6 STRUCTURAL EVALUATION OF A MULTI-CANISTER OVERPACK

Multi-canister overpacks were developed as part of the Hanford Spent Nuclear Fuel Project. The multi-canister overpacks facilitate the removal, processing, and storage of deteriorating spent nuclear fuel currently stored in the Hanford site's K-East and K-West Basins. Their purpose was to contain, confine, and maintain spent nuclear fuel (Goldmann, 2000a). The multi-canister overpack is a vessel that has internal components that maintain structural integrity while providing criticality control and spent nuclear fuel drying capability (Goldmann, 2000a,b; Garvin, 2001). In the following sections, the specific design of the multi-canister overpack, associated finite element analysis, and comparison with full-scale tests are discussed.

6.1 The Multi-Canister Overpack

The multi-canister overpack is a 304L stainless steel cylindrical vessel approximately 610 mm [24 in] in diameter and 4,216 mm [166 in] long, and the shell is fabricated from 12.7-mm [0.5-in]-thick stainless steel material. The bottom is forged from stainless steel with a diameter of 610 mm [24 in] and a thickness of 51 mm [2 in] {except at the center, which is 28.7 mm [1.13 in] thick}; this allows water to flow to the central process tube for removal. The bottom is welded to the vessel body to provide a permanent sealed closure. The multi-canister overpack is loaded through its top end, which is covered by a shield plug with four access ports and a locking ring. A cover plate is then welded on the top to seal the multi-canister overpack (Figure 6-1). The multi-canister overpack vessel must satisfy American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Subsection NB (Garvin, 2001). The multi-canister overpack is designed so a total of five or six fuel baskets can be stacked inside. The spent fuel is placed in the baskets while underwater in the K-Basins. The baskets are filled with fuel and scrap pieces and then loaded into the multi-canister overpack. After the baskets are loaded into the multi-canister overpack, the shield plug with the central process tube is placed into the open end of the multi-canister overpack, shielding workers when the transfer cask, containing the multi-canister overpack, is lifted from the pool. The cask and the multi-canister overpacks are then taken to the dewatering and drying operations. During these operations, the fuel will not be removed from the protection of the multi-canister overpack. Covers for the process ports may be installed or removed as needed per operating procedures (Garvin, 2001; Goldmann, 2000a). The dry weight of each multi-canister overpack is 8,281 kg [18,260 lb] with six Mark 1A baskets and 9,106 kg [20,080 lb] with five Mark IV baskets (Snow, 2003; Garvin, 2001). Details of each basket configuration are shown in Figure 6-2.

6.2 Finite Element Analysis and Full-Scale Tests

Rains (1999) uses finite element analysis to study the scenario in which the multi-canister overpack concentrically drops from a multi-canister overpack handling machine into the shipping cask. The multi-canister overpack is dropped from a height of 2.5 m [8.2 ft], concentrically enters the cask, and drops an additional 4 m [12.83 ft] to the cask bottom. Because of the tight fit between the canister and the cask, an air cushion effect develops and the velocity of the multi-canister overpack decreases. In addition to this cushioning effect, it is also noted that the shipping cask itself is supported by an impact absorber. Hollenbeck and Tu (1999) consider eccentric drops of the multi-canister overpack in which it impacts the edge of the storage tube case. Material failure is calculated from a triaxiality factor (Garvin, 2001), which utilizes uniaxial tensile failure data to be generalized to multiaxial stress and strain states.

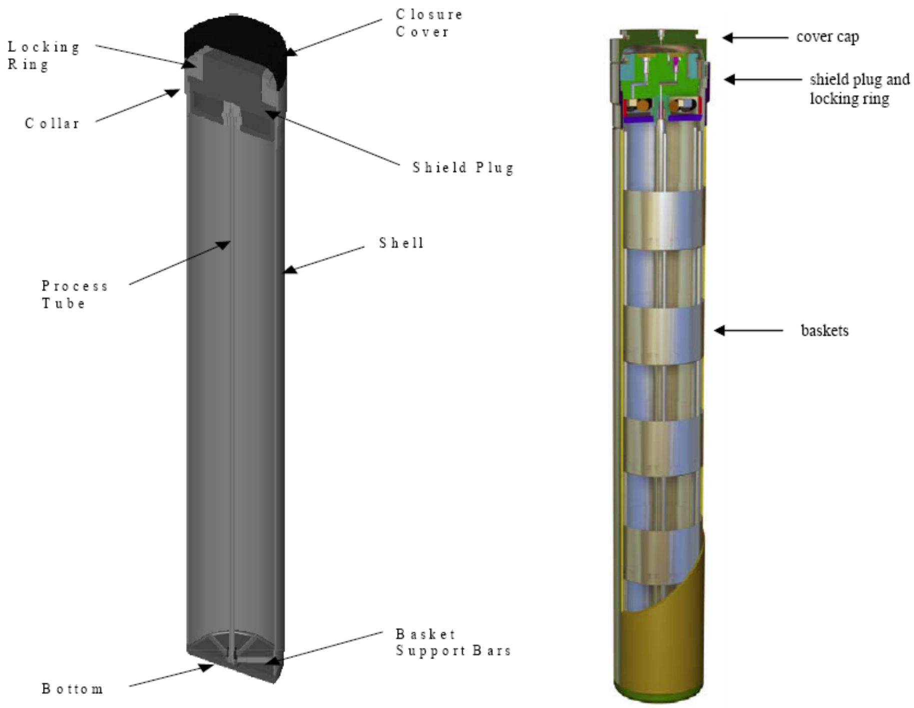


Figure 6-1. Typical Multi-Canister Overpack (Snow, et al., 2005, Used With Permission of the American Society of Mechanical Engineers, Copyright 2005; Morton, et al., 2006, Used With Permission of the American Nuclear Society, Copyright 2006)

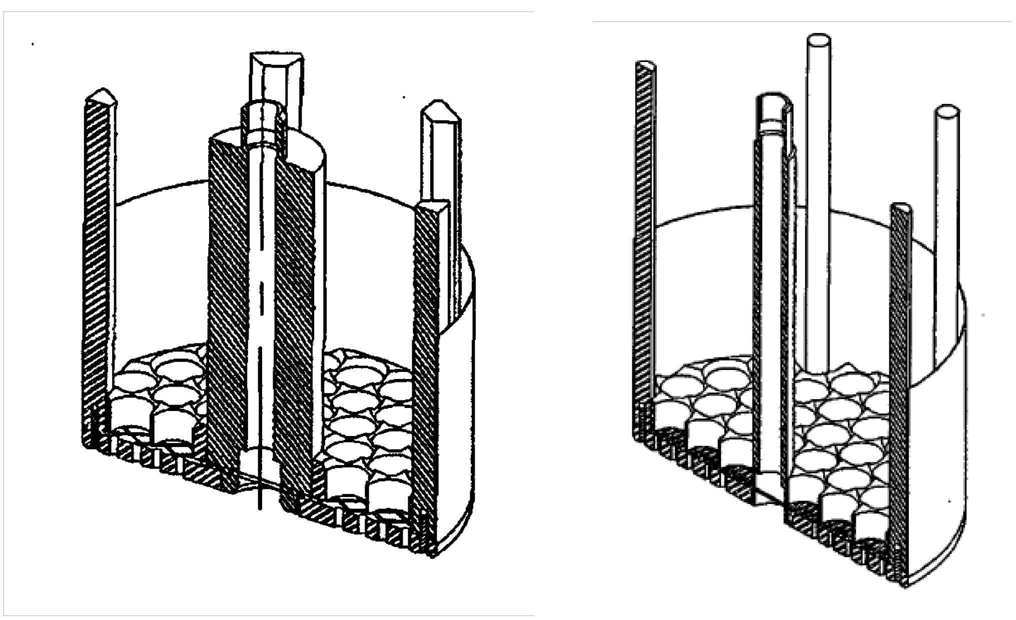


Figure 6-2. Mark 1A and Mark IV Multi-Canister Overpack Internal Basket Configurations (Garvin, 2001)

The concentric drop results presented by Rains (1999) show that the drop into the cask produces very large impact reactions on the multi-canister overpack and internals. Plastic strains are shown to occur in the bottom of the multi-canister overpack sides, the bottom two baskets' support plates, the outside support posts, and the center support post. These plastic strains are due to a large amount of energy that is absorbed by the multi-canister overpack internals during impact. The maximum equivalent plastic strain in the bottom of the multi-canister overpack is 2.0 percent. This is below the calculated effective failure strain of 15 percent, which is based on the multiaxial stress state (using the triaxiality factor) and strain rate (Rains, 1999). In the weld where the multi-canister overpack walls and bottom of the canister meet, equivalent plastic strains are approximately 2.1 percent, which is below the calculated effective failure strain of 12.5 percent for the weld material. Therefore, breaching of the multi-canister overpack containment boundary due to through-wall cracking is not expected (Rains, 1999).

For the case of an eccentric drop of the multi-canister overpack onto a standard storage tube (Hollenbeck and Tu, 1999), the maximum equivalent plastic strain in compression (negative triaxiality factor) is approximately 40 to 45 percent, which occurs on the bottom of the multi-canister overpack near the point of impact. Because compressive failure strain for the base metal is 80 percent, no through-wall cracking is predicted. The maximum predicted equivalent plastic strain at the multi-canister overpack lower weld is approximately 2 percent, which is below the weld metal failure strain of 10 percent. It is noted that the maximum tensile principal strain is in a radial direction, which would only lead to possible spalling or flaking of the outer surface rather than through-wall cracking (Hollenbeck and Tu, 1999).

Finite element analyses of 7-m [23-ft] and 0.61-m [2-ft] drop events, as specified in the Waste Acceptance System Requirements Document (DOE, 2002), are discussed in Snow (2004, 2003). The Waste Acceptance System Requirements Document only requires a vertical orientation for the 7-m [23-ft] drop event; however, Snow (2003) believed it would be prudent to also include orientations of 1° and 3° off vertical. The 0.61-m [2-ft] drop was analyzed for the most critical orientation as determined by previous analyses of the standardized canisters.

With regard to a failure criteria, Snow (2003) selected a value of 47 percent as the minimum elongation value considering all of the materials used in the canister (see Table 6-1) Similarly, the minimum fracture strain was taken as 118 percent. The use of these values by Snow (2003) should be considered a very conservative approach.

For the case of the 7-m [23-ft] drop, Table 6-2 shows the equivalent plastic strains for the drop orientations of vertical and 1° and 3° off vertical. For a vertical drop (0°), the equivalent plastic strain is 9 percent at the inside surface of the multi-canister overpack bottom, caused by the baskets bearing on the multi-canister overpack bottom, and the largest plastic strain in the other multi-canister overpack components is 4 percent or less. For the 3° off-vertical drop, the maximum plastic strain is 35 percent (Mark IV basket, outside surface of the bottom at the flange), showing that this orientation is the most critical; however, the maximum strain is still below the 47 percent minimum elongation. Therefore, the multi-canister overpack would maintain containment for all of the drop scenarios. It should be noted, however, that the Mark IV basket has strains approaching 100 percent. The Mark IV basket, which has a slender post as compared to the Mark 1A basket, undergoes significant deformation due to bending of the center post and perimeter bars and would result in significant damage to the fuel (Snow, 2003).

