APPENDIX 4.4

DESCRIPTION OF S300 PIPE OVERPACK

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4.4 Description of S300 Pipe Overpack

4.4.1 Introduction

The S300 pipe overpack is based closely on the standard pipe overpack described in Appendix 4.1 of the CH-TRU Payload Appendices. It differs from the standard pipe overpack through the addition of neutron shielding within the pipe component. It is intended for the shipment of sealed neutron sources in the TRUPACT-II and HalfPACT. Appendix 1.3.1 of the TRUPACT-II Safety Analysis Report (SAR), Appendix 1.3.1 of the HalfPACT SAR, and Section 2.9.5 of the Contact-Handled Transuranic Waste Authorized Methods for Payload Control (CH-TRAMPAC) describe the materials of construction, sizes, and other dimensional specifications for the S300 pipe overpack. Up to 14 S300 pipe overpacks may be shipped in the TRUPACT-II, and up to 7 S300 pipe overpacks may be shipped in the HalfPACT. This appendix describes the structural, criticality, and shielding basis of the S300 pipe overpack.

4.4.2 Description

The S300 pipe overpack consists of a neutron shield insert placed inside a standard 12-inch (in.) pipe component which is, in turn, located by cane fiberboard and plywood dunnage within a standard 55-gallon drum with a rigid polyethylene liner and lid. A schematic of the S300 pipe overpack is shown in Figure 4.4-1. All of the components of the S300 pipe overpack, except the neutron shield insert, are identical to the 12-in. version of the standard pipe overpack described in Appendix 4.1 of the CH-TRU Payload Appendices.

The neutron shield insert is a two-part assembly consisting of a cylindrical body and stepped lid. With the exception of necessary clearances, the insert fits within and fills the 12-in. pipe component. The insert lid is held in place by the lid of the pipe component. The insert is made from solid, high-density polyethylene (HDPE), and has a nominal wall thickness of 4.13 inches.

The pipe component provides three significant control functions: (1) criticality control, (2) shielding, and (3) confinement of the sealed neutron sources. The following sections demonstrate the effectiveness of the S300 pipe overpack design for normal conditions of transport (NCT) and hypothetical accident conditions (HAC). All demonstrations are by analysis or by reference to the standard pipe overpack analysis and testing, unless stated otherwise.

4.4.3 Structural Analysis for NCT

The structural effectiveness of the S300 pipe overpack for NCT is demonstrated by showing that the source material contents are confined within the pipe component. The structural effectiveness of the pipe component for NCT is bounded by the structural effectiveness evaluation for HAC given in Section 4.4.4. It is shown in Section 4.4.6 that an adequate level of biological shielding for NCT is afforded by the shield insert itself, with all other materials providing mainly a distance attenuation function. The maximum deflection and resulting radial shift of the pipe overpack array for the NCT side drop, which is limiting for shielding calculations, is bounded by the HAC side drop analysis provided in Section 4.4.4. Additionally,

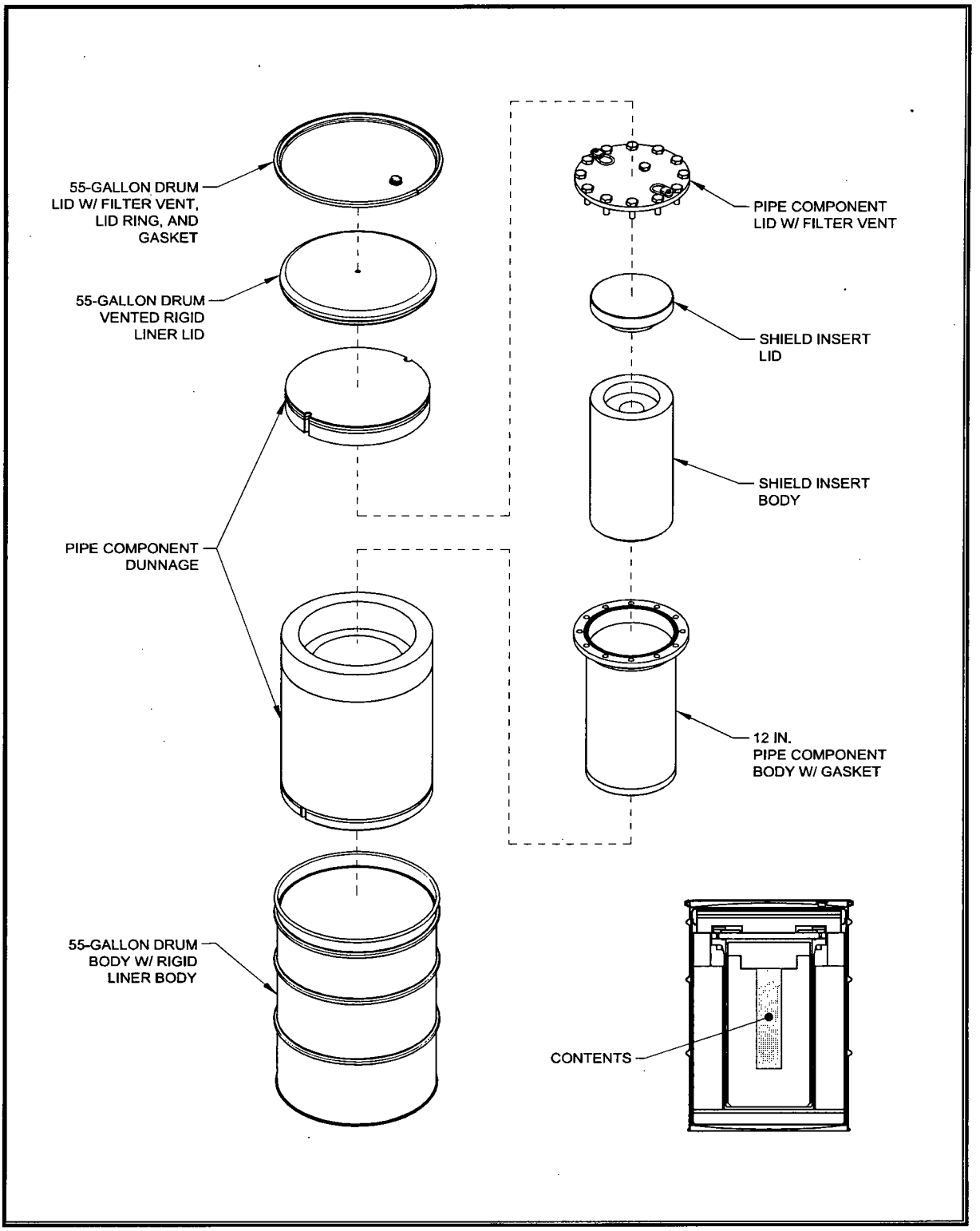

Figure 4.4-1 — S300 Pipe Overpack

the spacing between pipe components (i.e., effective drum diameter) utilized in the criticality analysis is also bounded by the HAC side drop analysis provided in Section 4.4.4.

4.4.4 Structural Analysis for HAC

The structural effectiveness of the S300 pipe overpack for HAC is demonstrated by showing that the source material contents remain confined within the pipe component. It is shown in Section 4.4.6 that an adequate level of biological shielding for HAC is afforded by distance attenuation considering the most conservative post-accident configuration of the TRUPACT-II or HalfPACT and pipe components. Since the shield insert in the S300 pipe overpack is not required for the HAC shielding analysis, the damage to the shielding material in the HAC free drop does not need to be quantified. However, the maximum deflection and resulting radial shift of the pipe overpack array will be quantified for the side drop orientation, which is limiting for shielding calculations. The following comparative analysis shows that the contents remain confined within the pipe component under the HAC free drop. Additionally, the analysis shows the maximum deflection of pipe overpacks and the resulting stacked array configuration resulting from the HAC free drop.

As shown in Table 2.9-7 of Section 2.9.2 of the CH-TRAMPAC, the weight of the 12-in. standard pipe component contents is bounded by a value of 225 lbs. Additionally, as shown in Table 2.9-20 of Section 2.9.5 of the CH-TRAMPAC, the total weight of the S300 pipe overpack shield insert and contents is also bounded by a weight of 225 lbs. Because the design of the standard and S300 pipe overpacks are structurally identical, and because the weight limit for items inside the pipe component are identical, all structural evaluations of the standard pipe overpack apply to the S300 pipe overpack. Ammerman and Bobbe, 1995,[1] demonstrates the leak tightness of the standard pipe overpack when subjected to HAC testing. Therefore, the source materials will remain confined within the pipe component under the HAC free drop.

Additionally, Ammerman and Bobbe[1] report a 20.250 in. minimum deformed pipe overpack diameter resulting from a free side drop orientation. Therefore, conservatively using a 20.000 in. 55-gallon drum diameter bounds the radial shift of the pipe component with respect to the S300 pipe overpack at (22.500 – 20.000)/2 = 1.250 in. The resulting stacked array of 14 S300 pipe overpacks resting against the TRUPACT-II inner containment vessel is accounted for in the HAC shielding analysis discussed in Section 4.4.6. The maximum drum crush values reported in Ammerman and Bobbe of 20.25 in. outside diameter by 29.62 in. height are directly utilized in the criticality analysis summarized in Section 4.4.5.

4.4.5 Criticality Analysis

A criticality analysis was performed for two different payload cases, depending on the quantities of special reflector materials in the payload container (see Chapter 6.0 of TRUPACT-II SAR or Chapter 6.0 of HalfPACT SAR for description of special reflector materials), as described below:

[1] Ammerman, D.J., and J.G. Bobbe, October 1995. "Rocky Flats Pipe Component Testing," TTC-1434, Sandia National Laboratories, Albuquerque, New Mexico.

- Case E: For Case E, the contents of the pipe overpack payload container contain less than or equal to 1% by weight quantities of special reflector materials. The pipe overpack payload container may contain greater than 1% by weight quantities of special reflector materials provided that one of the following conditions is met:
  - The special reflector materials are chemically or mechanically bound to the fissile material such that no reconfiguration or release of the bond is possible under normal or accident conditions, or
  - The special reflector materials are present in thicknesses and/or packing fractions that render them less effective than a 25% polyethylene/75% water equivalent reflector per the limits in Table 6.2-1 of the TRUPACT-II or HalfPACT SAR.
- Case F: For Case F, the contents of the pipe overpack payload container contain greater than 1% by weight quantities of special reflector materials that do not meet the exceptions listed for Case E.

The criticality analysis demonstrates that a TRUPACT-II shipment of 14 pipe overpacks with contents meeting the requirements of Case E at 200 FGE of ^{239}Pu each (for a total of 2,800 FGE per TRUPACT-II) or a HalfPACT shipment of 7 pipe overpacks with 200 FGE each (for a total of 1,400 FGE per HalfPACT) ensures compliance with the requirements of Title 10, Code of Federal Regulations (CFR), Sections 71.55 and 71.59 (10 CFR 71.55 and 71.59).[2] Additionally, shipments of pipe overpacks with contents meeting the requirements of Case F at 140 FGE for each payload container and 980 and 1960 FGE per HalfPACT and TRUPACT-II, respectively, ensure compliance with 10 CFR 71.55 and 71.59. Based on an infinite array of undamaged or damaged packages, the criticality transport index is 0.0.

The key parameters in the pipe overpack analysis for Case E are (1) the maximum fissile loading per pipe component is 200 FGE, (2) no more than 1% by weight quantities of special reflector materials are present or greater than 1% by weight quantities of special reflectors are either bound to the fissile material or meet the limits of Table 6.2-1 of the TRUPACT-II or HalfPACT SAR, (3) the spacing between the components (i.e., effective drum diameter) is reduced by the maximum amount reported in Section 4.4.4, and (4) the package arrays are infinite arrays stacked two high.

The key parameters in the pipe overpack analysis for Case F are (1) the maximum fissile loading per pipe component is 140 FGE, (2) the spacing between the components (i.e., effective drum diameter) is reduced by the maximum amount reported in Section 4.4.4, and (3) the package arrays are infinite arrays stacked two high.

[2] Packaging Technology, Inc., May 2004, "Pipe Overpack Criticality Analysis for the TRUPACT-II Package," ED-076, Packaging Technology, Inc., Tacoma, Washington.

The detailed analysis presented in Packaging Technology, 2004[2], presents the results of a series of SCALE 4.4 CSAS25 module[3] (KENO-Va version 4) calculations that establish a maximum system reactivity ($k_s + 2\sigma$) of less than 0.933 and the corresponding Upper Subcriticality Limit (USL) of 0.9377. Therefore, the shipment of 200 FGE or 140 FGE per pipe overpack for Cases E and F, respectively, in the TRUPACT-II and HalfPACT is safely subcritical.

4.4.6 Shielding Analysis

The payload of the S300 pipe overpack consists of neutron-emitting, actinide-bearing sealed sources, shown in Table 4.4-1. Source terms used in this analysis are for neutron emission and spectra for alpha-n reactions calculated by the SOURCES Version 4A computer code.[4] Of the sources shown in the table, the ^{238}Pu Be was determined to be the governing source for shielding calculations,[5] since it had the highest calculated unshielded dose rate of all the sources that will be transported in the S300.

Table 4.4-1 S300 Pipe Overpack Payloads

^{241}Am Be	^{238}Pu O	^{239}Pu Li	^{241}Am
^{238}Pu Be	^{239}Pu O	^{238}Pu B	^{238}Pu
^{239}Pu Be	^{244}Cm O	^{239}Pu F	^{239}Pu
^{241}Am O	^{241}Am Li	^{238}Pu ^{13}C	^{244}Cm

The radiation generated by the payload is in the form of neutrons and a relatively small amount of gamma radiation. Some additional gamma radiation is generated by capture of thermal neutrons in the neutron shielding. However, the gamma radiation remains a small fraction of the neutron radiation level.

Neutron shielding is provided by the shielding insert placed within the 12-in. pipe component. It has a minimum wall thickness of 4.06 in., and minimum end thicknesses of 3.58 in. at the bottom and 3.94 in. in the lid. None of the materials of construction of the S300 pipe overpack, including the neutron shielding material, generate hydrogen gas in excess of 10^{-10} moles hydrogen per second per liter of headspace as a consequence of neutron or gamma irradiation by the payload sources.[6] A combination of the neutron shielding material and the materials of

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[3] SCALE4.4., "Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation for Workstations and Personal Computers," RSICC code package C00545/MNYCP00, Oak Ridge National Laboratory, September 1998.

[4] Wilson, W.B., R.T. Perry, W. Charlton, et al., 1999, "SOURCES 4A: A Code for Calculating (alpha, n) Spontaneous Fission, and Delayed Neutron Sources and Spectra," LA-13639-MS, Los Alamos National Laboratory, Los Alamos, New Mexico.

[5] Gogol, S.L., and J.R. Bland, August 2002, "A Comparison of Dose Rates from (alpha, n) and Spontaneous Fission Neutron Sources," LA-UR-02-5120,.Los Alamos National Laboratory, Los Alamos, New Mexico.

[6] Bustos, L.D., W.F. Sandoval, R. Villarreal, and L.R. Field, October 2000, "Hydrogen Generation Rate Potential from Neutron and Gamma Ray Interactions with Shielding/Packaging Materials Contained in the S100 Pipe Component Overpack," Los Alamos National Laboratory, Los Alamos, New Mexico.

construction of the S300 pipe overpack provide sufficient shielding for both neutron and gamma radiation.

Dose rate calculations were performed for a single S300 pipe overpack and for a TRUPACT-II in both the as-loaded and post-NCT free drop configurations.[7] The results were used to determine the maximum loading of the S300 pipe overpack such that the regulatory dose rate limits will be met in each case for NCT and HAC. In the analysis, the bounding payload of ^{238}Pu Be was used, as discussed above. Source gamma radiation was negligible and was not included, but capture gamma dose rate contribution was included in the calculated integrated dose rate. Dose rate calculations were made for a single S300 pipe overpack as presented for loading into a TRUPACT-II, for a TRUPACT-II as presented for transport with a payload of 14 identical S300 pipe overpacks each having the maximum payload, and for a TRUPACT-II including a conservative representation of NCT free drop damage with a payload of 14 identical S300 pipe overpacks each having the maximum payload. (The HAC case is discussed below.) Dose rates were calculated at the surface and at defined distances from the containers as shown in Table 4.4-2. As shown in the table, the limiting dose is for the TRUPACT-II package at a distance of 5 meters from the package surface (the truck cab, a normally occupied space), and is equal to 2 millirem per hour (mrem/hr). The corresponding S300 pipe overpack surface dose limit is 155 mrem/hr. This means that, as long as the surface dose rate of any S300 pipe overpack transported in a TRUPACT-II is at or below 155 mrem/hr, then the dose rate external to the TRUPACT-II will not exceed 2 mrem/hr at a distance of 5 meters, nor will any of the other, less governing regulatory limits be exceeded. The TRUPACT-II calculations govern the case of the HalfPACT. Each S300 pipe overpack will be surveyed before loading into a TRUPACT-II or HalfPACT to ensure compliance with the limiting surface dose rate of 155 mrem/hr, as given in Section 3.2 of the CH-TRAMPAC.

[7] Packaging Technology, Inc., August 2002, "Dose Rate Calculations for the S300 Pipe Overpack," ED-072, Packaging Technology, Inc., Tacoma, Washington.

Table 4.4-2 Maximum Dose Rates for S300 Pipe Overpack and TRUPACT-II

	Maximum Dose Rate (mrem/hr)	Limits (mrem/hr)④
S300 Surface	155 ±0.36	200
TRUPACT-II side Surface (undamaged) ①②	64.5 ±0.62	200
TRUPACT-II 2 meters (undamaged)	8.06 ±0.10	10
TRUPACT-II 5 meters (undamaged)③	1.97 ±0.03	2
TRUPACT-II side Surface (damaged)	120 ±0.20	200
TRUPACT-II 2 meters (damaged)	9.83 ±0.12	10

Notes:
1. TRUPACT-II contains 14 identical S300 pipe overpacks, each with a maximum surface dose rate of 155 mrem/hr or less.
2. Side dose rate governs over top or bottom dose rates.
3. The 5 meter distance corresponds to the normally occupied space of the truck cab.
4. Limits established by CH-TRAMPAC (S300 surface) or 10 CFR 71.47(b) (TRUPACT-II).

The damage to the TRUPACT-II and payload under NCT is assumed to occur in the 3 ft. side drop, and is discussed in Section 4.4.4. The drums are modeled as resting on the inside of the TRUPACT-II ICV, which is resting on its side. Each drum is conservatively reduced in size to a diameter of 20.0 in., and the array is accordingly compressed and shifted to be in contact with the inside surface of the TRUPACT-II ICV.

For HAC, the drums, neutron shielding material, pipe components, and internal dunnage are conservatively removed from consideration in the shielding calculation, and the sum total of all activity in the S300 payload is concentrated as a single point source resting on the inside surface of the TRUPACT-II ICV. In accordance with 10 CFR 71.51(a)(2), the dose point is located 1 meter from the external surface of the package. This is equivalent to a total distance from the source of 1 meter plus the minimum crushed wall thickness of the TRUPACT-II or HalfPACT. For simplicity and conservatism, the calculations assume that there is no material of any kind between the source and the dose point. The crushed wall thickness is found by subtracting the HAC 30-foot free drop side orientation crush damage from the original wall thickness of the package as follows. The outer diameter of the package is 94.38 inches, and the inner diameter of the ICV is 73.63 inches, which gives an undamaged wall thickness of 10.38 inches. The maximum crush damage is found in Table 2.10.3-1 of the TRUPACT-II SAR for Test No. 2, as equal to 3.63 inches. The remaining wall thickness is then equal to 10.38 – 3.63 = 6.75 inches. In the shielding calculations, a value of 6.5 inches is conservatively used. As already discussed, no material is assumed to fill this space. The resulting maximum allowable activity within the

TRUPACT-II is a total of 406 Ci, and the resulting conservative dose rate is 999 mrem/hr at 1 meter from the crushed TRUPACT-II surface, which meets the requirements of 10 CFR 71.51(a)(2). As for NCT, the TRUPACT-II HAC calculations govern the case of the HalfPACT.

4.4.7 Authorized Payload Contents

As demonstrated in Section 4.4.6, when loaded with sealed neutron sources of the types specified in Table 4.4-1 (the authorized contents), the S300 pipe overpack meets all regulatory dose rate limits. The bounding payload is defined in three ways: (1) a maximum dose rate on the surface of the S300 pipe overpack of 155 mrem/hr for any S300 pipe overpack placed into the TRUPACT-II or HalfPACT, (2) a maximum activity of 406 Ci within a single TRUPACT-II or HalfPACT, and (3) a maximum payload of 200 FGE per S300 pipe overpack, or a total of 2,800 FGE per TRUPACT-II or 1,400 FGE per HalfPACT when the contents meet the requirements for Case E, or (4) a maximum payload of 140 FGE per S300 pipe overpack, or a total of 1,960 FGE per TRUPACT-II or 980 FGE per HalfPACT when the contents meet the reuqirements for Case F. Section 4.2.5 demonstrates that 200 FGE per S300 pipe overpack is safely subcritical for Case E contents and that 140 FGE per S300 pipe overpack is safely subcritical for Case E contents.

4.4.8 Conclusion

The S300 pipe overpack design is very closely based on the standard pipe overpack. It consists of a standard 12-in. pipe component within a 55-gallon drum, including a rigid liner and lid. A neutron shield insert is placed inside the pipe component. The analyses summarized in this appendix demonstrate the ability of the S300 pipe overpack to provide three significant control functions under NCT and HAC: (1) criticality, (2) shielding, and (3) confinement of the payload. The payload of the S300 is sealed neutron sources of the types listed in Table 4.4-1. The structural analysis shows that the source material remains confined within the pipe component in conservatively bounded NCT and HAC free drops. For criticality, it is shown that 200 FGE per S300 pipe overpack for Case E payloads is safely subcritical and 140 FGE per S300 pipe overpack is safely subcritical for Case F payloads. The shielding analysis shows that, with the maximum authorized contents, the dose rate limits for NCT and HAC (including appropriate shielding damage assumptions in each case) are met.

APPENDIX 4.5

DESCRIPTION OF SHIELDED CONTAINER

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4.5 Description of Shielded Container

4.5.1 Introduction

The shielded container is a vented carbon steel and lead cylindrical structure with a removable lid. It is designed to be used for the shipment of specific transuranic waste forms in the HalfPACT package. Drawing 163-008 in Appendix 1.3.1 of the HalfPACT SAR[1] and Section 2.9.10 of the *Contact-Handled Transuranic Waste Authorized Methods for Payload Control* (CH-TRAMPAC)[2] delineate the materials of construction, sizes, and other dimensional specifications for the shielded container and associated dunnage components.

The shielded container is intended for the shipment of transuranic waste forms with high gamma energies in the HalfPACT. The HalfPACT package can accommodate three (3) shielded containers. As configured for shipment, the shielded container payload assembly remains within the previously established design and certification bases and limits of the HalfPACT package for weight (7,600 pounds) and decay heat (30 watts). Limits on shielded container activity and fissile content are also set consistent with previously implemented and accepted analytic approaches.

This appendix describes the structural, thermal, shielding, and criticality basis of the shielded container payload.

4.5.2 Description

The shielded container, approximately the same size as a standard 55-gallon drum, consists of a twin-shelled, carbon steel cylindrical structure and a lid. Nominally, 1 inch of lead shielding is contained between the 7-gauge inner shell and 11-gauge outer shell. The shells are connected to an upper flange and a 3-inch thick solid steel bottom. The 3-inch thick solid steel lid integrates a silicone rubber gasket, fifteen 1/2-inch, alloy steel closure bolts, two alignment pins to facilitate remote assembly, and a lead-shielded filter port. Three removable lifting eyes are used for handling the shielded container prior to installation within a HalfPACT packaging.

The shielded container is designed to carry one 30-gallon steel payload drum. A partially exploded view of the shielded container, including its 30-gallon payload drum, is provided in Figure 4.5-1. In addition to the 30-gallon payload drum, the shielded container may optionally contain a mesh "bag" to facilitate remote installation of the 30-gallon payload drum into the shielded container. The shielded container must be installed with a filter vent; Section 2.5 of the CH-TRAMPAC provides the minimum specification for the shielded container filter vent.

As illustrated in Figure 4.5-2 and Figure 4.5-3, the shielded container payload system also includes a triangular spaceframe pallet, optional plastic stretch wrap and/or banding around the shielded containers, an optional plastic slipsheet below the three shielded containers, an optional plastic reinforcing plate above the three shielded containers, a radial dunnage assembly surrounding the three shielded containers, and an axial dunnage assembly below and above these components.

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[1] U.S. Department of Energy (DOE), *HalfPACT Shipping Package Safety Analysis Report*, USNRC Certificate of Compliance 71-9279, U.S. Department of Energy, Carlsbad Field Office, Carlsbad, New Mexico.

[2] U.S. Department of Energy (DOE), *Contact-Handled Transuranic Waste Authorized Methods for Payload Control* (CH-TRAMPAC), U.S. Department of Energy, Carlsbad Field Office, Carlsbad, New Mexico.

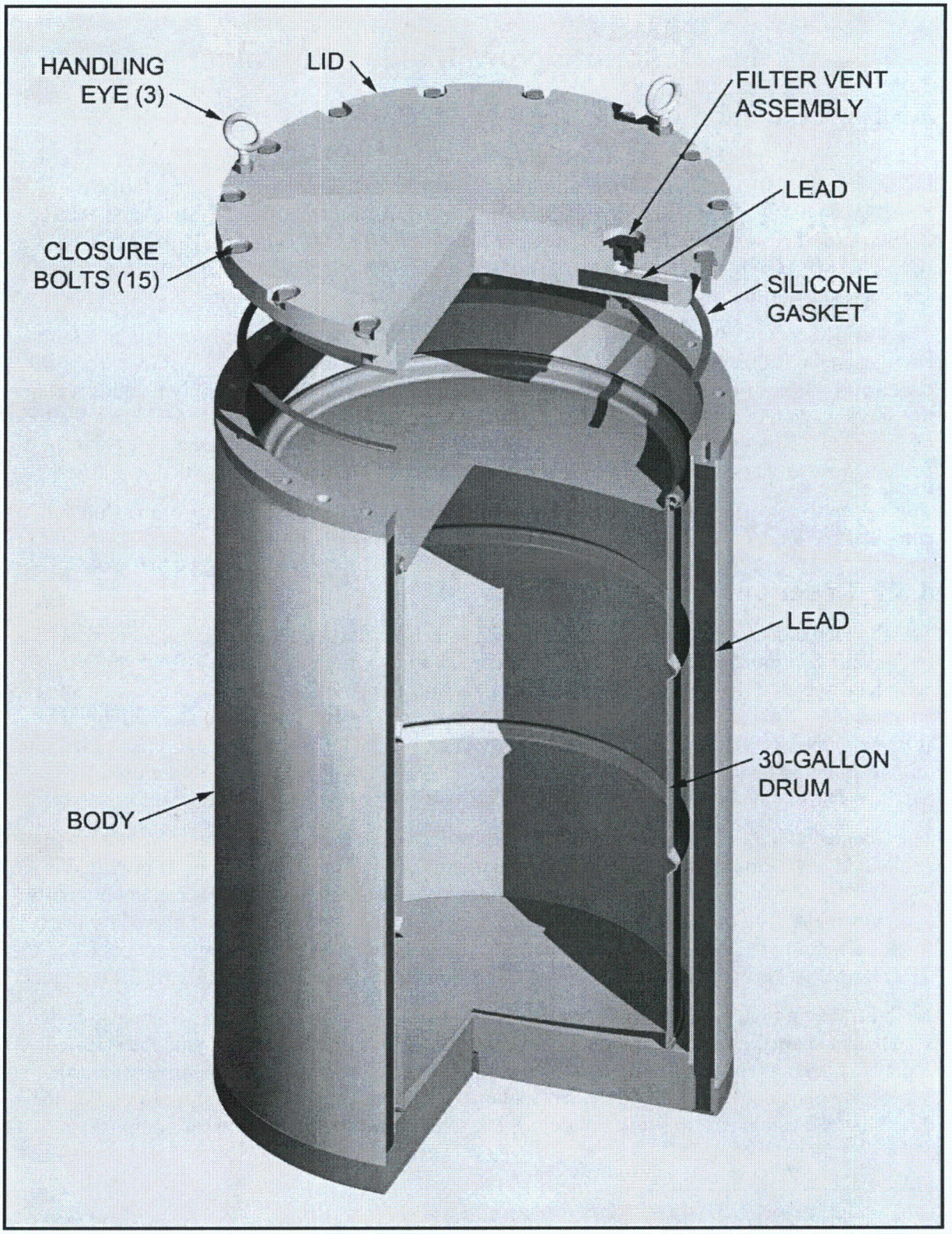

Figure 4.5-1 – Shielded Container Configuration

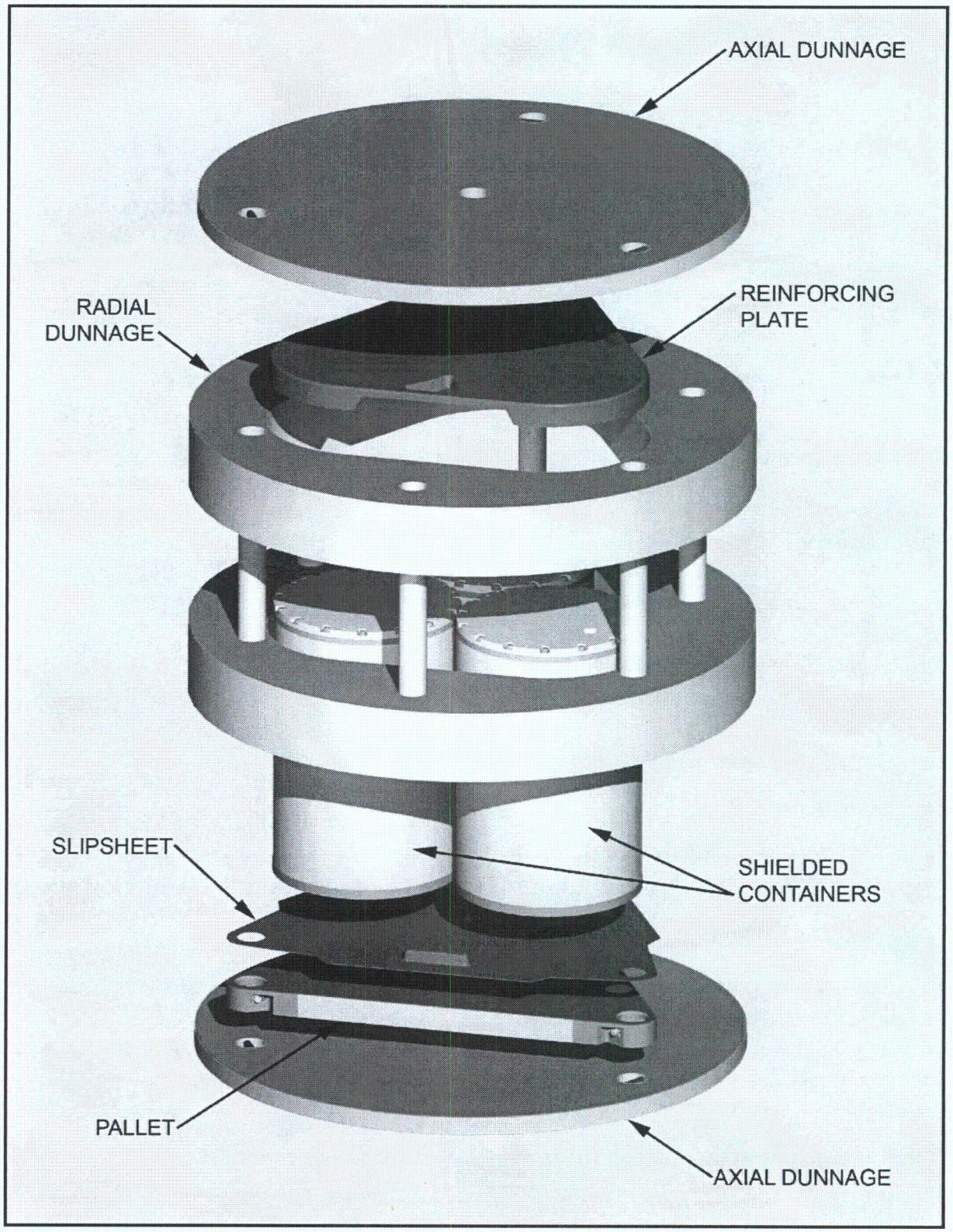

Figure 4.5-2 – Shielded Container Payload Components (Exploded View)

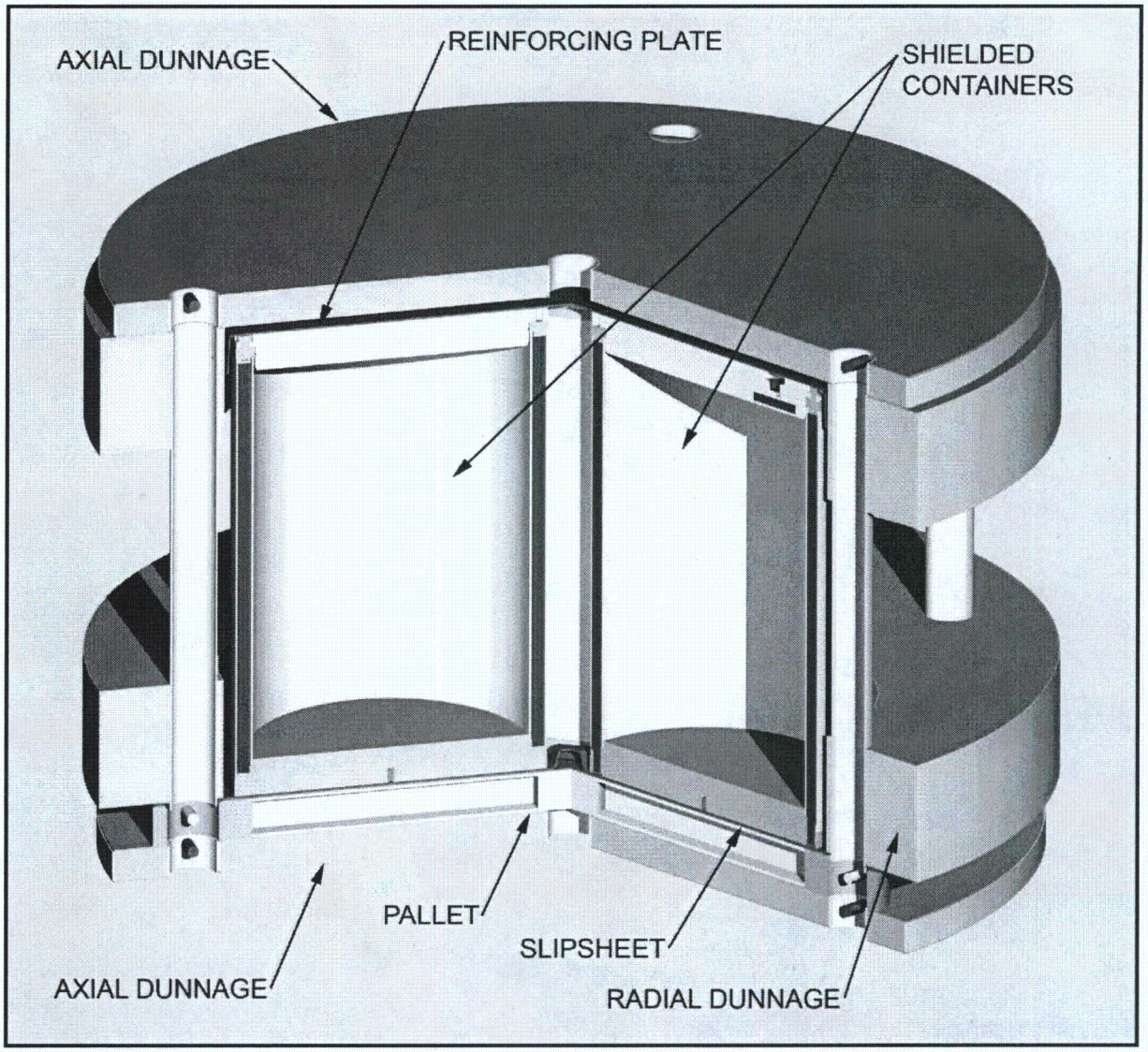

Figure 4.5-3 – Shielded Container Payload Components (Assembled View)

4.5.3 Structural Evaluation

Since the shielded container is vented, it is not subject to pressure loads present within the HalfPACT package's containment boundary.

Based on the following paragraphs, the shielded container payload configuration is, from a HalfPACT packaging perspective, bounded by previous certification testing for the HalfPACT package as currently presented in the HalfPACT SAR.

4.5.3.1 Structural Evaluation for Normal Conditions of Transport

Under normal conditions of transport (NCT), the shielded container maintains both confinement and shielding integrity. Since confinement and shielding integrity has been demonstrated for hypothetical accident conditions (HAC) without loss of fine particulate confinement or degradation

of the shielding material, as discussed in Section 4.5.3.2, *Structural Evaluation for Hypothetical Accident Conditions*, and HAC bounds NCT, demonstrations specific to NCT are not necessary.

4.5.3.2 Structural Evaluation for Hypothetical Accident Conditions

Under HAC, the shielded container confines its contents within its shielded boundary. To demonstrate confinement and shielding integrity of the shielded container, a full-scale test program was conducted.[3] Since confinement integrity was maintained, and the shielding material did not reconfigure during HAC testing, NCT is bounded by the HAC test program.

