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Hanford Sitewide Transportation Safety Document

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U.S. Department of Energy Richland

Date Published
June 2002

Prepared for the U.S. Department of Energy
Assistant Secretary for Environmental Management

Project Hanford Management Contractor for the
U.S. Department of Energy under Contract DE-AC06-96RL13200



**United States
Department of Energy**

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SITE TRANSPORTATION SAFETY DOCUMENT (TSD)**

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02-ABD-0085

APR - 4 2002

Mr. E. K. Thomson, President
Fluor Hanford, Inc.
Richland, Washington 99352

Dear Mr. Thomson:

**CONTRACT NO. DE-AC06-96RL13200 - APPROVAL OF THE HANFORD SITE
TRANSPORTATION SAFETY DOCUMENT (TSD)**

The attached Hanford Site TSD, DOE/RL-2001-0036, for onsite packaging and transportation activities has been approved by RL as meeting the requirements of 10 CFR 830.207(a) and DOE O 460.1A, Attachment 1, Criterion 5. The TSD was prepared using a collaborative approach involving FHI, BMI, BHI, CHG and RL staff. The basis for approval is documented in the attached Safety Evaluation Report.

Onsite packaging and transportation activities may be conducted using the methodology described in the TSD without further RL approval unless the proposed activity results in an Unreviewed Safety Question (USQ) using the methodology discussed in the TSD. Modifications to approved risk based packages resulting in an USQ shall be submitted to RL for approval. Risk based packages included in the TSD are approved. New risk-based packages must be submitted to RL for approval prior to use. Modifications to approved risk based packages resulting in a negative USQ determination per the methodology in the TSD shall be submitted to RL for information until further notice. The submittals should include documentation addressing all aspects of the risk-based USQ methodology described in the TSD.

In lieu of using the TSD for onsite shipments, Hanford contractors may elect to submit all packaging not meeting Department of Transportation requirements to RL for approval or submit an alternative TSD for RL approval in accordance with 10 CFR 830.207(a) and DOE O 460.1A, if deemed to be more cost effective than implementation of the TSD. The basis for such a determination shall be submitted to RL for approval prior to developing an alternative TSD.

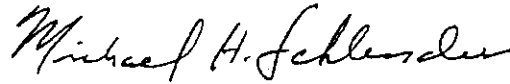
Mr. E. K. Thomson
02-ABD-0085

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If you have any questions, you may contact me, or your staff may contact Richard Self,
Authorization Basis Division, on (509) 372-2863.

Sincerely,



Keith A. Klein
Manager

(for)

ABD:RJS

Attachments

**Safety Evaluation Report
Hanford Transportation Safety Document
DOE/RL-2001-0036, Revision 0**

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1.0 EXECUTIVE SUMMARY

This safety evaluation report (SER) documents the basis for the U.S. Department of Energy (DOE), Richland Operations Office (RL) to approve the Hanford Sitewide Transportation Safety Document (TSD) for onsite Transportation and Packaging (T&P) at Hanford. Hanford contractors, on behalf of DOE-RL, prepared and submitted the Hanford Sitewide Transportation Safety Document, DOE/RL-2001-0036, Revision 0, (DOE/RL 2001), dated October 4, 2001, which is referred to throughout this report as the TSD.

In the context of the TSD, Hanford onsite shipments are the activities of moving hazardous materials, substances, and wastes between DOE facilities and over roadways where public access is controlled or restricted and includes intra-area and inter-area movements. The TSD sets forth requirements and standards for onsite shipment of radioactive and hazardous materials and wastes within the confines of the Hanford Site on roadways where public access is restricted by signs, barricades, fences, or other means including road closures and moving convoys controlled by Hanford Site security forces.

The Hanford Sitewide TSD was developed in accordance with the 10 CFR 830 (CFR 2001a) "safe harbor" Documented Safety Analysis (DSA) methodology for transportation activities, DOE Order 460.1A, *Packaging and Transportation Safety* (DOE 1996c), and DOE-STD-3009, *Preparation Guide for U. S. Department of Energy Non-reactor Nuclear Facility Safety Analysis Reports* (DOE 2000). The Hanford TSD was prepared in coordination with all Hanford Site contractors, DOE-RL, and DOE-Office of River Protection (-ORP). The Hanford TSD complies with 10 CFR 830 as a Documented Safety Analysis (DSA) for onsite transportation and packaging activities at Hanford. This TSD can be used by all tenants at the Hanford Site. Site tenants may also elect to develop their own Rule-compliant TSD, which will require separate DOE approval.

The format and content of this Safety Evaluation Report is based on the guidance provided in DOE-STD-1104-96, *Review and Approval of Nonreactor Nuclear Facility Safety Analysis Reports* (DOE 1996b). This approval is for Revision 0 to Hanford TSD and TSRs (DOE/RL 2001) dated March 11, 2002.

The TSD and TSRs was prepared to comply with the requirements of 10 CFR 830 (CFR 2001a), Subpart B. Consistent with Table 2, Appendix A of 10 CFR 830, Subpart B, the TSD was produced using "safe harbors" following the methodology presented in DOE-O 460.1A *Packaging and Transportation Safety* (DOE 1996c), including guidance from DOE-G 460.1-1, *Implementation Guide for Use with DOE O 460.1A, Packaging and Transportation Safety* (DOE 1997a). The Risk-Based package and transportation analysis followed the analytical methodology, and is compared to the Evaluation Guideline in DOE-STD-3009, *Preparation Guide for U. S. Department of Energy Non-reactor Nuclear Facility Safety Analysis Reports* (DOE 2000).

On February 5th, 2002, the DOE-RL issued the *Fluor Hanford Nuclear Safety Basis Strategy and Criteria* (DOE/RL 2002), which define the following consequence level guidelines:

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Table 1 – Safety Basis Criteria Consequence Levels (Bins)

Consequence	Public	Worker at 100 meters
High	>25 rem TEDE or >ERPG-2 / TEEL-2	>100 rem TEDE or >ERPG-3 / TEEL-3 @ facility boundary or prompt death to facility worker
Moderate	>1 rem TEDE or >ERPG-1 / TEEL-1	>25 rem TEDE or >ERPG-2 / TEEL-2 @ facility boundary, or serious injury to facility worker, or significant radiological or chemical exposure to facility workers
Low	<moderate consequences	<moderate

The consequence guidelines are incorporated into the following risk matrix to assist in identifying the most significant risk to designated receptors:

Table 2 - Risk Classes-Frequency versus Consequences

Consequence	Frequency Beyond 100 meters Below 100 meters	Frequency Beyond 100 meters Below 100 meters	Frequency Beyond 100 meters Below 100 meters	Frequency Beyond 100 meters Below 100 meters
High	Risk Class III	Risk Class II	Risk Class I	Risk Class I
Moderate	Risk Class IV	Risk Class III	Risk Class II	Risk Class I
Low	Risk Class IV	Risk Class IV	Risk Class III	Risk Class III

In addition to the Evaluation Guideline in DOE-STD-3009 (DOE 2000), the Risk-Based accident analyses are compared to the Hanford Safety Basis Criteria (DOE/RL 2002) to identify the level of risk associated with Risk-Based transportation activities, and assure sufficient controls are incorporated to maintain risk within DOE-RL expectations.

The TSD identifies three categories of packages, all of which achieve "equivalent safety" as required by DOE Order 460.1A (DOE 1996c) for onsite shipments. This includes two categories that specifically meet the intent of 10 CFR 71. The third category of packages used for onsite shipments relies on a Risk-Based approach consistent with DOE-STD-3009-94 (DOE 2000) to demonstrate an equivalent level of safety for the public and an acceptable level of safety for onsite workers:

1. **DOT Compliance:** For all non-radioactive and most radioactive hazardous materials, equivalent safety is achieved by implementing the U.S. Department of Transportation (DOT) regulations onsite. No specific accident analyses is required or provided in the TSD for shipments performed under DOT compliance.

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2. DOT-Equivalent Packaging: For radioactive materials, where full compliance with DOT regulations cannot be achieved, an equivalent method for achieving the accepted national level of safety will be implemented. When full compliance with DOT cannot be achieved because of technical or economic conditions, meeting site-specific standards and performance requirements is the preferred technique for achieving safety equivalent to that resulting from following DOT regulations when shipping in-commerce. No specific accident analyses are provided in the TSD for shipments performed with DOT-Equivalent Packaging. Rather, deterministic analyses are conducted to support Hanford Site performance-based certification. DOT-Equivalent packages are required to meet TSD-specified design and performance requirements that are equivalent to DOT-compliant packages for Hanford Site conditions. Package performance may be demonstrated by testing or analysis.

3. Non-equivalent Packaging (Risk-Based Packaging): When full compliance with DOT regulations or compliance with DOT-equivalent packaging for radioactive materials cannot be achieved, a Risk-Based method for demonstrating the accepted level of safety will be implemented. This technique must demonstrate safety equivalent to that resulting from following DOT regulations when shipping in commerce through the use of an approved DOE-RL and DOE-ORP Risk-Based approach, consistent with DOE-STD-3009-94. Specific bounding accident scenarios are evaluated to demonstrate that use of Risk-Based packaging does not exceed the DOE-STD-3009-94 Evaluation Guideline and the Consequence Guidelines of the Safety Basis Criteria (DOE/RL 2002). No safety-class structures, systems, or components (SSC's) are associated with Risk-Based packaging. Risk-Based packaging accidents are summarized below:

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Table 3 – Risk-Based Package Accident Summary – 100/200 Areas

Scenario	Inventory A ₁ (Curies)	Analysis Type	Frequency	Public Consequences		100 m Worker Consequences		Credited Controls
				Dose (rem)	Risk Class	Dose (rem)	Risk Class	
TRU (Pu - Alpha)								
Collision and Spill (Entire load)	30,000 A ₂ (162 Ci Pu-239)	Unmitigated	Extremely Unlikely	0.1	IV	170	II	
		Mitigated	Extremely Unlikely	0.01	IV	17	IV	(a) and (b)
Partial Truck Fire* (30% of load)	Involves 30% of load 9,000 A ₂ (48.6 Ci Pu-239)	Unmitigated	Unlikely	0.2	III	11	III	
		Mitigated	Unlikely	0.2	III	11	III	None
Collision and Fire (Entire load)	30,000 A ₂ (162 Ci Pu-239)	Unmitigated	Beyond Extremely Unlikely	0.7	IV	36	IV	
		Mitigated	Beyond Extremely Unlikely	0.7	IV	36	IV	None
Hydrogen Deflagration (Single Package)	3,000 A ₂ (16.2 Ci Pu-239)	Unmitigated	Anticipated	0.7	III	38	I	
		Mitigated	Extremely Unlikely	0.7	IV	38	III	(c) and (d)
Non-TRU (Cs, Sr - Beta-Gamma)								
Collision and Spill (Entire load)	30,000 A ₂ (405,000 Ci Cs-137 or 81,000 Ci Sr-90)	Unmitigated	Extremely Unlikely	0.1	IV	145	II	
		Mitigated	Extremely Unlikely	0.01	IV	14	IV	(a) and (b)
Partial Truck Fire* (30% of load)	Involves 30% of load 9,000 A ₂ (121,500 Ci Cs-137 or 24,300 Ci Sr-90)	Unmitigated	Unlikely	0.2	III	9	III	
		Mitigated	Unlikely	0.2	III	9	III	None
Collision and Fire (Entire load)	30,000 A ₂ (405,000 Ci Cs-137 or 81,000 Ci Sr-90)	Unmitigated	Beyond Extremely Unlikely	0.6	IV	30	IV	
		Mitigated	Beyond Extremely Unlikely	0.6	IV	30	IV	None
Hydrogen Deflagration (Single Package)	3,000 A ₂ (40,500 Ci Cs-137 or 8,100 Ci Sr-90)	Unmitigated	Anticipated	0.6	III	32	I	
		Mitigated	Extremely Unlikely	0.6	IV	32	III	(c) and (d)

* - This accident was not included in the submitted TSD. It is directed to be included in the TSD by this SER.

Credited Controls:

- (a) Maximum speed limited to 35 mph.
- (b) Sandbags or non-rad (dunnage) containers in front row of array, high dose equivalent-Curie (DE-Ci) containers in rear
- (c) Containers vented prior to shipment
- (d) Chemically reactive contents not present

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Table 4 – Risk-Based Package Accident Summary – 300 Area

Scenario	Inventory A ₁ (Curies)	Analysis Type	Frequency	Public Consequences		100 m Worker Consequences		Credited Controls
				Dose (rem)	Risk Class	Dose (rem)	Risk Class	
TRU (Pu - Alpha)								
Collision and Spill (Entire load)	10,000 A2 (54.1 Ci Pu-239)	Unmitigated	Unlikely	4.9	II	33	II	
		Mitigated	Unlikely	0.5	III	3	III	(a) and (b)
Partial Truck Fire* (30% of load)	Involves 30% of load 3,000 A2 (16.2 Ci Pu-239)	Unmitigated	Unlikely	0.7	III	4	III	
		Mitigated	Unlikely	0.7	III	4	III	None
Collision and Fire (Entire load)	10,000 A2 (54.1 Ci Pu-239)	Unmitigated	Extremely Unlikely	2.3	III	12	IV	
		Mitigated	Extremely Unlikely	2.3	III	12	IV	None
Hydrogen Deflagration (Single Package)	940 A2 (5.1 Ci Pu-239)	Unmitigated	Anticipated	2.3	I	12	III	
		Mitigated	Extremely Unlikely	2.3	III	12	IV	(c) and (d)
Non-TRU (Cs, Sr - Beta-Gamma)								
Collision and Spill (Entire load)	10,000 A2 (135,000 Ci Cs-137 or 27,000 Ci Sr-90)	Unmitigated	Unlikely	4.2	II	28	II	
		Mitigated	Unlikely	0.4	III	3	III	(a) and (b)
Partial Truck Fire* (30% of load)	Involves 30% of load 3,000 A2 (40,500 Ci Cs-137 or 8,100 Ci Sr-90)	Unmitigated	Unlikely	0.6	III	3	III	
		Mitigated	Unlikely	0.6	III	3	III	None
Collision and Fire (Entire load)	10,000 A2 (135,000 Ci Cs-137 or 27,000 Ci Sr-90)	Unmitigated	Extremely Unlikely	2.0	III	10	IV	
		Mitigated	Extremely Unlikely	2.0	III	10	IV	None
Hydrogen Deflagration (Single Package)	940 A2 (12,690 Ci Cs-137 or 2,538 Ci Sr-90)	Unmitigated	Anticipated	2.0	I	10	III	
		Mitigated	Extremely Unlikely	2.0	III	10	IV	(c) and (d)

* - This accident was not included in the submitted TSD. It is directed to be included in the TSD by this SER.

Credited Controls:

- (a) Maximum speed limited to 35 mph.
- (b) Sandbags or non-rad (dunnage) containers in front row of array, high dose equivalent-Curie (DE-Ci) containers in rear
- (c) Containers vented prior to shipment
- (d) Chemically reactive contents not present

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For Risk-Based packages, the TSD establishes requirements for the Transportation Unreviewed Safety Question (USQT) process to determine the approval authority for proposed changes to Risk-Based transportation activities and potential inadequacies in the TSD Risk-Based safety analyses. The USQT Process for Risk Based packages was evaluated and deemed compliant with 10 CFR 830 (CFR 2001a), Subpart B. It is noted that the guidance in TSD Appendix C, *Unreviewed Safety Question Process For Transportation* does not specifically address implied margins of safety. However, the TSD was prepared and submitted to the DOE for approval prior to issuance of DOE G 424.1-1, *Implementation Guide For Use In Addressing Unreviewed Safety Question Requirements* (DOE 2001), on October 24, 2001. This is addressed in the Technical Direction provided in Appendix B of this SER.

Package Specific Safety Documents (PSSDs) are identified in TSD Appendix A. Table 1 of TSD Appendix A lists existing Hanford Site PSSDs in use at the time the TSD was prepared. These PSSDs, at the revision level identified, are part of the TSD and are approved for continuing use. Documents that were previously used and do not appear on TSD Appendix A, Table 1, are not approved for use after the effective date of the TSD. Table 2 of TSD Appendix A lists packagings approved following the requirements of the TSD. Each table is divided into four parts. Part One identifies those packagings compliant with U.S. DOT packaging requirements. Part Two identifies those packagings judged to meet the DOT-equivalent packaging standards. Part Three contains those packagings judged to meet Risk-Based packaging standards. Part Four contains exemptions to the TSD requirements authorized by the DOE.

DOE-RL has determined that the TSD provides sufficient hazard analysis and controls to assure adequate safety of onsite shipments, and will reduce the risk to the Offsite Public and Onsite Workers to levels within the guidelines provided in DOE Order 460.1A (DOE 1996c), the Evaluation Guideline in DOE-STD-3009 (DOE 2000), and the Consequence Guidelines of the Hanford Safety Basis Criteria (DOE/RL 2002). DOE-RL concludes that the TSD, with incorporation of the Technical Direction provided in Appendix A of this SER, complies with DOE Order 460.1A (DOE 1996c), DOE-STD-3009 (DOE 2000), and Subpart B to 10 CFR 830 (CFR 2001a).

2.0 REVIEW PROCESS

Staff of the Authorization Basis Division in the Office of Assistant Manager for Safety and Engineering and DOE-RL Program Offices performed the review. For Risk-Based packages the DOE-RL review was performed consistent with DOE Standard 1104-96, *Review and Approval of Non-Reactor Facility Safety Analysis Reports* (DOE 1996b). Compliance was determined by comparing the appropriate section of the TSD or TSR to the specific federal citation using the review criteria contained in Section 1.4, "Issue Origination and Resolution" of that Standard. A review plan was not prepared.

Hanford contractors, on behalf of DOE-RL, prepared and submitted the TSD, DOE/RL-2001-0036, Revision 0 (DOE/RL 2001), dated March 11, 2002, to demonstrate compliance with 10 CFR 830 (CFR 2001a), Subpart B. This SER was prepared to evaluate the degree to which the TSD complies with 10 CFR 830, Subpart B, to document how assurance of such compliance was performed, and to document the basis for approval by the DOE-RL.

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The TSD was prepared in accordance with the 10 CFR 830 (CFR 2001a) "safe harbor" approach of DOE-O 460.1A, *Packaging and Transportation Safety* (DOE 1996c), including guidance from DOE-G 460.1-1, *Implementation Guide for Use with DOE O 460.1A, Packaging and Transportation Safety* (DOE 1997a), and in accordance with DOE-STD-3009, *Preparation Guide for U. S. Department of Energy Non-reactor Nuclear Facility Safety Analysis Reports* (DOE 2000). The review ensured that the DSA complies with all applicable regulations in Subpart B to 10 CFR 830, including compliance to "safe harbor" methodology, and therefore:

- meets the applicable requirements using the equivalent safety standard as defined in DOE Order 460.1A (DOE 1996c);
- identifies and addresses hazards associated with Risk-Based package and transportation activities in sufficient detail to show that the associated hazard controls are adequate to protect the workers, the public, and the environment, per the applicable requirements of DOE-STD-3009 (DOE 2000).

For Risk-Based packages and transportation activities, the TSD also establishes requirements for a transportation-specific USQ_T process to evaluate all proposed changes to Risk-Based transportation activities and potential inadequacies in the Risk-Based portion of the TSD and supporting safety analyses to determine the appropriate approval authority. The USQ_T process was also evaluated and deemed compliant with 10 CFR 830 (CFR 2001a), Subpart B.

The following documents were reviewed:

- DOE/RL-2001-0036, Hanford Sitewide Transportation Safety Document, Revision 0, dated October 4, 2001 (DOE/RL 2001)
- 10 CFR 71, *Packaging and Transportation of Radioactive Material* (CFR 2001b)
- 10 CFR 830, *Nuclear Safety Management* (CFR 2001b)
- 49 CFR Subchapter C, *Hazardous Materials Regulations* (CFR 2001c)
- DOE-O 460.1A, *Packaging and Transportation Safety* (DOE 1996c)
- DOE-G 460.1-1, *Implementation Guide for Use with DOE O 460.1A, Packaging and Transportation Safety* (DOE 1997a)
- DOE-STD-3009-94, *Preparation Guide for U. S. Department of Energy Non-reactor Nuclear Facility Safety Analysis Reports* (DOE 2000)

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3.0 APPROVAL BASIS

The TSD together with programmatic procedures complies with the requirements in 10 CFR 830 (CFR 2001a), Subpart B. The TSD and the Technical Direction provided in this SER provide adequate protection to the site workers, the public, and the environment for onsite transportation of hazardous materials. Consistent with Table 2, Appendix A of 10 CFR 830, Subpart B, the TSD was produced using "safe harbors" following the methodology presented in DOE-O 460.1A *Packaging and Transportation Safety* (DOE 1996c), including guidance from DOE-G 460.1-1, *Implementation Guide for Use with DOE O 460.1A, Packaging and Transportation Safety* (DOE 1997a). The Risk-Based package and transportation analysis followed the analytical methodology, and meets the Evaluation Guideline in DOE-STD-3009, *Preparation Guide for U. S. Department of Energy Non-reactor Nuclear Facility Safety Analysis Reports* (DOE 2000). The Risk-Based package and transportation analysis also meets the *Hanford Safety Basis Criteria* (DOE/RL 2002). Transportation accident analyses may continue to use ICRP 26/30 dose conversion factors for consistency with national transportation standards.

The specified codes and standards listed in TSD Chapter V, *External Regulations*, and TSD Chapter VI, *Site-Specific Standards, Procedures, and Instructions*, are appropriate and consistent with applicable DOE requirements. The TSD reflects an appropriate use of graded approach in the detail and rigor of its analysis for Risk-Based packaging.

For Risk-Based packages and transportation activities, a USQ_T process is established in the TSD, consistent with the requirements of 10 CFR 830.203.

The review and approval of the TSD is established through verification of adequacy of the following five approval bases:

- Base Information
- Hazard and Accident Analyses
- Safety Structures, Systems, and Components (SSCs)
- Derivation of Technical Safety Requirements (TSRs)
- Programmatic Controls

3.1 Adequacy of Base Information

The TSD sets forth requirements and standards for onsite T&P activities. These activities consist of onsite shipments of hazardous materials, substances, and wastes between DOE facilities and over roadways where public access is controlled or restricted by signs, barricades, fences, or other means including road closures and moving convoys controlled by Hanford Site security forces. Onsite shipment activities include intra-area and inter-area movements.

The Hanford Sitewide TSD encompasses all onsite shipments of government-owned hazardous materials, substances, and wastes within the Hanford Site, which includes radioactive materials and wastes. Only materials identified as hazardous by the DOT, and regulated if shipped in commerce, are covered. The TSD does not include movement of these government materials by hand or hand truck.

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DOE issued Subpart B to 10 CFR 830 (CFR 2001a) in January 2001. Subpart B codified as federal regulations previous requirements for safety analysis that were contained in various DOE orders. For DOE transportation activities, 10 CFR 830 identifies DOE Order 460.1A, *Packaging and Transportation Safety* (DOE 1996c), as a "safe harbor" for transportation DSA development methodology. For onsite transfers, DOE Order 460.1A, *Packaging and Transportation Safety*, requires compliance with the Hazardous Materials Regulations, or the site- or facility-specific cognizant Operations or Field Office approved Transportation Safety Document that describes the methodology and compliance process to meet equivalent safety for any deviation from the Hazardous Materials Regulations. The TSD defines the onsite T&P program at the Hanford Site, and the means by which T&P activities comply with the DOE transportation safety requirements specified in DOE Order 460.1A. The T&P program must comply with DOT regulations provided in Title 49 Code of Federal Regulations (CFR) Subchapter C, *Hazardous Materials Regulations* (CFR 2001c), and U.S. Nuclear Regulatory Commission (NRC) standards provided in 10 CFR 71, *Packaging and Transportation of Radioactive Material* (CFR 2001b), or provide an equivalent level of safety.

T & P activities specifically addressed by the TSD include the following:

- Identification of responsibilities, lines of authority, and program approval procedures.
- Definition of minimum safe packaging requirements including necessary design, fabrication, and quality assurance (QA) elements, using appropriate codes and standards.
- Description of the process and analysis used to ensure equivalent safety requirements are established.
- Description of the Hanford Site including maps identifying boundaries, railways, and roadways that clearly delineate offsite and onsite areas, and procedures for establishing access control for any area having occasional public access.
- Provisions for emergency response and recovery under credible accident conditions (AC).
- Process for accomplishing nonroutine T&P activities.
- Personnel Qualification and Training
- Documentation and Record Keeping
- Transport Vehicle Operations

In addition, the TSD formally establishes an Unreviewed Safety Question process specific to T&P activities, the USQ_T process. Consistent with 10 CFR 830, the USQ_T process permits a contractor to make physical and procedural changes to transportation activities without prior DOE approval, provided the changes do not explicitly or implicitly affect the safety basis as represented by the TSD (i.e., result in a negative USQ_T). The USQ_T process provides a means

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for keeping the transportation safety basis current by ensuring changes receive the appropriate level of review and approval. The USQ_T process evaluates all proposed changes to transportation activities, and potential inadequacies in the TSD and supporting safety analyses, to determine if DOE approval is required. The USQ_T process also specifies required actions if a potential inadequacy of the Risk-Based safety analyses is discovered. The process provides a method for contractors to determine if an USQ_T is involved, and the actions to take if the situation involves a positive USQ_T.

3.1.1 Documented Safety Analysis

The Documented Safety Analysis (DSA) for the Hanford Site Transportation consists of the TSD. The TSD defines the accepted level of risk, based on compliance with DOT regulations provided in Title 49 Code of Federal Regulations (CFR) Subchapter C, *Hazardous Materials Regulations* (CFR 2001c) and U.S. Nuclear Regulatory Commission (NRC) standards provided in 10 CFR 71, *Packaging and Transportation of Radioactive Material* (CFR 2001b) or an equivalent level of safety provide by authorized DOT-Equivalent deterministic packages. The Risk-Based package and transportation analysis followed the methodology of DOE-STD-3009, *Preparation Guide for U.S. Department of Energy Non-reactor Nuclear Facility Safety Analysis Reports* (DOE 2000). The USQ_T process provides the means for Hanford Site contractors to determine the approval authority for changes to packages and transportation activities.

The Hanford Sitewide TSD defines the onsite T&P program at the Hanford Site, and represents the DSA required under 10 CFR 830, *Nuclear Safety Management*, for Hanford transportation activities. The sitewide TSD applies to all DOE-RL and DOE-ORP Hanford Site prime contractors and their major subcontractors who perform hazardous materials (hazmat) T&P activities in association with their project or mission.

10 CFR 830 defines nuclear facility broadly to include all those facilities, activities, or operations that involve, or will involve, radioactive and/or fissionable materials in such form and quantity that a nuclear hazard potentially exists to the employees or the general public. As demonstrated in the TSD analysis of Risk-Based packages, accidents associated with onsite shipments of such packages can have measurable offsite consequences, and potentially significant onsite consequences. Therefore, Hanford onsite transportation activities are subject to 10 CFR 830 requirements for nuclear facilities.

The Hanford TSD only applies to onsite T&P activities. For DOT-compliant and DOT-equivalent packages, 10 CFR 830 identifies the "safe harbor" DSA methodology for transportation activities as U.S. Department of Energy (DOE) Order 460.1A, *Packaging and Transportation Safety*, and its associated guidance document. For onsite shipments of hazardous materials, DOE Order 460.1A, states:

"Onsite hazardous materials transfers shall comply with the Hazardous Materials Regulations, or the site- or facility-specific cognizant Operations or Field Office approved Transportation Safety Document that describes the methodology and compliance process to meet equivalent safety for any deviation from the Hazardous Materials Regulations. For multiple-tenant DOE sites, safety documents for several contractor organizations may be combined into a single

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document..."

The TSD identifies three categories of packages, all of which achieve "equivalent safety" as required by DOE Order 460.1A (DOE 1996c) for onsite shipments. This includes two categories that specifically meet the intent of 10 CFR 71. The third category of packages used for onsite shipments relies on a Risk-Based approach consistent with DOE-STD-3009-94 (DOE 2000) to demonstrate an equivalent level of safety for the public and an acceptable level of safety for onsite workers:

1. **DOT Compliance:** For all non-radioactive and most radioactive hazardous materials, equivalent safety is achieved by implementing the DOT regulations onsite. All portions of the regulations, other than those identifying the workings and administration of the DOT (e.g., enforcement) will be implemented. Equivalent administrative activities including approval of exemptions will be conducted by RL and ORP.
2. **DOT-Equivalent Packaging:** For radioactive materials, where full compliance with DOT regulations cannot be achieved, an equivalent method for achieving the accepted national level of safety will be implemented. Before implementing the DOT-equivalent packaging method, a documented evaluation showing that DOT compliance is not technically or economically practical must be prepared. For these radioactive hazardous materials, equivalent safety is achieved by implementing all portions of the regulations, except for regulations identifying the following:
 - Workings and administration of the DOT
 - How to obtain exemptions
 - Design and performance requirements for IP-2, IP-3, Type A, fissile and Type B packaging
 - Communication requirements associated with shipment of radioactive materials

Equivalent administrative activities, including approval of exemptions, will be conducted by RL and ORP. Performance requirements for onsite DOT-equivalent packages are identified in Chapter VI, Section 5.2 of the TSD. These requirements are developed to result in a package that produces physical performance under Hanford Site conditions equivalent to an offsite package subject to regulatory-established test conditions. Examples of Hanford Site-specific conditions that are different than DOT and Nuclear Regulatory (NRC) regulatory conditions are: environmental conditions (Site-specific temperature extremes), application of artificial cooling after a 30 minute fire (assumes Site fire department response), allowance for container venting (not permitted by DOT), and specific free drop characteristics (models the Central Waste Complex (CWC) storage pad). Specific details and justifications for Hanford Site-specific variances from the DOT and NRC regulatory conditions are provided in TSD Appendix B.

When full compliance with DOT cannot be achieved because of technical or economic conditions, meeting these site-specific standards and performance

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requirements is the preferred technique for achieving safety equivalent to that of following DOT regulations when shipping in commerce. DOT-equivalent packages are not used for transportation of non-radioactive hazardous materials.

3. **Non-equivalent Packaging (Risk-Based Packaging):** When full compliance with DOT regulations or compliance with DOT-equivalent packaging for radioactive materials cannot be achieved, a Risk-Based method for demonstrating the accepted national level of safety will be implemented. This technique must demonstrate safety equivalent to that resulting from following DOT regulations when shipping in commerce through the use of an approved DOE-RL and DOE-ORP Risk-Based approach, consistent with DOE-STD-3009-94. Specific bounding accident scenarios are evaluated to demonstrate that use of Risk-Based packaging is equivalent to the accepted national level of safety for DOT-complaint packaging by assuring that the DOE-STD-3009-94 Evaluation Guideline is not exceeded, and the Consequence Guidelines of the Safety Basis Criteria (DOE/RL 2002) are not exceeded. Before implementing the nonequivalent packaging method, a documented evaluation showing that both DOT compliance and DOT-equivalent packaging compliance are not technically or economically practical must be prepared. For these radioactive hazardous materials, equivalent safety is achieved by implementing all portions of the regulations, except for regulations identifying the following:

- Workings and administration of the DOT
- How to obtain exemptions
- Design and performance requirements for IP-2, IP-3, Type A, fissile and Type B packaging
- Communication requirements associated with shipment of radioactive materials

For Risk-Based packages, administrative activities, including the approval of exemptions, will be conducted by DOE-RL and DOE-ORP, as specified in the TSD. Performance requirements for onsite non-equivalent packages are identified in Chapter VI, Section 5.3 of the TSD. These requirements are developed to result in a package, which under Hanford Site conditions will produce radiological performance equivalent to an offsite package subject to regulatory-established tests and conditions. Risk-Based packages are not used for transportation of non-radioactive hazardous materials. The USQ_T process provides the means for Hanford Site contractors to determine the approval authority for changes to Risk-Based packages and transportation activities.

Conclusion: The Base Information presented in the TSD adequately describes the mission and scope of operations for Hanford onsite transportation activities, and specifically addresses the process for development and approval of DOT-equivalent and Risk-Based packages, based on the appropriate standards, guides and requirements. No issues were identified with respect to the TSD Base Information.

3.2 Adequacy of Hazards and Accident Analyses

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The Hanford Sitewide TSD encompasses all onsite shipments of government-owned hazardous materials, substances, and wastes within the Hanford Site, which includes radioactive materials and wastes. Only materials identified as hazardous by the DOT, and regulated if shipped in commerce, are covered. The TSD does not include movement of these government materials by hand or hand truck. This movement must be included in the contractor's industrial safety and health program (29 CFR 1910/1026, "Occupational Safety and Health Administration") and/or Hanford Safety Basis documents.

For all nonradioactive materials, equivalent safety is achieved by implementing the DOT regulations onsite. Therefore, specific consequence analyses are not provided for accidents involving nonradioactive hazardous materials.

Compliance with DOT requirements for shipments of radioactive materials in Type B packages (i.e., packages designed and constructed to contain greater than Type A quantities) requires implementation of applicable DOT regulations, or onsite equivalent requirements. Specifically, performance tests for DOT-equivalent packaging are the same as those specified in the regulations, except in some conditions the alternate Hanford Site-specific conditions are applied. The physical tests, documentation, and approval requirements for DOT-compliant and DOT-equivalent Type B packages assure an equivalent level of safety based on package performance under normal conditions to transport and credible accident conditions. Therefore, specific hazard and accident analyses are not provided for accidents involving DOT-compliant and DOT-equivalent packages.

3.2.1 Source Terms

Compliance with DOT requirements for shipments of radioactive materials in Type A packages is based, in part, on the A_1 and A_2 values of radionuclide inventories defined in 49 CFR § 173.435, or derived in accordance with the procedure prescribed in 49 CFR § 173.433. As defined by 49 CFR § 173, A_1 values are the maximum activity of special form Class 7 (radioactive) material permitted in a Type A package. A_2 values are the maximum activity of Class 7 (radioactive) material, other than special form, low-specific activity (LSA) or surface contaminated object (SCO), permitted in a Type A package.

Packages shipped onsite that contain greater than 1 A_2 value (i.e., Type B quantities) that do not meet DOT or DOT-equivalent Type B requirements must meet the TSD requirements specific to non-equivalent (Risk-Based) packaging. Such package/load configurations do not fully comply with DOT requirements, and are not permitted to be shipped "in commerce" (i.e., offsite).

For Risk-Based packages, the TSD originally proposed a public intake acceptance criterion of $1.0\text{E-}06 A_2$, based on the acceptable leakage rate associated with a leaking package during NCs at the location of the maximally exposed member of the public. In the derivation of the release rate from Type B packages during NCs of transport (IAEA 1990, p. 107-108), the release of $1.0\text{E-}06 A_2$ per hour is based on a member of the public spending 20% of the working time ($400 \text{ h/yr} = 20\% \times 2,000 \text{ h/yr}$) around a leaking certified Type B package in an enclosed vehicle of 50 m^3 volume with 10 air changes per hour. This scenario results in a maximum public intake over the course of one year of $1.0\text{E-}06 A_2$ from NCs.

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The TSD submitted for approval on October 4, 2001, provided the Risk-Based package analysis presented in TSD Table 6-7, *Dose Consequence and Release Frequency from Packages in Transportation*. This table establishes the maximum A_2 content for individual Risk-Based packages and the total content permitted in shipments of Risk-Based packages, which specifically serve to reduce the consequences to Onsite Workers. For Risk-Based packages containing transuranic (TRU) materials (e.g., Plutonium and Americium isotopes), the maximum content for an individual package is 3,000 A_2 values (16.2 Ci Pu-239) in the Hanford 100 and 200 areas, and 940 A_2 values (5.1 Ci Pu-239) in the Hanford 300 area. The total conveyance limit for TRU materials is 30,000 A_2 values (162 Ci Pu-239) in the Hanford 100 and 200 areas, and 10,000 A_2 values (54.1 Ci Pu-239) in the Hanford 300 area. For Risk-Based packages containing non-TRU materials (e.g., Sr-90 and Cs-137), the maximum content for an individual Risk-Based package is 940 A_2 values (12,690 Ci Cs-137 or 2,538 Ci Sr-90) in the Hanford 300 area. The total A_2 conveyance limit for non-TRU materials is 10,000 A_2 values (135,000 Ci Cs-137 or 27,000 Ci Sr-90) in the Hanford 300 area.

For Risk-Based packages containing non-TRU materials (e.g., Sr-90 and Cs-137), the TSD submitted for approval on October 4, 2001, proposed a maximum content for an individual Risk-Based package of 6,000 A_2 values (81,000 Ci Cs-137 or 16,200 Ci Sr-90) in the Hanford 100 and 200 areas. The total conveyance limit for non-TRU materials was submitted as 60,000 A_2 values (810,000 Ci Cs-137 or 162,000 Ci Sr-90) in the Hanford 100 and 200 areas. However, as discussed below, for accidents involving Risk-Based packages in the Hanford 100 and 200 areas these proposed A_2 values resulted in consequences that were deemed unacceptable by the DOE-RL. Therefore, the DOE-RL determined that appropriate limits for the 100 and 200 areas are 3,000 A_2 values (40,500 Ci Cs-137 or 8,100 Ci Sr-90) for an individual package and 30,000 A_2 values (405,000 Ci Cs-137 or 81,000 Ci Sr-90) for a conveyance limit. This is addressed in the Technical Direction provided in Appendix A of this SER.

The proposed Public intake acceptance criterion of $1.0\text{E-}06$ A_2 for Risk-Based packages is equivalent to 2.3 rem for Pu-239 (Solubility Class W). However, the distance between the public and a Risk-Based package on the Hanford site can vary from 300 meters (300 Area) to 11.7 kilometers (200 Area). Long distances to the public can substantially reduce the initial source term from an accident by several orders of magnitude through atmospheric dilution. Using this approach, the potential dose to Onsite Workers can be orders of magnitude higher than to the Public. For example, an accident in the 200 Area of the Hanford site that results in a dose of 2.3 rem to the Public could result in a dose of 3,785 rem to the Onsite Worker at 100 meters from the accident. Given the potential for unacceptable consequences to the Onsite Worker, the DOE-RL did not accept the public intake criterion of $1.0\text{E-}06$ A_2 as sufficient to protect onsite individuals. Though risk-based packaging must, at least, still meet the public intake criteria of $1.0\text{E-}06$ A_2 , DOE-RL determined that specific Risk-Based accident analyses performed in accordance with DOE-STD-3009 (DOE 2000) were required to demonstrate an acceptable level of risk to the Onsite Worker.

DOT-compliant and DOT-equivalent packages are designed and tested to meet specific performance criteria related to normal conditions (NCs) to transport and credible accident conditions (ACs). 10 CFR 71.51 requires that certified Type B packages be designed,

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constructed, and prepared for shipment so that under the tests specified in 10 CFR 71.71, "Normal conditions of transport," there would be no loss or dispersal of radioactive contents as demonstrated to a sensitivity of $1.0E-06$ A₂ per hour. It is possible that some Risk-Based packages fail only the DOT containment requirement for ACs; i.e., they may meet the DOT containment requirements for NCs. The NC tests specified in 10 CFR 71.71 include minimum and maximum ambient temperatures, minimum and maximum external pressures, maximum insolation, vibration, water spray, free drop, corner drop, and penetration.

3.2.2 Hazards Summary

For all nonradioactive materials, equivalent safety is achieved by implementing the DOT regulations onsite. The primary hazard associated with Hanford onsite transportation activities is the uncontrolled release of radioactive materials in quantities that exceed those expected for DOT-compliant packages. Risk-based packages, by definition do not comply with DOT requirements and are not equivalent to DOT-compliant packages. Therefore, breaches of Risk-Based packages are the greatest hazard addressed by the TSD.

A hazard analysis was conducted for Risk-Based packages to identify and evaluate the hazards associated with transporting radioactive material on the Hanford Site. The purpose of the hazard analysis was to review potential sources of hazards during transport and construct potential accident scenarios that could be applied to the shipments and packagings and lead to loss or dispersal of radioactive materials. A generic hazard "checklist" was used to guide the analysis and ensure a comprehensive review of the potential hazards. The results of the hazard analysis are documented in TSD Appendix G, "*Justification And Basis For Shipment Of Risk-Based Packages.*"

- Thermal energy from radioactive decay of the cargo must be dissipated to prevent package failure regardless of whether the package is equivalent or nonequivalent. Any engineered or administrative controls implemented to ensure adequate heat dissipation (e.g., source limits, cooling systems) must be adhered to.
- Extreme ambient conditions (heat and cold) that could lead to package failure were not included in the Hanford Site accident risk framework because they are already included in the approach for equivalent packages.
- Potential explosions were judged extremely unlikely as long as the requirements to prevent explosions for equivalent packages are adhered to.
- Immersion of a package under a head of water and subsequent failure due to the hydrostatic pressure load was judged to be incredible because the shipments would not be transported over or near a body of water of sufficient depth to be an immersion concern or a concern about retrieving the package in a timely manner.
- Potential energy hazards (e.g., package falling from a bridge or overpass and heavy objects being dropped on a package) were determined to be incredible. There are no bridges or overpasses on the Hanford Site that a package could fall from and heavy object lifting requirements preclude them from being lifted over a radioactive material package.

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except under controlled and previously approved conditions.

- Toxic and hazardous chemicals are, in general, not a threat to Hanford Site packages. The chemicals may be an issue related to industrial hygiene and worker protection, but do not represent a credible threat to the package. Similarly, corrosion was removed as a potential accident scenario because of the long-term nature of the threat and the existing package maintenance program in which inspections are conducted periodically to identify potential corrosion or other forms of package degradation.
- Potential acts of sabotage or terrorism were not considered further because of the absence of credible threats. Hanford shipments are not considered attractive to potential saboteurs and terrorists. Furthermore, the Hanford Site is provided with its own security forces that are capable of rapidly responding to and dealing with potential saboteurs and terrorists.
- Natural phenomena threats to packages include earthquakes, high winds, tornadoes, and floods. However, the likelihood of such events are low and the likelihood that such an event occurs at the same time a package is in transit, combined with the likelihood that the natural phenomena event is severe enough to threaten package integrity makes these incredible occurrences. Furthermore, shipping is restricted during periods of highly hazardous weather, such as dust storms and high winds that decrease visibility and snowstorms that lead to dangerous road conditions.

Based on the TSD hazards analysis for Risk-Based packages, the TSD addressed potential ACs including impact (such as vehicular crashes and rollovers), puncture, crush, and fire. These events are the same as the hypothetical ACs defined in 10 CFR 71. Package requirements and controls were developed based on the deterministic analyses conducted to support traditional package certification or Hanford Site performance-based certification.

3.2.3 Criticality

The TSD requires criticality accidents to be demonstrated to be not credible under all postulated NCs and ACs. The sequence of events leading to a potential criticality accident was judged to be incredible due to the engineering and administrative controls in place to prevent criticality accidents. These include, where appropriate, limiting the fissile inventory, preventing ingress of moderator materials (such as water) into the package, geometry controls, adding poisons, etc. These requirements remain in effect for equivalent as well as nonequivalent packages. Therefore, consequences of a criticality accident were not specifically calculated.

3.2.4 Worker Safety Summary

Protection of occupational workers from radiological hazards as a result of the packaging and transportation of radioactive materials is achieved by compliance with the TSD and the TSRs contained therein. Each Hanford Site contractor must implement a comprehensive Integrated Environment, Safety and Health Management System (ISMS) Plan and assess onsite T&P activities against ISMS requirements.

There are no programmatic controls discussed in the TSD as specifically providing protection to

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the workers from transportation hazards and credible accidents. Protection from standard industrial hazards is achieved by adherence to the requirements of 29 CFR 1910.

3.2.5 Accident Summary

For Risk-Based packages that weigh 500 kg (1,102 lb) or less, the TSD requires evaluation of the frequencies and consequences of eight accident scenarios that could lead to package failure and release of radioactive materials:

1. Impact-only
2. Impact followed by fire
3. Puncture-only
4. Puncture followed by fire
5. Crush-only
6. Crush followed by fire
7. Fire-only
8. Hydrogen deflagration.

For packages that weigh greater than 500 kg (1,102 lb), accident scenarios that involve crush-only and crush followed by fire need not be evaluated, in accordance with 10 CFR 71.

Collision is the most common type of transportation accident. In a severe collision accident, the truck hits an obstacle, causing the truck to decelerate at a rapid rate. The package, however, is still traveling at the initial velocity of the truck. The velocity of the package is reduced by the amount of energy required to break the tiedown hardware (which is small). The package then collides with the front of the conveyance, which had stopped because of the collision with the obstacle. The collision between the package and the conveyance is an impact event because the force on the package is localized on one side; this is the same type of force created when an object is dropped and hits the ground. If there are additional packages on the conveyance that are behind the front package, then an inertial crush force is generated as these packages collide with and push against the front package, which is being resisted by the front of the conveyance. Impact and crush forces are differentiated by the application of the force to the package. If the force occurs on one side of the package, it is considered an impact, and if it occurs on multiple sides of the package, it is considered crush.

Puncture is a type of impact in which a small diameter probe impacts the package. The TSD evaluates this type of event as occurring less frequently than a collision. The TSD considers the release from a puncture event to be bounded by the release from the impact described above. Therefore, a puncture event as a result of a collision is not specifically evaluated in the TSD.

A fire may follow a collision. Transportation statistics used for the TSD show that fire occurs in approximately 1.6% of all accidents. These fires have variable diameter, temperature, duration, and location relative to the cargo. The TSD states that un-vented Type A 210 L (55-gal) drums, on average, will lose containment 3 minutes after exposure to a fully-engulfing fuel fire. The TSD evaluates a post-collision fire that is assumed to involve the entire inventory of the conveyance.

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Hydrogen gas deflagrations may occur anytime, not just during transportation. Hydrogen can build up in a package due to chemical reactions between different materials of the payload and due to radiolytic decomposition of water or other hydrogenous materials. Another flammable gas (methane) can build up due to the anaerobic respiration by organisms.

Vehicle fires can occur due to electrical or fuel system malfunctions that are unrelated to a traffic accident. The TSD did not specifically evaluate such fires. Because this type of fire is not associated with a severe accident, and would therefore only be expected to involve a limited amount of fuel or combustible material, only a portion of the load would be expected to be involved in the fire. The Technical Direction in Appendix A specifically directs evaluation of a vehicle fire due to electrical or fuel system malfunctions.

Of the eight evaluated accidents, three credible accidents were determined to be bounding representative accidents: 1) a collision that results in a spill; 2) a collision that results in a vehicle fire; and 3) a hydrogen deflagration that involves a single package. All accident consequences were evaluated as unmitigated and mitigated.

The TSD did not specifically evaluate spontaneous vehicle fires due to electrical or fuel system malfunctions. Lacking specific data, such fires are assumed to occur at an Anticipated frequency ($>1.0E-02/\text{yr}$). The probability that such a fire could spread to involve radioactive material carried on the vehicle is affected by many conditions, including location and severity of the fire, availability of external combustibles, presence of radioactive packages and location on the vehicle, etc. Considering the compounded set of conditions necessary to result in the release of radioactive material from this type of fire, a frequency of Unlikely would be appropriate for this accident. Because this is a spontaneous fire during normal transportation activities, significant damage to the cargo prior to the fire is not assumed, and significant combustible material would not be expected to be present. Therefore, a damage ratio of 30% (i.e., 30% of the cargo is involved in the fire) is judged to be a reasonable assumption for this event. The Technical Direction in Appendix A specifically directs evaluation of a vehicle fire due to electrical or fuel system malfunctions as a fourth representative accident.

Highway accident frequency (probability of occurrence) was determined through evaluation of Hanford-specific and national highway accident statistics. The frequency of a hydrogen deflagration was taken from the Hanford Waste Receiving and Packaging facility DSA.

Atmospheric dispersion factors were calculated using the accepted Hanford GXQ code. Ground level release was assumed for spills. Fires and the hydrogen deflagration events assumed plume buoyancy for a 0.17 Mw fire, equivalent to combustion of 1 gallon of diesel fuel. For the severe fire involving the entire conveyance limit, this is considered extremely conservative.

Onsite co-located worker distance was taken as 100 meters from the point of release. Offsite consequences were calculated for the 100/200 areas, and the 300 area, based on the respective distances to the Hanford site boundaries. Immediate worker risks are evaluated qualitatively.

Results of the original Risk-Based package accident analyses and the approved Risk-Based package analysis are provided in Tables 5, 6A, and 6B:

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Table 5 - Submitted Risk Based Packaging Accident Consequences

Scenario	MAR (A ₂)	Damage Ratio	Airborne Release Fraction (Note 2)	Respirable Fraction (Note 2)	Breathing Rate (Note 2)	DCF rem/A ₂	MOI		100 m Worker		Frequency (/yr) (Note 5)	Accident Assumptions (Note 6)
							Y/Q s/m ³ (Note 3)	Consequences (rem) (Note 4)	Y/Q s/m ³ (Note 3)	Consequences (rem) (Note 4)		
TRU (Pu-Alpha) 100/200 Areas - Unmitigated												
Collision (Spill)	30000	1.0	1.00E-03	0.1	3.33E-04	2.32E+06	4.49E-05	1.0E-01	0.0732	1.7E+02	8.00E-05	Unmitigated
Fire	30000	1.0	1.45E-03	1.0	3.33E-04	2.32E+06	2.09E-05	7.0E-01	0.00106	3.6E+01	1.28E-06	Unmitigated
Hydrogen Deflagration	3000	1.0	2.15E-02	7.2E-01	3.33E-04	2.32E+06	2.09E-05	7.3E-01	0.00106	3.8E+01	Anticipated	Unmitigated
TRU (Pu-Alpha) 300 Area - Unmitigated												
Collision (Spill)	10000	1.0	1.00E-03	0.1	3.33E-04	2.32E+06	6.38E-03	4.9E+00	0.0421	3.3E+01	3.20E-04	Unmitigated
Fire	10000	1.0	1.45E-03	1.0	3.33E-04	2.32E+06	2.06E-04	2.3E+00	0.00108	1.2E+01	5.12E-06	Unmitigated
Hydrogen Deflagration	940	1.0	2.15E-02	7.2E-01	3.33E-04	2.32E+06	2.06E-04	2.3E+00	0.00108	1.2E+01	Anticipated	Unmitigated
Non-TRU (Cs, SR - Beta-Gamma) 100/200 Areas - Unmitigated												
Collision (Spill)	60000	1.0	1.00E-03	0.1	3.33E-04	1.98E+06	4.49E-05	1.8E-01	0.0732	2.9E+02	9.60E-06	Unmitigated
Fire	60000	1.0	1.45E-03	1.0	3.33E-04	1.98E+06	2.09E-05	1.2E+00	0.00106	6.1E+01	1.54E-07	Unmitigated
Hydrogen Deflagration	6000	1.0	2.15E-02	7.2E-01	3.33E-04	1.98E+06	2.09E-05	1.3E+00	0.00106	6.5E+01	Anticipated	Unmitigated
Non-TRU (Cs, SR - Beta-Gamma) 300 Area - Unmitigated												
Collision (Spill)	10000	1.0	1.00E-03	0.1	3.33E-04	1.98E+06	6.38E-03	4.2E+00	0.0421	2.8E+01	3.20E-04	Unmitigated
Fire	10000	1.0	1.45E-03	1.0	3.33E-04	1.98E+06	2.06E-04	2.0E+00	0.00108	1.0E+01	5.12E-06	Unmitigated
Hydrogen Deflagration	940	1.0	2.15E-02	7.2E-01	3.33E-04	1.98E+06	2.06E-04	2.0E+00	0.00108	1.0E+01	Anticipated	Unmitigated
TRU (Pu-Alpha) 100/200 Areas - Mitigated												
Collision (Spill)	30000	0.1	1.00E-03	0.1	3.33E-04	2.32E+06	4.49E-05	1.0E-02	0.0732	1.7E+01	8.00E-05	(a) and (b)
Fire	30000	1.0	1.45E-03	1.0	3.33E-04	2.32E+06	2.09E-05	7.0E-01	0.00106	3.6E+01	1.28E-06	Unmitigated
Hydrogen Deflagration	3000	1.0	2.15E-02	7.2E-01	3.33E-04	2.32E+06	2.09E-05	7.3E-01	0.00106	3.8E+01	Ext. Unlikely	(c) and (d)
TRU (Pu-Alpha) 300 Area - Mitigated												
Collision (Spill)	10000	0.1	1.00E-03	0.1	3.33E-04	2.32E+06	6.38E-03	4.9E-01	0.0421	3.3E+00	3.20E-04	(a) and (b)
Fire	10000	1.0	1.45E-03	1.0	3.33E-04	2.32E+06	2.06E-04	2.3E+00	0.00108	1.2E+01	5.12E-06	Unmitigated
Hydrogen Deflagration	940	1.0	2.15E-02	7.2E-01	3.33E-04	2.32E+06	2.06E-04	2.3E+00	0.00108	1.2E+01	Ext. Unlikely	(c) and (d)
Non-TRU (Cs, SR - Beta-Gamma) 100/200 Areas - Mitigated												
Collision (Spill)	60000	0.1	1.00E-03	0.1	3.33E-04	1.98E+06	4.49E-05	1.78E-02	0.0732	2.9E+01	9.60E-06	(a) and (b)
Fire	60000	1.0	1.45E-03	1.0	3.33E-04	1.98E+06	2.09E-05	1.20E+00	0.00106	6.1E+01	1.54E-07	Unmitigated
Hydrogen Deflagration	6000	1.0	2.15E-02	7.2E-01	3.33E-04	1.98E+06	2.09E-05	1.28E+00	0.00106	6.5E+01	Ext. Unlikely	(c) and (d)
Non-TRU (Cs, SR - Beta-Gamma) 300 Area - Mitigated												
Collision (Spill)	10000	0.1	1.00E-03	0.1	3.33E-04	1.98E+06	6.38E-03	4.21E-01	0.0421	2.8E+00	3.20E-04	(a) and (b)
Fire	10000	1.0	1.45E-03	1.0	3.33E-04	1.98E+06	2.06E-04	1.97E+00	0.00108	1.0E+01	5.12E-06	Unmitigated
Hydrogen Deflagration	940	1.0	2.15E-02	7.2E-01	3.33E-04	1.98E+06	2.06E-04	1.97E+00	0.00108	1.0E+01	Ext. Unlikely	(c) and (d)

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Table 6A – Approved Risk Based Packaging Accident Consequences – 100/200 Areas

Scenario	MAR (A ₁) (Note 1)	Damage Ratio	Airborne Release Fraction (Note 2)	Respirable Fraction (Note 2)	Breathing Rate	DCF rem/A ₁	MOI		100 m Worker		Frequency (Note 5)	Credited Controls (Note 6)
							Z/Q s/m ³	Consequences (rem)	Z/Q s/m ³	Consequences (rem)		
TRU (Pu-Alpha) 100/200 Areas - Unmitigated												
Collision (Spill)	30000	1.0	1.00E-03	0.1	3.33E-04	2.32E+06	4.49E-05	1.0E-01	0.0732	1.7E+02	Ext. Unlikely	Unmitigated
Fire	30000	1.0	1.45E-03	1.0	3.33E-04	2.32E+06	2.09E-05	7.0E-01	0.00106	3.6E+01	Ext. Unlikely	Unmitigated
Partial Fire	9000	1.0	1.45E-03	1.0	3.33E-04	2.32E+06	2.09E-05	2.1E-01	0.00106	1.1E+01	Unlikely	Unmitigated
Hydrogen Deflagration	3000	1.0	2.15E-02	7.2E-01	3.33E-04	2.32E+06	2.09E-05	7.5E-01	0.00106	3.8E+01	Anticipated	Unmitigated
TRU (Pu-Alpha) 100/200 Areas - Mitigated												
Collision (Spill)	30000	0.1	1.00E-03	0.1	3.33E-04	2.32E+06	4.49E-05	1.0E-02	0.0732	1.7E+01	Ext. Unlikely	(a) and (b)
Fire	30000	1.0	1.45E-03	1.0	3.33E-04	2.32E+06	2.09E-05	7.0E-01	0.00106	3.6E+01	Ext. Unlikely	Unmitigated
Partial Fire	10000	1.0	1.45E-03	1.0	3.33E-04	2.32E+06	2.09E-05	2.3E-01	0.00106	1.2E+01	Unlikely	Unmitigated
Hydrogen Deflagration	3000	1.0	2.15E-02	7.2E-01	3.33E-04	2.32E+06	2.09E-05	7.5E-01	0.00106	3.8E+01	Ext. Unlikely	(c) and (d)
Non-TRU (Cs, SR - Beta-Gamma) 100/200 Areas - Unmitigated												
Collision (Spill)	30000	1.0	1.00E-03	0.1	3.33E-04	1.98E+06	4.49E-05	8.9E-02	0.0732	1.4E+02	Ext. Unlikely	Unmitigated
Fire	30000	1.0	1.45E-03	1.0	3.33E-04	1.98E+06	2.09E-05	6.0E-01	0.00106	3.0E+01	BEU	Unmitigated
Partial Fire	10000	1.0	1.45E-03	1.0	3.33E-04	1.98E+06	2.09E-05	2.0E-01	0.00106	1.0E+01	Unlikely	Unmitigated
Hydrogen Deflagration	3000	1.0	2.15E-02	7.2E-01	3.33E-04	1.98E+06	2.09E-05	6.4E-01	0.00106	3.2E+01	Anticipated	Unmitigated
Non-TRU (Cs, SR - Beta-Gamma) 100/200 Areas - Mitigated												
Collision (Spill)	30000	0.1	1.00E-03	0.1	3.33E-04	1.98E+06	4.49E-05	8.88E-03	0.0732	1.4E+01	Ext. Unlikely	(a) and (b)
Fire	30000	1.0	1.45E-03	1.0	3.33E-04	1.98E+06	2.09E-05	5.99E-01	0.00106	3.0E+01	BEU	Unmitigated
Partial Fire	10000	1.0	1.45E-03	1.0	3.33E-04	1.98E+06	2.09E-05	2.00E-01	0.00106	1.0E+01	Unlikely	Unmitigated
Hydrogen Deflagration	3000	1.0	2.15E-02	7.2E-01	3.33E-04	1.98E+06	2.09E-05	6.38E-01	0.00106	3.2E+01	Ext. Unlikely	(c) and (d)

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Table 6B – Approved Risk Based Packaging Accident Consequences – 300 Area

Scenario	MAR (A ₂) (Note 1)	Damage Ratio	Airborne Release Fraction (Note 2)	Respirable Fraction (Note 2)	Breathing Rate	DCF rem/A ₂	MOI		100 m Worker		Frequency	Credited Controls (Note 6)
							Z/Q s/m ³ (Note 3)	Consequences (rem) (Note 4)	Z/Q s/m ³ (Note 3)	Consequences (rem) (Note 4)		
TRU (Pu-Alpha) 300 Area - Unmitigated												
Collision (Spill)	10000	1.0	1.00E-03	0.1	3.33E-04	2.32E+06	6.38E-03	4.9E+00	0.0421	3.3E+01	Unlikely	Unmitigated
Fire	10000	1.0	1.45E-03	1.0	3.33E-04	2.32E+06	2.06E-04	2.3E+00	0.00108	1.2E+01	Ext. Unlikely	Unmitigated
Partial Fire	3000	1.0	1.45E-03	1.0	3.33E-04	2.32E+06	2.06E-04	6.9E-01	0.00108	3.6E+00	Unlikely	Unmitigated
Hydrogen Deflagration	940	1.0	2.15E-02	7.2E-01	3.33E-04	2.32E+06	2.06E-04	2.3E+00	0.00108	1.2E+01	Anticipated	Unmitigated
TRU (Pu-Alpha) 300 Area - Mitigated												
Collision (Spill)	10000	0.1	1.00E-03	0.1	3.33E-04	2.32E+06	6.38E-03	4.9E-01	0.0421	3.3E+00	Unlikely	(a) and (b)
Fire	10000	1.0	1.45E-03	1.0	3.33E-04	2.32E+06	2.06E-04	2.3E+00	0.00108	1.2E+01	Ext. Unlikely	Unmitigated
Partial Fire	3000	1.0	1.45E-03	1.0	3.33E-04	2.32E+06	2.06E-04	6.9E-01	0.00108	3.6E+00	Unlikely	Unmitigated
Hydrogen Deflagration	940	1.0	2.15E-02	7.2E-01	3.33E-04	2.32E+06	2.06E-04	2.3E+00	0.00108	1.2E+01	Ext. Unlikely	(c) and (d)
Non-TRU (Cs, SR - Beta-Gamma) 300 Area - Unmitigated												
Collision (Spill)	10000	1.0	1.00E-03	0.1	3.33E-04	1.98E+06	6.38E-03	4.2E+00	0.0421	2.8E+01	Unlikely	Unmitigated
Fire	10000	1.0	1.45E-03	1.0	3.33E-04	1.98E+06	2.06E-04	2.0E+00	0.00108	1.0E+01	Ext. Unlikely	Unmitigated
Partial Fire	3000	1.0	1.45E-03	1.0	3.33E-04	1.98E+06	2.06E-04	5.9E-01	0.00108	3.1E+00	Unlikely	Unmitigated
Hydrogen Deflagration	940	1.0	2.15E-02	7.2E-01	3.33E-04	1.98E+06	2.06E-04	2.0E+00	0.00108	1.0E+01	Anticipated	Unmitigated
Non-TRU (Cs, SR - Beta-Gamma) 300 Area - Mitigated												
Collision (Spill)	10000	0.1	1.00E-03	0.1	3.33E-04	1.98E+06	6.38E-03	4.21E-01	0.0421	2.8E+00	Unlikely	(a) and (b)
Fire	10000	1.0	1.45E-03	1.0	3.33E-04	1.98E+06	2.06E-04	1.97E+00	0.00108	1.0E+01	Ext. Unlikely	Unmitigated
Partial Fire	3000	1.0	1.45E-03	1.0	3.33E-04	1.98E+06	2.06E-04	5.91E-01	0.00108	3.1E+00	Unlikely	Unmitigated
Hydrogen Deflagration	940	1.0	2.15E-02	7.2E-01	3.33E-04	1.98E+06	2.06E-04	1.97E+00	0.00108	1.0E+01	Ext. Unlikely	(c) and (d)

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- Note 1:** The MAR for the collision and fire scenarios are conveyance limits, while the MAR for the hydrogen deflagration is a package limit
- Note 2:** The ARF and RF shown are cumulative for multiple release mechanisms of the accident scenario. The payload materials are assumed to consist of surface contaminated combustible or non-combustible materials or powder, whichever is more conservative for the accident scenario.
- Note 3:** Uses the original Site fence line as the Site boundary. The χ/Qs for all fires and hydrogen deflagrations include the effects of plume rise from a small fire of 0.17 MW (1 gallon diesel fuel). A fire with a lower heat release rate gives conservatively higher doses than a larger heat release rate. The χ/Qs for all of the fires and hydrogen deflagrations are taken from DFSNQW-ECAL-100, χ/Qs for the collision in the 100/200 Areas are taken from TSD Table 7-3, and the χ/Qs for the collision in the 300 Area is taken from TSD Table 7-4 (to exclude building wake effects).
- Note 4:** Inhalation dose from 1.0E-06 A2: 2.32 rem from Pu-239 (Solubility Class W - nitrates), 1.98 rem from 50/50 mix of Sr/Cs [3.52 rem Sr-90 (Solubility Class Y - SrTiO₃), 0.431 rem Cs-137].
- Note 5:** Frequency calculation assumes 50 trips per year, 5 miles per trip for cases 1 and 5; 50 trips per year, 20 miles per trip for cases 2 and 6; 6 trips per year, 5 miles per trip for cases 3 and 7; and 50 trips per year, 20 miles per trip for cases 4 and 8. For simplicity, all transportation accidents are assumed to result in an impact, all impacts are assumed to result in at least partial failure to at least 1 package in an array, 1.6% of accidents result in fire, and fires result in at least partial failure of at least 1 package in an array. Frequency of an unmitigated hydrogen deflagration is assumed to be "anticipated," based on no programmatic controls to prevent occurrence. Frequency of the mitigated hydrogen deflagration is "extremely unlikely" (Ref. WRAP SAR, page 3-51).
- Note 6:** Credited Controls:
- (a) Maximum speed limited to 35 mph.
 - (b) Sandbags or non-rad (damage) containers in front row of array, high dose equivalent-Curie (DE-Ci) containers in rear
 - (c) Containers vented prior to shipment
 - (d) Chemically reactive contents not present

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As previously discussed, the TSD did not specifically evaluate spontaneous vehicle fires due to electrical or fuel system malfunctions. A spontaneous fire at an Unlikely frequency involving 30% of the total vehicle load is included in the approved summary tables for Risk-Based packages. Inclusion of this scenario is a condition of approval of the TSD, as discussed in Technical Direction provided in Appendix A of this SER.

As shown in Table 5, the consequences of a fire in the 100/200 Areas involving 60,000 A₂ values of non-TRU material (Cs-137 and SR-90), resulted in approximately 61 rem to the onsite at 100 meters. The consequences of a hydrogen deflagration in the 100/200 Areas involving 6,000 A₂ values of non-TRU material (Cs-137 and SR-90), resulted in approximately 65 rem to the onsite at 100 meters. The consequences of these accidents approach the 100 rem evaluation guideline in the Hanford Safety Basis Criteria (DOE/RL 2002) for the worker at 100 meters, and were therefore deemed unacceptable by the DOE-RL. The DOE-RL determined that appropriate limits for the 100 and 200 areas are 3,000 A₂ values (40,500 Ci Cs-137 or 8,100 Ci Sr-90) for an individual package and 30,000 A₂ values (405,000 Ci Cs-137 or 81,000 Ci Sr-90) for a conveyance limit. This is addressed in the Technical Direction provided in Appendix A of this SER.

3.2.6 Abnormal Events

The TSD did not specifically identify and evaluate abnormal events as discussed in 10 CFR 830.204.(b).(3). However, normal conditions to transport (NCs) include severe service conditions that could be considered representative of Abnormal Events (e.g., vibration, temperature and pressure extremes, vibration, water spray, free drop, corner drop, and penetration).

Conclusion: With incorporation of the Technical Direction provided in Appendix A of this SER, the TSD Hazard And Accident Analyses for Risk-Based packages is deemed adequate.

3.3 Adequacy of Safety Structure, Systems, And Components

The TSD was developed in accordance with the 10 CFR 830 "safe harbor" methodologies for transportation activities. DOE-compliant and DOT-equivalent packaging and transportation activities are addressed in the TSD in accordance with DOE-O 460.1A, "*Packaging and Transportation Safety*." Accidents involving Risk-Based packages were evaluated using the "safe harbor" methodologies of DOE-STD-3009, *Preparation Guide for U. S. Department of Energy Non-reactor Nuclear Facility Safety Analysis Reports* (DOE 2000). This SER provides a comparison of Risk-Based package accident consequences to DOE-STD-3009 requirements to provide perspective of relative risk and adequacy of controls.

Package Specific Safety Documents (PSSDs) are identified in TSD Appendix A. Table 1 of TSD Appendix A lists existing Hanford Site PSSDs in use at the time the TSD was prepared. These PSSDs, at the revision level identified, are part of the TSD and are approved for continuing use. Documents that were previously used and do not appear on TSD Appendix A, Table 1, are not approved for use after the effective date of the TSD. TSD Appendix A, Table 2 lists packaging approved following the requirements of the TSD. Each table is divided into four parts. Part One identifies those packaging compliant with U.S. DOT packaging requirements.

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Part Two identifies those packagings judged to meet the DOT-equivalent packaging standards identified in TSD Chapter VI, *Site-Specific Standards, Procedures, and Instructions*. Part Three contains those packagings judged to meet Risk-Based packaging standards. Part Four contains exemptions to the TSD requirements authorized by the DOE.

None of the analyzed accidents or abnormal events associated with Risk-Based packages cause significant offsite release and none approach the Evaluation Guide of 25 rem for safety-class (SC) structures, systems, or components (SSCs) given in Appendix A to DOE-Std-3009-94 (DOE 2000). As such, the TSD identifies no SSCs that would warrant safety-class designation.

Conclusion: Through the PSSDs identified in TSD Appendix A, the TSD adequately identifies equipment important to safety. No issues were identified with respect to Safety Structure, Systems, and Components.

3.4 Adequacy of Derivation Of Technical Safety Requirements

TSD Technical Safety Requirements (TSRs) are listed in TSD Appendix D. There are no Safety Limits, Limiting Control Settings, Limiting Conditions for Operation, or Surveillance Requirements identified for T&P activities. Five TSR-level Administrative Controls (TSR 5.1 through TSR 5.5) were established in the submitted TSD, which are summarized below:

5.1. Purpose

This Administrative Control generally defines the function of the TSD Administrative Controls, and state that these Administrative Controls apply to the activities associated with the T&P of radioactive materials on the Hanford Site in quantities exceeding 1 A₂.

5.2. Onsite Shipments and Transfers

This Administrative Control requires each contractor to have a formal program in place with an organizational structure to ensure sufficient staff to perform the actions identified in the TSD, and states that the overall responsibility for the TSD is assigned to RL. RL, through contract, shall identify those contractors and their roles in the performance of this TSD.

Contractors are specifically responsible to address the following organizational components of the TSD:

- T&P Functional Organization
- Contractor specific policies and procedures
- Document configuration control program (e.g., shipping papers, package specific safety documents, unreviewed safety questions, training and qualification records, etc.)
- Assessment program
- T&P manager responsibilities
- T&P activity manager responsibilities

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- TSD configuration manager responsibilities
- Unreviewed safety question process for transportation program
- Training program
- QA program

5.3 Criticality Safety Program Contractors are required to have a nuclear criticality safety program that covers T&P activities. The program must implement the criticality requirements defined in the TSD. Packaging is conducted under the originating facility safety analysis. The primary features of the criticality prevention program conducted under transportation activities are the package array and spacing.

This Administrative Control directly protects the fundamental accident analysis assumption that criticality accidents during onsite transportation activities are incredible under all normal and accident conditions.

5.4 Gas Accumulation Prevention Program

Contractors are required to have a gas accumulation prevention program that ensures package integrity will not be breached from over pressurization, deflagration, or explosion. The gas accumulation prevention program TSR is applicable to packages classified as transuranic or transuranic mixed. Causes of gas generation other than transuranic radiolysis are covered by standard safety programs and are not part of this TSR.

Methods used to achieve the TSR function must be documented and documentation maintained. Typical methods for prevention of gas accumulation are listed as:

- Venting
- Time limits of transport (shipping window)
- Package inventory controls (e.g., hydrogenous material limits)
- Chemical compatibility

This Administrative Control directly protects accident analysis assumptions for frequency of a hydrogen deflagration.

5.5 Damage Ratio Control for Risk-Based Packages

Contractors are required to have a program to control package damage ratio during transportation activities. The damage ratio control program is only applicable to Risk-Based packages.

Methods used to achieve the TSR function must be documented and documentation maintained. Typical methods for control of damage ratios for Risk-Based packages are listed as:

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- Speed limits
- Dunnage
- Tiedown systems

This Administrative Control directly protects accident analysis assumptions for damage ratios during collisions.

5.6 Surveillance and Maintenance Program

Contractors must implement a surveillance and maintenance program to ensure preservation of the key transportation safety functions for all packages currently used or planned for future use in transportation activities. The surveillance and maintenance program TSR is applicable to all packaging identified in the TSD or in use for onsite transportation.

The contractor's program must include provisions to:

- Identify surveillance and maintenance actions for packages not currently in use to maintain functionality of the package
- Perform surveillance and maintenance requirements identified in individual PSSD's for packages currently in use or prior to use following periods of non-use.
- Ensure actions are taken to ensure package integrity prior to use for those packages whose qualifications for transportation are in question
- Request deviations from this administrative control when appropriate
- Request removal of a package from the TSD

This administrative control protects assumptions in the TSD and associated PSSD's on package integrity.

5.7 Configuration Management Program

The contractor's transportation program must have a configuration control feature that ensures changes to packages that could affect the safety functions provided by the package design are done so in a controlled manner. This configuration management program is applicable to all packaging identified in the TSD or in use for onsite transportation.

The contractors program must address :

- All changes to packages
- Consideration of all safety functions provided by the package prior to initiating a change
- The configuration management process integration with the USQ process

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This Administrative Control protects accident analysis assumptions for package performance.

Conclusion: With the incorporation the Technical Direction provided in Appendix A of this SER, the TSD TSR Administrative Controls are appropriate and adequate for the activities and hazards associated with T&P activities.

3.5 Adequacy of Programmatic Controls

There are no programmatic controls specifically discussed in the TSD. Protection of occupational workers from radiological hazards as a result of the packaging and transportation of radioactive materials is achieved by compliance with the TSD and the TSRs contained therein. Each Hanford Site contractor must implement a comprehensive ISMS Plan and assess onsite T&P activities against ISMS requirements.

Conclusion: No issues were identified with respect to Programmatic Controls.

4.0 UNREVIEWED SAFETY QUESTION PROCESS

The TSD, in Chapter 6 and Appendix C, includes an adaptation of the traditional facility focused Unreviewed Safety Question (USQ_T) process to meet the needs of onsite transportation. The transportation USQ_T process described in the TSD retains the principle actions and questions of a facility focused USQ process, but also addresses how to apply the process to 10 CFR 71 compliant, equivalent safety, and risk based packages. It further defines roles, responsibilities, and actions to be taken by the multiple contractors involved, and RL, in communicating and coordinating results of changes to packages and USQ evaluations. The USQ_T process described appears adequate to elevate changes in risk to RL's attention, and in coordinating the interactions of multiple contractors, the Office of River Protection and RL.

The USQ_T process includes Categorical Exclusions (CX) as a primary screen, a more detailed screening worksheet for those items not categorically excluded, and a USQ_T determination worksheet for those items that have a yes answer on the screening worksheet. The CX's address those packages that do not contain sufficient radioactive material to be regulated by 10 CFR 830, and those packages either compliant with 10 CFR 71 or demonstrating equivalent safety to 10 CFR 71. The application of 10 CFR 71 criteria to onsite packaging is a unique feature of the USQ_T process.

The application of the USQ_T process following a package change or a discovery situation is described briefly below. The first step involves a determination whether the package meets 10 CFR 71 criteria. If a package still fully meets 10 CFR 71, whether or not it has a Certificate of Compliance from the Nuclear Regulatory Commission, it is categorically excluded from further USQ_T determination. The next step involves a determination of whether the package meets Hanford Packaging Standards (HPS). If a package demonstrates a level of safety equivalent to 10 CFR 71 through use of the HPS, it too is categorically excluded. If a package was previously either fully or equivalently compliant with 10 CFR 71 and it no longer is, due to a package change or discovery, it is then considered a risk based package. If the package was originally approved as 10 CFR 71 or HPS compliant, the change or discovery would require DOE approval

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as a positive USQ_T. If the package had previously been approved as a risk based package, a USQ_T determination would then be made to determine if DOE approval for the change or discovery is required.

The question of how equivalency to 10 CFR 71 is met is important in this logic, and is based on the HPS. The HPS are discussed in Chapter 6 and Appendix B of the TSD, and include the Hanford Standard Performance Tests and Hanford acceptance criteria. The Hanford Standard Performance Tests and Hanford acceptance criteria provide the basis for determining the equivalency of a package to specific testing required in 10 CFR 71 based on site specific environmental and test conditions. In some cases these are more stringent than those found in the federal regulation, such as ambient high temperatures. Table 1 in Appendix B of the TSD lists the original requirement and NRC acceptance criteria, along with the Hanford Standard Performance Test and Hanford acceptance criteria. Equivalency is shown by meeting the specific Hanford Performance tests and associated criteria in Appendix B, Table 1, and meeting the other Hanford Packaging Standards identified in Chapter 6 of the TSD. This approach meets the requirements of 10 CFR 830 and the documented safety analysis safe harbor, DOE 460.1A.

5.0 CONCLUSION

The DOE-RL has determined that the TSD provides sufficient hazard analysis and controls to assure adequate safety of onsite shipments, and will reduce the risk to the Offsite Public and Onsite Workers to levels within the guidelines provided in DOE Order 460.1A (DOE 1996c), the Evaluation Guideline in DOE-STD-3009 (DOE 2000), and the Consequence Guidelines of the Hanford Safety Basis Criteria (DOE/RL 2002). The DOE-RL concludes that the TSD, with incorporation of the Technical Direction provided in Appendix A of this SER, complies with DOE Order 460.1A (DOE 1996c), DOE-STD-3009 (DOE 2000), and Subpart B to 10 CFR 830 (CFR 2001a).

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6.0 REFERENCES

- CFR 2001a 10 CFR 830, Subpart A, *Safety Basis Requirements*, U. S. Code of Federal Regulations, as amended
- CFR 2001b 10 CFR 71, "*Packaging and Transportation of Radioactive Material*", U. S. Code of Federal Regulations, as amended
- CFR 2001c 49 CFR Subchapter C, *Hazardous Materials Regulations*, U. S. Code of Federal Regulations, as amended
- DOE 1991 *Unreviewed Safety Questions*, DOE Order 5480.21, U. S. Department of Energy, Washington, D. C., December 1991
- DOE 1994a *Nuclear Safety Analysis Reports*, DOE Order 5480.23, Change Notice No. 1, U. S. Department of Energy, Washington, D. C., March 1994
- DOE 1994b *Guidance for Preparation of DOE 5480.22 (TSR) and DOE 5480.23 (SAR) Implementation Plans*, DOE-STD-3011, U. S. Department of Energy, Washington, D. C., November 1994
- DOE 1994c *Airborne Release Fractions/Rates And Respirable Fractions For Nonreactor Nuclear Facilities*, DOE-HDBK-3010-94, U. S. Department of Energy, Washington, D. C., December 1994
- DOE 1996a *Technical Safety Requirements*, DOE Order 5480.22, Change Notice No. 2, U. S. Department of Energy, Washington, D. C., January 1996
- DOE 1996b *Review and Approval of Nonreactor Nuclear Facility Safety Analysis Reports*, STD-1104-96, U. S. Department of Energy, Washington, D. C., February 1996
- DOE 1996c *Packaging and Transportation Safety*, DOE Order 460.1A, Department of Energy, Washington, D. C., October 1996
- DOE 1997a *Implementation Guide for Use with DOE O 460.1A, Packaging and Transportation Safety*, DOE-G 460.1-1, Department of Energy, Washington, D. C., June 1997
- DOE 1997b *Hazard Categorization And Accident Analysis Techniques For Compliance With DOE Order 5480.23*, DOE-STD-1027-92, Change Notice No. 1, Nuclear Safety Analysis Reports, U. S. Department of Energy, Washington, D. C., September 1997
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- DOE 2001 *Implementation Guide For Use In Addressing Unreviewed Safety
Question Requirements*, DOE G 424.1-1, U. S. Department of
Energy, Washington, D. C., October 2001
- DOE/RL 2001 *Hanford Sitewide Transportation Safety Document*, DOE/RL-
2001-36, U. S. Department of Energy, Richland, Washington,
October 4, 2001
- DOE/RL 2002 DOE-RL letter 02-ABD-0053, *Contract Number DE-AC06-
96RL13200 – Fluor Hanford Nuclear Safety Basis Strategy and
Criteria*, U. S. Department of Energy, Richland Operations Office,
Richland, Washington, February 5, 2002
- IAEA 1990 *Explanatory Material for the IAEA Regulations for the Safe
Transport of Radioactive Material*, Safety Series No. 7, Second
Edition (as amended 1990), International Atomic Energy Agency,
Vienna.

**APPENDIX A
DIRECTED CHANGES TO THE TSD**

The TSD is approved, subject to the following conditions:

1. The evaluation of Risk-Based packages and transportation activities shall be modified to incorporate analysis of a vehicle fire due to electrical or fuel system malfunctions that involves up to 30% of the total radiological inventory in Risk Based packages.
2. The non-TRU limits for Risk-Based packages in the Hanford 100 and 200 areas shall be 3,000 A₂ values (40,500 Ci Cs-137 or 8,100 Ci Sr-90) for an individual package and 30,000 A₂ values (405,000 Ci Cs-137 or 81,000 Ci Sr-90) for a conveyance limit. The Risk-Based analysis and TSD Table 6-7 shall be updated to reflect these limits.
3. A TSR Administrative Control shall be incorporated to explicitly limit the inventory of individual Risk-Based packages and the total A₂ content for loads composed of Risk-Based packages to the specific A₂ values in TSD Table 6-7, as modified by approval condition 2 above.
4. Until further notice an informational copy of all Risk-Based packaging shall be submitted for RL review at least seven days prior to implementation. This includes packaging determined to represent a negative USQ using the methodology described in the TSD. For negative USQ changes no specific approval will be provided by RL, nor should implementation be delayed waiting for formal or informal RL concurrence. Submittals shall include all documentation generated using the methodology proposed in the TSD, including required determinations and the USQ evaluations. In addition, all other USQ evaluations shall be submitted to RL for review.

APPENDIX B
CHANGES TO BE INCLUDED IN THE TSD ANNUAL UPDATE

1. At the next Annual Update, the TSD shall be revised to incorporate all relevant aspects of DOE G 424.1-1, *Implementation Guide For Use In Addressing Unreviewed Safety Question Requirements*, including implied margins of safety, into the USQ_T Process.

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HANFORD SITEWIDE TRANSPORTATION SAFETY DOCUMENT

CHAPTER I. PURPOSE, SCOPE, AND APPLICABILITY

This chapter describes why the *Hanford Sitewide Transportation Safety Document* (TSD) is required, the scope of its coverage and applicability to contractors, user organizations, and individuals responsible for hazardous material and waste transportation and packaging (T&P) activities at the Hanford Site.

1.0 INTRODUCTION AND PURPOSE

This TSD defines the onsite T&P program at the Hanford Site, which complies with the U.S. Department of Energy (DOE) transportation safety requirements specified in DOE Order 460.1A, *Packaging and Transportation Safety*.

DOE Order 460.1A requires each DOE site to establish and maintain an onsite T&P program for the safe movement of hazardous materials, substances, and wastes. The T&P program is to be documented in an onsite TSD that is approved by the DOE-Richland Operations Office (RL). The program must provide a level of onsite safety equivalent to that achieved offsite under U.S. Department of Transportation (DOT) regulations provided in Title 49 *Code of Federal Regulations* (CFR) Subchapter C, "Hazardous Materials Regulations," and U.S. Nuclear Regulatory Commission (NRC) standards provided in 10 CFR 71, "Packaging and Transportation of Radioactive Material."

This TSD includes the following:

- Identification of responsibilities, lines of authority, and program approval procedures.
- Definition of minimum safe packaging requirements including necessary design, fabrication, and quality assurance (QA) elements, using appropriate codes and standards.
- Description of the process and analysis used to ensure equivalent safety requirements are established.
- Description of the Hanford Site including maps identifying boundaries, railways, and roadways that clearly delineate offsite and onsite areas, and procedures for clearly establishing access control for any area having occasional public access.
- Provisions for effective emergency response and recovery under credible accident conditions (AC).
- Process for accomplishing nonroutine T&P activities.

This TSD also addresses onsite "equivalent safety" by establishing a comprehensive set of onsite T&P performance standards and risk-based standards. The requirements and standards presented are equivalent to DOT and NRC standards (10 CFR 71) currently used for commercial offsite transportation. Hanford Site standards, however, are developed to be reflective of the unique transport environment within Hanford Site boundaries. Hanford Site restricted roadways, fenced facilities, Hanford Patrol inspection points, and barricades, as well as weather and physical location, shape Hanford's unique transportation environment. Additionally, Hanford Site contractors retain control over all onsite shipments, provide enhanced worker training, and possess unique site emergency preparedness and response capability that influence safety across the Hanford Site.

2.0 SCOPE AND APPLICABILITY

The TSD encompasses all onsite shipments of government-owned hazardous materials, substances, and wastes within the Hanford Site, which includes radioactive materials and wastes. Only materials identified as hazardous by the DOT, and regulated if shipped in commerce, are covered. The modes of transportation are defined in the DOT regulations (e.g., vehicles, vessel, air).

The TSD does not include movement of these government materials by hand or hand truck. This movement must be included in the contractor's industrial safety and health program (29 CFR 1910/1026, "Occupational Safety and Health Administration") and/or safety or authorization basis.

The movement of DOE-owned materials between onsite commercial facilities is covered by this TSD.

An onsite shipment is defined as the activity of moving hazardous materials, substances, and wastes between DOE facilities and over roadways where public access is controlled or restricted and includes intra-area and inter-area movements. An intra-facility transfer is the movement of material between individual buildings within a facility or facility complex or the surrounding compound area that is covered under a single facility safety analysis report (SAR) or other DOE-approved safety or authorization basis document.

This TSD sets forth requirements and standards for the safe onsite shipment of radioactive and hazardous materials and wastes within the confines of the Hanford Site on roadways where public access is restricted by signs, barricades, fences, or other means including road closures and moving convoys controlled by Hanford Site security forces.

This document and its provisions apply to DOE and all RL and Office of River Protection (ORP) Hanford Site prime contractors and subcontractors performing government work at the Hanford Site, unless specifically excluded in their respective contracts under RL and ORP.

Requests for a deviation to the requirements of this TSD must be submitted to RL for processing. The TSD exemption process is outlined in Chapter IX for nonroutine shipments.

The RL Manager is the Hanford Site Approval Authority. The RL Manager approves the content and applicability of this document.

3.0 ROADMAP THROUGH THE TSD

Chapter I. Purpose, Scope, and Applicability. This chapter describes why the Hanford Sitewide TSD is required, the scope of its coverage and applicability to contractors, user organizations, and individuals responsible for hazardous material and waste T&P activities at the Hanford Site.

Chapter II. Definitions and Acronyms. This chapter provides key definitions, common acronyms, and references specific regulatory sources for additional definitions that must be understood.

Chapter III. Site Description. This chapter describes the Hanford Site boundaries and roadways used to transport hazardous materials and wastes. It describes exclusion areas and facilities, and their principle missions. A Hanford Site map and Internet links are provided for more detailed site and area maps.

Chapter IV. Organizational Responsibilities. This chapter identifies RL and contractor responsibilities and interfaces within the transportation safety program as defined in this TSD, and specific package development and approval. It identifies Internet links to more detailed contractor charters that describe organizational responsibilities for program implementation.

Chapter V. External Regulations. This chapter identifies the external regulations that contain requirements applicable to onsite T&P activities. Several regulations are written specifically to address onsite packaging and transfer operations. Additional regulations apply to various activities that make up the overall process of T&P because they apply to the materials being handled or the activity being performed.

Chapter VI. Site-Specific Standards, Procedures, and Instructions. This chapter provides standards, requirements, programs, and processes that are needed to support safe T&P operations such as integrated safety management, QA, configuration management, and equivalent safety. Alternate design and performance requirements for onsite packagings are identified.

Chapter VII. Safety Assessment Methodology. This chapter identifies the methodologies used for achieving and demonstrating onsite safety equivalent to that accomplished by complying with the DOT regulations. In addition, it identifies specific evaluation methodology and acceptance criteria to apply when demonstrating compliance with the requirements found in Chapter VI for achieving equivalent onsite T&P activity safety.

Chapter VIII. Routine Transfers. This chapter describes the processes and procedures used to make routine onsite shipments. All packagings prepared and shipped as routine must be authorized and fully comply with the provisions of the TSD. Basic procedures and processes for handling onsite shipments of hazardous materials and wastes are included in this chapter.

Contractors provide more detailed facility/project operation procedures at the facilities where T&P activities occur.

Chapter IX. Nonroutine Transfers. This chapter describes the processes and procedures for dealing with nonroutine or emergency transfers. Conditions and circumstances for nonroutine or emergency transfers are defined, as well as applicable procedures and the approval process.

Chapter X. Personnel Qualification and Training. This chapter describes the qualification and training requirements for Hanford Site contractor hazardous material (hazmat) personnel, authorized shippers, drivers, and USQ_T evaluators. Individual facilities and organizations, to which personnel may be assigned, may establish additional qualifications and required training based on specific job task analysis, job classification, or individual training plan.

Chapter XI. Documentation and Record Keeping. This chapter identifies documentation and records requirements for different package categories. It also identifies or references processes and procedures for managing this documentation throughout the Hanford Site.

Chapter XII. Incident Reporting and Emergency Response. This chapter describes requirements for incident reporting, emergency response, and references Hanford Site-specific policies, manuals, and procedures for emergency management. It also addresses emergency planning, drills and exercises, and Hanford Site response capabilities.

Chapter XIII. Transport Vehicle Operations. This chapter describes the requirements for vehicle inspection and maintenance programs on the Hanford Site. References and requirements to specific manuals and procedures are provided for use by each contractor or subcontractor conducting transportation operations within Hanford Site boundaries.

Appendices A through H provide additional information and guidance to support application of the requirements found in the TSD.

Appendix A. Approved Package Specific Safety Documents List. This appendix provides listings of onsite package specific safety documents (PSSD).

Appendix B. Justification and Basis for Equivalency to DOT Regulations for Type B and Fissile Packages Transported on the Hanford Site. This appendix provides the justification and basis for demonstrating the equivalency of the TSD requirements to those identified in 10 CFR 71.

Appendix C. Unreviewed Safety Question Process for Transportation (USQ_T). This appendix identifies a procedure that establishes requirements for an USQ_T.

Appendix D. Technical Safety Requirements for Hanford Onsite T&P. This appendix identifies the TSRs established for Hanford onsite T&P.

Appendix E. Packaging Safety Analysis Assessment Report Contents and Format. This appendix provides guidance in formatting and preparing an onsite package safety analysis assessment (PSAA).

Appendix F. Example Application of Package Structural Design and Evaluation Requirements. This appendix provides an example application of onsite Type B package structural design and evaluation requirements to demonstrate application of the methodology provided in Chapter VII.

Appendix G. Justification and Basis for Shipment of Risked-Based Packages. This appendix provides the justification and basis used for demonstrating the equivalency of the TSD requirements to those identified in DOT Hazardous Materials Regulations.

Appendix H. Example Checksheets. This appendix contains examples of checklists used at the Hanford Site.

4.0 PRECEDENCE OF REQUIREMENTS

In cases where two or more similar requirements or limits occur impacting design, testing, analysis, fabrication or use of a packaging approved within the TSD standards, all requirements and limits must be met as applicable. If it appears one requirement is less restrictive than another the more restrictive requirement must be implemented, unless an exception is granted under the provisions of this TSD.

5.0 POLICY FOR PRE-TSD PACKAGINGS AND THEIR APPROVALS

The existing onsite T&P systems identified in Appendix A, Approved Package Specific Safety Documents List, Table 1, may continue to be used until specifically identified for cancellation by the RL Manager. Operation of the system shall be conducted in compliance with the existing PSSD (onsite safety analysis report for packaging [SARP] and onsite safety evaluation for packaging [SEP]) identified on the table. Other existing onsite T&P systems shall not be used until approved by the RL Manager.

6.0 DOCUMENT UPDATES, MAINTENANCE, AND DISTRIBUTION

The TSD must be updated annually to ensure the approved safety basis is maintained as regulations and requirements change, and as Hanford Site conditions and mission requirements mature.

All TSD changes must be performed in accordance with a TSD maintenance program. The TSD maintenance responsibility is assigned to Fluor Hanford, subject to contractual changes that may be exercised by DOE.

Maintenance includes minor changes and/or rewrites to provide clarifying language to the document. These are limited to inconsequential changes as discussed in Chapter VI, Section 6.1.2.2. Major rewrites, redesigns, or analytical work is considered an "update" and is handled separately based on Hanford Site programmatic needs.

Recommendations and changes must be submitted to RL in writing and demonstrate the required levels of safety are achievable based on packaging performance and/or risk assessment. The RL Manager must approve all changes to the TSD.

7.0 REFERENCES

10 CFR 71, "Packaging and Transportation of Radioactive Material," *Code of Federal Regulations*, as amended.

29 CFR 1910/1026, "Occupational Safety and Health Administration," U.S. Department of Labor, *Code of Federal Regulations*, as amended.

49 CFR, Subchapter C, "Hazardous Materials Regulations," *Code of Federal Regulations*, as amended.

DOE Order 460.1A, *Packaging and Transportation Safety*, U.S. Department of Energy, Washington, D.C.

CHAPTER II. DEFINITIONS AND ACRONYMS

This chapter provides key definitions, common acronyms, and references specific regulatory sources for additional definitions that must be understood.

1.0 DEFINITIONS

For additional definitions related to this subject, see the following CFRs:

- 10 CFR 71.4, "Definitions"
- 49 CFR 171, "General Information, Regulations, and Definitions"
- 49 CFR 171.8, "Definitions and abbreviations"
- 49 CFR 173, "Shippers-General Requirements for Shipments and Packagings"
- 49 CFR 173.403, "Definitions"
- 49 CFR 178, "Specifications for Packagings"
- 49 CFR 178(a), "Definitions"
- 49 CFR 350.105, "What definitions are used in this part?"
- 49 CFR 355.5, "Definitions"
- 49 CFR 365.201, "Definitions"
- 49 CFR 367.1, "Definitions"
- 49 CFR 368.2, "Definitions"
- 49 CFR 371.2, "Definitions"
- 49 CFR 372.107, "Definitions"
- 49 CFR 375.1, "Applicability and definitions"
- 49 CFR 376.2, "Definitions"
- 49 CFR 381.110, "What definitions are applicable to this part?"
- 49 CFR 382.107, "Definitions"
- 49 CFR 383.105, "Definitions"
- 49 CFR 384.105, "Definitions"
- 49 CFR 385.3, "Definitions"
- 49 CFR 386.2, "Definitions"
- 49 CFR 387.5, "Definitions"
- 49 CFR 389.3, "Definitions"
- 49 CFR 390.5, "Definitions"
- 49 CFR 393.5, "Definitions"
- 49 CFR 395.2, "Definitions"
- 49 CFR 397.65, "Definitions"
- 49 CFR 399.205, "Definitions"

A_1 means the maximum activity of special form Class 7 (radioactive) material permitted in a Type A package.

A₂ means the maximum activity of Class 7 (radioactive) material, other than special form, low-specific activity (LSA) or surface contaminated object (SCO), permitted in a Type A package. These values are listed in 49 CFR 173.435, "Table of A₁ and A₂ values for radionuclides," or derived in accordance with the procedure prescribed in 49 CFR 173.433, "Requirements for determining A₁ and A₂ values for radionuclides and for the listing of radionuclides on shipping papers and labels."

Approval Authority (contractor) means the contractor person or organization responsible for the overall onsite T&P program activities, including approving onsite PSSDs. Additionally, this person shall approve all contractor exceptions authorized in this standard.

Approval Authority is the RL Manager, which is responsible for the overall Hanford onsite T&P program, including approval of certain onsite PSSDs requiring RL Manager approval under the provisions of this TSD standard. Additionally, the RL Manager shall approve all exceptions authorized in this standard that require RL Manager approval and shall be responsible for sitewide assessments of the program.

Authorization Basis is safety documentation supporting the decision to allow a process or facility to operate. Included are corporate operational and environmental requirements as found in regulations and specific permits, and, for specific activities, work packages or job safety analyses. (See also nuclear safety authorization basis.)

Box means a packaging with complete rectangular or polygonal faces, made of metal, wood, plywood, reconstituted wood, fiberboard, plastic, or other suitable material. Holes appropriate to the size and use of the packaging, for purposes such as ease of handling or opening, or to meet classification requirements, are permitted as long as they do not compromise the integrity of the packaging during transportation, and are not otherwise prohibited in this subchapter.

C means Celsius or Centigrade.

Cargo tank means a bulk packaging which: (1) Is a tank intended primarily for the carriage of liquids or gases and includes appurtenances, reinforcements, fittings, and closures (for tank, see 49 CFR 178.345-1(c), "Definitions;" 178.337-1, "General requirements;" or 178.338-1, "General requirements;" as applicable); (2) Is permanently attached to or forms a part of a motor vehicle, or is not permanently attached to a motor vehicle but which, by reason of its size, construction or attachment to a motor vehicle is loaded or unloaded without being removed from the motor vehicle; and (3) Is not fabricated under a specification for cylinders, portable tanks, tank cars, or multi-unit tank car tanks.

Cargo tank motor vehicle means a motor vehicle with one or more cargo tanks permanently attached to or forming an integral part of the motor vehicle.

Class means hazard class. See hazard class.

Class 1. See 49 CFR 173.50 (Explosives)

Class 2. See 49 CFR 173.115 (Flammable gas)

Class 3. See 49 CFR 173.120 (Flammable liquid)

- Class 4.* See 49 CFR 173.124 (Flammable solid)
Class 5. See 49 CFR 173.128 (Oxidizer, organic peroxide)
Class 6. See 49 CFR 173.132 (Poisonous, infectious)
Class 7. See 49 CFR 173.403 (Radioactive)
Class 8. See 49 CFR 173.136 (Corrosive)
Class 9. See 49 CFR 173.140 (Miscellaneous hazardous material)

Closed transport vehicle means a transport vehicle or conveyance equipped with a securely attached exterior enclosure that during normal transportation restricts the access of unauthorized persons to the cargo space containing the Class 7 (radioactive) materials. The enclosure may be temporary or permanent, and, in the case of packaged materials, may be of the "see-through" type, and must limit access from top, sides, and bottom.

Closure means a device that closes an opening in a receptacle.

Combination packaging means a combination of packaging, for transport purposes, consisting of one or more inner packagings secured in a nonbulk outer packaging. It does not include a composite packaging.

Commerce means trade, traffic, transportation, or communication among the several States, or between a State and any place outside thereof, or within the District of Columbia, or a possession of the United States (other than the Trust Territory of the Pacific Islands), or between points in the same State but through a point outside thereof for profit.

Commercial motor vehicle (CMV) means a motor vehicle that has any of the following characteristics: (1) A gross vehicle weight, gross vehicle weight rating, gross combination weight, or gross combination weight rating of 4,537 kg (10,001 lb) or more; (2) Regardless of weight, is designed or used to transport 16 or more passengers, including driver; (3) Regardless of weight, is used in the transportation of hazardous materials and is required to be placarded pursuant to 49 CFR 172, Subpart F.

Composite packaging means a packaging consisting of an outer packaging and an inner receptacle, so constructed that the inner receptacle and the outer packaging form an integral packaging. Once assembled it remains thereafter an integrated single unit; it is filled, stored, shipped, and emptied as such.

Containment system means the assembly of components of the packaging intended to retain the radioactive contents during transportation.

Conveyance means (1) For transport by public highway or rail: any transport vehicle or large freight container; (2) For transport by water: any vessel, or any hold, compartment, or defined deck area of a vessel including any transport vehicle on board the vessel; and (3) For transport by aircraft, any aircraft.

Design means the description of a special form Class 7 (radioactive) material, a package, packaging, or LSA-III that enables those items to be fully identified. The description may

include specifications, engineering drawings, reports showing compliance with regulatory requirements, and other relevant documentation.

Designated facility means a hazardous waste treatment, storage, or disposal facility that has been designated on the manifest by the generator.

DOT/NRC/DOE-Approved Packagings (performance-based packagings): DOT, NRC, and DOE-approved and certified packagings must comply with all applicable federal regulations, specifications, and performance standards. When used onsite, the packages must be used in accordance with these agency authorizations.

Drum means a flat-ended or convex-ended cylindrical packaging made of metal, fiberboard, plastic, plywood, or other suitable material. This definition also includes packagings of other shapes made of metal or plastic (e.g., round taper-necked packagings or pail-shaped packagings) but does not include cylinders, jerricans, wooden barrels, or bulk packagings.

Employer means a person engaged in a business affecting commerce that has employees, but does not include the United States or any State or political subdivision of a State.

Employee means an employee of an employer who is employed in a business of their employer, which affects commerce.

Established Federal standard means any operative standard established by any agency of the United States and in effect on April 28, 1971, or contained in any Act of Congress in force on the date of enactment of the Williams-Steiger Occupational Safety and Health Act.

Excepted Packaging: See 49 CFR 173.403, Package, (1) "Excepted package means a packaging together with its excepted Class 7 (radioactive) materials as specified in 49 CFR 173.421 through 173.426 and 173.428."

Exclusive use (also referred to in other regulations as "sole use" or "full load") means sole use by a single consignor of a conveyance for which all initial, intermediate, and final loading and unloading are carried out in accordance with the direction of the consignor or consignee. The consignor and the carrier must ensure that any loading or unloading is performed by personnel having radiological training and resources appropriate for safe handling of the consignment. The consignor must issue specific instructions, in writing, for maintenance of exclusive-use shipment controls and include them with the shipping paper information provided to the carrier by the consignor.

Exemption onsite means a document issued by the RL Manager that authorizes a person to perform a function that is not otherwise authorized under this TSD.

F means degree Fahrenheit.

Facility Movement means the movement of material or packages between processing areas or storage areas within a specific facility or facility compound (e.g., Central Waste Complex

[CWC], T-Plant, 324 Building) covered by a facility SAR or other DOE-approved safety or authorization basis document(s). These movements are not considered onsite shipments or transfers. The safety basis for these facility movements must be adequately addressed in the facility SAR or other DOE-approved safety or authorization basis document(s).

Facility Worker means any individual located within the facility safety or authorization basis boundary or its emergency control area that is knowledgeable of the potential hazards and is trained to the facility emergency procedures. The consequences to the facility worker are typically evaluated qualitatively in the hazard evaluation rather than in the accident analysis. Any need for quantitative consequence analysis should be identified in the criteria document along with the basis for such a need.

Federal hazardous material transportation law means 49 USC 5101 et seq.

Fissile material means ^{238}Pu , ^{239}Pu , ^{241}Pu , ^{233}U , ^{235}U , or any combination of these radionuclides. The definition does not apply to unirradiated natural uranium and depleted uranium, and natural uranium or depleted uranium that has been irradiated in a thermal reactor. Certain additional exceptions are provided in 49 CFR 173.453.

Fissile material, controlled shipment means any shipment that contains one or more packages that have been assigned, in accordance with 49 CFR 173.457, nuclear criticality control transport indices greater than 10.

Gas means a material that has a vapor pressure greater than 300 kPa (43.5 psi) at 50 °C (122 °F) or is completely gaseous at 20 °C (68 °F) at a standard pressure of 101.3 kPa (14.7 psi).

Gross weight or Gross mass means the weight of a packaging plus the weight of its contents.

Hanford DOT-Equivalent Packagings (performance-based packagings): Hanford DOT-equivalent packagings must comply with Hanford-defined performance standards based on the Hanford Site transportation environment.

Hanford non-DOT-Equivalent Packagings (risk-based packagings): Hanford non-DOT-equivalent packagings are authorized under a dose consequence/risk assessment methodology approved by RL in this TSD.

Hazard class means the category of hazard assigned to a hazardous material under the definitional criteria of 49 CFR 173, "Shippers-General Requirements for Shipments and Packagings," and the provisions of the 49 CFR 172.101 table. A material may meet the defining criteria for more than one hazard class, but is assigned to only one hazard class.

Hazard Control: Within the TSD, the term hazard control is used in a broad context and is used to denote those requirements appropriate for the level of containment and communication, provided the possibility and consequences of credible accidents is taken into account. These control requirements should result in minimal acceptance of risk above the risks accepted in the context of existing Hazardous Material Regulations. For radioactive materials, appropriate

controls also need to be provided to ensure nuclear criticality safety and minimize personnel exposures in accordance with as low as reasonably achievable (ALARA) principles. The use of this term in the TSD in the broad context described above is consistent with its use in the transportation community and is not necessarily the same context contained in 10 CFR 830.

Hazardous material means a substance or material, which has been determined by the Secretary of Transportation to be capable of posing an unreasonable risk to health, safety, and property when transported in commerce, and which has been so designated. The term includes hazardous substances, hazardous waste, marine pollutants, and elevated temperature materials as defined in 49 CFR 171.8, materials designated as hazardous under the provisions of 49 CFR 172.101, "Purpose and use of hazardous materials table," and materials that meet the defining criteria for hazard classes and divisions in 49 CFR 173.

Hazardous substance for the purposes of this TSD, means a material, including its mixtures and solutions, that: (1) Is listed in Appendix A, 49 CFR 172.101; (2) Is in a quantity, in one package, which equals or exceeds the reportable quantity (RQ) listed in Appendix A in 49 CFR 172.101; and (3) When in a mixture or solution--

- (i) For radionuclides, conforms to paragraph 7 of Appendix A in 49 CFR 172.101.
- (ii) For other than radionuclides, is in a concentration by weight which equals or exceeds the concentration corresponding to the RQ of the material, as shown in the following table:

RQ pounds (kilograms)	Concentration by Weight	
	Percent	PPM
5000 (2270)	10	100,000
1000 (454)	2	20,000
100 (45.4)	0.2	2,000
10 (4.54)	0.02	200
1 (0.454)	0.002	20

The term does not include petroleum, including crude oil or any fraction thereof which is not otherwise specifically listed or designated as a hazardous substance in Appendix A to 49 CFR 172.101, and the term does not include natural gas, natural gas liquids, liquefied natural gas, or synthetic gas usable for fuel (or mixtures of natural gas and such synthetic gas).

Hazardous waste means, for the purposes of this TSD, any material that is subject to the Hazardous Waste Manifest Requirements of the U.S. Environmental Protection Agency (EPA) specified in 40 CFR 262, "Standards Applicable to Generators of Hazardous Waste."

Hazmat employee means a person who is employed by a hazmat employer and who in the course of employment directly affects hazardous materials transportation safety. This term includes an owner-operator of a motor vehicle that transports hazardous materials in commerce. This term includes an individual, including a self-employed individual, employed by a hazmat employer who, during the course of employment: (1) Loads, unloads, or handles hazardous materials; (2) Manufactures, tests, reconditions, repairs, modifies, marks, or otherwise represents containers, drums, or packagings as qualified for use in the transportation of hazardous materials; (3) Prepares hazardous materials for transportation; (4) Is responsible for safety of transporting hazardous materials; or (5) Operates a vehicle used to transport hazardous materials.

Hazmat employer means a person who uses one or more of its employees in connection with: transporting hazardous materials in commerce; causing hazardous materials to be transported or shipped in commerce; or representing, marking, certifying, selling, offering, manufacturing, reconditioning, testing, repairing, or modifying containers, drums, or packagings as qualified for use in the transportation of hazardous materials. This term includes an owner-operator of a motor vehicle that transports hazardous materials in commerce. This term also includes any department, agency, or instrumentality of the United States, a State, a political subdivision of a State, or an Indian tribe engaged in an activity described in the first sentence of this definition.

Head of Operations or Field Offices are terms used in DOE Order 460.1A to identify who is responsible for implementing the requirements of the order. For the Hanford Site, the RL Manager is identified as the individual responsible for implementing the DOE Order 460.1A requirements for RL and ORP.

Highway route controlled quantity means a quantity within a single package which exceeds: (1) 3,000 times the A_1 value of the radionuclides as specified in 49 CFR 173.435 for special form Class 7 (radioactive) material; (2) 3,000 times the A_2 value of the radionuclides as specified in 49 CFR 173.435 for normal form Class 7 (radioactive) material; or (3) 1,000 TBq (27,000 Ci), whichever is least.

IATA means International Air Transport Association.

ICAO means International Civil Aviation Organization.

IMO means International Maritime Organization.

Industrial Packaging: See 49 CFR 173.403, Package, (4) "Industrial package means a packaging that, together with its LSA material or SCO contents, meets the requirements of Sections 173.410 and 173.411. Industrial packages (IP) are categorized in 49 CFR 173.411 as either:

- (i) Industrial package Type 1 (IP-1)
- (ii) Industrial package Type 2 (IP-2)
- (iii) Industrial package Type 3 (IP-3)

Inner packaging means a packaging for which an outer packaging is required for transport. It does not include the inner receptacle of a composite packaging.

Inner receptacle means a receptacle that requires an outer packaging to perform its containment function. The inner receptacle may be an inner packaging of a combination packaging or the inner receptacle of a composite packaging.

Inter-area: An onsite shipment between two site-processing areas (i.e., 200 East Area to 200 West Area).

Intermediate packaging means a packaging that encloses an inner packaging or article and is itself enclosed in an outer packaging.

Intra-area: An onsite shipment between two facilities within a single site processing area (i.e., between T Plant and the CWC within the 200 West Area).

Intra-facility movement or intra-facility transfer means the movement of material between individual buildings within a facility or facility complex, or the surrounding compound area that is covered under a single facility SAR or other DOE-approved safety or authorization basis document.

Large truck means a truck over 4,536 kg (10,000 lb) gross vehicle weight rating including single unit trucks and truck tractors (Fatality Analysis Reporting System definition).

Limited quantity. 'Limited quantity,' when specified as such in a section applicable to a particular material, means the maximum amount of a hazardous material for which there is a specific labeling or packaging exception. See 49 CFR 171.8.

Limited quantity of Class 7 (radioactive) material means a quantity of Class 7 (radioactive) material not exceeding the materials package limits specified in 49 CFR 173.425 and conforming with requirements specified in 49 CFR 173.421. See 49 CFR 173.403.

Liquid means a material, other than an elevated temperature material, with a melting point or initial melting point of 20 °C (68 °F) or lower at a standard pressure of 101.3 kPa (14.7 psi). A viscous material for which a specific melting point cannot be determined must be subjected to the procedures specified in American Society for Testing and Materials (ASTM) D4359 "Standard Test Method for Determining Whether a Material is Liquid or Solid."

Liquid phase means a material that meets the definition of liquid when evaluated at the higher of the temperature at which it is offered for transportation or at which it is transported, not at the 37.8 °C (100 °F) temperature specified in ASTM D4359-84.

Low-Specific Activity: See 49 CFR 173.403, 'Low-Specific Activity' material means Class 7 (radioactive) material with limited specific activity which satisfies the descriptions and limits set forth below. Shielding materials surrounding the LSA material may not be considered in determining the estimated average specific activity of the package contents. LSA material must be in one of the following three groups:

- (1) LSA-I. (i) Ores containing only naturally occurring radionuclides (e.g., uranium, thorium) and uranium or thorium concentrates of such ores; or (ii) Solid unirradiated natural uranium or depleted uranium, or natural thorium or their solid or liquid compounds or mixtures; or (iii) Class 7 (radioactive) material, other than fissile material, for which the A_2 value is unlimited; or (iv) Mill tailings, contaminated earth, concrete, rubble, other debris, and activated material in which the Class 7 (radioactive) material is essentially uniformly distributed and the average specific activity does not exceed $10^{-6}A_2/g$.

- (2) LSA-II. (i) Water with tritium concentration up to 0.8 TBq/L (20.0 Ci/L); or (ii) Material in which the Class 7 (radioactive) material is distributed throughout and the average specific activity does not exceed $10^{-4} \text{ A}_2/\text{g}$ for solids and gases, and $10^{-5} \text{ A}_2/\text{g}$ for liquids.
- (3) LSA-III. Solids (e.g., consolidated wastes, activated materials) that meet the requirements of 49 CFR 173.468, "Test for LSA-III material," and which: (i) The Class 7 (radioactive) material is distributed throughout a solid or a collection of solid objects, or is essentially uniformly distributed in a solid compact binding agent (such as concrete, bitumen, ceramic, etc.); and (ii) The Class 7 (radioactive) material is relatively insoluble, or it is intrinsically contained in a relatively insoluble material so that, even under loss of packaging, the loss of Class 7 (radioactive) material per package by leaching when placed in water for 7 days would not exceed 0.1 A_2 ; and (iii) The average specific activity of the solid does not exceed $2 \times 10^{-3} \text{ A}_2/\text{g}$.

Low toxicity alpha emitters are: (1) Natural uranium, depleted uranium, and natural thorium; (2) Ores, concentrates or tailings containing ^{235}U , ^{238}U , ^{232}Th , ^{228}Th and ^{230}Th ; or (3) Alpha emitters with a half-life of less than 10 days.

Marking means a descriptive name, identification number, instructions, cautions, weight, specification, or UN marks, or combinations thereof, required by this subchapter on outer packagings of hazardous materials.

Material of trade means a hazardous material, other than a hazardous waste, that is carried on a motor vehicle: (1) For the purpose of protecting the health and safety of the motor vehicle operator or passengers; (2) For the purpose of supporting the operation or maintenance of a motor vehicle (including its auxiliary equipment); or (3) By a private motor carrier (including vehicles operated by a rail carrier) in direct support of a principal business that is other than transportation by motor vehicle.

Material poisonous by inhalation means: (1) A gas meeting the defining criteria in 49 CFR 173.115(c) and assigned to Hazard Zone A, B, C, or D in accordance with 49 CFR 173.116(a); (2) A liquid (other than as a mist) meeting the defining criteria in 49 CFR 173.132(a)(1)(iii) and assigned to Hazard Zone A or B in accordance with 49 CFR 173.133(a); or (3) Any material identified as an inhalation hazard by a special provision in column 7 of the 49 CFR 172.101 table.

Maximum Allowable Working Pressure or MAWP. For DOT specification cargo tanks used to transport liquid hazardous materials, see 49 CFR 178.345-1(k).

Maximum capacity means the maximum inner volume of receptacles or packagings.

Maximum net mass means the allowable maximum net mass of contents in a single packaging, or as used in 49 CFR 178, Subpart M, the maximum combined mass of inner packaging and the contents thereof.

Maximum normal operating pressure (MNOP) means the maximum gauge pressure that would develop in a receptacle in a period of 1 year, in the absence of venting or cooling, under the heat conditions specified in 10 CFR 71.71(c)(1).

Mixture means a material comprised of more than one chemical compound or element.

Mode means any of the following transportation methods; rail, highway, air, or water.

Motor carrier means a for-hire motor carrier or private motor carrier. The term includes a motor carrier's agents, officers, or representatives responsible for hiring, supervising, training, assigning, or dispatching a driver, or concerned with the installation, inspection, and maintenance of motor vehicle equipment or accessories or both.

Motor vehicle includes a vehicle, machine, tractor, trailer, or semi-trailer, or any combination thereof, propelled or drawn by mechanical power and used on the highways in the transportation of passengers or property. It does not include a vehicle, locomotive, or car operated exclusively on a rail or rails, or a trolley bus operated by electric power derived from a fixed overhead wire, furnishing local passenger transportation similar to street-railway service.

Multilateral approval means approval of a package or shipment by the relevant competent authority of the country of origin and of each country through or into which the package or shipment is to be transported. This definition does not include approval from a country over which Class 7 (radioactive) materials are carried in aircraft, if there is no scheduled stop in that country.

Name of contents means the proper shipping name as specified in 49 CFR 172.101.

National consensus standard means any standard or modification thereof which (1) has been adopted and promulgated by a nationally recognized standards-producing organization under procedures whereby it can be determined by the Secretary of Labor or by the Assistant Secretary of Labor that persons interested and affected by the scope or provisions of the standard have reached substantial agreement on its adoption; (2) was formulated in a manner which afforded an opportunity for diverse views to be considered; and (3) has been designated as such a standard by the Secretary or the Assistant Secretary, after consultation with other appropriate Federal agencies.

Natural thorium means thorium with the naturally occurring distribution of thorium isotopes (essentially 100 wt% of ²³²Th).

Navigable waters means waters of the United States, including the territorial seas.

"New Package" or "New" Package Specific Safety Document (New PSSD): Prior to implementation, this TSD requires RL Manager approval of all new packages or PSSDs that will contain fissile or greater than Type A quantities. The packages and PSSDs listed in Appendix A are considered to have RL Manager approval. Therefore, for the purpose of this TSD, a "New PSSD" is a safety document prepared to support the use of a new package or package design.

In addition, a “new package” or new package design means a completely new package design (including hardware and payload) or the circumstance where a feature of a previously approved package is changed such that its structural performance, under normal or accident conditions, compromises the margin of safety for criticality, shielding, or containment. Examples include the following: 1) A new cask or package design with unique payloads, not previously tested, analyzed, and/or approved is clearly considered a new packaging and would require a new PSSD approved by the RL Manager. 2) A previously approved package where a structural feature such as an overpack or impact limiter is removed that would compromise package performance under the accident conditions (e.g., drop, fire, etc.) would be considered a new package and would require a new PSSD approved by the RL Manager. 3) In contrast, a previously approved package where structural features are enhanced (e.g., changing material of construction from low-grade carbon steel to 304 stainless steel, adding an overpack, etc.) such that there is a negligible impact on performance. This circumstance might not be considered a new package based on this change alone and could be approved by the contractor assuming a negative USQ_T. Any significant design change should be discussed with RL to determine the package status in this regard.

Non-bulk packaging means a packaging which has: (1) A maximum capacity of 450 L (119 gal) or less as a receptacle for a liquid; (2) A maximum net mass of 400 kg (882 lb) or less and a maximum capacity of 450 L (119 gal) or less as a receptacle for a solid; or (3) A water capacity of 454 kg (1,000 lb) or less as a receptacle for a gas as defined in 49 CFR 173.115.

Non-fixed radioactive contamination means radioactive contamination that can be readily removed from a surface by wiping with an absorbent material. Non-fixed (removable) radioactive contamination is not significant if it does not exceed the limits specified in 49 CFR 173.443.

Non-reusable container means a packaging (container) whose reuse is restricted in accordance with the provisions of 49 CFR 173.28.

Normal form Class 7 (radioactive) material means Class 7 (radioactive) material that has not been demonstrated to qualify as “special form Class 7 (radioactive) material.”

North American Standard Inspection means the methodology used by State CMV safety inspectors to conduct safety inspections of CMVs. This consists of various levels of inspection of the vehicle or driver or both. The inspection criteria are developed by the Federal Motor Carrier Safety Administration in conjunction with the Commercial Vehicle Safety Alliance, an association of States, Canadian Provinces, and Mexico whose members agree to adopt these standards for inspecting CMVs in their jurisdiction.

N.O.S. means not otherwise specified.

N.O.S. description means a shipping description from the 49 CFR 172.101 table that includes the abbreviation n.o.s.

NPT means an American Standard taper pipe thread conforming to requirements of Federal Standard H28, part II, section VII. See 49 CFR 171.7(d)(12).

Nuclear safety authorization basis is the basis for the safe operation of a DOE nuclear facility. Nuclear safety authorization basis includes hazard classification documents, SARs, technical safety requirements (TSR), DOE-issued safety evaluation reports, and facility-specific commitments made to comply with DOE nuclear safety requirements.

Occupationally exposed hazmat employee means a hazmat employee whose duties involve exposure to ionizing radiation.

Offsite means outside the boundaries of the Hanford Site and on Hanford Site roadways, waterways, and land areas to which the public is given unrestricted access.

Offsite Public is represented by the maximum offsite individual, a hypothetical receptor located at or beyond the Hanford Site boundary (or public access area) at the distance and in the direction from the point of release at which the maximum dose occurs. This receptor location is used for evaluating the need for safety class structures, systems, or components (SSC). Additional receptor locations may be selected for reference purposes.

Offsite Shipment: A government, DOE contractor, or commercial shipment of radioactive or other hazardous material or waste leaving the Hanford Site, or shipment onsite to a non-DOE entity, or an onsite movement of these materials and wastes transported over roadways where the general public has unrestricted access. These shipments must be made in full compliance to applicable federal and state regulations. (Note that a road closure on the Hanford Site that restricts public access can change an offsite shipment into an onsite shipment).

Onsite: Inside the boundaries of the Hanford Site where public access is restricted.

Onsite Public is characterized by a hypothetical receptor within the Hanford Site boundary at locations bounded by: (1) The near bank of the Columbia River and (2) Highway 240 traversing the Hanford Site, in the direction and at the distance of highest dose or exposure. This receptor is used for reference purposes and is considered only when accident consequences at either of these locations exceed that for the maximum offsite individual.

*Onsite SARP*¹: The onsite SARP is a technical and safety document that demonstrates radioactive material packaging compliance to Hanford Site T&P standards. It describes the T&P system, the authorized payload and applicable operational controls. It also describes the testing and analyses performed to demonstrate compliance to the Hanford Site-equivalent T&P standards. An onsite SARP is generally issued for multi-purpose, multi-user-packaging systems.

*Onsite SEP*¹: The onsite SEP is a technical and safety document that demonstrates radioactive material packaging compliance to Hanford Site T&P standards. It describes the T&P system, the authorized payload, and applicable operational controls. It also describes the testing and analyses performed to demonstrate compliance to the Hanford Site-equivalent T&P standards.

¹ Obsolete term – replaced by PSSD.

This type of approval is generally based on a previously approved packaging. It is generally issued for single trip, single purpose, and single-user packaging systems.

Onsite Shipment or Onsite Transfer: A shipment of radioactive or other hazardous material or waste transported by motorized vehicle within the boundaries of the Hanford Site between DOE facilities, and over roadways where public access is controlled or otherwise restricted. These shipments are exempted from federal and state regulations, but must be controlled to provide equivalent safety. Onsite transfer is interchangeable with onsite shipment, but for clarity will not be used within this TSD.

Onsite Worker or “co-located” worker is represented by the maximum onsite individual, a hypothetical receptor located at the distance (at least 100 m [328 ft] or at the boundary of the facility or its emergency control area) and in the direction from the point of release at which the maximum dose occurs. This receptor location is used for evaluating the need for safety significant SSCs. Because particular factors affect the relevance of the onsite worker location, the criteria document should contain the basis for such selection.

Operations and Maintenance (O&M) Manual: A document issued for onsite packaging systems that identifies the packaging system and its components; the authorized payload including loading, closure and unloading instructions; inspection, testing and maintenance requirements and procedures; cask handling and tiedown for transport; and any other administrative or operational controls necessary to achieve required levels of safety.

Operator means a person who controls the use of an aircraft, vessel, or vehicle.

Outage or ullage means the amount by which a packaging falls short of being liquid full, usually expressed in percent by volume.

Outer packaging means the outermost enclosure of a composite or combination packaging together with any absorbent materials, cushioning and any other components necessary to contain and protect inner receptacles or inner packagings.

Overpack, except as provided in 49 CFR 178, Subpart K, means an enclosure that is used by a single consignor to provide protection or convenience in handling of a package or to consolidate two or more packages. Overpack does not include a transport vehicle, freight container, or aircraft unit load device. Examples of overpacks are one or more packages: (1) Placed or stacked onto a load board such as a pallet and secured by strapping, shrink wrapping, stretch wrapping, or other suitable means; or (2) Placed in a protective outer packaging such as a box or crate.

Package: See 49 CFR 171.8, ‘Package or Outside Package’ means a packaging plus its contents. For radioactive material, see 49 CFR 173.403, ‘Package’ means, for Class 7 (radioactive) materials, the packaging together with its radioactive contents as presented for transport.

Package Safety Analysis Assessment (PSAA) is a safety analysis assessment document prepared to demonstrate that an onsite package meets the packaging design and performance criteria

identified in the TSD. The PSAA will address the criteria that are applicable to the equivalent packaging method employed.

Package Specific Safety Document (PSSD) is a safety document prepared to demonstrate the safety associated with a specific package. PSSDs include existing RL approved onsite SARPs, onsite SEPs, Certificate of Compliance (CoC) or Certificate of Competent Authority for the NRC, DOT, DOD, and DOE packaging systems. For onsite packages, the PSSDs are evaluations that demonstrate how the criteria of the TSD are met.

Packaging: See 49 CFR 171.8, 'Packaging' means a receptacle and any other components or materials necessary for the receptacle to perform its containment function in conformance with the minimum packing requirements of this TSD. For radioactive material, see 49 CFR 173.403, 'Packaging' means, for Class 7 (radioactive) materials, the assembly of components necessary to ensure compliance with the packaging requirements of this subpart. It may consist of one or more receptacles, absorbent materials, spacing structures, thermal insulation, radiation shielding, and service equipment for filling, emptying, venting and pressure relief, and devices for cooling or absorbing mechanical shocks. The conveyance, tiedown system, and auxiliary equipment may sometimes be designated as part of the packaging.

Packaging system is packaging and ancillary equipment necessary to conduct shipments. It may include the transportation vehicle.

Packing group means a grouping according to the degree of danger presented by hazardous materials. Packing Group I indicates great danger; Packing Group II, medium danger; Packing Group III, minor danger. See 49 CFR 172.101(f).

Person means an individual, firm, co-partnership, corporation, company, association, joint-stock association, including any trustee, receiver, assignee, or similar representative thereof; or government, Indian tribe, or agency or instrumentality of any government or Indian tribe when it offers hazardous material for transportation in commerce or transports hazardous material to further a commercial enterprise, but such term does not include: (1) The United States Postal Service; (2) For the purposes of 49 USC 5123 and 5124, any agency or instrumentality of the Federal Government.

Portable tank means a bulk packaging (except a cylinder having a water capacity of 454 kg [1,000 lb] or less) designed primarily to be loaded onto, or on, or temporarily attached to a transport vehicle or ship and equipped with skids, mountings, or accessories to facilitate handling of the tank by mechanical means. It does not include a cargo tank, tank car, multi-unit tank car tank, or trailer carrying 3AX, 3AAX, or 3T cylinders.

Preferred route or Preferred highway is a highway for shipment of highway route controlled quantities of radioactive materials so designated by a State routing agency, and any Interstate System highway for which an alternative highway has not been designated by such State agency as provided by 49 CFR 397.103.

Primary hazard means the hazard class of a material as assigned in the 49 CFR 172.101 table.

Proper shipping name means the name of the hazardous material shown in Roman print (not italics) in 49 CFR 172.101.

P.s.i. or psi means pounds per square inch.

P.s.i.a. or psia means pounds per square inch absolute.

P.s.i.g. or psig means pounds per square inch gauge.

Public Access Roadway: A roadway where the general public has unrestricted access. This can include a state highway, county or city road, or Hanford Site roadway; e.g., Route 10 between the 1100 Area and the Wye Barricade.

Q component is a packaging part (component) identified during the design effort as important to safety. For guidance in identifying parts, see Appendix A of Regulatory Guide 7.10, *Establishing Quality Assurance Programs for Packaging Used in the Transport of Radioactive Material*.

Radiation level means the radiation dose-equivalent rate expressed in millisievert(s) per hour or mSv/h (millirem(s) per hour or mrem/h). Neutron flux densities may be converted into radiation levels according to Table 1.

Table 1--Neutron Fluence Rates to be Regarded as Equivalent to a Radiation Level of 0.01 mSv/h (1 mrem/h)¹.

Energy of Neutron	Flux density equivalent to 0.01 mSv/H (1 mrem/h) neutrons per square centimeter per second (n/cm ² /s)
Thermal (2.510E-8)MeV	272.0
1 keV	272.0
10 keV	281.0
100 keV	47.0
500 deV	11.0
1 MeV	7.5
5 MeV	6.4
10 MeV	6.7

¹ Flux densities equivalent for energies between those listed in this table may be obtained by linear interpolation.

Radioactive contents means a Class 7 (radioactive) material, together with any contaminated liquids or gases within the package.

Radioactive instrument or article means any manufactured instrument or article such as an instrument, clock, electronic tube or apparatus, or similar instrument or article having Class 7 (radioactive) material in gaseous or nondispersible solid form as a component part.

Radioactive material means any material having a specific activity greater than 70 Bq/g (0.002 μ ci/g) (see definition of "specific activity").

Receptacle means a containment vessel for receiving and holding materials, including any means of closing.

Reportable quantity (RQ) for the purposes of this TSD means the quantity specified in column 2 of the appendix to 49 CFR 172.101 for any material identified in column 1 of the appendix.

Residue means the hazardous material remaining in a packaging, including a tank car, after its contents have been unloaded to the maximum extent practicable and before the packaging is either refilled or cleaned of hazardous material and purged to remove any hazardous vapors.

RSPA means the Research and Special Programs Administration, DOT, Washington, D.C. 20590.

Safety basis means the documented safety analysis and hazard controls that provide reasonable assurance that a DOE nuclear facility can be operated safely in a manner that adequately protects workers, the public, and the environment. (The TSD safety basis refers to, but does not include PSSDs. The PSSDs for onsite packages are evaluations that demonstrate how the criteria of the TSD are met.)

SCF (standard cubic foot) means 1 ft³ of gas measured at 60 °F and 14.7 psia.

Shipment means the shipping or transporting of goods.

Shipping paper means a shipping order, bill of lading, manifest, or other shipping document serving a similar purpose and containing the information required by 49 CFR 172.202, 172.203 and 172.204.

Single packaging means a nonbulk packaging other than a combination packaging.

Site Controlled Access Roadway: Hanford Site roads where public access is controlled or otherwise restricted. Controls include barricades, fences, signs, and Hanford Patrol checkpoints. Road closures can be executed on certain roads to convert a public access roadway to a controlled access roadway.

Solid means a material that is not a gas or a liquid.

Solution means any homogeneous liquid mixture of two or more chemical compounds or elements that will not undergo any segregation under conditions normal to transportation.

Special form Class 7 (radioactive) material means Class 7 (radioactive) material that satisfies the following conditions: (1) It is either a single solid piece or is contained in a sealed capsule that can be opened only by destroying the capsule; (2) The piece or capsule has at least one dimension not less than 5 mm (0.2 in.); and (3) It satisfies the test requirements of 49 CFR 173.469. Special form encapsulations designed in accordance with the requirements of 49 CFR 173.389(g) in effect on June 30, 1983 (see 49 CFR 173, revised as of October 1, 1982), and constructed prior to July 1, 1985, and special form encapsulations designed in accordance

with the requirements of 49 CFR 173.403 in effect on March 31, 1996 (see 49 CFR 173, revised as of October 1, 1995), and constructed prior to April 1, 1997, may continue to be used. Any other special form encapsulation must meet the requirements of this paragraph.

Specific activity of a radionuclide means the activity of the radionuclide per unit mass of that nuclide. The specific activity of a material in which the radionuclide is essentially uniformly distributed is the activity per unit mass of the material.

Specification packaging means a packaging conforming to one of the specifications or standards for packagings in 49 CFR 178 or 179.

Standard means a standard that requires conditions, or the adoption or use of one or more practices, means, methods, operations, or processes, reasonably necessary or appropriate to provide safe or healthful employment and places of employment.

State means a State of the United States, the District of Columbia, the Commonwealth of Puerto Rico, the Commonwealth of the Northern Mariana Islands, the Virgin Islands, American Samoa, Guam, or any other territory or possession of the United States designated by the Secretary.

Strong outside container means the outermost enclosure that provides protection against the unintentional release of its contents under conditions normally incident to transportation.

Subsidiary hazard means a hazard of a material other than the primary hazard. (See primary hazard). Table in 49 CFR 172.101 or 172.101 table means the Hazardous Materials Table (HMT) in 49 CFR 172.101.

Surface Contaminated Object (SCO): See 49 CFR 173.403, 'Surface Contaminated Object' means a solid object that is not radioactive, but has Class 7 (radioactive) material distributed on any of its surfaces. SCO must be in one of two groups with surface activity not exceeding the following limits:

- (1) SCO-I: A solid object on which: (i) The nonfixed contamination on the accessible surface averaged over 300 cm² (or the area of the surface if less than 300 cm²) does not exceed 4 Bq/cm² (10⁻⁴ μCi/cm²) for beta and gamma and low toxicity alpha emitters, or 0.4 Bq/cm² (10⁻⁵ μCi/cm²) for alpha emitters; (ii) The fixed contamination on the accessible surface averaged over 300 cm² (or the area of the surface if less than 300 cm²) does not exceed 4 x 10⁴ Bq/cm² (1.0 μCi/cm²) for beta and gamma and low toxicity alpha emitters, or 4 x 10³ Bq/cm² (0.1 μCi/cm²) for all other alpha emitters; and (iii) The nonfixed contamination plus the fixed contamination on the inaccessible surface averaged over 300 cm² (or the area of the surface if less than 300 cm²) does not exceed 4 x 10⁴ Bq/cm² (0.1 μCi/cm²) for beta and gamma and low toxicity alpha emitters, or 4 x 10³ Bq/cm² (0.1 μCi/cm²) for all other alpha emitters.
- (2) SCO-II: A solid object on which the limits for SCO-I are exceeded and on which: (i) The nonfixed contamination on the accessible surface averaged over 300 cm² (or the area of the surface if less than 300 cm²) does not exceed 400 Bq/cm² (10⁻² μCi/cm²) for beta and

gamma and low toxicity alpha emitters or 40 Bq/cm^2 ($10^{-3} \text{ } \mu\text{Ci/cm}^2$) for all other alpha emitters; (ii) The fixed contamination on the accessible surface averaged over 300 cm^2 (or the area of the surface if less than 300 cm^2) does not exceed $8 \times 10^5 \text{ Bq/cm}^2$ ($20 \text{ } \mu\text{Ci/cm}^2$) for beta and gamma and low toxicity alpha emitters, or $8 \times 10^4 \text{ Bq/cm}^2$ ($2 \text{ } \mu\text{Ci/cm}^2$) for all other alpha emitters; and (iii) The nonfixed contamination plus the fixed contamination on the inaccessible surface averaged over 300 cm^2 (or the area of the surface if less than 300 cm^2) does not exceed $8 \times 10^5 \text{ Bq/cm}^2$ ($20 \text{ } \mu\text{Ci/cm}^2$) for beta and gamma and low toxicity alpha emitters, or $8 \times 10^4 \text{ Bq/cm}^2$ ($2 \text{ } \mu\text{Ci/cm}^2$) for all other alpha emitters.

Technical name means a recognized chemical name or microbiological name currently used in scientific and technical handbooks, journals, and texts. Generic descriptions are authorized for use as technical names, provided they readily identify the general chemical group or microbiological group. Examples of acceptable generic chemical descriptions are organic phosphate compounds, petroleum aliphatic hydrocarbons, and tertiary amines. For proficiency testing only, generic microbiological descriptions such as bacteria, mycobacterium, fungus, and viral samples may be used. Except for names that appear in Subpart B of 49 CFR 172, trade names may not be used as technical names.

Transfer: Any onsite shipment or onsite transport of hazardous materials, substances, and wastes, including those that are radioactive, between facilities where the facility boundaries are described and covered under a facility SAR or other DOE-approved safety or authorization basis document(s). For example, an onsite shipment between one of the 200 West Area "tank farms" to the CWC would be considered an onsite shipment or transfer. A movement of material or waste in or around the confines of the CWC would not be considered an onsite shipment or transfer from a transportation safety perspective. Such movements should be properly addressed in the CWC Facility SAR.

Transport index (TI) means the dimensionless number (rounded up to the next tenth) placed on the label of a package to designate the degree of control to be exercised by the carrier during transportation. The TI is determined as follows: (1) For nonfissile material packages, the number determined by multiplying the maximum radiation level in mSv/h at 1 m (3.3 ft) from the external surface of the package by 100 (equivalent to the maximum radiation level in mrem/h at 1 m [3.3 ft]); or (2) For fissile material packages, the number determined by multiplying the maximum radiation level in mSv/h at 1 m (3.3 ft) from any external surface of the package by 100 (equivalent to the maximum radiation level in mrem/h at 1 m [3.3 ft]) or, for criticality control purposes, the number obtained by dividing 50 by the allowable number of packages that may be transported together, whichever number is larger.

Transport vehicle means a cargo-carrying vehicle such as an automobile, van, tractor, truck, semitrailer, tank car, or rail car used for the transportation of cargo by any mode. Each cargo-carrying body (trailer, rail car, etc.) is a separate transport vehicle.

TSD Exemption: A documented exception or deviation to any TSD requirement approved by the RL Manager. The TSD exemption process is defined in Chapter IX for nonroutine shipments.

Type A Packaging: See 49 CFR 173.403, Package, (2) 'Type A package' means a packaging that, together with its radioactive contents limited to A_1 or A_2 as appropriate, meets the requirements of 49 CFR 173.410 and 173.412 and is designed to retain the integrity of containment and shielding required by this part under normal conditions of transport as demonstrated by the tests set forth in 49 CFR 173.465, "Type A packaging tests," or 173.466, "Additional tests for Type A packagings designed for liquids and gases," as appropriate. A Type A package does not require Competent Approval Authority.

Type B Packaging: See 49 CFR 173.403, Package, (3) 'Type B package' means a Type B packaging that, together with its radioactive contents, is designed to retain the integrity of containment and shielding required by this part when subjected to the normal conditions of transport and hypothetical accident test conditions set forth in 10 CFR 71.

Type A quantity means a quantity of Class 7 (radioactive) material, the aggregate radioactivity which does not exceed A_1 for special form Class 7 (radioactive) material or A_2 for normal form Class 7 (radioactive) material, where A_1 and A_2 values are given in 49 CFR 173.435 or are determined in accordance with 49 CFR 173.433.

Type B quantity means a quantity of material greater than a Type A quantity.

UN standard packaging means a packaging conforming to standards in the UN recommendations on the Transport of Dangerous Goods.

Unilateral approval means approval of a package solely by the competent authority of the country of origin.

United States means a State of the United States, the District of Columbia, the Commonwealth of Puerto Rico, the Commonwealth of the Northern Mariana Islands, the Virgin Islands, American Samoa, Guam, or any other territory or possession of the United States designated by the Secretary.

Unirradiated thorium means thorium containing not more than 10^{-7} grams ^{233}U per gram of ^{232}Th .

Unirradiated uranium means uranium containing not more than 10^{-6} grams plutonium per gram of ^{235}U and a fission product activity of not more than 9 MBq (0.24 mCi) of fission products per gram of ^{235}U .

Uranium--natural, depleted or enriched means the following: (1) "Natural uranium" means uranium with the naturally occurring distribution of uranium isotopes (approximately 0.711 wt% ^{235}U , and the remainder essentially ^{238}U). (2) "Depleted uranium" means uranium containing less ^{235}U than the naturally occurring distribution of uranium isotopes. (3) "Enriched uranium" means uranium containing more ^{235}U than the naturally occurring distribution of uranium isotopes.

USQ_T is an unreviewed safety question (USQ) process configured to evaluate onsite packaging and transportation issues.

Vessel includes every description of watercraft, used or capable of being used, as a means of transportation on the water.

2.0 ACRONYMS AND ABBREVIATIONS

AC	accident condition
ANSI	American National Standards Institute
ARF	airborne release fraction
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BPVC	boiler and pressure vessel code
Bq	becquerel
CDL	commercial driver's license
CFR	<i>Code of Federal Regulations</i>
Ci	curie
CMV	commercial motor vehicle
CoC	certificate of compliance
CSB	Canister Storage Building
CWC	Central Waste Complex
DCF	dose conversion factor
DDT	deflagration to detonation transition
DE-Ci	dose-equivalent curie
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
DR	damage ratio
EPA	U.S. Environmental Protection Agency
ES&H	environment, safety, and health
FMCSR	Federal Motor Carrier Safety Regulations
GIS	Geographic Information System
GSA	Government Services Administration
HAC	hypothetical accident condition
hazmat	hazardous material (commonly includes hazardous materials, substances, and wastes)
HEMP	<i>Hanford Emergency Management Plan</i>
HLW	high-level waste
HMPS	Hanford Minimum Packaging Standards
HMSR	<i>Hazardous Material Shipment Record</i>
HMT	Hazardous Materials Table
HNF	Hanford Nuclear Facility (document identifier)
IAEA	International Atomic Energy Agency
ICV	inner containment vessel
IP	industrial package
ISMS	Integrated Environment, Safety and Health Management System

ISO	International Standards Organization
LFL	lower flammability limit
LPF	leakpath factor
LSA	low-specific activity
MAR	material at risk
MNOP	maximum normal operating pressure
NC	normal condition
NCT	normal conditions of transport
NQA-1	Nuclear Quality Assurance
NRC	U.S. Nuclear Regulatory Commission
O&M	Operations and Maintenance
ORP	Office of River Protection
ORRSR	<i>Onsite Routine Radioactive Shipment Record</i>
POC	Patrol Operations Center
PSAA	Package Safety Analysis Assessment
psi	pounds per square inch
psia	pounds per square inch, absolute
PSSD	package specific safety document
QA	quality assurance
QAP	quality assurance program
RCRA	<i>Resource Conservation and Recovery Act of 1976</i>
RCT	radiological control technician
RF	respirable fraction
RL	Richland Operations Office
RQ	reportable quantity
RSR	<i>Radioactive Shipment Record</i>
s/m ³	seconds per cubic meter
SAR	safety analysis report (generally prepared for facilities or specific projects)
SARP	safety analysis report for packaging
SCO	surface contaminated object
SED	strain energy density
SEP	safety evaluation for packaging
SNM	special nuclear material
SSC	structures, systems, or components
SWITS	Solid Waste Information Tracking System
T&P	transportation and packaging
TI	transport index
TRU	transuranic
TSD	transportation safety document
TSL	Triaxial Strain Limit
TSR	technical safety requirement
UHWI	<i>Uniform Hazardous Waste Manifest</i>
USC	<i>United States Code</i>
USQ	unreviewed safety question
USQ _T	unreviewed safety question process for transportation
USQ _T D	unreviewed safety question process for transportation determination

USQ_TS unreviewed safety question process for transportation screening
V/R velocity/radius
WAC *Washington Administrative Code*
WLL working load limit

3.0 REFERENCES

- 10 CFR 71, "Packaging and Transportation of Radioactive Material," *Code of Federal Regulations*, as amended.
- 40 CFR 262, "Standards Applicable to Generators of Hazardous Waste," *Code of Federal Regulations*, as amended.
- 49 CFR 171, "General Information, Regulations, and Definitions," *Code of Federal Regulations*, as amended.
- 49 CFR 172, "Hazardous Materials Table, Special Provisions, Hazardous Materials Communications, Emergency Response Information, and Training Requirements," *Code of Federal Regulations*, as amended.
- 49 CFR 173, "Shippers-General Requirements for Shipments and Packagings," *Code of Federal Regulations*, as amended.
- 49 CFR 178, "Specifications for Packagings," *Code of Federal Regulations*, as amended.
- 49 CFR 179, "Specifications for Tank Cars," *Code of Federal Regulations*, as amended.
- 49 CFR 397, "Transportation of Hazardous Materials; Driving and Parking Rules," *Code of Federal Regulations*, as amended.
- 49 USC, "Transportation," *United States Code*, Washington, D.C.
- ASTM D4359, "Standard Test for Determining Whether a Material is a Liquid or a Solid," *American Society for Testing and Materials*, West Conshohocken, PA.
- DOE Order 460.1A, *Packaging and Transportation Safety*, U.S. Department of Energy, Washington, D.C.
- Regulatory Guide 7.10, *Establishing Quality Assurance Programs for Packaging Used in the Transport of Radioactive Material*, U.S. Nuclear Regulatory Commission, Washington, D.C.

CHAPTER III. SITE DESCRIPTION

This chapter describes the Hanford Site boundaries and roadways used to transport hazardous materials and wastes. It describes exclusion areas and facilities, and their principle missions. A Hanford Site map and Internet links are provided for more detailed site and area maps.

1.0 PHYSICAL DESCRIPTION

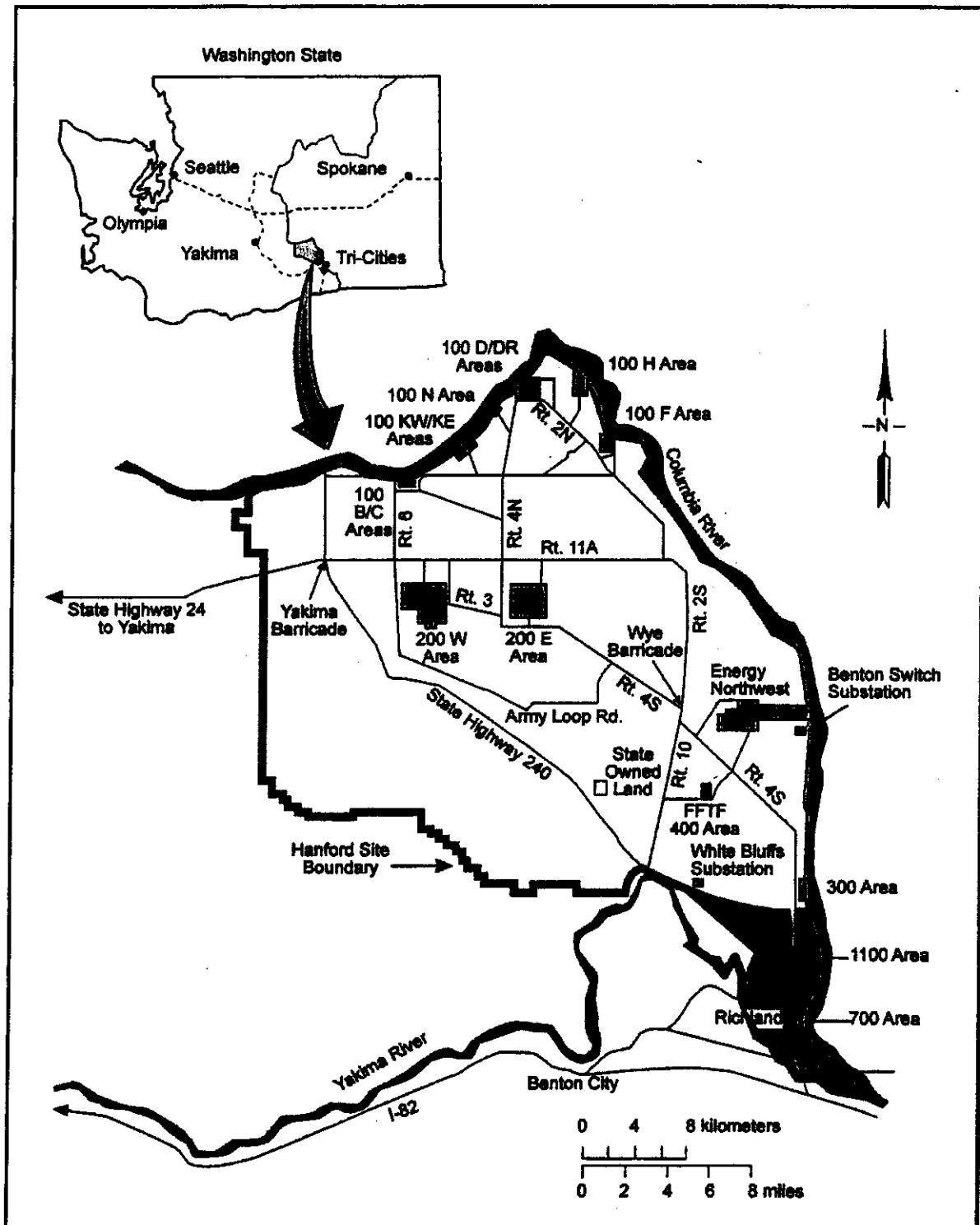
The Hanford Site covers approximately 1,450 km² (560 mi²) of semiarid land along the Columbia River in southeastern Washington State. It is owned by the U.S. Government and managed by the RL and ORP. The city of Richland adjoins the southeastern most portion of the Hanford Site boundary and is the nearest population center.

In early 1943, the U.S. Army Corps of Engineers selected the Hanford Site as the location for plutonium production for national defense. For over 20 years, activities were primarily dedicated to the continuation of plutonium production and managing the waste generated. In later years, activities became increasingly diverse, involving research and development for advanced reactors and renewable energy technologies. The end of the Cold War brought the shutdown of most of the Hanford Site's plutonium production and management facilities. Current missions are to safely cleanup and manage the legacy waste on the Hanford Site and to develop and deploy science and technology (DOE/RL-96-92, *Hanford Strategic Plan*).

The Hanford Site is divided into numerically designated areas. These areas served as the location for reactor, chemical separation, and related activities for the production and purification of special nuclear materials (SNM) and other nuclear activities. The reactors are located along the Columbia River in the 100 Areas. The reactor fuel reprocessing units are in the 200 Areas, which are on a plateau approximately 11 km (7 mi) from the Columbia River. The 300 Area, located adjacent to and north of Richland, contains the reactor fuel manufacturing plants and research and development laboratories. The 400 Area, 8 km (5 mi) northwest of the 300 Area, contains the Fast Flux Test Facility designed for testing liquid metal reactor systems. The 600 Area covers all locations within the Hanford Site boundary not specifically given an area designation. Adjacent to and north of Richland, the 1100 Area contains offices associated with administration, maintenance, transportation, and materials procurement and distribution. Offices also are located in the 700 Area, which is in downtown Richland.

The regional public highway network traversing the Hanford Site (Washington State Highways 24 and 240), nonrestricted access roadways (Route 10 and portions of Route 4S located south of the Wye Barricade), and restricted access roadways are shown in Figure 3-1. Roadways east of the Yakima Barricade, north of the Wye Barricade, and within the 300 and 400 areas are restricted to authorized personnel only. Other roadways are subject to such restrictions or closure as the DOE might require.

Figure 3-1. Hanford Site Major Roads.



T0108013.1

Figure 3-1 shows the major roads throughout the Hanford Site. These roads are classified as either primary or secondary routes. The primary routes include Routes 4N, 4S, 10, 2S, 3, 6, and 11A as well as various avenues within each area. The primary routes are constructed of bituminous asphalt (usually 5 cm [2 in.] thick, but the thickness of the asphalt layer will vary with each road) with an underlying aggregate base in accordance with DOT requirements. The secondary routes are constructed of layers of an oil-and-rock mixture with an underlying aggregate base. The aggregate base consists of various types and sizes of rock found onsite. The present load-bearing capacities of these roads are unknown; however, loads as large as 9.8 kg/cm² (140 psi) have been transported without observable damage to road surfaces. All roads meet the requirements for the American Association of State Highway and Transportation Officials HS-20-44 load rating (AASHTO 1983). An HS-20-44 loading represents a two-axle tractor (front-axle loading of 3,630 kg [8,000 lb] and rear-axle loading of 14,500 kg [32,000 lb] plus a single-axle trailer with a 14,500 kg [32,000 lb] axle loading).

Standard traffic control signs are used throughout the Hanford Site (e.g., octagonal stop signs and triangular yield signs). Speed limits are posted throughout the Hanford Site, and the maximum posted speed is 88 km/h (55 mi/h) on major thoroughfares. Inside the various areas, posted speeds are reduced to a maximum of 56 km/h (35 mi/h) and held to speeds as low as 24 km/h (15 mi/h).

The Hanford Site railroad is not discussed because all operations have ceased and the trackage and roadbed are no longer maintained.

2.0 PUBLIC ACCESS

The Hanford Site boundary (Figure 3-1) is marked by fences and signs. With the exception of one area, public access to other than roadways is restricted. Fences, signs, and security patrols are used to enforce the access restrictions.

On the south end of the Hanford Site, the public has access to a foot and bike path located near the Columbia River. The path extends between the south boundary of the Hanford Site and the 300 Area. The public has free access to the Columbia River where the river flows along the Hanford Site boundary. Access includes several islands located in that stretch of the river.

As noted in Chapter III, Section 1.0, Physical Description, several public highways pass through the Hanford Site. The public is permitted free access through Washington State Highways 24 and 240. In addition, the public also is permitted free access to the following Hanford Site roadways:

- Route 4S from the intersection with Horn Road to the Wye Barricade
- The two roads from 4S that go into the Energy Northwest area
- Kentucky Boulevard from 4S to the 400 Area
- Alabama Boulevard from Kentucky Boulevard to Route 10
- Route 10 from Washington State Highway 240 to Route 4S
- George Washington Way between Horn Road and Route 4S.

3.0 MAPS (LINKAGE TO OTHER SITE MAPS)

A complete set of site maps locating and describing the Hanford Site and the various exclusion areas (e.g., 200 East, 200 West) are located on the Hanford Internet site at:
<http://www.hanford.gov/top/maps.html>

4.0 PRIMARY ONSITE TRANSPORT ROUTES

Transportation routes are generally specified for each onsite packaging system in the onsite PSSD. When routes are not specified, it is expected that shipments be executed using the most direct route on principal site roadways. Back roads, unimproved roads, and shortcuts should not be used unless specifically authorized in the onsite PSSD transportation plan, project plan, health and safety plan, or are required by the activity, or other improved routes are not available.