Table 6-1. Multi-Canister Overpack Component and Basket Materials*

Component	Yield Strength (σ_y, psi)	Ultimate Strength (σ_u, psi)	Elongation (%)	Area Reduction (R)	Average Yield Strength ($\sigma_{y\ ave}$, psi)	True Fracture Strain ($\epsilon_{p\ true}$)
Multi-Canister Overpack Components						
Main Shell SA–240/ SA–312 (304/304L)	41,388 46,067 44,702	88,145 91,401 89,164	59.2 60.9 61.8	70.1 69.4 70.8	44,052	1.207 1.184 1.231
Collar SA–182 (F304/F304L)	43,200 40,700 40,200	86000 82,000 86,000	56.0 60.0 54.0	77.0 79.0 76.0	41,366	1.470 1.561 1.427
Bottom SA–182 (F304/F304L)	35,800 35,100 35,600 40,100	80,800 80,400 80,500 82,900	56.0 63.0 58.0 57.0	75.0 75.0 75.0 73.0	36,650	1.386 1.386 1.386 1.309
Cover SA–182 (F304L)	40,500 47,500	84,500 89,000	62.0 55.0	78.0 75.0	44,000	1.514 1.386
Mark 1A Basket Components						
Base Plate SA–182 (F304/F304L)	44,800 42,800 39,800	86,500 86,000 80,500	57.0 58.0 59.0	78.0 77.0 78.0	42,467	1.514 1.470 1.514
Perimeter Bar SA–479 (304/304L)	64,940 63,440	95,730 96,100	47.1 47.6	75.1 77.4	64,190	1.390 1.487
Center Post SA–479 (304/304L)	41,200 44,000 46,200	89,500 88,500 88,000	57.0 55.0 54.0	75.0 58.0 77.0	43,800	1.386 1.514 1.470
Mark IV Basket Components						
Base Plate A240 (304/304L)	33,252 35,100 34,719	79,969 85,500 83,504	62.5 60.0 62.3	74.9 78.0 73.3	34,357	1.382 1.514 1.470
Perimeter Bar SA–479 (304/304L)	39,200 48,500 55,000 42,200	91,200 95,200 94,000 89,500	54.1 55.3 48.0 48.8	74.6 77.2 74.0 72.0	46,225	1.370 1.478 1.347 1.273
Center Post A511 (304/304L)	33,649 45,282	80,239 84,788	54.8 49.6	Not Available	39,465	Insufficient Data

*Snow, S.D. "Analytical Evaluation of the MCO for Repository—Defined and Other Related Drop Events." EDF–NSNF–029. Rev. 0. New York City, New York: ASME. 2003.

Table 6-2. Maximum Plastic Strains in the Multi-Canister Overpack for 7 m [23 ft] Vertical and Near-Vertical Drop Events*								
		Maximum Equivalent Plastic Strain (%)						
Multi-Canister Overpack Basket Configuration	Drop Angle From Vertical (°)	Bottom at Flange		Main Shell		Collar		Lower Basket
		Outside Surface	Inside Surface	Outside Surface	Inside Surface	Outside Surface	Inside Surface	Maximum Anywhere
Mark 1A	0	4	5†	1	3	1	2	15
	1	18	7	2	13	<0.1	0.1	14
	3	34	13	13	28	0	0	14
Mark IV	0	5	5‡	1	3	1	2	99
	1	20	4	5	14	0.1	0.1	95
	3	35	14	13	29	0	0	93

*Snow, S.D. "Analytical Evaluation of the MCO for Repository—Defined and Other Related Drop Events." EDF–NSNF–029. Rev. 0. New York City, New York: ASME. 2003.
†For this model, the maximum strain of 9% in the bottom occurred under a basket support bar and was due to bearing loads.
‡For this model, the maximum strain of 8% in the bottom occurred under a basket support bar and was due to bearing loads.

Table 6-3 shows the maximum equivalent plastic strain for a 0.61-m [2-ft] drop at orientations of 60°, 90°, and 115° off vertical. The 60° orientation was previously shown to be the most critical slapdown event in terms of maximum equivalent plastic strains in the containment boundary (Blandford, 2003). The multi-canister overpack with Mark IV baskets was also evaluated for a 115° off-vertical orientation. This orientation takes into account that the multi-canister overpack with five Mark IV fuel baskets is not symmetric with respect to its center, and therefore the effect of slapdown of the multi-canister overpack bottom needed to be addressed (Blandford, 2003).

For the case of 60° off vertical on the bottom at which impact occurred, the maximum equivalent plastic strain of both multi-canister overpack canisters was 22 percent for the Mark IV basket and 20 percent for the Mark 1A basket. This maximum plastic strain is compressive and is located at the bottom of the multi-canister overpack where it contacts the surface. All other plastic strains for the 60° orientation of both multi-canister overpack components are comparable. Note that the plastic strain in the baskets themselves is 14 percent and 8 percent for the Mark IV and Mark 1A baskets, respectively. The higher compressive strains for the Mark IV basket is due to the fact the center post is smaller than the Mark 1A basket center post. Comparing the plastic strains of the Mark IV multi-canister overpack for the 60° and 115° orientations, the maximum plastic strain (22 percent) corresponds to the case of 60° and is located at the canister bottom. However, notice that the multi-canister overpack top components (i.e., collar, cover) and the main shell have plastic strains larger than those of the 60° off-vertical drop orientation. This is consistent because the top head makes first contact with the impact surface. However, the 60-percent orientation has the point of first contact at the bottom of canister, which leads to larger plastic strains in the bottom flange and impact area.

Table 6-3. Maximum Plastic Strains in the Multi-Canister Overpack for 0.61 m [2 ft] Worst Orientation Drops*						
		Maximum Equivalent Plastic Strain (%)				
		Mark 1A		Mark IV		
Component†	Surface	60% Off-Vertical Drop	90° Off-Vertical Drop	60% Off-Vertical Drop	90° Off-Vertical Drop	115° Off-Vertical Drop
Flange Bottom	outside	11	6	11	7	6
	inside	6	4	7	4	5
Bottom	outside	4	1	4	<1	1
	inside	6	7	6	7	11
Impact Area‡	outside	20	10	22	10	15
Main Shell	outside	4	3	4	2	10
	inside	7	10	6	6	15
Collar	outside	9	4	6	4	3
	inside	12	6	11	6	20
Cover	outside	5	2	5	2	14
	inside	4	<1	4	<1	0
Baskets§	(max.)	8	1	14	12	20

*Snow, S.D. "Analytical Evaluation of the MCO for Repository—Defined and Other Related Drop Events." EDF–NSNF–029. Rev. 0. New York City, New York: ASME. 2003.
†Components not listed in this table experienced zero or low plastic strains.
‡This was the side of the multi-canister overpack bottom where the impact occurred, which resulted in significant compressive strains.
§Peak strains in the Mark 1A baskets occurred in the basket walls or the top of the center post. Peak strains in the Mark IV baskets occurred in the base of the center post.

Snow (2003) compares the 610-mm [24-in]-diameter standardized canister and the multi-canister overpack because both canisters have the same diameter and a nominal wall thickness of 12.7 mm [0.5 in]. The standardized canister is 4.6 m [15 ft] long, and the multi-canister overpack is approximately 4.3 m [14 ft] long. However, the standardized canister has an energy-absorbing skirt with flanged and dished heads, while the multi-canister overpack has a thick, flat bottom and is approximately twice as heavy as the standardized canister. Table 6-4 shows the maximum equivalent plastic strains for both canisters [data for the standardized canister is from Blandford (2003)]. Note that the plastic strains are comparable for the two drop events. If a comparison is made between the critical case of 7° off vertical for the standardized canister versus the 3° off-vertical multi-canister overpack full-scale tests, the multi-canister overpack has 34- or 35-percent strain (Table 6-2) in the containment boundary, while the standardized canister has less than 1-percent strain [i.e., 0.7 percent given in Table 5-1 (Blandford, 2003)]. This shows once again that the skirt of the standardized canister does absorb a significant amount of energy and protects the containment boundary. Nevertheless, the predicted plastic strains in the multi-canister

Table 6-4. Maximum Strain Comparison—Multi-Canister Overpack Versus 610 mm [24 in] Standardized U.S. Department of Energy Spent Nuclear Fuel Canister*					
	Maximum Strain in 610-mm [24-in] Canister Containment Boundary			**Maximum Strain in Multi-Canister Overpack Containment Boundary**	
Drop Event	**Outside**	**Middle**	**Inside**	**Outside**	**Inside**
23-Foot Vertical Drop	6	0.6	4	5	5
2-Foot Worst Orientation Drop	23	15	16	22	11
*Snow, S.D. "Analytical Evaluation of the MCO for Repository—Defined and Other Related Drop Events." EDF–NSNF–029. Rev. 0. New York City, New York: ASME. 2003.					

overpack are below the minimum elongation strain of 47 percent and minimum fracture strain of 118 percent.

Similar finite element analyses investigate the puncture resistance of the multi-canister overpack subject to a 0.61- or 1-m [24- or 40-in] drop onto a 152-mm [6-in]-diameter rigid post (Snow, 2004). The multi-canister overpack with a Mark IV basket was chosen for the study because it has the largest design weight {9,107 kg [20,080 lb]}, and the Mark IV basket, with its smaller center support post and perimeter bars, has less stiffness to prevent puncture of the main shell. The performance of this multi-canister overpack should envelope (bound) all other multi-canister overpack canister configurations (Snow, 2004).

Two finite element models were analyzed: (i) the rigid post makes contact with the multi-canister overpack at a point halfway between two basket base plates (which will be referred to as center post) and (ii) the rigid post makes contact at approximately 25.4 mm [1 in] from the basket base plate (which will be referred to as offset post). The finite element models used in the analyses are shown in Figure 6-3. Table 6-5 shows the maximum equivalent plastic strains for all four of the finite element analyses. For the offset post analyses, the outside, midsurface (thickness), and inside maximum equivalent plastic strains are at the same location through the thickness, while for the centered post analysis, the maximum strains did not occur at the same location through the thickness (Snow, 2004).

Recall that in Snow (2003), the minimum elongation strain was 47 percent, and the minimum fracture strain level was 118 percent, considering the material properties of all the canister component properties. For the puncture analyses, Snow (2004) assumes that only the shell experiences significant strains. Therefore, from Table 6-1, the minimum elongation is 59 percent, and the minimum failure strain is 118 percent.

From Table 6-5, the maximum equivalent plastic strains from the centered post for both drop heights are low {i.e., 9- and 12-percent strain for the 0.61- and 1-m [24- and 40-in] drops, respectively}. These values indicate no rupture of the containment boundary is likely to occur. On the other hand, the offset analyses have significant strains for both drop heights. Specifically, the 610-mm [24-in] drop onto the offset post had maximum plastic strains of 47, 45, and 43 percent for the outside surface, midsurface, and inside surface, respectively.

