APPENDIX I

CASK DETAILS AND CERTIFICATES OF COMPLIANCE

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I.1 Cask Descriptions

This appendix provides a listing and brief description of the spent fuel transport casks that were considered for evaluation in this risk analysis. Also provided are the certificates of compliance for those casks selected for evaluation.

I.1.1 Truck Casks

GA-4	The Steel-DU-Steel design is stiffer than lead casks and has smaller deformations.
	The 4 PWR assembly capacity of this cask makes it the likely workhorse truck cask for any large transportation campaign.
	Elastomeric seals (ethylene propylene) allow larger closure deformations before leakage.
	Truck casks have polymer neutron shielding.
	Larger capacity allows for larger radioactive material inventory and possible larger consequences from an accident.
	The design is from the late 80s; General Atomics used finite element analyses in certification.
	The DU shielding is made from 5 segments, which could possibly result in segment-to-segment problems.
	The cask body has a square cross-section, which provides more possible orientations.
	The cask has an Aluminum honeycomb impact limiter.
NAC-LWT	The steel-lead-steel design is relatively flexible, which should result in plastic deformation of the body before seal failure.
	The NAC-LWT contains either a single PWR assembly or two BWR assemblies.
	The cask has both elastomeric and metallic seals. The low compression of the elastomeric seal (metallic is primary) allows little closure movement before leakage but may have better performance in a fire.
	The lead shielding could melt during severe fires, leading to loss of shielding.
	With liquid neutron shielding, the tank is likely to fail in extra-regulatory impacts.
	The bottom end impact limiter is attached to the neutron shielding tank, making side drop analysis more difficult.
	The NAC-LWT has an aluminum honeycomb impact limiter.
	The cask is very similar to the generic steel-lead-steel cask from NUREG 6672.
	The cask is being used for Foreign Research Reactor shipments.

I.1.2 Rail Casks

NAC-STC	The cask has a steel-lead-steel design, which is relatively flexible and should result in plastic deformation of the body before seal failure
	The NAC-STC is certified for both direct loaded fuel and for fuel in a welded canister.
	The cask can contain either 26 directly loaded PWR assemblies or 1 Transportable Storage Container (3 configurations, all for PWR fuel).
	The cask can have either elastomeric or metallic seals. A configuration must be chosen for analysis.
	The lead shielding used could melt during severe fires, leading to loss of shielding.
	The NAC-STC has polymer neutron shielding.
	The cask has a wood impact limiter (redwood and balsa).
	This cask is similar to the steel-lead-steel rail cask from NUREG 6672.
	Two casks have been built and are being used outside of the U.S.
NAC-UMS	The NAC –UMS has a steel-lead-steel design, which is relatively flexible and should result in plastic deformation of the body before seal failure.
	The fuel is in a welded canister.
	Baskets for 24 PWR assemblies or 56 BWR assemblies are available.
	Elastomeric seals allow larger closure deformations before leakage.
	The lead shielding could melt during severe fires, leading to loss of shielding.
	The cask has polymer neutron shielding.
	The cask has a wood impact limiter (redwood and balsa).
	The cask is similar to the steel-lead-steel rail cask from NUREG 6672.
	The NAC-UMS cask has never been built.
HI-STAR 100	The HI-STAR 100 cask has a layered all-steel design.
	The fuel is in a welded canister.
	Baskets for 24 PWR assemblies or 68 BWR assemblies are available.
	The cask has metallic seals, resulting in smaller closure deformations before leakage.
	The cask has polymer neutron shielding.
	The cask has aluminum honeycomb impact limiters.
	At least 7 of these casks have been built and are being used for dry storage; no impact limiters have been built.
	The HI-STAR 100 is proposed as the transportation cask for the Private Fuel Storage facility (PFS).

TN-68	The TN-68 has a layered all-steel design.
	Directly loaded fuel is used in the cask.
	The TN-68 has 68 BWR assemblies.
	Metallic seals result in smaller closure deformations before leakage.
	The cask has polymer neutron shielding.
	The cask h as a wood impact limiter (redwood and balsa).
	At least 24 TN-68 casks have been built and are being used for dry storage; no impact limiters have been built.
MP-187	The MP-187 has a steel-lead-steel design, which is relatively flexible and should result in plastic deformation of the body before seal failure.
	The fuel in a welded canister.
	There are 24 PWR assemblies.
	Metallic seals result in smaller closure deformations before leakage.
	The MP-187 has hydrogenous neutron shielding.
	The cask has aluminum honeycomb/polyurethane foam impact limiters (chamfered rectangular parallelepiped).
	This cask has never been built
MP-197	The MP-197 has a steel-lead-steel design, which is relatively flexible and should result in plastic deformation of the body before seal failure.
	The fuel in a welded canister.
	There are 61 BWR assemblies.
	Elastomeric seals allow larger closure deformations before leakage.
	The MP-197 has hydrogenous neutron shielding.
	The cask has a wood impact limiter (redwood and balsa).
	This cask has never been built.
TS125	The TS125 has a steel-lead-steel design, which is relatively flexible and should result in plastic deformation of the body before seal failure.
	The fuel in a welded canister.
	There are basket designs for 21 PWR assemblies or 64 BWR assemblies.
	Metallic seals result in smaller closure deformations before leakage.
	The TS125 has polymer neutron shielding.
	The cask has aluminum honeycomb impact limiters.
	This cask has never been built.

I.2 Certificates of Compliance

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9261	7	71-9261	USA/9261/B(U)F-96	1	OF	7

PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

ISSUED TO (*Name and Address*)

Holtec International
Holtec Center
555 Lincoln Drive West
Marlton, NJ 08053

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)* Revision 12, dated October 9, 2006, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: HI-STAR 100 System

(2) Description

The HI-STAR 100 System is a canister system comprising a Multi-Purpose Canister (MPC) inside of an overpack designed for both storage and transportation (with impact limiters) of irradiated nuclear fuel. The HI-STAR 100 System consists of interchangeable MPCs that house the spent nuclear fuel and an overpack that provides the containment boundary, helium retention boundary, gamma and neutron radiation shielding, and heat rejection capability. The outer diameter of the overpack of the HI-STAR 100 is approximately 96 inches without impact limiters and approximately 128 inches with impact limiters. Maximum gross weight for transportation (including overpack, MPC, fuel, and impact limiters) is 282,000 pounds. Specific tolerances germane to the safety analyses are called out in the drawings listed below. The HI-STAR 100 System includes the HI-STAR 100 Version HB (also referred to as the HI-STAR HB).

Multi-Purpose Canister

There are seven Multi-Purpose Canister (MPC) models designated as the MPC-24, MPC-24E, MPC-24EF, MPC-32,MPC-68, MPC-68F, and the MPC-HB. All MPCs are designed to have identical exterior dimensions, except 1) MPC-24E/EFs custom-designed for the Trojan plant, which are approximately nine inches shorter than the generic Holtec MPC design; and 2) MPC-HBs custom-designed for the Humboldt Bay plant, which are approximately 6.3 feet shorter than the generic Holtec MPC designs The two digits after the MPC designate the number of reactor fuel assemblies for which the respective MPCs are designed. The MPC-24 series is designed to contain up to 24 Pressurized Water Reactor (PWR) fuel assemblies; the MPC-32 is designed to contain up to 32 intact PWR assemblies; and the MPC-68 and MPC-68F are designed to contain up to 68 Boiling Water Reactor (BWR) fuel assemblies. The MPC-HB is designed to contain up to 80 Humboldt Bay BWR fuel assemblies

The HI-STAR 100 MPC is a welded cylindrical structure with flat ends Each MPC is an assembly consisting of a honeycombed fuel basket, baseplate, canister shell, lid, and closure ring The outer diameter and cylindrical height of each generic MPC is fixed. The outer diameter of the Trojan MPCs is the same as the generic MPC, but the height is approximately nine inches shorter than the generic MPC design. A steel spacer is used with the Trojan plant MPCs to ensure the MPC-overpack interface is bounded by the generic design The outer diameter of the Humboldt Bay MPCs is the same as the generic MPC, but the height is approximately 6.3 feet shorter than the generic MPC design. The Humboldt Bay MPCs are transported in a shorter version of the HI-STAR overpack, designated as the HI-STAR HB. The fuel basket designs vary based on the MPC model.

Overpack

The HI-STAR 100 overpack is a multi-layer steel cylinder with a welded baseplate and bolted lid (closure plate). The inner shell of the overpack forms an internal cylindrical cavity for housing the MPC. The outer surface of the overpack inner shell is buttressed with intermediate steel shells for radiation shielding. The overpack closure plate incorporates a dual O-ring design to ensure its containment function. The containment system consists of the overpack inner shell, bottom plate, top flange, top closure plate, top closure inner O-ring seal, vent port plug and seal, and drain port plug and seal.

Impact Limiters

The HI-STAR 100 overpack is fitted with two impact limiters fabricated of aluminum honeycomb completely enclosed by an all-welded austenitic stainless steel skin. The two impact limiters are attached to the overpack with 20 and 16 bolts at the top and bottom, respectively.

(3) Drawings

The package shall be constructed and assembled in accordance with the following drawings or figures in Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)*, Revision 12, as supplemented:

5 (a)(3) Drawings (continued)

(a) HI-STAR 100 Overpack	Drawing 3913, Sheets 1-9, Rev. 9
(b) MPC Enclosure Vessel	Drawing 3923, Sheets 1-5, Rev 16
(c) MPC-24E/EF Fuel Basket	Drawing 3925, Sheets 1-4, Rev. 5
(d) MPC-24 Fuel Basket Assembly	Drawing 3926, Sheets 1-4, Rev 5
(e) MPC-68/68F/68FF Fuel Basket	Drawing 3928, Sheets 1-4, Rev 5
(f) HI-STAR 100 Impact Limiter	Drawing C1765, Sheet 1, Rev. 4; Sheet 2, Rev. 3; Sheet 3, Rev. 4, Sheet 4, Rev. 4; Sheet 5, Rev. 2; Sheet 6, Rev. 3; and Sheet 7, Rev 1.
(g) HI-STAR 100 Assembly for Transport	Drawing 3930, Sheets 1-3, Rev. 2
(h) Trojan MPC-24E/EF Spacer Ring	Drawing 4111, Sheets 1-2, Rev. 0
(i) Damaged Fuel Container for Trojan Plant SNF	Drawing 4119, Sheet 1-4, Rev. 1
(j) Spacer for Trojan Failed Fuel Can	Drawing 4122, Sheets 1-2, Rev. 0
(k) Failed Fuel Can for Trojan	SNC Drawings PFFC-001, Rev. 8 and PFFC-002 Sheets 1 and 2, Rev. 7
(l) MPC-32 Fuel Basket Assembly	Drawing 3927, Sheets 1-4, Rev. 6
(m) HI-STAR HB Overpack	Drawing 4082, Sheets 1-7, Rev. 3
(n) MPC-HB Enclosure Vessel	Drawing 4102, Sheets 1-4, Rev. 1
(o) MPC-HB Fuel Basket	Drawing 4103, Sheets 1-3, Rev. 5
(p) Damaged Fuel Container HB	Drawing 4113, Sheets 1-2, Rev. 1

5.(b) Contents

(1) Type, Form, and Quantity of Material

(a) Fuel assemblies meeting the specifications and quantities provided in Appendix A to this Certificate of Compliance and meeting the requirements provided in Conditions 5.b(1)(b) through 5.b(1)(i) below are authorized for transportation.

5.(b)(1) Type, Form, and Quantity of Material (continued)

(b) The following definitions apply:

Damaged Fuel Assemblies are fuel assemblies with known or suspected cladding defects, as determined by review of records, greater than pinhole leaks or hairline cracks, empty fuel rod locations that are not filled with dummy fuel rods, missing structural components such as grid spacers, whose structural integrity has been impaired such that geometric rearrangement of fuel or gross failure of the cladding is expected based on engineering evaluations, or that cannot be handled by normal means. Fuel assemblies that cannot be handled by normal means due to fuel cladding damage are considered FUEL DEBRIS.

Damaged Fuel Containers (or Canisters) (DFCs) are specially designed fuel containers for damaged fuel assemblies or fuel debris that permit gaseous and liquid media to escape while minimizing dispersal of gross particulates.

The DFC designs authorized for use in the HI-STAR 100 are shown in Figures 1.2.10, 1.2.11, and 1.I.1 of the HI-STAR 100 System SAR, Rev. 12, as supplemented.

Fuel Debris is ruptured fuel rods, severed rods, loose fuel pellets, and fuel assemblies with known or suspected defects which cannot be handled by normal means due to fuel cladding damage, including containers and structures supporting these parts. Fuel debris also includes certain Trojan plant-specific fuel material contained in Trojan Failed Fuel Cans.

Incore Grid Spacers are fuel assembly grid spacers located within the active fuel region (i.e., not including top and bottom spacers).

Intact Fuel Assemblies are fuel assemblies without known or suspected cladding defects greater than pinhole leaks or hairline cracks and which can be handled by normal means. Fuel assemblies without fuel rods in fuel rod locations shall not be classified as intact fuel assemblies unless dummy fuel rods are used to displace an amount of water greater than or equal to that displaced by the original fuel rod(s). Trojan fuel assemblies not loaded into DFCs or FFCs are classified as intact assemblies.

Minimum Enrichment is the minimum assembly average enrichment. Natural uranium blankets are not considered in determining minimum enrichment.

Non-Fuel Hardware is defined as Burnable Poison Rod Assemblies (BPRA), Thimble Plug Devices (TPDs), and Rod Cluster Control Assemblies (RCCAs).

Planar-Average Initial Enrichment is the average of the distributed fuel rod initial enrichments within a given axial plane of the assembly lattice.

5.(b)(1)(b) Definitions (continued)

Trojan Damaged Fuel Containers (or Canisters) are Holtec damaged fuel containers custom-designed for Trojan plant damaged fuel and fuel debris as depicted in Drawing 4119, Rev. 1

Trojan Failed Fuel Cans are non-Holtec designed Trojan plant-specific damaged fuel containers that may be loaded with Trojan plant damaged fuel assemblies, Trojan fuel assembly metal fragments (e.g., portions of fuel rods and grid assemblies, bottom nozzles, etc.), a Trojan fuel rod storage container, a Trojan Fuel Debris Process Can Capsule, or a Trojan Fuel Debris Process Can. The Trojan Failed Fuel Can is depicted in Drawings PFFC-001, Rev. 8 and PFFC-002, Rev. 7.

Trojan Fuel Debris Process Cans are Trojan plant-specific canisters containing fuel debris (metal fragments) and were used to process organic media removed from the Trojan plant spent fuel pool during cleanup operations in preparation for spent fuel pool decommissioning. Trojan Fuel Debris Process Cans are loaded into Trojan Fuel Debris Process Can Capsules or directly into Trojan Failed Fuel Cans. The Trojan Fuel Debris Process Can is depicted in Figure 1.2.10B of the HI-STAR100 System SAR, Rev. 12, as supplemented.

Trojan Fuel Debris Process Can Capsules are Trojan plant-specific canisters that contain up to five Trojan Fuel Debris Process Cans and are vacuumed, purged, backfilled with helium and then seal-welded closed. The Trojan Fuel Debris Process Can Capsule is depicted in Figure 1.2.10C of the HI-STAR 100 System SAR, Rev. 12, as supplemented.

Undamaged Fuel Assemblies are fuel assemblies where all the exterior rods in the assembly are visually inspected and shown to be intact. The interior rods of the assembly are in place; however, the cladding of these rods is of unknown condition. This definition only applies to Humboldt Bay fuel assembly array/class 6x6D and 7x7C.

ZR means any zirconium-based fuel cladding materials authorized for use in a commercial nuclear power plant reactor.

(c) For MPCs partially loaded with stainless steel clad fuel assemblies, all remaining fuel assemblies in the MPC shall meet the more restrictive of the decay heat limits for the stainless steel clad fuel assemblies or the applicable ZR clad fuel assemblies.

(d) For MPCs partially loaded with damaged fuel assemblies or fuel debris, all remaining ZR clad intact fuel assemblies in the MPC shall meet the more

5.(b)(1)(b) Definitions (continued)

restrictive of the decay heat limits for the damaged fuel assemblies or the intact fuel assemblies

(e) For MPC-68s partially loaded with array/class 6x6A, 6x6B, 6x6C, or 8x8A fuel assemblies, all remaining ZR clad intact fuel assemblies in the MPC shall meet the more restrictive of the decay heat limits for the 6x6A, 6x6B, 6x6C, and 8x8A fuel assemblies or the applicable Zircaloy clad fuel assemblies

(f) PWR non-fuel hardware and neutron sources are not authorized for transportation except as specifically provided for in Appendix A to this CoC.

(g) BWR stainless-steel channels and control blades are not authorized for transportation.

(h) For spent fuel assemblies to be loaded into MPC-32s, core average soluble boron, assembly average specific power, and assembly average moderator temperature in which the fuel assemblies were irradiated, shall be determined according to Section 1.2.3.7.1 of the SAR, and the values shall be compared against the limits specified in Part VI of Table A.1 in Appendix A of this Certificate of Compliance.

(i) For spent fuel assemblies to be loaded into MPC-32s, the reactor records on spent fuel assemblies average burnup shall be confirmed through physical burnup measurements as described in Section 1.2.3.7.2 of the SAR.

5.(c) Criticality Safety Index (CSI)= 0.0

6. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) Each package shall be both prepared for shipment and operated in accordance with detailed written operating procedures. Procedures for both preparation and operation shall be developed. At a minimum, those procedures shall include the provisions provided in Chapter 7 of the HI-STAR SAR.

(b) All acceptance tests and maintenance shall be performed in accordance with detailed written procedures. Procedures for acceptance testing and maintenance shall be developed and shall include the provisions provided in Chapter 8 of the HI-STAR SAR.

7. The maximum gross weight of the package as presented for shipment shall not exceed 282,000 pounds, except for the HI-STAR HB, where the gross weight shall not exceed 187,200 pounds.

8. The package shall be located on the transport vehicle such that the bottom surface of the bottom impact limiter is at least 9 feet (along the axis of the overpack) from the edge of the vehicle.

9 The personnel barrier shall be installed at all times while transporting a loaded overpack.

10 The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71 17

11 Transport by air of fissile material is not authorized

12 Revision No 6 of this certificate may be used until May 31, 2010

13 Expiration Date: March 31, 2014

Attachment Appendix A

REFERENCES:

Holtec International Report No. HI-951251, *Safety Analysis Report for the Holtec International Storage, Transport, And Repository Cask System (HI-STAR 100 Cask System)*, Revision 12, dated October 9, 2006.

Holtec International supplements dated June 29, July 27, August 3, September 27, October 5, and December 18, 2007; January 9, March 19 and September 30, 2008; and February 27, 2009.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Eric J. Benner, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: May 8, 2009

APPENDIX A

CERTIFICATE OF COMPLIANCE NO. 9261, REVISION 7

MODEL NO. HI-STAR 100 SYSTEM

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Page	Table.	Description:
Page A-1 to A-23	Table A.1	Fuel Assembly Limits
Page A-1		MPC-24: Uranium oxide, PWR intact fuel assemblies listed in Table A.2.
A-2		MPC-68: Uranium oxide, BWR intact fuel assemblies listed in Table A.3 with or without Zircaloy channels
A-3		MPC-68: Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6X6C, 7x7A, or 8x8A
A-4		MPC-68: Mixed oxide (MOX), BWR intact fuel assemblies, with or without Zircaloy channels. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B.
A-5		MPC-68: Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. MOX BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B.
A-6		MPC-68: Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters
A-7		MPC-68F: Uranium oxide, BWR intact fuel assemblies, with or without Zircaloy channels. Uranium oxide BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A.
A-8		MPC-68F: Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A.
A-9		MPC-68F: Uranium oxide, BWR fuel debris, with or without Zircaloy channels, placed in damaged fuel containers. The original fuel assemblies for the uranium oxide BWR fuel debris shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A.

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Page:	Table:	Description.
A-10	Table A. 1 (Cont'd)	MPC-68F: Mixed oxide (MOX), BWR intact fuel assemblies, with or without Zircaloy channels. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B.
A-11		MPC-68F. Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. MOX BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B.
A-12		MPC-68F: Mixed Oxide (MOX), BWR fuel debris, with or without Zircaloy channels, placed in damaged fuel containers. The original fuel assemblies for the MOX BWR fuel debris shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B.
A-13		MPC-68F: Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters.
A-15		MPC-24E: Uranium oxide, PWR intact fuel assemblies listed in Table A.2.
A-16		MPC-24E: Trojan plant damaged fuel assemblies.
A-17		MPC-24EF: Uranium oxide, PWR intact fuel assemblies listed in Table A.2.
A-18		MPC-24EF: Trojan plant damaged fuel assemblies.
A-19		MPC-24EF: Trojan plant Fuel Debris Process Can Capsules and/or Trojan plant fuel assemblies classified as fuel debris.
A-20 to A-21		MPC-32: Uranium oxide, PWR intact fuel assemblies in array classes 15X15D, E, F, and H and 17X17A, B, and C as listed in Table A.2.
A-22 to A-23		MPC-HB: Uranium oxide, intact and/or undamaged fuel assemblies and damaged fuel assemblies, with or without channels, meeting the criteria specified in Table A.3 for fuel assembly array/class 6x6D or 7x7C.
A-24 to A-27	Table A.2	PWR Fuel Assembly Characteristics
A-28 to A-33	Table A.3	BWR Fuel Assembly Characteristics

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Page:	Table:	Description:
A-34	Table A.4	Fuel Assembly Cooling, Average Burnup, and Initial Enrichment MPC-24/24E/24EF PWR Fuel with Zircaloy Clad and with Non-Zircaloy In-Core Grid Spacers
A-34	Table A.5	Fuel Assembly Cooling, Average Burnup, and Initial Enrichment MPC-24/24E/24EF PWR Fuel with Zircaloy clad and with Zircaloy In-Core Grid Spacers
A-35	Table A.6	Fuel Assembly Cooling, Average Burnup, and Initial Enrichment MPC-24/24E/24EF PWR Fuel with Stainless Steel Clad.
A-35	Table A.7	Fuel Assembly Cooling, Average Burnup, and Initial Enrichment-MPC-68.
A-36	Table A.8	Trojan Plant Fuel Assembly Cooling, Average Burnup, and Initial Enrichment Limits.
A-36	Table A.9	Trojan Plant Non-Fuel Hardware and Neutron Source Cooling and Burnup Limits.
A-37	Table A.10	Fuel Assembly Cooling, Average Burnup, and Minimum Enrichment MPC-32 PWR Fuel with Zircaloy Clad and with Non-Zircaloy In-Core Grid Spacers.
A-37	Table A.11	Fuel Assembly Cooling, Average Burnup, and Minimum Enrichment MPC-32 PWR Fuel with Zircaloy Clad and with Zircaloy In-Core Grid Spacers.
A-38	Table A.12	Fuel Assembly Maximum Enrichment and Minimum Burnup Requirement for Transportation in MPC-32.
A-39	Table A.13	Loading Configurations for the MPC-32.
A-40		References.

Table A.1 (Page 1 of 23)
Fuel Assembly Limits

MPC MODEL: MPC-24

A. Allowable Contents

1. Uranium oxide, PWR intact fuel assemblies listed in Table A.2 and meeting the following specifications

a. Cladding type:	ZR or stainless steel (SS) as specified in Table A.2 for the applicable fuel assembly array/class
b. Maximum initial enrichment.	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly	
i. ZR clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.4 or A.5, as applicable.
ii. SS clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.6, as applicable.
d. Decay heat per assembly:	
i. ZR Clad:	≤833 Watts
ii. SS Clad:	≤488 Watts
e. Fuel assembly length:	≤ 176.8 inches (nominal design)
f. Fuel assembly width:	≤ 8.54 inches (nominal design)
g. Fuel assembly weight:	≤ 1,680 lbs

B. Quantity per MPC: Up to 24 PWR fuel assemblies.

C. Fuel assemblies shall not contain non-fuel hardware or neutron sources.

D. Damaged fuel assemblies and fuel debris are not authorized for transport in the MPC-24.

E. Trojan plant fuel is not permitted to be transported in the MPC-24.

II MPC MODEL: MPC-68

A Allowable Contents

1 Uranium oxide, BWR intact fuel assemblies listed in Table A.3, except assembly classes 6x6D and 7x7C, with or without Zircaloy channels, and meeting the following specifications:

a Cladding type	ZR or stainless steel (SS) as specified in Table A.3 for the applicable fuel assembly array/class
b Maximum planar-average initial enrichment.	As specified in Table A.3 for the applicable fuel assembly array/class.
c Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	
i. ZR clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.7, except for (1) array/class 6x6A, 6x6C, 7x7A, and 8x8A fuel assemblies, which shall have a cooling time ≥ 18 years, an average burnup ≤ 30,000 MWD/MTU, and a minimum initial enrichment ≥ 1.45 wt% ^{235}U, and (2) array/class 8x8F fuel assemblies, which shall have a cooling time ≥ 10 years, an average burnup ≤ 27,500 MWD/MTU, and a minimum initial enrichment ≥ 2.4 wt% ^{235}U.
ii. SS clad:	An assembly cooling time after discharge ≥ 16 years, an average burnup ≤ 22,500 MWD/MTU, and a minimum initial enrichment ≥ 3.5 wt% ^{235}U.
e. Decay heat per assembly:	
i. ZR Clad:	≤272 Watts, except for array/class 8X8F fuel assemblies, which shall have a decay heat ≤183.5 Watts.
a. SS Clad:	≤83 Watts
f. Fuel assembly length:	≤ 176.2 inches (nominal design)
g. Fuel assembly width:	≤ 5.85 inches (nominal design)
h Fuel assembly weight:	≤ 700 lbs, including channels

II MPC MODEL MPC-68 (continued)

A Allowable Contents (continued)

2 Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications.

a Cladding type	ZR
b Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.45 wt% ^{235}U.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels and damaged fuel containers

Table A.1 (Page 4 of 23)
Fuel Assembly Limits

II MPC MODEL: MPC-68 (continued)

A Allowable Contents (continued)

3 Mixed oxide (MOX), BWR intact fuel assemblies, with or without Zircaloy channels. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a Cladding type	ZR
b Maximum planar-average initial enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 400 lbs, including channels

Table A 1 (Page 5 of 23)
Fuel Assembly Limits

II MPC MODEL: MPC-68 (continued)

A Allowable Contents (continued)

4 Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers MOX BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a Cladding type	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for array/class 6x6B
c. Initial maximum rod enrichment.	As specified in Table A.3 for array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels and damaged fuel containers.

II MPC MODEL MPC-68 (continued)

A Allowable Contents (continued)

5 Thoria rods (ThO$_2$ and UO$_2$) placed in Dresden Unit 1 Thoria Rod Canisters (as shown in Figure 1.2.11A of the HI-STAR 100 System SAR, Revision 12) and meeting the following specifications

a. Cladding type	ZR
b. Composition:	98.2 wt.% ThO$_2$, 1.8 wt. % UO$_2$ with an enrichment of 93.5 wt. % ^{235}U.
c. Number of rods per Thoria Rod Canister:	$\leq$ 18
d. Decay heat per Thoria Rod Canister:	$\leq$ 115 Watts
e. Post-irradiation fuel cooling time and average burnup per Thoria Rod Canister:	A fuel post-irradiation cooling time $\geq$ 18 years and an average burnup $\leq$ 16,000 MWD/MTIHM.
f. Initial heavy metal weight:	$\leq$ 27 kg/canister
g. Fuel cladding O.D.:	$\geq$ 0.412 inches
h. Fuel cladding I.D.:	$\leq$ 0.362 inches
i. Fuel pellet O.D.:	$\leq$ 0.358 inches
j. Active fuel length:	$\leq$ 111 inches
k. Canister weight:	$\leq$ 550 lbs, including fuel

B. Quantity per MPC: Up to one (1) Dresden Unit 1 Thoria Rod Canister plus any combination of damaged fuel assemblies in damaged fuel containers and intact fuel assemblies, up to a total of 68.

C. Fuel assemblies with stainless steel channels are not authorized for loading in the MPC-68.

D. Dresden Unit 1 fuel assemblies (fuel assembly array/class 6x6A, 6x6B, 6x6C, or 8x8A) with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68. The Antimony-Beryllium source material shall be in a water rod location.

III MPC MODEL MPC-68F

A Allowable Contents

1 Uranium oxide. BWR intact fuel assemblies, with or without Zircaloy channels. Uranium oxide BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A and meet the following specifications:

a. Cladding type	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time ≥ 18 years, an average burnup ≤ 30,000 MWD/MTU, and a minimum initial enrichment ≥ 1.45 wt% ^{235}U.
e. Fuel assembly length:	≤ 176.2 inches (nominal design)
f. Fuel assembly width:	≤ 5.85 inches (nominal design)
g. Fuel assembly weight:	≤ 400 lbs, including channels

Table A.1 (Page 8 of 23)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

2. Uranium oxide, BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. Uranium oxide BWR damaged fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

a. Cladding type.	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.45 wt% ^{235}U.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels and damaged fuel containers

III MPC MODEL MPC-68F (continued)

A Allowable Contents (continued)

3 Uranium oxide, BWR fuel debris, with or without Zircaloy channels, placed in damaged fuel containers The original fuel assemblies for the uranium oxide BWR fuel debris shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications

a. Cladding type:	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for the applicable original fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable original fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTU, and a minimum initial enrichment $\geq$ 1.45 wt% ^{235}U for the original fuel assembly.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels and damaged fuel containers

III MPC MODEL MPC-68F (continued)

A Allowable Contents (continued)

4 Mixed oxide (MOX), BWR intact fuel assemblies. with or without Zircaloy channels MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a Cladding type:	ZR
b Maximum planar-average initial enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for fuel assembly array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 400 lbs, including channels

Table A.1 (Page 11 of 23)
Fuel Assembly Limits

III MPC MODEL: MPC-68F (continued)

A Allowable Contents (continued)

5. Mixed oxide (MOX), BWR damaged fuel assemblies, with or without Zircaloy channels, placed in damaged fuel containers. MOX BWR intact fuel assemblies shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications:

a. Cladding type	ZR
b. Maximum planar-average initial enrichment:	As specified in Table A.3 for array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels and damaged fuel containers

Table A.1 (Page 12 of 23)
Fuel Assembly Limits

III MPC MODEL: MPC-68F (continued)

A Allowable Contents (continued)

6 Mixed oxide (MOX), BWR fuel debris, with or without Zircaloy channels, placed in damaged fuel containers The original fuel assemblies for the MOX BWR fuel debris shall meet the criteria specified in Table A.3 for fuel assembly array/class 6x6B and meet the following specifications

a Cladding type	ZR
b Maximum planar-average initial enrichment:	As specified in Table A.3 for original fuel assembly array/class 6x6B.
c. Initial maximum rod enrichment:	As specified in Table A.3 for original fuel assembly array/class 6x6B.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time $\geq$ 18 years, an average burnup $\leq$ 30,000 MWD/MTIHM, and a minimum initial enrichment $\geq$ 1.8 wt% ^{235}U for the UO_2 rods in the original fuel assembly.
e. Fuel assembly length:	$\leq$ 135.0 inches (nominal design)
f. Fuel assembly width:	$\leq$ 4.70 inches (nominal design)
g. Fuel assembly weight:	$\leq$ 550 lbs, including channels and damaged fuel containers

III MPC MODEL: MPC-68F (continued)

A Allowable Contents (continued)

7 Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters (as shown in Figure 1.2 11A of the HI-STAR 100 System SAR, Revision 12) and meeting the following specifications

a Cladding Type	ZR
b Composition.	98.2 wt.% ThO_2, 1.8 wt. % UO_2 with an enrichment of 93.5 wt. % ^{235}U.
c. Number of rods per Thoria Rod Canister:	$\leq$ 18
d. Decay heat per Thoria Rod Canister:	$\leq$ 115 Watts.
e. Post-irradiation fuel cooling time and average burnup per Thoria Rod Canister:	A fuel post-irradiation cooling time $\geq$ 18 years and an average burnup $\leq$ 16,000 MWD/MTIHM.
f. Initial heavy metal weight:	$\leq$ 27 kg/canister
g. Fuel cladding O.D.:	$\geq$ 0.412 inches
h. Fuel cladding I.D.:	$\leq$ 0.362 inches
i. Fuel pellet O.D.:	$\leq$ 0.358 inches
j. Active fuel length:	$\leq$ 111 inches
k. Canister weight:	$\leq$ 550 lbs, including fuel

III MPC MODEL MPC-68F (continued)

B Quantity per MPC.

Up to four (4) damaged fuel containers containing uranium oxide or MOX BWR fuel debris The remaining MPC-68F fuel storage locations may be filled with array/class 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A fuel assemblies of the following type, as applicable

1 Uranium oxide BWR intact fuel assemblies
2 MOX BWR intact fuel assemblies;
3 Uranium oxide BWR damaged fuel assemblies placed in damaged fuel containers:
4 MOX BWR damaged fuel assemblies placed in damaged fuel containers; or
5 Up to one (1) Dresden Unit 1 Thoria Rod Canister

C Fuel assemblies with stainless steel channels are not authorized for loading in the MPC-68F.

D. Dresden Unit 1 fuel assemblies (fuel assembly array/class 6x6A, 6x6B, 6x6C or 8x8A) with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68F. The Antimony-Beryllium neutron source material shall be in a water rod location.

IV MPC MODEL: MPC-24E

A Allowable Contents

1 *Uranium oxide*, PWR intact fuel assemblies listed in Table A.2 and meeting the following specifications:

a Cladding type	ZR or stainless steel (SS) as specified in Table A.2 for the applicable fuel assembly array/class
b Maximum initial enrichment	As specified in Table A.2 for the applicable fuel assembly array/class
c. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly	
i. ZR clad:	Except for Trojan plant fuel, an assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.4 or A.5, as applicable.
ii. SS clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.6, as applicable.
iii. Trojan plant fuel	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.8.
iv Trojan plant non-fuel hardware and neutron sources	Post-irradiation cooling time, and average burnup as specified in Table A.9
d. Decay heat per assembly	
i. ZR Clad:	Except for Trojan plant fuel, decay heat ≤ 833 Watts. *Trojan plant fuel decay heat: ≤ 725 Watts*
ii. SS Clad:	≤ 488 Watts
e. Fuel assembly length:	≤ 176.8 inches (nominal design)
f. Fuel assembly width:	≤ 8.54 inches (nominal design)
g. Fuel assembly weight:	≤ 1,680 lbs, including non-fuel hardware and neutron sources

IV MPC MODEL: MPC-24E

A Allowable Contents (continued)

2 Trojan plant damaged fuel assemblies meeting the applicable criteria listed in Table A.2 and meeting the following specifications

a Cladding type	ZR
b Maximum initial enrichment	3.7% ^{235}U
c Fuel assembly post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment per assembly	An assembly post-irradiation cooling time, average burnup, and initial enrichment as specified in Table A.8 Decay Heat: ≤ 725 Watts
d. Fuel assembly length:	≤ 169.3 inches (nominal design)
e. Fuel assembly width:	≤ 8.43 inches (nominal design)
f. Fuel assembly weight:	≤ 1,680 lbs, including DFC or Failed Fuel Can

B. Quantity per MPC: Up to 24 PWR intact fuel assemblies. For Trojan plant fuel only, up to four (4) damaged fuel assemblies may be stored in fuel storage locations 3, 6, 19, and/or 22. The remaining MPC-24E fuel storage locations may be filled with Trojan plant intact fuel assemblies.

C. Trojan plant fuel must be transported in the custom-designed Trojan MPCs with the MPC spacer installed. Fuel from other plants is not permitted to be transported in the Trojan MPCs.

D. Except for Trojan plant fuel, the fuel assemblies shall not contain non-fuel hardware or neutron sources. Trojan intact fuel assemblies containing non-fuel hardware may be transported in any fuel storage location.

E. Trojan plant damaged fuel assemblies must be transported in a Trojan Failed Fuel Can or a Holtec damaged fuel container designed for Trojan Plant fuel.

F. One (1) Trojan plant Sb-Be and /or up to two (2) Cf neutron sources in a Trojan plant intact fuel assembly (one source per fuel assembly) may be transported in any one MPC. Each fuel assembly neutron source may be transported in any fuel storage location.

G. Fuel debris is not authorized for transport in the MPC-24E.

H. Trojan plant non-fuel hardware and neutron sources may not be transported in the same fuel storage location as a damaged fuel assembly.

V MPC MODEL: MPC-24EF

A Allowable Contents

1 Uranium oxide. PWR intact fuel assemblies listed in Table A.2 and meeting the following specifications

a. Cladding type	ZR or stainless steel (SS) as specified in Table A.2 for the applicable fuel assembly array/class.
b Maximum initial enrichment	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly	
i. ZR clad:	Except for Trojan plant fuel, an assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.4 or A.5, as applicable.
ii. SS clad:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.6, as applicable.
iii Trojan plant fuel:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.8.
iv Trojan plant non-fuel hardware and neutron sources:	Post-irradiation cooling time, and average burnup as specified in Table A.9.
d. Decay heat per assembly:	
a. ZR Clad:	Except for Trojan plant fuel, decay heat ≤ 833 Watts. Trojan plant fuel decay heat: ≤ 725 Watts.
b. SS Clad:	≤ 488 Watts
e. Fuel assembly length:	≤ 176.8 inches (nominal design)
f. Fuel assembly width:	≤ 8.54 inches (nominal design)
g. Fuel assembly weight:	≤ 1,680 lbs, including non-fuel hardware and neutron sources.

Table A 1 (Page 18 of 23)
Fuel Assembly Limits

V MPC MODEL: MPC-24EF

A Allowable Contents (continued)

2 Trojan plant damaged fuel assemblies meeting the applicable criteria listed in Table A.2 and meeting the following specifications:

a Cladding type	ZR
b Maximum initial enrichment	3 7% ^{235}U
c Fuel assembly post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time, average burnup, and initial enrichment as specified in Table A.8. Decay Heat: ≤ 725 Watts
d. Fuel assembly length:	≤ 169.3 inches (nominal design)
e. Fuel assembly width:	≤ 8.43 inches (nominal design)
f. Fuel assembly weight:	≤ 1,680 lbs, including DFC or Failed Fuel Can.

V MPC MODEL MPC-24EF

A Allowable Contents (continued)

3 Trojan Fuel Debris Process Can Capsules and/or Trojan plant fuel assemblies classified as fuel debris, for which the original fuel assemblies meet the applicable criteria listed in Table A.2 and meet the following specifications

a Cladding type	ZR
b Maximum initial enrichment.	3.7% ^{235}U
c. Fuel debris post-irradiation cooling time, average burnup, decay heat, and minimum initial enrichment per assembly:	Post-irradiation cooling time, average burnup, and initial enrichment as specified in Table A.8. Decay Heat: ≤ 725 Watts
d. Fuel assembly length:	≤ 169.3 inches (nominal design)
e. Fuel assembly width:	≤ 8.43 inches (nominal design)
f. Fuel assembly weight:	≤ 1,680 lbs, including DFC or Failed Fuel Can.

B. Quantity per MPC: Up to 24 PWR intact fuel assemblies. For Trojan plant fuel only, up to four (4) damaged fuel assemblies, fuel assemblies classified as fuel debris, and/or Trojan Fuel Debris Process Can Capsules may be stored in fuel storage locations 3, 6, 19, and/or 22. The remaining MPC-24EF fuel storage locations may be filled with Trojan plant intact fuel assemblies.

C. Trojan plant fuel must be transported in the custom-designed Trojan MPCs with the MPC spacer installed. Fuel from other plants is not permitted to be transported in the Trojan MPCs.

D. Except for Trojan plant fuel, the fuel assemblies shall not contain non-fuel hardware or neutron sources. Trojan intact fuel assemblies containing non-fuel hardware may be transported in any fuel storage location.

E. Trojan plant damaged fuel assemblies, fuel assemblies classified as fuel debris, and Fuel Debris Process Can Capsules must be transported in a Trojan Failed Fuel Can or a Holtec damaged fuel container designed for Trojan Plant fuel.

F. One (1) Trojan plant Sb-Be and /or up to two (2) Cf neutron sources in a Trojan plant intact fuel assembly (one source per fuel assembly) may be transported in any one MPC. Each fuel assembly neutron source may be transported in any fuel storage location.

G. Trojan plant non-fuel hardware and neutron sources may not be transported in the same fuel storage location as a damaged fuel assembly.

Table A.1 (Page 20 of 23)
Fuel Assembly Limits

VI MPC MODEL MPC-32

A Allowable Contents

1 Uranium oxide, PWR intact fuel assemblies in array/classes 15x15D, E, F, and H and 17x17A, B, and C listed in Table A.2 and meeting the following specifications

a Cladding type.	ZR
b. Maximum initial enrichment:	As specified in Table A.2 for the applicable fuel assembly array/class.
c. Post-irradiation cooling time, maximum average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time, average burnup, and minimum initial enrichment as specified in Table A.10 or A.11, as applicable.
d. Minimum average burnup per assembly (Assembly Burnup shall be confirmed per Subsection 1.2.3.7.2 of the SAR, which is hereby included by reference)	Calculated value as a function of initial enrichment. See Table A.12.
e. Decay heat per assembly:	≤ 625 Watts
f. Fuel assembly length:	≤ 176.8 inches (nominal design)
g. Fuel assembly width:	≤ 8.54 inches (nominal design)
h. Fuel assembly weight:	≤ 1,680 lbs

i. Operating parameters during irradiation of the assembly (Assembly operating parameters shall be determined per Subsection 1.2.3.7.1 of the SAR, which is hereby included by reference)

Core ave. soluble boron concentration:	≤ 1,000 ppmb
Assembly ave. moderator temperature:	≤ 601 K for array/classes 15x15D, E, F, and H ≤ 610 K for array/classes 17x17A, B, and C
Assembly ave. specific power:	≤ 47.36 kW/kg-U for array/classes 15x15D, E, F, and H ≤ 61.61 kW/kg-U for array/classes 17x17A, B, and C

Table A 1 (Page 21 of 23)
Fuel Assembly Limits

VI MP C MODEL. MPC-32 (continued)

B Quantity per MPC Up to 32 PWR intact fuel assemblies

C Fuel assemblies shall not contain non-fuel hardware

D Damaged fuel assemblies and fuel debris are not authorized for transport in MPC-32

E Trojan plant fuel is not permitted to be transported in the MPC-32.

