Final Draft

Review of DOE Planned Change Request for Shielded Containers for Remote-Handled Transuranic Waste

Contract Number EP-D-10-042
Work Assignment No. 1-02
Revision 4

Prepared for:

U.S. Environmental Protection Agency
Office of Radiation and Indoor Air
Center for Waste management and Regulations
1310 L Street N.W.
Washington, DC 20005

Jonathan Walsh
Work Assignment Manager

Prepared by:

S. Cohen & Associates
1608 Spring Hill Road
Suite 400
Vienna, Virginia 22182

December 29, 2010

TABLE OF CONTENTS

ATTACHMENTS

LIST OF TABLES

LIST OF FIGURES

LIST OF ACRONYMS

AA	Accident Analysis
AK	Acceptable Knowledge
ALARA	as low as reasonably achievable
AMAD	activity median aerodynamic diameter
AMWTF	Advanced Mixed Waste Treatment Facility
AMWTP	Advanced Mixed Waste Treatment Project
ANL	Argonne National Laboratory
ANL-CCP	Argonne National Laboratory - Central Characterization Project
ANLE-CCP	Argonne National Laboratory – East – Central Characterization Project
ANSI	American National Standards Institute
ARF	airborne release fraction
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials (ASTM International)
ATWIR	Annual Transuranic Waste Inventory Report
BCLDP	Battelle Columbus Laboratory Decommissioning Project
BDR	batch data report
BI	baseline inventories
C&T	Crawford and Taggart
C/kg	coulombs per kilogram
CBFO	Carlsbad Field Office (DOE)
CCA	Compliance Certification Application (WIPP)
CCDF	cumulative complementary distribution function
CCP	Central Characterization Project
CFR	*Code of Federal Regulations*
CH	contact handled
CH-TRAMPAC	Contact-Handled Transuranic Waste Authorized Methods for Payload Control
Ci	curie
CMS	configuration management system
CoC	Certificate of Compliance
cpm	counts per minute
CPR	cellulosics, plastics, and rubber

CRA	Compliance Recertification Application
CREL	Compliance Recertification Electronic Library
DBR	direct brine release
DOE	U.S. Department of Energy
DOT	U.S. Department of Transportation
DQO	data quality objective
DR	damage ratio
DRC	Document Review and Comment
DRF	document review forms
DSA	Documented Safety Analysis
DTC	dose-to-curie
EDTA	ethylene diamine tetraacetic acid
EPA	U.S. Environmental Protection Agency
FEP	feature, event, or process
FGE	fissile gram equivalent
FGR	Federal Guidance Report
FR	Federal Register
GA	gauge
GE VNC	General Electric Vallecitos Nuclear Center
GM	Geiger Mueller
Gs	standard acceleration of gravity
Gys	gray or grays
HP	health physics
Hz	Hertz
INL	Idaho National Laboratory
INL-CCP	Idaho National Laboratory - Central Characterization Project
keV	kilo electron volt; 1,000 electron volts
kg	kilogram
kPa	Kilo Pascal
LANL	Los Alamos National Laboratory
LANL-CCP	Los Alamos National Laboratory - Central Characterization Project
LBNL	Lawrence Berkley National Laboratory
Le/r	slenderness ratio

LHS	Latin hypercube sampling
LLNL	Lawrence Livermore National Laboratory
LPF	leak path factor
LWA	Land Withdrawal Act
m	meter
m^3	cubic meter
MAR	material at risk (pg. 67, App. B)
MCNP	Monte Carlo N Particle Transport Code
MeV	million electron volts
MgO	magnesium oxide
MOI	maximally exposed offsite individual
MPa	megapascal
mrem	milliroentgen equivalent man
mR	milliroentgen
M&TE	measurement and test equipment
NCT	normal conditions of transport
NDA	non-destructive assay
NDE	non-destructive examination
NRC	U.S. Nuclear Regulatory Commission
NTS	Nevada Test Site
OD	outside diameter
ORNL	Oak Ridge National Laboratory
ORNL- CCP	Oak Ridge National Laboratory - Central Characterization Project
PA	performance assessment
PABC	Performance Assessment Baseline Calculation
PACTD	Payload Assembly Container Transportation Certification Document
PAVT	Performance Assessment Verification Test
PCR	Planned Change Request
PE Ci	plutonium equivalent curies
PFP	Plutonium Finishing Plant
PNL	Pacific Northwest Laboratory
POC	pipe overpack container
QA	Quality Assurance

QAPD	Quality Assurance Program Document
QF	quality factor
R	Roentgen
RadCon	radiation control
REDC	Radiochemical Engineering Development Center
rem	roentgen equivalent man
RF	respirable fraction
RH	remote handled
RH-TRAMPAC	Remote-Handled Transuranic Waste Authorized Methods for Payload Control
SC	shielded container
SCA	shielded container assembly
SC&A	S. Cohen and Associates
SCPA	shielded container performance assessment
SI	International System of Units
SNL	Sandia National Laboratory
SRS	Savannah River Site
ST	source term
Sv	sievert
TAAC	TRU alpha activity concentration
TAER	Type A Evaluation Report
TED	total effective areas
TMU	total measurement uncertainty
TRAMPAC	Transuranic Waste Authorized Methods of Payload Control
TRU	transuranic
TRUPAC	Transuranic Package Transporter
USQ	Unreviewed Safety Question
VNC-CCP	Vallecitos Nuclear Center – Central Characterization Project
W	watt
WCO	waste certification official
WCPIP	Waste Characterization Program Implementation Plan
WDS	Waste Data System
WIPP	Waste Isolation Pilot Plant
WWIS	WIPP Waste Information System

EXECUTIVE SUMMARY

This report summarizes SC&A's review of the planned change request (PCR) submitted by the U.S. Department of Energy (DOE) to the U.S. Environmental Protection Agency (EPA) proposing the disposal of some remote-handled (RH) transuranic (TRU) waste in shielded containers on the floor of the disposal rooms at the Waste Isolation Pilot Plant (WIPP), rather than in canisters emplaced in the disposal room walls. The shielded containers have the approximate external dimensions of a 55-gallon drum, and the wall of the proposed container would consist of a lead-shielding layer sandwiched between an inner and outer layer of carbon steel. The dose rate at the surface of the shielded containers will be limited to 200 mrem/hr or less, as specified for contact-handled (CH) TRU waste in the WIPP Land Withdrawal Act, thus permitting DOE to handle the shielded containers as if they contained CH TRU waste. DOE's technical justification for the shielded container PCR was based, in part, on a special shielded container performance assessment (SCPA) designed to show that long-term performance of the repository was not significantly affected by the use of shielded containers.

The baseline selected for comparison with the SCPA was the 2004 Performance Assessment Baseline Calculation (PABC-2004), which provided the basis for EPA's first re-certification of the WIPP (as required every 5 years). At the time the PCR was submitted to EPA, the PABC-2004 was the most recently EPA-approved performance assessment (PA).[1] This baseline PA is designated as SCPA Scenario 1. Since the exact quantity of RH TRU waste that could be placed in shielded containers and meet the 200 mrem/hr surface dose rate criterion is uncertain, DOE chose a bounding approach for the SCPA in which all the RH TRU waste inventory would be emplaced on the floors of the disposal rooms. This was designated as SCPA Scenario 2. Scenario 3 assumed that half of the RH TRU waste would be placed on the disposal room floor in shielded containers and half would be in canisters placed in boreholes in the walls of the disposal rooms. DOE also examined a fourth scenario in which the 77 RH waste streams were treated individually, rather than aggregated into a single stream, which was the approach taken in Scenarios 1, 2, and 3. The mean cumulative complementary distribution functions (CCDFs)[2] for the four scenarios were virtually indistinguishable from each other.

In response to the PCR, EPA advised DOE of requirements that needed to be satisfied before the Agency would consider the PCR, including:

- NRC must approve the shipping container design
- The Department of Transportation must approve the shipping container design
- A safety analysis must be prepared by DOE

SC&A's evaluation of information provided by DOE indicates that these requirements have been adequately addressed.

SC&A conducted a detailed review of possible temperature effects in a shielded container and determined that the temperature rise within a container was small, being only a few degrees.

[1] On November 18, 2010, EPA re-certified that the WIPP remains in compliance with the requirements of 40 CFR 194 (75 **FR** 70584).

[2] CCDFs define the probability of exceeding radionuclide releases of various magnitudes.

SC&A also reviewed the changes to the WIPP baseline inventory and determined that the increases in the amounts of steel; lead; and cellulosics, plastics, and rubber (CPR) materials would increase by 42%, 83%, and 2%, respectively. SC&A also determined that inventory data were correctly integrated into the SCPA.

DOE used shielding calculations conducted with MicroShield® to estimate the quantities of RH TRU that could be considered as candidates for disposal in shielded containers. Based on the PABC-2004 inventory, about 32% of the final RH TRU waste form volume could be packaged in shielded containers, assuming that Cs-137 is the dominant photon emitter. Using an updated baseline inventory for the year ending December 31, 2007, this estimate increased to 97%. SC&A concluded that the MicroShield® analyses performed by DOE in support of the shielded container PCR are well-documented and appropriate for the intended use in scoping those RH TRU waste streams that are candidates for disposal in shielded containers.

Chemistry-related processes that could be affected by use of the shielded containers in the repository include gas generation from anoxic corrosion, gas generation caused by CPR degradation, redox conditions after repository closure, carbon dioxide (CO_2) consumption, and complexation of actinides by organic ligands. The principal effects of the shielded containers on WIPP repository chemistry would be expected to result from the large increases in the inventories of iron and lead, and small increases in cellulosics and plastics from the emplacement materials. Processes such as gas generation from anoxic corrosion of metals and microbial degradation of CPR, establishment of reducing conditions by metals corrosion and CPR degradation, CO_2 consumption by reaction with iron and lead, and competition of aqueous iron and lead species for organic ligand binding sites with actinides were evaluated. The results of this evaluation indicate that the increased iron, lead, and CPR inventories are not expected to have significant negative impacts on PA.

The approach used for the SCPA is similar to that used for previous PA calculation sets. The SCPA uses the same parameters and parameter values that were used in the PABC-2004, except for four constant parameters used only by the code CCDFGF for two of the three waste emplacement scenarios. The new parameters are derived from simple mathematical relationships that were checked for validity during this review and found to be accurately calculated and reported. The new emplacement scheme that would be used for RH waste in shielded containers also affects some of the stochastic variables used in CCDFGF calculations. For example, the probability of intruding RH waste will be affected by the new emplacement scheme. DOE revised the appropriate stochastic variables for the SCPA. The SCPA included three new sets of CCDFGF calculations using revised parameters. This review found no other changes that should have been made to CCDFGF. The run control scripts were reviewed and found to be complete. Furthermore, an independent check on the placement of selected CCDFGF simulations into Sandia National Laboratories' (SNL's) Configuration Management System (CMS) was made during an onsite visit.

A fundamental issue in implementing the use of shielded containers is the ability to determine that the surface dose rate does not exceed 200 mrem/hr. Procedures used throughout the DOE complex were reviewed and summarized. Based on this review, a list of factors that should be considered for inclusion in a complex-wide standardized procedure was proposed.

In summary, SC&A found that the design basis for the shielded container PCR was technically sound. However, important implementation questions must be addressed. For example, consideration should be given to adopting complex-wide procedures for characterizing the surface dose rate for shielded containers, and the procedures should account for measurement uncertainty to ensure that the 200 mrem/hr surface dose rate limit for CH TRU is not exceeded.

1.0 INTRODUCTION

On November 15, 2007, the U.S. Department of Energy (DOE) submitted to the U.S. Environmental Protection Agency (EPA) for approval a Planned Change Request[3] (PCR) proposing disposal of some remote-handled (RH) transuranic (TRU) waste in shielded containers on the floor of the disposal rooms at Waste Isolation Pilot Plant (WIPP), rather than in canisters emplaced in the disposal room walls as is presently done (Moody 2007). The currently approved approach for RH TRU disposal is to emplace canisters, each about 120-in long and 26-in in diameter with a 0.25-in steel shell, into holes bored into the walls of the disposal rooms (Dunagan et al. 2007). The empty canisters weigh up to 1,726 lb and can have a maximum gross weight of 8,000 lb when loaded (DOE 2006a).

In contrast, the shielded containers have the approximate external dimensions of a 55-gallon drum (23-in OD by 35.75-in high). The wall of the proposed container consists of an outer layer of carbon steel (0.125 in), a lead-shielding layer (1.0 in), and an inner carbon steel layer (0.188 in). Because various values for the thickness of inner and outer steel shells were quoted in different DOE documents, EPA questioned DOE as to the correct values. This issue has been resolved, as discussed in greater detail in Appendix A. The top and bottom of the container are 3-in thick carbon steel (Dunagan et al. 2007). Each shielded container weighs about 1,851 lb (Crawford and Taggart 2007). The weight of the package including waste contents is limited to 2,260 lb (NRC Certificate of Compliance 9279). The containers are vented to preclude internal pressure build-up. RH TRU waste in 30-gallon drums will be placed in the shielded containers.

The dose rate at the surface of the shielded containers will be limited to 200 mrem/hr or less, as specified for contact-handled (CH) TRU waste in the WIPP Land Withdrawal Act (LWA) (Public Law 102-579). This will permit DOE to process and handle the shielded containers as if they contained CH TRU waste. DOE has stated that even though some fraction of the RH TRU waste will be handled as if it were CH TRU waste, the total amount of RH TRU waste that can be disposed of in the WIPP will remain limited to 250,000 ft^3 (7,079 m^3), as specified in the Agreement for Consultation and Cooperation between DOE and the State of New Mexico (Moody 2007, DOE 1981).

