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(G) Reason	(H) Disp.	(J) Name	(K) Signature	(L) Date	(M) MSIN	(G) Reason	(H) Disp.	(J) Name	(K) Signature	(L) Date	(M) MSIN
		Design Authority				1	1	TA Dillhoff	[signature]	8/27/97	N2-57
		Design Agent				1	1	PC Ferrell	PC Ferrell	8/26/97	H1-15
1	1	Cog.Eng. WA McCormick	[signature]	8/26/97	H1-15	1	1	JG McFadden	JG McFadden	8/26/97	H1-15
1	1	Cog. Mgr. JG Field	[signature]	8/26/97	H1-15						
1	1	QA CR Hoover	[signature]	8/21/97	H1-15						
1	1	Safety DW McNally	DW McNally	8-26-97	H1-15						
1	1	RL Clawson	RL Clawson	8-26-97	H1-14						

18.	19.	20.	21. DOE APPROVAL (if required)
W. A. McCormick [signature] 8/26/97 Signature of EDT Originator / Date	Authorized Representative for Receiving Organization / Date	J. G. Field [signature] 8/26/97 Design Authority/ Cognizant Manager / Date	Ctrl. No. [] Approved [] Approved w/comments [] Disapproved w/comments

BD-7400-172-2 (05/96) GEF097

BD-7400-172-1

Safety Evaluation for Packaging (Onsite) Depleted Uranium Waste Boxes

W. A. McCormick
Waste Management Federal Services, Inc., Northwest Operations
Richland, Washington 99352
U.S. Department of Energy Contract DE-AC06-96RL13200

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Key Words: Fast Flux Test Facility, FFTF, 200 West Area, transport, steel boxes, wooden box, LSA quantities, Type B

Abstract: This safety evaluation for packaging (SEP) allows the one-time shipment of ten metal boxes and one wooden box containing depleted uranium material from the Fast Flux Test Facility to the burial grounds in the 200 West Area for disposal. This SEP provides the analyses and operational controls necessary to demonstrate that the shipment will be safe for the onsite worker and the public.

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LIST OF TERMS

ANSI	American National Standards Institute
ARF	airborne release fraction
ASME	American Society of Mechanical Engineers
Bq	becquerel
Bq/cm^2	becquerels per square centimeter
Ci	curie
cm	centimeter
cm^2	square centimeter
DOT	U.S. Department of Transportation
dpm/cm^2	disintegrations per minute per square centimeter
DU	depleted uranium
EDE	effective dose equivalent
FFTF	Fast Flux Test Facility
g/cm^3	grams per cubic centimeter
Gy	gray
Hz	hertz
IAEA	International Atomic Energy Agency
in.	inch
kg	kilogram
km	kilometer
km/h	kilometers per hour
kPa	kilopascal
lb	pound
LSA	low specific activity
m	meter
m^2	square meter
MeV	megaelectronvolt
$\mu Ci/cm^2$	microcuries per square centimeter
mi	mile
mi/h	miles per hour
mrem/h	millirem per hour
NTC	normal transport conditions
psi	pounds per square inch
psia	pounds per square inch, absolute
QA	quality assurance
QL	Quality Level
rem/h	rem per hour
RF	respirable fraction
SEP	safety evaluation for packaging
s/m^3	seconds per cubic meter
Sv	sievert
THI	Transportation Hazard Index
W	watt
W/Ci	watts per curie
yr	year

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SAFETY EVALUATION FOR PACKAGING (ONSITE) DEPLETED URANIUM WASTE BOXES

PART A: DESCRIPTION AND OPERATIONS

1.0 INTRODUCTION

1.1 GENERAL INFORMATION

The Fast Flux Test Facility has 11 boxes stored in a secured storage facility that contain depleted uranium (DU) material. This material has been discovered to contain small amounts of other radioactive nuclides and daughter products. It has been determined that the material must be moved to the burial grounds for disposal.

The material is currently contained in ten steel boxes and one wooden box. From observation of the box markings and general construction, it is determined that the boxes are strong, tight containers originally used to ship up to low specific activity (LSA) quantities of radioactive material (49 CFR 173). Laboratory analysis of smears on the material have determined that there is enough contamination in the uranium material to categorize each box quantity of radioactive material as Type B. The primary factor for this categorization is the presence of small quantities of ^{239}Pu, whose A_2 limit is 5.41×10^{-3} Ci. The largest amount of ^{239}Pu contained in any one box is 3.08×10^{-1} Ci.

Because (1) the DU material is already packaged, (2) repackaging of the material will be costly and result in worker exposure, and (3) this will be a one-time-only shipment to the burial ground, where the material will be subsequently placed in a high-integrity container, it has been determined that a safety evaluation for packaging (SEP) be prepared in order to authorize the shipment of these boxes as they currently exist. This SEP provides the analyses and operational controls necessary to demonstrate that the shipment will be safe for the onsite worker and the public.

1.2 SYSTEM DESCRIPTION

1.2.1 Metal Boxes

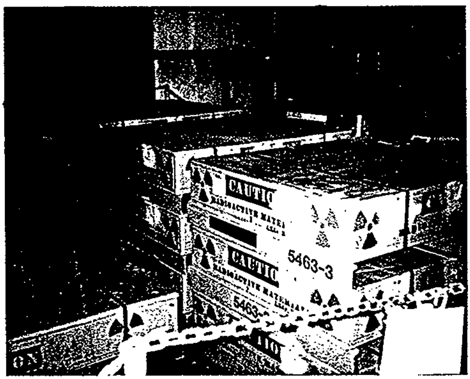
There are ten metal boxes. Nine of them are identical in size, shape, and construction. One is similar in construction and shape (5463-10); however, it is somewhat larger in size. Table A1-1 shows the container sizes and weights for all the boxes. Each of the metal containers is painted yellow with cyan radiation markings. The boxes are sealed with rubber gasket material and closed with hasps. In addition, each box is banded with two horizontal bands. There are forklift tine receivers welded to the bottom of each box as well. Figure A1-1 shows the boxes as they currently reside in the storage facility.

Table A1-1. Sizes and Weights.

Container No.	Size (length, width, height) cm (in.)	Tare weight kg (lb)	Gross weight kg (lb)
5463-1	114 x 135 x 34 (45 x 53 x 13.25)	197 (435)	2271 (5006)
5463-2	114 x 135 x 34 (45 x 53 x 13.25)	197 (435)	1957 (4314)
5462-3	114 x 135 x 34 (45 x 53 x 13.25)	197 (435)	2049 (4518)
5463-4	114 x 135 x 34 (45 x 53 x 13.25)	197 (435)	2693 (5936)
5463-5	114 x 135 x 34 (45 x 53 x 13.25)	197 (435)	2570 (5665)
5463-6	114 x 135 x 34 (45 x 53 x 13.25)	197 (435)	1697 (3741)
5463-7	114 x 135 x 34 (45 x 53 x 13.25)	197 (435)	1515 (3339)
5463-8	114 x 135 x 34 (45 x 53 x 13.25)	197 (435)	2336 (5149)
5463-9	114 x 135 x 34 (45 x 53 x 13.25)	197 (435)	2450 (5402)
5463-10	183 x 163 x 36 (72 x 64 x 14)	279 (615)	1876 (4135)
5463-11*	102 x 130 x 42 (40 x 51 x 16.5)	61 (135)	1858 (4097)

*Wooden box.

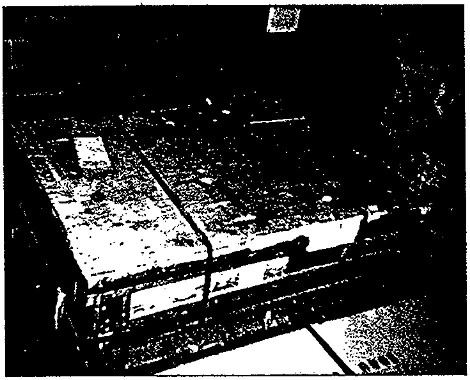
Figure A1-1. Metal Boxes As Currently Reside in Storage Facility.



1.2.2 Wooden Box

There is one wooden box (5463-11) containing the same type of material as the metal boxes. It is glued and nailed closed and contains black markings similar to the metal boxes. Additional banding has been placed on the box to further facilitate the lid seal. The bottom of the box has wooden boards attached to facilitate forklift handling. Table A1-1 shows the size and weight of the wooden box. Figure A1-2 depicts the box as it currently resides in the storage facility.

Figure A1-2. Wooden Box As Currently Resides in Storage Facility.



1.3 REVIEW AND UPDATE CYCLES

This SEP authorizes the one-time shipment of the identified payload. After completion of the shipment, the SEP will become void. No further reviews or updates will be required.

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2.0 PACKAGING SYSTEM

2.1 CONFIGURATION AND DIMENSIONS

The packaging dimensions are described in Part A, Section 1.0, of this SEP. From video taping of the inspection of several boxes while open, the DU material is braced within with sheet plywood and 2x4's.

2.2 MATERIALS OF CONSTRUCTION

Ten of the boxes are constructed of welded sheet metal and internal angle bracing. One of the boxes is constructed of heavy plywood approximately 2.54 cm (1.0 in.) thick (see figure A1-2).

2.3 MECHANICAL PROPERTIES OF MATERIALS

The mechanical properties of the materials cannot be readily established. Because the boxes are marked as LSA containers, by definition they meet the older U.S. Department of Transportation (DOT) definition of a strong, tight container, capable of containing its content mass and physical form without leaking.

2.4 DESIGN AND FABRICATION METHODS

The boxes were designed to meet the DOT definition of a strong, tight container. Since they were shipped to the Hanford Site on commercial roadways using DOT regulations for LSA quantities of radioactive material, it appears that the original manufacturer met the design and fabrication requirements for a strong, tight container.

2.5 WEIGHTS AND CENTERS OF GRAVITY

Package gross weights are given in Table A1-1. Based on observations of opened boxes, the center of gravity is in the approximate geometric center of the containers.

2.6 CONTAINMENT BOUNDARY

The containment boundary is normaly the DU metal itself; however, due to the age of the material, some surface oxidation has occurred. Therefore, the boxes are considered the containment boundary. The containment boundary for the metal boxes is the walls, lid, and rubber-gasketed seal. The containment boundary for the wooden boxes is the walls, lid and glue seal. The wooden box shall also be completely wrapped in plastic with a minimum 10-mil thickness and tape sealed prior to transport.

2.7 CAVITY SIZE

The cavity size for each box is shown in Table A2-1.

Table A2-1. Cavity Size.

Container No.	Cavity size (length, width, height) cm (in.)
5463-1	107 x 127 x 23.5 (42 x 50 x 9.25)
5463-2	107 x 127 x 23.5 (42 x 50 x 9.25)
5463-3	107 x 127 x 23.5 (42 x 50 x 9.25
5463-4	107 x 127 x 23.5 (42 x 50 x 9.25
5463-5	107 x 127 x 23.5 (42 x 50 x 9.25
5463-6	107 x 127 x 23.5 (42 x 50 x 9.25
5463-7	107 x 127 x 23.5 (42 x 50 x 9.25
5463-8	107 x 127 x 23.5 (42 x 50 x 9.25
5463-9	107 x 127 x 23.5 (42 x 50 x 9.25
5463-10	178 x 152 x 23 (70 x 60 x 9)
5463-11	102 x 127 x 30.5 (40 x 50 x 12)

2.8 HEAT DISSIPATION

Heat dissipation is achieved through passive thermal conduction and radiation. There are no artificial cooling mechanisms employed to dissipate payload decay heat. The highest heat output of any box payload is expected to be no more than 7.66×10^{-2} W, so decay heat is not a concern.

2.9 SHIELDING

No shielding is provided by the boxes other than the walls of the containers. Measured dose rates on surface contact of each container are shown in Table A2-2. The highest contact dose rate is 30.0 mrem/h, which is well within transportation limits.

Table A2-2. Dose Rates on Contact.

Container No.	Contact dose rate (mrem/h)
5463-1	4.0
5463-2	30.0
5463-3	4.0
5463-4	10.0
5463-5	3.0
5463-6	3.5
5463-7	3.0
5463-8	3.5
5463-9	3.5
5463-10	3.0
5463-11	4.0

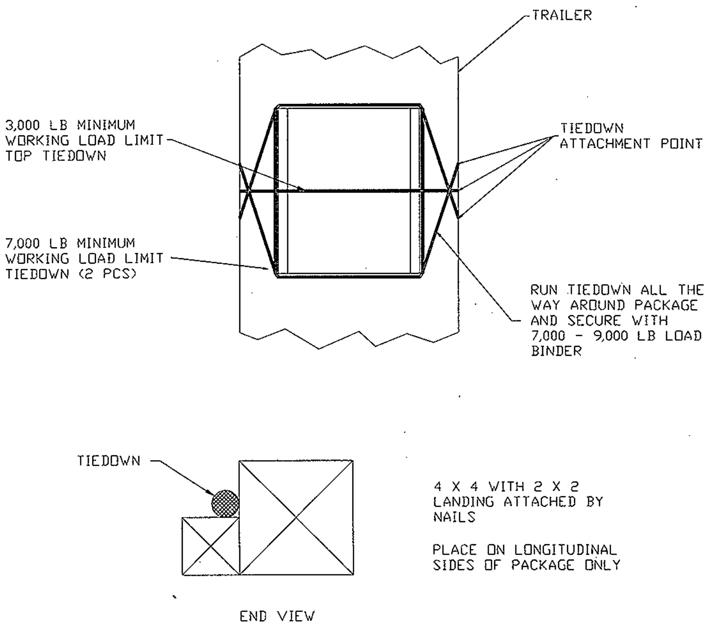
2.10 LIFTING DEVICES

The metal boxes feature forklift tine channels for lifting. The wood box is fitted with square lumber setoffs, which allow for forklift handling. There are no other lifting devices integral to the packagings.

2.11 TIEDOWN DEVICES

The are no tiedown devices integral to the metal or wooden boxes. Figure A2-1 shows the tiedown system.

Figure A2-1. Tiedown System.



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3.0 PACKAGE CONTENTS

3.1 GENERAL DESCRIPTION

Package contents consist of DU metal in the form of plates, discs, and blocks. In addition, wooden dunnage is used to support the metal in the void space of the containers.

3.2 CONTENT RESTRICTIONS

The contents are restricted to what is currently in the boxes for shipment to the burial ground.

3.2.1 Content Matrix

The content matrix consists almost entirely of DU, with small amounts of decay products and activated metal.

3.2.2 Radioactive Material

The radiological constituent source term for the worst-case box is shown in Table A3-1.

3.2.3 Nonradioactive Material

There is no nonradioactive material in the boxes other than the wood and paper used for dunnage and lining.

Table A3-1. Decay Heat and A_2 for the Depleted Uranium Waste Boxes, Ci.

Nuclide	W/Ci	A_2 (Ci)	Box 4		
			Ci	W	A_2s
^{60}Co	1.54 E-02	1.08 E+01	2.81 E-04	4.33 E-06	2.60 E-05
^{99}Tc	5.02 E-04	2.43 E+01	1.61 E-03	8.08 E-07	6.63 E-05
^{137}Cs	1.01 E-03	1.35 E+01	1.57 E-01	1.59 E-04	1.16 E-02
^{137m}Ba*	3.92 E-03	0.00	1.49 E-01	5.84 E-04	0.00
^{212}Bi*	1.68 E-02	0.00	2.44 E-02	4.10 E-04	0.00
^{212}Po*	5.21 E-02	0.00	4.27 E02	2.22 E-03	0.00
^{216}Po*	4.02 E-02	0.00	1.39 E-01	5.59 E-03	0.00
^{224}Ra*	3.37 E-02	0.00	2.10 E-01	7.08 E-03	0.00
^{220}Rn*	3.73 E-02	0.00	1.35 E-01	5.04 E-03	0.00
^{228}Th	3.21 E-02	1.08 E-02	2.07 E-01	6.64 E-03	1.92 E+01
^{234}Th	3.97 E-04	5.41 E+00	6.14 E-01	2.44 E-04	1.13 E-01
^{234}U	2.83 E-02	2.70 E-02	8.49 E-01	2.40 E-02	3.14 E+01
^{235}U	2.71 E-02	Unlimited	1.21 E-02	3.28 E-04	0.00
^{238}U	2.49 E-02	Unlimited	8.47 E-01	2.11 E-02	0.00
^{239}Pu	3.06 E-02	5.41 E-03	9.71 E-02	2.97 E-03	1.79 E+01
^{240}Pu	3.06 E-02	5.41 E-03	5.97 E-03	1.83 E-04	1.10 E+00
Total			3.49 E+00	7.66 E-02	6.97 E+01

*This radionuclide is a daughter as defined in 49 CFR 173.433; therefore, its activity was set to 0 for the A_2 calculations.

49 CFR 173.433, 1997, "Shippers--General Requirements for Shipments and Packagings," (.433) "Requirements for determination of A_{-1} and A_2 values for radionuclides," *Code of Federal Regulations*, as amended.

4.0 TRANSPORT SYSTEM

4.1 TRANSPORTER

The transporter shall consist of a flatbed, lowboy, or van tractor-trailer with a trailer cargo minimum capacity of 13,608 kg (30,000 lb).

4.2 TIEDOWN SYSTEM

The tiedown system shall meet the requirements of the 49 CFR 393, Subpart I. Blocking and bracing shall be used in addition to strapping to prevent lateral and longitudinal movement of the packages. Note that trailer stake-pockets, rub-rails, van sidings, and trailer racks are *not* considered acceptable tiedown attachment points unless they are specifically designed and certified to meet the regulatory requirements.

4.3 SPECIAL TRANSFER REQUIREMENTS

- The wooden LSA box shall be wrapped in plastic sheeting with a minimum 10-mil thickness and sealed with tape prior to vehicle movement.
- A road closure is required for this shipment.
- Transport during inclement weather (e.g., rain, snow, hail) is prohibited.
- The transport vehicles' speed shall not exceed 32 km/h (20 mi/h) inside the 200 West Area boundary.
- Permissible external contamination limits for the boxes are shown in Table A4-1.

Table A4-1. External Container Contamination Limits.