Table A.1 (Page 22 of 23)
Fuel Assembly Limits

VII MPC MODEL MPC-HB

A Allowable Contents

1 Uranium oxide, INTACT and/or UNDAMAGED FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, and FUEL DEBRIS, with or without channels, meeting the criteria specified in Table A.3 for fuel assembly array/class 6x6D or 7x7C and the following specifications:

a. Cladding type:	ZR
b. Maximum planar-average enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
c. Initial maximum rod enrichment:	As specified in Table A.3 for the applicable fuel assembly array/class.
d. Post-irradiation cooling time, average burnup, and minimum initial enrichment per assembly:	An assembly post-irradiation cooling time ≥ 29 years, an average burnup ≤ 23,000 MWD/MTU, and a minimum initial enrichment ≥ 2.09 wt% ^{235}U.
e. Fuel assembly length:	≤ 96.91 inches (nominal design)
f. Fuel assembly width:	≤ 4.70 inches (nominal design)
g. Fuel assembly weight:	≤ 400 lbs, including channels and DFC
h. Decay heat per assembly:	≤ 50 W
h. Decay heat per MPC:	≤ 2000 W

Table A 1 (Page 23 of 23)
Fuel Assembly Limits

VII MPC MODEL. MPC-HB (continued)

B Quantity per MPC-HB Up to 80 fuel assemblies

C Damaged fuel assemblies and fuel debris must be stored in a damaged fuel container Allowable Loading Configurations: Up to 28 damaged fuel assemblies/fuel debris in damaged fuel containers, may be placed into the peripheral fuel storage locations as shown in SAR Figure 6.I.3, or up to 40 damaged fuel assemblies/fuel debris, in damaged fuel containers, can be placed in a checkerboard pattern as shown in SAR Figure 6.I.4. The remaining fuel locations may be filled with intact and/or undamaged fuel assemblies meeting the above applicable specifications, or with intact and/or undamaged fuel assemblies placed in damaged fuel containers.

NOTE 1: The total quantity of damaged fuel or fuel debris permitted in a single damaged fuel container is limited to the equivalent weight and special nuclear material quantity of one intact assembly.

NOTE 2: Fuel debris includes material in the form of loose debris consisting of zirconium clad pellets, stainless steel clad pellets, unclad pellets, or rod segments up to a maximum of one equivalent fuel assembly. A maximum of 1.5 kg of stainless steel clad is allowed per cask.

Table A.2 (Page 1 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**14x14A**	**14x14B**	**14x14C**	**14x14D**	**14x14E**
Clad Material (Note 2)	ZR	ZR	ZR	SS	Zr
Design Initial U (kg/assy.) (Note 3)	≤ 407	≤ 407	≤ 425	≤ 400	≤ 206
Initial Enrichment (MPC-24, 24E, and 24EF) (wt % ^{235}U)	≤ 4.6 (24) ≤ 5.0 (24E/EF)	≤ 4.6 (24) ≤ 5.0 (24E/EF)	≤ 4.6 (24) ≤ 5.0 (24E/EF)	≤ 4.0 (24) ≤ 5.0 (24E/EF)	≤ 5.0
No. of Fuel Rod Locations	179	179	176	180	173
Fuel Clad O.D. (in.)	≥ 0.400	≥ 0.417	≥ 0.440	≥ 0.422	≥ 0.3415
Fuel Clad I.D. (in.)	≤ 0.3514	≤ 0.3734	≤ 0.3880	≤ 0.3890	≤ 0.3175
Fuel Pellet Dia. (in.)	≤ 0.3444	≤ 0.3659	≤ 0.3805	≤ 0.3835	≤ 0.3130
Fuel Rod Pitch (in.)	≤ 0.556	≤ 0.556	≤ 0.580	≤ 0.556	Note 6
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 144	≤ 102
No. of Guide Tubes	17	17	5 (Note 4)	16	0
Guide Tube Thickness (in.)	≥ 0.017	≥ 0.017	≥ 0.038	≥ 0.0145	N/A

Table A.2 (Page 2 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**15x15A**	**15x15B**	**15x15C**	**15x15D**	**15x15E**	**15x15F**
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note [illegible])	≤ 464	≤ 464	≤ 464	≤ 475	≤ 475	≤ 475
Initial Enrichment (MPC-24, 24E, and 24EF) (wt % ^{235}U)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)	≤ 4.1 (24) ≤ 4.5 (24E/EF)
Initial Enrichment (MPC-32) (wt. % ^{235}U) (Note 5)	N/A	N/A	N/A	(Note 5)	(Note 5)	(Note 5)
No. of Fuel Rod Locations	204	204	204	208	208	208
Fuel Clad O.D. (in.)	≥ 0.418	≥ 0.420	≥ 0.417	≥ 0.430	≥ 0.428	≥ 0.428
Fuel Clad I.D. (in.)	≤ 0.3660	≤ 0.3736	≤ 0.3640	≤ 0.3800	≤ 0.3790	≤ 0.3820
Fuel Pellet Dia. (in.)	≤ 0.3580	≤ 0.3671	≤ 0.3570	≤ 0.3735	≤ 0.3707	≤ 0.3742
Fuel Rod Pitch (in.)	≤ 0.550	≤ 0.563	≤ 0.563	≤ 0.568	≤ 0.568	≤ 0.568
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	21	21	21	17	17	17
Guide/Instrument Tube Thickness (in.)	≥ 0.015	≥ 0.015	≥ 0.0165	≥ 0.0150	≥ 0.0140	≥ 0.0140

Table A.2 (Page 3 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/ Class	**15x15G**	**15x15H**	**16x16A**	**17x17A**	**17x17B**	**17x17C**
Clad Material (Note 2)	SS	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 420	≤ 475	≤ 443	≤ 467	≤ 467	≤ 474
Initial Enrichment (MPC-24, 24E, and 24EF) (wt % ^{235}U)	≤ 4.0 (24) ≤ 4.5 (24E/EF)	≤ 3.8 (24) ≤ 4.2 (24E/EF)	≤ 4.6 (24) ≤ 5.0 (24E/EF)	≤ 4.0 (24) ≤ 4.4 (24E/EF)	≤ 4.0 (24) ≤ 4.4 (24E/EF) (Note 7)	≤ 4.0 (24) ≤ 4.4 (24E/EF)
Initial Enrichment (MPC-32) (wt. % ^{235}U) (Note 5)	N/A	(Note 5)	N/A	(Note 5)	(Note 5)	(Note 5)
No. of Fuel Rod Locations	204	208	236	264	264	264
Fuel Clad O.D. (in.)	≥ 0.422	≥ 0.414	≥ 0.382	≥ 0.360	≥ 0.372	≥ 0.377
Fuel Clad I.D. (in.)	≤ 0.3890	≤ 0.3700	≤ 0.3320	≤ 0.3150	≤ 0.3310	≤ 0.3330
Fuel Pellet Dia. (in.)	≤ 0.3825	≤ 0.3622	≤ 0.3255	≤ 0.3088	≤ 0.3232	≤ 0.3252
Fuel Rod Pitch (in.)	≤ 0.563	≤ 0.568	≤ 0.506	≤ 0.496	≤ 0.496	≤ 0.502
Active Fuel Length (in.)	≤ 144	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	21	17	5 (Note 4)	25	25	25
Guide/Instrument Tube Thickness (in.)	≥ 0.0145	≥ 0.0140	≥ 0.0400	≥ 0.016	≥ 0.014	≥ 0.020

Table A.2 (Page 4 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Notes:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class

2. ZR Designates cladding material made of Zirconium or Zirconium alloys

3. Design initial uranium weight is the nominal uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each PWR fuel assembly, the total uranium weight limit specified in this table may be increased up to 2.0 percent for comparison with users' fuel records to account for manufacturer tolerances.

4. Each guide tube replaces four fuel rods.

5. Minimum burnup and maximum initial enrichment as specified in Table A.12.

6. This fuel assembly array/class includes only the Indian Point Unit 1 fuel assembly. This fuel assembly has two pitches in different sectors of the assembly. These pitches are 0.441 inches and 0.453 inches.

7. Trojan plant-specific fuel is governed by the limits specified for array/class 17x17B and will be transported in the custom-designed Trojan MPC-24E/EF canisters. The Trojan MPC-24E/EF design is authorized to transport only Trojan plant fuel with a maximum initial enrichment of 3.7 wt.% ^{235}U.

Table A.3 (Page 1 of 6)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	6x6A	6x6B	6x6C	7x7A	7x7B	8x8A
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 110	≤ 110	≤ 110	≤ 100	≤ 195	≤ 120
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 2.7	≤ 2.7 for the UO_2 rods. See Note 4 for MOX rods	≤ 2.7	≤ 2.7	≤ 4.2	≤ 2.7
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 4.0	≤ 4.0	≤ 4.0	≤ 5.5	≤ 5.0	≤ 4.0
No. of Fuel Rod Locations	35 or 36	35 or 36 (up to 9 MOX rods)	36	49	49	63 or 64
Fuel Clad O.D. (in.)	≥ 0.5550	≥ 0.5625	≥ 0.5630	≥ 0.4860	≥ 0.5630	≥ 0.4120
Fuel Clad I.D. (in.)	≤ 0.5105	≤ 0.4945	≤ 0.4990	≤ 0.4204	≤ 0.4990	≤ 0.3620
Fuel Pellet Dia. (in.)	≤ 0.4980	≤ 0.4820	≤ 0.4880	≤ 0.4110	≤ 0.4910	≤ 0.3580
Fuel Rod Pitch (in.)	≤ 0.710	≤ 0.710	≤ 0.740	≤ 0.631	≤ 0.738	≤ 0.523
Active Fuel Length (in.)	≤ 120	≤ 120	≤ 77.5	≤ 80	≤ 150	≤ 120
No. of Water Rods (Note 11)	1 or 0	1 or 0	0	0	0	1 or 0
Water Rod Thickness (in.)	≥ 0	≥ 0	N/A	N/A	N/A	≥ 0
Channel Thickness (in.)	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.120	≤ 0.100

Table A.3 (Page 2 of 6)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	8x8B	8x8C	8x8D	8x8E	8x8F	9x9A
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 185	≤ 185	≤ 185	≤ 185	≤ 185	≤ 177
Maximum planar-average initial enrichment (wt % ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2	< 4.0	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	63 or 64	62	60 or 61	59	64	74/66 (Note 5)
Fuel Clad O.D. (in.)	≥ 0.4840	≥ 0.4830	≥ 0.4830	≥ 0.4930	≥ 0.4576	≥ 0.4400
Fuel Clad I.D. (in.)	≤ 0.4295	≤ 0.4250	≤ 0.4230	≤ 0.4250	≤ 0.3996	≤ 0.3840
Fuel Pellet Dia. (in.)	≤ 0.4195	≤ 0.4160	≤ 0.4140	≤ 0.4160	≤ 0.3913	≤ 0.3760
Fuel Rod Pitch (in.)	≤ 0.642	≤ 0.641	≤ 0.640	≤ 0.640	≤ 0.609	≤ 0.566
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 or 0	2	1 - 4 (Note 7)	5	N/A (Note 12)	2
Water Rod Thickness (in.)	≥ 0.034	> 0.00	> 0.00	≥ 0.034	≥ 0.0315	> 0.00
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.100	≤ 0.055	≤ 0.120

Table A.3 (Page 3 of 6)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	9x9B	9x9C	9x9D	9x9E (Note 13)	9x9F (Note 13)	9x9G
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 177	≤ 177	≤ 177	≤ 177	≤ 177	≤ 177
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rods	72	80	79	76	76	72
Fuel Clad O.D. (in.)	≥ 0.4330	≥ 0.4230	≥ 0.4240	≥ 0.4170	≥ 0.4430	≥ 0.4240
Fuel Clad I.D. (in.)	≤ 0.3810	≤ 0.3640	≤ 0.3640	≤ 0.3640	≤ 0.3860	≤ 0.3640
Fuel Pellet Dia. (in.)	≤ 0.3740	≤ 0.3565	≤ 0.3565	≤ 0.3530	≤ 0.3745	≤ 0.3565
Fuel Rod Pitch (in.)	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 (Note 6)	1	2	5	5	1 (Note 6)
Water Rod Thickness (in.)	> 0.00	≥ 0.020	≥ 0.0300	≥ 0.0120	≥ 0.0120	≥ 0.0320
Channel Thickness (in.)	≤ 0.120	≤ 0.100	≤ 0.100	≤ 0.120	≤ 0.120	≤ 0.120

Table A.3 (Page 4 of 6)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**10x10A**	**10x10B**	**10x10C**	**10x10D**	**10x10E**
Clad Material (Note 2)	ZR	ZR	ZR	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 186	≤ 186	≤ 186	≤ 125	≤ 125
Maximum planar-average initial enrichment (wt.% ^{235}U)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	92/78 (Note 8)	91/83 (Note 9)	96	100	96
Fuel Clad O.D. (in.)	≥ 0.4040	≥ 0.3957	≥ 0.3780	≥ 0.3960	≥ 0.3940
Fuel Clad I.D. (in.)	≤ 0.3520	≤ 0.3480	≤ 0.3294	≤ 0.3560	≤ 0.3500
Fuel Pellet Dia. (in.)	≤ 0.3455	≤ 0.3420	≤ 0.3224	≤ 0.3500	≤ 0.3430
Fuel Rod Pitch (in.)	≤ 0.510	≤ 0.510	≤ 0.488	≤ 0.565	≤ 0.557
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 83	≤ 83
No. of Water Rods (Note 11)	2	1 (Note 6)	5 (Note 10)	0	4
Water Rod Thickness (in.)	≥ 0.0300	> 0.00	≥ 0.031	N/A	≥ 0.022
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.055	≤ 0.080	≤ 0.080

Table A.3 (Page 5 of 6)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**6x6D**	**7x7C**
Clad Material (Note 2)	Zr	Zr
Design Initial U (kg/assy.)(Note 3)	≤ 78	≤ 78
Maximum planar-average initial enrichment (wt % ^{235}U)	≤ 2.6	≤ 2.6
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 4.0 (Note 14)	≤ 4.0
No. of Fuel Rod Locations	36	49
Fuel Clad O.D. (in.)	≥ 0.5585	≥ 0.486
Fuel Clad I.D. (in.)	≤ 0.505	≤ 0.426
Fuel Pellet Dia. (in.)	≤ 0.488	≤ 0.411
Fuel Rod Pitch (in.)	≤ 0.740	≤ 0.631
Active Fuel Length (in.)	≤ 80	≤ 80
No. of Water Rods (Note 11)	0	0
Water Rod Thickness (in.)	N/A	N/A
Channel Thickness (in.)	≤ 0.060	≤ 0.060

Table A.3 (Page 6 of 6)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Notes

1 All dimensions are design nominal values Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class

2 ZR designates cladding material made from Zirconium or Zirconium alloys

3 Design initial uranium weight is the uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each BWR fuel assembly, the total uranium weight limit specified in this table may be increased up to 1.5% for comparison with users' fuel records to account for manufacturer's tolerances.

4 $\leq$ 0.635 wt. % ^{235}U and $\leq$ 1.578 wt. % total fissile plutonium (^{239}Pu and ^{241}Pu), (wt. % of total fuel weight, i.e., UO_2 plus PuO_2).

5. This assembly class contains 74 total fuel rods; 66 full length rods and 8 partial length rods.

6. Square, replacing nine fuel rods.

7. Variable

8. This assembly class contains 92 total fuel rods; 78 full length rods and 14 partial length rods.

9. This assembly class contains 91 total fuel rods, 83 full length rods and 8 partial length rods.

10. One diamond-shaped water rod replacing the four center fuel rods and four rectangular water rods dividing the assembly into four quadrants.

11. These rods may be sealed at both ends and contain Zr material in lieu of water.

12. This assembly is known as "QUAD+" and has four rectangular water cross segments dividing the assembly into four quadrants.

13. For the SPC 9x9-5 fuel assembly, each fuel rod must meet either the 9x9E or 9x9F set of limits for clad O.D., clad I.D., and pellet diameter.

14. Only two assemblies may contain one rod each with an initial maximum enrichment up to 5.5 wt%.

Table A 4

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND INITIAL ENRICHMENT
MPC-24/24E/24/EF PWR FUEL WITH ZIRCALOY CLAD AND
WITH NON-ZIRCALOY IN-CORE GRID SPACERS

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Initial Enrichment (wt. % U-235)
≥ 9	≤ 24,500	≥ 2.3
≥ 11	≤ 29,500	≥ 2.6
≥ 13	≤ 34,500	≥ 2.9
≥ 15	≤ 39,500	≥ 3.2
≥ 18	≤ 44,500	≥ 3.4

Table A.5

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND INITIAL ENRICHMENT
MPC-24/24E/24EF PWR FUEL WITH ZIRCALOY CLAD AND
WITH ZIRCALOY IN-CORE GRID SPACERS

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Initial Enrichment (wt. % U-235)
≥ 6	≤ 24,500	≥ 2.3
≥ 7	≤ 29,500	≥ 2.6
≥ 9	≤ 34,500	≥ 2.9
≥ 11	≤ 39,500	≥ 3.2
≥ 14	≤ 44,500	≥ 3.4

Table A.6

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND INITIAL ENRICHMENT
MPC-24/24E/24EF PWR FUEL WITH STAINLESS STEEL CLAD

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Initial Enrichment (wt. % U-235)
≥ 19	≤ 30,000	≥ 3.1
≥ 24	≤ 40,000	≥ 3.1

Table A.7

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND INITIAL ENRICHMENT
MPC-68

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Initial Enrichment (wt. % U-235)
≥ 5	≤ 10,000	≥ 0.7
≥ 7	≤ 20,000	≥ 1.35
≥ 8	≤ 24,500	≥ 2.1
≥ 9	≤ 29,500	≥ 2.4
≥ 11	≤ 34,500	≥ 2.6
≥ 14	≤ 39,500	≥ 2.9
≥ 19	≤ 44,500	≥ 3.0

Table A.8

TROJAN PLANT FUEL ASSEMBLY COOLING, AVERAGE BURNUP. AND INITIAL ENRICHMENT LIMITS (Note 1)

Post-irradiation Cooling Time (years)	Assembly Burnup (MWD/MTU)	Assembly Initial Enrichment (wt.% ^{235}U)
≥16	≤42,000	≥3.09
≥16	≤37,500	≥2.6
≥16	≤30,000	≥2.1

NOTES:

1. Each fuel assembly must only meet one set of limits (i.e., one row)

Table A.9

TROJAN PLANT NON-FUEL HARDWARE AND NEUTRON SOURCES COOLING AND BURNUP LIMITS

Type of Hardware or Neutron Source	Burnup (MWD/MTU)	Post-Irradiation Cooling Time (Years)
BPRAs	≤15,998	≥24
TPDs	≤118,674	≥11
RCCAs	≤125,515	≥9
Cf neutron source	≤15,998	≥24
Sb-Be neutron source with 4 source rods, 16 burnable poison rods, and 4 thimble plug rods	≤45,361	≥19
Sb-Be neutron source with 4 source rods, 20 thimble plug rods	≤88,547	≥9

Table A 10

FUEL ASSEMBLY COOLING. AVERAGE BURNUP. AND MINIMUM ENRICHMENT MPC-32 PWR FUEL WITH ZIRCALOY CLAD AND WITH NON-ZIRCALOY IN-CORE GRID SPACERS

Post-irradiation cooling time (years)	Assembly burnup (MWD/MTU)	Assembly Initial Enrichment (wt. % U-235)
≥12	≤24,500	≥2.3
≥14	≤29,500	≥2.6
≥16	≤34,500	≥2.9
≥19	≤39,500	≥3.2
≥20	≤42,500	≥3.4

Table A.11

FUEL ASSEMBLY COOLING, AVERAGE BURNUP, AND MINIMUM ENRICHMENT MPC-32 PWR FUEL WITH ZIRCALOY CLAD AND WITH ZIRCALOY IN-CORE GRID SPACERS

Post-irradiation cooling time (years)	Assembly burnup (MWD/MTU)	Assembly Initial Enrichment (wt.% U-235)
≥8	≤24,500	≥2.3
≥9	≤29,500	≥2.6
≥12	≤34,500	≥2.9
≥14	≤39,500	≥3.2
≥19	≤44,500	≥3.4

Table A.12

FUEL ASSEMBLY MAXIMUM ENRICHMENT AND MINIMUM BURNUP REQUIREMENTS FOR TRANSPORTATION IN MPC-32

Fuel Assembly Array/Class	Configuration (Note 2)	Maximum Enrichment (wt.% U-235)	Minimum Burnup (B) as a Function of Initial Enrichment (E) (Note 1) (GWD/MTU)
15x15D, E, F, H	A	4.65	$B = (1.6733)*E^3-(18.72)*E^2+(80.5967)*E-88.3$
	B	4.38	$B = (2.175)*E^3-(23.355)*E^2+(94.77)*E-99.95$
	C	4.48	$B = (1.9517)*E^3-(21.45)*E^2+(89.1783)*E-94.6$
	D	4.45	$B = (1.93)*E^3-(21.095)*E^2+(87.785)*E-93.06$
17x17A,B,C	A	4.49	$B = (1.08)*E^3-(12.25)*E^2+(60.13)*E-70.86$
	B	4.04	$B = (1.1)*E^3-(11.56)*E^2+(56.6)*E-62.59$
	C	4.28	$B = (1.36)*E^3-(14.83)*E^2+(67.27)*E-72.93$
	D	4.16	$B = (1.4917)*E^3-(16.26)*E^2+(72.9883)*E-79.7$

NOTES:

1. E = Initial enrichment (e.g., for 4.05 wt.%, E = 4.05).
2. See Table A.13.
3. Fuel Assemblies must be cooled 5 years or more.

Table A 13

LO ADING CONFIGURATIONS FOR THE MPC-32

CONFIGURATION	ASSEMBLY SPECIFICATIONS
A	• Assemblies that have not been located in any cycle under a control rod bank that was permitted to be inserted during full power operation (per plant operating procedures); or • Assemblies that have been located under a control rod bank that was permitted to be inserted during full power operation (per plant operating procedures), but where it can be demonstrated, based on operating records, that the insertion never exceeded 8 inches from the top of the active length during full power operation.
B	• Of the 32 assemblies in a basket, up to 8 assemblies can be from core locations where they were located under a control rod bank, that was permitted to be inserted more than 8 inches during full power operation. There is no limit on the duration (in terms of burnup) under this bank. • The remaining assemblies in the basket must satisfy the same conditions as specified for configuration A.
C	• Of the 32 assemblies in a basket, up to 8 assemblies can be from core locations where they were located under a control rod bank, that was permitted to be inserted more than 8 inches during full power operation. Location under such a control rod bank is limited to 20 GWD/MTU of the assembly. • The remaining assemblies in the basket must satisfy the same conditions as specified for configuration A.
D	• Of the 32 assemblies in a basket, up to 8 assemblies can be from core locations where they were located under a control rod bank, that was permitted to be inserted more than 8 inches during full power operation. Location under such a control rod bank is limited to 30 GWD/MTU of the assembly. • The remaining assemblies in the basket must satisfy the same conditions as specified for configuration A.

REFERENCES:

Holtec International Report No HI-951251. *Safety Analysis Report for the Holtec International Storage, Transport. And Repository Cask System (HI-STAR 100 Cask System)*, Revision 12 dated October 6. 2006. as supplemented

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON D.C. 20555-0001

SAFETY EVALUATION REPORT

Docket No. 71-9261
Model No. HI-STAR 100 Package
Certificate of Compliance No. 9261
Revision No. 7

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SAFETY EVALUATION REPORT

Docket No. 71-9261
Model No. HI-STAR HB
Certificate of Compliance No 9261
Revision No 7

SUMMARY

By application dated October 5. 2006. as supplemented June 29. July 27. August 3, September 27. October 5. and December 18, 2007; January 9. March 19 and September 30, 2008; and February 27. 2009. Holtec International (Holtec or the applicant) requested a revision to the 10 CFR Part 71 Certificate of Compliance (CoC) No. 9261 for the Model No HI-STAR 100 system The spent fuel cask that is the subject of this amendment request is contained in a site specific license granted by NRC to PG&E for storage of specific fuel at the Humboldt Bay Power Station (HB) site (Docket No. 72-27). Additional supporting changes also requested include incorporating Metamic as an approved neutron absorber and updating the cask identification to B(U)F-96 in accordance with 10 CFR 71.19(e). The staff informed Holtec of the acceptance of the transportation application for technical review by letter dated November 9. 2006.

Based on the statements and representations in the application, as supplemented, and Revision 12 of the SAR, the staff concludes, per its evaluation described in this Safety Evaluation Report (SER), that the requested changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

1.0 GENERAL INFORMATION

The following sections summarize the applicant's change requests with respect to the packaging and its contents.

1.1 Packaging

With respect to the packaging, the applicant is proposing to:

- add the HI-STAR 100 Version HB (HI-STAR HB) for use at HB,
- change the Safety Analysis Report (SAR) and licensing drawings to add Metamic as a neutron absorber for use in the HI-STAR HB, and
- change the Package Identification Number of the HI-STAR 100 System from B(U)F-85 to B(U)F-96 in the CoC and Chapter 1 of the SAR.

A new shorter version of HI-STAR 100 was designed specifically for use at HB because the fuel assemblies are shorter than typical Boiling Water Reactor (BWR) fuel assemblies. The design includes a HI-STAR HB overpack, the multiple-purpose canister for HB (MPC-HB), and the impact limiters. The shorter design results in a reduced gross weight. Additionally, a HB specific Damaged Fuel Container (DFC) has been designed.

For the addition of Metamic as a neutron absorber for the HI-STAR HB, the drawings specified Metamic and the minimum B-10 loading, Metamic was added to SAR text, a description of the material was added. an updated criticality analysis was provided and the acceptance testing requirements were provided in SAR Chapter 8

To support the request to change the Package Identification Number of the HI-STAR 100 System from B(U)F-85 to B(U)F-96 in the CoC, staff reviewed the nineteen issues considered in the rulemaking process that resulted in the revised 10 CFR Part 71 dated January 26, 2004 The staff evaluated the applicant's request, as described below.

- Issue 1. Changing Part 71 to the International Systems of Units (SI) Only

  This proposal was not adopted in the final rule, and therefore no changes are needed in the package application or the CoC to conform to the new rule

- Issue 2. Radionuclide Exemption Values

  The final rule adopted radionuclide activity concentration values and consignments activity limits in TS-R-1 for the exemption from regulatory requirements for the shipment or carriage of certain radioactive low-level materials. In addition, the final rule adopted an exemption from regulatory requirements for certain natural material and ores containing naturally occurring radionuclides. Based on the design purpose of the package and the allowed contents specified in the certificate, this change is not applicable to the HI-STAR 100 package. Thus, no changes are needed to conform to the new rule.

- Issue 3, Revision of A_1 and A_2.

  The final rule adopted changes in the A_1 and A_2 values from TS-R-1, with the exception of two radionuclides. The A_1 and A_2 values were modified in TS-R-1 based on refined modeling of possible doses from radionuclides, and the NRC agreed that incorporating the latest in dosimetric modeling would improve transportation regulations. The applicant provided an updated containment analysis in Chapter 4 of the application incorporating the revised A_2 values, which are for radioactive material in normal form. In general, the A_2 values for radionuclides important to the containment requirements for spent fuel shipments were increased. Although the calculated maximum allowable leakage rates were increased, the applicant retained the maximum allowable leakage rate that is demonstrated for the cask through leak testing (4.3 E-6 atm cm^3/sec He). The staff agrees that the package meets the containment requirements of 10 CFR 71.51 considering the changes in the A_1 and A_2 values in Appendix A, Table A-1, of the revised 10 CFR Part 71.

- Issue 4, Uranium Hexafluoride (UF_6) Package Requirements.

  These changes are not applicable, since the package is not authorized for the transport of uranium hexafluoride. Therefore, no changes are needed to conform to the new rule.

- Issue 5, Criticality Safety Index (CSI).

  The final rule adopted the CSI requirement from TS-R-1. The applicant revised Chapters 1 and 6 to clearly distinguish between the CSI and the Transport Index (TI).

- Issue 6. Type C Packages and Low Dispersible Material

  This proposal was not adopted for the final rule. Thus, no changes are necessary.

- Issue 7. Deep Immersion Test

  The final rule adopted an extension of the previous version of 10 CFR 71.61 from packages for irradiated fuel to any Type B package containing activity greater than 10^5 A_2. Because the Model No. HI-STAR 100 package is designed to transport irradiated fuel, the applicant already complies with the deep immersion requirements of 10 CFR 71.61. Thus, no changes are necessary to conform to the new rule.

- Issue 8. Grandfathering Previously Approved Packages

  The final rule adopted a process for allowing continued use, for specific periods of time, of a previously approved packaging design without demonstrating compliance to the final rule. The applicant has decided in accordance with 10 CFR 71.19(e) to submit information demonstrating compliance with the final rule. Thus, grandfathering the design of the package is not necessary.

- Issue 9, Changes to Various Definitions.

  The final rule adopted several revised and new definitions. These changes were adopted to provide clarity to Part 71. No change is necessary to conform to the new rule.

- Issue 10, Crush Test for Fissile Material Packages.

  The revised 10 CFR 71.73 expanded the applicability of the crush test to fissile material packages. The crush test is required for packages with a mass not greater than 500 kilograms (1100 pounds). Since the Model No. HI-STAR 100 package has a mass greater than this, the crush test is not applicable. Therefore, no change is necessary to conform to the new rule.

- Issue 11, Fissile Material Package Design for Transport by Aircraft.

  The final rule adopted a new section, Section 71.55(f), which addresses packaging design requirements for packages transporting fissile material by air. The package is not authorized for shipment by air, and this requirement is not applicable to the Model No. HI-STAR 100 package.

- Issue 12, Special Package Authorizations.

  The final rule adopted provisions for special package authorization that will apply only in limited circumstances and only to one-time shipments of large components. This provision is not applicable to the Model No. HI-STAR 100 package. Thus, no change is necessary to conform to the new rule.

- Issue 13, Expansion of Part 71 Quality Assurance (QA) Requirements to Certificate Holders.

The final rule expanded the scope of Part 71 QA requirements to apply to any person holding or applying for a CoC. QA requirements apply to design, purchase, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair, and modification of components of packaging that are important to safety. The applicant must meet the QA program requirements of 10 CFR 71 101(a), (b), and (c). No change is needed to conform to the new rule.

- Issue 14, Adoption of the American Society of Mechanical Engineers (ASME) code

  This proposal was not adopted in the final rule. Thus, no change is needed to conform to the new rule

- Issue 15, Change Authority for Dual-Purpose Package Certificate Holders

  This proposal was not adopted for the final rule. Thus, no change is necessary to conform to the new rule.

- Issue 16, Fissile Material Exemptions and General License Provisions.

  The final rule adopted various revisions to the fissile material exemptions and the general license provisions in Part 71 to facilitate effective and efficient regulation of the transport of small quantities of fissile material. The criticality safety of the Model No. HI-STAR 100 package does not rely on limiting fissile materials to exempt or generally licensed quantities. Chapter 6 of the package application demonstrates criticality safety of the package with the authorized fissile contents. Therefore, no change is necessary to conform to the new rule.

- Issue 17, Double Containment of Plutonium.

  The final rule removed the requirement that packages with plutonium in excess of 0.74 terabecquerel (20 curies) have a second separate inner container. Holtec revised the package application to remove references and discussions related to the second inner container requirement. Additionally, the requirement for helium leak testing has been removed, since the MPC is no longer a containment boundary. Further, the CoC has been revised to delete the limits based on previous double containment requirement in the following condition:

  Condition 5(a)(2), deleted the words "BWR fuel debris may be shipped only in the MPC-68F;" "PWR spent fuel assemblies classified as fuel debris may be loaded only in MPC-24-EF;" and "For the HI-STAR 100 System transporting fuel debris in a MPC-68F or MPC-24EF, the MPC provides the second inner container, in accordance with 10 CFR 71.63. The MPC pressure boundary is a welded enclosure constructed entirely of a stainless steel alloy."

- Issue 18, Contamination Limits as Applied to Spent Fuel and High Level Waste Packages.

  This proposal was not adopted for the final rule. Thus, no change is needed to conform to the new rule.

- Issue 19, Modification of Events Reporting Requirements

  The final rule adopted modified reporting requirements. While the final rule is applicable to the package, no change is needed to either the CoC or the package application to conform to the new rule

"-96" Conclusion

Based on the statements and representations in the application, the staff concludes that the design has been adequately described and meets the requirements of the revised 10 CFR Part 71 Thus, the staff agrees that including the designation "-96" in the identification number is warranted To allow time to modify the packaging markings to include the "-96" designation in the package identification number, the certificate has been conditioned to allow use of packagings marked with the "-85" designation for a period of approximately one year. After May 31, 2009, the packaging must be marked with the package identification number including the "-96" designation.

1.2 Contents

The applicant has requested the following additions or changes to the contents to:

- add the HB fuel as authorized contents,
- change the definition of Damaged Fuel Assembly, and
- add a definition of Undamaged Fuel Assembly.

Changes were made to incorporate the fuel specific to the HB plant into the CoC.

The damaged fuel definition is changed to:

> Damaged Fuel Assemblies are fuel assemblies with known or suspected cladding defects, as determined by review of records, greater than pinhole leaks or hairline cracks, empty fuel rod locations that are not filled with dummy fuel rods, missing structural components such as grid spacers, whose structural integrity has been impaired such that geometric rearrangement of fuel or gross failure of the cladding is expected based on engineering evaluations, or that cannot be handled by normal means. Fuel assemblies that cannot be handled by normal means due to fuel cladding damage are considered FUEL DEBRIS.

The undamaged fuel definition added is:

> Undamaged Fuel Assemblies are fuel assemblies where all the exterior rods in the assembly are visually inspected and shown to be intact. The interior rods of the assembly are in place; however, the cladding of these rods is of unknown condition. This definition only applies to Humboldt Bay fuel assembly array/class 6x6D and 7x7C.

These changes were made for consistency with the corresponding definitions in the HI-STORM Storage CoC (72-1014).

1.3 Generic Changes

The applicant has proposed the following generic changes that are not specific to the HI-STAR HB:

- delete the contents of CoC Section 6.(a) and replace with a direct reference to SAR Chapter 7.
- in the revised Chapter 7, a change was made to the closure plate bolt torque-from 2895 ft.-lbs to 2000 ft.-lbs.
- delete the contents of CoC Section 6.(b) and replace with a direct reference to SAR Chapter 8
- in the revised Chapter 8, dimensional and B-10 loading requirements that are already specified on the licensing drawings are not repeated.
- change the minimum enrichment from 1.8 wt%^{235}U to 1.45 wt%^{235}U in CoC Appendix A, Sections II.A.1.d.i, II.A.2.d, III.A.1.d, III.A.2.d, and III.A.3.d,
- add two assembly cooling times, burnups, and initial enrichments in CoC Appendix A, Table A.7,
- change drawing to allow the use of SA 350 LF3 as an alternate material to SA203A for the HI-STAR Inner Containment Shell and Port Cover.

The modification of CoC Conditions 6.(a) and 6.(b) to reference SAR Chapters 7 and 8 were made to remove unnecessary duplication and details from the CoC and SAR and provide consistency with other 10 CFR Part 71 CoCs. The structural evaluations in SAR Chapter 2 have been revised to support the closure plate bolt torque.

The changes to cooling times, burnups, and initial enrichments are in response to client needs. SAR Chapter 5 was updated to reflect these changes.

The structural evaluations were updated to show that the Safety Factors remain acceptable for both Normal Conditions of Transport (NCT) and Hypothetical Accident Conditions (HAC) for the allowance of SA 350 LF3 as an alternate material to SA203A for the HI-STAR Inner Containment Shell and Port Cover.

There were other SAR changes that reflected a need to update the SAR and did not require a CoC change.

1.4 Drawings

The applicant has requested approval of changes to Drawing Nos. 3913, 3923, 3930, and 5014-C1765 and added Drawing Nos. 4082, 4102, 4103, and 4113. Drawing 3927 was updated to change its title.

The changes to Drawing No 3913 consist of specifying all necessary details of the buttress plate attachment holes in one callout on sheet 5 rather than spread throughout the drawing. The tolerance on the attachment bolt circle was corrected from +/- 1/8" on Sheet 5 to +/-0.06". The drawing changes corrected an editorial error in detailing the closure plate bolt threads as UNC. The drawing now reflects the actual threads (UN) that are used. The drawing also now reflects the allowance of the use of SA 350 LF3 as an alternate material to SA203A for the HI-STAR Inner Containment Shell and Port Cover.

The changes to Drawing No 3923 include replacing the thru hole in the upper fuel spacer plate with a threaded hole; making the lower plate optional for PWR fuel with control components; changes to eliminate the potential for an undersized condition in the shell, baseplate, and lid assembly; change to the optional lid diameter for the MPC-68 as justified in the structural evaluation; delete the requirements for a secondary containment on plutonium shipments (as allowed by the 10 CFR Part 71 rule change in 2004); and eliminate some redundancy

The changes to Drawing No. 5014-C1765 include changes to update the current licensing drawing to allow for the use of the HI-STAR HB impact limiters as supported in the structural evaluation and editorial changes.

Drawing Nos. 4082, 4102, 4103, and 4113 were added to include the needed packaging specifications for HB fuel.

The staff reviewed the revised set of licensing drawings and finds that the information on the drawings provides an adequate basis for its evaluation against 10 CFR Part 71 requirements. The information on the drawings is consistent with the package as described and evaluated in the SAR.

2.0 STRUCTURAL

Supplement 2.I to the application provides a structural evaluation of the HI-STAR HB package, a shortened version of HI-STAR 100. The organization of the supplement mirrors the format and content of Chapter 2 of the application for the approved HI-STAR 100 package, except it only contains material directly pertinent to the HI-STAR HB.

The staff reviewed the application to revise the Model No. HI-STAR 100 package structural design and evaluation to assess whether the package will remain within the allowable values or criteria for normal conditions of transport (NCT) and hypothetical accident conditions (HAC) as required in 10 CFR Part 71. This application was also reviewed to determine whether the package fulfills the acceptance criteria listed in Section 2 (Structural Review) of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel."

2.1 Structural Design

2.1.1 Design Feature Changes

The HI-STAR HB package consists of three principal structural components: (1) the HI-STAR HB overpack, (2) the MPC-HB, and (3) the impact limiters.

The HI-STAR HB overpack, including materials, configurations of inner and intermediate shells, top and bottom forgings, and closure lids, is structurally identical to those of the HI-STAR 100 except for the shorter overall length, lower package weight, reduced strength of impact limiter crush materials, and smaller-diameter threads on lifting trunnions. Other exceptions, which are determined to have insignificant adverse effects on the structural performance, include the optional use of the stacked SA350 LF3 ring forgings to replace the 2.5-inch thick inner shell and the decreased number and length of enclosure shell radial gussets, which connect the outer and intermediate shells to form the cavity for the placement of Holtite-A neutron shield material

The MPC-HB, which consists of an enclosure vessel (EV) and a fuel basket, is configured to hold up to 80 Humboldt Bay fuel assemblies Except for shorter length, the structural features of the EV are identical to those with the other HI-STAR 100 MPCs, including materials, shell diameters, and dimensions of base plate, lid, and penetrations. The MPC-HB fuel basket, as a honeycombed structural weldment, is similar in construction to other approved HI-STAR 100 MPC basket assemblies. It is held inside the EV cavity against the angle and bar spacers welded to the EV shell

The impact limiters of the HI-STAR HB package maintain identical design in form and dimensions to that of the HI-STAR 100 demonstrated structurally adequate for the 60-g design basis deceleration limits. The crush strengths of the aluminum honeycomb material are reduced, however, to accommodate lighter weight of the HI-STAR HB to ensure that the maximum cask decelerations remain to be bounded by the 60-g design basis limits for the 30-ft, free-drop, hypothetical accident conditions.

2.1.2 Design Criteria and Performance Overview

Section 2.I.0 of the supplement notes that the applicable codes, standards, and design criteria, including the design basis maximum cask decelerations of 60 g for the HI-STAR HB, are identical to those for the HI-STAR 100.

For overpack structural components other than the enclosure shell, the application argued that the reduced length and weight of the HI-STAR HB ensure that all stress-based evaluations performed on the HI-STAR 100 produced lower bound safety factors compared to those that would have been calculated for the HI-STAR HB. Sections 2.I.6.1 and 2.I.6.3 evaluate the modified enclosure shell for the heat and reduced external pressure conditions, respectively. The evaluation concluded and the staff agrees that there exist large stress safety factors against an internal pressure developing from off-gassing of the neutron shield material. The staff also concurs that the change associated with the radial gussets will have minor effects on the global response of the overpack subject to a lateral drop.