Three shielded containers were assembled on a triangular spaceframe pallet and installed, including axial and radial dunnage assemblies, within a HalfPACT inner containment vessel (ICV). The package was subjected to two 30-foot free drops onto a flat, essentially unyielding, horizontal surface: a vertical end drop and a horizontal side drop. The HalfPACT outer confinement assembly (OCA), with its energy absorbing polyurethane foam, was conservatively omitted from the tests. At the conclusion of the second 30-foot free drop, each shielded container was removed from the ICV, sprayed with water, and visually examined for the presence of fluorescein dye to verify confinement integrity. Each shielded container was subsequently opened and subjected to shielding integrity testing to verify shielding integrity.

To conservatively test to the maximum allowable payload weight of 7,600 pounds within a HalfPACT packaging, each test shielded container utilized a 30-gallon steel drum (approximately 35 pounds empty) filled with approximately 455 pounds of concrete and 70 pounds of sand, for a total loaded weight of 560 pounds.

To address shielded container performance and any potential for adverse effects on the HalfPACT packaging containment and confinement boundaries when transporting shielded containers, it is only necessary to perform 30-foot free drop tests for the flat bottom and side orientations. This is because both the radial dunnage assembly and the axial dunnage assemblies (acting in conjunction with the adjacent aluminum honeycomb end spacers) have been independently designed to absorb 100% of the payload energy associated with a 30-foot drop.

In an end drop orientation, virtually all payload related energy is absorbed by a combination of crushing the aluminum honeycomb end spacer (primary energy absorber) and an axial dunnage assembly (secondary energy absorber). As demonstrated by the bottom end drop testing, the payload pallet structure was minimally deformed and can therefore be assigned no significant energy absorbing role. The radial dunnage assembly also plays no significant energy absorbing role in an end drop and was undamaged.

Conversely, the payload pallet, axial dunnage, and honeycomb end spacer assemblies remain undamaged in a side drop, while the radial dunnage assembly absorbs all the energy associated with the three loaded shielded containers.

Any drop orientation other than end or side would partially crush both the axial and radial energy absorbing dunnage components, but each to a lesser degree than what occurs for the more limiting end and side drop tests. As such, for other drop orientations, loads on both the HalfPACT ICV as well as on the shielded containers themselves are more distributed (i.e., partially shared by both end and side structures) and of lesser magnitude than those experienced in flat end or side drops. Also, from a post-

[3] *Regulatory Hypothetical Accident Condition Type B Testing for the HalfPACT Shielded Container Payload*, WP 08-PT.15, Rev. 0, Washington TRU Solutions, December 2007.

HAC shielding point of view, the greatest shift of the shielded containers within the ICV will occur for the end and side drop tests. Finally, the relatively large post-drop residual radial and axial clearances that existed between the shielded containers and the ICV clearly demonstrated that there is no potential for the shielded containers to directly impact, or in any way compromise, the HalfPACT ICV.

Further technical justification for the selected drop orientations, testing at ambient temperature, and testing without internal pressure is provided in Section 5.0 of the test report.[3] The position of the three test shielded containers, B01, B02, and B03 is given in Figure 4.5-4.

4.5.3.2.1 End Drop

The end drop was performed using an unprotected HalfPACT ICV that was stiffened at its lower end to conservatively simulate a cold impact deceleration acting on the HalfPACT package of 409*g* if the OCA were present. Given the circumferentially uniform and permanent deformation that occurred just above the stiffeners at the lower end of the ICV shell in the shielded container end drop test (absent in all prior TRUPACT-II and HalfPACT testing that included an OCA), it is clear that stiffening of the ICV for the shielded container testing conservatively bounded the overall system deceleration that would exist if an OCA was present.

The total deformation to the lower end payload components was approximately 6.2 inches for the ambient temperature testing; the deformation would modestly increase to 6.51 inches for NCT hot conditions. Figure 4.5-5, Figure 4.5-6, Figure 4.5-7, and Figure 4.5-8 illustrate end drop damage to these components. Correspondingly, the maximum estimated acceleration to the shielded containers, including an adjustment for the worst-case cold temperature, was 62.1*g*. The end drop acceleration was sufficiently low to preclude any amount of lead movement from the end drop test, as demonstrated by subsequent shielding integrity testing and physical sectioning of one of the test shielded containers (see Section 4.5.3.2.3, *Post-Drop Shielding Integrity Testing and Destructive Disassembly*). Potential lead movement at NCT hot conditions is also addressed in Section 5.0 of the test report[3] and shown to be insignificant. No damage was visible on the exterior of the shielded containers as a result of the end drop test.

4.5.3.2.2 Side Drop

The side drop was also conservatively performed using an unprotected HalfPACT ICV (i.e., without the energy absorbing HalfPACT OCA). The aluminum honeycomb end spacer, axial dunnage, and pallet assemblies at the ends of the shielded containers remain undamaged in a side drop, but serve to maintain the relative position of the containers within the radial dunnage assembly and ICV. For this reason, the lower honeycomb end spacer and axial dunnage assembly that were damaged in the end drop test were removed and replaced prior to the side drop test with a steel space frame of the correct overall height to re-center the shielded containers within the ICV and radial dunnage assembly.

In the case of a side drop, the radial dunnage assembly must absorb all of the drop-induced kinetic energy of the shielded containers. To maximize damage to the radial dunnage assembly and maximize the load acting on a single shielded container for the side drop test, the shielded containers were oriented to place a single shielded container and the least amount of radial dunnage thickness (9⅜ inches thick) directly in line with the impact point. Figure 4.5-9, Figure 4.5-10, Figure 4.5-11, and Figure 4.5-12 depict the pre-side drop configuration and post-side drop damage.

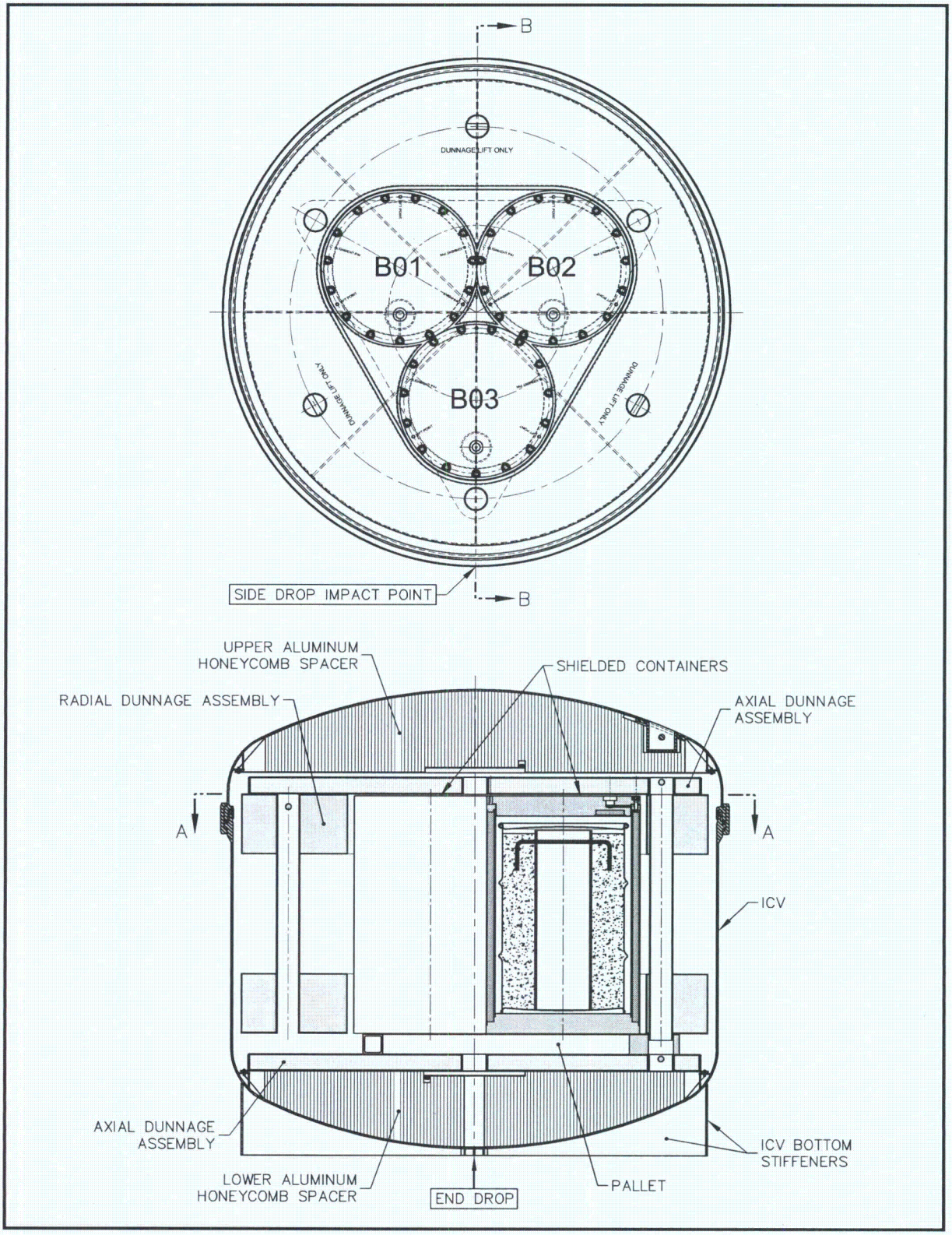

Figure 4.5-4 – Test Configuration and Orientations

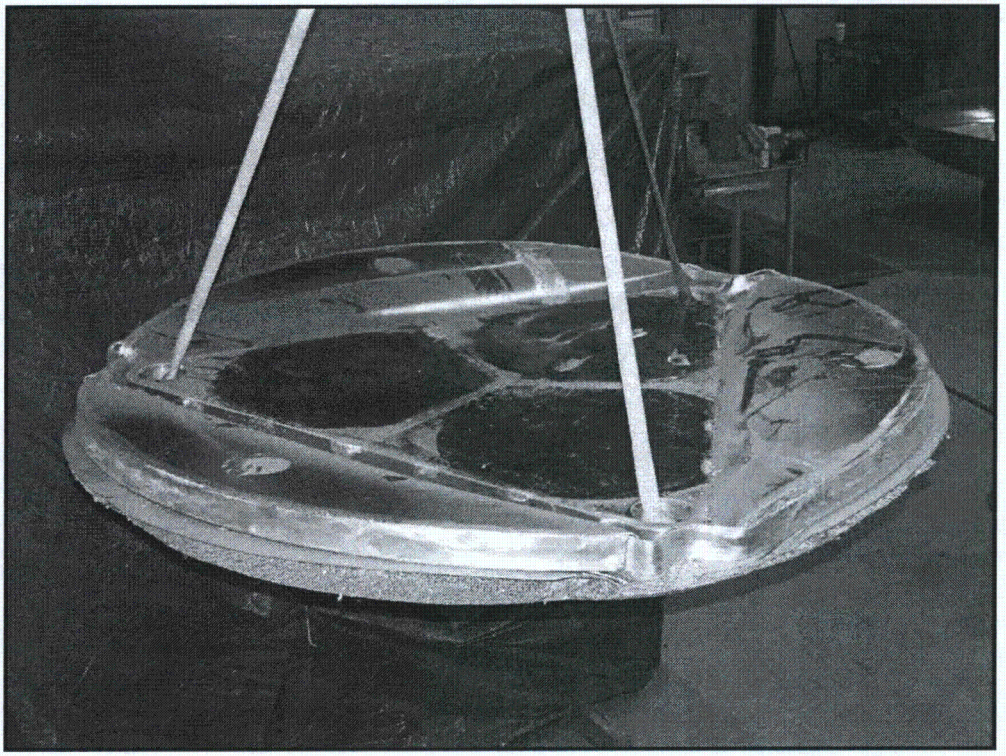

Figure 4.5-5 – End Drop Damage to the Lower Payload Components

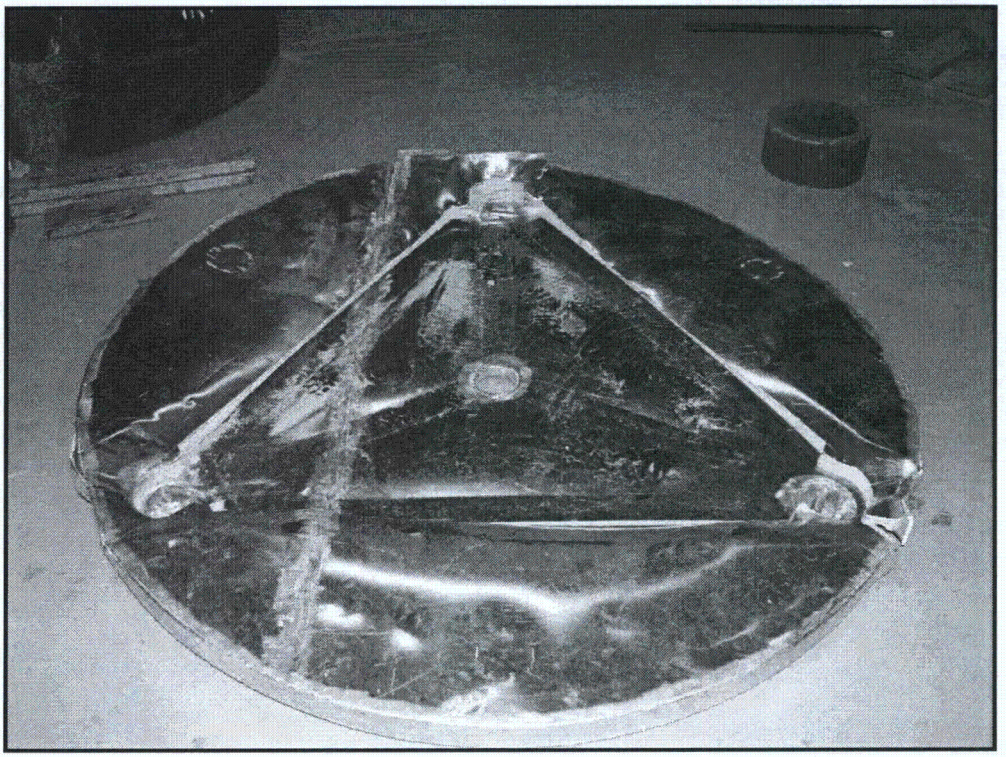

Figure 4.5-6 – End Drop Damage to the Lower Axial Dunnage

Figure 4.5-7 – End Drop Damage to the Lower Spacer (Top View)

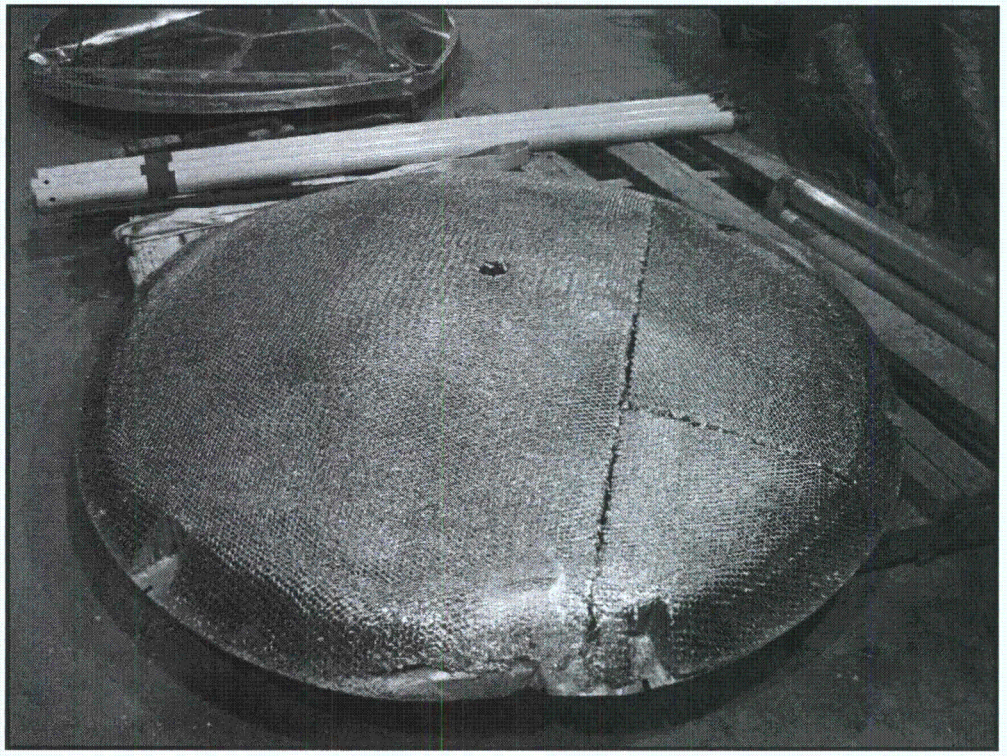

Figure 4.5-8 – End Drop Damage to the Lower Spacer (Bottom View)

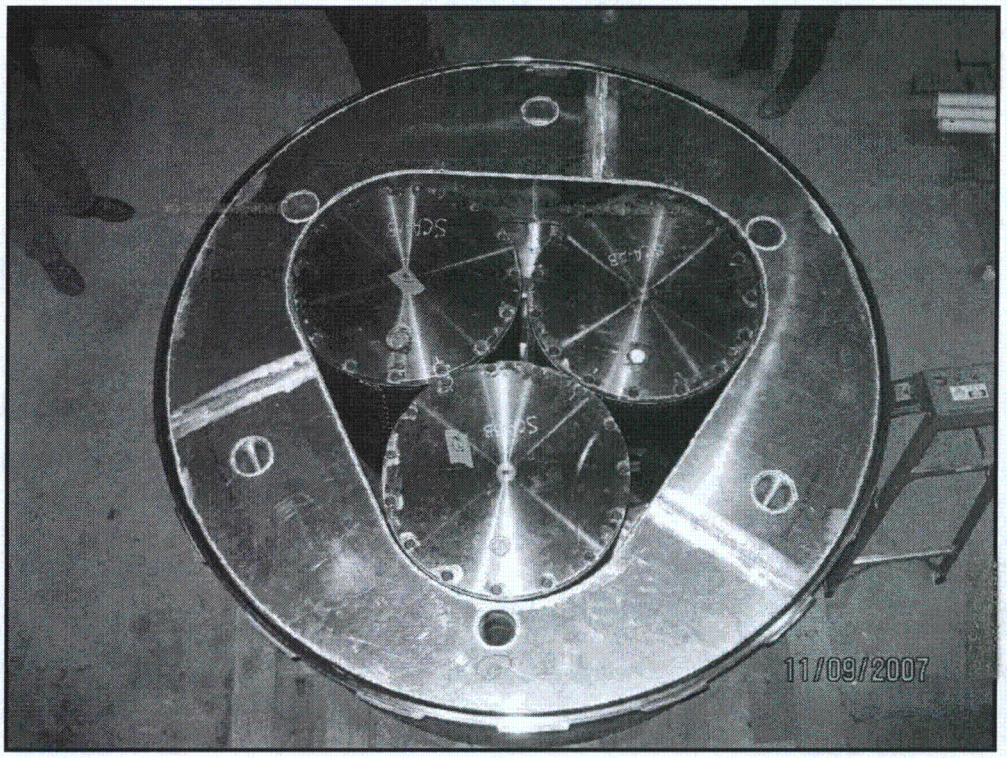

Figure 4.5-9 – Pre-Side Drop Radial Dunnage Configuration

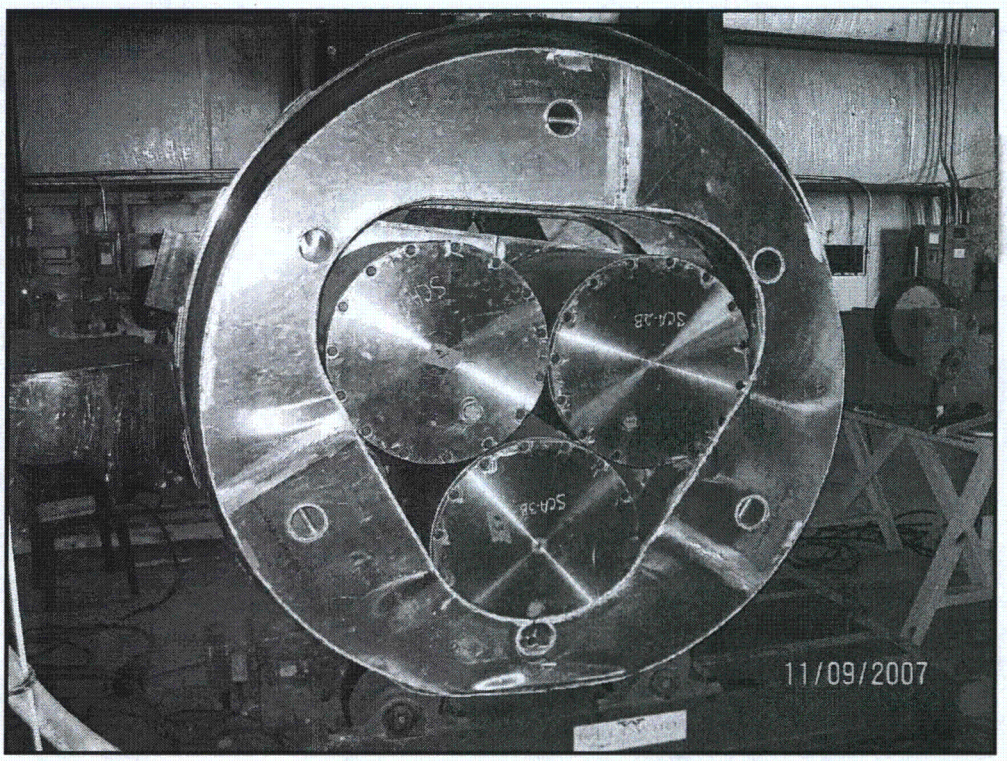

Figure 4.5-10 – Post-Side Drop Radial Dunnage Damage

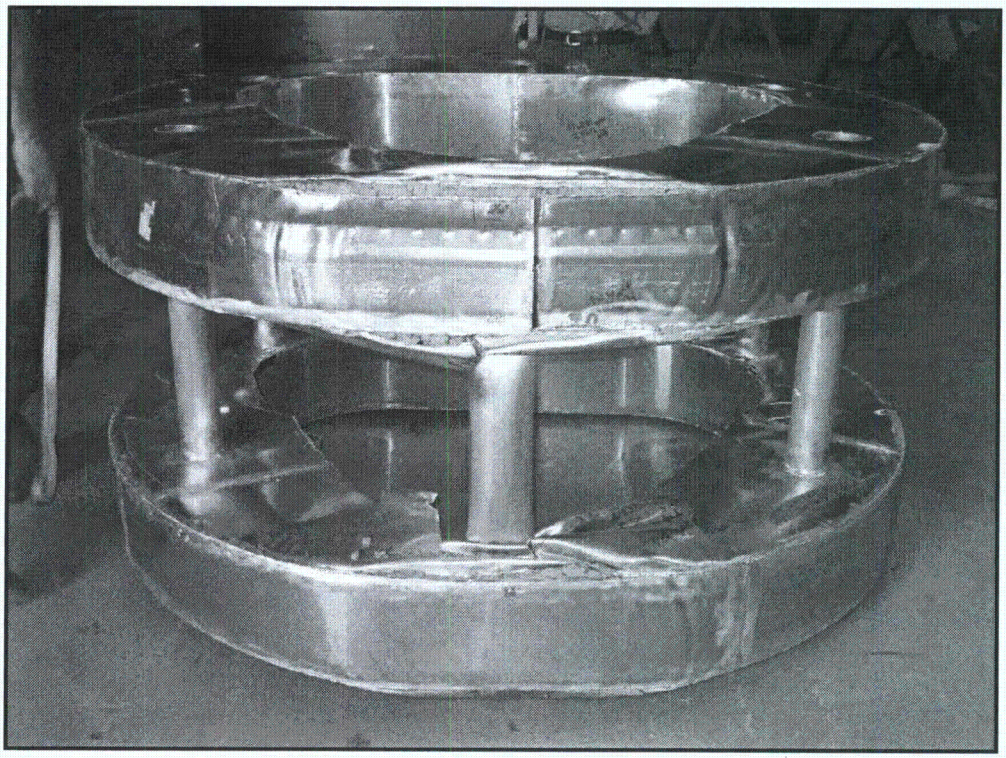

Figure 4.5-11 – Side Drop Damage to the Radial Dunnage (Outside)

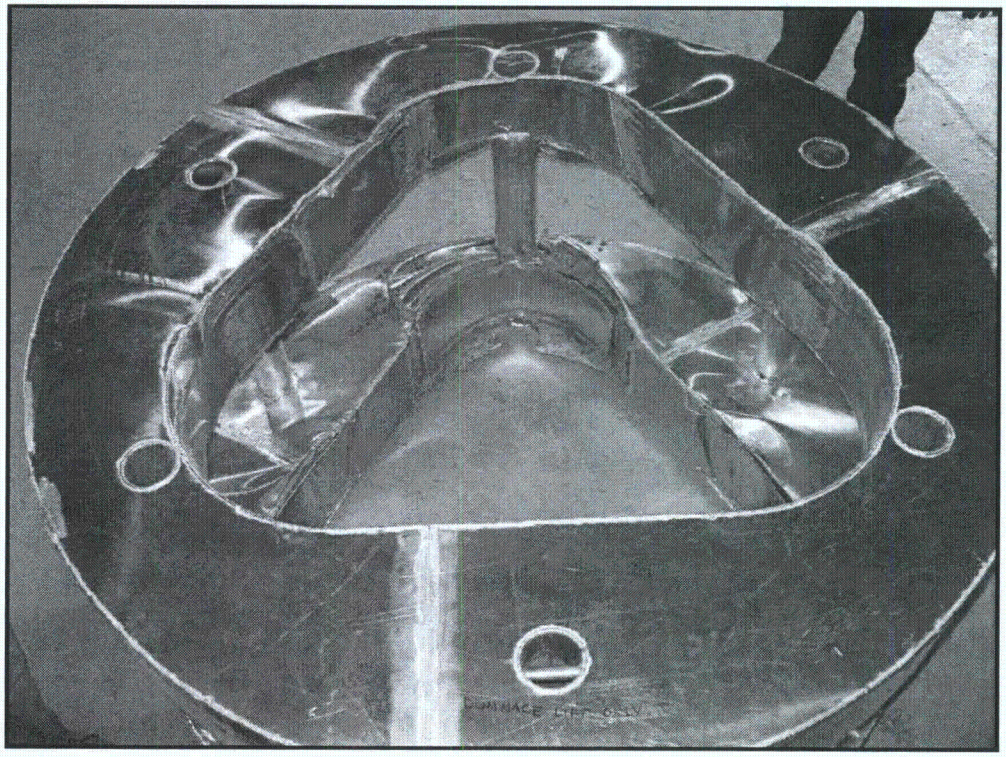

Figure 4.5-12 – Side Drop Damage to the Radial Dunnage (Inside)

The resulting measured deformation of the radial dunnage was 4⅝ inches. Adjusting for the worst-case hot temperature and the minimum allowed radial dunnage room temperature foam crush strength, the estimated maximum deformation becomes 7.07 inches, or 3/4 of the original thickness. Sufficient thickness remains to ensure that all the kinetic energy of the three shielded containers for the side drop event is fully absorbed by the radial dunnage.

Visible damage to the shielded containers was limited to localized flattening of the outer shell-to-flange/base welds that were in contact at shielded container to shielded container interface points during the side drop event.

Section 2.10.3.7.2.2 in the HalfPACT SAR[1] reports a side drop deformation of the OCA of 3¾ inches. Conservatively assuming no deformation of the radial dunnage occurs, the 3¾-inch deformation of the OCA sidewall can be used to establish a bounding lateral acceleration applicable to the shielded containers of 194*g*. Conservatively treating the shielded container as a simply supported beam, and ignoring any strength from the lead itself, the bending capacity of the shielded container's inner and outer shells is over 400*g*. Thus, significant design margin exists to prevent yielding of the shells.

4.5.3.2.3 Post-Drop Shielding Integrity Testing and Destructive Disassembly

Pre- and post-drop shielding integrity testing involved the use of a radiation detector and a Co-60 gamma source. A tripod apparatus was mounted to each shielded container's bolting flange and used to control detector/source spacing and location.

A gridded Mylar overlay allowed repeatability; grid spacing was set at 1½ inches or less, and the zero circumferential position was arbitrarily set at the outer shell's longitudinal seam weld. Each axial row consisted of 49 sets of readings around the circumference, with 24 total axial rows; 1,176 data points were taken to fully map the lead.

The pre-drop test and post-drop test curves at each axial row tend to track together, with little indication that either localized or global changes to the shielding occurred. With a few exceptions limited to the very ends of the containers, changes in measured dose rates were less than 10%.

In addition to shield integrity testing, two full length cross-sections (wall cut-outs) were taken from test shielded container B03, since it was the container subjected to the most cumulative damage. As can be seen in Figure 4.5-13 and Figure 4.5-14, lead slump did not occur, nor was movement of lead apparent anywhere along the cross-section. Additional detail relative to shielding integrity testing and sectioning of shielded container B03 is provided in Section 6.3 of the test report.[3]

4.5.3.2.4 Summary of Testing

Key test observations include the following:

1. Post-test visual inspection of the interior and exterior surfaces of the three shielded containers indicated no apparent global or localized deformation or damage to the shielded containers. The solid, concrete-filled rolling hoops in the 30-gallon test payload drums left no visible deformation of the shielded container's inner shell, even through these drums were loaded to exceed the 2,260-pound shielded container gross weight. Visible damage was limited to localized flattening (~2 inches long) of the outer shell-to-flange/base welds in contact at shielded container to shielded container interfaces during the side drop event.

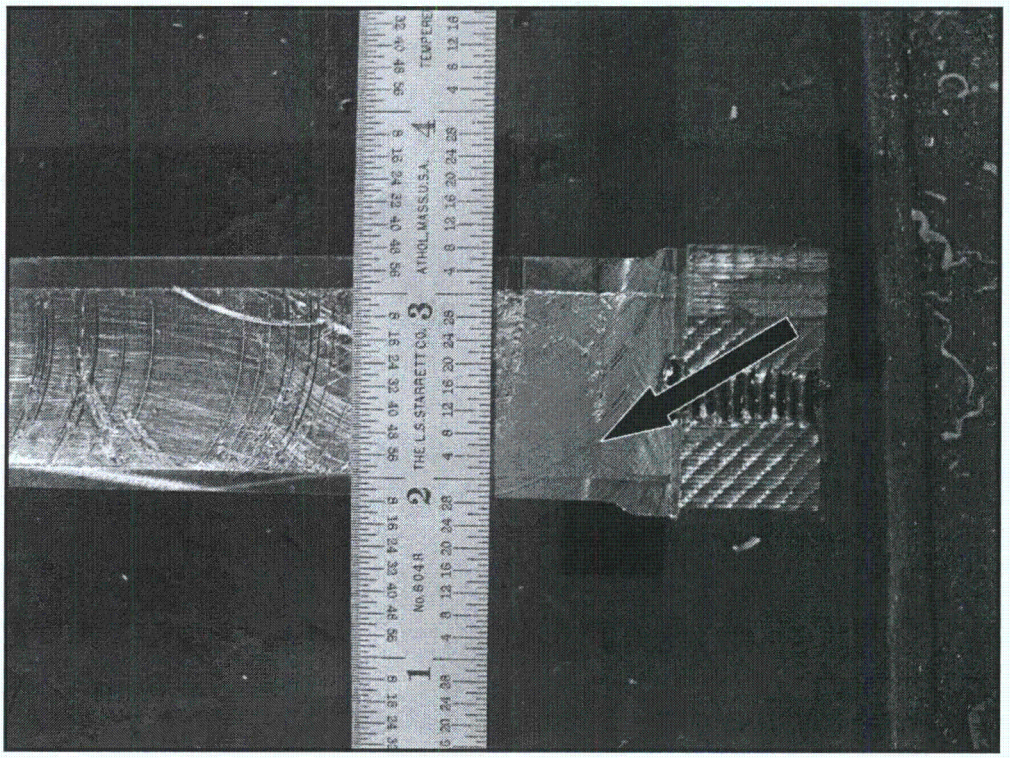

Figure 4.5-13 – Axial Slice in B03 at Lowest Point; Upper End

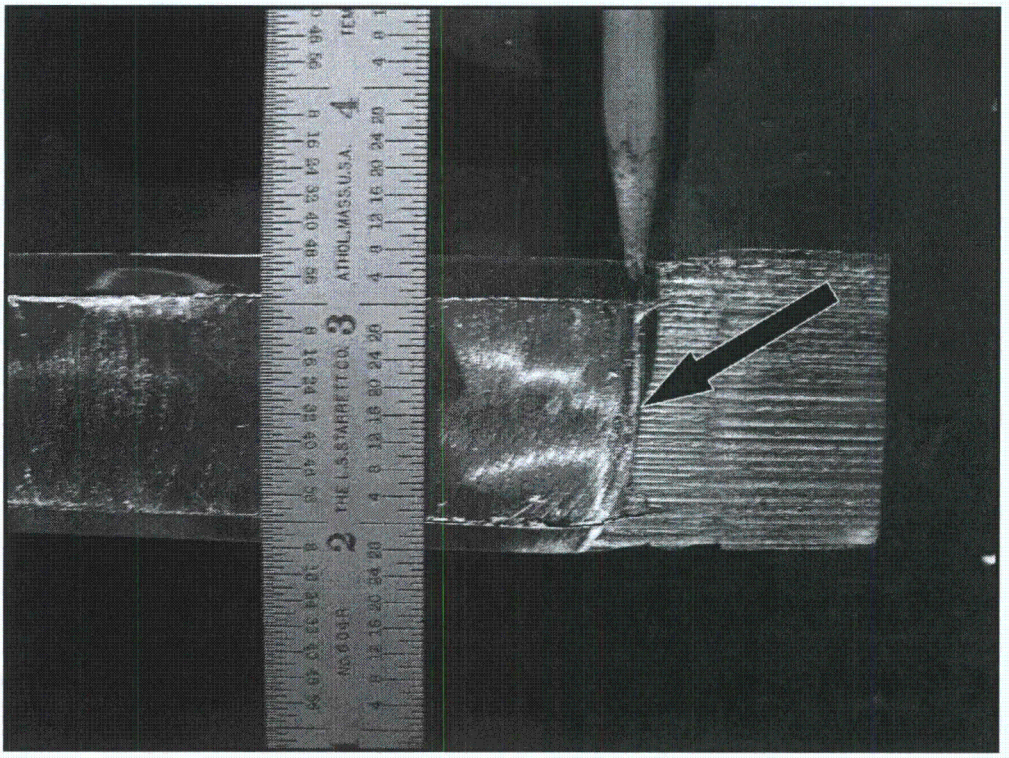

Figure 4.5-14 – Axial Slice in B03 at B02 Interface; Upper End

2. Post-test visual inspection of the HalfPACT ICV shell at its interface with payload dunnage components revealed no localized deformations that could in any way compromise containment integrity.
3. Subsequent to the performance of end and side drop testing, most closure bolts retained full residual torque, and all closure bolts retained some residual torque; 4 bolts on test shielded container B01, no bolts on test shielded container B02, and 1 bolt on test shielded container B03 lost a portion of their torque. In addition, the flour/fluorescein mixture placed within each shielded container was 100% retained throughout the testing. Collectively, these observations readily confirmed confinement integrity of the shielded containers.
4. Pre- and post-test shielding integrity tests coupled with destructive disassemblies of selected shielded container side walls showed no evidence of lead slump or changes of any significance to the shielding capabilities of the design. Post-test visual inspection of the shielded container wall cut-outs revealed some modest global and localized shell deformation, but the magnitudes were very limited, of no structural significance, and not coupled with measurable lead thinning or reduction in shielding.

In summary, the results of the testing program for the shielded containers demonstrate that under HAC the shielded containers maintain both confinement integrity and shielding integrity, and incur little visible damage.

4.5.4 Thermal Evaluation

4.5.4.1 Thermal Evaluation for Normal Conditions of Transport

The thermal analysis[4] model of the HalfPACT packaging was developed using the computer programs Thermal Desktop[5] and SINDA/FLUINT[6]. The thermal model of the shielded containers within the HalfPACT packaging is a composite of a newly generated 'solids' model of the shielded containers, the waste contents, and the associated dunnage, pallets, etc. and an existing two-dimensional lumped parameter model of the HalfPACT packaging. Using a feature of the SINDA/FLUINT computer program, these 'submodels' are combined into a single thermal model and solved simultaneously to generate a unified thermal solution. Since the thermal model of the shielded container and its pallet and dunnage assemblies represent a three-dimensional, 180° symmetry model, six segments of the two-dimensional, lumped parameter HalfPACT SAR model, each encompassing a 30° wide segment of the packaging circumference, are combined to form a matching 180° symmetry model of the packaging. Thermal connections between the individual 30° wide, two-dimensional segments are computed based on the model dimensions associated with each model node and the circumferential distance between adjacent segments. The model uses the same material properties for the package's components as used for the HalfPACT SAR analysis.[1]

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[4] G. J. Banken, *HalfPACT Shielded Container Thermal Analysis*, P04F.M2.02-03, Rev. 2, AREVA Federal Services LLC, Tacoma, WA, February 2009.

[5] Thermal Desktop®, Version 4.8, Cullimore & Ring Technologies, Inc., Littleton, CO, 2005.

[6] SINDA/FLUINT, Systems Improved Numerical Differencing Analyzer and Fluid Integrator, Version 4.8, Cullimore & Ring Technologies, Inc., Littleton, CO, 2005.

