are classified in accord with standards established by
THE AMERICAN NATIONAL STANDARDS INSTITUTE (ANSI) and
THE INTERNATIONAL ORGANIZATION FOR STANDARDIZATION (ISO)

CERTIFICATE NO: 95 DATE: 99-12-12

CAPSULE MODEL: C-188 (Bounding Conditions) CONTENTS: Cobalt -60

TEST REFERENCE REPORT: IN/TR1382 C188

CAPSULE MATERIAL: 316L Stainless steel OVERALL DIAMETER: 0.380"

ENCAPSULATION: Double OVERALL LENGTH: 17.777"

ANSI CLASSIFICATION AND PERFORMANCE STANDARD(1) ANSI 96:E64424

CLASSIFIED PERFORMANCE STANDARD(2)

TEST	CLASS	METHOD	REMARKS
TEMPERATURE	6	Test	Pass
EXTERNAL PRESSURE	4	Test	Pass
IMPACT	4	Test	Pass
VIBRATION	2	Test	Pass
PUNCTURE	4	Test	Pass

(1) See definition on reverse side
(2) See Table 1. Performance Standards on reverse side
(3) American National Standard N43.6-1997 is a revision of ANSI N542-1977

(4) Exceptions: Category II, III Gamma irradiation sources- Temperature class 5.

COMMENTS: Capsule integrity assured by Helium leak tests (ANS /HPS N43.6-1997, paragraphs A2.2.6 to A2.2.6.3.)

It is hereby certified that the described sealed source meets the specified standard as prescribed in (3) American National Standard N43.6-1997 "Sealed Radioactive sources, Classification". This standard complies with the classification and performance requirements of ISO 2919-1999(E), (see Note 4 above for exceptions).

Tested by [signature] Approved [signature]

Title Materials Specialist Title Manager, Package Engineering

Date 99-12-14 Date 99/12/15

MDS Nordion

SPECIAL FORM RADIOACTIVE MATERIAL TEST SUMMARY

The capsule model specified herein has been evaluated in accord with the International Atomic Energy Agency (I.A.E.A.) Safety Series No. 6, Regulations for the Safe Transport of Radioactive Material, 1985 Edition, (as amended 1990) Section VI, paragraphs 604-613 and 618.

TEST SUMMARY: 41 | DATE: 99-12-12

CAPSULE MODEL: C-188 (Bounding Conditions) | CONTENTS: Cobalt-60

TEST REFERENCE REPORT: IN/TR 1596 C188

CAPSULE MATERIAL: 316L Stainless Steel | OVERALL DIAMETER: 0.380"

ENCAPSULATION: Double | OVERALL LENGTH: 17.777"

SPECIAL FORM REQUIREMENTS(1)

TEST	PASS	FAIL	METHOD	REMARKS
IMPACT (607)(618)	X		Comparison	See Comments 3, 4, 5
PERCUSSION (608)	X		Comparison	See Comments 3, 4, 5
BENDING (609)	X		Test	Pass
HEAT (610)	X		Comparison	See Comments 2, 4, 5
LEACHING (612)(613)	---		---	See Comment 1

(1) See Special Form requirements on reverse side

COMMENTS: 1) Capsule integrity following bend test was verified by helium leak testing.

2) Paragraph 611(b), Safety series no. 6 specifies that the requirements of the Heat Test can be satisfied with the completion of a Class 6 Temperature Test per ISO 2919-1980 (E).

3) Paragraph 611(a), Safety series no. 6 specifies that the Impact and Percussion performance tests can be satisfied with the completion of a Class 4 Impact test per ISO 2919-1980 (E).

4) ISO 2919-1980 (E) Heat and Impact Tests are identical to ANS N43.6-1997.

5) The test reports for the Class 6 Temperature Test and the Class 4 Impact Tests are attached to ANSI certificate #95.

This summary verifies that the described capsule model meets the requirements of Special Form in accordance with the I.A.E.A. Safety Series No. 6, Regulations for the Safe Transport of Radioactive Material, 1985 Edition, (as amended 1990) Section VI, paragraphs 604-613 and 618.

Tested by J. Culbertson
Title Materials Specialist
Date 99-12-15

Approved M. Lyamil
Title Manager, Package and Facility Engineering
Date 99/12/16

Certification

Canadian Nuclear Safety Commission | **Commission canadienne de sûreté nucléaire**

SPECIAL FORM RADIOACTIVE MATERIAL CERTIFICATE NO. CDN/0010/S-96, (REV. 6)

30-A2-187-0 January 21, 2003

The Canadian Nuclear Safety Commission hereby certifies that the capsule, as described below, has been demonstrated to meet the regulatory requirements prescribed for special form radioactive material as defined in the Canadian *Packaging and Transport of Nuclear Substances Regulations* [1] and in the IAEA Regulations[2] subject to the following limitations, terms and conditions.

CAPSULE IDENTIFICATION

MDS Nordion C-188 Capsule, Types 1 to 13 inclusive.

CAPSULE DESCRIPTION

The C-188 capsule, Types 1 to 13 inclusive, as shown on MDS Nordion Drawing No. G130102-177, (Issue B) consists of an outer welded stainless steel body with solid end caps containing a variety of welded inner capsules. The overall length is 452 mm. The end cap diameters are 11.2 mm and the body diameter is 9.7 mm. The inner configurations consist of either one or two welded stainless steel or zircaloy capsules containing Cobalt-60 metal in slug, wafer or pellet form.

An illustration of the capsule is shown on attached specification Drawing No. C-188 (Issue 17).

AUTHORIZED RADIOACTIVE CONTENTS

This capsule is authorized to contain not more than 630 TBq (17,000 Ci) of Cobalt-60 in slug form or not more than 520 TBq (14,000 Ci) of Cobalt-60 in wafer or pellet form.

QUALITY ASSURANCE

The quality assurance program for the C-188 capsule shall be in accordance with the requirements of MDS Nordion Procedure No. IN/QA 0562 A000 (Issue 2) [3], "Sealed Source Quality Plan" and Technical Specifications Nos. IN/TS 1486 C188/C306 (1) [3] and IN/TS 0146 C188/C306 (5) [3] for design, manufacture, testing, documentation and inspection, as required by paragraph 310 of IAEA Regulations[2].

Canada

EXPIRY DATE

This certificate expires September 30, 2006.

P. Nelson
Designatted Officer pursuant to
subsection 37(2)(a) of the
Nuclear Safety and Control Act.

REFERENCES

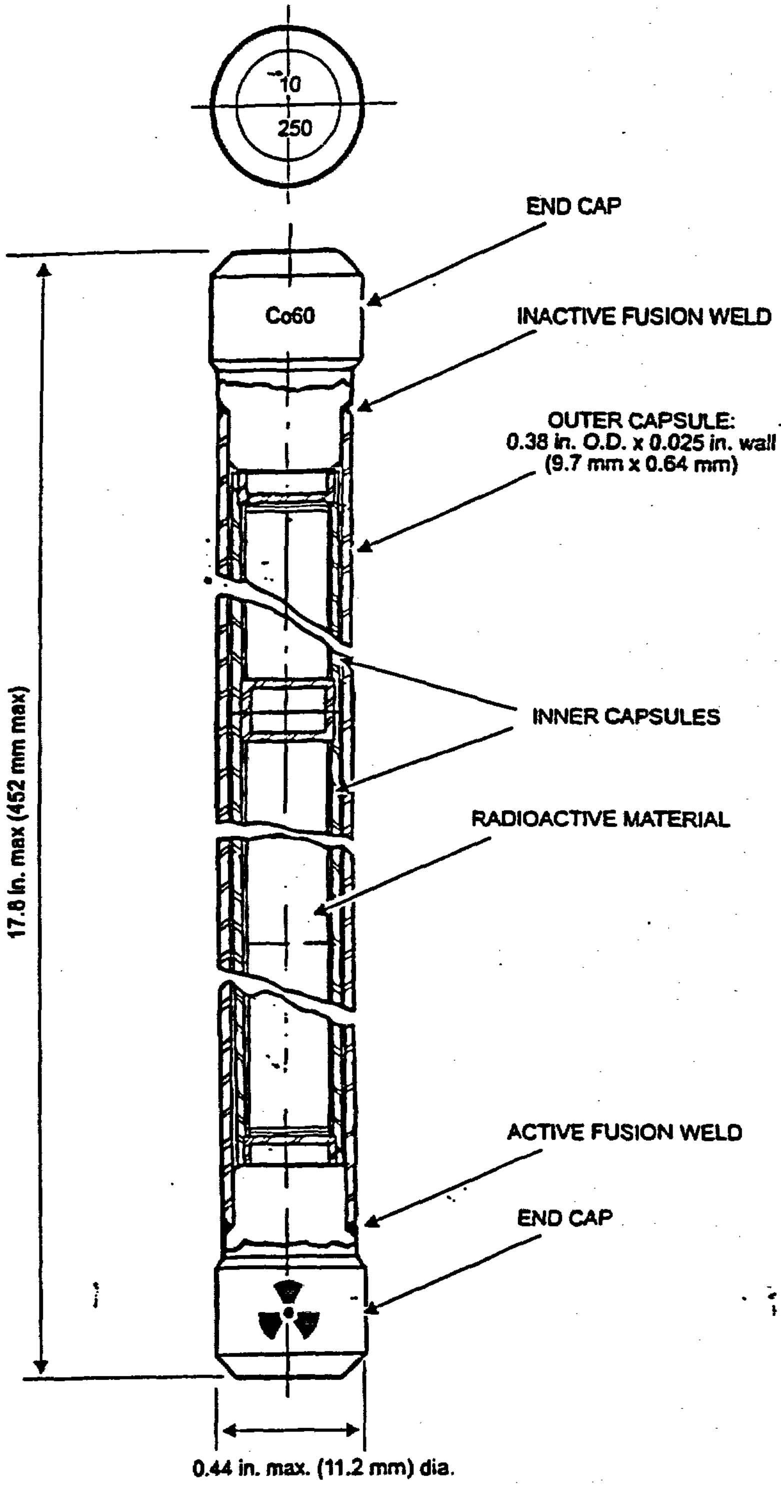
[1] Canadian *Packaging and Transport of Nuclear Substances Regulations*, SORS/2000-208, May 31, 2000

[2] International Atomic Energy Agency Safety Standards Series No. TS-R-1 (ST-1 Revised), Regulations for the Safe Transport of Radioactive Material, 1996 Edition (Revised).

[3] Or latest current revision

NOTES

1. Revision 0: November 09, 1989. Original certificate.
2. Revision 1: November 16, 1993. Issued to the IAEA 1985 Regulations.
3. Revision 2: October 04, 1994. Revised to include the "-85" suffix.
4. Revision 3: January 15, 1999. Certificate renewed.
5. Revision 4: March 07, 2000. Certificate revised.
6. Revision 5: September 26, 2002. Issued to the IAEA 1996 Regulations. Certificate renewed.
7. Revision 6: January 21, 2003. Revised to correct typographical error.



C-188 Type Number	Model Number of Inners
1.	C-177/C-177
2.	AC-191/AC-191
3.	AC-195/AC195
4.	C-246
5.	AC-339/AC-339
6.	AC-345/AC-348
7.	C-177/AC-191
8.	C-177/AC-195
9.	C-177/AC-339
10.	AC-191/AC-195
11.	AC-191/AC-339
12.	AC-195/AC-339
13.	See Note 6

Notes

1. Conforms to IAEA Special Form requirements. AECB Certificate No. CDN/0010/S-85.
2. Radioactive Material: Cobalt-60 in solid form.
3. Outer capsule material: Type 316L stainless steel.
4. All capsules are sealed by fusion welds.
5. Engraved on capsule:
   (A) Upper end cap
   face: serial number
   diameter: C188 Co60
   (B) Lower end cap diameter: MDSN X and Trefoil where X is material heat number.
6. Any inner design constructed from stainless steel or zircaloy consisting of one or more capsules containing Cobalt-60 pellets, slugs or wafers and of a design similar, but not identical to one or more of those contained in types 1 to 12.

MDS Nordion
447 March Road, P.O. Box 13500
Kanata, Ontario, Canada, K2K 1X8
Tel: (613) 592-2790 · Fax. (613) 592-6937

TITLE
C-188 Cobalt-60 Sealed Source

REF. G130102-177 IN/SS 1383 C188		REVISED FEB 00	DCN A-1688-D-01A
DATE FEB 67		No. C-188	ISSUE 17
DRAWN	CHECKED	APPROVED	SHEET 1 OF 1

REGISTRY OF RADIOACTIVE SEALED SOURCES AND DEVICES
SAFETY EVALUATION OF SEALED SOURCE
(AMENDED IN ITS ENTIRETY)

NO: NR-0220-S-103-S   DATE: 01/23/02

DEVICE TYPE: Gamma Irradiator Source

MODEL: C-188 (Series) Types 1 through 13
C-306 (Series) Types 1 and 2

MANUFACTURER/DISTRIBUTOR: MDS Nordion,
A division of MDS (Canada) Inc.
(formerly Nordion International, Inc. and Atomic Energy of Canada, Ltd.)
447 March Road
Ottawa, Ontario
Canada K2K 1X8

ISOTOPE: Cobalt-60

MAXIMUM ACTIVITY:
(C-188, slug material)
17,000 curies(629 TBq)
(C-188, wafer and pellet material)
14,000 curies(518 TBq)
(C-306)
8,500 curies(314.5 TBq)

LEAK TEST FREQUENCY: 6 months

PRINCIPAL USE: (M) Gamma Irradiator, Category IV

CUSTOM-SOURCE: ______ YES __X__ NO

DEVICE TYPE: Gamma Irradiator Source

DESCRIPTION:

The Models C-188 and C-306 are doubly encapsulated fusion welded sources consisting of one or two inner capsules and an outer capsule. The outer capsule is the same for each source model except for the length. The inner capsules vary according to the type and activity required.

The Model C-188 contains one or two inner capsules in various combinations according to type as shown in the table below:

C-188 Type Number	Model Number Of Inner(s)	C-188 Type Number	Model Number Of Inner(s)
1	C-177/C-177	7	C-177/AC-191
2	AC-191/AC-191	8	C-177/AC-195
3	AC-195/AC-195	9	C-177/AC-339
4	C-246	10	AC-191/AC-195
5	AC-339/AC-339	11	AC-191/AC-339
6	AC-345/C-348	12	AC-195/AC-339
		13	Two inners maximum.

The inner capsules vary according to the user requirements. The Model C-188 outer encapsulation is constructed of 316L stainless steel having dimensions as shown:

	Max.		Nominal		Min.	
	(inch)	(mm)	(inch)	(mm)	(inch)	(mm)
Overall length	17.9	454.7	17.8	452.1	17.7	449.6
Outside dia. at end caps	0.50	12.7	0.44	11.2	0.40	10.2
Outside dia. of body	0.40	10.2	0.38	9.7	0.37	9.4
Wall thickness of body	0.027	0.69	0.026	0.63	0.023	0.58

The end cap is attached to the main body using a fusion weld. Selection of the inner capsule/s varies according to user requirements in a configuration as shown above for Type Number 1 through 13. Source Models C-188 and C-306 have a consistent fit of minimum overall diameter and length dimensions between the inner and outer capsules to be within the range of a minimum diametrical clearance 0.001 inches (0.025 mm) and a minimum length clearance of 0.06 inches (1.5 mm) respectively.

DEVICE TYPE: Gamma Irradiator Source

DESCRIPTION (Cont'd):

This does not mean the diametrical gap will be 0.001 inches (0.025 mm) in all sources of Models C-188 and C-306. Previously distributed C-188 Types 1 to 12 and C-306 Types 1 and 2 have the same diametrical and longitudinal gap as they always had, as the designs of these sources have not changed.

The Model C-306 outer capsule is the same for all types. The inner capsule varies according to the user requirements. The Model C-306, Type 1 source contains one Model C-339 inner capsule; the Type 2 contains one C-177 inner capsule. Prior to December 21, 1998, the Model C-306, Type 1 source contained one Model AC-195 inner capsule, Type 2 contained one AC-191, and Type 3 contained one C-177 inner capsule **(Type 3 is no longer manufactured as of March 30, 2000)**.

The Model C-306 outer encapsulation is constructed of 316L stainless steel having dimensions as shown:

	Max.		Nominal		Min.	
	(inch)	(mm)	(inch)	(mm)	(inch)	(mm)
Overall length	9.6	243.9	9.5	241.3	9.4	238.8
Outside dia. at end caps	0.50	12.7	0.44	11.2	0.40	10.2
Outside dia. of body	0.40	10.2	0.38	9.7	0.37	9.4
Wall thickness of body	0.027	0.69	0.026	0.63	0.023	0.58

The end cap is attached to the main body using a fusion weld. Selection of the inner capsule varies according to user requirements; i.e., either Type 1 or Type 2 configuration. The fit of overall diameter and length dimensions between the inner and outer capsules is within the range of a minimum diametrical clearance 0.001 inches (0.025 mm) and a minimum length clearance of 0.06 inches (1.5 mm) respectively.

The inner capsules of source Models C-188 and C-306 have a maximum diameter of 0.32 inches (8.13 mm) and a minimum wall thickness of 0.015 inches (0.38 mm). The length of the inner capsules, the capsule material, and the radioactive source contents vary for each model as shown here in:

REGISTRY OF RADIOACTIVE SEALED SOURCES AND DEVICES
SAFETY EVALUATION OF SEALED SOURCE
(AMENDED IN ITS ENTIRETY)

DEVICE TYPE: Gamma Irradiator Source

DESCRIPTION (Cont'd):

Model No. of Inner	Radioactive Contents	Capsule Material	Max. length (inch)	(mm)
C-246	pellets or slugs	55316L	16.6	422
AC-339	pellets or slugs	Zircaloy 2/4	8.3	211
C-177	pellets or slugs	55316L	8.3	211
AC-191	pellets or slugs	55316L	8.3	211
AC-195	pellets or slugs	Zircaloy 2	8.3	211
AC-345	slugs	Zircaloy 2/4	11.3	287
C-348	slugs	55316L	5.3	134
C-188				
Type 13	slugs, pallets, or, wafers	Zircaloy 2~4 5S316L	16.6	422

The source material in the inner capsules is either 0.03-0.25 inches (0.76-6.35 mm) long and 0.03-0.25 inches (0.76-6.35 mm) diameter nickel plated cobalt pellets, or approximately 0.25 inches (6.35 mm) diameter and 0.03-0.5 inches (0.76-12.7mm) long nickel plated cobalt wafers, or approximately 0.25 inches (6.35mm) diameter and 0.5-3.0 inches (12.7-76.2 mm) long nickel plated slugs.

The majority of sources contain slug material as active contents. Occasionally, for low activity or sources requiring close tolerance dose outputs, material in pellet/wafer form is used as the active contents. The use of pellets/wafers makes it possible to mix pellets/wafers of various activities along with inactive pellets/wafers to accurately obtain required dose outputs. The pellets/wafers are of metallic form and nickel plated and, thus, indispersable in water.

LABELING:

Each model of the inner capsule assemblies is engraved on the end capsule with a serial number except the Model C-246 which is engraved on the body. All batches of inner capsules used in Model C-188 shall be traceable to the C-188 serial numbers. The Quality Assurance records for the inner capsules of all sources are maintained by MDS Nordion, Inc. The serial numbers for all sources shall be issued and controlled by MDS Nordion, Inc.

Labeling (Cont'd):

The Model C-188 and C-306 sources are engraved in the following manner:

- a unique serial number on the upper end cap face;
- either "C-188" or "C-306" and "Co 60" on the upper end cap diameter;
- the radiation trefoil and "MDSN X" (where MDSN designation of manufacturer, MDS Nordion, Inc., and X is the material heat number) on the lower end cap diameter. Sources manufactured under the earlier names of the company had been engraved correspondingly as "AECL" or "NII X."

DIAGRAM:

See Attachments 1 and 2.

CONDITIONS OF NORMAL USE:

The source Models C-188 and C-306 sources are designed primarily for use in wet source storage, pool type irradiators. Typical environments associated with the use of these irradiators include high temperatures, thermal shock due to sources being brought out of and into the water, and long term contact with water.

The sources may be used in dry source storage irradiators and environments for these devices would typically be less harsh. These uses would typically be medical facilities and laboratories fit for human occupancy. Therefore, the sources would be expected to be subjected to ambient temperatures and pressures. However, high activity sources may be exposed to elevated temperatures and temperature cycling due to internally generated heat.

PROTOTYPE TESTING:

The manufacturer conducted ANSI classification tests in order to classify the Model C-188 source. Category IV iradiators must have a minimum classification of E53424 to meet the requirements of 10 CFR 36.21 and ANSI N43.6. The Model C-188 source was successfully tested to E65646 in accordance with ANSI N542-1977, "Sealed Radioactive Sources, Classification." The tests were conducted by using dummy slug material rather than pellet material, because the slug configuration represented the more severe conditions. The manufacturer conducted an additional bend test, as specified in ANSI N43.10, "Safe design and Use of Panoramic Wet Source Storage gamma Irradiators." In the bend test, the Model C-188 source performed to Class 5.

The manufacturer stated that the Model C-306, Types 1 and 2, capsule met the standards of ANSI classification E54434 based on comparison with the Model C-188 capsule.

The manufacturer tested Model C-188 Type 13 for a worst-case scenario. Prototype models tested had a minimum 0.001 inches (0.025 mm) diametrical tolerance between the inner and the outer encapsulation. The inner length was represented by two equal solid rods of Type 316L stainless steel (stainless steel has a higher stiffness than Zircaloy). These solid bars exerted worst-case forces on the outer capsule during the ANSI testing. The solid bars were used to simulate the maximum resistance offered by the inner during testing. Tubing, without end caps, was used to simulate the least resistance by the inner during testing.

The outer source encapsulation retained its integrity over bounding stiffness range for the inner capsules. These worst-case prototypes were tested to E64424 classification and an additional class 5 bend test was done. The manufacturer test reports indicated that the outer encapsulation retained its integrity under these worst-case test conditions.

EXTERNAL RADIATION LEVELS:

A calculation of dose rates was done using the gamma radiation constant for cobalt-60 of 1.32 R/hr (13.2 mSv/hr) at one meter, per curie (39.4 in., per 37 GBq). A source containing maximum

DEVICE TYPE: Gamma Irradiator Source

EXTERNAL RADIATION LEVELS (Cont'd):

17,000 curies (629 TBq) would be expected to yield the following dose rates:

Distance from source	Radiation Level R/hr	Sv/hr
100cm/39.4 in	22,000	220
30cm/11.8 in	250,000	2,500
5cm/1.97 in	9,000,000	90,000

A source containing 14,000 curies (518 TBq) would be expected to yield the following dose rates:

Distance from source	Radiation Level R/hr	Sv/hr
100cm/39.4 in	18,500	185
30cm/11.8 in	205,000	2,050
5cm/1.97 in	7,400,000	74,000

A source containing 8,500 curies (314.5 TBq) would be expected to yield the following dose rates:

Distance from source	Radiation Level R/hr	Sv/hr
100cm/39.4 in	11,000	110
30cm/11.8 in	125,000	1,250
5cm/1.97 in	4,500,000	45,000

QUALITY ASSURANCE AND CONTROL:

MDS Nordion, Inc. (formerly AECL and Nordion International, Inc.) maintains a quality assurance and control program which has been deemed acceptable for licensing purposes by NRC. A copy of the program is on file with the NRC.

As a sole manufacturer and distributor of source Model C-188, Type 13, MDS Nordion, Inc. is committed to ensure that inners of Model C-188, Type 13 sources manufactured and supplied to MDS Nordion, Inc. by other manufacturers, who maintain a NRC license, shall meet the requirements outlined in this registration certificate.

QUALITY ASSURANCE AND CONTROL (Cont'd):

MDS Nordion, Inc. has committed to periodically conduct audits of subcontractors and source suppliers to ensure consistency and sustained production of quality products.. Subcontractor and suppliers shall be audited as needed in accordance with the provisions of ISO 9000 registered Quality Program. The results of such audits, follow-ups and corrective actions shall be recorded, retained, maintained and made available for inspections and audits.

LIMITATIONS AND/OR OTHER CONSIDERATIONS OF USE:

- The sources shall be distributed only to persons specifically licensed by the NRC or an Agreement State.

- Handling, storage, use, transfer, and disposal: To be determined by the licensing authority. In view that the sealed sources exhibit high surface dose rates when unshielded, they should be handled only by experienced licensed personnel using adequate remote handling equipment and procedures.

- These sources shall not be subjected to an environmental or other condition of use which would exceed an ANSI N542-1977 Classification of 77E54434.

- All C-188 Type 13 source will be tested to ANSI N542 classification 77E64424. It must also pass the additional Class 5 bend test prescribed in ISO 2919-1999(E). A Type 13 configuration shall not be used until these tests have been successfully completed and the simulated sources have been found leak tight.

- Any inner capsules acquired from other manufacturers are subject to MDS Nordion, Inc. approved quality requirements. MDS Nordion, Inc. is responsible for ensuring manufacturer's conformance to these requirements and requirements of other USA regulatory authorities.

- This registration sheet and the information contained within

DEVICE TYPE: Gamma Irradiator Source

LIMITATIONS AND/OR OTHER CONSIDERATIONS OF USE (Cont'd):

the references shall not be changed without the written consent of the NRC.

REVIEWER NOTE: Sources used in wet source storage irradiators shall be tested for contamination according to Section 36.59, 10 CFR Part 36.

REVIEWER NOTE: These sources may be used in dry source storage irradiators. Sources used in these devices shall be leak tested at intervals not to exceed six months using techniques capable of detecting 0.005 micro curie (185 Bq) of removable contamination.

SAFETY ANALYSIS SUMMARY:

Based on our review of the information and test data cited below, including the claimed ANSI classification, we continue to conclude that the Model C-188 and C-306 source designs are acceptable for licensing purposes.

Furthermore, we continue to conclude that the Model C-188 and C-306 sources would be expected to maintain their containment integrity for normal conditions of use and accidental conditions which might occur during the uses specified in this certificate.

REFERENCES:

The following supporting documents for the Models C-188 and C-306 sources are hereby incorporated by reference and are made a part of this registry document.

- Atomic Energy of Canada, Ltd. letters dated October 29, 1973, June 14, 1974, September 20, 1984, and July 18, 1985 with enclosures thereto.
- Nordion International, Inc. letters dated June 25, 1993, March 16, 1992, December 5, 1991, and October 3, 1988, with enclosures thereto, and letter received November 18, 1991, with enclosures thereto.

REFERENCES (Cont'd):

- MDS Nordion, Inc. letters dated January 13, 1998, February 16, 1998, April 3, 1998, December 21, 1999, and March 6, 2000, with enclosures thereto.
- **MDS Nordion letter dated December 14, 2001, requesting name and address change.**

ISSUING AGENCY:

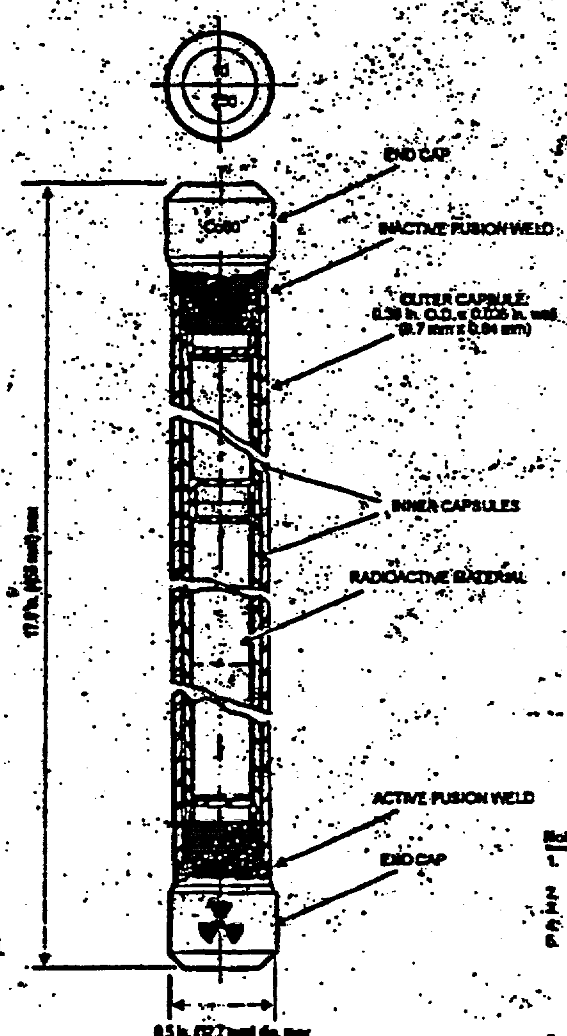
U.S. Nuclear Regulatory Commission

Date: 01/23/02 Reviewer: Michele L. Burgess
Michele L. Burgess

Date: 01/23/02 Concurrence: John P Jankovich
John P. Jankovich

REGISTRY OF RADIOACTIVE SEALED SOURCES AND DEVICES
SAFETY EVALUATION OF SEALED SOURCE
(AMENDED IN ITS ENTIRETY)

NO: NR-0220-S-103-S　　**DATE:** 01/23/02　　**ATTACHMENT 1**



Model

C-188 Type Number	Model Number of Inners
1.	C-177/C-177
2.	AC-191/AC-191
3.	AC-195/AC195
4.	C-346
5.	AC-339/AC-339
6.	AC-346/AC-346
7.	C-177/AC-191
8.	C-177/AC-195
9.	C-177/AC-339
10.	AC-191/AC-195
11.	AC-191/AC-339
12.	AC-195/AC-339
13.	See Note 6

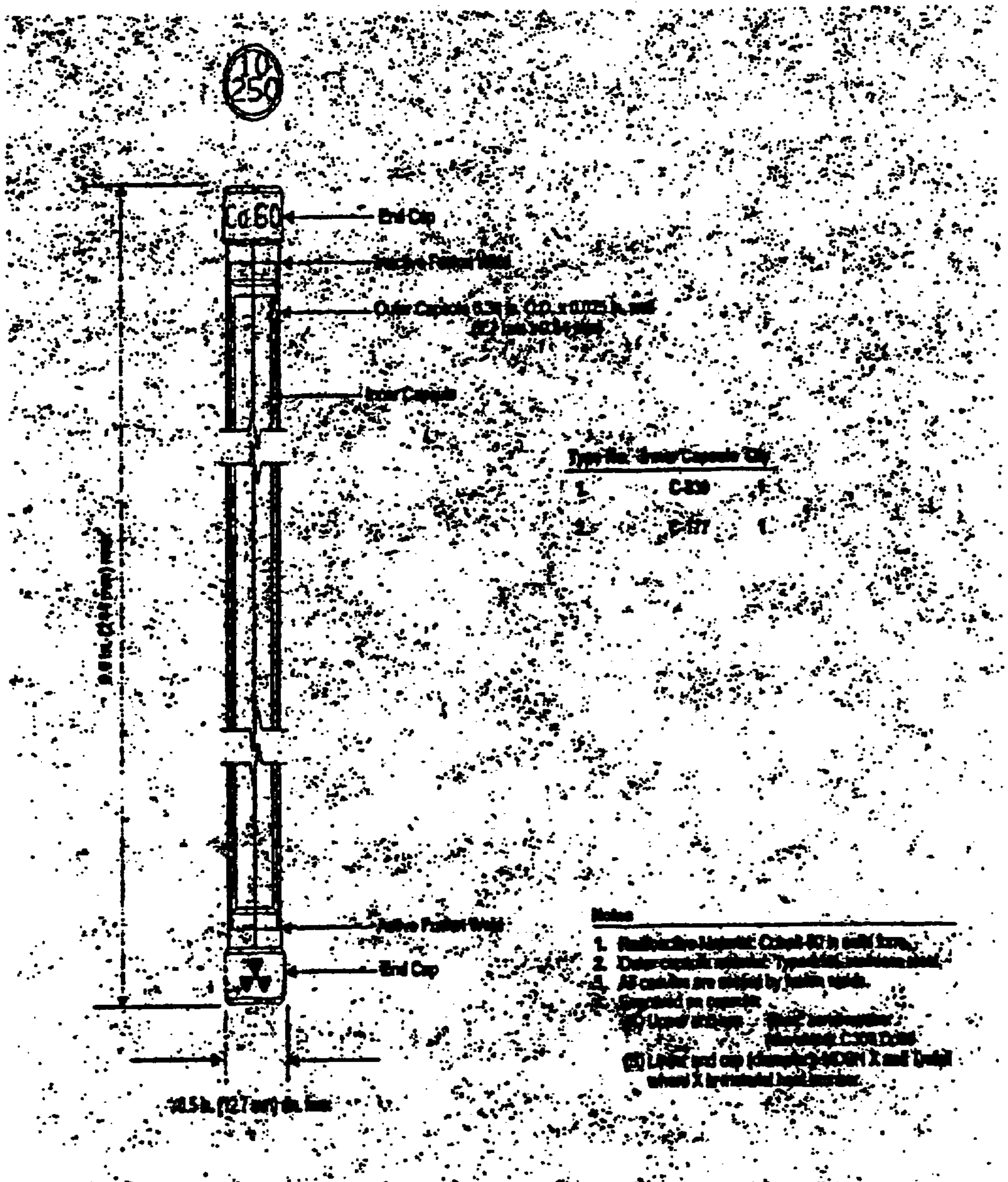
Notes

1. Conforms to IAEA Special Form requirements. AECB Certificate No. CDN/0010/S-85.
2. Radioactive Material: Cobalt-60 in solid form.
3. Outer capsule material: Type 316L stainless steel.
4. All capsules are sealed by fusion welds.
5. Engraved on capsule:
   (A) Upper end cap
   face: serial number
   diameter: C188 Co60
   (B) Lower end cap diameter: MDSN X and Trefoil where X is material heat number.
6. Any inner design constructed from stainless steel or zircaloy consisting of one or more capsules containing Cobalt-60 pellets, slugs or wafers and of a design similar, but not identical to one or more of those contained in types 1 to 12.

C-188 Sealed Source Assembly

REGISTRY OF RADIOACTIVE SEALED SOURCES AND DEVICES
SAFETY EVALUATION OF SEALED SOURCE
(AMENDED IN ITS ENTIRETY)

NO: NR-0220-S-103-S   DATE: 01/23/02   ATTACHMENT 2



C-306 Sealed Source Assembly

APPENDIX 4.4.3
STRESS ANALYSIS OF THE CONTAINMENT SYSTEM SUBJECT TO NORMAL CONDITIONS OF TRANSPORT OF F-294

CONTENTS

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1. INTRODUCTION

The inner shell assembly and the lid (closure plug) is defined as the containment system of F-294 package. Figure 4.4.3-F1 depicts the F-294 containment system.

During the normal conditions of transport (NCOT), the F-294 package with either the F-313 or F-457 source carrier is subjected to following:

1. Internal pressure in the cavity of 16 psig. the cavity walls are at 401°F. The closure plug is at 446°F (at the cavity end) and 288°F at the external surface.
2. In the 3 ft. drop, the F-294 is subjected to 130 g's in the top drop orientation.

The purpose of the closure plug bolted joint is to provide adequate shielding, but not necessary leaktightness, of the joint between the closure plug and the inner shell assembly. In this Appendix, it is demonstrated that the closure plug is retained when F-294 is subject to combination of forces during normal conditions of transport. Consequently the closure plug, which provides adequate shielding, shall be retained over the cavity of F-294, which houses Special Form cobalt-60 MDS Nordion C-188 sealed sources. As C-188 is certified Special Form RAM, the C-188 outer encapsulation provides the leaktightness for retention of radioactive material.

Therefore,

1. as C-188 is certified Special Form RAM and provides leak tight containment AND
2. as the closure plug is retained over the inner shell assembly, which houses the cobalt-60 C-188 sealed sources,

F-294 shall meet the NCOT containment system requirements (10 CFR 71.51 (a) (1)).

2. STRESS ANALYSIS OF CLOSURE PLUG BOLTED JOINT

2.1 CLOSURE PLUG BOLTED JOINT SUBJECT TO INTERNAL PRESSURE AND G-LOADS

Due to build-up of internal pressure, the main plug bolted closure is examined in detail. The internal pressure in the cavity is 17 psig (see Figure 4.4.3-F2).

2.1.1 Internal Pressure Load, W_{OP}

The internal pressure load, W_{OP} is calculated as follows:

$$\begin{aligned} W_{OP} &= \Delta P * \text{Area} \\ &= \Delta P * [\pi * G^2/4] \\ &= 16. * \pi * 15.91^2/4 \text{ [psi * in}^2\text{]} \\ &= 3{,}200 \text{ lb.} \end{aligned}$$

where

ΔP = 16 psi (internal pressure - outside the F-294 container at atmospheric pressure)

G = gasket reaction diameter = 15.91 in.

2.1.2 Gasket Seating Load, F_{SG}

$$F_{SG} = \pi * b * G * y$$

where

b = effective gasket seating width

G = gasket diameter

y = gasket seating stress = 200 psi
(Ref.[17] i.e., ASME VIII Div. I: Table UA-49-1)

Basic gasket seating width, b_o = actual width of gasket/2

= (16.38 – 15.44) x 0.5/2

= 0.235 in.

When $b_o \leq 1/4$ in., the effective gasket seating width, $b = b_o = 0.235$ in.

When $b_o \leq 1/4$ in., diameter at location of gasket reaction, G

G = Mean diameter of gasket contact face

= (16.38 + 15.44) x 0.5

= 15.91 in.

Gasket seating Load, F_{SG}

$F_{SG} = \pi * b * G * y$

$= \pi * 0.235 * 15.91 * 200$

= 2,400 lb.

Therefore, gasket seating and internal pressure load acting on the plug,

$W_{\text{bolt load required}}$ = FSG + WOP

= 2,400 + 3,200

= 5,600

Design check: what is the total bolt load available on basis of 2/3 * yield stress of the bolt material?

16 cap screws (1-8-UNC: UNBRAKO 1960) 1-in. are specified as closure plug bolts.

For UNBRAKO 1960 cap screw material, UTS = 180,000 psi; YS = 155,000 psi.

Bolt data - 1-in. nominal diameter

stress area per bolt = 0.551 in^2

root diameter = 0.838 in

UNC = coarse thread

8 threads per inch (8 tpi).

$W_{\text{bolt load available}}$ = no. of bolts x bolt area x allowable stress

= 16 x 0.551 x [2/3 x YS]

= 16 x 0.551 x 0.667 x 155,000

= 911,000 lb.

As $W_{\text{bolt load available}}$ (911,000 lb.) > $W_{\text{bolt load required}}$ (5,600 lb.), the closure plug bolting design is more than adequate to resist the forces on the closure plug due to internal pressure.

Now let us consider additional forces on the closure plug due to 130 g's on F-294 resulting from 3 ft. drop of F-294, in normal top end drop test.

$W_{G\text{-}LOAD}$ = Load due to G-load = W x G_{NCOT} = 1,135 lb. x 130 g's

= 147,600 lb.

where

$W = W_{PLUG} + W_{CONTENT}$ = 1,070 + 65 = 1,135 lb.

The total load on the closure plug required to maintain flanged, gasketed joint in NCOT is

$W_{\text{required closure plug,NCOT}}$ = $F_{SG} + W_{OP} + W_{G\text{-}LOAD}$

= 2,400 + 3,200 + 147,600 lb.

= 153,200

As $W_{\text{bolt load available}}$ (911,000 lb.) >$W_{\text{required closure plug,NCOT}}$ (153,200 lb.), the closure plug bolting design is more than adequate to resist the forces on the closure plug due to internal pressure and G-loads on F-294 arising from normal drop tests.

Safety factor (SF) = $W_{\text{bolt load available}}/W_{\text{required closure plug, NCOT}}$

= 911,000 lb./153,200

= 5.95

Margin of Safety = SF – 1 = 5.95 – 1 = 4.95
As the margin of safety (MF) is greater than zero, the bolted joint as specified shall be maintained during NCOT of F-294 package.

2.2 BOLT STRESS DUE TO APPLIED TORQUE.

100 ft.-lb. per bolt nominal torque is specified in the preparation for shipment procedure (see Chapter 7, Section 7.1) for the closure plug bolts of the F-294 container.

UNBRAKO catalogue specifies for bolt (screw) material: 155,000 yield stress at room temperature.

The direct bolt stress, s, is

$$\begin{aligned} s &= F/A_r \\ &= [T/(0.2 * D_{nom})]/A_r \\ &= 100 \times 12/(0.2 * 1.0)/0.551 \\ &= 10{,}890 \text{ psi} \end{aligned}$$

The shear stress t due to torque, T, is

$$\begin{aligned} t &= TD_r/J \\ &= [TD_r/2]/[\pi D_r^4/32] \\ &= 16T/[\pi * D_r^3] \\ &= 16 * 100 * 12/[\pi * 0.838^3] \\ &= 10{,}390 \text{ psi} \end{aligned}$$

The principal stress is:

$$\begin{aligned} s_{P1} &= (s/2) + \sqrt{[(s/2^2) + t^2]} \\ &= 10{,}890/2 + \sqrt{[(10{,}890/2)^2 + 10{,}390^2]} \\ &= 5{,}445 + 15{,}050 \\ &= 20{,}500 \text{ psi} \end{aligned}$$

The principal stress s_{P1} of 20,500 psi is less than 100,000 psi allowable stress for the bolt material (2/3 of yield stress of 155,000 psi).

$$\begin{aligned} \text{Safety Factor (SF)} &= \text{allowable stress/applied stress} \\ &= 100{,}000/20{,}500 \\ &= 4.87 \end{aligned}$$

Margin of Safety = SF –1 = 4.87 –1 = 3.87
where

F	= Applied force per bolt, lb.
	= T/CD
T	= applied bolt torque ft.-lb. or in.-lb.
D	= Nominal bolt diameter, 1 in.
C	= constant = 0.2 typical.
A_r	= area at minor dia. of thread = 0.551 in^2
D_r	= root diameter of external thread = 0.838 in.
J	= Polar moment of inertia for round shaft (in^4)

2.3 SHEAR OF THREADS IN THE BOLT HOLES

This analysis is intended to demonstrate that the threads in the flange bolt holes are strong enough to resist operating pressure, gasket seating and G loads. See Figure 4.4.3-F3 for force free body diagram.

Bolt hole - Internal thread specification:

- 1-8-UNC-2B
- 1.0 in. thread length of bolt engagement (i.e., bolt hole 1.25 in. deep)
- 8 threads per inch (tpi)
- 0.9188 basic pitch diameter
- 0.125 pitch (p)

Socket Head screw specification:

- 1-8-UNC, 2.0 in. long.
- Effective thread length = 1.0 in.

Shear area per bolt hole, A_S

$A_S = \pi * D_{min} * h$

where

D_{min} = basic pitch diameter

h = effective length of threads in shear

= 0.5 * number of threads engaged by screw * pitch

= 0.5 * 8 tpi * 1.0 * 0.125

= 0.5 in.

$A_S = \pi * 0.9188 * 0.5$

$= 1.44\ in^2$

$W_{required\ closure\ plug,\ NCOT} = F_{SG} + W_{OP} + W_{G\text{-}LOAD}$

= 2,400 + 3,200 + 147,000 lb.

= 152,600

τ, shear stress in the internal threads per bolt hole

$\tau = W_{required\ closure\ plug,NCOT}/[N*A_S]$

= 152,600/[16 * 1.44]

= 6,630 psi

For ss304L, minimum YS = 25,000 psi. at 100°F;

= 17,800 psi at 330°F

Allowable stress = 2/3 yield stress.

Safety factor (SF) = allowable stress/applied stress

= 11,860/6,630

= 1.78

Margin of Safety = SF – 1 = 1.78 – 1 = 0.78

As the shear stress of 6,630 psi is less than the allowable stress of 11,860 psi (i.e., 2/3 * yield stress of 17,800 psi), the bolt hole thread strength meets the requirements that the closure plug shall remain fastened to the inner shell assembly of F-294.

2.4 CLOSURE PLUG BOLTING IN F-294 SIDE DROP ATTITUDE

See Figure 4.4.3-F4.

In the side drop, the plug is held to the container by 16 fasteners of 1 in. dia., UNBRAKO (i.e., 180,000 UTS). The clearance between the head of the fastener and the unthreaded hole in the plug flange is 0.0625 in. The clearance between the plug and the container upper cavity is

0.040 in. Therefore in the side impact, the cylindrical side of the plug body will impact prior to the start of shearing of plug/container bolts (bolt head from bolt shank). Therefore, in this calculation it is not assumed that due to load sharing,

1. the weight of the ss flange of the plug assembly is resisted by 16 bolts AND
2. the weight of the cylindrical section of the plug (i.e., lead shielding, etc.) is resisted by the 1/3rd cylindrical arc of the plug and the container upper cavity.

See Figure 4.4.3-F5: Shear Force Diagram.

2.4.1 Plug Bolts

The weight of the plug is W_{PLUG} = 1070 lb. (Note: exclusive of the weight of contents 65 lb.)
The deceleration load is 130 g's. Therefore the impact load on the bolts is

$P_{IMPACT} = W_{PLUG} \times \text{G-load}$
$= 1,070 \times 130$
$= 139,100$ lb.

Shear area per bolt, $A_{SHEAR\ AREA}$

$A_{SHEAR\ AREA} = 0.551\ in^2$

Shear stress per bolt, t

$\tau = P_{IMPACT}/[A_{SHEAR\ AREA} \times \text{number of bolts}]$
$= 139,100/[0.551 \times 16]$
$= 139,100/8.816$
$= 15,780$ psi.

In addition, each bolt is subjected to gasket seating load and operating load due to internal pressure of 16 psig. the tensile stress is s

$s = F_{SG} + W_{OP}/\text{Total bolt area}$
$= 5,600/0.551 \times 16$
$= 635$ psi

The principal stress is:

$s_{P1} = (s/2) + \sqrt{[(s/2^2) + t^2]}$
$= 635/2 + \sqrt{[(635/2)^2 + 15,780^2]}$
$= 318 + 15,780$
$= 16,100$ psi

The principal stress s_{P1} of 16,100 psi is less than 100,000 psi allowable stress for the bolt material (2/3 of yield stress of 155,000 psi).

Safety Factor (SF) = allowable stress/applied stress
= 100,000/16,100
= 6.21

Margin of Safety = SF – 1 = 6.21 – 1 = 5.21

2.4.2 Plug Cylindrical Body/Container Upper Cavity Tube

One-third (1/3) of the cylindrical plug body resists the side impact.

W_{PLUG} = 1,070 lb.

The impact load, at deceleration of 130 g's is

$P_{IMPACT} = W_{PLUG} X$ 130 g's
= 1070 x 130
= 139,100 lb.

Area resisting the impact A = 0.33 x π x 14.790 x 10
= 154.7 in^2

Bearing stress, $s_{bearing}$

$s_{bearing}$ = P_{IMPACT}/A
= 139,100/154.7
= 900 psi.

Safety Factor (SF)

SF = allowable stress/applied compressive stress
= 2/3 x YS_{SS304L} at 330°F/$s_{bearing}$
= 2/3 x 17,800/900
= 11,860/900
= 13.1

Margin of Safety, $MS_{STRESS\text{-}BASED}$ = SF – 1 = 13.1 – 1 = 12.1

2.4.3 Plug Weld: WPC1

There are two (2) plug circumferential welds in the plug (WPC1 and WPC2)..However, only one plug weld (WPC1) is conservatively assumed to resist the side impact (see Figure 4.4.3-F6).

Weld area = π x 12.710 x 0.38 x 0.707 = 10.72 in^2

As the weld is fully radiographed, therefore joint efficiency is 100%.

Therefore the impact load is

P_{IMPACT} = W_{PLUG} x Deceleration load
= 1,070 x 130 g's
= 139,100 lb.

Shear stress, t = P_{IMPACT}/Weld area
= 139,100/10.72
= 12,970 psi

Safety Factor (SF)

SF = allowable stress/applied shear stress
= 2/3 x YS_{SS304L} at 330°F/t
= 2/3 x 17,800/t
= 11,860/ 12,970
= 0.91

Margin of Safety, $MS_{STRESS\text{-}BASED}$ = SF – 1 = 0.91 – 1 = -0.09

3. STRESS ANALYSIS OF INNER SHELL ASSEMBLY COMPONENTS DUE TO BUILD-UP OF INTERNAL PRESSURE AND G-LOADS

3.1 INNER SHELL ASSEMBLY COMPONENTS - INTERNAL PRESSURE

3.1.1 Lower Cavity Wall

See Figure 4.4.3-F7
The hoop stress in the lower cavity tube, without taking lead restraint into account, is as follows:

s_{hoop} = p d/2t
= 16 x 12/2 x 0.5 = 192 psi.

where

p = 16 psig internal pressure
d = mean diameter of lower cavity tube = 12.0 in.
t = 0.500 in.

For ss304L at 400°F, yield stress = 16,300 psi.

Safety Factor (SF) = allowable stress/applied stress
= 0.667 x YS/s_{hoop}
= 0.667 x 16,300/192
= 56.6

Margin of Safety (MS) = SF – 1 = 56.6 – 1 = 55.6

3.1.2 Upper Cavity Wall

See Figure 4.4.3-F8.
The hoop stress in the upper cavity tube, without taking lead restraint into account, is as follows:

s_{hoop} = p d/2t
= 16 x 15.2/2 x 0.5 = 243 psi.

where

p = 16 psig internal pressure
d = mean diameter of upper cavity tube = 15.2. in.
t = 0.500 in.

For ss304L at 400°F, yield stress = 16,300 psi.

Safety Factor (SF) = allowable stress/applied stress
= 0.667 x YS / s_{hoop}
= 0.667 x 16,300/243
= 44.7

Margin of Safety (MS) = SF – 1 = 44.7 – 1 = 43.7

3.1.3 Lower Cavity End Cap

See Figure 4.4.3-F9
The bending stress in the lower cavity end cap, without taking restraint of lead into account, is as follows:

s_b = $cp/[t/d]^2$
= $0.2 \times 16/[0.75/11.5]^2$ = 753 psi.

where

c	= constant based on joint geometry
	= 0.2 based on ASME VIII, Division 1, Figure UG = 34 (i)
p	= internal pressure = 16 psig
t	= thickness of end cap = 0.5 in.
d	= internal diameter of the tube = 11.5 in.

For Hastelloy C-276, YS, yield stress at 400 °F = 30,000 psi
(ASTM B-166)

Safety Factor (SF) = allowable stress/applied stress
= [2/3 x YS]/s_b
= [2/3 x 30,000]/753
= 26.5

Margin of Safety (MS) = SF – 1 = 26.5 – 1 = 25.5

Therefore, the inner assembly under build-up of pressure of 16 psig has sufficient margin of safety that the structural integrity of the inner assembly shall not be compromised.

3.1.4 Upper Cavity Ring Flange

See Figure 4.4.3-F10.
The bending stress in the upper cavity ring flange, without taking restraint of lead into account, is as follows:
Case 77, Table X of Ref[4]

s_r = $b\,w\,a^2/t^2$
= $0.0195 \times 16 * 7.892^2/0.5^2$ psi.
= 78 psi

where

s_r	= maximum radial stress
b	= constant depending upon a/b = 7.892/6.25 = 1.262,
	= 0.0195
w	= p = applied pressure = 16 psi
t	= thickness of ring flange = 0.5 in.
a	= 7.892 in. ring flange outside radius
b	= 6.25 in. ring flange inside radius

For ss304L forging to ASTM A-182, the material properties are:

Yield Stress (YS) = 16,300 psi. at 400°F

Safety Factor (SF) = allowable stress/applied stress
= 2/3 * YS/ 83
= 2/3 x 16,300 psi/78 psi
= 139

Margin of Safety (MS) = SF – 1 = 139 – 1 = 138

Therefore, the ring flange is well below the static value of YS.

3.2 INNER SHELL ASSEMBLY SUBJECT TO G-LOADS

3.2.1 Buckling of lower cavity tube

See Figure 4.4.3-F11
The weight of lead borne by the steel tube + cap,

$W_1 = \pi \times 6.25^2 \times 11.25 \times 0.41$

$W_1 = 566$ lb.

Assume the load is applied at the centre of the tube. Then, using Euler's formula, the collapse pressure load is

$P_C = \pi^2 EI/(L_e)^2$

where

$E = 28 \times 10^6$ psi
$L_e = 19.5$ in.
I = 2nd Moment of area = 340 in^4
$P_C = 247 \times 10^6$ lb.
Applied load = 566 lb. x g_{NORMAL}
= 566 X 130
= 73,580 lb.

Safety Factor (SF) = Collapse load to buckle cavity tube/Applied load
$= 247 \times 10^6/[73,580$ lb.]
$= 247 \times 10^6/73,580$
= 3,356

Margin of Safety (MS) = SF – 1 = 3,356 – 1 = 3,355
It is concluded that the lower cavity tube will not buckle.

3.2.2 Bending of Lower Cavity End Plate

See Figure 4.4.3-F12
G3 = 130 g's: G4 = 280 g's
The applied pressure on the cavity tube cap is

p = weight of lead x 130 g's/A_{CAP}
$= 566 \times 130/[\pi \times 6.25^2]$
= 73,500/122.73
= 600 psi

Is a 0.75 in. thick Hastelloy C-276 cap strong enough to resist 600 psi maximum applied pressure?
For Hastelloy C-276, the material properties are:

Yield Stress (YS) = 30,000 psi.
Ultimate Tensile Strength (UTS) = 100,000 psi.