Contaminant	Maximum permissible limits		
	Bq/cm^2	$\mu Ci/cm^2$	dpm/cm^2
Beta and gamma emitters and low toxicity alpha emitters	0.4	10^{-5}	22
All other alpha-emitting radionuclides	0.04	10^{-6}	2.2

Source: 49 CFR 173.443, 1997, "Shippers--General Requirements for Shipments and Packagings, *Code of Federal Regulations*, as amended.

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5.0 ACCEPTANCE OF PACKAGING FOR USE

5.1 INSPECTION

Prior to shipment, the shipper shall inspect the package and transport system to ensure the following.

- Verify that all penetrations in the boxes are plugged and sealed.
- Verify that the container closure mechanisms are properly fixed in the closed position.
- Verify that the box banding is in place and in good condition.
- Verify that the wooden box is adequately enclosed in plastic sheeting and sealed with tape.
- Verify that the external contamination limits are not exceeded.
- Verify that the boxes are not stacked.
- Verify that the tiedown system is in place as described in Part B, Section 10.0.

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6.0 OPERATING REQUIREMENTS

6.1 GENERAL REQUIREMENTS

The following requirements, as a minimum, shall be completed prior to or during transport, as applicable.

- Prior to shipment, a road closure shall be arranged.
- The boxes shall be properly marked and labeled.
- The shipping papers shall be completed.
- The transporters shall be properly placarded.
- No more than six boxes shall be placed on any one trailer.
- The boxes shall be blocked and braced to prevent lateral and longitudinal movement (see Part B, Section 10.0).
- The boxes shall be strapped to prevent vertical movement (see Part B, Section 10.0).
- The transport vehicles shall not exceed 32 km/h (20 mi/h) within the 200 West Area boundary.

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7.0 QUALITY ASSURANCE REQUIREMENTS

7.1 INTRODUCTION

This section describes the quality assurance (QA) requirements for the maintenance and use of the DU packaging system. The format and requirements for use on the Hanford Site are taken from WHC-CM-4-2, *Quality Assurance Manual*, and WHC-CM-2-14, *Hazardous Material Packaging and Shipping*.

7.2 GENERAL REQUIREMENTS

These requirements apply to activities that could affect the quality of the components of the packaging. The DU boxes are classified per the WHC-CM-2-14 with a Transportation Hazard Index (THI) of 2.

A THI 2 packaging represents the second highest level of hazard for the contents. A packaging assigned a THI 2 must be capable of mitigating a release that could result in a potential dose consequence of between 0.5 rem and 25 rem at the Hanford Site boundary, or greater than 5 rem within the Site, if fully released.

Each THI contains a Quality Level (QL) designator, consisting of an alpha designator and a numeric designator. The alpha designator assigns the fabrication, testing, use, and QA for each item, component, or activity associated with the packaging. The numeric designator following the alpha designator is the assigned THI number for the packaging. The following are definitions and requirements for each DU packaging item, component, or activity.

Quality Level A-2: Critical impact on safety and associated functional requirements: items or components whose failure or malfunction could directly result in an unacceptable condition of containment or confinement, shielding, or nuclear criticality.

This QL refers to the containment boundary of the DU boxes. The box bodies are the containment boundary. The requirements for operations and maintenance shall comply with the requirements in Part B, Section 7.2. Preventive maintenance and inspection of components shall be performed prior to shipment by personnel qualified to applicable standards specified in the safety documentation.

Any procurement of items shall be from a supplier with an approved QA program in accordance with, or equivalent to, appropriate basic requirements and supplements of the American Society of Mechanical Engineers (ASME) NQA-1 (ASME 1989). QA procurement clauses shall be imposed, as applicable, to ensure product quality. Specific requirements to be developed by the Packaging Quality Assurance engineer and the Packaging Engineering cognizant engineer.

Quality Level B-2: Major impact on safety and associated functional requirements. Components or activities whose failure or malfunction could indirectly result in an unacceptable condition of containment or confinement, shielding, or nuclear criticality. An unsafe condition could result only if the failure of this item or subsystem occurred in conjunction with the failure of other items or subsystem in A-2 or this level.

This QL refers to the seals of the DU boxes. The requirements for fabrication, operation, and maintenance shall comply with the requirements identified in Part B, Section 7.2. Preventive maintenance and inspection of components shall be performed prior to shipment and/or periodically (not to exceed one year) by personnel qualified to applicable standards specified in the safety documentation.

Any procurement of items shall be from a supplier with an approved QA program in accordance with, or equivalent to, appropriate basic requirements and supplements of ASME NQA-1 (ASME 1989). QA procurement clauses shall be imposed, as applicable, to ensure product quality. Specific requirements to be developed by the Packaging Quality Assurance engineer and Packaging Engineering cognizant engineer.

Quality Level C-2: Minor impact on safety and associated functional requirements. Items or components whose failure or malfunction would not reduce packaging effectiveness and would not result in an unacceptable condition of containment or confinement, shielding, or nuclear criticality, regardless of other failure in A-2, B-2, or this level.

This QL refers to the tiedowns and tiedown attachments for the DU boxes. The requirements for fabrication, operation, and maintenance shall comply with the requirements identified in Part B, Section 7.2, or Waste Management Federal Services, Inc., Northwest Operations' or Seller's prepared requirements media.

Materials for fabrication are required to be specified to recognized industrial, national, or international standards.

Any procurement of items shall be from a supplier with an approved QA program in accordance with, or equivalent to, appropriate basic requirements and supplements of ASME NQA-1 (ASME 1989). QA procurement clauses shall be imposed, as applicable, to ensure product quality. Specific requirements are to be developed by the Packaging Quality Assurance engineer and the Packaging Engineering cognizant engineer.

Documentation and review requirements are based upon the QL of each component or activity. Changes or discoveries of noncompliance for all QL A-2 and B-2 components and activities shall be reviewed by the unreviewed safety question screening process to ensure the quality and safety of the change or discovery. Changes to the SEP safety bases (contents, shielding, structural, containment, criticality) will require unreviewed safety question screening regardless of QL.

7.3 QUALITY REQUIREMENTS

Appropriate QA measures shall be used in the maintenance and use of the DU boxes.

7.3.1 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 the work and that quality achievement is verified by persons or organizations not directly responsible for performing the work.

Packaging Engineering, Loading Facility Operations, Radiological Protection managers, and Receiving Facility managers are responsible for the quality of the work performed by their respective organizations and for performing the following activities:

- Follow current requirements of this SEP
- Provide instructions for implementing QA requirements.

The Quality Assurance cognizant manager is responsible for establishing and administering the Hanford Site QA program, as stated in the WHC-CM-4-2, relative to the DU boxes.

7.3.2 Design Control

Not applicable.

7.3.3 Procurement and Fabrication Control

Not applicable.

7.3.4 Control of Inspection

Not applicable.

7.3.5 Control of Nonconforming Items

Not applicable.

7.3.6 QA Records and Document Control

Records that furnish documentary evidence of quality shall be specified, prepared, and maintained. All documents used to perform and/or verify quality-related activities are controlled. Controlled documents include (but are not limited to) the following: drawings, specifications, purchase orders, plans and procedures to inspect and test, reports, the SEP, 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 records associated with hazardous material packaging and transportation shall be retained for the life of the packaging. All lifetime storage records required for the DU boxes shall be stored with either Packaging Engineering or the user facility's engineering files, depending upon the purpose of the document. For records retention periods and location of records for the DU boxes (when used per this onsite SEP), see Table A7-1.

7.3.7 Audits

The following are possible activities and files to be audited during the use and maintenance of DU boxes and effective packaging components:

- Safety analysis records
- Operating procedures and acceptance and inspection records.

Table A7-1. Records Retention and Location.

Document	Retention period	Location
Safety analysis report for packaging	Lifetime	Waste Management Federal Services Inc., Northwest Operations, Packaging Engineering/Information Resource Management
Radiation surveys	5 years	User facility
Operating procedures	5 years	User facility
Quality assurance audits	Lifetime	User facility
Quality control inspection reports	Lifetime	User facility

7.3.8 Handling, Storage, and Shipping

Instructions for the handling, storage, and shipping are found in this SEP and the manufacturer's operating procedures for the trailers and the DU boxes. These requirements shall be implemented by the loading and unloading facility within operating procedures and maintenance procedures. The transport will be implemented by a radioactive shipment record for onsite transport of Type B material.

Special handling tools or lifting equipment (e.g., lift beams, straps, pins) for the DU boxes shall be used and controlled, as necessary, to ensure safe and adequate handling. Special handling tools and equipment shall be inspected and tested in accordance with the *Hanford Site Hoisting and Rigging Manual* (RL 1996) at specified time intervals to verify that the tools and equipment are adequately maintained. Operators of special handling and lifting equipment shall be experienced or trained in the use of the equipment.

Marking, labeling, and transport vehicle placarding for transport of the DU boxes shall be performed per the WHC-CM-2-14. Marking and labeling of the package for storage shall be maintained by storage maintenance procedures.

7.4 QA ACTIVITIES

Each cognizant engineer involved with use or maintenance of the DU boxes or related equipment is responsible for ensuring that the assigned tasks are performed in accordance with controlling plans and procedures, which must, in turn, conform to these QA requirements. Quality requirements for tasks are determined and documented in the plans and procedures used by the involved organizations.

The appropriate QL designators for components of the DU boxes are shown in Table A7-2.

Table A7-2. Quality Levels.

Equipment item or component	Quality Level
Depleted uranium box bodies	A-2
Seals (where used) of depleted uranium boxes	B-2
Tiedowns for depleted uranium boxes	C-2
Transporter tiedown attachments	C-2

7.5 SEP CONTROL SYSTEM

This SEP is a controlled supporting document. Only up-to-date approved versions of this SEP are used for transport. Any changes made to this SEP will be performed by Packaging Engineering and are incorporated and distributed to users through the Copy Control System.

Any review comment records produced during the initial release or subsequent changes will be on file with Packaging Engineering.

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8.0 MAINTENANCE

8.1 GENERAL REQUIREMENTS

The DU boxes will be shipped only once. Refer to Part A, Section 5.0, and Part A, Section 6.0, for inspection requirements prior to use.

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9.0 REFERENCES

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

49 CFR 393, 1997,"Parts and Accessories Necessary for Safe Operation," *Code of Federal Regulations*, as amended.

ASME, 1989, *Quality Assurance Program Requirements for Nuclear Facilities*, ASME NQA-1-1989 Edition, American Society of Mechanical Engineers, New York, New York.

RL, 1996, *Hanford Site Hoisting and Rigging Manual*, DOE/RL-92-36, U.S. Department of Energy, Richland Operations Office, Richland, Washington.

WHC-CM-2-14, *Hazardous Material Packaging and Shipping*, Westinghouse Hanford Company, Richland, Washington.

WHC-CM-4-2, *Quality Assurance Manual*, Westinghouse Hanford Company, Richland, Washington.

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PART B: PACKAGE EVALUATION

1.0 INTRODUCTION

Part B of this safety evaluation for packaging (SEP) contains evaluations necessary to show that the depleted uranium (DU) waste boxes currently stored at the Fast Flux Test Facility (FFTF) can be safely transported to the 200 West Area burial ground. The boxes have been discovered to contain small quantities of radionuclides other than DU, which has caused the quantity of material present to be categorized as Type B (49 CFR 173).

The box markings indicate that they were originally intended for the transport of low specific activity (LSA) radioactive material. The vintage of the boxes indicates that they were designed and fabricated as strong, tight containers (49 CFR 173). Due to time constraints, as low as reasonably achievable issues, and economic concerns, repackaging the material in onsite Type B packagings is not a desirable option.

Review of the actual quantity and type of material present, the physical form of the material, and the present condition of the boxes indicates that a one-time shipment of the packages, as they currently exist, is a reasonable and safe alternative.

1.1 SAFETY EVALUATION METHODOLOGY

The safety evaluation methodology for the DU waste boxes consists of two parts. The first part is a discussion of the ability of the containers to maintain containment, shielding, and nuclear criticality control during normal transport conditions (NTC). The second part consists of dose consequence and radiological risk analyses for onsite accident conditions.

For NTC, the discussion will be limited to the actual shipping scenario involved with this one-time shipment, rather than a generic analysis of the package as would normally be the case for routine or unlimited use of the containers.

For accident conditions, formal radiological risk and dose consequence analyses are performed to show that the payload can be transported without exceeding Hanford radiological exposure limits, given the comparative risk involved with this one-time shipping campaign.

1.2 EVALUATION SUMMARY AND CONCLUSIONS

Based on the evaluations in Part B of this SEP, the one-time transport of the contaminated DU in its existing containers is acceptable. The following itemized restrictions apply to this shipment.

- A road closure for south of the Wye Barricade restricting access to unauthorized personal shall be imposed.
- Transport vehicles' speed is limited to 32 km/h (20 mi/h) maximum inside the 200 West Area boundary.

1.3 REFERENCE

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

2.0 CONTENTS EVALUATION

2.1 CHARACTERIZATION

The contents to be transported in the 11 DU waste boxes shall consist of DU in the form of metal plates. The DU material is contaminated with greater-than-Type A quantities of radionuclides other than uranium. However, the contamination is largely nonremovable and nondispersible. The contents shall be limited to the maximum allowable source term shown in Table B2-1. Table B2-1 shows the radioactive inventory for each box along with the total inventory of each radionuclide for all 11 boxes (see sheet 4).

Table B2-1. Depleted Uranium Waste Boxes Inventory, Ci. (sheet 1 of 4)

Nuclide	Box 1		Box 2		Box 3	
	Bq	Ci	Bq	Ci	Bq	Ci
^{60}Co	8.95 E+07	2.42 E-03	7.33 E+06	1.98 E-04	7.40 E+07	2.00 E-03
^{99}Tc	3.81 E+05	1.03 E-05	4.18 E+07	1.13 E-03	3.40 E+05	9.18 E-06
^{137}Cs	8.62 E+07	2.33 E-03	4.11 E+09	1.11 E-01	7.40 E+07	2.00 E-03
^{137m}Ba	8.14 E+07	2.20 E-03	3.88 E+09	1.05 E-01	6.99 E+07	1.89 E-03
^{212}Bi	1.17 E+08	3.17 E-03	1.41 E+09	3.80 E-02	1.04 E+08	2.80 E-03
^{212}Po	1.17 E+08	3.17 E-03	1.11 E+09	3.01 E-02	1.04 E+08	2.80 E-03
^{216}Po	2.31 E+08	6.25 E-03	3.63 E+09	9.82 E-02	2.07 E+08	5.60 E-03
^{224}Ra	2.42 E+08	6.53 E-03	5.48 E+09	1.48 E-01	2.15 E+08	5.80 E-03
^{220}Rn	2.28 E+08	6.16 E-03	3.51 E+09	9.50 E-02	2.03 E+08	5.50 E-03
^{228}Th	2.28 E+08	6.16 E-03	5.40 E+09	1.46 E-01	2.03 E+08	5.50 E-03
^{234}Th	2.21 E+10	5.96 E-01	1.60 E+10	4.33 E-01	1.97 E+10	5.32 E-01
^{234}U	2.62 E+10	7.07 E-01	2.20 E+10	5.95 E-01	2.34 E+10	6.32 E-01
^{235}U	3.70 E+08	1.00 E-02	3.15 E+08	8.51 E-03	3.31 E+08	8.95 E-03
^{238}U	2.60 E+10	7.04 E-01	2.20 E+10	5.95 E-01	2.33 E+10	6.29 E-01
^{239}Pu	3.10 E+08	8.39 E-03	2.53 E+09	6.85 E-02	2.77 E+08	7.49 E-03
^{240}Pu	1.91 E+07	5.15 E-04	1.56 E+08	4.21 E-03	1.70 E+07	4.60 E-04
Total	7.64 E+10	2.06 E+00	9.16 E+10	2.48 E+00	6.83 E+10	1.84 E+00

Table B2-1. Depleted Uranium Waste Boxes Inventory, Ci. (sheet 2 of 4)

Nuclide	Box 4		Box 5		Box 6	
	Bq	Ci	Bq	Ci	Bq	Ci
^{60}Co	1.04 E+07	2.81 E-04	1.04 E+08	2.80 E-03	6.29 E+07	1.70 E-03
^{99}Tc	5.96 E+07	1.61 E-03	4.37 E+05	1.18 E-05	2.75 E+05	7.43 E-06
^{137}Cs	5.81 E+09	1.57 E-01	9.99 E+07	2.70 E-03	6.29 E+07	1.70 E-03
^{137m}Ba	5.51 E+09	1.49 E-01	9.43 E+07	2.55 E-03	5.96 E+07	1.61 E-03
^{212}Bi	9.03 E+08	2.44 E-02	1.33 E+08	3.60 E-03	8.51 E+07	2.30 E-03
^{212}Po	1.58 E+09	4.27 E-02	1.33 E+08	3.60 E-03	8.51 E+07	2.30 E-03
^{216}Po	5.14 E+09	1.39 E-01	2.59 E+08	7.00 E-03	1.66 E+08	4.50 E-03
^{224}Ra	7.77 E+09	2.10 E-01	2.77 E+08	7.50 E-03	1.74 E+08	4.70 E-03
^{220}Rn	5.00 E+09	1.35 E-01	2.59 E+08	7.00 E-03	1.63 E+08	4.40 E-03
^{228}Th	7.66 E+09	2.07 E-01	2.59 E+08	7.00 E-03	1.63 E+08	4.40 E-03
^{234}Th	2.27 E+10	6.14 E-01	2.52 E+10	6.82 E-01	1.59 E+10	4.31 E-01
^{234}U	3.14 E+10	8.49 E-01	2.98 E+10	8.06 E-01	1.88 E+10	5.08 E-01
^{235}U	4.48 E+08	1.21 E-02	4.25 E+08	1.15 E-02	2.69 E+08	7.26 E-03
^{238}U	3.13 E+10	8.47 E-01	2.98 E+10	8.06 E-01	1.89 E+10	5.10 E-01
^{239}Pu	3.59 E+09	9.71 E-02	3.46 E+08	9.36 E-03	2.22 E+08	5.99 E-03
^{240}Pu	2.21 E+08	5.97 E-03	2.13 E+07	5.75 E-04	1.36 E+07	3.68 E-04
Total	1.29 E+11	3.49 E+00	8.72 E+10	2.36 E+00	5.51 E+10	1.49 E+00

