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ENGINEERING DATA TRANSMITTAL

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Page 1 of 1

1. EDT

628163

2. To: (Receiving Organization) Safety Review Board	3. From: (Originating Organization) WMTS Engineering	4. Related EDT No.: NA
5. Proj./Prog./Dept./Div.: 772030/233, 324 Stabilization HMS40061	6. Design Authority/ Design Agent/Cog. Engr.: W. M. Boehnke	7. Purchase Order No.: NA
8. Originator Remarks: Original document transmitted to RL for review on 02/14/00. DOE and Fluor Hanford (FH) comments have been incorporated and are being retransmitted for approval.		9. Equip./Component No.: NA
11. Receiver Remarks: 11A. Design Baseline Document? ☐ Yes ☒ No		10. System/Bldg./Facility: NA
		12. Major Asm. Dwg. No.: NA
		13. Permit/Permit Application No.: NA
		14. Required Response Date: 06/02/00

15. DATA TRANSMITTED					(F)	(G)	(H)	(I)
(A) Item No.	(B) Document/Drawing No.	(C) Sheet No.	(D) Rev. No.	(E) Title or Description of Data Transmitted	Approval Designator	Reason for Transmittal	Originator Disposition	Receiver Disposition
1	HNF-4763	NA	0	Safety Analysis Report for Packaging (Onsite) Steel Waste Package	SQD	1	1	

16. KEY		
Approval Designator (F)	Reason for Transmittal (G)	Disposition (H) & (I)
E, S, Q, D or N/A (see WHC-CM-3-5, Sec 12.7)	1. Approval 2. Release 3. Information 4. Review 5. Post-Review 6. Dist. (Receipt Acknow. Required)	1. Approved 2. Approved w/comment 3. Disapproved w/comment 4. Reviewed no/comment 5. Reviewed w/comment 6. Receipt acknowledged

17. SIGNATURE/DISTRIBUTION (See Approval Designator for required signatures)									
(G) Reason	(H) Disp	(J) Name	(K) Signature	(L) Date	(M) MSIN	(G) Reason	(H) Disp	(J) Name	(K) Signature MSIN
		Design Authority				1	1	JA Reeves	6/7/00 H1-11
		Design Agent				1	1	DS Takasumi	6/7/00 L1-02
1	1	Cog. Eng. WM Boehnke		6/2/00	H1-11	1	1	DW Fritz	6/7/00 G1-29
1	1	Cog. Mgr. DW Bergmann		6/2/00	H1-11	1	1	RL BOGART	6/7/00 L1-02
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1	1	Ind. Review PC Ferrell*		6/7/00	H1-11				
1	1	BB Peters		6/2/00	H1-11				

18. Signature of EDT Originator W. M. Boehnke 6/7/00	19. Authorized Representative Date for Receiving Organization DL McCall 6/7/00	20. Design Authority Cognizant Manager L.T. Blackford 6/7/00	21. DOE APPROVAL (if required) Ctrl. No. 00-ABD-045 6-22-00 ☒ Approved ☐ Approved w/comments ☐ Disapproved w/comments
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memorandum

DATE: September 29, 2000

REPLY TO

ATTN OF: NE-20

SUBJECT: Removal of Applied Technology Designation from Specific Documents

TO: M. Wilson, Office of Scientific and Technical Information

The Office of Nuclear Energy, Science and Technology requests the Applied Technology (AT) designation be removed from the following documents::

- WHC-SD-TP-RPT-005, Rev. 1, Thermal Analysis Methods for Safety Analysis Reports for Packaging (SARP), January 11, 1996
- HEDL-TC-1613, Thermal Characterization of Ident 1578 Container, June 1982
- WHC-SD-FF-ER-050, Rev. 0, TAP-A Code Notebook, March 1990
- HNF-6550 T Plant Canyon Items Safety Evaluation Report (SEP), July 2000
- HNF-6743 T Plant Overpack Tiedown Evaluation, July 2000
- • HNF-4763 SARP (Onsite) Steel Waste Package, July 2000
- HNF-6807 Safety Analysis Report for Packaging (Onsite) for the Pipe Overpack Container
- 3I-SOP-REC-A-47
- 3I-SOP-REC-A-51
- 3I-SOP-REC-A-67
- Any other SARP, SER, or other document that is designated AT due to referencing one or more of the above documents.

Mr. O. A. Farabee's August 25, 2000, memorandum to Dr. Rob M. Versluis, on the subject of Applied Technology Designation Removal of Specific Documents, and its enclosure provides the explanation for making this request and is attached as part of the record. Also, Mr. Don Riley's e-mail to Rob Versluis dated September 11, 2000, is also attached as part of the record. I note further that the source AT document, which has caused a propagation of further AT document designations, contains technology information that, because of its age, is unlikely to provide commercial leverage to the Department in collaborative projects with other countries. The documents that received the AT designation by propagation (the rule is that a document that references an AT document is itself designated as AT) would not have been designated AT in their own right.

R. Shane Johnson, Acting Associate Director
for Technology and International Cooperation
Office of Nuclear Energy, Science
and Technology

Attachments

cc: O. A. Farabee, RL

Enclosure:
Applied Technology Designation Removal on Specific Documents

Waste Management Technical Services (WMTS) is assigned responsibility for developing and maintaining a program for the packaging and transportation of all hazardous material and wastes for the Hanford site. As part of this program, WMTS prepares and maintains safety analysis documentation for packaging, which includes Safety Analysis Reports for Packaging (SARPs) and Safety Evaluation Reports (SEPs).

One of the requirements of the SARPs and SEPs is to document safety of packaging from thermal environments, both normal and accident. WMTS has for the last five years included a reference to WHC-SD-TP-RPT-005, Rev. 1, *Thermal Analysis Methods for Safety Analysis Reports for Packaging*, January 11, 1996. Originally, this revised report was incorrectly cleared as a "Public Release" document on January 11, 1996. On March 3, 2000, the document was changed to Applied Technology (AT). The change was made due to the observation of an AT reference in the appendix of the document.

WMTS has many SARPs, and SEPs that have been released in the last five years that are public documents. Maintenance and revisions of these SARPs and SEPs will have the unnecessary AT designation added due to this reference. It is therefore requested that the documents causing this AT designation to the SARPs and SEPs have their AT designation removed, where necessary and permitted. Figure 1 shows a simple pictorial guide of effected documents and what drives the documents to be AT.

This request is to not remove all AT documents that are found in the references of the SARPs, but only those found in the upper three tiers of references in a SARP. It is proposed the SARP, what the SARP references, and what the references found in the SARP reference would all be made public documents. Below that tier there may likely remain AT references. The following discussion addresses each document as to the specific recommendations.

WHC-SD-TP-RPT-005, Rev. 1, *Thermal Analysis Methods for Safety Analysis Reports for Packaging*, January 11, 1996, was designated as Applied Technology information under the current guidelines because it references a document previously designated as Applied Technology, HEDL-TC-1613, *Thermal Characterization of Ident 1578 Container*, June 1982. This is what led to this documents redesignation earlier this year.

Under the current guidelines for designating Applied Technology information, it would not be appropriate to so designate HEDL-TC-1613. The document includes no references and provides no detailed engineering information about either fuel pins or the Ident 1578 fuel pin container. Electrical heaters were arrayed within the container and the container surface temperature was determined as a function of power input. The bulk of the document is a compilation of this test data. Therefore, there is no apparent reason to retain the Applied Technology designation for HEDL-TC-1613. Senior engineers of the FFTF technical team have provided basis for this conclusion. Their comments are included at the end of this enclosure.

WHC-SD-TP-RPT-005, Rev. 1, *Thermal Analysis Methods for Safety Analysis Reports for Packaging*, January 11, 1996, also references WHC-SD-FF-ER-050, Rev. 0, *TAP-A Code Notebook*, March 1990, which is currently identified as "Sponsor Limited." If further reviewed for unrestricted release, this document would have to be redesignated as "Applied Technology" due to a reference it makes to an AT document, HEDL-TC-2683, Rev. 0, *Evaluation of Natural Convective Air Cooling Test in FFTF Interim Decay Storage Vessel*, June 1985.

In view of the intended use of the report WHC-SD-TP-RPT-005, Rev. 1, *Thermal Analysis Methods for Safety Analysis Reports for Packaging*, January 11, 1996, the following action is recommended as reasonable and appropriate:

Identify the following documents as "base technology:"

- WHC-SD-TP-RPT-005, Rev. 1, *Thermal Analysis Methods for Safety Analysis Reports for Packaging*, January 11, 1996
- HEDL-TC-1613, *Thermal Characterization of Ident 1578 Container*, June 1982.
- WHC-SD-FF-ER-050, Rev. 0, *TAP-A Code Notebook*, March 1990,

Retain the "Applied Technology" designation on HEDL-TC-2683, Rev. 0, *Evaluation of Natural Convective Air Cooling Test in FFTF Interim Decay Storage Vessel*, June 1985, even though it is referenced in WHC-SD-FF-ER-050.

In addition, the following new documents that are currently AT would need to be identified as "base technology." These documents are AT due solely to the reference to the AT designated WHC-SD-TP-RPT-005 report above:

- HNF-6550 *T Plant Canyon Items SEP*, July 2000
- HNF-6743 *T Plant Overpack Tiedown Evaluation*, July 2000
- HNF-4763 *SARP (Onsite) Steel Waste Package*, July 2000
- 3I-SOP-REC-A-47
- 3I-SOP-REC-A-51
- 3I-SOP-REC-A-67
- Any other SARP, SEP, or related document that is designated AT due to the above reference (those anticipated to be released during this redesignation process).

Satisfactory resolution of this recommendation will stop the unnecessary AT designation from propagating further.

Figure 1

Applied Technology Reference Tree and Base Technology Identifier

Following are the comments and analyses received from the FFTF senior engineers.

FFTF Technical Staff review of HEDL-TC-1613, emailed from RB Baker, 7/31/00:

Does this document look like it could be down graded from Applied Technology to open literature?

This is basically a report on a cold thermal test of a mockup of the Ident 1578 shipping container for mixed-oxide fuel pins such as used in FFTF. It does not review designs, discuss theories or provide detailed evaluations using base technology. There looks to be very little unique or innovative information:

- no references are used (neither AP nor cleared)
- the subject Ident container is not described in any detail and appears to be generally a pipe with end closures
- some general information is noted on the FFTF mixed oxide driver fuel pins (diameter, wire wrap size, etc.) all of which is in the open literature
- The maximum decay heat expected for the 40 mixed oxide fuel pins while in the Ident 1578 are noted as limit for T3 cask-- nothing else is provided on the T3 (T3 is not AP per SA Chastain review)
- The thermal test data and analyses could be viewed as base data which though interesting and perhaps unique at the time (1982) it is not particularly innovative and the test could be easily repeated in a thermal lab anywhere.

I have not been trained in determining if a document is "Applied Technology" or not. However, I am of the opinion, based on this limited review, that at this point in time the subject report does not appear to contain what I would infer are "applied technology" subjects or topics from the FFTF/LMR point of view. This based on my over 15 years experience working with FFTF/LMR/ALMR projects.

RB Baker 7/31/00

FFTF Technical Staff review of HEDL-TC-1613, emailed from S. Chastain, 7/28/00:

Can the document be down graded from Applied Technology to open literature?

The test is basically a cold thermal test in the laboratory of the IDENT 1578-shipping container for mixed-oxide fuel pins.

- Does not reference documents that are Applied Technology
(Verified that the T-3 Cask SARP is an open literature document therefore not Applied Technology - Verification from Dan Johnston and Erik Neilsen)
- Information on FFTF fuel pin design is contained, however this information has previously been presented in open literature

I am not trained in classification of documents. However, this document doesn't seem to contain information that would qualify as FFTF/LMR Applied Technology.

Steve Chastain 7/28/00

1

U.S. DEPARTMENT OF ENERGY
RICHLAND OPERATIONS OFFICE
P. O. BOX 550 MSIN N2-36
825 JADWIN AVENUE
RICHLAND, WASHINGTON 99352

FFTF STANDBY PROJECT OFFICE
VERIFY: (509) 376-6314

FAX NUMBER: (509) 376-0177

DATE 10/2/00

TO: Yvonne Sherman 376-1563
Chris Williamson 376-4989

Don Riley 372-1435
Shirley Myers 372-1435
John Reeves 372-1435

FAX NUMBER _____

FROM: Bob Almqvist

TELEPHONE: 376-2171

This transmittal consists of 8 pages, including this cover sheet.

MESSAGE:

AT Approval

HNF-4763
Revision 0
EDT 628163
Copy No. ____

Safety Analysis Report for Packaging (Onsite) Steel Waste Package

Document Type: SARP

Division: WM

W. M. Boehnke

Waste Management Federal Services, Inc., A Subsidiary of GTS Duratek

Date Published

July 2000

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

Fluor Hanford

P.O. Box 1000

Richland, Washington

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References WHC-SD-TP-RPT-005, Applied Technology per Doug Bantle, 3/2000.			
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Key Words: SARP, 22-ton steel waste disposal box, SWDB

Abstract: The steel waste package is used primarily for the shipment of remote-handled radioactive waste from the 324 Building to the 200 Area for interim storage. The steel waste package is authorized for shipment of transuranic isotopes. The maximum allowable radioactive material that is authorized is 500,000 Ci. This exceeds the highway route controlled quantity (3,000 A₂s) and is a type B packaging.

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	0	Abstract
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REPORT FOR PACKAGING (SARP) (ONSITE) STEEL WASTE PACKAGE (SWP)

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JUN 22 2001

Mr. R. D. Hanson, President
Fluor Hanford, Inc.
Richland, Washington 99352

Dear Mr. Hanson:

CONTRACT NO. DE-AC06-96RL13200 - APPROVAL OF SAFETY ANALYSIS REPORT FOR
PACKAGING (SARP) (ONSITE) STEEL WASTE PACKAGE (SWP)

The SARP for the SWP, HNF-4763, Revision 0, is approved. This report authorizes use of the SWP for transport of remote-handled radioactive waste from the 324 Building to the 200 Area for interim storage. The attached Safety Evaluation Report (SER) documents the RL review of the SARP and identifies any limitations on use of the package.

If any direction is provided by a Contracting Officer's Representative (COR) which your company believes exceeds the COR's authority, you are to immediately notify the Contracting Officer and request clarification prior to complying with the direction.

If you have any questions or comments, please contact Richard Self, Authorization Basis Division, on (509) 372-2843.

Sincerely,

A handwritten signature in black ink, appearing to read "Peter M. Knollmeyer".

Peter M. Knollmeyer, Assistant Manager
for Nuclear Materials and Facility Stabilization

ABD:RJS

Attachment

cc w/attach:
E. S. Aromi, FHI
L. T. Blackford, FHI

U.S. DEPARTMENT OF ENERGY
RICHLAND OPERATIONS OFFICE
SAFETY EVALUATION REPORT FOR
SAFETY ANALYSIS REPORT FOR PACKAGING (ONSITE)
STEEL WASTE PACKAGE
JUNE 2000

Authorization Basis Division
U.S. Department of Energy, Richland Operations Office

Safety Evaluation Report

Date Published: June 2000

Safety Analysis Report for Packaging (Onsite)
Steel Waste Package

U.S. Department of Energy
Richland Operations Office

Approval

Sandra L. Johnson, Assistant Manager
for Safety and Engineering

C. J. Bosted,
Authorization Basis Division

Richard J. Self, Team Lead

Introduction

This Safety Evaluation Report (SER) documents the U.S. Department of Energy, Richland Operations Office (RL) review of the Safety Analysis Report for Packaging (SARP) (Onsite) Steel Waste Package, document HNF-4763, Rev. 0. The basis and any conditions of approval are identified.

The Steel Waste Package (SWP) is used to transfer remote-handled radioactive waste from the 324 Building to the 200 Area for interim storage. The SWP consists of a Steel Disposal Box Over pack and Steel Waste Disposal Box (SWDB).

The waste is expected to contain large quantities of ^{137}Cs and ^{90}Sr . It is also expected to contain small amounts of Transuranic (TRU) isotopes with total package contents exceeding the highway route controlled quantity ($3,000\text{A}_2$) of radioactive material. The maximum allowable radioactive material that will be shipped under this SARP is 500,000 Curies (Ci). Containment is provided by the cask during transport or accident conditions considered. No testing of the SWP was conducted. SARP evaluation was by analysis.

The SWP, as documented in the SARP, meets the requirements of DOE Order 460.1 *Packaging and Transportation Safety* for on site shipments. On site shipments are required to demonstrate equivalent level of safety to 10 CFR 71 requirements. The SARP analysis shows that the SWP maintains leak tight containment of the waste for all Normal Conditions of Transport (NCT) and for Hypothetical Accident Conditions (HAC) as required by 10CFR71. This packaging system therefore, meets the requirement for demonstrating an Equivalent Level of Safety to 10 CFR 71 safety criteria for on site transportation safety as established by DOE Order 460.1, *Packaging and Transportation Safety* and as further specified in HNF-PRO-157, *Radioactive Waste Shipments*, Rev 1A and HNF-PRO-154, *Responsibilities and Procedures*, for all Hazardous Materials, Rev 1.

Specific discussion of risk in Part B, Chapter 3.0 provides additional assurance of the ability to safely use the cask onsite. Based on this review, it is the judgement of the review team that shipments as described in the analyses and conditions identified in Steel Waste Package, SARP document HNF-4763, Rev. 0, present acceptable levels of risk and are approved.

Scope of Review

The SWP SARP has been subject to a previous RL review. The review resulted in approval for onsite use based on the given inventory and design. It is presumed that these reviews adequately verified supporting documentation, calculation methods, and design criteria as part of the review.

The current revision to the SWP SARP, the subject of this review, incorporates changes to the package's inventory. This review does not duplicate previous reviews nor delve into portions of the document that remained unchanged. Therefore, the scope of the review was limited to the possible effects the change in inventory have on containment, criticality, shielding, thermal, and structural analyses. Discussion of the Quality Assurance (QA), Operating Procedures, Acceptance Tests, and Maintenance programs, as they apply to the SWP, were also included in the review due to ongoing changes in the applicable contractor procedures.

Technical Review

The review team assessed the SARP sections, reviewed for understandability of the information as presented, and internal consistency of the document. In addition, there were specific areas of consideration unique to each topic as discussed below. Most observations or comments generated by the review team related to internal consistency of the document, clarity in the presentation of information, and general editorial changes. Changes incorporated due to the review rectified any inconsistencies or errors found.

Material and Fabrication Specifications Review

Part A of the SARP presents the design and specifications for the SWP and its approved payload. The design drawings and descriptions provided appear to meet regulatory requirements and are consistently applied in the analyses in Part B of the SARP. The design drawing presented in the SARP for the SWP are incomplete because key information (e.g. weights) have not been included. Furthermore, key package design specifications and drawings were provided in the SWP Procurement documents and specifications and were not included in the SARP. Other minor inconsistencies identified during the review have all been dispositioned.

Structural Review

Review of Part B, Chapter 7, Structural Evaluation, focused on material properties and stresses, structural performance of the package design under normal and accident conditions, and appropriateness of input parameters to ABAQUS, the computer code used. The review team performed limited manual confirmatory analyses.

The SWP meets the minimum transport requirements for both NCT and accident conditions within the Hanford Site. The structural evaluation of the SWP demonstrated that it will sustain the loads of site wide performance tests for both normal and accident conditions and maintain containment. The review team confirmed accurate inclusion of material properties and allowable stress. Maintaining containment after the puncture test was adequately demonstrated by evaluating the performance of the package by determining the minimum thickness required to prevent puncture. The resulting positive margins of safety show that there is sufficient material thickness to prevent puncture and maintain containment. Incorporation of thermal and pressure stresses from Chapters 8 and 9, and environmental heat loading were also verified.

Thermal Review

Review of Part B, Chapter 8, Thermal Evaluation, focused on specific changes from the previous document. ANSYS general finite element computer program was used for the thermal analysis. Input / model parameters used with the ANSYS analysis included: the Steel Waste Disposal Box (SWDB) and contents, the SWDB Over pack containment with stiffeners, and the SWDB Impact Limiter main structures were modeled. A quarter section model was developed to simplify the thermal model. The bounding decay heat load for the 22-Ton SWDB is assumed to be 1410 Watts.

Under NCT with no solar insolation, the outer surface of the SWDB Impact Limiter temperature reached 50°C (122°F), which is less than the 10CFR71 for exclusive-use shipments (i.e. 85°C (185°F) in the shade. For the structural evaluation, the external surface of the package reaches a temperature of 102°C (215°F) under solar insolation. Internally, the SWDB Over pack seal / flange temperature under solar insolation reaches temperatures up to 146°C (294°F). This temperature is well below the manufacturer's recommended continuous operating temperature limit of 204°C (400°F) for the Viton seal. The maximum SWDB Over pack containment boundary temperature under solar insolation was 151°C (304°F). Consequently, containment will be maintained during NCT.

During the Hypothetical Accident Condition (HAC) fire conditions (i.e. 800°C (1475°F) for thirty minutes, the temperature of the SWDB Over pack seal / flange reaches 123°C (253°F). The maximum temperature for the SWDB Over pack containment boundary during the accident fire was 131°C (267°F). This temperature was well below the manufacturer's recommended continuous maximum operating temperature limit of 204°C (400°F) for Viton. Consequently, during the accident-fire, the seal remains intact and the SWDB over pack containment boundary is maintained. Therefore, containment is maintained.

The review team confirmed that accurate materials, material properties and maximum and minimum temperatures were included in the structural evaluation section.

The review team noted that the minimum temperatures evaluated do not meet criteria identified in 10 CFR 71. -40°C (-40°F). The package's low-temperature performance was evaluated for Hanford on-site ambient minimum temperature limit of -33°C (-27°F). The containment boundary is constructed of austenitic stainless steel (Type 304). This alloy is not subject to low-temperature brittle fracture over the operating limits of the package. The alloy also maintains its ductility over the same range. The SWDB (containment boundary) Over pack double ring seal's working temperature range within the package's temperature operating limits. In accordance with NUREG / CR-1815 Section 5 *Qualification Procedures Associated with Categories of Safety*, a low temperature brittle fracture analysis was not conducted for the 32, 3/4-10 UNC ASTM A325 closure bolts for the Over pack lid because:

1. The bolts were not considered to be fracture critical components,
2. Multiple load paths exist, and
3. Bolted systems are designed to be redundant.

In other words, failure of one or more bolts can be tolerated since failure normally does not lead to penetration or rupture of contents. Consequently, under NCT and accident condition thermal loading, the containment boundary will maintain containment.

However, because the SWDB shielding is constructed from Carbon Steel manufactured to the requirements of ASTM A-36, brittle fracture is a concern in maintaining shielding within the package. Consequently based upon ASME BPVC Section VIII, Division 1 (ASME 1995e), guidelines for low temperature service, handling and transport of the SWDB package will not be permitted for metal temperatures less than 0°C (32°F). Details on the instrument of record, tolerance on the exact reading, and responses if temperatures decrease during normal and delayed shipments will be captured in contractor procedures.

Containment Review

The SWDB is designed and constructed to provide multiple confinement boundaries and a containment boundary during transport. The packaging system consists of a series of two or three confinement boundaries (depending upon the dispersability of the contents), *one within the other, providing multiple boundaries to prevent release of material*. Each confinement boundary is vented through NucFil² filters to prevent hydrogen buildup. These confinement boundaries are sealed within a leak tight containment boundary defined as the SWDB Over pack. The SWDB Over pack is designed and fabricated to meet the leak tight criteria of ANSI N14.5 per Part A, Section 6.5 (ANSI 1997) and maintains containment of the waste payload during NCT and evaluated HAC on the Hanford Site.

A satisfactory fabrication leak rate verification test of the SWP containment boundary will be performed by the manufacturer. The leak test will demonstrate that the SWP meets the 10^{-7} cc/s leak rate defined as leak tight under ANSI N14.5 per Part A, Section 6.5 (ANSI 1997). As evaluated in the SARP, no dose consequence or leak rate analyses were required because the SWP meets the ANSI N14.5 per Part A, Section 6.5 (ANSI 1997) definition of leak tight. Under these conditions, no containment source term was specified. The structural analyses summarized in Part B, Section 7.4, of the SARP adequately demonstrated that the SWP satisfies ASME (1995d) requirements for Service Levels A and C during NCT. Further, the structural analyses summarized in Part B, Section 7.5, of the SARP adequately demonstrated that the SWP satisfies ASME (1995d) requirements for Service Level D for evaluated HAC. Based upon these results the SWP is evaluated through Structural Analysis to be leak tight and that the SWP packaging system will maintain containment for on-site NCT and HACs.

Shielding Review

Review of the SARP's shielding evaluation supports the transport of wastes from the 300 Area to the 200 Area using the SWP. The maximum payload activity evaluated in the SARP shielding evaluation was 500,000 Ci. The shielding limits are 10 mSv/h (1000 mrem/h) at one meter (HAC); and 10 mSv/h (1000 mrem/h) at contact, 0.1 mSv/h (10 mrem/h) at two meters, and 0.02 – 5 mSv/h (2 – 5 mrem/h) at the driver location (NCT). At the driver location, (assumed to be 4 meters from the package), the limit is 0.02 mSv/h (2 mrem/h). If the driver is not a trained radiological worker, and 0.05 mSv/h (5 mrem/h) if the driver is a radiological worker. After a HAC resulting in partial loss of shielding the limit is 10 mSv/h (1000 mrem/h).

MCNP computer based simulations were conducted to determine gamma-ray dose rates at the SWP surface and beyond in both the vertical and horizontal directions. MCNP is a computer based software program that uses sampling methods based on randomly generated numbers (Monte Carlo method) to determine the transport of neutral particles such as gamma rays through shielding materials. Neutron dose rates were estimated using the procedure described by Nelson (1996) and the MCNP code.

The review team verified accuracy of source term and input parameters to MCNP. The SARP evaluated the following types Direct Radiation Sources:

Gamma Source

Beta Source

Neutron Source

The SARP shielding calculations demonstrated that SWP meets both NCT and HAC 10CFR 71 shielding limits for Engineered Container (EC) and Structural "Pieces" (SP) payloads with the following exceptions:

For the EC payload, the computed dose rate of 0.228 mSv/h (22.8 mrem/h at one meter directly below the SWP to ground level is of concern. Access to the region below the SWP will be restricted during NCT by Administrative Control.

For the EC payload, the computed gamma dose rate of 0.129 mSv/h (12.9 mrem/h at two meters above the SWP is not of concern. This area is not normally accessible by either workers or the public. Access to the region above the SWP will be restricted during NCT by Administrative Control.

For the Structural Pieces payload, the dose rates of 0.421 mSv/h (42.1 mrem/h) at two meters from the top, and the dose rate of 0.65 mSv/h (65.0 mrem/h) below the SWP at the one meter location and above ground level are considered to be excessive. Access to these regions above and below the SWP will be restricted during NCT by Administrative Control.

The minimum acceptable Structural Pieces payload mass is 362.8 kg (800 lb)

The HAC shielding analysis for both types of payload showed that 10 CFR 71 criteria were met for all locations one meter from the package for both EC and SP payloads.

Criticality Review

The likelihood of the allowed inventory of fissile material in the SWP payload becoming critical was evaluated as being incredible. The maximum allowable inventory of fissile material as given in Part B, Section 2.0, is 200 gm. This is below the sub critical mass limit for ^{239}Pu when present in an aqueous mixture of 480 gm (ANSI/ANS 8.1 –1983, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors* (ANSI / ANS 1983). This is a factor of 2.4 times greater than the 200 gm of ^{239}Pu allowed in the payload. This implies that 200 gm of ^{239}Pu will be sub critical under any conditions of water moderation and reflection. According to Crow (1996) concrete, plastic and some other common materials can decrease the minimum critical mass of a Plutonium system, but not to less than 300 gm. Of all the fissile isotopes as defined by 49 CFR 173.403, ^{239}Pu has the smallest minimum critical mass. Therefore, the sub criticality of 200 gm of this isotope or a mixture of fissile isotopes is assured, and criticality of the SWP payload is not possible.

Because of the dispersed nature of the Plutonium in the payload volume and because of the considerable separation between adjacent SWPs, if three SWPs are placed in closest proximity to each other, there is no credible way in which the fissile material within the three payloads could combine to form a critical mass.

Operating Procedures Review

Operating procedures as documented in the SARP, in Fluor Project Hanford and 324 Facility Procedures are considered adequate to ensure safe operation of the SWP.

Maintenance Program Review

No issues were identified during the review of the Maintenance section of the SARP and the 324 Facility Maintenance Program. However, the procedures to be utilized for maintenance of the SWP were not provided for RL review. The submittal of the procedures will be made a condition for RL approval.

Quality Assurance Requirements Review

The procurement, fabrication and maintenance Quality Assurance requirements as defined in the SWP SARP and 324 Facility Quality Assurance Program are considered adequate to ensure proper procurement, fabrication and operation of the SWP.

Conditions of Approval

The Review Team has identified the following conditions which are to be implemented by the contractor:

Prior to acceptance, a satisfactory fabrication leak rate verification test of the SWP containment boundary will be performed by the manufacturer. The leak test will demonstrate that the SWP meets the 10^{-7} cc/s leak rate defined as leak tight under ANSI N14.5 per Part A, Section 6.5 (ANSI 1997).

Prior to using a SWBD, the RTV-116 silicon sealant will be inspected to ensure that nowhere is the sealant applied in layers over 3/8 inches thick. The manufacturer's specifications state that RTV applied over 3/8 inches thick will not properly cure and set up.

Because the SWDB shielding is constructed from Carbon Steel manufactured to the requirements of ASTM A-36, brittle fracture is a concern in maintaining shielding within the package. The SWDB shielding may not meet 10CFR71 guidelines for equivalent level of safety for package operation. Consequently the SWDB package will not be used for transportation of hazardous radioactive waste when package metal temperatures are less than 0°C (32°F).

All operating and maintenance procedures involved in the use of this SARP, as well as specification and procurement of spare and replacement parts, will be verified to comply with the requirements of this SARP prior to their use.

Prior to use the SWDB must receive an on-site exception for package acceptance at the Central Waste Complex (CWC). The Hanford on-site SWP as currently approved has a maximum authorized activity of 500,000 Ci. The maximum authorized fissile content is 200 g. of material placed in the SWDB is restricted to dry waste. Total thermal generation is 1410 W, and the maximum Hydrogen generation rate is 1700 cm³/h. The shielding limits are 10 mSv/h (1000 mrem/h) at one meter (HAC); and 10 mSv/h (1000 mrem/h) at contact, 0.1 mSv/h (10 mrem/h) at two meters, and 0.02 – 5 mSv/h (2 – 5 mrem/h) at the driver location (NCT). These transport limits are in excess to the CWC Waste Acceptance Criteria (WAC) of 1 mSv/h (100 mrem / h). at 30 centimeters from the waste package and 2mSv/h (200 mrem/h) at any point on the surface of the package.

For the EC payload, the computed dose rate of 0.228 mSv/h (22.8 mrem/h at one meter directly below the SWP to ground level is of concern. Access to the region

below the SWP will be restricted during NCT by Administrative Control.

For the EC payload, the computed gamma dose rate of 0.129 mSv/h (12.9 mrem/h) at two meters above the SWP is not of concern. This area is not normally accessible by either workers or the public. Access to the region above the SWP will be restricted during NCT by Administrative Control.

For the Structural Pieces payload, the dose rates of 0.421 mSv/h (42.1 mrem/h) at two meters from the top, and the dose rate of 0.65 mSv/h (65.0 mrem/h) below the SWP at the one meter location and above ground level are considered to be excessive. Access to these regions above and below the SWP will be restricted during NCT by Administrative Control.

The minimum acceptable Structural Pieces payload mass is 362.8 kg (800 lb)

All Operating Requirements listed in the SWP SARP, Part B, Sections 6.1 through 6.5 shall be complied with.

All Quality Assurance Program Requirements listed in the SWP SARP, Part B, Sections 7.1 through 7.6 shall be complied with.

All Maintenance Requirements listed in the SWP SARP, Part B, Sections 8.1 through 8.3 and facility maintenance requirements shall be complied with. In addition, 324 Facility maintenance documents covering the SWP shall be provided to ABD for review prior to package use.

Conclusions

This SARP (Onsite) SWP, HNF-4763, Rev. 0, presents an adequate analysis of the safety and risks associated with onsite shipments of remote-handled radioactive waste in the Steel Waste Package from the 324 Building to the 200 Area for interim storage. Based on compliance with the requirements and conditions presented therein, shipments as described are authorized.

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LIST OF TERMS

ALARA	as low a reasonably achievable
ANSI	American National Standards Institute
ARF	airborne release fraction
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BPVC	<i>Boiler and Pressure Vessel Code</i>
Bq	becquerel
Btu/h-in-°F	British thermal units per hour-inch-degree Fahrenheit
Btu/lbm-°F	British thermal units per pound-mass-degree Fahrenheit
Ci	curie
Ci/g	curies per gram
cm	centimeter
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
dpm/cm ²	disintegrations per minute per cubic centimeter
DR	damage ratio
DRM	Documentation and Records Management
EC	Engineered Container
FGE	fissile gram equivalent
ft	foot
g	gram
g/cm ³	grams per cubic centimeter
gal	gallon
HAC	hypothetical accident condition
HRCQ	highway route controlled quantity
Hz	hertz
in.	inch
kg	kilogram
km/h	kilometers per hour
kPa	kilopascal
ksi	1000-lb per square inch
L	liter
lb	pound
lbf/in ²	pound-force per square inch
lbm/ft ³	pound-mass per cubic foot
lbm/in ³	pound-mass per cubic inch
m	meter
MCNP	Monte Carlo N-Particle (compute code)
mi/h	miles per hour
MPa	megapascal
mrem/h	millirem per hour
ms	millisecond
mSv/h	millisievert per hour

LIST OF TERMS, cont

nCi/g	nanocuries per gram
NCT	normal conditions of transport
PNL	Pacific Northwest Laboratory
psi	pounds per square inch
psia	pounds per square inch, absolute
psig	pounds per square inch, gauge
QA	quality assurance
QL	Quality Level
RGC	Rectangular Grout Container
RODC	Rectangular Overpack Disposal Container
SARP	safety analysis report for packaging
SS	stainless steel
std cm ³ /s	standard cubic centimeters per second
SWDB	Steel Waste Disposal Box
SWP	Steel Waste Package
THI	Transportation Hazard Indicator
TRU	transuranic
UNS	United Nations standard
vol%	volume percent
W	watt
W/Ci	watts per curie

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SAFETY ANALYSIS REPORT FOR PACKAGING (ONSITE) STEEL WASTE PACKAGE

PART A: PACKAGE DESCRIPTION AND OPERATIONS

1.0 INTRODUCTION

1.1 GENERAL INFORMATION

A shipping/storage container is required for the shipment of remote-handled radioactive waste from the 324 Building to the 200 Area for interim storage. Previous shipments of similar waste were made using the 22-Ton Steel Waste Package (SWP), which was originally designed by Pacific Northwest Laboratory (PNL) and approved for onsite use by PNL and the U.S. Department of Energy, Richland Operations Office, per PNL-MA-651, *Safety Analysis Report for the Steel Waste Package* (PNL 1988e). The criteria used for the SWP design and this safety analysis report for packaging (SARP) are documented in HNF-3723, *Packaging Design Criteria for the Steel Waste Package* (Boehnke 1999).

The waste (feedstock, glass logs, and dispersibles loaded in Engineered Containers [EC]) in previous shipments were loaded into a Rectangular Grout Container (RGC) and Rectangular Overpack Disposal Container (RODC) for shipment in the 22-Ton SWP.

The waste is expected to contain large quantities of ^{137}Cs and ^{90}Sr that exceed the previously authorized source term for the 22-Ton SWP. Additionally, it is expected to contain small amounts of transuranic (TRU) isotopes with total package contents exceeding the highway route controlled quantity (3,000A₂s) of radioactive material. The maximum allowable radioactive material that will be shipped under this SARP is 500,000 Ci as described in Part A, Section 3.0. The design of the 22-Ton Steel Waste Disposal Box (SWDB) Overpack (drawing E-D-003, *22 Ton Steel Waste Disposal Box Overpack Assembly* [see Part A, Section 10.0]) has been changed to serve as the containment boundary. Additionally, changes were made to the 22-Ton SWDB lifting bails to accommodate the changes made to the 22-Ton SWDB Overpack.

Onsite transport of the package is authorized by this SARP, which meets the requirements of HNF-PRO-154, *Responsibilities and Procedures for all Hazardous Material Shipments*.

1.2 SYSTEM DESCRIPTION

1.2.1 Packaging

The SWP consists of either six or seven components (see Figure A1-1), depending on the type of waste to be loaded, as follows:

- EC (optional for dispersible debris, drawing H-3-308669, *Engineered Container Assembly* [FDNW 1999])
- RGC (drawing H-3-70454, *Rectangular Grout Container Weldment/Sling Assembly* [PNL 1993a])
- RODC (drawing H-3-70453, *Rectangular Overpack Disposal Container/Weldment/Details* [PNL 1992, 1993b])
- 22-Ton SWDB (drawing H-3-56431, *22 Ton Steel Waste Disposal Box Assembly/Details* [PNL 1996c])
- 22-Ton SWDB Overpack Assembly (drawing E-D-003 [see Part A, Section 10.0])
- 22-Ton SWDB Shield Plate (drawing E-D-005, *22 Ton Steel Waste Disposal Box Shield Plate* [see Part A, Section 10.0])
- 22-Ton SWDB Impact Limiter (drawing H-3-56433, *22 Ton Steel Waste Dspl Box Impact Limiter Assy/Det* [PNL 1988b]).

The SWP has a maximum payload capacity of 3860 kg (8500 lb). The fully loaded gross weight of the package is 41,700 kg (92,000 lb).

1.2.1.1 EC. The EC is a tubular container made of 10-in. Schedule 10S (0.165-in. wall thickness) stainless steel pipe with a bolted flange on the top and a plate welded to the bottom. The EC has a NucFil¹ filter on the top. The EC is the primary container for dispersible debris. Typically, eight ECs are loaded as shown in Figure A1-2.

1.2.1.2 RGC. The RGC is a rectangular box made of 11-gauge (0.3cm [0.120-in.-] thick) carbon steel with an open top (no lid). The RGC is the primary container for nondispersible waste. For dispersible waste it is the container used to handle up to eight loaded ECs. No grout is used.

1.2.1.3 RODC. The RODC is a rectangular box made of 0.476 cm (3/16-in.-) thick steel with a sealed lid and NucFil filter for ventilation. The RODC serves as the first confinement boundary for nondispersible waste or as the first or second boundary for dispersible waste.

¹NucFil is a registered trademark of Nuclear Filter Technology, Inc., Lakewood, Colorado.

Figure A1-1. Steel Waste Package Components.

Figure A1-2. Payload Configuration
for Dispersible Debris.

1.2.1.4 22-Ton SWDB. The 22-Ton SWDB is a rectangular box made of steel plates that are bolted together and sealed to provide shielding and confinement for the SWP. The SWDB has a sealed lid with a NucFil filter for ventilation.

1.2.1.5 22-Ton SWDB Overpack. The 22-Ton SWDB Overpack surrounds the SWDB inside the SWDB Impact Limiter and serves as the containment boundary. It is a rectangular box made of 1.3 cm (0.5-in.-) thick stainless steel that is “leaktight” per American National Standards Institute (ANSI) N14.5, *American National Standard for Radioactive Materials—Leakage Tests on Packages for Shipment* (ANSI 1997).

1.2.1.6 22-Ton SWDB Shield Plate. The 22-Ton Shield Plate is installed beneath the 22-Ton SWDB Overpack inside the bottom of the SWDB Impact Limiter. It provides additional shielding.

1.2.1.7 22-Ton SWDB Impact Limiter. Impact resistance for the SWP is provided by the SWDB Impact Limiter consisting of a wood-filled, steel-clad core surrounded by a square steel-tube frame. The SWDB Impact Limiter is split in half (see Figure A1-1), and the two halves can be remotely locked together by locking bars actuated via two electric motors and drive screws (see Figure A1-3). There is an internally mounted NucFil filter with an isolation valve.

1.2.2 Transport

The package will be transported by tractor-trailer as shown in Figure A1-3. The SWP will be transported on a specially modified trailer as specified on drawing H-3-56435, *22 Ton Steel Waste Dspl Box Trailer Mod* (PNL 1988d), and shown in Figure A1-3. The trailer is authorized to carry a net capacity of 41,700 kg (92,000 lb) per WMNW-ENG-99-069, *Memo of Understanding – Capacity of Trailer for Steel Waste Package (HO-64-5261)* and email from R. Cummings to J. Pratt, dated May 19, 2000, *Trailer 64-5261* (see Part A, Section 10.0).

1.2.3 Additional Components

The SWP uses the 22-Ton SWDB lifting yoke (drawing H-3-56434, *22 Ton Steel Waste Disposal Box Lifting Yoke* (PNL 1988c).

1.3 REVIEW AND UPDATE CYCLES

This SARP is subject to periodic reviews and updates. The review shall be performed every 5 years to ensure that all SARP evaluations and other included information meet new or revised regulatory and/or company requirements (HNF-PRO-157, *Radioactive Material/Waste Shipments*).

Figure A1-3. Trailer Mounting for Steel Waste Package.

2.0 PACKAGING SYSTEM

2.1 CONFIGURATION AND DIMENSIONS

The SWP consists of six or seven components depending on the type of waste (refer to figure A1-1). If dispersible waste is to be loaded, the waste may be loaded first into ECs, which are then loaded into the RGC, or it may be loaded directly into the RGC. Nondispersible waste is loaded directly into the RGC. The RGC is then loaded into the RODC. The RODC is then loaded into the SWDB. The SWDB is loaded into the SWDB Overpack, which has already been loaded into the lower Impact Limiter with the SWDB Shield Plate installed and is on the trailer. After the lid is secured on the SWDB Overpack, the upper Impact Limiter is installed and secured to the lower Impact Limiter.

Each container inside the SWDB Overpack is passively vented either through a NucFil filter or through an open top (RGC). The SWDB Overpack is the containment boundary. The SWDB Overpack is sealed during transportation.

Table A2-1. Dimensions of Steel Waste Package.

Component	Length		Inside diam.		Outside diam.		Wall thickness	
	cm	in.	cm	in.	cm	In.	cm	in.
Engineered Container inner cavity	147.3	58.0	26.5	10.42	--	--	0.419	0.165
Engineered Container exterior	167.0	65.75	--	--	27.3	10.75	--	--
Component	Length		Width		Height			
	cm	in.	cm	in.	cm	in.	cm	in.
Rectangular Grout Container inner cavity	170.2	67.0	89.9	35.4	83.8	33.0		
Rectangular Grout Container exterior	175.9	69.3	95.6	37.6	84.6	33.3		
RODC inner cavity	178.4	70.3	98.0	38.6	87.1	34.3		
RODC exterior	184.4	72.6	104.1	41.0	91.8	36.1		
SWDB inner cavity	198.1	78.0	106.7	42.0	94.0	37.0		
SWDB exterior	254.0	100.0	152.4	60.0	129.5	51.0		
SWDB Overpack inner cavity	255.3	100.5	153.7	60.5	131.1	51.6		
SWDB Overpack exterior	276.9	109.0	177.8	70.0	141.7	55.8		
SWDB Shield Plate	259.1	102.0	160.0	63.0	7.6	3.0		
SWDB Impact Limiter inner cavity	278.1	109.5	179.1	70.5	161.0	63.4		
SWDB Impact Limiter exterior	361.7	142.4	261.4	102.9	229.9	90.5		

RODC = Rectangular Overpack Disposal Container.

SWDB = Steel Waste Disposal Box.

2.1.1 EC

The EC is a tubular container made of 10-in. Schedule 10S (0.165-in. thick) stainless steel pipe with a bolted flange on the top and a plate welded to the bottom. The outer dimensions

are 27.3 cm (10.75-in.) diameter by 167 cm (65.75-in.) length. Each EC weighs approximately 68 kg (150 lb). Typically, eight ECs are loaded as shown in Figure A1-3. See drawing H-3-308669 (FDNW 1999) for more details.

2.1.2 RGC

The RGC is a rectangular box made of 11-gauge (0.3 cm [0.12-in.-] thick) carbon steel with an open top (no lid). The outer dimensions are 175.9 cm (69.25 in.) long by 95.6 cm (37.6 in.) wide by 84.6 cm (33.3 in.) high. The RGC weighs approximately 230 kg (500 lb). No grout is used. See drawing H-3-70454 (PNL 1993a) for more details.

2.1.3 RODC

The RODC is a rectangular box made of 0.476 cm (3/16-in.-) thick steel with a sealed lid and NucFil filter for ventilation. The outer dimensions are 184.4 cm (72.6 in.) long by 104.1 cm (41.0 in.) wide by 91.8 cm (36.1 in.) high. The RODC weighs approximately 360 kg (800 lb). See drawing H-3-70453 (PNL 1992, 1993b) for more details.

2.1.4 22-Ton SWDB

The 22-Ton SWDB consists of thick steel plates that are bolted together and sealed to provide shielding for the SWP. The SWDB has a sealed lid with a NucFil filter for ventilation. The outer dimensions are 254.0 cm (100.0 in.) long by 152.4 cm (60.0 in.) wide by 129.5 cm (51.0 in.) high. The side walls are 22.9 cm (9 in.) thick, the bottom is 10.2 cm (4 in.) thick, and the top is 17.8 cm (7 in.) thick. The SWDB weighs approximately 20,150 kg (44,400 lb). See drawing H-3-56431 (PNL 1996c) for more details.

2.1.5 22-Ton SWDB Overpack

The 22-Ton SWDB Overpack surrounds the SWDB inside the Impact Limiter and serves as the containment boundary. It is a rectangular box made of 1.3 cm (0.5-in.-) thick stainless steel that is "leaktight" per ANSI (1997). The outer dimensions are 276.9 cm (109.0 in.) long by 177.8 cm (70.0 in.) wide by 141.7 cm (55.8 in.) high. The SWDB Overpack weighs approximately 3700 kg (8300 lb). See drawing E-D-003 (Part A, Section 10.0) for more details.

2.1.6 22-Ton SWDB Shield Plate

The SWDB Shield Plate is a rectangular carbon steel plate. It is 7.62 cm (3 in.) thick and measures 259.1 cm (102 in.) long by 160 cm (63 in.) wide by 7.6 cm (3 in.) high and weighs approximately 2500 kg (5500 lb). See drawing E-D-005 (Part A, Section 10.0) for more details.

2.1.7 22-Ton SWDB Impact Limiter

Impact resistance for the SWP is provided by an SWDB Impact Limiter consisting of a wood-filled, steel-clad core surrounded by a square steel-tube frame. The Impact Limiter is split in half (see Figure A1-1), and the two halves are remotely locked together by locking bars actuated via two electric motors and drive screws. The flange surface is sealed when locked, and there is an internally mounted NucFil filter with an isolation valve. The outer dimensions are approximately 361.7 cm (142.4 in.) long by 261.4 cm (102.9 in.) wide by 229.9 cm (90.5 in.) high. The SWDB Impact Limiter weighs approximately 10,886 kg (24,000 lb).

2.2 MATERIALS OF CONSTRUCTION

2.2.1 EC

The EC is constructed of 304L stainless steel. The lid is sealed with a Sepco² gasket and has a NucFil filter.

2.2.2 RGC

The RGC is constructed of A-36 carbon steel.

2.2.3 RODC

The RODC is constructed of A-36 carbon steel or stainless steel. The lid is sealed with a silicone rubber gasket and has a NucFil filter.

2.2.4 22-Ton SWDB

The SWDB is constructed of A-36 carbon steel. The joints are sealed with silicone sealant. The lid is sealed with ethylene propylene diene methylene (EPDM) gaskets and has a NucFil filter.

2.2.5 22-Ton SWDB Overpack Assembly

The redesigned SWDB Overpack assembly is the containment boundary of the 22-Ton SWDB transport package. The SWDB Overpack is constructed of Type 304 stainless steel. It consists of a box of 1.3 cm (0.5-in.) thickness, a forged flange, lid and lid bolts, and a Viton³ double O-ring seal. On the outer surfaces, there are a number of stiffeners lined vertically to

²Sepco is a registered trademark of the Sealing Equipment Products Co., Inc., Pelham, Alabama.

³Viton is a registered trademark of the E. I. Du Pont de Nemours & Company, Wilmington, Delaware.

protect the containment boundary from being intruded by the SWDB Impact Limiter guides during the impact.

2.2.5.1 22-Ton SWDB Overpack. The steel used in the SWDB Overpack and lid is Type 304 stainless steel. The stiffeners are constructed with Type 304 stainless steel.

2.2.5.2 SWDB Overpack Lid Bolts. The SWDB Overpack lid bolts are constructed of A-325 high-strength fastener steel.

2.2.5.3 SWDB Shield Plate. The SWDB Shield Plate is made of A-36 carbon steel.

2.2.6 SWDB Impact Limiter

The SWDB Impact Limiter is comprised of two, almost identical, upper and bottom sections. The upper section has additional lifting devices attached. The SWDB Impact Limiter is constructed of 0.64 cm (0.25-in.) steel liners with wood inserts. The outer frame of the SWDB Impact Limiter is constructed of rectangular steel tubes, to which the lifting and closing attachments are welded (drawing H-3-56433 [PNL 1988b]). The SWDB Impact Limiter liners and the bottom SWDB Impact Limiter guides are constructed of A-36 carbon steel. The outer tube frame is made of A-500 Grade B carbon steel. Wood fillers used in the SWDB Impact Limiter are 4 x 8 Douglas fir.

2.3 MECHANICAL PROPERTIES OF MATERIALS

The mechanical properties for the SWP are presented in Part B, Section 7.2.2.

2.4 DESIGN AND FABRICATION METHODS

The SWDB was designed by PNL on behalf of the U.S. Department of Energy (DOE).

The SWDB Overpack and Shield Plate were designed by Waste Management Technical Services for Fluor Hanford.

The SWDB Impact Limiter was designed by PNL on behalf of DOE.

The SWP is designed and constructed to provide multiple confinement boundaries and a containment boundary during transport. Designed as a system to the construction requirements of BPVC, Section VIII, Division 1 (ASME 1995e), the package meets the minimum transport performance requirements for both NCT and accident conditions within the Hanford Site. The packaging system consists of a series of two or three confinement boundaries (depending on dispersibility of the contents), one within the other, providing multiple boundaries to prevent release of material. Each confinement boundary is vented through Nucfil filters to prevent hydrogen buildup. For dispersible contents, there may be three confinement boundaries. From

inside to outside, they are the ECs, the RODC, and the 22-Ton SWDB. The ECs are long tubular members manufactured from stainless steel with a bolted flange on the top and a plate welded to the bottom. In the case of nondispersible contents, the contents are placed in the RGC, which is then placed into the RODC. Up to eight ECs are placed in the open-top RGC, which is then placed in the RODC. Nondispersible debris is placed directly into the RGC. The RODC is a welded steel box equipped with a gasketed closure. The RODC is then placed in the 22-Ton SWDB, which is a thick-walled carbon steel box providing both shielding and confinement. The SWDB is then sealed within the SWDB Overpack, which is the leaktight containment boundary. The SWDB Overpack is constructed of stainless steel and designed as the leaktight containment boundary of the package that meets the NCT and accident condition performance requirements for Hanford Site transport. In transport, the entire assemblage of containers is housed in an SWDB Impact Limiter. This SWDB Impact Limiter is tied to the conveyance and is considered as an integral part of the conveyance.

Major components of the SWDB Overpack containment boundary are constructed from austenitic stainless steel manufactured to the requirements of ASTM A-240, Type 304. As designed, the SWDB Overpack is a rectangular box with a bolted closure. A series of 5 cm (2-in.-) wide stiffeners (which are not part of the containment boundary) are welded to the four vertical sides to stiffen the side panels for pressure loading and to protect the containment boundary during impact. The stiffeners are constructed from stainless steel from ASTM A-240 Type 304 plate bent into a "U" shape with a plate of the same material welded on the backside. These stiffeners are skip welded to the panel, and in the case of the short side, they are skip welded together in pairs. The depth of the stiffeners on the short side is 9.53 cm (3.75 in.) and on the long side is 10.8 cm (4.25 in.). The bottom and side panels of the SWDB Overpack are constructed from 1.3 cm (0.5-in.-) thick plate manufactured to the requirements of ASTM A-240, Type 304 stainless steel. At the top is a 3.18 cm (1.25-in.-) thick, machined flange manufactured from ASTM A-240 or ASTM A-276, Type 304, is welded to the panels. Machined into the flange are two O-ring grooves for the containment seals and a leak test port. The containment boundary seal is provided by the two Viton O-rings. Two 1.9 cm (0.75-in.) round locating pins are provided on the flange for installation of the closure lid. For ease of installation, one locating pin is taller than the other one. Also machined into the flange are 32 through holes for the closure bolts with heavy hex nuts manufactured to ASTM A-194, Grade 8, requirements welded to the underside of the flange. The SWDB Overpack closure lid is a 3.18 cm (1.25-in.-) thick stepped plated constructed from ASTM A-240, Type 304 plate. The closure lid sealing surface landing of the lid is machined to a thickness of 2.87 cm (1.13 in.) with a seal surface finish. The stepped center portion of the plate provides additional shear loading resistance. The closure lid is provided with the two mating locating pin holes for lid installation and 32, 3/4-10 UNC ASTM A-325, closure bolts for lid closure. The lid is also equipped with four swivel hoist rings for lift points. In addition, the lid is equipped with two ports, which are a combination leak test, purge, and NucFil ports. Containment sealing of the ports is provided by the closure bolts and two fluorocarbon Stat-O-Seals⁴.

To protect the containment boundary during NCT and accident conditions, SWDB Impact Limiters are provided. The SWDB Impact Limiters are laminated wood encased in a steel casing that is constructed of welded structural steel members and steel plates. As designed, the SWDB

⁴Stat-O-Seal is a registered trademark of the Franklin Wolfe Company, Inc., Culver City, California.

Impact Limiters are of a split shell design with the bottom SWDB Impact Limiter permanently mounted and tied down to the conveyance. The SWDB Impact Limiters are equipped with a mechanism for locking the two halves together during transport. Operationally, the SWDB Overpack is installed in the SWDB Impact Limiter prior to any loading operations and remains in the SWDB Impact Limiter after unloading. Inside the SWDB Impact Limiter underneath the SWDB Overpack is a 7.62 cm (3-in.-) thick carbon steel plate for shielding. This shielding plate is manufactured from ASTM A-36 carbon steel and is not attached to either the SWDB Impact Limiter or SWDB Overpack.

2.5 WEIGHTS AND CENTER OF GRAVITY

The SWP has a maximum payload capacity of 3860 kg (8500 lb) for the nondispersible payload and 3300 kg (7300 lb) for the dispersible payload. The maximum total weight of the contents of the SWDB is 4500 kg (10,000 lb). The center of gravity is approximately located at the geometric center of the SWDB.

The individual component weights for the nondispersible payload and for dispersibles are shown in Tables A2-2 and A2-3, respectively.

Table A2-2. Component Weights—Nondispersible.

Component	Weight, kg (lb)
Rectangular Grout Container	230 (500)
Rectangular Overpack Disposal Container	360 (800)
Steel Waste Disposal Box (SWDB)	20,150 (44,400)
SWDB Overpack	3700 (8300)
SWDB Shield Plate	2500 (5500)
SWDB Impact Limiter	10,900 (24,000)
Maximum payload	3860 (8500)
Total	41,700 (92,000)

Table A2-3. Component Weights—Dispersibles.

Component	Weight, kg (lb)
Eight Engineered Containers	540 (1200)
Rectangular Grout Container	230 (500)
Rectangular Overpack Disposal Container	360 (800)
Steel Waste Disposal Box (SWDB)	20,150 (44,400)
SWDB Overpack	3700 (8300)
SWDB Shield Plate	2500 (5500)
SWDB Impact Limiter	10,900 (24,000)
Maximum payload	3300 (7300)
Total	41,700 (92,000)

2.6 CONTAINMENT BOUNDARY

The SWDB Overpack provides the containment boundary under the normal conditions of transport and hypothetical accident conditions specified in this SARP. The SWDB Overpack containment boundary consists of the following components:

- SWDB Overpack lid
- SWDB Overpack end, side, and bottom plates
- Weld joints of SWDB Overpack plates
- SWDB Overpack double O-ring seal
- Port covers
- O-ring seals for port covers
- Stat-O-Seals for port plugs.

2.7 CAVITY SIZE

The interior cavity of the RGC is 170.2 cm (67.0 in.) by 89.9 cm (35.4 in.) by 83.8 cm (33.0 in.). For dispersible debris, each EC has an internal cavity of 26.5 cm (10.42 in.) inside diameter by 147.3 cm (58.0 in.) long.

2.8 HEAT DISSIPATION

The SWP has a maximum total allowable decay heat load of 1410 W. There are no special features to dissipate heat. Heat is dissipated passively through the successive containers and surrounding SWDB Impact Limiter.

2.9 SHIELDING

Shielding is provided mainly by the SWDB. Additional shielding is provided by the steel shells of the SWDB Overpack, SWDB Shield Plate, and SWDB Impact Limiter.

2.10 LIFTING DEVICES

The SWP has a specially designed lifting yoke that can be used for lifting either the top half of the SWDB Impact Limiter or the SWDB. The SWDB Impact Limiter and SWDB are each equipped with two permanent lifting attachments that mate with the lifting yoke. The SWDB Overpack is equipped with four swivel hoist rings. The SWDB Overpack can be lifted only when empty. The SWDB Shield Plate has four threaded holes for hoist rings for lifting. During transport the hoist rings are replaced with plugs.

2.11 TIEDOWN DEVICES

The lower half of the SWDB Impact Limiter has five D-rings on each end that connect to turnbuckles to restrain the SWP on the bed of the transporter. The tiedown configuration is shown in Figure A1-2.

3.0 PACKAGE CONTENTS

The following describes the contents authorized by this SARP.

3.1 GENERAL DESCRIPTION

The contents of the SWP consist primarily of contaminated scrap equipment, fine dispersible material, and high-level vault filters removed from 300 Area facilities, such as the B Cell of the 324 Building. Equipment that may be present includes steel racks, tanks, pipes, hoses, valves, tank and pipe heel, and lead plugs from cell pipe penetrations. The dispersible material consists of small particulate solids (dust and powder). Dispersible debris may be loaded into an EC prior to loading into the RGC, or it may be loaded directly into the RGC. The dispersible debris payload configuration is shown in Figure A1-2. The nondispersible debris is loaded directly into the RGC. Nondispersible debris is defined as miscellaneous equipment that may have surface contamination. The high-level vault filters were generated during processing of waste from the 324 Building high-level vault tanks and are loaded directly into the RGC. The radionuclide contents of the SWP are restricted either by the activity limits shown in Table A3-1 or the sum of fractions described in Part A, Section 3.2.2.

3.2 CONTENT RESTRICTIONS

3.2.1 Radionuclide Inventory

Table A3-1 lists the bounding radionuclide inventory for the SWP. Because the inventory contains greater than 3,000 A₂s, it is a Type B highway route controlled quantity per 49 CFR 173, "Shippers—General Requirements for Shipments and Packagings," 173.403, "Definitions." The heat generation rate of this inventory is 1410 W. Because the total TRU activity is greater than 100 nCi/g, this inventory is radioactive TRU waste as defined in 40 CFR 191, "Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes," 191.02, "Definitions." The actual ratio of isotopes may vary provided that either the activity of each isotope is less than that in Table A3-1 or the sum-of-fractions equations in Part A, Section 3.2.2, are satisfied.

The expected maximum inventory contains 161 g of fissile material, as shown in Table A3-2, but may contain up to 200 g of fissile material. Because the quantity of fissile material is greater than 15 g per package, this payload is not fissile excepted per 49 CFR 173.453, "Fissile materials—exceptions." Fissile isotopes as defined by 49 CFR 173.403 include ²³³U, ²³⁵U, ²³⁸Pu, ²³⁹Pu, and ²⁴¹Pu. The actual ratio of fissile isotopes may vary provided that the sum-of-fractions equations in Part A, Section 3.2.2, are satisfied.

Table A3-1. Bounding Radionuclide Inventory of the Steel Waste Package.