The layout of the shielded containers within the HalfPACT packaging requires that a 180° symmetry model be used to allow the simulation of both an even distribution of decay heat loading as well as a case where all the decay heat is concentrated in a single shielded container. This level of analysis requires the modeling of 1½ shielded containers. The modeling uses a combination of surface and solid elements to simulate the container's base, the inner and outer shells, the lead shielding, and the lid. The modeling conservatively assumes a uniform air gap between the lead and the outer shell of the containers due to differential shrinkage between the lead and the carbon steel following lead pour; the gap is assumed to be 0.007 inches at room temperature, and 0 inches at 620 °F.

The 30-gallon drum payload is simulated as a solid with homogenous thermal properties and a volumetric heat generation. The interior dimensions of a generic 30-gallon drum (i.e., 18.25-inch inside diameter and 27.5-inch height) are used to set the volume and surface areas of the payload drum. The payload drum is assumed to be centered in the radial direction and resting on the bottom of the shielded container. However, since the payload drum is assumed to have a rolled rim, no direct contact is assumed between the base of the payload drum and the container. Instead, heat transfer between the payload drum and the container is assumed to be via conduction and radiation across an air gap, with a 1-inch gap at the top, a 1.075-inch gap at the side, and 0.45-inch gap at the bottom.

The pallet and dunnage assemblies are used to provide radial and axial restraint of the containers during transportation and during the regulatory drop events. Heat transfer between the shielded containers and the pallet and dunnage assemblies are modeled as conduction and radiation across the air gaps. The size of the air gaps are determined by the geometry of the design, with the exception that a 0.75-inch air gap is assumed between the surfaces of the shielded containers and the radial dunnage components.

Heat transfer between the shielded containers and the inner surfaces of the HalfPACT ICV are via conduction and radiation. Heat transfer between the shielded containers, the pallet, the axial and radial dunnage assemblies, and the ICV surfaces is simulated as conduction and radiation across the intervening air gaps.

Consistent with Section 3.1.3 of the HalfPACT SAR, maximum steady-state package temperatures with insolation are determined by using a combination of solar heating values. One steady-state analysis is made using the insolation values delineated in 10 CFR §71.71(c)(1), averaged over 24 hours. This action is intended to simulate the slow thermal response that the payload and internal package components have to a varying (i.e., cyclic) solar load. The presence of the HalfPACT's OCA polyurethane foam insulation and the relatively large thermal mass on the inside of the foam isolates (i.e., decouples) the thermal response of the internal components from the "12 hour on / 12 hour off" solar step function cycle applicable to the outside of the package. Thus, the peak temperatures of the components on the inside of the polyurethane foam are determined by applying the insolation values of 10 CFR §71.71(c)(1), averaged over 24 hours, to the exterior of the package. In contrast, the outer sections of the polyurethane foam and the OCA outer shell will respond more quickly to varying external solar loads. Therefore, the maximum steady-state temperatures of the polyurethane foam and OCA outer shell are estimated using another steady-state analysis and the 10 CFR §71.71(c)(1) insolation values averaged over 12 hours applied to the exterior of the package.

The payload within the shielded containers is conservatively assumed to be crumpled paper and to exhibit the thermal conductivity of air and possess zero thermal mass. This conservative representation of the payload bounds the potential temperature rise and temperature limit within a generic payload whose makeup prevents significant heat transfer via radiation and where its thermal conductance is dominated by trapped air spaces.

Table 4.5-1 summarizes the results for the NCT hot condition with insolation applied and 30 watts of decay heat distributed in three shielded containers (10 watts each), contrasted with the similar analysis in the HalfPACT SAR for seven 55-gallon drums. Similarly, Table 4.5-2 summarizes the results for the NCT hot condition with insolation applied and 30 watts of decay heat concentrated in one shielded container, contrasted with the similar analysis in the HalfPACT SAR for seven 55-gallon drums.

Table 4.5-1 – NCT Hot Temperatures with 30 Watts Evenly Distributed

Location	Solar Loading	Temperature (°F) for NCT Hot with Insolation	
		Three Shielded Containers	Seven 55-Gallon Drums (HalfPACT SAR)
Payload Drum Centerline			
• Shielded Container 1 / Center 55-Gallon Drum	24-hr avg	222	184
• Shielded Container 2 & 3 / Outer 55-Gallon Drums	24-hr avg	222	181
Payload Drum Bulk Average			
• Shielded Container 1 / Center 55-Gallon Drum	24-hr avg	179	170
• Shielded Container 2 & 3 / Outer 55-Gallon Drums	24-hr avg	178	168
Shielded Container 1 / Center 55-Gallon Drum			
• Base	24-hr avg	155	156
• Lid / Seal	24-hr avg	156	156
• Sidewall	24-hr avg	156	156
Shielded Container 2&3 / Outer 55-Gallon Drums			
• Base	24-hr avg	155	153
• Lid / Seal	24-hr avg	156	153
• Sidewall	24-hr avg	156	153
ICV Wall			
• Maximum	24-hr avg	150	153
• Average	24-hr avg	149	149
• Minimum	24-hr avg	148	146
ICV Air			
• Average	24-hr avg	154	151
Main O-ring Seals			
• ICV Maximum	24-hr avg	148	147
• OCV Maximum	24-hr avg	147	146
OCV Wall			
• Maximum	24-hr avg	149	150
• Average	24-hr avg	147	147
Polyurethane Foam			
• Maximum	12-hr avg	152	155
• Bulk Average	24-hr avg	128	129
OCA Outer Shell			
• Maximum	12-hr avg	152	155

Table 4.5-2 – NCT Temperatures with 30 Watts Concentrated

Location	Solar Loading	Temperature (°F) for NCT Hot with Insolation	
		All Heat in One Shielded Container	All Heat in Center 55-Gallon Drum (HalfPACT SAR)
Payload Drum Centerline			
• Shielded Container 1 / Center 55-Gallon Drum	24-hr avg	342	340
• Shielded Container 2 & 3 / Outer 55-Gallon Drums	24-hr avg	155	153
Payload Drum Bulk Average			
• Shielded Container 1 / Center 55-Gallon Drum	24-hr avg	224	252
• Shielded Container 2 & 3 / Outer 55-Gallon Drums	24-hr avg	154	153
Shielded Container 1 / Center 55-Gallon Drum			
• Base	24-hr avg	156	163
• Lid / Seal	24-hr avg	158	163
• Sidewall	24-hr avg	158	163
Shielded Container 2&3 / Outer 55-Gallon Drums			
• Base	24-hr avg	155	153
• Lid / Seal	24-hr avg	155	153
• Sidewall	24-hr avg	155	153
ICV Wall			
• Maximum	24-hr avg	150	154
• Average	24-hr avg	149	148
• Minimum	24-hr avg	148	145
ICV Air			
• Average	24-hr avg	154	151
Main O-ring Seals			
• ICV Maximum	24-hr avg	148	145
• OCV Maximum	24-hr avg	147	144
OCV Wall			
• Maximum	24-hr avg	149	150
• Average	24-hr avg	147	146
Polyurethane Foam			
• Maximum	12-hr avg	152	155
• Bulk Average	24-hr avg	128	128
OCA Outer Shell			
• Maximum	12-hr avg	152	155

As can be seen in Table 4.5-1 and Table 4.5-2, temperatures for the shielded container payload configuration are very similar to corresponding temperatures for the 55-gallon drum payload from the HalfPACT SAR, especially for the HalfPACT packaging components. As such, the HalfPACT packaging components maintain the same margin of safety when transporting a shielded container payload compared to transporting a 55-gallon drum payload (or any other currently licensed payload).

4.5.4.2 Thermal Evaluation for Hypothetical Accident Conditions

No safety evaluations for HAC are required for the shielded container payload since the results are bounded by those presented in the HalfPACT SAR. The basis for this conclusion is as follows:

1. The level of heat input into the HalfPACT package during the HAC event is a function of the package's exterior surface area, the thermal mass of the package components, etc., which is essentially unaffected by the makeup of the payload.
2. The temperature response within the payload is a function of its thermal mass and the amount of heat passed to it by the HalfPACT packaging.
3. Since the heat input to the HalfPACT packaging during the HAC event is essentially the same between a package containing a base payload evaluated in the HalfPACT SAR and a package containing a payload of three shielded containers, the thermal HAC response of the HalfPACT package will be bounded by that presented in HalfPACT SAR.
4. Given a similar temperature response for the ICV shell under HAC conditions for either the base payload or the shielded container payload, the thermal response of the shielded container will be inversely proportional to the thermal mass and directly proportional to the surface area of the shielded container payload versus that existing for the base payload. The maximum payload mass (i.e., the packaging contents which includes the payload containers, waste contents, dunnage, pallets, etc) of the shielded container payload is identical to the maximum base payload mass. However, the combined surface area of the shielded container payload is lower than the combined surface area of the base payload drum payload. As such, the rate of heat transfer between the ICV and the shielded containers will be lower under HAC conditions than seen with the base payload due to the lower area for radiation and convection/conduction heat transfer. A lower rate of heat transfer combined with an equal payload mass means the temperature rise experienced by the shielded container payload will be bounded by that experienced by the base payload.

As such, the transient thermal behavior and the rise in the temperatures for the shielded containers within the HalfPACT packaging during the HAC fire event is bounded by that (i.e., $\Delta T = 156°F$ to $290°F = +134°F$) reported for the base payload in Section 3.5.3 of the HalfPACT SAR. Since the melting point for lead is 620 °F and the temperature limit for the silicone rubber seal material is 450 °F[7], a HAC fire event would not reduce the effectiveness of the shielded container.

[7] ORD 5700, *Parker O-ring Handbook*, 1992, Parker Hannifin Corporation, Cleveland, OH.

4.5.5 Shielding Evaluation

The evaluation of compliance with the radiation dose rate limits for NCT and HAC required by 10 CFR §71.47 is presented in Chapter 5 of the HalfPACT SAR[1] for the shielded container payload configuration. When the HalfPACT package is loaded with an assembly of shielded containers containing gamma and/or neutron source terms that are limited per Section 3.3 of the CH-TRAMPAC[2], the package meets the NCT radiation dose rate requirements of 200 mrem/hr at the surface of the package and 10 mrem/hr at 2 meters from the surface of the package under exclusive use. As a result, the packages also comply with the HAC dose rate requirement of 1 rem/hr at 1 meter from the surface of the package.

4.5.6 Criticality Evaluation

Shielded containers are designed to transport TRU waste forms with high gamma energies within a HalfPACT package. A criticality evaluation[8] was performed for two different payload cases: (1) manually compacted waste, and (2) machine compacted waste. A maximum 325 fissile gram equivalent (FGE) of Pu-239 is justified for manually compacted waste, while a lower limit of 245 FGE is justified for machine compacted waste. The methodology and assumptions utilized in the existing HalfPACT SAR are also utilized in the current analysis. The following analyses demonstrate that this configuration complies with the requirements of 10 CFR §71.55 and §71.59. The criticality safety index, per 10 CFR §71.59, is 0.

Two general cases were developed. For Case G[9](manually compacted waste), the moderator was modeled as a composition of 25% polyethylene and 75% water (by volume). As polyethylene is a superior moderator than water, this composition results in higher reactivites than would be achieved by water moderation alone. This volume fraction of polyethylene is conservatively higher than the maximum value achievable for manually compacted (i.e., not machine compacted) waste determined by experiment. The reflector is modeled as a mixture of 25% polyethylene, 74% water, and 1% beryllium (by volume). Beryllium is a superior reflector than either water or polyethylene and the inclusion of beryllium is conservative, although at such a small volume fraction, the beryllium has only a small effect on the system reactivity.

For Case H (machine compacted waste), the moderator was modeled as 100% polyethylene. As polyethylene is a superior moderator than water, this composition results in higher reactivities than would be achieved by water moderation alone. The reflector is modeled as a mixture of 99% polyethylene and 1% beryllium (by volume). Beryllium is a superior reflector than polyethylene and the inclusion of beryllium is conservative, although at such a small volume fraction, the beryllium has only a small effect on the system reactivity.

Calculations for the HalfPACT package are performed using the three-dimensional Monte Carlo transport theory code, KENO-V.a v5.0.2, with the CSAS25 utility being used as a driver for the KENO-V.a code; both programs are part of the SCALE-PC v5[10] code system. In this role,

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[8] R. J. Migliore, *HalfPACT Shielded Container Criticality Analysis*, P04F.M2.02-02, Rev. 0, Packaging Technology, Inc., Tacoma, WA, December 2007.

[9] To avoid confusion, the case designations are selected to be additions to those utilized in the current HalfPACT SAR. Cases G and H are equivalent to Cases A and C, respectively, with modifications specific to the shielded container.

[10] *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, ORNL/TM-2005/39, Version 5, Vols. I-III, April 2005.

CSAS25 determines nuclide number densities, performs resonance processing, and automatically prepares the necessary input for the KENO-V.a code based on a simplified input description. The 238 energy-group (238GROUPNDF5), cross-section library based on ENDF/B-V cross-section data is used as the nuclear data library for the KENO-V.a code.

The upper subcritical limit (USL) for ensuring that the HalfPACT is acceptably subcritical, as determined in benchmark evaluations, is:

$$USL = 0.9377$$

The package is considered to be acceptably subcritical if the computed k_{safe} (k_s), which is defined as $k_{effective}$ (k_{eff}) plus twice the statistical uncertainty (σ), is less than the USL, or:

$$k_s = k_{eff} + 2\sigma < USL$$

In all models, the fissile material was assumed to form a single optimally moderated sphere. In actual practice, such a scenario is not credible because each shielded container is limited to 200 FGE. It is extremely unlikely that fissile material could escape from the shielded containers and reconfigure, or travel from one shielded container to another and reconfigure.

Conservative damage assumptions were utilized in both the NCT and HAC analysis. No credit was taken for the torispherical head of the HalfPACT, which would have increased separation distance in the array configuration. All foam and aluminum regions were replaced with reflector at the most reactive density. In the array models, the internal and external reflector densities were varied in order to maximize neutron interaction between packages.

This calculation modeled 1% by weight beryllium to account for the ≤1% by weight presence of any special reflector materials. Special reflectors (other than beryllium) that are in >1% by weight quantities are allowed if they are chemically or mechanically bound to the fissile material. Lead and steel were not considered "special reflectors," although these materials are more reflective than poly/water at large thicknesses. As the shielded containers have thick steel lids and bottoms, and lead side walls clad in steel, the presence of the shielded containers slightly increased the reactivity. Various configurations of fissile sphere and shielded container were utilized. The most reactive configuration (for both single package and array) always occurred when the fissile sphere was in a corner of a shielded container, which maximized reflection.

The maximum reactivity of the single package and infinite array models were nearly identical for most cases. This indicated that neutron communication between packages was rather limited, and the fissile material was largely isolated. Note that differences of approximately 0.002 between the various model results is often simply due to statistical fluctuation. The most reactive HAC single package model, with three shielded containers in a row, the fissile sphere shifted to the upper right corner in the center shielded container, and an H/Pu ratio of 900 (k_s = 0.9372), is statistically equivalent to the most reactive HAC array model, with one shielded container modeled at the side and top of the ICV, the fissile sphere located at the side and top of the shielded container, the OCA and external reflector modeled as a void, and an H/Pu ratio of 900 (k_s = 0.9368), although the arrangements within the package are quite different. Case G results in higher reactivities than Case H, although Case H has a much lower fissile mass. All results are below the USL of 0.9377.

Case G, the manually compacted waste stream, has a justifiable limit of 325 FGE per HalfPACT, and Case H, the machine compacted waste stream, has a justifiable limit of 245 FGE per HalfPACT. The corresponding results are summarized in Table 4.5-3.

Table 4.5-3 – Summary of Criticality Evaluation Results

Limit	Case G 325 FGE Manually Compacted Waste	Case H 245 FGE Machine Compacted Waste
Normal Conditions of Transport (NCT)		
	k_s	k_s
Single Unit Maximum	0.9354	0.9302
Infinite Array Maximum	0.9355	0.9313
Hypothetical Accident Conditions (HAC)		
	k_s	k_s
Single Unit Maximum	0.9372	0.9298
Infinite Array Maximum	0.9368	0.9340
USL = 0.9377		

4.5.7 Authorized Payload Contents for the Shielded Container

As demonstrated in Section 4.5.5, *Shielding Evaluation*, when loaded with gamma and/or neutron emitting isotopes with maximum activity limits summarized in the CH-TRAMPAC, the shielded container payload meets the NCT and HAC dose rate limits. As demonstrated in Section 4.5.6, *Criticality Evaluation*, when loaded with fissile material with maximum mass limits summarized for Cases G and H in Table 4.5-3, the shielded container payload meets the calculated reactivity limit and is safely subcritical.

4.5.8 Conclusion

The shielded container design consists of a vented carbon steel and lead cylindrical structure with a removable lid, surrounded by axial and radial dunnage, that is to be used for shipment of specific transuranic waste forms in the HalfPACT package.

The analyses summarized in this appendix demonstrate the ability of the shielded container to safely transport limited quantities of gamma and/or neutron emitting isotopes and fissile isotopes. Using geometries consistent with, or conservative with respect to, the structural and thermal analyses, the shielding evaluation showed that the dose rate limits for NCT and HAC (including appropriate shielding damage assumptions in each case) are met with the maximum authorized contents. In addition, the criticality evaluation showed that the reactivity limit is met for manually or machine compacted wastes with specified mass limits.

APPENDIX 4.6

DESCRIPTION OF CRITICALITY CONTROL OVERPACK

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4.6 Description of Criticality Control Overpack

4.6.1 Introduction

The Criticality Control Overpack (CCO) is an approximately 24-inch diameter, 35-inch tall steel 55-gallon drum with a stainless steel Criticality Control Container (CCC) and plywood Upper and Lower Dunnage assemblies. It is designed to be used for shipment of increased fissile waste contents in the TRUPACT-II and HalfPACT packages. Drawing 163-009 in Appendix 1.3.1 of the TRUPACT-II Safety Analysis Report[1] (SAR) and HalfPACT SAR[2] and Section 2.9.11 of the *Contact-Handled Transuranic Waste Authorized Methods for Payload Control* (CH-TRAMPAC)[3] delineate the materials of construction, sizes, and other dimensional specifications for the CCO.

The TRUPACT-II and HalfPACT packages can accommodate an assembly of fourteen (14) and seven (7) CCOs, respectively. As configured for shipment, the CCO payload assembly remains conservatively within the previously established design and certification bases and limits of the TRUPACT-II and HalfPACT packages for payload weight and decay heat. Limits on CCO activity and fissile content are also set consistent with previously implemented and accepted analytic approaches.

This appendix describes the structural, thermal, shielding, and criticality bases of the CCO payload.

4.6.2 Description

The CCO consists of a steel 55-gallon drum containing a CCC confinement vessel that is centrally positioned within the drum by laminated plywood dunnage. The CCC is constructed of 304/304L stainless steel 6-inch Class 150 standard blind and slip flanges and Schedule 40 pipe (NPS). The lid of the CCC is sealed with an aramid-inorganic/nbr standard ring gasket and retained with eight (8) 3/4-inch heavy hex head stainless steel bolts. The lid and base of the CCC are nominally 1 inch thick and the pipe shell is nominally 0.28 inches thick with an overall assembly height of approximately 29-1/2 inches. The CCO has an approximate tare weight of 230 pounds and a maximum gross weight of 350 pounds. A lifting attachment is optionally integrated into the CCC lid to facilitate handling.

Partially exploded views of the CCO and CCC are provided in Figure 4.6-1 and Figure 4.6-2, respectively. Both the 55-gallon drum and CCC must be fitted with a filter vent; Section 2.5 of the CH-TRAMPAC provides the minimum specification for the CCO filter vents.

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[1] U.S. Department of Energy (DOE), *TRUPACT-II Shipping Package Safety Analysis Report*, USNRC Certificate of Compliance 71-9218, U.S. Department of Energy, Carlsbad Field Office, Carlsbad, New Mexico.

[2] U.S. Department of Energy (DOE), *HalfPACT Shipping Package Safety Analysis Report*, USNRC Certificate of Compliance 71-9279, U.S. Department of Energy, Carlsbad Field Office, Carlsbad, New Mexico.

[3] U.S. Department of Energy (DOE), *Contact-Handled Transuranic Waste Authorized Methods for Payload Control* (CH-TRAMPAC), U.S. Department of Energy, Carlsbad Field Office, Carlsbad, New Mexico.

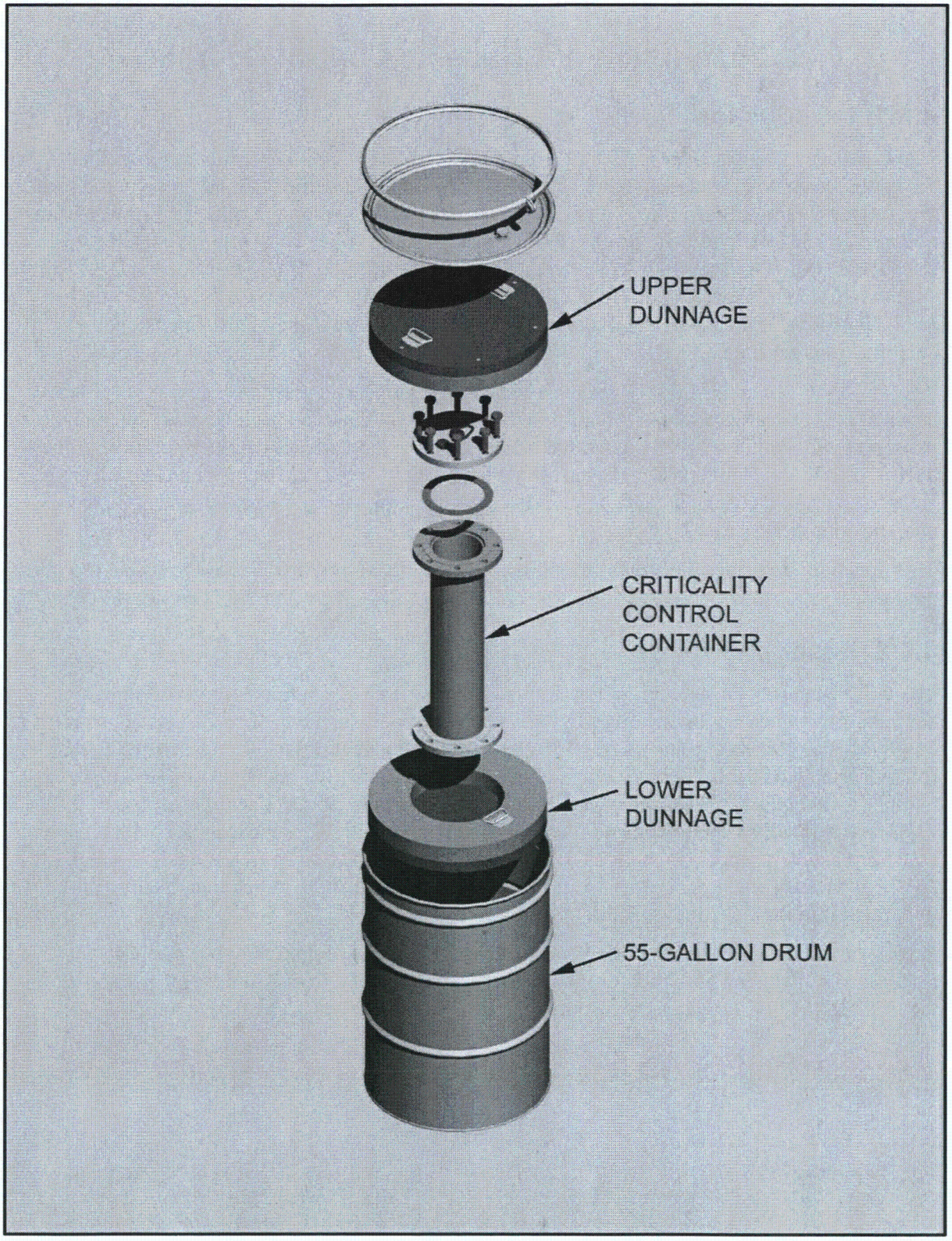

Figure 4.6-1 – Criticality Control Overpack

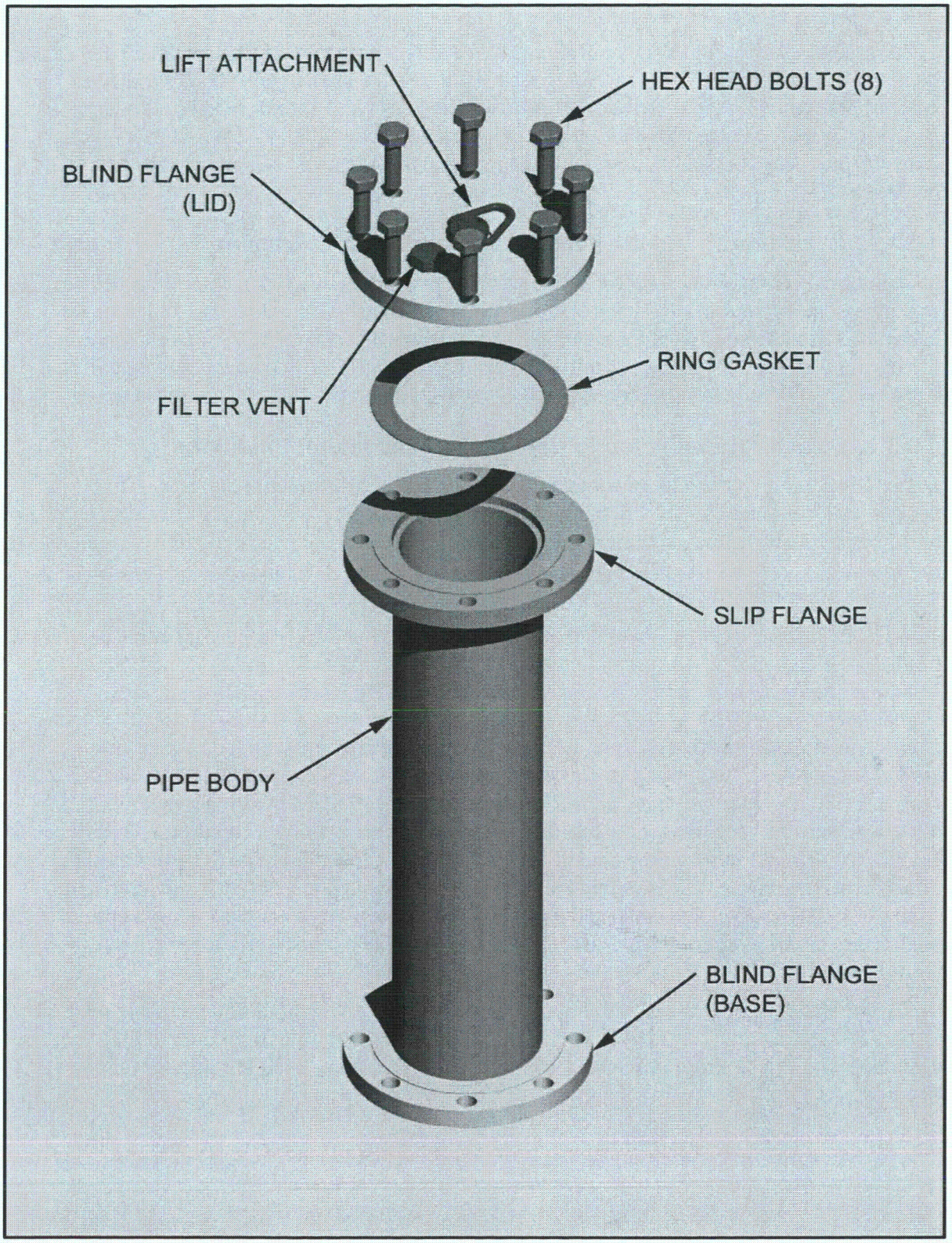

Figure 4.6-2 – Criticality Control Container

4.6.3 Structural Evaluation

The CCO payload configuration is, from a TRUPACT-II and HalfPACT packaging perspective, bounded by previous certification testing for the TRUPACT-II and HalfPACT package as currently presented in the TRUPACT-II and HalfPACT SARs. Therefore, the following structural evaluations are specific to determining the response of the CCOs when subject to transport and accident conditions when transported in the TRUPACT-II and HalfPACT packages.

4.6.3.1 Structural Evaluation for Normal Conditions of Transport

Under normal conditions of transport (NCT), the CCO maintains confinement, shielding integrity, and array spacing for criticality control. Since confinement, shielding integrity, and criticality control has been demonstrated for hypothetical accident conditions (HAC) without loss of fine particulate confinement, as discussed in Section 4.6.3.2, *Structural Evaluation for Hypothetical Accident Conditions*, and HAC bounds NCT, demonstrations specific to NCT are not necessary.

4.6.3.2 Structural Evaluation for Hypothetical Accident Conditions

Under HAC, the CCO retains its contents within the CCC confinement boundary. To demonstrate confinement, shielding integrity, and adequate array spacing for criticality control of the CCO, a full-scale test program was conducted.[4] Since confinement integrity was maintained, the shielding capability of the container was not reduced, and adequate array spacing was maintained for criticality control during HAC testing, NCT is bounded by the HAC test program.

Two CCO test articles were assembled and attached to side and end drop test fixtures comprised of plate steel to conservatively simulate the overburden forces and boundary conditions associated with a payload assembly of CCOs within the TRUPACT-II packaging Inner Containment Vessel (ICV). The impact-attenuating characteristics of the TRUPACT-II Outer Confinement Assembly (OCA), with its energy-absorbing polyurethane foam, was conservatively neglected from consideration in the tests. Each CCO test article and associated drop test fixture was subjected to a 30-foot free drop onto a flat, essentially unyielding, horizontal surface: one horizontal side drop and one vertical top-down end drop with the CCO at hot (>200 °F) conditions. At the conclusion of the 30-foot free drops, the CCO test articles were dimensionally inspected to determine the radial crush deformation due to the side drop and the axial crush deformation due to the end drop. The CCC test articles were also disassembled from the CCOs, misted with water, and visually examined for the presence of fluorescein dye to verify confinement integrity.

To conservatively test to the maximum allowable CCO gross weight of 350 pounds, each CCO test article was directly loaded with 137 pounds of a 50/50 (by volume) mixture of lead shot and sand topped with a flour/fluorescein indicator mixture, for a total CCO test article weight of 352 pounds.

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[4] Petersen Inc., *Criticality Control Overpack 30-Foot Free Drop Post-Test Summary Report*, Engineering Report 8448-R-001, Rev. 1, Ogden UT, March 2011.

4.6.3.2.1 Technical Basis for the Tests

The following sections supply the technical basis for the chosen free drop test orientations, the use of test fixtures as surrogates for the CCO payload assembly, and the free drop test temperatures and pressures.

4.6.3.2.1.1 Justification for Drop Orientations

To address CCO performance, it is only necessary to perform 30-foot free drop tests for the end and side orientations. Intermediate impact angles simply distribute the interaction forces at lower g-levels between the CCOs and the ICV, whereas the 0° and 90° impact orientations maximize the impact accelerations and localized bearing forces in a manner to maximize the potential for CCC confinement boundary and CCO dunnage crush damage. As such, side and end drop test orientations are bounding for the design.

The requirements of 10 CFR §71.73(c)(1) are satisfied as the drop test orientations are associated with positions for which maximum damage to the CCO is expected.

4.6.3.2.1.2 Justification for Use of Side and End Drop Test Fixtures

Due to the maximum gross weight limit of the CCO being 350 pounds, resulting in a total payload assembly weight that is significantly lower than the 7,265-pound and 7,600-pound payload capacity authorized for the TRUPACT-II and HalfPACT packagings, respectively, the packaging response to a CCO payload is bounded by the current packaging certification tests.

Both the side and end drops are performed on an unprotected (bare) CCO, resulting in higher deformations and acceleration loads to the CCO payload than if inside the impact-attenuating TRUPACT-II or HalfPACT packagings. Therefore, the CCO drop tests are a conservatively bounding determination of the minimum post-drop radial and axial CCO array spacing for the criticality evaluation.

4.6.3.2.1.2.1 Side Drop

As shown in Figure 4.6-3, the CCO side drop test fixture is designed to load the CCO test article with a compressive load equal to four CCOs. The basis for the loading configuration is derived from the configuration depicted in Figure 4.6-5. The total vertical load on the bottom CCO is based on the weight of two CCOs atop the bottom CCO, plus one-half the weight from each of the four side CCOs; the other half of the weight from each of the four side CCOs is assumed to be carried by the ICV. Note that the component of horizontal compressive loading into the bottom CCO caused by the four side CCOs would tend to oppose the vertical compressive load and is, therefore, ignored to maximize overall crush and minimize radial spacing.

4.6.3.2.1.2.2 End Drop

As shown in Figure 4.6-4, the CCO end drop test fixture is designed to load the CCO test article with a compressive axial load equal to one CCO. The end drop test fixture duplicates the two-high drum configuration in the TRUPACT-II package and will result in the maximum overall crush and minimum axial spacing.