Recognizing that the MPC-HB has been demonstrated capable of withstanding a side impact deceleration of 60 g for the Part 72 license for the Humboldt Bay ISFSI, the application concluded, and the staff agrees that no new analyses of the MPC-HB are required as long as other design and test conditions, such as the heat and cold pressure/temperature design bases, remain unchanged from those for the HI-STAR 100 enclosure vessel.

On the basis of the above, the staff focuses its review primarily on the impact limiter crush strengths modification for ensuring that maximum cask decelerations remain to be bounded by the design bases.

2.2 Weights and Centers of Gravity

Table 2.1.2.1 of the supplement lists weights of the impact limiters at 26,000 lbs, the loaded MPC-HB at 59,000 lbs, the combined weight at 161,200 lbs for the overpack plus loaded MPC-HB, and the total package weight at 187.200 lbs. The center of gravity of the loaded package is 61 4 inches above the base of the cask.

2.3 Mechanical Properties of Materials

This Materials Evaluation Report is part of an NRC certification review of Holtec's Model No. HI-STAR 100 as a spent nuclear fuel transportation package under requirements specified in accordance with 10 CFR Part 71

The changes proposed are changes that introduce the HB version of the HI-STAR system (HI-STAR HB and MPC-HB), generic changes that are necessary to support the HI-STAR HB (Metamic neutron absorber), but also apply to the other HI-STAR versions and minor changes not directly related to the HI-STAR HB. These minor changes do not affect the materials currently approved in the HI-STAR 100 System. Fuel assemblies for HB are shorter than typical BWR fuel assemblies; therefore, a shorter version of HI-STAR 100 was designed specifically for use at HB (i.e., one time transportation for use at HB only). The shorter design results in a reduced gross weight. Additionally, a HB specific DFC has been designed and fuel class arrays 6x6D and 7x7C have been characterized and analyzed for the HI-STAR HB.

The staff's materials evaluation of the proposed FSAR revisions is based on whether the applicant meets the applicable requirements of 10 CFR Part 71 for packaging and transportation of radioactive material. The evaluation focused on a brief review of the previously approved HI-STAR 100 System and the specific material modifications requested in the application. The objectives of this material review are to ensure adequate material properties exist.

2.3.1 Description

The following is a discussion of the HI-STAR 100 System (generic), the HI-STAR HB design, materials, and the changes proposed that introduce the HB version of the HI-STAR 100 system (HI-STAR HB and MPC-HB). Table 1.3.3 of the application, "Materials and Components of the HI-STAR 100 System," lists the specific material specifications. The HI-STAR 100 System was briefly reviewed for material properties generic to all components followed by a material review of specific material changes proposed that introduce the HB version HI-STAR HB and MPC-HB. No new determination on the adequacy of material properties was made unless it was used as the basis for the proposed revisions.

The HI-STAR 100 System in general consists of three primary components as follows: the multi-purpose canister (MPC), the overpack (HI-STAR) assembly and a set of impact limiters. The overpack confines the MPC and provides the containment boundary for transport conditions. The MPC is a hermetically sealed, welded cylinder with flat ends and an internal honeycomb fuel basket for storing and shipping spent nuclear fuel (SNF) within the overpack containment boundary. A set of impact limiters attached at both ends of the overpack provide energy absorption capability for NCT and HAC.

There are seven MPC models, all are designed to have similar exterior dimensions, one exception is custom designed for the HB plant (MPC-HB), which is approximately 6.3 feet shorter than the generic Holtec MPC designs Each corresponding fuel basket design varies based on the MPC model

2.3.2 HI-STAR 100 Overpack (generic):

The HI-STAR 100 overpack is a heavy-walled steel cylindrical vessel

The overpack containment boundary is formed by an inner shell welded to a bottom plate and to a heavy top flange with a bolted closure plate (all SA 350-LF3. 3½% nickel alloy steel with good impact toughness for use at low temperatures). The closure plate is machined with two concentric grooves for seals (Alloy X750 – NiCrFe).

The outer surface of the overpack inner shell is reinforced with intermediate shells (SA 516-70 carbon steel for moderate and low temperature service with improved notch toughness) of gamma shielding that are installed to ensure a permanent state of contact between adjacent layers. These layers provide additional strength to the overpack to resist puncture or penetration. Radial channels (SA 515-70, carbon steel intended for intermediate or high temperature service) are vertically welded to the outside surface of the outermost intermediate shell at equal intervals around the circumference acting as fins improving heat conduction to the overpack outer enclosure shell surface and as cavities for retaining and protecting the neutron shielding (Holtite-A). An enclosure shell (SA 515-70) is formed by welding panels between each of the radial channels forming additional cavities. These panels together with the exterior flats of the radial channels form the overpack outer enclosure shell. The Holtite-A is placed into each of the radial cavity segments formed, the outermost intermediate shell, and the enclosure shell panels. Pressure relief devices (rupture disks) are positioned in a recessed area on top of the outer enclosure to relieve internal pressure that may develop as a result of a fire accident and subsequent off-gassing of the neutron shield material. A layer of silicone sponge is positioned within each radial channel acting as thermal expansion foam to compress as the neutron shield expands in the axial direction.

The exposed steel surfaces, except seal seating surfaces, of the overpack and the intermediate shell layers are coated with Thermaline 450 and Carboline 890 to prevent corrosion based on expected service conditions. The inner cavity of the overpack is coated with a material appropriate to its high temperatures and the exterior of the overpack is coated with a material appropriate for fuel pool operations and environmental exposure. The coating applied to the intermediate shells acts as a surface preservative and is not exposed to the fuel pool or ambient environment.

Lifting trunnions are manufactured from a high strength alloy (SB 637 - NiCrFe) and installed in threaded openings to the overpack top flange for lifting and rotating the cask body between vertical and horizontal positions. Pocket trunnions were eliminated from the original design and are no longer considered qualified tie-down devices. For transportation, the HI-STAR 100 System is engineered to be mounted on a transport frame secured to the transporter bed.

2 3.3 Multi-Purpose Canisters (generic)

The HI-STAR 100 MPCs are welded cylindrical structures with flat ends. Each MPC is an assembly consisting of a honeycombed fuel basket, a baseplate, a canister shell, a lid with vent/drain ports and cover plates, and a closure ring (all from Alloy X) Generic MPCs are interchangeable, which have identical exterior dimensions. MPC baskets are formed from an array of plates (Alloy X) welded to each other, such that a honeycomb structure is created.

A series of basket supports (Alloy X) are welded to the inside of the shell position and support the MPC fuel basket Optional aluminum (Alloy 1100) heat conduction elements are installed in some early production models in the peripheral area formed by the basket, the MPC shell, and the basket supports A refined thermal analysis has allowed making this aluminum design feature "optional"

The MPC lid (Alloy X) is a circular plate (fabricated from one piece, or two pieces - split top and bottom) that is edge-welded to the MPC shell. Only the top piece is analyzed as part of the enclosure vessel pressure boundary if the two piece lid design is employed. The bottom piece acts primarily as a radiation shield and is attached to the top piece with a non-structural, non-pressure retaining weld. The MPC lid is equipped with vent and drain ports that are used to remove moisture and gas from the MPC and backfill the MPC with a specified pressure of inert helium gas.

Holtec states that the free volume of the MPC and the annulus between the external surface of the MPC and the inside surface of the overpack containment boundary are filled with 99.995% pure helium gas during fuel loading operations. Following MPC drying operations, the MPC is backfilled with a pre-determined amount of helium gas. The helium backfill ensures adequate heat transfer and provides an inert atmosphere for fuel cladding integrity. The helium gas also provides conductive heat transfer across any gaps between the metal surfaces inside the MPC and in the annulus between the MPC and the overpack containment boundary. Metal conduction transfers the heat throughout the MPC fuel basket, through the MPC aluminum heat conduction elements (if installed) and shell, through the overpack inner steel shell, intermediate steel shells, steel radial connectors, and finally, to the outer neutron shield enclosure steel shell.

The closure ring (Alloy X) is a circular ring edge-welded to the MPC shell and MPC lid. The lid provides sufficient rigidity to allow the entire MPC loaded with SNF to be lifted by the threaded holes in the MPC lid during transfer from the storage only HI-STORM 100 System to the HI-STAR 100 overpack for transportation. Threaded insert plugs (Alloy X) are installed to provide shielding when the threaded holes are not in use.

MPCs are designed to store and transport intact fuel assemblies, damaged fuel assemblies, and fuel classified as fuel debris. Intact SNF can be placed directly into the MPC. Damaged SNF and fuel debris must be placed into a Holtec DFC for transportation inside the MPC and the HI-STAR 100 overpack.

HB damaged fuel and fuel debris will be transported in the MPC-HB.

All MPCs are constructed entirely from stainless steel alloy materials except for the neutron absorber and aluminum vent and drain cap seal washers with no carbon steel parts. All structural components in a HI-STAR 100 MPC will be fabricated of Alloy X. For the MPC design and analysis, any steel part in an MPC may be fabricated from any of the acceptable Alloy X materials listed as follows, except that all steel pieces comprising the MPC shell must be fabricated from the same Alloy X stainless steel type: Type 316, Type 316LN, Type 304 and Type 304LN. Holtec states that the Alloy X approach is accomplished by qualifying the MPC for all mechanical, structural, neutronic, radiological, and thermal conditions using material thermophysical properties that are the least favorable and bounding for the entire group for the analysis in question

2.3.4 Impact Limiters (generic)

Once the HI-STAR 100 overpack is positioned and secured in the transport frame the overpack is fitted with (ALSTAR) aluminum honeycomb impact limiters, one at each end Impact limiters ensure the inertia loadings during NCT and HAC are maintained below design levels

2 3 5 Shielding (generic)

To minimize personnel exposure the HI-STAR 100 System is provided with shielding. Initial attenuation of gamma and neutron radiation emitted by the radioactive SNF is provided by the MPC fuel basket structure built from inter-welded plates (Alloy X) and Boral neutron poison panels with sheathing (Alloy X) attached to the fuel cell walls. The MPC canister shell, baseplate, and lid provide additional thicknesses of steel to further reduce gamma radiation and to a lesser extent, neutron radiation at the outer MPC surfaces.

Primary HI-STAR 100 shielding is located in the overpack and consists of neutron shielding (Holtite-A) and additional layers of steel for gamma shielding. Gamma shielding is provided by the overpack inner, intermediate and enclosure steel shells with additional axial shielding provided by both the bottom steel plate and top closure steel plate. Impact limiters provide an increase in gamma shielding and provide additional distance from the radiation source at the ends of the package during transport. A circular segment of neutron shielding is contained within each impact limiter to provide neutron attenuation.

Both Boral and Metamic are neutron absorber materials made of B_4C and Aluminum. Metamic and Boral are both approved materials for the HI-STAR HB only.

2.3.6 Holtite-A Neutron Shielding (Overpack, generic):

Holtec International states Holtite-A is a poured-in-place solid borated synthetic neutron-absorbing polymer. Holtite-A may be specified with a B_4C content of up to 6.5 wt%, however, Holtite-A is specified with a nominal B_4C loading (finely dispersed powder form) of 1 wt% percent for the HI-STAR 100 System. The nominal specific gravity of Holtite-A is 1.68 g/cm^3 and is reduced for shielding analysis by 4% to 1.61 g/cm^3 to conservatively bound and account for any potential weight loss at the design temperature and any inability to reach theoretical density. The nominal weight concentration of hydrogen is 6.0%. However, all shielding analyses will conservatively assume 5.9% hydrogen by weight in the calculations.

2.3.7 Gamma Shielding Material (Overpack, generic)

Carbon steel is used in successive layers of plate stock form with each layer of the intermediate shells constructed from two halves. Both halves of the shell are sheared, beveled, and rolled to the required radii. The two halves of the second layer are wrapped around the first shell. Each shell half is positioned in its location and while applying pressure using a uniquely engineered fixture, the halves are tack welded. The second layer is made by joining the two halves using two longitudinal welds. Successive layers are installed by repeating this process. The welding of every successive shell provides a certain inter-layer contact

2.3.8 Coolants (generic).

No coolants are used, however, helium is sealed within the MPC internal cavity The annulus between the MPC outer surface and overpack containment boundary is purged and filled with helium gas. The overpack annulus is backfilled with helium gas for heat transfer and seal testing. Concentric (Alloy X750 – NiCrFe) metallic seals in the overpack closure plate prevent leakage of the helium gas from the annulus and provide the containment boundary for the release of radioactive materials.

2.3.9 Chemical and Galvanic Reactions (generic):

Holtec states that no plausible mechanism exists for significant chemical or galvanic reactions in the HI-STAR 100 System during loading operations. The MPC, filled with helium, provides a non-aqueous and inert environment. Corrosion is a long-term time-dependent occurrence. The inert gas environment in the MPC precludes the incidence of corrosion during transportation. Additionally, the only dissimilar material groups in the MPC are (1) the neutron absorber material and stainless steel and (2) aluminum and stainless steel. Neutron absorber materials and stainless steel have been used in close proximity in wet storage for over 30 years. Many spent fuel pools at nuclear plants contain fuel racks, which are fabricated from neutron absorber materials and stainless steel materials, with geometries similar to the HI-STAR 100 MPC. Not one case of chemical or galvanic degradation has been found in fuel racks built by Holtec. This service experience provides a basis to conclude that corrosion will not likely occur in these materials. Furthermore, aluminum rapidly passivates in an aqueous environment, leading to a thin ceramic (Al_2O_3) barrier, which renders the material essentially inert and corrosion-free over long periods of application.

The HI-STAR 100 overpack combines low-alloy and nickel alloy steels, carbon steels, neutron and gamma shielding, thermal expansion foam, and bolting materials. All of these materials have a history of non-galvanic behavior within close proximity of each other. The internal and external carbon steel surfaces of the overpack and closure plates are sandblasted and coated to preclude surface oxidation. The coating does not chemically react with borated water. Therefore, chemical or galvanic reactions involving the overpack materials are unlikely and are not expected. Furthermore, the interfacing seating surfaces of the closure plate metallic seals are clad with stainless steel to assure long-term sealing performance and to eliminate the potential for localized corrosion.

2.3.10 Design Code Applicability (generic)

The ASME Boiler and Pressure Vessel Code (ASME Code), 1995 Edition with Addenda through 1997, is the governing code for the construction of the HI-STAR 100 System. The ASME Code is applied to each component consistent with the function of the structure, system, and components (SSCs) of the HI-STAR 100 System that are labeled Important to Safety (ITS). Some components perform multiple functions and in those cases, the most restrictive code is applied.

The HI-STAR overpack top flange, closure plate, inner shell, and bottom plate are designed and fabricated in accordance with the requirements of ASME Code, Section III, Subsection NB, to the maximum extent practical. The remainder of the HI-STAR overpack steel structure is designed and fabricated in accordance with the requirements of ASME Code, Section III, Subsection NF, to the maximum extent practical.

2.3.11 Acceptance Tests and Maintenance Program (generic).

Weld examinations shall be performed in accordance with written and approved procedures, by qualified personnel. All results, including relevant indications, shall be made a permanent part of the quality records by video, photographic, or other means providing an equivalent retrievable record of weld integrity.

ASME Code Section III and Regulatory Guides 7.11 and 7.12 require that certain materials be tested in order to assure that these materials are not subject to brittle fracture failures. Charpy V-notch testing shall be performed on each plate or forging for the HI-STAR 100 Package containment boundary (overpack inner shell, bottom plate, top flange, and closure plate) in accordance with ASME Code Section III, Subsection NB, Article NB-2300. Weld material used in fabricating the containment boundary shall be Charpy V-notch tested in accordance with ASME Section III, Subsection NB, Articles NB-2300 and NB-2430.

Non-containment portions of the overpack, as required, shall be Charpy V-notch tested in accordance with ASME Section III, Subsection NF, Articles NF-2300, and NE-2430. The non-containment materials to be tested include the intermediate shells, overpack port cover plates, and applicable weld materials.

2.3.12 HI-STAR 100 System, Version for HB:

Holtec states that the HI-STAR 100 System has been expanded to include options specific for use at PG&E's HB plant for dry storage and future transportation of SNF.

HB fuel has a cooling time of more than 25 years and relatively low burnup. Heat load and nuclear source terms of this fuel are substantially lower than the design basis fuel. Peak cladding temperatures and dose rates are below the regulatory limits with a significant margin. All major dimensions and features, such as diameter, wall thickness, flange design, top and bottom thicknesses, are maintained identical to the standard (generic) design.

The HI-STAR HB overpack is a heavy-walled, steel cylindrical vessel identical to the standard (generic) HI-STAR 100 overpack, except that the outer height is approximately 128 inches and the inner height is approximately 115 inches. The HI-STAR HB overpack does not contain radial channels vertically welded to the outside surface of the outermost intermediate shell as installed on the HI-STAR 100 overpack (generic) The HI-STAR HB overpack utilizes neutron shielding (Holtite-A) placed in the annulus region between the multi-layered shells and enclosure shell without connecting ribs. This feature is unique to the "HB" version. The annular shield, a thick layer of a low conductivity material, Holtite-A, retards the lateral transmission of fire heat during hypothetical accidents, which minimizes the heating of the HI-STAR HB package internals and the stored fuel during fires.

MPC-HB is similar to the generic MPC-68F, except it is approximately 114 inches high The MPC-HB is designed to transport up to 80 HB BWR SNF assemblies Damaged HB fuel and fuel debris must be transported in the Holtec custom designed HB DFC

Holtec considers that almost all of the HB fuel assemblies not classified as damaged are intact. however, the inspection records of the HB fuel assemblies precludes classifying the assemblies as intact fuel since the interior rods of the assembly are in an unknown condition These rods are classified as undamaged and can perform all fuel specific and system related functions. even with possible breaches or defects.

Applicable design codes, standards and criteria for the HI-STAR HB are identical to HI-STAR 100 except that the internal surfaces of the intermediate shells will not be coated with a silicone encapsulate due to its lower heat loads.

Differences between the HI STAR HB and HI-STAR 100 are limited to: shorter overall length, lower package weight, reduced strength of impact limiter crush materials, smaller diameter threads on lifting trunnions, and MPC-HB neutron absorber material as follows:

Holtec states that Metamic is a neutron absorber material developed by the Reynolds Aluminum Company for spent fuel reactivity control in dry and wet storage applications. Metamic is requested to be used in the HI-STAR HB. Metallurgically, Metamic is a metal matrix composite (MMC) consisting of a matrix of 6061 aluminum alloy (precipitation hardening, with magnesium and silicon as its major alloying elements) containing Type 1 ASTM C-750 boron carbide. Metamic is characterized by extremely fine aluminum (325 mesh or better) and boron carbide powder. Typically, the average B_4C particle size is between 10 and 15 microns. High performance and reliability of Metamic is derived from the particle size distribution of its constituents, rendered into a metal matrix composite by the powder metallurgy process. This yields uniform homogeneity. For the Metamic sheets used in the MPCs, the extruded form is rolled down into the required thickness.

Metamic has been subjected to an extensive array of qualification tests sponsored by the Electric Power Research Institute (EPRI) which evaluated the functional performance of the material at elevated temperatures (900°F) and radiation levels. Test results, documented in an EPRI report, indicate that Metamic maintains its physical and neutron absorption properties with little variation in its properties from the unirradiated state.

Time required to dehydrate a Metamic equipped MPC is expected to be less when compared to an MPC (generic) containing Boral, due to the absence of interconnected porosities. Analyses performed by Holtec show that the streaming due to particle size is virtually non-existent in Metamic. Metamic is a solid material, therefore, no capillary path through which spent fuel pool water can penetrate Metamic panels can chemically react with aluminum in the interior of the material to generate hydrogen. Chemical reaction of the outer surfaces of the Metamic neutron absorber panels that may occur with water to produce hydrogen transpires rapidly and reduces to an insignificant amount in a short period of time Nevertheless, combustible gas monitoring for Metamic equipped MPCs and purging or exhausting the space under the MPC lid during welding and cutting operations is required until sufficient field experience is gained that confirms that little or no hydrogen is released by Metamic during these operations.

Holtec states that each manufactured lot of neutron shield material shall be tested to verify that the material composition (aluminum and hydrogen), boron concentration, and neutron shield density (or specific gravity) meet specified requirements Testing and installation shall be performed in accordance with written and approved procedures and/or standards and shall become part of the QA record. The material manufacturer's QA program and its implementation shall be subject to review and ongoing assessment, including audits and surveillances. Procedures shall ensure that mix ratios and mixing methods are controlled in order to achieve proper material composition, boron concentration and distribution, and that pours are controlled in order to prevent gaps or voids from occurring in the material Neutron shield integrity shall be verified via measurements either at first use or with a check source over the entire surface of the neutron shield, including the impact limiters

2.3.13 Conclusion

For the purposes of the this review, the staff did revisit previously approved material properties used in the original HI-STAR 100 Cask System application and no new determination on the adequacy of those material properties was made unless it was used as the basis for the proposed revision requested in this application.

Furthermore, staff finds that the HI-STAR HB is composed of materials with a service proven history of use. The staff concludes that the materials and manufacture of Metamic as stated in the application for use in the HI-STAR HB are sufficient. The staff's conclusion regarding the manufacture, qualification, and use of Metamic, for the purposes of this review, is applicable to the HI-STAR HB. Definitions for Fuel Debris, Damaged Fuel Assemblies and Undamaged Fuel Assemblies, added to address the HB fuel assembly limited inspection records for packaging and transportation, are acceptable based on structural, containment, and criticality reviews.

The staff determined that, based on review of the application and supplements, that all material properties used by the applicant for the Model No. HI-STAR 100 for use at HB continue to meet the requirements of 10 CFR Part 71.

2.4 General Standards for All Packages

Section 2.I.4 of the supplement notes that the HI-STAR HB is a shorter and lighter version of the HI-STAR 100, and the design features presented in Section 2.4 of the application continue to apply to the HI-STAR HB. Therefore, the staff finds that the HI-STAR HB package meets the 10 CFR 71.43 requirements for the general standards for all packages.

2.5 Lifting and Tie-Down Standards

Section 2.I.5.1 of the supplement considers a bounding lifting weight of 161,200 lbs of the HI-STAR HB to recalculate the governing section moment and stresses for the trunnions with a slightly smaller diameter at the threaded portion of the trunnion than that for the HI-STAR 100. Considering the NUREG-0612 load multipliers of 6 and 10 for the yield and ultimate section load capacities, respectively, the application determines that all safety margins are greater than 1.0. This meets the 10 CFR 71.45(a) provision, which requires that lifting devices be sized to resist three times the design load without reaching material yield strength.

Section 2.I.5.2 notes that the tie-down devices and the reaction loads in Section 2.5 of the application bound those for the HI-STAR HB, which is shorter and lighter than the HI-STAR 100. On this basis, the staff agrees with the applicant's conclusion that no new analysis needs to be performed for the tie-down devices for satisfying the requirements of 10 CFR 71.45 (b)(1).

Section 2.I.5.3 determines that the ultimate bearing capacity at the trunnion-to-top forging interface is greater than the trunnion load limit. The ultimate moment capacity at and beyond the interface is also greater than the trunnion moment limit. This demonstrates that the trunnion shank reaches ultimate structural capacities prior to the top forging reaching its corresponding ultimate load capacities. Therefore, the staff agrees with the applicant's conclusion that failure of the external shank of the lifting trunnion will not cause loss of any other structural or shielding function of the overpack. This satisfies the excessive load requirements of 10 CFR 71.45(a) and 10 CFR 71.45(b)(3).

2.6 Normal Conditions of Transportation

Heat and Free-Drop. Section 2.I.6.1 of the supplement notes that the operating temperatures for the Humboldt Bay fuel, which are at or below comparable temperatures for the HI-STAR 100 analyses, give relatively higher at-temperature stress allowables for the HI-STAR HB. It includes a bounding stress evaluation of the cask enclosure shell. On this basis, the applicant argued and the staff agrees that, for the same cask free-drop deceleration limits and design pressures/temperatures, all other stress analyses of the HI-STAR HB would have resulted in additional margins compared to those for the HI-STAR 100. Section 2.I.6.7 notes that, as part of the Section 2.I.7 impact limiter drop analyses, the cask decelerations for the 1-ft free drops for the HI-STAR HB are shown to be less than the HI-STAR 100 design basis limits. Hence, the applicant's evaluations satisfy the requirements of 10 CFR 71(c)(1) and 10 CFR 71(c)(7) for the heat condition and free drop tests, respectively.

Cold, Reduced External Pressure, Increased Internal Pressure, and Vibration. Section 2.I.6.3 of the supplement refers to the Section 2.I.6.1, "Bounding Evaluation of the Enclosure Shell," which is also applicable to the reduced external pressure condition. As a result, Sections 2.I.6.2 through 2.I.6.5 note and the staff agrees that no new analyses or other calculations need to be performed for the normal conditions of transport cold, reduced external pressure, increased external pressure, and vibration, respectively, to satisfy the requirements of 10 CFR 71(c)(2), (3), (4), and (5).

Water Spray, Corner Drop, and Compression. The HI-STAR HB is identical to the HI-STAR 100 in all respects except for the length of the overpack. As such, Sections 2.I.6.6, 2.I.6.8, and 2.I.6.9 note and the staff agrees that the respective 10 CFR 71(c)(6), (8), and (9) conditions of water spray, corner drop, and compression are not applicable to the HI-STAR HB package.

2.7 Hypothetical Accident Conditions

30-Ft Free Drop. Section 2.7 of the application provides an evaluation of 10 CFR 71.73 hypothetical accident conditions for the HI-STAR 100, which is being modified with limited design feature changes for the HI-STAR HB. Since the applicable design criteria and safety analyses for the HI-STAR 100 remain to be bounding for the HI-STAR HB, the staff focuses its review primarily on the Holtec approach of using analysis alone to establish the design basis cask decelerations associated with the impact limiter modifications.

Drawing No. C1765, sheet 2 of 7, of the application lists reduced nominal crush strengths of aluminum honeycomb materials for the five section types, which range from about 50% to 60% of the corresponding strengths of the HI-STAR 100. The configuration, including overall geometry and attachment to the overpack of the HI-STAR HB impact limiter, is identical to that of the HI-STAR 100, with the sole difference being the impact limiter crush material strengths.

There is no scale model drop testing of the HI-STAR HB. Contrary to the common practice of qualifying impact limiters of spent fuel transportation packages by drop tests, the applicant adapted the HI-STAR 100 differential equation method for evaluating the HI-STAR HB free-drop accidents only by analysis. Section 2.I.7.1 of the supplement provides a summary description of the method as used for the HI-STAR 100. Essentially, the method entails a combined analysis and testing approach in three steps: (1) the 1/8-scale quasi-static tests of the impact limiter to establish its load-deflection characteristics; (2) the ¼-scale cask drop tests to demonstrate that the experimentally obtained cask rigid body decelerations were below the design bases; and (3) the development of analytical models capable of correlating the observed and calculated results.

The HI-STAR 100 analytical models involve either a single second-order differential equation for end drops or a set of three differential equations for side and slap-down drops. There exist, in each equation, one or two resistive force terms each formulated as product of the impact limiter static deformation and dynamic multiplier, also called dynamic correlation function. To implement the method for the HI-STAR HB, given that all other relevant physical attributes of the cask system can be incorporated into the differential equations directly, the dynamic correlation function remains the only parameter that is updated, at each time integration step, as a linear function of the concomitant crush velocity of the impact limiter. In a September 30, 2008, letter the applicant addressed the staff RAI on appropriate selection of the dynamic multipliers for the HI-STAR HB without relying on scale model drop tests. As discussed in Section 2.I.7.1, Revision 13c, of the supplement, the applicant further clarified it by noting that, based on manufacturer's catalog, no information suggesting that the dynamic multipliers in the differential equation method are a function of crush material strength. Therefore, the staff has reasonable assurance to agree with the applicant that the dynamic multipliers originally determined for the HI-STAR 100 remain valid for the HI-STAR HB analytical method, as discussed below.

The numerical simulation analysis of the HI-STAR HB impact limiters uses the same dynamic multipliers as those for the HI-STAR 100, for which, analysis results correlate adequately with the drop test results. Compared to the HI-STAR 100 impact limiters, the staff notes that the HI-STAR HB aluminum honeycomb sections are identically configured, constrained, and supported by an essentially rigid backbone structure. This ensures development of similar load paths for the resistive forces in both the HI-STAR 100 and HI-STAR HB impact limiters for same dynamic multipliers Thus, the staff has reasonable assurance that the HI-STAR 100 differential equation method is an acceptable predictive tool for determining HI-STAR HB cask decelerations and impact limiter crush depths. The staff also agrees with the applicant's assessment that the HI-STAR HB impact limiters will continue to remain attached to the cask if the calculated maximum cask decelerations are below 60 g, given that the HI-STAR 100 impact limiters remained attached to the overpack during the scale model drop tests

Table 2.I 7.1 of the supplement lists calculated maximum cask decelerations and impact limiter crush depths for the 30-ft free drop accidents. With nominal strengths of the crush material, the maximum decelerations are 56.5 g, 45.6 g, 34.8 g, 33.8 g, and 45.9 g for the top-end, bottom-end, side, C.G.-Over-Corner, and slapdown drops, respectively. Table 2.I.7.3 lists the crush strength sensitivity analysis results for ten drop cases. For a crush strength increase of 15% over the nominal, the maximum decelerations are 59 g, 38.5 g, and 49.2 g for the respective top-end, side, and slapdown drops, which are all below the design basis limit of 60 g For the 30-ft side drop, for a decrease of crush strength by 15%, the calculated crush depth of 15.2 inch indicates that the impact limiters experience some material lockup. This results in a cask side-drop deceleration incrrease from the nominal 34.8 g rise to 43 g, as it would be. However, the impact limiters still provide acceptable protection for the cask subject to a maximum side-drop deceleration far less than the 60 g design basis limits.

On the basis of the review above, the staff concludes that the HI-STAR HB design basis decelerations are bounded by the 60 g used also for the HI-STAR 100 package evaluation. As such, all relevant Section 2.7.1 evaluations for the HI-STAR 100 remain to be applicable to the HI-STAR HB in demonstrating its structural capabilities for meeting the 10 CFR 71.73 (c)(1) free-drop requirements.

Puncture. Section 2.I.7.2 of the supplement notes and the staff agree that the structure at the puncture locations is unchanged from the HI-STAR 100. Hence, no new or modified calculations need be performed for the HI-STAR HB for meeting the puncture test requirements of 10 CFR 71.73(c)(3).

Thermal. Section 2.I.7.3 of the supplement notes the thermal evaluation of fire accident. The staff agrees with the applicant's assessment that no new or modified structural calculations need be performed to qualify the HI-STAR HB for the 10 CFR 71.73(c)(4) fire test requirements.

Immersion - Fissile Material and Immersion – All Packages. The staff evaluated the structural feature differences between the HI-STAR HB and HI-STAR 100 packages and concurs that no new or modified calculations need be performed to qualify the HI-STAR HB for the subject immersion requirements of 10 CFR 71.73(c)(5) and (6).

2.8 Fuel Rods

The Humboldt Bay fuel is shorter than the design basis fuel carried by the HI-STAR 100 and will, therefore, exhibit larger structural margins against the side- and end-drop accidents. Thus, the staff concurs with the applicant's conclusion that the performance of the HI-STAR HB fuel rods is bounded by that of the HI-STAR 100, and is, therefore, acceptable. Table 1.0.1 of the CoC defines undamaged fuel assemblies as those where all exterior rods in the assembly are visually inspected and shown to be intact even if the cladding of interior rods are of unknown condition. Section 2.I.9 of the supplement notes that the exterior fuel rods serve to confine the interior fuel rods, thereby preventing interior fuel rods from dislocating and falling to the bottom of the fuel basket This potential, but limited, reconfiguration of interior fuel rods of undamaged fuel assembly will result in negligible change of fuel mass and center of gravity height, which has insignificant effects on the structural response of the HI-STAR HB. Furthermore, as reviewed in Section 6.4 of this safety evaluation, reconfigured interior fuel rods are acceptable based on the conditions analyzed for criticality control This justifies loading the "undamaged fuel assemblies" into the MPC-HB directly without placing them in the HI-STAR HB damaged fuel containers.

2.9 Review Findings

The staff reviewed the statements and representations in the application by considering the regulations, appropriate Regulatory Guides, applicable codes and standards, and acceptable engineering practices The staff concludes that the structural design has been adequately described and evaluated for meeting the requirements of 10 CFR Part 71

3.0 THERMAL

The staff reviewed the application to revise the Model No. HI-STAR 100 package thermal design and evaluation to assess whether the package temperatures will remain within their allowable values or criteria for NCT and HAC as required in 10 CFR Part 71. This application was also reviewed to determine whether the package fulfills the acceptance criteria listed in Section 3 (Thermal Review) of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel."

3.1 Thermal Review Description

The purpose of this Revision is to facilitate the transport of HB reactor spent fuel; however, some of the changes are generic in nature and apply to the entire family of contents of the HI-STAR 100. Besides the fuel type change, other changes impacting the thermal evaluation are: the addition of Metamic as a neutron absorber material, the addition of a higher emissivity value for stainless steel plates, and the request to remove the thermal acceptance test along with the thermal periodic test.

3.2 Thermal Evaluation

The fuel from HB is low burnup fuel that has been sitting in the spent fuel pool and/or dry cask storage for many years, and as a consequence, the decay heat limit for this physically smaller package is only 2 kW for its associated 80 fuel assemblies. The current approved HI-STAR 100 thermal design limit for BWR fuel is 18.5 kW for 68 fuel assemblies.

Additionally, the applicant has chosen to add Metamic as an additional neutron absorber in the SAR. Since Metamic is made of the same materials as Boral, it exhibits the same thermal conductivity, even though the manufacturing process is different. As a consequence, the same values of thermal conductivity for Boral are used.

A higher value emissivity (0.587) for stainless steel plates was provided, but it was stated, for conservatism, that a lower value of 0.36 continues to be used in the thermal calculations. The staff was concerned that this higher emissivity value may not be conservative for the HAC fire. However, after reviewing the Holtec response, the staff agrees that the starting point for the temperature distribution within the cask would be lower, and as a consequence, the impact of the 30 minute HAC fire would be alleviated by this and the thermal inertia of the massive transportation cask.

From the thermal analysis for HB, the NCT maximum temperature for the fuel cladding is 419°F and the overpack top plate is 129°F These values are considerably less than the design basis HI-STAR 100 BWR fuel of 713°F for the fuel cladding and 162°F for the overpack closure plate Also, since the decay heat load of the fuel for HB is significantly lower than the design basis heat load of the HI-STAR 100 and the HB overpack utilizes a heat shield. the HB thermal loading is bounded by the current design basis of the HI-STAR 100 for NCT and HAC

At the request of the staff, a description of the heat shield (referred to in Table 3 I.5) was added to SAR supplement Section 3.I.1. The heat shield is another term for the neutron shield where the previous radial support channels for the outer enclosure shell of the neutron shield were replaced with gussets at the top and bottom of the shield - so that only neutron shield material would be at the radial centerline of the cask to further inhibit heat input to the fuel during the HAC fire

Additionally, the applicant initially requested that the thermal acceptance test and the thermal periodic test be removed for all package configurations. Since the thermal acceptance test was limited to the first fabricated HI-STAR overpack, and that test was completed satisfactorily, the staff has no objection in removing this commitment. For the periodic thermal test, the staff is confident in the time dependent thermal performance of the HI-STAR 100 packaging materials except for the shielding material Holtite-A, and requested that its time dependent thermal characteristics be evaluated since Holtite-A is a polymer and, as such, is typically susceptible to heat and radiation degradation. In response to the staff's request HOLTEC submitted two reports entitled:

1) "Holtite-A: Results of Pre- and Post- Irradiation Tests and Measurements," HI-2002420, Rev. 1, dated 4/8/03, and

2) "Holtite-A Development History and Thermal Performance Data," HI-2002396, Rev. 3, dated 4/10/03.

The former report documents irradiation aging and performed physical observations; weight and density determinations; dimensional measurements; and neutron attenuation testing or chemical analyses on the Holtite-A samples. The latter report documents that the thermally aged samples of Holtite-A were free of large voids and gaps; visual examination confirmed material was stable (no warping, swelling, or cracking); and that weight loss results were less than 4%. Also, these tests were performed independently and did not evaluate the synergistic effects of dose and temperature. The staff did not find a direct correlation between these reports measurements/conclusions and providing assurance that the thermal conductivity of Holtite-A does not change with time. As a result, the staff requested Holtec to provide additional data of Holtite-A to determine the time dependent effect of radiation and heat on Holtite-A, or continue performing the periodic thermal test. Holtec chose to continue performing the periodic thermal test. This commitment is included in the CoC since the CoC requires that all procedures for acceptance testing and maintenance be developed from the provisions of SAR Chapter 8.

From a review of the proposed application of HB, the staff concludes that adequate justification has been presented to conclude that the material temperature limits of the HI-STAR HB transportation package have been satisfied considering the large design margins stemming from the relatively lower decay heat load and the approximate 300°F difference between the HAC cladding temperature and its limit of 1058°F

3.3 Conclusion

Based on the review of the application, the staff found reasonable assurance that the applicant has demonstrated that the HI-STAR HB package meets the thermal requirements for NCT and HAC as required by 10 CFR Part 71

4.0 CONTAINMENT

The staff reviewed the application to revise the Model No HI-STAR 100 package to verify that the package containment design has been described and evaluated under NCT and HAC as required in 10 CFR Part 71 This application was also reviewed to determine whether the package fulfills the acceptance criteria listed in Section 4 (Containment Review) of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel."

4.1 Containment System Design

The purpose of this Revision request is to facilitate the transport of HB reactor spent fuel. Besides the fuel type change, other changes that could impact the containment evaluation are: the revised A_2 values from 10 CFR Part 71, the removal of the MPC as a secondary containment (including the commitment to leak test it), updated procedures to change the closure bolt torque to 2000 ft-lbs from 2895 ft-lbs, and the changed definition of "damaged fuel assembly."

4.2 Containment Evaluation

The HB spent fuel is bounded by the design basis fuel for source term, and consequently, its reference leak rates are more than that of the design basis fuel and result in no impact on the previously approved HI-STAR 100 package. Furthermore, the reference leak rate calculations, presented in Table 4.1.1, "Summary of Containment Boundary Design Specifications," of the SAR continue to justify a leakage rate acceptance criterion of 4.3 E-6 atm cm^3/sec, He. This leakage rate acceptance criterion has been verified to be valid for the HB fuel, as well as for the changes in the A_2 values.

Since the double containment requirement for plutonium shipments has been removed from the regulations (ref. 10 CFR 71.63 dated 1/26/2004), the requirement to have the MPC serve as a second containment boundary and be leak tested is no longer required and that information has been removed from the SAR.

The lessening of the closure bolt torque values has been reviewed in the structural section and staff agrees that these new torque values still provide an adequate closure force to ensure integrity of the containment boundary.

The change in the definition of a damaged fuel assembly has no effect upon the containment evaluation because it has no effect on the source term.

Also, the latest version of ANSI N14.5 (i.e., 1997) was referenced by the applicant at the request of staff

4 3 Conclusion

Based on the statements and representations in the application, staff agrees that the applicant has shown that the use of the Model No. HI-STAR 100 for use at HB continues to meet the containment requirements of 10 CFR Part 71

5.0 SHIELDING

The staff reviewed the application to revise the Model No HI-STAR 100 package to verify that the shielding design has been described and evaluated under NCT and HAC, as required in 10 CFR Part 71 This application was also reviewed to determine whether the package fulfills the acceptance criteria listed in Section 5 (Shielding Review) of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel"

5 1 Description of Shielding Design

5.1.1 Packaging Design Features

Staff reviewed the changes to the design features proposed in the amendment request. The following changes were proposed that affect the shielding design:

Authorization of Metamic as a neutron absorber material was added to the application for use in the HI-STAR HB. The shielding analysis takes credit for the presence of the absorber plates. The currently approved design uses Boral as the neutron absorber.

Addition of the HB overpack, MPC, and basket (and HB-specific DFC). The design for the HB system differs from the standard HI-STAR 100 system in a few parameters. A significant difference is the reduced axial height of the overpack and the MPC. The shielding materials are unchanged. An additional difference is the lack of radial channels welded to the outermost intermediate steel shell of the HB overpack; thus, the neutron shield is not penetrated by steel channels that will result in neutron streaming paths through the neutron shield. Additionally, the HB overpack neutron shield thickness is a minimum of 4 inches, while the minimum thickness is 4.3 inches for the standard overpack. The MPC-HB basket holds 80 HB fuel assemblies (versus the 68 assemblies of the standard BWR MPC baskets).

No other proposed changes affect the shielding design. The staff reviewed the licensing drawings and descriptions of the HI-STAR 100 system as modified in the proposed amendment and finds there is sufficient detail to perform a shielding evaluation. The staff reviewed the proposed changes and finds them to be described in sufficient detail to perform an evaluation of the shielding design.

5.1.2 Codes and Standards

The applicant continues to use the flux-to-dose conversion factors from ANSI 6.1.1-1977. The staff finds use of these conversion factors to be acceptable.