Table B2-1. Depleted Uranium Waste Boxes Inventory, Ci. (sheet 3 of 4)

Nuclide	Box 7		Box 8		Box 9	
	Bq	Ci	Bq	Ci	Bq	Ci
^{60}Co	5.70 E+07	1.54 E-03	9.25 E+07	2.50 E-03	9.73 E+07	2.63 E-03
^{99}Tc	2.42 E+05	6.53 E-06	3.92 E+05	1.06 E-05	4.14 E+05	1.12 E-05
^{137}Cs	5.48 E+07	1.48 E-03	8.88 E+07	2.40 E-03	9.36 E+07	2.53 E-03
^{137m}Ba	5.18 E+07	1.40 E-03	8.40 E+07	2.27 E-03	8.84 E+07	2.39 E-03
^{212}Bi	7.40 E+07	2.00 E-03	1.21 E+08	3.27 E-03	1.28 E+08	3.45 E-03
^{212}Po	7.40 E+07	2.00 E-03	1.21 E+08	3.27 E-03	1.28 E+08	3.45 E-03
^{216}Po	1.47 E+08	3.97 E-03	2.39 E+08	6.45 E-03	2.51 E+08	6.79 E-03
^{224}Ra	1.54 E+08	4.15 E-03	2.49 E+08	6.74 E-03	2.63 E+08	7.10 E-03
^{220}Rn	1.45 E+08	3.91 E-03	2.35 E+08	6.35 E-03	2.48 E+08	6.69 E-03
^{228}Th	1.45 E+08	3.91 E-03	2.35 E+08	6.35 E-03	2.48 E+08	6.69 E-03
^{234}Th	1.40 E+10	3.78 E-01	2.27 E+10	6.14 E-01	2.39 E+10	6.47 E-01
^{234}U	1.65 E+10	4.46 E-01	2.68 E+10	7.25 E-01	2.82 E+10	7.63 E-01
^{235}U	2.36 E+08	6.38 E-03	3.81 E+08	1.03 E-02	4.03 E+08	1.09 E-02
^{238}U	1.65 E+10	4.45 E-01	2.68 E+10	7.24 E-01	2.83 E+10	7.65 E-01
^{239}Pu	1.97 E+08	5.33 E-03	3.20 E+08	8.65 E-03	3.37 E+09	9.12 E-02
^{240}Pu	1.21 E+07	3.27 E-04	1.96 E+07	5.31 E-04	2.07 E+08	5.60 E-03
Total	4.83 E+10	1.31 E+00	7.85 E+10	2.12 E+00	8.59 E+10	2.32 E+00

Table B2-1. Depleted Uranium Waste Boxes Inventory, Ci. (sheet 4 of 4)

Nuclide	Box 10		Box 11		Total - All Boxes	
	Bq	Ci	Bq	Ci	Bq	Ci
^{60}Co	6.92 E+07	1.87 E-03	7.77 E+07	2.10 E-03	7.40 E+08	2.00 E-02
^{99}Tc	2.93 E+05	7.91 E-06	3.29 E+05	8.90 E-06	1.04 E+08	2.82 E-03
^{137}Cs	6.62 E+07	1.79 E-03	7.40 E+07	2.00 E-03	1.06 E+10	2.87 E-01
^{137m}Ba	6.25 E+07	1.69 E-03	6.99 E+07	1.89 E-03	1.01 E+10	2.72 E-01
^{212}Bi	9.03 E+07	2.44 E-03	1.02 E+08	2.75 E-03	3.26 E+09	8.82 E-02
^{212}Po	9.03 E+07	2.44 E-03	1.02 E+08	2.75 E-03	3.65 E+09	9.86 E-02
^{216}Po	1.78 E+08	4.82 E-03	2.01 E+08	5.42 E-03	1.07 E+10	2.88 E-01
^{224}Ra	1.85 E+08	5.00 E-03	2.09 E+08	5.66 E-03	1.52 E+10	4.11 E-01
^{220}Rn	1.75 E+08	4.74 E-03	1.98 E+08	5.34 E-03	1.04 E+10	2.80 E-01
^{228}Th	1.75 E+08	4.74 E-03	1.98 E+08	5.34 E-03	1.49 E+10	4.02 E-01
^{234}Th	1.70 E+10	4.59 E-01	1.91 E+10	5.16 E-01	2.18 E+11	5.90 E +00
^{234}U	1.99 E+10	5.39 E-01	2.25 E+10	6.08 E-01	2.66 E+11	7.18 E +00
^{235}U	2.86 E+08	7.72 E-03	3.22 E+08	8.69 E-03	3.77 E+09	1.02 E-01
^{238}U	2.00 E+10	5.41 E-01	2.25 E+10	6.09 E-01	2.65 E+11	7.17 E+00
^{239}Pu	2.39 E+08	6.46 E-03	2.67 E+08	7.21 E-03	1.17 E+10	3.16 E-01
^{240}Pu	1.47 E+07	3.97 E-04	1.64 E+07	4.43 E-04	7.18 E+08	1.94 E-02
Total	5.85 E+10	1.58 E+00	6.59 E+10	1.78 E+00	8.45 E+11	2.28 E+01

The maximum number of A_2s and thermal characteristics of the contents are determined as shown in Table B2-2. Note that all 11 boxes contain Type B quantities of radioactive material, with some just slightly above Type A quantities, and the maximum of 69.7 A_2s being contained in Box 4. The maximum heat production of 0.0766 W also occurs for Box 4.

Table B2-2. Decay Heat and A_2 for the Depleted Uranium Waste Boxes, Ci. (sheet 1 of 6)

Nuclide	W/Ci	A_2 (Ci)	Box 1			Box 2		
			Ci	W	A_2s	Ci	W	A_2s
^{60}Co	1.54 E-02	1.08 E+01	2.42 E-03	3.73 E-05	2.24 E-04	1.98 E-04	3.05 E-06	1.83 E-05
^{99}Tc	5.02 E-04	2.43 E+01	1.03 E-05	5.17 E-09	4.24 E-07	1.13 E-03	5.67 E-07	4.65 E-05
^{137}Cs	1.01 E-03	1.35 E+01	2.33 E-03	2.35 E-06	1.73 E-04	1.11 E-01	1.12 E-04	8.22 E-03
^{137m}Ba*	3.92 E-03	0.00	2.20 E-03	8.62 E-06	0.00 E+00	1.05 E-01	4.12 E-04	0.00 E+00
^{212}Bi*	1.68 E-02	0.00	3.17 E-03	5.33 E-05	0.00 E+00	3.80 E-02	6.38 E-04	0.00 E+00
^{212}Po*	5.21 E-02	0.00	3.17 E-03	1.65 E-04	0.00 E+00	3.01 E-02	1.57 E-03	0.00 E+00
^{216}Po*	4.02 E-02	0.00	6.25 E-03	2.51 E-04	0.00 E+00	9.82 E-02	3.95 E-03	0.00 E+00
^{224}Ra*	3.37 E-02	0.00	6.53 E-03	2.20 E-04	0.00 E+00	1.48 E-01	4.99 E-03	0.00 E+00
^{220}Rn*	3.73 E-02	0.00	6.16 E-03	2.30 E-04	0.00 E+00	9.50 E-02	3.54 E-03	0.00 E+00
^{228}Th	3.21 E-02	1.08 E-02	6.16 E-03	1.98 E-04	5.70 E-01	1.46 E-01	4.69 E-03	1.35 E+01
^{234}Th	3.97 E-04	5.41 E+00	5.96 E-01	2.37 E-04	1.10 E-01	4.33 E-01	1.72 E-04	8.00 E-02
^{234}U	2.83 E-02	2.70 E-02	7.07 E-01	2.00 E-02	2.62 E+01	5.95 E-01	1.68 E-02	2.20 E+01
^{235}U	2.71 E-02	Unlimited	1.00 E-02	2.71 E-04	0.00 E+00	8.51 E-03	2.31 E-04	0.00 E+00
^{238}U	2.49 E-02	Unlimited	7.04 E-01	1.75 E-02	0.00 E+00	5.95 E-01	1.48 E-02	0.00 E+00
^{239}Pu	3.06 E-02	5.41 E-03	8.39 E-03	2.57 E-04	1.55 E+00	6.85 E-02	2.10 E-03	1.27 E+01
^{240}Pu	3.06 E-02	5.41 E-03	5.15 E-04	1.58 E-05	9.52 E-02	4.21 E-03	1.29 E-04	7.78 E-01
Total			2.06 E+00	3.94 E-02	2.85 E+01	2.48 E+00	5.41 E-02	4.91 E+01

Table B2-2. Decay Heat and A_2 for the Depleted Uranium Waste Boxes, Ci. (sheet 2 of 6)

Nuclide	W/Ci	A_2 (Ci)	Box 3			Box 4		
			Ci	W	A_2s	Ci	W	A_2s
^{60}Co	1.54 E-02	1.08 E+01	2.00 E-03	3.08 E-05	1.85 E-04	2.81 E-04	4.33 E-06	2.60 E-05
^{99}Tc	5.02 E-04	2.43 E+01	9.18 E-06	4.61 E-09	3.78 E-07	1.61 E-03	8.08 E-07	6.63 E-05
^{137}Cs	1.01 E-03	1.35 E+01	2.00 E-03	2.02 E-06	1.48 E-04	1.57 E-01	1.59 E-04	1.16 E-02
^{137m}Ba*	3.92 E-03	0.00	1.89 E-03	7.41 E-06	0.00 E+00	1.49 E-01	5.84 E-04	0.00 E+00
^{212}Bi*	1.68 E-02	0.00	2.80 E-03	4.70 E-05	0.00 E+00	2.44 E-02	4.10 E-04	0.00 E+00
^{212}Po*	5.21 E-02	0.00	2.80 E-03	1.46 E-04	0.00 E+00	4.27 E-02	2.22 E-03	0.00 E+00
^{216}Po*	4.02 E-02	0.00	5.60 E-03	2.25 E-04	0.00 E+00	1.39 E-01	5.59 E-03	0.00 E+00
^{224}Ra*	3.37 E-02	0.00	5.80 E-03	1.95 E-04	0.00 E+00	2.10 E-01	7.08 E-03	0.00 E+00
^{220}Rn*	3.73 E-02	0.00	5.50 E-03	2.05 E-04	0.00 E+00	1.35 E-01	5.04 E-03	0.00 E+00
^{228}Th	3.21 E-02	1.08 E-02	5.50 E-03	1.77 E-04	5.09 E-01	2.07 E-01	6.64 E-03	1.92 E+01
^{234}Th	3.97 E-04	5.41 E+00	5.32 E-01	2.11 E-04	9.83 E-02	6.14 E-01	2.44 E-04	1.13 E-01
^{234}U	2.83 E-02	2.70 E-02	6.32 E-01	1.79 E-02	2.34 E+01	8.49 E-01	2.40 E-02	3.14 E+01
^{235}U	2.71 E-02	Unlimited	8.95 E-03	2.43 E-04	0.00 E+00	1.21 E-02	3.28 E-04	0.00 E+00
^{238}U	2.49 E-02	Unlimited	6.29 E-01	1.57 E-02	0.00 E+00	8.47 E-01	2.11 E-02	0.00 E+00
^{239}Pu	3.06 E-02	5.41 E-03	7.49 E-03	2.29 E-04	1.38 E+00	9.71 E-02	2.97 E-03	1.79 E+01
^{240}Pu	3.06 E-02	5.41 E-03	4.60 E-04	1.41 E-05	8.50 E-02	5.97 E-03	1.83 E-04	1.10 E+00
Total			1.84 E+00	3.53 E-02	2.55 E+01	3.49 E+00	7.66 E-02	6.97 E+01

Table B2-2. Decay Heat and A_2 for the Depleted Uranium Waste Boxes, Ci. (sheet 3 of 6)

Nuclide	W/Ci	A_2 (Ci)	Box 5			Box 6		
			Ci	W	A_2s	Ci	W	A_2s
^{60}Co	1.54 E-02	1.08 E+01	2.80 E-03	4.31 E-05	2.59 E-04	1.70 E-03	2.62 E-05	1.57 E-04
^{99}Tc	5.02 E-04	2.43 E+01	1.18 E-05	5.92 E-09	4.86 E-07	7.43 E-06	3.73 E-09	3.06 E-07
^{137}Cs	1.01 E-03	1.35 E+01	2.70 E-03	2.73 E-06	2.00 E-04	1.70 E-03	1.72 E-06	1.26 E-04
^{137m}Ba*	3.92 E-03	0.00	2.55 E-03	1.00 E-05	0.00 E+00	1.61 E-03	6.31 E-06	0.00 E+00
^{212}Bi*	1.68 E-02	0.00	3.60 E-03	6.05 E-05	0.00 E+00	2.30 E-03	3.86 E-05	0.00 E+00
^{212}Po*	5.21 E-02	0.00	3.60 E-03	1.88 E-04	0.00 E+00	2.30 E-03	1.20 E-04	0.00 E+00
^{216}Po*	4.02 E-02	0.00	7.00 E-03	2.81 E-04	0.00 E+00	4.50 E-03	1.81 E-04	0.00 E+00
^{224}Ra*	3.37 E-02	0.00	7.50 E-03	2.53 E-04	0.00 E+00	4.70 E-03	1.58 E-04	0.00 E+00
^{220}Rn*	3.73 E-02	0.00	7.00 E-03	2.61 E-04	0.00 E+00	4.40 E-03	1.64 E-04	0.00 E+00
^{228}Th	3.21 E-02	1.08 E-02	7.00 E-03	2.25 E-04	6.48 E-01	4.40 E-03	1.41 E-04	4.07 E-01
^{234}Th	3.97 E-04	5.41 E+00	6.82 E-01	2.71 E-04	1.26 E-01	4.31 E-01	1.71 E-04	7.97 E-02
^{234}U	2.83 E-02	2.70 E-02	8.06 E-01	2.28 E-02	2.99 E+01	5.08 E-01	1.44 E-02	1.88 E+01
^{235}U	2.71 E-02	Unlimited	1.15 E-02	3.12 E-04	0.00 E+00	7.26 E-03	1.97 E-04	0.00 E+00
^{238}U	2.49 E-02	Unlimited	8.06 E-01	2.01 E-02	0.00 E+00	5.10 E-01	1.27 E-02	0.00 E+00
^{239}Pu	3.06 E-02	5.41 E-03	9.36 E-03	2.86 E-04	1.73 E+00	5.99 E-03	1.83 E-04	1.11 E+00
^{240}Pu	3.06 E-02	5.41 E-03	5.75 E-04	1.76 E-05	1.06 E-01	3.68 E-04	1.13 E-05	6.80 E-02
Total			2.36 E+00	4.51 E-02	3.25 E+01	1.49 E+00	2.85 E-02	2.05 E+01

Table B2-2. Decay Heat and A_2 for the Depleted Uranium Waste Boxes, Ci. (sheet 4 of 6)

Nuclide	W/Ci	A_2 (Ci)	Box 7			Box 8		
			Ci	W	A_2s	Ci	W	A_2s
^{60}Co	1.54 E-02	1.08 E+01	1.54 E-03	2.37 E-05	1.43 E-04	2.50 E-03	3.85 E-05	2.31 E-04
^{99}Tc	5.02 E-04	2.43 E+01	6.53 E-06	3.28 E-09	2.69 E-07	1.06 E-05	5.32 E-09	4.36 E-07
^{137}Cs	1.01 E-03	1.35 E+01	1.48 E-03	1.49 E-06	1.10 E-04	2.40 E-03	2.42 E-06	1.78 E-04
^{137m}Ba*	3.92 E-03	0.00	1.40 E-03	5.49 E-06	0.00 E+00	2.27 E-03	8.90 E-06	0.00 E+00
^{212}Bi*	1.68 E-02	0.00	2.00 E-03	3.36 E-05	0.00 E+00	3.27 E-03	5.49 E-05	0.00 E+00
^{212}Po*	5.21 E-02	0.00	2.00 E-03	1.04 E-04	0.00 E+00	3.27 E-03	1.70 E-04	0.00 E+00
^{216}Po*	4.02 E-02	0.00	3.97 E-03	1.60 E-04	0.00 E+00	6.45 E-03	2.59 E-04	0.00 E+00
^{224}Ra*	3.37 E-02	0.00	4.15 E-03	1.40 E-04	0.00 E+00	6.74 E-03	2.27 E-04	0.00 E+00
^{220}Rn*	3.73 E-02	0.00	3.91 E-03	1.46 E-04	0.00 E+00	6.35 E-03	2.37 E-04	0.00 E+00
^{228}Th	3.21 E-02	1.08 E-02	3.91 E-03	1.26 E-04	3.62 E-01	6.35 E-03	2.04 E-04	5.88 E-01
^{234}Th	3.97 E-04	5.41 E+00	3.78 E-01	1.50 E-04	6.99 E-02	6.14 E-01	2.44 E-04	1.13 E-01
^{234}U	2.83 E-02	2.70 E-02	4.46 E-01	1.26 E-02	1.65 E+01	7.25 E-01	2.05 E-02	2.69 E+01
^{235}U	2.71 E-02	Unlimited	6.38 E-03	1.73 E-04	0.00 E+00	1.03 E-02	2.79 E-04	0.00 E+00
^{238}U	2.49 E-02	Unlimited	4.45 E-01	1.11 E-02	0.00 E+00	7.24 E-01	1.80 E-02	0.00 E+00
^{239}Pu	3.06 E-02	5.41 E-03	5.33 E-03	1.63 E-04	9.85 E-01	8.65 E-03	2.65 E-04	1.60 E+00
^{240}Pu	3.06 E-02	5.41 E-03	3.27 E-04	1.00 E-05	6.04 E-02	5.31 E-04	1.62 E-05	9.82 E-02
Total			1.31 E+00	2.49 E-02	1.80 E+01	2.12 E+00	4.05 E-02	2.93 E+01

Table B2-2. Decay Heat and A_2 for the Depleted Uranium Waste Boxes, Ci. (sheet 5 of 6)