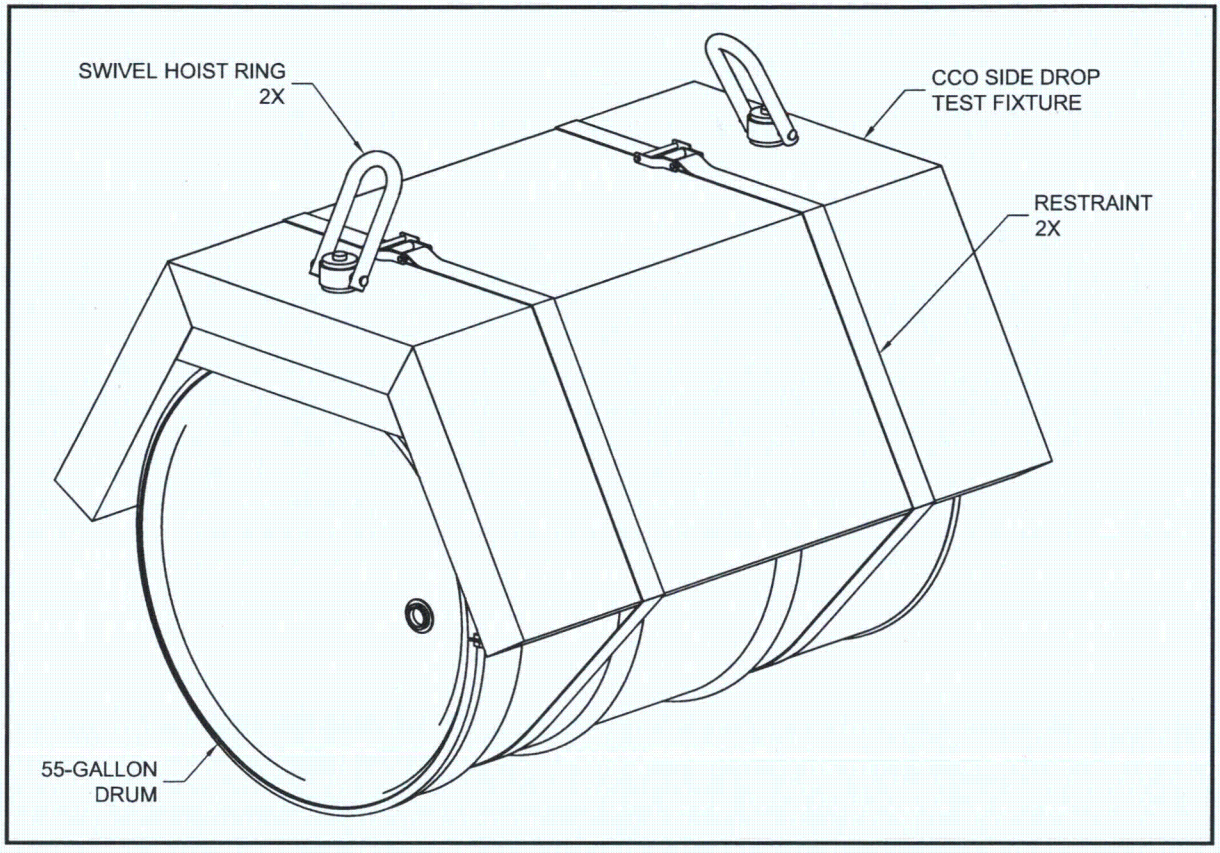

Figure 4.6-3 – Side Drop Test Configuration

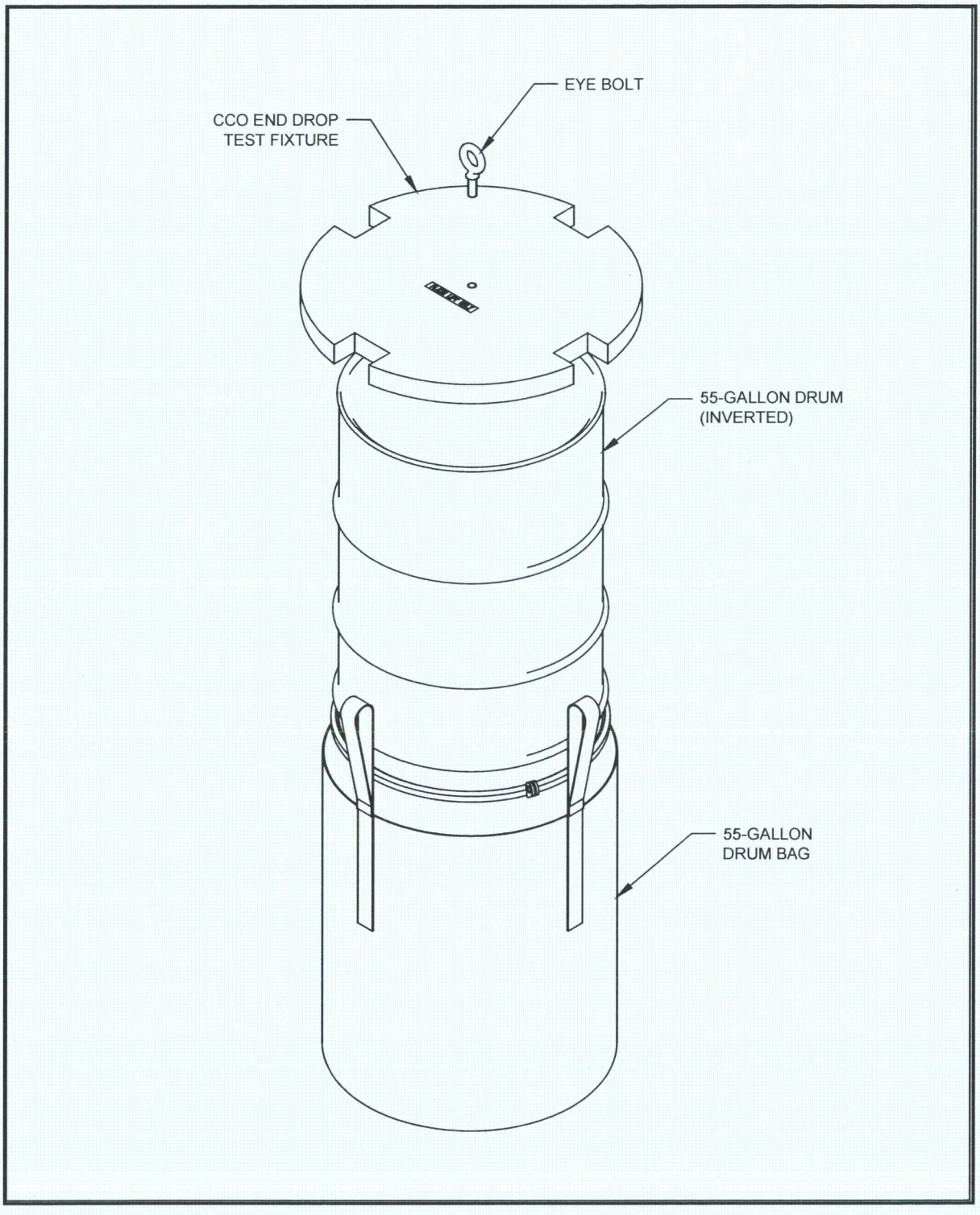

Figure 4.6-4 – End Drop Test Configuration

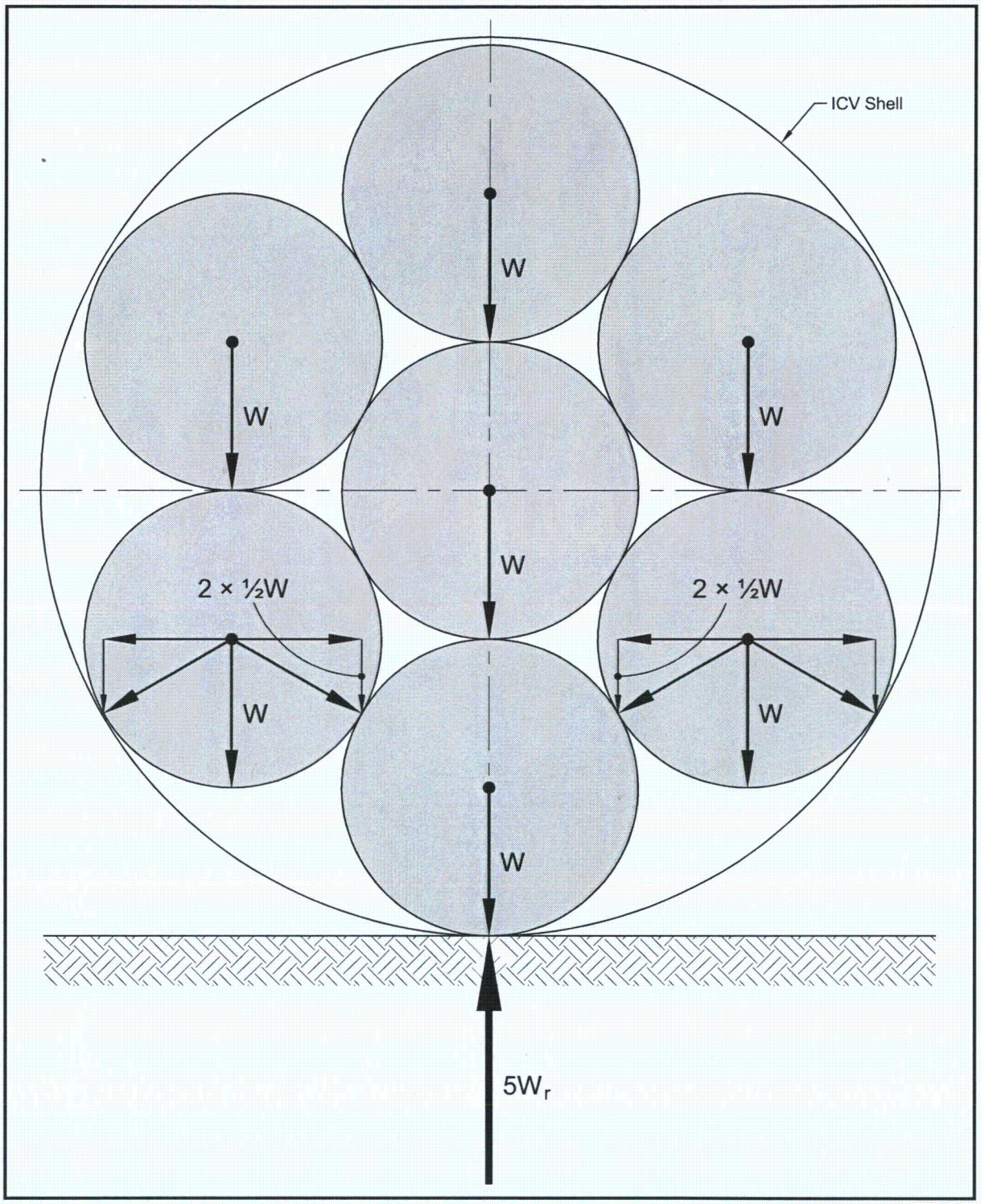

Figure 4.6-5 – Side Loading Free Body Diagram

4.6.3.2.1.3 Justification for Test Temperatures

The CCO side and end drops were performed with the CCO test article conditioned to >200 °F to bound the NCT hot condition. The elevated temperature maximizes the crush deformation of the plywood upper and lower dunnage assemblies, conservatively providing an upper bound on the post-drop CCO payload assembly array spacing for criticality purposes.

4.6.3.2.1.4 Justification for Test Pressure

These CCOs are vented and not subject to differential pressures; hence, internal pressurization of the CCO is not applicable.

4.6.3.2.2 Side Drop Results

The 30-foot hot side drop was performed using a CCO side drop test fixture weighing 1,452 pounds attached to the CCO side drop test article that was dropped in a horizontal orientation to radially impact directly onto the essentially unyielding drop pad surface. The test conservatively simulated the interaction between the lowest CCO and the six upper CCOs in a seven-pack array inside a TRUPACT-II or HalfPACT ICV. The test configuration neglected both the impact-attenuating characteristics of the packaging and the compliance of the upper CCOs in the array to maximize the impact accelerations on the CCC and crush deformation of the dunnage assemblies.

Post-drop inspection of the CCO indicated significant crushing of the 55-gallon drum and internal plywood dunnage assemblies that resulted in a minimum effective diameter of the CCO that measured 16¾ inches at the lid end and 15½ inches at the base end of the container (pre-drop diameter of the 55-gallon drum was ∅24 inches). The upper and lower plywood dunnage assemblies experienced a corresponding radial crush deformation, but the assemblies attenuated the impact and prevented any direct interaction between the CCC and the test fixture/pad. The measured accelerations due to impact were recorded with filtering of the data utilizing a low-pass Butterworth 10-pole filter having a 250 Hz cut-off frequency. A minimum impact acceleration peak, on average from two sensors, of approximately 233 g's was recorded.

Disassembly of the CCO and inspection of the CCC indicated no permanent plastic deformation of any of the confinement boundary components, essentially unaltered preload of the closure bolts (maximum bolt rotation to achieve installation torque value of 3.5°), and no loss of confinement as confirmed via black-light inspection of the assembly with no presence of the fluorescein indicator.

Figure 4.6-6 shows the CCO side drop test article and test fixture assembly prior to the drop test. Figure 4.6-7 shows the assembly during and after the side impact event. Figure 4.6-8 shows the post-test measurement of the effective diameter of the CCO and associated radial crushing of the dunnage assemblies. Figure 4.6-9 shows the CCC after the post-drop test confinement evaluation and bolt residual torque confirmation process, indicating no degradation of the confinement vessel.

Figure 4.6-6 – Side Drop Test Assembly

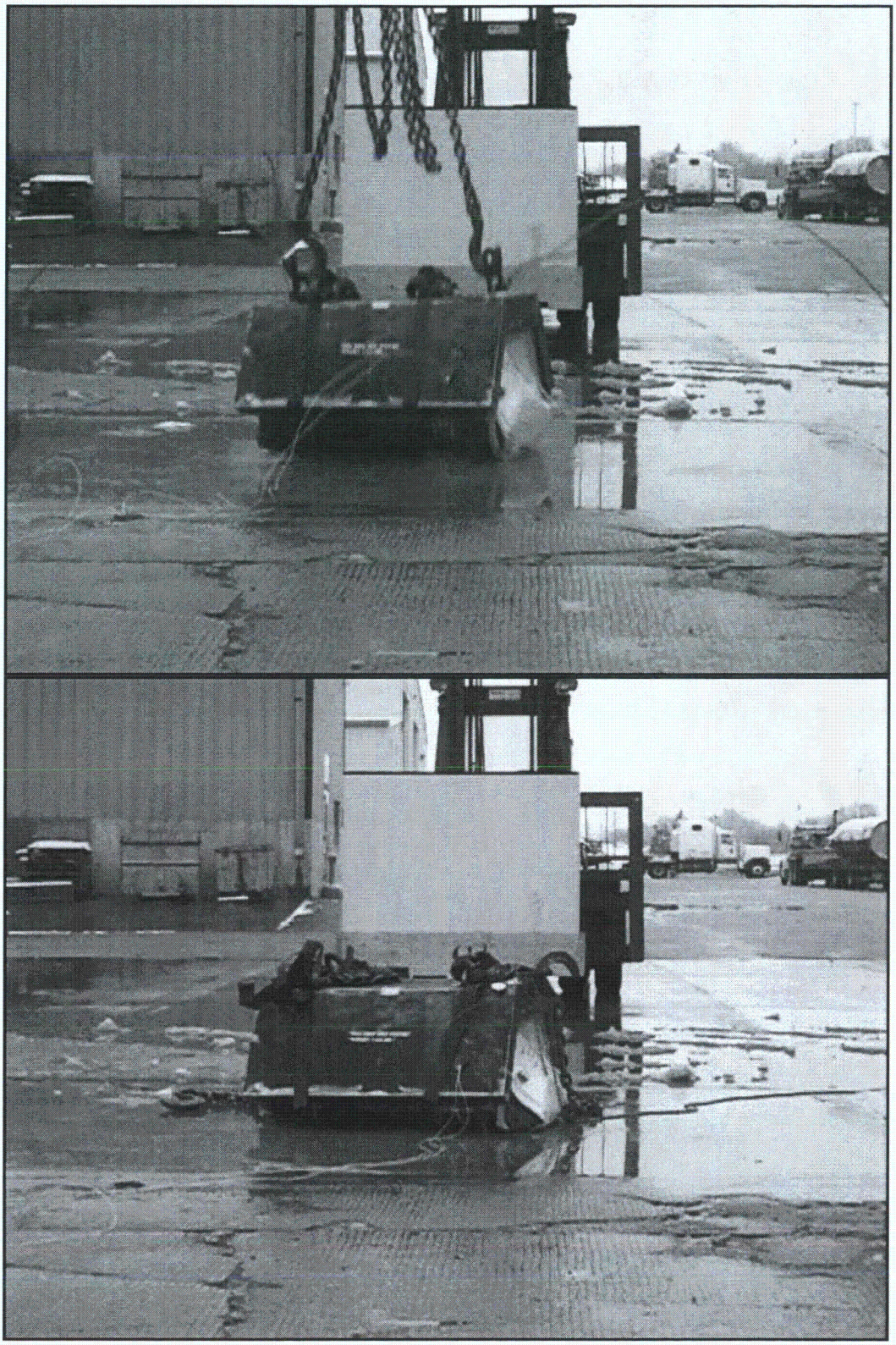

Figure 4.6-7 – Side Drop Test Assembly During and After Impact

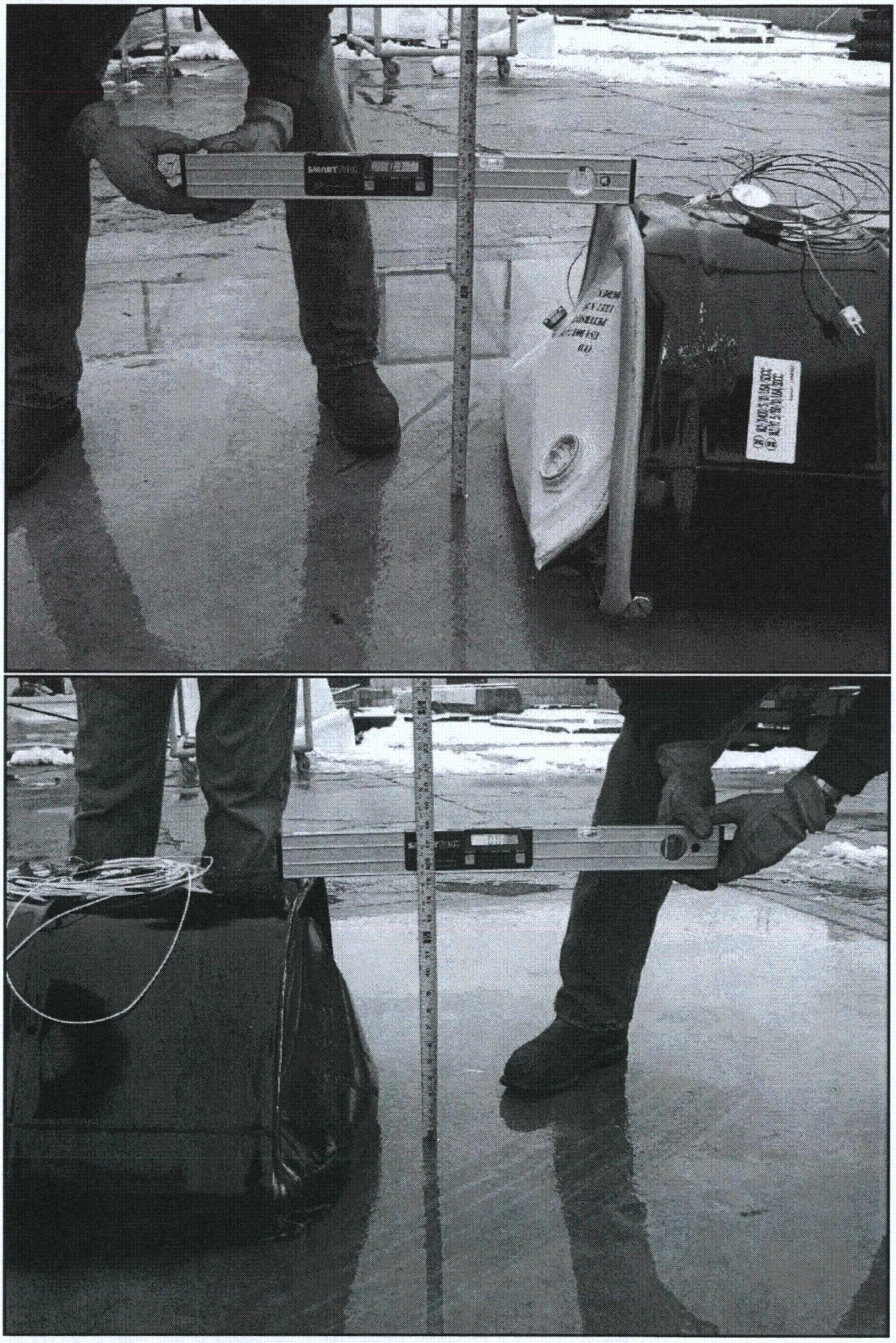

Figure 4.6-8 – Post-Side Drop Inspection of CCO Radial Deformation

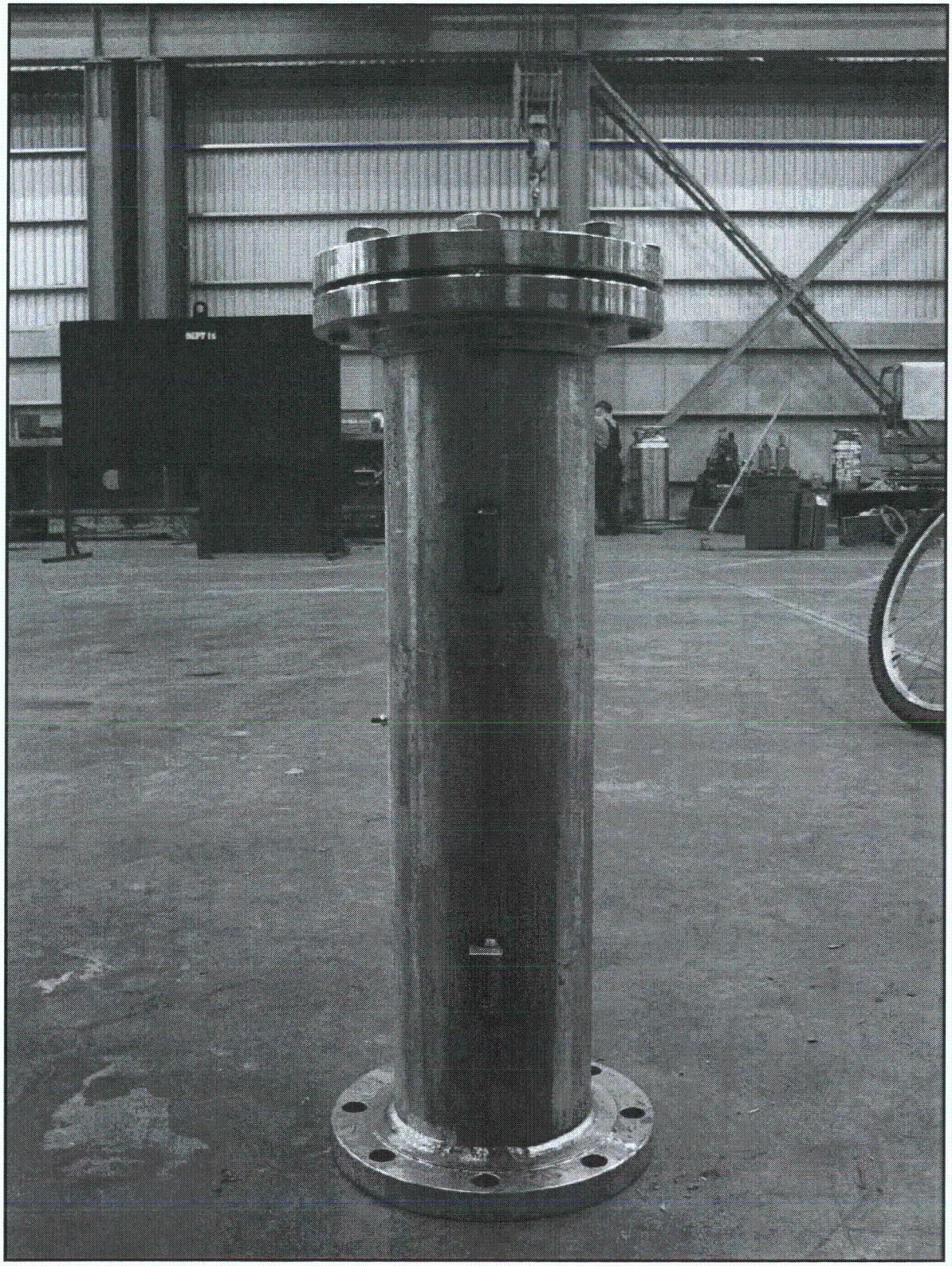

Figure 4.6-9 – Post-Side Drop Evaluation of CCC

4.6.3.2.3 End Drop Results

The 30-foot hot end drop was performed using a CCO end drop test fixture weighing 357 pounds attached to the CCO end drop test article that was dropped in an inverted vertical orientation to axially impact directly onto the essentially unyielding drop pad surface. The test conservatively simulated the interaction between a CCO in a lower seven-pack array and a CCO in an upper seven-pack array inside a TRUPACT-II. The test article was oriented to maximize the loads on the closure interface. The test configuration neglected both the impact-attenuating characteristics of the packaging (e.g., honeycomb payload spacers) and the compliance of the other axially adjacent CCO in the stacked 14-pack arrangement to maximize the impact accelerations on the CCC and crush deformation of the dunnage assemblies.

Post-drop inspection of the CCO indicated measurable crushing of the 55-gallon drum and internal plywood dunnage assemblies that resulted in a minimum effective height of the CCO that measured between 31⅛ inches and 32 inches (pre-drop height of the 55-gallon drum was 35 inches). The upper and lower plywood dunnage assemblies experienced a corresponding axial crush deformation, but the assemblies attenuated the impact and prevented any direct interaction between the CCC and the test fixture/pad. The measured accelerations due to impact were recorded with filtering of the data utilizing a low-pass Butterworth 10-pole filter having a 250 Hz cut-off frequency. A minimum impact acceleration peak, from one reporting sensor, of approximately 411 g's was recorded.

Disassembly of the CCO and inspection of the CCC indicated no permanent plastic deformation of any of the confinement boundary components, essentially unaltered preload of the closure bolts (maximum bolt rotation to achieve installation torque values of 5°), and no loss of confinement as confirmed via black-light inspection of the assembly with no presence of the fluorescein indicator.

Figure 4.6-10 shows the CCO end drop test article and test fixture assembly prior to the drop test. Figure 4.6-11 shows the assembly during and after the end impact event. Figure 4.6-12 shows the post-test measurement of the effective axial height of the CCO and associated axial crushing of the dunnage assemblies. Figure 4.6-13 shows the CCC after the post-drop test confinement evaluation and bolt residual torque confirmation process, indicating no degradation of the confinement vessel.

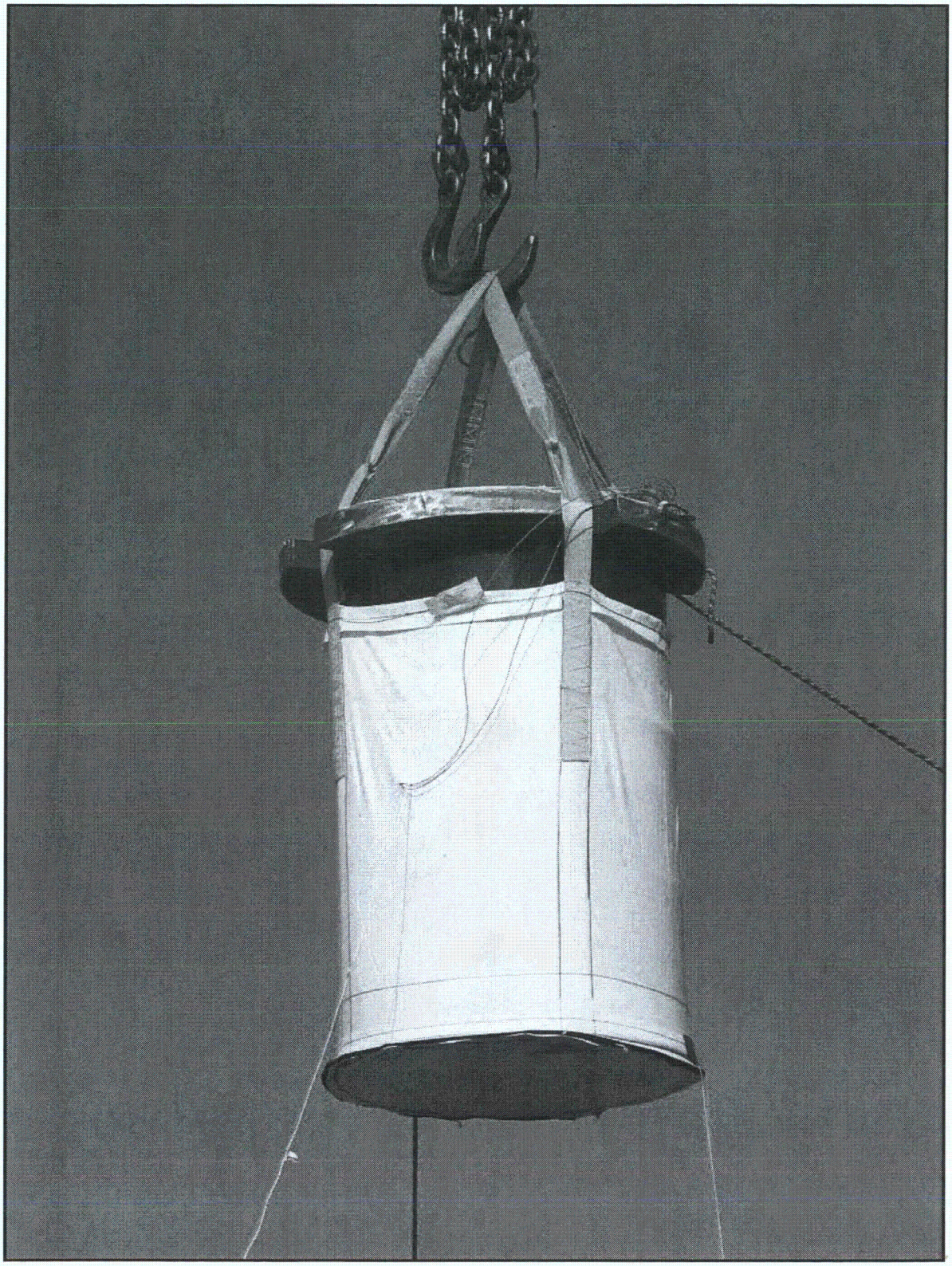

Figure 4.6-10 – End Drop Test Assembly

Figure 4.6-11 – End Drop Test Assembly During and After Impact

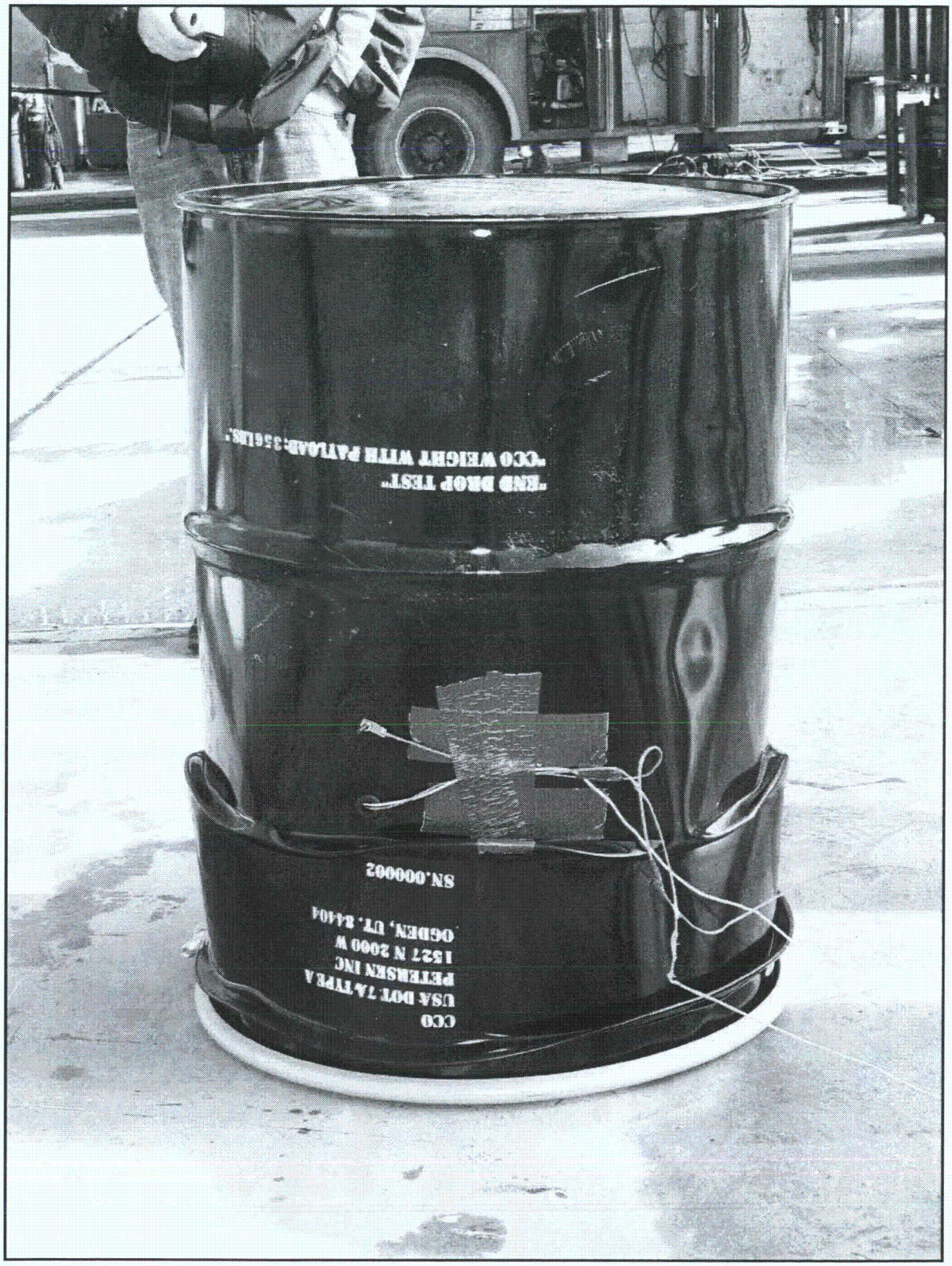

Figure 4.6-12 – Post-End Drop Inspection of CCO Axial Deformation

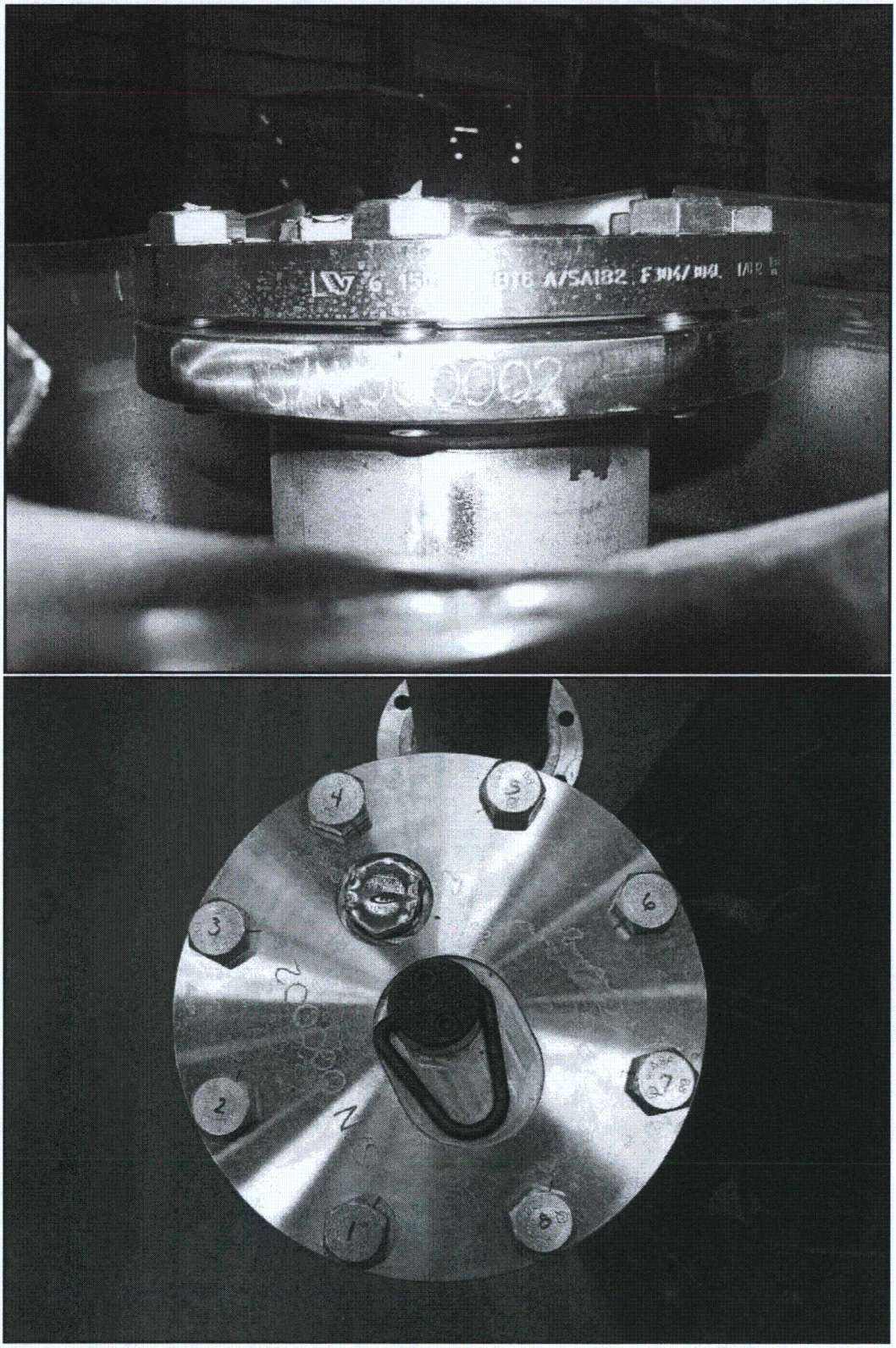

Figure 4.6-13 – Post-End Drop Evaluation of CCC

4.6.3.2.4 Summary of Testing

Key test observations include the following:

1. Post-test visual inspection of the exterior dimensions of the CCO indicated crushing of the upper and lower plywood dunnage assemblies such that the CCC experienced no direct impact with the external impacting surfaces of the test fixture and/or the test pad. The CCC experienced no measurable deformation.
2. The test conditions maximized the crushing of the upper and lower plywood dunnage assemblies by dropping at an elevated temperature using rigid test fixtures to simulate interaction with other CCOs in the payload assembly and by dropping directly onto an essentially unyielding surface rather than onto the impact-attenuating shells and polyurethane foam of the TRUPACT-II and/or HalfPACT ICV/OCA structures.
3. The CCC closure bolts retained essentially full residual torque, where the polar deviation to return the bolts to the initial installation torque was such that 99% of the gasket compression was retained after the drop event. In addition, the flour/fluorescein mixture placed within each CCC was 100% retained throughout the testing. Collectively, these observations readily confirmed confinement integrity of the CCO.

In summary, the results of the testing program for the CCO demonstrate that under HAC the CCO maintains confinement integrity, shielding integrity, and provides defined array spacing for criticality control purposes.

4.6.4 Thermal Evaluation

4.6.4.1 Thermal Evaluation for Normal Conditions of Transport

Thermal analysis models of the TRUPACT-II and HalfPACT packagings with a CCO payload were developed using the computer programs Thermal Desktop[5] and SINDA/FLUINT[6]. The thermal models are a three-dimensional half-symmetry (180°) finite element and finite difference solid element and surface/planar element and thermal entity representations of the packages. Complete details of the package thermal models, including model dimensions, material properties, boundary conditions, and decay configurations are provided in the calculation package.[7]

Identical modeling approaches were used for the thermal models for the TRUPACT-II and HalfPACT packages with appropriate modifications for the payload cavity height, number of installed payload containers, and decay heat limits and distribution. The CCO geometry is as given in Drawing 163-009 in Appendix 1.3.1 of the TRUPACT-II SAR[1] and HalfPACT SAR[2]. The TRUPACT-II was analyzed when transporting fourteen (14) CCOs, while the HalfPACT was analyzed for transporting seven (7) CCOs. The heat transfer between the various components within the CCOs, between the CCOs and the ICV, and between the ICV/OCA is via radiation and conduction. Convection within the payload cavity is conservatively ignored. Heat transfer between the exterior of the package and the environment is via radiation and natural convection.

The analyses assume that all packaging components are radially centered and all CCO components are radially and axially centered to maximize payload temperatures. Radially shifting the payload to initiate contact between the ICV and adjacent CCO would result in a negligible increase in packaging component and payload temperatures. More significant is the placement of decay heat loads within the packaging payload cavity. Similarly, a fractional shift of CCO components, radially and/or axially, would result in a negligible increase in payload temperatures.

All void spaces within the CCO and packaging cavity are assumed to be filled with air at atmospheric pressure. A paper-based waste stream with the effective conductivity of air is assumed for the payload with a maximum total decay heat loading of 40 watts for the TRUPACT-II package and 30 watts for the HalfPACT package. The distribution of decay heat is varied within the assembly of CCOs to establish the limiting case under the restriction that the maximum decay heat in any single CCO is limited to 20 watts. The decay heat within a CCO is assumed to be equally distributed within the waste volume on a volumetric basis.