5 1 3 Summary Table of Maximum Radiation Levels

The summary dose rate tables for the MPC-24, MPC-32, and MPC-68 are provided in Section 5.5 of the amended SAR Some of the dose rates for all three MPCs were modified due to a modification of the impact limiter model used in the analyses. Dose rates for the MPC-68 were modified to account for the proposed additional contents at dose locations where the proposed additional contents result in higher dose rates. The tables show maximum dose rates for the side, top, and base of the HI-STAR 100 to be below the 10 CFR Part 71 regulatory dose rate limits. Staff reviewed the dose rates presented in Section 5.5 of the SAR and finds they are consistent with those reported in the rest of the analysis and are the maximum dose rates

5.2 Source Specification

The staff reviewed the specifications for the contents proposed in the amendment request No changes were proposed to the allowable PWR contents. Proposed changes to the BWR contents include a decreased minimum allowable enrichment for intact and damaged 6x6A, 6x6C, 7x7A, and 8x8A assembly arrays/classes and fuel debris from these assembly classes (the proposed minimum enrichment is 1.45 wt.%) and two new maximum burnup, minimum cooling time, and minimum enrichment combinations for the remaining ZR-clad BWR assembly arrays/classes. One new combination is a maximum burnup of 10,000 MWd/MTU and a minimum of 5 years cooling for assemblies with a minimum enrichment of 0.7 wt.%, and the other combination is a maximum burnup of 20,000 MWd/MTU and a minimum of 7 years cooling for assemblies with a minimum enrichment of 1.35 wt.%.

The amendment also proposes to remove the HB assembly arrays/classes from the allowable contents of the MPC-68 and MPC-68F and allow transportation of these assemblies in the MPC-HB. The amendment proposes to allow transportation of the HB assemblies, 6x6D and 7x7C assembly arrays/classes, as intact, undamaged and damaged assemblies and fuel debris. Damaged fuel assemblies and fuel debris must be loaded in HB Damaged Fuel Canisters (DFC). The applicant proposes to load either 28 damaged assemblies in the peripheral basket locations or 40 damaged assemblies in a checkerboard pattern. These two configurations are illustrated in SAR Figures 6.I.3 and 6.I.4.

Based upon HB fuel records, there are no more than 337 linear inches of fuel fragments remaining at the HB reactor site. This fuel debris may include loose zirconium-clad pellets, stainless steel-clad pellets, unclad pellets and rod segments. Assuming all the fragments are clad with stainless steel and considering the cladding dimensions, the amount of stainless steel cladding that could be loaded is 1.25 kilograms. Therefore, the applicant proposed a limit for fuel debris of 1.5 kilograms of stainless steel cladding per cask. This limit is included in Section VII of Table A.1 in Appendix A to the CoC. The shielding analysis also includes the contribution of the steel cladding to the source terms used to demonstrate compliance with the regulatory dose rate limits. The design-basis HB assembly is the 6x6D class assembly with the characteristics provided in Table 5.I.1 of the SAR.

A definition for undamaged fuel was proposed to be added to the CoC. This definition is strictly limited in application to HB fuel assembly classes/arrays. Undamaged assemblies are those HB assemblies for which the condition of the assembly could not be completely verified to meet the CoC definition of intact fuel; however, visual examinations of these assemblies supported the determination that the outer rods of the assemblies are intact. These examinations also confirmed that the assembly interior rods are in place but could not confirm the condition of the cladding of the interior rods Thus, any changes to the assembly configuration due to accident conditions would only take place in the assembly interior and be confined by the outer rods to the assembly envelope (see Section 2.3.12 of this SER). The shielding analysis considers assemblies of this condition (see Section 5.4 of this SER).

5.2 1 Gamma Source

The applicant used the SAS2H and ORIGEN-S modules of the SCALE suite of codes to calculate the source terms, both neutron and gamma, for the proposed contents. SAR Table 5.2.5 lists the gamma source strength for design-basis ZR-clad BWR assemblies for the different maximum burnup, minimum cooling time, and minimum enrichment combinations analyzed. Table 5.2.6 lists the gamma source strength for the Dresden 1 assembly arrays/classes Table 5.I.3 lists the gamma source strength for the HB assembly arrays/classes Table 5.2.10 lists the Cobalt-60 source strength per assembly from assembly hardware, with Table 5.I.4 as the equivalent table for the HB assembly hardware. The staff reviewed the gamma source strengths provided in these tables and also performed confirmatory calculations of the gamma sources. Based upon its review and calculations, the staff finds the calculated source strengths to be acceptable for the proposed contents.

5.2.2 Neutron Source

The neutron source strengths for the design-basis ZR-clad BWR assemblies, the Dresden 1 assemblies, and the HB assemblies are provided in SAR Tables 5.2.13, 5.2.14, and 5.I.2, respectively. The staff reviewed these neutron source strengths and performed confirmatory calculations of the neutron sources. Based upon its review and calculations, the staff finds the calculated source strengths to be acceptable for the proposed contents.

5.3 Model Specification

The applicant uses the same shielding models as for analyses performed in the previously approved amendment, with the exception of a correction to the thickness of the impact limiter ribs. Sections 2 and 3 of this SER describe staff's evaluations of the structural and thermal performance of the HI-STAR 100 system as proposed in the amendment. None of the proposed changes were found to exceed the bounding conditions affecting shielding as evaluated in the previously approved amendment. Additionally, the shielding configuration of the HI-STAR HB is generally bounded by the shielding configuration of the design-basis HI-STAR 100. The neutron shield is 0.3 inches thinner for the HB overpack than it is for the standard HI-STAR 100 overpack, a condition which increases the neutron dose rates by about 30%. The applicant addressed this difference in the neutron shield in its evaluation of the HB system. Also, while the amendment proposes to allow use of Metamic as a neutron absorber, in addition to Boral, the models continue to use Boral. The modeled Boron-10 areal densities are the same and the plate thicknesses are essentially the same for the two absorbers; thus, the amount of aluminum and B_4C are essentially the same, resulting in no distinction between the two materials from a shielding perspective. Based on its review of these proposed changes, the staff finds that the shielding models, with the correction to the impact limiters, remain appropriate.

5.4 Evaluation

The applicant performed the shielding analysis with MCNP-4A, the same code the applicant used for the shielding analyses in the previously approved amendment. The applicant calculated the dose rates for design-basis BWR fuel with the proposed maximum burnup and minimum cooling time and enrichment combinations. These dose rates are presented in SAR Tables 5.4.9, 5.4.11, and 5.4.13 for the surface dose rates and two-meter dose rates for normal conditions and one-meter dose rates for accident conditions. The correction of the impact limiter thickness in the model affected the normal conditions dose rates at the cask base and top (surface and two-meter distance); therefore, updated dose rates for normal conditions are provided for all the MPCs in SAR Tables 5.4.8-11, 19, 20, 22-24, 26, 27, 29, 30, 32, and 33 These changes also necessitated updates to the summary tables, SAR Tables 5.5.1-3, used by the applicant to show compliance with the 10 CFR Part 71 regulatory dose limits. The maximum dose rates, however, continue to remain below the regulatory limits

Dose rates were not calculated for the Dresden 1 assemblies with the proposed minimum enrichment. Instead, the assembly neutron and gamma source strengths were compared on a source strength per inch basis with the source strengths of the design-basis BWR fuel assemblies (39,500 MWd/MTU and 14 years cooling). The comparison was made for damaged Dresden 1 assemblies that had reconfigured under accident conditions and was initially done using only the total neutron and gamma source strengths. This comparison showed that the design-basis BWR fuel source bounds the Dresden 1 fuel source. The applicant then also compared the source strengths on an energy-group basis. This second comparison showed that the design-basis neutron and gamma source strengths bound those of the damaged Dresden 1 assemblies over all energy groups except the 1.0 to 1.5 MeV gamma source. However, dose rate calculations showed that the dose rates (on the overpack side) from the damaged Dresden 1 assemblies are significantly lower than those from the design-basis intact BWR assemblies (by about 20%). The applicant concludes, therefore, that the dose rates from the Dresden 1 fuel assemblies will always be bounded by the dose rates from the design basis BWR fuel assemblies. The staff reviewed this information and performed a confirmatory analysis and finds that the dose rates from the design-basis intact BWR assemblies bound the dose rates from the Dresden 1 fuel assembly arrays/classes.

The applicant used a similar source term comparison (on an energy-group basis) between the design-basis intact BWR assemblies and the HB assemblies. In this case, however, since the MPC-HB and MPC-68 hold different numbers of assemblies, the source per inch was multiplied by the number of assemblies in the respective MPC. For accident conditions, the MPC-HB source per inch accounted for reconfiguration of the damaged HB assemblies. The comparison was made with design-basis intact BWR assemblies having a burnup of 24,500 MWd/MTU and 8 years cooling. The applicant's comparison indicated that the MPC-HB source strength over all energy groups (both neutron and gamma) is bounded by the source strength of the design-basis intact BWR assemblies, meaning the HI-STAR HB dose rates are bounded by the HI-STAR 100 containing a MPC-68 loaded with design-basis intact fuel.

This initial comparison looked solely at comparisons for the entire cask loading. However, staff questioned whether this comparison was sufficient given the different loading schemes for damaged fuel in the HB overpack and the relative importance of different MPC basket zones on the dose rates (interior versus exterior basket locations). For example, for side dose rates, fuel assemblies in the outermost rings of basket locations tend to dominate the gamma dose rates. This configuration will provide a better indication of the impact on dose rates of the 28 damaged assembly pattern in the HB overpack. In response to staff's questions, the applicant modified its evaluation to account for the relative importance of the different MPC basket zones to dose rates. The modified evaluation continued to indicate that the HI-STAR HB dose rates are bounded by the HI-STAR 100 containing the MPC-68 The modified evaluation also accounts for the difference in neutron shield thickness, increasing the neutron source by 30%, the assembly hardware, and the stainless steel cladding of fuel debris in the DFCs For accident conditions, the applicant performed further comparisons with the assumption that all assemblies in the HI-STAR HB are damaged. This latter comparison was performed as a bounding approach for addressing the condition of assemblies where the condition of the cladding could not be verified for fuel rods interior to the assembly lattice These assemblies are classified as undamaged, which classification applies only to the HB fuel arrays/classes. This comparison also indicates the dose rates from the HI-STAR HB are bounded by the dose rates from the HI-STAR 100 containing the MPC-68

The staff reviewed this information and also performed its own comparisons of the source terms. The comparisons included accident conditions for both allowable loading configurations of damaged fuel and the presence of undamaged fuel as well as a comparison of the source terms for assemblies loaded in equivalent outer basket cell regions, under both normal and accident conditions. Based on its review of the applicant's analysis and its own comparisons, the staff finds that the dose rates from the HI-STAR HB will be bounded by the dose rates from a HI-STAR 100 loaded with design-basis intact BWR fuel since the HI-STAR HB source strength, on a per inch basis, is bounded by that of the HI-STAR 100 containing design-basis intact BWR fuel in all the evaluated comparison scenarios.

5.5 Conclusion

Based on its review of the information and representations provided by the applicant in the amendment request and the SAR and independent analyses, the staff has reasonable assurance that the changes to the package design and contents satisfy the shielding requirements and dose limits in 10 CFR Part 71.

6.0 CRITICALITY

The purpose of this review is to verify that the proposed amendment meets the criticality safety requirements under normal conditions of transport and hypothetical accident conditions. The objectives include a review of the criticality design features and fuel specifications; review of the configuration and material properties for the HI-STAR HB Overpack; and a review of the methodology and results found in the criticality evaluation.

The staff reviewed the criticality safety analysis to ensure that all credible normal, off-normal, and accident conditions have been identified and their potential consequences on criticality considered such that the HI-STAR HB with the MPC-HB basket configuration meets the following regulatory requirements: 10 CFR 71.31, 71.33, 71.35, and 71.59. The staff's review also involved a determination on whether the cask system fulfills the acceptance criteria listed in Section 6 of NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel."

6.1 Description of Criticality Design

6.1.1 Packaging Design Features

The MPC-HB is a fuel basket with an increased capacity to accommodate up to 80 assemblies while maintaining the same MPC outer diameter. The criticality safety design continues to rely on the geometry of the fuel basket, fuel enrichment limits, and poison plates for criticality control, but has added Metamic as an alternative to the Boral poison plates

Results of the structural and thermal analyses show that the packaging design features important to criticality safety are not adversely affected by the tests specified in 10 CFR 71 71 and 71.73. The staff reviewed the description of the package design and found that the important features were appropriately identified and adequately described The engineering drawings and other information are sufficient to permit an independent evaluation

6.1.2 Summary Table of Criticality Evaluations

A summary of the criticality evaluation results for the HB fuel is reported in Table 6.I.1 of the SAR. The table includes results for a single package and arrays of undamaged and damaged packages including damaged fuel and fuel debris being transported in the MPC-HB.

The results show that the package design meets the requirements of 10 CFR Part 71 for criticality safety. All values of k_{eff}, after being adjusted for uncertainty and biases, fall below the acceptance limit of 0.95 given in the Standard Review Plan (SRP).

6.1.3 Criticality Safety Index

The applicant's analyses considered an infinite array of packages under both normal conditions of transport and hypothetical accident conditions and showed that they were below the acceptable limit. Therefore, the Criticality Safety Index is 0.0 for the package.

6.2 Spent Nuclear Fuel Contents

The applicant requested the addition of the 6x6D and 7x7C fuel assembly types with or without channels in the MPC-HB for transport in a modified (shorter) HI-STAR HB. The maximum assembly average enrichment is 2.6% for intact fuel, undamaged fuel, damaged fuel, and fuel debris. Undamaged fuel assemblies are assemblies where all of the exterior rods are shown to be intact; while the interior rods of the assembly are in place the condition of the cladding on these rods cannot be verified. Damaged fuel and fuel debris must be canned in a Holtec designed Damaged Fuel Container (DFC). Damaged fuel and fuel debris may be loaded into the 28 peripheral cell locations in the basket or into 40 cell locations in a checkerboard pattern in the basket.

The staff has reviewed the description of the spent fuel contents and concludes that it provides an adequate basis for the criticality evaluation.

6.3 General Considerations for Criticality Evaluations

6.3.1 Model Configuration

The applicant used the same general assumptions and modeling methods as previously reviewed for intact fuel, damaged fuel, and fuel debris. Notable modeling features used for this application are: (1) modeling the MPC-HB with 80 fuel assemblies and a shorter length than the current canister and transportation overpack, (2) analyzing variations of assembly positioning within the fuel basket cells where the fuel assemblies are shifted toward the basket center as well as centered in the fuel cells, (3) analyzing the damaged fuel and fuel debris as bare rods of fuel, (4) modeling undamaged fuel assemblies in place of intact assemblies to analyze the effects on reactivity for both approved basket configurations (peripheral and checkerboard), (5) modeling cases with Metamic and Boral plates separately to assess their analytic similarity, and (6) simulating fabrication damage to the poison plates as a hole up to 1-inch diameter

6.3.2 Material Properties

The material properties remained the same as previously reviewed except for the addition of Metamic where 90% credit was taken for the minimum boron content in this type of absorber plate.

6.3.3 Computer Codes and Cross Section Libraries

The applicant used the three-dimensional continuous energy code Monte Carlo N-Particle (MCNP4a) for the criticality analysis. The staff agrees that the codes and cross-section sets used in the analysis are appropriate for this application and fuel system.

6.3.4 Demonstration of Maximum reactivity

In the safety analysis, the applicant used fuel and basket dimensions within the tolerance limits which maximize k_{eff} as were found in previous analyses. These optimum conditions are: maximum active fuel length, maximum fuel pellet diameter, minimum cladding outside diameter, maximum cladding inside diameter, minimum guide type thickness, maximum channel thickness, minimum cell pitch, minimum cell inner dimensions, and nominal cell wall thickness.

The applicant performed a sensitivity analysis for the HB fuel and found that a package was more reactive when all fuel assemblies are shifted toward the center of the basket versus being centered in each cell but found no statistically significant difference between the cases of each poison plate being damaged with a 1 inch diameter hole in its middle versus an undamaged plate. In the final safety analyses, the applicant assumed all fuel assemblies were shifted toward the basket center and that all poison plates were damaged with a 1 inch diameter hole in the center.

Staff found the methods used to identify the parameters values which maximize k_{eff} to be appropriate and found the set of parameters used in the analysis to be acceptable.

6.3.5 Analysis Approach

The applicant performed comparative calculations for the two different HB fuel types and determined that the 6x6D was more reactive than the 7x7C assembly type. Subsequently, the

6x6D was used as the intact fuel type when analyzing a transportation package loaded with bounding combinations of intact, undamaged, and damaged fuel assemblies

In addition to analyses to support the inclusion of HB fuel in the HI-STAR 100, the applicant also analyzed the acceptability of the alternative poison plate material. Analyses were performed with Metamic poison plates and then compared to a cask fabricated with Boral poison plates In the criticality analysis, 75% of the minimum B-10 content is credited for Boral while 90% of the B-10 content is credited for the Metamic. The B-10 content in Metamic is chosen to be lower, and is chosen as the B-10 content so that both materials are the same in the analysis The difference in k_{eff} was not statistically significant

6.4 Single Package Evaluation

The applicant performed calculations for the MPC-HB which show the design is more reactive when internally flooded with full density water versus flooding with lower density water. Thus, a fully flooded package was used in the subsequent single package analyses. The applicant then performed a series of calculations for single HI-STAR HB packages containing the MPC-HB with 6x6D and 7X7C fuel for each case of an unreflected package, full reflection by water, and full reflection of the containment only. This was performed for two different cases: one with either all intact or undamaged assemblies; and one with damaged assemblies loaded with intact or undamaged assemblies. The bounding case was for the model with the damaged fuel (7x7 rod array) and the 6x6D undamaged fuel (in intact assembly positions) in an eccentric assembly positioning with assumed poison plate damage. The subsequent single package calculations used the results for a single unreflected package with full internal moderation but without the assumed poison damage. All of the applicant's results were below the acceptance level of 0.95 for k_{eff}.

6.5 Evaluation of Package Arrays Under Normal Conditions of Transport

The applicant performed calculations for an infinite array of undamaged packages containing the MPC-HB with intact and undamaged assemblies. All of the applicant's results were below the acceptance level of 0.95 for k_{eff} In these calculations, the packages were internally dry and had no moderator between packages.

6.6 Evaluation of Package Arrays Under Hypothetical Accident Conditions

The applicant performed calculations for an infinite array of damaged packages containing the MPC-HB with the limiting case of full internal moderation as found for the single package. All of the applicant's results were below the acceptance level of 0.95 for k_{eff}.

6.7 Benchmark Evaluations

The applicant's benchmarking procedures and methods have not changed and staff's evaluation is provided in a previous SER.

6.8 Evaluation Findings

The staff has reviewed the criticality description and evaluation of the package and concludes that it addressed the criticality safety requirements of 10 CFR Part 71.

The applicant analyzed the reactivity involving approved loading scenarios. The evaluation analyzed the effects of intact fuel, undamaged fuel, and damaged fuel loaded into peripheral and checkerboard loading configurations within the MPC-HB. A number of conservatives were used when determining the bounding condition. Reactivity results were evaluated for both approved loading configurations (peripheral and checkerboard) having bounding combinations of intact, undamaged, and/or damaged fuel.

Section 6.I.4.1 of the application states that assemblies with defects are considered damaged, and need to be placed into damaged fuel containers (DFCs). Calculations were performed in the application to ensure that assemblies with intact rods in the outermost rods but which may also have defects in the inner rods could be loaded without being placed into DFCs As part of the evaluation the rod pitch was varied inside the assembly to analyze the effects of reflection by water. The application shows that loading the MPC-HB with undamaged assemblies (in place of intact assemblies) along with damaged assemblies still yields results which are below the acceptance level of 0.95 for k_{eff}. Staff agree that the undamaged fuel assemblies with parameters described in the SAR report are acceptable to be loaded in the MPC-HB without being placed in a DFC as long as the overall integrity of the assembly (outer intact rod positioning, guide plates, etc.) remains structurally intact (See Section 2.3.12 of this SER).

Staff identified a minor error in the criticality evaluation regarding a supplemental report provided by the applicant (Holtec Report No: HI-2033010). Table C-1 stated that the bounding condition used for the 6x6 array with full water reflection came from C-55, which corresponds to a case where the model is centered. This is not consistent with the criticality evaluation found in Chapter 6 of the SAR, which states that the bounding case included the assembly being modeled in an eccentric manner. A telephone conference was conducted in which the applicant verified the assumption of NRC staff that it was merely a typographical error, and the applicant plans to correct this in the next revision of the report.

It should be noted that staff finds the 6x6D and 7x7C fuel assemblies, having parameters listed in the SAR, acceptable for loading into the MPC-HB. This is based on the conservatisms used in the analysis as well as the bounding reactivity in relation to the acceptance level of 0.95.

6.9 Conclusion

Based on the review of the presentations and information supplied by the applicant, staff finds reasonable assurance that the proposed amendment meets the criticality safety requirements of 10 CFR Part 71.

7.0 Package Operations

As part of the amendment, the applicant proposed a significant revision of the package operations. Instead of including the package operations explicitly in the CoC, the applicant proposed to place this information in Chapter 7 of the SAR and incorporate this chapter into the CoC by reference. Several items in Chapter 7 were modified as part of the proposed change in addition to inclusion of operations for the HI-STAR HB. Initially, the chapter included operations descriptions for dry loading as well as wet loading operations. However, due to concerns such as how dry loading operations would be performed and/or limited to prevent fuel oxidation, the applicant removed these operations.

The operations chapter also included a statement that indicated that HI-STAR 100 users could modify the sequence of, add or remove operations as necessary This statement introduces uncertainty into what constitutes the essential package operations, the delineation of which is the purpose of Chapter 7. In response to staff's questions, the applicant removed the statement Also, for operations for which the sequence does not impact the package preparation, the applicant modified Chapter 7 to explicitly indicate the operations which may be performed in any sequence in relation to each other. Staff also noted several important operations and the completion/acceptance criteria for other operations were removed from the descriptions in Chapter 7 as part of the proposed amendment. The applicant restored these descriptions in response to staff's questions

Based upon its review of the descriptions in the application, the staff finds that the package operations meet the requirements of 10 CFR Part 71 and that the operations are adequate to assure the package will be operated in a manner consistent with its evaluation for approval

8.0 Acceptance Tests and Maintenance Programs

As part of the amendment, the applicant proposed a significant revision of the acceptance test and maintenance program. Instead of including this information explicitly in the CoC, the applicant proposed to place this information in Chapter 8 of the SAR and incorporate this chapter into the CoC by reference. A number of items in Chapter 8 were modified as part of the proposed change. The acceptance tests and maintenance program is fully applicable to the HI-STAR HB without modification.

In its review of the Chapter 8 descriptions, the staff found that certain important acceptance criteria and tests related to shielding materials and shielding effectiveness had been removed or modified. In particular, the requirements for gamma shielding materials were not included in the proposed Chapter 8. However, staff considers that acceptance tests and criteria for all shielding materials should be included in the Chapter 8 program descriptions. The applicant modified the neutron shield acceptance test and periodic neutron shield integrity verification test to consist only of the radiation surveys performed prior to transport (and upon package receipt, for the periodic test) as described in Chapter 7 of the SAR. Staff considers that these surveys do not meet the purposes of the acceptance test and periodic integrity verification test of the neutron shield. The pre- and post-transport surveys only serve to ensure the 10 CFR Part 71 dose rate limits are not exceeded for a particular shipment. The acceptance and periodic verification tests ensure that the as fabricated neutron shield performs as designed, by comparing dose rates for given contents with the dose rates estimated by analysis for the same contents. Survey results that differ from estimated results, accounting for uncertainties in the measurements and the calculations, would indicate a problem with the as fabricated neutron shield. The tests in the currently approved CoC, fulfill these conditions. Based upon staff's considerations, the applicant included the acceptance tests and criteria for all shielding materials and modified the proposed acceptance and periodic neutron shield tests to retain the currently approved tests.

For the neutron shielding acceptance tests, the applicant describes two tests. The first test verifies shield integrity and is performed with either a check source (prior to first use) or the loaded contents at first use. This test examines the entire surface of the neutron shield, including the impact limiters. The measurements are compared with calculated values that are representative of either the check source or the loaded contents at first use. The second test, described as a shielding effectiveness test, is performed after the first fuel loading in a similar manner to the first test, except that the test does not cover the entire shield surface. This second test is performed when a check source is used for the shield integrity test Staff has reviewed these acceptance tests and finds that these tests are acceptable. This finding is based upon the verification of the performance of the cask's entire neutron shield, with the test measurements being compared versus calculated values for the respective test source

The applicant, in response to staff's questions, also modified the proposed visual inspections in the acceptance tests in a manner that clarified the tests and their acceptance criteria

Based upon its review of the descriptions in the application, the staff finds that the acceptance tests for the packaging meet the requirements of 10 CFR Part 71 and that the maintenance program is adequate to assure packaging performance during its service life.

CONDITIONS

The CoC has been revised as follows

Condition No. 5(a)(2):

A packaging description HI-STAR HB was added.

Condition Nos. 5(a)(3):

Nine drawings were revised.

Condition No. 5(b)(1):

The definitions of damaged fuel, damaged fuel containers, and fuel debris were modified. The definition of undamaged fuel was added to account for the fuel specific to HB.

Condition Nos. 6.(a) and 6.(b):

Revisions were made to reference SAR Chapters 7 and 8 in the CoC.

Condition No. 7:

Revision to add the maximum gross weight of the HI-STAR HB to the CoC.

Appendix A:

Revisions were made to add some cooling times, burnups, and initial enrichments, add the fuel specifications for the HB fuel, and provide some clarifications.

Condition No. 20:

Allows the use of Revision 6 of this certificate for one year.

CONCLUSION

The staff has reviewed the requested amendment to Certificate of Compliance No. 9261 Based on the statements and representations in the application, as supplemented, and Revision 12 of the SAR, the staff concludes that the requested changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71. Certificate of Compliance No. 9261 for the HI-STAR 100 transport package has been amended as requested by Holtec International

Issued with Certificate of Compliance No 9261, Revision No 7
on May 8, 2009

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a CERTIFICATE NUMBER	b REVISION NUMBER	c DOCKET NUMBER	d PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9235	11	71-9235	USA/9235/B(U)F-96	1 OF	12

2 PREAMBLE

a This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71 "Packaging and Transportation of Radioactive Material"

b This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported

3 THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a ISSUED TO *(Name and Address)*
NAC International
3930 East Jones Bridge Road, Suite 200
Norcross, Georgia 30092

b TITLE AND IDENTIFICATION OF REPORT OR APPLICATION
NAC International, Inc., application dated February 19, 2009

4 CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below

5 (a) Packaging

(1) Model No.: NAC-STC

(2) Description: For descriptive purposes, all dimensions are approximate nominal values. Actual dimensions with tolerances are as indicated on the Drawings.

A steel, lead and polymer (NS4FR) shielded shipping cask for (a) directly loaded irradiated PWR fuel assemblies, (b) intact, damaged and/or the fuel debris of Yankee Class or Connecticut Yankee irradiated PWR fuel assemblies in a canister, and (c) non-fissile, solid radioactive materials (referred to hereafter as Greater Than Class C (GTCC) as defined in 10 CFR Part 61) waste in a canister. The cask body is a right circular cylinder with an impact limiter at each end. The package has approximate dimensions as follows:

Cavity diameter	71 inches
Cavity length	165 inches
Cask body outer diameter	87 inches
Neutron shield outer diameter	99 inches
Lead shield thickness	3.7 inches
Neutron shield thickness	5.5 inches
Impact limiter diameter	124 inches
Package length:	
without impact limiters	193 inches
with impact limiters	257 inches

The maximum gross weight of the package is about 260,000 lbs.

The cask body is made of two concentric stainless steel shells. The inner shell is 1.5 inches thick and has an inside diameter of 71 inches. The outer shell is 2.65 inches thick and has

5.(a)(2) Description (Continued)

an outside diameter of 86.7 inches. The annulus between the inner and outer shells is filled with lead

The inner and outer shells are welded to steel forgings at the top and bottom ends of the cask. The bottom end of the cask consists of two stainless steel circular plates which are welded to the bottom end forging. The inner bottom plate is 6.2 inches thick and the outer bottom plate is 5.45 inches thick. The space between the two bottom plates is filled with a 2-inch thick disk of a synthetic polymer (NS4FR) neutron shielding material

The cask is closed by two steel lids which are bolted to the upper end forging The inner lid (containment boundary) is 9 inches thick and is made of Type 304 stainless steel. The outer lid is 5.25 inches thick and is made of SA-705 Type 630, H1150 or 17-4PH stainless steel. The inner lid is fastened by 42, 1-1/2-inch diameter bolts and the outer lid is fastened by 36, 1-inch diameter bolts. The inner lid is sealed by two O-ring seals. The outer lid is equipped with a single O-ring seal. The inner lid is fitted with a vent and drain port which are sealed by O-rings and cover plates. The containment system seals may be metallic or Viton. Viton seals are used only for directly-loaded fuel that is to be shipped without long-term interim storage.

The cask body is surrounded by a 1/4-inch thick jacket shell constructed of 24 stainless steel plates. The jacket shell is 99 inches in diameter and is supported by 24 longitudinal stainless steel fins which are connected to the outer shell of the cask body. Copper plates are bonded to the fins. The space between the fins is filled with NS4FR shielding material.

Four lifting trunnions are welded to the top end forging. The package is shipped in a horizontal orientation and is supported by a cradle under the top forging and by two trunnion sockets located near the bottom end of the cask.

The package is equipped at each end with an impact limiter made of redwood and balsa. Two impact limiter designs consisting of a combination of redwood and balsa wood, encased in Type 304 stainless steel are provided to limit the g-loads acting on the cask during an accident. The predominantly balsa wood impact limiter is designed for use with all the proposed contents. The predominately redwood impact limiters may only be used with directly loaded fuel or the Yankee-MPC configuration.

The contents are transported either directly loaded (uncanistered) into a stainless steel fuel basket or within a stainless steel transportable storage canister (TSC).

The directly loaded fuel basket within the cask cavity can accommodate up to 26 PWR fuel assemblies. The fuel assemblies are positioned within square sleeves made of stainless steel. Boral or TalBor sheets are encased outside the walls of the sleeves. The sleeves are laterally supported by 31, ½-inch thick, 71-inch diameter stainless steel disks. The basket also has 20 heat transfer disks made of Type 6061-T651 aluminum alloy. The support disks

5.(a)(2) Description (Continued)

and heat transfer disks are connected by six, 1-5/8-inch diameter by 161-inch long threaded rods made of Type 17-4 PH stainless steel.

The TSC shell, bottom plate, and welded shield and structural lids are fabricated from stainless steel. The bottom is a 1-inch thick steel plate for the Yankee-MPC and 1.75-inch thick steel plate for the CY-MPC The shell is constructed of 5/8-inch thick rolled steel plate and is 70 inches in diameter The shield lid is a 5-inch thick steel plate and contains drain and fill penetrations for the canister The structural lid is a 3-inch thick steel plate The canister contains a stainless steel fuel basket that can accommodate up to 36 intact Yankee Class fuel assemblies and Reconfigured Fuel Assemblies (RFAs), or up to 26 intact Connecticut Yankee fuel assemblies with RFAs, with a maximum weight limit of 35,100 lbs. Alternatively, a stainless steel GTCC waste basket is used for up to 24 containers of waste.

One TSC fuel basket configuration can store up to 36 intact Yankee Class fuel assemblies or up to 36 RFAs within square sleeves made of stainless steel. Boral sheets are encased outside the walls of the sleeves. The sleeves are laterally supported by 22 ½-inch thick , 69-inch diameter stainless steel disks, which are spaced about 4 inches apart. The support disks are retained by split spacers on eight 1.125-inch diameter stainless steel tie rods. The basket also has 14 heat transfer disks made of Type 6061-T651 aluminum alloy.

The second fuel basket is designed to store up to 26 Connecticut Yankee Zirc-clad assemblies enriched to 3.93 wt. percent, stainless steel clad assemblies enriched up to 4.03 wt. percent, RFAs, or damaged fuel in CY-MPC damaged fuel cans (DFCs). Zirc-clad fuel enriched to between 3.93 and 4.61 wt. percent, such as Westinghouse Vantage 5H fuel, must be stored in the 24-assembly basket. Assemblies approved for transport in the 26-assembly configuration may also be shipped in the 24-assembly configuration. The construction of the two basket configurations is identical except that two fuel loading positions of the 26-assembly basket are blocked to form the 24-assembly basket.

RFAs can accommodate up to 64 Yankee Class fuel rods or up to 100 Connecticut Yankee fuel rods, as intact or damaged fuel or fuel debris, in an 8x8 or 10x10 array of stainless steel tubes, respectively. Intact and damaged Yankee Class or Connecticut Yankee fuel rods, as well as fuel debris, are held in the fuel tubes. The RFAs have the same external dimensions as a standard intact Yankee Class, or Connecticut Yankee fuel assembly.

The TSC GTCC basket positions up to 24 Yankee Class or Connecticut Yankee waste containers within square stainless steel sleeves. The Yankee Class basket is supported laterally by eight 1-inch thick, 69-inch diameter stainless steel disks. The Yankee Class basket sleeves are supported full-length by 2.5-inch thick stainless steel support walls. The support disks are welded into position at the support walls. The Connecticut Yankee GTCC basket is a right-circular cylinder formed by a series of 1.75-inch thick Type 304 stainless steel plates, laterally supported by 12 equally spaced welded 1.25-inch thick Type 304 stainless steel outer ribs. The GTCC waste containers accommodate radiation activated and surface contaminated steel, cutting debris (dross) or filter media, and have the same external dimensions of Yankee Class or Connecticut Yankee fuel assemblies.

5.(a)(2) Description (Continued)

The Yankee Class TSC is axially positioned in the cask cavity by two aluminum honeycomb spacers. The spacers, which are enclosed in a Type 6061-T651 aluminum alloy shell, position the canister within the cask during normal conditions of transport. The bottom spacer is 14-inches high and 70-inches in diameter, and the top spacer is 28-inches high and also 70-inches in diameter.

The Connecticut Yankee TSC is axially positioned in the cask cavity by one stainless steel spacer located in the bottom of the cask cavity.

5.(a)(3) Drawings

(i) The cask is constructed and assembled in accordance with the following Nuclear Assurance Corporation (now NAC International) Drawing Nos.:

423-800, sheets 1-3, Rev. 14
423-802, sheets 1-7, Rev. 20
423-803, sheets 1-2, Rev. 8
423-804, sheets 1-3, Rev. 8
423-805, sheets 1-2, Rev. 6
423-806, Rev. 7
423-807, sheets 1-3, Rev. 3
423-811, sheets 1-2, Rev. 11
423-812, Rev. 6
423-900, Rev. 6
423-209, Rev. 0
423-210, Rev. 0
423-901, Rev. 2

(ii) For the directly loaded configuration, the basket is constructed and assembled in accordance with the following Nuclear Assurance Corporation (now NAC International) Drawing Nos.:

423-870, Rev. 5
423-871, Rev. 5
423-872, Rev. 6
423-873, Rev. 2
423-874, Rev. 2
423-875, sheets 1-2, Rev. 7

(iii) For the Yankee Class TSC configuration, the canister, and the fuel and GTCC waste baskets are constructed and assembled in accordance with the following NAC International Drawing Nos.:

455-800, sheets 1-2, Rev. 2
455-801, sheets 1-2, Rev. 3
455-820, sheets 1-2, Rev. 2
455-870, Rev. 5
455-871, sheets 1-2, Rev. 8
455-871, sheets 1-3, Rev. 7P2[1]
455-872, sheets 1-2, Rev. 12
455-872, sheets 1-2, Rev. 11P1[1]
455-873, Rev. 4
455-881, sheets 1-3, Rev. 8
455-887, sheets 1-3, Rev. 4
455-888, sheets 1-2, Rev. 8
455-891, sheets 1-2, Rev. 1
455-891, sheets 1-3, Rev. 2PO[1]
455-892, sheets 1-2, Rev. 3
455-892, sheets 1-3, Rev. 3P0[1]
455-893, Rev. 3
455-894, Rev. 2
455-895, sheets 1-2, Rev. 5
455-895, sheets 1-2, Rev. 5P0[1]
455-901, Rev. 0P0[1]
455-902, sheets 1-5, Rev. 0P4[1]
455-919, Rev. 2

[1]Drawing defines the alternate configuration that accommodates the Yankee-MPC damaged fuel can.

5.(a)(3) Drawings (Continued)

(iv) For the Yankee Class TSC configuration, RFAs are constructed and assembled in accordance with the following Yankee Atomic Electric Company Drawing Nos..

YR-00-060, Rev. D3
YR-00-061, Rev. D4
YR-00-062, sheet 1, Rev D4
YR-00-062, sheet 2, Rev D2
YR-00-062, sheet 3, Rev. D1
YR-00-063, Rev. D4
YR-00-064, Rev. D4
YR-00-065, Rev. D2
YR-00-066, sheet 1, Rev. D5
YR-00-066, sheet 2, Rev. D3

(v) The Balsa Impact Limiters are constructed and assembled in accordance with the following NAC International Drawing Nos..

423-257, Rev. 2
423-258, Rev. 2
423-843, Rev. 2
423-859, Rev. 0

(vi) For the Connecticut Yankee TSC configuration, the canister and the fuel and GTCC waste baskets are constructed and assembled in accordance with the following NAC International Drawing Nos.:

414-801, sheets, 1-2 Rev. 1
414-820, Rev. 0
414-870, Rev. 3
414-871, sheets 1-2, Rev. 6
414-872, sheets 1-3, Rev. 6
414-873, Rev. 2
414-874, Rev. 0
414-875, Rev. 0
414-881, sheets 1-2, Rev. 4
414-882, sheets 1-2, Rev. 4
414-887, sheets 1-4, Rev. 4
414-888, sheets 1-2, Rev. 4
414-889, sheets 1-3, Rev. 7
414-891, Rev. 3
414-892, sheets 1-3, Rev. 3
414-893, sheets 1-2, Rev. 2
414-894, Rev. 0
414-895, sheets 1-2, Rev. 4

(vii) For the Connecticut Yankee TSC configuration, DFCs and RFAs are constructed and assembled in accordance with the following NAC International Drawing Nos.:

414-901, Rev. 1
414-902, sheets 1-3, Rev. 3
414-903, sheets 1-2, Rev. 1
414-904, sheets 1-3, Rev. 0

5.(b) Contents

(1) Type and form of material

(i) Irradiated PWR fuel assemblies with uranium oxide pellets. Each fuel assembly may have a maximum burnup of 45 GWD/MTU. The minimum fuel cool time is defined in the Fuel Cool Time Table, below. The maximum heat load per assembly is 850 watts Prior to irradiation, the fuel assemblies must be within the following dimensions and specifications:

Assembly Type	14x14	15x15	16x16	17x17	17x17 (OFA)	Framatome-Cogema 17x17
Cladding Material	Zirc-4	Zirc-4	Zirc-4	Zirc-4	Zirc-4	Zirconium Alloy
Maximum Initial Uranium Content (kg/assembly)	407	469	402.5	464	426	464
Maximum Initial Enrichment (wt% ^{235}U)	4.2	4.2	4.2	4.2	4.2	4.5
Minimum Initial Enrichment (wt% ^{235}U)	1.7	1.7	1.7	1.7	1.7	1.7
Assembly Cross-Section (inches)	7.76 to 8.11	8.20 to 8.54	8.10 to 8.14	8.43 to 8.54	8.43	8.425 to 8.518
Number of Fuel Rods per Assembly	176 to 179	204 to 216	236	264	264	264[(1)]
Fuel Rod OD (inch)	0.422 to 0.440	0.418 to 0.430	0.382	0.374 to 0.379	0.360	0.3714 to 0.3740
Minimum Cladding Thickness (inch)	0.023	0.024	0.025	0.023	0.023	0.0204
Pellet Diameter (inch)	0.344 to 0.377	0.358 to 0.390	0.325	0.3225 to 0.3232	0.3088	0.3224 to 0.3230
Maximum Active Fuel Length (inches)	146	144	137	144	144	144.25

Notes:
(1) - Fuel rod positions may also be occupied by solid poison shim rods or solid zirconium alloy or stainless steel fill rods.

5.(b)(1)(i) Contents - Type and Form of Material - Irradiated PWR fuel assemblies (Continued)

FUEL COOL TIME TABLE
Minimum Fuel Cool Time in Years

	Fuel Assembly Burnup (BU)															
Uranium Enrichment (wt% U-235)	BU ≤ 30 GWD/MTU				30 < BU ≤ 35 GWD/MTU				35 < BU ≤ 40 GWD/MTU				40 < BU ≤ 45 GWD/MTU			
Fuel Type	14x14	15x15	16x16	17x17	14x14	15x15	16x16	17x17	14x14	15x15	16x16	17x17	14x14	15x15	16x16	17x17
1.7≤E<1.9	8	7	6	7	10	10	7	9	--	--	--	--	--	--	--	--
1.9≤E<2.1	7	7	5	7	9	9	7	8	12	13	9	11	--	--	--	--
2.1≤E<2.3	7	7	5	6	9	8	6	8	11	11	8	10	--	--	--	--
2.3≤E<2.5	6	6	5	6	8	8	6	7	10	10	8	9	14	15	12	14
2.5≤E<2.7	6	6	5	6	8	7	6	7	10	9	7	9	13	14	10	12
2.7≤E<2.9	6	6	5	5	7	7	5	6	9	9	7	8	12	12	9	11
2.9≤E<3.1	6	5	5	5	7	7	5	6	9	8	6	8	11	11	8	10
3.1≤E<3.3	5	5	5	5	7	6	5	6	8	8	6	7	10	10	8	9
3.3≤E<3.5	5	5	5	5	6	6	5	6	8	7	6	7	10	10	7	9
3.5≤E<3.7	5	5	5	5	6	6	5	6	7	7	6	7	9	9	7	9
3.7≤E<3.9	5	5	5	5	6	6	5	6	7	7	6	7	9	9	7	9
3.9≤E<4.1	5	5	5	5	6	6	5	6	7	7	6	7	8	9	7	9
4.1≤E≤4.2	5	5	5	5	5	6	5	6	6	7	6	7	8	8	7	9
4.2<E<4.3	--	--	--	5[(1)]	--	--	--	6[(1)]	--	--	--	7[(1)]	--	--	--	9[(1)]
4.3≤E≤4.5	--	--	--	5[(1)]	--	--	--	6[(1)]	--	--	--	7[(1)]	--	--	--	8[(1)]

Notes:
[(1)] - Framatome-Cogema 17x17 fuel only.