A group of three shielded containers will be assembled at the waste generator site into a three-pack using plastic stretch wrap and a special triangular shipping pallet (Sellmer 2007). It was originally proposed that the three-packs would be stacked three high on the disposal room floor and a supersack of magnesium oxide (MgO) would be placed on top of the stacked containers (Crawford and Taggart 2007). Subsequently, DOE determined that the three-packs would be stacked no more than two high and no MgO supersack would be placed on top of the upper three-pack. DOE will, nevertheless, insure that sufficient MgO is present in each disposal room to insure that the required safety factor of 1.2 is met. In addition, a three-pack will not be placed on another type of container, such as a standard waste box (Moody 2010).

DOE has stated that use of shielded containers will "increase the efficiency of utilization of the WIPP facility by easing the restrictions on waste handling needed during emplacement of RH

[3] EPA Docket: II-B2-31.

waste canisters in the walls of the rooms" (Moody 2007). DOE has elaborated on the reasons for the PCR as follows (DOE 2007b):

> *The emplacement of RH TRU waste in the walls of the disposal rooms is appropriate and necessary for higher activity waste streams; however, there are several reasons why an alternative disposal method is advantageous for lower activity RH TRU waste streams. The drilling and emplacement operations for the RH canisters impede direct access to a room. This is the result of the large specialized equipment required to emplace the canisters into boreholes. Borehole drilling is limited to drilling 1 to 2 boreholes per shift. The borehole drilling equipment also restricts access to the room. The operations are time consuming; it requires one 8-hour shift to emplace a single RH TRU waste canister. A single RH waste canister evolution from receipt of the RH TRU 72B until emplacement in the wall of the underground disposal room requires more than 10 hours. WIPP is limited to a maximum of 6 RH shipments per week just from the operational constraints. In contrast, the CH waste handling processes routinely allow 4-5 shipments (i.e., 3 HalfPACTs per shipment) per day to be received, unloaded and emplaced per day. Panels 1, 2 and 3 have been filled without emplacing any RH TRU waste canisters in the walls, limiting the available wall space for emplacement of RH TRU waste. Thus, the use of shielded containers can improve the efficiency of facility operations by minimizing the disruptions from in-the-wall emplacement of RH TRU waste canisters while providing additional storage locations for some of the RH TRU waste.*

This report summarizes SC&A's technical review of the shielded container PCR. Relevant regulatory and statutory issues regarding handling and emplacement of RH TRU waste are summarized in Section 2. DOE's compliance with approval requirements set by EPA (EPA 2007) are reviewed in Section 3, while DOE's technical justification that the PCR has no significant effect on repository performance is presented in Section 4. In Section 5, SC&A describes its review of how changes to the performance assessment (PA) parameters and codes were implemented. SC&A's review of the impact of the PCR on the waste inventory is provided in Section 6, and SC&A's review of the DOE approach for characterizing which RH TRU waste streams are likely candidates for emplacement in shielded containers is included in Section 7. In Section 8, SC&A reviews the impact of the PCR on repository chemistry, and in Section 9, SC&A reviews temperature effects associated with emplacement of RH TRU waste. Measurement uncertainty is discussed in Section 10, while overall conclusions regarding the technical review are incorporated into Section 11.

2.0 REGULATORY BACKGROUND

The WIPP LWA of 1992 (PL 102-579) defines RH TRU waste as TRU waste with a surface dose rate of 200 mrem/hr or greater, and CH TRU waste as TRU waste with a surface dose rate not greater than 200 mrem/hr. The LWA specifies the following with regards to RH TRU waste destined for disposal at the WIPP:

- Total activity – 5.1 million Ci

- Maximum activity concentration (averaged over canister volume) – 23 Ci/L
- Maximum surface dose – 1,000 rem/hr
- Surface dose limit – no more than 5% can exceed 100 rem/hr

In addition, the Consultation and Cooperation Agreement between DOE and the State of New Mexico limits the disposal volume for RH TRU waste to 250,000 ft^3 (DOE 1981). (This is converted to either 7,079 m^3 or 7,080 m^3 in various DOE documents.)

The acceptance criterion for the surface dose rate requirement of 200 mrem/hr for CH waste is documented in DOE 2007a (Section 3.3.5) as follows:

> *The external radiation dose equivalent rate of individual payload containers shall be ≤ 200 milliroentgen equivalent man (mrem)/hour (hr) at the surface with the exception of the S100 and S300 pipe overpacks which are limited to ≤ 179 mrem/hr and ≤ 155 mrem/hr, respectively, at the surface (Reference 9, Section 3.2; Reference 14, Section 5.6.2). Internal payload container shielding shall not be used to meet this criterion, except for authorized shielded payload container configurations such as the use of 55-gallon drums containing a pipe component (Reference 9, Section 2.9). Total dose equivalent rate and the neutron contribution to the total dose equivalent rate shall be reported for each payload container in the WWIS.*

The bases for the surface dose rates of 179 mrem/hr for the S100 pipe overpack container (POC) and 155 mrem/hr for the S300 POC are described in the CH TRU Payload Appendices, Rev. 1, May 2005 (http://www.wipp.energy.gov/documents_ntp.htm). In each case, they are a function of an NRC prescribed limit. For the S100 container, it is determined by an external dose requirement of not more than 10 mrem/hr at 2 m from a TRUPACT II container (holding 14 S100s) subjected to free drop damage under normal conditions of transport (NCT). For the S300 container, the limiting surface dose rate is determined by a requirement of an external dose of 2 mrem/hr at 5 m from an undamaged TRUPACT II container (as might be experienced by a truck driver). A very small statistical uncertainty is assigned to each of these limiting surface dose rates, e.g., 155 +/- 0.36 mrem/hr. This is the result of the fact that the analysis is based on Monte Carlo N Particle Transport Code (MCNP) runs whose uncertainty is a function of the number of particles tracked by the code (about 10 million particles).

The CH-TRAMPAC (Contact-Handled Transuranic Waste Authorized Methods for Payload Control) specifies that surface dose rate measurements shall be made with instruments traceable to a national standard (DOE 2005, Section 3.2.2).

The DOE PCR does not impact these statutory or regulatory requirements in any way. Any RH TRU waste that is packaged in shielded containers will be counted against the 7,079 m^3 limit.

Shipments of TRU waste are governed by regulations of the U.S. Nuclear Regulatory Commission (NRC) and the U.S. Department of Transportation (DOT). Shipments of RH TRU waste to WIPP are currently made in RH TRU waste canisters (payload containers) confined within an RH TRU 72-B cask. The RH-TRAMPAC (Remote-Handled Transuranic Waste

Authorized Methods for Payload Control) establishes numerous criteria that the waste must meet when shipped to WIPP. Examples include nuclear criticality requirements, limits on quantities of radionuclides that could be released in a hypothetical accident, and limits on heat from radioactive decay (DOE 2006a). The CH-TRAMPAC establishes companion requirements for CH TRU waste (DOE 2005).

DOE has demonstrated that the shielded containers meet DOT Type 7A packaging and NRC Type B drop tests, and has obtained approval from NRC for HalfPACT shipping containers[4] to transport shielded containers filled with RH TRU waste to WIPP. DOE proposes to ship three shielded containers packaged as a unit in a HalfPACT. EPA advised DOE on December 7, 2007, that the Agency was not prepared to make a final decision on the shielded container PCR until NRC and DOT approvals had been obtained for the HalfPACT (Reyes 2007). As discussed in greater detail in Section 3, these approvals have been obtained.

3.0 EPA REQUIREMENTS FOR PCR CONSIDERATION

After its preliminary review, EPA advised DOE of requirements that needed to be satisfied before the Agency would consider the PCR. These requirements were documented in a letter to DOE dated December 7, 2007 (Reyes 2007) as follows:

- NRC must approve the shipping container design
- The Department of Transportation must approve the shipping container design
- A safety analysis must be prepared by DOE

3.1 NRC Approval

On June 10, 2009, DOE provided EPA with NRC Certificate of Compliance (CoC) No. 9279 that authorized the inclusion of shielded containers in the HalfPACT waste shipping container. NRC required no changes in the design or materials of construction (DOE 2009a). Shielded containers were added to the existing CoC as Revision 5 that was approved by the NRC on May 15, 2009. [5]

Specific changes to the CoC to include shielded containers are as follows:

- 5(a)3 – Drawings – The shielded container is constructed in accordance with Packaging Technology, Inc., Drawing No. 163-008, sheets 1–6, Rev. 1.
- b(2) – Maximum quantity of material per package – (x) 2,260 pounds per shielded container
- 5(b)2 – Maximum number of payload containers per package and authorized packaging configurations – (ix) 3 shielded containers

[4] This cask is currently approved and used for the shipment of CH TRU waste.

[5] Revision 6 to the CoC was approved by the NRC on October 21, 2010 and extends the expiration date to October 31, 2015 "To renew a CoC, a CoC holder at the end of the 5-year approval period would submit a request to NRC with any necessary supporting information describing the capability of the package design to continue to meet technical requirements. After reviewing this information, the NRC would determine whether to grant a CoC renewal." http://www.nrc.gov/materials/transportation/certification.html#4

These changes do not constitute substantive revisions to the CoC.

3.2 Self-Certification by DOE of Department of Transportation Requirements

DOE's WIPP Waste Acceptance Criteria require that containers meet the requirements of DOT 7A packaging (DOE 2009b). In a January 21, 2009, letter to EPA (DOE 2009c), DOE documented that since the shielded container was a Type A package as defined by DOT, approval by DOT is not required. Rather, DOE can self-certify that the shielded container meets the requirements of 49 CFR 173.410 and 173.412. The shielded container must also meet tests prescribed in 49 CFR 173.465 or 173.466 and the requirements set out in 49 CFR 179.350.

DOE described studies conducted to demonstrate that shielded containers comply with Class A requirements in the June 2008 Type A Evaluation Report (TAER) (WTS 2008). The TAER identifies the analyses, tests, and evaluations performed on the shielded container to demonstrate compliance of the packaging design with the applicable requirements of 49 CFR 178.350.

DOE subjected two test containers to free drop tests from 4 ft above a rigid flat surface, as specified in 49 CFR 173.465. Prior to being dropped, the containers were oriented in positions that should cause maximum damage. Each container was dropped twice. During testing, no significant damage to the packages was sustained. After each drop test, examination of the containers showed that no release of the simulated contents had occurred. Containment of the radiological contents is the criterion for passing the drop test.

The two test containers were subjected to gamma scans to determine if the effectiveness of the shielding had been reduced by the drop tests, in order to inform shippers of potential changes to dose rate as the result of an accident. With the exception of two anomalous readings on one container (one location indicating approximately a 20% increase in shielding effectiveness and a second indicating approximately a 20% decrease in shielding effectiveness), the straight-line through-wall gamma scan readings in the body of the container demonstrated less than 20% change in shielding effectiveness as a result of the drop tests. Sectioning of the containers after testing revealed no indications of lead slump or lead movement as a result of the drop tests. The anomalous shielding readings were found to have resulted from imperfections in the casting of the prototypes.

In lieu of the physical stacking test required under § 173.465(d), DOE performed an analytical calculation assuming that six containers were stacked on top of the target container, thereby increasing the axial load by 20% above that specified in § 173.465(d). This axial load of 13,560 lbs was assumed to be borne solely by the inner cylindrical shell of the shielded container. Using procedures documented in ASME Boiler and Pressure Vessel Code Case N-284-1, DOE calculated that the axial load would not cause buckling of the inner shell.

Section 173.465(e) specifies a physical penetration test in which a 1.25-in diameter bar weighing 13.2 lb is dropped from a height of 3.3 ft or more onto the shielded container. In lieu of the physical test, DOE performed a calculation showing that the outer shell must be 0.015 in or less to be penetrated by the test bar. Since the thickness of the outer shell is 0.12 in, an ample margin of safety exists.

With regard to vibration testing, DOE asserts that because of the robust nature of the shielded container, it would meet the vibration testing requirement of 49 CFR 178.608. This requirement specifies that the waste package be tested on a vibrating table for 1 hour at a frequency that causes the shielded container to be raised above the table by about 0.063 in. DOE did not subject a shielded container to vibration testing and, instead, based its compliance opinion on the following reasoning (WTS 2008, Section 4.2.5):

> *The stiffness of the 3 in. thick lid and base, and the greater than 1 in. thick steel/lead/steel body sidewall is such that resonant frequencies would not be encountered during normal condition transport. The 15 closure bolts, when preloaded to the torque requirements referenced herein, would not loosen or otherwise be significantly affected by vibration conditions. Other miscellaneous components are either welded, press-fit, or otherwise secured in place and not significantly affected by vibration conditions and/or not critical components serving a containment or shielding function in the package.*

3.2.1 EPA Comments on Self-Certification Documentation

Based on its review of the self-certification testing described in Section 3.1 above, EPA submitted several comments to DOE for clarification/resolution. EPA's comments and DOE's responses are provided below (Patterson 2010):

EPA Comment 1

DOE described studies conducted to demonstrate that shielded containers comply with Class A requirements in the June 2008 Type A Evaluation Report (TAER) (WTS 2008). The TAER identifies the analyses, tests, and evaluations performed on the shielded container to demonstrate compliance of the packaging design with the applicable requirements of 49 CFR 178.350. According to Section 3.1 of the TAER:

> *Determination of the response of a point radiation source subject to movement within the package and any associated effects on radiation levels are not provided in this document. Drop test damage information is provided in Section 4.2.2 for use by the shipper in determining whether a significant change in radiation level would result for a specific payload.*

This raises the question as to how the shipper would determine whether a significant change in radiation could result for a specific payload. We presume that, if no significant damage occurs during drop testing and no tracers or other materials are released, significant changes in radiation levels would not result. If this is the intent, why is it not so stated? What is the burden imposed on the shipper? The TAER would benefit if this ambiguous statement were clarified.

This excerpt also raises the question as to where movement of point sources of radiation is addressed and what actions must be taken to prevent such movement.

Similarly, in Section 3.2, the authors note that:

> *Damage information is provided to assist the shipper in evaluating the possible dose rate changes at the surface of the package for the intended payloads to be shipped.*

The same concerns as outlined above apply here.

EPA Reference

WTS 2008. *Shielded Container Type A Evaluation Report*, ECO No. 11834, Rev. 0, Washington TRU Solutions LLC, Carlsbad, New Mexico, June 2008.