The package is mounted in an upright position on its transport trailer or railcar for NCT. This establishes the orientation of the exterior surfaces of the package for determining the free convection heat transfer coefficients and insolation loading. In addition, the bottom of the

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[5] Thermal Desktop®, *A CAD Based System for Thermal Analysis and Design*,Version 5.4, Cullimore & Ring Technologies, Inc., Littleton, CO, 2011.

[6] SINDA/FLUINT, *General Purpose Thermal/Fluid Network Analyzer*, Version 5.4, Cullimore & Ring Technologies, Inc., Littleton, CO, 2011.

[7] S. A. Porter, *Criticality Control Overpack Thermal Analysis*, CCO-CAL-0003, Rev. 1, Washington TRU Solutions LLC, Carlsbad, NM, April 2012.

dedicated transport trailer is open, and the bottom of the package is exposed to ambient air instead of resting on the ground or some other semi-adiabatic, conducting surface. Thus, heat transfer from the OCA exterior to the ambient is conducted via free convection and radiation exchange. The insolation values used for heat input to the exterior package surfaces for NCT, as specified in 10 CFR §71.71(c)(1) are as follows: 1) flat base of the package has no applied insolation, 2) all vertical OCA surfaces have an insolation value of 200 gcal/cm^2 (737.5 Btu/ft^2), and 3) the curved OCA torispherical head has an insolation value of 400 gcal/cm^2 (1,475 Btu/ft^2). Consistent with 10 CFR §71.71(b), a 100 °F ambient temperature is used. Insolation is applied in 12-hour "off/on" steps, i.e., a repeating 12-hour "off", 12-hour "on" cycle for a sufficient period of time (1,200 hours) to allow the hottest location within the package to reach pseudo-steady-state equilibrium.

As summarized below, a set of four primary TRUPACT-II package and four primary HalfPACT package cases were evaluated, both without and with insolation. Additionally included with each set is a zero watt decay heat load as a baseline case for determining the temperature/decay heat relationship for internal pressure calculations.

For the TRUPACT-II package:

- Case 0: 0 watts decay heat in 14 CCOs (0 watts each); baseline case for internal pressure calculations.
- Case 1: 40 watts decay heat evenly distributed in 14 CCOs (2.857 watts each); basic uniform decay heat load case for comparison with existing TRUPACT-II package and payload temperature results.
- Case 2: 40 watts decay heat evenly distributed in the 2 center CCOs (20 watts each); maximum heat load per CCO case with the highest thermal isolation (i.e., centered) that should produce the highest CCO contents temperature.
- Case 3: 40 watts decay heat evenly distributed in 2 axially aligned outer CCOs (20 watts each); highest ICV and OCV seal temperature case, along with Case 4.
- Case 4: 40 watts decay heat evenly distributed in 2 laterally aligned outer CCOs (20 watts each); highest ICV and OCV seal temperature case, along with Case 3.

For the HalfPACT package:

- Case 0: 0 watts decay heat in 7 CCOs; baseline case for internal pressure calculations.
- Case 1: 30 watts decay heat evenly distributed in 7 CCOs (4.286 watts each); basic uniform decay heat load case for comparison with existing HalfPACT package and payload temperature results.
- Case 2: 20 watts decay heat in the center CCO and 10 watts decay heat in one outer CCO; maximum heat load per CCO case with the highest thermal isolation (i.e., centered) that should produce the highest CCO contents temperature, along with Case 3.
- Case 3: 20 watts decay heat in the center CCO and 10 watts decay heat evenly distributed in the 6 outer CCOs; maximum heat load per CCO case with the highest thermal isolation (i.e., centered) that should produce the highest CCO contents temperature, along with Case 2.

- Case 4: 30 watts decay heat evenly distributed in 2 laterally aligned outer CCOs (15 watts each); highest ICV and OCV seal temperature case.

The results of the NCT thermal analyses are provided in Table 4.6-1, Table 4.6-2, Table 4.6-3, and Table 4.6-4. As can be seen in Table 4.6-2 for the TRUPACT-II package with insolation and Table 4.6-4 for the HalfPACT package with insolation, all maximum packaging temperatures and all maximum CCO temperatures are below the respective component maximum allowable temperatures. Thus, the thermal analysis results for the thermal performance of a CCO payload transported within a TRUPACT-II or HalfPACT packaging under NCT conditions demonstrate compliance with the requirements of 10 CFR §71.43(g) and §71.71.

Finally, 10 CFR §71.43(g) stipulates that maximum accessible surface temperatures must be less than 185 °F for a package based on 100 °F ambient conditions without insolation for exclusive use package shipments. As can be seen in Table 4.6-1 for the TRUPACT-II package without insolation and Table 4.6-3 for the HalfPACT package without insolation, all maximum packaging temperatures are below the maximum allowable temperature of 185 °F.

4.6.4.2 Thermal Evaluation for Hypothetical Accident Conditions

No safety evaluations for HAC are required for the CCO payload since HAC results are bounded by those presented in Chapter 3.0, *Thermal Evaluation*, of the TRUPACT-II and HalfPACT SARs[1,2]. The basis for this conclusion is as follows:

1. The level of heat input into the package during the HAC event is a function of the package's exterior surface area, the thermal mass of the package components, etc. Since maximum decay heat loads are unchanged, the level of heat input is essentially unaffected by the makeup of the payload.
2. The temperature response within the payload is a function of its thermal mass and the amount of heat passed to it by the packaging, which remains unchanged.
3. Since the heat input into the packaging during the HAC event is essentially the same between a package containing a base payload evaluated in the TRUPACT-II and HalfPACT SARs and a package containing a payload of CCOs, the HAC thermal response of the packages will be bounded by that currently presented in the TRUPACT-II and HalfPACT SARs, respectively.
4. Given that the CCC is radially and axially inset from the surface of the CCO's 55-gallon drum, the addition of these radial and axial air gaps will further isolate the payload from the drum temperatures encountered during the HAC transient event.

Temperatures for the CCOs within the TRUPACT-II and HalfPACT packagings during the HAC fire event may be determined by conservatively adding the experimentally measured temperature differential determined from HAC fire testing of each package to the worst-case initial temperatures from Table 4.6-1 of the TRUPACT-II package and Table 4.6-3 for the HalfPACT package. Table 4.6-5 and Table 4.6-6 for the TRUPACT-II and HalfPACT packages, respectively, summarize the predicted HAC temperatures for the major components in each package, none of which exceed defined temperature limits for the materials of construction. Thus, the CCO payload will not impact the safety basis of either the TRUPACT II or HalfPACT packages for HAC, and the requirements of 10 CFR §71.73(c)(3) are satisfied.

Table 4.6-1 – TRUPACT-II Package NCT Steady-State Temperatures (°F) without Insolation

Location	Component(s)	Case 0	Case 1	Case 2	Case 3	Case 4	Limit
Center CCOs	Contents Centerline Maximum①	100.0	147.4	284.6	124.5	124.0	N/A
	Contents Centerline Average③	100.0	144.7	266.0	124.2	123.6	④
	Contents Bulk Average③	100.0	136.6	208.8	124.2	123.6	④
	CCC Structure Maximum①	100.0	129.9	161.9	124.5	124.0	2,600
	CCC Gasket Maximum①	100.0	128.0	149.9	124.2	123.9	548
	Plywood Dunnage Maximum①	100.0	127.3	144.2	126.5	126.0	212
	55-Gallon Drum Maximum①	100.0	125.4	131.6	126.7	126.4	2,750
	55-Gallon Drum Average②	100.0	124.8	128.5	124.2	123.5	2,750
Outer CCOs (Heated)	Contents Centerline Maximum①	—	146.6	—	285.5	286.4	N/A
	Contents Centerline Average③	—	143.9	—	266.9	267.7	④
	Contents Bulk Average③	—	135.7	—	209.7	210.5	④
	CCC Structure Maximum①	—	129.1	—	162.8	163.8	2,600
	CCC Gasket Maximum①	—	127.1	—	150.9	151.8	548
	Plywood Dunnage Maximum①	—	126.4	—	145.3	145.3	212
	55-Gallon Drum Maximum①	—	124.9	—	132.9	132.9	2,750
	55-Gallon Drum Average②	—	123.9	—	129.6	130.3	2,750
Outer CCOs (Unheated)	Contents Centerline Maximum①	100.0	—	124.4	123.0	122.2	N/A
	Contents Centerline Average③	100.0	—	124.2	122.8	121.8	④
	Contents Bulk Average③	100.0	—	124.2	122.8	121.8	④
	CCC Structure Maximum①	100.0	—	124.4	123.0	122.2	2,600
	CCC Gasket Maximum①	100.0	—	124.1	122.7	122.2	548
	Plywood Dunnage Maximum①	100.0	—	126.1	124.3	123.9	212
	55-Gallon Drum Maximum①	100.0	—	126.3	124.5	124.1	2,750
	55-Gallon Drum Average②	100.0	—	124.2	122.7	123.4	2,750
All CCOs	Contents Centerline Average③	100.0	144.0	144.5	143.6	142.9	④
	Contents Bulk Average③	100.0	135.8	136.3	135.4	134.7	④
	55-Gallon Drum Average②	100.0	124.1	124.8	123.9	124.4	2,750
ICV Cavity Air	Bulk Average①②	100.0	122.9	123.4	122.8	122.9	N/A
ICV Structure	Maximum	100.0	122.4	122.3	126.9	125.0	800
	Bulk Average①	100.0	119.5	119.6	119.5	119.2	800
	Minimum	100.0	115.6	115.7	115.3	116.4	800
ICV O-ring Seal	Maximum	100.0	118.2	118.2	121.4	123.2	-40 to 225
OCV Structure	Maximum	100.0	121.2	121.1	125.1	122.7	800
	Bulk Average①	100.0	117.4	117.4	117.4	117.2	800
OCV O-ring Seal	Maximum	100.0	114.8	114.8	117.1	118.4	-40 to 225
Polyurethane Foam	Maximum	100.0	121.2	121.1	125.1	122.7	300
	Bulk Average①	100.0	108.9	108.9	108.9	108.7	300
OCA Outer Shell	Maximum	100.0	102.7	102.7	103.1	103.2	185

Notes:

① Temperature based on an arithmetical average.

② Temperature based on an area-weighted average.

③ Temperature based on a volume-weighted average.

④ Contents temperature limit based on Appendix 6.6 of the *CH-TRU Payload Appendices*.

Table 4.6-2 – TRUPACT-II Package NCT Steady-State Temperatures (°F) with Insolation

Location	Component(s)	Case 0	Case 1	Case 2	Case 3	Case 4	Limit
Center CCOs	Contents Centerline Maximum①	117.8	163.9	299.2	141.2	140.8	N/A
	Contents Centerline Average③	117.7	161.2	280.8	141.0	140.3	④
	Contents Bulk Average③	117.7	153.0	223.6	141.0	140.3	④
	CCC Structure Maximum①	117.8	146.3	176.4	141.2	140.8	2,600
	CCC Gasket Maximum①	117.8	144.5	165.1	141.0	140.7	548
	Plywood Dunnage Maximum①	117.8	143.9	159.8	143.1	142.7	212
	55-Gallon Drum Maximum①	117.8	142.1	147.9	143.3	143.1	2,750
	55-Gallon Drum Average②	117.7	141.5	145.0	140.9	140.2	2,750
Outer CCOs (Heated)	Contents Centerline Maximum①	—	163.1	—	300.1	301.1	N/A
	Contents Centerline Average③	—	160.4	—	281.7	282.6	④
	Contents Bulk Average③	—	152.2	—	224.5	225.4	④
	CCC Structure Maximum①	—	145.6	—	177.4	178.4	2,600
	CCC Gasket Maximum①	—	143.7	—	166.0	167.1	548
	Plywood Dunnage Maximum①	—	143.1	—	160.8	161.0	212
	55-Gallon Drum Maximum①	—	141.6	—	149.1	149.4	2,750
	55-Gallon Drum Average②	—	140.7	—	146.0	146.8	2,750
Outer CCOs (Unheated)	Contents Centerline Maximum①	117.8	—	141.2	139.8	139.1	N/A
	Contents Centerline Average③	117.7	—	141.0	139.6	138.6	④
	Contents Bulk Average③	117.7	—	141.0	139.6	138.6	④
	CCC Structure Maximum①	117.8	—	141.2	139.8	139.1	2,600
	CCC Gasket Maximum①	117.8	—	141.0	139.6	139.0	548
	Plywood Dunnage Maximum①	117.9	—	142.8	141.1	140.7	212
	55-Gallon Drum Maximum①	117.9	—	142.9	141.2	140.9	2,750
	55-Gallon Drum Average②	117.7	—	140.9	139.6	140.2	2,750
All CCOs	Contents Centerline Average③	117.7	160.5	161.0	160.1	159.4	④
	Contents Bulk Average③	117.7	152.3	152.8	151.9	151.3	④
	55-Gallon Drum Average②	117.7	140.8	141.5	140.7	141.1	2,750
ICV Cavity Air	Bulk Average①②	117.7	139.7	140.2	139.6	139.7	N/A
ICV Structure	Maximum	119.4	139.0	138.9	143.4	142.0	800
	Bulk Average①	117.9	136.7	136.8	136.7	136.3	800
	Minimum	116.5	134.4	134.5	133.6	134.2	800
ICV O-ring Seal	Maximum	118.4	135.9	135.8	138.9	140.7	-40 to 225
OCV Structure	Maximum	120.1	137.9	137.8	141.7	139.7	800
	Bulk Average①	118.4	135.1	135.2	135.1	134.9	800
OCV O-ring Seal	Maximum	120.1	133.9	133.9	136.1	137.3	-40 to 225
Polyurethane Foam	Maximum	150.6	150.8	150.8	150.8	150.8	300
	Bulk Average①	122.5	130.9	130.9	130.9	130.7	300
OCA Outer Shell	Maximum	150.6	150.8	150.8	150.8	150.8	800

Notes:

① Temperature based on an arithmetical average.

② Temperature based on an area-weighted average.

③ Temperature based on a volume-weighted average.

④ Contents temperature limit based on Appendix 6.6 of the *CH-TRU Payload Appendices*.

Table 4.6-3 – HalfPACT Package NCT Steady-State Temperatures (°F) without Insolation

Location	Component(s)	Case 0	Case 1	Case 2	Case 3	Case 4	Limit
Center CCO	Contents Centerline Maximum①	100.0	157.4	282.8	282.8	122.8	N/A
	Contents Centerline Average③	100.0	153.3	264.1	264.1	122.6	④
	Contents Bulk Average③	100.0	141.0	206.9	206.9	122.6	④
	CCC Structure Maximum①	100.0	131.1	160.1	160.0	122.8	2,600
	CCC Gasket Maximum①	100.0	128.2	147.9	147.9	122.5	548
	Plywood Dunnage Maximum①	100.0	126.5	141.2	141.2	124.2	212
	55-Gallon Drum Maximum①	100.0	123.8	128.1	127.2	124.8	2,750
	55-Gallon Drum Average②	100.0	123.3	126.3	126.3	122.5	2,750
Outer CCO(s) (Heated)	Contents Centerline Maximum①	—	156.4	203.7	135.7	243.9	N/A
	Contents Centerline Average③	—	152.3	194.2	134.1	229.8	④
	Contents Bulk Average③	—	140.0	165.6	129.4	186.9	④
	CCC Structure Maximum①	—	130.1	142.4	125.6	151.9	2,600
	CCC Gasket Maximum①	—	127.2	135.9	124.3	142.4	548
	Plywood Dunnage Maximum①	—	125.7	132.5	124.0	137.4	212
	55-Gallon Drum Maximum①	—	123.4	126.8	124.5	128.0	2,750
	55-Gallon Drum Average②	—	122.3	124.8	122.4	126.0	2,750
Outer CCOs (Unheated)	Contents Centerline Maximum①	100.0	—	122.2	—	120.5	N/A
	Contents Centerline Average③	100.0	—	122.0	—	120.4	④
	Contents Bulk Average③	100.0	—	122.0	—	120.4	④
	CCC Structure Maximum①	100.0	—	122.2	—	120.5	2,600
	CCC Gasket Maximum①	100.0	—	121.9	—	120.4	548
	Plywood Dunnage Maximum①	100.0	—	123.5	—	121.6	212
	55-Gallon Drum Maximum①	100.0	—	124.1	—	121.9	2,750
	55-Gallon Drum Average②	100.0	—	121.9	—	120.4	2,750
All CCOs	Contents Centerline Average③	100.0	152.4	152.6	152.7	151.9	④
	Contents Bulk Average③	100.0	140.1	140.4	140.4	139.7	④
	55-Gallon Drum Average②	100.0	122.4	122.9	123.0	122.3	2,750
ICV Cavity Air	Bulk Average①②	100.0	121.1	121.4	121.5	121.0	N/A
ICV Structure	Maximum	100.0	119.6	121.0	119.9	122.3	800
	Bulk Average①	100.0	117.6	117.7	117.7	117.6	800
	Minimum	100.0	114.6	114.6	114.6	114.3	800
ICV O-ring Seal	Maximum	100.0	117.0	118.0	117.0	119.1	-40 to 225
OCV Structure	Maximum	100.0	118.6	119.4	118.8	120.7	800
	Bulk Average①	100.0	115.6	115.6	115.6	115.6	800
OCV O-ring Seal	Maximum	100.0	113.9	114.6	113.9	115.6	-40 to 225
Polyurethane Foam	Maximum	100.0	118.6	119.4	118.8	120.7	300
	Bulk Average①	100.0	107.9	107.9	107.9	107.9	300
OCA Outer Shell	Maximum	100.0	102.4	102.5	102.4	102.6	185

Notes:

① Temperature based on an arithmetical average.

② Temperature based on an area-weighted average.

③ Temperature based on a volume-weighted average.

④ Contents temperature limit based on Appendix 6.6 of the *CH-TRU Payload Appendices*.

Table 4.6-4 – HalfPACT Package NCT Steady-State Temperatures (°F) with Insolation

Location	Component(s)	Case 0	Case 1	Case 2	Case 3	Case 4	Limit
Center CCO	Contents Centerline Maximum①	117.6	173.5	297.3	297.3	139.4	N/A
	Contents Centerline Average③	117.6	169.4	278.7	278.7	139.2	④
	Contents Bulk Average③	117.6	157.1	221.5	221.5	139.2	④
	CCC Structure Maximum①	117.6	147.1	174.5	174.5	139.4	2,600
	CCC Gasket Maximum①	117.6	144.4	162.9	162.9	139.2	548
	Plywood Dunnage Maximum①	117.7	142.8	156.6	156.6	140.7	212
	55-Gallon Drum Maximum①	117.7	140.2	144.3	143.5	141.3	2,750
	55-Gallon Drum Average②	117.5	139.8	142.6	142.6	139.0	2,750
Outer CCO(s) (Heated)	Contents Centerline Maximum①	—	172.5	219.1	152.2	258.8	N/A
	Contents Centerline Average③	—	168.5	209.8	150.6	244.9	④
	Contents Bulk Average③	—	156.2	181.1	145.8	201.9	④
	CCC Structure Maximum①	—	146.3	157.8	142.1	166.8	2,600
	CCC Gasket Maximum①	—	143.5	151.6	140.8	157.7	548
	Plywood Dunnage Maximum①	—	142.0	148.3	140.4	153.0	212
	55-Gallon Drum Maximum①	—	139.9	143.2	141.0	144.3	2,750
	55-Gallon Drum Average②	—	138.9	141.2	139.0	142.4	2,750
Outer CCOs (Unheated)	Contents Centerline Maximum①	117.7	—	138.8	—	137.2	N/A
	Contents Centerline Average③	117.6	—	138.7	—	137.2	④
	Contents Bulk Average③	117.6	—	138.6	—	137.1	④
	CCC Structure Maximum①	117.6	—	138.8	—	137.2	2,600
	CCC Gasket Maximum①	117.6	—	138.6	—	137.1	548
	Plywood Dunnage Maximum①	117.7	—	140.0	—	138.2	212
	55-Gallon Drum Maximum①	117.7	—	140.5	—	138.5	2,750
	55-Gallon Drum Average②	117.5	—	138.5	—	137.1	2,750
All CCOs	Contents Centerline Average③	117.6	168.6	168.8	168.9	168.2	④
	Contents Bulk Average③	117.6	156.3	156.6	156.6	155.9	④
	55-Gallon Drum Average②	117.5	139.0	139.5	139.5	138.9	2,750
ICV Cavity Air	Bulk Average①②	117.5	137.8	138.1	138.1	137.7	N/A
ICV Structure	Maximum	119.2	136.1	137.6	136.0	139.0	800
	Bulk Average①	117.8	134.7	134.8	134.8	134.8	800
	Minimum	116.6	133.2	133.0	133.2	132.4	800
ICV O-ring Seal	Maximum	118.1	134.4	135.4	134.4	136.4	-40 to 225
OCV Structure	Maximum	119.8	134.9	136.0	134.8	137.2	800
	Bulk Average①	118.3	133.3	133.3	133.3	133.3	800
OCV O-ring Seal	Maximum	119.8	132.7	133.4	132.6	134.3	-40 to 225
Polyurethane Foam	Maximum	150.6	150.7	150.7	150.7	150.8	300
	Bulk Average①	122.4	129.8	129.8	129.8	129.8	300
OCA Outer Shell	Maximum	150.6	150.7	150.7	150.7	150.8	800

Notes:

① Temperature based on an arithmetical average.

② Temperature based on an area-weighted average.

③ Temperature based on a volume-weighted average.

④ Contents temperature limit based on Appendix 6.6 of the *CH-TRU Payload Appendices*.

Table 4.6-5 – Predicted TRUPACT-II Package HAC Temperatures (°F) with a CCO Payload

Location	Component(s)	TRUPACT-II SAR①			CCO Analysis②		Limit
		Pre-Fire CTU-2	Post-Fire CTU-2	ΔT	Pre-Fire CCO	Post-Fire CCO	
Center Drums (CCOs)③	CCC Structure Maximum	—	—	—	161.9	194.9	2,600
	CCC Gasket Maximum	—	—	—	149.9	182.9	548
	Plywood Dunnage Maximum	—	—	—	144.2	177.2	212
	55-Gallon Drum Maximum	127	160	33	131.6	164.6	2,750
	55-Gallon Drum Average	127	150	23	128.5	151.5	2,750
Outer Drums (CCOs)④	CCC Structure Maximum	—	—	—	163.8	206.8	2,600
	CCC Gasket Maximum	—	—	—	151.8	194.8	548
	Plywood Dunnage Maximum	—	—	—	145.3	188.3	212
	55-Gallon Drum Maximum	127	170	43	132.9	175.9	2,750
	55-Gallon Drum Average	127	155	28	130.3	158.3	2,750
All Drums (CCOs)⑤	CCC Structure Maximum	—	—	—	163.5	205.5	2,600
	CCC Gasket Maximum	—	—	—	151.5	193.5	548
	Plywood Dunnage Maximum	—	—	—	145.1	187.1	212
	55-Gallon Drum Maximum	127	169	42	132.7	174.7	2,750
	55-Gallon Drum Average	127	153	26	130.0	156.0	2,750
ICV Cavity Air	Bulk Average	127	179	52	123.4	175.4	N/A
ICV Structure	Maximum	127	220	93	126.9	219.9	2,600
	Bulk Average	127	187	60	119.6	179.6	2,600
ICV O-ring Seal	Maximum	127	200	73	123.2	196.2	360
OCV Structure	Maximum	127	439	312	125.1	437.1	2,600
	Bulk Average	127	284	157	117.4	274.4	2,600
OCV O-ring Seal	Maximum	119	253	134	118.4	252.4	360

Notes:

① TRUPACT-II package pre-fire temperatures are taken from Section 3.5.2.2, *CTU-2 Package Conditions and Environment*, of the TRUPACT-II SAR and post-fire temperatures are taken from Table 3.5-5 of the TRUPACT-II SAR.

② CCO pre-fire temperatures are the maximum values reported in Table 4.6-1 for each TRUPACT-II package component. CCO post-fire temperatures are determined by adding ΔT to the pre-fire temperatures, with maximum post-fire temperatures for the CCC structure, CCC gasket, and plywood dunnage determined by adding ΔT for the 55-gallon drum maximum to the pre-fire temperatures of the corresponding component.

③ CCO temperatures for the inner CCOs are based on Case 2 values.

④ CCO temperatures for the outer CCOs are based on Case 4 values.

⑤ CCO temperatures for all CCOs are conservatively determined based on the volumetric average of Case 2 and Case 4 values.

Table 4.6-6 – Predicted HalfPACT Package HAC Temperatures (°F) with a CCO Payload

Location	Component(s)	HalfPACT SAR①			CCO Analysis②		Limit
		Pre-Fire CTU-2	Post-Fire CTU-2	ΔT	Pre-Fire CCO	Post-Fire CCO	
Center Drum (CCO)③	CCC Structure Maximum	—	—	—	160.1	209.7	2,600
	CCC Gasket Maximum	—	—	—	147.9	197.5	548
	Plywood Dunnage Maximum	—	—	—	141.2	190.8	212
	55-Gallon Drum Maximum	150.4	200	49.6	128.1	177.7	2,750
	55-Gallon Drum Average	148.8	200	51.2	126.3	177.5	2,750
Outer Drums (CCOs)④	CCC Structure Maximum	—	—	—	151.9	204.0	2,600
	CCC Gasket Maximum	—	—	—	142.4	194.5	548
	Plywood Dunnage Maximum	—	—	—	137.4	189.5	212
	55-Gallon Drum Maximum	147.9	200	52.1	128.0	180.1	2,750
	55-Gallon Drum Average	138.2	200	61.8	126.0	187.8	2,750
All Drums (CCOs)⑤	CCC Structure Maximum	—	—	—	153.1	204.8	2,600
	CCC Gasket Maximum	—	—	—	143.2	194.9	548
	Plywood Dunnage Maximum	—	—	—	137.9	189.6	212
	55-Gallon Drum Maximum	148.3	200	51.7	128.0	179.7	2,750
	55-Gallon Drum Average	139.7	200	60.3	126.0	186.3	2,750
ICV Cavity Air	Bulk Average	136.5	200	63.5	121.5	185.0	N/A
ICV Structure	Maximum	140.4	200	59.6	122.3	181.9	2,600
	Bulk Average	133.2	200	66.8	117.7	184.5	2,600
ICV O-ring Seal	Maximum	130.8	200	69.2	119.1	188.3	360
OCV Structure	Maximum	134.5	290	155.5	120.7	276.2	2,600
	Bulk Average	131.1	290	158.9	115.6	274.5	2,600
OCV O-ring Seal	Maximum	129.0	290	161.0	115.6	276.6	360

Notes:

① HalfPACT package pre-fire temperatures are taken from Table 3.5-1 of the HalfPACT SAR, and post-fire temperatures are taken from Section 3.5.3, *Package Temperatures*, of the HalfPACT SAR. As discussed in Section 3.5.3, pre-fire temperatures are based on a 90 °F difference between actual and theoretical pre-fire package temperatures.

② CCO pre-fire temperatures are the maximum values reported in Table 4.6-3 for each HalfPACT package component. CCO post-fire temperatures are determined by adding ΔT to the pre-fire temperatures, with maximum post-fire temperatures for the CCC structure, CCC gasket, and plywood dunnage determined by adding ΔT for the 55-gallon drum maximum to the pre-fire temperatures of the corresponding component.

③ CCO temperatures for the inner CCOs are based on Case 2 values.

④ CCO temperatures for the outer CCOs are based on Case 4 values.

⑤ CCO temperatures for all CCOs are conservatively determined based on the volumetric average of Case 2 and Case 4 values.

4.6.5 Shielding Evaluation

The evaluation of compliance with the radiation dose rate limits for NCT and HAC required by 10 CFR §71.47 is presented in Chapter 5 of the TRUPACT-II and HalfPACT SARs[1,2] for the CCO payload configuration. When the TRUPACT-II and HalfPACT packages are loaded with assemblies of CCOs containing gamma and/or neutron source terms that are limited per Section 3.3 of the CH-TRAMPAC[3], the package meets the NCT radiation dose rate requirements of 200 mrem/hr at the surface of the package and 10 mrem/hr at 2 meters from the surface of the package under exclusive use. As a result, the packages also comply with the HAC dose rate requirement of 1 rem/hr at 1 meter from the surface of the package.

4.6.6 Criticality Evaluation

CCOs are designed to transport transuranic (TRU) waste forms with high fissile material concentrations within the TRUPACT-II and HalfPACT packages. A criticality evaluation[8] was performed for payload contents that are manually compacted (i.e., not machine compacted) and contain less than or equal to 1% by weight quantities of special reflector materials. A maximum 380 fissile gram equivalent (FGE) of Pu-239 per CCO is justified for waste forms meeting these requirements. The methodology and assumptions utilized for other approved payloads in the existing TRUPACT-II and HalfPACT SARs are also utilized in the CCO analysis. The following analysis demonstrates that this configuration complies with the requirements of 10 CFR §71.55 and §71.59. The criticality safety index, per 10 CFR §71.59, is 0.

For Case I[9](manually compacted waste with less than or equal to 1% by weight quantities of special reflectors), the moderator and surrounding reflector within each CCC was modeled as a composition of 25% polyethylene, 74% water, and 1% beryllium (by volume). As polyethylene is a superior moderator to water, this composition results in higher reactivities than would be achieved by water moderation alone. This volume fraction of polyethylene is conservatively higher than the maximum value achievable for manually compacted (i.e., not machine compacted) waste determined by experiment. Beryllium is a superior reflector to either water or polyethylene, and the inclusion of beryllium is conservative, although at such a small volume fraction, the beryllium has only a small effect on the system reactivity.

The reactivity of CCOs in a HalfPACT package is bounded by the TRUPACT-II analysis. The HalfPACT and TRUPACT-II are essentially identical packages, although the HalfPACT payload region is approximately half the height of the TRUPACT-II. Therefore, the TRUPACT-II may transport 14 CCOs in two layers of seven, while the HalfPACT may transport only a single layer of seven CCOs. The radial spacing of the two packages is identical. Although the HalfPACT ICV is 30 inches shorter, the vertical spacing between the ICVs (e.g., the annular polyurethane insulation) is the same. Thus, the single layer of seven CCOs in the HalfPACT package will have a lower reactivity than the TRUPACT-II package, which has twice the fissile material. For this reason, calculations are performed only for the TRUPACT-II geometry.

Calculations for the TRUPACT-II package are performed using the three-dimensional Monte Carlo transport theory code, KENO-V.a v4.0, with the CSAS25 utility being used as a driver for the KENO-V.a code; both programs are part of the SCALE-PC v4.4[10] code system. In this role, CSAS25 determines nuclide number densities, performs resonance processing, and automatically prepares the necessary input for the KENO-V.a code based on a simplified input description. The 238 energy-group (238GROUPNDF5) cross-section library based on ENDF/B-V cross-section data is used as the nuclear data library for the KENO-V.a code.

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[8] R. J. Migliore, *Criticality Control Overpack Criticality Analysis for TRUPACT-II and HalfPACT*, 01937.01.M009-01, Rev. 0, AREVA Federal Services LLC, Federal Way, WA, February 2012.

[9] To avoid confusion, the case designator "I" is selected to be a sequential addition to those utilized in the current TRUPACT-II and/or HalfPACT SARs. Case I is equivalent to Case A with modifications specific to the CCO payload.

[10] *SCALE4.4: Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation for Workstations and Personal Computers*, RSICC code package C00545/MNYCP00, Oak Ridge National Laboratory, September 1998.

The upper subcritical limit (USL) for ensuring that the TRUPACT-II and HalfPACT is acceptably subcritical, as determined in benchmark evaluations, is:

$$USL = 0.9377$$

Each package is considered to be acceptably subcritical if the computed k_{safe} (k_s), which is defined as $k_{effective}$ (k_{eff}) plus twice the statistical uncertainty (σ), is less than the USL, or:

$$k_s = k_{eff} + 2\sigma < USL$$

In both the NCT and HAC cases, the TRUPACT-II is modeled with reduced outer dimensions consistent with a damaged package. The torispherical heads are also modeled as flat, which brings fuel in stacked packages into close proximity. The steel 55-gallon drum of the CCO is ignored in all models, which is conservative because the steel would absorb neutrons and lower the reactivity. In the NCT models, the spacing provided by the drums is preserved. In the HAC models, reduced drum dimensions consistent with accident geometry are modeled, which results in a highly compressed array within the package. Fuel is modeled as pure Pu-239 homogeneously mixed with a moderator consisting of 74% water, 25% polyethylene, and 1% beryllium (by volume). This is a more reactive moderator than pure water and bounds the observed polyethylene volume fraction in the waste stream. Special reflectors (other than beryllium) that are in >1% by weight quantities are allowed if they are chemically or mechanically bound to the fissile material. The height of the fissile mixture is varied to optimize the moderation. Also, because the drums are stacked in two layers within the ICV, fuel is conservatively arranged so that the fuel is at the bottom of the top layer and at the top of the bottom layer. In all models, the fissile material within each CCC was assumed to form a single optimally moderated cylinder.

The most reactive case is for the HAC array. The HAC array is modeled with an infinite number of packages in the x and y directions, and two packages in the z direction. Because of the infinite number of packages and internal moderation within the CCCs, the HAC array is most reactive with no reflecting material inside the package. The most reactive case has $k_s = 0.9357$, which is below the USL of 0.9377. Addition of any reflecting or moderating material causes a decrease in reactivity. The NCT array shows the same behavior. For the single package cases, maximum reactivity is achieved with some internal reflector. For the NCT single package, the most reactive condition is with the 74/25/1 water/polyethylene/beryllium mixture inside the CCC and full-density water in all other package regions. The presence of plywood dunnage has little influence on the result. For the HAC single package, the most reactive condition is similar to the NCT single package, except with void between the CCC and drum. The system behavior for the HAC single package differs from the NCT single package as a result of the reduced drum diameter in the HAC models. However, the single package reactivity is significantly less than the array reactivity.

Case I has a justifiable limit of 380 FGE per CCO. As such, when maximally loaded with CCOs, the TRUPACT-II and HalfPACT are limited to 5,320 FGE and 2,660 FGE, respectively. The criticality analysis results are summarized in Table 4.6-7.

Table 4.6-7 – Summary of Criticality Evaluation Results for 380 FGE per CCO

Normal Conditions of Transport (NCT)	
Case "I"	k_s
Single Unit Maximum	0.8033
Infinite Array Maximum	0.9002
Hypothetical Accident Conditions (HAC)	
Case "I"	k_s
Single Unit Maximum	0.8209
Infinite Array Maximum	0.9357
USL = 0.9377	

4.6.7 Authorized Payload Contents for the Criticality Control Overpack

As demonstrated in Section 4.6.5, *Shielding Evaluation*, when loaded with gamma and/or neutron emitting isotopes with maximum activity limits summarized in the CH-TRAMPAC, the CCO payload meets the NCT and HAC dose rate limits. As demonstrated in Section 4.6.6, *Criticality Evaluation*, when loaded with fissile material with maximum mass limits, the CCO payload meets the calculated reactivity limit as summarized for Case I in Table 4.6-7 and is safely subcritical.

4.6.8 Conclusion

The CCO design consists of a vented steel 55-gallon drum containing a CCC positioned within the drum by plywood dunnage that is to be used for shipment of specific TRU waste forms in the TRUPACT-II and HalfPACT packages.

The analyses summarized in this appendix demonstrate the ability of the CCO to safely transport limited quantities of gamma and/or neutron emitting isotopes and fissile isotopes. Using geometries consistent with, or conservative with respect to, the structural and thermal analyses, the shielding evaluation showed that the dose rate limits for NCT and HAC are met with the maximum authorized contents. In addition, the criticality evaluation showed that the reactivity limit is met for manually compacted wastes with specified mass limits.

APPENDIX 5.1

REAL-TIME RADIOGRAPHY PROCEDURES

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5.1 Real-Time Radiography Procedures

5.1.1 Description of Real-Time Radiography

Real-time radiography (RTR) is a nondestructive testing method that allows the RTR operator to ascertain the physical waste form within a payload container without opening it. The examination method utilizes X rays to inspect the payload container and contents and allows the operator to view events in progress (real time) such as wave motion of free liquids. A typical RTR system consists of:

1. An X-ray-producing device
2. An imaging system
3. An enclosure for radiation protection
4. A payload container handling system
5. An operator control station.

The X-ray-producing device has controls that allow the operator to vary the voltage, thereby controlling image quality. The voltage can be varied, typically between 150 and 400 kilovolts (kV), to provide an optimum degree of penetration through the waste. For example, high-density material (e.g., solidified liquid) is usually examined with the X-ray device set on the maximum voltage. This ensures maximum penetration through the payload container. Low-density material (e.g., plastics and cellulose) is usually examined at lower voltage settings to improve contrast and image definition. The imaging system typically utilizes a fluorescent screen and a low light television camera.

Payload containers are placed in the RTR vault. Waste drums are placed on a rotating platform. The platform or the X-ray tube and imaging system move up and down to allow total coverage of the drum. X rays are projected through the payload container and onto a fluorescent screen/image intensifier. The resultant image is transferred by a camera to a remotely located television screen. The operator conducts the examination by viewing the remote television screen. The operator scans the contents of the payload container during the examination. Waste boxes cannot be rotated during RTR inspection but are first inspected from one side and then rotated 180 degrees and inspected from the opposite side. The two-sided inspection is performed to compensate for magnification factors. Large magnification factors can occur depending upon the location of an object within a box. Scanning from two sides, 180 degrees apart, allows a higher degree of accuracy for determining sizes and quantities. The two scans, from opposite sides, also provide a higher degree of confidence for detection of objects that may be hidden when scanning from only one side. The waste payload is inspected for correct physical waste form description, sealed containers, pressurized containers, and free liquid waste forms.
The RTR operator documents the findings of the RTR examination of each waste payload by several means listed below:

- The results are recorded on an RTR examination form, and this form is included in the waste payload data package. The examination results may also be entered into a computerized data collection system.