5.(b)(1) Contents - Type and Form of Material (Continued)

(ii) Irradiated intact Yankee Class PWR fuel assemblies or RFAs within the TSC. The maximum initial fuel pin pressure is 315 psig. The fuel assemblies consist of uranium oxide pellets with the specifications, based on design nominal or operating history record values, listed below:

Assembly Manufacturer/Type	UN 16x16	CE [1] 16x16	West. 18x18	Exxon [2] 16x16	Yankee RFA	Yankee DFC
Cladding Material	Zircaloy	Zircaloy	SS	Zircaloy	Zirc/SS	Zirc/SS
Maximum Number of Rods per Assembly	237	231	305	231	64	305
Maximum Initial Uranium Content (kg/assembly)	246	240	287	240	70	287
Maximum Initial Enrichment (wt% ^{235}U)	4.0	3.9	4.94	4.0	4.94	4.97 [3]
Minimum Initial Enrichment (wt% ^{235}U)	4.0	3.7	4.94	3.5	3.5	3.5 [3]
Maximum Assembly Weight (lbs)	≤ 950	≤ 950	≤ 950	≤ 950	≤ 950	≤ 950
Maximum Burnup (MWD/MTU)	32,000	36,000	32,000	36,000	36,000	36,000
Maximum Decay Heat per Assembly (kW)	0.28	0.347	0.28	0.34	0.11	0.347
Minimum Cool Time (yrs)	11.0	8.1	22.0	10.0	8.0	8.0
Maximum Active Fuel Length (in)	91	91	92	91	92	N/A

Notes:

[1] Combustion Engineering (CE) fuel with a maximum burnup of 32,000 MWD/MTU, a minimum enrichment of 3.5 wt. percent ^{235}U, a minimum cool time of 8.0 years, and a maximum decay heat per assembly of 0.304 kW is authorized.

[2] Exxon assemblies with stainless steel in-core hardware shall be cooled a minimum of 16.0 years with a maximum decay heat per assembly of 0.269 kW.

[3] Stated enrichments are nominal values (fabrication tolerances are not included).

5.(b)(1) Contents - Type and Form of Material (Continued)

(iii) Solid, irradiated, and contaminated hardware and solid, particulate debris (dross) or filter media placed in a GTCC waste container, provided the quantity of fissile material does not exceed a Type A quantity, and does not exceed the mass limits of 10 CFR 71.15

(iv) Irradiated intact and damaged Connecticut Yankee (CY) Class PWR fuel assemblies (including optional stainless steel rods inserted into the CY intact and damaged fuel assembly reactor control cluster assembly (RCCA) guide tubes that do not contain RCCAs), RFAs, or DFCs within the TSC. The maximum initial fuel pin pressure is 475 psig. The fuel assemblies consist of uranium oxide pellets with the specifications, based on design nominal or operating history record values, listed below:

Assembly Manufacturer/Type	PWR [1] 15x15	PWR [2] 15x15	PWR [3]	CY-MPC RFA[4]	CY-MPC DFC[5]
Cladding Material	SS	Zircaloy	Zircaloy	Zirc/SS	Zirc/SS
Maximum Number of Assemblies	26	26	24	4	4
Maximum Initial Uranium Content (kg/assembly)	433.7	397.1	390	212	433.7
Maximum Initial Enrichment (wt% ^{235}U)	4.03	3.93	4.61	4.61[6]	4.61[6]
Minimum Initial Enrichment (wt% ^{235}U)	3.0	2.95	2.95	2.95	2.95
Maximum Assembly Weight (lbs)	≤ 1,500	≤ 1,500	≤ 1,500	≤ 1,600	≤ 1,600
Maximum Burnup (MWD/MTU)	38,000	43,000	43,000	43,000	43,000
Maximum Decay Heat per Assembly (kW)	0.654	0.654	0.654	0.321	0.654
Minimum Cool Time (yrs)	10.0	10.0	10.0	10.0	10.0
Maximum Active Fuel Length (in)	121.8	121.35	120.6	121.8	121.8

Notes:

1 Stainless steel assemblies manufactured by Westinghouse Electric Co., Babcock & Wilcox Fuel Co., Gulf Gen. Atomics, Gulf Nuclear Fuel, & Nuclear Materials & Man. Co.

2. Zircaloy spent fuel assemblies manufactured by Gulf Gen. Atomics, Gulf Nuclear Fuel, & Nuclear Materials & Man. Co., and Babcock & Wilcox Fuel Co.

3. Westinghouse Vantage 5H zircaloy clad spent fuel assemblies have an initial uranium enrichment > 3.93 % wt. U^{235}.

4 Reconfigured Fuel Assemblies (RFA) must be loaded in one of the 4 oversize fuel loading positions.

5. Damaged Fuel Cans (DFC) must be loaded in one of the 4 oversize fuel loading positions.

6 Enrichment of the fuel within each DFC or RFA is limited to that of the basket configuration in which it is loaded.

5.(b) Contents (Continued)

(2) Maximum quantity of material per package

(i) For the contents described in Item 5.(b)(1)(i) 26 PWR fuel assemblies with a maximum total weight of 39,650 lbs. and a maximum decay heat not to exceed 22.1 kW per package.

(ii) For the contents described in Item 5.(b)(1)(ii) Up to 36 intact fuel assemblies to the maximum content weight limit of 30,600 lbs with a maximum decay heat of 12.5 kW per package. Intact fuel assemblies shall not contain empty fuel rod positions and any missing rods shall be replaced by a solid Zircaloy or stainless steel rod that displaces an equal amount of water as the original fuel rod Mixing of intact fuel assembly types is authorized.

(iii) For intact fuel rods, damaged fuel rods and fuel debris of the type described in Item 5.(b)(1)(ii): up to 36 RFAs, each with a maximum equivalent of 64 full length Yankee Class fuel rods and within fuel tubes. Mixing of directly loaded intact assemblies and damaged fuel (within RFAs) is authorized. The total weight of damaged fuel within RFAs or mixed damaged RFA and intact assemblies shall not exceed 30,600 lbs. with a maximum decay heat of 12.5 kW per package.

(iv) For the contents described in Item 5.(b)(1)(iii): for Connecticut Yankee GTCC waste up to 24 containers of GTCC waste. The total cobalt-60 activity shall not exceed 196,000 curies. The total weight of the waste containers shall not exceed 18,743 lbs. with a maximum decay heat of 5.0 kW. For all others, up to 24 containers of GTCC waste. The total cobalt-60 activity shall not exceed 125,000 curies. The total weight of the waste and containers shall not exceed 12,340 lbs. with a maximum decay heat of 2.9 kW.

(i) For the contents described in Item 5.(b)(1)(iv): up to 26 Connecticut Yankee fuel assemblies, RFAs or damaged fuel in CY-MPC DFCs for stainless steel clad assemblies enriched up to 4.03 wt. percent and Zirc-clad assemblies enriched up to 3.93 wt. percent. Westinghouse Vantage 5H fuel and other Zirc-clad assemblies enriched up to 4.61 wt. percent must be installed in the 24-assembly basket, which may also hold other Connecticut Yankee fuel types. The construction of the two basket configurations is identical except that two fuel loading positions of the 26 assembly basket are blocked to form the 24 assembly basket. The total weight of damaged fuel within RFAs or mixed damaged RFAs and intact assemblies shall not exceed 35,100 lbs. with a maximum decay heat of 0.654 kW per assembly for a canister of 26 assemblies. A maximum decay heat of 0.321 kW per assembly for Connecticut Yankee RFAs and of 0.654 kW per canister for the Connecticut Yankee DFCs is authorized.

5.(c) Criticality Safety Index: 0.0

6 Known or suspected damaged fuel assemblies or rods (fuel with cladding defects greater than pin holes and hairline cracks) are not authorized, except as described in Item 5.(b)(2)(iii)

7. For contents placed in a GTCC waste container and described in Item 5.(b)(1)(iii): and which contain organic substances which could radiolytically generate combustible gases, a determination must be made by tests and measurements or by analysis that the following criteria are met over a period of time that is twice the expected shipment time:

The hydrogen generated must be limited to a molar quantity that would be no more than 4% by volume (or equivalent limits for other inflammable gases) of the TSC gas void if present at STP (i.e., no more than 0.063 g-moles/ft^3 at 14.7 psia and 70°F). For determinations performed by analysis, the amount of hydrogen generated since the time that the TSC was sealed shall be considered.

8. For damaged fuel rods and fuel debris of the quantity described in Item 5.(b)(2)(iii) and 5.(b)(2)(v): if the total damaged fuel plutonium content of a package is greater than 20 Ci, all damaged fuel shall be enclosed in a TSC which has been leak tested at the time of closure. For the Yankee Class TSC the leak test shall have a test sensitivity of at least 4.0 X 10^{-8} cm^3/sec (helium) and shown to have a leak rate no greater than 8.0 X 10^{-8} cm^3/sec (helium). For the Connecticut Class TSC the leak test shall have a test sensitivity of at least 1.0 X 10^{-7} cm^3/sec (helium) and shown to have a leak rate no greater than 2.0 X 10^{-7} cm^3/sec (helium).

9. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The package must be prepared for shipment and operated in accordance with the Operating Procedures in Chapter 7 of the application, as supplemented.

(b) Each packaging must be acceptance tested and maintained in accordance with the Acceptance Tests and Maintenance Program in Chapter 8 of the application, as supplemented, except that the thermal testing of the package (including the thermal acceptance test and periodic thermal tests) must be performed as described in NAC-STC Safety Analysis Report.

(c) For packaging Serial Numbers STC-1 and STC-2, only one of these two packagings must be subjected to the thermal acceptance test as described in Section 8.1.6 of the NAC-STC Safety Analysis Report.

10. Prior to transport by rail, the Association of American Railroads must have evaluated and approved the railcar and the system used to support and secure the package during transport.

11. Prior to marine or barge transport, the National Cargo Bureau, Inc., must have evaluated and approved the system used to support and secure the package to the barge or vessel, and must have certified that package stowage is in accordance with the regulations of the Commandant, United States Coast Guard.

12 Transport by air is not authorized.

13 Packagings must be marked with Package Identification Number USA/9235/B(U)F-96.

14 The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

15 Revision No. 9 of this certificate may be used until May 31. 2010.

16 Expiration date: May 31, 2014

REFERENCES

NAC International, Inc., application dated: February 19, 2009.

As supplemented June 3, 2009.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Eric J. Benner, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date: June 12, 2009

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON D.C. 20555-0001

SAFETY EVALUATION REPORT

Docket No. 71-9235
Model No. NAC-STC Package
Certificate of Compliance No. 9235
Revision No. 11

SUMMARY

On May 29, 2009, the U.S. Nuclear Regulatory Commission, issued Revision 10 to Certificate of Compliance (CoC) No. 9235 for the Model No. NAC-STC package. After issuing Revision 10 of CoC No. 9235, four typographical errors were identified in the title headings by the applicant in an e-mail dated June 3, 2009. On pages 3, 4, and 9, the Revision number in block b. was incorrectly left as 9. It should have been 10. Also, page 4 of 12 is incorrectly identified as page 4 of 122. The errors have been corrected. Changes made to the enclosed CoC are indicated by vertical lines in the margin.

CONDITIONS

No Conditions were changed. The typographical errors in the title headings have been corrected. The references were changed to add the June 3, 2009, supplement.

CONCLUSION

These changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

Issued with Certificate of Compliance No. 9235, Revision No. 11, on June 12, 2009.

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a CERTIFICATE NUMBER	b REVISION NUMBER	c DOCKET NUMBER	d PACKAGE IDENTIFICATION NUMBER	PAGE	PAGES
9226	3	71-9226	USA/9226/B(U)F-85	1 OF	9

2 PREAMBLE

a This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S Department of Transportation or other applicable regulatory agencies including the government of any country through or into which the package will be transported

3 THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a ISSUED TO *(Name and Address)*

General Atomics
3550 General Atomics Court
San Diego, California 92121-1122

b TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

General Atomics application dated
January 6, 2009

4 CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5

a. Packaging

(1) Model No.: GA-4

(2) Description

The GA-4 Legal Weight Truck Spent Fuel Shipping Cask consists of the packaging (cask and impact limiters) and the radioactive contents. The packaging is designed to transport up to four intact pressurized-water reactor (PWR) irradiated spent fuel assemblies as authorized contents. The packaging includes the cask assembly and two impact limiters, each of which is attached to the cask with eight bolts. The overall dimensions of the packaging are approximately 90 inches in diameter and 234 inches long

The containment system includes the cask body (cask body wall, flange, and bottom plate); cask closure; closure bolts; gas sample valve body; drain valve; and primary O-ring seals for the closure, gas sample valve, and drain valve.

<u>Cask Assembly</u>

The cask assembly includes the cask, the closure, and the closure bolts. Fuel spacers are also provided when shipping specified short fuel assemblies to limit the movement of the fuel. The cask is constructed of stainless steel, depleted uranium, and a hydrogenous neutron shield. The cask external dimensions are approximately 188 inches long and 40 inches in diameter. A fixed fuel support structure divides the cask cavity into four spent fuel compartments, each approximately 8.8 inches square and 167 inches long. The closure is recessed into the cask body and is attached to the cask flange with 12 1-inch diameter bolts. The closure is approximately 26 inches square, 11 inches thick, and weighs about 1510 lbs.

5.a. (2) (continued)

The cask has two ports allowing access to the cask cavity. The closure lid has an integral half-inch diameter port (hereafter referred to as the gas sample valve) for gas sampling, venting, pressurizing, vacuum drying, leakage testing, or inerting. A 1-inch diameter port in the bottom plate allows draining, leakage testing, or filling the cavity with water. A separate drain valve opens and closes the port The primary seals for the gas sample valve and drain valve are recessed from the outside cask surface as protection from punctures The gas sample valve and the drain valve also have covers to protect them during transport

Cask

The cask includes the containment (flange, cask body, bottom plate and drain valve seals), the cavity liner and fuel support structure; the impact limiter support structure; the trunnions and redundant lift sockets; the depleted uranium gamma shield; and the neutron shield and its outer shell. The cask body is square, with rounded corners and a transition to a round outer shell for the neutron shield. The cask has approximately a 1.5 inch thick stainless steel body wall, 2.6 inch thick depleted uranium shield (reduced at the corners), and 0.4 inch thick stainless steel fuel cavity liner.

The cruciform fuel support structure consists of stainless steel panels with boron-carbide (B_4C) pellets for criticality control. A continuous series of holes in each panel, at right angles with the fuel support structure axis, provides cavities for the B_4C pellets. The fuel support structure is welded to the cavity liner and is approximately 18 inches square by 166 inches long and weighs about 750 lbs.

The flange connects the cask body wall and fuel cavity liner at the top of the cask, and the bottom plate connects them at the bottom. The gamma shield is made up of five rings, which are assembled with zero axial tolerance clearance within the depleted uranium cavity, to minimize gaps. The impact limiter support structure is a slightly tapered 0.4 inch thick shell on each end of the cask. The shell mates with the impact limiter's cavity and is connected to the cask body by 36 ribs

The neutron shield is located between the cask body and the outer shell. The neutron shield design maintains continuous shielding immediately adjacent to the cask body under normal conditions of transport The details of the design are proprietary The design, in conjunction with the operating procedures, ensures the availability of the neutron shield to perform its function under normal conditions of transport.

Two lifting and tie-down trunnions are located about 34 inches from the top of the cask body, and another pair is located about the same distance from the bottom. The trunnion outside diameter is 10 inches, increasing to 11.5 inches at the cask interface. Two redundant lift sockets are located about 26 inches from the top of the cask body and are flush with the outer skin.

5.a. (2) (continued)

Materials
All major cask components are stainless steel, except the neutron shield, the depleted uranium gamma shield, and the B_4C pellets contained in the fuel support structure. All O-ring seals are fabricated of ethylene propylene

Impact Limiters
The impact limiters are fabricated of aluminum honeycomb, completely enclosed by an all-welded austenitic stainless steel skin. Each of the two identical impact limiters is attached to the cask with eight bolts. Each impact limiter weighs approximately 2,000 lbs

(3) Drawings
The packaging is constructed and assembled in accordance with the following GA Drawing Number:

Drawing No. 031348,
sheets 1 through 19, Revision D (Proprietary Version)
GA-4 Spent Fuel Shipping Cask Packaging Assembly

5.(b) Contents

(1) Type and Form of Material

(a) Intact fuel assemblies. Fuel with known or suspected cladding defects greater than hairline cracks or pinhole leaks is not authorized for shipment.

(b) The fuel authorized for shipment in the GA-4 package is irradiated 14x14 and 15x15 PWR fuel assemblies with uranium oxide fuel pellets. Before irradiation, the maximum enrichment of any assembly to be transported is 3.15 percent by weight of uranium-235 (^{235}U). The total initial uranium content is not to exceed 407 Kg per assembly for 14x14 arrays and 469 Kg per assembly for 15x15 arrays.

c) Fuel assemblies are authorized to be transported with or without control rods or other non-fuel assembly hardware (NFAH) Spacers shall be used for the specific fuel types, as shown on sheet 17 of the Drawings

(d) The maximum burnup for each fuel assembly is 35,000 MWd/MTU with a minimum cooling time of 10 years and a minimum enrichment of 3.0 percent by weight of ^{235}U or 45,000 MWd/MTU with a minimum cooling time of 15 years (no minimum enrichment).

(e) The maximum assembly decay heat of an individual assembly is 0.617 kW. The maximum total allowable cask heat load is 2.468 kW (including control components and other NFAH when present).

5.b. (1) (continued)

(f) The PWR fuel assembly types authorized for transport are listed in Table 1. All parameters are design nominal values.

(2) Maximum Quantity of Material per Package

(a) For material described in 5.b.(1): four (4) PWR fuel assemblies

(b) For material described in 5.b.(1): the maximum assembly weight (including control components or other NFAH when present) is 1,662 lbs The maximum weight of the cask contents (including control components or other NFAH when present) is 6,648 lbs., and the maximum gross weight of the package is 55,000 lbs.

Table 1 - PWR Fuel Assembly Characteristics

Fuel Type Mfr.-Array (Versions)	Design Initial U (kg/assy.)	No. of Fuel Rods	Fuel Rod Pitch (in.)	Pellet Diameter (in.)	Zr Clad Thickness (in.)	Active Fuel Length (in.)
W-15x15 (Std/ZC)	469	204	0.563	0.3659	0.0242	144
W-15x15 (OFA)	463	204	0.563	0.3659	0.0242	144
BW-15x15 (Mk.B,BZ,BGD)	464	208	0.568	0.3686	0.0265	142
Exx/A-15x15 (WE)	432	204	0.563	0.3565	0.030	144
CE-15x15 (Palisades)	413	204	0.550	0.358	0.026	144
CE-14x14 (Ft.Calhoun)	376	176	0.580	0.3765	0.028	128
W-14x14 (Model C)	397	176	0.580	0.3805	0.026	137
CE-14x14 (Std/Gen.)	386	176	0.580	0.3765	0.028	137
Exx/A-14x14 (CE)	381	176	0.580	0.370	0.031	137

5.b(2)(b)(continued)

Fuel Type Mfr.-Array (Versions)	Design Initial U (kg/assy.)	No. of Fuel Rods	Fuel Rod Pitch (in.)	Pellet Diameter (in.)	Zr Clad Thickness (in.)	Active Fuel Length (in.)
W-14x14 (OFA)	358	179	0.556	0.3444	0.0243	144
W-14x14 (Std/ZCA,/ZCB)	407	179	0.556	0.3674	0.0225	145.5
Exx/A-14x14 (WE)	379	179	0.556	0.3505	0.030	142

5.c. Criticality Safety Index (CSI): 100

6. Fuel assemblies with missing fuel pins shall not be shipped unless dummy fuel pins that displace an equal amount of water have been installed in the fuel assembly.

7. In addition to the requirements of Subpart G of 10 CFR 71:

   a. Each package shall be both prepared for shipment and operated in accordance with detailed written operating procedures. Procedures for both preparation and operation shall be developed using the specifications contained within the application. At a minimum, those procedures shall require the following provisions:

      (1) Identification of the fuel to be loaded and independent verification that the fuel meets the specifications of Condition 5.b of the CoC.

      (2) That before shipment the licensee shall:

         (a) Perform a measured radiation survey to assure compliance with 49 CFR 173.441 and 10 CFR 71.47 and assure that the neutron and gamma measurement instruments are calibrated for the energy spectrums being emitted from the package.
         (b) Verify that measured dose rates meet the following correlation to demonstrate compliance with the design bases calculated hypothetical accident dose rates: 3.4 x (peak neutron dose rate at any point on cask surface at its midlength) + 1.0 x (gamma dose rate at that location) $\leq$ 1000 mR/hr.
         (c) Verify that the surface removable contamination levels meet the requirements of 49 CFR 173.443 and 10 CFR 71.87.

7.a.(2) (continued)

(d) Inspect all containment seals and closure sealing surfaces for damage. Leak test all containment seals with a gas pressure rise test after final closure of the package. The leak test shall have a test sensitivity of at least 1 $\times 10^{-3}$ standard cubic centimeters per second of air (std-cm^3/sec) and there shall be no detectable pressure rise. A higher sensitivity acceptance and maintenance test may be required as discussed in Condition 7.b.(5), below

(3) Before leak testing, the following closure bolt and valve torque specifications:

(a) The cask lid bolts shall be torqued to 235 ± 15 ft-lbs.
(b) The gas sample valve and drain valve shall be torqued to 20 ± 2 ft-lbs.

(4) During wet loading operations and prior to leak testing, the removal of water and residual moisture from the containment vessel in accordance with the following specifications:

(a) Cask evacuation to a pressure of 0.2 psia (10 mm Hg) or less for a minimum of 1 hour.
(b) Verifying that the cask pressure rise is less than 0.1 psi in 10 minutes.

(5) Before shipment, independent verification of the material condition of the neutron shield as described in SAR Section 7.1.1.4 or 7.1.2.4.

b. All fabrication acceptance tests and maintenance shall be performed in accordance with detailed written procedures. Procedures for fabrication, acceptance testing, and maintenance shall be developed using the specifications contained within the application and shall include the following provisions:

(1) All containment boundary welds, except the final fabrication weld joint connecting the cask body wall to the bottom plate, shall be radiographed and liquid-penetrant examined in accordance with ASME Code Section III, Division 1, Subsection NB. Examination of the final fabrication weld joint connecting the cask body wall to the bottom plate may be ultrasonic and progressive liquid penetrant examined in lieu of radiographic and liquid penetrant examination.

(2) The upper lifting trunnions and redundant lifting sockets shall be load tested, in the cask axial direction, to 300 percent of their maximum working load (79,500 lbs. minimum) per trunnion and per lifting socket, in accordance with the requirements of ANSI N14.6. The upper and lower lifting trunnions shall be load tested, in the cask transverse direction, to 150 percent of their maximum working load (20,625 lbs. minimum) per trunnion, in accordance with the requirements of ANSI N14.6.

7.b.(continued)

(3) The cask containment boundary shall be pressure tested to 1.5 times the Maximum Normal Operating Pressure of 80 psig. The minimum test pressure shall be 120 psig.

(4) All containment seals shall be replaced within the 12-month period prior to each shipment.

(5) A fabrication leakage test shall be performed on all containment components including the O-ring seals prior to first use. Additionally, all containment seals shall be leak tested after the third use of each package and within the 12-month period prior to each shipment. Any replaced or repaired containment system component shall be leak tested. The leakage tests shall verify that the containment boundary leakage rate does not exceed the design leakage rate of 1×10^{-7} std-cm^3/sec. The leak tests shall have a test sensitivity of at least 5×10^{-8} std-cm^3/sec.

(6) The depleted uranium shield shall be gamma scanned with 100 percent inspection coverage during fabrication to ensure that there are no shielding discontinuities. The neutron shield supplier shall certify that the shield material meets the minimum specified requirements (proprietary) used in the applicant's shielding analysis.

(7) Qualification and verification tests to demonstrate the crush strength of each aluminum honeycomb type and lot to be utilized in the impact limiters shall be performed.

(8) The boron carbide pellets, fuel support structure and fuel cavity dimensions, and ^{235}U content in the depleted uranium shall be fabricated and verified to be within the specifications of Table 2 to ensure criticality safety.

Table 2

Specified Parameter	Minimum	Maximum
B_4C boron enrichment	96 wt% ^{10}B	N/A
Diameter of each B_4C pellet	0.426 in	0.430 in
Height of each B_4C pellet stack	7.986 in	8.046 in
Mass of ^{10}B in each B_4C pellet stack	31.5 g	N/A
Mass of each B_4C pellet stack	43.0 g	45.0 g
Diameter of each fuel support structure hole	0.432 in	0.44 in
Fuel support structure nominal hole pitch	N/A	0.55 in
Fuel support structure hole depth minus B_4C pellet-stack height (at room temperature)	0.009 in	0.129 in
Thickness of each fuel support structure panel	0.600 in	0.620 in
Fuel cavity width	N/A	9.135 in
^{235}U content in depleted uranium shielding material	N/A	0.2 wt%

8. Transport of fissile material by air is not authorized.

9. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.17.

10. Expiration Date: October 31, 2013.

NRC FORM 618
(8-2000)
10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE
FOR RADIOACTIVE MATERIAL PACKAGES

a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9226	3	71-9226	USA/9226/B(U)F-85	9	OF	9

REFERENCES

General Atomics Application for the GA-4 Legal Weight Truck Spent Fuel Shipping Cask, January 6, 2009.

FOR THE U.S NUCLEAR REGULATORY COMMISSION

Eric J. Benner, Chief
Licensing Branch
Division of Spent Fuel Storage and Transportation
Office of Nuclear Material Safety
and Safeguards

Date 2/5/09

APPENDIX II

DETAILS OF RISK ANALYSIS OF ROUTINE, INCIDENT-FREE TRANSPORTATION

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APPENDIX II

DETAILS OF RISK ANALYSIS OF ROUTINE, INCIDENT-FREE TRANSPORTATION

II.1 Introduction

NUREG-0170 (NRC, 1977) documented estimates of the radiological consequences and risks associated with the shipment by truck, train, plane, or barge of about 25 different radioactive materials, including power reactor spent fuel. The estimates were calculated using Version 1 of the RADTRAN code (Taylor and Daniel, 1977), which was developed for the NRC by Sandia National Laboratories specifically to support the conduct of the NUREG-0170 study. RADTRAN Version 6, integrated with the input file generator RADCAT, (Neuhauser, et al.[1], 2000; Weiner, et al., 2009) is the computational tool used in this study.

The basic risk assessment method employed in the RADTRAN code is widely accepted[2]. Changes to the code are tracked by a software quality assurance plan that is consistent with American National Standards Institute guidelines. The incident-free module of an earlier version of RADTRAN, RADTRAN 5.25, was validated by measurement (Steinman, et al, 2002); this module is the same in RADTRAN 6.0, the version used in the current study. Verification and validation of RADTRAN 6.0 are documented in Dennis, et al. (2008).

II.2 The RADTRAN Model of Routine Transportation

II.2.1 Description of the RADTRAN Program

RADTRAN calculates the radiological consequences and risks associated with the shipment of a specific radioactive material in a specific package along a specific route. Shipments that take place without the occurrence of accidents are routine, incident free shipments, and the radiation doses to various receptors (exposed persons) are called "incident-free doses." Since the probability of routine, incident-free shipment is essentially equal to one, RADTRAN calculates a dose rather than a risk for such shipments.[3] The dose from a routine shipment is based on the external dose from the part of the vehicle carrying the radioactive cargo, referred to as the "vehicle" in this discussion of RADTRAN. Doses to receptors from the external radiation from the vehicle depend on the distance between the receptor and the radioactive cargo being transported and the exposure time. Exposure time is the length of time the receptor is exposed to

---

[1] Neuhauser, et al (2000) is the technical manual for RADTRAN 5, and is cited because the basic equations for the incident-free analyses in RADTRAN 6 are the same as those in RADTRAN 5, and the technical manual for RADTRAN 6 is not yet available.

[2] RADTRAN was used ti calculate risks for, e.g., NUREG 0170 (NRC, 1977), the Yucca Mountain FEIS (DOE, 2002) , the re-certification of the Waste Isolation Pilot Plant. RADTRAN today has 600 registered users, about 25 precent of whom are U.S. government contractors and about 25 percent of whom are international. The list of users is proprietary.

[3] The probability of an incident or accident transportation depends on the trip length and would be about 10^{-3}. for a cross-country trip, the probability of routine transportation on such a trip is 1-0.001, or 0.999, or essentially one. For a shorter trip the probability of routine transportation is even closer to one

external emissions from the radioactive cargo. The doses in routine transportation depend only on the external dose rate from the cargo and not on the radioactive inventory of the cargo.

RADTRAN models the vehicle as a spherical radiation source traveling along the route. The source strength is the transport index (TI), 100 times the dose rate in mSv/hour[4] at 1 m from the cask, which is treated as an isotropically radiating virtual source at the center of the sphere, as shown in Figure II-1. (see Neuhauser, et al. for a detailed explanation)

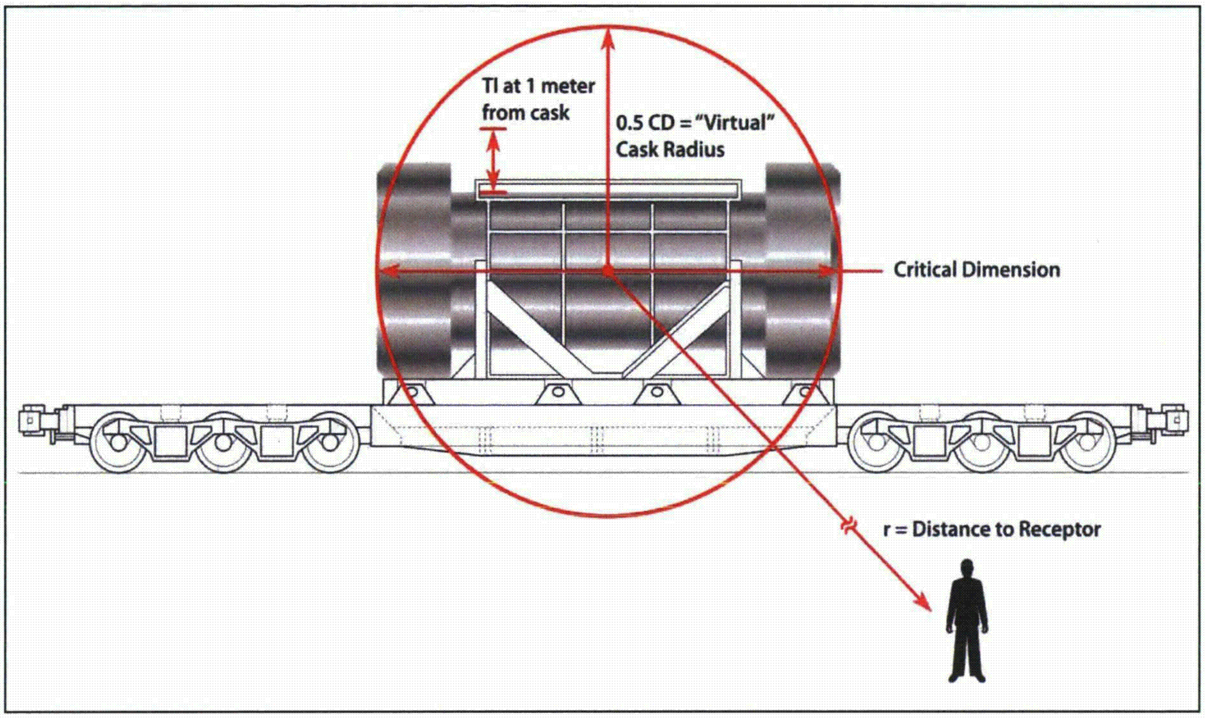

Figure II-1. RADTRAN model of the vehicle in routine, incident-free transportation.

When the distance to the receptor r is much larger than the critical dimension, RADTRAN models the dose to the receptor as proportional to $1/r^2$. When the distance to the receptor r is similar to or less than the critical dimension, as for crew or first responders, RADTRAN models the dose to the receptor as proportional to $1/r$. The TIs for the Rail-Lead and the Rail-Steel casks were calculated from the dose rates at 2 meters as reported in the Safety Analysis Reports of these casks (Holtec International, 2004, NAC international, 2004) and are shown in Table II-1.

The basic equation for calculating incident-free doses to a population along a transportation route is Equation II-1:

---

[4] One mSv = 100 mrem. Thus, 100 times the dose rate in mSv/hr at one meter from the package is equivalent to the dose rate in mrem/hr.

(II-1) $$D(x) = \frac{Qk_0 DR_v}{V} \int_{-\infty}^{\infty} \int_{xmin}^{xmax} \left\{ \frac{(\exp(-\mu r))(B(r))}{r\sqrt{r^2 - x^2}} \right\} dxdr$$

where x is the distance between the receptor and the source, perpendicular to the route

Q includes factors that correct for unit differences

k_0 is the package shape factor[5]

DR_v is the vehicle external dose rate: the TI

V is the vehicle speed

μ is the radiation attenuation factor

B is the radiation buildup factor

r is the distance between the receptor and the source along the route

Details of the application of this and similar equations may be found in Neuhauser et al (2000).

External radiation from casks carrying used nuclear fuel includes both gamma and neutron radiation. For calculating doses from gamma radiation, RADTRAN uses Equation II-2,

(II-2) $$(e^{-\mu r}) * B(r) = 1$$

for conservatism. For calculating doses from neutron radiation, on the other hand, RADTRAN uses Equation II-3

(II-3) $$(e^{-\mu r}) * B(r) = (e^{-\mu r}) * (1 + a_1 r + a_2 r^2 + a_3 r^3 + a_4 r^4)$$

where the coefficients are characteristics of the material.

Equation II-2 can be rewritten (Neuhausr et al, 2000) as Equation II-4

(II-4) $$D(x) = \frac{Qk_0 DR_v}{V} \left[f_\gamma * I_\gamma + f_n * I_n \right]$$

where f_γ is the gamma fraction of the external radiation

f_n is the neutron fraction of the external radiation

I_γ is the double integral in Equation II-1using Equation II-2

I_n is the double integral in Equation II-1using Equation II-3

---

[5] For details on the package shape factor, please see Equations 4 and 5 and accompanying text of Neuhauser et al., (2000).

Collective (population) doses are calculated by integrating over the band along the route where the population resides (the *x* integration in Equation II-1) and then integrating along the route from minus to plus infinity ($-\infty$ to ∞) (the *r* integration in Equation II-1). This is illustrated for a truck route in Figure II-2. The *x* integration limits in Figure II-2 are not to scale: *xmin* is usually 30 m. (200 m near a rail classification stop) and *xmax* is usually 800 m. Integration of x to distances greater than 800 results in risks not significantly different from integration to 800 meters, since the decrease in dose with distance is exponential.

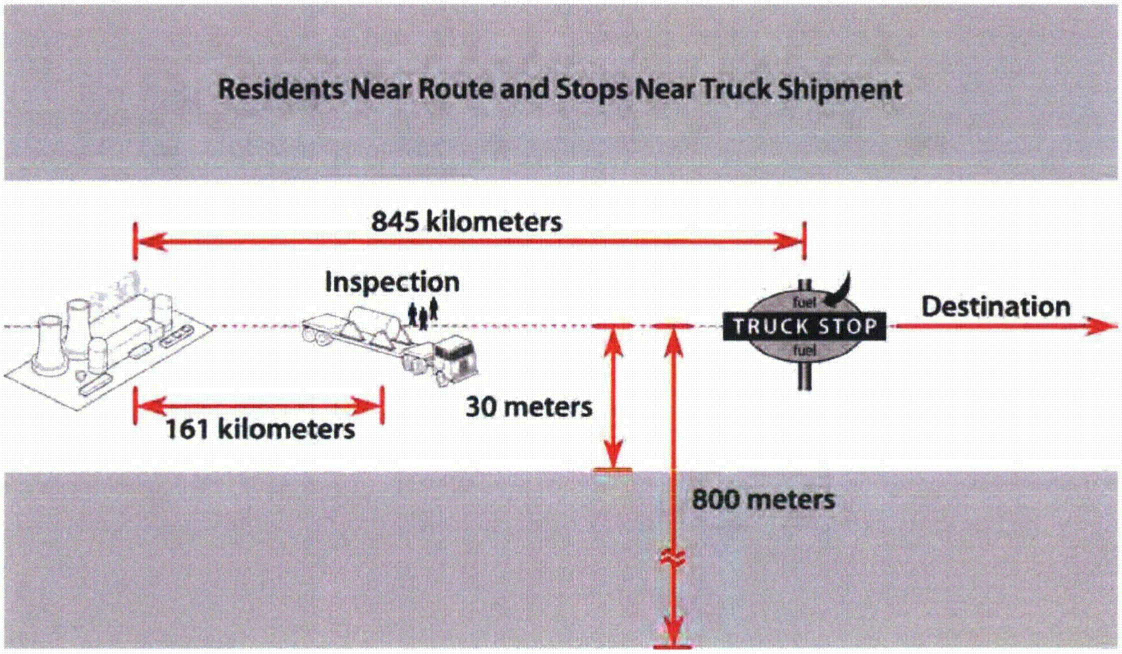

Figure II-2. Diagram of a truck route as modeled in RADTRAN; 845 km is the average distance a very large truck travels on half of its fuel capacity. The 161 km (100 miles) is the distance between spent fuel shipment inspections required by regulation (DOE, 2002).

Variants of Equation II-1 are used to calculate doses to members of the public at stops, vehicle crew members and other workers, occupants of vehicles that share the route with the vehicle carrying the radioactive cargo, and any other receptor identified. Figure II-3 is a diagram of the model used to calculate doses at truck stops. The inner circle defines the area occupied by people who are between the spent fuel truck and the building, and who are not shielded from the truck's external radiation. The dimensions of this circle and the average number of people who occupy it, along with the method used to determine these, are found in Griego et al. (1996).

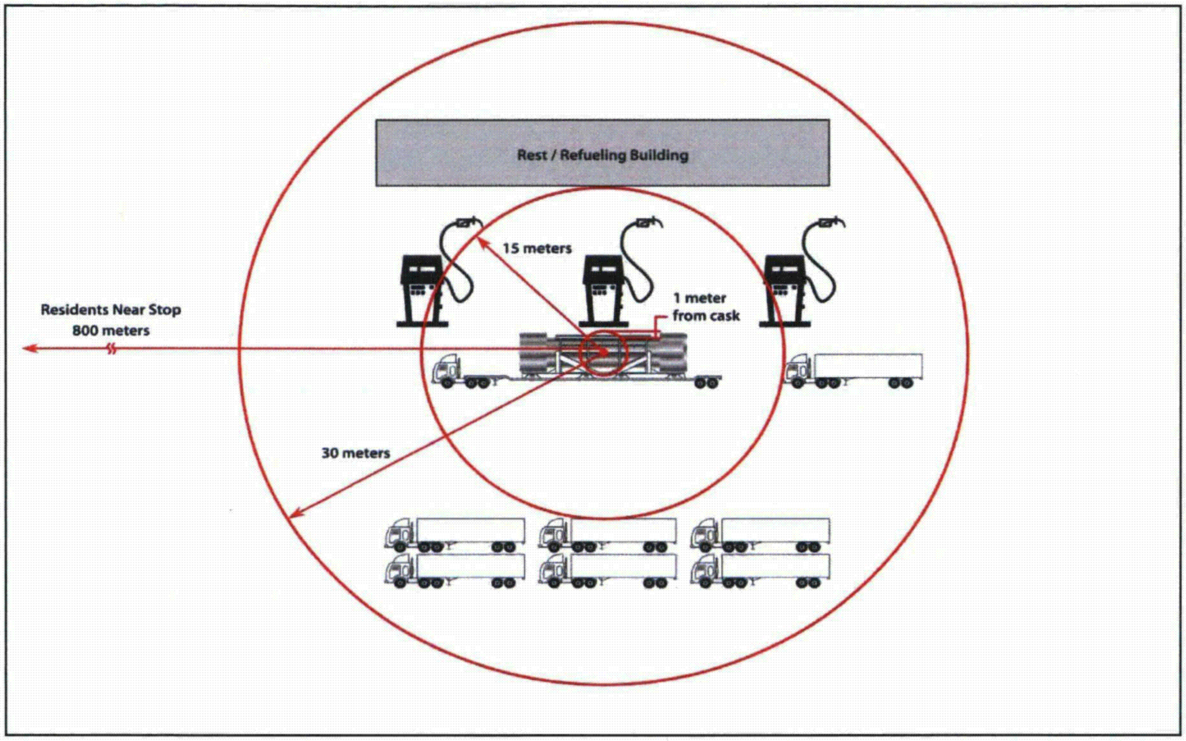

Figure II-3. Diagram of truck stop model.