Using ASME VIII, Division 1 (Ref[17])

$s = cp/[t/d]^2$
$= 0.2 \times 600/[0.75/11.5]^2$
= 28,213 psi.

where

c = constant depending on end tube to cap joint configuration
= 0.2 (ASME VIII, Division 1, Figure UG-34(i).

p = applied pressure = 600 psi

t = thickness of cap = 0.75 in.

d = 11.5 in. inside diameter

Safety Factor (SF) = allowable stress/applied stress
= [2/3 x YS]/12,193
= 2/3 x 30,000 psi/28,213 psi
= 20,000/28,213
= 0.71

Margin of Safety (MS) = SF – 1 = 0.71 – 1 = – 0.29

The maximum stress in the end cap s = 12,193 psi shall be below the allowable stress of 20,000 psi [i.e., 2/3 x YS]. Therefore, the end cap of the lower cavity of the inner shell assembly of F-294 shall not yield

3.2.3 Buckling of Upper Cavity Tube

See Figure 4.4.3.-F13. The weight of lead borne by the steel tube + flange,

W2 $= \pi \times [7.892^2 - 6.25^2] \times 31.75 \times 0.41$

W2 = 950 lb.

W1 + W2 = 566 + 950 = 1,516 lb.

The loads (W1 + W2) are applied at the centre of the tube. The line of action of deceleration force is in line with the centre of gravity of the upper tube. Therefore this is the case of compressive stress leading to buckling. Then the collapse load, using Euler's formula is

$P_C = \pi^2 EI/(L_e)^2$

where

E $= 28 \times 10^6$ psi

L_e = 11 in.

I = 2nd Moment of area $= \pi/4\,[7.89^4 - 7.39^4] = 701$ in^4

$P_C = 1{,}601 \times 10^6$ lb.

NCOT G-levels:

top of plug G1 = 116 g's

bottom of plug G2 = 113 g's

mean g's at mid height = 115 g's.

Applied load = 1,516 lb. x G-load
= 1,516 x 115
= 174,340 lb.

Safety Factor (SF) = collapse load to buckle cavity tube/applied load
$= 1{,}601 \times 10^6/[174{,}340$ lb.]
= 9,183

Margin of Safety (MS) = SF – 1 = 9,183 – 1 = 9,182

It is concluded that the upper cavity tube shall not buckle.

3.2.4 Bending of Upper Cavity Ring Flange

See Figure 4.4.3-F14.
In this model, the cavity ring flange plate is under external applied pressure as a result of impact.
The applied pressure acts on the upper cavity tube & the cavity ring flange.
Based on 116 g's deceleration load, the applied pressure on the upper cavity ring flange is

$$p = \text{weight of lead x 125 g's}/A_{RING\ FLANGE}$$
$$= 950 \times 116/[\pi \times (7.892^2 - 6.25^2)]$$
$$= 110{,}200/73$$
$$= 1{,}509 \text{ psi}$$

Is a 0.5 in. thick stainless steel forging (ring flange) strong enough to resist 1,509 psi maximum applied pressure?
For stainless steel (ss304L)A-182, the material properties are:
Yield Stress (YS) = 16,300 psi at 400°F.
Case 77, Table X of Ref[4]

$$s_r = b\,w\,a^2/t^2$$
$$s_r = b\,p * [a^2/t^2]\text{ psi.}$$
$$s_r = 0.0195 * 1{,}509 * 7.892^2/0.5^2\text{ psi.}$$
$$= 7{,}330\text{ psi}$$

where

- s_r = maximum radial stress
- b = constant depending upon a/b = 7.892/6.25 = 1.262, b = 0.0195
- w = p = applied pressure = 1,509 psi
- t = thickness of ring flange = 0.5 in.
- a = 7.892 in. ring flange outside radius
- b = 6.25 in. ring flange inside radius

Safety Factor (SF) = allowable stress/applied stress
$$= 2/3 * YS/7{,}330$$
$$= 2/3 * 16{,}300\text{ psi}/7{,}330\text{ psi}$$
$$= 1.48$$

Margin of Safety (MS) = SF – 1 = 1.48 – 1 = 0.48
The maximum stress in the upper cavity ring flange s_r = 7,330 psi is below 10,800 psi (the allowable stress of 2/3 x 16,300 psi [yield stress at 400°F]).

3.2.5 Container Top Flange

See Figure 4.4.3-F15
In this model, the container top flange plate is under external applied pressure as a result of impact.
The applied pressure acts on the container top flange & the conical shell

Based on 116 g's deceleration load, the applied pressure on the container top flange is

$$p = \text{weight of lead x 116 g's}/A_{RING\ FLANGE}$$
$$= (W_1 + W_2 + W_3) \times 116/[\pi \times (12.968^2 - 7.892^2)]$$
$$= (566 + 950 + 5{,}314) * 116/356.7$$
$$= 6830 \times 116/356.7$$
$$= 2{,}220\text{ psi}$$

Is a 1.5 in. thick stainless steel plate (ring flange) strong enough to resist 2,220 psi maximum applied pressure?
For stainless steel (ss304L)A-240, the material properties are:
Yield Stress (YS) = 17,800 psi at 330°F
Case 77, Table X, Roark (4th Edition)

$$s_r = b\,w\,a^2/t^2$$
$$= b\,w\,a^2/t^2$$
$$= 0.03 \times 2{,}220 \times 10.5^2/1.5^2$$
$$= 3{,}260 \text{ psi}$$

where

s_r	= maximum radial stress
a	= 10.5 in. ring flange outside radius (see Figure 4.4.3-F16)
b	= 7.892 in. ring flange inside radius
b	= constant depending upon a/b = 10.5/7.892 = 1.33, b = 0.03
w	= p = applied pressure = 2872 psi
t	= thickness of ring flange = 1.5 in.

Safety Factor (SF) = allowable stress/applied stress
= 2/3 x static YS/3,260
= 2/3 * 17,800 psi/3,260 psi
= 11,860/3,260
= 3.63

Margin of Safety (MS) = SF – 1 = 3.62 – 1 = 2.63

The maximum stress in the ring flange, s_r = 3,260 psi is below allowable of 2/3 * YS of 17,800 psi at 330°F. Therefore the ring flange shall not yield.

4. SUMMARY

The following is a summary of the status of each of the components of the inner shell assembly and the closure plug under appropriate driving forces and resulting in appropriate safety factors.

4.1 CLOSURE PLUG BOLTED JOINT

For the closure plug bolted joint components, the applied stresses, safety factors and margins of safety are as follows:

A. Closure plug bolted joint subject to internal pressure and G-loads:

$W_{\text{required closure plug,NCOT}}$ = 153,200 lb.
$W_{\text{bolt load available}}$ = 911,000 lb.
Safety factor (SF) = 5.95

As the Safety Factor (SF) is greater than 1, the bolted joint as specified shall be maintained during NCOT of F-294 package.

B. Bolt Stress due to applied torque of 100 ft.-lb. per bolt

Applied stress = 20,500 psi
Allowable stress = 100,000 psi
Safety factor (SF) = 4.87

Margin of Safety (MS) = SF – 1 = 4.87 – 1 = 3.87

C. Stress in the threads of the bolt hole

Applied stress = 6,320 psi
Allowable stress = 11,860 psi
Safety factor (SF) = 1.87

Margin of Safety (MS) = SF – 1 = 1.87 – 1 = 0.87

D. Closure plug: in side drop

1) Closure plug bolts:

Applied stress = 16,100 psi
Allowable stress = 100,000 psi
Safety Factor (SF) = 6.21

Margin of Safety (MS) = SF – 1 = 6.21 – 1 = 5.21

2) Plug cylindrical body: side impact:

Applied compressive stress = 900 psi
Allowable stress = 11,860 psi
Safety Factor, (SF) = 13.1

Margin of Safety (MS) = SF – 1 = 13.1 – 1 = 12.1

3 Plug weld WPC1:

Applied shear stress = 12,970 psi
Allowable stress =11,860 psi
Safety Factor (SF) = 0.91

Margin of Safety (MS) = SF – 1 = 0.91 – 1 = -0.09

4.2 INNER SHELL ASSEMBLY

For inner shell assembly and components, the stresses or collapse loads, safety factors and margins of safety are given below. The safety factor (SF) is defined as: allowable stress/applied stress. For buckling, the safety factor is defined as critical load or collapse load/applied load. The allowable stress is 2/3 of yield stress of material at appropriate temperature.

4.2.1 Summary of Stress Analysis of Inner Shell Assembly Components Under Internal Pressure

Various components identified in Figure 4.F2 were subjected to internal pressure of 16 psig and stress analyzed. The details of the calculations are given in Appendix 4-4-3. Summary is given here.

A. Lower Cavity wall due to build-up of internal pressure:

S_{hoop} = 192 psi; SF = 56.6; MS = 55.6

B. Upper Cavity wall due to build-up of internal pressure:

S_{hoop} = 243 psi; SF = 44.7; MS = 43.7

C. The bending stress in the lower cavity end cap, without taking restraint of lead into account, is

S_b = 1,693 psi; SF = 11.8; MS = 10.8

D. The bending stress in the upper cavity ring flange, without taking restraint of lead into account, is

s_r = 78 psi; SF = 139; MS = 138

4.2.2 Summary of Stress Analysis of Inner Shell Assembly Components Under NCOT G-Loads

Various components identified in Figure 4.F2 were subjected to G-loads of 116 – 130 g's depending on the location from the top closure and stress analyzed. The details of the calculations are given in Appendix 4-4-3. Summary is given here.

A. Lower cavity tube buckling:

Applied load = 73,580 lb.
Collapse load = 247 x 10^6 lb.

SF = 3,356; MS = 3,355

B. Lower cavity tube end cap:

Bending stress = 63,480 psi.
Allowable stress = 20,000 psi.

SF = 0.31; MS = – 0.7

C. Upper cavity tube buckling:

Applied load = 174,340 lb.
Collapse load = 1,601 x 10^6 lb.

SF = 9,183; MS = 9,182

D. Bending of upper cavity ring flange

Bending stress = 7,330 psi.
Allowable stress = 11,670 psi.

SF = 1.48; MS = 0.48

E. Container top ring flange

Bending stress = 3,260 psi
Allowable stress = 11,860 psi.

SF = 3.63; MS = 2.63

As the safety factors (SF) > 1 and as the margin for safety (MS) > 0, the bolted closure joint over the inner shell assembly of F-294 shall be maintained to resist internal pressure and G-loads encountered in the F-294 containment system during the normal conditions of transport of F-294.

The purpose of the closure plug bolted joint is to provide adequate shielding but not necessary leaktightness of the joint between the closure plug and the inner shell assembly. It has been demonstrated that the closure plug is retained when F-294 is subject to combination of forces during normal conditions of transport. Consequently the closure plug, which provides adequate shielding, shall be retained over the cavity of F-294, which houses Special Form cobalt-60 MDS Nordion C-188 sealed sources. As C-188 is certified Special Form RAM, the C-188 outer encapsulation provides the leaktightness for retention of radioactive material.

Therefore

1. as C-188 is certified Special Form RAM and provides leak tight containment AND
2. as the closure plug (the shielding) is retained over the inner shell assembly, which houses the cobalt-60 C-188 sealed sources,

F-294 shall meet the NCOT containment system requirements (10 CFR 71.51 (a) (1)).

Figure 4.4.3-F1
F-294 Containment System: Inner Shell Assembly and Closure Plug

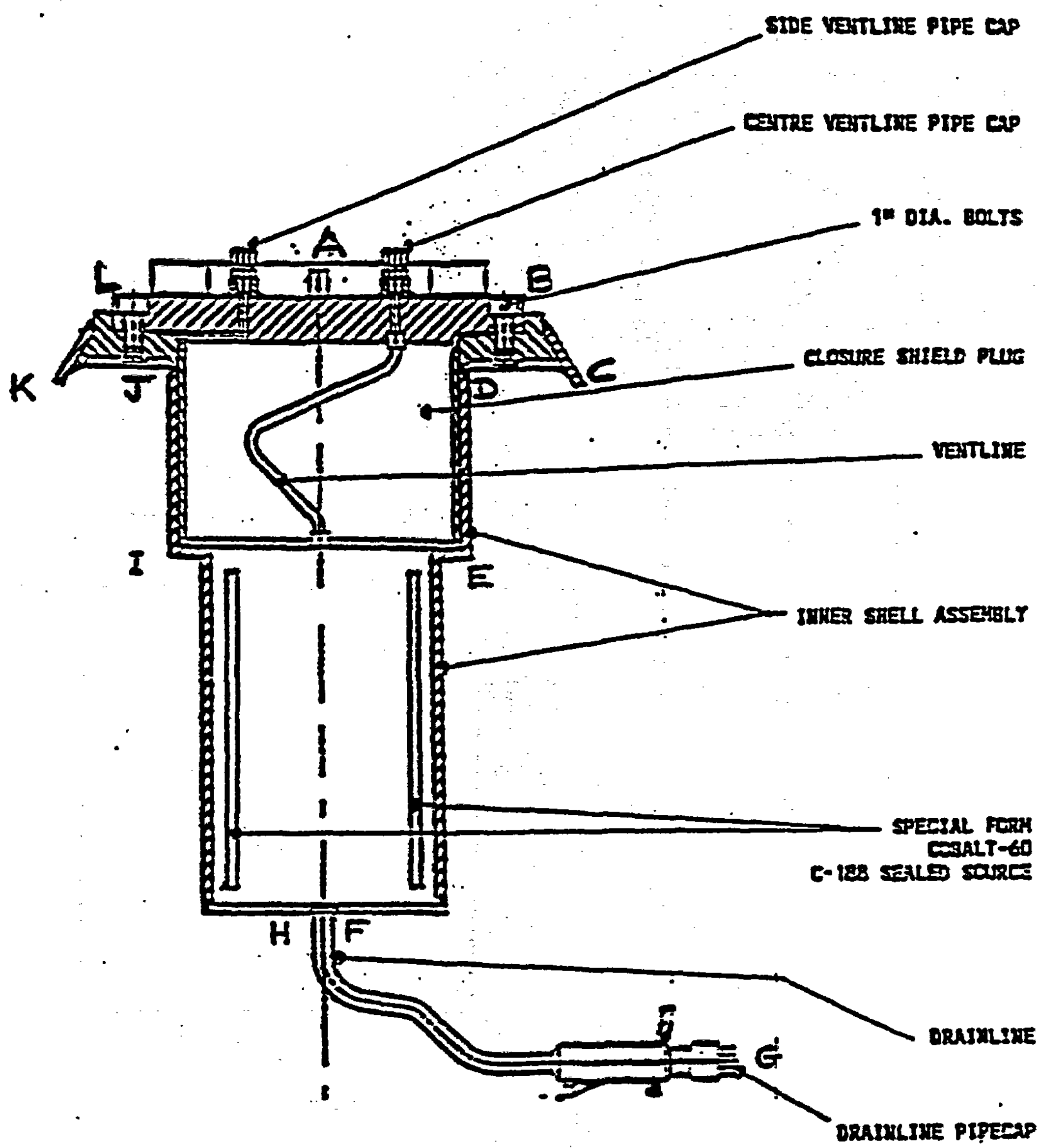


Figure 4.4.3-F2
Closure Plug Bolted Joint: Applied/Reactive Forces

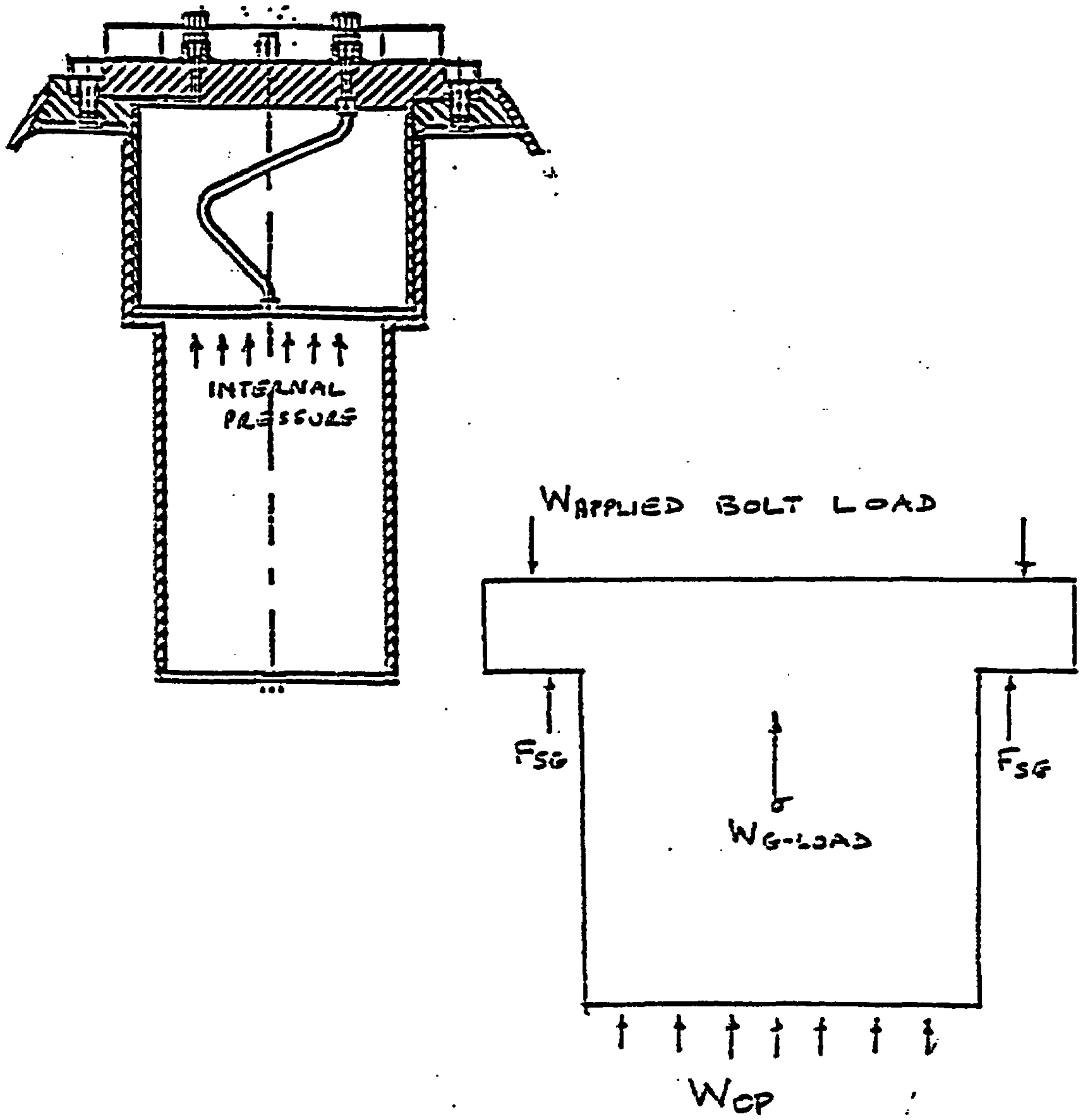


Figure 4.4.3-F3
Shear of the Threads of the Bolt Hole of Closure Plug Joint

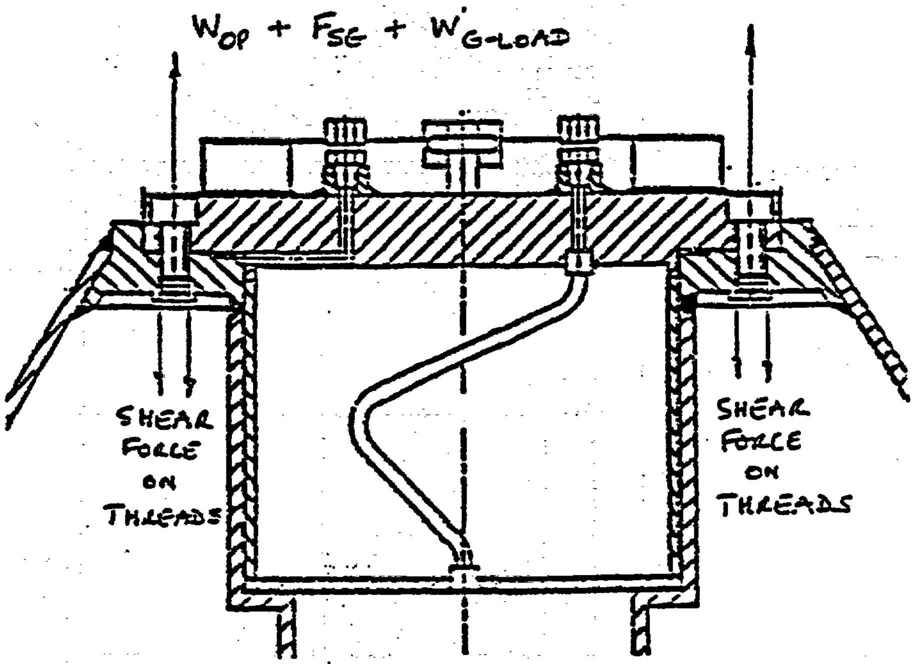


Figure 4.4.3-F4
Plug Cylindrical Body Impacting on Side: Plug Bolt Shearing

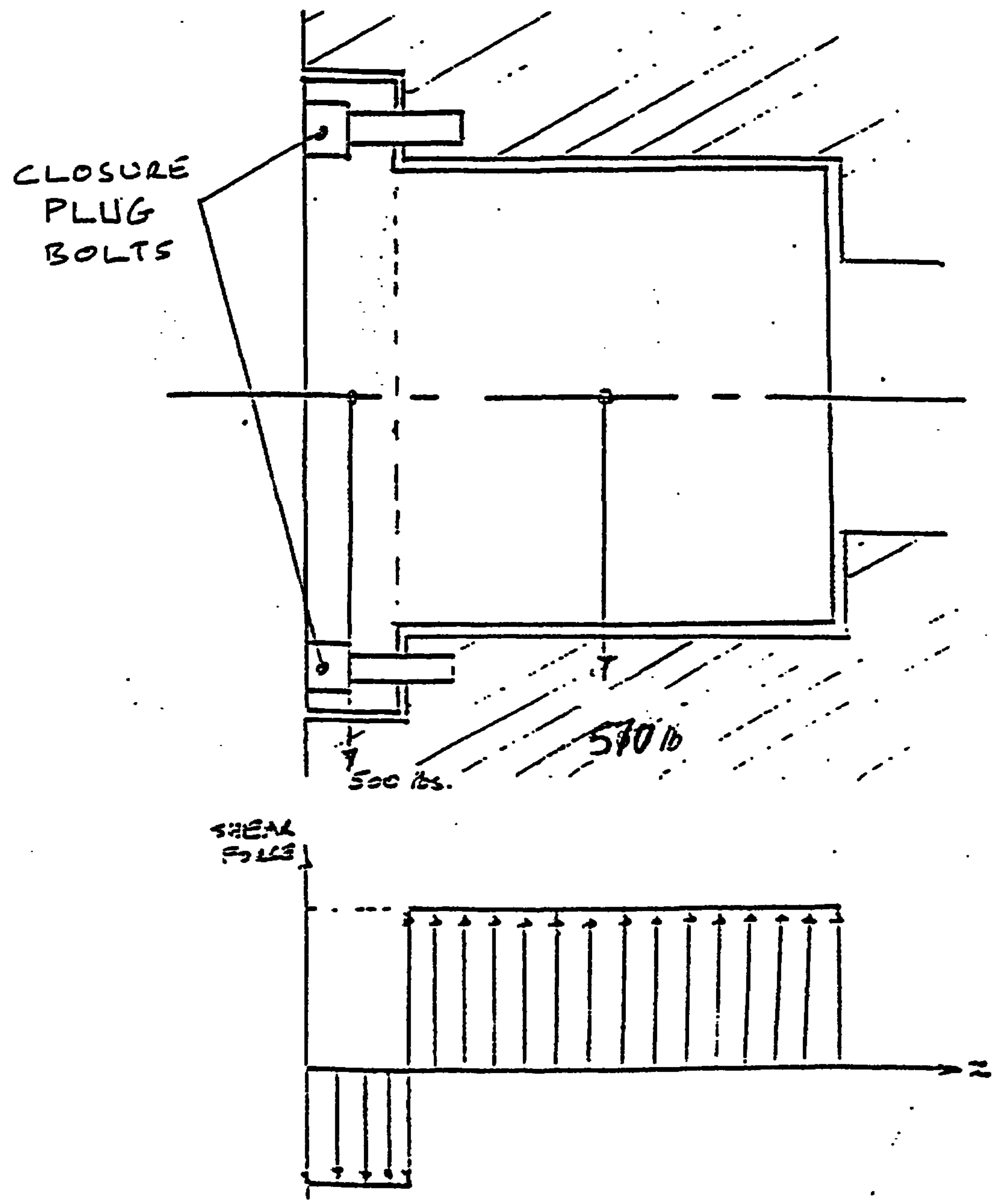


Figure 4.4.3-F5
Plug: Side Drop: Shear Force Diagram

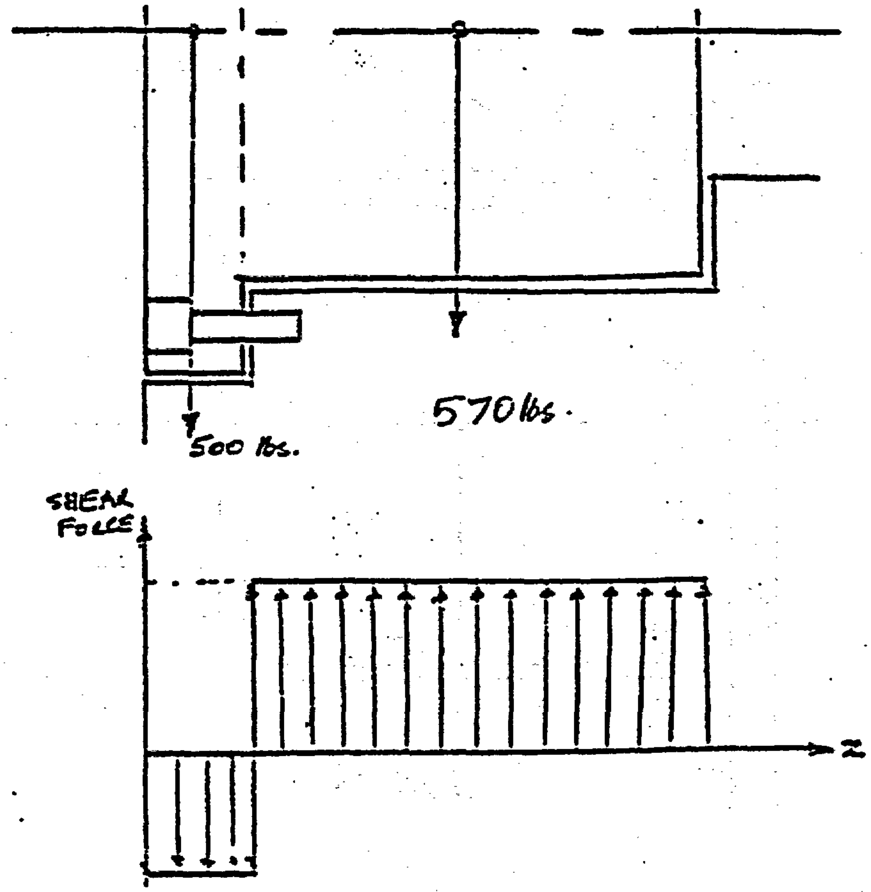


Figure 4.4.3-F6
Plug Cylindrical Body Impacting on Side: Plug Welds

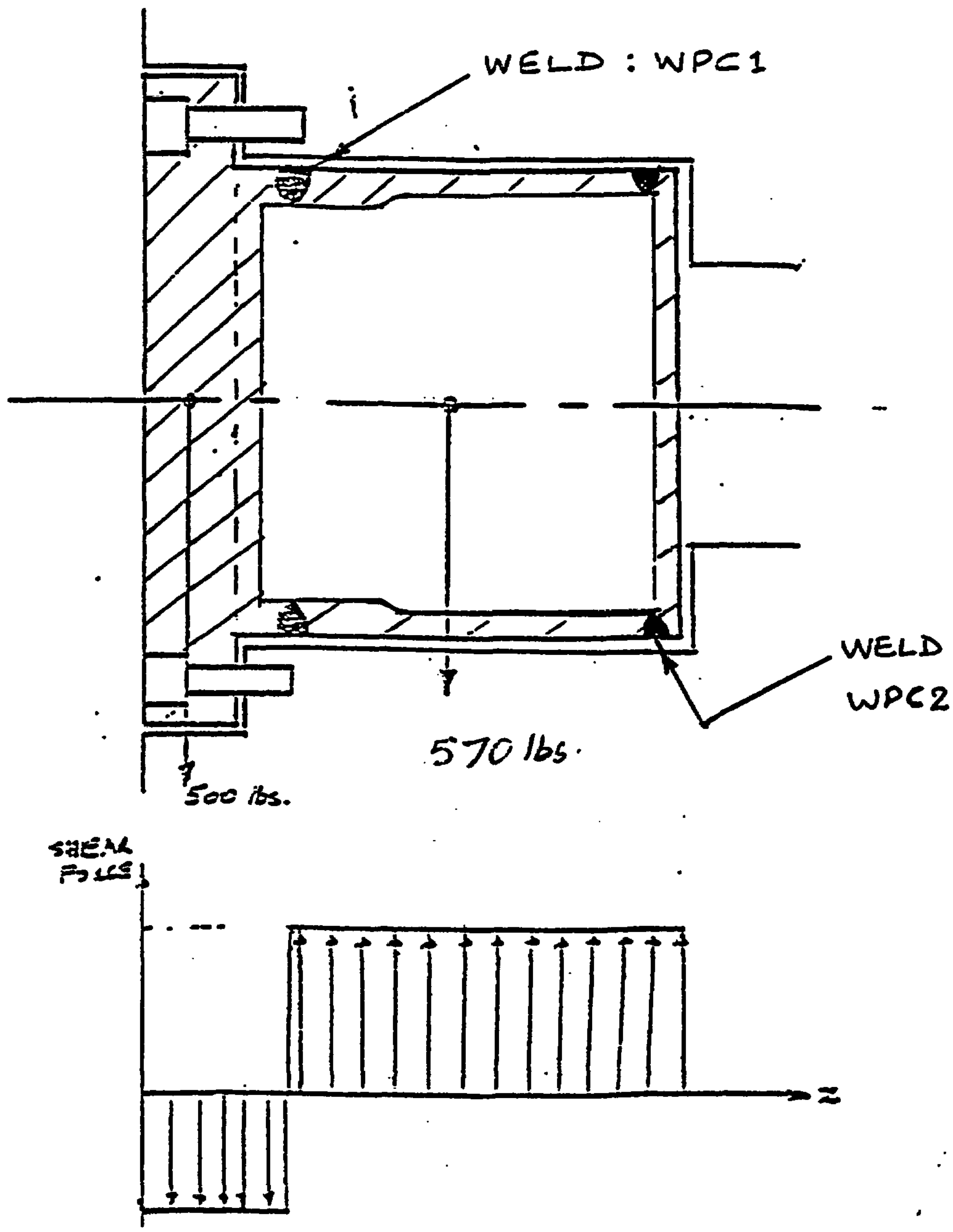


Figure 4.4.3-F7
Lower Cavity Tube

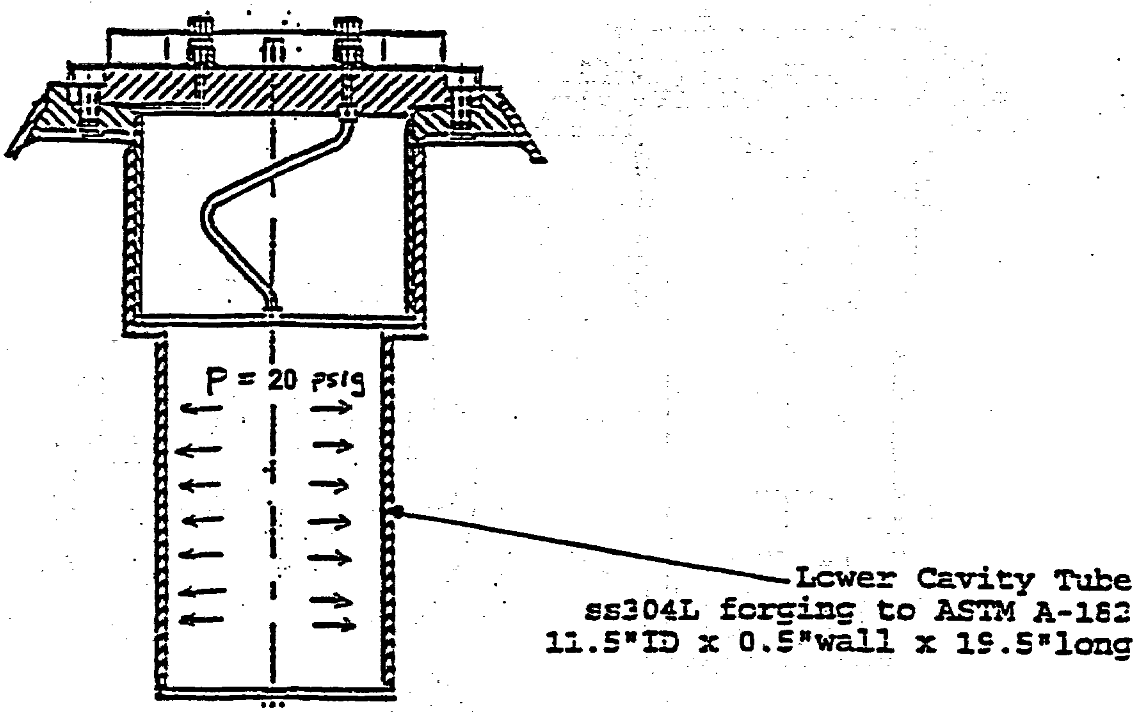


Figure 4.4.3-F8
Upper Cavity Tube

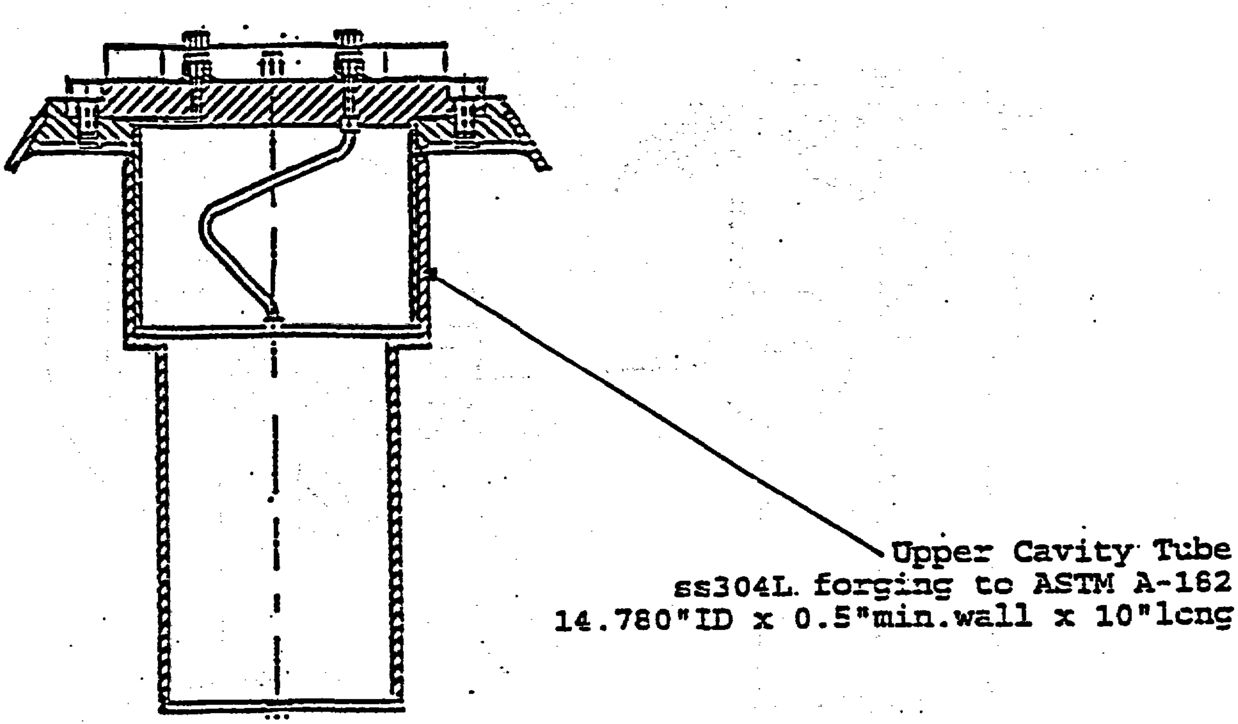


Figure 4.4.3-F9
Lower Cavity End Cap

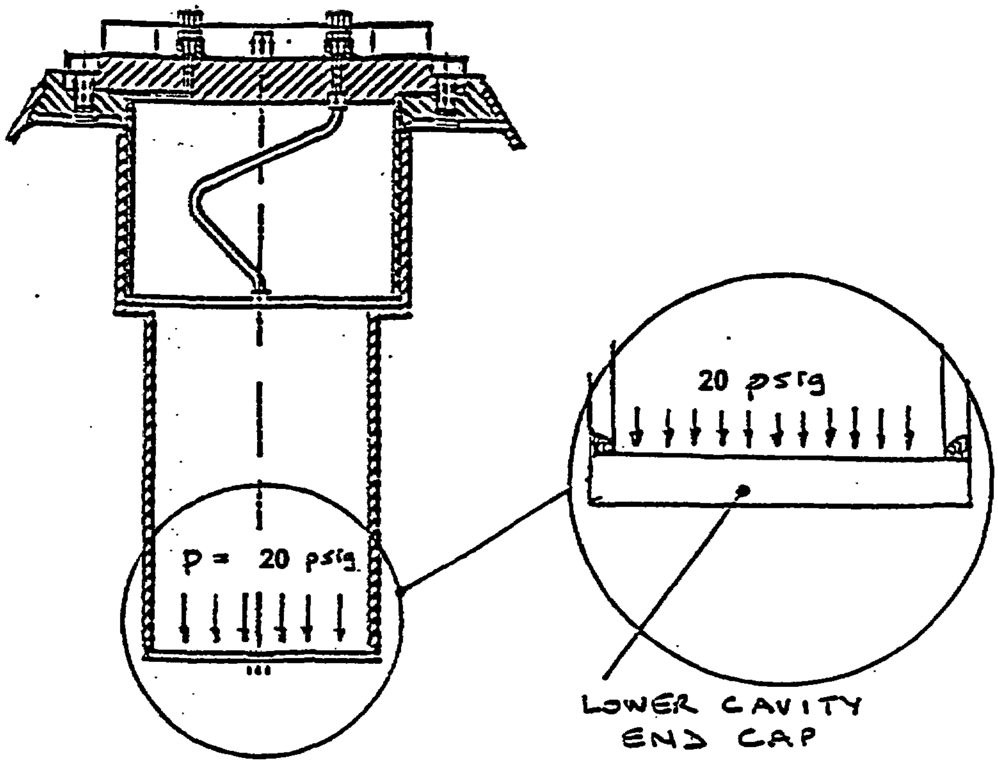


Figure 4.4.3-F10
Upper Cavity Ring Flange

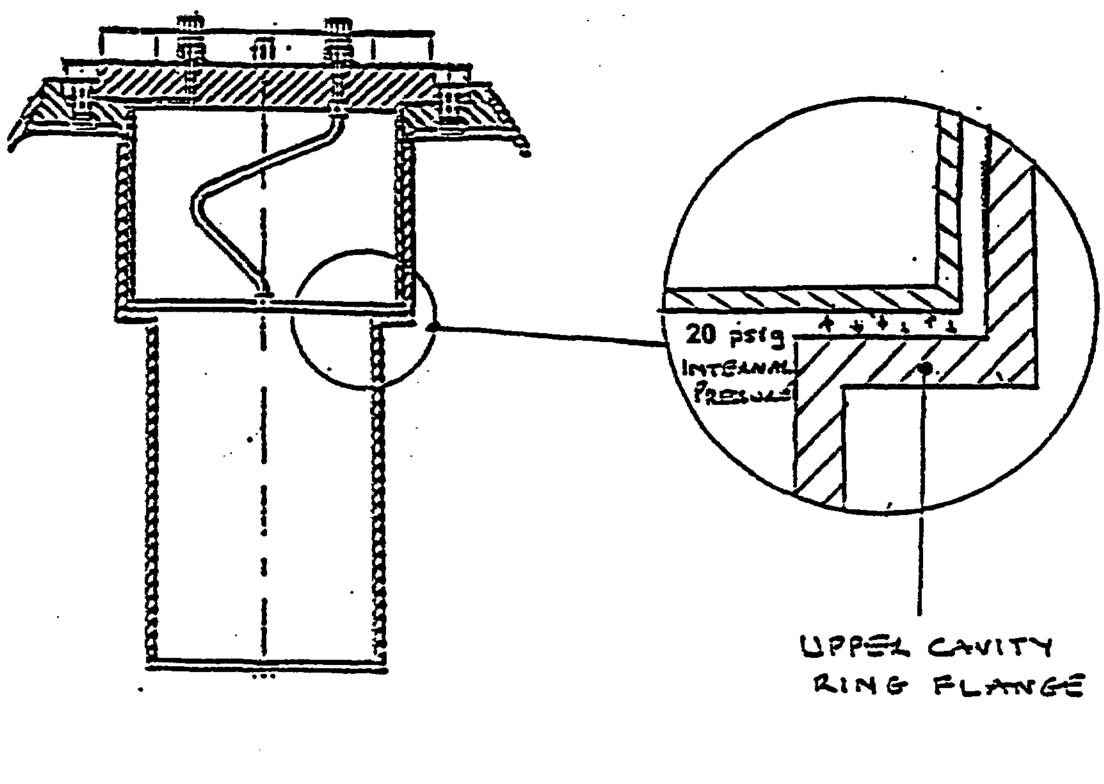


Figure 4.4.3-F11
Lower Cavity Tube Assembly under Axial Load

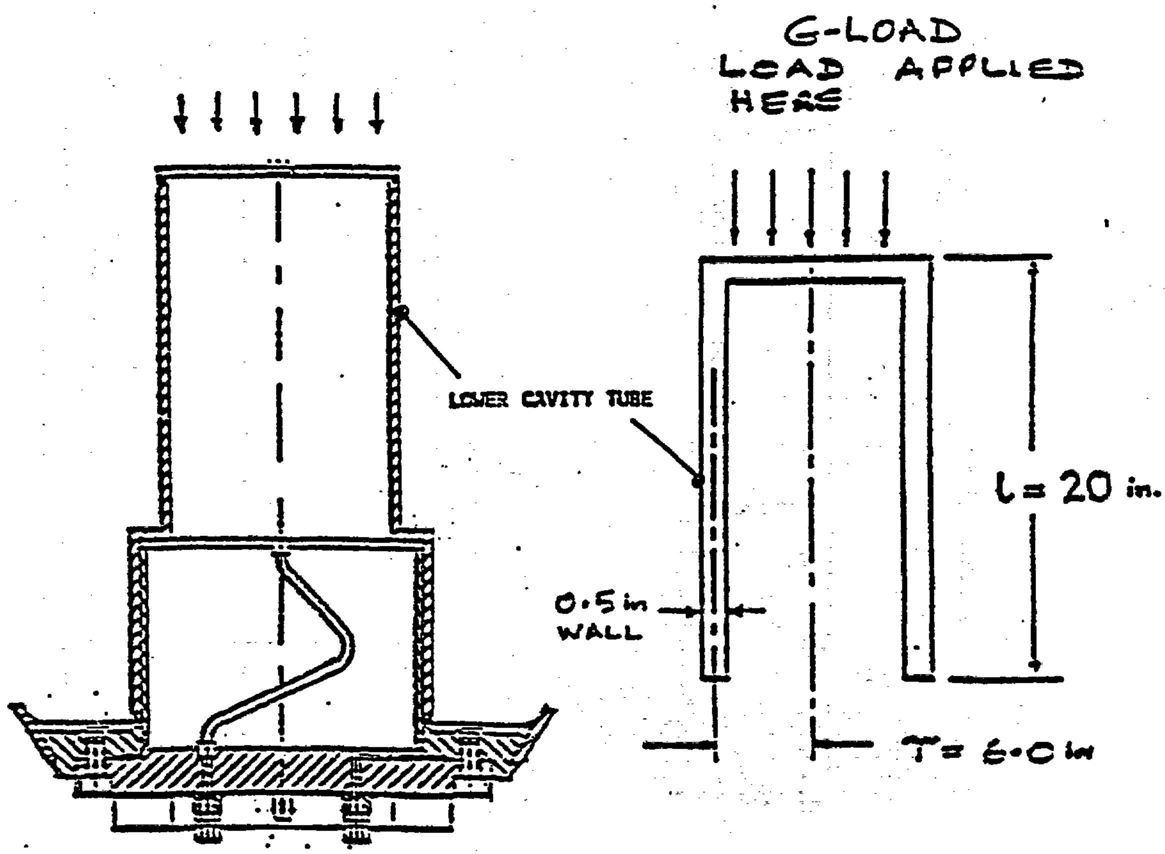


Figure 4.4.3-F12
Lower Cavity Tube End Cap

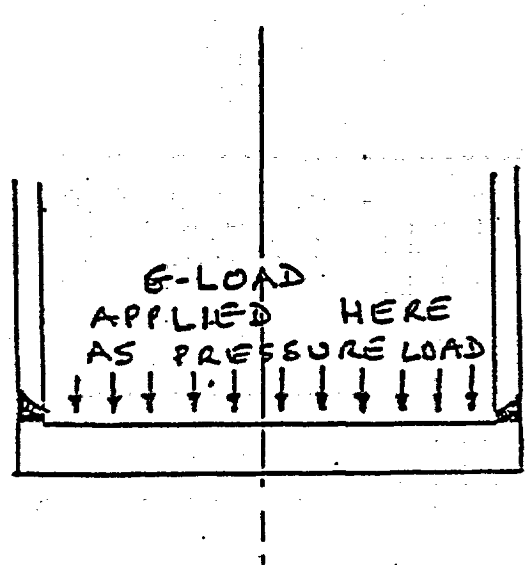


Figure 4.4.3-F13
Buckling of Upper Cavity Tube

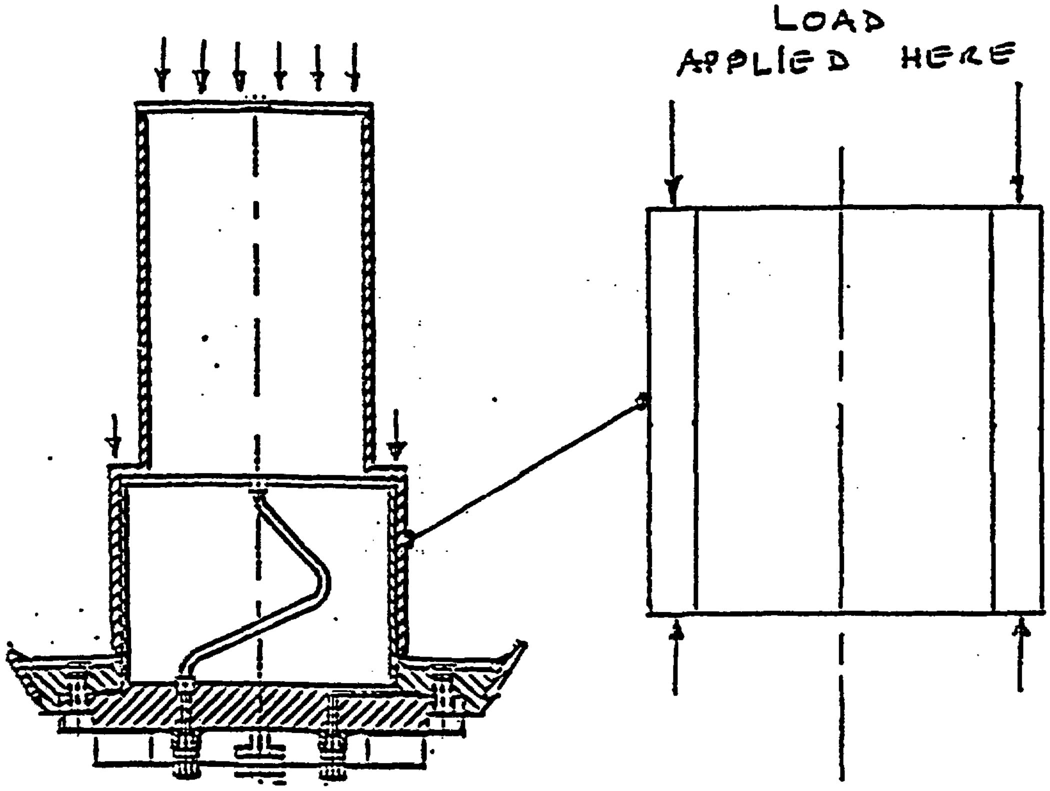


Figure 4.4.3-F14
Bending of Upper Cavity Ring Flange

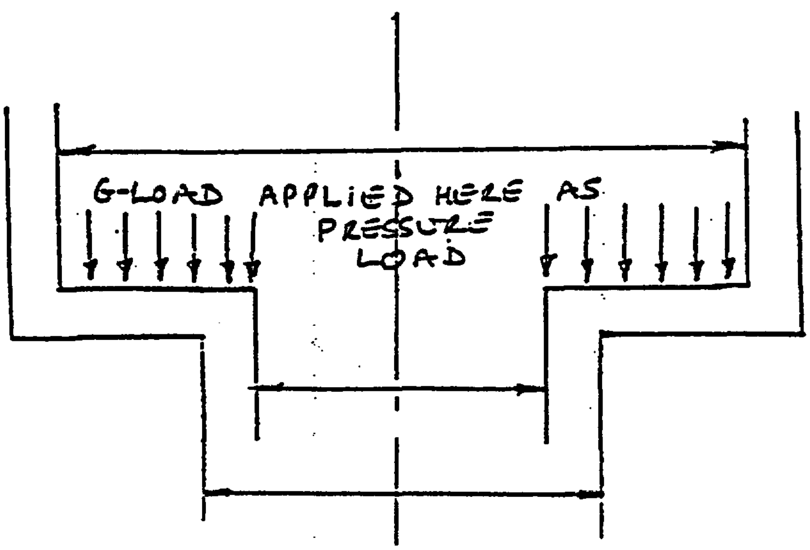


Figure 4.4.3-F15
Container Top Flange under Axial Load

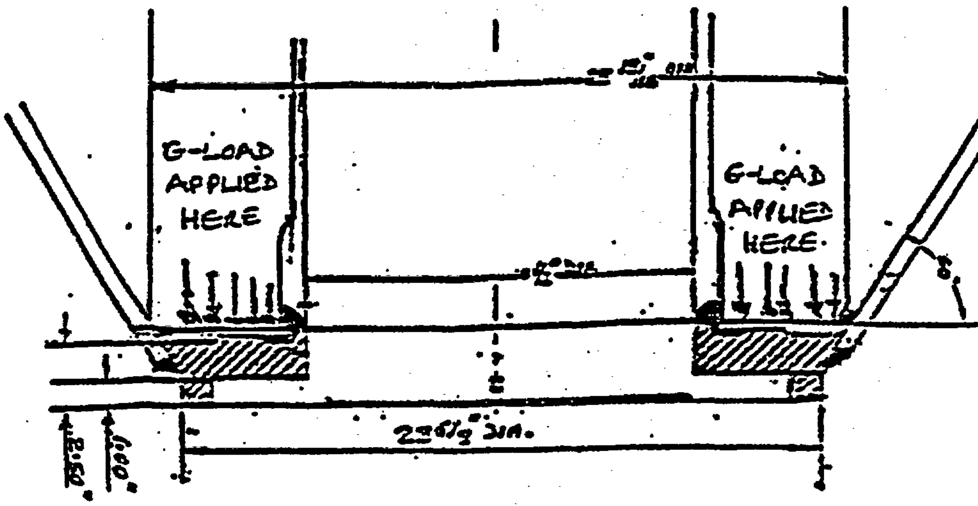
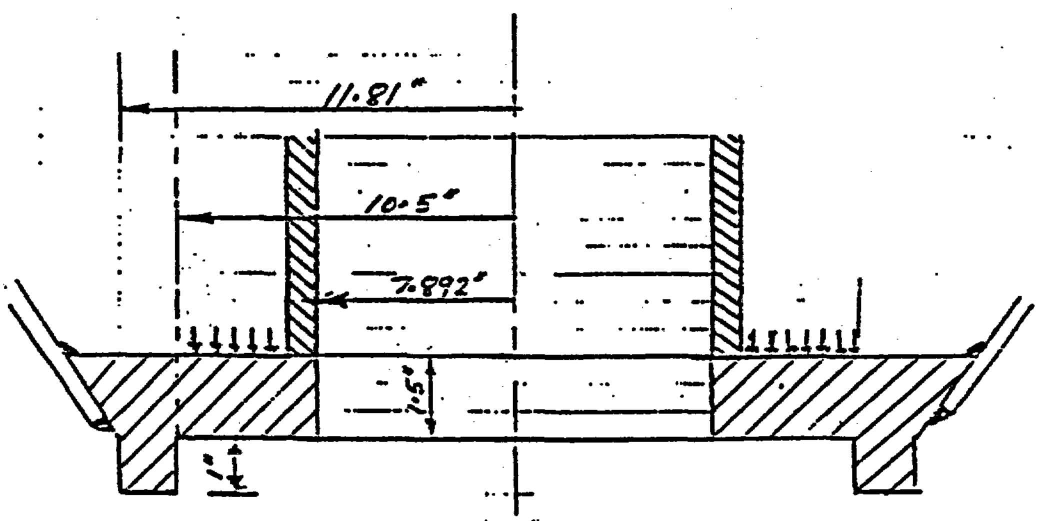


Figure 4.4.3-F16
Container Top Flange under Axial Load: Details



APPENDIX 4.4.4
C-188 STRUCTURAL INTEGRITY UNDER NORMAL CONDITIONS OF TRANSPORT OF F-294

1. INTRODUCTION

C-188 sources are loaded in the F-294 container cavity from a shielded cell at atmospheric pressure and the source equilibrium temperature or from an underwater pool facility at an external pressure of 30 ft. head of water and pool water temperature of 70°F – 100°F. When the C-188 sources are in the F-294/ F-313 container cavity, they reach an equilibrium temperature in about 24 hours and the highest temperature is 836°F for the 360 kCi case. In the F-294/F-457, this temperature is 842°F. As a result of the increase in the C-188 source temperature, the C-188 source is internally pressurized, based on natural gas laws. Estimates of pressure build up were presented in Chapter 3, Section 3.4 and are recaptured here.