- The examination is recorded on a videotape recorder.

- The RTR operator verbally describes the results of the RTR examination on the audio track of the videotape. The audio voice track on the videotape is an additional positive feature of RTR.

The advantage of viewing the examination in real time is that the X-ray device can be adjusted on the spot to obtain optimum imaging conditions, or the system can be stopped to focus on one object. RTR works extremely well for detecting free liquids due to its ability to view events in progress, such as wave motion. The presence of free liquids is verified by jogging the container or handling system (stopping and starting the container rotation) and then watching for the resulting wave motion. Free liquids in pressurized containers are easily detected, thereby assessing two parameters, liquids and pressurized containers. Items that may otherwise go undetected due to being shielded from the X rays by another object are often found because the operator is watching the inspection in real time while rotating or moving the payload container. Interpretation of results and disposition of the inspected payload containers are also accomplished at the time of inspection rather than waiting for X-ray film to be processed.

Operator training and experience are the most important control factor in ensuring the quality of RTR interpretation and inspection. Operator training, qualification, and certification are performed in accordance with Society for Nondestructive Testing (SNT)-TC-1A.[1] SNT-TC-1A is a nationally recognized guideline and is used by employers to train, qualify, and certify employees to perform specific nondestructive tests.

Recertification of operators is based upon evidence of continued satisfactory performance and is performed at least every two years. Unsatisfactory operator performance is cause for decertification. Retraining is required before an operator is again certified to interpret and disposition payload containers.

A training drum containing a variety of different container sizes and holding various amounts of liquid is periodically examined by each operator, as prescribed in the site waste certification and quality assurance (QA) procedures. The test videotape is then reviewed by supervision to ensure that the operator's interpretations remain consistent and accurate. The test tapes are also used for monitoring the imaging system characteristics and identifying other shapes or matrices.

QA oversight functions are performed by independent individuals who review videotape of examined payload containers. The frequency and number of payload containers included in these reviews is determined in accordance with Military Standard (MIL-STD)-105D, "Sampling Procedures and Tables for Inspection by Attributes".[2]

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[1] SNT-TC-1A, August 1984, "Recommended Practice No. SNT-TC-1A, Personnel Qualification and Certification in Nondestructive Testing."

[2] MIL-STD-105D, April 29, 1963, "Military Standard, Sampling Procedures and Tables for Inspection by Attributes."

RTR has several inherent limitations. X rays with a high enough energy level to penetrate high-density waste forms or shielded containers present an image with such a large latitude that low-density materials (i.e., liquids) will be indiscernible. Therefore, high-density waste forms or shielded payload containers must have the physical waste form verified by other methods (e.g., visual examination during packaging) or be rejected.

RTR inspection is a semi-quantitative examination that can identify and verify the payload container's physical contents. RTR cannot determine the chemical composition of the waste.

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APPENDIX 5.2

DOE ASSAY METHODS USED FOR DETERMINATION OF FISSILE MATERIAL CONTENT AND DECAY HEAT VALUES OF CH-TRU WASTES

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5.2 DOE Assay Methods Used for Determination of Fissile Material Content and Decay Heat Values of Contact-Handled Transuranic (CH-TRU) Wastes

5.2.1 Introduction

The isotopic composition of the waste may be determined from direct measurements taken on the product material during the processing or post-process certification at each site, analysis of the waste, or from existing records. The isotopic composition of the waste need not be determined by direct analysis or measurement of the waste unless process information is not available. Each U.S. Department of Energy (DOE) contractor CH-TRU waste generating and/or storage site may generate and/or store one or more types of waste forms (e.g., sludge, general laboratory waste, etc.) that must be assayed.

The major contractor sites that generate and/or store CH-TRU waste are listed in Table 5.2-1, and the assay techniques used by DOE contractor sites are listed in Table 5.2-2. The specific assay methods utilized at each site are given in Table 5.2-3.

Table 5.2-1 — DOE Contractor CH-TRU Waste Generator and/or Storage Sites

1.	Argonne National Laboratory - East (ANL-E)
2.	Idaho National Engineering and Environmental Laboratory (INEEL)
3.	Lawrence Livermore National Laboratory (LLNL)
4.	Los Alamos National Laboratory (LANL)
5.	Mound Facility (Mound)
6.	Nevada Test Site (NTS)
7.	Oak Ridge National Laboratory (ORNL)
8.	Richland Hanford (RH)
9.	Rocky Flats Environmental Technology Site (RFETS)
10.	Savannah River Site (SRS)

Table 5.2-2 — CH-TRU Waste Assay Methods

1.	Passive Gamma [HPGe, Ge(Li), NaI: transmission-corrected and noncorrected]
2.	Radiochemical assay: gross alpha and gamma spectrometry
3.	Passive neutron coincidence counting (PNCC)
4.	Passive/active neutron assay (PAN)
5.	Calorimetry

Table 5.2-3 — Summary of CH-TRU Waste Assay Methods Presently Used at DOE CH-TRU Waste Generating and Storage Sites[a]

Site	SGS or NaI	PNCC	PAN (drum)	PAN (box)	Radiochemistry	Mobile PAN
ANL-E	X				X	
Hanford	X		X			
INEEL			X	X		
LLNL	X				X	X
LANL	X	X	X		X	X
Mound	X				X	
NTS						X
ORNL	X		X			
RFETS	X	X	X	X	X	
SRS	X	X	X			

[a]Calorimetry method is also used to obtain quantitative radionuclide content.

The DOE and its site contractors have been historically, and continue to be, the dominant force in assay technology development and implementation, not only within the United States but internationally as well. Some of the assay technologies (passive gamma, radiochemistry, and passive neutron coincidence counting [PNCC]) are highly developed and have a long history of implementation first to nuclear products (in safeguards and material accounting) and eventually to nuclear scraps and wastes. Other assay technologies, such as passive/active neutron (PAN), are newer developments (circa 1980), and were developed especially for application to bulk TRU wastes assays under sponsorship of the DOE. Additional improvements to the assay technology continue to be made and implemented as indicated in this document.

Where practical, the DOE sites perform multiple independent assays of waste packages as well as real-time radiography (RTR) inspection. These independent assays generally take the form of a passive gamma assay (usually segmented gamma-ray scanning [SGS]) at the waste generator site followed by PAN assay at a central certification facility.

These practices, as well as quality assurance (QA) audits and administrative controls, provide assurances that correct values of fissile material and decay heat are assigned to each waste drum. In the case of special-case drums or of significant differences among independent assay measurements, personnel at each site review all available data, including the RTR information and assay records, to determine the appropriate action. If a reasonable assay value cannot be ascertained, remedial action is taken; either reassay if measurement errors are suspected, or repackaging if the drum is suspected of nonconformance with respect to fissile material content or decay heat.

This document describes the nondestructive and destructive assay methods for CH-TRU waste employed by the DOE sites. The assay methods employed by the DOE are shown to be reliable

and accurate means of determining fissile material, radionuclide, and decay heat content of CH-TRU wastes.

Assay topics addressed for an assay method include:

(1) An overview of the assay method
(2) Applicability to CH-TRU wastes
(3) Calibration standards and implementation
(4) Operator training requirements and practices
(5) Assay procedures
(6) Assay precision, bias, and limit of detection.

More details are presented for the SGS and PAN assay methods, which are the primary methods used within the DOE complex.

All systems or methods, except for PAN, have established American Society for Testing and Materials (ASTM), American National Standards Institute (ANSI), and/or U.S. Nuclear Regulatory Commission (NRC) guidelines or methods that describe proper calibration procedures, proper equipment set-up, etc. PAN is a new technique and does not yet have a guideline or method developed. However, comparisons of PAN data with the more established assay methods (e.g., SGS or radiochemistry) are discussed that demonstrate its reliability and accuracy.

QA and quality control (QC) practices used in assay methods are presented. New nondestructive assay developments such as neutron assay imaging are also discussed.

5.2.2 Assay Overview

This section describes the general features of nondestructive assay (NDA) and destructive assay methods used by the DOE site contractors to determine the TRU content of their bulk CH-TRU waste.

ANSI N15.20-1975[1] defines NDA to be "The observation of spontaneous or stimulated nuclear radiations, interpreted to estimate the content of one or more nuclides of interest in the item assayed, without affecting the physical or chemical form of the material.

active assay. Assay based on the observation of radiation(s) induced by irradiation from and external source.

passive assay. Assay based on the observation of naturally occurring or spontaneous nuclear radiation(s)."

[1] ANSI N15.20-1975, "American National Standard Guide to Calibrating Nondestructive Assay Systems."

Destructive assay refers to chemical analysis in which a sample aliquot is removed from the item (after assuring homogenization of the batch) to be assayed and prepared for alpha and/or gamma counting.

The NRC in NRC Regulatory Guide 5.11[2] describes the applicable NDA passive measurements: "Radiations attributable to alpha particle decay, to gamma-ray transitions following alpha and beta particle decay, and to spontaneous fission have served as the basis for practical passive NDA measurements."

Gamma rays, X-rays, and/or neutrons, as well as other subatomic particles, are emitted by the various TRU isotopes as they undergo de-excitation to their respective ground states or more stable energy levels. NDA techniques based on detection of each emitted radiation have been developed and utilized for CH-TRU bulk-waste assay.

The passive gamma, passive neutron coincidence counting, radiochemical, and calorimetric methods are techniques, which are described by the ASTM, ANSI, NRC, and American Society of Mechanical Engineers (ASME) standards, guidelines, and/or regulations. These documents[1,2,3,4,5,6,7,8,9,10] provide information to the user for proper implementation of these techniques.

Characteristics of any assay measurement include precision, bias, and detection limit. Proper calibration methods must also be employed to reduce or eliminate the bias of the assay results. Definitions of each of the above terms are given below and were obtained from

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[2] USNRC Regulatory Guide 5.11, "Nondestructive Assay of Special Nuclear Material Contained in Scrap and Waste," Revision 1, April 1984.

[3] ASTM C 853-82, "Standard Test Methods for Nondestructive Assay of Special Nuclear Materials Contained in Scrap and Waste."

[4] ANSI N15.35, "Guide to Preparing Calibration Material for Nondestructive Assay Systems that Count Passive Gamma Rays."

[5] ASTM C 696-80, "Methods for Chemical, Mass Spectrometric, and Spectrochemical Analysis of Nuclear-Grade Uranium Dioxide Powders and Pellets."

[6] ASTM C 697-86, "Methods for Chemical, Mass Spectrometric, and Spectrochemical Analysis of Nuclear-Grade Plutonium Dioxide Powders and Pellets."

[7] ASTM C 759-79, Methods for Chemical, Mass Spectrometric, Spectrochemical, Nuclear, and Radiochemical Analysis of Nuclear-Grade Plutonium Nitrate Solutions."

[8] American National Standard Calibration Techniques for the Calorimetric Assay of Plutonium Bearing Solids Applied to Nuclear Materials Control, ANSI-N15.22, American National Standards Institute, New York, 1987.

[9] Standard Test Method for Determination of Plutonium Isotopic Composition by Gamma-Ray Spectrometry, ASTM C 1030-84, ibid.

[10] Standard Test Method for Nondestructive Assay of Special Nuclear Material in Low Density Scrap and Waste by Segmented Passive Gamma-Ray Scanning, ASTM C 853. This draft standard has been referenced with permission from ASTM Subcommittee C-26.10.

ASTM C 859-87[11]. Examples for each discussed assay method are found in the appropriate sections.

(1) Precision: A generic term used to describe the dispersion of a set of measured values (also referred to as "random" or "statistical error").

(2) Bias: A persistent positive or negative deviation of the method average from the correct value or accepted reference value (also referred to as "constant" or "systematic error").

(3) Detection limit: A stated limiting value that designates the lowest concentration or mass that can be estimated or determined with confidence and that is specific to the analytical procedure used.

(4) Calibration: The determination of the values of the significant parameters by comparison with values indicated by a reference instrument or by a set of reference standards.

Estimates of precision can be calculated by standard error propagation techniques.[10] Radioactive decay is random and described by Poisson statistics. For Poisson statistics, the variance in measuring N events in a detector is equal to N. (The standard deviation is the square root of the variance.)

The precision of a nondestructive assay measurement is not strongly related to the measurement item's adherence to ideal matrix and nuclide density assumptions. For destructive assay methods (e.g., radiochemical), which require sampling, the precision of repeat measurements of a single item will be strongly influenced by a lack of adherence to ideal nuclide density assumptions. However, for SGS systems, measurement bias depends primarily on the adherence of the measurement item to the assumptions of small particle size and homogeneity. Negative assay bias (reported value less than actual value) will be encountered, for example, when the nuclide is present in lumps that attenuate their own radiation to a greater extent than the surrounding material (self-absorption). Radiochemical methods that dissolve material samples will not be affected by lumps. Matrix and nuclide density have no effect on calorimeter measurements. Techniques used to correct for self-absorption effects are used in PNCC and PAN assay techniques.[3,12,13] Positive assay bias (reported value greater than actual value) can occur when, for example, system multiplication effects become severe at high-plutonium (Pu) sample

[11] ASTM C 859-87, "Standard Terminology Relating to Nuclear Materials."

[12] J. T. Caldwell, R.D. Hastings, G.C. Herrera, W.E. Kunz, E.R. Shunk, "The Los Alamos Second-Generation System for Passive and Active Neutron Assays of Drum-Size Containers," Los Alamos Formal Report LA-10774-MS, September 1986.

[13] J. T. Caldwell, et al., "System Evaluation Including Assay Algorithm, Matrix Corrections, and Operational Performance of the Los Alamos Passive/Active Neutron Assay Systems," Los Alamos Technical Report N2-87-222WP.

loadings during PNCC measurements. Typical techniques used to control this interference are: (1) equivalent reference standards used for calibration, or (2) use of source addition techniques.[14]

Of course, to obtain precise assay measurements, count-rate-dependent losses resulting from phenomena such as pulse pileup and analyzer dead-time characteristics must be monitored and corrected. These corrections are not required for calorimeter measurements. Analyzer dead-time is defined as that period of time, which is unique to the analyzer, in which it is unable to accept input signals for analysis. This correction is accomplished through the use of a combination of electronic modules, and/or radioactive sources, and/or computer algorithms (which have been obtained through the assay of calibration standards).

The uncertainty (w) in a measurement is the composite error, including both the precision and bias of the measurement. The uncertainty in a quantity f that is a function of n independent variables x_i is given by:

$$w = \left[\sum_{i=1}^{n} w_i^2 \right]^{1/2}$$

where w_j is the uncertainty in the variable x_i.

Assay item preparation is generally limited to good waste/scrap segregation practices that produce relatively homogeneous items that are required for any successful waste/inventory management and assay scheme, regardless of the measurement method used.

5.2.3 Assay Methods Descriptions, Characteristics, and Limitations

This section describes the various assay methods, presents their characteristics (precision, bias, and detection limits), and discusses their limitations and applicability to assay of CH-TRU wastes. Assay methods discussed include calorimetry, passive gamma assay (e.g., SGS), radiochemical methods, PNCC, and PAN assay.

5.2.3.1 Calorimetry

Calorimetry has been used for many years in the nuclear weapons program for product assay of weapons grade (WG) Pu. Many of the NDA Pu standards in use throughout the DOE complex have been characterized by calorimetry. A large number of standard radiochemical and gravimetric assay comparisons have been performed to verify the accuracy of calorimetric assay measurements.

Basically, calorimeters measure the heat flow out of contained small packages. Experimental difficulties grow exponentially with package size, so this method is generally used only with small packages, a few liters in volume at most. The primary heat release in WG Pu materials is from alpha and beta decay, and with a knowledge of isotopic composition, precise Pu mass

[14] R. B. Perry, R. W. Brandenburg, and N. S. Beyer, 1972, "The Effect of Induced Fission on Plutonium Assay with a Neutron Coincidence Well Counter," Transactions of the American Nuclear Society, 15 674.

values are readily obtained from virtually any physical or chemical form of Pu material, without knowledge of precise compound stoichiometry (e.g., Pu-to-oxygen ratio).

The kinetic energy of the emitted alpha or beta particle and the energy of the emitted alpha or beta particle and the recoil nucleus is transformed into heat, together with some fraction of the gamma ray energies and conversion electrons that may be emitted by the excited daughter nucleus in lowering its energy to a more stable nuclear configuration. The electrons and low-energy gamma rays are totally absorbed, while the higher-energy gamma rays that may escape from the calorimeter chamber comprise less than 0.01% of the total decay energy. Thus, most of the energy associated with these transitions of the daughter nucleus to ground state, as well as all of the energy associated with the alpha particle and recoil nucleus, is absorbed within the calorimeter.

The calorimeter method measures the total decay heat produced by an item. The relative isotopic abundances of the Pu and americium (Am) nuclides in the mixture and the values of decay heat per gram for each nuclide are used to calculate the average decay heat per gram of nuclide mixture. The total measured decay heat divided by the average decay heat per gram yields the grams of nuclide mixture.

ANSI N-15.22[8] describes the calorimetry procedure and equipment used for the assay technique. This standard method is used in DOE facilities for calorimetry calibrations, setup, and as the guide to operational measurements.

5.2.3.2 Passive Gamma Assay

5.2.3.2.1 Segmented Gamma Scanning (SGS)

High Resolution (Hyperpure Germanium [HPGe], Lithium-drifted Germanium GE[Li]), Transmission and Count-Rate Corrected Assays.

Overview

The first NDA measurements of TRU isotopes using passive gamma rays were performed by DOE contractor personnel more than 40 years ago. Passive gamma-ray NDA of TRU isotopes is a highly developed technology, and is also the most widely implemented. The introduction more than 20 years ago of germanium solid state detectors and the subsequent incorporation of these detectors into computer-based detection packages has improved the resolution and reliability of these systems. Commercial manufacturers of these systems include Canberra of Meriden, Connecticut, and Nuclear Data of Schaumberg, Illinois.

The number of individual TRU isotopes or their daughters that can be assayed with SGS is large; uranium (U)-233, Pu-238, Pu-239, neptunium (Np)-237, Am-241, Am-243 being among the more common ones. In each case, one or more characteristic moderate-to-high energy gamma rays are emitted in sufficient intensity to permit estimates of quantities in low-to-moderate density waste packages as large as 208-liter (L) drums. The recommended (ASTM C 853-82)[11] experimental arrangement for SGS assays is shown in Figure 5.2-1. This

figure displays the essential elements required for SGS assays of TRU isotopes in any package size.

To minimize assay errors due to axial inhomogeneities, assays are performed in segments along a waste package's vertical axis. The effects of radial inhomogeneities are minimized by rotating the drum during the assay measurement. The detector is shielded in such a manner so as to allow the waste drum to be scanned in segments (typically 10 to 20 segments).

Gamma-ray attenuation is measured for each segment with a transmission source in the indicated geometry of Figure 5.2-1. The energy of this source is selected to match that of the gamma-ray line(s) being measured (e.g., Oak Ridge National Laboratory [ORNL] uses a mixed europium (Eu)-152/Eu-154 oxide source for its large array of gamma-emitting radioisotopes [50 keV to 1600 keV], and selenium-75 is typically used for Pu-239 assays).

State-of-the-art counting electronics allow dynamic counting rate ranges of factors of 10^4 to 10^5 or more, with dead-time corrections measured with a second small, low-energy source positioned near the detector. Waste packaged are automatically rotated about their vertical axes and cycled through the required segment heights with standardized, computer-controlled electronic motors and precision mechanical turntable/elevator hardware. SGS hardware-software packages are commercially available from several manufacturers.

Applicability to CH-TRU Wastes

A prime factor that determines applicability of SGS to perform assay measurements of CH-TRU waste packages is gamma-ray transmission through the package. Other factors affecting assay measurements include particle self-absorption and nonhomogeneity of the assayed item ("lumping"). Two conditions must be met to optimize assay results. First, the particles containing the nuclide must be small to minimize self-absorption of emitted gamma radiation. Second, the mixture of material within a package segment must be reasonably uniform in order to apply an attenuation correction factor, computed from a single measurement of item transmission through the segment. Variations in item composition and density within a vertical segment lead to indeterminate errors. Such variations should be minimized through strict scrap and waste segregation procedures.

A combination of analytical error analysis[3] and experimental usage over many years has determined that transmission factors greater than or equal to 0.5% are required for accurate SGS assays. The physical density of a waste package that this requirement defines depends greatly on the package size [i.e., the radial distance from the gamma-emitting source(s) to detector] and the energy of the gamma rays used for the analysis. Four-liter packages having densities as high as 2 grams per cubic centimeter (g/cm^3) meet the criterion, whereas 208 L packages are limited to densities of 0.5 g/cm^3 or less. To assure compliance with these limits, all current SGS software packages include an automatic warning indicating when the transmission factor for any sector falls below the prescribed limiting value. The routine practice at some sites is to calculate a contribution from that sector based on the lower-limit transmission (e.g., 0.5%).

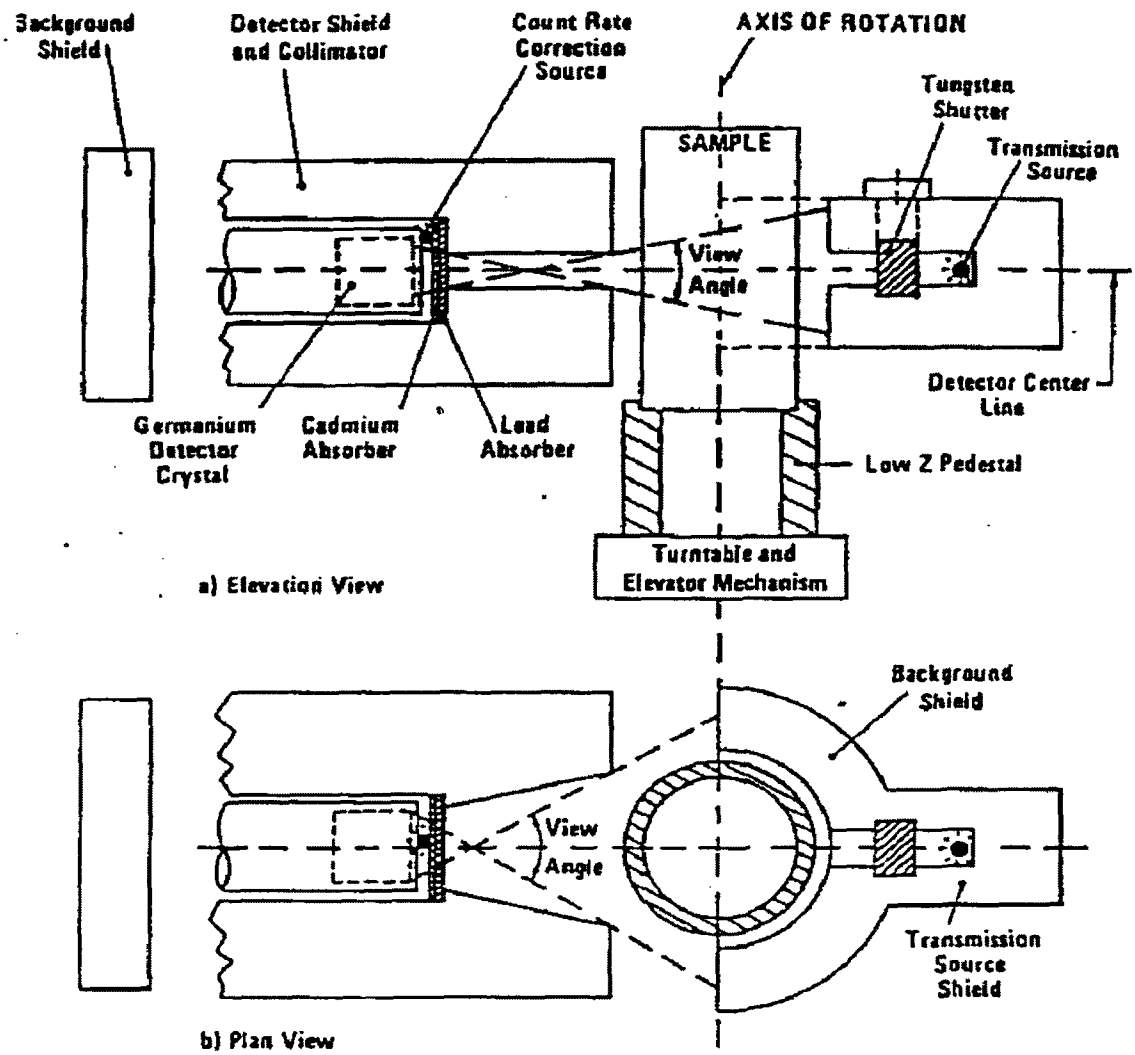


Figure 1
Schematic Arrangement for Segmented Gamma-Ray Scanning (SGS) System

Figure 5.2-1—Schematic Arrangement for Segmented Gamma-Ray Scanning (SGS) System

The reason for maintaining the assay value, rather than disregarding it, is because most SGS transmission failures occur for only one sector out of the 10 to 20 drum sectors assayed. This sector, on the average, contains only a small fraction of the waste drum's total TRU inventory of gamma-emitting isotopes.

On the average, estimating the TRU content for one or two failed segments in this fashion results in only a small overall error for the waste drum. Other sites (e.g., Hanford) flag such drums for management decision on whether the item should be disassembled, examined, and repackaged, or reassayed on a neutron sensing assay instrument, which is less sensitive to density variations. Since the SGS assay value for a transmission failure is truly a lower limit, and as is discussed in detail in Section 5.2.3.4, passive neutron assays generally provide upper limit assay values (especially for WG Pu); the combination of SGS and passive neutron assay methods tends to bracket the actual assay value.

Some matrix forms are inherently unsuitable for SGS analysis. Such forms may contain 'lumps' of nuclide, that is, nuclides contained in small volumes of matrix material having a localized density substantially different from the bulk density of the rest of the container. The dimensions of nuclide particles that constitute a lump vary with the energy of the emitted radiation used for the analytical measurement. The possible magnitude of the problem may be estimated from the following example of attenuating effects. A plutonium metal sphere 0.02 cm in diameter will absorb approximately 4% of the 414 keV, Pu-239 gamma rays produced. Approximately 15% of the 186 keV gamma rays of U-235 will be absorbed in a uranium metal sphere of the same diameter.

As mentioned previously, another condition that will cause measurement problems is presented by containers with several irregular regions, highly variable in density, that prevent the calculation of a valid attenuation correction based on the transmission measurement. In case of such a condition, an analytical method less sensitive to nuclide and matrix densities, such as PNCC, should be employed.

Careful inspection of the transmission and nuclide peak areas for each segment may provide clues when a measurement should be suspect. Sudden, discontinuous changes in the transmission values for adjacent segments or high nuclide count values for isolated segments are examples of signals indicating possible problem items.

Instrument Calibration, Standards Preparation, and Implementation

The recommended DOE facility standard guide used for preparation of SGS standards is ANSI N-15.35.[4] The recommended DOE facility standard for implementation of these standards is ANSI N-15.20.[1] ANSI N-15.20 calls for the preparation of the calibration material using intimate and stable mixtures of the TRU isotope with matrix material and for preparation of a suitable number of calibration standards to cover the anticipated isotopic concentration region of interest (ROI). In the case of Pu-239, this range is 5 to 200 grams (g) for 208 L drums.

When establishing a calibration curve for the SGS instrument, as least two calibration standards are used for each content code. One standards drum contains a TRU isotopic mass near the low

end of the ROI (e.g., 5g Pu-239) while the other contains a TRU isotopic mass near the high end of the ROI (e.g., 200g Pu-239). Both drums contain waste stream matrix mixtures and densities designed to simulate the waste streams. Some sites use more than the two drums described above to ensure a proper calibration factor. Other sites measurement control programs require standards drums to be measured by the assay instrument multiple times, both before and after each measurement session.

Acceptable ranges for calibration data are specified in the operating procedures (e.g., ORNL accepts a variance of +/- 5%). If assay measurement falls outside the acceptable range, no production assay measurement is performed until the issue has been resolved by a designated NDA expert.

Operator Training Requirements and Practices

Present-day commercial SGS systems, such as the Canberra and Nuclear Data models, are highly-automated, computer-based systems. The instruments are computer-controlled using relatively interactive ("user-friendly") software. Only trained personnel are allowed to operate the assay equipment. Personnel are qualified according to DOE Order 5480.5.[15]

Each site provides a specialized training program for NDA instrument operators. The operators are directed and/or assisted by a designated site NDA expert. Expertise is attained by education and experience.

Assay Procedures

The assay procedures cited in ASTM C 853-82, "Standard Test Methods for Nondestructive Assay of Special Nuclear Materials Contained in Scrap and Waste,"[3] are recommended for use at all DOE facilities. These procedures stress usage of proper calibration standards, proper equipment and equipment setup, avoidance of practices (such as misalignment of the waste package) known to result in inaccurate assays, attention to proper record keeping and equipment maintenance, and safe operation of the equipment.

Assay Precision, Bias, and Detection Limit

Assay precision is generally taken to mean measurement repeatability. In the case of typical SGS systems, operated and calibrated according to the recommended procedures, repeatability of results is limited only by statistical counting errors. Counting statistics, in turn, are a strong function of TRU isotopic loading and counting time.

ASTM C 853[10] discusses SGS precision and bias in detail. Some of that discussion follows. The precision of an SGS assay is a function of the precision of the peak areas measured for each segment. The precision of an assay is normally better when the following conditions can be obtained:

[15] DOE Order 5480.5, "Safety of Nuclear Facilities," September 23, 1986.

- Increased count time
- High transmission source activity
- Low attenuation for gamma radiation in the energy range of interest.

Typical SGS assay precisions for low-density wastes are listed in Table 5.2-4. Certain matrices, such as graphite molds and cemented insulation, whose densities are above the prescribed SGS-applicability limit of 0.5 g/cm^3 for 208 L packages, and drum handling (homogeneity of calibration standards may be jeopardized) can have a deleterious effect on assay precision.[3]

Table 5.2-4 — Typical SGS Assay Precisions for WG Pu in Low-Density Wastes Contained in 208-L Drum

WG Pu (g)	Precision
1	+/- 100%
10	+/- 10%
30	+/- 3%

(Table 5.2-4 values reflect the assumption that the guidelines given in ASTM C 853-82[3] were adhered to in acquiring the data.)

The precision of an assay performed by SGS is not strongly related to the measurement item's adherence to ideal matrix and nuclide density assumptions. However, measurement bias depends primarily on the adherence of the measurement item to the assumptions of small particle size and homogeneity. Negative bias will be encountered when the nuclide is present in lumps that attenuate their own radiation to a greater extent than the surrounding material. Positive bias can result from low transmission items with over-corrected end effects. Items containing high-density areas may be biased either high or low or be unbiased, depending on the relative position of the high density area and the nuclide of interest. In the majority of measurement situations, however, it is expected that when biases exist, measurement results will be lower than true values.

Several SGS and destructive assay comparison studies of several waste forms indicate SGS assay biases of 10% or less, at the 95% confidence level.[16] Assay biases for low-density waste matrices contained in 208-L drum packages are 5% or less. In small package applications (based on numerous Safeguards and Nuclear Materials Accounting applications), SGS assay biases of less than 0.5% have been reported.[16] The basic assay formalism associated with the SGS method, that is, transmission correction and the use of small segments, is conducive to very accurate results if recommended procedures are correctly followed. Heterogeneous matrices and isotopic concentration variations can have a severe and unpredictable effect on assay bias for a given waste drum.

Typical SGS assay biases for two types of wastes are summarized in Table 5.2-5.[16]

[16] Fleissner, John G. and Hume, Merril W., "Comparison of Destructive and Nondestructive Assay of Heterogeneous Salt Residues," RFP-3876, March 29, 1986.

Table 5.2-5 — Typical SGS Assay Biases

Waste Type	Biases
Heterogeneous salts	10%[a]
Low-density (e.g., combustibles)	5%

[a] Based upon assay of a number of 4-L waste packages prior to placing in a 208-L drum.

(Table 5.2-5 values reflect the assumption that the guidelines given in ASTM C 853-82 were adhered to in acquiring the data.)

SGS assay limit of detection for typical applicable wastes and standard counting times in current routine use in DOE facilities (30 min or less per assay measurement) is about 5g WG Pu. This is based on 30% assay precision. Usually SGS assays are only performed on 208-L drums when screening indicates 10 g or more WG Pu to be present.

SGS Assay Results Comparisons

Calorimetric assay measurements of heterogeneous molten salt residues have provided a total assay value for the Pu and Am, as an NDA comparison (referee) technique for an assessment of SGS assay of RFETS molten salts.[16] Reliable interpretation of the calorimetry measurements depended on an accurate determination of the Am/Pu ratio, since the relative amount of heat produced by the Am in these samples was typically 50% or more.

Gamma-ray spectral isotopic analysis coupled with calorimetry was performed at Mound on nine cans of molten salt, which were subsequently returned to RFETS for dissolution and solution quantification. Results of the Mound measurements show a relative standard deviation range from 0.032% to 0.50% for Pu values and 0.23% to 0.39% for Am values. No biases or statistical differences between pairs or measurements were noted.

5.2.3.2.2 NaI (low-resolution) Assays

Both transmission-corrected and transmission-uncorrected sodium iodide (NaI) assay units are used in DOE facilities. The transmission-uncorrected units are used, for example, at RFETS for low-density wastes containing up to 20 g Pu.

The NaI transmission-corrected assay instruments are special function units, servicing isolated waste streams producing a single type of waste under generally constant conditions. Typically, these NaI units consist of five individual collimated NaI detectors mounted at different heights that view a rotating drum, effecting a five-segment assay. Transmission source geometry is similar to that shown in Figure 5.2-1. The pertinent NDA guidelines outlined in ASTM C 853-82[3] are applied to these systems in a fashion similar to that described for SGS units. A two-window pulse height analysis is performed to correct for Am-241 and fission product interferences, and software indicators flag segments containing excessive Am or fission product amounts.

Calibration standards are carefully chosen for the particular waste stream being monitored, and assays are performed in a fashion similar to SGS assays. These units produce, on the average, reliable results as determined by numerous quantitative comparisons with SGS and PAN assays of the same waste drums.[17] Analytical studies of assay biases for these systems indicate +/-10% levels of bias. This has been verified with SGS and PAN comparisons of a large number of drums assayed with these NaI systems.[17] It should also be pointed out that assay standards practices and procedures are adhered to in accordance with ASTM C 853-82 and ANSI N-15.35. Duplicate assays with SGS or PAN (performed at the certification facility) provide additional assurance that proper TRU assay values are being generated with these systems.

5.2.3.3 Radiochemical Methods

The basic application of radiochemical methods in TRU waste assays is in quantifying radioisotopic content of process liquid or sludge waste forms. Before final drying or cementation, a batch of process sludge is contained in a single large tank. The sludge is then mixed for a sufficient period of time to assure a homogenous mixture. This mixture is then sampled at several points while circulating and the samples subjected to routine radiochemical processing and analysis (precipitation and separation followed by alpha and/or gamma spectrometry). The prepared aliquot samples are assayed in a standard alpha spectrometer. In the cases of higher-activity sludges, these samples are assayed using another method (e.g., passive gamma-ray spectroscopy). Assays of samples obtained from individual sludge drums may also be performed.

Using standard analyses, the individual TRU isotopic activities are determined; Pu-239, total Pu (WG Pu), and Am-241. These aliquot sample activities are then used to determine the original batch TRU activity, on an individual isotopic basis. By proper accounting for any volume reduction or increase produced in the drying or cementation process, these batch activities can then be used to determine an individual 208 L drum's specific Pu and Am content, simply by weighing the drum and accounting for the amount of original batch sludge that was deposited in that particular drum. Accurate final drum assays depend upon following the procedure outlined in a careful manner, with maintenance of a homogeneous mixture during both the crucial sampling and drum filling stages.

Standard test methods (e.g., ASTM C 696-80[5], ASTM C 697-86[6], and ASTM C 759-79[7]) describe the radiochemical standard aliquot sampling procedures used in the DOE facilities. Assay standards are prepared and used as indicated in the standard test methods. Sampling, weighing of the sample, and handling the sample are done under conditions that assure that the sample is representative of the lot or batch. A lot or batch is defined as any quantity of solution that is uniform in isotopic, chemical, and physical characteristics by virtue of having been mixed in such a manner as to be thoroughly homogeneous. All containers used for a lot or batch are positively identified as containing material from a particular homogeneous solution.

[17] F. J. Schultz and J. T. Caldwell 1988 DOE Model Conference paper.

Assay biases at the final filled-drum stage are difficult to estimate, since they depend primarily on the maintenance of homogeneous mixtures during the sampling, drying/cementing, and final drum-filling stages.

5.2.3.4 Passive Neutron Coincidence Counting (PNCC) Assays

5.2.3.4.1 Overview

PNCC assays include assays conducted on small packages that are summed to give the final reported assay value for a waste container, and the passive portion of the PAN assay system, which is discussed in detail in Section 5.2.3.5.