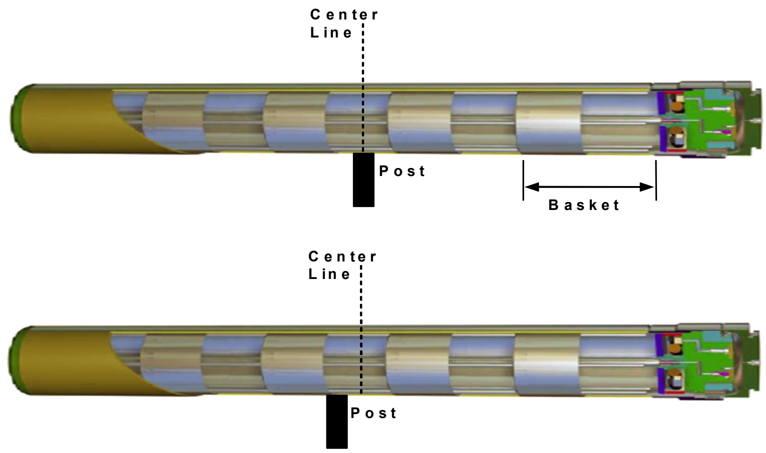


Figure 6-3. Multi-Canister Overpack Impact Post Locations (Morton, et al., 2006, Used with Permission of the Nuclear Society, Copyright 2006)

Table 6-5. Maximum Plastic Strains in the Multi-Canister Overpack Mark IV Main Shell for the Rigid Post Impact Events*			
Rigid Post Impact Height and Post Position	**Maximum Equivalent Plastic Strain (%) in the Multi-Canister Main Shell**		
	Outside Surface	**Midsurface**	**Inside Surface**
24-in† Centered Rigid Post	8	6	9
24-in† Offset Rigid Post	47	45	43
40-in‡ Centered Rigid Post	20	9	12
40-in‡ Offset Rigid Post	63	60	56

*Snow, S.D. "Analytical Evaluation of the Multi-canister Overpack for Puncture Drop Events." EDF–NSNF–039. Rev. 0. New York City, New York: ASME. 2004.
†Adjusted to 28-in to account for artificial energy losses (see Section 5.3.1).
‡Adjusted to 48-in to account for artificial energy losses (see Section 5.3.1).

Because these values are below the minimum elongation of 59 percent, the 610-mm [24-in]-offset drop does not result in a breach of the containment boundary. For the analysis of the 1-m [40-in]-offset drop, the largest maximum equivalent plastic strains are produced. Specifically, strain values of 63, 60, and 56 percent occur at the outside surface, midsurface, and inner surface, respectively.

Because the midsurface strain is just above the minimum elongation of 59 percent, a through crack and rupture of the containment boundary is likely to occur (Snow, 2004). Figure 6-4 shows the results for the 1,016-mm [40-in] drop height. In Figure 6-4(a), the deformation is

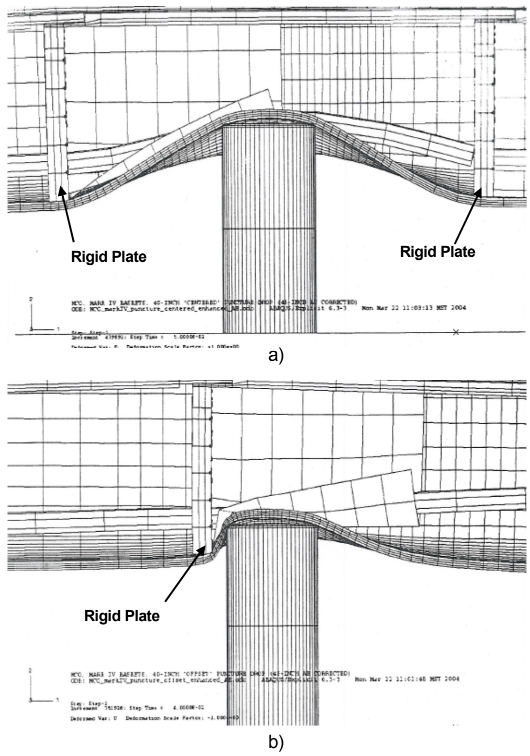


Figure 6-4. (a) Center Post Strike, (b) Offset Post Strike on Multi-Canister Overpack (Snow, 2004)

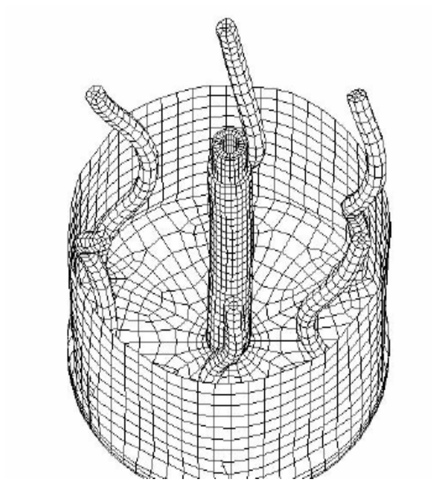
symmetric about the center post analysis, and because the deformation occurs between two rigid plates of the basket, the post is restrained uniformly by the canister containment boundary.

On the other hand, for the offset post analysis [Figure 6-4(b)], there is significantly higher (more severe) deformation (strains) to the left of the post. The rigid plate to the left of the post (at the bottom of the basket) restrains the deformation, resulting in a large amount of localized, asymmetric plastic strain.

Limited full-scale testing of a multi-canister overpack was undertaken in 2004 (Snow, et al., 2005; Morton, et al., 2006). Two multi-canister overpacks were fabricated, with each representing a typical Hanford multi-canister overpack and having the dimensions as given in Section 6.1. To account for strain-rate dependence of the stainless steel, the static true stress–strain curve was increased by 20 percent (see Section 4.3 for further discussion). For the finite element analysis, the multi-canister overpack had 4 baskets containing a 508-mm [20-in] diameter, 660-mm [26-in]-long bar with a center hole to fit around the center post of the basket. The fifth (bottom) basket contained 54 carbon steel bars 63.5 mm [2.5 in] in diameter to represent individual pieces of spent nuclear fuel. The tests consisted of one multi-canister overpack dropped from a height of 7 m [23 ft] in a vertical orientation, while the second multi-canister overpack was dropped from a height of 0.61 m [2 ft] with an initial angle of 60° off vertical (previously established as a critical orientation for slapdown to occur).

The first full-scale drop test {7-m [23-ft]} resulted in small deformations at the canister end {i.e., less than 6.35 mm [0.25 in] (Snow, et al., 2005)}. However, the majority of the deformation was in the internal components, with the impact energy absorbed by the fuel baskets, causing significant plastic deformation in the Mark IV bottom basket [Figure 6-5(a)]. Note that the tear in the multi-canister overpack basket was made for postdrop inspection and was not due to the drop (Snow, et al., 2005). The maximum equivalent plastic strains measured from the finite element analysis occurred at the bottom of the multi-canister overpack with a maximum surface strain of 3.5 percent and a midsurface strain of 2.9 percent. These small strains indicate there would be no breach of the multi-canister overpack containment boundary. However, as noted in the full-scale test results, a significant amount of deformation occurs in the bottom basket, as shown in Figure 6-5(b). For the multi-canister overpack dropped from a height of 0.61 m [2 ft] at an angle of 60° off vertical, there was minimal deformation. This limited height results in a small amount of impact energy, and as a result, only minor scuffing and flattening occurred along the bottom edge at the location of first impact. Due to the near-horizontal orientation, the top of the multi-canister overpack also contacted the surface, yet no noticeable deformations occurred (Snow, et al., 2005). Both of the multi-canister overpacks were helium leak tested and were determined to be leaktight.

(a)



(b)

Figure 6-5. Postdrop, Damaged Multi-Canister Mark IV Fuel Basket (a) Full-Scale Test, (b) Finite Element Analysis (Snow, et al., 2005, Used With Permission of the American Society of Mechanical Engineers, Copyright 2005)

7 STUDIES OF HIGH-LEVEL WASTE CANISTER DESIGNS

In the mid-1970s, a series of full-scale impact drop tests was initiated by Pacific Northwest Laboratories on canisters that contained high-level waste glass. The testing had two objectives: (i) to demonstrate that an impact due to an accidental drop would not breach the canister and possibly release the glass and (ii) to determine whether there was a breach such that the amount of respirable size (<10 μm diameter) glass particles released would be less than the amount allowed in the shipping cask (Peterson, et al., 1985). The drop tests demonstrated the structural integrity of the canister body, welds, fill nozzle, etc. when subjected to impact from an accidental drop. The drop height of these tests varied from 1 to 2.6 m [40 to 104 in] at a number of orientations. In the following sections, full-scale testing of actual canister designs and numerical modeling of purposed canister designs will be discussed.

7.1 Impact Testing of High-Level Waste Canisters

In 1983, full-scale tests were performed at Pacific Northwest Laboratory on representative Savannah River Laboratory high-level waste canisters. For these drop tests, two separate canisters were fabricated from 304L stainless steel, and one canister was fabricated from Grade 2 Titanium (Peterson, et al., 1985). Figure 7-1 shows typical canister dimensions. For these tests, the canister had a 610-mm [24-in] diameter and a length of 3 m [10 ft]. The canister was tested for both bottom and top drop orientations with the point of impact directly in line with the canister's center of gravity. The top and bottom drops were from a height of 9 m [30 ft], while the side drop was from a height of 1 m [40 in] onto a 152-mm [6-in]-diameter cylinder. The average filled weight of the canister was approximately 2,130 kg [4,740 lb].

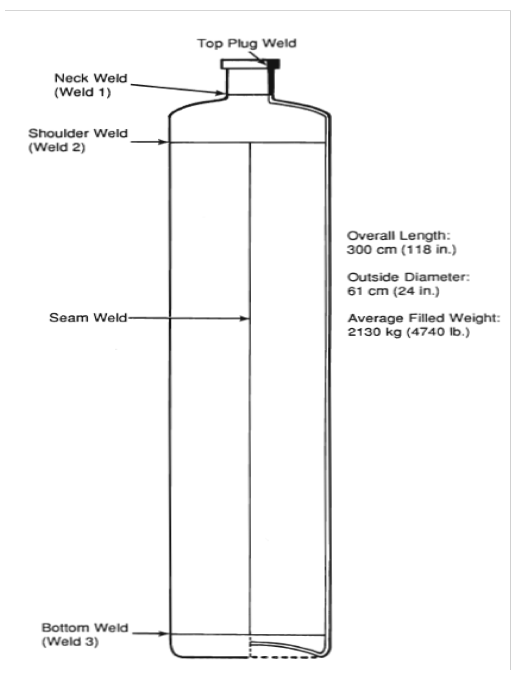


Figure 7-1. High-Level Waste Canister with Fabrication Details (Olson and Alzheimer, 1989)

Figure 7-2 shows the canister orientation for the top drop event. The titanium canister was dropped onto its bottom edge and produced asymmetric deformation as shown in Figure 7-3. The asymmetry was due to the point of impact being in line with the center of gravity; thus, the cylinder was not perfectly vertical. No rupture of the canister body was observed. For the top drop of the canister, the collapse of the nozzle also resulted in asymmetric deformation (Figure 7-4). Note that there is severe deformation of the canister top (nozzle), and a tear in the titanium body is observed. Peterson, et al. (1985) reasoned that the failure may be because titanium has less ductility than stainless steel, but they were not able to conclude the exact cause of the tear and suggested further testing. Strain circles were placed on the canister prior to the test. The strains near this location were approximately 14 percent, which was below the elongation to failure of 17 percent. However, it was noted that the failure mode appeared to be tearing of the material and not tensile failure. Direct measurements of the strain circles near the nozzle were not possible due to the severe deformation of the canister top.