Nuclide	W/Ci	A_2 (Ci)	Box 9			Box 10		
			Ci	W	A_2s	Ci	W	A_2s
^{60}Co	1.54 E-02	1.08 E+01	2.63 E-03	4.05 E-05	2.44 E-04	1.87 E-03	2.88 E-05	1.73 E-04
^{99}Tc	5.02 E-04	2.43 E+01	1.12 E-05	5.62 E-09	4.61 E-07	7.91 E-06	3.97 E-09	3.26 E-07
^{137}Cs	1.01 E-03	1.35 E+01	2.53 E-03	2.56 E-06	1.87 E-04	1.79 E-03	1.81 E-06	1.33 E-04
^{137m}Ba*	3.92 E-03	0.00	2.39 E-03	9.37 E-06	0.00 E+00	1.69 E-03	6.62 E-06	0.00 E+00
^{212}Bi*	1.68 E-02	0.00	3.45 E-03	5.80 E-05	0.00 E+00	2.44 E-03	4.10 E-05	0.00 E+00
^{212}Po*	5.21 E-02	0.00	3.45 E-03	1.80 E-04	0.00 E+00	2.44 E-03	1.27 E-04	0.00 E+00
^{216}Po*	4.02 E-02	0.00	6.79 E-03	2.73 E-04	0.00 E+00	4.82 E-03	1.94 E-04	0.00 E+00
^{224}Ra*	3.37 E-02	0.00	7.10 E-03	2.39 E-04	0.00 E+00	5.00 E-03	1.69 E-04	0.00 E+00
^{220}Rn*	3.73 E-02	0.00	6.69 E-03	2.50 E-04	0.00 E+00	4.74 E-03	1.77 E-04	0.00 E+00
^{228}Th	3.21 E-02	1.08 E-02	6.69 E-03	2.15 E-04	6.19 E-01	4.74 E-03	1.52 E-04	4.39 E-01
^{234}Th	3.97 E-04	5.41 E+00	6.47 E-01	2.57 E-04	1.20 E-01	4.59 E-01	1.82 E-04	8.48 E-02
^{234}U	2.83 E-02	2.70 E-02	7.63 E-01	2.16 E-02	2.83 E+01	5.39 E-01	1.53 E-02	2.00 E+01
^{235}U	2.71 E-02	Unlimited	1.09 E-02	2.95 E-04	0.00 E+00	7.72 E-03	2.09 E-04	0.00 E+00
^{238}U	2.49 E-02	Unlimited	7.65 E-01	1.90 E-02	0.00 E+00	5.41 E-01	1.35 E-02	0.00 E+00
^{239}Pu	3.06 E-02	5.41 E-03	9.12 E-02	2.79 E-03	1.69 E+01	6.46 E-03	1.98 E-04	1.19 E+00
^{240}Pu	3.06 E-02	5.41 E-03	5.60 E-03	1.71 E-04	1.04 E+00	3.97 E-04	1.21 E-05	7.34 E-02
Total			2.32 E+00	4.54 E-02	4.70 E+01	1.58 E+00	3.03 E-02	2.18 E+01

Table B2-2. Decay Heat and A_2 for the Depleted Uranium Waste Boxes, Ci. (sheet 6 of 6)

Nuclide	W/Ci	A_2 (Ci)	Box 11			Total for all 11 boxes		
			Ci	W	A_2s	Ci	W	A_2s
^{60}Co	1.54 E-02	1.08 E+01	2.10 E-03	3.23 E-05	1.94 E-04	2.00 E-02	3.08 E-04	1.85 E-03
^{99}Tc	5.02 E-04	2.43 E+01	8.90 E-06	4.47 E-09	3.66 E-07	2.82 E-03	1.42 E-06	1.16 E-04
^{137}Cs	1.01 E-03	1.35 E+01	2.00 E-03	2.02 E-06	1.48 E-04	2.87 E-01	2.90 E-04	2.13 E-02
^{137m}Ba*	3.92 E-03	0.00	1.89 E-03	7.41 E-06	0.00 E+00	2.72 E-01	1.07 E-03	0.00 E+00
^{212}Bi*	1.68 E-02	0.00	2.75 E-03	4.62 E-05	0.00 E+00	8.82 E-02	1.48 E-03	0.00 E+00
^{212}Po*	5.21 E-02	0.00	2.75 E-03	1.43 E-04	0.00 E+00	9.86 E-02	5.14 E-03	0.00 E+00
^{216}Po*	4.02 E-02	0.00	5.42 E-03	2.18 E-04	0.00 E+00	2.88 E-01	1.16 E-02	0.00 E+00
^{224}Ra*	3.37 E-02	0.00	5.66 E-03	1.91 E-04	0.00 E+00	4.11 E-01	1.39 E-02	0.00 E+00
^{220}Rn*	3.73 E-02	0.00	5.34 E-03	1.99 E-04	0.00 E+00	2.80 E-01	1.04 E-02	0.00 E+00
^{228}Th	3.21 E-02	1.08 E-02	5.34 E-03	1.71 E-04	4.94 E-01	4.02 E-01	1.29 E-02	3.72 E+01
^{234}Th	3.97 E-04	5.41 E+00	5.16 E-01	2.05 E-04	9.54 E-02	5.90 E+00	2.34 E-03	1.09 E+00
^{234}U	2.83 E-02	2.70 E-02	6.08 E-01	1.72 E-02	2.25 E+01	7.18 E+00	2.03 E-01	2.66 E+02
^{235}U	2.71 E-02	Unlimited	8.69 E-03	2.35 E-04	0.00 E+00	1.02 E-01	2.76 E-03	0.00 E+00
^{238}U	2.49 E-02	Unlimited	6.09 E-01	1.52 E-02	0.00 E+00	7.17 E+00	1.79 E-01	0.00 E+00
^{239}Pu	3.06 E-02	5.41 E-03	7.21 E-03	2.21 E-04	1.33 E+00	3.16 E-01	9.67 E-03	5.84 E+01
^{240}Pu	3.06 E-02	5.41 E-03	4.43 E-04	1.36 E-05	8.19 E-02	1.94 E-02	5.94 E-04	3.59 E+00
Total			1.78 E+00	3.41 E-02	2.45 E+01	2.28 E+01	4.54 E-01	3.66 E+02

This radionuclide is a daughter as defined in 49 CFR 173.433; therefore, its activity was set to 0 for the A_2 calculations.

49 CFR 173.433, 1997, "Shippers--General Requirements for Shipments and Packagings," (.433) "Requirements for determination of A_1 and A_2 values for radionuclides," *Code of Federal Regulations*, as amended.

2.1.1 Fissile Material Content

The contents to be transported in the waste boxes consists of DU, which is fissile excepted in accordance with the definition of fissile material in 49 CFR 173.403, which states that DU does not qualify as fissile material.

2.2 RESTRICTIONS

The contents as shown in Table B2-1 are the only contents authorized for the DU waste boxes.

2.3 SIZE AND WEIGHT

The maximum gross weight for the 11 boxes is 5936 lbs (2692 kg) and occurs for Box 4. The maximum quantity of 2495 kg (5500 lb) of DU is also contained in Box 4. Box 10 has the largest box dimensions (183 x 163 x 35.6 cm [72.0 x 64.0 x 14.0 in.]).

2.4 CONCLUSIONS

The contents to be transported in the DU waste boxes are fissile excepted and are Type B, non-highway route controlled quantities (49 CFR 173). The radiological risk evaluation in Part B, Section 3.0, and the shielding evaluation shown in Part B, Section 5.0, demonstrate that the DU waste boxes can ship the contents shown in Table B2-1 safely.

2.5 REFERENCE

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

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3.0 RADIOLOGICAL RISK EVALUATION

Eleven LSA boxes containing DU will be transported by highway from the FFTF to the 200 West Area burial grounds in a single shipment of two standard flatbed tractor-trailers. No more than six boxes of DU will be transported in a single tractor-trailer. The shipment will be made during daylight hours with closed roads, which will cover a distance of approximately 27.4 km (17.0 mi). The boxes will contain Type B quantities of material, but are not certified Type B shipping containers; therefore a radiological risk evaluation is made to determine the acceptability of the risks to the Hanford Site worker and to the public.

Onsite transportation safety requirements are outlined in WHC-CM-2-14, *Hazardous Material Packaging and Shipping,* and Mercado (1994). The acceptability of risks associated with onsite shipments is determined by a radiological risk evaluation, which applies risk acceptance criteria, package performance, and Hanford Site truck accident frequencies. A detailed risk analysis is not required, however, as long as a risk evaluation demonstrates that the risk acceptance criteria as outlined in Mercado (1994) are met.

Graded dose limitations for probable, credible, and incredible accident frequencies ensure safety in radioactive material packaging and transportation (Mercado 1994). The dose limitations to the offsite and onsite individuals are shown in Table B3-1.

Table B3-1. Risk Acceptance Criteria Limits.

Description	Annual frequency	Onsite dose limit* Sv (rem)	Offsite dose limit* Sv (rem)
Incredible	$<10^{-7}$	None	None
Incredible	10^{-7} to $<10^{-6}$	None	.25 (25)
Credible	10^{-6} to 10^{-3}	.05 (5)	.005 (.5)
Probable	10^{-3} to 1	.002 (.2)	.0001 (.01)

*Effective dose equivalent.

The Transportation Hazard Index (THI) evaluation shown in Part B, Section 4.6, determined that for the worst-case loading of one box of DU the public receptor dose is 0.034 Sv (3.4 rem) and the worker dose is 1.10 Sv (110 rem). If it is assumed that all six boxes on the tractor-trailer are breached in a potential accident and the dose consequence for one box is multiplied times the number of boxes in a tractor-trailer (six), the results can be compared to the risk acceptance criteria to determine the annual frequency limit. The total potential dose consequence is accordingly 0.21 Sv (21 rem) to the public and 7.0 Sv (700 rem) to the worker. The total potential dose consequence requires that the annual release frequency is less than 10^{-6}

In a radiological risk evaluation the total conditional probability of failure of the packaging is multiplied by the frequency of accidents per year to arrive at an annual accident release frequency. If the annual accident release frequency is below the required criteria, which in this case is a frequency of 10^{-6} releases per year, the shipment meets onsite transportation safety requirements (Mercado 1994).

The Hanford Site truck accident rate for all trucks including vans and lightweight pickup trucks is equal to 2.0×10^{-7} accidents per mile (Green et al. 1996). For a shipment of radioactive materials, which is carried out by trained truck drivers during daylight hours in good road conditions, a reduction factor of 20 can be applied to lower the rate to 1×10^{-8} (H&R 1995). The DU box shipment will be made in two tractor-trailers over a distance of 27.4 km (17.0 mi), which when multiplied by the accident rate and reduction factor, gives a frequency of 3.4×10^{-7} accidents per year. If no credit is taken for the performance of the boxes containing the material and the conditional release probability is assumed to be 1.0, the resulting accident release frequency is equal to 3.4×10^{-7} per year. The release frequency is below the required criterion of 10^{-6}. Therefore, the one-time shipment of DU boxes in two tractor-trailers meets onsite transportation safety requirements and presents no unacceptable risks to the worker or the public.

3.1 REFERENCES

Green, J. R., B. D. Flanagan, and H. W. Harris, 1996, *Hanford Site Truck Accident Rate, 1990-1995*, WHC-SD-TP-RPT-021, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

H&R, 1995, *Recommended Onsite Transportation Risk Management Methodology*, H&R522-1, H&R Technical Associates, Inc., Oak Ridge, Tennessee.

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.

WHC-CM-2-14, *Hazardous Material Packaging and Shipping*, Westinghouse Hanford Company, Richland, Washington.

3.2 APPENDIX: CHECKLIST FOR REVIEW

CHECKLIST FOR REVIEW

Document Reviewed: Radiological Risk Evaluation for the Depleted Uranium Waste Boxes

Scope of Review: entire document

Yes	No	NA		
[✓]	[ ]	[ ]	*	Previous reviews complete and cover analysis, up to scope of this review, with no gaps.
[✓]	[ ]	[ ]		Problem completely defined.
[✓]	[ ]	[ ]		Accident scenarios developed in a clear and logical manner.
[✓]	[ ]	[ ]		Necessary assumptions explicitly stated and supported.
[ ]	[ ]	[✓]		Computer codes and data files documented.
[✓]	[ ]	[ ]		Data used in calculations explicitly stated in document.
[✓]	[ ]	[ ]		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. Spreadsheet results should be treated exactly the same as hand calculations.
[ ]	[ ]	[✓]		Software input correct and consistent with document reviewed.
[ ]	[ ]	[✓]		Software output consistent with input and with results reported in document reviewed.
[✓]	[ ]	[ ]		Limits/criteria/guidelines applied to analysis results are appropriate and referenced. Limits/criteria/guidelines checked against references.
[✓]	[ ]	[ ]		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.
[ ]	[ ]	[✓]		Format consistent with appropriate NRC Regulatory Guide or other standards
[ ]		[✓]	*	Review calculations, comments, and/or notes are attached.
[✓]	[ ]	[ ]		**Document approved.**

J.E. Mercado J.E. Mercado — Aug. 18, 1997

Reviewer (Printed Name and Signature) — Date

* Any calculations, comments, or notes generated as part of this review should be signed, dated and attached to this checklist. Such material should be labeled and recorded in such a manner as to be intelligible to a technically qualified third party.

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4.0 CONTAINMENT EVALUATION

4.1 INTRODUCTION

The DU waste boxes are shown to meet the requirements for NTC containment as defined in Section 4.3 and the dose consequence acceptance criteria for accident conditions.

4.2 CONTAINMENT SOURCE SPECIFICATION

The containment source specification is described in Part B, Section 2.0, of this SEP. The physical form of the material is essentially solid, depleted, uranium metal.

4.3 NORMAL TRANSPORT CONDITIONS

The following sections describe Hanford onsite NTC evaluation requirements for general transport of Type B quantities of radioactive material in onsite packages. Because the DU boxes will be shipped only once and then permanently disposed of in a high-integrity container, rigorous evaluation to these requirements is not necessary. Section 4.5 will only discuss the specific transport conditions of this one-time shipment operation in comparison to the requirements listed in this section.

4.3.1 Conditions To Be Evaluated

4.3.1.1 Hot and Cold Conditions.

4.3.1.1.1 Hot Conditions. Evaluate package and payload performance for ambient temperature of 46.1 °C (115 °F) in still air with decay heat and solar insulation. Under hot conditions evaluate maximum exterior temperature of package for ambient temperature of 46.1 °C (115 °F) in still air with decay heat and in the shade. Use regulatory limits. Consider differential thermal expansion loads.

4.3.1.1.2 Cold Conditions. Evaluate package and payload performance for ambient temperature of -32.8 °C (-27 °F) in still air and in the shade. Consider differential thermal contraction loads.

4.3.1.2 Reduced and Increased External Pressure.

4.3.1.2.1 Reduced External Pressure. Evaluate package and payload performance for reduced external pressure of 95.2 kPa, absolute (13.81 psia). Determine differential pressure from the package internal pressure at operating temperature assuming it was initially loaded at an initial internal pressure of 101.4 kPa (14.7 psi) at 21.1 °C (70 °F).

4.3.1.2.2 Increased External Pressure. Evaluate package and payload performance for increased external pressure of 102.4 kPa, absolute (14.85 psia). Determine differential pressure from the package internal pressure at operating temperature assuming it was initially loaded at an initial internal pressure of 101.4 kPa (14.7 psi) at 21.1 °C (70 °F).

4.3.1.3 Vibration. Evaluate package fatigue performance for vibration normally incident to transport using response parameters given in American National Standards Institute (ANSI) N14.23 (ANSI 1992). For continuously used packages, assume infinite life as 10^6 cycles.

4.3.1.4 Water Spray. Evaluate package performance for resistance to water in-leakage. Consider gas leak-tight seals, as appropriate, for preventing in-leakage of water.

4.3.1.5 Inertial Loading. Inertial loading of packages is defined as the loads resulting from rough transport of the package. For rough transport assume the shock loading parameters from ANSI N14.23 (ANSI 1992) of 3.5*g* vertical, 2.3*g* longitudinal, and 1.6*g* lateral for rough transport. Note this is assumed to be a repetitive loading condition. Consequently, evaluate package fatigue response for a minimum of 1,000 cycles per shipment. Also note loading and unloading drops are not bounded by this condition and must be covered in the facility safety analysis report. If these conditions are applied, no in-route load transfer must be allowed in the safety analysis report for packaging.

4.3.1.6 Compression. For packages under 5,000 kg, evaluate package performance for a compression load of five times the weight of the package applied uniformly on the top and bottom of package.

4.3.1.7 Penetration. Evaluate package performance for a vertical steel cylinder with a hemispherical end of 3.2 cm in diameter and 6 kg of mass, dropping from a height of 1 m onto the package. The cylinder is assumed to strike the package's exposed surface in the region most vulnerable to puncture. The long axis of the cylinder is assumed to be perpendicular to the package surface.

4.3.1.8 Brittle Fracture. As a minimum, evaluate all metallic materials susceptible to brittle fracture per *ASME Boiler and Pressure Vessel Code* (ASME 1995) criteria.

4.3.1.9 Lifting and Tiedown. Evaluate package lifting performance of devices attached to the cask to the *Hanford Site Hoisting and Rigging Manual* (RL 1996) requirements. For tiedown attachments attached to the cask, evaluate performance either for (1) failure of the attachment prior to damage of the cask or (2) attachment meets the 10, 2, and 5 requirement of 10 CFR 71.45 (b).

4.4 ACCIDENT CONDITIONS

Accident conditions are evaluated for the DU waste boxes by radiological risk and dose consequence analysis. The radiological risk evaluation is given in Part B, Section 3.0, of this SEP. The dose consequence and associated transportation hazard index are given in Section 4.7.

4.5 CONTAINMENT EVALUATION AND CONCLUSIONS

4.5.1 Normal Transport Conditions

The DU waste boxes, based on the discussions in the following sections, are shown to be acceptable during NTC for this one-time shipment.

4.5.1.1 Hot and Cold Conditions. The intent of this requirement is to evaluate a packaging's ability to withstand differential thermal expansion and contraction loads over a wide temperature range. In addition, it evaluates the packaging material for brittle fracture at extremely low temperature and the packaging surface temperature at high temperatures to ensure that when combined with the internal decay heat of the packaging, the external surface of the packaging does not exceed 85 °C (185 °F).