Isotope	Relative amount (%)	Activity		Decay heat (W)	A ₂ value (Ci)	Inventory (A ₂)
		(Ci)	(Bq)			
⁶⁰ Co	5.37 E-04	2.68 E+00	9.92 E+10	4.13 E-02	1.08 E+01	2.48 E-01
⁹⁰ Sr	1.77 E+01	8.83 E+04	3.27 E+15	1.02 E+02	2.70 E+00	3.27 E+04
⁹⁰ Y+	1.77 E+01	8.83 E+04	3.27 E+15	4.89 E+02	daughter	daughter
¹³⁷ Cs	3.31 E+01	1.65 E+05	6.12 E+15	1.84 E+02	1.35 E+01	1.22 E+04
^{137m} Ba+	3.13 E+01	1.56 E+05	5.79 E+15	6.15 E+02	daughter	daughter
¹⁵⁴ Eu	2.26 E-02	1.13 E+02	4.18 E+12	1.03 E+00	1.35 E+01	8.37 E+00
²³⁸ Pu*	2.52 E-02	1.26 E+02	4.66 E+12	4.17 E+00	5.41 E-03	2.33 E+04
²³⁹ Pu*	1.91 E-03	9.54 E+00	3.53 E+11	2.97 E-01	5.41 E-03	1.76 E+03
²⁴⁰ Pu*	1.27 E-03	6.36 E+00	2.35 E+11	1.98 E-01	5.41 E-03	1.18 E+03
²⁴¹ Pu	7.87 E-02	3.93 E+02	1.45 E+13	1.25 E-02	2.70 E-01	1.46 E+03
²⁴² Pu*	2.64 E-06	1.32 E-02	4.88 E+08	3.89 E-04	5.41 E-03	2.44 E+00
²⁴¹ Am*	3.50 E-02	1.75 E+02	6.48 E+12	5.85 E+00	5.41 E-03	3.23 E+04
²⁴³ Cm*	7.29 E-04	3.64 E+00	1.35 E+11	1.33 E-01	8.11 E-03	4.49 E+02
²⁴⁴ Cm	5.03 E-02	2.51 E+02	9.30 E+12	8.80 E+00	1.08 E-02	2.33 E+04
Total	100	4.99 E+05	1.85 E+16	1.41 E+03	----	1.29 E+05

+Daughter nuclide. Isotopes contributing less than 1 E-3 Ci or 0.1 A₂ are not shown.

*Transuranic (TRU) isotope.

Table A3-2. Fissile Inventory of the Steel Waste Package.

Isotope	Activity		Mass (g)
	(Ci)	(Bq)	
²³⁸ Pu	1.26 E+2	4.66 E+12	7.41 E+0
²³⁹ Pu	9.54 E+0	3.53 E+11	1.54 E+2
²⁴¹ Pu	3.93 E+02	1.45 E+13	3.93 E+00
Total	5.28 E+02	1.95 E+13	1.65 E+02

The actual ratio of fissile isotopes may vary provided that Equation 2-2 provided in Part B, Section 2.1.2, is satisfied.

3.2.2 Sum of Fractions

An alternate approach to determine an acceptable source term of a particular payload is provided through the use of a sum of fractions equation. This approach is intended to be used when the activities of each isotope of a particular payload are not bounded for every isotope in Tables B2-1 or B2-3, or when unlisted isotopes are present.

Isotopic limits were derived for the isotopes identified in the dispersible materials (package identification numbers PNL-324-96-005 and -012) based on thermal, fissile mass, shielding, total activity, and gas generation constraints. Assuming only a single isotope is present in the payload, the activity limits determined from these constraints are shown in Table A3-3. The activity limit for an individual isotope is the minimum of the activity limits determined by considering each constraint separately. To ensure that the constraints are not exceeded for mixtures of isotopes, two "Sum of Fractions" equations are provided. Equation 3-1 is for nonfissile isotopes only, and Equation 3-2 is for fissile isotopes only. If both types of isotopes are present, then both equations must be satisfied. If the equations are satisfied, then the constraints will not be exceeded.

Table A3-3. Activity Limits L for Use in Sum-of-Fractions Equations 3-1 and 3-2.

Nonfissile isotope	L (Ci)
^{60}Co	3.4 E3
^{90}Sr	2.1 E5
^{137}Cs	2.5 E5
^{154}Eu	1.2 E4
^{234}U	1.2 E4
^{240}Pu	1.1 E4
^{241}Am	1.0 E4
$^{243/244}\text{Cm}$	9.8 E3
Fissile isotope	L (Ci)
^{233}U	1.9 E0
^{235}U	4.4 E-4
^{238}Pu	3.4 E3
^{239}Pu	1.2 E1
^{241}Pu	2.0 E4

Equation 3-1 applies to the nonfissile isotopes only.

$$\sum_i \frac{A_i}{L_i} \leq S \quad \text{Eq. 3-1}$$

where

- i = nonfissile isotope (^{60}Co , ^{90}Sr , ^{137}Cs , ^{154}Eu , ^{234}U , ^{240}Pu , ^{241}Am , and $^{243/244}\text{Cm}$)
- A_i = actual activity of isotope i present in the payload
- L_i = activity limit of isotope i (Table A3-4)
- S = sum of fractions; 0.9 if fissile material is present, 1.0 otherwise.

Equation 3-2 applies to the fissile isotopes only.

$$\sum_j \frac{A_j}{L_j} \leq 1.0 \quad \text{Eq. 3-2}$$

where

- j = fissile isotope (^{233}U , ^{235}U , ^{238}Pu , ^{239}Pu , and ^{241}Pu)
- A_j = actual activity of isotope j present in the payload
- L_j = activity limit of isotope j (Table A3-4)

The contents of the SWP is not limited to the isotopes in Table B2-4. A screening process is provided to eliminate isotopes present in small quantities that do not impact the evaluations in Part B. For isotopes not included in Table B2-4, if the activity of the isotope is less than 10% of its A_2 value, then that isotope may be transported in the SWP without inclusion in the sum of fractions calculation. If the activity of an unlisted isotope is greater than 10% of its A_2 value, then the activity limit of a surrogate isotope that conservatively bounds the radiation emitted by the unlisted isotope may be used. For predominantly gamma-emitting isotopes, the activity limit derived for ^{60}Co may be used. For predominantly beta-emitting isotopes, the activity limit derived for ^{90}Sr may be used. For other non-fissile isotopes, the activity limit for ^{240}Pu may be used. Alternatively, an additional analysis could be performed for particular inventories for which the exact isotopic characterization is known showing that the constraints in this section are satisfied. This analysis shall be performed and documented according to Hanford-Site approved quality assurance procedures.

3.2.3 Radioactive Materials

The maximum authorized activity of the SWP is 500,000 Ci. The maximum authorized fissile content is 200 g. The maximum authorized decay heat generation rate is 1410 W. The activities of the isotopes transported in the SWP are limited such that either the activity of each isotope is below that in the bounding inventory in Table A3-1 or that both Equations 3-1 and 3-2 are satisfied. Table A3-1 does not satisfy Equations 3-1 and 3-2 because the equations are more conservative. Payloads satisfying either Table A3-1 or Equations 3-1 and 3-2 meet all constraints.

No more than eight ECs will be transported in a single SWP. The SWP has a maximum payload capacity of 3860 kg (8500 lb) for the nondispersible payload and 3300 kg (7300 lb) for the dispersible payload. The maximum total weight of the contents of the SWDB is 4500 kg (10,000 lb).

3.2.4 Nonradioactive Materials

The contents of the SWP shall be restricted to dry waste from the 300 Area. Dry is defined as 5 percent or less water and organics by mass. Any containers that have contained liquids shall be completely drained and dried before loading. The hydrogenous content of the SWP is limited to a total of five (5) weight percent (wt%). This limit includes the following materials, but is not limited to, water, plastic, rubber, paper and other wood products, rags and other cotton products, grease and hydraulic fluids, ion exchange resins, and epoxys.

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4.0 TRANSPORT SYSTEM

4.1 TRANSPORTER

The SWP will be transported on a specially modified trailer supplied for exclusive use with the SWP as shown in drawing H-3-56435 (PNL 1988d). The modifications consist of the installation of tiedown points and blocking assemblies on the transporter bed. Immediately before shipment, the trailer shall be visually inspected for damage or deficiency(ies) that may affect the safety of the shipment. Any damage or deficiency(ies) that may affect the safety of the shipment shall be corrected before transport. The trailer is authorized to carry a net capacity of 41,700 kg (92,000 lb) per WMNW-ENG-99-069, *Memo of Understanding – Capacity of Trailer for Steel Waste Package (HO-64-5261)* and email from R. Cummings to J. Pratt, dated May 19, 2000, *Trailer 64-5261* (see Part A, Section 10.0).

4.2 TIEDOWN SYSTEM

The tiedown system for the SWP consists of structural carbon steel angles that are permanently welded to the bed of the transporter to provide lateral and longitudinal bracing. In addition, five turnbuckles on each end of the package provide vertical and additional longitudinal restraint (see Figures A2-2 and A2-3). Part A, Section 6.2, describes the tiedown configuration. Evaluation of the tiedown system is in Part B, Section 10.0.

4.3 SPECIAL TRANSFER REQUIREMENTS

4.3.1 General

Transport of the SWP shall meet the requirements of HNF-PRO-154 and HNF-PRO-157 as well as the following special transfer requirements.

4.3.2 Routing and Access Control

The SWP is authorized for onsite exclusive-use transfer only on the Hanford Site by tractor-trailer. Road closure to prevent public access to the shipment is required south of the Wye Barricade.

Transport of the package will be from the 300 Area to the 200 West Area Central Waste Complex. There are two transportation routes available. The first route is from Building 324 through the west gate of the 300 Area, north on Route 4 past the 200 East Area, west on Route 3 to the 200 West Area, and through the 200 West Area to the Central Waste Complex. The second route is from Building 324 through the west gate of the 300 Area, north on Route 4 to the

Wye Barricade, north on Route 2, west on Route 11A to the 200 West Area, and through the 200 West Area to the Central Waste Complex.

4.3.3 Radiological Limitations

The radiation dose on the external surface of the SWDB Impact Limiter shall not exceed 10 mSv/h (1000 mrem/h). The radiation dose at 2 m from any point on the SWP shall not exceed 0.1 mSv/h (10 mrem/h). Although not subject to the criteria, shielding analysis (Part B, Section 5.0) indicates that the axial dose rate (SWP top and bottom) may exceed the limit of 0.1 mSv/h (10 mrem/h) at 2 m and at the road surface, respectively.

At the driver location the dose rate shall not exceed 0.02 mSv/h (2 mrem/h) if the driver is not a trained radiological worker or 0.05 mSv/h (5 mrem/h) if the driver is a radiological worker.

Contamination on the exterior surfaces of the SWDB Impact Limiter shall not exceed 22 dpm/cm² for beta/gamma-emitting radionuclides and 2.2 dpm/cm² for alpha-emitting radionuclides as measured per 49 CFR 173.443, "Contamination control," 173.443(a).

4.3.4 Time Restrictions

As evaluated in Part B, Chapter 9.0, shipping time limits are imposed to ensure that shipment takes place in half the time required to reach a hydrogen concentration of 5 percent in the SWP. From the time that the SWDB Overpack is loaded and sealed, the SWP must be transported to its destination and vented within the time shown in Table A4-1 and Figure A4-1. The allowable shipping time is a function of the source term.

Table A4-1. Steel Waste Package Shipping Window.

Source term (Ci)	Shipping window (hours)
500,000	11
400,000	14
300,000	19
200,000	28
100,000	56
90,000	63
80,000	71
70,000	81
60,000	94
50,000	113
40,000	141
30,000	188
20,000	282
10,000	564
5,000	1129

Figure A4-1. Steel Waste Package Shipping Window.

4.3.5 Speed Limitations

The maximum speed of the tractor-trailer transporting the package shall be limited to 64 km/h (40 mi/h) on the straightaway.

4.3.6 Environmental Conditions

Transport shall not take place under icy conditions; when the temperature is at or below 0 °C (32 °F) or during periods of expected heavy traffic, dense fog, or adverse road conditions (such as snow or ice) as determined by the shipper.

4.3.7 Frequency of Use and Mileage Limitations

There are no frequency or mileage limitations.

4.3.8 Escort and Emergency Response

Escorts are not required for the SWP. A Hanford Site emergency response plan shall be in place before shipment occurs.

4.3.8 Escort and Emergency Response

Escorts are not required for the SWP. A Hanford Site emergency response plan shall be in place before shipment occurs.

4.3.9 Exclusive Use

During the transfer of the SWP with related equipment, the transport trailer shall not carry any other packagings containing hazardous material.

4.3.10 Driver Training Requirements

The driver of the vehicle shall meet the Site requirements for a heavy-duty driver, be qualified as a radiation worker, and have specific training for transporting highway route controlled quantities of radioactive material.

5.0 ACCEPTANCE OF PACKAGE FOR USE

Pre-use inspections shall be performed before each use of a SWP.

1. The user shall ensure that the packaging is in unimpaired physical condition before each use. No obvious, visually detectable flaws (e.g., broken welds, cracks, punctures, lid distortions) shall be present.
12. The user shall verify that:
 - a. All closure bolts are in place and in good condition with no obvious defects
 - b. Bolts that exhibit headmarks matching those on the DOE-Headquarters Suspect Fastener Headmark List are removed, segregated, reported, documented, and dispositioned in accordance with HNF-PRO-301, *Control of Suspect/Counterfeit Items*.
3. The user shall inspect the SWDB gasket and SWDB Overpack O-rings to ensure that they are in good condition with no obvious, visually detectable defects.

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6.0 OPERATING REQUIREMENTS

6.1 GENERAL REQUIREMENTS

1. The approved SARP shall be the controlling document for the shipment of 324 Facility material as defined in Part A, Section 3.0.
2. All applicable instructions and procedures for onsite shipment of radioactive material shall comply with HNF-PRO-157.
3. Operational controls presented in the SARP shall take priority over similar requirements presented in other Fluor Project Hanford manuals except where other Fluor Project Hanford requirements are more restrictive.
4. Written operating procedures shall be followed when packaging and transporting payloads in the SWP. These procedures shall provide guidance to ensure that the container is being used in accordance with the SARP.
5. Application of paint, plastic film, or similar surface coatings to cover surface contamination on the exterior surface of the SWDB Impact Limiter is not permitted.
6. Operating procedures shall require that the transport of the SWP be done in accordance with the "special transport requirements" noted in Part A, Section 4.0.
7. The SWDB Overpack shall not remain sealed for more than the allowed shipping window as shown in Table A4-1.
8. As low as reasonably achievable (ALARA) principles shall be followed during loading and unloading.
9. If less than eight ECs are to be transported, empty ECs, other dunnage, or other engineering-approved packing material (that stays within the safety analysis bounds provided in Part B) shall be placed inside the RGC, as necessary, to ensure contents do not shift during transport.
10. The SWP (including tiedown system) shall not be altered or revised without the concurrence of Packaging Engineering.
11. Under this SARP, the loaded SWP shall only be shipped onsite. The highway must include barricading to prevent public access (closed road routing) to the transport system south of the Wye Barricade. This is performed through the designated shipping organization.
12. Place tamper-indicating seals on the SWP prior to release for transportation.

6.2 TIEDOWN

The tiedowns shall be as shown in Figure A1-2. The tiedown attachments on the lower half of the SWDB Impact Limiter shall be visually inspected to ensure that there are no obvious, visually detectable defects. In addition, the turnbuckles shall be checked prior to every shipment for proper tension (5-7 ft-lb per PNL [1988d]).

6.3 RADIOLOGICAL LIMITATIONS

The radiation dose on the external surface of the SWP Impact Limiter shall not exceed 10 mSv/h (1000 mrem/h). The radiation dose at 2 m from any point on the SWP shall not exceed 0.1 mSv/h (10 mrem/h). Although not subject to the criteria, shielding analysis (Part B, Section 5.0) indicates that the axial dose rate (SWP top and bottom) may exceed the limit of 0.1 mSv/h (10 mrem/h) at 2 m and at the road surface, respectively.

At the driver location the dose rate shall not exceed 0.02 mSv/h (2 mrem/h) if the driver is not a trained radiological worker or 0.05 mSv/h (5 mrem/h) if the driver is a radiological worker.

Removable contamination on the external surfaces of the impact limiter shall be less than 22 dpm/cm² for beta/gamma-emitting radionuclides plus others and less than 2.2 dpm/cm² for all other alpha-emitting radionuclides per 49 CFR 173.443(a).

6.4 EMPTY PACKAGE (PACKAGING)

Preparation of the empty package for transportation shall consist of the following activities.

1. Inspect the interior of the SWDB Overpack to ensure there is no water or loose articles present.
2. Ensure that the surface dose level of the SWDB Impact Limiter is equal to or less than 0.005 mSv/h (0.5 mrem/h).
3. Perform a radiation survey on the SWP and trailer system to ensure that the radiation levels and contamination are below the specified limits. Contamination on all exterior surfaces of the SWDB Impact Limiter shall not exceed 22 dpm/cm² for beta/gamma-emitting radionuclides and 2.2 dpm/cm² for alpha-emitting radionuclides as measured per 49 CFR 173.443(a).
4. Ensure that the SWP is in unimpaired condition and securely closed to prevent leakage of radioactive material under normal conditions of transport.

6.5 LEAK TEST REQUIREMENTS

The SWDB Overpack shall be leak tested according to Table A6-1, per ANSI N14.5 (ANSI 1997).

Table A6-1. SWDB Overpack Leak Test Requirements.

Test	Frequency	Components	Criteria
Maintenance	After maintenance repair (such as weld repair), or replacement of components of the containment system	That portion of the containment system affected by the maintenance, repair, or component replacement	$1 \times 10^{-7} \text{ ref}\cdot\text{cm}^3/\text{s}$
Periodic	Within 12 months prior to each shipment	SWDB Overpack Assembly	$1 \times 10^{-7} \text{ ref}\cdot\text{cm}^3/\text{s}$
Preshipment	Before each shipment, after the contents are loaded and the package is closed	Seals that have been opened	$1 \times 10^{-3} \text{ ref}\cdot\text{cm}^3/\text{s}$

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7.0 QUALITY ASSURANCE REQUIREMENTS

7.1 INTRODUCTION

These sections describe the quality assurance (QA) requirements for the procurement, fabrication, and maintenance of the 22-Ton SWP. The requirements are taken from the following documents:

- HNF-MP-599, "*Project Hanford Quality Assurance Program Description*" (as implemented via Fluor Project Hanford procedures)
- 49 CFR 173.

7.2 GENERAL REQUIREMENTS

These requirements apply to activities, which include loading, unloading, and transportation operations, that could affect the quality of the packaging and associated hardware. The overall 22-Ton SWP assembly, Overpack, Impact Limiter, and specific components of the 22-Ton SWP assembly are assigned a Transportation Hazard Indicator (THI) and subordinate Quality Levels (QL) in accordance with HNF-PRO-154. The THI and QLs are used to show that each individual component meets the requirements of the overall assembly safety classification.

7.3 THI AND QL'S

The 22-Ton SWP is assigned a THI of 1 based upon criteria established in HNF-PRO-154.

SARP approval requirements are controlled by the THI assigned.

The alphanumeric QL defines the significance to safety of each component of a packaging system. QLs are defined as follows.

QL-A: Critical impact on safety and associated functional requirements: items or components whose single failure or malfunction could directly result in an unacceptable condition of containment, shielding, or nuclear criticality control. All applicable QA elements of HNF-MP-599 will apply to these activities as determined by the cognizant Quality Assurance engineer.

QL-B: Impact on safety and associated functional requirement: components whose failure or malfunction in conjunction with one other independent failure or malfunction could result in an unacceptable condition of containment, shielding, or nuclear criticality control. All applicable QA elements of HNF-MP-599 will apply to these activities as determined by the cognizant Quality Assurance engineer.

QL-C: Minor impact on safety and associated functional requirements: components whose failure or malfunction would not reduce packaging effectiveness and would not result in an unacceptable condition of containment, shielding, or nuclear criticality control regardless of other single failures. Selected application of any or all QA elements of HNF-MP-599 will be applied as designated by the package design authority to verify critical characteristics or functions.

7.4 ORGANIZATION

The organizational structure and the assignment of responsibility shall be such that quality is achieved and maintained by those who have been assigned responsibility for performing work. Quality achievement is to be verified by persons or organizations not directly responsible for performing the work.

- Follow current requirements of this SARP, HNF-MP-599, and applicable Fluor Project Hanford procedures.
- Provide instructions for implementing QA requirements.
- Quality Assurance organizations are responsible for establishing and administering the QA program as stated in HNF-PRO-154 and HNF-MP-599.

7.5 QA PLANNING AND ACTIVITIES

Each individual involved with the use or maintenance of the 22-Ton SWP is responsible for ensuring that the assigned tasks are performed in accordance with controlling plans and procedures, which must, in turn, conform to the requirements of this SARP. QA requirements for tasks are determined and documented in the plans and procedures used by the involved organizations.

Requirements are imposed on organizations by direct reference in plans and procedures. Table A7-1, "Functional Responsibility Matrix," shows documentation review requirements. This matrix assigns responsibility for review and approval to the appropriate organization either designing or using the 22-Ton SWP.

Table A7-1. Functional Responsibility Matrix.

Document	Packaging Engineering	User
Safety analysis report for packaging	P, R, A	R, A
Drawings/design changes	R, A	P, R, A
Purchase Order(s)	R, A	P, R, A
Operating procedures	R, A	P, R, A
Maintenance procedures	R, A	P, R, A
Nonconformance Reports	R, A	P, R, A

R = Review.

A = Approval.

P = Primary responsibility (organization responsible for developing the document).

7.5.1 Design Control

Fluor Hanford 324 Facility is the design authority for the package. Design Control shall be in accordance with HNF-PRO-244, *Engineering Data Transmittal Requirements*, HNF-PRO-440, *Engineering Document Change Control Requirements*, and HNF-PRO-1819, *PHMC Engineering Requirements*.

QLs for design control of 22-Ton SWP components are shown in Table A7-2. Classification of components into these quality categories is in accordance with HNF-PRO-154 and ensures that critical parameters of a given component are reviewed in a manner consistent with their importance. The design and analysis shall correspond to the QA requirements of the assigned QL.

Table A7-2. Design Quality Levels.

Component	Quality Level
Fasteners	A1
O-rings	A1
Swivel hoist ring	A1

7.5.1.1 Design Inputs. Design inputs for the 22-Ton SWP are taken from the component drawings as presented in this SARP. Any changes to these drawings shall be identified, documented, approved, and controlled in the same manner in which the drawings were originally released. A review shall be conducted to ensure that the design change meets the requirements of this SARP in accordance with applicable Hanford Site procedures.

7.5.1.2 Design Process. Cognizant personnel shall ensure that the 22-Ton SWP components meet the design requirements of the governing design specification.

7.5.1.3 Design Verification. Design adequacy of the 22-Ton SWP is demonstrated by this SARP.

In the event of design changes to the 22-Ton SWP, the cognizant engineering organization shall conduct a design review in accordance with HNF-PRO-445, *Design Verification Requirements*, as required by HNF-PRO-552, *Project Control*. Any changes to the final design of the 22-Ton SWP will require revision of this SARP.

7.5.2 Procurement and Fabrication Control

The design authority responsible for the 22-ton Steel Waste Package design initiates procurement documentation for 22-ton Steel Waste Package items. The standard purchase requisition shall contain both the technical and QA requirements. Specifications shall be written for complicated or critical component procurement items. The purchase requisition information is then converted to a purchase order by the purchasing organization. The cognizant Quality Assurance engineer or Quality Assurance manager reviews and approves all purchase requisitions and specifications per HNF-PRO-268, *Control of Purchased Items and Services*.

Changes to the purchase requisition or subsequent purchase order are subject to the same review and approval requirements as the original purchase requisition. Specifications, engineering drawings, or other QA documents shall impose QA requirements.

QLs for procurement of consumable components are shown in Table A7-2. Classification of components into these quality categories are in accordance with HNF-PRO-154 and ensures critical parameters of a given component are reviewed in a manner consistent with their importance. The component design and analysis shall correspond to the QA requirements of the assigned QL. Items procured or fabricated shall have Quality Control inspection prior to use for compliance with the purchase order, specification, and/or fabrication drawing. The cognizant engineer and cognizant Quality Assurance engineer shall define the acceptance criteria.

7.5.3 Control of Purchased Items and Services

The procurement of 22-Ton SWP items and services shall be documented and controlled to ensure conformance to specified requirements of HNF-PRO-268.

7.5.4 Control of Process

Instructions, procedures, drawings, checklists, or other appropriate means shall control processes affecting the safety basis or quality of the 22-Ton SWP. These means shall ensure that process parameters are controlled within defined limits and that specified controls will be applied as defined in HNF-MP-599.

Special processes performed onsite and by suppliers that control or verify quality, such as those used in welding, heat treating, and nondestructive examination of containment boundary items, shall be performed by qualified personnel using applicable approved procedures.

Records shall be maintained in accordance with HNF-PRO-222, *Quality Assurance Records*, for currently qualified personnel, processes, and equipment for each special process (if a procured item) or by the shop performing the process.

7.5.5 Inspection

In-process and final inspections shall be performed to the following guidelines. These inspections shall follow procedures written by the user and shall be documented on a traveler, quality inspection plan, or any other document written and approved by Quality Assurance and Engineering.

7.5.5.1 Inspection Personnel. Cognizant Quality Assurance personnel qualified in accordance with HNF-PRO-263, *Qualification and Certification of Inspection and Test Personnel*, shall perform all required inspections.

7.5.5.2 In-Process Inspection. In-process inspections for the 22-Ton SWP include (1) proper assembly and disassembly of the 22-Ton SWP and (2) leak testing of the SWDB and SWDB Overpack. Inspections shall be performed in accordance with HNF-PRO-283, *Control of Inspections*.

7.5.5.3 Shipment Inspections and Checks. The shipping procedures are established, as appropriate, for tests to be conducted by the storage facility to ensure the 22-Ton SWP maintains containment and is free of excessive contamination. The procedures also establish qualification criteria for responsible personnel who document and evaluate test results. Table A7-3 shows the package attributes that must be inspected or checked.

Table A7-3. Package Inspections and Checks.

Items 1 through 4 shall be verified:	
1	Steel Waste Package (SWP) closure for deterioration (visual inspection).
2	SWP contamination levels.
3	SWP external dose rates.
4	SWP markings for deterioration (visual inspection).

7.5.5.4 Final Inspection. Loading and unloading procedures shall be written by the user with appropriate reference to leak test procedures and maintenance procedures and shall be used to ensure adequate loading, operation, and maintenance of the packaging. The loading and unloading procedures shall identify actions required by loading personnel to safely and properly

load the 22-Ton SWP and Overpack in accordance with this SARP. Approved procedures and QA inspection plans shall be established to ensure that final inspection verifies compliance with the following items prior to each shipment.

- 22-Ton SWDB is properly assembled.
- 22-Ton SWDB Overpack is properly assembled.
- 22-Ton SWDB Impact Limiter is properly assembled.
- 22-Ton SWDB Overpack is leak tested, and testing satisfies the acceptance criteria.
- All acceptance criteria are met for the use of the new packaging.
- All shipping papers are properly completed.
- Packages are conspicuously and durably marked as required by DOT regulations.

Measures are established to ensure that an individual trained in DOT shipping requirements, designated by the user of the package, signs the shipping papers before authorization for shipping.

Operational and maintenance procedures are properly completed.

A checklist (see Table A7-4) is provided for the 22-Ton SWP operational procedures for which inspection and Quality Assurance overviews are required.

Table A7-4. Required Inspections and Verifications.

1	Visually inspect Steel Waste Disposal Box (SWDB) for damage.
2	Visually inspect SWDB Overpack for damage
3	Visually inspect SWDB Impact Limiter for damage
4	Visually inspect gaskets for damage.
5	Visually inspect sealing surfaces for damage.
6	Visually inspect nuts and bolts for damage.
7	Visually inspect tiedown components for loose connections or damage.
8	Visually inspect interior of Rectangular Grout Container to ensure there is no liquid or loose articles.
9	Verify acceptable leak test of SWDB Overpack.
10	Verify that contamination is within acceptance criteria.
11	Verify that dose rates are within acceptance criteria limits.
12	Verify storage location is acceptable.

7.5.6 Test Control

Test control will be in accordance with HNF-PRO-286, *Test Control*. See Part A, Section 7.5.5.2, for in-process inspections and checks that ensure the design has not been compromised.

7.5.7 Control of Measuring and Test Equipment

The requirements for measuring and test equipment are contained in HNF-PRO-490, *Control of Measuring and Test Equipment and Nondata Test Equipment*, and apply to all equipment used to determine acceptability of the 22-Ton SWP regardless of the assigned QL.

7.5.8 Handling, Shipping, and Storage

Handling, shipping and storage of the SWP will be in accordance with requirements in HNF-MP-599.

7.5.9 Inspection and Test Status

After completion of any required Quality Assurance/Quality Control inspections the 22-Ton SWP shall have the proper status tag(s) applied as required in accordance with HNF-PRO-297, *Inspection, Test, and Operating Status*.

7.5.10 Control of Nonconforming Items

Identification, documentation, evaluation and disposition of nonconforming items and activities shall be accomplished in accordance with HNF-PRO-298, *Nonconforming Item Reporting and Control*.

7.5.11 Corrective Action

Nonconforming conditions or conditions adverse to quality will be evaluated as described in section 7.4.6, and the need for corrective action is determined in accordance with HNF-PRO-052, *Corrective Action Management*.

7.5.12 Records and Document Control

Requirements are imposed on organizations by direct reference in plans and procedures. Table A7-1 shows documentation review requirements. This matrix assigns responsibility for document development, review, and approval to the appropriate organization for use of the 22-Ton SWP.

Records that furnish documentary evidence of quality shall be specified, prepared, and maintained in accordance with HNF-PRO-222. All documents used to perform and/or verify quality-related activities shall be controlled. Controlled documents include (but are not limited to) drawings, specifications, purchase orders, plans, inspection and testing procedures, reports, quality verification reports, nonconformance and corrective action reports, this SARP, and

operational and maintenance procedures. The Document Control System embodies the following features.

- Document changes are controlled in the same way as the original issue.
- Interfacing documents are properly coordinated and controlled.
- A reference system is in use that provides access to the current issues of project documents.

All QA records associated with the 22-Ton SWP shall be retained for the life of the packaging or 5 years, whichever is longer. All lifetime storage QA records required for the 22-Ton SWP shall be stored with the user depending on the purpose of the document. Table A7-5 shows QA record retention periods and record locations for the 22-Ton SWP.

Table A7-5. Retention and Location.		
Document	Retention period	Location
Safety analysis report for packaging	Lifetime	DRM
Quality control inspection reports	Lifetime	User/DRM
Nonconformance reports	Lifetime	User/DRM
Component procurement record and specifications	Lifetime	User/DRM

DRM = Documentation and Records Management.

7.5.13 Audits

Internal and external audit process shall be in accordance with HNF-MP-599.

7.6 SARP CONTROL SYSTEM

This SARP is a copy controlled supporting document to ensure that only up-to-date approved versions are used for transport. Any changes made to this SARP will be by engineering change notices, which are distributed through the Copy Control System and incorporated into this SARP.

Any review comment review records generated during the initial release or subsequent changes will be on file with Waste Management Technical Services, Engineering Services.

8.0 MAINTENANCE

8.1 GENERAL REQUIREMENTS

The SWP is a passive system designed to require minimal maintenance during its use. Maintenance of the SWP during use is limited to a visual inspection and a leak test. No changing of parts is necessary between uses unless degradation is noted or component service lives are exceeded.

8.2 INSPECTION AND VERIFICATION SCHEDULES

The following items shall be inspected or tested and appropriately maintained on the SWP prior to each use.

- Visually inspect the SWP and its components (nuts, studs, lift ring attachments, tiedown devices, gaskets, and O-rings) for damage and to ensure proper operating condition.
- Inspect and leak test the SWDB Overpack per Part A, Section 6.5.

The following items shall be performed annually.

- Ensure the trailer complies with applicable Hanford Site/DOT requirements.
- Replace the SWDB Overpack O-rings.
- Leak test the SWDB Overpack per ANSI N14.5 per Part A, Section 6.5 (ANSI 1997)

8.3 RECORDS AND DOCUMENTATION

See Part A, Section 7.5.12, for QA requirements for document retention.

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9.0 REFERENCES

- 40 CFR 191, "Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes, *Code of Federal Regulations*, as amended.
- 49 CFR 173, "Shippers—General Requirements for Shipments and Packagings," *Code of Federal Regulations*, as amended.
- HNF-MP-599, *Project Hanford Quality Assurance Program Description*, Fluor Hanford, Richland, Washington.
- HNF-PRO-052, *Corrective Action Management*, Fluor Hanford, Richland, Washington.
- HNF-PRO-154, *Responsibilities and Procedures for all Hazardous Material Shipments*, Fluor Hanford, Richland, Washington.
- HNF-PRO-157, *Radioactive Material/Waste Shipments*, Fluor Hanford, Richland, Washington.
- HNF-PRO-1819, *PHMC Engineering Requirements*, Fluor Hanford, Richland, Washington.
- HNF-PRO-222, *Quality Assurance Records Standards*, Fluor Hanford, Richland, Washington.
- HNF-PRO-244, *Engineering Data Transmittal Requirements*, Fluor Hanford, Richland, Washington.
- HNF-PRO-263, *Qualification and Certification of Inspection and Test Personnel*, Fluor Hanford, Richland, Washington.
- HNF-PRO-268, *Control of Purchased Items and Services*, Fluor Hanford, Richland, Washington.
- HNF-PRO-283, *Control of Inspections*, Fluor Hanford, Richland, Washington.
- HNF-PRO-286, *Test Control*, Fluor Hanford, Richland, Washington.
- HNF-PRO-297, *Inspection, Test, and Operating Status*, Fluor Hanford, Richland, Washington.
- HNF-PRO-298, *Nonconforming Item Reporting and Control*, Fluor Hanford, Richland, Washington.
- HNF-PRO-301, *Control of Suspect/Counterfeit Items*, Fluor Hanford, Richland, Washington.
- HNF-PRO-440, *Engineering Document Change Control Requirements*, Fluor Hanford, Richland, Washington.

HNF-PRO-445, *Design Verification Requirements*, Fluor Hanford, Richland, Washington.

HNF-PRO-490, *Control of Measuring and Test Equipment and Nondata Test Equipment*, Fluor Hanford, Richland, Washington.

HNF-PRO-552, *Project Control*, Fluor Hanford, Richland, Washington.

ANSI, 1997, *American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment*, ANSI N14.5, American National Standards Institute, New York, New York.

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

Boehnke, W. M., 1999, *Packaging Design Criteria for the Steel Waste Package*, HNF-3723, Rev. 0, prepared by Waste Management Federal Services, Inc., Northwest Operations for Fluor Daniel Hanford, Inc., Richland, Washington.

FDNW, 1999, *Engineered Container Assembly*, drawing H-3-308669, Rev. 0 (2 sheets), Fluor Daniel Northwest, Inc., Richland, Washington.

PNL, 1996c, *22 Ton Steel Waste Disposal Box Assembly/Details*, drawing H-3-56431, Rev. 1 (4 sheets), Pacific Northwest Laboratory, Richland, Washington.
Sheets 1 and 2, *22 Ton Steel Waste Disposal Box Assembly*
Sheets 3 and 4, *22 Ton Steel Waste Disposal Box Details*

PNL, 1993a, *Rectangular Grout Container Weldment/Sling Assembly*, drawing H-3-70454, Rev. 1 (2 sheets [plus Engineering Change Notice 656412]), Pacific Northwest Laboratory, Richland, Washington.
Sheet 1, *Rectangular Grout Container Weldment*
Sheet 2, *Rectangular Grout Container Sling Assembly*

PNL, *Rectangular Overpack Disposal Container/Weldment/Details*, drawing H-3-70453 (plus Engineering Change Notice 656412), Pacific Northwest Laboratory, Richland, Washington.
1992, Sheet 1, *Rectangular Overpack Disposal Container*, Rev. 0
1993b, Sheet 2, *Rectangular Overpack Disposal Container Weldment*, Rev. 1
Sheet 3, *Rectangular Overpack Disposal Container Details*, Rev. 1

PNL, 1988b, *22 Ton Steel Waste Dspl Box Impact Limiter Assy/Det*, drawing H-3-56433, Rev. 0 (9 sheets), Pacific Northwest Laboratory, Richland, Washington.
Sheet 1, *22 Ton Steel Waste Dspl Box Impact Limiter Assy*
Sheets 2 thru 9, *22 Ton Steel Waste Dspl Box Impact Limiter Det*

PNL, 1988c, *22 Ton Steel Waste Disposal Box Lifting Yoke*, drawing H-3-56434, Rev. 2, Pacific Northwest Laboratory, Richland, Washington.

PNL, 1988d, *22 Ton Steel Waste Dspl Box Trailer Mod*, H-3-56435, Rev. 0 (6 sheets), Pacific Northwest Laboratory, Richland, Washington.

PNL, 1988e, *Safety Analysis Report for the Steel Waste Package*, PNL-MA-651, Rev. 1, Pacific Northwest Laboratory, Richland, Washington.

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10.0 APPENDICES

10.1 MEMORANDUMS REGARDING STEEL WASTE PACKAGE TRAILER

Bogart, Rex L

From: Cummings, Robert J (Bob)
Sent: Friday, May 19, 2000 6:27 AM
To: Pratt, Jeffery (Jeff)
Cc: Bogart, Rex L; Boehnke, William M; Faubert, C L (Chuck)
Subject: Trailer 64-5261

Jeff,

Per our conversation on 5/18/00, this is to confirm that DynCorp fleet maintenance mechanics have inspected trailer # 64-5261, and the tires **do not** require replacement in order to safely operate the trailer. This is a change to the requirements that were listed in the memo of understanding WMNW-ENG-99-069, dated 6/16/99, that stated that tire replacement was recommended. Travel speed of 40 mph max. should still be observed.

If you have any questions, feel free to call me. 376-6537

Bob Cummings,
DynCorp Fleet Maint. Engineer

**WASTE MANAGEMENT FEDERAL SERVICES
NORTHWEST OPERATIONS**
A WASTE MANAGEMENT COMPANY

345 Hill Street
Richland, WA 99352
(509) 376-3035
(509) 372-1435 Fax

June 16, 1999

WMNW-ENG-99-069

To File

**MEMO OF UNDERSTANDING - CAPACITY OF TRAILER FOR STEEL WASTE
PACKAGE (HO-64-5261)**

This letter will serve as confirmation of the meeting held on June 3, 1999 between R. J. Cummings of DynCorp Tri-Cities Services and W. M. Boehnke, Waste Management Federal Services, Inc., Northwest Operations. During the meeting, it was agreed that the subject trailer in its current configuration is suitable to safely carry a net capacity of 92,000 lb at a maximum speed of 40 mph. The net capacity accounts for the weight of the modifications welded to the trailer. Tires must be replaced prior to initial shipment. The trailer is required to comply with applicable Hanford Site/U.S. Department of Transportation requirements (annual inspection must be current). Attached is a picture of the painted over nameplate, which shows the normal capacity to be 80,000 lb.

W. M. Boehnke

Date

R. J. Cummings Date

acp

Attachment

BWHC - B. L. Sasser L1-02

DYN - R. J. Cummings S4-56
T. C. Savage L4-19

WMNW - W. M. Boehnke H1-11
J. G. Field H1-11
R. J. Smith H1-11
WMB File/LB
BWHC/772030/121

A Division of Waste Management Federal Services, Inc.

10.2 DRAWINGS

1

WAS

SA
FILE

QTY REQ -540 -550 -560		PART/DASH NUMBER		DESCRIPTION		MATERIAL/REFERENCE		ITEM NO	
					OVERPACK ASSEMBLY			1	1
					BODY ASSEMBLY			5	2
					LID ASSEMBLY			2	3
					BOX ASSEMBLY			6	4
					NAME PLATE, SHEET, 16 GAUGE		STAINLESS STEEL	9	5
					NUCIL 013 VENTING FILTER		NUCLEAR FILTER TECH	2	6
					STIFFENER A, PLATE, 3/8"		ASTM A240 TYPE 304	7	7
					STIFFENER B, PLATE, 3/8"		ASTM A240 TYPE 304	7	8
					FLANGE		ASTM A240 OR A276 TYPE 304	8	9
					PLATE, 1 1/4" THK		ASTM A240 TYPE 304	2	10
					PLATE OR BAR		ASTM A240 OR A276 TYPE 304	8	11
					PLATE, 6 1/4" X 1 1/2" THK		ASTM A240 TYPE 304	5	12
					STIFFENER C, PLATE, 3/8"		ASTM A240 TYPE 304	7	13
					PORT COVER, PLATE, 1/4" THK		ASTM A240 OR A276 TYPE 304	4	14
					O-RING, 1/16", VTON		PARKER	2	15
					O-RING, 3/8", VTON		PARKER	8	16
					O-RING, 3/8", VTON		PARKER	8	17
					BOLT HY HEX, 3/4"-10UNC-ZA X 3 1/2" LG		ASTM A325 TYPE 1	1	18
					NUT, HY HEX, 3/4"-10UNC-2B		ASTM A194 GR 8	5	19
					SCREW, HEX CRUSONK CAP, #10		ASTM A449 TYPE 1	2	20
					24UNC-2A X 1 1/2" LG		ASTM A449 TYPE 1	2	21
					SCREW, HEX CAP, 1/4"		ASTM A449 TYPE 1	2	22
					20UNC-2A X 1 1/4" LG		ASTM A240 TYPE 304	7	23
					STIFFENER CAP A.C. SHEET, 16 GAUGE		ASTM A240 TYPE 304	7	24
					STIFFENER CAP B, SHEET, 16 GAUGE		ASTM A240 TYPE 304	7	25
					ROUND PIN, 3/4" DIA		CARR LANE, ASI 303	5	26
					ROUND PIN, 3/4" DIA X 3" LONG		CARR LANE, ASI 303	5	27
					HEAD		CARR LANE, ASI 303	5	28
					SWIVEL HOIST RING, WLL 4000 LB		CROSSBY	9	29
					5/8" - 11UNC X 2" LG		ASTM A240 TYPE 304	6	30
					BOTTOM PLATE, 1/2" THK		ASTM A240 TYPE 304	6	31
					END PLATE, 1/2" THK		ASTM A240 TYPE 304	6	32
					SIDE PLATE, 1/2" THK		ASTM A240 TYPE 304	6	33
					1/4" STAT-O-SEAL Im FLUOROCARBON		PARKER	2	34
					BUSHING, PRESS FIT, .770" ID X 1"		CARR LANE, ASI 416	3	35
					LONG		ASTM A240 OR A276 TYPE 304	4	36
					WELD PLUG, PLATE OR BAR		ASTM A240 OR A276 TYPE 304	4	37

GENERAL NOTES: (UNLESS OTHERWISE SPECIFIED)

1. ALL DIMENSIONS ARE IN INCHES. DIMENSIONING AND TOLERANCING PER ASME Y14.5M-94. TOLERANCES: FRACTIONAL = $\pm 1/8$, ANGULAR = $\pm 2^\circ$, DECIMAL .X = $\pm .01$, .XX = $\pm .03$, .XXX = $\pm .005$.
2. BREAK ALL SHARP EDGES AND REMOVE ALL BURRS.
3. MACHINED RADII .03 MAX.
4. ALL MACHINED SURFACES SHALL HAVE A 125/ OR BETTER FINISH.
5. FABRICATION SHALL BE IN ACCORDANCE WITH ASME B & PV CODE. SECTION VII, DIVISION 1. STAMP IS NOT REQUIRED. DESIGNATED MATERIALS SHALL CONFORM WITH ASTM STANDARDS. ALL MATERIAL SHALL BE AS SPECIFIED OR ENGINEER APPROVED EQUAL.
6. LIQUID PENETRANT EXAMINATION OF IDENTIFIED WELDS SHALL BE PERFORMED IN ACCORDANCE WITH ASME B & PV CODE, SECTION V, ACCEPTANCE CRITERIA SHALL CONFORM TO SECTION VII, APPENDIX B.

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DO NOT SCALE PRINT DIMENSIONS ARE IN INCHES UNLESS NOTED		DWG FILE ID: ED003		22 TON STEEL WASTE DISPOSAL BOX OVERPACK ASSEMBLY	
DATE: 12/14/99		DWT NO. 01156		WASTE MANAGEMENT TECHNICAL SERVICES	
DESIGNED BY: BD FLANNAGAN		CHECKED BY: JLB		DRAWING NUMBER E-D-003	
SUBMITTED DATE: 12/14/99		DATE: 12/14/99		SIZE: D	
DRAWN BY: WM BOEHMKE		DATE: 12/14/99		SHEET 1 OF 1	

① OVERPACK ASSEMBLY

SCALE: 1/10

REVISIONS					
REV	DATE	DESCRIPTION	DATE	APP'D BY	DATE

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REV NO	DESCRIPTION	REV BY	DATE	CHK BY	DATE	DATE

REVISIONS

3 LID ASSEMBLY
SCALE: 1/8

A10-9/10

#20 ±1.10
1/4" 20UNC-2B ±.5
FULL THD
#60 CBORE ±.35
#2.00 CBORE ±.25
SEAL AREA 32/ ONLY

#10 24-UNC-2B ±.5
FULL THD ON
#1.400 BC 4 PL
SEAL AREA 32/ ONLY

#20 THRU
1/4" 20UNC-2B ±.55
FULL THD
#60 CBORE ±.35
#2.00 CBORE ±.25

SECTION B-B

DETAIL 3

DETAIL 2

WASTE MANAGEMENT TECHNICAL SERVICES	22 TON STEEL WASTE DISPOSAL BOX OVERPACK ASSEMBLY	DO NOT SCALE PRINT DIMENSIONS ARE IN INCHES UNLESS NOTED	PROJECT NO. 772030 / 121	DATE FILED 01/03	DATE 12/14/99	BY BD FLANAGAN	CHECKED BY J. L. RILEY	DATE 12/14/99	REVISIONS
PROPRIETARY	NON-PROPRIETARY	THE DRAWING IS THE PROPERTY OF WASTE MANAGEMENT TECHNICAL SERVICES. IT IS NOT TO BE REPRODUCED, COPIED, OR USED IN ANY MANNER WITHOUT THE WRITTEN PERMISSION OF WASTE MANAGEMENT TECHNICAL SERVICES. AND IS TO BE RETURNED UPON REQUEST.	WASTE MANAGEMENT TECHNICAL SERVICES	WASTE MANAGEMENT TECHNICAL SERVICES	WASTE MANAGEMENT TECHNICAL SERVICES	WASTE MANAGEMENT TECHNICAL SERVICES	WASTE MANAGEMENT TECHNICAL SERVICES	WASTE MANAGEMENT TECHNICAL SERVICES	WASTE MANAGEMENT TECHNICAL SERVICES

REV NO	DESCRIPTION	REV BY	DATE	CHK BY	DATE	ENR

REVISIONS

32 WELD PLUG
SCALE: 2/1

14 PORT COVER
SCALE: 2/1

4 DETAIL 4

5 DETAIL 5

WASTE MANAGEMENT TECHNICAL SERVICES	
22 TON STEEL WASTE DISPOSAL BOX OVERPACK ASSEMBLY	
DO NOT SCALE PRINT DIMENSIONS ARE IN INCHES UNLESS NOTED	
PROJECT NO. 772030 / 121	DATE 01/15/99
DESIGNED BY BD FLANAGAN	CHECKED BY DE RILEY
ENGINEER WMA BOETTNER	DATE 1/14/99
REVISIONS	
REV NO	DESCRIPTION
1	END

SECTION C-C

SECTION D-D

② BODY ASSEMBLY

SCALE: 1/10

☐ PROPRIETARY ☒ NON-PROPRIETARY		PROJECT NO. 772630 / 121	
DO NOT SCALE PRINT DIMENSIONS ARE IN INCHES UNLESS NOTED		CAD FILE EL D0003 DTD NO. 01156 DRAWN BY 12/14/98
22 TON STEEL WASTE DISPOSAL BOX OVERPACK ASSEMBLY	
WASTE MANAGEMENT TECHNICAL SERVICES		REV. 0 DRAWING NUMBER E-D-003 SHEET 5 OF 5	
D		SCALE N/A DATE 1/1/99 CHECKED BY J. G. RILEY DRAWN BY WIN BOENKKE	

REVISIONS				
REV NO	DESCRIPTION	REV BY	CHEK BY	DATE

PRESS FIT
ROUND PIN

DETAIL 7

4 BOX ASSEMBLY
SCALE: 1/10

WASTE MANAGEMENT TECHNICAL SERVICES		DO NOT SCALE PRINT DIMENSIONS ARE IN INCHES UNLESS NOTED		WASTE MANAGEMENT TECHNICAL SERVICES	
PROJECT NO. 772030 / 121		DWG FILE NO. E0003		22 TON STEEL WASTE DISPOSAL BOX OVERPACK ASSEMBLY	
DATE: 12/11/99		DATE: 01/15/98		SIZE: D	
DRAWN BY: BD FLANAGAN		CHECKED BY: J. D. RILEY		DATE: 12/14/99	
DESIGNED BY: J. D. RILEY		APPROVED BY: W. BOEHMKE		DATE: 12/14/99	
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PAGE 1 OF 1		PAGE 1 OF 1		PAGE 1 OF 1	

REVISIONS			
NO.	DESCRIPTION	BY	DATE

WMTS	DATA CHANGE NOTICE (DCN)		DCN Controlled Number: <u>01009</u>	
			Release Information:	
DCN Category (mark one)	Originator's Name/Phone Number D. J. Reid/ 373-0106		Client/Company WMTS	
Supplemental ☒	Project Title		Required Response Date	
Direct Revision ☐	22 Ton Steel Waste Package			
Change DCN ☐	Document Numbers Changed by this DCN (include sheet no. and rev.)	Related DTF/DCN No(s)	Project/Crosswalk No.	
Cancel/Void ☐	E-D-003 Sheet 7 Rev. 0	NA	772030/233	
Design Baseline Document? ☒ Yes ☐ No	Affected Documents (Procedures, etc.) NA		Related PO Number	
Description of Change Modify Sections E-E and F-F as shown on page 2 of this DCN.				
Justification Electron Beam welding is not required for fabrication.				
Design Verification				
Design verification required? ☒ Yes ☐ No				
Design verification of this change was performed and the change(s) were found not to adversely impact the underlying safety basis.				
Verification performed by: <u>David Borman</u> <u>2/1/2000</u> (Name) (Date)				
File Information (see QP 17-1, Attachment A): ☒ Lifetime ☐ Non-Permanent				
SIGNATURES				
Title	Name	Signature	Date	
Originator	D. J. Reid	<i>D. J. Reid</i>	1/31/00	
Project/Task Manager	W. M. Boehnke	<i>W. M. Boehnke</i>	1/31/00	
QA	K. A. Johnson	<i>K. A. Johnson</i>	1/31/00	
Responsible Manager	J. A. Reeves	<i>J. A. Reeves</i>	1/31/00	
Safety	D. L. McCall Tino Romano	<i>D. L. McCall</i> for Tino Romano per Telecon 2/1/00	2/1/2000	
[X] COPY TO FILE

DATA CHANGE NOTICE (DCN) Continuation Sheet	DCN Controlled Number: 01039 Page 2 of 2
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Modify Sections E-E and F-F of drawing E-D-003 Sheet 7 Rev. 0 as shown below.

13 STIFFENER C
SCALE: 1/2

E SECTION E-E

F SECTION F-F

22 STIFFENER CAP A, C
SCALE: 4/5

23 STIFFENER CAP B
SCALE: 4/5

7 STIFFENER A
SCALE: 1/2

8 STIFFENER B
SCALE: 1/2

REV	NO	DESCRIPTION	REV	DATE	CHK	DATE	APP	DATE

☐ PROPRIETARY	☒ NON-PROPRIETARY	DO NOT SCALE PRINT DIMENSIONS ARE IN INCHES UNLESS NOTED	WM WASTE MANAGEMENT TECHNICAL SERVICES	
PROJECT NO. 772030 / 121		QSD FILE NO. E0003	22 TON STEEL WASTE DISPOSAL BOX OVERPACK ASSEMBLY	
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DESIGNED BY BD FLANAGAN		CHECKED BY J. RILEY		
DATE 12/14/99		DATE 12/14/99		
FORWARDED BY WIN BOCHANCE		DATE 12/14/99	SCALE 1/2	SHEET 7 OF 8

8 DETAIL 8

SECTION G-G

REVISIONS				
REV NO	DESCRIPTION	DATE BY	DATE APP'D	DATE

BOX LIFTING CONFIGURATION

SERIAL NUMBER: XXXXXXXX
DRAWING: WMTS / E-D-003 REV. X
MANUFACTURED: XX/XX/XXXX
MANUFACTURER: COMPANY NAME
PURCHASE ORDER: XXXXXX

5 NAME PLATE

4 PL 26

LID LIFTING CONFIGURATION

SUPPORT LID SUFFICIENTLY AND IN SUCH A MANNER AS TO PREVENT DAMAGE TO SEALING SURFACE WHEN REMOVED FROM BOX.

PROPRIETARY	DO NOT SCALE PRINT	WASTE MANAGEMENT TECHNICAL SERVICES
NON-PROPRIETARY	UNLESS NOTED OTHERWISE	
PROJECT NO. 772030 / 121	CAD FILE NO. E0003	22 TON STEEL WASTE DISPOSAL BOX OVERPACK ASSEMBLY
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DATE: 12/14/99	DATE: 12/14/99	DATE: 12/14/99

PART B: PACKAGE EVALUATION

1.0 INTRODUCTION

Part B of this safety analysis report for packaging (SARP) provides the evaluations that show the adequacy of the Steel Waste Package (SWP) in meeting the requirements of HNF-PRO-154, *Responsibilities and Procedures for all Hazardous Material*, for the onsite transfer of highway route controlled quantity (HRCQ) Type B, radioactive material and HNF-3723, *Packaging Design Criteria for the Steel Waste Package* (Boehnke 1999). The onsite normal conditions of transport (NCT) and accident conditions were analyzed as documented in this part. The SWP meets the criteria for those conditions.

1.1 SAFETY EVALUATION METHODOLOGY

The analyses detailed in this part for NCT include a 0.3 m (1.0-ft) drop onto a concrete surface in an orientation to cause the most damage, elevated and reduced external pressure, hot and cold conditions, penetration, vibration normally incident to transportation, lifting, and tiedown loadings. Accident conditions evaluated include a 9.1 m (30-ft) drop onto a concrete surface; free drop of 1 m (40 in.) onto the upper end of a 15 cm (6-in.-) diameter mild steel bar; and a 30-minute, fully engulfing, 800 °C (1475 °F) fire. The analyses show that the SWP maintains leaktight containment of the waste material through all NCT and accident conditions. Therefore, the SWP system meets onsite transportation safety criteria (WHC-SD-TP-RPT-001, *Report on Equivalent Safety for Transportation and Packaging of Radioactive Materials* [Mercado 1994]) based on SWP system design and operational control.

1.2 EVALUATION SUMMARY AND CONCLUSIONS

The SWP is safe for the onsite transfer of 300 Area remote-handled waste as demonstrated by this SARP. All loading/unloading, testing, and maintenance shall be followed and documented as stated in Part A of this SARP. As demonstrated in this part, the SWP meets the acceptance criteria for an onsite, HRCQ, Type B, radioactive material package.

1.2.1 Contents

The contents evaluated as acceptable for this SARP are a maximum heat load of 1410 W, a maximum content of 500,000 Ci total activity including activities of any daughter products, and a maximum fissile content of less than 200 g. Approval of this SARP authorizes shipment of this source term in the SWP under the conditions imposed in Part A of this SARP.

1.2.2 Radiological Risk

A radiological risk evaluation was not required because the SWP meets the applicable onsite packaging criteria as defined in Boehnke (1999).

1.2.3 Containment

Containment (Part B, Section 4.0) for the SWP is maintained by the Steel Waste Disposal Box (SWDB) Overpack during all NCT and hypothetical accident conditions (HAC) events. Fabrication leakage rate testing and annual retesting are performed per American National Standards Institute (ANSI) N14.5, *American National Standard for Radioactive Materials—Leakage Tests on Packages for Shipment* (ANSI 1997). Preshipment assembly air leak testing to a pressure of 26.6 kPa, gauge (4 psig), is performed prior to transport of each package. The leakage testing will demonstrate that the SWP was fabricated and assembled in a manner in which leaktightness (per ANSI [1997]) is ensured. Part B, Sections 4.0 and 7.0, document that a leaktight containment boundary is maintained for the postulated onsite accidents.

1.2.4 Shielding

For NCT the external dose rate, as shown in Part B, Section 5.0, is maintained below 10 mSv/h (1000 mrem/h) on the accessible surface of the SWP and for the vehicle surface, less than 0.1 mSv/h (10 mrem/h) at 2 m (6.5 ft), and less than 0.04 mSv/h (5 mrem/h) at the driver's location (4 m [13 ft]). If the driver location dose rate exceeds 0.02 mSv/h (2 mrem/h), the driver must be a trained radiological worker. For HAC, the dose rate limit of 10 mSv/h (1000 mrem/h) at 1 m (3.3 ft) is also met. Therefore, the SWP meets the acceptance criteria for shielding of Boehnke (1999).

1.2.5 Criticality

Criticality safety (Part B, Section 6.0) of the SWP is demonstrated based on the package containing a maximum payload fissile content of 200 g.

1.2.6 Structural

The SWP structural evaluation (Part B, Section 7.0) meets all NCT and HAC criteria identified in Boehnke (1999) with positive margins of safety. As prepared for transport, the SWP payload is protected by the packaging and its SWDB Impact Limiter.

1.2.7 Thermal

The SWP contents have a maximum decay heat of 1410 W. The SWP provides thermal protection to the contents during NCT and during an HAC fire event. Containment and shielding are maintained as evaluated per the criteria listed in Part B, Section 8.0.

1.2.8 Gas Generation

As shown in Part B, Section 9.0, water and organic material within the waste will generate hydrogen gas by radiolytic interactions. A payload containing the worst-case source term will generate sufficient hydrogen to reach 5 vol% within 26 hours. Shipping time limits are established to ensure that hydrogen concentrations within the SWP do not exceed 5 vol% in a period twice the shipping window as required by Boehnke (1999).

The total pressure inside the SWP will not exceed its 110.3 kPa, gauge (16-psig), design pressure during transportation.

1.2.9 Tiedown System

The SWP tiedown system (Part B, Section 10.0) was designed specifically for the SWP. The SWP tiedown system is demonstrated to be capable of restraining the SWP without producing stresses at or above the yielding strength of the materials when subjected to normal transportation loading conditions.

1.3 REFERENCES

HNF-PRO-154, *Responsibilities and Procedures for all Hazardous Material*, Fluor Hanford, Richland, Washington.

ANSI, 1997, *American National Standard for Radioactive Materials—Leakage Tests on Packages for Shipment*, ANSI N14.5, American National Standards Institute, New York, New York.

Boehnke, W. M., 1999, *Packaging Design Criteria for the Steel Waste Package*, HNF-3723, Rev. 0, prepared by Waste Management Federal Services, Inc., Northwest Operations for Fluor Daniel Hanford, Inc., Richland, Washington.

Mercado, J. E., 1994, *Report on Equivalent Safety for Transportation and Packaging of Radioactive Materials*, WHC-SD-TP-RPT-001, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

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2.0 CONTENTS EVALUATION

2.1 CHARACTERIZATION AND CLASSIFICATION

The contents of the SWP consist primarily of contaminated scrap equipment and fine dispersible material removed from 300 Area facilities, such as from the B Cell of the 324 Building. Equipment that may be present includes steel racks, tanks, pipes, hoses, valves, tank and pipe heel, and lead plugs from cell pipe penetrations. The dispersible material consists of small particulate solids (dust and powder).