DOE Response

Changes in radiation levels at the surface of the container are dependent on two factors during transportation. One is damage to the packaging, which is addressed in the SCA (Shielded Container Assembly) TAER with testing performed before and after the 4-foot drop tests, and discussed in Section 4.2.2.4, Shielding. There was no significant reduction of shielding effectiveness as a result of the drop tests. The other factor is the location of a point gamma radiation source within the SCA. It is the responsibility of the shipper to ensure that there is adequate bracing within the 30-gallon internal payload container such that the point radiation source doesn't move during transportation to cause a significant increase (20%) in the external radiation levels. This is addressed in Sections 2.4, 5.1, and Appendix A, Section 4.6.3. Specific loading instructions are not addressed in the SCA TAER, as that is not the intent of the document. The SCA TAER references the SCA Handling and Operation Manual, WP 08-PT.16, which does provide specific loading instructions, in Sections 2.4, on page 4-11, and in Section 5.1. WP 08-PT.16 is also referenced on WTS drawing, 165-F-026. These references are provided to inform the shipper that the SCA must be loaded and closed in accordance with those specific instructions in order for a loaded SCA to be certified as a Type A Packaging. While changes to the SCA TAER are not deemed necessary, the Handling and Operation Manual will be revised to further instruct the shipper to securely fasten and position contents within the 30-gallon internal payload container in a manner to prevent a significant increase in the level of radiation at the external surface of the SCA as a result of movement during transport.

DOE References

SCA Handling and Operation Manual, WP 08-PT.16
WTS drawing, 165-F-026

EPA Comment 2

With regard to vibration testing, DOE asserts that because of the robust nature of the shielded container, it would meet the vibration testing requirement of 49 CFR 178.608. This requirement specifies that the waste package be tested on a vibrating table for 1 hour at a frequency that causes the shielded container to be raised above the table by about 0.063 in. DOE did not subject a shielded container to vibration testing and, instead, based its compliance opinion on the following reasoning (WTS 2008, Section 4.2.5):

> *The stiffness of the 3 in. thick lid and base, and the greater than 1 in. thick steel/lead/steel body sidewall is such that resonant frequencies would not be encountered during normal condition transport. The 15 closure bolts, when preloaded to the torque requirements referenced herein, would not loosen or otherwise be significantly affected by vibration conditions. Other miscellaneous components are either welded, press-fit, or otherwise secured in place and not significantly affected by vibration conditions and/or not critical components serving a containment or shielding function in the package.*

This quotation contains several unsupported statements including:

- Resonant frequencies would not be generated
- Properly torqued bolts would not be loosened by vibration
- Press fit items would not be significantly affected by vibration

DOE needs to provide the technical basis for these suppositions.

DOE Response

(1) Generally a resonant frequency of 500Hz or more is considered inconsequential to normal transport conditions. A simplified calculation as well as a finite element analysis (FEA) model both show that the lid has a resonant frequency of approximately 1,300Hz, which is therefore out of this range.

(2) A calculation of the worst-case clamping load of the 15 closure bolts shows a force of 88,000 lbs. This corresponds to 262 Gs of acceleration required to lift the lid.

(3) The only two press fit items are protective plugs used to keep water from collecting in two areas, the filter port and the threaded lift interface holes. During transport, the SCA is required to have a DOT Type A compliant filter vent installed; therefore, the filter port plug is not used and is of no concern. The threaded lift interface holes are still plugged during transport, but only to prevent the collection of water or debris, since they are not part of the containment boundary (reference WTS drawing, 165-F-026-W1, see Section C-C). It should also be noted that for our intended use of the SCA, inside of a Type B package during transport, the container will not be exposed to the elements either.

DOE Reference

WTS drawing, 165-F-026

EPA Comment 3

In lieu of the physical stacking test required under § 173.465(d), DOE performed an analytical calculation assuming that six containers were stacked on top of the target container, thereby increasing the axial load by 20% above that specified in § 173.465(d). This axial load of 13,560 lbs was assumed to be borne solely by the inner cylindrical shell of the shielded container. Using procedures documented in ASME Boiler and Pressure Vessel Code Case N-284-1, DOE calculated that the axial load would not cause buckling of the inner shell. However, NRC's current position on Code Case N-284-1 (NRC 2007) is that use of this case by licensees to evaluate canisters and transportation casks is permissible only if it has been reviewed and approved by NRC. Given NRC's concerns about the Case, DOE should demonstrate that the errata, misprints, recommendations, and errors identified in NRC 2007 do not unfavorably affect the TAER calculations.

Reference

NRC (U.S. Nuclear Regulatory Commission) 2007. *ASME Code Cases Not Approved for Use*. Regulatory Guide 1.193, Rev. 2. October 2007.

DOE Response

When analyzed by calculating the slenderness ratio (Le/r) of the geometry, the SCA has a ratio of 9.36, which is well below the limit for a geometry to be considered short and wide (Le/r<30). Therefore this geometry would be bounded by a normal axial stress calculation, which shows a safety factor to yield when loaded to 13,560 lbs of 19:1 against the outer shell, which has the smaller cross sectional area of the two shells. Therefore it is more than reasonable to assume that the design of the SCA is not affected by the regulatory stacking test by a large margin.

It should also be noted that NRC Regulatory Guide 1.193 is specific to 10 CFR 50, which governs NRC-licensed "Nuclear Power Plants." Also, in the HalfPACT SAR, Section 2.6.7, ASME Code Case N-284 is specifically used and cited for the buckling calculations of the containment boundary of the HalfPACT, which is an NRC-approved and licensed Type B package.

3.2.2 DOE Self-Certification Conclusions

Based on review of the DOE self-certification activities and responses to EPA comments, SC&A concludes that DOE has met the self-certification requirements for SCAs.

3.3 Safety Analysis

Subsequent to EPA's letter of December 7, 2007 (EPA 2007), discussions were held between the Agency and DOE to refine the safety analysis requirements. Based on these discussions, it was agreed that "the results of typical Unreviewed Safety Question (USQ) analysis, written such that a technical reviewer without a nuclear safety background could more easily understand the safety analysis, will adequately address their [EPA's] concerns" (DOE 2010a). Based on this understanding, DOE provided four documents as being equivalent to the technical content of a typical USQ analysis:

- *Summary of Nuclear Criticality Safety Evaluation for Shielded Containers at the WIPP*, WIPP-025 (DOE 2010a, Enclosure 3).

- *WIPP Accident Analysis Calculations for Events Involving Releases from the Gamma Shielded Container*, WIPP-031 (DOE 2010a, Enclosure 4).
- *Fire Analysis of the Shielded Container for the Waste Isolation Pilot Plant, Carlsbad, New Mexico*, WIPP-032 (DOE 2010a, Enclosure 2).
- A white paper entitled: *Summary of the Safety Impact Analysis for the Lead Shielded Container* (DOE 2010a, Enclosure 1). This paper summarizes the results from WIPP-025, WIPP-031, and WIPP-032.

Based on concerns that the original white paper did not adequately cover loss of lead shielding from a fire, DOE provided *Summary of a Revised Safety Impact Analysis for the Lead Shielded Container Assembly* in December 2010 (DOE 2010b).

The safety analysis determined whether use of shielded containers would result in any changes to the radiation dose received by certain onsite workers or the maximally exposed offsite individual in various accident scenarios. The basis of comparison was the WIPP Documented Safety Analysis (DSA) that considered a variety of waste containers. The WIPP DSA evaluated fires, explosions, loss of confinement, direct radiation exposures, criticality, and externally initiated and natural phenomena. From these scenarios, DOE determined those that were applicable to shielded containers.

Twenty-seven fire-related scenarios were determined to be applicable to shielded containers. Of these, DOE analyzed the worst-case fire scenarios and found that the worst-case scenarios for shielded containers had no greater consequences than scenarios already evaluated in the DSA. Since this was the case, DOE concluded that it was not necessary to evaluate the other selected fire scenarios, since the worst-case scenarios were bounding. The bounding scenario was a fire resulting in the loss of both the lead shielding and of the SCA's ability to physically contain waste. In its accident analysis, DOE calculated the radiation exposure of both the maximally exposed offsite individual (MOI) and an onsite worker, using the same process employed for other waste containers. It should be noted that direct gamma radiation exposure resulting from loss of lead shielding was not evaluated (DOE 2010a, Enclosure 2, Section 7.1). Because any fire would force workers to leave the immediate area, and because of the greater dose impact of inhaled radionuclides, the *Preparation Guide for U.S Department of Energy Nonreactor Nuclear Facility Documented Safety Analyses* (DOE 2006e, Appendix A, Section A.3) identifies the airborne pathway as being of primary concern for nonreactor nuclear facility safety. The DOE analysis of fire accidents is considered in detail in Appendix B of this document.

Four criticality scenarios were considered for shielded containers. DOE determined that the possibility of nuclear criticality being achieved was incredible (i.e., had a probability of $<10^{-6}$/yr of occurring). DOE also examined the probability of radiation exposure from surface contamination or from direct radiation and determined that the frequency of occurrence of such events was no greater for shielded containers than had been determined for other containers in the DSA. The shielded containers will have the same limits for Fissile Gram Equivalent (FGE) mass and Plutonium-Equivalent Curies as a 55-gallon drum containing CH TRU waste. These limits are set by the HalfPACT Safety Analysis Report (DOE-CBFO 2008). Thus, there is no increase in the quantity of transuranic radionuclides that could be released from a disruptive physical event.

In summary, DOE concluded that the handling and emplacement of shielded containers does not increase the probability or consequences of any event considered in the DSA. Based on the review presented here, SC&A agrees with this conclusion.

4.0 DOE JUSTIFICATION FOR USE OF SHIELDED CONTAINERS

DOE's technical justification for the shielded container PCR was based, in part, on a special shielded container performance assessment (SCPA) designed to show that long-term performance of the repository was not significantly affected by the use of shielded containers (Dunagan et al. 2007). The SCPA involved five sequential steps, as quoted in the referenced document (Dunagan et al. 2007):

(1) Evaluate WIPP PA baseline assumptions, models, and parameters to determine which are affected by the use of shielded containers

(2) Develop an analysis design to incorporate necessary modifications to the baseline approach

(3) Develop necessary parameters for the SCPA

(4) Execute WIPP PA codes

(5) Conduct an analysis of results, including a comparison with baseline predictions of long-term repository performance

4.1 Evaluation of Impact of Shielded Containers on WIPP Baseline PA Assumptions

In Step 1 of the SCPA, DOE considered nine components of the WIPP PA that might be affected by the PCR. DOE's list of components and their possible impact on PA is reproduced here as Table 3-1 (Dunagan et al. 2007, Table 6). Based on a detailed discussion of each item, DOE developed the SCPA modeling approach summarized in Table 4-2 (Dunagan et al. 2007, Table 8). EPA has reviewed the basis for the modeling decisions and is in general agreement with the technical arguments presented by DOE. Further discussion of inventory changes is presented in Section 6 of this report, while a detailed discussion of chemical conditions is presented in Section 8. Codes used for the SCPA modeling are discussed in Section 5.

Table 4-1: Components of WIPP PA that could Potentially be Affected by Shielded Containers

Component	Possible Implementation Issue
Contents of waste materials	Will the use of shielded containers affect the contents of waste materials and waste material mechanical properties?
Emplacement and container materials	Will the use of shielded containers affect the amount of steel and CPR materials associated with emplacement and container materials?
Room closure	Will the use of shielded containers affect room closure and the porosity of the waste areas?
Chemical conditions	Will the presence of lead in the shielded containers affect chemical

Table 4-1: Components of WIPP PA that could Potentially be Affected by Shielded Containers

Component	Possible Implementation Issue
	conditions and actinide solubilities?
Waste emplacement	Will loading schemes and disposal schedules associated with the shielded containers present inconsistencies with the assumption of random waste emplacement?
Repository temperature	Will the use of shielded containers affect the repository temperature and heat distribution?
Impact of waste location on release mechanisms	Will emplacement RH waste on the floor of disposal rooms make it more/less accessible to release mechanisms?
Impact of shielded container properties on release mechanisms	Will the physical characteristics of the shielded containers affect release mechanisms?
Location of RH waste streams	Will the location of RH waste streams, on the floor versus in the walls, affect normalized release from the repository?

Table 4-2: SCPA Approach to Modeling Issues Listed in Table 4-1

Component	Possible Implementation Issue
Contents of waste materials	DOE does not propose to modify the waste materials that will be emplaced in the WIPP, so the SCPA will use the waste inventory and mechanical parameters from the CRA-2004 PABC, since they will not be affected by the use of shielded containers.
Emplacement and container materials	The SCPA will use the CRA-2004 PABC emplacement parameters for steel and CPR materials, since the small change to these quantities caused by the use of shielded containers will not significantly affect repository performance.
Room closure	The SCPA will use the CRA-2004 PABC porosity surfaces that were calculated with the Standard Waste Model, since repository performance is relatively insensitive to the structural rigidity of waste and waste containers.
Chemical conditions	Because the presence of lead is expected to have a generally beneficial effect on chemical conditions and decrease actinide solubilities, the SCPA will conservatively use the CRA-2004 PABC actinide solubilities that were calculated without explicitly including the effect of lead.
Heterogeneity of waste emplacement	The SCPA will assume that the stacks of CH and shielded containers are randomly distributed, since mean releases are insensitive to uncertainty in the spatial arrangement of the waste.
Repository temperature	The use of shielded containers does not affect any baseline assumptions pertaining to repository temperatures, so the SCPA will not make any modifications to these baseline assumptions.
Impact of waste location on release mechanisms	The SCPA will assume that RH waste in shielded containers is accessible to all release mechanisms.
Impact of shielded container properties on release mechanisms	The SCPA will follow the baseline approach of assuming that all waste containers instantly fail, so the SCPA conservatively will not take credit for the physical properties of the shielded containers.
Location of RH waste streams	The SCPA will model RH emplacement with three scenarios: (1) All RH waste in the walls (2) All RH waste on the floor (3) Half of the RH waste in the wall and half of the RH waste on the floor These three scenarios will use a single, “average” RH waste stream.