The most common routes for transporting hazardous materials, substances, and wastes are as follows:

- 1100 Area to 300 Area: Take Route 4S north to 300 Area
- 1100 Area to 400 Area: Take Route 4S north to 400 Area
- 1100 Area to 200E: Take Route 4S north to 200E Area
- 1100 Area to 200W: Take Route 4S north to Route 3, then west to 200W Area

- 300 Area to 1100 Area: Take Route 4S south to 1100 Area
- 300 Area to 200W Area: Take Route 4S north to Route 3, then west to 200W Area

- 200E Area to 200W Area: Take Route 3 west to 200W Area
- 200E Area to 300 Area: Take Route 4S to 300 Area
- 200E Area to 1100 Area: Take Route 4S south to 1100 Area

- 200W Area to 200E Area: Take Route 3 east to 200E Area
- 200W Area to 300 Area: Take Route 3 east to Route 4S, then south to 300 Area
- 200W Area to 1100 Area: Take Route 3 east to Route 4S, then south to 1100 Area

- 100 Areas to 200E Area: As appropriate, take Route 1 or Route 2N to Route 4N, then turn south on Route 4N to Route 3, then turn east into 200E Area

- 100 Areas to 200W: As appropriate, take Route 1 or Route 2N to Route 4N, then turn south on Route 4N to Route 3, then turn west to 200W Area

- 100 Areas to 1100 Area: As appropriate, take Route 1 or Route 2N to Route 4N, then turn south onto Route 4N. Route 4N turns into Route 4S. Follow Route 4S to 1100 Area.

5.0 REFERENCES

AASHTO, 1983, American Association of State Highway and Transportation Officials
HS-20-44, Washington, D.C.

DOE/RL-96-92, *Hanford Strategic Plan*, U.S. Department of Energy, Richland Operations
Office, Richland, Washington.

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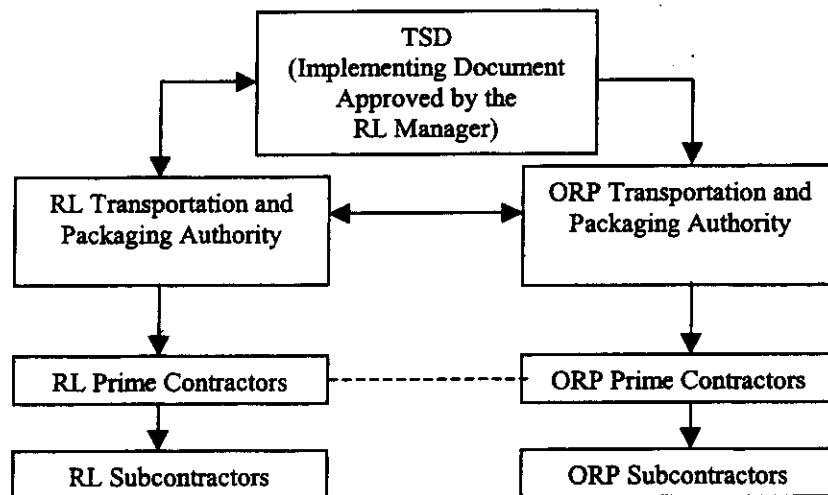
CHAPTER IV. ORGANIZATIONAL RESPONSIBILITIES

This chapter identifies RL and contractor responsibilities and interfaces within the transportation safety program as defined in this TSD and specific package development and approval. It identifies Internet links to more detailed contractor charters that describe organizational responsibilities for program implementation.

1.0 SITE ORGANIZATIONAL STRUCTURE

RL and ORP have sitewide responsibility for onsite T&P activities. RL and ORP implement the Hanford Sitewide T&P safety program through the RL Manager-approved sitewide TSD (see Figure 4-1). The sitewide TSD applies to all RL and ORP, and RL and ORP Hanford Site prime contractors and their major subcontractors who perform hazardous material T&P activities in association with their project or mission.

Figure 4-1. Hanford Site TSD Program.



The Hanford Site Internet sites at <http://www.hanford.gov> identify the RL, ORP, and prime contractor structure at the Hanford Site and identify appropriate senior management contacts. Senior managers or their delegates provide detailed organizational charts, identify key positions, roles and responsibilities, and lines of authority.

Within this organizational structure, RL is responsible for and recommends approval of the technical content and methodology used to provide adequate levels of control and safety for the onsite shipment of hazardous materials, substances, and wastes. Each prime contractor is responsible for complying with the provisions of this TSD.

2.0 RL AND ORP ROLES AND INTERFACES

RL and ORP are jointly responsible for implementing the Hanford Site hazardous materials T&P program identified in this TSD. The RL and ORP interact with each other and their contractors to accomplish implementation. The RL Manager, by agreement with ORP, serves as the T&P approval authority for the Hanford Site. RL acts as DOE contact for all requests for TSD changes or exemptions.

ORP provides RL with review and comments on the TSD and any ORP PSSD identified in the TSD as requiring RL Manager approval processing.

The RL Manager will review and approve all new fissile and Type B PSSDs.

ORP interfaces with their contractors to require implementation of and compliance with applicable requirements identified in the TSD and RL interfaces with their contractors to require implementation of and compliance with applicable requirements identified in the TSD.

RL recommended approval of all standards and requirements established in this TSD. The Hanford Sitewide TSD review and approval is documented in a safety evaluation report. The RL Manager approved the TSD. All subsequent revisions, with the exception of minor administrative and editorial corrections, will be processed for review and approval by RL. The RL Manager will approve all changes to the TSD.

RL and ORP will perform assessments of the respective contractor's program and T&P operations to ensure compliance with this TSD and applicable DOE orders (i.e., DOE Order 460.1A, DOE Order 460.2, *Departmental Materials Transportation & Packaging Management*, and DOE regulations in 10 CFR 830, "Nuclear Safety Management").

The RL Manager will approve all exceptions to technical standards or administrative requirements that have the potential to impact worker safety, public safety, or environmental compliance. Requests for exceptions must be submitted in writing to the RL Manager through RL after obtaining contractor approval authority concurrence. The TSD exemption process is further defined in Chapter IX, Nonroutine Shipments.

3.0 CONTRACTOR ROLES AND INTERFACES

Each contractor performing T&P activities must have a formal organization, program, and procedure in place to ensure compliance with the TSD standards that are applicable to the onsite shipment or transfer of hazardous materials, substances, and wastes as defined by applicable regulations.

Each contractor must identify an organization to manage the program and an approval authority to approve packaging systems authorized under the TSD.

Each contractor must complete a compliance or implementation matrix that identifies organizational responsibilities and key personnel.

Each contractor must list manuals, policies, procedures, processes, and systems to implement the TSD requirements and standards, and demonstrate compliance.

Each contractor must develop, test or evaluate, and document all new or modified packaging systems in compliance with the provisions of this TSD and applicable DOE orders.

Each contractor must implement and maintain a formal system for documenting the packaging and payload evaluation process to ensure that all packagings are approved for the payload and packaging operating and maintenance requirements are met with each shipment, and to ensure package configuration control is maintained within the packaging authorization.

Each contractor is to implement an unreviewed safety question process for transportation (USQ_T) as defined in Chapter VI, Section 6.1 and Appendix C of this TSD. All positive USQ_T determinations (USQ_TD) must be submitted to RL for approval prior to implementation.

Each contractor must have a system for performing required updates and maintenance activities on specific PSSDs (e.g., onsite SARPs and SEPs). The T&P system safety basis for each design must be assessed fully during document updating and maintenance.

Each contractor may use another contractor's PSSD after documented review and approval of the PSSD by their T&P approval authority.

Each contractor generating a new fissile or Type B PSSD must obtain RL approval prior to implementation.

Each contractor may use another contractor's transportation system after documented review and approval by their transportation approval authority.

Each contractor must establish an annual management assessment of their T&P safety program/operation to ensure compliance with the TSD, applicable regulations, DOE orders, and to identify opportunities to improve program performance.

4.0 REFERENCES

10 CFR 830, "Nuclear Safety Management," *Code of Federal Regulations*, as amended.

DOE Order 460.1A, *Packaging and Transportation Safety*, U.S. Department of Energy, Washington D.C.

DOE Order 460.2, *Departmental Materials Transportation & Packaging Management*, Change 1, U.S. Department of Energy, Washington, D.C.

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CHAPTER V. EXTERNAL REGULATIONS

This chapter identifies the external regulations that contain requirements applicable to onsite T&P activities. Several regulations are written specifically to address onsite packaging and transfer operations. Additional regulations apply to various activities that make up the overall process of T&P because they apply to the materials being handled or the activity being performed.

1.0 PACKAGING AND TRANSPORTATION, SPECIFIC REQUIREMENTS

The following external requirements specifically identify conditions for the onsite T&P of hazardous materials. Section 1.1 describes all hazardous materials transport requirements identified by the DOE. Section 1.2 describes radioactive materials requirements.

1.1 DOE ORDER 460.1A, PACKAGING AND TRANSPORTATION SAFETY

DOE Order 460.1A is the primary requirement for establishing and maintaining an onsite T&P program for the safe movement of hazardous materials, hazardous substances and wastes, including radioactive materials and wastes. This order requires that onsite hazardous material shipments comply with the DOT Hazardous Materials Regulations, or the site- or facility-specific cognizant operations or RL-approved TSD. The site- or facility-specific document must describe the methodology and compliance process used to provide equivalent safety for any deviation from the DOT Hazardous Materials Regulations. This document fulfills that requirement.

1.2 10 CFR 830, NUCLEAR SAFETY MANAGEMENT

For radioactive materials, 10 CFR 830 governs the conduct of DOE contractors, DOE personnel, and other persons conducting activities (including providing items and services) that affect, or may affect, the safety of DOE nuclear facilities. This includes transportation activities not regulated by the DOT. The rule permits treating transportation activities separate from the nuclear facilities the activities are conducted to support. In these cases, a separate documented safety analyses (such as SARs), as well as plans and programs, must be prepared. The rule also permits integrating the nuclear requirements and transportation requirements by integrating those activities into site or facility analyses and plans. The rule specifically identifies two "safe harbor" methods in Table 2 of Appendix A to Subpart B for transportation activities covered by this rule. The safe harbor methods endorse the methods and processes described in DOE Order 460.1A and its associated guide, and DOE Order 461.1, *Packaging and Transportation of Materials of National Security Interest*, and its associated manual. Both methods are identified as acceptable ways to satisfy the requirements for transportation activities covered by the provisions of this rule. This document takes advantage of the safe harbor provided by DOE Order 460.1A and its associated guide.

2.0 INDIRECT T&P REQUIREMENTS

The following are external safety requirements that may apply to onsite hazardous material T&P activities. Their applicability depends on the following conditions:

- Where the activity is performed
- What type of material is packaged
- How much material is handled
- Is the material considered waste or non-waste
- Physical form/chemical form
- Package used
- Method of transport.

2.1 NATIONAL REGULATIONS

This section identifies requirements that are known to influence onsite T&P. The organization performing the T&P activity must ensure that applicable requirements are identified and implemented.

10 CFR 71, "Packaging and Transportation of Radioactive Material," *Code of Federal Regulations*, as amended.

10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," *Code of Federal Regulations*, as amended.

10 CFR 835, "Occupational Radiation Protection," *Code of Federal Regulations*, as amended.

40 CFR 261, "Identification and Listing of Hazardous Waste," *Code of Federal Regulations*, as amended.

42 USC 9601 et seq., *Comprehensive Environmental Response, Compensation and Liability Act of 1980*.

49 CFR 40, "Procedures for Transportation Workplace Drug Testing Programs, U.S. Department of Transportation, Washington, D.C.

49 CFR 100 through 199, "Research and Special Programs Administration, Department of Transportation," U.S. Department of Transportation, Washington, D.C.

49 CFR 325 through 399, "Federal Highway Administration, Department of Transportation," U.S. Department of Transportation, Federal Highway Administration, Washington, D.C.

DOE Order 151.1A, *Comprehensive Emergency Management System*, U.S. Department of Energy, Washington, D.C.

DOE Order 232.1A, *Occurrence Reporting and Processing of Operations Information*, U.S. Department of Energy, Washington, D.C.

DOE Order 414.1A, *Quality Assurance*, U.S. Department of Energy, Washington, D.C.

DOE Order 460.1A, *Packaging and Transportation Safety*, U.S. Department of Energy, Washington, D.C.

DOE Order 460.2, *Departmental Materials Transportation & Packaging Management*, Change 1, U.S. Department of Energy, Washington, D.C.

DOE Order 461.1, *Packaging and Transfer or Transportation of Materials of National Security Interest*, U.S. Department of Energy, Washington, D.C.

DOE Order 474.1A, *Control and Accountability of Nuclear Materials*, U.S. Department of Energy, Washington, D.C.

DOT, *North American Emergency Response Guidebook*, U.S. Department of Transportation, Research and Special Programs Administration, Office of Hazardous Materials Initiatives and Training.

2.2 STATE REGULATIONS

WAC 173-303, "Dangerous Waste Regulations," *Washington Administrative Code*, as amended.

WAC 173-303-180, "Manifest," *Washington Administrative Code*, as amended.

WAC 173-303-220(2), "Generator Reporting," *Washington Administrative Code*, as amended.

WAC 173-303-370, "Manifest System," *Washington Administrative Code*, as amended.

WAC 446-50-080(1), "Transportation of Hazardous Materials," *Washington Administrative Code*, as amended.

2.3 LOCAL REGULATIONS

DOE-0223, *Emergency Implementation Procedures*, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

DOE/RL-91-28, *Hanford Facility Dangerous Waste Permit Application*, General Information Portion, Rev. 4, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

DOE/RL-94-02, *Hanford Emergency Management Plan*, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

RL 95-SWT-186 (Guidance letter, March 29, 1995), Director of Procurement Services Division, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

RLID 473.1, *Protection of Safeguards and Security Interests*, Implementing Directive, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

3.0 POTENTIAL INDIRECT T&P REQUIREMENTS

This section identifies regulatory or activity driven requirements that are not explicitly directed at transportation activities, but which result in imposition of requirements on transportation related activities. Their applicability depends on the following conditions:

- Where the activity is performed
- What type of material is packaged
- How much material is handled
- Is material considered waste or non-waste
- Physical form/chemical form
- Package used
- Method of transport.

The organization conducting the T&P activity should ensure that applicable requirements are identified and implemented.

Table 5-1 provides the number and title of DOE orders/notices. Table 5-2 provides the document number and title for RL tailored contractor requirements documents.

Table 5-1. DOE Orders and Notices.

Order/Notice Number	Changes	Title
CRD 151.1	1&2	Comprehensive Emergency Management System
CRD 200.1		Information Management Program
CRD 225.1A	1	Accident Investigations
CRD 231.1	1&2	Environment, Safety, and Health Reporting
CRD 232.1A		Occurrence Reporting and Processing of Operations Information
DOE M 232.1-1A		Occurrence Reporting and Processing of Operations Information
CRD 241.1		Scientific and Technical Information Management
CRD 251.1A		Directives System Order
CRD 252.1		Technical Standards Program
CRD 413.1		Management Control Program
CRD 414.1A		Quality Assurance
CRD 420.1	1&2	Facility Safety
CRD 435.1		Radioactive Waste Management

Table 5-1. DOE Orders and Notices.

Order/Notice Number	Changes	Title
CRD 440.1A		Worker Protection Management for DOE Federal and Contractor Employees
CRD 440.2	1&2	Aviation
CRD 460.2	1	Departmental Materials Transportation & Packaging Management
CRD 461.1		Packaging and Transfer or Transportation of Materials of National Security Interest
CRD 470.2A		Security and Emergency Management Independent Oversight and Performance Assurance Program
DOE M 471.1-1		Identification and Protection of Unclassified Controlled Nuclear Information Manual
CRD 471.1A		Identification and Protection of Unclassified Controlled Nuclear Information
CRD 471.2A		Information Security Program
DOE M 471.2-1B		Classified Matter Protection and Control Manual
DOE M 471.2-2		Classified Information Systems Security Manual
CRD 472.1B		Personnel Security Activities
CRD 473.4		DOE Badges
DOE N 473.5		Security Area Vouching and Piggybacking
DOE N 473.6		Security Conditions
CRD 473.7		Explosive Detection Program
DOE M 474.1-1		Manual for Control and Accountability of Nuclear Materials
CRD 474.1		Control and Accountability of Nuclear Materials
DOE M 474.1- 2	2	Nuclear Materials Management and Safeguards System Reporting and Data Submission
DOE M 475.1-1		Identifying Classified Information
DOE 1270.2B		Safeguards Agreement with the International Atomic Energy Agency
DOE 4330.4B		Maintenance Management Program
DOE 5400.1	1	General Environmental Protection Program
DOE 5400.5	1&2	Radiation Protection of the Public and the Environment
DOE 5480.19	1	Conduct of Operations Requirements for DOE Facilities
DOE 5480.20A		Personnel Selection, Qualifications, and Training Requirements for DOE Nuclear Facilities
DOE 5480.21		Unreviewed Safety Questions
DOE 5480.22	1&2	Technical Safety Requirements
DOE 5480.23	1	Nuclear Safety Analysis Reports
DOE 5480.30		Nuclear Reactor Safety Design Criteria
DOE 5480.4	1-4	Environmental Protection, Safety, and Health Protection Standards
DOE 5530.1A		Accident Response Group
DOE 5530	2	Nuclear Emergency Search Team
DOE 5530.3	1	Radiological Assistance Program
DOE 5610.13		Joint Department of Energy/Department of Defense Nuclear Weapon Safety, Security, and Control Program
DOE 5610.2	1	Control of Weapon Data
DOE 5632.1C		Protection and Control of Safeguards and Security Interests
DOE M 5632.1C-1	1	Manual for Protection and Control of Safeguards and Security Interests (Except Chapter III, paragraphs 1, 2, and 4 through 9)
DOE 5660.1B		Management of Nuclear Materials
DOE 6430.1A		General Design Criteria
DOE/RW-0333P	Rev. 8	Quality Assurance Requirements and Descriptions
SEN-22-90		DOE Policy on Signatures of RCRA Permit Applications

Table 5-2. RL Tailored Contractor Requirements Documents.

Document Number	Changes	Title
RLM 440.2		Aviation Manual
CRD 470.1 (supplemented)		Safeguards and Security Program Requirements
RLID 473.1		Protection of Safeguards and Security Interests
RLID 473.2		Hanford Site Access Eligibility
RLID 1210.1		Hanford Visitor Policies and Procedures
RLID 1300.1D		Facility Representative Program
RLID 5480.19		Conduct of Operations Requirements for RL
RLID 5630.3A		Protection of Hanford Facilities against Radiological and Toxicological Sabotage
RLID 5632.1B		Asset Protection Requirements
RLID 5633.3		Control and Accountability of Nuclear Materials RL Awareness Program Plan
DOE/RL-92-49		Radiological Assistance Program Plan - Region 8
DOE/RL-94-02		Hanford Emergency Management Plan
DOE/RL-92-36		Hanford Site Hoisting and Rigging Manual
DOE/RL-SOD-INST L&T.001	2	Hanford Site Lockout/Tagout Program
DOE/RL-94-97		Selection of Analytical Methods for Mixed Waste Analysis at Hanford
DOE/RL-96-68	2	Hanford Analytical Services Quality Assurance Requirements Document
DOE-0223		RL Emergency Implementing Procedures
DOE-0225		Hanford Emergency Assessment Resource Manual

4.0 STANDARDS

4.1 GOVERNMENT

DOE Guide 421.1-1, 1999, *Criticality Safety Good Practices Program Guide for DOE Nonreactor Nuclear Facilities*, U. S. Department of Energy, Washington, D.C.

DOE-HDBK-3010-94, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities*, Change Notice 1, U. S. Department of Energy, Washington, D.C.

DOE/RL-92-36, *RL Hoisting and Rigging Manual*, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

DOE-STD-3009-94, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports*, Change Notice No. 1, U. S. Department of Energy, Washington, D. C.

DOH, *Radiological Health Handbook*, U.S. Department of Health, Education, and Welfare, Rockville, Maryland.

EPA, 1992, *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents*, U.S. Environmental Protection Agency, Washington, D.C.

IAEA, 1990, *Explanatory Material for the IAEA Regulations for the Safe Transport of Radioactive Material*, Safety Series No. 7, Second Edition (As Amended 1990), International Atomic Energy Agency, Vienna.

ICRP, 1975, *Report of the Task Group on Reference Man*, International Commission on Radiological Protection, Report No. 23, Elmsford, New York.

MIL-HDBK-5G, *Military Handbook-Metallic Materials and Elements for Aerospace Vehicle Structures*, U.S. Department of Defense, Washington, D.C.

NUREG-1609, *Standard Review Plan for Transportation Packages for Radioactive Material*, Chapter 7, Spent Fuel Project Office, U.S. Nuclear Regulatory Commission, Washington, D.C.

NUREG/CR-1815, *Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers up to Four Inches Thick*, U.S. Nuclear Regulatory Commission, Washington, D.C.

NUREG/CR-3019, *Recommended Welding Criteria for Use in the Fabrication of Shipping Containers for Radioactive Materials*, U.S. Nuclear Regulatory Commission, Washington, D.C.

NUREG/CR-3332 and ORNL-5968, 1983, *Radiological Assessment: A Textbook on Environmental Dose Analysis*, prepared by Radiological Assessments Corporation, Neeses, South Carolina, and Oak Ridge National Laboratory, Oak Ridge, Tennessee, for the U.S. Nuclear Regulatory Commission, Washington, D.C.

NUREG/CR-3854, *Fabrication Criteria for Shipping Containers*, U.S. Nuclear Regulatory Commission, Washington, D.C.

NUREG/CR-4775, *Guide for Preparing Operating Procedures for Shipping Packages*, U.S. Nuclear Regulatory Commission, Washington, D.C.

NUREG/CR-4829, *Shipping Container Response to Severe Highway and Railway Accident Conditions*, Lawrence Livermore National Laboratory, Livermore, California.

NUREG/CR-5502, *Engineering Drawings for 10 CFR Part 71 Package Approvals*, Nuclear Regulations, U.S. Nuclear Regulatory Commission, Washington, D.C.

NUREG/CR-5661, *Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packaging for Radioactive Material*, U.S. Nuclear Regulatory Commission, Washington, D.C.

NUREG/CR-6007, *Stress Analysis of Closure Bolts for Shipping Casks*, U.S. Nuclear Regulatory Commission, Washington, D.C.

NUREG/CR-6407, *Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety*, U.S. Nuclear Regulatory Commission, Washington, D.C.

NUREG/CR-6672, *Reexamination of Spent Fuel Shipment Risk Estimates*, Sandia National Laboratories, Albuquerque, New Mexico.

Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*, U.S. Nuclear Regulatory Commission, Washington, D.C.

Regulatory Guide 7.6, *Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels*, U.S. Nuclear Regulatory Commission, Washington, D.C.

Regulatory Guide 7.10, *Establishing Quality Assurance Programs for Packaging Used in the Transport of Radioactive Material*, U.S. Nuclear Regulatory Commission, Washington, D.C.

Regulatory Guide 7.11, *Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 Inches (0.1 m)*, U.S. Nuclear Regulatory Commission, Washington, D.C.

Regulatory Guide 7.12, *Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Wall Thickness Greater than 4 Inches (0.1 m) But Not Exceeding 12 Inches (0.3 m)*, U.S. Nuclear Regulatory Commission, Washington, D.C.

4.2 INDUSTRY

ANSI/ANS-8.1-1998, *American National Standard for Nuclear Criticality Safety in Operations with Fissionable Material Outside Reactors*, American Nuclear Society, La Grange Park, Illinois.

ANSI N14.5, "Radioactive Materials - Leakage Test on Packages for Shipment," *American National Standards Institute*, La Grange Park, Illinois.

ASME, 1995, "Boiler and Pressure Vessel Code," Section VIII, Division 1, *American Society of Mechanical Engineers*, New York, New York.

ASME, 1995, "Boiler and Pressure Vessel Code," Section III, Subsection NE, *American Society of Mechanical Engineers*, New York, New York.

ASME, 1995, "Boiler and Pressure Vessel Code," Section II, Part D, *American Society of Mechanical Engineers*, New York, New York.

ISO 14001, *Integrated Organization for Standardization, Environmental Management System*, International Standards Organization, Geneva, Switzerland.

MIL-HDBK-695, *Rubber Products: Recommended Shelf Life*, Military Handbook, Aerospace Industries Association, Washington, D.C.

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CHAPTER VI. SITE-SPECIFIC STANDARDS, PROCEDURES, AND INSTRUCTIONS

This chapter provides standards, requirements, programs, and processes that are needed to support safe T&P operations such as integrated safety management, QA, configuration management, and equivalent safety. Alternate design and performance requirements for onsite packagings are identified.

1.0 OPERATING REQUIREMENTS

Specific standards and procedures are to be followed when conducting onsite shipments at the Hanford Site. The initial discussion is an overview of how the TSD achieves equivalent safety. In addition, the initial discussion provides an overview of requirements to be implemented. This is followed by identifying specific standards, procedures, and operating instructions that are to be implemented by operating procedures when conducting onsite T&P activities.

1.1 BASES FOR EQUIVALENT SAFETY

T&P activities are conducted onsite using three techniques to achieve safety equivalent to that resulting from following the DOT regulations when shipping in commerce. The following sections identify the three techniques.

1.1.1 DOT Compliance

For all nonradioactive and most radioactive hazardous materials, equivalent safety is achieved by implementing the DOT regulations onsite. All portions of the regulations, other than those identifying the workings and administration of the DOT (e.g., enforcement) will be implemented. Equivalent administrative activities including approval of exemptions will be conducted by RL and ORP. This is the preferred technique for achieving onsite equivalent safety when conducting onsite T&P activities.

1.1.2 DOT-Equivalent Packaging

For radioactive materials, where full compliance with DOT regulations cannot be achieved, an equivalent method for achieving the accepted national level of safety will be implemented. Before implementing the DOT-equivalent packaging method, a documented evaluation showing that DOT compliance is not technically or economically practical must be prepared and retained on file for the life of the package. For these radioactive hazardous materials, equivalent safety is

achieved by implementing all portions of the regulations, except for regulations identifying the following:

- Workings and administration of the DOT
- How to obtain exemptions
- DOT design and performance requirements for IP-2, IP-3, Type A, fissile and Type B packaging
- Communication requirements associated with shipping radioactive materials.

Equivalent administrative activities including approval of exemptions will be conducted by RL and ORP. Performance requirements for onsite DOT-equivalent packages are identified in Section 5.2 of this chapter. These requirements are developed to result in a package that produces physical performance under Hanford Site conditions equivalent to an offsite package subjected to regulatory-established test conditions. In addition, alternate communications methods are authorized as defined in this TSD.

When full compliance with DOT cannot be achieved because of technical or economic conditions, meeting these site-specific standards and performance requirements is the preferred technique for achieving safety equivalent to that resulting from following DOT regulations when shipping in commerce.

1.1.3 NonEquivalent Packaging

When full compliance with DOT regulations or compliance with DOT-equivalent packaging for radioactive materials cannot be achieved, an equivalent method for demonstrating the accepted level of safety will be implemented. Before implementing the nonequivalent packaging method, a document evaluation showing that DOT compliance and DOT-equivalent packaging compliance are not technically or economically practical must be prepared and retained on file for the life of the package. For these radioactive hazardous materials, equivalent safety is achieved by implementing all portions of the regulations, except for regulations identifying the following:

- Workings and administration of the DOT
- How to obtain exemptions
- DOT design and performance requirements for IP-2, IP-3, Type A, fissile and Type B packaging
- Communication requirements associated with shipping radioactive materials.

Equivalent administrative activities including the approval of exemptions will be conducted by RL for RL and ORP. Risk-based performance requirements for nonequivalent packages are identified in Section 5.3 of this chapter. The requirements are developed to result in a package, which under site conditions will produce radiological and toxicological performance equivalent to an offsite package subjected to regulatory-established tests and conditions.

When full compliance with DOT or DOT-equivalent packaging cannot be achieved because of technical or economic conditions, this technique may be used for achieving equivalent safety. The technique must demonstrate safety equivalent to that resulting from following DOT regulations when shipping in commerce using an approved RL risk-based approach.

2.0 QA PROGRAM

RL, ORP, and contractors must implement a quality assurance program (QAP). The QAP for contractors must meet contract requirements. Where no contract requirement exists, use 10 CFR 830, Subpart A. In addition, for design and fabrication of onsite fissile and Type B packagings, the onsite packaging requirements require implementing Nuclear Quality Assurance (NQA-1).

3.0 INTEGRATED SAFETY MANAGEMENT SYSTEM

RL, ORP, and each Hanford Site contractor must implement a comprehensive integrated environment, safety and health management system (ISMS) plan and assess onsite T&P activities against ISMS requirements.

The ISMS plan must establish a safety and environmental management system that integrates environment, safety and health (ES&H) requirements into the work planning and execution processes to effectively protect workers, the public, and environment.

The ISMS must incorporate best practices of the following policies, standards, and initiatives:

- RL ES&H Policy
- DOE's *Safety Management System Policy* (DOE Policy 450.4)
- DOE's *Line Environment, Safety and Health Oversight* (DOE Policy 450.5)
- DOE's *Secretarial Policy Statement Environment, Safety and Health* (DOE Policy 450.6)
- International Standards Organization (ISO) 14001.

The contractor ISMS plan must be approved by the appropriate RL or ORP Division.

4.0 CONFIGURATION MANAGEMENT PROCESS

4.1 PACKAGE SYSTEM AUTHORIZATIONS

Review and approval of this TSD by the RL Manager results in a safety basis document for onsite transportation activities that establish safety bounds for onsite hazardous materials T&P activities. This TSD includes the PSSD identified in Appendix A. The PSSDs for onsite packages are evaluations that demonstrate how the criteria of the TSD are met. RL, ORP, and contractors may make physical and procedural changes and conduct tests and experiments without prior RL approval, provided changes do not explicitly or implicitly affect the authorization basis identified in this TSD.

4.1.1 Approval of Activities and Packagings Bounded by TSD Approval

Changes to this TSD not bounded by the supporting safety evaluation and existing supporting package safety analysis documentation must be approved by the RL Manager. Changes bounded by the supporting safety evaluation and existing supporting package safety analysis documentation are approved by the contractor responsible for the T&P activity. The USQ_T identified in Section 6.1 of this chapter must be used to evaluate which changes must be approved by the RL Manager. All new fissile and Type B PSSDs must be approved by the RL Manager.

5.0 ONSITE T&P EQUIVALENT SAFETY REQUIREMENTS

5.1 DOT COMPLIANCE

For all nonradioactive (unless otherwise allowed in Chapter VIII) and most radioactive hazardous materials, equivalent safety is achieved onsite by implementing DOT regulations. All portions of the regulations, other than those identifying the workings of the DOT, will be implemented. Equivalent administrative activities including approval of exemptions will be conducted by RL and ORP. Table 6-1 identifies methods of implementation to the subpart level of the regulations found in 49 CFR 100 through 185. Table 6-2 identifies methods of implementation to the subpart level of the regulations found in 49 CFR 350 through 399 – “Federal Motor Carrier Safety Administration.” The tables differentiate which regulations are DOT administrative or other activities not applicable to onsite shipments, which regulations are written such that onsite implementation (compliance) without modification can be achieved, and which regulations are implemented onsite by emulation of the requirements. Emulation is required where regulations require an action on the part of DOT that is not applicable to onsite T&P activities. For example, DOT approval of an exemption to regulatory requirements for onsite shipments is not applicable. Emulation is also used to modify requirements to meet site conditions.

Table 6-1. 49 CFR Regulations – Transportation.

49 CFR Parts 100 to 185 – Transportation	DOT Administrative	Compliance	Emulate
CHAPTER I, SUBCHAPTER A	---	---	---
Part 100 [Reserved]	---	---	---
Part 101 [Reserved]	---	---	---
Part 106, Rulemaking procedures	---	---	---
106 Subpart A, General	X		
106 Subpart B, Procedures for Adoption of Rules	X		
Part 107, Hazardous materials program procedures	---	---	---
107 Subpart A, General Provisions	X		
107 Subpart B, Exemptions			X
107 Subpart C, Preemption	X		
107 Subpart D, Enforcement			X
107 Subpart E, Designation of Approval and Certification Agencies			X
107 Subpart F, Registration of Cargo Tank and Cargo Tank Motor Vehicle Manufacturers and Repairers and Cargo Tank Motor Vehicle Assemblers			X
107 Subpart G, Registration of persons Who Offer or Transport Hazardous Materials	X		
107 Subpart H, Approvals, Registrations and Submissions			X
Part 110, Hazardous materials public sector training and planning grants	X		
CHAPTER I, SUBCHAPTER B	---	---	---
Part 130, Oil spill prevention and response plans			X
CHAPTER I, SUBCHAPTER C	---	---	---
Part 171, General information, regulations, and definitions		X	
Part 172, Hazardous materials tables and hazardous materials communications regulations	---	---	---
172 Subpart A, General	X		
172 Subpart B, Table of Hazardous Materials and Special Provisions		X	
172 Subpart C, Shipping Papers			X
172 Subpart D, Marking			X
172 Subpart E, Labeling			X
172 Subpart F, Placarding			X
172 Subpart G, Emergency Response Information			X
172 Subpart H, Training		X	
172 App A, Office of Hazardous Materials Transportation Color Tolerance Charts and Tables		X	
172 App B, Trefoil Symbol		X	
172 App C, Dimensional Specifications for Recommended Placard Holder		X	
Part 173, Shippers – General requirements for shipments and packagings	---	---	---
173 Subpart A, General	X		
173 Subpart B, Preparation of Hazardous Materials for Transportation			X

Table 6-1. 49 CFR Regulations – Transportation.

49 CFR Parts 100 to 185 – Transportation	DOT Administrative	Compliance	Emulate
173 Subpart C, Definitions, Classification and Packaging for Class 1		X	
173 Subpart D, Definitions, Classification, Packing Group Assignments and Exceptions for Hazardous Materials Other Than Class 1 and Class 7		X	
173 Subpart E, Non-bulk Packaging for Hazardous Materials Other Than Class 1 and Class 7		X	
173 Subpart F, Bulk Packaging for Hazardous Materials Other Than Class 1 and Class 7		X	
173 Subpart G, Gases; Preparation and Packaging			X
173 Subpart H [Reserved]	---	---	---
173 Subpart I, Class 7 (Radioactive) Materials			X
173 Subpart J-O [Reserved]	---	---	---
173 App A, Appendix A - [Reserved]	---	---	---
173 App B, Appendix B – Procedure for Testing Chemical Compatibility and Rate of Permeation in Plastic Packaging and Receptacles	X		
173 App C, Appendix C – Procedure for Base-level Vibration Testing	X		
173 App D, Appendix D – Test Methods for Dynamite (Explosive, Blasting, Type A)	X		
173 App E, Appendix E – [Reserved]	---	---	---
173 App F, Appendix F – [Reserved]	---	---	---
173 App G, Appendix G – [Reserved]	---	---	---
173 App H, Appendix H to Part 173 – Method of Testing for Sustained Combustibility	X		
Part 174, Carriage by rail	N/A	N/A	N/A
174 Subpart A, General Requirements	X		
174 Subpart B, General Operating Requirements	X		
174 Subpart C, General Handling and Loading Requirements	X		
174 Subpart D, Handling of Placarded Rail Cars, Transport Vehicles and Freight Containers	X		
174 Subpart E, Class 1 (Explosive) Materials	X		
174 Subpart F, Detailed Requirements for Class 2 (Gases) Materials	X		
174 Subpart G, Detailed Requirements for Class 3 (Flammable Liquid) Materials	X		
174 Subparts H-I [Reserved]	---	---	---
174 Subpart J, Detailed Requirements for Division 6.1 (Poisonous) Materials	X		
174 Subpart K, Detailed Requirements for Class 7 (Radioactive) Materials	X		
Part 175, Carriage by aircraft	N/A	N/A	N/A
175 Subpart A, General Information and Regulations	X		
175 Subpart B, Loading, Unloading and Handling	X		
175 Subpart C, Specific Regulations Applicable According to Classification of Material	X		

Table 6-1. 49 CFR Regulations – Transportation.

49 CFR Parts 100 to 185 – Transportation	DOT Administrative	Compliance	Emulate
Part 176, Carriage by vessel	N/A	N/A	N/A
176 Subpart A, General	X		
176 Subpart B, General Operating Requirements	X		
176 Subpart C, General Handling and Stowage	X		
176 Subpart D, General Segregation Requirements	X		
176 Subpart E, Special Requirements for Transport Vehicles Loaded With Hazardous Materials and Transported on Board Ferry Vessels	X		
176 Subpart F, Special Requirements for Barges	X		
176 Subpart G, Detailed Requirements for Class 1 (Explosive) Materials	X		
176 Subpart H, Detailed Requirements for Class 2 (Compressed Gas) Materials	X		
176 Subpart I, Detailed Requirements for Class 3 (Flammable) and Combustible Liquid Materials	X		
176 Subpart J, Detailed Requirements for Class 4 (Flammable Solids), Class 5 (Oxidizers and Organic Peroxides), and Division 1.5 (Blasting Agents) Materials	X		
176 Subpart K [Reserved]	—	—	—
176 Subpart L, Detailed Requirements for Division 2.3 (Poisonous Gas) and Division 6.1 (Poisonous) Materials	X		
176 Subpart M, Detailed Requirements for Radioactive Materials	X		
176 Subpart N, Detailed Requirements for Class 8 (Corrosive Materials) Materials	X		
176 Subpart O, Detailed Requirements for Cotton and Vegetable Fibers, Motor Vehicles, and Asbestos	X		
Part 177, Carriage by public highway	—	—	—
177 Subpart A, General Information and Regulations			X
177 Subpart B, Loading and Unloading			X
177 Subpart C, Segregation and Separation Chart of Hazardous Materials			X
177 Subpart D, Vehicles and Shipments in Transit; Accidents			X
177 Subpart E, Regulations Applying to Hazardous Material on Motor Vehicles Carrying Passengers for Hire			X
Part 178, Specifications for packagings			X
178 Subpart A [Reserved]	—	—	—
178 Subpart B, Specifications for Inside Containers, and Linings		X	
178 Subpart C, Specifications for Cylinders		X	
178 Subparts D-G [Reserved]	—	—	—
178 Subpart H, Specifications for Portable Tanks		X	
178 Subpart I [Reserved]	—	—	—
178 Subpart J, Specifications for Containers for Motor Vehicle Transportation		X	
178 Subpart K, Specifications for packagings for Class 7 (Radioactive) Materials		X	
178 Subpart L, Non-bulk Performance-Oriented Packaging Standards		X	

Table 6-1. 49 CFR Regulations – Transportation.

49 CFR Parts 100 to 185 – Transportation	DOT Administrative	Compliance	Emulate
178 Subpart M, Testing of Non-bulk Packagings and Packages		X	
178 Subpart N, Intermediate Bulk Container Performance-Oriented Standards		X	
178 Subpart O, Testing of Intermediate Bulk Containers		X	
178 App A, Appendix A – Specifications for Steel		X	
178 App B, Appendix B – Alternative Leakproofness Test Methods		X	
178 App C, Appendix C – Nominal and Minimum Thickness of Steel Drums and Jerricans		X	
Part 179, Specifications for tank cars	---	---	---
179 Subpart A, Introduction, Approvals and Reports	X		
179 Subpart B, General Design Requirements			X
179 Subpart C, Specifications for Pressure Tank Car Tanks			X
179 Subpart D, Specifications for Non-Pressure Tank Car Tanks (Classes DOT-103, 104, 111AF, 111AW, and 115AW)			X
179 Subpart E, Specifications for Multi-Unit Tank Car Tanks (Classes DOT-106A and 110AW)			X
179 Subpart F, Specification for Cryogenic Liquid Tank Car Tanks and Seamless Steel Tanks (Classes DOT-113 and 107A)			X
179 App A, Appendix A to Part 179 – Procedures for Tank-Head Puncture-Resistance Test	X		
179 App B, Appendix B to Part 179 – Procedures for Simulated Pool and Torch-Fire Testing	X		
Part 180, Continuing qualification and maintenance of packagings	---	---	---
180 Subpart A, General	X		
180 Subparts B-C [Reserved]	---	---	---
180 Subparts D, Qualification and Maintenance of Intermediate Bulk Containers			X
180 Subpart E, Qualification and Maintenance of Cargo Tanks			X
180 Subpart F, Qualification and Maintenance of Tank Cars			X
180 App A, Internal Self-closing Stop Valve Emergency Closure Test for Liquefied Compressed Gasses			X
180 App B, Acceptable Internal Self-closing Stop Valve Leakage Tests for Cargo Tanks Transporting Liquefied Compressed Gasses			X
Part 185 [Reserved]	---	---	---

Table 6-2. 49 CFR Regulations - Federal Motor Carrier Safety Regulations.

49 CFR Parts 350 to 399 – Federal Motor Carrier Safety Regulations	DOT Administrative	Compliance	Emulate
CHAPTER III, SUBCHAPTER B	---	---	---
Part 350, Commercial Motor Carrier Safety Assistance Program	---	---	---
350 Subpart A, General	X		
350 Subpart B, Requirements for Participation	X		
350 Subpart C, Funding	X		
Part 355, Compatibility of State Laws and Regulations Affecting Interstate Motor Carrier Operations	---	---	---
355 Subpart A, General Applicability and Definitions	X		
355 Subpart B, Requirements	X		
Part 356, Motor Carrier Routing Regulations	X		
Part 360, Fees for Motor Carrier Registration and Insurance	X		
Part 365, Rules Governing Applications for Operating Authority	X		
365 Subpart A, How to Apply for Operating Authority	X		
365 Subpart B, How to Oppose Requests for Authority	X		
365 Subpart C, General Rules Governing the Application Process	X		
365 Subpart D, Transfer of Operating Rights Under 49 USC 10926	X		
Part 366, Designation of Process Agent	X		
Part 367, Standards for Registration with States	X		
Part 368, Applications for Certificates of Registration by Foreign Motor Carriers and Foreign Motor Private Carriers under 49 USC 10530	X		
Part 370, Principles and Practices for the Investigation and Voluntary Disposition of Loss and Damage Claims and Processing Salvage	X		
Part 371, Brokers of Property	X		
Part 372, Exemptions, Commercial Zones, and Terminal Areas	---	---	---
372 Subpart A, Exemptions	X		
372 Subpart B, Commercial Zones	X		
372 Subpart C, Terminal Areas	X		
Part 373, Receipts and Bills	---	---	---
373 Subpart A, Motor Carrier Receipts and Bills	X		
373 Subpart B, Freight Forwarders, Bills of Lading	X		
Part 374, Passenger Carrier Regulations	---	---	---
374 Subpart A, Discrimination in Operation of Interstate Motor Common Carriers of Passengers	X		
374 Subpart B, Limitation of Smoking on Interstate Passenger Carrier Vehicles	X		
374 Subpart C, Adequacy of Intercity Motor Common Carrier Passenger Service	X		
374 Subpart D, Notice of and Procedures for Baggage Excess Value Declaration	X		
374 Subpart E, Incidental Charter Rights	X		
Part 375, Transportation of Household Goods in Interstate or Foreign Commerce	X		
Part 376, Lease and Interchange of Vehicles	---	---	---
376 Subpart A, General Applicability and Definitions	X		

Table 6-2. 49 CFR Regulations - Federal Motor Carrier Safety Regulations.

49 CFR Parts 350 to 399 - Federal Motor Carrier Safety Regulations	DOT Administrative	Compliance	Emulate
376 Subpart B, Leasing Regulations	X		
376 Subpart C, Exemptions for the Leasing Regulations	X		
376 Subpart D, Interchange Regulations	X		
376 Subpart E, Private Carriers and Shippers	X		
Part 377, Handling of C.O.D. Shipments	---	---	---
377 Subpart A, Handling of C.O.D. Shipments	X		
377 Subpart B, Extension of Credit to Shippers by Motor Common Carriers, Water Common Carriers, and Household Goods Freight Forwards	X		
Part 378, Procedures Governing the Processing, Investigation, and Disposition of Overcharge, Duplicate Payment or Over Collection Claims	X		
Part 379, Preservation of Records	X		
Part 381, Waivers, Exemptions, and Pilot Programs	---	---	---
381 Subpart A, General	X		
381 Subpart B, Procedures for Requesting Waivers	X		
Part 382, Controlled Substances and Alcohol Use and Testing	---	---	---
382 Subpart A, General			X
382 Subpart B, Prohibitions			X
382 Subpart C, Tests Required			X
382 Subpart D, Handling of Test Results, Record Retention, and Confidentiality			X
382 Subpart E, Consequences for Drivers Engaging in Substance Use-Related Conduct			X
382 Subpart F, Alcohol Misuse and Controlled Substances Use Information, Training, and Referral			X
Part 383, Commercial Driver's License Standards; Requirements and Penalties	---	---	---
383 Subpart A, General		X	
383 Subpart B, Single License Requirement		X	
383 Subpart C, Notification Requirements and Employer Responsibilities		X	
383 Subpart D, Driver Disqualifications and Penalties		X	
383 Subpart E, Testing and Licensing Procedures		X	
383 Subpart F, Vehicle Groups and Endorsements		X	
383 Subpart G, Required Knowledge and Skills		X	
383 Subpart H, Tests		X	
383 Subpart I, [Reserved]	---	---	---
383 Subpart J, Commercial Driver's License Document		X	
Part 384, State Compliance With Commercial Driver's License Program	---	---	---
384 Subpart A, General	X		
384 Subpart B, Minimum Standards for Substantial Compliance by States	X		
384 Subpart C, Procedures for Determining State Compliance	X		
384 Subpart D, Consequences of State Noncompliance	X		
Part 385, Safety Fitness Procedures	X		

Table 6-2. 49 CFR Regulations - Federal Motor Carrier Safety Regulations.

49 CFR Parts 350 to 399 - Federal Motor Carrier Safety Regulations	DOT Administrative	Compliance	Emulate
Part 386, Rules of Practice for Motor Carrier Safety and Hazardous Materials Proceedings	---	---	---
386 Subpart A, Scope of Rules; Definitions	X		
386 Subpart B, Commencement of Proceedings, Pleadings	X		
386 Subpart C, Compliance and Consent Orders	X		
386 Subpart D, General Rules and Hearings	X		
386 Subpart E, Decision	X		
386 Subpart F, Injunctions and Imminent Hazards	X		
386 Subpart G, Penalties	X		
Part 387, Minimum Levels of Financial Responsibility for Motor Carriers	---	---	---
387 Subpart A, Motor Carriers of Property	X		
387 Subpart B, Motor Carriers of Passengers	X		
387 Subpart C, Surety Bonds and Policies of Insurance for Motor Carriers and Property Brokers	X		
387 Subpart D, Surety Bonds and Policies of Insurance for Freight Forwarders	X		
Part 388, Cooperative Agreements with States	X		
Part 389, Rulemaking Procedures - Federal Motor Carrier Safety Regulations	-	---	---
389 Subpart A, General	X		
389 Subpart B, Procedures for Adoption of Rules	X		
Part 390, Federal Motor Carrier Safety Regulations; General	---	---	---
390 Subpart A, General Applicability and Definitions			X
390 Subpart B, General Requirements and Information			X
390 [Removed and Reserved]	---	---	---
390 [Removed and Reserved]	---	---	---
Part 391, Qualifications of Drivers	---	---	---
391 Subpart A, General			X
391 Subpart B, Qualification and Disqualification of Drivers			X
391 Subpart C, Background and Character			X
391 Subpart D, Examinations and Tests			X
391 Subpart E, Physical Qualifications and Examinations			X
391 Subpart F, Files and Records			X
391 Subpart G, Limited Exemptions			X
Part 392, Driving of Commercial Motor Vehicles	---	---	---
392 Subpart A, General		X	
392 Subpart B, Driving of Commercial Motor Vehicles		X	
392 Subpart C, Stopped Commercial Motor Vehicles		X	
392 Subpart D, Use of Lighted Lamps and Reflectors		X	
392 Subpart E, License Revocation; Duties of Driver		X	
392 Subpart F, Fueling Precautions		X	

Table 6-2. 49 CFR Regulations - Federal Motor Carrier Safety Regulations.

49 CFR Parts 350 to 399 – Federal Motor Carrier Safety Regulations	DOT Administrative	Compliance	Emulate
392 Subpart G, Prohibited Practices		X	
Part 393, Parts and Accessories Necessary for Safe Operation	---	---	---
393 Subpart A, General		X	
393 Subpart B, Lighting Devices, Reflectors, and Electrical Equipment		X	
393 Subpart C, Brakes		X	
393 Subpart D, Glazing and Window Construction		X	
393 Subpart E, Fuel Systems		X	
393 Subpart F, Coupling Devices and Towing Methods		X	
393 Subpart G, Miscellaneous Parts and Accessories		X	
393 Subpart H, Emergency Equipment		X	
393 Subpart I, Protection Against Shifting or Falling Cargo		X	
393 Subpart J, Frames, Cab and Body Components, Wheels, Steering, and Suspension Systems		X	
Part 395, Hours of Service of Drivers			X
Part 396, Inspection, Repair, and Maintenance			X
Part 397, Transportation of Hazardous Materials; Driving and Parking Rules	---	---	---
397 Subpart A, General		X	
397 Subpart B, Not used by DOT	---	---	---
397 Subpart C, Routing of Non Radioactive Hazardous Materials			X
397 Subpart D, Routing of Class 7 (Radioactive) Materials			X
397 Subpart E, Preemption Procedures	X		
Part 398, Transportation of Migrant Workers	X		
Part 399, Employee Safety and Health Standards		X	

5.2 DOT-EQUIVALENT (Equivalent Packaging)

For some radioactive materials requiring IP-2, IP-3, Type A, fissile or Type B packagings, equivalent safety is achieved onsite by implementing a mixture of DOT regulations and onsite equivalent requirements. Sections 5.2.1 through 5.2.6.14 identify 49 CFR Transportation chapters, subchapters, parts, subparts that will be met by compliance with DOT regulations, and those where equivalent actions will be used.

5.2.1 Identification of DOT Chapters and Parts when using DOT-Equivalent Packaging

Table 6-1 identifies to the subpart level the regulations found in 49 CFR 100 through 185. Table 6-2 identifies to the subpart level the regulations found in 49 CFR 350 through 399. The tables differentiate which regulations are DOT administrative activities not applicable to onsite shipments, which regulations are written such that onsite implementation (compliance) without

modification can be achieved, and which regulations are implemented onsite by emulating the requirements.

When using alternate onsite packaging requirements, substitutions are made for regulations in Table 6-1 that identify the requirements for the following:

- DOT design and performance requirements for IP-2, IP-3, Type A, fissile and Type B packagings
- Communication requirements associated with shipment of radioactive materials.

Alternate requirements are used under the equivalent safety provision of DOE Order 460.1A.

Before implementing the DOT-equivalent packaging method, a documented evaluation showing that DOT compliance is not technically or economically practical must be prepared and retained on file for the life of the package.

5.2.2 Identification of Onsite Radioactive Materials Packaging Requirements

When using the alternate onsite packaging requirements, the substitutions identified in Sections 5.2.3, 5.2.4, 5.2.5, 5.2.6, and 5.3.2 are used in place of the packaging regulations identified in Table 6-1.

5.2.3 Performance Tests for Type A, IP-3, and IP-2 Packagings

Performance tests for DOT-equivalent Type A, IP-3, and IP-2 packagings are the same as those specified in the regulations, except in some conditions alternate Hanford Site-specific conditions are applied. The variations from the regulations are based on alterations developed for onsite Type B and fissile packaging. The description of and justification for deviations to regulatory requirements for Type B and fissile packagings are provided in Appendix B, Justification and Basis for Equivalency to DOT Regulations for Type B and Fissile Packages Transported on the Hanford Site. The following sections are performance tests for solids, liquids, and gases to demonstrate equivalency to the regulations.

5.2.3.1 Evaluations and Test Conditions.

5.2.3.1(a) Compliance with the test requirements in Sections 5.2.3.2 through 5.2.3.9 and the design requirements in Section 5.2.4 must be shown by any of the methods described in this section, or by a combination of these methods appropriate for the particular feature being evaluated:

5.2.3.1(a)(1) Performance of tests with prototypes or samples of the packaging, in which case the contents of the packaging for the test must simulate as closely as practicable the expected range of physical properties of the radioactive contents or packaging to be tested, must be

prepared as normally presented for transport. The use of nonradioactive substitute contents is encouraged, provided results of the testing take into account radioactive characteristics of the contents for which the package is being tested;

5.2.3.1(a)(2) Reference to a previous, satisfactory demonstration of compliance of a sufficiently similar nature. (When using this method, the sufficiently similar nature of previous tests will be justified by the primary content retention and shielding features, dimensions, weight, center of gravity, materials of construction, target surface, and free drop being the same as the package under evaluation);

5.2.3.1(a)(3) Performance of tests with models of appropriate scale incorporating those features that are significant with respect to the item under investigation when engineering experience has shown results of those tests to be suitable for design purposes. When a scale model is used, the need for adjusting certain test parameters, such as penetrator diameter or compressive load, must be taken into account. (When using this method, use of scale model test units shall be justified and defined by evidence of use for such scale modeling for other packages and the guidelines applied in International Atomic Energy Agency (IAEA) Safety Series No. 37 for scale model testing shall be applied. The validity and demonstration of scale modeling shall include a definition of the scale factor, the model represents with sufficient accuracy the details of the package, list of components not in the model, justification of deletion of a component, and justification of the similitude criteria used. A scale factor of no less than 1:4 will be used.); or

5.2.3.1(a)(4) Calculations or reasoned evaluation using reliable and conservative procedures and parameters. (When using this method, use of calculations or reasoned evaluations shall be justified and defined by providing the assumption, calculation methods, dynamic load factors, stress-strain or load-deformation information, verification and validation of any finite element analysis computer codes used, reference to source of calculations, procedures, and parameters, and verification of independent reviewer.)

5.2.3.1(b) With respect to the initial conditions for tests provided in Sections 5.2.3.2 through 5.2.3.9, compliance must be based on the assumption that the package is in equilibrium at an ambient temperature of 38 °C (100 °F).

5.2.3.2 Package Qualification and Testing. Type A equivalent packagings must be designed to withstand all tests delineated below. IP-2 equivalent packagings must be designed to withstand the tests identified in Sections 5.2.3.5, 5.2.3.6, and 5.2.3.9. IP-3 equivalent packagings must be designed to meet requirements identified for IP-2 equivalent packagings plus the test identified in Section 5.2.3.3. Methods by which compliance with these requirements may be addressed are identified in Section 5.2.3.1. The Type A, IP-3, and IP-2 packagings are typically demonstrated to have acceptable performance by exposing several prototype test units to the test conditions. Tests that are to be performed for onsite equivalent packagings are described briefly below.

5.2.3.3 Reduced Pressure Test. The reduced pressure test is intended to simulate the reduction of ambient (external) pressure to the package to 25 kPa (3.5 psia). This test is conducted to verify the ability of the package to meet the identified design requirement. This verification may

be achieved by pressurizing the internal cavity of the packaging to 78 kPa (11.2 psig). Leak detection by visual identification of loss of material, soap bubble, pressure change, or other method will be used.

5.2.3.4 Water Spray Test. The water spray test simulates exposure to rainfall of approximately 5 cm/h (2 in./h) for at least 1 hour. This test must precede each of the other tests or test sequences described in Sections 5.2.3.5 through 5.2.3.7. The time interval between the end of the water spray test and beginning of the next test shall be such that the water has soaked into the maximum extent without appreciable drying of the exterior of the package. The time interval is 2 hours if the spray is applied from four different directions simultaneously. Other tests will follow immediately if the spray is applied from each of the four directions consecutively.

5.2.3.5 Free Drop Test. The free drop test consists of a fall onto a flat, horizontal, rigid surface. The rigid surface shall be reinforced concrete. The reinforced concrete surface is defined as a 20 cm (8-in.) thick 3,000 psi concrete pad, two-way reinforced with No. 7 rebar on 30 cm (12 in.) centers. Orientation of the test package is such that the fall will cause maximum damage to the package and its safety features. The distance of the fall will be measured from the lowest part of the packaging to the upper surface of the target. The distance of the fall is determined by the gross weight of the packaging, packaging material type, and material form to be shipped. The fall distances are identified in Table 6-3.

Table 6-3. Free Drop Distance for Testing Packages to NC.

Packaging mass kilograms (pounds)	Free drop distance	
	meters	(feet)
<Mass 5,000 (11,000)	1.2	(4)
5,000 (11,000) Mass to 10,000 (22,000)	0.9	(3)
10,000 (22,000) Mass to 15,000 (33,000)	0.6	(2)
> 15,000 (33,000) Mass	0.3	(1)

For fiberboard or wood rectangular packages with a mass of 50 kg (110 lb) or less, a separate specimen must be subjected to a free drop onto each corner from a height of 0.3 m (1 ft).

For cylindrical fiberboard packages with a mass of 100 kg (220 lb) or less, a separate specimen must be subjected to a free drop onto each of the quarters of each rim from a height of 0.3 m (1 ft).

5.2.3.6 Stacking Test. The compression test lasts for a period of at least 24 hours. The compressive load is equivalent to either five times the mass of the actual package or 13 kPa (2 psi) multiplied by the vertically projected area of the package, whichever is greater. The load is applied uniformly to two opposite sides of the package, one of which must be the base on which the package would normally stand.

5.2.3.7 Penetration Test. For the test, the package must be placed on a rigid, flat, horizontal surface that will not move significantly while the test is being performed. A bar (penetration bar), 3.2 cm (1.25 in.) in diameter with a hemispherical end, and mass of 6 kg (13 lb), with its

longitudinal axis vertical, is dropped onto the center of the weakest part of the package, so that if it penetrates far enough, it will hit the containment system. The bar must not be deformed by the test.

The distance of the fall of the bar is measured from its lower end to the upper surface of the package. The determination of this distance depends on the basic physical form of the contents for which the packaging is designed. The penetration drop height for packagings designed to contain only solids is 1 m (3.3 ft).

5.2.3.8 Additional Tests for Type A Packagings Designed for Liquids and Gases.

5.2.3.8(a) In addition to the tests described in Sections 5.2.3.4 through 5.2.3.7, Type A packagings designed for liquids and gases must be capable of withstanding the following tests:

5.2.3.8(a)(1) Free Drop Test. The packaging specimen must drop onto the target so as to suffer the maximum damage to its containment. The height of the drop measured from the lowest part of the packaging specimen to the upper surface of the target must be 9 m (30 ft) or greater. The target must be as specified in Section 5.2.3.5.

5.2.3.8(a)(2) Penetration Test. The specimen must be subjected to the test specified in Section 5.2.3.7, except that the height of the drop must be 1.7 m (5.5 ft).

5.2.3.9 Vibration Standard. A randomly selected sample packaging is filled and closed as for shipment. The sample packaging is placed on a vibrating platform that has a vertical or rotary double-amplitude (peak-to-peak displacement) of 2.54 cm (1 in.). The package should be constrained horizontally to prevent it from falling off the platform, but must be free to move vertically, bounce, and rotate.

The test must be performed for at least 1 hour at a frequency that causes the package to be raised from the vibrating platform so that a piece of material of approximately 1.6 mm (0.063 in.) thickness can be passed between the bottom of any package and the platform.

Immediately following the period of vibration, the package must be removed from the platform, turned on its side, and observed for any evidence of leakage. The package must be demonstrated to meet the conditions identified in paragraph 5.2.4.2(j), items (1) and (2).

5.2.4 Design and Construction Requirements for Type A, IP-3, and IP-2 Packagings

This section defines the design and construction requirements used on the Hanford Site to achieve safety equivalent to that resulting from following the DOT regulations when shipping in commerce. It should be noted that, to ensure compliance, acceptance criteria and evaluation requirements outlined in the following sections should be factored into development of the packaging design and construction.

5.2.4.1 General Design Requirements. The general design requirements are applicable to Type A, IP-3, and IP-2 packagings.

5.2.4.1(a) The package can be easily handled and properly secured in or on a conveyance during transport.

5.2.4.1(b) Each lifting attachment that is a structural part of the package must be designed with a minimum safety factor of three against yielding when used to lift the package in the intended manner, and it must be designed so that failure of any lifting attachment under excessive load would not impair the ability of the package to meet other requirements of this TSD. Any other structural part of the package that could be used to lift the package must be capable of being rendered inoperable for lifting the package during transport or must be designed with strength equivalent to that required for lifting attachments.

5.2.4.1(c) The external surface, as far as practicable, will be free from protruding features and will be easily decontaminated.

5.2.4.1(d) The outer layer of packaging will avoid, as far as practicable, pockets or crevices where water might collect.

5.2.4.1(e) Each feature that is added to the package will not reduce the safety of the package.

5.2.4.1(f) The package will be capable of withstanding the effects of any acceleration, vibration, or vibration resonance that may arise under normal conditions (NC) without any deterioration in the effectiveness of the closing devices on various receptacles or in the integrity of the package as a whole and without loosening or unintentionally releasing the nuts, bolts, or other securing devices even after repeated use.

5.2.4.1(g) The materials of construction of the packaging and any components or structures will be physically and chemically compatible with each other and with the package contents. The behavior of the packaging and package contents under irradiation will be taken into account.

5.2.4.1(h) All valves through which the package contents could escape will be protected against unauthorized operation.

5.2.4.2 Additional Design Requirements. All additional design requirements are applicable to Type A packagings. Only requirements in paragraphs 5.2.4.2(a) through (j) apply to the IP-3. The additional design requirements are not applicable to IP-2 packagings:

5.2.4.2(a) The outside of the packaging incorporates a feature, such as a seal, that is not readily breakable, and that while intact, is evidence that the package has not been opened. In the case of packages shipped in closed transport vehicles in exclusive use, the cargo compartment, instead of the individual packages, may be sealed.

5.2.4.2(b) The smallest external dimension of the package is not less than 10 cm (4 in.).

5.2.4.2(c) Containment and shielding is maintained during transportation and storage in a temperature range of $-33\text{ }^{\circ}\text{C}$ ($-27\text{ }^{\circ}\text{F}$) and $46\text{ }^{\circ}\text{C}$ ($115\text{ }^{\circ}\text{F}$). Special attention shall be given to liquid contents and to the potential degradation of the packaging materials within the temperature range.

5.2.4.2(d) The packaging must include a containment system securely closed by a positive fastening device that cannot be opened unintentionally or by pressure that may arise within the package during normal transport. Special form Class 7 (radioactive) material may be considered as a component of the containment system. If the containment system forms a separate unit of the package, it must be securely closed by a positive fastening device that is independent of any other part of the package.

5.2.4.2(e) For each component of the containment system, account is taken, where applicable, of radiolytic decomposition of materials and generation of gas by chemical reaction and radiolysis.

5.2.4.2(f) The containment system will retain its radioactive contents under the reduction of ambient pressure to 25 kPa (3.6 psi).

5.2.4.2(g) Each valve, other than a pressure relief device, is provided with an enclosure to retain any leakage.

5.2.4.2(h) Any radiation shield that encloses a component of the packaging specified as part of the containment system will prevent the unintentional escape of that component from the shield.

5.2.4.2(i) Failure of any tiedown attachment that is a structural part of the packaging, under both NC and AC, must not impair the ability of the package to meet other requirements of this subpart.

5.2.4.2(j) When evaluated against the performance requirements of this section and tests specified in Section 5.2.4.1 or using any of the methods authorized by Section 5.2.3.1, the packaging will prevent the following:

5.2.4.2(j)(1) Loss or dispersal of the radioactive contents.

5.2.4.2(j)(2) A significant increase in radiation levels recorded or calculated at the external surfaces for the condition before the test.

5.2.4.2(k) Each packaging designed for liquids will be as follows:

5.2.4.2(k)(1) Designed to provide for ullage (head space) to accommodate variations in temperature of the contents, dynamic effects, and filling dynamics;

5.2.4.2(k)(2) Meet conditions prescribed in subparagraph (j) of this section when subjected to the tests specified in Section 5.2.4.1 or evaluated against these tests by any of the methods authorized by Section 5.2.3.1; and

5.2.4.2(k)(3) Either:

5.2.4.2(k)(3)(i) Have sufficient suitable absorbent material to absorb twice the volume of the liquid contents. The absorbent material must be compatible with the package contents and suitably positioned to contact the liquid in the event of leakage; or

5.2.4.2(k)(3)(ii) Have a containment system comprised of primary inner and secondary outer containment components designed to ensure retention of the liquid contents within the secondary outer component in the event the primary inner component leaks.

5.2.4.2(l) Each package designed for gases, other than tritium not exceeding 40 TBq (1,000 Ci) or noble gases not exceeding the A_2 value appropriate for the noble gas, will be able to prevent loss or dispersal of contents when the package is subjected to tests prescribed in Section 5.2.4.1 or evaluated against these tests by any of the methods authorized by Section 5.2.3.1.

5.2.4.3 Radiation Level Limitations.

5.2.4.3(a) Except as provided in paragraph 5.2.4.3(b), each package of Class 7 (radioactive) materials offered for transportation must be designed and prepared for shipment, so that under conditions normally incident to transportation, the radiation level does not exceed 2 mSv/h (200 mrem/h) at any point on the external surface of the package, and the TI does not exceed 10.

5.2.4.3(b) A package that exceeds the radiation level limits specified in paragraph 5.2.4.3(a) of this section must be transported by exclusive-use shipment, and the radiation levels for such shipment may not exceed the following during transportation:

5.2.4.3(b)(1) 2 mSv/h (200 mrem/h) on the external surface of the package unless the following conditions are met, in which case the limit is 10 mSv/h (1,000 mrem/h):

5.2.4.3(b)(1)(i) The shipment is made in a closed transport vehicle.

5.2.4.3(b)(1)(ii) The package is secured within the vehicle so that its position remains fixed during transportation.

5.2.4.3(b)(1)(iii) There are no loading or unloading operations between the beginning and end of the transportation.

5.2.4.3(b)(2) 2 mSv/h (200 mrem/h) at any point on the outer surfaces of the vehicle, including the top and underside of the vehicle; or in the case of a flat-bed style vehicle, at any point on the vertical planes projected from the outer edges of the vehicle, on the upper surface of the load or enclosure if used, and on the lower external surface of the vehicle.

5.2.4.3(b)(3) 0.1 mSv/h (10 mrem/h) at any point 2 m (6.6 ft) from the outer lateral surfaces of the vehicle (excluding the top and underside of the vehicle); or in the case of a flat-bed style vehicle, at any point 2 m (6.6 ft) from the vertical planes projected by the outer edges of the vehicle (excluding the top and underside of the vehicle).

5.2.4.3(b)(4) 0.02 mSv/h (2 mrem/h) in any normally occupied space, except that this provision does not apply to carriers if they operate under the provisions of a radiation protection program and if personnel under their control who are in such an occupied space wear radiation dosimetry devices.

5.2.4.3(c) For shipments made under the provisions of paragraph (b) of this section, instructions for maintenance, onsite transport controls and to avoid actions that will unnecessarily delay delivery or result in increased radiation levels or radiation exposure to workers or the general public shall be provided in the safety analysis documentation and operating procedures.

5.2.4.4 Thermal Limitations.

5.2.4.4(a) Heat generated within the package by the radioactive contents will not, during conditions normally incident to transport, affect the integrity of the package; and

5.2.4.4(b) Temperature of the accessible external surfaces of the loaded package will not, assuming still air in the shade at an ambient temperature of 38 °C (100 °F), exceed:

5.2.4.4(b)(1) 50 °C (122 °F) in other than an exclusive-use shipment; or

5.2.4.4(b)(2) 85 °C (185 °F) in an exclusive-use shipment.

5.2.4.5 Demonstration of Compliance with Tests. Following the test sequences, which expose prototypes to applicable tests, performance of the packaging is evaluated. The package must show no loss or dispersal of radioactive contents and no significant increase in radiation levels recorded or calculated at the external surfaces for the conditions before the test. A 20% increase is considered significant. If determined acceptable, the packaging is approved for use.

5.2.4.6 Lead Shielding Design and Construction Requirements. Poured lead shielding, if used, must meet the following conditions:

5.2.4.6(a) The grade and quality of the lead shall be in accordance with ASTM B-29, "Standard Specification for Refined Lead."

5.2.4.6(b) The design will be such that the lead pour is continuous and free of dross.

5.2.4.6(c) Preheating and cooling of the shielding cavity walls will be specified to produce a sound shield and to minimize thermal stresses in the container walls.

5.2.4.6(d) Supports for container walls will be provided to ensure concentricity and the required lead thickness.

5.2.4.6(e) Impingement of molten lead on container walls will be minimized.

5.2.4.6(f) Provision will be provided for venting and topping off the lead.

5.2.4.6(g) As a minimum, gamma scanning, probing, or ultrasonic testing will be used to demonstrate soundness of the shielding.

5.2.4.7 Marking. Each onsite Type A package must be marked on the outside "Onsite Type A," "Radioactive Material," and with the name and address of the packaging manufacturer.

5.2.5 Performance Tests for Fissile and Type B Packagings

In accordance with DOE Order 460.1A and supplemental guidance from DOE Guide 460.1-1, *The Packaging and Transportation Safety Implementation Guide*, minimum requirements for Type B onsite packaging that provide an equivalent degree of safety to that provided by the DOT regulations were developed. The requirements are based on DOT regulatory requirements defined in 10 CFR 71, and NRC interpretations and supplementary guidance. However, the requirements and NRC interpretations and supplementary guidance have been modified for Hanford Site-specific conditions. In addition, Hanford Site institutional, administrative, security, procedural controls, and communication were also considered in modifying the regulatory requirements. Description of and justification for deviations to the regulatory requirements are provided in Appendix B, Justification and Basis for Equivalency to DOT Regulations for Type B and Fissile Packages Transported on the Hanford Site. Demonstration of compliance with the requirements for DOT-equivalent packaging can be by testing, analysis, or a combination of both.

Normal conditions of transport (NCT) and hypothetical accident conditions (HAC) are defined in the regulations. For DOT-equivalent packaging, to eliminate confusion between regulatory requirements and onsite shipment requirements, the terminology is normal condition (NC) and accident condition (AC), respectively.

Performance tests for DOT-equivalent packaging are the same as those specified in the regulations, except in some conditions, the alternate Hanford Site-specific conditions are applied. The following are performance tests for NC and AC to demonstrate equivalency to the regulations.

5.2.5.1 Normal Conditions.

5.2.5.1(a) Evaluation of each package design must include a determination of the effect on that design condition and test specified. Separate units may be used for free drop tests, compression tests, and penetration tests. However, the units must be subjected to the water spray test before being subjected to other tests.

5.2.5.1(b) Initial conditions of the tests and demonstration of compliance shall be based on the ambient temperature preceding and following the tests remaining constant at a Hanford Site temperature extreme value between -33°C (-27°F) and 46°C (115°F). The specified temperature shall be the most unfavorable for the feature under consideration. The initial internal pressure within the containment system must be considered to be MNOP, unless a lower

internal pressure consistent with the ambient temperature considered to precede and follow the test is more unfavorable.

5.2.5.1(c) Conditions and Tests.

5.2.5.1(c)(1) Heat. An ambient temperature of 46 °C (115 °F) in still air and insolation according to that specified in Table 6-4:

Table 6-4. Insolation Table.

Form and location of surface	Total insolation for a 12-hour period (g-cal/cm ²)	Total insolation for 12-hour period (BTU/ft ²)
Flat surface/horizontal transport (base)	None	None
Flat surface/horizontal transport (other surfaces)	647	2,386
Flat surface/non-horizontal transport	162	597
Curved surfaces	324	1,193

5.2.5.1(c)(2) Cold. An ambient temperature of -33 °C (-27 °F) in still air and shade.

5.2.5.1(c)(3) Reduced External Pressure. An external pressure of 25 kPa (3.5 psia).

5.2.5.1(c)(4) Increased External Pressure. An external pressure of 140 kPa (20 psia).

5.2.5.1(c)(5) Vibration. Vibration normally incident to transport.

5.2.5.1(c)(6) Water Spray. A water spray that simulates exposure to rainfall of approximately 5 cm/h (2 in./h) for at least 1 hour.

5.2.5.1(c)(7) Free Drop. Between 1½ and 2½ hours after the conclusion of the water spray test, a free drop through the distance specified in Table 6-5 onto a reinforced concrete surface, striking the surface in a position that the package is normally transported. The reinforced concrete surface is defined as a 20 cm (8-in.) thick, 20,684 kPa (3,000 psi), concrete pad, two-way reinforced with No. 7 rebar on 30 cm (12 in.) centers.

Table 6-5. Criteria for Free Drop Test (Weight/Distance).

Package weight		Free drop distance	
Kilograms	Pounds	Meters	Feet
5,000 or less	11,000	1.2	4
5,000 to 10,000	11,000 to 22,000	0.9	3
10,000 to 15,000	22,000 to 33,000	0.6	2
More than 15,000	More than 33,000	0.3	1

5.2.5.1(c)(8) Compression. For packages weighing up to 5,000 kg (11,000 lb), the package shall be subjected to a compressive load applied uniformly to the top and bottom of the package in the position in which the package would be normally transported for a period of not less than 24 hours. The compressive load shall be the greater of five times the package weight or 13 kPa (2 psi) multiplied by the vertically projected area of the package.

5.2.5.1(c)(9) Penetration. Impact of the hemispherical end of a vertical steel cylinder of 3.2 cm (1.25 in.) diameter and 6 kg (13 lb) mass, dropped from a height of 1 m (40 in.) onto the exposed surface of the package that is expected to be most vulnerable to puncture. The long axis of the cylinder must be perpendicular to the package surface.

5.2.5.2 Accident Conditions.

5.2.5.2(a) Evaluations for AC shall be based on sequential application of the tests specified, and in the order indicated to determine their cumulative effect on a package or array of packages. Only in the case of the immersion test (paragraph 5.2.5.2[c][6]) may an undamaged unit be used for the test or evaluation.

5.2.5.2(b) Initial conditions for the tests, except for water immersion, to demonstrate compliance with the DOT-equivalent packaging requirements shall be as follows:

- Ambient temperature before and after the tests must remain constant at a value between Hanford Site temperature extremes of -33°C (-27°F) to 46°C (115°F)
- Specified temperature shall be the most unfavorable for the feature under consideration
- Initial internal pressure within the containment system must be considered to be the MNOP, unless a lower internal pressure consistent with the ambient temperature considered to precede and follow the test is more unfavorable.

5.2.5.2(c) Tests for AC shall be conducted in the following order on a single specimen:

5.2.5.2(c)(1) Free Drop. A free drop of 9 m (30 ft) onto a reinforced concrete surface, striking the surface in a position for which maximum damage is expected. The reinforced concrete surface is defined as a 20 cm (8-in.) thick, 20,684 kPa (3,000 psi), concrete pad, two-way reinforced with No. 7 rebar on 30 cm (12 in.) centers.

5.2.5.2(c)(2) Crush. The specimen is subjected to a dynamic crush test by positioning the test unit onto the concrete surface defined in paragraph 5.2.5.2(c)(1) with the long axis horizontal to the concrete surface. A specimen with the same weight, dimension, configuration, and in the same orientation is dropped onto the specimen from a height of 9 m (30 ft). This test is only required when the package has a mass not greater than 500 kg (1,100 lb), an overall density not greater than $1,000\text{ kg/m}^3$ (62.4 lb/ft^3) based on external dimensions, and radioactive contents greater than 1,000 A₂ not as special form radioactive material.

5.2.5.2(c)(3) Puncture. A free drop of the specimen through a distance of 1 m (40 in.) in a position for which maximum damage is expected, onto the upper end of a solid, vertical, cylindrical, mild steel bar, welded to steel plate and mounted onto the concrete surface specified in paragraph 5.2.5.2(c)(1). The long axis of the bar must be vertical. The bar must be 15 cm (6 in.) in diameter with the top horizontal and its edges rounded to a radius of not more than 0.6 cm (0.25 in.) and of a length as to cause maximum damage to the package. As a minimum, the bar shall be welded to a 2.5 cm (1-in.) thick by 45.7 cm (18-in.) by 45.7 cm (18-in.) square plate that is mounted to the concrete surface.

5.2.5.2(c)(4) Thermal. Exposure of the specimen fully engulfed in a hydrocarbon fuel/air fire of sufficient extent, and in sufficiently quiescent ambient conditions, to provide an average emissivity coefficient of at least 0.9, with an average flame temperature of at least 800 °C (1,475 °F) for a period of 30 minutes. The fuel source shall extend horizontally at a minimum of 1 m (40 in.), but not more than 3 m (10 ft) beyond the external surface of the specimen. The specimen shall be positioned 1 m (40 in.) above the surface of the fuel source. For purposes of calculation, the external surface absorptivity coefficients shall be nationally accepted published values at fire temperature. Cooling shall be applied after 30 minutes from the start of the fire.

5.2.5.2(c)(5) Immersion-Fissile Material. For fissile material packages where water inleakage has not been assumed for criticality analysis, the specimen shall be subjected to immersion under a water head of at least 0.9 m (3 ft) in the attitude for which maximum leakage is expected.

5.2.5.2(c)(6) Immersion-All Packages. A separate, undamaged specimen shall be subjected to water pressure equivalent to immersion under a water head of at least 15 m (50 ft). For test purposes, an external pressure of water of 150 kPa (21.7 psig) is considered to meet these conditions.

5.2.6 Design and Construction Requirements for Fissile and Type B Packagings

This section defines the fissile and Type B packaging design and construction requirements used on the Hanford Site to achieve safety equivalent to that resulting from following the DOT regulations when shipping in commerce. It should be noted that, to ensure compliance, the acceptance criteria and evaluation requirements outlined in the following sections should be factored into development of the packaging design and construction.

5.2.6.1 General Packaging Requirements.

5.2.6.1(a) The smallest overall dimension of a package may not be less than 10 cm (4 in.).

5.2.6.1(b) The outside of a package must incorporate a feature, such as a seal, that is not readily breakable and that, while intact, would be evidence that the package has not been opened by unauthorized persons.

5.2.6.1(c) Each package must include a containment or confinement system securely closed by a positive fastening device that cannot be opened unintentionally or by a pressure that may arise within the package.

5.2.6.1(d) A package must be made of materials and construction that ensures there is no significant chemical, galvanic, or other reactions among the packaging components and the package contents, including possible reactions from inleakage of water, to the maximum credible extent. Account must be taken of the behavior of materials under irradiation.

5.2.6.1(e) A package valve or other device, the failure of which would allow radioactive contents to escape, must be protected against unauthorized operations and, except for a pressure relief device, must be provided with an enclosure to retain any leakage.

5.2.6.1(f) A package must be designed, constructed, and prepared for transport so that under the NC tests specified, there would be no loss or dispersal of radioactive contents, no significant increase in external surface radiation levels, and no substantial reduction in the effectiveness of the packaging.

5.2.6.1(g) A package must be designed, constructed, and prepared for transport so that in still air at 46 °C (115 °F) and in the shade, no accessible surface of a package would have a temperature exceeding the exclusive-use temperature of 85 °C (185 °F) and the nonexclusive-use temperature of 50 °C (122 °F).

5.2.6.1(h) A package may be designed, constructed, and prepared for transport with a filtered venting feature. Filtered venting will be provided by a high-efficiency particulate air filter capable of sustaining NC and AC loadings. The filter will have a minimum aerosol efficiency of 99.97% for 0.3 µm diameter particles of dioctylphthalate smoke or equivalent. The minimum hydrogen diffusivity will be 1.90 E-06 mole/sec/mole fraction at 25 °C (77 °F). The minimum airflow rate will be 200 mL/min (12.20 in³/min) at a pressure differential of 1.9 mmHg (1.0 in. H₂O). Filter housing seals will be manufactured from materials that have a minimum shelf life of 5 to 10 years, as specified by military specification MIL-HDBK-695, or be equivalent to the filter media. The filter housing seal will be leak tight as defined in American National Standards Institute (ANSI) N14.5. The filter frame will be resistant to atmospheric corrosion by the use of zinc electroplating per section or by the inherent nature of the material (e.g., stainless steel or polyethylene).

5.2.6.2 Lifting Device Requirements.

5.2.6.2(a) Any lifting attachment that is a structural part of a package must be designed and tested in accordance with DOE/RL-92-36 and ANSI N14.6, *Radioactive Materials – Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 kg) or More*.

5.2.6.2(b) The design of the lifting device will be such that failure of the lifting device or any component under excessive load would not impair the ability of the package to meet other requirements of this section.

5.2.6.2(c) Any structural part of the package that could be used to lift the package must be capable of being rendered inoperable for lifting the package during transport, or must be designed with strength equivalent to that required for the lifting attachments.

5.2.6.3 Tiedown Devices.

5.2.6.3(a) For a system of tiedown devices that is a structural part of the package, the system will be capable of withstanding a static force applied at the center of gravity of the package having a vertical component of 2 times the gross weight of the package, a horizontal component along the direction in which the conveyance travels of 10 times the gross weight of the package, and horizontal component in the transverse direction of 5 times the gross weight of the package. The application of this load will not generate stresses in any material of the package in excess of its yield strength.

5.2.6.3(b) Any other structural part of the package that could be used to tiedown the package must be capable of being rendered inoperable for tying down the package during transport, or must be designed with strength equivalent to that required for tiedown devices.

5.2.6.3(c) Each tiedown device that is a structural part of a package will be designed so that failure of the device under excessive load would not impair the ability of the package to meet other requirements of this section.

5.2.6.4 External Radiation Requirements.

5.2.6.4(a) Each radioactive material package for "nonexclusive" use onsite transport must be designed and prepared for transport such that, under NC, the radiation level does not exceed 2 mSv/h (200 mrem/h) at any point on the external surface of the package, and the TI does not exceed 10.

5.2.6.4(b) A package that exceeds the radiation level limits specified in paragraph 5.2.6.4(a) must be transported onsite by exclusive-use transport only and the radiation levels will exceed the following limits only under the required conditions:

5.2.6.4(b)(1) The package will be secured to the conveyance so that its position remains fixed during NC. There will be no loading or unloading operations between the beginning and end of the onsite transport. Only under these conditions does the following limit of 10 mSv/h (1,000 mrem/h) on the external surface of the package apply.

5.2.6.4(b)(2) At any point on the outer external surface of the conveyance, including top and underside, radiation levels will not exceed 2 mSv/h (200 mrem/h). In the case of a flat-bed style conveyance, the top is defined as the top of the load, the sides are defined as the vertical planes projecting from the outer edges of the conveyance, and the underside is defined as the lower external surface of the conveyance.

5.2.6.4(b)(3) At any point 2 m (80 in.) from the outer lateral surfaces of the conveyance (excluding the top and underside of the conveyance), the radiation levels will not exceed 0.1 mSv/h (10 mrem/h).

5.2.6.4(b)(4) In any normally occupied space, the radiation level will not exceed 0.02 mSv/h (2 mrem/h).

5.2.6.4(b)(5) Instructions for maintenance, onsite transport controls, and to avoid actions that will unnecessarily delay delivery or result in increased radiation levels or radiation exposure to workers or the general public will be provided in the safety analysis documentation and operating procedures.

5.2.6.5 NC Subcriticality Assurance Requirements. All radioactive material packages containing fissile material must be designed, constructed, and contents so limited that under NC the following requirements are met to ensure subcriticality:

5.2.6.5(a) The contents will remain subcritical under all normal performance tests conditions.

5.2.6.5(b) The geometric form of the package contents would not be substantially altered under all normal performance test conditions.

5.2.6.5(c) There will be no leakage of water into the containment or confinement system unless an evaluation of an undamaged package is performed which assumes that moderation is present to such an extent as to cause maximum reactivity consistent with the chemical and physical form of the material.

5.2.6.5(d) There will be no substantial reduction in the effectiveness of the packaging defined as follows:

5.2.6.5(d)(1) No more than a 5% reduction in the total effective volume of the packaging on which nuclear safety is assessed.

5.2.6.5(d)(2) No more than a 5% reduction in the effective spacing between the fissile contents and the outer surface of the package.

5.2.6.5(d)(3) No occurrence of an aperture in the outer surface of the packaging large enough to permit the entry of a 10 cm (4-in.) cube.

5.2.6.6 AC Subcriticality Assurance Requirements. All radioactive material packages containing fissile material must be designed, constructed, and contents so limited that subcriticality is ensured under the following conditions:

5.2.6.6(a) The fissile material is in the most reactive credible configuration consistent with the damaged condition of the package, and the chemical and physical form of the contents.

5.2.6.6(b) Water moderation occurs to the most reactive credible extent consistent with the damaged condition of the package and the chemical and physical form of the contents.

5.2.6.6(c) There is full reflection by water on all sides, as close as is consistent with the damaged condition.

5.2.6.7 Package Array Subcriticality Assurance Requirements. Radioactive material packages that are transported in arrays and contain fissile material must be designed, constructed, and contents so limited that the following requirements are met to ensure subcriticality:

5.2.6.7(a) Fissile material packages must be controlled by the operating procedures to ensure that an array of packages remains subcritical. To establish this control, the designer of the package shall derive a number "N" based on the following conditions and assuming packages are stacked in any arrangement and with close full reflection on all sides of the stack by water.

5.2.6.7(a)(1) Five times "N" undamaged packages with nothing between the packages shall be subcritical.

5.2.6.7(a)(2) Two times "N" damaged packages must be subcritical with optimum interspersed hydrogenous moderation.

5.2.6.7(a)(3) The value of "N" shall not be less than 0.5.

5.2.6.7(b) The TI based on nuclear criticality control must be obtained by dividing the number 50 by the value of "N" derived using the procedure specified above in paragraph 5.2.6.7(a). The value of the TI for nuclear criticality control may be zero provided that an unlimited number of packages is subcritical such that the value of "N" is effectively equal to infinity under the procedure specified in paragraph 5.2.6.7(a). Any TI greater than zero must be rounded up to the first decimal place.

5.2.6.7(c) When a fissile material package is assigned a nuclear criticality control TI, the following requirements shall be applied:

5.2.6.7(c)(1) For a package TI not greater than 10, procedures must be in place to limit the sum of the TIs to 50 for a nonexclusive-use onsite shipment, and to 100 in an exclusive-use onsite shipment.

5.2.6.7(c)(2) For a package TI greater than 10, procedures must in be place to limit the sum of the TIs to 100 in an exclusive-use conveyance.

5.2.6.8 Normal Transport Conditions. Under NC, there shall be no loss or dispersal of radioactive material as demonstrated to a sensitivity of 10^{-6} A₂ per hour, no significant increase in external radiation levels, and no reduction in the effectiveness of the packaging.

Note: The sensitivity requirement 10^{-6} A₂ per hour is not applicable to Type A fissile.

5.2.6.9 Accident Conditions. Under AC, there shall be no release of krypton-85 exceeding 10 A₂ in 1 week, and no release of other radioactive material exceeding a total 1 A₂ in 1 week. In addition, the external radiation dose rate shall not exceed 10 mSv/h (1 rem/h) at 1 m (40 in.) from the external surface of the package.

5.2.6.10 Containment or Confinement Boundary Requirements. Confinement, in accordance with 10 CFR 72.3, means those systems and components of a package, including filtration devices, which act as barriers between radioactive substances contained by the package and the environment.

Containment, in accordance with 10 CFR 71.4, means those system and components of the packaging intended to retain radioactive material during transport with a leak testable boundary.

5.2.6.10(a) As a minimum, the primary containment or confinement boundary will be designed and constructed in accordance with the requirements of American Society for Mechanical Engineers (ASME) boiler and pressure vessel code (BPVC), Section VIII, Division 1. It is not intended that the ASME Code stamp be applied to the shipping container or containment or confinement boundary. Design loadings and load combination on the containment or confinement boundary will be as specified in Table 6-6.

Table 6-6. Design Loadings and Load Combination on the Containment or Confinement Boundary.

Normal or Accident Condition	Applicable Initial Conditions							Maximum Internal Pressure	Maximum Weight of Contents*
	Ambient Temperature		Insolation		Decay Heat				
	46 °C (115 °F)	-33°C (-27 °F)	Max	-17.8 °C (0 °F)	Max	-17.8 °C (0 °F)			
Normal Conditions									
Hot Ambient			X		X		X		
Cold Ambient				X	X		X		
				X		X	X		
Minimum External Pressure	X		X		X		X		
		X		X	X		X		
Vibration and Shock Normally incident to the mode of transport	X		X		X		X		
		X		X	X		X		
		X		X		X	X		
Free Drop	X		X		X		X	X	
		X		X	X		X	X	
		X		X		X	X	X	
Accident Conditions									
Free Drop (9 m [30 ft])	X		X		X		X	X	
		X		X	X		X	X	
		X		X		X	X	X	
Puncture	X		X		X		X	X	
		X		X	X		X	X	
		X		X		X	X	X	
Thermal (Fire Accident)	X		X		X		X		

* It should be noted that local structural responses might be greater during an impact test if the weight of the contents is less than maximum. Subsequently, free drop testing should also be conducted at minimum content weights.

5.2.6.10(b) Cover plates and lids will be recessed or otherwise protected.

5.2.6.10(c) The containment or confinement boundary will be designed such that vessel, closures, penetration closures, and closure seals are leak testable. In addition, if full, drain or test ports utilize quick-disconnect valves, such valves will not preclude leak testing of their containment seals.

5.2.6.10(d) For vented packages, the containment or confinement boundary will consist of the containment vessel, closure, closure seals, filter housing, and filter housing seals, all of which must be leak testable.

5.2.6.10(e) Containment or confinement boundary seals will be specified to be compatible with the packaging and contents, and remain below allowable limits under both NC and AC. In addition, as a minimum, the seals must not degrade under irradiation from the contents for a period of 5 years from the time of loading.

5.2.6.10(f) As a minimum, containment or confinement boundary seals will have a shelf life of 10 years.

5.2.6.10(g) The materials of construction of seals and ancillary components will be manufactured to nationally or internationally recognized standards, such as ASTM, ANSI, and ISO.

5.2.6.11 Lead Shielding Design and Construction Requirements.

5.2.6.11(a) The grade and quality of the lead shall be in accordance with ASTM B-29.

5.2.6.11(b) The design will be such that the lead pour is continuous and free of dross.

5.2.6.11(c) Preheating and cooling of shielding cavity walls will be specified to produce a sound shield and to minimize thermal stresses in the container walls.

5.2.6.11(d) Supports for container walls will be provided to ensure concentricity and the required lead thickness.

5.2.6.11(e) Impingement of molten lead on container walls will be minimized.

5.2.6.11(f) Provision will be provided for venting and topping off the lead.

5.2.6.11(g) As a minimum, gamma scanning, probing, or ultrasonic testing will be used to demonstrate soundness of the shielding.

5.2.6.12 Pressure Testing Prior to Use. Prior to initial use of the packaging, the containment/confinement boundary will be pressure tested to 150% of its rated MNOP to ensure structural integrity at that pressure.

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Subject: **CONTRACT NO. DE-AC06-96RL13200 – APPROVAL OF THE HANFORD
SITE TRANSPORTATION SAFETY DOCUMENT (TSD)**

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APR - 4 2002

Mr. E. K. Thomson, President
Fluor Hanford, Inc.
Richland, Washington 99352

Dear Mr. Thomson:

**CONTRACT NO. DE-AC06-96RL13200 – APPROVAL OF THE HANFORD SITE
TRANSPORTATION SAFETY DOCUMENT (TSD)**

The attached Hanford Site TSD, DOE/RL-2001-0036, for onsite packaging and transportation activities has been approved by RL as meeting the requirements of 10 CFR 830.207(a) and DOE O 460.1A, Attachment 1, Criterion 5. The TSD was prepared using a collaborative approach involving FHI, BMI, BHI, CHG and RL staff. The basis for approval is documented in the attached Safety Evaluation Report.

Onsite packaging and transportation activities may be conducted using the methodology described in the TSD without further RL approval unless the proposed activity results in an Unreviewed Safety Question (USQ) using the methodology discussed in the TSD. Modifications to approved risk based packages resulting in an USQ shall be submitted to RL for approval. Risk based packages included in the TSD are approved. New risk-based packages must be submitted to RL for approval prior to use. Modifications to approved risk based packages resulting in a negative USQ determination per the methodology in the TSD shall be submitted to RL for information until further notice. The submittals should include documentation addressing all aspects of the risk-based USQ methodology described in the TSD.

In lieu of using the TSD for onsite shipments, Hanford contractors may elect to submit all packaging not meeting Department of Transportation requirements to RL for approval or submit an alternative TSD for RL approval in accordance with 10 CFR 830.207(a) and DOE O 460.1A, if deemed to be more cost effective than implementation of the TSD. The basis for such a determination shall be submitted to RL for approval prior to developing an alternative TSD.

Mr. E. K. Thomson
02-ABD-0085

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APR - 4 2002

If you have any questions, you may contact me, or your staff may contact Richard Self,
Authorization Basis Division, on (509) 372-2863.

Sincerely,

A handwritten signature in cursive script that reads "Michael A. Schleuder".

Keith A. Klein
Manager

(for)

ABD:RJS

Attachments

**Safety Evaluation Report
Hanford Transportation Safety Document
DOE/RL-2001-0036, Revision 0**

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1.0 EXECUTIVE SUMMARY

This safety evaluation report (SER) documents the basis for the U.S. Department of Energy (DOE), Richland Operations Office (RL) to approve the Hanford Sitewide Transportation Safety Document (TSD) for onsite Transportation and Packaging (T& P) at Hanford. Hanford contractors, on behalf of DOE-RL, prepared and submitted the Hanford Sitewide Transportation Safety Document, DOE/RL-2001-0036, Revision 0, (DOE/RL 2001), dated October 4, 2001, which is referred to throughout this report as the TSD.

In the context of the TSD, Hanford onsite shipments are the activities of moving hazardous materials, substances, and wastes between DOE facilities and over roadways where public access is controlled or restricted and includes intra-area and inter-area movements. The TSD sets forth requirements and standards for onsite shipment of radioactive and hazardous materials and wastes within the confines of the Hanford Site on roadways where public access is restricted by signs, barricades, fences, or other means including road closures and moving convoys controlled by Hanford Site security forces.

The Hanford Sitewide TSD was developed in accordance with the 10 CFR 830 (CFR 2001a) "safe harbor" Documented Safety Analysis (DSA) methodology for transportation activities, DOE Order 460.1A, *Packaging and Transportation Safety* (DOE 1996c), and DOE-STD-3009, *Preparation Guide for U. S. Department of Energy Non-reactor Nuclear Facility Safety Analysis Reports* (DOE 2000). The Hanford TSD was prepared in coordination with all Hanford Site contractors, DOE-RL, and DOE-Office of River Protection (-ORP). The Hanford TSD complies with 10 CFR 830 as a Documented Safety Analysis (DSA) for onsite transportation and packaging activities at Hanford. This TSD can be used by all tenants at the Hanford Site. Site tenants may also elect to develop their own Rule-compliant TSD, which will require separate DOE approval.

The format and content of this Safety Evaluation Report is based on the guidance provided in DOE-STD-1104-96, *Review and Approval of Nonreactor Nuclear Facility Safety Analysis Reports* (DOE 1996b). This approval is for Revision 0 to Hanford TSD and TSRs (DOE/RL 2001) dated March 11, 2002.

The TSD and TSRs was prepared to comply with the requirements of 10 CFR 830 (CFR 2001a), Subpart B. Consistent with Table 2, Appendix A of 10 CFR 830, Subpart B, the TSD was produced using "safe harbors" following the methodology presented in DOE-O 460.1A *Packaging and Transportation Safety* (DOE 1996c), including guidance from DOE-G 460.1-1, *Implementation Guide for Use with DOE O 460.1A, Packaging and Transportation Safety* (DOE 1997a). The Risk-Based package and transportation analysis followed the analytical methodology, and is compared to the Evaluation Guideline in DOE-STD-3009, *Preparation Guide for U. S. Department of Energy Non-reactor Nuclear Facility Safety Analysis Reports* (DOE 2000).

On February 5th, 2002, the DOE-RL issued the *Fluor Hanford Nuclear Safety Basis Strategy and Criteria* (DOE/RL 2002), which define the following consequence level guidelines:

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Table 1 – Safety Basis Criteria Consequence Levels (Bins)

Consequence	Public	Worker at 100 meters
High	>25 rem TEDE or >ERPG-2 / TEEL-2	>100 rem TEDE or >ERPG-3 / TEEL-3 @ facility boundary or prompt death to facility worker
Moderate	>1 rem TEDE or >ERPG-1 / TEEL-1	>25 rem TEDE or >ERPG-2 / TEEL-2 @ facility boundary, or serious injury to facility worker, or significant radiological or chemical exposure to facility workers
Low	<moderate consequences	<moderate

The consequence guidelines are incorporated into the following risk matrix to assist in identifying the most significant risk to designated receptors:

Table 2 - Risk Classes-Frequency versus Consequences

Consequence	Frequency Bin			
	Below 10 ⁻⁵ / yr	10 ⁻⁵ to 10 ⁻⁴ / yr	10 ⁻⁴ to 10 ⁻³ / yr	Above 10 ⁻³ / yr
High	Risk Class III	Risk Class II	Risk Class I	Risk Class I
Moderate	Risk Class IV	Risk Class III	Risk Class II	Risk Class I
Low	Risk Class IV	Risk Class IV	Risk Class III	Risk Class III

In addition to the Evaluation Guideline in DOE-STD-3009 (DOE 2000), the Risk-Based accident analyses are compared to the Hanford Safety Basis Criteria (DOE/RL 2002) to identify the level of risk associated with Risk-Based transportation activities, and assure sufficient controls are incorporated to maintain risk within DOE-RL expectations.

The TSD identifies three categories of packages, all of which achieve “equivalent safety” as required by DOE Order 460.1A (DOE 1996c) for onsite shipments. This includes two categories that specifically meet the intent of 10 CFR 71. The third category of packages used for onsite shipments relies on a Risk-Based approach consistent with DOE-STD-3009-94 (DOE 2000) to demonstrate an equivalent level of safety for the public and an acceptable level of safety for onsite workers:

1. **DOT Compliance:** For all non-radioactive and most radioactive hazardous materials, equivalent safety is achieved by implementing the U.S. Department of Transportation (DOT) regulations onsite. No specific accident analyses is required or provided in the TSD for shipments performed under DOT compliance.

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2. DOT-Equivalent Packaging: For radioactive materials, where full compliance with DOT regulations cannot be achieved, an equivalent method for achieving the accepted national level of safety will be implemented. When full compliance with DOT cannot be achieved because of technical or economic conditions, meeting site-specific standards and performance requirements is the preferred technique for achieving safety equivalent to that resulting from following DOT regulations when shipping in commerce. No specific accident analyses are provided in the TSD for shipments performed with DOT-Equivalent Packaging. Rather, deterministic analyses are conducted to support Hanford Site performance-based certification. DOT-Equivalent packages are required to meet TSD-specified design and performance requirements that are equivalent to DOT-compliant packages for Hanford Site conditions. Package performance may be demonstrated by testing or analysis.

3. Non-equivalent Packaging (Risk-Based Packaging): When full compliance with DOT regulations or compliance with DOT-equivalent packaging for radioactive materials cannot be achieved, a Risk-Based method for demonstrating the accepted level of safety will be implemented. This technique must demonstrate safety equivalent to that resulting from following DOT regulations when shipping in commerce through the use of an approved DOE-RL and DOE-ORP Risk-Based approach, consistent with DOE-STD-3009-94. Specific bounding accident scenarios are evaluated to demonstrate that use of Risk-Based packaging does not exceed the DOE-STD-3009-94 Evaluation Guideline and the Consequence Guidelines of the Safety Basis Criteria (DOE/RL 2002). No safety-class structures, systems, or components (SSC's) are associated with Risk-Based packaging. Risk-Based packaging accidents are summarized below:

5.2.6.13 Tiedown System Requirements.

5.2.6.13(a) For transport of all onsite packagings having a gross weight in excess of 454 kg (1,000 lb), an engineered tiedown system will be provided.

5.2.6.13(b) Design of the tiedown system will comply with the requirements of either DOT regulations 49 CFR 393.100 or the *North American Cargo Securement Standard*. The tiedown system design must conform to requirements of either the regulations or the standard, but not a combination of both. In application of the regulations or the *North American Cargo Securement Standard*, the requirements of paragraphs 5.2.6.13(c) through 5.2.6.13(m) will apply.

5.2.6.13(c) In the case of onsite packages, each package must be secured by one of the following:

- A single package, a group of articles, or a combination of packages loaded side by side across the width of the conveyance must be secured by at least one tiedown assembly over its top for at least 2.4 m (8 ft) of its length, and at least two tiedown assemblies securing each individual package or combination of packages banded or otherwise secured together and handled as a unit. However, articles that have a length of 2.4 m (8 ft) or less and are securely butted against each other in the fore and aft direction may be secured by metal angles secured by tiedown assemblies, or they may be secured by a timber having a nominal cross section of 10 cm x 10 cm (4 in. x 4 in.) or more placed longitudinally over the packages and secured by tiedown assemblies.
- If packages are tiered and each tiered article rests securely on the one beneath it, the tier may be secured in the same manner as a single level of those articles in accordance with the rules outlined in this section.
- These rules do not apply to packages that are fastened together by special methods and properly secured to the conveyance in accordance with the loading requirements of 49 CFR 393.100 or the *North American Cargo Securement Standard*.
- These rules do not apply for packages transported on conveyances specifically designed and constructed for transport of the package.

5.2.6.13(d) In all cases, packages will be blocked and braced to prevent longitudinal and lateral movement under the loads specified in DOT regulations 49 CFR 393.100 or the *North American Cargo Securement Standard* by one of the following:

- Chains used as blocking that are wrapped around the base of the package or secured to the bottom of the package and attached to the conveyance. The aggregate working load limit (WLL) of chocking chains must be greater than the loads specified in the regulations or the *North American Cargo Securement Standard*.

- Commercial grade lumber nailed to the deck of the conveyance may be used as blocking. The commercial grade lumber will be a minimum of 2 x 4 sections. Lumber must be predrilled and nailed to the deck by a minimum of 10 d common nails driven in right angles to the conveyance deck.
- Packages may also be secured against lateral movement by securely blocking or bracing against the side, sideboards, or stakes of the conveyance.

5.2.6.13(e) In all cases for tiedowns, load binders and hardware must have a strength equal to or greater than the minimum strength specified for the tiedowns. The handles of "breakover" load binders used to tighten chains will be secured with two loops of No. 12 gage wire or with the loose end of the chain to prevent opening of the binder during transport. If ratchet type load binders are used, it is not necessary to bind the handles.

5.2.6.13(f) Hooks, bolts, welds, or other connectors by which a tiedown assembly is attached to the conveyance, including the mountings, must have a strength equal to or greater than the tiedown assembly in the direction the tiedown assembly is loaded.

5.2.6.13(g) Anchorages of winch or other fastening devices mounted on a conveyance and used in conjunction with a tiedown assembly must have a combined strength equal to or greater than the strength of the tiedown assembly.

5.2.6.13(h) A tiedown assembly and its associated connectors and attachment devices must be designed, constructed, and maintained so the driver of the conveyance can tighten them. However, the rules of this subparagraph do not apply to a securement system in which the tiedown assembly consists of steel strapping or to a tiedown assembly that is not subject to the rules of this section.

5.2.6.13(i) Intermodal-type cargo containers having integral securement devices must be fastened to the chassis of the conveyance with securement devices that prevent them from being unintentionally unfastened. Securement devices must restrain the container from moving more than .6 cm (.25 in.) forward or aft, and more than .6 cm (.25 in.) laterally when the container is subjected to the following accelerations relative to the conveyance:

Direction of force relative to longitudinal axis of vehicle	Acceleration in <i>g</i> 's
Downward	1.70
Upward	0.50
Lateral	0.30
Longitudinal	1.80

5.2.6.13(j) Tiedown materials specified must have minimum WLLs listed in the table located in 49 CFR 393.102(b)(2). These limits are to be used when the tiedown material is not marked by the manufacturer with a WLL. Tiedown materials that are marked by the manufacturer with WLLs that differ from the table will be considered to have a WLL equal to the value for which they are marked.

5.2.6.13(k) The package assembly will have, or be provided with, a base of such size that the load on the conveyance bed does not exceed the rated unit loading or the rated capacity of the conveyance. In cases where the unit load rating information is not available for a particular conveyance, $2,442 \text{ kg/m}^2$ (500 lb/ft^2) will be assumed as the maximum unit load rating.

5.2.6.13(l) Proper placement of the load on the conveyance to avoid exceeding allowable axle loadings is the responsibility of the driver. The location of the attachment points on the conveyance and dimensions of the tiedown members will be selected accordingly.

5.2.6.13(m) To reduce wear and contact loads, wooden blocks or rubber bumpers will be placed between the package and tiedowns, tiedowns and the edge of the conveyance, and the tiedowns.

5.2.6.14 Marking. Each onsite Type B package must be marked on the outside "Onsite Type B" "Radioactive Material," and with the name and address of the packaging manufacturer. Each onsite fissile package must be marked on the outside "Onsite Fissile Type *," "Radioactive Material," and with the name and address of the packaging manufacturer.

5.3 NONEQUIVALENT (EQUIVALENT RADIOLOGICAL AND TOXICOLOGICAL PROTECTION)

For some radioactive materials requiring IP-2, IP-3, Type A, fissile or Type B packagings, equivalent safety is achieved onsite by implementing a mixture of the DOT regulations and equivalent requirements. Sections 5.3.1 through 5.3.2.3 identify 49 CFR Transportation chapters, subchapters, parts, subparts and sections that will be met by full compliance with DOT regulations. Those regulations where equivalent actions will be used to result in radiological and toxicological safety equivalent to that resulting offsite from full compliance with DOT regulations are identified in Section 5.3.1.

5.3.1 Identification of DOT Chapters and Parts When using Nonequivalent Packaging

Table 6-1 identifies to the subpart level the regulations, found in 49 CFR 100 through 185. Table 6-2 identifies to the subpart level the regulations, found in 49 CFR 350 through 399. The tables differentiate which regulations are DOT administrative activities not applicable to onsite shipments (transfers), which regulations are written such that onsite implementation (compliance) without modification can be achieved, and which regulations are implemented onsite by emulating the requirements.

* Use A or B as applicable.

When using the alternate onsite nonequivalent packaging requirements, substitutions are made for the regulations on the table that identify the requirements for:

- DOT design and performance requirements for IP-2, IP-3, Type A, fissile and Type B packagings
- Communication requirements associated with shipment of radioactive materials.

The alternate requirements are used under the equivalent safety provision of DOE Order 460.1A.

Before implementing the nonequivalent packaging method, a documented evaluation showing that both DOT compliance and DOT-equivalent packaging compliance are not technically or economically practical must be prepared and retained on file for the life of the package.

5.3.2 Identification of Onsite Radioactive Materials Nonequivalent Packaging Requirements

When using the alternative onsite nonequivalent packaging requirements, the following requirements are substituted in place of the regulations identified in Table 6-1. These requirements may be used for demonstrating equivalent safety for IP-2, IP-3, Type A, fissile and Type B packagings.

5.3.2.1 Consequence Requirements. When compliance to DOT or onsite equivalent packaging standards cannot be demonstrated, radiological and toxicological safety, and/or acceptable risk may be used to demonstrate an equivalent level of safety. Nonequivalent, risk-based packaging, by definition, does not maintain the same level of containment as DOT or equivalent packaging. Therefore, equivalent packaging containment performance requirements are identified. The equivalent containment performance requirements consider the specific characteristics and controls of the shipment of onsite risk-based packages. Limits are set for exposure and the intake of radionuclides into the body. The intake limits are set at the same amount of intake from AC and NC releases used in evaluating shipments made in commerce. The following limits apply when preparing a consequence evaluation to demonstrate equivalent radiological and toxicological safety. No limits are identified for what is considered an acceptable level of risk.

- The consequence analysis shall quantitatively evaluate impact and fire accidents. The severity of these accidents shall be at least as severe as to cause complete failure of the package, but no more severe than the accident test conditions for certified Type B packages described in 10 CFR 71.73, "Hypothetical accident conditions."
- The immediate worker is defined as any individual within a 100 m (328 ft) radius of the accident that receives an uncontrolled dose as a result of the accident.
- The co-located worker is defined as the maximally-exposed individual within the Hanford Site boundary and at or beyond a 100 m (328 ft) radius of the accident.

- The offsite receptor is defined as the maximally-exposed member of the public at or beyond the Hanford Site boundary for an exposure duration of not less than 2 hours.
- The consequence analysis shall evaluate the inhalation exposure pathway for the co-located worker and offsite receptors. The inhalation dose shall be qualitatively evaluated for the immediate worker. The external photon, beta, and neutron dose for the immediate worker shall also be evaluated.
- The evaluation shall account for the physical and chemical form of the waste material that leaks from the damaged package during the AC through the use of appropriate airborne release fractions and respirable fractions taken from DOE-HDBK-3010-94.
- The consequence analysis may include the mitigating effect of the damaged package after the AC through the use of damage ratios or leak path factors, provided that a structural analysis supports the mitigating effect. These factors shall be clearly identified where the results are reported. If these factors are used, then the dose consequences, with and without these mitigating factors, shall be reported.
- The consequence analysis shall determine that the maximum dose to the offsite receptor does not exceed that in Table 6-7. Note that the values in Table 6-7 are based, in part, on not exceeding an intake by the offsite receptor of 10^{-6} A₂. Furthermore, the consequence analysis shall determine that the dose to the co-located worker is no greater than the dose to the worker (onsite, 100 m [328 ft]) for the applicable scenario in Table 6-7. Use of Table 6-7 is limited to packages, together with the planned payload, that satisfies the Type A packaging tests. The bases for a limit of 10^{-6} A₂ intake by the offsite receptor and a risk limit for the onsite worker are given in Appendix G.
- Criticality requirements for risk-based packaging are those identified for equivalent packaging given in Sections 5.2.6.5, 5.2.6.6, and 5.2.6.7.

The applicant is required to assess the frequencies and consequences of up to eight accident scenarios. For packages that weigh 500 kg (1,102 lb) or less, the seven accident scenarios that could lead to package failure and release of radioactive materials that must be quantified are as follows:

- Impact-only
- Impact followed by fire
- Puncture-only
- Puncture followed by fire
- Crush-only
- Crush followed by fire
- Fire-only
- Hydrogen deflagration.

Note: A criticality accident shall be demonstrated to be not credible under all postulated NC and AC.

Table 6-7. Dose Consequence and Release Frequency from Packages in Transportation

Material	MAR ¹ A ₀	DR	ARF ²	RF ²	LPF	BR m ³ /h	MOI (Offsite)		Calculated Worker Consequences rem	Annual Frequency ⁴ /yr	Accident-specific Assumptions ⁵
							Y/Q ³ cm ³	Consequences rem			
1. TRU (Pu - Alpha), 100/200 Areas, Unmitigated											
Collision (spill)	30000	1	1E-03	1E-01	1	3.33E-04	4.40E-06	1.04E-01	7.32E-02	1.70E+02	Unmitigated
Fire	30000	1	1.45E-03	1	1	3.33E-04	2.00E-05	7.03E-01	1.08E-03	3.57E+01	Unmitigated
Hydrogen Degradation	3000	1	2.145E-02	7.2E-01	1	3.33E-04	2.00E-05	7.40E-01	1.08E-03	3.80E+01	Unmitigated
2. TRU (Pu - Alpha), 300 Area, Unmitigated											
Collision (spill)	10000	1	1E-03	1E-01	1	3.33E-04	6.30E-05	4.03E+00	4.21E-02	3.20E+01	Unmitigated
Fire	10000	1	1.45E-03	1	1	3.33E-04	2.00E-04	2.31E+00	1.08E-03	1.21E+01	Unmitigated
Hydrogen Degradation	940	1	2.145E-02	7.2E-01	1	3.33E-04	2.00E-04	2.31E+00	1.08E-03	1.21E+01	Unmitigated
3. non-TRU (Ca, Sr - Beta-Gamma), 100/200 Areas, Unmitigated											
Collision (spill)	60000	1	1E-03	1E-01	1	3.33E-04	4.40E-05	1.77E-01	7.32E-02	2.08E+02	Unmitigated
Fire	60000	1	1.45E-03	1	1	3.33E-04	2.00E-05	1.20E+00	1.08E-03	6.07E+01	Unmitigated
Hydrogen Degradation	6000	1	2.145E-02	7.2E-01	1	3.33E-04	2.00E-05	1.27E+00	1.08E-03	6.47E+01	Unmitigated
4. non-TRU (Ca, Sr - Beta-Gamma), 300 Area, Unmitigated											
Collision (spill)	10000	1	1E-03	1E-01	1	3.33E-04	8.30E-05	4.20E+00	4.21E-02	2.77E+01	Unmitigated
Fire	10000	1	1.45E-03	1	1	3.33E-04	2.00E-04	1.97E+00	1.08E-03	1.03E+01	Unmitigated
Hydrogen Degradation	940	1	2.145E-02	7.2E-01	1	3.33E-04	2.00E-04	1.97E+00	1.08E-03	1.03E+01	Unmitigated
5. TRU (Pu - Alpha), 100/200 Areas, Mitigated											
Collision (spill)	30000	1E-01	1E-03	1E-01	1	3.33E-04	4.40E-05	1.04E-02	7.32E-02	1.70E+01	(a) and (b)
Fire	30000	1	1.45E-03	1	1	3.33E-04	2.00E-05	7.03E-01	1.08E-03	3.57E+01	Unmitigated
Hydrogen Degradation	3000	1	2.145E-02	7.2E-01	1	3.33E-04	2.00E-05	7.40E-01	1.08E-03	3.80E+01	(c) and (d)
6. TRU (Pu - Alpha), 300 Area, Mitigated											
Collision (spill)	10000	1E-01	1E-03	1E-01	1	3.33E-04	6.30E-05	4.03E-01	4.21E-02	3.20E+00	(a) and (b)
Fire	10000	1	1.45E-03	1	1	3.33E-04	2.00E-04	2.31E+00	1.08E-03	1.21E+01	Unmitigated
Hydrogen Degradation	940	1	2.145E-02	7.2E-01	1	3.33E-04	2.00E-04	2.31E+00	1.08E-03	1.21E+01	(c) and (d)
7. non-TRU (Ca, Sr - Beta-Gamma), 100/200 Areas, Mitigated											
Collision (spill)	60000	1E-01	1E-03	1E-01	1	3.33E-04	4.40E-05	1.77E-02	7.32E-02	2.88E+01	(a) and (b)
Fire	60000	1	1.45E-03	1	1	3.33E-04	2.00E-05	1.20E+00	1.08E-03	8.07E+01	Unmitigated
Hydrogen Degradation	6000	1	2.145E-02	7.2E-01	1	3.33E-04	2.00E-05	1.27E+00	1.08E-03	8.47E+01	(c) and (d)
8. non-TRU (Ca, Sr - Beta-Gamma), 300 Area, Mitigated											
Collision (spill)	10000	1E-01	1E-03	1E-01	1	3.33E-04	6.30E-05	4.03E-01	4.21E-02	2.77E+00	(a) and (b)
Fire	10000	1	1.45E-03	1	1	3.33E-04	2.00E-04	1.97E+00	1.08E-03	1.03E+01	Unmitigated
Hydrogen Degradation	940	1	2.145E-02	7.2E-01	1	3.33E-04	2.00E-04	1.97E+00	1.08E-03	1.03E+01	(c) and (d)

Notes:

- 1) The MAR for the collision and fire scenarios are conveyance limits, while the MAR for the hydrogen degradation is a package limit.
- 2) The ARF and RF are shown for multiple release mechanisms of the accident scenario. The payload materials are assumed to consist of surface contaminated combustible or noncombustible materials or powder, whichever is more conservative for the accident scenario.
- 3) Uses the original Site fence line as the Site boundary. The Y/Qs for all fires and hydrogen degradations include the effects of plume rise from a small fire of 0.17 MW (1 gallon diesel fuel). A fire with a lower heat release rate gives conservatively higher doses than a larger heat release rate. The Y/Qs for all of the fires and hydrogen degradations are taken from DFNW-ECAL-100, the Y/Qs for the collision in the 100/200 Areas are taken from Table 7-3, and the Y/Qs for the collision in the 300 Area is taken from Table 7-4 (to exclude building wake effects).
- 4) Frequency calculation assumes 60 trips per year, 9 miles per trip for cases 1 and 5; 80 trips per year, 20 miles per trip for cases 2 and 6; 6 trips per year, 5 miles per trip for cases 3 and 7; and 50 trips per year, 20 miles per trip for cases 4 and 8. For simplicity, all transportation accidents are assumed to result in at least partial failure of at least 1 package in an array, 1.6% of accidents result in fire, and the results in at least partial failure of at least 1 package in an array. Frequency of an unmitigated H₂ degradation is assumed to be "anticipate," frequency of mitigated H₂ degradation is "extremely unlikely" (Ref. WRSAP SAR, p. 3-61).
- 5) Accident-specific assumptions
 - a) Maximum speed limited to 55 mph
 - b) Sandbags or non-rad (dunnage) containers in front row of array, high DE-CI containers in rear
 - c) Containers vented prior to shipment
 - d) Chemically reactive contents not present
- 6) Inhalation dose from 10³ A₀: 2.32 rem from Pu-239 (Class W-nitrate), 1.98 rem from 60/60 mix of Sr/Cs (3.52 rem Sr-90, 0.431 rem Cs-137).

For packages that weigh greater than 500 kg (1,102 lb), accident scenarios that involve crush-only and crush followed by fire need not be evaluated, in accordance with 10 CFR 71.

An important element of the risk-based approval process is providing justification for the package failure thresholds that are used to determine the conditional probabilities of encountering ACs that are more severe. The failure thresholds should be determined in the structural and thermal analysis sections of the application for approval of a risk-based package. References to the appropriate sections of the application where the failure thresholds are determined should be provided. For cases in which failure thresholds are not or cannot be determined, an appropriate conservative assumption must be provided and justified. In this case, analogy to a similar package design could provide a reasonable technical basis for the failure thresholds. The application should include references to an evaluation of a similar package design in which failure thresholds are determined, a discussion of the differences that exist between the packages, and defensible yet conservative failure thresholds. For all package applications, ensure that any controls taken to reduce the frequencies of the accident scenarios are appropriately addressed.

Because the specific accident environments encountered will lead to different release characteristics, the accident scenario frequencies must be correlated with the consequences of each specific accident scenario. For example, a package exposed to impact-only ACs may have different release characteristics than a package exposed to an impact and fire environment. The applicant must ensure that the consequences calculated using the methodology described in Chapter VII, Section 3.2.1.1 correspond to the ACs for which quantitative accident frequency information has been developed.

5.3.2.3 Risk. For a specific package, the consequence and frequency data are combined, to identify the risk associated with the transportation. For generic Type A drums or boxes, Table 6-7 shows the bounding consequence and frequency of release. The table identifies essential data used in the evaluations conducted for the identified packages. The evaluation that supports Table 6-7 is presented in Appendix G, Section 2.0. The evaluation must consist of the following: An evaluation report based on the applicable equivalent package requirements of Section 5.2.2 containing the applicable information and formatted as identified in Chapter VII, Section 3.1. This report should document the package's performance; showing both the requirements met and those failed. Where the package does not meet an applicable requirement, discuss the performance provided. To this basic report, add a risk-based evaluation following the methodology identified in Chapter VII, Section 3.2.

6.0 ADDITIONAL SAFETY REQUIREMENTS

Onsite T&P activities are subject to the requirements identified in 10 CFR 830 (see Section 4.0). Implementation of the QA requirements identified in 10 CFR 830, Subpart A, as applied to fissile and Type B packages, are discussed in Chapter VII, Section 3.1.2.9 of the TSD. Requirements of 10 CFR 830, Subpart B, Safety Basis Requirements, for performance of work, safety basis, documented safety analysis, and RL approval are addressed by the TSD as a whole.

The following sections identify the implementation of the requirements for an USQ_T and for TSRs.

6.1 USQ_T

RL, ORP, and each contractor must implement the USQ_T defined in this section. Appendix C, Unreviewed Safety Question Process for Transportation, provides an example USQ_T evaluation process and implementation guide. The example USQ_T evaluation process identifies methods for evaluating changes to onsite T&P activities to ensure they are bounded by the TSD and existing supporting PSSD.

6.1.1 USQ_T Requirements

- (a) RL, ORP, and each contractor responsible for a T&P activity that involves greater than 1 A₂ of radioactive material must establish, implement, and take actions consistent with an USQ_T that meets the requirements of this section. Each contractor must submit for RL approval a procedure for its USQ_T.
- (b) RL, ORP, and each contractor responsible for a T&P activity that involves greater than 1 A₂ of radioactive material must implement the RL-approved USQ_T procedure in situations where there is a:
 - (1) Temporary or permanent change in the T&P activity as described in the existing documented safety analysis;
 - (2) Temporary or permanent change in the procedures as described in the existing documented safety analysis;
 - (3) Test or experiment not described in the existing documented safety analysis; or
 - (4) Potential inadequacy of the documented safety analysis because the analysis potentially may not be bounding or may be otherwise inadequate.
- (c) RL, ORP, and each contractor responsible for a T&P activity that involves greater than 1 A₂ of radioactive material must obtain RL Manager approval prior to taking any action determined to involve a positive USQ_T finding.
- (d) RL, ORP, and each contractor responsible for a T&P activity that involves greater than 1 A₂ of radioactive material must annually submit to RL a summary of the USQ_TD performed since the prior submission.

- (e) If RL, ORP, or a contractor responsible for a T&P activity that involves greater than 1 A₂ of radioactive material discovers or is made aware of a potential inadequacy of the documented safety analysis, it must:
- (1) Take action, as appropriate, to place or maintain the T&P activity in a safe condition until an evaluation of the safety of the situation is completed;
 - (2) Notify RL and other contractors of the situation;
 - (3) Perform an USQ_TD and notify RL and other contractors promptly of the results; and
 - (4) Submit the evaluation of the safety of the situation to RL prior to removing any operational restrictions initiated to meet paragraph (e)(1) of this section.

6.1.2 USQ_T Categorical Exclusions

6.1.2.1 T&P Activity.

- (a) Shipments of Type A packages that contain less than or equal to 1 A₂ of material per package will be categorically excluded.
- (b) Packaging systems that are 10 CFR 71 compliant will be categorically excluded.
- (c) Packaging systems that demonstrate equivalent safety to 10 CFR 71 (Hanford Packaging Standards) will be categorically excluded.
- (d) Packaging systems that move from 10 CFR 71 compliant to Hanford Packaging Standard compliant will be categorically excluded.
- (e) Nonradioactive hazardous material packaging systems will comply with 49 CFR requirements and will be categorically excluded.

6.1.2.2 Inconsequential Changes. Inconsequential changes are purely editorial changes (no technical change) that have no impact on safety, will be categorically excluded. (This applies only to the TSD. Changes to the TSR or SER require RL approval.) Examples of these changes include the following:

- Spelling
- Typographical errors
- Grammatical changes
- Clarifications
- Additional notes or references
- Organizational name changes.

Any change that alters the physical condition of a transport activity is consequential. This includes not only packaging modifications, but also includes procedural steps that direct manipulation of packaging equipment such as manipulation of closures.

Additionally, any change that alters documentation of the physical condition of a transport activity, such as operator readings or observations or their intervals, is consequential. Steps that only verify the physical condition of a transport activity status are inconsequential. In other words, while adding (or deleting) a step to close a package vent would be consequential, adding a step to verify that a package vent was closed would be inconsequential. However, adding (or deleting) a signature block for verification of the package vent position would be consequential, so would a change in the frequency of documenting verification. Changes in frequency that are not documented would be inconsequential.

6.1.3 USQ_TD of Approval

The USQ_T is an important tool to evaluate whether changes affect the safety basis. RL, ORP, and each contractor must use the USQ_T to ensure that the safety basis for a T&P activity that involves greater than 1 A₂ of radioactive material is not undermined by changes in the facility, work performed, associated hazards, or other factors that support the adequacy of the safety basis.

The USQ_T permits RL, ORP, and each contractor to make physical and procedural changes to a T&P activity that involves greater than 1 A₂ of radioactive material and to conduct tests and experiments without prior approval, provided these changes do not cause a positive USQ_T. The USQ_T provides RL, ORP, and each contractor with the flexibility needed to conduct day-to-day operations by requiring only those changes and tests with a potential to impact the safety basis (and therefore the safety of the T&P activity that involves greater than 1 A₂ of radioactive material) be approved by the RL Manager. This allows RL to focus its review on those changes significant to safety. The USQ process helps keep the safety basis current by ensuring appropriate review of and response to situations that might adversely affect the safety basis. Appendix C, Unreviewed Safety Question Process for Transportation, provides the RL Manager's expectations for an USQ_T.

6.2 TSR

RL, ORP, and each contractor responsible for a T&P activity that involves greater than 1 A₂ of radioactive material must implement the following:

- TSRs that are derived from this TSD
- Prior to use, obtain RL Manager approval of any change to TSRs
- Notify RL of any violation of a TSR.

The TSRs are identified in Appendix D, Technical Safety Requirements for Hanford Onsite T&P, for activities involving greater than 1 A₂ of radioactive material.

RL, ORP, and each contractor may take emergency actions that depart from an approved TSR when no actions consistent with the TSR are immediately apparent, and when these actions are needed to protect the workers, public, or environment from imminent and significant harm. Such actions must be approved by a person in authority (e.g., certified shipper) as designated in the TSRs. RL, ORP, and each contractor must report the emergency actions to RL as soon as practicable.

7.0 REFERENCES

- 10 CFR 71, "Packaging and Transportation of Radioactive Material," *Code of Federal Regulations*, as amended.
- 10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," *Code of Federal Regulations*, as amended.
- 10 CFR 830, "Nuclear Safety Requirements," *Code of Federal Regulations*, as amended.
- 49 CFR 100 through 185, "Research and Special Programs Administration, U.S. Department of Transportation," *Code of Federal Regulations*, as amended.
- 49 CFR 350 through 399, "Federal Motor Carrier Safety Administration, U.S. Department of Transportation," *Code of Federal Regulations*, as amended.
- 49 CFR 393, "Parts and Accessories Necessary for Safe Operation," *Code of Federal Regulations*, as amended.
- ANSI N14.5, "Radioactive Materials - Leakage Test on Packages for Shipment," *American National Standards Institute*, La Grange Park, Illinois.
- ANSI N14.6, "Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4,500 kg) or More," *American National Standards Institute*, La Grange Park, Illinois.
- ASTM B-29, "Standard Specification for Refined Lead," *American Society for Testing and Materials*, West Conshohocken, Pennsylvania.
- DOE Order 460.1A, *Packaging and Transportation Safety*, U.S. Department of Energy, Washington, D.C.
- DOE Guide 460.1-1, *The Packaging and Transportation Safety Implementation Guide*, U.S. Department of Energy, Washington, D.C.
- DOE Policy 450.4, *Safety Management System Policy*, U.S. Department of Energy, Washington, D.C.

DOE Policy 450.5, *Line Environment, Safety and Health Oversight*, U.S. Department of Energy, Washington, D.C.

DOE Policy 450.6, *Secretarial Policy Statement Environment, Safety and Health*, U.S. Department of Energy, Washington, D.C.

DOE/RL-92-36, *RL Hoisting and Rigging Manual*, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

DOE-HDBK-3010-94, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities*, Change Notice 1, U. S. Department of Energy, Washington, D.C.

ISO 14001, *Integrated Organization for Standardization, Environmental Management System*, International Standards Organization, Geneva, Switzerland.

MIL-HDBK-695, *Rubber Products: Recommended Shelf Life*, Military Handbook, Aerospace Industries Association, Washington, D.C.

North American Cargo Securement Standard, Canadian Council of Motor Transport Administrators, in collaboration with Canadian and American partners in public and private sectors, Quebec, Canada.

CHAPTER VII. SAFETY ASSESSMENT METHODOLOGY

This chapter identifies the methodologies used for achieving and demonstrating onsite safety equivalent to that accomplished by complying with the DOT regulations. In addition, it identifies specific evaluation methodology and acceptance criteria to apply when demonstrating compliance with the requirements found in Chapter VI for achieving equivalent onsite T&P activity safety.

1.0 METHODOLOGY FOR EQUIVALENT SAFETY

Three different methodologies are used to achieve and demonstrate equivalent safety for onsite T&P activities at the Hanford Site. The three methodologies are compliant with DOE Order 460.1A, Paragraph 4.b, "Onsite Safety Requirements." All three methodologies ensure safety equivalent to that resulting from following the DOT regulations when shipping a package in commerce. The following sections identify the three methodologies.

2.0 COMPLIANCE METHODS

2.1 DOT COMPLIANCE

For all nonradioactive and most radioactive hazardous materials, equivalent safety is achieved by implementing the DOT regulations onsite. All portions of the regulations, other than those identifying the workings of the DOT (e.g., enforcement activities), will be implemented. Equivalent administrative activities including the approval of exemptions and exceptions will be conducted by RL and ORP. By emulating the DOT regulation for onsite hazardous material T&P activities, an equivalent level of safety is achieved. Under this methodology, RL will implement those actions identified for the DOT within the DOT regulations. RL, ORP, and the Hanford Site contractors will implement those actions identified for the user of the DOT regulations. No alternative evaluation or acceptance criteria are needed when using this methodology.

2.2 DOT-EQUIVALENT PACKAGING

For some radioactive materials requiring IP-2, IP-3, Type A, fissile or Type B packagings, the methodology for demonstrating equivalent onsite safety is to apply a mixture of emulating the DOT regulations and using alternate package performance requirements. The alternate package requirements are chosen to produce a package, which when tested or evaluated to onsite conditions, performs equivalent to a DOT package subjected to the DOT/NRC performance requirements. The alternate packaging performance requirements are identified in Chapter VI, Site-Specific Standards, Procedures, and Instructions. The following sections identify the evaluation techniques and acceptance criteria developed for this methodology. Evaluation

techniques, when applied to T&P activities conducted using the requirements identified in Chapter VI, result in achieving and demonstrating compliance with the equivalent safety requirements of DOE Order 460.1A for T&P activities.

Chapter VI contains the Hanford packaging performance standards. Appendix B contains technical arguments or justification and references to documentation that validate the equivalence of packaging performance for onsite and offsite packages under their respective conditions.

2.3 NONEQUIVALENT PACKAGING (Equivalent Radiological and Toxicological Protection)

For some radioactive materials requiring IP-2, IP-3, Type A, fissile or Type B packagings, the methodology for demonstrating equivalent onsite safety is to apply a mixture of emulating the DOT regulations and using alternate package performance requirements. Alternate package requirements are chosen to produce a package, which when tested or evaluated to onsite conditions result in radiological and toxicological safety equivalent to that resulting offsite from full compliance with DOT/NRC regulations governing the loss of material under NCs and for Type B packages, ACs. Alternate packaging performance requirements are identified in Chapter VI. The following sections identify evaluation techniques and acceptance criteria developed for this methodology. Evaluation techniques, when applied to T&P activities conducted using the requirements identified in Chapter VI, result in achieving and demonstrating compliance with the equivalent safety requirements of DOE Order 460.1A for T&P activities.

Chapter VI contains the Hanford packaging performance standards. Appendix G contains the documentation that validates the equivalence of packaging performance for onsite and offsite packages under their respective conditions.

3.0 EVALUATION AND ACCEPTANCE CRITERIA

When the alternate package testing and design criteria presented in Chapter VI are used to design a packaging, the following evaluation methodologies and acceptance criteria must be used to demonstrate compliance. For DOT compliant packagings, the evaluation and acceptance criteria identified in the regulations must be followed.

3.1 DOT-EQUIVALENT PACKAGINGS

3.1.1 Evaluation Requirements and Acceptance Criteria for IP-2, IP-3, and Type A Packagings

This section outlines the elements that are to be addressed for an onsite packaging safety assessment to demonstrate acceptable performance based on the performance tests in Chapter VI,

Section 5.2.3. It also defines the acceptable performance criteria for testing and/or analysis to be applied to demonstrate the packaging provides an equivalent degree of safety to the regulations in maintaining containment/confinement, subcriticality, and shielding. It should be noted that in any onsite packaging safety assessment, claims of adequacy of designs or design methods must be supported by technical bases; i.e., by an appropriate engineering evaluation or description of actual tests. No specific format is identified for IP-2, IP-3, and Type A onsite package safety documentation.

3.1.1.1 General Information. The general information of the onsite packaging safety assessment must include a purpose, detailed packaging description, detailed definition and description of the containment/confinement boundary, contents, and assessment of the general design requirements for packaging (Chapter VI, Section 5.2.4.1).

3.1.1.1(a) Package Description. The package description in the onsite packaging safety assessment must include the following information:

3.1.1.1(a)(1) Detailed description [addressed in paragraph 3.1.1.1(b) and (c) below] of the packaging design in sufficient detail to provide an adequate basis for evaluation.

3.1.1.1(a)(2) Sketches or engineering drawings of the packaging must be prepared to conform to ASME Y14.5, "Geometric Dimensioning and Tolerancing Principles," drafting requirements.

3.1.1.1(a)(3) Detailed description [addressed in paragraph 3.1.1.1(d)] of the contents in sufficient detail to provide an adequate basis for evaluation.

3.1.1.1(a)(4) Detailed description of the loading, unloading, and transport operations of the package.

3.1.1.1(a)(5) Identification of established codes and standards for package design, fabrication, assembly, testing, maintenance, and use.

3.1.1.1(b) Packaging Description. A detailed description of the packaging must include the following:

3.1.1.1(b)(1) General packaging summary, including maximum weight and overall dimensions.

3.1.1.1(b)(2) Containment/confinement features [see paragraph 3.1.1.1(c)].

3.1.1.1(b)(3) Shielding features.

3.1.1.1(b)(4) Structural and other features, including gaps and coolants, for transfer and dissipation of heat.

3.1.1.1(b)(5) Other structural features, including support structures, lifting and tiedown devices, and impact limiters.

3.1.1.1(b)(6) Description of any special fabrication processes not addressed by a nationally recognized or Hanford Site standard.

3.1.1.1(c) Containment/Confinement Boundary Definition and Description. A detailed description of the containment boundary must include the following:

3.1.1.1(c)(1) Definition of the exact boundary of the containment/confinement system. This includes containment/confinement vessel, welds, drain or fill ports, valves, seals, test ports, lids, cover plates, and other closure devices.

3.1.1.1(c)(2) If multiple seals are used for a single closure, the seal defined as the containment system seal must be clearly defined.

3.1.1.1(c)(3) A sketch of the containment/confinement system must be provided.

3.1.1.1(c)(4) For a vented and filtered package, the venting and filtration system must be described in detail. The manufacturer, type, model, rating, and performance characteristics of the filter and filter housing seals must be provided.

3.1.1.1(c)(5) All containment/confinement boundary components must be shown on the sketch or engineering drawings.

3.1.1.1(d) Content Description. A detailed description of the contents must include the following:

3.1.1.1(d)(1) Identification and maximum quantity (radioactivity or mass and specific activity) of the radioactive material.

3.1.1.1(d)(2) Chemical and physical form, including density and moisture content.

3.1.1.1(d)(3) Location and configuration of contents within the packaging.

3.1.1.1(d)(4) Any material subject to chemical, galvanic, or other reaction, including the generation of combustible gases.

3.1.1.1(d)(5) Maximum normal operating and design pressure.

3.1.1.1(d)(6) Maximum weight.

3.1.1.1(d)(7) Maximum decay heat.

3.1.1.1(e) Assessment of General Design Requirements for Packaging. General design requirements for the packaging must be demonstrated to meet the requirements outlined in Chapter VI, Section 5.2.4.1.

3.1.1.2 Structural Evaluation. This section outlines the elements that are to be addressed for an onsite packaging safety assessment to demonstrate acceptable structural performance based on the performance tests defined in Chapter VI, Section 5.2.3 and structural requirements of Chapter VI, Section 5.2.4. The section also defines the acceptable performance criteria for testing and/or analysis to be applied to demonstrate the packaging provides an equivalent degree of safety to the regulations in maintaining containment/confinement and shielding. It should be noted that in any onsite packaging safety assessment, claims of adequacy of designs or design methods must be supported by technical bases; i.e., by an appropriate engineering evaluation or description of actual tests.

3.1.1.2(a) Description of Structural Design. The following information must be provided in the structural description of the packaging:

3.1.1.2(a)(1) Dimensions, tolerances, and materials.

3.1.1.2(a)(2) Weights and center of gravity of the packaging and major subcomponents.

3.1.1.2(a)(3) Maximum weight of contents.

3.1.1.2(a)(4) MNOP.

3.1.1.2(a)(5) Closure system and containment/confinement boundary and its components must be clearly identified and described.

3.1.1.2(a)(6) Description of handling and transport requirements.

3.1.1.2(a)(7) Fabrication methods.

3.1.1.2(a)(8) Codes and standards for the packaging design must be identified, properly applied, and appropriate for the intended purpose.

3.1.1.2(b) Material Properties and Specifications. The following information must be provided and/or demonstrated in describing the materials of construction of this packaging:

3.1.1.2(b)(1) Appropriate specification for the materials of construction must be identified for control of the material.

3.1.1.2(b)(2) Materials and their properties must be consistent with the design codes or standards selected. If no standard is available, adequately documented material properties and specifications for the design and fabrication of the packaging must be provided.

3.1.1.2(b)(3) Materials of the containment/confinement boundary must have sufficient fracture toughness to preclude brittle fracture under NC. In demonstrating fracture toughness of a material, guidance from either ASME BPVC, Section VIII, Division 1; ASME BPVC, Section III; Regulatory Guide 7.11; or Regulatory Guide 7.12 may be used.

3.1.1.2(b)(4) Material properties must be appropriate for the load conditions (e.g., static or dynamic impact loads, hot or cold temperatures, and wet or dry conditions).

3.1.1.2(b)(5) Temperature at which allowable limits are defined must be consistent with minimum and maximum service temperatures.

3.1.1.2(b)(6) Force-deformation properties for impact limiters must be provided and must be based on appropriate test conditions and temperatures.

3.1.1.2(b)(7) Packaging materials and coatings must not produce a significant chemical or galvanic reaction among packaging components, packaging contents, or between the packaging components or packaging content.

3.1.1.2(b)(8) Changes resulting from inleakage of water must be evaluated.

3.1.1.2(b)(9) The possibility of generating hydrogen and other flammable gases must be evaluated.

3.1.1.2(b)(10) The possibility of galvanic interactions and the formation of eutectics must be evaluated for metallic components that may come into physical contact. Such interactions may occur with depleted uranium, lead, or aluminum.

3.1.1.2(b)(11) Any damaging effects of radiation on the packaging materials must be evaluated. These effects may include degradation of seals, sealing material, coatings, adhesives, and structural materials.

3.1.1.2(c)' Fabrication. Information addressing the manufacturing process of construction (e.g., forming, fitting, aligning, welding, brazing, heat treatment, and mechanical joints) must be provided to demonstrate proper packaging fabrication.

3.1.1.2(c)(1) When fabrication specifications are prescribed by appropriate codes and standards, the code or standards must be identified in the engineering drawings.

3.1.1.2(c)(2) Unless otherwise justified, specification of the code or standard used for design must also be used for fabrication.

3.1.1.2(c)(3) For components in which no code or standard is applicable, the specifications on which the evaluation depends must be identified. In addition, the method of control to ensure the specifications are achieved must be described.

3.1.1.2(d) Examination. Information addressing the examination methods and criteria by which the fabrication of the packaging is determined to be acceptable must be provided.

3.1.1.2(d)(1) Unless otherwise justified, the specification of the code or standard used for fabrication must also be used for examination.

3.1.1.2(d)(2) For components in which no code or standard is applicable, the method and acceptance criteria must be provided herein.

3.1.1.2(e) Lifting Devices. When lifting devices are a structural part of the package, their connection with the package body, and the package body in the local area around the lifting devices must be evaluated.

3.1.1.2(e)(1) The lifting devices must be demonstrated to comply with the design requirements of Chapter VI, paragraph 5.2.4.1(b).

3.1.1.2(f) Tiedown Devices. When tiedown devices are a structural part of the package, their connection with the package body and the package body in the local area around the tiedown device must be evaluated.

3.1.1.2(f)(1) The tiedown devices must be demonstrated to comply with the design requirements of Chapter VI, paragraph 5.2.4.2(i).

3.1.1.2(g) General Consideration. The structural evaluation must demonstrate that the package meets the performance test requirements defined in Chapter VI, Section 5.2.3 and the design requirements defined in Chapter VI, Section 5.2.4. The evaluation must also address the following:

- The most limiting initial conditions must be used as defined in Chapter VI, Section 5.2.3.1.
- The evaluation methods are appropriate for the loading conditions considered and accepted practices and precepts are followed.
- The most damaging orientations have been considered. It should be noted that the most damaging orientation for one component may not be the most damaging for another component.

3.1.1.2(g)(1) Evaluation by Testing. If the package performance is demonstrated by testing, the onsite packaging safety assessment must include the following:

3.1.1.2(g)(1)(i) Description of the test surface (e.g., material, mass, dimensions) used for the free drop, penetration, and crush tests. The test surface must be, as a minimum, in accordance with Chapter VI, Section 5.2.3.5. The total mass of the test surface must be greater than 10 times the mass of the package.

3.1.1.2(g)(1)(ii) Description of the steel bar (e.g., material, dimension, orientation) used for the penetration test.

3.1.1.2(g)(1)(iii) The test bar must have sufficient length to cause maximum damage to the package and meet the requirements of Chapter VI, Section 5.2.3.7.

3.1.1.2(g)(1)(iv) The test specimen must have been fabricated using the same materials, methods, and QA as specified in the design. Any differences must be identified and the effects evaluated. Substitutes for the contents should have the same representative weight as the actual contents.

3.1.1.2(g)(1)(v) Selected free drop orientations must be in the orientation for which maximum damage to the feature being tested is expected. These selections must be justified.

3.1.1.2(g)(1)(vi) All test results must be evaluated and their implications interpreted, including interior and exterior damage of the test specimen. Unexpected or unexplained test results indicating possible testing problems or non-reproducible specimen behavior must be described and evaluated.

3.1.1.2(g)(1)(vii) Videos and photographs of tests should be provided as records and maintained available for review.

3.1.1.2(g)(1)(viii) Margins of safety of the package design must be evaluated against the test results.

3.1.1.2(g)(1)(ix) Criteria for evaluating pass/fail for the test conditions must be provided and used as a comparison with the test results.

3.1.1.2(g)(2) Evaluation by Analysis. If the package performance is demonstrated by analysis, the onsite packaging safety assessment must include the following:

3.1.1.2(g)(2)(i) A clear description of calculations and all assumptions must be provided.

3.1.1.2(g)(2)(ii) The response of the package to loads, in terms of stress and strain to the components and structural members must be provided, and the structural stability of individual members must be demonstrated.

3.1.1.2(g)(2)(iii) The analytical methods must evaluate impact at any angle, rigid-body rotation, and secondary impacts (slapdown).

3.1.1.2(g)(2)(iv) When used, computer codes must be appropriately used, verified, and benchmarked.

3.1.1.2(g)(2)(v) When using quasi-static analysis techniques, appropriate dynamic amplification factors must be developed and applied.

3.1.1.2(g)(2)(vi) Material properties and models must be appropriate for the load combinations evaluated. Material properties must be consistent with the analytical methods. The strain rate at which the properties are given must be justified.

3.1.1.2(g)(2)(vii) A summary of results must be provided which compare results of the analyses with the acceptable performance criteria established below in paragraph 3.1.1.2(g)(3).

3.1.1.2(g)(3) Acceptable Performance Criteria for Evaluation by Analysis. The following methods may be used for evaluation. Alternate benchmarked methods may be used provided they meet requirements identified in paragraph 3.1.1.2(g)(2).

NC acceptable packaging performance shall be assessed in accordance with the ASME BPVC, Section III, Subsection NE requirements for Service Level A and C Limits. Combined stress intensities from temperature, pressure, and free drop are evaluated against the requirements defined for Service Level C Limits. Stress intensities from all other load combinations are to be evaluated against the requirements defined for Service Level A Limits. For structural evaluation purposes, maintenance of containment/confinement is defined as the ability of the packaging systems to sustain applied loading without exceeding ASME allowable stress intensity values. In addition, at closure locations, the seals must be demonstrated to remain sufficiently compressed under NC loads to maintain containment/confinement.

Onsite equivalent packaging requirements also allow plastic deformation of the packaging when it is demonstrated by either testing or analysis that containment/confinement and shielding are maintained under NC. If plastic analysis methods are used to demonstrate maintenance of containment/confinement and shielding, the loads cannot be superimposed. Because ASME BPVC criteria are inappropriate for plastic analysis, packaging performance will be assessed by the combination of the following two methods.

The first and simplest acceptance criteria screening is strain energy density (SED). The maximum SED of the material is also known as toughness and at material failure can be conservatively estimated from the results of uniaxial tensile test data. A conservative estimate of SED at material failure is as follows:

$$SED = \frac{(\sigma_y + \sigma_u)}{2} \epsilon_f$$

where: σ_y = yield strength of the material
 σ_u = ultimate strength of the material
 ϵ_f = ultimate tensile elongation.

These parameters are available from engineering stress-strain data. Use of the SED also minimizes concerns over unavoidable material variations such as heat-to-heat variations in yield strength and strain hardening behavior.

To ensure against ductile failure of the material, the maximum SED on the package is limited to 70% of the SED failure of the material or material toughness. This is in keeping with the intent of Regulatory Guide 7.6 to prevent large unconfined strain plasticity and system failure due to loss of dimensional stability at key locations. The limit is based on applying the 70% Service Level D stress limit for material loading specified in ASME BPVC, Section III, Appendices to the ultimate elongation specified in ASME BPVC, Section II material properties at operating

temperatures. This results in the SED limit of 70% of the maximum material toughness at operating temperature. The allowable SED in equation form is defined as:

This is compared against the von Mises stress times, the cumulative strain at the location of

$$SED_{all} = 0.70 \frac{(\sigma_y + \sigma_u)}{2} \epsilon_f$$

evaluation. It is used for global assessment of package performance under dynamic impact loading conditions and identifying high loading areas for detail assessment. Consequently, as a second screening at critical location and in areas of high SED values, a more comprehensive measure of performance is used to assess equivalent cumulative strains against a limiting value.

The most comprehensive criteria demonstrating acceptable cumulative strain levels to prevent material rupture are based on establishing limits on the maximum equivalent cumulative strain. The strain limits are developed based on Manjoine (1983), *Transaction of the ASME, Journal of Pressure Vessel Technology*, that the tensile elongation measured in a tension test can be adjusted to predict multi-axial failure strain by the following relation:

$$\frac{\epsilon}{\epsilon_t} = 2^{(1-TF)} \quad \text{for } TF \geq 0 \quad \text{and} \quad \frac{\epsilon}{\epsilon_t} = 2 \quad \text{for } TF \leq 0$$

where: ϵ = the effective strain at failure

ϵ_t = the ultimate tensile elongation of the material from tension tests, and

TF = the triaxiality factor.

The triaxiality factor is defined by the principal stresses ($\sigma_1, \sigma_2, \sigma_3$) as:

$$TF = \frac{\sqrt{2}(\sigma_1 + \sigma_2 + \sigma_3)}{\sqrt{[(\sigma_1 - \sigma_2)^2 + (\sigma_2 - \sigma_3)^2 + (\sigma_3 - \sigma_1)^2]}}$$

Based on the ductility versus triaxiality factor curve presented in Manjoine (1983), the following conservative failure strain limits, termed the Triaxial Strain Limit (TSL), can be established for any ductile material under multi-axial loading. Additionally, in keeping with the intent of Regulatory Guide 7.6 to prevent large unconfined strain plasticity and system failure due to loss of dimensional stability at key locations, the ultimate strain is limited to 70% of the reported ultimate tensile elongation of the material. In addition, welds do not have the ductility of the base metal and the presence of residual weld stresses, in weld areas the TSL is one-half the limit established above. Consequently, to demonstrate acceptable performance in non-weld areas, the TSL is established as equal to the triaxiality factor times 70% of the ultimate tensile elongation of the material. The TSL in weld areas is established as one-half times the triaxiality factor times 70% of the ultimate tensile elongation of the material.

In equation form, the TSLs are:

$$\text{TSL} = 0.70 \times \text{TF} \times e_t \text{ in nonweld areas}$$

and

$$\text{TSL} = 0.35 \times \text{TF} \times e_t \text{ in weld areas}$$

To demonstrate acceptable performance of the packaging, the cumulative strain at all locations must remain below the established TSL at the location of evaluation. Also, at closure locations, the cumulative strains must be demonstrated not to result in loss of containment or decompression of the seals by flange distortion or closure bolt elongation.

3.1.1.2(h) Structural Evaluation of NC. Evaluation of the package under NC is based on the effects of tests and conditions specified in Chapter VI, Section 5.2.3.1. These tests must not result in any decrease in package effectiveness in maintaining containment/confinement and shielding. Under these test conditions, ambient air temperature before and after the test must remain constant at that specified in Chapter VI, paragraph 5.2.3.1(b). Initial internal pressure on containment components must be considered normal operating pressure, unless a lower internal pressure consistent with the selected ambient temperature is more severe.

3.1.1.2(h)(1) Heat Condition. Adequacy of the packaging under heat conditions must be demonstrated by the following evaluations at the hot temperature specified in Chapter VI, paragraph 5.2.4.2(c).

3.1.1.2(h)(1)(i) Maximum temperatures under heat conditions must be consistent with the thermal section [Chapter VI, paragraph 5.2.4.2(c)].

3.1.1.2(h)(1)(ii) Normal operating pressure, in combination with the maximum internal heat load and any residual fabrication stresses, must be evaluated.

3.1.1.2(h)(1)(iii) Any differential thermal expansion and possible geometric interference must be evaluated.

3.1.1.2(h)(1)(iv) The containment/confinement boundary must be evaluated.

3.1.1.2(h)(2) Cold Condition. Adequacy of the packaging under cold conditions must be demonstrated by the following evaluations at the cold temperature specified in Chapter VI, paragraph 5.2.4.2(c):

3.1.1.2(h)(2)(i) Minimum temperatures with no internal heat loading must be consistent with the Thermal section (Chapter VI, paragraph 5.2.4.2(c)).

3.1.1.2(h)(2)(ii) The evaluation must consider minimum internal pressure with minimum internal heat load and any residual fabrication stresses.

3.1.1.2(h)(2)(iii) Any differential thermal expansion and possible geometric interference must be evaluated.

3.1.1.2(h)(2)(iv) Evaluate effects of the possible freezing of liquids.

3.1.1.2(h)(2)(v) The containment/confinement boundary must be evaluated.

3.1.1.2(h)(3) Reduced External Pressure. Adequacy of the packaging under reduced external pressure conditions specified in Chapter VI, Section 5.2.3.3 must be demonstrated by the following evaluations:

3.1.1.2(h)(3)(i) The packaging must be evaluated to demonstrate the containment components are capable of sustaining the differential pressure loading specified in Chapter VI, paragraph 5.2.4.2(f) without loss of containment capability. The evaluations must consider the greatest possible pressure differential between the inside and outside of the containment components.

3.1.1.2(h)(3)(ii) The containment/confinement boundary must be evaluated.

3.1.1.2(h)(4) Vibration. Adequacy of the packaging under the effects of vibration normally incident to transport tests as specified in Chapter VI, Section 5.2.3.9 must be demonstrated by the following evaluations:

3.1.1.2(h)(4)(i) Fatigue must be evaluated at highly stressed components, considering the combined stresses due to vibration, temperature, and pressure loads.

3.1.1.2(h)(4)(ii) If closure bolts are reused, the bolts must be evaluated for fatigue and the bolt preload must be included.