2.1.1 Expected Radionuclide Inventory

The source term for the SWP is drawn from data on the dispersible material. It is assumed that the relative amounts of the isotopes in the dispersible material are representative of the relative amounts of the isotopes on the scrap equipment as well. Characterization data on the dispersible material are taken from PNL-324-96-005, *Manifest 96004: Mixed Waste EC-14,15,17, Feed Cans 1-13&15 Shipped 3/6/96* (PNL 1996a), and PNL-324-96-012, *Manifest 96006: Dispersible ED-16, 18, 19, 21, 22, 23, 22, and 25: 22.5 Ton Box Shipment 3/1/96* (PNL 1996b). These manifests document the shipment of 11 Engineered Containers (EC) loaded with dispersible material in the SWP on March 6, 1996, and March 1, 1996. Included in the manifests are solid waste storage/disposal records; spreadsheet calculations; and other worksheets, checklists, and forms. Data included in the manifests on waste other than the dispersible material are not considered in this source term evaluation.

The manifests report concentrations of ^{137}Cs , ^{90}Sr , ^{60}Co , ^{154}Eu , ^{238}Pu , ^{239}Pu , ^{240}Pu , and ^{241}Am for the 11 containers of dispersible material shipped. The expected source term, Table B2-1, is derived from a mixture consisting of these isotopes in the same proportions as the maximum concentrations of each isotope from all containers. This mixture is then decayed 3 years using the computer program Radcalc for Windows¹ version 2.01 (McFadden et al. 1998). This decay time corresponds to the amount of time since the shipments were made and allows sufficient time to build in concentrations of the prominent daughter products. The relative amounts of the isotopes in the mixture, including the daughter products, are then scaled to the maximum authorized activity of 500,000 Ci to obtain the expected activities.

Recent characterization data show the possibility of the presence of ^{241}Pu , ^{242}Pu , ^{243}Cm , and ^{244}Cm in wastes originating in B-Cell of the 324 Building. Hill and Hobart (*Data Analysis and Radionuclide Scaling Factor for the B-Cell Waste Stream* [2000]) have derived scaling factors for determining the activities of ^{241}Pu and ^{242}Pu based on the combined activity of ^{239}Pu and ^{240}Pu , and the activities of ^{243}Cm and ^{244}Cm based on the ^{137}Cs activity. These scaling factors are 24.7, 8.28E-4, 2.203E-5, and 1.520E-3 for ^{241}Pu , ^{242}Pu , ^{243}Cm , and ^{244}Cm , respectively. Uranium isotopes were not found to be present in the B-Cell characterization samples (Section 2.3.2, Hill and Hobart, 2000).

¹Windows is a registered trademark of the Microsoft Corporation, Redmond, Washington.

Table B2-1 lists the expected radionuclide inventory for the SWP. If this inventory does not bound the activity of each isotope of a particular shipment, then the sum-of-fractions approach in Part B, Section 2.1.2, can be used. If this approach is also unsatisfactory, an additional analysis can be performed for particular inventories for which the isotopic characterization is known to determine if the constraints in Part B, Section 2.1.2, are satisfied. This analysis shall be performed and documented according to Hanford-Site-approved quality assurance procedures.

Because the inventory contains greater than 3,000 A₂s, it is a Type B HRCQ quantity per 49 CFR 173, "Shippers—General Requirements for Shipments and Packagings," 173.403, "Definitions." The heat generation rate of this inventory is 1410 W. Because the total transuranic (TRU) activity is greater than 100 nCi/g, this inventory is radioactive TRU waste as defined in *Radioactive Waste Management* (DOE 1988) and 40 CFR 191, "Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes," 191.02, "Definitions."

Table B2-1. Expected Radionuclide Inventory of the Steel Waste Package.

Isotope	Relative amount (%)	Activity		Decay heat (W)	A ₂ value (Ci)	Inventory (A ₂)
		(Ci)	(Bq)			
⁶⁰ Co	5.37 E-04	2.68 E+00	9.92 E+10	4.13 E-02	1.08 E+01	2.48 E-01
⁹⁰ Sr	1.77 E+01	8.83 E+04	3.27 E+15	1.02 E+02	2.70 E+00	3.27 E+04
⁹⁰ Y+	1.77 E+01	8.83 E+04	3.27 E+15	4.89 E+02	daughter	daughter
¹³⁷ Cs	3.31 E+01	1.65 E+05	6.12 E+15	1.84 E+02	1.35 E+01	1.22 E+04
^{137m} Ba+	3.13 E+01	1.56 E+05	5.79 E+15	6.15 E+02	daughter	daughter
¹⁵⁴ Eu	2.26 E-02	1.13 E+02	4.18 E+12	1.03 E+00	1.35 E+01	8.37 E+00
²³⁸ Pu*	2.52 E-02	1.26 E+02	4.66 E+12	4.17 E+00	5.41 E-03	2.33 E+04
²³⁹ Pu*	1.91 E-03	9.54 E+00	3.53 E+11	2.97 E-01	5.41 E-03	1.76 E+03
²⁴⁰ Pu*	1.27 E-03	6.36 E+00	2.35 E+11	1.98 E-01	5.41 E-03	1.18 E+03
²⁴¹ Pu	7.87 E-02	3.93 E+02	1.45 E+13	1.25 E-02	2.70 E-01	1.46 E+03
²⁴² Pu*	2.64 E-06	1.32 E-02	4.88 E+08	3.89 E-04	5.41 E-03	2.44 E+00
²⁴¹ Am*	3.50 E-02	1.75 E+02	6.48 E+12	5.85 E+00	5.41 E-03	3.23 E+04
²⁴³ Cm*	7.29 E-04	3.64 E+00	1.35 E+11	1.33 E-01	8.11 E-03	4.49 E+02
²⁴⁴ Cm	5.03 E-02	2.51 E+02	9.30 E+12	8.80 E+00	1.08 E-02	2.33 E+04
Total	100	4.99 E+05	1.85 E+16	1.41 E+03	----	1.29 E+05

+Daughter nuclide. Isotopes contributing less than 1 E-3 Ci or 0.1 A₂ are not shown.

*Transuranic (TRU) isotope.

The expected inventory contains 165 g of fissile material, as shown in Table B2-2, but may contain up to 200 g of fissile material. This payload is not fissile excepted per 49 CFR 173.453, "Fissile materials—exceptions." Fissile isotopes as defined by 49 CFR 173.403 are

^{233}U , ^{235}U , ^{238}Pu , ^{239}Pu , and ^{241}Pu . The actual ratio of fissile isotopes may vary provided that Equation 2-2 in Part B, Section 2.1.2, is satisfied.

Table B2-2. Fissile Inventory of the Steel Waste Package.

Isotope	Activity		Mass (g)
	(Ci)	(Bq)	
^{238}Pu	1.26 E+2	4.66 E+12	7.41 E+0
^{239}Pu	9.54 E+0	3.53 E+11	1.54 E+2
^{241}Pu	3.93 E+02	1.45 E+13	3.93 E+00
Total	5.28 E+02	1.95 E+13	1.65 E+02

The actual ratio of fissile isotopes may vary provided that Equation 2-2 provided in Part B, Section 2.1.2, is satisfied.

A source term was developed specifically for the shielding evaluation such that the highest dose rates would be obtained. Other inventories, subject to the limits in this section, cause lower dose rates. The expected source term consists primarily of ^{137}Cs and ^{90}Sr and their daughters. Because the dose per curie from ^{137}Cs is greater than that from ^{90}Sr , a shielding source term was derived that maximizes ^{137}Cs at the expense of ^{90}Sr and is shown in Table B2-3.

Table B2-3. Bounding Radionuclide Inventory for the Shielding Evaluation.

Isotope	Relative amount (%)	Activity		Decay heat (W)	A ₂ value (Ci)	Inventory (A ₂)
		(Ci)	(Bq)			
^{60}Co	5.36 E-04	2.68 E+00	9.92 E+10	4.13 E-02	1.08 E+01	2.48E-01
^{90}Sr	0.00 E+00	0.00 E+00	0.00 E+00	0.00 E+00	2.70 E+00	0.00E+00
^{90}Y	0.00 E+00	0.00 E+00	0.00 E+00	0.00 E+00	daughter	daughter
^{137}Cs	5.13 E+01	2.56 E+05	9.49 E+15	2.85 E+02	1.35 E+01	1.90 E+04
$^{137\text{m}}\text{Ba}$	4.85 E+01	2.43 E+05	8.97 E+15	9.53 E+02	daughter	daughter
^{154}Eu	2.26 E-02	1.13 E+02	4.18 E+12	1.03 E+00	1.35 E+01	8.37 E+00
^{238}Pu	2.52 E-02	1.26 E+02	4.66 E+12	4.17 E+00	5.41 E-03	2.33 E+04
^{239}Pu	1.91 E-03	9.54 E+00	3.53 E+11	2.97 E-01	5.41 E-03	1.76 E+03
^{240}Pu	1.27 E-03	6.36 E+00	2.35 E+11	1.98 E-01	5.41 E-03	1.18 E+03
^{241}Pu	7.85 E-02	3.93 E+02	1.45 E+13	1.25 E-02	2.70 E-01	1.45 E+03
^{242}Pu	2.63 E-06	1.32 E-02	4.87 E+08	3.88 E-04	5.41 E-03	2.43 E+00
^{241}Am	3.50 E-02	1.75 E+02	6.48 E+12	5.85 E+00	5.41 E-03	3.23 E+04
^{243}Cm	7.28 E-04	3.64 E+00	1.35 E+11	1.33 E-01	8.11 E-03	4.49 E+02
^{244}Cm	5.03 E-02	2.51 E+02	9.30 E+12	8.80 E+00	1.08 E-02	2.33 E+04
Total	100	5.00 E+05	1.85 E+16	1.26 E+03	----	1.03 E+05

2.1.2 Sum-of-Fractions Derivation

An alternate approach to determine an acceptable source term of a particular payload is provided through the use of a sum of fractions equation. This approach is intended to be used when the activities of each isotope of a particular payload are not bounded for every isotope in Tables B2-1 or B2-3, or when unlisted isotopes are present.

Isotopic limits were derived for the isotopes identified in the dispersible materials (package identification numbers PNL-324-96-005 and -012) based on thermal, fissile mass, shielding, total activity, and gas generation constraints. Assuming only a single isotope is present in the payload, the activity limits determined from these constraints are shown in Table B2-4. The activity limit for a particular isotope is the maximum allowable activity determined from the most restrictive of the five constraints listed in this section. To ensure that the constraints are not exceeded for mixtures of isotopes, two "Sum of Fractions" equations are provided. Equation 2-1 is for nonfissile isotopes only, and Equation 2-2 is for fissile isotopes only. If both types of isotopes are present, then both equations must be satisfied. If the equations are satisfied, then the constraints will not be exceeded.

Table B2-4. Activity Limits L for Use in Sum-of-Fractions Equations 2-1 and 2-2.

Nonfissile isotope	L (Ci)
^{60}Co	3.4 E3
^{90}Sr	2.1 E5
^{137}Cs	2.5 E5
^{154}Eu	1.2 E4
^{234}U	1.2 E4
^{240}Pu	1.1 E4
^{241}Am	1.0 E4
$^{243/244}\text{Cm}$	9.8 E3
Fissile isotope	L (Ci)
^{233}U	1.9 E0
^{235}U	4.4 E-4
^{238}Pu	3.4 E3
^{239}Pu	1.2 E1
^{241}Pu	2.0 E4

Equation 2-1 applies to the nonfissile isotopes only.

$$\sum_i \frac{A_i}{L_i} \leq S \quad \text{Eq. 2-1}$$

where

- i = nonfissile isotope (^{60}Co , ^{90}Sr , ^{137}Cs , ^{154}Eu , ^{234}U , ^{240}Pu , ^{241}Am , and $^{243/244}\text{Cm}$)
- A_i = actual activity of isotope i present in the payload
- L_i = activity limit of isotope i (Table B2-4)
- S = sum of fractions; 0.9 if fissile material is present, 1.0 otherwise.

Equation 2-2 applies to the fissile isotopes only.

$$\sum_j \frac{A_j}{L_j} \leq 1.0 \quad \text{Eq. 2-2}$$

where

- j = fissile isotope (^{233}U , ^{235}U , ^{238}Pu , ^{239}Pu , and ^{241}Pu)
- A_j = actual activity of isotope j present in the payload
- L_j = activity limit of isotope j (Table B2-4)

The contents of the SWP is not limited to the isotopes in Table B2-4. A screening process is provided to eliminate isotopes present in small quantities that do not impact the evaluations in Part B. For isotopes not included in Table B2-4, if the activity of the isotope is less than 10% of its A_2 value, then that isotope may be transported in the SWP without inclusion in the sum of fractions calculation. If the activity of an unlisted isotope is greater than 10% of its A_2 value, then the activity limit of a surrogate isotope that conservatively bounds the radiation emitted by the unlisted isotope may be used. For predominantly gamma-emitting isotopes, the activity limit derived for ^{60}Co may be used. For predominantly beta-emitting isotopes, the activity limit derived for ^{90}Sr may be used. For other non-fissile isotopes, the activity limit for ^{240}Pu may be used. Alternatively, an additional analysis could be performed for particular inventories for which the exact isotopic characterization is known showing that the constraints in this section are satisfied. This analysis shall be performed and documented according to Hanford-Site approved quality assurance procedures.

The distinction between fissile and nonfissile isotopes is necessary to maximize the payload of the box. Because the fissile isotopes are limited to 200 g total, the contribution to both the heat generation and total activity by the fissile isotopes is less than 10 percent of the limits. If fissile mass was not considered separately from the other constraints, then the sum of fractions could be exceeded with less than 10 percent of the maximum authorized thermal generation rate and the maximum authorized activity. The use of two equations allows the maximum fissile mass plus additional nonfissile isotopes within the thermal, shielding, total activity, and gas generation constraints.

The activity limit for a particular isotope is the maximum allowable activity determined from the most restrictive of the five constraints listed in this section. As a result, the use of the sum-of-fractions equations to determine the acceptability of a particular inventory is more restrictive than an analysis for that inventory. The wider range of applicability of the sum-of-fractions approach is considered to offset its greater restriction.

Establishing activity limits based on the minimum determined from the four constraints on the non-fissile isotopes introduces conservatism that would not be present if separate equations and activity limits were given for each constraint. However, this conservatism was deemed acceptable compared to the cumbersome checking of five equations. If a particular inventory does not satisfy either Table B2-1 or both Equations 2-1 and 2-2, then an additional analysis can be performed for particular inventories for which the isotopic characterization is known to determine if the constraints in Part B, Section 2.1.2, are satisfied. This analysis shall be performed and documented according to Hanford-Site-approved quality assurance procedures.

The constraints are as follows:

- 1410 W total thermal generation
- 200 g total fissile mass for the fissile isotopes as defined in 49 CFR 173.403
- 500,000 Ci total activity including activities of any daughter products

- Shielding limits of 10 mSv/h (1000 mrem/h) at 1 m (HAC) and 10 mSv/h (1000 mrem/h) at contact, 0.1 mSv/h (10 mrem/h) at 2 m, and 0.02-5 mSv/h (2-5 mrem/h) at the driver location (NCT)
- 1700 cm³/h hydrogen generation rate.

Note that no dose consequence constraints are considered, because the box is designed to survive all hypothetical accident conditions.

The isotope limited by thermal generation was ⁹⁰Sr. The five fissile isotopes were limited by the fissile mass constraint. The isotope ¹³⁷Cs was limited by the total activity constraint.

The shielding-based limits were obtained from Monte Carlo N-Particle (MCNP) (Briesmeister 1997, McCoy 1998) calculations using the models for NCT and HAC from Part B, Section 5.0. Based on the results of the shielding analysis in Part B, Section 5.0, the most restrictive dose rate limit, i.e., the dose rate limit reached for the lowest activity inventory, is for the structural pieces payload during HAC at 1 m from the bottom surface. This case was rerun with unit source strengths of the strong gamma emitters ⁶⁰Co and ¹⁵⁴Eu to determine the dose rate per unit activity for each isotope. The allowable activity, based on shielding limits, was then back calculated from this factor and the 10 mSv/h (1000 mrem/h) dose rate limit. The other strong gamma emitter, ¹³⁷Cs, was analyzed, at its activity limit in Part B, Section 5.0, and was found to have met the shielding limits. The other isotopes in the expected source term of Table B2-1 are limited by constraints other than the shielding limits. The detailed shielding and gas generation models are given in Part B, Sections 5.8 and 9.4, respectively. The model input lines that were changed in the models are included in Part B, Section 2.6 (appendices). The gas generation limits were obtained from Radcalc calculations. The source was replaced with single isotopes of unit strength and then scaled to a hydrogen gas generation rate of 1700 cm³/h. The isotopes limited by this constraint consisted of ²³⁴U, ²⁴⁰Pu, ²⁴¹Am, and ^{243/244}Cm.

2.2 RESTRICTIONS

The maximum authorized activity of the SWP is 500,000 Ci. The maximum authorized fissile content is 200 g. The maximum authorized decay heat generation rate is 1410 W. The maximum authorized increase in pressure due to radiolytic gas generation is 96 kPa, gauge (14 psig). The activities of the isotopes transported in the SWP are limited such that either the activity of each isotope is below that in the nominal inventory in Table B2-1 or that both Equations 2-1 and 2-2 are satisfied. Table B2-1 does not satisfy Equations 2-1 and 2-2 because the equations are more conservative. Payloads satisfying either Table B2-1 or Equations 2-1 and 2-2 meet all constraints.

Material placed in the SWDB shall be restricted to dry waste. Dry is defined as 5 percent or less water and organics. Any containers that have contained liquids shall be completely drained and dried before being loaded into the RGC. No more than eight ECs will be transported in a single SWP. The SWP has a maximum payload capacity of 3860 kg (8500 lb) for the

nondispersible payload and 3300 kg (7300 lb) for the dispersible payload. The maximum total weight of the contents of the SWDB is 4500 kg (10,000 lb).

2.3 SIZE AND WEIGHT

Waste contents are limited to those that fit within the RGC for nondispersible and within the RGC or ECs for dispersible waste. The waste contents are limited to 3860 kg (8500 lb) for nondispersible and 3300 kg (7300 lb) for dispersible waste.

2.4 CONCLUSIONS

Radionuclide inventories of the SWP satisfying both Equations 2-1 and 2-2 are fissile, Type B, TRU, and HRCQ. Inventories in which each isotope is below its expected loading of Table B2-1 are also fissile, Type B, TRU, and HRCQ.

2.5 REFERENCES

40 CFR 191, "Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes," *Code of Federal Regulations*, as amended.

49 CFR 173, 1997, "Shippers--General Requirements for Shipments and Packagings," *Code of Federal Regulations*.

Briesmeister, J. F., Editor, 1997, *MCNP--A General Monte Carlo Code N-Particle Transport Code, Version 4B*, LA-12625-M, Los Alamos National Laboratory, Los Alamos, New Mexico.

DOE, 1988, *Radioactive Waste Management*, DOE 5820.2A, U. S. Department of Energy, Washington, D. C.

Hill, R. L., and R. L. Hobart, *Data Analysis and Radionuclide Scaling Factor for the B-Cell Waste Stream*, HNF-4904, Rev. 0-A, Fluor Hanford, Richland, Washington.

McCoy, J. C., 1998, *WMNW Computer Program Verification for MCNP4B*, EBU-SQA-002, Revision 1, Waste Management Federal Services, Inc., Northwest Operations, Richland, Washington.

McFadden, J. G., K. E. Hillesland, and J. G. Field, 1998, *Radcalc for Windows Version 2.01, Volume I: User's Manual, and Volume II: Technical Manual*, HNF-2549, Rev. 0, Waste Management Federal Services, Inc., Northwest Operations, Richland, Washington.

PNL, 1996a, *Manifest 96004: Mixed Waste, EC-14,15,17, FEED Cans 1-13&14 Shipped 3/6/96*, PNL-324-96-005, Pacific Northwest Laboratories, Richland, Washington.

PNL, 1996b, *Manifest 96006: Dispersible EC-16, 18, 19, 21, 22, 23, 24, and 25, 22.5 Ton Box Shipment 3/1/96*, PNL-324-96-012, Pacific Northwest Laboratories, Richland, Washington.

2.6 APPENDICES

2.6.1 Calculations of Activity Limits

This section describes how the activity limits in Table B2-4 are generated. The values for the heat production factor and specific activity were taken from Radcalc's (McFadden et al. 1998) data library. The limits, given in Part B, Section 2.1.2, are reiterated below for convenience:

- 1410 W total thermal generation
- 200 g total fissile mass for the fissile isotopes as defined in 49 CFR 173.403
- 500,000 Ci total activity including activities of any daughter products
- Shielding limits of 10 mSv/h (1000 mrem/h) at 1 m (HAC) and 10 mSv/h (1000 mrem/h) at contact, 0.1 mSv/h (10 mrem/h) at 2 m, and 0.02-5 mSv/h (2-5 mrem/h) at the driver location (NCT)
- 1700 cm³/h hydrogen gas generation rate.

The activity limit based on the thermal generation constraint is calculated by dividing the thermal limit of 1410 W by the heat production factor for a particular isotope.

The activity limit based on the fissile mass constraint is calculated by multiplying the fissile mass limit of 200 g by the specific activity. This constraint is only applicable to the fissile isotopes as defined by 49 CFR 173.403: ²³³U, ²³⁵U, ²³⁸Pu, ²³⁹Pu, and ²⁴¹Pu.

The activity limit based on the total activity constraint for isotopes with short-lived daughter products includes the activity of the daughter product at its secular equilibrium concentration. Both ⁹⁰Sr and ¹³⁷Cs have short-lived daughter products, ⁹⁰Y and ^{137m}Ba, respectively. This activity limit is calculated by dividing the total activity limit of 500,000 Ci by the sum of one (for the parent isotope) and the branching fraction to the short-lived daughter product. The branching fraction for ⁹⁰Y is 1 and for ^{137m}Ba is 0.946.

The activity limit based on the shielding constraints requires running MCNP with a unit activity of a particular isotope using the models developed in Part B, Section 5.0, to determine the dose rate per unit activity. This calculation is described in Part B, Section 2.1.2. The resulting dose rates per unit activity are given in Part B, Section 2.6.2, for ⁶⁰Co and ¹⁵⁴Eu.

Part B, Section 2.6.5, contains the ORIGEN-S and MCNP input files for ^{244}Cm along with the resulting dose rates per unit activity and activity limit. This activity limit is obtained by dividing the most restrictive dose rate limit by the dose rate per unit activity at that location.

The activity limit based on the gas generation constraint is calculated in Radcalc using the gas generation model developed in Part B, Section 9.0. The hydrogen gas generation limit of $1700\text{ cm}^3/\text{h}$, is divided by the gas generation rate of hydrogen gas due to the radiolysis of hydrogenous materials per unit activity of a particular isotope to obtain this activity limit. Part B, Section 2.6.3, gives the hydrogen gas generation rates for isotopes producing significant amounts of hydrogen. Only alpha emitters are likely to be limited by this constraint.

Table B2-5 shows the activity limits for use in sum-of-fractions Equations 2-1 and 2-2.

Table B2-5. Calculation of Activity Limits for Use
in Sum-of-Fractions Equations 2-1 and 2-2.

Nuclide	Heat Prod. (W/Ci)	Sp. Act. (Ci/g)	Heat (Ci)	Fissile (Ci)	Shldg. (Ci)	Total Act. (Ci)	Gas Gen. (Ci)	Max. Allow. (Ci)	Limiting
^{60}Co	1.54 E-02		9.16 E+04		3.47 E+03	5.00 E+05	1.29 E+05	3.47 E+03	Shldg.
^{90}Sr	6.70 E-03		2.10 E+05			2.50 E+05	2.10 E+05	2.10 E+05	Heat
^{137}Cs	4.83 E-03		2.92 E+05			2.57 E+05	3.57 E+05	2.57 E+05	Total Act.
^{154}Eu	9.08 E-03		1.55 E+05		1.23 E+04	5.00 E+05	1.57 E+05	1.23 E+04	Shldg.
^{233}U	2.91 E-02	9.70 E-03	4.85 E+04	1.94 E+0		5.00 E+05	1.21 E+04	1.94 E+00	Fissile
^{234}U	2.88 E-02		4.90 E+04			5.00 E+05	1.22 E+04	1.21 E+04	H ₂ gas
^{235}U	2.77 E-02	2.20 E-06	5.09 E+04	4.40 E-4		5.00 E+05	1.30 E+04	4.40 E-04	Fissile
^{238}Pu	3.26 E-02	1.70 E+01	4.33 E+04	3.40 E+3		5.00 E+05	1.06 E+04	3.40 E+03	Fissile
^{239}Pu	3.11 E-02	6.20 E-02	4.53 E+04	1.24 E+1		5.00 E+05	1.13 E+04	1.24 E+01	Fissile
^{240}Pu	3.11 E-02		4.53 E+04			5.00 E+05	1.13 E+04	1.11 E+04	H ₂ gas
^{241}Pu	3.10 E-05	1.03 E+02	4.55 E+07	2.06 E+4		5.00 E+05	2.72 E+07	2.06 E+04	Fissile
^{241}Am	3.34 E-02		4.22 E+04			5.00 E+05	1.06 E+04	1.04 E+04	H ₂ gas
$^{244/243}\text{Cm}$	3.66 E-02		3.85 E+04		1.9 E+04	5.00 E+05	9.88 E+03	9.88 E+03	H ₂ gas

2.6.2 Replacement Lines for MCNP Input File 22Ton52 in Part B, Section 5.0

^{60}Co :

```
202 c/x 0.0 40.0 20.0 $ tally in air
212 c/y 0.0 40.0 20.0 $ tally in air
222 cz      20.0 $ tally in air
sdef erg=d7 x=d1 y=d2 z=d3 wgt=7.3956E+10
sc7 Co-60 source
si7 1 1.1732 1.3325
sp7 .999 .99982
sd212 1256.6 1. 1256.6 1. 2513.3 1.
```

This source in the structural "pieces" model gives 0.288 mrem/h (fractional uncertainty of 0.0186) at 1 m from the bottom surface during HAC.

¹⁵⁴Eu:

```

202 c/x 0.0 40.0 20.0 $ tally in air
212 c/y 0.0 40.0 20.0 $ tally in air
222 cz          20.0 $ tally in air
sdef erg=d7 x=d1 y=d2 z=d3 wgt=5.69245E+10
sc7  Eu-154 source
si7  1.123 .248 .592 .723 .757 .873 .996 1.0048 1.2745 1.594
sp7   .405 .066 .048 .197 .043 .1145 .103 .179 .355 .028
sd212 1256.6 1. 1256.6 1. 2513.3 1.

```

This source in the structural “pieces” model gives 0.0812 mrem/h (fractional uncertainty of 0.0263) at 1 m from the bottom surface during HAC.

2.6.3 Replacement Lines for Radcalc Input File in Part B, Section 9.3.2²³⁴U:

Source from input:

Radionuclide:	Curies:	Becquerels:	Grams:
U-234	4.90E+004	1.81E+015	7.88E+006
Total Activity:	4.90E+004	1.81E+015	
Total Minus Daughters:	4.90E+004	1.81E+015	

²³⁴U Output:

HYDROGEN:

The sealed package will contain 2.50 % hydrogen in 0.14 days

H2 Volume (20 C, 101.325 kPa): 2.31E+004 cc

H2 Rate (20 C, 101.325 kPa) : 6.82E+003 cc/hour

²⁴⁰Pu:

Source from input:

Radionuclide:	Curies:	Becquerels:	Grams:
Pu-240	4.53E+004	1.68E+015	1.97E+005
Total Activity:	4.53E+004	1.68E+015	
Total Minus Daughters:	4.53E+004	1.68E+015	

²⁴⁰Pu Output:

HYDROGEN:

The sealed package will contain 2.50 % hydrogen in 0.14 days

H2 Volume (20 C, 101.325 kPa): 2.31E+004 cc

H2 Rate (20 C, 101.325 kPa) : 6.83E+003 cc/hour

²⁴¹Am:

Source from input:

Radionuclide:	Curies:	Becquerels:	Grams:
Am-241	4.22E+004	1.56E+015	1.23E+004
Total Activity:	4.22E+004	1.56E+015	
Total Minus Daughters:	4.22E+004	1.56E+015	

²⁴¹Am Output:

HYDROGEN:

The sealed package will contain 2.50 % hydrogen in 0.14 days

H2 Volume (20 C, 101.325 kPa): 2.31E+004 cc

H2 Rate (20 C, 101.325 kPa) : 6.77E+003 cc/hour

²³⁸Pu:

Source from input:

Radionuclide:	Curies:	Becquerels:	Grams:
Pu-238	3.40E+003	1.26E+014	2.00E+002
Total Activity:	3.40E+003	1.26E+014	
Total Minus Daughters:	3.40E+003	1.26E+014	

²³⁸Pu Output:

HYDROGEN:

The sealed package will contain 2.50 % hydrogen in 1.76 days

H2 Volume (20 C, 101.325 kPa): 2.31E+004 cc

H2 Rate (20 C, 101.325 kPa) : 545. cc/hour

^{243/244}Cm:

Source from input:

Radionuclide:	Curies:	Becquerels:	Grams:
Cm-243	9.88E+003	3.66E+014	1.91E+002
Total Activity:	9.88E+003	3.66E+014	
Total Minus Daughters:	9.88E+003	3.66E+014	

^{243/244}Cm Output:

HYDROGEN:

The sealed package will contain 2.50 % hydrogen in 0.57 days

H2 Volume (20 C, 101.325 kPa): 2.31E+004 cc

H2 Rate (20 C, 101.325 kPa) : 1.70E+003 cc/hour

CHECKLIST FOR CHECKING OF ANALYSIS/CALCULATIONS

Document Number/Revision: HNF-4763, Rev. 0Document Title: SWP SARP / Part B, Section 2.6.3

Yes	No	N/A	
☐	☐	☐	Problem completely defined.
☒	☐	☐	Appropriate analytical method used.
☒	☐	☐	Necessary assumptions are appropriate and explicitly stated.
☒	☐	☐	Computer codes and data files documented.
☒	☐	☐	Data used in calculations explicitly stated in document.
☐	☐	☒	Sources of non-standard formula/data are referenced and the correctness of the reference verified.
☒	☐	☐	Data checked for consistency with original source information as applicable.
☒	☐	☐	Mathematical derivations checked including dimensional consistency of results.
☒	☐	☐	Models appropriate and used within range of validity or use outside range of established validity justified.
☐	☐	☒	Hand calculations checked for errors.
☒	☐	☐	Code run streams correct and consistent with analysis documentation.
☒	☐	☐	Code output consistent with input and with results reported in analysis documentation.
☒	☐	☐	Acceptability limits on analytical results applicable and supported. Limits checked against sources.
☒	☐	☐	Safety Margins consistent with good engineering practices.
☒	☐	☐	Conclusions consistent with analytical results and applicable limits.
☒	☐	☐	Results and conclusions address all points required in the problem statement.

I have checked the analysis/calculation and it is complete and accurate to the best of my knowledge.

Engineer/Checker E. Neilsen 2-4-2000 Date

NOTE: Any hand calculations, notes or summaries generated as part of this check should be signed, dated, and attached to this checklist. Material should be labeled and recorded so that it is intelligible to a technically qualified third party.

2.6.4 Data on Dispersible Material from Manifests No. 96004 and No. 96006

Sheet4

Calculation page 1

SCW Shipment of mixed waste all data was taken from analytical smears (see attached sheet). The following values were calculated by converting uCi/g into Curies (ie multiplying by the weight and dividing by 1000.													
Container	ersible Ne	Cs-137	Sr-90	Co-60	Eu-154	Pu-238	Pu-239	Pu-240	Am-241				
16	84.9	1.12E+04	6.82E+03		5.57E+00	6.63E-01							Pu-239/240
Curies		9.51E+02	5.79E+02		4.73E-01	5.63E-02	0.022974	0.015316	5.88E+00				4.51E-01
									4.99E-01				3.83E-02
19	86.4	1.36E+04	4.42E+03		9.34E+00	1.27E+01							1.44E+00
Curies		1.18E+03	3.82E+02		8.07E-01	1.10E+00	0.07465	0.049766					1.24E-01
21	120.86	1.29E+04	9.30E+03		7.67E+00	8.35E-01			8.93E+00				6.20E-01
Curies		1.56E+03	1.12E+03		9.27E-01	1.01E-01	0.04496	0.029973	1.08E+00				7.48E-02
22	98.7	1.75E+04	8.08E+03	0.15	7.91E+00	1.58E+00			1.00E+01				1.00E+00
Curies		1.73E+03	7.97E+02	1.48E-02	7.81E-01	1.56E-01	0.05922	0.03948	9.87E-01				9.87E-02
23	98.7	1.44E+04	7.79E+03	0.2	1.41E+01	2.02E+00			1.73E+01				1.39E+00
Curies		1.42E+03	7.69E+02	1.97E-02	1.39E+00	1.99E-01	0.082316	0.054877	1.71E+00				1.37E-01
24	103.09	1.02E+04	6.13E+03	0.2	1.41E+00	3.27E-01			4.29E+00				1.72E-01
Curies		1.05E+03	6.32E+02	2.06E-02	1.45E-01	3.37E-02	0.010639	0.007093	4.42E-01				1.77E-02
25	122.86	1.22E+04	7.80E+03	0.39	1.33E+01	2.24E+00			1.52E+01				1.56E+00
Curies		1.50E+03	9.58E+02	4.79E-02	1.63E+00	2.75E-01	0.114997	0.076665	1.87E+00				1.92E-01
18	50	1.31E+04	6.82E+03	0.0047	7.98E+00	3.02E+00			8.34E+00				8.98E-01
Curies		652.65	341.15	0.0047	0.39915	0.15115	0.026937	0.017958	0.41675				0.044895
Total Curies		1.00E+04	5.58E+03	1.08E-01	6.56E+00	2.07E+00	0.436692	0.291128	7.00E+00				7.28E-01
The Pu-239 and Pu-240 was split apart based upon the curie ratio of Point Beach fuel sheared during NWVP.													
The curie ratio was calculated to be 60% Pu-239 and 40% Pu-240.													

NOTE: No samples were taken from 18. Container 18 curie contents were average from the other containers

Page 1

Sheet4

SCW Shipment of dispersible non mixed containers. All data was taken from analytical samples (see original list). The following values were calculated by converting uCi/g into Curies (ie multiplying by the weight and dividing by 1000.										
Container	Net Wt	Pu-238	Pu-239	Pu-240	Sr-90	Cs-137	Eu-154	Am-241		
14 uCi/g	109.22	1.39E+00	6.12E-01	4.08E-01	7.16E+03	8.43E+03	1.05E+01	1.25E+01		239-240
14 Curies		1.52E-01	6.68E-02	4.46E-02	7.82E+02	9.21E+02	1.15E+00	1.37E+00		1.02E+00
15uCi/g	76.1	7.58E+00	4.20E-01	2.80E-01	5.11E+03	1.48E+04	4.00E+00	0.00E+00		7.00E-01
15 Curies		5.77E-01	3.20E-02	2.13E-02	3.89E+02	1.13E+03	3.04E-01	0.00E+00		
17uCi/g	98.2	8.95E-01	3.76E-01	2.50E-01	5.62E+03	1.53E+04	6.03E+00	9.25E+00		6.26E-01
17 Curies		0.087889	0.036884	0.024589	551.884	1502.46	0.592146	0.90835		
Total Curies		8.17E-01	1.36E-01	9.05E-02	1.72E+03	3.55E+03	2.04E+00	2.27E+00		
The Pu-239 and Pu-240 was split apart based upon the curie ratio of Point (Beach Fuel sheared during NWVP. The ratio was calculated to be 60% Pu-239 and 40% Pu-240.										

X Note: Calculation page 1

2.6.5 ORIGEN-S and MCNP Input Files for ^{244}Cm

ORIGEN-S input file for unit activity of ^{244}Cm

```
#ORIGENS
0$$ E T
DECAY CASE
3$$ 21 1 1 22 A16 4 A33 0 E T
35$$ 0 T
56$$ A2 3 A6 1 A10 0 A13 1 A15 3 E
57** 0 E T
CM-244
1 CI
60** .1 .3 1
61** F.00001
65$$
'GRAM-ATOMS GRAMS CURIES WATTS-ALL WATTS-GAMMA
21Z
3Z 3Z 1 0 0 3Z 3Z 6Z
21Z
84** 1.492E+7 1.22E+7 1.E+7 8.18E+6 6.36E+6
4.96E+6 4.06E+6 3.01E+6 2.46E+6 2.35E+6
1.83E+6 1.11E+6 5.5E+5 1.11E+5 3.35E+3
5.83E+2 1.01E+2 2.9E+1 1.01E+1 3.059999E+0
1.12E+0 4.14E-1 9.999998E-3
73$$ 962440
74** 1
75$$ 2 T
56$$ F0 T
END
```

MCNP input file for unit activity of ^{244}Cm

This input file for the Engineered Containers model gives a dose rate of 5.18×10^{-4} mrem/h/Ci (fractional uncertainty of 0.0729) at 2 m from the top surface of the package during normal conditions of transport, which for ^{244}Cm is the most restrictive dose rate limit and transport condition. The dose at 2 m from the lateral surface (longest side) is 4.52×10^{-4} mrem/h/Ci (fractional uncertainty of 0.0508).

```
Steel Waste Package Shielding Model neutrons/NCT
c
c      neutron src, 4 MeV
c
c      This model based on Lee Carter's ibulk file (Carter 1998)
c
c      Wilson mods use surface and cell #'s >300
c      Wilson modification adds impact limiter and ground scatterer
c
c      Overpack/containment design is new and includes added
c      shielding at base (Boehnke 1999). Steel box thicknesses
c      include thin-walled steel liners into which
c      payload is placed
c
c      the liners are: "grout" container 0.120" sides, 0.1875" btm
c      and "disposal" container 0.1875" sides, btm and lid
c      for total: 0.78105 cm sides, 0.9525 cm btm, 0.47625 cm lid
c
c      A steel plate that is part of truck attachment frame is included
c      in the NCT shielding model and is part of base of impact limiter
c
c      Dose rates for NCT referenced to impact limiter
c      Dose rates for HAC referenced to overpack
c
c      Source from J. Boles, Ci inventory from Table B2-4
```

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```

c      500,000 Ci total inventory (no Sr-90/Y-90)
c
c      Douglas fir composition: assumed same as celotex C6H1005
c      (LANL primer), with 0.512 g/cc density from CRC 43rd ed.
c
c      ===== CELL BLOCK =====
c      inside box, 8 source cans, source, steel for each
1  58 -1.15 -101 -92  91      $ source
2 304 -8.00 -111 -93  94 (101:92:-91) $ steel can
3  58 -1.15 -102 -92  91      $ source
4 304 -8.00 -112 -93  94 (102:92:-91) $ steel can
5  58 -1.15 -103 -92  91      $ source
6 304 -8.00 -113 -93  94 (103:92:-91) $ steel can
7  58 -1.15 -104 -92  91      $ source
8 304 -8.00 -114 -93  94 (104:92:-91) $ steel can
9  58 -1.15 -105 -92  91      $ source
10 304 -8.00 -115 -93  94 (105:92:-91) $ steel can
11  58 -1.15 -106 -92  91      $ source
12 304 -8.00 -116 -93  94 (106:92:-91) $ steel can
13  58 -1.15 -107 -92  91      $ source
14 304 -8.00 -117 -93  94 (107:92:-91) $ steel can
15  58 -1.15 -108 -92  91      $ source
16 304 -8.00 -118 -93  94 (108:92:-91) $ steel can
19  7 0.0000512 -1  8 -31 38 -61 68 $ beyond cans, inside box
      (93:-94:(111 112 113 114 115 116 117 118))
c      carbon steel box -- cells going outward for importances
31 33 -7.85 -2  9 -32 39 -62 69 (1:-8:31:-38:61:-68)
32 33 -7.85 -3 10 -33 40 -63 70 (2:-9:32:-39:62:-69)
33 33 -7.85 -4 11 -34 41 -64 71 (3:-10:33:-40:63:-70)
34 33 -7.85 -5 12 -35 42 -65 71 (4:-11:34:-41:64) $ no -z
35 33 -7.85 -6 13 -36 43 -66 71 (5:-12:35:-42:65) $ no -z
36 33 -7.85 -7 14 -37 44 -66 71 (6:-13:36:-43) $ no -z, +za
c      air-gap between carbon steel box and overpack
41  7 0.0000512 -16 18 -46 48 -76 71 (7:-14:37:-44:66)
c      overpack with 3" shield at base (includes 3/8" liner btm)
42 304 -8.0 -17 19 -47 49 -77 792 (16:-18:46:-48:76:-793) $ sides and top
      (4:-11:34:-41:71) $ of overpack
43  33 -7.85 -16 18 -46 48 -71 79 $ bottom 3.375" shield plate
431 33 -7.85 -16 18 -46 48 -79 791 $ bottom shield plate
433 33 -7.85 -16 18 -46 48 -791 793 $ bottom shield plate
432 304 -8.0 -4 11 -34 41 -793 792 $ bottom 0.5" overpack btm
c      air beyond, first 4 cells for sides and above, next 4 cells are below
61  7 0.0000512 -22 26 -52 56 -82 86 (501:-508:511:-518:521) $ to 1 m
62  7 0.0000512 -23 27 -53 57 -83 86 (22:-26:52:-56:82) $ to 2 m
63  7 0.0000512 -24 28 -54 58 -84 86 (23:-27:53:-57:83) $ to 3 m
631 7 0.0000512 -29 30 -59 60 -89 86 (24:-28:54:-58:84) $ to 4 m
64  7 0.0000512 -99      (29:-30:59:-60:89) $ beyond
65  7 0.0000512 -501 508 -511 518 -528 86 $ to 1 m
66  58 -1.7 -29 30 -59 60 -86 87 $ silica for ground scatter to 2 m
67  7 0.0000512 -29 30 -59 60 -87 88 $ to 3 m
68  7 0.0000512 -29 30 -59 60 -88 -99 $ beyond 3 m
c      gap between overpack and impact limiter
71  7 0.0000512 505 -504 515 -514 -524 792
      (17:-19:47:-49:77:-792)
c      impact limiter
c      inner steel plate of impact limiter excluding bottom (cell 810)
81  33 -7.85 -503 506 -523 526 -513 516
      (504:-505:524:-792:514:-515) (-11:4:-41:34:-526:792)
c      Douglas fir timbers between steel plates
c      (density from 43rd CRC, p.1569) excludes bottom (cell 820)
82  100 -0.512 -502 507 -522 527 -512 517
      (503:-506:523:-526:513:-516) (-11:4:-41:34:-527:526)
c      outer steel plate of impact limiter excludes bottom (cell 830)
83  33 -7.85 -501 508 -521 528 -511 518
      (502:-507:522:-527:512:-517) (-11:4:-41:34:-528:527)
810 33 -7.85 11 -4 41 -34 526 -792 $ btm piece beneath cell 432
820 100 -0.512 11 -4 41 -34 527 -526 $ btm piece beneath cell 810
830 33 -7.85 11 -4 41 -34 528 -527 $ btm piece beneath cell 820
c
c      outside world

```

```

99  0          99

c  ===== SURFACE BLOCK =====
c  carbon steel box, 78" x 42" x 37" inner, 96" x 60" x 48" outer
  1 px  98.279  $ wall includes 0.78105 cm liner
  2 px  102.87
  3 px  106.68
  4 px  110.49
  5 px  114.30
  6 px  118.11
  7 px  121.92
  8 px -98.279  $ wall includes 0.78105 cm liner
  9 px -102.87
 10 px -106.68
 11 px -110.49
 12 px -114.30
 13 px -118.11
 14 px -121.92
c  overpack
 16 px  127.635
 17 px  128.905
 18 px -127.635
 19 px -128.905
c  tally surfaces referenced to impact limiter
 22 px  260.655
 23 px  360.655  $ tally, 2 m
 24 px  460.655
 26 px -260.655
 27 px -360.655  $ tally, 2 m
 28 px -460.655
 29 px  560.655  $ tally at 4 m added for driver
 30 px -560.655  $ tally at 4 m added for driver
c  impact limiter
 501 px  160.655  $ outer steel plate surface
 502 px  160.02   $ outer Doug Fir surface
 503 px  139.7    $ inner steel plate surface
 504 px  139.065  $ inner steel plate surface
 505 px -139.065  $ inner steel plate surface
 506 px -139.7    $ inner steel plate surface
 507 px -160.02   $ outer Doug Fir surface
 508 px -160.655  $ outer steel plate surface
c
c  carbon steel box, 78" x 42" x 37" inner, 96" x 60" x 48" outer
 31 py  52.559  $ wall includes 0.78105 cm liner
 32 py  57.15
 33 py  60.96
 34 py  64.77
 35 py  68.58
 36 py  72.39
 37 py  76.20
 38 py -52.559  $ wall includes 0.78105 cm liner
 39 py -57.15
 40 py -60.96
 41 py -64.77
 42 py -68.58
 43 py -72.39
 44 py -76.20
c  overpack
 46 py  76.835
 47 py  78.105
 48 py -76.835
 49 py -78.105
c  tallies referenced to impact limiter
 52 py  211.125
 53 py  311.125  $ tally, 2 m
 54 py  411.125
 56 py -211.125
 57 py -311.125  $ tally, 2 m
 58 py -411.125
 59 py  511.125
 60 py -511.125

```

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```

c      impact limiter
511 py 111.125 $ outer steel plate surface
512 py 110.49 $ outer Doug Fir surface
513 py 90.17 $ inner steel plate surface
514 py 89.535 $ inner steel plate surface
515 py -89.535 $ inner steel plate surface
516 py -90.17 $ inner steel plate surface
517 py -110.49 $ outer Doug Fir surface
518 py -111.125 $ outer steel plate surface

c
c      carbon steel box, 78" x 42" x 37" inner, 96" x 60" x 48" outer
c      7" lid, 4" bottom
61 pz 93.5038 $ includes 0.47625 cm liner steel
62 pz 97.536
63 pz 101.092
64 pz 104.648
65 pz 108.204
66 pz 111.76
68 pz 0.00 $ floor of steel box
69 pz -3.387
70 pz -6.773
71 pz -10.16

c      overpack (55.25" or 140.335cm overall height)
76 pz 117.158 $ inside surf. overpack top
77 pz 120.333 $1.25" top thickness per new design
79 pz -13.16 $ 3.0" overpack base shield VR surf
791 pz -16.66 $ 3.0" overpack base shield VR surf
793 pz -18.7325 $ 3.375" overpack base shield btm (includes 3/8" liner)
792 pz -20.0025 $ 0.5" overpack bottom

c      -z tallies referenced to btm frame plate of IL
c      the impact limiter rests on this plate
82 pz 262.6275 $ tally, 1 m
83 pz 362.6275 $ tally, 2 m
84 pz 462.6275
86 pz -144.1325 $ tally, 1 m
87 pz -244.1325 $ tally, 2 m
88 pz -344.1325
89 pz 562.6275

c      impact limiter
c      the impact limiter surfaces have moved because of 3" steel btm plate
c      and also to account for 3/8" liners -inner IL height is 161.04cm
521 pz 162.6275 $ outer steel plate surface
522 pz 161.9925 $ outer Doug Fir surface
523 pz 141.6725 $ inner steel plate surface
524 pz 141.0375 $ inner steel plate surface
c      surf 792 at -20.0025 is the inner surf of IL inner steel plate
c      btm IL surfaces moved due to 3" steel + 3/8" liners -also move top IL surfaces
526 pz -20.6375 $ inner steel plate surface (.25" plate)
527 pz -40.9575 $ outer Doug Fir surface (8")
528 pz -44.1325 $ outer steel plate + 1" frame base (1.25")

c
c      surfaces for source
91 px -74.6125 $ -x end of source with 0.625" plug
92 px 74.6125 $ +x end of source with 0.625" plug
93 px 76.20 $ +x end of pipes (60 in. long)
94 px -76.20 $ -x end of pipes (60 in. long)

c      outside world
99 so 1000.0

c      surfaces for source, inner radius first
101 c/x -27.5000 13.7500 13.2334
102 c/x 0.0 13.7500 13.2334
103 c/x 27.5000 13.7500 13.2334
104 c/x -13.7500 37.5657 13.2334
105 c/x 13.7500 37.5657 13.2334
106 c/x -27.5000 61.3814 13.2334
107 c/x 0.0 61.3814 13.2334
108 c/x 27.5000 61.3814 13.2334
111 c/x -27.5000 13.7500 13.6525
112 c/x 0.0 13.7500 13.6525
113 c/x 27.5000 13.7500 13.6525
114 c/x -13.7500 37.5657 13.6525

```

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```

115 c/x 13.7500 37.5657 13.6525
116 c/x -27.5000 61.3814 13.6525
117 c/x 0.0 61.3814 13.6525
118 c/x 27.5000 61.3814 13.6525
c tally circles
201 c/x 0.0 40.0 30.0 $ for contact
202 c/x 0.0 40.0 60.0 $ in air
211 c/y 0.0 40.0 30.0 $ for contact
212 c/y 0.0 40.0 60.0 $ in air
221 cz 30.0 $ for contact
222 cz 60.0 $ in air

c =====DATA BLOCK=====
c nps 10000
ctme 20
print 10 110
prtmp j -800
mode n
c imp:p 1 16r 4 16 64 512 2048 8192 8192 16384 256
c 16384 4r 16384 16384 2r 16384 3r 1024 4096 16384 0
imp:n 1 44r 0
phys:n
sdef erg=d1 cel=d2 pos fcel d3 axs=1 0 0 ext=d4
rad=d5 wgt=1.39e+5
sc1 neutron spectrum (alpha,n), sp. fission
sil s 20 21
spl 0.0076 0.9924
sc20 (alpha,n) neutron source on SiO2 from Jacobs & Liskien
c Energy of the alpha particles =5.5 MeV
# si20 sp20
0.0 0.0
0.1 2.44
0.2 2.50
0.3 2.44
0.4 2.77
0.5 3.57
0.6 4.47
0.7 4.67
0.8 4.78
0.9 4.38
1.0 3.89
1.1 3.51
1.2 2.76
1.3 2.23
1.4 1.83
1.5 1.66
1.6 1.48
1.7 1.37
1.8 1.51
1.9 1.73
2.0 1.86
2.1 2.09
2.2 2.27
2.3 2.46
2.4 2.51
2.5 2.44
2.6 2.52
2.7 2.57
2.8 2.54
2.9 2.37
3.0 2.23
3.1 1.98
3.2 1.69
3.3 1.47
3.4 1.27
3.5 1.14
3.6 1.00
3.7 0.76
3.8 0.49
3.9 0.41
4.0 0.31

```