Table 4-2: SCPA Approach to Modeling Issues Listed in Table 4-1

Component	Possible Implementation Issue
	One additional calculation will model all 77 individual RH waste streams on the floor.

4.2 Modeling Scenarios

The baseline selected for comparison with the SCPA is the Performance Assessment Baseline Calculation 2004 (PABC-2004), which was the most recent EPA-approved PA at the time the PCR was submitted. The PABC-2004 served as the basis for EPA's recertification of the WIPP in 2006. The PABC-2004 results demonstrated that RH TRU waste, when emplaced in the walls of the disposal rooms, is a minor contributor to possible releases from the WIPP repository. For example, there is a 10% probability that total normalized releases from all waste will exceed 8.86E-02 EPA units, while there is also a 10% probability that normalized releases from RH TRU waste will exceed 3.55E-04 EPA units (Dunagan et al. 2007, Table 5). Thus the impact of RH TRU waste on PA is more than two orders of magnitude less than that of CH TRU waste when RH waste is emplaced in the room walls. The PABC-2004 baseline is designated SCPA Scenario 1 by DOE.

Since PABC-2004 has been issued, two additional major PAs have been completed by DOE—the 2009 Compliance Recertification Application PA (CRA-2009) and the Agency-mandated PABC-2009. EPA examined the documentation related to these two PAs as part of its recertification review and concluded in a November 18, 2010, Federal Register notice that the WIPP remained in compliance with the regulatory requirements. The mean CCDF developed for CRA-2009 shows a very slight increase in the mean normalized releases as compared to the PABC-2004 (DOE 2009d, Appendix PA, Figure PA-84). The CCDFs for PABC-2009 were marginally different from those for the CRA-2009, with total normalized releases being slightly lower at high probabilities and higher at low probabilities (Camphouse 2010). Table 4-3 compares the mean normalized releases from the three PAs:

Table 4-3. Mean Total Releases from Various Performance Assessments

Probability	Performance Assessment	Mean Total Release (EPA Units)	Release Limit (EPA Units)
0.1	PABC-2004	0.09	1
0.1	CRA-2009 PA	0.1	1
0.1	PABC-2009	0.09	1
0.001	PABC-2004	0.60	10
0.001	CRA-2009 PA	0.72	10
0.001	PABC-2009	1.10	10

Thus, it does not appear that more recent modeling renders the use of PABC-2004 as the shielded container analysis baseline as inappropriate.

Since the exact quantity of RH TRU waste that could be placed in shielded containers and meet the 200 mrem/hr surface dose rate criterion is uncertain, DOE chose a bounding approach for the SCPA in which all the RH TRU waste would be emplaced on the floors of the disposal rooms. This was designated as SCPA Scenario 2. SCPA Scenario 3 assumed that half of the RH TRU waste would be placed on the disposal room floor in shielded containers and half would be in canisters placed in boreholes in the walls of the disposal rooms.

DOE compared the mean CCDFs for Scenarios 2 and 3 against the PABC-2004 Baseline (Scenario 1). The comparisons were made based on mean total releases and mean releases from cuttings and cavings, direct brine releases, and spallings releases separately. Differences among the three scenarios were essentially indistinguishable, based both on consideration of specific release mechanisms and total releases (Dunagan et al. 2007, Figures 8, 9, 10, and 11). Although the differences were very small, direct brine releases from Scenario 2 were slightly less than similar releases calculated in the PABC-2004. DOE attributed this slight difference to the fact that the conditional probability of a drilling intrusion hitting an excavated area, given that the intrusion was within the repository berm area, was less for Scenario 2, because the excavated area was smaller (i.e., less RH area in the walls of the disposal rooms).

In Scenarios 1, 2, and 3, DOE assumed that all of the RH TRU waste was combined into a single composite waste stream. To test the significance of this assumption, DOE implemented Scenario 4 that was similar to Scenario 2, except that all 77 waste RH TRU waste streams were treated individually. As in the PABC 2004, waste was assumed to be emplaced in a random, heterogeneous manner, because PA release mechanisms are insensitive to the spatial arrangement of emplaced waste (Hansen at al. 2004). DOE showed that there were no differences between the mean total releases for Scenarios 2 and 4 (Dunagan et al. 2007, Figure 12).

5.0 REVIEW OF SCPA CALCULATIONAL PROCEDURES

The previous section described DOE's general approach to modeling the impacts of shielded containers on repository performance. In this section, we take a detailed look at the parameters used; the computer codes used, including their qualification; and the results obtained. The SCPA takes a bounding approach to assessing the impact of using shielded containers for the disposal of RH waste. The results of PABC-2004 (Leigh et al. 2005) represent performance of the repository with all of the RH waste emplaced in the walls of the repository. The additional set of calculations that were performed for the SCPA assume that some fraction of the RH waste is placed on the disposal room floors in shielded containers and the remaining fraction is placed in the walls of the disposal rooms. To isolate the impact of the shielded containers on repository performance, the SCPA is designed to deviate as little as possible from the PABC-2004.

The SCPA analysis is intended to answer the following question:

- How is long-term repository performance affected by emplacement of RH waste in shielded containers?

As discussed in Section 4.1, the only baseline parameters and assumptions that DOE modified to represent the emplacement of RH waste in shielded containers were those parameters and assumptions related to the location of RH waste. No other parameters or assumptions from the PABC-2004 were modified. Consequently, the only PA codes that DOE activated for the SCPA were those codes related to the execution of CCDFGF and EPAUNI. The CCDFGF code determines the consequences of the releases for the various scenarios, and EPAUNI calculates radionuclide decay. These codes were rerun as necessary for the SCPA scenarios described in Section 4.2.

The approach used for the SCPA is similar to that used for previous PA calculation sets. The SCPA begins with an analysis of the features, events, and processes (FEPs) that may or may not have a bearing on the performance of the repository. As discussed in Section 4.1, the use of shielded containers does not have any major impact on PA implementation, and no FEPs are affected. The "retained" FEPs are formulated into scenarios that are modeled. Scenarios are modeled using conceptual models that represent the physical and chemical processes of the repository. The conceptual models are implemented through a series of computer simulations and associated parameters that describe the natural and engineered components of the disposal system (e.g., site characteristics, waste forms, waste quantities, and engineered features). The computer simulations are developed from conceptual models. The results of the simulations quantify the magnitude and probability of potential releases of radioactive materials from the disposal system to the accessible environment over the 10,000-year regulatory period.

The following subsections detail the approach that the SCPA used.

5.1 Parameters

The SCPA uses the same parameters and parameter values that were used in the PABC-2004, except for a few constant parameters that are used only by the code CCDFGF for two of the three waste emplacement scenarios, as follows:

- Scenario 2 – Placing all of the RH waste with the CH waste in the disposal area
- Scenario 3 – Placing half of the RH waste with the CH waste in the disposal area, while half of the RH waste remains in the walls

To evaluate these scenarios, the parameters that were changed are REFCON:FVW and REFCON:AREA_RH (Dunagan 2007).

The parameter REFCON:FVW represents the fraction of the repository volume occupied by waste in the CCDFGF. In the PABC-2004, REFCON:FVW is set to 0.385 and is unitless (Kirchner 2007a).

REFCON:FVW was calculated for the PABC-2004 by dividing the total volume of CH wastes, REPOSIT:VOLCHW, by the excavated storage volume of the repository, REFCON:VREPOS. Because the repository volume occupied by the waste will be different from that assumed in the PABC-2004, DOE developed four new parameters to simulate the two scenarios described above.

Two of the four new parameters represent the fraction of the repository volume occupied by waste. One of the new parameters, REFCON:FVW_ALLRH, represents the repository volume occupied by waste with all of the RH waste inventory volume included with the total CH waste volume on the repository floor. This parameter is calculated in a manner similar to

REFCON:FVW, but with the inclusion of the total volume of RH wastes, and has a value of 0.402.

The second of the new parameters represents the repository volume occupied when half of the RH waste inventory volume is included with the total CH waste volume. This parameter is calculated in a manner similar to REFCON:FVW, but with the inclusion of half of the value of parameter REPOSIT:VOLRHW. This parameter, REFCON:FVW_HALFRH, has a value of 0.394.

The two remaining new parameters represent that area of RH waste disposal in the model CCDFGF called REFCON:AREA_RH in the PABC-2004 simulations. The value for this parameter in the PABC-2004 is 1.576×10^4 m^2 and is representative of all of the RH wastes in the wall of the repository (Helton 1996). This value is calculated by multiplying the expected number of canisters of RH wastes in the repository (8,000) by the footprint of an RH waste canister, 1.97 m^2, to determine the area of RH waste disposal (Helton 1996).

The fourth parameter represents the area of RH waste disposal when all of the RH waste is included with the CH waste on the repository floors. This parameter is called REFCON:AREA_NORH and has a value of 0.

Table 5-1 lists the new parameters that were created for the SCPA and their values. These parameters were used in the place of REFCON:FVW and REFCON:AREA_RH, since REFCON:FVW and REFCON:AREA_RH were calculated assuming that all RH waste would be placed in the walls.

Table 5-1: Parameters Created for the SCPA

Material	Property	Value (Units)	Description
REFCON	FVW_ALLRH	0.402 (none)	Fraction of repository volume occupied by CH and RH waste in CCDFGF model (Scenario 2).
REFCON	FVW_HALFRH	0.394 (none)	Fraction of repository volume occupied by CH waste and half of total RH waste in CCDFGF model (Scenario 3).
REFCON	AREA_NORH	0 (m^2)	Area for RH waste disposal in CCDFGF model when all RH waste is included with CH waste on repository floors (Scenario 2).
REFCON	AREA_HAFRH	7.880E+03 (m^2)	Area for RH waste disposal in CCDFGF model when half of total RH waste is included with CH waste on repository floors (Scenario 3).

DOE presents the derivations of the new parameters in Dunagan (2007). The parameters are derived from simple mathematical relationships that were checked for validity during this review and found to be accurately calculated and reported.

5.2 PA Computer Codes

5.2.1 Latin Hypercube Parameter Sampling

The LHS computer code is used to sample those PA parameters which are represented by distributions in order to account for subjective uncertainty. Three replicates of 100 vectors each are used for the SCPA, as well as the identical random seed and parameter ordering from the PABC-2004 Latin hypercube sampling (LHS) calculations. Use of these random seeds and orderings results in identical sampled parameter values for parameters that are common to both the PABC-2004 and SCPA. Consequently, results from the SCPA can be compared with those from the PABC-2004 on a vector-by-vector basis.

The SCPA uses the same sampled parameters that were used in the PABC-2004, and the SCPA does not introduce any new parameters that need to be sampled. Thus, LHS was not re-run, and the SCPA uses the results from the PABC-2004 LHS calculations. These results are documented in Kirchner (2005).

DOE's approach of keeping the random seed identical to the PABC-2004 LHS calculations allows meaningful comparisons to be made between the resulting vectors. Otherwise, it would be difficult to differentiate between changes due to parameters sampled independently from those associated with the shielded container scenarios.

LHS Version 2.41 was used to support the SCPA and PABC-2004. In January 2005, the LHS code was revised to Version 2.42 in order to describe the normal, lognormal, student, and logstudent distributions. These revisions to the code would not have impacted the SCPA results obtained with LHS 2.41.

5.2.2 Inventory: EPAUNI

Leigh et al. (2005) gives a comprehensive description of the inventory that was used for the PABC-2004. Modifications to this inventory to account for the use of shielded containers are described in Section 6.

For the PABC-2004, the code EPAUNI is used to calculate waste stream activities at a set of times (Fox 2005), and the results of these calculations are input to the CCDFGF code. The code is run once for CH waste streams and once for RH waste streams. The SCPA required three new sets of EPAUNI calculations, as follows:

(1) For Scenario 2, all waste streams (CH and the "average" RH) are included in a single EPAUNI calculation

(2) For Scenario 3, all CH waste streams and half of the "average" RH waste stream are included in a single EPAUNI calculation

(3) For Scenario 4, the 77 individual RH waste streams are included in a single EPAUNI calculation

This review found that there are no other changes that should have been made to EPAUNI. The run control scripts in Appendix B of Dunagan et al. 2007 for EPAUNI were reviewed and found to be complete. Furthermore, an independent check on the placement of selected EPAUNI simulations into SNL's Configuration Management System (CMS) was made during an onsite visit on August 6, 2008.

5.2.3 Salado Flow: BRAGFLO

The two-phase flow code BRAGFLO simulates the brine and gas flow in and around the WIPP repository and incorporates the effects of disposal room consolidation and closure, gas generation, brine consumption, and inter-bed fracturing in response to gas pressure. The SCPA uses the PABC-2004 waste material parameters. In addition, the SCPA BRAGFLO calculations use the baseline porosity surface calculations from SANTOS. Thus, the presence of the shielded containers will not affect PABC-2004 BRAGLFO calculations. The SCPA uses PABC-2004 BRAGFLO results documented by Nemer and Stein (2005).

The new waste configuration may result in a baseline porosity surface that is different than that calculated with SANTOS and input into BRAGFLO for the PABC-2004. The impacts of waste containers with a higher rigidity than that of the Standard Waste Model used in the CCA was considered by both the Agency and DOE during the approval of supercompacted waste. Park and Hansen (2003) considered the impacts of incompressible containers on the results of PA. The rigidity of the containers was found to diminish room closure, resulting in higher porosities. These higher porosities would cause lower room pressures which, in turn, would result in smaller release volumes. Hansen et al. (2004) showed that the results of PA are relatively insensitive to container rigidity, and that the Standard Waste model results in the highest repository pressures and releases.

Therefore, although the BRAGFLO result may have been different if the rigidity of the shielded containers was reflected in the model, using the PABC-2004 approach for the SPCA would lead to conservative results.