3.1.1.2(h)(4)(iii) Evaluations must include demonstration that resonant vibration conditions, which may cause rapid fatigue damage, are not present in the packaging components.

3.1.1.2(h)(4)(iv) Effects of package internal components and contents must be considered in the evaluations.

3.1.1.2(h)(4)(v) The containment/confinement boundary must be evaluated.

3.1.1.2(h)(4)(vi) Guidance for vibrational loading parameters is provided in ANSI N14.23, "(Draft) Design Basis/Resistance to Shock/Vibration," and ASTM D4169, "Standard Practice for Performance Testing of Shipping Containers and Systems."

3.1.1.2(h)(5) Water Spray. Adequacy of the packaging under the effects of the water spray test as specified in Chapter VI, Section 5.2.3.4 must be demonstrated by evaluating the effects on material properties of the package. The evaluation must also demonstrate that the water spray results in no loss of radioactive material from the package or increase in radiation levels at the surface of the package.

3.1.1.2(h)(6) Free Drop. Adequacy of the packaging under the effects of the free drop tests as specified in Chapter VI, Section 5.2.3.5 must be demonstrated by the following evaluations:

3.1.1.2(h)(6)(i) Design of the packaging must be evaluated for the structural response for the specified free drop height in combination with internal pressure.

3.1.1.2(h)(6)(ii) Closure system design with the combined effects of free drop impact force, internal pressure, gasket compression force, and bolt preload must be evaluated.

3.1.1.2(h)(6)(iii) Other important components, such as the port-cover, port-cover plates, and shielding enclosures must be demonstrated not to decrease in effectiveness when subjected to the combination of impact force and internal pressure.

3.1.1.2(h)(6)(iv) The containment/confinement boundary must be evaluated for the combination of impact force and internal pressure.

3.1.1.2(h)(6)(v) The closure system design seals must be demonstrated not to decompress from bolt stretch or flange deflection to the extent that containment/confinement is lost under the combined loadings.

3.1.1.2(h)(7) Compression (Stacking). Adequacy of the packaging under the effects of the stacking test as specified in Chapter VI, Section 5.2.3.6 must be demonstrated by evaluating the structural response of the package to the static loads specified.

3.1.1.2(h)(8) Penetration. Adequacy of the packaging under the effects of the penetration test as specified in Chapter VI, Section 5.2.3.7 must be demonstrated by evaluating the effects of the package at the most vulnerable location.

3.1.2 Evaluation Requirements and Acceptance Criteria for Fissile and Type B Packagings

This section describes elements that are to be addressed for an onsite PSAA to demonstrate acceptable performance based on performance tests and design requirements presented in Chapter VI, Sections 5.2.5 and 5.2.6. It also defines acceptable performance criteria for testing and/or analysis to be applied to demonstrate the packaging provides an equivalent degree of safety to the regulations in maintaining containment/ confinement, subcriticality, and shielding. It should be noted that in any onsite PSAA, claims of adequacy of designs or design methods must be supported by technical bases; i.e., by an appropriate engineering evaluation or description of actual tests. The techniques and methodology provided in this section must be used to demonstrate compliance with the requirements from Chapter VI, Sections 5.2.5 and 5.2.6.

Guidance for formatting and preparing the onsite PSAA report is provided in Appendix E, Package Safety Analysis Assessment Report Contents and Format. The PSAA report is divided into 11 chapters. The contents of the chapters correspond to contents identified in Sections 3.1.2.1 through 3.1.2.11 below. An example of the application of structural design and evaluation requirements to a shielded drum is provided in Appendix F, Example Application of Package Structural Design and Evaluation Requirements.

3.1.2.1 General Information. General information of the onsite PSAA must include purpose, detailed packaging description, detailed definition and description of the containment/confinement boundary, contents, and assessment of the general design requirements for packaging (Chapter VI, Section 5.2.6.1).

3.1.2.1(a) Package Description. The package description in onsite PSAAs must include the following information:

3.1.2.1(a)(1) A detailed description [addressed in paragraph 3.1.2.1(b) and (c) below] of the packaging design in sufficient detail to provide an adequate basis for evaluation.

3.1.2.1(a)(2) Engineering drawings of the packaging must be prepared in accordance with NUREG/CR-5502.

3.1.2.1(a)(3) A detailed description [addressed in paragraph 3.1.2.1(d)] of the contents in sufficient detail to provide an adequate basis for evaluation.

3.1.2.1(a)(4) A detailed description of loading, unloading, and transport operations of the package.

3.1.2.1(a)(5) Identification of established codes and standards for package design, fabrication, assembly, testing, maintenance, and use.

3.1.2.1(b) Packaging Description. A detailed description of the packaging must include the following:

3.1.2.1(b)(1) General packaging summary, including maximum weight and overall dimensions.

3.1.2.1(b)(2) Containment/confinement features [see paragraph 3.1.2.1(c)].

3.1.2.1(b)(3) Shielding features.

3.1.2.1(b)(4) Criticality control features, including neutron poisons, moderators, and spacers.

3.1.2.1(b)(5) Structural and other features, including gaps and coolants, for transfer and dissipation of heat.

3.1.2.1(b)(6) Other structural features, including support structures, lifting and tiedown devices, and impact limiters.

3.1.2.1(b)(7) A description of any special fabrication processes not addressed by a nationally recognized or Hanford Site standard.

3.1.2.1(c) Containment/Confinement Boundary Definition and Description. A detailed description of the containment boundary must include the following:

3.1.2.1(c)(1) Definition of the exact boundary of the containment/confinement system. This includes containment/confinement vessel, welds, drain or fill ports, valves, seals, test ports, lids, cover plates, and other closure devices.

3.1.2.1(c)(2) If multiple seals are used for a single closure, the seal defined as the containment system seal must be clearly defined.

3.1.2.1(c)(3) A sketch of the containment/confinement system must be provided.

3.1.2.1(c)(4) For vented and filtered packages, the venting and filtration system must be described in detail. The manufacturer, type, model, rating, and performance characteristics of the filter and filter housing seals must be provided.

3.1.2.1(c)(5) All containment/confinement boundary components must be shown on the engineering drawings.

3.1.2.1(d) Content Description. A detailed description of the contents must include the following:

3.1.2.1(d)(1) Identification and maximum quantity (radioactivity or mass) of the radioactive material.

3.1.2.1(d)(2) Identification and maximum quantity of fissile material.

3.1.2.1(d)(3) Chemical and physical form, including density and moisture content, and the presence of other moderating constituents.

3.1.2.1(d)(4) Location and configuration of contents within the packaging.

3.2.2.1(d)(5) Identification and quantity of nonfissile materials used as neutron absorbers or moderators.

3.1.2.1(d)(6) Any material subject to chemical, galvanic, or other reaction, including generation of combustible gases.

3.1.2.1(d)(7) Maximum normal operating and design pressure.

3.1.2.1(d)(8) Maximum weight.

3.1.2.1(d)(9) Maximum decay heat.

3.1.2.1(e) Assessment of General Design Requirements for Packaging. General design requirements for packaging must be demonstrated to meet the requirements outlined in Chapter VI, Section 5.2.6.1.

3.1.2.2 Structural Evaluation. This section outlines the elements that are to be addressed for an onsite PSAA to demonstrate acceptable structural performance based on the performance tests defined in Chapter VI, Section 5.2.5 and structural requirements of Chapter VI, Section 5.2.6. The section also defines acceptable performance criteria for testing and/or analysis to be applied to demonstrate the packaging provides an equivalent degree of safety to the regulations in maintaining containment/confinement, subcriticality, and shielding. It should be noted that in any onsite PSAA, claims of adequacy of designs or design methods must be supported by technical bases; i.e., by an appropriate engineering evaluation or description of actual tests.

3.1.2.2(a) Description of Structural Design. The following information must be provided in the structural description of the packaging:

3.1.2.2(a)(1) Dimensions, tolerances, and materials.

3.1.2.2(a)(2) Weights and center of gravity of the packaging and major subcomponents.

3.1.2.2(a)(3) Maximum weight of contents.

3.1.2.2(a)(4) MNOP.

3.1.2.2(a)(5) Closure system and containment/confinement boundary and its components must be clearly identified and described.

3.1.2.2(a)(6) Description of handling and transport requirements.

3.1.2.2(a)(7) Fabrication methods.

3.1.2.2(a)(8) Codes and standards for the packaging design must be identified and must be properly applied and appropriate for the intended purpose.

3.1.2.2(b) Material Properties and Specifications. The following information must be provided and/or demonstrated in describing the materials of construction of this packaging:

3.1.2.2(b)(1) Appropriate specification for the materials of construction must be identified for control of the material.

3.1.2.2(b)(2) Materials and their properties must be consistent with the design codes or standards selected. If no standard is available, adequately documented material properties and specifications for design and fabrication of the packaging must be provided.

3.1.2.2(b)(3) Materials of the containment/confinement boundary must have sufficient fracture toughness to preclude brittle fracture under NC and AC. In demonstrating fracture toughness of

a material, guidance from ASME BPVC, Section VIII, Division 1; ASME BPVC, Section III; Regulatory Guide 7.11; or Regulatory Guide 7.12 may be used.

3.1.2.2(b)(4) Material properties must be appropriate for the load conditions (e.g., static or dynamic impact loads, hot or cold temperatures, and wet or dry conditions).

3.1.2.2(b)(5) Temperature at which allowable limits are defined must be consistent with minimum and maximum service temperatures.

3.1.2.2(b)(6) Force-deformation properties for impact limiters must be provided and must be based on appropriate test conditions and temperatures.

3.1.2.2(b)(7) Packaging materials and coatings must not produce a significant chemical or galvanic reaction among packaging components, among packaging contents, or between the packaging components or packaging content.

3.1.2.2(b)(8) Reactions resulting from the inleakage of water must be evaluated.

3.1.2.2(b)(9) The possibility of generating hydrogen and other flammable gases must be evaluated.

3.1.2.2(b)(10) The possibility of galvanic interactions and the formation of eutectics must be evaluated for metallic components that may come into physical contact. Such interactions may occur with depleted uranium, lead, or aluminum.

3.1.2.2(b)(11) Any damaging effects of radiation on the packaging materials must be evaluated. These effects may include degradation of seals, sealing material, coatings, adhesives, and structural materials.

3.1.2.2(c) Fabrication. Information addressing the manufacturing process of construction (e.g., forming, fitting, aligning, welding, brazing, heat treatment, and mechanical joints) must be provided to demonstrate proper packaging fabrication.

3.1.2.2(c)(1) When fabrication specifications are prescribed by appropriate codes and standards, the code or standards must be identified in engineering drawings.

3.1.2.2(c)(2) Unless otherwise justified, the specification of the code or standard used for design must also be used for fabrication.

3.1.2.2(c)(3) For components in which no code or standards are applicable, the specifications on which the evaluation depends must be identified. In addition, the method of control to ensure specifications are achieved must be described.

3.1.2.2(d) Examination. Information addressing examination methods and criteria by which fabrication of the packaging is determined to be acceptable must be provided.

3.1.2.2(d)(1) Unless otherwise justified, the specification of the code or standard used for fabrication must also be used for examination.

3.1.2.2(d)(2) For components in which no code or standards are applicable, the method and acceptance criteria must be provided herein and a summary provided in Section 3.1.2.11.

3.1.2.2(e) Lifting Devices. When lifting devices are a structural part of the package, their connection with the package body, and the package body in the local area around the lifting devices, must be evaluated.

3.1.2.2(e)(1) Lifting devices must be demonstrated to comply with the design requirements of Chapter VI, Section 5.2.6.2.

3.1.2.2(f) Tiedown Devices. When tiedown devices are a structural part of the package, their connection with the package body and the package body in the local area around the tiedown device must be evaluated.

3.1.2.2(f)(1) Tiedown devices must be demonstrated to comply with the design requirements of Chapter VI, Section 5.2.6.3.

3.1.2.2(g) General Consideration. The structural evaluation must demonstrate the package meets the performance test requirements defined in Chapter VI, Section 5.2.5 and design requirements defined in Chapter VI, Section 5.2.6. The evaluation must also address the following:

- The most limiting initial conditions must be used as defined in Table 6-5 of Chapter VI, paragraph 5.2.6.10(a)
- The evaluation methods are appropriate for the loading conditions considered and follow accepted practices and precepts
- The most damaging orientations have been considered. It should be noted that the most damaging orientation for one component may not be the most damaging for another component.

3.1.2.2(g)(1) Evaluation by Testing. If the package performance is demonstrated by testing, the onsite PSAA must include the following:

- Description of the test surface (e.g., material, mass, dimensions) used for the free drop and crush tests. As a minimum, the test surface must be in accordance with Chapter VI, paragraph 5.2.5.1(c)(7). The total mass of the test surface must be at least 10 times the mass of the package.
- Description of the steel bar (e.g., material, dimension, orientation, method of mounting) used for the puncture test. The test bar must have sufficient length to cause maximum damage to the package and meet the requirements of Chapter VI, paragraph 5.2.5.2(c)(3).

- The test specimen must have been fabricated using the same materials, methods, and QA as specified in the design. Any differences must be identified and the effects evaluated. Substitutes for the contents should have the same representative weight as the actual contents.
- Of the selected free drop orientations, one must be in the orientation for which maximum damage is expected. This selection must be justified.
- All test results must be evaluated and their implications interpreted, including interior and exterior damage, for the test specimen. Unexpected or unexplained test results indicating possible testing problems or non-reproducible specimen behavior must be described and evaluated.
- Videos and photographs of the tests must be provided as records and maintained available for review for the life of the package.
- Criteria for evaluating pass/fail for the test conditions must be provided and used as a comparison with the test results.

3.1.2.2(g)(2) Evaluation by Analysis. If the package performance is demonstrated by analysis, the onsite PSAA must include the following:

- A clear description of calculations and all assumptions must be provided.
- Response of the package to loads, in terms of stress and strain to the components and structural members, must be provided and the structural stability of individual members must be demonstrated.
- Analytical methods must evaluate impact at any angle, rigid-body rotation, and secondary impacts (slapdown).
- Computer codes must be appropriately used, verified, and benchmarked.
- When using quasi-static analysis techniques, appropriate dynamic amplification factors must be developed and applied.
- Material properties and models must be appropriate for the load combinations evaluated. Material properties must be consistent with the analytical methods. The strain rate at which the properties are given must be justified.
- A summary of results must be provided which compare results of the analyses with the acceptable performance criteria established in paragraph 3.1.2.2(g)(3).

3.1.2.2(g)(3) Acceptable Performance Criteria for Evaluation by Analysis. NC acceptable packaging performance will be assessed in accordance with the ASME BPVC, Section III, Subsection NE requirements for Service Level A and C Limits. The combined stress intensities

from temperature, pressure, and free drop are evaluated against the requirements defined for Service Level C Limits. Stress intensities from all other load combinations are to be evaluated against the requirements defined for Service Level A Limits. For structural evaluation purposes, maintenance of containment/confinement is defined as the ability of the packaging system to sustain the applied loading without exceeding ASME allowable stress intensity values. Also, at closure locations, the seals must be demonstrated to remain sufficiently compressed under NC loads as to maintain containment/confinement.

Under AC, the package must be subjected to the sequence of performance tests in the following order: free drop, crush (when applicable), puncture, fire, and immersion (fissile packages). Where the package loading is linear elastic, loads can be combined by superimposition and the performance assessed to the requirements of ASME BPVC, Subsection NE, Service Level D Limits.

Onsite equivalent package requirements also allow plastic deformation of the packaging when it is demonstrated by either testing or analysis that containment/confinement, shielding, and subcriticality are maintained under AC. When packaging performance is demonstrated by elastic plastic analysis, the loads are required to be applied in the sequence as specified in Chapter VI, paragraph 5.2.5.2(c) to assess cumulative damage of the packaging. If plastic analysis methods are used to demonstrate maintenance of containment/confinement, shielding, and subcriticality, the loads cannot be superimposed and cumulative damage must be assessed after application of all the loads to the worst-case locations. Because ASME BPVC criteria are inappropriate for plastic analysis, packaging performance will be assessed by the combination of the following two methods:

The first and simplest acceptance criteria screening is SED. The maximum SED of the material is also known as toughness and at material failure can be conservatively estimated from the results of uniaxial tensile test data. A conservative estimate of SED at material failure is as follows:

$$SED = \frac{(\sigma_y + \sigma_u)}{2} \epsilon_f$$

where: σ_y = yield strength of the material
 σ_u = ultimate strength of the material
 ϵ_f = ultimate tensile elongation.

These parameters are available from engineering stress-strain data. Use of the SED also minimizes concerns over unavoidable material variations such as heat-to-heat variations in yield strength and strain hardening behavior.

To ensure against ductile failure of the material, the maximum SED on the package is limited to 70% the SED to failure of the material or material toughness. This is in keeping with the intent of the Regulatory Guide 7.6 to prevent large unconfined strain plasticity and system failure due to loss of dimensional stability at key locations. The limit is based on applying the 70% Service

Level D stress limit for material loading, specified in ASME BPVC, Section III, Appendices to the ultimate elongation specified in ASME BPVC, Section II material properties at operating temperatures. This results in the SED limit of 70% of the maximum material toughness at operating temperature. The allowable SED in equation form is defined as:

$$SED_{all} = 0.70 \frac{(\sigma_y + \sigma_u)}{2} \varepsilon_f$$

This is compared against the von Mises stress times the cumulative strain at the location of evaluation. It is used for global assessment of package performance under dynamic impact loading conditions and identifying high loading areas for detail assessment. Consequently, as a second screening at critical locations and in areas of high SED values, a more comprehensive measure of performance is used to assess equivalent cumulative strains against a limiting value.

The most comprehensive criteria demonstrating acceptable cumulative strain levels to prevent material rupture are based on establishing limits on the maximum equivalent cumulative strain. The strain limits are developed based on Manjoine (1983) observation that the tensile elongation measured in a tension test can be adjusted to predict multi-axial failure strain by the following relation:

$$\frac{\varepsilon}{e_t} = 2^{(1-TF)} \quad \text{for } TF \geq 0 \quad \text{and} \quad \frac{\varepsilon}{e_t} = 2 \quad \text{for } TF \leq 0$$

where: ε = the effective strain at failure

e_t = the ultimate tensile elongation of the material from tension tests

TF = the triaxiality factor.

The triaxiality factor is defined by the principal stresses ($\sigma_1, \sigma_2, \sigma_3$) as:

$$TF = \frac{\sqrt{2}(\sigma_1 + \sigma_2 + \sigma_3)}{\sqrt{[(\sigma_1 - \sigma_2)^2 + (\sigma_2 - \sigma_3)^2 + (\sigma_3 - \sigma_1)^2]}}$$

Based on the ductility versus triaxiality factor curve presented in Manjoine (1983), the following conservative failure strain limits, termed the TSL, can be established for any ductile material under multi-axial loading. In addition, in keeping with the intent of the Regulatory Guide 7.6 to prevent large unconfined strain plasticity and system failure due to loss of dimensional stability at key locations, the ultimate strain is limited to 70% of the reported ultimate tensile elongation of the material. Also, welds do not have the ductility of the base metal and the presence of residual weld stresses, in weld areas the TSL is half the limit established above. Consequently, to demonstrate acceptable performance in non-weld areas, the TSL is established as equal to the triaxiality factor times 70% of the ultimate tensile elongation of the material. The TSL in weld

areas is established as one-half times the triaxiality factor times 70% of the ultimate tensile elongation of the material. In equation form, the TSLs are:

$$\text{TSL} = 0.70 \times \text{TF} \times e_t \text{ in nonweld areas}$$

and

$$\text{TSL} = 0.35 \times \text{TF} \times e_t \text{ in weld areas}$$

To demonstrate acceptable performance of the packaging, the cumulative strain at all locations must remain below the established TSL at the location of evaluation. Also, at closure locations, the cumulative strains must be demonstrated not to result in loss of containment or decompression of the seals by flange distortion or closure bolt elongation.

3.1.2.2(g)(4) Pressure Loading. The following pressure evaluations must be provided in the onsite PSAA to demonstrate the structural integrity of the packaging containment components under pressure:

- As stated in Chapter VI, Section 5.2.6.12 the packaging must be demonstrated to sustain a pressure of 150% of the rated MNOP.
- The package must be evaluated to demonstrate that the containment components are capable of sustaining the differential pressure loading specified in Chapter VI, paragraphs 5.2.5.1(c)(3) and 5.2.5.1(c)(4) without loss of containment capability.
- The evaluations must consider the greatest possible pressure differential between the inside and outside of the containment components.

3.1.2.2(h) Structural Evaluation of NC. Evaluation of the package under NC is based on the effects of tests and conditions specified in Chapter VI, Section 5.2.5.1. These tests must not result in any decrease in package effectiveness in maintaining containment/confinement, shielding, and subcriticality. Under these test conditions, the ambient air temperature before and after the test must remain constant at that value between the specified temperatures in Chapter VI, paragraphs 5.2.5.1(c)(1) and 5.2.5.1(c)(2) which is the most unfavorable for the feature being evaluated. Initial internal pressure on the containment components must be considered to be MNOP, unless a lower internal pressure consistent with the selected ambient temperature is more severe.

3.1.2.2(h)(1) Heat Condition. The adequacy of the package under heat conditions must be demonstrated by the following evaluations at the hot temperature specified in Chapter VI, paragraph 5.2.5.1(c)(1):

- The maximum temperatures under heat conditions must be consistent with the thermal evaluation section (3.1.2.3).
- The MNOP, in combination with the maximum internal heat load and any residual fabrication stresses, must be evaluated.

- Any differential thermal expansion and possible geometric interference must be evaluated.
- The containment/confinement boundary must be evaluated for the load combinations specified in Chapter VI, paragraph 5.2.6.10(a), and Table 6-6.

3.1.2.2(h)(2) Cold Condition. The adequacy of the package under cold conditions must be demonstrated by the following evaluations at the cold temperature specified in Chapter VI, paragraph 5.2.5.1(c)(2):

- The minimum temperatures with no internal heat loading must be consistent with the thermal evaluation section (3.1.2.3).
- The evaluation must consider the minimum internal pressure with the minimum internal heat load and any residual fabrication stresses.
- Any differential thermal expansion and possible geometric interference must be evaluated.
- Evaluate the effects of the possible freezing of liquids.
- The containment/confinement boundary must be evaluated for the load combinations specified in Chapter VI, paragraph 5.2.6.10(a), and Table 6-6.

3.1.2.2(h)(3) Reduced External Pressure. The adequacy of the package under reduced external pressure conditions specified in Chapter VI, paragraph 5.2.5.1(c)(3) must be demonstrated by the following evaluations:

- The greatest possible pressure difference between the inside and outside of the containment components must be evaluated.
- The containment/confinement boundary must be evaluated for the load combinations specified in Chapter VI, paragraph 5.2.6.10(a), and Table 6-6.

3.1.2.2(h)(4) Increased External Pressure. Adequacy of the package under increased external pressure conditions specified in Chapter VI, paragraph 5.2.5.1(c)(4) must be demonstrated by the following evaluations:

- The evaluation must consider the loading condition in combination with the minimum internal pressure.
- The greatest possible pressure difference between the inside and outside of the containment components must be evaluated.
- The possibility of buckling must be evaluated.
- The containment/confinement boundary must be evaluated for the load combinations specified in Chapter VI, paragraph 5.2.6.10(a), and Table 6-6.

3.1.2.2(h)(5) Vibration. Adequacy of the package under the effects of vibration normally incident to transport tests as specified in Chapter VI, paragraph 5.2.5.1(c)(5) must be demonstrated by the following evaluations:

- Fatigue must be evaluated at highly stressed components, considering the combined stresses due to vibration, temperature, and pressure loads.
- If the closure bolts are reused, the bolts must be evaluated for fatigue and the bolt preload must be included.
- Evaluations must include a demonstration that resonant vibration conditions, which may cause rapid fatigue damage, are not present in the package components.
- The effects of package internal components and contents must be considered in the evaluations.
- The containment/confinement boundary must be evaluated for the load combinations specified in Chapter VI, paragraph 5.2.6.10(a), and Table 6-6.
- Guidance for vibrational loading parameters is provided in ANSI N14.23 and ASTM D4169.

3.1.2.2(h)(6) Water Spray. Adequacy of the package under the effects of water spray test as specified in Chapter VI, paragraph 5.2.5.1(c)(6) must be demonstrated by evaluating the effects on the material properties of the package.

3.1.2.2(h)(7) Free Drop. Adequacy of the package under the effects of the free drop tests as specified in Chapter VI, paragraph 5.2.5.1(c)(7) must be demonstrated by the following evaluations:

- Design of the package must be evaluated for the structural response for the specified free drop height in combination with internal pressure, thermal stress, and residual stresses.
- The closure system design with the combined effects of the free drop impact force, internal pressure, thermal stress, residual stress, O-ring compression force, and bolt preload must be evaluated.
- Other important components (such as the port-cover, port-cover plates, and shielding enclosures) must be demonstrated not to decrease in effectiveness when subjected to the combination of impact force, internal pressure, residual stress, and thermal stress.
- The containment/confinement boundary must be evaluated for the load combinations specified in Chapter VI, paragraph 5.2.6.10(a), and Table 6-6.
- The closure system design seals must be demonstrated not to decompress from bolt stretch or flange deflection to the extent that containment/confinement is lost under the combined loadings.

3.1.2.2(h)(8) Compression. Adequacy of the package under the effects of the compression test as specified in Chapter VI, paragraph 5.2.5.1(c)(8) must be demonstrated by evaluating the structural response of the package to the static loads specified.

3.1.2.2(h)(9) Penetration. Adequacy of the package under the effects of the penetration test as specified in Chapter VI, paragraph 5.2.5.1(c)(9) must be demonstrated by evaluating the effects of the package at most the vulnerable location.

3.1.2.2(i) Structural Evaluation of AC. Evaluation of the package under AC is based on the sequential application of tests and conditions specified in Chapter VI, Section 5.2.5.2, in the order indicated, to determine their cumulative effect on the package. The evaluation of the ability of a package to withstand any one of these tests must consider the damage that resulted from the previous test. In addition, the NC tests must not affect the package's ability to withstand the AC tests. Under these test conditions, the ambient air temperature before and after the test must remain constant as specified in Chapter VI, paragraphs 5.2.5.1(c)(1) and 5.2.5.1(c)(2), which is the most unfavorable for the feature being evaluated. The initial internal pressure on the containment components must be considered to be the MNOP, unless a lower internal pressure consistent with the selected ambient temperature is more severe.

The evaluation must demonstrate the package has sufficient structural integrity to maintain containment/confinement, shielding, and subcriticality under the AC specified. Results of this evaluation must also be considered for the following areas:

- Deformation of shielding components must be included in the shielding evaluation.
- Deformation of components required for heat transfer must be included in the thermal evaluation.
- Deformation of components required for subcriticality must be included in the criticality evaluation.

3.1.2.2(i)(1) Free Drop. In demonstrating the adequacy of the package under the effects of the free drop tests as specified in Chapter VI, paragraph 5.2.5.2(c)(1), the following must be included:

- The free drop orientation that causes the most severe damage, including center of gravity over corner, oblique orientation with secondary impact, side drop, and free drop onto the closure must be evaluated.
- It must be noted that the most damaging orientation for one component may not be the most damaging orientation for another component.

- When a feature such as a tiedown component is a structural part of the package, it must be included in the selection of the free drop test configuration and the free drop orientation.
- For a package with lead shielding, the effects of lead slump must be evaluated. The lead slump determined must be consistent with that used in the shielding evaluation.

3.1.2.2(i)(2) Crush. When required, the adequacy of the package under the effects of the dynamic crush test as specified in Chapter VI, paragraph 5.2.5.2(c)(2) must be demonstrated by evaluating the structural response of the package to the loads specified. The package must have been previously subjected to the free drop test.

3.1.2.2(i)(3) Puncture. In demonstrating adequacy of the package under the effects of the puncture tests as specified in Chapter VI, paragraph 5.2.5.2(c)(3), the following must be included:

- Position for which maximum damage would be expected must be evaluated.
- Any damage from the free drop and crush conditions must be considered.
- Punctures at oblique angles, near a support, at a valve, and at a penetration must be considered.

3.1.2.2(i)(4) Thermal. In demonstrating the adequacy of the package under the effects of the fully engulfing fire thermal test as specified in Chapter VI, paragraph 5.2.5.2(c)(4), the following must be included:

- Any damage resulting from the free drop, crush, and puncture conditions must be incorporated into the initial conditions of the package for the fire test.
- Determination of the maximum pressure in the package during and after the test considers the temperature resulting from the fire and any increase in gas inventory caused by combustion or decomposition processes.
- Evaluate maximum thermal stresses that can occur either during or after the fire.

3.1.2.2(i)(5) Immersion-Fissile Material. When water in-leakage is not assumed in the criticality evaluation, the adequacy of the package under the effects of immersion for fissile material packages as specified in Chapter VI, paragraph 5.2.5.2(c)(5) must be demonstrated by evaluating the package for in-leakage of water. A water head of 0.9 m (3 ft) in the orientation that maximum leakage is expected must be applied to the package.

3.1.2.2(i)(6) Immersion-All Packages. In demonstrating the adequacy of the package under the effects of the immersion tests as specified in Chapter VI, paragraph 5.2.5.2(c)(6), the following must be included:

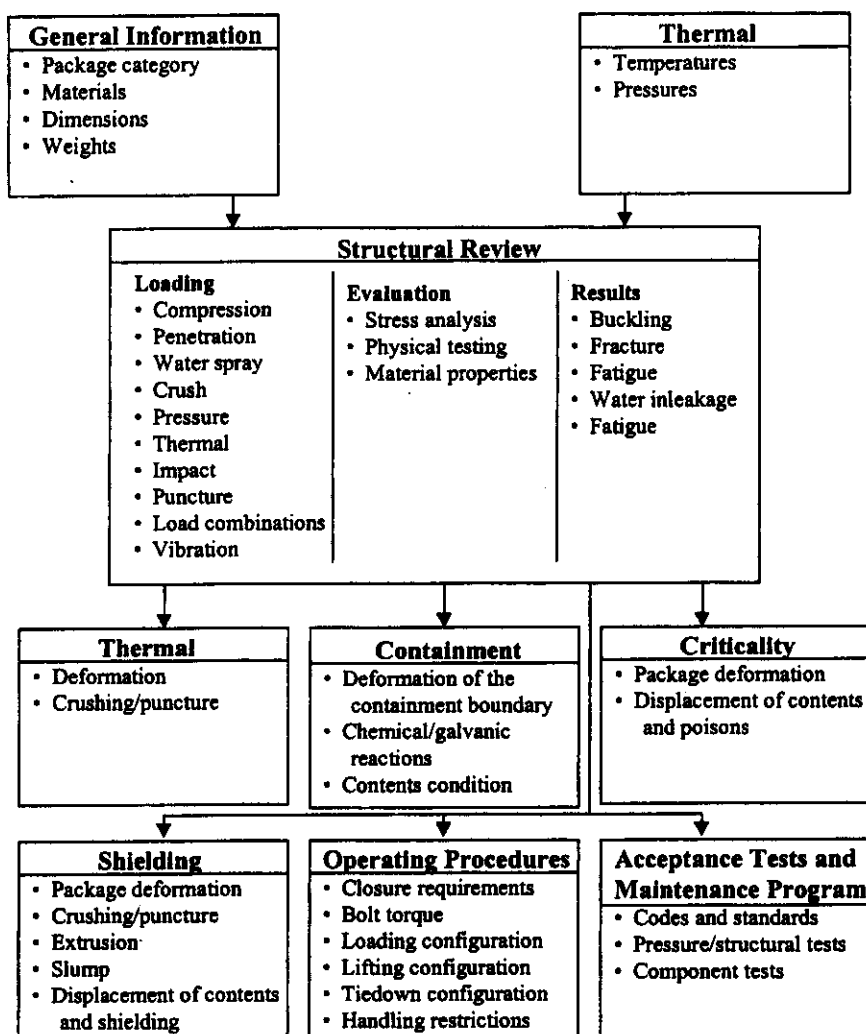
- An undamaged specimen must be subjected to a water pressure equivalent to immersion under a head of water of at least 15 m (50 ft). This is equivalent to an external pressure of 150 kPa (21.7 psig).
- The evaluation must consider the loading condition in combination with the minimum internal pressure.
- The greatest possible pressure difference between the inside and outside of the containment components must be evaluated.
- The possibility of buckling must be evaluated.

3.1.2.2(j) Evaluation Procedures. Structural evaluation of a package is based in part on the description and evaluations presented in the general information and thermal evaluation sections. In addition, results of the structural evaluation are considered in the evaluation of all other sections. As an aid to the user, an information flow diagram for the structural evaluation is presented in Figure 7-1.

3.1.2.3 Thermal Evaluation. This section outlines elements that are to be addressed in an onsite PSAA to demonstrate acceptable thermal performance based on performance tests defined in Chapter VI, Section 5.2.5. The section also defines the acceptable performance criteria for testing and/or analysis to be applied to demonstrate the package provides an equivalent degree of safety to the regulations in maintaining containment, subcriticality, and shielding. It should be noted that in any onsite PSAA, claims of adequacy of designs or design methods must be supported by technical bases; i.e., by an appropriate engineering evaluation or description of actual tests.

The package shall be evaluated for all NC thermal evaluation events -- heat, cold, reduced external pressure, and increased external pressure -- by either testing or analysis. These tests do not need to be performed in a specific sequence. The package shall also be evaluated for the AC thermal condition (fire) by either testing or analysis, and must be preceded by the free drop, crush (when applicable), and puncture performance tests.

Figure 7-1. Structural Evaluation Flow Diagram.



3.1.2.3(a) Description of Thermal Design.

3.1.2.3(a)(1) Design Features. Design features that are important to the thermal performance include the following:

- Package geometry and materials of construction.
- Structural and mechanical features that may affect heat transfer, such as cooling fins, insulating materials, surface conditions of the package components, and gaps or physical contacts between internal components.

3.1.2.3(a)(2) Contents Decay Heat. The decay heat will be determined from the maximum allowed radioactive contents.

3.1.2.3(a)(3) Summary Tables of Temperatures. Summary tables of the maximum or minimum temperatures that affect structural integrity, containment, shielding, and criticality will be presented for both NC and AC. For fire test conditions, the tables will also include the following:

- Maximum temperatures and the time at which they occur after fire initiation
- Maximum temperatures of the post-fire steady-state condition.

The temperatures will be consistent with the structural evaluation (3.1.2.2) and containment/confinement (3.1.2.4) sections.

3.1.2.3(a)(4) Summary Tables of Maximum Pressures in the Containment System. The summary tables will include the MNOP and the maximum pressure under AC. These pressures will be consistent with those in the general information (3.1.1.1), structural evaluation (3.1.2.2), containment/confinement (3.1.2.4), and acceptance tests and maintenance program (3.1.2.11) sections.

3.1.2.3(b) Material Properties and Component Specification.

3.1.2.3(b)(1) Material Properties. Appropriate thermal properties will be specified for materials that affect heat transfer both within the package and from the package to the environment. These materials include any liquids or gases within the package and gases external to the package for AC.

Thermal absorptivities and emissivities must be appropriate for the package surface conditions and each thermal condition being evaluated. If a property is reported as a single value, this value must bound the equivalent temperature-dependent property.

3.1.2.3(b)(2) Component Specifications.

- Maximum allowable service temperatures or pressures will be specified for each package component, as appropriate.
- Minimum allowable service temperature of all the package components will be less than or equal to -33°C (-27 °F).
- Technical specifications for applicable package components (such as pressure relief valves and fusible plugs) will be provided.

3.1.2.3(c) General Considerations.

- Thermal evaluation of the package design can be performed by analysis or testing, or by a combination of both.

- A packaging may be tested or evaluated in accordance with the requirements of Chapter VI or the regulations.
- Because Chapter VI provides an equivalent degree of safety to the regulations, the performance test conditions and parameters are not interchangeable. In other words, the package is either analyzed or tested to the test conditions and parameters of Chapter VI or the regulations, not a combination of the two.
- Testing of the package will be conducted, as a minimum, in accordance with performance test conditions defined in Chapter VI.

3.1.2.3(c)(1) Evaluation by Test. For evaluation by testing, the test package, test facility, and test procedures must be described in detail. An approved QAP must be in place for the fabrication of the test package, operation of the test facility, and the gathering and evaluation of the test data. To ensure adequate prototypical testing in accordance with the performance test conditions set forth in Chapter VI or the regulations:

- Thermal performance of the test package, including simulated contents, test instrumentation, and attachment hardware, will be representative of the actual package design.
- Temperature-sensing instrumentation will be located to measure the maximum temperature of critical package components and to characterize the significant heat transfer pathways.
- Test instrumentation will be mounted at locations that minimize their effects on local test temperatures.
- The pre-fire condition for testing will be the maximum NC steady-state temperature.
- For performance tests, quenching and fire suppression of the AC fire will be initiated after a minimum fire duration of 30 minutes.

Test results, as a minimum, will include the following:

- Initial conditions (e.g., temperature, pressure) and changes in the package resulting from structural tests.
- Maximum steady-state temperatures and pressures (e.g., pre-fire conditions).
- Maximum temperature and pressure during the fire and post-fire periods.
- Physical changes in the package condition resulting from the test, such as changes in packaging material properties caused by combustion or melting of packaging components.

Conditions, such as ambient temperature, decay heat, and packaging absorptivity and emissivity, may not be exactly represented in thermal testing. Corrections or evaluations to account for these differences must be provided in the safety assessment report. For example, a thermal

evaluation must be provided to correct for temperature if the ambient temperature at the onset of the fire test is lower than 46 °C (115 °F).

3.1.2.3(c)(2) Evaluation by Analysis. If the package performance is demonstrated by analysis, the onsite PSAA must include the following:

- The methods used are properly referenced or developed, and correctly applied.
- Assumptions in modeling heat sources and heat transfer paths and modes must be clearly stated and justified.
- The appropriate thermal properties for the package materials are correctly incorporated.
- The evaluation considers changes in package geometry and material properties resulting from structural and thermal tests under NC and AC.
- The required temperature and thermal boundary conditions for NC and AC are applied correctly.
- The time interval after the fire test is adequate to ensure the maximum component temperatures and post-fire steady-state temperature has been achieved.
- The maximum temperature and pressure of each component do not exceed their allowable values.
- Combustion of the package components is considered, including the heat produced.
- Thermal evaluation should assume the heat transfer medium is air, and effects of air on the contents and packaging components (e.g., oxidation of depleted uranium shielding) should be addressed.

If analysis is chosen in lieu of testing, the analysis will demonstrate satisfactory performance of the package to the performance test conditions defined in Chapter VI. The analysis performed will meet the following requirements:

- Computer numerical simulations of package performance under the specified test conditions will have been benchmarked and properly documented.
- Thermal quenching will be applied 30 minutes after initiation of the fully engulfing fire.
- Combustion of package components will be accounted for in the evaluation, including the heat produced.
- The thermal evaluation will assume that the heat transfer medium is air, and the effects of air on the contents and packaging components (e.g., oxidation) must be addressed.

3.1.2.3(c)(3) Acceptance Criteria for Evaluation by Analysis.

- Under NC, structural integrity and safe performance of the package will demonstrate that the package maintains containment, shielding, and subcriticality for the specified performance tests.
- In addition, packages will be demonstrated to maintain an exterior packaging temperature of 50 °C (122 °F) for nonexclusive use and 85 °C (185 °F) for exclusive use shipments under the hot extremes of Hanford Site conditions in the shade.
- Maximum seal and shielding material temperatures will be demonstrated to be below the rated maximum continuous use temperature of the material for NC.
- Also, component material temperatures will be demonstrated to be above the rated minimum continuous use temperature of the material for NC.
- Maximum normal pressures and temperatures will be established for key packaging components and will be consistent with the pressures and material temperatures used in the structural section.
- Structural integrity and safe performance of the package is demonstrated by evaluating the cumulative damage from the previous AC free drop, crush (when applicable), and puncture loads combined with the thermal stresses, pressure, and differential thermal expansion loadings from the fire. This evaluation is to be provided in the structural section.
- Maximum accident pressures and temperatures will be established for key packaging components and will be consistent with the pressures and material temperatures used in the structural section. In the case of the package closure mechanisms, such as bolts, the cumulative strains from the entire sequence of accident loads will be evaluated to demonstrate that containment is maintained.
- Under fire conditions, the limiting value to demonstrate containment by the primary seals is based on the manufacturers stated high temperature operation and duration being below the maximum fire accident temperature and duration at the seal.
- Maintenance of package shielding and criticality control will be demonstrated by the ability of the package to maintain structural integrity and effectiveness under accident loading conditions after the sequence of accidents specified in Chapter VI.
- As an example, lead-shielding effectiveness will be demonstrated in this section, as well as the structural and shielding sections, by demonstrating no loss of shielding effectiveness from lead slump resulting from the sequence of accident loads.

3.1.2.3(c)(4) Margins of Safety.

- Margins of safety must be established for temperature, pressure, and thermal stresses for all packaging components important to safety.
- In establishing the margins of safety, the evaluation of package performance must address the effects of uncertainties in thermal properties, test conditions, diagnostics, and analytical methods.
- If package performance is assessed by testing, it must be demonstrated that the test results are reliable and repeatable.

3.1.2.3(c)(5) Thermal Evaluation under NC. The thermal evaluation should demonstrate that the tests for NC do not result in significant reduction in packaging effectiveness, including the following:

- Degradation of the heat-transfer capability of the packaging (such as creation of new gaps between components).
- Changes in material conditions or properties (e.g., expansion, contraction, gas generation, and thermal stresses) that affect the structural performance.
- Changes in the package that affect containment, shielding, or criticality such as thermal decomposition or melting of materials.
- Ability of the packaging to withstand the tests under NC.

It will be verified that the component temperatures and pressures do not exceed their allowable values.

3.1.2.3(c)(5)(i) Heat.

- The Hanford Site extreme hot temperature is 46 °C (115 °F).
- The maximum allowable package external temperature in the shade is 85 °C (185 °F) for exclusive use and 50 °C (122 °F) for nonexclusive use shipments.

3.1.2.3(c)(5)(ii) Cold. The Hanford Site cold temperature extreme is -33 °C (-27 °F).

3.1.2.3(c)(5)(iii) MNOP. The MNOP shall be determined when the package has been subjected to the steady-state heat condition. The MNOP calculation will consider all possible sources of gases such as the following:

- Gases initially present in the package.
- Saturated vapor, including water vapor from the contents or packaging.

- Helium from the radioactive decay of the contents.
- Hydrogen or other gases resulting from thermal- or radiation-induced decomposition of materials such as water or plastic.

3.1.2.3(c)(5)(iv) Hydrogen or other flammable gases comprising greater than 5% by volume of the total gas inventory within any confined volume will be evaluated.

3.1.2.3(c)(5)(v) Evaluation of the effects of flammable gas contents will include deflagration, deflagration to detonation transition, and the resultant pressures as described in the gas generation section.

3.1.2.3(c)(5)(vi) Thermal stresses that are caused by constrained interfaces among package components resulting from temperature gradients and different thermal expansions will be evaluated. The evaluation will include thermal stresses as well as cyclic stresses during the service life of the package.

3.1.2.3(d) Thermal Evaluation under AC.

3.1.2.3(d)(1) Initial Conditions. Prior to the fire test, the package design should be evaluated for the effects of the drop, crush (if applicable), and puncture tests. The initial physical condition of the package design used in the thermal evaluation should consider these effects. The most unfavorable conditions will be used:

- An ambient temperature between -27°C and 115°C (-16°F and 239°F).
- An internal pressure of the package equal to the MNOP unless a lower internal pressure, consistent with the ambient temperature, is less favorable.
- Contents at its maximum decay heat unless a lower heat, consistent with the temperature and pressure, is less favorable.

The initial steady-state temperature distribution should be consistent with the thermal evaluation under NC.

3.1.2.3(d)(2) Thermal Performance Test. The following conditions should be met for the thermal performance test:

- The package will be fully engulfed in a fire with an average emissivity coefficient of at least 0.9, with an average flame temperature of at least 800°C ($1,475^{\circ}\text{F}$) for a period of 30 minutes.
- The fuel source must extend horizontally at least 1 m (40 in.), but may not extend more than 3 m (10 ft) beyond any external surface of the package.

- The package must be positioned 1 m (40 in.) above the surface of the fuel source.
- For purposes of calculation, the surface absorptivity coefficient must be the value that the package may be expected to possess when exposed to the fire specified, and the convective coefficient must be that value which may be demonstrated to exist if the package were exposed to the fire specified.
- Artificial cooling (quenching) may be applied 30 minutes after the initiation of the fire.
- Thermal insulation must be applied before and after the fire.

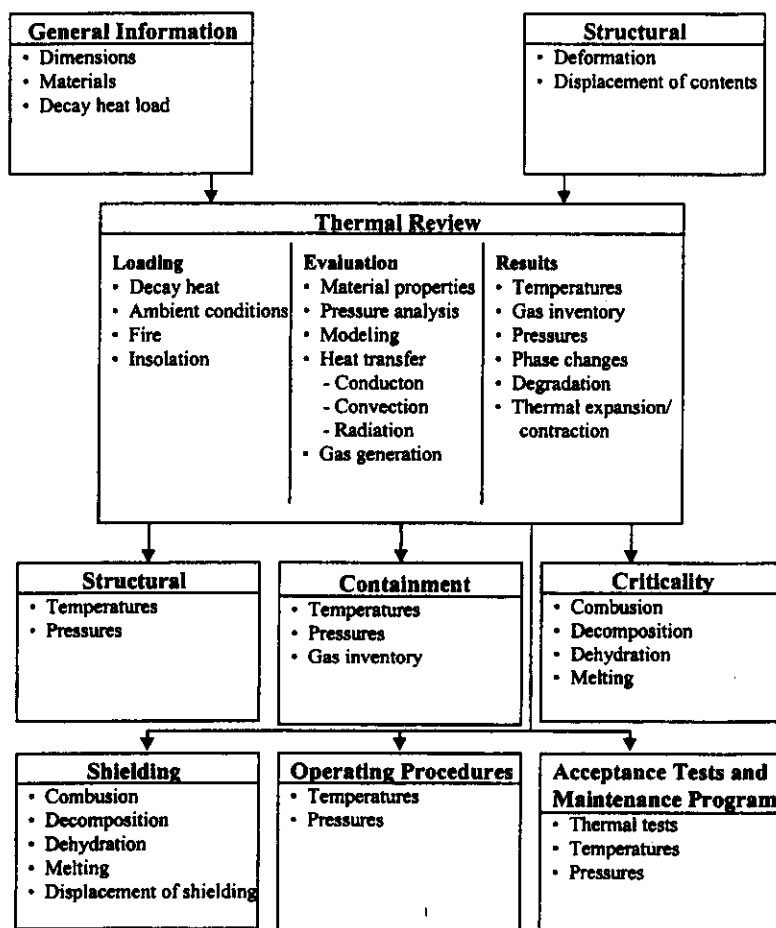
3.1.2.3(d)(3) Maximum Temperature and Pressure.

- The evaluation will appropriately determine both the transient peak temperatures of the package components as a function of time after the fire, and the maximum temperatures from the post-fire, steady-state condition.
- The lead shielding will not reach melting temperature.
- The evaluation of the maximum pressure in the package design will be based on the MNOP as it is affected by fire-caused increases in package component temperatures.
- Possible increases in the gas inventory, caused by fire-induced thermal combustion or decomposition processes, will be accounted for in the pressure determination and in the gas generation section.

3.1.2.3(d)(4) Maximum Thermal Stresses. There will be an evaluation of the thermal stresses caused by constrained interfaces among the package components resulting from temperature gradients and differential thermal expansions.

3.1.2.3(e) Evaluation Procedures. The thermal evaluation of a package is based in part on the description and evaluations presented in the general information (3.1.1.1) and the structural evaluation (3.1.2.2) sections. In addition, results of the thermal evaluation are considered in the evaluation of all other sections. As an aid to the user, an information flow diagram for the thermal evaluation is presented in Figure 7-2.

Figure 7-2. Thermal Evaluation Flow Diagram.



3.1.2.4 Containment/Confinement Evaluation. This section outlines the elements that are to be addressed for an onsite PSAA to demonstrate acceptable containment/confinement performance based on the performance tests defined in Chapter VI, Section 5.2.5 and containment/confinement requirements of Chapter VI, Section 5.2.6. The section also defines the acceptable performance criteria for testing and/or analysis to be applied to demonstrate the package provides an equivalent degree of safety to the regulations in maintaining containment/confinement, subcriticality, and shielding. It should be noted that in any onsite PSAA, claims of adequacy of designs, or design methods must be supported by technical bases; i.e., by an appropriate engineering evaluation or description of actual tests.

3.1.2.4(a) Description of Containment/Confinement System. The following information must be provided in the containment/confinement system description of the packaging.

3.1.2.4(a)(1) The exact containment/confinement boundary must be defined. This includes the containment/confinement vessel, welds, seals, lids, cover plates, valves, and other closure devices. In the case of filtered confinement system packages, containment/confinement components such as the vessel, closure lid, filter housing, and seals must be defined as the containment/confinement boundary.

3.1.2.4(a)(2) All components of the containment/confinement boundary must be shown and specified on the drawings.

3.1.2.4(a)(3) The following information provided on the containment/confinement boundary must be consistent with that provided in the structural and thermal evaluations:

- Material of construction
- Welds
- Applicable codes and standards (e.g., ASME BPVC code specifications for the vessel)
- Bolt torques required to maintain positive closure
- Maximum and minimum allowable temperatures of components, including seals
- Maximum and minimum temperatures of components under the tests for NCs and ACs.

3.1.2.4(a)(4) All containment/confinement boundary penetrations and their method of closure must be adequately described.

3.1.2.4(a)(5) Performance specifications for components such as filters, valves, and pressure relief devices must be identified.

3.1.2.4(a)(6) If penetration into the containment/confinement boundary are closed by two seals (e.g., to enable leak testing), the containment/confinement boundary component seal must be defined.

3.1.2.4(a)(7) The radionuclides and physical form of the contents must be provided and consistent with Section 3.1.2.1. Any significant daughter products must be provided.

3.1.2.4(a)(8) The constituents of the releasable source term, including radioactive gases, liquids, and powdered aerosols must be identified and described.

3.1.2.4(b) General Requirements.

3.1.2.4(b)(1) Acceptable performance can be demonstrated either by testing or analytically.

3.1.2.4(b)(2) The package must be demonstrated to satisfy the quantified release rates specified in Chapter VI, Sections 5.2.6.8 and 5.2.6.9.

3.1.2.4(b)(3) The maximum permissible volumetric leakage rates based on the allowed release rates specified in Chapter VI, Sections 5.2.6.8 and 5.2.6.9 must be determined for both NC and AC. ANSI N14.5 provides an acceptable method for this determination.

3.1.2.4(b)(4) The volumetric leakage rates must be converted to standard air leakage in accordance with ANSI N14.5.

3.1.2.4(b)(5) The evaluation of any combustible gas generation is to be performed in accordance with Section 3.1.2.7, Gas Generation.

3.1.2.4(c) Acceptance Criteria. The package design must be demonstrated to meet the release requirements specified in Chapter VI, Sections 5.2.6.8 and 5.2.6.9 for both filtered and sealed packages. The package must be demonstrated to satisfy the following requirements:

3.1.2.4(c)(1) Any filters and valves (except pressure relief valves) on the package must be demonstrated to be protected against unauthorized operation.

3.1.2.4(c)(2) For filtered packages, the filter must be demonstrated to meet the requirements of Chapter VI, paragraph 5.2.6.1(h). The filters must also be demonstrated to maintain confinement effectiveness under both NC and AC.

3.1.2.4(c)(3) Cover plates and lids must be demonstrated to be protected by recessing or other methods.

3.1.2.4(c)(4) All containment seals and penetrations, including drain and vent ports, must be demonstrated to be leak testable. If fill, drain, or test ports utilize quick-disconnect valves they will be demonstrated not to preclude leak testing of the containment seals.

3.1.2.4(c)(5) Demonstrate no galvanic, chemical, or other reactions will occur between the seal and the packaging or its contents, and that the seal will not degrade due to irradiation.

3.1.2.4(c)(6) Demonstrate that the proper dimensions of the seal grooves are specified for the type and size of seal specified.

3.1.2.4(c)(7) Demonstrate that the temperature of the containment boundary component seals will remain within their specified allowable limits under both NC and AC.

3.1.2.4(c)(8) Demonstrate that the containment system components are securely closed by a positive fastening device that cannot be opened unintentionally or by a pressure that may arise within the package.

3.1.2.4(c)(9) If less than 100% of the contents are considered releasable a justification for the lower fraction must be provided.

3.1.2.4(c)(10) Elastomeric seal compression must be maintained under both NC and AC loading conditions so as to prevent a permanent compression set of the elastomeric material.

3.1.2.4(c)(11) The containment/confinement system must be demonstrated to satisfy the requirements of Chapter VI, Section 5.2.6.10.

3.1.2.4(c)(12) For packages equipped with filters, the total release rate must be demonstrated to be with the limits specified in Chapter VI, Sections 5.2.6.8 and 5.2.6.9. This includes accounting for the leakage rate from the containment components of the confinement system.

3.1.2.4(d) Containment under NCs.

3.1.2.4(d)(1) The package must be demonstrated to satisfy the allowable release rate requirements specified in Chapter VI, Section 5.2.6.8.

3.1.2.4(d)(2) The MNOP and maximum temperature must be consistent with those determined in Sections 3.1.2.2, 3.1.2.3, and 3.1.2.7.

3.1.2.4(d)(3) Using the above pressure and temperature, the maximum permissible leakage rate must be converted to the reference air leakage rate in standard cubic centimeters per second (std cc/s), as defined in ANSI N14.5.

3.1.2.4(d)(4) If compliance is demonstrated by testing, the leakage rate of the package subjected to the tests specified in Chapter VI, Section 5.2.5.1 must not exceed the maximum allowable leakage rate for NC. Leakage testing of scale-model package may be applicable, but is not necessarily sufficient demonstration of compliance.

3.1.2.4(d)(5) If compliance is demonstrated by analysis, it must be demonstrated that the package containment boundary components, seal regions, and closure bolts do not undergo any deformations that could result in a breach of the containment boundary or loss of elastomeric seal compression of less than 0.018 cm (0.007 in.). In addition, the materials of the containment/confinement system must be demonstrated not to exceed their maximum allowable temperature limits when subjected to the conditions specified in Chapter VI, Section 5.2.5.1.

3.1.2.4(d)(6) As specified in Section 3.1.2.11, a leak test must be performed during acceptance testing to demonstrate that the package meets the maximum allowable leakage rate.

3.1.2.4(d)(7) The amount of decompression of the seal, when subjected to the conditions specified in Chapter VI, Section 5.2.5.1 must be determined and must be sufficient to maintain containment of the contents.

3.1.2.4(e) Containment under ACs.

3.1.2.4(e)(1) The packaging must be demonstrated to satisfy the allowable release rate requirements specified in Chapter VI, Section 5.2.6.9.

3.1.2.4(e)(2) The temperatures, pressures, and physical conditions of the package (including contents) must be consistent with those determined in Sections 3.1.2.2, 3.1.2.3, and 3.1.2.7.

3.1.2.4(e)(3) Using the above pressure and temperature, the maximum permissible leakage rate must be converted to the air leakage rate in standard cubic centimeters per second (std cc/s), as defined in ANSI N14.5.

3.1.2.4(e)(4) If compliance is demonstrated by testing, the leakage rate of the package subjected to the tests specified in Chapter VI, Section 5.2.5.2 must not exceed the maximum allowable leakage rate for AC. Leakage testing of scale-model package may be applicable, but is not necessarily sufficient demonstration of compliance.

3.1.2.4(e)(5) If compliance is demonstrated by analysis, it must be demonstrated that the package containment boundary components, seal regions, and closure bolts do not undergo any deformations that could result in a breach of the containment boundary or loss of elastomeric seal compression of less than 0.018 cm (0.007 in.). In addition, the materials of the containment/confinement system must be demonstrated not to exceed their maximum allowable temperature limits when subjected to the conditions specified in Chapter VI, Section 5.2.5.2.

3.1.2.4(e)(6) The amount of decompression of the seal when subjected to the conditions specified in Chapter VI, Section 5.2.5.2 must be determined and must be sufficient to maintain containment of the contents.

3.1.2.4(f) Leakage Rate Tests.

3.1.2.4(f)(1) Based on the reference air leakage rate, it will be demonstrated by providing data that the allowable leakage rate for the following conditions is determined in accordance with ANSI N14.5:

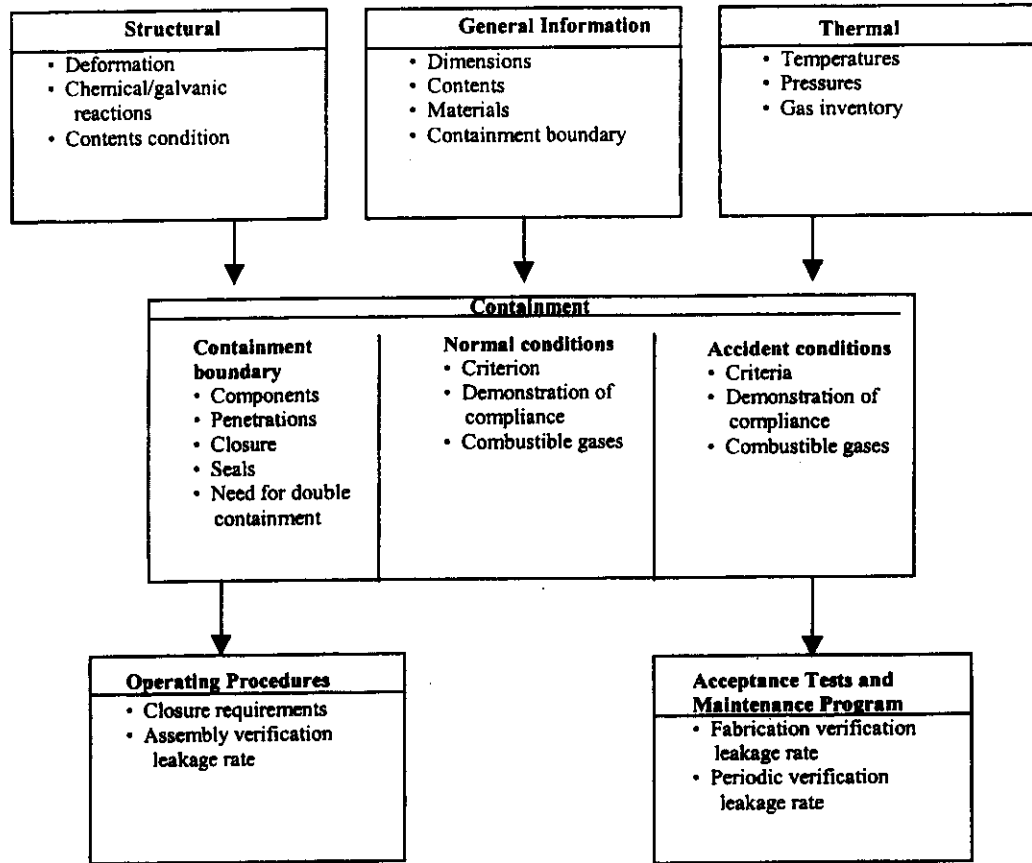
- Fabrication leakage rate test
- Periodic leakage rate tests
- Assembly (pre-shipment) leakage rate tests.

3.1.2.4(f)(2) The fabrication and periodic leakage rate tests must be included in Section 3.1.2.11.

3.1.2.4(f)(3) The pre-shipment leakage rate test for assembly verification must be included in Section 3.1.2.11.

3.1.2.4(g) Evaluation Procedures. The containment/confinement evaluation of a package is based in part on the description and evaluations presented in the general information and the structural, thermal, and gas generation evaluations. In addition, results of the containment/confinement evaluation are considered in the evaluation of all other sections. As an aid to the user, an information flow diagram for the containment/confinement evaluation is presented in Figure 7-3.

Figure 7-3. Containment/Confinement Evaluation Flow Diagram.



3.1.2.5 Shielding Evaluation.

3.1.2.5(a) Chapter VI Requirements. This section outlines the elements that are to be addressed for an onsite PSAA to demonstrate shielding performance.

3.1.2.5(b) Description of Shielding System Design. The description of the shielding system design will include the following:

- Dimensions, tolerances, and densities of material for neutron or gamma shielding, including those packaging components considered in the shielding evaluation.
- Mass density, atomic density, or areal density of materials used as neutron absorbers.
- Structural components that maintain the contents in a fixed position within the package.
- Dimensions of the transport vehicle that are to be considered in the evaluation.

Maximum dose rates shall be presented for NC and AC at the appropriate locations. If the package is designed for multiple types of contents, the contents producing the highest external dose rate at each location will be clearly identified and evaluated.

3.1.2.5(c) Radiation Source.

3.1.2.5(c)(1) Gamma Source. The maximum gamma source strength and spectra will be calculated in the evaluation. The source contribution from radioactive daughter products will be accounted for. The production of secondary gamma reactions (e.g., from $[n, \gamma]$ reactions in the shield material) will also be included in the source term.

3.1.2.5(c)(2) Neutron Source. The calculation for the neutron source will consider both spontaneous fission and (α, n) reactions, as appropriate. If either of these source contributions is considered negligible, an appropriate justification will be provided. The production of neutrons from subcritical multiplication will be calculated as part of the evaluation or otherwise appropriately included in the source term. The contribution from spontaneous fission and (α, n) will be separately identified, along with the actinides or light nuclei that are significant for these processes.

3.1.2.5(d) Shielding Model. If the contents of the package can be positioned at varying locations or with varying densities, the location and physical properties of the contents used in the evaluation will be those resulting in the maximum external radiation levels. Any changes in configuration (e.g., displacement of source or shielding, reduction in shielding) that result under NC or AC will be included, as appropriate.

3.1.2.5(d)(1) Material Properties. The appropriate material properties (e.g., mass densities and atom densities) will be used in the shielding models of the packaging, contents, and conveyance (if applicable). Any changes resulting under NC and AC will be included as appropriate. For example, the melting of lead is not acceptable under either NC or AC. Shielding properties will not degrade during the service life of the packaging (e.g., degradation of foam or dehydration of hydrogenous materials).

If the shielding model considers a homogeneous source region, rather than a detailed heterogeneous model of the contents, the approach will be justified. It will also be verified that the homogenized mass densities are correct for NC and AC.

3.1.2.5(e) Shielding Evaluation.

3.1.2.5(e)(1) Methods. The methods used for the shielding evaluation will be appropriate. Standard computer programs used will be referenced and evidence of verification and benchmarking shall be provided. Other codes or methods will be described in the document, and the appropriate supplemental information shall be provided. The number of dimensions of the code will be appropriate for the package geometry.

The cross-section library used by the code will be applicable for the shielding calculations. The code will account for subcritical multiplication and secondary gamma production unless these conditions have been considered elsewhere (e.g., in the source term specification).

3.1.2.5(e)(2) Input and Output Data. Key input data for the shielding calculations will be identified. The key input data depend on the type of code, as well as the code itself. The document will include representative input files used in the analyses.

Representative output files will be included in the PSSD. The calculated dose rates from the output files will agree with those called out in the text. For computer programs that are based on the Monte Carlo method, proper convergence will be achieved.

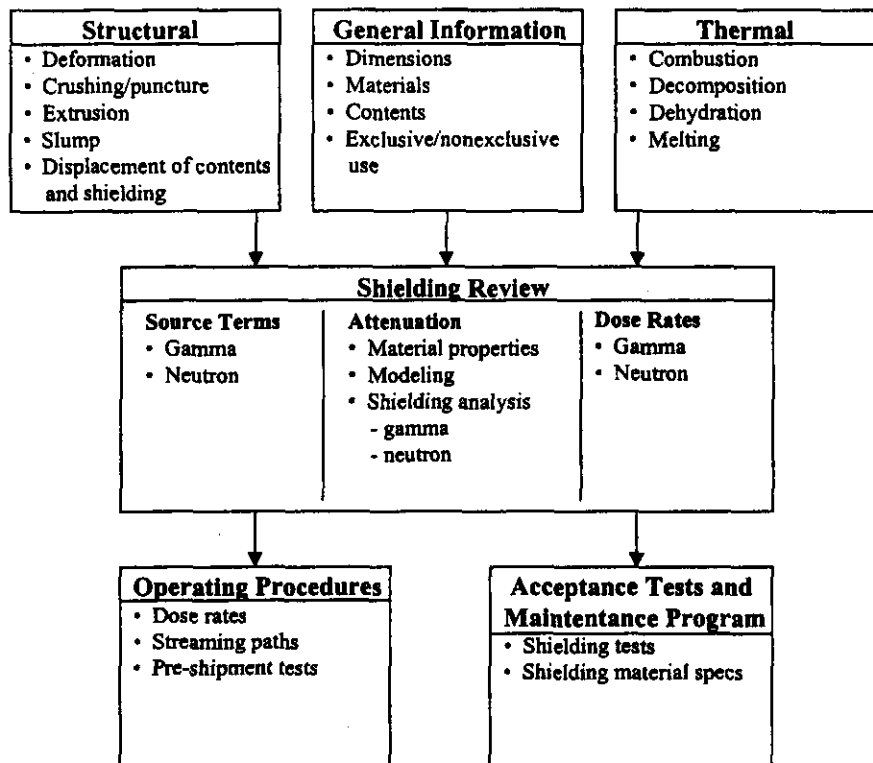
3.1.2.5(e)(3) Flux to Dose Rate Conversion. Gamma and neutron flux will be properly converted to dose rates. This conversion should generally use ANSI/ANS 6.1.1-1977, "Neutron and Gamma-Ray Flux-to-Dose-Rate Factors," although other conversions may be used for point-kernel gamma calculations. Note that use of the conversions in ANSI/ANS 6.1.1-1991 can result in a significant underestimation of external dose rates. Additionally, the dose rates determined with the 1991 standard do not correspond physically to dose rates measured by typical radiation monitoring instruments.

3.1.2.5(e)(4) External Radiation Levels. External radiation levels under NC and AC will meet the appropriate limits. The analyses will show the locations selected are those with the maximum dose rates. To determine maximum dose rates, radiation levels may be averaged over the cross-sectional area of a probe of reasonable size. For packages with streaming paths or voids, averaging should not be used to reduce the radiation levels resulting from such features.

The evaluation will address damage to the shielding under NC and AC. Any damage under NC will not exceed allowable limits. Any increase will be explained and justified as not significant.

3.1.2.5(e)(5) Evaluation Procedures. The shielding evaluation of a package is based in part on the description and evaluations presented in paragraph 3.1.2.5(e). In addition, results of the shielding evaluation are considered in the evaluation of all other sections. As an aid to the user, an information flow diagram for the shielding evaluation is presented in Figure 7-4.

Figure 7-4. Shielding Evaluation Flow Diagram.



3.1.2.6 Criticality Evaluation.

3.1.2.6(a) Equivalent and Nonequivalent Packaging Requirements. A package used for the shipment of fissile material must be designed and constructed, and its contents limited so that it would be subcritical if water were to leak into the containment system, or liquid contents were to leak out of the containment system. Consequently, the contents shall be demonstrated to remain subcritical under the following conditions:

1. The most reactive credible configuration consistent with the chemical and physical form of the material.
2. Moderation by water to the most reactive credible extent.
3. Close full reflection of the containment system by water on all sides, or such greater reflection of the containment system as may additionally be provided by the surrounding material of the packaging.

A package that is used for the shipment of fissile material will be demonstrated to remain subcritical under NC by demonstrating that:

1. The contents will be subcritical.
2. The geometric form of the package contents will not be substantially altered.

3. There will be no leakage of water into the containment system unless, in the evaluation of undamaged packages, it has been assumed that moderation is present to such an extent as to cause maximum reactivity consistent with the chemical and physical form of the material.
4. There will be no substantial reduction in the effectiveness of the packaging including:
 - No more than 5% reduction in the total effective volume of the packaging on which nuclear safety is assessed.
 - No more than 5% reduction in the effective spacing between the fissile contents and the outer surface of the packaging.
 - No occurrence of an aperture in the outer surface of the packaging large enough to permit the entry of a 10 cm (4-in.) cube.

A package that is used for the shipment of fissile material will be demonstrated to remain subcritical under AC. To demonstrate this, the following assumptions will be made:

1. The fissile material is in the most reactive credible configuration consistent with the damaged condition of the package, and the chemical and physical form of the contents.
2. Water moderation occurs to the most reactive credible extent consistent with the damaged condition of the package, and the chemical and physical form of the contents.
3. There is full reflection by water on all sides, as close as is consistent with the damaged condition of the package.

A fissile material package must be controlled by the shipper or the carrier during transport to ensure that the array of packages remains subcritical. To verify this control, the fissile material package will be demonstrated to have a number "N" based on the following conditions being satisfied, assuming the packages are stacked together in any arrangement and with close full reflection on all sides of the stack by water:

1. Five times "N" undamaged packages with nothing between the packages would be subcritical.
2. Two times "N" damaged packages, if each package were subjected to the AC tests would be subcritical with optimum interspersed hydrogenous moderation.
3. The value of "N" cannot be less than 0.5.

3.1.2.6(b) Description of Criticality Control Design. Design features that will be included in the criticality evaluation are as follows:

- Dimensions and tolerance of the containment system for fissile material.
- Structural components that maintain the fissile material or neutron poisons in a fixed position within the package or in a fixed position relative to each other.
- Location, dimensions, and concentration of neutron absorbing materials and moderating materials, including neutron poisons and shielding material.
- Dimensions and tolerances of floodable voids and flux traps within the package.
- Dimensions and tolerances of the overall package that affect the physical separation of the fissile material contents in package arrays.

The following cases will be addressed: (1) a single package; (2) an array of undamaged packages; and (3) an array of damaged packages. Based on the number of packages evaluated in the arrays, the appropriate N should be determined and the criticality TI calculated.

3.1.2.6(c) Fissile Content. Specifications that will be included in the criticality evaluation include fissile material mass, dimensions, enrichment, physical and chemical composition, density, moisture, and other characteristics that are dependent on the specific contents. Because a partially filled container may allow more room for moderators (e.g., water), the most reactive case may be for a mass of fissile material that is less than the maximum allowable contents.

If the package is designed for multiple types of contents, a separate criticality evaluation and proposal for different criticality controls for each content type may be included. Any assumptions that certain contents need not be evaluated because they are less reactive than evaluated contents will also be properly justified.

3.1.2.6(d) General Considerations. The following considerations are applicable to the criticality evaluations of a single package and arrays of packages under NC and AC.

3.1.2.6(d)(1) Model Configuration. The models used in the criticality calculation will be consistent with the effects on the packaging and its contents during NC and AC. The dimensions of the contents and the packaging used in the model will be correct. For some types of packagings and contents (e.g., powders), the contents can be positioned at varying locations and densities. The relative location and physical properties of the contents within the packaging will be justified as those resulting in the maximum multiplication factor. Dimensional tolerances (e.g., for cavity sizes and poison thickness) will be considered in a manner that maximizes reactivity.

3.1.2.6(d)(2) Material Properties. The appropriate mass densities and/or atom densities will be provided for materials used in the models of the packaging and contents. Material properties will

be consistent with the condition of the package under the NC and AC tests, and any difference between the NC and AC will be addressed.

The materials will be relevant to the criticality design (e.g., poisons, foams, plastics, and other hydrocarbons) and properly specified. No more than 75% of the specified minimum neutron poison concentration should generally be considered in the criticality evaluation. The materials will not degrade during the service life of the packaging.

3.1.2.6(d)(3) Computer Codes and Cross-Section Libraries. An appropriate computer code, or other acceptable method, will be used for the criticality evaluation. Standard codes will be referenced. Other codes or methods will be described and any appropriate supplemental information will be provided.

Criticality evaluations will also use an appropriate cross-section library. If multi-group cross sections are used, it will be confirmed that the neutron spectrum of the package has been appropriately considered and that the cross sections are properly processed to account for resonance absorption and self-shielding.

The code will be properly used for the criticality evaluation. Key input data for the criticality calculations will be identified. These include the number of neutrons per generation, number of generations, convergence criteria, mesh selection, etc., depending on the code that is used. At least one representative input file for a single package, undamaged array, and damaged array evaluation will be included in the document. It will be verified that the information from the criticality model, material properties, and cross sections is properly input into the code, and that the calculation has properly converged.

3.1.2.6(d)(4) Demonstration of Maximum Reactivity. The analysis will demonstrate that the most reactive configuration for a single package, an array of undamaged packages, and an array of damaged packages has been analyzed. Any assumptions and approximations will be clearly identified and justified.

The analysis will determine the optimum combination of internal moderation (within the package) and interspersed moderation (between packages), as applicable. It will be confirmed that preferential flooding of different regions within the package is considered as appropriate. Note that the maximum allowable fissile material quantity is not necessarily the most reactive contents.

3.1.2.6(e) Single Package Evaluation.

3.1.2.6(e)(1) Configuration. The criticality evaluation will demonstrate that a single package is subcritical under both NC and AC. The evaluations will consider the following:

- Fissile material in its most reactive credible configuration consistent with the condition of the package, and the chemical and physical form of the contents.
- Water moderation to the most reactive credible extent, including water inleakage to the containment system.

- Full water reflection on all sides of the containment system, or reflection by the package materials, whichever results in the maximum reactivity.

3.1.2.6(e)(2) Results. The package shall meet the requirements outlined in paragraph 3.1.2.6(a). If a package can be shown to be subcritical by reference to a standard in lieu of calculations, it will be verified that the standard is applicable to the package conditions.

3.1.2.6(f) NC Evaluation of Package Arrays.

3.1.2.6(f)(1) Configuration. The criticality evaluation will demonstrate that an array of 5N packages is subcritical under NC. The evaluation will consider the following:

- The most reactive configuration of the array (e.g., pitch and package orientation) with nothing between the packages.
- The most reactive credible configuration of the packaging and its contents under NC. The analysis for the array of undamaged packages may assume that the packages are dry internally, provided that there is no water leakage into the package, including the containment system, when the package is subjected to the tests specified in 10 CFR 71.71.
- Full water reflection on all sides of a finite array.

3.1.2.6(f)(2) Results. The most reactive array conditions will be clearly identified and the results of the evaluation consistent with information presented in the document.

The appropriate N value will be used to determine the TI. The appropriate N should be the smaller value, which ensures subcriticality for 5N packages under NC or 2N packages under AC.

3.1.2.6(g) AC Evaluation of Package Arrays.

3.1.2.6(g)(1) Configuration. The criticality evaluation will demonstrate that an array of 2N packages is subcritical under AC. The evaluation will consider the following:

- The most reactive configuration of the array (e.g., pitch, package orientation, and internal moderation).
- Optimum interspersed hydrogenous moderation.
- The most reactive credible configuration of the packaging and its contents under AC, including inleakage of water.
- Full water reflection on all sides of a finite array.

3.1.2.6(g)(2) Results. The most reactive array conditions will be clearly identified and the results of the evaluation will be consistent with information presented in the document.

The appropriate N value will be used to determine the TI. The appropriate N should be the smaller value that ensures subcriticality for 2N packages under AC or 5N packages under NC.

3.1.2.6(h) Benchmark Evaluations. Computer codes for criticality calculations will be benchmarked against critical experiments. The analysis of the benchmark experiments will use the same computer code, hardware, and cross-section library as those used to calculate the k_{eff} values for the package.

3.1.2.6(h)(1) Applicability of Benchmark Experiments. Benchmark experiments will be applicable to the actual packaging design and contents. Benchmark experiments will have, to the maximum extent possible, the same materials, neutron spectra, and configuration as the package evaluations. Key package parameters that will be compared with those of the benchmark experiments include type of fissile material, enrichment, hydrogen/uranium ratio, poison, and configuration. Differences between the package and benchmarks will be identified and properly considered.

Furthermore, overall quality of the benchmark experiments and uncertainties in the experimental data (e.g., mass, density, dimensions) will be addressed.

3.1.2.6(h)(2) Bias Determination. Results of the calculations for the benchmark experiments and the method used to account for the biases will be discussed, including the contribution from uncertainties in the experimental data.

A sufficient number of appropriate benchmark experiments will be analyzed, and the results of the benchmark calculations will be used to determine an appropriate bias for the package calculations. The benchmark evaluations will address trends in the bias with respect to parameters such as pitch-to-rod diameter, assembly separation, neutron absorber material, etc. Only negative biases (results that under predict k_{eff}) will be considered, with positive bias results treated as zero bias.

Statistical and convergence uncertainties of benchmark calculations will also be considered. The uncertainties will not significantly affect the results.

3.1.2.6(i) Evaluation Criteria for Subcriticality.

3.1.2.6(i)(1) Summary. The CFR does not define the evaluation criteria to determine what is safely subcritical; subsequently, the following criteria are to be used for onsite transport evaluations:

- For equivalency with NRC regulatory guidelines, a k_{eff} limit of 0.95, including consideration of bias and uncertainties, will be used in the criticality evaluation of packages for transportation on the Hanford Site. Packages that are demonstrated to have a k_{eff} limit that exceeds 0.95 require RL approval, except under the following conditions:
 - A k_{eff} limit of 0.965 and 0.98 will be used for unirradiated uranium systems enriched to ≤ 1.25 and ≤ 0.95 weight % ^{235}U , respectively.

- For systems in which the total mass of fissile material per shipment is less than the minimum critical mass of ^{239}Pu (or ^{235}U if no plutonium is present) in aqueous mixtures as specified in ANSI/ANS-8.1-1998 ("Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," p. 5), the criticality evaluation will not include extensive computer evaluations because of the wide acceptance of the limits in ANSI/ANS-8.1-1998.
- For all other systems, the use of a k_{eff} limit greater than 0.95 requires that the criticality safety margin of the system be defined in terms of a parametric curve of the maximum calculated k_{eff} vs the number of packages per shipment. If an increase in the number of packages per shipment results in a small increase in k_{eff} , then the system is on the plateau of the parametric curve, and a higher k_{eff} limit may be justified; however, a k_{eff} limit greater than 0.98 shall not be used.

3.1.2.6(i)(2) Systems With a Small Fissile Mass Content. Based on ANSI/ANS-8.1-1998, extensive computer evaluations of systems are not required when the total fissile mass content per shipment is less than the subcritical limit for fissile isotopes in solution as shown in Table 1 of ANSI/ANS-8.1-1998. For ^{239}Pu , this limit is 0.48 kg (480 g). For ^{235}U this limit is .76 kg (760 g). For mixtures of fissile isotopes, the limit for ^{239}Pu may be used. An administrative safety margin may be used to reduce the maximum allowable fissile mass to account for uncertainties in the fissile mass measurement or for other uncertainties.

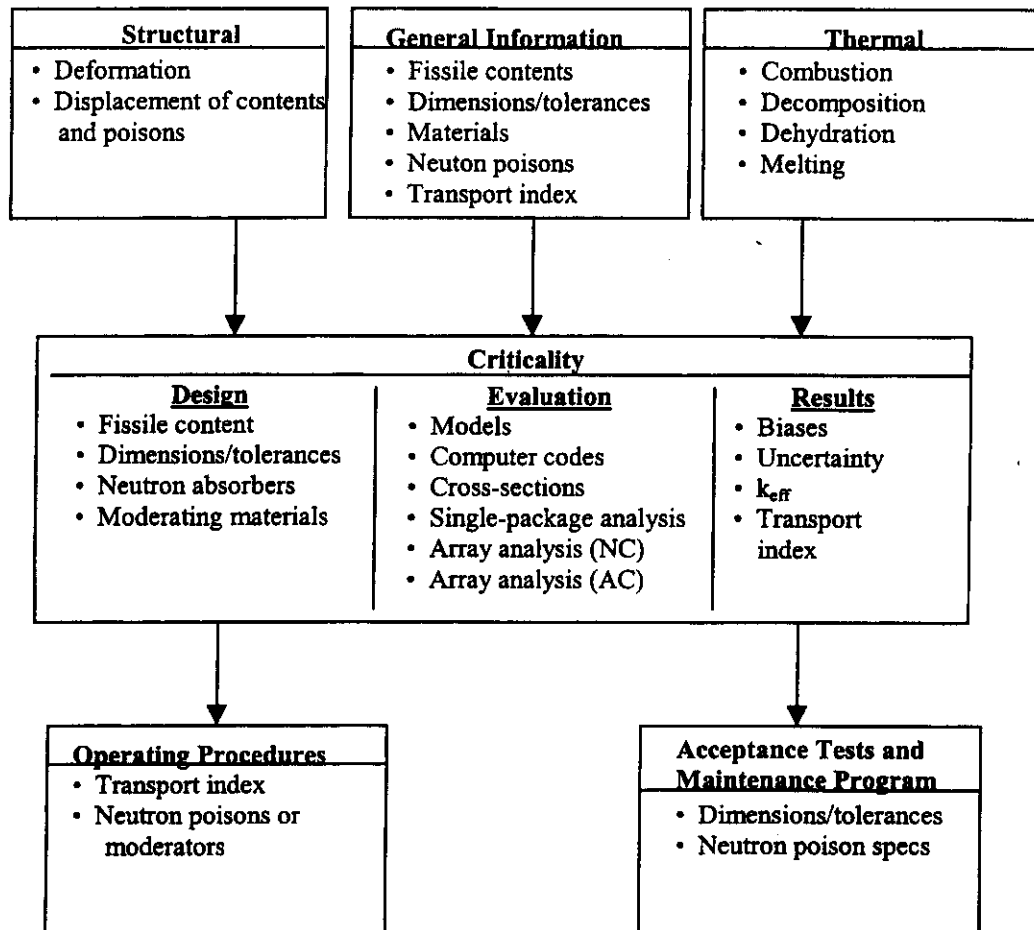
3.1.2.6(j) Evaluation Procedures. The criticality evaluation relies on other inputs/evaluations. In addition, results of the criticality evaluation are considered in the evaluation of other sections. As an aid to the user, an information flow diagram for the criticality evaluation is presented in Figure 7-5.

3.1.2.7 Gas Generation. This section defines the requirements to demonstrate acceptable performance for onsite packaging in which the potentially flammable gas mixture within any confined region of the package exceeds the lower flammability limit (LFL) of 5% (by volume) of the free gas volume. For packages with no potential for flammable gas mixtures, the requirements of this section are not applicable. The acceptable performance requirement is that the package must maintain containment/confinement, subcriticality, and shielding of the contents under both NC and AC. The section also defines the acceptable performance criteria for testing and/or analysis to be applied to demonstrate the packaging provides an equivalent degree of safety to the regulations. It should be noted that in any onsite PSAA, claims of adequacy of designs or design methods must be supported by technical bases, i.e., by an appropriate engineering evaluation or description of actual tests.

3.1.2.7(a) Description of the Package. The following information must be provided in the description of the package:

3.1.2.7(a)(1) The containment/confinement boundary must be exactly defined and confined spaces must be identified.

Figure 7-5. Criticality Evaluation Flow Diagram.



3.1.2.7(a)(2) Design pressure of the containment components must be identified.

3.1.2.7(a)(3) Content of the package and sources of flammable gas generation must be identified.

3.1.2.7(a)(4) Radioactive material source terms and moisture content of the payload must be identified.

3.1.2.7(a)(5) Rate of flammable gas generation.

3.1.2.7(a)(6) Any venting features of the containment/confinement boundary to prevent pressurization.

3.1.2.7(a)(7) Any operational controls such as purging with inert gas prior to shipment or use of getters and/or recombiners shall be identified.

3.1.2.7(a)(8) The void volume of the package must be defined.

3.1.2.7(b) Material Properties and Characteristics. The following information must be provided and/or demonstrated in describing the material properties of the contents and the materials of construction of the packaging:

3.1.2.7(b)(1) Material properties of the packaging must be provided.

3.1.2.7(b)(2) Description of the gas generating sources of the contents must be provided.

3.1.2.7(b)(3) Any potentially flammable mixture of vapors or gases that could form in the containment vessel of a package must be identified and chemical characteristics provided.

3.1.2.7(c) Acceptable Performance Criteria. The fundamental approach is to allow for shipment of potentially flammable mixtures inside a shipping package, provided the primary requirements of containment/confinement, subcriticality, and shielding are satisfied.

3.1.2.7(c)(1) Acceptable performance can be demonstrated by either testing or by analytical methods.

3.1.2.7(c)(2) Packages with potential flammable gas generation of greater than 5% by volume of the containment/confinement boundary gas void over of period of time that is twice the expected shipment time under normal operating conditions must be vented, inerted, or designed to withstand deflagration.

3.1.2.7(c)(3) MNOP, including the maximum gas generated over the life of the package, must be less than the design pressure for the package containment boundary. For a vented package the MNOP is defined as the pressure in the confinement boundary, including the maximum gas generated over a non-vented period of 30 days, with a sealed vent at normal operating temperatures. This pressure must also be less than the design pressure of the package containment portions of the confinement boundary.

3.1.2.7(c)(4) Maximum design pressure for the package containment boundary or the confinement boundary with a sealed vent must be less than the burst pressure.

3.1.2.7(c)(5) Containment boundary components must be demonstrated to maintain containment/confinement, shielding, subcriticality under the sequence of AC performance tests specified in Chapter VI, Section 5.2.5.2.

3.1.2.7(c)(6) Maximum deflagration pressure from the worst-case gas mixture must be less than the containment vessel and components design pressure at normal operating temperatures.

3.1.2.7(c)(7) Containment/confinement boundary burst pressure must not be exceeded under ACs.

3.1.2.7(c)(8) The internal pressure increase from potential flammable gas generation within containment/confinement boundary must not exceed the design pressure under normal operating conditions.

3.1.2.7(c)(9) A deflagration to detonation transition (DDT) must not occur under worst-case gas generation conditions under NC.

3.1.2.7(c)(10) Filter and other pressure relieving components must be capable of sustaining deflagration pressures as well as all NCs and ACs without loss of effectiveness.

3.1.2.7(d) General Considerations. The evaluation of flammable gas mixtures within shipping packages is primarily a gas generation and structural evaluation of the containment/confinement boundary. In general, the evaluation of radiation shielding and criticality should not be impacted by the gas mixture inside the containment/confinement boundary. The evaluation must demonstrate that the package meets the performance test requirements defined in Chapter VI, Section 5.2.5; design requirements defined in Chapter VI, Section 5.2.6; and the requirements provided in paragraph 3.1.2.7(c) above. The evaluation must also address the following:

- The most limiting initial conditions must be used as defined in Chapter VI, paragraph 5.2.6.10(a), and Table 6-6.
- Evaluation methods are appropriate for the loading conditions considered and follow accepted practices and precepts.
- When high-efficiency particulate air filters are used, the minimum airflow rate must be 200 mL/min (12.20 in³/min) at a pressure differential of 1.9 mmHg (1.0 in. H₂O). A 100% verification of flow capacity shall be made on the filters. The flow capacity and traceability to the applicable filters tested must be provided. The minimum hydrogen diffusivity must be 1.90 E-06 mole/sec/mole fraction at 25 °C (77 °F). The filtering efficiencies must be a minimum of 99.97% when subjected to 0.3 µm (0.0118 mil) particles of dioctylphthalate smoke or equivalent. A 100% verification of efficiency must be made on the filters. Documented verification of filtering efficiency and traceability to the applicable items tested must be provided. Each filter must be uniquely numbered.
- When getters and/or recombiners are used, issues and/or conditions that must be addressed include, but are not limited to, the following:
 - Bounding credible scenarios for calculations and/or testing must be assumed.
 - Getter and/or recombiner operational life and capacity before and after shipment must be identified. Communication pathways between the getter and/or recombiner and hydrogen gas (e.g., the pathway for hydrogen gas in a bag inside the package to the getter located in the inner containment vessel headspace) must be identified.
 - Getters and/or recombiners must be compatible with the payload and packaging.
 - Generation of free liquids (water) from the getter and/or recombiner must be addressed.
 - Temperature effects of getter and/or recombiner reaction on the packaging and payload must be considered.

- Gettering reversibility must be addressed.
- Minimum operating pressure related to the getter and/or recombiner performance must be addressed.
- Getter materials and/or recombiner must be engineered into a form (i.e., matrix, structure, and package) so that it can be located within the confines of the package or inner containment vessel.
- It must be shown that any increase in pressure or chemical reactions within the containment vessel due to these vapors or gases could not significantly reduce package effectiveness.
- The maximum quantity of fission gas products that could be available for release in the containment vessel shall be identified. The source term shall be specified, all parameters and assumptions pertinent to the calculation of fission gas products presented, and any data used supported by appropriate references. When it is appropriate, sample calculations will be presented.

3.1.2.7(d)(1) Evaluation by Testing. If the package performance is demonstrated by testing, the onsite PSAA must include the following:

- Objective and description of the test and procedures used. The description must also include initial pressure and temperatures. For packages equipped with filters, ambient pressure may be used. However, normal operating temperature must be used.
- Description of the test equipment used (i.e., ranges, accuracy, gas mixing devices, gas mixture, gas generate rate, and response time).
- Deflagration tests on the containment/confinement vessel may be performed on a separate undamaged unit.
- The sequence of tests specified in Chapter VI, Section 5.2.5.2 must be performed with the initial pressure of the deflagration pressure.
- The test specimen must have been fabricated using the same materials, methods, and QA as specified in the design. Any differences must be identified and the effects evaluated. Substitutes for the contents should have the same representative weight as the actual contents.
- All test results must be evaluated and their implications interpreted, including interior and exterior damage of the test specimen. Unexpected or unexplained test results indicating possible testing problems or non-reproducible specimen behavior must be described and evaluated.
- Videos, photographs, and test data of the tests must be provided as records and maintained available for review.

- Margins of safety of the package design must be evaluated against the test results.
- Criteria for evaluating pass/fail for the test conditions must be provided and used as a comparison with the test results.

3.1.2.7(d)(2) Evaluation by Analysis. If the package performance is demonstrated by analysis, the onsite PSAA must include the following:

- A clear description of the calculations and all assumptions must be provided.
- A clear description of the calculational parameters and methods used to determine gas generation rate and heat generated.
- The methods used for the gas generation evaluation will be appropriate. Any computer programs used will be referenced and evidence of verification and benchmarking shall be provided. Other codes or methods will be described in the document, and the appropriate supplemental information will be provided. The number of dimensions of the code will be appropriate for the package geometry. The models used in the gas generation calculation will be consistent with the effects on the packaging and its contents during NC and AC.
- All source terms will be specified, all parameters and assumptions pertinent to the calculation of the internal pressure presented, and any data used supported by appropriate references. When it is appropriate, sample calculations will be presented.
- The response of the package to loads, in terms of stress, strain, pressure, and temperature to the components and structural members must be provided, and the structural stability of individual members must be demonstrated.
- When used, computer codes must be appropriately used, verified, and benchmarked.
- When using quasi-static analysis techniques, appropriate dynamic amplification factors must be developed and applied.
- Material properties and models must be appropriate for the load combinations evaluated. The material properties must be consistent with the analytical methods. The strain rate at which the properties are given must be justified.
- A summary of results must be provided that compare results of the analyses with the acceptable performance criteria established in paragraph 3.1.2.2(g)(3). For deflagration only, ASME BPVC, Section III, Service Level C limits must be applied as a minimum.

3.1.2.7(e) NC Evaluation.

3.1.2.7(e)(1) Potentially flammable gas generated must be determined for a period of time that is twice the expected shipment time under normal operating conditions. If the generated flammable

gases within this period exceed 5% by volume of the containment vessel component gas void, then the package must be vented, inerted, or designed to withstand deflagration.

3.1.2.7(e)(2) The MNOP, including the maximum gas generated over the life of the package, must be demonstrated to be less than the design pressure for the package containment boundary. For a vented package, the MNOP is defined as the pressure from the maximum gas generation rate over a period of 30 days with a sealed vent. The MNOP must be demonstrated to be less than the design pressure of the containment components of the confinement system.

3.1.2.7(e)(3) Design pressure on the containment boundary or a sealed confinement boundary must be demonstrated to be less than the burst pressure of the system.

3.1.2.7(e)(4) A DDT must be demonstrated not to occur under the worst-case NC gas generation conditions.

3.1.2.7(e)(5) The maximum deflagration pressure from the worst-case gas mixture must be less than the design pressure of the containment boundary or a sealed confinement boundary under normal operating conditions.

3.1.2.7(f) AC Evaluation.

3.1.2.7(f)(1) Containment boundary or sealed components of a confinement boundary must be demonstrated to maintain containment, subcriticality, and shielding under the sequence of AC performance tests.

3.1.2.7(f)(2) Containment boundary or sealed components of a confinement boundary burst pressure must be demonstrated not to be exceeded under ACs. This evaluation must be based on the greatest possible pressure difference.

3.1.2.7(f)(3) Initial pressure at which the sequence of AC tests are to be evaluated must be equal to at least the MNOP under ambient conditions.

3.1.2.8 Tiedown and Load Securement. This section outlines the elements that are to be addressed for an onsite PSAA to demonstrate acceptable tiedown system performance based on the requirements defined in Chapter VI, Section 5.2.6.13. The section also defines the acceptable performance criteria for analysis to be applied. It should be noted that in any tiedown performance assessment, claims of adequacy of designs or design methods must be supported by technical bases; i.e., by an appropriate engineering evaluation or description of actual tests.

3.1.2.8(a) Description of Tiedown System. The following information must be provided in the tiedown description for the package:

3.1.2.8(a)(1) Description of the tiedown system securing the package to the conveyance.

3.1.2.8(a)(2) Weights and center of gravity of the package.

3.1.2.8(a)(3) Description of all tiedown hardware and system for blocking and bracing the package.

3.1.2.8(a)(4) Tiedown system design drawings, sketches, and free body diagrams will be provided for all but the simplest tiedowns used for packages with a small mass.

3.1.2.8(a)(5) Restrictions and other concerns for securing the package will be stated in this section.

3.1.2.8(b) Evaluation Requirements. The following information must be provided in the tiedown evaluation for the package:

3.1.2.8(b)(1) All assumptions will be clearly identified in the evaluation. The basis for and justification of the conservatism for the assumptions shall be provided.

3.1.2.8(b)(2) Required WLLs and ratings for all tiedown assembly components will be determined based on the loadings established by DOT regulations specified in 49 CFR 393.100 or the *North American Cargo Securement Standard*. It should be noted that in the case of the DOT tiedown regulations, no consideration will be given for frictional load resistance between the deck of the conveyance and the package.

3.1.2.8(b)(3) WLLs and/or ratings of all tiedown components (including the conveyance attachment points) and the margins of safety will be provided in tabulated form.

3.1.2.8(b)(4) Minimum load capacity of the conveyance will be established.

3.1.2.8(b)(5) Flexibility of the conveyance and road induced vibration may cause loosening or failure of the tiedowns. As a consequence, the tiedown load paths will be demonstrated to be direct as possible without compromising any required configuration constraints.

3.1.2.8(b)(6) The evaluation will identify all restrictions to the tiedown system and conveyance that must be incorporated into the operating procedures. These restrictions may include interface materials, blocking restrictions, allowable bearing surfaces, feasibility of placing cargo on the package, stacking, temperature restrictions, center of gravity locations, package orientation, or electrical power, pneumatic, or hydraulic requirements.

3.1.2.8(b)(7) The evaluation will identify when a direct tiedown (low pretension, stretching permitted) is being used and when an indirect tiedown (high pretension, limited stretching) is being used.

3.1.2.8(b)(8) As part of the evaluation, an inspection checklist will be developed for incorporation into Section 3.1.2.10 of the safety documentation. As a minimum, the inspection will include a visual inspection of the main conveyance beam, web to flange welds, beam webs, and tiedown components for any signs of distortion or failure prior to release of the package for shipment.

3.1.2.8(b)(9) The evaluation will address the conveyance tiedown points. As conveyance tiedown points are not always rated, the evaluation must identify or determine the load capacity and geometry of loading. The evaluation must also demonstrate conservatism of and basis for any assumptions concerning the capacity of the tiedown points.

3.1.2.9 Quality Assurance. This section identifies the quality requirements to be used when implementing the equivalent packaging requirements. DOE QA requirements for onsite T&P activities are identified in 10 CFR 830, Subpart A; DOE Order 414.1A, and other older orders. The use of alternatives to DOT packagings for onsite shipments is authorized in DOE Order 460.1A. The following paragraphs identify QA requirements to be included in the PSAA and in QAPs for activities associated with the design fabrication, maintenance, storage, and use of onsite Type B and fissile packages.

3.1.2.9(a) Quality Requirements. The QAP should contain the following information:

1. A requirement to develop a QAP by applying the QA criteria specified in paragraph 3.1.2.9(b)(2). The QAP shall include a discussion of how the criteria of paragraph 3.1.2.9(b)(2) will be satisfied. The criteria of paragraph 3.1.2.9(b)(2) shall be applied using a graded approach. The implementing organization(s) shall use appropriate standards, wherever applicable, to develop and implement its QAP. For fissile and Type B quantity packagings, the standard should be NQA-1. The implementing QAP will have the approval of the applicable DOE organization. If the implementing QAP has not been approved by DOE, it will be submitted for approval.
2. An implementing organization may, at any time, make changes to an approved QAP. Changes made over the previous year will be submitted annually to the applicable DOE organization for review. A submittal will identify the changes, the pages affected, the reason for the changes, and the basis for concluding that the revised QAP continues to satisfy the requirements of this section. Changes made to correct spelling, punctuation, or other editorial items do not require explanation.
3. Implementation plans and QAPs will be regarded as approved by DOE 90 days after submittal, unless approved or rejected by DOE at an earlier date, and shall include any modification made or directed by DOE.

3.1.2.9(b) Quality Program and Acceptance Criteria. The package safety documentation will identify that the following criterion are to be implemented by a QAP that implements the quality requirements necessary to ensure the safety associated with the package.

3.1.2.9(b)(1) QA Organization. The users of a packaging are responsible for the establishment and execution of the QAP. The user may delegate to others, such as contractors, agents, or consultants, the work of establishing and executing the QAP, or any part of the QAP, but shall retain responsibility for the program. The user will clearly establish and delineate, in writing, the authority and duties of persons and organizations performing activities affecting the safety-related functions of SSCs. These activities include performing the functions associated with attaining quality objectives and the QA functions.

- The QA functions are as follows:
 - Ensuring that an appropriate QAP is established and effectively executed
 - Verifying, by procedures such as checking, auditing, and inspection, that activities affecting the safety-related functions have been performed correctly.
- The persons and organizations performing QA functions must have sufficient authority and organizational freedom to:
 - Identify quality problems
 - Initiate, recommend, or provide solutions
 - Verify implementation of solutions.
- The persons and organizations performing QA functions will report to a management level that ensures the required authority and organizational freedom, including sufficient independence from cost and schedule, when opposed to safety considerations, are provided.
- Because of the many variables involved, such as the number of personnel, the type of activity being performed, and the location or locations where activities are performed, the organizational structure for executing the QAP may take various forms, provided that the persons and organizations assigned the QA functions have the required authority and organizational freedom.
- Irrespective of the organizational structure, the individual(s) assigned the responsibility for assuring effective execution of any portion of the QAP, at any location where activities subject to this section are being performed, must have direct access to the levels of management necessary to perform this function.

3.1.2.9(b)(2) QAP. The user will establish, at the earliest practicable time consistent with the schedule for accomplishing the activities, a QAP that complies with the requirements of the paragraph. The user shall document the QAP by written procedures or instructions and will carry out the program in accordance with those procedures throughout the period during which the packaging is used. The user will identify the material and components to be covered by the QAP, the major organizations participating in the program, and the designated functions of these organizations.

The user, through its QAP, will provide control over activities affecting the quality of the identified materials and components to an extent consistent with their importance to safety, and as necessary to ensure conformance to the approved design of each individual package used for the shipment of radioactive material. The user will ensure that activities affecting quality are accomplished under suitably controlled conditions. Controlled conditions include the use of appropriate equipment; suitable environmental conditions for accomplishing the activity, such as adequate cleanliness; and assurance that all prerequisites for the given activity have been

satisfied. The user will take into account the need for special controls, processes, test equipment, tools, and skills to attain the required quality, and the need for verification of quality by inspection and test.

The user will base the requirements and procedures of its QAP on the following considerations concerning the complexity and proposed use of the package and its components:

- Impact of malfunction or failure of the item to safety
- Design and fabrication complexity or uniqueness of the item
- Need for special controls and surveillance over processes and equipment
- Degree to which functional compliance can be demonstrated by inspection or test
- Quality history and degree of standardization of the item.

The user will provide for indoctrination and training of personnel performing activities affecting quality, as necessary, to ensure that suitable proficiency is achieved and maintained. The user will review the status and adequacy of the QAP at established intervals. Management of other organizations participating in the QAP will regularly review the status and adequacy of that part of the QAP they are executing.

3.1.2.9(b)(3) Package Design Control. The user will establish measures to ensure that applicable regulatory requirements and the package design, as specified in the approval for those materials and components to which this section applies, are correctly translated into specifications, drawings, procedures, and instructions. These measures must include provisions to ensure that appropriate quality standards are specified and included in design documents and that deviations from standards are controlled. Measures must be established for the selection and review for suitability of application of materials, parts, equipment, and processes that are essential to the safety-related functions of the materials, parts, and components of the packaging.

The user will establish measures for the identification and control of design interfaces and for coordination among participating design organizations. These measures must include the establishment of written procedures, among participating design organizations, for the review, approval, release, distribution, and revision of documents involving design interfaces. The design control measures must provide for verifying or checking the adequacy of design by methods such as design reviews, alternate or simplified calculational methods, or by a suitable testing program. For the verifying or checking process, the user will designate individuals or groups other than those who were responsible for the original design, but who may be from the same organization. Where a test program is used to verify the adequacy of a specific design feature in lieu of other verifying or checking processes, the user shall include suitable qualification testing of a prototype or sample unit under the most adverse design conditions. The user shall apply design control measures to items such as the following:

- Criticality physics, radiation shielding, stress, thermal, hydraulic, and accident analyses
- Compatibility of materials
- Accessibility for in-service inspection, maintenance, and repair
- Features to facilitate decontamination
- Delineation of acceptance criteria for inspections and tests.

The user will subject design changes, including field changes, to design control measures commensurate with those applied to the original design.

3.1.2.9(b)(4) Procurement Document Control. The user will establish measures to ensure that adequate quality is required in the documents for procurement of material, equipment, and services, whether purchased by the user or by its contractors or subcontractors. To the extent necessary, the user will require contractors or subcontractors to provide a QAP consistent with the applicable provisions of this TSD.

3.1.2.9(b)(5) Instructions, Procedures, and Drawings. The user will prescribe activities affecting quality by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall require that these instructions, procedures, and drawings be followed. The instructions, procedures, and drawings must include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

3.1.2.9(b)(6) Document Control. The user will establish measures to control the issuance of documents such as instructions, procedures, and drawings, including changes, which prescribe all activities affecting quality. These measures must ensure that documents, including changes, are reviewed for adequacy, approved for release by authorized personnel, and distributed and used at the location where the prescribed activity is performed. These measures must ensure that changes to documents are reviewed and approved.

3.1.2.9(b)(7) Control of Purchased Material, Equipment, and Services. The user will establish measures to ensure that purchased material, equipment, and services, whether purchased directly or through contractors and subcontractors, conform to the procurement documents. These measures must include provisions, as appropriate, for source evaluation and selection, objective evidence of quality furnished by the contractor or subcontractor, inspection at the contractor or subcontractor source, and examination of products on delivery.

The user will have available documentary evidence that material and equipment conform to the procurement specifications before installation or use of the material and equipment. The user will retain, or have available, this documentary evidence for the life of the package to which it applies. The user will ensure that the evidence is sufficient to identify the specific requirements met by the purchased material and equipment.

The user will assess the effectiveness of the control of quality by contractors and subcontractors at intervals consistent with the importance, complexity, and quantity of the product or services.

3.1.2.9(b)(8) Identification and Control of Materials, Parts, and Components. The user will establish measures for the identification and control of materials, parts, and components. These measures must ensure that identification of the item is maintained by heat number, part number, or other appropriate means, on the item or on records traceable to the item, as required throughout fabrication, installation, and use of the item. These identification and control measures must be designed to prevent the use of incorrect or defective materials, parts, and components.

3.1.2.9(b)(9) Control of Special Processes. The user will establish measures to ensure that special processes, including welding, heat treating, and nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements.

3.1.2.9(b)(10) Internal Inspection. The user will establish and execute a program for inspection of activities affecting quality by or for the organization performing the activity, to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity. The inspection must be performed by individuals other than those who performed the activity being inspected. Examination, measurements, or tests of material or products processed must be performed for each work operation where necessary to ensure quality. If direct inspection of processed material or products is not carried out, indirect control by monitoring processing methods, equipment, and personnel must be provided. Both inspection and process monitoring must be provided when quality control is inadequate without both. If mandatory inspection hold points, which require witnessing or inspecting by the user's designated representative and beyond that work should not proceed without the consent of its designated representative are required, the specific hold points must be indicated in appropriate documents.

3.1.2.9(b)(11) Test Control. The user will establish a test program to ensure that all testing required to demonstrate that the packaging components will perform satisfactorily in service is identified and performed in accordance with written test procedures that incorporate the requirements of this TSD and the requirements and acceptance limits contained in the package approval. The test procedures must include provisions for ensuring that all prerequisites for the given test are met, that adequate test instrumentation is available and used, and that the test is performed under suitable environmental conditions. The user will document and evaluate the test results to ensure that test requirements have been satisfied.

3.1.2.9(b)(12) Control of Measuring and Test Equipment. The user will establish measures to ensure that tools, gauges, instruments, and other measuring and testing devices used in activities affecting quality are properly controlled, calibrated, and adjusted at specified times to maintain accuracy within necessary limits.

3.1.2.9(b)(13) Handling, Storage, and Shipping Control. The user will establish measures to control, in accordance with instructions, the handling, storage, shipping, cleaning, and preservation of materials and equipment to be used in packaging to prevent damage or deterioration. When necessary for particular products, special protective environments, such as inert gas atmosphere, and specific moisture content and temperature levels, must be specified and provided.

3.1.2.9(b)(14) Inspection, Test, and Operating Status. The user will establish measures to indicate, by the use of markings such as stamps, tags, labels, routing cards, or other suitable means, the status of inspections and tests performed upon individual items of the packaging. These measures must provide for the identification of items that have satisfactorily passed required inspections and tests where necessary to preclude inadvertent bypassing of the inspections and tests.

The user will establish measures to identify the operating status of components of the packaging, such as tagging valves and switches, to prevent inadvertent operation.

3.1.2.9(b)(15) Nonconforming Materials, Parts, or Components. The user will establish measures to control materials, parts, or components that do not conform to the user's requirements to prevent their inadvertent use or installation. These measures must include, as appropriate, procedures for identification, documentation, segregation, disposition, and notification to affected organizations. Nonconforming items must be reviewed and accepted, rejected, repaired, or reworked in accordance with documented procedures.

3.1.2.9(b)(16) Corrective Action. The user will establish measures to ensure that conditions adverse to quality, such as deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of a significant condition adverse to quality, the measures must ensure that the cause of the condition is determined and corrective action taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken must be documented and reported to appropriate levels of management.

3.1.2.9(b)(17) Quality Assurance Records. The user will maintain sufficient written records to describe the activities affecting quality. The records must include the instructions, procedures, and drawings required by paragraph 3.1.2.9(b)(5) to prescribe QA activities and must include closely related specifications such as required qualifications of personnel, procedures, and equipment. The records must include the instructions or procedures that establish a records retention program that is consistent with applicable regulations and designates factors such as duration, location, and assigned responsibility. The user will retain these records for 3 years beyond the date when the user last engages in the activity for which the QAP was developed. If any portion of the written procedures or instructions is superseded, the user will retain the superseded material for 3 years after it is superseded.

3.1.2.9(b)(18) Audits. The user will carry out a comprehensive system of planned and periodic audits to verify compliance with all aspects of the QAP, and to determine the effectiveness of the program. The audits must be performed in accordance with written procedures or checklists by appropriately trained personnel not having direct responsibilities in the areas being audited. Audited results must be documented and reviewed by management having responsibility in the area audited. Follow-up action, including reaudit of deficient areas, must be taken where indicated.

3.1.2.9(c) Quality Program Contents and Format.

3.1.2.9(c)(1) Graded Application of the QAP. For fissile and Type B quantity packaging, the package safety documentation will (using NQA-1 including the supplements as the bases) provide a cross-referencing index, which demonstrates that each of the 18 criteria identified in paragraphs 3.1.2.9(b)(1) through (18) are addressed by written procedures. An example of such a matrix is presented in Table 1, Regulatory Guide 7.10. Because of the inter-relationship of the 18 criteria, more than one quality procedure generally will be applicable to each criterion.

The graded application process shall determine the appropriate level of analysis, documentation, and actions necessary to comply with the requirements through the consideration of prescribed factors, such as the following:

- Nuclear safety classification or hazard category of the item or activity
- Relative importance to safety, safeguards, and security
- Magnitude of any hazard or risk involved
- Adequacy of existing safety documentation
- Impact/consequences of failure on programmatic mission of a facility
- Particular characteristics of a package or activity
- Life-cycle stage of a package
- Complexity of items, services, or processes involved
- History of problem at a facility, or with an item, service, or process.

The graded application process will not be used to circumvent applicable QA, legal, or contractual requirements. Rather, the grading determines the extent to which controls within the QA criteria are applied. The graded application of QAP requirements is normally achieved through a combination of the following:

- Extent to which procedures, instructions, or specifications define the processes or work methods involved
- Extent of assessment, verification, review, or oversight activities
- Extent of documentation required
- Degree of control over activities.

For all packagings, the importance to safety should be based primarily on the ability of the package to provide the following:

- Containment of radioactive material
- Subcriticality of fissile material
- Shielding of radiation.

The graded approach should consider the complexity and proposed use of the package and its components as described in the following:

- Impact of malfunction or failure of the item to safety
- Design and fabrication complexity or uniqueness of the item
- Need for special controls and surveillance over processes and equipment
- Degree to which functional compliance can be demonstrated by inspection or test
- Quality history and degree of standardization of the item.

3.1.2.9(c)(2) Graded Approach for SSCs Important to Safety. For all fissile and Type B quantity packaging, the package safety documentation will provide a package-specific listing (Q-List) of all SSCs important to safety. The listing for SSCs will be consistent with the parts list or similar information presented in the packaging drawings. Justification should be provided for any item identified on the drawings, but not defined as important to safety in the Q-list. The package safety documentation will identify a quality category (e.g., A, B, C) for each SSC important to safety and these categories will be appropriately defined. The assigned categories will be properly justified based on their definition, the package type, and the safety function of each SSC. Guidance on the application of categories and QA requirements is provided in paragraph 3.1.2.9(c)(1). Additional guidance can be found in Appendix A of Regulatory Guide 7.10. Definitions of typical categories and representative safety classifications for SSCs of transportation packagings are also presented in Table 2 and Table 5, respectively, of NUREG/CR-6407.

3.1.2.9(c)(3) Package-Specific Quality Criteria and Package Activities. For fissile and Type B quantity packages, the package safety documentation will address each of the 18 quality criteria in NQA-1 as they apply to the proposed package. The package safety documentation will identify for each criterion, as applicable, the appropriate level of effort for package activities based on their importance to safety. Guidance on QA requirements applicable to each category is provided in Appendix A of Regulatory Guide 7.10. Other guidance is presented in Chapter 4 of NUREG/CR-6407, which also describes typical design and fabrication records maintained for each QA category. Table 7-1 identifies typical levels of effort for each of the 18 criteria of NQA-1 that should be considered in the review, based on quality category. Note that the omission of Category C items from the QA effort may not be appropriate if they involve a condition of approval specified in the package approval document.

Table 7-1. Typical Level of QA Effort by Quality Category.

QA Element/Level of Effort	Category A	Category B	Category C
1. QA Organization			
Responsibility established	X	X	X
Authority and duties written	X	X	X
QA functions executed	X	X	X
Reporting levels clearly defined	X	X	X
Independence from cost and schedule assured	X	X	X
2. QA Program			
Procedures written	X	X	X
Activities affecting quality controlled	X	X	X
Graded approach established	X	X	X
Indoctrination and training provided	X	X	X
3. Design Control			
Most stringent codes and standards	X		
Codes and standards	X	X	
Prototype test and/or analysis	X	X	
Formal design review	X	X	
Internal peer review	X	X	
Off-the-shelf items			X
Conditions of approval controlled	X	X	X
4. Procurement Document Control			
Traceability	X	X	
Qualified vendor lists	X		
Off-the-shelf items			X
5. Instructions, Procedures, and Drawings			
Written and documented	X	X	
Qualitative or quantitative acceptance criteria	X	X	
Changes to conditions of approval listed in certificate controlled	X	X	X
6. Document Control			
Controlled issue	X	X	
Controlled changes	X	X	
7. Control of Purchased Material, Equipment, and Services			
Source evaluation and selection	X		
Inspection at contractor	X		
Formal receiving inspection	X	X	
Audits or surveillance at vendor plants	X		
Evidence of QA at contractor	X	X	
Objective proof that all specifications are met	X	X	X
Incoming inspection for damage only			X
8. Identification and Control of Materials, Parts, and Components			
Positive identification and traceability	X		
Identification and traceability to heats, lots, or other groupings		X	
Identification to end use drawings			X
9. Control of Special Processes			
Welding, heat treating, and nondestructive examination performed by qualified personnel	X		
Qualification records and training of personnel	X		
Only specified critical operations by qualified persons		X	
No special processes			X

Table 7-1. Typical Level of QA Effort by Quality Category.

QA Element/Level of Effort	Category A	Category B	Category C
10. Internal Inspection			
Documented inspection of all specifications	X		
Process monitoring if required by quality	X		
Examination, measurement, or test of material or processed product to assure quality	X	X	
Inspectors independent of those performing operations	X	X	
Qualified inspectors only	X	X	
Visual receiving inspection only			X
11. Test Control			
Written test program	X	X	
Written test procedures	X	X	
Documentation of testing and evaluation	X	X	
Observation of supplier acceptance tests as appropriate	X		
12. Control of Measuring and Test Equipment			
Tools, gauges, and instruments in formal calibration program	X	X	
13. Handling, Storage, and Shipping Control			
Written plans and procedures	X	X	
Routine handling			X
14. Inspection, Test, and Operating Status			
Individual items identified as to status or condition	X	X	
Status indicated by stamps, tags, labels, etc.	X	X	
Visual examination only			X
15. Nonconforming Materials, Parts, or Components			
Written procedures to prevent inadvertent use	X	X	
Nonconformance documented and closed	X	X	
Disposal without records			X
16. Corrective Action			
Conditions adverse to quality identified and corrected	X	X	X
Cause and corrective action documented	X	X	
Safety significant events reported	X	X	X
17. QA Records			
Design and use records	X	X	
Results of reviews, inspections, tests, audits, surveillances, and materials analysis	X	X	
Personnel qualifications	X	X	
Records of fabrication retained for life of package plus 3 years	X	X	
Records of acceptance testing retained for life of package plus 3 years	X	X	
Records of maintenance retained for life of package plus 3 years	X	X	
Shipping records retained for 3 years after shipment	X	X	X
Records managed by a written procedure for retention and disposal	X	X	X
18. Audits			
Written plan of periodic audits	X	X	X
Implementation by written procedures	X	X	X
Lead auditor certified	X	X	
All auditors certified	X		

3.1.2.9(c)(4) Requirements for Quality Categories for Packagings. Quality categories will be based on the relative safety significance of each Q component and item and, where appropriate, their subcomponent parts. The category should be identified as "A" for components and items that are critical to safe operation. For components and items with a major impact on safety, the category will be identified as "B." Category C should be used for components and items with a minor impact on safety. Examples of Category A items are SSCs whose failure or malfunction could result directly in a condition adversely affecting public or worker health and safety. Adverse conditions are defined as loss of containment, loss of shielding, or unsafe geometry compromising criticality control. Category B items could be SSCs whose failure or malfunction could indirectly result in a condition adversely affecting public or worker health and safety. For the failure of a Category B item to result in an adverse condition, a second failure of a component or item would need to occur. Category C items would be those SSCs whose failure or malfunction would not significantly reduce the packaging effectiveness and would be unlikely to create a condition adversely affecting public or worker health and safety. Each component, item, structure, or subsystem of a packaging will be tabulated and designated with the proper category. The QA requirements will also be identified for each component, item, structure, or subsystem.

3.1.2.10 Operating Procedures. This section addresses the contents of the operating procedures that are to be provided.

3.1.2.10(a) Operating Procedures for DOT-Equivalent Packaging. The guidance outlines the elements that are to be addressed for an onsite PSAA to demonstrate the operating procedures for safe operations of Type B and fissile packages. The guidance, when applied using a graded approach, is appropriate for any packagings requiring safety documentation.

3.1.2.10(b) Summary of Operation Requirements and Restrictions. This section will provide requirements and restrictions related to package operations. All requirements and restrictions related to package operations will be provided including: drawing and revision numbers for all package components; a description of the form (solid, liquid, powder, etc.) and fissile load of the contents; any package handling restrictions (such as lifting height limits or dual load path requirements; any neutron poison, moderator, and gap requirements; expected gamma and neutron radiation levels, and locations of any streaming paths; and any closure (such as closure lid bolt torques necessary for containment) and component testing (such as pressure taps) requirements.

3.1.2.10(c) General Information. The operation procedures will list any special equipment required for handling the particular package and include all pertinent details listed below:

3.1.2.10(c)(1) The operating procedures will include appropriate quantitative and qualitative criteria for determining that important activities have been satisfactorily accomplished.

3.1.2.10(c)(2) Detailed implementing procedures will be provided. It is recommended that the packaging safety documentation identify NUREG/CR-4775 as guidance for preparation of those procedures.

3.1.2.10(d) Package Loading. Loading of the package will include inspections, tests, and special preparations. Inspections will include those made prior to loading the package to determine that the package is not damaged and surface contamination levels are within the allowable limits. The procedures will discuss the inspection of the gaskets, criteria for replacement, and, if applicable, procedures for repair. The inspection of each closure device and criteria for replacement will also be included.

The operating procedures will provide instructions on package loading. Instructions about the following items (as applicable) will be included in the loading instructions:

3.1.2.10(d)(1) Preparation.

- Appropriate documents to be reviewed by operating personnel
- The package will be loaded and closed in accordance with written procedures
- Describe any special controls and precautions for handling
- Verify that the package is in unimpaired physical condition and that all required periodic maintenance has been performed
- Ensure that the package is conspicuously and durably marked with the model number, serial number, gross weight, and package identification number, as applicable
- Ensure use of the package complies with all other conditions of approval
- Determine the package is proper for the contents to be shipped, including the need for canning of damaged fuel or for a second containment vessel, if applicable
- Ensure packaging sealing surfaces have been properly prepared and protected
- Check that all payload treatment processes performed subsequent to package loading are appropriate for the particular payload in question, and process equipment operators are familiar with both processing procedures and task operating procedures
- Check that packaging interior contamination levels are not so excessive that significant contamination could be imparted to the payload itself
- Check that all components operate as designed and have been tested as necessary
- Identify any special equipment that may be needed (listing) when loading the package.

3.1.2.10(d)(2) Loading of Contents.

- Describe any special controls and precautions for loading
- Indicate the method of loading the contents
- Ensure that any required moderator or neutron absorber is present and in proper condition
- When applicable, specify the method to remove water from the package, as appropriate
- Vent gases from the package or add fill gas, as appropriate
- Ensure that each closure device of the package, including seals and gaskets, is properly installed, secured, and free of defects
- Identify bolt torques consistent with those identified by the design, and/or shown on the drawings
- Confirm the package has been loaded and closed appropriately
- Conduct any required package testing prior to release for transport
- Decontaminate exterior surfaces of the packaging and transport vehicle.

3.1.2.10(e) Shipment Preparation. *The operation procedures will provide instructions on shipment preparation. The instructions will include the following items, as a minimum:*

- Securing the package to vehicle, including acceptable tiedown configurations
- Rendering unused lifting devices inoperable
- Conducting the package radiation survey, including a determination of surface contamination
- When applicable, conducting the temperature survey to verify that limits specified are not exceeded
- Specify the assembly verification leakage rate, and how to ensure package closures are leak tested in accordance with applicable closure standards such as ANSI N14.5
- Ensure that any system for containing liquid is properly sealed and has adequate space or other specified provision for expansion of the liquid
- Verify that any pressure relief device is operable and set

- Ensure that any structural component that could be used for lifting or tiedown during transport is rendered inoperable for those purposes unless it meets the required design requirements
- Ensure that the tamper-indicating device is installed
- Specify the attachment of impact limiters, personnel barriers, or similar devices, as applicable
- For a fissile material shipment, identify any special controls and precautions for transport, loading, unloading, and handling, and any appropriate actions in case of an accident or delay
- Identify any special controls that will be provided to the carrier for a package shipped by exclusive use under the provisions
- Identify any special controls that will be provided to the carrier for a fissile-material package
- Describe any special instructions that should be provided to the consignee for opening the package
- Ensure that the TI for each package and the sum of the TIs for the shipment are appropriate for the type of shipment.

3.1.2.10(f) Package Receipt. The operation procedures will give instructions to the package recipient on the following items:

- Ensure that the package is examined for visible damage, status of the tamper-indicating device, surface contamination, and external radiation levels are made
- Describe any special actions to be taken if the package is damaged, if the tamper-indicating device is not intact, or if surface contamination or radiation survey levels are too high
- Identify any special handling equipment needed
- Describe any proposed special controls and precautions for handling and unloading.

3.1.2.10(g) Package Unloading. Package unloading will include inspections, tests, and special preparations. As applicable, the procedures used to ensure the safety removal of fission gases, contaminated coolant, and solid contaminants will be discussed. Also, as applicable, any required cool-down procedures will be described, and it will be shown that it does not affect the continued use of the package.

Operating procedures will provide detailed instructions for package unloading. Instructions on the following items shall be included:

- A list of any special equipment that may be needed
- Preparation of the package for removal from the transport vehicle
- Proper removal of the package from the transport vehicle
- Package transfer from the transportation vehicle to the unloading site
- Describe the appropriate method to open the package
- Identify the appropriate method to remove the contents
- Ensure that the contents are completely removed
- Inspection of the interior of the packaging after contents have been removed
- Possible preparation of packaging for short- or long-term empty storage.

3.1.2.10(h) Preparation of Empty Packages for Transport. A discussion of inspections, tests, and special preparations of the packaging necessary to ensure safe transport of the empty package will be included. Instructions on the following items will be included:

- Verify package is empty
- Verify requirements for non-fixed radioactive contamination limits have been met
- Verify requirements for fixed-radioactive contamination limits have been met
- Properly close packaging
- Ensure packaging is delivered, to storage or a carrier, in a condition such that subsequent transport will not reduce the effectiveness of the packaging (e.g., damage to sealing surfaces caused by the freezing of moisture not properly removed).

3.1.2.10(i) Inspections and Maintenance. The operating procedure will establish how the maintenance and acceptance requirements are to be implemented.

3.1.2.10(j) Records and Reporting Requirements.

3.1.2.10(j)(1) Records for Each Shipment. The DOE requires the maintenance of records for each shipment. The operating procedures will identify the applicable DOE and company procedures, and the records they require to be maintained.

3.1.2.10(j)(2) Records of Package History. Historic records will be maintained for the lifetime of the package plus 3 years. Historic records will consist of sufficient QA records to furnish documentary evidence of the quality of packaging components that have safety significance and of services effecting quality. Records to be maintained include results of the following determinations: new package shielding meets design requirements; containment will maintain integrity at 1.5 times the MNOP (where applicable); packaging was fabricated in accordance with the approved design; and results of monitoring, inspections, and auditing of work

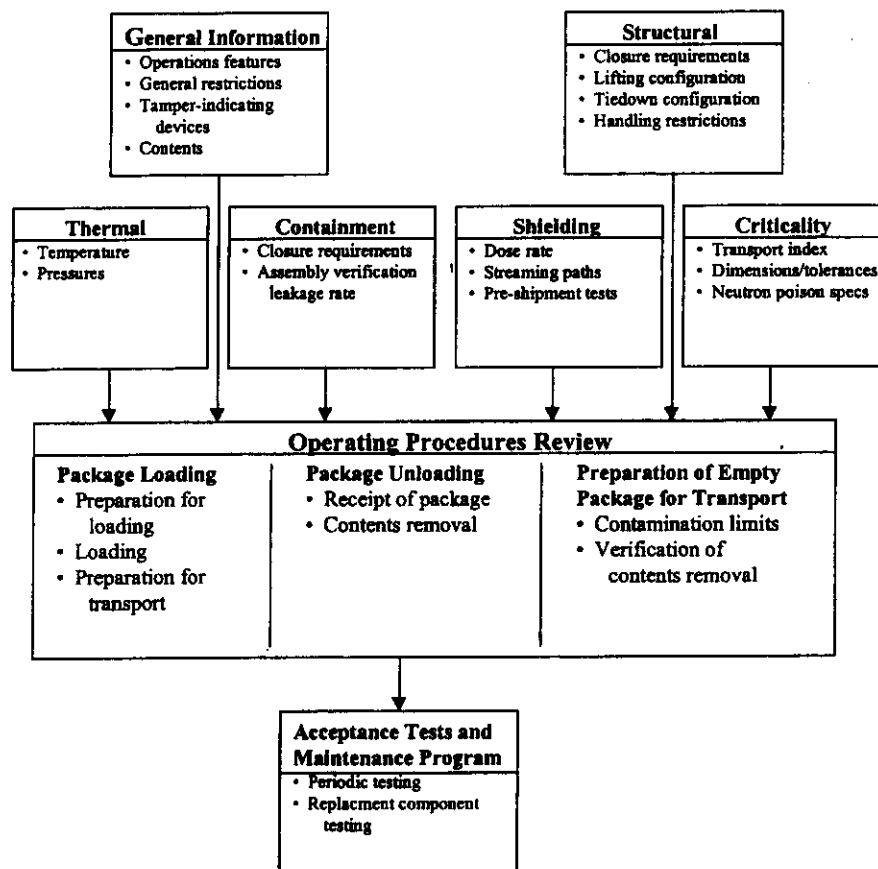
performance during design, fabrication, assembly, testing, modification, maintenance and repair of the packaging.

3.1.2.10(j)(3) Reports. Operating procedures will identify package specific conditions that require reporting identified in the DOE orders or Hanford Site procedures.

3.1.2.10(k) Operation Checklist. Operating procedures will present a checklist by the organization shipping the package. That organization is responsible for ensuring that operating procedures are in place before using the container. The checklist will identify the requirements that must be incorporated into procedures at the time of shipment.

3.1.2.10(l) Evaluation Procedures. The operations evaluation of a package is based in part on the description and evaluations presented in other evaluations. In addition, results of the operations evaluation are considered in the evaluation of all other sections. As an aid to the user, an information flow diagram for the operations evaluation is presented in Figure 7-6.

Figure 7-6. Operations Evaluation Flow Diagram.



3.1.2.11 Acceptance Tests and Maintenance Program. This section addresses the contents of the acceptance and maintenance procedures.

The guidance outlines the elements that are to be addressed in the onsite PSAA to demonstrate the acceptance tests and maintenance program required for safe operation of Type B and fissile packages. In addition, the guidance, when applied using a graded approach, is appropriate for any packaging requiring safety documentation:

- Structural and pressure tests
- Leakage rate tests
- Shielding tests
- Thermal tests
- Other tests.

3.1.2.11(a) Acceptance Test Procedures. Test procedures associated with the acceptance of the package for use that are required to be conducted prior to and during the service life of the packaging will be identified. When practicable, national standard procedures will be utilized. In the absence of such codes, the basis and rationale used to formulate the QAP will be described in establishing the requirements. The acceptance tests will include the following:

- Visual inspections and measurements
- Weld examinations
- Component tests
- Material tests.

The frequency of the tests will be identified. The requirements should provide flexibility in the schedule to provide for times when the shipping window will extend beyond the due date for the test.

The acceptance test will support the following verifications:

- Before first use, fabrication of each packaging must be verified to be in accordance with the approved design. [49 CFR 173.474 and 10 CFR 71.85(c)]
- Before first use, each packaging must be inspected for cracks, pinholes, uncontrolled voids, or other defects that could significantly reduce its effectiveness. [49 CFR 173.474 and 10 CFR 71.85(a)]
- Before first use, if the MNOP of a package exceeds 35 kPa (5 psig), the containment system of each packaging must be tested at an internal pressure at least 50% higher than MNOP to verify its ability to maintain structural integrity at that pressure. [10 CFR 71.85(b)]

- When applicable, before first use, each packaging must be conspicuously and durably marked with its model number, serial number, gross weight, and a package identification number. [10 CFR 71.85(c)]
- Additionally, perform any tests deemed appropriate by the approval authority. [10 CFR 71.93(b)]

3.1.2.11(a)(1) Acceptance Tests. When the following acceptance tests are specified, present (as a minimum) a description of the test and its acceptance criteria. Standards will be referenced or an alternate basis provided.

3.1.2.11(a)(1)(i) Visual Inspections and Measurement. Provide inspections to verify that the packaging has been fabricated and assembled in accordance with the drawings. Dimensions and tolerances specified on the drawings will be confirmed by measurement.

3.1.2.11(a)(1)(ii) Weld Examinations. Provide welding examinations to verify fabrication in accordance with the drawings, codes, and standards specified in the design documents. Location, type, and size of the welds will be confirmed by measurement. When appropriate, specifications for weld performance, nondestructive examination, and acceptance will be identified.

3.1.2.11(a)(1)(iii) Structural and Pressure Tests. Identify and describe any applicable structural or pressure tests. Such tests will comply with 10 CFR 71.85(b), as well as applicable codes or standards specified in the design documentation.

3.1.2.11(a)(1)(iv) Leakage Tests. When applicable, identify that the containment system of the packaging will be subjected to a fabrication leakage test. The sensitivity of these tests must be identified. Those tests specified in ANSI N14.5 should be used when applicable. Require that all closures, including drains and vents, be leak-tested. The acceptable leakage criterion will be consistent with that identified in Section 3.1.2.4, Containment/Confinement Evaluation.

3.1.2.11(a)(1)(v) Component and Material Tests. Appropriate tests and acceptance criteria will be specified for components that affect package performance. Examples of such components include seals, gaskets, valves, fluid transport systems, and rupture disks or other pressure-relief devices. Components will be tested to meet the performance specifications shown on the engineering drawing of the package. When tests adversely affect the continued performance of a component, applicable QA procedures will be described to justify that the tested component is equivalent to the component that will be used in the packaging.

Verify that appropriate tests and acceptance criteria are specified for packaging materials. Tests for neutron absorbers (e.g., boron, gadolinia) and insulating materials (e.g., foams, fiberboard) will ensure that minimum specifications for density and isotopic content are achieved. Materials will be tested to meet the performance specifications shown on the engineering drawings.

3.1.2.11(a)(1)(vi) Shielding Tests. Appropriate shielding tests will be specified for both neutron and gamma radiation. The tests and acceptance criteria will be sufficient to ensure that no voids or streaming paths exist in the shielding.

3.1.2.11(a)(1)(vii) Thermal Tests. Appropriate tests will be specified to demonstrate the heat transfer capability of the packaging. These tests will confirm that the heat transfer performance determined in the evaluation is achieved in the fabrication process.

3.1.2.11(b) Maintenance Program. A maintenance program for ensuring continuous safe performance of the packaging will be identified. The packaging must be maintained in unimpaired physical condition except for superficial defects such as marks or dents [49 CFR 173.475 and 10 CFR 71.87(b)]. The program will include periodic testing, inspection, and replacement schedules, as well as criteria for replacement and repair of components and subsystems on a routine or as-needed basis. Frequency of maintenance will be identified. Requirements will provide flexibility in the schedule to provide for times when the shipping window will extend beyond the due date for the maintenance.

3.1.2.11(b)(1) Structural and Pressure Tests. If required by the package, structural and pressure tests to be performed and the frequency of performance will be identified. When a test(s) is used, the instrumentation and test sensitivity must be identified. The periodic and maintenance leakage rate tests (when applicable) will be those specified in ANSI N14.5. The acceptable leakage rate criterion will be consistent with that identified in Section 3.1.2.4, Containment/Confinement Evaluation. When appropriate, replacement schedules for seals will be identified.

3.1.2.11(b)(2) Leak Tests. Leak tests that are to be performed and the frequency of performance will be identified. For most packaging systems, this would include a test of the package before each shipment and an annual test of each packaging. The sensitivity of these tests must be identified. Use of the leakage rate tests specified in ANSI N14.5 is recommended. The acceptable leakage rate criterion will be consistent with that identified in the containment section (3.1.2.4). When appropriate, replacement schedules for seals will be identified.

3.1.2.11(b)(3) Subsystem Maintenance. The test and replacement schedule to be used for packaging subsystems (e.g., auxiliary cooling systems and neutron shield tanks) whose inadequate performance could impair the total package safety will be described. The schedules established will be justified using verifiable test or manufacturer's data.

3.1.2.11(b)(4) Valves, Rupture Disks, and Gasket on the Containment Vessel. The test and replacement schedule for the components on the containment vessel will be specified. The schedules that are established will be justified using verifiable test or manufacturer's data. For most systems, this would include, as a minimum, a visual inspection before each closure, and an annual gasket and seal replacement.

3.1.2.11(b)(5) Shielding. The test and inspection schedules will be described, as well as the corrective action that is to be used to ensure adequate shielding performance. Both gamma and neutron sources will be considered.

3.1.2.11(b)(6) Thermal. The tests proposed and the frequency of the tests that will be performed on the total system will be described. It will be shown the proposed total frequency will detect degradation in the thermal performance of the packaging before compromising package safety.

3.1.2.11(b)(7) Miscellaneous. Any additional test not previously considered that will be performed periodically on components and subsystems will be described.

3.1.2.11(c) Acceptance Criteria.

3.1.2.11(c)(1) Visual Inspection. Visual inspections will be performed and the intended purpose for each inspection discussed. The criteria will be clearly stated for acceptance of each inspection, as well as the action that will be taken if noncompliance is encountered.

3.1.2.11(c)(2) Structural and Pressure Tests. There will be a description of how the tests are to be performed. The acceptance criteria will be included in the document. The actions that are to be taken when the prescribed criteria are not met will also be included. An estimate of the sensitivity of the tests will be provided.

3.1.2.11(c)(3) Leak Tests. A description of the leak tests to be performed will be included in the document. Leak tests will be performed on the containment vessel as well as auxiliary equipment such as shield tanks. The criteria for acceptance and the action to be taken, if the criteria are not met, will be described. The sensitivity of the tests will be estimated, and a basis for the estimate given.

3.1.2.11(c)(4) Component Tests. The tests for the components will be discussed in the document. Acceptance criteria will be provided and the action to be taken if the criteria are not met will be discussed.

3.1.2.11(c)(4)(i) Valves, Rupture Disks, and Fluid Transport Devices. Valves, rupture disks, and fluid transport devices will be tested under the most severe service conditions for which the package design assumes their acceptable performance. When the tests are presumed to adversely affect the continued performance of a component, the results of the tests on components of the same model and type may be substituted.

3.1.2.11(c)(4)(ii) Gaskets. Gaskets will be tested under conditions simulating the most severe service conditions under which the gaskets are to perform. Because these acceptance tests may degrade the performance of the gasket under test or the package into which it is assembled, or both, the tests are not necessarily performed on gaskets or packages to be put into service. The simulation system must ensure adequate representation of those conditions that would prevail if the actual system were used in a test. Gaskets must be procured under a QAP that is adequate to ensure acceptance testing of a given gasket device is equivalent to acceptance testing of all gaskets supplied and identified by that manufacturer as that model gasket.

3.1.2.11(c)(4)(iii) Miscellaneous. Any component that is not listed in paragraphs 3.1.2.11(c)(4)(i) and 3.1.2.11(c)(4)(ii) whose failure would impair the package effectiveness will be tested under the most severe conditions for which the package was designed. Because these acceptance tests may degrade the performance of the component under test or the system into which it is assembled, or both, the tests are not necessarily performed on components or systems to be put into service. The simulation system shall ensure adequate representation of those conditions that would prevail if the actual system were used in this test. Furthermore, components must be procured under a QAP adequate to ensure that acceptance testing of a given component device is equivalent to acceptance testing of all devices supplied and identified by that manufacturer as that model device.

3.1.2.11(c)(5) Tests for Shielding Integrity. The tests that are to be performed to establish shielding for both gamma and neutron sources will be discussed. The discussion will include the dimensions of the grid pattern or a description of the scanning procedure that demonstrates the inspection of 100% of the package surface area. The acceptance criteria, as well as the action to be taken if the criteria are not met, will be described.

3.1.2.11(c)(6) Thermal Acceptance Tests. The tests to verify that each package performs within some defined variance in accordance with the results of the thermal analyses or tests for NC will be discussed.

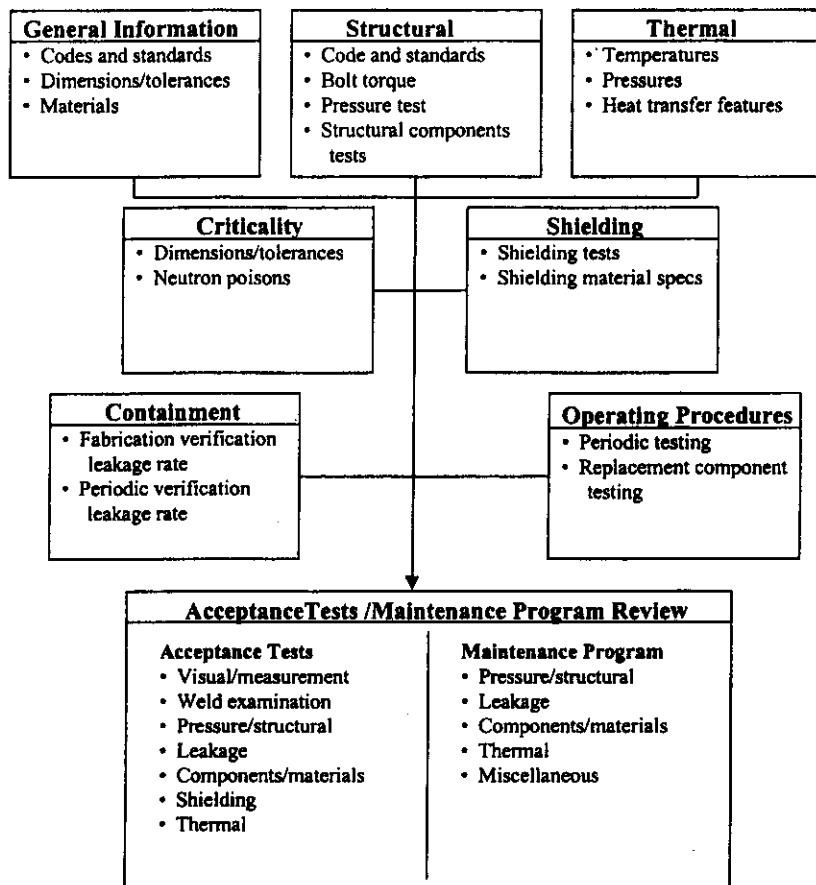
3.1.2.11(c)(6)(i) Discussion of Test Setup. The description of the tests will include heat source, instrumentation, and schematic showing thermocouple and heat source locations, as well as the placement of other test equipment. The test sensitivity will be estimated based on instrumentation, test item, and environmental variations.

3.1.2.11(c)(6)(ii) Test Procedures. The procedures used in testing and data recording will be discussed. The frequency of data recording during the test will be reported. The criteria used to define the steady-state (thermal equilibrium) condition of the test item will also be discussed.

3.1.2.11(c)(6)(iii) Acceptance Criteria. The thermal acceptance criteria and the method employed to compare the acceptance test results with the predicted thermal performance will be discussed. The action to be taken if the thermal acceptance criteria are not met by the packaging unit will also be discussed.

3.1.2.11(d) Evaluation Procedures. The acceptance tests and maintenance program evaluation of a package is based in part on the description and evaluations presented in the above sections. In addition, results of the acceptance tests and maintenance program evaluation are considered in the evaluation of all other sections. As an aid to the user, an information flow diagram for the acceptance tests and maintenance program evaluation is presented in Figure 7-7.

Figure 7-7. Acceptance Tests and Maintenance Program Evaluation Flow Diagram.



3.2 NONEQUIVALENT PACKAGINGS

When using the alternative onsite nonequivalent packaging requirements, the following evaluation and acceptance criteria must be used. These requirements may be used for demonstrating equivalent safety for IP-2, IP-3, Type A, fissile and Type B packagings.

Before conducting the consequence and frequency evaluations, the proposed package must be evaluated against the applicable equivalent package requirements of Chapter VI, Section 5.2. Applicability shall be based on proposed contents. The evaluation shall be documented in a report and formatted as identified in Section 3.1.2. This report shall document the package's performance; showing both the requirements met and those failed. Where the package does not meet an applicable requirement, discuss the performance provided. To this basic report, add the consequence and failure frequency information. The consequence and failure frequency evaluations shall be developed in accordance with the methodologies provided in Sections 3.2.1 and 3.2.2. The consequence and failure frequency information provided shall be detailed enough to support the risk-based approval decision.

3.2.1 Evaluation Requirements and Acceptance Criteria for Nonequivalent Risk-Based Packages

This section outlines the elements that are to be addressed for an onsite PSAA to demonstrate acceptable radiological and toxicological safety, and/or acceptable risk based on the requirements identified in Chapter VI, Section 5.3.2. It also defines the acceptable analysis to be conducted to demonstrate the packaging provides an equivalent degree of safety and acceptable level of risk.

Appendix G, Justification and Basis for Shipment of Risk-Based Packages, contains the technical arguments or justification to validate the equivalence of the packaging performance. It also presents the accident scenario data to be applied when conducting an analysis.

3.2.1.1 Consequence Analysis Methodology. This section covers the calculation of dose consequences and package specific release limits. Package specific release limits are calculated such that the maximum dose to the offsite receptor and co-located worker do not exceed that in Table 6-7. The calculation of the package specific release limits is similar to the calculation of dose consequences resulting from an intake, so both are discussed in this section.

To promote consistency between consequence analyses for different packages, an approved methodology for calculating the intake and dose consequences from transportation accidents is provided.

The consequence analysis determines the radiological consequence, or dose, to the maximally exposed onsite and offsite receptors from a radioactive material release during an accident. Note that the maximally exposed onsite receptor is defined as a person within the Hanford Site boundary at least 100 m (328 ft) from the accident. As a minimum, the consequence analysis shall quantitatively evaluate impact and fire accidents. The severity of these accidents shall be at least as severe as to cause complete failure of the package, but no more severe than the accident test conditions for certified Type B packages described in 10 CFR 71.73. Although puncture, crush, and immersion tests are also described in the regulations, the impact event is normally more severe than the puncture and crush events, and immersion is neglected because it is not a credible accident scenario on the Hanford Site.

The consequence analysis considers the credible pathways through which the onsite and offsite receptors may be exposed to radiation. In the case where one pathway dominates the dose, the evaluation of other pathways may be less rigorous or qualitative. All receptors may be exposed to radioactive material by inhaling particulate matter and being submersed in a particulate aerosol carried downwind. Direct external dose from unshielded photons, beta particles, and neutrons is normally negligible at distances of 100 m (328 ft) or greater.

Exposure from consuming contaminated food or drinking contaminated water is not considered, because state and federal emergency response procedures (DOE-0223, *Emergency Plan Implementing Procedure*; WSDOA 1994, *Fixed Nuclear Facility Emergency Response Procedure*; WSDOH 1993, "Response Procedures for Radiation Emergencies;" EPA 1992, *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents*) are in place to

prevent ingestion of contaminated food and water in the event of an accident. The primary determinant of exposure from the ingestion pathway is the effectiveness of public health measures (i.e., interdiction) rather than the severity of the accident itself.

The material that leaks from the damaged package shall account for the physical and chemical form of the waste material during the AC using appropriate airborne release fractions and respirable fractions. The consequence analysis may include the mitigating effect of the damaged package after the AC using damage ratios or leak path factors, provided that a structural analysis supports the mitigating effect. If these factors are used, then both the mitigated and the unmitigated dose consequences shall be reported.

In addition to the onsite receptor, immediate workers in the vicinity of the accident (i.e., less than 100 m [328 ft]) may be exposed to radiation. The immediate worker for transportation cannot be defined in the same sense as for facilities. This is because the location of a release of radioactive material during transportation may potentially occur at any point along the transportation route between the originating and receiving facilities. The shipment may pass by several facilities along the route, and each shipment may pass by different facilities depending on the particular originating and receiving facilities. In addition, the shipment may pass by other vehicles along the route. In the event of an accident, it is foreseeable that people in other vehicles may stop to try to assist the driver involved in the accident, unknowingly being exposed to the released radioactive material. Furthermore, calculations of inhalation dose are subject to widely varying uncertainties. Therefore, because the immediate worker is not easily defined, consequence evaluations to workers in the immediate vicinity of an accident (i.e., less than 100 m [328 ft]), are addressed qualitatively.

The offsite receptor is the maximally exposed member of the public at the Hanford Site boundary for an exposure duration of not less than 2 hours. This location and duration is the same as required in facility evaluations for determining the safety classification of SSCs (DOE-STD-3009-94, p. A-3). On the Hanford Site, only badged traffic is allowed to pass through the barricades, although members of the public with visitor's badges may be escorted onsite, and public access is not restricted on state highways 24, 240, and 243. These highways are considered throughways with only transient public usage as described in Scott (1995), *Clarification of Hanford Site Boundaries for Current and Future Use in Safety Analyses*, and Route 4 South and Route 10 leading up to the Wye Barricade are also treated in this manner. In the 300 Area, members of the public may be on Route 4 South and on Cypress St., but beyond the Central Badging Office, signs state that badges are required, and at the fenceline signs state that only government vehicles are allowed beyond the fenceline. Public access to the Columbia River within the Hanford Site boundaries is also not restricted. Direction from RL in Klein (2000) *Contract No. DE-AC06-96RL13200 - Revision of Criteria for Section of Safety-Class Structures, Systems and Components for Fluor Hanford, Inc.*, specifically excludes considering onsite locations where a member of the public might be at the time of the accident as the site boundary for consequence evaluations, although the contractor shall evaluate the accident dose consequences at the near shore of the Columbia River and Highway 240 or other locations in order to recommend prudent protective features over or beyond those protective measures already invoked. This site boundary definition is consistent with the accident analysis in the facility SARs, although SARs include the consequences at other locations for information.

3.2.1.1.1 Inhalation and Submersion Dose. Because risk-based packages are by definition not shown to maintain containment during ACs, all or a fraction of the contents may escape the containment barrier. The amount of material that could escape the containment barrier and be released to the environment depends on the accident scenario, the mitigative capability of the packaging, the physical and chemical form of the waste material, and the total radiological inventory of the package.

The dose consequence to the offsite receptor and the co-located worker at 100 m (328 ft) due to inhalation is calculated with the following equation as described in HNF-SD-WM-ISB-007, *Central Waste Complex Interim Safety Basis*, Section 5.4, or through the use of a computer program such as GENII v. 1.485 (PNL-6584, *GENII - The Hanford Environmental Radiation Dosimetry Software System, Volume 2: Users' Manual*; WHC-SD-WM-TI-596, *Verification Tests for the July 1993 Revision to the GENII Radionuclide and Dose Increment Libraries*; and WHC-SD-WM-TI-675, *Test Documentation for GENII Software Version 1.485*).

$$D = S \times \frac{\chi}{Q'} \times BR \times DCF$$

where:

- D = 50-year committed effective dose equivalent due to inhalation and submersion, in rem
- S = airborne source term, respirable, in dose-equivalent curie (DE-Ci)
- χ/Q' = time-integrated atmospheric dispersion coefficient, in seconds/cubic meter (s/m^3)
- BR = breathing rate, in m^3/s
- DCF = inhalation dose conversion factor, in rem/DE-Ci

Explanatory Material for the IAEA Regulations for the Safe Transport of Radioactive Material (IAEA 1990) bases its allowable release rate during NC and allowable release during AC on an intake by a member of the public of 10^{-6} A₂. Intake is defined as the amount of radioactive material taken into the body. The intake by the offsite receptor is the product of the airborne respirable source term, the atmospheric dispersion coefficient, and the breathing rate.

$$\text{Intake} = S \times \frac{\chi}{Q'} \times BR$$

Thus, the dose is simply the intake multiplied by the appropriate dose conversion factor.

$$D = \text{Intake} \times DCF$$

The source term for inhalation calculations is the amount of respirable material that has escaped containment and is available for dispersive transport to a receptor location. A source term for a specific release mechanism is equal to the product of the material at risk (MAR), the damage ratio (DR), the airborne release fraction (ARF), the respirable fraction (RF), and the appropriate

leakpath factor (LPF) (DOE-HDBK-3010-94, Section 1.2). The general equation for the source term is as follows:

$$S = \text{MAR} \times \text{DR} \times \text{ARF} \times \text{RF} \times \text{LPF}$$

MAR is the maximum radiological inventory of the package or array of packages involved in the accident. Daughter products of the parent isotopes should be considered in the MAR. The DR is the fraction of the MAR (i.e., the fraction of the contents of a single package, or the fraction of the packages in the array) actually affected by the accident-generated conditions and may conservatively be assumed to be 100%, unless information is available to justify a lower DR.

The bases for the ARFs and RFs associated with the postulated accident scenario are taken from DOE-HDBK-3010-94. This DOE document provides reasonable factors for estimating the radiological source terms in deterministic analyses of upper-bound, worst-case, accident scenarios postulated for nonreactor facilities using experimental data for different release mechanisms.

The largest release fraction for solid payloads is for loose surface-contamination on metal equipment in an impact. According to Section 5.1 of DOE-HDBK-3010-94, contaminated equipment that does not undergo brittle fracture responds to an impact event by shock-induced vibration of the substrate, and an ARF of 1×10^{-3} and RF of 1.0 are considered bounding. Other solid payloads have lower release fractions. For example, according to Section 4.1 of DOE-HDBK-3010-94, for chemically-nonreactive (i.e., noncombustible) compounds in powder form subjected to thermal stress, an ARF of 6×10^{-3} and RF of 0.01 are considered bounding based on the entrainment of powder by the upward flow from the heated surface. Release fractions for various material forms and accidents are shown in Table 7-2.

The LPF is the cumulative fraction of airborne radioactive material that escapes through the containment boundary; therefore, the LPF accounts for fractional release of contents. In general, the LPF may conservatively be assumed to be 100%, unless information is available to justify a lower number. When the leak path cross-sectional area through the containment boundary is small ($\sim 1 \text{ mm}^2$), such as that which may occur from the failure of an elastomer seal when subjected to high mechanical loads, the mass deposition rate of vapors and particles onto cask interior surfaces is rapid compared to the mass rate of their release from the cask to the environment (NUREG/CR-6672, p. 7-21). Thus, unless cask depressurization is rapid (which may occur in a fire scenario), deposition of vapors and large particles onto cask interior surfaces will be efficient, which will lower the amount of aerosol released.

If credit is taken for the capability of some or all of the packages in the array to withstand the accident generated conditions, or for a reduction in the airborne source term due to filtration and deposition that the aerosol may experience prior to escape through the containment boundary, then these factors shall be clearly identified. Furthermore, both the mitigated and the unmitigated dose consequences shall be reported.

Table 7-2. Summary of Release Fractions for Common Payloads.