```

      4.1      0.25
      4.2      0.14
      4.3      0.10
      4.4      0.05
      4.5      0.04
sc21  spontaneous fission source spectrum (Cm 244)
sp21  -3  0.906  3.848
si2   1  1 3 5 7 9 11 13 15
sp2   d  1 7r
ds3   1  0.0    -27.5000  13.7500    0.0    0.0    13.7500
      0.0    27.5000  13.7500    0.0   -13.7500  37.5657
      0.0    13.7500  37.5657    0.0   -27.5000  61.3814
      0.0    0.0000  61.3814    0.0    27.5000  61.3814
si4   -74.6125  74.6125
sp4   -21  0
si5   13.2334
sp5   -21  1
c
m7    8016.50c  0.22000  7014.50c  0.78000  $ Air
m33   26000.55c -.9775  24000.50c -.010  25055.50c -.0075
      42000.50c -.002  6000.50c -.003  $ carbon steel
m58   8016.50c 0.66667  14000.50c 0.33333  $ simulate glass
c     SS-304L from Nuclear Systems Materials Handbook Rev. 36
m304  6000.50c -0.0003  26000.55c -0.6872  15031.60c -0.01
      25055.50c -0.02  28000.50c -0.0925  24000.50c -0.19
c     Douglas fir (celotex composition-LANL criticality primer)
m100  1001.50c -.0622  6000.50c -.4444  8016.50c -0.4934
mt100 lwtr.01t
c     ansi/ans-6.1.1-1977 fluence-to-dose,neuts(mrem/hr/(n/cm**2/s)
de0   log  2.5-8 1-7 1-6 1-5 1-4 .001 .01 .1 .5 1 2.5 5 7 10 14 20
df0   log  .00367 .00367 .00446 .00454 .00418 .00376 .00356 .0217
      .0926 .132 .125 .156 .147 .147 .208 .227
c
fc2   dose (mrem/hr) on contact x surface, 1st entry
f2:n  501 508 t
fs2   -201
sd2   2827.4 1. 2827.4 1. 5654.8 1.
fc22  dose (mrem/hr) at 2 meter in x, 1st entry
f22:n 23 27 t
fs22  -202
sd22  11310. 1. 11310. 1. 22620. 1.
fc42  dose (mrem/hr) at 4 meter in x, 1st entry
f42:n 29 30 t
fs42  -202
sd42  11310. 1. 11310. 1. 22620. 1.
fc102 dose (mrem/hr) on contact y surface, 1st entry
f102:n 511 518 t
fs102 -211
sd102 2827.4 1. 2827.4 1. 5654.8 1.
fc122 dose (mrem/hr) at 2 meter in y, 1st entry
f122:n 53 57 t
fs122 -212
sd122 11310. 1. 11310. 1. 22620. 1.
fc202 dose (mrem/hr) on contact z surface, 1st entry
f202:n 521 528 t
fs202 -221
sd202 2827.4 1. 2827.4 1. 5654.8 1.
fc212 dose (mrem/hr) at 1 meter in z, 1st entry
f212:n 82 86 t
fs212 -222
sd212 11310. 1. 11310. 1. 22620. 1.
fc222 dose (mrem/hr) at 2 meter in z, 1st entry
f222:n 83 87 t
fs222 -222
sd222 11310. 1. 11310. 1. 22620. 1.

```

CHECKLIST FOR CHECKING OF ANALYSIS/CALCULATIONS

Document Number/Revision: HNF-4763, Rev 0Document Title: Safety Analysis Report for Packaging (Onsite) Steel Waste PackageSection: Contents - Part B Section 2 (Contents), Section 5.8.5 (Shielding) for addition of 4 new isotopes (^{241}Pu , ^{242}Pu , ^{243}Cm , ^{244}Cm)

Yes	No	N/A	
☒	☐	☐	Problem completely defined.
☒	☐	☐	Appropriate analytical method used.
☒	☐	☐	Necessary assumptions are appropriate and explicitly stated.
☒	☐	☐	Computer codes and data files documented.
☒	☐	☐	Data used in calculations explicitly stated in document.
☐	☐	☒	Sources of non-standard formula/data are referenced and the correctness of the reference verified.
☒	☐	☐	Data checked for consistency with original source information as applicable.
☒	☐	☐	Mathematical derivations checked including dimensional consistency of results.
☒	☐	☐	Models appropriate and used within range of validity or use outside range of established validity justified.
☒	☐	☐	Hand calculations checked for errors.
☒	☐	☐	Code run streams correct and consistent with analysis documentation.
☒	☐	☐	Code output consistent with input and with results reported in analysis documentation.
☒	☐	☐	Acceptability limits on analytical results applicable and supported. Limits checked against sources.
☒	☐	☐	Safety Margins consistent with good engineering practices.
☒	☐	☐	Conclusions consistent with analytical results and applicable limits.
☒	☐	☐	Results and conclusions address all points required in the problem statement.

I have checked the analysis/calculation, and it is complete and accurate to the best of my knowledge.

Engineer/Checker Anthony Savino

Date 5/15/2000

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3.0 RADIOLOGICAL RISK EVALUATION

3.1 INTRODUCTION

The U. S. Department of Energy Order 460.1A (DOE 1996) requires that onsite transfers of Type B radioactive material packagings comply with the Hazardous Materials Regulations of the Department of Transportation (49 CFR 171-180) or show that a degree of safety equivalent to the regulations is demonstrated. 49 CFR 173.413 requires Type B packages to be designed and constructed to meet the requirements of 10 CFR 71. 10 CFR 71.51 requires Type B packages to be designed so that under the normal conditions of transport (NCT) specified in 10 CFR 71.71, there would be no loss or dispersal of radioactive contents, as demonstrated to a sensitivity of 10^{-6} A₂ per hour, no significant increase in external surface radiation levels, and no substantial reduction in the effectiveness of the packaging. Additionally, under the hypothetical accident conditions (HAC) in 10 CFR 71.73, the package must be designed so that there would be no escape of krypton-85 exceeding 10 A₂ in 1 week, and no external radiation dose rate exceeding 10 mSv/h (1 rem/h) at 1 m (40 in) from the external surface of the package.

Part B, Section 4.0, "Containment Evaluation," address the containment requirement for both NCT and HAC, and concludes that the SWP packaging system containment boundary maintains leaktightness for both NCT and HAC.

Part B, Section 5.0, "Shielding Evaluation," address the shielding requirement for both NCT and HAC, and concludes that the external dose rates from the SWP at the maximum authorized inventory during NCT and after HAC are below the dose rate limits.

Part B, Section 2.0, "Contents Evaluation," shows that 10 A₂ of krypton-85 is not authorized for shipment in the SWP; therefore, this requirement is not applicable.

Because the SWP meets the Hazardous Materials Regulations of the DOT, the safety requirements for onsite transfers are met per DOE (1996).

3.2 REFERENCES

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

10 CFR 71.51, "Additional Requirements for Type B Packages," *Code of Federal Regulations*, as amended.

10 CFR 71.71, "Normal Conditions of Transport," *Code of Federal Regulations*, as amended.

10 CFR 71.73, "Hypothetical Accident Conditions," *Code of Federal Regulations*, as amended.

49 CFR 173, "Shippers--General Requirements for Shipments and Packagings," *Code of Federal Regulations*, as amended.

49 CFR 173.413, "Requirements for Type B Packages," *Code of Federal Regulations*, as amended.

DOE, 1996, *Packaging and Transportation Safety*, DOE O 460.1A, U. S. Department of Energy, Washington, D. C.

4.0 CONTAINMENT EVALUATION

4.1 INTRODUCTION

The 22-Ton SWDB is designed and constructed to provide multiple confinement boundaries and a containment boundary during transport. The packaging system consists of a series of two or three confinement boundaries (depending on dispersibility of the contents), one within the other, providing multiple boundaries to prevent release of material. Each confinement boundary is vented through NucFil² filters to prevent hydrogen buildup. These confinement boundaries are sealed within a leaktight containment boundary defined as the SWDB Overpack. The SWDB Overpack is designed and fabricated to meet the leaktight criteria of ANSI (1997) and maintains containment of the waste during NCT and postulated accident conditions on the Hanford Site.

As designed, the SWDB Overpack is a rectangular box with a bolted closure. Machined into the flange are two Viton³ O-ring grooves for the containment seals and a leak test port. The containment boundary seal is provided by the two Viton O-rings. Two 1.9 cm (0.75-in.) round locating pins are provided on the flange for installation of the closure lid. For ease of installation, one locating pin is taller than the other one. Also machined into the flange are 32 through holes for the closure bolts with heavy hex nuts. The SWDB Overpack closure lid is a 3.18 cm (1.25-in.-) thick stepped plate with the sealing surface landing machined to a thickness of 2.87 cm (1.13 in.). The stepped center portion of the plate provides additional shear loading resistance. The closure lid is provided with the two mating locating pinholes for lid installation and 32, 3/4-10 unified national coarse thread (UNC) American Society for Testing and Materials (ASTM) A-325, closure bolts for lid closure. The lid is also equipped with four swivel hoist rings for lift points. In addition, the lid is equipped with two ports that are a combination leak test, purge, and NucFil ports. Containment sealing of the ports is provided by the closure bolts and two fluorocarbon Stat-O-Seals⁴ to maintain a leaktight (10^{-7} std cc/s) seal.

The primary purpose for the SWDB is to be the interim storage container for the 324 Building RH mixed and TRU waste until it is treated and/or repacked for disposal. The SWP retains the waste under all normal and accident conditions.

4.2 CONTAINMENT SOURCE SPECIFICATION

A fabrication verification leak test of the SWP containment boundary will be performed at the manufacturer. That leak test will demonstrate that the SWP can meet the 10^{-7} std cc/s leak rate defined as leaktight by ANSI (1997).

²NucFil is a registered trademark of Nuclear Filter Technology, Inc., Lakewood, Colorado.

³Viton is a registered trademark of the E. I. du Pont de Nemours & Company, Wilmington, Delaware.

⁴Stat-O-Seal is a registered trademark of The Franklin C. Wolfe Company, Inc., Culver City, California.

No dose consequence or leak rate analyses were required because the SWP will be verified to meet the ANSI (1997) definition of leaktight. Therefore, no containment source term was specified.

4.3 NCT

4.3.1 Conditions To Be Evaluated

The conditions to be evaluated for NCT are pressurization of the SWDB Overpack, seal temperature range, and seal integrity after a 0.3 m (1-ft) drop. The SWDB Overpack must remain leaktight and must provide containment under all NCT.

The NCT involves exposure to temperatures as low as -33 °C (-27 °F) and as high as 46 °C (115 °F) while exposed to maximum solar insolation. The containment boundary must survive external pressure fluctuations as low as 95.21 kPa, absolute (13.81 psia), and as high as 102.38 kPa, absolute (14.85 psia). The containment boundaries also must survive a water spray; vibration normally incident to transfer; impact of a 3.2 cm (1.25-in.-) diameter, 6 kg (12-lb), vertical steel cylinder dropped from a height of 1 m; and a 0.3 m (1-ft) free drop onto an essentially hard, unyielding surface.

4.3.2 Containment Acceptance Criteria

The SWP containment shall meet the ANSI (1997) leaktight criterion with a leakage rate less than 10^{-7} std cm³/s. Leaktightness is proven when the SWP containment boundary is shown to meet American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (BPVC), Section III, Subsection NE (ASME 1995d), Service Levels A and C, criteria during all NCT.

As stated in Part B, Section 4.2, the SWDB Overpack will undergo a fabrication leak test to demonstrate a leaktight design.

4.3.3 Containment Model

The structural analyses of the SWP are provided in Part B, Section 7.0. The results of the NCT analyses are presented in Part B, Section 7.4. These structural analyses demonstrate that the SWP maintains a leaktight containment boundary during NCT.

4.3.4 Containment Calculations

The structural analyses summarized in Part B, Section 7.4, demonstrate that the SWP satisfies ASME (1995d) Service Levels A and C criteria during NCT. Because the Service Levels A and C stress allowables are met, the SWDB Overpack remains leaktight during NCT.

4.4 ACCIDENT CONDITIONS

4.4.1 Conditions To Be Evaluated

The conditions to be evaluated for accident scenarios are pressurization of the SWP and performance of the SWP seal during the drop, puncture, and fire.

The impact accident is simulated by a free drop of 9.0 m (30 ft) of the package onto a hard unyielding surface. The puncture is a 1 m (40-in.) free drop of the package onto a 15 cm (6.-in.-) diameter mild steel bar. The thermal accident is simulated by exposure of the package to a 30-minute, 800 °C (1475 °F) engulfing fire.

4.4.2 Containment Acceptance Criteria

The package must satisfy external release rate requirements for defined accident conditions. The SWP containment boundary remains leaktight during defined accident scenarios. The containment boundary must be shown to satisfy ASME (1995d) Service Level D stress allowables during the accident conditions. Proper squeeze must also be maintained on the SWP lid seal. Squeeze is defined as the percent reduction in the cross-sectional diameter of the O-ring due to compression.

4.4.3 Containment Model

The structural analysis of the SWP is provided in Part B, Section 7.0. The results of the accident condition analyses are presented in Part B, Section 7. 5, and the thermal evaluation is presented in Part B, Section 8.0. These analyses demonstrate that the SWP maintains a leaktight containment boundary and provides containment during all accident conditions.

4.4.4 Containment Calculations

The structural analyses summarized in Part B, Section 7.5, demonstrate the SWP satisfies ASME (1995d) Service Level D stress allowables during all accident conditions. Because the Service Level D stress allowables are met and proper compression is maintained on all SWP seals (as shown in Part B, Section 4.7.1), the SWDB Overpack meets containment criteria.

4.5 CONTAINMENT EVALUATION AND CONCLUSIONS

Structural analyses show that leaktightness of the SWP packaging system containment is maintained for onsite NCT and for defined accident conditions.

4.6 REFERENCES

ANSI, 1997, *American National Standard for Radioactive Materials - Leakage Tests on Packages for Shipment*, ANSI N14.5, American National Standards Institute, New York, New York.

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

4.7 APPENDIX: 22-TON BOX CONTAINMENT SEAL ENGINEERING SAFETY EVALUATION

ENGINEERING SAFETY EVALUATION

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Preparer: SS Shiraga <i>[Signature]</i>	Date: 01/11/00
Checker: WS Josephson <i>[Signature]</i>	Date: 01/13/00

1.0 OBJECTIVE

The objective of this evaluation is to evaluate the primary containment sealing system performance of the closure lid and seal under normal transport conditions (NTC) and accident conditions (AC). Acceptable performance is demonstrated by the capability of the seals to remain in compression under NTC and AC, thus maintaining the containment.

2.0 REFERENCES

Parker Seal Group, 1991, *Parker O-Ring Handbook*, Parker Seal Group, O-Ring Division, Lexington, Kentucky.

WMTS, 1999, *22 Ton Steel Waste Disposal Box Shield Plate*, drawing E-D-005, Rev. 0, Waste Management Technical Services, Richland, Washington.

3.0 ASSUMPTIONS, RESULTS, AND CONCLUSIONS

In this evaluation, the following assumptions are made:

- The closure lid O-rings are assumed to be stretched into the grooves as recommended by the manufacturer. The assumed stretch of the O-rings is 5%. This results in a cross-sectional diametrical reduction of the O-ring of approximately 3.4%.
- The worst-case loading results from the top edge drop of the package under both NTC and AC.
- The shelf life of the seals has not been exceeded and the elastomeric O-rings remain elastic and pliable.

This evaluation is based on determining the range of initial O-ring compression. For conservatism, the low end of the range is used as the basis of evaluation. The amount of compression of the seal is then compared with the average plastic elongation of a set of closure bolts in the most highly loaded section of the closure lid to flange joint. The fundamental basis for using the average plastic elongation of a set of closure bolts is that the O-ring is elastic, the closure lid is 1.13 in. thick at it narrowest location, and the mating flange has a thickness of

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1.5 in. When clamped together, these thick plates have a high degree of stiffness and make up a highly rigid joint. The closure lid/flange joint stiffness and the bolts surrounding the plastically deformed bolt act to maintain the joint and seal compression. Complete loss of compression of an O-ring section would require the entire section of the closure lid/flange and a set of bolts to experience large unrestrained plastic deformations; however, this does not occur as demonstrated in the structural analysis (Part B, Section 7.0). Subsequently, large unrestrained plastic deformation on one bolt would not result in decompression of the seal. However, large unrestrained plastic deformation of an entire set of bolts would result in loss of seal compression of a section of the O-ring seal, which could result in loss of containment.

The approach taken in this evaluation is to assess the maximum accumulated true plastic strains on the most highly loaded set of bolts reported in the structural evaluation (Part B, Section 7.0) for NTC and AC. In this assessment, the average maximum accumulated true plastic strain is determined. The length of the bolt to determine the average bolt elongation for most highly loaded set of bolts is then multiplied by this value. It should be noted that the bolt length in this evaluation is not the actual length of the bolt, but the finite element analysis (FEA) model bolt length, which is from the top of the closure lid to top of the flange.

Based on the manufacturer's recommendations (Parker 1991), the minimum compression of the O-ring is greater than 0.007 in. and less than the 30% maximum recommended. Consequently, the sizing of the O-ring seals is adequate for optimum performance. In order to prevent the material from taking a permanent compressive set, the manufacturer establishes the requirement for the minimum compression of 0.007 in.

Results of this evaluation show that for a set of six bolts, the average maximum accumulated elongation under NTC is well below the minimum compression of the O-ring seal. This is demonstrated by a margin of safety of 30.4. In the AC case, the margin of safety for the same set of six bolts is 0.05. This results in a less than optimal seal compression of 0.002 in. after a worst-case AC drop. At this compression, some seal compression would be maintained to prevent loss of containment. These margins of safety are based on the worst loading condition of a top edge drop with a maximum internal pressure of 16 psig. Based on the positive margins of safety of maintenance of seal compression, the containment performance of the elastomeric O-ring sealing system is demonstrated for NTC and AC.

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4.0 EVALUATION

22 Ton Box Containment Seal Evaluation:

Determine range of cross section, assuming a 5% stretch (Parker 1991, A4-7):

Free cross sectional diameter range: $\text{csd}_1 := 0.341 \cdot \text{in}$ $\text{csd}_2 := 0.375 \cdot \text{in}$ Assumed inside stretch: $\text{st} := 5.0$ % reduction in cross section: $\text{yrd} := 0.56 + 0.59 \cdot \text{st} - 0.0046 \cdot \text{st}^2$ $\text{yrd} = 3.395$ Stretched cross section range: $\text{csdn}_1 := \text{csd}_1 \cdot \left(1 - \frac{\text{yrd}}{100}\right)$ $\text{csdn}_1 = 0.329 \cdot \text{in}$ $\text{csdn}_2 := \text{csd}_2 \cdot \left(1 - \frac{\text{yrd}}{100}\right)$ $\text{csdn}_2 = 0.362 \cdot \text{in}$ Range of groove depth: $\text{gd}_1 := 0.276 \cdot \text{in}$ $\text{gd}_2 := 0.284 \cdot \text{in}$

Range of O-ring compression:

Least diameter with greatest depth: $\text{comp}_1 := \text{csdn}_1 - \text{gd}_2$ $\text{comp}_1 = 0.0454 \cdot \text{in}$ Greatest diameter with least depth: $\text{comp}_2 := \text{csdn}_2 - \text{gd}_1$ $\text{comp}_2 = 0.0863 \cdot \text{in}$

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$$\text{Nominal compression: } \text{comp}_{\text{nom}} := \frac{\text{comp}_1 + \text{comp}_2}{2} \quad \text{comp}_{\text{nom}} = 0.0658 \text{ in}$$

$$\text{Greatest \% compression: } \frac{\text{comp}_2}{\text{csdn}_1} = 26.2 \%$$

$$\text{Least \% compression: } \frac{\text{comp}_1}{\text{csdn}_2} = 12.5 \%$$

$$\text{Length of bolt based on FEA element length: } l_b := 1.13 \text{ in}$$

NTC Seal Compression Determination:

Average logarithmic (or true plastic) strain for six highest strained bolts under worst case loading Section 7.4.5 Table B7-12:

$$\epsilon_{n1} := 0.00207 \frac{\text{in}}{\text{in}} \quad \epsilon_{n2} := 0.00045 \frac{\text{in}}{\text{in}} \quad \epsilon_{n3} := 0.00280 \frac{\text{in}}{\text{in}}$$

$$\epsilon_{n4} := 0.00054 \frac{\text{in}}{\text{in}} \quad \epsilon_{n5} := 0.00001 \frac{\text{in}}{\text{in}} \quad \epsilon_{n6} := 0.00182 \frac{\text{in}}{\text{in}}$$

$$\epsilon_{n\text{ave}} := \frac{\epsilon_{n1} + \epsilon_{n2} + \epsilon_{n3} + \epsilon_{n4} + \epsilon_{n5} + \epsilon_{n6}}{6} \quad \epsilon_{n\text{ave}} = 0.00128 \frac{\text{in}}{\text{in}}$$

$$\text{Average lengthening of bolts across most highly loaded region: } l_{n\text{inc}} := l_b \cdot \epsilon_{n\text{ave}}$$

$$l_{n\text{inc}} = 0.00145 \text{ in}$$

NTC Margin of Safety Based on Lowest Compression:

$$\text{MS}_{\text{ntc}} := \frac{\text{comp}_1}{l_{n\text{inc}}} - 1 \quad \text{MS}_{\text{ntc}} = 30.4$$

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AC Seal Compression Determination:

Average logarithmic (or true plastic) strain for six highest strained bolts under worst case loading Section 7.5.5 Table B7-16:

$$\epsilon_1 := 0.00043 \cdot \frac{\text{in}}{\text{in}} \quad \epsilon_2 := 0.00608 \cdot \frac{\text{in}}{\text{in}} \quad \epsilon_3 := 0.02748 \cdot \frac{\text{in}}{\text{in}}$$

$$\epsilon_4 := 0.08160 \cdot \frac{\text{in}}{\text{in}} \quad \epsilon_5 := 0.05372 \cdot \frac{\text{in}}{\text{in}} \quad \epsilon_6 := 0.06068 \cdot \frac{\text{in}}{\text{in}}$$

$$\epsilon_{\text{ave}} := \frac{\epsilon_1 + \epsilon_2 + \epsilon_3 + \epsilon_4 + \epsilon_5 + \epsilon_6}{6} \quad \epsilon_{\text{ave}} = 0.03833 \cdot \frac{\text{in}}{\text{in}}$$

Average lengthening of bolts across most highly loaded region:

$$l_{\text{inc}} := l_b \cdot \epsilon_{\text{ave}}$$

$$l_{\text{inc}} = 0.04331 \cdot \text{in}$$

AC Margin of Safety Based on Lowest Compression:

$$MS_{\text{ac}} := \frac{\text{comp}_1}{l_{\text{inc}}} - 1$$

$$MS_{\text{ac}} = 0.05$$

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5.0 SHIELDING EVALUATION

5.1 INTRODUCTION

This shielding evaluation supports the transport of wastes from the 300 Area to the 200 Area using the SWP. The contaminated material is from 300 Area facilities, such as the B cell of Building 324. There are two waste payload configurations. One payload consists of contaminated fine dispersible material that is loaded into stainless steel cylinders (ECs) prior to loading into the SWP. The other payload consists of small pieces of contaminated scrap loaded into a steel container (the Rectangular Grout Container [RGC]) and placed into the steel shielding box of the SWP. The maximum payload activity for the shielding evaluation is 500,000 Ci.

The limits for this package are 10 mSv/h (1000 mrem/h) at any point on the exterior surface of the package and 0.1 mSv/h (10 mrem/h) at lateral distances of 2 m from the package. At the driver location, assumed to be 4 m from the package, the limit is 0.02 mSv/h (2 mrem/h) if the driver is not a trained radiological worker, and 5 mrem/h if the driver is a radiological worker. After an HAC resulting in a partial loss of shielding, the limit is 10 mSv/h (1000 mrem/h).

MCNP (Briesmeister 1997, McCoy 1998a) simulations were done to determine gamma-ray dose rates at the SWP surface and beyond in both the vertical and horizontal directions. MCNP is a computer program that uses sampling methods based on randomly generated numbers (the Monte Carlo method) to determine the transport of neutral particles, such as gamma rays, through shielding materials. The MCNP model is based on an earlier model developed by Carter (Mercado 1998). Neutron dose rates were estimated using the procedure described by Nelson (1996) and with the MCNP code.

5.2 DIRECT RADIATION SOURCE SPECIFICATION

The sources used for these evaluations represent the worst-case shielding condition based on information provided by Fluor Hanford. It is expected that the primary radionuclide will be ^{137}Cs with its associated gamma-emitting daughter product $^{137\text{m}}\text{Ba}$. The source term considered in this evaluation is shown in Table B5-1. It is taken from Part B, Section 2.0, Table B2-3. Other source terms meeting the conditions of Part B, Section 2.0, are conservatively derived to produce smaller dose rates than the source in Table B5-1.

Table B5-1. Radioisotopes in Source.

Isotope	Activity (Ci)
^{60}Co	2.68 E+00
^{90}Sr	0.00 E+00
^{90}Y (daughter)	0.00 E+00
^{137}Cs	2.56 E+05
$^{137\text{m}}\text{Ba}$ (daughter)	2.43 E+05
^{154}Eu	1.13 E+02
^{234}U	1.09 E-03
^{238}Pu	1.26 E+02
^{239}Pu	9.54 E+00
^{240}Pu	6.36 E+00
^{241}Pu	3.93 E+02
^{242}Pu	1.32 E-02
^{241}Am	1.75 E+02
^{243}Cm	3.64 E+00
^{244}Cm	2.51 E+02
Total	5.00 E+05

5.2.1 Gamma Source

The gamma source for the SWP payload consists primarily of the gamma ray from the ^{137}Cs daughter, $^{137\text{m}}\text{Ba}$. The gamma rays from ^{60}Co and ^{154}Eu are low in intensity, but high in energy and can become important for thick shields. Gamma rays from the TRU radionuclides are negligible in intensity and low in energy. ORIGEN-S (Bowman 1997, McCoy 1998b) was used to calculate the gamma-ray source term for all the radionuclides in Table B5-1 including bremsstrahlung from energetic beta particles. The ORIGEN-S calculation of bremsstrahlung production assumes that the matrix material for beta-particle slowing down is uranium oxide. This produces a conservative bremsstrahlung estimate. The ORIGEN-S input file used to produce the gamma-ray source term is included in Part B, Section 5.8.3. The energy distribution of the gamma source term is shown in Table B5-2.

5.2.2 Beta Source

Virtually all beta particles originating in the source are from ^{137}Cs . They do not contribute to the dose rate outside the SWP because of the shielding provided by the Steel Waste Disposal Box, overpack, and impact limiter. This shielding is described in Part B, Section 5.4.3. The bremsstrahlung radiation produced by the deceleration of the beta particles in the source is accounted for in the photon emission rates calculated using the ORIGEN-S computer code. However, this bremsstrahlung is not expected to contribute to the dose rate outside the SWP because of its low energy.

Table B5-2. Energy Distribution of the Gamma-Ray Source Term.

Upper energy bound (MeV)	Gamma-ray intensity (photons/s)	Upper energy bound (MeV)	Gamma-ray intensity (photons/s)
5.0000 E-02	0.0000 E+00	9.9633 E-01	4.3100 E+11
1.0000 E-01	1.2950 E+14	1.0047 E+00	2.6700 E+09
2.0000 E-01	5.5800 E+13	1.0049 E+00	7.2700 E+11
3.0000 E-01	1.1400 E+13	1.1731 E+00	3.3000 E+10
4.0000 E-01	3.8700 E+12	1.1733 E+00	9.9100 E+10
5.0000 E-01	8.6500 E+11	1.2744 E+00	3.9100 E+10
6.6163 E-01	7.6000 E+11	1.2746 E+00	1.4800 E+12
6.6165 E-01	8.0600 E+15	1.3324 E+00	1.5000 E+09
7.2329 E-01	1.9800 E+11	1.3326 E+00	9.9100 E+10
7.2331 E-01	8.2500 E+11	1.5000 E+00	3.1500 E+10
8.7318 E-01	2.7100 E+11	2.0000 E+00	6.6100 E+10
8.7320 E-01	4.8100 E+11	2.5000 E+00	1.0600 E+06
9.9631 E-01	7.1100 E+10	Total	8.2600 E+15

5.2.3 Neutron Source

The actinides contained in the payload (^{234}U , ^{238}Pu , ^{239}Pu , ^{240}Pu , ^{241}Am from Table B5-1) produce neutrons through spontaneous fission and (α ,n) reactions. The total neutron source from spontaneous fission and (α ,n) reactions was calculated to be 1.61×10^6 n/s using Tables 1 and 3 in Nelson (1996) and the activities from Table B5-1. Because the allowed quantities of the fissile isotopes are much less than required for criticality, for even the most favorable conditions, the subcritical multiplication factor is conservatively assumed to be 0.5. This calculation used the methods described in *Estimation of Neutron Dose Rates from Nuclear Waste Packages* (Nelson 1996) and is outlined in Part B, Section 5.8.1.

A scoping calculation of the neutron dose rate was done using the Nelson method. This method produced a very conservative dose rate estimate that equaled nearly 20 percent of the smallest computed gamma dose rate. This percentage was considered too large and therefore a MCNP calculation was done to more accurately estimate the neutron dose rate. The MCNP calculation shows that the neutron doses are less by a factor of 20 or more than those estimated by the Nelson method. The neutron dose rates computed by MCNP are given in Table B5-11 of Part B, Section 5.8.1. For each of the points where gamma dose rates are computed, the MCNP neutron dose rates are less by at least a factor of 300. The MCNP computed dose rates are smaller than those computed with the Nelson method because now the moderating and shielding provided by the steel box and the impact limiter is included in the calculation. Also, with MCNP the source is represented accurately geometrically, rather than as a point as required by the Nelson method. The neutron source energy was taken as 4 MeV because this was found to

produce higher dose rates than a scoping calculation done with a ^{240}Pu spontaneous fission neutron spectrum and because the neutrons produced by (α, n) reactions in oxygen are less energetic than 4 MeV.

As a result of the MCNP calculations, the neutron dose rates can be neglected compared to the gamma dose rates and are not included in the shielding analysis.

5.3 SUMMARY OF SHIELDING PROPERTIES OF MATERIALS

The shielding attenuation properties for the bulk materials used in this analysis are contained in the photon cross-section libraries that comprise the MCNP computer code system (Briesmeister 1997). The composition and densities of the shielding materials used in the calculational models are given in Part B, Section 5.4.3. Figure B5-1 is an elevation view of the MCNP model for the SWP. It shows that the primary gamma-ray shield is the steel box. Shielding is also provided by the SWDB Overpack and Impact Limiter. A 7.62 cm (3-in.-) thick plate has been added to the bottom of the SWDB Overpack to supplement what initial analysis showed to be deficient shielding at the base of the steel box. This plate appears as part of the steel box in Figure B5-1. A 2.54 cm (1-in.-) thick steel plate that is part of a frame on the transport vehicle that supports the SWP during transport is included at the base of the SWDB Impact Limiter.

5.4 NCT

5.4.1 Conditions To Be Evaluated

The dose rates will be calculated at the outer surfaces of the SWDB Impact Limiter. These surfaces include those on the long and short sides and at the top and bottom. The bottom surface is taken to be the bottom of the support plate of the frame into which the SWP is placed on the transport vehicle. The 2.54 cm (1-in.-) thick support plate must be in place during transport. The dose rates will also be computed at 2 m from the side and top surfaces, 1 m from the bottom (at the ground surface), and 4 m from the surface in the long direction, a conservative location for the vehicle driver.

Figure B5-1. Elevation View of Monte Carlo N-Particle Geometry for the Steel Waste Package Gamma Dose Rate Simulation Model for the Engineered Containers Payload.

5.4.2 Acceptance Criteria

The placement of the SWP on the truck bed during NCT could allow workers and the public to be exposed to both lateral and axial doses. However, it is unlikely that worker or public exposure would occur in the axial direction because these locations are not easily accessible. The lateral dose rates from the package must meet the limits of 10 mSv/h (1000 mrem/h) on the exterior surface of the package and 0.1 mSv/h (10 mrem/h) at a distance of 2 m from the package. Axial dose rates may exceed these limits but in such cases should be made inaccessible by administrative controls. These controls will be in accordance with HNF-5173, *PHMC Radiological Control Manual* (Giltz 2000). Additionally, the dose rate in any normally occupied space of the transfer vehicle must be less than 0.02 mSv/h (2 mrem/h) if the worker has no radiological training or 0.05 mSv/h (5 mrem/h) if the worker is radiologically trained.

5.4.3 Shielding Model

5.4.3.1 Assumptions and Information Used.

- The photon source was derived from the bounding radionuclide inventory for the shielding analysis as defined in Part B, Section 2.1, Table B2-3.
- The ANSI/ANS-6.1.1-1991 (ANSI/ANS 1991) flux-to-dose conversion factors were as given in Table B5-3 where these conservatively assume an anterior-posterior exposure.
- The carbon steel box dimensions are those reported in a previous analysis by Carter (Mercado 1998) and as verified from engineering drawings.
- The SWDB Overpack dimensions are from a new design (E-D-003 [see Part A, Section 10.0]).
- The inner dimensions of the SWDB Impact Limiter are from Table A2-1 of Crow (1996).
- Outer dimensions of the SWDB Impact Limiter were computed assuming an 20.3 cm (8-in.) thickness of wood and 0.64 cm (0.25-in.) thicknesses of carbon steel for inner and outer walls.
- The dispersible source material is assumed to be in eight stainless steel containers as specified in Mercado (1998). The silica composition and mass density of 1.15 g/cm³ for the source material was suggested in Part B, Section 2.3, of Crow (1996).

- The structural pieces source material is assumed to have an associated mass of 362.8 kg (800 lb) and consist of silicon dioxide, a conservative choice of material with respect to gamma-ray attenuation. This mass is the minimum allowed and is about 29 percent of the EC's payload mass.
- Material densities and elemental mass fractions are as shown in Table B5-4.

Table B5-3. Photon Fluence-to-Dose Conversion Factors.

Energy (MeV)	Flux to dose rate (mrem/h)/(photon/cm ² /s)	Energy (MeV)	Flux to dose rate (mrem/h)/(photon/cm ² /s)
1.00 E-02	2.232 E-05	5.00 E-01	9.144 E-04
1.50 E-02	5.625 E-05	6.00 E-01	1.076 E-03
2.00 E-02	8.568 E-05	8.00 E-01	1.379 E-03
3.00 E-02	1.184 E-04	1.00 E+00	1.656 E-03
4.00 E-02	1.314 E-04	1.50 E+00	2.246 E-03
5.00 E-02	1.382 E-04	2.00 E+00	2.758 E-03
6.00 E-02	1.440 E-04	3.00 E+00	3.672 E-03
8.00 E-02	1.624 E-04	4.00 E+00	4.500 E-03
1.00 E-01	1.919 E-04	5.00 E+00	5.292 E-03
1.50 E-01	2.797 E-04	6.00 E+00	6.012 E-03
2.00 E-01	3.708 E-04	8.00 E+00	7.488 E-03
3.00 E-01	5.616 E-04	1.00 E+01	8.892 E-03
4.00 E-01	7.416 E-04	1.20 E+01	1.040 E-02

Source: ANSI/ANS, 1991, *Neutron and Gamma-ray Fluence-to-dose Factors*, ANSI/ANS-6.1.1-1991, American National Standards Institute/American Nuclear Society, New York, New York.

Table B5-4. Material Composition and Densities for Monte Carlo N-Particle Model.

Material	Density (g/cm ³)	Element	Mass Fraction
Carbon steel	7.85	Iron	0.9775
		Chromium	0.01
		Manganese	0.0075
		Molybdenum	0.002
		Carbon	0.003
Stainless steel	8.00	Iron	0.6872
		Chromium	0.19
		Nickel	0.0925
		Manganese	0.02
		Phosphorus	0.01
		Carbon	0.0003
Structural "Pieces" source material	0.188	Silicon	0.3333 ^a
		Oxygen	0.6667 ^a
Dispersible source material	1.15	Silicon	0.3333 ^a
		Oxygen	0.6667 ^a
Douglas fir	0.512 ^b	Hydrogen	0.0622
(Celotex) ^c		Carbon	0.4444
		Oxygen	0.4934
Air (sea level)	1.23 E-03	Nitrogen	0.78
		Oxygen	0.22

^aThese are atom fractions.

^bDensity from CRC, 1961, *Handbook of Chemistry and Physics*, 43rd Edition, Chemical Rubber Publishing Company, Cleveland, Ohio, p 1569.

^cCelotex composition from Harmon, C. D. II, R. D. Busch, J. F. Briesmeister, and R. A. Forster, *Criticality Calculations with MCNPTM: A Primer*, LA-12827-M, Los Alamos National Laboratory, Los Alamos, New Mexico.

Celotex is a registered trademark of The Celotex Corporation, Tampa, Florida.

5.4.3.2 Shielding Model for Dispersible Material in ECs. Plan and elevation views of the SWP shielding model are shown in Figures B5-1 and Figures B5-2, respectively. The model is based on the assumptions and information given in Part B, Section 5.4.3.1. Material composition and mass densities are given in Table B5-4.

The eight ECs are placed inside the steel box according to the payload configuration used by Carter (Mercado 1998). As shown in Figure B5-1, the containers are arranged so that their outer surfaces are just touching. The 500,000 Ci of source material is uniformly distributed within each EC. The gamma-ray energy spectrum is given in Table B5-2. The thin steel liners identified as “Rectangular Grout Container” and “Rectangular Overpack Disposal Container” in referenced drawings (drawings H-3-70454 [PNL 1993a] and H-3-70453 [PNL 1992, 1993b]) into which the ECs are placed prior to loading into the steel box are not explicitly included in the MCNP model. However, the steel of these liners is included in the wall thickness of the steel box. The dimensions of the MCNP model components are listed below.

- Source containers for dispersible material:
 - Eight stainless steel cylinders 27.31 cm (10.75 in.) outside diameter
 - 0.419 cm (0.165-in.) wall, 152.4 cm (60 in.) long
 - 0.9525 cm (0.375-in.-) thick plugs at each end
- Dispersible source material:

Silica dust with mass density 1.15 g/cm³ filling interior volume of cylindrical source containers
- Inside of carbon steel box containing the payload (liners added inside to top and sides, but liner bottom added to shield plate thickness of overpack):

196.56 cm (77.385 in.) long by 105.12 cm (41.385 in.) wide by 94 cm (37 in.) high
- Carbon steel box wall thicknesses (liners included in sides and top):
 - 23.641 cm (9.3075-in.) sides, 18.256 cm (7.1875-in.) top,
 - 10.16 cm (4.0-in.) bottom (liner bottom added to overpack)
- Inside of stainless steel overpack that contains the carbon steel box:

255.27 cm (100.5 in.) long by 153.67 cm (60.5 in.) wide by 127.32 cm (50.125 in.) high
- Overpack plate thicknesses:

1.27 cm (0.5-in.) sides, 3.175 cm (1.25-in.) top, 9.8425 cm (3.875-in.) bottom (bottom includes 7.62 cm (3-in.) carbon steel shield, 0.95 cm (0.375 in.) liner thickness, and 1.27 cm (0.5-in.) stainless steel overpack base)

Figure B5-2. Plan View of Monte Carlo N-Particle Geometry for the Steel Waste Package Gamma Dose Rate Simulation Model for the Engineered Containers Payload.

- Inside of impact limiter that contains overpack and steel box with payload:
278.13 cm (109.5 in.) long by 179.07 cm (70.5 in.) wide by 161.04 cm (63.4 in.) high
- Impact limiter consists of inner 0.635 cm (0.25-in.-) thick carbon steel plate
followed by 20.32 cm (8-in.) thickness of Douglas fir
followed by outer 0.635 cm (0.25-in.) thickness of carbon steel plate
- Outside of impact limiter:
321.31 cm (126.5 in.) long by 222.25 cm (87.5 in.) wide by 204.22 cm (80.4-in.) high
- Base plate of carbon steel beneath impact limiter:
2.54 cm (1 in.) thick
- Air surrounds impact limiter above and on sides
- Air between base plate and silica ground surface:
Ground 1 m below base plate, modeled as a 1 m thickness, with a lateral extent of +/- approximately 5 m
- Dose points referenced to outer surfaces of impact limiter except on bottom where they are referenced to the outer surface of the base plate.

As shown in Figures B5-1 and B5-2, the steel shielding box is subdivided into several layers in the MCNP model. This is done so that importance sampling can be employed to improve statistical precision of the computed dose rates. For the initial design of the SWP, the bottom portion of the steel box was thinner than in the present design. Because dose rates were expected to be much greater at the bottom of the SWP, scattering of gamma rays from the ground to the dose points of interest at the sides was a concern. Carter (Mercado 1998) first identified this concern. In order to include ground scatter contributions to the gamma-ray dose rates, a soil material was added at the 1 m distance corresponding to the height of the bottom base plate above the ground when the SWP is loaded onto the transport vehicle.

Scoping calculations were done for the original design with and without the ground-scattering material 1 m beneath the SWP. Results showed that the ground-scattered contributions to the surface dose rates on the sides were about 10 percent of the total, while at the 2 m location the majority of the dose rate was from ground scattering. The dose rates above the SWP are not affected by ground scattering. The scoping calculations further showed that the dose rate limit 1 m below the SWP for the hypothetical accident condition was greatly exceeded.

Results of the scoping calculations led to a design modification in which a 7.62 cm (3-in.-) thick steel plate was added to the bottom of the overpack (E-D-005 [see Part A,

Section 10.0). This dramatically reduces the dose rates below the SWP and causes ground scatter to the side dose points to be very small. Also, the new overpack walls and bottom are 1.27 cm (0.5-in.) thick, which is greater than the original design thickness of 0.64 cm (0.25 in.) (Crow 1996) because it now serves as a containment boundary for the SWP. The thickness of the overpack lid has increased to 3.18 cm (1.25 in.) (E-D-003). All shielding models and dose rate results presented in this analysis are based on the new SWP design. The dimensions presented earlier in this section and the geometry shown in Figures B5-2 and B5-1 are for the new design.

The simulation of gamma-ray transport to the several dose points includes the production of bremsstrahlung by energetic electrons that are produced by gamma-ray interactions in the shielding materials. This is a secondary bremsstrahlung production mechanism that is unrelated to the bremsstrahlung associated with beta decay in the source and computed by ORIGEN-S. Scoping calculations done with and without the use of the bremsstrahlung option in MCNP show that its use increases the simulated dose rates by several percent.

Dose rates are referenced to the outer steel plate surface of the SWDB Impact Limiter, except at the bottom where there is an additional 2.54 cm (1-in.) thickness of steel that represents the base of the frame in which the lower half of the SWDB Impact Limiter rests. Flux-to-dose conversion is done by MCNP using the values in Table B5-3.

5.4.3.3 Shielding Model for Payload Consisting of Structural Pieces. The MCNP model for a payload of structural pieces uses a different source representation. There are no ECs. The payload is placed directly into the RGC that is loaded into the SWP. This payload is also limited to 500,000 Ci, but can consist of a variety of miscellaneous materials: chunks of glass, metal scrap, filter media, steel cans, hoses, manipulator boots, and other items. It was assumed that the payload is a silica material and that the mass is uniformly distributed inside the RGC. The silica mass density of 0.188 g/cm³ corresponds to a payload mass of 362.8 kg (800 lb). This is the minimum payload mass for the 500,000 Ci activity that produces dose rates on the sides that are within the 10 mSv/h (1,000 mrem/h) limit at 1 m for the HAC. A smaller payload mass of this activity will exceed the HAC dose rate limit at an activity of 500,000 Ci. The silica material composition produces bounding dose rate estimates because other materials, such as steel, of the same mass will produce lower dose rates. The MCNP file is identical to that for the EC payload, attached as Part B, Section 5.8.2, except that the source is now uniformly distributed within the interior volume of the steel box.

5.4.4 Shielding Calculations

5.4.4.1 Dose Rates for the EC Payload. The calculated dose rates at each lateral surface of the SWP and at a 2 m distance in each direction are shown in Table B5-5. These dose rates are for the engineered container payload. Also given is the dose rate at 4 m, from the long side, corresponding to the front edge of the trailer on which the SWP is loaded. The driver location will be beyond the 4-m point and therefore this location gives a conservative estimate of the dose rate to the driver.

Table B5-5. Steel Waste Package Gamma-Ray Dose Rates for Engineered Container Payload.

Distance from container surface	Dose rate limit (mrem/h)	Dose rate (mrem/h) ^a	
		Long side	Short side
Contact	1,000	5.83 (0.027)	15.2 (0.015)
2 m	10	0.682 (0.024)	2.18 (0.012)
Driver location (4 m)	2 or 5 ^b	0.235 (0.030)	-----

^aValues in parentheses are fractional statistical uncertainty (one standard deviation) in the Monte Carlo N-Particle- (MCNP-) calculated dose rate (MCNP file 22ton43)

^bThe limit of 2 mrem/h is for a driver without radiological worker training; the 5 mrem/h limit applies to a driver with radiological worker training.

All of the side-surface contact dose rates are well below the 10 mSv/h (1000 mrem/h) limit. Although not subject to these limits, the top and bottom surface-contact dose rates are 0.935 mSv/h (93.5 mrem/h) and 0.625 mSv/h (62.5 mrem/h), respectively. The 2 m dose rate of 0.0218 mSv/h (2.18 mrem/h) for the short sides (direction perpendicular to the length of the transport vehicle) and 0.00682 mSv/h (0.682 mrem/h) for the long sides, along the length of the transport vehicle, are well below the 0.1 mSv/h (10 mrem/h) limit. The 0.00235 mSv/h (0.235 mrem/h) dose rate at the driver location is within the 0.02 mSv/h (2 mrem/h) limit for a worker without radiological training.

The dose rate at 2 m above the top surface is 0.129 mSv/h (12.9 mrem/h). The dose rate at the ground surface directly beneath the SWP is 0.228 mSv/h (22.8 mrem/h). Neither of these locations (above and below the SWP) is subject to the dose limits; and neither location is normally accessible to the public or workers. However, administrative controls should be considered to ensure that both locations are inaccessible to workers and the public.

5.4.4.2 Dose Rates for the Structural “Pieces” Payload. The corresponding dose rates computed for the “pieces” payload are shown in Table B5-6. The homogenized mass density of 0.188 g/cm³ corresponds to the minimum payload mass of 362.8 kg (800 lb). As discussed earlier in part B, Section 5.4.3.3, a payload with the same activity, but lower mass will produce dose rates above the HAC limit and therefore is not allowed. The only exception to this lower mass limit is the case where the activity can be documented as less than 500,000 Ci. In this case, a special analysis would have to be done with the measured payload mass and activity to determine if the dose rate limits are met.

All of the side-surface contact dose rates for the “pieces” payload are well below the 10 mSv/h (1000mrem/h limit). Although not subject to these limits, the top- and bottom-surface contact dose rates are 2.48 mSv/h (248 mrem/h) and 1.44 mSv/h (144 mrem/h), respectively. The 2 m dose rate of 0.0712 mSv/h (7.12 mrem/h) for the short sides (direction perpendicular to the length of the transport vehicle) and 0.0461 mSv/h (4.61 mrem/h) for the long sides, along the length of the transport vehicle, are below the 0.1 mSv/h (10 mrem/h) limit. The 0.0164 mSv/h (1.64 mrem/h) dose rate at the driver location is within the 0.02 mSv/h (2 mrem/h) limit for a worker without radiological training.

For the “pieces” payload, the dose rate at 2 m above the top surface is 0.421 mSv/h (42.1 mrem/h). The dose rate at the ground surface directly beneath the SWP is 0.650 mSv/h (65.0 mrem/h). Neither of these locations (above and below the SWP) is subject to the dose limits; and neither location is normally accessible to the public or workers. However, administrative controls should be considered to ensure that both locations are inaccessible to workers and the public.

Table B5-6. SWP Gamma-Ray Dose Rates for “Pieces” Payload Represented as a Homogeneous Distribution Within the Steel Box and With the Minimum Allowed Mass Density of 0.188 g/cm³.

Distance from container surface	Dose rate limit (mrem/h)	Dose rate (mrem/h) ^a	
		Long side	Short side
Contact	1,000	38.0 (0.016)	38.4 (0.014)
2 m	10	4.61 (0.014)	7.12 (0.010)
Driver location (4 m)	2 or 5 ^b	1.64 (0.019)	----

^aValues in parentheses are fractional statistical uncertainty (one standard deviation) in the Monte Carlo N-Particle- (MCNP-) calculated dose rate (MCNP file 22ton51)

^bThe limit of 2 mrem/h is for a driver without radiological worker training, the 5 mrem/h limit applies to a driver with radiological worker training.

5.5 HAC

5.5.1 Conditions To Be Evaluated

The HAC assumes the loss of the SWDB Impact Limiter that includes the shielding provided by its steel plates and 20.32 cm (8-in.-) thick wood interior, as well as the 2.54 cm (1-in.-) thick steel plate at the bottom that is part of the transport vehicle frame. The dose rates modeled for the HAC are referenced to the outer surfaces of the steel overpack.

Because the gamma-ray dose rates are much greater than the neutron dose rates, as determined in Part B, Section 5.2.3, calculations were only made for the gamma-ray source term.

5.5.2 Acceptance Criteria

The dose rate at a distance of 1 m from any surface of the overpack following an accident in which the impact limiter is lost will not exceed 1,000 mrem/h at 1 m.

5.5.3 Shielding Model

The shielding models for the HAC are slightly modified versions of the NCT model. The material of the SWDB Impact Limiter is replaced with air, and the steel base plate beneath the SWP that is part of the transport vehicle support frame is also replaced with air. The ground is also replaced with air. Ground-scattering effects are neglected for the HAC dose rate calculations, so the SWP without SWDB Impact Limiter is surrounded completely with air. A scoping calculation determined that ground scattering is not significant when the SWP is placed on its side on the ground following the accident, at least not for the bounding dose rate at 1 m from the exposed bottom surface. One model contains the engineered payload source, and the other contains the structural “pieces” payload. Dose rates for the HAC are referenced to the intact overpack surfaces.

5.5.4 Shielding Calculations

The dose rates are shown in Table B5-7 for both the EC and the structural “pieces” payloads. The dose rates on all surfaces are within the 10 mSv/h (1000 mrem/h) limit at 1 m. The dose rates are higher for the structural “pieces” payload because the mass of 362.8 kg (800 lb) is at the minimum allowed such that dose rates are just within the HAC limit.

Table B5-7. Steel Waste Package Gamma-Ray Dose Rates for the Accident Condition (Impact Limiter, 2.54 cm (1-in.) Frame Plate, and Ground-Scatter Material Removed).

Distance from overpack surface	Dose rate limit (mrem/h)	Payload	Dose rate (mrem/h)*			
			Long side	Short side	Top	Bottom
1m	1,000	Engineered containers	6.67 (0.023)	21.9 (0.0130)	146 (0.011)	368 (0.009)
1m	1,000	Structural “pieces”	46.0 (0.021)	65.4 (0.016)	441 (0.013)	987 (0.011)

*Values in parentheses are fractional statistical uncertainty (one standard deviation) in the Monte Carlo N-Particle (MCNP) calculated dose rate (MCNP file 22ton42 for engineered containers, 22ton52 for structural pieces)

5.6 SHIELDING EVALUATION AND CONCLUSIONS

5.6.1 EC Payload

The shielding calculations have shown that for NCT and for the 500,000 Ci payload of ECs, the surface dose rates are all less than the 10 CFR 71, "Packaging and Transportation of Radioactive Material," limit of 10 mSv/h (1000 mrem/h).

The shielding calculations have shown that for NCT and for the 500,000 Ci payload of ECs, the dose rates 2 m from all lateral surfaces are less than the 10 CFR 71 limit of 0.1 mSv/h (10 mrem/h).

The computed dose rate of 0.228 mSv/h (22.8 mrem/h) 1 m directly below at ground level is of concern. Access to the region below the SWP should be restricted during NCT.

The gamma dose rate of 0.129 mSv/h (12.9 mrem/h) at 2 m above the SWP is not of concern. This location is not normally accessible by workers or the public. Access to the region above the SWP should be restricted during NCT.

The HAC shielding analysis for the EC payload shows that 10 CFR 71 criteria are met for all 1m locations, including the case when the accident causes the SWP to rest on its side so that the bottom surface is exposed.

5.6.2 Structural "Pieces" Payload

For a structural "pieces" payload with the minimum mass of 362.8 kg (800 lb), the computed surface dose rates of the SWP are all within 10 CFR 71 limits. Larger payload masses (with a 500,000 Ci activity) will result in lower dose rates and are allowed. The mass must not be less than the 362.8 kg (800 lb) minimum.

The dose rates of 0.421 mSv/h (42.1 mrem/h) at 2 m from the top and the dose rate of 0.65 mSv/h (65.0 mrem/h) below the SWP at the 1 m location of the ground level are considered excessive. Although the dose rates in these axial directions are not limited by the acceptance criteria, it is recommended that administrative controls be implemented to restrict access to the regions above and below the SWP during NCT for both types of payload.

The HAC shielding analysis for the minimum mass structural "pieces" payload shows that 10 CFR 71 criteria are met for all surfaces, including the case when the accident causes the SWP to rest on its side so that the bottom surface is exposed.

5.7 REFERENCES

- 10 CFR 71, "Packaging and Transportation of Radioactive Material," *Code of Federal Regulations*, as amended.
- ANSI/ANS, 1991, *Neutron and Gamma-ray Fluence-to-dose Factors*, ANSI/ANS-6.1.1-1991, American National Standards Institute/American Nuclear Society, New York, New York.
- Briesmeister, J. F., editor, 1997, *MCNP – A General Monte Carlo N-Particle Transport Code, Version 4B*, LA -12625-M, Los Alamos National Laboratory, Los Alamos, New Mexico.
- Bowman, S. M., 1997, *SCALE 4.3 Code Collection*, NUREG/CR-0200 (under Oak Ridge National Laboratory Contract to the U.S. Nuclear Regulatory Commission), U.S. Nuclear Regulatory Commission, Washington, D.C. (Documentation is contained on EBU-SQA-001-CD.)
- CRC, 1961, *Handbook of Chemistry and Physics*, 43rd Edition, Chemical Rubber Publishing Company, Cleveland, Ohio.
- Crow, S. R., 1996, *Addendum to the Safety Analysis Report for the Steel Waste Packaging*, WHC-SD-TP-DA-004, Rev. 1, Westinghouse Hanford Company, Richland, Washington.
- Giltz, T. P., 2000, *PHMC Radiological Control Manual*, HNF-5173, Rev. 0, Fluor Hanford, Richland, Washington.
- Harmon, C. D. II, R. D. Busch, J. F. Briesmeister, and R. A. Forster, 1994, *Criticality Calculations with MCNPTM: A Primer*, LA-12827-M, Los Alamos National Laboratory, Los Alamos, New Mexico.
- McCoy, J. C., 1998a, *WMNW Computer Program Verification for MCNP4B*, EBU-SQA-002, Rev. 1, Waste Management Federal Services, Inc., Northwest Operations, Richland, Washington.
- McCoy, J. C., 1998b, *WMNW Computer Program Verification for SCALE 4.3*, EBU-SQA-001, Rev. 1, Waste Management Federal Services, Inc., Northwest Operations, Richland, Washington.
- Mercado, J. E., 1998, *Scoping Thermal and Shielding Evaluation for the 22.5 Ton Box*, EBU-RPT-008, Rev. 0, Waste Management Northwest, Richland, Washington.
- Nelson, J. V., 1996, *Estimation of Neutron Dose Rates from Nuclear Waste Packages* (internal memo 8M730-JVN-96-007 to J. R. Green, March 8, 1996), Westinghouse Hanford Company, Richland, Washington.

5.8 APPENDICES

5.8.1 Neutron Dose Rate Calculations

In the method described in *Estimation of Neutron Dose Rates from Nuclear Waste Packages* (Nelson 1996), the total neutron source term $S(ST)$, which accounts for neutron multiplication, is determined by adding the spontaneous fission source term ($S(SF)$) and the (α, n) source term ($S(\alpha, n)$) and dividing by 1 minus the k_{eff} .

$$S(ST) = \frac{(S(SF) + S(\alpha, n))}{(1 - k_{eff})}$$

$S(SF)$ and $S(\alpha, n)$ are computed using neutron production rate tables in Nelson (1996). After $S(ST)$ is determined, it is used to determine the dose rate in the equation;

$$D(r) = \frac{0.01 \times S(T)}{r^2}$$

where r is the distance from the source and $D(r)$ is the dose rate in mrem/h as a function of r .

The total neutron production from the Steel Waste Package payload is shown in Table B5-8 for each of the transuranic radionuclides. The $S(ST)$ equation above was used to compute these values, along with the production rates of Tables 1 and 3 in Nelson (1996), and the activities shown in Table B5-1. The (α, n) neutron source term was calculated assuming that the source material was in a pure oxygen matrix. This is a conservative assumption because it does not take credit for alpha particle slowing down in other elements present in the source material. Because the allowed quantities of the fissile radionuclides are much less than required for criticality for even the most favorable conditions, the subcritical multiplication factor is conservatively assumed to be 0.5. Then the $S(ST)$ equation gives a total neutron source of 1.61×10^6 n/s.

Table B5-8. Neutron Source Term for Steel Waste Package Payload.

Isotope	Activity (Ci)	Neutron production rate (n/s)
²³⁴ U	1.09 E-03	1.61
²³⁸ Pu	1.26 E+02	3.25 E+05
²³⁹ Pu	9.54 E+00	1.86 E+04
²⁴⁰ Pu	6.36 E+00	3.74 E+04
²⁴¹ Am	1.75 E+02	4.25 E+05
Total	---	8.06 E+05

The D(r) equation was used to estimate the neutron dose rate at the outer surfaces of the Steel Waste Disposal Box Impact Limiter assuming the source is located at the midpoint of the actual payload. The values are shown in Table B5-9.

Table B5-9. Neutron Dose Rate for Steel Waste Package Payload Using the Nelson (1996) Method.

Dose point location	Distance from source midpoint (cm)	Neutron dose rate (mrem/h)
Long-dimension side surface	160.7	0.6
Short-dimension side surface	111	1.3
Top surface	133.4	0.9
Bottom surface	82.0	2.4

Nelson, J. V., 1996, *Estimation of Neutron Dose Rates from Nuclear Waste Packages* (internal memo 8M730-JVN-96-007 to J. R. Green, March 8, 1996), Westinghouse Hanford Company, Richland, Washington.

Because these estimates are more than 5 percent of the gamma doses computed at the side surfaces, it is necessary to use a more sophisticated neutron dose rate calculation with the Monte Carlo N-Particle (MCNP) code (Briesmeister 1997). An MCNP calculation for the same dose points was done that includes the shielding provided by the materials surrounding the payload and the detailed geometry of the payload. Using a neutron energy of 4 MeV, which produces conservative dose rate estimates (greater than those obtained with spontaneous fission and α, n sources) the dose rates of Table B5-11 are obtained. The lines of the input file that must replace those in the MCNP gamma model in Part B, Section 5.8.2, to give the neutron model are given in Part B, Section 5.8.4. The neutron dose rates in Table B5-10 are much less than those estimated from the simple Nelson method in Table B5-9.

Table B5-10. Neutron Dose Rate for Steel Waste Package
Payload Using Monte Carlo N-Particle.

Dose point location	Neutron dose rate (mrem/h)	Fractional statistical precision
Long-dimension side surface	0.016	0.035
Short-dimension side surface	0.029	0.025
Top surface	0.034	0.041
Bottom surface	0.050	0.036

Source: Monte Carlo N-Particle input file 22ton48.

5.8.1.1 References.

Briesmeister, J. F., editor, 1997, *MCNP – A General Monte Carlo N-Particle Transport Code, Version 4B*, LA -12625-M, Los Alamos National Laboratory, Los Alamos, New Mexico.

Nelson, J. V., 1996, *Estimation of Neutron Dose Rates from Nuclear Waste Packages* (internal memo 8M730-JVN-96-007 to J. R. Green, March 8, 1996), Westinghouse Hanford Company, Richland, Washington.

5.8.2 MCNP Input File 22ton43—SWP Shielding Model for NCT and for the EC Payload

Steel Waste Package: NCT with Engineered Container Payload

c
c This model based on Lee Carter's ibulk file (Carter 1998)
c
c Wilson mods use surface and cell #'s >300
c Wilson modification adds impact limiter and ground scatterer
c
c Overpack/containment design is new and includes added
c shielding at base (Boehnke 1999). Steel box thicknesses
c include thin-walled steel liners into which
c payload is placed
c
c the liners are: "grout" container 0.120" sides, 0.1875" btm
c and "disposal" container 0.1875" sides, btm and lid
c for total: 0.78105 cm sides, 0.9525 cm btm, 0.47625 cm lid
c
c A steel plate that is part of truck attachment frame is included
c in the NCT shielding model and is part of base of impact limiter
c
c Dose rates for NCT referenced to impact limiter
c Dose rates for HAC referenced to overpack
c
c Source from J. Boles, Ci inventory from Table B2-3
c 500,000 Ci total inventory (no Sr-90/Y-90)
c
c Douglas fir composition: assumed same as celotex C6H10O5
c (LANL primer), with 0.512 g/cc density from CRC 43rd ed.
c

c
 c ===== CELL BLOCK =====
 c inside box, 8 source cans, source, steel for each
 1 58 -1.15 -101 -92 91 \$ source
 2 304 -8.00 -111 -93 94 (101:92:-91) \$ steel can
 3 58 -1.15 -102 -92 91 \$ source
 4 304 -8.00 -112 -93 94 (102:92:-91) \$ steel can
 5 58 -1.15 -103 -92 91 \$ source
 6 304 -8.00 -113 -93 94 (103:92:-91) \$ steel can
 7 58 -1.15 -104 -92 91 \$ source
 8 304 -8.00 -114 -93 94 (104:92:-91) \$ steel can
 9 58 -1.15 -105 -92 91 \$ source
 10 304 -8.00 -115 -93 94 (105:92:-91) \$ steel can
 11 58 -1.15 -106 -92 91 \$ source
 12 304 -8.00 -116 -93 94 (106:92:-91) \$ steel can
 13 58 -1.15 -107 -92 91 \$ source
 14 304 -8.00 -117 -93 94 (107:92:-91) \$ steel can
 15 58 -1.15 -108 -92 91 \$ source
 16 304 -8.00 -118 -93 94 (108:92:-91) \$ steel can
 19 7 0.0000512 -1 8 -31 38 -61 68 \$ beyond cans, inside box
 (93:-94:(111 112 113 114 115 116 117 118))
 c carbon steel box -- cells going outward for importances
 31 33 -7.85 -2 9 -32 39 -62 69 (1:-8:31:-38:61:-68)
 32 33 -7.85 -3 10 -33 40 -63 70 (2:-9:32:-39:62:-69)
 33 33 -7.85 -4 11 -34 41 -64 71 (3:-10:33:-40:63:-70)
 34 33 -7.85 -5 12 -35 42 -65 71 (4:-11:34:-41:64) \$ no -z
 35 33 -7.85 -6 13 -36 43 -66 71 (5:-12:35:-42:65) \$ no -z
 36 33 -7.85 -7 14 -37 44 -66 71 (6:-13:36:-43) \$ no -z, +za
 c air-gap between carbon steel box and overpack
 41 7 0.0000512 -16 18 -46 48 -76 71 (7:-14:37:-44:66)
 c overpack with 3" shield at base (includes 3/8" liner btm)
 42 304 -8.0 -17 19 -47 49 -77 792 (16:-18:46:-48:76:-793) \$ sides and top
 (4:-11:34:-41:71) \$ of overpack
 43 33 -7.85 -16 18 -46 48 -71 79 \$ bottom 3.375" shield plate
 431 33 -7.85 -16 18 -46 48 -79 791 \$ bottom shield plate
 433 33 -7.85 -16 18 -46 48 -791 793 \$ bottom shield plate
 432 304 -8.0 -4 11 -34 41 -793 792 \$ bottom 0.5" overpack btm
 c air beyond, first 4 cells for sides and above, next 4 cells are below
 61 7 0.0000512 -22 26 -52 56 -82 86 (501:-508:511:-518:521) \$ to 1 m
 62 7 0.0000512 -23 27 -53 57 -83 86 (22:-26:52:-56:82) \$ to 2 m
 63 7 0.0000512 -24 28 -54 58 -84 86 (23:-27:53:-57:83) \$ to 3 m
 631 7 0.0000512 -29 30 -59 60 -89 86 (24:-28:54:-58:84) \$ to 4 m
 64 7 0.0000512 -99 (29:-30:59:-60:89) \$ beyond
 65 7 0.0000512 -501 508 -511 518 -528 86 \$ to 1 m
 66 58 -1.7 -29 30 -59 60 -86 87 \$ silica for ground scatter to 2 m
 67 7 0.0000512 -29 30 -59 60 -87 88 \$ to 3 m
 68 7 0.0000512 -29 30 -59 60 -88 -99 \$ beyond 3 m
 c gap between overpack and impact limiter
 71 7 0.0000512 505 -504 515 -514 -524 792
 (17:-19:47:-49:77:-792)
 c impact limiter
 c inner steel plate of impact limiter excluding bottom (cell 810)
 81 33 -7.85 -503 506 -523 526 -513 516
 (504:-505:524:-792:514:-515) (-11:4:-41:34:-526:792)
 c Douglas fir timbers between steel plates
 c (density from 43rd CRC, p.1569) excludes bottom (cell 820)
 82 100 -0.512 -502 507 -522 527 -512 517
 (503:-506:523:-526:513:-516) (-11:4:-41:34:-527:526)
 c outer steel plate of impact limiter excludes bottom (cell 830)
 83 33 -7.85 -501 508 -521 528 -511 518
 (502:-507:522:-527:512:-517) (-11:4:-41:34:-528:527)
 810 33 -7.85 11 -4 41 -34 526 -792 \$ btm piece beneath cell 432

820 100 -0.512 11 -4 41 -34 527 -526 Sbtm piece beneath cell 810
 830 33 -7.85 11 -4 41 -34 528 -527 Sbtm piece beneath cell 820

c
 c outside world
 99 0 99

c ===== SURFACE BLOCK =====

c carbon steel box, 78" x 42" x 37" inner, 96" x 60" x 48" outer

1 px 98.279 \$ wall includes 0.78105 cm liner
 2 px 102.87
 3 px 106.68
 4 px 110.49
 5 px 114.30
 6 px 118.11
 7 px 121.92
 8 px -98.279 \$ wall includes 0.78105 cm liner
 9 px -102.87
 10 px -106.68
 11 px -110.49
 12 px -114.30
 13 px -118.11
 14 px -121.92

c overpack

16 px 127.635
 17 px 128.905
 18 px -127.635
 19 px -128.905

c tally surfaces referenced to impact limiter

22 px 260.655
 23 px 360.655 \$ tally, 2 m
 24 px 460.655
 26 px -260.655
 27 px -360.655 \$ tally, 2 m
 28 px -460.655
 29 px 560.655 \$ tally at 4 m added for driver
 30 px -560.655 \$ tally at 4 m added for driver

c impact limiter

501 px 160.655 \$ outer steel plate surface
 502 px 160.02 \$ outer Doug Fir surface
 503 px 139.7 \$ inner steel plate surface
 504 px 139.065 \$ inner steel plate surface
 505 px -139.065 \$ inner steel plate surface
 506 px -139.7 \$ inner steel plate surface
 507 px -160.02 \$ outer Doug Fir surface
 508 px -160.655 \$ outer steel plate surface

c

c carbon steel box, 78" x 42" x 37" inner, 96" x 60" x 48" outer

31 py 52.559 \$ wall includes 0.78105 cm liner
 32 py 57.15
 33 py 60.96
 34 py 64.77
 35 py 68.58
 36 py 72.39
 37 py 76.20
 38 py -52.559 \$ wall includes 0.78105 cm liner
 39 py -57.15
 40 py -60.96
 41 py -64.77
 42 py -68.58
 43 py -72.39
 44 py -76.20

c overpack

46 py 76.835
 47 py 78.105
 48 py -76.835
 49 py -78.105
 c tallies referenced to impact limiter
 52 py 211.125
 53 py 311.125 \$ tally, 2 m
 54 py 411.125
 56 py -211.125
 57 py -311.125 \$ tally, 2 m
 58 py -411.125
 59 py 511.125
 60 py -511.125
 c impact limiter
 511 py 111.125 \$ outer steel plate surface
 512 py 110.49 \$ outer Doug Fir surface
 513 py 90.17 \$ inner steel plate surface
 514 py 89.535 \$ inner steel plate surface
 515 py -89.535 \$ inner steel plate surface
 516 py -90.17 \$ inner steel plate surface
 517 py -110.49 \$ outer Doug Fir surface
 518 py -111.125 \$ outer steel plate surface
 c
 c carbon steel box, 78" x 42" x 37" inner, 96" x 60" x 48" outer
 c 7" lid, 4" bottom
 61 pz 93.5038 \$ includes 0.47625 cm liner steel
 62 pz 97.536
 63 pz 101.092
 64 pz 104.648
 65 pz 108.204
 66 pz 111.76
 68 pz 0.00 \$ floor of steel box
 69 pz -3.387
 70 pz -6.773
 71 pz -10.16
 c overpack (55.25" or 140.335cm overall height)
 76 pz 117.158 \$ inside surf. overpack top
 77 pz 120.333 \$1.25" top thickness per new design
 79 pz -13.16 \$ 3.0" overpack base shield VR surf
 791 pz -16.66 \$ 3.0" overpack base shield VR surf
 793 pz -18.7325 \$ 3.375" overpack base shield btm (includes 3/8" liner)
 792 pz -20.0025 \$ 0.5" overpack bottom
 c -z tallies referenced to btm frame plate of IL
 c the impact limiter rests on this plate
 82 pz 262.6275 \$ tally, 1 m
 83 pz 362.6275 \$ tally, 2 m
 84 pz 462.6275
 86 pz -144.1325 \$ tally, 1 m
 87 pz -244.1325 \$ tally, 2 m
 88 pz -344.1325
 89 pz 562.6275
 c impact limiter
 c the impact limiter surfaces have moved because of 3" steel btm plate
 c and also to account for 3/8" liners -inner IL height is 161.04cm
 521 pz 162.6275 \$ outer steel plate surface
 522 pz 161.9925 \$ outer Doug Fir surface
 523 pz 141.6725 \$ inner steel plate surface
 524 pz 141.0375 \$ inner steel plate surface
 c surf 792 at -20.0025 is the inner surf of IL inner steel plate
 c btm IL surfaces moved due to 3" steel + 3/8" liners -also move top IL surfaces
 526 pz -20.6375 \$ inner steel plate surface (.25" plate)
 527 pz -40.9575 \$ outer Doug Fir surface (8")