BRAGFLO Version 5.0 was used to support the SCPA and PABC-2004. In 2007, a number of changes were made to BRAGFLO from Version 5.00 to Version 6.00 in accordance with a paper by Hansen and Stein (2005), which describes changes that should be made to PA models to accommodate a more realistic evolution of the WIPP underground. The changes made to BRAGFLO included changes to the disturbed rock zone, brine availability, MgO precipitation, room closure, and formulations pertaining to the capillary pressure versus saturation, which impact the physical and chemical characteristics of the WIPP disposal rooms. These revisions to the code would not fundamentally change the SPCA results obtained with BRAGFLO 5.0.

5.2.4 Actinide Mobilization: PANEL

Actinides mobilize in WIPP brines as dissolved species and species sorbed onto colloids as calculated with the PANEL code. The SCPA uses the PABC-2004 actinide solubilities, so the presence of the shielded containers does not affect actinide mobilization calculations by PANEL from the PABC-2004. DOE uses the PABC-2004 PANEL results, documented in Garner and Leigh 2005, for the SCPA.

PANEL Version 4.02 was used to support the SCPA and PABC-2004. In April 2005, PANEL was revised to Version 4.03 to allow setting the default panel brine volume via MATSET. These revisions to the code would not have impacted the SCPA results obtained with PANEL Version 4.02.

5.2.5 Salado Transport: NUTS and PANEL

The WIPP radioisotope mobilization and decay code NUTS is used to simulate the transport of radionuclides through the Salado Formation. DOE maintains that the use of shielded containers will not affect PABC-2004 NUTS results, since the SCPA uses the PABC-2004 BRAGFLO results. The SCPA uses the PABC-2004 NUTS results, as documented in Lowry 2005.

The DOE approach is reasonable, and no changes to NUTS for the SCPA appear to be warranted.

NUTS Version 2.05A was used to support the SCPA and PABC-2004. In 2007, NUTS 2.05A was revised to NUTS 2.05C, where the format of the date and time was revised in order to avoid run-time errors, because the time argument of DATE_AND_TIME was too short. These revisions to the code would not have impacted the SCPA results obtained with NUTS 2.05A.

Radionuclide transport to the Culebra for the E1E2 intrusion scenario (BRAGFLO scenario S6) is calculated by running the PANEL code in "intrusion mode" (PANEL_INT). DOE indicates that the use of shielded containers does not affect PABC-2004 PANEL_INT results, since the SCPA uses the PABC-2004 BRAGFLO results. The PABC-2004 PANEL_INT results used in the SPCA are documented in Garner and Leigh 2005.

The DOE approach is reasonable and no changes to NUTS and PANEL for the SCPA appear to be warranted.

5.2.6 Culebra Flow and Transport: MODFLOW and SECOTP2D

SCPA Culebra flow and transport calculations are identical to PABC-2004 results, since DOE assumes that their conceptual models are not affected by the presence of shielded containers. DOE therefore uses the Culebra flow and transport results from the PABC-2004 for the SCPA. These results are documented in Lowry and Kanney 2005.

This approach is reasonable and no changes to MODFLOW and SECOTP2D for the SCPA appear to be warranted.

SECOTP2D Version 1.41 was used to support the SCPA and PABC-2004. In 2005, SECOTP2D was revised to SECOTP2D 1.41A to correct several formatting issues. These revisions to the code would not have impacted the SCPA results obtained with SECOTP2D 1.41.

5.2.7 Spallings: DRSPALL

Spallings volumes from a single borehole intrusion are calculated by DRSPALL at initial repository pressures of 10, 12, 14, and 14.8 MPa. Since the PABC-2004, as well as the SCPA, conservatively assumes that the shielded containers (and all others) instantaneously fail when the WIPP facility is closed, the presence of shielded containers will not impact either the waste material properties or the DRSPALL results. Thus, DOE uses the spallings results that were calculated by DRSPALL for the PABC-2004 and documented in Vugrin 2005a for the SCPA.

The effect that placement of RH waste on the waste room floors has on spallings releases is addressed in the SCPA CCDFGF calculations (Section 5.2.10).

DRSPALL Version 1.0 was used to support the SCPA and PABC-2004. In January 2004, DRSPALL Version 1.0 was revised to DRSPALL 1.10, in which the following modifications were made:

- Cosmetic changes
- Bypassing the bounds checking
- Changing the upper bound on the far-field stress from 15E+06 to 18E+06 to accommodate future initial conditions

These revisions to the code would not have impacted the SCPA results obtained with DRSPALL 1.0.

5.2.8 Cuttings and Cavings: CUTTINGS_S

The code CUTTINGS_S has two major functions for WIPP PA: (1) calculation of cuttings and cavings volumes from a single borehole intrusion, and (2) interpolation of DRSPALL volumes to calculate spall volumes in the scenarios for drilling intrusions. Since the SCPA conservatively assumes that the shielded containers (and all others) instantaneously fail when the WIPP facility is closed, the presence of shielded containers will impact neither the waste material properties nor the cuttings and cavings volumes predicted in PABC-2004 CUTTINGS_S. Thus, DOE uses the PABC-2004 CUTTINGS_S results, documented in Vugrin 2005b, for the SCPA. The effect that placement of RH waste in the waste rooms has on spallings and cuttings and cavings releases is addressed in the SCPA CCDFGF calculations (Section 5.2.10).

This approach is reasonable for the SCPA.

CUTTINGS Version 5.04 was used to support the SCPA and PABC-2004. Since 2004, CUTTINGS has undergone a number of upgrades involving the removal of unneeded functionality in order to improve maintainability, changes to reduce the number of input files needed, and the determination of the radius of turbulent flow. The most recent version of CUTTINGS is 6.02. These revisions to the code would not fundamentally change the SPCA results obtained with CUTTINGS 5.04.

5.2.9 Direct Brine Release: BRAGFLO_DBR

For the PABC-2004, BRAGFLO is run in the direct brine release (DBR) mode (BRAGFLO_DBR) to calculate the volumes of brine in DBRs. DOE assumes that since the SCPA uses the PABC-2004 BRAGFLO and CUTTINGS_S results, the presence of shielded containers will not impact the DBR results. Therefore, the SCPA uses the BRAGFLO_DBR results from the PABC-2004. These results are documented in Stein et al. 2005.

This is a reasonable approach, and no changes to BRAGFLO_DBR for the SCPA appear to be warranted.

5.2.10 CCDF Construction: CCDFGF

The code CCDFGF assembles the release estimates from all other components of the WIPP PA system to generate CCDFs of releases. In doing so, the code incorporates the stochastic uncertainty associated with drilling events. Stochastic uncertainty pertains to unknowable future events, such as intrusion times and locations that may affect repository performance, and is treated by generating random sequences of future events.

The new emplacement scheme that would be used for RH waste in shielded containers affects a few of the stochastic variables used in CCDFGF calculations. For example, the probability of intruding RH waste will be affected by the new emplacement scheme. Thus, DOE re-evaluated all of the parameters and stochastic variables used by CCDFGF for the SCPA to determine if they would be affected by the emplacement of shielded containers, and appropriate changes were included in the SCPA. The SCPA includes three new sets of CCDFGF calculations using revised parameters. One set of calculations was completed for each of the emplacement scenarios that are discussed in Section 4.2 and above. This review found that there are no other changes that should have been made to CCDFGF. The run control scripts in Appendix B of Dunagan et al. (2007) for CCDFGF were reviewed and found to be complete. Furthermore, an independent check on the placement of selected CCDFGF simulations into SNL's CMS was made during an onsite visit on August 6, 2008.

CCDFGF Version 5.0A was used to support the SCPA and PABC-2004. In June 2004, the code was changed from Version 5.0A to Version 5.01 to reflect changes involving the confidence intervals assigned to the drilling rate that were changed from 90% to 99.5%. The most recent version of the code, however, was issued in December 2004 (Version 5.02). This new version includes changes to the Function FindSeries (i.e., a block in the IF-THEN-ELSE construction was removed and a check is made within the remaining block to ensure that 0 is never returned). These revisions to the code would not have impacted the SCPA results obtained with CCDFGF 5.0A.

5.3 PA Results

The PABC-2004 results represent the repository performance using baseline assumptions and parameters (Scenario 1). DOE compared the results of SCPA Scenarios 2 and 3 with the PABC-2004 results to assess the potential impacts of shielded containers on repository performance. The results from the SCPA Scenarios 2 and 4 were compared to assess the sensitivity of the explicit representation of individual waste streams on releases. For each set of comparisons, DOE evaluated the CCDFs generated from the modeling studies.

DOE presents total normalized releases for the SCPA scenarios, with a discussion of the cuttings and cavings, spallings, and direct brine releases (Dunagan et al. 2007). Specifically, the mean CCDFs for Replicate 1 of these release mechanisms are presented. The releases through groundwater transport for all scenarios were found to be insignificant, as in the PABC-2004.

DOE ran SCPA Scenario 2 for three replicates, while Scenarios 3 and 4 were run for one replicate. DOE asserts that the trends observed for Replicate 1 comparisons hold for Replicates

2 and 3, as well, so they are not shown. The mean CCDFs for Replicates 2 and 3 of Scenario 2, however, are shown in Appendix A of Dunagan et al. 2007.

DOE's results from SCPA Scenarios 2 and 3 are compared to the PABC-2004 results in Section 5.3.1 below. These scenarios were modeled to determine if the packaging of RH waste in shielded containers affects baseline estimates of releases. In Section 5.3.2, DOE's comparison for SCPA Scenarios 2 and 4 is discussed, indicating whether the explicit representation of RH waste streams yields significantly different results from calculations that model RH waste with a single composite waste stream.

5.3.1 Comparison of Scenarios 2 and 3 to the PABC-2004

The results of the SCPA Scenario 2 represent the long-term repository performance when all of the RH waste is placed in shielded containers on the floor of the repository. The results of the SCPA Scenario 3 represent the long-term repository performance when half of the RH waste is placed in shielded containers on the floor of the repository and the remaining half is placed in canisters in the walls of the repository. When SCPA Scenarios 2 and 3 are compared to the results from PABC-2004, the effects of the location of the RH waste can be evaluated.

As part of this review, a visual comparison was made of the mean CCDFs for total cuttings and cavings, spallings, and DBR normalized releases for Replicate 1 of SCPA Scenarios 2 and 3 with those from the PABC-2004. This comparison found that the means are virtually indistinguishable between SCPA scenarios and the baseline. In addition, the horsetail plots for Scenario 2, Replicate 2 of the SCPA, were visually compared with Replicate 2 of the PABC-2004 (Vugrin and Dunagan 2005). The overall distributions were comparable, indicating higher percentile values for the distributions were similar.

DOE indicates that the mean DBR CCDFs have no significant differences between SCPA scenarios and the PABC-2004. Based upon Figure 5-1 (from Dunagan et al. 2007), this assertion also appears to be supported. However, as shown in that figure, the mean DBRs for SCPA Scenario 2 are consistently the smallest, and the mean DBRs for the PABC-2004 are consistently the largest, but the differences are minor.

The frequency and magnitude of DBRs are typically highest when a brine pocket has previously been intruded, since the additional brine that enters the repository in these scenarios generally results in high pressures and brine saturations, conditions leading to DBRs. Thus, DOE explains the trend in DBRs by evaluating the probability of hitting a brine pocket. Given that an intrusion intersects the berm area, the conditional probability that a borehole penetrates a Castile brine pocket is the product of two factors: (1) P(E|B), the conditional probability that the excavated area is intruded given that the berm area is intersected, and (2) the sampled GLOBAL:PBRINE value. The PABC-2004 and SCPA scenarios used identical sampled GLOBAL:PBRINE values, so the trend in mean DBRs is caused by the differences in P(E|B). P(E|B) is defined to be the sum of the RH and CH excavated areas, divided by the berm area as shown below.

$$P(E \mid B) = \frac{\text{CH Area} + \text{RH Area}}{\text{Berm Area}}$$

Since the PABC-2004 has the largest RH area, the probability of intersecting a brine pocket (when the berm area is intersected) is higher than for Scenarios 2 and 3 and, consequently, the PABC-2004 model resulted in the largest mean DBRs. Similarly, SCPA Scenario 2 had the lowest DBRs, since that calculation used the smallest RH area.

DOE also indicates that even though the PABC-2004 had the largest mean DBRs at all probabilities, the difference between the DBRs for the PABC-2004 and the SCPA scenarios was still extremely small and not large enough to discernibly impact mean total releases.

DOE satisfactorily explains the differences between the various scenarios, and their assertion that the differences are small is supported by the results presented in Figure 5-1.