1.2 WITH 360 KCI CASE:

In the C-188 assembly, the pressure build up is as follows:

T_1 = Ambient temperature of C-188 prior to loading in F-294 =70°F in the pool
P_1 = Pressure of C-188 prior to loading in F-294 = 14.7 psia
T_2 = Temperature of C-188 after loading in F-294 = 842°F
P_2 = Pressure of C-188 after loading in F-294 = ? (unknown) psia

$$
\begin{aligned}
P_2 &= P_1 \times [T_2 + 460]/[T_1 + 460] \\
&= 14.7 \times [842 + 460]/[70 + 460] \\
&= 14.7 \times 1{,}302/530 \\
&= 36.11 \text{ psia} \\
&= 21.4 \text{ psig} \\
&\approx 22 \text{ psig}
\end{aligned}
$$

During normal conditions of transport of C-188s in the F-294, the C-188 has an internal pressure of 22 psig and maximum temperature of 842°F.

2. STRESS ANALYSIS OF C-188 UNDER NORMAL CONDITIONS OF TRANSPORT F-294

Step 1:

Internal pressure of 22 psig.
Capsule Temperature = 842°F

Step 2:

See Figure 4.4.4.F1 depicting the C-188 loading under internal pressure.

Step 3:

Three (3) distinct regions shall be stress analyzed. They are:

1. The cylindrical tube with 0.025 in. thick wall and 0.380 in. OD, away from the end cap region.
2. The cylindrical tube with 0.017 in. thick wall and 0.380 in. OD, at the end cap region.
3. The end cap.

Step 4:

Cylindrical Tube away from end cap region.

the maximum hoop stress due to internal pressure is given by:

σ_2 = pd/2t

where

p = internal pressure = 22 psi

d = mean diameter = 0.380 – 0.025 = 0.355 in.

t = wall thickness away from the end cap region = 0.025 in.

σ_2 = 22 x 0.355/2 x 0.025

= 157 psi

Step 5:

Cylindrical Tube at the end cap region.

The maximum hoop stress due to internal pressure is given by:

σ_2 = pd/2t

where

p = internal pressure = 22 psi

d = mean diameter = 0.380 – 0.017 = 0.363 in.

t = wall thickness of tube at the end cap region = 0.017 in.(min.)

σ_2 = 22 x 0.363/2 x 0.017

= 235 psi

Step 6:

Stress in the end cap.

Based on ASME VIII, Div. 1, UG-34, "Unstayed Flat Head and Covers", the maximum bending stress due to internal pressure is given by:

σ_b = cp/[t/d]^2

where

p = internal pressure = 22 psi

d = Internal diameter of end cap = 0.337 in.

t = thickness of the end cap = 0.450 in. (min.)

c = constant, based on joint configuration.

The C-188 end cap joint approximates to that of Figure UG-34 (h), ASME VIII, Div. 1. = 0.33

σ_b = 0.33 x 22 /[0.450/0.337]^2

= 4 psi

Step 7:

Factor of Safety and Margin of Safety.

For ss316L, at 842°F, (from Ref. D.E. Fraser)

σ_{YS} = 16,500 psi

σ_{UTS} = 60,000 psi

Factor of Safety (FS)

FS = Allowable stress/Applied stress

= Yield Stress of ss316L at 842°F / hoop stress in tube at joint

= 16,500 psi/235 psi

= 70.2

Margin of Safety, MS

MS = FS – 1 = 70.2 – 1 = 69.2

Step 8:

Summary.

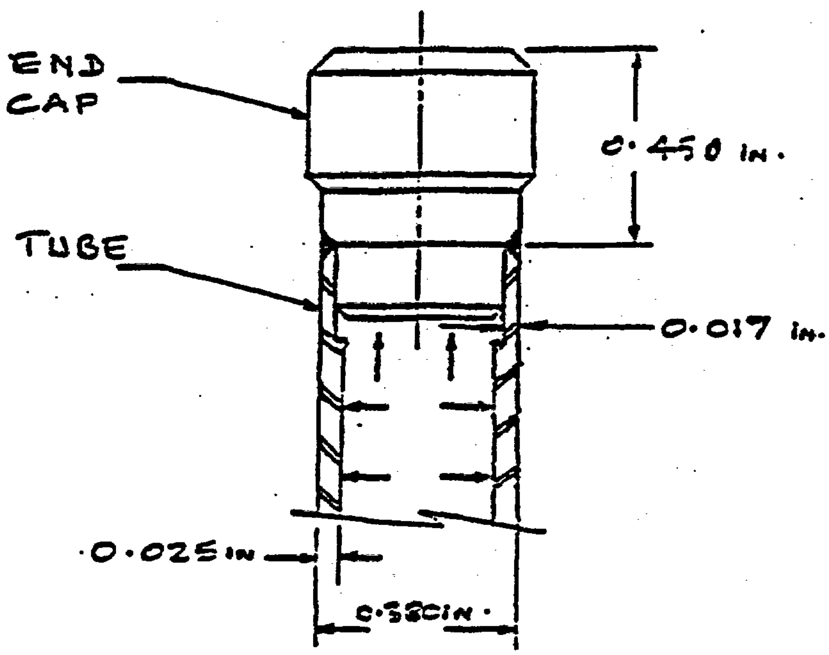
Due to internal pressure of 22 psig in the C-188 during NCOT of F-294,

1. the hoop stress in the tube away from joint = 157 psi
2. the hoop stress in the tube at the joint = 235 psi
3. the bending stress in the end cap = 4 psi.

Based on yield stress of 16,500 psi for ss316L at 842°F, the C-188 has a Factor of Safety of 70.2 and Margin of Safety of 69.2

Therefore the containment, i.e., the outer assembly of the C-188 sealed source, shall maintain its structural integrity.

Figure 4.4.4.F1
C-188 Under Internal Pressure Loading in F-294



APPENDIX 4.4.5
C-188 STRUCTURAL INTEGRITY UNDER HYPOTHETICAL ACCIDENT CONDITIONS OF TRANSPORT (HACOT) OF F-294

1. INTRODUCTION

C-188 sources are loaded in the F-294 container cavity from a shielded cell at atmospheric pressure and the source equilibrium temperature <u>or</u> from an underwater pool facility at the external pressure of 30 ft. head of water and pool water temperature of 70°F – 100°F. When the C-188 sources are in the F-294/ F-313 container cavity, they reach an equilibrium temperature of 836°F for 360 kCi case in about 24 hours. In the F-294/F-457, this temperature is 842°F. As a result of the increase in the C-188 source temperature, the C-188 source is internally pressurized, based on natural gas laws. Subsequently when the F-294 package is subjected to a thermal test under HACOT, there is additional internal pressure build up in the C-188 source. Estimates of this pressure build up were presented in Chapter 3, Section 3.5 and are recaptured here.

1.1 360 KCI CASE:

In the C-188 assembly, the pressure build up is as follows:

T_1 = Temperature of C-188 in underwater pool = 70°F

P_1 = Internal Pressure of C-188 in underwater pool = 14.7 psia

T_2 = Temperature of C-188 in HACOT of F-294 = 955°F

P_2 = Pressure of C-188 in HACOT of F-294 = ? (unknown) psia

$$
\begin{aligned}
P_2 &= P_1 \times [T_2 + 460]/[T_1 + 460] \\
&= 14.7 \times [955 + 460]/[70 + 460] \\
&= 14.7 \times 1{,}415/530 \\
&= 39.2 \text{ psia} \\
&= 24.5 \text{ psig} \\
&\approx 27 \text{ psig}
\end{aligned}
$$

During Hypothetical Accident Conditions Of Transport (HACOT) of C-188s in the F-294, the C-188 has an internal pressure of 27 psig and maximum temperature of 955°F.

2. STRESS ANALYSIS OF C-188 UNDER INTERNAL PRESSURE

Step 1:

Internal pressure of 27 psig.
Capsule Temperature = 955°F

Step 2:

See Figure 4.4.5.F1 depicting the C-188 loading under internal pressure.

Step 3:

Three (3) distinct regions shall be stress analyzed. They are:

1. The cylindrical tube with 0.025 in. thick wall and 0.380 in. OD, away from the end cap region.
2. The cylindrical tube with 0.017 in. thick wall and 0.380 in. OD, at the end cap region.
3. The end cap.

Step 4:

Cylindrical Tube away from end cap region.
The maximum hoop stress due to internal pressure p is given by:

$\sigma_2 = pd/2t$

where

p = internal pressure = 27 psig
d = mean diameter = 0.380 – 0.025 = 0.355 in.
t = wall thickness away from the end cap region = 0.025 in.
σ_2 = 27 x 0.355/2 x 0.025
= 192 psi

Step 5:

Cylindrical Tube at the end cap region.
The maximum hoop stress due to internal pressure p is given by:

$\sigma_2 = pd/2t$

where

p = internal pressure = 27 psig
d = mean diameter = 0.380 - 0.017 = 0.363 in.
t = wall thickness of tube at the end cap region = 0.017 in.(min.)
σ_2 = 27 x 0.363 / 2 x 0.017
= 288 psi

Step 6:

Stress in the end cap.
Based on ASME VIII, Div. 1, UG-34, "Unstayed Flat Head and Covers", the maximum bending stress due to internal pressure p is given by:

$\sigma_b = cp/[t/d]^2$

where

p = internal pressure = 27 psig
d = Internal diameter of end cap = 0.337 in.
t = thickness of the end cap = 0.450 in. (min.)
c = constant, based on joint configuration.

The C-188 end cap joint approximates to that of Figure UG-34 (h), ASME VIII, Div. 1. = 0.33

$\sigma_b = 0.33 \times 27 /[0.450/0.337]^2$
= 5 psi

Step 7:

Factor of Safety and Margin of Safety.
For ss316L, at 955°F, (from Ref. D.E. Fraser)

σ_{YS} = 15,000 psi
σ_{UTS} = 55,000 psi

Factor of Safety (FS)

FS = Allowable stress/Applied stress
= Yield Stress of ss316L at 955°F/hoop stress in tube at joint
= 15,000 psi/288 psi
= 52.1

Margin of Safety, MS

MS = FS – 1 = 52.1 – 1 = 51.1

Step 8:

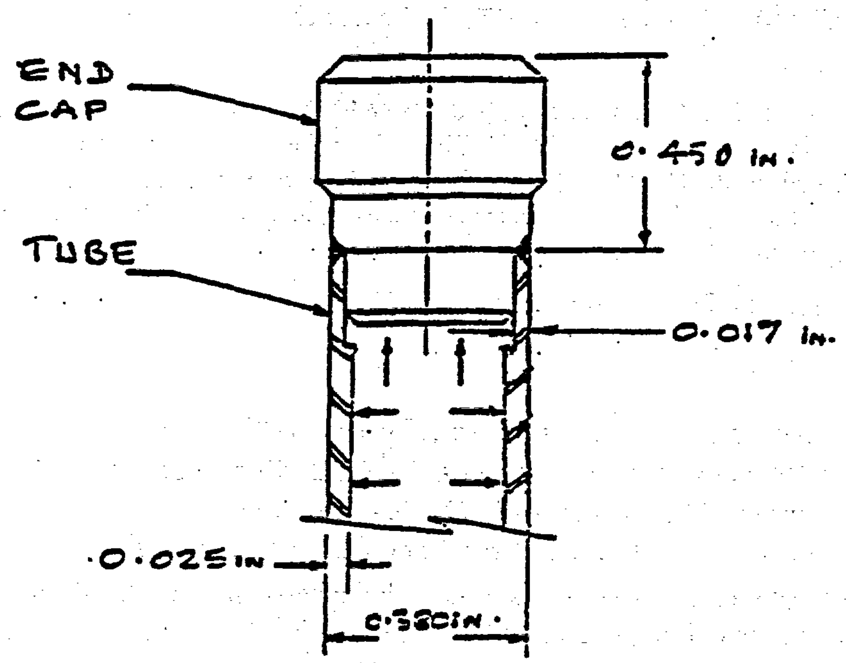
Summary.
Due to internal pressure of 27 psig in the C-188 during HACOT of F-294,

1. the hoop stress in the tube away from joint = 192 psi
2. the hoop stress in the tube at the joint = 288 psi
3. the bending stress in the end cap = 5 psi.

Based on yield stress of 15,000 psi for ss316L at 955°F, the C-188 has a Factor of Safety of 52 and a Margin of Safety of 51.

Therefore the containment, i.e., the outer assembly of the C-188 sealed source, shall maintain its structural integrity.

Figure 4.4.5-F1
C-188 under Internal Pressure Loading in F-294



3. C-188 SEALED SOURCE UNDER TOP END IMPACT

The case of the C-188 sealed source capsule under top impact is presented here. The model chosen to represent this case is depicted in Figure 4.4.5-F2.

The stress developed in the capsule outer shell will be calculated due to the deceleration of mass of the entire C-188 at 130 g's.

The weight of C-188 is $W_{C\text{-}188}$ = 130 g x 0.5 lb. = 65 lb. (G-load of 130 g's is as per Chapter 2, Appendix 2.10.12).

During the drop tests, the normal temperature of the C-188, based on 360 kCi of cobalt-60 case, is 824°F in the F-294/F-313 and 842°F in the F-294/F-457.

Compression

The support reaction P due to 130-g level is:

$$P = -W_{C\text{-}188} \times \text{G-load} = -0.5 \times 130 = 65 \text{ lb.}$$

The axial stress for this condition is given by:

$$\begin{aligned}\sigma_{comp.} &= P/A_{\text{outer shell, min. wall}} \\ &= 65/(\pi/4\,(0.376^2 - 0.338^2)) \\ &= 65/0.0213 \\ &= 3{,}052 \text{ psi.}\end{aligned}$$

Safety Factor = Allowable Stress/Applied stress
= UTS for ss316L at 842°F/applied stress
= 60,000 psi/3,052 psi
= 19.6

Margin of Safety = SF – 1 = 19.6 – 1 = 18.6

The yield stress of ss316L at 842°F = 16,500 psi.

As the $\sigma_{comp.}$ (3,052 psi) < Yield Stress (16,500 psi), the tube shall not yield nor buckle in the top end drop.

Buckling

Let us examine the C-188 sealed source capsule under end impact for buckling. The model chosen to represent this case is shown in Figure 4.4.5-F2. The ends of the capsule are free to rotate, translation is fixed because the ends of the C-188 are trapped between the bottom plate of the source carrier and the shield plug. Any restraining of the C-188 sealed source capsule by intermediate spacer plates of the source holder has been ignored.

The critical buckling load (Euler load) is given by

$$P_{cr} = \pi^2 EI/kl$$

where

E = modulus of elasticity = 22.6×10^6 psi at 955°F
I = 2nd moment of area = $\pi/64\,(0.376^4 - 0.325^4) = 4.335 \times 10^{-4}$ in^4
l = length of the column = 17.777 – 0.9 = 16.877 in.
k = effective length factor, dependent on the conditions of fixity of the column.

In this case the column is free to rotate (i.e., hinge; zero moment reaction), but translation is zero. Therefore, the column end condition code is "pin-jointed and fixed end".

In this case K = 1.2 (Ref.[5] CISC Handbook 1967).

Therefore

$$P_{cr} = \pi^2\, 22.6 \times 10^6 \times 4.335 \times 10^{-4}/(1.2\text{x}\ 16.877)^2$$
$$= 235 \text{ lb.}$$

The weight of the C-188, $W_{C\text{-}188} = 0.5$ lb.

Therefore, the G-load which may initiate buckling

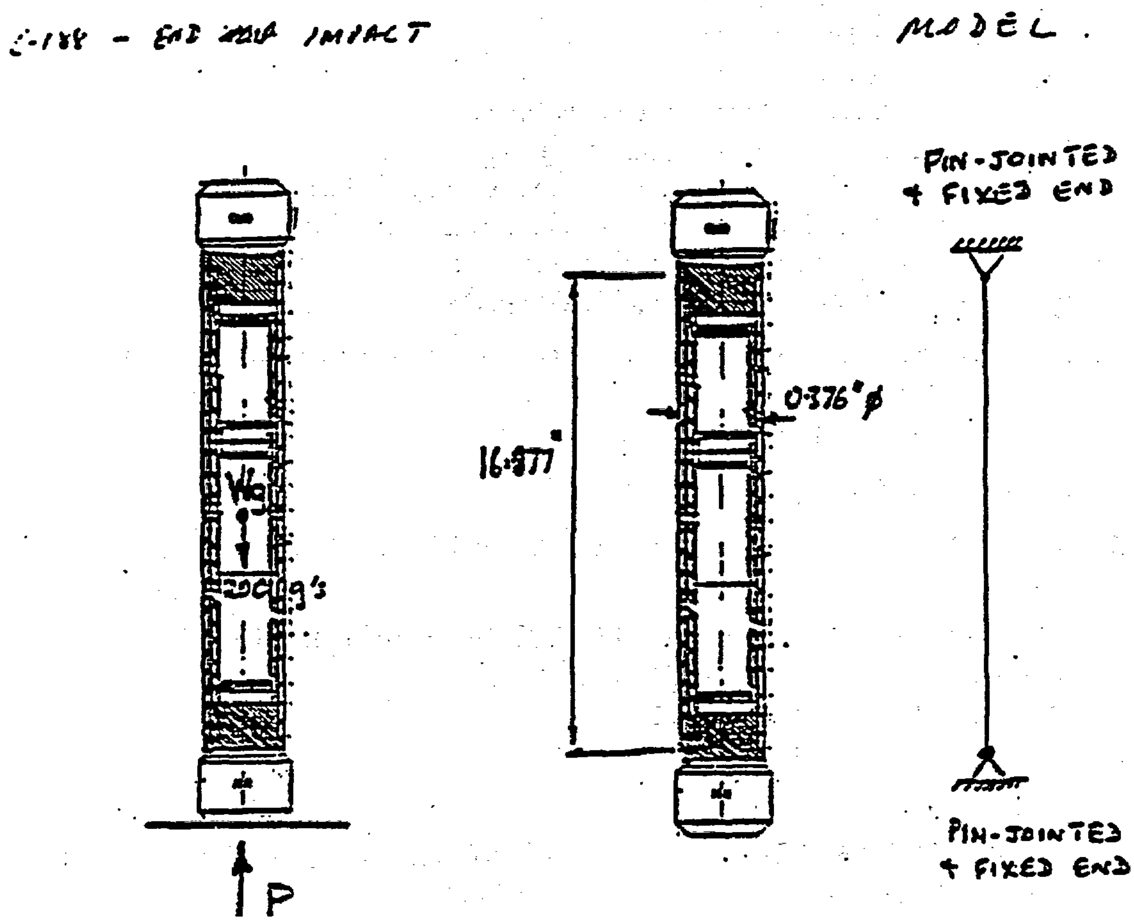
$$\text{G-load}_{buckling} = P_{cr}/W_{C\text{-}188}$$
$$= 235/0.5$$
$$= 470 \text{ g's.}$$

$$SF_{G\text{-}LOAD\ BASED} = \text{Allowable g-load/Applied g-load}$$
$$= 470/130$$
$$= 3.6$$

$$MS_{G\text{-}LOAD\ BASED} = SF - 1 = 3.6 - 1 = 2.6$$

Therefore, it is concluded that the C-188 sealed source integrity, in the top end drop impact, shall be maintained as the Margin of Safety, either stress based or load based, exceeds 1.

Figure 4.4.5-F2
C-188 under Top End Impact



4. C-188 IN SIDE DROP IMPACT

In a side oblique drop test, the deceleration loads in the F-294 cavity were 54 g's (at bottom of cavity) and unknown (at top of cavity) as the signal cables were cut during the drop test. Therefore, the measured G-loads are not clearly established. Consequently, to assess C-188 stresses, it is assumed that the cavity is subjected to 1,000 g's. In a side drop, the C-188 loading case is represented in the model as per Figure 4.4.5-F3. The C-188 is normally held within the F-313 carrier with three clear spans of 5 in. and the balance of column length is free (i.e., overhang).

The dimensions of the C-188 are:

Length $L_{C\text{-}188}$	= 17.777 in.
OD	= 0.380/0.376 in.
ID	= 0.330/0.326 in.

The weight of the C-188 is 0.5 lb. If this mass is assumed to be distributed uniformly along the length of the C-188, the distributed load ω due to design load of 1000-g is:

$$\omega = 1000 \times 0.5/17.777 = 28.126 \text{ lb./in.}$$

The three-moment equation in terms of three consecutive support points (i, j, k) is (see Figure 4.4.5-F4):

$$M_iL_i + 2M_j(L_i + L_j) + M_kL_j = -6A_{ij}a_{ij}/L_{ij} - 6A_{jk}b_{jk}/L_{jk}$$

where

M_i	= moment at support i
M_j	= moment at support j
M_k	= moment at support k
L_{ij}	= length of span between supports i and j
L_{jk}	= length of span between supports j and k
A_{ij}	= Area of BM diagram of applied load over span of length L_{ij}
A_{jk}	= Area of BM diagram of applied load over span of length L_{jk}
a_{ij}	= distance of centroid of A_{ij} from support i
b_{jk}	= distance of centroid of A_{jk} from support k

For a simple beam under uniform load, the bending moment diagram is a parabola with maximum ordinate $\omega l^2/8$. The area of the parabolic segment is

$$A = 2/3 l \times \omega l^2/8 = \omega l^3/12$$

and its centroid is at mid-span, so that $a = l/2$.

On the C-188 model, supports are identified as 0, 1, 2, 3, and free end as 4.

For Spans 01 and 12:

$$0 + 2M_1(2L_1) + M_2L_1 = -6x(\omega L_1^3/12 \times L_1/2)/L_1 - 6x(\omega L_1^3/12 \times L_1/2)/L_1$$
$$4M_1L_1 + M_2L_1 = -\omega L_1^3/4 - \omega L_1^3/4$$
$$4M_1L_1 + M_2L_1 = -\omega L_1^3/2$$
$$4M_1 + M_2 = -\omega L_1^2/2$$

For Spans 12 and 23

$$M_1L_1 + 2M_2(2L_1) + M_3L_1 = -6x(\omega L_1^3/12 \times L_1/2)/L_1 - 6x(\omega L_1^3/12 \times L_1/2)/L_1$$
$$M_1L_1 + 4M_2L_1 + M_3L_1 = -\omega L_1^3/4 - \omega L_1^3/4$$
$$M_1L_1 + 4M_2L_1 + M_3L_1 = -\omega L_1^3/2$$
$$M_1 + 4M_2 + M_3 = -\omega L_1^2/2$$

For Span 34

$M_3 = -\omega L_2^2/2$

Therefore, 3 equations for 3 unknown moments M_1, M_2, M_3 are:

$4M_1 + M_2 = -\omega L_1^2/2$

$M_1 + 4M_2 + M_3 = -\omega L_1^2/2$

$M_3 = -\omega L_2^2/2$

Substituting

$\omega = 28.126$ lb./in

$L_1 = 5.0$ in.

$L_2 = 2.77$ in.

We get

$M_1 = -77.54$ in-lb.

$M_2 = -41.40$ in-lb.

$M_3 = -108.5$ in-lb.

The reactions at the supports are:

$R_0 = 54.85$ lb.

$R_1 = 163.35$ lb.

$R_2 = 120.0$ lb.

$R_3 = 161.83$ lb.

The shear and bending moment diagrams were constructed as per Figure 4.4.5-F5

The maximum stress was then evaluated.

$M_{max} = 108.5$ in-lb.

$\sigma_{bending} = MC/I$

where

$C = 0.376/2 = .188$

$I = \pi/64\,[0.376^4 - 0.326^4] = 4.268 \times 10^{-4}\ in^4$

$\sigma_{bending} = 108.5 \times 0.188/[4.268 \times 10^{-4}] = 47{,}800$ psi

During the drop tests, the maximum normal temperature of C-188, based on 360 kCi of cobalt-60 case, is 842°F. At 842°F, for ss316L,

YS = 16,500 psi

UTS = 60,000 psi

SF = allowable stress/applied stress

= UTS of ss316L at 842°F / $\sigma_{bending}$

= 60,000/47,800

= 1.25

Margin of Safety, (MS) = SF – 1 = 1.25 – 1 = 0.25

The C-188 is stressed beyond yield stress but not beyond static UTS. Therefore, the C-188 shall not fail.

The shear stress, τ is evaluated next.

The maximum reaction force $R_{max} = 163.4$ lb.

The area of shear,

$A_{shear} = \pi/4\,(0.376^2 - 0.326^2) = 2.757 \times 10^{-2}\ in^2$

Therefore shear stress, τ

τ $= R_{max} / A_{shear} = 163.4$ lb./ 2.757×10^{-2} in^2 = 5,926.5 psi

SF = allowable stress / applied stress

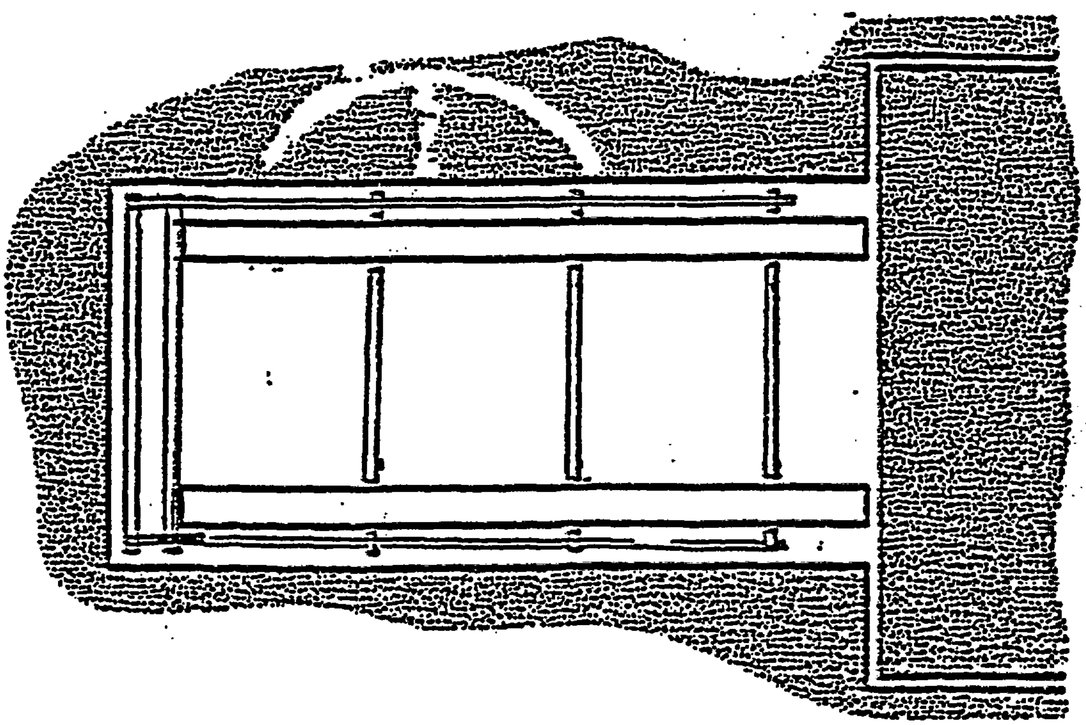
= UTS of ss316L at 842°F/τ

= 60,000 / 5,927

= 10.1

Margin of Safety, MS = SF – 1 = 10.1 – 1 = 9.1

Figure 4.4.5-F3
C-188 Model in a F-294 Side Drop Impact



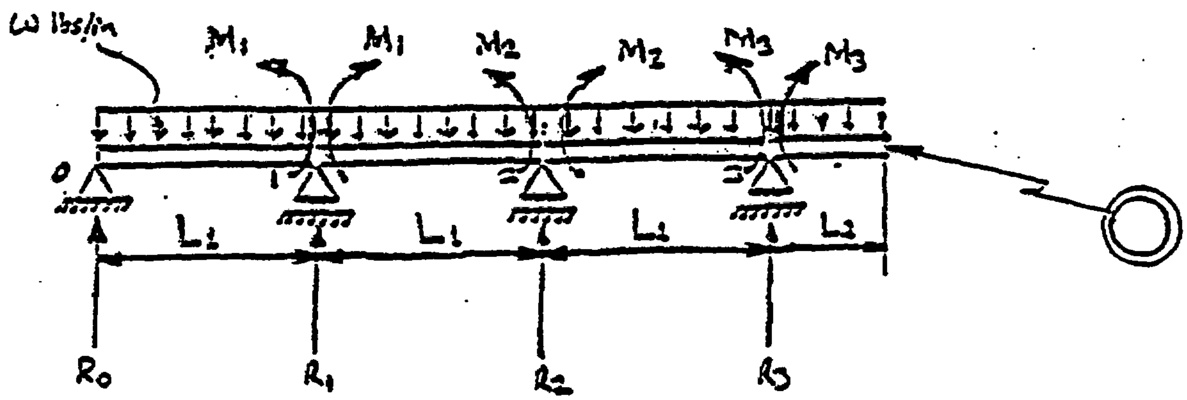


Figure 4.4.5-F4
Three Moment Equation: Identification of Terms

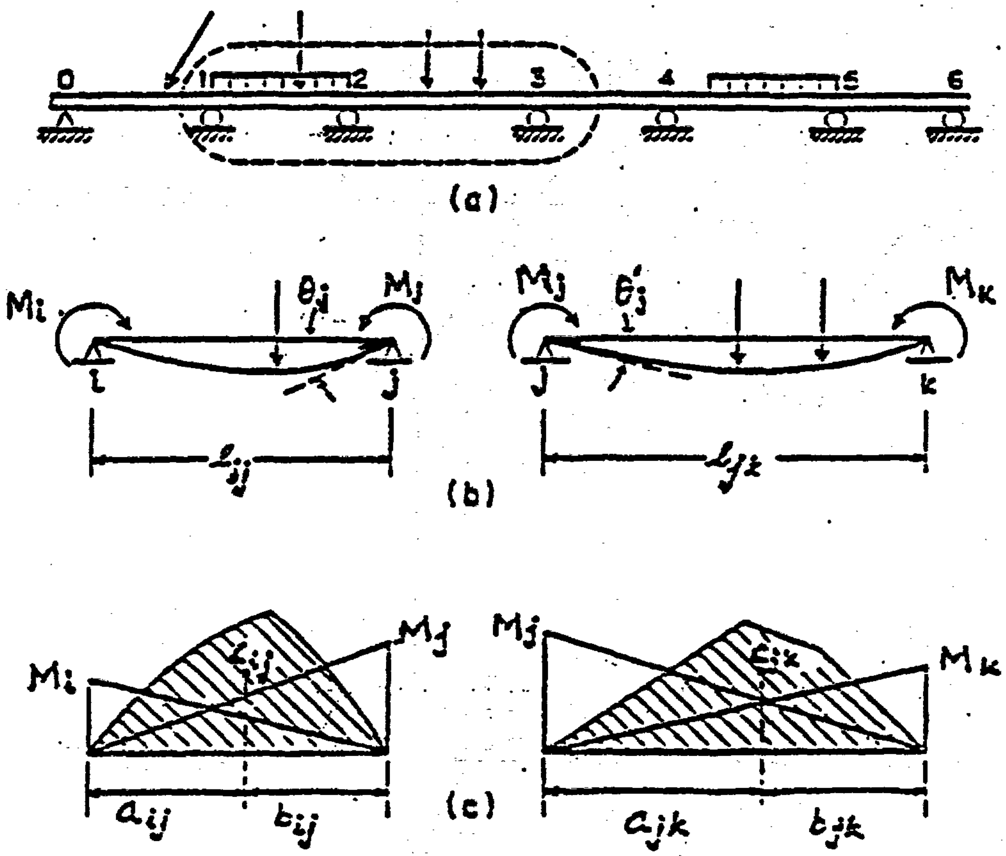
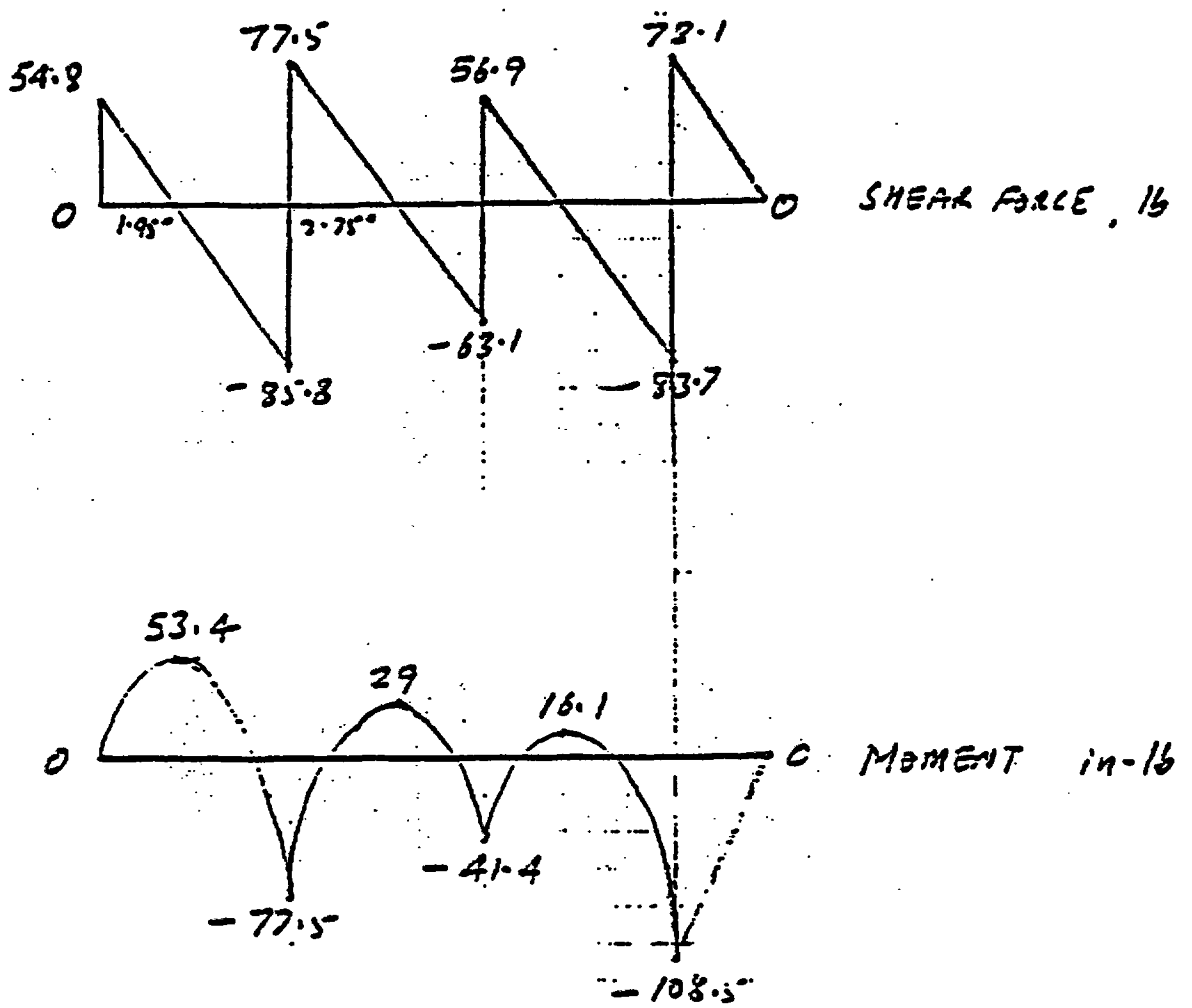


Figure 4.4.5-F5
Shear Force and Moment Diagram of C-188 in a F-294 Side Drop



APPENDIX 4.4.6
STRESS ANALYSIS OF THE CONTAINMENT SYSTEM SUBJECT TO HYPOTHETICAL ACCIDENT CONDITIONS OF TRANSPORT OF F-294

CONTENTS

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1. INTRODUCTION

The inner shell assembly and the lid (closure plug) is defined as the containment system of F-294 package. Figure 4.4.6-F1 depicts the F-294 containment system.

During the hypothetical accident conditions of transport (HACOT), the F-294 package with either the F-313 or the F-457 source carrier is subjected to following:

1. Internal pressure in the cavity of 20 psig.

   The cavity walls are at 500°F design (the actual cavity temperature is 505°F [263°C]). See Chapter 3, Section 3.5.

   The top closure is at 500°F design (the actual top closure temperature is 471°F [244°C]).
2. In the actual 30 ft. free drop, the F-294 is subjected to 132 g's deceleration in the top drop orientation or 136 g's deceleration in the side oblique drop orientation.

The purpose of the closure plug bolted joint is to provide adequate shielding, but not necessarily leaktightness, of the joint between the closure plug and the inner shell assembly. In this Appendix it is demonstrated that the closure plug is retained when F-294 is subject to combination of forces during hypothetical accident conditions of transport. Consequently the closure plug, which provides adequate shielding, shall be retained over the cavity of F-294, which houses Special Form cobalt-60 MDS Nordion C-188 sealed sources. As C-188 is certified Special Form RAM, the C-188 outer encapsulation provides the leaktightness for retention of radioactive material.

Therefore,

1. as C-188 is certified Special Form RAM and provides leak tight containment AND
2. as the closure plug is retained over the inner shell assembly, which houses the cobalt-60 C-188 sealed sources,

F-294 shall meet the HACOT containment system requirements (10 CFR 71.51 (a) (2)).

2. STRESS ANALYSIS OF CLOSURE PLUG BOLTED JOINT

2.1 CLOSURE PLUG BOLTED JOINT SUBJECT TO INTERNAL PRESSURE AND G-LOADS

Due to build-up of internal pressure, the main plug bolted closure is examined in detail. The internal pressure in the cavity is 20 psig (see Figure 4.4.6-F2).

2.1.1 Internal Pressure Load, W_{OP}

The internal pressure load, W_{OP} is calculated as follows:

$$\begin{aligned} W_{OP} &= \Delta P * \text{Area} \\ &= \Delta P * [\pi * G^2/4] \\ &= 20 * \pi * 15.91^2/4\ [\text{psi} * \text{in}^2] \\ &= 4{,}000\ \text{lb.} \end{aligned}$$

where

ΔP = 20 psi (internal pressure - outside the F-294 container at atmospheric pressure)

G = gasket reaction diameter = 15.91 in.

2.1.2 Gasket Seating Load, F_{SG}

$F_{SG} = \pi * b * G * y$

where

- b = effective gasket seating width
- G = gasket diameter
- y = gasket seating stress = 200 psi
  (Ref.[17] i.e., ASME VIII Div. I: Table UA-49-1)

Basic gasket seating width, b_O = actual width of gasket/2
= (16.38 – 15.44) x 0.5/2
= 0.235 in.

When $b_O \leq 1/4$ in., the effective gasket seating width, $b = b_O = 0.235$ in.

When $b_O \leq 1/4$ in., diameter at location of gasket reaction, G

G = Mean diameter of gasket contact face
= (16.38 + 15.44) x 0.5
= 15.91 in.

Gasket seating Load, F_{SG}

$F_{SG} = \pi * b * G * y$
$= \pi * 0.235 * 15.91 * 200$
= 2,400 lb.

Therefore, gasket seating and internal pressure load acting on the plug,

$W_{\text{bolt load required}}$ = FSG + WOP
= 2,400 + 4,000
= 6,400

Design check: what is the total bolt load available on basis of UTS of the bolt material?
16 cap screws (1-8-UNC: UNBRAKO 1960) 1-in. are specified as closure plug bolts.

For UNBRAKO 1960 cap screw material, UTS = 180,000 psi; YS = 155,000 psi.

Bolt data - 1-in. nominal diameter
stress area per bolt = 0.551 in^2
root diameter = 0.838 in
UNC = coarse thread
8 threads per inch (8 tpi).

$W_{\text{bolt load available}}$ = no. of bolts x bolt area x allowable stress
= 16 x 0.551 x [UTS]
= 16 x 0.551 x 180,000
= 1,586,800 lb.

As $W_{\text{bolt load available}}$ (1,586,800 lb.) > $W_{\text{bolt load required}}$ (6,400 lb.), the closure plug bolting design is more than adequate to resist the forces on the closure plug due to internal pressure.

Now let us consider additional forces on the closure plug due to 132 g's on F-294 resulting from 30 ft. free drop test of F-294, in top end drop orientation.

$W_{G\text{-}LOAD}$ = Load due to G-load = W x G_{HACOT} = 1,135 lb. x 132 g's
= 149,900 lb.

where

$W = W_{PLUG} + W_{CONTENT}$ = 1,070 + 65 = 1,135 lb.

The total load on the closure plug required to maintain flanged, gasketed joint in HACOT is

$W_{\text{required closure plug,HACOT}} = F_{SG} + W_{OP} + W_{G\text{-}LOAD}$
$= 2{,}400 + 4{,}000 + 149{,}900$ lb.
$= 156{,}300$

As $W_{\text{bolt load available}}$ (1,586,800 lb.) > $W_{\text{required closure plug,HACOT}}$ (156,300 lb.), the closure plug bolting design is more than adequate to resist the forces on the closure plug due to internal pressure and G-loads on F-294 arising from hypothetical accident drop tests.

Safety factor (SF) $= W_{\text{bolt load available}} / W_{\text{required closure plug, HACOT}}$
$= 1{,}586{,}800$ lb./156,300
$= 10.15$

Margin of Safety = SF – 1 = 10.15 – 1 = 9.15

As the margin of safety (MS) is greater than zero, the bolted joint as specified shall be maintained during HACOT of F-294 package.

2.2 BOLT STRESS DUE TO APPLIED TORQUE PLUS G-LOAD (132 G's)

100 ft.-lb. per bolt nominal torque is specified in the preparation for shipment procedure (see Chapter 7, Section 7.1) for the closure plug bolts of the F-294 container.

UNBRAKO catalogue specifies for bolt (screw) material: 155,000 yield stress at room temperature.

The direct bolt stress, s, due to torque is

$s = F/A_r$
$= [T/(0.2 * D_{nom})]/A_r$
$= 100 \times 12/(0.2 * 1.0)/0.551$
$= 10{,}890$ psi

s_1, additional direct stress on the bolt due to G-load

s_1 = force per bolt/stress area of bolt
$= W_{G\text{-}LOAD}$ per bolt$/A_r$
$= 147{,}200/16 * 0.551$
$= 16{,}700$ psi

The shear stress, t, due to torque, T, is

$t = TD_r/J$
$= [TD_r/2]/[\pi D_r^4/32]$
$= 16T/[\pi * D_r^3]$
$= 16 * 100 * 12/[\pi * 0.838^3]$
$= 10{,}390$ psi

The principal stress is:

$s_{P1} = ((s + s_1)/2) + \sqrt{[((s + s_1)/2)^2) + t^2]}$
$= (10{,}890 + 16{,}700)/2) + \sqrt{[((10{,}890 + 16{,}700)/2)^2 + 10{,}390^2]}$
$= 13{,}800 + 17{,}260$
$= 31{,}060$ psi

The principal stress s_{P1} of 31,060 psi is less than 180,000 psi UTS or 155,000 yield stress for the bolt material.

Safety Factor (SF) = allowable stress/applied stress
$= 180{,}000/31{,}060$
$= 5.79$

Margin of Safety = SF −1 = 5.79 −1 = 4.79
where

F	=	Applied force per bolt, lb.
	=	T/CD
T	=	applied bolt torque ft.-lb. or in.-lb.
D	=	nominal bolt diameter, 1 inch
C	=	constant = 0.2 typical.
A_r	=	area at minor dia. of thread = 0.551 in.^{2}
D_r	=	root diameter of external thread = 0.838 in.
J	=	polar moment of inertia for round shaft (in^4)

2.3 SHEAR OF THREADS IN THE CONTAINER BOLT HOLES

Are the threads in the container flange bolt holes strong enough to resist operating pressure, gasket seating and G loads? See Figure 4.4.6-F3 for force free body diagram.

Bolt hole - Internal thread specification:

- 1-8-UNC-2B
- 1.0 in. thread length of bolt engagement (i.e., bolt hole 1.25 in. deep)
- 8 threads per inch (tpi)
- 0.9188 basic pitch diameter
- 0.125 pitch (p)

Socket Head screw specification:

- 1-8-UNC, 2.0 in. long.
- Effective thread length = 1.0 in.

Shear area per bolt hole, A_S

$A_S = \pi * D_{min} * h$

where

D_{min} = basic pitch diameter

h = effective length of threads in shear
= 0.5 * number of threads engaged by screw * pitch
= 0.5 * 8 tpi * 1.0 * 0.125
= 0.5 in.

$A_S = \pi * 0.9188 * 0.5$
$= 1.44$ in^2

$W_{required\ closure\ plug,\ HACOT} = F_{SG} + W_{OP} + W_{G\text{-}LOAD}$
= 2,400 + 4,000 + 149,900 lb.
= 156,300

τ, shear stress in the internal threads per bolt hole

$\tau = W_{required\ closure\ plug,\ HACOT}/[N*A_S]$
= 156,300/[16 * 1.44]
= 6,800 psi

For ss304L,
UTS = 70,000 psi. at 100°F;
Minimum YS = 25,000 at 100°F
YS = 17,800 at 330°F

Safety factor (SF) = allowable stress/applied stress
= 70,000/6,800
= 10.29

Margin of Safety = SF − 1 = 10.29− 1 = 9.29

As the shear stress of 6,800 psi is less than the stress of 17,800 psi (i.e., yield stress), the bolt hole thread will not yield.

As the Margin of Safety (MS) > 0, the bolt hole threads will not strip. Therefore, the bolt hole thread design meets the requirement that the closure plug shall remain fastened to the inner shell assembly of F-294.

2.4 CLOSURE PLUG BOLTING IN F-294 SIDE DROP ORIENTATION

See Figure 4.4.6-F4.
In the side drop, the plug is held to the container by 16 fasteners of 1 in. dia., UNBRAKO (i.e., 180,000 UTS). The clearance between the head of the fastener and the unthreaded hole in the closure plug flange is 0.0625 in. The clearance between the plug and the container upper cavity is 0.040 in. Therefore in the side impact, the cylindrical side of the plug body will impact prior to the start of shearing of plug/container bolts (bolt head from bolt shank). Normally, due to load sharing,

1. 500 lb., the weight of the ss flange of the plug assembly, is resisted by 16 bolts AND
2. 570 lb., the weight of the cylindrical section of the plug (i.e., lead shielding, etc.) is resisted by the 1/3rd cylindrical arc of the plug and the container upper cavity.

However, in this calculation it is assumed that

1. the entire weight of the closure plug (1,070 lb.) is resisted by 16 bolts and
2. the entire weight of the closure plug (1,070 lb.) is resisted by 1/3rd arc of the container upper cavity.

This is the conservative approach.
See Figure 4.4.6-F5: Shear Force Diagram.

2.4.1 Plug Bolts

The weight of the plug is W_{PLUG} = 1,070 lb. (Note: exclusive of the weight of contents 65 lb.)

In the side oblique drop orientation, the measured deceleration load is 136 g's. Therefore the impact load on the bolts is

$$P_{IMPACT} = W_{PLUG} \times \text{G-load}$$
$$= 1{,}070 \times 136$$
$$= 145{,}520 \text{ lb.}$$

Shear area per bolt, $A_{SHEAR\ AREA}$

$$A_{SHEAR\ AREA} = 0.551 \text{ in}^2$$

Shear stress per bolt, τ

$$\tau = P_{IMPACT}/[A_{SHEAR\ AREA} \times \text{number of bolts}]$$
$$= 145{,}520/[0.551 \times 16]$$
$$= 145{,}520/8.816$$
$$= 16{,}500 \text{ psi.}$$

In addition, each bolt is subjected to gasket seating load and operating load due to internal pressure of 20 psig. the tensile stress is s

$$s = (F_{SG} + W_{OP})/\text{total bolt area}$$
$$= 6{,}400/0.551 \times 16$$
$$= 726 \text{ psi}$$

The principal stress is:

$$s_{P1} = (s/2) + \sqrt{[(s/2^2) + \tau^2]}$$
$$= 726/2 + \sqrt{[(726/2)^2 + 16{,}500^2]}$$
$$= 363 + 15{,}780$$
$$= 16{,}500 \text{ psi}$$

The principal stress s_{P1} of 16,500 psi is less than 180,000 psi allowable stress for the bolt material (UTS of 180,000 psi).

Safety Factor (SF) = allowable stress/applied stress
= 180,000/16,500
= 10.9

Margin of Safety = SF – 1 = 10.9 – 1 = 9.9

2.4.2 Plug Cylindrical Body/Container Upper Cavity Tube

One-third (1/3) of the cylindrical plug body resists the side impact.

W_{PLUG} = 1,070 lb.

The impact load, at deceleration of 136 g's is

$P_{IMPACT} = W_{PLUG}$ X 136 g's
= 1070 x 136
= 145,520 lb.

Area resisting the impact A = 0.33 x π x 14.790 x 10
= 154.7 in^2

Bearing stress, $s_{bearing}$

$s_{bearing}$ = P_{IMPACT}/A
= 145,520/154.7
= 940 psi.

Safety Factor (SF)

SF = allowable stress/applied compressive stress
= UTS/$s_{bearing}$
= 70,000/940
= 70,000/940
= 74.4

Margin of Safety, $MS_{STRESS\text{-}BASED}$ = SF – 1 = 74.4 – 1 = 73.4

2.4.3 Plug Weld: WPC1

There are two (2) plug circumferential welds in the plug (WPC1 and WPC2).However, only one plug weld (WPC1) is conservatively assumed to resist the side impact (see Figure 4.4.6-F6).

Weld area = π x 12.710 x 0.38 x 0.707 = 10.72 in^2

As the weld is fully radiographed, therefore joint efficiency is 100%.

Therefore the impact load is

P_{IMPACT} = W_{PLUG} x Deceleration load
= 1,070 x 136 g's
= 145,520 lb.

Shear stress, τ = P_{IMPACT}/Weld area
= 145,520/10.72
= 13,600 psi

Safety Factor (SF)

SF = allowable stress/applied shear stress
= UTS/τ
= 70,000/τ
= 70,000/13,600
= 5.14

Margin of Safety, $MS_{STRESS\text{-}BASED}$ = SF – 1 = 5.14 – 1 = 4.14

3. STRESS ANALYSIS OF INNER SHELL ASSEMBLY COMPONENTS DUE TO BUILD-UP OF INTERNAL PRESSURE AND G-LOADS

3.1 INNER SHELL ASSEMBLY COMPONENTS - INTERNAL PRESSURE

3.1.1 Lower Cavity wall.

See Figure 4.4.6-F7

The hoop stress in the lower cavity tube, without taking lead restraint into account, is as follows:

s_{hoop} = p d/2t
= 20 x 12/2 x 0.5 = 240 psi.

where

p = 20 psig internal pressure
d = mean diameter of lower cavity tube = 12.0 in.
t = 0.500 in.

For ss304L at 600°F, UTS = 60,000 psi.

Safety Factor (SF) = allowable stress/applied stress
= UTS/s_{hoop}
= 60,000/240
= 250

Margin of Safety (MS) = SF – 1 = 250 – 1 = 249

3.1.2 Upper Cavity Wall

See Figure 4.4.6-F8.

The hoop stress in the upper cavity tube, without taking lead restraint into account, is as follows:

s_{hoop} = p d/2t
= 20 x 15.2/2 x 0.5 = 304 psi.

where

p = 20 psig internal pressure
d = mean diameter of upper cavity tube = 15.2. in.
t = 0.500 in.

For ss304L at 600°F, UTS = 60,000 psi.

Safety Factor (SF) = allowable stress/applied stress
= UTS/s_{hoop}
= 60,000/304
= 197

Margin of Safety (MS) = SF – 1 = 197 – 1 = 196

3.1.3 Lower Cavity End Cap

See Figure 4.4.6-F9

The bending stress in the lower cavity end cap, without taking restraint of lead into account, is as follows:

s_b = $cp/[t/d]^2$
= $0.2 \times 20/[0.75/11.5]^2$ = 940 psi.

where

c	= constant based on joint geometry
	= 0.2 based on ASME VIII, Division 1, Figure UG = 34 (i)
p	= internal pressure = 20 psig
t	= thickness of end cap = 0.75 in.
d	= internal diameter of the tube = 11.5 in.

For Hastelloy C-276, UTS, yield stress at 600 °F = 70,000 psi
(ASTM B-166)

Safety Factor (SF) = allowable stress/applied stress
= UTS/s_b
= 70,000/940
= 74.4

Margin of Safety (MS) = SF – 1 = 74.4 – 1 = 73.4

Therefore, the inner assembly under build-up of pressure of 20 psig has sufficient margin of safety that the structural integrity of the inner assembly shall not be compromised.

3.1.4 Upper Cavity Ring Flange

See Figure 4.4.6-F10.
The bending stress in the upper cavity ring flange, without taking restraint of lead into account, is as follows:
Case 77, Table X of Ref[4]

s_r = $b\,w\,a^2/t^2$
= $0.0195 \times 20 * 7.892^2/0.5^2$ psi.
= 97.5 psi

where

s_r	= maximum radial stress
b	= constant depending upon a/b = 7.892/6.25 = 1.262,
	= 0.0195
w	= p = applied pressure = 20 psi
t	= thickness of ring flange = 0.5 in.
a	= 7.892 in. ring flange outside radius
b	= 6.25 in. ring flange inside radius

For ss304L forging to ASTM A-182, the material properties are:

Ultimate Tensile Strength (UTS) = 60,000 psi. at 600°F

Safety Factor (SF) = allowable stress/applied stress
= UTS/98
= 60,000 psi/98 psi
= 615

Margin of Safety (MS) = SF – 1 = 615 – 1 = 614

Therefore, the ring flange is well below the static value of UTS.