```

528 pz -44.1325 $ outer steel plate+1"frame base (1.25")
c
c   surfaces for source
91 px -74.6125 $ -x end of source with 0.625" plug
92 px 74.6125 $ +x end of source with 0.625" plug
93 px 76.20 $ +x end of pipes (60 in. long)
94 px -76.20 $ -x end of pipes (60 in. long)
c   outside world
99 so 1000.0
c   surfaces for source, inner radius first
101 c/x -27.5000 13.7500 13.2334
102 c/x 0.0 13.7500 13.2334
103 c/x 27.5000 13.7500 13.2334
104 c/x -13.7500 37.5657 13.2334
105 c/x 13.7500 37.5657 13.2334
106 c/x -27.5000 61.3814 13.2334
107 c/x 0.0 61.3814 13.2334
108 c/x 27.5000 61.3814 13.2334
111 c/x -27.5000 13.7500 13.6525
112 c/x 0.0 13.7500 13.6525
113 c/x 27.5000 13.7500 13.6525
114 c/x -13.7500 37.5657 13.6525
115 c/x 13.7500 37.5657 13.6525
116 c/x -27.5000 61.3814 13.6525
117 c/x 0.0 61.3814 13.6525
118 c/x 27.5000 61.3814 13.6525
c   tally circles
201 c/x 0.0 40.0 30.0 $ for contact
202 c/x 0.0 40.0 60.0 $ in air
211 c/y 0.0 40.0 30.0 $ for contact
212 c/y 0.0 40.0 60.0 $ in air
221 cz 30.0 $ for contact
222 cz 60.0 $ in air

c =====DATA BLOCK=====
c nps 10000
ctme 800
print 10 110
prtmp j -800
mode p
imp:p 1 16r 2 8 50 300 2+3 1+4 1+4 $ends with cell 41 (gap box/ovpk)
      2+4 200 1+3 5+3 7+3 $ends with cell 432 (btm ovpk)
      20+4 4r 9+4 9+4 2r $ends with cell 68
      2+4 4+4 12+4 20+4 3+4 7+4 9+4 0 $starts w. gap ovpk/IL
c imp:p 1 16r 2 14 100 800 7000 6+4 6+4 8+4 128
c      6+5 3r 1024 3r 8+4 2+5 3+5 6+5 0
phys:p j 0
sdef erg=d7 cel=d1 pos fcel d2 axs=1 0 0 ext=d3
      rad=d4 wgt=8.26e+15
si1 1 1 3 5 7 9 11 13 15
sp1 d 1 7r
ds2 1 0.0 -27.5000 13.7500 0.0 0.0 13.7500
      0.0 27.5000 13.7500 0.0 -13.7500 37.5657
      0.0 13.7500 37.5657 0.0 -27.5000 61.3814
      0.0 0.0000 61.3814 0.0 27.5000 61.3814
si3 -74.6125 74.6125
sp3 -21 0
si4 13.2334
sp4 -21 1
sc7 ORIGEN source from J.Boles with Ci inventories of Table B2-3
# si7 sp7
  0.0 0.0

```

```

5.0000E-02  0.0
1.0000E-01  1.2950E+14
2.0000E-01  5.58E+13
3.0000E-01  1.14E+13
4.0000E-01  3.87E+12
5.0000E-01  8.65E+11
6.6163E-01  7.60E+11
6.6165E-01  8.06E+15
7.2329E-01  1.98E+11
7.2331E-01  8.25E+11
8.7318E-01  2.71E+11
8.7320E-01  4.81E+11
9.9631E-01  7.11E+10
9.9633E-01  4.31E+11
1.0047E+00  2.67E+09
1.0049E+00  7.27E+11
1.1731E+00  3.30E+10
1.1733E+00  9.91E+10
1.2744E+00  3.91E+10
1.2746E+00  1.48E+12
1.3324E+00  1.50E+09
1.3326E+00  9.91E+10
1.5000E+00  3.15E+10
2.0000E+00  6.61E+10
2.5000E+00  1.06E+06
c      total 8.26E+15
c
m7  8000.02p 0.22000 7000.02p 0.78000 $ Air
m33 26000.02p -9775 24000.02p -.010 25000.02p -.0075
    42000.02p -.002 6000.02p -.003 $ carbon steel
m58 8000.02p 0.66667 14000.02p 0.33333 $ simulate glass
c  SS-304L from Nuclear Systems Materials Handbook Rev. 36
m304 6000.02p -0.0003 26000.02p -0.6872 15000.02p -0.01
    25000.02p -0.02 28000.02p -0.0925 24000.02p -0.19
c  Douglas fir (celotex composition-LANL criticality primer)
m100 1000.02p -.0622 6000.02p -.4444 8000.02p -.04934
c  ansi/ans-6.1.1-1991 AP fluence-to-dose,photons(mrem/h/(p/cm**2/s))
de0  log .01 .015 .02 .03 .04 .05
    .06 .08 .10 .15 .20 .30
    .40 .50 .60 .80 1.0 1.5
    2.0 3.0 4.0 5.0 6.0 8.0
    10. 12.
df0  log 2.232e-5 5.652e-5 8.568e-5 1.184e-4 1.314e-4 1.382e-4
    1.440e-4 1.624e-4 1.919e-4 2.797e-4 3.708e-4 5.616e-4
    7.416e-4 9.144e-4 1.076e-3 1.379e-3 1.656e-3 2.246e-3
    2.758e-3 3.672e-3 4.500e-3 5.292e-3 6.012e-3 7.488e-3
    8.892e-3 1.040e-2
fc2  dose (mrem/h) on contact x surface, 1st entry
f2:p 501 508 t
fs2  -201
sd2  2827.4 1. 2827.4 1. 5654.8 1.
fc22 dose (mrem/h) at 2 meter in x, 1st entry
f22:p 23 27 t
fs22 -202
sd22 11310. 1. 11310. 1. 22620. 1.
fc42 dose (mrem/h) at 4 meter in x, 1st entry
f42:p 29 30 t
fs42 -202
sd42 11310. 1. 11310. 1. 22620. 1.
fc102 dose (mrem/h) on contact y surface, 1st entry
f102:p 511 518 t
fs102 -211

```

```

sd102 2827.4 1. 2827.4 1. 5654.8 1.
fc122 dose (mrem/h) at 2 meter in y, 1st entry
f122:p 53 57 t
fs122 -212
sd122 11310. 1. 11310. 1. 22620. 1.
fc202 dose (mrem/h) on contact z surface, 1st entry
f202:p 521 528 t
fs202 -221
sd202 2827.4 1. 2827.4 1. 5654.8 1.
fc212 dose (mrem/h) at 1 meter in z, 1st entry
f212:p 82 86 t
fs212 -222
sd212 11310. 1. 11310. 1. 22620. 1.
fc222 dose (mrem/h) at 2 meter in z, 1st entry
f222:p 83 87 t
fs222 -222
sd222 11310. 1. 11310. 1. 22620. 1.

```

5.8.3 ORIGEN-S Input File for Calculation of the SWP Source Energy Spectrum Used in the MCNP Models

```

#ORIGENS
0$$$ E T
DECAY CASE
3$$$ 21 1 1 0 A16 4 A33 25 E T
35$$$ 0 T
56$$$ A2 1 A6 1 A10 0 A13 9 A15 3 E
57** 0 E T
22.5 TON SWP SHIELDING SOURCE TERM
500 KCI TOTAL INVENTORY
60** .1
61** F1E-3
65$$$
'GRAM-ATOMS GRAMS CURIES WATTS-ALL WATTS-GAMMA
  3Z 3Z 1 0 0 3Z 3Z 6Z
  3Z 3Z 1 0 0 3Z 3Z 6Z
  3Z 3Z 1 0 0 3Z 3Z 6Z
81$$$ 2 0 26 1 E
82$$$ 2
83** 1.E+7 3.E+6 2.5E+6 2.E+6 1.5E+6 1.3326E+6 1.3324E+6 1.2746E+6
1.2744E+6 1.1733E+6 1.1731E+6 1.0049E+6 1.0047E+6 9.9633E+5 9.9631E+5
8.732E+5 8.7318E+5 7.2331E+5 7.2329E+5 6.6165E+5 6.6163E+5 5.E+5 4.E+5
3.E+5 2.E+5 1.E+5
73$$$ 270600 551370 561371 631540 922340 942380 942390 942400
952410
74** 2.68 2.567E5 2.429E5 113 1.09E-3 126 9.54 6.36 175
75$$$ 1 1 1 1 2 2 2 2 2 T
22.5 TON SWP SHIELDING SOURCE TERM TIMESTEP 1
56$$$ F0 T
END

```

5.8.4 MCNP Input File 22ton48. SWP Neutron Shielding Model for the EC Payload—Only Those Input Lines Differing From 22ton43 Are Included

Steel Waste Package Neutron Shielding Model

```

c
c   neutron src, 4 MeV
c
c   =====DATA BLOCK=====
mode   n
imp:n   1 44r 0
phys:n
sdef   erg=4   cel=d1 pos fcel d2  axs=1 0 0 ext=d3
       rad=d4 wgt=8.06e+5
c   sp8   -3 .83367 4.43166 $ watt spectrum Pu-240
c
m7     8016.50c 0.22000 7014.50c 0.78000      $ Air
m33    26000.55c -.9775 24000.50c -.010 25055.50c -.0075
       42000.50c -.002 6000.50c -.003      $ carbon steel
m58    8016.50c 0.66667 14000.50c 0.33333      $ simulate glass
m304   6000.50c -0.0003 26000.55c -0.6872 15031.60c -0.01
       25055.50c -0.02 28000.50c -0.0925 24000.50c -0.19
m100   1001.50c -.0622 6000.50c -.4444 8016.50c -0.4934
mt100  lwtr.01t
c   ansi/ans-6.1.1-1977 fluence-to-dose,neuts(mrem/h)/(n/cm**2/s)
de0    log 2.5-8 1-7 1-6 1-5 1-4 .001 .01 .1 .5 1 2.5 5 7 10 14 20
df0    log .00367 .00367 .00446 .00454 .00418 .00376 .00356 .0217
       .0926 .132 .125 .156 .147 .147 .208 .227
f2:n   501 508 t
f12:n  22 26 t
f22:n  23 27 t
f32:n  24 28 t
f42:n  29 30 t
f102:n 511 518 t
f112:n 52 56 t
f122:n 53 57 t
f132:n 54 58 t
f202:n 521 528 t
f212:n 82 86 t
f222:n 83 87 t

```

5.8.5 Evaluation of ^{241}Pu , ^{242}Pu , ^{243}Cm , and ^{244}Cm

Recent characterization data (Hill and Hobart 2000) show the possibility of the presence of ^{241}Pu , ^{242}Pu , ^{243}Cm , and ^{244}Cm in wastes originating in B-Cell of the 324 Building. The shielding evaluation did not consider these additional isotopes. This evaluation shows that these isotopes contribute insignificantly to the dose rate outside of the package during NTC and HAC.

The scaling factors (Hill and Hobart 2000) in Table B5-11 were used to obtain the maximum activities of these nuclides for a total payload activity of 500,000 Ci. The scaled activities are based on the expected source term in Table B2-1, which lists 1.65E5 Ci of ^{137}Cs , 9.54E0 Ci of ^{239}Pu , and 6.36 Ci of ^{240}Pu .

Table B5-11. Scaling Factors for ^{241}Pu , ^{242}Pu , ^{243}Cm , and ^{244}Cm .

Nuclide	Activity Scaling Factor	Relative to	Activity (Ci)
^{241}Pu	24.7	$^{239}\text{Pu} + ^{240}\text{Pu}$	3.93 E2
^{242}Pu	8.28 E-4	$^{239}\text{Pu} + ^{240}\text{Pu}$	1.32 E-2
^{243}Cm	2.203 E-5	^{137}Cs	3.64 E0
^{244}Cm	1.520 E-3	^{137}Cs	2.51 E2

The computer code ORIGEN-S of the SCALE version 4.3 code package (Bowman 1997, McCoy 1998) was used to calculate the photon source term in Table B5-12 and the neutron source term in Table B5-13 for these additional isotopes at the scaled activities.

Table B5-12. Photon Source Term for ^{241}Pu , ^{242}Pu , ^{243}Cm , and ^{244}Cm .

Average Energy (MeV)	Source Strength (photons/s)
1.00 E-02	1.51 E+12
3.00 E-02	7.82 E+03
5.50 E-02	2.65 E+09
8.50 E-02	1.13 E+10
1.20 E-01	4.93 E+10
1.70 E-01	1.97 E+03
3.00 E-01	3.26 E+10
6.50 E-01	6.72 E+07
1.13 E+00	2.66 E+07
1.58 E+00	1.30 E+07
2.40 E+00	6.36 E+06
2.80 E+00	3.86 E+06
3.25 E+00	2.28 E+06
3.75 E+00	1.32 E+06
4.25 E+00	7.67 E+05
4.75 E+00	4.44 E+05
5.50 E+00	4.02 E+05

The computer code ISO-PC version 2.1 (Rittmann 1995, 1996) was used to evaluate the gamma dose rate outside the package from these additional isotopes. ISO-PC uses the point kernel integration method to calculate the gamma dose rate around simple geometric shapes due to gamma rays, X-rays, and bremsstrahlung radiation. Only the side with the minimum amount of shielding was evaluated, as this dose rate will bound that of any other side. The side with the minimum amount of shielding is the bottom after HAC, where the impact limiter and the steel plate that is part of the trailer is lost, as described in Part B, Section 5.5.1. This leaves the 7.875 in. of steel from the grout container, rectangular overpack disposal container, steel waste disposal box, and the steel waste disposal box overpack.

The source was assumed to be uniformly distributed throughout the interior volume of the rectangular grout container. The dimensions of the rectangular grout container are taken from Table A2-1. The source matrix was conservatively assumed to be soil at 0.4 g/cm^3 , which gives a total source mass of 513 kg. This mass is less than what is expected to be shipped, which results in reduced self-shielding, and the material consists of a lower atomic number, which reduces attenuation.

Gamma-ray fluence-to-dose conversion factors were taken from American National Standard ANSI/ANS-6.1.1-1991, *Neutron and Gamma-Ray Fluence-to-Dose Factors* (ANS 1991), for an anterior-to-posterior irradiation pattern and are shown in Table B5-3. This irradiation pattern is the most conservative of all the patterns in ANS (1991).

The gamma dose rate calculated at the bottom surface of the package damaged by the HAC was $3.14\text{E-}08 \text{ Sv/h}$ ($3.14\text{E-}06 \text{ rem/h}$). At 1 m from the bottom surface, the dose rate was $1.46\text{E-}08 \text{ Sv/h}$ ($1.46\text{E-}06 \text{ rem/hr}$). These are both much less than the dose rate calculated for the remainder of the isotopes in Table B5-1, which was $3.68\text{E-}3 \text{ Sv/h}$ (0.368 rem/h) at the bottom surface after HAC. Therefore, it is concluded that, for both NTC and HAC, the isotopes ^{241}Pu , ^{242}Pu , ^{243}Cm , and ^{244}Cm at the activities in Table B5-1 do not contribute significantly to the gamma dose rate outside the package.

The neutron source term was calculated in ORIGEN-S considering production from spontaneous fission and (α, n) reactions. Calculation of the neutrons produced by the absorption of alpha particles requires knowledge of the waste matrix. Neutron production by this mechanism occurs only in elements of low atomic number. The only such element present in the SWP in an appreciable quantity is oxygen. For simplicity and conservatism, a uranium dioxide waste matrix was assumed. Subcritical multiplication of the neutron population due to fissions in fissile nuclei was neglected based on the limit on fissile mass of 200 g spread throughout the waste matrix. Table B5-13 gives the calculated neutron source term for the four additional isotopes.

Table B5-13. Neutron Source Term for ^{241}Pu , ^{242}Pu , ^{243}Cm , and ^{244}Cm .

Nuclide	Activity (Ci)	(α, n)	SF	Total
^{241}Pu	3.93 E+02	5.22 E+00	1.88 E-01	5.41 E+00
^{242}Pu	1.32 E-02	7.39 E+00	5.76 E+03	5.77 E+03
^{243}Cm	3.64 E+00	3.96 E+03	8.59 E+01	4.05 E+03
^{244}Cm	2.51 E+02	2.65 E+05	3.46 E+07	3.49 E+07
Total		2.69 E+05	3.46 E+07	3.49 E+07

The dose rate from neutrons, excluding the contributions from ^{241}Pu , ^{242}Pu , ^{243}Cm , and ^{244}Cm , was evaluated in Section 5.8.1 and given in Table B5-10. Because the model used simplistic assumptions for the neutron energy spectrum, the dose rates evaluated previously can simply be scaled up by the ratio of the total neutron source strength (from Tables B5-8 and B5-13) to the source strength without ^{241}Pu , ^{242}Pu , ^{243}Cm , and ^{244}Cm . This ratio is 43.3 ($=[3.49\text{E}7+8.06\text{E}5]/8.06\text{E}5$). Table B5-14 gives the scaled neutron dose rates at the surface of

the package during NCT. Dose rates after HAC are limited at 1 m from the damaged package surface and would be less.

Table B5-14. Neutron Dose Rates Including ^{241}Pu , ^{242}Pu , ^{243}Cm , and ^{244}Cm .

Detector Location	Neutron Dose Rate (mrem/h)	Fractional Statistical Precision
Long-dimension side surface	0.693	0.035
Short-dimension side surface	1.26	0.025
Top surface	1.47	0.041
Bottom surface	2.17	0.036

Therefore, it is concluded that, for both NTC and HAC, the isotopes ^{241}Pu , ^{242}Pu , ^{243}Cm , and ^{244}Cm at the activities in Table B5-1 do not contribute significantly to the neutron dose rate outside the package.

5.8.6 References

- ANS, 1991, *Neutron and Gamma-Ray Fluence-to-Dose Factors*, ANSI/ANS-6.1.1-1991, American Nuclear Society, La Grange Park, Illinois.
- Bowman, S. M., 1997, *SCALE 4.3 Code Collection*, NUREG/CR-0200 (under Oak Ridge National Laboratory Contract to the U.S. Nuclear Regulatory Commission), U.S. Nuclear Regulatory Commission, Washington, D.C.
- Hill, R. L., and R. L. Hobart, *Data Analysis and Radionuclide Scaling Factor for the B-Cell Waste Stream*, HNF-4904, Rev. 0-A, Fluor Hanford, Richland, Washington.
- McCoy, J. C., 1998, *WMNW Computer Program Verification for SCALE 4.3*, EBU-SQA-001, Revision 1, Waste Management Federal Services, Inc., Northwest Operations, Richland, Washington.
- Rittmann, P. D., 1995, *ISO-PC Version 1.98 - User's Guide*, WHC-SD-WM-UM-030, Westinghouse Hanford Company, Richland, Washington.
- Rittmann, P. D., 1996, "Summary of Changes to ISO-PC Version 2.1," CCC-636, *ISO-PC 2.1: Kernel Integration Code System for General Purpose Isotope Shielding Analyses*, Radiation Safety Information Computational Center, Oak Ridge, Tennessee.

5.8.7 Appendix

5.8.7.1 ORIGEN-S Input File

```
#ORIGENS
0$$ E T
DECAY CASE
3$$ 21 1 1 22 A16 4 A33 25 E T
```

```

35$$$ 0 T
56$$$ A2 1 A6 1 A10 0 A13 4 A15 3 E
57** 0 E T
22.5 TON SWP SHIELDING SOURCE TERM
500 KCI TOTAL INVENTORY
60** .1
61** F1E-3
65$$$
'GRAM-ATOMS GRAMS CURIES WATTS-ALL WATTS-GAMMA
      3Z      3Z      1 0 0      3Z      3Z      6Z
      3Z      3Z      1 0 0      3Z      3Z      6Z
      3Z      3Z      1 0 0      3Z      3Z      6Z
81$$$ 2 0 26 1 E
82$$$ 2
83** 1.E+7 3.E+6 2.5E+6 2.E+6 1.5E+6 1.3326E+6 1.3324E+6 1.2746E+6
      1.2744E+6 1.1733E+6 1.1731E+6 1.0049E+6 1.0047E+6 9.9633E+5 9.9631E+5
      8.732E+5 8.7318E+5 7.2331E+5 7.2329E+5 6.6165E+5 6.6163E+5 5.E+5 4.E+5
      3.E+5 2.E+5 1.E+5
84** 1.492E+7 1.22E+7 1.E+7 8.18E+6 6.36E+6
      4.96E+6 4.06E+6 3.01E+6 2.46E+6 2.35E+6
      1.83E+6 1.11E+6 5.5E+5 1.11E+5 3.35E+3
      5.83E+2 1.01E+2 2.9E+1 1.01E+1 3.059999E+0
      1.12E+0 4.14E-1 9.999998E-3
73$$$ 962440 962430 942410 942420
74** 251 3.64 393 0.0132
75$$$ 2 2 2 2 T
22.5 TON SWP SHIELDING SOURCE TERM TIMESTEP 1
56$$$ F0 T
END

```

5.8.7.2 ISO-PC Input File