II.2.2 The RADTRAN Software

This section is a brief description of the RADTRAN software program. A full description of the software and how to use it may be found in the RADCAT User Guide (Weiner, et al., 2009).

The equations that RADTRAN uses, variants of Equation II-1, are programmed in FORTRAN 95. RADTRAN reads in

- an input text file that contains the input parameters as defined by the RADTRAN user,
- a text file that contains an internal library of 148 radionuclides with their associated dose conversion factors and half-lives,
- a binary file that contains the societal ingestion doses for one curie of each radionuclide in the internal radionuclide library, and
- dilution factors and isopleths areas for several weather patterns.

Only the first of these is used in calculating doses from incident-free transportation; the other three are used in the accident analysis and will be discussed in Appendix V.

The input text file can be written directly using a text editor, or can be constructed using the input file generator RADCAT. RADCAT, programmed in XML and running under Java Webstart, provides a series of screens that guide the user in entering values for RADTRAN input parameters. Figure II-4 shows a RADCAT screen.

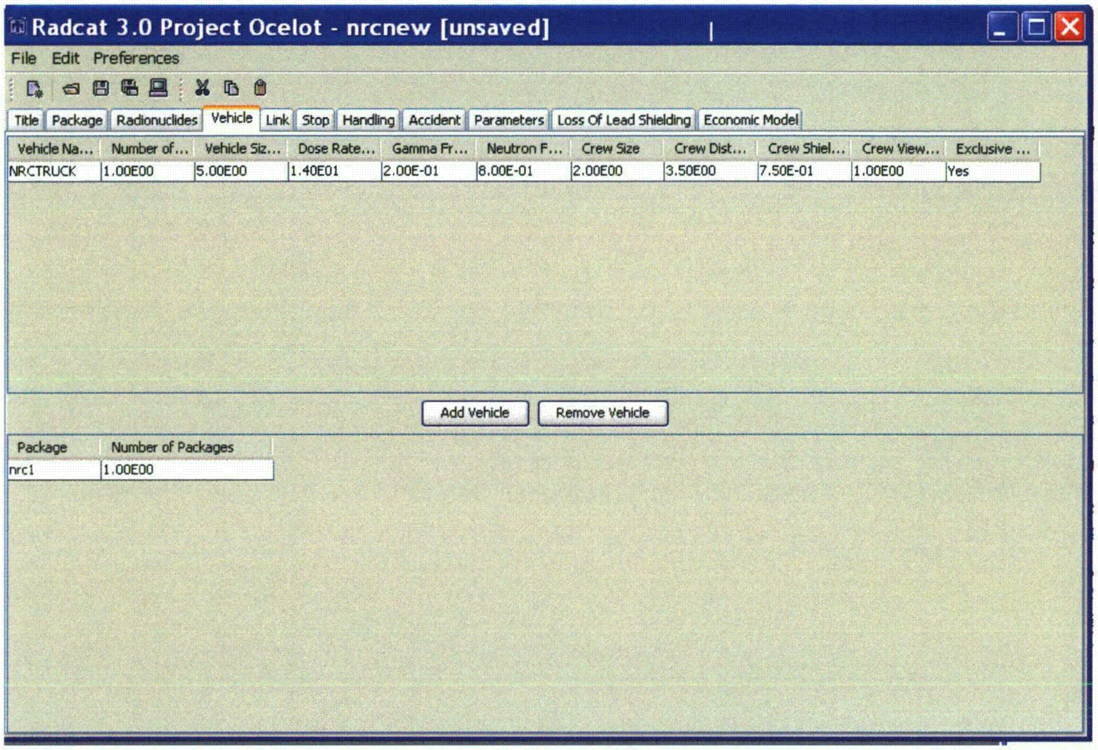

Figure II-4. RADCAT vehicle screen.

RADTRAN output is a text file that can be saved as text or as a spreadsheet.

II.3 RADTRAN Input Parameters

II.3.1 Vehicle-specific Input Parameters

RADTRAN does not allow for the offset of the package from the trailer edge, so the physical dimensions of the package are considered the physical dimensions of the vehicle. Table II-1 shows the vehicle-specific input parameters to RADTRAN and shows the parameter values used in this analysis. The Rail-Steel is modeled transporting canistered fuel; the Rail-Lead is modeled transporting uncanistered fuel. The Truck-DU is a truck cask; the other two are rail casks. In this analysis, the Truck-DU is assumed to be transported by truck; the Rail-Lead and the Rail-Steel, by rail.

Table II-1. Vehicle-specific parameters

	Truck-DU	Rail-Lead	Rail-Steel
Transportation mode	Highway	Rail	Rail
Length (critical dimension)	5.94 m	4.90m	5.08 m
Diameter ("crew view")	2.29 m	2.5 m	3.2 m
Distance from cargo to crew cab	3.5 m	150 m minimum	150 m minimum
TI	14	14.02	10.34
Gamma fraction	0.77	0.89	0.90
Neutron fraction	0.23	0.11	0.10
Number of packages per vehicle	1 per truck	1 per railcar	1 per railcar
Number of crew	2	3	3
Exclusive use?	yes	NA	NA
Dedicated rail	NA	yes	yes
17 × 17 PWR assemblies	4	26	24

II.3.2 Route-Specific Input Parameters

The route parameters are shown in Table II-2 for a unit risk calculation. They are the common input parameters for the 16 specific rail routes and 16 specific truck routes analyzed.

Table II-2. Route parameters for unit risk calculation (DOT 2004a, b)

Parameter	Interstate Highway	Freight Rail
Rural Vehicle speed (U.S. average kph)	108	40.4
Suburban Vehicle speed (U.S. average kph)	102	40.4
Urban Vehicle speed (U.S. average kph)	97	24
Rural Vehicle density (U.S. average vehicles/hr)	1119	17[a]
Suburban Vehicle density (U.S. average vehicles/hr)	2464	17
Urban Vehicle density (U.S. average vehicles/hr)	5384	17
Persons per vehicle	1.5	2
Farm fraction	0.5	0.5
Minimum distance of stop from nearby residents (m)	30	200
Maximum distance of stop from nearby residents (m)	800	800
Stop time for classification (hours)	NA	27
Stop time in transit for railroad change (hours)	NA	variable
Stop time for truck inspections (hours)	0.75	NA
Stop time at truck stops (hours)	0.83	NA
Average number of people sharing the stop	6.9[b]	NA
Minimum distance to people sharing the stop (m)	1[b]	NA
Maximum distance to people sharing the stop (m)	15[b]	NA
Truck stop worker distance from cask (m)	15	NA
Truck stop worker shielding factor	0.018	NA
Truck crew shielding factor	0.377	NA
Escort distance from cask (m)	4	16

[a]Railcars per hr
[b]Griego et al, 1996.

II.3.3. Other Parameters

RADTRAN includes a set of parameters whose values are not generally known by the user and which have been used routinely in transportation risk assessments. RADTRAN contains default values for these parameters, but all default values can be changed by the user. Table II-3 lists the parameter values used in the incident-free analysis.

Table II-3. Parameter values in the RADTRAN 6 analysis

Parameter	Value
Shielding factor for residents (fraction of energy impacting the receptor): R= rural, S=suburban, U=urban	R=1.0 S=0.87 U=0.018
Fraction of outside air in urban buildings	0.25
Fraction of urban population on sidewalk	0.48
Fraction of urban population in buildings	0.52
Ratio of non-residents to residents in urban areas	6
Distance from in-transit shipment for maximum exposure in m. (RMEI exposure)	30
Vehicle speed for maximum exposure in km/hr. (RMEI exposure)	24
Distance from in-transit shipment to nearest resident in rural and suburban areas, m	30
Distance from in-transit shipment to nearest resident in urban areas, m	27
Population bandwidth, m	800
Distance between vehicles (trains), m	3.0
Minimum number of rail classification stops	1

Additional input parameters are rural, suburban, and urban route lengths and population densities, characteristics of stops along a route and the TI of the package.

II.3.4 RADTRAN input and output files

Figure II-5 shows the incident-free unit risk input file for the Truck-DU cask. Figure II-6 shows the incident-free unit risk input files for the Rail-lead-steel and Rail-steel casks. In the interests of space, only the portion of the input files relevant to routine incident-free transportation are shown.

```
OUTPUT CI_REM
FORM UNIT
DIMEN 1 0 18
PARM 0 1 3 0
PACKAGE GA4 14.0 0.77 0.22999999999999998 5.94
VEHICLE -1 GA_4 1.400000E01 0.77 0.23 5.94 1.0 2.0 3.5 0.38 2.29
GA4 1.0
FLAGS
IACC 2
IUOPT 2
REGCHECK 0
MODSTD
DISTOFF FREEWAY 3.000000E01 3.000000E01 8.000000E02
DISTOFF SECONDARY 2.700000E01 3.000000E01 8.000000E02
DISTOFF STREET 5.000000E00 8.000000E00 8.000000E02
DISTON
FREEWAY 1.500000E01
SECONDARY 3.000000E00
STREET 3.000000E00
ADJACENT 4.000000E00
MITDDIST 3.000000E01
MITDVEL 2.400000E01
RR 1.000000E00
RU 1.800000E-02
RS 8.700000E-01
LINK R GA_4 1.0 108.0 1.5 1.0 1119.0 1.0 0.0 R 1 0.5
LINK S GA_4 1.0 108.0 1.5 1.0 2464.0 1.0 0.0 S 1 0.0
LINK U GA_4 0.9 102.0 1.5 1.0 5384.0 1.0 0.0 U 1 0.0
LINK U_RUSH GA_4 0.1 51.0 1.5 1.0 10760.0 1.0 0.0 U 1 0.0
STOP STOP_1 GA_4 9180.0 1.0 15.0 1.0 0.83
STOP RURAL GA_4 1.0 30.0 800.0 1.0 0.83
STOP SUBURBAN GA_4 1.0 30.0 800.0 0.87 0.83
STOP URBAN GA_4 1.0 30.0 800.0 0.018 0.83
EOF
```

Figure II-5. RADTRAN unit risk input file for the Truck-DU cask.

```
RADTRAN 6       July 2008
 && SEE APPENDIX A.2 FOR DETAILS
 && UNIT RISK FACTOR
 && INCIDENT-FREE TRANSPORT
 && AVERGE TI
 && PWR 35GHWD/MTHM BURNUP   10 YEAR COOLED  24 ASSEMBLIES
 && REMARK
 TITLE NAC-STC
 OUTPUT CI_REM
 FORM UNIT
 DIMEN 1 0 18
 PARM 0 1 3 0
 PACKAGE NAC-STC 14.02 0.89 0.10999999999999999 4.9
 PACKAGE HI-STAR_100 10.034 0.9 0.09999999999999998 5.08
VEHICLE -2 NAC-STC 1.400000E01 0.89 0.11 4.9 1.0 3.0 150.0 1.0 2.5
       NAC-STC 1.0
       HI-STAR 0.0
 VEHICLE -2 HI-STAR 1.030000E01 0.9 0.1 5.08 1.0 3.0 150.0 1.0 3.2
       NAC-STC 0.0
       HI-STAR 1.0
 FLAGS
    IACC 2
    ITRAIN 2
    IUOPT 2
    REGCHECK 0
 MODSTD
    DDRWEF 1.800000E-03
    FMINCL 1.000000E00
    DISTOFF RAIL 3.000000E01 3.000000E01 8.000000E02
    DISTON
     RAIL 3.000000E00
    MITDDIST 3.000000E01
    MITDVEL 2.400000E01
    RPD 6.000000E00
    RR 1.000000E00
    RU 1.800000E-02
    RS 8.700000E-01
 LINK NAC_R NAC-STC 1.0 40.4 3.0 1.0 8.0 1.0 0.0 R 3 0.5
 LINK NAC_S NAC-STC 1.0 40.4 3.0 1.0 8.0 1.0 0.0 S 3 0.0
 LINK NAC_U NAC-STC 1.0 24.0 3.0 1.0 17.0 1.0 0.0 U 3 0.0
 LINK HISTAR_R HI-STAR 1.0 40.4 3.0 1.0 8.0 1.0 0.0 R 3 0.5
 LINK HISTAR_S HI-STAR 1.0 40.4 3.0 1.0 8.0 1.0 0.0 S 3 0.0
 LINK HISTAR_U HI-STAR 1.0 24.0 3.0 1.0 17.0 1.0 0.0 U 3 0.0
 STOP CLASSIFICATION NAC-STC 1.0 200.0 800.0 1.0 27.0
 STOP CLASSIFICATION HI-STAR 1.0 200.0 800.0 1.0 27.0
 EOF
```

Figure II-6. RADTRAN unit risk input file for the Rail-Lead and Rail-Steel casks.

II.4. Routes

This study analyzes both the per-km doses from a single shipment on rural, suburban, and urban route segments and doses to receptors from a single shipment between 16 representative pairs of origins and destinations, chosen to represent a range of route lengths and a variety of populations. The actual truck and rail routes were selected for a number of reasons. The combination of four origins and four destinations represent a variety of route lengths and population densities and both private and government facilities, a large number of states, and includes origins and destinations that were included in the analyses of NUREG/CR-6672, thereby permitting comparison of results between the studies.

Power reactor spent fuel (SNF) and high-level radioactive waste (HLW) are currently stored at 77 locations in the U.S. (67 nuclear generating plants, five storage facilities at sites of decommissioned nuclear plants, and five DOE defense facilities). The origin sites (Table II-4) include two nuclear generating plants (Indian Point and Kewaunee), a storage site (Maine Yankee), and a National Laboratory (Idaho National Laboratory). The destination sites include the two proposed repository sites not characterized (Deaf Smith County, TX, and Hanford, WA) (DOE, 1986), the site of the proposed Private Fuel Storage facility (Skull Valley, UT), and a National Laboratory site (Oak Ridge, TN). The routes modeled are shown in Table II-4. The populations within 800 meters of the route were determined from output of the WebTRAGIS (Johnson and Michelhaugh, 2003) routing code, modified by the increase in population between 2000 and 2006 (see Table II-5). Both truck and rail versions of each route are analyzed. These routes are used for illustrative purposes. No actual spent fuel shipments on these routes are occurring or planned.

Table II-4. Specific routes modeled (urban kilometers are included in total kilometers)

Origin	Destination	Population within 800 m (1/2 mile)		Total Kilometers		Urban Kilometers	
		Rail	Truck	Rail	Truck	Rail	Truck
Maine Yankee Site, ME	Hanford, WA	1,647,190	1,129,685	5084	5013	355	116
	Deaf Smith County, TX	1,321,024	1,427,973	3362	3596	211	165
	Skull Valley, UT	1,451,325	1,068,032	4068	4174	207	115
	Oak Ridge, TN	1,146,478	1,137,834	2125	1748	161	135
Kewaunee NP, WI	Hanford, WA	476,914	423,163	3028	3453	60	52
	Deaf Smith County, TX	677,072	494,920	1882	2146	110	60
	Skull Valley, UT	806,115	505,226	2755	2620	126	58
	Oak Ridge, TN	779,613	646,034	1395	1273	126	92
Indian Point NP, NY	Hanford, WA	961,026	869,763	4781	4515	229	97
	Deaf Smith County, TX	1,027,974	968,282	3088	3074	204	109
	Skull Valley, UT	1,517,758	808,107	3977	3672	229	97
	Oak Ridge, TN	1,146,245	561,723	1264	1254	207	60
Idaho National Lab, ID	Hanford, WA	164,399	132,662	1062	959	20	15
	Deaf Smith County, TX	298,590	384,912	1913	2291	40	52
	Skull Valley, UT	169,707	132,939	455	466	26	19
	Oak Ridge, TN	593,680	569,240	3306	3287	75	63

Route segments and population densities are provided by WebTRAGIS. WebTRAGIS uses census data from the 2000 census. Updated population data to 2006 were provided in the 2008 Statistical Abstract (U.S. Census Bureau, 2008). Table 13 of U.S. Census Bureau (2008) shows the percent increase in population for each of the 50 states of the United States, as well as for the U. S. as a whole, and Table 20 shows the percent increase in population for all metropolitan areas in the U.S. with more than 250,000 people. Data from these two tables were combined to give population multipliers for states along routes for which the collective dose and the population increase were significant enough to make a correction.

The population multipliers used are shown in Table II-5. "Significant" was taken to mean that the population difference was more than 1% (i.e., multipliers between 0.99 and 1.01 were not considered significant). The state-specific multiplier was applied to rural and suburban routes through the state (even though some of these routes would be within the largest metropolitan area), and the multiplier for the largest metropolitan area in that state was applied to the urban route segments (even though some of the urban segments may not be within the largest metropolitan area). For states without a metropolitan area with more than 250,000 people (Delaware, Montana, North Dakota, South Dakota, Vermont and Wyoming), the state-wide increase was used. [6]

[6] For the final version of this report, the routes will be re-run using the 2010 census data. This will eliminate the need for population multipliers.

Table II-5. Population multipliers

State	Rural, Suburban, Urban Designation	Population Multiplier	State	Rural, Suburban, Urban Designation	Population Multiplier
Arkansas	Rural, Suburban	1.051	New Hampshire	Rural, Suburban	1.064
	Urban	1.069		Urban	1.058
Colorado	Rural, Suburban	1.105	New Jersey	Rural, Suburban	1.037
	Urban	1.105		Urban	1.027
Connecticut	Rural, Suburban	1.029	New Mexico	Rural, Suburban	1.075
	Urban	1.020		Urban	1.119
Delaware	Rural, Suburban	1.089	New York	Rural, Suburban	1.017
	Urban	1.089		Urban	1.027
District of Columbia	Rural, Suburban	1.017	Ohio	Rural, Suburban	1.011
	Urban[a]	1.017		Urban	0.984
Idaho	Rural, Suburban	1.133	Oklahoma	Rural, Suburban	1.037
	Urban	1.221		Urban	1.070
Illinois	Rural, Suburban	1.033	Oregon	Rural, Suburban	1.082
	Urban	1.045		Urban	1.109
Indiana	Rural, Suburban	1.038	Pennsylvania	Rural, Suburban	1.013
	Urban	1.092		Urban	1.025
Iowa	Rural, Suburban	1.019	South Dakota	Rural, Suburban	1.036
	Urban	1.110		Urban	1.036
Kansas	Rural, Suburban	1.028	Tennessee	Rural, Suburban	1.061
	Urban	1.037		Urban	1.109
Kentucky	Rural, Suburban	1.041	Texas	Rural, Suburban	1.127
	Urban	1.051		Urban	1.163
Maine	Rural, Suburban	1.037	Utah	Rural, Suburban	1.142
	Urban	1.054		Urban	1.102
Maryland	Rural, Suburban	1.060	Vermont	Rural, Suburban	1.025
	Urban	1.103		Urban	1.025
Massachusetts	Rural, Suburban	1.014	Virginia	Rural, Suburban	1.080
	Urban	1.014		Urban	1.103
Minnesota	Rural, Suburban	1.050	Washington	Rural, Suburban	1.085
	Urban	1.069		Urban	1.072
Missouri	Rural, Suburban	1.044	Wisconsin	Rural, Suburban	1.036
	Urban	1.036		Urban	1.006
Montana	Rural, Suburban	1.047	Wyoming	Rural, Suburban	1.043
	Urban	1.047		Urban	1.043
Nebraska	Rural, Suburban	1.033			
	Urban	1.072			

[a]For the urban areas within the District of Columbia, the growth rate of the District was used rather than the growth rate of metropolitan Washington. The growth rate of metropolitan Washington was used for urban areas within Maryland and Virginia.

Parameters like population density and route segment lengths, that are specific to each route, were developed using WebTRAGIS. Figure II-7 through Figure II-10 are WebTRAGIS maps showing the routes.

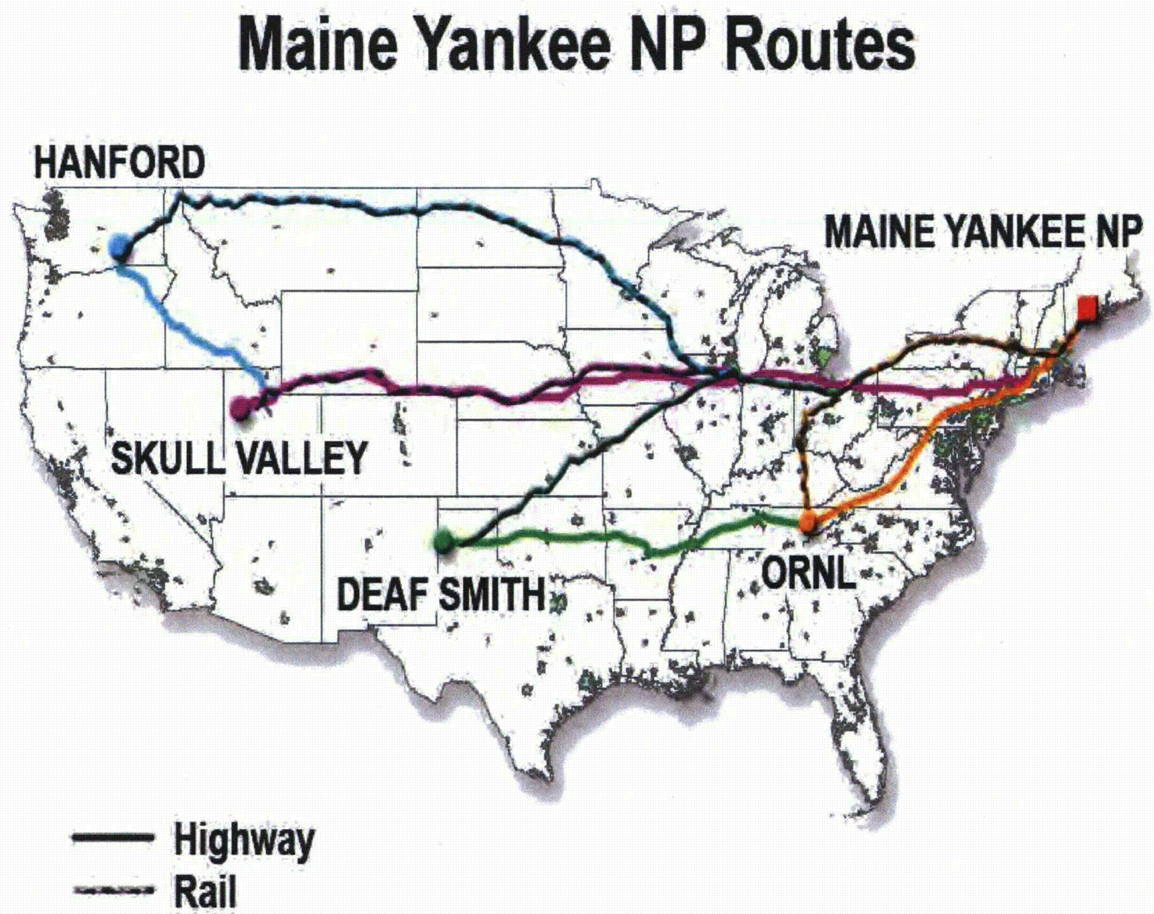

Figure II-7. Highway and rail routes from the Maine Yankee Nuclear Plant (NP) site.

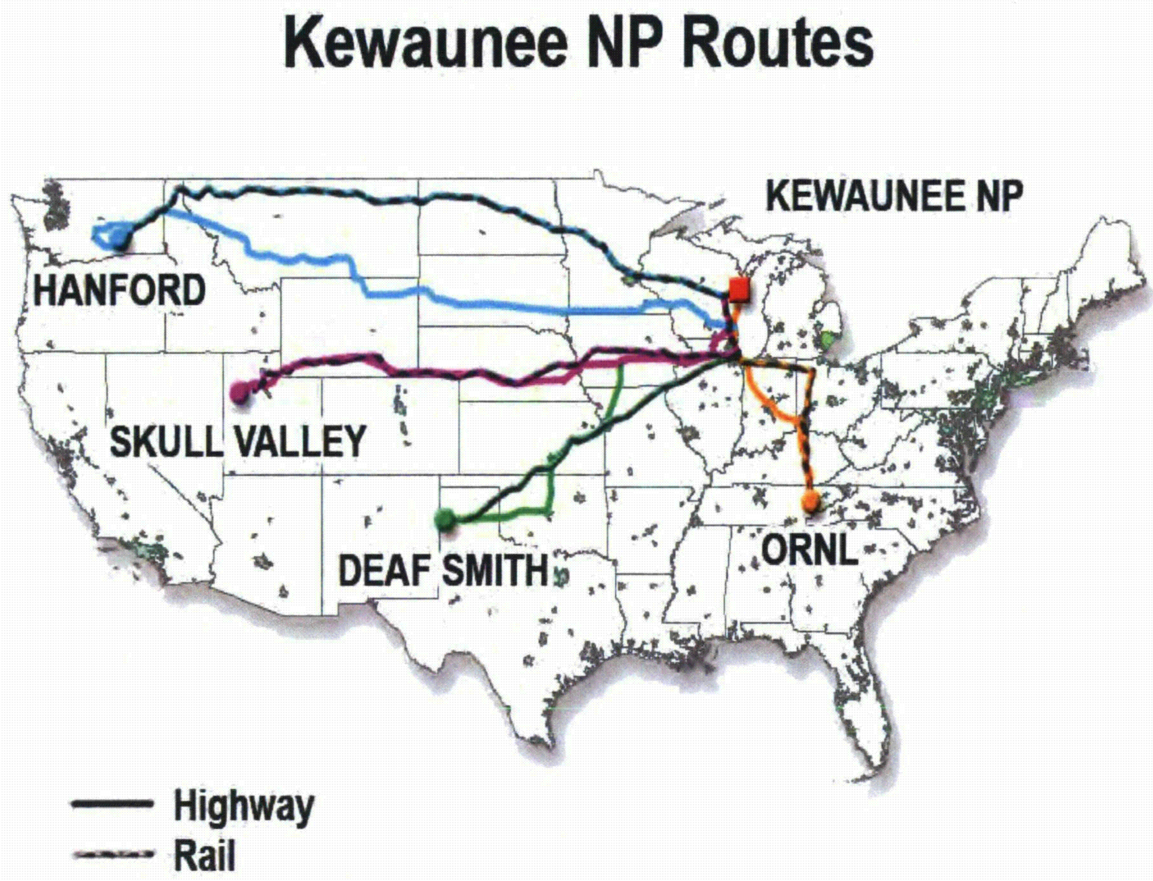

Figure II-8. Highway and rail routes from the Kewaunee Nuclear Plant (NP) site.

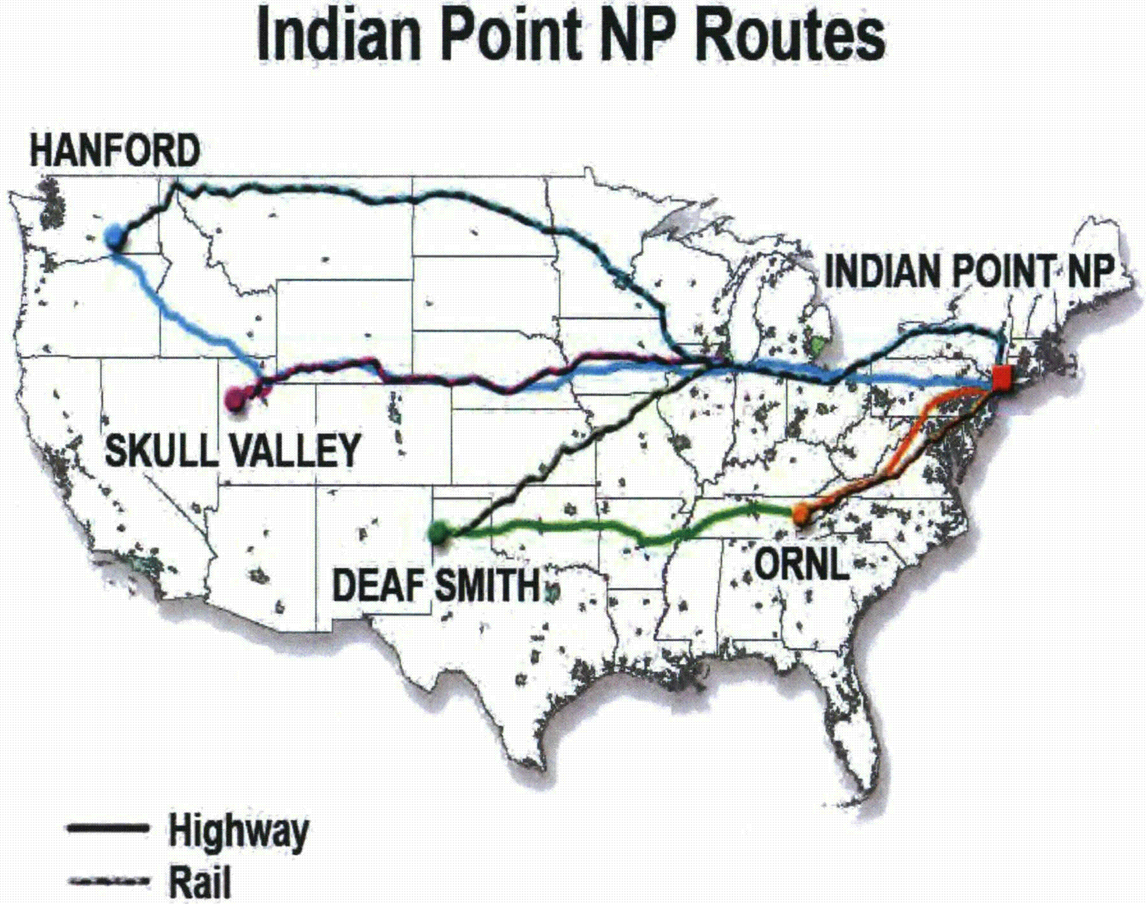

Figure II-9. Highway and rail routes from Indian Point Nuclear Plant (NP).

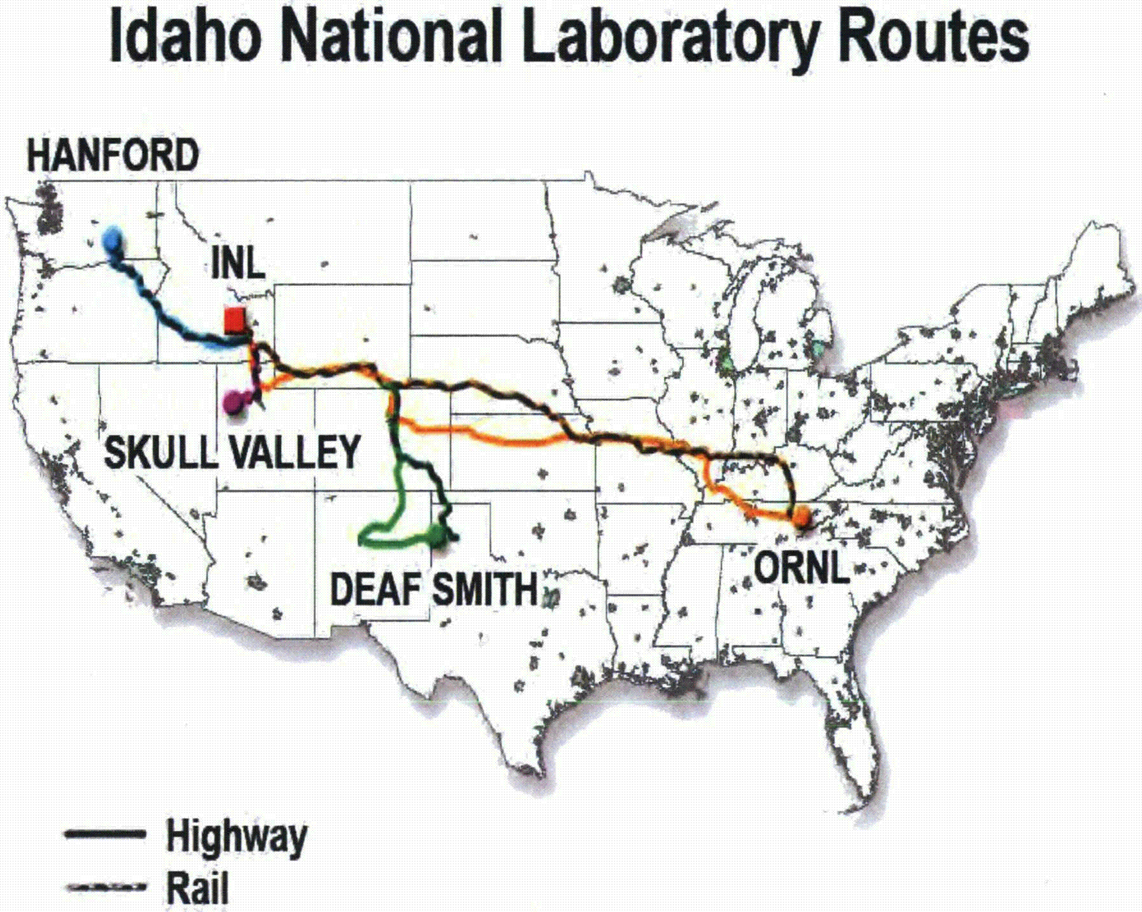

Figure II-10. Highway and rail routes from Idaho National Laboratory (INL).

II.5 Results

II.5.1 Maximally Exposed Resident In-Transit Dose

The largest dose from a moving vehicle to an individual member of the public is sustained when that individual is 30 meters (a conservative estimate of the interstate right-of-way) from the moving vehicle, and the vehicle is moving at the slowest speed it would be likely to maintain. This speed is 24 kph (16 mph) for both rail and truck. Table II-6 shows the maximum individual dose, in Sv, for each package. These doses are directly proportional to the external dose rate (TI) of each package. For comparison, a single dental x-ray delivers a dose of 4×10^{-5} Sv (Stabin, 2009), about 7000 times the doses shown in Table II-6.

Table II-6. Maximum individual doses.

Package (mode)	Dose in Sv
Truck-DU (truck)	6.7×10^{-9}
Rail-Lead (rail)	5.7×10^{-9}
Rail-Steel (rail)	4.3×10^{-9}

Figure II-11 and Figure II-12 show the portion of the RADTRAN output file that reflects these doses.

```
RUN DATE: [ 03-02-2010    AT 18:07 ]              PAGE    11
                    TRUCK URF -- PUBLIC

            MAXIMUM INDIVIDUAL IN-TRANSIT DOSE
                     GA_4          6.70E-07 REM
```

Figure II-11. RADTRAN output for maximum individual truck doses.

```
RUN DATE: [ 02-19-2010    AT 10:55 ]              PAGE    10
                        NAC-STC
            MAXIMUM INDIVIDUAL IN-TRANSIT DOSE


                    NAC-STC      5.67E-07 REM

                    HI-STAR      4.30E-07 REM
```

Figure II-12. RADTRAN output for maximum individual rail doses.

II.5.2 Unit Risk: Rail Routes

The doses to railyard workers along the route, to residents and others along the route, and to occupants of vehicles that share the route from a single shipment (one rail cask) traveling one km past a population density of one person/km^2 are shown in Table II-7. The dose units are person-Sv. The doses are calculated assuming one cask on a train, because railcar-km is the unit usually used to describe freight rail transport. The data in this table may be used to calculate collective doses along routes as follows:

- This is a conservatively calculated dose that assumes that the railyard crew receives a fraction of the classification yard dose when the train stops. The railyard crew dose is multiplied by the length of each type of route traveled. The classification yard occupational collective dose (Wooden, 1986), assuming a 30-hour classification stop, is integrated into RADTRAN. This integrated dose was adjusted to reflect the 27-hour stop (Table II-3) (DOT, 2004b).

- The area of the band occupied by the population along the route is equal to the product of the kilometers traveled and the band width (usually 800 m on each side of the route). RADTRAN calculates the doses to residents along the route by integrating over this area. This "unit dose to a resident along the route" is multiplied by the area of the band and the appropriate population density (obtained from a routing code like WebTRAGIS).

Table II-7. Individual doses ("unit doses" or "unit risks") to various receptors for rail routes. The units of the dose to the residents near a railyard where the train has stopped, Sv- km^2/hour, reflect the output of the RADTRAN stop model, which incorporates the area occupied.

Cask and route type	Resident along route, Sv-km[a]	Resident near railyard Sv-km^2/hour[b]	Occupants of vehicles sharing the route person-Sv/km[c]
Rail-Lead rural	7.3E-10	3.5E-07	1.6E-08
Rail-Lead suburban	6.3E-10	3.5E-07	1.6E-08
Rail-Lead urban	2.2E-11	3.5E-07	4.6E-08
Rail-Steel rural	5.6E-10	2.7E-07	1.2E-08
Rail-Steel suburban	4.8E-10	2.7E-07	1.2E-08
Rail-Steel urban	1.7E-11	2.7E-07	3.5E-08

[a]To obtain the collective dose to residents along the route, multiply this number by the route length and the population density.
[b]To obtain the collective dose to residents near a railyard, multiply this number by the population density and the stop duration.
[c]To obtain the collective dose to occupants of vehicles sharing the route, multiply this number by the route length (the vehicle density and occupants/vehicle used are the national average)

Figure II-13 shows the RADTRAN output for Table II-7 (in rem). The relevant data in the output are in bold print.

```
          IN-TRANSIT POPULATION EXPOSURE IN PERSON-REM

            LINK          CREW        OFF LINK    ON LINK
          NAC_R         4.32E-05     7.29E-08    1.63E-06
          NAC_S         4.32E-05     6.34E-08    1.63E-06
          NAC_U         7.28E-05     2.21E-09    4.63E-06
          HISTAR_R      3.27E-05     5.55E-08    1.24E-06
          HISTAR_S      3.27E-05     4.83E-08    1.24E-06
          HISTAR_U      5.51E-05     1.68E-09    3.50E-06

                      STOP EXPOSURE IN PERSON-REM

                ANNULAR AREA     CLASSIFICA    3.51E-05
                ANNULAR AREA     CLASSIFICA    2.66E-05
```

Figure II-13. RADTRAN output for Table II-7.

II.5.3 Unit risk: truck routes

The doses to truck crew, residents and others along the route, and to occupants of vehicles that share the route from a single shipment (one truck cask) traveling one kilometer past a population density of one person/km^2 are shown in Table II-8. The dose units are person-Sv. Rural, suburban, and urban doses to residents living near stops are calculated by multiplying the appropriate stop dose -truck stops are not typically located in urban areas) by the appropriate population density (obtained from a routing code like WebTRAGIS). The number of stops on each route segment is calculated by dividing the length of the route segment by 845 km (average distance between refueling stops for a large semi-detached trailer truck (DOE, 2002, Appendix J). The area of the band occupied by the population along the route is equal to the kilometers traveled multiplied by, e.g., 1.6 for a band width of 800 m on each side of the route.

Table II-8. Individual dose ("unit risk") to various receptors along truck routes. Unit risks are shown for each EPA region (see Table II-18).

	Resident along route Sv-km	Resident near stops Sv-km^2/hour	Occupants of vehicles sharing route, person-Sv/km									
			EPA Regions									
			1	2	3	4	5	6	7	8	9	10
Truck-DU rural	3.1E-10	3.26E-08	4.8E-08	1.1E-07	2.3E-07	1.6E-07	1.3E-07	9.9E-08	1.0E-07	8.8E-08	1.6E-07	1.2E-07
Truck-DU suburban	2.7E-10	2.84E-08	8.0E-08	2.3E-07	4.0E-07	3.1E-07	2.7E-07	1.7E-07	1.8E-07	2.2E-07	4.1E-07	2.9E-07
Truck-DU urban	5.2E-12		2.4E-07	4.6E-07	6.4E-07	6.2E-07	4.9E-07	3.3E-07	2.7E-07	4.1E-07	8.4E-07	6.3E-07
Truck-DU urban rush hour[d]	1.2E-12		2.2E-07	4.3E-07	5.9E-07	5.8E-07	4.5E-07	3.1E-07	2.5E-07	3.8E-07	7.7E-07	5.8E-07
6.9 people sharing stop (person-Sv)		2.3E-04										

[a]To obtain the collective dose to residents along the route, multiply this number by the route length and the population density.

[b]To obtain the collective dose to residents near a railyard, multiply this number by the population density and the stop duration.

[c]To obtain the collective dose to occupants of vehicles sharing the route, multiply this number by the route length (the vehicle density and occupants/vehicle used are the regional average)

[d]One-tenth of the urban route segment is considered "rush-hour km" – equivalent to the truck spending 10% of the urban transit distance during rush hour. RADTRAN has historically assumed that the vehicle density doubles during rush hour and the vehicle speed is halved. The slower vehicle speed impacts the dose to urban residents along the route, but the vehicle density does not. Both factors influence the dose to occupants of vehicles sharing the route. The unit risk factors in the table incorporate the 9/10 of the distance during non-rush hour and 1/10 of the distance during rush hour, so both factors should be multiplied by the total urban distance.

II.5.4 Doses along selected routes.

Doses to receptors along the routes shown in Table II-4 are presented below.

II.5.4.1 Collective doses to receptors along the route

Using route data from Web TRAGIS, collective doses from incident-free transportation were calculated. For rural and suburban route segments, collective doses calculated were doses sustained by the resident population. Non-resident populations were included with residents as receptors along the urban segments of the routes. Table II-9 to Table II-12 show collective doses along rail routes and Table II-13 to Table II-16, along highway routes. Blank cells in the tables indicate that no route miles for that population type was present along that route (e.g., not all routes transit urban areas in all states).