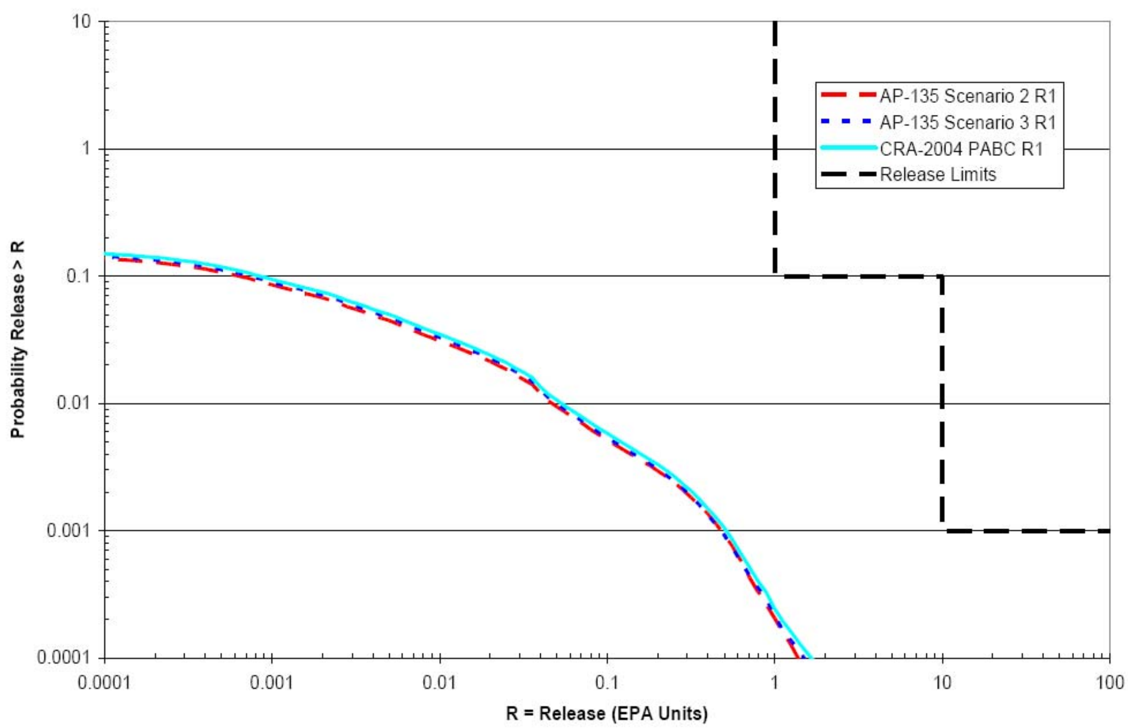

Figure 5-1: Mean DBRs from SCPA Scenarios 2 and 3 and PABC-2004, Replicate 1

(Dunagan et al. 2007)

5.3.2 Comparison of Scenarios 2 and 4 to the PABC-2004

DOE compares the results of the SCPA Scenario 2 and Scenario 4 to determine the impact on releases when RH waste is modeled with two different approaches. SCPA Scenario 2 calculations use a single composite RH waste stream to represent all RH waste streams, while SCPA Scenario 4 calculations explicitly represent all of the 77 individual RH waste streams. There are no other differences between the two calculations.

The total normalized releases for SCPA Scenarios 2 and 4, Replicate 1, are compared in Figure 5-2. DOE concludes that modeling the use of shielded containers with explicit representation of all 77 individual RH waste streams caused no differences of practical significance on the mean releases.

DOE's conclusion is supported by the results presented in Figure 5-2, in which the mean total releases for each of the scenarios are virtually indistinguishable from one another.

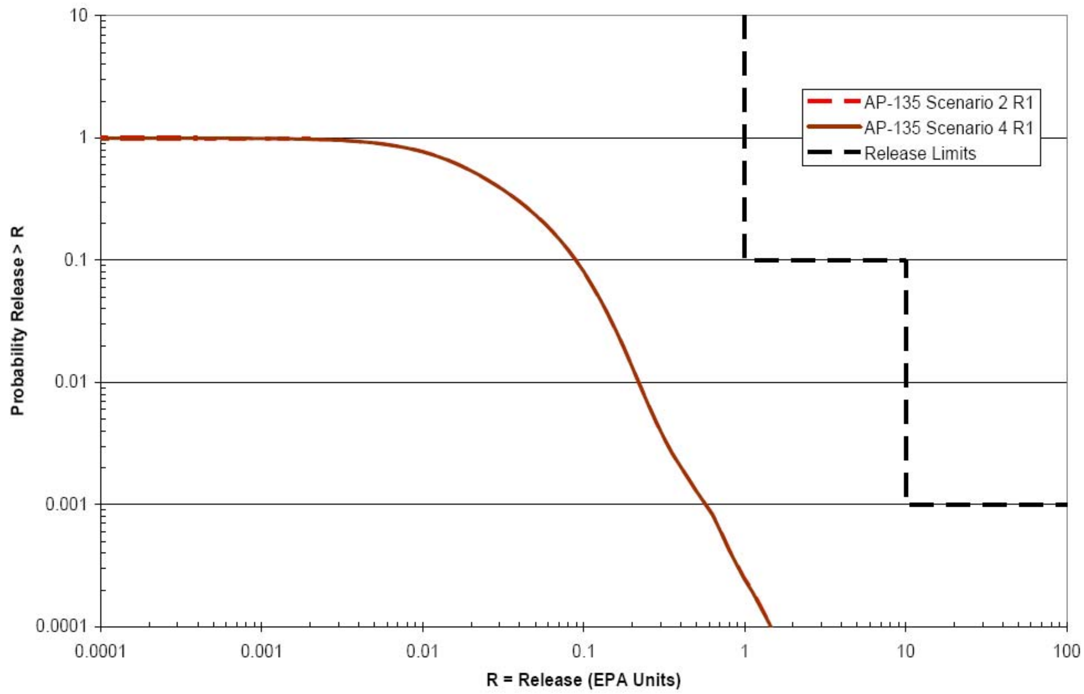

Figure 5-2: Mean Total Releases from SCPA Scenario 2 and Scenario 4, Replicate 1

(Dunagan et al. 2007)

5.3.3 Deviations from Analysis Plan AP-135

Prior to conducting the SCPA, DOE prepared analysis plan AP-135 documenting the proposed modeling studies (Dunagan and Vugrin 2007). DOE made four deviations from AP-135. One deviation was the container design, where the thickness of the steel lid and bottom were increased from 2.75 in to 3 in. This change was captured in the steel estimates (Crawford and Taggart 2007).

The second deviation from AP-135 was the addition of calculations to assess the impact of RH waste on releases. DOE reran CCDFGF using the PABC-2004 files (Leigh et al. 2005) with the fraction of repository volume occupied by waste (REFCON:FVW) set equal to zero. By setting this parameter equal to zero, the contributions from cuttings, cavings, and spallings releases from CH waste were eliminated from the total release results. This allowed DOE to evaluate the impact of the RH waste on total releases, because the current baseline (PABC-2004) only attributes RH cuttings releases to the total release results. Therefore, the cuttings output of this set of calculations represents the total releases from RH waste.

The third deviation from AP-135 is that additional qualified codes that were not explicitly called out in AP-135 were used in the SCPA calculations (Table 5-2). DOE used these additional codes to capture the parameter changes that were required for the analysis.

The fourth deviation pertains to DOE’s consideration of several changes to the WIPP PA technical baseline, as described in AP-132 (Vugrin and Nemer 2007). The Shielded Container Analysis Plan (AP-135) indicates that the DOE intended to submit a PCR to the EPA that would contain a set of modifications to WIPP PA models and parameters that the DOE would like to include in the WIPP PA technical baseline. DOE performed those calculations, but does not present the results. DOE has decided not to continue with the peer review of the proposed PA changes.

5.4 Software

The SCPA was performed using the same codes used in the previous compliance performance calculations (i.e., PABC-2004) and listed in Table 5-2. These codes were executed on the WIPP PA Alpha Cluster, which is described in Table 5-3.

The verification, testing, and documentation of all of these codes have been conducted during previous reviews and found to be adequate (TEA 2005).

Additionally, commercial off-the-shelf software, such as MATHEMATICA®, MATLAB®, MATHCAD®, Excel®, Access®, Grapher®, or Kaleidagraph®, running on MS Windows XP®-based PC workstations may be utilized.

Table 5-2: Computer Codes Used for SCPA

Code	Version	Build Date	Executable
CCDFGF	5.02	13-DEC-2004	CCDFGF_QB0502.EXE
EPAUNI	1.15A	03-JUL-2003	EPAUNI_QA0115A.EXE
PRECCDFGF	1.01	07-JUL-2005	PRECCDFGF_QA0101.EXE
POSTLHS	4.07A	25-APR-2005	POSTLHS_QA0407A.EXE
MATSET	9.10	29-NOV-2001	MATSET_QA0910.EXE
GENMESH	6.08	31-JAN-1996	GM_PA96.EXE

Table 5-3: WIPP PA Alpha Cluster

Node	Hardware Type	CPU	Operating System
CCR	HP AlphaServer ES45 Model 2	Alpha EV68	Open VMS 8.2
TDN	HP AlphaServer ES45 Model 2B	Alpha EV68	Open VMS 8.2
BTO	HP AlphaServer ES40	Alpha EV68	Open VMS 8.2
CSN	HP AlphaServer ES40	Alpha EV68	Open VMS 8.2
GNR	HP AlphaServer ES47	Alpha EV7	Open VMS 8.2
MC5	HP AlphaServer ES47	Alpha EV7	Open VMS 8.2
TRS	HP AlphaServer ES47	Alpha EV7	Open VMS 8.2

TBB	HP AlphaServer ES47	Alpha EV7	Open VMS 8.2

5.5 Applicable QA Procedures

A review of the applicable WIPP QA procedures was conducted and is summarized below.

- Analyses will be conducted and documented in accordance with the requirements of NP 9-1, *Analyses*.

  A review of the documents relevant to NP 9-1 was conducted and we found that the appropriate procedures were followed, thereby meeting the NP 9-1 requirements (Dunagan et al. 2007, Long 2007).

- All software used will meet the requirements laid out in NP 19-1, *Software Requirements* and NP 9-1, as applicable.

  A previous EPA review of the software concluded that the requirements of NP 19-1 were met for each of the codes used to support SCPA (TEA 2005).

- The analyses will be reviewed using NP 6-1, *Document Review Process*.

  Documentation of the review process was examined and found to be appropriate and consistent with NP 6-1 requirements (Kirchner 2007b and 2007c, Chavez 2007a and 2007b, Moo Lee 2007a and 2007b, Fox 2007, Trone 2007, and Clayton 2008).

- All required records will be submitted to the WIPP Records Center in accordance with NP 17-1, *Records*.

  During a site visit on August 6, 2008, the records were reviewed and found to be properly archived in the WIPP Records Center in accordance with NP 17-1.

- New and revised parameters will be created as discussed in NP 9-2, *Parameters*.

  A review of the documents relevant to the parameter development was conducted and found that the procedures followed satisfactorily met NP 9-2 (Dunagan 2007, Chavez 2007a, Moo Lee 2007a, and Chavez 2007b).

6.0 IMPACT OF WASTE INVENTORY

As was discussed in Section 4 of this report, DOE made a bounding assessment to be used in the PA for the PCR that all of the RH TRU waste would be placed in shielded containers on the disposal room floor, rather than in canisters in the boreholes in the room walls. Estimates of the resulting inventory changes were made based on the assumption that all RH TRU waste was packaged in shielded containers.

The PCR will have no impact on the inventory of radionuclides contained in the RH TRU waste. The only difference is that some portion of the RH TRU radioactivity will be emplaced on the disposal room floor, rather than in boreholes in the walls of the disposal rooms. However, the quantities of lead, steel (iron) and cellulosics, plastics, and rubber (CPR) will be affected. An estimate of the quantities of these materials, based on the assumption that all of the RH TRU waste is emplaced in shielded containers, was developed by Crawford and Taggart (2007). Their estimates are summarized in Table 6-1, together with comparable estimates from the 2004 PABC-2004 (Leigh et al. 2005, Table 12), the 2007 Baseline Inventory (ATWIR 2007), and the 2008 Baseline Inventory[6] (ATWIR 2008). From Table 6-1, it can be seen that the PCR would add more lead, steel, and CPR to the WIPP inventory than included in any of the three reference inventories.

Table 6-1: Estimates of Quantities of Steel, Lead and CPR in RH TRU Waste Packaging and Emplacement Materials

Material	Based on PCR		Based on PABC-2004 Inventory		Based on 2007 Inventory		Based on 2008 Inventory	
	Packaging (kg)	Emplacement (kg)	Packaging (kg)	Emplacement (kg)[a]	Packaging (kg)	Emplacement (kg)	Packaging (kg)	Emplacement (kg)
Lead	2.70E07	0	2.97E06	0	3.82E04	0	2.48E04	0
Steel	2.56E07	0	3.82E06	0	4.32E06	0	4.46E06	0
Cellulosics	0	2.40E04	0	0	0	0	0	0
Plastics	1.13E05	4.29E05	2.19E04	0	7.79E04	0	9.91E04	0
Rubber	0	0	0	0	0	0	0	0

a - All emplacement materials identified in Leigh et al. 2005 assumed to be associated with CH TRU waste.

However, these inventory increases must be examined from the perspective of the total quantities of each material associated with both RH TRU and CH TRU waste, not just those associated with RH TRU packaging and emplacement materials. The PABC-2004 inventory provides an appropriate basis for comparison, since the PABC-2004 CCDFs are used as the baseline for comparison with the SCPA CCDFs. Details from the PABC-2004 are included in Table 6-2 (Leigh et al. 2005). Material densities are converted to masses, assuming a volume of 1.685E05 m^3 for CH–TRU waste and a volume of 7.079E03 m^3 for RH TRU waste.

[6] This inventory is also documented in PAIR 2008 and was used in the PABC-2009.

Table 6-2: Masses of Selected Materials in the PABC-2004 Inventory

Material	CH Waste		RH Waste		Total
	Density (kg/m^3)	Mass (kg)	Density (kg/m^3)	Mass (kg)	Mass (kg)
Steel in Waste	110	1.85E07	59	4.18E05	1.89E07
Cellulosics in Waste	60	1.01E07	9.3	6.58E04	1.01E07
Rubber in Waste	13	2.19E06	6.7	4.74E04	2.24E06
Plastics in Waste	43	7.24E06	8.0	5.66E04	7.30E06
Lead in Waste	150[a]	2.53E07	74[a]	5.24E05	2.58E07
Steel in Packaging	170	2.86E07	540	3.82E06	3.24E07
Cellulosics in Packaging	0	0	0	0	0
Rubber in Packaging	0	0	0	0	0
Plastics in Packaging	17	2.86E06	3.1	2.19E04	2.88E06
Lead in Packaging	0.013	Negligible	420	2.97E06	2.97E06
Cellulosics in Emplacement	—	2.07E05	—	0	2.07E05
Plastics in Emplacement	—	1.48E06	—	0	1.48E06

a - Assumes that “lead/cadmium” waste stream is all lead (Crawford 2005).