The DU waste boxes will be transported in the month of September 1997. Differential thermal expansion and contraction loads will be minimal due to the minimal ambient air temperature transitions this time of the year on the Hanford Site. Low-temperature brittle fracture of the thin-walled boxes and the wooden box is not a concern. The decay heat of the worst-case payload is less than 1 W. In the unlikely event that the ambient temperature during the shipment exceeds 37.8 °C (100 °F), the surface of the boxes will not exceed 85 °C (185 °F).

4.5.1.2 Reduced and Increased External Pressure. The intent of this requirement is to ensure structural integrity of the packaging during large variations in barometric pressure due to changes in altitude and atmospheric temperature. It also evaluates the ability of the packaging to withstand increased internal pressure due to gas generation and payload decay heat.

None of these conditions will exist during this one-time shipment. There will be minimal changes in altitude and temperature during the short trip (FFTF to the 200 West Area). Pressure buildup within the boxes due to payload decay heat or hydrogen gas generation does not exist.

4.5.1.3 Vibration. The vibration evaluation is intended to investigate the fatigue failure potential for packagings used on a relatively frequent basis. The design fatigue life of the packaging is one million cycles, and the evaluation uses the ANSI N14.23 (ANSI 1992) criteria for analysis.

The DU waste boxes will be shipped one time approximately 27.4 km (17.0 mi). Using the ANSI N14.23 (ANSI 1992) applied cyclic rate of 2 Hz and conservatively averaging the transporter speed at 32.2 km/h (20 mi/h), the number of vibration cycles incurred will be 6,120. This less than 0.7% of a normal fatigue life evaluation period.

The boxes have been in indoor storage for many years. The previous usage history is unknown. It is known that they were shipped to the Hanford Site from an offsite source. Visual inspection of the containers indicate that they are in good condition. The additional small number of transport vibration cycles is unlikely to significantly reduce the structural integrity of the packagings.

4.5.1.4 Water Spray. Ten of the 11 DU waste boxes are constructed of steel with rubber gasketed seals. Water in-leakage is not a potential problem. The wooden box will be wrapped in plastic with minimum 10-mil thickness with tape seal closure. In addition, shipment during inclement weather is specifically prohibited in the operational controls set for in Part A of this SEP. Water in-leakage is not a concern.

4.5.1.5 Inertial Loading. The inertial loading evaluation is intended to investigate potential failure of a package during rough transport conditions. The evaluation provides vertical, longitudinal, and lateral loading criteria in order to evaluate the packaging's ability to maintain structural integrity when the transport vehicle encounters physical features of the roadway (e.g., potholes, bumps, washboard) that transmit shock loads through the vehicle chassis to the packaging. In addition, these loads are intended to simulate hard braking of the transport vehicle and high *g*-load cornering. The design baseline for these loadings is 1,000 cycles per shipment.

For this one-time shipment of the DU boxes, the effects of shock and vibration are not analyzed. The roadway between the FFTF and the 200 West Area is relatively smooth and will not produce a large number of higher-than-normal shock loadings. The roadway inside the 200 West Area is not as smooth; however, the short duration of the trip should not significantly impact the structural integrity of the boxes.

Wood dunnage is provide on the inside of the boxes to minimize impact loadings resulting from the payload shifting during incidental shock loads.

4.5.1.6 Compression. The DU waste boxes are restricted to a single height for transport. They are currently stored in stacks of three with no visible indication of excessive stress. Failure due to compression is not a concern.

4.5.1.7 Penetration. The penetration evaluation is intended for packagings that are routinely handled during the course of transport operations. It includes bumping by machinery, setting the container on a surface object, falling tools, and other mishaps coincident to loading, in-transit handling, and unloading operations.

Visual inspection and historical knowledge of these types of containers indicate that the puncture test is not a concern. For this one-time shipment the penetration evaluation is not performed.

4.5.1.8 Brittle Fracture. The material used in construction of the DU waste boxes is not susceptible to brittle fracture per ASME (1995) criteria.

4.5.1.9 Lifting and Tiedown. There are no tiedown attachment points integral to the DU waste boxes. Forklift tine slots are provided on the bottom of the metal boxes to facilitate handling by a forklift. The wooden box provides for forklift handling with wooden timbers attached to the bottom of the box. There are no other lifting attachment features integral to the packagings; therefore, no further evaluation is required.

4.5.2 Accident Conditions

Based on the radiological risk evaluation in Part B, Section 3.0, of this SEP and the dose consequence evaluation given in Section 4.6, the DU waste boxes can be transported one time to the burial ground while still remaining within the acceptable limits for onsite and offsite receptor doses.

4.6 SUMMARY OF DOSE CONSEQUENCE RESULTS

This engineering analysis documents the dose consequence calculations used to support the THI evaluation for the DU waste boxes. There are a total of 11 LSA boxes that will be used to transfer DU from the FFTF to the 200 West Area burial grounds.

Table B4-1 shows the dose consequence summary results from each exposure pathway for the maximum authorized contents for the DU waste boxes. The table also includes the dose to each receptor, which is obtained by summing the dose contributions from each pathway. Because the public receptor dose is greater than 0.5 rem and the onsite worker dose is greater than 5 rem, the packaging must be designed to THI 2 requirements. The criteria for a THI of 2, as stated in WHC-CM-2-14 is:

> "THI-2: This represents the second highest level of hazard from the contents. A packaging system assigned this level transports material that has the potential of causing a dose consequence to an individual between 0.5 rem and 25 rem at the Hanford Site boundary, or greater than 5 rem within the site, if fully released."

Table B4-1. Summary of Doses for the Depleted Uranium Waste Boxes, rem (Sv).

Exposure pathway	Hanford Site worker at 3 m	Public receptor*
External photon dose	7.0 E-05 (7.0 E-07)	NA
External dose from β-particles	2.3 E-04 (2.3 E-06)	NA
Inhalation and submersion from the airborne transport pathway	110 (1.1)	3.4 (0.034)
Total effective dose equivalent	110 (1.1)	3.4 (0.034)

Note: 100 rem = 1 sievert (Sv).
*This receptor is located 100 m N of the 400 Area.

4.6.1 Introduction and Overview

Eleven LSA boxes at FFTF contain DU, two of which were irradiated during their use as shielding, while the remaining nine are unirradiated. These boxes will be transported together for a one-time shipment from the FFTF to the 200 West Area burial grounds.

An estimate of the dose consequences for various exposure pathways is necessary to determine the THI for the DU waste boxes. Section 4.5.2 discusses the general methodology used to perform the dose consequence calculations. Section 4.5.3 addresses the source term, and Sections 4.5.4 through 4.5.9 summarize the results for various exposure pathways.

4.6.2 Dose Consequence Analysis Methodology

IAEA (1990) defines a standardized approach for evaluating transportation packaging requirements, called the Q-system. The Q-system methods, as outlined in IAEA (1990), have been incorporated into *Report on Equivalent Safety for Transportation and Packaging of Radioactive Materials* (Mercado 1994). Mercado (1994) is used to demonstrate that onsite shipments meet onsite transportation safety requirements per WHC-CM-2-14.

In the Q-system, the following five exposure pathways are considered: (1) external exposure to photons, (2) external exposure to β-particles, (3) inhalation, (4) skin contamination and ingestion, and (5) submersion in a cloud of gaseous isotopes. In special cases, such as α-particle or neutron emitters, other exposure routes are considered. In some cases a pathway will be judged to be small with respect to the others, and consideration will be minimal. Modifications to the International Atomic Energy Agency (IAEA) scenarios are incorporated to more closely describe the particular conditions of the shipment. Detailed calculations for the postulated accident are performed whenever possible. However, in some cases, the worst-case rules-of-thumb of IAEA (1990) are used.

The Q-system was developed as an all-encompassing generalized methodology using only the isotope as the defining variable. In this report, the specifics of the package are considered. Some of the dose pathways may be considered incredible (frequency $<10^{-6}$/yr), and although these pathways are covered in IAEA (1990), they are disregarded in the analysis.

In the IAEA system, the Q-values that are calculated are the radionuclide activities corresponding to each exposure route that causes the individual to receive the effective dose equivalent (EDE) limit. The minimum Q-values define the A_2 values for the shipped materials. In the case of nondispersible materials (limited by the A_1 values), only the first two Q-values (based on exposure to external photon and external beta particles) are used. Note that for all radiation except neutrons, protons, and heavier charged particles (including α-particles), 1 gray (Gy) = 1 sievert (Sv), and 1 rad = 1 rem.

There are two receptors of interest in the Q-system: the Hanford Site worker and the public receptor. The Hanford Site worker is assumed to be located about 3 m from the package. The public receptor is assumed to be located at the nearest point of public access, which is 100 m N of the 400 Area.

4.6.3 Source Term

The DU material is contaminated with greater-than-Type A quantities of radionuclides other than uranium. However, the contamination is mostly nonremovable and nondispersible. Table B4-2 shows the radioactive inventory for each box along with the total for each nuclide for all 11 boxes (see Table B4-2, sheet 2). Upon examination of Table B4-2, it was found that Box 4 has the worst radioactive inventory from a dose consequence standpoint. Although the direct gamma dose from Box 5 is higher than that for Box 4 due to the higher ^{60}Co content, the direct beta and inhalation dose from Box 4 is higher than that for Box 5 due to the higher ^{137}Cs and ^{239}Pu content. Because the inhalation dose dominates the total dose, Box 4 has the worst radioactive inventory from a dose consequence standpoint.

Table B4-2. Depleted Uranium Waste Boxes Inventory, Ci. (sheet 1 of 2)

Nuclide	Box 1	Box 2	Box 3	Box 4	Box 5	Box 6
^{60}Co	2.42 E-03	1.98 E-04	2.00 E-03	2.81 E-04	2.80 E-03	1.70 E-03
^{99}Tc	1.03 E-05	1.13 E-03	9.18 E-06	1.61 E-03	1.18 E-05	7.43 E-06
^{137}Cs	2.33 E-03	1.11 E-01	2.00 E-03	1.57 E-01	2.70 E-03	1.70 E-03
^{137m}Ba	2.20 E-03	1.05 E-01	1.89 E-03	1.49 E-01	2.55 E-03	1.61 E-03
^{212}Bi	3.17 E-03	3.80 E-02	2.80 E-03	2.44 E-02	3.60 E-03	2.30 E-03
^{212}Po	3.17 E-03	3.01 E-02	2.80 E-03	4.27 E-02	3.60 E-03	2.30 E-03
^{216}Po	6.25 E-03	9.82 E-02	5.60 E-03	1.39 E-01	7.00 E-03	4.50 E-03
^{224}Ra	6.53 E-03	1.48 E-01	5.80 E-03	2.10 E-01	7.50 E-03	4.70 E-03
^{220}Rn	6.16 E-03	9.50 E-02	5.50 E-03	1.35 E-01	7.00 E-03	4.40 E-03
^{228}Th	6.16 E-03	1.46 E-01	5.50 E-03	2.07 E-01	7.00 E-03	4.40 E-03
^{234}Th	5.96 E-01	4.33 E-01	5.32 E-01	6.14 E-01	6.82 E-01	4.31 E-01
^{234}U	7.07 E-01	5.95 E-01	6.32 E-01	8.49 E-01	8.06 E-01	5.08 E-01
^{235}U	1.00 E-02	8.51 E-03	8.95 E-03	1.21 E-02	1.15 E-02	7.26 E-03
^{238}U	7.04 E-01	5.95 E-01	6.29 E-01	8.47 E-01	8.06 E-01	5.10 E-01
^{239}Pu	8.39 E-03	6.85 E-02	7.49 E-03	9.71 E-02	9.36 E-03	5.99 E-03
^{240}Pu	5.15 E-04	4.21 E-03	4.60 E-04	5.97 E-03	5.75 E-04	3.68 E-04

Table B4-2. Depleted Uranium Waste Boxes Inventory, Ci. (sheet 2 of 2)

Nuclide	Box 7	Box 8	Box 9	Box 10	Box 11	Total
^{60}Co	1.54 E-03	2.50 E-03	2.63 E-03	1.87 E-03	2.10 E-03	2.00 E-02
^{99}Tc	6.53 E-06	1.06 E-05	1.12 E-05	7.91 E-06	8.90 E-06	2.82 E-03
^{137}Cs	1.48 E-03	2.40 E-03	2.53 E-03	1.79 E-03	2.00 E-03	2.87 E-01
^{137m}Ba	1.40 E-03	2.27 E-03	2.39 E-03	1.69 E-03	1.89 E-03	2.72 E-01
^{212}Bi	2.00 E-03	3.27 E-03	3.45 E-03	2.44 E-03	2.75 E-03	8.82 E-02
^{212}Po	2.00 E-03	3.27 E-03	3.45 E-03	2.44 E-03	2.75 E-03	9.86 E-02
^{216}Po	3.97 E-03	6.45 E-03	6.79 E-03	4.82 E-03	5.42 E-03	2.88 E-01
^{224}Ra	4.15 E-03	6.74 E-03	7.10 E-03	5.00 E-03	5.66 E-03	4.11 E-01
^{220}Rn	3.91 E-03	6.35 E-03	6.69 E-03	4.74 E-03	5.34 E-03	2.80 E-01
^{228}Th	3.91 E-03	6.35 E-03	6.69 E-03	4.74 E-03	5.34 E-03	4.02 E-01
^{234}Th	3.78 E-01	6.14 E-01	6.47 E-01	4.59 E-01	5.16 E-01	5.90 E+00
^{234}U	4.46 E-01	7.25 E-01	7.63 E-01	5.39 E-01	6.08 E-01	7.18 E+00
^{235}U	6.38 E-03	1.03 E-02	1.09 E-02	7.72 E-03	8.69 E-03	1.02 E-01
^{238}U	4.45 E-01	7.24 E-01	7.65 E-01	5.41 E-01	6.09 E-01	7.17 E+00
^{239}Pu	5.33 E-03	8.65 E-03	9.12 E-02	6.46 E-03	7.21 E-03	3.16 E-01
^{240}Pu	3.27 E-04	5.31 E-04	5.60 E-03	3.97 E-04	4.43 E-04	1.94 E-02

4.6.4 External Dose Due to Photon (Gamma) Exposure

The IAEA scenario assumes that a person is exposed to a damaged transport package following an accident. The shielding of the package is assumed to be completely lost in the accident. This analysis will be done assuming a person remains 3 m from the source for 15 minutes.

The computer code ISO-PC (Rittmann 1995) was used to calculate the dose rate 3 m from the source. The fluence-to-dose conversion factors used were the anterior-to-posterior irradiation pattern as outlined in ANSI standard ANSI/ANS-6.1.1-1991 (ANS 1991).

As discussed in Section 4.6.3, the radioactive inventory for Box 4 was used for this analysis. Box 4 is 107 cm (42 in.) long, 127 cm (50 in.) wide, and 23.5 cm (9.25 in.) high. The payload for Box 4 consists of 2495 kg of DU plates that are packed closely together in the box with wooden shoring above (and possibly below) to keep the plates from moving vertically. The plates essentially fill the inside box dimensions, thus, preventing horizontal movement. Assuming a density of 18 g/cm^3 for the uranium metal, the approximate height of the DU plates is 10.2 cm (4 in.). Therefore, the source was assumed to be homogeneously distributed throughout a volume of 42 x 50 x 4 in. (107 x 127 x 10.2 cm). No credit is taken for any shielding afforded by the box or shoring material.

The resulting dose rate from ISO-PC is 2.8 x 10^{-4} rem/hr (2.8 x 10^{-4} Sv/hr) at 3 m from the unshielded source. Therefore the maximum total external gamma EDE for the Hanford Site worker is 7.0 x 10^{-5} rem (7.0 x 10^{-7} Sv) for a 15-minute exposure period. The ISO-PC input deck is shown in Section 4.8.1.

4.6.5 External Dose Due to β-Particle Emitters

Because of the limited range of β-particles relative to that of photons, a shielding factor is used by the IAEA to account for residual shielding from material such as package debris. Except for this factor, no effort is made to account for either self-shielding or shielding from an accurate model of the damaged package. Shielding and dose rate factors are graphed in the IAEA Safety Guide No. 7 (IAEA 1990) as a function of the maximum energy of the β-particle. The IAEA beta dose rate calculation methods are based on an individual located 1 m from the unshielded source.

This analysis assumes an individual remains at a distance of 3 m from the source for a 15-minute exposure period. A factor will be applied to the dose rates calculated using the IAEA method to account for the difference between the 1-m distance assumed in developing the shielding factors and the 3-m distance in this analysis. This factor was conservatively taken to be 0.333 [(1 m/3 m)] since the dose rate falls off between $1/r^2$ and 1/r, where r is the distance from the source. This also conservatively ignores any attenuation of the beta particles over the 3-m distance.

Table B4-3 shows the β-particle dose calculations for the inventory listed in Table B4-2 for Box 4. The total β-particle dose rate to the skin for an individual located 3 m from the source is 9.2 x 10^{-2} rem/hr (9.2 x 10^{-4} Sv/hr). This results in a β-particle dose of 2.3 x 10^{-2} rem (2.3 x 10^{-4} Sv) to the skin for a 15-minute exposure. Because the tissue weighting factor for the skin is 0.01 (ICRP 1991), the whole body EDE is then 2.3 x 10^{-4} rem (2.3 x 10^{-6} Sv).

4.6.6 Inhalation and Ingestion Dose

Radioactive material may be inhaled following an accident due to resuspension or volatization of radioactive material released from the package. This section addresses the dose received by workers and the public due to exposure to airborne radioactivity during a postulated accident event. Although there is some external contamination present on the surface of the DU as a result of oxidation, the radioactive material is mostly nonremovable and nondispersible. Therefore, only a fire scenario will be considered for this portion of the analysis. Note that the results for a fire scenario conservatively bound those for any nonfire scenario that could be postulated that would result in an airborne release.

4.6.6.1 Selection of Airborne Release Fraction. An airborne release fraction (ARF) of 1 x 10^{-3} and a respirable fraction (RF) of 1 is applied to the material at risk to obtain the quantity of radioactive material that is made airborne for the fire scenario. The ARF/RF was taken from DOE (1994), "Summary of Analysis Data, Thermal Stress: Uranium," and is the recommended bounding value associated with oxidation of uranium at elevated temperatures. Note that the use of this value for the postulated fire scenario is conservative because the actual experiments involve uranium rods spaced apart in an array and suspended in air while being subjected to cyclic thermal stress. It is unlikely that any accident would result in the DU plates being broken or spaced apart and subject to cyclic thermal stress as was the case for the experimental arrangement.