```

0      3      bottom - CONTACT 22-ton SWP 0.5 g/cc soil payload
4 new isotopes: Pu-241,Pu-242,Cm-243,Cm-244, at 500kCi total
&Input Next=1, ISpec=3, IGeom=10, ICONC=0, SFACT=1, DUNIT=1,
NTheta=30, NPsi=30, NShld=2, JBuf=2, OPTION=1,
Slth=170.2, y=89.9, X=115., 214., T(1)=83.8, T(2)=20.00,
source(1,1)= 1.52E+12, 1.00E-02, 0.,
source(1,2)= 7.82E+03, 3.00E-02, 0.,
source(1,3)= 2.67E+09, 5.50E-02, 0.,
source(1,4)= 1.14E+10, 8.50E-02, 0.,
source(1,5)= 4.96E+10, 1.20E-01, 0.,
source(1,6)= 1.97E+03, 1.70E-01, 0.,
source(1,7)= 3.28E+10, 3.00E-01, 0.,
source(1,8)= 6.78E+07, 6.50E-01, 0.,
source(1,9)= 2.68E+07, 1.13E+00, 0.,
source(1,10)= 1.31E+07, 1.58E+00, 0.,
source(1,11)= 6.41E+06, 2.40E+00, 0.,
source(1,12)= 3.89E+06, 2.80E+00, 0.,
source(1,13)= 2.30E+06, 3.25E+00, 0.,
source(1,14)= 1.33E+06, 3.75E+00, 0.,
source(1,15)= 7.73E+05, 4.25E+00, 0.,
source(1,16)= 4.48E+05, 4.75E+00, 0.,
source(1,17)= 4.05E+05, 5.50E+00, 0. &
soil 21 0.4

```

1steel 9 7.86

End of Input

&Input Next=6 &

Model of 22 ton box after HAC (most limiting).

Dimensions are for the rectangular grout container, taken from Table A2-1.

Source is for 393 Ci Pu241, 0.0132 Ci Pu242, 3.66 Ci Cm243, 253 Ci Cm244.

See file \origen.arp\swp\swp5.out. These nuclides were not included in

the expected source term in Table B2-1. Activities are based on activity

ratios relative to Cs-137 activity in Table B2-1. Ratios taken from

HNF-4904, Rev. 0-A.

CHECKLIST FOR CHECKING OF ANALYSIS/CALCULATIONS

Document Number/Revision: HNF-4763Document Title: Safety Analysis Report for Packaging (Onsite) Steel Waste Package
Part B Section 5.0, Shielding Evaluation and Part B Section 6.0, Criticality

Yes	No	N/A	
☒	☐	☐	Problem completely defined.
☒	☐	☐	Appropriate analytical method used.
☒	☐	☐	Necessary assumptions are appropriate and explicitly stated.
☒	☐	☐	Computer codes and data files documented.
☒	☐	☐	Data used in calculations explicitly stated in document.
☐	☐	☒	Sources of non-standard formula/data are referenced and the correctness of the reference verified.
☒	☐	☐	Data checked for consistency with original source information as applicable.
☐	☐	☒	Mathematical derivations checked including dimensional consistency of results.
☒	☐	☐	Models appropriate and used within range of validity or use outside range of established validity justified.
☒	☐	☐	Hand calculations checked for errors.
☒	☐	☐	Code run streams correct and consistent with analysis documentation.
☒	☐	☐	Code output consistent with input and with results reported in analysis documentation.
☒	☐	☐	Acceptability limits on analytical results applicable and supported. Limits checked against sources.
☒	☐	☐	Safety Margins consistent with good engineering practices.
☒	☐	☐	Conclusions consistent with analytical results and applicable limits.
☒	☐	☐	Results and conclusions address all points required in the problem statement.

I have checked the analysis/calculation, and it is complete and accurate to the best of my knowledge.

Engineer/Checker J. L. Boles Date 6/28/99**NOTE:** Any hand calculations, notes, or summaries generated as part of this check should be signed, dated, and attached to this checklist. Material should be labeled and recorded so that it is intelligible to a technically qualified third party.

CHECKLIST FOR CHECKING OF ANALYSIS/CALCULATIONS

Document Number/Revision: HNF-4763, Rev 0Document Title: Safety Analysis Report for Packaging (Onsite) Steel Waste PackageSection: Contents - Part B Section 2 (Contents), Section 5.8.5 (Shielding) for addition of 4 new isotopes (^{241}Pu , ^{242}Pu , ^{243}Cm , ^{244}Cm)

Yes	No	N/A	
☒	☐	☐	Problem completely defined.
☒	☐	☐	Appropriate analytical method used.
☒	☐	☐	Necessary assumptions are appropriate and explicitly stated.
☒	☐	☐	Computer codes and data files documented.
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☐	☐	☒	Sources of non-standard formula/data are referenced and the correctness of the reference verified.
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☒	☐	☐	Conclusions consistent with analytical results and applicable limits.
☒	☐	☐	Results and conclusions address all points required in the problem statement.

I have checked the analysis/calculation, and it is complete and accurate to the best of my knowledge.

Engineer/Checker Anthony Savino

Date 5/15/2000

6.0 CRITICALITY EVALUATION

The maximum allowed inventory of fissile material as given in Part B, Section 2.0, is 200 g. The subcritical mass limit for ^{239}Pu when present as an aqueous mixture is 480 g as given in ANSI/ANS-8.1-1983, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors* (ANSI/ANS 1983). This is a factor of 2.4 greater than the 200 g of ^{239}Pu allowed in the payload. This implies that 200 g of ^{239}Pu will be subcritical under any conditions of water moderation and reflection. According to Crow (1996) concrete, plastic, and some other common materials can decrease the minimum critical mass of a plutonium system, but not to less than 300 g. Of all the fissile isotopes as defined by 49 CFR 173.403, ^{239}Pu has the smallest minimum critical mass. Therefore, the subcriticality of 200 g of these isotopes is assured, and criticality of the SWP payload is not possible.

Because of the dispersed nature of the plutonium in the payload volume and because of the considerable separation between adjacent SWPs, if three SWPs are placed in closest proximity to each other, there is no credible way in which the plutonium of the three payloads could combine to produce a critical mass.

Therefore, there is no criticality concern for the allowed inventory of fissile material in the SWP.

6.1 REFERENCES

49 CFR 173, "Shippers—General Requirements for Shipments and Packagings," *Code of Federal Regulations*, as amended.

ANSI/ANS, 1983, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*, ANSI/ANS-8.1-1983, American National Standards Institute/American Nuclear Society, La Grange Park, Illinois.

Crow, S. R., 1996, *Addendum to the Safety Analysis Report for the Steel Waste Packaging*, WHC-SD-TP-DA-004, Rev. 1, Westinghouse Hanford Company, Richland, Washington.

CHECKLIST FOR CHECKING OF ANALYSIS/CALCULATIONS

Document Number/Revision: HNF-4763Document Title: Safety Analysis Report for Packaging (Onsite) Steel Waste Package
Part B Section 5.0, Shielding Evaluation and Part B Section 6.0, Criticality

Yes	No	N/A	
☒	☐	☐	Problem completely defined.
☒	☐	☐	Appropriate analytical method used.
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☐	☐	☒	Sources of non-standard formula/data are referenced and the correctness of the reference verified.
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☒	☐	☐	Conclusions consistent with analytical results and applicable limits.
☒	☐	☐	Results and conclusions address all points required in the problem statement.

I have checked the analysis/calculation, and it is complete and accurate to the best of my knowledge.

Engineer/Checker J. L. BolesDate 6/28/99

NOTE: Any hand calculations, notes, or summaries generated as part of this check should be signed, dated, and attached to this checklist. Material should be labeled and recorded so that it is intelligible to a technically qualified third party.

7.0 STRUCTURAL EVALUATION

7.1 INTRODUCTION

The SWP will transport Type B, HRCQ of fissile remote-handled waste and special-case waste from the 324 Building to the 200 Area for interim storage. Although the 200 Area is the primary destination, the SARP will authorize transportation to any site within the Hanford Site. The contents of the SWP will consist primarily of contaminated scrap equipment removed from B Cell of the 324 Building: racks, tanks, pipes, hoses, valves, tank and pipe heel, filters, lead plugs from cell pipe penetrations, and dispersible material.

7.2 STRUCTURAL EVALUATION OF PACKAGE

7.2.1 Structural Design Features

The SWP is designed and constructed to provide multiple confinement boundaries and a containment boundary during transport. Designed as a system to the construction requirements of BPVC, Section VIII, Division 1 (ASME 1995e), the package meets the minimum transport performance requirements for both NCT and accident conditions within the Hanford Site. The packaging system consists of a series of two or three confinement boundaries (depending on dispersibility of the contents), one within the other, providing multiple boundaries to prevent release of material. Each confinement boundary is vented through Nucfil filters to prevent hydrogen buildup. For dispersible contents, there may be three confinement boundaries. From inside to outside, they are the ECs, the RODC, and the 22-Ton SWDB. The ECs are long tubular members manufactured from stainless steel with a bolted flange on the top and a plate welded to the bottom. In the case of nondispersible contents, the contents are placed in the RGC, which is then placed into the RODC. Up to eight ECs are placed in the open-top RGC, which is then placed in the RODC. Nondispersible debris is placed directly into the RGC. The RODC is a welded carbon steel box equipped with a gasketed closure. The RODC is then placed in the 22-Ton SWDB, which is a thick-walled carbon steel box providing both shielding and confinement. The SWDB is then sealed within the SWDB Overpack, which is the leaktight containment boundary. The SWDB Overpack is constructed of stainless steel and designed as the leaktight containment boundary of the package that meets the NCT and accident condition performance requirements for Hanford Site transport. In transport, the entire assemblage of containers is housed in an SWDB Impact Limiter. This SWDB Impact Limiter is tied to the conveyance and is considered as an integral part of the conveyance.

Major components of the SWDB Overpack containment boundary are constructed from austenitic stainless steel manufactured to the requirements of ASTM A-240, Type 304. As designed, the SWDB Overpack is a rectangular box with a bolted closure. A series of 5 cm (2-in.-) wide stiffeners (which are not part of the containment boundary) are welded to the four vertical sides to stiffen the side panels for pressure loading and to protect the containment boundary during impact. The stiffeners are constructed from stainless steel from ASTM A-240

Type 304 plate bent into a “U” shape with a plate of the same material welded on the backside. These stiffeners are skip welded to the panel, and in the case of the short side, they are skip welded together in pairs. The depth of the stiffeners on the short side is 9.53 cm (3.75 in.) and on the long side is 12.1 cm (4.75 in.). The bottom and side panels of the SWDB Overpack are constructed from 1.3 cm (0.5-in.-) thick plate manufactured to the requirements of ASTM A-240, Type 304 stainless steel. At the top is a 3.18 cm (1.25-in.-) thick, machined flange manufactured from ASTM A-240 or ASTM A-276, Type 304, is welded to the panels. Machined into the flange are two O-ring grooves for the containment seals and a leak test port. The containment boundary seal is provided by the two Viton O-rings. Two 1.9 cm (0.75-in.) round locating pins are provided on the flange for installation of the closure lid. For ease of installation, one locating pin is taller than the other one. Also machined into the flange are 32 through holes for the closure bolts with heavy hex nuts manufactured to ASTM A-194, Grade 8, requirements welded to the underside of the flange. The SWDB Overpack closure lid is a 3.18 cm (1.25-in.-) thick stepped plate constructed from ASTM A-240, Type 304 plate. The closure lid sealing surface landing of the lid is machined to a thickness of 2.87 cm (1.13 in.) with a seal surface finish. The stepped center portion of the plate provides additional shear loading resistance. The closure lid is provided with the two mating locating pin holes for lid installation and 32, $\frac{3}{4}$ -10 UNC ASTM A-325, closure bolts for lid closure. The lid is also equipped with four swivel hoist rings for lift points. In addition, the lid is equipped with two ports, which are a combination leak test, purge, and NucFil ports. Containment sealing of the ports is provided by the closure bolts and two fluorocarbon Stat-O-Seals.

To protect the containment boundary during NCT and accident conditions, SWDB Impact Limiters are provided. The SWDB Impact Limiters are laminated wood encased in a steel casing that is constructed of welded structural steel members and steel plates. As designed, the SWDB Impact Limiters are of a split shell design with the bottom SWDB Impact Limiter permanently mounted and tied down to the conveyance. The SWDB Impact Limiters are equipped with a mechanism for locking the two halves together during transport. Operationally, the SWDB Overpack is installed in the SWDB Impact Limiter prior to any loading operations and remains in the SWDB Impact Limiter after unloading. Inside the SWDB Impact Limiter underneath the SWDB Overpack is a 7.62 cm (3-in.-) thick carbon steel plate for shielding. This shielding plate is manufactured from ASTM A-36 carbon steel and is not attached to either the SWDB Impact Limiter or SWDB Overpack.

7.2.2 Mechanical Properties of Materials

The 22-Ton SWDB Overpack containment boundary is constructed of materials with mechanical properties as shown in Table B7-1. Table B7-2 shows the stress intensities acceptance criteria for NCT, and Tables B7-3 and B7-4 show the allowable stress intensities for ASME (1995d) Service Levels A and C. Table B7-5 shows the stress intensity acceptance criteria for accident conditions, and Table B7-6 shows the allowable stress intensities for Service Level D. In most cases, the mechanical properties are extracted from the BPVC, Section II, Part D (ASME 1995a), for Class 2 and 3 service. For this evaluation, the 304 stainless steel was assumed as ASTM A-240. When mechanical properties were not specifically stated in the ASME (1995a) section, they were developed from room-temperature properties published in the

ASTM standards and expanded relative to temperature by the methods outlined in MIL-HDBK-5G, *Military Handbook-Metallic Materials and Elements for Aerospace Vehicle Structures* (DOD 1994). In the case of the O-ring seals, the properties are extracted from ORD 5700, *Parker O-Ring Handbook* (Parker Seal 1992).

The specified O-rings are manufactured from Viton. This rubber has a manufacturer's working temperature range of -40°C (-40°F) to 204°C (400°F) and will take temperature excursions up to 316°C (600°F) for up to one-half hour without loss of sealing capabilities. The manufacturer's recommended useful age for this material is up to 20 years.

Table B7-1. Steel Waste Package Containment Boundary Properties.

Component/ material/UNS number	Temper- ature (°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)
Overpack assembly and flange/304 SS (SA-240)/ S30400/ ^{18}Cr - ^8Ni	70	30.00	75.00	18.80	28.30	----
	100	29.10	71.25	18.80	28.10	8.55
	200	25.80	60.75	16.70	27.60	8.79
	300	22.50	55.50	15.00	27.00	9.00
	400	20.40	52.50	13.80	26.50	9.19
	500	18.90	51.75	12.90	25.80	9.37
	600	18.30	51.75	12.10	25.30	9.53
Bolts/ASTM A325/1/2<t<1/ Grade 1: K02706, Grade 2: K11900/ C	70	92.00	120.00	23.00	29.30	----
	100	91.08	118.80	23.00	29.10	6.50
	200	87.40	117.60	23.00	28.60	6.67
	300	83.72	116.40	23.00	28.10	6.87
	400	80.04	114.00	23.00	27.50	7.07
	500	76.36	111.60	23.00	27.10	7.25
	600	72.68	106.80	23.00	26.50	7.42

ASTM = American Society for Testing and Materials.

UNS = United Nations Standard

SS = stainless steel.

Table B7-2. Containment Boundary Stress Evaluation Criteria for Normal Conditions of Transport.

	Stress category	Normal conditions of transport	
		Service Level A	Service Level C
Components other than bolts	Primary membrane stress intensity ^(a)	S_{mc}	Greater of 1.2 S_{mc} or 1.0 S_y
	Primary membrane + bending stress intensity ^(a)	$1.5 S_{mc}$	Greater of 1.8 S_{mc} or 1.5 S_y
	Range of primary + secondary stress intensity ^(b)	$3.0 S_{mc}$	$3.6 S_{mc}$
	Pure primary shear stress ^(c)	$0.6 S_{mc}$	$0.6 S_{mc}$
Bolts ^(d)	Average tension	S_{mc}	S_{mc}
	Average shear	$0.6 S_{mc}$	$0.6 S_{mc}$
	Average tension + average shear Stress ratio of average stress/allowable Rt: Stress ratio for average tensile Rs: Stress ratio for average shear	$Rt^2 + Rs^2 < 1$	$Rt^2 + Rs^2 < 1$
	Bolts having tensile strength of >100 ksi Tension + shear + bending + residual torsion	$1.35 S_{mc}$	$1.35 S_{mc}$

^aDefinitions according to NRC, 1978, *Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels*, Regulatory Guide 7.6, Rev. 1, Paragraphs B.4 and C.2, U.S. Nuclear Regulatory Commission, Washington, D.C.

^bDefinitions according to NRC (1978), Paragraph C.4.

^cDefinitions from ASME, 1995d, *Boiler and Pressure Vessel Code*, Section III, Subsection NE-3227.2, American Society of Mechanical Engineers, New York, New York.

^dDefinitions from Mok, G. C., 1989, *Stress Analysis of Closure Bolts for Shipping Casks*, NUREG/CR-6007, Table 6.1 (under Lawrence Livermore National Laboratory contract to the U.S. Nuclear Regulatory Commission), U.S. Nuclear Regulatory Commission, Washington, D.C.

Table B7-3. Containment Boundary Normal Condition Allowable Stresses
for Service Level A, MPa (ksi).

	Stress category	70 °F	100 °F	200 °F	300 °F	400 °F	500 °F	600 °F
Compo- nents other than bolts	Material	304 SS (SA-240)						
	Primary membrane stress intensity	142.58 (20.68)	142.58 (20.68)	126.66 (18.37)	113.76 (16.50)	104.66 (15.18)	97.84 (14.19)	91.77 (13.31)
	Primary membrane + bending stress intensity	213.88 (31.02)	213.88 (31.02)	190.02 (27.56)	170.65 (24.75)	156.99 (22.77)	146.79 (21.29)	137.69 (19.97)
	Range of primary + secondary stress intensity	427.75 (62.04)	427.75 (62.04)	379.97 (55.11)	341.29 (49.50)	313.99 (45.54)	293.51 (42.57)	275.31 (39.93)
	Pure primary shear stress	85.56 (12.41)	85.56 (12.41)	75.98 (11.02)	68.26 (9.90)	62.81 (9.11)	58.67 (8.51)	55.09 (7.99)
Bolts	Material	ASTM A325						
	Average tension stress	174.44 (25.30)	174.44 (25.30)	174.44 (25.30)	174.44 (25.30)	174.44 (25.30)	174.44 (25.30)	174.44 (25.30)
	Average shear stress	104.66 (15.18)	104.66 (15.18)	104.66 (15.18)	104.66 (15.18)	104.66 (15.18)	104.66 (15.18)	104.66 (15.18)
	Average stress	348.87 (50.60)	348.87 (50.60)	348.87 (50.60)	348.87 (50.60)	348.87 (50.60)	348.87 (50.60)	348.87 (50.60)

ASTM = American Society for Testing and Materials.
SS = stainless steel.

Table B7-4. Containment Boundary Normal Condition Allowable Stresses
for Service Level C, MPa (ksi).

	Stress category	70 °F	100 °F	200 °F	300 °F	400 °F	500 °F	600 °F
Compo- nents other than bolts	Material	304 SS (SA-240)						
	Primary membrane stress intensity	206.84 (30.00)	200.64 (29.10)	177.88 (25.80)	155.13 (22.50)	140.65 (20.40)	130.31 (18.90)	126.17 (18.30)
	Primary membrane + bending stress intensity	310.26 (45.00)	300.96 (43.65)	266.83 (38.70)	232.70 (33.75)	210.98 (30.60)	195.47 (28.35)	189.26 (27.45)
	Range of primary + secondary stress intensity	513.31 (74.45)	513.31 (74.45)	455.95 (66.13)	409.55 (59.40)	376.80 (54.65)	352.18 (51.08)	330.40 (47.92)
	Pure primary shear stress	85.56 (12.41)	85.56 (12.41)	75.98 (11.02)	68.26 (9.90)	62.81 (9.11)	58.67 (8.51)	55.09 (7.99)
Bolts	Material	ASTM A325						
	Average tension stress	174.44 (25.30)	174.44 (25.30)	174.44 (25.30)	174.44 (25.30)	174.44 (25.30)	174.44 (25.30)	174.44 (25.30)
	Average shear stress	104.66 (15.18)	104.66 (15.18)	104.66 (15.18)	104.66 (15.18)	104.66 (15.18)	104.66 (15.18)	104.66 (15.18)
	Average stress	348.87 (50.60)	348.87 (50.60)	348.87 (50.60)	348.87 (50.60)	348.87 (50.60)	348.87 (50.60)	348.87 (50.60)

ASTM = American Society for Testing and Materials.
SS = stainless steel.

Table B7-5. Containment Boundary Stress Evaluation Criteria for Accident Conditions.

	Stress category	Accident conditions
		Service Level D
Components other than bolts	Primary membrane stress intensity ^(a)	Lesser of $2.4S_{mc}$ and $0.7S_u$
	Primary membrane + bending stress intensity ^(b)	Lesser of $3.6S_{mc}$ and S_u
	Range of primary + secondary stress intensity ^(b)	----
	Pure primary shear stress ^(c)	$0.42S_u$
Bolts	Average tension	$0.7S_u$ or S_y
	Average shear	$0.6S_y$ or $0.42S_u$
	Average tension + average shear Stress ratio of average stress/allowable Rt: Stress ratio for average tensile Rs: Stress ratio for average shear	$Rt^2 + Rs^2 < 1$

Definitions according to NRC, 1978, *Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels*, Regulatory Guide 7.6, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, D.C.

^aNRC (1978), Paragraph C.6.

^bNRC (1978) Paragraphs C.4 and C.6.

^cDefinitions from ASME, 1995d, *Boiler and Pressure Vessel Code*, Section III, Subsection NE-3227.2, American Society of Mechanical Engineers, New York, New York.

^dDefinitions from Mok, G. C., 1989, *Stress Analysis of Closure Bolts for Shipping Casks*, NUREG/CR-6007, Table 6.3 (under Lawrence Livermore National Laboratory contract to the U.S. Nuclear Regulatory Commission), U.S. Nuclear Regulatory Commission, Washington, D.C.

Table B7-6. Containment Boundary Accident Condition Allowable Stresses for Service Level D, MPa (ksi).

	Stress category	70 °F	100 °F	200 °F	300 °F	400 °F	500 °F	600 °F
Components other than bolts	Material	304 SS (SA-240)						
	Primary membrane stress intensity	342.19 (49.63)	342.19 (49.63)	293.23 (42.53)	267.86 (38.85)	251.18 (36.43)	234.84 (34.06)	220.22 (31.94)
	Primary membrane + bending stress intensity	513.31 (74.45)	491.25 (71.25)	418.86 (60.75)	382.66 (55.50)	361.97 (52.50)	352.18 (51.08)	330.40 (47.92)
	Range of primary + secondary stress intensity	2827 (410)	2806 (407)	2758 (400)	2696 (391)	2648 (384)	2579 (374)	2530 (367)
	Pure primary shear stress	217.18 (31.50)	206.36 (29.93)	175.95 (25.52)	160.72 (23.31)	152.03 (22.05)	149.89 (21.74)	149.89 (21.74)
Bolts	Material	ASTM A325						
	Average tension stress	579.16 (84.00)	573.37 (83.16)	567.58 (82.32)	561.78 (81.48)	550.20 (79.80)	526.48 (76.36)	501.11 (72.68)
	Average shear stress	347.50 (50.40)	344.05 (49.90)	340.53 (49.39)	337.08 (48.89)	330.12 (47.88)	315.92 (45.82)	300.68 (43.61)

ASTM = American Society for Testing and Materials.

SS = stainless steel.

7.2.3 Chemical and Galvanic Reactions

Stainless steel, unlike lower alloy or carbon steel, does not rust under ordinary atmospheric exposure, making it less prone to contamination. Stainless steel is an iron-based alloy that contains at least 10.5-percent chromium. In the presence of oxygen, an invisible, protective layer of chromium oxide forms on its surface, which protects the stainless steel and keeps it from oxidizing in the presence of moisture.

7.2.4 Size of Package and Cavity

The RGC has inside dimensions of approximately 89.9 cm (35.4 in.) wide by 170.2 cm (67.0 in.) long by 83.8 cm (33.0 in.) high and an approximate volume of 1.28 m³ (45.3 ft³). The SWDB exterior measures approximately 152.4 cm (60 in.) wide by 254 cm (100 in.) long by 129.5 cm (51.0 in.) high. The SWDB Impact Limiter exterior measures approximately 261.4 cm (102.9 in.) wide by 361.7 cm (142.4 in.) long by 229.1 cm (90.5 in.) high.

7.2.5 Weights and Center of Gravity

The SWP has a maximum payload capacity of 3860 kg (8500 lb) for the nondispersible payload and 3300 kg (7300 lb) for the dispersible payload. The maximum total weight of the contents of the SWDB is 4500 kg (10,000 lb). The maximum gross weight of the loaded SWDB assembly shall be limited to 37.2 metric tons (41 tons). The maximum loaded handling weight of the SWDB shall be limited to 27.2 metric tons (30 tons) due to the lifting bail analysis.

7.2.6 Tamper-Indicating Feature

A tamper-indicating seal will be installed to the Steel Waste Package to provide visual evidence of tampering. Inadvertent opening is prevented because of the size and number of fasteners on the impact limiter and inner containers and the series of deliberate procedures that would be needed to open the package.

7.2.7 Positive Closure

Positive closure of the 22-Ton SWDB is provided by the lids of the 22-Ton SWDB and SWDB Overpack. The 22-Ton SWDB lid weighs 4379 kg (9655 lb) and is bolted to the 22-Ton SWDB with 16 bolts. The SWDB Overpack lid weighs 1100 kg (2500 lb) and is bolted to the 22-Ton SWDB Overpack with 32, ASTM A325 ¾-10UNC-2A bolts. Also, the SWDB Overpack is encased in the SWDB Impact Limiter that is clamped together by locking device. This device requires specialized tooling to unlock. Consequently, inadvertent opening of the package is not possible without specialized lifting and handling equipment.

7.2.8 Lifting and Tiedown Devices

The package lifting device performance is evaluated against the requirements of DOE-RL-92-36, *Hanford Site Hoisting and Rigging Manual* (RL 1999). For tiedown attachments that are attached to the package, performance shall be evaluated for either (1) failure of the attachment before damage of the package or (2) attachment meets the 10, 2, and 5 requirements of 10 CFR 71.45(b).

Lifting and tiedown attachments shall meet the technical requirements of 10 CFR 71, 10 CFR 71.45, and RL (1999). The lifting attachments for the packaging shall be capable of lifting three times the total suspended weight without generating a combined stress or maximum tensile stress at any point in the load path in excess of the corresponding minimum yield strength of their materials of construction. The lifting attachments shall be compatible with the cranes at the 324 Facility and portable cranes, as applicable, to permit field lifting of the package.

A second set of lifting devices is provided on the SWDB for loading and unloading of a fully loaded SWDB from and into the SWDB Overpack. These lifting devices consist of two, large, heavy-construction lifting lugs fixed to each short side of the SWDB. These lifting devices are designed to the requirements of RL (1999), which requires a 3-to-1 design factor based on yield. Also, the lifting devices are designed with spring assists for remotely raising and are collapsible such that the SWDB fits within the confines of the SWDB Overpack. The ability of the lifting devices to meet the RL (1999) requirements is demonstrated in Part B, Section 7.9.1, of this SARP.

7.2.9 Brittle Fracture

The package low-temperature performance is evaluated at an ambient temperature of -33 °C (-27 °F). Because the containment boundary is constructed of austenitic stainless steel, brittle fracture of the containment boundary is not a concern. Austenitic stainless steel is not susceptible to low-temperature brittle fracture and maintains its ductility at extremely low temperatures. 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* (NRC 1991), the U.S. Nuclear Regulatory Commission recognizes this behavior and does not require low temperature ductility test of stainless steel for packages. The containment boundary includes the SWDB Overpack double g-ring seal. These Viton O-rings have a manufacturer's working temperature range of -40 °C (-40 °F) (Parker Seal 1992). Consequently, under NCT and accident condition loading, the containment boundary will maintain containment.

However, because the SWDB shielding is constructed from carbon steel manufactured to the requirements of ASTM A-36, brittle fracture is a concern in maintenance of shielding. Consequently, based on BPVC Section VIII, Division 1 (ASME 1995e), guidelines for low-temperature service, handling and transport of the SWDB package is not permitted for a metal temperature of less than 0 °C (32 °F).

7.3 STRUCTURAL MODEL

7.3.1 ABAQUS/Explicit Model Description

ABAQUS is a general-purpose finite element analysis program that can be used for elastic, plastic, and other inelastic analysis of structural members. This finite element analysis program can be used in one-, two-, and three-dimensional analysis. The ABAQUS provides two subsets of analysis scheme; one is ABAQUS/Standard, and the other is ABAQUS/Explicit (HKS 1998). The latter is used in this impact evaluation.

The ABAQUS' explicit dynamic finite element program, ABAQUS/Explicit (HKS 1998), is fully vectorized for use on high-end computers. In order to achieve a faster computation, some limitations of element intrinsic characteristics have been incorporated, such as a smaller subset of elements (compared to ABAQUS/Standard) to be allowed in ABAQUS/Explicit, the use of diagonal mass matrix.

The most important feature of ABAQUS/Explicit (HKS 1998) is its ability in solving contact problems as in the simulation of transport packaging impact problems. Other important features of ABAQUS are as follows.

7.3.1.1 Restart Option. The ability to restart at a desired analytical step is essential for this type of analysis due to the limitation of computer resources available for this type of analysis. The restart also offers the user some control of the convergence of analytical results. In addition, the plastic deformation of steel and the intrinsic crushing and cracking characteristics of brittle materials that can generate unrealistic results requires the restart to allow the user to go back to an acceptable step and start over again from there. This restart option allows the user to restart from a step/iteration from a saved restart file. The restart file can also be used to generate additional results that are not contained in the results file or time history file.

7.3.1.2 Post Processing. The dynamic nonlinear analysis generates a large amount of data that can be controlled by the user in the input file and saved in the results file. The ABAQUS post-processor can be used to retrieve the needed data in the form of tables or plots. The ABAQUS plot processor is very versatile in processing the needed data into historical plots, contour plots, deformation plots, etc. Only the data saved in the results file and history file can be processed; however, additional data can also be generated from the restart file.

7.3.1.3 Convergence Control. The selection of step size in the nonlinear analysis for convergence control of the solution to ensure reasonable results is handled internally in ABAQUS/Explicit (HKS 1998). The program selects the initial time step size based on the element characteristic length and then increases or decreases the time step size automatically as the analysis proceeds. In an ABAQUS time history analysis, a large number of steps, increments, and iterations are used to ensure proper convergence of solution.

7.3.2 Material Models

7.3.2.1 22-Ton SWDB and Overpack Assembly. The redesigned SWDB Overpack assembly is the containment boundary of the 22-Ton SWDB transport package. The SWDB Overpack is constructed of Type 304 stainless steel. It consists of a box of 1.3 cm (0.5-in.) thickness, a forged flange, lid and lid bolts, and O-ring seals. On the outer surfaces, there are a number of stiffeners lined vertically to protect the containment boundary from being intruded by the SWDB Impact Limiter guides during the impact.

7.3.2.1.1 Overpack Assembly. The steel used in the SWDB Overpack and lid is Type 304 stainless steel. The stiffeners are constructed with Type 304 or 304L stainless steel. The material properties of Type 304 stainless steel are obtained from ASME (1995a, 1995b). The mechanical properties are as follows:

- $E = 28.0 \times 10^6$ psi at 70 °F
- Poisson's ratio = 0.3
- Density = 500 lbm/ft³.

The ASME (1995b) Class 2 and 3 component material minimum ultimate strength (S_u), minimum yield strength (S_y), and allowable stress (S_m) from ASME (1995c), Tables U, Y-1, and 2A, are shown in Table B7-7.

Table B7-7. SA-276, Type 304 Stainless Steel.

Temperature (°F)	S_u (ksi)	S_y (ksi)	S_m (ksi)
Up to 100	70.0	30.0	20.0
200	66.2	25.0	20.0
300	61.5	22.5	20.0
400	60.0	20.7	18.7
500	59.3	19.4	17.5

Source: ASME, 1995b, *Boiler and Pressure Vessel Code*, Section II, "Material," "Part D—Properties," Tables U, Y-1, and 2A, American Society of Mechanical Engineers, New York, New York.

The plasticity of Type 304 stainless steel is shown in Table B7-8.

Table B7-8. Plasticity of Type 304 Stainless Steel.

Stress (psi)	Plastic strain	Stress (psi)	Plastic strain
21870	0.0	45930	0.070
25000	0.002	47800	0.080
27510	0.004	49570	0.090
28830	0.006	51210	0.100
29810	0.008	54980	0.125
30590	0.010	58300	0.150
33840	0.020	60970	0.175
36680	0.030	63320	0.200
39310	0.040	66600	0.250
41640	0.050	69000	0.300
43870	0.060	70000	0.350

7.3.2.1.2 SWDB Overpack Lid Bolts. The SWDB Overpack lid bolts are constructed of A-325 high-strength fastener steel. It has a modulus of elasticity of 29×10^6 psi, a Poisson's ratio of 0.3, a yield strength of 634,318 kPa (92,000 psi), an ultimate tensile strength of 827,371 kPa (120,000 psi), and minimum elongation of 14 percent. The density of the steel is 0.29 lbm/in³. The stress-strain relationship is modeled as a bilinear curve with up to the maximum 14-percent plastic strain.

7.3.2.1.3 SWDB. The SWDB is modeled by the ABAQUS rigid element. No material properties are necessary.

7.3.2.2 SWDB Impact Limiter. The SWDB Impact Limiter is comprised of two, almost identical, upper and bottom sections. The upper section has additional lifting devices attached. The SWDB Impact Limiter is constructed of 0.64 cm (0.25-in.) steel liners with wood inserts. The outer frame of the SWDB Impact Limiter is constructed of rectangular steel tubes, to which the lifting and closing attachments are welded (drawing H-3-56433, *22 Ton Steel Waste Dspl Box Impact Limiter Assy/Det* [PNL 1988b]). The impact analysis model of the SWDB Impact Limiter is based on drawings H-3-56430, *22 Ton Steel Waste Disposal Box Arrangement* (PNL 1988a); H-3-56431, *22 Ton Steel Waste Disposal Box Details* (PNL 1996c); and H-3-56433 (PNL 1988b). The lifting devices are neglected in the impact analysis model.

7.3.2.2.1 ASTM A-36 Carbon Steel. The SWDB Impact Limiter liners and the bottom SWDB Impact Limiter guides are constructed of A-36 carbon steel. The outer tube frame is made of A-500 Grade B carbon steel, but in this analysis it is modeled as A-36 carbon steel. This is also true for the assumption of connections, such as the hooks between the upper and bottom SWDB Impact Limiters.

The A-36 carbon steel has a modulus of elasticity of 29×10^6 psi, a Poisson's ratio of 0.3, a yield strength of 248,211 kPa (36,000 psi), an ultimate tensile strength of 413,685 kPa

(60,000 psi), and a minimum elongation of 20 percent. The density of the steel is 0.29 lbm/in³. The stress-strain relationship is modeled by a bilinear curve up to the maximum strain of 20 percent.

7.3.2.2.2 SWDB Impact Limiter Wood Fillers. Wood fillers used in the SWDB Impact Limiter are 4 x 8 Douglas fir. The wood material properties are obtained from ANON (1980). The wood is a highly orthotropic material depending on the orientation of wood fiber. In this analysis, the wood is assumed as an isotropic elastic material with a modulus of elasticity of 1.95×10^6 psi and a Poisson's ratio of 0.30. The Douglas fir has a density of approximately 30 lbm/ft³.

7.3.2.3 SWDB Shield Plate. A 7.62 cm (3-in.-) thick shield plate is placed at the bottom of the lower SWDB Impact Limiter. It is made of A-36 carbon steel. The material properties are the same as described previously.

7.3.3 Finite Element Models and Analysis Results

Taking advantage of the model symmetry, a half-section model was developed in this analysis. The entire package remains in its upright position, but the unyielding target is rotated and positioned for different impact orientations.

The overpack containment boundary model uses ABAQUS three-dimensional shell element S4R for the bottom and wall sections and ABAQUS three-dimensional solid element C3D8R for the flange and lid. ABAQUS B31 beam element is used for the lid bolts. The stiffeners on the outside of the overpack use ABAQUS three-dimensional shell element S4R to model. For the SWDB Impact Limiter, the model uses ABAQUS three-dimensional solid element C3D8 for wood inserts and three-dimensional shell element S4R for steel liners and outer steel tube frames. The hooks between the upper and bottom SWDB Impact Limiters are simplified as connecting bolts using ABAQUS B31 beam element. The 8 cm (3-in.) SWDB Shield Plate is modeled using ABAQUS three-dimensional solid element C3D8R. As noted before, the SWDB is modeled by ABAQUS rigid element.

The entire model is shown in Figure B7-1 for the general view of the side impact orientation. Relevant element information is shown in Figures B7-2 through B7-12.

Figure B7-1. General View of Side Impact.

Figure B7-2. 22-Ton Steel Waste Disposal Box Containment.

Figure B7-3. 22-Ton Steel Waste Disposal Box Containment Bottom.

Figure B7-4. 22-Ton Steel Waste Disposal Box Containment Wall.

Figure B7-5. 22-Ton Steel Waste Disposal Box Containment Flange.

Figure B7-6. 22-Ton Steel Waste Disposal Box Containment Lid.

Figure B7-7. 22-Ton Steel Waste Disposal Box Containment Lid Bolts.

Figure B7-8. 22-Ton Steel Waste Disposal Box Containment Stiffeners.

Figure B7-9. 22-Ton Steel Waste Disposal Box Shield Plate.

Figure B7-10. 22-Ton Steel Waste Disposal Box Simulation

Figure B7-11. Steel Waste Disposal Box Impact Limiter Bottom Section.

Figure B7-12. Steel Waste Disposal Box Impact Limiter Top Section.

7.4 NCT

The performances of the 22-Ton SWDB transport package under the NCT free drop are evaluated in this analysis. The cases analyzed herein include the 0.3 m (1-ft) drops of the transport package impacting on its short side and top edge onto an unyielding surface. The models developed in the accident conditions are used for this analysis with only a change of the initial impact velocity in each case.

7.4.1 Conditions To Be Evaluated

The following conditions or events are intended to bound the effects of Hanford Site NCT. The structural verification of the 22-Ton SWDB design is contained in this section.

- Reduced external pressure of 24.1 kPa, absolute (3.5 psia)
- Increased external pressure of 137.9 kPa, absolute (20 psia)
- Free drop from a height of 0.3 m (1 ft) onto a hard, unyielding surface
- Package impact in the orientation expected to cause maximum damage.

The most critical conditions and events are evaluated for the worst-case environmental conditions. Load combinations for these evaluations are taken from Regulatory Guide 7.8, *Load Combinations for the Structural Analysis of Shipping Casks* (NRC 1977).

7.4.2 Acceptance Criteria

NCT performance acceptance criteria shall be based on BPVC, Section III, Subsection NE (ASME 1995d). Maintenance of containment is defined as the ability of the packaging systems to sustain the applied loading without exceeding ASME allowables.

7.4.3 Initial Conditions

7.4.3.1 Environmental Heat Loading. In Part B, Section 8.0, the temperatures of the 22-Ton SWDB containment boundary under maximum ambient conditions is determined. The worst-case temperature of 152 °C (306 °F) under NCT with solar insolation is at the outside of the overpack boundary. Without solar insolation the maximum temperature at the SWDB Overpack boundary is 129 °C (264 °F).

7.4.3.2 Maximum Thermal and Pressure Stresses. The thermal stresses from component temperature differences under NCT are negligible as demonstrated by the following evaluation. Considering thermal stresses as secondary (as defined in ASME [1995d]), the lowest primary plus secondary stress intensity allowable of $3s_m$ is 291.23 MPa (42.24 ksi). Based on this value, the calculated value for an allowable temperature differential from the equation $3s_m/E\alpha$ is 48 °C (119 °F). This is substantially higher than the temperature differences between adjacent points on the SWDB Overpack. Consequently, the effects of thermal stressing of internal components

are negligible. Due to the uncertainty of the contents, the maximum differential normal operating pressure at temperature has been evaluated in Part B, Section 7.9.2. The maximum differential normal operating pressure that can be retained by the SWDB without reducing the performance of the package from other loads is 110 kPa (16 psi) under normal operating temperatures (as shown in Part B, Section 7.9.2).

7.4.4 0.3 m (1-ft) Side Impact

The performance of the 22-Ton SWP under NCT of 0.3 m (1-ft) side impact onto an unyielding target is evaluated in this subsection. The total analysis time in this case is 20 ms, and the rebound occurs at 12.8 ms into the event. Using impact force data of the entire package, the peak acceleration is determined to be 82.6g. This impact force data is processed through a double filtration process to remove high-frequency impact noise as recommended by Safety Series No. 37, *Advisory Material for the IAEA Regulations for the Safe Transport of Radioactive Material* (IAEA 1990). The first filtration has a 10,000 Hz filter, and the second filtration has a 1,000 Hz filter. A sine-Butterworth filter with six poles in ABAQUS postprocessing is used. The average acceleration of the package is determined when the entire package reaches zero velocity. The average acceleration is found to be 19.5g; i.e., $[(96.3 \text{ in}^2/\text{sec})/(0.0128 \text{ sec} \cdot \text{g})]$. The acceleration, velocity, and displacement histories of the entire package are shown in Figures B7-13, B7-14, and B7-15, respectively. Pertinent acceleration data are shown in Table B7-9.

Figure B7-13. Acceleration History of the 22-Ton Steel Waste Package in the 0.3 m (1-ft) Side Impact.

Figure B7-14. Velocity History of the 22-Ton Steel Waste Package in the 0.3 m (1-ft) Side Impact.

Figure B7-15. Displacement History of the 22-Ton Steel Waste Package in the 0.3 m (1-ft) Side Impact.

Table B7-9. 22-Ton Steel Waste Package Acceleration Data, 0.3 m (1-ft) Side Impact.

Case		Peak acceleration (g)	Average acceleration (g)
Side impact	0.3 m (1-ft) drop	82.6	19.5

At the containment pressure boundary of the SWDB Overpack, the stresses and strains are evaluated. Impact-induced high frequencies of stress components are filtered out by using a double filtration process that is the same as that used for determination of the acceleration. The first filtration frequency is 10,000 Hz, and the second filtration frequency is 1000 Hz. A sine-Butterworth filter with six poles in ABAQUS postprocessing is used for stress filtration; however, the strains are not filtered. Also, readings of the stress and strain are taken at the end of the run when the responses are stabilized. The stresses and strains in this case are relatively low except at highly localized areas at the junction of the wall and the bottom plate on the opposite side of the impact end. The center portion of this junction has been pushed inward slightly by the reaction of the inner liner of the SWDB Impact Limiter and the Shield Plate. The highest end-of-impact stress intensities are approximately 344.74-372.32 MPa (50-54 ksi) at the junctions of the bottom plate to side wall. Superimposing the stress from pressure at these locations, the combined loading maximum stress intensity becomes approximately 393 MPa (57 ksi). At this location the maximum stress intensities are considered a combination of primary membrane, bending, and secondary (due to the discontinuity). Consequently, the ASME (1995d) Service Level C limit in these localized areas is 409.55 MPa (59.4 ksi). The highest strain is approximately 10 percent in the localized deformed regions, indicating regions of highly localized deformation, but there is no breach of the boundary resulting in loss of containment. The worst-case combined stress on the closure bolts is 448.85 MPa (65.1 ksi). This less than the ASME (1995d) maximum stress allowable of 523.31 MPa (75.9 ksi). Subsequently, the bolts will maintain the lid seal and thereby maintain containment. The critical elements in the containment parts are shown in Table B7-10.

Figure B7-16 shows the history of stress intensity at element (EL) 2, an element in the containment bottom. Figure B7-17 shows the history of logarithmic strain at EL 1001, an element of the containment wall. Figure B7-18 shows the history of stress intensity at the lid bolt EL 8910. Figure B7-19 shows the history of logarithmic strain at the lid bolt EL 8913. All stresses and strains except those in highly loaded areas are minor. The deformation of the SWDB containment boundary of the transport package is also negligible as shown in Figure B7-20.

Table B7-10. Results of American Society of Mechanical
Engineers Stress Intensity and Logarithmic Strain
0.3 m (1-ft) Side Impact.

Element	S.I. (ksi)	Log. strain	Element	S.I. (ksi)	Log. strain
Containment bottom			Containment wall		
1	43.81	0.03669	1001	45.20	0.10330
2	53.77	0.03345	2970	24.36	0.01570
38	16.35	0.00250	2971	28.13	0.00745
188	13.75	0.00278	2972	31.70	0.00663
238	9.36	0.00276	2979	40.58	0.00974
588	8.40	0.00234	2980	38.99	0.01116
638	5.13	0.00244	2983	40.26	0.01168
688	19.29	0.00190	----	----	----
Containment flange			Containment lid		
4369	7.49	0.00257	7493	3.73	0.00119
4370	13.46	0.00806	7543	5.49	0.00110
4869	7.24	0.00315	----	----	----
4870	18.76	0.00841	----	----	----
Containment lid bolt					
8910	39.31	0.00185	8914	19.92	0.00249
8911	32.95	0.00203	8915	19.18	0.00247
8912	26.05	0.00228	8916	18.99	0.00248
8913	9.20	0.00279	----	----	----

Figure B7-16. Stress Intensity of Element 2 in 0.3 m (1-ft) Side Impact.

Figure B7-17. Logarithmic Strain of Element 1001 in 0.3 m (1-ft) Side Impact.

Figure B7-18. Stress Intensity of Lid Bolt Element 8910
in 0.3 m (1-ft) Side Impact.

Figure B7-19. Logarithmic Strain of Lid Bolt Element 8913
in 0.3 m (1-ft) Side Impact.

Figure B7-20. Deformation Plot of 0.3 m (1-ft) Side Impact.

7.4.5 0.3 m (1-ft) Top-Edge Impact

The performance of the 22-Ton SWP under NCT of 0.3 m (1-ft) top-edge impact onto an unyielding target is evaluated. The total analysis time in this case is 50 ms. The rebound time is estimated to be 18.15 ms in this 0.3 m (1-ft) top-edge impact. Using the impact force data of the entire package, the peak acceleration is determined to be 30.4g by a double filtration process to remove high-frequency impact noise as recommended by IAEA (1990). The first filtration has a 10,000 Hz filter, and the second filtration has a 1,000 Hz filter. A sine-Butterworth filter with six poles in ABAQUS postprocessing is used. The average acceleration of the package is calculated when the entire package reaches zero velocity. The average acceleration is found to be 13.7g; i.e., (96.3/0.01815)g. The acceleration, velocity, and displacement histories of the entire package are shown in Figures B7-21, B7-22, and B7-23, respectively. Pertinent acceleration data are shown in Table B7-11.

At the containment pressure boundary of the SWDB Overpack, stresses and strains are evaluated. Impact-induced high frequencies of stress components are filtered out by using a double filtration process that is the same as that used in determination of the acceleration. The first filtration frequency is 10,000 Hz, and the second filtration frequency is 1,000 Hz. A sine-Butterworth filter with six poles in ABAQUS postprocessing is used here. As with the previous impact evaluation, the strains are not filtered. Also, the readings of the stress and strain are taken at the end of the run when the responses are stabilized. In this 0.3 m (1-ft) top-edge impact, the critical locations exist at the junction at the corner of the containment wall and at the containment lid on the impact side. The maximum stress intensities at the junction corner of containment wall and at the middle of the lid at the impact side are approximately 193.05 MPa (28 ksi). Superimposing the stress from pressure at these locations, the combined loading maximum stress intensity is approximately 216.5 MPa (31.4 ksi). At these locations the maximum stress intensity

is considered a combination of primary membrane, primary bending, and secondary. Consequently, the ASME (1995d) Service Level C limit in these areas is 409.5 MPa (59.4 ksi). The highest strains are approximately 4.2 percent at the containment lid and 3.0 percent on the wall and are highly localized. This indicates regions of highly localized deformations, but there is no breach of the containment boundary, which would result in a loss of containment. The worst-case combined stress on the closure bolts is 442.0 MPa (64.1 ksi). This is less than the ASME (1995d) maximum stress allowable of 523.3 MPa (75.9 ksi). Subsequently, the bolts will maintain the lid seal and thereby maintain containment. The critical elements in the containment parts are shown in Table B7-12.

Figure B7-21. Acceleration History of the 22-Ton Steel Waste Package in the 0.3 m (1-ft) Top-Edge Impact.

Figure B7-22. Velocity History of the 22-Ton Steel Waste Package in the 0.3 m (1-ft) Top-Edge Impact.

Figure B7-23. Displacement History of the 22-Ton Steel Waste Package in the 0.3 m (1-ft) Top-Edge Impact.

Table B7-11. 22-Ton Steel Waste Package Acceleration
Data, 0.3 m (1-ft) Top-Edge Impact.

Case		Peak acceleration (g)	Average acceleration (g)
Top-edge impact	0.3 m (1-ft) drop	30.4	13.7

Table B7-12. Results of American Society of Mechanical
Engineers Stress Intensity and Logarithmic Strain
0.3 m (1-ft) Top-Edge Impact.

Element	S.I. (ksi)	Log. strain	Element	S.I. (ksi)	Log. strain
Containment bottom			Containment wall		
586	13.51	0.00045	2870	23.15	0.02151
587	3.64	0.00143	2871	20.07	0.00872
588	19.21	0.00049	2872	10.31	0.01147
636	14.42	0.00084	2960	7.95	0.02240
637	5.06	0.00178	2970	28.37	0.03006
686	14.47	0.00104	2971	24.28	0.01488
687	9.26	0.00168	2972	18.38	0.00542
688	11.13	0.00321	----	----	----
Containment flange			Containment lid		
4170	14.74	0.00349	7044	27.81	0.04204
4270	16.47	0.00388	----	----	----
4370	12.46	0.01445	----	----	----
4769	15.32	0.01240	----	----	----
4770	12.95	0.01470	----	----	----
4869	10.88	0.01051	----	----	----
4870	12.47	0.01913	----	----	----
Containment lid bolt					
8907	38.31	0.00184	8914	8.20	0.00054
8911	32.59	0.00207	8915	14.40	0.00001
8912	5.33	0.00045	8916	16.54	0.00182
8913	10.43	0.00280	----	----	----

Figures B7-24 and B7-25 show the history of stress intensity and logarithmic strain at EL 2970, an element on the containment wall. Figures B7-26 and B7-27 show the history of stress intensity and logarithmic strain at EL 7044, which is on the containment lid. Figure B7-28 shows the history of stress intensity at the lid bolt EL 8907. Figure B7-29 shows the history of logarithmic strain at the lid bolt EL 8913. The deformation of the containment boundary in this case is negligible as shown in Figure B7-30.

Figure B7-24. Stress Intensity of Element 2970
in 0.3 m (1-ft) Top-Edge Impact.

Figure B7-25. Logarithmic Strain of Element 2970
in 0.3 m (1-ft) Top-Edge Impact.

Figure B7-26. Stress Intensity of Element 7044
in 0.3 m (1-ft) Top-Edge Impact.

Figure B7-27. Logarithmic Strain of Element 7044
in 0.3 m (1-ft) Top-Edge Impact.

Figure B7-28. Stress Intensity of Lid Bolt Element 8907
in 0.3 m (1-ft) Top-Edge Impact.

Figure B7-29. Logarithmic Strain of Lid Bolt Element 8913
in 0.3 m (1-ft) Top-Edge Impact.

Figure B7-30. Deformation Plot of 0.3 m (1-ft) Top-Edge Impact.

7.5 ACCIDENT CONDITIONS

The performance of the 22-Ton SWP in the accident drop conditions is evaluated in this section. The worst loading cases evaluated are presented herein and are the 9 m (30-ft) drops of the transport package impacting on its short side and on its top edge onto an unyielding surface.

7.5.1 Conditions To Be Evaluated

The following conditions or events are intended to envelope the effects of Hanford Site accident transport conditions. The following scenarios related to accident conditions are addressed:

- Free drop from a height of 9 m (30 ft) onto a hard, unyielding surface.
- Package impact orientation expected to cause maximum damage.

The most critical conditions and events are evaluated for the worst-case environmental conditions and external pressures.

7.5.2 Acceptance Criteria

Accident condition performance acceptance criteria are based on ASME (1995d). Maintenance of containment is defined as the packaging system's ability to sustain the loadings without exceeding the ASME (1995d) Service Level D limits.

7.5.3 Initial Conditions

7.5.3.1 Environmental Heat Loading. In Part B, Section 8.0, the temperatures of the 22-Ton SWDB containment boundary under maximum ambient conditions are determined. The worst-case temperature is 152 °C (306 °F) under accident conditions at the SWDB Overpack boundary.

7.5.3.2 Maximum Thermal and Pressure Stresses. The thermal stresses from component temperature difference under NCT are negligible as demonstrated by the following evaluation. Considering thermal stresses as secondary (as defined in ASME [1995d]), the lowest primary plus secondary stress intensity allowable of $3s_m$ is 291.23 MPa (42.24 ksi). Based on this value, the calculated value for an allowable temperature differential from the equation $3s_m/E\alpha$ is 48 °C (119 °F). This is substantially higher than the temperature differences between adjacent points on the SWDB Overpack. Consequently, the effects of thermal stressing of internal components are negligible. Due to the uncertainty of the contents, the maximum differential normal operating pressure at temperature has been evaluated in Part B, Section 7.9.2. The maximum differential normal operating pressure that can be retained by the SWDB without reducing the performance of the package from other loads is 110 kPa (16 psi) under normal operating temperatures as shown in Part B, Section 7.9.2.

7.5.4 9 m (30-ft) Side Impact

The performance of the 22-Ton SWP in the accident condition of 9 m (30-ft) side impact onto an unyielding target is evaluated. The total analysis time in this case is 20 ms, and the rebound occurs at 7.35 ms. Using the impact force data of the entire package, the peak acceleration is determined found to be 280.3g. This impact force data is processed through a double filtration process to remove high-frequency impact noises as recommended by IAEA (1990). The first filtration has a 10,000 Hz filter, and the second filtration has a 1,000 Hz filter. A sine-Butterworth filter with six poles in ABAQUS postprocessing is used. The average acceleration of the package is calculated when the entire package reaches zero velocity. The average acceleration is determined to be 185.9g; i.e., $[(528 \text{ in}^2/\text{sec})/(0.00735 \text{ sec} \cdot g)]$. The acceleration, velocity, and displacement histories of the entire package are shown in Figures B7-31, B7-32, and B7-33, respectively. Pertinent acceleration data are shown in Table B7-13.

Figure B7-31. Acceleration History of the 22-Ton Steel Waste Package in the 9 m (30-ft) Side Impact.

Figure B7-32. Velocity History of the 22-Ton Steel Waste Package in the 9 m (30-ft) Side Impact.

Figure B7-33. Displacement History of the 22-Ton Steel Waste Package in the 9 m (30-ft) Side Impact.

Table B7-13. 22-Ton Steel Waste Package Acceleration
Data, 9 m (30-ft) Side Impact.

Case		Peak acceleration (g)	Average acceleration (g)
Side impact	9 m (30-ft) drop	280.3	185.9

At the containment pressure boundary of the SWDB Overpack, stresses and strains are evaluated in this subsection. Impact-induced high frequencies of stress components are filtered out by using a double filtration process that is the same as that used for determination of the acceleration. The first filtration frequency is 10,000 Hz, and the second filtration frequency is 1,000 Hz. A sine-Butterworth filter with six poles in ABAQUS postprocessing is for stress filtration; however, the strains are not filtered. Also, readings of the stress and strain are taken at the end of the run when the responses are stabilized. In this 9 m (30-ft) side impact, the critical locations are at the junction of the corner with the containment wall and at the flange on the impact side. Due to the highly localized discontinuity at the junction corner, stress concentration effect causes very high local stress in this area. The highest impact stress intensities are approximately 468.84-482.63 MPa (68-70 ksi). When combined with internal pressure, the total stress intensities are within the Service Level D limits for peak stress range. The highest strains are approximately 33.5 percent at the containment boundary of the flange and 29.5 percent on the wall and below the ultimate rupture strain of stainless steel, which is 35 percent. The strains induced by pressure are in the elastic range, are very low in comparison to the above strains, and are considered as negligible. The critical elements in the containment parts are shown in Table B7-14.

Figures B7-34 and B7-35 show the history of stress intensity and logarithmic strain at EL 2970, an element in the containment wall. Figure B7-36 shows the history of stress intensity at EL 4373, and Figure B7-37 shows the history of logarithmic strain at EL 4371. Both elements are on the containment flange inside surface. Figure B7-38 shows the history of stress intensity at the lid bolt EL 8913. Figure B7-39 shows the history of logarithmic strain at the lid bolt EL 8914. The stresses in this 9 m (30-ft) side drop are high, near the material rupture strength. The high stress and strain, however, only exist at local regions where the containment boundary faces are joined together. Due to the local stress/strain concentration effect, these stresses and strains are classified as peak stress in ASME stress classification, which also includes secondary stresses. Consequently, these peak stresses are below the Service Level D allowable range. In general, as a result of this effect, the local loading existing after the impact will not cause the breach of the containment boundary because the strains are below the ultimate rupture strain. However, large deformations of the SWDB Overpack occur in these highly strained localized regions. The deformation of the SWDB containment boundary of the transport package is shown in Figure B7-40. The deformation of containment boundary stiffeners on the impact side is shown in Figure B7-41 in which crushes of two outside stiffeners can be readily seen in this symmetrical half-section plot.

Table B7-14. Results of American Society of Mechanical Engineers Stress Intensity and Logarithmic Strain
9 m (30-ft) Side Impact.

Element	S.I. (ksi)	Log. strain	Element	S.I. (ksi)	Log. strain
Containment bottom			Containment wall		
37	34.69	0.01362	2870	60.69	0.14490
38	33.63	0.05129	2871	52.14	0.02329
188	25.99	0.04512	2872	46.58	0.00909
388	13.44	0.04590	2960	49.63	0.22090
588	11.96	0.07265	2970	70.21	0.29540
637	12.38	0.05220	2971	59.33	0.05742
638	25.56	0.05587	2972	45.70	0.05442
Containment flange			Containment lid		
4370	62.55	0.31580	7792	34.30	0.01951
4371	48.00	0.33520	7793	33.58	0.08036
4372	61.29	0.30740	7794	28.92	0.08889
4373	68.91	0.28540	7842	29.85	0.04941
4870	52.32	0.18550	7843	30.07	0.04443
4871	49.66	0.16650	7844	16.81	0.01097
4872	51.38	0.10720	----	----	----
Below not containment boundary			----	----	----
4270	66.17	0.30620	----	----	----
4271	35.44	0.36720	----	----	----
4272	31.56	0.34870	----	----	----
Containment lid bolt					
8911	23.44	0.00232	8914	9.60	0.04919
8912	25.24	0.03641	8915	22.02	0.04095
8913	38.72	0.00208	8916	13.27	0.02731

Figure B7-34. Stress Intensity of Element 2970 in 9 m (30-ft) Side Impact.

Figure B7-35. Logarithmic Strain of Element 2970 in 9 m (30-ft) Side Impact.

Figure B7-36. Stress Intensity of Element 4373 in 9 m (30-ft) Side Impact.

Figure B7-37. Logarithmic Strain of Element 4371 in 9 m (30-ft) Side Impact.

Figure B7-38. Stress Intensity of Lid Bolt Element 8913
in 9 m (30-ft) Side Impact.

Figure B7-39. Logarithmic Strain of Lid Bolt Element 8914
in 9 m (30-ft) Side Impact.

Figure B7-40. Deformation Plot of 22-Ton Steel Waste Package in 9 m (30-ft) Side Impact.

Figure B7-41. Deformation Plot of 22-Ton Steel Waste Disposal Box Overpack Containment Boundary Stiffener in 9 m (30-ft) Side Impact.

7.5.5 9 m (30-ft) Top-Edge Impact

The performance of the 22-Ton SWP in the accident condition of 9 m (30-ft) top-edge impact onto an unyielding target is evaluated herein. The total analysis time in this case is 50 ms. The rebound time is estimated to be 17.0 ms in this 9 m (30-ft) top-edge impact. Using the impact force data of the entire package, the peak acceleration is determined to be 111.4g after undergoing a double filtration process to remove high-frequency impact noises as recommended by IAEA (1990). The first filtration has a 10,000 Hz filter, and the second filtration has a 1000 Hz filter. A sine-Butterworth filter with six poles in ABAQUS postprocessing is used. The average acceleration of the package is calculated when the entire package reaches zero velocity. The average acceleration is determined to be 80.4g, i.e., (528/0.0170)g. The acceleration, velocity, and displacement histories of the entire package are shown in Figures B7-42, B7-43, and B7-44, respectively. Pertinent acceleration data are shown in Table B7-15.

Figure B7-42. Acceleration History of the 22-Ton Steel Waste Package in the 9 m (30-ft) Top-Edge Impact.

Figure B7-43. Velocity History of the 22-Ton Steel Waste Package in the 9 m (30-ft) Top-Edge Impact.

Figure B7-44. Displacement History of the 22-Ton Steel Waste Package in the 9 m (30-ft) Top-Edge Impact.

Table B7-15. 22-Ton Steel Waste Package Acceleration
Data, 9 m (30-ft) Top-Edge Impact.

Case		Peak acceleration (g)	Average acceleration (g)
Top-edge impact	9 m (30-ft) drop	111.4	80.4

At the containment pressure boundary of the SWDB Overpack, stresses and strains are evaluated. Impact-induced high frequencies of stress components are filtered out by using a double filtration process that is the same as that used in determination of the acceleration. The first filtration frequency is 10,000 Hz, and the second filtration frequency is 1,000 Hz. A sine-Butterworth filter with six poles in ABAQUS postprocessing is used to filter the stress component; however, the strains are not filtered. Also, the readings of the stress and strain are taken at the end of the run when the responses are stabilized. In this 9 m (30-ft) top-edge impact, the critical locations exist at the junction corner of the containment wall and the flange on the impact side. Due to the highly localized discontinuity at the junction corner, stress concentration effect causes very high local stress in this area. Stresses at the critical location are not as severe as those of the 9 m (30-ft) side impact. The highest impact stress intensities are approximately 413.69 MPa (60 ksi). When combined with internal pressure, the total stress intensities are approximately 434.37 MPa (63 ksi). This is within the Service Level D limits for the peak stress range for Type 304 stainless steel. The highest strains are approximately 31.4 percent at the containment boundary of the flange and 23.2 percent on the wall and below the ultimate rupture strain of stainless steel, which is 35 percent. The critical elements in the containment parts are shown in Table B7-16.

Figure B7-45 shows the history of stress intensity at EL 2971, and Figure B7-46 show the logarithmic strain at EL 2970. Both elements are in the containment wall. Figures B7-47 and B7-48 show the history of stress intensity and logarithmic strain at EL 4370, which is on the containment flange inside surface. Figures B7-49 and B7-50 show the history of stress intensity and logarithmic strain at the lid bolt EL 8914. As with the 9 m (30 ft) side drop, the stresses in the 9 m (30-ft) edge drop are high and near the material rupture strength. The high stress and strain, however, only exist at local regions where the containment boundary faces are joined together. Due to the local stress/strain concentration effect, these stresses and strains are classified as peak stress in ASME stress classification, which also includes secondary stresses. Consequently, these peak stresses are below the Service Level D allowable range. In general, as a result of this effect, the local loading existing after the impact will not cause the breach of the containment boundary because the strains are below the ultimate rupture strain. However, large deformations of the SWDB Overpack occur in these highly strained localized regions. The deformation of the SWDB containment boundary of the transport package is shown in Figure B7-51. The deformation of containment boundary stiffeners on the impact side is shown in Figure B7-52 in which crushes of outside stiffeners can be readily seen in this symmetrical half-section plot. Not shown in Figure B7-52, the center stiffener on the other side from the impact end is crushed completely.

Table B7-16. Results of American Society of Mechanical Engineers Stress Intensity and Logarithmic Strain
9 m (30-ft) Top-Edge Impact.

Element	S.I. (ksi)	Log. strain	Element	S.I. (ksi)	Log. strain
Containment bottom			Containment wall		
588	20.81	0.01598	2860	9.97	0.10810
637	21.53	0.01124	2960	17.09	0.18040
638	15.65	0.01538	2970	47.85	0.23170
401	25.49	0.04924	2971	56.78	0.08938
451	43.98	0.03200	2972	16.40	0.07708
501	34.85	0.03173	----	----	----
551	11.66	0.06115	----	----	----
601	36.60	0.05212	----	----	----
Containment flange			Containment lid		
4370	61.24	0.33380	7792	30.06	0.01959
4371	34.21	0.33120	7793	20.57	0.08781
4372	45.47	0.29130	7794	30.13	0.09485
4373	51.03	0.25900	7842	19.45	0.04637
4770	41.58	0.32650	7843	26.89	0.06155
4771	34.51	0.30240	7844	17.31	0.05265
4772	39.12	0.28310	----	----	----
Below not containment boundary			----	----	----
4672	38.14	0.49390	----	----	----
4673	25.49	0.47750	----	----	----
4674	26.47	0.45610	----	----	----
Containment lid bolt					
8911	6.09	0.00043	8914	72.48	0.08160
8912	21.23	0.00608	8915	46.74	0.05372
8913	22.76	0.02748	8916	23.83	0.06068

Figure B7-45. Stress Intensity of Element 2971
in 9 m (30-ft) Top-Edge Impact.

Figure B7-46. Logarithmic Strain of Element 2970
in 9 m (30-ft) Top-Edge Impact.

Figure B7-47. Stress Intensity of Element 4370
in 9 m (30-ft) Top-Edge Impact.

Figure B7-48. Logarithmic Strain of Element 4370
in 9 m (30-ft) Top-Edge Impact.

Figure B7-49. Stress Intensity of Lid Bolt Element 8914
in 9 m (30-ft) Top-Edge Impact.

Figure B7-50. Logarithmic Strain of Lid Bolt Element 8914
in 9 m (30-ft) Top-Edge Impact.

Figure B7-51. Deformation Plot of 22-Ton Steel Waste Package in 9 m (30-ft) Top-Edge Impact.

Figure B7-52. Deformation Plot of 22-Ton Steel Waste Disposal Box Overpack Containment Boundary Stiffener in 9 m (30-ft) Top-Edge Impact.

7.6 PUNCTURE

In accordance with Boehnke (1999), the 22-Ton SWDB is evaluated for puncture (see Part B, Section 7.9.3 for analysis). Essentially, the 22-Ton SWDB is dropped a distance of 1 m (40 in.) onto the upper end of a solid, vertical, cylindrical, mild steel 15 cm (6-in.-) diameter pin. As shown in Part B, Section 7.9.3, the SWP meets the criteria for the puncture analysis.

7.7 STRUCTURAL EVALUATION AND CONCLUSIONS

From the above evaluations, it is shown that the drop orientations that cause the most severe loading to the exterior of the 22-Ton SWDB are the side and top-edge impacts. In these cases, the exterior locations of the 22-Ton SWDB are subject to highly localized stresses; however, these stresses do not affect the containment performance of the 22-Ton SWDB because the stresses are within allowable ASME stress limits and accumulated strains are below the maximum material rupture strain. The maximum normal operating temperature of the 22-Ton SWDB is 152 °C (306 °F). Table B7-17 shows the containment boundary for the 22-Ton SWDB.