From Table 6-2, it can be calculated that the total steel mass in the PABC-2004 inventory is 5.14E07 kg. If all of the steel in the RH packaging (3.82E06 kg) were replaced by an amount of steel assuming that all of the RH waste was packaged in shielded containers (2.56E07 kg per Crawford and Taggart 2007, Table 8), the net increase in steel inventory would be 2.18E07 kg (2.56E07 – 0.38E07) and the total steel inventory would be 7.32E07 kg (5.14E07 kg + 2.18E07 kg), an increase of 42%.

Similar results for other materials of interest are summarized in Table 6-3. Data in columns 2 and 3 are derived from Table 6-2; data in column 4 are taken from Table 6-1; data in column 5 are the differences between columns 3 and 4; data in column 6 are the sum of columns 2 and 5; and column 7 results are obtained by dividing data in column 5 by those in column 1. It is clear from Table 6-3 that large increases in the masses of steel and lead will result from the PCR, but that the increase in the total CPR mass is only about 2%.

Table 6-3: Increase in Mass of Steel, Lead, and CPR Materials Used in PCR as Compared to PABC-2004

Material	Total Mass in PABC-2004 (kg)	RH Mass in PABC-2004 Pack/Emplace (kg)	RH Mass in PCR Pack/Emplace (kg)	Change in Mass PCR vs. PABC-2004 Pack/Emplace (kg)	Total PCR Mass (kg)	Increase in Mass PCR vs. PABC-2004 (%)
Steel	5.14E07	3.82E06	2.56E07	2.18E07	7.32E07	42
Lead	2.88E07	2.97E06	2.70E07	2.40E07	5.28E07	83
Cellulosics	1.04E07	0	2.40E04	2.40E04	1.04E07	0.23
Plastics	1.17E07	2.19E04	5.42E05	5.20E05	1.22E07	4.4
Rubber	2.24E06	0	0	0	2.24E06	0

The very small increase in CPR is of no consequence, because DOE adjusts the amount of MgO added to the disposal rooms to maintain a constant MgO Excess Factor of 1.2 (Reyes 2008). This procedure ensures that sufficient MgO is available to sequester all CO_2 generated by microbial degradation of CPR.

Comparable data to that in Table 6-2 are included in Table 6-4 for the PABC-2009 inventory.

Table 6-4: Masses of Selected Materials in the PABC-2009 Inventory

Material	CH Waste		RH Waste		Total
	Density (kg/m^3)	Mass (kg)	Density (kg/m^3)	Mass (kg)	Mass (kg)
Steel in Waste	81	1.36E07	170	1.20E06	1.48E07
Cellulosics in Waste	40	6.74E06	22	1.56E05	6.90E06
Rubber in Waste	5.6	9.44E05	6.6	4.67E04	9.91E05
Plastics in Waste	38	6.40E06	28	1.98E05	6.60E06
Lead in Waste	150[a]	2.53E07	74[a]	5.24E05	2.58E07
Steel in Packaging	190	3.20E07	630	4.46E06	3.65E07
Cellulosics in Packaging	5.1	8.59E05	0	0	0
Rubber in Packaging	0	0	0	0	0
Plastics in Packaging	16	2.86E06	14	9.91E04	2.96E06
Lead in Packaging	0.0	0	3.5	2.48E04	2.48E04
Cellulosics in Emplacement	1.34	2.26E05	—	0	2.26E05
Plastics in Emplacement	6.59	1.11E06	—	0	1.11E06

a - Assumes that “lead/cadmium” waste stream is all lead (Crawford 2005).

From Table 6-4, it can be calculated that the total steel mass in the PABC-2009 inventory is 5.13E07 kg, a value virtually identical with 5.14E07 kg for PABC-2004. No new data were available for lead in the waste, so the same estimate was used as in Table 6-2. As will be discussed in Section 8, the presence of iron and lead in the repository is expected to have a beneficial effect by contributing to the establishment and maintenance of reducing conditions. Consequently, any incremental changes associated with the PAIR 2008 inventory used in the PABC-2009 were not evaluated further.

As part of this review, the inventory-related data in Crawford and Taggart 2007 and Dunagan et al. 2007 were examined. The inventory data and calculations in Sections 4 and 6 of Crawford and Taggart 2007, including data in Tables 4, 5, 6, 7, and 8, were verified and traced to their sources. The inventory data in Dunagan et al. 2007, particularly in Sections 3.1.2 and 3.1.4.1 of that report, were similarly examined and verified.

7.0 RADIOLOGICAL CHARACTERIZATION OF CANDIDATE WASTE STREAMS FOR SHIELDED CONTAINERS

Due to the fact that the dose rate would be limited to 200 mrem/hr on the surface of shielded containers, not all of the RH TRU waste streams identified in DOE/TRU-2006-3344 (DOE 2006c) could be packaged in shielded containers. DOE (i.e., Los Alamos National Laboratory-

Carlsbad Operations, or LANL-Carlsbad) performed a scoping analysis (Crawford and Taggart 2007) to estimate which of the RH TRU waste streams are potential candidates for disposal in shielded containers. The final determination as to which RH TRU waste would actually be packaged in shielded containers would be made based upon the measured surface dose rate once the waste has been placed inside the container. The DOE analysis is for planning purposes only.

For the scoping analysis, LANL-Carlsbad focused only on the gamma component of RH TRU waste, and ignored any neutron contribution to the surface dose rate. This is appropriate for this level of scoping, as the neutron dose contribution to the total surface dose rate is expected to be small. For example, DOE 2006b, Section 2.4.1.2, states that, "The shielding [for the hot cell complex] is designed for an internal gamma surface dose rate of 400,000 Rem/hr and for an internal neutron surface dose rate of 45 Rem/hr." In other words, the expected neutron dose rate is only about 0.01125% of the total estimated dose. A similar statement was made in the WIPP technical basis for external dosimetry (Bradley et al. 1993):

> *The neutron component of the total external radiation exposure for RH TRU is expected to be a small percentage of the total gamma exposure. If one looks at the ratio of the maximum neutron dose rate to the maximum total allowed dose rate (270 mrem/hr per 1,000,000 mrem/hr), the expected neutron dose rate is only 0.027% of the total estimated dose.* (See discussion in Section 6.3 below).

Since (1) the scoping analysis is limited to photon/gamma rays, (2) the shielded containers have a simple cylindrical geometry, and (3) the container walls are sufficiently thick to shield any weak gamma rays (i.e., Bremsstrahlung is not a concern), this analysis is ideally suited for the point-kernel analysis method used by MicroShield®:

7.1 MicroShield® Quality Assurance

MicroShield® is a comprehensive photon/gamma ray shielding and dose assessment program that is widely used for designing radiation shields.[7] It was originally developed by Grove Engineering, which was acquired by Areva NP (formerly Framatome ANP, Inc.). MicroShield is an enhanced adaptation of ISOSHLD II, a proven mainframe gamma shielding code developed at Pacific Northwest Laboratory (PNL) in the mid-1960s.

On file at the LANL-Carlsbad records center are (1) LOC-MIC-01, Rev. 0, *MicroShield 6 Requirements Memorandum*, (2) LOC-MIC-02, Rev. 0, *MicroShield Verification & Validation Report*, supplied by Grove Engineering, and (3) LOC-MIC-03, Rev. 0, *MicroShield 6 User's Manual*, also supplied by Grove. The LANL-Carlsbad records center also has the Installation and Checkout forms (QP19-2-2) that document the installation of MicroShield on two computers (PN1164783 and PN1165337), including the successful results running each of the 24 test cases on each computer. Each of these documents is also available in the Compliance Recertification Electronic Library (CREL), and have been reviewed and found to be acceptable.

[7] To avoid redundancy, the registered trademark symbol ® is not included with subsequent references to MicroShield, but is understood as included.

Additionally, Document Review Forms (DRF) are completed whenever a problem or error occurs during the MicroShield (or any other software) installation. Presently, there are nine MicroShield-related DRFs on file at the LANL-Carlsbad records center. A check of these DRFs during a visit to the LANL-Carlsbad records center found them to be properly implemented.

7.2 MicroShield Dimensions

Figure 7-1 shows a schematic of the RH TRU waste shielded container (Day 2008). It has been designed to hold a single 30-gallon drum having outside dimensions consistent with those of a 55-gallon drum. For comparison purposes, the dimensions of DOT-17C 30-gallon and 55-gallon drums are given in Table 7-1.

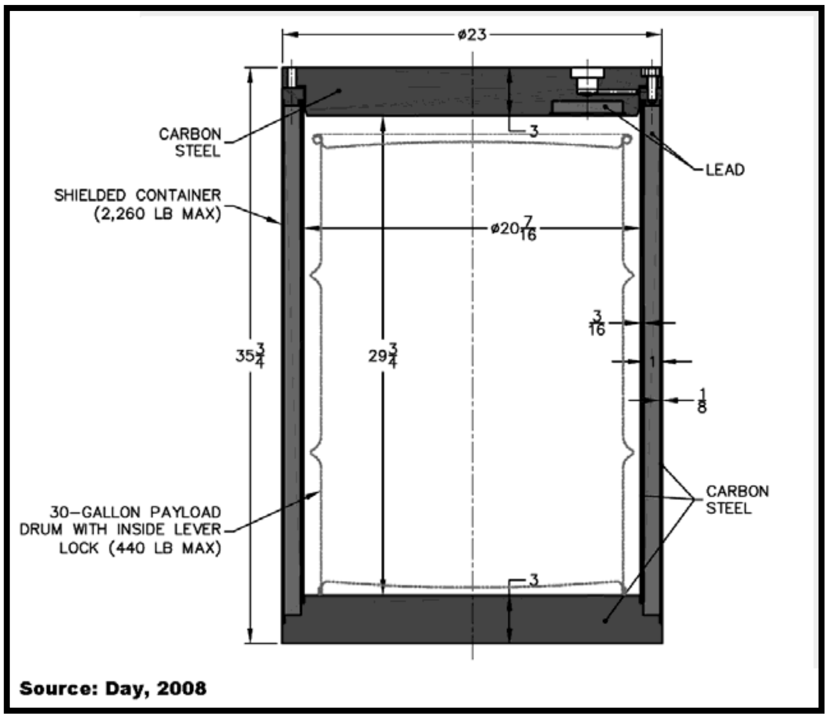

Figure 7-1: Shielded Container

Table 7-1: DOT-17C Drum Dimensions

Dimension	30-gallon drum		55-gallon drum	
	Interior	Exterior	Interior	Exterior
Height (in)	28.0	29.5	33.25	35.00
Diameter (in)	18.0	20.0	22.25	24.00

Source: DOE 1996

Table 7-2 gives the dimensions LANL used in MicroShield, and indicates that LANL has slightly over-estimated the radius of a 30-gallon drum, resulting in a drum volume closer to 35 gallons. The effect of this on the MicroShield analysis is two-fold. First, the larger radius would place the source nearer to the dose receptor, thereby tending to slightly increase the calculated dose. Second, since a 1-Ci source was specified for each MicroShield analysis, the larger volume would result in a reduced concentration, thereby slightly decreasing the calculated dose. In reality, it is unknown where the source would be inside the 30-gallon drum, and that uncertainty would be much larger than any uncertainty introduced by this slight overestimate of the radius.

Table 7-2 also indicates that the distance to the dose point was slightly shorter (i.e., conservative), when compared to the distance estimated by SC&A.

Table 7-2: MicroShield Dimensions

Parameter	LANL	SC&A
30-gallon drum Dimensions		
Height (in)	27.7	28
Radius (in)	9.6	9
Volume (in^3)	8,092	7,125
Volume (gal)	35.03	30.85
Shielded Container Dimensions		
Diameter (in)	20.4375	N/A
Height (in)	29.75	N/A
Side Dose Point Distance (in)		
Waste	9.645	9.000
Air	0.542	1.219
Steel	0.188	0.188
Lead	1.000	1.000
Steel	0.125	0.125
Total	11.50	11.53
Top Dose Point Distance (in)		
Waste	27.69	28.00
Steel	3.00	3.00
Air	1.06	1.75
Total	31.75	32.75

LANL Source: Crawford and Taggart 2007, page 27

7.3 RH TRU Significant Radionuclides

The WIPP RH TRU waste streams contain approximately 126 radionuclides (DOE 2006c). Rather than perform the MicroShield analysis for each radionuclide, LANL decided to limit the analysis to only those radionuclides that have a significant contribution to the dose, either because of their relative abundance in the waste and/or because they are significant photon/gamma ray emitters.

LANL used 'gamma factors' from the "Radiological Health Handbook" (Shleien 1992) to determine which radionuclides to include in their MicroShield analysis. This resulted in the list of 14 radionuclides given in Table 7-3, taken from INV-SAR-08, Table 3 (Crawford and Taggart

2007). Rather than repeat the LANL calculations, it was decided to rank the radionuclides based on information provided in *Federal Guidance Report No. 12* (FGR 12) (Eckerman and Ryman 1993).

FGR 12, Table A.1, presents (along with other data) the average photon energy (MeV) per nuclear transformation. The Table A.1 photon energy was used to rank the 126 RH TRU radionuclides, with the resulting rankings shown in column 3 of Table 7-3 for the same suite of radionuclides as initially selected by LANL. The relative rankings were based on the product of the average photon energy and the curie density from DOE 2006c, Table 32.

Table 7-3: Significant RH TRU Radionuclides

Nuclide	Curies per 200 mrem/hr‡	Energy Rank	Neutron*	RH/CH Ratio
Cs-137	2.0	1	No	
Co-60	0.12	2	No	
Eu-152	0.39	3	No	
Eu-154	0.29	4	No	
Am-241	208,855	5	Yes	1.4× Lower
Cs-134	0.64	6	No	
Cm-247	10	9	No	
Pu-238	2,855,368	10	Yes	15.9× Lower
Pu-240	9,365,050	11	Yes	2.5× Lower
Cm-244	697,800	13	Yes	4.1× Higher
Pu-243	1,396	17	No	
Cm-243	213	28	Yes	29.6× Higher
Th-229	4,435	39	No	3.2× Higher
Pu-239	151,240	97	Yes	4.6× Lower

‡ Source: Crawford and Taggart 2007, Table 3
* Source: DOE 2006c, Table 5.1

Using the photon energy ranking, 10 of the same 14 radionuclides would be selected as significant. The four LANL significant radionuclides not selected by the FGR 12 photon energy ranking were Pu-243, Cm-243, Th-229, and Pu-239. The four radionuclides selected by the FGR 12 photon energy ranking, but not included as significant by LANL, were Eu-155 (7), Tl-208 (8), Bi-212 (12), and Sb-125 (14). As will be discussed subsequently, additional MicroShield runs were made to determine the significance of the four radionuclides identified by the FGR 12 photon energy ranking.