Table B4-3. β-Particle Dose Rate for Beta Emitters Contributing > 0.01% to the Total Dose.[c]

Isotope	Activity (Ci)	Activity (Bq)	Branching ratio	E_{max} (MeV)	Dose rate factor[a]	Shielding factor[b]	Dose rate (rem/h)	% dose
^{137}Cs	1.57 E-01	5.82 E+09	0.946	0.51155	1.8 E-04	100	8.94 E-03	9.71
			0.054	1.17320	3.1 E-04	6	1.46 E-02	15.91
^{212}Bi	2.44 E-02	9.03 E+08	0.0344	0.62536	1.8 E-04	100	5.04 E-05	0.05
			0.0261	0.73330	1.8 E-04	20	1.91 E-04	0.21
			0.08	1.51880	3.1 E-04	3	6.72 E-03	7.31
			0.484	2.24600	3.1 E-04	2	6.09 E-02	66.35
^{234}Th	6.14 E-01	2.27 E+10	0.068	0.09578	1.0 E-05	500	2.78 E-05	0.03
			0.185	0.09620	1.0 E-05	500	7.56 E-05	0.08
			0.725	0.18858	1.0 E-05	500	2.97 E-04	0.32
Totals for beta emitters contributing > 0.01%							9.18 E-02	99.98
Totals for all beta emitters							9.21 E-02	100.00

[a] Dose rate factor in units of Gy/hr or Sv/hr for a 1-m Ci source from IAEA (1990).
[b] Shielding factor from IAEA (1990).
[c] Note that a factor of 0.333 is applied to the dose rates to account for a source-to-receptor distance of 3 m for this analysis, versus the 1-m distance assumed in the development of the dose rate factors from IAEA (1990).

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

The ARF x RF of 1 x 10^{-3} is applied to the material at risk, which is conservatively taken to be 100% of the Box 4 inventory from Table B4-2. The quantity of airborne radioactive material released is shown in Table B4-4.

Note that DOE (1994) states that the uranium material made airborne during the fire is in oxide form; i.e., is associated with the "Y" solubility class (dissolution halftimes in simulated interstitial lung fluids of >100 days).

4.6.6.1.1 Discussion of Integrated Normalized Air Concentration Value (χ/Q'). After the radioactive material becomes airborne, it is transported downwind and inhaled by onsite workers or the public. The concentration of this material is reduced, or diluted, as it is being transported due to atmospheric mixing and turbulence. χ/Q' (s/m^3) is used to characterize the dilution of the airborne contaminants during atmospheric transport and dispersion. It is equal to the time-integrated normalized air concentration at the receptor. χ/Q' is a function of the atmospheric conditions (i.e., wind speed, stability class) and the distance to the receptor.

Bounding χ/Q' values are generated consistent with the methods described in *Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,* Regulatory Guide 1.145 (NRC 1982). Since atmospheric conditions fluctuate, a bounding atmospheric condition is determined to be that condition that causes a downwind concentration of airborne contaminants that is exceeded only a small fraction of time because of weather

Table B4-4. Accident Airborne Release Quantities.

Nuclide	Ci
^{60}Co	2.81 E-07
^{99}Tc	1.61 E-06
^{137}Cs	1.57 E-04
^{137m}Ba	1.49 E-04
^{212}Bi	2.44 E-05
^{212}Po	4.27 E-05
^{216}Po	1.39 E-04
^{244}Ra	2.10 E-04
^{220}Rn	1.35 E-04
^{228}Th	2.07 E-04
^{234}Th	6.14 E-04
^{234}U	8.49 E-04
^{235}U	1.21 E-05
^{238}U	8.47 E-04
^{239}Pu	9.71 E-05
^{240}Pu	5.97 E-06

fluctuations. Regulatory Guide 1.145 (NRC 1982) defines this fraction of exceedance as 0.5% for each sector or 5% for the overall Hanford Site. The Hanford Site is broken up into 16 sectors that represent 16 compass directions (i.e., S, SSW, SW, . . ., ESE, SE, SSE). χ/Q' values are generated for weather conditions that result in downwind concentrations exceeded only 0.5% of the time in the maximum sector or 5% of the time for the overall Site. These χ/Q' values are also referred to as 99.5% maximum sector and 95% overall Site χ/Q' values. The greater of these two values is called the bounding χ/Q' value and is used to assess the dose consequences for accident scenarios. The bounding χ/Q' value represents minimum dispersing conditions that result in maximum downwind concentrations; i.e., concentrations exceeded only a very small fraction of the time. This χ/Q' value will therefore result in very conservative estimates of accident consequences.

The χ/Q' values in this report were generated using the GXQ computer program, Version 3.1C (Hey 1993a, 1993b). The meteorological data used by GXQ are in the form of joint frequency tables. The joint frequency data are the most recent data available; they are nine-year-averaged data (1983-1991) from the Hanford Site meteorology towers located in the 200 and 400 Areas. The χ/Q' values are generated using the methods described in Regulatory Guide 1.145 (NRC 1982) for a ground release with no credit taken for plume rise, plume meander, plume depletion, or any other models. This is conservative because all of these models reduce the airborne concentration at the downwind receptor locations.

Although we are interested in the dose to a Hanford Site worker at 3 m, the dose to an onsite receptor located 100 m from the release point is calculated using the worst-case χ/Q' value at 100 m. This dose is then multiplied by a factor of 30 to obtain the dose to the Hanford Site worker at 1 m in accordance with IAEA (1990). This approach is taken because the Gaussian equation, along with the parameters used to calculate the χ/Q' values, are only valid for distances of 100 m or greater. Although this analysis assumes the transport worker remains 3 m from the package, the inhalation portion of the transport worker dose is conservatively taken to be that calculated using the IAEA method for a worker located 1 m from the package.

The DU waste boxes will be transported from the 400 Area to the 200 West Area. The maximum χ/Q' value for an onsite receptor is 3.41×10^{-2} s/m^3 and occurs for an individual located 100 m E of the release point in the 200 Area.

The 400 Area is not a public exclusion area. Even though the roads may be closed during movement of the DU waste boxes, members of the public may be in the area. Therefore, it is conservatively assumed for this analysis that the public receptor is located in the 400 Area, 100 m from the release point in any compass direction. The maximum public receptor χ/Q' value is 3.17×10^{-2} s/m^3 and occurs for an individual located 100 m N of the release point in the 400 Area. The GXQ input files for the maximum χ/Q' cases are listed in Section 4.8.2. The titles of the joint frequency files used by GXQ is listed below:

- 200 AREA (HMS) - 10 M - Pasquill A - G (1983 - 1991 Average)
- 400 AREA (FFTF) - 10 M - Pasquill A - G (1983 - 1991 Average).

4.6.6.1.2 Inhalation and Submersion Dose Calculations. Because the GENII computer code Version 1.485 (Napier 1988) is the Site standard computer code for environmental release dose calculations, it was used to calculate the inhalation and submersion dose for the maximum onsite and public receptors. The airborne release quantities used in GENII are shown in Table B4-4. An example GENII input deck is listed in Section 4.8.3. Note that the uranium made airborne during the fire was in oxide form; i.e., Y solubility class. This is also assumed to apply to the plutonium in the source term; therefore, the Pacific Northwest National Laboratory solubility library was used. The GENII libraries used were as follows:

- GENII Default Parameter Values (28-Mar-90 RAP)
- Radionuclide Library - Times<100 years (23-July-93 PDR)
- External Dose Factors for GENII in person Sv/yr per Bq/n (8-May-90)
- Pacific Northwest National Laboratory Solubilities, Yearly Dose Increments (23-Jul-93 PDR).

The EDE from GENII for the inhalation and submersion pathways is 6.4 rem (6.4×10^{-2} Sv) for the maximum onsite receptor at 100 m NNE of the 200 Area. The inhalation dose contribution to the EDE is based on a 50-year dose commitment period. The maximum χ/Q' value from GENII was 6.0×10^{-2} s/m^3 for the maximum onsite receptor. The dose rates calculated by GENII are proportional to the χ/Q' values. The GXQ code calculates the 99.5% maximum sector and 95% overall Site χ/Q' values consistent with Regulatory Guide 1.145 (NRC 1982) methods, while GENII is inconsistent with Regulatory Guide 1.145 methods. As mentioned in the previous section, the maximum onsite receptor χ/Q' value from GXQ is 3.41×10^{-2} s/m^3. Therefore, the EDE for the inhalation and submersion pathways is 3.6 rem (3.6×10^{-2} Sv) for the maximum onsite receptor at 100 m using the GXQ χ/Q' value for the 200 Area. This value was obtained by multiplying the GENII dose rate by the ratio of the GXQ χ/Q' value to the GENII χ/Q' value. Similarly, the maximum public receptor dose is 3.4 rem (3.4×10^{-2} Sv) using the GXQ χ/Q' value of 3.17×10^{-2} s/m^3 for a receptor located 100 m N of the release point in the 400 Area.

To compensate for the fact that the onsite dose is calculated at a source-to-receptor distance of 100 m, this dose is multiplied by a factor of 30 to obtain the dose to the transport worker at 1 m in accordance with IAEA (1990). Although this analysis assumes the transport worker remains 3 m from the package, the inhalation portion of the transport worker dose is conservatively taken to be that calculated using the IAEA method for a worker located 1 m from the package. This results in an EDE of 110 rem (1.1 Sv) for the Hanford Site worker. Table B4-5 shows the doses for the postulated accident scenario.

Table B4-5. Inhalation and Submersion Dose (rem).

	Hanford worker (at 3 m)	Public receptor*
Effective dose equivalent	110	3.4

Note: 100 rem = 1 Sv.
*This receptor is located 100 m N of the 400 Area.

4.6.6.1.3 Ingestion and Ground Shine Dose. The other potential internal exposure pathway for the public receptor is the ingestion pathway. Exposure through the ingestion pathway occurs when radioactive materials that have been deposited offsite during passage of the plume are ingested either by eating crops grown in, or animals raised on, contaminated soil or through drinking contaminated water. There are U.S. Department of Energy; U.S. Department of Energy, Richland Operations Office; state; and federal programs in place to prevent ingestion of contaminated food in the event of an accident (RL 1994, WSDOH 1993, WS 1994, EPA 1992). The primary determinant of exposure from the ingestion pathway is the effectiveness of public health measures (i.e., interdiction) rather than the severity of the accident itself. The ingestion pathway, if it occurs, is a slow-to-develop pathway and is not considered an immediate threat to an exposed population in the same sense as airborne plume exposures.

The ground shine pathway is an additional potential external exposure pathway for the public receptor. Ground shine refers to the external dose received by a person standing on ground contaminated by radioactive materials deposited during passage of the airborne radioactive plume. Similar to the ingestion pathway, the primary determinant of exposure from the ground shine pathway is the effectiveness of public health measures (i.e., interdiction) rather than the severity of the accident itself. The ground shine pathway is a slow-to-develop pathway and is not considered an immediate threat to an exposed population in the same sense as airborne plume exposures.

Because of the radioactive inventory contained in the boxes, it is argued that in the event of an accident scenario that results in the release of a large portion of the inventory, interdictive measures (RL 1994, WSDOH 1993, WS 1994, EPA 1992) would be taken to prevent ingestion of contaminated food and exposure through the ground shine pathway. Therefore, the ingestion and ground shine pathway doses were not calculated in this report.

4.6.7 Skin Contamination and Ingestion Dose

In the IAEA guide (IAEA 1990), it is assumed that 1% of the package contents are spread over an area of 1 m^2 and handling of debris results in contamination of the hands to 10% of this level. It is further assumed that the worker is not wearing gloves, but that the individual recognizes the possibility of contamination and washes the hands within 5 hours. The EDE to the skin received by the individual is estimated from a graph provided in the IAEA guide.

The IAEA scenario for the uptake of activity due to ingestion of the material assumes that the person ingests all of the contamination from 10 cm^2 of skin over a 24-hour period. Because the dose per unit uptake via inhalation is generally the same order or larger than that via ingestion, the inhalation pathway will normally be limiting for internal contamination due to β-ray emitters. In particular, if the skin contamination dose is much larger than the inhalation dose, the ingestion pathway is not considered.

Both these pathways are ordinarily neglected when calculating the dose consequences from an onsite transportation accident. The transportation workers are trained in the appropriate response to protect themselves from experiencing unnecessary radiation exposure, including preventing skin contamination and ingestion.

4.6.8 Submersion Dose Due to Gaseous Vapor

This exposure pathway is caused by submersion in a cloud of gaseous isotopes that are not taken into the body. A rapid release of 100% of the package contents is assumed. The IAEA guide (IAEA 1990) concentrates entirely on releases within confined structures. No guidance is given for outside releases.

There are no gaseous vapors present in the boxes; therefore, this exposure pathway is not applicable.

4.6.9 Special Considerations

Alpha particle emitters are not of significance in the material considered in this report. The alpha particle emitters are of a low concentration, and their effect will be through the mechanism of inhalation that has been considered separately. Therefore, they are not addressed in this report. The quantity of radon present in the fuel is insignificant; therefore, radon is not addressed in this report.

The fuel (e.g., uranium, plutonium) contained in the boxes emit neutrons through (α,n) and spontaneous fission reactions. These neutron emitters will contribute to the dose received by the Hanford Site worker, but will have a negligible impact on the public receptor. A conservative estimate of the neutron dose was made using the method described in Nelson (1996). The results indicate that the neutron dose contribution is very small compared to the gamma dose. Therefore, the neutron dose was not calculated separately in this report.

Bremsstrahlung has been included in the consideration of photon effects, and the effects of short-lived daughter products have been included in all of the calculations. Where these isotopes are significant, they are assumed to be in equilibrium with their longer-lived parent isotopes.

4.6.10 Total Dose

Table B4-1 in Section 4.6 shows the dose from each exposure pathway. The table also includes the dose to each receptor, which is obtained by summing the dose contributions from each pathway.

4.7 REFERENCES

10 CFR 71, 1997, "Packaging and Transportation of Radioactive Material," *Code of Federal Regulations*, as amended.

ANS, 1991, *Neutron and Gamma-Ray Fluence-to-Dose Factors,* ANSI/ANS-6.1.1-1991, American Nuclear Society, La Grange Park, Illinois.

ANSI, 1992, *American National Standard Design Basis for Resistance to Shock and Vibration of Radioactive Material Packages Greater than One Ton in Truck Transport*, ANSI N14.23, DRAFT, American National Standards Institute, New York, New York.

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4.8 APPENDICES

4.8.1 ISO-PC Input File

```
0       2 Depleted Uranium in LSA Boxes
Dose Rate at 3 m from Unshielded Box Surface
&Input Next= 1 , IGeom= 10, OPTION=0, DUNIT=7,
 NTheta= 40 , NPsi= 40, NShld= 1 , JBuf= 1 , SFact= 1.0 ,
 Slth= 127 ,
 T(1)= 10.2 ,
 Y= 107  ,
 X= 310.2 ,
 WEIGHT(472) =   2.81E-04 ,
 WEIGHT(141) =   1.61E-03 ,
 WEIGHT(335) =   1.57E-01 ,
 WEIGHT(336) =   1.49E-01 ,
 WEIGHT(525) =   2.44E-02 ,
 WEIGHT(522) =   2.10E-01 ,
 WEIGHT(523) =   1.35E-01 ,
 WEIGHT(442) =   2.07E-01 ,
 WEIGHT(530) =   6.14E-01 ,
 WEIGHT(520) =   8.49E-01 ,
 WEIGHT(476) =   1.21E-02 ,
 WEIGHT(526) =   8.47E-01 ,
 WEIGHT(493) =   9.71E-02 ,
 WEIGHT(494) =   5.97E-03 , &
1DU     15    18.
END OF INPUT
&Input Next= 6 &
```