Payload	ARF x RF
Grout	
Drop, 9 m (30 ft) drop height (p. 4-6)	4.14E-05
Fire (pp. 4-5, 4-48)	1.30E-05
Equipment, surface contaminated metal	
Drop (p. 5-7)	1.00E-03
Fire (p. 5-5)	6.00E-05
Trash, surface contaminated combustible	
Drop (p. 5-4)	1.00E-04
Fire (p. 5-1)	5.00E-04
Powder, non-reactive	
Drop, 3 m (10 ft) (p. 4-9)	6.00E-04
Fire (p. 4-7)	6.00E-05
Liquid, aqueous solution	
Drop (p. 3-4)	1.00E-04
Fire, boiling (p. 3-1)	2.00E-03
Liquid, slurry or sludge	
Drop (p. 3-4)	4.00E-05

Pages from DOE-HDBK-3010-94, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities*, Change Notice 1, U.S. Department of Energy, Washington, D. C.

The airborne emission not only leads to an inhalation dose at the receptor location while the plume passes, but also some long-term inhalation due to resuspended activity. The dose from inhalation of resuspended activity is shown in HNF-SD-SNF-TI-059, *A Discussion on the Methodology for Calculating Radiological and Toxicological Consequences for the Spent Nuclear Fuel Project at the Hanford Site* (p. 3-14), to be small compared to the dose from inhalation of the passing plume. Factors that reduce the dose from inhalation of resuspended activity include a low rate of deposition from the plume to the ground, a low resuspension factor of deposited material, weathering of surface contamination into locations that are not exposed, and a low exposure time due to removal of any measured contamination following surveys of the accident zone. Therefore, the dose, due to resuspension, can be ignored.

Once the material has been released to the environment, it then is transported through the air to downwind receptors. Atmospheric transport of aerosols is discussed in NUREG/CR-3332, and ORNL-5968, Chapter 2). For safety analyses, the process is conservatively simplified to neglect mechanisms that remove radionuclides from the airborne plume, such as deposition onto ground, vegetation, or buildings; rainout; and washout.

With the source term identified, the other terms in the consequence equation can be evaluated. In the equation for the offsite receptor, the parameter χ/Q' (sometimes referred to as the atmospheric dispersion coefficient) defines the downwind concentration, integrated over time, normalized to the amount released, at a particular downwind location. This parameter characterizes the dilution between a release point and a receptor and is calculated in accordance with NRC guidance (Regulatory Guide 1.145). Hanford Site-specific meteorological data on

wind speed, direction, and atmospheric stability have been tabulated over the years 1983 to 1991 and are used to generate χ/Q' values for various distances in each of 16 compass sectors from potential release locations on the site. Note that the Gaussian diffusion theory on which χ/Q' is based is not considered valid at distances less than 100 m (328 ft). The values are derived such that there is a 99.5% probability that the meteorological conditions in any sector, on any given day, are bounded. The calculation of χ/Q' may be performed by a computer program such as GXQ v. 4.0 (WHC-SD-GN-SWD-30002, *GXQ 4.0 Program Users' Guide*; WHC-SD-GN-SWD-30003, *GXQ Program Verification and Validation*), as long as Regulatory Guide 1.145 is implemented.

The locations of the maximally exposed offsite individuals for several potential release locations onsite are shown in Table 7-3, as well as some onsite locations, along with the distance to these receptors and the associated atmospheric dispersion coefficients. The values in Table 7-3 are taken from the applicable facility SAR for ground level releases.

The dose consequence to the offsite receptor is calculated assuming the release occurs in the location that causes the largest dose consequence to the offsite receptor. If the shipment occurs anywhere onsite, then the largest dose consequence to the offsite receptor occurs for a release in the 300 Area, due to its proximity to the Columbia River. Table 7-4 shows the value of χ/Q' at distances of 100 m (328 ft) to 800 m (2,652 ft) from a release in the 300 Area calculated using the computer code GXQ v. 4.0 (WHC-SD-GN-SWD-30002 and WHC-SD-GN-SWD-30003). The GXQ output file is given in Appendix G, Section 1.5. The northern sector, i.e., the sector directly north of a release, bounds the airborne concentration in all sectors for each of these distances.

The third term in the equation for the dose consequence to the offsite receptor is the breathing rate. The breathing rate for the reference man during light activity from the International Commission on Radiological Protection (ICRP 1975) (20 L/min or $3.33 \times 10^{-4} \text{ m}^3/\text{s}$) normally applies during the 16 hours of the day when the man is assumed to be awake. This value exceeds the average breathing rate during resting (7.5 L/min or $1.25 \times 10^{-4} \text{ m}^3/\text{s}$), which applies during the 8 hours of the day when the man is assumed to be asleep. The 24-hour average breathing rate ($2.64 \times 10^{-4} \text{ m}^3/\text{s}$) normally applies to offsite receptors when the release is continuous for at least 24 hours. Breathing rates are also given for an adult woman and a 10-year old child, but these are less than the rate for an adult man, which would reduce the intake and would be less conservative.

Table 7-3. Locations and Atmospheric Dispersion Coefficients for Onsite and Maximally Exposed Offsite Individuals.

Release Location Receptor Location	χ/Q' , s/m ³
100 K Area, K West Reactor and Adjacent Buildings ^a	
Onsite Worker, 100 m (328 ft) E	7.32×10^{-2}
Columbia River, 520 m (1,706 ft) W	3.55×10^{-3}
100 Area Fire Station, 3,750 m ESE	1.60×10^{-4}
Hanford Site boundary, 10,070 m (33,038 ft) W	4.49×10^{-5}
100 K Area, Cold Vacuum Drying Facility ^a	
Onsite Worker, 100 m (328 ft) E	7.32×10^{-2}
Columbia River, 650 m (2,132 ft) W	2.44×10^{-3}
100 Area Fire Station, 3,750 m (12,303 ft) ESE	1.60×10^{-4}
Hanford Site boundary, 10,090 m (33,103 ft) W	4.48×10^{-5}
200 East Area, Canister Storage Building ^a	
Onsite Worker, 100 m (328 ft) E	3.41×10^{-2}
Highway 240, 9,280 m (30,446 ft) W	2.36×10^{-3}
Hanford Site boundary, 17,390 m (57,054 ft) E	1.30×10^{-5}
200 West Area, Central Waste Complex ^b	
Onsite Worker, 800 m (2,625 ft), SE	8.99×10^{-4}
Hanford Site boundary, 11,700 m (38,386 ft) W	1.76×10^{-5}
200 West Area, Plutonium Finishing Plant ^c	
Onsite Worker, 950 m (3,117 ft) ESE (fire)	5.98×10^{-5}
Onsite Worker, 450 m (1,476 ft) WNW (rupture)	1.22×10^{-3}
Hanford Site boundary, 17,400 m (57,087 ft) NNW (fire)	1.16×10^{-6}
Hanford Site boundary, 12,500 m (41,010 ft) W (rupture)	1.62×10^{-5}
300 Area, 324 Building ^d	
Onsite Worker, 100 m (328 ft) N	1.4×10^{-2}
Hanford Site boundary, west bank of the Columbia River, 305 m (1,001 ft) E (N was used)	2.9×10^{-3}
300 Area, 327 Building ^e	
Onsite Worker, 100 m (328 ft) (N is assumed)	4.26×10^{-2}
Hanford Site boundary, west bank of the Columbia River, 549 m (1,801 ft) E	4.32×10^{-3}

^a HNF-3553, *Spent Nuclear Fuel Project, Final Safety Analysis Report*, Rev. 0, Fluor Hanford, Richland, Washington, p. 3-23.

^b HNF-SD-WM-ISB-007, *Central Waste Complex Interim Safety Basis*, Rev. 1-C, Fluor Hanford, Richland, Washington, p. 5-8.

^c HNF-SD-CP-SAR-021, *Plutonium Finishing Plant Final Safety Analysis Report*, Rev. 2, Fluor Hanford, Richland, Washington, p. T9-23. Note that separate χ/Q' s are given for fire and mechanical rupture accidents, where the χ/Q' s for fire take credit for plume rise. The worst-case meteorology for buoyant releases is different than for non-buoyant releases, which is why the receptor locations are not the same.

^d HNF-SD-SPI-SAR-001, *324 Building Safety Analysis Report*, Rev. 3, Fluor Hanford, Richland, Washington, p. 6-47; also see HNF-4608, *324 Building Calculation Notebook*, Rev. 1, Fluor Hanford, Richland, Washington. The χ/Q' at the site boundary was selected as the highest sector value at the minimum distance "in accordance with guidance contained in HNF-PRO-704."

^e HNF-SD-SPI-SAR-002, *327 Building Safety Analysis Report*, Rev. 2A, Fluor Hanford, Richland, Washington, p. B-22. Note that the χ/Q' s for the 327 Building are based on 95% worst-case meteorology, whereas the χ/Q' s for all other facilities are based on 99.5% worst-case meteorology.

Table 7-4. χ/Q' as a Function of Direction and Distance From a Release in the 300 Area.

Sector	Distance							
	100 m	200 m	300 m	400 m	500 m	600 m	700 m	800 m
S	3.18E-02	9.58E-03	4.81E-03	2.96E-03	2.04E-03	1.50E-03	1.16E-03	9.29E-04
SSW	1.16E-02	3.49E-03	1.75E-03	1.08E-03	7.41E-04	5.47E-04	4.23E-04	3.38E-04
SW	4.39E-03	1.32E-03	6.62E-04	4.08E-04	2.81E-04	2.07E-04	1.60E-04	1.28E-04
WSW	3.88E-03	1.13E-03	5.56E-04	3.36E-04	2.28E-04	1.66E-04	1.22E-04	9.81E-05
W	1.51E-02	4.44E-03	2.23E-03	1.38E-03	9.52E-04	7.05E-04	5.48E-04	4.41E-04
WNW	2.90E-02	8.74E-03	4.39E-03	2.70E-03	1.86E-03	1.37E-03	1.06E-03	8.48E-04
NW	3.28E-02	9.88E-03	4.96E-03	3.05E-03	2.10E-03	1.55E-03	1.20E-03	9.59E-04
NNW	3.29E-02	9.92E-03	4.98E-03	3.07E-03	2.11E-03	1.56E-03	1.20E-03	9.63E-04
N	4.21E-02	1.27E-02	6.38E-03	3.93E-03	2.70E-03	1.99E-03	1.54E-03	1.23E-03
NNE	2.89E-02	8.73E-03	4.38E-03	2.70E-03	1.86E-03	1.37E-03	1.06E-03	8.47E-04
NE	2.93E-02	8.82E-03	4.43E-03	2.73E-03	1.88E-03	1.38E-03	1.07E-03	8.57E-04
ENE	1.60E-02	4.78E-03	2.40E-03	1.48E-03	1.02E-03	7.52E-04	5.82E-04	4.67E-04
E	3.16E-02	9.51E-03	4.78E-03	2.94E-03	2.02E-03	1.49E-03	1.15E-03	9.23E-04
ESE	2.99E-02	9.03E-03	4.53E-03	2.79E-03	1.92E-03	1.42E-03	1.09E-03	8.76E-04
SE	3.16E-02	9.53E-03	4.79E-03	2.95E-03	2.03E-03	1.49E-03	1.16E-03	9.25E-04
SSE	3.23E-02	9.74E-03	4.89E-03	3.01E-03	2.07E-03	1.53E-03	1.18E-03	9.46E-04

The last term in the equation for the dose consequence to the offsite receptor is the dose conversion factor. This factor relates the amount of material inhaled (intake) to the 50-year committed effective dose equivalent. This factor has units of rem/DE-Ci inhaled, where the DE-Ci is a conversion factor derived in HNF-EP-0063, *Hanford Site Solid Waste Acceptance Criteria*, to normalize the exposure risk from inhalation of various radionuclides to the risk from inhalation of ^{239}Pu . Because the risk depends on the lung clearance class, which is determined by the chemical form of the plutonium, two different dose conversion factors are given. For plutonium oxide or plutonium that is oxidized during a fire, which is an insoluble form, the dose conversion factor is 3.08×10^8 rem/DE-Ci (HNF-SD-WM-ISB-007). For non-oxidized plutonium, which is considered soluble (lung clearance class W), the dose conversion factor is 4.292×10^8 rem/DE-Ci (HNF-SD-WM-ISB-007, Section 5.4). For other elements, the consequence evaluation shall use the DCF for non-oxidized plutonium, because that is the DCF used to derive the DE-Ci conversion factors listed in HNF-EP-0063.

The inhalation dose to immediate workers can be qualitatively evaluated by multiplying the inhalation dose at 100 m (328 ft) by a factor of 30 (IAEA, 1990, p. 78). Note that this is only an estimate because of the large uncertainties in the airborne concentration at distances within a 100 m (328 ft) radius.

3.2.1.1.2 Intake from ACs. To show an equivalent degree of safety to the DOT Hazardous Material Regulations, the intake of radioactive material by the maximally exposed individual at the Hanford Site boundary is limited to 10^{-6} A₂ from either NC or AC. This intake is the basis for the allowable release rates in 10 CFR 71.51.

Intake is the product of the airborne respirable source term, the integrated normalized downwind airborne concentration (referred to as the atmospheric dispersion coefficient in SARs), and the breathing rate. The derivation in IAEA considered airborne respirable release fractions in the range of 10^{-3} to 10^{-2} , and uptake fractions (i.e., the fraction of the released aerosol that is taken into the body) in the range of 10^{-4} to 10^{-3} . The uptake fraction is defined as the product of the breathing rate and the atmospheric dispersion coefficient. For an accident with a respirable airborne fraction of 10^{-3} , an uptake of 10^{-3} would be required to get an equivalent intake of 10^{-6} A_2 as considered in the IAEA derivation.

The following equations summarize the definitions of intake and uptake.

$$\text{Intake} = Q \times \text{ARF} \times \text{RF} \times \text{Uptake}$$

where:

Intake = the amount released that is taken into the body = 10^{-6} A_2
 Q = $\text{MAR} \times \text{DR} \times \text{LPF}$ = total amount released from package
 = 1 A_2 maximum allowed by DOT
 Uptake = fraction of the respirable aerosol taken into body = 10^{-3}
 $\text{ARF} \times \text{RF}$ = fraction of the amount released that is respirable aerosol = 10^{-3} .

$$\text{Uptake} = \chi/Q' \times \text{BR}$$

where:

BR = breathing rate = 3.3×10^{-4} m^3/s
 χ/Q' = atmospheric dispersion coefficient, s/m^3 , dependent on release location.

A degree of safety equivalent to the regulations for ACs is shown by restricting the intake by the maximally exposed member of the public at the Hanford Site boundary to the same as that associated from a certified package in ACs on public highways, namely 10^{-6} A_2 . This is done by restricting the total amount in the package available to release. Substituting for the uptake and solving for Q gives:

$$Q = \frac{\text{Intake}}{\text{BR} \times \chi/Q' \times \text{ARF} \times \text{RF}}$$

Because of the increased distance to the public from a release on the Hanford Site compared to a release on public highways, a release larger than 1 A_2 must occur to cause an equivalent public intake of 10^{-6} A_2 as considered in the IAEA derivation of the A_2 values used by DOT. Equivalency as defined in this document depends on the distance to and location of the offsite receptor and the location of the release; therefore, Q will vary by area.

In the 300 Area, the location of the maximally exposed individual along the Hanford Site boundary is off the near bank of the Columbia River, which is as close as 300 m (984 ft) to the route of a possible shipment. The atmospheric dispersion coefficient at 300 m (984 ft) from a release in the 300 Area is 6.38×10^{-3} s/m^3 , as calculated with the computer program GXQ v. 4.0

(WHC-SD-GN-SWD-30002 and WHC-SD-GN-SWD-30003). Note that GXQ incorporates the recommendations of the NRC Regulatory Guide 1.145. Based on these conservative assumptions, the maximum amount that may be released from a risk-based package in an accident in the 300 Area that results in an equivalent public intake of $10^{-6} A_2$ is $471 A_2$, as shown in Table 7-5.

Table 7-5. Example Calculations for Risk-Based Packages Based on Equivalent Public Exposure to a Release from ACs.

Area of Shipment	Maximum AC Release for Negative USQ_T	Public A_2 Intake	Airborne respirable fraction	Breathing Rate, m^3/s	χ/Q' s/m^3
300 Area	$4.71 \times 10^2 A_2$	10^{-6}	10^{-3}	3.33×10^{-4}	6.38×10^{-3}
200 Areas only	$1.71 \times 10^5 A_2$	10^{-6}	10^{-3}	3.33×10^{-4}	1.76×10^{-5}
100 K Area	$6.70 \times 10^4 A_2$	10^{-6}	10^{-3}	3.33×10^{-4}	4.48×10^{-5}

In the 200 Area, the location of the maximally exposed individual along the Hanford Site boundary is 11.7 km (7.3 mi) west of the 200 West Area (HNF-SD-WM-ISB-007), with an atmospheric dispersion coefficient of $1.76 \times 10^{-5} s/m^3$. The maximum amount that may be released from a risk-based package in an accident in the 200 Area that results in an equivalent public intake of $10^{-6} A_2$ is $1.71 \times 10^5 A_2$, as shown in Table 7-5.

The atmospheric dispersion coefficient at locations in between areas is bounded by the coefficient at either the originating or receiving facility.

The limits for risk-based packages depend on where the package may be shipped, as shown in Table 7-5. In this example, an unmitigated release is assumed, although a mitigated release may be justified provided a structural analysis or other technical basis supports the mitigating effect. Furthermore, the fraction of the contents that forms a respirable aerosol is conservatively taken to be 10^{-3} , whereas the respirable airborne release fraction for grout, surface contaminated equipment, surface contaminated trash, and powders are all less than or equal to that value (DOE-HDBK-3010-94).

The example calculations in Table 7-5 are derived only from equivalency to the ACs containment requirement for public exposure; criteria derived from equivalency to the NCs containment requirement for public exposure are given in the following section.

3.2.1.1.3 Intake from NCs. The previous section addresses the public during ACs, but not during NCs. 10 CFR 71.51 requires that certified Type B packages be designed, constructed, and prepared for shipment so that under the tests specified in 10 CFR 71.71, "Normal conditions of transport," there would be no loss or dispersal of radioactive contents as demonstrated to a sensitivity of $10^{-6} A_2/h$. It is possible that some risk-based packages fail only the DOT containment requirement for ACs; i.e., they may meet the DOT containment requirements for NCs. For risk-based packages that do not meet the containment requirements for NCs, the intake associated with a leaking package during NCs at the location of the maximally exposed member of the public shall not exceed the intake of a member of the public from a certified DOT package

in the immediate vicinity of a package in commerce. In the derivation of the release rate from Type B packages during NCs of transport (IAEA 1990, p. 107-108), the release of 10^{-6} A₂/h is based on a member of the public spending 20% of his working time (400 h/yr = 20% x 2,000 h/yr) working around a leaking certified Type B package in an enclosed vehicle of 50 m³ volume with 10 air changes per hour. Note that in commerce, the public includes those who work in transportation. This scenario results in a maximum public intake over the course of 1 year of 10^{-6} A₂ from NCs.

The intake equation used for NCs is similar to that used for ACs, except that the total release Q is replaced by the product of the release rate Q' and exposure time t, as shown below. Also, all leakage during NCs is assumed to be completely respirable, because the diameter of the leak path is assumed to be small. In the event that the diameter of the leak path during NCs is not small, then some of the aerosol may be too large to be respirable, in which case the inhalation dose would be reduced. Therefore, this is a conservative assumption.

$$Q' = \frac{\text{Intake}}{t \times \text{BR} \times \chi/Q}$$

In the 300 Area, the nearest site boundary is the Columbia River. Because the Columbia River serves as a recreational area, a member of the public may be as close as the near bank. This location is as close as 300 m (984 ft) to the route of a possible shipment. In this scenario, the member of the public is assumed to spend 400 h/yr at the near bank of the Columbia River, or about 1 h/d. This exposure time is consistent with the time used in the IAEA derivation of the allowable release rate during NC from certified DOT Type B packages, and it is conservative considering the river's use as a recreational area. According to Table 7-4, the atmospheric dispersion coefficient at 300 m (984 ft) from a release in the 300 Area is 6.38×10^{-3} s/m³. The release rate during NC from a risk-based package in the 300 Area causing an equivalent public intake of 10^{-6} A₂/yr for this person is 1.18×10^{-3} A₂/h, as shown in Table 7-6.

Table 7-6. Example Calculations for Risk-Based Packages Based on Equivalent Public Exposure during NCs of Transport.

Area of Shipment	Maximum NC Release Rate for Negative USQ _T	Public A ₂ Intake	Airborne Respirable Fraction	Exposure Time, hr	Breathing Rate, m ³ /s	χ/Q ¹ s/m ³
300 Area ²	1.18×10^{-3} A ₂ /hr	10^{-6}	1	400	3.33×10^{-4}	6.38×10^{-3}
300 Area ³	3.51×10^{-4} A ₂ /hr	10^{-6}	1	8766	2.64×10^{-4}	1.23×10^{-3}
200 Areas only	2.46×10^{-2} A ₂ /hr	10^{-6}	1	8766	2.64×10^{-4}	1.76×10^{-3}
100 K Area	9.65×10^{-3} A ₂ /hr	10^{-6}	1	8766	2.64×10^{-4}	4.48×10^{-5}

¹ Note that the χ/Q 's reported are valid for short-term releases during accidents using 99.5% worst-case meteorology. For extended releases, these χ/Q 's are not appropriate and result in overly conservative results. It is more appropriate to use an average (24-hour or annual) χ/Q , although the SARs do not always report average χ/Q 's.

² This calculation is based on a member of the public at the near bank of the river, 300 m (984 ft) from the location of the release, for 400 h/yr.

³ This calculation is based on a member of the public at the far bank of the river, 800 m (2,625 ft) from the location of the release, for 8,766 h/yr.

Another location considered for a member of the public is a resident on the far bank of the Columbia River across from the 300 Area. The exposure time considered for this person is a full year, or 8,766 hours. This location is at least 800 m (2,625 ft) from the route of a possible shipment. According to Table 7-4, the atmospheric dispersion coefficient at this location is $1.23 \times 10^{-3} \text{ s/m}^3$. The 24-hour average (2.64×10^{-4}) breathing rate applies at this location to account for 8 hours of resting per day. The release rate from NCs from a risk-based package in the 300 Area causing an equivalent public intake of $10^{-6} \text{ A}_2/\text{yr}$ for this person is $3.51 \times 10^{-4} \text{ A}_2/\text{h}$, as shown in Table 7-6. Thus, for a release that is continuous and uniform over a 1-year duration, the release rate based on a full-time resident at the far bank of the river is more restrictive than that for the recreational user on the waters off the near bank of the Columbia River.

In the 200 Area, the location of the maximally exposed individual along the Hanford Site boundary is 11.7 km (7.3 mi) west of the 200 West Area (HNF-SD-WM-ISB-007), with an atmospheric dispersion coefficient of $1.76 \times 10^{-5} \text{ s/m}^3$. The exposure time considered for this person is a full year, or 8,766 hours, and the 24-hour average breathing rate applies. The release rate from NCs from a risk-based package in the 200 Area causing an equivalent public intake of $10^{-6} \text{ A}_2/\text{yr}$ for this person is $2.46 \times 10^{-2} \text{ A}_2/\text{h}$, as shown in Table 7-6.

The example calculations in Table 7-6 are derived only from equivalency to the NCs containment requirement for public exposure; criteria derived from equivalency to the ACs containment requirement for public exposure are given in the preceding section.

3.2.1.1.4 External Dose. The external dose due to photons, beta particles, and neutrons from a package damaged by an impact or fire is calculated for the immediate worker for a justifiable duration. The co-located worker and offsite receptors are located at a sufficient distance from the release such that the external dose to the offsite receptor is negligible. Information from the structural evaluation may be used to determine the configuration of the package after ACs, or it may be conservatively assumed that all of the shielding provided by the packaging is lost. The analysis may use a computer program, provided it satisfies the contractor's QA requirements.

All sources of photons should be considered, although the analysis may quantitatively analyze only one source if it is shown to dominate the dose rate. Sources of photons include gamma rays emitted by the nucleus and its daughter products during radioactive decay, bremsstrahlung radiation produced by the deceleration of beta particles, and photons produced by capture or inelastic scattering of neutrons.

Calculations of the dose to the skin from beta particles may either be performed by hand calculation following the guidance in IAEA (1990, Section AI.4.2), or through the use of a computer program. Note that the location of the receptor in the IAEA guidance is 1 m (3 ft), whereas 3 m (10 ft) is used in the consequence evaluation. To correct for the greater distance, a geometric attenuation factor based on either a point source or a line source may be included. Note also that beta particles have finite ranges, which depend on their energy. Beta particles lose energy as they are transported through air. Some beta particles will lose all their energy before reaching the onsite receptor. Thus, beta particles with a maximum range of less than 3 m (10 ft) can be neglected. Because the range is a function of the beta particle endpoint energy E_{max} , a threshold energy can be determined below which the range is less than 3 m (10 ft). From the

Radiological Health Handbook (DOH, pp. 122-123), a beta particle with energy of 1 MeV has a range of approximately 3 m (10 ft). Thus, only isotopes that emit beta particles with an endpoint energy of greater than 1 MeV need to be included in the dose from this pathway.

Sources of neutrons that should be considered are isotopes that undergo spontaneous fission, isotopes of low atomic number that emit neutrons after absorbing alpha particles, and the subcritical multiplication of these neutrons by fissile isotopes. However, in most cases, the external dose from neutrons is a small fraction of the inhalation dose. Hand calculations following the guidance in Nelson (1996), "Estimation of Neutron Dose Rates from Nuclear Waste Packages," may be used to quickly estimate the contribution of the neutron dose rate, and, if it is a small fraction of the inhalation dose, a more extensive analysis is not necessary. Otherwise, a qualified computer program may be used.

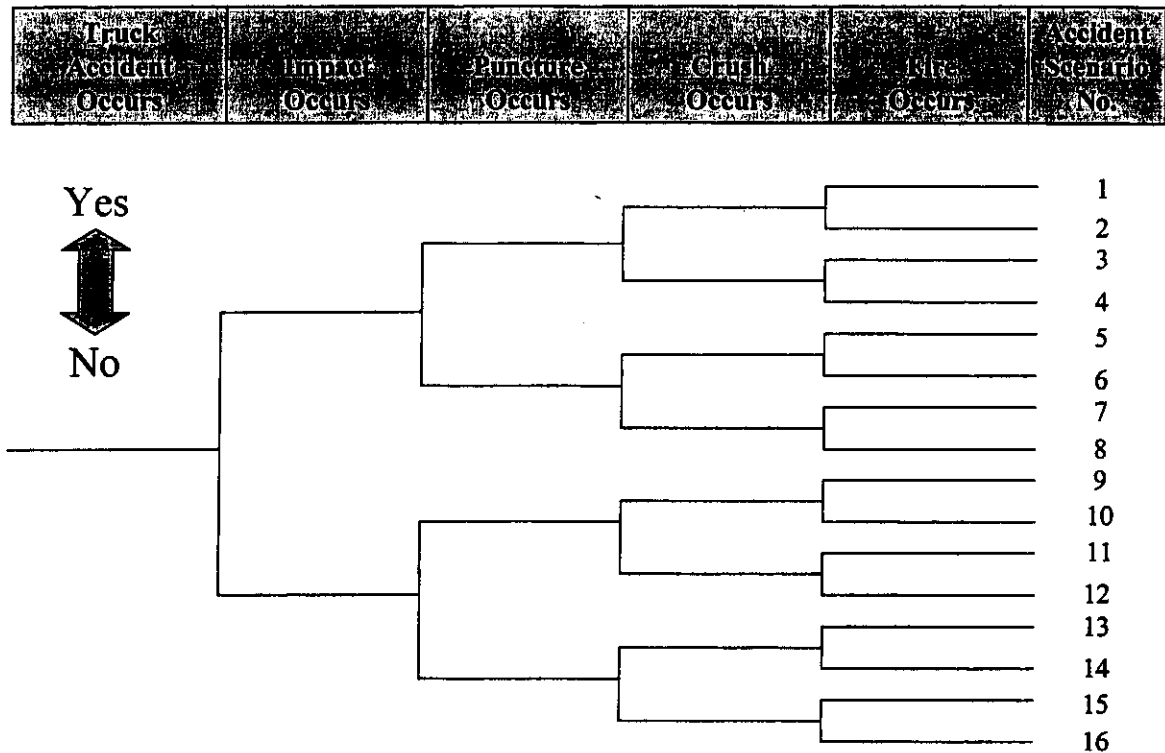
3.2.2 Frequency Analysis Methodology

If the release rates from the package exceed the package specific release limits derived using the methodology in Section 3.2.1, then an equivalent degree of safety based on limiting the public intake is not shown, and it is left to a case-by-case basis to determine if the dose consequences and frequency of a release are acceptable. Thus, a method is needed to calculate the frequencies of various potential accident scenarios to support the risk evaluation. The accident frequencies must be commensurate with the severities of the accident scenarios to provide a valid evaluation. This section describes the approach and provides data to be used to evaluate the frequencies of truck transportation accidents on the Hanford Site.

3.2.2.1 Accident Frequency Analysis Approach. The overall approach to determining Hanford Site accident frequencies is to start with the assumption that an accident has occurred based on the initiating event frequency as discussed in this section. This includes all accidents, ranging from minor "fender benders" to high-speed collisions and fires. The hazards analysis in Appendix G highlighted the accident scenarios that must be evaluated quantitatively and reduced the number of accident scenarios that need to be evaluated. However, it only eliminated those that were less than credible and those that are prevented through engineering and administrative controls (i.e., less than $1 \times 10^{-7}/\text{yr}$), the remaining accident scenarios represent the bulk of the risk of transporting radioactive materials on the Hanford Site.

An event tree approach is used to establish the overall framework for calculating the frequencies of truck accidents that could lead to a release of radioactive material on the Hanford Site (Figure 7-8). An event tree starts with an initiating event (in this case, the initiating event is a truck transportation accident) and follows with a series of ACs that describe the mechanical (impact, puncture, and crush) and thermal (fire) environments that could potentially be generated in an accident. For consistency with the 10 CFR 71, Type B ACs, four potential accident environments are defined in chronological order, i.e., the first event is impact, second is puncture, third is crush, and fourth is fire. Each of these subsequent events (or environments) is conditional upon the prior occurrence of the preceding event(s), as are the associated probabilities of occurrence of each environment. Note that immersion, the final HAC defined in 10 CFR 71, is not applicable to the Hanford Site (see Appendix G).

Figure 7-8. "Unpruned" Event Tree Model of Potential Transportation Accident Conditions.



The third (puncture) and fourth (crush) events are treated independently from each other because of the way in which the available probabilistic information is presented. It was not possible, given the data, to determine the conditional probability of a crush event given a collision or overturn that generates puncture conditions. In other words, the available data did not support separation of the two events so they could be independent of each other. Consequently, they are evaluated separately when determining accident frequencies. This is conservative because it tends to double-count the events that involve both puncture and crush. However, this conservatism is believed to be small because only a small fraction of actual accidents would involve both puncture and crush conditions, given the occurrence of a collision or overturn.

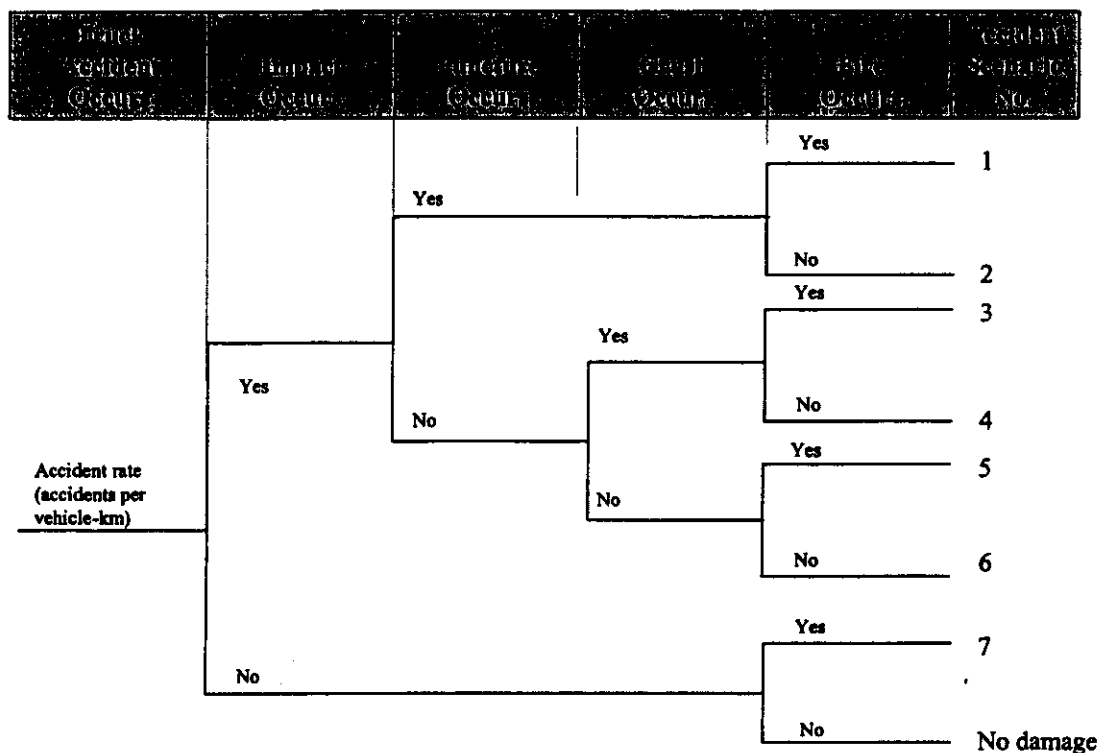
The fifth event represents fire conditions that may occur in accidents. All five events are aligned in chronological order at the top of the event tree shown in Figure 7-8. The event tree diagram was then constructed by first drawing a branch to represent the occurrence of any accident, then drawing a second set of branches to represent the impact event, and so on. The top branch of the impact event represents the occurrence of impact (i.e., collision or overturn) conditions and the bottom branch represents non-collision accidents. Similarly, for subsequent events, the top branches represent the occurrence of the specific condition and the bottom branch represents its absence from the particular scenario. Probabilistic information is then inserted on each branch to quantify the event tree. The probabilistic information may be in the form of a point value or a distribution. The probability of the top branch is 1 minus the probability of the bottom branch.

The next step in developing the event tree is to “prune” it or remove branches to simplify the tree. Note that each accident scenario in Figure 7-8 is numbered. The rationale for removing certain accident scenarios from further consideration are presented below:

- Scenarios 5 through 8 were deleted and replaced by two branches that represent the occurrence of impact conditions but no puncture or crush, or fire conditions. Because crush and puncture conditions are treated independently of each other, Scenarios 5 through 8 collapse into two branches that represents non-occurrence of puncture and crush conditions (i.e., impact-only) followed by fire or non-fire conditions.
- Scenarios 9 through 14 were completely removed from the event tree because they represent scenarios that are not physically possible. For example, puncture probes cannot be generated in accidents that do not also involve a collision or overturn; i.e., the occurrence of a puncture probe that strikes the package is dependent on the prior occurrence of an impact event. Similarly, for a crush environment to occur, prior occurrence of a collision or overturn is required. Therefore, the event tree branches that represent accident scenarios defined by non-impact events, followed by puncture and crush conditions were deleted.

This leaves a total of 8 potential accident scenarios in the event tree model (Figure 7-9). Note that the final scenario at the bottom of Figure 7-9 represents all accidents that do not result in package failure. Also note that the accident scenarios in Figure 7-9 have been renumbered, rather than retaining the scenario numbers shown in Figure 7-8, to avoid confusion in the following sections.

Figure 7-9. Simplified Event Tree Model of Potential Transportation Accidents.



The following general equation is used to calculate the frequencies of the various accident scenarios depicted in Figure 7-9:

$$F_j = RNDP_j$$

where:

- F_j = The frequency of accident scenario "j"
- R = Accident rate (accidents/vehicle-km)
- N = Number of shipments/yr
- D = One-way shipping distance (km)
- P_j = Conditional probability of Accident Scenario "j" given an accident has occurred.

The different scenarios in Figure 7-9 can be modeled by substituting different expressions for P_j , as follows:

- $P_j = P_i$ for impact-only accident scenarios
- $P_j = P_{p/i}$ for impact scenarios that also lead to puncture conditions
- $P_j = P_{c/i}$ for impact scenarios that also lead to crush conditions
- $P_j = P_{f/i}$ for impact scenarios that also involve fires
- $P_j = P_{f/p}$ for impact/puncture scenarios that also involve fires
- $P_j = P_{f/c}$ for impact/crush scenarios that also involve fires
- $P_j = P_f$ for accident scenarios that involve fire-only conditions (no collision).

In other words, the calculational approach involves estimating the frequency of all accidents (product of the accident rate, shipping distance, and annual number of shipments) and then multiplying this product by a series of fractions that represent the probabilities of encountering conditions that could compromise the integrity of each package's containment system. The rest of this chapter describes the derivation process for the accident rate (parameter "R") and conditional probabilities (values for P_j) used in the calculations. The remaining parameters, including N and D , are obtained by consulting with potential users of the package.

3.2.2.1.1 Truck Accident Rate (R). Several studies have been conducted that determined highway accident rates for transportation risk assessment. Table 7-7 presents a side-by-side display of the accident rate information provided in the reviewed documents. Although the bases for the accident rates differ from one study to the other and should not be compared directly, the more recent values are within the range from about 2×10^{-7} to 4×10^{-7} accidents per vehicle-km (i.e., from 2 to 4 accidents/10 million km traveled). This illustrates that all of the values in this range would be acceptable for developing accident frequency estimates and none would result in a serious under- or over-estimate of Hanford Site accident frequencies.

Table 7-7. Summary of Truck Accident Rates from Previous Studies.

Source	Time period covered	Rate (accidents per vehicle-km) ^(a)	Comment
SLA-74-0001	1966 - 70	2.5×10^{-6}	National average data for trucks and delivery vans
NUREG/CR-4829	1968 - 81	4.0×10^{-6}	National average data; American Petroleum Institute data for heavy trucks
ANL/ESD/TM-68	1986 - 88	2.6×10^{-7}	Washington State primary highways; heavy trucks
		3.94×10^{-7}	National average primary highways; heavy trucks
ANL/ESD/TM-150	1994 - 96	1.75×10^{-7}	Washington State primary highways; heavy trucks
		3.66×10^{-7}	National average primary highways; heavy trucks
WHC-SD-TP-RPT-021	1990 - 95	2.0×10^{-7}	Hanford Site rate for trucks, vans, and pickups combined
NUREG/CR-6672	1984, 1986 - 88	2.2×10^{-7}	Nation-wide data, rural areas; adjusted Bureau of Motor Carrier Safety data

^(a)Because of different bases that were used to derive the accident rates, such as the definition of an accident, the values are not directly comparable. See the source studies for additional information on the bases for the derived accident rates.

Sources:

ANL/ESD/TM-68, *Longitudinal Review of State-Level Accident Statistics for Carriers of Interstate Freight*, Argonne National Laboratory, Argonne, Illinois.

ANL/ESD/TM-150, *State-Level Accident Rates of Surface Freight Transportation: A Reexamination*, Argonne National Laboratory, Argonne, Illinois.

NUREG/CR-4829, *Shipping Container Response to Severe Highway and Railway Accident Conditions*, U.S. Nuclear Regulatory Commission, Washington, D.C.

NUREG/CR-6672, *Reexamination of Spent Fuel Shipment Risk Estimates*, Sandia National Laboratories, Albuquerque, New Mexico.

SLA-74-0001, *Severities of Transportation Accidents Volume I--Summary*, Change 1, Sandia National Laboratories, Albuquerque, New Mexico.

WHC-SD-TP-RPT-021, *Hanford Site Truck Accident Rate*, Westinghouse Hanford Company, Richland, Washington.

The most recent evaluation of truck accident rates is the basis for this document. Note that highway shipments on the Hanford Site may be conducted using a number of different truck types, including semi-tractor/trailer combination rigs, flatbed trucks, and vans. For this analysis, a single accident rate is used to represent the accident rates of all types of highway vehicles. The available data did not support separating the overall highway accident rate into separate rates for specific types of vehicles (e.g., heavy-combination trucks, delivery vans).

The overall truck accident rate (parameter "R" in equation 1) used in the risk evaluation is represented by the nation-wide average accident rates for heavy trucks developed by NUREG/CR-6672 (Figure 3.10a). The rate was developed using Bureau of Motor Carrier Safety data that has been adjusted to isolate accidents that occurred in cities and non-city regions. The database used to develop the highway accident rates was large, which increases the confidence in the estimates. The rural (non-city) accident rate is used for Hanford Site evaluations because of its generally rural nature. The accident rate given by NUREG/CR-6672 for rural highways is 2.2×10^{-7} accidents per vehicle-km (3.2×10^{-7} accidents/vehicle-mile), which is the approximate 50% confidence interval (i.e., there is a 50% likelihood that the actual accident rate is 2.2×10^{-7} /km or less). This represents the rate of all accidents, regardless of severity, ranging

from minor fender-benders to severe collisions and fires. The annual frequency of all accidents for a given packaging system is the product of the highway accident rate, number of shipments per year, and the shipping distance. Note that only the one-way shipping distance is used in the Hanford Site accident frequency calculations because the package is loaded only when traveling in one direction; i.e., the package is empty for the return trip.

3.2.2.1.2 Impact Probability (P_i). Impact conditions generated in an accident are a function of many variables, including impact velocity, orientation, and impact surface characteristics. NUREG/CR-6672 and NUREG/CR-4829 constructed impact velocity distributions that considered the impact orientation (i.e., corner, end, or side impact) and the characteristics of various real surfaces (e.g., hard rock, soil, concrete, roadbed). The distributions were constructed to develop factors to adjust the overall accident rate to reflect real ACs and to determine the fractions of accidents that exceed specified impact velocities. This type of data is needed to calculate Hanford Site accident frequencies.

According to NUREG/CR-6672, the probability of a collision or overturn, given an accident occurs, is 0.7412. The probability of a non-collision accident is $(1 - 0.7412)$ or 0.2588, given an accident occurs. The impact velocity distributions constructed by NUREG/CR-6672 and NUREG/CR-4829 were constructed assuming that a collision/overturn accident occurs. Thus, the following distribution is used to determine the fraction of collision and overturn accidents that result in impact velocities of a specified value. For this methodology, the specified impact velocity would be the impact failure threshold determined via structural analysis.

To construct the distributions, NUREG/CR-6672 assumed that the impacts at any angle are equally probable. They used data from Geographic Information System (GIS) interrogation of the digitized U.S. Department of Agriculture maps to characterize the wayside characteristics adjacent to three representative truck shipping routes. They then conducted Monte Carlo analyses to develop the cumulative distributions that are used here to assign the conditional probabilities of experiencing varying levels of impact severity. The distributions used for Hanford Site evaluations are shown in Table 7-8. The cumulative distribution is also illustrated in Figure 7-10 to assist in interpolation between the impact velocities listed in Table 7-8.

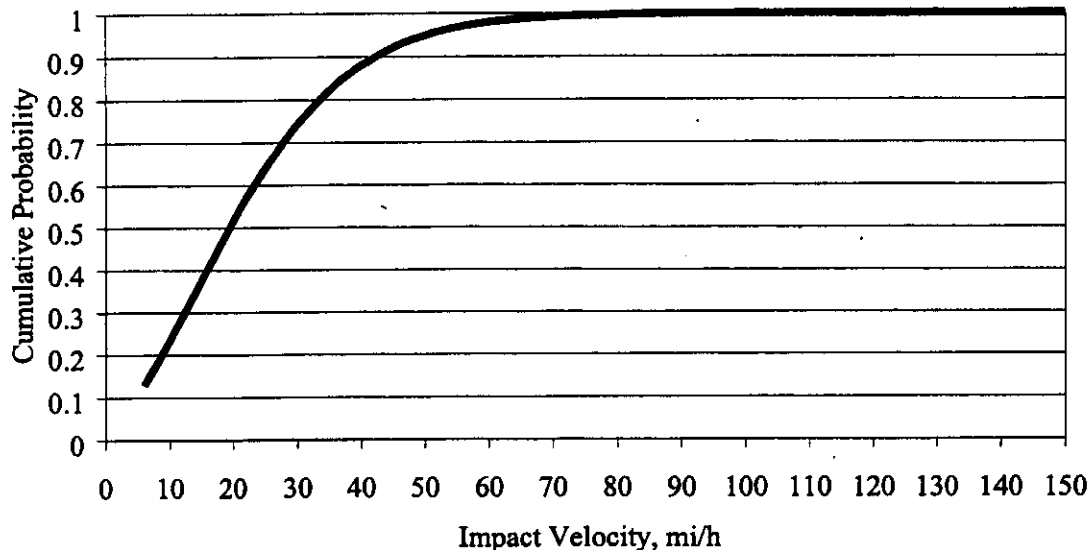
Table 7-8. Cumulative Truck Accident Velocity Distribution.

Initial Truck Velocity Adjusted for Braking							
Velocity (mi/h)	Cumulative Probability	Velocity (mi/h)	Cumulative Probability	Velocity (mi/h)	Cumulative Probability	Velocity (mi/h)	Cumulative Probability
0.0	0.0	30.0	0.74353	62.0	0.98383	94.0	0.99979
2.0	0.03834 ^(a)	34.0	0.80877	66.0	0.98908	98.0	0.99990
6.0	0.12916	38.0	0.86020	70.0	0.99261	102.0	0.99995
10.0	0.23508	42.0	0.89961	74.0	0.99503	106.0	0.99998
14.0	0.34886	46.0	0.92881	78.0	0.99670	110.0	0.99999
18.0	0.46237	50.0	0.95009	82.0	0.99825	150.0	1.0
22.0	0.56877	54.0	0.96547	86.0	0.99910		
26.0	0.66345	58.0	0.97634	90.0	0.99956		

Source: NUREG/CR-6672.

^(a) Probability that the accident or impact velocity is less than or equal to the listed velocity. For example, about 0.038 (3.8%) of all accidents have an impact velocity of 3 km/h (2 mi/h) or less.

Figure 7-10. Plot of Cumulative Distribution of Impact Velocities.



The velocity distribution shown in Table 7-8 is for accidents that occur on level ground. This is representative of the Hanford Site because there are no bridges of concern for a truck to fall from and strike the ground below the bridge. There are also no significant embankments for the truck and packaging system to fall from and strike an object or surface below. The distribution credits a reduction in velocity due to braking. The distribution also incorporates the probability of striking surfaces of various characteristics, such as hard rock, soft rock, soil, and the roadbed. The data used by NUREG/CR-6672 to characterize the impact surface was developed from U.S. Department of Agriculture digitized maps using GIS interrogation techniques. This corrected some potential non-conservatisms in the NUREG/CR-4829 study. The GIS techniques were applied to four illustrative routes, including one cross-country route between Florida and Washington State. These data were accounted for in the NUREG/CR-6672 analysis.

3.2.2.1.3 Puncture (P_{pn}). Puncture conditions can be generated in a collision accident that generates sharp objects. The probability that such a condition is generated is extremely speculative, as it depends on the probability that a probe is generated that is capable of causing puncture will be formed, that the probe strikes the surface of the packaging system in an orientation that could lead to failure, and finally that the probe does not break before it causes failure. NUREG/CR-6672 stated that the probability that a puncture probe capable of causing failure of a Type B spent fuel shipping cask was assumed to be 0.001, based on a review of puncture and shear failures of rail tank cars. This was calculated by multiplying the probability of failure of rail tank car wall (4 out of 100 accidents reviewed or 0.04 per accident) by itself to represent two 2.54 cm (1-in.) thick walls of a representative spent fuel shipping cask. This is not directly applicable to Hanford Site accidents because rail shipments are not conducted on the Hanford Site and few onsite packaging systems are as robust as a spent fuel shipping cask.

Previous work provided in SLA-74-0001, *Severities of Transportation Accidents Volume I--Summary*, was used to form the basis for determining the probability of encountering puncture conditions. SLA-74-0001 concluded that puncture environments were most likely to occur in collision and overturn accidents. Based on a review of accident reports, they also determined that 72% of the accident reviewed resulted in cargo damage and, of those, 31.1% were puncture accidents. These percentages were updated using the following information:

- Probability of collision, given an accident = 0.7412
- Probability of overturn, given an accident = 0.2588 (probability of non-collision accident) * 0.5336 (probability of impact with roadbed given non-collision accident) * 0.6046 (probability of overturn given impact with roadbed and non-collision accident) = 0.0835
- Probability of collision and overturn, given an accident = 0.7412 + 0.0835 = 0.8246

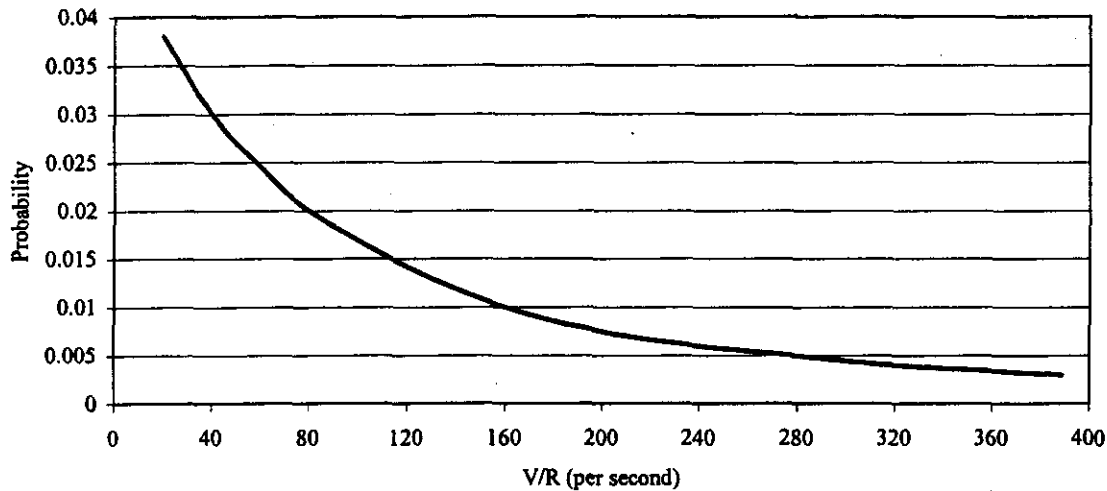
Then, combining the above probability of collision and overturn with the SLA-74-0001 data on puncture results in the following overall probability of puncture given and collision or overturn accident:

$$0.8246 \frac{\text{collision or overturn}}{\text{accident}} \times 0.72 \frac{\text{cargo damage}}{\text{collision or overturn}} \times 0.311 \frac{\text{puncture accident}}{\text{cargo damage accident}} \\ = 0.18 \frac{\text{puncture accidents}}{\text{accident}}$$

This indicates that about 18% of all accidents result in some puncture damage to the cargo. This is the probability that at least one package in a shipment of relatively soft containers (e.g., 210 L [55-gal] drums) is damaged by puncture conditions. SLA-74-0001 also determined from accident reports that about 6.8% of such containers are penetrated in a puncture accident that involves a shipment with multiple "soft" containers. Thus, 18% of all collision or overturn accidents result in a puncture-type failure of at least one soft packaging system and, of those, 6.8% of the containers in multiple-container shipments are failed, given that at least one package has failed in this manner. These fractions are applied to Type A Packages and assume that the package provides no resistance to puncture (i.e., assumes containment system failure if a puncture environment is applied to the package).

For hardened packaging systems (taken to be Type B packages), the relationship is much more complex. SLA-74-0001 chose to describe the puncture environment in terms of the ratio of the relative velocity between the probe and package to the radius of the probe (V/R). SLA-74-0001 provided Monte Carlo simulation modeling to determine the distribution of V/R for hardened, Type B packages. Monte Carlo simulations were conducted for various probe radius distributions and velocity distributions to determine the V/R distribution shown in Figure 7-11.

Figure 7-11. Probability Distribution of Puncture Accident Severity.



This probability distribution can be coupled with the impact velocity distribution given in Figure 7-10 to calculate the probability of creating a puncture probe that is sufficiently hardened and oriented properly to damage a Type B packaging system. The failure thresholds calculated for nonequivalent packaging systems determine the appropriate velocity to use in the calculations. Furthermore, it is concluded in SLA-74-0001 that the probability of causing failure of Type B packaging systems is essentially independent of the probe radius.

3.2.2.1.4 Crush. The hypothetical crush test for Type B packages is applied only to packages that weigh less than 500 kg (1,102 lb) (see 10 CFR 71). Thus, for packages greater than 500 kg (1,102 lb), this branch of the event tree shown in Figure 7-8 is not applicable.

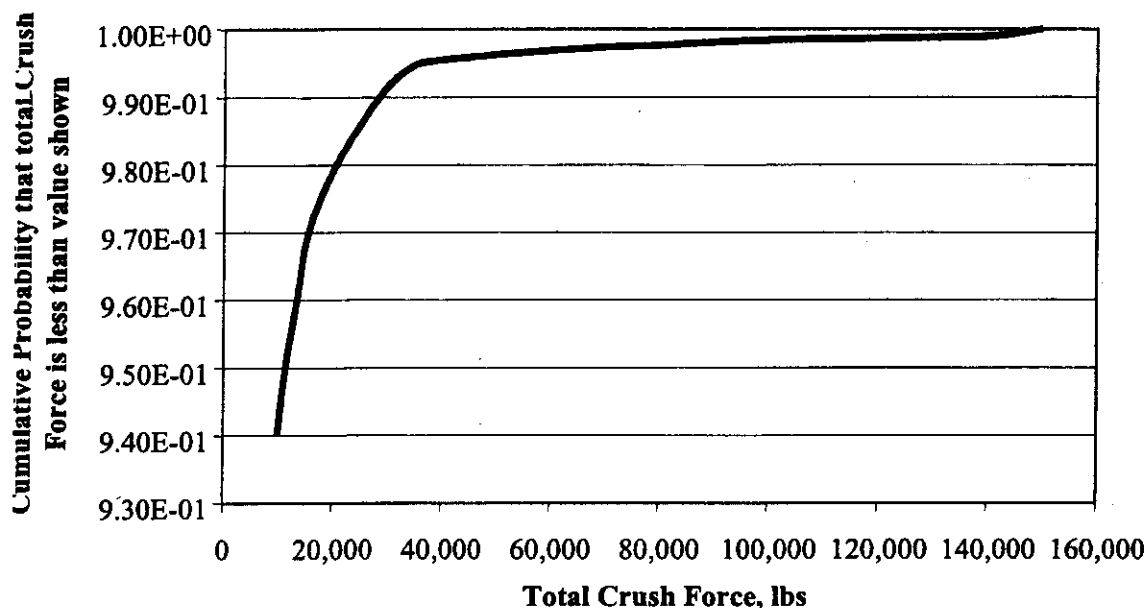
A crush environment can be generated in three ways. The first way results from an overturn accident in which the vehicle or trailer comes to rest on top of the packaging system while it is still attached to the vehicle (i.e., tiedown system remains in place). The second way results from a collision accident in which one or more packages are ejected from the vehicle or trailer and the vehicle, trailer, or other heavy object comes to rest on top of the package. Both of these scenarios result in a static crush environment with a heavy load applied to the top of the package (i.e., the package is underneath the vehicle); the load would be resisted by an equal force in the opposite direction exerted on the package by the ground. The crush load applied to the package is a function of the weight of the object resting on top of the package (including the weight of the trailer and cargo where applicable), and the amount of energy absorbed by the crushing object (e.g., trailer wall deformation will reduce the applied crush load).

The third way of generating a crush load on the package is referred to as "inertial" crush. This environment results from the transfer of momentum from one package to another within a cargo compartment. For example, the deceleration forces produced during a collision would cause a load of multiple 208 L (55-gal) drum packages to press together, producing a crush force. In addition to the packages pushing together, the row of packages adjacent to the trailer wall are

also crushed from one direction by the weight of the other packages and by the trailer wall itself. This is referred to as a dynamic crush environment. In the dynamic crush environment, the load applied to the package is a function of the weight of the other cargo, the relative acceleration between the trailer and the packages, and the loading configuration. The analyses prepared in SLA-74-0001 concluded that the probability of static crush is insignificant relative to the inertial crush probability.

The results are shown in Figure 7-12. The analyst determines the failure threshold of the package with respect to the crush environment and then selects the appropriate conditional probability of generating crush forces that are equal to or greater than threshold.

Figure 7-12. Cumulative Distribution of Total Crush Load, Given a Collision or Overturn Accident.

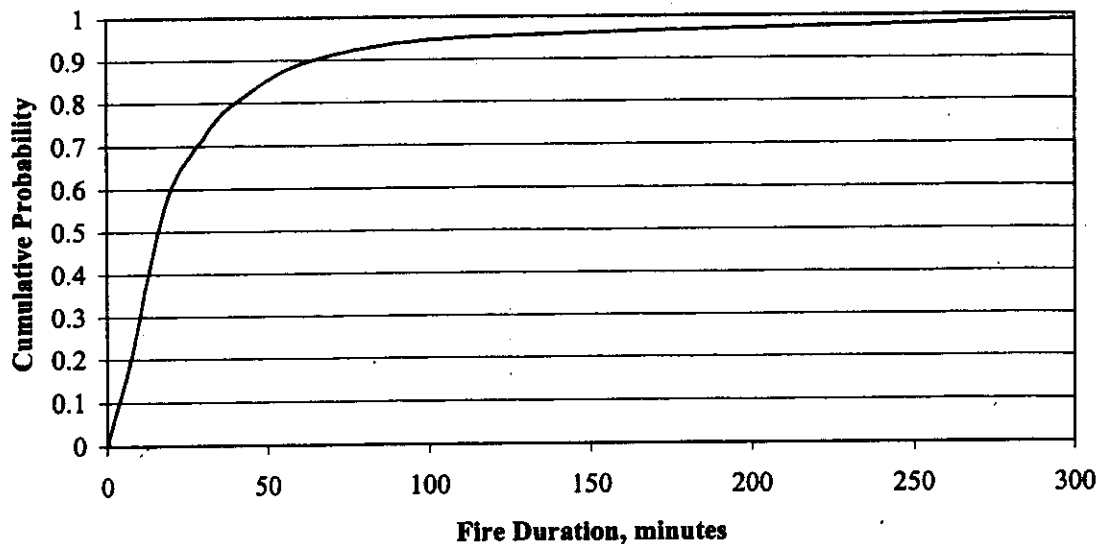


3.2.2.1.5 Fire. The effects of fires are a function of the temperature and duration of the fire, the fire orientation with respect to the package (i.e., fire and container are co-located versus offset from each other), and the thermal properties of the package (e.g., thermal resistance of the seals, emissivity of the container surfaces). NUREG/CR-6672, NUREG/CR-4829, and SAND 93-2580, *A Statistical Description of the Types and Severities of Accidents Involving Tractor Semi-Trailers*, have examined accident reports and modeled fire effects to characterize the probabilities of various fire environments. Typically, these studies generated a number of probabilistic distributions to generate the approximate probabilities of fires of varying severities. These distributions are based on nation-wide truck transport and are assumed to be applicable to the Hanford Site in general. However, certain distributions are not applicable, such as the probabilities of fires occurring after truck/train grade-crossing accidents (rail transport is no longer used on the Hanford Site).

This section describes the information needed to estimate the conditional probabilities of generating various fire scenarios. As with the previous distributions, the first distributions require the ACs to be isolated from the packaging integrity evaluations conducted in the studies listed above.

There are two types of fire scenarios modeled in Figure 7-13. One is a fire that occurs in conjunction with or following a mechanical impact event, such as a collision or overturn. The other is a fire that occurs in non-collision accidents. The former scenarios are intended to model the cumulative damage from mechanical and thermal ACs while the latter models thermal-only ACs (i.e., no mechanical damage sustained prior to fire initiation). The conditional probabilities of these scenarios are developed in the following subsections.

Figure 7-13. Cumulative Probability Distribution of Fire Duration for Accidents Involving Mechanical Damage (i.e., collisions, overturns).



3.2.2.1.6 Conditional Probability of Fire Given Mechanical Accident. In SAND 93-2580, cumulative distributions of fire durations were developed based on reviews of accident reports, curve-fitting techniques, and Monte Carlo analyses. A cumulative distribution of fire duration was developed that will be used to select the conditional probability of encountering such conditions on the Hanford Site. Note that the distributions are based on nation-wide accident data. Little can be done to adjust these distributions to more-closely reflect Hanford Site road conditions, as the fire severity probabilities are mostly dependent on the type of vehicle struck (e.g., truck tanker, passenger car, etc.) rather than road wayside conditions. The following assumptions are implied in the use of this distribution:

- The fire temperature reached is assumed to be the regulatory hypothetical fire test temperature (800 °C [1,472 °F]). No credit is taken for the probability that the fire temperature is lower than that value. For perspective, SAND 93-2580 concluded that essentially all fires have average fire temperatures greater than 650 °C (1,202 °F) and that only one fire in two reaches an average temperature of 1,000 °C (1,832 °F).
- It is assumed that the fire is engulfing. For perspective, SAND 93-2580 determined that no more than one fire in two is engulfing.
- The distribution includes the data on fires involving collisions with trains. For perspective, SAND 93-2580 determined that collisions with trains are the most likely scenarios to result in an engulfing fire condition (or fires with diameters greater than 7.6 m [25 ft]). Collisions with trucks, cars, and fixed or non-fixed objects were determined to have an 80% probability of being less than 7.6 m (25 ft) diameter. In other words, inclusion of train-accident initiated fires lead to a higher probability of being engulfing than would be calculated for the Hanford Site where rail shipping is no longer conducted.

These assumptions and the perspectives describe key conservatisms that demonstrate the bounding nature of this methodology and data.

3.2.2.1.7 Conditional Probability of Fire-only Accident. The cumulative distribution used to select the conditional probabilities of fire-only accident scenarios is shown in Figure 7-14. This figure was developed using NUREG/CR-6672 data that represents the cumulative probability of fires lasting less than or equal to some duration for non-collision accidents. The numerical values used to construct the distribution are shown in Table 7-9.

Figure 7-14. Cumulative Probability Distribution of Fire Duration for Non-collision Accidents.

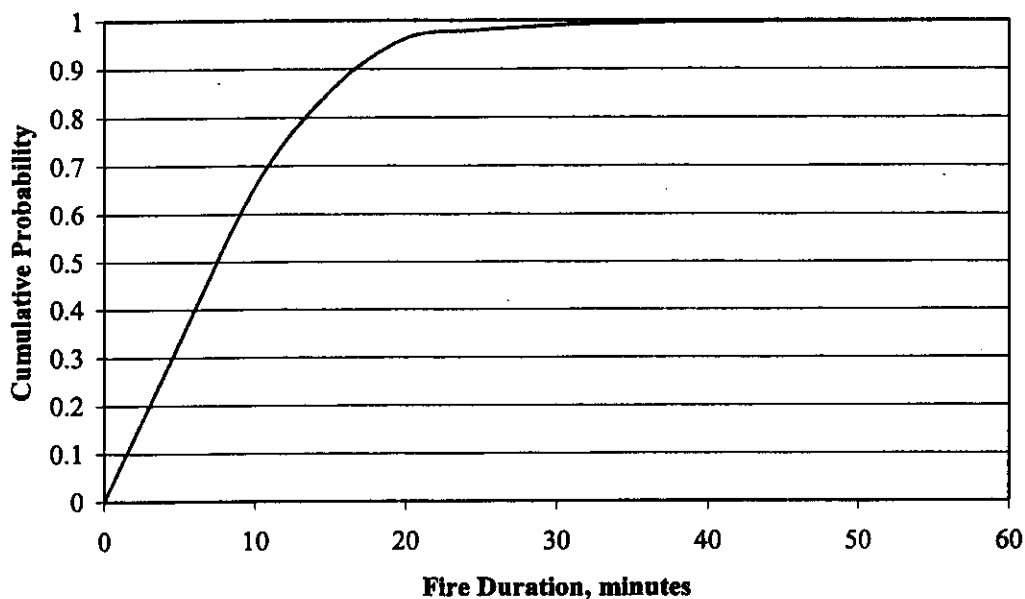
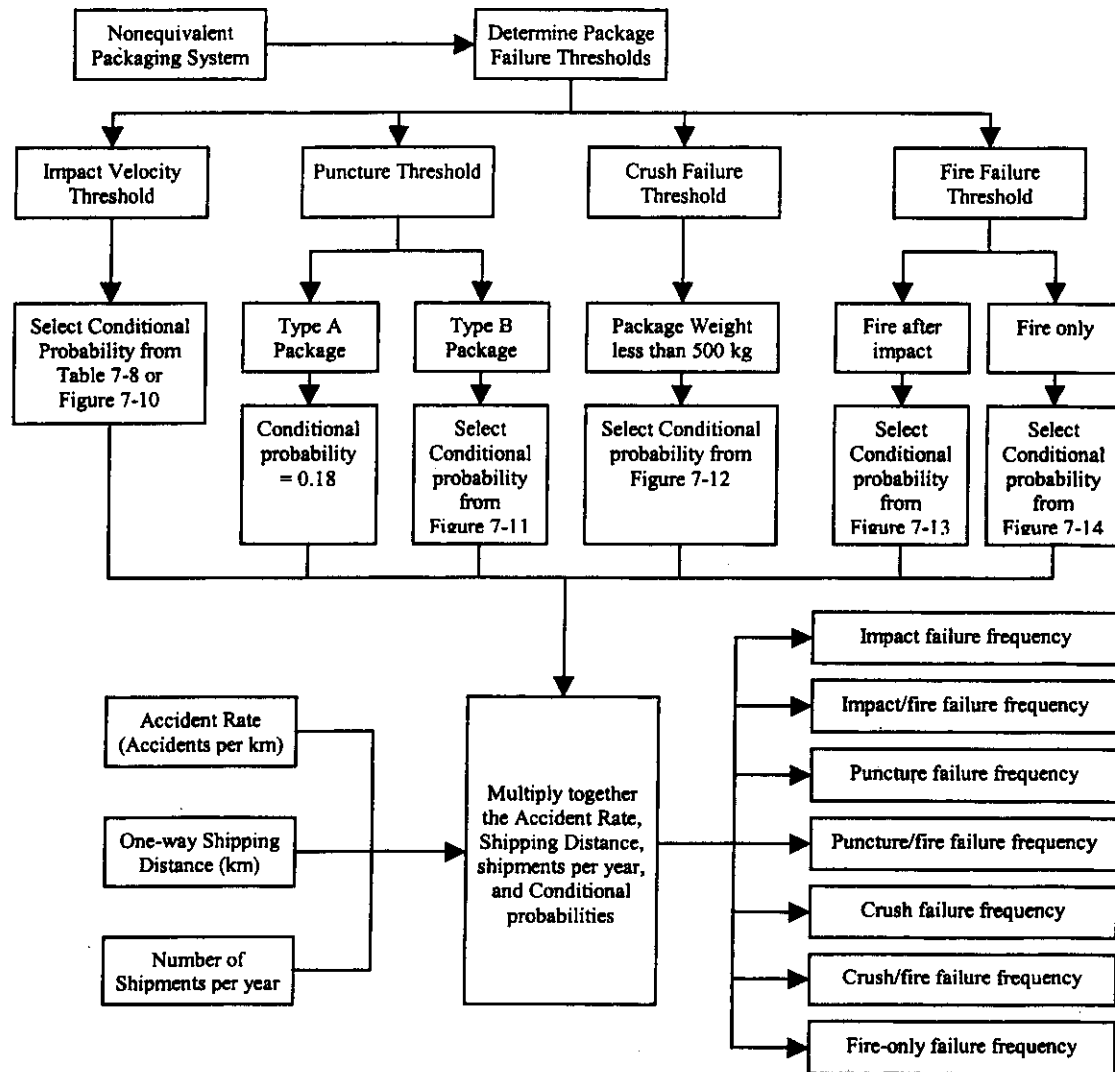


Table 7-9. Cumulative Fire Duration Probability Given Collision or Overturn.

Duration, minutes	Cumulative Probability	Duration, minutes	Cumulative Probability	Duration, minutes	Cumulative Probability	Duration, minutes	Cumulative Probability
0	0	20	0.9625	40	0.9970	60	0.9998
5	0.3311	25	0.9801	45	0.9985	65	0.99991
10	0.6596	30	0.9897	50	0.9992	70	0.99996
15	0.8551	35	0.9944	55	0.9996	75	~1

3.2.2.2 Hanford Site Accident Frequencies. Figure 7-15 shows the overall calculation process for determining the frequencies of severe accidents that could result in package failures. Up to five accident scenario frequencies are calculated for Type A packages and five for Type B packages (note that a different set of data is used to calculate the probabilities of puncture for Type A packages than for Type B). The frequencies of severe accidents on the Hanford Site are calculated by multiplying together the accident rate, number of shipments per year of the particular cargo to be transported, one-way shipping distance, and the conditional probabilities of the various ACs described above. An example evaluation is provided in Appendix G.

Figure 7-15. Accident Frequency Calculation Process.



4.0 REFERENCES

- 10 CFR 71, "Packaging and Transportation of Radioactive Material," *Code of Federal Regulations*, as amended.
- 10 CFR 830, "Nuclear Safety Requirements," *Code of Federal Regulations*, as amended.
- 49 CFR 173, "Shippers--General Requirements for Shipments and Packagings," *Code of Federal Regulations*, as amended.
- 49 CFR 393, "Parts and Accessories Necessary for Safe Operation," *Code of Federal Regulations*, as amended.
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HNF-SD-SPJ-SAR-002, *327 Building Safety Analysis Report*, Fluor Hanford, Richland, Washington.

HNF-SD-WM-ISB-007, *Central Waste Complex Interim Safety Basis*, Fluor Hanford, Richland, Washington.

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CHAPTER VIII. ROUTINE TRANSFERS

This chapter describes the processes and procedures used to make routine onsite shipments. All packagings prepared and shipped as routine must be authorized and fully comply with the provisions of the TSD. Basic procedures and processes for handling onsite shipments of hazardous materials and wastes are included in this chapter. Contractors provide more detailed facility/project operation procedures at the facilities where T&P activities occur.

1.0 COMMON HAZARDOUS MATERIALS AND WASTES SHIPPED ONSITE

All government-owned hazardous materials and wastes regulated by the DOT when shipped in commerce must be packaged and transported onsite in accordance with the provisions of this TSD regardless of the material or waste type or waste category. Generally, all onsite shipments made in accordance with this TSD are considered routine. Approximately 500 to 700 routine onsite shipments are made each month across the Hanford Site.

Nonroutine shipments are those that fall outside the provisions of the TSD or are shipments that meet the criteria of an "emergency move." Procedures for handling nonroutine shipments and emergency movements are further addressed in Chapter IX, Nonroutine Transfers.

The Hanford Site is a large industrial site with many chemical and nuclear processing facilities, laboratories, and waste management treatment, storage, and disposal sites. As a result, hundreds of different chemicals and chemical products are brought to the Hanford Site each year to support various projects and their missions. Materials commonly shipped onsite between facilities include the following:

- Flammable, nonflammable, and toxic compressed gases
- Corrosive liquids and solids
- Flammable liquids and solids
- Reactive metals, oxidizers, and explosives.

These materials and their resultant wastes represent all the DOT hazard classes (Class 1 through Class 9), and federal and state waste categories (EPA hazardous wastes, Washington Department of Ecology dangerous wastes, and extremely hazardous wastes.)

Additionally, radioactive materials and wastes are generated, used, and transported across the Hanford Site. Radioactive materials commonly shipped onsite include the following:

- Uranium and plutonium metals and oxides
- Sealed radiation, irradiation, and industrial test sources
- Process and environmental samples
- All categories of radioactive wastes
- Mixed waste.

Over 50 years of defense programs, nuclear energy production, and energy research have generated and continue to generate radioactive waste as Hanford Site facilities are decommissioned and cleanup projects move forward. These wastes are generated, packaged, and shipped onsite to waste management treatment, storage, and disposal facilities. These radioactive wastes consist of the following categories:

Spent Nuclear Fuel - The majority of this material was generated at the nuclear production reactors along the Columbia River and are now stored in the 100 K East and 100 K West fuel storage basins. This fuel is now being extracted from the fuel storage basins, placed into specially designed canisters and casks, and transported to the Canister Storage Building (CSB) located in 200 East Area. The onsite cask used for this transfer is called the multi-canister overpack cask. The fuel-filled canisters will remain at the CSB until they can be transferred to a licensed spent nuclear fuel repository.

High-Level Waste - The majority of the Hanford Site high-level waste (HLW) is stored in underground storage tanks located in the 200 East and 200 West areas. It is expected this waste will be extracted from the tanks and processed into vitrified glass, which, in turn will be placed into specially designed canisters for long-term storage and disposal. The low-activity waste generated from this process will be placed in a burial trench located in the 200 East Area. The HLW will be placed into special canisters and casks and transferred to the CSB in the 200 East Area and to similar buildings for long-term storage. The HLW will remain in storage until it can be transferred to a licensed HLW repository. Casks for the movement of the low-activity waste and HLW are currently being designed.

Transuranic Waste - This waste can be either remote-handled or contact-handled waste. It is generally confined to a small number of facilities that processed or handled transuranic (TRU) radionuclides, such as plutonium and americium, in support of defense program activities conducted at the Hanford Site. Much of this waste is located at the Hanford Site Plutonium Finishing Plant, CWC, and at the Waste Receiving and Processing facility. Generally, this waste is placed into authorized metal drums and metal box-based packaging systems and shipped onsite. Eventually, much of this waste will be prepared and shipped to the Waste Isolation Pilot Plant located near Carlsbad, New Mexico.

Low-Level Waste - This waste is generated at Hanford Site facilities where radioactive materials are processed or handled. This waste is usually a secondary waste stream generated because of other work or waste generated from a specific cleanup project. At the Hanford Site, this waste is usually made up of trash, building rubble, soil, laboratory supplies, and personal protective equipment potentially contaminated with mixed fission products or activation products. This waste is generally packaged in metal drums and metal boxes for contact-handled waste or specially designed casks or other container systems for remote-handled wastes. The wastes are transported onsite from the point of origin to the appropriate disposal facility or burial trench.

Mixed Waste or Radioactive Mixed Waste - This waste is generally radioactive low-level waste that is also cross contaminated with other hazardous constituents regulated under EPA or state rules. These wastes are generated throughout the Hanford Site and must be treated, stored, and disposed under special waste management rules. Generally, these wastes are packaged in metal

drums or metal box containers or specially designed high-integrity containers. They are transported onsite to designated mixed waste facilities located in the 200 West Area.

2.0 COMMON PACKAGINGS USED

At the Hanford Site, the policy and expectation is that all hazardous materials and hazardous wastes (except onsite radioactive materials and wastes, or otherwise as stated in Sections 3.1.2 and 3.1.4) must be packaged and transported onsite and offsite in full compliance with DOT regulations. Onsite radioactive material and waste shipments must either meet DOT regulations or provide an equivalent degree of safety as defined and demonstrated in this TSD.

Most hazardous materials and wastes are packaged in nonbulk packagings authorized in accordance with the DOT regulations under 49 CFR 173. Sections 173.201, 173.202, 173.203 for liquid hazardous materials meeting packing groups I, II, and II. Sections 173.211, 173.212, and 173.213 for solid hazardous materials meeting packing groups I, II, and III. Commonly used containers or packagings are identified below:

- *Class 1 Explosives* – 49 CFR 173 Subpart C for details
- *Class 2 Division 2.1, 2.2, and 2.3* – Cylinders meeting DOT Specification 3A, 3AA, 3AX, 4B and 39 as common examples
- *Class 3 Flammable Liquids* – DOT Specification UN1A1 or UN1A2 single packagings, or authorized combination packagings
- *Class 4 Flammable Solids* – DOT Specification UN1A1 or UN1A2 single packagings, or authorized combination packagings
- *Class 5 Oxidizers* – DOT Specification UN1A1 or UN1A2 single packagings, or authorized combination packagings
- *Class 6 Poisonous Materials* – DOT Specification UN1A1 or UN1A2 single packaging, or authorized combination packagings
- *Class 7 Radioactive* – Excepted packagings, industrial packagings IP-1, IP-2, IP-3, DOT Specification 7A Type A packagings, and NRC or DOT Certified Type B packages, or in onsite packaging systems authorized under the provision of this TSD. See Appendix A for a list of currently authorized onsite packaging systems
- *Class 8 Corrosives* – DOT Specification UN1A1 or UN1A2 single packagings, or authorized combination packagings
- *Class 9 Misc. Hazardous Materials* – Excepted packagings, and DOT Specification UN1A1 or UN1A2, or authorized combination packagings.

Some hazardous materials and wastes, including those that are radioactive, are transported onsite in bulk packagings. These include large wooden and metal boxes meeting appropriate specifications, and DOT Specification MC 312 and MC 412 cargo tanks. Table 8-1 provides a list of currently used cargo tank equipment.

Table 8-1. Current Cargo Tank-Trailer Fleet.

Tanker Type	Identification Number	Material of Construction	Capacity L (gal)
Polar ^(a) Tank Trailer, DOT MC 312	64-03858	316 Stainless Steel	19,000 (5,000)
Polar Tank Trailer, DOT MC 312	64-03859	316 Stainless Steel	19,000 (5,000)
Superior ^(b) Tank Trailer, DOT MC 412	64-04268	304 Stainless Steel	3,800 (1,000)
Polar Tank Trailer, DOT MC 312	64-04275	316 Stainless Steel	19,000 (5,000)
Walker ^(c) Tank Trailer, DOT MC 312	64-04294	316 Stainless Steel	19,000 (5,000)
Beall ^(d) Tank Trailer, DOT MC 412	64-05385	316 Stainless Steel	30,400 (8,000)
Beall Tank Trailer, DOT MC 412	64-05386	316 Stainless Steel	30,400 (8,000)

^(a)Polar is a trademark of Polar Tank Trailer, Inc., Holdingford, Minnesota.

^(b)Superior is a trademark of Superior Steel Products, Inc., Caldwell, Idaho.

^(c)Walker is a trademark of Walker Stainless Steel Equipment Company, Inc., New Lisbon, Wisconsin.

^(d)Beall is a trademark of Beall Corp., Portland, Oregon.

The MC 312 cargo tanks were fabricated to the MC 312 DOT Specification per 49 CFR 178.340 and 178.343 (1989 standards). The MC 412 units were fabricated to the MC 412 DOT Specifications per 49 CFR 178.345 and 178.348 (current standards). Each cargo tank is ASME certified, and has a CoC that states the cargo tank was designed, constructed, and tested in accordance with DOT Motor Vehicle Cargo Tank Specifications for cargo tanks used for the transportation of liquids classified as hazardous material.

3.0 ONSITE HAZARDOUS AND RADIOACTIVE MATERIAL SHIPMENTS - PROCESS AND PROCEDURES

The previous two sections identified the commonly shipped hazardous materials and wastes, and the common packagings used to ship them. The following sections identify the procedures that are to be followed when conducting onsite shipments of hazardous materials.

3.1 REQUIREMENTS

Onsite shipments of hazardous materials and wastes including radioactive materials and wastes must be executed in accordance with this TSD and in accordance with contractor implementing procedures.

RL, ORP, and site contractors that conduct onsite hazardous material T&P activities must prepare and implement procedures that incorporate the following general and specific activities when applicable.

3.1.1 General Requirements for All Shipments

3.1.1.1 Contractor Program, Project, and Facility Procedures. Hazardous material shipments must be performed in accordance with applicable operating procedures that provide for nuclear safety, environmental protection, QA, security, radiological exposure and contamination control, and T&P requirements, as well as applicable federal, state, and contractor requirements. Examples include the following:

- DOE orders
- 49 CFR, "Transportation"
- This TSD
- PSSD (e.g., SARP, SEP, test and evaluation reports)
- CoC
- *Washington Administrative Code* (WAC) 173-303, "Dangerous Waste Regulations"
- 10 CFR 835, "Occupational Radiation Protection."

All PSSD requirements from applicable onsite package documents, CoCs, and/or applicable federal regulations governing the use of each package must be verified and incorporated into applicable program, project, or facility operating procedures. Compliance with these procedural requirements is mandatory.

The safety evaluations conducted for T&P activities are conducted for completed packages. The point where the safety responsibility transfers between a facility and the T&P activity is identified in Table 8-2.

Table 8-2. Transportation to Facility Safety Responsibility Interfaces.

Shipping Facility	Receiving Facility	TSD Interface Boundaries
Facility has safety or authorization basis document that defines facility boundary within which all activities including T&P activities are addressed.	Facility has safety or authorization basis document that defines facility boundary within which all activities including T&P activities are addressed.	Transfer of responsibility occurs at shipping facility safety or authorization basis boundary and at the receiving facility safety or authorization basis boundary.
Facility has not defined a boundary in a safety or authorization basis document within which T&P activities are addressed.	Facility has safety or authorization basis document that defines facility boundary within which all activities including T&P activities are addressed.	Transfer of responsibility occurs at shipping facility loading dock/area and at the receiving facility safety or authorization basis boundary.
Facility has not defined a boundary in a safety or authorization basis document within which T&P activities are addressed.	Facility has not defined a boundary in a safety or authorization basis document within which T&P activities are addressed.	Transfer of responsibility occurs at shipping facility loading dock/area and at the receiving facility loading/area dock.

All interface requirements resulting from transport of hazardous materials must be incorporated into operating procedures. Packing and unpacking of a package is conducted under the facility safety or authorization basis documentation. Procedures for those tasks, however, must incorporate steps that result in the completed package meeting all applicable T&P requirements. Packages prepared in full compliance with DOT requirements may be stored in compliance with DOT storage requirements if permitted by the facility safety or authorization basis documentation. All onsite packages must be stored in compliance with the facility safety or authorization basis documentation. All onsite packages must be loaded on the transport vehicle in compliance with both the facility safety or authorization basis documentation and the PSSD. Most PSSDs do not evaluate the hazards associated with packing or loading activities.

New or revised procedures for activities conducted at an interface must be subjected to the USQ_T.

3.1.1.2 Packaging Authorization. All hazardous material and waste (except radioactive material and waste, or otherwise as allowed in Sections 3.1.2 and 3.1.4) shipped onsite shall be packaged and shipped in accordance with DOT hazardous material regulations as set forth in 49 CFR 171 through 185, and conform to the requirements of this TSD.

Radioactive materials and wastes shipped onsite shall be packaged and shipped in accordance with DOT Hazardous Material Regulations as set forth in 49 CFR 171 through 185 or may be shipped under a DOE-approved alternative providing an equivalent degree of safety as specified in this TSD.

3.1.1.3 Personnel Delegation and Training.

- All personnel involved with preparing hazardous material for transport including operators, drivers, waste handlers, and managers are considered hazmat employees and must be trained in accordance with 49 CFR 172.704, "Training requirements." See Chapter X, Personnel Training and Qualification, for details.
- Contractor operating organizations making shipments must establish a process for authorizing and delegating shipper responsibility.
- Delegated (authorized) shippers must be appropriately trained and delegated in accordance with the contractor's delegation process; 49 CFR 172, "Hazardous Materials Table; Special Provisions;" "Hazardous Materials Communications;" "Emergency Response Information;" "Training Requirements;" and Chapter X, Personnel Training and Qualification.
- The authorizing signature on a shipping document shall be that of an individual who is on the contractors' delegated (authorized) shippers list.
- Drivers/waste handlers must be appropriately trained in accordance with 49 CFR 172.704; 177.816; and 397.101(e); "Requirements for motor carriers and drivers;" and Chapter X, Personnel Training and Qualification.

- Drivers of vehicles hauling hazardous material onsite must have a current and appropriate driver's license (e.g., commercial driver's license [CDL] when required), medical card, and if transporting greater than highway route controlled quantity, proof of training in their possession while on duty.
- USQ_T evaluators must be appropriately trained in accordance with 49 CFR 172.704 and the requirements of the contractor's USQ_T.

3.1.1.4 Operational Controls.

- Authorized shippers should use a shipment checklist when preparing hazardous material and waste shipments, radioactive material and waste shipments, and mixed waste shipments.
- Transport of government-owned hazardous material by Hanford Site contractors shall be performed using a commercial carrier or approved government vehicle properly equipped to transport hazardous material with a trained and qualified driver. Transport of nongovernment-owned hazardous materials must meet the same requirements as when transported offsite.
- Only properly maintained and regulatory-compliant motor vehicles may be used for the transportation of hazardous material.
- The contractor shall define acceptable conditions for weather, road conditions, and other natural phenomena.
- A copy of the current Hazardous Material - Certificates of Registration must be provided in the cab of each vehicle that may require placards for the transport of hazardous material (49 CFR 107, Subpart G).

3.1.1.5 Documentation and Records. Source documentation used for physical, chemical, and radiological characterization of material and waste must be maintained as described in the contractor's records management system. Source documentation may include laboratory analysis, Material Safety Data Sheets, contents inventory sheets, waste designations, radiological characterization data, radiological calculation data, survey reports, radioactive material area certifications, Solid Waste Information Tracking System (SWITS) 310 reports, and/or Land Disposal Restriction Certifications/Notifications.

An auditable record of the inspection, maintenance, and certification for a packaging that documents the package is ready for use must be maintained at the packager's facility. This includes a copy of the current PSSD (e.g., SARP, SEP, test and evaluation report, CoC, or O&M instructions), and should include applicable package testing documentation and package closure instructions.

The shipper must be able to document that each package content meets the authorized payload limits as applicable in 49 CFR, the CoC issued by the NRC or DOE, or the onsite package documentation.

Records generated because of transporting hazardous material must be maintained. This includes a file of supporting documentation used as a basis for making the shipment (e.g., current CoC, PSSD). All appropriate internal procedures must reflect the requirements of these documents prior to making the initial shipment. Records should be retained for the same period as defined for the equivalent offsite shipment.

3.1.1.6 Safeguards and Security. Shipments of accountable nuclear materials must have contractor and RL Safeguard's approval prior to shipment.

Shipments of Category I and II SNM must be scheduled in advance to ensure availability of equipment, security, and properly cleared drivers.

When road closures are required by the PSSD or by safeguards and security requirements, these closures shall be coordinated and scheduled through the transportation service provider lead or designee.

3.1.1.7 Nuclear and Radiological Safety. All commercial vehicles operated by non-Hanford Site contractors will be free released back into commerce using DOT limits. Contractors using their own vehicles will also be released using DOT limits. All other onsite shipments using government vehicles will be free released per limits specified in 10 CFR 835.

Dose rates and contamination limits for onsite shipments must comply with, as applicable, 49 CFR 173; 10 CFR 835, Appendix D; or under limits authorized by this TSD within the specific PSSD (e.g., onsite SARP or SEP).

3.1.1.8 Planning and Communications. Each shipper of fissile radioactive material, Type B quantity of radioactive material, or gas poisonous by inhalation must provide the receiving organization with an estimated time of arrival and notification of shipment release. If a shipment of fissile radioactive material, Type B quantity radioactive material, or gas poisonous by inhalation is not received at its intended destination near the expected estimated time of arrival, the receiving organization must contact the originator to determine reason for delay and to determine if corrective actions are necessary.

A 1-day advance notice of shipment to involved organizations is recommended. The notice is to allow time for adequate regulatory review and coordination for onsite shipments. Special shipments (e.g., waste shipments, road closure, or escorted shipments) may require longer lead times.

Emergency response information must be provided in the cab of the vehicle. This includes the appropriate emergency response guide page from the *North American Emergency Response Guidebook*, the shipping papers, or a current copy of the guidebook. Shipping papers must provide an emergency contact telephone number.

Contractor management must be notified of all transportation-related incidents as soon as practical. Formal reporting of all incidents must be completed in accordance with the Hanford Site incident and occurrence reporting system requirements.

3.1.1.9 Checklists. The contractor should develop a shipment checklist. The checklists should include applicable requirements from DOE orders, DOT regulations, and onsite shipment requirements adequate to ensure that material is characterized, described, packaged, marked, labeled, and can be transported in accordance with applicable requirements. A checklist should be completed prior to each shipment not covered by an *Onsite Routine Radioactive Shipment Record* (ORRSR). An example of the checklist can be found in Appendix H, Example Checklists.

3.1.2 Radioactive Material Shipments – General Provisions

3.1.2.1 Fissile Material and SNM. The shipment of fissile material requires certain controls in addition to those needed for nonfissile radioactive material. Fissile material requires controls for criticality safety and accountability and because DOT-designated fissile material is also SNM of national strategic importance, additional controls for security and safeguards are mandatory. Fissile material as defined per 49 CFR 173.403 identifies the following radionuclides and isotopes: ^{238}Pu , ^{239}Pu , ^{241}Pu , and ^{233}U , ^{235}U , or combinations of these radionuclides.

Fissile-Excepted Material - Fissile material is classified as either fissile or fissile-excepted material. Fissile-excepted material is exempted from criticality safety packaging requirements. These shipments must be in accordance with both shipping and receiving facility fissionable material limits and restrictions. The most common exception is no more than .015 kg (15 g) of fissile material in a package. If material is transported in bulk, the .015 kg (15 g) limitation applies to the conveyance. Additional fissile material exceptions given in 49 CFR 173.453, “Fissile materials-exceptions” may be used for onsite shipments.

Shippers must evaluate fissile material packages and shipments per limitations specified in the following:

- 10 CFR 71.18, “General License: Fissile material limited quantity per package,” through 10 CFR 71.24, “General License: Fissile material, limited moderator, controlled shipment”
- 10 CFR 71.35, “Package evaluation”
- 49 CFR 173.457, “Transportation of fissile material, controlled shipments—specific requirements”
- 49 CFR 173.459, “Mixing of fissile material packages.”

This requires the shipper to evaluate activity and packaging per these section limitations.

Fissile Material - All onsite shipments (e.g., >.015 kg [15 g]) of fissile material not packaged in offsite packages shall be packaged in packages authorized by the TSD and reviewed and approved by the contractor approval authority. Appropriate ES&H and QA organizations shall review onsite packages for criticality safety concerns. Onsite package design when coupled with other considerations shall provide criticality safety protection equivalent to protection provided by approved offsite packages.

Onsite shipments shall be made in accordance with both shipping and receiving facility criticality safety specifications or facility fissile material status and the package fissile contents limitations based on the PSSD. The PSSD shall be available for use within the loading and unloading areas or any place where packages containing fissionable material are stored.

Accountable Material - Accountable material is nuclear material so designated by the Secretary of Energy and currently includes depleted uranium, enriched uranium, ^{241}Am , ^{243}Am , curium, berkelium, californium, $^{238-242}\text{Pu}$, ^6Li , ^{233}U , normal uranium, ^{237}Np , deuterium, tritium, and thorium. The requirements of the contractor's nuclear material accountability plan describe accountable material as any material that contains 0.5 times or more of the reporting unit of one or more of the above listed materials, elements, or isotopes. In more general terms, some of this material is also referred to as the following:

- Nuclear material
- SNM
- Source material.

Accountable nuclear material must be controlled and accounted for as directed by DOE Order 474.1A, *Control and Accountability of Nuclear Materials*. Material shipments and transfers must be documented and controlled in conformity with this order.

Approvals for Accountable Radioactive Material - Certain onsite movements of accountable radioactive material require Safeguard's concurrence and RL approval. Nonreportable radioactive material shipments and some small amounts of accountable radioactive material may be covered by a blanket approval obtained at the beginning of each fiscal year; approvals for larger amounts of accountable material are generated on a case-by-case basis. (RLID 473.1)

The shipment originator shall initiate application for Safeguard's concurrence and RL approval of the shipment. (RLID 473.1)

Categorization and Physical Protection for SNM Shipments - The degree of security imposed on SNM shipments is determined by the category amounts of SNM. DOE Order 474.1A defines the minimum requirements for safeguarding SNM by categories while in transit.

The contractor must manage all Category I shipments and their documentation as classified. Some Category II shipments may also be classified based on the criteria established in DOE Order 474.1A.

3.1.2.2. Radioactive Mixed Waste. Radioactive mixed waste is radioactive waste contaminated with waste that is hazardous/dangerous as defined in 40 CFR 261 and/or WAC 173-303. The radioactive component of mixed waste is regulated under the *Atomic Energy Act of 1954*. The nonradioactive hazardous component of mixed waste is regulated under the *Resource Conservation and Recovery Act of 1976* (RCRA). To the extent that RCRA is not inconsistent with the *Atomic Energy Act of 1954*, mixed waste must be handled in accordance with the requirements of both federal laws (except as noted below).

Note: Normally scheduled routine, radioactive mixed waste shipped solely within the 300 Area (other than those radioactive mixed wastes requiring placarding per 49 CFR 172.504, Table 1) may be packaged and transported in compliance with contractor approved procedures; those radioactive mixed wastes where placarding is required per Table 1 shall be packaged (including marking and labeling) in accordance with DOT hazardous material regulations when shipped solely within the 300 Area.

3.1.2.3 Shipment of Packages on Public Access Roadways. All radioactive material transported over onsite roadways that are open to the public (DOT definition of "in-commerce") must be in compliance with DOT regulations. For a shipment that does not meet these regulations, transporting on the Hanford Site may be done during off-peak hours with the roads closed and/or crossings manned by Benton County Sheriff or Hanford Patrol to prevent public access to the shipment.

This provision, allowed by DOT, removes the "in-commerce" clause by physically eliminating unrestricted public access by the use of gates and/or guards. This process has been reviewed and approved by the DOT and DOE. For these shipments, packages approved only for onsite use by PSSD (e.g., SARP, SEP) are authorized. In addition to the road closure requirements, the applicable PSSD may require other safety constraints such as speed restrictions and/or escorts for the shipment.

Each contractor shall implement a 'Road Closure' procedure prior to performing a shipment requiring restricting public access. The procedure shall direct how Hanford Site areas and roadways to which the public has access are to be closed so that onsite shipments can be conducted. The procedure shall ensure, as a minimum, the following requirements:

- Public access will be restricted for a hazardous materials shipment
- Authorization is required before shipment
- Roads and areas to be closed
- Escort requirements
- Roads are barricaded to exclude the public
- Movement of the shipment into controlled areas and how roads will be controlled
- No traffic on the roads or in the area before shipment
- Emergency vehicle access during a road or area closure
- 'Safe secure trailer' shipment has access during a road closure
- Road closure occurs during off-peak hours.

3.1.2.4 Radiological Vehicle Control Program. Radiological controlled vehicles, formerly referred to as "regulated vehicles," do not qualify as radioactive material shipments by DOT definition and, therefore, are not documented on an onsite *Radioactive Shipment Record* (RSR). Movement and management of these vehicles onsite are covered under an approved program. This program identifies responsibilities and requirements for managing all radiological controlled vehicles including survey requirements and identification of the vehicles as per 10 CFR 835.603, "Radiological areas and radioactive material areas," paragraph (g), which requires the placement of a decal bearing a trefoil symbol and the words "Caution Radioactive Material" posted on the front, back, and both sides of the vehicle.

3.1.2.5 Radioactive Instruments and Articles (Radiological Sources). Radioactive material that meets the definition as described in 49 CFR 173.424, "Excepted packages for radioactive instruments and articles," (i.e., radiological sources) shall be packaged/transported by or under the direction of authorized personnel. In order to use the provisions of 49 CFR 173.424, items (a) through (h) have to be satisfied. This includes the packaging requirements referenced in 49 CFR 173.410, "General design requirements," and notice requirements in 49 CFR 173.422, "Additional requirements for excepted packages containing Class 7 (Radioactive) materials," which requires information on the outside of the packaging. The ORRSR may be used in place of normal package documentation to control onsite movement of these sources.

3.1.3 Onsite Routine Radioactive Shipment Record

Because of their frequency and uniformity of contents and packaging, some onsite radioactive material shipments are considered routine and the contractor may authorize movement and documentation by using a special shipping paper known as an ORRSR.

The ORRSR must provide preprinted shipment requirements and limitations. The preprinted requirements and limitations specified must be adhered to when making a shipment using the ORRSR. Violations or misuse must result in withdrawal of the ORRSR by the contractor.

Requirements, limitations, and supporting documentation specified on the ORRSR shall be reviewed and approved by the following organizations:

- Operations management (management of organization requesting ORRSR)
- Requesting facility's radiological engineering safety
- The transportation service provider lead or designee.

The ORRSR may allow qualified personnel as well as authorized radioactive material shippers to certify that the shipment meets all the requirements as set forth on the ORRSR. The transportation lead or designee shall certify the packaging system meets applicable requirements for onsite shipment.

3.1.4 Onsite Nonradioactive Hazardous Material/Hazardous Waste Shipments

The contractor shall ensure the onsite nonradioactive hazardous materials/hazardous waste shipments are made in compliance with applicable DOT requirements (except as noted below). The contractor is responsible to obtain a DOE approval or waiver from RL for any requirement that cannot be met in accordance with the provisions of this TSD.

Note: Normally scheduled routine, hazardous materials shipped solely within the 300 Area (other than those hazardous materials requiring placarding per 49 CFR 172.504, Table 1) may be packaged and transported in compliance with contractor approved procedures; those hazardous materials where placarding is required per Table 1 shall be packaged (including marking and labeling) in accordance with DOT hazardous material regulations when shipped solely within the 300 Area.

Procedures for onsite shipment of nonradioactive material/waste must include directions on how to accomplish the following tasks:

1. Obtain source documentation for proposed package contents.
2. Determine if the material to be transported meets the definition of a hazardous material as defined in 49 CFR 171.8, "Definitions and abbreviations."
3. Determine how the material to be shipped is defined by chemical, technical, or generic name.
 - a. Determine or define hazardous properties.
 - b. Determine quantity of material to be shipped (by weight, volume, or otherwise, as appropriate).
4. If material to be shipped is currently packaged, determine if packaging is in unimpaired condition and meets DOT regulations for the material contained.
5. If material is a waste, determine if it is a federally regulated hazardous waste (40 CFR 261 and WAC 173-303-070) or a state-only dangerous waste regulated only by WAC 173-303-070 in accordance with WAC 173-303.
6. Consult the HMT in 49 CFR 172.101 "Subpart B-purpose and use of hazardous materials table."
 - a. Determine if material contains a RQ of a hazardous substance under 49 CFR 172.101, Appendix 1, "Hazardous substances other than radionuclides."
 - b. Determine hazard class and/or division (Column 3 of HMT).
 - c. Establish proper shipping name (Column 2 of HMT or WAC 173-303-180(7) for state-only dangerous waste).
 - d. Determine identification number (Column 4 of HMT).
 - e. Determine packing group (Column 5 of HMT).
 - f. Determine labeling requirements that apply to the shipment (Column 6 of HMT).
 - g. Determine special provisions applicable to the shipment (Column 7 of HMT).

Note: 49 CFR 173, contains specific requirements for packaging hazardous material or waste identified in the HMT. These requirements should be closely examined because they specify types of packaging, loading requirements and restrictions, internal packaging, and markings specific to the commodity being shipped.

7. If material is unpackaged, or requires repackaging, select and obtain a container based on the following considerations:
 - a. The applicable determinations of Steps 1 through 5 above.
 - b. Size, shape, weight, and volume of material to be shipped.
 - c. Destination, mode of transport, and consignee's ability to handle receipt of the shipment (air carriers may have different requirements for size and weight limits for sample shipments).
 - d. Availability of shipping containers.
8. Complete a packaging, marking, labeling, and loading instructions checklist if used.
9. Inspect the package to make sure it is in unimpaired physical condition. Verify that the package meets specifications of Column 8 in the HMT. If the package is a DOT performance-oriented package, verify the manufacturer's notification is available. If it is not available, obtain the notification.
10. Fill and close the package following the manufacturer's notification if applicable.
11. Inspect marking and labeling on containers and correct deficiencies as necessary.
12. If shipment is to be made from a radiation control area, obtain a radiation survey to determine radiation and contamination information and to ensure the exterior surfaces are free of contamination.
13. When required, contact the receiving facility and arrange for receipt of the load.
 - a. Make all shipment pre-notifications to the receiving facility.
 - b. Notify the consignee (receiver) of the dates of the shipment, the expected date of arrival, and any special loading and unloading instructions.
14. Coordinate the shipment with transportation and waste handling organizations.
15. Determine the appropriate shipping paper to be used: *Hazardous Material Shipment Record* (HMSR), *Uniform Hazardous Waste Manifest* (UHWM), RSR.
16. When a HMSR is the shipping paper used, complete Form 54-3000-596 or equivalent, i.e., shipment record generated in SWITS database.
17. When a UHWM is the shipping paper used, complete the form or equivalent, i.e., shipment record generated in SWITS database. The completed manifest, and other pertinent shipping papers or instructions, must accompany the shipment. Distribution of the UHWM can be found on the back of the form.

Note: If an outdated UHWM is being used, EPA has authorized the continued use of the forms. The Federal Register has published this authorization allowing preparers of the UHWM to line out void expiration dates and write in the new extension date along with the preparer's initials in the upper right corner.

3.1.5 Onsite Radioactive Material/Waste Shipments

Procedures for the onsite shipment of radioactive material/waste must include directions on how to accomplish the following tasks:

1. Determining if the material to be shipped is radioactive material to be regulated by DOT or this TSD. Refer to 49 CFR 173.403, "Radioactive material."
 - a. Determine the radionuclides present (e.g., ^{239}Pu , ^{233}U , ^{137}Cs).
 - b. Determine the total activity in terms of becquerels (Bq).
 - c. When fissile radionuclides are present (49 CFR 173.403, "Definitions"), determine the grams of fissile material for criticality control.
 - d. Determine if material qualifies as a Hazardous Substance RQ. (See 49 CFR 171.8, "Definitions and abbreviations," and 49 CFR 172.101, Appendix A, Tables 1 and 2, 'Other than radionuclides' and 'Radionuclides'.)
 - e. Determine if material will be shipped as 'normal form' or 'special form' (49 CFR 173.403).
 - f. If normal form, determine the physical form (e.g., solid, liquid, gas) and the chemical form (e.g., nitrate, oxide, elemental) of the radioactive material.
 - g. Determine if other hazardous material is present or if other hazardous characteristics are exhibited (e.g., corrosive, flammable liquid, oxidizing material).
 - h. If the material is a waste, determine if it is a federally regulated hazardous waste (40 CFR 261 and WAC 173-303-070) or a state-only dangerous waste regulated only by WAC 173-303-070 in accordance with WAC 173-303.
 - i. Identify appropriate A_1 or A_2 values for radionuclides present.
 - Determine A_1 values for radionuclides in special form (49 CFR 173.433, "Requirements for determining A_1 and A_2 values for radionuclides and for the listing of radionuclides on shipper papers and labels," and 49 CFR 173.435, "Table of A_1 and A_2 values for radionuclides.").
 - Determine A_2 values for radionuclides in normal form (49 CFR 173.433 and 49 CFR 173.435).
 - j. Determine the quantity category based on the radionuclides present, their A_1 or A_2 values as appropriate, and the total activity in Bq (TBq is also acceptable) in each package (e.g., cask, box, drum). See the following sections of 49 CFR for the quantities listed:
 - LSA I, II, or III radioactive material, or SCO I or II (49 CFR 173.403).

- Excepted Package, 49 CFR 173.403 and 49 CFR 173.421 through 173.426.
- Type A Quantity, 49 CFR 173.431, "Activity limits for Type A and Type B packages," paragraph (a).
- Type B Quantity, 49 CFR 173.431, paragraph (b).
- Highway Route Controlled Quantity, 49 CFR 173.403.

Note: (If a LSA or SCO shipment contains fissile radionuclides, they must meet the requirements for fissile-excepted per 49 CFR 173.453.)

2. Determine the proper shipping name, hazard class, and the United Nations/North American Hazardous Material Identification number. (49 CFR 172.101, HMT, Columns 2, 3 and 4 or WAC 173-303-180(7) for state-only dangerous waste)
3. Determine the packaging, marking, and labeling requirements.
4. A listing of commonly used packages for shipping radioactive material onsite can be found in Appendix A, Package Specific Safety Documentation.
 - a. Select a packaging according to either 49 CFR or this TSD based on the following considerations:
 - Preference to use DOT/DOE/NRC compliant package, or if unavailable, an onsite package approved under this TSD
 - Quantity, type, form, and classification of material to be packaged
 - Need for shielding and/or cooling
 - Size, shape, and weight of material to be shipped; if other hazards present, compatibility with those hazards
 - Destination, mode of transport, and consignee's ability to receive the shipment
 - Dose rate and contamination considerations
 - Availability of shipping packages
 - The additional requirements of HNF-EP-0063 and HNF-3172, *Liquid Waste Processing Facilities Waste Acceptance Criteria*, Rev. 1.
 - b. Determine marking and labeling requirements for each package in compliance with 49 CFR or the PSSD, where applicable. Because of high dose rates, the requirement to mark and/or label may be waived according to the applicable PSSD (e.g., SARP or SEP).

5. Determine all special considerations such as suggested routings, mode of transport, special vehicle requirements, safeguards and security considerations, destination, requested delivery dates, and any other pertinent information concerning the shipment.
6. Arrange for a radiological survey of the package(s) to determine that dose rate limits and removable contamination limits are not exceeded.

Note: Certain packaging systems are exempt from onsite limits and are instead controlled by limits specified in the approved PSSD.

7. Determine the category and the security requirement if the material is SNM (see DOE Order 474.1A). Obtain Safeguard's review if necessary.
8. Determine the shipping paper to be used.
9. Determine if any requirements found in Chapter VIII, Section 3.1.2, Radioactive Material Shipments - General Provisions, apply to the material.
10. Identify package, loading, and closure instructions.
11. Ensure the appropriate emergency response phone number is documented on the shipment record (e.g., RSR, Hazardous Waste Manifest).
12. Schedule onsite transportation from the originating facility to receiving facility. Schedule transportation equipment suitable for the shipment. Schedule radiological control technician (RCT) support or road closure support if required.
13. Arrange for a RCT to perform a removable contamination and deep dose radiation survey of the vehicle and vehicle cab for release per applicable limits and swipe requirements.
14. Ensure the vehicle is appropriate for the load and free of defects that could affect the safe transport of the material. Verify that a copy of the current Hazardous Material Certificate of Registration is in the cab of each vehicle that may require placards for the transport of hazardous material or is a commercial motor vehicle with a gross vehicle weight rating of 4,356 kg (10,000 lb).
15. Ensure preventative maintenance or equivalent inspections are current and required safety equipment is on the transport vehicle.
16. Ensure required emergency equipment is on the transport vehicle: Warning devices for stopped vehicles such as bi-directional reflective triangles or flares, spare fuses (if equipped) and charged fire extinguisher rated 10 BC or more.
17. Ensure the RCT has verified the radiation level in the cab is within acceptable limits (e.g., less than 0.02 mSv/hr).

18. Ensure supervision of the loading and securing of packages into or onto the transport vehicle to ensure compliance with applicable loading and tiedown requirements as specified in 49 CFR 393, "Subpart I-Protection Against Shifting or Falling Cargo;" 49 CFR 177.834(a), "General requirements;" or procedures found in the PSSD (e.g., SARP or CoC). Ensure riggers or truck drivers are aware of any tiedown requirements that may be listed in associated CoCs, PSSD, work procedures.
19. Ensure the carrier has received instructions pertaining to any administrative controls specified in the PSSD for transport of the package(s).
20. Advise the receiver of the estimated time of arrival and request confirmation of delivery on inter-area shipments exceeding Type A quantities.

Note: Depending on the quantities shipped, some SNM shipments shall require classified notifications. Information and quantity limits can be found in DOE Order 474.1A and should be consulted prior to all SNM shipment.

21. When applicable, make the following notifications when performing SNM shipments:
 - Notify the on-duty Hanford Fire Department battalion chief of the starting time, estimated duration of the shipment, route to be followed, and destination to expedite assistance, if needed. Examples are road closure, and "no fire" provision.
 - Notify Hanford Patrol of the departure time, route, destination, and estimated time of arrival. Request escort service as required.
22. Ensure completion of the applicable shipment checklist.
23. Resolve all questions concerning any special services required while the package is being transported to the consignee's facility (e.g., cooling water in case of delay enroute, control of personnel because of a radiation dose, escort by Hanford Patrol or Benton County Sheriff for traffic control).
24. Ensure the original shipping paper(s) are provided to the driver making the shipment.
25. Ensure the proper placard is identified for the material to be transported.
26. Ensure the transport vehicle is placarded "Radioactive," when required, and as appropriate for other hazardous material being transported in accordance with DOT regulations or as required when utilizing PSSD (e.g., SARPs, SEPs). Ensure placarding is documented on shipping paper in appropriate box.
27. A copy of the shipping document shall be maintained as a record.

4.0 REFERENCES

- 10 CFR 71, "Packaging and Transportation of Radioactive Material," *Code of Federal Regulations*, as amended.
- 10 CFR 835, "Occupational Radiation Protection," *Code of Federal Regulations*, as amended.
- 40 CFR 261, "Identification and Listing of Hazardous Waste," *Code of Federal Regulations*, as amended.
- 49 CFR 107, "Hazardous Materials Program Procedures," *Code of Federal Regulations*, as amended.
- 49 CFR 171, "General Information, Regulations, and Definitions," *Code of Federal Regulations*, as amended.
- 49 CFR 172, "Hazardous Materials Table, Special Provisions, Hazardous Materials Communications, Emergency Response Information, and Training Requirements," *Code of Federal Regulations*, as amended.
- 49 CFR 173, "Shippers--General Requirements for Shipments and Packagings," *Code of Federal Regulations*, as amended.
- 49 CFR 177, "Carriage by Public Highway," *Code of Federal Regulations*, as amended.
- 49 CFR 178, "Specifications for Packagings," *Code of Federal Regulations*, as amended.
- 49 CFR 393, "Parts and Accessories Necessary for Safe Operation," *Code of Federal Regulations*, as amended.
- DOE Order 474.1A, *Control and Accountability of Nuclear Materials*, U.S. Department of Energy, Washington, D.C.
- DOT, *North American Emergency Response Guidebook*, U.S. Department of Transportation, Research and Special Programs Administration, Office of Hazardous Materials Initiatives and Training.
- HNF-EP-0063, *Hanford Site Solid Waste Acceptance Criteria*, Fluor Hanford, Richland, Washington.
- HNF-3172, *Liquid Waste Processing Facilities Waste Acceptance Criteria*, Fluor Hanford, Richland, Washington.
- RLID 473.1, *Protection of Safeguards and Security Interests*, Implementing Directive, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

WAC 173-303, "Dangerous Waste Regulations," *Washington Administrative Code*, Washington State Department of Ecology, Olympia, Washington.

CHAPTER IX. NONROUTINE TRANSFERS

This chapter describes the processes and procedures for dealing with nonroutine or emergency transfers. Conditions and circumstances for nonroutine or emergency transfers are defined, as well as applicable procedures and the approval process.

1.0 NONROUTINE AND EMERGENCY SHIPMENT PROCEDURES

This chapter identifies how nonroutine and emergency shipments (transfers) are handled under the onsite hazardous materials T&P safety program. Nonroutine onsite shipments (transfers) and emergency transfers are unusual and unpredictable events that may occur onsite. A nonroutine shipment or transfer may be necessary to facilitate the movement of a piece of equipment that cannot be easily cleaned or dismantled. For example, a test loop contaminated with sodium. Offsite, a request for an exemption would be submitted to DOT by way of DOE. Onsite, DOE must grant the exemption to the onsite requirement. In an emergency, to mitigate a spill or release event, and/or provide immediate protection to workers, the public, or to the environment as a result of an accident or incident, immediate action may be required. For example, a cargo tank could be transported onsite and a road obstruction or collision causes damage to the containment boundary allowing a small spill to begin. The Incident Commander for emergency responders and site management may decide to trans-load the liquid contents to another cargo tank to mitigate the event, minimize the clean up needed, and facilitate recovery operations. Emergency procedures will define such events and protocols to be used in making such a decision.

1.1 NONROUTINE SHIPMENTS

Nonroutine onsite shipments (transfers) are handled by emulating the DOT regulations associated with exemptions (see 49 CFR 107.105). To request an exemption from the onsite hazardous material T&P requirement identified in this TSD the requester shall prepare a request following the format identified in 49 CFR 107.105(c), "Description of exemption proposal." The exemption request should address the specific regulation in the TSD from which relief is being sought. The request shall include the information identified in 49 CFR 107.105. The request shall be submitted to RL. The length of an onsite exemption granted by the RL Manager will correspond with the periods identified in 49 CFR 107.113(h). Requests for extensions shall be submitted to RL containing the information as identified in 49 CFR 107.109.

1.2 EMERGENCY SHIPMENTS

The need for an emergency shipment will be determined in accordance with DOE/RL-94-02, *Hanford Emergency Management Plan* (HEMP). The Emergency Duty Officer shall have the responsibility for event classification and activation of the Hanford Emergency Response Organization as appropriate. Shipments (transfers) are handled in accordance with the site

emergency preparedness plan. Additional information about the HEMP is presented in Chapter XII, Incident Reporting and Emergency Response.

2.0 REFERENCES

49 CFR 107, "Hazardous Materials Program Procedures," *Code of Federal Regulations*, as amended.

DOE/RL-94-02, *Hanford Emergency Management Plan*, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

CHAPTER X. PERSONNEL TRAINING AND QUALIFICATION

This chapter describes the qualification and training requirements for Hanford Site contractor hazmat personnel, authorized shippers, drivers, and USQ_T evaluators. Individual facilities and organizations, to which personnel may be assigned, may establish additional qualifications and required training based on specific job task analysis, job classification, or individual training plan.

1.0 REQUIREMENTS

RL, ORP, and each Hanford Site contractor must have a personnel training and qualification program that emulates the requirements identified in 49 CFR 172, Subpart H, "Training."

1.1 TRAINING PROGRAM REQUIREMENTS

All hazmat employees involved in T&P activities must be trained and tested to the training requirements identified in 49 CFR 172, Subpart H. Training requirements include the following:

- General awareness/familiarization training designed to enable the employee to recognize and identify hazardous materials consistent with the hazard communication requirements in 49 CFR 172 and this TSD.
- Function-specific training specifically applicable to the functions the employee performs. This includes modal-specific training requirements as described in 49 CFR 174 through 177.
- Safety training concerning the following:
 - Emergency response information
 - Measures to protect the employee from the hazardous materials to which they may be exposed
 - Methods and procedures for avoiding accidents.

Drivers or other persons operating commercial motor vehicles as defined in the Federal Motor Carrier Safety Regulations (FMCSR) must be trained and tested in accordance with 49 CFR 177.816, "Driver training."

A new hazmat employee, or a hazmat employee changing job functions, must complete initial training within 90 days after employment or a change in job function. The employee may perform those functions prior to the completion of required training -- provided the employee performs those functions under the direct supervision of a properly trained and knowledgeable hazmat employee.

DOT and this onsite emulation require this training and certification to occur every 3 years [49 CFR 172.704(c)(2)]. Each hazmat employer is responsible for ensuring hazmat employees are aware of any regulatory changes that may affect their specific functions in the onsite T&P program.

RL, ORP, and each Hanford Site contractor's training program must identify and qualify the following personnel through adequate training and testing:

- All hazmat employees (e.g., operators, technicians, warehousemen, drivers, managers, supervisors) who in the course of their employment affect transportation safety as defined by the DOT must be trained in accordance with 49 CFR 172.704. This training must include general awareness, function specific and safety training, and driver training when the employee operates a motor vehicle used to transport hazardous materials, substances, and wastes.
- Designated shippers, authorized shippers, or shipping agents acting on behalf of the contractor or RL and ORP must receive advanced hazmat training to address characterization and classification of material for shipment, selection and use of packagings, hazard communications, regulatory compliance, operational procedures, and transport protocols. Training must be given commensurate with the employee's responsibilities and authority to execute shipments.
- Drivers or other persons operating commercial motor vehicles as defined in the FMCSR must possess a CDL and must be trained and tested in accordance with 49 CFR 177.816. Drivers must also receive training on contractor or project procedures and protocols.
- USQ_T evaluators must be trained and tested in the USQ_T and its application with the TSD. The following training courses, academic degree, and experience requirements, as judged by the contractor, must be met to qualify as an USQ_T evaluator. Except as noted, exemptions from and the use of equivalencies to the identified requirements may be authorized by contractors, provided the same level of competence is obtained:
 - DOE-Headquarters packaging Certification Course
 - DOE USQ Course
 - Hanford Specific USQ_T Course (no exemption allowed)
 - Education – BS degree
 - Experience – 2 years – 1 in transportation
 - On-the-job training of at least two USQ_TD (may be satisfied with practical examination).

1.2 EMPLOYER CERTIFICATION

RL, ORP, and each Hanford Site contractor must have a formal process in place to certify that each hazmat employee has been trained and tested as required in 49 CFR 172.704(d)(5) and this TSD.

1.3 SHIPPER AUTHORIZATION

RL, ORP, and each Hanford Site contractor must have a formal process in place to authorize employees as designated shippers, authorized shippers, or shipping agents acting on behalf of the contractor and/or RL, ORP. This process must include the following:

- Identification of qualifications and training
- Management delegation of signature authority.

Each contractor must maintain an Authorized Shippers List, which is updated regularly.

Only persons holding delegated shipper authority may be authorized to sign shipping documents on behalf of the contractor and/or RL, ORP.

RL, ORP, and each Hanford Site contractor authorizing shippers may, in addition to DOT training requirements, identify other federal, state, and local requirements necessary to maintain "authorized shipper" status.

1.4 USQ_T EVALUATOR AUTHORIZATION

RL, ORP, and each Hanford Site contractor that has an USQ_T must have a process for designating USQ_T evaluators.

2.0 DOCUMENTATION AND RECORDS REQUIREMENTS

Documentation and records requirements for onsite hazmat employees are to be in accordance with the requirements identified in 49 CFR, Subpart H and Chapter XI, Documentation and Record Keeping.

3.0 REFERENCES

49 CFR 172, "Hazardous Materials Table, Special Provisions, Hazardous Materials Communications, Emergency Response Information, and Training Requirements," *Code of Federal Regulations*, as amended.

49 CFR 174, "Carriage by Rail," *Code of Federal Regulations*, as amended.

49 CFR 175, "Carriage by Aircraft," *Code of Federal Regulations*, as amended.

49 CFR 176, "Carriage by Vessel," *Code of Federal Regulations*, as amended.

49 CFR 177, "Carriage by Public Highway," *Code of Federal Regulations*, as amended.

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CHAPTER XI. DOCUMENTATION AND RECORD KEEPING

This chapter identifies documentation and records requirements for different package categories. It also identifies or references processes and procedures for managing this documentation throughout the Hanford Site.

1.0 DOCUMENTATION REQUIREMENTS

Record requirements include retention of items such as packaging documentation, training records, vehicle maintenance and inspection records, and documentation associated with the onsite transfer of hazardous material.

Records shall be legible throughout the retention period and copies shall be reproducible and legible. Measures shall be taken to preserve records in such a manner as to preclude deterioration of the records.

2.0 PACKAGING DOCUMENTATION

Packaging documentation for onsite transportation of hazardous materials shipments (transfers) must be in full compliance with applicable DOT regulations or the onsite emulation of the applicable regulations.

Packagings that fully meet a DOT specification or DOT/NRC performance standard (i.e., UN1A2, MC 412, DOT-7A Type A) when used onsite must meet all the documentation requirements for that DOT specification or performance standard as required in 49 CFR 171 through 179.

Packagings that do not meet a DOT specification or DOT performance standard when used onsite must be approved by RL and ORP, or when applicable, the Hanford Site contractor. The documentation must meet the onsite implementation of the DOT documentation requirements, those identified in this TSD, and those identified in the PSSD.

2.1 EXCEPTED PACKAGING DOCUMENTS

Excepted packagings, frequently referred as "strong tight packagings," have no formal documentation requirements associated with the procurement or use of any individual package. The shipper must use his or her professional judgment to determine if the packaging and package as offered for transportation meets 49 CFR 173.24, "General requirements for packagings and packages;" 173.24a, "Additional general requirements for nonbulk packagings and packages;" and 173.24b, "Additional general requirements for bulk packagings;" as appropriate. For radioactive materials, excepted packagings must also meet 49 CFR 173.410, "General design

requirements.” In addition to determining if the package meets the requirements, consider also the needs of the offeror and receiver, including their need for stacking.

2.2 INDUSTRIAL PACKAGING DOCUMENTS

For industrial packagings except IP-1 as required by 49 CFR 173.411, “Industrial packagings,” the shipper must maintain on file for at least 1 year after the latest shipment complete documentation of tests and an engineering evaluation or comparative data showing that the construction methods, packaging design, and materials of construction comply with the applicable packaging specification (i.e., IP-2, IP-3).

2.3 TYPE A PACKAGING DOCUMENTS

For 7A Type A packagings as required by 49 CFR 173.415, “Authorized Type A packages,” the shipper must maintain on file for at least 1 year after the latest shipment, complete documentation of tests and an engineering evaluation or comparative data showing that the construction methods, packaging design, and materials of construction comply with the applicable packaging specification.

2.4 TYPE B AND FISSILE PACKAGING DOCUMENTS

For packagings used onsite for the shipment of Type B and/or fissile materials and wastes, the contractor must have a NRC or DOE-approved SARP or develop and maintain an onsite PSSD,

Onsite PSSDs are generally developed and approved for multi-purpose, multi-use, or multi-user packagings. See Chapter VII, Safety Assessment Methodology, Section 3.1.2 for application content details.

3.0 TRAINING AND QUALIFICATION RECORDS

RL, ORP, and each Hanford Site contractor offering transportation, or transports hazardous material on roads to which the public has unrestricted access, must meet the requirements defined in 49 CFR 107, Subpart G, “Registration of Persons Who Offer or Transport Hazardous Materials.” For onsite shipments (transfers), an onsite emulation of this regulation has been implemented.

Each organization shall maintain training records in accordance with 49 CFR 172.704 (d) for all personnel who have a direct effect on transportation safety.

3.1 HAZMAT EMPLOYEES

RL, ORP, and each Hanford Site contractor with a hazmat employee involved in onsite T&P activities must meet the documentation and record keeping requirements defined in 49 CFR 172, Subpart H, "Training."

3.2 DRIVERS

RL, ORP, and each Hanford Site contractor driver used to transport hazardous materials onsite must meet the documentation and record keeping requirements defined in 49 CFR 177.816, "Driver training;" 49 CFR 383, Subpart J, "Commercial Driver's License Document;" and 49 CFR 391, "Qualifications of Drivers."

RL, ORP, and each Hanford Site contractor shall also maintain an accident register and accident report for drivers, in accordance with 49 CFR 390.15, for a period of 1 year after an accident occurs.

3.3 USQ_T EVALUATORS

RL, ORP, and each Hanford Site contractor with an USQ_T must meet the documentation and record requirements identified in 10 CFR 830 for the USQ process.

4.0 VEHICLE MAINTENANCE AND INSPECTION RECORDS

RL, ORP, and each Hanford Site contractor motor carrier that uses motor vehicles onsite to ship hazardous materials must meet the requirements defined in 49 CFR 396, "Inspection, Repair, and Maintenance," for motor vehicles.

5.0 SHIPPING PAPERS

The requirements for documentation and record keeping shall be established from U.S.C., Title 49 CFR -- Transportation, Subtitle III -- General and Intermodal Programs, Chapter 51 -- "Transportation of Hazardous Material," Section 5110, Shipping papers and disclosure.

For onsite shipments, the HMSR, which is filled out by the shipper, may serve as the only documentation to accompany the shipment onsite.

6.0 REFERENCES

10 CFR 830, "Nuclear Safety Management," *Code of Federal Regulations*, as amended.

- 49 CFR 107, "Hazardous Materials Program Procedures," *Code of Federal Regulations*, as amended.
- 49 CFR 171, "General Information, Regulations, and Definitions," *Code of Federal Regulations*, as amended.
- 49 CFR 172, "Hazardous Materials Table, Special Provisions, Hazardous Materials Communications, Emergency Response Information, and Training Requirements," *Code of Federal Regulations*, as amended.
- 49 CFR 173, "Shippers--General Requirements for Shipments and Packagings," *Code of Federal Regulations*, as amended.
- 49 CFR 174, "Carriage by Rail," *Code of Federal Regulations*, as amended.
- 49 CFR 175, "Carriage by Aircraft," *Code of Federal Regulations*, as amended.
- 49 CFR 176, "Carriage by Vessel," *Code of Federal Regulations*, as amended.
- 49 CFR 177, "Carriage by Public Highway," *Code of Federal Regulations*, as amended.
- 49 CFR 178, "Specifications for Packagings," *Code of Federal Regulations*, as amended.
- 49 CFR 179, "Specifications for Tank Cars," *Code of Federal Regulations*, as amended.
- 49 CFR 383, "Commercial Driver's License Standards; Requirements and Penalties," *Code of Federal Regulations*, as amended.
- 49 CFR 390, "Federal Motor Carrier Safety Regulations; General," *Code of Federal Regulations*, as amended.
- 49 CFR 391, "Qualifications of Drivers," *Code of Federal Regulations*, as amended.
- 49 CFR 396, "Inspection, Repair, And Maintenance," *Code of Federal Regulations*, as amended.

CHAPTER XII. INCIDENT REPORTING AND EMERGENCY RESPONSE

This chapter describes requirements for incident reporting, emergency response, and references Hanford Site-specific policies, manuals, and procedures for emergency management. It also addresses emergency planning, drills and exercises, and Hanford Site response capabilities.

1.0 REPORTING REQUIREMENTS

Actions taken to implement incident reporting and emergency response requirements associated with onsite T&P activities are identified in the HEMP (DOE/RL-94-02). This TSD requires compliance with the requirements identified in DOE/RL-94-02 by RL and ORP, RL and ORP primary contractors and subcontractors. DOE maintains policies, programs, and procedures (DOE-0223) to implement the HEMP for onsite T&P activities.

2.0 HANFORD SITE EMERGENCY MANAGEMENT PROGRAM BASIS

The comprehensive Hanford Site emergency management program is based on and commensurate with the hazards and consequences associated with facilities and activities on the Hanford Site (i.e., developed consistent with a graded approach), offsite facilities that may impact the Hanford Site, and onsite and offsite DOE transportation emergency preparedness activities involving radiological and nonradiological hazardous materials.

3.0 HANFORD TRANSPORTATION EMERGENCY PREPAREDNESS PROGRAM

The Hanford Transportation Emergency Preparedness Program provides the framework for response to onsite and offsite transportation incidents involving radiological and nonradiological hazardous material. For transportation planning purposes, shipments transported on roadways and in areas of the Hanford Site where public access is restricted are exempt from the DOT regulations found in 49 CFR. Shipments transported on roadways and in areas where the public is granted unrestricted access are considered "in commerce" and shall be regulated under the DOT regulations found in 49 CFR. When public access control is extended to areas within the Hanford Site boundary, to which the public had previously been granted unrestricted access (e.g., road closure), shipments may be made under onsite control. Onsite controls may be used on roadways and in areas for as long as the public access restriction remains in force.

For transportation incidents that occur on the Hanford Site, an Incident Command System is used to mitigate the situation. This includes shipments of nongovernment material being moved in full compliance with DOT/NRC requirements. Upon notification of the event, the Hanford Fire Department shall assume incident command responsibilities. The Emergency Duty Officer shall have the responsibility for event classification and activation of the Hanford Emergency Response Organization as appropriate.

4.0 HEMP SUMMARY

The HEMP (DOE/RL-94-02) details the Hanford Site emergency organization, authorities, and responsibilities for response to and mitigation of emergency events involving facilities and activities on the Hanford Site including transportation. These events include the full spectrum of operational emergencies, natural phenomena, transportation events, and safeguard and security emergencies. This HEMP also describes the authorities, responsibilities, and agreements for response to offsite and near-site facility emergencies that have the potential for detrimentally affecting the health of personnel and safety of operations at the Hanford Site.

DOE maintains procedures necessary to implement the emergency management program described in the HEMP. The procedures contain detailed information and specific instructions, including response actions, associated precautions and prerequisites, and identification of responsible individuals needed to perform the appropriate action during a drill, exercise, or actual emergency.

For the Hanford Site, these procedures include, but are not be limited to, the following:

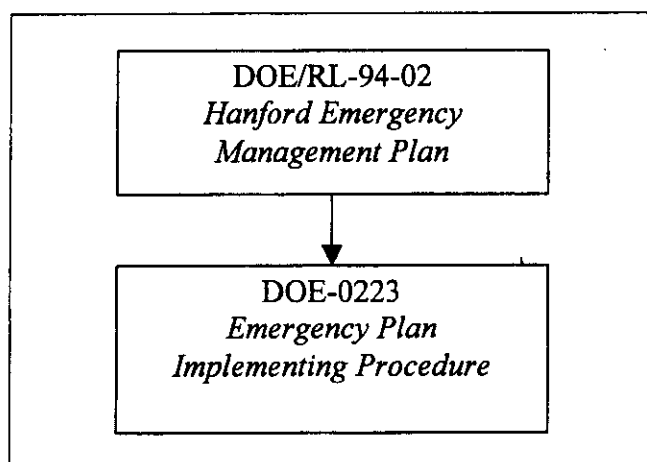
- The operation of the Hanford Incident Command System and responsibilities of the Incident Command Organization
- The responsibilities for the DOE Hanford Emergency Operations Center, which includes the Policy Team, Site Management Team, and the Joint Information Center
- Recognition, categorization/classification, and notification of emergencies and other incidents
- Protective action recommendations
- Response to nonradiological hazardous substance spills or releases during transportation incidents occurring onsite, which are not covered by unit-specific contingency plans or building emergency plans
- Response to polychlorinated biphenyl spills or releases in accordance with 40 CFR 761, "Polychlorinated Biphenyls (PCBs) Manufacturing, Processing, Distribution in Commerce, and Use Prohibitions"
- Termination, reentry, and recovery for DOE order emergencies and events that meet RCRA contingency plan implementation criteria
- Response to onsite and offsite shipments of DOE-owned radiological and nonradiological hazardous materials.

In addition to the program for response to and mitigation of emergencies, the HEMP also provides direction on the activities necessary to ensure emergency preparedness on the Hanford Site such as training, drills, exercises, and assessments. The authority and responsibility for interfaces with offsite organizations responsible for protecting the public and the environment, including those agencies that may provide or request support in the event of an emergency, is also delineated.

The DOE responsibility to provide, upon request, radiological advice and assistance to other federal, tribal, state, or local governments under the Radiological Assistance Program is defined in DOE/RL-92-49, *U.S. Department of Energy Radiological Assistance Program Response Plan Region 8*.

Figure 12-1 provides the hierarchy for emergency preparedness documentation for transportation.

Figure 12-1. Transportation Emergency Preparedness Documentation Hierarchy.



5.0 REFERENCES

40 CFR 761, "Polychlorinated Biphenyls (PCBs) Manufacturing, Processing, Distribution in Commerce, and Use Prohibitions," *Code of Federal Regulations*, as amended.

49 CFR, "Transportation," *Code of Federal Regulations*, as amended.

DOE-0223, *Emergency Plan Implementing Procedure*, U.S. Department of Energy, Richland Field Office, Richland, Washington.

DOE/RL-92-49, *U.S. Department of Energy Radiological Assistance Program Response Plan Region 8*, U.S. Department of Energy, Richland, Operations Office, Richland, Washington.

DOE/RL-94-02, *Hanford Emergency Management Plan*, U.S. Department of Energy, Richland, Operations Office, Richland, Washington.

CHAPTER XIII. TRANSPORT VEHICLE OPERATIONS

This chapter describes the requirements for motor vehicle inspection and maintenance programs for T&P activities on the Hanford Site. References to specific requirements and procedures are provided for use by each contractor or subcontractor conducting transportation operations within Hanford Site boundaries.

1.0 ONSITE TRANSPORTATION PROGRAM REQUIREMENTS

For transportation, equivalent safety is achieved by compliance with DOT requirements or by emulating the DOT regulations associated with hazardous materials transportation.

Modes of transportation used for the onsite shipment of hazardous materials regulated by the DOT for hazardous materials shipped in commerce are covered by the requirements of this TSD. DOT requirements cover shipments by rail, air, water, motor vehicle, and pipeline.

1.1 GENERAL

RL, ORP, and each Hanford Site contractor that transports onsite hazardous material shipments must do so in a safe and regulatory compliant manner. Compliance with the DOT regulations for shipment by rail, air, water, and motor vehicle or the onsite emulation of them is required. Contractors must conduct transport operations in accordance with this TSD, which imposes regulatory requirements, best industry practices, and other site procedures deemed necessary by the DOE to achieve equivalent safety and be in accordance with applicable federal and state regulations.

1.2 MODAL APPLICATION

1.2.1 Rail

Onsite rail shipments of hazardous materials are currently suspended. If onsite rail transportation is started, shipments must be made in full compliance with applicable DOT regulations or RL and ORP-approved emulation of the applicable regulations.

1.2.2 Air

No onsite air transportation is currently performed. If onsite air transportation is started, shipments must be made in full compliance with applicable DOT regulations or RL and ORP-approved emulation of the applicable regulations.

1.2.3 Water

All onsite transportation by water (carriage by vessel) is performed in areas open to the public. All onsite shipment by water must be in full compliance with the applicable DOT regulations.

1.2.4 Motor Vehicle

Onsite transportation of hazardous materials shipments by motor vehicle must be made in full compliance with applicable DOT regulations or the onsite emulation of the applicable regulations. When the emulation of DOT requirements is used, RL will implement those actions identified for the DOT within the DOT regulations. RL, ORP, and their contractors will implement those actions identified for the user of the DOT regulations.

Compliance with applicable DOT regulations or equivalent standards must be met for the operation, inspection and maintenance of vehicles, and/or the selection, qualification and training of management, drivers, inspection personnel, and maintenance staff.

1.2.5 Pipeline

This TSD does not include onsite transportation of hazardous materials by pipeline. The safety of onsite transportation of hazardous materials by pipeline is covered in a facility SAR or other DOE-approved safety or authorization basis documents.

1.3 TRANSPORT VEHICLE OPERATION

RL, ORP, and their contractors performing transportation operations, fleet operations, or vehicle maintenance for onsite transportation of hazardous material must implement programs, policies, and procedures necessary to meet the provisions of this TSD. Program requirements must address the following key elements as applicable to the specific onsite transportation operation:

- Transportation workplace drug testing program
- Controlled substance and alcohol use and testing
- Driver licensing standards
- Compliance with FMCSRs or approved alternatives
- Qualification of drivers
- Driving of Commercial Motor Vehicles as defined by DOT
- Equipment necessary for safe operation
- Hours of service for drivers
- Inspection, repair, and maintenance
- Transportation of hazardous materials, driving, and parking rules

- Employee safety and health standards
- Training.

1.4 MOTOR VEHICLE

The following is a list of transportation program requirements that must be considered in each Hanford Site contractor's transportation program. All transport vehicle operations involving the onsite transfer of hazardous materials, substances, and wastes must comply with the FMCSR found in 49 CFR 40, "Procedures for Transportation Workplace Drug Testing Programs," Sections .325, .355 through .399; DOT Hazardous Material Regulations in 49 CFR 177; or must be performed under a RL-approved program meeting the intent of these regulations, which provides an equivalent degree of safety for site personnel, the public, and the environment. All variances from the FMCSR must be documented in the contractor's transportation operations manual, policies, and/or procedures, and must be approved by RL.

Programs performed under this TSD must include all applicable items from the following listing:

- Vehicles used to transport hazardous materials, substances, and wastes onsite must be in accordance with FMCSR specifications and standards when procured and placed into service, and must be inspected and maintained in accordance with these requirements.
- Management personnel, vehicle drivers, inspection personnel, and maintenance staff must be trained and qualified in accordance with the FMCSR and the DOT Hazardous Material Regulations when applicable.
- All vehicle maintenance must be performed and meet applicable DOT and Government Services Administration (GSA) requirements and schedules for the type of vehicle and type of service for which it is used.
- Periodic inspections must be performed on motor vehicles in accordance with DOT regulations at intervals not to exceed 12 months.
- Cargo tank trailers controlled by Hanford Site contractors must be inspected, tested, and maintained in accordance with applicable DOT regulations.
- Preventative maintenance schedules must be maintained for each type of vehicle. The type of motor vehicle determines the frequency and level of maintenance. Maintenance must be provided in accordance with the GSA vehicle maintenance schedules.

1.5 OPERATOR DUTIES

The following list provides the transportation program requirements associated with operator duties identified as key to accomplishing the required level of safety in Hanford Site contractor transportation programs. Programs performed under this TSD must include all applicable items from the following listing:

- Drivers must inspect all commercial motor vehicles in accordance with DOT pre-trip and post-trip requirements.
- Drivers hauling placarded loads must have a current and properly endorsed CDL and meet all driver qualifications including participation in a drug and alcohol testing program meeting applicable federal standards.
- Drivers must report all vehicle accidents or incidents, driving violations, fines, or penalties immediately to their respective contractor management.
- Drivers must comply with all FMCSR or approved alternatives including driver hours of service, driving and parking rules, pre-trip and post-trip inspections, maintenance of shipping documents and emergency response guides, and other safety equipment and tiedown/load securement requirements.

2.0 INSPECTION AND MAINTENANCE PROGRAM REQUIREMENTS

RL, ORP, and their contractors must have programs, policies, and procedures in place to ensure all transport vehicles used for the onsite transport of hazardous materials, substances, and wastes undergo daily and periodic inspections and maintenance in accordance with DOT and FMCSR regulations and in accordance with applicable GSA schedules. Variance from these federal standards must be documented and approved by RL.

A list of the inspection and maintenance program requirements, which must be considered in the Hanford Site contractor's transportation programs, is provided below. Programs performed under this TSD must include all applicable items from the following:

- Assignment of responsibilities
- Identify number and type of vehicles and other transport equipment available for service and identify equipment load ratings
- Identify type of service each vehicle is assigned
- Identify GSA inspection and maintenance schedules applicable to each vehicle or vehicle type

- Identify location of all inspection and maintenance records
- Specify frequency of audits, findings, and corrective actions.

3.0 ONSITE DRIVER QUALIFICATION AND TRAINING

RL and ORP contractors operating vehicles for onsite transport of hazardous materials, substances, and wastes must have programs, policies, and procedures in place ensuring all drivers operating the vehicles are properly trained and qualified in accordance with FMCSR and DOT Hazardous Materials Regulations. Variance from these federal standards must be documented and approved by RL.

A list of driver requirements identified as key to accomplishing the required level of safety in previous Hanford Site contractor transportation programs is provided below. Programs performed under this TSD must include all applicable items from the following:

- Assignment of responsibilities
- Basic driver qualification (consider age, health, driving experience)
- Drug and alcohol testing
- Licensing and endorsements
- Initial and recurrent training
- Driver personal and work driving record
- Maintenance of driver records.

5.0 REFERENCES

49 CFR 40, "Procedures for Transportation Workplace Drug Testing Programs," *Code of Federal Regulations*, as amended

49 CFR 177, "Carriage by Public Highway," *Code of Federal Regulations*, as amended.

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APPENDIX A
APPROVED PACKAGE SPECIFIC SAFETY DOCUMENTS LIST

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Table 2. Hanford Site (Onsite) PSSDs.....	App A-20

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APPENDIX A

APPROVED PACKAGE SPECIFIC SAFETY DOCUMENTS LIST

The following tables provide listings of onsite package specific safety documents (PSSD). Table 1, Existing Hanford Site (Onsite) PSSDs, contains packagings being used at the time this Hanford Sitewide Transportation Safety Document (TSD) was prepared. These documents, at the revision level identified, by inclusion on this listing, are made a part of this TSD and are approved for continuing use. Documents that were previously used that do not appear on this list may not be used after the effective date of this TSD.

Table 2, Hanford Site (Onsite) PSSDs, contains packagings approved following the requirements of this TSD. Documents approved as required by this TSD may be added to this listing. Requests for changes to Table 2 shall be submitted to the company contracted to maintain the TSD.

The listings in the tables are divided into four parts. Part One contains those packagings compliant with U.S. Department of Transportation packaging requirements. Part Two contains those packagings judged to meet the equivalent packaging standards identified in Chapter VI, Site-Specific Standards, Procedures, and Instructions, of the TSD. Part Three contains those packagings judged to meet the nonequivalent packaging standards. Part Four contains exemptions to the TSD requirements authorized by the RL Manager.

Table 1. Existing Hanford Site (Onsite) PSSDs.

Document Number	Current Revision	Clearance Date	Title	Configuration Manager	Package Custodian
Part One -- DOT Compliant Packaging Listing of August 16 2001					
NRC					
USA/0361/B(U)F-85	7		Model No. PAT-1		
USA/4888/B()	11		Model No. SENTINEL-25A, LCG-25A; SENTINEL-25B, LCG-25B; SENTINEL-25C, LCG-25C; SENTINEL-25C3, -25D, -25E, and -25F		
USA/4986/AF	38		Model No. RA-3		
USA/5086/B(U)F	11		Model No. UNC-2600		
USA/5149/B()F	10		Model No. 814A		
USA/5580/B()F	7		Model No. S5W Power Unit		
USA/5607/B()F	11		Model No. T-2		
USA/5796/B(U)	14		Model Nos. 181375 and 181361		
USA/5797/B(U)F	14		Model No. Inner High Flux Isotope Reactor (HFIR) Unirradiated Fuel Element Shipping Container, and outer HFIR Unirradiated Fuel Element Shipping Container		
USA/5805/B()	20		Model No. CNS 3-55		
USA/5830/B()	8		Model No. SNAP-21		
USA/5862/B()	8		Model No. SENTINEL-100F		
USA/5926/B()F	17		Model No. GE-100		
USA/5939/B()F	28		Model No. 1500		
USA/5957/B()F	27		Model No. BMI-1		
USA/5979/B()	10		Model No. 5979		
USA/6058/B()F	14		Model No. B-3		
USA/6078/AF	28		Model Nos. 927A1 and 927C1		
USA/6206/AF	27		Model No. Model B		
USA/6280/B()	7		Model No. A-0109 Irradiator in A-0117 Overpack		
USA/6294/AF	26		Model No. UNC-2901		

Table 1. Existing Hanford Site (Onsite) PSSDs.

Document Number	Current Revision	Clearance Date	Title	Configuration Manager	Package Custodian
USA/6346/B()F	28		Model No. FSV-1		
USA/6347/AF	9		FSV-3		
USA/6357/AF	8		Model No. NNFD-10		
USA/6400/B()F	25		Model No. 6400		
USA/6553/AF	17		Model No. PADUCAH TIGER		
USA/6574/B()	27		Model No. 3-82B		
USA/6581/AF	31		Model No. 51032-1		
USA/6613/B(U)	8		Model No. 702		
USA/6642/B()	6		Model No. 4.5-TON CF		
USA/6717/B(U)	10		Model No. 6717-B		
USA/6786/B()F	6		Model Nos. URIPS-8A and URIPS-8B		
USA/9001/B()F	35		Model No. IF-300		
USA/9009/B()F	18		Model No. FL 10-1		
USA/9010/B()F	40		Model No. NLI-1/2		
USA/9015/B()F	20		Model Nos. TN-8 and TN-8L		
USA/9016/B()F	12		Model No. TN-9		
USA/9023/B()F	8		Model No. NLI-10/24		
USA/9027/B(U)-85	16		Model Nos. 741-OP		
USA/9030/B()	9		Model Nos. MW-3000 and SENTINEL-8		
USA/9034/AF	12		Model No. TRIGA-I		
USA/9035/B(U)-85	17		Model No. 680-OP		
USA/9036/B(U)-85	10		Model No. C-1		
USA/9037/AF	12		Model No. TRIGA-II		
USA/9039/B(U)	10		Model No. 715		
USA/9056/B(U)	11		Model No. SPEC 2-T		
USA/9067/B()F	6		Model No. BCL-3		
USA/9068/B()F	6		Model No. BCL-2		
USA/9069/B()	11		Model No. MO-1		
USA/9070/B(U)	15		Model No. N-55		
USA/9071/B()	6		Model No. AP-101		
USA/9081/B()	13		Model No. CNS 1-13C		

Table 1. Existing Hanford Site (Onsite) PSSDs.

Document Number	Current Revision	Clearance Date	Title	Configuration Manager	Package Custodian
USA/9098/B()	9		Model Nos. CI-20WC-2 and CI-20WC-2A		
USA/9099/B(U)F-85	9		Model No. ATR		
USA/9102/B()	8		Model No. NPI-20WC-6		
USA/9107/B(U)	6		Model No. 771		
USA/9132/B(M)F	14		Model No. T-3		
USA/9148/B(U)	5		Model No. 770		
USA/9150/B(U)-85	6		Model No. PAT-2		
USA/9152/B()F	13		Model No. CNS 1-13C II		
USA/9157/B(U)-85	8		Model No. IR-100		
USA/9165/B(U)	4		Model No. 855		
USA/9168/B(U)	11		Model No. CNS 8-120B		
USA/9183/B()F	13		Model No. NAC-1		
USA/9184/B(U)	5		Model No. PAS-1		
USA/9185/B(U)-85	5		Model No. OP-100		
USA/9187/B(U)	4		Model No. 865		
USA/9196/AF-85	18		Model No. UX-30		
USA/9200/B(M)F	10		Model No. 125-B		
USA/9202/B(U)F	7		Transnuclear Model TN-BRP		
USA/9203/AF	12		Model No. DHTF		
USA/9204/B(U)-85	6		Model No. CNS 10-160B		
USA/9206/B(U)F	7		Model No. TN-REG		
USA/9212/B(M)F-85	0		RH-TRU 72-B		
USA/9215/B(U)	5		Model No. NPI-20WC-6 MkII		
USA/9216/B()F	7		Model No. CNS 1-13G		
USA/9217/AF	10		Model No. ANF-250		
USA/9218/B(U)F-85	12		Model No. TRUPACT-II		
USA/9225/B(U)F-85	28		Model No. NAC-LWT		
USA/9226/B(U)F-85	0		Model No. GA-4		
USA/9228/B(U)F-85	19		Model No. 2000		
USA/9233/B(U)	5		Model No. TN-RAM		
USA/9234/B(U)F	15		Nuclear Containers Model NCI-21PF-1		

Table 1. Existing Hanford Site (Onsite) PSSDs.

Document Number	Current Revision	Clearance Date	Title	Configuration Manager	Package Custodian
USA/9235/B(U)F	2		Model No. NAC-STC		
USA/9239/AF	10		Model Nos. MCC-3, MCC-4, and MCC-5		
USA/9246/AF	2		Model No. ST		
USA/9248/AF	16		Model Nos. SP-1, SP-2, and SP-3		
USA/9250/B(U)F-85	7		Model No. Babcock and Wilcox Model No. NNFD 5X22		
USA/9251/AF	10		Model No. BW-2901		
USA/9252/AF	4		Model No. 51032-2		
USA/9253/B(U)F	4		Model No. TN-FSV		
USA/9255/B(U)F-85	5		Model No. NUHOMS MP187 Multi-Purpose Cask		
USA/9258/B(U)-85	0		MDS Nordian Model No. F-294		
USA/9261/B(U)F-85	1		Model No. HI-STAR 100 System		
USA/9263/B(U)-85	3		Model No. SPEC-150		
USA/9269/B(U)-85	3		Model No. 650L		
USA/9272/AF-85	5		Model No. CE-B1		
USA/9274/AF	5		Model No. ABB-2901		
USA/9277/B(U)F	1		Model No. FSV-1 UNIT 3		
USA/9279/B(U)F-85	0		HalfPACT Waste Shipping Container		
USA/9280/AF-85	1		Model No. UBE-1		
USA/9281/AF-85	2		Model No. UBE-2		
USA/9282/B(U)-85	0		Model No. SPEC-300		
USA/9283/B(U)-85	0		Model No. OPL-660 and OP-660.		
USA/9284/B(U)F-85	1		ESP-30X Protective Shipping Package for 30-Inch UF6 Cylinders		
USA/9285/AF-85	1		Model No. SRP-1		
USA/9287/B(U)-85	0		Model No. Sterigenics Eagle		
USA/9288/AF-85	3		Model No. ECO-PAK OP Uranium Oxide Transport Unit Package		
USA/9289/B(U)F-85	1		Model No. WE-1		

Table 1. Existing Hanford Site (Onsite) PSSDs.

Document Number	Current Revision	Clearance Date	Title	Configuration Manager	Package Custodian
USA/9292/AF-85	2		Model No. PATRIOT		
USA/9293/B(U)-85	1		Model No. TN-68 Transport Package		
USA/9294/AF-85	0		Model No. NPC		
USA/9296/B(U)-85	1		Model No. 880 Series Packages		
USA/9511/B(U)	2		Model No. Beneficial Uses Shipping System Cask (BUSS) R-1		
DOE					
USA/5320-3/B()F (DOE)	16		PU Oxide and AM Oxide Shipping Cask		
USA/5467/AF-85 (DOE)	21		Model: Steel Banded Wooden Shipping Containers, Models G-4214, G-4255, G-4273 and G-4292		
USA/9099/B(U)F (DOE)	15		ATR Fuel Element Shipping Container		
USA/9184/B(U) (DOE)	2		PAS-1		
USA/9225/B(U)-85 (DOE)	1		Model No. NAC-LWT		
USA/9511/B(U) (DOE)	6		BUSS Model R-1		
USA/9516/B(U)F-85 (DOE)	6		Model No. Mound 1KW		
USA/9904/B(U)F-85 (DOE)	3		Model No. RTG Package		
USA/9932/B(U) (DOE)	8		Model No. UC-609		
USA/9965/B(U)F (DOE)	9		Model No. 9965		
USA/9975/B(M)F-85 (DOE)	5		Model N0. 9975		
DOT					
USA/0018/S	7		SR-CF-100 Series Neutron Source Manufactured by either the Savannah River Laboratory or the Oak Ridge National Laboratory (ORNL)		
USA/0036/S	6		NRD Model A-001		
USA/0043/S	9		Monsanto Research Corporation Model 2720		

Table 1. Existing Hanford Site (Onsite) PSSDs.

Document Number	Current Revision	Clearance Date	Title	Configuration Manager	Package Custodian
USA/0058/S	6		General Electric CF-100 Series		
USA/0061/B(U)	17		Theratron 78, T780, T780-C, T780-E T780IEC, Phoenix, Theratron 1000 and T1000E Radiotherapy Heads		
USA/0062/S	6		General Electric Standard Teletherapy Source		
USA/0065/S	7		Model SR CF-1000 Series Neutron Source Manufactured by either the Savannah River Laboratory or ORNL		
USA/0066/S	6		3M Model 4F6H		
USA/0071/S	5		3M Model 4D6L Capsules Manufactured Prior to August 3, 1989.		
USA/0073/S	7		General Electric Bulk CO-60 Container		
USA/0074/S	5		3M Model 4F6P		
USA/0077/S	6		3M Model 4F6S. "Use Only" for service life of capsule		
USA/0078/S	8		Gulf Nuclear Model CSV. NRC has restricted application to "Use Only." Gulf Nuclear no longer makes source capsule		
USA/0080/S	3		Monsanto Research Corporation Capsule Identified by MRC Drawing No. SK19502A0 (A Prototype of Model No. 24132)		
USA/0087/S	4		Dresser Atlas Model No. DA-5.		
USA/0088/S	5		Dresser Atlas Model No. DA-20		
USA/0095/S	8		Source Production and Equipment Co. Series B, G, R and T Model Sources		
USA/0112/S	5		Schlumberger Model NSR-GB		
USA/0113/S	8		Model Nos. NSR-F, NSR-D and NSR-R		
USA/0114/S	5		Gulf Nuclear, Inc. Model No. AMBE 71-1		
USA/0115/S	8		Gulf Nuclear, Inc. Model VL-1		

Table 1. Existing Hanford Site (Onsite) PSSDs.