Table II-9. Collective doses to residents along the route (person-Sv) from rail transportation; shipment origin INL

DEST. AND ROUTE	Rail-Lead				Rail-Steel			
	Rural	Suburban	Urban	Total	Rural	Suburban	Urban	Total
ORNL								
Colorado	1.3E-07	5.8E-07		7.1E-07	1.0E-07	4.4E-07		5.4E-07
Idaho	1.7E-06	7.6E-06	3.4E-07	9.7E-06	1.3E-06	5.8E-06	2.6E-07	7.4E-06
Illinois	1.8E-06	1.7E-05	4.5E-07	1.9E-05	1.3E-06	1.3E-05	3.4E-07	1.4E-05
Indiana	1.7E-06	8.2E-06	1.8E-07	1.0E-05	1.3E-06	6.3E-06	1.4E-07	7.7E-06
Kansas	1.3E-06	6.6E-06	1.6E-07	8.1E-06	9.7E-07	5.0E-06	1.2E-07	6.1E-06
Kentucky	2.6E-06	2.1E-05	8.6E-07	2.5E-05	2.0E-06	1.6E-05	6.6E-07	1.9E-05
Missouri	2.4E-06	2.2E-05	1.1E-06	2.6E-05	1.8E-06	1.7E-05	8.7E-07	2.0E-05
Nebraska	3.5E-06	1.2E-05	3.5E-07	1.6E-05	2.7E-06	9.5E-06	2.6E-07	1.2E-05
Tennessee	1.2E-06	7.8E-06	4.2E-08	9.1E-06	9.4E-07	6.0E-06	3.2E-08	6.9E-06
Wyoming	1.4E-06	8.8E-06	2.1E-07	1.0E-05	1.1E-06	6.7E-06	1.6E-07	7.9E-06
DEAF SMITH								
Colorado	3.3E-06	4.1E-05	1.7E-06	4.6E-05	2.5E-06	3.2E-05	1.3E-06	3.5E-05
Idaho	1.7E-06	7.6E-06	3.4E-07	9.7E-06	1.3E-06	5.8E-06	2.6E-07	7.4E-06
Oklahoma	1.1E-07	1.8E-07		2.9E-07	8.3E-08	1.4E-07		2.2E-07
Texas	4.1E-07	3.4E-06	5.9E-08	3.8E-06	3.1E-07	2.6E-06	4.4E-08	2.9E-06
Wyoming	1.1E-06	6.0E-06	1.5E-07	7.3E-06	8.5E-07	4.6E-06	1.2E-07	5.6E-06
HANFORD								
Idaho	3.7E-06	1.6E-05	**6.0E-07**	2.0E-05	2.8E-06	1.2E-05	4.6E-07	1.5E-05
Oregon	1.4E-06	9.2E-06	2.2E-07	1.1E-05	1.1E-06	7.0E-06	1.7E-07	8.3E-06
Washington	1.2E-07	4.4E-06	2.6E-07	4.7E-06	8.9E-08	3.3E-06	2.0E-07	3.6E-06
SKULL VALLEY								
Idaho	1.4E-06	6.6E-06	3.3E-07	8.3E-06	1.1E-06	5.0E-06	2.5E-07	6.3E-06
Utah	1.6E-06	1.9E-05	1.1E-06	2.2E-05	1.2E-06	1.4E-05	8.5E-07	1.6E-05

Sample calculation: Urban route from INL to Hanford through Idaho, Rail-Lead cask

RADTRAN output (unit risk): 2.21E-09 person-rem (from Figure II-13)
Population density: 2281 persons/km^2
Population multiplier: 1.133
Route segment length: 10.5 km
Population (collective) dose = 2.21E-09*2281*1.133*10.5 = 6.00E-05 person-rem
Convert to SI units: 6.00E-05 person-rem*0.01person-Sv/person-rem = 6.00E-07 person-Sv
Blank cell indicates no route miles of this population density.

Table II-10. Collective doses to residents along the route (person-Sv), rail transportation, shipment origin Indian Point

DEST. AND ROUTES	Rail-Lead				Rail-Steel			
	Rural	Suburban	Urban	Total	Rural	Suburban	Urban	Total
ORNL								
Delaware	1.2E-08	7.3E-06	8.2E-07	8.2E-06	9.5E-09	5.6E-06	6.3E-07	6.2E-06
DC	3.2E-09	8.9E-07	4.5E-07	1.3E-06	2.4E-09	6.8E-07	3.5E-07	1.0E-06
Maryland	6.9E-07	2.2E-05	2.0E-06	2.5E-05	5.3E-07	1.7E-05	1.5E-06	1.9E-05
New Jersey	4.0E-07	1.2E-05	1.4E-06	1.4E-05	3.1E-07	9.1E-06	1.1E-06	1.0E-05
New York	3.0E-08	1.6E-06	3.4E-06	5.0E-06	2.3E-08	1.3E-06	2.5E-06	3.8E-06
Pennsylvania	4.9E-08	8.6E-06	3.2E-06	1.2E-05	3.8E-08	6.5E-06	2.4E-06	9.0E-06
Tennessee	2.2E-06	3.0E-05	6.6E-07	3.3E-05	1.7E-06	2.3E-05	5.0E-07	2.5E-05
Virginia	4.1E-06	5.9E-05	2.3E-06	6.5E-05	3.1E-06	4.5E-05	1.7E-06	5.0E-05
DEAF SMITH								
Illinois	1.5E-06	2.7E-05	2.4E-06	3.1E-05	1.1E-06	2.0E-05	1.8E-06	2.3E-05
Indiana	2.1E-06	1.1E-05	**5.4E-07**	1.4E-05	1.6E-06	8.6E-06	4.1E-07	1.1E-05
Iowa	3.0E-07	6.2E-07	3.1E-08	9.5E-07	2.3E-07	4.8E-07	2.4E-08	7.2E-07
Kansas	2.0E-06	1.8E-05	7.9E-07	2.1E-05	1.5E-06	1.4E-05	6.0E-07	1.6E-05
Missouri	1.2E-06	7.0E-06	2.4E-07	8.5E-06	9.2E-07	5.4E-06	1.8E-07	6.5E-06
New York	5.5E-06	6.1E-05	4.9E-06	7.1E-05	4.2E-06	4.6E-05	3.7E-06	5.4E-05
Ohio	2.5E-06	3.2E-05	2.3E-06	3.6E-05	1.9E-06	2.4E-05	1.7E-06	2.8E-05
Oklahoma	4.5E-07	4.0E-06	5.2E-08	4.5E-06	3.4E-07	3.1E-06	3.9E-08	3.4E-06
Pennsylvania	4.1E-07	9.3E-06	4.9E-07	1.0E-05	3.1E-07	7.1E-06	3.7E-07	7.8E-06
Texas	7.3E-07	5.1E-06	1.2E-07	6.0E-06	5.6E-07	3.9E-06	9.4E-08	4.5E-06
HANFORD								
Idaho	9.8E-07	6.7E-06	9.6E-08	7.8E-06	7.5E-07	5.1E-06	7.3E-08	5.9E-06
Illinois	1.4E-06	2.1E-05	2.2E-06	2.4E-05	1.0E-06	1.6E-05	1.7E-06	1.9E-05
Indiana	2.1E-06	1.1E-05	5.4E-07	1.4E-05	1.6E-06	8.6E-06	4.1E-07	1.1E-05
Minnesota	3.2E-06	2.9E-05	1.2E-06	3.4E-05	2.4E-06	2.2E-05	8.9E-07	2.6E-05
Montana	2.2E-06	1.3E-05	1.4E-07	1.6E-05	1.7E-06	1.0E-05	1.1E-07	1.2E-05
New York	5.5E-06	6.1E-05	4.9E-06	7.1E-05	4.2E-06	4.6E-05	3.7E-06	5.4E-05
North Dakota	1.0E-06	8.2E-06	2.6E-07	9.5E-06	7.7E-07	6.3E-06	2.0E-07	7.2E-06
Ohio	2.5E-06	3.2E-05	2.3E-06	3.6E-05	1.9E-06	2.4E-05	1.7E-06	2.8E-05
Pennsylvania	4.1E-07	9.3E-06	4.9E-07	1.0E-05	3.1E-07	7.1E-06	3.7E-07	7.8E-06
Washington	1.1E-06	1.3E-05	6.5E-07	1.5E-05	8.5E-07	1.0E-05	5.0E-07	1.2E-05
Wisconsin	1.7E-06	8.2E-06	3.7E-07	1.0E-05	1.3E-06	6.2E-06	2.8E-07	7.8E-06
SKULL VALLEY								
Colorado	1.3E-07	5.8E-07		7.1E-07	1.0E-07	4.4E-07		5.4E-07
Illinois	1.3E-06	2.1E-05	2.7E-06	2.5E-05	9.9E-07	1.6E-05	2.1E-06	1.9E-05
Indiana	2.1E-06	1.1E-05	5.4E-07	1.4E-05	1.6E-06	8.6E-06	4.1E-07	1.1E-05
Iowa	4.0E-06	1.8E-05	3.4E-07	2.2E-05	3.1E-06	1.4E-05	2.6E-07	1.7E-05
Nebraska	4.2E-06	2.0E-05	6.2E-07	2.5E-05	3.2E-06	1.5E-05	4.7E-07	1.9E-05
New York	5.5E-06	6.1E-05	4.9E-06	7.1E-05	4.2E-06	4.6E-05	3.7E-06	5.4E-05
Ohio	2.5E-06	3.2E-05	2.3E-06	3.6E-05	1.9E-06	2.4E-05	1.7E-06	2.8E-05
Pennsylvania	4.1E-07	9.3E-06	4.9E-07	1.0E-05	3.1E-07	7.1E-06	3.7E-07	7.8E-06
Utah	1.3E-06	1.8E-05	1.1E-06	2.0E-05	9.7E-07	1.4E-05	8.6E-07	1.6E-05
Wyoming	1.4E-06	9.5E-06	2.3E-07	1.1E-05	1.0E-06	7.2E-06	1.8E-07	8.4E-06

Sample calculation: Urban route from Indian Point to Deaf Smith through Indiana, Rail-Lead cask

RADTRAN output (unit risk): 2.21E-09 person-rem (from Figure II-13)
Population density: 2305.9 persons/km^2
Population multiplier: 1.0
Route segment length: 10.6 km
Population (collective) dose = 2.21E-09*2305.9*1.0 *10.6 = 5.40 E-05 person-rem
Convert to SI units: 5.40E-05 person-rem*0.01person-Sv/person-rem = 5.40E-07 person-Sv

Table II-11. Collective doses to residents along the route (person-Sv) rail transportation; shipment origin Kewaunee

DEST. AND ROUTES	Rail-Lead				Rail-Steel			
	Rural	Suburban	Urban	Total	Rural	Suburban	Urban	Total
ORNL								
Illinois	2.4E-07	2.1E-05	2.5E-06	2.3E-05	1.8E-07	1.6E-05	1.9E-06	1.8E-05
Indiana	2.1E-06	1.1E-05	5.4E-07	1.4E-05	1.6E-06	8.6E-06	4.1E-07	1.1E-05
Kentucky	3.2E-06	1.6E-05	7.0E-07	2.0E-05	2.4E-06	1.3E-05	5.3E-07	1.5E-05
Ohio	2.2E-06	3.0E-05	1.4E-06	3.3E-05	**1.6E-06**	2.3E-05	1.1E-06	2.5E-05
Tennessee	7.4E-07	5.0E-06	4.1E-08	5.7E-06	5.6E-07	3.8E-06	3.1E-08	4.4E-06
Wisconsin	1.9E-06	2.5E-05	1.5E-06	2.9E-05	1.5E-06	1.9E-05	1.1E-06	2.2E-05
DEAF SMITH								
Illinois	1.6E-06	3.5E-05	3.0E-06	4.0E-05	1.2E-06	2.7E-05	2.3E-06	3.0E-05
Iowa	3.0E-07	6.2E-07	3.1E-08	9.5E-07	2.3E-07	4.8E-07	2.4E-08	7.2E-07
Kansas	2.0E-06	1.8E-05	7.9E-07	2.1E-05	1.5E-06	1.4E-05	6.0E-07	1.6E-05
Missouri	1.2E-06	7.0E-06	2.8E-07	8.5E-06	9.2E-07	5.4E-06	2.2E-07	6.5E-06
Oklahoma	4.5E-07	4.0E-06	5.2E-08	4.5E-06	3.4E-07	3.1E-06	3.9E-08	3.4E-06
Texas	7.3E-07	5.1E-06	1.2E-07	6.0E-06	5.6E-07	3.9E-06	9.4E-08	4.6E-06
Wisconsin	1.9E-06	2.5E-05	1.5E-06	2.9E-05	1.5E-06	1.9E-05	1.1E-06	2.2E-05
HANFORD								
Idaho	9.8E-07	6.7E-06	9.6E-08	7.8E-06	7.5E-07	5.1E-06	7.3E-08	5.9E-06
Minnesota	3.3E-06	3.0E-05	9.2E-07	3.5E-05	2.5E-06	2.3E-05	7.0E-07	2.6E-05
Montana	2.2E-06	1.3E-05	1.4E-07	1.3E-05	1.7E-06	1.0E-05	1.1E-07	1.0E-05
North Dakota	1.0E-06	8.2E-06	2.6E-07	9.5E-06	7.7E-07	6.3E-06	2.0E-07	7.2E-06
Washington	1.1E-06	1.3E-05	6.5E-07	1.5E-05	8.5E-07	1.0E-05	5.0E-07	1.2E-05
Wisconsin	3.5E-06	2.2E-05	8.9E-07	2.6E-05	2.7E-06	1.7E-05	6.8E-07	2.0E-05
SKULL VALLEY								
Colorado	1.3E-07	5.8E-07	0.0E+00	7.1E-07	1.0E-07	4.4E-07	0.0E+00	5.4E-07
Illinois	1.4E-06	2.7E-05	2.8E-06	3.1E-05	1.1E-06	2.1E-05	2.1E-06	2.4E-05
Iowa	4.0E-06	1.8E-05	3.4E-07	2.2E-05	3.1E-06	1.4E-05	2.6E-07	1.7E-05
Nebraska	4.2E-06	2.0E-05	6.2E-07	2.5E-05	3.2E-06	1.5E-05	4.7E-07	1.9E-05
Utah	1.3E-06	1.8E-05	1.1E-06	2.0E-05	9.7E-07	1.4E-05	8.6E-07	1.6E-05
Wisconsin	1.9E-06	2.5E-05	1.5E-06	2.9E-05	1.5E-06	1.9E-05	1.1E-06	2.2E-05
Wyoming	1.4E-06	9.5E-06	2.3E-07	1.1E-05	1.0E-06	7.2E-06	1.8E-07	8.4E-06

Sample calculation: Rural route from Kewaunee to ORNL through Ohio, Rail-steel cask

RADTRAN output (unit risk): 5.55E-08 person-rem (from Figure II-13)
Population density: 14.8 persons/km^2
Population multiplier: 1.0
Route segment length: 200.6 km
Population (collective) dose = 5.55E-08*14.8*1.0*200.6 = 1.65E-04 person-rem
Convert to SI units: 1.65E-04 person-rem*0.01person-Sv/person-rem = 1.65E-06 person-Sv

Table II-12. Collective doses to residents along the route (person-Sv) rail shipment origin Maine Yankee

DEST. AND ROUTES	Rail-Lead				Rail-Steel			
	Rural	Suburban	Urban	Total	Rural	Suburban	Urban	Total
ORNL								
Kentucky	3.2E-06	1.6E-05	7.0E-07	2.0E-05	2.4E-06	1.3E-05	5.3E-07	1.5E-05
Maine	9.3E-07	1.6E-05	6.2E-07	1.7E-05	7.1E-07	1.2E-05	4.7E-07	1.3E-05
Massachusetts	1.3E-06	2.9E-05	1.8E-06	3.2E-05	1.0E-06	2.2E-05	1.4E-06	2.4E-05
New Hampshire	3.8E-07	7.5E-06	2.5E-07	8.2E-06	2.9E-07	5.7E-06	1.9E-07	6.2E-06
New York	4.8E-06	5.2E-05	1.8E-06	5.9E-05	3.7E-06	4.0E-05	1.4E-06	4.5E-05
Ohio	3.6E-06	4.9E-05	3.2E-06	5.6E-05	2.7E-06	3.7E-05	2.5E-06	4.3E-05
Pennsylvania	4.2E-07	9.4E-06	5.1E-07	1.0E-05	3.2E-07	7.2E-06	3.9E-07	7.9E-06
Tennessee	7.4E-07	5.0E-06	4.1E-08	5.7E-06	5.6E-07	3.8E-06	3.1E-08	4.4E-06
Vermont	6.7E-08	5.2E-07		5.9E-07	5.1E-08	4.0E-07		4.5E-07
DEAF SMITH								
Illinois	1.5E-06	2.7E-05	2.4E-06	3.1E-05	1.1E-06	2.0E-05	1.8E-06	2.3E-05
Indiana	2.1E-06	1.1E-05	5.4E-07	1.4E-05	1.6E-06	8.6E-06	4.1E-07	1.1E-05
Iowa	3.0E-07	6.2E-07	3.1E-08	9.5E-07	2.3E-07	4.8E-07	2.4E-08	7.2E-07
Kansas	2.0E-06	1.8E-05	7.9E-07	2.1E-05	1.5E-06	1.4E-05	6.0E-07	1.6E-05
Maine	9.3E-07	1.6E-05	6.2E-07	1.7E-05	7.1E-07	1.2E-05	4.7E-07	1.3E-05
Massachusetts	1.3E-06	2.9E-05	1.8E-06	3.2E-05	1.0E-06	2.2E-05	1.4E-06	2.4E-05
Missouri	1.2E-06	7.0E-06	2.4E-07	8.5E-06	9.2E-07	5.4E-06	1.8E-07	6.5E-06
New Hampshire	3.8E-07	7.5E-06	2.5E-07	8.2E-06	2.9E-07	5.7E-06	1.9E-07	6.2E-06
New York	4.8E-06	5.2E-05	1.8E-06	5.9E-05	3.7E-06	4.0E-05	1.4E-06	4.5E-05
Ohio	2.5E-06	3.2E-05	2.3E-06	3.6E-05	1.9E-06	2.4E-05	1.7E-06	2.8E-05
Oklahoma	4.3E-07	3.9E-06	4.9E-08	4.5E-06	3.3E-07	3.0E-06	3.7E-08	3.4E-06
Pennsylvania	4.1E-07	9.3E-06	4.9E-07	1.0E-05	3.1E-07	7.1E-06	3.7E-07	7.8E-06
Texas	7.3E-07	5.1E-06	1.2E-07	6.0E-06	5.6E-07	3.9E-06	9.4E-08	4.5E-06
Vermont	6.7E-08	5.2E-07		5.9E-07	5.1E-08	4.0E-07		4.5E-07
HANFORD								
Idaho	9.8E-07	6.7E-06	9.6E-08	7.8E-06	7.5E-07	5.1E-06	7.3E-08	5.9E-06
Illinois	1.4E-06	2.1E-05	2.2E-06	2.4E-05	1.0E-06	1.6E-05	1.7E-06	1.9E-05
Indiana	2.1E-06	1.1E-05	5.4E-07	1.4E-05	1.6E-06	8.6E-06	4.1E-07	1.1E-05
Maine	9.3E-07	1.6E-05	6.2E-07	1.7E-05	7.1E-07	1.2E-05	4.7E-07	1.3E-05
Massachusetts	1.3E-06	2.9E-05	1.8E-06	3.2E-05	1.0E-06	2.2E-05	1.4E-06	2.4E-05
Minnesota	3.2E-06	2.9E-05	1.2E-06	3.4E-05	2.4E-06	2.2E-05	8.9E-07	2.6E-05
Montana	2.2E-06	1.3E-05	1.4E-07	1.6E-05	1.7E-06	1.0E-05	1.1E-07	1.2E-05
New Hampshire	3.8E-07	7.5E-06	2.5E-07	8.2E-06	2.9E-07	5.7E-06	1.9E-07	6.2E-06
New York	4.8E-06	5.2E-05	2.2E-06	5.9E-05	3.7E-06	4.0E-05	1.7E-06	4.5E-05
North Dakota	1.0E-06	8.2E-06	2.6E-07	9.5E-06	7.7E-07	6.3E-06	2.0E-07	7.2E-06
Ohio	2.5E-06	3.2E-05	2.3E-06	3.6E-05	1.9E-06	2.4E-05	1.7E-06	2.8E-05

Table II-12. Collective doses to residents along the route (person-Sv) rail shipment origin Maine Yankee -- continued

DEST. AND ROUTES	Rail-Lead				Rail-Steel			
	Rural	Suburban	Urban	Total	Rural	Suburban	Urban	Total
HANFORD (cont.)								
Pennsylvania	4.1E-07	9.3E-06	4.9E-07	1.0E-05	3.1E-07	7.1E-06	3.7E-07	7.8E-06
Vermont	6.7E-08	5.2E-07		5.9E-07	5.1E-08	4.0E-07		4.5E-07
Washington	1.1E-06	1.3E-05	6.5E-07	1.5E-05	8.5E-07	1.0E-05	5.0E-07	1.2E-05
Wisconsin	1.7E-06	8.2E-06	3.7E-07	1.0E-05	1.3E-06	6.2E-06	2.8E-07	7.8E-06
SKULL VALLEY								
Colorado	1.3E-07	5.8E-07		7.1E-07	1.0E-07	4.4E-07		5.4E-07
Illinois	1.3E-06	2.2E-05	2.7E-06	2.6E-05	1.0E-06	1.7E-05	2.0E-06	2.0E-05
Indiana	2.1E-06	1.1E-05	5.4E-07	1.4E-05	1.6E-06	8.6E-06	4.1E-07	1.1E-05
Iowa	4.0E-06	1.8E-05	3.4E-07	2.2E-05	3.1E-06	1.4E-05	2.6E-07	1.7E-05
Maine	9.6E-07	1.6E-05	1.6E-07	1.7E-05	7.3E-07	1.2E-05	1.2E-07	1.3E-05
Massachusetts	6.5E-07	2.8E-05	8.5E-07	3.0E-05	4.9E-07	2.1E-05	6.5E-07	2.3E-05
Nebraska	4.2E-06	2.0E-05	6.2E-07	2.5E-05	**3.2E-06**	1.5E-05	4.7E-07	1.9E-05
New Hampshire	1.1E-07	3.7E-06	4.9E-08	3.8E-06	8.6E-08	2.8E-06	3.7E-08	2.9E-06
New York	4.8E-06	5.2E-05	1.8E-06	5.9E-05	3.7E-06	4.0E-05	1.4E-06	4.5E-05
Ohio	2.5E-06	3.2E-05	2.3E-06	3.6E-05	1.9E-06	2.4E-05	1.7E-06	2.8E-05
Pennsylvania	4.1E-07	9.3E-06	4.9E-07	1.0E-05	3.1E-07	7.1E-06	3.7E-07	7.8E-06
Utah	1.3E-06	1.8E-05	1.1E-06	2.0E-05	9.7E-07	1.4E-05	8.6E-07	1.6E-05
Vermont	6.7E-08	5.2E-07	0.0E+00	5.9E-07	5.1E-08	4.0E-07	0.0E+00	4.5E-07
Wyoming	1.4E-06	9.5E-06	2.3E-07	1.1E-05	1.0E-06	7.2E-06	1.8E-07	8.4E-06

Sample calculation: Rural route from Maine Yankee to Skull Valley through Nebraska, Rail-steel cask

RADTRAN output (unit risk): 5.55E-08 person-rem (from Figure II-13)
Population density: 9.3 persons/km^2
Population multiplier: 1.0
Route segment length: 621.7 km
Population (collective) dose = 5.55E-08*9.3*1.0*621.7 = 3.21E-04 person-rem
Convert to SI units: 3.17E-04 person-rem*0.01person-Sv/person-rem = 3.21E-06 person-Sv

Table II-13. Collective doses to residents along the route (person-Sv) from truck transportation (Truck-DU); shipment origin Maine Yankee

DESTINATION	ROUTES	Rural	Suburban	Urban	Urban Rush Hour	TOTAL
ORNL	Connecticut	2.8E-07	1.5E-05	5.6E-07	1.2E-07	1.58E-05
	Maine	4.0E-07	7.3E-06	6.6E-08	1.5E-08	7.74E-06
	Maryland	3.4E-08	1.3E-06	1.3E-08	3.0E-09	1.33E-06
	Massachusetts	2.7E-07	1.2E-05	2.0E-07	4.5E-08	1.23E-05
	New Hampshire	4.7E-08	1.5E-06	7.4E-09	1.6E-09	1.59E-06
	New Jersey	1.8E-07	6.4E-06	3.1E-07	6.9E-08	6.92E-06
	New York	**2.1E-09**	1.7E-06	5.1E-07	1.1E-07	2.28E-06
	Pennsylvania	1.1E-06	1.3E-05	1.6E-07	3.6E-08	1.42E-05
	Tennessee	7.6E-07	9.4E-06	7.9E-08	1.8E-08	1.02E-05
	Virginia	1.8E-06	1.9E-05	1.3E-07	2.8E-08	2.13E-05
	West Virginia	1.1E-07	2.6E-06	7.2E-09	1.6E-09	2.73E-06
DEAF SMITH	Connecticut	1.5E-06	9.9E-06	9.5E-08	2.1E-08	1.15E-05
	Maine	2.9E-07	1.5E-05	5.7E-07	1.3E-07	1.60E-05
	Maryland	4.0E-07	6.8E-06	3.7E-08	8.2E-09	7.28E-06
	Massachusetts	3.4E-08	1.3E-06	1.3E-08	3.0E-09	1.34E-06
	New Hampshire	2.7E-07	1.2E-05	2.0E-07	4.5E-08	1.23E-05
	New Jersey	4.7E-08	1.5E-06	7.4E-09	1.6E-09	1.59E-06
	New York	2.4E-07	8.6E-06	1.7E-07	3.9E-08	9.03E-06
	Oklahoma	2.4E-08	4.3E-06	2.0E-07	4.5E-08	4.53E-06
	Pennsylvania	1.6E-06	8.4E-06	1.1E-07	2.4E-08	1.02E-05
	Tennessee	9.4E-07	1.1E-05	1.3E-07	3.0E-08	1.19E-05
	Texas	2.9E-06	2.5E-05	3.9E-07	8.7E-08	2.83E-05
	Virginia	3.9E-07	2.2E-06	9.5E-08	2.1E-08	2.76E-06
	West Virginia	1.7E-06	1.8E-05	1.2E-07	2.7E-08	2.03E-05

Table II-13. Collective doses to residents along the route (person-Sv) from truck transportation (Truck-DU); shipment origin Maine Yankee -- continued

DESTINATION	ROUTES	Rural	Suburban	Urban	Urban Rush Hour	Total
HANFORD	Connecticut	2.7E-07	1.1E-05	3.0E-07	6.6E-09	1.18E-05
	Idaho	1.4E-06	6.6E-06	9.6E-08	2.1E-09	8.15E-06
	Illinois	8.1E-07	6.5E-06	1.2E-07	2.7E-09	7.42E-06
	Indiana	8.3E-07	7.1E-06	1.2E-07	2.6E-09	8.10E-06
	Iowa	1.9E-06	6.8E-06	5.8E-08	1.3E-09	8.73E-06
	Maine	4.0E-07	6.8E-06	3.7E-08	8.2E-10	7.28E-06
	Massachusetts	2.8E-07	1.2E-05	2.1E-07	4.6E-09	1.28E-05
	Nebraska	2.0E-06	5.4E-06	8.8E-08	2.0E-09	7.48E-06
	New Hampshire	4.7E-08	1.5E-06	7.4E-09	1.6E-10	1.59E-06
	New York	2.8E-07	5.7E-06	5.0E-08	1.1E-09	6.05E-06
	Ohio	1.3E-06	1.2E-05	1.7E-07	3.8E-09	1.39E-05
	Oregon	8.1E-07	2.9E-06	2.6E-08	5.8E-10	3.74E-06
	Pennsylvania	2.0E-06	1.1E-05	8.2E-08	1.8E-09	1.33E-05
	Utah	6.3E-07	4.0E-06	1.8E-08	4.1E-10	4.68E-06
	Washington	8.9E-08	8.4E-07	5.0E-08	1.1E-09	9.92E-07
	Wyoming	9.1E-07	3.6E-06	3.5E-08	7.8E-10	4.54E-06
SKULL VALLEY	Connecticut	2.7E-07	1.1E-05	3.0E-07	6.6E-09	1.18E-05
	Illinois	8.1E-07	6.5E-06	1.2E-07	2.7E-09	7.42E-06
	Indiana	8.3E-07	7.1E-06	1.2E-07	2.6E-09	8.10E-06
	Iowa	1.9E-06	6.8E-06	5.8E-08	1.3E-09	8.73E-06
	Maine	4.0E-07	6.8E-06	3.7E-08	8.2E-10	7.28E-06
	Massachusetts	2.7E-07	1.2E-05	2.0E-07	4.5E-09	1.23E-05
	Nebraska	2.0E-06	5.4E-06	8.8E-08	2.0E-09	7.48E-06
	New Hampshire	4.7E-08	1.5E-06	7.4E-09	1.6E-10	1.59E-06
	New York	2.8E-07	5.7E-06	5.0E-08	1.1E-09	6.05E-06
	Ohio	1.3E-06	1.2E-05	1.7E-07	3.8E-09	1.39E-05
	Pennsylvania	2.0E-06	1.1E-05	8.2E-08	1.8E-09	1.33E-05
	Utah	5.2E-07	4.6E-06	2.0E-07	4.5E-09	5.40E-06
	Wyoming	9.1E-07	3.6E-06	3.5E-08	7.8E-10	4.54E-06

Sample calculation: Rural route from Maine Yankee to ORNL through New York, truck cask

RADTRAN output (unit risk): 3.05E-08 person-rem (from Table II-8)
Population density: 4.4 persons/km^2
Population multiplier: 1.0
Route segment length: 1.6 km
Population (collective) dose = 3.05E-08*4.4*1.0*1.6 = 2.15E-07 person-rem
Convert to SI units: 2.15E-07 person-rem*0.01person-Sv/person-rem = 2.15E-09 person-Sv

Table II-14 Collective doses to residents along the route (person-Sv) from truck transportation (Truck-DU); shipment origin Indian Point.

DESTINATION	ROUTES	Rural	Suburban	Urban	U. Rush Hour	Total
ORNL	Maryland	5.4E-08	2.1E-06	3.0E-09	3.0E-09	1.3E-06
	New Jersey	3.9E-07	1.4E-05	3.9E-08	3.9E-08	9.0E-06
	New York	7.5E-08	7.0E-06	4.9E-08	4.9E-08	4.7E-06
	Pennsylvania	9.4E-07	1.1E-05	3.0E-08	3.0E-08	1.2E-05
	Tennessee	7.9E-07	9.7E-06	1.5E-08	1.5E-08	1.1E-05
	Virginia	1.7E-06	1.8E-05	2.7E-08	2.7E-08	2.0E-05
	West Virginia	1.1E-07	2.6E-06	1.6E-09	1.6E-09	2.7E-06
DEAF SMITH	Arkansas	2.3E-06	1.6E-05	2.2E-08	2.2E-08	1.1E-05
	Maryland	5.4E-08	2.1E-06	3.0E-09	3.0E-09	1.3E-06
	New Jersey	3.9E-07	1.4E-05	3.9E-08	3.9E-08	9.0E-06
	New York	4.7E-08	4.3E-06	4.9E-08	4.9E-08	4.7E-06
	Oklahoma	1.7E-06	8.7E-06	2.6E-08	2.6E-08	1.0E-05
	Pennsylvania	9.4E-07	1.1E-05	3.0E-08	3.0E-08	1.2E-05
	Tennessee	2.9E-06	2.5E-05	3.9E-07	8.7E-08	2.8E-05
	Texas	2.9E-06	2.5E-05	8.7E-08	2.1E-08	2.8E-06
	Virginia	3.9E-07	2.2E-06	2.1E-08	2.7E-08	2.0E-05
	West Virginia	1.7E-06	1.8E-05	2.7E-08	1.6E-09	2.7E-06
HANFORD	Idaho	**1.4E-06**	6.6E-06	1.6E-09	2.1E-08	8.1E-06
	Illinois	7.8E-07	6.3E-06	2.1E-08	2.7E-08	7.4E-06
	Indiana	8.6E-07	7.1E-06	2.7E-08	2.6E-08	8.1E-06
	Iowa	1.9E-06	6.8E-06	2.6E-08	1.3E-08	8.7E-06
	Nebraska	2.0E-06	5.4E-06	1.3E-08	2.0E-08	7.5E-06
	New Jersey	2.6E-07	6.7E-06	2.0E-08	3.2E-08	7.1E-06
	New York	4.7E-08	4.3E-06	3.2E-08	4.9E-08	4.7E-06
	Ohio	1.3E-06	1.2E-05	4.9E-08	3.8E-08	1.4E-05
	Oregon	8.9E-07	3.2E-06	3.8E-08	5.8E-09	4.1E-06
	Pennsylvania	1.8E-06	8.7E-06	5.8E-09	8.1E-09	1.1E-05
	Utah	6.3E-07	4.0E-06	8.1E-09	4.1E-09	4.7E-06
	Washington	8.9E-08	8.4E-07	4.1E-09	1.1E-08	9.9E-07
	Wyoming	9.1E-07	3.6E-06	1.1E-08	7.8E-09	4.5E-06
SKULL VALLEY	Illinois	8.1E-07	6.5E-06	7.8E-09	2.7E-08	7.4E-06
	Indiana	8.3E-07	7.1E-06	2.7E-08	2.6E-08	8.1E-06
	Iowa	1.9E-06	6.8E-06	2.6E-08	1.3E-08	8.7E-06
	Nebraska	2.0E-06	5.4E-06	1.3E-08	2.0E-08	7.5E-06
	New Jersey	2.6E-07	6.7E-06	2.0E-08	3.2E-08	7.1E-06
	New York	4.7E-08	4.3E-06	3.2E-08	4.9E-08	4.7E-06
	Ohio	1.3E-06	1.2E-05	4.9E-08	3.8E-08	1.4E-05
	Pennsylvania	1.8E-06	8.7E-06	3.8E-08	8.1E-09	1.1E-05
	Utah	5.1E-07	4.6E-06	8.1E-09	4.5E-08	5.4E-06
	Wyoming	9.1E-07	3.6E-06	4.5E-08	7.8E-09	4.5E-06

Sample calculation: Rural route from Indian Point to Hanford through Idaho, truck cask

RADTRAN output (unit risk): 3.05E-08 person-rem (from Table II-8)
Population density: 11.3 persons/km^2
Population multiplier: 1.133
Route segment length: 357 km
Population (collective) dose = 3.05E-08*11.3*1.133*357= 1.39E-04 person-rem
Convert to SI units: 1.39E-04 person-rem*0.01person-Sv/person-rem = 1.39E-06 person-Sv

Table II-15. Collective doses to residents along the route (person-Sv) from truck transportation (Truck-DU); shipment origin INL

DESTINATION	ROUTES	Rural	Suburban	Urban	Urban Rush Hour	Total
ORNL	Colorado	1.0E-06	4.7E-06	1.4E-07	3.1E-08	5.86E-06
	Idaho	6.3E-07	2.7E-06	2.4E-08	5.4E-09	3.32E-06
	Illinois	9.1E-07	6.0E-06	1.9E-08	4.3E-09	6.98E-06
	Kansas	1.6E-06	6.7E-06	1.1E-07	2.5E-08	8.50E-06
	Kentucky	5.8E-07	2.3E-06	1.8E-09	4.1E-10	2.86E-06
	Missouri	1.2E-06	1.5E-05	3.1E-07	6.9E-08	1.65E-05
	Tennessee	1.4E-06	8.8E-06	1.1E-07	2.5E-08	1.03E-05
	Utah	6.6E-07	4.2E-06	1.8E-08	4.1E-09	4.89E-06
	Wyoming	7.9E-07	2.5E-06	2.5E-08	5.5E-09	3.30E-06
DEAF SMITH	Colorado	1.3E-06	1.6E-05	4.4E-07	9.8E-08	1.74E-05
	Idaho	6.3E-07	2.7E-06	2.4E-08	5.4E-09	3.32E-06
	New	1.2E-06	5.6E-06	1.8E-07	4.1E-08	7.02E-06
	Texas	5.3E-08	9.4E-08			1.47E-07
	Utah	6.6E-07	**4.2E-06**	1.8E-08	4.1E-09	4.89E-06
	Wyoming	7.9E-07	2.5E-06	2.5E-08	5.5E-09	3.30E-06
HANFORD	Idaho	1.7E-06	8.7E-06	1.1E-07	2.5E-08	1.05E-05
	Oregon	3.0E-06	2.5E-08	5.6E-10	5.6E-09	3.83E-06
	Washington	8.4E-07	5.0E-08	1.1E-09	1.1E-08	9.92E-07
SKULL VALLEY	Idaho	6.3E-07	2.7E-06	2.4E-08	5.4E-09	3.32E-06
	Utah	6.0E-07	7.4E-06	2.4E-07	5.4E-08	8.33E-06

Sample calculation: Suburban route from INL to Deaf Smith through Utah, truck cask

RADTRAN output (unit risk): 2.73E-08 person-rem (from Table II-8)
Population density: 260.1persons/km^2
Population multiplier: 1.102
Route segment length: 53.4 km
Population (collective) dose = 2.73E-08*260.1*1.102*53.4= 4.18E-04 person-rem
Convert to SI units: 4.18E-04 person-rem*0.01person-Sv/person-rem = 4.18E-06 person-Sv

Table II-16. Collective doses to residents along the route (person-Sv) from truck transportation (Truck-DU); shipment origin Kewaunee.

DESTINATION	ROUTES	Rural	Suburban	Urban	Urban Rush Hour	Total
ORNL	Illinois	2.1E-07	9.8E-06	3.6E-07	7.9E-08	1.05E-05
	Indiana	1.3E-06	1.2E-05	2.0E-07	4.5E-08	1.31E-05
	Kentucky	1.2E-06	1.0E-05	1.3E-07	2.9E-08	1.17E-05
	Ohio	5.9E-08	8.3E-07	1.1E-08	2.4E-09	9.04E-07
	Tennessee	3.6E-07	6.2E-06	7.8E-08	1.7E-08	6.67E-06
	Wisconsin	9.8E-07	7.6E-06	3.5E-07	7.8E-08	9.00E-06
DEAF SMITH	Illinois	7.5E-07	3.3E-06	1.2E-08	2.7E-09	4.02E-06
	Iowa	1.4E-06	7.4E-06	6.3E-08	1.4E-08	8.90E-06
	Kansas	1.0E-06	6.9E-06	1.6E-07	3.6E-08	8.16E-06
	Missouri	6.5E-07	6.1E-06	6.2E-08	1.2E-08	6.81E-06
	Oklahoma	1.1E-06	6.0E-06	1.8E-09	2.1E-08	7.18E-06
	Texas	3.9E-07	2.2E-06	9.5E-08	2.1E-08	2.76E-06
	Wisconsin	1.2E-06	7.6E-06	2.8E-07	6.1E-08	9.20E-06
HANFORD	Idaho	2.1E-07	4.1E-06	6.0E-08	1.1E-08	4.03E-06
	Minnesota	1.7E-06	2.7E-06	1.3E-08	2.9E-09	4.46E-06
	Montana	2.1E-06	8.2E-06	1.0E-07	2.2E-08	1.03E-05
	South Dakota	1.5E-06	3.9E-06	3.1E-08	6.9E-09	5.35E-06
	Washington	1.0E-06	9.2E-06	2.0E-07	4.5E-08	1.05E-05
	Wisconsin	2.1E-06	1.1E-05	2.7E-07	6.0E-08	1.38E-05
	Wyoming	5.6E-07	1.6E-06	2.2E-08	4.9E-09	2.21E-06
SKULL VALLEY	Illinois	7.5E-07	3.3E-06	1.2E-08	2.7E-09	4.02E-06
	Iowa	1.9E-06	6.8E-06	5.8E-08	1.3E-08	8.73E-06
	Nebraska	2.0E-06	5.4E-06	8.8E-08	2.0E-08	7.48E-06
	Utah	5.1E-07	4.6E-06	2.0E-07	4.5E-08	5.39E-06
	Wisconsin	1.2E-06	7.6E-06	**2.8E-07**	6.1E-08	9.20E-06
	Wyoming	9.1E-07	3.6E-06	3.5E-08	7.8E-09	4.54E-06

Sample calculation: Urban route from Kewaunee to Skull Valley through Wisconsin, truck cask, not during rush hour

RADTRAN output (unit risk): 5.22E-10 person-rem (from Table II-8)
Population density: 2660 persons/km^2
Population multiplier: 1.0
Route segment length: 19.9 km
Population (collective) dose = 5.22E-10*2660*1.0*19.9 = 2.76E-05 person-rem
Convert to SI units: 2.76E-05 person-rem*0.01person-Sv/person-rem = 2.76E-07 person-Sv

Collective dose is best used in making comparisons; e.g., in comparing the risks of routine transportation along different routes. All collective doses modeled are of the order of 10^{-5} person-Sv or less. The tables show that, in general, urban residents sustain a slightly larger dose from rail transportation than from truck transportation on the same state route, even though urban population densities are similar; e.g., for the Maine urban segment of the Maine Yankee-to-ORNL route,

- the truck route urban population density is 2706 persons/km^2 and the collective dose is 6.6×10^{-8} person-Sv
- the rail route urban population density is 2527 persons/km^2, but the collective dose is 6.2×10^{-7} person-Sv from the Rail-Lead cask is almost 10 times larger than the dose from the Truck-DU cask, even though the external dose rates from the two casks are nearly the same.