4.8.2 GXQ Input File

```
200 Area - Sector 99.5% X/Q Values - 100 m
c GXQ Version 4.0 Input File
c  mode
  1
c
c MODE CHOICE:
c mode = 1 then X/Q based on Hanford site specific meteorology
c mode = 2 then X/Q based on atmospheric stability class and wind speed
c mode = 3 then X/Q plot file is created
c
c LOGICAL CHOICES:
c  ifox  inorm icdf  ichk  isite ipop
   T    F    F    F    F    F
c ifox  = t then joint frequency used to compute frequency to exceed X/Q
c       = f then joint frequency used to compute annual average X/Q
c inorm = t then joint frequency data is normalized (as in GENII)
c       = f then joint frequency data is un-normalized
c icdf  = t then cumulative distribution file created (CDF.OUT)
c       = f then no cumulative distribution file created
c ichk  = t then X/Q parameter print option turned on
c       = f then no parameter print
c isite = t then X/Q based on joint frequency data for all 16 sectors
c       = f then X/Q based on joint frequency data of individual sectors
c ipop  = t then X/Q is population weighted
c       = f then no population weighting
c
c X/Q AND WIND SPEED ADJUSTMENT MODELS:
c  ipuff idep  isrc  iwind
   0    0    0    0
c DIFFUSION COEFFICIENT ADJUSTMENT MODELS:
c  iwake ipm   iflow ientr
   0    0    0    0
c EFFECTIVE RELEASE HEIGHT ADJUSTMENT MODELS:
c  (irise igrnd)iwash igrav
   0    0    0    0
c ipuff = 1 then X/Q calculated using puff model
c       = 0 then X/Q calculated using default continuous plume model
c idep  = 1 then plume depletion model turned on (Chamberlain model)
c isrc  = 1 then X/Q multiplied by scalar
c       = 2 then X/Q adjusted by wind speed function
c iwind = 1 then wind speed corrected for plume height
c isize = 1 then NRC RG 1.145 building wake model turned on
c       = 2 then MACCS virtual distance building wake model turned on
c ipm   = 1 then NRC RG 1.145 plume meander model turned on
c       = 2 then 5th Power Law plume meander model turned on
c       = 3 then sector average model turned on
c iflow = 1 then sigmas adjusted for volume flow rate
c ientr = 1 then method of Pasquill used to account for entrainment
c irise = 1 then MACCS buoyant plume rise model turned on
c       = 2 then ISC2 momentum/buoyancy plume rise model turned on
c igrnd = 1 then Mills buoyant plume rise modification for ground effects
c iwash = 1 then stack downwash model turned on
c igrav = 1 then gravitational settling model turned on
c       = 0 unless specified otherwise, 0 turns model off
c
c PARAMETER INPUT:
c             reference              frequency
c  release    anemometer   mixing       to
c  height      height      height     exceed
c  hs(m)       ha(m)       hm(m)      Cx(%)
c  ___________ ___________ ___________ ___________
  0.00000E+00  1.00000E+01  1.00000E+03  5.00000E-01
c
c  initial     initial                  gravitational
c  plume       plume      release    deposition   settling
c  width       height     duration   velocity     velocity
c  Wb(m)       Hb(m)      trd(hr)    vd(m/s)      vg(m/s)
```

```
c  __________ __________ __________ __________ __________
  0.00000E+00  0.00000E+00  0.00000E+00  1.00000E-03  1.00000E-03
c
c             initial    initial              convective
c  ambient      plume      plume     release    heat release
c  temperature  temperature  flow rate  diameter    rate(1)
c  Tamb(C)      T0(C)      V0(m3/s)   d(m)       qh(w)
c  __________ __________ __________ __________ __________
  2.00000E+01  2.20000E+01  1.00000E+00  1.00000E+00  0.00000E+00
c
c (1) If zero then buoyant flux based on plume/ambient temperature difference.
c
c  X/Q        Wind
c  scaling    Speed
c  factor     Exponent
c  c(?)       a(?)
c  __________ __________
  1.00000E+00  7.80000E-01
c
c RECEPTOR DEPENDENT DATA (no line limit)
c FOR MODE    make       RECEPTOR DEPENDENT DATA
c 1 (site specific)       sector distance receptor-height
c 2 (by class & wind speed) class windspeed distance offset receptor-height
c 3 (create plot file)     class windspeed xmax imax ymax jmax xqmin power
c
c RECEPTOR PARAMETER DESCRIPTION
c sector = 0, 1, 2... (all, S, SSW, etc.)
c distance = receptor distance (m)
c receptor height = height of receptor (m)
c class = 1, 2, 3, 4, 5, 6, 7 (P-G stability class A, B, C, D, E, F, G)
c windspeed = anemometer wind speed (m/s)
c offset = offset from plume centerline (m)
c xmax = maximum distance to plot or calculate to (m)
c imax = distance intervals
c ymax = maximum offset to plot (m)
c jmax = offset intervals
c xqmin = minimum scaled X/Q to calculate
c power = exponent in power function step size
0 100 0

400 Area - Sector 99.5% X/Q Values - 100 m
c GXQ Version 4.0 Input File
c  mode
  1
c
c MODE CHOICE:
c mode = 1 then X/Q based on Hanford site specific meteorology
c mode = 2 then X/Q based on atmospheric stability class and wind speed
c mode = 3 then X/Q plot file is created
c
c LOGICAL CHOICES:
c  ifox inorm icdf ichk isite ipop
   T   F    F    F    F    F
c ifox  = t then joint frequency used to compute frequency to exceed X/Q
c       = f then joint frequency used to compute annual average X/Q
c inorm = t then joint frequency data is normalized (as in GENII)
c       = f then joint frequency data is un-normalized
c icdf  = t then cumulative distribution file created (CDF.OUT)
c       = f then no cumulative distribution file created
c ichk  = t then X/Q parameter print option turned on
c       = f then no parameter print
c isite = t then X/Q based on joint frequency data for all 16 sectors
c       = f then X/Q based on joint frequency data of individual sectors
c ipop  = t then X/Q is population weighted
c       = f then no population weighting
c
c X/Q AND WIND SPEED ADJUSTMENT MODELS:
c  ipuff idep isrc iwind
   0    0    0    0
```

```
c DIFFUSION COEFFICIENT ADJUSTMENT MODELS:
c  iwake ipm   iflow ientr
  0    0    0    0
c EFFECTIVE RELEASE HEIGHT ADJUSTMENT MODELS:
c  (irise igrnd)iwash igrav
  0    0    0    0
c ipuff  = 1 then X/Q calculated using puff model
c        = 0 then X/Q calculated using default continuous plume model
c idep   = 1 then plume depletion model turned on (Chamberlain model)
c isrc   = 1 then X/Q multiplied by scalar
c        = 2 then X/Q adjusted by wind speed function
c iwind  = 1 then wind speed corrected for plume height
c isize  = 1 then NRC RG 1.145 building wake model turned on
c        = 2 then MACCS virtual distance building wake model turned on
c ipm    = 1 then NRC RG 1.145 plume meander model turned on
c        = 2 then 5th Power Law plume meander model turned on
c        = 3 then sector average model turned on
c iflow  = 1 then sigmas adjusted for volume flow rate
c ientr  = 1 then method of Pasquill used to account for entrainment
c irise  = 1 then MACCS buoyant plume rise model turned on
c        = 2 then ISC2 momentum/buoyancy plume rise model turned on
c igrnd  = 1 then Mills buoyant plume rise modification for ground effects
c iwash  = 1 then stack downwash model turned on
c igrav  = 1 then gravitational settling model turned on
c        = 0 unless specified otherwise, 0 turns model off
c
c PARAMETER INPUT:
c              reference                frequency
c  release      anemometer   mixing        to
c  height       height       height       exceed
c  hs(m)        ha(m)        hm(m)        Cx(%)
c  ___________  ___________  ___________  ___________
  0.00000E+00  1.00000E+01  1.00000E+03  5.00000E-01
c
c  initial      initial                   gravitational
c  plume        plume        release      deposition   settling
c  width        height       duration     velocity     velocity
c  Wb(m)        Hb(m)        trd(hr)      vd(m/s)      vg(m/s)
c  ___________  ___________  ___________  ___________  ___________
  0.00000E+00  0.00000E+00  0.00000E+00  1.00000E-03  1.00000E-03
c
c              initial      initial                   convective
c  ambient      plume        plume        release      heat release
c  temperature  temperature  flow rate    diameter     rate(1)
c  Tamb(C)      T0(C)        V0(m3/s)     d(m)         qh(w)
c  ___________  ___________  ___________  ___________  ___________
  2.00000E+01  2.20000E+01  1.00000E+00  1.00000E+00  0.00000E+00
c
c (1) If zero then buoyant flux based on plume/ambient temperature difference.
c
c  X/Q          Wind
c  scaling      Speed
c  factor       Exponent
c  c(?)         a(?)
c  ___________  ___________
  1.00000E+00  7.80000E-01
c
c RECEPTOR DEPENDENT DATA (no line limit)
c FOR MODE     make       RECEPTOR DEPENDENT DATA
c 1 (site specific)       sector distance receptor-height
c 2 (by class & wind speed) class windspeed distance offset receptor-height
c 3 (create plot file)    class windspeed xmax imax ymax jmax xqmin power
c
c RECEPTOR PARAMETER DESCRIPTION
c sector = 0, 1, 2... (all, S, SSW, etc.)
c distance = receptor distance (m)
c receptor height = height of receptor (m)
c class = 1, 2, 3, 4, 5, 6, 7 (P-G stability class A, B, C, D, E, F, G)
c windspeed = anemometer wind speed (m/s)
```

```
c offset = offset from plume centerline (m)
c xmax = maximum distance to plot or calculate to (m)
c imax = distance intervals
c ymax = maximum offset to plot (m)
c jmax = offset intervals
c xqmin = minimum scaled X/Q to calculate
c power = exponent in power function step size
0 100 0
```

4.8.3 GENII Input File

```
######################### Program GENII Input File ############ 8 Jul 88 ####
Title: FFTF Box 4 Onsite
    \SAMPL\G-AIR.AC                    Created on 01-22-1990 at 07:30
OPTIONS============================ Default
==========================================
F  Near-field scenario?    (Far-field)  NEAR-FIELD: narrowly-focused
F  Population dose?        (Individual)        release, single site
T  Acute release?          (Chronic)    FAR-FIELD: wide-scale release,
   Maximum Individual data set used               multiple sites
               Complete                          Complete
TRANSPORT OPTIONS============ Section  EXPOSURE PATHWAY OPTIONS===== Section
T Air Transport              1   F Finite plume, external      5
F Surface Water Transport    2   T Infinite plume, external    5
F Biotic Transport (near-field) 3,4  F Ground, external        5
F Waste Form Degradation (near) 3,4  F Recreation, external    5
                              T Inhalation uptake       5,6
REPORT OPTIONS========================  F Drinking water ingestion    7,8
T Report AEDE only              F Aquatic foods ingestion    7,8
F Report by radionuclide        F Terrestrial foods ingestion 7,9
F Report by exposure pathway       F Animal product ingestion    7,10
F Debug report on screen          F Inadvertent soil ingestion

INVENTORY ######################################################################

4  Inventory input activity units: (1-pCi 2-uCi 3-mCi 4-Ci 5-Bq)
0  Surface soil source units (1- m2 2- m3 3- kg)
   Equilibrium question goes here

   --------|----Release Terms------|----------Basic Concentrations---------|
   Use when|  transport selected   |  near-field scenario, optionally      |
   --------|-----------------------|---------------------------------------|
   Release |     Surface Buried |    Surface Deep   Ground  Surface|
   Radio-  |Air    Water  Waste |Air    Soil   Soil   Water   Water |
   nuclide |/yr    /yr    /m3   |/m3    /unit  /m3    /L      /L    |
   --------|------- ------- -------|------- ------- ------- ------- -------|
   CO60    2.81E-07
   TC99    1.61E-06
   CS137   1.57E-04
   BI212   2.44E-05
   RA224   2.10E-04
   TH228   2.07E-04
   TH234   6.14E-04
   U 234   8.49E-04
   U 235   1.21E-05
   U 238   8.47E-04
   PU239   9.71E-05
   PU240   5.97E-06
   --------|----Derived Concentrations-----|
   Use when|  measured values are known   |
   --------|------------------------------|
   Release |Terres. Animal  Drink   Aquatic|
   Radio-  |Plant   Product Water   Food   |
   nuclide |/kg     /kg     /L      /kg    |
   --------|------- ------- ------- -------|
```

```
TIME ##########################################################################

1   Intake ends after (yr)
50  Dose calc. ends after (yr)
1   Release ends after (yr)
0   No. of years of air deposition prior to the intake period
0   No. of years of irrigation water deposition prior to the intake period

FAR-FIELD SCENARIOS (IF POPULATION DOSE) ######################################

0        Definition option: 1-Use population grid in file POP.IN
0                           2-Use total entered on this line.

NEAR-FIELD SCENARIOS ##########################################################

         Prior to the beginning of the intake period: (yr)
0          When was the inventory disposed?  (Package degradation starts)
0          When was LOIC?  (Biotic transport starts)
0        Fraction of roots in upper soil (top 15 cm)
0        Fraction of roots in deep soil
0        Manual redistribution: deep soil/surface soil dilution factor
0        Source area for external dose modification factor (m2)
TRANSPORT #####################################################################
         ====AIR TRANSPORT====================================SECTION
1=====
              0-Calculate PM           |0      Release type (0-3)
3        Option: 1-Use chi/Q or PM value  |F   Stack release (T/F)
              2-Select MI dist & dir   |0      Stack height (m)
              3-Specify MI dist & dir  |0      Stack flow (m3/sec)
1        Chi/Q or PM value             |0      Stack radius (m)
10        MI sector index (1=S)        |0      Effluent temp. (C)
100.       MI distance from release point (m)|0   Building x-section (m2)
T        Use jf data, (T/F) else chi/Q grid|0  Building height (m)

         ====SURFACE WATER TRANSPORT==========================SECTION 2=====
0        Mixing ratio model: 0-use value, 1-river, 2-lake
0        Mixing ratio, dimensionless
0        Average river flow rate for: MIXFLG=0 (m3/s), MIXFLG=1,2 (m/s),
0        Transit time to irrigation withdrawl location (hr)
         If mixing ratio model > 0:
0          Rate of effluent discharge to receiving water body (m3/s)
0          Longshore distance from release point to usage location (m)
0          Offshore distance to the water intake (m)
0          Average water depth in surface water body (m)
0          Average river width (m), MIXFLG=1 only
0          Depth of effluent discharge point to surface water (m), lake only

         ====WASTE FORM AVAILABILITY==========================SECTION 3=====
0        Waste form/package half life, (yr)
0        Waste thickness, (m)
0        Depth of soil overburden, m

         ====BIOTIC TRANSPORT OF BURIED SOURCE================SECTION 4=====
T        Consider during inventory decay/buildup period (T/F)?
T        Consider during intake period (T/F)?  | 1-Arid non agricultural
0        Pre-Intake site condition.............| 2-Humid non agricultural
                                               | 3-Agricultural

EXPOSURE ######################################################################

         ====EXTERNAL EXPOSURE================================SECTION
5=====
         Exposure time:              |  Residential irrigation:
0          Plume (hr)                | T     Consider: (T/F)
0          Soil contamination (hr)   | 0     Source: 1-ground water
0          Swimming (hr)             |               2-surface water
0          Boating (hr)              | 0     Application rate (in/yr)
0          Shoreline activities (hr) | 0     Duration (mo/yr)
0        Shoreline type: (1-river, 2-lake, 3-ocean, 4-tidal basin)
```

```
0       Transit time for release to reach aquatic recreation (hr)
1.0     Average fraction of time submersed in acute cloud (hr/person hr)

        ====INHALATION=========================================SECTION
6=====
8766.0  Hours of exposure to contamination per year
0       0-No resus-  1-Use Mass Loading        2-Use Anspaugh model
0         pension    Mass loading factor (g/m3)  Top soil available (cm)

        ====INGESTION POPULATION==============================SECTION 7=====
0       Atmospheric production definition (select option):
0         0-Use food-weighted chi/Q, (food-sec/m3), enter value on this line
          1-Use population-weighted chi/Q
          2-Use uniform production
          3-Use chi/Q and production grids (PRODUCTION will be overridden)
0       Population ingesting aquatic foods, 0 defaults to total (person)
0       Population ingesting drinking water, 0 defaults to total (person)
F       Consider dose from food exported out of region (default=F)

        Note below: S* or Source: 0-none, 1-ground water, 2-surface water
                     3-Derived concentration entered above
        ==== AQUATIC FOODS / DRINKING WATER INGESTION=========SECTION 8====

F       Salt water? (default is fresh)

        USE      TRAN- PROD-   -CONSUMPTION- |
        ? FOOD   SIT   UCTION  HOLDUP  RATE |
        T/F TYPE hr    kg/yr   da      kg/yr |  DRINKING WATER
        --- ------ ----- ------- ------ ----- |-------------------------
        F FISH   0.00 0.0E+00  0.00   0.0 | 0   Source (see above)
        F MOLLUS 0.00 0.0E+00  0.00   0.0 | T   Treatment? T/F
        F CRUSTA 0.00 0.0E+00  0.00   0.0 | 0   Holdup/transit(da)
        F PLANTS 0.00 0.0E+00  0.00   0.0 | 0   Consumption (L/yr)

        ====TERRESTRIAL FOOD INGESTION========================SECTION 9=====

        USE      GROW  --IRRIGATION--        PROD-   --CONSUMPTION--
        ? FOOD   TIME  S RATE  TIME   YIELD  UCTION  HOLDUP  RATE
        T/F TYPE da    * in/yr mo/yr  kg/m2  kg/yr   da      kg/yr
        --- ------ ----- - ----- ----- ----- ------- ------ ------
        F LEAF V 0.00 0  0.0  0.0    0.0  0.0E+00   0.0    0.0
        F ROOT V 0.00 0  0.0  0.0    0.0  0.0E+00   0.0    0.0
        F FRUIT  0.00 0  0.0  0.0    0.0  0.0E+00   0.0    0.0
        F GRAIN  0.00 0  0.0  0.0    0.0  0.0E+00   0.0    0.0

        ====ANIMAL PRODUCTION CONSUMPTION=====================SECTION 10====

        ---HUMAN---- TOTAL  DRINK  ------------STORED FEED-------------
USE     CONSUMPTION PROD-  WATER  DIET  GROW -IRRIGATION--   STOR-
? FOOD   RATE HOLDUP UCTION CONTAM FRAC- TIME S RATE TIME  YIELD AGE
T/F TYPE kg/yr  da   kg/yr  FRACT.  TION  da  * in/yr mo/yr kg/m3 da
--- ------ ------ ----- ------ ------- ---- ---- - ----- ----- ----- -----
F BEEF    0.0  0.0  0.00   0.00 0.00  0.0 0  0.0 0.00  0.00  0.0
F POULTR  0.0  0.0  0.00   0.00 0.00  0.0 0  0.0 0.00  0.00  0.0
F MILK    0.0  0.0  0.00   0.00 0.00  0.0 0  0.0 0.00  0.00  0.0
F EGG     0.0  0.0  0.00   0.00 0.00  0.0 0  0.0 0.00  0.00  0.0
                          ------------FRESH FORAGE-----------
  BEEF                     0.00  0.0 0  0.0 0.00  0.00  0.0
  MILK                     0.00  0.0 0  0.0 0.00  0.00  0.0

##########################################################################
```

4.8.4 Checklist for Technical Peer Review

CHECKLIST FOR TECHNICAL PEER REVIEW

Document: Transportation Hazard Index (THI) Analysis for the Depleted Uranium Waste Boxes SEP," by A. V. Savino, August 12, 1997.

Scope: Entire analysis.