Document Number	Current Revision	Clearance Date	Title	Configuration Manager	Package Custodian
USA/0116/S	4		Halliburton Services Model X-602-04-101 Neutron Source		
USA/0124/B(U)	12		MDS Nordion International Inc. F-245 Shipping Container, Serial Nos. 1 to 5 and 7 to 26		
USA/0125/B(U)	10		Nordion International Inc. Model No. F-327/F-247 Serial Nos. 1 to 10 Inclusive, 12 to 41 Inclusive		
USA/0126/B(U)-85	14		MDS Nordion Inc. F-327/F-251 and F-327/F-318 Transport Packages. F-327/F-251: all serial numbers 1 and above. F-327/F-318: Serial numbers 2-15, 17-27, 30-46, 48, 49, 51-61, 63-82, 84-94.		
USA/0135/S	7		Model Nos. NSR-M and NSR-L.		
USA/0137/S	4		3M 4D6P Source Capsules Manufactured Prior to August 3, 1989		
USA/0141/S	9		General Electric Company Neutron Source Model GEN-CF-1X or 2765-AAOO		
USA/0149/S	5		Gulf Nuclear Model AMBE 71-2A Source Capsules Manufactured Prior to March 8, 1988		
USA/0152/S	4		New England Nuclear/Du Pont Company Models NER-570 and NER-580		
USA/0154/S	7		Amersham Corporation Model Nos. 60001, 60004, 60006, 60017, 60018, 60020, 60021 and 68310		
USA/0158/S	4		E.I. Du Pont/Nen Model NER-479C		
USA/0159/S	4		E.I. Du Pont/Nen Model NER-478C		
USA/0161/S	1		New England Nuclear Model NER-550		
USA/0165/S	5		Amersham Corporation Model Nos. A424-2 through A424-19 (excluding A424-6 and A424-9) and A453-1 through A453-10 (excluding A453-3 and A453-4). They		

Table 1. Existing Hanford Site (Onsite) PSSDs.

Document Number	Current Revision	Clearance Date	Title	Configuration Manager	Package Custodian
			are composed of Tech/Ops Capsule Styles 60000, 60001, 60002, 60003, 60004, 60011, 60012, 60013		
USA/0166/S	8		Gamma Industries Model Nos. VD, VD(HP), NB, NBG, NB(HP), Single Encapsulation Universal Source, Double Encapsulation universal source, and single encapsulation with side manufactured prior to August 1, 1993		
USA/0169/B(U)	8		U.K. Design No. 0666S		
USA/0174/S	4		Gulf Nuclear, Inc. Model CS-2		
USA/0179/S	7		Amersham Series 900 Iridium Capsule.		
USA/0192/S	4		Isomedix Model ISO-1000 Manufactured prior to June 30, 1998		
USA/0208/B(U)F-85	7		Model No. JRC-80Y-20T		
USA/0214/B(U)	12		MDS Nordion Inc. Model F-168-X, Shipping Flask Nos. 22-X, 23-X, 24-X, 25-X, 26-X, and 41-X		
USA/0220/AF-85	11		BU-J		
USA/0221/S	6		IPL Line Source		
USA/0226/B(U)	7		U.K. Design No. 1933A		
USA/0227/B(U)	9		U.K. Design No. 1933B		
USA/0228/B(U)	6		U.K. Design No. 1934A		
USA/0242/S	3		Monsanto Research Corporation Model 24154-C		
USA/0245/S	7		AB ELEKTA Model 43047 and AB ELEKTA Model 43685 Source Capsule		
USA/0250/B(U)	10		U.K. Design No. 0924BP		
USA/0257/S	4		Amersham Corporation Model 849		
USA/0263/S	2		Monsanto Research Corp. Model No. 24195		
USA/0269/B(U)	10		U.K. Design No. 0666AY		
USA/0272/B(U)	6		U.K. Design No. 1935A		

Table 1. Existing Hanford Site (Onsite) PSSDs.

Document Number	Current Revision	Clearance Date	Title	Configuration Manager	Package Custodian
USA/0277/S	3		Neutron Products Model Nos. BN-450-14 and BN-450-14-A		
USA/0283/S	3		3M Model 3FIG Capsules Manufactured Prior To August 3, 1989		
USA/0292/S	5		Neutron Product, Inc. Model Nos. NPTT Series, NPI-XX-XXXXW, and NPI-XX-XXXX		
USA/0297/S	3		Industrial Nuclear Company Model A Source Capsule		
USA/0301/B(U)	5		U.K. Design No. 0924W		
USA/0302/B(U)	8		U.K. Design No. 0666AW		
USA/0304/B(U)	7		U.K. Design No. 0666T		
USA/0307/B(U)	7		U.K. Design No. 0666W		
USA/0316/B(U)	6		U.K. Design No. 0924BZ		
USA/0331/S	4		Gammatron Model AN-HP Source Capsule		
USA/0335/S	5		Amersham Corporation Model 875 Series Source Capsule Assemblies		
USA/0336/S	7		IPL Model XFB-3		
USA/0337/B(U)-85	10		Croft Associates Model 2773A		
USA/0348/B(U)	9		MDS Nordion Inc. Model F-231, Serial Numbers 7, 8, 9, 11 to 24 inclusively		
USA/0350/S	4		IPL Model 343		
USA/0351/S	4		IPL Model N-252		
USA/0352/S	4		IPL Model No. 295		
USA/0353/S	4		IPL Model No. 193		
USA/0354/S	4		IPL No. 274-1		
USA/0356/S	8		IPL Model Nos. 225, 416, A3000, A3015, A3023, A3024, A3030		
USA/0357/S	7		IPL Model Nos. A3214 and A3203 Source Capsules		

Table 1. Existing Hanford Site (Onsite) PSSDs.

Document Number	Current Revision	Clearance Date	Title	Configuration Manager	Package Custodian
USA/0361/B(U)F-85	4		PAT-1, Plutonium Air Transportable Package		
USA/0363/S	4		Amercham Corporation Capsule Code X38/4		
USA/0366/S	4		Gammatron Model GT-GHP		
USA/0367/S	5		Frontier Technology Corporation Model 10 Series and Model 100 Series		
USA/0376/S	3		Specification SS-2050		
USA/0377/S	4		TECH/OPS RPD, Inc. or Amersham Corporation Models 60011, 60012, 60013		
USA/0381/B(U)F-85	5		Transport Container No. GNS 11		
USA/0382/B(U)-85	8		Croft Model No. 2835A		
USA/0383/S	2		Coratomic Type X Source for Cardiac Pacemaker		
USA/0392/S	5		Amersham Corporation Series 875 Capsule		
USA/0393/S	2		CIS-US, Inc. Model 791		
USA/0394/S	2		Amersham Corporation Model No. 922		
USA/0401/B/(U)F-85	5		Model No. JMS-87Y-18.5T		
USA/0404/B(U)-85	5		Gammacell 40 MK2 Irradiator		
USA/0406/AF-85	9		21PF-1 Overpacks as Specified in Japanese Certificate of Approval of Package Design No. J/27/AF-85, Rev. 2 and in Japanese Certificate of Approval for Packaging No. J/27/AF-85, Rev. 2		
USA/0407/B(U)	5		U.K. Design No. 3100A		
USA/0408/B(U)-85	6		U.K. Design No. 3300A		
USA/0411/AF	7		Model Nos. 5A, 5B, 8A, 12A, 12B, 30A, 30B, 48A, 48F, 48G 48H, 48X, 48HX and 48Y		
USA/0412/AF-85	8		Model No. BU-D		
USA/0419/S	2		3M Model 4P6E		
USA/0420/S	2		3M Model 4P6M		

Table 1. Existing Hanford Site (Onsite) PSSDs.

Document Number	Current Revision	Clearance Date	Title	Configuration Manager	Package Custodian
USA/0427/S	3		CIS-US, Inc. Models 772 and 774 Source Capsules		
USA/0442/AF-85	11		NT-IX		
USA/0444/B(U)	5		MDS Nordion Inc. Model F-271 Transport Package, Serial Nos. 1 to 10		
USA/0452/B(U)F-85	7		JRF-90Y-950K		
USA/0459/B(U)-85	4		Model No. Theratronics International F147 (85) Transfer Case, Serial Numbers 61 and Higher		
USA/0461/B(U)-85	5		MDS Nordion Inc., F-168 (1985) Transport Package, Serial Number 53 to 76 inclusive; 83 and up		
USA/0462/S	3		IPL Models 3021 and 3027		
USA/0463/S	1		J. L. Sheperd & Associates Model 7810-109-BP		
USA/0464/S	1		J.L. Shepherd & Associates Model 5810-190		
USA/0468/B(U)-85	3		MDS Nordion Inc. Model F-168-X (1985), Shipping Flask Nos. 77-X, 78-X, 79-X, 80-X, 81-X, and 82-X		
USA/0469/B(U)-85	4		MDS Nordion Inc. Gammacell 1000 and 3000, (Shipping Models) Serial Numbers 42 and up		
USA/0474/B(U)-85	1		JAERI Model No. TPL-92Y-45K		
USA/0475/B(U)	2		Nordion International, Inc. Gammacell 1000 Irradiator (Shipping Model in the 20WC-5 Overpack) and Nordion International, Inc. Gammacell 3000 Irradiator (Shipping Model in the 20WC-5 Overpack)		
USA/0477/B(U)-85	3		Nordion Gammacell 1000 and 3000, (Shipping Models in the 20WC-5 Overpack) Serial Nos. 42 and up		
USA/0480/AF	2		AECL Model: Maple 4 Enriched Fuel Bundle Shipping Package, Serial Nos. 1 to 7		

Table 1. Existing Hanford Site (Onsite) PSSDs.

Document Number	Current Revision	Clearance Date	Title	Configuration Manager	Package Custodian
USA/0485/B(U)F	1		AECL Model 4H Enriched Fuel Bundle Shipping Package, Serial Nos. 1 through 8		
USA/0490/AF-85	4		J/37/AF-85, Rev.3		
USA/0494/S	1		OmniTron International, Inc. Source Wire Model Nos. SL-777 and SL-777V		
USA/0495/AF-85	2		RAJ-II		
USA/0497/S	1		Amersham Model X.444		
USA/0498/S	1		IPL Model HEG-1		
USA/0500/S	1		Amersham Model X.1065		
USA/0501/S	1		Amersham Model X.44		
USA/0502/S	2		Amersham Model X.54 and X.540		
USA/0508/S	1		IPL Model A3906		
USA/0509/B(U)-85	3		MDS Nordion Models F-127, F-127-X and RAI/F-127 Transport Packages, Serial Nos. 59 and Up		
USA/0513/S	1		Amersham Model X.560		
USA/0514/S	0		AN-HP		
USA/0515/S	1		IPL Model GFS-3 Source Capsule		
USA/0516/S	1		IPL Model Nos. A3224-01, A3224-02, A3224-03, A3224-11, A3224-12, A3224-13 and A3807 Source Capsules		
USA/0517/S	1		IPL Model Nos. A3224-04, A3224-14, A3901-1 and A3901-2 Source Capsules		
USA/0518/S	1		IPL Model A3908 Source Capsule		
USA/0523/S	0		JL Sheppard 78100484-1		
USA/0526/S	0		JL Sheppard 7810-0109-R		
USA/0530/S	0		JLS&A 8810-AMBE-154		
USA/0531/S	0		Model DSK 2384		
USA/0532/B(U)-85	2		Ganuk Model GA-01 Transport Container		

Table 1. Existing Hanford Site (Onsite) PSSDs.

Document Number	Current Revision	Clearance Date	Title	Configuration Manager	Package Custodian
USA/0539/S	0		AMBE MJ-1L and AMBE MJ-1S		
USA/0540/S	0		J. L. Shepherd Model No. 7810-9		
USA/0541/S	0		J.L. Sheperd Model 7810-8		
USA/0543/S	0		Sperry Sun Source Plug Assembly No. 009100		
USA/0544/S	0		CIS-US Model 789		
USA/0545/B(U)	1		U.K. Design No. 3605C		
USA/0551/B(U)F-85	3		GNS-16 Spent Fuel Cask		
USA/0552/B(U)F-85	0		AEA Technology Neutron Source Container		
USA/0553/B(U)-85	0		AECL-CRL Irradiated Material Transport Package		
USA/0554/B(U)-85	3		Theratronics Int. LTD. Radiotherapy Heads: Theratron 780C (T780C), S/N 274 & up; Theratron 780E (T780E), S/N 601 & UP; Theratron 1000E (T1000E), S/N 44 & up; Phoenix, S/N 145 & up; Theratron Elite 80, S/N 1 & up; & Theratron Elite 100, S/N 1 & up		
USA/0556/B(U)-85	2		KATY		
USA/0558/B(U)F-85	0		Model No. JMS-87Y-18.5T		
USA/0559/S	0		Model Nos. 6810G - Types I, II, III, IV, V and VI		
USA/0560/S	0		JLS & A 7810-150		
USA/0562/B(U)-85	5		ZA/CNS1005/B(U)-85		
USA/0563/AF-85	2		BNFL Model 3516 Uranium Transport Package		
USA/0566/S	0		Source Production and Equipment Co. Model Nos. G and T (Spec. Drawing 19B002, Rev. 1)		
USA/0567/AF-85	1		Model Nos. DOT Specification 21PF-1 and 21PF-1B Protective Overpacks		
USA/0569/B(M)-85	0		Model No. NR-10		
USA/0570/S	1		CNS0010-192 Brachytherapy Source		

Table 1. Existing Hanford Site (Onsite) PSSDs.

Document Number	Current Revision	Clearance Date	Title	Configuration Manager	Package Custodian
USA/0571/S	0		Varian Model VS-2000		
USA/0573/B(U)F-85	0		TN 7-2 Irradiated Fuel Assembly Cask		
USA/0575/H(U)-96	0		2000 Med Package		
USA/0577/B(U)F-85	0		COG-OP-30B Overpack		
USA/0578/B(U)-85	0		MDS Nordion Inc. Model F-231 (1985) and Model F-231 MK 2 Transport Packages, Serial Nos. 11 and higher		
USA/0587/B(U)-85	0		Gammacell 40 MK3 Irradiator, Serial No. 11 and Subsequent		
USA/0590/B (U)-85	0		U.K. Design No. 3605A		
USA/0592/B(U)-85	0		U.K. Design No. 3605B		
USA/0593/B(U)-85	0		U.K. Design No. 3605C		
USA/0594/B(U)-85	0		U.K. Design No. 3605M		
USA/0597/S	0		Model X.2050 Series source capsules		
USA/4986/AF	28		Model No. RA-3		
USA/5467/AF-85	1		SBWSC (Steel Banded Wooden Shipping Container)		
USA/5796/B(U)	12		Model Nos. 181375 and 181361		
USA/5979/B()	7		Alpha Omega Services, Inc. Model No. 5979		
USA/6050/B(U)	12		MDS Nordion Inc. Model F-144 Shipping Container, Serial Nos. 1, 5, and 9, and Model F-144-AC, Serial No. 3		
USA/6078/AF	1		Model Nos. 927A1 and 927C1		
USA/6125/B(U)	12		Nordion Gammacell 220 Irradiator, Serial Nos. 1 to 256 Inclusive		
USA/6162/B(U)	16		Nordion International Inc. F-127 J-Rod Shipping Container, Serial Nos. 50, 52, and 54		
USA/6214/B(U)	16		Nordion International Inc. F-112 Shipping Containers, Serial Nos. 10, 11, 20, 29,31, 33, 34, 36, 37, 40, 42, 43, 61,62, 64, 65, 68. Nordion International Inc. F-113,		

Table 1. Existing Hanford Site (Onsite) PSSDs.

Document Number	Current Revision	Clearance Date	Title	Configuration Manager	Package Custodian
			Shipping Containers, Serial Nos. 1, 4-10, 13, 15-18, 20, 22, 59, 60, 61, 63-66, 68, 69, 72-74, 76-78, 105-110		
USA/6217/B(U)	15		MDS Nordion Inc. Model F-143 Transfer Case, Serial Nos. 20, 50, 53, 54, 59, 62, and 64; or Model F-158 Transfer Case, Serial Nos. 3, 4, 5, 6, 8, 9, 10 and 14		
USA/6306/B(U)	14		Nordion International Inc. F-168 Shipping Flask Nos. 20, 21, 28, 31, 32, 33, 36, 38, 39, 42 to 52		
USA/6355/B(U)	12		Theratron International F-147 Transfer Case Serial Numbers 1 to 60		
USA/6400/B()F	1		Model No. 6400 (Super Tiger)		
USA/6581/AF-85	25		Siemens Power Corporation Model No. 51032-1		
USA/6613/B(U)	8		Amersham Model No. 702		
USA/6717/B(U)	13		Amersham Corporation Model 6717-B		
USA/6788/B(U)F-85	5		Croft Associates Model 2799E		
USA/9019/AF	26		General Electric Company Model BU-7		
USA/9027/B(U)-85	14		Model No. 741-OP		
USA/9032/B(U)-85	6		Amersham Model 650		
USA/9034/AF-85	12		Model No. TRIGA-1		
USA/9035/B(U)-85	10		Model No. 680-OP		
USA/9036/B(U)-85	12		Model Spec. C-1		
USA/9037/AF-85	12		Model No. TRIGA-2		
USA/9039/B(U)	11		AEA Technology Model No. 715		
USA/9056/B(U)-85	11		Source Production & Equipment Company, Inc. Model Spec 2-T		
USA/9107/B(U)-85	6		Model No. 771 Shipping Container		
USA/9148/B(U)	4		Model 770 Packaging		
USA/9150/B(U)-85	5		Model No. PAT-2		
USA/9157/B(U)	5		Model No. IR-100		

Table 1. Existing Hanford Site (Onsite) PSSDs.

Document Number	Current Revision	Clearance Date	Title	Configuration Manager	Package Custodian
USA/9165/B(U)	5		Model No. 855		
USA/9185/B(U)-85	5		Model No. OP-100		
USA/9187/B(U)	5		Amersham Model No. 865		
USA/9196/AF-85	21		Model No. UX-30		
USA/9204/B(U)-85	1		CNS 10-160B		
USA/9215/B(U)	5		Model No. NPI-20WC-6 MkII		
USA/9217/AF	11		American Nuclear Fuels Corporation Model ANF-250		
USA/9225/B(U)F-85	23		Model No. NAC-LWT		
USA/9228/B(U)F-85	11		General Electric Model 2000		
USA/9234/B(U)F	11		Nuclear Containers Model NCI-21PF-1		
USA/9239/AF	12		Model Nos. MCC-3, MCC-4, and MCC-5 Packagings		
USA/9245/B(U)-85	5		Model No. 420		
USA/9248/AF	16		Framatome ANP Richland Division, Inc. Model Nos. SP-1, SP-2 and SP-3		
USA/9250/B(U)F-85	4		BWX Technologies Inc, Model No. NNFD 5X22		
USA/9258/B(U)-85	0		MDS Nordian Model No. F-294		
USA/9263/B(U)-85	5		Model No. Spec-150		
USA/9269/B(U)-85	3		AEA Technology/QSA Inc. Model 650L		
USA/9272/AF-85	0		CE-B1		
USA/9274/AF	3		AAB-2901		
USA/9282/B(U)-85	0		SPEC-300		
USA/9283/B(U)-85	0		AEA Technology Model Nos. OPL-660 and OP-660		
USA/9284/B(U)F-85	0		ESP-30X Protective Shipping Package		
USA/9285/AF-85	1		SRP-1		
USA/9288/AF-85	2		ECO-PAK OP Uranium Oxide Transport Unit (OP-TU)		
USA/9292/AF-85	1		PATRIOT		
USA/9294/AF-85	1		Global Nuclear Fuel Model No. NPC		

Table 1. Existing Hanford Site (Onsite) PSSDs.

Document Number	Current Revision	Clearance Date	Title	Configuration Manager	Package Custodian
USA/9296/B(U)-85	0		AEA Technology 880 Series Packages		
USA/9516/B(U)F-85	2		Mound 1KW		
Part Two -- Equivalent Packaging					
BHI-01476	0	05/08/01	PAS-1 Cask Onsite SARP for 105F/H Fuel Storage Basin Spent Nuclear Fuel	BHI	BHI
HNF-4763	0-A	03/07/01	Steel Waste Package	FH T&P	FH T&P
HNF-5919	0-B	04/20/01	ATG Nuclear Services 382-B Cask	FH T&P	ATG Nuclear Services
HNF-SD-TP-SARP-001	2-C	09/25/00	Sample Pig Transport System	FH T&P	CHG
HNF-SD-TP-SARP-017	2-C	05/10/01	Multicanister Overpack Cask	FH T&P	FH T&P
HNF-SD-TP-SARP-020	0	03/26/97	T-3 Cask	FH T&P	FH T&P
HNF-SD-TP-SARP-021	0-C	12/03/97	CASTOR GSF Cask	FH T&P	FH T&P
HNF-SD-TP-SARP-022	0-B	05/14/98	GNS 12 Packaging	FH T&P	FH T&P
HNF-SD-TP-SARP-023	0-D	09/29/99	Doorstop Sample Carrier System	FH T&P	CHG
SD-RE-SAP-054	3-B	07/01/97	21-PF-1 Overpack/EBR-II Spent Fuel Cask	FH T&P	FH T&P
WHC-SD-TP-SARP-008	0-B	10/01/96	NRF TRIGA Packaging	FH T&P	FH T&P
WHC-SD-TP-SARP-010	0	01/30/96	Interim Storage Cask	FH T&P	FH T&P
WHC-SD-TP-SEP-058	0-B	12/28/99	SERF Cask	FH T&P	FH T&P
WHC-SD-WM-SARP-001	1-G	08/25/00	Lead-Lined Drum 21 PF-1 Packaging System	FH T&P	All Users
Part Three -- Nonequivalent Packaging					
HNF-2209	0-A	06/05/01	Steel Drum	FH T&P	All Users
HNF-2760	0-D	09/16/99	Ion Exchange Modules	FH T&P	All Users
HNF-3341	0	09/29/98	Decontaminated Equipment Self Container	FH T&P	All Users
HNF-3786	0	09/23/99	ATG Nuclear Services 382-B Cask	FH T&P	ATG Nuclear Services
HNF-6550	0	07/14/00	T Plant Canyon Items	FH T&P	FH T&P
HNF-6807	1	06/15/01	Pipe Overpack Container	FH T&P	FH T&P
HNF-7538	0-A	05/31/01	Finished Fuel Assembly	FH T&P	FH T&P
HNF-SD-TP-SARP-007	1	09/29/00	Flexible Material Packaging	FH T&P	All Users
HNF-SD-TP-SARP-013	0-A	11/07/00	Long-Length Contaminated	FH T&P	CHG

Table 1. Existing Hanford Site (Onsite) PSSDs.

Document Number	Current Revision	Clearance Date	Title	Configuration Manager	Package Custodian
			Equipment Transport System		
HNF-SD-TP-SARP-026	0	09/25/97	Transuranic Performance Demonstration Program Sample Packaging	FH T&P	FH T&P
HNF-SD-TP-SARP-027	0-A	03/19/98	WESF Ion Exchange Module	FH T&P	All users
WHC-SD-TP-SARP-002	1-F	02/03/00	Onsite Transfer Cask	FH T&P	CHG
WHC-SD-TP-SARP-006	0-A	10/06/95	85-Gallon Retrieval Drum	FH T&P	All Users
WHC-SD-TP-SARP-009	0-A	07/22/98	LR56/H Cask System	FH T&P	All Users
WHC-SD-TP-SARP-015	0-D	07/06/98	SARP (Onsite) Double-Shell Tank Car	FH T&P	FH T&P
WHC-SD-TP-SARP-018	0-E	10/19/99	Type B Quantities of Radioactive Material in Type A Boxes	FH T&P	All Users
WHC-SD-TP-SEP-022	1	10/10/95	Neutron source Holder Assembly	FH T&P	FH T&P
WHC-SD-TP-SEP-048	0-B	03/10/00	Transport of LSA-II Liquids in MC-312 Cargo Tanks	FH T&P	FH T&P
Part Four -- Exemptions					

AECL = Atomic Energy of Canada, Ltd., Ontario

ATG = Applied Technology Group

BHI = Bechtel Hanford Inc.,

BNFL = British Nuclear Fuels

BUSS = Beneficial Uses Shipping system Cask

CHG = CH2M Hill Group

CNS = Chem-Nuclear System

DOE-RL = U.S. Department of Energy, Richland Operations Office

FH = Fluor Hanford

GNS = Gesellschaft für Nuklear-Service mbH

HFIR = High Flux Isotope Reactor

HNF = Hanford Nuclear Facility (document identifier)

IPL = Isotope Products Laboratories

LSA = low specific activity

NRF = Neutron Radiography Facility

ORNL = Oak Ridge National Laboratory

SARP = Safety analysis report for packaging

SERF = Special Environmental Radiometallurgy Facility

T&P = transportation and packaging

TRIGA = Test Reactor and Isotope Production General Atomics

WESF = Waste Encapsulation and Storage Facility

WHC = Westinghouse Hanford Company (document identifier)

Table 2. Hanford Site (Onsite) PSSDs.

Document Number	Current Revision	Clearance Date	Title	Configuration Manager	Package Custodian
Part One -- DOT Compliant Packaging					
Onsite use of a package identified by DOT, NRC, and DOE for use in commerce and not identified in Table 1 requires the approval of the RL Manager. DOT approved packagings are identified in 49 CFR 173, Subpart I. A listing of NRC, DOE, and DOT approval certificates for Type B and Fissile packagings on the RAMPAC web site (RAMPAC.com). Packagings on this listing are approved for use in commerce.					
No packages listed in this section					
Part Two -- Equivalent Packaging					
No packages listed in this section					
Part Three -- Nonequivalent Packaging					
No packages listed in this section					
Part Four -- Exemptions					
No packages listed in this section					

DOE = U.S. Department of Energy
 DOT = U.S. Department of Transportation
 NRC = U.S. Nuclear Regulatory Commission
 RL = U.S. Department of Energy, Richland Operations Office.

APPENDIX B

**JUSTIFICATION AND BASIS FOR EQUIVALENCY TO DOT REGULATIONS FOR
TYPE B AND FISSILE PACKAGES TRANSPORTED ON THE HANFORD SITE**

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FIGURE

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TABLE

Table 1. Hanford Onsite Package System Acceptable Performance Criteria.....	App B-15

LIST OF TERMS

AC	accident condition
ASME	American Society of Mechanical Engineers
BPVC	Boiler and Pressure Vessel Code
CFR	<i>Code of Federal Regulations</i>
CSB	Canister Storage Building
CWC	Central Waste Complex
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
HAC	hypothetical accident condition
FEA	Finite Element Analysis
IAEA	International Atomic Energy Agency
MCNP	Monte Carlo (computer code for) neutron photon (transport)
NC	normal condition
NCT	normal conditions of transport
NRC	U.S. Nuclear Regulatory Commission
NUREG	U.S. Nuclear Regulatory Commission Regulation
RL	Richland Operations Office
SED	strain energy density
TSL	triaxial strain limit

APPENDIX B

JUSTIFICATION AND BASIS FOR EQUIVALENCY TO DOT REGULATIONS FOR TYPE B AND FISSILE PACKAGES TRANSPORTED ON THE HANFORD SITE

1.0 INTRODUCTION AND OBJECTIVE

The objective of this Appendix is to demonstrate equivalency to regulations specified in 10 *Code of Federal Regulations* (CFR) 71, "Packaging and Transportation of Radioactive Material," of the acceptable performance criteria for design, construction, and evaluation of Type B packages for use on the Hanford Site. The proposed Hanford Site acceptable performance criteria are based on U.S. Nuclear Regulatory Commission (NRC) staff guidelines established in Regulatory Guide 7.6, *Design Criteria for Structural Analysis of Shipping Cask Containment Vessels*, to demonstrate acceptable performance of spent fuel and Category I package design under performance tests specified in 10 CFR 71.71, "Normal conditions of transport," and 10 CFR 71.73, "Hypothetical accident conditions." These acceptable performance criteria, adopted by the U.S. Department of Energy (DOE) – Richland Operations Office (RL), and Office of River Protection, provide a degree of safety equivalent to that provided by 10 CFR 71 performance tests in ensuring public and worker safety under normal and accident transport conditions on the Hanford Site. In practice, equivalent safety is to be demonstrated by subjecting the package design to 10 CFR 71 performance tests as amended for Hanford Site conditions and evaluating the package design against criteria equivalent to NRC acceptable performance criteria.

To alleviate confusion, the terminology and acronym for normal conditions of transport (NCT) and hypothetical accident conditions (HAC) specified in the regulations are changed for onsite performance test conditions. The terminology and acronym for Hanford Site normal and accident conditions are normal conditions (NC) and accident conditions (AC), respectively.

2.0 BACKGROUND

Performance criteria stipulated in 10 CFR 71 for the specified performance tests are that containment, subcriticality, and shielding shall be maintained by the package. The specified performance tests can be performed analytically or by physical testing. However, NRC staff recognizes the cost of full-scale testing may be prohibitive and scale model-testing technology is not fully developed. Consequently, NRC has developed acceptable performance criteria for design and evaluation of Type B packages to demonstrate conformance with performance requirements specified in 10 CFR 71. The NRC, using a graded approach, has established three categories of acceptable package performance based on the content of the package as described in NUREG-1609, *Standard Review Plan for Transportation Packages for Radioactive Material*.

These categories are based on Sections III and VIII of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPVC). NUREG-1609 indicates that spent fuel and Category I packages be designed and constructed to the requirements of ASME BPVC, Section III, Subsection NB. Subsection NB is used for critical safety components of a nuclear reactor and requires the highest level of quality technologically available. NUREG-1609 also states that Category II and III packages be designed and constructed to the requirements of ASME BPVC, Section III, Subsection ND and ASME BPVC, Section VIII, Division 1.

To demonstrate acceptable spent fuel and Category I package performance under performance test conditions defined in the regulations, the NRC has established acceptance criteria outlined in Regulatory Guide 7.6. This is used in conjunction with Regulatory Guide 7.8, *Load Combinations for the Structural Analysis of Shipping Casks*, and NUREG/CR-6007, *Stress Analysis of Closure Bolts for Shipping Casks* to demonstrate acceptable NCT and HAC performance of the package, used in general commerce. Regulatory Guide 7.6 is based on Section III, Subsection NB of ASME BPVC and establishes standards and criteria for evaluating structural integrity of Type B packages. The basic assumption of ASME BPVC and Regulatory Guide 7.6 is the components remain linearly elastic and the principle of superposition can be applied to determine combined loads. The NRC recognizes that Section III, Subsection NB contains requirements for nuclear power plants. Subsequently, some requirements are not applicable for package design and evaluation. However, the NRC has adopted portions of ASME BPVC that use a "design-by-analysis" approach to form performance acceptance criteria for the containment boundary of spent fuel and Category I packages. By Regulatory Guide 7.6, evaluation criteria for NCT are defined as criteria for Service Level A Limits (NCs) and HAC are defined as the criteria for Service Level D Limits (faulted conditions). In practice, the NRC staff recognizes that to satisfy these requirements, the structural behavior of critical components on the containment boundary must remain elastic under all loading conditions.

In development of Hanford Site analytical acceptable performance criteria, three key factors are considered:

1. For components on the containment boundary to remain elastic under ACs is extremely conservative. This requirement ignores the energy absorption capability of the material of construction from localized strains without breach of containment or loss of dimensional stability.
2. Development of the regulations by the U.S. Department of Transportation (DOT) and NRC for safe transport in commerce is fundamentally based on maintaining containment and requiring that radiological safety be engineered into the package. The basis is as follows:
 - Packages are shipped via a variety of carriers and conveyances. Therefore, reliance for safe transport must be placed in the package because commercial carriers do not always have adequately trained operators and properly maintained equipment.
 - Operators for commercial carriers may not have training in handling radioactive materials.

- Local and state authorities may not have adequately trained accident emergency response units readily available.
 - The carriers, and local and state authorities determine transport routes.
 - All transport is on public roadways.
3. Transport of fissile and Type B packages on the Hanford Site are not in commerce when north of Wye Barricade or when shipped under road closure. Subsequently, the following provisions are in place to enhance onsite transportation safety:
- Access is controlled within the Wye Barricade. Subsequently, the majority of transport is not on public roads.
 - Trained emergency response teams are available and dedicated to the Hanford Site.
 - Roads are closed for transport of uncertified fissile or Type B packages from areas south of the Wye Barricade to the Wye Barricade.
 - Maintenance of equipment and training of personnel are required, monitored, and controlled by Hanford Site authorities.
 - These factors are also considered by the NRC in development of regulations (10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste") for commercial storage facilities, which are onsite operations.

3.0 JUSTIFICATION FOR PERFORMANCE

3.1 HANFORD SITE FREE DROP ONTO CONCRETE SURFACE

The free drop target surface specified in the regulations is a hard and essentially unyielding surface. In practice, this has been defined as a thick reinforced concrete pad with a thick (>5 cm [2-in.]) steel plate firmly fixed to the concrete, all of which must be set on compact soil. The International Atomic Energy Agency (IAEA) Safety Series No. 37 Guide states that the total weight of the concrete and steel plate for a proper free drop test must be at minimum 10 times the weight of package. This surface is specified in the regulations for conservatism and test reproducibility. In Finite Element Analysis (FEA) evaluations, the surface is defined as an infinitely rigid and unmovable surface. This FEA theoretical surface absorbs no energy from the impact and reflects all force waves.

On the Hanford Site, the free drop performance test target surface is established as the Central Waste Complex (CWC) storage pad. The pad is a 20 cm (8-in.) thick 21 MPa (3,000 psi)

concrete pad, reinforced with No. 7 rebar on 30 cm (12 in.) centers. This free drop surface is the most rigid surface encountered during transport of packages on the Hanford Site. In addition, this surface provides an additional margin of safety for the most common and numerous packages transported onsite, drums and boxes. The additional margin of safety is derived from the behavior of relatively small light packages (less than 4,534 kg [10,000 lb]) impacting a reinforced concrete surface. In such an impact, the small package behaves as though the impact was on a hard unyielding surface. The explanation for this is that small light packages, such as drums, have a hardness value far less than a reinforced concrete surface. Consequently, the drum tends to absorb a majority of the energy of impact in deformation and the concrete surface tends to act as a hard unyielding surface. The larger robust packages, such as spent fuel casks, have a hardness value much higher than the specified concrete surface. This results in a concrete surface absorbing much of the energy of impact. In essence, by specifying this surface, the most numerous and less robust Type B packages transported on the Hanford Site are subjected to the more conservative loading.

3.2 NC ACCEPTABLE PERFORMANCE CRITERIA

On the Hanford Site, packages are used only for onsite transport and storage. In lieu of testing, analytical evaluation and numerical simulations of the package are used to assess a majority of the packages to specified performance tests. Consequently, the NC performance acceptance criteria for all Type B packages will be based on ASME BPVC, Section III, Subsection NE in lieu of Subsection NB specified in Regulatory Guide 7.6. This is justified on the basis that Subsection NE is applicable to the containment vessel of a nuclear facility and has provisions tailored to the special features of bolted flange connections. Subsection NB is applicable to the reactor coolant boundary and has provisions tailored to pumps, vessels, valves, and piping. From a structural standpoint, there is no significant difference between the two code sections relative to determination of allowable stress intensities and combining of loads. Also, there is no major difference between the stress intensity criteria of these two subsections except that the stress intensity value in Subsection NB is S_m while it is defined as S_{mc} in Subsection NE. For materials related issues including welding, fabrication, and examination, Subsection NE is sufficient because the package is not subjected to the same high temperature, high pressure, and intense radiation as the reactor coolant boundary. The temperature, pressure, and radiation level of the package are closer to those of the secondary metal containment vessel for a nuclear reactor facility.

Also, NC performance acceptance criteria are based on ASME BPVC, Section III, Subsection NE Service Level A and C Limits. Combined stress intensities from temperature, pressure, and free drop are evaluated against requirements defined for Service Level C Limits. Stress intensities from all other load combinations are to be evaluated against requirements defined for Service Level A Limits. For structural evaluation purposes, maintenance of containment is defined as the ability of packaging systems to sustain the applied loading without exceeding ASME allowable stress intensity values.

The use of Service Level C Limits for free drop loads combined with other loads is justified on the basis that during onsite transport a free drop of a package would be considered at best an

occurrence and at worst an event. In either case, the situation would dictate an emergency response on the Hanford Site. For this type of scenario, the package would be required to maintain containment, shielding, and subcriticality. After such an occurrence or event, the package would be transported under emergency conditions to a facility for removal and repacking of the contents into a new package. This sequence of actions after a free drop is consistent with the definition for Service Level C as defined in ASME BPVC, Section III, Subsection NCA, General Requirements for Division 1 and 2. In Subsection NCA under subparagraph NCA-2142.2, Service Level C Limits permit large deformation in areas of structural discontinuity without loss of function, which may necessitate removal of the component from service for inspection or repair of damage. Application of this limit is defined in Subsection NE, subparagraph NE-3113.3 as applicable loads subject to Service Level A and B Limits in combination with loads due to natural phenomena for which safe shutdown is required.

3.3 BRITTLE FRACTURE ACCEPTABLE PERFORMANCE CRITERIA

For package containment boundaries and critical components that are constructed from ferritic steels, low temperature brittle fracture of the materials need to be addressed in a similar manner to certified Type B packages. To address this issue, the NRC has developed fracture toughness criteria in Regulatory Guide 7.11, *Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 Inches (0.1 m)*. This regulatory guide establishes material fracture toughness requirements for three categories based on curie content of the payload. The highest category, Category I, requires material fracture toughness testing to requirements specified in NUREG/CR-1815, *Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers up to Four Inches Thick*. In lieu of these requirements, low temperature material service requirements of ASME BPVC, Section VIII, Division 1 are applied for design and evaluation of Type B packages to demonstrate resistance to brittle fracture based on minimum metal temperature for safe handling and transport of packages. The justification is as follows:

- Methods and requirements provided in Section VIII, Division 1 are based on accumulated years of experience in design, material usage, construction, and operation of pressure vessels by members of ASME.
- Basis for Section III and Regulatory Guide 7.11 requirements are derived from the requirements first established in Section VIII, Division 1.
- Methods and requirements outlined in Section VIII, Division 1, can be readily applied by the engineer without the expense of fracture toughness testing the materials of construction. The number of laboratories qualified to perform fracture toughness tests is limited and testing is expensive.

For packages and components constructed from austenitic stainless steels, no fracture toughness related restrictions are placed on the material. This is justified on the basis that Regulatory Guide 7.11 states that austenitic stainless steel is not susceptible to brittle failure and containment vessels constructed of stainless steel are acceptable to the NRC without testing.

3.4 ACCEPTABLE PERFORMANCE CRITERIA FOR FREE DROP AND PUNCTURE AC

The NRC regulatory position for demonstration of compliance of spent fuel and Category I package design to HAC performance tests defined in 10 CFR 71 is for the package to meet Service Level D Limits specified in ASME BPVC, Section III, Subsection NB as amended by Regulatory Guide 7.6. The intent of Regulatory Guide 7.6 is to demonstrate containment by showing there are no large unconfined plastic strains and system failure due to loss of dimensional stability using linear elastic methods and criteria defined in ASME BPVC. However, accident free drop and puncture loads on Type B packages without impact absorption devices cause highly nonlinear material responses. In particular, contact stresses from impact imposed on packages (especially without impact limiters) when dropped, exceed the yield strength of the material. Evaluating the material response in an impact event assuming a linear elastic material response is inappropriate. Application of ASME criteria to the results of a nonlinear analysis is also inappropriate. Therefore, it is necessary to generate acceptance criteria that go beyond the assumptions of linear elasticity and quasi-static load control. Acceptance criteria used must be consistent with elastic-plastic analysis techniques that are applied to ductile materials (austenitic stainless steel) that are subjected to highly localized impact loadings. Puncture and free drop impacts of Type B packages are energy-controlled events. Subsequently, the use of energy-based parameters as acceptance criteria are required to accurately assess the structural integrity, containment performance, and dimensional stability of the package. At present, there are two acceptance criteria based on energy parameters that have been used at the Hanford Site to assess package performance under accident free drop and puncture loads. They are strain energy density (SED) and triaxial strain limit (TSL).

Both acceptance criteria are energy based. The SED criterion is based on material toughness and the TSL criterion is based on the ultimate elongation of material and principal stresses. The choice in using these criteria has been left to the structural evaluation engineer. In most cases the SED criterion is used for shells, closures lids, and closure bolts. The TSL criterion is used at the more critical locations of joints and at discontinuities. In the case where the package is fitted with impact absorption devices or for all other loading combination cases (without free drop or puncture loads), performance is assessed to the requirements of ASME BPVC, Subsection NE, Service Level D Limits.

3.4.1 Criterion 1

The first and simplest acceptance criteria parameter is SED. The SED (V) of a material is defined as:

$$V = \int \sigma d\epsilon$$

In other words, the integral of the material stress (σ) versus strain (ϵ) curve.

The maximum SED of the material is also commonly known as toughness and, at material failure, can be conservatively estimated from results of uniaxial tensile test data. A conservative estimate of SED at material failure is as follows:

$$U_t = \frac{(\sigma_y + \sigma_u)}{2} \epsilon_f$$

where: σ_y = yield strength of the material

σ_u = ultimate strength of the material

ϵ_f = elongation at failure.

All these parameters are available from engineering stress-strain data. Use of the SED also minimizes concerns over unavoidable material variations such as heat-to-heat variations in yield strength and strain hardening behavior. In addition, the SED on the material from impact loads can be easily determined from stress and strain data retrieved from the FEA numerical simulation of the impact.

To ensure against ductile failure of the material, the maximum SED on the package is limited to 70% of the SED to failure of the material or material toughness. This is in keeping with the intent of Regulatory Guide 7.6 to prevent large unconfined strain plasticity and system failure due to loss of dimensional stability at key locations. The 70% limit is based on Service Level D limits for material loading, specified in ASME BPVC, Section III.

3.4.2 Criterion 2

In the SED measure of performance, no distinction is made between compressive and tensile strains, and all strains are treated as tensile. Consequently, this measure of performance is very conservative because it does not account for the less severe effects of compressive strains (with respect to breaching of the material). A more time consuming, but more accurate measure of performance is comparing the cumulative principal strains or equivalent plastic strains against a limiting value that could initiate local fracture. These cumulative strains generated by FEA numerical simulations of free drop and puncture events are the difference between the largest and smallest principal logarithmic strains at a specific element. This is, by definition, the strain counterpart of the ASME stress intensity. The equivalent plastic strain is the plastic strains combined in a von Mises manner.

Criteria demonstrating acceptable cumulative strain levels to prevent material rupture are based on establishing limits on the maximum principal cumulative strain. Strain limits are developed based on the Manjoine (1983), *Transaction of the ASME, Journal of Pressure Vessel Technology* observation that tensile elongation measured in a tension test can be adjusted to predict multi-axial failure strain by the following relation:

$$\frac{\epsilon}{\epsilon_t} = 2^{(1-TF)} \quad \text{for } TF \geq 0 \quad \text{and} \quad \frac{\epsilon}{\epsilon_t} = 2 \quad \text{for } TF \leq 0$$

where: ϵ = the effective strain at failure
 e_t = the tensile elongation from tension tests, and
 TF = the triaxiality factor.

The triaxiality factor is defined by the principal stresses ($\sigma_1, \sigma_2, \sigma_3$) as:

$$TF = \frac{\sqrt{2}(\sigma_1 + \sigma_2 + \sigma_3)}{\sqrt{[(\sigma_1 - \sigma_2)^2 + (\sigma_2 - \sigma_3)^2 + (\sigma_3 - \sigma_1)^2]}}$$

Based on the ductility versus triaxiality factor curve presented in Manjoine (1983), the following conservative failure strain limits can be established for any ductile material. For a triaxiality factor of 1.0 (uniaxial tension), the cumulative strain limit is the reported ultimate strain of the material at tensile rupture. For a triaxiality factor of 2.0 (maximum value for biaxial tension at the surface), the cumulative strain limit is half the reported ultimate strain of the material at tensile rupture. For negative values of triaxiality factor (significant compressive stress), the cumulative strain limit is double the reported ultimate strain of the material at tensile rupture. This criterion was developed and used by the Canister Storage Building (CSB) Expert Review Panel for evaluating multiccanister overpack free drops in the CSB.

3.5 FIRE ACCIDENT ACCEPTABLE PERFORMANCE CRITERIA

In the case of HAC fire, performance test requirements specified in 10 CFR 71.73 stipulate the fire temperature (800 °C [1,475 °F]), duration (30 minutes), emissivity (0.9), and package absorptivity (0.8). It also stipulates that no artificial cooling be applied and the package must cool naturally. In the case of package absorptivity, the NRC staff position is that testing real emissivities of the package may be used, but must be accounted for by increasing the fire temperature. Also, the sequence of events leading up to the fire is specified. As with all other performance tests, Regulatory Guide 7.6 is the acceptable performance criteria along with maintenance of shielding. For seal integrity, the NRC staff has traditionally used the manufacturers' recommended continuous operating temperature as the limiting temperature of the seal material.

For onsite transportation, it is proposed that the fire AC parameters for evaluation be as follows:

- Fire temperature of 800 °C (1,475 °F)
- Duration of 30 minutes
- Artificial water cooling is applied to the package immediately after the 30-minute fire
- Effect of solar radiation can be ignored for the entire event
- Emissivity of the fire is 0.9

- Absorptivity of the package is the published absorptivity of the outer surface material at temperature
- Sequence of events leading to the fire is the same as specified in the regulations.

Hanford Site parameters that are not identical to the regulations are justified on the following basis:

- Because there are dedicated Fire Fighting Units trained in fighting radioactive material fires on the Hanford Site, cooling after a 30-minute fire duration is realistic.
- Lower absorptivities of the outer shell of the packaging are justified on the basis that measurements of material emissivities at temperature are much lower than the 0.9 absorptivity specified in the regulations; as an example, the absorptivity of 316 stainless steel is 0.6 at 538 °C (1,000 °F).

Under fire conditions, the limiting value to demonstrate containment performance of the seal is based on the manufacturers' stated high temperature operation for a duration of 1 hour. Structural integrity and safe performance of the package is demonstrated by evaluating thermal loading on the package to the requirements of ASME BPVC Section III, Subsection NE for Service Level D Limits.

3.6 DESIGN AND CONSTRUCTION STANDARD FOR PACKAGES

The NRC has established standards and criteria for design and construction of packages in Regulatory Guides, NUREG/CR-3854, *Fabrication Criteria for Shipping Containers*, and NUREG/CR-3019, *Recommended Welding Criteria for Use in the Fabrication of Shipping Containers for Radioactive Materials*. These standards and criteria are based on the graded-approach application of Section III of ASME BPVC. As an example, spent fuel and Category I package containment vessels are to be designed and constructed to the requirements of ASME BPVC, Section III, Subsection NB. Category II package containment vessels are to be designed and constructed to the requirements of ASME BPVC, Section III, Subsection ND, and Category III package containment vessels to ASME BPVC, Section VIII, Division 1. Package design and construction quality assurance requirements are defined in 10 CFR 71, Subpart H. The intent of the NRC in establishing these various standards, criteria, and requirements is to ensure quality of the package by imposing controls on construction processes of packages and components.

For the design and construction of packages for transport and storage on the Hanford Site, ASME BPVC, Section VIII, Division 1 or 2 is to be used for the containment vessel of the package. The containment vessel is required to be constructed by an ASME BPVC, Section VIII, certified shop and in some cases the design engineer may require application of a U or U-2 Stamp. In all cases, pressure and leak tests are required. The bases for this are as follows:

- Quality control of basic vessel construction processes such as design, material selection, material handling, inspection, welding, assembly, and heat treatment are fundamentally the same as ASME BPVC, Section III, except additional restrictions are applied depending on the subsection (Subsection NB is the most restrictive).
- Vessels designed and constructed to the requirements of ASME BPVC, Section VIII, Division 1 have an inherent margin of 4. Vessels designed and constructed to the requirements of ASME BPVC, Section III, Subsection NB and Section VIII, Division 2 have an inherent margin of 3.
- Original basis of Section III Subsections are ASME BPVC, Section VIII, Divisions 1 and 2.
- Code philosophies in both sections are the same.
- Weld procedures and welders are qualified to ASME BPVC, Section IX under both Sections III and VIII.
- Nondestructive examination of materials and components are performed to the requirements of ASME BPVC, Section V under Section III and Section VIII.
- Both sections require the same known materials, proven design practices, known fabrication practices for forming, bending, welding and postweld heat treatment, proven nondestructive examination procedures, and pressure testing.
- Containment vessels, when properly designed to the requirements of ASME BPVC, Section VIII, Division 1 or 2 for internal pressure loadings of 1.4 to 2.8 MPa (200 to 400 psig) with standard material sizes, can be demonstrated to meet the proposed acceptable performance criteria for NC and AC loadings. This includes free drops.
- Pressure, radiation, and temperature extremes for a package are far less than for the primary coolant system of a reactor.
- Each vessel must be pressure tested.
- ASME BPVC, Section VIII, Division 1 or 2 provides a known commercial standard from which the containment vessel can be constructed that meets fundamental design and construction requirements of regulatory guides and NUREGs.

As an example of such a package, a design of a typical small Type B shielded drum package is provided in Appendix F, Example Application of Package Structural Design and Evaluation Requirements, of this TSD. Total weight of the package is approximately 2,041 kg (4,500 lb) and gross weight of containment vessel is approximately 181 kg (400 lb). The containment vessel is designed for an internal pressure of 1.4 MPa (200 psig) at 204 °C (400 °F), with a maximum normal operating pressure of 0.8 MPa (100 psig) at 93 °C (200 °F). The vessel is designed in accordance with design rules stipulated in ASME BPVC, Section VIII, Division 1, using stock sizes of materials. The package is then evaluated for NC and AC structural integrity and containment maintenance using the proposed acceptable performance criteria.

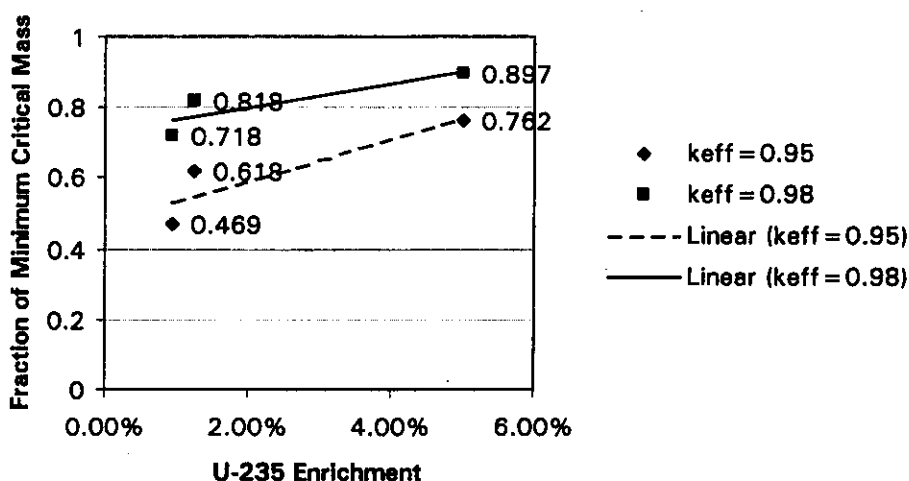
3.7 HANFORD SITE CRITICALITY LIMITS FOR PACKAGES

The NRC staff position is that transportation packages must have, in addition to the bias and uncertainties, a minimum administrative subcritical margin of 0.05 Δk (p. 2, NUREG/CR-5661, *Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packaging for Radioactive Material*); i.e., a maximum k_{eff} limit of 0.95 considering bias and uncertainties. However, if there is sufficient critical data to adequately determine calculational bias and uncertainty, the subcritical margin may be increased by some amount (NUREG/CR-5661, p. 18).

The NRC recommendations are based on transportation of commercial reactor fuel, which has a typical ^{235}U enrichment of 3 to 5 wt%. Because the minimum mass required for criticality increases greatly as the enrichment decreases to near the natural enrichment of 0.71 wt%, the fraction of the minimum critical mass for a given k_{eff} limit is much higher at 0.95 and 1.25 wt% ^{235}U than at the typical enrichment of commercial reactor fuel. Therefore, an equivalent degree of safety to the NRC recommendations can be achieved by using the same fraction of the minimum critical mass associated with commercial reactor fuel to calculate a k_{eff} limit for uranium systems enriched to 0.95 and 1.25 wt% ^{235}U .

The fraction of the minimum critical mass for a k_{eff} of 0.95 tends to decrease as the system size at critical increases. Because very low-enriched uranium systems have large critical masses, the fraction of critical mass for a k_{eff} of 0.95 is smaller than is typically true for higher enriched uranium systems, as shown in Figure 1. Thus, in terms of relative mass, there is a greater mass margin for low-enriched uranium for a fixed Δk_{eff} margin. WHC-SD-SQA-CSA-20330, *Use of a K_{eff} Subcritical Limit of 0.98*, addresses Audit Finding No. CS.2-1 by the Tiger Team for Westinghouse Hanford Facilities and quotes fractions of critical mass ($k_{\text{eff}}=0.95$) of 0.469, 0.618, and 0.762 for optimally spaced uranium rods in water at 0.95%, 1.25%, and 5.00% enrichments, respectively. These fractions of critical mass increase to 0.718, 0.818, and 0.897, respectively, for $k_{\text{eff}}=0.98$. Hence, the 5.0% enriched uranium at a $k_{\text{eff}}=0.95$ has a fraction of critical mass greater than the 0.95% enriched uranium at a $k_{\text{eff}}=0.98$. Interpolating in their tables obtains a fraction of critical mass of 0.710 for a 1.25% enrichment at ($k_{\text{eff}}=0.965$). This is also less than the 5.0% enriched uranium fraction of critical mass of 0.762 at ($k_{\text{eff}}=0.95$).

Figure 1. Fraction of Minimum Critical Mass for Uranium Systems.



A justification for a smaller criticality margin for low-enriched uranium is that the low enrichment implies that a large system is required to obtain criticality. The large system means that the fraction of critical mass at $k_{eff}=0.95$ is reduced compared to that for smaller systems. It can be demonstrated that the same phenomenon occurs for high-enriched uranium, if a poison is also present, to force a large system for criticality. For example, a 94% enriched uranium metal (with Godiva compositions) sphere reflected with water, has a calculated fraction of critical mass of 0.812 at $k_{eff}=0.95$. If the enriched uranium metal composition were homogeneously mixed with water (using theoretical densities to obtain the mix) to obtain a $H/^{235}U$ atomic ratio of 500 with this homogenized sphere again reflected with water, the fraction of critical mass decreases to 0.795 at $k_{eff}=0.95$. Adding a little ^{10}B to the homogeneous uranium-water mix to obtain a $^{235}U/^{10}B$ atomic ratio of 15, while retaining the $H/^{235}U$ atomic ratio of 500, will increase the critical radius because of neutron absorption in the ^{10}B , and the fraction of critical mass decreases to 0.684 at $k_{eff}=0.95$. Adding a little more ^{10}B to give a $^{235}U/^{10}B$ atomic ratio of 10 further decreases the fraction of critical mass to 0.497 at $k_{eff}=0.95$, where the critical ($k_{eff}=1.0$) sphere radius increases from 26.110 cm (10.279 in.) to 41.291 cm (16.256 in.) with this further ^{10}B increase.

The 0.684 and 0.497 fractions of critical mass for the high-enriched uranium-water mix with ^{10}B poison brackets the 0.618 ($k_{eff}=0.95$) for optimally spaced uranium rods in water at 1.25% enrichment as given in WHC-SD-WM-SAR-062, *K Basins Safety Analysis Report*. The amount of ^{10}B in the high-enriched uranium-water mix could be further adjusted by interpolation and additional calculations to obtain the 0.618 of the low-enriched uranium, if so desired. Neutronically, the high-enriched uranium in water with ^{10}B and the 1.25% enriched uranium rods have similarities. For the 1.25% enriched uranium rods, the large amount of ^{238}U acts as a poison and increases the volume required for criticality. For high-enriched uranium with ^{10}B , the ^{10}B is the poison and increases the volume. An important difference is that for the 1.25% enriched uranium rods, ^{238}U is intimately mixed with the ^{235}U so that the $^{235}U/U$ ratio is maintained even if fuel rods become rubble in an accident. Therefore, if the criticality analysis is based on optimal fuel rod diameter and spacing, damage can only decrease k_{eff} . It is typically much more difficult to guarantee the mixing, and retention of the ^{10}B (or some other poison) for

ACs, so such poisoning is often not practical for enriched systems. This high-enriched uranium with ^{10}B is given as an example to show the neutronic similarities, and that the fraction of critical mass ($k_{\text{eff}}=0.95$) decreases as the critical mass becomes large.

In summary, a smaller margin for criticality for very low-enriched uranium is justified for two reasons: (1) very low-enriched uranium systems have large critical masses (or volumes), which means that in terms of relative mass, there is a greater mass margin for low-enriched uranium for a fixed Δk_{eff} margin; and (2) ^{238}U is intimately mixed with ^{235}U so that the $^{235}\text{U}/\text{U}$ ratio is maintained even if fuel rods become rubble in an accident, ensuring that the neutronic poisoning from ^{238}U is still effective. Comparisons of relative mass margins for optimal uranium rods in water indicate that a Δk_{eff} margin of 0.02 for uranium enrichments ≤ 0.95 wt% ^{235}U has at least as large a relative mass margin as a Δk_{eff} margin of 0.05 for a uranium enrichment of 5.0 wt% ^{235}U . Similarly, a Δk_{eff} margin of 0.035 for uranium enrichments ≤ 1.25 wt% ^{235}U has at least as large a relative mass margin as a Δk_{eff} margin of 0.05 for a uranium enrichment of 5.0 wt% ^{235}U . Therefore, these margins are recommended at the lower enrichments.

It should be noted that there are two difficulties in demonstrating a smaller margin for criticality for very low-enriched uranium: (1) the Δk_{eff} uncertainties, both statistical in the Monte Carlo calculations and experimental for the benchmarks, tend to be about the same as for higher enrichments so with a smaller margin for criticality they tend to become relatively more important; and (2) there are few measurements at very low uranium enrichments because the required experimental systems must be so large. WHC-SD-SNF-ANAL-013, *MCNP Criticality Validation and Bias for LEU Systems*, provides a criticality validation study for experiments with low uranium enrichments using results of calculations, with Monte Carlo (computer code for) neutron photon (transport) (MCNP) using ENDF/B-V cross sections, reported in the literature. Fortunately, there is good agreement between the calculations and experiment, and WHC SD-SNF-ANAL-013 obtained a bias plus uncertainty of about 0.010.

By definition, a system for which the effective multiplication constant (k_{eff}) is exactly 1 is critical, and the system is subcritical if the k_{eff} is less than 1. Good engineering practices impose a safety margin on the system to ensure an accidental critical condition is not attained. This safety margin is typically imposed on the k_{eff} . For example, for the transport of NRC-licensed radioactive material, the NRC (NUREG/CR-5661) recommends the use of a k_{eff} limit of 0.95, including consideration of bias and uncertainties. As another example, storage and handling operations at K Basins are based on a k_{eff} limit of 0.98 (WHC-SD-WM-SAR-062, p. 6-1). The basis for use of 0.98 is included in HNF-SD-SNF-CSER-010, *Criticality Safety Evaluation Report for Storage and Removal of Spent Nuclear Fuel from K Basin*.

For transport of arrays of packages, regardless of the actual k_{eff} limit, k_{eff} is not controlled directly, but instead indirectly by controlling enrichment, mass, geometry, other process parameters, or a combination thereof that influences k_{eff} . At the time of evaluation, the values of these physical parameters are fixed, and the criticality evaluation controls k_{eff} by controlling the number of packages on a single shipment. Typically, the magnitude of the safety margin in

terms of the number of packages is not reported for a given safety margin in k_{eff} . However, DOE G 421.1-1 (p. 69), *Criticality Safety Good Practices Program Guide for DOE Nonreactor Nuclear Facilities*, recommends that k_{eff}

“should be used to assess safety only in conjunction with its relation to parametric curves representing the system under consideration. For example, the often-referenced “safe” condition of $k_{\text{eff}} = 0.95$ could be overly conservative at a point on the flat of a k_{eff} vs. H/X parametric curve, but very non-conservative on the steep upward slope of such a curve.”

Applying this concept to Hanford Site packages, the adequacy of a safety margin can be demonstrated by a parametric curve of k_{eff} versus the number of packages. For a hypothetical situation where the k_{eff} of X number of packages in ACs is 0.94, but the k_{eff} of an array of 2X packages is 0.96, a k_{eff} limit of 0.95 may be unnecessarily conservative based on as low as reasonably achievable, cost, or schedule considerations because the addition of a large number of packages to the shipment does not increase the k_{eff} significantly (and in no cases will a k_{eff} limit higher than 0.98 be used). An equivalent degree of safety based on a sufficient margin of safety in terms of the controllable physical parameter (the number of packages) may be demonstrated in this manner, where the k_{eff} is on the plateau of the parametric curve, and a k_{eff} limit ≤ 0.98 in this example is both safe and non-arbitrary.

Besides the safety margin in the k_{eff} limit, an additional component of safety analysis is using conservative assumptions. However, an analysis that is excessively conservative is not necessarily desirable. As noted in DOE G 421.1-1, (pp. 64), “excessive conservatism is any approximation or combination thereof that results in an excessive safety margin and thus needlessly hinders the mission of the facility or usurps resources.” Furthermore, DOE G 421.1-1, (p. 66) states, “appropriateness in conservatism means not only relevance, but also implies (1) realism in all aspects of evaluations, and (2) cost effectiveness that does not preclude consideration of reasonable alternatives.” Thus, conservative assumptions should be relevant, realistic, and not needlessly usurp resources. Criticality safety evaluations should determine whether assumptions are reasonably or excessively conservative.

3.8 HANFORD SITE ENVIRONMENTAL CONDITIONS

The equivalent package standards provide an equivalent degree of safety to those provided by 10 CFR 71 and supplemental NRC regulatory guidance. Equivalency is based on site-specific environmental and test conditions, which in the majority of cases, are identical to the regulations. However, as shown in Table 1, in a few cases the site-specific worst-case conditions are less severe and in two cases are more severe than those specified in the regulations. The following sections identify and justify the site environmental condition use in the equivalent standards.

Table 1. Hanford Onsite Package System Acceptable Performance Criteria.

10 CFR 71 Standard Performance Test		NRC Regulatory Acceptance Criteria	Hanford Standard Performance Test	Hanford Acceptance Criteria
10 CFR 71.43 General standards for all packages				
10 CFR 71.43 (a)	Smallest overall dimension not less than 10 cm (4 in).	Self-explanatory.	Same as 10 CFR 71.43 (a).	Same as NRC.
10 CFR 71.43 (b)	Outside of package must have tamper-indicating devices.	Self-explanatory.	Same as 10 CFR 71.43 (b).	Same as NRC.
10 CFR 71.43 (c)	Each package must include a positive fastening device.	Self-explanatory.	Same as 10 CFR 71.43 (c).	Same as NRC.
10 CFR 71.43 (d)	Package must be made of materials and construction that assure no significant chemical, galvanic, or other reaction among packaging components and package contents.	Self-explanatory.	Same as 10 CFR 71.43 (d).	Same as NRC.
10 CFR 71.43 (e)	Package valve or other device must be protected against unauthorized operation and must be provided with an enclosure to retain any leakage.	Self-explanatory.	Same as 10 CFR 71.43 (e).	Same as NRC.
10 CFR 71.43 (f)	Package must not have loss or dispersal of radioactive contents, no significant increase in external surface radiation levels, and no substantial reduction in the effectiveness of the packaging.	Self-explanatory.	Same as 10 CFR 71.43 (f).	Same as NRC.
10 CFR 71.43 (g)	Package must have, in still air at 38 °C (100 °F) and in the shade, no accessible surface temperature exceeding 50 °C (122 °F) in a nonexclusive use shipment or 85 °C (185 °F) in an exclusive use shipment.	Self-explanatory.	Same as 10 CFR 71.43 (g), except still air requirement is at the higher onsite temperature of 46 °C (115 °F).	Same as NRC.

Table 1. Hanford Onsite Package System Acceptable Performance Criteria.

10 CFR 71 Standard Performance Test		NRC Regulatory Acceptance Criteria	Hanford Standard Performance Test	Hanford Acceptance Criteria
10 CFR 71.43 (h)	Package may not incorporate a feature intended to allow continuous venting.	Self-explanatory.	Packages with high hydrogen-gas generation rates are allowed to have a vent or package must be demonstrated to withstand deflagration.	Package has HEPA filter, demonstrated to not allow liquid passage and maintain regulatory release limits under all performance tests. Sustaining deflagration demonstrated by meeting Service Level D requirements or test with no loss of containment.
10 CFR 71.45 Lifting and tiedown standards for all packages				
10 CFR 71.45 (a)	(1) Lifting attachment must have a minimum safety factor of 3 against yielding. Failure of device not to impair ability of package to meet other safety requirements. (2) Similar parts made inoperable or meet design requirements.	(1) ANSI N14.6 (a) Lifting 3 x total weight without exceeding yield strength. OR 5 x total weight without exceeding ultimate strength. (b) If failure of devices affects safety-related equipment, device designed for twice above loads or dual load path system provided. (2) Self-explanatory.	(1a) RL Hoisting and Rigging Manual. (1b) RL Hoisting and Rigging Manual. (2) Same as 10 CFR 71.45 (a).	(1a) RL Hoisting and Rigging Manual requirements. (1b) RL Hoisting and Rigging Manual requirements. (2) Same as NRC.
10 CFR 71.45 (b) Tiedown devices	(1) System of tiedown devices that is a structural part of the package must be capable of withstanding static loads of 2, 10, and 5 times the gross weight of the package applied to the center of gravity of the package. (2) Must be capable of being rendered inoperable for tying down the package during transport, or must be designed with strength equivalent to that required for tiedown devices. (3) Failure of the device under excessive load would not impair the ability of the package to meet other requirements.	(1) In this case, the tiedowns to critical components are to meet these requirements when attached to critical components. Loads are to be evaluated to Regulatory Guide 7.6 or applicable ASME subsection per NUREG-1609, for Service Level A. (2) Self-explanatory. (3) Self-explanatory.	(1) Same as 10 CFR 71.45 (b). (2) Same as 10 CFR 71.45 (b). (3) Same as 10 CFR 71.45 (b).	(1) Same as NRC, except demonstrated by meeting ASME BPVC, Section III, Subsection NE, for Service Level A. (2) Same as NRC. (3) Self-explanatory.

Table 1. Hanford Onsite Package System Acceptable Performance Criteria.

10 CFR 71 Standard Performance Test		NRC Regulatory Acceptance Criteria	Hanford Standard Performance Test	Hanford Acceptance Criteria
10 CFR 71.71 Normal conditions of transport				
10 CFR 71.71 (a) Evaluation	Evaluation of each package design must include a determination of the effect on that design of the conditions and tests.	Demonstrated by meeting requirements of Regulatory Guide 7.6 or applicable ASME subsection per NUREG-1609, for Service Level A. Load combination per Regulatory Guide 7.8	Same as 10 CFR 71.71 (a).	Same as NRC, except demonstrated by meeting ASME BPVC, Section III, Subsection NE, for Service Level A.
10 CFR 71.71 (b) Initial conditions	Ambient temperature preceding and following tests must remain constant at value between -29 °C (-20 °F) and 38 °C (100 °F). The initial internal pressure must be MNOP consistent with the ambient temperature considered to precede and follow the tests.	Demonstrated by meeting requirements of Regulatory Guide 7.6 or applicable ASME subsection per NUREG-1609, for Service Level A.	Same as 10 CFR 71.71 (b) except extreme temperature range is -33 °C (-27 °F) to 46 °C (115 °F) for the Hanford Site.	Same as NRC, except demonstrated by meeting ASME BPVC, Section III, Subsection NE, for Service Level A.
10 CFR 71.71 (c) Conditions and tests, (1) Heat	An ambient temperature of 38 °C (100 °F) in still air and insolation in accordance with: Form and location of surface/total insolation for a 12-hour period: - Flat surfaces transported horizontally - base/none - other surfaces/800 g-cal/cm² - Flat surfaces not transported horizontally/200 g-cal/cm² - Curved surfaces/400 g-cal/cm²	Package accessible surfaces in still air at 38 °C (100 °F) and in the shade must not exceed 50 °C (122 °F) in nonexclusive use shipments and 85 °C (185 °F) in exclusive use shipment. Also, worst case to be used in combination with other loads as defined in Regulatory Guide 7.8 to develop NCT load combinations.	NC Heat (Hanford Site) - Ambient Temperature= 46 °C (115 °F). - Still air. - Decay Heat. - Solar insolation for 12 h/d specific to the Hanford Site.	Same as NRC except 46 °C (115 °F) ambient temperature.
10 CFR 71.71 (c) (2) Cold	An ambient temperature of -40 °C (-40 °F) in still air and shade.	Criteria for Regulatory Guide 7.11 or 7.12 material brittle fracture requirements. Also, condition for evaluation of NCT and HAC performance.	Hanford Site - Ambient Temperature = -33 °C (-27 °F). - Still air. - In the shade.	ASME BPVC, Section VIII, Division 1 criteria per UCS-66 for brittle fracture. Also, conditions for evaluation of NC and AC.

Table 1. Hanford Onsite Package System Acceptable Performance Criteria.

10 CFR 71 Standard Performance Test		NRC Regulatory Acceptance Criteria	Hanford Standard Performance Test	Hanford Acceptance Criteria
10 CFR 71.71 (c) (3) Reduced external pressure	External pressure of 25 kPa (3.5 psi) absolute.	Considered as part of Regulatory Guide 7.8 to establish MNOP for NCT load combination development. Acceptance criterion of the worst-case load combination is ASME BPVC applicable subsection per NUREG-1609, Service Level A.	Same as 10 CFR 71.71 (c).	Same as NRC, except demonstrated by meeting ASME BPVC, Section III, Subsection NE, for Service Level A for sealed packages or evaluate for plugged filter for vented packages.
10 CFR 71.71 (c) (4) Increased external pressure	External pressure of 140 kPa (20 psi) absolute.	Considered as part of Regulatory Guide 7.8 to develop worst-case NCT load combinations. Acceptance criterion of the worst-case load combination is ASME BPVC applicable subsection per NUREG-1609, Service Level A.	Same as 10 CFR 71.71 (c).	Same as NRC, except demonstrated by meeting ASME BPVC, Section III, Subsection NE, for Service Level A when evaluated for packages with plugged filter and external pressure.
10 CFR 71.71 (c) (5) Vibration	Vibration normally incident to transport.	Acceptance criterion is applicable ASME BPVC subsection per NUREG-1609, Service Level A fatigue analysis. For transport vibration, NRC has accepted the argument that the NCT drop is bounding.	Same as 10 CFR 71.71 (c).	Same as NRC, except demonstrated by meeting ASME BPVC, Section III, Subsection NE, for Service Level A. NC free drop is used as an argument to bound vibration.
10 CFR 71.71 (c) (6) Water spray	Water spray that simulates exposure to rainfall of approximately 5 cm/h (2 in./h) for at least 1 hour.	Normally this is demonstrated by the argument that package is sealed leaktight and remains leaktight during NCT; therefore, it is watertight. Vulnerable areas to water intrusion such as impact limiters and neutron shielding fire vent ports are evaluated.	Same as 10 CFR 71.71 (c).	Same as NRC, except that since venting is allowed, filters must be evaluated.

Table 1. Hanford Onsite Package System Acceptable Performance Criteria.

10 CFR 71 Standard Performance Test		NRC Regulatory Acceptance Criteria	Hanford Standard Performance Test	Hanford Acceptance Criteria
10 CFR 71.71 (c) (7) Free drop	Between 1.5 and 2.5 hours after the water spray test, a free drop through a distance specified onto a flat, essentially unyielding, horizontal surface, striking the surface in a position for which maximum damage is expected.	For this case, all applicable combined loads from above as specified in Regulatory Guide 7.8. NRC acceptance criteria are Regulatory Guide 7.6 or applicable ASME BPVC subsection per NUREG-1609, Service Level A to demonstrate containment. For criticality control, load combinations on key components are measured against ASME BPVC, Section III, Subsection NG. Shielding limits are regulatory requirements.	Hanford Site requirements the same except 1.2 m (4.0 ft) to 0.3 m (1.0 ft) onto a worst-case Hanford Site target established as a 20 cm (8-in.) thick concrete pad, sufficiently large enough to contain the entire impact event. Concrete shall have a specified minimum strength of 21 MPa (3,000 psi) reinforced with No. 7 rebar on 30 cm (12 in.) centers over compact Hanford soil.	Same as NRC, except demonstrated containment by meeting ASME BPVC, Section III, Subsection NE, for Service Level C with all applicable load combinations. For criticality control, the same as the NRC. Shielding limits based on Hanford Site requirements.
10 CFR 71.71 (c) (8) Corner drop	Free drop from a height of 0.3 m (1 ft) onto flat, essentially unyielding, horizontal surface. Test applied only to fiberboard, wood, or fissile material rectangular packages not exceeding 50 kg (110 lb) and fiberboard, wood or fissile material cylindrical packages not exceeding 100 kg (220 lb).	Addressed on a case-by-case basis.	Same as 10 CFR 71.71 (c) (8) except the free drop surface target is the same as described above.	Same as NRC.
10 CFR 71.71 (c) (9) Compression	For packages weighing up to 5,000 kg (11,000 lb), the package must be subjected for 24 hours to a compressive load applied uniformly to top and bottom of package in the position the package would normally be transported. The compressive load must be the greater of the following: (i) The equivalent of 5 times the weight of the package; or (ii) The equivalent of 13 kPa (2 lbf/in ²) multiplied by the vertically projected area of the package.	Not normally performed for large package. For small packages, load combinations above are evaluated with compression to applicable ASME BPVC subsection per NUREG-1609 for Service Level A to demonstrate maintenance of containment.	Same as 10 CFR 71.71 (c).	Same as NRC, except demonstrated by meeting ASME BPVC, Section III, Subsection NE, for Service Level A.

Table 1. Hanford Onsite Package System Acceptable Performance Criteria.

10 CFR 71 Standard Performance Test		NRC Regulatory Acceptance Criteria	Hanford Standard Performance Test	Hanford Acceptance Criteria
10 CFR 71.71 (c) (10) Penetration	Impact of the hemispherical end of a vertical steel cylinder of 3.2 cm (1.25 in.) diameter and 6 kg (13 lb) mass, dropped from a height of 1 m (40 in.) onto the exposed surface of the package that is expected to be most vulnerable to puncture. The long axis of the cylinder must be perpendicular to the package surface.	Normally, for large packages, this is evaluated only when ancillary equipment, such as shut off valves required to NCT containment, is judged to be vulnerable to hard and light falling projectiles. Small packages with thin walls and/or vulnerable locations must be evaluated to demonstrate containment with only highly localized deformation in the impact region.	Same as 10 CFR 71.71 (c).	Same as NRC.
10 CFR 71.73 Hypothetical accident conditions				
10 CFR 71.73 (a) Test procedures	Evaluation for HAC is based on sequential application of the tests, in order indicated, to determine their cumulative effect. An undamaged specimen may be used for the water immersion tests.	Demonstrated by meeting requirements of Regulatory Guide 7.6 or applicable ASME BPVC subsection per NUREG-1609, for Service Level D.	Same as 10 CFR 71.73 (a).	Same as NRC except demonstrated no loss of containment by showing SED or accumulated plastic strain below allowable limits for the material. As an alternative, use ASME BPVC, Section III, Subsection NE, for Service Level D. For criticality control, load combinations on key components are measured against ASME BPVC, Section III, Subsection NG. Loss of shielding limits is the same as the NRC.

Table 1. Hanford Onsite Package System Acceptable Performance Criteria.

10 CFR 71 Standard Performance Test		NRC Regulatory Acceptance Criteria	Hanford Standard Performance Test	Hanford Acceptance Criteria
10 CFR 71.73 (b) Test conditions.	Except for water immersion tests to demonstrate compliance with the requirements during testing, the ambient air temperature before and after the tests must remain constant at the value between -29 °C (-20 °F) and 38 °C (100 °F). The initial internal pressure within the containment system must be the MNOP, unless a lower internal pressure, consistent with the ambient temperature assumed to precede and follow the tests, is more unfavorable.	Self-explanatory.	Same as 10 CFR 71.73 (b) except temperature range for Hanford Site is -33 °C (-27 °F) to 46 °C (115 °F).	Self-explanatory.
10 CFR 71.73 (c) Tests	Tests for HAC must be conducted as follows:			
10 CFR 71.73 (c) (1) Free drop	A free drop through a distance of 9 m (30 ft) onto a flat, essentially unyielding, horizontal surface, striking the surface in a position for which maximum damage is expected.	NRC acceptance criteria are Regulatory Guide 7.6 or applicable ASME BPVC subsection per NUREG-1609, Service Level D to demonstrate containment. For criticality control, load combinations on key components are measured against ASME BPVC, Section III, Subsection NG. Also shielding must be within HAC limit of $\leq 1,000$ mR/h, 1 m (40 in.) from external surface of package.	AC Hanford Site Requirements the same except, free drop onto a worst-case Hanford Site target established as a 20 cm (8-in.) thick concrete pad, sufficiently large enough to contain the entire impact event. Concrete shall have a specified minimum strength of 21 MPa (3,000 psi) reinforced with No. 7 rebar on 30 cm (12 in.) centers over compact Hanford soil.	Same as NRC, except demonstrating no loss of containment by showing the SED or accumulated plastic strain is below allowable limits for the material. As an alternative, use ASME BPVC, Section III, Subsection NE, for Service Level D. For criticality control, load combinations on key components are measured against ASME BPVC, Section III, Subsection NG. Loss of shielding limits is the same as the NRC.

Table 1. Hanford Onsite Package System Acceptable Performance Criteria.

10 CFR 71 Standard Performance Test		NRC Regulatory Acceptance Criteria	Hanford Standard Performance Test	Hanford Acceptance Criteria
10 CFR 71.73 (c) (2) Crush	Subjection to a dynamic crush test by positioning the specimen on a flat, essentially unyielding, horizontal surface so as to suffer maximum damage by the drop of a 500 kg (1,100 lb) mass from 9 m (30 ft) onto the specimen. The mass must consist of a solid mild steel plate 1 m (40 in.) by 1 m (40 in.) and must fall in a horizontal attitude. Test is required only when the specimen has a mass not greater than 500 kg (1,100 lb), and overall density not greater than 1,000 kg/m³ (62.4 lb/ft³), and radioactive contents greater than 1,000 A₂ not as special form radioactive material.	NRC acceptance criteria are Regulatory Guide 7.6 or applicable ASME BPVC subsection per NUREG-1609, Service Level D and must follow free drop (sequentially) to demonstrate containment. For criticality control, load combinations on key components are measured against ASME BPVC, Section III, Subsection NG. Also shielding must be within HAC limit of $\leq 1,000$ mR/h 1 m (40 in.) from external surface of package.	Same as 10 CFR 71.73 (c) (2) except the target surface for the Hanford Site is as specified above and crushing object is an identical package dropping from a height of 9 m (30 ft) onto the damaged package in the orientation to cause the most damage. Specimen on a concrete surface as defined above.	Same as NRC, except demonstrating no loss of containment by showing the SED or accumulated plastic strain is below allowable limits for the material. As an alternative, use ASME BPVC, Section III, Subsection NE, for Service Level D. For criticality control, load combinations on key components are measured against ASME BPVC, Section III, Subsection NG. Loss of shielding limits is the same as the NRC.
10 CFR 71.73 (c) (3) Puncture	A free drop through a distance of 1 m (40 in.) in a position for which maximum damage is expected, onto the upper end of a solid, vertical, cylindrical, mild steel bar mounted on an essentially unyielding, horizontal surface. The bar must be 15 cm (6 in.) in diameter, with the top horizontal and its edge rounded to a radius of not more than 6 mm (0.25 in.), and of a length as to cause maximum damage to the package, but not less than 20 cm (8 in.) long. The long axis of the bar must be vertical.	NRC acceptance criteria are Regulatory Guide 7.6 or applicable ASME BPVC subsection per NUREG-1609, Service Level D and must follow free drop and crush (when required) (sequentially) to demonstrate containment. For criticality control, load combinations on key components are measured against ASME BPVC, Section III, Subsection NG. Also shielding must be within HAC limit of $\leq 1,000$ mR/hr 1 m (40 in.) from external surface of package.	Same as 10 CFR 71.73, except puncture rod mounted on concrete slab described for free drop.	Same as NRC, except demonstrating no loss of containment by showing the SED or accumulated plastic strain is below allowable limits for the material. As an alternative, use ASME BPVC, Section III, Subsection NE, for Service Level D. For criticality control, load combinations on key components are measured against ASME BPVC, Section III, Subsection NG. Loss of shielding limits is the same as the NRC.

Table 1. Hanford Onsite Package System Acceptable Performance Criteria.

10 CFR 71 Standard Performance Test		NRC Regulatory Acceptance Criteria	Hanford Standard Performance Test	Hanford Acceptance Criteria
10 CFR 71.73 (c) (4) Thermal	Exposure of the specimen fully engulfed in a hydrocarbon fuel/air fire of sufficient extent to provide an average emissivity coefficient of at least 0.9, with an average flame temperature of at least 800 °C (1,475 °F) for a period of 30 minutes. The fuel source must extend horizontally at least 1 m (40 in.), but may not extend more than 3 m (10 ft), beyond any external surface of the specimen, and the specimen must be positioned 1 m (40 in.) above the surface of the fuel source. The surface absorptivity coefficient must be either that value which the package may be expected to possess if exposed to the fire specified or 0.8, whichever is greater. Artificial cooling may not be applied after cessation of external heat input, and any combustion of materials of construction, must be allowed to proceed until it terminates naturally.	NRC acceptance criteria is Regulatory Guide 7.6 or applicable ASME BPVC subsection per NUREG-1609, Service Level D and must follow free drop, crush (when required), and puncture (sequentially) to demonstrate containment. For criticality control, load combinations on key components are measured against ASME BPVC, Section III, Subsection NG. Also shielding must be within HAC limit of $\leq 1,000$ mR/h, 1 m (40 in.) from external surface of package. Also must maintain seal and shielding effectiveness and shielding must not melt and re-cool. Seals not allowed to reach temperature above maximum rated continuous use temperature.	AC Hanford Site: - Exposure of the entire package for not less than 30 minutes. - To a heat flux of not less than a radiation environment of 800 °C (1,475 °F). - With a flame emissivity coefficient of at least 0.9. - Package surface absorptivity must be published value at temperature. - Significant, convective heat must be included based on still ambient air at 800 °C (1,475 °F). - Artificial cooling of the package is applied after 30 minutes. - Combustion is terminated after 30 minutes.	Same as NRC, except demonstrated by meeting ASME BPVC, Section III, Subsection NE Service Level D for thermal loading. For criticality control, load combinations on key components are measured against ASME BPVC, Section III, Subsection NG. Also in this case, water applied after 30 minutes from initiation of fire. Seals must maintain containment and properties for 1 hour. Loss of shielding limits is the same as the NRC.
10 CFR 71.73 (c) (5) Immersion – fissile material	In cases where water leakage has not been assumed for criticality analysis, immersion under a head of water of at least 0.9 m (3 ft) in the attitude for which maximum leakage is expected.	NRC acceptance criteria are Regulatory Guide 7.6 or applicable ASME BPVC subsection per NUREG-1609, Service Level D and must follow free drop, crush (when required), puncture, and thermal fire (sequentially) to demonstrate containment. For criticality control, load combinations on key components are measured against ASME BPVC, Section III, Subsection NG.	Same as 10 CFR 71.71 (c) when no restriction on transport over large and deep bodies of water.	Same as NRC, except demonstrated by meeting ASME BPVC, Section III, Subsection NE Service Level D for external pressure loading. For criticality control, load combinations on key components are measured against ASME BPVC, Section III, Subsection NG. Vented packages require fully flooded conditions for criticality evaluation.

Table 1. Hanford Onsite Package System Acceptable Performance Criteria.

10 CFR 71 Standard Performance Test		NRC Regulatory Acceptance Criteria	Hanford Standard Performance Test	Hanford Acceptance Criteria
10 CFR 71.73 (c) (6) Immersion – all packages	A separate, undamaged specimen must be subjected to water pressure equivalent to immersion under a head of water of at least 15 m (50 ft). For test purposes, an external pressure of water of 150 kPa (21.7 lbf/in ²) gauge is considered to meet these conditions.	NRC acceptance criteria are Regulatory Guide 7.6 or applicable ASME subsection per NUREG-1609, Service Level D to demonstrate containment. For criticality control, load combinations on key components are measured against ASME BPVC, Section III, Subsection NG.	Same as 10 CFR 71.73 (c) when no restriction on transport over large and deep bodies of water.	Same as NRC, except demonstrated by meeting ASME BPVC, Section III, Subsection NE Service Level D for external pressure loading. For criticality control, load combinations on key components are measured against ASME BPVC, Section III, Subsection NG. Vented packages require fully flooded conditions for criticality evaluation.

AC = accident condition
 ANSI = American National Standards Institute
 ASME = American Society of Mechanical Engineers
 BPVC = Boiler and Pressure Vessel Code
 CFR = Code of Federal Regulations
 HAC = hypothetical accident condition
 HEPA = high-efficiency particulate air (filter)
 NC = normal condition
 NCT = normal conditions of transport
 NRC = U.S. Nuclear Regulatory Commission
 MNOP = maximum normal operating pressure
 RL = U.S. Department of Energy, Richland Operations Office
 SED = strain energy density

Sources:

DOE/RL-92-36, *RL Hoisting and Rigging Manual*, U.S. Department of Energy, Richland Operations Office, Richland, Washington.
 NUREG-1609, *Standard Review Plan for Transportation Packages for Radioactive Material*, Chapter 7, Spent Fuel Project Office, U.S. Nuclear Regulatory Commission, Washington, D.C.
 Regulatory Guide 7.6, *Design Criteria for Structural Analysis of Shipping Cask Containment Vessels*, U.S. Nuclear Regulatory Commission, Washington, D.C.
 Regulatory Guide 7.8, *Load Combinations for the Structural Analysis of Shipping Casks for Radioactive Material*, U.S. Nuclear Regulatory Commission, Washington, D.C.
 Regulatory Guide 7.11, *Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 Inches (0.1 m)*, U.S. Nuclear Regulatory Commission, Washington, D.C.
 Regulatory Guide 7.12, *Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Wall Thickness Greater than 4 Inches (0.1 m) But Not Exceeding 12 Inches (0.3 m)*, U.S. Nuclear Regulatory Commission, Washington, D.C.

3.8.1 Conditions for Package External Temperature

The regulation states that the maximum package external temperature will be determined in the shade at an ambient still air temperature of 38 °C (100 °F). Based on climatological data reported from 1912 to 1980 in WHC-SD-TP-RPT-004, *Environmental Conditions for On-Site Hazardous Materials Packages*, the highest recorded temperature on the Hanford Site is 46 °C

(115 °F). Consequently, site-specific determination of maximum allowable package external temperature is based on a more severe condition than for the regulations. However, the maximum allowable package external temperature is unchanged from the temperatures specified in the regulation of 85 °C (185 °F) for exclusive use and 50 °C (122 °F) for nonexclusive use shipments.

3.8.2 Heat and Cold Temperature Test Conditions

The range of heat and cold environmental temperature test conditions specified in the regulations is 38 °C to –40 °C (100 °F to –40 °F). Based on Hanford climatological data provided in WHC-SD-TP-RPT-004, the highest and lowest recorded temperatures on the Hanford Site are 46 °C and –33 °C (115 °F and –27 °F). Consequently, site-specific package heat test condition high temperature extreme is more severe than the regulations. In the case of site-specific package cold test condition, the Hanford Site low temperature extreme is only slightly less severe than the regulations. Equivalency to the regulations is also provided by institutional controls at the Hanford Site, in that at such low temperature extremes all operations (including onsite transport) would be limited to only critical and emergency operations for facility and personnel safety.

3.8.3 Package Lifting

The DOE/RL-92-36, *RL Hoisting and Rigging Manual*, is required for all Hanford Site lifting and handling operations and is equivalent to packaging regulatory requirements.

3.8.4 Initial Conditions for NCs of Transport

The range of NC test condition temperature specified in the regulations is 38 °C to –29 °C (100 °F to –20 °F). Based on Hanford Site climatological data provided in WHC-SD-TP-RPT-004, the highest and lowest recorded temperatures on the Hanford Site are 46 °C and –33 °C (115 °F and –27 °F). Consequently, site-specific package initial conditions for NC tests are more severe than the regulations for the range of environmental temperatures.

3.8.5 Solar Insolation

The Hanford Site solar insolation values are based on a 25-year average of direct and diffuse solar radiation measurements (WHC-SD-TP-RPT-004). These values are normalized to the requirements of 10 CFR 71. Because these values are specific to the Hanford Area, they are used in lieu of the regulatory values.

3.8.6 Free Drop Surface

The regulatory requirement for free drop surfaces is a hard, essentially unyielding surface, which according to IAEA guidelines is defined as a target with a mass of 10 times that of the package. As defined by the IAEA the target is to be covered by a 5 cm (2-in.) thick steel plate embedded in concrete. On the Hanford Site, no such hard unyielding surfaces exist. The worst-case accident impact surface under all transportation conditions is the CWC storage pad. This is defined as the Hanford Site equivalent to a hard unyielding surface due to its mass and relative hardness to other objects. The CWC storage pad is a 20 cm (8-in.) thick concrete slab, two-way reinforced with number 7 rebar on 30 cm (12-in.) centers constructed for heavy truck loads. This is based on Lawrence Livermore National Laboratory studies in NUREG/CR-4829, *Shipping Container Response to Severe Highway and Railway Accident Conditions*, which has also shown that only impacts onto large massive objects generate significant forces on the package.

In addition, the studies in NUREG/CR-4829 have shown that 48 km/h (30 mi/h) impact speeds onto hard unyielding surfaces are equivalent to impact speeds of 97 km/h (60 mi/h) onto a "medium" hard surface, such as concrete. Because speeds on the Hanford Site are limited to 88 km/h (55 mi/h), this would limit maximum package impact velocity onto a massive concrete surface to an equivalent velocity onto a hard unyielding surface. Also, as demonstrated by Belgium and French Atomic Energy Commission (SAND87-1903, *Summary of Radioactive Materials Package Storage*) tiedown restraints failure and driver reaction time would further limit the package impact velocities to approximately one-half the initial conveyance speed. Based on these studies, the test conditions are equivalent to the regulations and conservatively bound onsite transportation ACs.

A NC free drop, unlike the regulatory scenario where normal operations would continue and be reported later, a NC free drop of a package on the Hanford Site would institutionally and administratively be considered at best an unusual occurrence and at worst an event. Consequently, all operations would stop, proper personnel notified, the area would be secured, steps would be taken to mitigate any release, and recovery from the condition would be initiated. In all cases, the package would be removed from normal transportation service for assessment of damage. These institutional and administrative controls provide a measure of control exceeding any controls applied in commercial transport of radioactive material packages for a NC free drop. In addition, transient load transfers are not routinely allowed onsite, and for those requiring load transfer, due to conveyance break down or unusual circumstances, will be monitored and performed to approved procedures. The majority of routine packaging handling operations is conducted at the shipping or receiving facilities.

3.8.7 Puncture Surface

The test puncture bar surface is mounted in the same concrete surface as the free drop condition, and in the prescribed manner as the regulatory puncture bar is mounted onto an essentially hard unyielding surface. Equivalency of the puncture bar test to the regulations is based on the same rationale for the free drop surface, in that no essentially unyielding surface exists on the Hanford Site. However, there are abutments and concrete surfaces at the Hanford Site on which objects

firmly mounted to a concrete surface can act as a puncture bar. Assuming a puncture bar is firmly mounted onto a massive concrete surface would bound any puncturing object on the Hanford Site. Also, at the low impacting velocity specified in the regulations, a package impacting a puncture bar mounted to a massive concrete surface would behave in the same manner as a package impacting a puncture bar mounted on a hard unyielding surface. This is due to localized affect of the loading on the package and the distribution of the load from the mounting base.

3.8.8 Fire AC

The fire test is specified at the same temperature as the regulatory fire test. Also, as specified in the regulations, the package is to be fully engulfed in flames for a duration of 30 minutes. Consequently, the bounding fire is defined as an 800 °C (1,475 °F) temperature fire with a flame emissivity of 0.9, with the specified package material surface emissivities and absorptivities obtainable in published thermal properties data. As specified in the regulations, and package combustibles shall be allowed to completely burn and the packages cooled by natural convection and radiation only. For onsite conditions, based on a dedicated fire department and conservative response time, the package is assumed to have quenching applied after 30 minutes. Accounting for these mitigating factors, the fire AC parameters provides an equivalent degree of safety as provided by the regulations.

4.0 REFERENCES

- 10 CFR 71, "Packaging and Transportation of Radioactive Material," *Code of Federal Regulations*, as amended.
- 10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," *Code of Federal Regulations*, as amended.
- DOE G 421.1-1, 1999, *Criticality Safety Good Practices Program Guide for DOE Nonreactor Nuclear Facilities*, U. S. Department of Energy, Washington, D.C.
- HNF-SD-SNF-CSER-010, *Criticality Safety Evaluation Report for Storage and Removal of Spent Nuclear Fuel from K Basin*, Fluor Hanford, Richland, Washington.
- Manjoine, M. J., 1983, "Damage and Failure at Elevated Temperature," *Transaction of the ASME Journal of Pressure Vessel Technology*, Volume 105, pp. 58-62, February 1983.
- NUREG-1609, *Standard Review Plan for Transportation Packages for Radioactive Material*, Chapter 7, Spent Fuel Project Office, U.S. Nuclear Regulatory Commission, Washington, D.C.

NUREG/CR-1815, *Recommendations for Protecting Against Failure by Brittle Fracture in Ferritic Steel Shipping Containers up to Four Inches Thick*, U.S. Nuclear Regulatory Commission, Washington, D.C.

NUREG/CR-3019, *Recommended Welding Criteria for Use in the Fabrication of Shipping Containers for Radioactive Materials*, U.S. Nuclear Regulatory Commission, Washington, D.C.

NUREG/CR-3854, *Fabrication Criteria for Shipping Containers*, U.S. Nuclear Regulatory Commission, Washington, D.C.

NUREG/CR-4829, UCID-20733, *Shipping Container Response to Severe Highway and Railway Accident Conditions*, Lawrence Livermore National Laboratory, Livermore, California.

NUREG/CR-5661, *Recommendations for Preparing the Criticality Safety Evaluation of Transportation Packaging for Radioactive Material*, U.S. Nuclear Regulatory Commission, Washington, D.C.

NUREG/CR-6007, *Stress Analysis of Closure Bolts for Shipping Casks*, U.S. Nuclear Regulatory Commission, Washington, D.C.

Regulatory Guide 7.6, *Design Criteria for Structural Analysis of Shipping Cask Containment Vessels*, U.S. Nuclear Regulatory Commission, Washington, D.C.

Regulatory Guide 7.8, *Load Combinations for the Structural Analysis of Shipping Casks for Radioactive Material*, U.S. Nuclear Regulatory Commission, Washington, D.C.

Regulatory Guide 7.11, *Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 Inches (0.1 m)*, U.S. Nuclear Regulatory Commission, Washington, D.C.

SAND87-1903, *Summary of Radioactive Materials Package Storage*, Sandia National Laboratories, Albuquerque, New Mexico.

WHC-SD-SNF-ANAL-013, *MCNP Criticality Validation and Bias for LEU Systems*, Rev 0, Westinghouse Hanford Company, Richland, Washington.

WHC-SD-SQA-CSA-20330, *Use of a k_{eff} Subcritical Limit of 0.98*, Westinghouse Hanford Company, Richland, Washington.

WHC-SD-TP-RPT-004, *Environmental Conditions for On-Site Hazardous Materials Packages*, Westinghouse Hanford Company, Richland, Washington.

WHC-SD-WM-SAR-062, 2000, *K Basins Safety Analysis Report*, Rev. 3L, Fluor Hanford, Richland, Washington.

APPENDIX C

UNREVIEWED SAFETY QUESTION PROCESS FOR TRANSPORTATION (USQ_T)

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TABLE

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LIST OF TERMS

CFR	<i>Code of Federal Regulations</i>
CoC	Certificate of Compliance
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
JCO	justification for continued operation
NRC	U.S. Nuclear Regulatory Commission
PISA	potential inadequacy in the safety analyses
POC	point of contact
PSSD	package specific safety document
SARP	safety analysis report for packaging
SEP	safety evaluation for packaging
SRB	safety review board
TSD	<i>Hanford Sitewide Transportation Safety Document</i>
TSR	technical safety requirement
T&P	transportation and packaging
USQ _T	unreviewed safety question process for transportation
USQ _T D	unreviewed safety question process for transportation determination
USQ _T S	unreviewed safety question process for transportation screening

APPENDIX C

UNREVIEWED SAFETY QUESTION PROCESS FOR TRANSPORTATION (USQ_T)

1.0 PURPOSE

This procedure establishes the requirements for the Unreviewed Safety Question Process for Transportation (USQ_T). Attachments 1 through 9 provide additional information and guidance when preparing and submitting an USQ_T.

The USQ_T is to determine level of approval and not determine safety. The USQ_T permits a contractor to make physical changes and procedural changes to transportation activities without prior U.S. Department of Energy (DOE), Richland Operations Office (RL) approval, provided the changes do not explicitly or implicitly affect the *Hanford Sitewide Transportation Safety Document* (TSD). The USQ_T provides a contractor with the flexibility needed to conduct day-to-day operations by requiring that only those changes with a potential to impact the safety basis be approved by RL. This allows RL to focus its review on those changes significant to safety. The USQ_T is an important tool for keeping the transportation safety basis current by ensuring changes receive the appropriate level of review and approval. The USQ_T also specifies required actions if a potential inadequacy of the safety analyses is discovered. The process provides a method for contractors to determine if an USQ_T is involved and the actions to take if the situation involves an USQ_T.

2.0 SCOPE

This procedure is applicable to all transportation and packaging (T&P) activities with greater than 1 A₂ of radioactive material. The USQ_T is applicable to new transportation activities approved by RL.

Any change to the technical safety requirements (TSR) requires RL approval; therefore, the USQ_T does not apply. The USQ_T cannot be used to modify a portion or statement in TSRs or safety evaluation reports.

3.0 IMPLEMENTATION

This procedure is effective 90 days after publication to ensure adequate time for training and qualification.

4.0 REQUIREMENTS

4.1 SITUATIONS REQUIRING THE IMPLEMENTATION OF THE USQ_T

1. The USQ_T shall be implemented in situations where there is a:
 - a. Temporary or permanent change in the T&P activity as described in the existing documented safety analysis (for T&P activities this is the TSD);
 - b. Temporary or permanent change in the procedures as described in the existing documented safety analysis;
 - c. Test or experiment not described in the documented safety analysis; or
 - d. Potential inadequacy of the documented safety analysis because the analysis may not be bounding or may be otherwise inadequate.
2. The USQ_T must be completed before implementing a proposed T&P activity (design modification, activity change, procedure change). For modifications, the USQ_T determination (USQ_TD) is completed before the modified document is released to perform work in modifying packaging systems. If the proposed activity involves a positive USQ_T, RL approval must be obtained before implementing the proposed activity. If the proposed activity does not involve an USQ_T, the contractor may approve and implement the proposed activity.

4.2 ACTIONS REQUIRED FOR POTENTIAL INADEQUACIES IN DOCUMENTED SAFETY ANALYSES (PISAs)

If the contractor T&P program management discovers or is made aware and has reason to believe that there is a potential inadequacy of the safety analyses, contractor T&P program management must:

1. Take action, as appropriate, to place or maintain the T&P packaging system or activity in a safe condition until an evaluation of the safety of the situation is completed.
2. Notify RL and other contractors of the situation.
3. Perform an USQ_TD and notify RL and other contractors promptly of the results.
4. Submit the evaluation of the safety of the situation to RL prior to removing any operational restrictions initiated to place or maintain the facility in a safe condition.

If an USQ_T is determined to be present, RL approval is required before the operational restrictions are removed.

4.3 USQ_T TRAINING AND QUALIFICATION REQUIREMENTS

1. All personnel responsible for evaluating or approving USQ_T documentation shall receive training on this process, the TSD, and changes thereof.
2. USQ_T evaluators and other personnel directly involved are designated in writing by the contractor T&P manager and must meet the minimum training and qualification requirements defined in Chapter X of the TSD.

4.4 IMPLEMENTING DOCUMENTATION

The contractor T&P program manager shall establish T&P-specific implementing documentation which:

1. Documents application of any categorical exclusions.
2. Defines change control processes of physical changes.
3. Integrates the USQ_T into change control and work control processes, such as design, engineering, maintenance, inspection, and operation.
4. Requires an independent review by a qualified evaluator of each USQ_TD.
5. Requires contractor T&P program manager approval of changes that require USQ_TD so that management is informed of the determination results.

4.5 JUSTIFICATION FOR CONTINUED OPERATIONS (JCO)

The specific purpose of a JCO is to document the technical bases for continuing to operate outside of conditions defined in the documented safety analysis under specific conditions and to request RL approval for continued or temporary operation under those conditions. The JCO amends the T&P safety basis upon RL approval for the interim operations.

4.6 SAFETY REVIEW BOARD

The contractor's T&P management may choose to form a safety review board (SRB) consisting of multi-discipline experts to advise the manager on implementation of the USQ_T. Qualification of members is at the discretion of the manager.

4.7 REPORT AND RECORD REQUIREMENTS

1. The contractor's T&P manager shall annually submit to RL a summary of the USQ_TD performed since the prior submission. This summary is submitted to RL in support of RL's annual update of the TSD.
2. Contractor T&P program management shall maintain records as shown in Section 7.0. USQ_TD records shall be maintained for the operational life of the Hanford Site and transferred to any subsequent contractor before or at the same time the responsibility for the function is transferred.

4.8 PROCESS REQUIREMENTS

1. Pre-screening/Screening Proposed Activities

For the purposes of the USQ_T, "activity" means any change to procedures or equipment (including prior undocumented changes), or activities.

- a. **Applicability Assessment:** Applicability may be used to determine that the proposed T&P activity does not require the performance of a documented USQ_T screening (USQ_TS) or USQ_TD. The two types of applicability assessment are as follows:
 - 1) For proposed activities that must be submitted to RL for approval, the performance of USQ_TD is not applicable. Examples include major new transportation activities and TSR changes.
 - 2) For types of changes approved as Categorical Exclusions in the TSD, Chapter VI, Section 6.1.2, USQ_TD are not applicable. Examples include changes that are purely editorial and make no technical change, etc. Each category being excluded must receive an USQ evaluation, must be approved by RL, and must be documented in TSD.
 - 3) Application of the applicability assessment criterion to a specific change does not require a qualified USQ_T trained evaluator; however, the person performing this function must have familiarity with the process and categorical exclusions.
- b. **Screening**
 - 1) Documented USQ_TS of proposed activities may be used to determine whether USQ evaluations are required. An USQ_TD is not required if:
 - Any proposed change is not a temporary or permanent change in the activity or in the procedures as described in the safety basis, and
 - Any proposed test or experiment is described in the safety basis.

- 2) This screening procedure will specify the screening questions to be used.
- 3) Because knowledge of the documented safety analysis and the intent of the USQ_T is required, concurrence of two qualified USQ_T evaluators is required for an USQ_TS.
- 4) For all changes, which cannot be eliminated from the USQ_T by prescreening or screening above, an USQ_TD shall be performed.

2. Identify PISA

A PISA is defined as an existing condition that is identified as potentially being outside of the bounded conditions of the safety basis. The identification of PISAs may come from multiple sources. For example, occurrence screenings, questions from workers or management, questions from a regulatory agency, etc., may be used to indicate a potential PISA exists.

3. Perform USQ_TD

- a. A proposed activity involves an USQ_T if the situation is not bound by the criteria established in the TSD.
- b. The USQ_TD shall address the above condition by answering the following questions as a minimum. If the answer is **No** proceed to the next question. If the answer to **all** questions by both evaluators is "No", the situation does not involve an USQ_T. If **any** question is **Yes**, it is an USQ_T.
 - 1) Could the design modification, activity, change, or issue increase the probability of occurrence or consequences of an accident or malfunction previously evaluated in the TSD?
 - 2) Could the design modification, activity, change, or issue create the possibility of an accident or malfunction different from any previously evaluated in the TSD?
 - 3) Does the design modification, activity, change, or issue reduce the margin of safety as defined in the TSD?
- c. The determination is completed by one qualified evaluator and reviewed and concurred with by a second qualified evaluator. It records the scope of the determination and the logic for determining whether or not an USQ_T exists. A defensible explanation must be documented for the answers given. (The justification should stand alone on its technical merit.)
- d. If a proposed activity involves an USQ_T, obtain RL approval prior to implementation. If the proposed activity does not involve an USQ_T, the proposed activity may be implemented.

5.0 PROCESS

This section presents the basic USQ_T for proposed activities and for PISAs.

5.1 PROCESS FOR PROPOSED ACTIVITIES

For the purposes of the USQ_T, "activity" means any change to packaging systems and/or procedures (including prior undocumented changes), or any new activities.

<i>Actionee</i>	<i>Step</i>	<i>Action</i>
USQ _T Qualified Evaluator	5.1.1	Conduct an applicability assessment. IF this is a categorical exclusion exit this procedure and document on the appropriate configuration control document (e.g., engineering change notice, engineering data transmittal, procedure change form, etc.), ELSE continue.
First USQ _T Qualified Evaluator	5.1.2	Conduct screen for USQ _T . If screen is positive go to Step 5.1.4, ELSE continue.
Second USQ _T Qualified Evaluator	5.1.3	Perform independent review of USQ _T screen. IF screen is negative complete documentation and exit this procedure, ELSE, IF screen is positive go to Step 5.1.4.
First USQ _T Qualified Evaluator	5.1.4	Perform an USQ _T D. (Refer to determination questions in paragraph 4.8.3.b.)
Second USQ _T Qualified Evaluator	5.1.5	Perform independent review of the USQ _T D. Note: If the answer to any determination questions (paragraph 4.8.3.b) by both evaluators is "yes", the activity involves an USQ _T . Note: If the differences cannot be resolved, submit to the contractor T&P manager for resolution.
Contractor T&P Program Manager	5.1.6	IF a proposed activity involves a positive USQ _T , THEN modify the proposed activity and return to Step 5.1.3, OR go to Step 5.1.7. OR, cancel the proposed activity and exit ELSE, go to Step 5.1.10.
	5.1.7	Obtain RL approval of the proposed activity and any necessary safety basis changes.
	5.1.8	Notify the TSD configuration manager.
TSD Configuration Manager	5.1.9	Notify the contractor's T&P point of contact (POC).
	5.1.10	Implement the proposed activity.

5.2 PROCESS FOR PISAs

<i>Actionee</i>	<i>Step</i>	<i>Action</i>
Contractor T&P Activity Managers	5.2.1	Determine whether the event or condition indicates a PISA. IF a determination as to whether the event or condition indicates a PISA cannot be made OR, the event or condition indicates a PISA, THEN go to Step 5.2.2. ELSE, no further USQ _T action is required.
	5.2.2	Notify the contractor T&P program manager of the PISA.
	5.2.3	Take action, as appropriate, to place and maintain the T&P activity in a safe condition.
	5.2.4	Notify RL of the situation as an off normal occurrence under the occurrence reporting system.
Contractor T&P Program Manager	5.2.5	Notify the TSD configuration manager.
TSD Configuration Manager	5.2.6	Notify the contractor's T&P POC.
First USQ _T Qualified Evaluator	5.2.7	Perform an USQ _T D. (Refer to evaluation questions in paragraph 4.8.3.b.)
Second USQ _T Qualified Evaluator	5.2.8	Perform independent review of the USQ _T D. Note: If the answer to any determination questions (paragraph 4.8.3.b) by both evaluators is "yes", the activity involves an USQ _T . Note: If the differences cannot be resolved, submit to the contractor T&P manager for resolution.
Contractor T&P Program Manager	5.2.9	Submit the USQ _T evaluation to RL and the TSD configuration manager. IF a situation involves a positive USQ _T , THEN go to Step 5.2.10. ELSE, go to Step 5.2.14.
TSD Configuration Manager	5.2.10	Notify the contractor's T&P POC.
Contractor T&P Program Manager	5.2.11	Obtain RL approval before removing operational restrictions.

<i>Actionee</i>	<i>Step</i>	<i>Action</i>
TSD Configuration Manager	5.2.12	IF operation is to continue with an USQ _T , THEN prepare a modification to the TSD or a JCO and submit to RL for approval. ELSE, continue.
	5.2.13	Obtain RL approval of the revised safety basis.
	5.2.14	Remove any operational restrictions initiated (in Step 5.2.3).
	5.2.15	Incorporate changes in the documented safety basis in the next annual update, as appropriate.

6.0 FORMS

Unreviewed Safety Question Process for Transportation (USQ_T) Screening form (Attachment 3)
Unreviewed Safety Question Process for Transportation (USQ_T) Determination form (Attachment 4)

7.0 RECORD IDENTIFICATION

Table 7-1. Records Capture Table

Name of Document	Submittal Responsibility	Retention Responsibility
USQ _T screen/determination.	USQ _T evaluator	Contractor T&P program manager
T&P program implementing documents.	Contractor T&P program manager	Contractor T&P program manager
Training records related to USQ process.	Trainers	Contractor T&P program manager

8.0 REFERENCES

8.1 SOURCE REFERENCES

10 CFR 830, "Nuclear Safety Management," *Code of Federal Regulations*, as amended.

DOE Guide 424.X-X, *Implementation Guide for Use in Addressing Unreviewed Safety Question (USQ) Requirements*, U.S. Department of Energy, Washington, D.C.

DOE Guide 460.1-1, *The Packaging and Transportation Safety Implementation Guide*, U.S. Department of Energy, Washington, D.C.

DOE Order 460.1A, *Packaging and Transportation Safety*, U.S. Department of Energy, Washington, D.C.

8.2 WORKING REFERENCES

DOE/RL-2001-0036, *Hanford Sitewide Transportation Safety Document*, Fluor Hanford, Richland, Washington.

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Attachment 1: Responsibilities

TSD Configuration Manager

1. Ensure that PISA notifications are appropriately distributed to the contractor T&P POCs.
2. Ensure that USQ_T evaluations are appropriately distributed to the contractor T&P POCs.
3. Maintain configuration control of the TSD.
4. Prepare the annual TSD update incorporating the contractor's annual USQ report.
5. Maintain the safety basis current as required by DOE Order 460.1A and 10 CFR 830.

Contractor T&P Manager – is the functional manager for T&P and acts as the contractor T&P POC. May be the contractor T&P activity manager.

1. May establish their facility SRBs (membership, quorum, etc.).
2. Assign personnel as USQ_T evaluators, taking into consideration educational background, overall work experience, work experience in transportation, current work scope, and completion of all required training. The USQ_T evaluator designation and justification of personnel must be documented. At a minimum, those personnel designated as design authorities and cognizant individuals assigned to apply categorical exclusions must be designated as USQ_T evaluators.
3. Implement the requirements for training and retraining USQ_T evaluators as specified in Section 4.3 of this procedure.
4. Prepare and submit the annual USQ report to the TSD configuration manager.
5. Ensure USQ_TS and evaluation files (including SRB meeting minutes) are maintained within the project as quality assurance documents.
6. Qualify and designate USQ_T evaluators.
7. Qualify and designate members of the SRB if applicable.

Contractor T&P Activity Manager – is the line manager responsible for T&P activities (e.g., project manager).

1. Ensure that USQ_TS and evaluations are appropriately conducted, documented, and distributed.
2. Before implementing a proposed action, obtain RL approval for situations determined to involve an USQ_T or otherwise requiring RL approval.
3. Take actions for a PISA in accordance with Sections 4.2 and 5.2 of this procedure.

USQ Evaluators

1. Screen and/or evaluate proposed changes, tests, experiments, and occurrences against the current safety basis to determine if a potential USQ_T exists.
2. Ensure that the contractor's T&P manager allows for the use of USQ_TS to rule out the existence of an USQ_T for physical modifications prior to performing such modifications.
3. Obtain peer review by a second USQ (reviewer) of the work performed in the determination. The USQ_T and/or determination must document the logic and defensible justification for determining whether or not an USQ_T exists.
4. Refer to the transportation manager those USQ_TD when an USQ_T cannot be ruled out.
5. Ensure qualification (training and transportation manager designation) prior to performing any USQ_T function.

Safety Review Board (SRB)

1. Review USQ_TD for referred proposed changes, tests, experiments, and occurrences and advise the T&P activity manager.

Note: The SRB cannot overturn a determination, but may request a reevaluation if further information is identified.

2. Approve JCO before obtaining further approvals.
3. Approve categorical exclusions for items that are to be exempted prior to submission to RL for approval.

Attachment 2: Definitions and Acronyms

Categorical exclusion means a screening method that defines a category or categories of items or procedures that may be excluded from the USQ_T. The justification for a categorical exclusion must define the conditions required for the exclusion to be valid. Categorical exclusions must be approved by the SRB and RL before they become effective. A listing of categorical exclusions is contained in Chapter VI of the TSD.

Change, as used in the USQ_T, means a change is created by any of the following conditions:

1. A physical modification (permanent or temporary) to a transportation activity as described in the documented safety analysis, or a revision to a transportation activity procedure (permanent or temporary), or other policies or procedures as described in the documented safety analysis;
2. Introduction of new tests or experiments; and/or
3. A TSR basis revision.

The change must have the credible potential to impact the RL-approved documented safety analysis in order for it to be considered a change that requires consideration in the USQ_T.

Documented safety analysis means a documented analysis of the extent to which a packaging system or transportation activity can be operated safely with respect to workers, the public, and the environment, including a description of the conditions, safe boundaries, and hazard controls that provide the basis for ensuring safety. For transportation, this is the TSD.

Hazard controls means measures to eliminate, limit, or mitigate hazards to workers, the public, or the environment that are derived from the safety analyses, including:

1. Physical, design, structural, and engineering features;
2. Safety structures, systems, and components;
3. Safety management programs;
4. TSR; and
5. Other controls necessary to provide adequate protection from hazards.

Occurrence means an event (e.g., package rupture, valve failure, loss of containment, environmental spills) or a condition (e.g., an as-found state, whether or not resulting from an event), that may have adverse safety, health, quality assurance, security, operational or environmental implications.

Occurrence report means a documented evaluation of an event or condition that is prepared in sufficient detail to enable the reader to assess its significance, consequences, or implications and to evaluate the actions being proposed or employed to correct the condition or to avoid recurrence. All occurrence reports are screened for an USQ_T in accordance with this procedure.

Package Specific Safety Document (PSSD) is a safety document prepared to demonstrate the safety associated with a specific package. For onsite packages, the PSSDs are evaluations that demonstrate how the criteria of the TSD are met. PSSDs include existing RL approved onsite safety analysis report for packaging (SARP), safety evaluation for packaging (SEP), Certificate of Compliance (CoC) or Certificate of Competent Authority for U.S. Nuclear Regulatory Commission (NRC), U.S. Department of Transportation (DOT), U.S. Department of Defense, and DOE packaging systems.

Potential Inadequacy in the Safety Analyses (PISA) means situations of concern wherein T&P activity management has information that gives reason to believe that there is the potential that the current safety analyses may not be bounding or the approved safety basis may be otherwise inadequate.

Safety Basis means the documented safety analysis and hazard controls that provide reasonable assurance that a DOE T&P activity can be conducted safely in a manner that adequately protects workers, the public, and the environment. [Safety basis replaces authorization basis.]

Transportation Safety Document. The TSD process identified in DOE O 460.1A and DOE Guide 460.1-1 provides a safe harbor approach to the requirements in 10 CFR 830.204 Documented Safety Analysis. The TSD provides the methodology and compliance process to meet equivalent safety for any deviation from DOT regulations. The TSD provides bounding conditions based on minimum packaging standards and existing onsite SARP or SEP, and includes NRC, DOT, and RL-approved packaging systems. The TSD defines the requirements for the USQ_T and the TSR identification process.

Technical Safety Requirement (TSR). For the purposes of this procedure, TSRs are also known as operational safety requirements, interim operational safety requirements, and technical specifications and standards as identified in the facility specific safety basis.

Acronyms

CFR	<i>Code of Federal Regulations</i>
CoC	certificate of compliance
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
IG	Implementation Guide (DOE G 424.X-X)
JCO	justification for continued operations
NRC	U.S. Nuclear Regulatory Commission
PISA	potential inadequacy in the safety analyses
PSSD	package specific safety document
SARP	safety analysis report for packaging
SEP	safety evaluation for packaging
SRB	safety review board
USQ _T	unreviewed safety question process for transportation
USQ _T D	unreviewed safety question process for transportation determination

USQ_TS unreviewed safety question process for transportation screening
TSR technical safety requirement

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Attachment 3: USQ_TS Forms (Screening)**USQ_TS WORKSHEET****IDENTIFICATION**

PACKAGING SYSTEM NAME OR TRANSPORTATION ACTIVITY OR PROCEDURE NUMBER:
CHANGE TITLE OR PROCEDURE TITLE:
CHANGE NUMBER:
USQ _T SCREENING NUMBER:

RESULTS SUMMARY

BASED ON THE INFORMATION DEVELOPED IN THE FOLLOWING WORKSHEET: ☐ This issue was processed as an USQ_TS ☐ The determination was Negative (an USQ _T D IS NOT required) ☐ The determination was Positive (an USQ _T D IS required)

PREPARATION, REVIEW, AND APPROVAL SIGNATURES

USQ _T S First Evaluator's Name (Printed or Typed):	
USQ _T S First Evaluator's Signature:	Date:
USQ _T S Second Evaluator's Name (Printed or Typed):	
USQ _T S Second Evaluator's Signature:	Date:

List all documents directly related to this activity, or procedure that were reviewed (e.g., TSD, PSSD, Package Specific Evaluation, Categorical Exclusion, etc.):

1. Does the proposed activity or occurrence represent a temporary or permanent change to the TSD or procedures as described in the TSD?

☐ Yes ☐ No

WHY (explain the technical basis for the answer):

2. Does the proposed activity or occurrence represent conditions that have not been analyzed in the TSD?

☐ Yes ☐ No

WHY (explain the technical basis for the answer):

3. Does the proposed activity or occurrence represent a test or experiment NOT described in the TSD that may affect the safe operation of the facility?

☐ Yes ☐ No

WHY (explain the technical basis for the answer):

Attachment 4: Screening Guidance

The following are suggested screening questions:

1. Does the proposed activity or occurrence represent a temporary or permanent change to the facility or procedures as described in the documented safety analysis?
2. Does the proposed activity or occurrence represent conditions that have not been analyzed in the documented safety analysis?
3. Does the proposed activity or occurrence represent a test or experiment NOT described in the documented safety analysis that may affect the safe operation of the facility?

The following guidance should be considered when completing an USQ_{TS}. This guidance should not be considered all-inclusive; additional factors may need to be considered depending on the nature of the proposed activity.

Does the proposed activity accomplish the following, or is the occurrence the result of a prior action that did:

1. Alter the design of a transportation activity as described in the safety basis?
2. Modify, add, or delete the description of operation, operating environment, or analyses of any system or component described in the safety basis?
3. Modify, add, delete, or conflict with any of the design bases stated in the safety basis?
4. Conflict with the principle or general design criteria stated in the safety basis?
5. Modify, add, or delete a flow diagram or transportation activity drawing provided in the safety basis?

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Attachment 5: USQ_TD Forms (Determination)**USQ_TD WORKSHEET****IDENTIFICATION**

PACKAGING SYSTEM NAME OR TRANSPORTATION ACTIVITY:
CHANGE TITLE:
CHANGE NUMBER:
USQ _T DETERMINATION NUMBER:

RESULTS SUMMARY

BASED ON THE INFORMATION DEVELOPED IN THE FOLLOWING THE WORKSHEET: ☐ This issue was processed as a routine USQ_TD (that is, a proposed change) ☐ The determination was Negative (USQ_TD IS NOT involved with this change) ☐ The determination was Positive (an USQ_TD IS involved with this change) ☐ This issue was processed as a special PISA USQ_TD ☐ The determination was Negative (an actual inadequacy of the safety analyses DOES NOT exist) ☐ The determination was Positive (an actual inadequacy of the safety analyses DOES exist)

PREPARATION, REVIEW, AND APPROVAL SIGNATURES

USQ _T D First Evaluator's Name (Printed or Typed):	
USQ _T D First Evaluator's Signature:	Date:
USQ _T D Second Evaluator's Name (Printed or Typed):	
USQ _T D Second Evaluator's Signature:	Date:

List all documents directly related to this activity, change or PISA that were reviewed (e.g., TSD, PSSD, Package Specific Evaluation, Categorical Exclusion, etc.):

Has a payload specific evaluation been conducted? If not, one must be conducted before proceeding with this USQ_TD (format and required contents are found in Appendix H).

1. Could the proposed design modification, activity, change, or issue increase the probability of occurrence or consequences of an accident or malfunction previously evaluated in the TSD?	☐ Yes ☐ No

WHY (explain the technical basis for the answer):

2. Could the proposed design modification, activity, change, or issue create the possibility of an accident or malfunction different from any previously evaluated in the TSD?	☐ Yes ☐ No

WHY (explain the technical basis for the answer):

3. Does the proposed design modification, activity, change, or issue reduce the margin of safety as defined in the TSD?	☐ Yes ☐ No

WHY (explain the technical basis for the answer):

Attachment 6: USQ_TD Guidance

A safety analysis (payload specific evaluation) is required at a level adequate to show the safety of the change prior to conducting the USQ_TD. This attachment provides guidance for answering the questions identified in the USQ_TD form. If any of these questions are answered "yes", the proposed activity or issue is considered to constitute an USQ_T.

Consistent with the intent of the USQ_T, the following provides general guidance for each question on the USQ_TD form.

1. For each determination question provide a "Yes" or "No" response.
2. Review the payload specific evaluation before conducting the USQ_TD.
3. Provide a separate basis for each question to justify the answer to the question.
 - A restatement of the question does not constitute a satisfactory justification or basis.
 - An adequate justification provides sufficient explanation such that an independent reviewer could reach the same conclusion.
4. State all references used to support the questions/basis conclusions. References may include applicable sections of the TSD identified in the description section, documents directly referenced (first generation) in TSD, and any other applicable technical and safety documentation.

QUESTIONS:

1. **Could the proposed design modification, activity, change, or issue increase the probability of occurrence or consequences of an accident or malfunction previously evaluated in the TSD?**

In answering this question, the first step is to determine the accidents that may be affected by the proposed activity, which have been evaluated in the previously approved safety basis. By focusing on the initiators of the previously evaluated accidents, an evaluation is made as to whether there is an increased likelihood that a given accident would occur. The following questions may provide a useful approach in making this evaluation.

- a. Will the proposed activity meet the design, material, and construction standards applicable to the packaging system being modified? If the answer is "yes," this aspect of the proposed activity is judged not to increase the likelihood of the occurrence of an accident. If the answer is "no" to any of the items, either a justification for saying there is no increase in the likelihood of the occurrence of an accident will need to be developed or it is concluded that the likelihood of the occurrence of an accident is increased.

- b. Does the proposed activity affect overall packaging system performance in a manner that could increase the probability of a previously analyzed accident? Possible questions to ask are:
- 1) Does the proposed activity use instrumentation with accuracies or response characteristics that are different from those of existing instrumentation such that an accident is more likely to occur?
 - 2) Does the proposed activity cause packaging systems to be operated outside their design or testing limits?
 - 3) Does the proposed activity cause system vibration, fatigue, corrosion, thermal cycling, or degradation of the environment for packaging systems that would exceed the design limits?
 - 4) Does the proposed activity cause a change to any packaging system interface in a way that could increase the likelihood of an accident?

When evaluating for "increases in consequences" of an accident, if the previously bounding case for that family of accidents remains the bounding case, then generally there is no increase in the consequences within the USQ_T. In this regard, it is important the family of accidents be related, in addition to being of the same type (e.g., fires), but also use the same set of preventative and mitigative measures.

In answering this question, the first step is to determine which accidents evaluated in the safety analyses may have their radiological and hazardous material consequences altered as a direct result of the activity. The next step is to determine whether the activity does, in fact, increase the consequences of any of the accidents evaluated in the existing safety analyses. It is important to note that consequences to workers (in-facility and outside, or co-located), as well as to the public, are considered. Examples of questions that assist in this evaluation are:

- a. Does the proposed activity degrade or prevent safety functions described or assumed in the existing safety analyses?
- b. Does the proposed activity alter any assumptions previously made in evaluating the radiological and hazardous material consequences in the existing safety analyses?
- c. Does the proposed activity play a direct role in mitigating the radiological or hazardous material consequences assumed in the existing safety analyses?
- d. Does the proposed activity affect the integrity or function of any fission product barrier or any radioactive or hazardous material barriers?

2. Could the proposed design modification, activity, change, or issue create the possibility of an accident or malfunction different from any previously evaluated in the TSD?

In answering this question, the first step is to determine the types of accidents evaluated in the existing safety analyses. The types of credible accidents that the activity could create can then be identified and listed. Evaluating the differences between the two lists will determine the answer to the question. The accidents evaluated in the existing safety analyses are generally chosen to be bounding for a broad class of credible accidents. Thus, comparison of a new accident to the existing analyses may require referral to the underlying hazard analyses.

An accident or malfunction that involves an initiator or failure not considered in the existing safety analyses is potentially an accident or malfunction of a different type.

To answer this question, the types of failure modes of packaging systems that have been previously evaluated in the existing safety analyses and that would be affected by the activity are identified. Then the types of failure modes that the activity could create need to be identified. Comparing the two lists can provide an answer to the question.

3. Does the proposed design modification, activity, change, or issue reduce the margin of safety as defined in the TSD?

To answer this question, first determine whether a margin of safety is involved. To do this, the basis for each proposed activity should be identified. If a margin of safety is defined in the safety basis, a margin of safety is involved and the effects of the activity on it should be assessed.

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Attachment 7: Guidance for Justifications for Continued Operations (JCO)

When Continuing Operations is Justifiable

In cases in which the USQ_T evaluation of an event or condition leads to the determination of an USQ_T, the T&P activity manager has the inherent responsibility for ensuring the plant is in a known-safe status and within the safety basis if possible. In some cases, this may require T&P activity shutdown.

However, there are cases in which safe facility operation can be ensured despite the existence of an USQ_T. In addition, there are facilities (e.g., underground storage tanks) for which the concept and implications of a shutdown are nebulous.

In some of these cases, shutdown should be interpreted as cessation of all activities except for those that clearly could not result in events or risk that have not been accepted by RL in the safety basis. Some or all of the secured activities might later be shown to be appropriate, and therefore resumed with proper precautions and RL approval.

What a JCO Shows

The many different situations that may lead to a JCO make it impossible to provide detailed instructions for preparing the document itself. However, in each case in which safe interim operations may be requested, the objective of the justification is to document the analysis supporting the conclusion that safe operations may continue.

The justification should convincingly demonstrate that risk acceptance by RL can be recommended or that the increase in risk has sufficient quantification for RL to evaluate for acceptance. Otherwise, the JCO should show the incremental risk of continued operation or conduct of the activity is small compared to the risk previously accepted in the safety basis.

Use of JCO

The specific purpose of a JCO is to document the bases for continuing to operate facilities before the ultimate resolution of the safety issue associated with an USQ_T or to permit resumption of specific activities suspended when the USQ_T safety question was determined. The JCO amends the safety basis (upon RL approval) until it is terminated.

PREPARING JCO

Address the Safety Question

The declaration of a PISA USQ_T means the facility or operation is outside the facility-identified safety basis. The JCO should convincingly demonstrate the proposed activity (continuation or

resumption of an operation or activity) is consistent with the requirement that all activities and operations represent acceptable public and onsite safety risk.

CONTENT OF A JCO

The JCO content should include the following:

1. A description of the issue determined to involve a USQ_T.
2. The interim controls and/or restrictions based on the best available information, analysis, and engineering judgment; including any restriction on the use of the JCO to authorize additional work or activity.
3. An estimate of the risk associated with the interim operation.
4. A description of the margins of safety to be maintained.
5. A description of how existing controls/restrictions maintain safe conditions.
6. A description of the additional controls needed and how they ensure safe conditions.
7. The date or event that will permit closure or termination of the JCO. The JCO may remain in effect until the safety basis has been updated and approved by RL, or the discovered condition that initiated the USQ_T and JCO has been eliminated.

Show that Overall Risk is Acceptable

For example, if a failure to satisfy seismic design requirements (generally established on the basis of many years of facility operation) is discovered, an USQ_T would be determined if the PISA significantly increases the accident consequences or chance of occurring. But, continued operations for a short period before correcting the deficiency (hours, days, perhaps even months) may well be acceptable if the chance of some minimum-severity seismic event can be shown to be adequately remote. The justification is that the period of exposure to an event is very low and so the risk (probability times consequence) is also very low, and is therefore acceptable to RL.

Place Limits on the Operations

It may be possible to bound (restrict) the operations so that the increased likelihood of an accident is acceptable. Process limitations can be imposed to regain the previously acceptable low-accident consequence or probability.

For example, a reduction in allowed radioactive material inventory may provide an acceptable basis for continued operation, even if an exhaust filter efficiency was inaccurately used in the safety analyses. Here the justification is that operating restrictions can be used to make the

design basis accident less severe compared to as analyzed in the final safety analysis (based on the worst-case initial conditions for the accident).

Additional Administrative Controls

Special administrative steps can sometimes be specified to avert a potential hazard. For example, continuous operator surveillance for a short period may be shown to be an adequate substitute for an annunciator assumed to trigger operator response in the final documented safety analysis. Here the justification is that the critical function (alerting the operator) can be maintained in the presence of the failed component.

Additional Analyses

Reanalyzing the accident may show a capability for safe operation when credit is taken for design features that were not used in the original analysis of an accident. These features may be shown to equal or even outweigh the effects of a discovered deficiency. Here, the justification is made by showing operation is within the safety basis. Credit may also be taken for current measured parameters rather than the conservatively assumed values in the safety analyses, e.g., measured vs. assumed concrete strength.

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Attachment 8: Payload Specific Evaluation Format and Contents

1.0 PURPOSE

A payload-specific evaluation is performed for a specific package or campaign when the requirements and safety basis for the shipment are exceeded or changed. Examples of when to conduct an evaluation include an unauthorized source term, an increase in payload weight, the addition of hydrogenous material affecting gas generation, an unevaluated transport vehicle, and a change to the package configuration.

2.0 JUSTIFICATION

A detailed justification shall be prepared explaining why the DOT or the Hanford Site equivalent packaging requirements have not been met. The justification may include discussions involving limiting circumstances such as costs, technical information, operational requirements, and programmatic needs.

3.0 PACKAGE EVALUATIONS

Each safety basis evaluation topic listed below will be reviewed for possible changes affecting the stated requirements because of the change being evaluated. The examples listed are not meant to be a complete listing, but rather typical aspects relative to payload-specific evaluations.

3.1 CONTENTS

A review of the contents evaluation shall include changes to the radioactive isotopes, fissile material, physical form (solid, liquid, gas, powder, fixed contamination), absorbents, and hydrogenous material. The contents evaluation performed for the specific payload shall be compared to the requirements in Chapter VI, Section 5.0 of the TSD.

3.2 CONTAINMENT

A review of the containment evaluation shall include changes to the contents that could affect a leak rate calculation. The containment evaluation results performed for the specific payload shall be compared to the requirements in Chapter VI, Section 5.0 of the TSD.

3.3 RISK

A review of the risk evaluation for risk-based packages shall include changes affecting the dose consequence evaluation or accident frequency evaluation.

3.3.1 Dose Consequence

The dose consequence evaluation may be affected by changes to the contents such as increasing the payload limits, physical form of the payload, changes to the packaging, or changes to the location of the package. This evaluation shall discuss the payload-specific dose consequences utilizing the methodology identified in Chapter VII and compare them to the requirements in Chapter VI, Section 5.0 of the TSD.

3.3.2 Accidents

The accident frequency evaluation may be affected by changes to the number of shipments, mileage, administrative controls, and gross weight of the vehicle. This evaluation shall discuss the payload-specific impacts to any previously analyzed frequency calculations utilizing the methodology in Chapter VI, Section 5.0 of the TSD.

3.4 SHIELDING

A review of the shielding evaluation shall include changes to the contents, packaging, or vehicle. The shielding evaluation results performed for the specific payload shall be compared to the requirements in Chapter VI, Section 5.0 of the TSD.

3.5 CRITICALITY

A review of the criticality evaluation shall include changes to the fissile material contents, packaging, number of packages per vehicle, or proximity to bodies of water. The criticality evaluation results performed for the specific payload shall be compared to the requirements in Chapter VI, Section 5.0 of the TSD.

3.6 STRUCTURAL

A review of the structural evaluation shall include changes to the packaging, lift fixtures, payload weight, packages per vehicle, allowable transport vehicle. Furthermore, changes to the contents that could affect the thermal and gas generation evaluations may also affect the material properties and internal pressure. The structural evaluation results performed for the specific payload shall be compared to the requirements in Chapter VI, Section 5.0 of the TSD.

3.7 THERMAL

A review of the thermal evaluation shall include changes to the contents and the package heat loading, and possibly a change to the receiving facility's requirements. The thermal evaluation results performed for the specific payload shall be compared to the requirements in Chapter VI, Section 5.0 of the TSD.

3.8 GAS GENERATION

A review of the gas generation evaluation shall include changes to the contents, allowable shipping window, or payload configuration. Items that directly affect the radiolytic production of hydrogen gas are the radioactive contents, hydrogenous material, "G" value, and void space. The gas generation evaluation results performed for the specific payload shall be compared to the requirements in Chapter VI, Section 5.0 of the TSD.

3.9 TIEDOWN

A review of the tiedown evaluation shall include changes to the package weight, transport vehicle, and number of packages. The tiedown evaluation results performed for the specific payload shall be compared to the requirements in Chapter VI, Section 5.0 of the TSD.

4.0 ADDITIONAL CONTROLS

Based on the results of the package-specific evaluation, additional controls may be placed on the transport of the package.

4.1 ADMINISTRATIVE CONTROLS

The addition of an administrative control must be documented in the applicable procedures and/or transportation requirement. An example of an administrative control would be limiting the speed of the transport vehicle such that the maximum speed is less than the failure threshold of the package.

4.2 ENGINEERING CONTROLS

The addition of an engineering control must be documented in the applicable procedure and or transportation requirement. An example of an engineering control would be to vent the package through a filter rather than sealed to extend the shipping window.

5.0 CONCLUSION

For a payload-specific evaluation, it is necessary to provide a clear justification explaining why the DOT or Hanford Site equivalent packaging requirements were not met and perform the affected safety basis evaluations. A complete review must be performed for all safety basis evaluations supporting the shipment following any payload specific changes to verify full compliance with all requirements.

Attachment 9: Guidance for Annual Summary Submittals

Summarize each USQ_TD performed since the last annual summary and provide the following:

1. Identifying number of the evaluation and/or change.
2. Title of the evaluation.
3. Date of the evaluation.
4. Summary of the change and any pertinent basis information.
5. Results of the evaluation.

Note: Any positive USQ_T evaluations should have already been resolved separately from the annual submittal.

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APPENDIX D

TECHNICAL SAFETY REQUIREMENTS FOR HANFORD ONSITE T&P

APPENDIX D

TECHNICAL SAFETY REQUIREMENTS FOR HANFORD ONSITE T&P

PREFACE

The technical safety requirements (TSR) for Hanford Onsite Transportation and Packaging (T&P) define the administrative controls and limits for the T&P of radioactive materials and waste in quantities greater than 1 A₂. These controls and limits ensure that the T&P of hazardous materials are completed in a safe manner and provide protection to the environment, Hanford Site personnel, and the public. Controls required for public safety, significant worker safety, and for maintaining radiological consequences below the limits and guidelines are included.

In the T&P program, packaging system design is passive and the TSRs are administrative. The TSRs are based on the preventive and mitigative features determined to be required in the body of this *Hanford Sitewide Transportation Safety Document* (TSD). The TSRs constitute an agreement or contract between the U.S. Department of Energy (DOE), Richland Operations Office (RL) and Hanford Site contractors regarding the safe packaging and transportation of hazardous materials on the Hanford Site. As such, the TSRs cannot be changed without the approval of RL.

The format and content of the TSRs are based on (1) DOE O 5480.22, *Technical Safety Requirements*; (2) the guidance provided by contractor policy; (3) NUREG 1431, *Standard Technical Specifications, Westinghouse Plants*; (4) HNF-SD-MP-TSR-001, *TSR Writers' Guide*; (5) DOE G 423.X, *Implementation Guide for Use in Developing Technical Safety Requirements*. The TSRs are maintained as part of this TSD, which is a controlled document.

The TSRs do not apply to offsite shipments of hazardous materials. Offsite shipments are done in full compliance to U.S. Department of Transportation (DOT) requirements as found in 49 *Code of Federal Regulations* (CFR) and 10 CFR.

Protection of occupational workers from radiological hazards as a result of the packaging and transportation of radioactive materials is achieved by compliance with this TSD and the TSRs contained herein.

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List of Terms

CFR	<i>Code of Federal Regulations</i>
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
LCO	limiting conditions for operation
LCS	limited control settings
NUREG	U.S. Nuclear Regulatory Commission Regulatory Guide
ORP	Office of River Protection
PSSD	package specific safety document
RL	Richland Operations Office
SL	safety limit
SR	surveillance requirement
T&P	transportation and packaging
TSD	Hanford Site-wide Transportation Safety Document
TSR	technical safety requirement

SECTION 1

USE AND APPLICATION

Section 1 USE AND APPLICATION

1.1 Definitions

-----NOTE-----

The defined terms of this section are unique definitions. They appear in CAPITALIZED type and are applicable throughout these technical safety requirements (TSRs) and BASES. Some terms in this section refer the user to another section for the definition. This approach will prevent a shortened definition from being supplied and used out of context.

<u>Term</u>	<u>Definition</u>
ACTIONS	ACTIONS shall be that part of the limiting conditions for operation (LCO) that prescribes required actions to be taken under designated conditions within specified completion times.
ADMINISTRATIVE CONTROL	Refer to Section 1.2, ADMINISTRATIVE CONTROLS.
BASES	Refer to Attachment 1, BASES.
DESIGN FEATURES	DESIGN FEATURES are those design characteristics, primarily passive in nature, that are of a special importance to maintaining adequate control, shielding, or containment of radiological or toxicological material for which indiscriminate changes are to be prevented. See Attachment 2, "DESIGN FEATURES," for DESIGN FEATURES criteria.
VIOLATION	Refer to Section 1.5, Technical Safety Requirement VIOLATIONS.

Section 1 USE AND APPLICATION

1.2 Administrative Controls

ADMINISTRATIVE CONTROLS are the provisions relating to organization and management, procedures, recordkeeping, reviews, audits, and specific program requirements for risk reduction necessary to ensure safe operation of the facility. The TSRs (i.e., safety limit [SL], limited control setting [LCS], LCO, and ADMINISTRATIVE CONTROLS) establish administrative requirements that ensure TSRs are met in the operation of the facility and the procedures that are followed should a TSR not be met. ADMINISTRATIVE CONTROLS are generally written at the program level and contain program key elements, as applicable. ADMINISTRATIVE CONTROLS are established if (1) a safety function is best satisfied by a program instead of a hardware system; (2) control of a condition is not measured in real-time or near to real-time; (3) control of a condition is not under the immediate control of the operator; (4) a condition does not require immediate action and sufficient recovery time exists to permit mitigating action; or (5) a condition requires an evaluation based on prevalent conditions. ADMINISTRATIVE CONTROLS do not require ACTION statements or surveillance requirements (SR). SRs necessary to demonstrate compliance with an ADMINISTRATIVE CONTROL and the actions taken, should an ADMINISTRATIVE CONTROL requirement not be met, are performed according to procedures.

ADMINISTRATIVE CONTROLS are established for those programs required to prevent or mitigate accidents with unacceptable radiological consequences to the offsite public or onsite worker, to provide significant defense in depth, or provide significant worker safety.

1.3 SL, LCO, and Surveillance

For T&P activities, the packaging system design is passive. No SLs are expected for transportation activities because there are no processes or activities in which the operator causes a process variable to be manipulated, which if left unchecked or uncontrolled would result in catastrophic failure of a passive safety barrier. For example, there are no operator-initiated processes to increase temperature, pressure, electrical, or mechanical insult to the cargo that could lead to catastrophic failure. Because there are no SLs or LCOs, surveillances are not required.

1.4 Alternate Emergency Actions

In an emergency, if a situation develops that is not addressed by the TSRs, staff members are expected to use their training and expertise to take actions to correct or mitigate the situation. Also, staff members may take actions that depart from a requirement in the TSRs provided that:

- An emergency situation exists;
- These actions are needed immediately to protect the worker and public health and safety; and
- No ACTION consistent with the TSR can provide adequate or equivalent protection.

Such actions shall be approved, as a minimum, by the contractor's T&P approval authority or the RL Manager. If emergency actions are taken, verbal notifications shall be made to RL within 2 hours and by written reports to Headquarters within 24 hours. (DOE 5480.22, paragraph 9.i, and Attachment 1, paragraph 3.)

1.5 Technical Safety Requirements VIOLATION

SLs, LCSs, LCOs, or SRs are not associated with the T&P program; therefore, there are no ACTION statements. Accordingly, a VIOLATION of a TSR occurs as a result of failure to comply with an ADMINISTRATIVE CONTROL.

A VIOLATION of an AC occurs when a specific requirement in the AC is not met. A VIOLATION of an AC also occurs when discrepancies in a program or procedure non-compliances indicate a programmatic breakdown, which occurs if they are significant enough to render the safety analysis assumptions invalid. Individual programmatic discrepancies or procedure non-compliances do not by themselves constitute a VIOLATION of an AC.

An AC VIOLATION shall require the following actions to restore AC compliance:

- (a) Suspend those activities that cannot be conducted safely in light of the AC VIOLATION;
 - (b) Notify DOE of the VIOLATION in accordance with Occurrence Reporting requirements;
 - (c) Prepare an Occurrence Report in accordance with Occurrence Reporting requirements; and
 - (d) Prepare and implement a Recovery Plan describing the steps leading to compliance with an AC.
-

1.6 Reporting Requirements

In the event a TSR VIOLATION occurs, make notifications and report the event to RL. Reporting requirements are contained in DOE 232.1, *Occurrence Reporting and Processing of Operations Information*.

Additionally, should a discrepancy between the facility and a TSR be discovered, it must be reported to RL in accordance with DOE 232.1 and an evaluation must be performed to determine if an unreviewed safety question exists as defined in 10 CFR 830.

SECTION 2

SAFETY LIMITS

2.0 SAFETY LIMITS

There are no SLs associated with the T&P program.

SECTION 3

LIMITING CONDITIONS FOR OPERATION AND LIMITED CONTROL SETTINGS

3.0 LIMITING CONDITIONS FOR OPERATION/LIMITED CONTROL SETTINGS

There are no LCO and LCS associated with the T&P program.

SECTION 4

SURVEILLANCE REQUIREMENTS

4.0 SURVEILLANCE REQUIREMENTS

There are no SRs associated with the T&P program.

SECTION 5

ADMINISTRATIVE CONTROLS

5.0 ADMINISTRATIVE CONTROLS

5.1 Purpose

- 5.1.1 The purpose of the ADMINISTRATIVE CONTROL is to state the provisions relating to organization and management, procedures, record keeping, reviews, audits, and specific program requirements for risk reduction necessary to ensure safe packaging and transportation of radioactive materials on the Hanford Site.

5.1.2 Applicability

These ADMINISTRATIVE CONTROLS apply to the activities associated with the T&P of radioactive materials on the Hanford Site in quantities exceeding 1 A₂.

5.0 ADMINISTRATIVE CONTROLS

5.2 Onsite Shipments and Transfers

5.2.1 Each contractor must have a formal program in place and sufficient staffing to perform the actions identified in the TSD.

5.2.2 Applicability

Overall responsibility for this TSD is assigned to RL. RL, through contract, shall identify those contractors and their roles in the performance of this TSD.

5.2.3 Contractor Responsibilities

The contractor shall address the following organizational components as addressed in Chapter IV of the TSD and Appendix C, Unreviewed Safety Question Process for Transportation:

- A T&P Functional Organization
 - Contractor specific policies and procedures
 - A document configuration control program (e.g., shipping papers, package specific safety documents, unreviewed safety questions, training and qualification records, etc.)
 - Assessment program
 - T&P manager responsibilities
 - T&P activity manager responsibilities
 - TSD configuration manager responsibilities
 - Unreviewed safety question process for transportation program
 - Training program
 - QA program.
-

5.0 ADMINISTRATIVE CONTROLS

5.3 Nuclear Criticality Safety

5.3.1 The contractor program (Section 5.2) must contain provisions for nuclear criticality safety that implements the criticality requirements defined in the TSD.

5.3.2 Background

Packaging is conducted under the originating facility safety analysis. The primary features of the criticality prevention conducted under transportation activities are the package array and spacing. Other features that could be utilized include over packing with neutron absorbers. A more detailed discussion is contained in Chapter VII, Section 3.1.2.6 of the TSD.

5.0 ADMINISTRATIVE CONTROLS

5.4 Gas Accumulation Minimization

- 5.4.1 The contractor program (Section 5.2) must contain provisions to identify gas accumulation controls in packaging to minimize the potential for loss of package integrity from over-pressurization, deflagration, or explosion during transportation.

5.4.2 Applicability

The gas accumulation minimization TSR is only applicable to risk-based packages classified as transuranic or transuranic mixed. Causes of gas generation other than transuranic radiolysis are covered by standard safety programs and are not part of this TSR.

5.4.3 Key Features

Methods commonly used to achieve the TSR function include those listed below. The specific method used must be documented and documentation maintained.

- Venting
 - Time limits of transport (shipping window)
 - Package inventory controls (e.g., hydrogenous material limits)
 - Chemical compatibility.
-

5.0 ADMINISTRATIVE CONTROLS

5.5 Package Damage Ratio Minimization for Risk-Based Packages

5.5.1 The contractor program (Section 5.2) must contain provisions to identify controls to minimize the potential for damage to packages from accidents during transportation activities.

5.5.2 Applicability

The package damage ratio minimization TSR is only applicable to risk-based packages.

5.5.3 Key Features

Methods commonly used to achieve the TSR function include those listed below. These features may be used singularly or in combination to minimize adverse impacts of transportation related accidents.

- Speed limits
 - Dunnage
 - Tiedown systems.
-

5.0 ADMINISTRATIVE CONTROLS

5.6 Surveillance and Maintenance Program

5.6.1 The contractor must implement a surveillance and maintenance program to ensure preservation of the key transportation safety functions provided by the package.

5.6.2 Applicability

This administrative control applies to all packaging identified in the TSD or that is in use for onsite transportation. Packages that are no longer used for transportation, but only for storage of material are exempted from these surveillance and maintenance program requirements.

5.6.3 Key Features

The program will:

1. Identify specific surveillance and maintenance requirements, applicable when a package is not currently in use for transportation, but has the potential for future transportation use, such that the package does not degrade and is readily available for use.
 2. Ensure the specific surveillance and maintenance requirements identified in the individual PSSDs are met during package use, and prior to initial use following periods when they were not in use to transport material.
 3. Assure that packages shall not be used for transportation if the surveillance and maintenance has lapsed or if the qualifications for transportation use have otherwise fallen into question unless the following steps have been completed:
 - a) A plan has been developed that outlines the actions that must be performed by the user to ensure package integrity during transportation; and
 - b) DOE approval of the plan has been obtained; and
 - c) The plan has been implemented prior to shipping the package.
 4. Have a process to request deviations from the surveillance and maintenance program requirements. Such deviations shall be submitted to RL for approval.
 5. Have a process to request removal of a package from the TSD, when the package is not in current use and will not be used in the future for transportation. Such requests shall be submitted to RL for approval.
-

5.0 ADMINISTRATIVE CONTROLS

5.7 Configuration Management Program

5.7.1 The contractor's transportation program must have a configuration management feature that ensures changes to packages are evaluated for their impact on all safety functions identified in the PSSD that are provided by the package design.

5.7.2 Applicability

This administrative control applies to all packaging identified in the TSD or is in use for onsite transportation. Packages that are no longer used for transportation, but only for storage of material are exempted from these surveillance and maintenance program requirements.

5.7.3 Key Features

The configuration management feature of the contractor's transportation program must address the following key features.

1. All changes to a package will be controlled through the configuration management process.
 2. All safety functions provided by the package design, as identified in the PSSD, will be considered prior to initiating a change.
 3. The configuration management process will interface with the Unreviewed Safety Question to identify the need for RL approval of package changes.
-

SECTION 6

REFERENCES

Section 6 REFERENCES

10 CFR 830, "Nuclear Safety Requirements," *Code of Federal Regulations*, as amended.

10 CFR 71, "Packaging and Transportation of Radioactive Material," *Code of Federal Regulations*, as amended.

DOE 232.1, *Occurrence Reporting and Processing of Operations Information*, U.S. Department of Energy, Washington, D.C.

DOE G 423.X, *Implementation Guide for Use in Developing Technical Safety Requirements*, approved for technical use, U.S. Department of Energy, Washington D.C.

DOE O 5480.22, *Technical Safety Requirements*, U.S. Department of Energy, Washington D.C.

HNF-SD-MP-TSR-001, *TSR Writers' Guide*, Fluor Hanford, Richland, Washington.

NUREG 1431, *Standard Technical Specifications, Westinghouse Plants*, U.S. Nuclear Regulatory Commission, Washington, D.C.

ATTACHMENT 1

BASES

Attachment 1 BASES

According to DOE O 5480.22, a BASES attachment is required to summarize the reasons for establishing SLs, LCOs, LCSs, and associated SRs. This attachment does not apply to the T&P program because there are no SLs, LCOs, LCSs, or SRs. BASES for the ADMINISTRATIVE CONTROLS of these TSRs are derived from this TSD.

ATTACHMENT 2
DESIGN FEATURES

Attachment 2 DESIGN FEATURES

DESIGN FEATURES are those features not covered elsewhere in the TSRs and that, if altered or modified, would have a significant effect on safety. There are no DESIGN FEATURES in the T&P program.

APPENDIX E
PACKAGE SAFETY ANALYSIS ASSESSMENT CONTENTS AND FORMAT

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APPENDIX E

PACKAGE SAFETY ANALYSIS ASSESSMENT CONTENTS AND FORMAT

This appendix provides guidance in formatting and preparing an onsite package safety analysis assessment (PSAA) to demonstrate the adequacy of the packaging in providing an equivalent degree of safety as required in the regulations. It also aids in ensuring the completeness of the information for timely review and approval. This PSAA is the principal document in which information and bases are provided for determining if the package provides an equivalent degree of safety to the regulations.

1.0 PACKAGE EVALUATION

1.1 GENERAL INFORMATION

This section will present an introduction and general package description.

1.2 INTRODUCTION

This section will include the use of the package, model number, number of packages, and any other pertinent introductory information.

1.3 EVALUATION SUMMARY AND CONCLUSIONS

Briefly summarize the results of each section of the document (e.g., structural, thermal). This section will be considered an executive summary of the evaluations performed. Any specific restrictions, payloads, etc., will be identified and explained.

1.4 REFERENCES

Provide section references here.

2.0 STRUCTURAL EVALUATION

2.1 INTRODUCTION

This section will identify, describe, discuss, and analyze the structural engineering design of the packaging, components, and systems.

2.2 STRUCTURAL EVALUATION OF THE PACKAGE

2.2.1 Structural Design and Features

This section will describe the package, materials, and fabrication methods; discuss the use of codes and/or standards; and specify any special design features.