Figure 7-5 shows the deformation of the top end of one of the stainless steel canisters. Note that the deformation is rather symmetric because there is no failure at the nozzle. Figure 7-6 shows the bottom canister deformation resulting from the bottom end drop test. Note again that the asymmetry of the deformation is the result of the point of impact being directly in line with the center of gravity of the canister (i.e., the canister is not in a perfectly vertical orientation). Strains measured from the test specimens showed tensile strains on the order of 12 to 14 percent. Figure 7-6 shows the stainless steel canister after the side impact with no rupture of the canister body observed. Helium leak testing and dye penetrant testing of the welds showed that the canisters were able to maintain its structural integrity (Peterson, et al., 1985).

In 1989, a series of full-scale impact tests was performed on Defense Waste Processing Facility canisters to show the canisters would not breach when subject to a 7-m [23-ft] drop onto

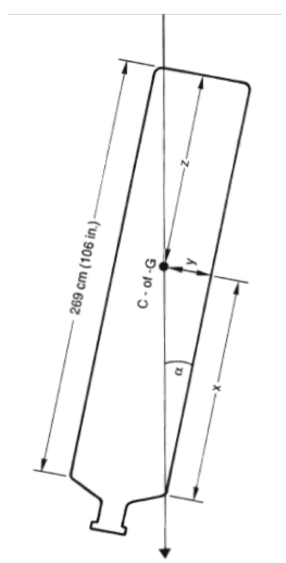


Figure 7-2. Canister Orientation for Center of Gravity Aligning With Corner Shoulder of High-Level Waste Canister (Olson and Alzheimer, 1989)

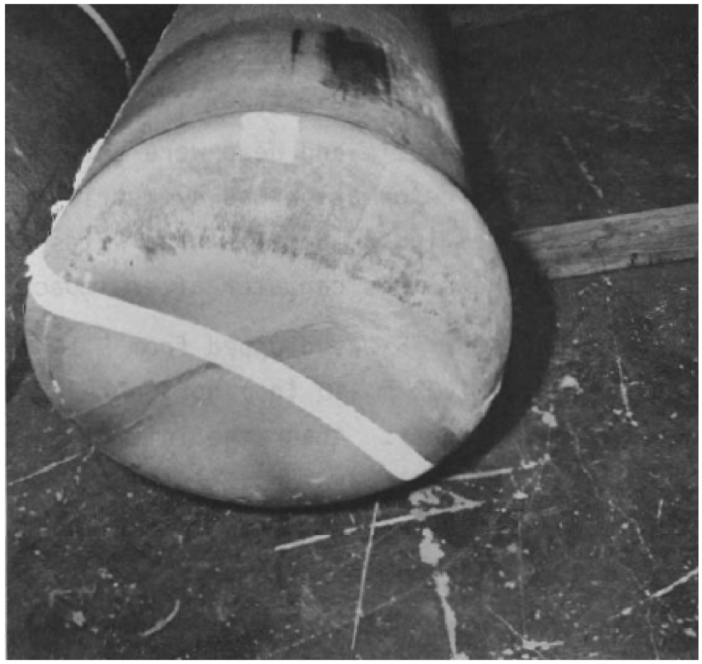


Figure 7-3. Asymmetric Deformation of Titanium Canister Bottom. Bottom Drop Event From a Height of 9 m [30 ft] (Peterson, et al., 1985).

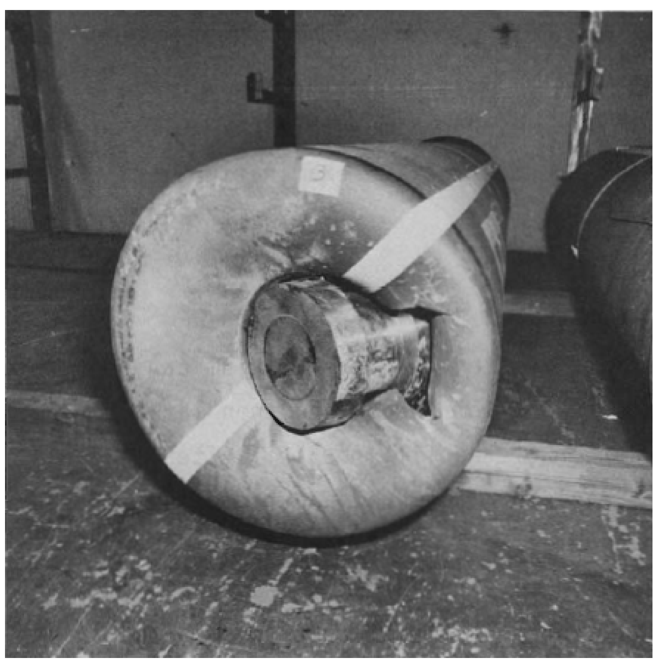


Figure 7-4. Asymmetric Deformation of Titanium Canister Showing Rupture at the Nozzle. Top Drop Event From a Height of 9 m [30 ft] (Peterson, et al., 1985).

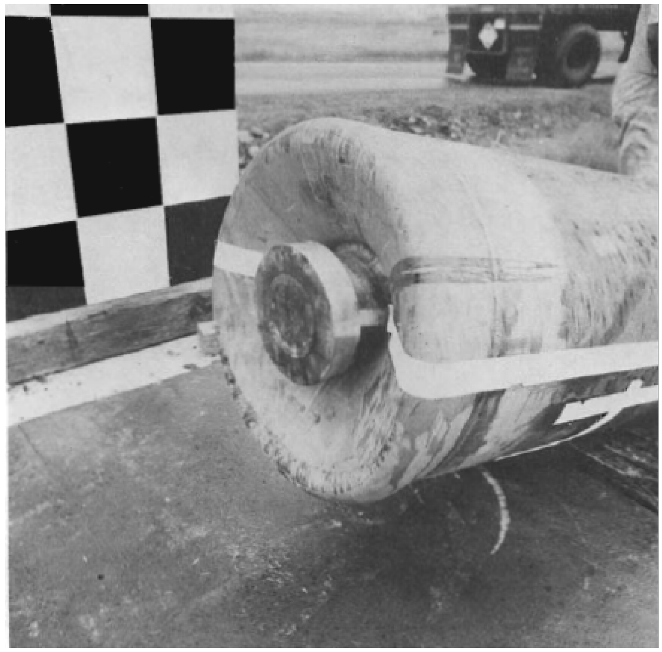


Figure 7-5. Symmetric Deformation of Stainless Steel Canister Nozzle Deformation. Top Drop Event From a Height of 9 m [30 ft] (Peterson, et al., 1985).

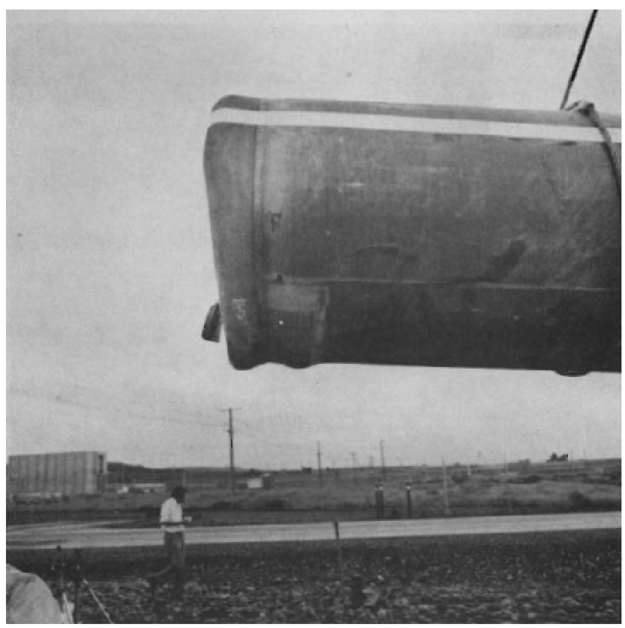


Figure 7-6. Asymmetric Deformation of Stainless Steel Canister Bottom. Bottom Drop Event From a Height of 9 m [30 ft] (Peterson, et al., 1985).

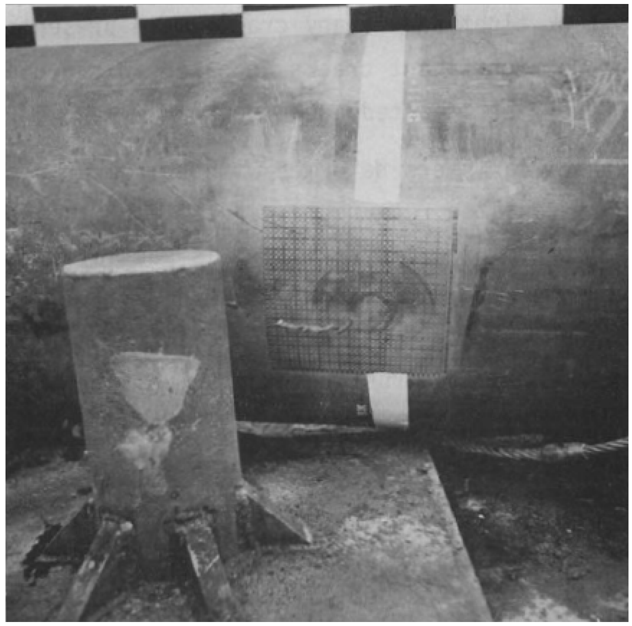


Figure 7-7. Side Deformation of a Stainless Steel Canister Due to Horizontal Drop Onto a 0.15-m [6-in] Cylinder. Drop Height is 1 m [40 in] (Peterson, et al., 1985).

an unyielding surface (Olson and Alzheimer, 1989). The test containers were filled at the Savannah River Laboratory with simulated waste glass to approximately 85 percent of their capacity with a total weight of approximately 2,150 kg [4,740 lb]. Each canister was 3 m [10 ft] long with a diameter of 610 mm [24 in]. The impact surface was a 216-mm [8.5-in]-thick steel plate placed on a 1.8-m [5.8-ft] reinforced concrete slab, which provided an essentially unyielding surface. Test orientations were similar to those described previously.

A small amount of deformation was observed for the vertically oriented (0° off center), bottom drop test. There also was observable rebound of approximately 750 mm [30 in] after first striking the impact surface. The amount of deformation was expected to be small because of the waste glass inside the canisters. Strain measurements were on the order of 2 percent or less. The height of the canisters on average decreased by 0.3 percent {i.e., 7.5 to 10 mm [0.3 to 0.4 in]} and the average diameter increased by 2.5 percent {i.e., 10 to 15 mm [0.4 to 0.6 in]} in the area close to the bottom of the canister (Olson and Alzheimer, 1989).