Table B7-17. Containment Boundary Components and Materials for the 22-Ton Steel Waste Disposal Box.

Component	Material
Overpack assembly/flange	304 SS (ASTM SA-240)
Bolts	ASTM A325

ASTM = American Society for Testing and Materials.
SS = stainless steel.

7.7.1 NCT Evaluation and Conclusions

Under the Hanford Site normal conditions (see Part B, Section 7.4.), it must be demonstrated that the design of the 22-Ton SWDB maintains containment and shielding. The containment boundary is defined in Table B7-24. Stress allowables used in the analyses are taken from Regulatory Guide 7.6, *Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels* (NRC 1978), and are compatible with the (ASME 1995d) Article NE-3000. These stress allowables, combined with both classical and analytical methods, are used to demonstrate containment during NCT. ASME (1995b) Service Levels A and C criteria are used for analytical acceptance. The allowable stresses are given in Table B7-2.

The ASME BPVC stress intensity used for comparison to ASME allowables is defined as the largest difference among the principal stresses. In all evaluation cases, the results of the finite element calculations are retrieved and processed to obtain the ASME BPVC stress intensities. Comparison of ASME BPVC stress intensities versus the ASME BPVC allowables are used to demonstrate the structural integrity and containment performance of the cask except

in critical areas where plastic deformation is a primary factor. In critical areas of plastic deformation, the material strain is evaluated.

Tables B7-3 and B7-4 show the containment boundary normal condition allowables for Levels A and Level C, respectively.

7.7.1.1 0.3 m (1-ft) Side Impact. The stresses and strains for the 0.3 m (1-ft) side impact are relatively low except at local areas of discontinuity, such as at the junction of the wall and bottom plate on the opposite side of the impact end. The center portion of this junction is slightly pushed inward by the reaction of the inner line of the SWDB Impact Limiter and Shield Plate. The highest combined stresses on the containment bottom is located at elements 1 (322.74 MPa [46.81 ksi]) and 2 (391.42 MPa [56.77 ksi]). The highest combined stresses on the containment wall are located at elements 1001 (332.33 MPa [48.20 ksi]), 2979 (321.16 MPa [46.58 ksi]), and 2983 (318.95 MPa [46.26 ksi]). The stresses on the containment flange and lid are low as compared with the other containment boundary wall stresses. The combined highest stresses on the bolts occur at elements 8910 (450.30 MPa [65.31 ksi]), 8911 (405.07 MPa [58.75 ksi]), and 8912 (357.49 MPa [51.85 ksi]).

7.7.1.2 0.3 m (1-ft) Top-Edge Impact. The critical stress locations for the 0.3 m (1-ft) top-edge impact are at the junction corner of the containment wall and the containment lid. The highest combined stresses on the containment wall are located at elements 2970 (216.29 MPa [31.37 ksi]) and 2971 (188.09 MPa [27.28 ksi]). The highest combined stress on the containment lid is located at element 7044 (287.51 MPa [41.70 ksi]). The stresses on the containment bottom and flange are low compared with the other containment boundary stresses. The highest stresses on the bolts occur at elements 8907 (441.95 MPa [64.10 ksi]) and 8911 (402.65 MPa [58.40 ksi]).

7.7.1.3 Stress Intensity Allowables. The 22-Ton SWDB meets the stress intensity allowables that are specified in Table B7-4 for Service Level C requirements when impacted on the side and top-edge from a height of 0.3 m (1 ft) under maximum differential pressure loading. Because the highest stresses occur at locations of gross discontinuity, the allowable range of primary and secondary stress intensity is used. In all cases, the stress intensities are below the Service Level C allowables for the range of primary and secondary stresses.

The 22-Ton SWDB screws are manufactured to the requirements of ASTM A-325, which has an average stress intensity of 348.87 MPa (50.60 ksi) at 151 °C (306 °F). For the 0.3 m (1-ft) side and top-edge impacts, all stresses were below the Service Level C maximum combined stress allowable.

7.7.2 Accident Conditions Evaluation and Conclusions

Under accident conditions, verification of the design must demonstrate that the 22-Ton SWDB maintains containment and shielding. Structural adequacy is demonstrated by showing the stress in the containment boundary does not exceed the limits specified for Service Level D limits of the ASME (1995d). Guidance for stress allowables is obtained from NRC (1978).

NRC (1978) recommends the containment boundary stress allowables in Table B7-5, which are comparable to Article F-1000 of the ASME (1995c) appendices. These stress allowables coupled with linear-elastic or plastic-elastic analysis are used to demonstrate maintenance of containment during accident conditions loading. The ASME (1995d) Service Level D criteria are used for analytical acceptance. Table B7-6 shows the allowable stress as a function of temperature.

7.7.2.1 9 m (30-ft) Side Impact. The critical locations for the 9 m (30-ft) side impact occur at the junction corner of the containment wall and flange on the impact side. Due to the highly localized discontinuity at the junction corner, stress concentration effects cause high local stress in this area and are considered by ASME (1995b) as peak stresses. The highest combined stresses at the containment wall are located at elements 2970 (504.77 MPa [73.21 ksi]), 2870 (439.13 MPa [63.69 ksi]), and 2971 (429.75 MPa [62.33 ksi]). The highest combined stresses at the containment flange are located at elements 4373 (485.46 MPa [70.41 ksi]), 4370 (441.61 MPa [64.05 ksi]), and 4372 (432.65 MPa [62.75 ksi]). The stresses on the containment bottom and lid are low compared with the other containment boundary stresses. The highest stress on the bolts occurs at element 8913 (444.85 MPa [64.52 ksi]).

7.7.2.2 9 m (30-ft) Top-Edge Impact. The critical locations for the 9 m (30-ft) top-edge impact occur at the junction corner of the containment wall and the flange on the impact side. Due to the highly localized discontinuity at the junction corner, stress concentration effect causes high local stress in this area and is considered by ASME (1995d) as peak stresses. The highest combined stresses at the containment wall are located at elements 2971 (412.17 MPa [59.78 ksi]) and 2970 (350.60 MPa [50.85 ksi]). The highest combined stresses at the flange are located at elements 4370 (432.58 MPa [62.74 ksi]) and 4373 (362.18 MPa [52.53 ksi]). The stresses on the containment bottom and lid are low compared with the other containment boundary stresses. The highest combined average tensile stress on the bolts occurs at element 8914 (561.78 MPa [81.48 ksi]).

7.7.2.3 Stress Intensity Allowables. The 22-Ton SWDB closure bolts are manufactured to the requirements of ASTM A-325, which has an average tension stress allowable of 577.23 MPa (83.72 ksi) at 151 °C (306 °F). The combined loading in all accident condition cases is shown to not be significant. The closure bolts remain elastic, therefore, maintaining containment of the 22-Ton SWDB.

In the above evaluations, the Service Level D primary membrane and bending allowable stress intensities are exceeded at one or more locations. These locations are at gross discontinuities; subsequently, NRC (1978) permits using the peak stress intensity range allowable at 10 cycles when all loads are considered. As shown by the values in Table B7-6, the estimated stress intensities are well below the permissible limit. However, under dynamic load conditions, the stress values at a particular location fluctuate greatly relative to time and do not stabilize until well after the impact. This is due to the wave action in the member generated by the force and the various elasticity and plasticity states the material experiences as the load varies with time. The result is more of an estimation of the stress intensity rather than a direct reading for an assessment of the loading on the member. Subsequently, in this evaluation a more applicable assessment criterion for the components is to evaluate the accumulated strain

generated by the impact and combine it with the material strains generated from the internal pressure. Assessing the accumulated combined strains provides a more stable and accurate assessment of the material damage at the location of concern.

Consequently, in all cases the significant localized deformations will occur in the member at gross discontinuities. However, because the combination of maximum stress intensity and maximum strains in the members is below the allowable material stress intensity and ultimate strain, no material rupture will occur, and containment is maintained.

7.7.3 Conclusions

This structural evaluation of the 22-Ton SWDB demonstrates that it will sustain the loads of the packaging performance tests of Boehnke (1999) under both normal and accident conditions and maintain containment. Ability of the package to sustain the NCT performance test loads is demonstrated by meeting the Service Levels A and C stress intensity limits defined in ASME (1995d) and by demonstrating the accumulated strains remain below material rupture. Maintaining containment when subject to the performance test loads for accident conditions is demonstrated by meeting Service Level D stress intensity limits defined in ASME (1995d) and by demonstrating the accumulated strains remain below material rupture. Maintaining containment after a puncture test is demonstrated by evaluating the performance of the package by determining the minimum thickness required to prevent puncture. The resulting positive margins of safety shows there is sufficient material thickness to prevent puncture and maintain containment.

7.8 REFERENCES

- ANON, 1980, *The Encyclopedia of Wood*, Coles Publishing Company Limited, Toronto.
- ASME, 1995a, *Boiler and Pressure Vessel Code*, Section II, "Material," "Part A—Ferrous Materials," American Society of Mechanical Engineers, New York, New York.
- ASME, 1995b, *Boiler and Pressure Vessel Code*, Section II, "Material," "Part D—Properties," American Society of Mechanical Engineers, New York, New York.
- ASME, 1995c, *Boiler and Pressure Vessel Code*, Section III, Division 1, "Appendices," American Society of Mechanical Engineers, New York, New York.
- ASME, 1995d, *Boiler and Pressure Vessel Code*, Section III, Subsection NE, American Society of Mechanical Engineers, New York, New York.
- ASME, 1995e, *Boiler and Pressure Vessel Code*, Section VIII, Division 1, "Pressure Vessels," American Society of Mechanical Engineers, New York, New York.

ASTM, 1993, *ASTM Standard Specification for Structural Steel*, Designation: A36/A36M-93a, American Society for Testing and Materials, Philadelphia, Pennsylvania.

ASTM, 1992, *ASTM Standard Specification for Electric-Resistance-Welded Carbon and Alloy Steel Mechanical Tubing*, Designation: A513-92, American Society for Testing and Materials, Philadelphia, Pennsylvania.

Boehnke, W. M., 1999, *Packaging Design Criteria for the Steel Waste Package*, HNF-3723, Rev. 0, prepared by Waste Management Federal Services, Inc., Northwest Operations for Fluor Daniel Hanford, Inc., Richland, Washington.

DOD, 1994, *Military Handbook-Metallic Materials and Elements for Aerospace Vehicle Structures*, MIL-HDBK-5G, U.S. Department of Defense, Washington, D.C.

HKS, 1998, *ABAQUS/Explicit User's Manual*, Version 5.8, Hibbitt, Karlsson & Sorensen, Inc., Pawtucket, Rhode Island.

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.

Mok, G. C., 1989, *Stress Analysis of Closure Bolts for Shipping Casks*, NUREG/CR-6007 (under Lawrence Livermore National Laboratory contract to the U.S. Nuclear Regulatory Commission), U.S. Nuclear Regulatory Commission, Washington, D.C.

NRC, 1991, *Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 Inches*, Regulatory Guide 7.11, U.S. Nuclear Regulatory Commission, Washington, D.C.

NRC, 1978, *Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels*, Regulatory Guide 7.6, Rev. 1, U.S. Nuclear Regulatory Commission, Washington, D.C.

NRC, 1977, *Load Combinations for the Structural Analysis of Shipping Casks*, Regulatory Guide 7.8, U.S. Nuclear Regulatory Commission, Washington, D.C.

Parker Seal, 1992, *Parker O-Ring Handbook*, ORD 5700, Parker Seal Group, O-Ring Division, Parker Hannifin Corporation, Cleveland, Ohio.

PNL, 1996c, *22 Ton Steel Waste Disposal Box Assembly/Details*, drawing H-3-56431, Rev. 1 (4 sheets), Pacific Northwest Laboratory, Richland, Washington.
Sheets 1 and 2, *22 Ton Steel Waste Disposal Box Assembly*.
Sheets 3 and 4, *22 Ton Steel Waste Disposal Box Details*.

PNL, 1988a, *22 Ton Steel Waste Disposal Box Arrangement*, drawing H-3-56430, Rev. 0, Pacific Northwest Laboratory, Richland, Washington.

PNL, 1988b, *22 Ton Steel Waste Dspl Box Impact Limiter Assy/Det*, drawing H-3-56433, Rev. 0 (9 sheets), Pacific Northwest Laboratory, Richland, Washington.

Sheet 1, *22 Ton Steel Waste Dspl Box Impact Limiter Assy*.

Sheets 2 thru 9, *22 Ton Steel Waste Dspl Box Impact Limiter Det*.

RL, 1993, *Hanford Site Hoisting and Rigging Manual*, DOE/RL-92-36, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

CHECKLIST FOR CHECKING OF ANALYSIS/CALCULATIONS
Document Number/Revision: <u>HNF-4763/0 ABAQUS/Explicit 22 Ton Package FEA Structural Analysis</u>			
Document Title: <u>Safety Analysis Report for Packaging (Onsite) Steel Waste Package</u>			
Yes	No	N/A	
☒	☐	☐	Problem completely defined.
☒	☐	☐	Appropriate analytical method used.
☒	☐	☐	Necessary assumptions are appropriate and explicitly stated.
☒	☐	☐	Computer codes and data files documented.
☒	☐	☐	Data used in calculations explicitly stated in document.
☐	☐	☒	Sources of non-standard formula/data are referenced and the correctness of the reference verified.
☒	☐	☐	Data checked for consistency with original source information as applicable.
☐	☐	☒	Mathematical derivations checked including dimensional consistency of results.
☒	☐	☐	Models appropriate and used within range of validity or use outside range of established validity justified.
☐	☐	☒	Hand calculations checked for errors.
☒	☐	☐	Code run streams correct and consistent with analysis documentation.
☒	☐	☐	Code output consistent with input and with results reported in analysis documentation.
☒	☐	☐	Acceptability limits on analytical results applicable and supported. Limits checked against sources.
☒	☐	☐	Safety Margins consistent with good engineering practices.
☒	☐	☐	Conclusions consistent with analytical results and applicable limits.
☒	☐	☐	Results and conclusions address all points required in the problem statement.
I have checked the analysis/calculation and it is complete and accurate to the best of my knowledge.			
Engineer/Checker <u>Scott Shiraga</u>			Date <u>02/04/00</u>
NOTE: Any hand calculations, notes or summaries generated as part of this check should be signed, dated, and attached to this checklist. Material should be labeled and recorded so that it is intelligible to a technically qualified third party.			

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7.9 APPENDICES

7.9.1 Lifting Bail Design

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Checker: DL Riley <i>DL Riley</i> per <i>Don Riley per Tekon</i>	Date: 9/22/99

1.0 OBJECTIVE

The objective of this analysis is to design and analyze the lifting bail of the 22 Ton Steel Waste Disposal Box. The new design takes the original lifting bail and adds a hinge allowing the bail to fold over towards the lid. A spring is used to hold the top of the bail at a 52 degree angle so the bail can be remotely pushed to the vertical position. Since the new design does not change the attachment to the box or the top of the bail which interfaces with the original lifting yoke, this analysis is limited to the design of the hinge.

The lifting bails must withstand a factor of safety for lifting of 3:1 to yield.

2.0 REFERENCES

- AISC, 1989, *Manual of Steel Construction*, Allowable Stress Design, 9th Edition, American Institute of Steel Construction, Inc., Chicago, Illinois.
- PNL, 1988, *22 Ton Steel Waste Disposal Box Arrangement*, drawing H-3-56431, Rev. 1, Pacific Northwest Laboratory, Richland, Washington.
- Shigley, Joseph E, and Charles R. Mischke, 1989, *Mechanical Engineering Design*, Fifth Edition, McGraw-Hill, Inc. New York, New York.
- DWS, 1999, *Lifting Bail Spring*, (Fax to D. Reid, 9/23/99, 13:29), 800-238-3510, Diamond Wire Spring Co., Tyler, Texas.

3.0 ASSUMPTIONS, RESULTS, AND CONCLUSIONS

The lifting bail design consists of three main components. The components are the top and bottom parts of the bail, the hinge pin, and the spring. The maximum weight of the package that is used in the calculations is conservatively estimated to be 30,000 lb on each lifting bail (60,000 lb total). The margin of safety for each evaluation includes a lifting factor of 3:1 to yield.

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The hinge pin is made of AISI 41L40 with a yield stress of 120,000 psi. A two step process was used to determine the bending of the pin for this design. The first step used the formulas for continuous beams, with the theory of three moments. The second step utilizes the simple beam formulas for a uniformly distributed load and variable end moments. The end moments are taken from the moments determined from the first step and added in the second step to find the maximum moment for each span. The moments are symmetric about the center so only one side of the maximum moments are determined. The maximum moment is found on the forth span to be 2,151 in-lb. The maximum bending stress in the pin is thus 32,706 psi which gives a margin of safety of 0.22. The largest shear stress of the pin is 6,603 psi with a margin of safety of 2.03. The largest bending stress at the reaction points is 9,032 psi with a margin of safety of 3.43. From the bending and shear stresses the principle stresses are determined and are shown to be 9,871 psi and 6,603 psi for shear. The margin of safety is 3.05 and 0.22 respectively. The bearing stress of the pin is 4,898 psi with a margin of safety of 3.08.

The top and bottom bails were evaluated for direct tensile stress, shear tear-out stress, and bearing stress using AISI 1040 with a yield stress of 42,000 psi. Both the top and bottom bails have the same area for hinge teeth. Therefore, stresses will be the same for both top and bottom bails. The direct tensile stress is 3,820 psi with a margin of safety of 2.67. The shear tear-out stress is 4,193 psi with a margin of safety of 0.67. The bearing stress is 4,881 psi with a margin of safety of 1.87.

The maximum moment required to lift the top bail by the spring was calculated to be 825 in-lb about the center of the pin. At 60 degrees the moment required is 413 in-lb. Spring $\phi 2.0 \times 3.86 \times \phi .374$ 17-7 by Diamond Wire Spring Co. was determined to sufficiently provide the required moments and allows the lifting bail to be positioned at a steady state angle of 52 degrees.

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Evaluation of Pin

Using the equation for the theorem of three moments (AISC 1989, 2-294):

$$M_1 L_1 + 2M_2 (L_1 + L_2) + M_3 L_2 + \frac{w L_1^3}{4} + \frac{w L_2^3}{4}$$

Assume there are no gaps between the hinge teeth.

$$F := 30000$$

$$L_1 := 1.17 \quad L_2 := 1.75 \quad L_3 := 4.4$$

$$w_1 := \frac{F}{6 \cdot 1.17}$$

$$w_2 := \frac{-F}{4 \cdot 1.75}$$

$$w_1 = 4273.5$$

$$w_2 = -4285.71$$

$$w_3 := 0$$

$$M_1 := 0$$

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Guess values
for Mathcad:

$$M_1 := 0 \quad M_7 := -150 \quad M_{13} := 0$$

$$M_2 := 500 \quad M_8 := 600$$

$$M_3 := 450 \quad M_9 := 450$$

$$M_4 := 450 \quad M_{10} := 450$$

$$M_5 := 600 \quad M_{11} := 500$$

$$M_6 := -150 \quad M_{12} := 200$$

Given due to symmetry

$$M_2 = M_{11} \quad M_4 = M_9 \quad M_6 = M_7$$

$$M_3 = M_{10} \quad M_5 = M_8$$

Equation 1

$$M_1 \cdot L_1 + 2 \cdot M_2 \cdot (L_1 + L_2) + M_3 \cdot L_2 + \frac{w_1 \cdot L_1^3}{4} + \frac{w_2 \cdot L_2^3}{4} = 0$$

Equation 2

$$M_3 \cdot L_1 + 2 \cdot M_4 \cdot (L_1 + L_2) + M_5 \cdot L_2 + \frac{w_1 \cdot L_1^3}{4} + \frac{w_2 \cdot L_2^3}{4} = 0$$

Equation 3

$$M_5 \cdot L_1 + 2 \cdot M_6 \cdot (L_1 + L_3) + M_7 \cdot L_3 + \frac{w_1 \cdot L_1^3}{4} + \frac{w_3 \cdot L_3^3}{4} = 0$$

Equation 4

$$M_7 \cdot L_1 + 2 \cdot M_8 \cdot (L_1 + L_2) + M_9 \cdot L_2 + \frac{w_1 \cdot L_1^3}{4} + \frac{w_2 \cdot L_2^3}{4} = 0$$

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Equation 5

$$M_9 \cdot L_1 + 2 \cdot M_{10} (L_1 + L_2) + M_{11} \cdot L_2 + \frac{w_1 \cdot L_1^3}{4} + \frac{w_2 \cdot L_2^3}{4} = 0$$

Equation 6

$$M_{12} - w_1 \frac{L_1^2}{8} + \frac{M_{11} + M_{13}}{2} - \frac{(M_{11} - M_{13})^2}{2 \cdot w_1 \cdot L_1^2} = 0$$

$$\begin{bmatrix} M_1 \\ M_2 \\ M_3 \\ M_4 \\ M_5 \\ M_6 \\ M_7 \\ M_8 \\ M_9 \\ M_{10} \\ M_{11} \\ M_{12} \\ M_{13} \end{bmatrix} := \text{Find}(M_1, M_2, M_3, M_4, M_5, M_6, M_7, M_8, M_9, M_{10}, M_{11}, M_{12}, M_{13})$$

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Results:

$$\begin{array}{lll}
 M_1 = 0 & M_6 = -154.84 & M_{11} = 559.1 \\
 M_2 = 559.1 & M_7 = -154.84 & M_{12} = 478.4 \\
 M_3 = 437.65 & M_8 = 594.05 & M_{13} = 0.02 \\
 M_4 = 424.56 & M_9 = 424.56 & \\
 M_5 = 594.05 & M_{10} = 437.65 &
 \end{array}$$

Largest moment is $M_5 = M_8 = 594.05$ in-lb.

Find maximum moment between spans

Span 1

$$M_{\max 1} := \frac{w_1 \cdot L_1^2}{8} - \frac{M_1 + M_2}{2} + \frac{(M_1 - M_2)^2}{2 \cdot w_1 \cdot L_1^2}$$

$$M_{\max 1} = 478.42$$

Span 2

$$M_{\max 2} := \frac{w_2 \cdot L_2^2}{8} - \frac{M_2 + M_3}{2} + \frac{(M_2 - M_3)^2}{2 \cdot w_2 \cdot L_2^2}$$

$$M_{\max 2} = -2139.57$$

Span 3

$$M_{\max 3} := \frac{w_1 \cdot L_1^2}{8} - \frac{M_3 + M_4}{2} + \frac{(M_3 - M_4)^2}{2 \cdot w_1 \cdot L_1^2}$$

$$M_{\max 3} = 300.16$$

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Span 4

$$M_{\max 4} := \frac{w_2 \cdot L_2^2}{8} - \frac{M_4 + M_5}{2} + \frac{(M_4 - M_5)^2}{2 w_2 L_2^2}$$

$$M_{\max 4} = -2151.02$$

Span 5

$$M_{\max 5} := \frac{w_1 \cdot L_1^2}{8} - \frac{M_5 + M_6}{2} + \frac{(M_5 - M_6)^2}{2 w_1 L_1^2}$$

$$M_{\max 5} = 559.58$$

Span 6

$$M_{\max 6} := \frac{w_3 \cdot L_3^2}{8} - \frac{M_6 + M_7}{2}$$

$$M_{\max 6} = 154.84$$

Maximum moment is $M_{\max 4} = -2151.02$ in-lb.

Find reaction forces to determine shear:

Span 1

$$R_1 := \frac{w_1 \cdot L_1}{2} - \frac{M_1 - M_2}{L_1}$$

$$R_1 = 2977.87$$

$$R_{2a} := \frac{w_1 \cdot L_1}{2} + \frac{M_1 - M_2}{L_1}$$

$$R_{2a} = 2022.13$$

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Span 2

$$R_{2b} := \frac{w_2 \cdot L_2}{2} + \frac{M_2 - M_3}{L_2}$$

$$R_{3a} := \frac{w_2 \cdot L_2}{2} - \frac{M_2 - M_3}{L_2}$$

$$R_{2b} = -3680.6$$

$$R_{3a} = -3819.4$$

Span 3

$$R_{3b} := \frac{w_1 \cdot L_1}{2} + \frac{M_3 - M_4}{L_1}$$

$$R_{4a} := \frac{w_1 \cdot L_1}{2} - \frac{M_3 - M_4}{L_1}$$

$$R_{3b} = 2511.19$$

$$R_{4a} = 2488.81$$

Span 4

$$R_{4b} := \frac{w_2 \cdot L_2}{2} - \frac{M_4 - M_5}{L_2}$$

$$R_{5a} := \frac{w_2 \cdot L_2}{2} + \frac{M_4 - M_5}{L_2}$$

$$R_{4b} = -3653.15$$

$$R_{5a} = -3846.85$$

Span 5

$$R_{5b} := \frac{w_1 \cdot L_1}{2} + \frac{M_5 - M_6}{L_1}$$

$$R_{6a} := \frac{w_1 \cdot L_1}{2} - \frac{M_5 - M_6}{L_1}$$

$$R_{5b} = 3140.07$$

$$R_{6a} = 1859.93$$

Span 6

$$R_{6b} := \frac{M_6 - M_7}{L_3}$$

$$R_{7a} := \frac{M_6 - M_7}{L_3}$$

$$R_{6b} = 0$$

$$R_{7a} = 0$$

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Add reactions together for each span

$$R_2 := R_{2a} + R_{2b}$$

$$R_4 := R_{4a} + R_{4b}$$

$$R_6 := R_{6a} + R_{6b}$$

$$R_3 := R_{3a} + R_{3b}$$

$$R_5 := R_{5a} + R_{5b}$$

$$R_1 = 2977.87$$

$$R_2 = -1658.47$$

$$R_3 = -1308.21$$

$$R_4 = -1164.34$$

$$R_5 = -706.78$$

$$R_6 = 1859.93$$

Maximum reaction force is $R_1 = 2977.87$ lbf.

Determine Shear Stresses at reaction points from the beam:

$$A_{pin} := \frac{\pi}{4} (.875)^2$$

$$A_{pin} = 0.601 \text{ in}^2$$

$$\tau_1 := \frac{4}{3} \frac{R_1}{A_{pin}}$$

$$\tau_1 = 6602.95 \text{ psi}$$

$$\tau_2 := \frac{4}{3} \frac{R_2}{A_{pin}}$$

$$\tau_2 = -3677.39 \text{ psi}$$

$$\tau_3 := \frac{4}{3} \frac{R_3}{A_{pin}}$$

$$\tau_3 = -2900.75 \text{ psi}$$

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$$\tau_4 := \frac{4}{3} \frac{R_4}{A_{pin}} \quad \tau_4 = -2581.74 \text{ psi}$$

$$\tau_5 := \frac{4}{3} \frac{R_5}{A_{pin}} \quad \tau_5 = -1567.17 \text{ psi}$$

$$\tau_6 := \frac{4}{3} \frac{R_6}{A_{pin}} \quad \tau_6 = 4124.1 \text{ psi}$$

Largest Shear Stress is at $\tau_1 = 6602.95 \text{ psi}$

Determine the bending stress at the reaction points:

$$c = 0.44 \text{ in} \quad I = 0.0288 \text{ in}^4$$

$$\sigma_1 := \frac{M_1 \cdot c}{I} \quad \sigma_1 = 0 \text{ psi}$$

$$\sigma_2 := \frac{M_2 \cdot c}{I} \quad \sigma_2 = 8500.97 \text{ psi}$$

$$\sigma_3 := \frac{M_3 \cdot c}{I} \quad \sigma_3 = 6654.36 \text{ psi}$$

$$\sigma_4 := \frac{M_4 \cdot c}{I} \quad \sigma_4 = 6455.26 \text{ psi}$$

$$\sigma_5 := \frac{M_5 \cdot c}{I} \quad \sigma_5 = 9032.29 \text{ psi}$$

$$\sigma_6 := \frac{M_6 \cdot c}{I} \quad \sigma_6 = -2354.23 \text{ psi}$$

Largest bending stress at the reaction points is at $\sigma_5 = 9032.29 \text{ psi}$

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Determine the maximum principal stresses at the reaction points:

$$\sigma_{p1} := \frac{\sigma_1}{2} + \left[\left(\frac{\sigma_1}{2} \right)^2 + \tau_1^2 \right]^{.5} \quad \sigma_{p1} = 6602.95 \text{ psi}$$

$$\sigma_{p2} := \frac{\sigma_2}{2} + \left[\left(\frac{\sigma_2}{2} \right)^2 + \tau_2^2 \right]^{.5} \quad \sigma_{p2} = 9870.97 \text{ psi}$$

$$\sigma_{p3} := \frac{\sigma_3}{2} + \left[\left(\frac{\sigma_3}{2} \right)^2 + \tau_3^2 \right]^{.5} \quad \sigma_{p3} = 7741.3 \text{ psi}$$

$$\sigma_{p4} := \frac{\sigma_4}{2} + \left[\left(\frac{\sigma_4}{2} \right)^2 + \tau_4^2 \right]^{.5} \quad \sigma_{p4} = 7360.79 \text{ psi}$$

$$\sigma_{p5} := \frac{\sigma_5}{2} + \left[\left(\frac{\sigma_5}{2} \right)^2 + \tau_5^2 \right]^{.5} \quad \sigma_{p5} = 9296.48 \text{ psi}$$

$$\sigma_{p6} := \frac{\sigma_6}{2} + \left[\left(\frac{\sigma_6}{2} \right)^2 + \tau_6^2 \right]^{.5} \quad \sigma_{p6} = 3111.68 \text{ psi}$$

Largest Principal Stress is $\sigma_{p2} = 9870.97 \text{ psi}$

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Determine Maximum Principal Shear Stress:

$$\tau_{p1} := \left[\left(\frac{\sigma_1}{2} \right)^2 + \tau_1^2 \right]^{.5} \quad \tau_{p1} = 6602.95 \text{ psi}$$

$$\tau_{p2} := \left[\left(\frac{\sigma_2}{2} \right)^2 + \tau_2^2 \right]^{.5} \quad \tau_{p2} = 5620.48 \text{ psi}$$

$$\tau_{p3} := \left[\left(\frac{\sigma_3}{2} \right)^2 + \tau_3^2 \right]^{.5} \quad \tau_{p3} = 4414.12 \text{ psi}$$

$$\tau_{p4} := \left[\left(\frac{\sigma_4}{2} \right)^2 + \tau_4^2 \right]^{.5} \quad \tau_{p4} = 4133.16 \text{ psi}$$

$$\tau_{p5} := \left[\left(\frac{\sigma_5}{2} \right)^2 + \tau_5^2 \right]^{.5} \quad \tau_{p5} = 4780.33 \text{ psi}$$

$$\tau_{p6} := \left[\left(\frac{\sigma_6}{2} \right)^2 + \tau_6^2 \right]^{.5} \quad \tau_{p6} = 4288.8 \text{ psi}$$

Largest Principal Shear Stress is $\tau_{p1} = 6602.95 \text{ psi}$

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Determine Bending Stresses from maximum bending moments between reaction points:

Maximum moment is: $M_{max4} = -2151.02 \text{ in} \cdot \text{lb}$

$$c := .5 \cdot \frac{7}{8} \quad I := \frac{3}{4} (c)^4$$

$$\sigma_b := |M_{max4}| \cdot \frac{c}{I}$$

$$\sigma_b = 32705.52 \text{ psi}$$

Determine Bearing Stress on pin:

Area length of teeth on hinge $L := 7.0 \text{ in}$ Diameter of pin $D_p := .875 \text{ in}$

$$\text{Bearing stress } \sigma_{br} := \frac{F}{L D_p} \quad \sigma_{br} = 4897.96 \text{ psi}$$

Pin Allowable Stresses:

AISI 41L40 - 120,000 psi yield strength

Allowable including a 3:1 Factor of Safety for lifting

$$\sigma_{all} := \frac{1}{3} \cdot 120000 \quad \sigma_{all} = 40000 \text{ psi}$$

Shear allowable using maximum shear stress theory predicts yield stress for shear at $\frac{1}{2} \sigma_{all}$ (Shigley, 1989)

$$\tau_{all} := .5 \cdot \sigma_{all} \quad \tau_{all} = 20000 \text{ psi}$$

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Compare maximum stresses to allowables:

Largest shear stress at reaction points

$$\tau_1 = 6602.95 \text{ psi} \quad MS_a := \frac{\tau_{all}}{\tau_1} - 1 \quad MS_a = 2.03$$

Largest bending stress at reaction points and margin of safety

$$\sigma_5 = 9032.29 \text{ psi} \quad MS_b := \frac{\sigma_{all}}{\sigma_5} - 1 \quad MS_b = 3.43$$

Largest principal stresses at reaction points and margin of safety

$$\sigma_{p2} = 9870.97 \text{ psi} \quad MS_c := \frac{\sigma_{all}}{\sigma_{p2}} - 1 \quad MS_c = 3.05$$

Largest principal shear stress at reaction points and margin of safety

$$\tau_{p1} = 6602.95 \text{ psi} \quad MS_d := \frac{\tau_{all}}{\tau_{p1}} - 1 \quad MS_d = 0.22$$

Largest bending stress between reaction points and margin of safety

$$\sigma_b = 32705.52 \text{ psi} \quad MS_d := \frac{\sigma_{all}}{\sigma_b} - 1 \quad MS_d = 0.22$$

Bearing stress on pin and margin of safety

$$\sigma_{br} = 4897.96 \text{ psi} \quad MS_e := \frac{\tau_{all}}{\sigma_{br}} - 1 \quad MS_e = 3.08$$

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Evaluation of Lifting Bails

Both top and bottom lifting bails have the same total length of teeth of 7.0 in. Therefore, stresses will be equal. Loading for each bail is 30,000 lbf.

Width of bail: $W := 2.0 \text{ in}$ Total length of teeth: $L := 7.0 \text{ in}$ Diameter of shaft hole: $\phi h := .878 \text{ in}$

Length of shear tear-out path:

$$L_s := \left(.90 \text{ in} - \frac{\phi h}{2} \right) + \left(1.0 \text{ in} - \frac{\phi h}{2} \right) \quad L_s = 1.02 \text{ in}$$

Minimum tensile area: $A_t := L(W - \phi h) \quad A_t = 7.85 \text{ in}^2$ Shear tear-out area: $A_s := L_s \cdot L \quad A_s = 7.15 \text{ in}^2$ Bearing area: $A_b := L \cdot \phi h \quad A_b = 6.15 \text{ in}^2$ Tensile Stress: $\sigma_t := \frac{F}{A_t} \quad \sigma_t = 3819.7 \text{ psi}$ Shear Tear-Out Stress: $\tau_t := \frac{F}{A_s} \quad \tau_t = 4193.45 \text{ psi}$ Bearing Stress: $\sigma_{bs} := \frac{F}{A_b} \quad \sigma_{bs} = 4881.22 \text{ psi}$

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Lifting Bail Allowable Stresses:

AISI 1040 - 42,000 psi yield strength

Allowable including a 3:1 Factor of Safety for lifting

$$\sigma_{b\ all} := \frac{1}{3} 42000 \text{ psi} \quad \sigma_{b\ all} = 14000 \text{ psi}$$

Shear allowable using maximum shear stress theory predicts yield stress for shear at $\frac{1}{2} \sigma_{b\ all}$ (Shigley, 1989)

$$\tau_{b\ all} := .5 \sigma_{b\ all} \quad \tau_{b\ all} = 7000 \text{ psi}$$

Compare maximum stresses to allowables:

Tensile stress and margin of safety

$$\sigma_t = 3819.7 \text{ psi} \quad MS_f := \frac{\sigma_{b\ all}}{\sigma_t} - 1 \quad MS_f = 2.67$$

Shear tear-out stress and margin of safety

$$\tau_t = 4193.45 \text{ psi} \quad MS_g := \frac{\tau_{b\ all}}{\tau_t} - 1 \quad MS_g = 0.67$$

Bearing stress and margin of safety

$$\sigma_{bs} = 4881.22 \text{ psi} \quad MS_h := \frac{\sigma_{b\ all}}{\sigma_{bs}} - 1 \quad MS_h = 1.87$$

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Evaluation of Torsional Spring Force

The torsional spring for this application was analyzed with spring information provided by Diamond Wire Spring Co. The spring has been designed by Diamond Wire Spring Co. to satisfy all of the requirements of the application. The material chosen is stainless steel. The model/part number assigned by Diamond Wire Spring Co. is $\phi 2.0 \times 3.86 \times \phi .374$ 17-7 (DWS 1999).

Angle between spring legs at the relaxed state: $\theta = 55 \text{ deg}$

Width of the spring from end to end: $w = 3.86 \text{ in.}$

Number of turns in the spring: $n = 9.35$

Diameter of the spring wire used: $d = .37 \text{ in.}$

Outer diameter of the spring: $D = 2.00 \text{ in.}$

Volume of top lifting bail (derived by AutoCAD): $V := 404.47 \text{ in}^3$

Density of steel (Shigley, 1989): $\rho := .283 \frac{\text{lb}}{\text{in}^3}$

Weight of top lifting bail: $W := \rho \cdot V$ $W = 114.06 \text{ lb}$

Centroid location (derived by AutoCAD): $Y_{\text{bail}} := 7.21 \text{ in}$

Maximum torque to overcome (at 0 deg):

$$T_0 := W \cdot Y_{\text{bail}} \cos(0 \text{ deg}) \quad T_0 = 825.29 \text{ in lb}$$

Torque at 45 deg:

$$T_{45} := W \cdot Y_{\text{bail}} \cos(45 \text{ deg}) \quad T_{45} = 583.57 \text{ in lb}$$

Torque at 60 deg:

$$T_{60} := W \cdot Y_{\text{bail}} \cos(60 \text{ deg}) \quad T_{60} = 412.65 \text{ in lb}$$

ENGINEERING SAFETY EVALUATION

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Preparer: BD Flanagan *BD*

Date: 9/22/99

Checker: DL Riley *DLR for*

Date: 9/22/99

Determine the steady state angle of the lifting bail.

Spring rate $S_r := 9.7067 \frac{\text{in} \cdot \text{lb}}{\text{deg}}$

Weight of top lifting bail $W = 114.47 \text{ lb}$

Centroid location bail $Y_{\text{bail}} := 7.21 \text{ in}$

Steady state angle

$$T_a := S_r \cdot \phi \cdot \text{deg}$$

$$T_b := W \cdot Y_{\text{bail}} \cos(\phi \cdot \text{deg})$$

$$T_a := T_b$$

$$\phi := 52.16$$

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AutoCAD output for top lifting bail

Volume: 404.4687
 Bounding box: X: -9.5450 -- 9.5450
 Y: -1.0000 -- 16.1000
 Z: -1.0000 -- 1.0000
 Centroid: X: 0.0000
 Y: 7.2082
 Z: 0.0000
 Moments of inertia: X: 30125.8997
 Y: 9383.4752
 Z: 39237.3121
 Products of inertia: XY: 0.0000
 YZ: 0.0000
 ZX: 0.0000
 Radii of gyration: X: 8.6303
 Y: 4.8166
 Z: 9.8493
 Principal moments and X-Y-Z directions about centroid:
 I: 9110.7225 along [1.0000 0.0000 0.0000]
 J: 9383.4752 along [0.0000 1.0000 0.0000]
 K: 18222.1349 along [0.0000 0.0000 1.0000]

WMNW-0013-99

7.9.2 SWDB Overpack Pressure Evaluation **ENGINEERING SAFETY EVALUATION**

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Preparer: SS Shiraga <i>[Signature]</i>	Date: 08/03/99
Checker: RJ Smith <i>[Signature]</i>	Date: 08/10/99

1.0 OBJECTIVE

The primary objective of this evaluation is to demonstrate that the SWDB Overpack maintains containment under the maximum differential design pressure of 16 psi at a maximum normal operating temperature of 300 °F. A second objective is to determine the stresses from the specified pressure loading for evaluation of load combination.

2.0 REFERENCES

- ASME, 1992, *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.
- Blodgett, O. W., 1966, *Design of Welded Structures*, The James F. Lincoln Arc Welding Foundation, Cleveland, Ohio.
- Timoschenko, S., 1959, *Theory of Plates and Shells*, McGraw-Hill Book Company, New York, New York.
- Young, W. C., 1989, *Roark's Formulas for Stress and Strain*, Sixth Edition, McGraw-Hill Book Company, New York, New York

3.0 ASSUMPTIONS, RESULTS, AND CONCLUSIONS

The following parameters are assumed in this evaluation.

- The maximum normal operating temperature is 300 °F.
- The maximum differential pressure across the boundary is 16 psi at the maximum normal operating temperature.
- The material of construction is ASTM A-240, Type 304 stainless steel.

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- The closure bolts are heavy hex structural steel bolts, manufactured to the requirements of ASTM A-325, Type 1. The O-rings are elastomeric and, by definition, are self-energizing. Therefore, no load is put on the bolts.

Results of the evaluation show that under the above assumed pressure loading the margins of safety at the worst-case locations and on the closure bolts are positive. These margins of safety are based on ASME Section III, Subsection NE, Service Level A limits. In most cases, the margins of safety are large (greater than 5). However, at the flange to panel junction the lowest margin of safety is 0.04. The lowest margin of safety on the closure bolts is 0.5 under the combination of tension, bending, and preload. As input for combined loading determinations, the worst-case short-side panel stress is 3.2 ksi, at the junction of the flange and short side panel is 1.5 ksi, and the combined loading on the bolts is 24 ksi.

The evaluation shows that the SWDB Overpack remains leak tight under the differential design pressure loading of 16 psi. This demonstrated by the positive margins of safety based on the Service Level A limits from ASME Section III, Subsection NE.

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Overall height of stiffener section: $h_{stif} := h_1 + t_w$ Inside bend radius: $r := 0.38 \text{ in}$

Cross sectional area of panel between stiffeners: $A_{panel} := b_{stif} t_w$ $A_{panel} = 4.65 \text{ in}^2$

Cross sectional area of stiffeners:

$$A_{stif} := 2 \cdot \left[\left[h_1 - (r + t_{stif}) \right] \cdot t_{stif} \right] + \left[\left(w_1 - 2 \cdot t_{stif} \right) \cdot t_{stif} \right] + \frac{2 \cdot \pi \cdot \left[t_{stif} + r \right]^2 - r^2}{4} + \left[w_1 - 2 \cdot (t_{stif} + r) \right] \cdot t_{stif}$$

Moment of inertia of plate about its centerline: $I_{pl} := \frac{1}{12} \cdot b_{stif} t_w^3$

Determine centroid of section from centerline of panel:

Distance from C.L. of panel to backing plate C.L.: $y_1 := \frac{t_w}{2} + \frac{t_{stif}}{2}$

Distance from C.L. of panel to straight section of stiffener C.L.: $y_2 := \frac{t_w}{2} + \frac{[h_1 - (r + t_{stif})]}{2}$

Distance from C.L. of panel to centroid of corner radius:

$$y_3 := \frac{t_w}{2} + (h_1 - r) + r \cdot \left[\frac{2}{3 \cdot \frac{\pi}{2} \cdot \left(2 - \frac{t_{stif}}{r} \right)} + \left(1 - \frac{t_{stif}}{r} \right) \cdot \frac{4}{3 \cdot \pi} \right]$$

Distance from C.L. of panel to top plate C.L.: $y_4 := \frac{t_w}{2} + h_1 - \frac{t_{stif}}{2}$

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Distance of centroid from CL of panel:

$$d_{st} := \frac{\left[\left[(w_1 - 2 \cdot t_{stif}) \cdot t_{stif} y_1 \right] + \left[2 \cdot \left[\left[h_1 - (r + t_{stif}) \right] \cdot t_{stif} \right] \cdot y_2 \right] + \left[\frac{2 \cdot \pi \cdot \left[t_{stif}^2 + r^2 - r^2 \right]}{4} \cdot y_3 \right] + \left[(w_1 - 2 \cdot (t_{stif} + r)) \cdot t_{stif} y_4 \right] \right] \cdot 2}{A_{panel} + A_{stif}^2} \quad d_{st} = 1.25 \text{ in}$$

Outer fiber distances: $c_p := \frac{A_{stif} d_{st}}{A_{stif} + A_{panel}} + \frac{t_w}{2} \quad c_s := h_{stif} - c_p$

Moment of inertia of stiffener:

Moment of backing plate about CL: $I_{bpl} := \frac{(w_1 - 2 \cdot t_{stif}) \cdot t_{stif}^3}{12}$

Moment of side plates about CL: $I_{spl} := \frac{\left[h_1 - (t_{stif} + r) \right]^3 \cdot t_{stif}}{12}$

Moment of radius corners about CL: $I_{rad} := \frac{2 \cdot \pi \cdot \left[(r + t_{stif})^4 - r^4 \right]}{16}$

Moment of top plate about CL: $I_{tpl} := \frac{(w_1 - 2 \cdot r) \cdot t_{stif}^3}{12}$

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Moment of inertia of stiffener about CL of side plates:

$$I_{stifcl} = \left[2 \cdot I_{spl} + I_{bpl} + I_{rad} + I_{tpl} + \left[w_1 - 2 \cdot t_{stif} \right] \cdot t_{stif} \left[\frac{h_1 - (t_{stif} + r)}{2} - \frac{t_{stif}}{2} \right]^2 \right. \\ \left. + \frac{2 \cdot \pi \cdot \left[(t_{stif} + r)^2 - r^2 \right]}{4} \left[\frac{h_1 - (t_{stif} + r)}{2} + r \right] \left[\frac{2}{3 \cdot \frac{\pi}{2} \cdot 2 - \frac{t_{stif}}{r}} + 1 - \frac{t_{stif}}{r} \cdot \frac{4}{3 \cdot \pi} \right]^2 \right. \\ \left. + \left[w_1 - 2 \cdot (t_{stif} + r) \right] \cdot t_{stif} \cdot \left(\frac{h_1}{2} - \frac{t_w}{2} \right)^2 \right]$$

Moment of inertia of stiffener about centroid of stiffener and panel:

$$I_{stif} = 2 \cdot I_{stifcl} + 2 \cdot A_{stif} \left[\frac{h_1 - (t_{stif} + r)}{2} - \left(d_{st} - \frac{t_w}{2} \right) \right]^2$$

$$\text{Maximum moment: } M_{max} = \frac{p_{des} \cdot b_{stif} \cdot H_{pl}^2}{8} \quad M_{max} = 3910 \cdot \text{lbft}$$

$$\text{Stress on panel: } \sigma_{panel} = \frac{M_{max} \cdot c_p}{I_{stif}} \quad \sigma_{panel} = 3.2 \cdot \text{ksi}$$

$$\text{Stress on stiffener: } \sigma_{stif} = \frac{M_{max} \cdot c_s}{I_{stif}} \quad \sigma_{stif} = 13.9 \cdot \text{ksi}$$

$$\text{Maximum deflection: } \delta = \frac{5 \cdot p_{des} \cdot b_{stif} \cdot H_{pl}^4}{384 \cdot E_{sst} \cdot I_{stif}} \quad \delta = 0.039 \cdot \text{in}$$

Since maximum deflection is less than 40% of the thickness of the plate, small deflection theory applies to the panel evaluation.

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To determine the maximum stress at the junction of the flange to the panel plate at the closure, idealize the section of the panel between the stiffeners is a flat rectangular plate simply supported on three sides with the flange joint end considered as clamped and fixed.

Use the solutions outlined in Timoshenko 1959, page 240.

Maximum moments at edge corrected for Poisson's Ratio:

Aspect ratio of panel section: $\frac{H_{pl}}{b_{stif}} = 5.41$ Therefore treat as infinitely long plate in Table 52.

Moment at edge: $M_{ed} := 0.1250 \cdot \left(\frac{1.3}{1.2}\right) \cdot p_{des} \cdot (b_{stif} - 4 \cdot in)^2$ $M_{ed} = 61 \cdot lbf$

Stress at edge: $\sigma_b := 6 \cdot \frac{M_{ed}}{t_w^2}$ $\sigma_b = 1.5 \cdot ksi$

Margin of Safety: $MS_{ed} := \frac{1.5 \cdot s_{mc2}}{\sigma_b} - 1$ $MS_{ed} = 17.8$

Long Panel dimensions:

Wall thickness: $t_w := 0.5 \cdot in$ Height: $H_{pl} := 50.25 \cdot in$ Length: $L_{pl} := 101.5 \cdot in$

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Stiffener dimensions:

Height: $h_1 := 4.25 \cdot \text{in}$ Width: $w_1 := 2 \cdot \text{in}$ Wall thickness: $t_{\text{stif}} := 0.375 \cdot \text{in}$ Length: $l_{\text{stif}} := 50.37 \cdot \text{in}$ Distance between stiffeners: $b_{\text{stif}} := 2 \cdot 15.83 \cdot \text{in}$ Overall height of stiffener section: $h_{\text{stif}} := h_1 + t_w$ Inside bend radius: $r := 0.38 \cdot \text{in}$ Cross sectional area of panel between stiffeners: $A_{\text{panel}} := b_{\text{stif}} t_w$ $A_{\text{panel}} = 15.83 \cdot \text{in}^2$

Cross sectional area of stiffeners:

$$A_{\text{stif}} := 2 \cdot \left[\left[h_1 - (r + t_{\text{stif}}) \right] \cdot t_{\text{stif}} \right] + \left[\left[w_1 - 2 \cdot t_{\text{stif}} \right] \cdot t_{\text{stif}} \right] + \frac{2 \cdot \pi \cdot \left[(t_{\text{stif}} + r)^2 - r^2 \right]}{4}$$

Moment of inertia of plate about its centerline: $I_{\text{pl}} := \frac{1}{12} \cdot b_{\text{stif}} t_w^3$ **Determine centroid of section from centerline of panel:**Distance from CL of panel to backing plate CL: $y_1 := \frac{t_w}{2} + \frac{t_{\text{stif}}}{2}$ Distance from CL of panel to straight section of stiffener CL: $y_2 := \frac{t_w}{2} + \frac{[h_1 - (r + t_{\text{stif}})]}{2}$

Distance from CL of panel to centroid of corner radius :

$$y_3 := \frac{t_w}{2} + (h_1 - r) + r \cdot \left[\frac{2}{3 \cdot \frac{\pi}{2} \cdot \left(2 - \frac{t_{\text{stif}}}{r} \right)} + \left(1 - \frac{t_{\text{stif}}}{r} \right) \cdot \frac{4}{3 \cdot \pi} \right]$$

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Distance from CL of panel to top plate CL: $y_4 := \frac{t_w}{2} + h_1 - \frac{t_{stif}}{2}$

Distance of centroid from CL of panel:

$$d_{st} := \frac{\left[\left[\frac{w_1 - 2 \cdot t_{stif} \cdot t_{stif}}{4} \cdot y_1 \right] + \left[2 \cdot \left[h_1 - (r + t_{stif}) \right] \cdot t_{stif} \cdot y_2 \right] + \left[\frac{2 \cdot \pi \cdot \left[t_{stif}^2 + r^2 \right]}{4} \cdot y_3 \right] + \left[w_1 - 2 \cdot (t_{stif} + r) \right] \cdot t_{stif} \cdot y_4 \right]}{A_{panel} + A_{stif}} \quad d_{st} = 0.46 \text{ in}$$

Outer fiber distances: $c_p := \frac{A_{stif} d_{st}}{A_{stif} + A_{panel}} + \frac{t_w}{2} \quad c_s := h_{stif} - c_p$

Moment of inertia of stiffener:

Moment of backing plate about CL: $I_{bpl} := \frac{(w_1 - 2 \cdot t_{stif} \cdot t_{stif})^3}{12}$

Moment of side plates about CL: $I_{spl} := \frac{[h_1 - (t_{stif} + r)]^3 \cdot t_{stif}}{12}$

Moment of radius corners about CL: $I_{rad} := \frac{2 \cdot \pi \cdot \left[(r + t_{stif})^4 - r^4 \right]}{16}$

Moment of top plate about CL: $I_{tpl} := \frac{(w_1 - 2 \cdot r) \cdot t_{stif}^3}{12}$

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Moment of inertia of stiffener about CL of side plates:

$$I_{stifcl} = \left[2 \cdot I_{spl} + I_{bpl} + I_{rad} + I_{tpl} + \left[w_1 - 2 \cdot t_{stif} \cdot t_{stif} \right] \left[\frac{h_1 - (t_{stif} + r)}{2} - \frac{t_{stif}}{2} \right]^2 + \frac{2 \cdot \pi \cdot \left[(t_{stif} + r)^2 - r^2 \right]}{4} \left[\frac{h_1 - (t_{stif} + r)}{2} + r \cdot \left[\frac{2}{3 \cdot \frac{\pi}{2} \cdot \left(2 - \frac{t_{stif}}{r} \right)} + 1 - \frac{t_{stif}}{r} \cdot \frac{4}{3 \cdot \pi} \right] \right]^2 + \left[w_1 - 2 \cdot (t_{stif} + r) \right] \cdot t_{stif} \cdot \left(\frac{h_1}{2} - \frac{t_w}{2} \right)^2 \right]$$

Moment of inertia of stiffener about centroid of stiffener and panel:

$$I_{stif} = I_{stifcl} + A_{stif} \left[\frac{h_1 - (t_{stif} + r)}{2} - \left(d_{st} - \frac{t_w}{2} \right) \right]^2$$

$$\text{Maximum moment: } M_{max} := \frac{p_{des} \cdot b_{stif} \cdot H_{pl}^2}{8} \quad M_{max} = 13324 \cdot \text{lbft}$$

$$\text{Stress on panel: } \sigma_{panel} := \frac{M_{max} \cdot c_p}{I_{stif}} \quad \sigma_{panel} = 3.3 \cdot \text{ksi}$$

$$\text{Stress on stiffener: } \sigma_{stif} := \frac{M_{max} \cdot c_s}{I_{stif}} \quad \sigma_{stif} = 43.1 \cdot \text{ksi}$$

$$\text{Maximum deflection: } \delta := \frac{5 \cdot p_{des} \cdot b_{stif} \cdot H_{pl}^4}{384 \cdot E_{sst} \cdot I_{stif}} \quad \delta = 0.095 \cdot \text{in}$$

Since maximum deflection is less than 40% of the thickness of the plate, small deflection theory applies to the panel evaluation.

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To determine the maximum stress at the junction of the flange to the panel plate at the closure, idealize the section of the panel between the stiffeners is a flat rectangular plate simply supported on three sides with the flange joint end considered as clamped and fixed.

Use the solutions outlined in Timoshenko 1959, page 240.

Maximum moments at edge corrected for Poisson's Ratio:

$$\text{Aspect ratio of panel section: } \frac{H_{pl}}{b_{stif}} = 1.6$$

$$\text{Moment at edge: } M_{ed} := 0.0722 \cdot \frac{1.3}{1.2} \cdot p_{des} \cdot (b_{stif} - 2 \cdot \text{in})^2 \quad M_{ed} = 1101 \cdot \text{lb}\cdot\text{in}$$

$$\text{Stress at edge: } \sigma_b := 6 \cdot \frac{M_{ed}}{t_w^2} \quad \sigma_b = 26.4 \cdot \text{ksi}$$

$$\text{Margin of Safety: } MS_{ed} := \frac{1.5 \cdot s_{mc2}}{\sigma_b} - 1 \quad MS_{ed} = 0.04$$

Closure lid center:

Idealize the closure lid as a rectangular plate uniformly loaded and all edges simply supported under a pressure loading.

$$\text{Assume the plate thickness at the center: } t_{cpl} := 1.5 \cdot \text{in}$$

$$\text{Dimensions of the closure plate: } Wid := (67.3 - 2 \cdot (3.6)) \cdot \text{in} \quad Len := (107.3 - 2 \cdot (3.6)) \cdot \text{in}$$

Evaluate per method outlined in Roark 1989.

$$\text{Aspect ratio: } \frac{Len}{Wid} = 1.7 \quad \text{By Table 26: } \beta := \frac{0.5172 + 0.5688}{2} \quad \beta = 0.543$$

$$\alpha := \frac{0.0906 + 0.1017}{2} \quad \alpha = 0.096$$

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Deflection check: $\delta_{pl1} := \frac{\alpha \cdot p_{des} \cdot Wid^4}{E_{sst} \cdot t_{cpl}^3}$ $\delta_{pl1} = 0.22 \text{ in}$

Deflection less than 50% of plate thickness, therefore small deflection theory applies.

Stress at center: $\sigma_{cplb} := \frac{\beta \cdot p_{des} \cdot Wid^2}{t_{cpl}^2}$ $\sigma_{cplb} = 13.9 \text{ ksi}$

Closure Lid Margin of Safety: $MS_{cen} := \frac{1.5 \cdot s_{mc2}}{\sigma_{cplb}} - 1$ $MS_{cen} = 1$

Closure lid edge:

Idealize the closure lid as a rectangular plate uniformly loaded and all edges fixed under a pressure loading.

Assume the plate thickness at the center: $t_{cpl} := 1.125 \text{ in}$

Dimensions of the closure plate: $Wid := (67.3 - 2 \cdot (1.15)) \text{ in}$ $Len := (107.3 - 2 \cdot (1.15)) \text{ in}$

Evaluate per method outlined in Roark 1989.

Aspect ratio: $\frac{Len}{Wid} = 1.6$ By Table 26: $\beta_1 := \frac{0.4680 + 0.4872}{2}$ $\beta_1 = 0.478$

$\beta_2 := \frac{0.2286 + 0.2406}{2}$ $\beta_2 = 0.235$

$\alpha := \frac{0.0251 + 0.0267}{2}$ $\alpha = 0.026$

Deflection: $\delta_{max1} := \frac{\alpha \cdot p_{des} \cdot Wid^4}{E_{sst} \cdot t_{cpl}^3}$ $\delta_{max1} = 0.192 \text{ in}$

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Deflection less than 50% of plate thickness, therefore small deflection theory applies.

$$\text{Stress at center of long edge: } \sigma_{\text{elong}} := \frac{\beta_1 \cdot p_{\text{des}} \cdot \text{Wid}^2}{t_{\text{cpl}}^2} \quad \sigma_{\text{elong}} = 25.5 \text{ ksi}$$

$$\text{Margin of safety: } MS_{\text{elong}} := \frac{1.5 \cdot s_{\text{mc2}}}{\sigma_{\text{elong}}} - 1 \quad MS_{\text{elong}} = 0.1$$

Loading on closure bolts from pressure and preload:

To determine prying moment on the bolts from pressure, conservatively idealize the plate as a uniform fixed edge plate. Assume the maximum moment at the center of an edge is the uniformly distributed along the length of the plate and distributed uniformly to each closure bolt.

Bolts are ASTM A-325, Type 1.

$$\text{Nominal bolt diameter: } d_b := 0.75 \text{ in} \quad \text{Pitch: } \text{pit} := \frac{10}{\text{in}}$$

$$\text{Total number of bolts: } N_{\text{tot}} := 32$$

$$\text{Yield strength: } s_{yb} := 83.7 \text{ ksi} \quad \text{Ultimate strength: } s_{ub} := 116.4 \text{ ksi}$$

$$\text{NE allowable: } s_{bc} := 1.1 \cdot 23 \text{ ksi} \quad s_{bc} = 25.3 \text{ ksi}$$

$$\text{Elastic modulus: } E_b := 28.1 \cdot 10^6 \text{ psi}$$

Evaluate bolt loading as outlined in Bickford 1990.

$$\text{Pressure area of lid: } A_{\text{pres}} := \text{Len} \cdot \text{Wid} \quad \text{Total circumference of lid: } \text{cir} := 2 \cdot \text{Wid} + 2 \cdot \text{Len}$$

$$\text{Total force on lid: } F_{\text{tten}} := p_{\text{des}} \cdot A_{\text{pres}}$$

$$\text{Unit circumferential loading: } F_{\text{tc}} := \frac{F_{\text{tten}}}{\text{cir}}$$

$$\text{Number of bolts long side (conservatively ignore end bolts): } N_{\text{lb}} := 8$$

$$\text{Number of bolts short side (conservatively ignore end bolts): } N_{\text{sb}} := 6$$

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Worst case span between bolts on long side: $t_{cl} := 12.22 \text{ in}$

Distance from outside edge to inside edge of flange: $s_e := 3.60 \text{ in}$

Worst case eccentric loading on long side: $F_{eten} := \frac{F_{tc} \cdot Len}{N_{lb}} \quad F_{eten} = 4215 \text{ lbf}$

Tensile load per bolt: $F_{ten} := \frac{F_{tten}}{N_{tot}} \quad F_{ten} = 3413 \text{ lbf}$

Preload on bolts:

Maximum preload torque: $T_{pre} := 180 \text{ lbf ft}$ Conservatively assume a nut factor of: $k_n := 0.2$

Preload force on bolt: $F_{pre} := \frac{T_{pre}}{d_b \cdot k_n} \quad F_{pre} = 14400 \text{ lbf}$

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Tensile stress area of bolt: $A_t := 0.785 \cdot \left(d_b - \frac{0.9743}{\text{pitch}} \right)^2$

Eccentric load factor:

Minimum thickness: $t_{\min} := t_{\text{cpl}}$ Flange thickness: $t_f := 1.5 \cdot \text{in}$

Diameter of contact between bolt head and closure lid (min): $D_b := 1.25 \cdot \text{in}$

Diameter of bolt hole: $D_h := 0.94 \cdot \text{in}$

$t_{\min} + D_b = 2.375 \cdot \text{in}$ is less than s_c and less than t_{cl} , therefore:

$b_{\text{sec}} := D_b + t_{\min}$ and $A_j := b_{\text{sec}} \cdot (D_b + t_{\min})$ and $D_j := \text{Wid}$

Since D_b is less than D_j and D_j is less than $3D_b$ and head diameter is the same as the contact diameter, then:

$$A_c := \frac{\pi}{4} \cdot (D_b^2 - D_h^2) + \frac{\pi}{8} \cdot \left(\frac{D_j}{D_b} - 1 \right) \cdot \left[\frac{D_b \cdot (t_{\text{cpl}} + t_f)}{5} + \frac{(t_{\text{cpl}} + t_f)^2}{100} \right]$$

Joint stiffness: $K_{jc} := \frac{E_{\text{sst}} \cdot A_c}{t_{\text{cpl}} + t_f}$

Joint resilience:

Distance from centerline of bolt to edge of contact: $s_{ie} := \frac{s_c}{2} - \frac{1.15 \cdot \text{in}}{2} - \frac{D_b}{2}$

Distance from load to back of bolt head: $a_x := \frac{3.6 \cdot \text{in}}{2} + s_{ie} + \frac{D_b}{2}$ $a_x = 3.02 \cdot \text{in}$

Radius of gyration for a rectangular section: $R_g := 0.209 \cdot \text{Wid}$

Factor: $\lambda := \frac{s_{ie} \cdot A_c}{R_g \cdot A_j}$

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Concentric resilience:
$$r_j := \frac{1}{K_{jc}} \cdot \left(1 + \frac{s_{ic}^2 \cdot A_c}{R_g^2 \cdot A_j} \right)$$

Eccentric resiliences:
$$r_{j1} := r_j \cdot (1 + \lambda^2) \quad r_{j2} := r_j \cdot \left(1 + \frac{a_x}{s_{ic}} \cdot \lambda^2 \right)$$

Elastic modulus of bolt at temperature: $E_b := 28.1 \cdot 10^6 \text{ psi}$

Resilience of the bolt:

Length of bolt: $L_{bolt} := 3.25 \text{ in}$ Length of thread: $L_t := 1.75 \text{ in}$

Length of the shank: $L_b := L_{bolt} - L_t$

Length of bolt between nut and head: $L_g := t_{min} + t_f$

Thickness of nut: $T_{nut} := \frac{47}{64} \text{ in}$ Thickness of head: $T_{head} := \frac{1}{2} \text{ in}$

Nominal diameter area of bolt: $A_{nd} := \frac{\pi \cdot d_b^2}{4}$

Resilience of bolt:
$$r_b := \frac{L_b - \frac{T_{head}}{2}}{E_b \cdot A_{nd}} + \frac{L_g - L_b + \frac{T_{nut}}{2}}{E_b \cdot A_t}$$

Eccentricity factor:
$$\phi := \frac{r_{j2}}{r_{j1} + r_b} \quad \phi = 0.026$$

Prying loading on bolts: $F_{pry} := \phi \cdot F_{eten} \quad F_{pry} = 108 \text{ lbf}$

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Tensile load on bolts: $\sigma_{tenb} := \frac{F_{ten}}{A_t} \quad \sigma_{tenb} = 10.2 \text{ ksi}$

Bending load on bolts: $\sigma_{benb} := \frac{F_{pry}}{A_t} \quad \sigma_{benb} = 0.3 \text{ ksi}$

Residual stress from preload (assume only half of preload torque is effective):

$$\sigma_{resid} := \frac{1}{2} \cdot \frac{F_{pre}}{A_t} \quad \sigma_{resid} = 21.5 \text{ ksi}$$

Combined stress: $\sigma_{comb} := \sqrt{(\sigma_{tenb} + \sigma_{benb})^2 + \sigma_{resid}^2} \quad \sigma_{comb} = 24 \text{ ksi}$

Margins of Safety:

Average tension stress: $MS_{b1} := \frac{s_{bc}}{\sigma_{tenb}} - 1 \quad MS_{b1} = 1.5$

Tension plus bending plus residual torsion: $MS_{b2} := \frac{1.5 \cdot s_{bc}}{\sigma_{comb}} - 1 \quad MS_{b2} = 0.6$

7.9.3 Outer Shell Pin Puncture

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1.0 OBJECTIVE

The objective of this analysis is to determine the puncture resistance performance of the 22-Ton SWDB overpack when subjected to the test parameters of 10 CFR 71.73.

2.0 REFERENCES

10 CFR 71.73, "Hypothetical accident conditions," *Code of Federal Regulations*, as amended.

ORNL, 1970, *Cask Designers Guide: A Guide for the Design, Fabrication, and Operation of Shipping Casks for Nuclear Applications*, ORNL-NSIC-68, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Roark, R. J., and W. C. Young, 1975, *Formulas for Stress and Strain*, Fifth Edition, McGraw-Hill Book Company, New York, New York.

3.0 ASSUMPTIONS, RESULTS, AND CONCLUSIONS

In this evaluation, the following parameters are assumed.

- The SWDB assembly is dropped from 40 in. onto a 6 in. diameter carbon steel pin that is sufficiently long to penetrate the impact limiter and impact the overpack (Figure 1).
- The entire assembly (including impact limiter) impacts the puncture pin.
- The puncture pin is manufactured from A-36 carbon steel with a flow stress that is the average of the yield strength and ultimate strength (ORNL 1970).
- The worst case impact is on the bottom plate without the shield plate, since this is the area of the package with the least support and is the thinnest section.
- The 4 in. thick SWDB provides support to the overpack bottom walls during impact.

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- The dynamic flow stress is used in lieu of the yield strength since the loading from impacts must be considered as dynamic loading. During dynamic loading, the stainless steel plastically flows outward from the contact point instead of buckling.

Results of this evaluation show that there would be significant localized deformation of the overpack bottom section, as shown by the large elastically computed stresses. However, this damage is localized and the bottom plate does not collapse or puncture. This is demonstrated by the 0.1 margin of safety based on the collapsing strength of the overpack bottom section with the backing of the SWDB.

Figure 1.

BOX PERPENDICULAR DROP TO CYLINDER

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4.0 EVALUATION

22 Ton Package Puncture Evaluation :

Evaluate the puncture performance of the 22 ton SWDB Overpack in accordance with the performance test defined in 10 CFR 71.73 without the impact limiter and for a bottom end drop onto the pin. In a test, the Overpack wall is being backed up by the SWDB.

Thickness of the Overpack and SWDB: $t := 4.5 \cdot \text{in}$

Assumed Poissons ratio of stainless steel: $\nu := 0.3$

Bottom plate inside dimensions: $a := 100.5 \cdot \text{in}$ $b := 60.5 \cdot \text{in}$

Radius of puncture pin: $r_o := 3 \cdot \text{in}$

Elastic modulus of 304 SS (SA-240) at 300 °F: $E := 27 \cdot 10^6 \cdot \text{psi}$

Yield strength of 304 SS at 300 °F: $\sigma_y := 22.5 \cdot \text{ksi}$

Ultimate strength of 304 SS at 300 °F: $\sigma_u := 66 \cdot \text{ksi}$

Dynamic flow stress: $\sigma_f := \frac{\sigma_y + \sigma_u}{2}$ $\sigma_f = 44.25 \cdot \text{ksi}$

Weight of package with impact limiter: $w := 82000 \cdot \text{lb}$

When the Overpack falls on the steel pin, the pin supports the inertial weight of the Overpack. The force exerted by the pin is its dynamic stress times its area. Using an assumed dynamic flow stress of 47 ksi (taken as the average of yield strength [36 ksi] and ultimate strength [58 ksi] of a mild steel [A36]), the force is:

Dynamic flow stress: $\text{DFS} := 47 \cdot \text{ksi}$

Yield strength: $\text{AYS} := 36 \cdot \text{ksi}$

Ultimate strength: $\text{US} := 58 \cdot \text{ksi}$

Force applied by pin: $F := \text{DFS} \cdot (\pi \cdot r_o^2)$ $F = 1329 \cdot \text{kip}$

The deceleration of the Overpack is: $g_{op} := \frac{F}{w}$ $g_{op} = 16.2$

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Idealize as impact on a fixed edge rectangular plate with load point from pin at center. Use method outlined in Roark 1975, page 392:

Roark plate factors:

Aspect ratio: $\frac{a}{b} = 1.7$ Plate factors: $\beta_1 := 0.061$ $\beta_2 := 0.995$ $\alpha := 0.078$

Equivalent contact radius:

Since $r_o > 0.5 t$ then: $r_o := r_o$

Impact - fixed-edge rectangle with load on 6 in. diameter centered circle.

Maximum bending stress at center: $\sigma_b := \frac{3 \cdot F}{2 \cdot \pi \cdot t^2} \left[(1 + \nu) \cdot \ln \frac{2 \cdot b}{\pi \cdot r_o} + \beta_1 \right]$ $\sigma_b = 106 \text{ ksi}$

Maximum deflection at center: $y := \frac{\alpha \cdot F \cdot b^2}{E \cdot t^3}$ $y = 0.1542 \text{ in}$

The maximum bending stress is beyond yield, therefore plastic flow occurs on the overpack. Consequently determine load to collapse bottom plate.

Determination of collapse of bottom plate:

Bottom plate - worst case.

Load required to collapse or puncture plate is based on dynamic flow stress of material. Any shape, any size, concentrated load at any point (Roark and Young 1975, page 410). Since this is a dynamic loading case, the dynamic flow stress is substituted for the yield strength in the equation.

Collapsing load: $W_{ut} := \sigma_f \left(\frac{1}{2} \cdot \pi \cdot t^2 \right)$ $W_{ut} = 1408 \text{ kip}$

Margin of safety against puncture:

$$MS := \frac{W_{ut}}{F} - 1 \quad MS = 0.1$$

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8.0 THERMAL EVALUATION

8.1 INTRODUCTION

The SWP is designed to transport Type B, HRCQ of fissile remote-handled waste and special-case waste from the 324 Building to the 200 Area for interim storage. The Central Waste Complex is the primary destination; however, this SARP authorizes transportation to any facility within the Hanford Site. The contents of the SWP consists primarily of contaminated scrap equipment removed from B Cell of the 324 Building: racks, tanks, pipes, hoses, valves, tank and pipe heel, filters, lead plugs from cell pipe penetrations, and dispersible material. The SWP consists of an RGC, RODC, 22-Ton SWDB, SWDB Overpack, and 22-Ton SWDB Impact Limiter. An alternative configuration includes ECs with dispersible waste placed in the RGC (see Part A, Section 1.2). The evaluated configuration is bounding for all authorized package configurations.

This section presents the thermal evaluation of the safety performance for onsite transportation. Evaluation parameters include nationally recognized standards, regulations, and criteria for radioactive packaging. The results of these analyses performed demonstrate the 22-Ton SWP is safe for onsite transport of the intended payload. The purpose of this thermal evaluation is to determine the temperature performance of the 22-Ton SWP under the NCT and the fire accident condition with a content heat load of 1410 W. The ambient temperature is assumed to be the maximum Hanford Site temperature of 46 °C (115 °F). The package is analyzed with and without the solar insolation for the NCT and without the solar insolation for the fire accident condition. The approximate dimensions of the transportation package are 320.68 cm (126.25 in.) by 219.10 cm (86.25 in.) by a height of 203 cm (80 in.). For this evaluation the package is assumed in its normal transport position.

8.1.1 Package Description

The 22-Ton SWP is evaluated based on the configuration shown in drawings H-3-56430 (PNL 1988a), H-3-56431 (PNL 1996c), H-3-56433 (PNL 1988b), and E-D-003, 22 Ton SWDB Overpack Assembly (see Part A, Section 10.0 for drawing E-D-003). The SWDB is constructed with thick plates made of A-36 steel that are bolted together to confine disposable radioactive wastes for shipping to a disposal site. The SWDB Overpack is constructed with Type 304 stainless steel and is the containment boundary. The SWDB Overpack assembly and a 7.62 cm (3-in.-) thick shield plate (A-36 steel) are enclosed by the SWDB Impact Limiter, which is constructed of two halves. The SWDB Impact Limiter consists of a steel liner with Douglas fir wood impact material. The outside of the SWDB Impact Limiter liner has a steel tube framing that is used for structural protection and for attachments of locking devices and lifting mechanisms. This outer frame is not considered in this thermal evaluation.

A thermal analysis model has been developed on ANSYS Finite Element Computer Program, Version 5.5.3 (SAS 1999), for this evaluation. In this evaluation, the SWDB, waste

contents, SWDB Overpack containment boundary with its stiffeners, and the SWDB Impact Limiter main structures are modeled. A quarter-section model is developed to simplify the thermal model.

8.2 THERMAL SOURCE PARAMETERS

The bounding decay heat load of the 22-Ton SWDB radioactive waste contents is assumed as 1410 W. The volume of the waste contents is assumed to be the inside volume of the SWDB cavity; i.e., 198 cm (78 in.) long by 107 cm (42 in.) wide by 94 cm (37 in.) high. Therefore, the waste volume is 1.99 m^3 ($121,212 \text{ in}^3$), and the volumetric heat deposition density is $0.00341 \text{ Btu/h-in}^3$.

For conservatism the solar insolation is assumed as that specified in Boehnke (1999). The total insolation over a 12-hour period is 800 g-cal/cm^3 on the exposed flat surfaces transported horizontally, 200 g-cal/cm^3 on the flat surfaces not transported horizontally, and 400 g-cal/cm^3 on curved surfaces. This is equivalent to a heat flux of $1.7068 \text{ Btu/h-in}^2$ for the 800 g-cal/cm^3 value. Over a 12-hour period and assuming the SWP is loaded on top of the transporter, the top surface of the package receives 800 g-cal/cm^3 , the four side surfaces receive 200 g-cal/cm^3 , and the bottom surface does not receive any solar loading.

The thermal radiation and thermal convection on the outside surfaces of the SWDB Impact Limiter are as follows. The top and four side surfaces are assumed to be surrounded by a 46°C (115°F) environment, while the bottom is assumed to be surrounded by a 38°C (100°F) environment. During the fire accident, the entire exterior of the package is assumed to be engulfed by an 800°C (1475°F) temperature fire.

8.3 SUMMARY OF THERMAL PROPERTIES OF MATERIALS

8.3.1 SWDB Overpack Containment Boundary

The SWDB Overpack containment boundary and stiffeners are constructed of Type 304 stainless steel. Its thermal material properties are obtained from WHC-SD-TP-RPT-005, *Thermal Analysis Methods for Safety Analysis Reports for Packaging* (Irwin 1996), Table A-12a, and are shown in Table B8-1.

Table B8-1. Material Properties of Type 304 Stainless Steel.

Temperature (°F)	Thermal conductivity (Btu/h-in-°F)	Specific heat (Btu/lbm-°F)	Density (lbm/in ³)
32	0.644	0.120	0.286
70	0.713	0.109	0.286
100	0.727	0.111	0.286
200	0.774	0.116	0.286
400	0.860	0.125	0.286
600	0.943	0.130	0.286
800	1.022	0.135	0.286
1200	1.170	0.140	0.286
1400	1.238	0.145	0.286
1500	1.274	0.149	0.286

Source: Irwin, J. J., 1996, *Thermal Analysis Methods for Safety Analysis Reports for Packaging*, WHC-SD-TP-RPT-005, Rev. 0, Table A-12a, Westinghouse Hanford Company, Richland, Washington.

8.3.2 SWDB, Shield Plate, and Impact Limiter Liners

The SWDB and Impact Limiter liners are constructed of ASTM A-36 steel. Its thermal material properties are based Irwin (1996), Table A-18, as shown in Table B8-2.

Table B8-2. Material Properties of A-36 Carbon Steel.

Temperature (°F)	Thermal conductivity (Btu/h-in-°F)	Specific heat (Btu/lbm-°F)	Density (lbm/in ³)
-40	1.917	0.096	0.283
70	1.970	0.106	0.283
200	2.033	0.118	0.283
300	2.033	0.123	0.283
400	2.018	0.128	0.283
500	1.975	0.133	0.283
600	1.927	0.137	0.283
700	1.869	0.143	0.283
800	1.811	0.149	0.283
900	1.744	0.156	0.283
1000	1.667	0.165	0.283
1100	1.559	0.172	0.283
1200	1.517	0.189	0.283
1300	1.392	0.220	0.283

Source: Irwin, J. J., 1996, *Thermal Analysis Methods for Safety Analysis Reports for Packaging*, WHC-SD-TP-RPT-005, Rev. 0, Table A-18, Westinghouse Hanford Company, Richland, Washington.

8.3.3 Wood

The wood inserts used in the SWDB Impact Limiter are assumed to have similar thermal properties as the neutralized current acid waste end spacer in Irwin (1996), Table 33. These are shown in Table B8-3.

Table B8-3. Material Properties of Wood.

Temperature (°F)	Thermal conductivity (Btu/h-in-°F)	Specific heat (Btu/lbm-°F)	Density (lbm/in ³)
0	0.007056	0.570	0.022
450	0.007056	0.570	0.022

Source: Irwin, J. J., 1996, *Thermal Analysis Methods for Safety Analysis Reports for Packaging*, WHC-SD-TP-RPT-005, Rev. 0, Table 33, Westinghouse Hanford Company, Richland, Washington.

8.3.4 Air