PNCC method for determination of Pu assay in product materials has been used for Safeguards verification purposes within the DOE complex for more than 20 years. PNCC has also been applied to the assay of TRU-bearing wastes and scraps for many years (ANSI N15-20-1975[1], subsections 20-28). In addition, the NRC Regulatory Guide 5.11[2] describes NDA techniques acceptable to the NRC for assay of wastes and scraps, which includes PNCC. These standards and regulatory guides are used to ensure proper application by the DOE of PNCC to scrap and waste assay. In fact, DOE laboratories, primarily Los Alamos National Laboratory (LANL), have been largely responsible for the development of PNCC as a reliable assay technique for TRU wastes, as well as for scraps and product.

The prototypical PNCC is comprised of a high-efficiency neutron detector large enough to accommodate the waste package of interest. It operates by detecting the number of time-correlated neutrons being emitted spontaneously by the assay item. In fission events, bursts of 2,3,4,5,... neutrons are emitted simultaneously, and the detection of two or more of these in time coincidence serves to identify the original fission event within the material being measured. Specialized counting electronics (e.g., shift register) have been devised to accomplish and record these measurements. These are discussed in detail in ASTM C 853-82[3].

Any TRU isotope that undergoes spontaneous fission at a measurable rate can be quantified by PNCC. Comprising this category are the even isotopes of Pu, curium (Cm), and californium (Cf). Most commonly within the DOE complex, the different grades of Pu [WG, reactor grades (RG) of different isotopic compositions, heat source grade (HSG)] are quantified by coincidence counting of the included mixture of even isotopes of Pu; predominantly Pu-240 for WG and RG, and Pu-238 for HSG. Thus, with knowledge of the Pu isotopics, the observed coincidence rates can be interpreted to yield total Pu mass.

5.2.3.4.2 Applicability to CH-TRU Wastes

The primary requirement for application of PNCC to CH-TRU waste assay is knowledge of the included isotopics, since normally the quantity of interest is the total elemental mass (i.e., total Pu mass) rather than the even isotope masses only. In addition, the wastes should not include mixed-element spontaneous fission emitters. For instance, it is undesirable to have Cm and Cf isotopes present in the same assay item containing Pu isotopes. Most DOE CH-TRU wastes contain Pu even-isotope spontaneous fission emitters. A typical average WG Pu isotope mix

contains 5.8% Pu-240. Pu-240 is responsible for more than 99% of coincidence neutrons detected in typical WG Pu wastes.

5.2.3.4.3 Instrument Calibration, Standards Preparation, and Implementation

Calibration of PNCC instruments, similar to SGS, is obtained by establishing a curve of instrument response versus isotopic mass.[1,3] A minimum of four calibration points are obtained over the mass range of interest using standards that are representative of the materials being measured. Within each content code, or waste category, the variation due to interference effects within the boundaries defining the limits of that category is measured. Calibration standards are constructed using containers identical to those for the scrap or waste, with contents that are representative of the range of matrix conditions to be encountered. It is not recommended to extrapolate beyond the calibration range established during instrument calibration. Encapsulated Cf-252 sources, such as those used at ORNL for passive PAN calibration, are available to be used for PNCC calibration purposes.

Acceptable ranges for calibration data are specified in the operating procedures. If assay measurement values fall outside the acceptable range, no production assay measurements are performed until the issue has been resolved. Operators contact a designated NDA expert for consultation.

5.2.3.4.4 Operator Training Requirements and Practices

Present-day commercial PNCC systems, such as the JOMAR or National Nuclear models, are highly-automated, computer-based systems. The instruments are computer-controlled using relatively interactive ("user-friendly") software. Only trained personnel are allowed to operate the assay equipment. Personnel are qualified according to DOE Order 5480.5.[15]

Each site provides a specialized training program for NDA instrument operators. The operators are directed and/or assisted by a designated site NDA expert. Expertise is attained by education and experience.

5.2.3.4.5 Assay Procedures

The assay procedures cited in ASTM C 853-82, "Standard Test Methods for Nondestructive Assay of Special Nuclear Materials Contained in Scrap and Waste,"[3] are recommended for use at all DOE facilities. These procedures stress usage of proper calibration standards, proper equipment and equipment setup, avoidance of practices (such as misalignment of the waste package) known to result in inaccurate assays, attention to proper record keeping and equipment maintenance, and safe operation of the equipment.

5.2.3.4.6 Assay Precision, Bias, and Limit of Detection

Most PNCC units are used to assay small packages (4-L size), which are then placed into larger waste containers, such as 208-L drums. Assuming proper administrative control of drum filling, this practice greatly reduces the assay errors associated with all PNCC performance effects except counting statistics and isotopics. Additional errors caused by self-multiplication or system dead-time are significant only when strong neutron sources are present.

Sources of assay biases and measurement uncertainties include:

(a) Counting statistics: A significant source of error at both extremes of count rate. Measurement uncertainty can be significant at very low count rates for all assay conditions. For every high count rates, when the rate is due primarily to a strong (alpha, n) internal source or induced fissions, assay bias is increased.

(b) Isotopics: For WG Pu waste assay biases produced by systematically incorrect actual Pu isotopics are 3% or less, based on use of historic average WG Pu isotopics. Uncertainties in the measurement of the isotopic composition, generally considered to be unbiased, increase the uncertainty of the assay value.

(c) Self-multiplication (or induced fissions): Generally a problem when fairly large Pu amounts are present in conjunction with strong (alpha, n) sources within the same drum (measurement value greater than actual value). This phenomenon is a source of potential bias producing uncertainty in the assay value. Multiplication effects should not be significant when TRU gram loadings are low and waste volumes are large.

(d) System dead-time: A problem when strong neutron sources are present (measurement value is less than actual value). This phenomenon is a source of potential bias with an associated uncertainty.

(e) Calibration: Typically, assay uncertainties produced by uncertainties in calibration are 3% or less.

(f) Matrix effects: Matrix effects include neutron poisons (e.g., boron, cadmium) and other neutron emitters [species that spontaneous fission or have strong (alpha, n) reactions] and neutron moderators. See Caldwell et al., 1986[12] for a discussion of moderator error estimation techniques.

Table 5.2-6 provides a summary of typical PNCC assay error contributions for low-density waste forms.

Table 5.2-6 — Summary of Typical PNCC Assay Error Contributions for Low-Density Waste Forms

Error Contribution	WG Pu	Typical Errors
Counting statistics	1 g 10 g 30 g 100 g	+/- 50% +/- 10% +/- 5% +/- 3%
Self-multiplication		+/- 5%
System dead-time		+/- 3%
Isotopics		+/- 3%
Calibration		+/- 3%
Matrix (low-density)		+/- 5-20%

Estimates of PNCC assay uncertainty can be calculated by standard error propagation techniques from the various bias contribution variances. The grams of plutonium calculated by PNCC is a function of net passive coincidence neutron count rate (gross neutron coincidence count rate minus accidental neutron coincidence rate) (SIGP), self-multiplication (MULT), system dead time (SYSDT), isotopics (ISOP), calibration (CALIBP), and moderator index (MI).

For 30 grams of plutonium, the uncertainty in the 30 grams of plutonium is given by (see Section 5.2.2):

$$\begin{aligned} w &= [f(SIGP)^2 + f(MULT)^2 + f(SYSDT)^2 + f(ISOP)^2 + f(CALIBP)^2 + f(MI)^2]^{1/2} \\ &= [(0.05)^2 + (0.05)^2 + (0.03)^2 + (0.03)^2 + (0.03^2 + (0.20)^2]^{1/2} \\ &= 0.218 \text{ or } 21.8\%. \end{aligned}$$

The passive mass assay value reported by the PNCC assay algorithm would then be:

Passive mass (grams) = 30.00 +/- 6.55.

Anderson and Lemming, 1982, MLM-3009, Table 5, p. 33, [18] shows overall neutron production rates for several of the more common TRU isotopes and several of the more common matrices that produce significant (alpha, n) reactions. For example, WG Pu, which has an average alpha energy of 5.15 megaelectron-volts (MeV), produces approximately 2 neutrons per second per millicuries-alpha (n/s/mCi-alpha) in an oxide matrix and 215 n/s/mCi-alpha in a fluoride matrix. Pu-238 and Am-241, which have an average alpha particle energy of 5.5 MeV, produce approximately 2.5 n/s/mCi-alpha in an oxide matrix and approximately 310 n/s/mCi-alpha in a fluoride matrix. These values are representative of pure chemicals and alloys. Neutron production rates for waste materials will be less, since the TRU isotopes are more widely dispersed and the alpha particles are less likely to encounter a productive target.

The more usual (alpha, n) reactions that can cause passive assay concerns consist of normal WG Pu in which a sizeable fraction of the Pu is chemically bound to either fluorides or bound in a salt mixture containing aluminum or magnesium. Typically, metal oxide or nitrate forms of TRU isotopes (which produce approximately 0.7 to 2 n/s/mCi-alpha) present no problems for passive neutron assays (both passive PAN and PNCC). In practice, rates ranging to 20 n/s/mCi-alpha do not decrease passive assay precisions drastically as long as the alpha sources present are only those associated with WG or RG Pu. However, waste streams that include additional Am-241 can be difficult to assay passively even if the TRU chemical form is the oxide.

[18] M. E. Anderson and J. F. Lemming, "Selected Measurement Data for Plutonium and Uranium," MLM-3009 (or ISPO-157), November 1982.

5.2.3.5 Passive-Active Neutron (PAN) Assay Systems

5.2.3.5.1 Overview

PAN assay systems consist of two independent assay units—passive and active neutron. The combination of passive and active neutron assays within a common system provides a unique set of information.

The passive assay portion of the PAN method described below is an adaptation of the PNCC method using "self-measurement" matrix corrections. Two complete passive assay detection systems are maintained with separate counting electronics. Two detection systems are used separately or in combination depending upon the neutron count rate. The passive coincidence measurement provides quantitative information on even isotopes present in the waste container, such as Pu-240. The passive singles neutron count rate (difference between total neutron rate and that due to spontaneous fission events) provides semi-quantitative information on alpha particle emitters present in the waste container, such as Am-241. The active assay provides quantitative information on the Pu-239 and other fissile isotope constituents. See Schultz et al., 1984[19] and WIPP-DOE-157, 1989[20] for a more complete description of the system.

For WG Pu, the passive coincidence and active assays provide independent total Pu assay values. This fact has been extremely important in verifying the accuracy or determining the bias of the PAN assay measurement technique, as presented in Caldwell, et al., 1986.[12] This formalism has also been verified by extensive comparisons of both passive and active neutron assays with SGS.[10] PAN assays systems have been developed for both drums and boxes (see Section 5.2.5 for discussion of box PAN assay systems). For these relatively large waste containers, effects of the waste material (matrix) on the neutron signals observed cannot be neglected.

5.2.3.5.2 Instrumentation

A basic cross-sectional view of a typical LANL PAN detection system, showing the schematic "interwoven" layout of the two distinct types of neutron detection packages (bare He-3 and cadmium (Cd)-shielded He-3 detector tubes), is shown in Figure 5.2-2.

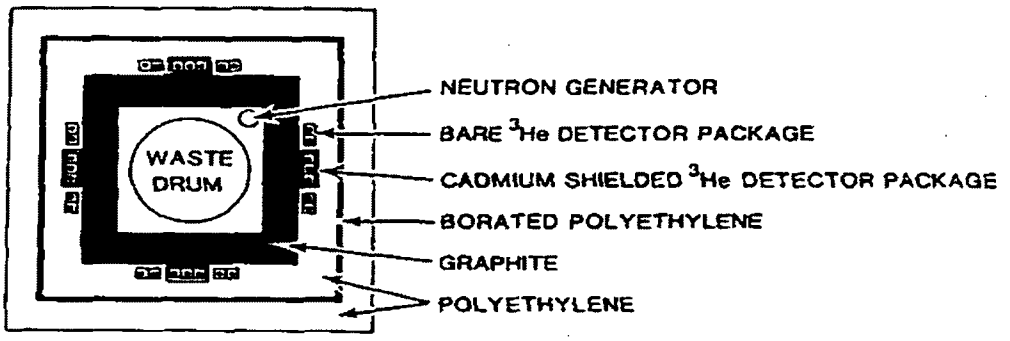
Passive Assay Portion

The passive portion of the PAN assay system utilizes the two types of detection packages to:

(a) determine a MI used to determine a correction to the assay calculation to account for the matrix characteristics, and

[19] F. J. Schultz, et al., Oak Ridge National Laboratory; J. T. Caldwell et al., Los Alamos National Laboratory, "First-Year Evaluation of a Nondestructive Assay System for the Examination of ORNL TRU Waste," ORNL-6007, April 1984.

[20] "Data Package Format for Certified Transuranic Waste for the Waste Isolation Pilot Plant," WIPP-DOE-157, Rev. 2, January 1989.



Cross-sectional views of the second-generation assay chamber show the layout and relative positions of both shielded and bare ^{3}He detector packages.

Figure 2
Schematic PAN System Layout

Figure 5.2-2—Schematic PAN System Layout

(b) optimize counting statistics depending on the actual relative neutron sources encountered.

For low count-rate waste containers, all counts detected by the neutron detector packages are summed to yield the lowest assay limit of detection possible. All detector count rates (acquired by both bare and shielded detectors) are summed electronically to obtain a "System Totals" neutron detection efficiency of approximately 12%.

For waste containers with higher Pu loadings (e.g., 100 g or more) coupled with strong (alpha, n) backgrounds, the Cd-shielded detectors are summed independently, and the "Shielded Totals" count rate is formed with a resulting neutron detection efficiency of 2.9%. However, this detection package possesses a much faster "die-away" or "neutron-collection" time, approximately six times faster than that of the slower "Systems Totals", which is approximately 15 microseconds. At low count rates, the slower collection time is of no consequence [i.e., accidental coincidences due to (alpha, n) reactions are small] and, thus, the Systems Totals provides not only a more sensitive but also statistically more precise passive assay measurement.

At higher count rates the faster die-away time of the Shielded Totals gains a higher precision than the less specific count rate (Systems Totals). As a consequence, at high neutron count rates the Shielded Totals Coincidence rate is used to obtain the more precise passive assay measurement value.

The cross-over count rate (i.e., the count rate at which the assay measurement value obtained by the Shielded Totals supplants the Systems Totals) has been experimentally determined to be approximately 2000 counts per second (cps) (Systems Total count rate), and this value is used in the assay algorithm. There is a substantial range in which either Systems Coincidence or Shielded Coincidence rates both provide precise assay values. Many data comparisons have been performed in this cross-over region to verify the self-consistency between the two coincidence measurements.[17]

Active Assay Portion

The active portion of PAN systems performs a high-sensitivity, pulsed thermal neutron interrogation assay of waste drums. As shown schematically in Figure 5.2-2, a small 14-MeV neutron generator placed within the assay chamber between the waste drum and moderating walls provides short pulses (5-10 microseconds [ms] of high-energy interrogating neutrons. In approximately 0.5 ms, all original fast neutrons in this interrogating pulse have been thermalized by multiple collisions with the graphite and polyethylene walls and moderating materials within the waste drum. This "thermalized interrogating pulse" persists ($T^{1/2}$ about 400 ms) for some time, during which induced fissions within the waste drum are produced, primarily in Pu-239 or other fissile isotopes. These events, in turn, result in prompt-fission, spectrum neutrons being emitted by each fissioning nucleus.

The cadmium-shielded detection packages have been designed to reject an external thermal neutron flux to 1 part in 10^7, but to respond sensitively to fission spectrum neutrons. The

summed shielded detector packages shown in Figure 5.2-2 detect about 3% of all induced fission events that are produced within typical waste drums.

An additional measurement feature not shown in Figure 5.2-2, but discussed at length in Caldwell, et al., 1986,[12] is the set of thermal flux monitors, one Cd-shielded and collimated and the other bare, that are also positioned inside the assay chamber between the waste drum and the moderating walls. As discussed at length in Caldwell, et al., 1986,[12] the ratio of these flux monitors is highly sensitive to the neutron absorption characteristics of the waste drum contents. This ratio is used to form a drum "Absorption Index" (AI) (see Section 5.2.3.5.3).

5.2.3.5.3 PAN Assay Matrix Corrections

Two types of matrix effects can interfere with the active neutron measurements: absorption and moderation.[12] The absorption effects occur almost entirely as an attenuation of the interrogating thermal neutrons, caused by the presence of various neutron poisons within the waste matrix (e.g., boron, cadmium, chlorine, etc.).

Moderation effects occur at two stages of the measurement. The original burst of 14-MeV neutrons can be moderated to a considerable extent during passage through the waste matrix. Generally, this results in a larger thermal neutron interrogation flux than would have been produced in the absence of matrix. After the interrogation flux has produced fission reactions within the waste matrix, the same moderating materials can attenuate the prompt-fission signal neutrons resulting in a decrease in observed response relative to the no-matrix case. This attenuation of signal fission-neutrons also is the primary matrix effect for the passive measurement.

The approach to matrix corrections has been to base corrections on measured quantities determined as adjuncts to the primary active and passive TRU assay measurements. The systematic matrix correction algorithm is based on an analytic fit to assay measurements obtained for different positions of the source within a matrix drum. These analytic fits then provide estimates of uncertainty for the active and passive assay data.

The absorption matrix correction approach used by the PAN systems employs a ratio of an unshielded in-chamber flux monitor to a cadmium-collimated, in-chamber flux monitor (designated the barrel flux monitor). This ratio is termed the AI. The barrel flux monitor detects those neutrons that have undergone drum matrix interactions. The ratio of the monitors strongly reflects the neutronic properties of the matrix.

$$AI = \frac{[\text{flux monitor response (0.7-4.7 ms)}]}{[\text{barrel flux monitor response (0.7-4.7 ms)}]} \quad (1)$$

The MI depends upon the responses of the two detection systems (Cd-shielded and bare) to moderated neutrons. The shielded detectors are insensitive to thermal neutrons, while the bare detectors are very sensitive to the thermal neutron flux. In turn, the thermalized fraction depends very strongly on the moderator density of the matrix. To use this relationship in obtaining matrix correction factors, the ratio is normalized so that a value of zero is obtained when no moderator is present and, in addition, a small correction is made to account for self-absorption effects.

The general equation for a moderator index is given in Equation (2).

$$MI = \{1 - [(\text{shielded totals})/(\text{system totals})]/A_o\} \times \{A_1 + A_2 \times \ln(AI)\} \quad (2)$$

The term within the first set of brackets is the basic raw spectral data, and the term within the second set of brackets is the correction term for matrix absorption effects. The same MI values are used for both active and passive matrix corrections.

In order to obtain data to construct analytical models of matrix correction factors, 19 simulated waste matrices were fabricated,[12] and active and passive calibration standards were placed in known locations throughout the waste matrix drums. Both active and passive assay matrix response measurements were obtained as a function of location (radius, r, and height, z) of the standards. The resulting matrix response values varied smoothly as a function of r and z. These studies determined that the systematic effects are due only to gross neutron absorber and moderator amounts and are independent of the actual nature of the materials themselves. That is, a drum filled with Raschig rings (borated glass) produces the same responses as a drum filled with vermiculite mixed with an equally absorbing amount of borax.

Most of the observed distributions have been found to fit a power law as given in Equation (3):

$$y = A + Br^N \quad (3)$$

where A, B, and N are the fit parameters and r is the drum radius.

Volume-weighted average values were calculated using this equation, representing the most probable measurement result for either a totally uniform or a totally random distribution of source material within the matrix.

The matrix correction factor (MCFA) for an active assay measurement is a function of the AI and MI.

$$MCFA = MCFA(AI) \times MCFA(MI) \quad (4)$$

The MCFA values were fit to the power law (equation 3) as a function of their AI values for the 19 simulated waste matrices. The following set of equations describing the absorption portion of the active assay matrix correction factor were obtained for one PAN system:

$$MCFA(AI) = 1.00 \quad (5)$$

for the AI less than or equal to 2.272, and

$$MCFA(AI) = 0.54x(AI)^{0.612} \quad (6)$$

for the AI greater than 2.72.

The moderator portion, MCFA(MI), of the active assay matrix correction factor is obtained by dividing the total measured MCFA values by the calculated MCFA(AI) values obtained in equations (5) or (6).

The analytic representation of these data is thus of the form

$$MCFA(MI) = 1.00, \quad (7)$$

for the MI less than or equal to 0.40,

$$MCFA(MI) = 0.483\exp[1.817(MI)] \quad (8)$$

for the MI greater than 0.40.

The passive neutron matrix corrections are determined by systematic drum matrix measurements in a manner similar to the active measurements discussed previously. The passive matrix correction factors, MCFP, are a strong function of the MI.

The MCFP analytic fits to the four independent quantities measured during a passive assay scan are given below.

$$MCFP(\text{system totals}) = 1.00, \quad (9)$$

for the MI less than or equal to 0.355,

$$MCFP(\text{system totals}) = -0.16 + 3.28(MI), \quad (10)$$

for the MI greater than 0.355,

$$MCFP(\text{shielded totals}) = 1/[1 - MI], \quad (11)$$

$$MCFP(\text{system coincidence}) = [(0.5967)/(1 - MI) + 0.4187]^2, \quad (12)$$

$$MCFP(\text{shielded coincidence}) = [(0.8902)/(1 - MI) + 0.2337]^2. \quad (13)$$

The matrix correction equations given above or variations thereof are contained in the present PAN assay systems algorithms used throughout the DOE. Some sites perform additional matrix-dependent corrections to the assay results as discussed in Section 5.2.3.5.9.

Figure 5.2-3 shows the "Moderator Index" (see Caldwell, et al., 1986[12] for detailed discussion of the MI) obtained with mock matrix drums containing various hydrogen densities spanning the region of interest for general CH-TRU wastes. As can be seen, the MI varies smoothly with

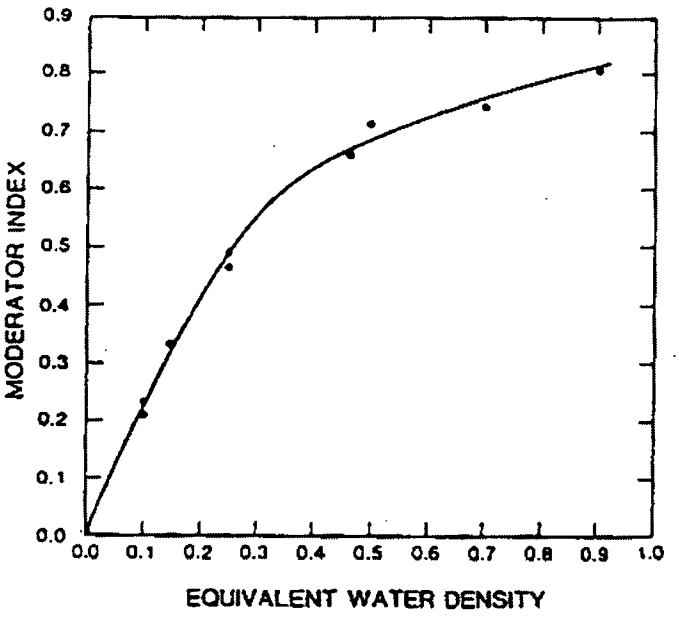


Figure 5.2-3—Moderator Index Measures for PAN Systems Using the Detector Ratio Method

average hydrogen density within a 208-L drum. Sludges display one of the higher average hydrogen densities of any CH-TRU waste form, with correspondingly high MIs (0.4 to 0.8). Lightly moderating matrices, such as combustibles, have MIs falling typically in the 0.1 to 0.3 region, and miscellaneous metals matrices, which generally contain no moderating materials, have measured MIs near 0.0.

Figure 5.2-4 shows the actual moderator correction factor (MCF) data[12] for the PNCC portion of the PAN systems as implemented at INEEL, Hanford, and SRS. The MCF value is the multiplicative factor required to normalize a given matrix measurement to the empty drum level of PNCC sensitivity. As can be seen, the MCF value varies smoothly as a function of the MI; Figure 5.2-4 can be used to estimate typical MCF values. For example,

(a) Miscellaneous metals, MCF = 1.0 (i.e., same sensitivity as with empty drum),

(b) Combustibles matrix, MCF = 1.35, and

(c) Sludges, MCF = 3.6.

The MCF range observed for a 3,000 CH-TRU drum sludge assay campaign at INEEL was 1.8 to 10.0.

When performing PNCC assays of highly moderating matrices, such as sludges, measurement of a MCF value is essential for accurate assay results to be obtained. A "calibration" based on a "typical" sludge drum would result in assay errors of hundreds of percent for some drums because of the large hydrogen density variations observed.

Figures 5.2-5a and 5.2-5b show plots of the systematic active assay correction factors. As can be seen in Figures 5.2-5a and 5.2-5b, some waste materials require no matrix correction (relative to a standard response measured in an empty drum). Examples of these waste matrices are cellulose-based combustibles, graphite molds and scarfings, aluminum scrap, dry-to-moderately-wet dirt, and silica.

5.2.3.5.4 Assay Algorithm and Data Acquisition System

All PAN units utilize a similar assay algorithm. At present, all drum- size units are equipped with IBM/PC-based data acquisition systems as described in Kuckertz, et al, 1987.[21] The system operating program (NEUT) controls all data acquisition and contains the assay algorithm.

Each data acquisition consists of sequential active and passive neutron assays, preceded by a user interactive initialization stage in which drum identification, content code information, drum weight, etc., can be entered from the PC keyboard, from a bar code reader, or from an RS-232 port by direct interaction with a site's data management computer. The weight of the drum's contents is used in calculating the nanocuries per gram (nCi/g) assay value that differentiates between TRU and non-TRU wastes. The content code input is used to flag difficult-to-assay

[21] T. H. Kuckertz et al., "Making Transuranics Assay Measurements Using Modern Controllers," Proceedings 9th ESARDA Symposium on Safeguards and Nuclear Material Management, London UK, pages 389-393, May 1987.

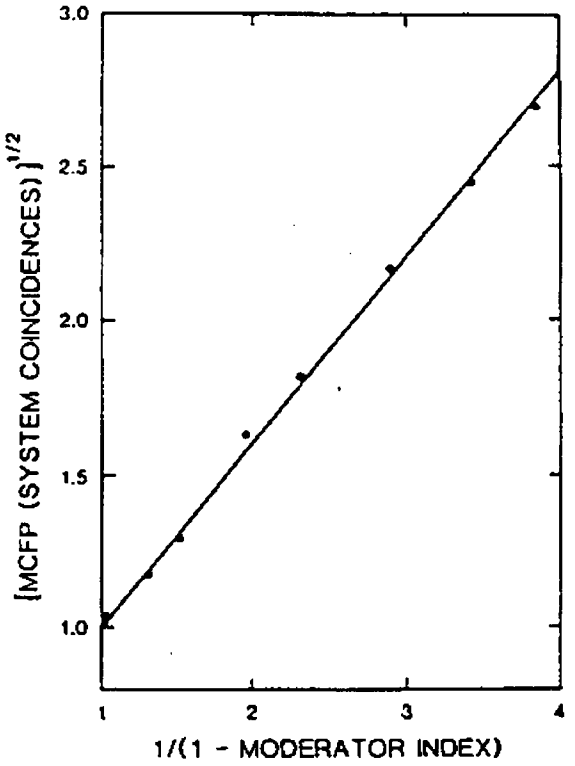
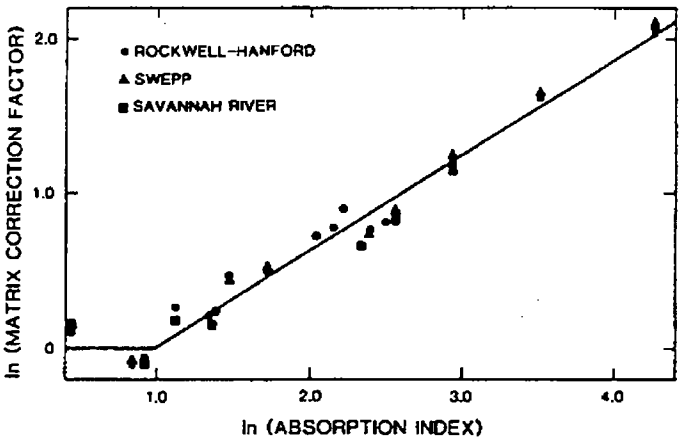


Figure 5.2-4—Use of the Moderator Index to Determine Passive Neutron Coincidence Matrix Correction Factors for a PAN System



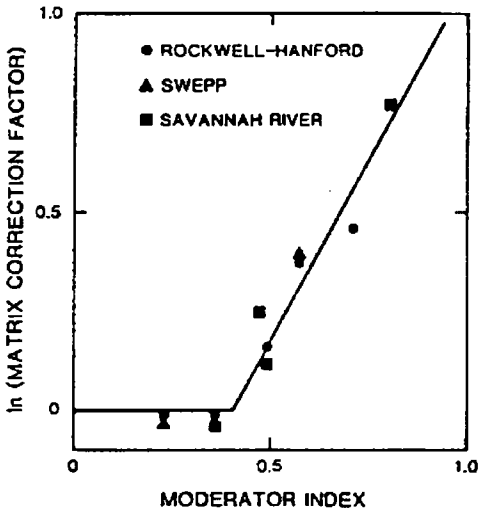


Figure 5.2-5—PAN System Active Assay Matrix Correction Factors Measured at Hanford, INEEL, and SRS with a Set of 20 Standard Matrix Drums

matrices or "special case" drums (see Section 5.2.3.5.11). All data input modes are in current use at the various sites. However obtained, that information becomes part of the permanent record stored with the TRU assay and matrix measurement data.

Modifications and upgrades have been performed at various times since the original algorithm was written in 1982. The development and upgrade of hardware and software has continued.[12,22,23,24] The software revisions can be readily accomplished within the Fortran software framework of NEUT.

Measured data as well as all initialization information, date and time stamp from the internal PC clock/calendar, and final analyzed results are archived. An on-line hard copy printout of the assay parameters and results is also generated. All background and calibration measurements are routinely recorded and archived in the same fashion as normal assays. Thus, a continuous and traceable record of all data is maintained. Most sites, in fact, are maintaining this complete set of data in interactive databases (e.g., Lotus 1,2,3 or dBASE III), wherein all single-run assay data occupies one row in a single spreadsheet or database record. In some cases, 10,000 such records exist at a site (e.g., INEEL). This data archiving technique is an extremely important development as such extensive waste drum assay databases have not been developed previously in the NDA field, much less put into such readily accessible form. This greatly facilitates internal consistency checks and comparisons of large numbers of individual drum assays results obtained with different assay techniques.

RG Pu (i.e., Pu containing high Pu-240 content) is accommodated within the same PAN algorithm as is used for WG Pu. The PAN operator is queried before each assay as to whether the drum contains WG Pu and, if not, what is the correct Pu-240 percentage for that waste drum. Once the Pu-240 percentage is entered, the algorithm automatically corrects both passive and active assays for the different Pu isotopics.

The basic Pu algorithm cannot, however, directly accommodate Pu-238 or HSG Pu. Those sites assaying HSG Pu waste exercise a special algorithm option in their Main Menus that allows for analysis of the basic passive and active data in terms of HSG Pu. If selected this option:

(a) Interprets the active assay results in terms of a Pu isotopic mix consisting of 18% Pu-239 and 82% Pu-238. Since only the Pu-239 fraction is fissile, total Pu mass is obtained by dividing the active assay result by the factor 0.18.

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[22] J. T. Caldwell, J.M. Bieri, and A.P. Colarusso, "The Los Alamos Second-Generation Passive-Active Neutron Assay System--FY86 Operations Record and System Evaluation", Los Alamos Technical Report LA-Q2TN-86-106, September 1986.

[23] A. P. Colarusso, et al., "Mobile Nondestructive Assay System," Proceedings of 28th Annual INMM Meeting, Newport Beach, Ca, July 12-15, 1987.

[24] K. L. Coop, J. T. Caldwell, and C. A. Goulding, "Assay of Fissile Materials Using a Combined Thermal/Epithermal Neutron Interrogation Technique," Third International Conference on Facility Operations-Safeguards Interface, San Diego, CA, November 29 - December 4, 1987.

(b) Interprets the passive assay data similarly. Pu-238 undergoes spontaneous fission at a rate of 2600 neutrons per second per gram (n/s/g) (for comparison, the Pu-240 rate is 990 n/s/g). Thus, passive coincidence counts can be used to obtain an estimate of Pu-238 mass.

(c) As in all cases the Systems Totals Passive Singles rate can be used, assuming oxide as the dominant chemical form, to estimate a total alpha particle emission rate. This estimate can then be used to calculate the Pu-238 mass.

The SRS possesses most of the DOE's Pu-238 waste, and is currently evaluating their Pu-238 algorithm.

5.2.3.5.5 Applicability to CH-TRU Wastes

The PAN systematic matrix correction factors discussed in Caldwell, et al., 1986[12], and Section 5.2.3.5.3, and now implemented in all drum-size PAN units enables the quantitative assay of virtually all DOE wastes presently packaged in 208-L drums. At present, these six implemented PAN units have been used to assay, collectively, about 20,000 CH-TRU waste drums at the various sites, including 2,000 drum assays performed with the mobile drum unit at the Nevada Test Site (NTS) and Lawrence Livermore National Laboratory (LLNL).

5.2.3.5.6 Instrument Calibration, Standards Preparation, and Implementation

Calibration of PAN units includes a thorough initial calibration after fabrication and routine calibrations using secondary standards.

Caldwell, et al., 1986[12], lists the standards used in all the present PAN units, for which all passive and active calibration standards have National Bureau of Standards (NBS)-traceable or NBS-referenceable origins. Absolute and matrix standards calibrations were conducted of the PAN unit. The PAN units were then each provided a set of secondary standards (placed in "Pink Drums" for conspicuous identification) consisting of standard, NBS-referenceable Cf-252 passive assay and U-235 active assay materials. A baseline reference data set for both passive and active assays was obtained for each PAN unit with these unique "Pink Drum" standards, and each unit has subsequently performed standard Pink Drum assays prior to each set of PAN waste drum assays.

A typical set of these standards measurements performed with the INEEL PAN unit and extending for almost a three-year period is shown in Figure 5.2-6. The individual passive and active standards measurements fall well within a +/- 5% window, with no measurable systematic drift during the three-year operational history. Caldwell, et al., 1986[12], lists the corresponding Pink Drum measurements for Hanford, SRS, and the mobile drum unit. All display the same basic stability of response.

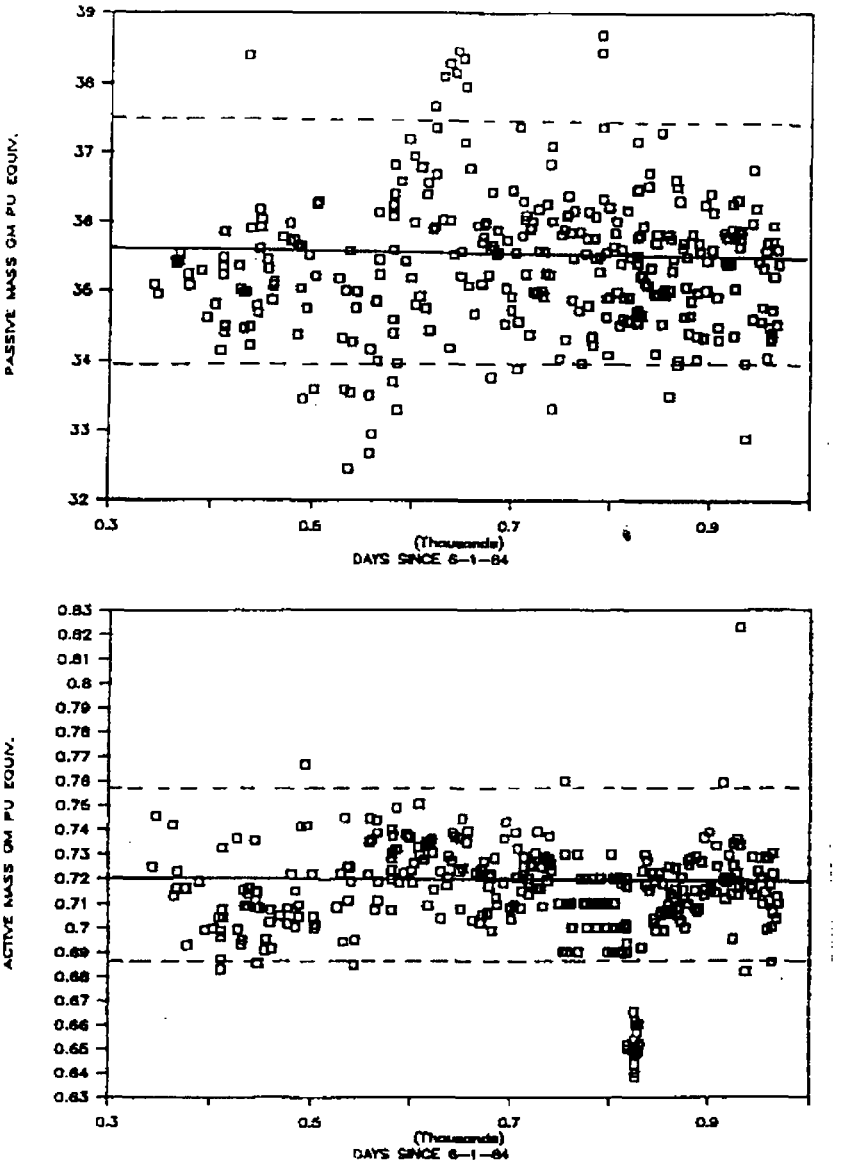

Figure 5.2-6—INEEL PAN Standards Measurements (Pink Drum) Performed Over a Three-Year Period.

(Top Graph Shows Passive Standard and Bottom Graph Shows Active Standard. Dashed Lines Indicate a +/-5% Measurement Error Band About Expected Standards Assay Values.)