3.2 INNER SHELL ASSEMBLY SUBJECT TO G-LOADS

3.2.1 Buckling of lower cavity tube

See Figure 4.4.6-F11
The weight of lead borne by the steel tube + cap,

$W_1 = \pi \times 6.25^2 \times 11.25 \times 0.41$

$W_1 = 566$ lb.

Assume the load is applied at the centre of the tube. Then, using Euler's formula, the collapse pressure load is

$P_C = \pi^2 EI/(L_e)^2$

where

$E = 28 \times 10^6$ psi

$L_e = 19.5$ in.

I = 2nd Moment of area = 340 in^4

$P_C = 247 \times 10^6$ lb.

Applied load $= 566$ lb. x $g_{ACCIDENT}$
$= 566 \text{ X } 132$
$= 74{,}712$ lb.

Safety Factor (SF) = Collapse load to buckle cavity tube/applied load
$= 247 \times 10^6/[74{,}712 \text{ lb.}]$
$= 247 \times 10^6/74{,}712$
$= 3{,}306$

Margin of Safety (MS) = SF – 1 = 3,306 – 1 = 3,305
It is concluded that the lower cavity tube will not buckle.

3.2.2 Bending of Lower Cavity End Plate

See Figure 4.4.6-F12

$G = 132$ g's

The applied pressure on the cavity tube cap is

p = weight of lead x 132 g's/A_{CAP}
$= 566 \times 132/[\pi \times 6.25^2]$
$= 74{,}712/122.73$
$= 609$ psi

Is a 0.75 in. thick Hastelloy C-276 cap strong enough to resist 609 psi maximum applied pressure?
For Hastelloy C-276, the material properties are:

Yield Stress (YS) = 30,000 psi.

Ultimate Tensile Strength (UTS) = 100,000 psi.

Using ASME VIII, Division 1 (Ref[17])

$s = cp/[t/d]^2$
$= 0.2 \times 609/[0.75/11.5]^2$
$= 28{,}636$ psi.

where

c	= constant depending on end tube to cap joint configuration
	= 0.2 (ASME VIII, Division 1, Figure UG-34(i).
p	= applied pressure = 609 psi
t	= thickness of cap = 0.75 in.
d	= 11.5 in. inside diameter

Safety Factor (SF) = allowable stress/applied stress
= UTS/28,630
= 70,000 psi/28,636 psi
= 2.4

Margin of Safety (MS) = SF – 1 = 2.4 – 1 = 1.4

The maximum stress in the end cap s = 28,636 psi shall be below the allowable stress of 70,000 psi UTS. Therefore, the end cap of the lower cavity of the inner shell assembly of F-294 shall not rupture.

3.2.3 Buckling of Upper Cavity Tube

See Figure 4.4.6.-F13. The weight of lead borne by the steel tube + flange,

$$W2 = \pi \times [7.892^2 - 6.25^2] \times 31.75 \times 0.41$$

$$W2 = 950 \text{ lb.}$$

$$W1 + W2 = 566 + 950 = 1{,}516 \text{ lb.}$$

The loads (W1 + W2) are applied at the centre of the tube. The line of action of deceleration force is in line with the centre of gravity of the upper tube. Therefore, this is the case of compressive stress leading to buckling. Then the collapse load, using Euler's formula is

$$P_C = \pi^2 EI/(L_e)^2$$

where

E	$= 28 \times 10^6$ psi
L_e	= 11 in.
I	= 2nd Moment of area $= \pi/4\,[7.89^4 - 7.39^4] = 701$ in^4
P_C	$= 1{,}601 \times 10^6$ lb.

HACOT G-levels:

top of plug G1	= 132 g's
Applied load	= 1,516 lb. x G-load
	= 1,516 x 132
	= 200,112 lb.

Safety Factor (SF) = collapse load to buckle cavity tube/applied load
$= 1{,}601 \times 10^6/[200{,}112 \text{ lb.}]$
= 8,000

Margin of Safety (MS) = SF – 1 = 8,000 – 1 = 7,999

It is concluded that the upper cavity tube shall not buckle.

3.2.4 Bending of Upper Cavity Ring Flange

See Figure 4.4.6-F14.

In this model, the cavity ring flange plate is under external applied pressure as a result of impact.
The applied pressure acts on the upper cavity tube & the cavity ring flange.

Based on 132 g's deceleration load, the applied pressure on the upper cavity ring flange is

p = weight of lead x 132 g's/$A_{RING\ FLANGE}$
= $950 \times 132/[\pi \times (7.892^2 - 6.25^2)]$
= 125,400/73
= 1,718 psi

Is a 0.5 in. thick stainless steel forging (ring flange) strong enough to resist 1,718 psi maximum applied pressure?
For stainless steel (ss304L)A-182, the material properties are:
Ultimate Tensile Stress (UTS) = 60,000 psi at 600°F.
Case 77, Table X of Ref[4]

s_r = $b\ w\ a^2/t^2$
s_r = $b\ p * [a^2/t^2]$ psi.
s_r = $0.0195 * 1{,}718 * 7.892^2/0.5^2$ psi.
= 8,345 psi

where

s_r = maximum radial stress
b = constant depending upon a/b = 7.892/6.25 = 1.262, b = 0.0195
w = p = applied pressure = 1,718 psi
t = thickness of ring flange = 0.5 in.
a = 7.892 in. ring flange outside radius
b = 6.25 in. ring flange inside radius

Safety Factor (SF) = allowable stress/applied stress
= UTS/7,330
= 60,000 psi/8,345 psi
= 7.18

Margin of Safety (MS) = SF – 1 = 7.18 – 1 = 6.18

The maximum stress in the upper cavity ring flange s_r = 8,345 psi is below the allowable stress of 60,000 psi (UTS at 600°F). Also, the maximum stress of 8,345 psi is below the yield stress 15,000 psi at 600°F. Therefore, the upper cavity ring flange will not yield.

3.2.5 Container Top Flange

See Figure 4.4.6-F15
In this model, the container top flange plate is under external applied pressure as a result of impact.
The applied pressure acts on the container top flange & the conical shell

Based on 132 g's deceleration load, the applied pressure on the container top flange is

p = weight of lead x 132 g's/$A_{RING\ FLANGE}$
= $(W_1 + W_2 + W_3) \times 132/[\pi \times (12.968^2 - 7.892^2)]$
= (566 + 950 + 5,314) * 132/356.7
= 6,830 x 132/356.7
= 2,527 psi

Is a 1.5 in. thick stainless steel plate (ring flange) strong enough to resist 2,220 psi maximum applied pressure?

For stainless steel (ss304L)A-240, the material properties are:
UTS = 60,000 psi at 600°F
Case 77, Table X, Roark (4th Edition)

s_r $= b\ w\ a^2/t^2$
$= b\ w\ a^2/t^2$
$= 0.03 \times 2{,}527 \times 10.5^2/1.5^2$
$= 3{,}710$ psi

where

s_r = maximum radial stress
a = 10.5 in. ring flange outside radius (see Figure 4.3.3-F16)
b = 7.892 in. ring flange inside radius
b = constant depending upon a/b = 10.5/7.892 = 1.33, b = 0.03
w = p = applied pressure = 2,527 psi
t = thickness of ring flange = 1.5 in.

Safety Factor (SF) = allowable stress/applied stress
= UTS/3,260
= 60,000 psi/3,710 psi
= 60,000/3,710
= 161

Margin of Safety (MS) = SF – 1 = 161 – 1 = 160

The maximum stress in the ring flange, s_r = 3,710 psi is below allowable YS of 17,800. Therefore, the ring flange shall not yield. Also, the maximum stress of 3,710 psi is less than UTS of 60,000 psi. Therefore, the ring flange will not rupture (crack).

4. SUMMARY

The following is a summary of the status of each of the components of the inner shell assembly and the closure plug under appropriate driving forces and resulting in appropriate safety factors.

4.1 CLOSURE PLUG BOLTED JOINT

For the closure plug bolted joint components, the applied stresses, safety factors and margins of safety are as follows:

A. Closure plug bolted joint subject to internal pressure and G-loads:

$W_{\text{required closure plug, HACOT}}$ = 156,300 lb.
$W_{\text{bolt load available}}$ = 1,586,000 lb.
Safety factor (SF) = 10.15

As the Safety Factor (SF) is greater than 1, the bolted joint as specified shall be maintained during HACOT of F-294 package.

B. Bolt Stress due to applied torque of 100 ft.-lb. per bolt + G-loads (132 g's)

Applied stress = 31,060 psi
Allowable stress = 180,000 psi
Safety factor (SF) = 5.79

Margin of Safety (MS) = SF – 1 = 5.79 – 1 = 4.79

C. Stress in the threads of the bolt hole

Applied stress = 6,800 psi
Allowable stress = 70,000 psi
Safety factor (SF) = 10.29

Margin of Safety (MS) = SF – 1 = 10.29 – 1 = 9.29

D. Closure plug: in side drop

1) Closure plug bolts:

Applied stress = 16,500 psi
Allowable stress = 180,000 psi
Safety Factor (SF) = 10.9

Margin of Safety (MS) = SF – 1 = 10.9 – 1 = 9.9

2) Plug cylindrical body: side impact:

Applied compressive stress = 940 psi
Allowable stress = 70,000 psi
Safety Factor, (SF) = 74.4

Margin of Safety (MS) = SF – 1 = 74.4 – 1 = 73.4

3 Plug weld WPC1:

Applied shear stress = 13,600 psi
Allowable stress = 70,000 psi
Safety Factor (SF) = 5.14

Margin of Safety (MS) = SF – 1 = 5.14 – 1 = 4.14

4.2 INNER SHELL ASSEMBLY

For inner shell assembly and components, the stresses or collapse loads, safety factors and margins of safety are given below. The safety factor (SF) is defined as: allowable stress/applied stress. For buckling, the safety factor is defined as critical load or collapse load/applied load. The allowable stress is ultimate tensile stress of material at appropriate temperature.

4.2.1 Summary of Stress Analysis of Inner Shell Assembly Components Under Internal Pressure

Various components identified in Figure 4.4.6-F2 were subjected to internal pressure of 20 psig and stress analyzed. The details of the calculations are given in Appendix 4.4.6. Summary is given here.

A. Lower Cavity wall due to build-up of internal pressure:

s_{hoop} = 240 psi; SF = 250; MS = 249

B. Upper Cavity wall due to build-up of internal pressure:

s_{hoop} = 304 psi; SF = 197; MS = 196

C. The bending stress in the lower cavity end cap, without taking restraint of lead into account, is

s_b = 940 psi; SF = 74.4; MS = 73.4

D. The bending stress in the upper cavity ring flange, without taking restraint of lead into account, is

s_r = 98 psi; SF = 615; MS = 614

4.2.2 Summary of Stress Analysis of Inner Shell Assembly Components under HACOT G-Loads

Various components identified in Figure 4.4.6-F2 were subjected to G-loads of 132 g's and stress analyzed. The details of the calculations are given in Appendix 4.4.6. Summary is given here.

A. Lower cavity tube buckling:
   Applied load = 74,710 lb.
   Collapse load = 247×10^6 lb.
   SF = 3,306; MS = 3,305

B. Lower cavity tube end cap:
   Bending stress = 28,636 psi.
   Allowable stress = 70,000 psi.
   SF = 2.4; MS = 1.4

C. Upper cavity tube buckling:
   Applied load = 200,100 lb.
   Collapse load = $1{,}601 \times 10^6$ lb.
   SF = 8,000; MS = 7,999

D. Bending of upper cavity ring flange
   Bending stress = 8,345 psi.
   Allowable stress = 70,000 psi.
   SF = 7.18; MS = 6.18

E. Container top ring flange
   Bending stress = 3,710 psi
   Allowable stress = 60,000 psi.
   SF = 161; MS = 160

As the safety factors (SF) > 1 and as the margin for safety (MS) > 0, the bolted closure joint over the inner shell assembly of F-294 shall be maintained to resist internal pressure and G-loads encountered in the F-294 containment system during the hypothetical accident conditions of transport of F-294.

The purpose of the closure plug bolted joint is to provide adequate shielding but not necessarily leaktightness of the joint between the closure plug and the inner shell assembly. It has been demonstrated that the closure plug is retained when F-294 is subject to combination of forces during hypothetical accident conditions of transport. Consequently the closure plug, which provides adequate shielding, shall be retained over the cavity of F-294, which houses Special Form cobalt-60 MDS Nordion C-188 sealed sources. As C-188 is certified Special Form RAM, the C-188 outer encapsulation provides the leaktightness for retention of radioactive material.

Therefore

1. as C-188 is certified Special Form RAM and provides leak tight containment AND
2. as the closure plug (the shielding) is retained over the inner shell assembly, which houses the cobalt-60 C-188 sealed sources,

F-294 shall meet the hypothetical accident conditions of transport (HACOT) containment system requirements (10 CFR 71.51 (a) (2)).

Figure 4.4.6-F1
F-294 Containment System: Inner Shell Assembly and Closure Plug

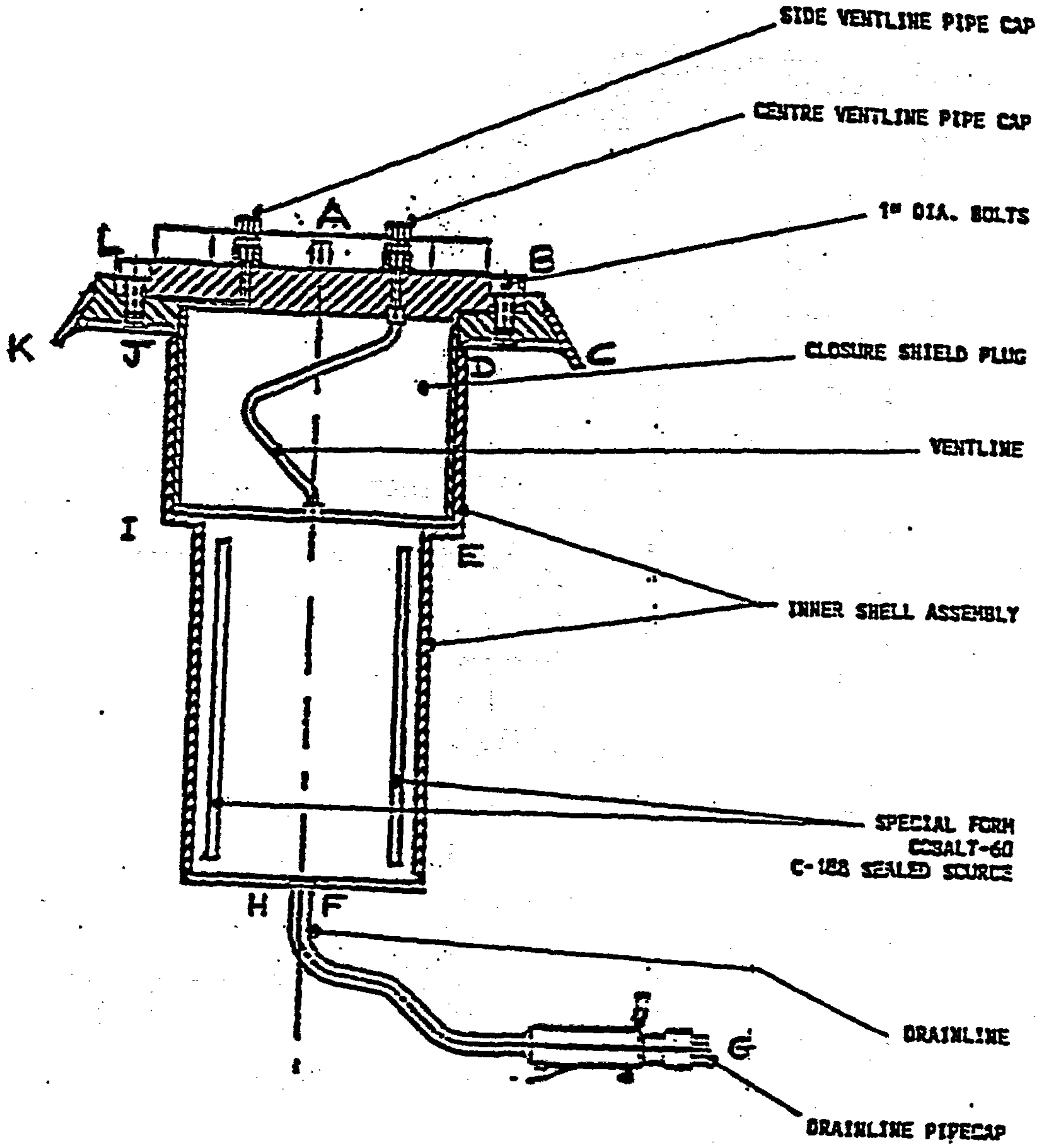


Figure 4.4.6-F2
Closure Plug Bolted Joint: Applied/Reactive Forces

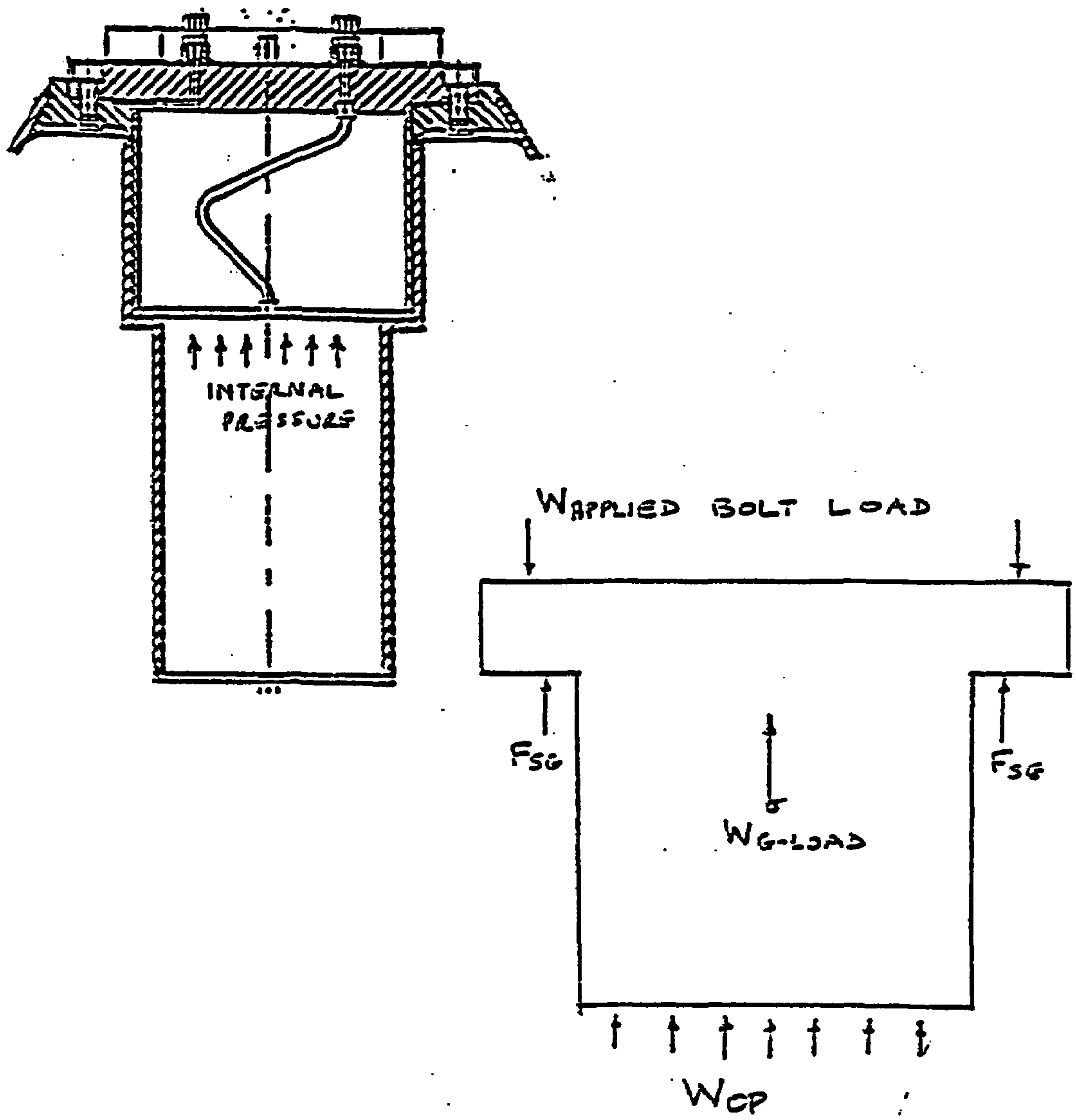


Figure 4.4.6-F3
Shear of the Threads of the Bolt Hole of Closure Plug Joint

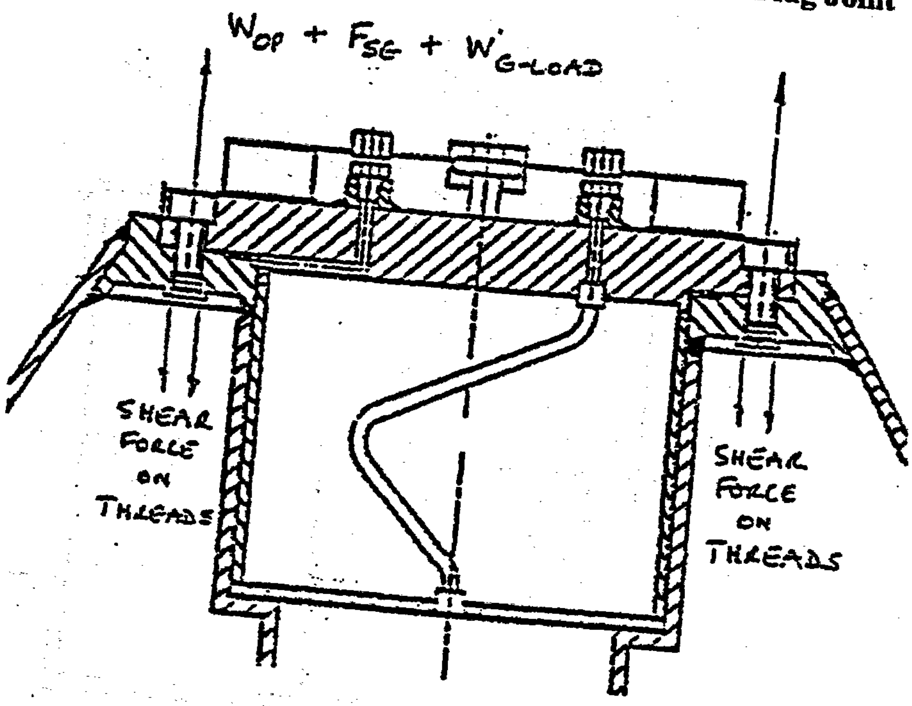


Figure 4.4.6-F4
Plug Cylindrical Body Impacting on Side: Plug Bolt Shearing

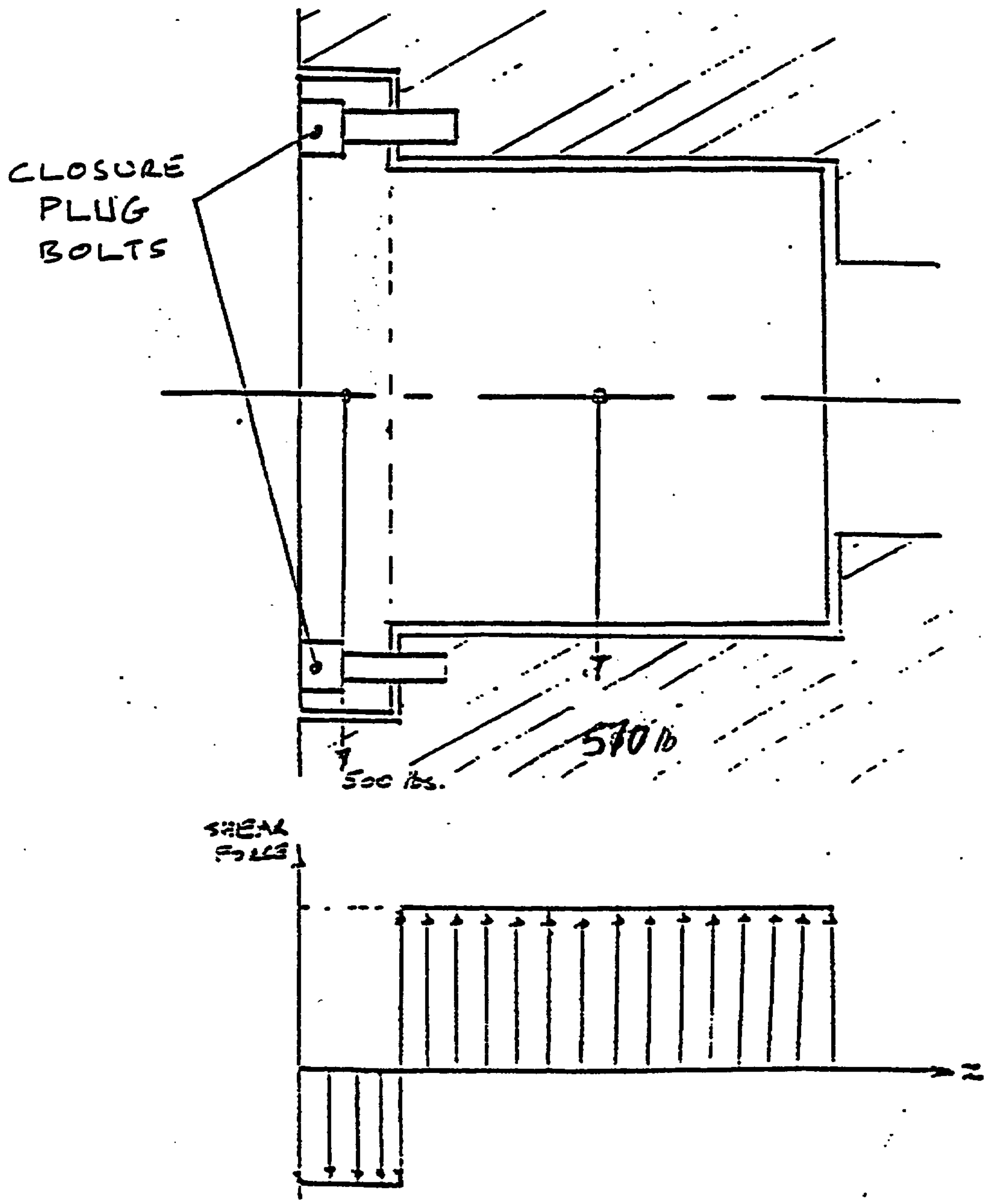


Figure 4.4.6-F5
Plug: Side Drop: Shear Force Diagram

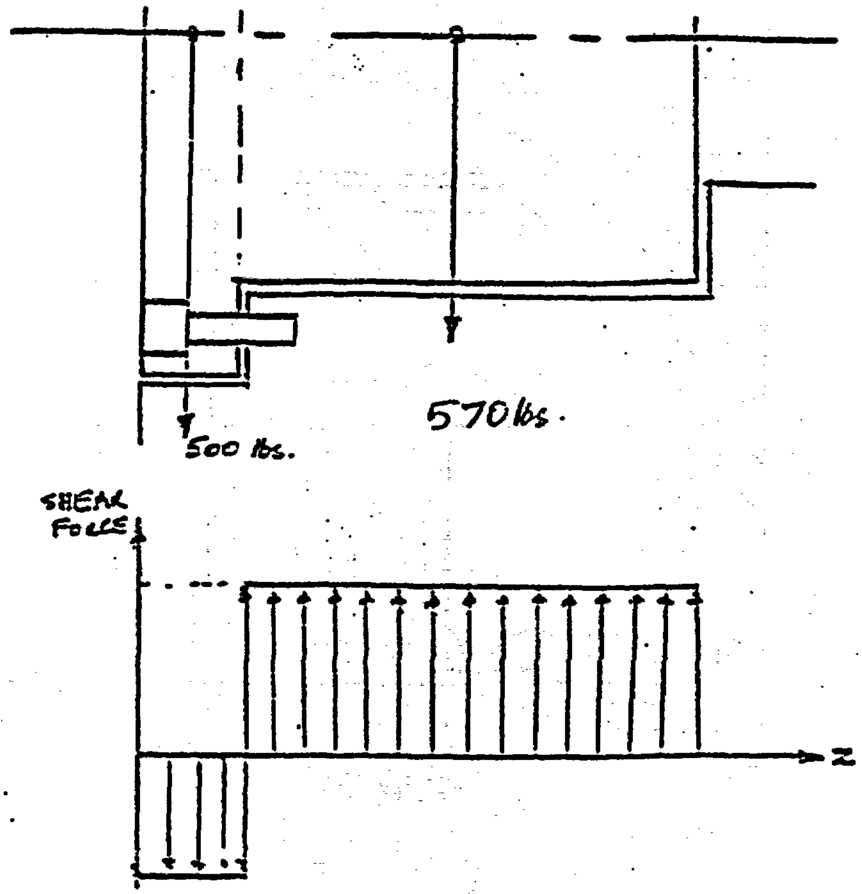


Figure 4.4.6-F6
Plug Cylindrical Body Impacting on Side: Plug Welds

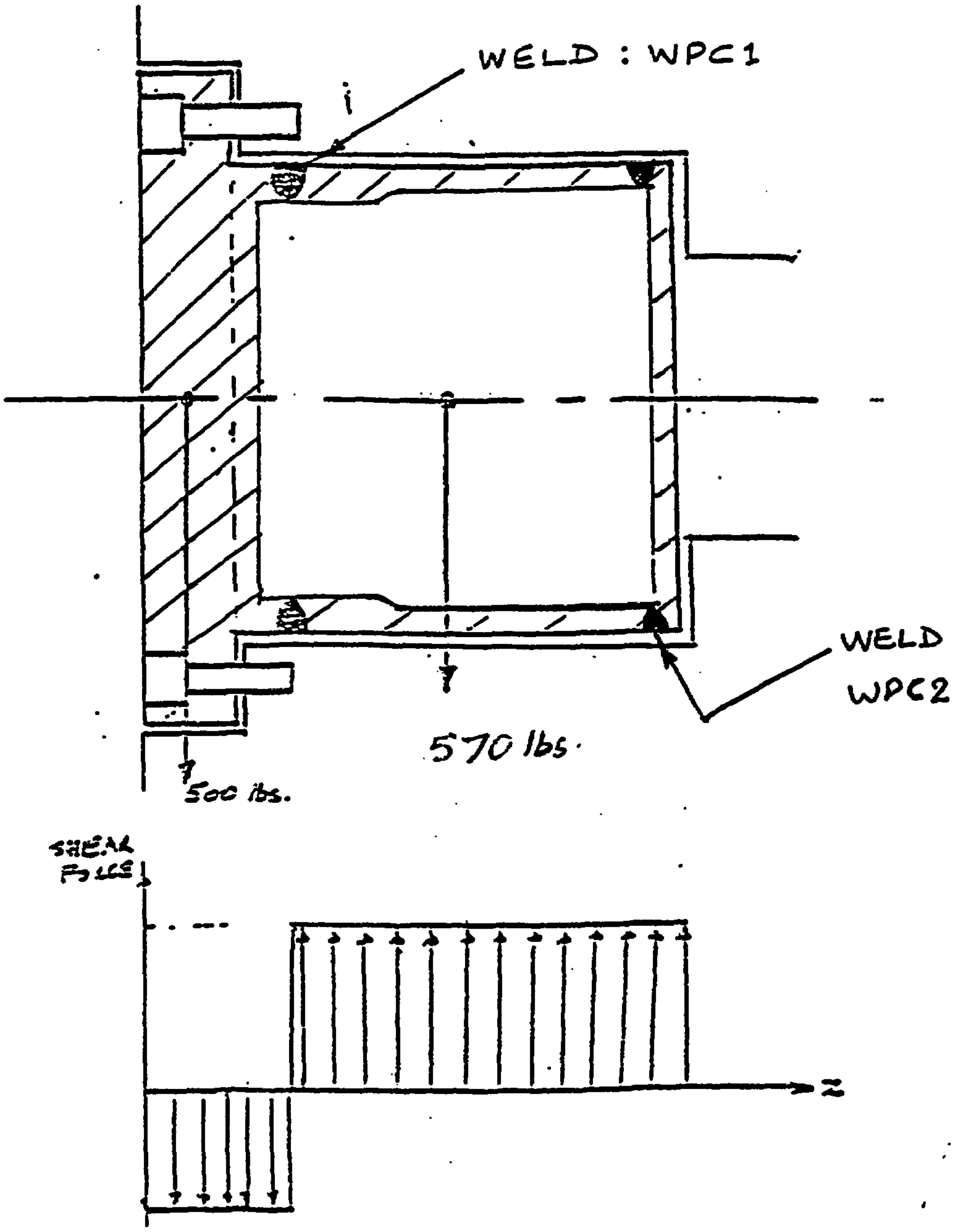


Figure 4.4.6-F7
Lower Cavity Tube

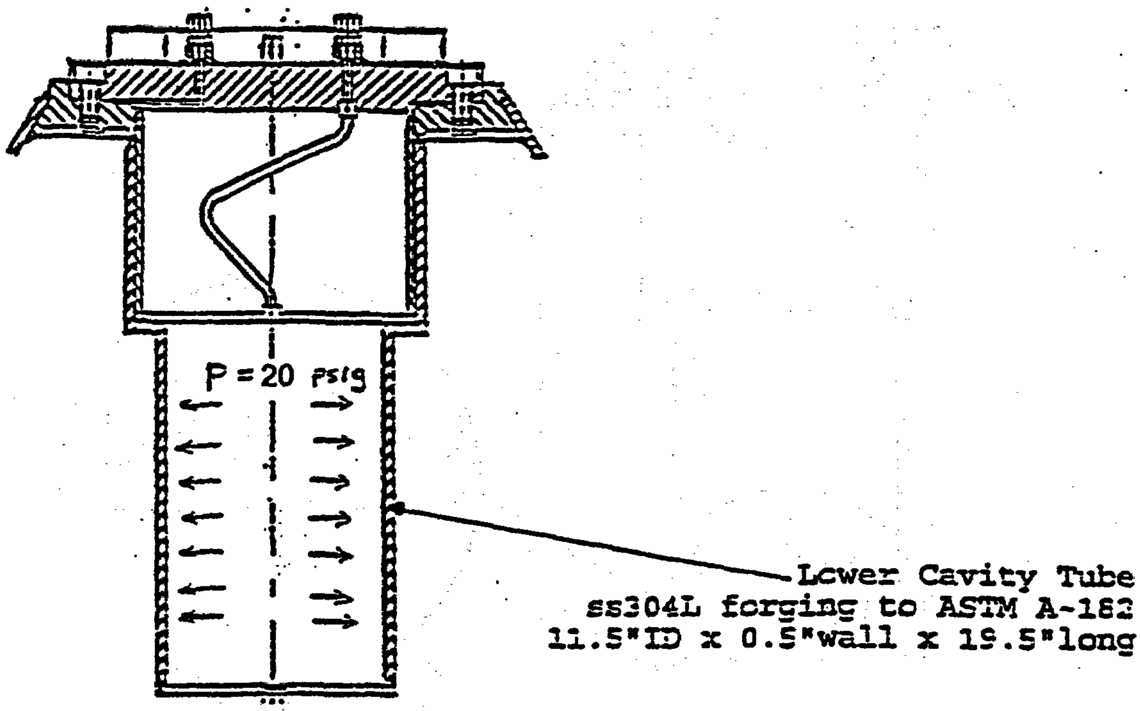


Figure 4.4.6-F8
Upper Cavity Tube

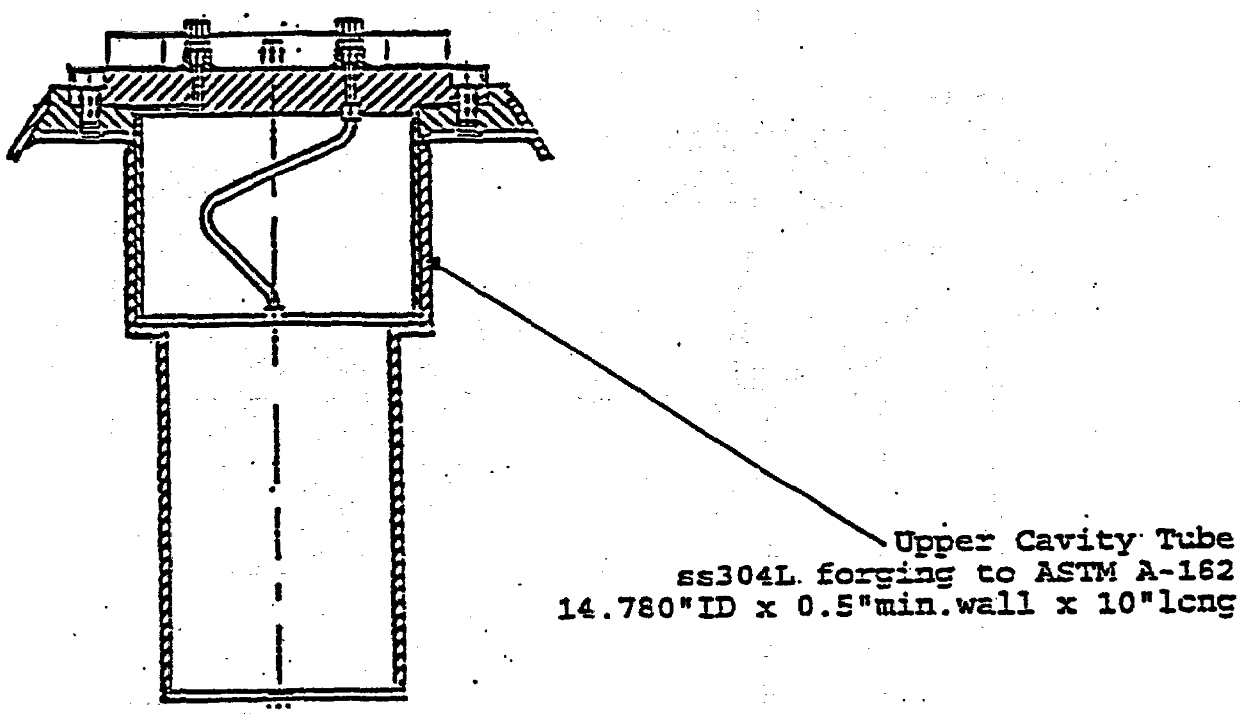


Figure 4.4.6-F9
Lower Cavity End Cap

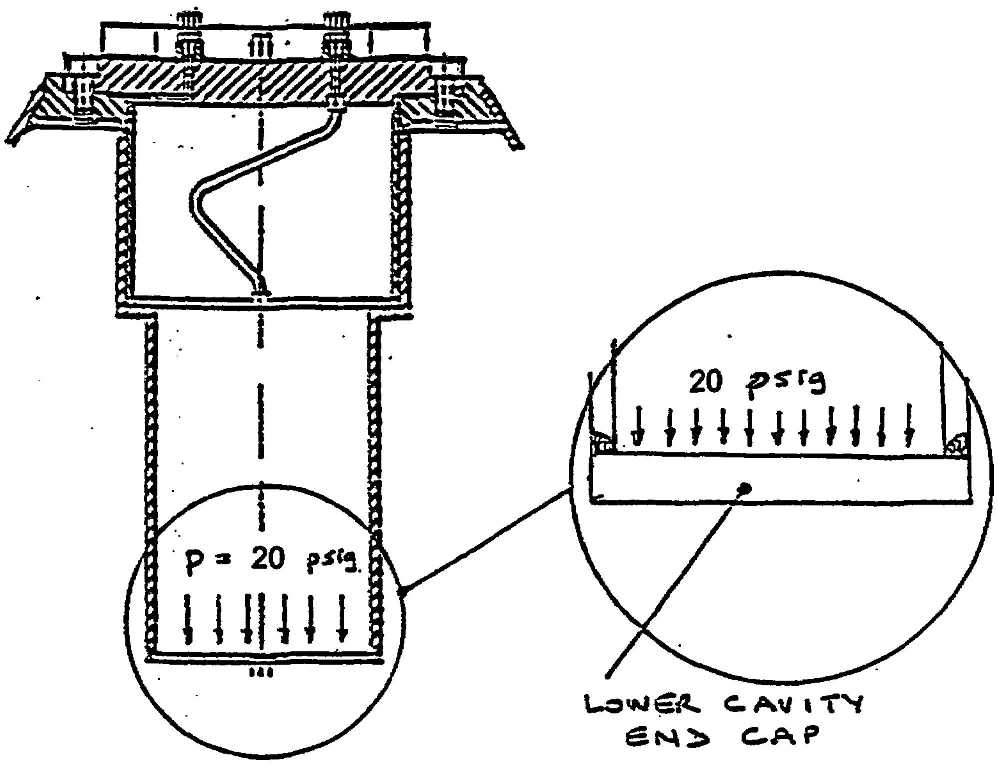


Figure 4.4.6-F10
Upper Cavity Ring Flange

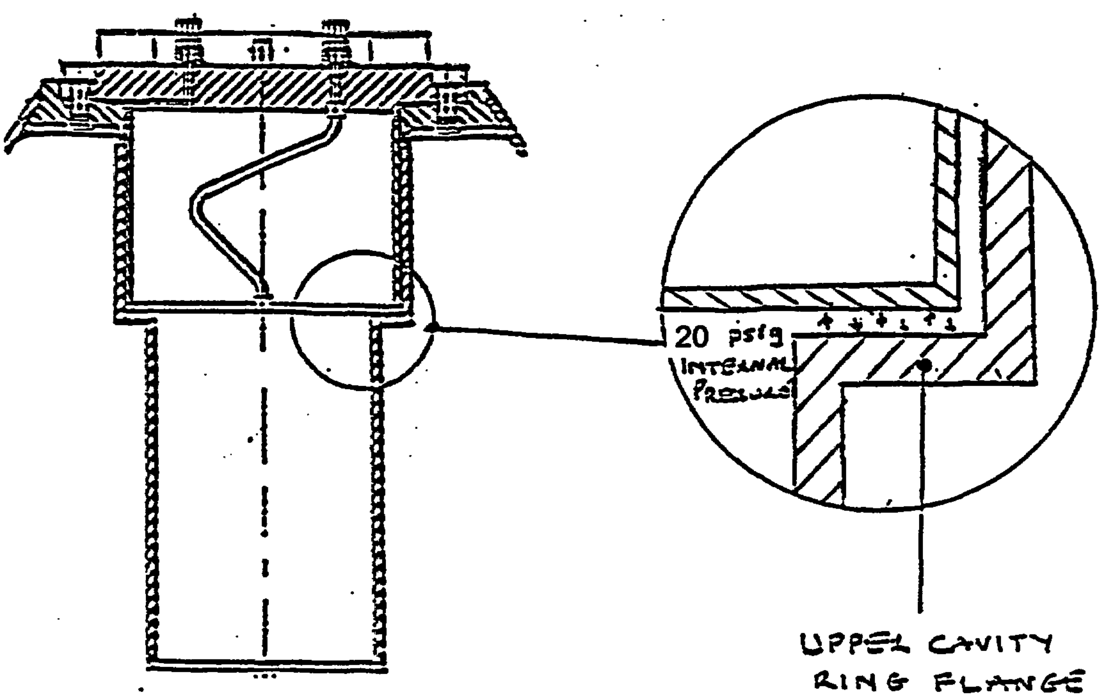


Figure 4.4.6-F11
Lower Cavity Tube Assembly under Axial Load

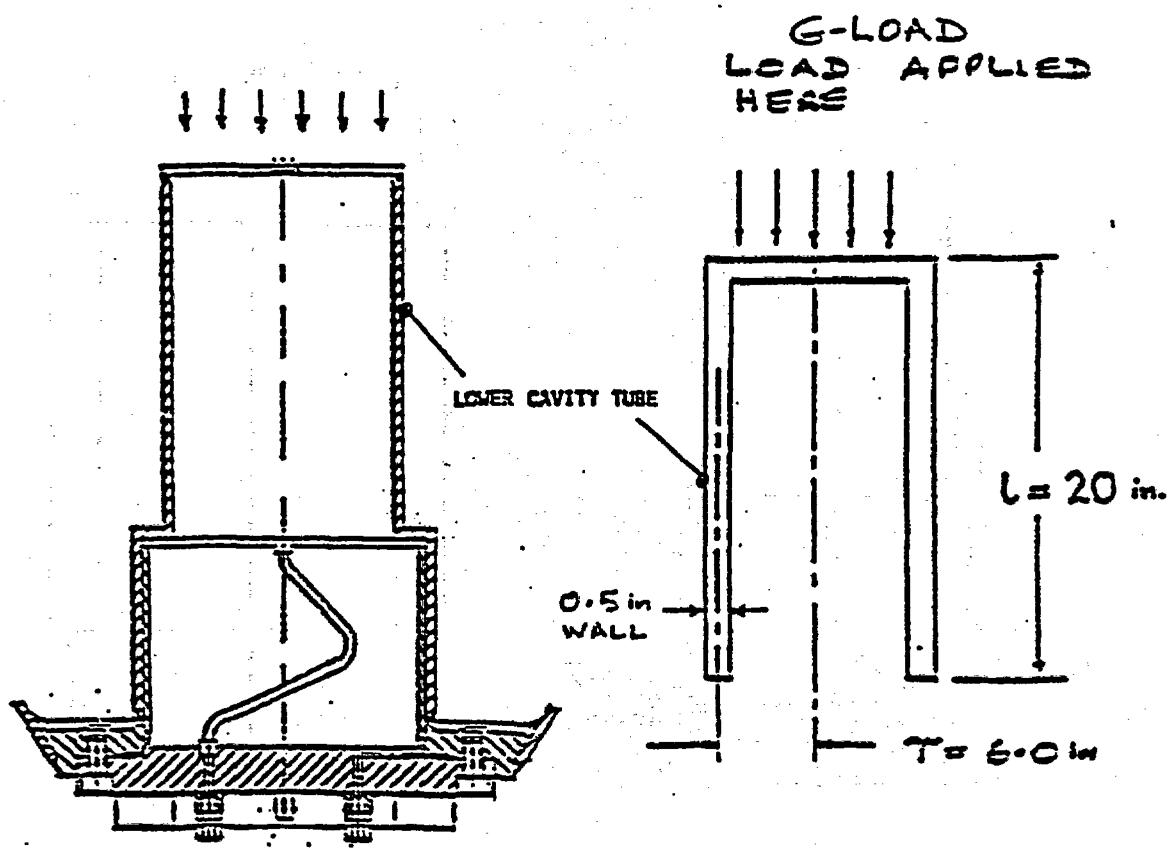


Figure 4.4.6-F12
Lower Cavity Tube End Cap

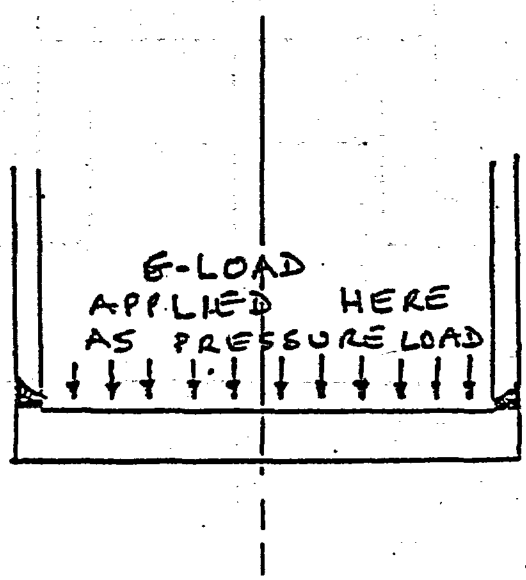


Figure 4.4.6-F13
Buckling of Upper Cavity Tube

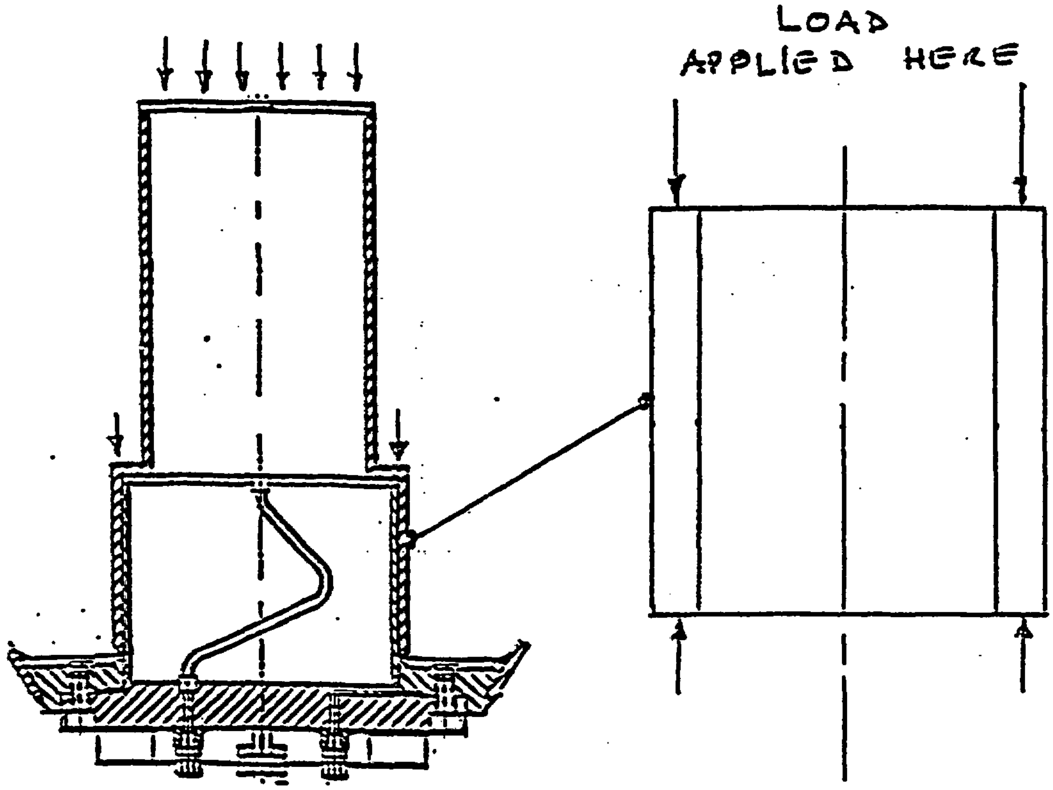


Figure 4.4.6-F14
Bending of Upper Cavity Ring Flange

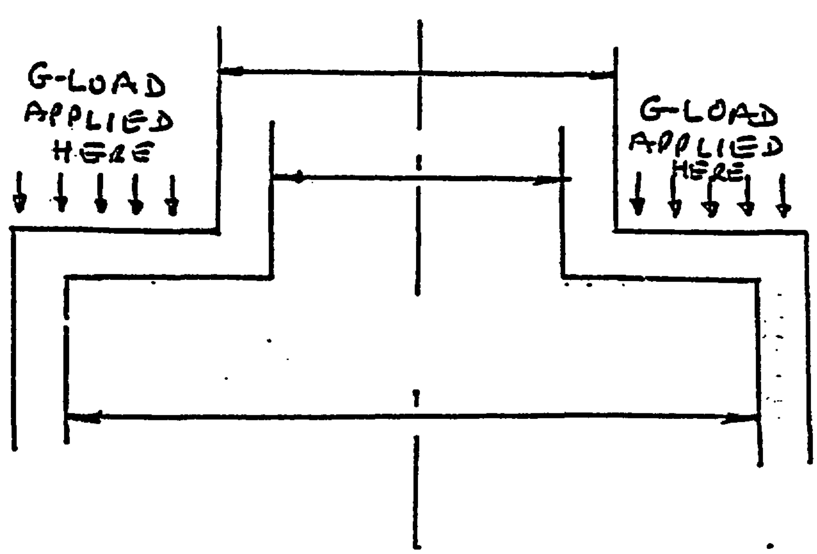


Figure 4.4.6-F15
Container Top Flange under Axial Load

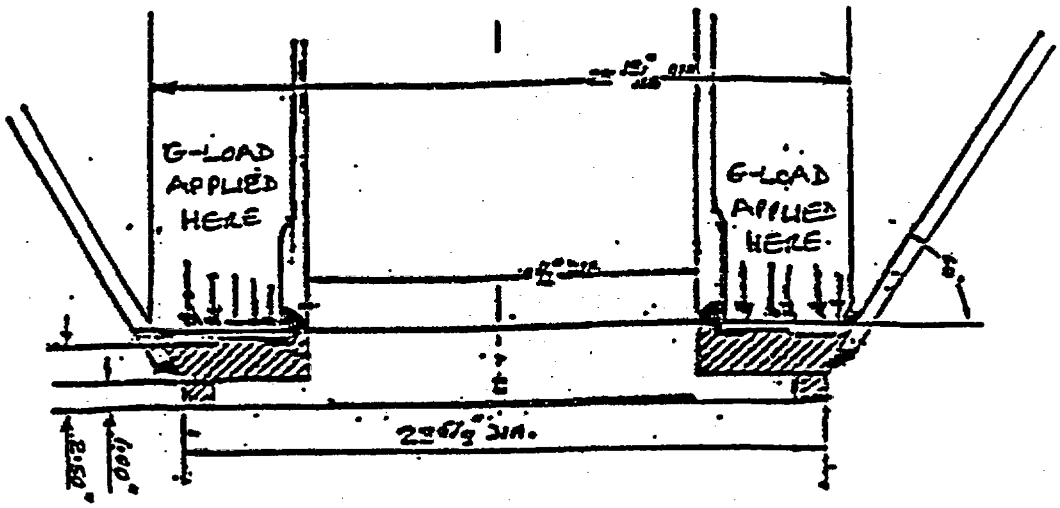
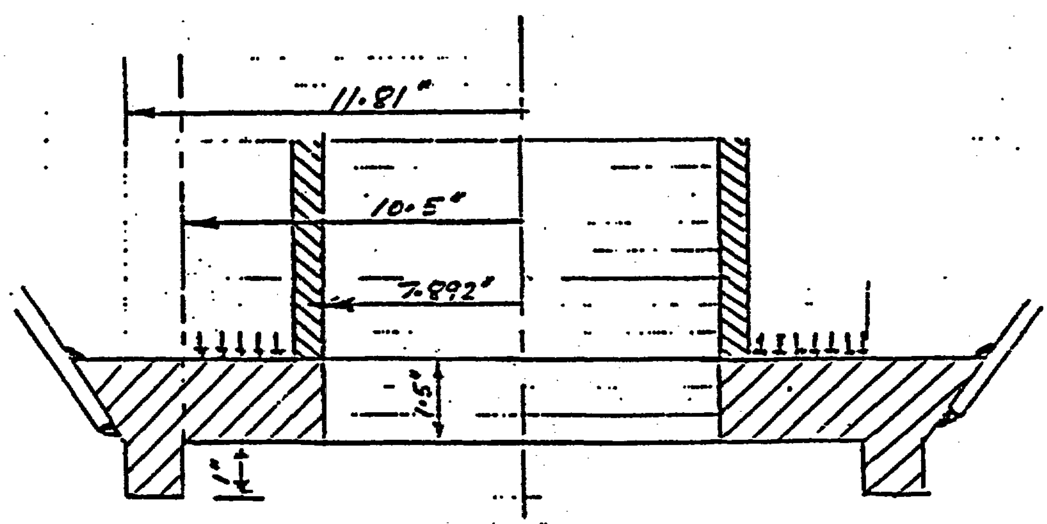


Figure 4.4.6-F16
Container Top Flange under Axial Load: Details



APPENDIX 4.4.7
F-294 PROTOTYPE CONTAINER TESTING: DROP TEST DATA RELEVANT TO THE CONTAINMENT SYSTEM

1. INTRODUCTION

On 25 February 1998, a single full-scale prototype F-294 container was subjected to a series of eight (8) drop tests as listed below:

Test #1:	Normal Free Drop Test: top end drop orientation
Test #2:	30 ft. Free Drop: side oblique drop orientation
Test #3C:	Puncture Test: impact on the zone near lift lug fin #4
Test #4:	Puncture Test: impact cylindrical fireshield
Test #5:	Puncture Test: impact on fixed skid lower plate
Test #6:	30 ft. Free Drop Test: top end drop orientation.
Test #7:	Puncture Test: impact on the crush shield upper plate
Test #8:	Puncture Test: impact cylindrical fireshield (nameplate zone)

The full details of the drop test and results are given in Chapter 2, Appendix 2.10.12. In this Appendix, the key findings of the drop test program, applicable to the containment system, are re-captured.