Doses from rail transportation through urban areas are larger than those from truck transportation because train transportation was designed, and train tracks were laid, to go from city center to city center. Trucks carrying spent fuel, on the other hand, are required to use the interstate highway system, and to use bypasses around cities where such bypasses exist. In the example presented, the truck traverses 5 km of urban route while the train traverses 13 urban km. In addition, the average urban train speed is 24 km/hour (15 mph) while the average urban truck speed is 102 km/hour (63.4 mph). A truck carrying a cask through an urban area at about four times the speed of a train carrying a similar cask will deliver ¼ the dose of the rail cask.

II.5.4.2 Doses to occupants of vehicles sharing the route

The dose to occupants of vehicles sharing a highway route (the on-link dose) consists of the sum of three components:

- dose to persons in vehicles traveling in the opposite direction to the shipment
- dose to persons in vehicles traveling in the same direction as the shipment, and
- dose to persons in passing vehicles.

In the case of rail, there is a dose only to occupants of railcars (the rail analog to highway vehicles) traveling in the opposite direction, since passing on parallel track is rarely the case. RADTRAN uses Equation II-4 to calculate the dose to occupants of vehicles traveling in the opposite direction. The result is

II-5 $$D_{opp} = 2*\left(\frac{N*PPV}{V}\right)*\frac{Qk_0DR_v}{V}\left[f_\gamma * I_\gamma + f_n * I_n\right]$$

Where D_{opp} is the dose to occupants of railcars traveling in the opposite direction
N is the number of railcars sharing the route
PPV is the number of passengers per railcar

And the other terms are defined as in preceding equations. The factor of two is included to account for the vehicle moving toward the radioactive cargo and then away from it and an additional factor of (*N*PPV/V*) for the dose to people in the oncoming vehicle, which is assumed to be traveling at the same speed as the cargo. N is the number of oncoming vehicles per hour and P is the number of persons per vehicle.

Rail

The dose to occupants of railcars other than the railcar carrying the radioactive cargo, and moving in the opposite direction, is provided in Table II-17. The vehicle occupancies used to calculate the table, one person on rural and suburban segments, and five people on urban segments, have been used historically in RADTRAN since 1988. The occupancy is consistent with the following considerations:

- Freight trains carry a crew of three, but all but one or two of the 60 to 120 cars on a freight train are unoccupied.
- Urban track carries almost all passenger rail traffic.
- Dose is calculated for one cask on a train, and rail statistics are per railcar, not per train.

The dose to occupants of other trains depends on train speed and the external dose rate from the spent fuel cask. Train speeds are available only for the entire U.S., not for each state. Therefore the doses to occupants of trains that share the route with either a loaded Rail-Lead cask or a loaded Rail-Steel cask are shown in Table II-17 for rural, suburban, and urban segments of each entire route, rather than state-by-state.

Table II-17. Collective doses (person-Sv) to occupants of trains sharing the route.

SHIPMENT ORIGIN/ DESTINATION	Rail-Lead Cask				Rail-Steel Cask			
	Rural	Suburban	Urban	Total	Rural	Suburban	Urban	Total
MAINE YANKEE								
ORNL	2.0E-05	1.2E-05	7.5E-06	4.0E-05	1.5E-05	9.3E-06	5.6E-06	3.0E-05
DEAF SMITH	3.8E-05	1.3E-05	9.7E-06	6.1E-05	2.9E-05	1.0E-05	7.4E-06	4.6E-05
HANFORD	6.2E-05	1.7E-05	1.6E-05	9.0E-05	4.7E-05	1.3E-05	1.2E-05	6.8E-05
SKULL VALLEY	4.8E-05	1.6E-05	**9.6E-06**	7.4E-05	3.6E-05	1.2E-05	7.3E-06	5.5E-05
KEWAUNEE								
ORNL	1.4E-05	7.0E-06	5.8E-06	2.7E-05	1.0E-05	5.3E-06	4.4E-06	2.0E-05
DEAF SMITH	2.4E-05	5.2E-06	5.1E-06	3.4E-05	1.8E-05	4.0E-06	3.9E-06	2.6E-05
HANFORD	4.2E-05	6.7E-06	2.8E-06	5.2E-05	3.2E-05	5.1E-06	2.1E-06	3.9E-05
SKULL VALLEY	3.5E-05	7.8E-06	5.8E-06	4.9E-05	2.7E-05	5.9E-06	4.4E-06	3.7E-05
INDIAN POINT								
ORNL	9.2E-06	8.1E-06	9.6E-06	2.7E-05	7.0E-06	6.1E-06	7.2E-06	2.0E-05
DEAF SMITH	3.6E-05	1.1E-05	9.4E-06	5.6E-05	2.8E-05	8.2E-06	7.1E-06	4.3E-05
HANFORD	6.0E-05	1.4E-05	1.1E-05	8.5E-05	4.6E-05	1.1E-05	8.0E-06	6.5E-05
SKULL VALLEY	4.8E-05	1.3E-05	1.1E-05	6.5E-05	3.6E-05	1.0E-05	8.0E-06	4.9E-05
INL								
ORNL	4.6E-05	7.1E-06	3.4E-06	5.7E-05	3.5E-05	5.4E-06	2.6E-06	4.3E-05
DEAF SMITH	2.7E-05	3.2E-06	1.9E-06	3.2E-05	2.1E-05	2.5E-06	1.4E-06	2.5E-05
HANFORD	1.5E-05	1.7E-06	9.3E-07	1.8E-05	1.2E-05	1.3E-06	7.0E-07	1.4E-05
SKULL VALLEY	5.5E-06	1.5E-06	1.2E-06	8.2E-06	4.2E-06	1.1E-06	9.0E-07	6.2E-06

Sample calculation: Urban segment from Maine Yankee to Skull Valley, rail-lead cask

RADTRAN output (unit risk): 4.63E-6 person-rem (from Figure II-13)
Route urban length: 207 km (from Table II-4)
Population (collective) dose = 4.63E-6*207= 9.58E-04 person-rem
Convert to SI units: 9.58E-04 person-rem*0.01person-Sv/person-rem = 9.58E-06 person-Sv

The number of occupants of other trains, the train speed, and the railcars per hour are incorporated into the RADTRAN calculation. This is then multiplied by the total rural, suburban, and urban kilometers, respectively, of the route.

<u>Truck</u>

Vehicle density data for large semi-detached trailer trucks traveling U.S. interstates and primary highways is available and well qualified. Every state records traffic counts on major (and most minor) highways and publishes these routinely. Average vehicle density data from each of the 10

EPA regions was used (Weiner, et al., 2009, Appendix D). The EPA regions were used because they include all of the "lower 48" U.S. states (Alaska and Hawaii are included in EPA Region 10 but are not considered in this risk assessment because no spent fuel will be shipped to or from them). Table II-18 shows the 10 EPA regions.

Table II-18. States comprising the ten EPA regions

Region	States Included in Region	Vehicles per Hour		
		Rural	Suburban	Urban
1	Connecticut, Massachusetts, Maine, New Hampshire, Rhode Island, Vermont	439	726	2129
2	New Jersey, New York	1015	2094	4163
3	Delaware. Maryland, Pennsylvania, Virginia, West Virginia	2056	3655	5748
4	Alabama, Florida, Georgia, Kentucky, Mississippi, North Carolina, South Carolina, Tennessee	1427	2776	5611
5	Illinois, Indiana, Michigan, Minnesota, Ohio, Wisconsin	1200	2466	4408
6	Arkansas, Louisiana, New Mexico, Oklahoma, Texas	897	1498	3003
7	Iowa, Kansas, Missouri, Nebraska	926	1610	2463
8	Colorado, Montana, North Dakota, South Dakota, Utah, Wyoming	795	1958	3708
9	Arizona, California, Nevada	1421	3732	7517
10	Idaho, Oregon , Washington	1123	2670	5624

The calculation of doses to occupants sharing the highway route with the radioactive materials truck includes the dose to vehicles passing the radioactive cargo and vehicles in an adjoining lane, as well as vehicles traveling in the opposite direction. The equations that describe this calculation are Equations 28 and 34 of Neuhauser et al, 2000.

Figure II-14 is the diagram accompanying these equations and shows the parameters used in the calculation. Parameter values are in Table II-1.

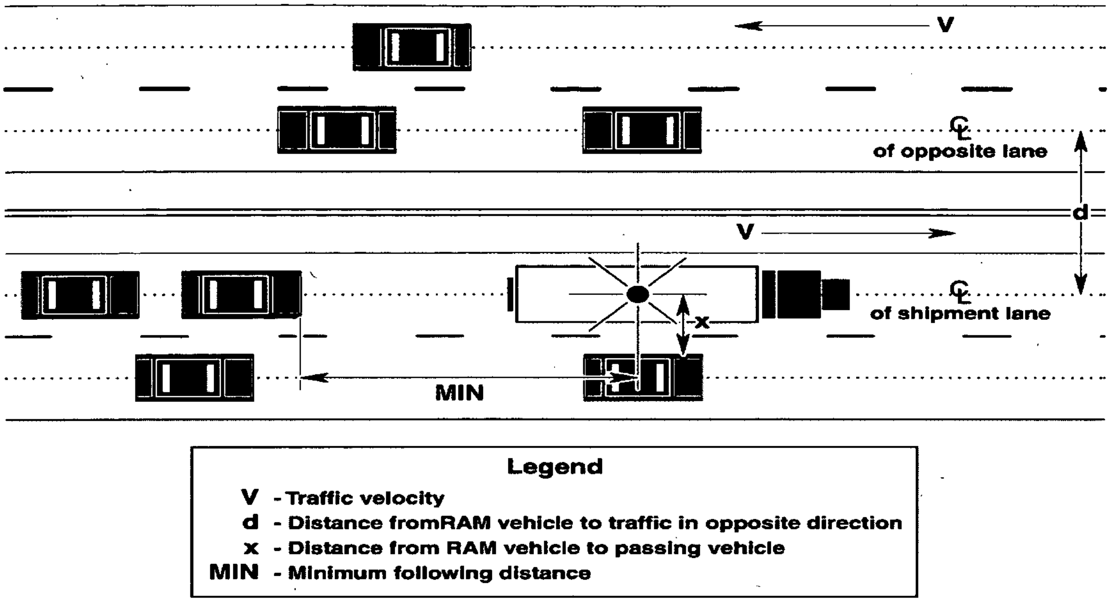

Figure II-14. Parameters for calculating doses to occupants of highway vehicles sharing the route with the radioactive shipment (from Figure 3-2 of Neuhauser, et al., 2000).

Table II-19 to Table II-22 show the doses to individuals in vehicles sharing the highway route with the truck carrying a loaded Truck-DU cask. The number of occupants of other vehicles, the vehicle speed, and the vehicles per hour are incorporated into the RADTRAN calculation. This is then multiplied by the rural, suburban, and urban kilometers, respectively, of each state transited.

Table II-19. Collective doses to persons sharing the route (person-Sv) from truck transportation (Truck-DU); shipment origin Maine Yankee

DESTINATION	ROUTES	Rural	Suburban	Urban	Urban Rush Hour	Total
ORNL	Connecticut	**2.0E-06**	9.2E-06	9.2E-06	8.5E-06	2.9E-05
	Maine	2.9E-06	6.7E-06	1.1E-06	1.0E-06	1.2E-05
	Maryland	1.3E-06	4.9E-06	9.0E-07	8.3E-07	8.0E-06
	Massachusetts	1.7E-06	8.7E-06	3.4E-06	3.2E-06	1.7E-05
	New Hampshire	3.7E-07	1.4E-06	1.9E-07	1.8E-07	2.1E-06
	New Jersey	4.5E-06	1.6E-05	6.6E-06	6.1E-06	3.3E-05
	New York	7.5E-07	2.1E-06	1.3E-05	1.2E-05	2.7E-05
	Pennsylvania	3.0E-05	4.8E-05	7.0E-06	6.5E-06	9.2E-05
	Tennessee	1.7E-05	3.2E-05	4.2E-06	3.9E-06	5.6E-05
	Virginia	6.4E-05	9.3E-05	6.2E-06	5.7E-06	1.7E-04
	West Virginia	2.8E-06	1.2E-05	4.5E-07	4.1E-07	1.5E-05
DEAF SMITH	Arkansas	3.1E-05	2.1E-05	2.8E-06	2.6E-06	5.8E-05
	Connecticut	2.0E-06	9.2E-06	9.2E-06	8.5E-06	2.9E-05
	Maine	2.9E-06	6.8E-06	7.3E-07	6.8E-07	1.1E-05
	Maryland	1.3E-06	4.9E-06	9.0E-07	8.3E-07	8.0E-06
	Massachusetts	1.7E-06	8.7E-06	3.4E-06	3.2E-06	1.7E-05
	New Hampshire	3.7E-07	1.4E-06	1.9E-07	1.8E-07	2.1E-06
	New Jersey	4.5E-06	1.6E-05	6.6E-06	6.1E-06	3.3E-05
	New York	7.5E-07	6.8E-06	6.9E-06	6.4E-06	2.1E-05
	Oklahoma	4.2E-05	1.6E-05	2.8E-06	2.6E-06	6.4E-05
	Pennsylvania	3.0E-05	4.8E-05	7.0E-06	6.5E-06	9.2E-05
	Tennessee	7.8E-05	8.6E-05	2.0E-05	1.8E-05	2.0E-04
	Texas	2.2E-05	3.1E-06	2.4E-06	2.2E-06	2.9E-05
	Virginia	6.4E-05	9.3E-05	6.2E-06	5.7E-06	1.7E-04
	West Virginia	2.8E-06	1.2E-05	4.5E-07	4.1E-07	1.5E-05

Table II-19. Collective doses to persons sharing the route (person-Sv) from truck transportation (Truck-DU); shipment origin Maine Yankee -- continued

DESTINATION	ROUTES	Rural	Suburban	Urban	Urban Rush Hour	Total
HANFORD	Connecticut	1.7E-06	8.0E-06	5.1E-06	4.7E-06	2.0E-05
	Idaho	4.4E-05	2.3E-05	4.6E-06	4.2E-06	7.6E-05
	Illinois	2.4E-05	2.0E-05	5.0E-06	4.6E-06	5.4E-05
	Indiana	1.8E-05	2.6E-05	4.6E-06	4.3E-06	5.3E-05
	Iowa	4.0E-05	1.7E-05	1.4E-06	1.3E-06	6.0E-05
	Maine	2.9E-06	6.8E-06	7.3E-07	6.8E-07	1.1E-05
	Massachusetts	1.7E-06	8.7E-06	3.4E-06	3.2E-06	1.7E-05
	Nebraska	6.7E-05	1.3E-05	1.9E-06	1.8E-06	8.4E-05
	New Hampshire	3.7E-07	1.4E-06	1.9E-07	1.8E-07	2.1E-06
	New York	2.5E-06	4.6E-06	1.1E-06	9.9E-07	9.2E-06
	Ohio	2.8E-05	6.9E-05	4.0E-06	3.7E-06	1.0E-04
	Oregon	3.7E-05	9.5E-06	1.4E-06	1.3E-06	4.9E-05
	Pennsylvania	8.7E-05	6.9E-05	4.0E-06	3.7E-06	1.6E-04
	Utah	1.6E-05	1.1E-05	6.2E-07	5.7E-07	2.8E-05
	Washington	7.6E-06	2.1E-06	2.6E-06	2.4E-06	1.5E-05
	Wyoming	7.5E-05	1.0E-05	2.1E-06	2.0E-06	8.9E-05
SKULL VALLEY	Connecticut	1.7E-06	8.0E-06	5.1E-06	4.7E-06	2.0E-05
	Illinois	2.4E-05	2.0E-05	5.0E-06	4.6E-06	5.4E-05
	Indiana	1.8E-05	2.6E-05	4.6E-06	4.3E-06	5.3E-05
	Iowa	4.0E-05	1.7E-05	1.4E-06	1.3E-06	6.0E-05
	Maine	2.9E-06	6.8E-06	7.3E-07	6.8E-07	1.1E-05
	Massachusetts	1.7E-06	8.7E-06	3.4E-06	3.2E-06	1.7E-05
	Nebraska	6.7E-05	1.3E-05	1.9E-06	1.8E-06	8.4E-05
	New Hampshire	3.7E-07	4.8E-06	4.8E-07	4.4E-06	1.0E-05
	New York	5.8E-06	1.3E-05	2.1E-06	1.9E-06	2.3E-05
	Ohio	2.8E-05	4.1E-05	7.3E-06	6.7E-06	8.3E-05
	Pennsylvania	8.7E-05	6.9E-05	4.0E-06	3.7E-06	1.6E-04
	Utah	1.8E-05	8.1E-06	6.1E-06	5.6E-06	3.8E-05
	Wyoming	7.5E-05	1.0E-05	2.1E-06	2.0E-06	8.9E-05

Sample Calculation: Rural segment from Maine Yankee to ORNL through Connecticut (Region 1)

Unit risk (From Table II-8: 4.80E-08 Sv

Rural route segment length: 40.7 km

Dose to occupants of vehicles sharing the route: 4.80E-08*40.7 = 1.95E-06

Table II-20. Collective doses to persons sharing the route (person-Sv) from truck transportation (Truck-DU); shipment origin Indian Point.

DESTINATION	ROUTES	Rural	Suburban	Urban	Urban RH	Total
ORNL	Maryland	1.3E-06	4.9E-06	9.0E-07	8.3E-07	7.9E-06
	New Jersey	4.5E-06	1.6E-05	6.6E-06	6.1E-06	3.3E-05
	New York	1.3E-06	6.5E-06	7.6E-06	7.0E-06	2.2E-05
	Pennsylvania	3.0E-05	4.8E-05	7.0E-06	6.5E-06	9.2E-05
	Tennessee	1.7E-05	3.4E-05	3.8E-06	3.5E-06	5.8E-05
	Virginia	6.4E-05	9.3E-05	6.2E-06	5.7E-06	1.7E-04
	West Virginia	6.4E-05	1.2E-05	4.5E-07	4.1E-07	7.7E-05
DEAF SMITH	Arkansas	3.1E-05	2.1E-05	2.8E-06	2.6E-06	5.7E-05
	Maryland	1.3E-06	4.9E-06	9.0E-07	8.3E-07	7.9E-06
	New Jersey	4.5E-06	1.6E-05	6.6E-06	6.1E-06	3.3E-05
	New York	1.3E-06	6.5E-06	7.6E-06	7.0E-06	2.2E-05
	Oklahoma	4.2E-05	1.6E-05	2.8E-06	2.6E-06	6.3E-05
	Pennsylvania	3.0E-05	4.8E-05	7.0E-06	6.5E-06	9.2E-05
	Tennessee	7.8E-05	8.6E-05	2.0E-05	1.8E-05	2.0E-04
	Texas	2.2E-05	3.1E-06	2.4E-06	2.2E-06	3.0E-05
	Virginia	6.4E-05	9.3E-05	6.2E-06	5.7E-06	1.7E-04
	West Virginia	2.8E-06	1.2E-05	4.5E-07	4.1E-07	1.6E-05
HANFORD	Idaho	4.4E-05	2.3E-05	4.6E-06	**4.2E-06**	7.6E-05
	Illinois	2.4E-05	2.0E-05	5.0E-06	4.6E-06	5.4E-05
	Indiana	1.8E-05	2.6E-05	4.6E-06	4.3E-06	5.3E-05
	Iowa	4.0E-05	1.7E-05	1.4E-06	1.3E-06	6.0E-05
	Nebraska	6.7E-05	1.3E-05	1.9E-06	1.8E-06	8.4E-05
	New Jersey	4.8E-06	1.3E-05	5.6E-06	5.2E-06	2.9E-05
	New York	1.3E-06	6.5E-06	7.6E-06	7.0E-06	2.2E-05
	Ohio	1.5E-06	7.6E-06	8.1E-06	7.4E-06	2.5E-05
	Oregon	3.7E-05	9.5E-06	1.4E-06	1.3E-06	4.9E-05
	Pennsylvania	8.0E-05	5.7E-05	2.2E-06	2.0E-06	1.4E-04
	Utah	1.6E-05	1.1E-05	6.2E-07	5.7E-07	2.8E-05
	Washington	7.6E-06	2.1E-06	2.6E-06	2.4E-06	1.5E-05
	Wyoming	7.5E-05	1.0E-05	2.1E-06	2.0E-06	8.9E-05
SKULL VALLEY	Illinois	2.4E-05	2.0E-05	5.0E-06	4.6E-06	5.4E-05
	Indiana	1.8E-05	2.6E-05	4.6E-06	4.3E-06	5.3E-05
	Iowa	4.0E-05	1.7E-05	1.4E-06	1.3E-06	6.0E-05
	Nebraska	6.7E-05	1.3E-05	1.9E-06	1.8E-06	8.4E-05
	New Jersey	5.6E-06	1.5E-05	5.9E-06	5.5E-06	3.2E-05
	New York	1.5E-06	7.6E-06	8.1E-06	7.4E-06	2.5E-05
	Ohio	2.8E-05	4.1E-05	7.3E-06	6.7E-06	8.3E-05
	Pennsylvania	8.0E-05	5.7E-05	2.2E-06	2.0E-06	1.4E-04
	Utah	1.8E-05	8.1E-06	6.1E-06	5.6E-06	3.8E-05
	Wyoming	7.5E-05	1.0E-05	2.1E-06	2.0E-06	8.9E-05

Sample Calculation: Urban rush hour segment from Indian Point to Hanford through Idaho (Region 10)
Unit risk (From Table II-8: 5.80E-07Sv
Urban route segment length: 7.3 km
Dose to occupants of vehicles sharing the route: 5.80E-07*7.3 = 4.23E-06

Table II-21. Collective doses to persons sharing the route (person-Sv) from truck transportation (Truck-DU); shipment origin INL

DESTINATION	ROUTES	Rural	Suburban	Urban	Urban Rush Hour	Total
ORNL	Colorado	3.1E-05	1.1E-05	4.0E-06	3.7E-06	5.0E-05
	Idaho	2.2E-05	8.0E-06	1.3E-06	1.2E-06	3.3E-05
	Illinois	2.5E-05	2.4E-05	1.1E-06	1.0E-06	5.1E-05
	Kansas	6.2E-05	1.4E-05	2.7E-06	2.5E-06	8.1E-05
	Kentucky	1.8E-05	1.1E-05	1.2E-07	1.2E-07	2.9E-05
	Missouri	2.5E-05	2.3E-05	7.2E-06	6.7E-06	6.2E-05
	Tennessee	3.3E-05	3.5E-05	5.2E-06	4.8E-06	7.8E-05
	Utah	1.3E-05	1.1E-05	6.2E-07	5.7E-07	2.5E-05
	Wyoming	7.5E-05	1.0E-05	2.1E-06	2.0E-06	8.9E-05
DEAF SMITH	Colorado	3.9E-05	3.6E-05	1.9E-05	1.8E-05	1.1E-04
	Idaho	2.2E-05	8.0E-06	1.3E-06	1.2E-06	3.3E-05
	New Mexico	6.4E-05	9.8E-06	4.8E-06	4.4E-06	8.3E-05
	Texas	7.7E-06	1.7E-07			7.9E-06
	Utah	1.3E-05	**1.1E-05**	6.2E-07	5.7E-07	2.5E-05
	Wyoming	7.0E-05	7.6E-06	1.5E-06	1.4E-06	8.1E-05
HANFORD	Idaho	5.5E-05	6.3E-05	5.4E-06	5.0E-06	1.3E-04
	Oregon	3.7E-05	2.0E-05	1.4E-06	1.3E-06	6.0E-05
	Washington	7.6E-06	2.1E-06	2.6E-06	2.4E-06	1.5E-05
SKULL VALLEY	Idaho	2.2E-05	8.0E-06	1.3E-06	1.2E-06	4.2E-05
	Utah	1.5E-05	1.5E-05	7.2E-06	6.6E-06	4.4E-05

Sample Calculation: Suburban segment from INL to Deaf Smith through Utah (Region 8)
Unit risk (From Table II-8: 2.20E-07 Sv
Suburban route segment length: 53.4 km
Dose to occupants of vehicles sharing the route: 2.15E-07*53.4 = 1.148E-05

Table II-22. Collective doses to persons sharing the route (person-Sv) from truck transportation (Truck-DU); shipment origin Kewaunee.

DESTINATION	ROUTES	Rural	Suburban	Urban	Urban Rush Hour	Total
ORNL	Illinois	3.7E-06	2.0E-05	1.4E-05	1.3E-05	5.1E-05
	Indiana	3.3E-05	3.8E-05	8.3E-06	7.7E-06	8.7E-05
	Kentucky	2.7E-05	4.3E-05	7.2E-06	6.7E-06	8.4E-05
	Ohio	1.4E-06	2.5E-06	5.4E-07	5.0E-07	4.9E-06
	Tennessee	1.1E-05	1.8E-05	4.4E-06	4.1E-06	3.8E-05
	Wisconsin	2.0E-05	2.1E-05	1.3E-05	1.2E-05	6.6E-05
DEAF SMITH	Illinois	2.0E-05	1.2E-05	5.9E-07	5.4E-07	3.3E-05
	Iowa	3.2E-05	1.6E-05	1.6E-06	1.4E-06	5.1E-05
	Kansas	2.9E-05	1.2E-05	3.5E-06	3.2E-06	4.8E-05
	Missouri	1.4E-05	1.1E-05	1.3E-06	1.2E-06	2.8E-05
	Oklahoma	3.4E-05	1.1E-05	2.8E-06	2.6E-06	5.0E-05
	Texas	2.2E-05	3.1E-06	2.4E-06	2.2E-06	3.0E-05
	Wisconsin	2.5E-05	2.3E-05	9.8E-06	9.0E-06	6.7E-05
HANFORD	Idaho	9.3E-06	1.1E-05	3.0E-06	2.8E-06	2.6E-05
	Minnesota	5.2E-05	1.3E-05	5.4E-07	5.0E-07	6.6E-05
	Montana	9.6E-05	3.0E-05	5.4E-06	5.0E-06	1.4E-04
	South Dakota	5.3E-05	1.2E-05	1.0E-06	9.5E-07	6.7E-05
	Washington	4.6E-05	3.0E-05	1.1E-05	1.0E-05	9.7E-05
	Wisconsin	4.6E-05	4.0E-05	9.9E-06	9.2E-06	1.1E-04
	Wyoming	4.0E-05	4.1E-06	1.4E-06	1.3E-06	4.7E-05
SKULL VALLEY	Illinois	2.0E-05	1.2E-05	5.9E-07	5.4E-07	3.3E-05
	Iowa	4.0E-05	1.7E-05	1.4E-06	1.3E-06	6.0E-05
	Nebraska	6.7E-05	1.3E-05	1.9E-06	1.8E-06	8.4E-05
	Utah	2.4E-05	1.0E-05	8.8E-06	8.1E-06	4.4E-05
	Wisconsin	2.5E-05	2.3E-05	**9.8E-06**	9.0E-06	6.7E-05
	Wyoming	7.5E-05	1.0E-05	2.1E-06	2.0E-06	8.9E-05

Sample Calculation: Urban segment from Kewaunee to Skull Valley through Wisconsin (Region 5) not during rush hour
Unit risk (From Table II-8: 4.90E-07 Sv
Urban route segment length: 19.9 km
Dose to occupants of vehicles sharing the route: 4.90E-07*19.9 = 9.75E-06

II.5.4.3 Doses from stopped vehicles

Rail

Trains are stopped in classification yards at the origin and destination of the trip. The usual length of these classification stops is 27 hours. The collective dose to the railyard workers at these classification stops from the radioactive cargo is calculated internally by RADTRAN and is based on calculations of Wooden (1986) which authors of this document have verified. This "classification yard dose" for the two rail casks studied is:

- For the Rail-Lead: 1.5×10^{-5} person-Sv
- For the Rail-Steel: 1.1×10^{-5} person-Sv
- These collective doses include doses to the train crew while the train is in the yard.

The collective dose to people living near a classification yard is calculated by multiplying the average dose from the rail cask to an individual living near a classification yard, as shown in Table II-7, by the population density between 200 and 800 meters from the rail yard. The population density is obtained from WebTRAGIS, and the integration from 200 to 800 meters (Table II-2) is performed by RADTRAN.

Most train stops along any route are shown in the WebTRAGIS output for that route. The stops on the rail route from Maine Yankee to Hanford are shown in Table II-23 as an example.

Table II-23. Example of rail stops on the Maine Yankee-to-Hanford rail route

Stop	Reason	Route type (R, S, U)[a] and State	Time (hours)
Classification	Initial classification	S, ME	27
1	Railroad transfer (short line to ST)	S, ME	4.0
2	Railroad transfer (ST to CSXT)	R, NY	4.0
3	Railroad transfer (CSXT to IHB)	S, IL	2.0
4	Railroad transfer (IHB to BNSF)	S, IL	<<1
5	Railroad transfer (BNSF to UP)	S, WA	<<1
Classification	Final classification	S, WA	27

[a]Determined from the WebTRAGIS output

Railyard worker collective doses can then be calculated for Stops 1, 2, and 3 in Table II-23. Parameter values are from Table II-23 and the classification yard dose above:

Dose: $(4/27)*(1.5 \times 10^{-5}) = 2.2 \times 10^{-6}$ person- Sv for the Rail-Lead cask.

Dose: $(4/27)*(1.1 \times 10^{-5}) = 1.6 \times 10^{-6}$ person-Sv for the Rail-Steel cask.

The factor of 4/27 is in the equation because the classification stop doses are calculated by RADTRAN for activities lasting a total of 27 hours, and the in-transit stops are for only four hours.

The average dose to an individual living 200 to 800 meters from a classification yard, as calculated by RADTRAN, is

- 3.5×10^{-7} Sv from the Rail-Lead cask.
- 2.7×10^{-7} Sv from the Rail-Steel cask.

Collective doses to residents near a yard (a classification yard or railroad stop) are then calculated from the general expression:

Dose (person-Sv) = (Population density)*(Dose/hr to resident near yard)*(Stop time)*(shielding factor)

Thus, for a suburban population density of 373.8 persons/km^2 (the suburban population density through Maine along the Maine Yankee-to-Hanford route) living near Stop 1 in Table II-23,

Dose = (373.8 persons/km^2)*(3.5×10^{-7} Sv-km^2/hour)*(4 hours)*0.87 = 4.6×10^{-4} person-Sv.

Results for the stops are given in Table II-24.

Table II-24. Doses at rail stops on the Maine Yankee-to-Hanford rail route

Stop	Route type (R, S, U)[a] and State	Time (hours)	Railyard worker dose (person-Sv)[b]		Residents near stop (person-Sv)	
			Rail-Lead	Rail-Steel	Rail-Lead	Rail-Steel
Classification, origin	S, ME	27	1.5E-05	1.1E-05	2.3E-05	1.8E-05
1	S, ME	4.0	2.16E-06	1.61E-06	4.6E-04	3.5E-04
2	R, NY	4.0	2.16E-06	1.61E-06	2.5E-05	1.9E-05
3	S, IL	2.0	1.08E-06	8.05E-07	2.9E-04	2.2E-04
Classification, destination	S, WA	27	1.5E-05	1.1E-05	1.9E-05	1.4E-05

[a]Determined from the WebTRAGIS output
[b]The yard worker dose depends only on the length of time the railcar is stopped in the yard, independent of population density and shielding factor.

<u>Truck</u>

Doses at truck stops are calculated differently. There are two types of receptors at a truck stop, in addition to the truck crew: residents who live near the stop and people who share the stop with the refueling truck. Griego, et al. (1996) conducted some time and motion studies at a number of truck stops. They found that the average number of people at a stop between the gas pumps and the nearest building was 6.9, the average distance from the fuel pump to the nearest building was 15 meters, and the longest refueling time for a large semi-detached trailer truck was 0.83 hour (50 minutes). With these parameters, the collective dose to the people sharing the stop would be

2.3×10^{-4} person-Sv (Table II-8). The relationship between the collective dose and the number of receptors is not linear in this case.

The collective dose to residents near the stop is calculated in the same way as for rail transportation, using data in Table II-8, the population density of the region around the stop, and the stop time.

Dose (person-Sv) = (Population density)*(Dose/hr to resident near stop)*(Stop time)

Thus, for a rural population density of 15.4 persons/km^2 (the average along the Maine Yankee-to-Hanford truck route):

Dose/stop = (15.4 persons/km^2)*(3.3×10^{-8} Sv-km^2/hour)*(0.83 hours) = 4.2×10^{-5} person-Sv.

The population density used in the calculation is the density around the truck stop; appropriate residential shielding factors are used in the calculation. Unlike a train, the truck will stop several times on any truck route to fill the fuel tanks. Very large trucks generally carry two 80-gallon tanks each and stop for fuel when the tanks are half empty. A semi carrying a Truck-DU cask can travel an average of 845 km (DOE, 2002) before needing to refuel. The number of refueling (and rest) stops depends on the length of each type of route segment. The following equations are used in this calculation

Route segment length (km)/(845 km/stop) = stops/route segment

Dose (person-Sv) = (population/km^2)*(dose to resident near stop(Sv-km^2/hr))*
(stops/route segment)*(hours/stop)

Table II-25 shows the collective doses to residents near stops for the rural and suburban segments of the 16 truck routes in Table II-4. Trucks carrying Truck-DU casks of spent fuel are unlikely to stop in urban areas.

The rural and suburban population densities in Table II-25 are the averages for the entire route. An analogous calculation can be made for each state traversed. However, in neither case can one determine beforehand exactly where the truck will stop to refuel. In some cases (e.g., INL to Skull Valley) the truck may not stop at all because the total distance from INL to the Skull Valley site is only 466.2 km. The route from Indian Point to ORNL illustrates another situation. This route is 1028 km long, and would thus include one truck stop, which could be in either a rural or a suburban area.

Table II-25. Collective doses to residents near truck stops

Origin	Destination	Type	Persons/ km^2	Average number of stops	Person-Sv		
					Residents near Stops	Persons Sharing Stops	Total
Maine Yankee	ORNL	Rural	19.9	1.73	5.6E-07	3.9E-04	3.9E-04
		Suburban	395	2.09	1.2E-05	4.7E-04	4.8E-04
	Deaf Smith	Rural	18.6	2.47	1.5E-06	5.6E-04	5.6E-04
		Suburban	371	1.6	1.7E-05	3.6E-04	3.8E-04
	Hanford	Rural	15.4	4.33	1.9E-06	9.7E-04	9.8E-04
		Suburban	325	1.5	1.2E-05	3.4E-04	3.5E-04
	Skull Valley	Rural	16.9	3.5	1.9E-06	7.9E-04	7.9E-04
		Suburban	332.5	1.3	1.2E-05	2.9E-04	3.0E-04
Kewaunee	ORNL	Rural	19.8	0.81	5.2E-07	1.8E-04	1.8E-04
		Suburban	361	0.59	6.0E-06	1.3E-04	1.4E-04
	Deaf Smith	Rural	13.5	2.0	8.6E-07	4.5E-04	4.5E-04
		Suburban	339	0.52	5.0E-06	1.2E-04	1.2E-04
	Hanford	Rural	10.5	3.4	1.2E-06	7.7E-04	7.7E-04
		Suburban	316	0.60	5.4E-06	1.4E-04	1.4E-04
	Skull Valley	Rural	12.5	2.6	1.1E-06	5.9E-04	5.9E-04
		Suburban	324.5	0.44	4.1E-06	9.9E-05	1.0E-04
Indian Point	ORNL	Rural	20.5	0.71	4.7E-07	1.6E-04	1.6E-04
		Suburban	388	0.71	7.8E-06	1.6E-04	1.7E-04
	Deaf Smith	Rural	17.1	2.3	1.3E-06	5.2E-04	5.2E-04
		Suburban	370	1.2	1.3E-05	2.7E-04	2.8E-04
	Hanford	Rural	13.0	4.1	1.8E-06	9.2E-04	9.2E-04
		Suburban	338	1.1	1.1E-05	2.5E-04	2.6E-04
	Skull Valley	Rural	14.2	3.3	1.5E-06	7.4E-04	7.4E-04
		Suburban	351	0.93	9.3E-06	2.1E-04	2.2E-04
INL	ORNL	Rural	12.4	3.1	1.3E-06	7.0E-04	7.0E-04
		Suburban	304	0.72	6.3E-06	1.6E-04	1.7E-04
	Deaf Smith	Rural	7.8	2.3	5.8E-07	5.2E-04	5.2E-04
		Suburban	339	0.35	3.4E-06	7.9E-05	8.2E-05
	Hanford	Rural	6.5	0.43	2.0E-07	9.7E-05	9.7E-05
		Suburban	200	0.57	9.4E-07	1.3E-04	1.3E-04
	Skull Valley	Rural	10.1	0.42	1.4E-07	9.5E-05	9.5E-05
		Suburban	343	0.11	1.1E-06	2.5E-05	2.6E-05

[a]The number of stops is the kilometers of the route segment divided by 845 km, the distance between stops, so that it may be a fraction. Retaining the fraction allows the calculation to be repeated.

Sample Calculation: Rural stop from Maine Yankee to ORNL

Stop dose from RADTRAN output: 3.26E-06 rem = 3.26E-08 Sv. This takes into account the 30-to-800 m. bandwidth.

Average rural population density: 19.9 persons/km^2

Total rural km = 731
Distance between truck stops :845 km (DOE, 2002)
Number of truck stops: 731/845 = 0.865
Collective dose: 19.9 * 0.865*3.26E-08 = 5.6E-07

II.5.4.4 Occupational Doses

Occupational doses from routine, incident-free radioactive materials transportation include doses to truck and train crew, railyard workers, inspectors, and escorts. Not included are workers who handle spent fuel containers in storage, loading and unloading casks from vehicles or during intermodal transfer, and attendants who would refuel trucks, because truck refueling stops in the U.S. no longer have such attendants.[7]

Table II-26 summarizes the occupational doses.

Table II-26. Occupational doses per shipment from routine incident-free transportation

Cask and route type	Train crew in transit[a]: 3 people; person-Sv	Truck crew in transit 2 people; person-Sv	Escort: Sv/hour[a]	Inspector: Sv per inspection	Truck stop worker: Sv per shipment	Rail classification yard workers: person-Sv
Rail-Lead rural/suburban	5.4E-09		5.8E-06			1.5E-05
Rail-Lead urban	9.1E-08		5.8E-06			
Rail-Steel rural/suburban	4.1E-09		4.4E-06			1.1E-05
Rail-Steel urban	6.8E-09		4.4E-06			
Truck-DU rural/suburban		3.8E-07	4.9E-09	3.7E-04	6.7E-06	
Truck-DU urban		3.6E-07	4.9E-09			

[a]The truck crew is shielded while in transit to sustain a maximum dose of 0.02 mSv/hour

The doses to rail crew and rail escorts are similar. Spent fuel may be transported in dedicated trains so that both escorts and train crew are assumed to be within a railcar of the railcar carrying the spent fuel. Escorts in the escort car are not shielded, because they must maintain line-of-sight to the railcar carrying spent fuel. Train crew members are in a crew compartment and were assumed to have some shielding, resulting in an estimated dose about 25 percent less than the escort. The largest collective doses are to railyard workers. The number of workers in railyards is not a constant, and the number of activities that brings these workers into proximity with the shipment varies as well. This analysis assumes the dose to the worker doing an activity for each activity: inspection, coupling and decoupling the railcars, moving the railcar into position for

[7] The States of Oregon and New Jersey still requires gas station attendants to refuel cars and light duty vehicles, but heavy truck crew do their own refueling.

coupling, etc. The differences between doses in the Rail-Lead case and the Rail-Steel case reflect the differences in cask dimensions and in external dose rate.

Truck crew members are shielded so that they receive a maximum dose of 2.0×10^{-5} Sv per hour. This regulatory maximum was imposed in the RADTRAN calculation. Truck inspectors generally spend about an hour within one meter of the cargo (Weiner and Neuhauser, 1992), resulting in a relatively large dose. An upper bound to the duration of a truck refueling stop is about 50 minutes (0.83 hours) (Griego et al,, 1996). The truck stop worker whose dose is reflected in Table II-26 is assumed to be outside (unshielded) at 15 meters from the truck during the stop. Truck stop workers that are in concrete or brick buildings would be shielded from any radiation.

II.6 Interpretation of Collective Dose

Collective dose is essentially the product of an average radiation dose and the number of people who receive that average dose. The following example – a state suburban segment on a particular route – is typical of all routes in all states; only the specific numbers change.

The following parameters characterize a representative segment of the Maine Yankee-to-Hanford truck route; the suburban segment through Illinois, shown below, is a representative example:

- Route segment length: 73 km
- Suburban population density: 324 persons/km^2
- Area occupied by that population: $0.800 \text{ km} \times 2 \times 73 = 116.8 \text{ km}^2$
- Total suburban population exposed to the shipment = 37,800 people
- From Table II-13, the collective radiation dose to that population, from routine, incident-free transportation, is 6.5×10^{-6} person-Sv.
- U.S. background is 0.0036 Sv per year or 4.1×10^{-7} Sv per hour. At an average speed of 108 kph, the population is exposed for 0.675 hour.

The background dose sustained by each member of this population is 2.8×10^{-7} Sv for a total collective dose of 0.0105 person-Sv. The total collective dose is thus 0.0105065 person-Sv with the shipment, and 0.0105000 person-Sv without the shipment. The collective dose from routine, incident-free, transport is a very small increase in the collective dose the population continually receives from natural sources.