2.2.2 Mechanical Properties of Materials

This section will evaluate the strength of the specified materials and other physical properties related to structural design of the packaging.

2.2.3 Chemical and Galvanic Reactions

This section will discuss potential reactions between packaging materials, contents, and other factors that may impact the integrity of the packaging.

2.2.4 Size of the Package and Cavity

2.2.5 Weights and Center of Gravity

This section will include the packaging gross weight, weight of the package with maximum payload, center of gravity of the package, and center of gravity of the fully loaded package.

2.2.6 Tamper-Indicating Feature

2.2.7 Positive Closure

This section will describe the closure system and why the package cannot be inadvertently opened.

2.2.8 Lifting and Tiedown Features

The section will describe all devices and attachments that can be used to lift the package or its lid, identify all devices that are a structural part of the package and can be used as tiedowns, and discuss the overall tiedown system.

2.2.9 Brittle Fracture

This section will include a discussion of the affects of brittle fracture on the fabricated packaging materials. ASME BPVC, Section VIII, Division 1 will be used to evaluate brittle fracture.

2.3 STRUCTURAL MODEL

This section will discuss model description and assumptions, and the code description and validation.

2.4 NORMAL CONDITIONS

2.4.1 Conditions to be Evaluated

This section will discuss specific tests that will be performed and/or analyzed (drop tests, penetration tests) for normal conditions.

2.4.2 Acceptance Criteria

This section will provide the allowable leak rates, stress categories, and failure modes.

2.4.3 Initial Conditions

2.4.3.1 Environmental Heat Loadings. This section will discuss the worst-case temperature with and without solar insolation for normal conditions.

2.4.3.2 Maximum Thermal and Pressure Stresses. This section will discuss the effects of thermal and pressure stresses on the components for normal conditions.

2.5 ACCIDENT CONDITIONS

2.5.1 Conditions to be Evaluated

This section will discuss specific tests that will be performed and/or analyzed (drop tests, penetration tests) for accident conditions.

2.5.2 Acceptance Criteria

This section will provide the allowable leak rates, stress categories, and failure modes.

2.5.3 Initial Conditions

2.5.3.1 Environmental Heat Loadings. This section will discuss the worst-case temperature with and without solar insolation for accident conditions.

2.5.3.2 Maximum Thermal and Pressure Stresses. This section will discuss effects of thermal and pressure stresses on the components for accident conditions.

2.6 PUNCTURE

As applicable to packagings, this section will discuss the affects of puncture.

2.7 STRUCTURAL EVALUATION AND CONCLUSIONS

This section will discuss the drop orientation with the most severe loading.

2.7.1 Normal Conditions Evaluation and Conclusions

This section will discuss all normal conditions drop test results.

2.7.2 Accident Conditions Evaluation and Conclusions

This section will discuss all accident conditions drop test results.

2.8 REFERENCES

Provide section references here.

2.9 APPENDICES

If required, provide section appendices here.

3.0 THERMAL EVALUATION

3.1 INTRODUCTION

This section will identify, describe, discuss, and analyze thermal engineering design of the packaging, components, and systems.

3.2 THERMAL SOURCE SPECIFICATION

3.3 SUMMARY OF THERMAL PROPERTIES OF MATERIALS

This section will list thermal properties of all materials used.

3.4 THERMAL EVALUATION FOR NORMAL CONDITIONS

3.4.1 Conditions to be Evaluated

This section will indicate specific tests that will be performed and/or analyzed (drop test, penetration tests) to support thermal evaluation of the package for normal conditions.

3.4.2 Acceptance Criteria

This section will discuss failure modes, thermal conditions, etc., as required to provide a basis for accepting thermal data from tests and/or analyses.

3.4.3 Thermal Model

This section will provide model description and assumptions, and code description and validation.

3.4.4 Thermal Analyses

3.4.4.1 Internal Temperature.

3.4.4.2 Maximum Surface Temperature.

3.5 THERMAL EVALUATION FOR ACCIDENT CONDITIONS

3.5.1 Conditions to be Evaluated

This section will indicate specific tests that will be performed and/or analyzed (drop test, fire tests) to support thermal evaluation of the package for accident conditions.

3.5.2 Acceptance Criteria

This section will discuss failure modes, thermal conditions, etc., as required to provide a basis for accepting thermal data from tests and/or analyses.

3.5.3 Package Conditions and Environment

3.5.4 Thermal Model

This section will provide model description and assumptions, and code description and validation.

3.5.5 Thermal Analyses

3.5.5.1 Internal Temperatures.

3.5.5.2 Maximum Surface Temperatures.

3.6 CONCLUSIONS

This section will include evaluations of designated tests and analyses performed to assure the package is thermally acceptable following designated normal conditions and accident conditions during transfer.

3.7 REFERENCES

Provide section references here.

3.8 APPENDICES

If required, provide section appendices here.

4.0 CONTAINMENT/CONFINEMENT EVALUATION

4.1 INTRODUCTION

This section will discuss the methodology used to evaluate containment and/or confinement performance of the package (e.g., package leakage or release rates).

4.2 CONTENTS INFORMATION

4.2.1 Characterization

This section will discuss content information, present specific characterization data, and provide evaluations for the content analysis data used to determine qualities of material, fissile content, and other data that are acceptable.

4.2.2 Restrictions

This section will provide a discussion of content restrictions and include information to assure that generic or specific content restrictions are sufficient.

4.2.3 Size and Weight

This section will evaluate size and weight of the contents to assure they are correctly listed and in accordance with the packaging size and structural load requirements.

4.2.4 Conclusions

This section will provide a conclusion to evaluations performed for the contents section.

4.3 CONTAINMENT SOURCE SPECIFICATION

This section will provide a detailed description of the contents of the package and assess dispersability of the material if required. A detailed listing of the contents will be provided in this section.

4.4 NORMAL CONDITIONS

4.4.1 Conditions to be Evaluated

This section will indicate the specific tests that will be performed and/or analyzed (e.g., drop test, penetration test) to support the evaluation of the package containment system for normal transfer conditions.

4.4.2 Containment Acceptance Criteria

This section will discuss allowable leak rates or dose consequences, failure modes, etc., as required to provide a basis for accepting data from tests or analyses. If the data are qualitative, the basis of evaluation must be justified.

4.4.3 Containment Model

This section will include the release rate or dose consequence model description, release assumptions, and code description and evaluation.

4.4.4 Containment Calculations

4.5 ACCIDENT CONDITIONS

4.5.1 Conditions to be Evaluated

This section will indicate specific tests that will be performed and/or analyzed (e.g., drop test, fire test) to support evaluation of the package containment system for accident conditions.

4.5.2 Containment Acceptance Criteria

This section will discuss the allowable leak rates or dose consequences, failure modes, etc., as required to provide a basis for accepting data from tests or analyses. If data are qualitative, the basis of evaluation must be justified.

4.5.3 Containment Model

This section will include release rate or dose consequence model descriptions, release assumptions, and code description and evaluation.

4.5.4 Containment Calculations

4.6 CONCLUSIONS

This section will contain a discussion on the containment system and determine if the package will safely contain contents within acceptable release or dose consequence limits for the designated normal conditions and accident conditions identified.

4.7 REFERENCES

Provide section references here.

4.8 APPENDICES

If required, provide section appendices here.

5.0 SHIELDING EVALUATION

5.1 INTRODUCTION

This section will identify, describe, discuss, and analyze shielding design of the packaging, components, and systems.

5.2 DIRECT RADIATION SOURCE SPECIFICATION

This section will state contents, and gamma and neutron source terms used in the analyses.

5.2.1 Gamma Source

This section will state the quantity of radioactive material included as contents. The method used to determine gamma source strength and distribution will be described in detail.

5.2.2 Beta Source

This section will state the quantity of radioactive material included as contents. The method used to determine beta source and distribution will be described in detail.

5.2.3 Neutron Source

This section will state the quantity of radioactive material included as contents. The method used to determine neutron source strength and distribution will be described in detail.

5.3 SUMMARY OF SHIELDING PROPERTIES OF MATERIALS

5.4 NORMAL CONDITIONS

5.4.1 Conditions to be Evaluated

This section will indicate specific tests that will be performed and/or analyzed (e.g., drop test, penetration test) to support evaluation of the package shielding system for normal conditions.

5.4.2 Acceptance Criteria

This section will include failure modes, dose rates, radiation levels, etc., as required to provide a basis for accepting data from tests and/or analyses.

5.4.3 Shielding Model

This section will include model description and assumptions, and code description and validation.

5.4.4 Shielding Calculations

5.5 ACCIDENT CONDITIONS

5.5.1 Conditions to be Evaluated

This section will indicate specific tests that will be performed and/or analyzed (e.g., drop test, fire test) to support evaluation of the package shielding system for accident conditions.

5.5.2 Acceptance Criteria

This section will include failure modes, dose rates, radiation levels, etc., as required to provide a basis for accepting data from tests and/or analyses.

5.5.3 Shielding Model

This section will include model description and assumptions, and code description and validation.

5.5.4 Shielding Calculations

5.6 CONCLUSIONS

This section will include evaluations for designated tests and analyses performed to verify integrity of the package shielding during designated normal conditions and accident conditions.

5.7 REFERENCES

Provide section references here.

5.8 APPENDICES

If required, provide section appendices here.

6.0 CRITICALITY EVALUATION

6.1 INTRODUCTION

This section will identify, describe, discuss, and analyze principal criticality engineering-physics design of the packaging, components, and system important to safety and necessary to comply with demonstration of package subcriticality.

6.2 CRITICALITY SOURCE SPECIFICATION

6.3 SUMMARY OF CRITICALITY PROPERTIES OF MATERIALS

6.4 NORMAL CONDITIONS

6.4.1 Conditions to be Evaluated

This section will indicate specific tests that will be performed and/or analyzed (e.g., drop test, penetration test) to support criticality evaluation of the package for normal conditions.

6.4.2 Acceptance Criteria

This section will include failure modes, criticality conditions, etc., as required to provide a basis for accepting criticality data from tests and/or analyses.

6.4.3 Criticality Model

This section will include the model description and assumptions, and code description and validation.

6.4.4 Criticality Calculations

6.5 ACCIDENT CONDITIONS

6.5.1 Conditions to be Evaluated

This section will indicate specific tests that will be performed and/or analyzed (e.g., drop test, fire test, etc.) to support criticality evaluation of the package for accident conditions.

6.5.2 Acceptance Criteria

This section will include failure modes, criticality conditions, etc., as required to provide a basis for accepting criticality data from tests and/or analyses.

6.5.3 Criticality Model

This section will include model description and assumptions, and code description and validation.

6.5.4 Criticality Calculations

6.6 CRITICAL BENCHMARK EXPERIMENTS

This section will provide justification for the validity of the calculational method and neutron cross-section values used in the analysis by presenting results of the calculations for selected critical benchmark experiments.

6.6.1 Benchmark Experiments and Applicability

This section will provide a general description of selected critical benchmark experiments that are to be analyzed. Applicability of the benchmarks in relation to the package and its contents will be shown.

6.6.2 Details of Benchmark Calculations

Actual nuclear and geometric input parameters used for benchmark calculations will be provided.

6.6.3 Results of Benchmark Calculations

Results of the benchmark calculations will be given.

6.7 CONCLUSIONS

This section will include evaluations of designated tests and analyses performed to assure radioactive contents of the package remains consistent with applicable limits and controls during designated normal conditions and accident conditions.

6.8 REFERENCES

Provide section references here.

6.9 APPENDICES

If required, provide section appendices here.

7.0 GAS GENERATION EVALUATION

7.1 GAS GENERATION

This section will include a discussion of hydrogen gas build-up and provide an evaluation of data to assure packaging design specifications are acceptable to prevent excessive hydrogen gas build-up. Consideration will include the use of venting and/or devices such as catalysts.

7.2 PACKAGE PRESSURE

This section will include a discussion of estimated internal package pressure and provide an evaluation of data to assure that packaging design specifications are acceptable to prevent excessive internal pressures. Consideration will include the use of pressure relief systems, vents, or other devices. It will be assured the package design is sufficient to safely contain the contents at the estimated operation pressure of the system during transfers.

7.3 REFERENCES

Provide section references here.

7.4 APPENDICES

If required, provide section appendices here.

8.0 TIEDOWN EVALUATION

8.1 SYSTEM DESIGN

This section will discuss the general tiedown system, evaluate tiedown system design drawings (if provided), and indicate restrictions and other concerns for safely securing the package to the vehicle.

8.2 ATTACHMENTS AND RATINGS

This section will provide an evaluation of load ratings for attachment devices on the vehicle and straps/chains or other devices used to tiedown the package.

8.3 REFERENCES

Provide section references here.

8.4 APPENDICES

If required, provide section appendices here.

9.0 PACKAGING QUALITY ASSURANCE

9.1 INTRODUCTION

This section is considered as a guideline for developing quality assurance plans and other documents to assure the packaging was designed and fabricated in accordance with identified codes, standards, and associated criteria.

9.2 QUALITY REQUIREMENTS

In this section, assigned safety class(s) and approval designators will be identified. In addition, specific guidance for development of quality assurance plans, inspection and maintenance plans, and training plans will be provided.

9.3 ORGANIZATION

In this section, information and charts to describe the quality assurance organization that must be in place to develop quality assurance plans, complete inspections, control changes, and maintain records will be provided.

9.4 REFERENCES

Provide section references here.

9.5 APPENDICES

If required, provide section appendices here.

10.0 OPERATING PROCEDURES

10.1 GENERAL REQUIREMENTS

This section will contain recommended guidelines for using the packaging. As necessary, specific operating procedures will be noted and included. Specific requirements that could adversely affect the integrity of the packaging should be directed to operations. Packagings that may be reused will have requirements discussed in sufficient detail to ensure the packaging is not damaged during transfer and unloading operations.

10.2 PACKAGE LOADING

This section will contain information on installing package contents, closure of package, and lifting of package.

10.3 PACKAGE UNLOADING

This section will contain information on lifting the package, opening package, and removing package contents.

10.4 EMPTY PACKAGE (PACKAGING)

This section will contain information on cleaning, inspecting, closing, handling, storage, and transfer.

10.5 REFERENCES

Provide section references here.

10.6 APPENDICES

If required, provide section appendices here.

11.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

11.1 NEW PACKAGING

11.1.1 Acceptance Requirements

This section will describe requirements that a new packaging must meet prior to approving the package for transport of contents. This section will include the extent of inspection required, specific requirements, etc.

11.1.2 Inspection and Testing

This section will indicate specific inspection and testing (if applicable) necessary to accept a new packaging for use, describe how the task should be performed, and criteria that should be used for accepting results of the inspection and testing.

11.1.3 Documentation

This section will indicate specific documents that will be developed to ensure packaging is acceptable, who will provide the documents, and where and how long they will be maintained.

11.2 PACKAGING FOR REUSE

11.2.1 Acceptance Requirements

The type of packaging and circumstances under which packaging will be reused will be considered. The requirements under which the packaging will be acceptable for reuse will be determined and identified. Requirements will include physical inspections, possible testing and/or retesting, new gaskets or other seals, painting, load tests, corrosion checks, etc.

11.2.2. Inspection and Testing

This section will contain specific inspections and testing (if applicable) to accept a package for reuse, describe how tasks should be performed, and criteria that should be used for accepting the results of inspection and testing.

11.2.3 Documentation

This section will contain specific documents that will be developed to ensure the packaging is acceptable, indicate who will provide the documents, and where and how they will be maintained.

11.3 GENERAL MAINTENANCE REQUIREMENTS

This section will discuss tasks that will be completed to ensure that a new or reused packaging is maintained in good condition for use. Requirements will include periodic inspections and testing for conditions such as corrosion and physical damage.

11.4 INSPECTION AND VERIFICATION SCHEDULES

This section will indicate specific inspection and testing, describe how tasks will be performed, and criteria that will be used for accepting the results of inspection and testing.

11.5 RECORDS AND DOCUMENTATION FOR MAINTENANCE

This section will indicate specific documents that will be developed to ensure the packaging is acceptable, who will provide the documents, and how long they will be maintained.

11.6 REFERENCES

Provide section references here.

11.7 APPENDICES

If required, provide section appendices here.

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APPENDIX F

EXAMPLE APPLICATION OF PACKAGE STRUCTURAL DESIGN AND EVALUATION REQUIREMENTS

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LIST OF TERMS

AC	accident condition
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BPVC	<i>Boiler and Pressure Vessel Code</i>
CY	Calendar Year
IAEA	International Atomic Energy Agency
MNOP	maximum normal operating pressure
NC	normal condition
NDE	nondestructive examination
NRC	U.S. Nuclear Regulatory Commission
NUREG	U.S. Nuclear Regulatory Commission regulation
PEEQ	equivalent plastic strain
QA	quality assurance
SED	strain energy density
TF	triaxiality factor
TSL	triaxial strain limit
UNC	(pg F-3)

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APPENDIX F

EXAMPLE APPLICATION OF PACKAGE STRUCTURAL
DESIGN AND EVALUATION REQUIREMENTS

1.0 INTRODUCTION

To demonstrate that economical Type B packages can be constructed to meet the Hanford Site minimum packaging standards, a typical generic package design is presented in this report. The most numerous Type B packages (with large A_2 quantities) in use on the Hanford Site are 210 L (55-gal) drums that are shielded with either lead or concrete. In this example, a typical lead shielded drum package has been designed and evaluated to the requirements of the minimum packaging standards as follows:

- The cost of such a drum in moderate quantities (5 to 20) would be approximately \$15,000 to \$17,000 each (Calendar Year [CY] 1999 dollars).
- The primary new feature of this design is that the containment vessel is designed and constructed to the requirements of American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPVC), Section VIII, Division 1 for a reasonably high design pressure of 1.4 MPa (200 psig) at a design temperature of 204 °C (400 °F).
- The inherent safety factor by the "Design by Rule" section of the ASME BPVC approach is 4 under normal operating pressures. This is to ensure structural integrity and maintenance of containment under normal conditions (NC) and the accident conditions (AC) free drop.
- The package is evaluated at a maximum normal operating pressure (MNOP) of 0.7 MPa (100 psig) and at an operating temperature of 93°C (200 °F) to requirements of ASME BPVC, Section III, Subsection NE, Service Level A, ensuring adequate design.
- The rationale for this design approach is the pressure energy retention capability of the design is approximately 1.1×10^5 ft-lb and the NC 1 m (4-ft) free drop energy is 1.2×10^3 ft-lb. Under AC, the 9 m (30-ft) free drop energy is 9×10^3 ft-lb, whereas the burst pressure energy capability of the design is 6.2×10^5 ft-lb. Consequently, the NC pressure energy retention capability and AC vessel burst pressure energy are nearly two orders of magnitude greater than the corresponding NC and AC free drop energies.

The main advantages of this type of design approach and the use of ASME BPVC, Division 1 as the design basis are:

- The "Design by Rule" method in the ASME BPVC provides quick, simple, and a nationally recognized method for design and construction of containment boundaries. This reduces engineering costs for package design.

- ASME BPVC, Division 1 is a commercial standard and is nationally recognized.
- Vessels designed by this approach provide a greater degree of safety than other packages previously used at the Hanford Site. This provides cost savings in developing and implementing administrative controls on shipments.
- Packaging designed and constructed by this approach can be easily demonstrated to meet minimum packaging standards. This provides a large cost savings in analyses, and in developing and defending the safety documentation. It also avoids costly safety basis design iterations and modifications.
- The standard defines all the welding, manufacturing processes, and testing requirements. This provides cost savings in developing engineering specifications.
- The quality assurance (QA) requirements for design and construction are already defined in the ASME BPVC and amendable to Hanford Site QA requirements.
- The design and construction of the package meets the intent of 10 CFR 71, Subpart H with regard to consistency in quality and control of design, construction, and nondestructive examination (NDE) processes.

2.0 GENERAL EVALUATION APPROACH

For the NC performance evaluations, the stress intensities developed from the MNOP evaluation are combined by superimposition on the stress intensities developed from the NC free drop (at the operating temperature) as stipulated by Regulatory Guide 7.6, *Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels* and Regulatory Guide 7.8, *Load Combinations for the Structural Analysis of Shipping Casks for Radioactive Material*. The combined stress intensity values with free drop are then measured against ASME BPVC, Section III, Subsection NE, Service Level C limits to demonstrate that containment is maintained. Load combinations without free drops are assessed against ASME BPVC, Section III, Subsection NE, Service Level A limits. In addition, to ensure that containment is maintained, the cumulative strains at key locations of the package are evaluated. In this case, the acceptable performance criterion is that there are no unconfined plastic strains that would result in loss of dimensional stability of the closure system. NC free drop stress intensities and cumulative strains are developed from numerical simulations of the free drop impact onto the specified concrete surface using ABAQUS/Explicit.¹

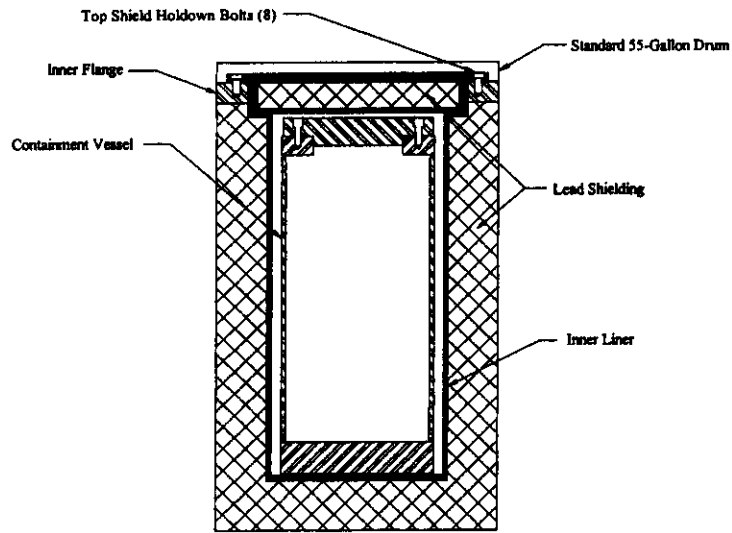
¹ ABAQUS/Explicit version 5.8-1 was used for the numerical modeling and impact simulations in this document. Validation and verification of this computer program is documented in Waste Management Technical Services document ENG-VV-010, *ABAQUS Version 5.8-1 Verification on PX17265 Computer* (WMTS 1999).

For AC evaluations, package performance under free drop and puncture conditions are evaluated for the limiting values of strain energy density (SED) and triaxial strain limit (TSL). The SED is used to assess the damage to the package and ensure no breach or rupture of the package occurs. The TSL is used to assess cumulative strains at key locations of the package and ensure no loss of dimensional stability or rupture that would result in loss of containment. In both cases, containment by the package is demonstrated by the SED and cumulative strains at all locations being below the limits established for the material. For the fire accident-loading case, the loading on the package is assessed against ASME BPVC, Section III, Subsection NE, Service Level D limits to demonstrate containment by the structure. Material maximum high temperature limits for 1 hour after the fire are used to demonstrate seal integrity. The temperature at the centerline of the shielding is to demonstrate that shielding integrity is maintained by comparison to the melting point of lead and assessment of lead slump at the centerline temperature.

3.0 DESCRIPTION OF PACKAGE

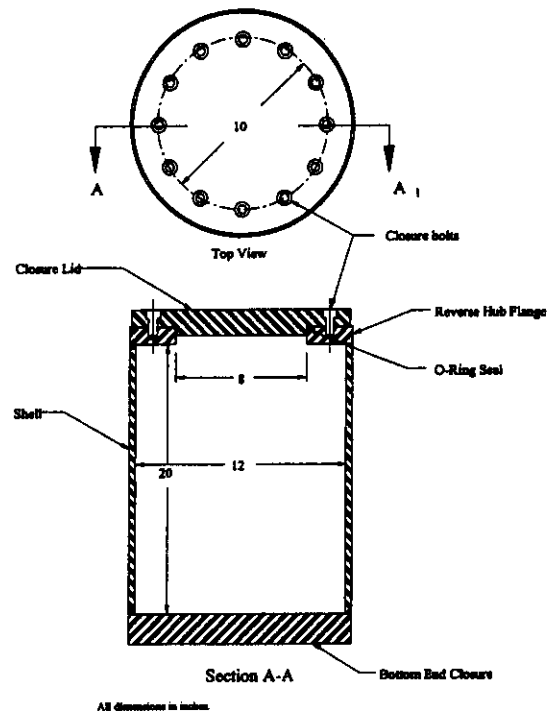
The outer shell of the package is a UN1A2 210 L (55-gal) galvanized drum with a removable gasketed head cover and bolted locking ring. The drum and head cover are manufactured from 18-gage American Society for Testing and Materials (ASTM) A-366 galvanized steel. Inside the drum is a welded carbon steel liner with a bolting ring, and lead filled shielding lid. The cavity between the liner is filled with lead. Figure 1 shows an assembly of the package. The containment boundary shell is constructed of a 30 cm (12-in.) Schedule 40 by 51 cm (20-in.) high stainless steel pipe welded on the bottom to a 2.54 cm (1-in.) thick ASTM SA-240, Type 304L plate. The shell is constructed of ASTM SA-312, Type TP304L stainless steel with an outside diameter of 32.4 cm (12.75 in.), inside diameter of 30.32 cm (11.938 in.), and a wall thickness of 1.03 cm (0.406 in.). The reverse hub flange is constructed of ASTM SA-240, Type 304L plate with an outside diameter of 32.4 (12.75 in.), inside diameter of 30 cm (8 in.), and a wall thickness of 3.8 cm (1.5 in.). The closure lid, also constructed of ASTM SA-240, Type 304L plate, has a diameter of 32.38 cm (12.75 in.), and a height of 2.54 cm (1 in.). The stepped in section of the closure lid has a diameter of 30 cm (8 in.) and a height of 1.3 (0.5 in.). The lid is bolted down by 12, 1/2-13 UNC SA-193, Grade B8 bolts. Figure 2 shows an example of a generic shield drum package containment vessel with these dimensions.

Figure 1. Lead Shielded Drum Package.



Lead Shielded Drum Package

Figure 2. Generic Shielded Drum Package.



4.0 CONTAINMENT VESSEL MATERIAL PROPERTIES

In most cases, the mechanical properties for the generic shielded package containment vessel are extracted from the ASME BPVC, Section II, Part D (ASME 1995c). When mechanical properties were not specifically stated in the ASME BPVC section, they were developed from room temperature properties published in the ASTM standards and expanded relative to temperature by the methods outlined in the *Military Handbook-Metallic Materials and Elements for Aerospace Vehicle Structures* (MIL-HDBK-5G). Table 1 shows the containment boundary materials and properties. Table 2 shows the stress evaluation criteria used for NC Service Levels A and C, and Tables 3 and 4 show the allowable stresses for the materials used for construction of the generic shielded drum package containment vessel.

Table 1. Minimum Packaging Containment Boundary Materials and Properties for Generic Shielded Drum Package, ASME BPVC, Class 2/3.

Component/ Material/UNS Number	Temperature (°F)	Yield Strength S _y (ksi)	Ultimate Strength S _u (ksi)	Membrane Allowable S _m (ksi)	Elastic Modulus (10 ⁶ psi)	Coefficient of Expansion (10 ⁶ in/in-F) (A/C)
SA-240, 304L, plate, 18Cr-8Ni, S30403 [p. 90, 33, permitted by III and VIII-1]. closure lid, reverse hub flange, and bottom-end closure	100	25.0	70.0	16.7	28.1	8.63/0.0031
	200	21.3	66.2	16.5	27.6	9.08/0.0137
	300	19.1	60.9	15.3	27.0	9.46/0.0248
	400	17.5	58.5	14.7	26.5	9.80/0.0364
	500	16.3	57.8	14.4	25.8	10.10/0.0483
	600	15.5	57.0	14.0	25.3	10.38/0.0606
SA-312, TP304L, smls pipe, 18Cr-8Ni, S30403 [p. 94, 1, permitted by VIII- 1]. shell.	100	25.0	70.0	16.7	28.1	8.63/0.0031
	200	21.3	66.2	16.5	27.6	9.08/0.0137
	300	19.1	60.9	15.3	27.0	9.46/0.0248
	400	17.5	58.5	14.7	26.5	9.80/0.0364
	500	16.3	57.8	14.4	25.8	10.10/0.0483
	600	15.5	57.0	14.0	25.3	10.38/0.0606
SA-193, B8, 18Cr- 8Ni, S30400, ≤3/4 [p. 390, 31]. bolts.	100	97.0	120.0	25.0	28.1	8.63/0.0031
	200	86.0	101.3	25.0	27.6	9.08/0.0137
	300	75.0	92.5	25.0	27.0	9.46/0.0248
	400	68.0	87.5	25.0	26.5	9.80/0.0364
	500	63.0	87.5	25.0	25.8	10.10/0.0483
	600	62.0	87.5	25.0	25.3	10.38/0.0606

Table 2. Confinement Boundary Stress Evaluation Criteria for Normal Transfer Conditions.

	Stress category	Normal transfer conditions	
		Service level A	Service level C
Components other than bolts	General membrane (Average primary stress across solid section. Excludes discontinuities and concentrations.)	S_{mc}	greater of $1.0 S_y$ and $1.2 S_{mc}$
	Local membrane (Average stress across solid section. Considers discontinuities but not concentrations.)	$1.5 S_{mc}$	greater of $1.5 S_y$ and $1.8 S_{mc}$
	Local + bending (Component of primary stress proportional to distance from centroid of solid section. Excludes discontinuities and concentrations.)	$1.5 S_{mc}$	greater of $1.8 S_{mc}$ and $1.5 S_y$
Bolts	Yield	$2/3 S_y$	---

Table 3. Confinement Boundary Normal Condition Allowable Stresses for Service Level A for the Generic Shielded Drum Package (ksi).

	Stress Category	30 °C (100 °F)	93 °C (200 °F)	149 °C (300 °F)	204 °C (400 °F)	260 °C (500 °F)	315 °C (600 °F)
Components other than bolts	Material	SA-240, 304L					
	General membrane	18.37	18.15	16.83	16.17	15.84	15.40
	Local membrane	27.56	27.23	25.25	24.26	23.76	23.10
	Local + bending	27.56	27.23	25.25	24.26	23.76	23.10
	Material	SA-312, TP304L					
	General membrane	18.37	18.15	16.83	16.17	15.84	15.40
	Local membrane	27.56	27.23	25.25	24.26	23.76	23.10
	Local + bending	27.56	27.23	25.25	24.26	23.76	23.10
Bolts	Material	SA-193, B8					
	Yield	64.67	57.33	50.00	45.33	42.00	41.33

Table 4. Confinement Boundary Normal Condition Allowable Stresses for Service Level C for the Generic Shielded Drum Package (ksi).

	Stress Category	30 °C (100 °F)	93 °C (200 °F)	149 °C (300 °F)	204 °C (400 °F)	260 °C (500 °F)	315 °C (600 °F)
Components other than bolts	Material	SA-240, 304L					
	General membrane	25.00	21.78	20.20	19.40	19.01	18.48
	Local membrane	37.50	32.67	30.29	29.11	28.51	27.72
	Local + bending	37.50	32.67	30.29	29.11	28.51	27.72
	Material	SA-312, TP304L					
	General membrane	25.00	21.78	20.20	19.40	19.01	18.48
	Local membrane	37.50	32.67	30.29	29.11	28.51	27.72
	Local + bending	37.50	32.67	30.29	29.11	28.51	27.72
Bolts	Material	SA-193, B8					
	NA	---	---	---	---	---	---

5.0 SUMMARY OF RESULTS

The results of this evaluation show that a shielded drum package with a containment vessel designed and constructed in accordance with ASME BPVC, Section VIII, Division 1 will meet minimum performance standards performance requirements. Results of this evaluation demonstrate that the package maintains containment under both normal and ACs. Maintenance of containment is demonstrated by the stress intensities and strains from the various loading conditions remaining below allowable limits established in the minimum performance standards acceptable performance criteria. Evaluation results for NC show that all stress intensities from the specified free drop load combinations remain below ASME BPVC, Section III, Subsection NE, Service Level C limits. For the specified load combinations, excluding free drop, the stress intensities are shown to remain below Service Level A limits. This is demonstrated by the ASME BPVC MNOP evaluation of the containment vessel provided in this section. Under AC, the package loads remain under established allowable SED and cumulative strain limits. In this evaluation, the free drop orientation resulting in the most severe damage is determined. The package is then subjected to the accident sequence of a NC free drop followed by an AC free drop, and finally a puncture drop in this orientation. For this package, the most severely damaging orientation is the horizontal side drop. Results of this sequential loading show that, although the package and containment vessel are severely deformed, containment is maintained.

It must be noted that in the case of the U.S. Nuclear Regulatory Commission (NRC) cask evaluations, the sequencing of the ACs is not significant, because by imposition of Subsection NB, the material response is primarily elastic. However, for onsite evaluations where the elastic-plastic response of the material is used to demonstrate containment, shielding, and subcriticality, evaluation of the package to loadings from the specified sequence of AC tests are of critical importance. As demonstrated in the results and summarized in Table 5, the cumulative strains are approximately additive. However, the maximum cumulative strains do not always occur in the same location in each of the loading conditions. Consequently, the total cumulative strains at key locations from a sequential application of accident performance tests are of critical importance in demonstrating containment, shielding, and subcriticality maintenance for onsite packages.

5.1 DESIGN CALCULATIONS

ASME BPVC, Section VIII, Division 1 was used to design the generic shielded drum package containment vessel. In addition to finding the thickness of the various components, materials are also specified for a design pressure of 1.4 MPa (200 psi) at a design temperature of 204 °C (400 °F).

The cylindrical shell is treated as a thin-walled vessel. A 30 cm (12 in.) Schedule 40 pipe with an outside diameter of 32.4 cm (12.75 in.), inside diameter of 30.32 cm (11.938 in.), and a wall thickness of 1.03 cm (0.406 in.) is specified for the body. The specified shell material of construction is SA-312, TP 304L stainless steel seamless pipe.

Table 5. Worst-Case Strain Results for Each Drop Orientation.

Drop	Element	Strain (in/in)	Location
1 m (4 ft) horizontal side drop	31081	1.637 E-02	bottom junction
	32161	2.268 E-02	top junction
1 m (4 ft) bottom-end drop	32851	9.309 E-03	flange
	32852	1.060 E-02	flange
1 m (4 ft) bottom corner drop	32121	1.746 E-03	shell wall
	32161	3.260 E-03	top junction
1 m (4 ft) top corner drop	31081	5.736 E-03	bottom junction
	31082	1.812 E-03	bottom junction
9 m (30 ft) horizontal side drop	31081	5.416 E-02	bottom junction
	31123	5.554 E-02	shell wall
9 m (30 ft) bottom-end drop	31081	4.556 E-02	bottom junction
	31092	4.563 E-02	bottom junction
9 m (30 ft) bottom corner drop	32121	9.707 E-03	shell wall
	32161	5.443 E-03	top junction
9 m (30 ft) top corner drop	31081	3.148 E-02	bottom junction
	32321	2.742 E-02	top junction
Puncture drop	30741	9.366 E-02	bottom junction
	31081	8.427 E-02	bottom junction

The closure lid and bottom-end closure are constructed from SA-240, 304L stainless steel plate. The bottom-end closure is a flat, circular unstayed blind flange with a diameter of 32.4 cm (12.75 in.) and a thickness of 3.8 cm (1.5 in.). The closure lid has a diameter of 32.4 cm (12.75 in.) and a thickness of 3.8 cm (1.5 in.). Twelve closure bolts constructed of SA-193, Grade B8 are used to bolt the lid in place.

The reverse hub flange is an integral flange with an outside diameter of 32.4 cm (12.75 in.), an inside diameter of 20 cm (8 in.), and a thickness of 3.8 cm (1.5 in.). The bolt circle diameter is 25 cm (10 in.). The flange is also constructed from SA-240, 304L stainless steel plate.

5.2 MNOP PRESSURE EVALUATION

The MNOP for the generic shielded drum containment vessel is specified to be 0.7 MPa (100 psi) at a maximum normal temperature of 93 ° C (200 °F). For the evaluation, the containment vessel is conservatively idealized as a cylindrical shell with two flat circular closure plates. The vessel is evaluated to ASME BPVC, Section III, Subsection NE for Service Level A in accordance with ASME BPVC, Section III. The bolts are evaluated to NUREG/CR-6007, *Stress Analysis of Closure Bolts for Shipping Casks*. The margins of safety for the applied internal pressure load at key locations on the containment vessel are as follows:

- 9.9 for the shell body
- 29.5 for the junction of the shell to the flange
- 4.4 in the center of the top and bottom closure plates
- 1.3 for tension in the bolts
- 3.2 for shear in the bolts

- 0.2 for tension + shear in the bolts
- 0.8 for maximum stress intensity in the bolts.

Furthermore, stress intensities, principal strains, and strain energy densities at key locations are as follows:

- 1.5 ksi stress intensity on the shell body
- 1.6 ksi stress intensity at the junctions of the shell body
- 4.55×10^{-5} principal elastic strain on the shell body
- 9.46×10^{-5} principal elastic strain at the junctions on the shell body
- 0.2 lb-in/in³ SED on the shell wall
- 0.4 lb-in/in³ SED at the junctions of the shell body.

These values are used in this evaluation for specified load combinations for both NC and AC evaluations. The values obtained from the ABAQUS/Explicit numerical simulation are superimposed on the pressure loads to develop the necessary load combinations.

5.3 ABAQUS/EXPLICIT NC AND AC FREE DROP NUMERICAL SIMULATIONS

As part of this example, the typical drum package design is numerically simulated in the ABAQUS/Explicit computer program. Element maps of the numerical model are provided in Figures 3 and 4. The ABAQUS/Explicit package model is subjected to numerically simulated NC and AC free drops onto a hard unyielding surface. In this case, a hard unyielding surface is used because the robust concrete surface specified is much harder and stiffer than a lead-lined drum. Consequently, the concrete would act as a hard unyielding surface relative to the drum. To put this into perspective, the drum is essentially a sheet metal can filled with lead. If such a sheet metal can is dropped onto a concrete surface, little damage would occur to the concrete, and the sheet metal can and relatively soft lead would absorb most of the energy by deformation.

The typical drum package is subjected to four NC and four AC free drops at different orientations. The free drop orientations are: (1) horizontal side, (2) flat bottom-end, (3) center of gravity over bottom corner, and (4) center of gravity over top corner.

Also, for the worst-case orientation, the package is numerically subjected to NRC interpretation of the regulations for AC free drop tests. The NRC interpretation is that the NC free drop must be followed by the 9 m (30-ft) AC free drop in the same orientation and at the same impact location. To meet the accident event sequence specified in the regulations, the most severely damaged package is then subjected to the specified puncture test and fire test. In all free drop orientations, the combined loads from the appropriate NCs will be superimposed on the accident loads.

Figure 3. Element Map Diagram for Generic Shielded Package Containment Vessel.

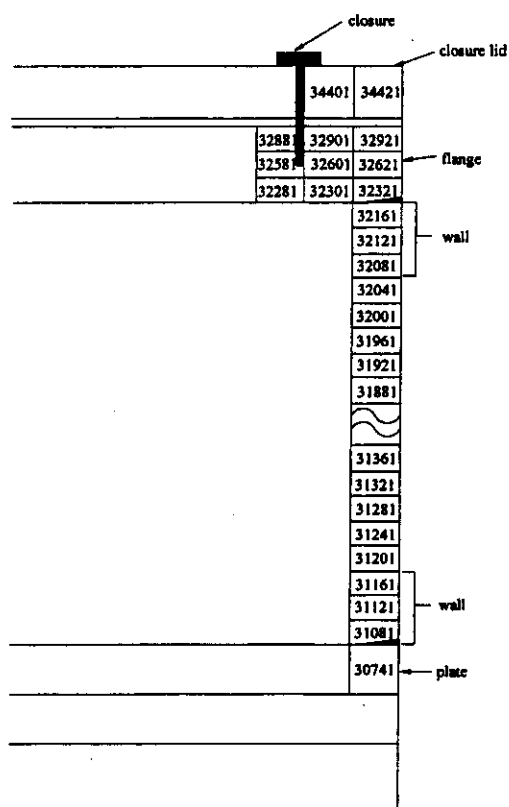
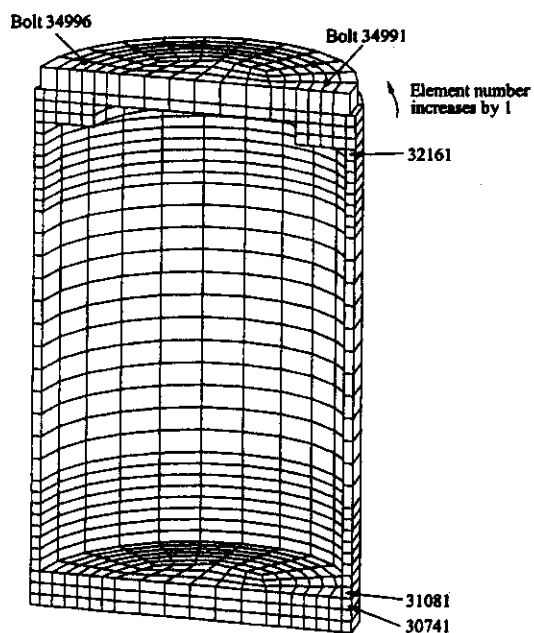


Figure 4. Generic Shielded Drum Package Containment Vessel Elements.



5.4 NC AND AC EVALUATIONS

In these evaluations, the deceleration, velocity, and displacement time histories of the package are retrieved and processed after the ABAQUS run. The deceleration time history is based on the contact forces between the typical drum package and the unyielding target. This deceleration time history contains high frequency numerical and vibration noise that needs to be filtered out for accurate assessment of the deceleration forces involved. In this analysis, the deceleration time histories undergo a two-stage filtration using a six-pole sine-Butterworth filter. The first filtration filters out frequencies at or above 10,000 Hz. This filters out the computational numerical induced high frequency noise. The second filtration uses a cutoff frequency recommended by the International Atomic Energy Agency (IAEA) (1990), *Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material*, Safety Series No. 37. For a package in the range of 1,814 kg (4,000 lb), the cutoff frequency range is 380-760 Hz. In the following analyses, all deceleration time histories are filtered with a final cutoff frequency of 500 Hz.

The stress intensity, cumulative strain, and SED results are subjected to a single-stage of 500 Hz filtration in these analyses to eliminate high frequency noise. The stress intensities reported are ASME BPVC stress intensities, which are defined as the range between the largest principal stress and the smallest principal stress at an evaluation location. In conjunction with this, the cumulative strains reported are defined as the range of the largest and the smallest of logarithmic principal strain. This, by definition, is the strain counterpart of the ASME stress intensity.

5.5 ACCEPTANCE CRITERIA

5.5.1 Normal Transport Conditions

As previously stated, the acceptable performance criteria in maintaining containment during NC is ASME BPVC, Section III, Subsection NE, Service Level A limits for all combined loads except for those involving free drop. The acceptable performance criteria for those combined loads including free drop is ASME BPVC, Section III, Subsection NE, Service Level C limits.

5.5.2 Accident Conditions

In this typical shielded drum package evaluation, the limiting SED is defined as 70% of the material toughness. This is in keeping with the primary membrane stress intensity limit established in ASME BPVC for Service Level D limits. Based on this criterion, the maximum SED of the containment vessel is 1.16×10^4 in-lb/in³ and 1.53×10^4 in-lb/in³ for the containment closure bolts.

The TSL of the equivalent plastic strain (PEEQ) on the containment vessel material are for:

- Uniaxial tension loading (TF=1), the materials limit is 35%.
- Biaxial tension loading (TF=2), the materials limit is 17%.

- Significant compression (TF negative), the materials limit is 70%.
- For conservatism, the materials limit in weld areas is $\frac{1}{2}$ the above values.

To ensure containment by the containment closure bolts, the total cumulative plastic strain on each bolt is limited to 50% of the seal compression. In this case, the cumulative plastic strain limit on the closure bolts is 2%, whereas the ultimate elongation of the bolt material under uniaxial tension is 20%.

5.6 1 M (4-FT) NC FREE DROPS

5.6.1 Horizontal Side Drop

This case evaluates the typical shielded drum package performance for a 1 m (4-ft) horizontal side free drop onto a horizontal unyielding surface (see Figure 5). The total numerical simulation time in this case is 20 ms. The resulting impact deceleration time history is shown in Figure 6. From Figure 6, the peak acceleration is determined to be 215.6gs with a 3.27 ms rebound time of the package. Based on the rebound time and change in velocity, the average deceleration is calculated to be 152.4gs. The velocity time history profile of the package is shown in Figure 7 and the displacement time history profile of the package is shown in Figure 8. Pertinent impact deceleration data are shown in Table 6.

Figure 5. Shielded Drum Package 1 m (4 ft) Horizontal Side Drop.

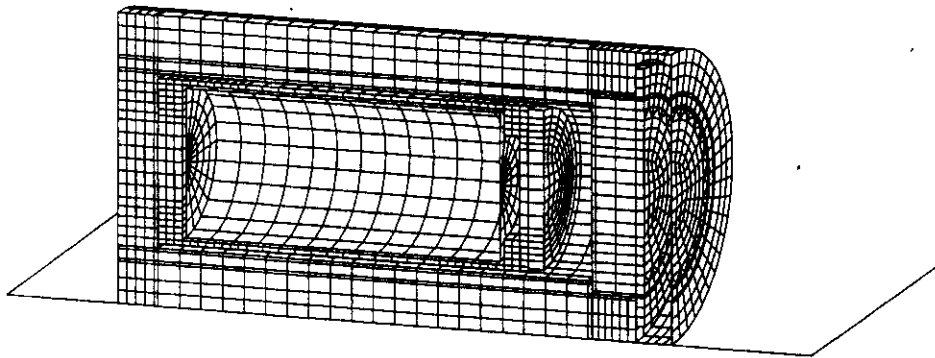


Figure 6. Shielded Drum Package 1 M (4 ft) Horizontal Side Drop Acceleration History.

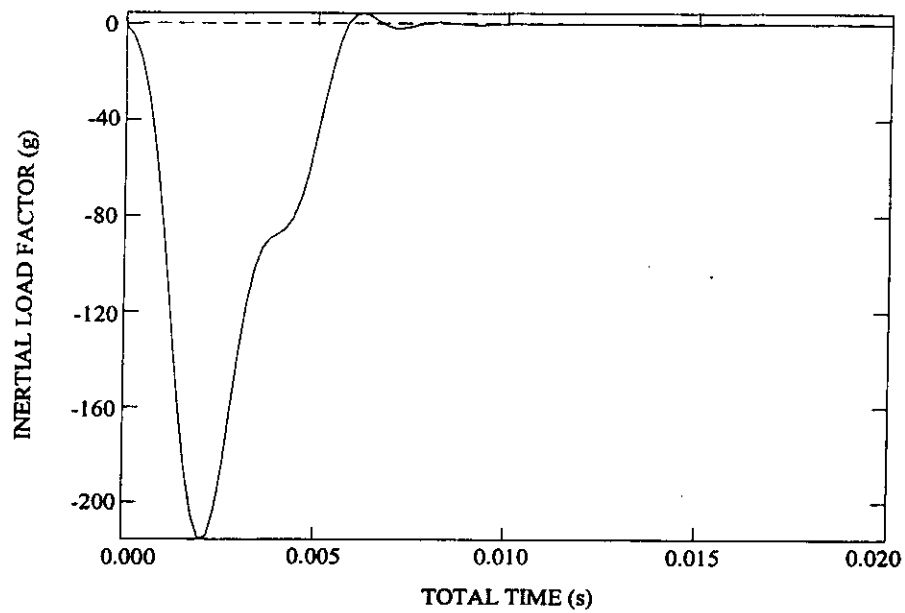


Figure 7. Shielded Drum Package 1 m (4 ft) Horizontal Side Drop Velocity History.

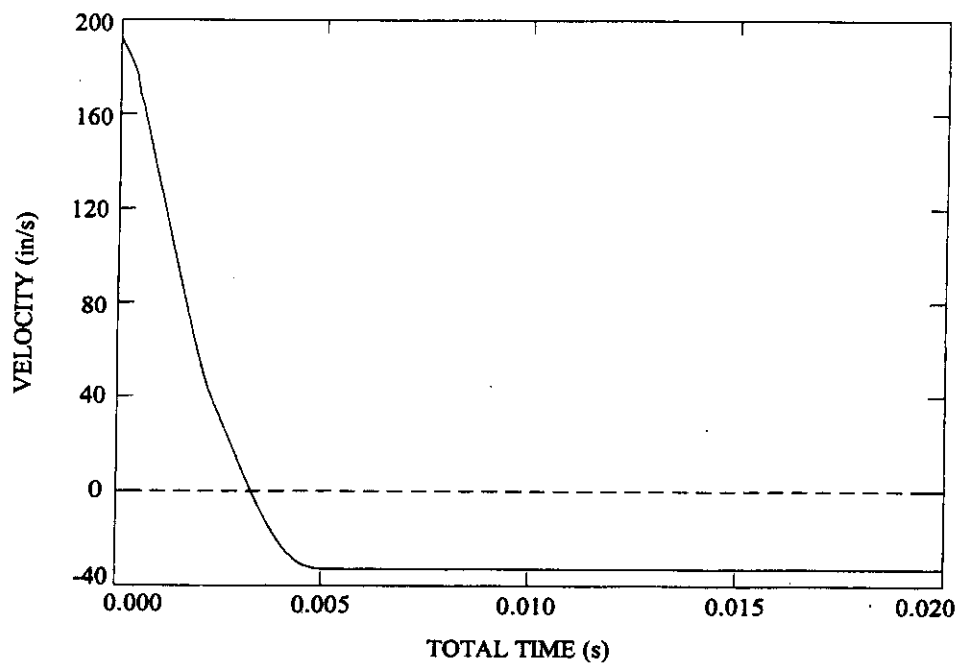


Figure 8. Shielded Drum Package 1 m (4 ft) Horizontal Side Drop Displacement History.

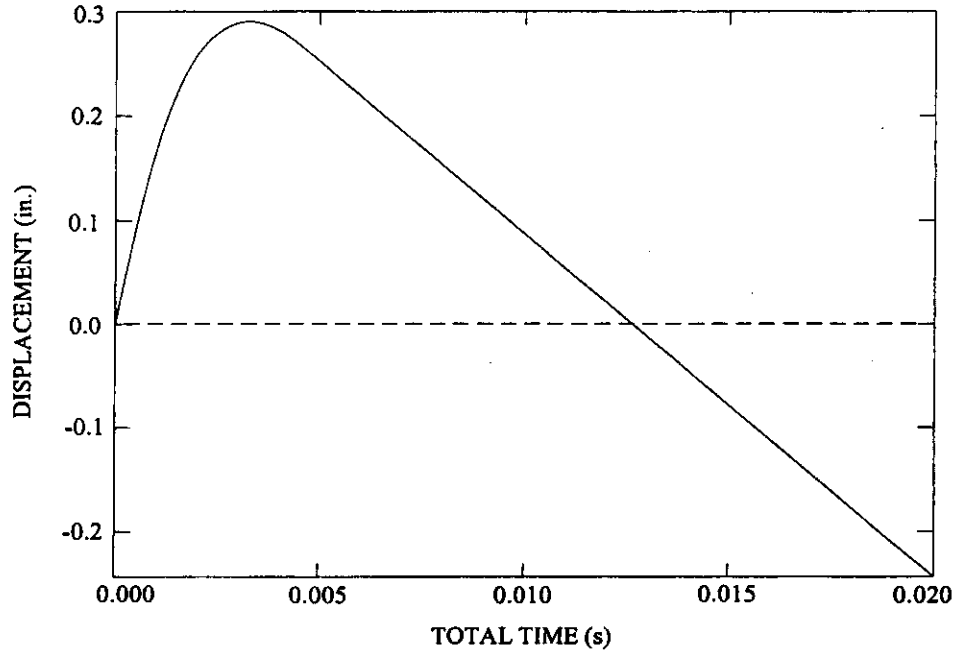


Table 6. Impact Acceleration Data of the 1 m (4-ft) Horizontal Side Drop.

Case	Average rebound-time acceleration (g)	Peak acceleration (g)	Rebound time (ms)
1 m (4 ft) horizontal side drop	152.4	215.6	3.27

The ASME BPVC stress intensities at the most highly stressed containment vessel locations and at all containment closure bolts are shown in Table 7. The stress intensity time history at the highest stressed location of the containment vessel is shown in Figure 9. The stress intensity time history at the highest stressed containment closure bolt is shown in Figure 10.

The cumulative strains at the most severely strained locations of the containment vessel and at all containment closure bolts are shown in Table 8. The largest cumulative strain on the containment vessel occurs in the shell wall connection to the reverse hub flange. However, in this case, the cumulative strain is at a discontinuity and only slightly above the strain at yield in the range of 2.3%. A typical cumulative strain time history at EL 32161 is shown in Figure 11. The largest cumulative strain on the containment closure bolts is almost negligible at approximately 0.1% (see Figure 12).

Table 7. ASME Code Stress Intensity at Containment Vessel from the 1 m (4-ft) Horizontal Side Drop.

Containment		Closure bolt	
Element	Stress intensity (ksi)	Element	Stress intensity (ksi)
30741	22.74	34991	16.53
30742	18.74	34992	6.52
30743	18.88	34993	5.00
31081	28.88	34994	6.37
31082	29.07	34995	7.73
31083	27.70	34996	7.24
31121	27.90		
31122	28.76		
31123	27.95		
31161	21.22		
31162	26.05		
31163	27.49		
32081	19.30		
32082	23.30		
32083	22.96		
32121	25.30		
32122	24.81		
32123	22.22		
32161	30.44		
32162	27.32		
32163	22.28		
32921	24.02		
32922	14.12		
32923	12.99		

Figure 9. Shielded Drum Package 1 m (4-ft) Horizontal Side Drop Stress Intensity History at the Containment Location (EL 32161).

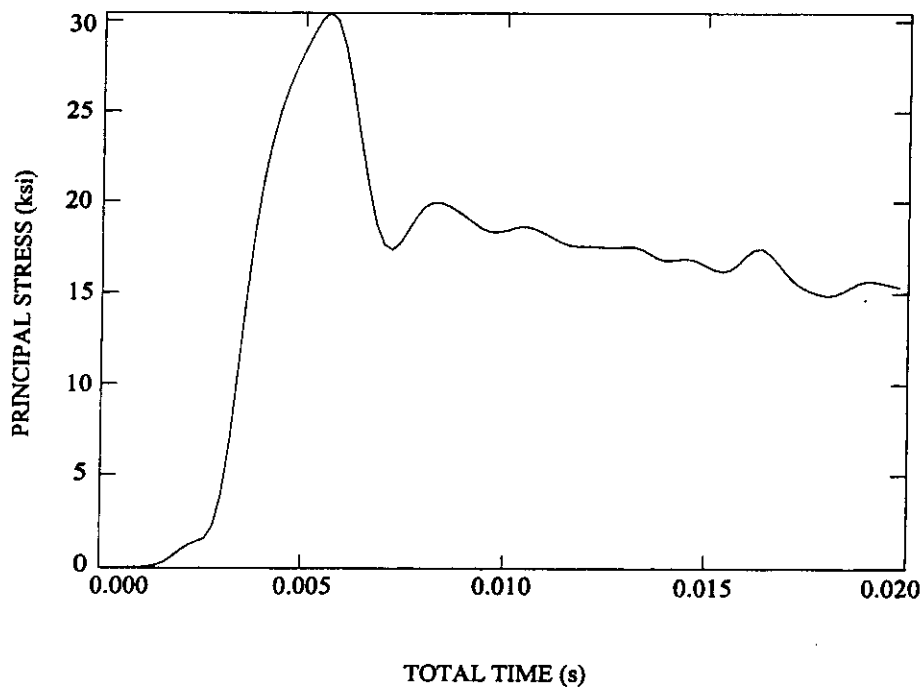


Figure 10. Shielded Drum Package 1 m (4 ft) Horizontal Side Drop Stress Intensity History at Closure Bolt No. 1 (EL 34991).

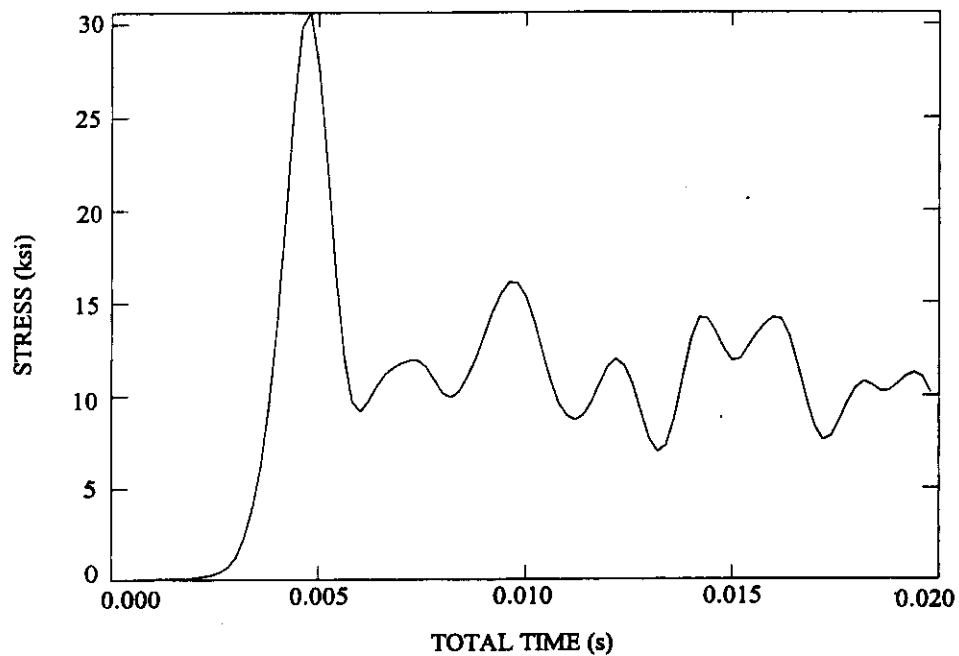


Table 8. Accumulated Peak Strains at Containment Vessel from the 1 m (4-ft) Horizontal Side Drop.

Containment		Closure bolt	
Element	Strain (in/in)	Element	Strain (in/in)
30741	3.295 E-03	34991	1.109 E-03
30742	1.354 E-03	34992	4.371 E-04
30743	8.614 E-04	34993	3.356 E-04
31081	1.637 E-02	34994	4.274 E-04
31082	1.195 E-02	34995	5.186 E-04
31083	5.931 E-03	34996	4.855 E-04
31121	6.534 E-03		
31122	7.760 E-03		
31123	7.291 E-03		
31161	2.425 E-03		
31162	5.282 E-03		
31163	6.900 E-03		
32081	4.452 E-03		
32082	2.961 E-03		
32083	1.847 E-03		
32121	1.360 E-02		
32122	7.395 E-03		
32123	1.927 E-03		
32161	2.268 E-02		
32162	1.262 E-02		

Table 8. Accumulated Peak Strains at Containment Vessel from the 1 m (4-ft) Horizontal Side Drop.

Containment		Closure bolt	
Element	Strain (in/in)	Element	Strain (in/in)
32163	1.694 E-03		
32921	8.035 E-03		
32922	7.889 E-04		
32923	5.666 E-04		

Figure 11. Shielded Drum Package 1 m (4-ft) Horizontal Side Drop Strain at Containment Location (EL 32161).

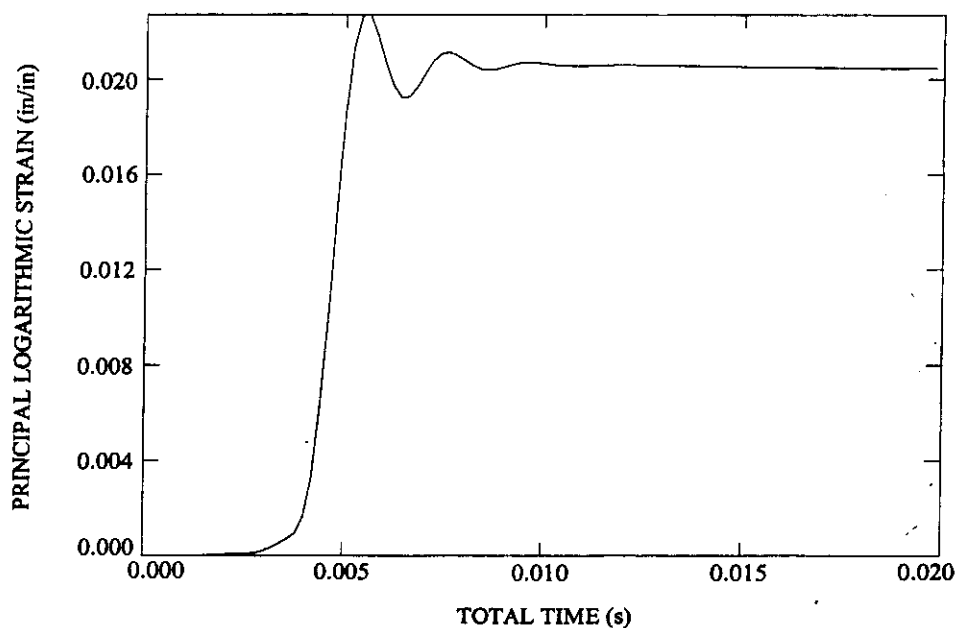
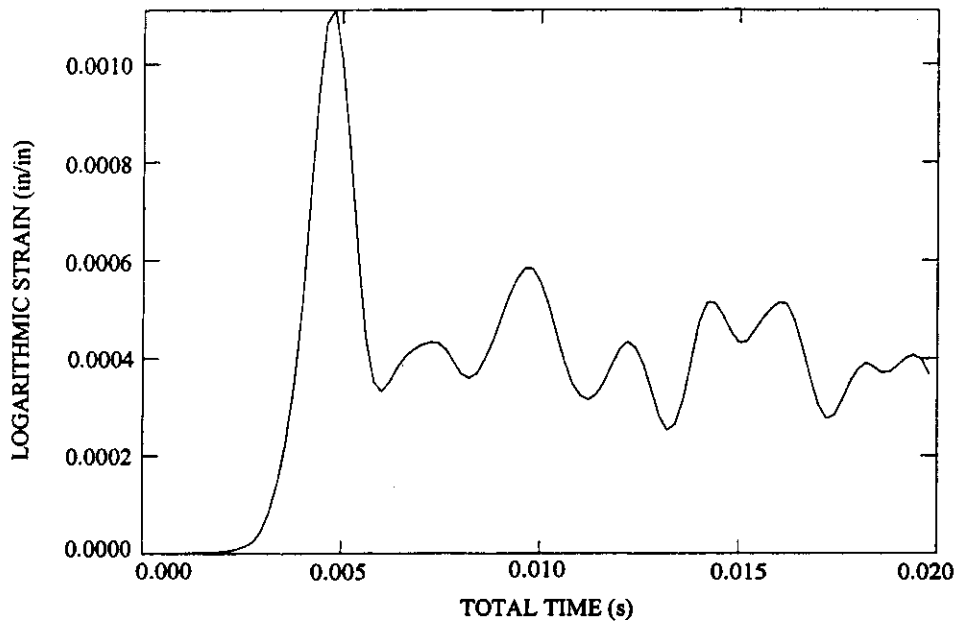
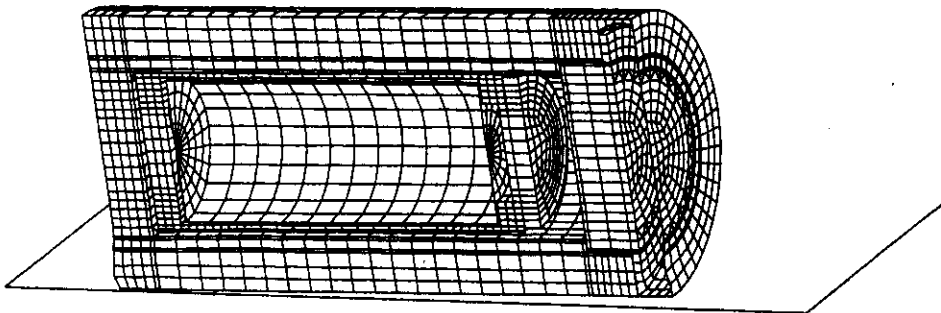


Figure 12. Shielded Drum Package 1 m (4-ft) Horizontal Side Drop Strain at Closure Bolt No. 1 (EL 34991).



The results show high stress locations on the containment vessel occur at the juncture of the cylindrical wall and the closure ends. In this case, the top-end stress intensities and cumulative strains are slightly higher than at the bottom end. These areas of high loading occur at discontinuities, and are very localized and concentrated around the impact edge and joint. Figure 13 shows the deformation of the drum and containment vessel at the end of the initial impact.

Figure 13. Shielded Drum Package 1 m (4-ft) Horizontal Side Drop Deformation Plot.



5.6.2 Flat Bottom-End Free Drop

This case evaluates the typical shielded drum package performance for a 1 m (4-ft) bottom-end drop onto a horizontal hard unyielding surface (see Figure 14). The total numerical simulation time of this case is 20 ms. The resulting impact deceleration time history is shown in Figure 15. From Figure 15, the peak acceleration is determined to be 255.8gs with a rebound time of 1.82 ms. Based on rebound time and the change in velocity, the average deceleration is calculated to be 273.9gs. The velocity time history of the package is shown in Figure 16 and the displacement time history of the package is shown in Figure 17. Pertinent impact acceleration data are shown in Table 9.

Figure 14. Shielded Drum Package 1 m (4-ft) Bottom-End Drop.

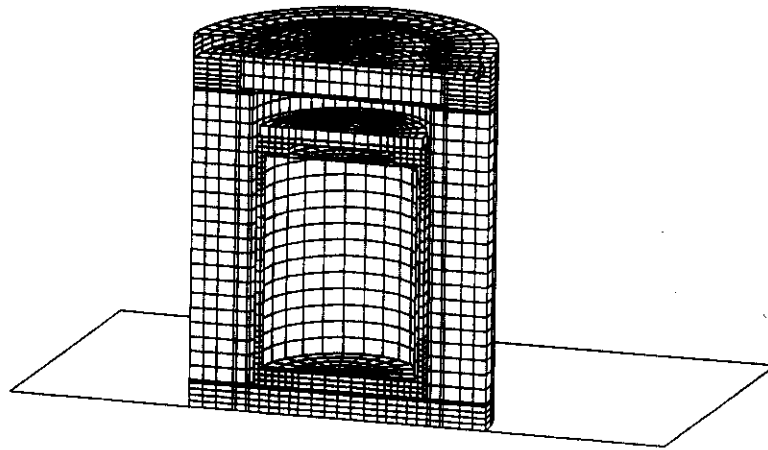


Figure 15. Shielded Drum Package 1 m (4-ft) Bottom-End Drop Acceleration History.

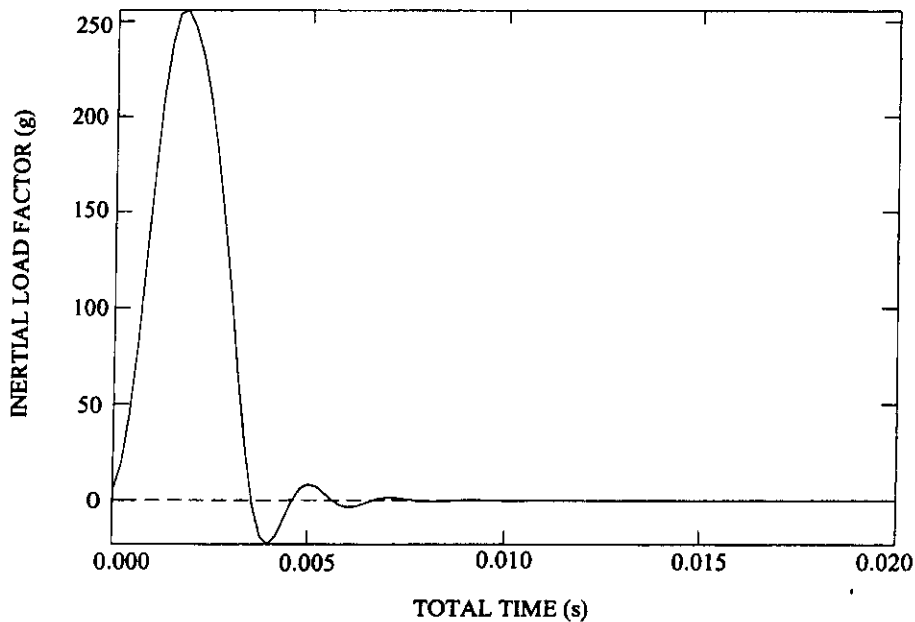


Figure 16. Shielded Drum Package 1 m (4-ft) Bottom-End Drop Velocity History.

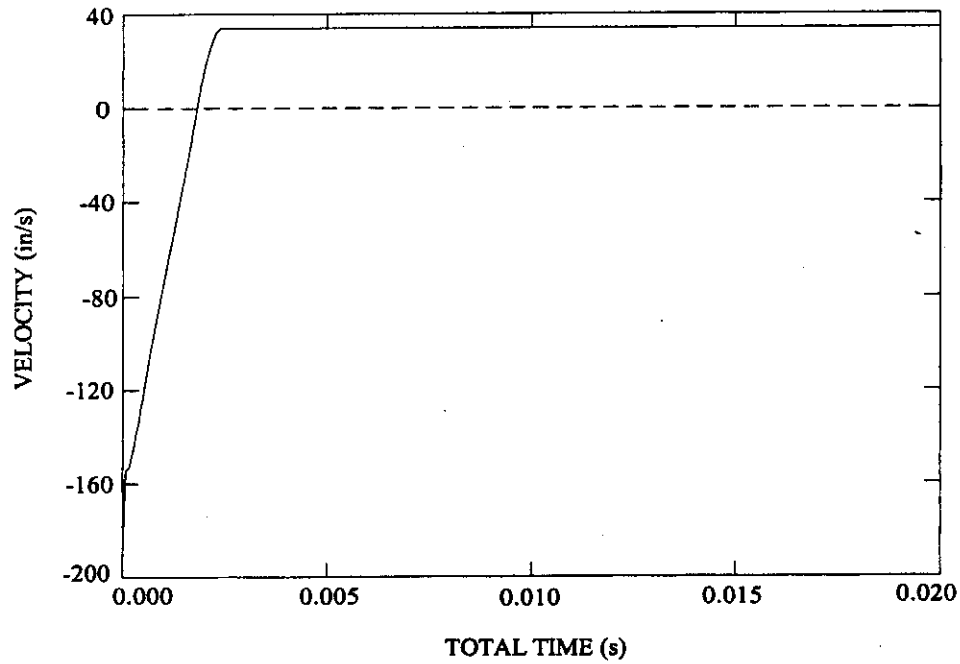


Figure 17. Shielded Drum Package 1 m (4-ft) Bottom-End Drop Displacement History.

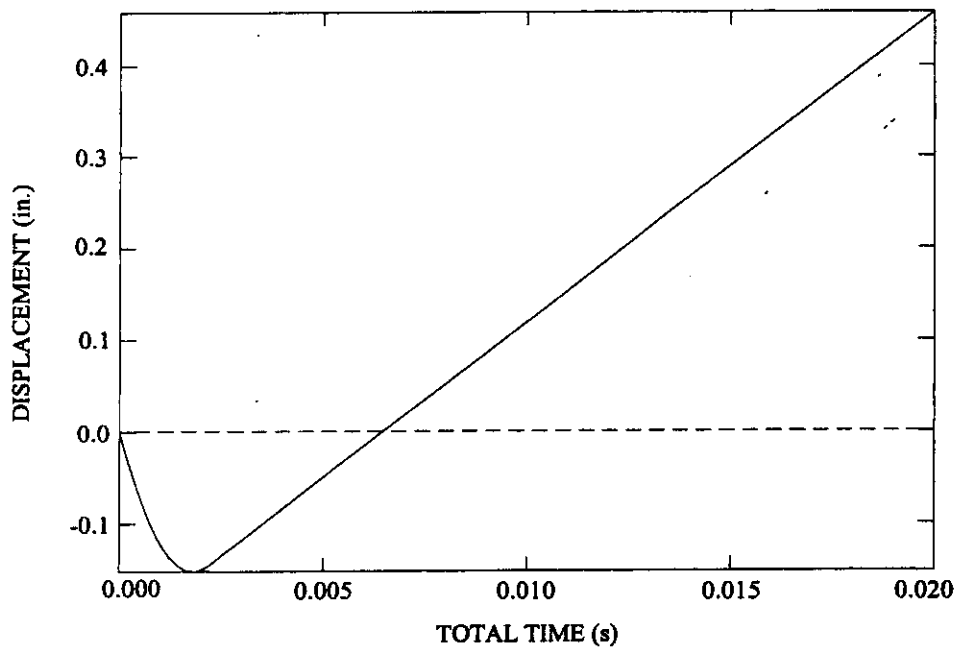


Table 9. Impact Acceleration Data of the 1 m (4 ft) Bottom-End Drop.

Case	Average rebound-time acceleration (g)	Peak acceleration (g)	Rebound time (ms)
1 m (4-ft) bottom-end drop	273.9	255.8	1.82

The ASME BPVC stress intensities at high stress locations on the containment vessel and all containment closure bolts are shown in Table 10. The stress intensity time history at the most severely stressed location on the containment vessel (EL 32851) is shown in Figure 18. The stress intensity time history at the highest stressed containment closure bolt is shown in Figure 19.

The cumulative strains at the high strain locations on the containment vessel and at all containment closure bolts are shown in Table 11. The highest cumulative strain on the containment vessel occurs at the reverse hub flange at EL 32852. This cumulative strain is approximately 1.1 % as shown in Figure 20. The highest cumulative strain on the containment closure bolts is approximately 0.9 % as shown in Figure 21.

Table 10. ASME Code Stress Intensity at Containment Vessel from the 1 m (4 ft) Bottom-End Drop.

Containment		Closure bolt	
Element	Stress intensity (ksi)	Element	Stress intensity (ksi)
30603	15.38	34991	18.79
30604	15.33	34992	19.82
30609	16.57	34993	19.19
30610	16.49	34994	19.26
31081	13.43	34995	19.18
31082	13.20	34996	18.89
31091	13.15		
31092	13.44		
31121	13.07		
31122	12.97		
31131	13.25		
31132	13.20		
31161	10.90		
31162	12.04		
31171	11.92		
31172	10.90		
32161	12.22		
32162	11.99		
32171	12.36		
32172	12.18		
32841	10.49		
32842	14.14		
32843	17.65		
32844	18.59		
32849	9.58		
32850	14.26		
32851	19.55		
32852	16.66		

Figure 18. Shielded Drum Package 1 m (4-ft) Bottom-End Drop Stress Intensity History at the Containment Location (EL 32851).

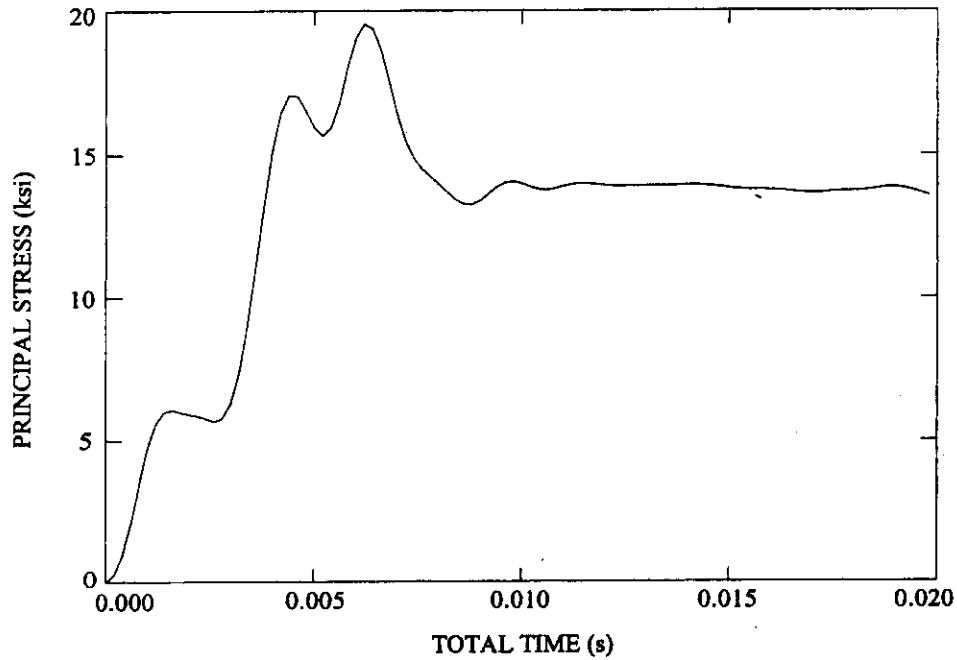


Figure 19. Shielded Drum Package 1 m (4-ft) Bottom-End Drop Stress Intensity at Closure Bolt No. 2 (EL 34992).

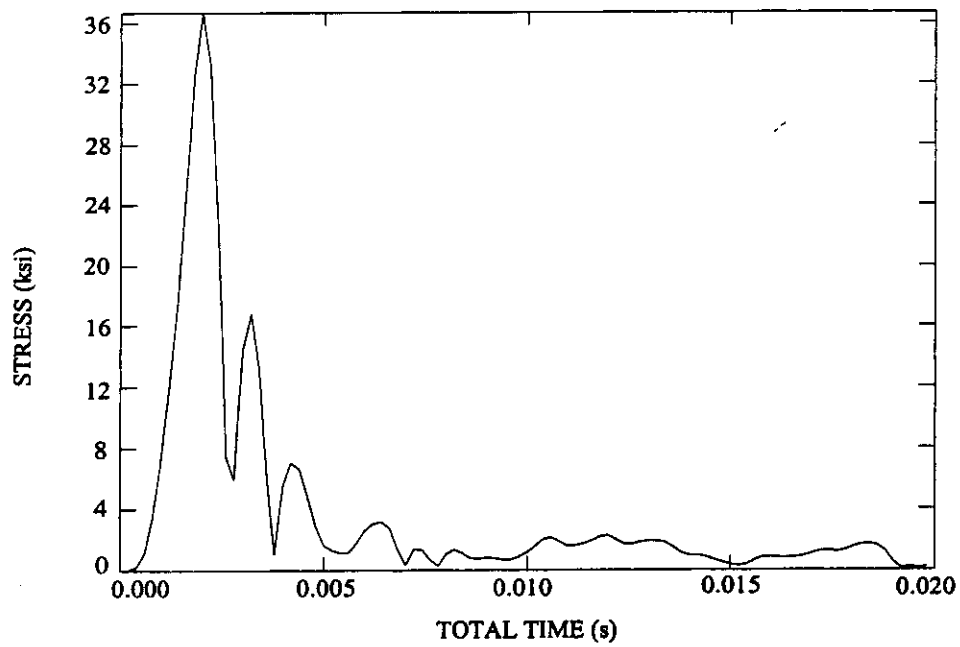


Table 11. Accumulated Peak Strains at Containment Vessel from the 1 m (4 ft) Bottom-End Drop.

Containment		Closure bolt	
Element	Strain (in/in)	Element	Strain (in/in)
30603	6.447 E-03	34991	8.817 E-03
30604	6.476 E-03	34992	8.844 E-03
30609	5.533 E-03	34993	8.765 E-03
30610	5.541 E-03	34994	8.880 E-03
31081	4.645 E-03	34995	8.910 E-03
31082	4.571 E-03	34996	8.931 E-03
31091	4.555 E-03		
31092	4.636 E-03		
31121	5.463 E-03		
31122	5.390 E-03		
31131	5.383 E-03		
31132	5.455 E-03		
31161	4.078 E-03		
31162	4.045 E-03		
31171	4.045 E-03		
31172	4.104 E-03		
32161	1.315 E-03		
32162	1.313 E-03		
32171	1.324 E-03		
32172	1.345 E-03		
32841	3.264 E-03		
32842	7.271 E-03		
32843	7.308 E-03		
32844	1.328 E-03		
32849	2.081 E-03		
32850	4.651 E-03		
32851	9.309 E-03		
32852	1.060 E-02		

Figure 20. Shielded Drum Package 1 m (4-ft) Bottom-End Drop Strain at the Containment Location (EL 32852).

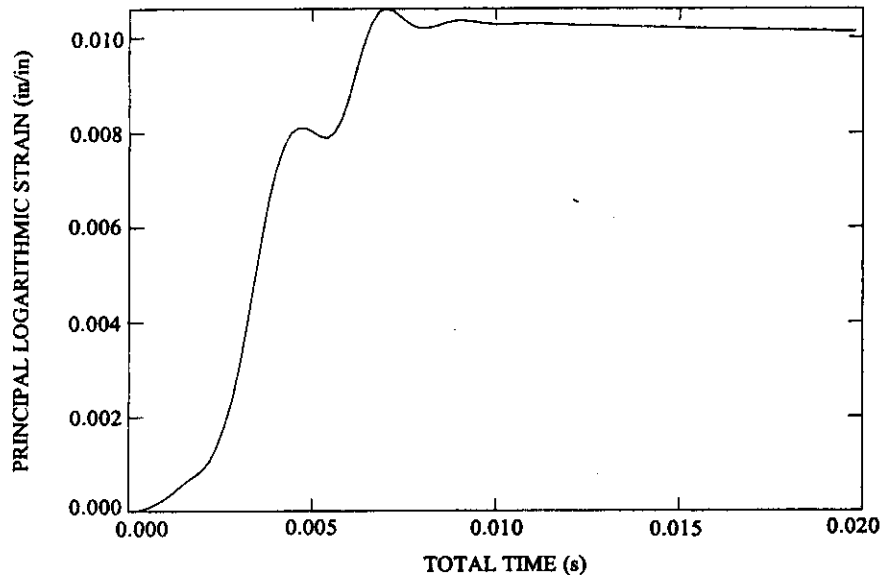
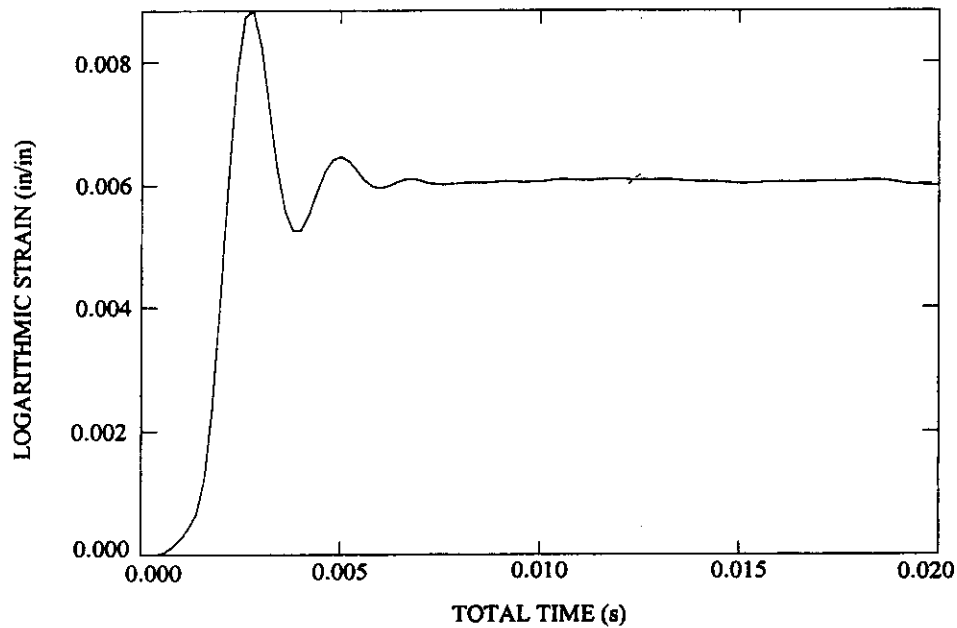


Figure 21. Shielded Drum Package 1 m (4-ft) Bottom-End Drop Strain at Closure Bolt No. 2 (EI 34992).



The results show the highest stresses on the containment vessel are within ASME BPVC, Section III, Subsection NE, Service Level C limits. The cumulative strains are only slightly above the strain at yield. Figure 22 shows the deformation of the entire package at the end of impact. Figures 23 and 24 show the deformation of the containment vessel, and the inner liner and lead shielding, respectively. From Figure 23, the containment vessel is shown to have no significant deformation. The drum lid is shown to deflect slightly downward and at the bottom the drum wall is pushed slightly outward by the hydrostatic action of the lead shielding.

In this case, the highly stressed locations of the containment vessel occur at the bottom closure flange and the top reverse hub flange. The top reverse hub flange is subjected to slightly higher stresses and strains than the bottom closure flange. The containment closure lid bolts are also subjected to a slight degree of plastic deformation due to the propagation of force waves from the area of impact upward through the containment wall and into the reverse hub flange and closure bolts.

Figure 22. Shielded Drum Package 1 m (4-ft) Bottom-End Drop Entire Package Deformation Plot.

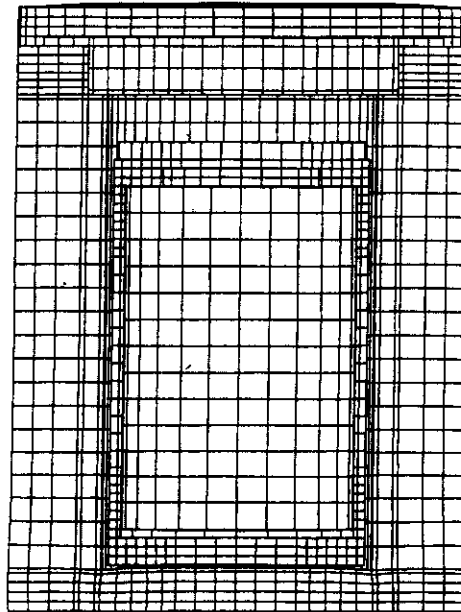


Figure 23. Shielded Drum Package 1 m (4-ft) Bottom-End Drop Containment Deformation Plot.

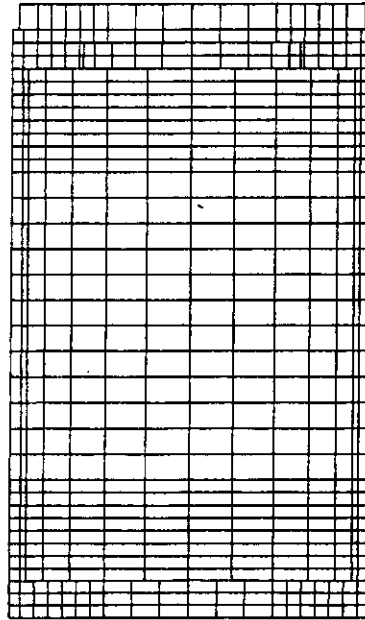
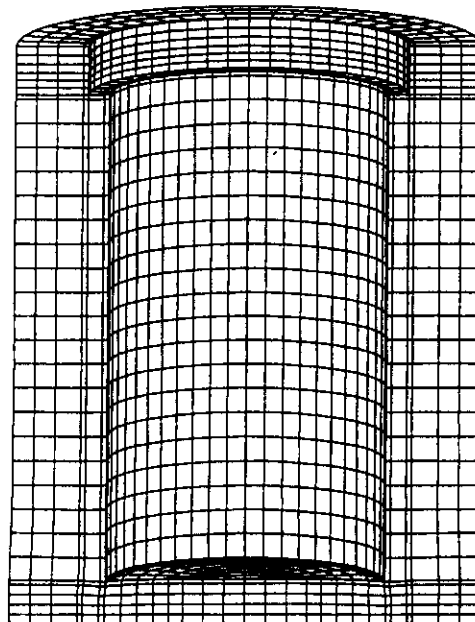


Figure 24. Shielded Drum Package 1 m (4-ft) Bottom-End Drop Inner Liner and Lead Shielding Deformation Plot.



5.6.3 Center of Gravity Over Bottom Corner Free Drop

This case evaluates the shielded drum package performance for a 1 m (4 ft) center of gravity over the bottom corner drop onto a horizontal hard unyielding surface (see Figure 25). The total numerical simulation time of this case is 20 ms. The resulting impact deceleration time history is shown in Figure 26. From Figure 26, the peak deceleration is determined to be 53.1gs with a rebound time of 4.11 ms. Based on the rebound time and velocity change, the average deceleration is calculated to be 35.3gs. The velocity time history of the package is shown in Figure 27 and the displacement time history of the package is shown in Figure 28. Pertinent impact acceleration data are shown in Table 12.

Figure 25. Shielded Drum Package 1 m (4 ft) Bottom Corner Drop.

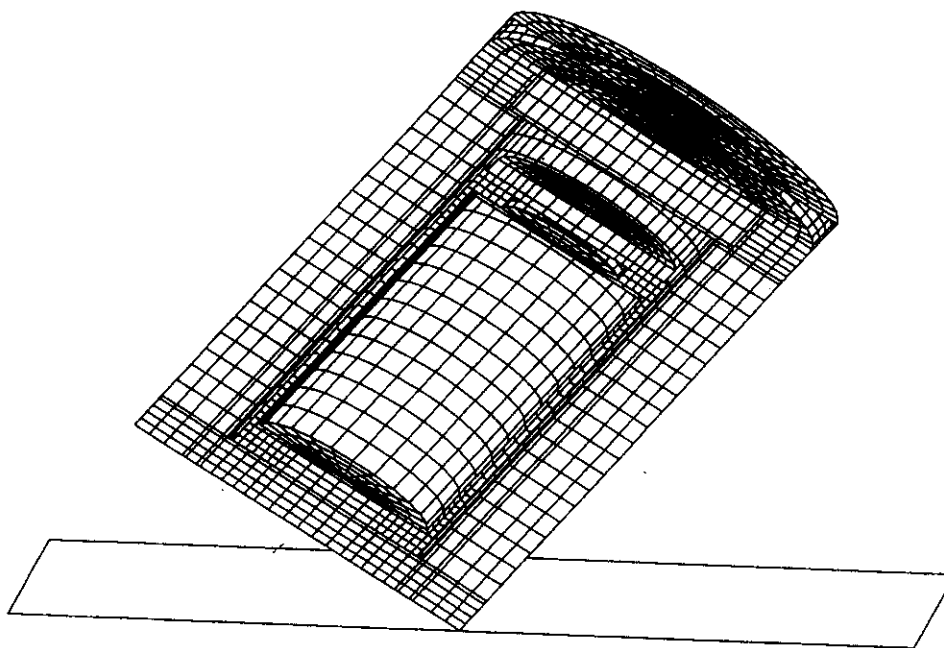


Figure 26. Shielded Drum Package 1 m (4-ft) Bottom Corner Drop Acceleration History.

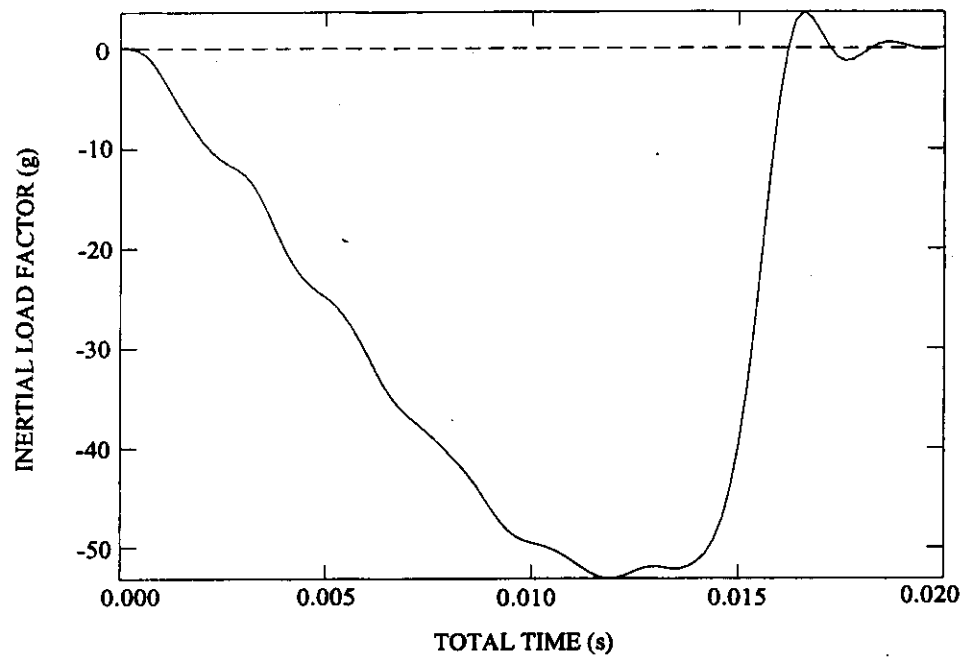


Figure 27. Shielded Drum Package 1 m (4-ft) Bottom Corner Drop Velocity History.

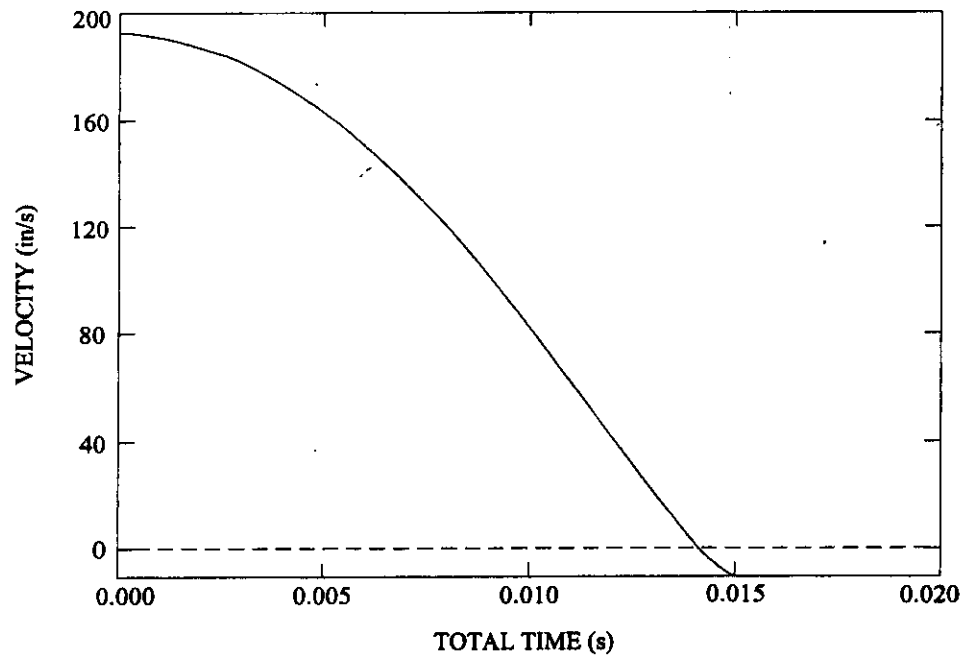


Figure 28. Shielded Drum Package 1 m (4-ft) Bottom Corner Drop Displacement History.

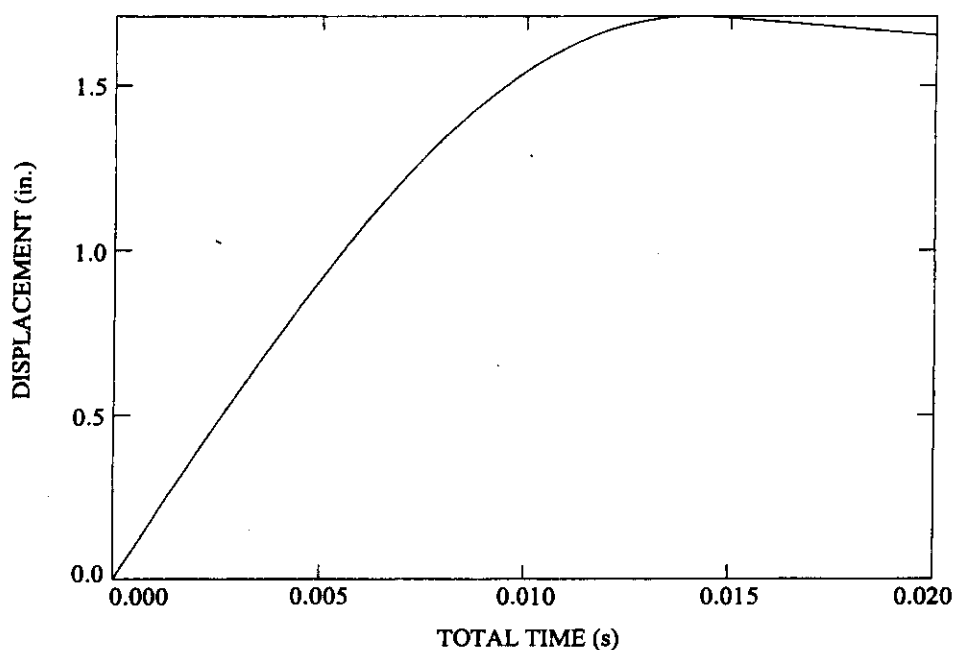


Table 12. Impact Acceleration Data of the 1 m (4-ft) Bottom Corner Drop.

Case	Average rebound-time acceleration (g)	Peak acceleration (g)	Rebound time (ms)
1 m (4-ft) bottom corner drop	35.3	53.1	14.11

The ASME BPVC stress intensities at highly stressed containment vessel locations and all containment closure bolts are shown in Table 13. The stress intensity time history at the most severely stressed location of the containment vessel (EL 32161) is shown in Figure 29. The stress intensity time history at the most severely stressed containment closure bolt is shown in Figure 30.

The cumulative strains at the highly strained locations of the containment vessel and all containment closure bolts are shown in Table 14. The largest cumulative strain on the containment vessel (EL 32161) occurs at the top end of the wall. However, the cumulative strain is only in the range of 0.33 % as shown in Figure 31. The largest cumulative strain on the containment closure bolts is approximately 0.06 % (see Figure 32).

Table 13. ASME Code Stress Intensity at Containment Vessel from the 1 m (4 ft) Bottom Corner Drop.

Containment		Closure bolt	
Element	Stress intensity (ksi)	Element	Stress intensity (ksi)
30741	16.51	34991	8.72
30742	9.82	34992	5.14
30743	6.11	34993	4.98
31081	13.07	34994	3.82
31082	11.10	34995	4.46
31083	7.03	34996	7.48
31121	13.02		
31122	10.03		
31123	7.88		
31161	9.55		
31162	8.96		
31163	7.27		
32081	15.22		
32082	11.70		
32083	9.62		
32121	16.52		
32122	15.91		
32123	11.75		
32161	18.58		
32162	15.96		
32163	10.84		
32921	13.68		
32922	6.74		
32923	3.83		

Figure 29. Shielded Drum Package 1 m (4-ft) Bottom Corner Drop Stress Intensity History at the Containment Location (EL 32161).

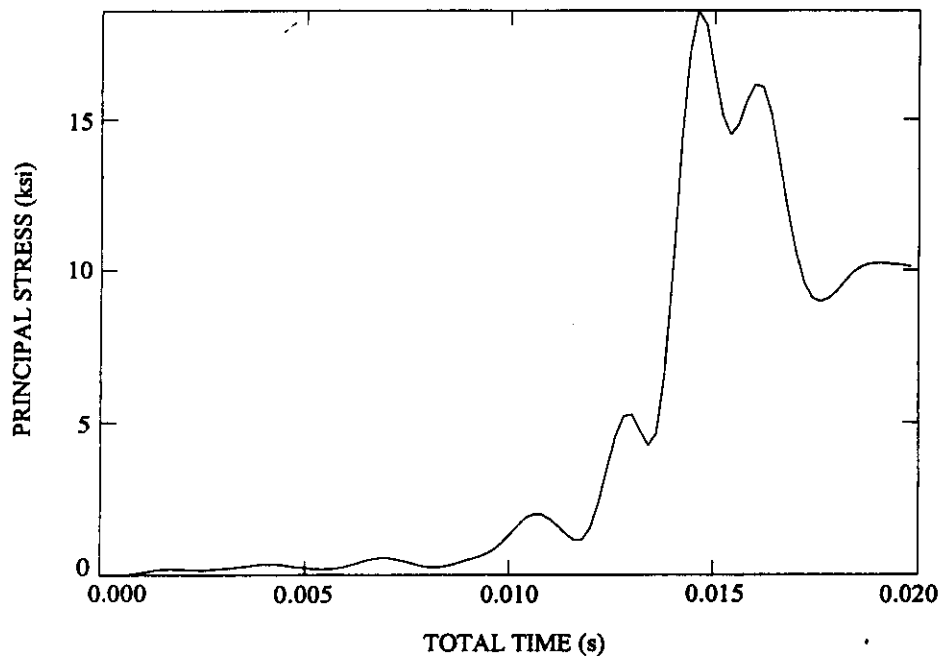


Figure 30. Shielded Drum Package 1 m (4-ft) Bottom Corner Drop Stress Intensity History at Closure Bolt No. 1 (EL 34991).

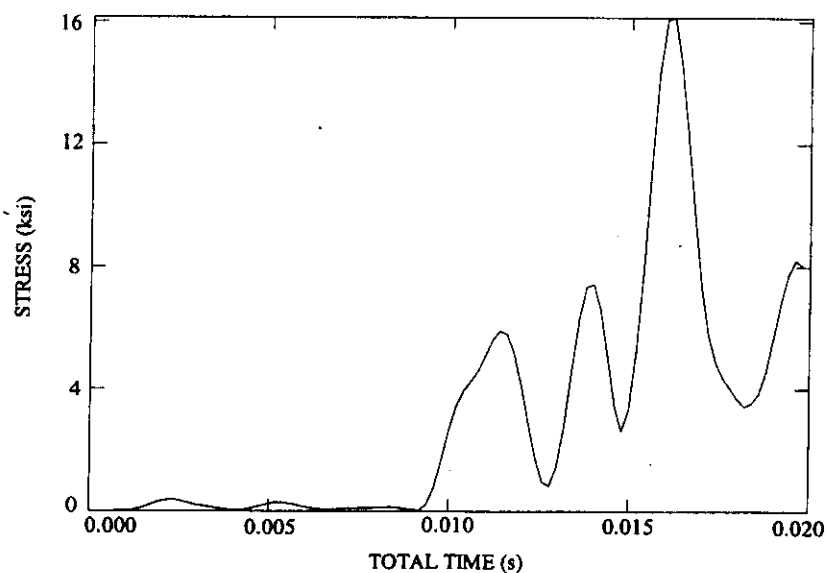


Table 14. Accumulated Peak Strains at Containment Vessel from the 1 m (4 ft) Bottom Corner Drop.

Containment		Closure bolt	
Element	Strain (in/in)	Element	Strain (in/in)
30741	1.245 E-03	34991	5.846 E-04
30742	7.041 E-04	34992	3.451 E-04
30743	5.198 E-04	34993	2.979 E-04
31081	9.985 E-04	34994	2.560 E-04
31082	6.064 E-04	34995	2.992 E-04
31083	5.209 E-04	34996	5.018 E-04
31121	7.052 E-04		
31122	6.477 E-04		
31123	5.374 E-04		
31161	5.918 E-04		
31162	5.783 E-04		
31163	4.942 E-04		
32081	8.338 E-04		
32082	6.753 E-04		
32083	5.929 E-04		
32121	1.746 E-03		
32122	9.072 E-04		
32123	6.847 E-04		
32161	3.260 E-03		
32162	1.111 E-03		
32163	6.358 E-04		
32921	1.247 E-03		
32922	3.794 E-04		
32923	4.315 E-04		

Figure 31. Shielded Drum Package 1 m (4-ft) Bottom Corner Drop Strain at the Containment Location (EL 32161).

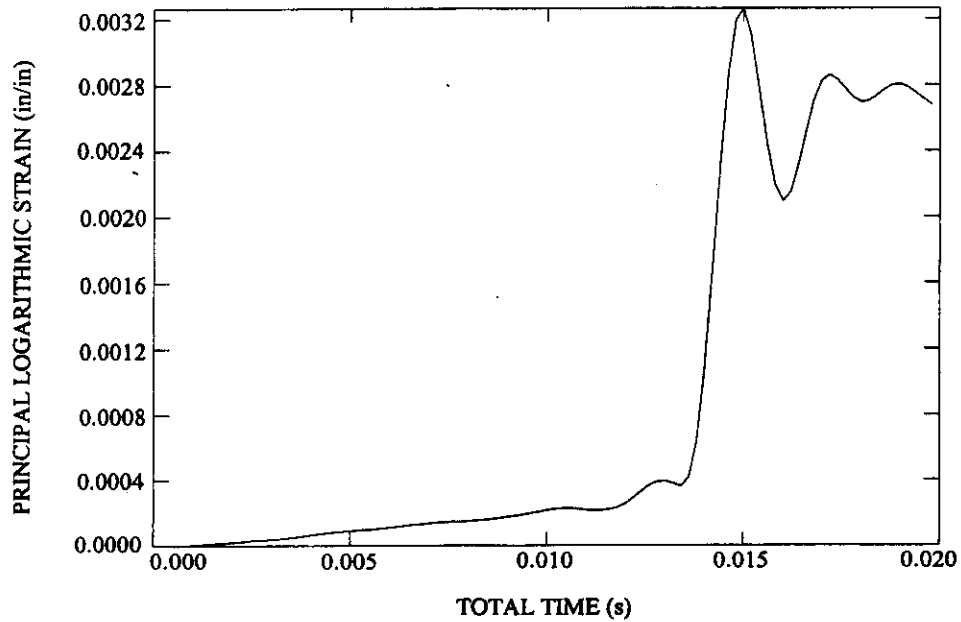
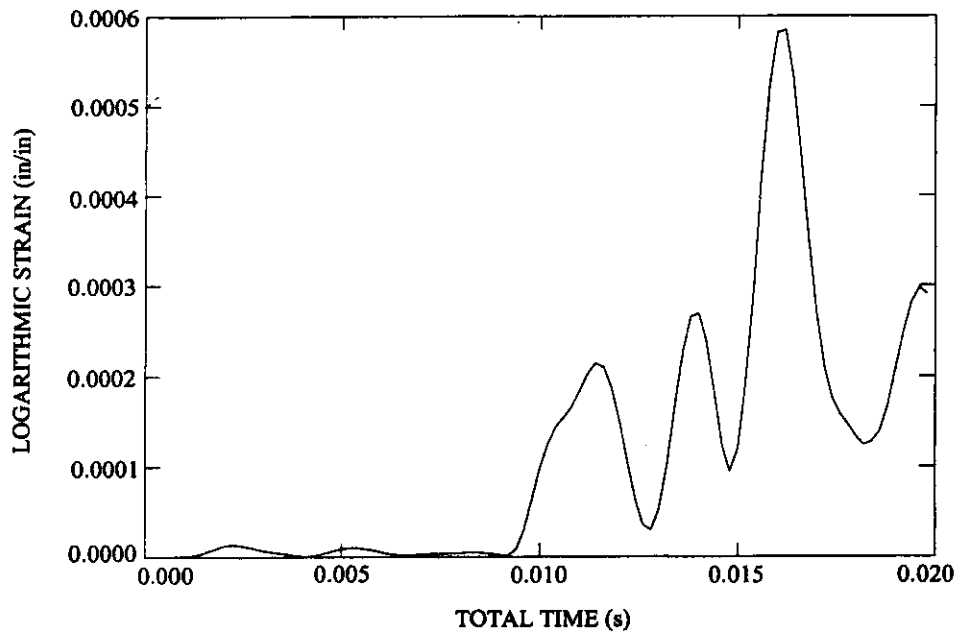


Figure 32. Shielded Drum Package 1 m (4-ft) Bottom Corner Drop Strain at Closure Bolt No. 1 (El 34991).



The stress intensities on the containment vessel are well within Service Level C limits and the cumulative strains are very low. The containment closure bolt stresses are well within Service Level C limits and strains are very low. Figure 33 shows the deformation of the package at the end of the free drop. Figures 34 and 35 show the deformation of the containment vessel, and the inner liner and lead shielding, respectively. Figure 35 shows that the lead shielding deforms, but not severely enough to cause any significant loading on the containment boundary.

Figure 33. Shielded Drum Package 1 m (4-ft) Bottom Corner Drop Entire Package Deformation Plot.

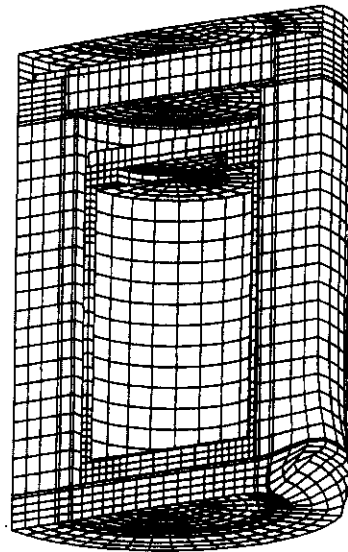


Figure 34. Shielded Drum Package 1 m (4-ft) Bottom Corner Drop Containment Deformation Plot.

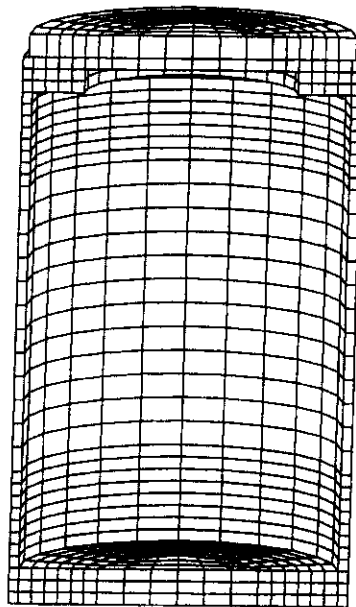
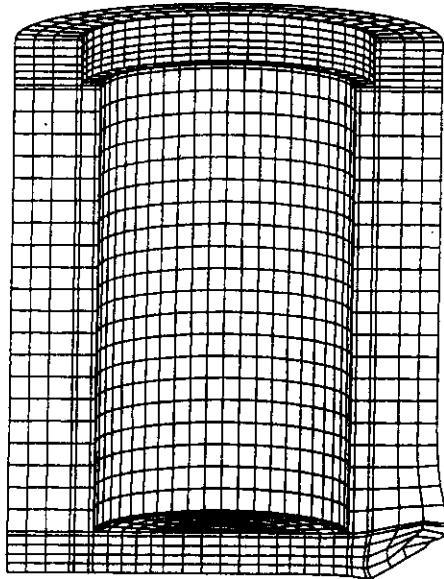


Figure 35. Shielded Drum Package 1 m (4-ft) Bottom Corner Drop Lead Shielding Deformation Plot.



In this 1 m (4 ft) center of gravity over bottom corner drop case, the high stress locations on the containment vessel occur at the bottom-end closure flange, and at the junction of the shell wall and the reverse hub flange. This high stress region is very localized and on the impact side. The stresses and strains at the top portion of the shell wall are slightly higher than those on the bottom-end closure flange. The containment closure lid bolts are not significantly stressed in this case.

5.6.4 Center of Gravity Over Top Corner Free Drop

This case evaluates the shield drum package performance for a 1 m (4 ft) center of gravity over the top corner drop onto a hard unyielding surface (see Figure 36). The total numerical simulation time of this case is 30 ms. The resulting impact deceleration time history is shown in Figure 37. From Figure 37, the peak acceleration is determined to be 80.5gs with a rebound time of 17.30 ms. Based on the rebound time and velocity change, the average deceleration is calculated to be 28.8gs. The velocity time history of the package is shown in Figure 38 and the displacement time history of the package is shown in Figure 39. Pertinent impact acceleration data are shown in Table 15.

Figure 36. Shielded Drum Package 1 m (4 ft) Top Corner Drop.

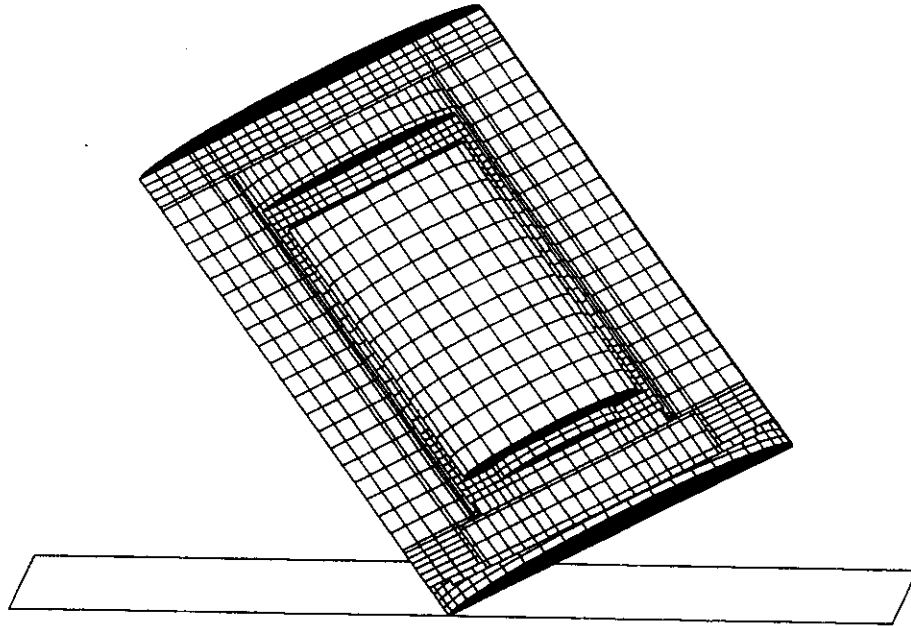


Figure 37. Shielded Drum Package 1 m (4-ft) Top Corner Drop Acceleration History.

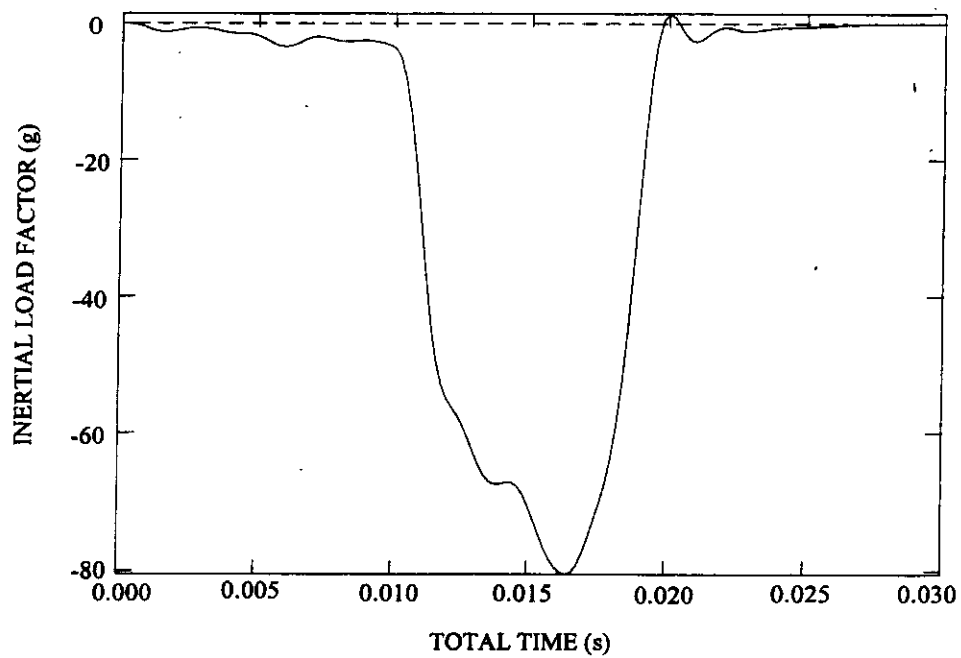


Figure 38. Shielded Drum Package 1 m (4-ft) Top Corner Drop Velocity History.

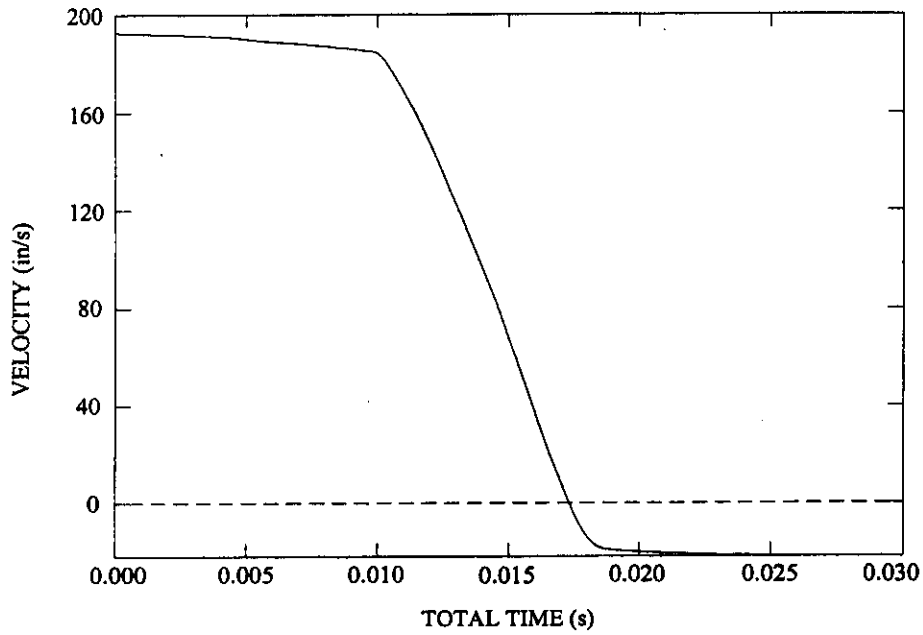


Figure 39. Shielded Drum Package 1 m (4-ft) Top Corner Drop Displacement History.

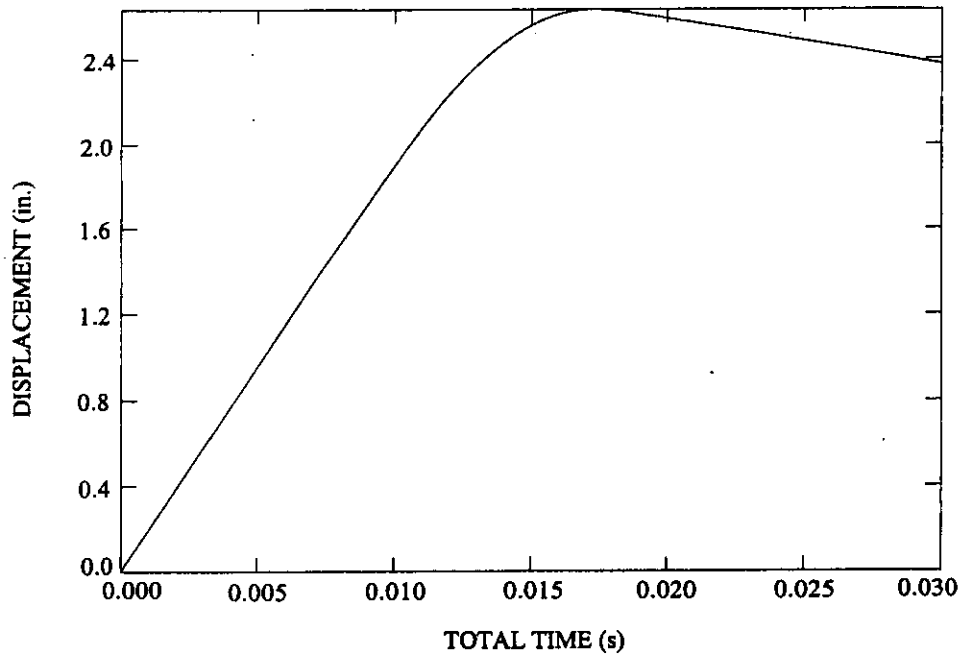


Table 15. Impact Acceleration Data of the 1 m (4 ft) Top Corner Drop.

Case	Average rebound-time acceleration (g)	Peak acceleration (g)	Rebound time (ms)
1 m (4 ft) top corner drop	28.8	80.5	17.30

The ASME BPVC stress intensities at the highly stressed containment vessel locations and all containment closure bolts are shown in Table 16. The stress intensity time history at the highest stress location on the containment vessel (EL 31081) is shown in Figure 40. The stress intensity time history at the highest stress containment closure bolt is shown in Figure 41.

The cumulative strains at the high strain locations on the containment vessel and the containment closure bolts are shown in Table 17. The largest cumulative strain occurs at the junction of the bottom-end closure to the shell wall. However, this cumulative strain is very low, in the range of 0.57 % at EL 31081 as shown in Figure 42. The largest cumulative strain on containment closure bolt is approximately 0.07 % as shown in Figure 43.

Table 16. ASME Code Stress Intensity at Containment Vessel from the 1 m (4 ft) Top Corner Drop.

Containment		Closure bolt	
Element	Stress intensity (ksi)	Element	Stress intensity (ksi)
30741	12.54	34991	10.04
30742	7.95	34992	7.76
30743	3.99	34993	6.86
31081	20.79	34994	6.25
31082	15.21	34995	4.78
31083	9.23	34996	3.12
31121	15.97		
31122	13.88		
31123	10.05		
31161	12.81		
31162	10.37		
31163	7.46		
32081	13.31		
32082	11.03		
32083	8.06		
32121	14.37		
32122	10.63		
32123	7.24		
32161	13.29		
32162	10.46		
32163	7.36		
32921	15.44		
32922	9.10		
32923	5.20		

Figure 40. Shielded Drum Package 1 m (4-ft) Top Corner Drop Stress Intensity History at the Containment Location (EL 31081).

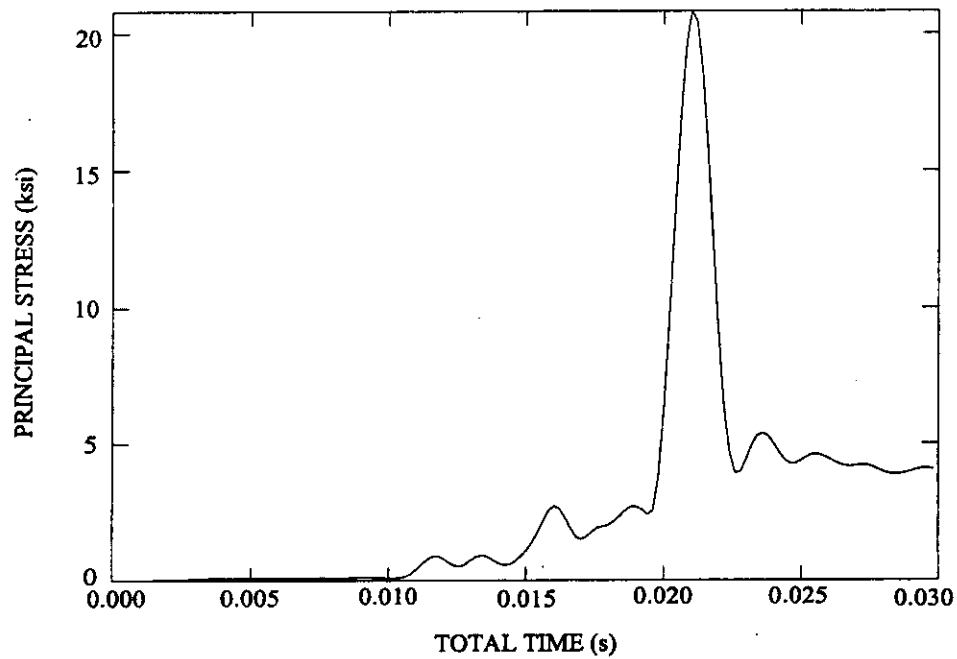


Figure 41. Shielded Drum Package 1 m (4-ft) Top Corner Drop Stress Intensity History at Closure Bolt No. 1 (EL 34991).

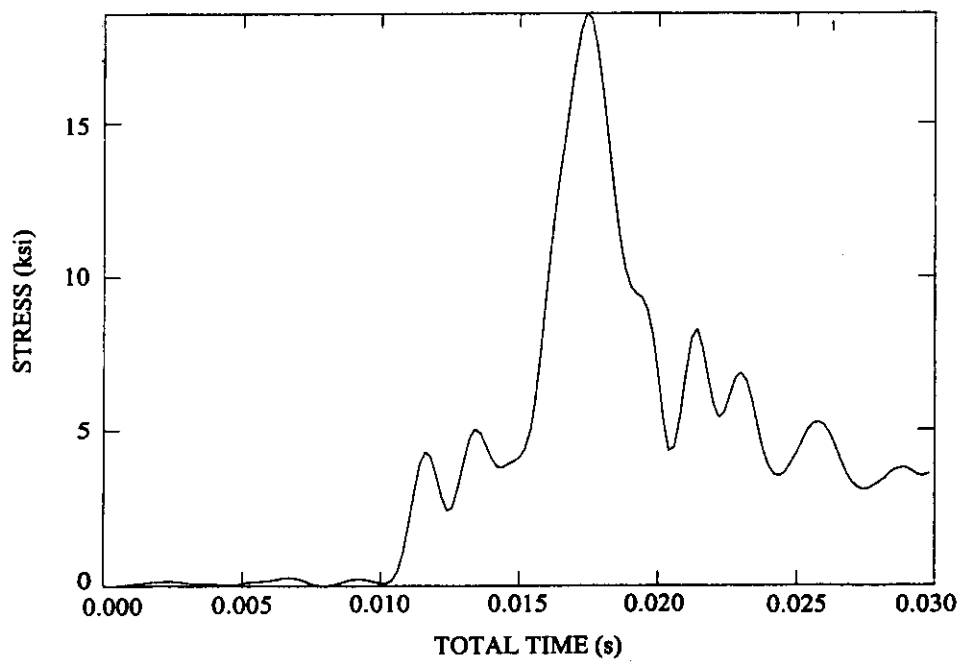


Table 17. Accumulated Peak Strains at Containment Vessel from the 1 m (4 ft) Top Corner Drop.

Containment		Closure bolt	
Element	Strain (in/in)	Element	Strain (in/in)
30741	7.391 E-04	34991	6.737 E-04
30742	7.419 E-04	34992	5.208 E-04
30743	6.847 E-04	34993	4.603 E-04
31081	5.736 E-03	34994	4.196 E-04
31082	1.812 E-03	34995	3.207 E-04
31083	7.005 E-04	34996	2.091 E-04
31121	1.112 E-03		
31122	8.608 E-04		
31123	6.321 E-04		
31161	7.018 E-04		
31162	5.587 E-04		
31163	5.876 E-04		
32081	8.134 E-04		
32082	6.775 E-04		
32083	5.940 E-04		
32121	9.020 E-04		
32122	6.942 E-04		
32123	5.679 E-04		
32161	9.354 E-04		
32162	7.004 E-04		
32163	5.931 E-04		
32921	8.215 E-04		
32922	5.724 E-04		
32923	6.083 E-04		

Figure 42. Shielded Drum Package 1 m (4 ft) Top Corner Drop Strain at the Containment Location (EL 31081).

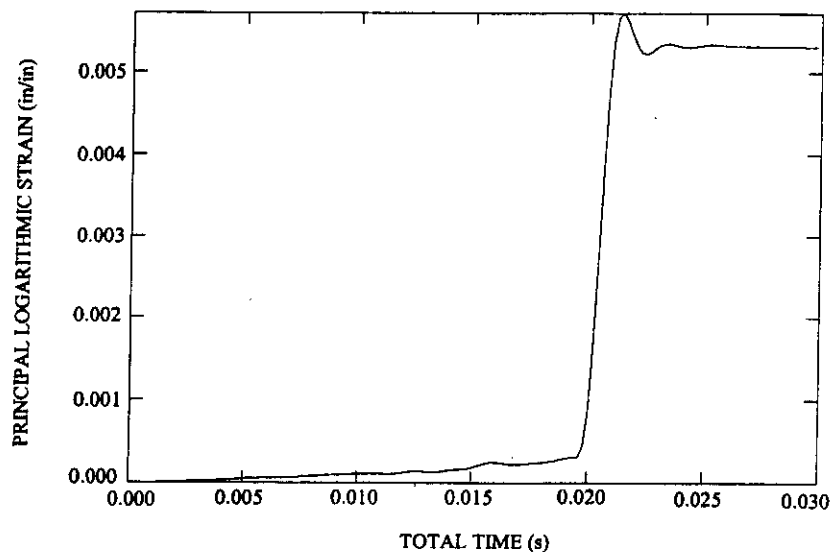
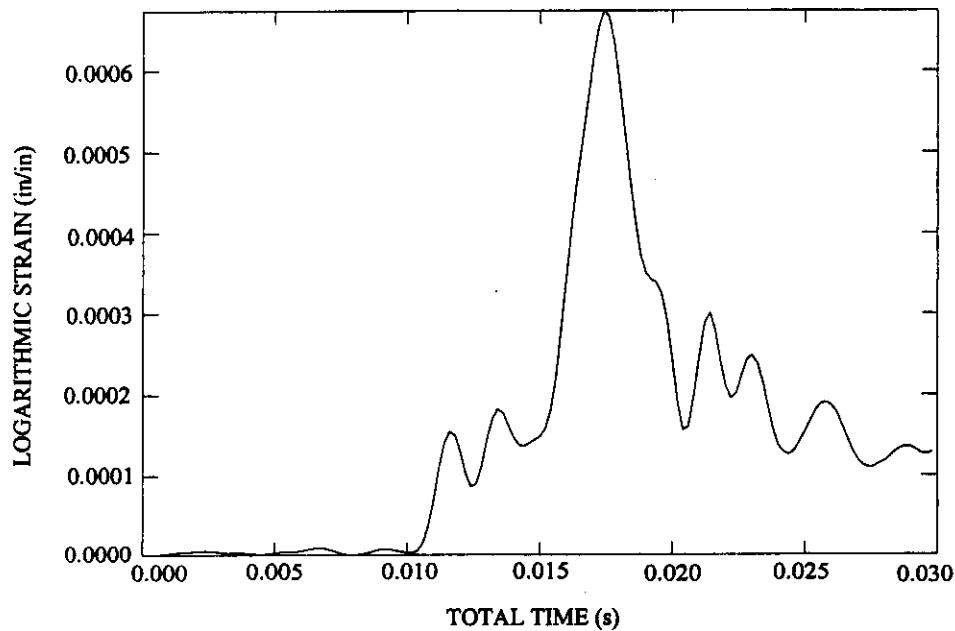


Figure 43. Shielded Drum Package 1 m (4 ft) Top Corner Drop Strain at Closure Bolt No. 1 (EL 34991).



The stresses on the containment vessel are well within Service Level C limits. The cumulative strains are minor and only slightly above the offset yield point. The containment closure bolt stresses are well within Service Level C limits and the strains are very low. Figure 44 shows the deformation of the entire package at the end of the free drop. Figures 45 and 46 show the deformation of the containment vessel, and the inner liner and lead shielding, respectively. The lead shielding deforms, but not severely enough to cause any significant loading of the containment vessel.

In this 1 m (4 ft) center of gravity over top corner drop case, the stress and strain on the containment vessel are low because of the deformation of the lead shielding. The high stress locations on the containment vessel occur at the junction of the shell wall with the bottom-end closure flange. The containment closure lid bolts are not significantly stressed in this case.

Figure 44. Shielded Drum Package 1 m (4 ft) Top Corner Drop Entire Package Deformation Plot.

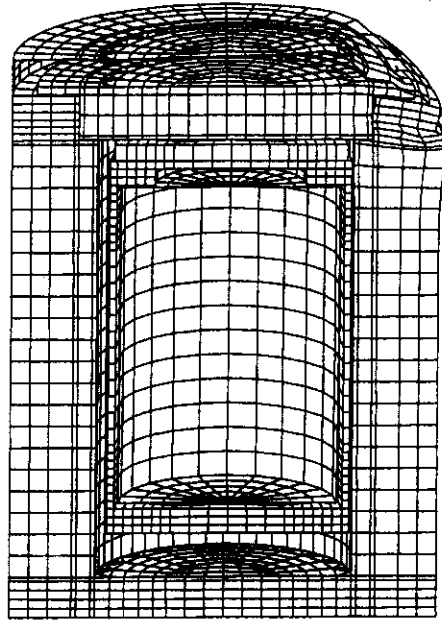


Figure 45. Shielded Drum Package 1 m (4 ft) Top Corner Drop Containment Deformation Plot.

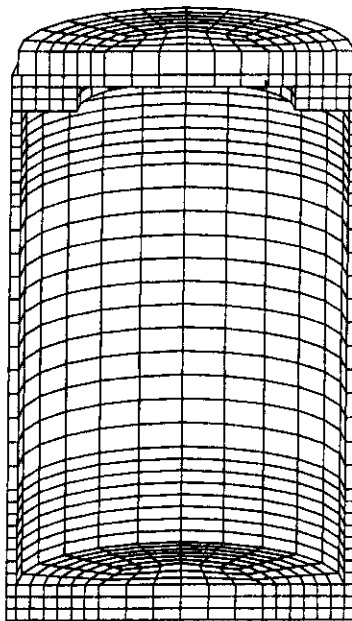
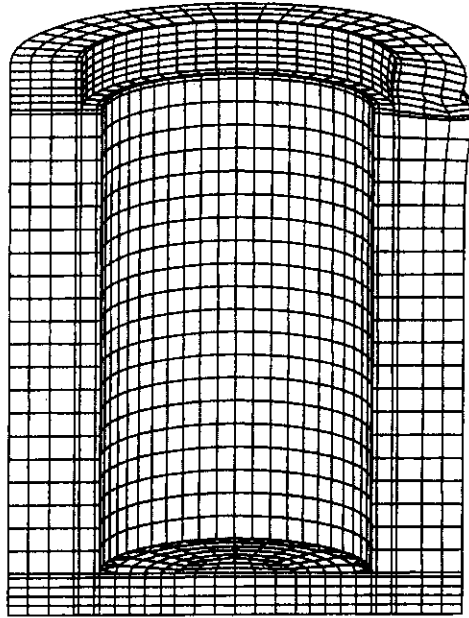


Figure 46. Shielded Drum Package 1 m (4 ft) Top Corner Drop Lead Shielding Deformation Plot.



5.7 9 M (30 FT) AC FREE DROPS

5.7.1 Horizontal Side Free Drop

This case evaluates the typical shielded drum package performance for a 9 m (30 ft) horizontal side free drop onto a horizontal hard unyielding surface (see Figure 47). The total numerical simulation time of the impact is 20 ms. The resulting impact deceleration time history is shown in Figure 48. From Figure 48, the peak deceleration is determined to be 377.7gs with a rebound time of 4.8 ms. Based on the rebound time and the change in velocity, the average deceleration is calculated to be 284.7gs. The velocity time history of the package is shown in Figure 49 and the displacement time history is shown in Figure 50. Pertinent impact acceleration data are shown in Table 18.

Figure 47. Shielded Drum Package 9 m (30 ft) Horizontal Side Drop.

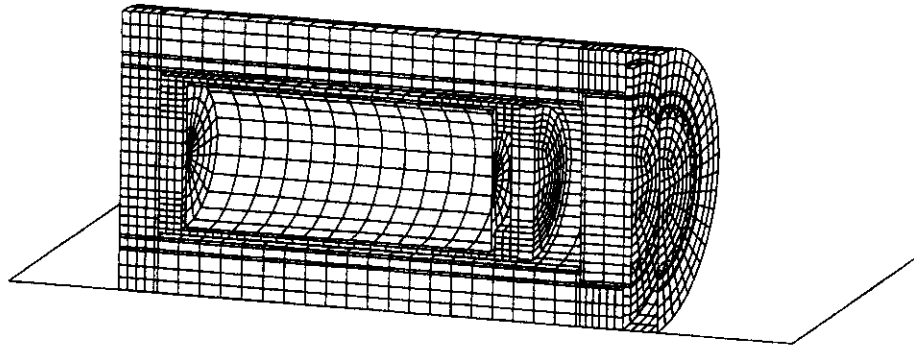


Figure 48. Shielded Drum Package 9 m (30-ft) Horizontal Side Drop Acceleration History.

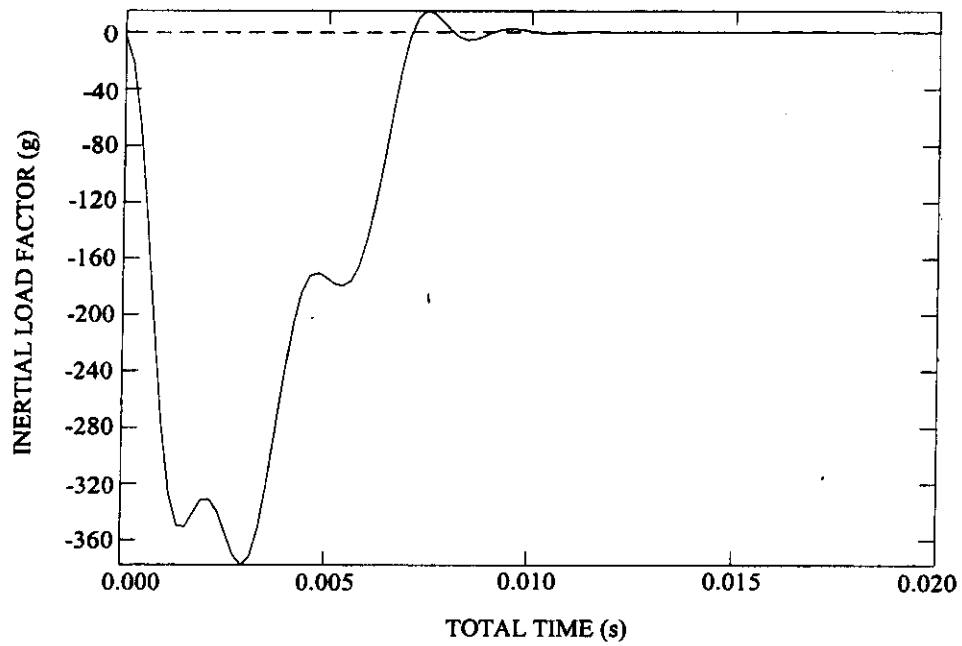


Figure 49. Shielded Drum Package 9 m (30-ft) Horizontal Side Drop Velocity History.

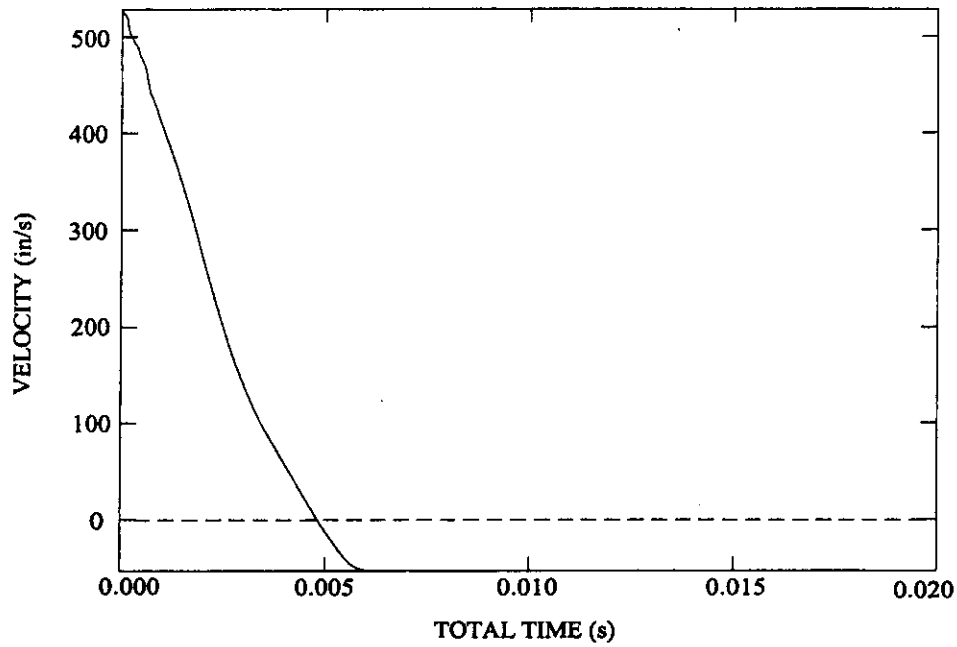


Figure 50. Shielded Drum Package 9 m (30-ft) Horizontal Side Drop Displacement History.

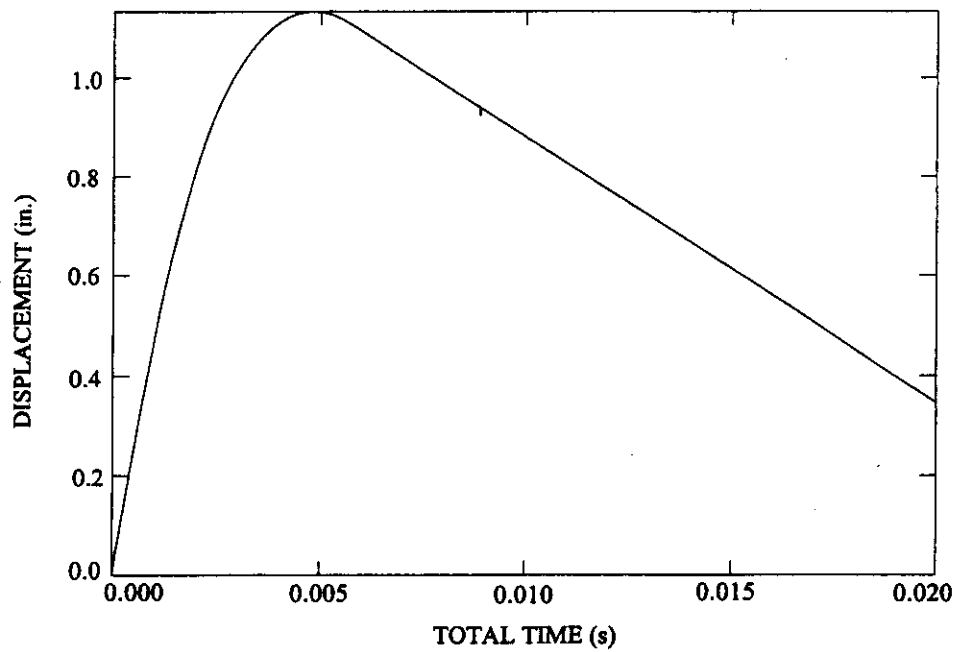


Table 18. Impact Acceleration Data of the 9 m (30 ft) Horizontal Side Drop.

Case	Average rebound-time acceleration (g)	Peak acceleration (g)	Rebound time (ms)
9 m (30 ft) horizontal side drop	284.7	377.7	4.8

The ASME BPVC stress intensities at high stress locations on the containment vessel and at all containment closure bolts are shown in Table 19. The stress intensity time history at the most severely stressed location of the containment vessel is shown in Figure 51. The stress intensity time history at the most highly stressed containment closure bolt is shown in Figure 52.

The cumulative strains at the high strain locations on the containment vessel and at all containment closure bolts are shown in Table 20. The highest cumulative strain occurs at the juncture of the shell wall and the bottom-end plate. However, this cumulative strain is only slightly above the yield strain with a value of approximately 5.6%. A typical cumulative strain time history at EL 31081 is shown in Figure 53. The highest cumulative strain on the containment closure bolts is approximately 0.2% as shown in Figure 54.

Table 19. ASME Code Stress Intensity at Containment Vessel from the 9 m (30 ft) Horizontal Side Drop.

Containment		Closure bolt	
Element	Stress intensity (ksi)	Element	Stress intensity (ksi)
30741	38.80	34991	29.98
30742	28.74	34992	10.34
30743	29.92	34993	3.85
31081	43.23	34994	8.48
31082	41.92	34995	11.45
31083	41.66	34996	15.06
31121	34.04		
31122	38.01		
31123	41.97		
31161	30.73		
31162	35.44		
31163	40.25		
32081	25.93		
32082	24.41		
32083	28.43		
32121	28.21		
32122	26.98		
32123	28.79		
32161	29.98		
32162	25.57		
32163	27.27		
32921	34.88		
32922	22.35		
32923	22.33		

Figure 51. Shielded Drum Package 9 m (30-ft) Horizontal Side Drop Stress Intensity History at Containment Location (EL 31081).

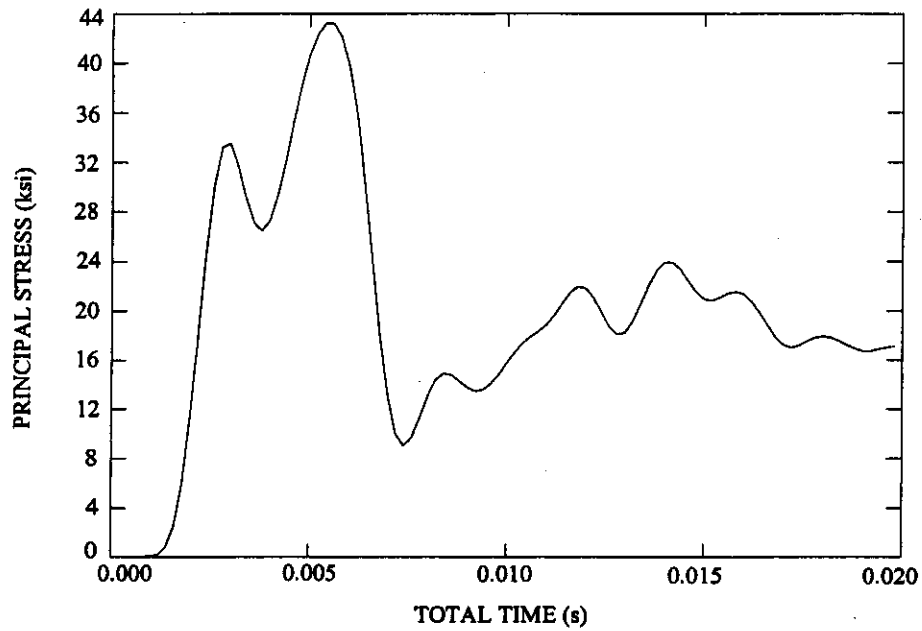


Figure 52. Shielded Drum Package 9 m (30-ft) Horizontal Side Drop Stress Intensity History at Closure Bolt No. 1 (El 34991).

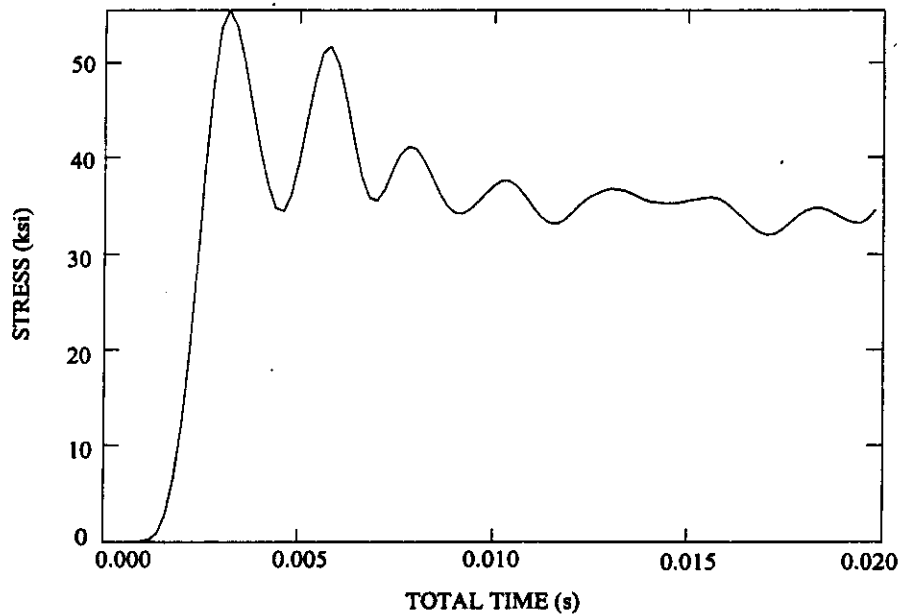


Table 20. Accumulated Peak Strains at Containment Vessel from the 9 m (30 ft) Horizontal Side Drop.

Containment		Closure bolt	
Element	Strain (in/in)	Element	Strain (in/in)
30741	3.680 E-02	34991	2.013 E-03
30742	7.566 E-03	34992	6.936 E-04
30743	8.739 E-03	34993	2.583 E-04
31081	5.416 E-02	34994	5.693 E-04
31082	4.878 E-02	34995	7.685 E-04
31083	5.193 E-02	34996	1.010 E-03
31121	1.351 E-02		
31122	3.204 E-02		
31123	5.554 E-02		
31161	8.941 E-03		
31162	2.098 E-02		
31163	4.285 E-02		
32081	4.765 E-03		
32082	3.105 E-03		
32083	8.937 E-03		
32121	7.959 E-03		
32122	5.090 E-03		
32123	9.211 E-03		
32161	8.910 E-03		
32162	5.581 E-03		
32163	7.203 E-03		
32921	2.652 E-02		
32922	3.033 E-03		
32923	1.419 E-03		

Figure 53. Shielded Drum Package 9 m (30 ft) Horizontal Side Drop Strain at Containment Location (EL 31081).

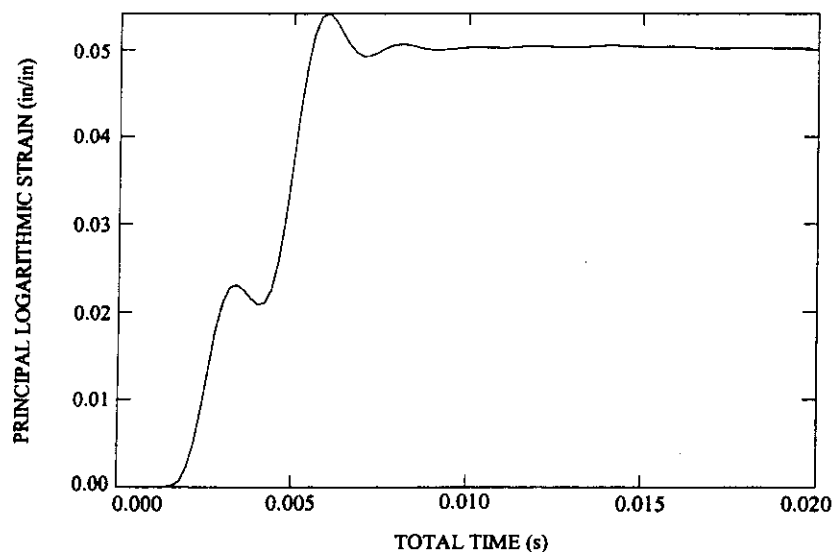
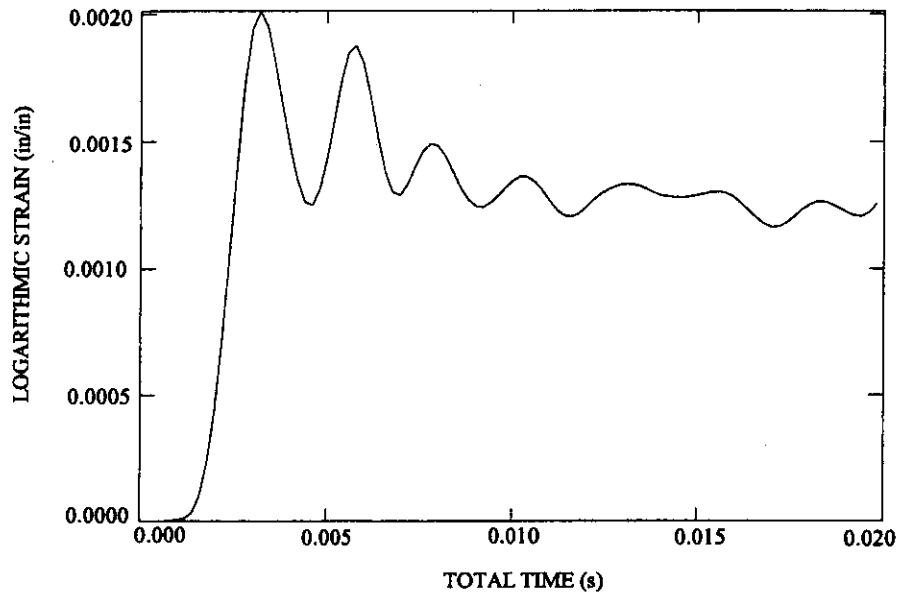


Figure 54. Shielded Drum Package 9 m (30 ft) Horizontal Side Drop Strain at Closure Bolt No. 1 (EL 34991).