2. INTEGRITY OF THE CAVITY OF F-294

See Figure 4.4.7-F1

Prior to the drop test, the cavity of F-294 was air pressure tested at internal pressure of 45 psig. at 20°C for a period of two hours. No loss of pressure was observed. Subsequently the cavity was subjected to helium leak test. The cavity was leak tight to 4. x 10^{-7} atm cc/sec.

The F-294 was then subjected to a series of eight (8) drop tests.

After the drop tests, before opening the closure plug, the container was subjected to air pressure test and helium leak test. To conduct these tests, the drainline cap was fastened and torqued to 50 ft.-lb. torque. All other torques were not disturbed. The cavity was pressure tested at 45 psig at 20°C for a period of two hours; there was no loss of pressure. Subsequently the cavity was pressurized to 15 psig and subjected to sniffer helium leak test. The cavity was leaktight to 1 x 10^{-6} atm cc/sec.

Therefore, it is concluded that the integrity of the F-294 cavity is sound before the drop tests and after the drop tests.

3. INTEGRITY OF THE CLOSURE PLUG

3.1 FASTENERS OF THE CONTAINMENT SYSTEM

The bolt torques:

<u>Prior to the drop tests</u>, the torques on the fasteners of the containment system were as follows:

1-	Closure Plug Bolts:	100 ft.-lb. ± 10%.
2	Vent #1:	open, to permit accelerometer cables to go through.
3.	Vent #2:	20 ft.-lb. ± 10%.
4.	Drainline Cap:	open, to permit accelerometer cables to go through.

<u>After the drop tests</u>, the torques on the fasteners of the containment system were as follows:

1.	Closure Plug Bolts:	opening torques of the bolts are recorded in Table 4.4.7-T1. The bolt numbering is as per Figure 4.4.7-F2.
2.	Vent #1:	open, to permit accelerometer cables to go through.
3.	Vent #2:	20 ft.-lb. ±- 10%
4.	Drainline Cap:	open, to permit accelerometer cables to go through.

3.2 DAMAGE TO THE CLOSURE PLUG

1. After the drop tests, the closure plug bolts were not damaged.
2. After the drop tests, 15 of 16 closure plug bolts were not loose. One (1) closure plug of sixteen (16) appeared to be loose.
3. Before the drop tests, closure plug lifting lug height was = 2.5 in. After the drop tests, the closure plug lifting lug height is = 1.875 in. In other words, the lift lug of the closure plug was "compressed" by 0.625 in. (i.e., 2.5 – 1.875). This damage was primarily due to three drop tests:

   Test #1: Normal free drop test: top end orientation

   Test #6: 30 ft. free drop test: top end orientation

   Test #7: Puncture pin (top): impact on crush shield upper plate.

   The puncture pin impact was arrested by the lift lug of the closure plug, resulting in compression of 5/8 in. of the lift lug of the closure plug. Despite this puncture pin impact on the closure plug, the closure plug was not significantly damaged, with the exception of one specific location (i.e., lift lug of closure plug).

   After the drop tests, the bolt torques of the closure plug were of sufficient magnitude such that the F-294 cavity passed 45 psig air pressure test and also 15 psig helium leak test. Therefore, the integrity of the F-294 closure plug is sound.

Table 4.4.7-T1
Post-Drop: Opening Torques for F-294 Closure Plug

Bolt No.	Opening Torque (ft.-lb.)
#1	30.
#2	70
#3	90
#4	90
#5	20.
#6	30
#7	5
#8	30
#9	40
#10	0
#11	140+
#12	140+ ---- 220 - binding
#13	120 ------binding
#14	140+ ---- 180
#15	140+ ----- 220
#16	140 + —210 (2nd torque wrench)

4. THE INTEGRITY OF DUMMY C-188S

Prior to the drop tests of F-294, eight (8) dummy full-scale C-188s were helium leak tested.

The same eight dummy C-188s were located in a F-313 carrier and installed in the cavity of F-294. See Figure 4.4.7-F3. The F-313 carrier was buffered between the top of the carrier and the bottom of the plug to protect the accelerometer cables during the drop tests. The C-188s were not restrained and were free to bounce up or down and sideways. However, the carrier was restrained.

After the drop test of F-294, the dummy C-188s were helium leak tested. They were leaktight to a level of 1×10^{-9} atm cc/sec.

Therefore, it is concluded that the integrity of C-188s is sound.

5. CONCLUSIONS

The containment system of F-294 is defined as the inner assembly of F-294 (from closure plug to bottom of the cavity), inclusive of C-188 cobalt-60 sealed sources. It has been demonstrated by tests that:

1. the integrity of C-188s is sound
2. the integrity of the cavity of F-294 is sound (i.e., no cracks, etc.)
3. the integrity of the closure plug is sound, as it did not come loose after eight (8) drop tests.

Given that the F-457 source carrier, loaded with 80 C-188 capsules, only weighs 23 lb. more than the fully loaded F-313, test results are expected to be the same for the F-294 with the F-457 source carrier as they are for the F-294 with the F-313 source carrier.

Figure 4.4.7-F1
The Cavity of F-294

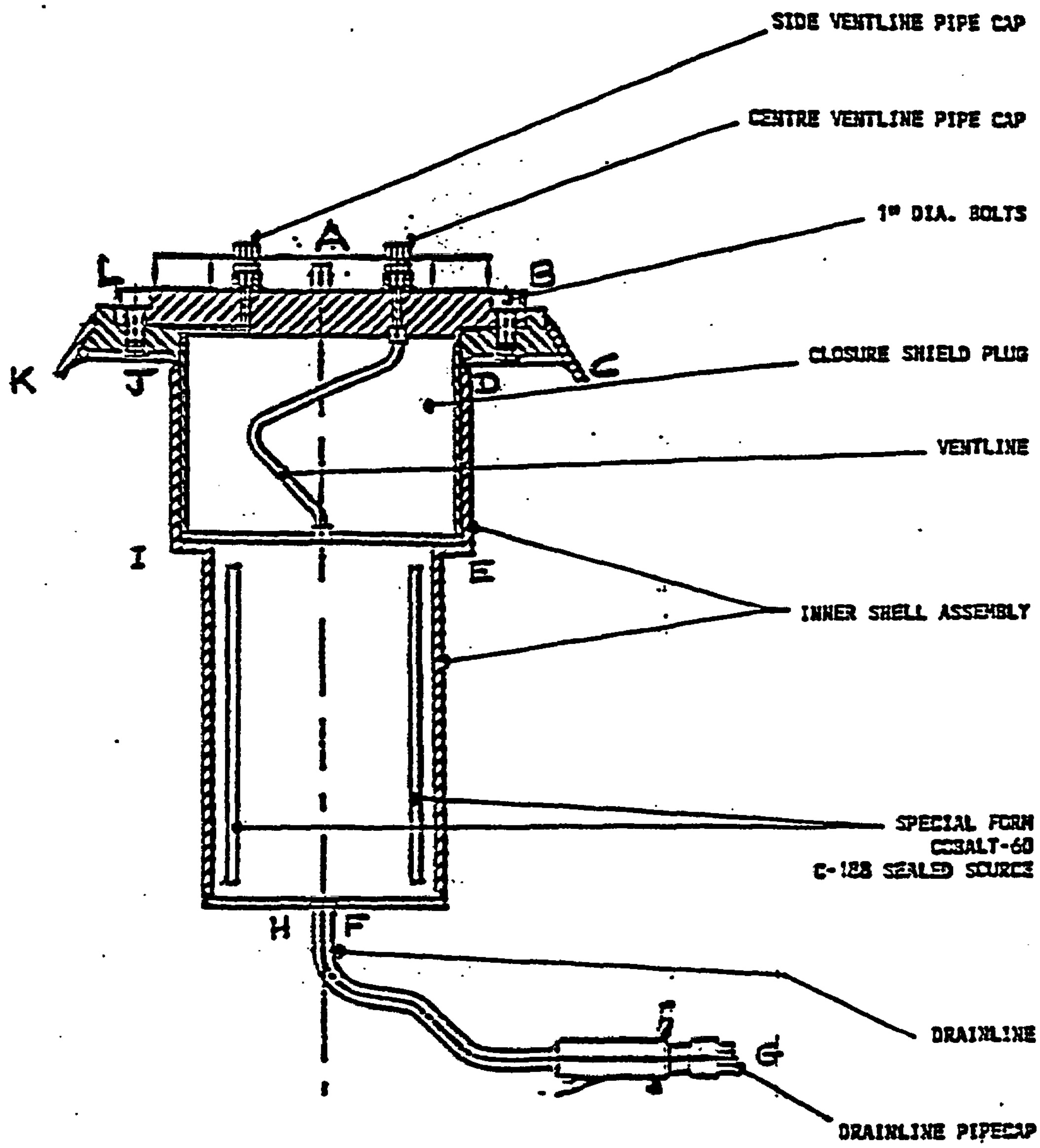


Figure 4.4.7-F2
The Numbering of Closure Plug Bolts

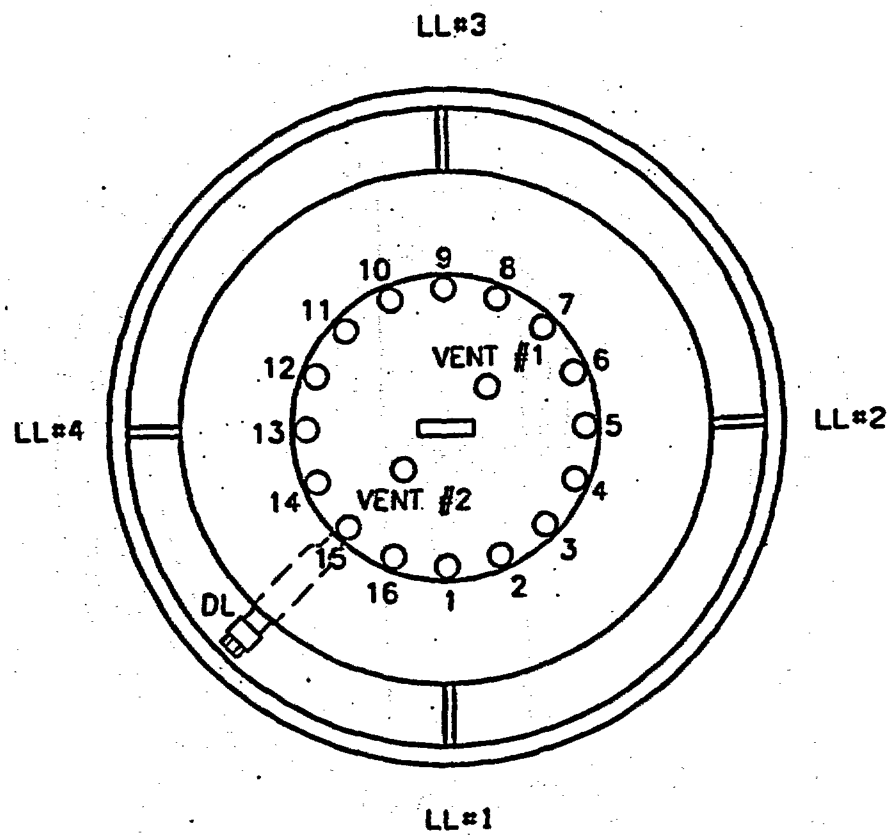
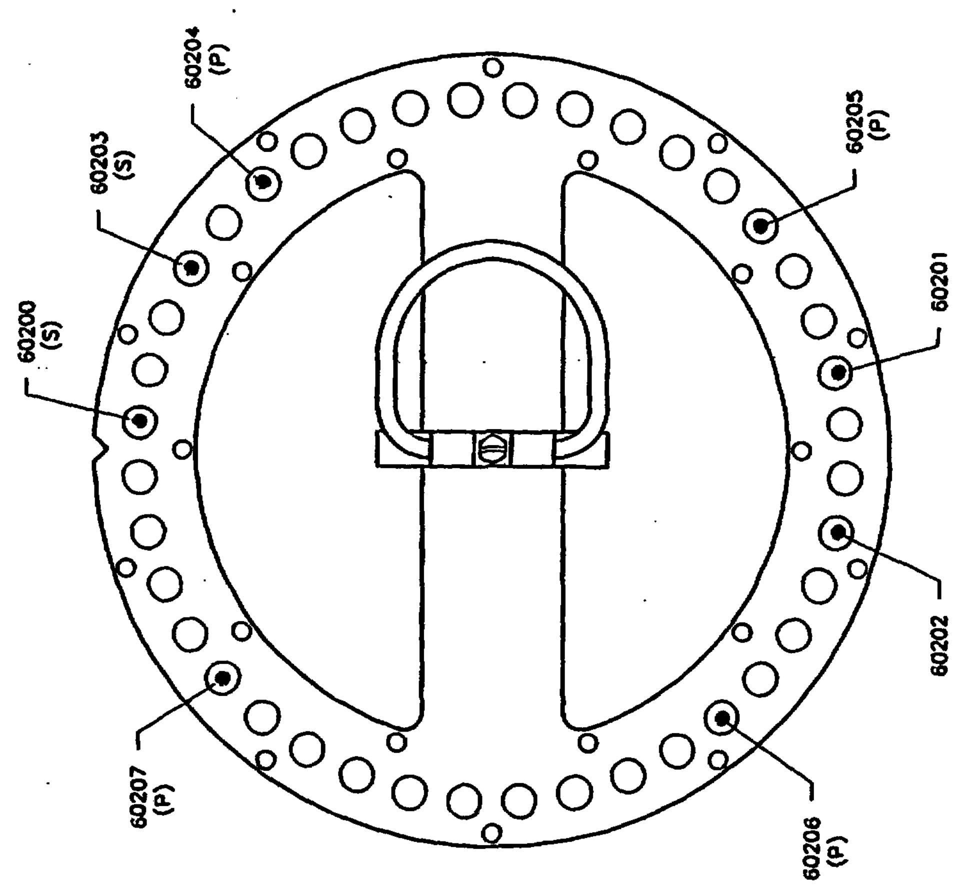
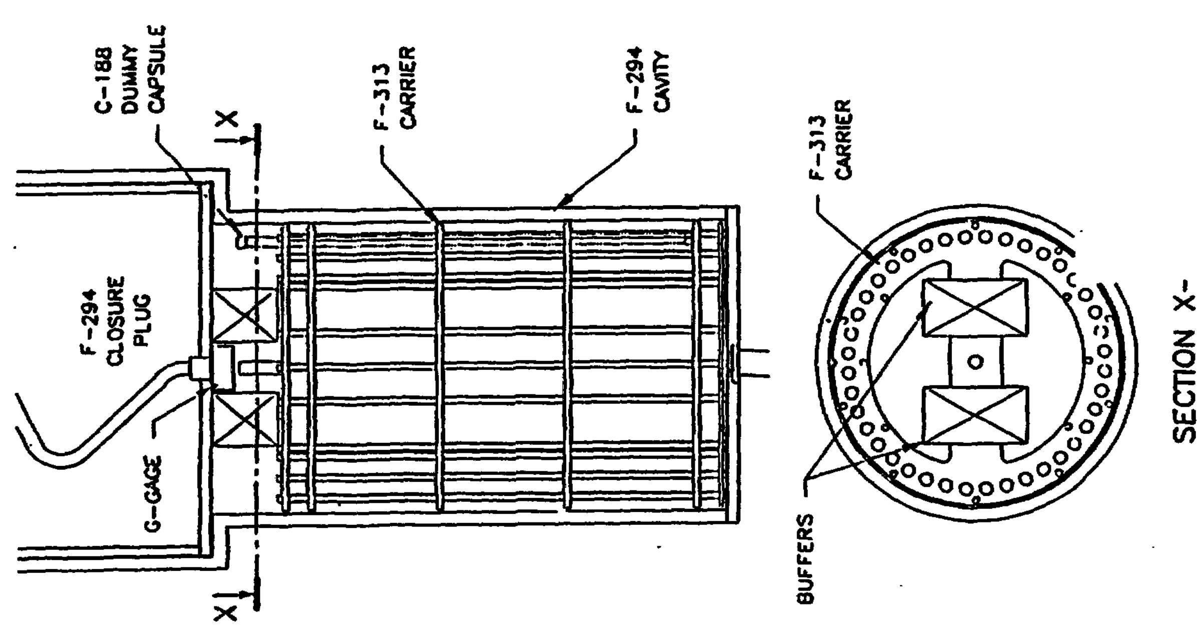


Figure 4.4.7-F3
Location of Dummy C-188s in F-313 Carrier during F-294 Drop Tests





CHAPTER 5 – SHIELDING EVALUATION

5.1 DISCUSSION AND RESULTS

The F-294 package has been designed for 360 kCi ^{60}Co. A vertical cross-section of the F-294 package assembly is illustrated in Fig. 5.1-F1. There are two main shielding components of the F-294 flask: steel-encased lead (11.25 in. [286 mm] Pb minimum typically) is used to reduce dose equivalent rates due to radiation propagating through a solid shield (i.e., no ducts) to acceptable levels. The outer cylindrical jacket is a 0.5 in. (12.7 mm) thick stainless steel plate; the second main component is the plug which is used to reduce dose equivalent rates above the unit. The plug consists of 11.0 in. (279 mm) Pb minimum typically.

This shipping configuration is designed to reduce dose equivalent rates at any point on the external surface to less than 200 mrem/h (2 mSv/h) and the TI to less than 10.

Radiation fields are summarized in Tables 5.1-T1 and 5.1-T2 for the F-294 prototype flask with 375,510 Ci as of January 06, 1998, in the form of 40 C-188 capsules in an F-313 source carrier. This source was used for F-294 external radiation measurements. The surface measurements (mR/h) and at 1m have not been normalized to 360,000 Ci ^{60}Co, and therefore represent a conservative estimate of the external radiation fields. All measurements were performed using two radiation survey meters: Berthold Rato/F G-M Tube, and a Nuclear Enterprises PDM-1 Ion Chamber. The location of the field point measurements are given in Figure 5.1-F2. The full-scale F-294 package was subjected to eight drop tests. The details of all the measurements are given in Appendix 5.1-1 (Pre-drop) and Appendix 5.1-2 (Post-drop). The source activity for each date of radiation survey measurement is given in Table 5.1-T1. Tables 5.1-T2 and 5.1-T3 summarize the Pre-drop and Post-drop radiation survey using the Berthold Rato/F survey meter.

Table 5.1-T1
Source Activity for Various Radiation Survey Dates

Radiation Measurement	Source Activity (Ci)	on Date
Initial Source Activity	375,510	1998 Jan. 06
Pre-drop Test Survey 1 (with Crush Shield and Fireshield in place)	375,360	1998 Jan. 07
Pre-drop Test Survey 2 (without Crush Shield and Fireshield)	374,684	1998 Jan. 12
Post Drop Test Survey 1 (with Crush Shield and Fireshield in place)	365,221	1998 Mar. 24
Post Drop Test Survey 2 (without Crush Shield and Fireshield)	364,958	1998 Mar. 26

Table 5.1-T2
Radiation Fields for the F-294 Package Before the Drop (1998 January 07) and After the Drop (1998 March 24)

Field Point	Surface Measurement (mR/h) With Crush Shield & Fireshield		Field at 1 m from Surface (mR/h) With Crush Shield & Fireshield	
	Before Drop	After Drop	Before Drop	After Drop
1	2.2	4.0	0.4	0.8
2	2.0	2.6	0.4	0.8
3	1.0	1.6	0.4	0.8
4	2.8	5.0	1.0	1.4
5	12	16	1.8	1.8
6	6.5	5.5	1.2	1.6
7	0.3	0.6	0.8	1.0
8	14	30	1.4	1.8

Table 5.1-T3
Radiation Fields for the F-294 container Before the Drop (1998 January 07) and After the Drop (1998 March 24)

Field Point	Surface Measurement (mR/h) Without Crush Shield & Fireshield		Field at 1 m from Surface (mR/h) Without Crush Shield & Fireshield	
	Before Drop	After Drop	Before Drop	After Drop
1	6.5	9.0	0.4	1.2
2	9.0	14	0.4	1.4
3	10.0	22	0.4	1.6
4	5.5	8.5	1.0	1.4
5	24	26	1.8	3.5
6	0.6	0.7	1.2	3.0
7	0.5	0.7	0.8	1.2
8	14	30	1.4	1.6

Figure 5.1-F1
Vertical Cross-Section of the F-294 Package Assembly

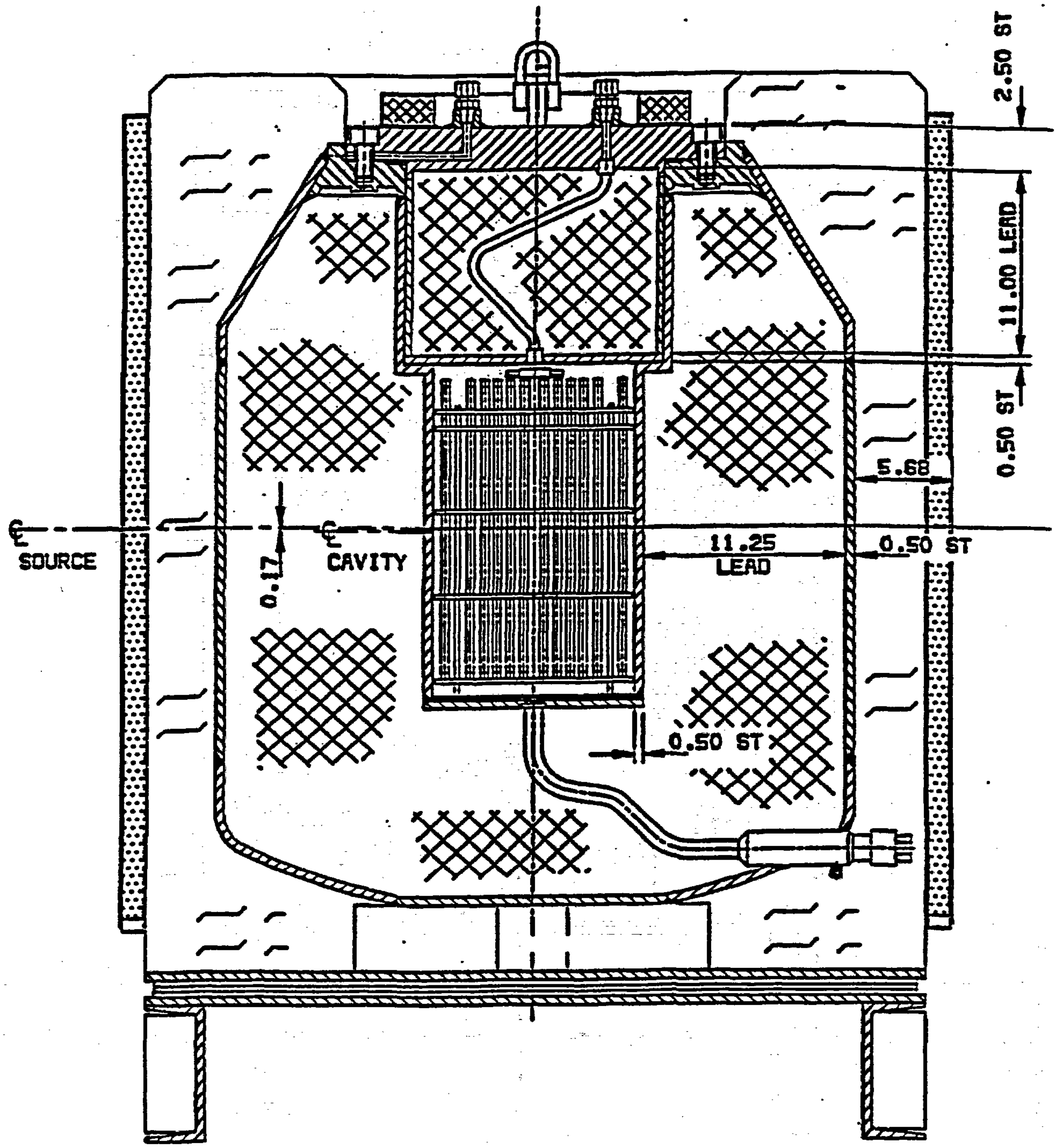
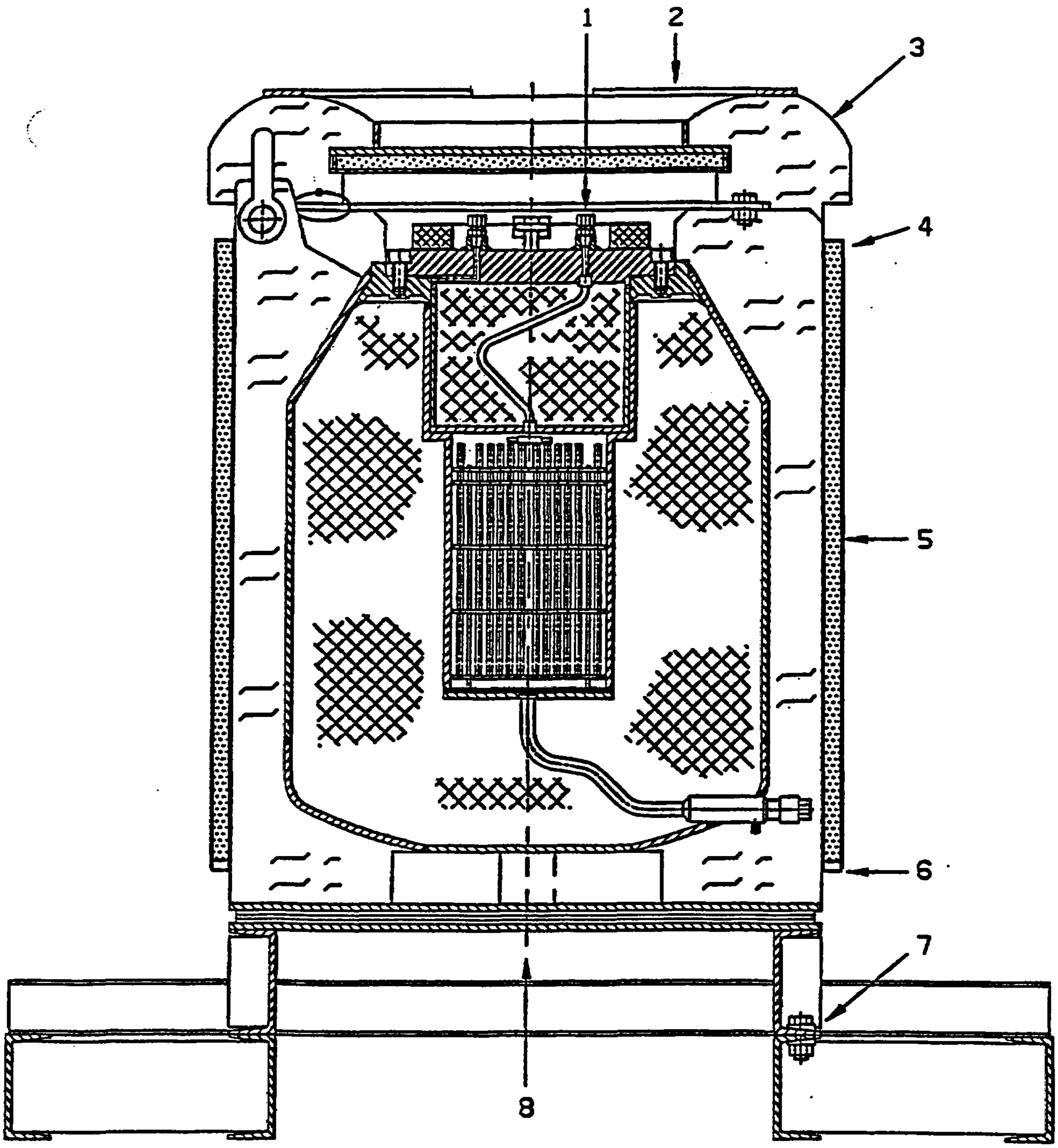


Figure 5.1-F2
Radiation Survey Field Points



5.2 SOURCE SPECIFICATION

The capacity of the F-294 is 360 kCi ^{60}Co. The number of nuclear transformations (disintegrations) per second corresponding to 360 kCi is:

$$360{,}000 \text{ Ci} \times 3.7 \times 10^{10} \text{ disintegrations/s/Ci} = 1.332 \times 10^{16} \text{ dis/s}$$

The ^{60}Co decay scheme is shown in Figure 5.2-F1 (see Ref. [1]). The average energy of the most probable beta particle (99.9%) is approximately 96 keV, and therefore will not escape the double layers of encapsulation and the source material; it will not be considered further since it will not contribute to external radiation fields.

Using the 99.9% and 100% probabilities for the 1.173 MeV and 1.332 MeV gamma transitions respectively, Table 5.2-T1 summarizes the source strength as a function of energy.

Table 5.2-T1
Source Strength for 360 kCi ^{60}Co

Gamma Energy (MeV)	MeV/sec	Photons/sec
1.173	1.561×10^{16}	1.331×10^{16}
1.332	1.774×10^{16}	1.332×10^{16}

MDS Nordion determines the contained activity of the C-188 source from the measurements of the exposure rate at approximately four meters from the side of the C-188 source. The measured exposure rate is converted to activity (Ci) using the 1.29 R·m^2/h/Ci (see Appendix 5.5.2) exposure rate constant for ^{60}Co and applying corrections for distance, self-absorption and attenuation due to the double encapsulation of the source capsule. The self-absorption factor for slugs or pellets used in the C-188 source capsule was determined from Monte Carlo calculations.

5.3 MODEL SPECIFICATION

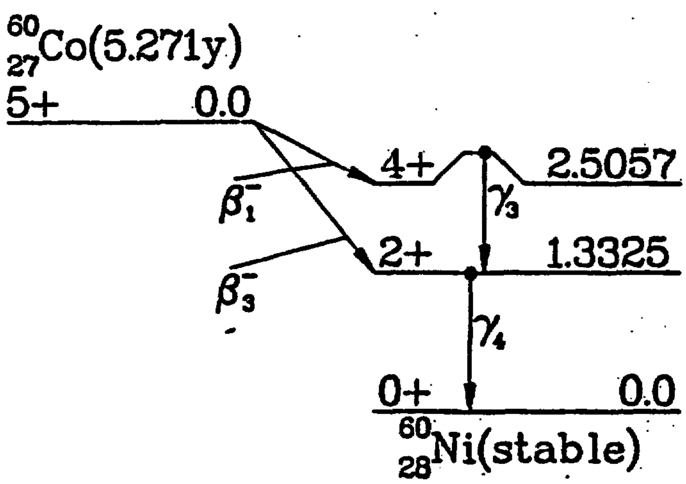
5.3.1 DESCRIPTION OF THE RADIAL AND AXIAL SHIELDING CONFIGURATION

Figure 5.1-F1 shows the dimensions (to scale) of the radial and axial shielding materials. Radiation surveys of F-294 prototype packaging show the highest fields on contact are below the package and in the horizontal plane through the source centre.

5.3.2 SHIELD REGIONAL DENSITIES

The mass densities for lead and iron used in Section 5.4 are 11.3 g/cm^3 and 7.86 g/cm^3 respectively. For ease of calculation, steel was approximated by iron.

Figure 5.2-F1
Cobalt-60 Decay Scheme



27 COBALT-60

HALFLIFE = 5.271 YEARS 29-MAR-78
DECAY MODE(S): β-

RADIATION	y(i) $(Bq\cdot s)^{-1}$	E(i) (MeV)	y(i)×E(i)
β- 1	9.99E-01	9.577E-02*	9.57E-02
β- 3	8.00E-04	6.258E-01*	5.01E-04
γ 3	9.99E-01	1.173E 00	1.17E 00
ce K, γ 3	1.50E-04	1.165E 00	1.75E-04
γ 4	1.00E 00	1.332E 00	1.33E 00
ce K, γ 4	1.14E-04	1.324E 00	1.50E-04

LISTED X, γ AND γ± RADIATIONS	2.50E 00
OMITTED X, γ AND γ± RADIATIONS**	1.14E-04
LISTED β, ce AND Auger RADIATIONS	9.65E-02
OMITTED β, ce AND Auger RADIATIONS**	4.73E-05
LISTED RADIATIONS	2.60E 00
OMITTED RADIATIONS**	1.61E-04

* AVERAGE ENERGY (MeV)
** EACH OMITTED TRANSITION CONTRIBUTES <0.100% TO Σy(i)×E(i) IN ITS CATEGORY.
NICKEL-60 DAUGHTER IS STABLE.

5.4 SHIELDING EVALUATION

Figure 5.4-F1 illustrates the field point locations before and after the Normal Conditions of Transport test (3 ft. free drop in top end drop orientation).

Figure 5.4-F2 illustrates the field point locations in the Hypothetical Accident Conditions of Transport tests (two [2] 30 ft. free drops and five [5] puncture pin drop tests.

The shielding evaluations in sections 5.4.1 and 5.4.2 were performed with the activity distributed in 40 C-188 sources in a single ring F-313 source carrier. When this activity is distributed in 80 C-188 sources in a double ring F-457 source carrier, the shielding evaluation results can be assumed to be the same. This is a conservative assumption based on the fact that the fields from the 40 sources on the inner ring of the double ring carrier will be partially shielded by the 40 sources on the outer ring.

5.4.1 SHIELDING EVALUATION OF F-294 PACKAGE IN NORMAL CONDITIONS OF TRANSPORT (NCOT)

The shielding evaluation of the F-294 package in normal conditions of transport (NCOT) was done for the following case:

> NCOT Case 1. External fields were evaluated near the top plug area where the source cage moved 1 in. (25 mm) closer to the outside, and the crush-shield was deformed by approximately 0.5 in. (12.7 mm). The effect of this reduction in source-to-dose distance is presented here.
>
> Before the drop test: Top of the crush shield (beyond top of the plug): Contact reading = 2.2 mrem/h (375,360 Ci as of 1998-Jan-07).
>
> On contact: Top of the crush shield (beyond top of the plug): Calculated dose = 1.7 mrem/h.
>
> After the NCOT test assessment: Top of the plug: Contact calculated dose rate = 1.9 mrem/h (See Appendix 5.5.6).

Therefore, the increase in radiation dose after the normal free drop test compared to the radiation dose before the normal free drop test = (1.9 mrem/h)/ (1.7 mrem/h) = 1.12 = 12 % increase. This complies with 10 CFR, Para 71.43 (f).

5.4.2 *SHIELDING EVALUATION OF F-294 PACKAGE UNDER HYPOTHETICAL CONDITIONS OF TRANSPORT (HACOT)*

The shielding evaluation of the F-294 package in hypothetical conditions of transport (HACOT) was done for the following two cases:

HACOT Case 1. The side where the source cage moved 0.25 in. (6 mm) closer to the outside, and the fireshield moved-in by approximately 3.88 in. (98.6 mm); and,

HACOT Case 2. The bottom where 1.4 in. (35.6 mm) of Pb slump was evaluated (Reference: 1997 March 18 letter from Cass R. Chappell (Package Certification Chief) to Joe Stirling (MDS Nordion).

Other shielding thicknesses (e.g., steel encasement) are assumed to be unaffected by the drop. MicroShield Version 5 was used to calculate these fields (Ref. [2]). This method calculates radiation fields in exposure rate. A 0.97 factor is used to convert from exposure to dose equivalent. The reduction in external radiation fields due to self-absorption was 10%, and mutual absorption factors were assumed to be 0% and 30% for top (plug geometry) and side geometry respectively. For this calculation, 360 kCi ^{60}Co is distributed in a point source at the centre of the cavity. ANS 6.4.3 attenuation coefficients and build-up factors are used by MicroShield Version 5.01 (Refs. [2] and [3]). Using the geometry shown in Figure 5.4-F1, the results are summarized in Table 5.4-T1.

Table 5.4-T1
Calculated Dose Equivalent Rates

Case Number (Field Location)	Dose Equivalent Rate on Contact (mrem/h)	Dose Equivalent Rate at 100 cm from Contact (mrem/h)
NCOT Case 1 (Plug Area)	1.9 (calculated)	0.7
HACOT Case 1 (Side Shield Area)	33.4	3.7
HACOT Case 2 (Bottom Area, with hypothetical 1.4 in. [35.6 mm] lead slump[3]	224	35.7

5.4.3 *IMPACT OF THE ALL STAINLESS STEEL CRACK SHIELD VERSUS STEEL-ENCASED-LEAD CRACK SHIELD.*

The F-294 test packaging crack shield assembly (located at top of the closure plug) is made up of steel-encased lead. The F-294 transport package crack shield is all stainless steel assembly. In Appendix 5.5.8, it is demonstrated that this design change has minimum impact from view point of the external radiation fields on the F-294 package.

[3] 1.4 inches of lead slump is only hypothetical. 0.44 inches was measured, so the fields would double in magnitude from 30 mR/h to 60 mR/h (at the bottom).

FIGURE WITHHELD UNDER 10 CFR 2.390

FIGURE WITHHELD UNDER 10 CFR 2.390

5.5 APPENDICES

The build-up factor calculation for heterogeneous media is usually the largest source of error in shielding calculations. During the integration (MicroShield), the build-up factor is calculated on all materials between the source kernel and the dose point. The shielding mean free paths for all material between the source and the dose point are used with the build-up characteristics of the dominant material (lead). Typical output from MicroShield 5 is given in Appendix 5.5.4 (Ref. [2]). For most shielding calculations investigated in Chapter 5, the fields calculated by MicroShield are in good agreement with the measured values.

Appendices

5.5.1 List of References for Chapter 5
5.5.2 Pre-drop Radiation Survey
5.5.3 Post-drop Radiation Survey
5.5.4 Converting Exposure to Dose Equivalent
5.5.5 Exposure Rate Constant for ^{60}Co
5.5.6 Typical MicroShield Output
5.5.7 Shielding Evaluation for the Hypothetical Accident conditions of Transport
5.5.8 Worst-case Estimate of the Increase in External Radiation Fields for the Re-designed Crack Shield Assembly

APPENDIX 5.5.1
LIST OF REFERENCES FOR CHAPTER 5

[1] ICRP Publication 38, *Radionuclide Transformations. Energy and Intensity of Emissions*, Pergamon Press, 1983.

[2] MicroShield Version 5.01, Grove Engineering, Inc., Rockville, Maryland, 1995-1996.

[3] ANSI/ANS-6.4.3, *Gamma-Ray Attenuation Coefficients and Build-up Factors for Engineering Materials*, Chicago, Illinois, 1992.

[4] CCEMRI, 1985, Comite consultatif pour les Etalone de Mesure des rayonnement ionisants, Rapport de la 11 session (BIPM) pR 157.

[5] Hubbell, J. H., "Photon Mass Attenuation and Energy-Absorption Coefficients from 1 keV to 20 MeV", International Journal of Applied Radiation Isotopes, Volume 33, Pergamon Press Ltd., 1982.

[6] ICRU Report 31, *Average Energy Reuired to Produce an Ion Pair*, 1979.

APPENDIX 5.5.2
PRE-DROP RADIATION SURVEY

Report for F-294 Regulatory Tests of IN/QA 1368 F294 (1)

Test # 5.1.9 Radiation Survey Before the Drop

The F-294 was loaded with 375,510 curies Cobalt-60 on 1998 January 06 in the form of forty (40) C-188 sealed sources as per the loading diagram attached (see Figure 1). The loading was done as any typical preparation for shipment, complete with a cavity argon purge and all fasteners appropriately torqued in Cell 06 within Industrial Operations, MDS Nordion, Kanata.

The upper crushshield and fireshield were assembled in place. The loaded flask was then moved on to the levelator for access to the underside (position #8 on the survey report), which is also an area of low activity levels, providing for accurate survey results. The F-294 shipping package was surveyed with two calibrated instruments as outlined in the Radiation Integrity for New Transport Packaging Procedure CO-QC/TP-0001 (2). Based on the type of instrumentation and the last calibration data, all readings would be within ± 5% of the actual. The highest reading for each elevation/location on the F-294 was recorded on the attached forms CO-QC/TPF4-0001 (2).

The initial survey was completed with the upper crushshield and fireshield in place on 1998 January 07 (see Figure 2). The second survey was completed on the container with them removed on 1998 January 12 (see Figure 3).

There were no unusual readings with either instrument for both configurations. All readings indicated complete and sound shielding integrity.

Configuration	Max. Reading on Contact	Max. Reading @ 1 Meter
F-294 Package	14 mR/h	1.8 mR/h
F-294 Container	24 mR/h	3.0 mR/h

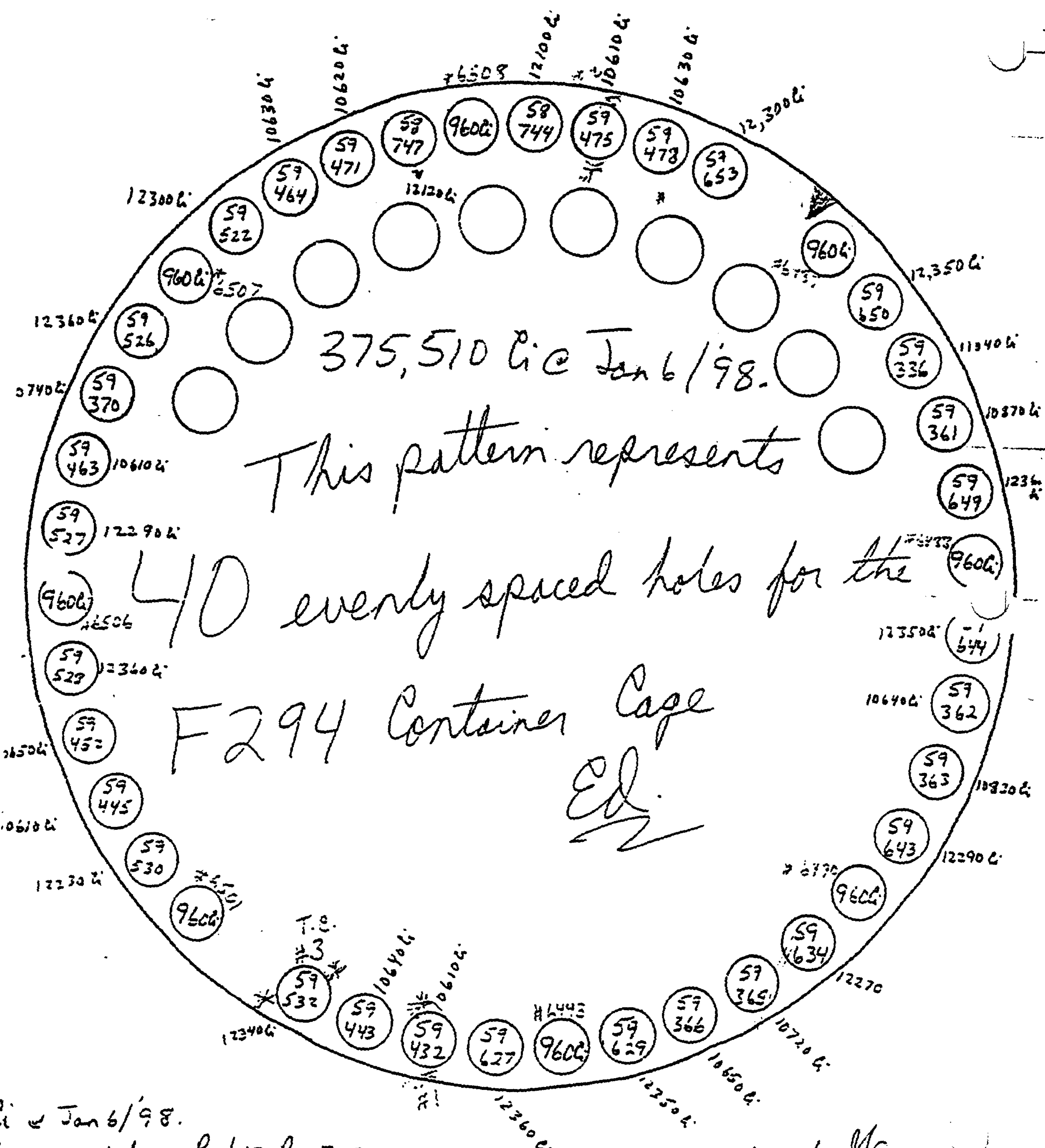
Conclusion: Both configurations meet the acceptance criteria of 200 mR/h on contact and 10 mR/h at 1 meter.

Prepared By: [signature] Date: 98.03.11.
D. Whitby, Industrial QC

Reviewed By: [signature] Date: 98 Mar. 12
K. O'Hara, Industrial Engineering, Physics

Approved By: [signature] Date: 98-MAR.-12
V. Shah, Package Engineering

MASTER



Ci @ Jan 6/98.
Low pencils from Pool 13 Pos E-2.
* Thermocouples #

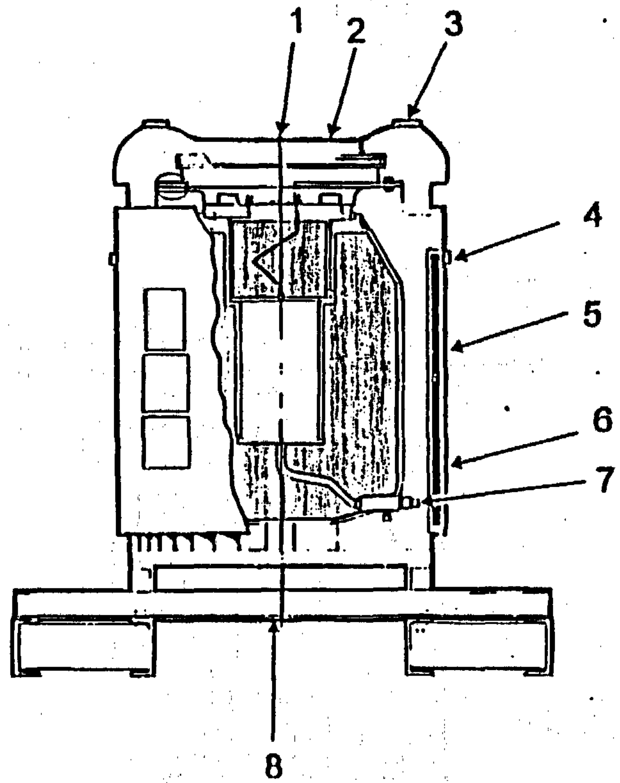
FIGURE 1

Acceptable
98/01/06

SHIPPING CONTAINER - RADIATION SURVEY FORMAT

RADIATION SURVEY
READINGS IN MILLIROENTGENS PER HOUR

POSITION	SURVEY METER S/N 2000 — GM Tube	SURVEY METER S/N 200 — Ion Chamber
	Make: BERTHOLD	Make: NUCLEAR ENTERPRISES
	Model: RATO/F	Model: PDM-1
	Calibration Date: 97.7.8	Calibration Date: 97.8.22
AT 1 METRE FROM SURFACE		
1	0.40 mR/h	7 μSv/h (0.7 mR/h)
2	0.40	4 (0.4)
3	0.35	4 (0.4)
4	0.95	6 (0.6)
5	1.8	12 (1.2)
6	1.2	9 (0.9)
7	0.80	5 (0.5)
8	1.4	14 (1.4)
METER IN CONTACT WITH SURFACE		
1	2.2 mR/h	22 μSv/h (2.2 mR/h)
2	2.0	18 (1.8)
3	1.0	11 (1.1)
4	2.8	23 (2.3)
5	12	90 (9.0)
6	6.5	50 (5.0)
7	0.30	8 (0.8)
8	14	120 (12.0)



CONTAINER F-294 # .... PROTOTYPE

SURVEY AND ACTIVITY DATE: 98.01.07

SOURCE DESCRIPTION: C-188's

CURIES: 375,510 Ci 98Jan06

REMARKS: CRUSH SHIELD AND FIRESHIE[LD] ARE IN PLACE. NO 'HOT SPOTS' OBSERVED; HIGHES[T] SIDE READING OCCURRED AT SOURCE HT.

SURVEYED BY: [signature]

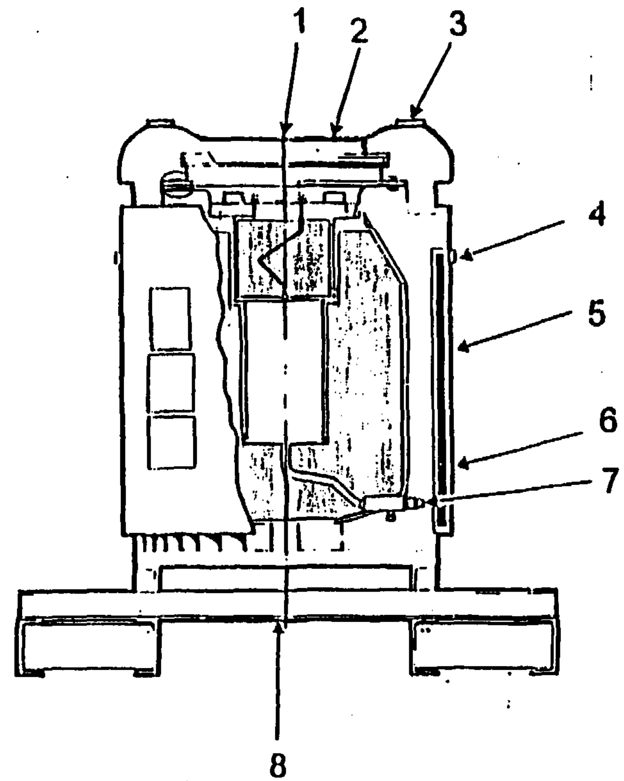
APPROVED BY: [signature] 98 Mar 12 — Senior Radiation Physicist [initials] 98 Mar 12 and [signature] 98-03-12 MAR. — Package & Facility Engineering

NOTE: Measurements on contact and at 100 cm from contact are made with the GM Tube and Ion Chamber instruments.

SHIPPING CONTAINER - RADIATION SURVEY FORMAT

RADIATION SURVEY
READINGS IN MILLIROENTGENS PER HOUR

POSITION	SURVEY METER S/N 2000 — GM Tube	SURVEY METER S/N 300 — Ion Chamber
	Make: BERTHOLD	Make: NUCLEAR ENTERPRISES
	Model: RATO/E	Model: PDM-1
	Calibration Date: 97.7.8	Calibration Date: 97.8.22
AT 1 METRE FROM SURFACE		
1	1.0 mR/h	6 μSv/h (0.6 mR/h)
2	1.0	6 (0.6)
3	1.4	9 (0.9)
4	1.4	14 (1.4)
5	3.0	23 (2.3)
6	2.2	18 (1.8)
7	1.2	8 (0.8)
8	1.4	10 (1.0)
METER IN CONTACT WITH SURFACE		
1	6.5 mR/h	70 μSv/h (7.0 mR/h)
2	9.0	70 (7.0)
3	10	80 (8.0)
4	5.5	40 (4.0)
5	24	150 (15)
6	0.6	50 (5.0)
7	0.5	8 (0.8)
8	14	110 (11)



CONTAINER F-294 # .... PROTOTYPE

SURVEY AND ACTIVITY DATE: 98.01.12

SOURCE DESCRIPTION: C-188's

CURIES: 375,510 Ci 98 Jan 06

REMARKS: CRUSHSHIELD + FIRESHIELD REMOVED.