5.2.3.5.7 Operator Training Requirements and Practices

The present generation of PAN units are highly-automated, computer-based systems. The instruments are computer-controlled using relatively interactive ("user-friendly") software. Only trained personnel are allowed to operate the assay equipment. Personnel are qualified according to DOE Order 5480.5.[15]

Standardized training requirements and guidelines for all DOE assay operators are based upon such already-existing industry standard training requirements, such as SNT-TC-1A. Each site provides a specialized training program for NDA instrument operators. The operators are directed and/or assisted by a designated site NDA expert. Expertise is attained by education and experience.

5.2.3.5.8 Assay Procedures

As of 1989, the PAN assay systems were comparatively recent additions (approximately six years) to NDA instrumentation and, as a consequence, ASTM and ANSI standards have not been developed for PAN assay systems. Active assay techniques have been used for approximately 18 years, but the 14-MeV thermalized neutron assay (active portion of PAN) is comparatively recent. Of course, the passive coincidence portion of PAN is similar to the PNCC assay technique and, therefore, PNCC ASTM, ANSI, and NRC standard practices and guidelines are followed for that portion of the PAN system.

All PAN standard operating procedures instruct operators to acquire a background and a "Pink Drum" data set before any assays on waste containers are performed. These data sets are checked for consistency and, if the results fall outside a predetermined (e.g., +/-10%) acceptance window, remedial action is taken. The remedial action can include a repetition of the background and/or standards measurements. If the second measurement is successful, general assays can resume. If the problem persists qualified personnel are contacted to "debug" the system. No CH-TRU waste drum assays can resume until the problem is satisfactorily resolved. If the background or standards measurement is outside the acceptance window, the diagnostic generally assumed is that a hardware problem exists.

The assay procedure for PAN units equipped with the IBM/PC data acquisition system is relatively straightforward. An operator inserts a waste drum into the PAN unit and enters all drum identification information via an interactive dialogue (PAN assay system software, NEUT, prompts the operator for the specific information). Once NEUT has checked the information for correct format, the assay record and programmable electronics hardware are properly indexed, gates set, etc., NEUT then sends a message to the operator (via the CRT screen) that the system is ready to begin an assay.

At this point the operator presses one button, the start sequence button on the MA165C neutron generator controller unit. This initiates the PAN active assay. At the conclusion of the active assay, NEUT automatically records all data and initiates the PAN passive assay. At the conclusion of the PAN passive assay, all data is recorded, analyzed and printed out for

immediate inspection. The operator is then informed (via the CRT Screen) that the system is ready to perform another assay.

5.2.3.5.9 Assay Precision, Bias, and Limit of Detection

The PAN assay algorithm contains a calculation of the measurement uncertainty[12] that combines statistical uncertainties and estimated systematic biases based on the measured matrix correction factor. For a generally heterogeneous matrix and TRU materials distribution, the larger the indicated matrix correction, the larger the expected assay uncertainty. These values are reported with the actual assay values, for both passive and active neutron assays. For many well-characterized waste streams a typical value for the estimated uncertainty (not including the statistical contribution to the error) is 20%.

When a systematic matrix correction formalism is used, the corresponding systematic uncertainty in the passive assay measurement can be decreased to 5% or less. This low an uncertainty is valid for dry, combustible, low-hydrogen content waste, such as general laboratory waste. The passive assay value uncertainty is calculated as for PNCC. The algorithm used in the passive coincidence portion of the PAN units calculates a composite assay uncertainty based on combining all the effects discussed above, which becomes part of the permanent archived assay record.

The active assay value uncertainty estimate includes a systematic bias contribution, which is a function of the MCF (AI and MI). For reasonably uniform TRU isotope distributions (such as are found in sludges), AI measurements indicate assay uncertainties of +/- 10%. For nonuniform TRU isotope distributions, the uncertainty is a function of the magnitude of the MCF.[12] That is, the larger the MCF, the larger is the associated assay uncertainty. The effects on the assay measurement of concentrated TRU activity in different drum locations have been calculated and plotted as a function of the total MCF. For example, a matrix correction factor of approximately five yields a corresponding uncertainty of 50% in the assay measurement.

Extensive comparisons have been performed for passive and active neutron assays of the same drum, for a great variety of matrix types (e.g., four types of sludges, job-control wastes, combustibles, graphite scarfings, miscellaneous metals, tantalum crucibles, glassware, molten salts, filter media, dirt, and others.) Some of these comparisons are shown in the figures of this document and are discussed in Fleissner, et al., 1986, and Schultz and Caldwell, 1988.[16,17] It should be noted that the matrix corrections applied to passive and active assays for a given type of matrix (except where no matrix corrections are necessary) are quite distinct. Thus, there is a very low probability of obtaining agreement by accident between active and passive neutron assays for wastes with significant moderator and absorber amounts. If one obtains agreement, both independent PAN assay techniques are considered to yield unbiased assay measurement values.

The assay limit of detection for the active neutron portion of the PAN unit can be as low as a few milligrams of Pu-239 placed anywhere within a typical 208-L waste drum.

5.2.3.5.10PAN Assay Results Comparisons

Comparisons of PAN assay results with SGS or radiochemical assay methods have been performed. The PAN assays, both passive and active, have been compared with SGS and radiochemistry assay results for (a) matrices requiring little or no matrix corrections, such as graphite molds and general laboratory wastes, and (b) homogeneous matrices (e.g., sludges).

Two papers[13,17] detail several comparisons of PAN and SGS assay measurements. Caldwell, et al.[13] includes a total database of some 5,000 assays performed at NTS, LLNL, Hanford, INEEL, LANL, and SRS. The drum assay and matrix correction formalism presented in Caldwell et al., 1986[12] was extensively evaluated for all types of waste matrices and waste content codes being generated within the DOE complex. Schultz and Caldwell 1988[17] encompasses an even larger database, but is confined to INEEL PAN assays and comparisons with RFETS SGS assays.

Figures 5.2-7a, 5.2-7b, and 5.2-7c show a recent PAN/SGS comparison performed by LANL personnel[11] using a mobile PAN drum-sized unit. The data was acquired at LLNL from assays of a set of some 200, 208-L, WG Pu waste drums consisting of general laboratory wastes (e.g., glassware, cellulosics, plastics, etc.) that had been assayed using the LLNL SGS unit. Figures 5.2-7a, 5.2-7b, and 5.2-7c show the PAN passive neutron, PAN active neutron, and SGS assay measurements comparisons. A statistical analysis of this data set indicates systematic agreement between both PAN neutron data sets and the SGS assay results at the 5% level (95% confidence level). Figure 5.2-8 shows a plot of similar waste stream assays performed with the Hanford PAN system, comparing passive and active neutron assay values for a set of 300 waste drums.

Figures 5.2-9a and 5.2-9b show a set of over 300 "graphite molds" matrix waste drums (RG Pu) assayed with the PAN unit at INEEL and also with an SGS unit located at the RFETS. A statistical analysis of this data set indicates systematic agreement of all three independent assay methods to within 10% on the average, at the 95% confidence level.

Quantitative comparisons between radiochemical Pu (WG) and Am-241 determinations and active PAN have been performed at the INEEL facility (SWEPP).[13] These comparison studies of approximately 1,300 drums of RFETS aqueous sludges comprise more than 100 individual sludge batches. These sludges contain low Pu and relatively high Am concentrations. The results of these comparisons are shown in Figure 5.2-10.

The batch-average drum Pu assay as determined at RFETS was compared to the same quantity as measured with the INEEL PAN unit. The indicated straight line shows the relationship RFETS Pu Mass = PAN Pu Mass. A statistical analysis of this same data indicates a best straight-line fit relationship of

$$\text{RFETS Pu Mass} = -0.06 + 0.85 * \text{PAN Pu Mass},$$

with a correlation factor of 0.51. The calculated correlation factor indicates that 51% of the variance does fit a straight line. The 0.85 constant indicates an approximately 15% average measurement bias between the two assay techniques.

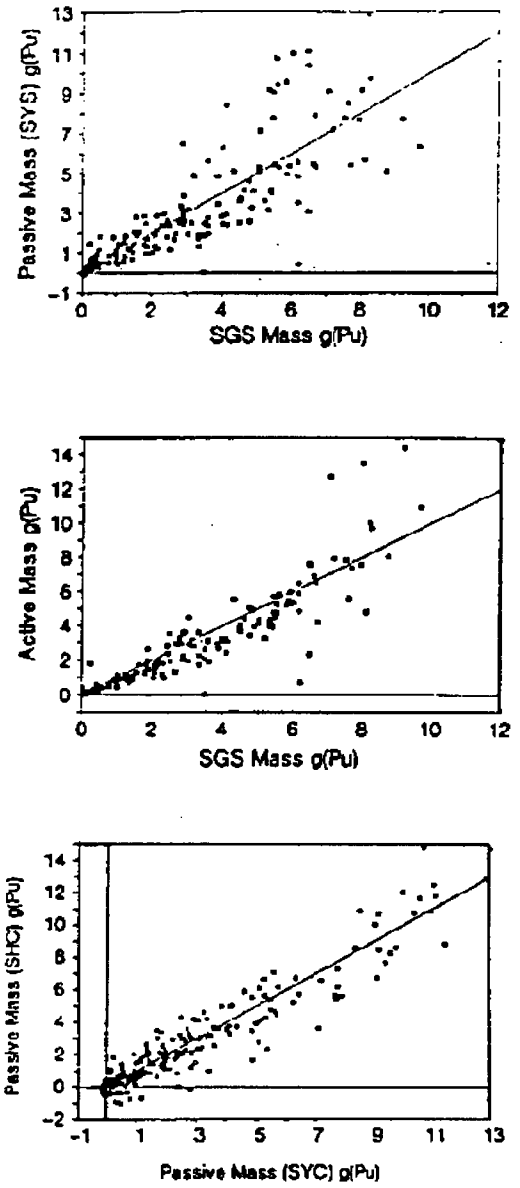

Figure 5.2-7—Comparison of Assay Data Sets of 200 LLNL CH-TRU Waste Drums.
Waste Matrix in Non-Segregated General Laboratory Waste.
Top Graph Shows Passive Neutron (PAN) Compared to SGS. Middle Graph Shows Active Neutron (PAN) Compared to SGS. Bottom Graph Shows Comparison of the Two Independent PAN Passive Neutron Assay Systems. Straight Line (X=Y) Depicts Ideal Case, Where Assay Techniques Shown Yield Identical Assay Measurement Results.

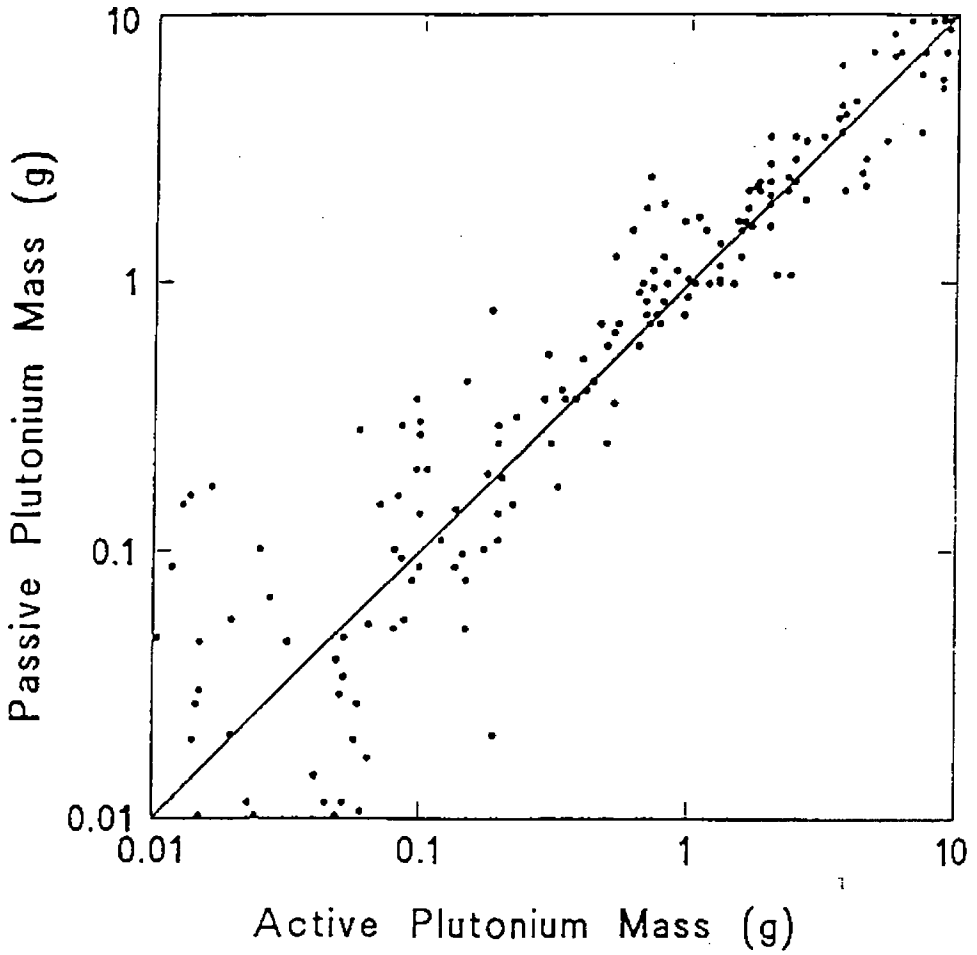

Figure 5.2-8—Comparison of 300 Hanford CH-TRU Waste Drum Assays Performed with the PAN System, Passive Neutron Compared to Active Neutron.

Straight Line (X=Y) Depicts Ideal Case, Where Assay Techniques Shown Yield Identical Assay Measurement Results.

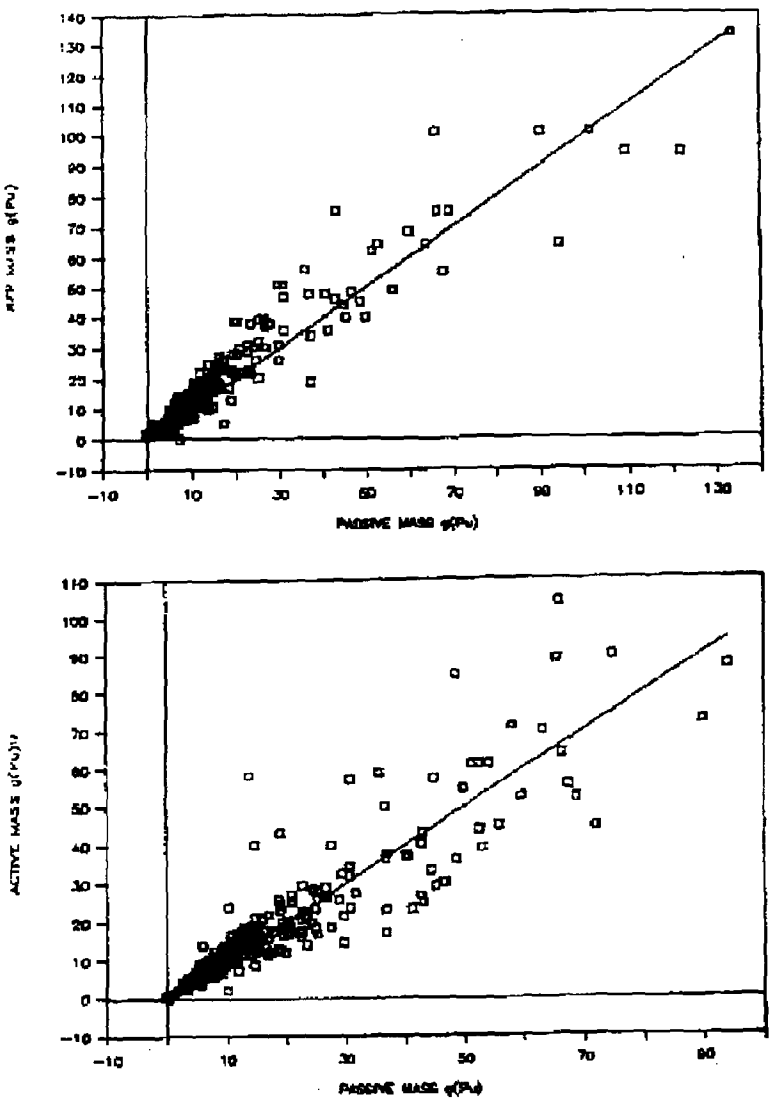

Figure 5.2-9—Assay Comparisons of a Set of 300 RFETS CH-TRU Waste Drums (Graphite Molds Matrix) Showing RFETS SGS System Assays Compared to INEEL PAN Unit Passive Neutron Assays (Top) and INEEL PAN Active Neutron and Passive Neutron Comparisons (Bottom).

Straight Lines (X=Y) Depict Ideal Case, Where Assay Techniques Shown Yield Identical Assay Measurement Results.

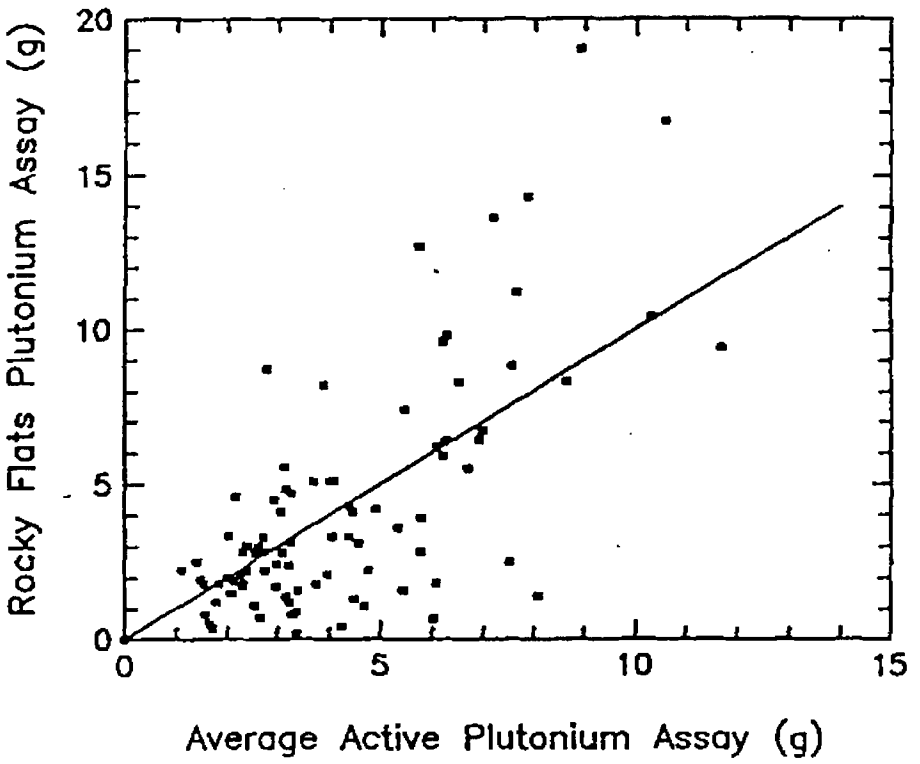

Figure 5.2-10—Batch Average Pu Assays of 1300 Sludge Drums Performed at RFETS Compared to PAN Assays of the Same Drums Done at INEEL.

Straight Line is a Linear Least Squares Fit to Data.

The individual PAN systematic measurement error (discussed in detail in Caldwell, et al., 1986[12]) for a typical sludge drum measurement is approximately 10% on average, due primarily to possible systematic errors in the matrix correction formalism. That error estimate is based on the observed standard deviation found for mockup sludge drum calibrations. PAN systems are able to measure the total uncorrelated neutrons, but cannot measure the individual contributions from mixed, uncorrelated sources of neutrons [e.g., (alpha, n) reactions due to Cm-244 and Am-241]. The contributions from Am-241, for example, can be calculated (obtained from passive single neutron rate) if no other (alpha, n) sources are present in the waste.

The two assay measurement techniques appear to agree within probable systematic errors associated with each technique (assuming a +/- 10% systematic error for both techniques). Other similar individual sludge drum NDA comparisons (passive and active PAN neutron assays of high-Pu, low-Am activity sludge drums[13] verify the basic PAN matrix corrections at the +/-10% level.

Schultz and Caldwell, 1988[17] details additional such comparisons for a great variety of matrix types, including heterogeneous matrices and highly neutron-absorbing matrices. In all cases the PAN assays are highly correlated with SGS assays and with each other. Comparison studies reported in Schultz and Caldwell, 1988[17] indicate a slightly better agreement between PAN and radiochemistry assay methods than the 10-15% discussed above.

5.2.3.5.11 Choice of Passive or Active Assay Value

Two assay values, a "passive" mass and an "active" mass, are obtained with each PAN assay. A choice of the values to be reported as the PAN assay value is performed within the assay algorithm by an analytical evaluation of assay conditions. Content-code-specific algorithm options are developed by a site based upon an evaluation by the assay data reviewer or expert of the individual site's waste content codes.

When specific content code or matrix information is available that indicates, for example, that passive assay results are more reliable than active assay results for that content code, then the algorithm selects the passive assay results when that content code is entered via the PAN operating software, NEUT. Similar overrides involve the statistical accuracy of a measurement. For example, if the passive measurement has a large error associated with it, then the active measurement is selected.

The default PAN assay algorithm is known to underestimate the absorption correction factor for various waste streams (e.g., tantalum crucibles). This phenomenon is due to neutron tunneling effects caused by the stacked arrangement of the crucibles. A correction factor based on the RFETS SGS assay value is used to adjust the PAN assay values.

A similar method is used at INEEL to modify the MI obtained during the assay of sludge drums. For sludge drums having low Pu content, an average MI is used in the assay value calculations. The average MI value used was obtained from previous assays of the same waste stream containing higher Pu loadings and, thus, higher count rates and improved counting statistics.

One algorithm option that is used by Hanford is based on experimental results indicating that fissile self-absorbing effects in several of their content codes were small for Pu mass loadings of 10 g or less, but increasingly probable for successively higher Pu mass loadings. It also took into account the experimental fact that Pu loadings in excess of 10 g led to statistically more precise passive assay measurements. Of course, the exact "cross-over" point between using the PAN active or passive assay measurement depends on several experimental factors such as waste form, (alpha, n) source strength, isotopics, etc.

INEEL has modified their PAN algorithm option to reflect the phenomenon that sufficiently strong (alpha, n) sources cause experimental measurement problems for the PAN passive assay measurement, rending it biased high. If the passive assay error bar exceeds 40 g Pu, then the option selects the self-absorption corrected active assay value. This algorithm option also reflects the development of a fissile self-absorption model that is applied to all PAN active assays. This model is purely empirical and assumes that the probability of fissile self-absorption is a monotonically increasing function of the total fissile loading in a waste drum. The function is determined from evaluations of large sets of actual CH-TRU waste drum assays employing PAN active and passive neutron and SGS measurements.[12] The derived functions, however, produce large error bars at the higher fissile mass values. In the "near 200 g Pu" regime, for example, the self-absorption corrected active assays may have relative errors of 40%.

PAN assay results may also be evaluated by the site NDA expert when special-case waste drums are encountered. The various factors that determine what is a "special case" drum include nominal assay values for CH-TRU waste drums near or in excess of the 200 g criticality limit and a lack of tag isotopics information (e.g., Pu-240 content). Both of these situations preclude proper interpretation of PAN assay measurement data.

Approximately one out of 500 RFP CH-TRU waste drums assayed with the INEEL PAN system are assigned an assay value near or in excess of the 200 g criticality limit. Typically, these drums contain very high (alpha, n) radioisotopic sources, which due to the high numbers of neutrons present, results in poor passive PAN coincidence measurements. The large numbers of neutrons (approximately 10^4 neutrons per second or greater) decrease the signal to noise ratio to below acceptable limits. Content codes 409 and 411 display this characteristic and also contain "lumps" of Pu that invalidates the active neutron assay. A panel of 3 or 4 INEEL experts examines and compares all assay and RTR information available, including a critical evaluation of the "tag" assay values. After a consensus of the expert panel is achieved a suitable resolution is proposed and the appropriate action taken. This action may include acceptance of the less than 200 g tag value or the waste drum is returned to the waste generator for repackaging. Both of these actions have occurred.

Hanford waste drum tag values obtained by using an SGS and the certification or check of the SGS assays with a PAN unit indicate assay values near or in excess of the 200 g criticality limit. The approach used at Hanford is basically the same one employed at INEEL, but a two-expert panel is used to evaluate the available data. The return-to-generator option is used frequently with drums having assay values near the 200 g limit.

Invalid or unavailable isotopics information occurred when a small set of Pacific Nuclear Laboratory (PNL) waste drums were assayed at Hanford using a PAN unit. These drums contained Cm-244, but this isotope was not originally listed on the accompanying data sheets. The PAN assay results indicated very high passive coincidence assay values. Many of the assay values were well in excess of the nominal 200 g limit. Experts resolved the problem through direct dialogues with the waste generator who, subsequently, agreed to provide the proper isotopic information that would then allow these drums to be properly assayed.

5.2.4 Fissile Material Content and Decay Heat Value Calculations

The fissile or fissionable isotope content for CH-TRU waste containers is expressed in terms of Pu-239 fissile gram equivalents (FGE), as defined in ANSI/ANS-8.1[25] and ANSI/ANS-8.15.[26] The standards list the maximum subcritical mass limits for fissile and non-fissile actinide nuclides. The fissile material (e.g., Pu-239, U-235, etc.) or fissionable content of a CH-TRU waste container may be obtained by using any of the previously described assay techniques. However, to obtain the number of Pu-239 FGE present, the isotopic composition contained in the waste form must be known.

The PNCC assay method detects the coincident neutrons emitted by the even-number TRU isotopes (e.g., Pu-240 and Cm-244). (See Section 5.2.3.4 for a detailed discussion of the PNCC assay technique.) Once the coincident neutron (spontaneous fission) emitters have been quantified and the proper correction factors applied (e.g., self-multiplication, system dead-time, etc.), one can calculate the fissile material content by applying the known isotopic ratios.

After the mass of each TRU isotope present has been determined, the decay heat can be calculated by multiplying the mass of each isotope by the decay heat per gram of the isotope. For the general case of alpha and beta decay, the decay heat per gram can be calculated by using Eq. 10 in ANSI-N15.22.[8,20]

The original PAN system algorithm used a very conservative means to estimate a waste drum's decay heat. First, it was assumed that only the drum's alpha particle inventory was responsible for the waste drum's decay heating. Second, the conservative assumption was made that all neutrons detected were produced by (alpha, n) reactions within the drum's waste matrix. In the original algorithm this assumption led to the value of 98,000,000 MeV of alpha decay heat energy being associated with each neutron emitted. This estimate is conservative for two reasons:

(1) In all cases of WG Pu, a fraction of the neutrons detected are produced by Pu-240 or other spontaneous fission reactions, with a much lower decay-heat-per-neutron factor. Typically each spontaneous fission neutron is associated with approximately 100 MeV of decay heat energy.

[25] ANSI/ANS-8.1, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."

[26] ANSI/ANS-8.15, "Nuclear Criticality Control of Special Actinide Elements."

(2) In many waste packages the actual (alpha, n) production factor is higher than the conservative value (2 n/s/mCi-alpha) used in the PAN algorithm. Production factors as high as 14 n/s/mCi-alpha were observed for RFETS CH-TRU waste drums when assayed with the INEEL PAN system.[27] For the various fluoride, magnesium (Mg), aluminum (Al) salt, and typical sludge matrices values as high as 100 n/s/mCi-alpha are observed.

To obtain more realistic estimates of the decay heat value for sludge wastes and those containing fluorides, INEEL has used an experimentally determined (alpha, n) production factor. For example, an experimentally determined production factor is being used for RFETS aqueous sludges.[26] This derived production factor has been incorporated into the PAN assay algorithm.

Hanford and SRS use an (alpha, n) production factor based on given Pu isotopics contained in the waste drum. The modified PAN assay algorithm assumes that the entire 241 mass (i.e., sum of Pu-241 and Am-241) is in the form of Am-241. This assumption is very conservative since Am-241 produces approximately 35 times the decay heating associated with Pu-241.

ORNL uses a PAN decay heat calculation algorithm based upon the subtraction of the spontaneous fission neutron portion (passive neutron coincidence) from the observed total neutron count rate.

Error bars associated with the decay heat calculation propagate in the same fashion as that described for PAN Pu mass calculation (see Section 5.2.3.5.9).

The administrative classification of 200 millirem per hour (mrem/h) for container surface dose rate imposed for CH-TRU waste automatically limits the decay heat contributions from beta- and gamma-emitting radioisotopes.

To calculate the fission product inventories required to generate a surface dose rate of 200 mr/h one can assume the total external surface dose rate is produced by fission products (the beta-and gamma-emitters). Any TRU radioisotopes present are conservatively assumed not to contribute to the observed external dose rate. One also assumes that the short-lived fission products have decayed sufficiently to be of no concern. This is a conservative assumption since most waste drums are more than one year old. Consequently, only a small number of "pure" beta-emitters would then be present (e.g., strontium [Sr]-90). The remaining predominant radioisotopes producing the heat-generating radiation other than Sr-90 would then be cesium (Cs)-137 and cobalt (Co)-60.

Consider a CH-TRU waste drum containing 100 kilograms (kg) of medium atomic number wastes. Assume the maximum allowable container external dose rate of 200 mrem/h and that this rate is attributable solely to beta- and gamma-emitters (conservative assumption). Also, assume that the radioisotope inventory of this waste drum is a mixture of the dominant long-lived fission product species Sr-90 and Cs-137. For long-term decay 6.5% Cs-137 and 2.2% Sr-

[27] C. E. Moss and J. T. Caldwell, "Assay of TRU Wastes Containing (alpha, n) Sources," LA UR 86-2220, June 22, 1986.

90 are produced by reactor fissions.[28] This fission product mixture produces a decay heating of 0.91 MeV, 50% derived from beta emission and 50% derived from gamma-ray emission.[29]

If one also makes the following conservative assumptions: (1) that only the Cs-137 0.67 MeV gamma ray exits the waste drum, (2) the Cs-137 is located in the center of the waste drum, and (3) the mass attenuation factor is very conservative (i.e., 0.74 cm^2/g), one can calculate the curies required to produce an container external dose rate of 200 mrem/h. These assumptions and calculations yield an estimate of 0.25 Ci of Cs-137 present in the example waste drum. Using the isotopic ratio discussed earlier, one calculates a beta and gamma decay heating of 0.001 watt. This decay heating value is equivalent to that produced by the alpha emissions of 0.4 g WG Pu.

Detailed gamma-ray spectral studies of ORNL and RFETS CH-TRU waste drums[19,26,30] indicate that the detected gamma rays were attributable to TRU isotopes and not to fission products. Consequently, from the calculations above and the small quantities of observed fission products the contributions to the waste drum decay heat from beta- and gamma-emitters is negligible.

5.2.5 New Assay Developments

In keeping with the DOE's general policy of upgrading NDA hardware, software and procedures, the DOE continues to support a vigorous NDA development program, with specific support of TRU waste assay-related instrumentation. One recent example of this development is a significantly improved digital processing unit (upgraded shift register) for use with PNCC units that greatly reduces basic dead-time limitations associated with older PNCC units. The improvement considerably lowers processing errors. These newer processors are now either installed or in the process of being installed in all DOE facilities using PNCC assay techniques.

Another development nearing the implementation phase in support of the PAN assay method is "neutron imaging" of 208 L waste drums.[31] With very little change in the basic PAN hardware, 208 L waste drum assay data may be acquired in a fashion that allows it to be processed with existing imaging software similar to that used in medical CAT scans. Neutron imaging has already been demonstrated using an upgraded DOE PAN unit,[30] the Mobile Drum System. Improved matrix and self-absorption corrections to the basic assay data will then be possible using the neutron imaging technique. Accurate determination of the TRU material distribution should result in considerably improved assay accuracies since it is known that incomplete matrix and self-absorption corrections are a major source of assay errors.

PAN box assay systems have been used in the DOE complex for several years. The earlier versions are discussed in several of the references[13,21,22] that include assay campaigns at NTS and RFETS. Recently, some improvements in the box PAN systems have been made and

[28] "Chart of the Nuclides," Knolls Atomic Power Laboratory.

[29] David C. Kocher, "Radioactive Decay Data Tables," DOE/TIC-11026, 1981.

[30] F. J. Schultz, et al., "Neutron and Gamma-Ray Nondestructive Examination of Contact-Handled Transuranic Waste at the ORNL TRU Waste Drum Assay Facility," ORNL-6103, March 1985.

[31] San Horton, "Neutron Imaging," U. S. Army, Ph.D. thesis, 1988.

implemented at some sites. The new box assay unit at INEEL and the Mobile System box assay unit are examples of this implemented PAN technology. The improvements have been in four areas:

(a) Improved matrix corrections made possible by development and deployment of a "moving" shielded flux monitor that samples the emergent interrogating flux along the entire length of a box.

(b) Improved detection uniformity throughout the box volume made possible by a better detector layout.

(c) Smaller matrix inhomogeneity assay errors, achieved by implementation of a source location algorithm, which effects matrix corrections correlated with the location of the TRU material is based on crude neutron imaging, and

(d) Implementation of IBM/PC data acquisition/controller hardware and software.

Improvement (a) is similar to the shielded flux monitor described in Section 5.2.3.5 that is used in the drum-size PAN. These new developments are being implemented simultaneously with the institution of the standard waste box.

5.2.6 QA and QC Practices

QA and QC are important functions at all DOE facilities, especially in special nuclear materials (SNM) accounting areas. The 18 elements of NQA-1 are being implemented throughout the DOE.[15] TRU waste assay QA and QC practices adhere to the guidelines established in NQA-1. Some of these practices have been discussed previously in relationship to other subjects. A brief listing of some of these QA/QC practices in common use to assay newly generated and stored CH-TRU waste throughout the DOE complex are:

(1) Expert NDA personnel check of NDA records before assay values assigned to the waste package (NQA-1 Elements 1. Organization and 10. Inspection). (The functions of these site NDA reviewers or experts have been discussed previously in Sections 5.2.3.2, 5.2.3.4, and 5.2.3.5),

(2) Assay standards used before and after waste assays (NQA-1 Element 12. Control of Measuring and Test Equipment). Detailed discussions of the instrument calibration and calibrations standards preparation and implementation have been discussed in detail for each assay method in Sections 5.2.3.2.1.3, 5.2.3.4.3, and 5.2.3.5.6),

(3) Automatic electronic system gain stabilization. (Element 3. Design Control). For each assay method manufacturer's system instructions and site operating procedures suggest to the assay system operators the correct settings for proper system gain and stabilization,

(4) Automatic software flagging and tagging for special-case waste containers (NQA-1 Element 5. Instructions, Procedures, and Drawings). For example, less than 0.5%

transmission segments are flagged for SGS assay measurements and special-case waste matrices are flagged by the computer algorithm in PAN assay measurements,

(5) Administrative tagging of failed segment assays as "LOWER LIMITS" (NQA-1 Element 15. Control of Nonconforming Items). This procedure is performed for those SGS waste drums that qualify.

(6) Original segment assay and calibration data archived for future reference (NQA-1 Elements 6. Document Control and 17. Quality Assurance Records). PAN assay algorithm automatically archives all calibration data acquired (see Section 5.2.3.5.6),

(7) Detailed data sheets are prepared for each drum and accompany the drum through all NDA, RTR, etc. stations. Some sites (e.g., INEEL) use automated, computerized data sheets and bar codes (NQA-1 Element 9. Control of Processes),

(8) Most sites' SGS assays verified by one or two other independent NDA measurements (at the certification facility). For example, RFETS SGS assays are verified by the INEEL PAN passive and active assay measurements (NQA-1 Elements 9. Control of Processes and 18. Audits),

(9) Nonconforming drums returned for repackaging (NQA-1 Element 15. Control of Nonconforming Items). When a drum is found to contain non-conforming items after RTR inspection or to contain, for example, greater than the acceptable nuclear criticality limit, it is returned to the waste generator for repackaging,

(10) RTR inspection is used in appraisal of NDA "Special-Case" waste containers to aid in the evaluation of matrix problems (NQA-1 Element 16. Corrective Action). For example, special-case waste drums (e.g., INEEL. tantalum crucible content code) are flagged by the PAN assay algorithm through previous evaluation and identification of the content code by the site NDA reviewer (expert). See Section 5.2.3.5.11 for a more detailed discussion of the special-case waste drums,

(11) RTR inspection is used routinely to verify or evaluate waste form or content code of all waste drums (NQA-1 Element 8. Identification and Control of Items). RTR inspections are used to confirm the proper identification of waste content codes.

(12) General practice of upgrading NDA hardware, software, and procedures whenever available and fiscally possible is pursued (NQA-1 Elements 2. Quality Assurance Program and 3. Design Control). See Section 5.2.5 for a discussion of new NDA developments.

It is DOE policy to conduct periodic audits of all Waste Isolation Pilot Plant certification activities at each site.[32] The audit teams consist of technical NDA (and nondestructive

[32] WIPP-DOE 120, Quality Assurance.

examination [NDE]) and administrative personnel knowledgeable in several of the NDA technologies discussed in Section 5.2.3. The purpose of these audits is to provide independent monitoring and evaluation of each site's NDA and NDE activities on a regular basis and to foster compliance with certification plans. These in-depth evaluations take place on site, and a detailed audit team report, including recommendations for improvements in areas judged deficient, follow each such audit. In addition, each site conducts independent internal audits on at least an annual basis, covering the same overall procedures as performed in the DOE external audits. The effect of these audits is to provide considerable independent oversight of each DOE site's NDA and NDE operations, as overlays to each site's routine operations activities.