The SEDs at critical regions on the containment vessel and all containment closure bolts are tabulated in Table 21. Figure 55 shows the SED time history at the most severely loaded location, EL 31081. Figure 56 shows the SED time history at the most severely loaded closure bolt element, EL 34991. Figure 57 shows the damage and deformations to the package and containment vessel at the end of the impact. Figures 58 through 60 are the triaxiality factor (TF), PEEQ, and the composite TF and PEEQ time history profiles at element EL 31081, respectively.

The SEDs on the containment vessel are well below the maximum material toughness. The cumulative strains are also well below the ultimate material strain and, as shown in Figure 60, are primarily compressive. The bolt stresses and strains are also well below the maximum material allowable values for both SED and TSL.

In this 9 m (30 ft) side free drop case, the high loading locations on the containment vessel occur at the junction of bottom-end plate and shell wall at the area of impact. Longitudinal bending of the shell wall is also observed. However, this high loading is very localized at the discontinuities and concentrated around the impact edge and joint.

Table 21. Strain Energy Density at Containment Vessel from the 9 m (30 ft) Horizontal Side Drop.

Containment		Closure bolt	
Element	Strain energy density (in-lb/in ³)	Element	Strain energy density (in-lb/in ³)
30741	6.213 E+02	34991	3.094 E+01
30742	1.010 E+02	34992	4.555 E+00
30743	1.302 E+02	34993	1.530 E+00
31081	1.133 E+03	34994	6.836 E+00
31082	9.758 E+02	34995	6.799 E+00
31083	9.131 E+02	34996	1.157 E+01
31121	3.026 E+02		
31122	5.854 E+02		
31123	1.033 E+03		
31161	1.520 E+02		
31162	3.384 E+02		
31163	7.616 E+02		
32081	9.085 E+01		
32082	4.366 E+01		
32083	1.123 E+02		
32121	1.954 E+02		
32122	1.064 E+02		
32123	1.260 E+02		
32161	2.259 E+02		
32162	1.155 E+02		
32163	1.058 E+02		
32921	4.119 E+02		
32922	4.730 E+01		
32923	1.230 E+01		

Figure 55. Shielded Drum Package 9 m (30 ft) Horizontal Side Drop Strain Energy Density at Containment Location (EL 31081).

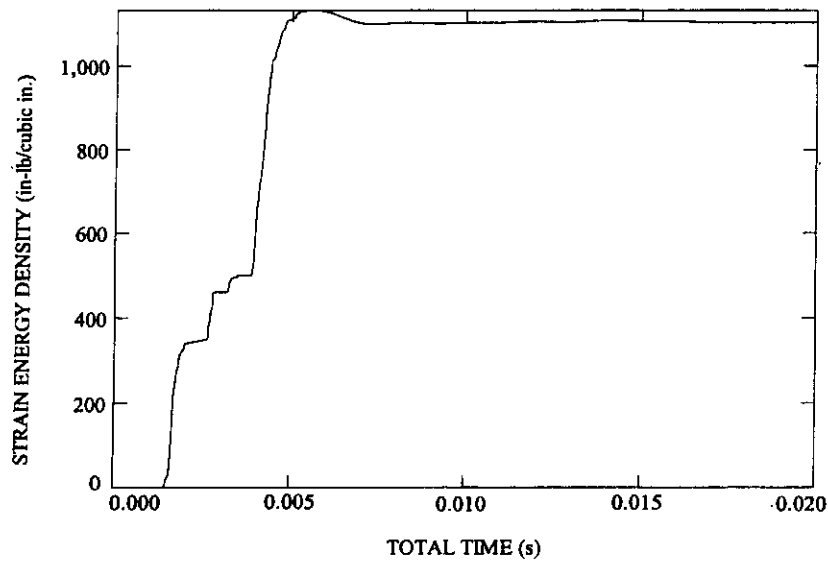


Figure 56. Shielded Drum Package 9 m (30 ft) Horizontal Side Drop Strain Energy Density at Closure Bolt No. 1 (EL 34991).

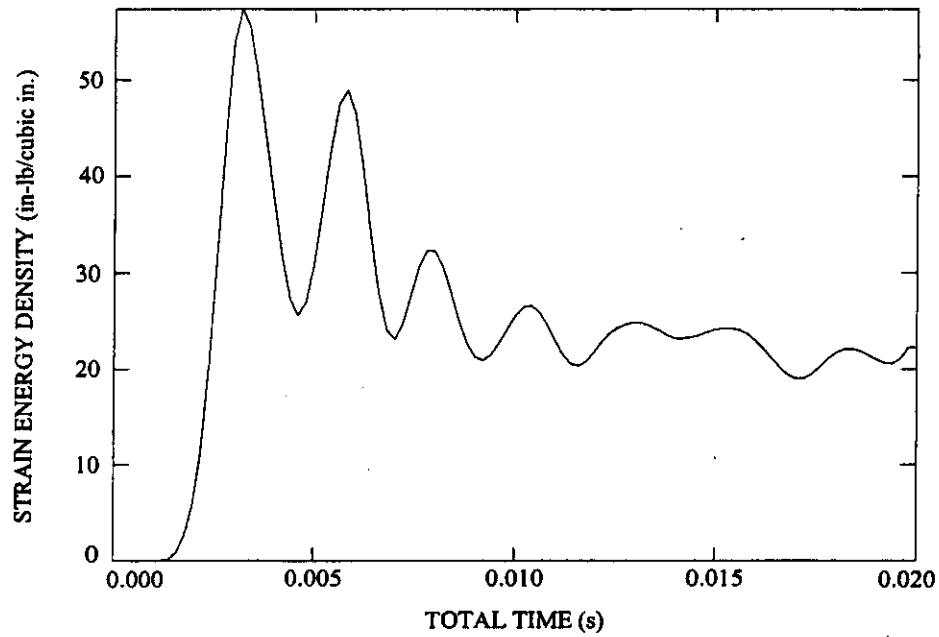


Figure 57. Shielded Drum Package 9 m (30 ft) Horizontal Side Drop Deformation Plot.

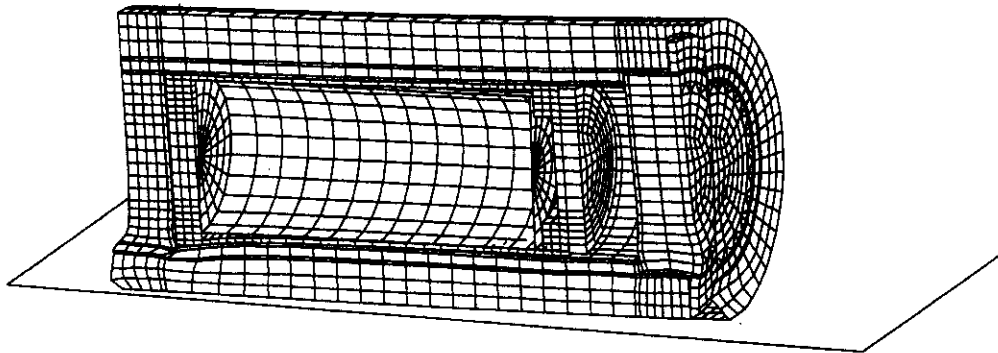


Figure 58. Triaxiality Factor Plot at EL 31081.

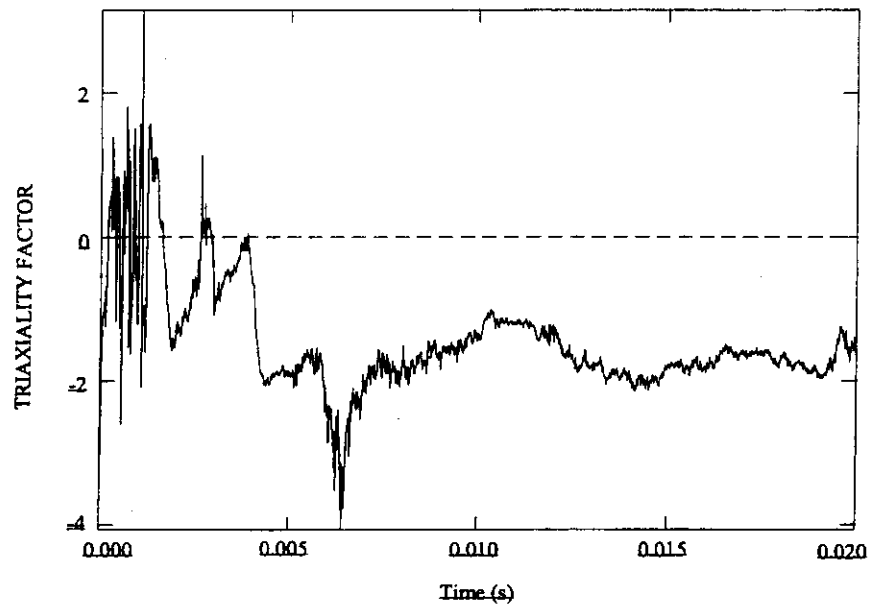


Figure 59. Equivalent Plastic Strain Plot at El 31081.

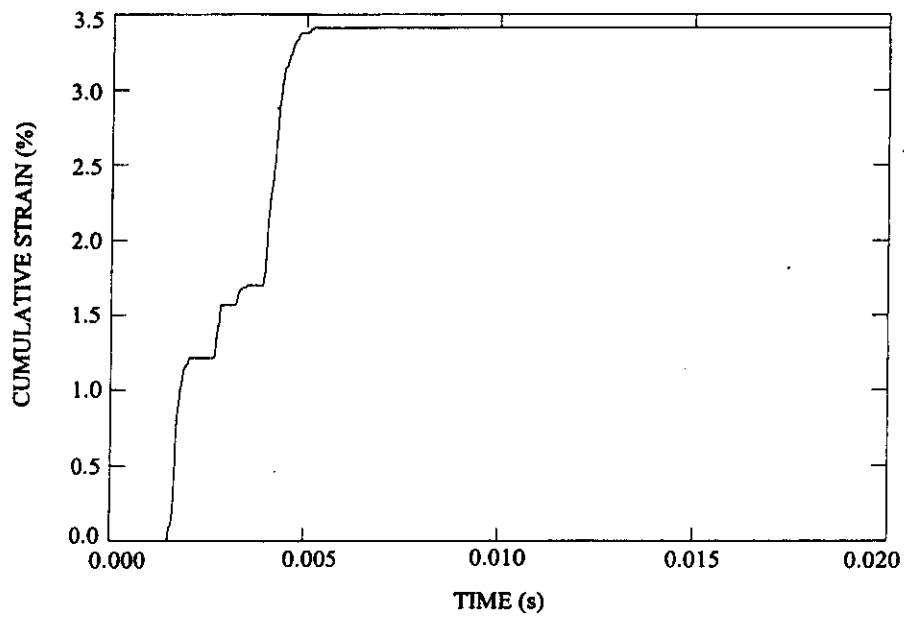
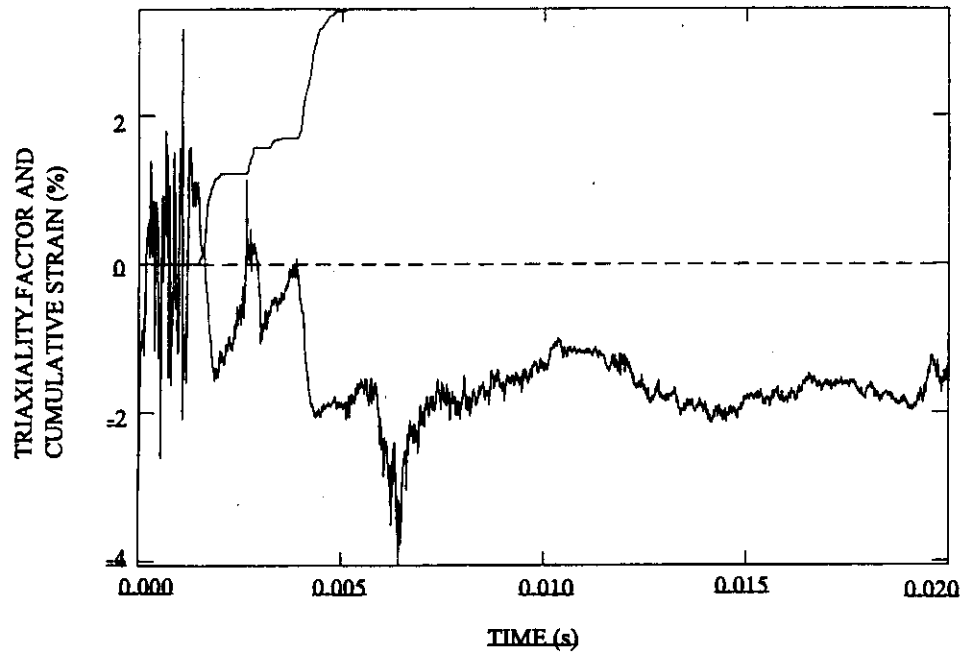


Figure 60. Combined Triaxiality Factor and Equivalent Plastic Strain Plot at EL 31081.



5.7.2 Flat Bottom-End Free Drop

This case simulates the typical shielded drum package performance for a 9 m (30 ft) flat bottom-end free drop onto a hard unyielding surface (see Figure 61). The total numerical simulation time of this case is 20 ms. The resulting deceleration time history is shown in Figure 62. From Figure 62, the peak deceleration is determined to be 349.6gs with a rebound time of 3.8 ms. Based on the rebound time and velocity change, the average deceleration is calculated to be 359.6gs. The velocity time history of the package is shown in Figure 63 and the displacement time history of the package is shown in Figure 64. Pertinent impact acceleration data are shown in Table 22.

Figure 61. Shielded Drum Package 9 m (30 ft) Bottom-End Drop.

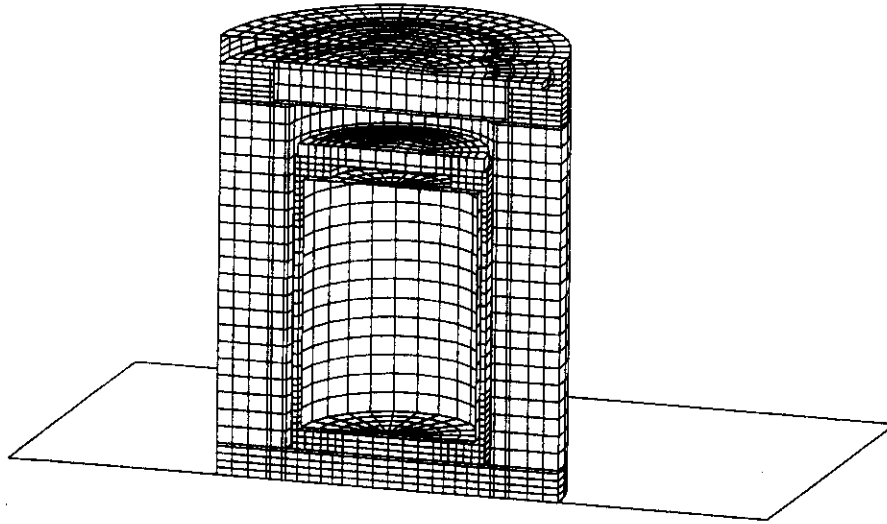


Figure 62. Shielded Drum Package 9 m (30 ft) Bottom-End Drop Acceleration History.

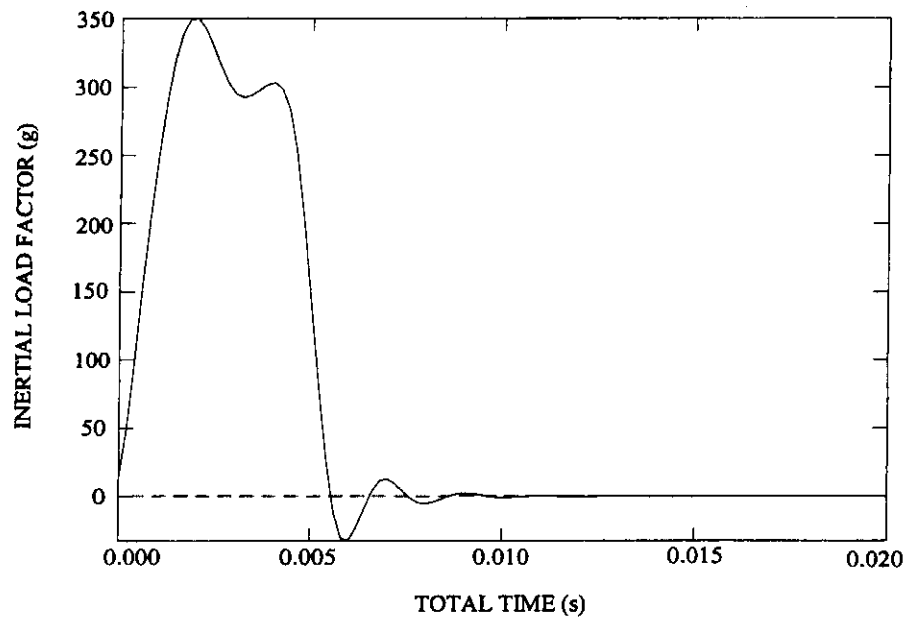


Figure 63. Shielded Drum Package 9 m (30 ft) Bottom-End Drop Velocity History.

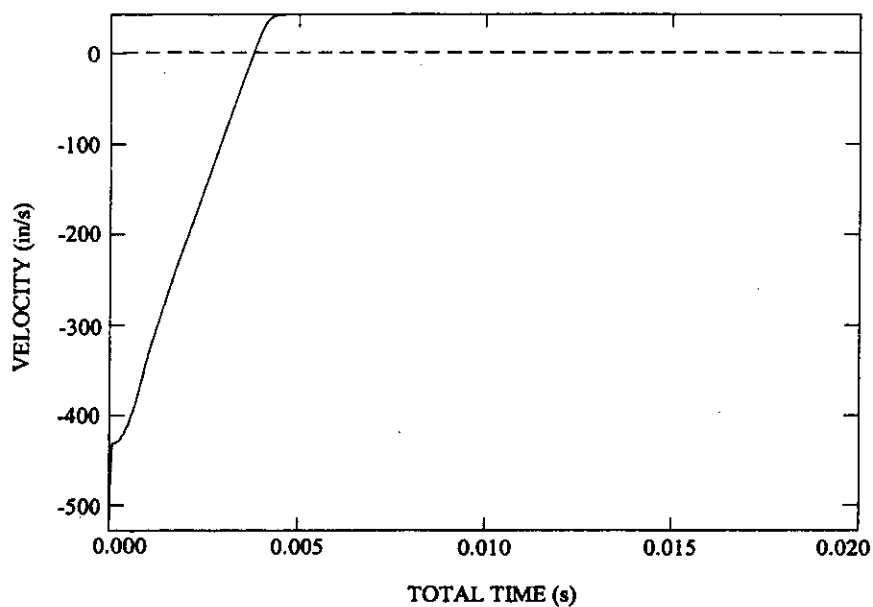


Figure 64. Shielded Drum Package 9 m (30 ft) Bottom-End Drop Displacement History.

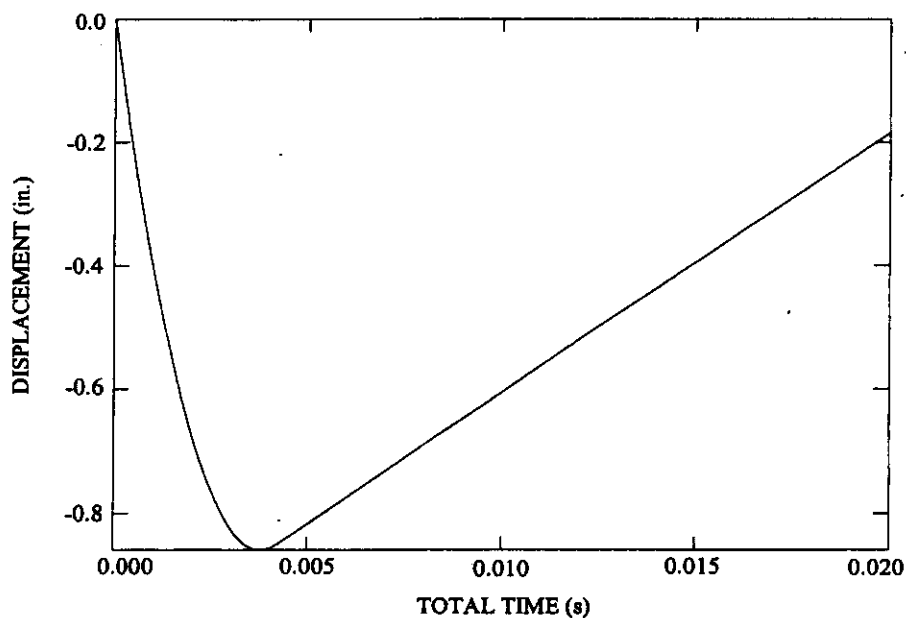


Table 22. Impact Acceleration Data of the 9 m (30 ft) Bottom-End Drop.

Case	Average rebound-time acceleration (g)	Peak acceleration (g)	Rebound time (ms)
9 m (30 ft) bottom-end drop	359.6	349.6	3.8

The ASME BPVC stress intensities at highly stressed containment vessel locations and at all containment closure bolts are shown in Table 23. The stress intensity time history at a typical high-stress location of the containment vessel (EL 31081) is shown in Figure 65. The stress intensity time history of a typical closure bolt is shown in Figure 66.

The cumulative strains at the highly strained locations of the containment vessel and at all containment closure bolts are shown in Table 24. The largest cumulative strain on the containment vessel occurs in the shell wall juncture to the bottom-end closure flange. This cumulative strain is only in the range of 4.6 %, which is only slightly above the offset yield point of the material. Shell bending at this location extends to at least three thicknesses from the weld. A typical cumulative strain time history at EL 31081 is shown in Figure 67. The largest cumulative strain on the containment closure bolts is approximately 0.83 % as shown in Figure 68.

Table 23. ASME Code Stress Intensity at Containment Vessel from the 9 m (30 ft) Bottom-End Drop.

Containment		Closure bolt	
Element	Stress intensity (ksi)	Element	Stress intensity (ksi)
30603	20.72	34991	13.10
30604	21.12	34992	11.56
30609	23.08	34993	12.08
30610	24.00	34994	12.54
31081	35.69	34995	11.70
31082	36.93	34996	12.86
31083	37.80		
31084	37.57		
31089	37.71		
31090	37.66		
31091	37.24		
31092	36.43		
31121	27.38		
31122	27.91		
31131	29.07		
31132	29.20		
31161	23.14		
31162	22.73		
31171	22.31		
31172	23.15		
32161	17.36		
32162	17.13		
32171	16.90		
32172	17.51		
32841	13.93		
32842	13.88		
32851	14.68		
32852	11.53		

Figure 65. Shielded Drum Package 9 m (30 ft) Bottom-End Drop Stress Intensity History at the Containment Location (EL 31081).

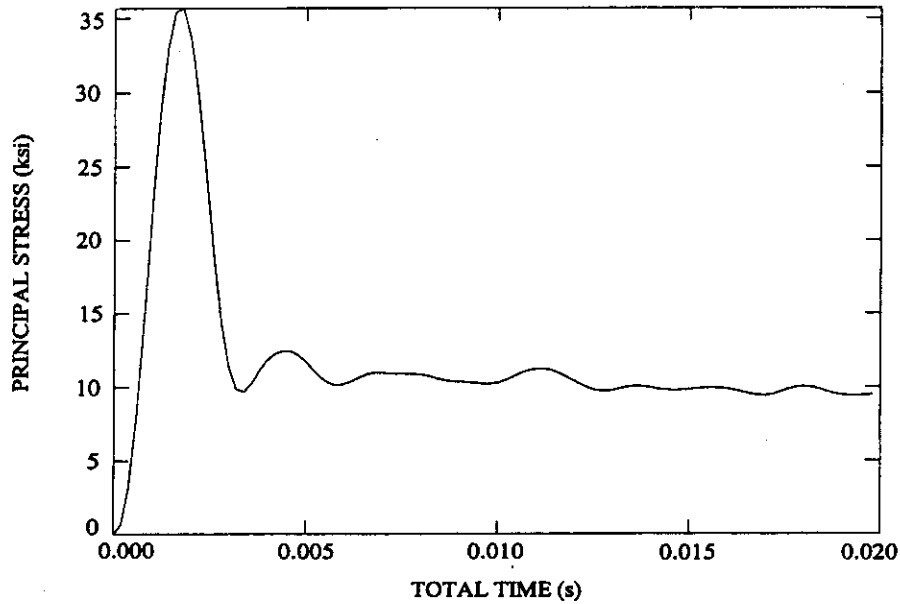


Figure 66. Shielded Drum Package 9 m (30 ft) Bottom-End Drop Stress Intensity History at Closure Bolt No. 6 (EL 34996).

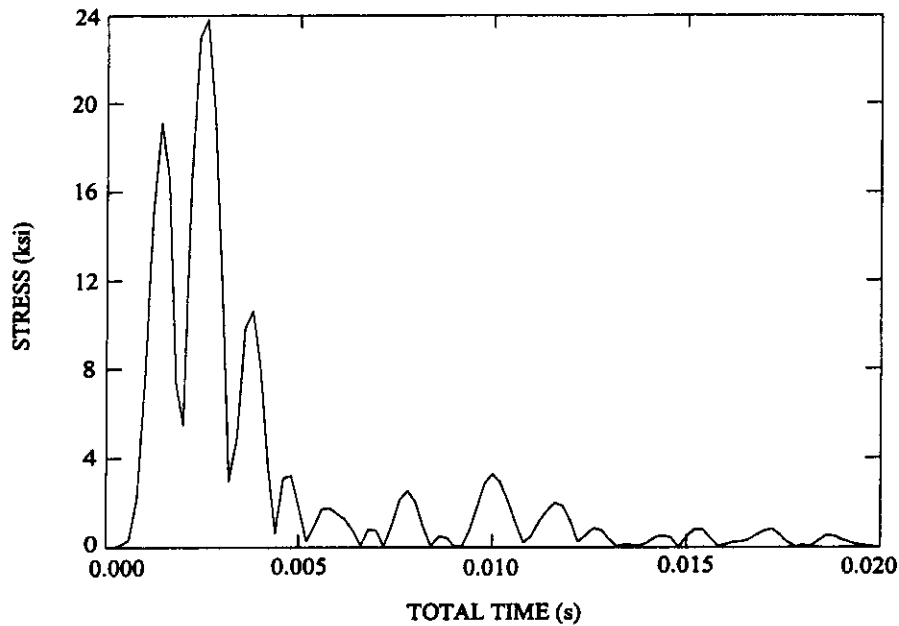


Table 24. Accumulated Peak Strains at Containment Vessel from the 9 m (30 ft) Bottom-End Drop.

Containment		Closure bolt	
Element	Strain (in/in)	Element	Strain (in/in)
30603	3.225 E-02	34991	8.158 E-03
30604	3.222 E-02	34992	8.085 E-03
30609	2.743 E-02	34993	8.204 E-03
30610	2.732 E-02	34994	8.261 E-03
31081	4.556 E-02	34995	8.213 E-03
31082	4.528 E-02	34996	8.311 E-03
31083	4.510 E-02		
31084	4.513 E-02		
31089	4.521 E-02		
31090	4.517 E-02		
31091	4.530 E-02		
31092	4.563 E-02		
31121	4.366 E-02		
31122	4.335 E-02		
31131	4.353 E-02		
31132	4.390 E-02		
31161	3.575 E-02		
31162	3.551 E-02		
31171	3.563 E-02		
31172	3.567 E-02		
32161	2.631 E-03		
32162	2.614 E-03		
32171	2.691 E-03		
32172	2.632 E-03		
32841	6.906 E-03		
32842	4.151 E-03		
32851	6.330 E-03		
32852	8.245 E-03		

Figure 67. Shielded Drum Package 9 m (30 ft) Bottom-End Drop Strain at the Containment Location (EL 31081).

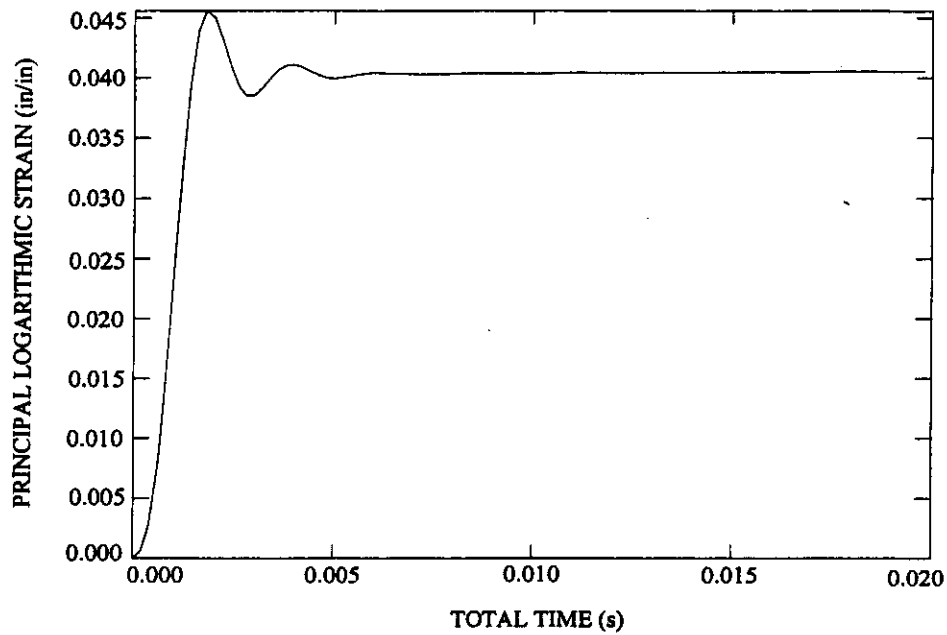
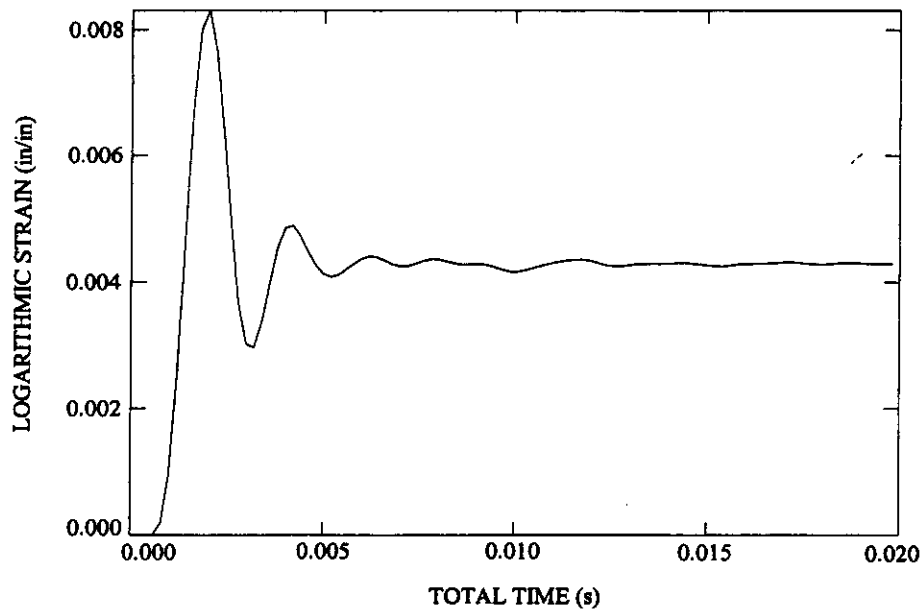


Figure 68. Shielded Drum Package 9 m (30 ft) Bottom-End Drop Strain at Closure Bolt No. 6 (El 34996).



The SEDs in critical regions of the containment vessel and all containment closure bolts are presented in Table 25. Figure 69 shows the SED time history at EL 31081. Figure 70 shows the SED time history at containment closure bolt element EL 34996. Figure 71 shows the deformation of the entire shielded drum package at the end of the free drop. Figures 72 and 73 show the deformation of the containment vessel, drum, and lead shielding at the end of the free drop, respectively. Figures 74 through 76 are the TF, the PEEQ at EL 31081, and the TF and PEEQ overlay time history profiles at EL 31081, respectively.

The SEDs on the containment vessel are well below the maximum material toughness. The cumulative strains are also well below the ultimate material strain and, as shown in Figure 74, are primarily compressive. The bolt stresses and strains are also well below the maximum material allowable values for both SED and TSL.

Table 25. Strain Energy Density at Containment Vessel
from the 9 m (30 ft) Bottom-End Drop.

Containment		Closure bolt	
Element	Strain energy density (in-lb/in ³)	Element	Strain energy density (in-lb/in ³)
30603	5.913 E+02	34991	1.334 E+03
30604	5.896 E+02	34992	1.351 E+03
30609	4.765 E+02	34993	1.361 E+03
30610	4.744 E+02	34994	1.362 E+03
31081	8.050 E+02	34995	1.361 E+03
31082	7.993 E+02	34996	1.365 E+03
31083	7.958 E+02		
31084	7.960 E+02		
31089	7.976 E+02		
31090	7.972 E+02		
31091	8.009 E+02		
31092	8.100 E+02		
31121	7.536 E+02		
31122	7.441 E+02		
31131	7.487 E+02		
31132	7.633 E+02		
31161	5.876 E+02		
31162	5.795 E+02		
31171	5.825 E+02		
31172	5.871 E+02		
32161	5.915 E+01		
32162	5.742 E+01		
32171	5.955 E+01		
32172	5.804 E+01		
32841	1.068 E+02		
32842	8.440 E+01		
32851	1.097 E+02		
32852	1.203 E+02		

Figure 69. Shielded Drum Package 9 m (30 ft) Bottom-End Drop Strain Energy Density at the Containment Location (EL 31081).

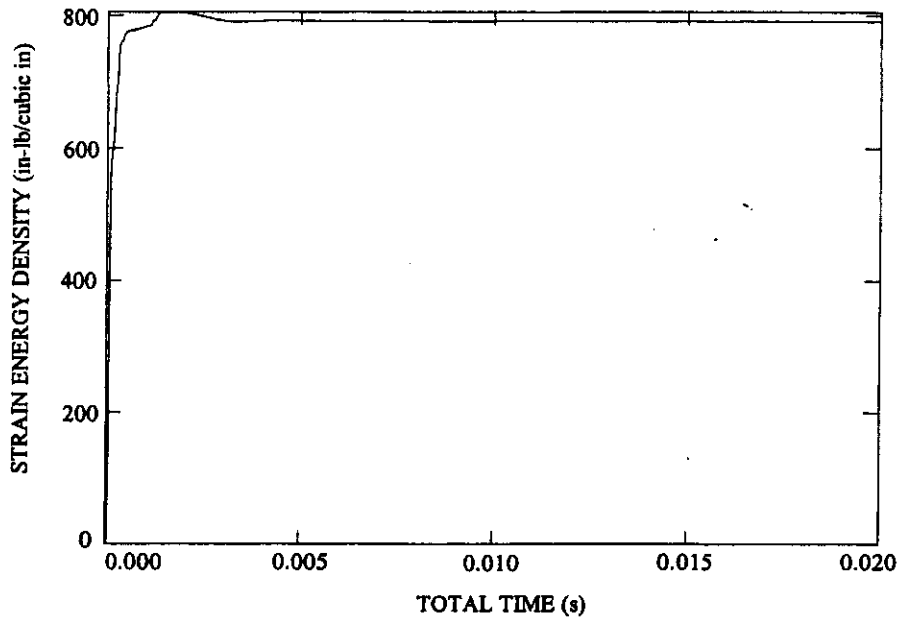


Figure 70. Shielded Drum Package 9 m (30 ft) Bottom-End Drop Strain Energy Density at Closure Bolt No. 6 (El 34996).

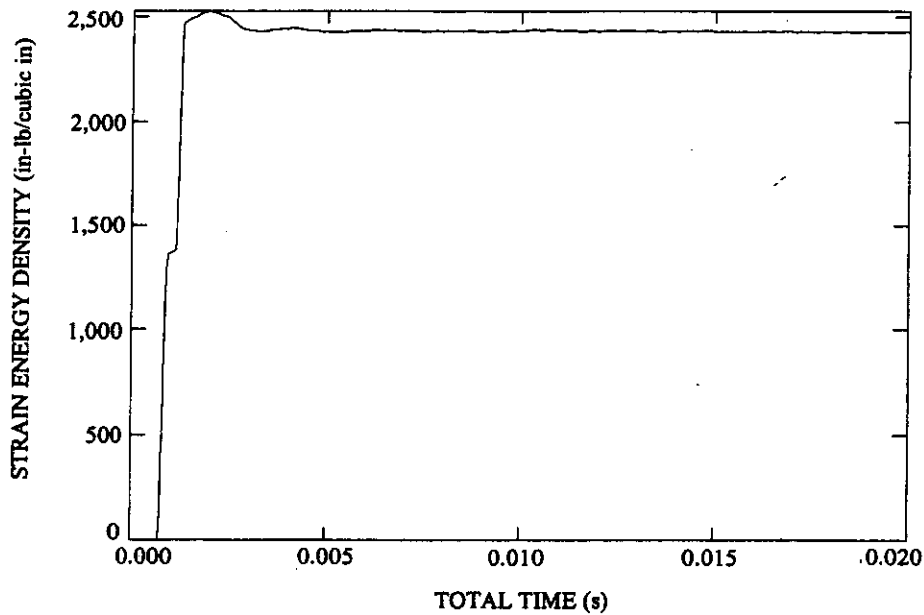


Figure 71. Shielded Drum Package 9 m (30 ft) Bottom-End Drop Entire Package Deformation Plot.

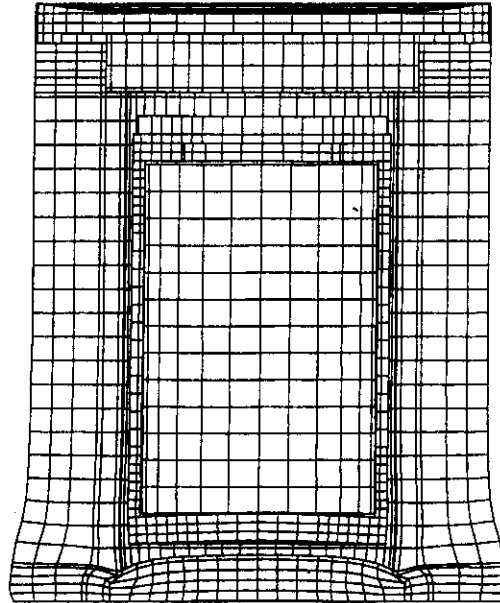


Figure 72. Shielded Drum Package 9 m (30 ft) Bottom-End Drop Containment Deformation Plot.

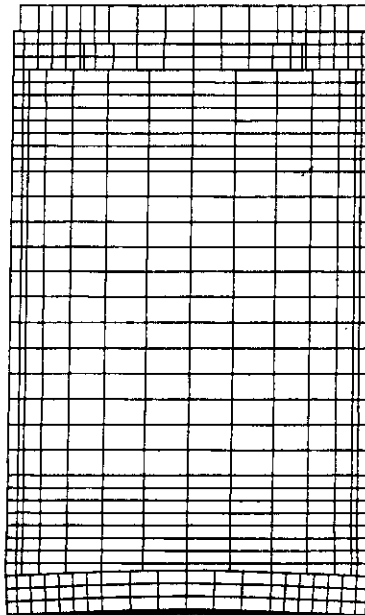


Figure 73. Shielded Drum Package 9 m (30 ft) Bottom-End Drop Inner Liner and Lead Shielding Deformation Plot.

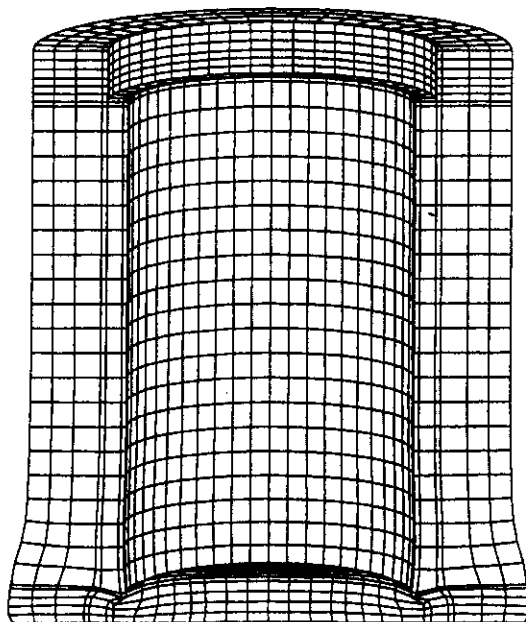


Figure 74. Triaxility Factor Plot at EL 31081.

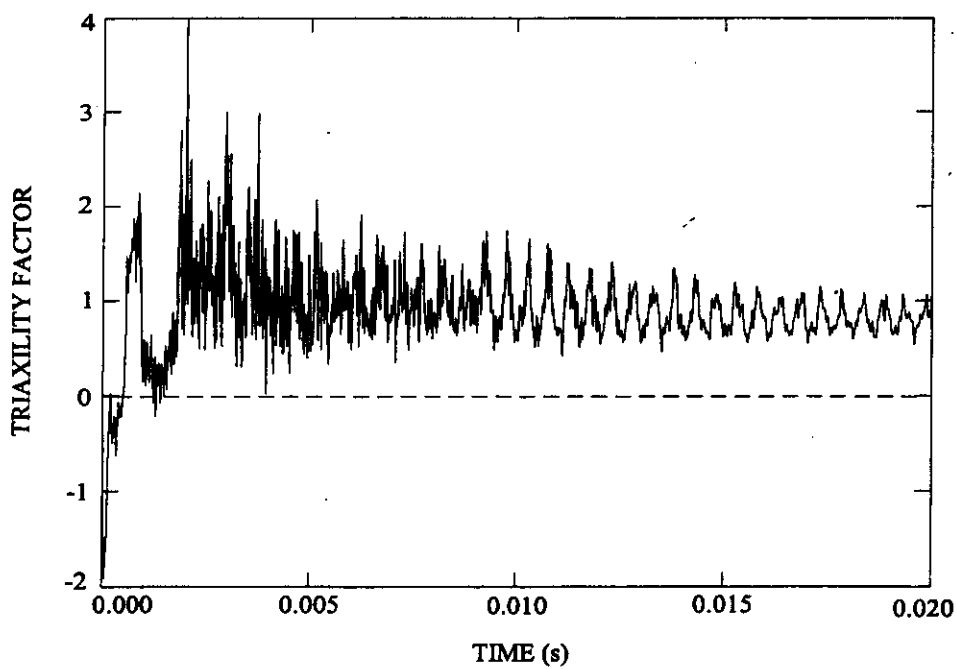


Figure 75. Equivalent Plastic Strain Plot at EL 31081.

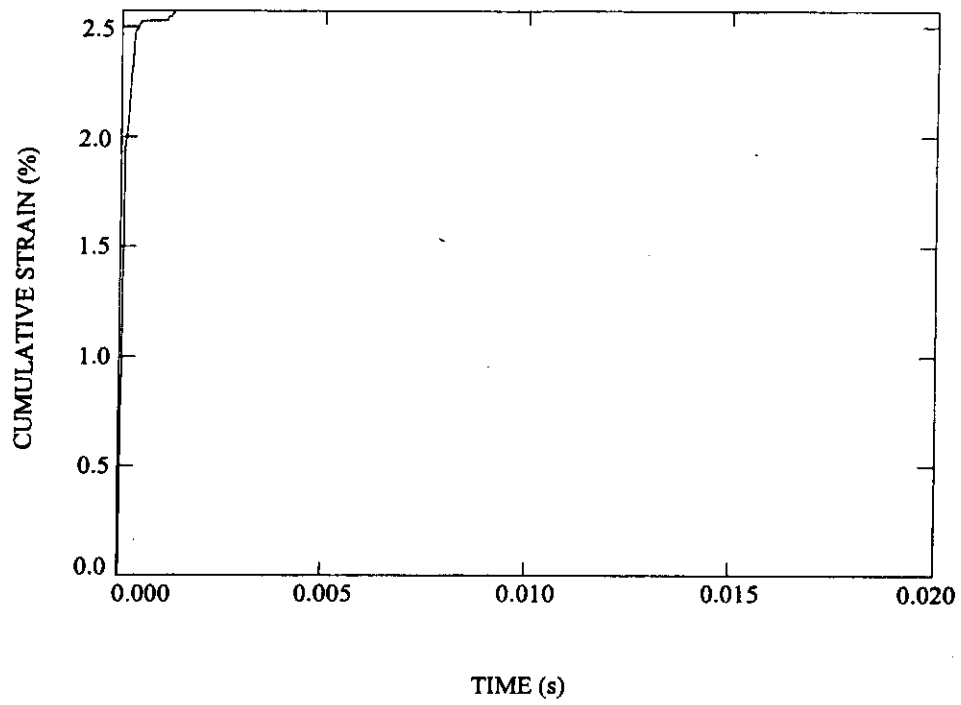
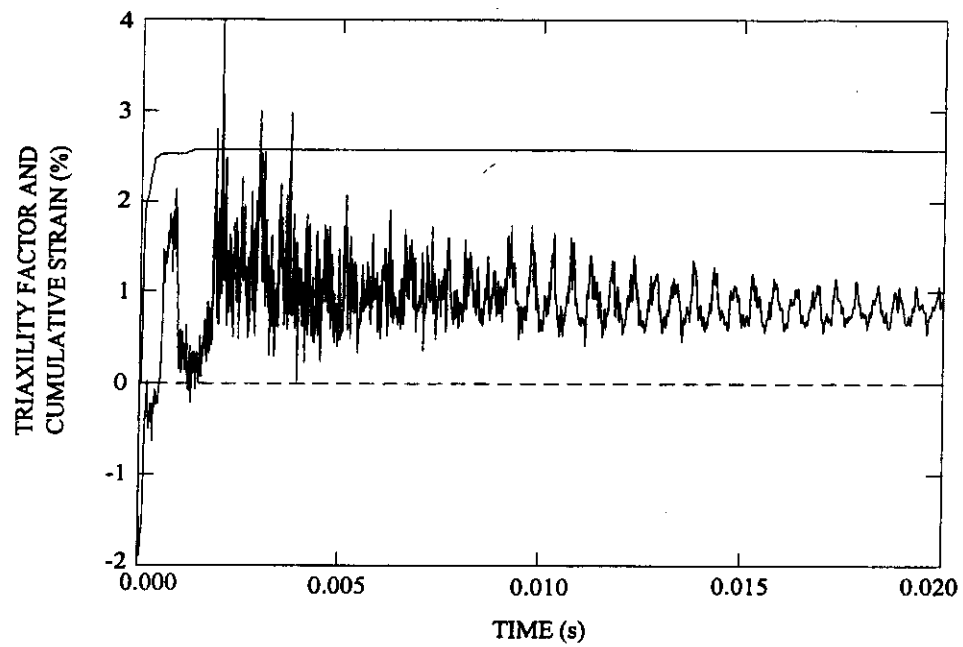


Figure 76. Combined Triaxiality Factor and Equivalent Plastic Strain Plot at EL 31081.



In the 9 m (30 ft) bottom-end free drop, the high stress locations on the containment vessel occur at the junction of bottom flange plate and shell wall. Localized bending deformation of the bottom flange plate and shell wall has been observed in the region of the joint. However, in these regions, the high stresses on the containment vessel are well within Service Level D limits. The cumulative strains are only slightly above offset yield in the area near the impact and are very low in comparison to the ultimate strain. For the containment closure bolts, the stresses and strains are very low relative to the material allowable limits.

5.7.3 Center of Gravity over Bottom Corner Free Drop

This case simulates the shielded drum package performance for a 9 m (30 ft) center of gravity over the bottom corner free drop onto a hard unyielding surface (see Figure 77). The total numerical simulation time is 20 ms. The resulting impact deceleration time history is shown in Figure 78. From Figure 78, the peak deceleration is determined to be 184.3gs, and the rebound time is 11.87 ms. Based on the rebound time and change in velocity, the average deceleration is 115.1gs. The velocity time history of the package is shown in Figure 79 and the displacement time history of the package is shown in Figure 80. Pertinent impact acceleration data are shown in Table 26.

Figure 77. Shielded Drum Package 9 m (30 ft) Bottom Corner Drop.

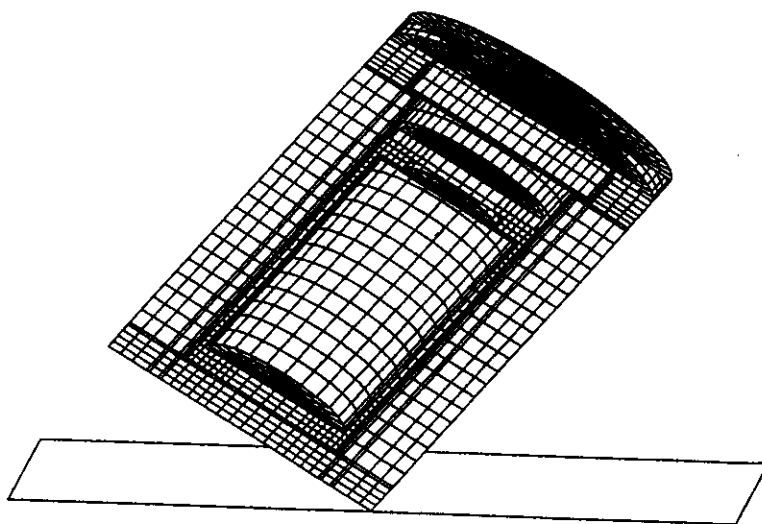


Figure 78. Shielded Drum Package 9 m (30 ft) Bottom Corner Drop Acceleration History.

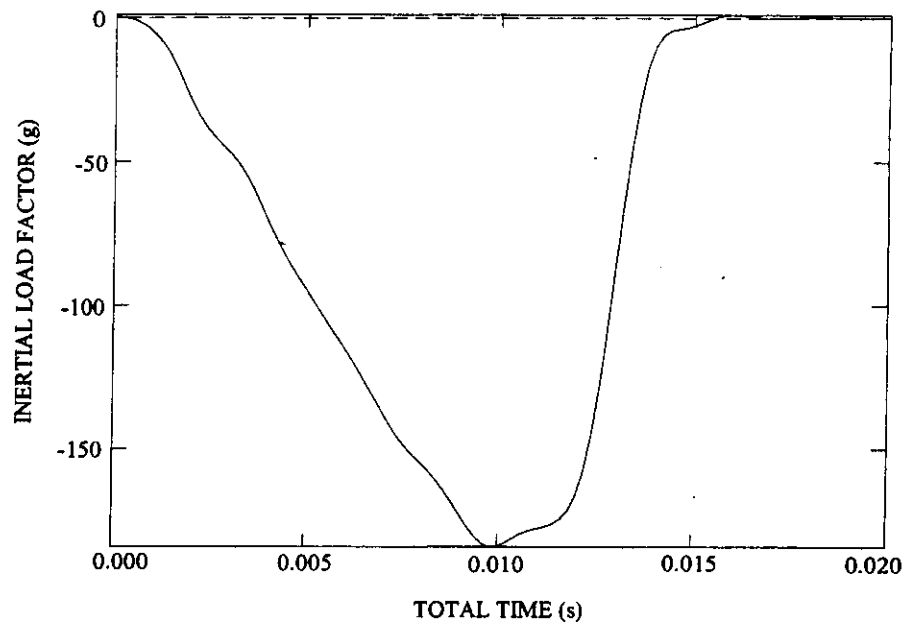


Figure 79. Shielded Drum Package 9 m (30 ft) Bottom Corner Drop Velocity History.

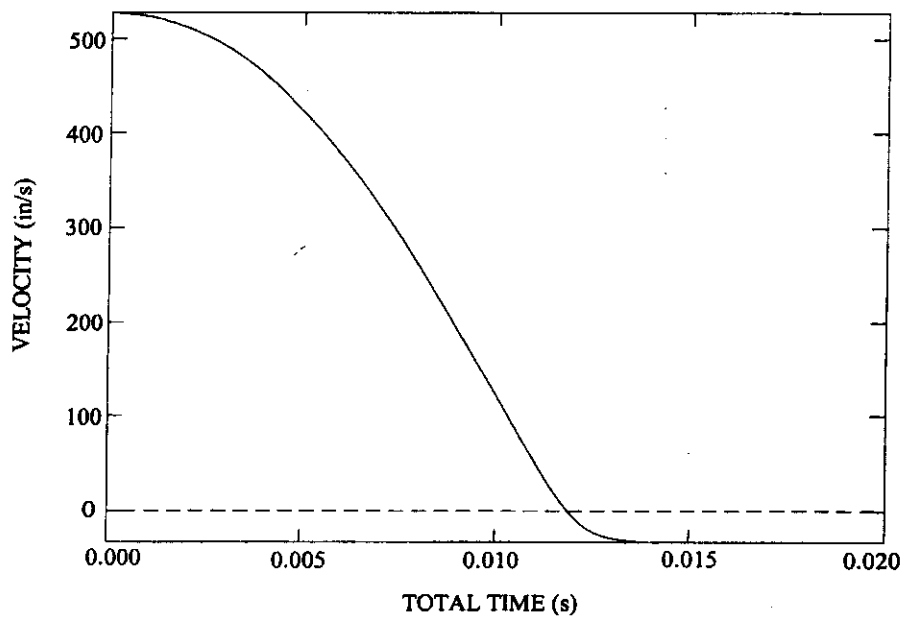


Figure 80. Shielded Drum Package 9 m (30 ft) Bottom Corner Drop Displacement History.

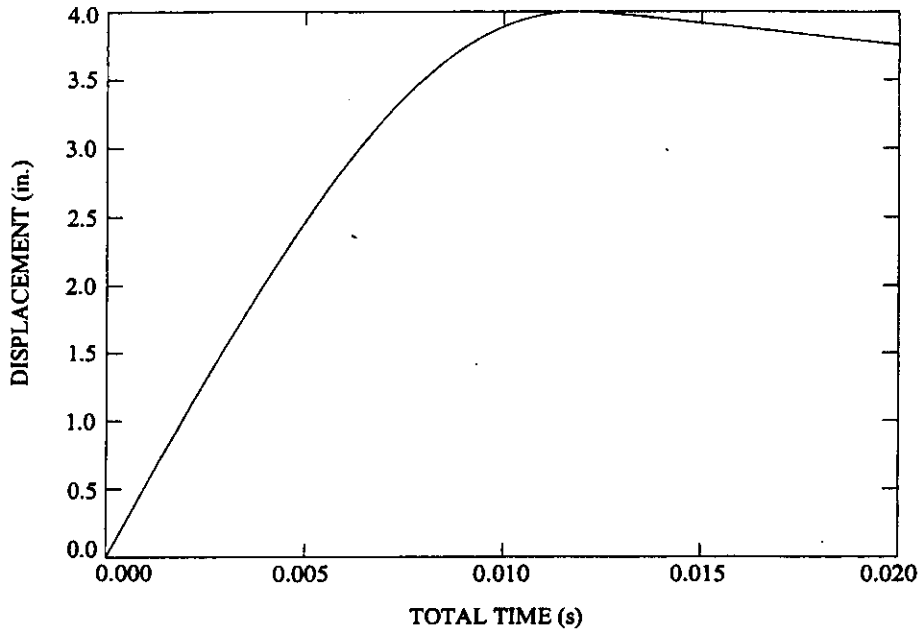


Table 26. Impact Acceleration Data of the 9 m (30 ft) Bottom Corner Drop.

Case	Average rebound-time acceleration (g)	Peak acceleration (g)	Rebound time (ms)
9 m (30 ft) bottom corner drop	115.1	184.3	11.87

The ASME BPVC stress intensities at the high stress locations on the containment vessel and at all containment closure bolts are shown in Table 27. The stress intensity time history at the most severely stressed location on the containment vessel (EL 32121) is shown in Figure 81. The stress intensity time history at a typical closure bolt is shown in Figure 82.

The cumulative strains at the high strain locations on the containment vessel and at all containment closure bolts are shown in Table 28. The largest cumulative strain occurs near the top of the shell wall at the junction with the reverse hub flange. This cumulative strain is at EL 32121 and only in the range of 0.97 % and shown in Figure 83. The largest cumulative strain on the containment closure bolts is only approximately 0.06 % as shown in Figure 84.

The SEDs at the critical region of the containment vessel and the containment closure bolts are provided in Table 29. Figure 85 shows the SED time history at the worst-case location of EL 32121. Figure 86 shows the SED time history at closure bolt element EL 34996. Figure 87 shows the deformation of the entire package at the end of the free drop. Figures 88 and 89 show the deformation of the containment vessel and lead shielding at the end of the free drop. Figures 90 through 92 are the TF, the PEEQ, and combined TF and PEEQ time histories at element EL 32121, respectively.

Table 27. ASME Code Stress Intensity at Containment Vessel from the 9 m (30 ft) Bottom Corner Drop.

Containment		Closure bolt	
Element	Stress intensity (ksi)	Element	Stress intensity (ksi)
30741	17.90	34991	5.85
30742	13.34	34992	5.05
30743	9.23	34993	4.57
31081	21.20	34994	5.55
31082	18.05	34995	7.09
31083	14.19	34996	8.56
31121	22.44		
31122	20.15		
31123	14.23		
31161	21.43		
31162	17.42		
31163	11.09		
32081	14.95		
32082	11.65		
32083	9.35		
32121	27.47		
32122	15.93		
32123	9.66		
32161	21.16		
32162	18.88		
32163	13.80		
32921	16.21		
32922	8.31		
32923	4.56		

Figure 81. Shielded Drum Package 9 m (30 ft) Bottom Corner Drop Stress Intensity History at the Containment Location (El 32121).

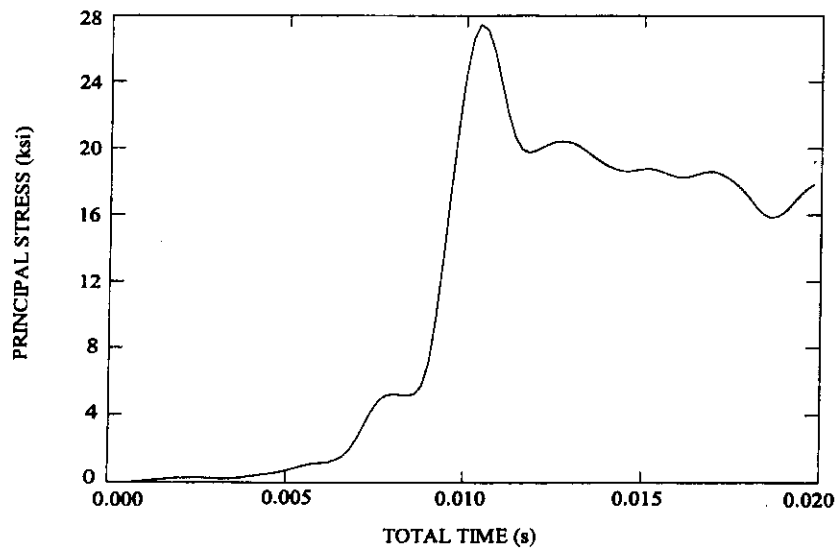


Figure 82. Shielded Drum Package 9 m (30 ft) Bottom Corner Drop Stress Intensity History at Closure Bolt No. 6 (EL 34996).

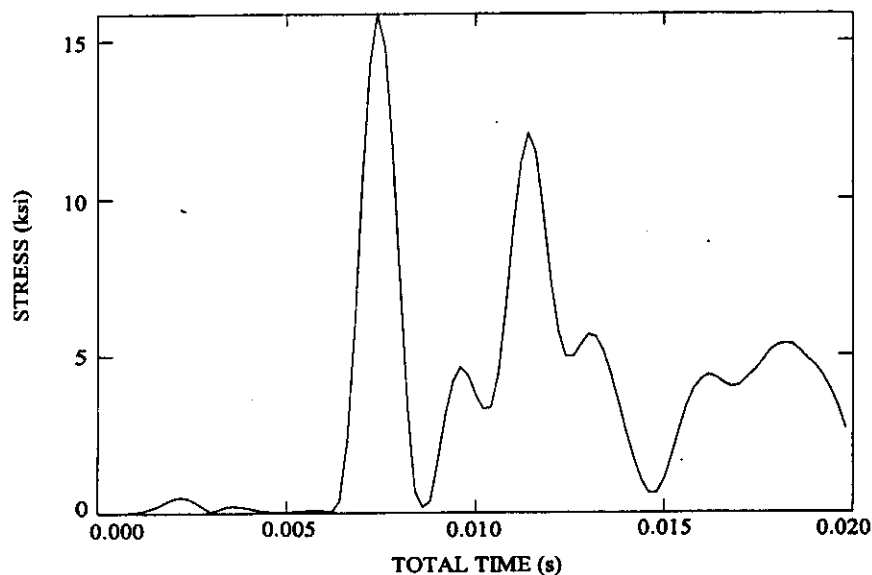


Table 28. Accumulated Peak Strains at Containment Vessel from the 9 m (30 ft) Bottom Corner Drop.

Containment		Closure bolt	
Element	Strain (in/in)	Element	Strain (in/in)
30741	9.292 E-04	34991	3.929 E-04
30742	7.695 E-04	34992	3.387 E-04
30743	5.704 E-04	34993	3.069 E-04
31081	1.957 E-03	34994	3.721 E-04
31082	9.573 E-04	34995	4.756 E-04
31083	6.915 E-04	34996	5.741 E-04
31121	2.755 E-03		
31122	1.125 E-03		
31123	6.980 E-04		
31161	1.593 E-03		
31162	9.040 E-04		
31163	6.203 E-04		
32081	7.216 E-04		
32082	6.163 E-04		
32083	4.824 E-04		
32121	9.707 E-03		
32122	4.334 E-03		
32123	6.216 E-04		
32161	5.443 E-03		
32162	4.594 E-03		
32163	9.986 E-04		
32921	2.321 E-03		
32922	3.577 E-04		
32923	4.379 E-04		

Figure 83. Shielded Drum Package 9 m (30 ft) Bottom Corner Drop Strain at Containment Location (EL 32121).

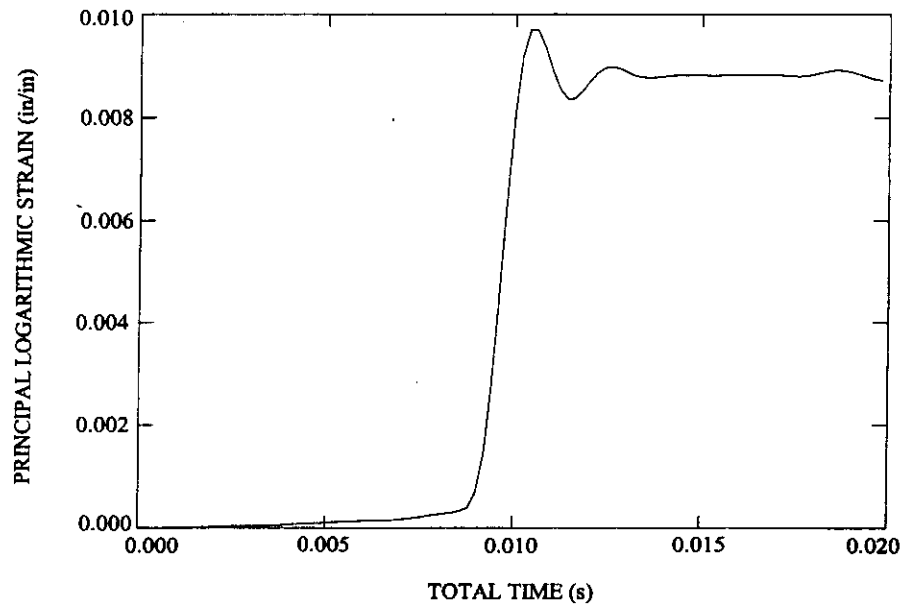


Figure 84. Shielded Drum Package 9 m (30 ft) Bottom Corner Drop Strain at Closure Bolt No. 6 (El 34996).

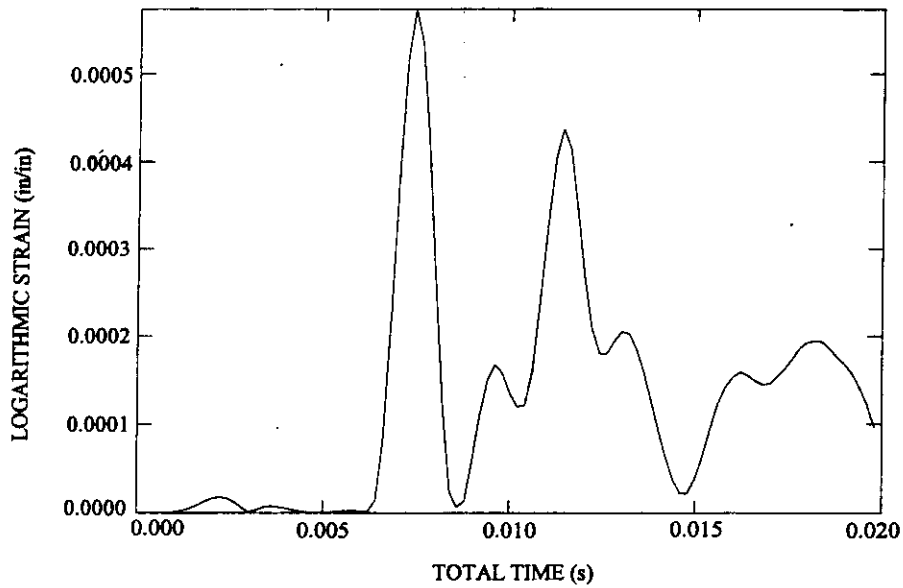


Table 29. Strain Energy Density at Containment Vessel from the 9 m (30 ft) Bottom Corner Drop.

Containment		Closure bolt	
Element	Strain energy density (in-lb/in ³)	Element	Strain energy density (in-lb/in ³)
30741	1.646 E+01	34991	2.963 E+00
30742	2.329 E+00	34992	2.270 E+00
30743	1.193 E+00	34993	1.600 E+00
31081	2.192 E+01	34994	2.725 E+00
31082	6.942 E+00	34995	4.489 E+00
31083	2.664 E+00	34996	6.404 E+00
31121	3.384 E+01		
31122	9.990 E+00		
31123	2.701 E+00		
31161	2.064 E+00		
31162	5.900 E+00		
31163	1.724 E+00		
32081	3.359 E+00		
32082	4.086 E+00		
32083	1.599 E+00		
32121	1.500 E+02		
32122	6.319 E+01		
32123	4.977 E+00		
32161	7.630 E+01		
32162	6.604 E+01		
32163	1.117 E+01		
32921	3.262 E+01		
32922	1.246 E+00		
32923	3.395 E-01		

Figure 85. Shielded Drum Package 9 m (30 ft) Bottom Corner Drop Strain Energy Density at the Containment Location (EL 32121).

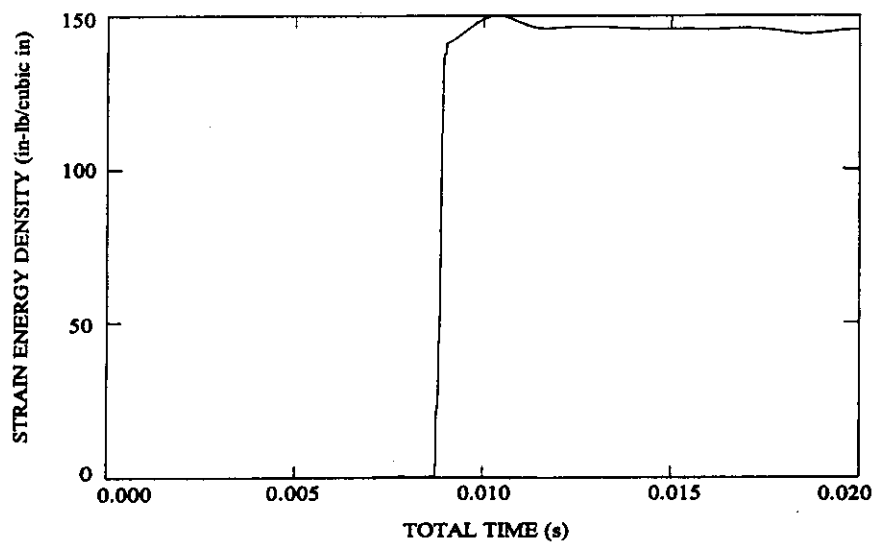


Figure 86. Shielded Drum Package 9 m (30 ft) Bottom Corner Drop Strain Energy Density at Closure Bolt No. 6 (EL 34996).

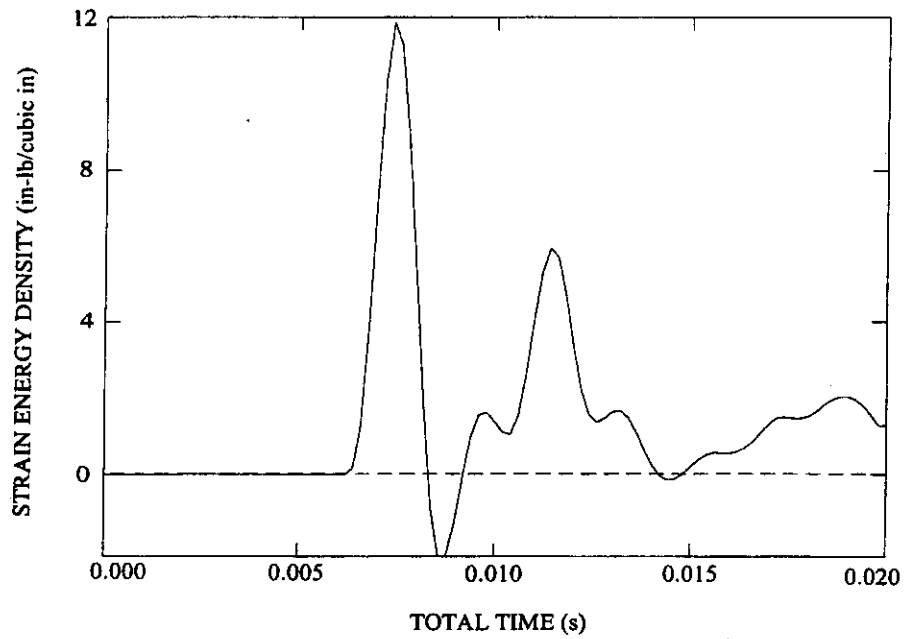


Figure 87. Shielded Drum Package 9 m (30 ft) Bottom Corner Drop Entire Package Deformation Plot.

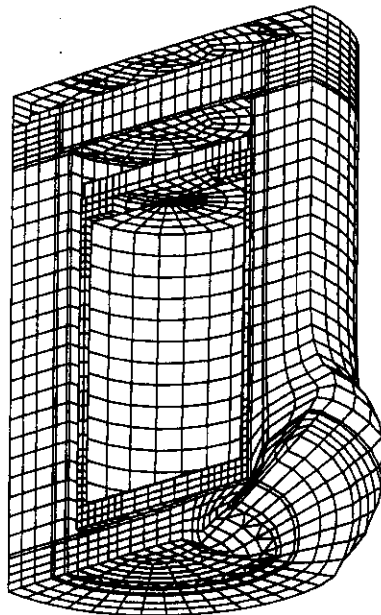


Figure 88. Shielded Drum Package 9 m (30 ft) Bottom Corner Drop Containment Deformation Plot.

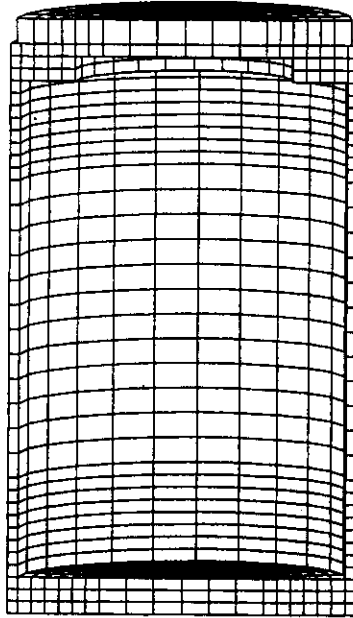


Figure 89. Shielded Drum Package 9 m (30 ft) Bottom Corner Drop Inner Liner and Lead Shielding Deformation Plot.

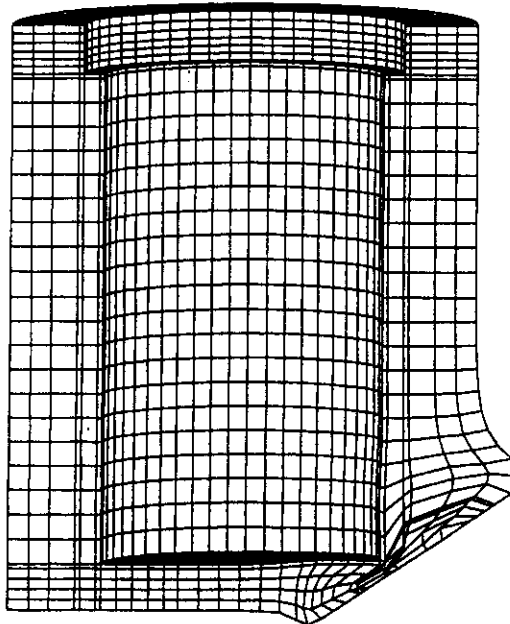


Figure 90. Triaxility Factor Plot at EL 32121.

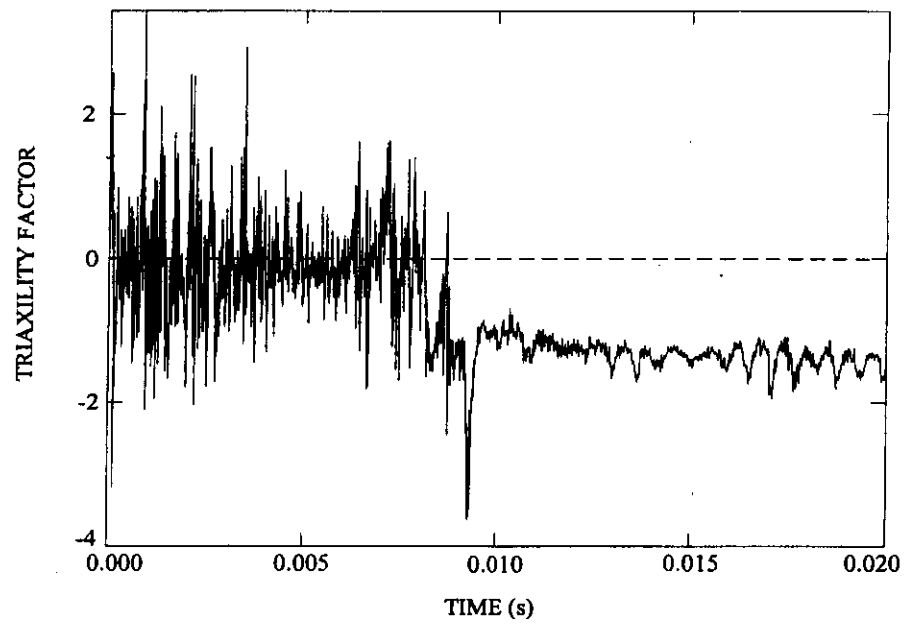


Figure 91. Equivalent Plastic Strain Plot at EL 32121.

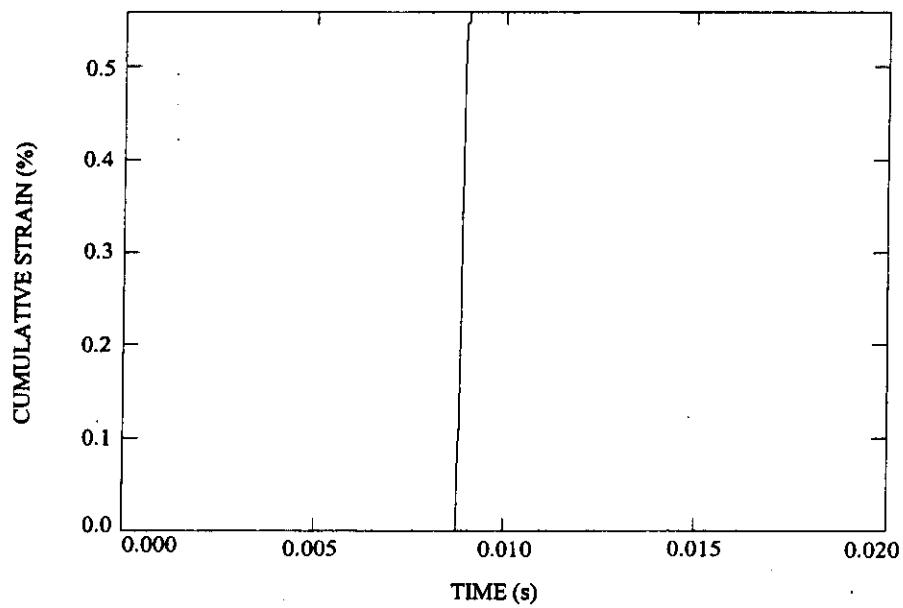
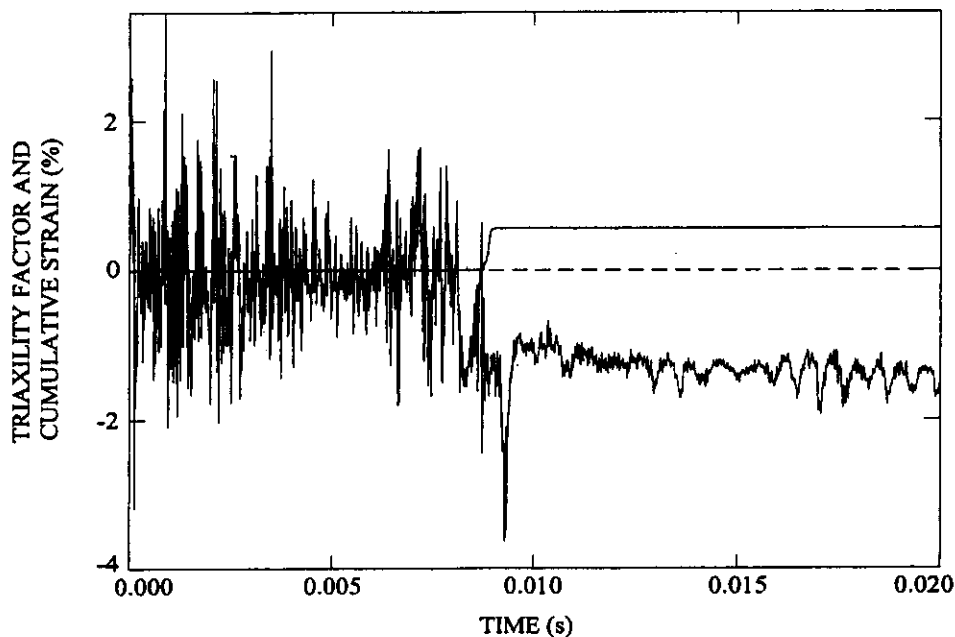


Figure 92. Combined Triaxility Factor and Equivalent Plastic Strain Plot at EL 32121.



The SEDs on the containment vessel are well below the maximum material toughness. The cumulative strains are also well below the ultimate material strain and, as shown in Figure 90, and are primarily compressive. The bolt stresses and strains are also well below maximum material allowable values for both SED and TSL.

In this free drop case, the high stress location on the containment vessel occurs at the shell wall to the reverse hub flange joint. The highest-stressed location does not occur at the weld, but a few inches away from the weld. This high stress region is very localized on the impact side of the containment vessel. The stresses on the containment vessel are well within ASME BPVC, Subsection NE limits for Service Level D. The cumulative strains are only slightly above the offset yield point. The containment closure bolt stresses are well within ASME BPVC, Subsection NE limits and the strains are not significant. The lead shielding severely deformed at the impact location, but the deformation is not sufficient to apply significant loading on the containment vessel. Subsequently, the loading on the containment vessel remains relatively low, since the majority of the impact energy is absorbed by the deformation of the lead shielding.

5.7.4 Center of Gravity Over Top Corner Free Drop

This case simulates the shielded drum package performance for a 9 m (30 ft) center of gravity over the top corner free drop onto a hard unyielding surface (see Figure 93). The total numerical simulation time of this case is 30 ms. The resulting deceleration time history is shown in Figure 94. From Figure 94, the peak acceleration is determined to be 196.5gs with a rebound time of 12.66 ms. Based on the rebound time and velocity change, the average deceleration is calculated to be 107.9gs. The velocity time history of the package is shown in Figure 95 and the displacement time history of the package is shown in Figure 96. Pertinent impact acceleration data are shown in Table 30.

The ASME BPVC stress intensities at the highly stressed containment vessel locations and all containment closure bolts are shown in Table 31. The stress intensity time history at the highest stress location on the containment vessel (EL 32321) is shown in Figure 97. The stress intensity time history at a typical containment closure bolt is shown in Figure 98.

The cumulative strains at highly strained locations on the containment vessel and all containment closure bolts are shown in Table 32. The largest cumulative strain on the containment vessel occurs at the joint of the shell wall to the reverse hub flange. This cumulative strain at EL 32321 is very low in the range of 2.74 % as shown in Figure 99. The largest cumulative strain on the containment closure bolts is approximately 0.35 % at the end of the free drop as shown in Figure 100.

Figure 93. Shielded Drum Package 9 m (30 ft) Top Corner Drop.

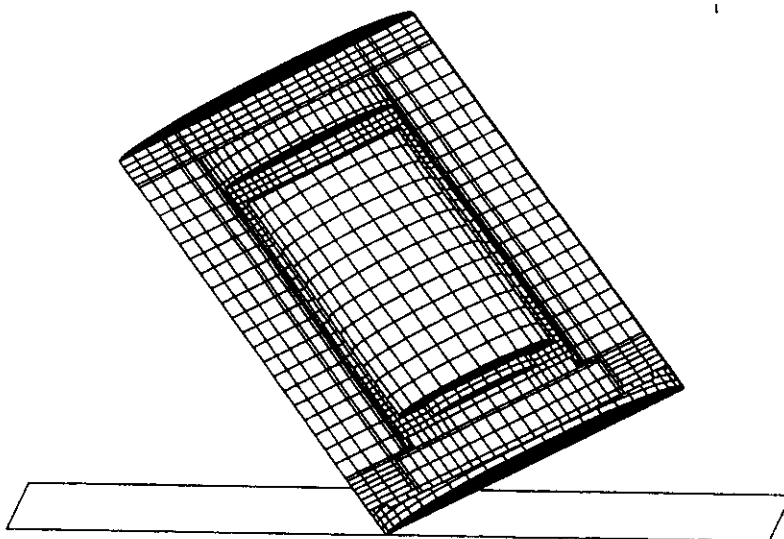


Figure 94. Shielded Drum Package 9 m (30 ft) Top Corner Drop Acceleration History.

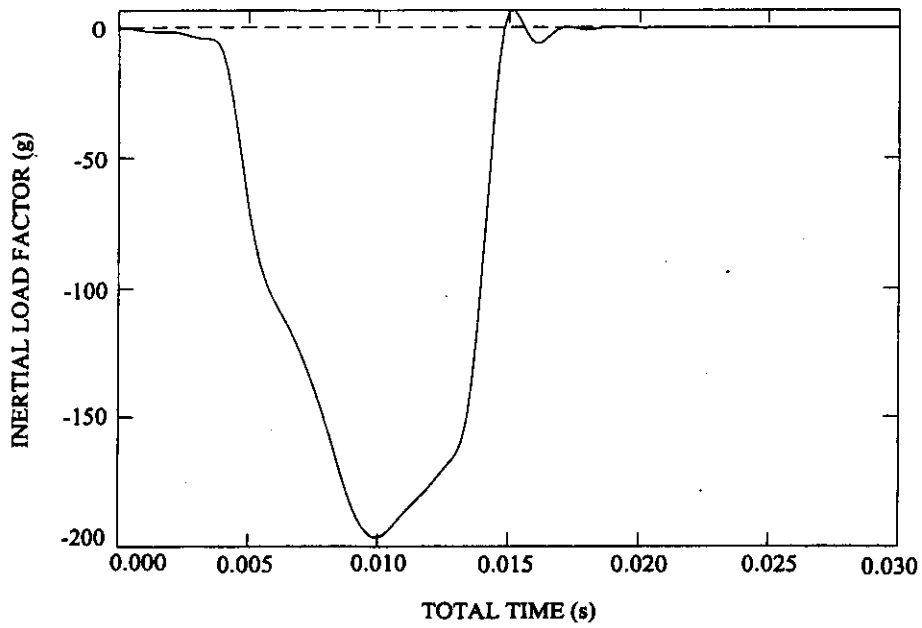


Figure 95. Shielded Drum Package 9 m (30 ft) Top Corner Drop Velocity History.

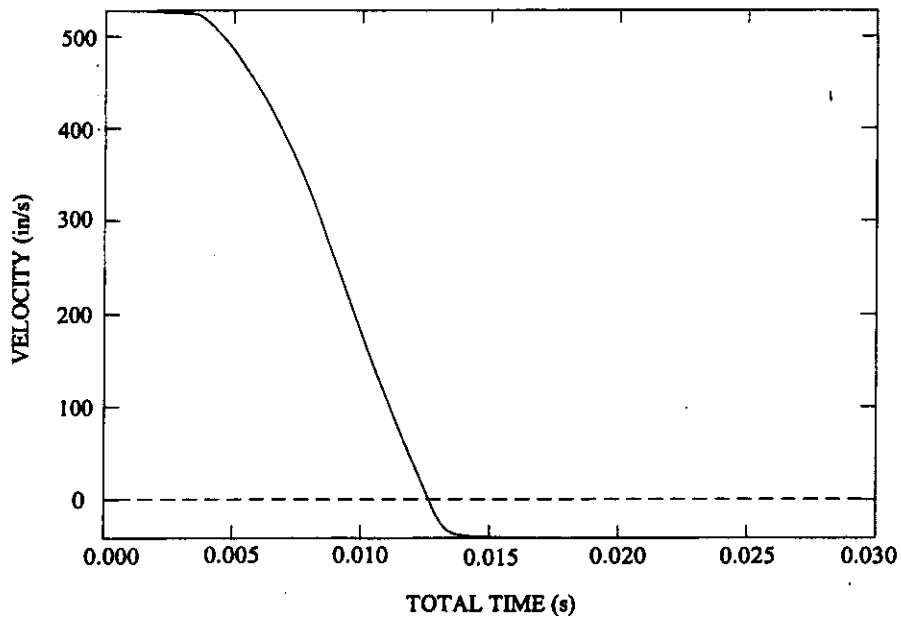


Figure 96. Shielded Drum Package 9 m (30 ft) Top Corner Drop Displacement History.

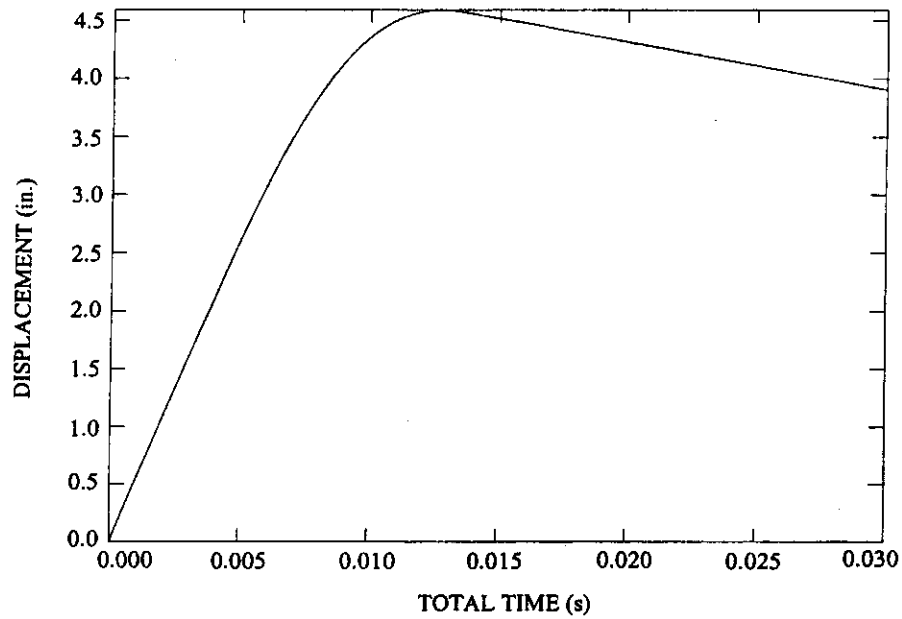


Table 30. Impact Acceleration Data of the 9 m (30 ft) Top Corner Drop.

Case	Average rebound-time acceleration (g)	Peak acceleration (g)	Rebound time (ms)
9 m (30 ft) top corner drop	107.9	196.5	12.66

Table 31. ASME Code Stress Intensity at Containment Vessel from the 9 m (30 ft) Top Corner Drop.

Containment		Closure bolt	
Element	Stress intensity (ksi)	Element	Stress intensity (ksi)
30741	13.64	34991	36.14
30742	14.29	34992	16.92
30743	9.30	34993	10.13
31081	19.20	34994	15.18
31082	19.13	34995	16.36
31083	17.14	34996	13.21
31121	21.61		
31122	16.10		
31123	15.46		
31161	17.83		
31162	16.77		
31163	15.98		
32081	31.23		
32082	30.27		
32083	27.60		
32121	31.19		
32122	29.80		
32123	27.63		
32161	34.63		
32162	34.45		
32163	30.05		
32321	36.00		
32322	32.94		
32323	28.05		
32921	23.60		
32922	24.42		
32923	17.89		

Figure 97. Shielded Drum Package 9 m (30 ft) Top Corner Drop Stress Intensity History at the Containment Location (EL 32321).

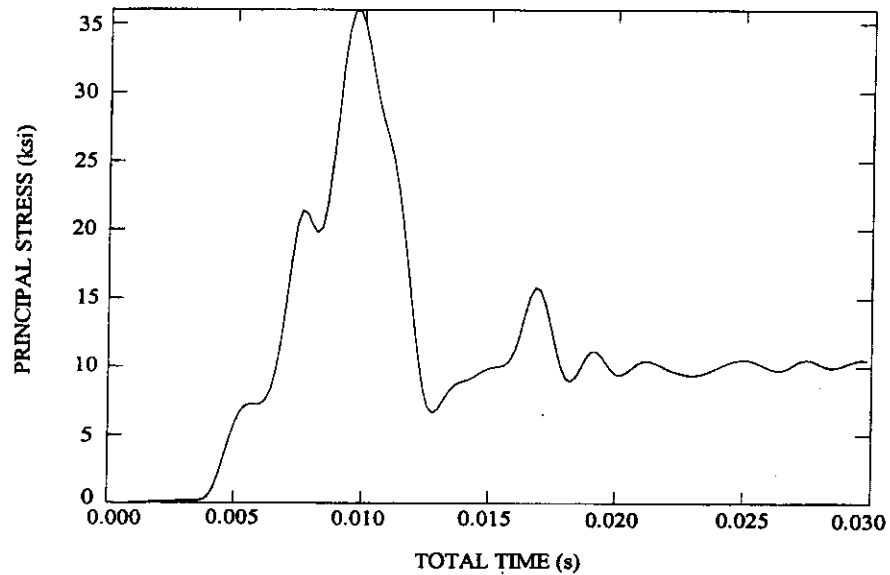


Figure 98. Shielded Drum Package 9 m (30 ft) Top Corner Drop Stress Intensity History at Closure Bolt No. 1 (EL 34991).

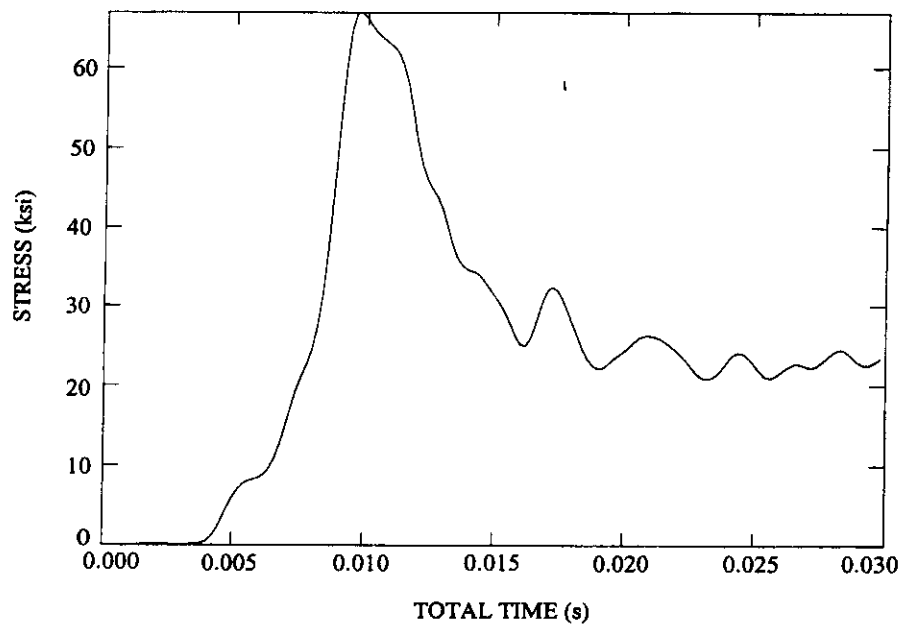


Table 32. Accumulated Peak Strains at Containment Vessel from the 9 m (30 ft) Top Corner Drop.

Containment		Closure bolt	
Element	Strain (in/in)	Element	Strain (in/in)
30741	1.735 E-03	34991	2.425 E-03
30742	1.266 E-03	34992	1.135 E-03
30743	8.076 E-04	34993	6.798 E-04
31081	3.148 E-02	34994	2.234 E-03
31082	1.891 E-02	34995	3.384 E-03
31083	3.516 E-03	34996	3.530 E-03
31121	7.542 E-03		
31122	6.737 E-03		
31123	1.822 E-03		
31161	3.916 E-03		
31162	3.597 E-03		
31163	1.541 E-03		
32081	1.334 E-02		
32082	1.179 E-02		
32083	8.322 E-03		
32121	1.219 E-02		
32122	1.134 E-02		
32123	9.240 E-03		
32161	2.204 E-02		
32162	1.842 E-02		
32163	1.105 E-02		
32321	2.742 E-02		
32322	1.646 E-02		
32323	6.598 E-03		
32921	2.378 E-03		
32922	2.044 E-03		
32923	9.328 E-04		

Figure 99. Shielded Drum Package 9 m (30 ft) Top Corner Drop Strain at the Containment Location (EL 32321).

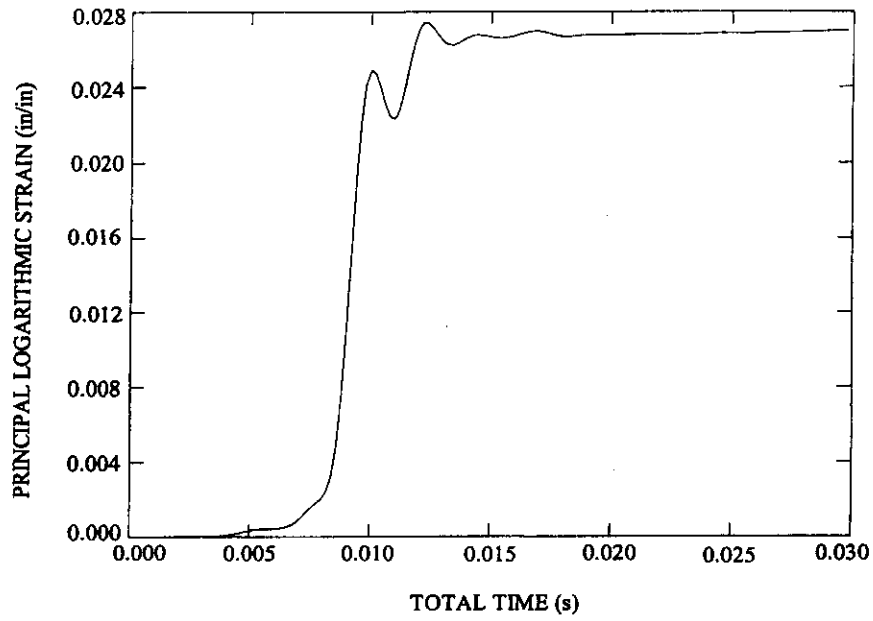
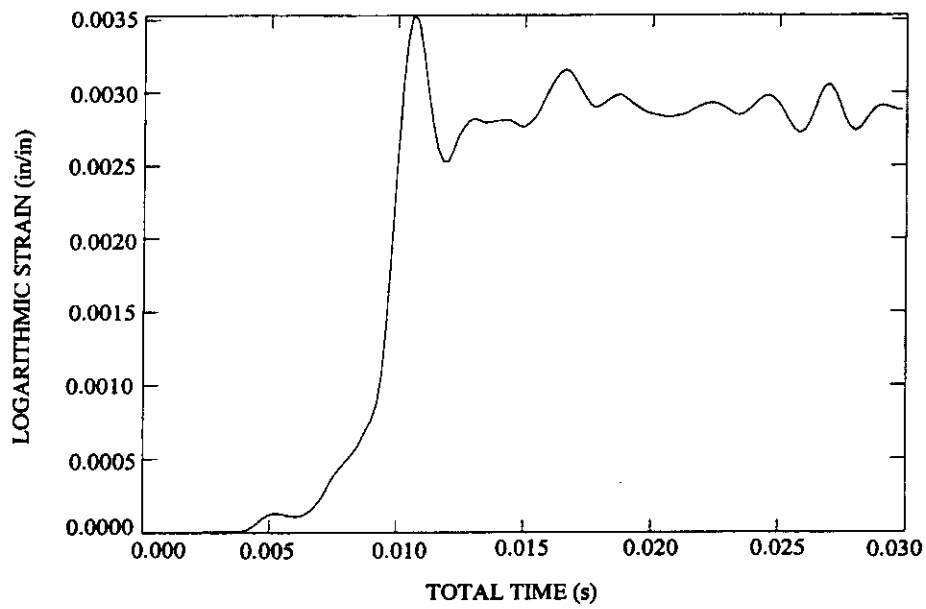


Figure 100. Shielded Drum Package 9 m (30 ft) Top Corner Drop Strain at Closure Bolt No. 6 (EL 34996).



The SEDs at the critical region on the containment vessel and the containment closure bolts are presented in Table 33. Figure 101 shows the SED time history at EL 32321. Figure 102 shows the SED time history at containment closure bolt element EL 34996.

Table 33. Strain Energy Density at Containment Vessel from the 9 m (30 ft) Top Corner Drop.

Containment		Closure bolt	
Element	Strain energy density (in-lb/in ³)	Element	Strain energy density (in-lb/in ³)
30741	1.608 E+01	34991	4.304 E+01
30742	9.449 E+00	34992	9.520 E+00
30743	1.166 E+00	34993	9.963 E+00
31081	4.740 E+02	34994	8.521 E+01
31082	2.728 E+02	34995	1.418 E+02
31083	4.835 E+01	34996	1.499 E+02
31121	1.157 E+02		
31122	9.432 E+01		
31123	2.770 E+01		
31161	5.300 E+01		
31162	5.000 E+01		
31163	1.932 E+01		
32081	2.218 E+02		
32082	1.800 E+02		
32083	1.192 E+02		
32121	3.036 E+02		
32122	2.008 E+02		
32123	1.386 E+02		
32161	7.509 E+02		
32162	4.340 E+02		
32163	1.777 E+02		
32321	5.268 E+02		
32322	2.897 E+02		
32323	8.236 E+01		
32921	3.247 E+01		
32922	2.043 E+01		
32923	4.192 E+00		

Figure 101. Shielded Drum Package 9 m (30 ft) Top Corner Drop Strain Energy Density at the Containment Location (EL 32321).

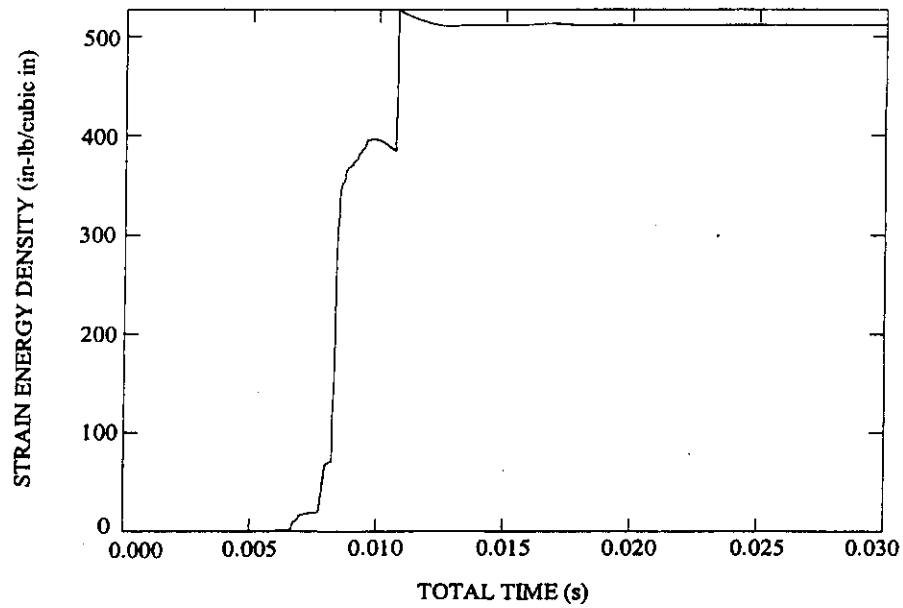
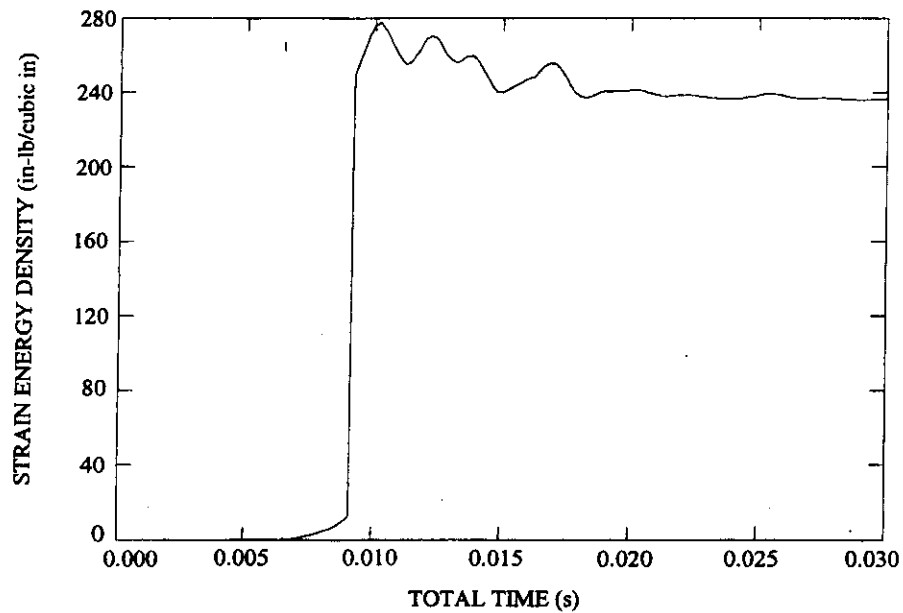


Figure 102. Shielded Drum Package 9 m (30 ft) Top Corner Drop Strain Energy Density at Closure Bolt No. 6 (EL 34996).



The stresses on the containment vessel are well within Service Level D limits. The cumulative strains are very low and only slightly above the offset yield point of the material. The containment closure bolt stresses are well within Service Level D limits and the strains are well within acceptable limits. Figures 103 and 104 show the deformation of the package at the end of the free drop. Figures 105 and 106 show the deformation of the containment and the lead shielding at the end of the free drop, respectively. Figure 106 shows severe deformation of the lead shielding in the area of impact. Figures 107 through 109 are the TF, the PEEQ, and the TF and PEEQ combined time history at EL 32321, respectively.

In this free drop case, the high stress locations on the containment vessel occur at the juncture of the shell wall to the reverse hub flange. This high stress region extends several inches from the weld on the impact side of the containment vessel.

Figure 103. Shielded Drum Package 9 m (30 ft) Top Corner Drop Entire Package Deformation Plot, Front View.

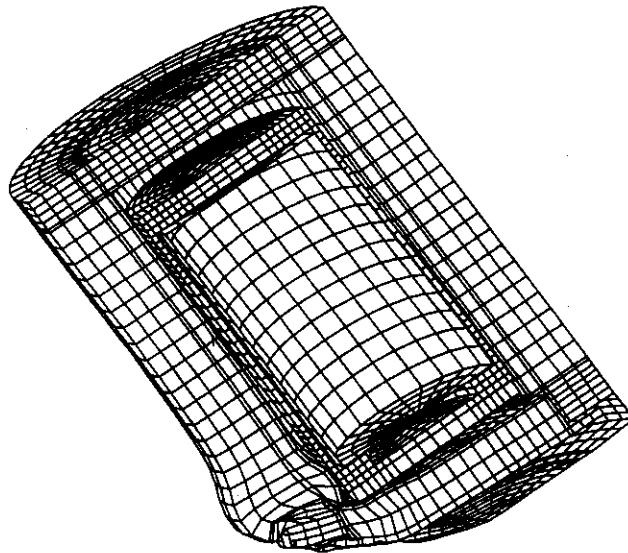


Figure 104. Shielded Drum Package 9 m (30 ft) Top Corner Drop Entire Package Deformation, Back View.

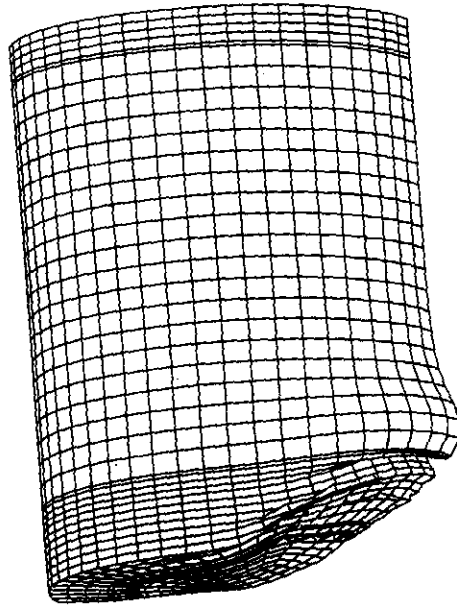


Figure 105. Shielded Drum Package 9 m (30 ft) Top Corner Drop Containment Deformation Plot.

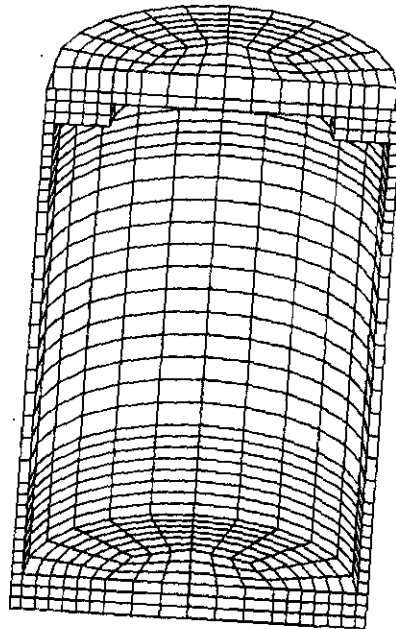


Figure 106. Shielded Drum Package 9 m (30 ft) Top Corner Drop Inner Liner and Lead Shielding Deformation Plot.

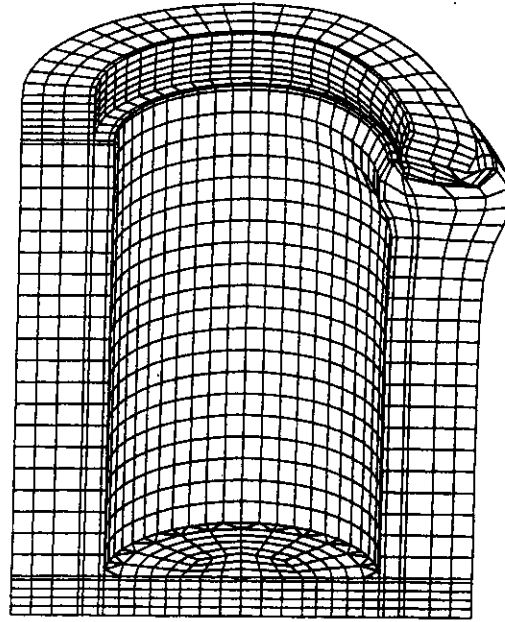


Figure 107. Triaxility Factor Plot at EL 32321.

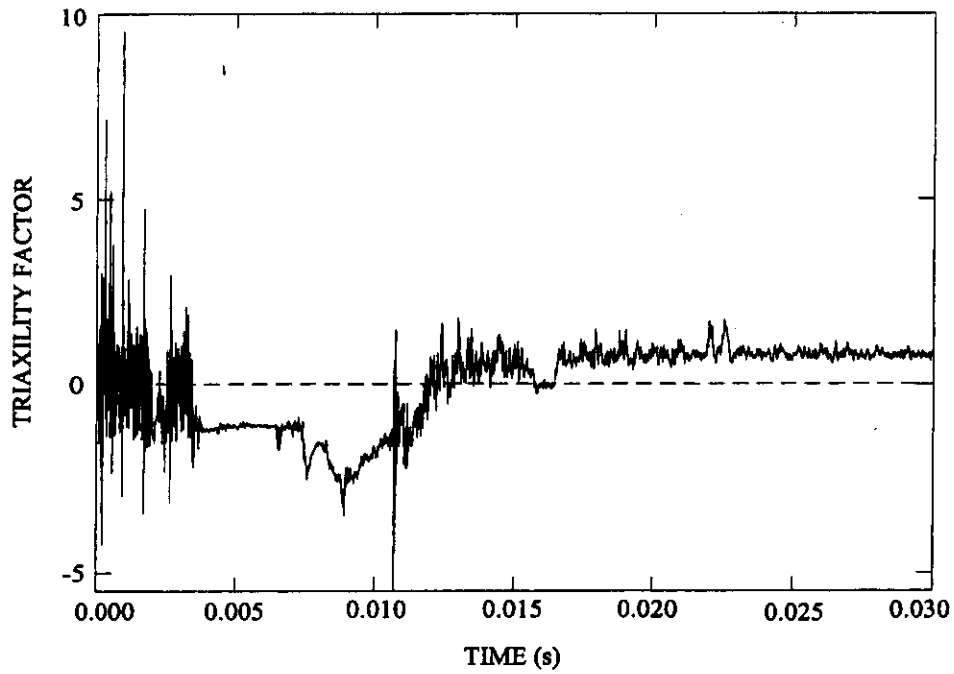


Figure 108. Equivalent Plastic Strain Plot at EL 32321.

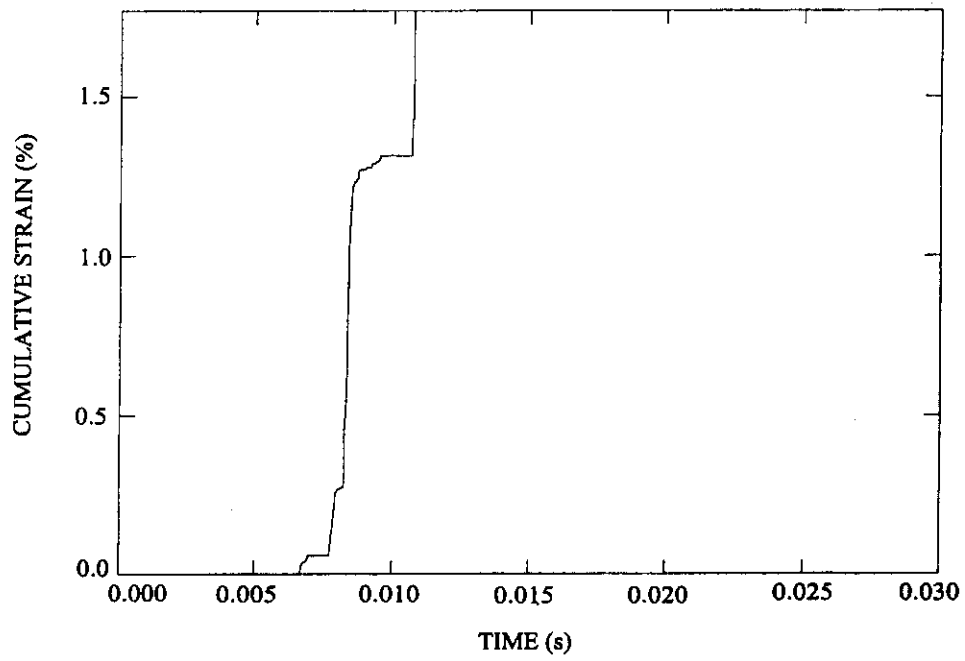
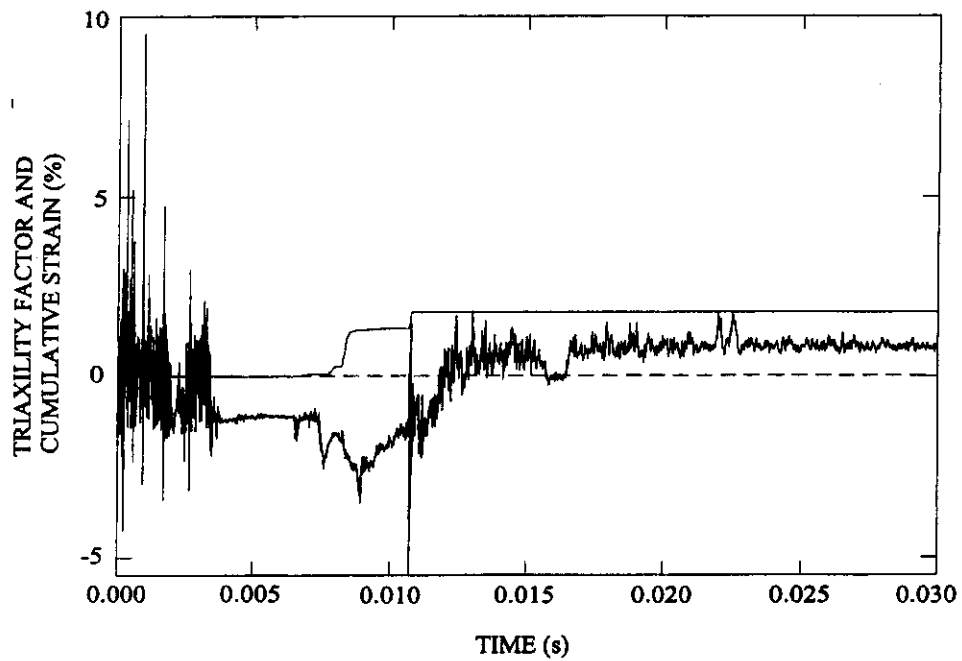


Figure 109. Combined Triaxiality Factor and Equivalent Plastic Strain Plot at EL 32321.



5.8 1 M (4 FT) AND 9 M (30 FT) SEQUENTIAL FREE DROPS FOLLOWED BY PUNCTURE

In keeping with the accident sequence specified in 10 CFR 71.73, "Physical Protection of Plants and Materials," and meeting the NRC interpretation, the shielded drum package is numerically simulated to be subjected to a sequence of free drops followed by a puncture test. The shielded drum package is subjected to a 1 m (4 ft) horizontal side NC free drop followed by a 9 m (30 ft) horizontal side AC free drop with impact in the same location as the NC free drop. Following the sequence of free drops, the package is subjected to the regulatory puncture test. For the puncture test, the package is dropped 1 m (40 in.) horizontally onto the puncture bar in the same planar orientation as the free drops. Reported in this section are the numerical simulation results of package puncture after a numerically simulated series of free drops. In this evaluation, the horizontal orientation was chosen since the previous free drop data showed this to result in the worst-case loading to the package. The data in this case are filtered in the same manner as for the previous free drop cases.

The shielded drum package is subjected to a 1 m (40 in.) puncture drop onto a vertical mild steel 15 cm (6 in.) diameter by 20 cm (8 in.) long bar mounted on a horizontal unyielding surface (see Figure 110). The puncture drop of the package is numerically simulated following a numerically simulated 1 m (4 ft) horizontal side drop followed by a 9 m (30 ft) horizontal side drop. This numerically simulates the cumulative damage to the package following the sequence of regulatory performance tests. For the puncture test, the puncture bar is oriented onto the mid-section of the package on the same side as the previous free drop impacts. The total numerical simulation time of this puncture drop is 30 ms, starting from an end of 9 m (30 ft) free drop at 40 ms and ending at 70 ms. The resulting impact deceleration time history is shown in Figure 111. From Figure 111, the peak deceleration is determined to be 48.0gs with a rebound time of 10.7 ms. Based on the rebound time and the change in velocity, the average deceleration from puncture is calculated to be 42.5gs. The velocity time history of the package is shown in Figure 112 and the displacement time history of the package is shown in Figure 113. Pertinent deceleration data are shown in Table 34.

Figure 110. Shielded Drum Package Puncture Drop.

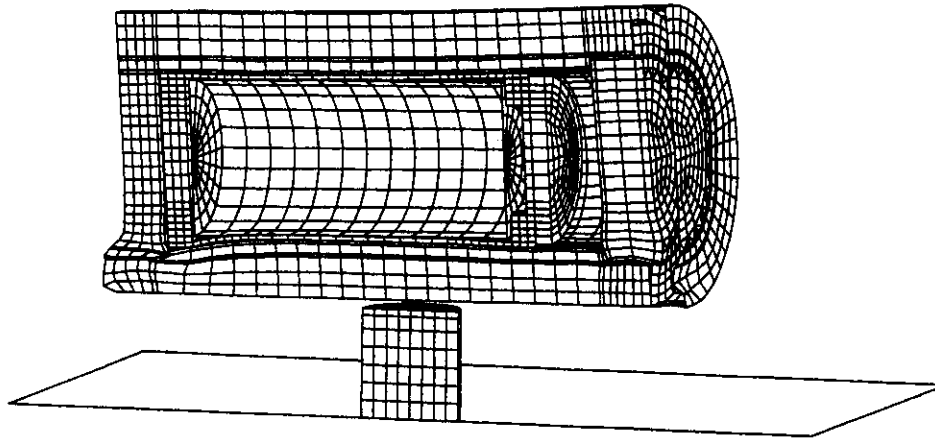


Figure 111. Shielded Drum Package Puncture Drop Acceleration History.

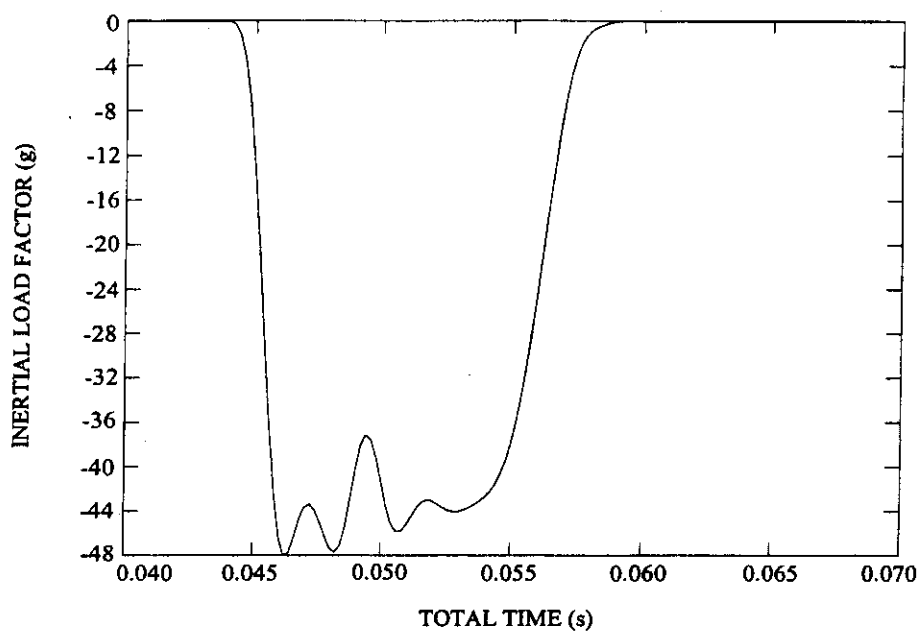


Figure 112. Shielded Drum Package Puncture Drop Velocity History.

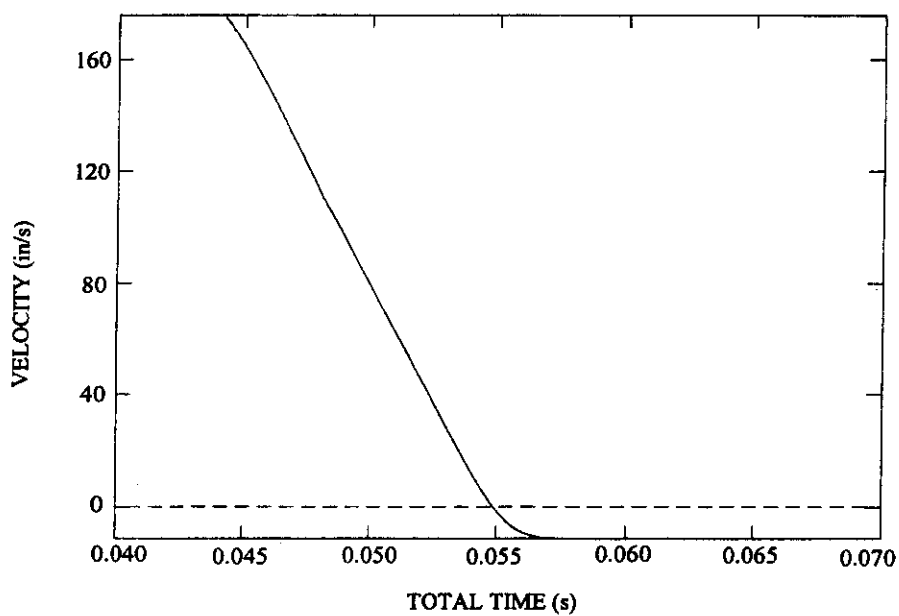


Figure 113. Shielded Drum Package Puncture Drop Displacement History.

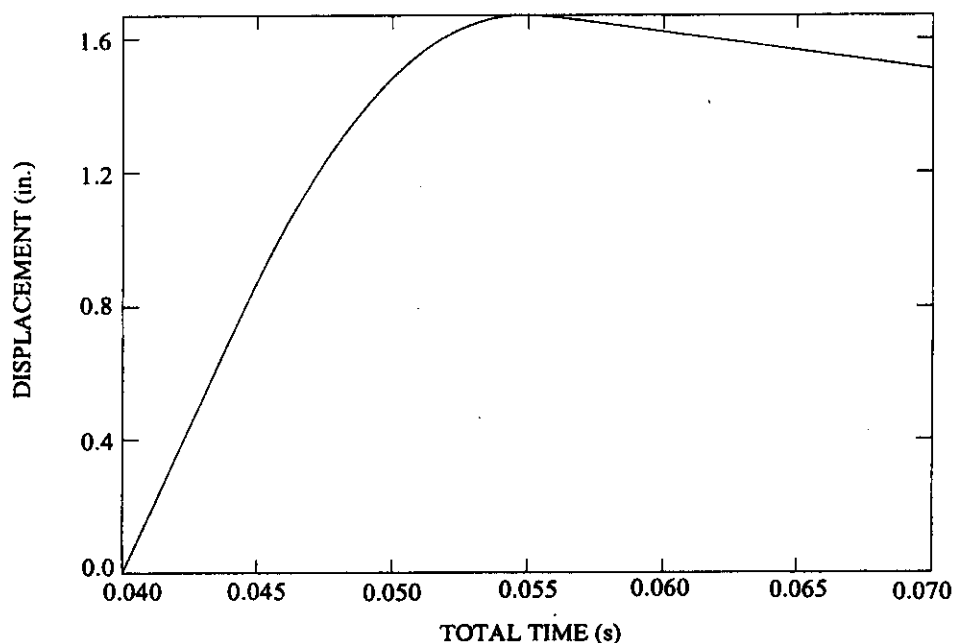


Table 34. Impact Acceleration Data of the Puncture Drop.

Case	Average rebound-time acceleration (g)	Peak acceleration (g)	Rebound time (ms)
1 m (40 in.) puncture drop	42.5	48.0	10.71

All ASME Code stress intensities at the highly stressed containment vessel locations and all containment closure bolts are shown in Table 35. The stress intensity time history at the most severely stressed location on the containment vessel is shown in Figure 114. The stress intensity time history at the most severely stressed containment closure bolt is shown in Figure 115.

The total cumulative strains at the highly strained locations on the containment vessel and at all containment closure bolts are shown in Table 36. The largest total cumulative strain on the containment vessel occurs at the junction of the bottom plate and shell wall. The maximum total cumulative strain is in the range of 9.4%. A typical cumulative strain time history at EL 30741 is shown in Figure 116. The largest total cumulative strain on containment closure bolt is approximately 0.1% as shown in Figure 117.

The SEDs at the critical region of the containment vessel and at all containment closure bolts are provided in Table 37. Figure 118 shows the SED time history at EL 31081. Figure 119 shows the SED time history at containment closure bolt element EL 34991.

Table 35. ASME Code Stress Intensity at Containment Vessel from the Puncture Drop.

Containment		Closure bolt	
Element	Stress intensity (ksi)	Element	Stress intensity (ksi)
30741	50.70	34991	30.25
30742	37.01	34992	5.05
30743	35.84	34993	2.23
31081	46.16	34994	1.02
31082	34.83	34995	9.51
31083	36.88	34996	18.05
31121	35.34		
31122	39.51		
31123	38.84		
31161	34.09		
31162	33.74		
31163	22.82		
32081	23.68		
32082	34.42		
32083	37.70		
32121	21.59		
32122	35.56		
32123	36.86		
32161	29.72		
32162	37.44		
32163	36.30		
32921	34.20		
32922	24.96		
32923	24.87		

Figure 114. Shielded Drum Package Puncture Drop Stress Intensity History at the Containment Location (EL 30741).

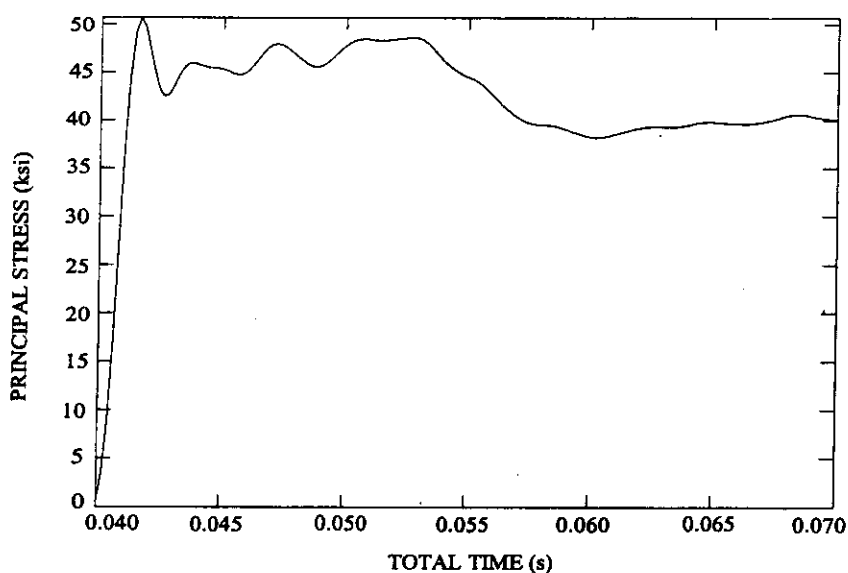


Figure 115. Shielded Drum Package Puncture Drop Stress Intensity History at Closure Bolt No. 1 (EL 34991).

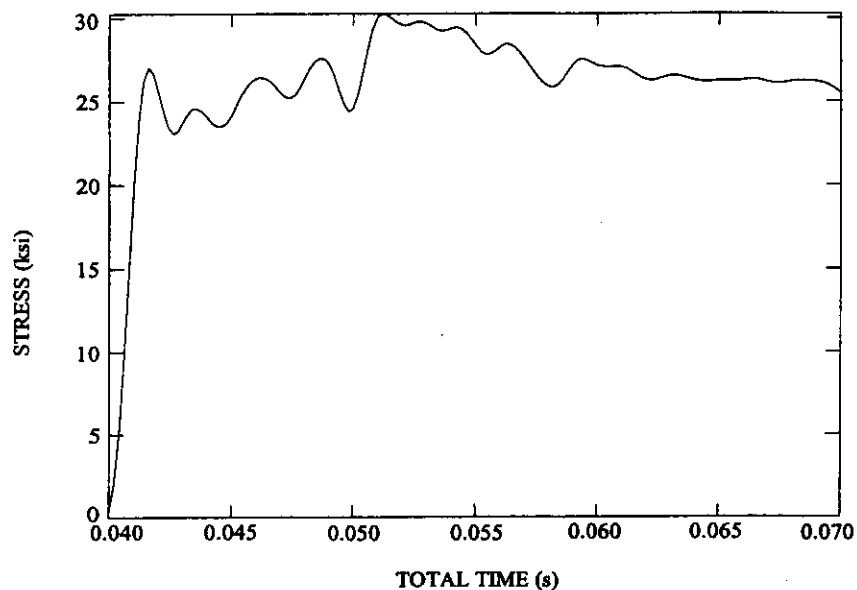


Table 36. Accumulated Peak Strains at Containment Vessel from the Puncture Drop.

Containment		Closure bolt	
Element	Strain (in/in)	Element	Strain (in/in)
30741	9.366 E-02	34991	1.096 E-03
30742	2.673 E-02	34992	1.828 E-04
30743	1.667 E-02	34993	8.070 E-05
31081	8.427 E-02	34994	7.302 E-05
31082	6.712 E-02	34995	3.444 E-04
31083	7.568 E-02	34996	6.541 E-04
31121	2.148 E-02		
31122	4.260 E-02		
31123	7.394 E-02		
31161	1.504 E-02		
31162	3.150 E-02		
31163	5.542 E-02		
32081	6.699 E-03		
32082	1.457 E-02		
32083	2.956 E-02		
32121	1.858 E-02		
32122	1.347 E-02		
32123	2.597 E-02		
32161	2.294 E-02		
32162	1.688 E-02		
32163	2.541 E-02		
32921	1.095 E-02		
32922	2.458 E-03		
32923	2.113 E-03		

Figure 116. Shielded Drum Package Puncture Drop Strain at the Containment Location (EL 30741).

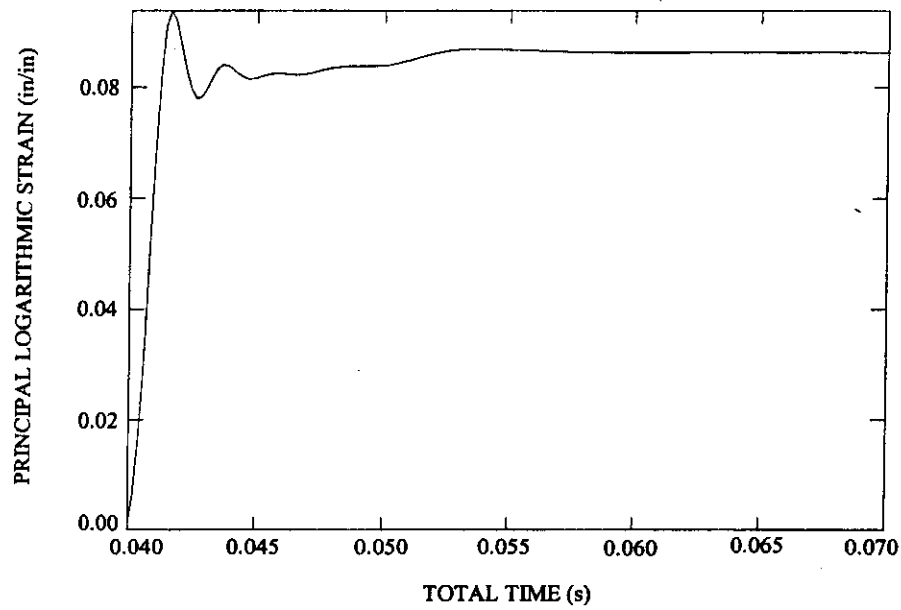


Figure 117. Shielded Drum Package Puncture Drop Strain at Closure Bolt No. 1 (EL 34991).

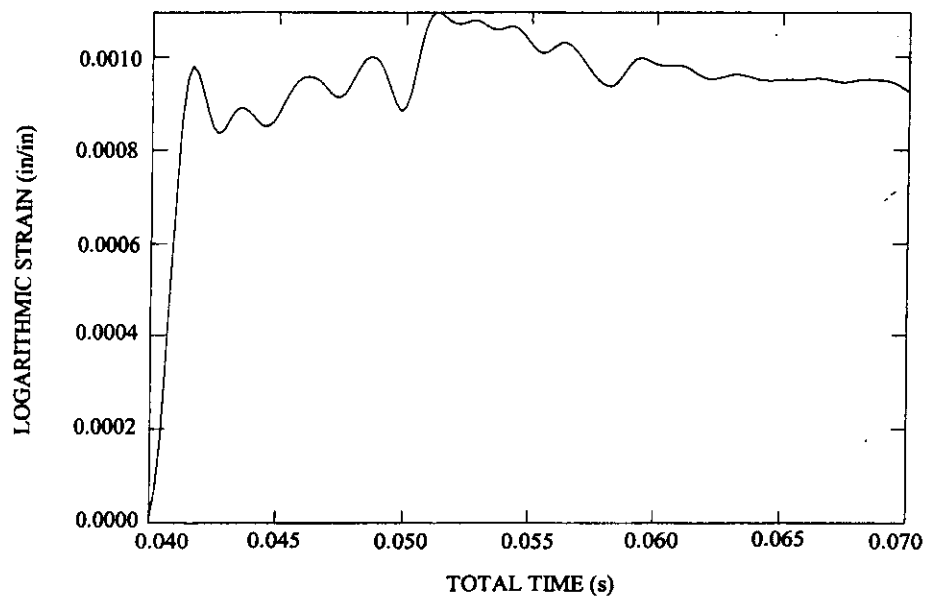


Table 37. Strain Energy Density at Containment Vessel from the Puncture Drop.

Containment		Closure bolt	
Element	Strain energy density (in-lb/in ³)	Element	Strain energy density (in-lb/in ³)
30741	1.899 E+03	34991	1.659 E+01
30742	5.655 E+02	34992	5.512 E-01
30743	2.783 E+02	34993	8.849 E-02
31081	2.453 E+03	34994	8.132 E-02
31082	1.574 E+03	34995	1.671 E+00
31083	1.298 E+03	34996	6.163 E+00
31121	6.770 E+02		
31122	7.927 E+02		
31123	1.268 E+03		
31161	3.058 E+02		
31162	4.984 E+02		
31163	8.915 E+02		
32081	1.312 E+02		
32082	2.504 E+02		
32083	4.925 E+02		
32121	3.792 E+02		
32122	3.344 E+02		
32123	4.351 E+02		
32161	7.009 E+02		
32162	4.885 E+02		
32163	4.358 E+02		
32921	2.661 E+02		
32922	4.105 E+01		
32923	1.770 E+01		

Figure 118. Shielded Drum Package Puncture Drop Strain Energy Density at the Containment Location (EL 31081).

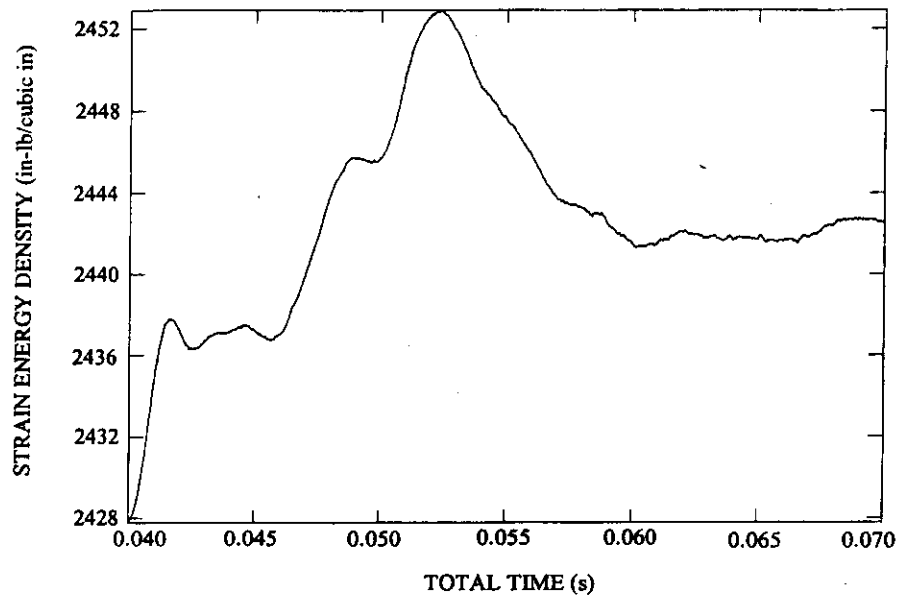
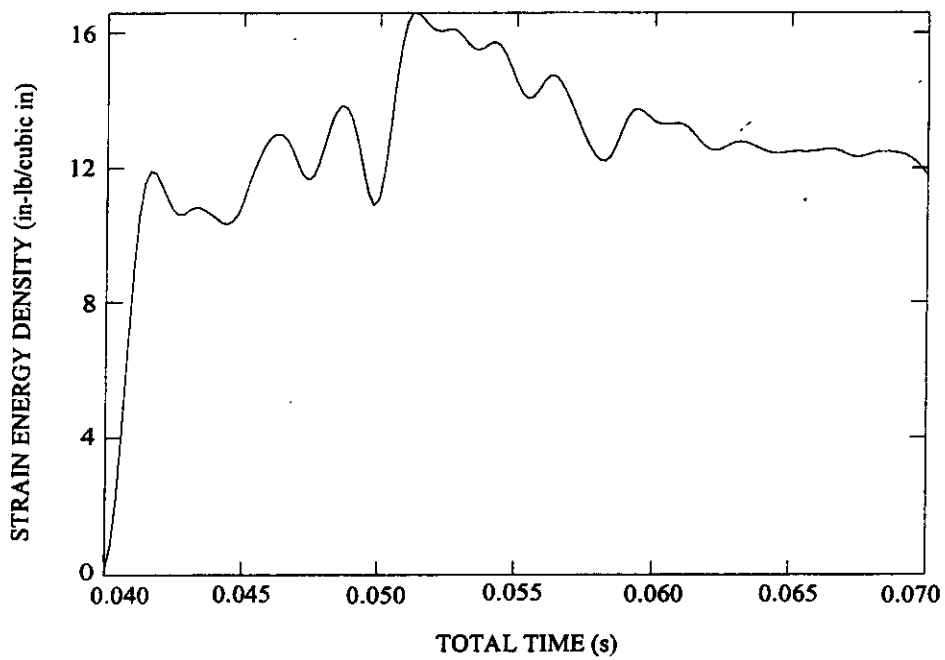


Figure 119. Shielded Drum Package Puncture Drop Strain Energy Density at Closure Bolt No. 1 (EL 34991).



The high stresses on the containment vessel are below the Service Level D limit of the material. However, the cumulative strains are much less than the ultimate strain of the material and are primarily compressive. The containment closure bolt stresses and cumulative strains are not significant and are not sufficient to completely decompress the seal, thereby maintaining containment. Figure 120 shows the deformation of the entire package at the end of series of numerical simulations. Figures 121 and 122 show the deformation of the lead shielding and containment vessel, respectively. Figures 123 through 125 show the TF, PEEQ, and the combined time history, respectively, at EL 30741.

In this puncture drop case, the highly stressed location of the containment vessel occurs at the junction of bottom flange plate and shell wall on the impact side. The high-localized stress magnitudes and regions are approximately the same as after the 9 m (30 ft) free drop. Consequently, for this package, puncture does not contribute significant loading on the containment vessel and only reduces shielding in a localized area surrounding the impact point.

Figure 120. Shielded Drum Package Puncture Drop Entire Package Deformation Plot.

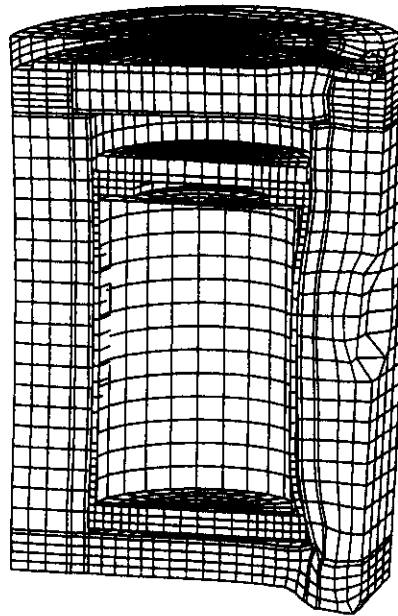


Figure 121. Shielded Drum Package Puncture Drop Drum Deformation Plot.

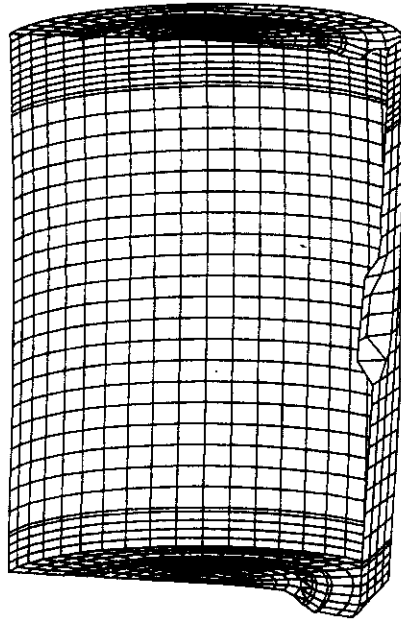


Figure 122. Shielded Drum Package Puncture Drop Containment Deformation Plot.

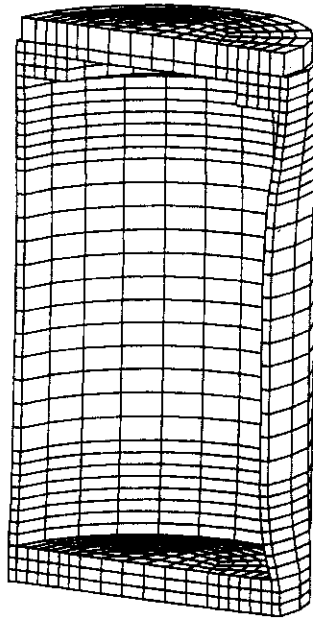


Figure 123. Triaxility Factor Plot at EL 30741.

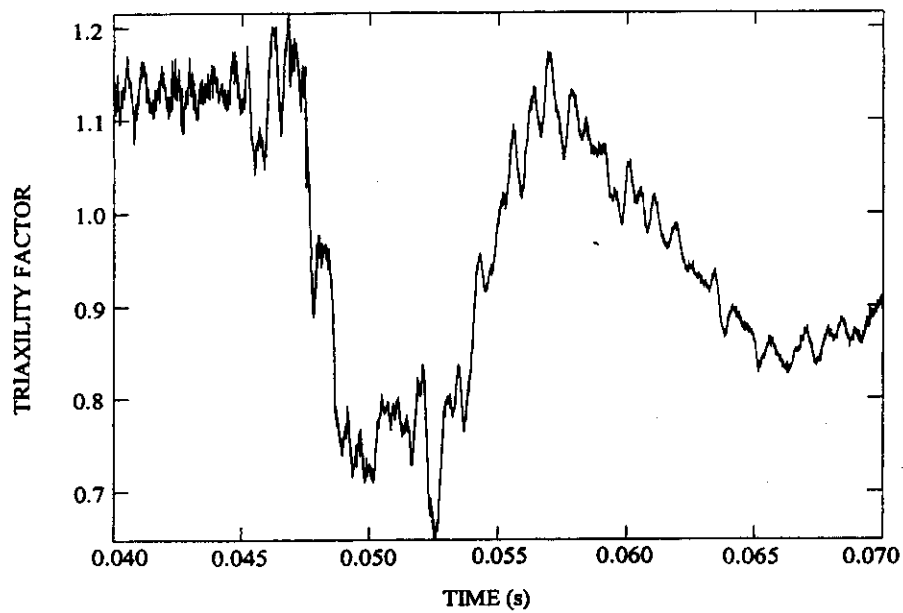


Figure 124. Equivalent Plastic Strain Plot at EL 30741.

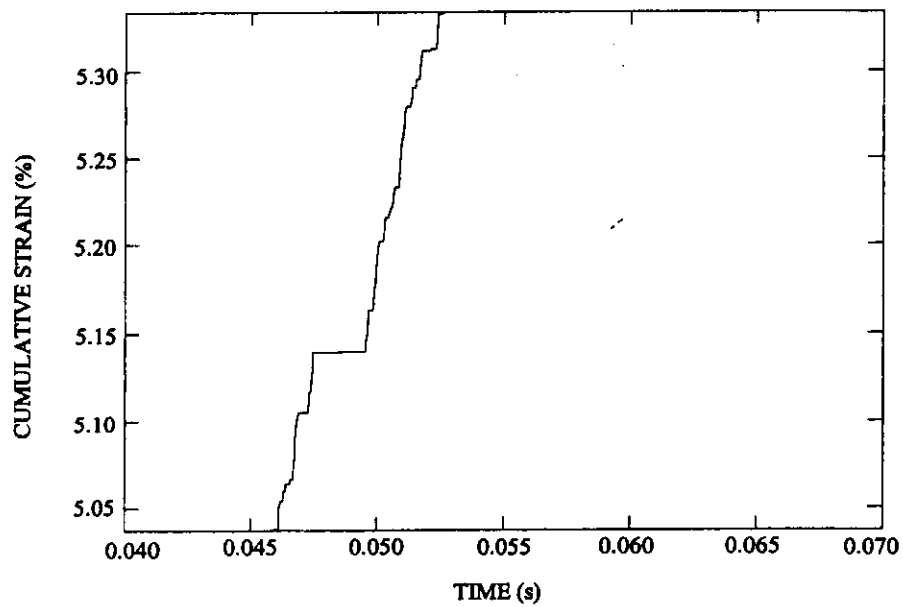
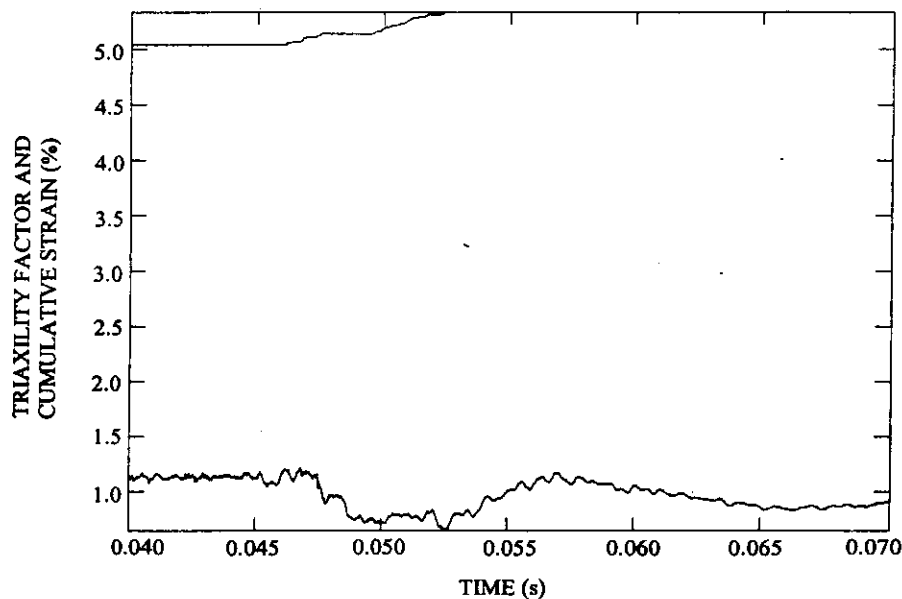


Figure 125. Combined Triaxiality Factor and Equivalent Plastic Strain Plot at EL 30741.



6.0 DISCUSSION OF RESULTS AND CONCLUSION

Based on this example, a shielded drum package containment vessel can be designed and constructed to ASME BPVC, Section VIII, Division 1 requirements which meets the minimum packaging standards. When the containment vessel is designed for a pressure of 1.4 MPa (200 psi) and for a higher temperature than the normal operating temperature, it has sufficient structural integrity to meet the severe loading requirements from the accident free drops defined in the minimum packaging standards. The cost of such a design in quantities of 20 or more would cost only \$2,100 and could be built by any commercial shop familiar with the ASME BPVC. Such a shop would have all the processes, qualifications, and QA in place to easily build the vessel.

The package is subjected to the minimum performance standards performance tests and evaluated to the criteria defined in the minimum performance standards basis and justification. The results show the containment vessel easily meets NC and AC requirements for transport. Under NC, the worst-case combined loading is from a 1 m (4 ft) horizontal side free drop at EL 32161. The combined loading from the free drop and pressure result in a stress intensity of 32 ksi at the junction of the reverse hub flange and shell wall. Consequently, this stress intensity can be treated as the combination of primary membrane plus primary bending plus secondary stress intensity. EL 32161 in the NC horizontal free drop is also the worst-case cumulative strain location. At this location, the maximum cumulative strain is 2.3% and when combined with the elastic strain from internal pressure loading. The total strain at this location is approximately 2.31%. This indicates that some minor deformation takes place at the joint. However, this deformation is highly localized and is only slightly above the elastic yield point of the material, which is permissible under Service Level C. The worst-case combined loading on the

containment closure bolts is from a flat bottom-end free drop at EL 34992. The stress intensity at this bolt is 19.82 ksi from the free drop and 25 ksi from internal pressure. The combined stress intensity of 44.82 ksi is less than the Service Level A limit of 57.3 ksi. The highest cumulative strain on the containment closure bolts is at EL 34996 with a value of 0.89% resulting in only a 0.033 cm (0.013 in.) maximum elongation of the bolt. Considering there is a 30% compression of the seal, the compressive displacement of the seal is 0.191 cm (0.075 in.). Consequently there is a large margin of safety. Summaries of NC and AC results at worst-case locations are provided in Tables 38 and 39.

Table 38. Normal Transport Conditions Load Combinations.

Drop orientation	Containment vessel		Closure bolts	
	Element number	Margin of safety	Element number	Margin of safety
1 m (4 ft) horizontal side drop	32161	0.54	34991	0.39
1 m (4 ft) bottom-end drop	32851	0.27	34992	0.28
1 m (4 ft) bottom corner drop	32161	1.45	34991	large margin
1 m (4 ft) top corner drop	31081	1.21	34991	0.64

For the NC case, only stress intensity was evaluated.

Table 39. Accident Conditions Load Combinations.