HIGHEST READING OCCURRED AT SOURCE HEIGHT (~32" ABOVE FLOOR)

SURVEYED BY: [signature]

APPROVED BY: [signature] 98 Mar 12 — Senior Radiation Physicist and [signature] 18-MAR-12 — Package & Facility Engineering

NOTE: I... ements on contact and at 100 cm from contact ... de with the GM Tube and Ion Chamber instruments.

CO-QC/TPF4-... v 2

APPENDIX 5.5.3
POST-DROP RADIATION SURVEY

Report for F-294 Regulatory Tests of IN/QA 1368 F294 (1)

Test # 5.3.9 Radiation Survey After the Drop

Prepared By: Date: 98.04.06
D. Whitby, Industrial QC

Reviewed By: Date: 98 04 23
K. O'Hara, Industrial Engineering, Physics

Approved By: Date: 98/04/23
V. Shah, Package Engineering

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List of Tables

Test # 5.3.9 Radiation Survey After the Drop

The F-294 was loaded in the same manner as the pre-drop survey; with the same forty (40) C-188 sources decayed to 366,160 curies Cobalt-60 on 1998 March 17 as per the loading diagram attached (see Figure 1). The loading was again done as any typical preparation for shipment, complete with a cavity argon purge and all fasteners appropriately torqued in Cell 06 within Industrial Operations, MDS Nordion, Kanata. After the thermal testing, the upper crushshield was set in place as well as close as it could be to its proper damaged position. The three segments of the cylindrical sectioned fireshield were assembled and secured in place. The loaded flask was then moved on to the levelator for access to the underside (position #8 on the survey report), which is also an area of low background activity levels, providing for accurate survey results. The second and third survey were performed with the fireshield and crushshield removed.

The F-294 shipping package was surveyed with the same two calibrated instruments as the pre-drop survey as outlined in the Radiation Integrity for New Transport Packaging Procedure CO-QC/TP-0001 (2). The highest reading for each elevation/location on the F-294 was recorded on the attached forms CO-QC/TPF4-0001 (2).

The first post-drop survey was completed with the upper crushshield and the sectioned fireshield in place on 1998 March 24 (see Figure 2).
The second survey was completed on the container with both the fireshield and the crushshield removed on 1998 March 25 (see Figure 3). Both the first and second survey included contact readings within the damaged zones.
A varied third survey was completed on 1998 March 26 (see Figure 4). The damaged fins of the F-294 permitted a more intimate contact reading with the survey meter on the container wall. Therefore, the readings gathered on the first two surveys are the highest readings attained per CO-QC/TP-0001, but are not necessarily good for before and after comparison readings, as the proximity of the meter to the sources is closer on the damaged F-294. Therefore, the third survey was to attain readings as if the fins were not damaged; the contact readings were taken at approximately the same distance from the container wall as the pre-drop survey. Readings were recorded, only if different from the second post-drop survey.

There were no unusually high localized readings with either instrument for all configurations.

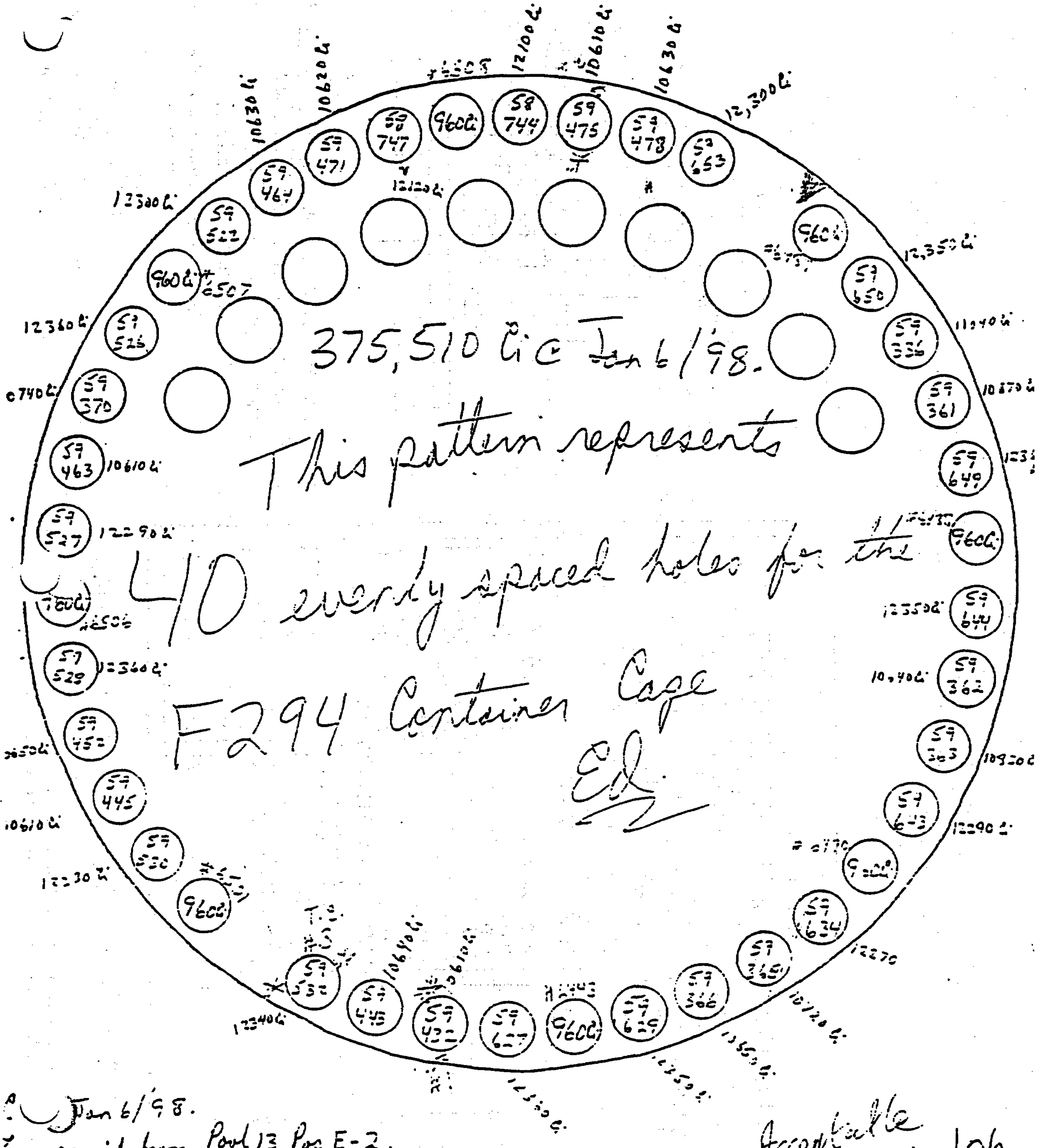
Configuration	Max. Reading on Contact	Max. Reading @ 1 Meter	Figure
F-294 Package	30 mR/h	1.9 mR/h	2
F-294 Container	30 mR/h	3.5 mR/h	3
F-294 Container (varied)	26 mR/h	N/A	4

Special additional measurements around the F-294 damaged zones were taken. These are recorded in Table 1.

Observations: There was a moderate increase in the fields around all areas, with a considerable increase at the bottom center of the container:
(post- drop reading) / (pre-drop reading) = %increase
30 / 14 = 2.14 or approximately 200% increase.

Conclusions: All configurations meet the acceptance criteria of 1 rem/h at 1 meter from the external surface of the package. [10CFR 71.51 (a) (2)]

MASTER



Jan 6/98.
Low pencils from Pool 13 Pos E-2.
* Thermocouples *

FIGURE 1

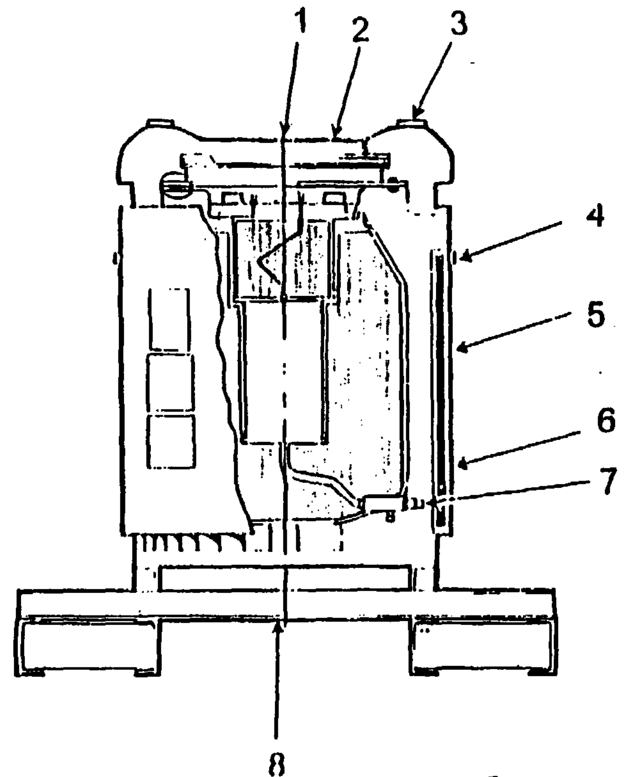
Acceptable
98/01/06

FIG. 2

SHIPPING CONTAINER - RADIATION SURVEY FORMAT

RADIATION SURVEY
READINGS IN MILLIROENTGENS PER HOUR

POSITION	S/N 2020 SURVEY METER GM Tube Make: BERTHOLD Model: RATE/F Calibration Date: 98·03·05	S/N 200 Ion Chamber Make: NUCLEAR ENTERPRISES Model: PDM-1 Calibration Date: 98·03·03
AT 1 METRE FROM SURFACE		
1	0.8 mR/h	9 μSv/h (0.9 mR/h)
2	0.8	8 (0.8)
3	0.8	8 (0.8)
4	1.4	13 (1.3)
5	1.8	19 (1.9)
6	1.6	13 (1.3)
7	1.0	10 (1.0)
8	1.8	13 (1.3)
METER IN CONTACT WITH SURFACE		
1	4.0 mR/h [PUNCTURE ZONE DEPRESSION]	40 μSv/h (4.0 mR/h)
2	3.6 mR/h	20 (2.0)
3	1.6	10 (1.0)
4	5.0 [DEPRESSION HERE " " "]	24 (2.4)
5	16	100 (10.0)
6	5.5	40 (4.0)
7	0.6	5 (0.5)
8	30 [PUNCTURE ZONE DEPRESSION]	210 (21)



CONTAINER F-294 # .... PROTOTYPE {POST DROP}
SURVEY AND ACTIVITY DATE: 98 MARCH 24
SOURCE DESCRIPTION: C-188's
CURIES: 365,221 Ci
REMARKS: CRUSHSHIELD + FIRESHIELD ARE IN PLACE.

SURVEYED BY: ______________

APPROVED BY: ______________ Senior Radiation Physicist and ______________ Package & Facility Engineering

NOTE: [illegible]ements on contact and at 100 cm from contact [illegible]le with the GM Tube and Ion Chamber instruments.

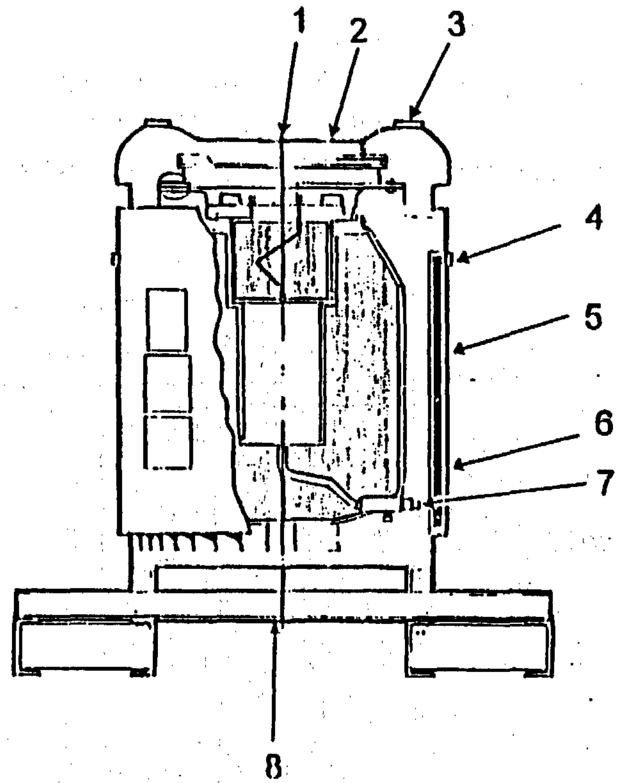
CO-QC/TPF4-[illegible] v 2

F 3

SHIPPING CONTAINER - RADIATION SURVEY FORMAT

RADIATION SURVEY
READINGS IN MILLIROENTGENS PER HOUR

POSITION	SURVEY METER S/N 2000 — GM Tube	SURVEY METER S/N 200 — Ion Chamber
	Make: BERTHOLD	Make: NUCLEAR ENTERPRISES
	Model: RATE/E	Model: PDM-1
	Calibration Date: 98·03·05	Calibration Date: 98·03·03
AT 1 METRE FROM SURFACE		
1	1.2 mR/h	6 µSv/h (0.6 mR/h)
2	1.4	6 (0.6)
3	1.6	9 (0.9)
4	1.4	14 (1.4)
5	3.5	25 (2.5)
6	3.0	20 (2.0)
7	1.2	9 (0.9)
8	1.6	13 (1.3)
METER IN CONTACT WITH SURFACE		
1	9.0 mR/h [BESIDE WELDED SHACKLE]	90 µSv/h (9.0 mR/h)
2	14 [NR. L.L. #1]	90 (9.0)
3	22 [HOLE PATTERN NR. L.L. #1]	110 (11)
4	8.5 [BETWEEN FINS @ L.L. # R E H+E]	50 (5.0)
5	26 [BETWEEN DAMAGED FINS]	140 (14)
6	0.9	50 (5.0)
7	0.7	5 (0.5)
8	30	200 (20)



CONTAINER F-294 # .... PROTOTYPE – POST DROP

SURVEY AND ACTIVITY DATE: 98·03·26

SOURCE DESCRIPTION: C-188's

CURIES: 364,958 Ci

REMARKS: CRUSHSHIELD + FIRESHIELD REMOVED.

SURVEYED BY: [signature]

APPROVED BY: [signature] Senior Radiation Physicist and [signature] Package & Facility Engineering

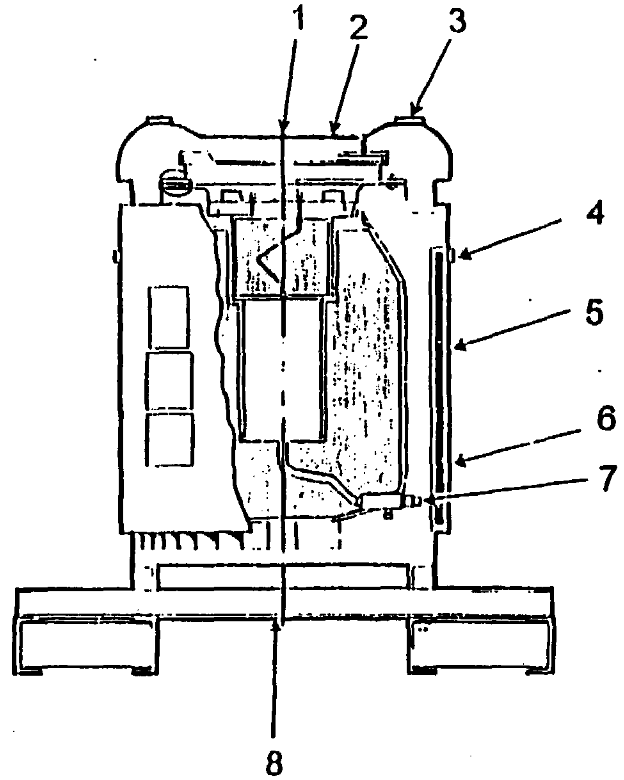
NOTE: Measurements on contact and at 100 cm from contact are made with the GM Tube and Ion Chamber instruments.

FIG. 4

SHIPPING CONTAINER - RADIATION SURVEY FORMAT

RADIATION SURVEY
READINGS IN MILLIROENTGENS PER HOUR

POSITION	SURVEY METER S/N 2000 — GM Tube	SURVEY METER S/N 290 — Ion Chamber
	Make: BERTHOLD	Make: NUCLEAR ENTERPRISES
	Model: Rate/F	Model: PDM-1
	Calibration Date: 98·03·05	Calibration Date: 98 03 03
AT 1 METRE FROM SURFACE		
1		
2		
3		
4		
5		
6		
7		
8		
METER IN CONTACT WITH SURFACE *		
1		
2		
3		
4	5.5 mR/h NR LL#3	
5	27 NR LL#3	130 µSv/h (13 mR/h)
6		
7		
8	26	



CONTAINER F-294 # .... PROTOTYPE POST DROP

SURVEY AND ACTIVITY DATE: 98·03·26

SOURCE DESCRIPTION: C-188's

CURIES: 364,958 Ci

REMARKS: CRUSHSHIELD + FIRESHIELD REMOVED.
* PLACEMENT OF METER AS IF FINS WERE NOT DAMAGED.

SURVEYED BY: [signature]

APPROVED BY: [signature] Senior Radiation Physicist and [signature] Package & Facility Engineering

NOTE: M... ...ements on contact and at 100 cm from contact ar... ...e with the GM Tube and Ion Chamber instruments.

Radiation Survey at the F 294 Damaged Zones

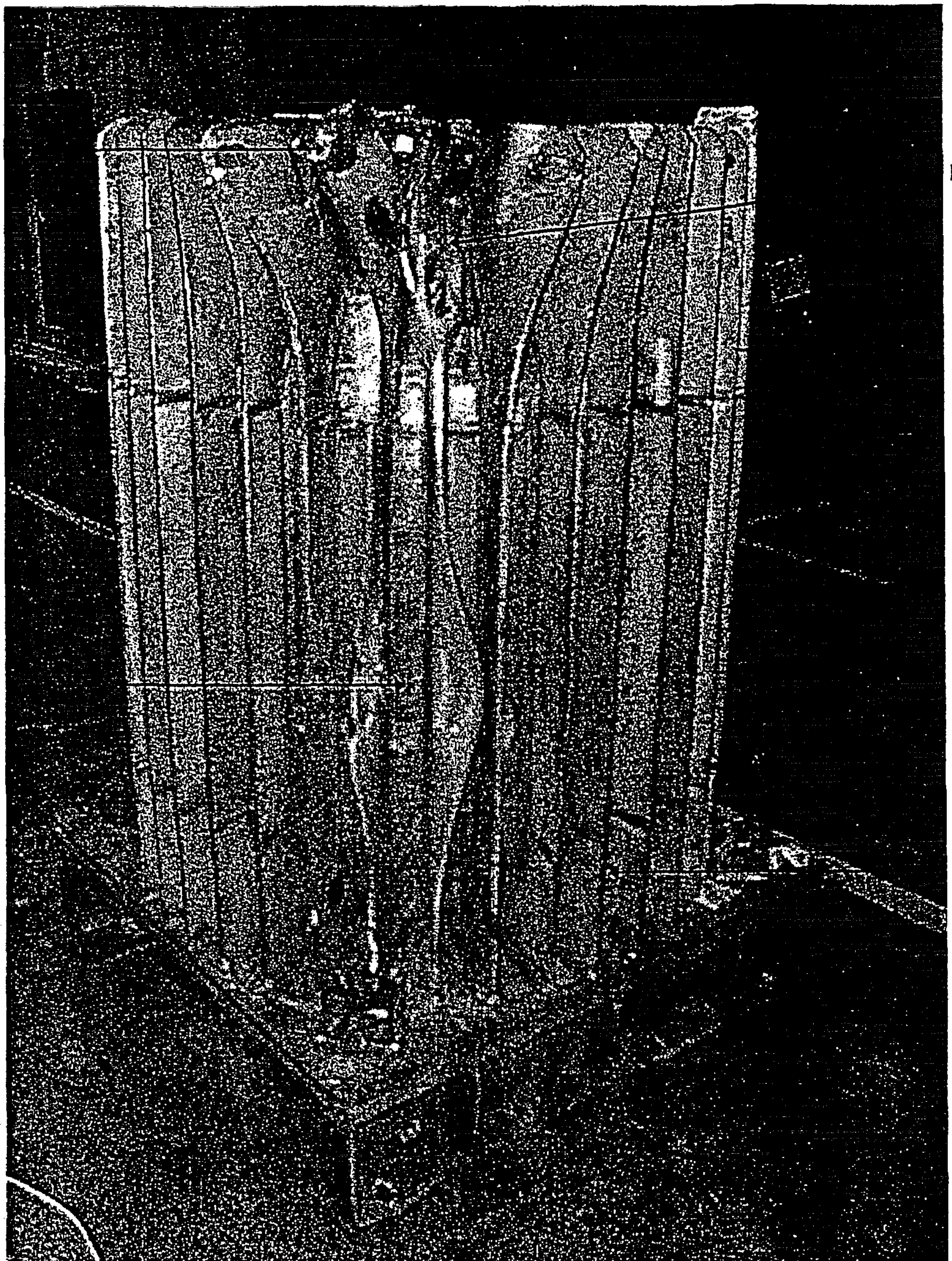
ADDITIONAL READINGS

	LOCATION	CONTACT READING	1 METER READING	OPPOSITE POSITION	CONTACT READING	1 METER READING
Figure 5 Position #1	Upper damaged zone at lift lug #4.	8.5 mR/h	1.4 mR/h	Upper section, at lift lug #2	3.0 mR/h	1.0 mR/h
Figure 6 Position #2	Puncture pin damage zone near lift lug #3	26 mR/h	3.5 mR/h	Mid section, at lift lug #1	12 mR/h	2.0 mR/h
Figure 5 Position #3	Puncture pin midsection lift lug #4	26 mR/h	2.4 mR/h	mid-section near lift lug #2.	10.4 mR/h	2.2 mR/h
Figure 7 Position #4	Top of fins of damaged zone, 16" from center	4.0 mR/h	1.4 mR/h	Top of fins, between lift lugs #2 & #3.	3.0 mR/h	1.4 mR/h
	Highest reading on top of plug (over bolt hole pattern)	22 mR/h	1.6 mR/h	Over bolt holes on opposite side.(nr L.L #1)	10 mR/h	1.4 mR/h
	Over vent line	10 mR/h				

Table 1

Post Drop Test

ug #4



Position #1

tion #3

Drainline

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Figure 5

Post Drop Test

ift lug #2

Lift lug #3

Position #2

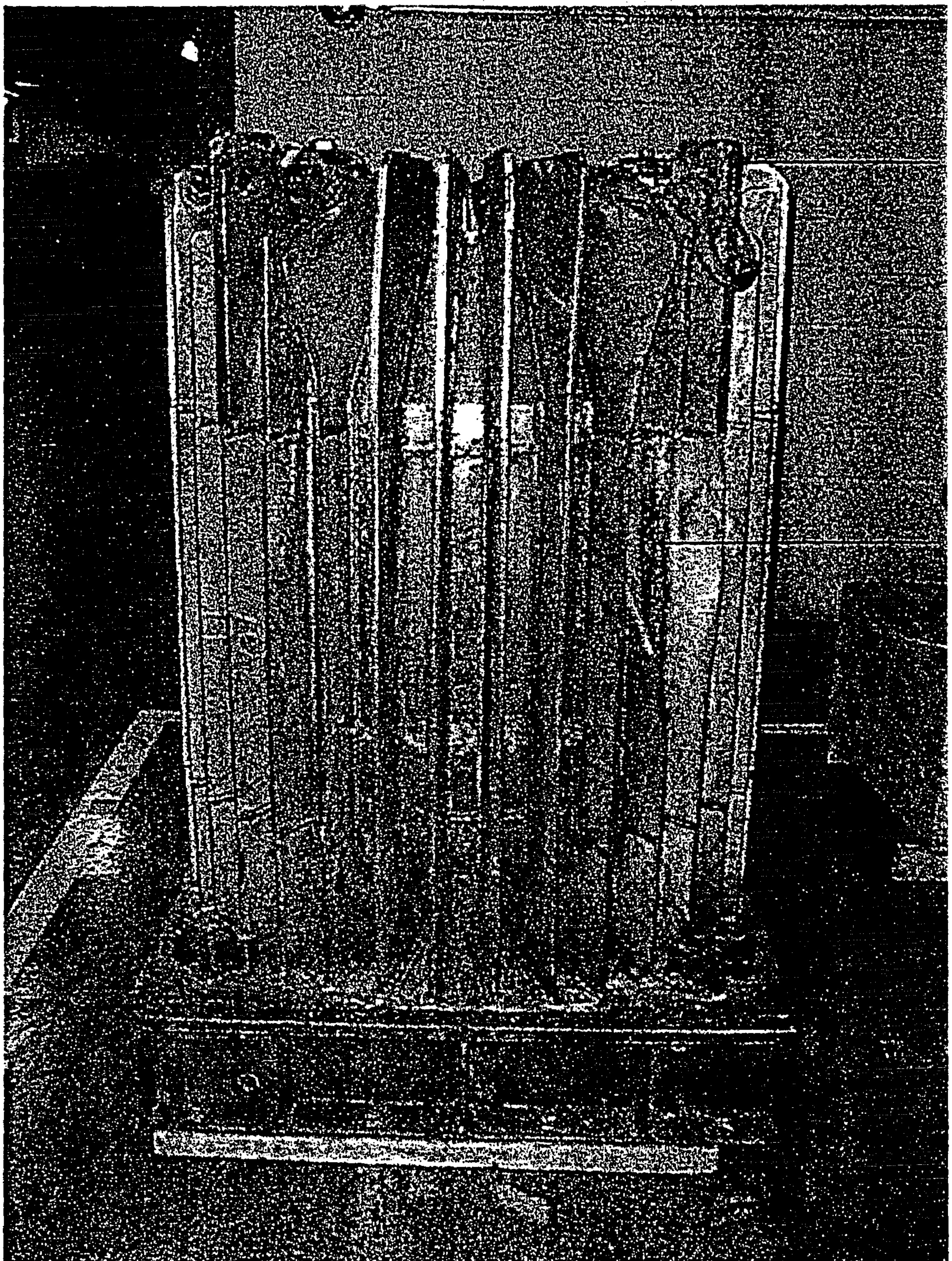

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Figure 6

Post Drop Test

Position #4

:Lug #1

Lift Lug #2

line this

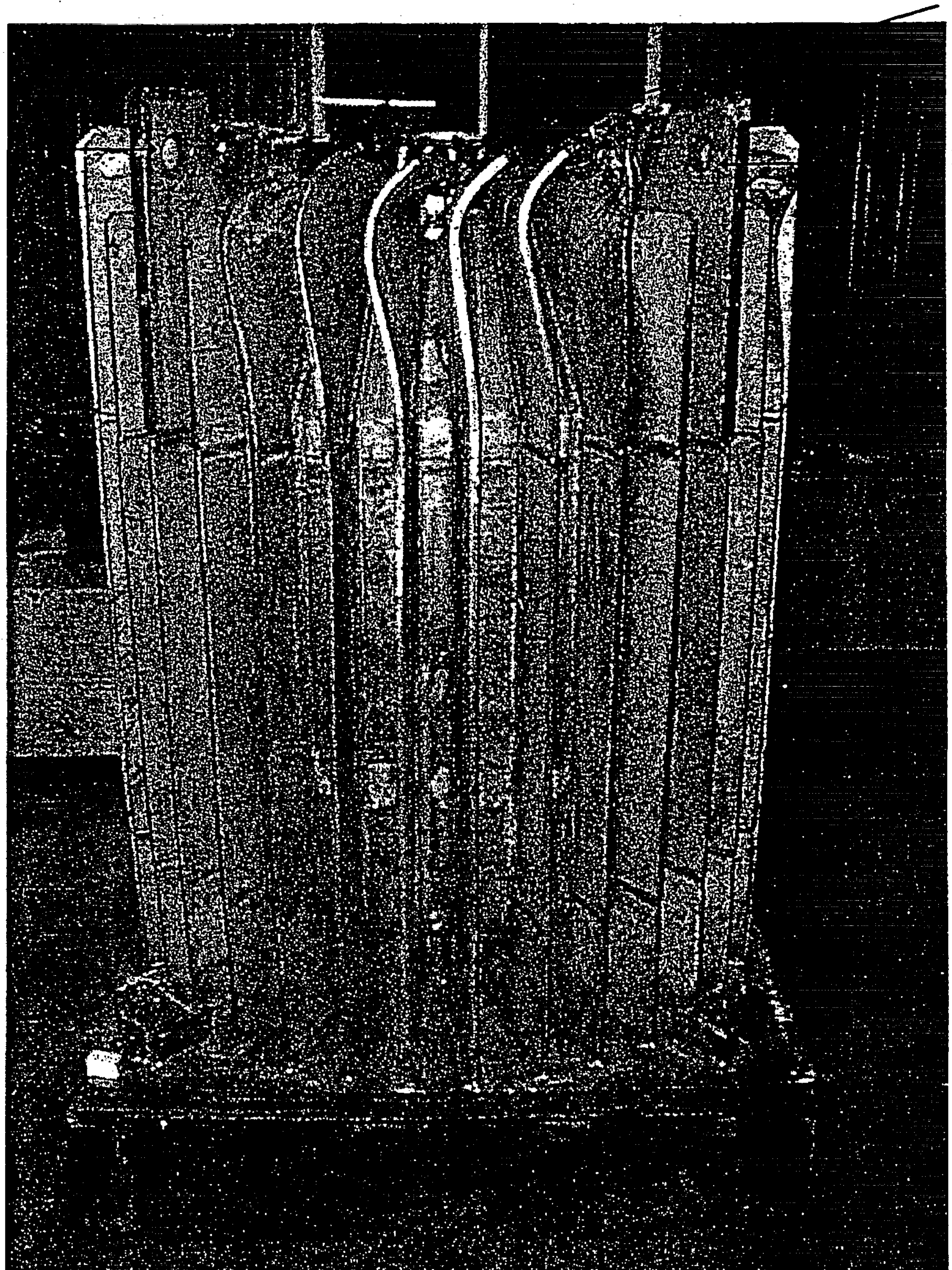

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Figure 7

APPENDIX 5.5.4
CONVERTING EXPOSURE TO DOSE EQUIVALENT

The radiation fields are measured in exposure rate (mR/h). The conversion factor from exposure (mR) to dose equivalence (mrem) is 0.97.
1 R ≡ 2.58 x 10^{-4} C/kg (by definition)
W/e = 33.97 J/C (average energy expended is dry air per ion pair formed) (see Ref. [6]).

1 R ≡ 2.58 x 10^{-4} C/kg x 33.97 J/C = 8.76 x 10-3 J/kg
1 rad = 0.01 J/kg (by definition)
Therefore, 1 R = 0.876 rad (in air)
To convert from absorbed dose in air to absorbed dose in water:

$$\frac{D_{water}}{D_{air}} = \frac{(\mu_{en} / \rho)_{water}}{(\mu_{en} / \rho)_{air}} = \frac{0.0297\ cm^2/g}{0.0267\ cm^2/g}$$

where
(μ_{en} / ρ), the mass-energy absorption coefficients, have been interpolated from Hubbell, 1982 (see Ref. [5]), assuming 1.25 MeV gamma radiation.

Therefore,
1 R = 0.876 rad (in air) = 0.876 x 1.112 rad (in water)
= 0.97 rad (in water)

Finally, the quality factor for gamma radiation is 1, so the 0.97 factor can be used to convert from exposure (R) to dose equivalence (rem).

1 mR ≡ 0.97 mrem

APPENDIX 5.5.5
EXPOSURE RATE CONSTANT FOR ^{60}Co

1. W/e = 33.97 J C^{-1} (BIPM, 1985) (Ref. [4])
2. 1 = 1.173 MeV, 99.9% probability
   2 = 1.332 MeV, 100% probability (ICRP 38, Radionuclide Transformations)
3. The corresponding mass-energy absorption coefficients $\frac{(\mu_{en})}{\rho}$ in dry air near sea level were interpolated from Hubbell (1982). (Ref. [5])

   $$\frac{(\mu_{en})}{\rho} = 0.270 \text{ cm}^2 \text{ g}^{-1} \text{ for } 1.173 \text{ MeV}$$

   $$\frac{(\mu_{en})}{\rho} = 0.262 \text{ cm}^2 \text{ g}^{-1} \text{ for } 1.332 \text{ MeV}$$

4. $1\text{R} = 2.58 \times 10^{-4}$ C/kg by definition

The exposure rate constant for ^{60}Co is:

$$\frac{3.7 \times 10^{10} \text{ disintegrations}}{\text{sec. Ci}} \times \frac{3600 \text{ sec}}{\text{h}} \times \frac{1}{4\pi (100 \text{ cm})^2}$$

$$\times \frac{1}{2.58 \times 10^{-4} \ \frac{\text{C}}{\text{kg R}}} \times \frac{10^3 \text{ g}}{\text{kg}} \times \frac{1}{33.97 \text{ J C}^{-1}} \times \frac{1.6022 \times 10^{-19} \text{ J}}{\text{eV}}$$

$$\times\ 10^6 \text{ eV/MeV } (1.173 \text{ MeV} \times 0.999 \times 0.0270 \text{ cm}^2/\text{g} + 1.332 \text{ MeV} \times 1.000 \times 0.0262 \text{ cm}^2/\text{g})$$

$$= 1.29 \frac{\text{Rm}^2}{\text{h.Ci}}$$

APPENDIX 5.5.6
TYPICAL MICROSHIELD OUTPUT

MicroShield v5.01 (5.01-00157)
MDS Nordion

Page : 1
DOS File: NCOT1A.MS5
Run Date: May 25, 1998
Run Time: 3:16:01 PM
Duration: 00:00:00

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

Case Title: NCOT Case 1
Description: Ncot Case 1 (F294 Package) - After Drop
Geometry: 1 - Point

Dose Points

	X	Y	Z
# 1	76.835 cm	0 cm	0 cm
	2 ft 6.2 in	0.0 in	0.0 in

Shields

Shield Name	Dimension	Material	Density
Shield 1	22.86 cm	Air	0.00122
Shield 2	1.27 cm	Iron	7.86
Shield 3	27.94 cm	Lead	11.34
Shield 4	6.35 cm	Iron	7.86
Shield 5	1.905 cm	Iron	7.86
Shield 6	13.97 cm	Air	0.00122
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels
Co-60	3.6000e+005	1.3320e+016

Buildup
The material reference is : Shield 3

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.6938	2.173e+12	2.463e-09	1.145e-08	4.755e-12	2.211e-11
1.1732	1.332e+16	1.919e+01	1.335e+02	3.429e-02	2.385e-01
1.3325	1.332e+16	1.464e+02	1.110e+03	2.540e-01	1.925e+00
TOTALS:	2.664e+16	1.656e+02	1.243e+03	2.883e-01	2.164e+00

MicroShield v5.01 (5.01-00157)
MDS Nordion

```
Page    : 1                              File Ref: ________
DOS File: NCOT1.MS5                          Date: ________
Run Date: May 25, 1998                         By: ________
Run Time: 3:10:05 PM                      Checked: ________
Duration: 00:00:00
```

Case Title: NCOT Case 1
Description: Ncot Case 1 (F294 Package) - Before Drop
Geometry: 1 - Point

Dose Points

	X	Y	Z
# 1	80.645 cm	0 cm	0 cm
	2 ft 7.7 in	0.0 in	0.0 in

Shields

Shield Name	Dimension	Material	Density
Shield 1	10.0 in	Air	0.00122
Shield 2	.5 in	Iron	7.86
Shield 3	11.0 in	Lead	11.34
Shield 4	2.5 in	Iron	7.86
Shield 5	.75 in	Iron	7.86
Shield 6	6.0 in	Air	0.00122
Air Gap		Air	0.00122

Source Input
Grouping Method : Actual Photon Energies

Nuclide	curies	becquerels
Co-60	3.6000e+005	1.3320e+016

Buildup
The material reference is : Shield 3

Results

Energy MeV	Activity photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.6938	2.173e+12	2.235e-09	1.039e-08	4.315e-12	2.006e-11
1.1732	1.332e+16	1.741e+01	1.211e+02	3.112e-02	2.164e-01
1.3325	1.332e+16	1.329e+02	1.007e+03	2.305e-01	1.747e+00
TOTALS:	2.664e+16	1.503e+02	1.128e+03	2.616e-01	1.964e+00

APPENDIX 5.5.7
SHIELDING EVALUATION FOR THE HYPOTHETICAL ACCIDENT THERMAL CONDITIONS

The vertical cross-section of the F-294 package assembly (Figure 5.1-F1) shows 11.25 in. lead minimum typically. The thermal model calculates no lead melt (see Section 3.5.6). The calculation, therefore, is identical to the one that calculated dose equivalent rates before impact in Section 5.4.

MicroShield has been used to calculate these fields. The results are shown below in Table 5.5-T1.

Table 5.5-T1
Calculated Dose Equivalent Rates Before and After the Hypothetical Thermal Conditions

On Contact		At One Meter	
mrem/h	µSv/h	mrem/h	µSv/h
17.7	177	3.6	36

APPENDIX 5.5.8
WORST-CASE ESTIMATE OF THE INCREASE IN EXTERNAL RADIATION FIELDS FOR THE RE-DESIGNED CRACK SHIELD ASSEMBLY

The F-294 test packaging crack shield assembly is comprised of steel-encased lead (3/16 in. [0.48 cm] steel, 1.5 in. (38 mm) lead and 0.25 in. (6 mm steel). The increase in external radiation fields is estimated if the 1.5 in. Pb is replaced by 1.5 in. steel.

The worst-case scenario is established by assuming no energy degradation as the ^{60}Co γ-rays propagate through the crack and strike the crack shield assembly. The transmission of ^{60}Co radiation (1.25 MeV average energy assumed) through 1.5 in. Pb and 1.5 in. steel (approximated by iron) will be calculated.

Case I - Transmission of ^{60}Co Radiation through 3.81 cm Pb

mfp = (0.0572 cm^2/g) (11.3 g/cm^3) (3.81 cm) = 2.46

Build-up (^{60}Co in 2.46 mfp Pb) ≈ 1.86

Transmission Factor ≈ 1.86 exp (–2.46) = 0.16

Case II - Transmission of ^{60}Co Radiation through 3.81 cm Fe

mfp = (0.0532 cm^2/g) (7.86 cm^2/g) (3.81 cm) = 1.59

Build-up (^{60}Co in 1.59 mfp Fe) ≈ 2.33

Transmission Factor ≈ 2.33 exp (–1.59) = 0.48

The estimated worst-case increase in external radiation fields is 0.48/0.16 ≈ 3.0. Table 5.5-T2 summarizes the estimated external radiation fields at 5 cm and 100 cm in the plug area for 360 kCi ^{60}Co.

Table 5.5-T2
Estimated Radiation Fields for the Steel-Encased Lead and Pure Steel Crack Shield Assemblies

	Radiation Fields (mR/h) at Different Field Locations	
Crack Shield Assembly Materials	@ 5 cm	@ 100 cm
Steel-encased Lead	1.52	1.2
Steel	4.56	3.6[1]

[1] The radiation field at 1 meter will not increase three-fold since the crack-leakage contribution to the field at 1 meter is small.

CHAPTER 6 – CRITICALITY

The requirements of this chapter are not applicable since F-294 package does not contain any fissile material.

CHAPTER 7 – OPERATING PROCEDURES

This chapter describes the operating procedures for the F-294 package. There are two (2) transport scenarios: i) international shipments and, ii) domestic shipments within the USA.

i) International shipments

For international shipments *originating from the Customer's site* to MDS Nordion Inc., Ottawa, Ontario, Canada, the F-294 is loaded and prepared for shipment as per MDS Nordion procedure IN/OP 0283 F294 (Ref. [1]).

ii) Domestic shipments within the USA

For domestic shipments within the USA, there can be four distinct phases in a single round trip:

Phase 1: An empty F-294 package with appropriate equipment is sent from MDS Nordion, Ottawa, Ontario, Canada to a Wet Source Storage Gamma Irradiator (Category IV) customer's site A.

Phase 2: At customer's site A in the USA, the F-294 is loaded with the appropriate number of C-188 sealed sources and prepared for shipment from customer's site A to customer's site B as per MDS Nordion Procedure IN/OP 0283 (Ref. [1]).

Phase 3: At customer's site B, the sources from the F-294 are unloaded and transferred to the irradiator at site B as per MDS Nordion Procedure IN/OP 0284 F294 (Ref. [2]). At the conclusion of this phase the F-294 package is empty.

Phase 4: An empty F-294 package is shipped back to MDS Nordion, Ottawa, Ontario, Canada as per MDS Nordion Procedure IN/OP 0285 F294 (Ref. [3]).

In addition, the F-294 package can be used for multiple round trips from a Wet Source Storage Gamma Irradiator customer's site A and to a Wet Source Storage Gamma Irradiator customer's site B or vice versa.

Phase 1: An empty F-294 package is sent to a customer's site A from MDS Nordion.

Phase 2: The F-294 is loaded at the customer's site A and prepared for shipment to the customer's site B.

Phase 3: The F-294 is unloaded at the customer's site B. The empty F-294 is returned to the customer's site A for a second trip.

Phase 4: The F-294 is loaded (second load) at the customer's site A and prepared for shipment to the customer's site B.

Phase 5: The F-294 is unloaded (second unload) at the customer's site B. The empty F-294 is then returned to the customer's site A for a third trip (third load) or returned empty to MDS Nordion, Ottawa, Ontario, Canada.

7.1 PROCEDURES FOR LOADING THE PACKAGE

The underwater loading of the C-188 sealed sources in the F-294 package at a customer's site A and the preparation for shipment of F-294 transport package is carried out as follows:

7.1.1 PURPOSE

This operating procedure is to ensure that the underwater loading of C-188 cobalt-60 sealed sources and the preparation for shipment of the F-294 transport package from a customer's irradiator site A to the customer's site B is in compliance with the design specifications, handling requirements, and regulatory requirements. The F-294 transport package shall be transported as "exclusive use" shipment as per 10 CFR Part 71 regulations.

7.1.2 SCOPE

This operating procedure describes the following operations:

1. The underwater loading of the F-313 cage (carrier) or F-457 cage (carrier), containing C-188s (cobalt-60) sources, into the cavity of the F-294 package.
2. Preparation for shipment of the F-294 package.
3. Instructions for securing the F-294 package on the road vehicle.

These are the minimum requirements that must be achieved.

7.1.3 COMPLIANCE AND RESPONSIBILITY

It is the responsibility of MDS Nordion International personnel or its agent to ensure that the F-294 transport package is prepared for transport in compliance with the regulatory requirements.

It is the responsibility of the customer, in the role of the consignor, to approve the release of the F-294 shipment.

It is the responsibility of the pertinent regulatory authority to enforce compliance as per the F-294 transport package license.

7.1.4 TRANSFERRING F-294 EMPTY PACKAGING FROM THE IRRADIATOR SITE TO THE POOL

1. Verify that the F-294 packaging is empty.
2. Remove the shipping skid.
3. Remove the crush shield and the cylindrical fireshield from the packaging.
4. Inspection.
   - 4.1 Inspection and replacement for top closure components.
     - a) Inspect the neoprene gasket, the sealing surfaces and the closure bolts.
     - b) If necessary, replace the gasket or closure bolts.
   - 4.2 Inspection and replacement for the vent line closure components.
     - a) Inspect the vent line caps, gasket and seal surfaces.
     - b) If necessary, replace the gasket.

     Ref: Apply 20 ft.-lb. ± 2 ft.-lb. torque to close the vent line cap.
   - 4.3 Inspection and replacement for the drain line closure components.
     - a) Inspect the vent line caps, gasket and seal surfaces.
     - b) If necessary, replace the gasket.

     Ref: Apply 50 ft.-lb. ± 5 ft.-lb. torque to close the drain line cap.
   - 4.4 Inspection of the cylindrical fireshield.
     - a) Ensure that the openings of the fireshield that allow air to flow through the transport package are unobstructed.
5. Confirm that the source rack is lowered into the pool and is fully disabled.
6. Sling the F-294 to the crane.
7. The F-294 packaging can now be transferred into the irradiator shielded building.
8. Lift the container and lower it into the pool.
9. Lift the shield plug clear of the container, giving access to the cavity of the F-294.

> *Note*
> *When handling sources/source carriers underwater,*
> *always have an audible radiation monitor at your work position*
> *for warning if, by accident, any source/carrier is lifted too close to the pool surface.*
> *When removing any item from the pool, always check it with the radiation monitor*
> *as the item is being brought to the surface.*

7.1.5 UNDERWATER LOADING OF C-188S

1. Load C-188s into the source carrier as per loading diagram, see Figure 7.1- F1 a, for the F-313 source carrier or 7.1-F1b for the F-457 source carrier. Do not exceed 360 kCi of Cobalt-60, the licensed limit of the F-294 transport package. Retain all wipes for the possibility of further testing. The activity should be distributed evenly around the carrier.

> *Note*
> *The wipes of C-188s and source carrier*
> *are taken underwater at the customer's site.*
> *The wipes are returned to MDS Nordion or its agent for confirmatory measurements.*

2. Lift the source carrier and place it on the bottom of the cavity of the F-294 container.
3. Replace the shield plug. Fasten two out of sixteen bolts with the F-294 at the bottom of the pool.
4. Slowly raise the F-294 package out of the water from the bottom of the pool to the top of the pool.

> Warning
> **Stand clear as steam and hot water may blow out of the vent plug holes**
> **when the container is raised out of the water.**

5. Ensure that the shield plug is fully secured. Torque each bolt to 100 ± 10 ft.-lb. (133 ± 13 N m).
6. Blow argon through the vent line, forcing out ALL the remaining water in the F-294 cavity through the drainline back into the pool. The following steps outline the procedure for purging F-294 cavity with argon and the criteria for ensuring all the water has been drained from the F-294 cavity.
   - 6.1 Connect the equipment as per Figure 7.1-F2.
   - 6.2 Open Valve VX to set argon pressure at 30 psi and let argon flow from the center vent line connection through the F-294 cavity.
   - 6.3 Check for water or water spray flowing from the drain line outlet.
   - 6.4 Repeat blowing argon until there is no water spray exiting from drain line outlet. Use a rag or paper towel for detecting water.
7. Seal off vent line connection using gasket, vent line shield insert, cap. Use 20 ft.-lb. ± 2 ft.-lb. torque on the cap. Seal off drain line connection using gasket and cap. Use 50 ft.-lb. ± 5 ft.-lb. torque on the cap.
8. The F-294 container is transferred from inside the irradiator building to outside the irradiator building. The F-294 can be placed on the truck trailer bed or a designated staging area for a specified period (Ref. [2]) before the final argon purge.

7.1.6 PERFORM THE CAVITY WATER FLUSH TEST PROCEDURE

1. Remove the crush shield from the top of the container. Remove the cylindrical fireshield from the container.
2. Remove the drainline cap and the top vent center cap and the vent-line shield plug from the container. If any water drains from the container, collect the water and check it for any contamination. Document and report the drainage of discolored water from the container.
3. Thread the brass adapter with the filter into top center vent plug.
4. Connect the brass adapter supplied to the lower drain tube of the F-294 package.
5. Close the spigot valve on the US 10-gallon plastic container, check all hose connections for tightness and fill 4/5th full with de-ionized water. Use of two (2) US 5-gallon plastic containers in place of one (1) US 10-gallon container is permitted.
6. With plastic tubing attached to each end of the filter, secure one end of plastic tube to the lower draintube adapter on the F-294 package and the other end to the spigot on the US 10-gallon plastic container.

> *Note*
> *For optimum flow of the water into the cavity,*
> *open the spigot with the water-filled plastic container below the drain-tube level*
> *to allow water to fill the plastic tube,*
> *then raise the water-filled plastic container to the top of the transport package.*

7. Support the bottom of the water-filled plastic container above, or level with, the top of the F-294 package.
8. Remove the cap of the US 10-gallon plastic container and open the spigot, allowing water to slowly fill the F-294 package cavity.

> **Warning**
> **Steam and hot water may blow out of the vent holes as the cavity fills.**

9 Monitor the filter on the top vent hole during filling for increase in the radiation fields. If the radiation field at the monitoring position increases, immediately close the spigot to prevent further entry of water into the cavity. Evacuate the area, taking care to prevent the possible spread of contamination. Restrict access to the area. Notify the facility's Radiation Safety Officer, the local competent authority and MDS Nordion.

10. As soon as the cavity and the vent tube(s) are full, lower the plastic container to the ground and allow all the water to drain back into the plastic container through the filter. Monitor the filter in the plastic tube with the survey meter.
11. When all the water has drained from the cavity, close the spigot on the plastic container, and disconnect the plastic tube from the lower drain tube adapter on the F-294 package and the brass adapter with the filter from the top center vent plug.
12. Remove the filters to a low background area and monitor the filters. Wearing protective gloves, carefully cut open each filter and remove the filter material. Place the filter material in a labeled and lockable plastic bag and slowly scan for radioactive contamination. Record the highest count rate during the scan.

> *Note*
> *1. The instrument must be used in the geometry (or as close as practically possible) for which the instrument conversion factor was established.*
> *The filter material, therefore, should be as flat as possible for the scan.*
> *2. Care must be taken to ensure that the instrument is allowed to reach equilibrium before the reading is made.*

13. Removable contamination test evaluation:
    - 13.1 If the net count rate corresponds to an activity less than 5 nCi (185 Bq), the test is negative. If the test is negative, no further action is required except proper record-keeping. Retain all wipes/filters for the possibility of further testing.
    - 13.2 If the net count rate corresponds to an activity more than 5 nCi (185 Bq), the test is positive. If the test is positive, inform the local Radiation Safety Officer and MDS Nordion for further disposition.
    - 13.3 If the net count rate is greater than 0.5 nCi, but less than 5 nCi, contact MDS Nordion. Do not proceed further until authorized to do so.
14. Remove the brass adapter from the end of the lower drain tube of the shipping container.
15. If the F-294 package is to be transferred to the pool in the irradiator shield building, proceed to Section 5 of this procedure.
16. If the task is delayed, the cavity must be properly drained and purged with Argon before vent and drain cap may be replaced.

7.1.6.1 Notifying MDS Nordion of Deficiencies

MDS Nordion, Ottawa, Ontario, Canada, or its agent, shall be notified immediately if any of the following deficiencies are evident.

1. Deficiencies of the F-294 on arrival at the customer's site B, including:
   - a) Radiation levels over 200 mrem/h at the surface of the transport package.
   - b) Transport Index (TI) greater than 10.
   - c) Non-fixed contamination from the external surface showing a wipe reading greater than 3 nCi.
2. Drainage of discolored water from the container. If this occurs, a sample of water shall be taken and returned to MDS Nordion for analysis.
3. Contamination found to be above 0.5 nCi when using the cavity water flush test procedure.
4. Any other abnormalities that are indicated when following the step-by-step procedures.

7.1.7 PREPARATION FOR SHIPMENT OF LOADED F-294

1. After a specified waiting period for F-294 as per Figure 7.1-F3, a final argon purge of the F-294 cavity can begin. The following steps outline the procedure for purging the F-294 cavity with argon and the criteria for ensuring all the water has been drained from the F-294 cavity.
   - 1.1 Open the closures on vent lines and drain line.
   - 1.2 Connect the equipment as per Figure 7.1-F2.
   - 1.3 Open Valve VX to set argon pressure at 30 psi and let argon flow from the center vent line connection through the F-294 cavity.
   - 1.4 Check for water or water spray flowing from the drain line outlet.

1.5 Repeat blowing argon until there is no water spray exiting from drain line outlet. Use a rag or paper towel for detecting water.

1.6 Seal off drain line connection using gasket and cap. Use 50 ft.-lb. ± 5 ft.-lb. torque on the cap.

1.7 Disconnect the argon line to center ventline.

1.8 Seal off vent line connection using gasket, vent line shield insert and cap. Use 20 ft.-lb. ± 2 ft.-lb. torque on the cap.

2. Check all external surfaces on the package for contamination. The level of non-fixed contamination shall be determined by wiping an area of 300 cm^2 of the external surface by hand with a dry filter paper or a wad of dry cotton wool or any other material of this nature. The maximum permissible level of removable contamination on the wipe is 0.37 Bq/cm^2 (10^{-5} $\mu Ci/cm^2$). This translates to a wipe reading of 110 Bq (3 nCi).

3. Perform a radiation survey of the assembled package. Radiation levels shall not exceed 2 mSv/h (200 mrem/h) on the external surface of the package or 0.1 mSv/h (10 mrem/h) at any point one meter from the surface of the package. See Table 7.1-T1.

4. Secure the cylindrical fireshield. Torque each bolt to 200 ± 20 ft.-lb. (272 ± 27 N·m).

5. Secure the crush shield on the package using sixteen (16) fasteners.
   1. Secure with eight (8) top fasteners using torque of 200 ± 20 ft.-lb. (272 ± 27 N·m) on each fastener.
   2. Secure with eight (8) side fasteners, using 50 ± 5 ft.-lb. (68 ± 7 N·m) on each fastener.

6. Heat screen is not required as F-294 accessible surface temperatures are less than 82°C (182°F) and as F-294 package is transported as "exclusive use" shipment.

7. Insert two wire seals through the crush shield and container ring. Join the wire with the lead pellet.

8. Secure the container to the shipping skid. Torque each bolt to 200 ± 20 ft.-lb. (272 ± N·m).

9. Attach the completed "DANGER - This Package is Loaded with High Activity Source" placard.

10. Affix Category labels as per Table 7.1-T1.

11. Affix one UN2916 Label next to each of the Radioactive Category labels.

12. Transport vehicles and freight containers carrying radioactive material transport packages must display placards in accordance with the applicable transport regulations. In case of road transport within North America, the trailer of the transport vehicle must display placards on both sides, and front and rear, indicating that it carries radioactive materials.