Yes	No	NA	
[ ]	[ ]	[X]	Previous reviews complete and cover analysis, up to scope of this review, with no gaps.
[X]	[ ]	[ ]	Problem completely defined.
[X]	[ ]	[ ]	Accident scenarios developed in a clear and logical manner.
[X]	[ ]	[ ]	Necessary assumptions explicitly stated and supported.
[X]	[ ]	[ ]	Computer codes and data files documented.
[X]	[ ]	[ ]	Data used in calculations explicitly stated in document.
[X]	[ ]	[ ]	Data checked for consistency with original source information as applicable.
[X]	[ ]	[ ]	Mathematical derivations checked including dimensional consistency of results.
[X]	[ ]	[ ]	Models appropriate and used within range of validity or use outside range of established validity justified.
[X]	[ ]	[ ]	Hand calculations checked for errors. Spreadsheet results should be treated exactly the same as hand calculations.
[X]	[ ]	[ ]	Software input correct and consistent with document reviewed.
[X]	[ ]	[ ]	Software output consistent with input and with results reported in document reviewed.
[X]	[ ]	[ ]	Limits/criteria/guidelines applied to analysis results are appropriate and referenced. Limits/criteria/guidelines checked against references.
[X]	[ ]	[ ]	Safety margins consistent with good engineering practices.
[X]	[ ]	[ ]	Conclusions consistent with analytical results and applicable limits.
[X]	[ ]	[ ]	Results and conclusions address all points required in the problem statement.
[X]	[ ]	[ ]	Format consistent with appropriate NRC Regulatory Guide or other standards
[ ]		[X]	Review calculations, comments, and/or notes are attached.
[X]	[ ]	[ ]	**Document approved.**

Janet G. McFadden Janet G. McFadden 8/18/97

Reviewer (Printed Name and Signature) Date

Any notes and/or comments should be attached.

4.8.5 HEDOP Review Checklist

HEDOP REVIEW CHECKLIST
for
Radiological and Nonradiological Release Calculations

Document: Transportation Hazard Index (THI) Analysis for the Depleted Uranium Waste Boxes SEP," by A. V. Savino, August 12, 1997.

Scope of Review: Entire Analysis

YES	NO*	N/A		
[X]	[ ]	[ ]	1.	A detailed technical review and approval of the environmental transport and dose calculation portion of the analysis has been performed and documented.
[ ]	[ ]	[X]	2.	Detailed technical review(s) and approval(s) of scenario and release determinations have been performed and documented.
[ ]	[X]	[ ]	3.	HEDOP-approved code(s) were used.
[X]	[ ]	[ ]	4.	Receptor locations were selected according to HEDOP recommendations.
[X]	[ ]	[ ]	5.	All applicable environmental pathways and code options were included and are appropriate for the calculations.
[X]	[ ]	[ ]	6.	Hanford site data were used.
[X]	[ ]	[ ]	7.	Model adjustments external to the computer program were justified and performed correctly.
[X]	[ ]	[ ]	8.	The analysis is consistent with HEDOP recommendations.
[X]		[ ]	9.	Supporting notes, calculations, comments, comment resolutions, or other information is attached. (Use the "Page 1 of X" page numbering format and sign and date each added page.)
[X]	[ ]		10.	**Approval is granted on behalf of the Hanford Environmental Dose Overview Panel.**

* All "NO" responses must be explained and use of nonstandard methods justified.

(review revised analysis) KR 8/22/97

Kathy Rhoads K Rhoads 8/19/97
HEDOP-Approved Reviewer (Printed Name and Signature) Date

COMMENTS (add additional signed and dated pages if necessary):

Item 3: GXQ used for environmental transport calculations; GENII results are included for comparison.

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5.0 SHIELDING EVALUATION

5.1 INTRODUCTION

The measured contact dose rates for the DU waste boxes are within the maximum allowables authorized by the HSRCM-1. No further evaluation is required.

5.2 REFERENCE

HSRCM-1, *Hanford Site Radiological Control Manual*, Pacific Northwest National Laboratory, Richland, Washington.

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6.0 CRITICALITY EVALUATION

6.1 INTRODUCTION

The DU waste box payload is characterized as fissile excepted per 49 CFR 173. Therefore, a criticality evaluation is not required.

6.2 REFERENCE

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

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7.0 STRUCTURAL EVALUATION

7.1 INTRODUCTION

The following sections summarize the structural evaluation for the DU waste boxes.

7.2 NORMAL TRANSPORT CONDITIONS

Based on the fact that this is a one-time shipment, a structural evaluation for NTC is not performed. See Part B, Section 4.0, for a discussion of NTC containment.

7.3 ACCIDENT CONDITIONS

Accident conditions are evaluated for the DU waste boxes by radiological risk and dose consequence analyses. The radiological risk evaluation is given in Part B, Section 3.0, of this SEP. The dose cosequence and associated THI are given in Part B, Section 4.0, of this SEP.

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8.0 THERMAL EVALUATION

8.1 INTRODUCTION

The following sections summarize the thermal evaluation for the DU waste boxes.

8.2 THERMAL SOURCE SPECIFICATION

The worst-case payload decay heat is identified as that contained in box number 5463-4, which is 0.0766 W.

8.3 THERMAL EVALUATION FOR NORMAL TRANSPORT CONDITIONS

Due to the very low internal decay heat loading of the payload, and based on the fact that this is a one-time shipment, a thermal evaluation for NTC is not required. See Part B, Section 4.0, for a discussion of NTC containment.

8.4 THERMAL EVALUATION FOR ACCIDENT CONDITIONS

Accident conditions are evaluated for the DU waste boxes by radiological risk and dose consequence analyses. The radiological risk evaluation is given in Part B, Section 3.0, of this SEP. The dose consequence and associated THI are given in Part B, Section 4.0, of this SEP.

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9.0 PRESSURE AND GAS GENERATION EVALUATION

9.1 GAS GENERATION

Hydrogen gas generation due to radiolysis requires the presence of hydrogen atoms in a molecular matrix that releases hydrogen atoms when energized with ionizing radiation. Normally this occurs when the package contains water, organic material, and polyethylene or other plastic material. The wood dunnage and small quantities of paper (< 0.45 kg [1.0 lb]) present in each container generally will not produce significant quantities of hydrogen gas when exposed to the gamma photon emissions from the radioactive constituents of the payload. In-leakage of water will not be a factor for this shipment.

Therefore, it can be concluded that no further evaluation is required.

9.2 PACKAGE PRESSURE

Package pressurization can result from radiolytic hydrogen gas generation, thermal heat buildup, and major changes in barometric pressure. As stated previously, hydrogen gas generation is not a concern. Decay heat from the payload is minimal (< 1 W); therefore, thermal heat buildup is not a concern. No major changes in barometric pressure will occur during this shipment; therefore, pressure buildup via this mechanism is also not a concern.

No further evaluation is required.

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10.0 TIEDOWN EVALUATION

10.1 SYSTEM DESIGN

This one-time shipment will employ an engineered tiedown system compliant with the DOT transportation regulations found in 49 CFR 393, Subpart I. No further evaluation is required. Section 10.4 provides the analyses that demonstrate the tiedowns meet the acceptance criteria.

10.2 TIEDOWN SYSTEM

The tiedown system used is a system of blocking, bracing, and strapping as described in Section 10.4.

10.3 REFERENCE

49 CFR 393, 1997, "Parts and Accessories Necessary for Safe Operation," *Code of Federal Regulations*, as amended.

10.4 APPENDIX: ENGINEERING ANALYSIS

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ENGINEERING SAFETY EVALUATION

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Section Chief: P. C. Ferrell	Date 08-22-97

1.0 OBJECTIVE

The objective of these analyses is to show that the engineered tiedown system for the depleted uranium waste boxes meets the requirements of 49 CFR 393.102 for tiedown devices. This regulation requires that the tiedown working load limit not be exceeded when subjected to a load of one-half times the weight of the packaging for a given configuration.

2.0 REFERENCES

49 CFR 393, 1997, "Parts and Accessories Necessary for Safe Operation," *Code of Federal Regulations*, as amended.

3.0 ASSUMPTIONS, RESULTS, AND CONCLUSIONS

3.1 ASSUMPTIONS

The following assumptions are made to the positioning, orientation, blocking and bracing, and strapping of the depleted uranium waste boxes on the transporter.

- The boxes will be placed with the long axis of the package in the longitudinal direction of the trailer centered along the approximate longitudinal centerline of the trailer.
- Two horizontal tiedowns will be used for each box (one started from each side of the trailer), which together will completely block the container on all sides.
- Each horizontal tiedown used shall cross over itself or meet at the same point on the same side of the trailer for attachment. If the horizontal tiedown crosses over itself for attachment, the attachment point shall not be further along the side of the trailer than the lateral edge of the container.
- For each horizontal tiedown, the cross-over attachment point distances will be approximately equi-distant from the centerline of the container.
- The minimum working load limit for each horizontal tiedown is 7,000 lbf.

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- One vertical tiedown with a 3,000-lb minimum working load limit will be utilized on each box to prevent movement in the vertical direction.
- 4x4 lumber with 2x2 landings attached will be employed for horizontal tiedown standoff on the longitudinal edges of each container.
- The tiedown attachment devices are at least as strong as the tiedowns in the direction they are employed.
- Box 5463-11 will be blocked and braced in a similar manner to the other boxes, and the tiedowns do not require analysis for loadings in the lateral and longitudinal directions. This is due to the bounding geometry and the weight of the other boxes.
- Tipping of the box is not a concern due to geometry.

3.2 RESULTS

The analyses in Section 4.0 show that a positive factor of safety is maintained when the tiedowns are subject to applied loadings of one-half the weight of each package in the lateral, longitudinal, and vertical directions, applied nonsimultaneously.

3.3 CONCLUSIONS

The tiedown system, as analyzed, meets the requirements of 49 CFR 393.102.

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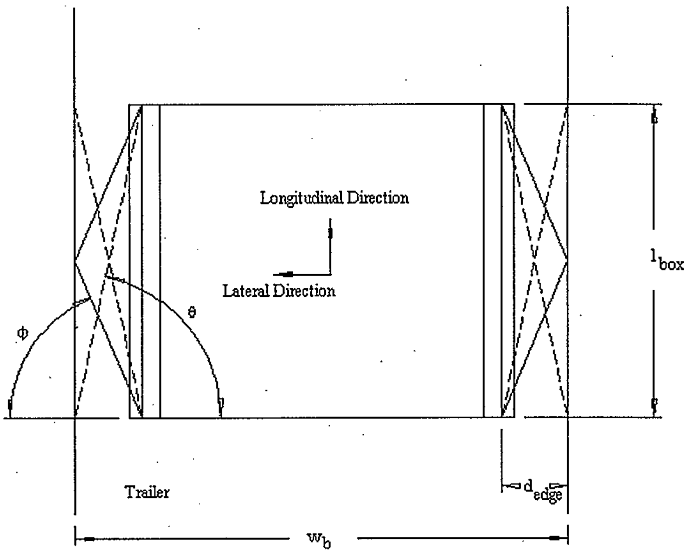
4.0 EVALUATION

Evaluation of DU Boxes:



Lateral and Longitudinal Loading (Boxes 5463-1 thru 5463-9):

Width of trailer: $w_b := 8 \cdot ft$ Distance from edge of box to edge of trailer: $d_{edge} := 22 \cdot in$

Maximum weight of box: $W_b := 5936 \cdot lbf$ Length of box: $l_{box} := 53 \cdot in$

DOT loading requirement for securement: $g_{fac} := 0.5$

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For tiedowns crossed over each other:

Horizontal angle: $\theta := \mathrm{atan}\left(\frac{l_{box}}{d_{edge}}\right)$ $\theta = 67.46 \cdot deg$

Lateral loading tension on tiedown leg: $T_{lat} := \frac{g_{fac} \cdot W_b}{2 \cdot \cos(\theta)}$ $T_{lat} = 3871 \cdot lbf$

Longitudinal loading tension on tiedown leg: $T_{long} := \frac{g_{fac} \cdot W_b}{2 \cdot \sin(\theta)}$ $T_{long} = 1607 \cdot lb$

Assuming tiedown working capacity is 7,000 lb: Factor of Safety: $T_{cap} := 7000 \cdot lbf$

$$SF_1 := \frac{T_{cap}}{T_{lat}} \qquad SF_1 = 1.8$$

For tiedowns meeting on centerline:

Horizontal angle: $\phi := \mathrm{atan}\left[\frac{\frac{l_{box}}{2}}{d_{edge}}\right]$ $\phi = 50 \cdot deg$

Lateral loading tension on tiedown leg: $T_{lat} := \frac{g_{fac} \cdot W_b}{2 \cdot \cos(\phi)}$ $T_{lat} = 2323 \cdot lbf$

Longitudinal loading tension on tiedown leg: $T_{long} := \frac{g_{fac} \cdot W_b}{2 \cdot \sin(\phi)}$ $T_{long} = 1929 \cdot lb$

Assuming tiedown working capacity is 7,000 lb: Factor of Safety: $T_{cap} := 7000 \cdot lbf$

$$SF_1 := \frac{T_{cap}}{T_{lat}} \qquad SF_1 = 3$$

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Lateral and Longitudinal Loading (Boxes 5463-10):

Width of trailer: $w_b := 8 \cdot ft$ Distance from edge of box to edge of trailer: $d_{edge} := 12.5 \cdot in$

Maximum weight of box: $W_b := 4135\,lbf$ Length of box: $l_{box} := 72 \cdot in$

DOT loading requirement for securement: $g_{fac} := 0.5$

For tiedowns crossed over each other:

Horizontal angle: $\theta := \operatorname{atan}\left(\frac{l_{box}}{d_{edge}}\right)$ $\theta = 80 \cdot deg$

Lateral loading tension on tiedown leg: $T_{lat} := \frac{g_{fac} \cdot W_b}{2 \cdot \cos(\theta)}$ $T_{lat} = 6043 \cdot lbf$

Longitudinal loading tension on tiedown leg: $T_{long} := \frac{g_{fac} \cdot W_b}{2 \cdot \sin(\theta)}$ $T_{long} = 1049 \cdot lbf$

Assuming tiedown working capacity is 7,000 lb: Factor of Safety: $T_{cap} := 7000\,lbf$

$$SF_1 := \frac{T_{cap}}{T_{lat}} \qquad SF_1 = 1.2$$

For tiedowns meeting on centerline:

Horizontal angle: $\phi := \operatorname{atan}\left[\frac{\frac{l_{box}}{2}}{d_{edge}}\right]$ $\phi = 71 \cdot deg$

Lateral loading tension on tiedown leg: $T_{lat} := \frac{g_{fac} \cdot W_b}{2 \cdot \cos(\phi)}$ $T_{lat} = 3152 \cdot lbf$

Longitudinal loading tension on tiedown leg: $T_{long} := \frac{g_{fac} \cdot W_b}{2 \cdot \sin(\phi)}$ $T_{long} = 1094 \cdot lbf$

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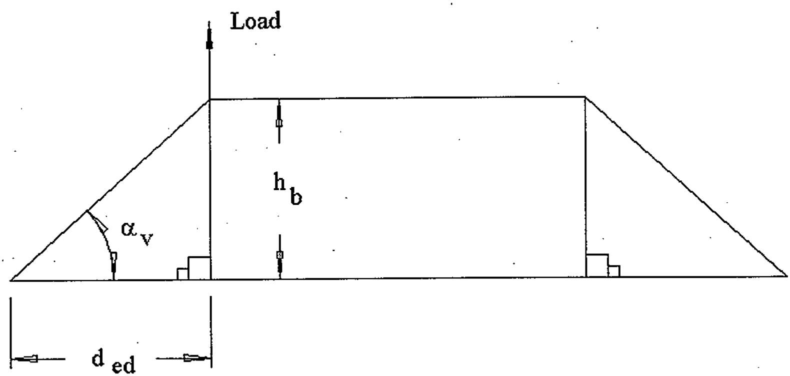
Assuming tiedown working capacity is 7,000 lb: Factor of Safety: $T_{cap} := 7000\,\mathrm{lbf}$

$$SF_1 := \frac{T_{cap}}{T_{lat}} \qquad SF_1 = 2.2$$

Vertical loading (Boxes 5463-1 thru 5463-9):



Height of box: $h_b := 13.25\,\mathrm{in}$ Distance from edge: $d_{ed} := 25.5\,\mathrm{in}$

Vertical angle: $\alpha_v := \mathrm{atan}\left(\frac{h_b}{d_{ed}}\right)$ $\alpha_v = 27.5 \cdot \mathrm{deg}$

Vertical loading tension in tiedown: $T_v := \frac{g_{fac} \cdot W_b}{2 \cdot \sin(\alpha_v)}$ $T_v = 2242 \cdot \mathrm{lbf}$

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Assuming tiedown working capacity is 3,000 lb: Factor of Safety: $T_{cap} := 3000 \cdot lb$

$$SF_1 := \frac{T_{cap}}{T_v} \qquad SF_1 = 1.3$$

Vertical loading (Boxes 5463-10):

Height of box: $h_b := 14 \cdot in$ Distance from edge: $d_{ed} := 16 \cdot in$

Vertical angle: $\alpha_v := \operatorname{atan}\left(\frac{h_b}{d_{ed}}\right)$ $\alpha_v = 41.2 \cdot deg$

Vertical loading tension in tiedown: $T_v := \frac{g_{fac} \cdot W_b}{2 \cdot \sin(\alpha_v)}$ $T_v = 1570 \cdot lbf$

Assuming tiedown working capacity is 3,000 lb: Factor of Safety: $T_{cap} := 3000 \cdot lb$

$$SF_1 := \frac{T_{cap}}{T_v} \qquad SF_1 = 1.9$$

Vertical loading (Boxes 5463-11):

Height of box: $h_b := 16.5 \cdot in$ Distance from edge: $d_{ed} := 22.5 \cdot in$

Vertical angle: $\alpha_v := \operatorname{atan}\left(\frac{h_b}{d_{ed}}\right)$ $\alpha_v = 36.3 \cdot deg$

Vertical loading tension in tiedown: $T_v := \frac{g_{fac} \cdot W_b}{2 \cdot \sin(\alpha_v)}$ $T_v = 1748 \cdot lbf$

Assuming tiedown working capacity is 3,000 lb: Factor of Safety: $T_{cap} := 3000 \cdot lb$

$$SF_1 := \frac{T_{cap}}{T_v} \qquad SF_1 = 1.7$$

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DISTRIBUTION SHEET

To	From	Page 1 of 1
Distribution	Packaging Engineering	Date Aug. 21, 1997
Project Title/Work Order		EDT No. 621891
Safety Evaluation for Packaging (Onsite) Depleted Uranium Waste Boxes (HNF-SD-TP-SEP-066 Rev. 0)		ECN No. N/A

Name	MSIN	Text With All Attach.	Text Only	Attach./ Appendix Only	EDT/ECN Only
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T. A. Dillhoff	N2-57	X			
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