The SWDB is assumed to be placed on top of the overpack containment, and the Overpack containment is assumed to be placed on top of the shield plate. The shield plate is placed directly on the top inside surface of the lower impact limiter. No air gaps exist. Air gaps exist between other surfaces of the SWDB and the overpack containment and between other surfaces of the overpack containment and the impact limiter. The thermal physical properties of air are taken from Irwin (1996), Table B-5. They are shown in Table B8-4.

Table B8-4. Material Properties of Air.

Temperature (°F)	Thermal conductivity (Btu/h-in-°F)	Specific heat (Btu/lbm-°F)	Density (lbm/in ³)
0	1.11E-3	0.239	4.98E-5
32	1.17E-3	0.240	4.69E-5
100	1.28E-3	0.240	4.11E-5
200	1.45E-3	0.241	3.47E-5
300	1.61E-3	0.243	3.01E-5
400	1.77E-3	0.245	2.66E-5
500	1.93E-3	0.247	2.38E-5
600	2.08E-3	0.250	2.16E-5
700	2.23E-3	0.253	1.97E-5
800	2.38E-3	0.256	1.82E-5
900	2.53E-3	0.259	1.68E-5
1000	2.66E-3	0.262	1.57E-5
1500	3.33E-3	0.276	1.17E-5

Source: Irwin, J. J., 1996, *Thermal Analysis Methods for Safety Analysis Reports for Packaging*, WHC-SD-TP-RPT-005, Rev. 0, Table B-5, Westinghouse Hanford Company, Richland, Washington.

8.3.5 Waste Contents

The waste contents are conservatively assumed to have the same thermal material properties as the Disposable Solid Waste Cask payload in Irwin (1996), Table 45. The thermal properties of waste contents are shown in Table B8-5.

Table B8-5. Material Properties of Waste Contents.

Temperature (°F)	Thermal conductivity (Btu/h-in-°F)	Specific heat (Btu/lbm-°F)	Density (lbm/in ³)
32	0.645	0.12	0.286
212	0.786	0.122	0.286
932	1.048	0.13	0.286

Source: Irwin, J. J., 1996, *Thermal Analysis Methods for Safety Analysis Reports for Packaging*, WHC-SD-TP-RPT-005, Rev. 0, Table B-5, Westinghouse Hanford Company, Richland, Washington.

8.4 THERMAL EVALUATION FOR NCT

8.4.1 Conditions To Be Evaluated

The NCT evaluation is designed to measure thermal performance of the 22-Ton SWP under the high and low extremes and the Hanford climate. Under NCT an ambient temperature of 46 °C (115 °F) without solar insolation and with solar insolation is evaluated. An ambient temperature of -33 °C (-27 °F) is evaluated in Part B, Section 7.2.9.

8.4.2 Acceptance Criteria

Acceptance criteria for NCT are based on Boehnke (1999). The 22-Ton SWP must not show any appreciable degradation of shielding, containment, or structural properties. The acceptance criteria for the NCT assumed that the outer surface of the SWDB Impact Limiter must be less than 85 °C (185 °F) in the shade.

8.4.3 Thermal Model for NCT

ANSYS general finite element computer program (SAS 1998, SAS 1999) is used to develop the thermal model for both normal and accident conditions. A quarter-section model is used for this analysis. The general view of the entire model is shown in Figure B8-1. The SWDB and contents, the SWDB Overpack containment with stiffeners, and the SWDB Impact Limiter model are shown in Figures B8-2 through B8-4, respectively.

Figure B8-1. General View of 22-Ton Steel Waste Disposal Box
Thermal Finite Element Model.

Figure B8-2. Finite Element Model of Steel Waste
Disposal Box and Waste Contents.

Figure B8-3. Finite Element Model of Steel Waste Disposal Box Overpack Boundary With Stiffeners.

Figure B8-4. Finite Element Model of Steel Waste Disposal Box Impact Limiter.

ANSYS element SHELL57, three-dimensional thermal shell is used for the overpack containment shells and SWDB Impact Limiter inside and outside liners. ANSYS SOLID70, three-dimensional solid element is used for the SWDB, waste contents, SWDB Shield Plate, and SWDB Impact Limiter wood inserts. Air is also modeled with SOLID70 elements. The thermal radiation and thermal convection of the SWDB Impact Limiter outside surface to environment is modeled through surface element SURF152. The SURF152 surface effect element is a shell element connecting four nodes. It has an optional extra node, which can be used for heat transfer purposes. The surface element can be used for heat transfer of thermal convection and thermal radiation. It can also be used to account for the heat flux through these surfaces. The thermal radiation in the air gaps around the SWDB Overpack containment is modeled using the radiation matrices MATRIX50.

The emissivity factors for carbon steel and Type 304 stainless steel are 0.9 and 0.6, respectively. On the outer surface of the SWDB Overpack exposed to the solar insolation, an absorption factor of 0.80 is used (Irwin [1996], Table A-51). The natural convection heat transfer coefficient equation of the external surface of SWDB Overpack to the environment is taken from Irwin (1996), Appendix B. The temperature-dependent convection heat transfer coefficients are shown in the tables below. Tables B8-6 through B8-8 are the heat transfer coefficients of NCT.

Table B8-6. Convection Heat Transfer Coefficient
of Impact Limiter Vertical Side Planes:
Normal Conditions of Transport.

Temperature difference (°F)	Convection heat transfer coefficient (Btu/h-in ² -°F)
100	0.00396
200	0.00471
300	0.00522
400	0.00560
500	0.00593
600	0.00620
700	0.00645
800	0.00666
900	0.00686
1000	0.00705

Source: Irwin, J. J., 1996, *Thermal Analysis Methods for Safety Analysis Reports for Packaging*, WHC-SD-TP-RPT-005, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

Table B8-7. Convection Heat Transfer Coefficient
of Impact Limiter Top Surface: Normal
Conditions of Transport.

Temperature difference (°F)	Convection heat transfer coefficient (Btu/hr-in ² -°F)
100	0.00344
200	0.00409
300	0.00452
400	0.00486
500	0.00514
600	0.00538
700	0.00559
800	0.00578
900	0.00595
1000	0.00611

Source: Irwin, J. J., 1996, *Thermal Analysis Methods for Safety Analysis Reports for Packaging*, WHC-SD-TP-RPT-005, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

Table B8-8. Convection Heat Transfer Coefficient
of Impact Limiter Bottom Surface: Normal
Conditions of Transport.

Temperature difference (°F)	Convection heat transfer coefficient (Btu/h-in ² -°F)
100	0.0015
200	0.0018
300	0.0020
400	0.0022
500	0.0023
600	0.0024
700	0.0025
800	0.0026
900	0.0026
1000	0.0027

Source: Irwin, J. J., 1996, *Thermal Analysis Methods for Safety Analysis Reports for Packaging*, WHC-SD-TP-RPT-005, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

8.4.4 Thermal Analysis Results for NCT

The results for NCT without solar insolation are shown in Figures B8-5 through B8-8 for the seal/flange, SWDB Overpack, SWDB, and SWDB Impact Limiter outer liner, respectively. The results for NCT with solar insolation are shown in Figures B8-9 through B8-12 for seal/flange, containment, SWDB, and SWDB Impact Limiter outer shell, respectively. The maximum temperatures are shown in Table B8-9.

Figure B8-5. Temperature Distribution in Flange for Normal Conditions of Transport Without Solar Insolation.

Figure B8-6. Temperature Distribution in Steel Waste Disposal Box Overpack Containment Boundary for Normal Conditions of Transport Without Solar Insolation.

Figure B8-7. Temperature Distribution in Steel Waste Disposal Box for Normal Conditions of Transport Without Solar Insolation.

Figure B8-8. Temperature Distribution in Impact Limiter Outer Liner for Normal Conditions of Transport Without Solar Insolation.

Figure B8-9. Temperature Distribution in Steel Waste Disposal Box Overpack Flange for Normal Conditions of Transport With Solar Insolation.

Figure B8-10. Temperature Distribution in Steel Waste Disposal Box
Overpack Containment Boundary for Normal Conditions
of Transport With Solar Insolation.

Figure B8-11. Temperature Distribution in Steel Waste Disposal Box for
Normal Conditions of Transport With Solar Insolation.

Figure B8-12. Temperature Distribution in Impact Limiter Outer Liner for Normal Conditions of Transport With Solar Insolation.

Table B8-9. Normal Condition Thermal Analysis Results Maximum Temperature.

	No solar insolation temperature °C (°F)	Solar insolation temperature °C (°F)
Seal/Flange of SWDB Overpack containment	121 (249)	146 (294)
SWDB Overpack boundary	129 (264)	152 (306)
SWDB and contents	132 (270)	156 (312)
External surface of SWDB Impact Limiter	50 (122)	102 (215)

SWDB = Steel Waste Disposal Box.

8.5 THERMAL EVALUATION FOR HYPOTHETICAL FIRE ACCIDENT CONDITION

8.5.1 Conditions To Be Evaluated

As a bounding case prior to the fire accident, the 22-Ton SWP is assumed to be at a steady-state temperature of 46 °C (115 °F). The fire temperature is 800°C (1475 °F) for a duration of 30 minutes with an effective flame emissivity of 0.9. The emissivity of 0.9 for the

outer carbon steel surfaces of the SWDB Impact Limiter is used. For conservative consideration, the effective emissivity is assumed to be 0.9, and the absorption is assumed to be 1.

8.5.2 Acceptance Criteria

For onsite transport, the 22-Ton SWP must maintain adequate containment and shielding capabilities during and after a fire-based accident event. The package must have no appreciable degradation of shielding, containment, or structural properties.

8.5.3 Package Conditions and Environment

For onsite transport, the accident-conditions fire is assumed as an independent event. Consequently, the package is assumed to be fully engulfed within an accident-conditions fire in the undamaged condition. Prior to the fire, the package is assumed to be at steady state with an exterior temperature of 50 °C (122 °F) ambient condition without solar insolation (Boehnke 1999). The fire is assumed to be at a temperature of 800 °C (1475 °F), have an effective emissivity of 0.9, and have a duration of 30 minutes minimum (Boehnke 1999). Cooldown is assumed to have a duration in excess of 3 hours with ambient conditions at 46 °C (115 °F).

8.5.4 Thermal Model for Accident Conditions

The thermal model in the fire accident condition is the same as that of the NCT model except for some minor changes. The thermal convection heat transfer coefficients are modified to account for the higher airflow around the package. Also, the transient thermal load steps are added.

The temperature-dependent convection heat transfer coefficients are shown in the tables below. Tables B8-10 through B10-12 are the heat transfer coefficients of the HAC (fire).

Table B8-10. Convection Heat Transfer Coefficient of Impact Limiter
Vertical Side Planes: Fire Accident Condition.

Temperature difference (°F)	Convection heat transfer coefficient (Btu/h-in ² -°F)	Temperature difference (°F)	Convection heat transfer coefficient (Btu/h-in ² -°F)
100	0.00603	900	0.01245
200	0.00758	1000	0.01289
300	0.00867	1100	0.01331
400	0.00953	1200	0.01369
500	0.01026	1300	0.01406
600	0.01089	1400	0.01441
700	0.01146	1500	0.01474
800	0.01198		

Table B8-11. Convection Heat Transfer Coefficient of Impact Limiter
Top Surface: Fire Accident Condition.

Temperature difference (°F)	Convection heat transfer coefficient (Btu/h-in ² -°F)	Temperature difference (°F)	Convection heat transfer coefficient (Btu/h-in ² -°F)
100	0.00571	900	0.01180
200	0.00718	1000	0.01222
300	0.00821	1100	0.01261
400	0.00903	1200	0.01297
500	0.00972	1300	0.01332
600	0.01032	1400	0.01365
700	0.01086	1500	0.01396
800	0.01135		

Table B8-12. Convection Heat Transfer Coefficient of Impact Limiter
Bottom Surface: Fire Accident Condition.

Temperature Difference (°F)	Convection heat transfer coefficient (Btu/h-in ² -°F)	Temperature Difference (°F)	Convection heat transfer coefficient (Btu/h-in ² -°F)
100	0.00698	900	0.01442
200	0.00878	1000	0.01493
300	0.01003	1100	0.01541
400	0.01103	1200	0.01586
500	0.01188	1300	0.01628
600	0.01261	1400	0.01668
700	0.01327	1500	0.01707
800	0.01387		

8.5.5 Maximum Temperatures

Table B8-13 shows the maximum temperature conditions at selected points during and after the accident-conditions fire. During a fire, the maximum temperature of the SWDB Overpack containment boundary reaches 131 °C (267 °F); the SWDB and contents reaches 134 °C (273 °F); and the outer surface of the impact limiter reaches 800 °C (1472 °F). The maximum temperature in the SWDB Overpack seal/flange region is 123 °C (253 °F).

8.5.6 Thermal Analysis Results for Accident Condition

The temperature histories of the fire accident at the payload, containment bottom, containment lid, and containment seal/flange locations are shown in Table B8-13 and Figure B8-13. The SWDB Impact Limiter outer surface is shown in Figure B8-14.

Table B8-13. Fire Accident Thermal Analysis Results.

Case	Fire accident °C (°F)
Seal/Flange of SWDB Overpack containment	123 (253)
SWDB Overpack boundary	131 (267)
SWDB and contents	134 (273)
Outer surface of SWDB Impact Limiter	800 (1472)

SWDB = Steel Waste Disposal Box

Figure B8-13. Maximum Temperature History at the Payload, Containment Bottom, Lid, and Seal/Flange of 22-Ton Steel Waste Package.

Figure B8-14. Maximum Temperature History at the Steel Waste Disposal Box Impact Limiter Outer Surface.

8.6 THERMAL EVALUATION AND CONCLUSIONS

In the NCT under no solar insolation, the outer surface of SWDB Impact Limiter temperature reaches 50 °C (122 °F), which is less than the regulatory limit for exclusive-use shipments of 85 °C (185 °F) in the shade. For the structural evaluation the exterior surface of the package reaches a temperature of 102 °C (215 °F) under solar insolation. Internally the SWDB Overpack seal/flange temperature under solar insolation reaches up to 146 °C (294 °F). This temperature is well below the manufacturer's recommended continuous operating temperature limit of 204 °C (400 °F) for Viton. Consequently, containment will be maintained during NCT. The maximum temperature of the SWDB Overpack containment boundary under solar insolation is 151 °C (304 °F).

During the accident-conditions fire, the temperature of the SWDB Overpack seal/flange area reaches 123 °C (253 °F). The maximum temperature of the SWDB Overpack containment boundary during an accident-conditions fire is 131 °C (267 °F). This temperature is well below the manufacturer's recommended continuous operating temperature limit of 204 °C (400 °F) for Viton. Consequently, during an accident-conditions fire, since the seal remains intact and the SWDB Overpack containment boundary is manufactured from stainless steel, containment will be maintained.

8.7 REFERENCES

- Boehnke, W. M., 1999, *Packaging Design Criteria for the Steel Waste Package*, HNF-3723, Rev. 0, prepared by Waste Management Federal Services, Inc., Northwest Operations for Fluor Daniel Hanford, Inc., Richland, Washington.
- Irwin, J. J., 1996, *Thermal Analysis Methods for Safety Analysis Reports for Packaging*, WHC-SD-TP-RPT-005, Rev. 0, Westinghouse Hanford Company, Richland, Washington.
- PNL, 1996c, *22 Ton Steel Waste Disposal Box Assembly/Details*, drawing H-3-56431, Rev. 1 (4 sheets), Pacific Northwest Laboratory, Richland, Washington.
 Sheets 1 and 2, *22 Ton Steel Waste Disposal Box Assembly*
 Sheets 3 and 4, *22 Ton Steel Waste Disposal Box Details*
- PNL, 1988a, *22 Ton Steel Waste Disposal Box Arrangement*, drawing H-3-56430, Rev. 0, Pacific Northwest Laboratory, Richland, Washington.
- PNL, 1988b, *22 Ton Steel Waste Dspl Box Impact Limiter Assy/Det*, drawing H-3-56433, Rev. 0 (9 sheets), Pacific Northwest Laboratory, Richland, Washington.
 Sheet 1, *22 Ton Steel Waste Dspl Box Impact Limiter Assy*
 Sheets 2 thru 9, *22 Ton Steel Waste Dspl Box Impact Limiter Det*

SAS, 1999, *ANSYS CD ROM Media, Revision 5.5.3*, ANSYS, Inc., Canonsburg, Pennsylvania.

SAS, 1998, *ANSYS Basic Analysis Procedure Guide for Revision 5.5*, ANSYS, Inc., Canonsburg, Pennsylvania.

WM			CHECKLIST FOR CHECKING OF ANALYSIS/CALCULATIONS	
Document Number/Revision: <u>HNF-4763/0 ANSYS 22 Ton Package FEA Thermal Analysis</u>				
Document Title: <u>Safety Analysis Report for Packaging (Onsite) Steel Waste Package</u>				
Yes	No	N/A		
☒	☐	☐	Problem completely defined.	
☒	☐	☐	Appropriate analytical method used.	
☒	☐	☐	Necessary assumptions are appropriate and explicitly stated.	
☒	☐	☐	Computer codes and data files documented.	
☒	☐	☐	Data used in calculations explicitly stated in document.	
☐	☐	☒	Sources of non-standard formula/data are referenced and the correctness of the reference verified.	
☒	☐	☐	Data checked for consistency with original source information as applicable.	
☐	☐	☒	Mathematical derivations checked including dimensional consistency of results.	
☒	☐	☐	Models appropriate and used within range of validity or use outside range of established validity justified.	
☐	☐	☒	Hand calculations checked for errors.	
☒	☐	☐	Code run streams correct and consistent with analysis documentation.	
☒	☐	☐	Code output consistent with input and with results reported in analysis documentation.	
☒	☐	☐	Acceptability limits on analytical results applicable and supported. Limits checked against sources.	
☒	☐	☐	Safety Margins consistent with good engineering practices.	
☒	☐	☐	Conclusions consistent with analytical results and applicable limits.	
☒	☐	☐	Results and conclusions address all points required in the problem statement.	
I have checked the analysis/calculation and it is complete and accurate to the best of my knowledge.				
Engineer/Checker <u>Scott Shiraga</u>			Date <u>02/02/00</u>	
NOTE: Any hand calculations, notes or summaries generated as part of this check should be signed, dated, and attached to this checklist. Material should be labeled and recorded so that it is intelligible to a technically qualified third party.				

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9.0 PRESSURE AND GAS GENERATION EVALUATION

Hydrogenous materials in the waste will generate hydrogen gas by radiolytic interactions. There are no waste forms authorized for the SWP that are expected to chemically generate hydrogen gas. The SWP has a series of nested containers, each vented by a NucFil-013 high-efficiency particulate air filter. The hydrogen gas will be generated within the innermost container and diffuse through the high-efficiency particulate air filters to the outer containers. The SWDB package is sealed within the SWDB Overpack. Calculations were made to determine the radiolytic hydrogen gas generation rate and the allowable shipping window to ensure that no void space within the packaging will exceed the 5-percent hydrogen gas criteria in twice the expected shipping window. Pressure calculations were also prepared to determine if the maximum pressure rise within the SWP during transport was below the 110.3 kPa, gauge (16-psig), design pressure (as shown in Part B, Section 7.9.2).

9.1 GAS GENERATION

The Radcalc for Windows computer code (McFadden et al. 1999) was used to determine the hydrogen generation rate from radiolytic interactions. Radcalc utilizes the G-value method to calculate the radiolytic hydrogen gas generation rate. $G(H_2)$ is equal to the number of molecules of H_2 generated per 100 eV of ionizing radiation absorbed. The $G(H_2)$ value used for the hydrogenous portion of the waste was assumed to be equivalent to that for mineral oil. The waste is restricted to dry waste from the 300 Area facilities. As given in Part B, Section 2.0, "dry" is defined as 5 percent or less of water and organics combined. The effective $G(H_2)$ value was determined by multiplying the $G(H_2)$ values for mineral oil by the mass fraction of hydrogenous material. A conservative approach was used. Diffusion through the Nucfil-013 filters was found to be less than 4 percent of the generation rate; therefore, no credit was taken for diffusion out of the RODC.

The gas generation rate resulting from the worst-case source term ($420,000 \text{ Ci } ^{90}\text{Sr} + ^{90}\text{Y}$) is $1700 \text{ cm}^3/\text{h}$. Given this hydrogen gas generation rate and an internal void volume of 0.9 m^3 , the time to reach 5-percent hydrogen is 1.12 days. The allowable shipping window is half the time required to reach 5-percent hydrogen or 0.56 days (13 hours) for this maximum source term.

The allowable shipping window is inversely related to the source term. That is, if the source term is reduced by half, the gas generation rate is reduced by half, and the shipping window is doubled. The allowable shipping window as a function of the waste package radionuclide inventory is shown in Table B9-1. Part A, Section 4.3.5, includes administrative controls to limit the amount of time that the package is sealed to ensure that hydrogen gas concentration does not exceed 5 percent.

Table B9-1. Steel Waste Package Shipping Window.

Source term (Ci)	Shipping window (hours)
500,000	11
400,000	14
300,000	19
200,000	28
100,000	56
90,000	63
80,000	71
70,000	81
60,000	94
50,000	113
40,000	141
30,000	188
20,000	282
10,000	564
5,000	1129

9.2 PACKAGE PRESSURE

The maximum allowable hydrogen gas partial pressure (at twice the shipping window) that may be present in the SWP is 5.33 kPa (5 vol% hydrogen). Accounting for oxygen gas generated at a 1:2 ratio to hydrogen, the maximum SWP internal differential pressure will be 8.0 kPa, gauge (1.16 psig), at 20 °C (68 °F).

As given in Part B, Section 8.0, under normal solar insolation conditions, the maximum internal temperature will be 156 °C (313 °F). If the SWP were sealed under ambient conditions of 20 °C (68 °F) and allowed to heat up to 156 °C (313 °F), the maximum air pressure inside the SWP would be 1.58 atm (absolute). At the maximum SWP air temperature of 156 °C (313 °F), the maximum internal differential pressure will be 58.2 kPa, gauge (8.44 psig), which is within the design pressure of 110.3 kPa (16 psig) and meets the acceptance criteria.

9.3 REFERENCE

McFadden, J. G., K. E. Hillesland, and A. V. Savino, 1999, *Radcalc for Windows Version 2.11, Volume I: User's Manual and Volume II: Technical Manual*, ENG-RPT-034, Rev. 0, Waste Management Technical Services, Inc., Richland, Washington.

9.4 GAS GENERATION AND PRESSURE EVALUATION

ENGINEERING SAFETY EVALUATION

Subject: Gas Generation & Pressure Evaluation in the Steel Waste Package Page 1 of 16

Preparer: B. Peters *B. Peters* Date 01/13/2000

Checker: E. Neilsen *E. Neilsen* Date 01/14/2000

1.0 Objectives:

The Steel Waste Package will be used to transport remote-handled radioactive waste from the 324 Building to the 200 Area for disposal. This evaluation will determine the rate of radiolytic hydrogen gas generation, the time to reach 2.5% hydrogen in the package, and the maximum internal pressure in the cask.

2.0 References:

1. McFadden, J. G., K. E. Hillesland, and A. V. Savino, 1999, *Radcalc for Windows, Version 2.11*, ENG-RPT-034, Rev. 0, Waste Management Technical Services, Inc., Richland, Washington.
2. NucFil Filter Brochure provided by NFT Incorporated, Lakewood, Colorado.
3. Liekhus, K. J., 1994, *Transport Characteristics Across Drum Filter Vents and Polymer Bags*, EGG-WM-11454, Idaho National Engineering Laboratory, EG&G Idaho Inc., Idaho Falls, Idaho.
4. Cussler, E. L., 1984, *Diffusion Mass Transfer In Fluid Systems*, Cambridge University Press, Cambridge.

3.0 Assumptions, Results, and Conclusions:

The Steel Waste Package (SWP) consists of three major parts. The first part, the Steel Waste Disposal Box (SWDB) Overpack, provides the containment during transportation and some shielding. The second part of the packaging, the SWDB, is placed in the SWDB Overpack and provides the primary shielding. The SWDB is removed from the Overpack when placed in storage, thus the SWDB provides primary containment during storage. The SWDB is vented into the SWDB Overpack through a 7-inch long 0.25-inch diameter channel and a Nucfil-013 HEPA filter. The third part of the packaging is the Rectangular Overpack Disposal Container (RODC) which is placed inside the SWDB. It contains an open-top rectangular grout container (RGC). Non-dispersible waste is placed directly in the RGC. Dispersible waste is first placed in an engineered container (EC) and then in the RGC. The EC is a sealed 10-inch diameter schedule 40 pipe, 60 inches long, vented through a Nucfil-013 HEPA filter. The RODC has a lid and is vented into the SWDB through a Nucfil-013 HEPA filter.

HYDROGEN GENERATION

Hydrogen generation from the SWP contents was evaluated assuming a uniformly distributed source term in eight (8) ECs placed in the RGC. The hydrogen gas generation

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rate for waste containing 5% hydrogenous material and the worst case source term was 1700 cm³/hr. The time to reach 5% hydrogen in the void space of the RODC is 1.12 days. The Radcalc output file is given in Appendix 5.1. Based on half the time to reach 5% hydrogen (i.e. 2.5%) in the sealed SWP, the shipping window for the worst case source term (420,000 Ci ⁹⁰Sr + ⁹⁰Y) is 13 hours.

The allowable time for shipment is an inverse function of the source term (assuming a constant hydrogenous material content); that is if the source term is cut in half, only half as much hydrogen is generated, and it will take twice as long to reach 5% hydrogen gas. Results of applying this relationship to a range of inventories are shown in Table 1.

Table 1. Steel Waste Package Shipping Window

Source Term (Ci)	Shipping Window (Hrs)
500,000	11
400,000	14
300,000	19
200,000	28
100,000	56
90,000	63
80,000	71
70,000	81
60,000	94
50,000	113
40,000	141
30,000	188
20,000	282
10,000	564
5,000	1129

PRESSURE

The maximum allowable hydrogen gas partial pressure (at twice the shipping window) which may be present in the SWP is 5.33 kPa (5 vol% hydrogen). Accounting for oxygen gas generated at a 1:2 ratio to hydrogen, the maximum SWP internal differential pressure will be 8.0 kPa (1.16 psig) at 20 °C. As given in Part B, Section 8, under normal solar insolation conditions the maximum internal temperature will be 156 °C (313 °F). If the SWP were sealed under ambient conditions of 20 °C (68 °F) and allowed to heat up to 156 °C, the maximum air pressure inside the SWP would be 1.58 atm (absolute). At the

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maximum cask air temperature of 156 °C, the maximum internal differential pressure will be 58.2 kPa (8.44 psig) which is within the design pressure of 16 psig.

4.0 Engineering Evaluation:

4.1 **Hydrogen Generation Rate**

Waste Volume and Void Volume

As described in Part A Section 2, there are two waste configurations to be shipped in the SWP. These are dispersible waste in ECs (maximum of 8) then in the RGC or non-dispersible waste placed directly in the RGC. The configuration that gives the least void space (8 ECs) and thus the shortest time to reach the LFL will be the limiting case for gas generation and is assumed in this evaluation.

The configuration of the SWP is shown in Figure A1-1. The following dimensions and weights are taken from Part A, Section 1 of the SARP. The ECs are assumed to have no available void space because they will be prepared well in advance of being placed in the SWB. The void spaces within the ECs will be in equilibrium with the hydrogen generation within each and are assumed to have no capacity for additional gas storage. This is not quite accurate but is conservative.

Assumptions/Constants

$$\text{carbon steel density} \quad \rho_{cs} := 7.85 \frac{\text{g}}{\text{cm}^3}$$

$$\text{stainless steel density} \quad \rho_{ss} := 8.00 \frac{\text{g}}{\text{cm}^3}$$

Engineered Containers (EC)

There are 8 ECs per package. The carbon steel container outer dimensions are 10.75" OD and 60" long

$$\text{Vol}_{EC} := 8 \cdot \pi \cdot \frac{(10.75 \text{ in})^2}{4} \cdot 60 \text{ in} \quad \text{Vol}_{EC} = 0.714 \text{ m}^3$$

Rectangular Grout Container (RGC)

$$W_{RGC} := 500 \text{ lb} \cdot 453.6 \frac{\text{g}}{\text{lb}}$$

$$\text{metalVol}_{RGC} := \frac{W_{RGC}}{\rho_{cs}} \quad \text{metalVol}_{RGC} = 0.029 \text{ m}^3$$

ENGINEERING SAFETY EVALUATION

Subject: Gas Generation & Pressure Evaluation in the Steel Waste Package Page 4 of 16Preparer: B. Peters *BP* Date 01/13/2000Checker: E. Neilsen *EN* Date 01/14/2000**Rectangular Overpack Disposal Container (RODC)**

outer dimensions are 72-5/8" x 41" x 34-5/8" and walls are 3/16" carbon steel

$$\text{Vol}_{\text{RODC}} := 72.625 \text{ in} \cdot 41 \text{ in} \cdot 34.625 \text{ in} \quad \text{Vol}_{\text{RODC}} = 1.69 \text{ m}^3$$

$$W_{\text{RODC}} := 800 \text{ lb} \cdot 453.6 \frac{\text{g}}{\text{lb}}$$

$$\text{metalVol}_{\text{RODC}} := \frac{W_{\text{RODC}}}{\rho_{\text{cs}}} \quad \text{metalVol}_{\text{RODC}} = 0.046 \text{ m}^3$$

$$\text{innerVol}_{\text{RODC}} := \text{Vol}_{\text{RODC}} - \text{metalVol}_{\text{RODC}} \quad \text{innerVol}_{\text{RODC}} = 1.643 \text{ m}^3$$

Steel Waste Disposal Box (SWDB)

outer dimensions are 96" x 60" x 48" and walls are stainless steel

$$\text{Vol}_{\text{SWDB}} := 96 \text{ in} \cdot 60 \text{ in} \cdot 48 \text{ in} \quad \text{Vol}_{\text{SWDB}} = 4.531 \text{ m}^3$$

$$W_{\text{SWDB}} := 44400 \text{ lb} \cdot 453.6 \frac{\text{g}}{\text{lb}}$$

$$\text{metalVol}_{\text{SWDB}} := \frac{W_{\text{SWDB}}}{\rho_{\text{ss}}} \quad \text{metalVol}_{\text{SWDB}} = 2.517 \text{ m}^3$$

$$\text{innerVol}_{\text{SWDB}} := \text{Vol}_{\text{SWDB}} - \text{metalVol}_{\text{SWDB}} \quad \text{innerVol}_{\text{SWDB}} = 2.013 \text{ m}^3$$

Steel Waste Disposal Box Overpack (OP)

outer dimensions are 109" x 70" x 52.8" and walls are stainless steel

$$\text{Vol}_{\text{OP}} := 109 \text{ in} \cdot 70 \text{ in} \cdot 52.8 \text{ in} \quad \text{Vol}_{\text{OP}} = 6.602 \text{ m}^3$$

$$W_{\text{OP}} := 8300 \text{ lb} \cdot 453.6 \frac{\text{g}}{\text{lb}}$$

$$\text{metalVol}_{\text{OP}} := \frac{W_{\text{OP}}}{\rho_{\text{ss}}} \quad \text{metalVol}_{\text{OP}} = 0.471 \text{ m}^3$$

$$\text{innerVol}_{\text{OP}} := \text{Vol}_{\text{OP}} - \text{metalVol}_{\text{OP}} \quad \text{innerVol}_{\text{OP}} = 6.131 \text{ m}^3$$

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Void Volumes

There are three areas of void space within the package, each separated by a Nucfil-013 HEPA filter. The first area (1) is the void space within the Rectangular Overpack Disposal Container. Assume that the engineered containers have no void (conservative). The second area (2) is the void space outside the RODC but within the Steel Waste Disposal Box. The RODC and the SWDB both have Nucfil-013 filters. The third void space (3) is outside the SWDB but within the SWDB Overpack. The void volume calculations are shown below.

$$\text{Void}_1 := \text{innerVol}_{\text{RODC}} - \text{metalVol}_{\text{RGC}} - \text{Vol}_{\text{EC}} \quad \text{Void}_1 = 0.9 \text{ m}^3$$

$$\text{Void}_2 := \text{innerVol}_{\text{SWDB}} - \text{Vol}_{\text{RODC}} \quad \text{Void}_2 = 0.324 \text{ m}^3$$

$$\text{Void}_3 := \text{innerVol}_{\text{OP}} - \text{Vol}_{\text{SWDB}} \quad \text{Void}_3 = 1.6 \text{ m}^3$$

$$\text{TotalVoid} := \text{Void}_1 + \text{Void}_2 + \text{Void}_3 \quad \text{TotalVoid} = 2.825 \text{ m}^3$$

Diffusion Path and Cross-Sectional Area

There are three primary resistances to diffusion in the SWP. These are the two Nucfil-013 filters (F1 & F3) and the 0.25-inch diameter channel (R2) through the SWDB lid. The diffusion of hydrogen is a function of both the diffusional path and the cross-sectional area of the flow. These values are calculated below for the three primary diffusional resistances.

"F1" and "F3": First and third resistances are Nucfil-013 vent filters.

$$\text{filterthickness} := 0.6 \text{ in} \quad \text{filterdiam} := 0.73 \text{ in}$$

$$\text{tortuosity} \quad \tau := \sqrt{2} \quad (\text{approximation for randomly packed bed})$$

Filter void fraction was adjusted to give results matching laboratory diffusion rate as provided in reference 3 (Liekhus 1994, $1.17\text{E-}5$ mol/s/mole fraction).

$$\text{filtervoid} := 38.0\%$$

$$\text{pathF1} := \text{filterthickness} \cdot \tau \quad \text{pathF3} := \text{filterthickness} \cdot \tau$$

$$\text{areaF1} := \text{filtervoid} \cdot \pi \cdot \frac{\text{filterdiam}^2}{4} \quad \text{areaF1} = 1.026 \text{ cm}^2$$

$$\text{areaF3} := \text{filtervoid} \cdot \pi \cdot \frac{\text{filterdiam}^2}{4} \quad \text{areaF3} = 1.026 \text{ cm}^2$$

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Calculation of diffusion rate through Nucfil-013 vent filter

$$Q_{ref} := 0.611 \frac{\text{cm}^2}{\text{s}} \quad \text{diffusion constant of hydrogen in air at 273K, from ref. 4 (Cussler 1984)}$$

$$T_{ref} := 273 \text{ K} \quad T = 293 \text{ K}$$

$$P_3 := \frac{64.4}{2.54} \text{ in}_\text{Hg} \quad T_3 := 298 \text{ K} \quad (\text{lab conditions from ref. 3})$$

$$Q_3 := Q_{ref} \frac{T_3^{1.75}}{T_{ref}^{1.75}} \quad Q_3 = 0.712 \frac{\text{cm}^2}{\text{s}} \quad = \text{diffusion constant at 298K}$$

$$Q := Q_{ref} \frac{T^{1.75}}{T_{ref}^{1.75}} \quad Q = 0.691 \frac{\text{cm}^2}{\text{s}} \quad = \text{diffusion constant at 293K}$$

$$d\text{Conc} := \frac{P_3}{R_g \cdot T_3} \quad d\text{Conc} = 3.455 \cdot 10^{-5} \frac{\text{mol}}{\text{cm}^3}$$

$$N_{F1} := \text{areaF1} \cdot Q_3 \frac{d\text{Conc}}{\text{pathF1}} \quad N_{F1} = 1.172 \cdot 10^{-5} \frac{\text{mol}}{\text{s}}$$

=> This is close to the reference value (ref. 3) of 1.17E-5 mol/s/mole fraction) and assumes that the diffusional resistance of the filter inlet and outlet are insignificant relative to that of the filter media.

"R2": The second resistance is a 0.25" ID channel angle cut through the SWDB lid. This is shown in drawing H-3-56431, sheet 2 of 4, rev 1. This channel consists of two parts the first is cut right-to-left through a 3" piece of the lid at an angle of 30 degrees from vertical. The second part is cut left-to-right through a 4" piece of the lid at an angle of 30 degrees from vertical. This second cut however opens into the Nucfil-013 filter which is countersunk 1" such that this second cut only penetrates 3" of this part of the lid. Both portions of the channel can be considered to be 3 inches long for a total of 6 inches 30 degrees from vertical.

$$\text{pathR2} := \frac{2 \cdot (3 \text{ in})}{\cos\left(\frac{30}{180} \pi\right)} \quad \text{pathR2} = 6.928 \text{ in}$$

$$\text{areaR2} := \pi \frac{(0.25 \text{ in})^2}{4} \quad \text{areaR2} = 3.167 \cdot 10^{-5} \text{ m}^2$$

ENGINEERING SAFETY EVALUATION

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Radcalc for Windows (McFadden et al. 1999) was used to calculate the radiolytic production of hydrogen gas in the SWP. Radcalc uses the G value method to quantify the volume of hydrogen produced in the drum. The G value of a material is defined as the number of molecules formed or disassociated per 100 eV energy absorbed in the waste matrix. G values are dependant on the type of radioactive decay and the absorbing material. The $G(H_2)$ value used for the hydrogenous portion of the waste was assumed to be equivalent to that for mineral oil which has an gamma $G(H_2)$ of 2.7 (ref. 1). The $G(H_2)$ value for beta and alpha were assumed to be equal to that for gamma (2.7) and 4 times that for gamma (10.8), respectively. These are conservative values and are higher than that for water. The waste is restricted to dry waste from the 324 Building. As given in Part B, Section 2, "dry" is defined as 5% or less of water and organics combined. The effective $G(H_2)$ value was determined by multiplying the mineral oil values by the mass fraction of hydrogenous material.

The software Radcalc for Windows assumes that 100% of the energy emitted from alpha and beta radiation is absorbed in the matrix generating hydrogen. For gamma radiation, however, a percent of the emitted energy escapes the media. The software has pre-calculated absorption fractions for packages frequently used on the Hanford Site. For this study, the 4'x4' liner was used as it most closely approximates the shape and internal volume of the rectangular grout container.

Source Term

As seen in Part B section 2, there are two ways to define the maximum source term. There is an expected maximum radionuclide inventory derived from characterization data from eleven earlier waste packages as shown in Table B2-1. Alternatively a maximum radionuclide inventory can be calculated using a sum-of-fractions equation. This is described in Part B section 2.1.2 with the individual isotope activity limits shown in Table B2-4.

As described above, Radcalc for Windows accounts for gamma radiation that escapes from the waste package without interacting with the media. The worst case source term should therefore be one generating only beta/alpha radiation (100% interaction). As shown in Table B2-1, the primary expected radionuclides are ^{90}Sr and ^{137}Cs . These two radionuclides with their daughter products make up 99.2% of the expected source term. Considering only these two radionuclides, a source term containing only ^{90}Sr will produce more hydrogen than one containing a mixture. The worst case source term therefore will be ^{90}Sr set at the maximum activity limit from Table B2-4 (210,000 Ci).

The hydrogen generation rate calculated for the worst case source term is $1700 \text{ cm}^3/\text{hr}$. The Radcalc output file is given in Appendix 5.1. The hydrogen generation rate

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calculated for the expected maximum source term (given in Table B2-1) is 1560 cm³/hr.
The Radcalc output file is given in Appendix 5.2

Hydrogen Diffusion

Each region is a component of the diffusion and can be treated as a series resistance. First calculate the hydrogen generation rate as gmoles per second, then for each component calculate the concentration gradient necessary to drive the hydrogen diffusion rate equal to the generation rate. The concentration gradients necessary for each component are added to determine the maximum gradient and thus the maximum concentration of hydrogen. Fick's law of diffusion is used in the following form:

$$\text{flux per unit area } j = \frac{Q \cdot \partial C}{L}$$

$$\text{diffusional flow rate } n = A \cdot j$$

$$\text{rearranging } \partial C = \frac{n \cdot L}{Q \cdot A}$$

where n = diffusional flowrate (mol/sec)
 Q = hydrogen diffusion constant (0.611 cm²/s at 273K per Cussler 1988)
 L = diffusional path length (cm)
 A = diffusional cross-sectional area (cm²)
 ∂C = concentration gradient (mol/cm³)

The next step, given the diffusion path lengths and cross-sectional areas calculated above, is to determine the concentration gradient across each of the resistances required to give a diffusional flowrate equal to the gas generation rate of 1700 cm³/hr calculated for the worst case source term (see appendix 5.1).

Calculate hydrogen molar generation rate in 22 Ton Box

$V := 1700 \frac{\text{cm}^3}{\text{hr}}$ This is the hydrogen generation rate calculated for the maximum source term (see appendix 5.1, Radcalc file 22TONBX1.RAD).

$$P_0 := 1 \cdot \text{atm} \quad T := 293 \cdot \text{K} \quad R_g := 82.06 \frac{\text{atm} \cdot \text{cm}^3}{\text{mol} \cdot \text{K}}$$

$$n := \frac{P_0 \cdot V}{R_g \cdot T} \quad n = 1.964 \cdot 10^{-5} \frac{\text{mol}}{\text{s}} \quad = \text{Hydrogen molar generation rate at STP}$$

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Now calculate the concentration gradient across each segment of the overall diffusion path within the SWP.

$$\begin{aligned}
 Q &= 0.691 \frac{\text{cm}^3}{\text{s}} & dC_1 &:= n \frac{\text{pathF1}}{(Q \cdot \text{areaF1})} & dC_1 &= 59.66 \frac{\text{mol}}{\text{m}^3} \\
 & & dC_2 &:= n \frac{\text{pathR2}}{(Q \cdot \text{areaR2})} & dC_2 &= 1.578 \cdot 10^3 \frac{\text{mol}}{\text{m}^3} \\
 & & dC_3 &:= n \frac{\text{pathF3}}{(Q \cdot \text{areaF3})} & dC_3 &= 59.66 \frac{\text{mol}}{\text{m}^3} \\
 dC &:= dC_1 + dC_2 + dC_3 & dC &= 1.698 \cdot 10^3 \frac{\text{mol}}{\text{m}^3} \\
 PH_2 &:= dC \cdot R_g \cdot T & PH_2 &= 40.817 \text{ atm}
 \end{aligned}$$

As can be seen, the concentration gradient required for diffusion to equal the hydrogen generation rate is excessive (40 atm) and will not occur. It is also apparent that the primary resistance to diffusion is the channel in the SWDB lid but that the Nucfil-013 filters will also provide significant resistance at these flowrates.

Next the diffusional flowrate through a single Nucfil-013 filter will be determined for a differential concentration equal to 5 volume percent hydrogen.

$$\begin{aligned}
 \Delta C &:= \frac{(0.05 \text{ atm})}{R_g \cdot T} \\
 n_1 &:= \text{areaF1} \cdot Q \cdot \frac{\Delta C}{\text{pathF1}} & n_1 &= 6.846 \cdot 10^{-7} \frac{\text{mol}}{\text{s}} \\
 V_1 &:= n_1 \cdot R_g \cdot \frac{T}{P_0} & V_1 &= 59.256 \frac{\text{cm}^3}{\text{hr}}
 \end{aligned}$$

As seen, the diffusional flowrate through a single Nucfil-013 filter at 5% hydrogen is less than 60 cm³/hr. This is compared to the generation rate of 1700 cm³/hr. Based on these calculations, even the first Nucfil-013 filter will present a significant resistance to the diffusion of hydrogen from the RODC. The hydrogen gas concentration calculations will assume the package void volume is only that within the RODC (0.90 m³, Void₁ above).

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Therefore, as determined in the Radcalc model (see Appendix 5.1), the time to reach 2.5% hydrogen is 0.56 days (13 hrs) for the worst case source term (420,000 Ci ^{90}Sr + ^{90}Y). The allowable shipping window is inversely related to the source term. That is, if the source term is reduced by half, the gas generation rate is reduced by half and the allowable shipping window is doubled. The allowable shipping window as a function of the waste package radionuclide inventory was given in Table 1.

4.2 Maximum Cask Pressure

The maximum hydrogen concentration in the SWP will be 5% hydrogen (a shipping window will be established to limit this to 2.5%). It is assumed that oxygen gas is also produced from radiolysis. The stoichiometric quantity is 0.5 moles of oxygen for each mole of hydrogen. Therefore the maximum pressure in the SWP (at standard temperature of 20 °C) will be:

$$P_1 := P_0 + \left[\left(0.05 + \frac{0.05}{2} \right) \cdot \text{atm} \right] \quad P_1 = 1.075 \text{ atm}$$

As given in Part B, Section 8, under normal solar insolation conditions the maximum internal temperature will be 156 °C (313 °F). Now calculate the maximum internal pressure at the maximum temperature accounting for the gas generation.

$$T_{\text{max}} := (273.15 + 156) \cdot \text{K} \quad T_{\text{max}} = 429.15 \text{ K} \quad T := 293 \cdot \text{K}$$

$$P_{\text{max}} := P_1 \cdot \frac{T_{\text{max}}}{T} \quad P_{\text{max}} = 1.575 \text{ atm}$$

$$P_{\text{max}} - P_0 = 8.443 \text{ psi}$$

This is less than the design pressure of 16 psig and the SWP meets the acceptance criteria for maximum pressure.

5.0 Appendices:

5.1 SWP Hydrogen Generation: Worst Case Source Term (Radcalc Output file)

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Radcalc for Windows 2.11

Date: 01-12-00 18:44

Performed By: B Peters

Checked By: E Neilsen

File: 22TONBX1.RAD

===== Input Information =====

Source from input:

Radionuclide:	Curies:	Becquerels:	Grams:
Sr-90	2.10E+005	7.77E+015	1.49E+003
Y-90	2.10E+005	7.77E+015	3.87E-001 (Daughter)

Total Activity:	4.20E+005	1.55E+016
Total Minus Daughters:	2.10E+005	7.77E+015

Waste Form: Normal
 Physical Form: Solid
 Container Type: 4 x 4 Liner

Package Void Volume: 9.00E+005 cc
 Waste Volume: 7.14E+005 cc
 Waste Mass: 8.21E+005 g
 Waste Void Volume: 0.000 cc

Days to decay source before seal time: 0.00 days
 Calculate number of days sealed until 2.50% hydrogen is reached.

G Value Material Selection:

Contribution	Weight	G-Alpha	G-Beta	G-Gamma	Name
5.00%	0.05	10.8	2.7	2.7	Mineral oils
95.00%	0.95	0	0	0	USER

G Values are calculated from list averaging:

G Alpha	G Beta	G Gamma
0.54	0.135	0.135

Comments:

Mass of hydrogenous material set at 5%. Assume the maximum radiolysis will be equal to mineral oil. Waste density set at 1.15 g/cc (50% of theoretical density). The void volume is 900,000 cc (0.9 m3). Assume no waste void volume.

===== Calculated Results =====

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HYDROGEN:

The sealed package will contain 2.50 % hydrogen in 0.56 days

H2 Volume (20 C, 101.325 kPa): 2.31E+004 cc

H2 Rate (20 C, 101.325 kPa): 1.70E+003 cc/hour

DECAY HEAT AND PRESSURE:

Heat Generated at seal time: 1.41E+003 Watts

Partial Pressure (H2): 2.60 kPa

Partial Pressure (O2): 1.30 kPa (if H2O is present in waste)

Total Pressure (H2 + Air): 104. kPa

Total Pressure (H2 + O2 + Air): 105. kPa (if H2O is present in waste)

TRANSPORTATION:

Note: Transportation classifications assume three significant figures.

Calculations are made at user-specified decay time.

Radioactive: Yes (5.12E+008 nCi/g)

Effective A2 for Mixture: 2.70 Ci

Type Determination: B (from unity fraction 77778.)

Limited Quantity: No

LSA-I Determination: No (from LSA-I unity fraction 94724.)

LSA-II Determination: No (from LSA-II unity fraction 947.24)

LSA-III Determination: No (from LSA-III unity fraction 47.362)

HRC Quantity Determination: Yes

Fissile Quantity: 0.000 g

Fissile Excepted: NA

There are no fissile radionuclides present per 49CFR173.403

Container Category: I (Container Category per Reg Guide 7.11)

TRU Waste: No (Transuranic Waste per DOE Order 5820.2a)

Reportable Quantity: Yes (from RQ unity fraction 2.10E+006)

Source at start of seal time:

Radionuclide:	Curies:	Becquerels:	Grams:
Sr-90	2.10E+005	7.77E+015	1.49E+003
Y-90	2.10E+005	7.77E+015	3.87E-001 (Daughter)
Total Activity:	4.20E+005	1.55E+016	
Total Minus Daughters:	2.10E+005	7.77E+015

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Subject: Gas Generation & Pressure Evaluation in the Steel Waste Package Page 13 of 16

Preparer: B. Peters *BP* Date 01/13/2000Checker: E. Neilsen *EN* Date 01/14/2000

Shipping Papers and Labels:

ISOTOPE	Number of A2s	Fraction of Total A2s	Cumulative Total A2s
* Sr-90	7.78E+004	1.00E+000	1.00000
Y-90	0.00E+000	0.00E+000	1.00000

* Contains 95% of the total A2s and must be included per 49 CFR 173.433.

5.2 SWP Hydrogen Generation: Maximum Source Term (Radcalc Output file)

Radcalc for Windows 2.11

Date: 01-09-00 22:01

Performed By: B Peters

Checked By: E Neilsen

File: 22TONBOX.RAD

===== Input Information =====

Source from input:

Radionuclide:	Curies:	Becquerels:	Grams:
Co-60	2.68E+000	9.92E+010	2.37E-003
Sr-90	8.83E+004	3.27E+015	6.25E+002
Y-90	8.83E+004	3.27E+015	1.62E-001 (Daughter)
Cs-137	1.66E+005	6.15E+015	1.91E+003
Ba-137m	1.57E+005	5.81E+015	2.92E-004 (Daughter)
Eu-154	1.13E+002	4.18E+012	4.18E-001
U-234	1.09E-003	4.03E+007	1.75E-001
Pu-238	1.26E+002	4.66E+012	7.41E+000
Pu-239	9.54E+000	3.53E+011	1.54E+002
Pu-240	6.36E+000	2.35E+011	2.77E+001
Am-241	1.75E+002	6.47E+012	5.10E+001

Total Activity:	5.00E+005	1.85E+016
Total Minus Daughters:	2.55E+005	9.43E+015

Waste Form: Normal
 Physical Form: Solid
 Container Type: 4 x 4 Liner

Package Void Volume: 9.00E+005 cc
 Waste Volume: 7.14E+005 cc
 Waste Mass: 8.21E+005 g

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Subject: Gas Generation & Pressure Evaluation in the Steel Waste Package Page 14 of 16

Preparer: B. Peters *BP* Date 01/13/2000Checker: E. Neilsen *EN* Date 01/14/2000

Waste Void Volume: 0.000 cc

Days to decay source before seal time: 0.00 days

Calculate number of days sealed until 2.50% hydrogen is reached.

G Value Material Selection:

Contribution	Weight	G-Alpha	G-Beta	G-Gamma	Name
5.00%	0.05	10.8	2.7	2.7	Mineral oils
95.00%	0.95	0	0	0	USER

G Values are calculated from list averaging:

G Alpha	G Beta	G Gamma
0.54	0.135	0.135

Comments:

Mass of hydrogenous material set at 5%. Assume the maximum radiolysis will be equal to mineral oil. Waste density set at 1.15 g/cc (50% of theoretical density). The void volume is 900,000 cc (0.9 m3). Assume no waste void volume.

===== Calculated Results =====

HYDROGEN:

The sealed package will contain 2.50 % hydrogen in 0.62 days

H2 Volume (20 C, 101.325 kPa): 2.31E+004 cc

H2 Rate (20 C, 101.325 kPa) : 1.56E+003 cc/hour

DECAY HEAT AND PRESSURE:

Heat Generated at seal time: 1.40E+003 Watts

Partial Pressure (H2): 2.60 kPa

Partial Pressure (O2): 1.30 kPa (if H2O is present in waste)

Total Pressure (H2 + Air): 104. kPa

Total Pressure (H2 + O2 + Air): 105. kPa (if H2O is present in waste)

TRANSPORTATION:

Note: Transportation classifications assume three significant figures.

Calculations are made at user-specified decay time.

Radioactive: Yes (6.09E+008 nCi/g)

Effective A2 for Mixture: 2.46 Ci

Type Determination: B (from unity fraction 1.0357E+005)

Limited Quantity: No

ENGINEERING SAFETY EVALUATION

Subject: Gas Generation & Pressure Evaluation in the Steel Waste Package Page 15 of 16Preparer: B. Peters Date 01/13/2000Checker: E. Neilsen Date 01/14/2000

LSA-I Determination: No (from LSA-I unity fraction 1.0509E+005)

LSA-II Determination: No (from LSA-II unity fraction 1050.9)

LSA-III Determination: No (from LSA-III unity fraction 52.546)

HRC Quantity Determination: Yes

Fissile Quantity: 161. g

Fissile Excepted: Yes

5g fissile radionuclides per 10 liters or less per 49CFR173.453(d)

Container Category: I (Container Category per Reg Guide 7.11)

TRU Waste: Yes (Transuranic Waste per DOE Order 5820.2a)

Reportable Quantity: Yes (from RQ unity fraction 1.08E+006)

Source at start of seal time:

Radionuclide:	Curies:	Becquerels:	Grams:
Co-60	2.68E+000	9.92E+010	2.37E-003
Sr-90	8.83E+004	3.27E+015	6.25E+002
Y-90	8.83E+004	3.27E+015	1.62E-001 (Daughter)
Cs-137	1.66E+005	6.15E+015	1.91E+003
Ba-137m	1.57E+005	5.81E+015	2.92E-004 (Daughter)
Eu-154	1.13E+002	4.18E+012	4.18E-001
U-234	1.09E-003	4.03E+007	1.75E-001
Pu-238	1.26E+002	4.66E+012	7.41E+000
Pu-239	9.54E+000	3.53E+011	1.54E+002
Pu-240	6.36E+000	2.35E+011	2.77E+001
Am-241	1.75E+002	6.47E+012	5.10E+001
Total Activity:	5.00E+005	1.85E+016	
Total Minus Daughters:	2.55E+005	9.43E+015	

Shipping Papers and Labels:

ISOTOPE	Number of A2s	Fraction of Total A2s	Cumulative Total A2s
* Sr-90	3.27E+004	3.16E-001	0.315573
* Am-241	3.23E+004	3.12E-001	0.627885

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Subject: Gas Generation & Pressure Evaluation in the Steel Waste Package Page 16 of 16Preparer: B. Peters *BP* Date 01/13/2000Checker: E. Neilsen *EN* Date 01/14/2000

* Pu-238	2.33E+004	2.25E-001	0.852750
* Cs-137	1.23E+004	1.19E-001	0.971541
Pu-239	1.76E+003	1.70E-002	0.988566
Pu-240	1.18E+003	1.14E-002	0.999916
Eu-154	8.37E+000	8.08E-005	0.999997
Co-60	2.48E-001	2.40E-006	1.00000
U-234	4.04E-002	3.90E-007	1.00000
Ba-137m	0.00E+000	0.00E+000	1.00000
Y-90	0.00E+000	0.00E+000	1.00000

* Contains 95% of the total A2s and must be included per 49 CFR 173.433.

10.0 PACKAGE TIEDOWN SYSTEM EVALUATION

The tiedown system for the 22-Ton SWP consists of structural carbon steel angles that are permanently welded to a specially modified conveyance to provide lateral and longitudinal securement during transport. In addition, five turnbuckles on each longitudinal end of the package provide vertical and additional longitudinal restraint (shown in Figure B10-1). Although the package is a performance-based package meeting all the test requirements of 10 CFR 71, it is not a licensed package. Consequently, transport of the package is restricted to the Hanford Site. Loading of the SWP will take place at the 324 Building, located in the 300 Area, and off-loading will take place in the 200 Area. Because the package is not a licensed package, the tiedown requirements are established based on the more restrictive requirements of the Draft *Uniform Cargo Securement Standard for Highway Transport Vehicles* (CCMTA 1997).

10.1 SYSTEM DESIGN

The tiedown system is designed for use with a specially modified trailer that is shown on drawing H-3-56435 (PNL 1988d). As designed, the system consists of structural carbon steel angles 20.3 cm (8 in.) by 20.3 cm (8 in.) by 1.54 cm (1 in.) that are permanently welded to the bed of the conveyance and surround the base of the lower half of the SWDB Impact Limiter. Five, 3.8 cm (1.5-in.) turnbuckles are connected between the D-rings on each end of the lower half of the SWDB Impact Limiter and D-rings on the trailer.

The package is restrained to prevent movement and shifting during all NCT and in response to emergency situations, such as panic stops and emergency evasive maneuvers. Consequently, the package restraint system is demonstrated to be capable of withstanding the following forces (applied independently) as defined by the Canadian Council of Motor Transport Administrators in CCMTA (1997):

- 0.8g times the weight of the package in a forward direction
- 0.5g times the weight of the package in a rearward direction
- 0.5g times the weight of the package in the lateral direction
- 0.2g times the weight of the package relative to the load in a vertical direction, discounting gravity.

Figure B10-1. Tiedown Details of Steel Waste Package.

10.2 RATING AND ATTACHMENTS

The 22-Ton SWP securement system is an engineered tiedown system specifically designed for this package. Components specified for the tiedown system have no specific rating because each is manufactured from steel for uses other than cargo securement.

Consequently, in order to maintain safe securement of the package, the design margins of safety are based on the yield strength of the material. As demonstrated by the positive margins of safety presented in Part B, Section 10.4, the package can be safely transported on the Hanford Site without loss of cargo. The lowest margin of safety of the system is 0.3 at the chock angle welds based on the standard published yield strength of the material. In addition, the specified design loadings exceed the U.S. Department of Transportation cargo securement requirements specified in 49 CFR 393, "Parts and Accessories Necessary for Safe Operation," 393.100, "General rules for protection against shifting cargo."

10.3 REFERENCES

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

49 CFR 393, "Parts and Accessories Necessary for Safe Operation," *Code of Federal Regulations*, as amended.

CCMTA, 1997, *A Uniform Cargo Securement Standard for Highway Transport Vehicles* (DRAFT), Canadian Council of Motor Transport Administration, Ottawa, Ontario, Canada.

PNL, 1988d, *22 Ton Steel Waste Dspl Box Trailer Mod*, drawing H-3-56435, Rev. 0 (6 sheets), Pacific Northwest Laboratory, Richland, Washington.

10.4 APPENDIX: TIEDOWN REQUIREMENTS **ENGINEERING SAFETY EVALUATION**

Subject: 22 Ton Steel Waste Disposal Box Tiedown Requirements	Page: 1 of 11
Preparer: JE Scott	Date: 10/21/99
Checker: SS Shiraga	Date: 10/21/99

1.0 OBJECTIVE

The objective of this evaluation is to determine the tiedown performance of the 22.5 ton steel waste disposal box (SWDB) to the conveyance against the Canadian Council of Motor Transport Administration (CCMTA) parameters.

2.0 REFERENCES

- AISC, 1989, *Manual of Steel Construction*, Ninth Edition, American Institute of Steel Construction, New York, New York.
- ASTM, 1994, *Annual Book of Standards*, Volume 15.08, Section 15, American Society for Testing and Materials, Philadelphia, Pennsylvania.
- AWS, 1988, *Structural Welding Code – Steel*, Eleventh Edition, D1.1-88, American Welding Society, Inc., Miami, Florida.
- CCMTA, 1997, *A Uniform Cargo Securement Standard for Highway Transport Vehicles*, Canadian Council of Motor Transport Administration. Ottawa, Ontario, Canada.
- Shigley, Joseph E, and Charles R. Mischke, 1989, *Mechanical Engineering Design*, Fifth Edition, McGraw-Hill, Inc. New York, New York.
- WMNW, 1999, *22 Ton Steel Waste Disposal Box Arrangement*, drawing E-D-03, Waste Management Northwest, Richland, Washington.
- PNL, 1988b, *22 Ton Steel Waste Disposal Box Details*, drawing H-3-56431, Rev. 1, Pacific Northwest Laboratory, Richland, Washington.
- PNL, 1988c, *22 Ton Steel Waste Dspl Box Impact Limiter Det*, drawing H-3-56433, Rev. 0, Pacific Northwest Laboratory, Richland, Washington.
- PNL, 1988d, *22 Ton Steel Waste Disposal Box Lifting Yoke*, drawing H-3-56434, Rev. 2, Pacific Northwest Laboratory, Richland, Washington.
- PNL, 1988e, *22 Ton Steel Waste Dspl Box Trailer Mod*, drawing H-3-56435, Rev. 0, Pacific Northwest Laboratory, Richland, Washington.
- Carr Lane, 1998, *Component Parts of Jigs and Fixtures*, Carr Lane Manufacturing Co., St. Louis, Missouri.

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Subject: 22 Ton Steel Waste Disposal Box Tiedown Requirements	Page: 2 of 11
Preparer: JE Scott <i>JE Scott</i>	Date: 10/21/99
Checker: SS Shiraga <i>SS Shiraga</i>	Date: 10/21/99

3.0 ASSUMPTIONS, RESULTS, AND CONCLUSIONS

The SWDB must be secured to the conveyance during transport. The securement requirements are taken from the Canadian Council of Motor Transport Administrators. Friction is assumed to be negligible and is not considered in this analysis. The assumed inertial load factors of the SWDB are as follows:

- 0.8g deceleration in the forward direction.
- 0.5g deceleration in the backward direction.
- 0.5g deceleration in the lateral direction.
- 0.2g in the vertical or upward direction.

The center of gravity for the SWDB is assumed to be the geometric center of the 22.5 ton SWDB. Using the computed center of gravity height of 37.03 in., analysis of the moments created on the SWDB by the inertial loads confirms that it will not tip over laterally or longitudinally. A vertical force balance analysis shows that the turnbuckles and tiedowns have sufficient strength to hold the SWDB down. The safety hoist tiedown has the same allowable as the turnbuckles so the margin of safety for both is 7.3. A longitudinal force balance analysis verifies that the 10 turnbuckles will successfully restrain the SWDB from forward or backward motion with a margin of safety of 1.0.

The welds are assumed to fail before any bending of the angles occurs. For this reason, analysis of the angle is unnecessary. In both the lateral and longitudinal weld evaluation, only the outer 0.25 in. fillet weld is considered. The corner welds are neglected for conservatism. The allowable for the weld is acquired by multiplying the yield strength of the base metal by 0.3. This procedure is taken from the *Structural Welding Code – Steel* (AWS 1988). The margin of safety for the lateral welds and longitudinal welds is 5.4 and 1.9, respectively. The fasteners used to secure the tiedown are conservatively assumed to be machined from annealed stock. This yield strength of 30 ksi is obtained from the *Annual Book of Standards* (ASTM 1994, vol 15.08). The axial and shear stresses imposed on each fastener are analyzed and a margin of safety of 3.0 and 2 is obtained.

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Preparer: JE Scott *JE Scott* **Date:** 10/21/99
Checker: SS Shiraga *SS Shiraga* **Date:** 10/21/99

4.0 EVALUATION

Total weight of the SWDB:
from drawing H-3-56435 (PNL, 1988e)

$$W_{\text{total}} := 92000 \text{ lbf}$$

Center of gravity height of SWDB:

$$H_{\text{cg1}} := 24.03 \text{ in}$$

Height of package from conveyance:
from drawings H-3-56435 (PNL, 1988e),
H-3-56433 (PNL, 1988c), H-3-56430
(PNL 1988a), X (WMNW, 1999)

$$H_p := 13 \text{ in}$$

Center of gravity height relative to conveyance:

$$H_{\text{cg2}} := H_{\text{cg1}} + H_p$$

$$H_{\text{cg2}} = 37.03 \text{ in}$$

Figure 1. Height of Packaging.

Tipping of SWDB:

Lateral: $M_l := .5 \cdot W_{\text{total}} \cdot H_{\text{cg2}} - W_{\text{total}} \cdot 21 \text{ in}$

Moment Eqn

about origin O

$$M_l = -2.29 \cdot 10^5 \text{ lbf} \cdot \text{in}$$

The resulting moment is negative which indicates that the SWDB will not tip

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Moment Eqn
about origin O $M_2 = -8.63 \cdot 10^5 \text{ lbf-in}$

Figure 3. Longitudinal Tipping.

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Vertical Load Analysis:

Angle between turnbuckle and vertical axis: $\theta := 44.5^\circ$

Force in each turnbuckle due to the vertical load. There are 10 turnbuckles attached to the SWDB (T_v derived from a force balance eqn)

$$T_v := \frac{.2 \cdot W_{total}}{10 \cdot \cos(\theta)}$$

$$T_v = 2.58 \cdot 10^3 \text{ lbf}$$

Safe working load obtained from drawing H-3-56435 (PNL, 1988e)

$$T_{v_allowable} := 21400 \text{ lbf}$$

Since $T_v < T_{v_allowable}$, this is acceptable

$$\text{Margin_of_Safety} := \left(\frac{T_{v_allowable}}{T_v} - 1 \right)$$

$$\text{Margin_of_Safety} = 7.3$$

Figure 4. Vertical Load.

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Longitudinal Load Analysis:

Angle between turnbuckle and vertical axis is the same

Force in each turnbuckle due to the forward load. There are 10 turnbuckles attached to the SWDB (T_h derived from a force balance eqn)

$$T_h := \frac{.8 \cdot W_{\text{total}}}{10 \cdot \sin(\theta)}$$

$$T_h = 1.05 \cdot 10^4 \text{ lbf}$$

Safe working load obtained from drawing H-3-56435 (PNL, 1988e)

$$T_{h_allowable} := 21400 \text{ lbf}$$

Since $T_h < T_{h_allowable}$, this is acceptable

$$\text{Margin_of_Safety} := \left(\frac{T_{h_allowable}}{T_h} - 1 \right)$$

$$\text{Margin_of_Safety} = 1.04$$

Figure 5. Longitudinal Load.

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The force of the vertical load is translated through the turnbuckles to the safety hoist tiedown attachment. The tiedown hoist, which will observe the same load as the turnbuckles, has a safe working load of 20,000 lbf. This load is specified in drawing H-3-56435 (PNL, 1988e).

$$T_h := 10500.65 \text{ lbf}$$

$$T_{h_allowable} := 20000 \text{ lbf}$$

Since $T_h < T_{h_allowable}$, this is acceptable

$$\text{Margin_of_Safety} := \left(\frac{T_{h_allowable}}{T_h} - 1 \right)$$

$$\text{Margin_of_Safety} = 0.9$$

The .25 inch fillet weld runs the full length of the chocking on all four sides. The corner welds and spot welds, which would add additional strength and support, have been neglected. The lateral and longitudinal fillet welds are analyzed separately to insure that they provide sufficient strength.

Lateral Fillet Weld Analysis:

	Dist. to cg of weld	Length of weld leg	Length of weld
Obtained from drawing H-3-56435 (PNL, 1988e)	$y := 77.5 \text{ in}$	$h := .25 \text{ in}$	$d := 155 \text{ in}$

Weld throat area: $A := .707 \cdot h \cdot d$
 $A = 27.4 \text{ in}^2$

Total shear stress
on each fillet weld
due to axial stress
and bending stress

$$\tau := \frac{.5 \cdot W_{\text{total}}}{A}$$

$$\tau = 1.68 \cdot 10^3 \text{ psi}$$

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The allowable shear stress is assumed to be .3 times the yield strength of the base metal (A-36 steel). This procedure is outlined in the *Structural Welding Code - Steel* (AWS, 1988).

$$\tau_{\text{allowable}} := 10800 \text{ psi}$$

$$\text{Margin_of_Safety} := \left(\frac{\tau_{\text{allowable}}}{\tau} - 1 \right)$$

$$\text{Margin_of_Safety} = 5.43$$

Since $\tau < \tau_{\text{allowable}}$, this is acceptable

Longitudinal Fillet Weld Analysis:

	Dist. to cg of weld	Length of weld leg	Length of weld
Obtained from drawing H-3-56435 (PNL, 1988e)	$y := 77.5 \text{ in}$	$h := .25 \text{ in}$	$d := 112 \text{ in}$

Weld throat area:

$$A := .707 \cdot h \cdot d$$

$$A = 19.8 \text{ in}^2$$

Total shear stress
on each fillet weld
due to axial stress
and bending stress

$$\tau := \frac{.8 \cdot W_{\text{total}}}{A}$$

$$\tau = 3.72 \cdot 10^3 \text{ psi}$$

The allowable shear stress is assumed to be .3 times the yield strength of the base metal (A-36 steel). This procedure is outlined in the *Structural Welding Code - Steel* (AWS, 1988).

$$\tau_{\text{allowable}} := 10800 \text{ psi}$$

$$\text{Margin_of_Safety} := \left(\frac{\tau_{\text{allowable}}}{\tau} - 1 \right)$$

$$\text{Margin_of_Safety} = 1.9$$

Since $\tau < \tau_{\text{allowable}}$, this is acceptable

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Hex Bolt Analysis

The 5/8 inch hex socket cap screw, lock washer, flat washer, and hex nuts are assumed to have a yield strength of 30 ksi. This was obtained from the *Annual Book of Standards* (ASTM, 1994 vol 15.08). The allowable shear strength is assumed to be the yield strength multiplied by 0.6. This is specified in *Mechanical Engineering Design* (Shigley, 1989). There are four cap screws in each safety hoist tiedown. The maximum load taken by the safety hoist tiedown is 10,500 lbf. This force is assumed to be distributed between the four fasteners.

Fastener yield strength in tension: $\sigma_{\text{allowable}} := 30000 \text{ psi}$

Fastener yield strength in shear: $\tau_{\text{allowable}} := 18000 \text{ psi}$

Force per fastener: $F_{\text{per fastener}} := \frac{10500 \text{ lbf}}{4}$

$$F_{\text{per fastener}} = 2.63 \cdot 10^3 \text{ lbf}$$

Cross-sectional area: $A_c := \pi \cdot .3125^2 \text{ in}^2$

$$A_c = 0.31 \text{ in}^2$$

Figure 6. Hex Bolts.

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The total vertical load on the fasteners is the vertical component plus the added force caused by the moment. The total vertical load is greatest on fasteners 1 and 2 because the direction of the vertical force and moment are the same. The moment opposes the vertical load on fasteners 3 and 4.

Vertical component of force on one fastener:

$$F_{\text{vertical}} := F_{\text{per fastener}} \cdot \sin(45.5 \cdot \text{deg})$$

$$F_{\text{vertical}} = 1.87 \cdot 10^3 \cdot \text{lbf}$$

Distance from fastener to centroid of the bolt group:

$$r_1 := 1.2057 \cdot \text{in} \quad r_2 := 1.2057 \cdot \text{in} \quad r_3 := 1.2057 \cdot \text{in} \quad r_4 := 1.2057 \cdot \text{in}$$

$$M := F_{\text{per fastener}} \cdot \sin(45.5 \cdot \text{deg}) \cdot 1.031 \cdot \text{in}$$

Force due to the moment load:

$$F_{\text{moment}} := \frac{M \cdot r_1}{r_1^2 + r_2^2 + r_3^2 + r_4^2}$$

$$F_{\text{moment}} = 400.25 \cdot \text{lbf}$$

Normal stress of fastener 1:

$$\sigma_{\text{tension}} := \frac{F_{\text{vertical}} + F_{\text{moment}}}{A_c}$$

$$\sigma_{\text{tension}} = 7.41 \cdot 10^3 \cdot \text{psi}$$

$$\text{Margin_of_Safety} := \left(\frac{\sigma_{\text{allowable}}}{\sigma_{\text{tension}}} - 1 \right)$$

$$\text{Margin_of_Safety} = 3.05$$

Since $\sigma_{\text{tension}} < \sigma_{\text{allowable}}$, this is acceptable

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The shearing force is the horizontal component of the total force on one fastener.
 It is the same across all 4 hex cap screws, therefore, the shearing stress is the same.

Shear stress of one fastener:

$$\tau := \frac{F_{\text{per_fastener}} \cdot \cos(45.5 \cdot \text{deg})}{A_c}$$

$$\tau = 6 \cdot 10^3 \text{ psi}$$

$$\text{Margin_of_Safety} := \left(\frac{\tau_{\text{allowable}}}{\tau} - 1 \right)$$

$$\text{Margin_of_Safety} = 2$$

Since $\tau < \tau_{\text{allowable}}$, this is acceptable

The horizontal component of total maximum load on one fastener causes bearing stress.
 The rectangular bearing stress area is the the height of the hoist multiplied by the diameter of the fastener.

Height of the tiedown hoist: $h := 1.875 \text{ in}$

Diameter of the fastener: $d := .625 \text{ in}$

Bearing surface area of one fastener:

$$A_b := h \cdot d$$

$$A_b = 1.17 \text{ in}^2$$

Bearing stress of one fastener:

$$\sigma_{\text{bearing}} := \frac{F_{\text{per_fastener}} \cdot \cos(45.5 \cdot \text{deg})}{A_b}$$

$$\sigma_{\text{bearing}} = 1.57 \cdot 10^3 \text{ psi}$$

$$\text{Margin_of_Safety} := \left(\frac{\sigma_{\text{allowable}}}{\sigma_{\text{bearing}}} - 1 \right)$$

$$\text{Margin_of_Safety} = 18.11$$

Since $\sigma_{\text{bearing}} < \sigma_{\text{allowable}}$, this is acceptable