Table 7.1-T1
Package Label Requirements

Label	Radiation Level at External Surface of Package	Transport Index (T.I.)[1]	Radiation Level at External Surface of Vehicle
Radioactive I (white)	≤ 5.0 μ Sv/h (0.5 mrem/h)	-	-
Radioactive II (yellow)	>5.0 μ Sv/h (0.5 mrem/h) ≤ 500 μ Sv/h (50 mrem/h)	≤1.0	
Radioactive III (yellow)	>500 μ Sv/h (0.5 mrem/h) ≤ 2,000 μ Sv/h (200 mrem/h)	> 1 and ≤10.0	
Radioactive III (yellow)[2]	>2,000 μ Sv/h (200 mrem/h) ≤ 10,000 μ Sv/h (1,000 mrem/h)	–	≤ 2,000 μ Sv/h (200 mrem/h)

[1] T.I. - Radiation level in microsieverts per hour at 1m from the external surface of the package divided by 10 (mrem/h at 1 m).

[2] Exclusive use condition, road transport only. For further information on road transport, see IAEA TS-R-1, *Regulations for the Safe Transport of Radioactive Material*, 1996 Edition (Revised).

7.1.8 INSTRUCTIONS FOR SECURING THE PACKAGE ON ROAD VEHICLES

1. The F-294 transport package has been designed so that it can be secured to the transport vehicle. As the transport package is a heavy load, local regulations relevant to the security of the load during transport may apply. Due to the high heat content of the F-294 transport package, the following is mandatory:
   - it is prohibited to cover the F-294 transport package
   - it is prohibited to transport F-294 transport package in a closed vehicle.
2. The F-294 transport package should be positioned on the vehicle bed with skid channels parallel to the direction of travel. Shocks should be used at the base of the skid channels (front and back, in the direction of travel). These should be firmly fastened to the bed of the vehicle as described in MDS Nordion Specification IN/GI 0006 Z000 (Ref. [4]). Section 2.3 (attachment) and 2.4 (lumber).
3. Bracing, if applicable, shall be in accordance with regulations of the State from where the shipment originates.
4. If the package is tied down (rather than braced), one-inch shackles with load binders or turnbuckles and minimum 3/8 in. chains shall be used. The angle of the chain to the vertical should be between 50 and 60 degrees.
5. Tension the chains equally, to the point that each one is taut, with all visible sag removed.
6. The appropriate reference documents may be supplied to the carrier by the shipper, if not already in their possession. Other guidelines and regulations may apply in other jurisdictions.

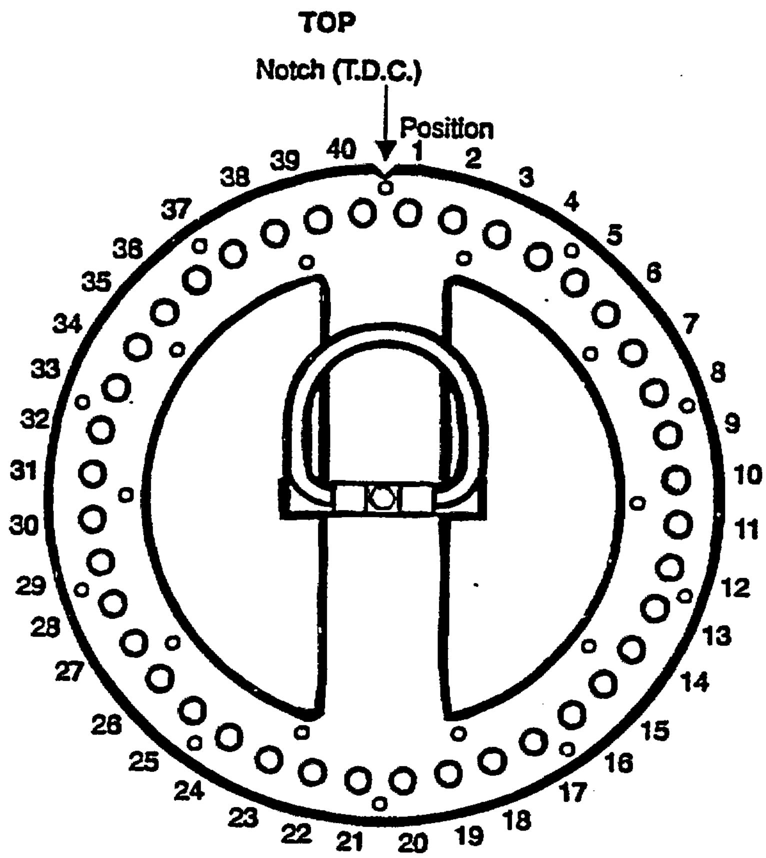
7.1.9 ADDITIONAL INSTRUCTIONS

1. Any other additional instructions with respect to the shipment as per USNRC Certificate of Compliance shall be applicable.
2. The following documents shall be approved prior to the shipment departure.
   a) Release for Shipment
   b) Radiation Survey
   c) Record of C-188 Sources, Activity and Serial Numbers. Ensure that the total activity is less than the licensed capacity of the F-294 transport package.
3. Appropriate documents shall be provided to the carrier or his agent.
4. F-294 transport package shall be transported as "exclusive-use" shipment.
5. Any instruction with respect to the safe dissipation of heat.

7.1.10 LOADING PROCEDURE

This loading procedure is formalized in the MDS Nordion Operating Procedure IN/OP 0283 F294 (Ref. [1]) and only persons properly trained and authorized to handle the F-294 transport package are permitted to carry out this work. This ensures the effectiveness of the operating procedure and thereby the safety of the package.

Figure 7.1-F1a
Loading Diagram for the F-313 Source Carrier



Notes:

1. Mark hole with X if hole position is empty.
2. Mark hole with C-188 S/N if hole position is filled.

CONTAINER NO:

CARRIER NO:

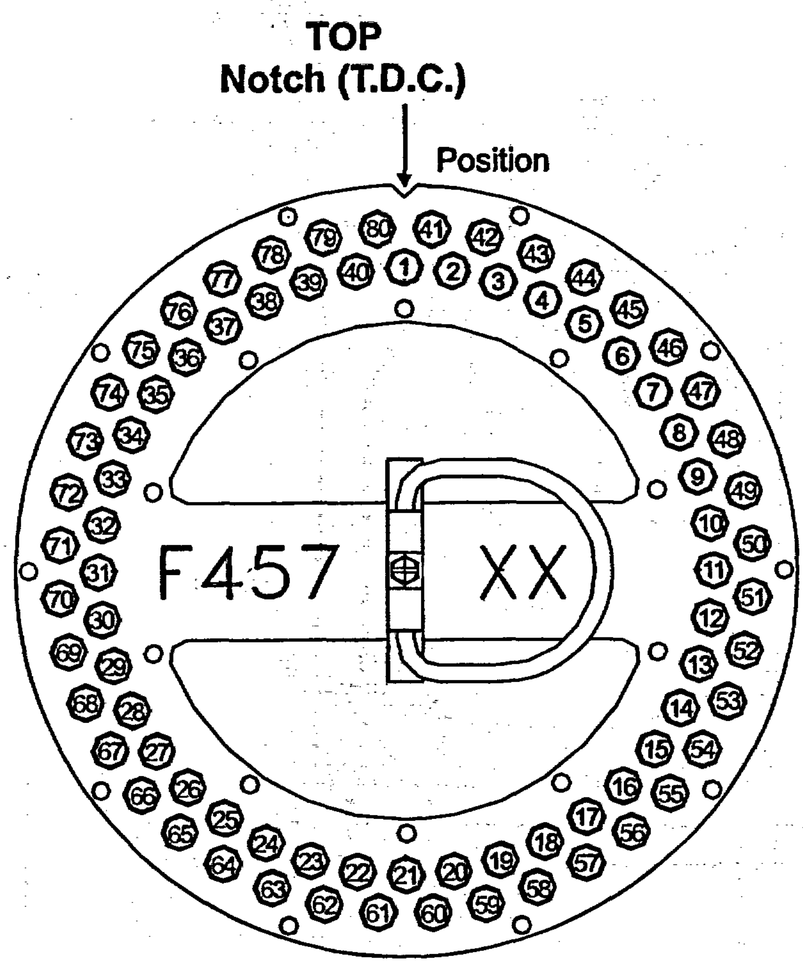
LOADING DATE

P&S NO:

WARRANTY YRS.

ACTIVITY CURIES

Figure 7.1-F1b
Loading Diagram for the F-457 Source Carrier



Notes:

1. Mark hole with X if hole position is emply.
2. Mark hole with C-188 S/N if hole position is filled.

CONTAINER NO:
CARRIER:
LOADING DATE:
P&S NO:
WARRANTY: **YRS.**
ACTIVITY: **CURIES**

Figure 7.1-F2
Argon Purging of F-294 Cavity

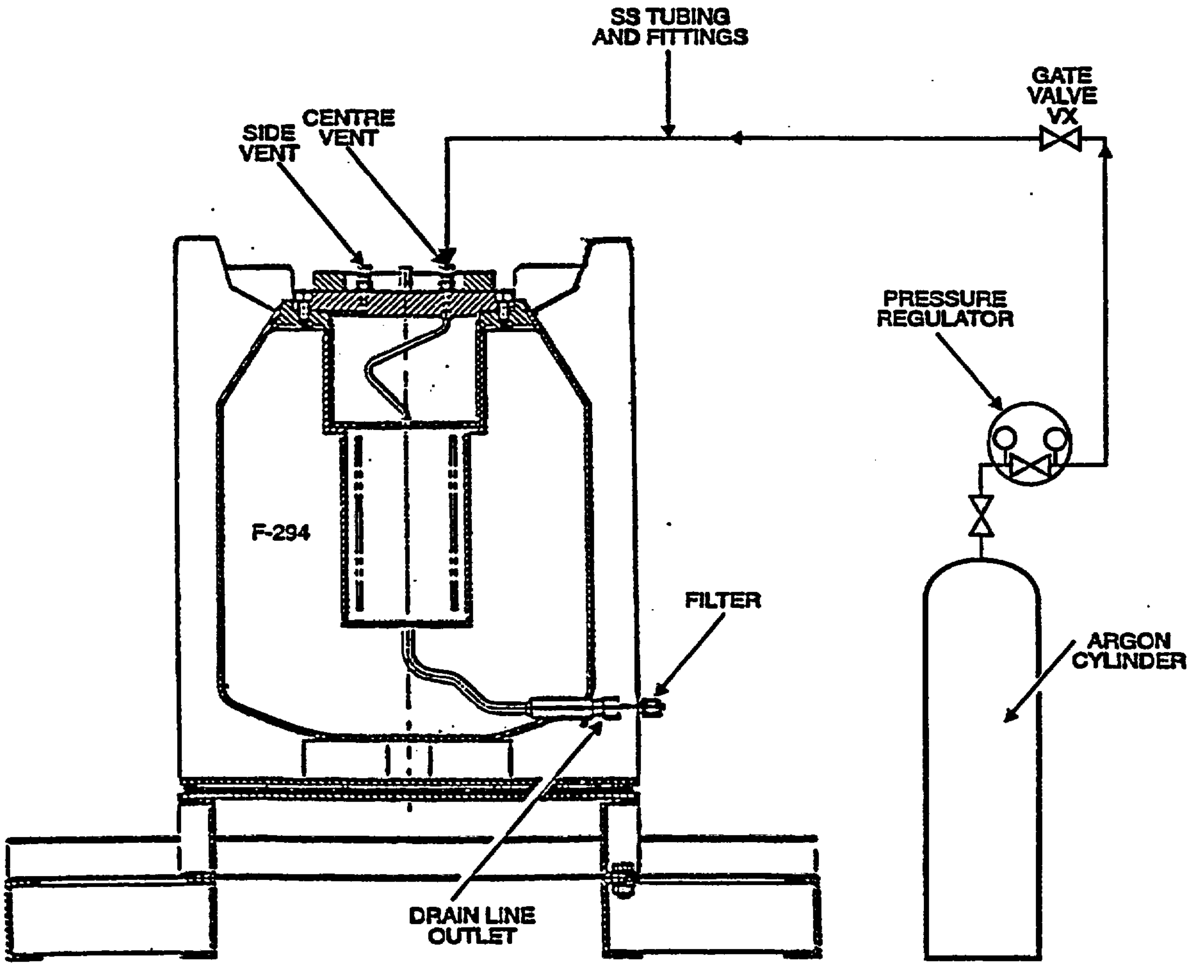
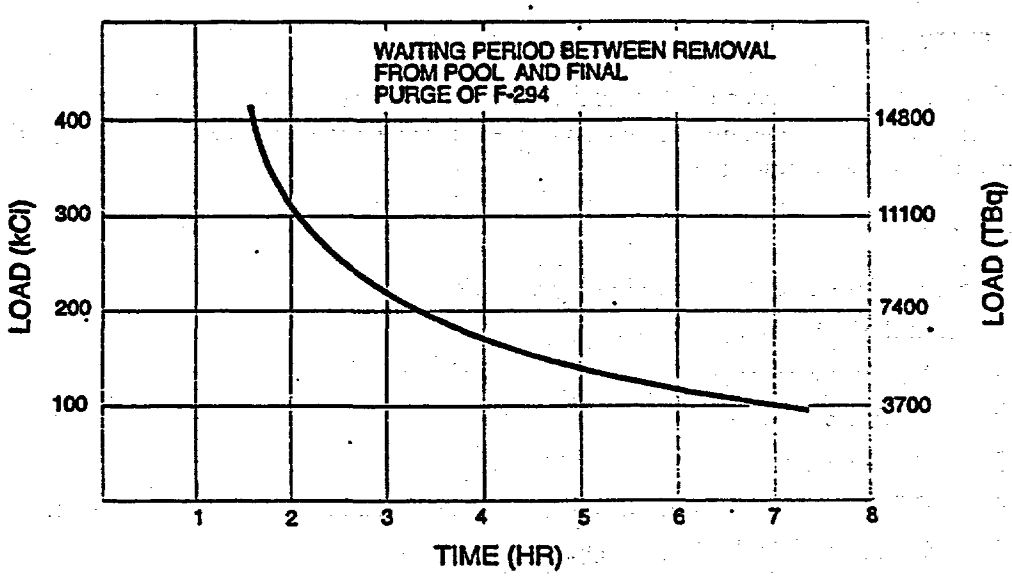


Figure 7.1-F3
Waiting Period Between Removal from Pool and Final Purge of F-294 Package



7.2 PROCEDURES FOR UNLOADING THE PACKAGE

The unloading of the C-188 sealed sources from the F-294 package at a customer's site B is carried out as follows:

7.2.1 PURPOSE

This operating procedure is to ensure that the underwater unloading of C-188 (cobalt-60) sources from the F-294 transport package at a customer's irradiator site B is in compliance with the design specifications, handling requirements and regulatory requirements.

7.2.2 SCOPE

This operating procedure describes the following operations:

1. Receipt of the F-294 transport package.
   - 1.1 Visual examination.
   - 1.2 Surface wipe test
   - 1.3 Radiation survey.
   - 1.4 F-294 cavity water flush test for contamination and to verify the source integrity.
   - 1.5 Cleaning of the F-294 package.
2. Transfer of loaded F-294 package to the bottom of the source storage pool.
3. Unloading of C-188s from the F-294 package.
4. Removal of the empty F-294 package from the pool.

These are the minimum requirements that must be achieved.

7.2.3 COMPLIANCE AND RESPONSIBILITY

1 It is the responsibility of MDS Nordion International personnel or its qualified agent to ensure that the operations described by these procedures are followed and the F-294 transport package is prepared for shipment in compliance with the regulatory requirements.

2. It is the responsibility of the customer at site B, in the role of the consignee, to accept the receipt of the F-294 shipment.

3. It is the responsibility of the pertinent regulatory authority to enforce compliance as per F-294 transport package license.

7.2.4 RECEIPT OF F-294 TRANSPORT PACKAGE

7.2.4.1 Visual Inspection

Visually inspect the F-294 transport package for damage and deterioration. Damage and deterioration, if any, are designated as either superficial or integrity-related.

The fireshield, the crush shield and the packaging are to be inspected for the quality of the paint finish, punctures and dents, and cracks or corrosion of welds.

Check and verify that the tamper-proof seal is intact. If the tamper-proof seal is not intact, contact the RSO at the customer's site or MDS Nordion for further disposition.

> *Note*
> *Immediately contact MDS Nordion's Package Engineering Group regarding any damage or deterioration that may be integrity-related.*

Ensure that any damage or deterioration is clearly documented in the Installation Report.

7.2.4.2 Surface Wipe Test

Check all external surfaces on the package for contamination. The level of non-fixed contamination shall be determined by wiping an area of 300 cm^2 of the external surface by hand with a dry filter paper or a wad of dry cotton wool or any other material of this nature. The maximum permissible level of removable contamination on the wipe is 0.37 Bq/cm^2 ($10^{-5}\mu Ci/cm^2$). This translates to a wipe reading of 110 Bq (3 nCi).

7.2.4.3 Radiation Survey

Perform a radiation survey of the assembled package. Radiation levels shall not exceed 2 mSv/h (200 mrem/h) on the external surface of the package or 0.1 mSv/h (10 mrem/h) at any point one meter from the surface of the package. See Table 7.1-T1.

7.2.4.4 Perform the Cavity Water Flush Test Procedure

1. Remove the crush shield from the top of the container. Remove the cylindrical fireshield from the container.
2. Remove the drainline cap and the top vent center cap and the vent-line shield plug from the container. If any water drains from the container, collect the water and check it for any contamination. Document and report the drainage of discolored water from the container.
3. Thread the brass adapter with the filter into top center vent plug.
4. Connect the brass adapter supplied to the lower drain tube of the F-294 package.
5. Close the spigot valve on the US 10-gallon plastic container, check all hose connections for tightness and fill 4/5th full with de-ionized water. Use of two (2) 5-gallon (US) plastic containers in place of one (1) 10-gallon (US) container is permitted.
6. With plastic tubing attached to each end of the filter, secure one end of plastic tube to the lower draintube adapter on the F-294 package and the other end to the spigot on the 10-gallon (US) plastic container.

> *Note*
> *For optimum flow of the water into the cavity,*
> *open the spigot with the water-filled plastic container below the drain-tube level*
> *to allow water to fill the plastic tube,*
> *then raise the water-filled plastic container to the top of the transport package.*

7. Support the bottom of the water-filled plastic container above, or level with, the top of the F-294 package.
8. Remove the cap of the 10-gallon (US) plastic container and open the spigot, allowing water to slowly fill the F-294 package cavity.

> **Warning**
> **Steam and hot water may blow out of the vent holes as the cavity fills.**

9. Monitor the filter on the top vent hole during filling for an increase in the radiation fields. If the radiation field at the monitoring position increases, immediately close the spigot to prevent further entry of water into the cavity. Evacuate the area, taking care to prevent the possible spread of contamination. Restrict access to the area. Notify the facility's Radiation Safety Officer, the local competent authority and MDS Nordion.
10. As soon as the cavity and the vent tube(s) are full, lower the plastic container to the ground and allow all the water to drain back into the plastic container through the filter. Monitor the filter in the plastic tube with the survey meter.
11. When all the water has drained from the cavity, close the spigot on the plastic container, and disconnect the plastic tube from the lower drain tube adapter on the F-294 package and the brass adapter with the filter from the top center vent plug.
12. Remove the filters to a low background area and monitor the filters. Wearing protective gloves, carefully cut open each filter and remove the filter material. Place the filter material in a labeled and lockable plastic bag and slowly scan for radioactive contamination. Record the highest count rate during the scan.

> *Note*
> *1. The instrument must be used in the geometry (or as close as practically possible) for which the instrument conversion factor was established.*
> *The filter material, therefore, should be as flat as possible for the scan.*
> *2. Care must be taken to ensure that the instrument is allowed to reach equilibrium before the reading is made.*

13. Removable contamination test evaluation:

    13.1 If the net count rate corresponds to an activity less than 5 nCi (185 Bq), the test is negative. If the test is negative, no further action is required except proper record-keeping. Retain all wipes/filters for the possibility of further testing.

    13.2 If the net count rate corresponds to an activity more than 5 nCi (185 Bq), the test is positive. If the test is positive, inform the local Radiation Safety Officer and MDS Nordion for further disposition.

    13.3 If the net count rate is greater than 0.5 nCi, but less than 5 nCi, contact MDS Nordion. Do not proceed further until authorized to do so.
14. Remove the brass adapter from the end of the lower drain tube of the shipping container.
15. If the F-294 package is to be transferred to the pool in the irradiator shield building, proceed to Section 5 of this procedure.
16. If the task is delayed, the cavity must be properly drained and purged with Argon before vent and drain cap may be replaced.

7.2.4.5 Notifying MDS Nordion of Deficiencies

MDS Nordion, Ottawa, Ontario, Canada, or its agent, shall be notified immediately if any of the following deficiencies are evident.

1. Deficiencies of the F-294 on arrival at the customer's site B, including:
   a) Radiation levels over 200 mrem/h at the surface of the transport package.
   b) Transport Index (IT) greater than 10.
   c) Non-fixed contamination from the external surface giving a wipe reading greater than 3 nCi.
2. Drainage of discolored water from the container. If this occurs, a sample of the water shall be taken and returned to MDS Nordion for analysis.
3. Contamination found to be above 0.5 nCi when using the cavity water flush test procedure.
4. Any other abnormalities that are indicated when following the step-by-step procedures.

7.2.5 TRANSFER OF LOADED F-294 PACKAGE TO THE BOTTOM OF SOURCE STORAGE POOL

After successful completion of checking that the C-188 source integrity is sound, perform the following operations on the F-294 package.

7.2.5.1 Transfer F-294 from Outside to Inside the Irradiator Building

1. Remove the shipping skid.
2. Sling the F-294 to the crane.
3. Confirm that the source rack is lowered into the pool and fully disabled.
4. The F-294 package can now be transferred from the external grounds to the irradiator shielded building.
5. Remove the roof plug of the irradiator shield building.
6. Lower the F-294 package in the irradiator building into the pool.
7. Lift the shield plug clear of the container, giving access to the cavity of F-294.
8. Lift the source carrier out of the F-294 container cavity and place the source carrier at the bottom of the pool, clear of the container.
9. Replace the shield plug.
10. Lift the F-294 container and place it in the designated staging area outside the irradiator building.
11. If the F-294 container has to be returned empty to MDS Nordion, see Section 7.3.

7.2.6 EMERGENCY ACTION FOLLOWING A SUSPECTED RADIATION INCIDENT

1. Do not try to clean up the suspected contamination.
2. Leave the area of suspected high fields, taking care to prevent the possible spread of contamination. Post warning signs to restrict access to the area.
3. Check all operating personnel for possible contamination.

4. Inform the local Radiation Safety Officer, the pertinent regulatory authority, and MDS Nordion.
5. MDS Nordion will investigate every report of a suspected radiation incident. MDS Nordion may request the licensee to perform additional tests and arrange for qualified personnel to visit the site and assess the situation. MDS Nordion will confirm or disprove the presence of contamination, and report their findings to the pertinent regulatory authority.

7.2.7 UNLOADING PROCEDURE

This unloading procedure is formalized in the MDS Nordion Procedure IN/OP 0284 F294 (Ref. [2]) and only persons properly trained and authorized to handle the F-294 transport package are permitted to carry out this work. This ensures the effectiveness of the operating procedure and thereby the safety of the package.

7.3 PREPARATION OF AN EMPTY PACKAGE FOR TRANSPORT

From a customer's site, an empty F-294 package is prepared for shipment to MDS Nordion, Ottawa, Ontario, Canada as follows:

7.3.1 PURPOSE

This operating procedure is to ensure that the preparation for shipment of the empty F-294 transport packaging to MDS Nordion, Ottawa, Ontario, Canada, from a customer's irradiator site B is in compliance with the design specifications, handling requirements, and regulatory requirements.

7.3.2 SCOPE

This operating procedure describes the following operations:

- Preparation for shipment of the empty F-294 transport packaging.
- Instructions for securing the F-294 packaging on the road vehicle.

These are the minimum requirements that must be achieved.

7.3.3 COMPLIANCE AND RESPONSIBILITY

1. It is the responsibility of MDS Nordion personnel or its agent to ensure that the F-294 transport packaging is prepared for shipment in compliance with the regulatory requirements.
2. It is the responsibility of the pertinent regulatory authority to enforce compliance as per F-294 transport package license.

7.3.4 OPERATIONS ON THE EMPTY F-294 TRANSPORT PACKAGING

1. After the container is empty and on the trailer truck or in the designated staging area, monitor the radiation around the container to verify that it is definitely empty.

   1.1 Surface Wipe Test.

   Check all external surfaces on the package for contamination. The level of non-fixed contamination shall be determined by wiping an area of 300 cm^2 or the external surface by hand with a dry filter paper or a wad of dry cotton wool or any other material of this nature. The maximum permissible level of removable contamination on the wipe is 0.37 Bq/cm^2 ($10^{-5} \mu Ci/cm^2$).

2. Secure the shield plug. Torque each bolt to 100 ± 10 ft.-lb. (133 ± 13 N·m)
3. After purging, ensure that vent and drain lines are sealed.

4. Secure the cylindrical fireshield to the retaining bracket on the fixed skid. Torque each bolt to 200 ± 20 ft.-lb. (272 ± 27 N·m).
5. Secure the crush shield on the package using sixteen (16) fasteners.
   1. Secure with eight (8) top fasteners using torque of 200 ± 20 ft.-lb. (272 ± 27 N·m) on each fastener.
   2. Secure with eight (8) side fasteners, using 50 ± 5 ft.-lb. (68 ± 7 N·m) on each fastener.
6. Secure the container to the shipping skid. Torque each bolt to 200 ± 20 ft.-lb. (272 ± N.·m).
7. Cover the "Radiation Caution" plates with the "EMPTY" labels.
8. Remove the Category III labels.
9. Affix MDS Nordion return address labels on two opposite sides of the container.
10. The F-294 packaging is now ready for EMPTY shipment to MDS Nordion.

7.3.5 INSTRUCTIONS FOR SECURING THE EMPTY F-294 ON ROAD VEHICLES

1. The F-294 transport package has been designed so that it can be secured to the transport vehicle. As the transport package is a heavy load, local regulations relevant to the security of the load during transport may apply.
2. The F-294 package should be positioned on the vehicle bed with skid channels parallel to the direction of travel. Shocks should be used at the base of the skid channels (front and back in the direction of travel). These should be firmly fastened to the bed of the vehicle as described in MDS Nordion Specification IN/GI 0006 Z000, Section 2.3 (attachment) and 2.4.4 (lumber) Ref. [4].
3. Bracing, if applicable, shall be in accordance with Regulations of the State from where the shipment originates.
4. If the package is tied down (rather than braced), one-inch shackles with load binders or turnbuckles and minimum 3/8 in. chains shall be used. The angle of the chain to the vertical should be between 50 and 60 degrees.
5. Tension the chains equally, to the point that each one is taut, with all visible sag removed.
6. The appropriate reference documents may be supplied to the carrier by the shipper, if not already in their possession. Other guidelines and regulations may apply in other jurisdictions.

7.3.6 ADDITIONAL INSTRUCTIONS

1. Any other additional instructions with respect to the shipment as per USNRC Certificate of Compliance shall be applicable.
2. "Release for Shipment" document for the empty F-294 shall be approved prior to the shipment departure.
3. Appropriate documents shall be provided to the carrier or his agent.

7.3.7 OPERATING PROCEDURE

The procedure for handling the F-294 package is formalized as MDS Nordion Operating Procedure IN/OP 0285 F294 (Ref. [3]) and only persons properly trained and authorized to handle the F-194 transport package are permitted to carry out this work. This ensures the effectiveness of the operating procedure and thereby the safety of the package.

7.4 REFERENCES

[1] MDS Nordion Procedure IN/OP 0283 F294, "Operating Procedure for the Underwater Loading and the Preparation for Shipment of the F-294 Transport Package from the Customer's Site A in the USA".

[2] MDS Nordion Procedure IN/OP 0284 F294, "Operating Procedure for the Underwater Unloading of the F-294 Transport Package at the Customer's Site B in the USA".

[3] MDS Nordion Procedure IN/OP 0285 F294, "Preparation for Shipment of the Empty F-294 Transport Package".

[4] MDS Nordion Specification IN/GI 0006 Z000, "Guidelines for Securing Radioactive Packages Shipped by Road".

CHAPTER 8 – ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

This chapter discusses the acceptance test and maintenance program used on the F-294 transport package, in compliance with the applicable subsections of 10 CFR Part 71. MDS Nordion Inc. has a quality assurance program in place governing all aspects of the F-294 packaging (design, manufacturing, testing, use, inspection and maintenance etc.) which meets the requirements of 10 CFR Part 71 and is approved by USNRC: (Quality Assurance Program Approval No. 0703, docket No. 71-0703, expiration date July 31, 2005).

8.1 ACCEPTANCE TESTS

All inspections and tests of the F-294 package, prior to its first use, are an integral part of the manufacturing process as described in the Technical Specification included in Chapter 1, Appendix 1.3.1 to this report.

8.1.1 VISUAL INSPECTION

The package is visually examined for any non-conformance in materials or fabrication using applicable codes, standards and drawings. In particular it is ensured that:

1. there is no interference fit between parts other than that called for in the engineering drawings, (see Chapter 1, Appendix 1.3.1, Technical Specification for F-294).
2. all required fasteners are in place and properly installed; and
3. all safety features are in place.

8.1.2 STRUCTURAL AND PRESSURE TESTS

Inspections and tests to ensure the structural integrity of the F-294 package are an integral part of the manufacturing and quality assurance program. All critical materials, components, welding supplies, fasteners etc. are subject to the quality program and in particular the requirements of the F-294 Technical Specification in Chapter 1, Appendix 1.3.1 All critical welds surrounding the lead shielding are inspected by radiography and/or by liquid penetrant examination procedures. All critical components and subassemblies are inspected to engineering drawings and/or specifications at key points in the manufacturing process.

The F-294 cavity is pressure tested at an internal pressure of 45 psig using air at 20°C. The purpose of this pressure test is primarily to detect any leaks in the cavity welded structure and drainline assembly. Should an inspection or test fail to meet the prescribed criteria, the quality assurance program (i.e., non-conformances) and section 7 of the F-294 Technical Specification (Chapter 1, Appendix 1.3.1) formally describe the corrective action to be taken.

8.1.3 LEAK TESTS

The following leak tests are specified for the F-294 container.

1. The leak test of the cavity of F-294 container assembly using air at 45 psig. at 20°C. (See Appendix 8.3.2, MDS Nordion Procedure IN/MP 0019 Z000).
2. The leak test of the cylindrical fireshield using air at 10 psig at 20°C. (See Appendix 8.3.2, MDS Nordion Procedure IN/MP 0019 Z000).
3. The leak test of the cavity of the F-294 container assembly using helium at 14.7 psig to meet acceptance standard of leak rate of less than 10^{-4} atm cc/sec at 20°C (see Appendix 8.3.4, MDS Nordion Procedure for Helium Leak Test of F-294 Cavity).

8.1.4 COMPONENT TESTS

8.1.4.1 Valves, Rupture Discs, and Fluid Transfer Devices

This section is not applicable since there are no valves, rupture discs or fluid transfer devices on the F-294 package.

8.1.4.2 Gaskets

There are the following gaskets on the F-294 package:

1. "Neoprene" gasket between the shield plug and the container body.
2. "Cajon" nickel gasket in the vent blind cap closures.
3. "Neoprene" gasket in the drain blind cap closure.

The F-294 container undergoes inspection and maintenance prior to each shipment from MDS Nordion, Ottawa, Ontario, Canada. The gaskets are visually examined for defects. The seal surfaces are visually examined for nicks or damage. New gaskets are installed on the F-294 during the regular or annual inspection and maintenance procedure.

8.1.4.3 Miscellaneous

The F-294 package is designed and has been tested and analyzed to demonstrate that it meets all requirements of use and safety prescribed by the regulations when used in the intended manner. The quality program governing all aspects of the F-294 package ensure that it is both manufactured and used in compliance with the prescribed requirements. Therefore, the requirements of this section are met in that there are no additional components not already considered whose failure would impair the package effectiveness.

8.1.5 TESTS FOR SHIELDING INTEGRITY

The F-294 package is tested to ensure it meets the requirements for radiation shielding as specified in section 7 of the Technical Specification, included as Chapter 1, Appendix 1.3.1 of this report. The shielding test is performed by MDS Nordion as per MDS Nordion Specification CO-QC/TP-0001 (2), attached in Appendix 8.3.1. The acceptance criteria prescribed is:

1. maximum of 200 mrem/h at any accessible external surface of the F-294 package, loaded with 360 kCi of cobalt-60, AND
2. maximum of 10 mrem/h at 1 meter from any accessible external surface of the F-294 package, loaded with 360 kCi of cobalt-60.

Extrapolation of dose rates is permitted provided the test source is not less than 60% of the maximum license limit of the package.

If the results of the test do not meet the acceptance criteria prescribed, then section 7 of the Technical Specification (Chapter 1, Appendix 1.3.1) and the general Quality Assurance Program describe the actions to be taken.

There are no neutron sources in the F-294 package.

8.1.6 THERMAL ACCEPTANCE TESTS

A thermal acceptance test is required for each F-294 package before first use. A thermal acceptance test of the F-294 package before first use shall be carried out as per Appendix 8.3.3., MDS Nordion Procedure IN/OP 0597 F294. See Chapter 3 for a full discussion of the thermal design and performance of the F-294 package.

8.1.6.1 Discussion of Test Setup

See Appendix 8.3.3, MDS Nordion Procedure. See Figure 8-F1.

8.1.6.2 Test Procedure

See Appendix 8.3.3, MDS Nordion Procedure.

8.1.6.3 Acceptance Criteria

8.1.6.3.1. Acceptance or rejection of test data

The following acceptance criteria assumes a 360 kCi of cobalt-60 load; environment temperature of 38°C and no solar heat load. The results of the thermal test shall be extrapolated to these conditions by MDS Nordion.

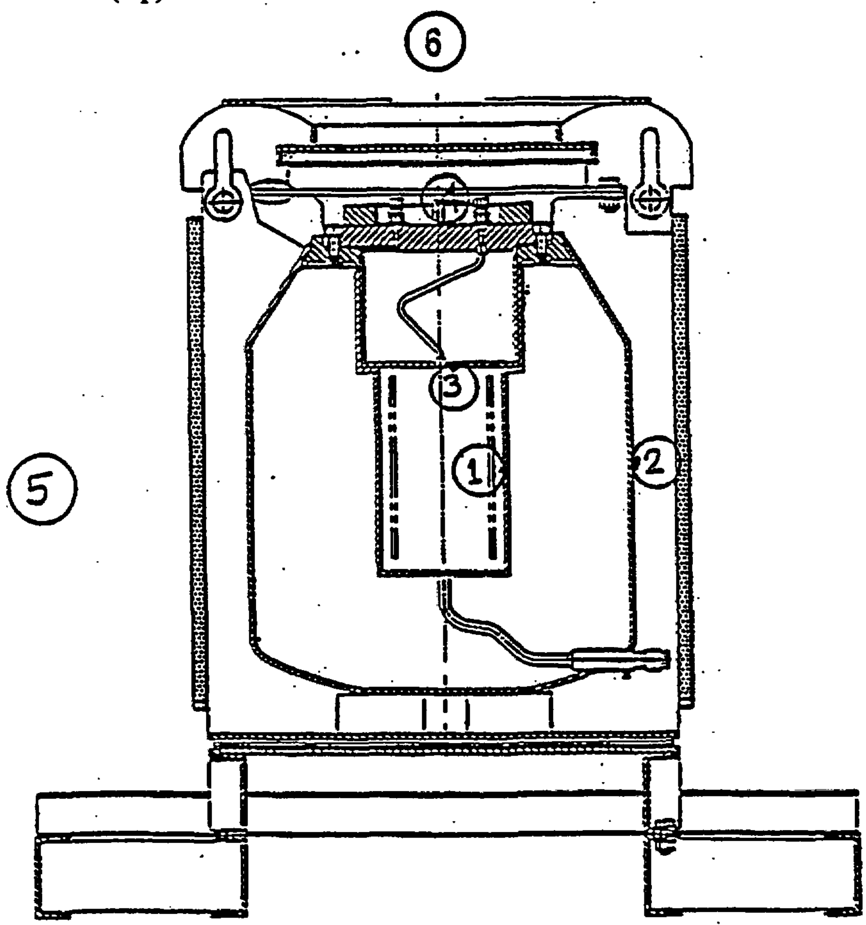
1. The maximum closure plug temperature at the bottom shall be less than 230°C.
2. The temperature gradient between the top and bottom surfaces of the closure plug shall not exceed 100°C.
3. The maximum cavity wall surface temperature, at mid-height of the cavity, shall not exceed 200°C.
4. The radial temperature gradient at the mid-height of the lead-shielded cask shall not exceed 80°C.

Deviations from these criteria may be acceptable if the maximum lead temperatures do not exceed 310°C, when the unit is subjected to the hypothetical fire test in accordance with the thermal model presented in Chapter 3, Appendix 3.6.4 for the F-313 source carrier and Appendix 3.6.7 for the F-457 source carrier.

In such a case, the deviations shall be documented and approved in accordance with the Quality Assurance program as provide in Chapter 9, Appendix 9.3.2.

Figure 8-F1
Locations of Thermocouples on F-294 Transport Package

T/C 1	=	Cavity Wall. Drainline Side
T/C 2	=	Outside Wall of Container, Mid Height
T/C 3	=	Underside of Closure Plug
T/C 4	=	Top of Closure Plug
T/C 5	=	Ambient (side)
T/C 6	=	Ambient (top)



8.2 MAINTENANCE PROGRAM

After the F-294 is "IN-SERVICE", this section describes the maintenance program used to ensure the continued performance of the F-294 package. The F-294 package is inspected and maintained prior to every reload of cobalt-60 sources. The inspection and maintenance is carried out as per MDS Nordion Procedure IN/MP 0019 Z000, attached in Appendix 8.3.2.

8.2.1 STRUCTURAL AND PRESSURE TESTS

Prior to shipment of the package, all critical components are visually inspected to ensure that they are undamaged and continue to meet the requirements of the applicable engineering drawings and specifications. The pressure tests of the F-294 cavity and the cylindrical fireshield are carried out.

8.2.2 LEAK TESTS

The following leak tests are carried out for the F-294 container:

1. The leak test of the cavity of F-294 using air at 45 psig. at 20°C. (See Appendix 8.3.2, MDS Nordion Procedure IN/MP 0019 Z000).
2. The leak test of the cylindrical fireshield using air at 10 psig at 20°C. (See Appendix 8.3.2, MDS Nordion Procedure IN/MP 0019 Z000).

8.2.3 SUBSYSTEM MAINTENANCE

The F-313 or F-457 source carrier for C-188 sources is thoroughly cleaned and decontaminated after each use.

8.2.4 VALVES, RUPTURE DISCS, AND GASKETS ON THE CONTAINMENT VESSEL

Some of the requirements of this section are not applicable to the F-294 package since it does not have any valves, rupture discs on its containment system.

8.2.4.1 Gaskets for Containment System (F-294 Inner Shell Assembly)

The F-294 container undergoes inspection and maintenance prior to each shipment from MDS Nordion, Ottawa, Ontario, Canada. The gaskets are visually examined for defects. The seal surfaces are visually examined for nicks or damage. New gaskets are installed on the F-294 during the regular or annual inspection and maintenance procedure.

The replacement schedule for gaskets is as follows:

1. Neoprene gasket container to plug seal joint: Prior to each round trip of F-294.
2. Nickel gaskets vent-line caps: Prior to each round trip of F-294.
3. Neoprene gasket drain line cap: Prior to each round trip of F-294.

8.2.5 SHIELDING

The shielding tests used prior to any shipment of the F-294 package are the same as those used for the initial shipment as described in section 8.1.5 of this report.

8.2.6 THERMAL

Once every five (5)-year period of operation after F-294 is declared in-service, a heat removal capability of the package shall be established using procedure in Appendix 8.3.3. The heat removal capability of the F-294 shall be established by comparing the baseline temperature map of the F-294 before 1st use and the temperature map of the F-294 every 3 years thereafter.

8.2.7 MISCELLANEOUS

This section is not applicable to the F-294 package.

8.3 APPENDICES

Appendix 8.3.1	MDS Nordion Specification, CO-QC/TP-0001, Procedure for Radiation Integrity of New Shipping Containers
Appendix 8.3.2	MDS Nordion Procedure, IN/MP 0019 Z000, Radioactive Material Transport Packaging Inspection and Maintenance Procedure
Appendix 8.3.3	MDS Nordion procedure, IN/OP 0597 F294, Procedure for F-294 Steady State Thermal Test
Appendix 8.3.4	MDS Nordion Procedure, IN/OP 0598 F294, Procedure for Helium Leak Test of F-294 Cavity

APPENDIX 8.3.1
MDS NORDION PROCEDURE FOR RADIATION INTEGRITY OF NEW SHIPPING CONTAINERS

Radiation Integrity for new Transport Packages Procedure

1. SCOPE

1. This specification covers the radiation survey conditions, instruments and procedures required for verifying the radiation shield integrity of Transport Packages for transporting sources, principally Cobalt 60.

2. PURPOSE

To ensure that the dose-equivalent rate when loaded with the design quantity of activity, does not exceed the limits specified in "IAEA TS-R-1 - Regulations for the Safe Transport of Radioactive Material - 1996 Edition (Revised)". The survey meter described in Section 5.1 is used to verify these limits.

and/or

To provide data to verify shielding calculations. In this case the technical data in the design specifications identifies the amount of Cobalt 60 to be used to verify the Transport Packages shielding capacity for other radionuclides.

3. EQUIPMENT REQUIRED

1. Radiation survey meters approved for use in surveying Transport Packages are as follows:
    - Nuclear Enterprises survey meter PDM1 which has an ionization chamber volume of 450 cc and a cross section (window) of 100 cm^2 or a comparable instrument.
    - A Berthold Rato/F, Bicron Surveyor 2000 or a comparable instrument. This meter is used to identify any crack leakage.
    - Survey meters (equivalent in area and sensitivity) to either of the above having detector cross sections approaching but not exceeding 100 cm^2 and 10 cm^2, for the 100 cm and 5 cm measurement positions respectively. This meter is used to qualify section 2.1.
    - GM Tube and Ion Chamber type instruments are used for all field measurements.

Note:

1. For the 5 cm field measurement, it is acceptable for the survey meter to be in intimate contact with the Transport Package.
2. A one meter stick should be used for maintaining an accurate distance from the surface of the Transport Package.
3. Kodak Ready Pack X-ray film, type Industrial AA for identifying the distribution of leakage radiation at any hot spot.
4. Radiation Survey Format.

4. PREPARATION

1. Remove fireshield and plug shielding plates for crack leakage tests.

2. The instruments defined in 3.1 will be calibrated according to SERA-OP-001-009. Instruments should be calibrated one month prior to the survey. Ensure that the field reading has been adjusted, based on the calibration factor.

5. TRANSPORT PACKAGE SURVEY

5.1 Preliminary Survey

1. The Transport Package is loaded with one or more sources at 5% to 15% of the maximum limit and surveyed by the Industrial Operations Monitor. If surface fields of 100 mR/h or greater are detected, notify a Quality Control Technician and Package Engineering.

Note: Ensure that the fireshield and plug shielding plates (if applicable)have been removed.

5.2 Determination of Crack Leakage or Voids in Container Shielding

1. To achieve the purposes of the survey, the Transport Package must be loaded with at least 90% of the design capacity. The source will be that for which the Transport Package was designed. The radiation source will occupy a volume within the transport package's cavity as close as possible to the typical shipping configuration. Details of the source and all meter readings will be recorded on the pertinent Radiation Survey Format.

2. The Monitor will survey the container in the cell to ensure that it is safe to approach and will notify the Quality Control Technician of the location of the highest radiation fields on the container.

3. The first survey will be of the outer surface of the Transport Package, without the fireshield and plug shielding plates (if applicable). GM Tube and Ion Chamber measurements will be used. The instrument will be placed in between any external fins of the Transport Package if possible.

4. The entire surface of the Transport Package must be scanned. Because most Transport Packages are cylindrically symmetrical, a constant reading is expected at each particular elevation around the Transport Package. These elevations are identified on the appropriate Radiation Survey Format as 4, 5 and 6.

5. **Scanning the Cylindrical Surface** - By scanning the entire vertical surface, the highest readings at each elevation will be identified and recorded on the appropriate Radiation Survey Format.

6. **Scanning Top and Bottom Surfaces** - By scanning over the entire top and bottom surfaces, the highest readings at position 1, (through the Transport Package plug) position 2, (up the interface crack between the plug and body), position 3, (through the top surface of the body) and position 8, (through the bottom surface) will be identified and recorded on the appropriate Radiation Survey Format.

7. **Scanning Special Areas** - Special areas, such as the vent line exit, drain line exit, (position 7) will be monitored, and the readings recorded, if in excess of that from adjacent areas.

8. **Allowable Surface Radiation** - If any region exceeds 200 mrem/h (when the container is fully loaded), the need to radiograph that region using the x-ray film will be determined in conjunction with Physics and Package Engineering. The typical exposure time is calculated to produce 1R exposure.

9. X-ray film, (e.g., Kodak AA 14" x 17" X-ray) is normally used on the top surface and the bottom surface of the transport package. The film's exposure should be about 1 R. Record the following information on the film:

   - Date:
   - Transport Package type:
   - Transport Package #:
   - Top or Bottom:
   - Orientation:
   - Start Time of Exposure:
   - Finish Time of Exposure:
   - Source Activity: _____Ci
   - Source Type:

10. The developed films are retained by the Quality Control Technician.

5.3 Verification of Transport Package Requirements

1. The survey will be repeated as in steps 7.2.2 to 7.2.5 (positions 1 to 8) at the surface and at 1 meter from the surface using the instrument with a detector cross section of 100 cm^2 or less. The measurements will be recorded on the appropriate Radiation Survey Format. Similarly, any hot spots will be also recorded.

2. Allowable Radiation at 1 meter from Surface - With the transport package loaded to the designed capacity, the radiation fields at 1 meter must not exceed 10 mrem/h, (according to 2.1).

3. If any reading, when properly corrected for instrument calibration factor, exceeds 200 mrem/h on the surface or 10 mrem/h at 1 meter from the surface, the completed handwritten Radiation Survey Form together with the film records and a deviation report, will be brought to the immediate attention of Package Engineering and the Physics Department.

6. DOCUMENTATION REQUIREMENTS

1. The Radiation Survey Format must be completed in full by the Quality Control Technician, signed and dated.

2. Attach a copy of the instruments' calibration certificates to the Radiation Survey Format.

3. The Radiation Survey Report will be checked, signed and dated by the Quality Control Measurement Technician, a Radiation Physicist and Package Engineering as a complete and correct record of the unit survey. The Quality Control Measurement Technician retains a copy.

APPENDIX 8.3.2
MDS NORDION RADIOACTIVE MATERIAL TRANSPORT PACKAGING INSPECTION AND MAINTENANCE PROCEDURE

Radioactive Material Transport Packaging Inspection and Maintenance Procedure

1. PURPOSE

This procedure is to be followed to ensure that Type B radioactive material transport packagings are inspected and maintained in accordance with the design specifications and the regulatory requirements for safety in transport [1,2].

2. SCOPE

This procedure is applicable to all returnable packagings used by Industrial Operations.

3. GENERAL REQUIREMENTS

3.1 General

1. There are two types of inspections. Routine inspections are completed after every shipment. Annual inspections are completed at least once per year. For some components, they are completed after every shipment. Any component that has not undergone a required inspection shall not be shipped until the applicable inspection procedure has been successfully completed.
2. The inspection procedure shall be completed as specified on the procedure flowchart and in the applicable procedure.
3. Records of inspections are to be maintained on the ROMIS Container Management System by the Technician responsible for inspection and maintenance.
4. A tag system shall be used to provide visual approval status of each packaging. The tag system consists of any or all of the following:
   - Inspection satisfactory: Blue tag stating "Acceptable for Service" .
   - Maintenance/Repair: Red tag stating "Maintenance/Repair Required".
   - Not Contaminated: Yellow tag stating "No Removable Contamination Inside or Outside".
   - Cleaning: Pink tag stating "Container Cleaned".
   - Routine Testing: White tag stating "Routine Testing Completed".

4. INSPECTION PROCEDURE

In general, inspections are carried out in the four stages identified in Table 1.

Table 1
Inspection Stages

Inspection Type	Form Status
Receiving	R
Quarantine	Q
Decontamination	D
Annual	I

The following sections define the inspection requirements in general terms. The procedure is described in Figure 1.

4.1 Receiving Inspection

1. Receiving inspections take place after the package components have been checked for contamination and radioactive material. After receiving inspection the yellow "No Removable Contamination Inside or Outside" Tag is attached.

   If the component is not free of contamination and radioactive material, it is quarantined and decontaminated and/or emptied.

2. Any defects observed during a receiving inspection are logged on the inspection checklist. If defects are observed, a red "Maintenance/Repair" tag is attached to the component.

   Following receiving inspection, the packaging is transferred to the Decontamination Service Area, or the Container Maintenance Room.

4.2 Decontamination Inspection

1. The Decontamination Inspection checklist includes all outstanding items on the Receiving Inspection checklist, plus the additional requirements on the Decontamination Inspection Checklist. If applicable, a white "Routine Testing Completed" tag is attached.
2. If applicable, decontamination of the component is completed prior to the decontamination inspection. Once the component has been cleaned, the pink "Container Cleaned" tag is attached.
3. Any defects observed during the inspection are logged on the inspection checklist. If defects are observed, the red "Maintenance/Repair Required" tag is attached to the component.
4. If no defects are observed and if the component is not due for an annual inspection, the component is moved into "Available" status and the blue "Acceptable for Service" tag is applied.
5. All repair procedures are required to be documented on the Transport Packaging Repair Report (See reference 5), and require the approval of Package Engineering unless the repair has been authorized as a Pre-approved Repair.
6. Once repairs have been completed; the repaired area of the component is re-tested. If the component has been successfully repaired, the Transport Packaging Repair Report pertaining to the repair is forwarded to the Manager, Industrial Quality Control for a documentation and process review verification. Once the Manager, Industrial Quality Control has reviewed and accepted the repair outlined in the Transport Packaging Repair Report, the red "Maintenance/ Repair Required" tag is removed and the Transport Packaging Repair Report is stored in the Corporate Records Unit History File.

   If the component fails the repair inspection, a new IN/OP 0524 F000 F1 is generated and shall contain a reference to the previous IN/OP 0524 F000 F1. Then steps (5) and (6) are repeated until the component passes the inspection or is removed from service.

4.3 Annual Inspection

1. Annual inspection includes receiving and decontamination inspections plus the additional requirements defined on the component inspection checklist.
2. If no defects are observed, the annual inspection checklist is completed, the component is moved into "available status" and the blue "Acceptable for Service" tag is attached. If defects are observed, steps 3, 4 and 5 must be completed.
3. All repair procedures are required to be documented on the Transport Packaging Repair Report, and require the approval of Package Engineering unless the repair has been authorized as a Pre-approved Repair.

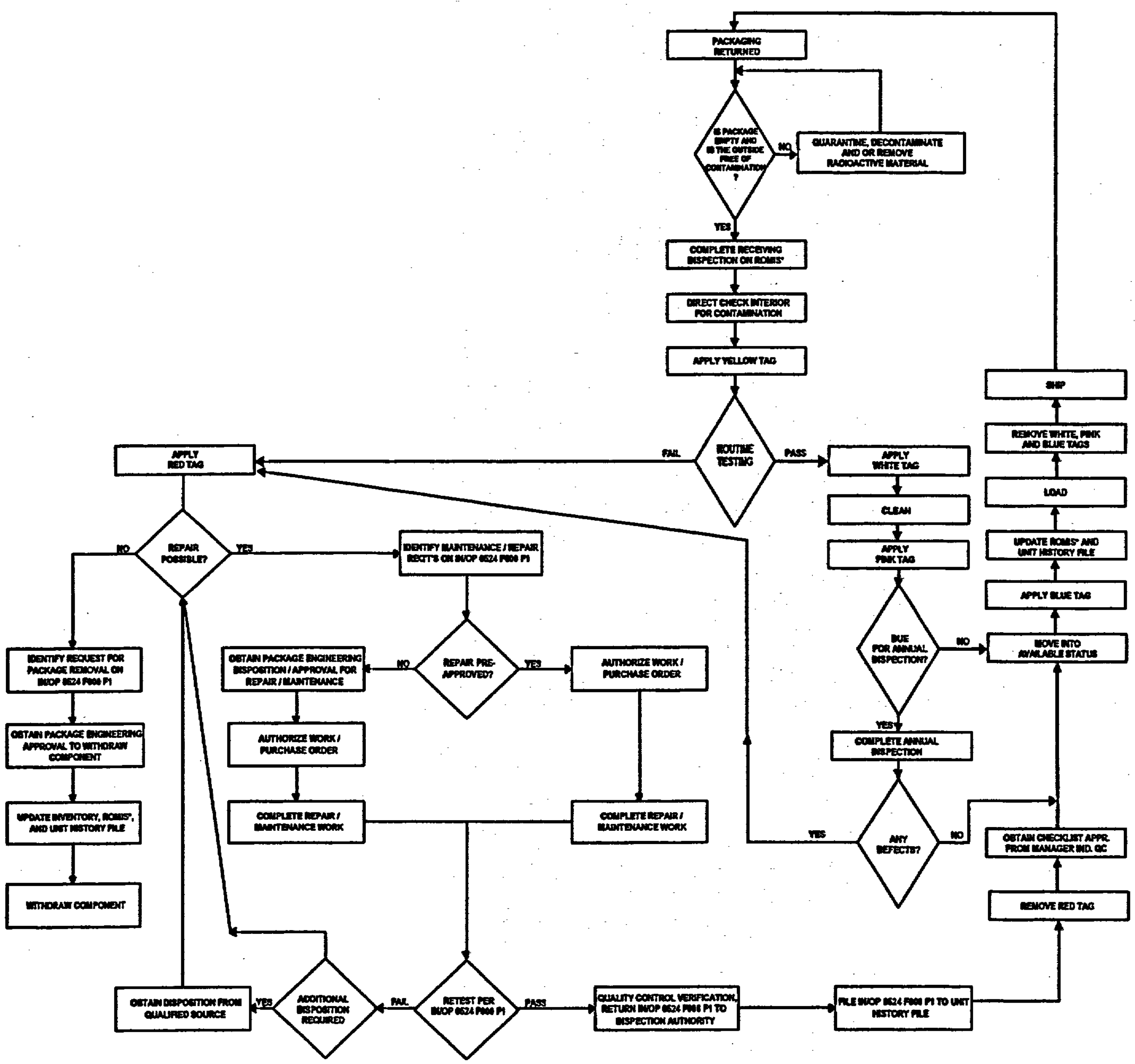
4. Once repairs have been competed; the repaired area of the component is re-tested. If the component has been successfully repaired, the Transport Packaging Repair Report pertaining to the repair is forwarded to the Manager, Industrial Quality Control for a documentation and process review verification. Once the Manager, Industrial Quality Control has reviewed and accepted the repair outlined in the Transport Packaging Repair Report, the red "Maintenance/ Repair Required" tag is removed and the Transport Packaging Repair Report is stored in the Corporate Records Unit History File.