A second orientation was chosen in which the center of gravity was in line with the shoulder of the canister [i.e., where the canister body meets the curved top just above the shoulder (Weld 2, Figures 7-1 and 7-2)]. All of the canisters exhibited a significant amount of deformation to the canister top flange/nozzle (Figure 7-8) (Olson and Alzheimer, 1989). For this case, approximately 150 mm [6 in] of rebound after first impact was observed. This markedly smaller amount of rebound is to be expected because there was significant plastic deformation of the canister top. As in the previous study, strain circles were used to measure the amount of strain produced by the impact. The largest measured strain for the top drop test was 52 percent, which occurred perpendicular to the weld where the top meets the canister body (see

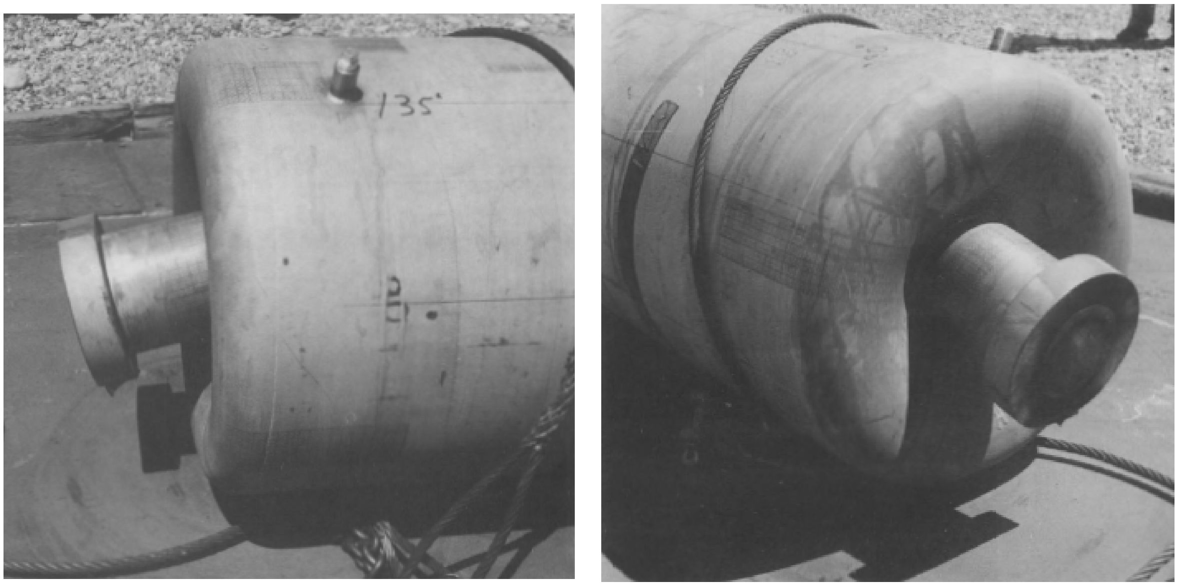


Figure 7-8. End Deformation of Stainless Steel Test Canister Subjected to a Top Drop With Center of Gravity in Line With Canister Shoulder (Olson and Alzheimer, 1989)

Figure 7-1, Weld 2); however, the strains in this location were compressive. Olson and Alzheimer (1989) state that tensile strains would be produced on the inside of the canister, but they would be lower in magnitude. Any preexisting flaws of the weld on the inside of the canister could open, but they would not propagate through the thickness because of the compressive strains on the outside surface. With respect to the changes in canister dimensions, there was a decrease in height of 5 percent {i.e., 140 to 164 mm [5.5 to 6.5 in]}, which was mainly due to the nozzle being pushed back into the canister top, and a 1-percent increase in diameter from 17 to 24 mm [0.4 to 0.6 in]. All of the canisters remained leaktight as measured by the standard helium test. Dye penetrant testing of the welds also showed no cracks developed in the welds. Thus, all canisters showed no indication of breach.

In 1994, five containers from the West Valley Demonstration Project to store high-level waste glass were similarly tested by Pacific Northwest Laboratory (Whittington, et al., 1995). The canisters were approximately 3 m [118 in] long and 610 mm [24 in] in diameter. Each canister, in a vertical orientation, was dropped from a height of 7 m [23 ft] onto an unyielding impact surface constructed at the Pacific Northwest Laboratory test facilities.

When each of the canisters were drop tested, three remained upright, while two canisters experienced slapdown. Each canister rebounded from the impact surface by 152 to 203 mm [6 to 8 in]. All of the canisters showed a minimum amount of deformation after impact with an increase in diameter located approximately 305 mm [12 in] from the bottom head (Whittington, et al., 1995). Each canister was marked with strain circles to measure the amount of strain at regions that were expected to produce the maximum amount of deformation along a line that extended from the bottom to a distance of 254 mm [10 in] above the bottom. The strain circles measured strains of 1 to 3 percent. At a distance above 254 mm [10 in], the strains were somewhat constant and smaller than those measured at the lower side wall of the canister (Whittington, et al., 1995). Most of the axial strains were compressive, as expected, with some tensile strains present due to bending of the canister. The axial strains measured were in the range of 5 percent compressive and 3 percent tensile. The strains measured in the hoop direction were tensile with a maximum hoop strain of 7 percent. These strains were well below the 56 percent failure strain of 304L stainless steel.

The amount of canister deformation was quantified by measuring the change in diameter and height and were used to judge the straightness of the canisters. This is necessary because the damaged canister was required to fit into a 635-mm [25-in]-diameter cylindrical cavity without forcing. For all of the canisters, there was a small amount of increase in diameter on the order of 12.7 mm [0.5 in] (2 percent) at the bottom head. The change in the height of the canister was decreased by approximately 12 mm [0.47 in] (0.4 percent) on average. With these small changes in canister dimension, the canisters were able to fit into the outside sleeve. Helium leak testing was performed on each of the canisters, and the canisters were considered leaktight (Whittington, et al., 1995).

7.2 A Numerical Study of Conceptual High-Level Waste Canisters

This section presents a number of conceptual canister designs described in Hill, et al. (2004). These canisters are used to dispose of calcine, a vitrified (particulate) form of high-level waste. All of the work presented in this section is purely conceptual and, therefore, studied solely using finite element models of the proposed canisters. The validity and accuracy of using the finite element method to estimate the deformation behavior of standardized canisters has been discussed in previous sections. Four types of canister designs will be discussed: (i) 610 mm [24 in]; (ii) 1,676 mm [66 in]; (iii) 1,676-mm [66-in] donut; and (iv) 1,676-mm [66-in] flat bottom high-level waste canisters.

The 610-mm [24-in] high-level waste canister model is quite similar to the 610-mm [24-in]-diameter DOE standardized spent nuclear fuel canister design, although there are a few design elements that would be different [i.e., the top head is modified for the loading of the calcine waste, and there would be no internal impact plates (located at each end)]. The 610-mm [24-in]-diameter canister has a wall thickness of 12.7 mm [0.5 in] or 9.525 mm [0.375 in] for these conceptual designs. This canister would also have a length of 4,572 mm [180 in] consistent with the standardized spent nuclear fuel canister.

The design of the 1,676-mm [66-in]-diameter canister has the same structural modifications as the 610-mm [24-in] canister. The wall thicknesses to be analyzed are 9.5, 12.7, 19.1, 25.4, and 35 mm [0.375, 0.50, 0.75, 1.00, and 1.375 in], and the 1,676-mm [66-in]-diameter canister length is 5,334 mm [210 in] (Hill, et al., 2004). The so-called donut design (Figure 7-9) is similar to the 1,676-mm [66-in] option, except the canister has a cylindrical opening at the center with an approximate inside diameter of 470 mm [18.5 in]. This cylindrical opening would allow the codisposal of a DOE standardized spent fuel canister. Each of these three canisters would have a 686-mm [27-in] skirt around each end.

Unlike the previous three canisters, the proposed 1,676-mm [66-in] flat bottom design (Figure 7-10) would not have a skirt on each head of the canister and has a design similar to the Navy Long Spent Fuel Canister (Hill, et al., 2004). The 1,676-mm [66-in] flat bottom canister has a wall thickness of 25.4, 19.05, 12.1, and 9.5 mm [1.0, 0.75, 0.5, and 0.375 in], and the top and bottom heads have 381-mm [15.0-in] and 89-mm [3.5-in]-thick plates, respectively. The overall length of the canister is 5,334 mm [210 in].

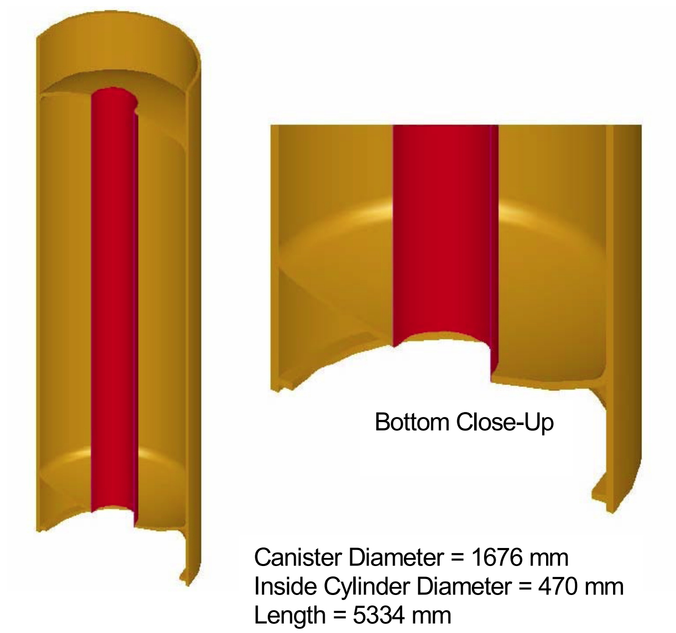


Figure 7-9. 1,676-mm [66-in] Donut Canister (Hill, et al., 2004 Used With Permission of WM Symposia, Inc., Copyright 2005) [25.4 mm = 1 in]

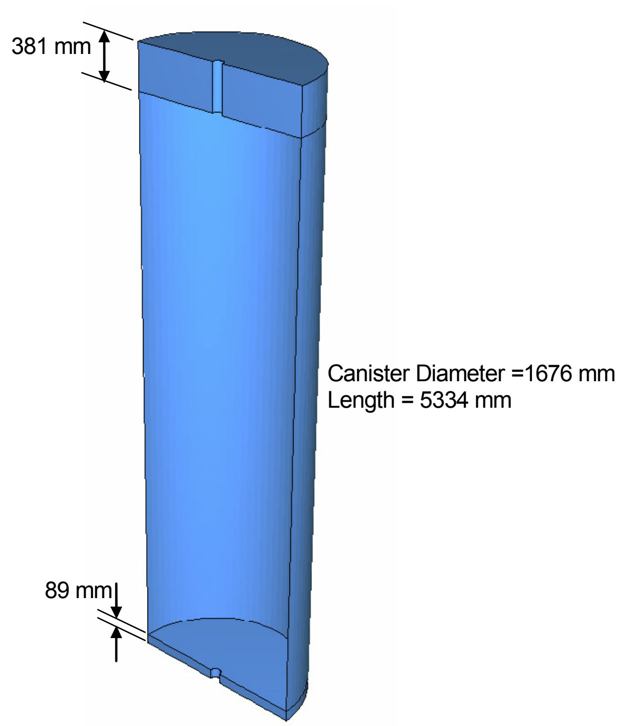


Figure 7-10. 1,676-mm [66-in] Flat Bottom Canister (Hill, et al., 2004 Used With Permission of WM Symposia, Inc., Copyright 2004) [25.4 mm = 1 in]

As in the standardized spent nuclear fuel canister designs, the material chosen for these conceptual canisters is 316L stainless steel. To account for the high strain rates, the dynamic true stress–strain curve is obtained from increasing its values by 20 percent.