Drop orientation	Containment vessel (element number/margin of safety)			Closure bolts (element number/margin of safety)		
	Stress intensity	Strain	Strain energy density	Stress intensity	Strain	Strain energy density
9 m (30 ft) horizontal side drop	31081/0.03	31123/5.25	31081/9.23	34991/1.37	34991/large margin	34991/large margin
9 m (30 ft) bottom-end drop	31083/0.18	31092/2.83	31092/13.30	34991/4.41	34996/large margin	34996/10.20
9 m (30 ft) bottom corner drop	32121/0.60	32121/large margin	32121/large margin	34996/7.28	34996/large margin	34996/large margin
9 m (30 ft) top corner drop	32321/0.23	31081/10.08	32161/14.44	34991/0.96	34996/large margin	34996/large margin
puncture drop	—	30741/0.89	31081/3.73	—	34991/large margin	34991/large margin

As demonstrated in the results, the worst-case loading results from the horizontal side drop. Consequently, the package is subjected to NC and AC horizontal side free drops in sequence, followed by a puncture drop on the impact side. Results from the numerical simulation of the test sequence with the superimposition of the internal pressure load demonstrate the satisfactory structural performance of the package in accordance with both the regulations and minimum packaging standards. The test sequence based on regulations and NRC interpretations is a

normal free drop followed by an accident free drop followed by a puncture, all at the same location on the package. As a result, for this evaluation, the free drop orientation, which results in the most severe damage to the package, is used for this sequence of events. In this evaluation, it was determined that the most severe loads on the package are from a horizontal side free drop. The results of this evaluation show that in the case of sequential free drops and puncture loadings, stress is of little use in assessing damage to the containment vessel since there is no sustained loading. The only meaningful parameter for assessing damage to the containment vessel is cumulative strain. Based on the cumulative strain after puncture, the highest strain occurs at EL 30741 on the bottom-end closure at the juncture with the shell wall. The value of the cumulative strain is approximately 9.4%, indicating maximum damage to the containment vessel at this location. The time history of the TF shows that during the time of loading, the maximum TF is less than 1, indicating less than uniaxial tension. Consequently, the maximum adjusted cumulative strain at this location is conservatively assumed as 9.4%. The total accumulated triaxial strain limit for this location condition, since it is at a weld, is 17.5%. The margin of safety at this most highly strained location is approximately 86%. The largest SED is 2,453 in-lb/in³ at EL 31081, which occurs after puncture on the shell wall juncture with the bottom-end closure. The minimum permissible SED of the material is 1.16×10^4 in-lb/in³. As a result, the lowest margin of safety against material rupture is very large. For the closure bolts, the largest cumulative strain occurs after the flat bottom-end drop at EL 34996 with a value of 0.83%. Combining this value with the NC condition maximum cumulative strain, the total elongation of the bolt is 0.066 cm (0.026 in.). Based on the compression of the seal, the margin of safety against seal decompression remains large. Also, the total maximum cumulative strain is much less than the uniaxial tensile ultimate elongation of the bolt material. This is further evidenced by the large margin of safety on the SED of the bolt material. The worst SED on the containment closure bolt is only 2,527 in-lb/in³, as compared with the specified allowable limit of 1.53×10^4 in-lb/in³. As demonstrated in the results, the containment vessel is damaged and experiences some deformation, but maintains containment.

Design, validation, and weight calculations are provided in Section 8.0.

7.0 REFERENCES

- 10 CFR 71, "Packaging and Transportation of Radioactive Material," *Code of Federal Regulations*, as amended.
- ASME, 1995a, *Boiler and Pressure Vessel Code*, Section VIII, Division 1, American Society of Mechanical Engineers, New York, New York.
- ASME, 1995b, *Boiler and Pressure Vessel Code*, Section III, Subsection NE, American Society of Mechanical Engineers, New York, New York.
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IAEA, 1990, *Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material*, IAEA Safety Guides, Safety Series No. 37, International Atomic Energy Agency, Vienna, Austria.

MIL-HDBK-5G, *Military Handbook-Metallic Materials and Elements for Aerospace Vehicle Structures*, MIL-HDBK-5G, U.S. Department of Defense, Washington, D.C.

NUREG/CR-6007, *Stress Analysis of Closure Bolts for Shipping Casks*, 1993, (under Lawrence Livermore National Laboratory contract to the NRC), U.S. Nuclear Regulatory Commission, Washington, D.C.

Regulatory Guide 7.6, *Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels*, U.S. Nuclear Regulatory Commission, Washington, D.C.

Regulatory Guide 7.8, *Load Combinations for the Structural Analysis of Shipping Casks for Radioactive Material*, U.S. Nuclear Regulatory Commission, Washington, D.C.

WMTS, 1999, *ABAQUS Version 5.8-1 Verification on PX17265 Computer*, ENG-VV-010, Waste Management Technical Services, Richland, Washington, July 29, 1999.

8.0 APPENDICES

8.1 DESIGN CALCULATIONS



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1.0 OBJECTIVE

The objective of this calculation is to determine the design for a generic shielded drum packaging containment vessel for the Hanford Site that meets the performance tests of the minimum packaging standards.

2.0 REFERENCES

- ASME, 1992, *Boiler and Pressure Vessel Code*, Section VIII, Division 1, American Society of Mechanical Engineers, New York, New York.
- Bickford, J. H., 1990, *An Introduction to the Design and Behavior of Bolted Joints*, Second Edition, Marcel, Dekker, Inc., New York, New York.
- Mok, G. C., L. E. Fischer, and S. T. Hsu, 1993, *Stress Analysis of Closure Bolts for Shipping Casks*, (prepared by Lawrence Livermore National Laboratory and Kaiser Engineering for the U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards), NUREG/CR-6007, U.S. Nuclear Regulatory Commission, Washington, D.C., January 1993.

3.0 ASSUMPTIONS, RESULTS, AND CONCLUSIONS

In this analysis, the following assumptions were made:

- The internal design pressure is assumed to be 200 psi.
- The design temperature is assumed to be 400 °F.
- The joint efficiency, as determined from the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPVC), Section VIII, Division 1, is 0.6. Note that the welds are full penetration, but not radiographed.
- The gasket for the shielded drum package is an O-ring and is self-energizing.
- The closure bolt nut factor is 0.2.



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The ASME BPVC, Section VIII, Division 1 was used to design a generic shielded drum package containment vessel for the Hanford Site. This design includes specifying material and thickness for all structures associated with the containment vessel (see Figure 1).

The cylindrical shell is a thin-walled vessel. The analysis shows a minimum wall thickness of 0.14 in. It was determined that a 12 in. Schedule 40 pipe, with an outside diameter of 12.75 in., inside diameter of 11.938 in., and a wall thickness of 0.406 in. would be used. The shell is constructed SA-312, TP304L stainless steel.

The bottom-end closure is a flat, circular unstayed blind flange with a diameter of 12.75 in. A minimum thickness of 1.1 in. was calculated; therefore, it was determined the thickness of the bottom-end closure would be 1.5 in. The bottom-end closure is constructed of SA-240, 304L stainless steel plate.

The closure lid diameter is 12.75 in. The minimum thickness of the closure lid was determined to be 1.12 in. Therefore, the thickness used for the design will be 1.5 in. and the material of construction is SA-240, 304L stainless steel plate. It was determined that 12 closure bolts would be used, constructed of SA-193, B8.

The reverse hub flange used for the shielded drum package is an integral flange. The outside diameter is 12.75 in. and the inside diameter is 8 in. The bolt circle diameter is 10 in. The thickness was determined to be 1.5 in. and the flange is constructed of SA-240, 304L stainless steel plate.

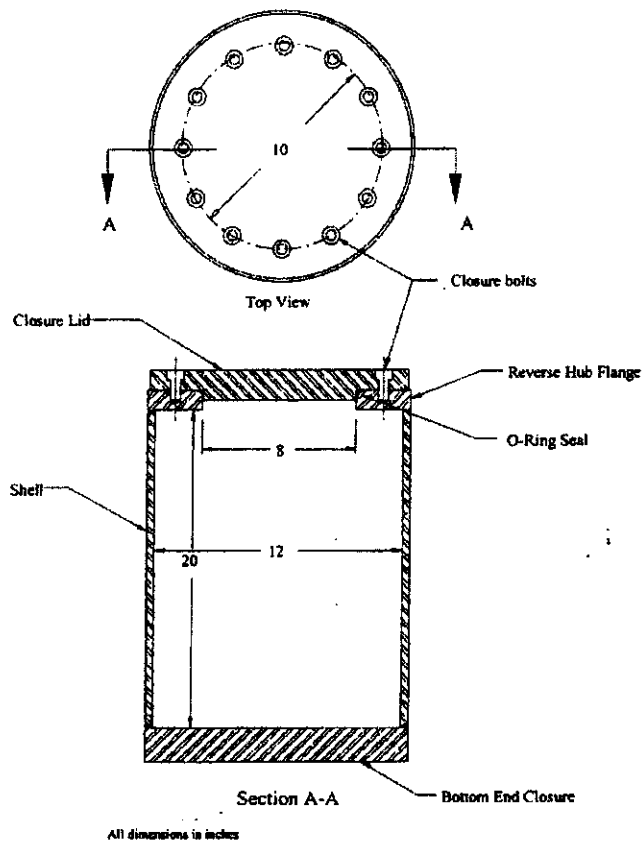
The flange was also checked for rigidity and determined to be within the limit of the rigidity index. The J factor is less than 1. This calculation demonstrates that the joint has sufficient rigidity to provide a leak tight joint.



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Figure 1.





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4.0 EVALUATION

MINIMUM PACKAGING STANDARDS

Use ASME BPVC, Section VIII, Division 1, Subsection A, Part UG, Subpart UG-27. This is to check the shell thickness.

internal design pressure: $P := 200 \text{ psi}$

inside radius of shell course under consideration: $R := \frac{12 \text{ in}}{2}$ $R = 6 \text{ in}$

maximum allowable stress value (assume material is SA-312, TP 304L) at 400 °F: $S := 14.7 \text{ ksi}$

joint efficiency for appropriate joint in cylindrical shell (from ASME BPVC, Section VIII, Division 1, Subsection A, Part UW, Subpart UW-12: $E := 0.6$
 Note that welds are full penetration groove, but not radiographed.

Cylindrical shells:

(1) circumferential stress (longitudinal joints) [Note: this step is not needed since SA-312 smls pipe is used.]:

$$t_1 := \frac{P \cdot R}{S \cdot E - 0.6 \cdot P} \quad t_1 = 0.14 \text{ in}$$

(2) longitudinal stress (circumferential joints):

$$t_2 := \frac{P \cdot R}{2 \cdot S \cdot E + 0.4 \cdot P} \quad t_2 = 0.07 \text{ in}$$

The vessel will be thin-walled since the ratio of the mean radius to the thickness is > or equal to 10:

$$\frac{R}{t_2} = 88.6$$

A 12 in. Schedule 40 pipe with the following dimensions will be used: OD 12.75 in., ID 11.938 in., and wall thickness of 0.406 in.



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For the bottom-end closure, use ASME BPVC, Section VIII, Division 1, Subsection A, Part UG, Subpart UG-34. The bottom-end closure is a flat, circular unstayed blind flange. It is assumed to be constructed of SA-240, 304L.

C factor - use Figure UG-34 (h): $C := 0.33$

diameter: $d := 12.75 \text{ in}$

internal design pressure - given above: $P = 200 \text{ psi}$

maximum allowable stress value (assume material is SA-240, 304L) at 400 °F: $S_{be} := 14.7 \text{ ksi}$

joint efficiency - given above: $E = 0.6$

Use equation 1 to find the minimum thickness: $t_{be} := d \cdot \sqrt{\frac{C \cdot P}{S \cdot E}}$ $t_{be} = 1.1 \text{ in}$

Assume a shell thickness of 1.5 in.

To find the number and required strength of the closure bolts, ASME BPVC, Section VIII, Division 1, Appendix 2, Subpart 2-5.

diameter at location of gasket load reaction: $G := 8.25 \text{ in}$

internal design pressure - given above: $P = 200 \text{ psi}$

effective gasket or joint-contact-surface seating width: $b := \frac{1}{4} \text{ in}$

gasket factor (is an O-ring and self-energizing): $m := 0$

Required bolt load for operating conditions:

$$W_{m1} := 0.785 \cdot G^2 \cdot P + (2 \cdot b \cdot 3.14 \cdot G \cdot m \cdot P) \quad W_{m1} = 10685.8 \text{ lbf}$$



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gasket seating load: $y := 0 \cdot \text{psi}$

Minimum initial bolt load required: $W_{m2} := 3.14 \cdot b \cdot G \cdot y$ $W_{m2} = 0 \cdot \text{lbf}$

Total required and actual bolt areas:

allowable bolt stress at atmospheric temperature (100 °F): $S_a := 25 \cdot \text{ksi}$

allowable bolt stress at design temperature (400 °F): $S_b := 25 \cdot \text{ksi}$

nominal diameter of bolt: $D := \frac{1}{2} \cdot \text{in}$ threads per inch: $n := \frac{13}{\text{in}}$

number of bolts: $N_b := 12$

cross-sectional area of bolts using
root diameter of thread: $A_b := 0.7854 \cdot \left(D - \frac{1.3}{n}\right)^2$ $A_b = 0.13 \cdot \text{in}^2$

total cross-sectional area of bolts at root of thread,
required for operating conditions: $A_{m1} := \frac{W_{m1}}{S_b}$ $A_{m1} = 0.4 \cdot \text{in}^2$

total cross-sectional area of bolts at root of thread,
required for gasket seating: $A_{m2} := \frac{W_{m2}}{S_a}$ $A_{m2} = 0 \cdot \text{in}^2$

A_m is the greater of the two: $A_m := A_{m1}$

number of bolts needed: $N_{\text{need}} := \frac{W_{m1}}{S_b \cdot A_b}$ $N_{\text{need}} = 3$

$A_t := N_b \cdot A_b$ $A_t = 1.51 \cdot \text{in}^2$

Assume 12 bolts are used.



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Flange design bolt load:

for operating conditions: $W_1 := W_{m1}$

for gasket seating: $W_2 := \frac{(A_m + A_b) \cdot S_a}{2}$ $W_2 = 6913.7 \text{ lbf}$

For the closure lid thickness, use ASME BPVC, Section VIII, Division 1, Subsection A, Part UG, Subpart UG-34. The closure flange is a bolted flat circular unstayed blind flange. It is assumed to be constructed on of SA-240, 304L.

C factor - use Figure UG-34 (k): $C_1 := 0.3$

diameter - same as above for bottom plate: $d = 12.75 \text{ in}$

internal design pressure - same as above: $P = 200 \text{ psi}$

maximum allowable stress value - same as above: $S_{be} = 14.7 \text{ ksi}$

joint efficiency - same as above: $E = 0.6$

total bolt load given for circular heads - given above: $W_1 = 10685.8 \text{ lbf}$

gasket moment arm: $h_G := 5 \text{ in} - 4 \text{ in} - \frac{1}{8} \text{ in}$ $h_G = 0.88 \text{ in}$

Use equation 2 to find minimum required thickness:

$$t_{cl} := d \cdot \sqrt{\frac{C_1 \cdot P}{S_{be} \cdot E} + \frac{1.9 \cdot W_1 \cdot h_G}{S_{be} \cdot E \cdot d^3}} \quad t_{cl} = 1.12 \text{ in}$$

Closure lid is assumed to be 1.5 in. thick.



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Initial Preload Torque

nut factor: $K := 0.2$

Bickford recommendation for initial preload torque to achieve 25 ksi (in-lb) - size 1/2-13 UNC

$$B_{rec} := 354.8 \text{ in-lbf}$$

nominal diameter of closure bolt - given above: $D = 0.5 \text{ in}$

From NUREG/CR-6007, determine preload bolt force per bolt:

$$F_a := \frac{B_{rec}}{K \cdot D}$$

$$F_a = 3548 \text{ lbf} \quad \text{for 1 bolt.}$$

$$F_a \cdot 12 = 42576 \text{ lbf} \quad \text{for 12 bolts. This is greater than } W_{m1} \text{ and } W_2.$$

Thickness of Reverse Hub Flange. The reverse hub flange is assumed to be constructed of SA-240, 304L.

Depth of bolt holes should be about 2.5 times the diameter of the bolt:

$$\frac{1}{2} \text{ in} \cdot 2.5 = 1.25 \text{ in}$$

For conservatism, and considering the bottom flange will be 1.5 in. thick, consider the reverse hub flange to be 1.5 in. thick.

Use ASME BPVC, Section VIII, Division 1, Appendix 2, Subpart 2-13 to check flange thickness:

inside diameter of shell: $B := 12 \text{ in}$ inside diameter of reverse flange: $B' := 8 \text{ in}$

bolt circle diameter: $C_b := 10 \text{ in}$ outside diameter of flange: $A := 12.75 \text{ in}$



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total hydrostatic end force on attached component: $H := 0.785 \cdot G^2 \cdot P$ $H = 10685.8 \cdot \text{lbf}$

hydrostatic end force on area inside of flange: $H_D := 0.785 \cdot B^2 \cdot P$ $H_D = 22608 \cdot \text{lbf}$

difference between hydrostatic end force on attached
 component and hydrostatic end force on area inside
 of flange: $H_T := H - H_D$
 $H_T = -11922.2 \cdot \text{lbf}$

thickness of hub: $g_o := 0.406 \cdot \text{in}$ $g_1 := g_o$

radial distance from bolt circle to circle on which H_D acts:

$$h_D := \frac{(C_b + g_1 - 2 \cdot g_o - B)}{2} \quad h_D = -1.2 \cdot \text{in}$$

factor: $h_{or} := \sqrt{A \cdot g_o}$ $h_{or} = 2.28 \cdot \text{in}$

radial distance from bolt circle to circle on which H_T acts:

$$h_T := \frac{1}{2} \cdot \left(C_b - \frac{B + G}{2} \right) \quad h_T = -0.0625 \cdot \text{in}$$

ratio of outside diameter of flange to inside diameter of flange:

$$K := \frac{A}{B} \quad K = 1.6$$

$$Z := \frac{K^2 + 1}{K^2 - 1} \quad Z = 2.3$$



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$$\text{factor: } Y := \frac{1}{K-1} \left(0.66845 + 5.71690 \cdot \frac{K^2 \cdot \log(K)}{K^2 - 1} \right) \quad Y = 4.3$$

$$\text{factor: } T := \frac{K^2 \cdot (1 + 8.55246 \cdot \log(K)) - 1}{(1.04720 + 1.9448 \cdot K^2) \cdot (K - 1)} \quad T = 1.7$$

$$\text{factor: } U := \frac{K^2 \cdot (1 + 8.55246 \cdot \log(K)) - 1}{1.36136 \cdot (K^2 - 1) \cdot (K - 1)} \quad U = 4.8$$

$$\alpha_r := \frac{\left[1 + \frac{0.668 \cdot (K + 1)}{Y} \right]}{K^2} \quad \alpha_r = 0.6$$

$$T_r := \left(\frac{Z + 0.3}{Z - 0.3} \right) \cdot \alpha_r \cdot T \quad T_r = 1.2$$

$$U_r := \alpha_r \cdot U \quad U_r = 2.6$$

$$Y_r := \alpha_r \cdot Y \quad Y_r = 2.4$$

hub length: $h := 0 \cdot \text{in}$



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For V:

$$\frac{h}{h_{or}} = 0$$

$$V := 0.550103$$

$$\frac{g_1}{g_0} = 1$$

$$d_r := \frac{U_r \cdot h_{or} \cdot g_0^2}{V}$$

$$d_r = 1.79 \cdot \text{in}^3$$

$$F := 0.908920$$

$$e_r := \frac{F}{h_{or}}$$

$$e_r = 0.4 \cdot \text{in}^{-1}$$

$$f := 1$$

$$\text{thickness: } t := 1.5 \cdot \text{in}$$

$$\text{factor: } L_r := \frac{t \cdot e_r + 1}{T_r} + \frac{t^3}{d_r}$$

$$L_r = 3.2$$

radial distance from gasket load reaction to bolt circle:

$$h_G := \frac{(C_b - G)}{2}$$

$$h_G = 0.88 \cdot \text{in}$$

$$\text{gasket load: } H_G := W_1 - H$$

$$H_G = 0 \cdot \text{lbf}$$



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component of moment due to H_D : $M_D := H_D \cdot h_D$ $M_D = -27197.5 \text{ in-lbf}$

component of moment due to H_G : $M_G := H_G \cdot h_G$ $M_G = 0 \text{ in-lbf}$

component of moment due to H_T : $M_T := H_T \cdot h_T$ $M_T = 745.1 \text{ in-lbf}$

total moment acting on the flange, for operating conditions:

$$M_o := |M_D + M_G + M_T| \quad M_o = 26452.3 \text{ in-lbf}$$

Stresses at Outside Diameter:

$$S_H := \frac{f M_o}{L_r \cdot g_l^2 \cdot B'} \quad S_H = 6227.6 \text{ psi}$$

$$S_R := \frac{(1.33 \cdot t \cdot e_r + 1) \cdot M_o}{L_r \cdot t^2 \cdot B'} \quad S_R = 819.8 \text{ psi}$$

$$S_{T1} := \left(\frac{Y_r \cdot M_o}{t^2 \cdot B'} \right) - \frac{Z \cdot S_R \cdot (0.67 \cdot t \cdot e_r + 1)}{(1.33 \cdot t \cdot e_r + 1)} \quad S_{T1} = 2043.8 \text{ psi}$$



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Stress at Inside Diameter B'

$$ST_2 := \left(\frac{M_o}{t^2 \cdot B'} \right) \cdot \left[Y - \frac{2 \cdot K^2 \cdot \left(1 + \frac{2}{3} \cdot t \cdot e_r \right)}{(K^2 - 1) \cdot L_r} \right] \quad ST_2 = 4272.3 \text{ psi}$$

Check of Flange Rigidity - ASME BPVC, Section VIII, Division 1, Appendix S.

elastic modulus of flange at design temperature: $E_m := 26.5 \cdot 10^6 \text{ psi}$

rigidity factor for integral flange: $K_I := 0.3$

inside diameter of flange: $B_{id} := 8 \text{ in}$

factor: $h_o := \sqrt{B_{id} \cdot g_o} \quad h_o = 1.8 \text{ in}$

e (for an integral flange): $e_{if} := \frac{F}{h_o} \quad e_{if} = 0.5 \text{ in}^{-1}$

factor (for integral flange): $d_{if} := \frac{U}{V} \cdot h_o \cdot g_o^2 \quad d_{if} = 2.6 \text{ in}^3$

factor: $L := \frac{t \cdot e_{if} + 1}{T} + \frac{t^3}{d_{if}} \quad L = 2.4$

integral flange: $J := \left(\frac{52.14 \cdot M_o \cdot V}{L \cdot E_m \cdot g_o^2 \cdot h_o \cdot K_I} \right) \quad J = 0.14$

Therefore, J is within the limit of the rigidity index.

8.2 VALIDATION OF DESIGN CALCULATIONS



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1.0 OBJECTIVE

The objective of this calculation is to confirm the adequacy of the design and stress intensities for the normal transfer condition pressure load for a generic shielded drum packaging containment vessel for the Hanford Site per American Society of Mechanical Engineering (ASME) *Boiler and Pressure Vessel Code* (BPVC) Section III, Subsection NE. Also, stresses and strains will be developed from pressure for combination with normal and accident loads.

2.0 REFERENCES

- ASME, 1992a, *Boiler and Pressure Vessel Code*, Section III, Appendices, American Society of Mechanical Engineers, New York, New York.
- ASME, 1992b, *Boiler and Pressure Vessel Code*, Section III, Subsection NE, American Society of Mechanical Engineers, New York, New York.
- Bickford, J. H., 1990, *An Introduction to the Design and Behavior of Bolted Joints*, Second Edition, Marcel, Dekker, Inc., New York, New York.
- Mok, G. C., L. E. Fischer, and S. T. Hsu, 1993, *Stress Analysis of Closure Bolts for Shipping Casks*, (prepared by Lawrence Livermore National Laboratory and Kaiser Engineering for the U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards), NUREG/CR-6007, U.S. Nuclear Regulatory Commission, Washington, D.C., January 1993.

3.0 ASSUMPTIONS, RESULTS, AND CONCLUSIONS

In this analysis, the following assumptions were made:

- The maximum normal operating pressure (MNOP) is 100 psi.
- The maximum normal temperature is 200 °F.



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- The containment vessel is conservatively idealized as a cylindrical shell with two flat circular closure plates. Note that the top and bottom of the containment vessel are conservatively assumed to have the same thickness (1.5 in.). See Figures 1 and 2.
- The vessel is evaluated to ASME BPVC, Section III, Subsection NE with methods outlined in ASME BPVC, Section III, Appendix F. The bolts are evaluated to NUREG/CR-6007.

The following table delineates the margins of safety from the applied internal pressure load:

Component	Margin of Safety
Shell	
body	9.9
junction to flange	29.5
Top and bottom closure plates	
center of plate	4.4
Bolts	
tension	1.3
shear	3.2
tension + shear	0.2
maximum stress intensity	0.8

For superposition of the loads (normal and accident conditions load combinations), the following stress intensities, principal strains, and strain energy density at key locations are provided:

Component	Value
stress intensity on shell body	1.5 ksi
stress intensity at junctions	1.6 ksi
principal elastic strain on shell body	4.55×10^{-5}
principal elastic strain at junction	9.46×10^{-5}
strain energy density at wall	0.2 in-lb/in ³
strain energy density at junction	0.4 in-lb/in ³

This calculation of the containment vessel as designed will maintain containment as demonstrated by the positive margin of safety. The generic shielded drum analysis confirms that the material and thickness of each component will pass with positive margins of safety.



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Figure 1. Cylindrical Shell.

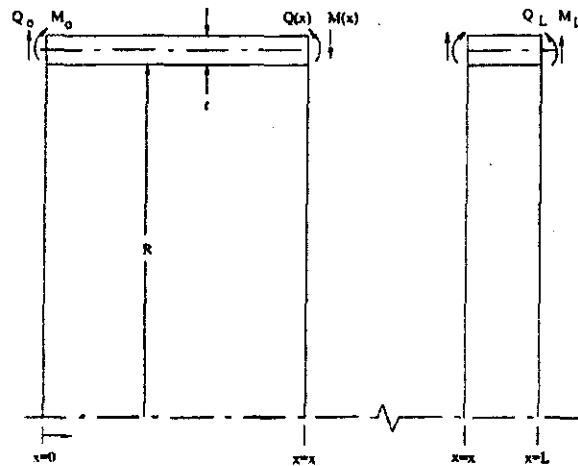
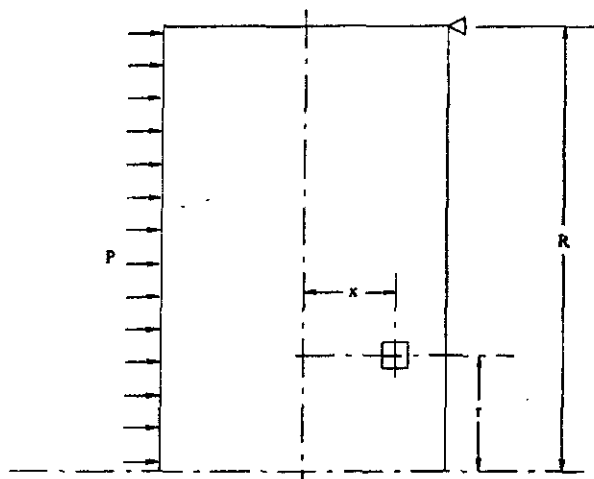


Figure 2. Flat Circular Heads.





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4.0 EVALUATION

MINIMUM PACKAGING STANDARDS . Confirming adequacy of design and stress intensities for NTC load combinations.

Note: The top and bottom are analyzed together since they are conservatively assumed the same.

maximum normal operating pressure (at maximum normal temperature of 200 °F): MNOP := 100 ·psi

membrane allowable at 200 °F: Sm := 16.5 ·ksi

The containment vessel is conservatively idealized as a cylindrical shell with two flat circular closure plates.

The shell principal stresses are determined using ASME BPVC, Section III, Appendices, Article A-2000, Subarticle A-2212, Principal Stresses.

internal pressure - given above: p := MNOP

outside radius of shell: $OR_s := \frac{12.75 \cdot \text{in}}{2}$ inside radius of shell: $IR_s := \frac{11.938 \cdot \text{in}}{2}$

ratio of outside radius to an intermediate radius: $Z := \frac{OR_s}{\left(\frac{OR_s + IR_s}{2} \right)}$ Z = 1.03

ratio of outside radius to inside radius: $Y := \frac{OR_s}{IR_s}$ Y = 1.07

Tangential (circumferential) stress component:

$$\sigma_1 := \frac{p \cdot (1 + Z^2)}{(Y^2 - 1)} \quad \sigma_1 = 1469.4 \cdot \text{psi}$$

Longitudinal (meridional) stress component:

$$\sigma_2 := \frac{p}{(Y^2 - 1)} \quad \sigma_2 = 710.9 \cdot \text{psi}$$



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Radial stress component:

$$\sigma_3 := \frac{p \cdot (1 - Z^2)}{(Y^2 - 1)} \quad \sigma_3 = -47.5 \text{ psi}$$

ASME BPVC, Section III, Appendices, Article A-2000, Subarticle A-2221, General Primary Membrane Stress Intensity (shell).

inside radius: $R := IR_s$ thickness of cylinder: $t := 0.406 \text{ in}$

$$S_{pm} := \left(\frac{p \cdot R}{t} \right) + \left(\frac{p}{2} \right) \quad S_{pm} = 1.5 \text{ ksi}$$

Margin of safety: $MS_{pm} := \frac{S_m}{S_{pm}} - 1 \quad MS_{pm} = 9.9$

ASME BPVC, Section III, Appendices, Article A-2000, Subarticle A-2222, Maximum Value of Primary Plus Secondary Stress Intensity (on shell wall).

$$S_{psm} := 2 \cdot p \cdot \frac{Y^2}{Y^2 - 1} \quad S_{psm} = 1.6 \text{ ksi}$$

Margin of safety: $MS_{psm} := \frac{3 \cdot S_m}{S_{psm}} - 1 \quad MS_{psm} = 29.5$

Determine principal stresses from pressure at the center of the conservatively idealized closure flanges. ASME BPVC, Section III, Appendices, Article A-5000, Subarticle A-5212.

pressure - same as above: $p = 100 \text{ psi}$ thickness of plate: $t_p := 1.5 \text{ in}$

longitudinal distance from midplane of plate: $x := \frac{t_p}{2} \quad x = 0.75 \text{ in}$

Poisson's ratio: $\nu := 0.29$

outside radius of plate (note that the closure lid is idealized as a solid plate with no cut-outs): $R_p := \frac{12.75 \text{ in}}{2} \quad R_p = 6.38 \text{ in}$

radial distance at center of plate: $r := 0 \text{ in}$



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Radial bending stress: $\sigma_r := p \cdot \frac{3 \cdot (x)}{4 \cdot t_p^3} \cdot \left[(3 + \nu) \cdot (R_p^2 - r^2) \right]$ $\sigma_r = 2.2 \text{ ksi}$
 $\sigma_{3a} := \sigma_r$

Tangential bending stress: $\sigma_t := p \cdot \frac{3 \cdot (x)}{4 \cdot t_p^3} \cdot \left[(3 + \nu) \cdot R_p^2 - (1 + 3 \cdot \nu) \cdot r^2 \right]$
 $\sigma_t = 2.2 \text{ ksi}$ $\sigma_{1a} := \sigma_t$

Longitudinal stress: $\sigma_{ls} := \left(\frac{p}{t_p} \right) \cdot \left(x - \frac{t_p}{2} \right)$ $\sigma_{ls} = 0 \text{ ksi}$
 $\sigma_{2a} := \sigma_{ls}$

Using ASME BPVC, Section III, Subsection NE, Paragraph NE-3215 (c), the stresses are calculated at the center of plate:

$S_{12} := |\sigma_{1a} - \sigma_{2a}|$ $S_{12} = 2.2 \text{ ksi}$
 $S_{23} := |\sigma_{2a} - \sigma_{3a}|$ $S_{23} = 2.2 \text{ ksi}$
 $S_{31} := |\sigma_{3a} - \sigma_{1a}|$ $S_{31} = 0 \text{ ksi}$

The largest stress intensity is S_{12} .

Use ASME BPVC, Section III, Subsection NE, Table NE-3217-1 to determine the type of stress intensity. It is primary bending.

$MS_{pb} := \frac{S_m}{S_{12}} - 1$ $MS_{pb} = 6.4$



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Use ASME BPVC, Section III, Appendices, Article A-5000, Subarticle A-5223.2.

$$f := \frac{0.406 \cdot \text{in}}{\left(\frac{12.75 \cdot \text{in}}{2} \right)} \quad f = 0.06$$

$$\text{geometry constant: } F_3 := \frac{3}{8} \cdot (1 - \nu) \cdot (2 - f) \cdot (8 - f(4 - f) \cdot (1 - \nu)) \quad F_3 = 4$$

elastic modulus of plate at 200°F: $E_p := 27.6 \cdot 10^6 \cdot \text{psi}$

outside radius of plate: $R_o := \frac{12.75 \cdot \text{in}}{2} \quad R_o = 6.4 \cdot \text{in}$

Use ASME BPVC, Section III, Appendices, Article A-2000, Subarticle A-2243. The top and bottom plates are considered the same.

modulus of elasticity at 200°F of cylinder wall (SA-312, TP304L): $E := 27.6 \cdot 10^6 \cdot \text{psi}$

$$D := \frac{E \cdot t^3}{12 (1 - \nu^2)} \quad D = 168057.7 \cdot \text{in} \cdot \text{lbf}$$

inside radius: $R = 6 \cdot \text{in}$ thickness of the cylinder: $t = 0.406 \cdot \text{in}$

$$\beta := \left[\frac{3 \cdot (1 - \nu^2)}{\left(R + \frac{t}{2} \right)^2 \cdot t^2} \right]^{\frac{1}{4}} \quad \beta = 0.8 \cdot \text{in}^{-1}$$

axial distance measured from the reference end of cylinder: $x_1 := 0$

length of cylinder: $L := 20 \cdot \text{in}$



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$$B_{11} := \frac{(\sinh(2 \cdot \beta \cdot L) - \sin(2 \cdot \beta \cdot L))}{2 \cdot (\sinh(\beta \cdot L)^2 - \sin(\beta \cdot L)^2)} \quad B_{11} = 1$$

$$B_{12} := \frac{(\cosh(2 \cdot \beta \cdot L) - \cos(2 \cdot \beta \cdot L))}{2 \cdot (\sinh(\beta \cdot L)^2 - \sin(\beta \cdot L)^2)} \quad B_{12} = 1$$

$$B_{22} := \frac{(\sinh(2 \cdot \beta \cdot L) + \sin(2 \cdot \beta \cdot L))}{2 \cdot (\sinh(\beta \cdot L)^2 - \sin(\beta \cdot L)^2)} \quad B_{22} = 1$$

$$G_{11} := -\frac{(\cosh(\beta \cdot L) \cdot \sin(\beta \cdot L) - \sinh(\beta \cdot L) \cdot \cos(\beta \cdot L))}{(\sinh(\beta \cdot L)^2 - \sin(\beta \cdot L)^2)} \quad G_{11} = 0$$

$$G_{12} := -\frac{2 \cdot \sinh(\beta \cdot L) \cdot \sin(\beta \cdot L)}{(\sinh(\beta \cdot L)^2 - \sin(\beta \cdot L)^2)} \quad G_{12} = 0$$

$$G_{22} := \frac{-2 \cdot (\cosh(\beta \cdot L) \cdot \sin(\beta \cdot L) + \sinh(\beta \cdot L) \cdot \cos(\beta \cdot L))}{(\sinh(\beta \cdot L)^2 - \sin(\beta \cdot L)^2)} \quad G_{22} = 0$$

Use ASME BPVC, Section III, Appendices, Article A-5000, Subarticle A-5523. The top and bottom plates are considered the same.

$$\text{plate factor: } F_1 := \frac{3 \cdot (1 - \nu) \cdot (2 - f^2) \cdot (1 - f)^2 \cdot (8 - f(4 - f) \cdot (1 - \nu))}{16 \cdot (2 - f)} \quad F_1 = 0.9$$

$$\text{rotational displacement: } \theta_p := \frac{F_1}{E_p \cdot \left(\frac{t_p}{R_o}\right)^3} \cdot p \quad \theta_p = 2.617 \cdot 10^{-4}$$



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radial displacement: $w_p := -\left(\frac{t_p}{2}\right) \cdot \theta_p$ $w_p = -1.963 \cdot 10^{-4} \text{ in}$

Given

radial displacement (top end):

$$\frac{2 \cdot F_3}{3 \cdot E_p \cdot \left(\frac{t_p}{R_o}\right)} \cdot Q_o + \frac{F_3}{E_p \cdot R_o \cdot \left(\frac{t_p}{R_o}\right)^2} \cdot M_o + w_p = \left(\frac{B_{11}}{2 \cdot \beta^3 \cdot D}\right) \cdot Q_o + \left(\frac{B_{12}}{2 \cdot \beta^2 \cdot D}\right) \cdot M_o + \left(\frac{G_{11}}{2 \cdot \beta^3 \cdot D}\right) \cdot Q_L + \left(\frac{G_{12}}{2 \cdot \beta^2 \cdot D}\right) \cdot M_L$$

rotational displacement:

$$\left[\frac{-F_3}{E_p \cdot R_o \cdot \left(\frac{t_p}{R_o}\right)^2} \cdot Q_o + \frac{-2 \cdot F_3}{E_p \cdot R_o^2 \cdot \left(\frac{t_p}{R_o}\right)^3} \cdot M_o \right] + \theta_p = \left(\frac{B_{12}}{2 \cdot \beta^2 \cdot D}\right) \cdot Q_o + \left(\frac{B_{22}}{\beta \cdot D}\right) \cdot M_o + \left(\frac{G_{12}}{2 \cdot \beta^2 \cdot D}\right) \cdot Q_L + \left(\frac{G_{22}}{2 \cdot \beta \cdot D}\right) \cdot M_L$$

radial displacement (bottom end):

$$\frac{2 \cdot F_3}{3 \cdot E_p \cdot \left(\frac{t_p}{R_o}\right)} \cdot Q_L + \frac{F_3}{E_p \cdot R_o \cdot \left(\frac{t_p}{R_o}\right)^2} \cdot M_L + w_p = \left(\frac{G_{11}}{2 \cdot \beta^3 \cdot D}\right) \cdot Q_o + \left(\frac{G_{12}}{2 \cdot \beta^2 \cdot D}\right) \cdot M_o + \left(\frac{B_{11}}{2 \cdot \beta^3 \cdot D}\right) \cdot Q_L + \left(\frac{B_{12}}{2 \cdot \beta^2 \cdot D}\right) \cdot M_L$$



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rotational displacement:

$$\frac{-F_3}{E_p \cdot R_o \cdot \left(\frac{t_p}{R_o}\right)^2} Q_L + \frac{-2 \cdot F_3}{E_p \cdot R_o^2 \cdot \left(\frac{t_p}{R_o}\right)^3} M_L + \theta_p = \left(\frac{G_{12}}{2 \cdot \beta^2 \cdot D}\right) Q_o + \left(\frac{G_{22}}{2 \cdot \beta \cdot D}\right) M_o + \left(\frac{B_{12}}{2 \cdot \beta^2 \cdot D}\right) Q_L + \left(\frac{B_{22}}{\beta \cdot D}\right) M_L$$

$$\begin{bmatrix} Q_o \\ Q_L \\ M_o \\ M_L \end{bmatrix} := \text{find}(Q_o, Q_L, M_o, M_L)$$

$$Q_o = -133.68 \cdot \frac{\text{lbf}}{\text{in}}$$

$$Q_L = -129.4 \cdot \frac{\text{lbf}}{\text{in}}$$

$$M_o = 119.4 \cdot \text{lbf}$$

$$M_L = 114.04 \cdot \text{lbf}$$



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Use ASME BPVC, Section III, Appendices, Article A-2000, Subarticle A-2260 to determine the principal stresses on the inside and outside of the shell at the junction to the closure flanges.

tangential stress:

$$\text{inside surface of cylinder: } \sigma_{t2} := \frac{E_p \cdot w_p}{\left(R + \frac{t}{2}\right)} + \frac{6 \cdot v \cdot M_o}{t^2} \quad \sigma_{t2} = 0.4 \text{ ksi}$$

$$\text{outside surface of cylinder: } \sigma_{t2a} := \frac{E_p \cdot w_p}{\left(R + \frac{t}{2}\right)} - \frac{6 \cdot v \cdot M_o}{t^2} \quad \sigma_{t2a} = -2.1 \text{ ksi}$$

longitudinal stress:

$$\text{inside surface of cylinder: } \sigma_{l2} := \frac{6 \cdot M_o}{t^2} \quad \sigma_{l2} = 4.3 \text{ ksi}$$

$$\text{outside surface of cylinder: } \sigma_{l2a} := -\left(\frac{6 \cdot M_o}{t^2}\right) \quad \sigma_{l2a} = -4.3 \text{ ksi}$$

$$\text{radial stress: } \sigma_{r2} := 0 \text{ ksi}$$



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Primary + secondary principal stresses:

tangential: inside: $\sigma_{ps1} := \sigma_1 + \sigma_{t2}$ $\sigma_{ps1} = 1.9 \cdot \text{ksi}$

 outside: $\sigma_{ps1a} := \sigma_1 + \sigma_{t2a}$ $\sigma_{ps1a} = -0.7 \cdot \text{ksi}$

longitudinal: inside: $\sigma_{ps2} := \sigma_2 + \sigma_{l2}$ $\sigma_{ps2} = 5.1 \cdot \text{ksi}$

 outside: $\sigma_{ps2a} := \sigma_2 + \sigma_{l2a}$ $\sigma_{ps2a} = -3.6 \cdot \text{ksi}$

radial: $\sigma_{ps3} := \sigma_3 + \sigma_{r2}$ $\sigma_{ps3} = 0 \cdot \text{ksi}$

Determine strain using Hooke's Law (find principal strains of the joint):

$\epsilon_1 := \frac{1}{E} [\sigma_{ps1} - \nu (\sigma_{ps2} + \sigma_{ps3})]$ $\epsilon_1 = 0.000014$

$\epsilon_2 := \frac{1}{E} [\sigma_{ps2} - \nu (\sigma_{ps1} + \sigma_{ps3})]$ $\epsilon_2 = 0.000164$

$\epsilon_3 := \frac{1}{E} [\sigma_{ps3} - \nu (\sigma_{ps1} - \sigma_{ps2})]$ $\epsilon_3 = 0.000032$

effective strain at the joint:

$\epsilon_{\text{eff}} := \frac{\sqrt{2}}{3} \sqrt{(\epsilon_1 - \epsilon_2)^2 + (\epsilon_2 - \epsilon_3)^2 + (\epsilon_3 - \epsilon_1)^2}$ $\epsilon_{\text{eff}} = 9.46 \cdot 10^{-5}$



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Primary + secondary stress intensities:

inside: $PS_{1i} := |\sigma_{ps1} - \sigma_{ps2}|$ $PS_{1i} = 3.21 \text{ ksi}$

$PS_{2i} := |\sigma_{ps2} - \sigma_{ps3}|$ $PS_{2i} = 5.1 \text{ ksi}$

$PS_{3i} := |\sigma_{ps1} - \sigma_{ps3}|$ $PS_{3i} = 1.9 \text{ ksi}$

outside: $PS_{1o} := |\sigma_{ps1a} - \sigma_{ps2a}|$ $PS_{1o} = 2.97 \text{ ksi}$

$PS_{2o} := |\sigma_{ps2a} - \sigma_{ps3}|$ $PS_{2o} = 3.6 \text{ ksi}$

$PS_{3o} := |\sigma_{ps1a} - \sigma_{ps3}|$ $PS_{3o} = 0.6 \text{ ksi}$

The largest stress on the inside is PS_{2i} and the largest stress on the outside is PS_{2o} .

The S_m value for SA-240, 304L plate at 200 °F: $S_m := 16.5 \text{ ksi}$

Margin of Safety for primary + secondary stresses (for shell):

inside: $MS_2 := \frac{3 \cdot S_m}{PS_{2i}} - 1$ $MS_2 = 8.7$

outside: $MS_3 := \frac{3 \cdot S_m}{PS_{2o}} - 1$ $MS_3 = 12.8$



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Use ASME BPVC, Section III, Appendices, Article A-5000, Subarticle A-5224.3 to determine stresses at the center of the closure plate.

$$F_4 := \frac{3}{8} \cdot (1 - \nu) \cdot (2 - f) \cdot (8 - f(4 - f) \cdot (1 - \nu)) \quad F_4 = 4.03$$

stresses due to radial load and moment (and tangential):

$$\sigma_{rt} := \frac{F_4}{t_p} \cdot \left(1 - \frac{6 \cdot x}{t}\right) \cdot Q_0 - \frac{12 \cdot F_4 \cdot x}{t_p^3} \cdot M_0 \quad \sigma_{rt} = 2.3 \text{ ksi}$$

ASME BPVC, Section III, Appendices, Article A-5000, Subarticle A-5224.2.

$$\sigma_{ln} := \left(\frac{p}{t_p}\right) \cdot \left(x - \frac{t_p}{2}\right) \quad \sigma_{ln} = 0 \text{ ksi}$$

Find stresses:

$$\text{tangential:} \quad SI_1 := |\sigma_{rt} + \sigma_t| \quad SI_1 = 4.6 \text{ ksi}$$

$$\text{radial:} \quad SI_2 := |\sigma_{rt} + \sigma_r| \quad SI_2 = 4.6 \text{ ksi}$$

$$\text{longitudinal:} \quad SI_3 := |\sigma_{ln} + \sigma_{ls}| \quad SI_3 = 0 \text{ ksi}$$

The largest stress intensity is SI_1 .

Margin of safety:

$$MS_4 := \frac{1.5 \cdot S_m}{SI_1} - 1 \quad MS_4 = 4.4$$



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Use NUREG/CR-6007, Stress Analysis of Closure Bolts for Shipping Casks to find the torque and pressure loading. Use Tables 4.1 and 4.3.

nominal diameter of closure bolt: $D_b := \frac{1}{2} \cdot \text{in}$ nut factor: $K := 0.2$

Bickford recommendation for initial preload torque to achieve 25 kis (in-lb) - size 1/2-13 UNC: $Q := 354.8 \cdot \text{in} \cdot \text{lbf}$

closure lid diameter at location of gasket load reaction: $D_{lg} := 8.25 \cdot \text{in}$

closure lid diameter at bolt circle: $D_{lb} := 10 \cdot \text{in}$

Poisson's ratio of closure lid material: $\text{NUI} := \nu$

pressure difference between the inside and outside of the containment boundary: $\Delta P := \text{MNOP}$

total number of closure bolts: $N_b := 12$

Young's modulus of closure lid material (at 200°F): $E_l := 27.6 \cdot 10^6 \cdot \text{psi}$ $E_{lf} := E_l$

Young's modulus of cask wall material (at 200°F): $E_c := 27.6 \cdot 10^6 \cdot \text{psi}$

thickness of closure lid: $t_l := 1.5 \cdot \text{in}$

thickness of cask wall: $t_c := \text{OR}_g - 4 \cdot \text{in}$ $t_c = 2.4 \cdot \text{in}$

Table 4.1. Applied preload using a torque wrench.

Non-prying tensile force per bolt:

Preload force per bolt: $F_a := \frac{Q}{K \cdot D_b}$ $F_a = 3548 \cdot \text{lbf}$

Torsional bolt moment per bolt: $M_t := 0.5 \cdot Q$ $M_t = 177.4 \cdot \text{in} \cdot \text{lbf}$

Table 4.3. Bolt Forces/Moments Generated by Pressure Loads.

Non-prying tensile force per bolt: $F_{a_p} := \frac{\pi \cdot D_{lg}^2 \cdot \Delta P}{4 \cdot N_b}$ $F_{a_p} = 445.5 \cdot \text{lbf}$



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Shear bolt force per bolt:

$$F_s := \frac{\pi \cdot E_l \cdot t_l \cdot \Delta P \cdot D_{lb}^2}{2 \cdot N_b \cdot E_c \cdot t_c \cdot (1 - \text{NUI})} \quad F_s = 1164.4 \cdot \text{lbf}$$

Fixed-edge force:

$$F_f := \frac{D_{lb} \cdot \Delta P}{4} \quad F_f = 250 \frac{\text{lbf}}{\text{in}}$$

Moment:

$$M_f := \frac{\Delta P \cdot D_{lb}^2}{32} \quad M_f = 312.5 \cdot \text{lbf}$$

Table 2.1. Maximum Prying Force Generated by Applied Loads.

$$C_1 := 1$$

closure lid diameter at the outer edge: $D_{lo} := 12.75 \cdot \text{in}$

closure lid diameter at inner edge: $D_{li} := 11.938 \cdot \text{in}$

thickness of the closure flange: $t_{lf} := 1.5 \cdot \text{in}$

bolt length between top and bottom surfaces of closure lid at bolt circle: $L_b := \frac{5}{8} \cdot \text{in}$

nominal diameter of closure bolt: $D_b := D_b$

Young's modulus of closure bolt material at 200 °F: $E_b := 27.6 \cdot 10^6 \cdot \text{psi}$



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$$C2 := \left[\frac{8}{3 \cdot (Dlo - Dlb)^2} \right] \cdot \left[\frac{E \cdot t^3}{1 - \text{NUI}} + \frac{(Dlo - Dlb) \cdot E \cdot t^3}{Dlb} \right] \cdot \left(\frac{Lb}{Nb \cdot Db^2 \cdot Eb} \right) \quad C2 = 0.4$$

Bolt preload per unit length of bolt circle: $P := Fa \cdot \left(\frac{Nb}{\pi \cdot Dlb} \right) \quad P = 1355.2 \frac{\text{lbf}}{\text{in}}$

Ff is less than P; therefore B=P B := P

$$Fap := \left(\frac{\pi \cdot Dlb}{Nb} \right) \cdot \left[\frac{2 \cdot \text{Mf}}{(Dlo - Dlb)} - \frac{C1 \cdot (B - Ff) - C2 \cdot (B - P)}{C1 + C2} \right]$$

$Fap = -1518.1 \cdot \text{lbf}$ therefore: $Fap_a := 0 \cdot \text{lbf}$

Table 2.2. Maximum Bending Bolt Moment Generated by Applied Loads.

$$Kb := \left(\frac{Nb}{Lb} \right) \cdot \left(\frac{Eb}{Dlb} \right) \cdot \left(\frac{Db^4}{64} \right) \quad Kb = 51750.1 \cdot \text{lbf}$$

$$Kl := \frac{E \cdot t^3}{3 \cdot \left[\left(1 - \text{NUI}^2 \right) + \left(1 - \text{NUI} \right)^2 \cdot \left(\frac{Dlb}{Dlo} \right)^2 \right] \cdot Dlb} \quad Kl = 2532638.3 \cdot \text{lbf}$$

bending bolt moment per bolt:

$$\text{Mbb} := \left(\frac{\pi \cdot Dlb}{Nb} \right) \cdot \left(\frac{Kb}{Kb + Kl} \right) \cdot \text{Mf} \quad \text{Mbb} = 16.4 \cdot \text{in} \cdot \text{lbf}$$

Table 4.9. Combining Bolt Forces from Loads.

$$Fa = 3548 \cdot \text{lbf}$$

Sum of Non-Prying Tensile Bolt Forces:

$$Fa_p = 445.5 \cdot \text{lbf} \quad \text{Therefore use } Fa.$$



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Sum of Prying Tensile Forces:

Fixed-edge force:

$$F_f = 250 \cdot \frac{\text{lb} \cdot \text{f}}{\text{in}}$$

Moment:

$$M_f = 312.5 \cdot \text{lb} \cdot \text{f}$$

Total Tensile Bolt Force:

$$TT := F_a + F_{ap_a}$$

$$TT = 3548 \cdot \text{lb} \cdot \text{f}$$

Shear Bolt Force:

$$F_s = 1164.4 \cdot \text{lb} \cdot \text{f}$$

Table 5.1. Stress Evaluation.

bolt thread pitch: $n := 13$

$$p_1 := \frac{1}{n} \cdot \text{in}$$

$$p_1 = 0.08 \cdot \text{in}$$

bolt diameter: $Db_a := Db - 0.9743 \cdot p_1$

$$Db_a = 0.425 \cdot \text{in}$$

average tensile stress caused by the tensile bolt force:

$$S_{ba} := \frac{1.2732 \cdot TT}{Db_a^2}$$

$$S_{ba} = 25 \cdot \text{ksi}$$

average shear stress caused by shear bolt force:

$$S_{bs} := \frac{1.2732 \cdot F_s}{Db_a^2}$$

$$S_{bs} = 8.2 \cdot \text{ksi}$$



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maximum bending stress caused by bending bolt moment:

$$S_{bb} := \frac{10.186 \cdot M_{bb}}{D_{ba}^3} \quad S_{bb} = 2.2 \text{ ksi}$$

maximum shear stress caused by torsional bolt moment:

$$S_{bt} := \frac{5.093 \cdot M_t}{D_{ba}^3} \quad S_{bt} = 11.8 \text{ ksi}$$

maximum stress intensity caused by tension + shear + bending + torsion:

$$S_{bi} := \left[(S_{ba} + S_{bb})^2 + 4 \cdot (S_{bs} + S_{bt})^2 \right]^{0.5} \quad S_{bi} = 48.3 \text{ ksi}$$

Table 6.3. Stress Analysis of Closure Bolts.

minimum yield strength of bolt material (200 °F): $S_y := 86 \text{ ksi}$
 (SA-193, B8)

tension: $MS_{b1} := \frac{\frac{2}{3} \cdot S_y}{S_{ba}} - 1 \quad MS_{b1} = 1.3$

shear: $MS_{b2} := \frac{0.6 \cdot \frac{2}{3} \cdot S_y}{S_{bs}} - 1 \quad MS_{b2} = 3.2$

tension + shear: $MS_{b3} := \left(\frac{S_{ba}}{\frac{2}{3} \cdot S_y} \right)^2 + \left(\frac{S_{bs}}{0.6 \cdot \frac{2}{3} \cdot S_y} \right)^2 \quad MS_{b3} = 0.2 < 1$



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$$\text{maximum stress intensity: } MS_{sbi} := \frac{1.5 \cdot \frac{2}{3} \cdot S_y}{S_{bi}} - 1 \quad MS_{sbi} = 0.8$$

Calculate strain energy density at joint discontinuity:

Von Mises stress:

$$\sigma_o := \sqrt{\sigma_{ps1}^2 + \sigma_{ps2}^2 + \sigma_{ps3}^2 - \sigma_{ps1} \cdot \sigma_{ps2} - \sigma_{ps2} \cdot \sigma_{ps3} - \sigma_{ps3} \cdot \sigma_{ps1}}$$

$$\sigma_o = 4.5 \cdot \text{ksi}$$

$$\text{SED at the joint: } SED_{joint} := \sigma_o \cdot \epsilon_{eff} \quad SED_{joint} = 0.4 \cdot \frac{\text{lbfin}}{\text{in}^3}$$

Calculate strain energy density at shell wall:

Von Mises stress:

$$\sigma_{ow} := \sqrt{\sigma_1^2 + \sigma_2^2 + \sigma_3^2 - \sigma_1 \cdot \sigma_2 - \sigma_2 \cdot \sigma_3 - \sigma_3 \cdot \sigma_1}$$

$$\sigma_{ow} = 1.3 \cdot \text{ksi}$$

$$\text{strain: } \epsilon_1 := \frac{p \cdot R}{E \cdot t} \cdot \left(1 - \frac{\nu}{2}\right) \quad \epsilon_1 = 0.000046$$

$$\epsilon_2 := \frac{p \cdot R}{E \cdot t} \cdot (0.5 - \nu) \quad \epsilon_2 = 0.000011$$

$$\epsilon_{eff2} := \frac{\epsilon_1 + \epsilon_2}{2} + \frac{1}{2} \sqrt{(\epsilon_1 - \epsilon_2)^2} \quad \epsilon_{eff2} = 4.55 \cdot 10^{-5}$$

$$\text{SED at shell wall: } SED_{shell} := \sigma_o \cdot \epsilon_{eff2} \quad SED_{shell} = 0.2 \cdot \frac{\text{lbfin}}{\text{in}^3}$$

8.3 WEIGHT CALCULATIONS



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1.0 OBJECTIVE

The objective of this calculation is to determine the weight for a generic shielded drum packaging containment vessel for the Hanford Site.

2.0 REFERENCE

ASME, 1992, *Boiler and Pressure Vessel Code*, Section II, Part D, American Society of Mechanical Engineers, New York, New York.

3.0 ASSUMPTIONS, RESULTS, AND CONCLUSIONS

In this analysis, the following assumptions were made:

- The density of stainless steel is assumed to be 0.29 lbf/in³.
- The weight of the payload is assumed to be 150 lb.
- The shell of the containment vessel is constructed of a 12 in. Schedule 40 pipe of SA-312, TP304L.
- The bottom-end and top-end closures, and the reverse hub flange are constructed of SA-240, 304L.

The following table summarizes the weights for the package:

Component	Weight
Shell	91.3 lbf
Bottom-end closure	55.5 lbf
Closure lid	44.3 lbf
Reverse hub flange	33.7 lbf
Total weight of package (without payload)	224.8 lbf
Total weight of package (with payload)	374.8 lbf



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4.0 EVALUATION

MINIMUM PACKAGING STANDARDS - weight calculation

payload weight: $w_{\text{pay}} := 150 \cdot \text{lb}$

density of stainless steel: $\text{Den}_{\text{ss}} := 0.29 \frac{\text{lb}}{\text{in}^3}$

1. **Shell.** 12 in. Schedule 40. Assume it is constructed of SA-312, TP304L. Assume to be a cylinder.

outside diameter: $\text{OD}_s := 12.75 \cdot \text{in}$ inside diameter: $\text{ID}_s := 11.938 \cdot \text{in}$

wall thickness: $\text{WT}_s := 0.406 \cdot \text{in}$ length: $L_s := 20 \cdot \text{in}$

Volume of shell: $V_s := \frac{\pi}{4} \cdot L_s \cdot (\text{OD}_s^2 - \text{ID}_s^2)$ $V_s = 314.9 \cdot \text{in}^3$

Weight: $W_s := V_s \cdot \text{Den}_{\text{ss}}$ $W_s = 91.3 \cdot \text{lb}$

2. **Bottom-End Closure.** Assume it is constructed of SA-240, 304L. It is a solid, stainless steel plate, 1.5 in. thick. Assume perfect disk.

diameter: $d_{\text{bec}} := 12.75 \cdot \text{in}$ height: $h_{\text{bec}} := 1.5 \cdot \text{in}$

volume: $V_{\text{bec}} := \frac{\pi}{4} \cdot h_{\text{bec}} \cdot d_{\text{bec}}^2$ $V_{\text{bec}} = 191.5 \cdot \text{in}^3$

weight: $W_{\text{bec}} := V_{\text{bec}} \cdot \text{Den}_{\text{ss}}$ $W_{\text{bec}} = 55.5 \cdot \text{lb}$



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3. Closure Lid. Assume it is constructed of SA-240, 304L plate.

Top section of lid:

diameter: $d_{cf} := 12.75 \text{ in}$ height: $h_{cf} := 1.0 \text{ in}$

Bottom section of lid:

diameter: $d_{bcf} := 8 \text{ in}$ height: $h_{bcf} := 0.5 \text{ in}$

$$\text{volume: } V_{cf} := \left(\frac{\pi}{4} \cdot h_{cf} \cdot d_{cf}^2 \right) + \left(\frac{\pi}{4} \cdot h_{bcf} \cdot d_{bcf}^2 \right) \quad V_{cf} = 152.8 \text{ in}^3$$

$$\text{weight: } W_{cf} := V_{cf} \cdot \text{Den}_{ss} \quad W_{cf} = 44.3 \text{ lbf}$$

4. Reverse Hub Flange. Assume it is constructed of SA-240, 304L plate.

outside diameter: $OD_{rhf} := 12.75 \text{ in}$ inside diameter: $ID_{rhf} := 8 \text{ in}$

wall thickness: $WT_{rhf} := 1.5 \text{ in}$

$$\text{volume: } V_{rhf} := \frac{\pi}{4} \cdot WT_{rhf} \cdot (OD_{rhf}^2 - ID_{rhf}^2) \quad V_{rhf} = 116.1 \text{ in}^3$$

$$\text{weight: } W_{rhf} := V_{rhf} \cdot \text{Den}_{ss} \quad W_{rhf} = 33.7 \text{ lbf}$$



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TOTAL EMPTY WEIGHT :

$$TW_{mp} := W_s + W_{bec} + W_{cf} + W_{rhf}$$

$$TW_{mp} = 224.8 \text{ •lb}$$

TOTAL WEIGHT WITH PAYLOAD :

$$TW_{wp} := TW_{mp} + w_{pay}$$

$$TW_{wp} = 374.8 \text{ •lb}$$

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APPENDIX G
JUSTIFICATION AND BASIS FOR SHIPMENT OF RISK-BASED PACKAGES

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LIST OF TERMS

AC	accident conditions
ARF	airborne release fraction
CFR	<i>Code of Federal Regulations</i>
DE-Ci	dose-equivalent curie
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
HMR	Hazardous Materials Regulations
NC	normal conditions
NRC	U.S. Nuclear Regulatory Commission
PSSD	package specific safety documents
RF	respirable fraction
RL	Richland Operations Office
SAR	safety analysis report
TRU	transuranic
TSD	<i>Hanford Sitewide Transportation Safety Document</i>
USQ _T	unreviewed safety question process for transportation
V/R	velocity/radius

APPENDIX G

JUSTIFICATION AND BASIS FOR SHIPMENT OF RISK-BASED PACKAGES

1.0 INTRODUCTION

Section 1.1 summarizes the impacts of the unreviewed safety question process for transportation (USQ_T) on the approval process of risk-based packages. Section 1.2 summarizes the U.S. Department of Transportation (DOT) Hazardous Materials Regulations (HMR) and the argument of equivalent safety for risk-based packages. Section 1.3 describes the methodology and data set that may be used to calculate the accident release frequency from transportation accident scenarios.

1.1 HIGHLIGHTS OF THE USQ_T FOR RISK-BASED PACKAGES

The shipment of risk-based packages is subject to approval requirements. Appendix A, Tables 1 and 2, Part Three, list package specific safety documents (PSSD) for risk-based packages authorized for onsite transportation. Each PSSD has specific requirements, limits, and controls governing the shipment of that particular risk-based package. The shipment of one of these risk-based packages within the bounds of the PSSD is approved by the U.S. Department of Energy (DOE), Richland Operations Office (RL). The shipment of one of these packages outside of those bounds, whether a modification to the design of the package, or changes in the requirements, limits, or controls of the package, is subject to the USQ_T prior to shipment of the package to determine whether the change is within the safety basis envelope of the *Hanford Sitewide Transportation Safety Document* (TSD). If the change is within the safety basis, then the USQ_T is negative, and the shipment of the package with the changes considered in the USQ_T is approved. If the change is outside the safety basis, then the USQ_T is positive, and RL approval is required prior to the shipment of the package. Details of the USQ_T are given in Chapter VI and Appendix C.

To assist in RL's review of the PSSD for packages not listed in Appendix A, Table 1, Part Three, a methodology is provided in Chapter VII of the TSD for determining whether the package provides an equivalent degree of safety to the DOT HMR. If the package exceeds the release limits in the PSSD calculated with this methodology, then equivalency is not shown, and it is left to a case-by-case basis to determine if the dose consequences and frequency of a release are acceptable, or if repackaging into a more robust package is required. The values in Table 1 are derived using conservative assumptions to provide an example of the release limits that are calculated with this methodology. As discussed in the following sections, the release limits in each PSSD depend on the area of shipment, dispersability of the payload, damage to the package, potential leak paths, and 99.5% worst-case meteorology.

Table 1. Example Limits Based on Equivalent Public Intake.

Release Location	Release Rate (A_2 /hr) Normal Transfer Conditions	Release (A_2) Accident Conditions
300 Area	3.51×10^{-4}	4.71×10^{-2}
200 Area	2.46×10^{-2}	1.71×10^{-3}
100 K Area	9.65×10^{-3}	6.70×10^{-4}
In Commerce	1.00×10^{-6}	1.00×10^0

Note that the calculated shipment-specific release limits such as the examples in Table 1 are not considered to result in an acceptable public exposure and are not to be treated as design acceptance criteria. Rather, they are meant to result in an intake that is equivalent to that in the DOT HMR. Use of these limits shall not prevent the transportation operation from including design features to attenuate the release of material during accident conditions (AC). The packaging is only the first line of defense in the transportation operation, and all other available administrative and engineering controls that reasonably reduce the frequency of an accidental release and/or the intake by the maximally exposed member of the public should be implemented as practical, consistent with the as low as reasonably achievable principle. The combination of the calculated shipment-specific release limits and the administrative and engineering controls designed to preclude and mitigate a release provides a degree of safety equivalent to the DOT HMR for onsite shipments of radioactive material packages.

1.2 APPROACH FOR SHOWING EQUIVALENCY TO DOT HMR

DOE Order 460.1A, *Packaging and Transportation Safety*, requires that onsite shipments of radioactive material packages comply with the DOT HMR (49 *Code of Federal Regulations* [CFR] 171-180) or be shown to meet a degree of safety equivalent to the regulations. The purpose of the HMR is to protect the public, which includes those who work in transportation, from harmful exposure to radiation. To protect the public, the DOT HMR requires that the packaging provide adequate shielding, maintain subcriticality, and contain its contents within specified limits during normal conditions (NC) and ACs. The criticality and shielding requirements for risk-based packaging are not substantially different than the requirements for DOT and equivalent packaging given in Chapter VI of the TSD. However, risk-based packaging, by definition, does not maintain the same level of containment as DOT or equivalent packaging. Therefore, equivalent packaging containment performance requirements are derived that consider the specific characteristics and controls of onsite shipments of risk-based packages that limit the intake of radionuclides into the body to the same amount of intake from accidental releases in commerce. One of these controls is that the public is not in the immediate vicinity of a damaged package, as is assumed in the derivation of the DOT containment requirements, because of restricted public access during onsite shipments and the large distances to the Hanford Site boundary. Other assumptions behind the DOT containment requirements that can be revised for onsite shipments include the dispersability of the payload, damage to the package, and potential leak paths. The effect of the characteristics and controls of onsite shipments of risk-based packages is that package-specific release limits may be calculated such that the intake

of radioactive material by the maximally exposed member of the public is no greater than the intake by the public from the DOT allowable release limits in commerce.

49 CFR 173.467, "Tests for demonstrating the ability of Type B and fissile materials packaging to withstand ACs in transportation," requires that each certified Type B packaging or packaging for fissile material meet the test requirements prescribed in 10 CFR 71, "Packaging and Transportation of Radioactive Material," for ability to withstand ACs in transportation. Additionally, Type B packages that meet the applicable requirements of 10 CFR 71 and approved by the U.S. Nuclear Regulatory Commission (NRC) are authorized for shipment per 49 CFR 173.416, "Authorized Type B packages." Therefore, to show an equivalent degree of safety, NRC regulations in 10 CFR 71 must be considered in addition to the DOT HMR.

For shipments in commerce, 10 CFR 71.51, "Additional requirements for Type B packages," requires that certified Type B packages be designed, constructed, and prepared for shipment so that under the tests specified in 10 CFR 71.73, "Hypothetical accident conditions," there would be no escape of radioactive material exceeding a total amount A_2 in one week. This leakage is taken to mean that no more than 1 A_2 per accident may be released, rather than a continuous uniform release rate during the week the accident occurs. The intake of the release by the public, rather than the release itself, causes the exposure and must be controlled in order to show an equivalent degree of safety. For an accident on a public highway, the public could be in the immediate vicinity of the package. In the derivation of the allowable release during ACs, the release of 1 A_2 is based on the release of 10^{-3} to 10^{-2} as a respirable aerosol combined with an uptake into the body of 10^{-4} to 10^{-3} of the respirable aerosol (IAEA 1990, *Explanatory Material for the IAEA Regulations for the Safe Transport of Radioactive Material*, p. 78), for a total intake of approximately 10^{-6} A_2 . The release of 1 A_2 is defined such that the dose to a person in the vicinity of a transport package following an accident does not exceed the annual dose limit for radiation workers recommended by the International Commission on Radiological Protection, namely 5 rem (IAEA 1990, p. 72 and p. 110).

10 CFR 71.51 also requires that certified Type B packages be designed, constructed, and prepared for shipment so that under the tests specified in 10 CFR 71.71, "Normal conditions of transport," there would be no loss or dispersal of radioactive contents as demonstrated to a sensitivity of 10^{-6} A_2 /h. It is possible that some risk-based packages fail only the DOT containment requirement for ACs (i.e., they may meet the DOT containment requirements for NCs). For risk-based packages that do not meet the containment requirements for NCs, the intake associated with a leaking package during NCs at the location of the maximally exposed member of the public shall not exceed the intake of a member of the public from a certified DOT package in the immediate vicinity of a package in commerce. In the derivation of the allowable release rate from Type B packages during NCs (IAEA 1990, p. 107-108), the release of 10^{-6} A_2 /h is based on a member of the public spending 20% of his working time ($400 \text{ h/yr} = 20\% \times 2,000 \text{ h/yr}$) working around a leaking certified Type B package in an enclosed vehicle of 50 m^3 ($1,766 \text{ ft}^3$) volume with 10 air changes per hour. Note that in commerce, the public includes those who work in transportation. This scenario results in a maximum public intake over the course of 1 year of 10^{-6} A_2 from NCs.

Within the boundaries of the Hanford Site, DOE has established radiation protection standards, limits, and program requirements for protecting individuals from ionizing radiation resulting from DOE activities. 10 CFR 835.1302, "Emergency exposure situations," regulates emergency exposure situations. These regulations state the following:

- The risk of injury to those individuals involved in rescue and recovery operations shall be minimized.
- Operating management shall weigh actual and potential risks against the benefits to be gained.
- No individual shall be required to perform a rescue action that might involve substantial personal risk.
- Each individual authorized to perform emergency actions likely to result in occupational doses exceeding the values of the limits provided in 10 CFR 835.202(a), "Occupational dose limits for general employees," shall be trained in accordance with 10 CFR 835.901(b), "Radiation safety training," and briefed beforehand on the known or anticipated hazards to which the individual will be subjected.

Therefore, minimizing worker risk (consisting of dose consequence and frequency) is a requirement of 10 CFR 835, "Occupational Radiation Protection." Administrative and engineered controls can be used to reduce the frequency of a release, the severity of the forces on the package, and/or the consequences of a release.

A degree of safety equivalent to the DOT HMR can be shown by limiting the public intake of radioactive material from a risk-based package to 10^{-6} A₂ from both NCs and ACs, which is the same intake on which the DOT containment limits are based. Furthermore, worker safety is achieved by minimizing the radiological risk (dose consequence and frequency) from a transportation accident.

1.3 ONSITE TRANSPORTATION ACCIDENTS

1.3.1 Overall Framework

The overall framework for estimating the frequencies of accidents was tailored to consider Hanford Site-specific road conditions where possible. The Hanford Site is a relatively benign area with respect to potential accident scenarios and subsequent damage to packaging systems. The Hanford Site is characterized by moderate terrain; there are no long, steep grades; few rocky hills and valleys; no rail traffic; predominantly soft (sand) wayside structures; and no bridges, tunnels, freeway overpasses or underpasses. Consequently, the likelihood of encountering road conditions that can lead to severe accidents is less than that on a typical interstate or state highway in the United States. Consequently, it can be concluded that the frequencies of severe accidents on the Hanford Site are lower than on interstate and state highways. Furthermore,

because the Hanford Site is a controlled access area, officials can implement controls to reduce the frequencies of accidents even further, e.g., by closing the road when a shipment occurs.

The other key attribute of the accident frequency framework is that it must be sensitive to the calculated failure thresholds of the packaging systems involved. In other words, if a certain package is determined to fail when subjected to a 48 km/h (30 mph) impact event, the methodology must be flexible enough to calculate the frequency of accidents on the Hanford Site in which the impact velocity is 48 km/h (30 mi/h) or greater. Similarly, the methodology must be capable of calculating the frequencies of accidents that involve fires of varying duration to reflect the thermal resistance of specific packaging systems.

1.3.2 Hazard Analysis

A hazard analysis was conducted to identify and evaluate the hazards associated with transporting radioactive material on the Hanford Site. The purpose of the hazard analysis was to review potential sources of hazards during transport and construct potential accident scenarios that could be applied to the shipments and packagings and lead to loss or dispersal of radioactive materials. A generic hazard "checklist" was used to guide the analysis and ensure a comprehensive review of the potential hazards. A columnar format was used to review the hazards and record their presence or absence with respect to Hanford Site transportation. If a hazard was determined to be present, one or more accident scenarios were constructed that describe the potential effects of the hazard on a packaging system and the subsequent effects on the radioactive materials in the package. In addition, the engineering and administrative barriers that prevent or mitigate the potential accident scenarios are recorded. Information on the engineering and administrative barriers was recorded to support a determination of whether the potential accident scenarios were credible events.

The hazard analysis developed to support the Hanford Site accident frequency analysis framework is presented in Table 2. The following conclusions were derived from the information in Table 2:

- DOE considers the occurrence of an inadvertent criticality event as unacceptable, regardless of the projected consequences of such an event. For this reason, DOE emphasizes prevention of inadvertent criticality and requires packages to be designed such that inadvertent criticality accidents will not occur even under severe conditions that promote a nuclear reaction (e.g., optimum moderation). The sequence of events leading to a potential criticality accident was judged to be incredible due to the engineering and administrative controls in place to prevent criticality accidents. These include, where appropriate, limiting the fissile inventory, preventing ingress of moderator materials (such as water) into the package, geometry controls, adding poisons, etc. These requirements remain in effect for equivalent as well as nonequivalent packages.

Table 2. Hazards Analysis of Hanford Site Transportation Activities

Hazard	Potential Accident	Available Controls		Remarks
		Physical	Administrative	
Electricity	Package struck by high voltage power line	Packages protect contents from alternating current		Not credible (judgment)
Explosives	Solid or liquid explosive ignites and fails package due to overpressure or missile	Package design provides resistance to pressure and missiles, commensurate with hazard	Solid and liquid explosives packaged in accordance with DOT requirements; segregated from radioactive materials	Not credible (judgment)
	Hydrogen gas generated by radiolysis ignites within package	Package strength Vents in packages	Shipping windows; Liquid content limits; Source limits; Venting procedures	Gas generation analysis required; controls ensure accident is not credible
Cryogenic systems	Liquid nitrogen, liquid argon not present when transporting radioactive materials			Not credible (judgment)
Direct radiation sources	Loss of shielding occurs as a result of a transportation accident	Package design, resistance to loss of shielding; personnel barriers	Exposure distance restrictions; Radiation protection training; Detection instrumentation; Emergency Preparedness Program and Procedures	Shielding is generally an integral component of package and not likely to become degraded as a result of an accident
Radioactive materials	Release of radioactive materials from a transportation accident	Package design and containment requirements commensurate with radioactive hazard	Package certification; Radiation protection training; Emergency Preparedness Program and Procedures	Containment of radioactive materials under NCs and ACs addressed in SAR for Packaging; public access restrictions
	Uncontrolled nuclear reaction (i.e., inadvertent criticality)	Geometry controls; Presence of neutron absorbers, poisons	Inventory limits; Exclusion of water; Emergency Preparedness Program and Procedures	Accident involving water immersion incredible at arid Hanford Site; Firefighters trained not to fight fires that involve fissile materials with water; Criticality controls ensure accident is not credible
Flammable materials	Fire affects integrity of transportation package	Package design, thermal resistance, commensurate with radioactive hazard	Hanford Fire Department Emergency Preparedness Program and Procedures	Most likely sources are gasoline, diesel vehicle fuels; Flammable liquids are shipped in accordance with DOT requirements
Compressed gases	Compressed gas tank or bottle explodes and affects integrity of radioactive material transportation package	Package design commensurate with radioactive hazard	Compressed gases not shipped with radioactive materials	Packages resistant to external pressure; Compressed gases shipped in accordance with DOT requirements
Thermal energy (internal decay heat source)	Package overheats, fails due to inadequate or loss of cooling	Package design	Thermal power limits	Packages normally designed to dissipate heat naturally

Table 2. Hazards Analysis of Hanford Site Transportation Activities

Hazard	Potential Accident	Available Controls		Remarks
		Physical	Administrative	
Thermal energy (local temperature extreme)	Package failure to high ambient temperature	Package design resists hot ambient conditions	None	Hanford requirements exceed DOT and NRC NCs of transport; Degradation limited by short trip duration
Thermal energy (extreme cold)	Package degradation due to cold ambient temperature causes failure in transit	Package design resists cold ambient conditions	None	Extreme cold has little effect on package integrity; Degradation limited by short trip duration
High pressure	Traffic accident leads to package immersion under a head of water	Package design commensurate with hazard	None	Incredible event because river depth not sufficient to threaten package integrity
Kinetic energy	Package involved in traffic accident (impact, rollover) that fails containment	Package design, impact resistance, commensurate with radioactive hazard; Impact limiters; Tiedown systems	Driver training awareness; Road closure if necessary; Emergency Preparedness Program and Procedures; Time-of-day restrictions to avoid traffic congestion	Credible event, impact covered by DOT and NRC regulations; Low probability of striking a "hard" target on Hanford Site
	Package involved in traffic accident that generates puncture conditions that fail containment	Package design, puncture resistance, commensurate with hazard; Impact limiters; Tiedown systems	Driver training awareness; Road closure if necessary; Emergency Preparedness Program and Procedures; Time-of-day restrictions to avoid traffic congestion	Credible event; Puncture covered by DOT and NRC regulations
	Package involved in traffic accident that generates crush condition that fails containment	Package design, resistance to crush, commensurate with hazard; Impact limiters; Tiedown systems	Driver training awareness; Road closure if necessary; Emergency Preparedness Program and Procedures; Time-of-day restrictions to avoid traffic congestion	Applicable to only small (<500 kg [1,102 lb]) packages; Credible event; Crush covered by DOT and NRC regulations
	Material handling accident	Not applicable	Not applicable	Handling accidents while loading and unloading packages are covered in facility SAR
Potential energy	Package falls from bridge or overpass	See "Remarks"	See "Remarks"	No bridges or overpasses at the Hanford Site
	Package dropped during loading/unloading	Package design commensurate with hazard; Lifting equipment design	Control of heavy loads requirements; Operating procedures; Personnel training	Credible event, free drop covered by DOT and NRC regulations; Addressed in facility SAR
	Heavy object dropped on package	Package design, puncture resistance	Personnel training (heavy load controls)	Not credible to lift heavy object over package and drop it; Addressed in facility SAR

Table 2. Hazards Analysis of Hanford Site Transportation Activities

Hazard	Potential Accident	Available Controls		Remarks
		Physical	Administrative	
Toxic, hazardous chemicals	Toxic or hazardous chemical co-released with radioactive materials	Package design resists release of toxic, hazardous chemicals	Emergency preparedness procedures and training; Driver awareness of package contents	Unless co-contaminants, radioactive and toxic/hazardous materials are shipped separately; DOT packaging requirements applicable to toxic/hazardous material shipments apply
Corrosion	Corrosion leads to package degradation and failure in transit	Package design considers internal and external corrosion; Corrosion-resistant materials	Visual inspection; Package maintenance program requirements	Package internal atmosphere inverted when required to prevent cargo degradation
Sabotage	Terrorist or sabotage incident threatens package integrity	Package design commensurate with hazard (shipment of highly-radioactive payloads are in accident-resistant packages); Hanford security personnel and equipment	Public access restrictions	Hanford shipments not highly-attractive to terrorists/saboteurs; Judged to be incredible event
Natural phenomena	Earthquake, tornado, high wind, flood, meteorite, or other natural phenomena causes package failure in transit	Package design commensurate with hazard	Emergency preparedness procedures and training; Driver awareness of weather conditions; Travel restrictions under extreme weather conditions	Incredible event because of low probability of such conditions occurring at the same time a package is in transit; Hanford Meteorological Station
High-intensity magnetic fields; Lasers; High noise levels; Biohazards; Non-ionizing radiation sources; Inert atmosphere	Not applicable or not capable of causing package damage			

AC = accident condition.

DOT = U.S. Department of Transportation.

NC = normal condition.

NRC = U.S. Nuclear Regulatory Commission.

SAR = safety analysis report.

- Thermal energy from radioactive decay of the cargo must be dissipated to prevent package failure regardless of whether the package is equivalent or nonequivalent. Any engineered or administrative controls implemented to ensure adequate heat dissipation (e.g., source limits, cooling systems) must be adhered to.

- Extreme ambient conditions (heat and cold) that could lead to package failure were not included in the Hanford Site accident risk framework because they are already included in the approach for equivalent packages.
- Potential explosions were judged extremely unlikely as long as the requirements to prevent explosions for equivalent packages are adhered to. The primary source of the explosion hazard is radiolytic decomposition of water or plastic within a package that generates hydrogen gas. The hydrogen gas concentration could build within the package to the lower explosive limit and then be ignited by some mechanism (e.g., spark created by a traffic accident). An analysis of hydrogen generation is required for each package, equivalent or nonequivalent, and any engineered or administrative barriers required to prevent a hydrogen explosion must be followed. Alternatively, the package may be shown by analysis or test to be capable of withstanding the most energetic hydrogen explosion capable of being generated within the package. For these reasons, the potential for explosions involving the package are not addressed in the Hanford Site framework.
- Immersion of a package under a head of water and subsequent failure due to the hydrostatic pressure load was judged to be incredible. This is because the shipments would not be transported over or near a body of water of sufficient depth to be an immersion concern or a concern about retrieving the package in a timely manner. Thus, immersion scenarios are not considered further.
- Potential energy hazards (e.g., package falling from a bridge or overpass and heavy objects being dropped on a package) were determined to be incredible. There are no bridges or overpasses on the Hanford Site that a package could fall from and heavy object lifting requirements preclude them from being lifted over a radioactive material package except under controlled and previously approved conditions. Other elements of heavy lifting (e.g., package dropped during loading or unloading) are addressed in facility-specific safety analysis reports (SAR).
- Toxic and hazardous chemicals are, in general, not a threat to Hanford Site packages. The chemicals may be an issue related to industrial hygiene and worker protection, but do not represent a credible threat to the package. Similarly, corrosion was removed as a potential accident scenario because of the long-term nature of the threat and the existing package maintenance program in which inspections are conducted periodically to identify potential corrosion or other form of package degradation.
- Potential acts of sabotage or terrorism were not considered further because of the absence of credible threats. Hanford shipments are not considered attractive to potential saboteurs and terrorists because most shipments are in support of cleanup that is supported by most groups and because of the absence of a potentially large body count sought by terrorist organizations. Furthermore, the Hanford Site is provided with its own security forces that are capable of rapidly responding to and dealing with potential saboteurs and terrorists.
- Natural phenomena threats to packages include earthquakes, high winds, tornadoes, and floods. However, the likelihood of such events are low and the likelihood that such an event

occurs at the same time a package is in transit, combined with the likelihood that the natural phenomena event is severe enough to threaten package integrity makes these incredible occurrences. Furthermore, shipping is restricted during periods of highly hazardous weather, such as dust storms and high winds that decrease visibility and snowstorms that lead to dangerous road conditions. For these reasons, natural phenomena hazards were eliminated from further consideration.

Based on the above, the Hanford Site accident frequency analysis framework must address potential ACs including impact (such as vehicular crashes and rollovers), puncture, crush, and fire. Note that these events are the same as the hypothetical ACs defined in 10 CFR 71. However, package requirements and controls that were developed based on the deterministic analyses conducted to support traditional package certification or Hanford Site performance-based certification must also be met (e.g., criticality controls, hydrogen gas generation limitations).

1.4 EXAMPLE

Assume that a packaging system for Type B quantities of radioactive material has been determined to be nonequivalent via detailed structure and thermal modeling. The loaded weight of the package is 2,000 kg (4,409 lb). The detailed modeling determined the following containment failure thresholds:

- Mechanical failure occurs when the package is subjected to a 40 km/h (25 mi/h) collision with an unyielding target
- Puncture failure occurs when the package is subjected to a 9 m/sec (30 ft/sec) impact onto a 20 cm (8-in.) diameter puncture probe
- Failure occurs after a 25-minute fire following a collision or after 45 minutes when no collision occurs (i.e., fire only).

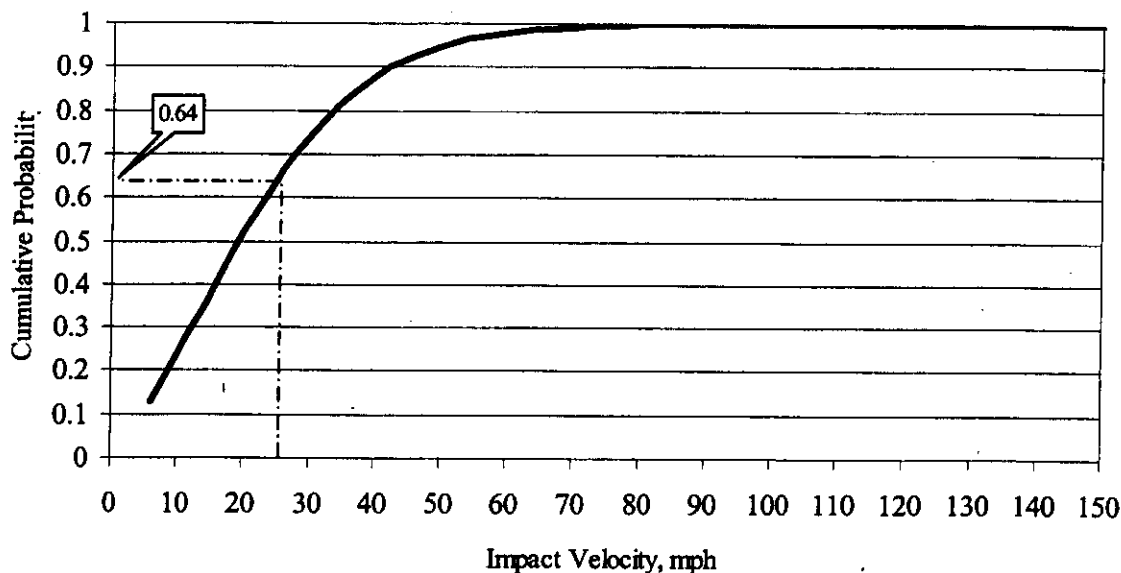
In addition, it is known that the package will be used to transport waste materials from the 300 Area to the Low-Level Waste Burial Ground in the 200 West Area, a distance of approximately 25 km (40 mi), and is planned to be used twice per year.

Solution

1. Determine Impact Failure Frequency:

- a. Determine Conditional Probability. The conditional probability of a collision/overtake is 0.7412. Referring to Figure 1, which was constructed using Figure 7-10, Chapter VII of the TSD, it can be seen that the probability of collision/overtake that produces a 40 km/h (25 mi/h) impact velocity or less is about 0.64. Therefore, the probability of encountering collisions that could potentially fail the package (i.e., at 40 km/h [25 mi/h] or greater) is about $(1 - 0.64)$ or 0.36.

Figure 1. Example Illustrating Collision Failure Probability.



- b. Calculate Frequency. The frequency of collision failure is the product of the accident rate, one-way distance, number of annual shipments, the conditional probability of a collision/overtake occurring, and the conditional probability that the impact velocity generated in a collision/overtake accident is 40 km/h (25 mi/h) or greater. The frequency is:

$$\begin{aligned} \text{Frequency} &= (3.2 \times 10^{-7} \text{ accidents/mile})(40 \text{ miles/trip})(2 \text{ trips/yr})(0.7412)(0.36) \\ &= 6.8 \times 10^{-6} \text{ collision/overtake failures per year} \end{aligned}$$

2. Determine Puncture Accident Failure Frequency:

- a. Determine Puncture Failure Probability. This is a Type B package so the puncture failure probability is determined using Figure 7-13, Chapter VII of the TSD. The structural

analysis of the package determined that failure could occur when the package is subjected to a 30 ft/sec drop onto a 20 cm (8-in.) diameter puncture probe. Thus, V/R is equal to $(30 \text{ ft})(12 \text{ in/ft})/8 \text{ in.} = 45 \text{ sec}^{-1}$. Referring to Figure 7-11 from the TSD, Chapter VII, the probability of encountering puncture conditions characterized by $V/R = 45 \text{ sec}^{-1}$ is about 0.028, given that a collision/overtake accident occurs.

- b. Calculate Frequency. Similar to Step 1.b above, the annual frequency of a release due to a collision followed by puncture is:

$$\begin{aligned}\text{Frequency} &= (3.2 \times 10^{-7} \text{ accidents/mi.})(40 \text{ miles/trip})(2 \text{ trips/yr})(0.741) (0.028) \\ &= 7.2 \times 10^{-7} \text{ puncture failures per year}\end{aligned}$$

3. Determine Crush Failure Frequency:

- a. Determine Crush Failure Probability. Crush is not applicable, since the package weight exceeds 500 kg (1,102 lb)

4. Determine Fire Failure Frequency:

- a. Determine Fire Failure Probability. This fire failure probability is determined using Figures 7-13 and 7-14, Chapter VII of the TSD. The thermal analysis of the package determined that failure of a damaged package (i.e., following the collision/overtake impact event discussed above) could occur when subjected to a 25-minute regulatory fire or after 45 minutes for a non-damaged package. Referring to Chapter VII, Figure 7-13, the probability of a fire lasting 25 minutes or less following a collision or overturn is about 0.7. The probability that a fire lasts 25 minutes or longer, and threatens the integrity of the package, is $(1 - 0.7)$ or about 0.3. For non-collision accidents, the probability that a fire lasts longer than 45 minutes is $(1 - 0.9985) = 0.0015$.
- b. Calculate Frequency. Similar to Step 1.b above, the annual frequency of a release due to a collision followed by 25 minute fire or longer is:

Collision and Fire

$$\begin{aligned}\text{Frequency} &= (3.2 \times 10^{-7} \text{ accidents/mile})(40 \text{ miles/trip})(2 \text{ trips/yr}) \\ &\quad (0.7412)(0.36) (0.3) \\ &= 2.4 \times 10^{-6} \text{ failures per year}\end{aligned}$$

Collision/Puncture and Fire

$$\begin{aligned}\text{Frequency} &= (3.2 \times 10^{-7} \text{ accidents/mile})(40 \text{ miles/trip})(2 \text{ trips/yr}) \\ &\quad (0.7412)(0.028) (0.3) \\ &= 1.6 \times 10^{-7} \text{ failures per year}\end{aligned}$$

Fire-Only

$$\begin{aligned}\text{Frequency} &= (3.2 \times 10^{-7} \text{ accidents/mile})(40 \text{ miles/trip}) \\ &\quad (2 \text{ trips/yr})(0.2588)(0.0015) \\ &= 1.0 \times 10^{-8} \text{ failures per year}\end{aligned}$$

5. Summary:

The accident frequencies used to compare against the risk evaluation guidelines are as follows:

1. Impact failure – 6.8×10^{-6} per year
2. Puncture failure – 7.2×10^{-7} per year
3. Impact and fire failure – 5.7×10^{-6} per year
4. Impact/puncture and fire – 1.6×10^{-7} per year
5. Fire-only failure – 1.0×10^{-8} per year

The consequences of this accident would then be determined and the (frequency, consequence) pair plotted on the risk evaluation guideline plot to determine the acceptability of such an accident. Note that the consequences are determined separately using the methodology and data described in the TSD, Chapter VII, Section 3.2.1. The consequence estimates must consider the specific conditions generated in each accident scenario. For example, the amount of material released from an impact-only package failure should be calculated using release fractions that are representative of releases from mechanical damage to the material form. For scenarios involving fires, the release fractions that represent releases from applying thermal stress to the material form should be selected. Additional information used to develop consequence estimates are presented in the TSD, Chapter VII, Section 3.2.1.

1.5 GXQ OUTPUT FILE

Current Input File Name: temp.IN

GXQ Version 4.0
December 19, 1994

General Purpose Atmospheric Dispersion Code
Produced by Radiological & Toxicological Analysis
Westinghouse Hanford Company

Users Guide documented in WHC-SD-GN-SWD-3002 Rev. 1.
Validation documented in WHC-SD-GN-SWD-3003 Rev. 1.
Code Custodian is Brit E. Hey, WHC, ext. 376-2921.

Run Date = 06/04/01
Run Time = 13:48:18.67

INPUT ECHO:
300 Area - Sector 99.5% X/Q Values
c GXQ Version 4.0 Input File
c mode
1


```

c
c MODE CHOICE:
c mode = 1 then X/Q based on Hanford site specific meteorology
c mode = 2 then X/Q based on atmospheric stability class and wind speed
c mode = 3 then X/Q plot file is created
c
c LOGICAL CHOICES:
c   ifox  inorm icdf  ichk  isite ipop
c       T    F    F    F    F    F
c ifox = t then joint frequency used to compute frequency to exceed X/Q
c      = f then joint frequency used to compute annual average X/Q
c inorm = t then joint frequency data is normalized (as in GENII)
c      = f then joint frequency data is un-normalized
c icdf  = t then cumulative distribution file created (CDF.OUT)
c      = f then no cumulative distribution file created
c ichk  = t then X/Q parameter print option turned on
c      = f then no parameter print
c isite = t then X/Q based on joint frequency data for all 16 sectors
c      = f then X/Q based on joint frequency data of individual sectors
c ipop  = t then X/Q is population weighted
c      = f then no population weighting
c
c X/Q AND WIND SPEED ADJUSTMENT MODELS:
c   ipuff idep isrc iwind
c       0    0    0    0
c DIFFUSION COEFFICIENT ADJUSTMENT MODELS:
c   iwake ipm  iflow ientr
c       0    0    0    0
c EFFECTIVE RELEASE HEIGHT ADJUSTMENT MODELS:
c   (irise igrnd)iwash igrav
c       0    0    0    0
c ipuff = 1 then X/Q calculated using puff model
c      = 0 then X/Q calculated using default continuous plume model
c idep  = 1 then plume depletion model turned on (Chamberlain model)
c isrc  = 1 then X/Q multiplied by scalar
c      = 2 then X/Q adjusted by wind speed function
c iwind = 1 then wind speed corrected for plume height
c isize = 1 then NRC RG 1.145 building wake model turned on
c      = 2 then MACCS virtual distance building wake model turned on
c ipm   = 1 then NRC RG 1.145 plume meander model turned on
c      = 2 then 5th Power Law plume meander model turned on
c      = 3 then sector average model turned on
c iflow = 1 then sigmas adjusted for volume flow rate
c ientr = 1 then method of Pasquill used to account for entrainment
c irise = 1 then MACCS buoyant plume rise model turned on
c      = 2 then ISC2 momentum/buoyancy plume rise model turned on
c igrnd = 1 then Mills buoyant plume rise modification for ground effects
c iwash = 1 then stack downwash model turned on
c igrav = 1 then gravitational settling model turned on
c      = 0 unless specified otherwise, 0 turns model off
c
c PARAMETER INPUT:
c
c   release          reference          mixing          frequency
c   height           anemometer         height          to
c   hs (m)           height             height          exceed
c                   ha (m)              hm (m)         Cx (%)
c

```

	0.00000E+00	1.00000E+01	1.00000E+03	5.00000E-01	
c					
c	initial	initial			gravitational
c	plume	plume	release	deposition	settling
c	width	height	duration	velocity	velocity
c	Wb (m)	Hb (m)	trd (hr)	vd (m/s)	vg (m/s)
c					
	0.00000E+00	0.00000E+00	0.00000E+00	1.00000E-03	1.00000E-03
c					
c		initial	initial		convective
c	ambient	plume	plume	release	heat release
c	temperature	temperature	flow rate	diameter	rate (1)
c	Tamb (C)	T0 (C)	V0 (m3/s)	d (m)	qh (w)
c					
	2.00000E+01	2.20000E+01	1.00000E+00	1.00000E+00	0.00000E+00
c					
c	(1) If zero then buoyant flux based on plume/ambient temperature difference.				
c					
c	X/Q	Wind			
c	scaling	Speed			
c	factor	Exponent			
c	c (?)	a (?)			
c					
	1.00000E+00	7.80000E-01			
c					
c	RECEPTOR DEPENDENT DATA (no line limit)				
c	FOR MODE	make	RECEPTOR DEPENDENT DATA		
c	1 (site specific)		sector distance receptor-height		
c	2 (by class & wind speed)		class windspeed distance offset receptor-height		
c	3 (create plot file)		class windspeed xmax imax ymax jmax xqmin power		
c					
c	RECEPTOR PARAMETER DESCRIPTION				
c	sector = 0, 1, 2... (all, S, SSW, etc.)				
c	distance = receptor distance (m)				
c	receptor height = height of receptor (m)				
c	class = 1, 2, 3, 4, 5, 6, 7 (P-G stability class A, B, C, D, E, F, G)				
c	windspeed = anemometer wind speed (m/s)				
c	offset = offset from plume centerline (m)				
c	xmax = maximum distance to plot or calculate to (m)				
c	imax = distance intervals				
c	ymax = maximum offset to plot (m)				
c	jmax = offset intervals				
c	xqmin = minimum scaled X/Q to calculate				
c	power = exponent in power function step size				

MODE:

Site specific X/Q calculated.

LOGICAL CHOICES:

Joint frequency used to calculate X/Q based on frequency of exceedance.

No normalization of joint frequency.

X/Q calculated for single sector.

MODELS SELECTED:

Default Gaussian plume model selected.

WARNING/ERROR MESSAGES:

JOINT FREQUENCY DATA:

300 AREA - 10 M - Pasquill A - G (1983 - 1991 Average)

Created 8/26/92 KR

300 Area - Sector 99.5% X/Q Values

SECTOR	DISTANCE (m)	RECEPT HEIGHT (m)	SECT. FREQ. (%)	POPULATION	TOTAL POPULATION SCALED X/Q (s/m3)	AVERAGE INDIVIDUAL SCALED X/Q (s/m3)	ATM. STAB. CLASS	WIND SPEED (m/s)
S	100	0	11.55	1	3.18E-02	3.18E-02	F	0.89
SSW	100	0	5.07	1	1.16E-02	1.16E-02	F	2.65
SW	100	0	2.39	1	4.39E-03	4.39E-03	F	7.15
WSW	100	0	1.82	1	3.88E-03	3.88E-03	C	0.89
W	100	0	4.20	1	1.51E-02	1.51E-02	E	0.89
WNW	100	0	6.49	1	2.90E-02	2.90E-02	G	2.65
NW	100	0	8.60	1	3.28E-02	3.28E-02	F	0.89
NNW	100	0	5.97	1	3.29E-02	3.29E-02	F	0.89
N	100	0	8.76	1	4.21E-02	4.21E-02	F	0.89
NNE	100	0	8.50	1	2.89E-02	2.89E-02	G	2.65
NE	100	0	9.64	1	2.93E-02	2.93E-02	G	2.65
ENE	100	0	6.13	1	1.60E-02	1.60E-02	G	4.70
E	100	0	4.55	1	3.16E-02	3.16E-02	F	0.89
ESE	100	0	2.86	1	2.99E-02	2.99E-02	G	2.65
SE	100	0	5.20	1	3.16E-02	3.16E-02	F	0.89
SSE	100	0	7.95	1	3.23E-02	3.23E-02	F	0.89
S	200	0	11.55	1	9.58E-03	9.58E-03	F	0.89
SSW	200	0	5.07	1	3.49E-03	3.49E-03	F	2.65
SW	200	0	2.39	1	1.32E-03	1.32E-03	F	7.15
WSW	200	0	1.82	1	1.13E-03	1.13E-03	C	0.89
W	200	0	4.20	1	4.44E-03	4.44E-03	E	0.89
WNW	200	0	6.49	1	8.74E-03	8.74E-03	G	2.65
NW	200	0	8.60	1	9.88E-03	9.88E-03	F	0.89
NNW	200	0	5.97	1	9.92E-03	9.92E-03	F	0.89
N	200	0	8.76	1	1.27E-02	1.27E-02	F	0.89
NNE	200	0	8.50	1	8.73E-03	8.73E-03	G	2.65
NE	200	0	9.64	1	8.82E-03	8.82E-03	G	2.65
ENE	200	0	6.13	1	4.78E-03	4.78E-03	G	4.70
E	200	0	4.55	1	9.51E-03	9.51E-03	G	2.65
ESE	200	0	2.86	1	9.03E-03	9.03E-03	G	2.65
SE	200	0	5.20	1	9.53E-03	9.53E-03	F	0.89
SSE	200	0	7.95	1	9.74E-03	9.74E-03	F	0.89
S	300	0	11.55	1	4.81E-03	4.81E-03	F	0.89
SSW	300	0	5.07	1	1.75E-03	1.75E-03	F	2.65
SW	300	0	2.39	1	6.62E-04	6.62E-04	F	7.15
WSW	300	0	1.82	1	5.56E-04	5.56E-04	C	0.89
W	300	0	4.20	1	2.23E-03	2.23E-03	E	0.89
WNW	300	0	6.49	1	4.39E-03	4.39E-03	G	2.65
NW	300	0	8.60	1	4.96E-03	4.96E-03	F	0.89
NNW	300	0	5.97	1	4.98E-03	4.98E-03	F	0.89
N	300	0	8.76	1	6.38E-03	6.38E-03	F	0.89

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NNE	300	0	8.50	1	4.38E-03	4.38E-03	G	2.65
NE	300	0	9.64	1	4.43E-03	4.43E-03	G	2.65
ENE	300	0	6.13	1	2.40E-03	2.40E-03	G	4.70
E	300	0	4.55	1	4.78E-03	4.78E-03	G	2.65
ESE	300	0	2.86	1	4.53E-03	4.53E-03	G	2.65
SE	300	0	5.20	1	4.79E-03	4.79E-03	F	0.89
SSE	300	0	7.95	1	4.89E-03	4.89E-03	F	0.89
S	400	0	11.55	1	2.96E-03	2.96E-03	F	0.89
SSW	400	0	5.07	1	1.08E-03	1.08E-03	F	2.65
SW	400	0	2.39	1	4.08E-04	4.08E-04	F	7.15
WSW	400	0	1.82	1	3.36E-04	3.36E-04	C	0.89
W	400	0	4.20	1	1.38E-03	1.38E-03	E	0.89
WNW	400	0	6.49	1	2.70E-03	2.70E-03	G	2.65
NW	400	0	8.60	1	3.05E-03	3.05E-03	F	0.89
NNW	400	0	5.97	1	3.07E-03	3.07E-03	F	0.89
N	400	0	8.76	1	3.93E-03	3.93E-03	F	0.89
NNE	400	0	8.50	1	2.70E-03	2.70E-03	G	2.65
NE	400	0	9.64	1	2.73E-03	2.73E-03	G	2.65
ENE	400	0	6.13	1	1.48E-03	1.48E-03	G	4.70
E	400	0	4.55	1	2.94E-03	2.94E-03	G	2.65
ESE	400	0	2.86	1	2.79E-03	2.79E-03	G	2.65
SE	400	0	5.20	1	2.95E-03	2.95E-03	F	0.89
SSE	400	0	7.95	1	3.01E-03	3.01E-03	F	0.89
S	500	0	11.55	1	2.04E-03	2.04E-03	F	0.89
SSW	500	0	5.07	1	7.41E-04	7.41E-04	F	2.65
SW	500	0	2.39	1	2.81E-04	2.81E-04	F	7.15
WSW	500	0	1.82	1	2.28E-04	2.28E-04	C	0.89
W	500	0	4.20	1	9.52E-04	9.52E-04	E	0.89
WNW	500	0	6.49	1	1.86E-03	1.86E-03	G	2.65
NW	500	0	8.60	1	2.10E-03	2.10E-03	F	0.89
NNW	500	0	5.97	1	2.11E-03	2.11E-03	F	0.89
N	500	0	8.76	1	2.70E-03	2.70E-03	F	0.89
NNE	500	0	8.50	1	1.86E-03	1.86E-03	G	2.65
NE	500	0	9.64	1	1.88E-03	1.88E-03	G	2.65
ENE	500	0	6.13	1	1.02E-03	1.02E-03	G	4.70
E	500	0	4.55	1	2.02E-03	2.02E-03	F	0.89
ESE	500	0	2.86	1	1.92E-03	1.92E-03	G	2.65
SE	500	0	5.20	1	2.03E-03	2.03E-03	F	0.89
SSE	500	0	7.95	1	2.07E-03	2.07E-03	F	0.89
S	600	0	11.55	1	1.50E-03	1.50E-03	F	0.89
SSW	600	0	5.07	1	5.47E-04	5.47E-04	F	2.65
SW	600	0	2.39	1	2.07E-04	2.07E-04	F	7.15
WSW	600	0	1.82	1	1.66E-04	1.66E-04	C	0.89
W	600	0	4.20	1	7.05E-04	7.05E-04	E	0.89
WNW	600	0	6.49	1	1.37E-03	1.37E-03	G	2.65
NW	600	0	8.60	1	1.55E-03	1.55E-03	F	0.89
NNW	600	0	5.97	1	1.56E-03	1.56E-03	F	0.89
N	600	0	8.76	1	1.99E-03	1.99E-03	F	0.89
NNE	600	0	8.50	1	1.37E-03	1.37E-03	G	2.65
NE	600	0	9.64	1	1.38E-03	1.38E-03	G	2.65
ENE	600	0	6.13	1	7.52E-04	7.52E-04	G	4.70
E	600	0	4.55	1	1.49E-03	1.49E-03	G	2.65
ESE	600	0	2.86	1	1.42E-03	1.42E-03	G	2.65
SE	600	0	5.20	1	1.49E-03	1.49E-03	F	0.89
SSE	600	0	7.95	1	1.53E-03	1.53E-03	F	0.89
S	700	0	11.55	1	1.16E-03	1.16E-03	F	0.89
SSW	700	0	5.07	1	4.23E-04	4.23E-04	F	2.65

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SW	700	0	2.39	1	1.60E-04	1.60E-04	F	7.15
WSW	700	0	1.82	1	1.22E-04	1.22E-04	E	4.70
W	700	0	4.20	1	5.48E-04	5.48E-04	E	0.89
WNW	700	0	6.49	1	1.06E-03	1.06E-03	G	2.65
NW	700	0	8.60	1	1.20E-03	1.20E-03	F	0.89
NNW	700	0	5.97	1	1.20E-03	1.20E-03	F	0.89
N	700	0	8.76	1	1.54E-03	1.54E-03	F	0.89
NNE	700	0	8.50	1	1.06E-03	1.06E-03	G	2.65
NE	700	0	9.64	1	1.07E-03	1.07E-03	G	2.65
ENE	700	0	6.13	1	5.82E-04	5.82E-04	G	4.70
E	700	0	4.55	1	1.15E-03	1.15E-03	G	2.65
ESE	700	0	2.86	1	1.09E-03	1.09E-03	G	2.65
SE	700	0	5.20	1	1.16E-03	1.16E-03	F	0.89
SSE	700	0	7.95	1	1.18E-03	1.18E-03	F	0.89
S	800	0	11.55	1	9.29E-04	9.29E-04	F	0.89
SSW	800	0	5.07	1	3.38E-04	3.38E-04	F	2.65
SW	800	0	2.39	1	1.28E-04	1.28E-04	F	7.15
WSW	800	0	1.82	1	9.81E-05	9.81E-05	E	4.70
W	800	0	4.20	1	4.41E-04	4.41E-04	E	0.89
WNW	800	0	6.49	1	8.48E-04	8.48E-04	G	2.65
NW	800	0	8.60	1	9.59E-04	9.59E-04	F	0.89
NNW	800	0	5.97	1	9.63E-04	9.63E-04	F	0.89
N	800	0	8.76	1	1.23E-03	1.23E-03	F	0.89
NNE	800	0	8.50	1	8.47E-04	8.47E-04	G	2.65
NE	800	0	9.64	1	8.57E-04	8.57E-04	G	2.65
ENE	800	0	6.13	1	4.67E-04	4.67E-04	G	4.70
E	800	0	4.55	1	9.23E-04	9.23E-04	G	2.65
ESE	800	0	2.86	1	8.76E-04	8.76E-04	G	2.65
SE	800	0	5.20	1	9.25E-04	9.25E-04	F	0.89
SSE	800	0	7.95	1	9.46E-04	9.46E-04	F	0.89
S	900	0	11.55	1	7.64E-04	7.64E-04	F	0.89
SSW	900	0	5.07	1	2.78E-04	2.78E-04	F	2.65
SW	900	0	2.39	1	1.06E-04	1.06E-04	F	7.15
WSW	900	0	1.82	1	8.08E-05	8.08E-05	E	4.70
W	900	0	4.20	1	3.64E-04	3.64E-04	E	0.89
WNW	900	0	6.49	1	6.97E-04	6.97E-04	G	2.65
NW	900	0	8.60	1	7.88E-04	7.88E-04	F	0.89
NNW	900	0	5.97	1	7.92E-04	7.92E-04	F	0.89
N	900	0	8.76	1	1.01E-03	1.01E-03	F	0.89
NNE	900	0	8.50	1	6.96E-04	6.96E-04	G	2.65
NE	900	0	9.64	1	7.04E-04	7.04E-04	G	2.65
ENE	900	0	6.13	1	3.84E-04	3.84E-04	G	4.70
E	900	0	4.55	1	7.59E-04	7.59E-04	G	2.65
ESE	900	0	2.86	1	7.20E-04	7.20E-04	G	2.65
SE	900	0	5.20	1	7.60E-04	7.60E-04	F	0.89
SSE	900	0	7.95	1	7.77E-04	7.77E-04	F	0.89
S	1000	0	11.55	1	6.41E-04	6.41E-04	F	0.89
SSW	1000	0	5.07	1	2.33E-04	2.33E-04	F	2.65
SW	1000	0	2.39	1	8.87E-05	8.87E-05	F	7.15
WSW	1000	0	1.82	1	6.80E-05	6.80E-05	E	4.70
W	1000	0	4.20	1	3.06E-04	3.06E-04	E	0.89
WNW	1000	0	6.49	1	5.85E-04	5.85E-04	G	2.65
NW	1000	0	8.60	1	6.61E-04	6.61E-04	F	0.89
NNW	1000	0	5.97	1	6.65E-04	6.65E-04	F	0.89
N	1000	0	8.76	1	8.51E-04	8.51E-04	F	0.89
NNE	1000	0	8.50	1	5.84E-04	5.84E-04	G	2.65
NE	1000	0	9.64	1	5.91E-04	5.91E-04	G	2.65

ENE	1000	0	6.13	1	3.23E-04	3.23E-04 G	4.70
E	1000	0	4.55	1	6.37E-04	6.37E-04 G	2.65
ESE	1000	0	2.86	1	6.05E-04	6.05E-04 G	2.65
SE	1000	0	5.20	1	6.38E-04	6.38E-04 F	0.89
SSE	1000	0	7.95	1	6.52E-04	6.52E-04 F	0.89

2.0 SUMMARY

This section contains the assumptions and methodologies used to develop Table 6-7, which is found in Chapter VI. For future risk-based package evaluations, specific characteristics of the package or contents may warrant different assumptions.

Bounding radiological dose consequence and release frequency evaluations were performed for three accident scenarios for a single package containing generic solid payloads and generic transuranic (TRU) and non-TRU wastes. Results are reported for both unmitigated and mitigated accidents. With the assumptions given, the mitigated dose consequence and release frequency do not challenge the guidelines contained in DOE-STD-3009-94, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports*, Appendix A. These results are for bounding accidents selected to ensure the analyses cover the classes of activities conducted under transportation. The combined release and resultant consequences from multiple packages on the conveyance, if applicable, are bounded by the inventory limit and controlled by administrative controls. Administrative controls should be designed to mitigate the consequences, mitigate the severity of the accident, and/or reduce the frequency of the accident to satisfy the assumptions in these calculations. These controls are required when the shipment payload contains greater than 1 A₂ of radioactive material.

2.1 PACKAGES

The packages considered in this report are the same package types considered and accepted by the DOT, plus those unique packages that DOE has accepted as having acceptably low levels of risk. The content limits based on the mitigated results should be applied only to packages that are shown to pass the performance requirements for DOT Type A packages. The content limits, based on the unmitigated results, may be applied to all packages including those that do not meet the performance requirements for Type A packages, because no credit is taken for the package in ACs in the table of unmitigated results.

2.2 ACCIDENT SCENARIOS

The accident scenarios selected are those selected by DOT as representing the deterministic conditions required to be survived during transportation. A hazard analysis and accident selection for transportation is described in Section 1.3.

Collision is the most common type of transportation accident. In a collision accident, the truck hits an obstacle, causing the truck to decelerate at a rapid rate. The package, however, is still traveling at the initial velocity of the truck. The velocity of the package is reduced by the amount of energy required to break the tiedown hardware (which is small). The package then collides with the front of the conveyance, which had stopped because of the collision with the obstacle. The collision between the package and the conveyance is an impact event because the force on the package is localized on one side; this is the same type of force created when an object is dropped and hits the ground.

If there are additional packages on the conveyance that are behind the front package, then an inertial crush force is generated as these packages collide with and push against the front package, which is being resisted by the front of the conveyance. Impact and crush forces are differentiated by the application of the force to the package. If the force occurs on one side of the package, it is considered an impact, and if it occurs on multiple sides of the package, it is considered crush.

Puncture is a type of impact in which a small diameter probe impacts the package. It occurs less frequently than a collision, as described in Chapter VII, Sections 3.2.2.1.2 and 3.2.2.1.3, and the release from puncture is bounded by the release from the impact described above.

A fire may follow the collision. Transportation statistics show that fire occurs in approximately 1.6% of all accidents, as described in Chapter VII, Sections 3.2.2.1.5 and 3.2.2.1.6. These fires have variable diameter, temperature, duration, and location relative to the cargo. Unvented Type A 210 L (55-gal) drums, on average, will lose containment 3 minutes after exposure to a fully-engulfing fuel fire.

Hydrogen gas deflagrations may occur anytime, not just during transportation. Hydrogen can build up in a package due to chemical reactions between different materials of the payload and due to radiolytic decomposition of water or other hydrogenous materials. Another flammable gas (methane) can build up due to the anerobic respiration by organisms on organic material, although this mechanism has never been known to produce significant quantities of flammable gas in drums at the Hanford Site.

2.3 PAYLOADS

The payload of the package is assumed to consist of debris generated during cleanup activities at the Hanford Site. This debris most likely consists of surface contaminated equipment and combustible items. It is assumed for conservatism that particulate material may be present, which is modeled as powder. For simplicity, the isotopic composition is assumed to consist of either TRU or non-TRU material, which is modeled as either plutonium or a mixture of cesium and strontium, respectively.

2.4 RISK-BASED PACKAGING SYSTEMS ASSUMPTIONS

The assumptions described here serve to mitigate the consequences, mitigate the severity of the accident, or reduce the frequency of the accident. These assumptions are discussed quantitatively in the results section.

The speed of the truck is assumed 56 km/h (35 mi/h). This assumption limits the energy that must be dissipated in a collision accident. More importantly, it reduces the severity of the injuries to the driver because of the collision.

For arrays of packages, the least-hazardous packages (measured by dose-equivalent curie [DE-Ci]) are assumed to be placed at the front of the conveyance and the most hazardous packages are assumed to be placed at the rear of the conveyance, although this arrangement shall not result in an unacceptable direct external dose rate to the driver. This assumption mitigates the total amount released, because the packages in the front experience the largest crush forces.

For arrays of packages, impact-limiting devices, such as sandbags or empty drums, are assumed to be placed at the front of the conveyance. This limits the damage to the packages by absorbing more energy than the front structure of the conveyance, thus mitigating the release.

Fuel tanker trucks are assumed not staged within 10 m (32 ft) of the road. This limits the size and duration of the fire.

The road is assumed to be closed to all other traffic during the shipment. This applies to all shipments, not just those areas where public access is unrestricted. This reduces the frequency of multi-vehicle accidents and limits the total fuel available for a fire to only the fuel in the transport vehicle's fuel tank.

Packages are assumed to be vented prior to shipping, and chemically reactive contents are assumed not to be present to preclude a buildup of hydrogen above its lower flammability limit. The PSSD shall evaluate the buildup of hydrogen during shipment to establish a shipping window.

2.5 RESULTS

The dose consequences and the release frequency from an array of packages due to a collision, fire, or hydrogen deflagration are shown in Chapter VI, Table 6-7. A brief description of the calculations is provided in the following subsections.

2.5.1 Data Sources

The material at risk is the total material at risk on the conveyance, except for the case of hydrogen deflagration, which is the total material at risk in one package. Therefore, Chapter VI, Table 6-7 imposes two limits: 1) a shipment-level limit and 2) a package-level limit.

Because the purpose of Table 6-7 is to provide a generic indicator of the risks from transportation accidents of nonequivalent Type B packages, conservative assumptions must be used so that the results bound the risk from the majority of packages. Use of Table 6-7 is limited to packages, together with the planned payload, that satisfies the Type A packaging tests. Packages with payloads that are capable of withstanding accident environments that are more severe than the Type A packaging tests would retain more of its contents in an accident and would be bounded by the results in Table 6-7. The 210 L (55-gal) drums are considered in the calculations as a representative Type A package because they are commonly used onsite.

2.5.1.1 Fire Release Fraction. Studies of 210 L (55-gal) drums in fire tests show that in a fully engulfing fire, the lid may be completely blown off the drum, or the lid may rupture but be partially held in place by the closure ring. In either case, failure occurred in an average of 3 minutes. Drums not completely engulfed by the fire fail by lid seal degradation (WHC-SD-WM-TRP-246, *Solid Waste Drum Array Fire Performance*, Section 8.1). Therefore, two components of the airborne source term for the fire scenario are considered. The first component is the airborne release caused by the burning of ejected combustible waste. The second component is the airborne release caused by the burning of the combustible waste remaining in the package.

The fraction of material estimated to be ejected during a fully engulfing fire from a drum with the lid completely blown off is 10%. This value is used in HNF-SD-W026-SAR-002, *WRAP Final Safety Analysis Report*, and is based on experimental hydrogen deflagration testing of drums.

In a fire, a combustible payload will have a greater airborne release than powder or noncombustible payloads. Although many packages will contain a mixture of combustibles, noncombustibles, and powder, the release fraction for combustibles is used in the fire release calculations so as not to exclude packages that do not contain combustibles exclusively. The airborne release fraction (ARF) and respirable fraction (RF) for the burning of contents ejected from the package is 1×10^{-2} and 1, respectively, from DOE-HDBK-3010-94, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities*, Section 5.1, which is the bounding value for uncontained cellulose or largely cellulosic mixed waste. This is also the value used in HNF-SD-W026-SAR-002. The ARF and RF for the burning of contents remaining in the package is 5×10^{-4} and 1, respectively from DOE-HDBK-3010-94, Section 5.1, which is the bounding value for packaged mixed waste.

The resulting combined ARF, taking into account both the ejected and retained components of the fire stress, is 1.45×10^{-3} ($= 0.1 \times 1 \times 10^{-2} + 0.9 \times 5 \times 10^{-4}$). The average RF is 1.

2.5.1.2 Hydrogen Deflagration Release Fraction. Three components of the airborne source term for the hydrogen deflagration scenario are considered. The first component is the airborne release caused by the blast effects of the deflagration, which applies to all of the contents of the package. The other two components are the same as for the fire release: the burning of ejected combustible waste and the burning of combustible waste remaining in the package.

In a hydrogen deflagration, a powder payload will have a greater airborne release due to the explosive stress than combustible or noncombustible payloads. However, the combustible payload will have a greater airborne release due to the subsequent burning after the hydrogen deflagration. Although many packages will contain a mixture of combustibles, noncombustibles, and powder, the release fraction for powder is used for the explosive stress component, and the release fraction for combustibles is used for the fire components, so as not to exclude packages that do not contain powder or combustibles exclusively.

The ARF and RF for the explosive stress component of the deflagration are estimated in DOE-HDBK-3010-94 based on venting data. For the venting of powders or confinement failure at pressures in the range of 0.18 to 3.5 MPa_g (~500 psig), DOE-HDBK-3010-94 shows a bounding ARF and RF of 0.1 and 0.7, respectively. However, the DOE Rocky Flats Field Office has determined that an ARF of 2×10^{-2} is appropriate for the hydrogen deflagration scenario based on a difference between the experimental setup on which the values in DOE-HDBK-3010-94 is based and actual packaging configurations. This value was determined in consultation with the primary author, Jofu Mishima, who felt this value was still too conservative for this application (Stithem 2001, personal communication, *Drum Hydrogen Deflagration Modeling and Fire Modeling*).

The ARF and RF for the burning of ejected and retained contents caused by the deflagration are the same as for the fire release described previously.

The resulting combined ARF, taking into account the explosive stress and both the ejected and retained components of the fire stress, is 2.145×10^{-2} ($= 2 \times 10^{-2} + 0.1 \times 1 \times 10^{-2} + 0.9 \times 5 \times 10^{-4}$). The average RF is 0.72 [$= 0.7 \times 2 \times 10^{-2} + 1 \times 1.45 \times 10^{-3}$] / 2.145×10^{-2}].

The damage ratios are 1 for the unmitigated accidents and the mitigated fire and hydrogen deflagration accidents, and 0.1 for the mitigated collision accidents. These values are generally accepted as reasonable for the assumptions given. A more detailed discussion on the assumptions follows. Suffice it to say here that, with no discussion of assumed controls, the NRC (NUREG-0170, *Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes, Volume 1*) has identified damage ratios of 1 to 10% for Type A packages in Type B (i.e., severe) accident scenarios.

The χ/Q 's are determined in compliance with Regulatory Guide 1.145, *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants*. All are taken from facility SARs with two exceptions. The value at 300 m (984 ft) from the 300 Area was calculated separately. This is because the 324 Facility factored in the increased turbulence caused by the wake effect of the 324 Building. The calculated value is higher than that in the facility SAR, which results in a conservatively higher dose. The χ/Q 's for fire and hydrogen deflagration were calculated using the computer code GXQ version 4.0 for a thermal plume rise assuming the Maccs plume rise model, the Mills buoyant plume rise correction, the Pasquill entrainment model, and wind speed corrected for plume height. It is noted that the plume rise from a hydrogen deflagration is driven by momentum, which is mechanistically different than from a fire, which is driven by thermal effects; this was neglected for simplicity.

The breathing rate, $3.33 \times 10^{-4} \text{ m}^3/\text{s}$, is taken from ICRP Publication 23, *Report of the Task Group on Reference Man*, for the reference man during light activity.

The DE-Ci factors are taken from HNF-EP-0063, *Hanford Site Solid Waste Acceptance Criteria*.

The accident frequency is taken from Chapter VII, Section 3.2.2.1.1, which is from NUREG/CR-6672, *Reexamination of Spent Fuel Shipment Risk Estimates*, with the conditional probability of fire taken from SLA-74-0001, *Severities of Transportation Accidents Volume III--Motor Carriers*. The probability of a drum deflagration is taken from SAR 1996, *Rocky Flats Environmental Technology Site Safety Analysis Report*.

In addition to the consequences calculated for the site boundary at 328 m (100 m) from the release location, shown in Table 6-7, consequences are also calculated for the near bank of the Columbia River and along Highways 240 and 24 for releases in the 100 and 200 areas. The calculated consequences in Table 3 are provided for information only and are not used as limits. All of the input parameters in Table 3 are the same as in Table 6-7, except for the χ/Q' values, which are dependent on location. The computer code GXQ v 4.0 (WHC-SD-GN-SWD-30002, *GXQ 4.0 Program Users' Guide* and WHC-SD-GN-SWD-30003, *GXQ Program Verification and Validation*) was used to calculate the χ/Q' values, and these calculations are documented in DFSNW-ECAL-100, χ/Q Calculations in Support of the TSD. Potential release locations considered were the 100 N, 100 K, 200 East, and 200 West areas. Note that the χ/Q' values are calculated assuming a ground-level release and have a frequency of exceedance of 0.5% for the maximum sector or 5% for the overall site, whichever is worse.

2.5.2 Effect of Assumptions

2.5.2.1 Collision. To mitigate the consequences from a collision accident, the speed of the truck shall be limited to 56 km/h (35 mi/h), the drums at the front of the conveyance shall contain no radioactive material, and the most hazardous drums (measured by DE-Ci) shall be placed at the rear.

In a head-on collision accident, the front row of drums impacts the front of the conveyance, and the following rows of drums impact the front row of drums. This creates a crush environment, which is defined as forces impinging on more than one side of the package. A crush environment is usually more severe than an impact environment, such as a drop, in which the forces impinge on only one side of the package. Sacrificial dunnage drums in the front row will act as an impact limiter at the front of the conveyance to absorb most of the kinetic energy of the drums in following rows and ensure that the highest crush forces occur on drums containing no radioactive material. The magnitude of the crush force depends on the speed and mass of the moving object; the lower the speed or mass, the lower the crush force. Therefore, in a head-on collision, the lowest crush forces will be at the back of the trailer. The last row of drums will not be crushed, because there is nothing behind it; the drums will only experience the impact when they collide with the row of drums in front of them. Therefore, the most hazardous drums in terms of amount release should be placed at the back of the array to minimize the crush force.

Table 3. Dose Consequence at Highway and River Worst-Case Locations.

Material	MAR A ₂	DR	ARF	RF	LPF	BR m ² /s	near River bank		Highway 240/24 Consequences rem	Annual Frequency /yr	Accident-specific Assumptions
							Y/Q ² s/m ³	Z/Q ² s/m ³			
1. TRU (Pu - Alpha), 100 N. K Areas, Unmitigated											
Collision (epi)	30000	1	1E-03	1E-01	1	3.33E-04	1.94E-02	4.50E+01	1.00E-01	8.00E-05	Unmitigated
Fire	30000	1	1.45E-03	1	1	3.33E-04	1.29E-04	4.31E+00	6.90E-01	1.28E-06	Unmitigated
Hydrogen Degradation	3000	1	2.145E-02	7.2E-01	1	3.33E-04	1.29E-04	4.50E+00	7.34E-01	Anticipated	Unmitigated
2. TRU (Pu - Alpha), 200 Areas, Unmitigated											
Collision (epi)	30000	1	1E-03	1E-01	1	3.33E-04	3.39E-05	7.84E-02	4.31E-01	3.20E-04	Unmitigated
Fire	30000	1	1.45E-03	1	1	3.33E-04	2.29E-05	7.60E-01	2.50E+00	6.12E-06	Unmitigated
Hydrogen Degradation	3000	1	2.145E-02	7.2E-01	1	3.33E-04	2.29E-05	8.10E-01	2.87E+00	Anticipated	Unmitigated
3. nonTRU (Ca, Sr - Beta-Gamma), 100 N. K Areas, Unmitigated											
Collision (epi)	60000	1	1E-03	1E-01	1	3.33E-04	1.94E-02	7.00E+01	1.80E-01	9.80E-06	Unmitigated
Fire	60000	1	1.45E-03	1	1	3.33E-04	1.29E-04	7.33E+00	1.17E+00	1.54E-07	Unmitigated
Hydrogen Degradation	6000	1	2.145E-02	7.2E-01	1	3.33E-04	1.29E-04	7.81E+00	1.26E+00	Anticipated	Unmitigated
4. nonTRU (Ca, Sr - Beta-Gamma), 200 Areas, Unmitigated											
Collision (epi)	60000	1	1E-03	1E-01	1	3.33E-04	3.39E-05	1.34E-01	7.35E-01	3.20E-04	Unmitigated
Fire	60000	1	1.45E-03	1	1	3.33E-04	2.29E-05	1.29E+00	4.28E+00	5.12E-06	Unmitigated
Hydrogen Degradation	6000	1	2.145E-02	7.2E-01	1	3.33E-04	2.29E-05	1.39E+00	4.54E+00	Anticipated	Unmitigated
5. TRU (Pu - Alpha), 100 N. K Areas, Mitigated											
Collision (epi)	30000	1E-01	1E-03	1E-01	1	3.33E-04	1.94E-02	4.50E+00	1.00E-02	8.00E-05	(a) and (b)
Fire	30000	1	1.45E-03	1	1	3.33E-04	1.29E-04	4.31E+00	6.90E-01	1.28E-06	Unmitigated
Hydrogen Degradation	3000	1	2.145E-02	7.2E-01	1	3.33E-04	1.29E-04	4.50E+00	7.34E-01	Ext Unlikely	(c) and (d)
6. TRU (Pu - Alpha), 200 Areas, Mitigated											
Collision (epi)	30000	1E-01	1E-03	1E-01	1	3.33E-04	3.39E-05	7.84E-02	4.31E-02	3.20E-04	(a) and (b)
Fire	30000	1	1.45E-03	1	1	3.33E-04	2.29E-05	7.60E-01	2.50E+00	6.12E-06	Unmitigated
Hydrogen Degradation	3000	1	2.145E-02	7.2E-01	1	3.33E-04	2.29E-05	8.10E-01	2.87E+00	Ext Unlikely	(c) and (d)
7. nonTRU (Ca, Sr - Beta-Gamma), 100 N. K Areas, Mitigated											
Collision (epi)	60000	1E-01	1E-03	1E-01	1	3.33E-04	1.94E-02	7.00E+00	1.80E-02	9.80E-06	(a) and (b)
Fire	60000	1	1.45E-03	1	1	3.33E-04	1.29E-04	7.33E+00	1.17E+00	1.54E-07	Unmitigated
Hydrogen Degradation	6000	1	2.145E-02	7.2E-01	1	3.33E-04	1.29E-04	7.81E+00	1.26E+00	Ext Unlikely	(c) and (d)
8. nonTRU (Ca, Sr - Beta-Gamma), 200 Areas, Mitigated											
Collision (epi)	60000	1E-01	1E-03	1E-01	1	3.33E-04	3.39E-05	1.34E-02	7.35E-02	3.20E-04	(a) and (b)
Fire	60000	1	1.45E-03	1	1	3.33E-04	2.29E-05	1.29E+00	4.28E+00	5.12E-06	Unmitigated
Hydrogen Degradation	6000	1	2.145E-02	7.2E-01	1	3.33E-04	2.29E-05	1.39E+00	4.54E+00	Ext Unlikely	(c) and (d)

Calculations in previous safety analysis reports for packaging (HNF-2209, *Safety Analysis Report for Packaging (Onsite) Steel Drum*, Section 7.4) conservatively modeled inertial crush as a drum sandwiched between two hard unyielding plates; one an immobile surface representing the front of the trailer, and the other a 2,268 kg (5,000-lb) plate with an initial velocity of 40 km/h (25 mi/h) representing the column of drums behind the front drum. This is an idealized scenario designed to estimate conservatively the potential damage to that drum. These calculations show substantial failure of the front drum caused by buckling of the drum wall near the drum bottom plate. Evidence of crush failure first begins to show in this conservative model when the 2,268 kg (5,000-lb) plate has an initial speed of 27.8 km/h (17.3 mi/h).

Further calculations in HNF-2209 (Section 7.5) evaluate the response of an array of drums to a collision. This calculation was an analytical model as opposed to the finite-elements model as described above. The coupled, second-order differential equations of motion for an array of three drums crushed against a hard unyielding surface were solved in closed-form modeling the drums as a series of masses, springs, and dashpots. This idealized model is a simplification of the problem compared to the finite elements model described above, which is a dynamic analysis. The three drums were modeled as having an equal mass of 658 kg (1,450 lb), and the initial system velocity was 48 km/h (30 mi/h).

The results of the model indicate that the first drum has a much larger deceleration after striking the unyielding surface than the second or third drums, which are tightly coupled such that they move with nearly the same velocity and the same acceleration. The kinetic energy of the second and third drum is transformed into dissipative energy in the first drum, and because of the large deceleration of the first drum, the first drum absorbs much more energy than the second or third drum. The first drum would likely be ruptured, but the second and third drums most likely would not be ruptured. These results are conservative because they neglect the energy absorbed by the transport vehicle, energy that would not be transferred to the drums. The model considered a stationary unyielding surface, when no surface is truly unyielding. Also, the maximum weight of the drum was modeled, when in practice actual weights vary from nearly empty up to the maximum mass. Although only three drums were considered in the analysis in HNF-2209 (Section 7.5), it was concluded that it would take many more rows of drums before the second row of drums would fail. An uncertainty analysis was conducted to account for the variable mass of the drums in an array, and it was concluded with greater than 95% confidence that none of the drums would fail for an initial system velocity of 48 km/h (30 mi/h), a 70 to 80% confidence that the first drum only would fail for an initial system velocity of 56 km/h (35 mi/h), and a 30 to 40% confidence that the first drum only would fail for an initial system velocity of 64 km/h (40 mi/h).

By limiting the speed of the truck to 56 km/h (35 mi/h), it is concluded that only the first row of drums fail in a collision. Note that the impact-limiting function of the front row of drums is effective only where the collision occurs along the longitudinal axis of the trailer. In a jackknife accident, the trailer could overturn, and the drums could impact the ground. However, the shoulder and surrounding ground are yielding, which would mitigate the forces on the package. Furthermore, the percentage of jackknives among all collisions is small, and side-impact collisions by other vehicles are prevented due to road closure.

2.5.2.2 Fire. Assumed controls are not credited with any reduction in the consequence or frequency of a fire accident.

2.5.2.3 Hydrogen. To mitigate the consequences from a hydrogen deflagration, the drums shall be vented prior to shipment, and chemically reactive contents shall be prohibited. Chemically reactive contents include the co-mingling of uranium metal and water. Experience at Fernald indicates that the rate of hydrogen gas generation by the corrosion of uranium metal can overwhelm the diffusive capability of a drum vented by a NucFil filter. The rate of hydrogen gas generation by radiolysis of hydrogenous material is much smaller than that from corrosion, and would not normally be appreciable in drums for a few days' time. However, for drums with high heat generation rates and containing significant amounts of water or other hydrogenous materials, venting during shipment may be required. The PSSD will have specific requirements for venting based on an analysis of hydrogen gas generation for the given contents.

Although the lower flammability limit of hydrogen is 4%, physical testing (WSRC-TR-90-165, *TRU Drum Hydrogen Explosion Tests (U)*) showed that in the five cases where the lid was blown off, the hydrogen concentration was at minimum 17%. In the five cases where the hydrogen exploded at less than 17% (the minimum was 13%), only bulging of the drum top and bottom occurred. Measurements in pressure vessels show that the deflagration severity is a function of the volume percent of hydrogen present, with a maximum 260 psig at approximately 35 vol% hydrogen. Based on the venting requirement and the prohibition of chemically reactive contents, the hazards analysis in Chapter VII, Section 3.2.2 concluded that the occurrence of a flammable gas deflagration is an incredible event. Consequently, the probability that more than one drum on the conveyance would experience a hydrogen deflagration is also incredible.

3.0 REFERENCES

- 10 CFR 71, "Packaging and Transportation of Radioactive Material," *Code of Federal Regulations*, as amended.
- 10 CFR 835, "Occupational Radiation Protection," *Code of Federal Regulations*, as amended.
- 49 CFR 171 through 180, Subtitle B, "Other Regulations Relating to Transportation," *Code of Federal Regulations*, as amended.
- 49 CFR 173, "Shippers--General Requirements for Shipments and Packagings," *Code of Federal Regulations*, as amended.
- DOE Order 460.1A, *Packaging and Transportation Safety*, U. S. Department of Energy, Washington, D. C.

- DOE-HDBK-3010-94, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities*, Change Notice 1, U.S. Department of Energy, Washington, D.C.
- DOE-STD-3009-94, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports*, Change Notice No. 1, U.S. Department of Energy, Washington, D.C.
- HNF-2209, *Safety Analysis Report for Packaging (Onsite) Steel Drum*, Fluor Hanford, Richland, Washington.
- HNF-EP-0063, *Hanford Site Solid Waste Acceptance Criteria*, Rev. 5, Fluor Hanford, Richland, Washington.
- HNF-SD-W026-SAR-002, *WRAP Final Safety Analysis Report*, Rev. 2, Fluor Hanford, Richland, Washington.
- ICRP, 1975, *Report of the Task Group on Reference Man*, International Commission on Radiological Protection Report No. 23, Elmsford, New York.
- IAEA, 1990, *Explanatory Material for the IAEA Regulations for the Safe Transport of Radioactive Material*, Safety Series No. 7, Second Edition (as amended 1990), International Atomic Energy Agency, Vienna.
- NUREG-0170, *Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes, Volume 1*, U.S. Nuclear Regulatory Commission, Washington, D.C.
- NUREG/CR-6672, *Reexamination of Spent Fuel Shipment Risk Estimates*, Sandia National Laboratories, Albuquerque, New Mexico.
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- SAR, 1996, *Rocky Flats Environmental Technology site Safety Analysis Report, Volume I, Site Description and Characteristics*, Kaiser-Hill Company, Rocky Flats Environmental Technology Site, Golden Colorado.
- SLA-74-0001, *Severities of Transportation Accidents Volume III--Motor Carriers*, 1976, Sandia Laboratories, Albuquerque, New Mexico.
- Stithem, A., 2001, *Drum Hydrogen Deflagration Modeling and Fire Modeling*, personal communication with J. L. Boles and D. W. Bergmann, email, August 9, 2001.

WHC-SD-GN-SWD-30002, *GXQ 4.0 Program Users' Guide*, Rev. 1, Westinghouse Hanford Company, Richland, Washington.

WHC-SD-GN-SWD-30003, *GXQ Program Verification and Validation*, Rev. 1, Westinghouse Hanford Company, Richland, Washington.

WHC-SD-WM-TRP-246, *Solid Waste Drum Array Fire Performance*, 1995, Rev. 0, Los Alamos Technical Associates, Kennewick, Washington, and Hughes Associates, Baltimore, Maryland, for Westinghouse Hanford Company, Richland, Washington.

WSRC-TR-90-165, *TRU Drum Hydrogen Explosion Tests (U)*, Westinghouse Savannah River Company, Aiken, South Carolina.

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APPENDIX H
EXAMPLE CHECKLISTS

CONTENTS

Hazardous Material/Waste Checklist - Highway Milkrun Shipments	App H-2
Checklist for Radioactive Material By Highway.	App H-3
Radioactive Mixed Waste Checklist.	App H-5
Non-radioactive Hazardous Waste Checklist, Truck Freight (1162 Bldg. Review).	App H-7

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This appendix contains examples of checklist used at the Hanford Site. The examples may be used as a starting point for developing checklists to be used when conducting onsite shipments. Additional examples of checklists may be found on the Hanford Internet at http://www.hanford.gov/pss/t&p/shipper_checklists.html.

The following example checklists are provided in this appendix.

- Hazardous Material/Waste Checklist - Highway Milkrun Shipments
- Checklist for Radioactive Material By Highway
- Radioactive Mixed Waste Checklist
- Non-radioactive Hazardous Waste Checklist, Truck Freight (1162 Bldg. Review).

Shipper: _____

1 over 1: _____

Shipment #:
Charge Code:**HAZARDOUS MATERIAL/WASTE CHECKLIST - HIGHWAY
MILKRUN SHIPMENTS**

1. No breaches in Containment			
2. No Excessive Paint chips or surface tears			
3. Closed Securely per manufacturer's instructions or procedures			
4. No rust or creases or bulges			
5. Correct package for the material shipped			
6. Overall construction satisfactory (See Reverse)			
7. If shipment shipped under an exemption check the following:			
8. Do you have a current copy of the exemption on file?			
9. Check expiration date on exemption. Is it current?			
10. Is exemption valid for the product being shipped in the package?			
11. A copy of exemption must accompany the shipment			
Shipping Labels			
12. Shipper			
13. Receiver			
14. Word "Waste" marked in association with PSN (if required)			
15. Letters "RQ" marked in association with PSN (if required)			
16. Proper Shipping Name			
17. Identification Number			
18. N.O.S. constituents in parentheses			
19. Poison constituents in parentheses			
20. Hazardous substance in parentheses			
21. Primary label			
22. Subsidiary Label without Hazard Class			
23. Container Specification Marking			
24. Exemption Number on package			
25. Max. permissible gross wgt. of container (verify manufacturer's embossed marking)			
26. Gross weight, compare weight(s) to SWTS			
27. Orientation Arrows (combination packages of liquid)			

Rev. 2 01/18/00

Date:
RSR #:

Shipper: _____
1 over 1: _____

CHECKLIST FOR RADIOACTIVE MATERIAL BY HIGHWAY			
1. Charge code for billing			
2. Charge code for labor			
3. RSR			
4. Other characterization documentation			
5. Is the material greater than 70 Bq per gram			
6. Does the material conform to any exceptions			
7. Is it specifically listed Uranium or Thorium			
8. Is it a RCRA waste			
9. Is it special form with certificate			
10. Material is fissile or fissile excepted			
11. If being shipped as "Empty", internal smears done			
12. Is a RCRA waste (Note: cannot be shipped by air)			
13. Meets definition of other hazard class			
14. Radiation and contamination levels conform to PSN			
15. Class 7 calculations reviewed/attached			
16. CoC reviewed			
17. SARP reviewed if required			
18. Exemptions reviewed if applicable and current			
19. Full name and address of shipper			
20. Full name and address of receiver that is authorized with license on file			
21. Number of pages shown			
22. Word "Waste" if RCRA waste			
23. RQ in association with PSN			
22. Proper Shipping Name			
24. Uranium or Thorium specifically listed PSN (note: the words, "Radioactive Material" must be written in association with PSN. Be shown in conjunction with PSN)			
25. Emergency Response Guide number			
26. Emergency Response Phone number			
27. Hazard class and UN number			
28. Subsidiary hazard and packing group (optional)			
29. Physical and chemical form			
30. LSA or SCO group			
31. Number of packages, type, activity in Bq and T.J. for each package			
32. Name or symbol for radionuclides			
33. CoC number			
34. Note if the package is an overpack			
35. Category of label shown and correct for dose rates (Radioactive I, II, III)			
36. Exclusive Use Shipment			
37. Fissile excepted/Fissile material controlled shipment			
38. Package identification marks shown and copy of certificate included			

Date:
RSR #:

Shipper: _____
1 over 1: _____

CHECKLIST FOR RADIOACTIVE MATERIAL BY HIGHWAY			
39. Emergency contact number listed and guide number provided			
40. TID seal number listed			
41. Radiation survey completed and signed by HPT			
42. Signature of shipper			
43. One over one review completed			
44. Packaging is correct for the material classification with manufacturer certification			
45. Packaging is in unimpaired condition			
46. Package closed per manufacturer instructions or appropriate procedure			
47. Exemption for package, is current, and covers the RAD material in			
48. Name and address of Shipper and Consignee/Receiver			
49. "Waste" and/or "RC" marked in association with PSN.			
50. Proper Shipping Name			
51. ID #			
52. Excepted package statement in or on package or with shipping papers			
53. Primary and subsidiary labels applied near PSN. Two each on package			
54. Tamper seal affixed to package			
55. Orientation arrows (combination packages of liquids)			
56. "Type A" or "Type B" with trefoil marking on package			
57. Max. permissible gross wgt. of container (verify manufacturer's embossed marking)			
58. Gross Weight			
59. Carrier Print: _____			
60. Vehicle number Print: _____			
61. Vehicle inspection date within 1 year of shipment Print: _____			
62. Vehicle tires meet tread requirements			
63. Hazardous material certificate of registration current			
64. Latest emergency response guidebook in vehicle			
65. Driver name Print: _____			
66. CDL with correct endorsements Expiration Date: _____			
67. DOT medical card Expiration Date: _____			
68. Load properly braced, shored and secured			
69. Vehicle placarded as required (Yellow-III/HRCQ/LSA Exclusive Use)			
70. All hazardous materials segregated as per Segregation Table			
70. POC and ONC notification made (provide shipping papers & applicable attachment)			

Rev. 3 05/24/00 T0005032.4

Shipper: _____

1 over 1: _____

RADIOACTIVE MIXED WASTE CHECKLIST			
1. The consignee is authorized to receive the material			
2. Generator's US EPA ID #			
3. Manifest #			
4. Generator's name and address			
5. Transporter (1) Company name & EPA ID #			
6. Transporter (2) Company name & EPA ID #			
7. Designated facility name and site address			
8. Designated facility EPA ID #			
9. Transporter's phone numbers			
10. Designated facility phone number			
11. "RQ" or "X" in HM column (as required)			
12. Proper Shipping Name shown (*If Excepted Qty RAM SEE SECTION BELOW) - PSN/Hazard Class/ID #/PG (if subsidiary hazard) - RQ if hazardous substance			
13. The words "Radioactive Material" unless in proper shipping name (172.203(d)(1))			
14. Name or symbol of radionuclides			
15. Description of physical & chemical form, or special form			
16. The group for LSA-I-II-III and SCO-II			
17. The activity in Bq			
18. Type of radioactive material label (i.e., White I, Yellow II/III) identified?			
19. Transport Index (TI) listed for each package			
20. If fissile shipment, the weight in grams or kilograms of fissile isotopes			
21. Fissile Controlled warning			
22. "Fissile Excepted"			
23. Is activity within limits for Type A: Type B:			
24. If Type B, is Certificate of Compliance number noted? Has CoC been checked for compliance with quantity or other restrictions and current?			
25. "Highway Route Controlled Quantity" in association with Basic Description?			
26. Subsidiary hazard listed (class/division)			
27. If HRCQ, Route Plan attached?			
28. Number of packages			
29. Type of packages			
30. Weight in Kg			
31. Waste codes listed			
32. Is emergency contact number noted?			
33. Seal number noted?			
34. Carrier possesses most current ERG guide			
35. Placarding noted			
36. "Exclusive Use Shipment", paperwork attached?			
37. Page # on manifest			
38. Generator signed and dated manifest			
39. Transporter (1) signed and dated manifest			
40. Transporter (2) signed and dated manifest			

Shipper: _____

1 over 1: _____

RADIOACTIVE MIXED WASTE CHECKLIST			
Item	Y	N	NA
41. If a waste shipment review HWM, SWITS, 310 Report			
42. If not a Waste Review Bill of Lading, HMSR			
43. Does number and type of package match source documents?			
44. Hazardous waste sticker with manifest #, PSN, Hazard Class, ID #, PG (If subsidiary hazard), waste codes, and RQ (If hazardous substance)			
45. Gross Weight (≥ 50 Kgs or 110 lbs)			
46. Gross Weight, compare weight(s) to SWITS			
47. Max. permissible gross wgt. of container (verify manufacturer's embossed marking)			
48. Type A or B noted, if applicable - Manufacturer or shipper certification			
49. Hazardous and subsidiary risk labels applied - 2 labels filled out correctly - Category of RAM label applied [] None [] White I [] Yellow II [] Yellow III			
50. Package free of damage & surveyed by HPT			
51. Certificate of Compliance number attached if Type B - Serial number - Radiation symbol			
52. Seal(s) attached			
53. Check external cask temperature (max. 122°F for LTL, 180° F for Ex Use)			
54. Vehicle given visual inspection			
55. Package secured in or on vehicle			
56. Vehicle placarded as required or orange panels or PCB marking			
57. If HRCQ or TL quantities of radioactive, has carrier had a Motor Carrier Evaluation?			
58. Use Non-Radioactive Hazardous Waste Checklist. "Limited quantity radioactive material" must appear on the shipping paper (or manifest) in association with the basic description			
59. Carrier			
60. Vehicle Number			
61. Vehicle inspection date within 1 year of shipment			
62. Vehicle Tires meet Thread requirements			
63. Hazardous Material Certificate of Registration current			
64. Current Emergency Response Guidebook in vehicle			
65. Driver Name			
66. Current Commercial Driver License with proper endorsements and Physical Card			
67. Load properly braced, shored and secured			
68. Vehicle placarded as required or orange panels or PCB marking			
69. All Hazardous materials segregated as per Segregation Table			
70. POC and ONC notification made			

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Shipper: _____

1 over 1: _____

NON-RADIOACTIVE HAZARDOUS WASTE CHECKLIST TRUCK FREIGHT (1162 BLDG. REVIEW)			
1. Generator's EPA ID #			
2. Manifest #			
3. Generator's name and address			
4. Transporter (1) Company name & EPA ID #			
5. Transporter (2) Company name & EPA ID #			
6. Designated facility name and site address			
7. Designated facility EPA ID #			
8. Transporter's phone numbers			
9. Designated facility phone number			
10. "RQ" or "X" in HM Column (as required)			
11. The word "Waste" for Federal Waste Codes			
12. Proper Shipping Name			
13. Hazard class or division number			
14. Identification Number			
15. Packing Group			
16. N.O.S. entries			
17. The word "Poison" (if required) (6.1 PG I or II) (as in 172.203(K) and (M))			
18. Poison constituents (2.3, 6.1, PG I or II)			
19. Hazardous substances			
20. Additional requirements in 172.203 (Poison Inhalation and Hazard Zone are shown)			
21. Exemption number (copy of exemption on file and with shipment, if required)			
22. Limited quantity			
23. Empty packages may be worded, "Residue last contained..."			
24. Number of packages			
25. Type of package (DM, CF, etc.)			
26. Weight listed (in kg or other vol.)			
27. Waste codes listed (when required)			
28. Emergency contact phone # (box (J) on manifest) including area code			
29. Does carrier possess most current ERG guide			
30. Vehicle placarded as required or orange panels or PCB marking			
31. Page numbers shown on manifest			
32. Generator signed and dated manifest			
33. Transporter (1) signed and dated manifest			
34. Transporter (2) signed and dated manifest			
35. If a waste shipment review HWM, SWTS, 310 Report			
36. If not a Waste Review Bill of Lading, HMSR			
37. Does number and type of package match source documents?			

Shipper: _____

1 over 1: _____

NON-RADIOACTIVE HAZARDOUS WASTE CHECKLIST TRUCK FREIGHT (1162 BLDG. REVIEW)			
38. Authorized in Column 8 of 172.101 or 173.12			
39. No Damage			
40. If shipment shipped under an exemption check the following:			
41. Do you have a current copy of the exemption on file?			
42. Check expiration date on exemption			
43. Is exemption valid for the product being shipped in the package?			
44. A copy of exemption must accompany the shipment			
45. Hazardous Waste Sticker with manifest #, PSN, ID # and waste codes (when required)			
46. Marked Inhalation Hazard (when required)			
47. Exemption packages marked with DOT-E #			
48. Address of shipper			
49. Combination packages of liquid marked with orientation arrows			
50. Labpack list attached to container (when required)			
51. Max. permissible gross wgt. of container (verify manufacturer's embossed marking)			
52. Gross weight, compare weight(s) to SWTS			
53. Packages labeled with primary and secondary labels without number (as required)			
54. Containers surveyed by 1100 Area HPT and PTR form signed off by 1100 Area HPT			
55. Segregation of hazardous materials (complies with segregation table 49 CFR 177.848)			
56. Vehicle is placarded (if required)			
57. Driver trained per DOT requirements (CDL with proper endorsements and medical card)			
58. Vehicle meets all requirements of the Federal Motor Carrier Safety Regulations (annual inspection current)			
59. Package secured in or on the vehicle			
60. If hazardous waste, has the carrier had a Motor Carrier Evaluation?(Possess EPA ID #)			
61. Has bulk container been tested within the required time limits and the container approved and properly tested			
62. Carrier			
63. Vehicle Number			
64. Vehicle inspection date within 1 year of shipment			
65. Vehicle Tires meet Tread requirements			
66. Hazardous Material Certificate of Registration current			
67. Latest Emergency Response Guidebook in vehicle			
68. Driver Name			
69. Current Commercial Driver License with proper endorsements and Physical Card			
70. Load properly braced, shored and secured			
71. Vehicle placarded as required			
72. All Hazardous materials segregated as per segregation table			
73. POC and ONC notification made			

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