   If the component fails the repair inspection, a new IN/OP 0524 F000 F1 is generated and shall contain a reference to the previous IN/OP 0524 F000 F1. Then steps (3) and (4) are repeated until the component passes the inspection or is removed from service. [Once the repair has been approved by Package Engineering, the red tag is removed. The Transport Packaging Repair Report and the final inspection checklist are submitted to the Supervisor, Industrial Quality Control for review and approval.

5. Following QC approval, the component is moved into "Available" status and the blue "Acceptable for Service" tag is attached.

5. DOCUMENTATION

Electronic copies of all inspection records are kept on the ROMIS container management system. There is no requirement to generate hard copies of inspection checklists.



* ROMIS: RADIOISOTOPE ORDER MANAGEMENT AND INFORMATION SYSTEM

FIGURE 1
Procedure Flowchart for the F-294

APPENDIX 8.3.3
MDS NORDION PROCEDURE FOR F-294
STEADY STATE THERMAL TEST

Procedure for the F-294 Steady State Thermal Test

1. INTRODUCTION

The F-294 steady state thermal test is a method for measuring the temperature of the F-294 container at specified locations. The temperature test information serves as a method of establishing heat removal capability of the F-294 container.

The decay heat shall be provided by using cobalt-60 MDS Nordion C-188 sealed sources. The F-294 shall be loaded between 100% to 110% of the rated capacity of 360 kilo-curies of cobalt-60.

2. EQUIPMENT AND FACILITY REQUIRED

A shielded cell facility and qualified operators are required before this test can be planned and undertaken. In addition the equipment consists of:

1. F-294 container test specimen
2. Type K Thermocouples
3. Temperature Recorder
4. C-188 Sealed sources
5. F-313 or F-457 source carrier

3. DIAGRAM

See Figure F1 for Location of thermocouples on F-294 transport package.

4. PROCEDURE

The standard operating procedures for the operation of a shielded cell in the Cobalt Operations Facility, MDS Nordion, Kanata are applicable. The following procedure outlines in general the steps required for loading the C-188 sealed sources in F-294, conducting the thermal test and unloading the sealed sources from F-294.

1. Plan the thermal test of F-294 with the co-operation of Cobalt Operations Department, MDS Nordion.
2. Load the C-188 sources in a F-294 container in a shielded cell. Record the time/date when sealed sources were loaded.
3. Install the closure plug. Torque 2 bolts out of 16. 100 ft.-lb. ± 10 ft-lbs torque on each bolt.
4. Remove the F-294 container from the shielded cell.
5. Ensure that the radiation fields are within allowable limits for working around the container.
6. Torque all 16 closure plug bolts . Use torque of 100 ft-lbs ± 10 ft-lbs.
7. Install the thermocouples to the specified locations on the container as shown in Figure F1.
8. Connect the thermocouples to the temperature recorder(s).
9. Set the recorder(s) to record each thermocouple at every 0.5 hour or 1.0 hour.
10. Continue the measurements for 48 hours or shorter period if the temperature data suggests that the temperature measurements have reached steady state.
11. Suspend the test. Disconnect the temperature instrumentation.

12. Return the container to the shielded cell. Unload sources.
13. Remove empty F-294 from the shielded cell.
14. Test complete.

5. TEST REPORT

A test report shall document the results of the test. It shall include the following information:

1. A brief description of the test specimen (container serial number).
2. Date of test; test conducted by (name of the operator);
3. Source loading chart, the number of curies; the number of C-188's, the location and their serial numbers; time/date when the F-294 was loaded.
4. The number of thermocouples & their locations; type of thermocouples;
5. The connection details of the thermocouple joints.
6. The measured temperature data; plots if any.
7. The conclusions:

1) The maximum & range of temperature measurement at each specified thermocouple location.

6. ACCEPTANCE OR REJECTION OF F-294 THERMAL PERFORMANCE

The following acceptance criteria assumes a 360 kCi of cobalt -60 load; environment temperature of 38°C and no solar heat load. The results of the thermal test shall be extrapolated to these conditions by MDS Nordion.

1) The maximum closure plug temperature at the bottom shall be less than 230°C.
2) The temperature gradient between the top and bottom surfaces of the closure plug shall not exceed 100°C.
3) The maximum cavity wall surface temperature (at mid-height of the cavity) shall not exceed 200°C.
4) The radial temperature gradient at the mid-heights of the lead-shielded cask shall not exceed 80°C.

Deviations from these criteria may be acceptable if the maximum lead temperatures do not exceed 310°C, when the unit is subjected to the hypothetical fire test in accordance with the thermal model presented in Chapter 3, Appendix 3.6.4 of Ref. [1]. In such a case, the deviations shall be documented and approved in accordance with the Quality Assurance program as provide in Chapter 9, Appendix 9.3.2 of Ref. [1].

7. REFERENCED DOCUMENTATION

1. SAR : IN/TR-9301-F294
2. Omega Temperature Recorder Operators Manual

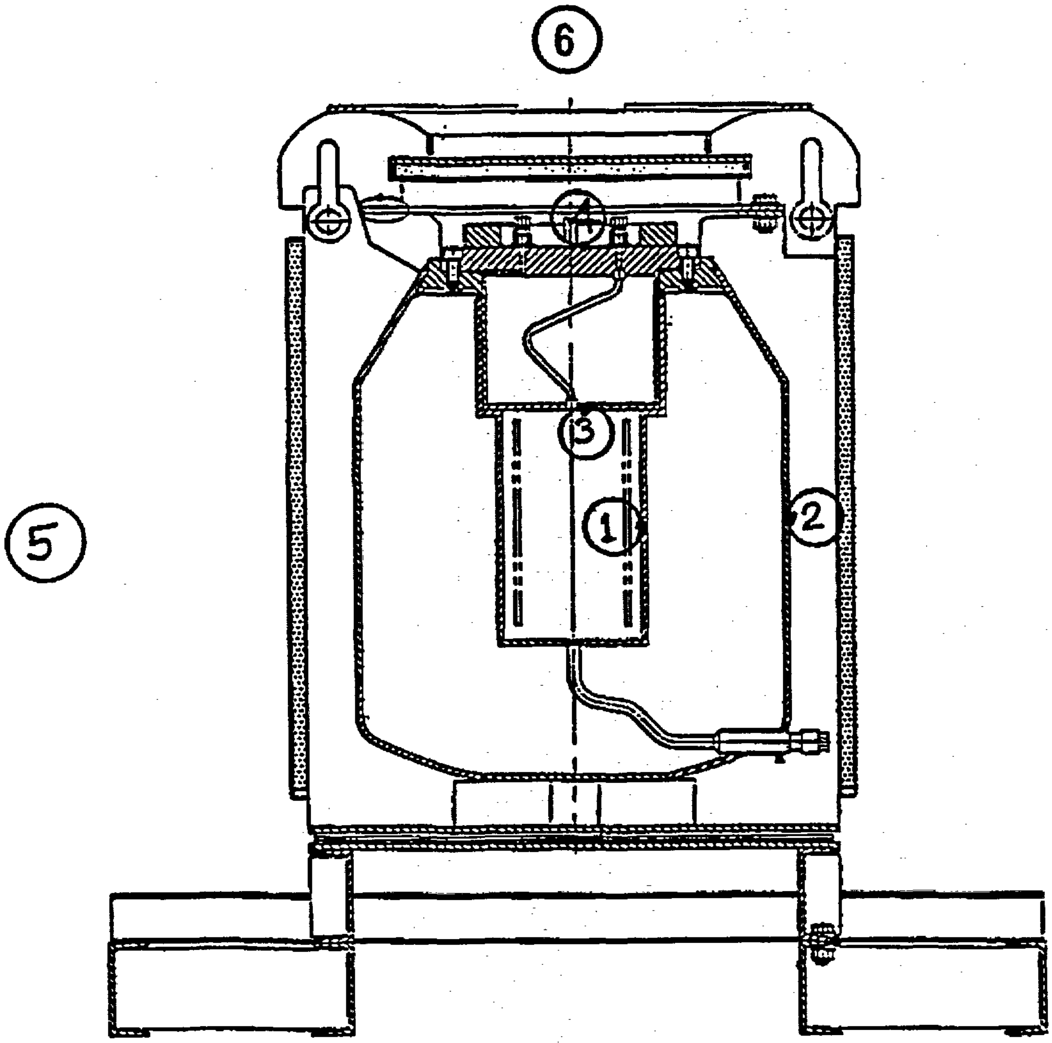


Figure F1
Locations of Thermocouples on F-294 Transport Package

T/C 1	=	CAVITY WALL. DRAINLINE SIDE
T/C 2	=	OUTSIDE WALL OF CONTAINER, MID HEIGHT
T/C 3	=	UNDERSIDE OF CLOSURE PLUG
T/C 4	=	TOP OF CLOSURE PLUG
T/C 5	=	AMBIENT (SIDE)
T/C 6	=	AMBIENT (TOP)

APPENDIX 8.3.4
MDS NORDION PROCEDURE FOR HELIUM LEAK TEST OF F-294 CAVITY

Procedure for Helium Leak Test of F-294 Cavity

1. INTRODUCTION

The Helium Leak Detection Sniffer Method Test (HLDSMT) is a simple method of identifying leaks in a containment system. The test is applicable to mostly large vessels where the completed evacuation would be time consuming. This test is applicable to sealed volumes greater than 1000cc.

2. EQUIPMENT & APPARATUS

The equipment & apparatus consists of:

1. Helium leak detector comes with sniffer probe.
2. Helium gas cylinder with regulator. Commercial grade helium is acceptable.
3. Valves, tubing, tube fittings.
4. The test specimen (i.e. the F-294 container).

3. DIAGRAM

See Figure 1.

4. PROCEDURE

1. Connect the equipment as shown in Figure 1. Seal and torque the joints appropriately:
    1) plug closure bolts: 100 ft-lbs ± 10 ft-lbs,
    2) vent line cap: 20 ft-lbs ± 2 ft-lbs,
    3) drain line cap: 50 ft-lbs ± 5 ft-lbs.
2. A positive pressure of helium 15 ± 10% psig is applied on the inside of the cavity of a container.
3. A probe is used to "sniff" the sealed joints of the cavity of the container.
4. Observation:
    1) If a leak exists the helium is "sniffed" and detected. It is relatively easy to pin point very small leaks by this test.

5. REPORT

A test report shall document the results of the tests. It shall include the following information:

1. A brief description of the test specimen including the type of seal, serial number of the container, closure plug bolt torques, vent line cap torque, drain line cap torque etc.
2. The date of the test.
3. The ambient temperature.
4. The pressure differential.
5. The length of time the specimen was examined.
6. The result of the test.
7. The operator's signature.
8. Any other relevant information.

6. ACCEPTANCE STANDARD

F-294 cavity assembly shall meet leak tightness to 1×10^{-4} atm. cc/sec at 20°C of Helium.

7. REFERENCES

1. ANSI-N14.5-88: American National Standard for Radioactive Materials - Leakage Tests on Packages for Shipment.

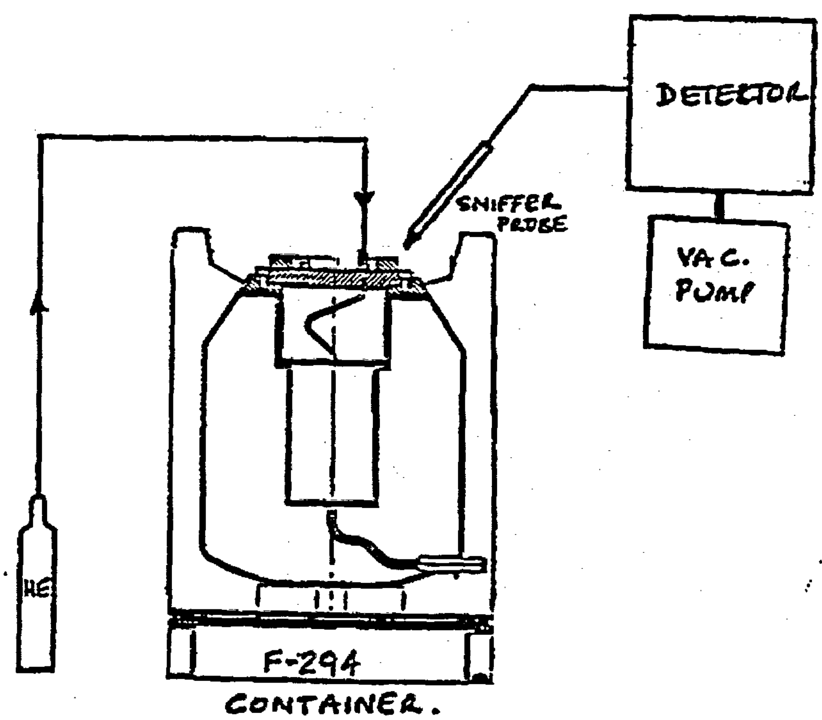


Figure 1:
Flow diagram for Helium Leak Detection sniffer Method Test(HLDSMT).

CHAPTER 9 – QUALITY ASSURANCE

This section describes the Quality Program in place at MDS Nordion Inc., Ottawa, Ontario, Canada, as it applies to the F-294 Transport Package in compliance with the applicable sections of 10 CFR Part 71.101 requirements. MDS Nordion Inc. has a quality assurance program in place governing all aspects of the Transport Packaging (design, manufacturing, testing, use, inspection and maintenance etc.) that meets the requirements of 10 CFR Part 71 and is approved by USNRC; (Quality Assurance Program Approval No. 0703, Docket No. 71-0703, dated April 7, 2003).

9.1 QUALITY ASSURANCE PROGRAM AT MDS NORDION

The following is the history of status of QA program at MDS Nordion or its predecessors.

#1. The Quality Assurance Program in existence at MDS Nordion or its predecessors in May 1981 is as per document attached in Appendix 9.1.

#2. The Quality Assurance Program in existence currently at MDS Nordion is as per MDS Nordion document IN/QA 0224 Z000 "RADIOACTIVE MATERIAL TRANSPORT PACKAGE QUALITY PLAN", attached in Appendix 9.2.

9.2 MANUFACTURING HISTORY OF THE F-294

9.2.1 PROTOTYPE F-294

Manufacture:	Start Date: Jan. 1984 (approx.). Completion Date: Oct. 1985.
Manufacturer:	AECL-MEDICAL
Purchase Order or Equal Document:	MorS 432-83150
QA Program:	CSA Standards Z299.3-1979 and Z299.2-1979

Radiation and Thermal tests (steady state) were conducted around March 1986.

9.3 APPENDICES

Appendix 9.3.1 Quality Assurance Program at AECL Commercial Products* in 1981.
Appendix 9.3.2 Quality Assurance Program at MDS Nordion since 1992.

* Predecessors' History
- Operated as AECL Commercial Products from 1952 to 1985. In 1985 the Company was split into two divisions:
  1. AECL RadioChemical Company
  2. AECL Medical Co.
- In 1990 AECL-Radio Chemical Company was re-named Nordion International.
- In 1991 Nordion International was privatized and sold to MSD Inc. of Toronto.
- In 1997, Nordion International Inc. was re-named MDS Nordion Inc.

Appendix 9.3.1
Quality Assurance Program at AECL Commercial Products in 1981

VALIDATION

ATOMIC ENERGY OF CANADA LIMITED - COMMERCIAL PRODUCTS
QUALITY ASSURANCE PROGRAM
FOR DELIVERY OF A RADIOACTIVE MATERIAL PACKAGE
TO A CARRIER FOR TRANSPORT

Prepared by: ______________________ May 25, 1981
Manager, Product Integrity Date

Accepted by: ______________________ May 25, 1981
General Manager, Medical Products Date

______________________ May 25, 1981
General Manager, Industrial Products Date

______________________ 25 May, 1981.
Manager, Regulatory Affairs Date

Approved by: ______________________ 25 May, 1981
General Manager, Quality Assurance Date

Authorized by: ______________________ 25 May '81
Vice President ... / Mr. John Beddoes Executive Vice President
Executive Vice-President Date

Original Issue Date 1981 May 25

1981 May 25

ATOMIC ENERGY OF CANADA LIMITED - COMMERCIAL PRODUCTS
QUALITY ASSURANCE PROGRAM
FOR DELIVERY OF A RADIOACTIVE MATERIAL PACKAGE
TO A CARRIER FOR TRANSPORT

TABLE OF CONTENTS

1. INTRODUCTION

1.1 Purpose

The purpose of this plan is to define Atomic Energy of Canada Limited - Commercial Products' (AECL-CP) quality assurance program for radioactive material packages used to deliver radioactive material to a carrier for transport as required by 10 CFR71.12. The program is limited to the company-operational dependent activities in the shipment of packages containing radioactive materials; therefore, the package dependent elements of the quality assurance program as defined in 10 CFR71 - Appendix E are not addressed.

1.2 Scope

1.2.1 The scope of this program is limited to the operational elements for packages intended for use to ship Type B quantities of radioactive material.

1.2.2 As a general guideline, packages normally used by AECL-CP which are under this quality program include, but are not limited to, those described in the following USNRC and Agreement State Licences:

54-00300-04
54-00300-09
54-00300-12
54-00300-13
12-18482-01
37-19318-01
5-2623 (Texas)
G.A. 695-1 (Georgia)

Also specific package design models as follows:

F-121	F-154
F-127	F-158
F-127X	F-168
F-131	F-168X
F-143	F-245
F-144	F-247
F-146	F-251
F-147	F-254

1.2.3 This program applies to the integrity of the package as achieved by loading, closure, monitoring, and labelling prior to delivery to a carrier for transport.

2. ORGANIZATION

AECL-CP organization is shown in Figure 1. The structure of the Quality Assurance Division is illustrated in Figure 2. Specific responsibilities as they relate to the program's activities are summarized below.

2.1 Executive Vice-President

The Executive Vice-President is responsible for establishing the company's Quality Assurance Policy and ensuring that all company operations are carried out in full compliance with that policy; this responsibility includes the quality assurance program described herein.

2.2 General Manager, Quality Assurance

The General Manager, Quality Assurance, reports directly to the Executive Vice-President and is responsible for providing quality assurance program management for shipment of radioactive material packages. He is the final authority and represents AECL-CP on all quality matters. His specific duties with respect to packages include:

(a) The responsibility for the promulgation and execution of instructions, policies and procedures.

(b) The administration of the quality assurance program.

(c) The authority to stop activities when the seriousness of a condition may adversely affect quality or safety.

(d) The overall activities related to licensing of packages.

(e) The performance of quality assurance audits of operations and the reporting of non-conformances to established quality assurance practices and procedures.

(f) Conducting audit follow-up and monitoring corrective action.

2.3 Manager, Regulatory Affairs

The Manager, Regulatory Affairs reports to the General Manager, Quality Assurance, and is responsible for the required activities for the licensing and provision of quality assurance documents of packages. He is a licensed source handler and also serves as the Radiation Safety Officer for all field operations, and as such, assures that packages used have been approved by the Competent Authority of Canada and re-validated by the U.S. Department of Transportation for use in transporting radioactive by-product material.

2.4 General Manager, Industrial Products

The General Manager, Industrial Products reports to the Executive Vice-President. He is responsible for the activities and facilities related to the use of packages incident to transport of radioactive materials for use in products under his responsibility. He is responsible for ensuring that these activities are carried out in compliance with the quality assurance program and procedures.

2.5 General Manager, Medical Products

The General Manager, Medical Products reports to the Executive Vice-President. He is responsible for the activities and facilities related to the use of packages used for transport of radioactive materials to be installed in medical products. He administers, through the Manager, Medical Installation and Service, and the Area Service Managers, the operations of the Area Offices. He ensures that these activities are carried out in compliance with the quality assurance program and procedures.

2.6 Manager, Product Integrity

The Manager, Product Integrity reports to the General Manager, Quality Assurance. He is responsible for performing audits of those activities related to the program and reporting non-conformances to the responsible managers. He also follows up audits, and requests and monitors corrective actions.

3. QUALITY ASSURANCE PROGRAM FOR OPERATIONS

3.1 The quality assurance program for shipping packages is implemented through the use of written procedures, instructions and training. Also, it includes the documentation of work done, checks, inspections, personnel and procedure qualifications and audits. To comply with the quality assurance program requirements, procedures and instructions are available and controlled for the following areas: training of authorized personnel, loading, closure, monitoring and labelling of packages and licensing.

3.2 Training

Personnel performing activities affecting quality and safety are formally trained in proper handling techniques for the packages and also receive indoctrination in.

(a) Principles and practices of radiation protection.

(b) Radioactivity measurement standardization, monitoring techniques and instruments.

(c) Mathematics and calculations basic to the use and measurement of radioactivity, and

(d) Biological effects of radiation.

3.3 Personnel Qualifications

Personnel handling or preparing shipping packages for delivery to a carrier for transport are competent and duly licensed source handlers, and are also appropriately trained and qualified.

3.4 Management Review

Review of the scope, status, implementation and effectiveness of the quality assurance program is conducted by management on that portion of the program for which they have designated responsibility. The reviews are conducted and documented at least once every two years.

3.5 Revision

The General Manager, Quality Assurance is responsible for maintaining the currency of quality assurance program documents. New or revised quality assurance program requirements are implemented within 90 days following issue or as determined by the General Manager, Quality Assurance.

Temporary deviations or additions to this document may be made with the approval of the General Manager, Quality Assurance and authorized by the Executive Vice-President and accepted by the responsible managers. It is the responsibility of the Manager, Regulatory Affairs to ensure that the revisions to these documents are approved by the same signatories as the original document and that, where necessary, it is lodged with the appropriate international competent authorities.

3.6 Package Configuration

The Radiation Safety Officer assures that modifications to packages are approved by the competent authority in Canada and are submitted to the U.S. Department of Transportation for re-validation.

4. DOCUMENT CONTROL

All documents related to shipping packages are controlled through the use of written procedures. All changes to documents are performed according to written procedures approved by management. The Radiation Safety Officer ensures that all relevant quality assurance program procedures and revisions are provided to each Area Office and that line responsibilities are conducted in accordance with those procedures. Each of the Service Managers shall assure that the source handlers are aware of the latest procedures and are required to satisfy the RSO of same.

5. CONTROL OF MEASURING AND TEST EQUIPMENT

For package operations a portable radiation survey instrument is used to establish the radiation level of the shipping package and to determine if the package is contaminated with radioactive material. The procedures used to control the standard of the measuring instrument provide for:

5.1 Identification of the instrument.

5.2 Calibration of sources used for calibrating the instruments in accordance with standards established by the National Research Council of Canada.

5.3 Establishment of frequency of calibration of the instrument.

5.4 Maintenance of calibration records.

5.5 The removal from service and repair of damaged or inaccurate instrumentation.

6. HANDLING, STORAGE AND SHIPPING

For package operations, written procedures provide for:

6.1 Work instructions for handling, preservation, storage, cleaning, packaging, labelling, monitoring and shipping requirements to be completed by duly licensed source handlers.

6.2 Verification by the source handler that the activities in Item 6.1 have been completed and that the USNRC and U.S. Department of Transportation shipping requirements are properly satisfied prior to consignment to a carrier for transport.

6.3 All shipping documentation (certification, acceptances, etc.) to be prepared prior to shipment by AECL-CP at the Ottawa Head Office of AECL-CP.

6.4 The assurance that a duly licensed source handler performs all the critical handling, monitoring, storage and preparation for transport operations.

6.5 Emergency procedures by both AECL-CP (the consignor) and the commercial carrier.

7. INSPECTION, TEST AND OPERATING STATUS

7.1 Inspection, test and operating status of the package are indicated and controlled by written procedures in conjunction with the handling, storage, monitoring, labelling and shipping operations.

7.2 A check list is prepared and signed by the source handler and maintained both at the U.S. Office and at AECL-CP, Ottawa, for each shipment. The check list identifies the regulatory required inspections and tests as in the written procedures.

7.3 Each Area Service Manager ensures that these functions are performed. The Radiation Safety Officer, by verification of documentation, and periodic inspection, assures that these functions have been performed.

7.4 If the package is not suitable for shipment because of damage or non-conformance the status and disposition of the package is maintained by written procedure.

8. QUALITY ASSURANCE RECORDS

8.1 Sufficient records are prepared and maintained in accordance with the written procedures of the quality assurance program to furnish objective evidence of the integrity and safety of the shipping package. These records are identifiable to written procedures and traceable to the package and its movements.

8.2 Records attesting to the personnel training and qualifications are maintained at each Area Office and at AECL-CP.

8.3 Records are stored at AECL-CP Central Records and at the applicable Area Office for use as working records. Record retention times are based on established procedures consistent with commitments to the competent authority.

9. AUDITS

9.1 Planned Quality Assurance audits are performed by personnel who are appropriately trained and have no direct responsibilities in the areas audited. The audits are performed in accordance with written procedures and established quality audit techniques.

9.2 The audits determine the degree of conformance to approved procedures and quality assurance documents and provide objective evidence to evaluate the effectiveness of the program.

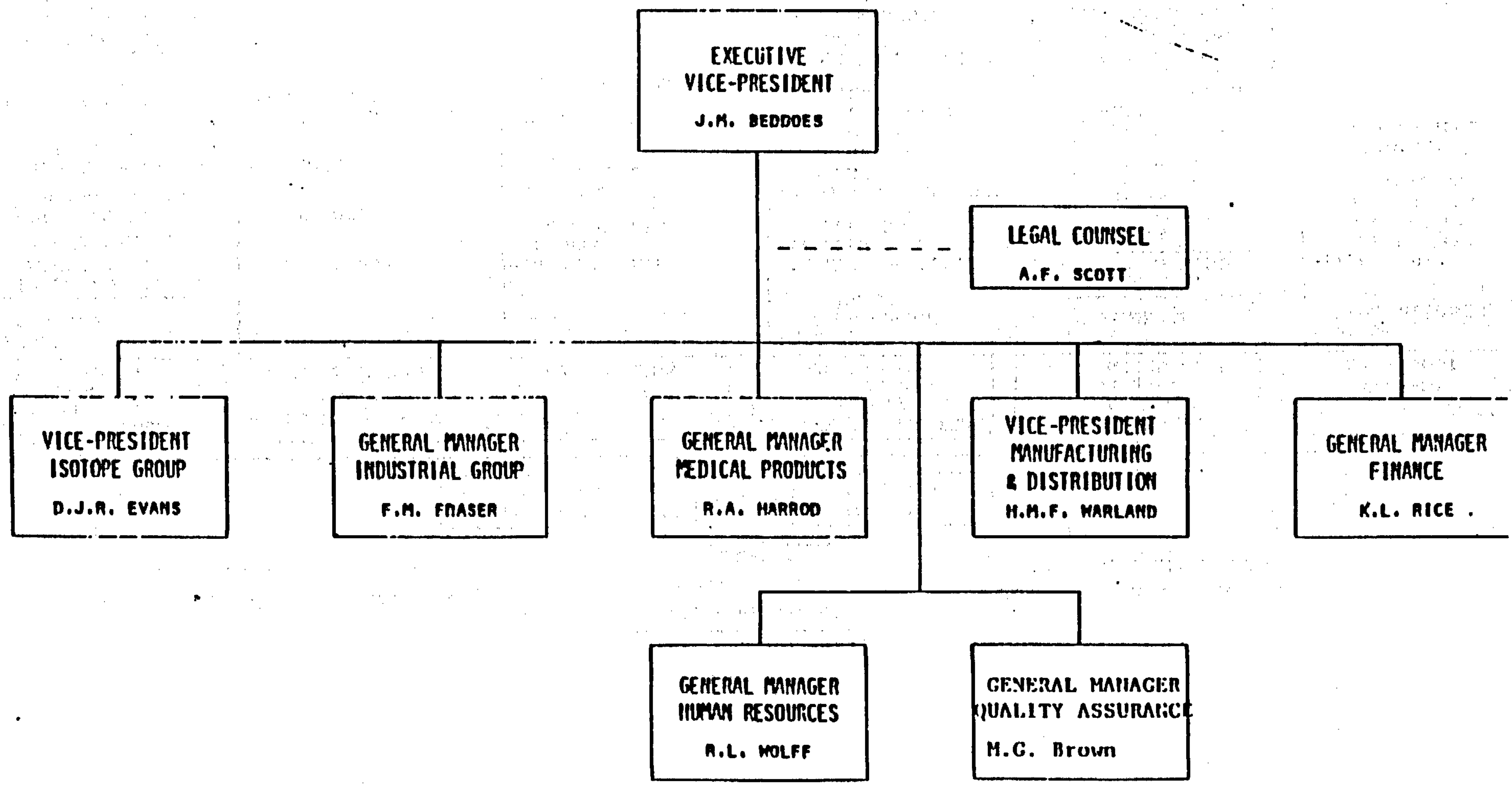
9.3 Corrective action for non-conformance is requested from the responsible manager and completion of the corrective action is verified. Uncorrected non-conformances are carried as Open Audit Items until corrected.

9.4 Audit frequency is based on the status, safety, importance or problems. At least one activity is audited each year.

9.5 Copies of audit reports are provided to responsible managers and senior management. Audit reports and related corrective action are maintained in Central Records at the Ottawa Head Office of AECL-CP.

FIGURE 1

ATOMIC ENERGY OF CANADA LIMITED - COMMERCIAL PRODUCTS
EXECUTIVE ORGANIZATION

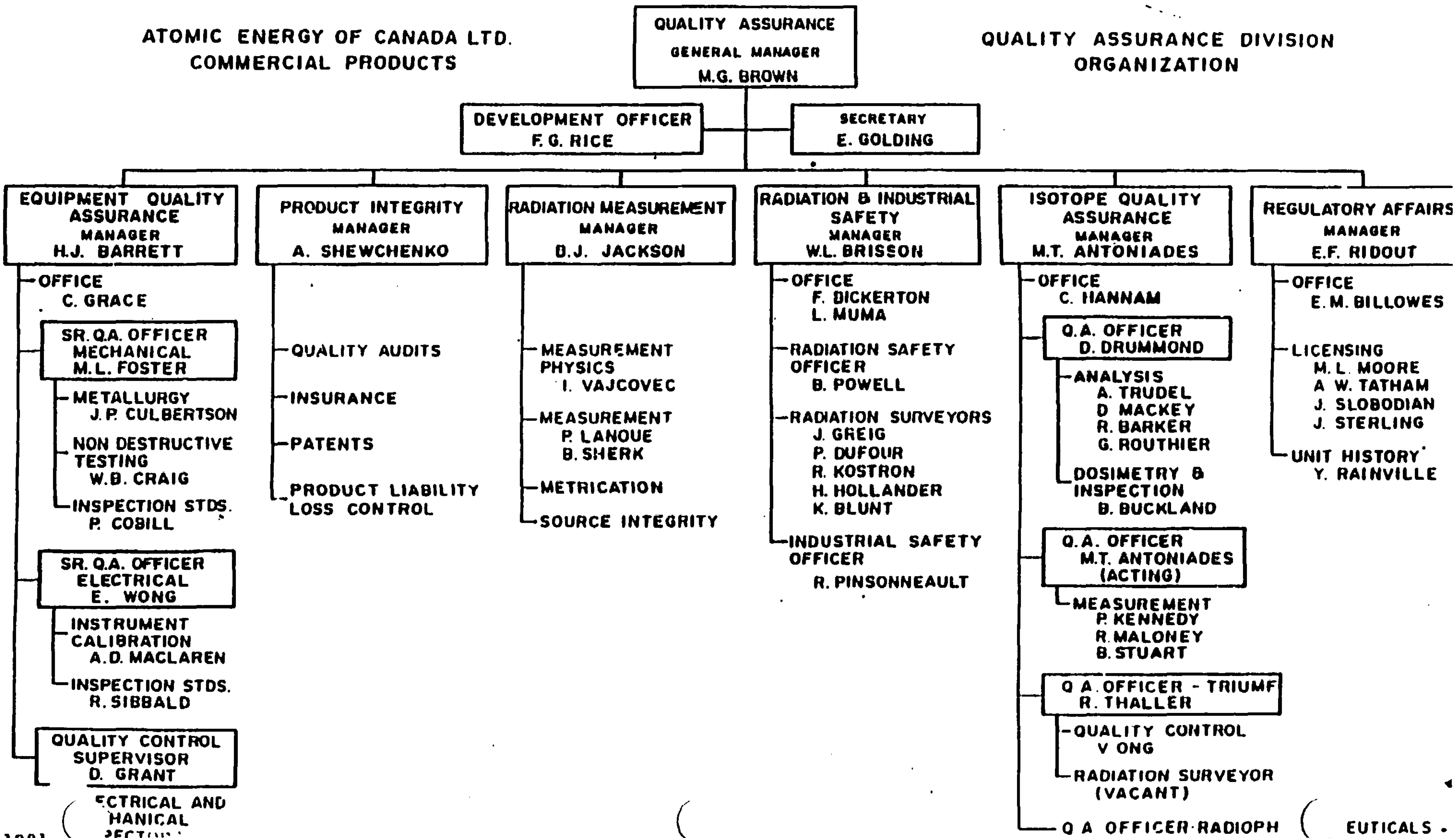


J.M. BEDDOES
1981 APRIL

FIGURE 2

ATOMIC ENERGY OF CANADA LTD.
COMMERCIAL PRODUCTS

QUALITY ASSURANCE DIVISION
ORGANIZATION



April 1981

APPENDIX 9.3.2
QUALITY ASSURANCE PROGRAM AT MDS NORDION SINCE 1992

Radioactive Material Transport Package Quality Plan

1. PURPOSE

This Quality Plan describes the activities associated with the design, fabrication, assembly, testing, maintenance, repair, modification, and use of MDS Nordion radioactive material (RAM) transport packaging. It identifies the activities, responsibilities, and action necessary to ensure that a transport package meets all regulatory, customer, and internal Quality Assurance Program requirements.

2. SCOPE

This plan is applicable to all MDS Nordion Transport Packages.

3. APPLICABLE DOCUMENTS

1. MDS Nordion Specification QSF 00, Ion Technologies Quality Manual
2. MDS Nordion SOP 5.00 –QA-00, Therapy Systems Quality Assurance Manual
3. MDS Nordion Specification QAP AP- 00, Nuclear Medicine Quality Manual
4. IAEA Safety Standard Series, Regulations for the Safe Transport of Radioactive Material, Regulation No. TS-R-1 (ST-1 Revised), 1996 Edition, (Revised)
5. IAEA Safety Guides, Safety Guide TS-G-1.1(ST-2) Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material (2002)
6. US-NRC Regulation 10 CFR 71 Packaging and Transportation of Radioactive Material
7. Packaging and Transport of Nuclear Substances Regulations, Nuclear Safety and Control Act
8. International Standard ISO 9001-2000 Quality Systems Management - Requirements
9. MDS Nordion Specification QAG-01, Device Design Control

4. PROCEDURE

4.1 Management Responsibility

4.1.1 Quality Policy

The MDS Nordion Quality Policy is maintained in the Business Unit Quality Manuals [1,2,3] and is approved by senior management. It is the responsibility of all applicable staff, at all levels, to be aware of and understand the Quality Policy and supporting procedures.

This Quality Plan is used to ensure that the specified requirements of transport packaging for radioactive material comply with pertinent regulatory requirements. This Plan defines the standard operating practices that affect radioactive material transport package quality. It establishes a documented system of management controls that provide confidence in the quality of all associated work activities.

4.1.2 Organization

Responsibility and Authority

Senior management responsibilities are described in references 1,2 and 3. Responsibility assignments for other functions involved in quality related activities for transport packages are described below:

Director, Quality Assurance

The authority for the administration of the Business Unit Quality Assurance Programs is delegated to the Business Unit Directors of Quality Assurance, who report to the Vice-President, Quality and Regulatory Affairs.

Director, Regulatory Affairs

The Director, Regulatory Affairs reports to the Vice-President Quality and Regulatory Affairs. The Director is responsible to the identification of regulatory requirements and coordinates the communications between MDS Nordion and the competent authorities.

Manager, Package Engineering

The Manager, Package Engineering reports to the Director, Radiopharmaceutical Operations, Nuclear Medicine. The Manager is responsible for the design, specification and testing of radioactive material packages and ongoing technical support. The Manager is responsible for verifying that all regulatory submissions for RAM transport package certification are accurate and complete.

Project Engineer

The Project Engineer reports to the Manager, Package Engineering, and is responsible for design projects from inception to completion. This includes the preparation or review of design documentation, design qualification testing, the preparation of specifications for manufacturing, and the review of Manufacturing, Inspection and Test Plans (MITP).

4.1.3 Management Review

Procedures for management review of the quality system are described in the Business Unit Quality Assurance Manuals. [1,2,3]

4.2 Quality Plan and Quality Systems

The Business Unit Quality Systems are described in the Business Unit Quality Assurance Manuals. [1,2,3] Requirements specific to transport packages are defined in this Quality Plan.

4.3 Contract Review

Authority for selection of transport packaging is through the operating divisions. Proposed uses of transport packaging which may not meet the specified requirements are referred to the Manager, Package Engineering for review. Each nonstandard application is reviewed for its capability to meet the standard, code, and regulatory requirements. For Type B packages, regulatory approval is obtained before use of any new packaging configurations.

4.4 Design Control

4.4.1 Design Planning and Activity Assignment

Written plans are prepared for design and verification activities for each new or modified RAM package design at project initiation. [9] The Manager, Package Engineering assigns the project to qualified personnel equipped with the resources necessary to prepare a fully compliant design.

4.4.2 Design Input

The Project Engineer verifies design input requirements for adequate identification and documentation. Design input verification involves, but is not limited to:

a) performance and functional criteria, including operational requirements

b) applicable codes and standards

The standard for the design of a radioactive transport package is reference [4]. However, other national codes shall be considered, as applicable [6,7][

c) regulatory requirements, including applicable Package Design Approval or Special Form Radioactive Material Certificates

d) environmental conditions

e) documentation, training, maintenance and inspection plans

f) the need for Special Form Material Certification for the sealed source if initial evaluation of radioactive material transport packaging/device or sealed source/packaging combinations indicate a need

g) where applicable, the results of contract review.

Incomplete, ambiguous, or conflicting requirements are resolved by the Project Engineer with the responsibility for drawing up the requirements.

4.4.3 Design Output

Design output is documented and expressed in terms of requirements, calculations, tests, and analysis. The design output shall:

a) meet the design input specification

b) show design analysis in sufficient detail to allow verification of the adequacy of the design and conformity to appropriate regulatory requirements whether or not these have been stated in the Design Plan,

c) include a Safety Analysis Report (SAR) in suitable detail to meet requirements of regulatory guidelines. The extent of the analysis and testing chosen must be sufficient to prove the validity of the design

All new RAM transport package designs must be evaluated to the applicable requirements. For Type B designs, the evaluation is part of a safety analysis report submitted to the competent authority. No Type B packaging design can be used before the evaluation is complete and a license has been issued. The Project Engineer is responsible for preparing an application for each new or modified Type B RAM transport package design.

d) include engineering drawings and operating procedures

e) include a Technical Specification for Type B packages

f) include a Design, Manufacturing and Operating Specification.

Safety Analysis Report

The application for certification of a Type B radioactive material transport package includes at least the following:

a) package description detailing radioactive contents, containment system, shielding, and operational features

b) structural evaluation including but not limited to:

1. structural design
   - design criteria referencing requirements for packages as defined in the design inputs
   - mechanical properties of structural materials
   - weights and centres of gravity
2. general requirements for packages such as:
   - lifting devices
   - closure methods
   - tiedown devices
   - external pressure
   - chemical and galvanic reactions
3. conditions of transport

c) thermal evaluation

d) accident analysis, based on IAEA or national competent authority regulatory requirements

e) overview drawing

f) preparation for shipment and inspection and maintenance procedures ,

g) Special Form evaluation, as applicable

h) test results, and/or engineering justification

i) the Design, Manufacture and Operating Specification.

Technical Specification

The Technical Specification is an integral part of the design documentation for a Type B package. It establishes the overall technical requirements for manufacture, assembly, inspection, test and delivery for each model of transport packaging. The Specification defines:

a) applicable engineering drawings

b) applicable standards

c) MDS Nordion specifications and procedures

d) quality program standards and codes required for manufacture. The manufacturer's quality program is normally subject to International Standard ISO 9001-2000 or CSA CAN3-Z299.2-85: The code requirement for welding and welder qualification is normally ASME BPV Code Section IX or CSA W59.

e) where applicable, requirements for welding, fitting and machining, surface finish and cleanliness, lead pouring and bonding, and painting

f) nonconformance and corrective action

g) inspection requirements

h) tests for welds, mechanical operation, lead bonding, radiation shielding, and leakage testing

i) requirements for supplier history file.

<u>Design, Manufacture and Operating Specification</u>

Each new project involving a Type B package shall include the issue of a Design, Manufacture and Operating Specification. The purpose of this document is to provide the design envelope to the applicable regulatory authority. The design envelope summarizes regulatory commitments and includes, at least:

a) the authorized contents, isotope, activity and form
b) the possible range of external dimensions
c) materials used and their thickness
d) a list of engineering drawings and procedures
e) reference to an information drawing which identifies the top manufacturing assembly drawing
f) marking and labeling
g) requirements for quality assurance in manufacture
h) requirements for use, including preparation for shipment and inspection and maintenance.

When design changes result in changes to the Design, Manufacture and Operating Specification, regulatory approval is required prior to transfer of the design to production.

4.4.4 Design Verification and Review

Designs and associated documents are verified and/or reviewed to ensure that they meet specified design requirements. Design verification is performed by qualified staff by conducting testing or by comparison to similar designs. The Manager, Package Engineering determines the extent of verification and review required. This decision is based on complexity, novelty, degree of standardization, and safety implications. The Design Plan identifies the verification and review requirements. All verification and review activities are documented. The nature of the verification process must conform to applicable codes and standards. The process involves:

a) qualification testing or comparison review according to applicable IAEA standards and competent regulatory authority regulations. Requirements, procedures, data, assumptions, and results are documented and filed. Results are evaluated against specified acceptance criteria. The conclusions of the tests or comparisons are recorded and filed in the transport design history files.

b) design review by qualified persons other than those who executed the design. The reviews determine if the design methods are appropriate and correctly applied. The reviewers verify that the assumptions and simplifications used are justifiable, and the design interfaces are properly addressed. Reviews are conducted before design release. They are documented, and include decisions.

4.4.5 Design Changes

A design change system is used for the control of drawings and supporting documentation. All changes to the design of RAM transport packages are reviewed and approved by the Manager, Package Engineering, Director, Quality Assurance; and where required, the representatives from the affected operating divisions. The Project Leader documents the change description, the reasons for it, and the implications. The method and extent of the design verification are dependent upon the extent and nature of the changes. The Project Leader identifies the necessary recipients of the revised design documents.

4.5 Document and Data Control

Documents and data that relate to the needs of the Quality Plan are controlled according to established procedures. [1,2, or 3, as applicable]

4.6 Purchasing

Documented policies and procedures are maintained to ensure that the purchased RAM transport packaging, components, materials, and services conform to specified requirements. [1,3]

Suppliers are selected based on their ability to meet the quality requirements. Measures are in place, through purchasing and QA policies and procedures, for the evaluation, selection and approval of suppliers. It is the responsibility of the Manager, Package Engineering to ensure that the purchasing documents clearly describe the material required. The key document for the information necessary is the purchase requisition, which references technical requirements such as the technical specification, drawings etc.

Each order for a transport packaging requires certain control activities and records, specifically:

a) purchase requisitions for transport packages are reviewed by Quality Assurance for adherence to the quality assurance procedures, and requirements

b) selected suppliers are on an approved vendors list

c) suppliers' Inspection and Test Plans are reviewed and approved by Package Engineering

d) Incoming inspection is performed according to the Inspection and Test Plan

e) requests for disposition of nonconformances must be submitted to the purchasing department in writing. Disposition is decided by the Manager, Package Engineering, or designate, and the Quality Engineer.

4.7 Control of Customer Supplied Product

Transport packaging supplied by customers is used in accordance with design and licensing documentation. As a minimum requirement, the customer is required to provide:

a) copies of relevant transport certificates, including certification of Special Form Radioactive Material Approval Certificates, if applicable

b) Operating Procedures.

4.8 Product Identification and Traceability

Each Type B package is identified with a model number and a unique serial number. During manufacture, special processes are qualified by the manufacturer.

4.9 Process Control

This quality element does not apply to transport packages.

4.10 Inspection and Testing

Inspection Plans are written for inspections performed during the life cycle of RAM transport packaging. These plans outline the type of inspection or testing to be undertaken. For each returnable RAM transport packaging type, an inspection and maintenance plan is prepared. The plans include, as applicable:

a) new packaging first-off inspection and acceptance requirements

b) periodic inspections after shipment, and before reuse

c) annual inspection and maintenance

d) inspection and maintenance checklists

e) instructions for special tests such as: leak testing, pressure tests, shielding tests, etc.

f) quality records to be kept.

Procedures for ongoing inspection and maintenance are normally prepared by the project engineer, and require review by the division responsible for the implementation of the inspection and maintenance procedure and the approval of the Manager, Package Engineering.

4.10.1 Package Qualification

Type B RAM packaging and components are inspected in accordance with the Technical Specification. Inspection and testing during manufacture are carried out by a qualified supplier using the Inspection and Test Plan approved by the Project Engineer. Supplier Manufacturing, Inspection and Test Plans are retained with the Unit History File.

Confirmation testing and/or review of supplier generated records to specific requirements is carried out at MDS Nordion prior to final acceptance of the RAM package. As a minimum requirement, a radiation survey must be performed prior to release for shipment. Documentation of the results is maintained in the unit manufacturing history file.

4.10.2 Inspection and Maintenance

In use packaging is periodically inspected to the appropriate Inspection and Maintenance Procedure by the applicable operating division.

4.11 Control of Inspection, Measuring and Test Equipment

Procedures for the control of Inspection, Measuring and Test Equipment are described in the Business Unit Quality Assurance Manuals. [1,2,3]

4.12 Inspection and Test Status

During manufacture, the inspection and test status of packagings is maintained in accordance with the qualified supplier's procedures. Systems are in place to ensure that the inspection and test status of all packagings is maintained.

4.13 Control of Nonconforming Product

Following the procedures in the relevant Quality Manual, disposition of nonconforming material is reviewed and the activity recorded. The system requires that the disposition of nonconformances be requested in writing. [1,3]

4.14 Corrective and Preventive Action

Documented procedures are in place for implementing corrective and preventive action. [1,2,3]

4.15 Handling, Storage, Packaging, Preservation and Delivery

The Project Engineer identifies, in the Technical Specification for Type B packages, the requirements for handling and storing by MDS Nordion's suppliers.

All RAM transport packages are prepared for shipment in accordance with Preparation for Shipment procedures. These procedures provide instructions to ensure the units are prepared for shipment in accordance with the requirements of the package Safety Analysis Report and include requirements for contamination testing, radiation surveys and labeling.

Documented procedures are in place to receive transport packages.

4.16 Control of Quality Records

Design Files

Project design files are normally filed with the design control documentation.

Manufacturing History File

Records are maintained to show that the specified quality requirements were met, and the quality system is operating correctly. Pertinent supplier quality records are an element of these data. The nature of the quality records is identified in the manufacturing plans. These records are maintained as a Unit History File for each Type B radioactive material transport packaging purchase. As a minimum, the following records are prepared by the supplier to form a Manufacturing History File:

a) Table of Contents

b) Copies of Purchase Orders and all amendments, if applicable

c) Supplier History File for components used, including, but not limited to:
- MDS Nordion QA Release Form
- Inspection and Test Plans
- completed inspection records
- list of drawings and specifications with current revisions in effect at the time of manufacturing and the serial numbers of the units supplied,
- copies of Deviation Disposition Requests
- Certified Material Test Reports, Certificates of Compliance or similar
- Certified Non-destructive Examination (NDE) reports,
- Welders' qualification certificates
- radiation survey data.

d) Reference to Manufacturing Plan used

e) Release for Shipment forms

f) Records are maintained in secured areas with limited access. The retention period for these records is the life of the packaging + 15 years.

<u>Service History</u>

Service history is maintained for each transport package design. This includes inspection and maintenance records.

Details of retention periods for specific records are detailed in the Quality Records procedure in the relevant Quality Manual. [1,2,3]

4.17 Quality Audits

Internal quality system audits are planned and performed in accordance with the Division Quality Assurance Manuals. [1,2,3]

4.18 Training

Training requirements are defined for all key roles affecting safety of RAM transport. Actual training is tracked against requirements including technical knowledge, control of process, specific skills and general theory in quality issues, safety, and company policies.

Detailed instruction for the carrying out of training is provided in each Division's administrative procedures. The significant requirements are summarized below:

a) all personnel involved in the transport of radioactive material receive training in radiation safety and transport regulations

b) specific qualification, training, and certification requirements are determined on an individual basis by line management. This determination is based on: the type of work, potential effect on quality, and the applicability of codes, standards or regulations

c) the line managers are responsible, with Human Resources, for maintaining records of staff selection, qualification, certification, and training. They also provide for the necessary training, and evaluate needs during staff performance reviews.

4.19 Servicing

Routine inspection and maintenance to transport packages discussed in section 4.10.

4.20 Statistical Techniques

This quality element does not apply to transport packages.

CHAPTER 10 – ABILITY OF THE F-294 TRANSPORT PACKAGE TO MEET THE REQUIREMENTS OF TS-R-1

The following revised and additional requirements of TS-R-1 were considered applicable to the F-294 with respect to package performance requirements.

- Inclusion of ambient temperature and pressure requirements (Paragraph 615)
- Change in reduced ambient pressure requirement (Paragraph 619 and 643)

10.1 GENERAL INFORMATION

There were no changes to the basic radionuclide (A_1 and A_2) values for Cobalt-60 (Co-60). No changes in the package designation or allowable contents are needed.

There are no impacts on the F-294 package with respect to package designation, drawings, and authorized contents as the result of the changes in the IAEA transport regulations.

10.2 STRUCTURAL EVALUATION

The requirements of Paragraphs 615, 619, and 643 of TS-R-1 state that:

615. The design of the package shall take into account ambient temperatures and pressures that are likely to be encountered in routine conditions of transport.

619. Packages containing radioactive material transported by air shall have a containment system able to withstand without leakage a reduction in ambient pressure to 5 kPa.

643. The containment system shall retain its radioactive contents under a reduction of ambient pressure to 60 kPa.

Paragraph 615 is a new requirement in TS-R-1. With respect to the structural integrity under the new requirement of Paragraph 615, the design of the F-294 takes into account ambient temperatures and pressures likely to be encountered during routine conditions of transport. In the Safety Analysis Report (SAR), it is shown that the F-294 package passes the tests demonstrating normal conditions of transport as specified in 10 CFR 71.71.

The requirement of Paragraph 619 is similar to the requirement in Paragraph 518 of SS6 which was applicable only to packages containing liquid radioactive materials. However, in TS-R-1, the requirement in Paragraph 619 applies irrespective of the radioactive material form. Therefore, the requirement in Paragraph 619 of TS-R-1 is applicable to the F-294 package which may be transported by air. Primary containment is provided by the sealed sources and the effect of a pressure drop to 5 kPa can be assessed as follows:

1. For typical applications, ISO 2919 (and companion standards ANSI N542, ANSI n43.6) requires sealed sources to remain leak free after being subjected to class 3 pressure test, which includes a pressure range from 25 kPa to 2 Mpa (0.7 to 290 psi).
2. The C-188 capsules have been tested and found to meet the requirements of a class 4 pressure test, which includes a pressure range from 25 kPa to 7 Mpa (0.7 to 1015 psi).
3. If the sources were loaded at 20°C and the final temperature is 1000°C, the normal operating pressure would be about 50 psi. If the pressure drops to 0 kPa (even less than the 5 kPa required) the pressure differential would be about 65 psi.

Successful completion of either the ISO 2919 class 3 or 4 pressure tests provides a practical demonstration of the ability of a sealed source to meet the requirements of Paragraph 619 of TS-R-1.

The requirement in Paragraph 643 changes the reduced external pressure requirement from 25 kPa in Paragraph 534 of SS6 to 60 kPa. This requirement is bounded by the requirement in Paragraph 619.

The F-294 complies with the pressure requirements in Paragraphs 615, 619, and 643 of TS-R-1.

10.3 THERMAL EVALUATION

The design of the F-294 takes into account ambient temperatures and pressures likely to be encountered during routine conditions of transport. In the Safety Analysis Report (SAR), it is shown that the F-294 package is in compliance with the thermal requirements under normal conditions as specified in 10 CFR 71.43(g) and 10 CFR 71.71. As such, the F-294 package meets the thermal requirement in Paragraph 615 of TS-R-1.

No other sections of the SAR are impacted by the new and revised requirements in TS-R-1.