The proposed canisters were subjected to a number of drop simulations using the finite element code ABAQUS/Explicit. Specific details of each canister's finite element model are described in Hill, et al. (2004). A number of drop orientations were analyzed similar to those used in the finite element analysis of the nine standardized spent fuel canisters discussed in Chapter 3. Some of the parameters used in this study to evaluate the proposed designs were maximum allowable design pressure, maximum plastic strain, and maximum deformation. The maximum plastic strain and the amount of deformation will be discussed next.

Table 7-1 shows the maximum equivalent plastic strain. Rupture is said to occur if the midplane strain exceeds 60 to 80 percent, which is based upon the correlation of the finite element analyses with the full-scale testing of Snow, et al. (1999). Note that for all cases, the midplane strain is below the rupture strain. The 1,676-mm [66-in] canister and the 1,676-mm [66-in] donut canister have the lowest strains of all the designs. These canisters have the flange at each end of the canister. The 1,676-mm [66-in] flat bottom canister has a maximum containment pressure boundary strain of 72 percent at the surface, but because the head plates are so thick, the canister does not incur a significant amount of deformation (Hill, et al., 2004).

Figure 7-11 shows the representative deformations of a canister resulting from a corner impact (i.e., its center of gravity is directly over the contact corner). Note that the deformation is limited to the skirt, and the amount of deformation depends on the wall thickness. A summary of the deformation acceptability is given in Table 7-2. The determination of whether the canister deformation is acceptable or not was based strictly on the judgment of Hill, et al. (2004).

The final conclusion was that the 610- or 1,676-mm [24- or 66-in] canisters (with the dished head and skirt) had comparable deformations. The 610-mm [24-in] canister would provide ease of handling, while the 1,676-mm [66-in] canister would result in fewer canisters required for waste disposal.

Table 7-1. Surface and Midplane Plastic Strains*

Model	Maximum Equivalent Plastic Strain (%)	
	Surface	**Midplane**
610-mm [24-in] Canister	65	32
1,676-mm [66-in] Canister	50	22
1,676-mm [66-in] Donut Canister	48	22
1,676-mm [66-in] Flat Bottom	72	27

*Hill, T.J., T.E. Rahl, D.K. Morton, K.C. Coughlan, J.T. Beck, and M.W. Patterson. "Canister Design for Direct Disposal of HLW Calcine Produced at the Idaho National Engineering and Environmental Laboratory." Waste Management 2004 Conference, Tucson, Arizona, February 29. Table VIII. Tucson, Arizona: Waste Management Symposium, Inc. 2004. Used With Permission of WM Symposia, Inc., Copyright 2004.

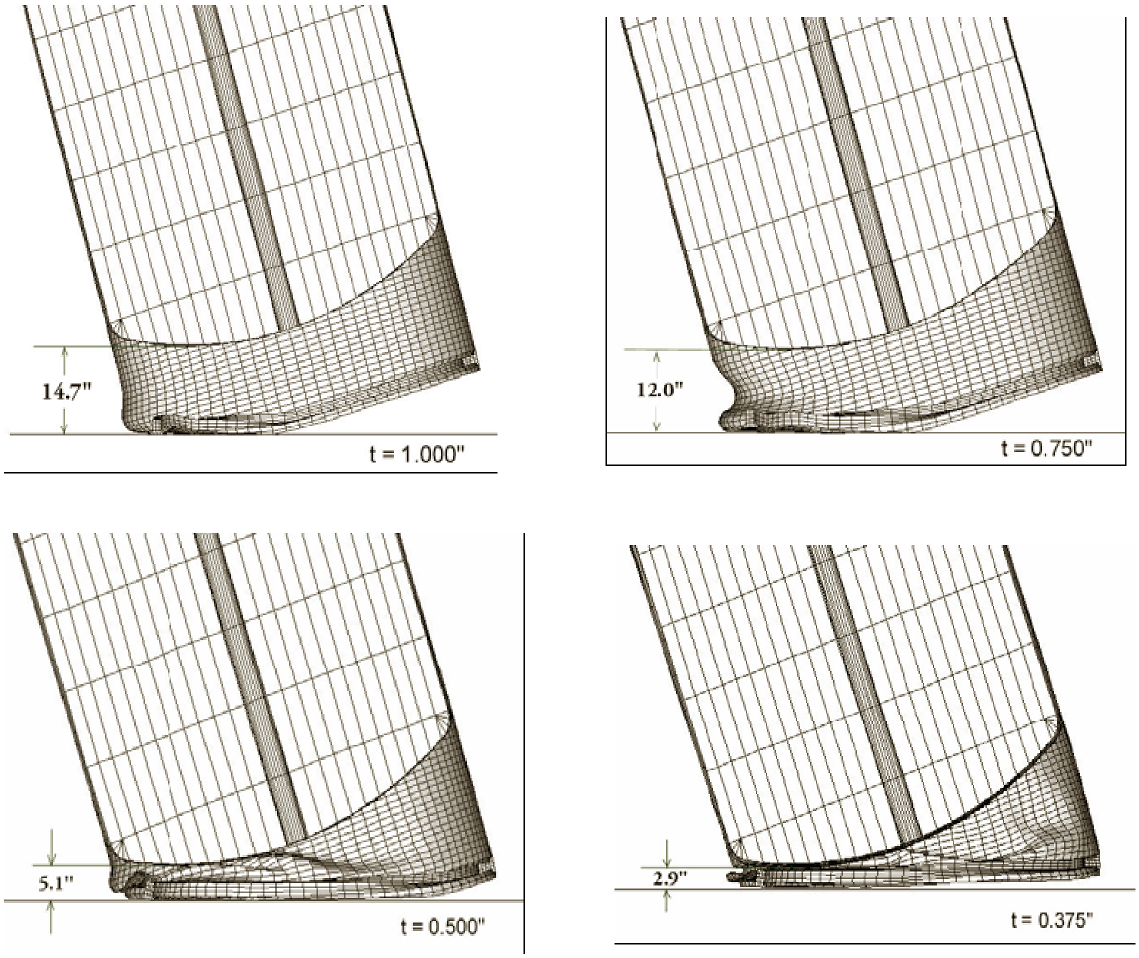


Figure 7-11. Corner Drop for Different Skirt Thickness (Hill, et al., 2004, Used With Permission of WM Symposia, Inc., Copyright 2004) [25.4 mm = 1 in]

Table 7-2. Summary of Observed and Predicted Acceptability of Deformation*			
		Max Deformation	
Model	**Thickness Variation**	**Acceptable**	**Not Acceptable**
610-mm [24-in] Canister	$t_{wall} = t_{head}$ = 12.7 mm [0.500 in]	X	
	$t_{wall} = t_{head}$ = 9.5 mm [0.375 in]	X	
1,676-mm [66-in] Canister	$t_{wall} = t_{head}$ = 35 mm [1.375 in]	X	
	$t_{wall} = t_{head}$ = 25.4 mm [1.000 in]	X	
	$t_{wall} = t_{head}$ = 19 mm [0.750 in]	X	
	$t_{wall} = t_{head}$ =12.7 mm [0.500 in]		X
	$t_{wall} = t_{head}$ = 9.5 mm [0.375 in]		X
1,676-mm [66-in] Donut Canister	$t_{wall} = t_{head}$ = 35 mm [1.375 in]	X	
	$t_{wall} = t_{head}$ = 25.4 mm [1.000 in]	X	
	$t_{wall} = t_{head}$ = 19 mm [0.750 in]	X	
	$t_{wall} = t_{head}$ =12.7 mm [0.500 in]		
	$t_{wall} = t_{head}$ = 9.5 mm [0.375 in]		X
1,676-mm [66-in] Flat Bottom	t_{wall} = 25.4 mm [1.0 in] $t_{bottom\ head}$ = 89 mm [3.50 in]	X	
	t_{wall} =19 mm [0.75 in] $t_{bottom\ head}$ = 89 mm [3.5 in]	Marginal	Marginal
	t_{wall} = 12.7 mm [0.50 in] $t_{bottom\ head}$ = 89 mm [3.5 in]		X
	t_{wall} = 9.5 mm [0.375 in] $t_{bottom\ head}$ = 89 mm [3.5 in]		X

*Hill, T.J., T.E. Rahl, D.K. Morton, K.C. Coughlan, J.T. Beck, and M.W. Patterson. “Canister Design for Direct Disposal of HLW Calcine Produced at the Idaho National Engineering and Environmental Laboratory.” Waste Management 2004 Conference, Tucson, Arizona, February 29, 2004. Tucson, Arizona: Waste Management Symposium, Inc. 2004.

8 CANISTER FABRICATION AND WELDING

In the previous chapters, various canister types (i.e., standardized, multi-canister overpack, and high-level waste) were reviewed in the literature to determine whether the canisters were able to withstand dynamic loads in the form of drop events. These canisters were evaluated in terms of maximum equivalent plastic strains and their ability to pass a leak test. Utilizing longitudinally welded 316L stainless steel pipe provided limited experimental testing of the welds. By satisfying the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, fabrication requirements, this should produce a canister without any significant flaws. However, it is possible that a flaw may be introduced because the materials may become embrittled during welding. As part of the evaluation of the canister subject to dynamic loads (i.e., a drop event), the fracture toughness of the welds is important. One method used to measure toughness is the Charpy V-Notch impact energy test. The following sections will discuss the mechanical and impact properties of the welds.

8.1 DOE Standardized Spent Nuclear Fuel Canister

The fabrication and welding process is described in DOE/SNF/REP–011 for the DOE standardized spent nuclear fuel canister (DOE, 1999a,b). The DOE spent nuclear fuel canisters may be made of SA–312 type 316L stainless steel for the shell and SA–240 type 316L for all other parts, including the heads, labels, and lifting rings. The optional plugs may be SA–479 type 316L stainless steel. All stainless steel materials may be annealed and pickled. A pressure boundary wall thickness reduction of 1.27 mm [0.050 in] has been established as the corrosion and erosion value to be used for canister design purposes. This value reflects the full design lifetime of 100 years. It is assumed that once the DOE spent nuclear fuel canister is placed inside the waste package, insignificant corrosion or erosion will occur for the next 50-year interval (DOE, 1999a).

Both the inner and outer surfaces of the canisters may have a finished condition for acceptable nondestructive examinations. The size of any burrs, sharp edges, and weld edges may be controlled. The interior surfaces may be smooth enough to allow easy loading of any DOE spent nuclear fuel or internals so as to not damage the spent nuclear fuel. During accidental drop events, high stress and strain values may be expected in local regions of the DOE spent nuclear fuel canisters where internal structures contact the canister shell pressure boundary. Internal structures (including baskets, spacers, sleeves, dividers, cans, welds) and fuel elements may contain sharp, stiff elements that could puncture the canister containment.

The DOE spent nuclear fuel canisters may be designed, fabricated, and examined per the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Division 3, Subsections WA and WB, 1997 Edition (or the most current edition approved by the NRC provided the specific code changes) for the loads and environments identified in this design specification. Certain changes to Section III, Division 3 need to be made for the DOE spent nuclear fuel canisters to be fabricated as N-stamped vessels, including (i) allowing field operations, (ii) allowing the actual N-stamping prior to the spent nuclear fuel being loaded into the DOE spent nuclear fuel canisters, (iii) allowing the use of ultrasonic examination for the final closure weld, and (iv) testing for helium leaks in lieu of pressure test requirements, after loading the spent nuclear fuel and final closure welding, and allowing helium leak testing in lieu of pressure testing for low design pressure vessels.

Due to the presence of the spent nuclear fuel after loading, the DOE spent nuclear fuel canisters may not be required to satisfy pressure test requirements of Section III, Division 3, after loading, final seal, and welding. The final canister weld may implement a welding procedure that can be qualified to yield leaktight welds. A leaktight weld may be considered equal to or better than the required leak rate necessary to satisfy the applicable 10 CFR Parts 71 and 72 requirements.

Morton and Snow (2000) indicate that all possible situations were adjusted to inflict the most damage to the test canister welds during their drop tests. The test canisters contained a number of weld joints that did not receive any postweld heat treatment. Although a seamless pipe can be used, the pipe used to fabricate the test canister bodies and skirts was longitudinally welded pipe. By using this type of pipe, the weld joint would be the logical first location to check if a problem were to develop.

A combination of manual tungsten inert gas and manual pulse metal arc (wire feed) welding techniques may be used. All of the pressure boundary welds existing before loading the test canisters with internals may be volumetrically examined using radiography testing and liquid penetrant examinations. Ultrasonic testing methods may be used to perform volumetric examinations of the final closure welds. After the drop tests are completed, the same two welds may be reexamined using both ultrasonic testing and radiography testing.

Schuster, et al. (2000) indicate that base metal flaws such as laminations (e.g., caused by multiple weld passes) may cause the breach of a dropped canister. An undetected flaw of sufficient size in specific locations with a particular orientation could lead to a canister breach. Based on the American Society of Mechanical Engineers Code, construction, examination requirements, and operational experience, a maximum undetected cracklike flaw size of more than 1 mm [0.04 in] is the weld flaw criteria for which canister drop survivability is judged (Smith, 2003).

The designer and fabricator of the DOE spent nuclear fuel canisters may establish, maintain, and execute a quality assurance program based on the criteria necessary to satisfy the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Division 3 construction criteria, 10 CFR Part 71 (Subpart H), and Part 72 (Subpart G) quality assurance requirements.

8.2 Multi-Canister Overpack Design

Goldmann (2000a) describes the multi-canister overpack designs, and related detailed design information is further given in Garvin (2001) and Goldmann (2000b). The multi-canister overpack shell may be fabricated from type 304/304L stainless steel. All components welded to the multi-canister overpack shell may be made of austenitic stainless steels compatible for welding to 304L stainless steel. A mechanically attached shield plug and any components thereof may be made from either 304L, 304N B&S, or B&SA. All materials may be American Society of Mechanical Engineers/American Standard Testing and Materials certified. A provision may be made to preclude metal-to-metal galling in threaded multi-canister overpack components.

All multi-canister overpack fabrication pressure boundary welds may be made in accordance with American Society of Mechanical Engineers Code, Section III, requirements. All welds may be sufficiently smooth to enable easy decontamination. Butt welds may be ground flush to within 0.76 mm [0.03 in] of base metal. Weld joint designs may avoid potential contamination traps to the greatest extent practicable. All multi-canister overpack pressure boundary welds and welds bearing the weight of the fully loaded multi-canister overpack may be designed for and pass nondestructive examination per American Society of Mechanical Engineers Code, Section III, Division I requirements.

Materials corrosion in terms of a reduction of the thickness should be considered in drop event analyses. Some potential corrosion modes include passive general corrosion, localized corrosion, galvanic corrosion, stress corrosion cracking, and hydrogen embrittlement. However, no quantitative penetration depth is given that can be potentially used in a drop event analysis.

Quality assurance requirements from the Project Hanford Office of Civilian Radioactive Waste Management Quality Assurance Program Plan may be applied to applicable fabrication, inspection, testing, handling, cleaning, shipping, and storage. The Quality Assurance Program Plan requires that Office of Civilian Radioactive Waste Management related spent nuclear fuel project activities comply with DOE/RW–0333P, Quality Assurance Requirements and Description for the Civilian Radioactive Waste Management Program (Quality Assurance Requirements and Description). Retrofit of the multi-canister overpack design to comply with the Office of Civilian Radioactive Waste Management is not required.

8.3 High-Level Waste Glass Canister

High-level waste glass canisters have been planned to be produced from West Valley Demonstration Project, Defense Waste Process Facilities, Hanford Waste Vitrification Plant, or Idaho Chemical Processing Plant. A summary of design specifications and processing guidance is presented in Ahn (1999). The majority of the high-level waste glass canisters are made of 304L stainless steel. Gas tungsten arc welding and upset resistance welding are used. Details are similar to other canister processes.

8.4 NRC Interim Staff Guidance

NRC issued Interim Staff Guidance on alternatives to the American Society of Mechanical Engineers Code (NRC, 2000) and on materials evaluation (NRC, 2001). These interim staff guidances are primarily for storage and transportation (10 CFR Parts 71 and 72).

ISG–10, Revision 1, Code Section III, may be used as an acceptable standard for the design and fabrication of a canister. Because a canister is not a pressure vessel, American Society of Mechanical Engineers Code, Section III, cannot be implemented without allowing some alternatives to its requirements. Specific alternatives may be used on a case-by-case basis for those requirements that are not applicable or practical to implement for the canister. The proposed alternatives may provide an acceptable level of quality and safety. Compliance with the specified requirements of American Society of Mechanical Engineers Code, Section III, may result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The materials evaluation chapter, ISG–15, ensures quality and uniformity in reviews performed. There are two nationally recognized codes that address welding: American Society of Mechanical Engineers and AWS D1.1. The American Society of Mechanical Engineers Code governs welded pressure vessels, from domestic water heaters to nuclear reactors. The AWS D1.1 Structural Welding Code is the applicable code for welding structural steel, such as the steel used for bridges and steel-framed skyscrapers. The various construction codes differ in their requirements for materials and welding procedures because each code is specialized with a particular application in mind. Standard weld and nondestructive examination symbols may be found in AWS A2.4. Except for welded closure lids, all welds of the confinement shell need to be full penetration welds. The nondestructive examination for these confinement welds is volumetric.

For designs employing lid materials and welds, either volumetric or multipass liquid penetrant testing inspection methods are acceptable. For either ultrasonic testing or liquid penetrant examination, the minimum detectable flaw size may be demonstrated to be less than the critical flaw size. The critical flaw size may be calculated in accordance with American Society of Mechanical Engineers XI methodology; however, net section stress may be governing for austenitic stainless steels and may not violate American Society of Mechanical Engineers Code, Section III, requirements. Flaws in austenitic stainless steels are expected to exceed the thickness of one weld bead. If using ultrasonic testing, the ultrasonic testing acceptance criteria are the same as those of NB–5332 for preservice examination. In accordance with code practice for supplementing volumetric examinations with a surface examination, a ultrasonic testing examination may be performed in conjunction with a root pass and cover pass liquid penetrant testing examination. If liquid penetrant testing is specified (i.e., no volumetric inspection), a stress reduction factor of 0.8 must be applied to the weld design.

8.5 Mechanical and Impact Properties of Welds

In the impact (i.e., drop) analysis/modeling of the 316L stainless steel canister, mechanical properties are important as input parameters. It is likely that 316L is used in the design of a currently considered canister (e.g., DOE, 2006). Therefore, data for 316L and its analogs are summarized here. The mechanical properties include yield strength, tensile strength, ductility, fracture toughness, and Charpy V-Notch impact energy. These mechanical properties vary depending on temperature, strain rate, fabrication, and welding. Recently, Dunn, et al. (2003) summarized relevant mechanical property data of 316L stainless steel or its analogs for various material and test conditions. Dunn, et al. (2003) summarizes the 316L stainless steel and discusses a few important parameter values associated with fabrication and welding for potential use in reviewing the assessment of the integrity of the 316L stainless steel canister under drop conditions:

> "The effects of fabrication processes on the mechanical properties of austenitic stainless steels are receiving considerable attention as a result of the use of these materials in nuclear power plants. Multiple studies have investigated the yield strength, tensile strength, and ductility of austenitic stainless steel base metals and welds. Although a marginal loss of ductility is observed for welded materials, welded austenitic stainless steels remain quite ductile. Fracture toughness measurements have been conducted to investigate the effects of crack orientation, cold work, inclusion content, temperature, weld metal composition, and welding method.

"Although heat-to-heat variations are large, the fracture toughness of austenitic stainless steel base metals and welds is generally high owing to their relatively low strength, strain hardening, and high ductility. Nevertheless, the fracture toughness of wrought austenitic stainless steel is dependent on the inclusion content, cold work, and crack orientation. A minor amount of cold work can result in a substantial decrease in fracture toughness. Inclusions have the greatest impact on cracks oriented along the rolling direction and parallel to the surface of wrought plate. Welds in austenitic stainless steels also have high fracture toughness, but the toughness value is dependent on the welding processes. Welds produced with methods that result in a minimal increase in the inclusion content, such as gas tungsten-arc welding and gas metal-arc welding retain fracture toughness similar to that of the wrought base material. Welding methods that result in a substantial increase in inclusion content such as submerged-arc welding, reduce the fracture toughness of the weld. Nevertheless, the fracture toughness of welded stainless steels is generally sufficiently high to preclude fracture-dominated failure."

In the impact analysis/modeling, the use of Charpy V-Notch impact energy may be more important. Dunn, et al. (2003) attempted to correlate the fracture toughness to the Charpy V-Notch impact energy. For example, the fracture toughness of ferritic pressure vessel steels with yield strength of more than 690 MPa (100 ksi) can be related to the Charpy V-Notch impact energy through an empirical relationship

$$[K_{IC}/\sigma_{YS}]^2 = 5.0\ [CVN/\sigma_{YS} - 0.05] \tag{8-1}$$

where K_{IC} is fracture toughness, σ_{YS} is yield strength, and CVN is Charpy V-Notch impact energy. Detailed correlation in 316L welded stainless steel is not established. However, the Charpy V-Notch impact energy of welded 316L stainless steel is available in the literature, as shown in Figure 8-1.

8.6 Preclosure Safety Analysis

The primary function of the canister (e.g., DOE, 2006) would be the containment of radionuclides during preclosure operations. Currently, the canister is considered to be important to safety in the Preclosure Safety Analysis during preclosure operations (NRC and DOE, 2006a,b) in the disposal. However, under canister drop conditions from various accidents, the canister may fail. The failed canister could release radionuclides. The dose to the worker and the public from the radionuclide release needs to meet the performance objectives in 10 CFR Part 63.111 (a). The consequence analysis of radionuclide release from the failed canister needs the radionuclide source term. In the Preclosure Safety Analysis, the source term for a given event sequence depends on release fraction and leak path factor, among several others (Kamas, et al., 2006). Release fraction is the fraction of each radionuclide released from the inventory. Leak path factor is the product of various factors involved in the subsequent mitigation of radionuclide release and dispersion from the release fraction (e.g., failed canister mitigation factor, building discharge fraction, and HEPA filter mitigation factor).

The failed canister mitigation factor was determined primarily by how much canister surface area would be exposed upon the canister failure. The surface area opening will in turn depend on the impact energy. For example, Sprung, et al. (2000) indicate a radionuclide particulate

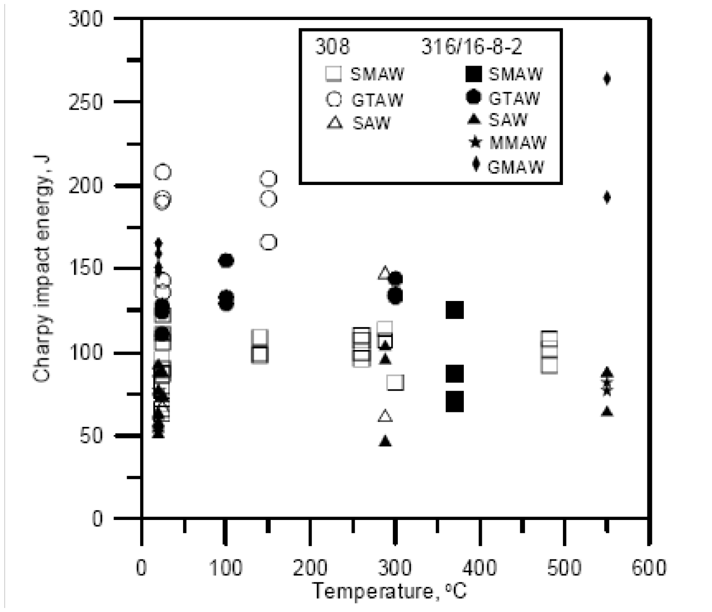


Figure 8-1. Charpy V-Notch Impact Energy for Austenitic Stainless Steel Welds (Gavenda, et al., 1996) [°F = 9/5 °C + 32, 1 Ft-lb = Joule/1.35] (SMAW = Shielded Metal-Arc Weld; GTAW = gas; Tungsten = ARC Weld; SAW = Submerged Arc Weld; GMAW = Gas Metal-Arc Weld; MMAW = Manual Metal-Arc Weld)

leak path factor of 0.02 for a impact speed of 96.5 meters per hour [60 miles per hour] pressurized to 5 atmospheres (506.6 kPa). In the drop case, this impact speed is determined primarily by the drop height. Alternatively, the drop height or the impact speed gives the impact energy onto the canister.

The Charpy V-Notch impact energy shown in Table 8-1 suggests how the open surface area of the canister could be estimated upon impact. The American Standard Testing and Materials accepted Charpy V-Notch specimen has a normal size of 1 × 1 × 5.5 cm [0.4 × 0.4 × 2.2 in] (American Standard Testing and Materials, 2007) with a 0.2-cm [0.08-in] notch. Assuming the total impact energy is used to create the surface area of the notched ligament, a correlation of the impact energy input and the open surface area can be estimated. For example, 100 J of Charpy V-Notch impact energy may create a fraction of ~0.03 of the surface area. The specific impact energy in this case would be higher than normally expected from the canister drop test at a 7-m [23-ft] height. This type of estimate can be normally used for ductile materials such as 316L stainless steel. The background information for the estimate of the specific impact energy for a canister might be obtained from Sprung (2000), and Kamas, et al. (2006) presented some exercise results. Recently, DOE conservatively assessed the particulate leak path factor of radionuclides as 0.1 (Dexheimer, et al., 2006).

Table 8-1. Effect of Inclusions and Specimen Orientation on Yield Strength, Tensile Strength, and Ductility*						
	Type 316L Heat A Inclusion Number: 41			Type 316L Heat B Inclusion Number: 85		
Orientation	**Yield Strength MPa [ksi]**	**Tensile Strength MPa [ksi]**	**Elongation Percent**	**Yield Strength MPa [ksi]**	**Tensile Strength MPa [ksi]**	**Elongation Percent**
S-L	293 [42.5]	571 [82.8]	73	281 [40.8]	445 [64.5]	15
T-L	303 [43.9]	597 [86.6]	73	279 [40.5]	558 [80.9]	74
L-T	310 [44.9]	591 [85.7]	76	279 [40.5]	561 [81.4]	78

*Dunn, D.S., Y.-M. Pan, D. Daruwalla, and A. Csontos. "The Effects of Fabrication Processes on the Mechanical Properties of Waste Packages—Progress Report." San Antonio, Texas: CNWRA. 2003.

9 SUMMARY

This report compiles literature evaluating the structural behavior of DOE spent nuclear fuel, Idaho Spent Fuel Project canisters, multi-canister overpack, and high-level waste canisters.

In several of these studies, finite element analyses were initially used to determine a preliminary canister design that would preserve the canister containment boundary (i.e., no breach). Small- and full-scale drop tests, performed by DOE, of proposed standardized canister designs were used to verify and validate the finite element analyses; accuracy was evaluated by comparing deformations of the test canister and those of the finite element model. Once the preliminary design was selected, two full-scale tests on 457- and 610-mm [18- and 24-in] canisters were performed. Finite element analyses of both drop tests were once again validated by comparison with the full-scale tests. Based upon these full-scale tests and corresponding finite element analyses, DOE chose the standardized canister design with the notable design feature of a flanged and dished head with a skirt welded around each end.

Based upon the previous small- and limited-full-scale tests, DOE performed a broader set of nine full-scale drop tests on the proposed standardized spent fuel canister. One test objective was to verify that the skirt attached to each end absorbed the impact energy, as designed. Each full-scale test measured the amount of canister deformation and whether or not the containment boundary was affected. Visual inspection and pressure tests were used to determine whether the containment boundary was intact. Finite element analysis of each drop test was performed, and DOE observed very good agreement, in terms of deformation, between the test and analysis. Surface and midsurface (thickness) maximum plastic strains obtained from the finite element analyses also verified no breach of the containment boundary. For a drop height of 9 m [30 ft], a maximum equivalent plastic surface strain of 57 percent occurred in the canister upper head, a maximum midplane (thickness) strain of 19 percent occurred, and a maximum inside surface strain of 42 percent occurred for the canister orientation with the largest slapdown effect. The important quantity was the maximum midplane strain (19 percent), which controls breach. These strains were below the DOE-selected conservative ultimate strain value of 48 percent.

The Idaho Project Spent Nuclear Fuel canister was based upon the standardized spent nuclear fuel canisters with modifications limited to fabrication details (i.e., wall thickness, welding of the dished head and skirt, etc.). The basic design of the canister is unchanged—particularly the welded skirt at each end of the canister. After establishing the accuracy of finite element analyses to simulate drop tests, drop events similar to those used in the standardized canister were performed. Deformations and maximum plastic strains obtained from these analyses were evaluated to verify that the canister would not breach. Midsurface strains were especially important because they indicated whether any surface damage would propagate through the containment boundary and result in a breach. For the drop event of 80° off vertical, the upper head (due to slapdown) had 48, 24, and 25 percent maximum surface, midsurface, and inner surface strains, respectively. In all cases, the maximum plastic strains of the containment pressure boundary were below the failure strain.

The standardized canister and the multi-canister overpack were compared because both canisters have the same diameter, nominal wall thickness, and approximately the same length. However, the standardized canister has an energy-absorbing skirt with flanged and dished heads, while the multi-canister overpack has a thick flat bottom and is approximately twice as heavy as the standardized canister. For the 7-m [23-ft] repository drop, the plastic strains are comparable for the two canisters. The multi-canister overpack had strains of 5 percent at both the outside and inside surfaces. The standardized canister had outside and inside surface strains of 6 and 4 percent, respectively. However, a comparison was made for the critical case of 7° off vertical of the standardized canister versus the 3° off vertical multi-canister overpack orientation; the multi-canister overpack had 34- or 35-percent strain in the containment boundary, while the standardized canister had less than 1-percent strain. This shows once again that the skirt of the standardized canister does absorb a significant amount of energy and protects the containment boundary. Nevertheless, the predicted plastic strains in the multi-canister overpack are below the minimum elongation strain of 47 percent and minimum fracture strain of 118 percent.

High-level waste canisters were evaluated for their structural behavior when subject to a drop event. Pacific Northwest Laboratories performed full-scale impact tests on canisters that contained high-level waste glass. The tests consisted of a vertical bottom drop where the bottom contacts the surface and a vertical top drop where the corner of the top shoulder and fill nozzle contacts the impact surface. Both tests were oriented such that the center of gravity of the canister is over the point of impact. The test canisters were constructed of titanium or stainless steel. The bottom drop of the titanium and stainless steel canisters showed asymmetric deformation of the canister bottom with no failure. However, for the case of the top drop, the titanium canister had a visible failure that appeared to be due to tearing around the fill nozzle. A similar test at the same orientation for the stainless steel canister did not have the tearing failure. The difference in ductility between titanium and stainless steel may have attributed to the failure.

Finite element studies of proposed canister designs for disposal of high-level waste were evaluated. The 610- and 1,676-mm [24- and 66-in] dished bottom canisters have an energy-absorbing skirt attached to the bottom, and one 1,676-mm [66-in] canister had a flat bottom. The canisters with the skirt had maximum surface strains of 65 to 48 percent for the 610-mm [24-in] and 1,676-mm [66-in] dished bottom, respectively. These strains are below the 60- to 80-percent maximum strain. One design in particular—the 1,676-mm [66-in] diameter, flat bottom canister—appears to be somewhat similar to the proposed transportation, aging, and disposal canister. Current specifications proposed for the transportation, aging, and disposal canister are a flat bottom; a diameter of approximately 1,676 mm [66 in]; and a length of 5.5 m [18 ft] as specified in the Revision B of the Preliminary Transportation, Aging, and Disposal Canister System Performance Specification. The 1,676-mm [66-in] flat bottom canister (with no skirt) had a maximum surface strain of 72 percent, which is in between the 60 to 80 percent maximum strain. Midsurface strains for all canisters averaged 30 percent, which is well below failure; therefore, no surface cracks would propagate through the thickness.

The analyses described here have demonstrated that the various canister types can withstand dynamic loads due to drop events of 9 m [30 ft] or below. These canisters were evaluated in terms of changes in geometry, maximum equivalent plastic strains, and their ability to pass a leak test. The standardized canister, for example, utilized longitudinally welded 316L stainless steel pipe as opposed to seamless pipe, and there are welds where the skirt joins the canister body. Using this type of pipe provided very limited experimental testing of the welds. The mechanical properties of the welds and the factors that may affect the weld were discussed. The fracture toughness of welds was discussed, including one method used to measure toughness: the Charpy V-Notch impact energy, which may provide estimates of the weld open surface area.

Extensive results and discussion with regard to full-scale testing and finite element simulations of canister drop events has been documented. This review has demonstrated that finite element analysis can be used to accurately analyze canister behavior for other loading scenarios such that full-scale testing may not be required. Knowledge gained from the full-scale testing and the corresponding finite element analyses of the canisters in this review should prove useful when evaluating the proposed transportation, aging, and disposal canisters for the Yucca Mountain Repository.

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