Table 7-3 also shows which of the 14 LANL-identified significant radionuclides are neutron emitters, as identified by RH TRU 72-B Cask SAR, Rev. 4, Table 5.1 (DOE 2006d). As Table 7-3 demonstrates, half of the radionuclides identified by LANL as significant for analysis of the shielded container are not neutron emitters, and four of the seven neutron emitters have lower average RH concentrations than CH concentrations, as indicated in the right-hand column of Table 7-3 (the concentrations used to perform this comparison were taken from DOE 2006c, Table 32). The right-hand column of Table 7-3 shows that, with the exception of Cm-243, Cm-244, and Th-229, the average neutron dose rate from RH TRU is expected to be less than from

CH TRU due to the lower concentrations of neutron emitters. This is additional confirmation that ignoring the neutron contribution to the shielded container surface dose rate is appropriate for this scoping analysis. Again, it is pointed out that the final determination as to which RH TRU waste would actually be packaged in shielded containers would be made based upon the measured gamma and neutron dose rates.

7.4 MicroShield Gamma Energy Distribution

MicroShield is provided with a library of photon/gamma ray distributions for 497 radionuclides, including all of the radionuclides listed in Table 7-3. MicroShield also provides the user with the capability to specify a unique photon/gamma ray distribution for up to 25 energies. By comparing the MicroShield supplied gamma distributions to the latest distributions available from the Lawrence Berkeley National Laboratory (LBNL) website (LBNL 2008), LANL determined that the MicroShield photon/gamma ray library provides accurate data for most of the significant RH TRU radionuclides, with the exception of the five radionuclides listed in Table 7-4.

Table 7-4: MicroShield Gamma Distributions

Nuclide	MeV	(Photons/sec)/Ci	% of Total Energy Activity
Am-241	0.026345	8.8800E+08	2.853%
	0.033205	3.9220E+07	0.159%
	0.059537	1.3283E+10	96.429%
	0.069231	6.6348E+07	0.560%
Cm-244	0.056867	1.0544E+07	100.000%
Pu-238	0.055303	1.7504E+07	100.000%
Pu-239	0.11291	1.7606E+07	100.000%
Pu-240	0.054327	1.9412E+07	100.000%

LANL obtained the latest gamma distributions for these 5 radionuclides from the LBNL website, and adjusted them as necessary to correspond to MicroShield's 25 energy distribution limits. Crawford and Taggart (2007), pages 31 through 35, show the resulting energy distributions determined by LANL.

A check of the LANL-derived Pu-238 gamma spectrum is presented in Table 7-5. The left two columns of Table 7-5 show the gamma energy data obtained from the LBNL website for Pu-239, while the middle column is simply the product of the left two columns. The second from the right column (% of Total Energy Activity) is the value from the middle column divided by the sum of the middle column values. The right-hand column adjusts the "% of Total Energy Activity" values to account for the gammas, whose "% of Total Energy Activity" values were too small to be included directly into MicroShield. This adjustment is necessary, because the LBNL website lists 172 gamma energies associated with Pu-239, whereas MicroShield can only accept 25 different gamma energies. Because these 147 gammas range from 13.8 keV to over 1 MeV, it is appropriate to distribute them over the entire range of gamma energies.

Comparing the "Adjusted" column of Table 7-5 to the Pu-239 energy distribution determined by LANL on page 34 of Crawford and Taggart 2007 shows good agreement. Similar agreement was found when the LANL-derived gamma spectra for Am-241, Cm-244, Pu-238, and Pu-240 were checked.

Table 7-5: Calculated Gamma Spectrum for Pu-239

MeV	Fraction	Energy Act (Mev)	% of Total Energy Activity	Adjusted
0.038661	0.000105	4.06E-06	6.907%	7.527%
0.046204	7.4E-06	3.42E-07	0.582%	0.634%
0.051624	0.000271	1.40E-05	23.804%	25.942%
0.056828	1.13E-05	6.42E-07	1.093%	1.191%
0.077598	4.1E-06	3.18E-07	0.541%	0.590%
0.09878	1.22E-05	1.21E-06	2.050%	2.235%
0.103032	2.3E-06	2.37E-07	0.403%	0.439%
0.11537	4.6E-06	5.31E-07	0.903%	0.984%
0.116258	5.97E-06	6.94E-07	1.181%	1.287%
0.129297	6.31E-05	8.16E-06	13.882%	15.129%
0.144201	2.83E-06	4.08E-07	0.694%	0.757%
0.203545	5.69E-06	1.16E-06	1.971%	2.148%
0.332842	4.94E-06	1.64E-06	2.798%	3.049%
0.336113	1.12E-06	3.76E-07	0.641%	0.698%
0.345008	5.56E-06	1.92E-06	3.264%	3.557%
0.367072	8.9E-07	3.27E-07	0.556%	0.606%
0.368557	8.8E-07	3.24E-07	0.552%	0.601%
0.375045	1.55E-05	5.83E-06	9.917%	10.807%
0.380173	3.05E-06	1.16E-06	1.973%	2.150%
0.38275	2.59E-06	9.91E-07	1.687%	1.838%
0.39256	2.05E-06	8.05E-07	1.369%	1.492%
0.393136	3.5E-06	1.38E-06	2.341%	2.551%
0.413707	1.47E-05	6.06E-06	10.319%	11.246%
0.422598	1.22E-06	5.16E-07	0.877%	0.956%
0.451483	1.89E-06	8.55E-07	1.455%	1.586%
Various	3.28E-05	4.84E-06	8.241%	N/A

7.5 Potential Shielded Container RH TRU Waste Volume

SC&A independently ran MicroShield with Cs-137 and obtained exactly the same results as obtained by LANL. SC&A also ran MicroShield for the four radionuclides identified above in Section 6.3, but not analyzed by LANL (i.e., Eu-155, Tl-208, Bi-212, and Sb-125).

To evaluate the potential contribution of each radionuclide to the RH TRU shielded container surface dose rate, the maximum activity loadings for the 17 radionuclides were compared to the 30-gallon drum loading, based on the volume-averaged RH TRU waste concentration from DOE/TRU-2006-3344, Table 32 (DOE 2006c). As Table 7-6 shows, Cs-137 is the only radionuclide whose drum activity loading (column 4) based on its volume-averaged concentration (column 3) exceeds its MicroShield calculated maximum activity loading for the 200 mrem/hr limit (column 2).

Table 7-6: Surface-Dose Limiting Versus Volume-Averaged Activity Loading

Nuclide	Curies per 200 mrem/hr‡	RH Conc (Ci/m^3)†	RH Drum Activity (Ci)††	Drum Activity/Limit
Cs-137	2	60	6.78	3.39
Co-60	0.12	0.26	0.02938	0.244833
Eu-152	0.39	0.33	0.03729	0.095615
Eu-154	0.29	0.16	0.01808	0.062345
Cs-134	0.64	0.015	0.001695	0.002648
Tl-208*	0.0539	0.00069	7.8E-05	0.001446
Bi-212*	1.77	0.0019	0.000215	0.000121
Cm-247	10	0.0067	0.000757	7.57E-05
Sb-125*	6.66	0.00069	7.8E-05	1.17E-05
Am-241	208,855	2	0.226	1.08E-06
Pu-239	151,240	0.74	0.08362	5.53E-07
Pu-243	1,396	0.0066	0.000746	5.34E-07
Cm-243	213	0.000071	8.02E-06	3.77E-08
Cm-244	697,800	0.15	0.01695	2.43E-08
Pu-238	2,855,368	0.54	0.06102	2.14E-08
Pu-240	9,365,050	0.22	0.02486	2.65E-09
Th-229	4,435	0.000026	2.94E-06	6.62E-10
Eu-155*	6.50E+10	0.049	0.005537	8.52E-14

‡ Source: Crawford and Taggart 2007, Table 3
† Source: DOE 2006c, Table 32 (decayed to 2001)
†† Product of column 3 and volume of 30-gallon drum
* Radionuclide not included in the LANL analysis

Although the average Cs-137 RH TRU waste concentration would exceed the surface dose rate limit if it were used to fill a 30-gallon drum, because the actual Cs-137 RH TRU waste concentration is expected to range from 1.6×10^{-6} to 7.530 Ci/m^3, some volume of RH TRU waste containing Cs-137 would be suitable for shielded container disposal. In fact, Figure 7-2 shows that about 32% of the final RH TRU waste form volume could be packaged in shielded containers, assuming that Cs-137 is the dominant photon emitter. Figure 7-2 is based on the individual RH TRU waste stream Cs-137 concentrations given in DOE 2006b, Appendix J, and a total RH TRU disposal volume of 7,080 m^3. Figure 7-2 shows two curves; the solid line curve is based on the Cs-137 concentrations for all of the RH TRU waste streams presented in DOE 2006c, Appendix J, while the dotted curve is based on only those RH TRU waste streams listed in Crawford and Taggart 2007, Table 10. Table 10 of Crawford and Taggart (2007) lists only those waste streams that would meet the 2-Ci limit for Cs-137. The large "step" in the middle of the DOE 2006c Appendix J curve is due to a single waste stream, RP-W016 "PUREX TRU

Cladding Removal Solids," which has a final form volume of 3,943.6 m^3 and a Cs-137 concentration of 19.1 Ci/m^3.[8]

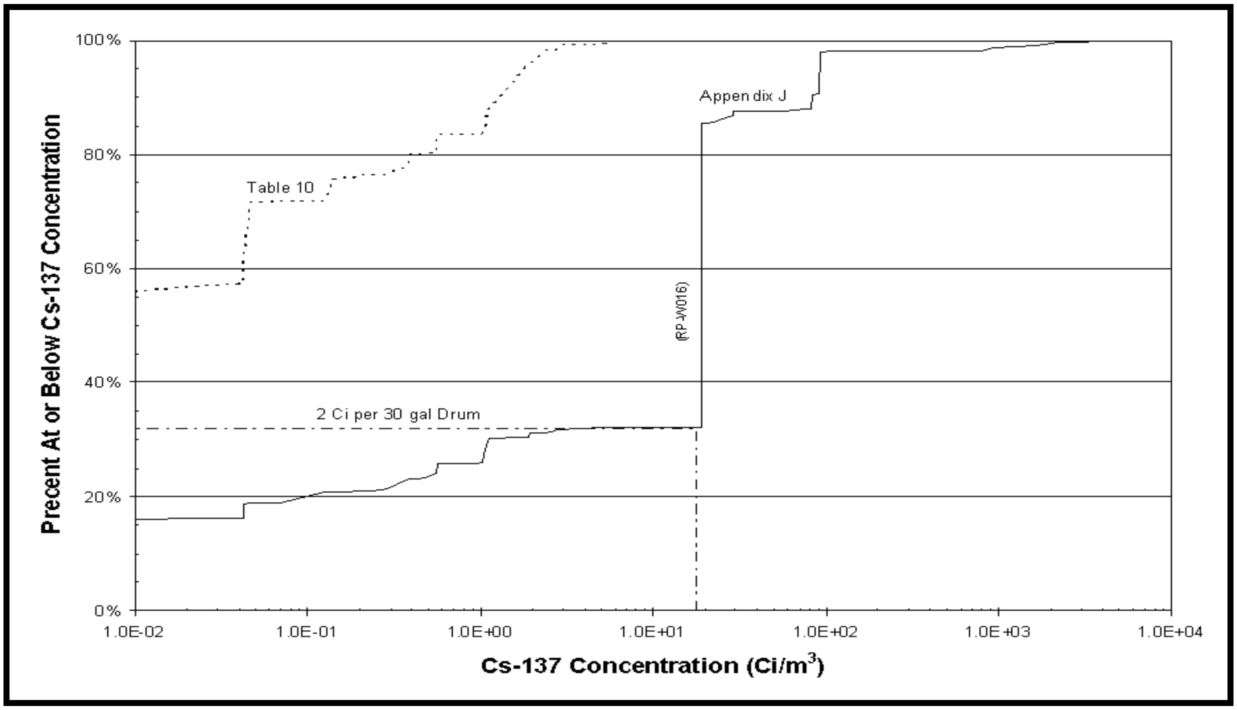

Figure 7-2: RH TRU Waste Stream At or Below the Limiting Cs-137 Concentration

Of course, the LANL analysis (Crawford and Taggart 2007) and this critique are for planning purposes only, and the final determination as to which RH TRU waste would actually be packaged in shielded containers would be made based upon the measured gamma and neutron dose rates.

7.6 Impact of 2008 Revised Transuranic Waste Baseline Inventory

Crawford and Taggart (2007) performed all of their analyses based upon the WIPP RH TRU inventory as given in DOE 2006c (i.e., the PABC-2004 inventory). In 2009, DOE issued the 2008 inventory report, Performance Assessment Inventory Report – 2008 (INV-PA-08),[9] which updated and revised both the CH TRU and RH TRU WIPP waste inventories as of December 31, 2007. This section looks at the effect, if any, of the revised RH TRU inventory on the shielded container analysis.

First, the 121 radionuclides that are reported in both PABC-2004 and INV-PA-08 were compared. Table 7-7 shows that for 55 radionuclides, the INV-PA-08 inventories are lower than the PABC-2004 inventory, while for 66 radionuclides, they are higher. The table also shows that

[8] This waste stream, which is a Hanford tank waste, has been removed from the 2007 inventory (ATWIR 2007), together with all other tank wastes, until a determination is made as to whether these wastes are TRU or high-level waste.

[9] Also referred to as PAIR 2008, which was the baseline inventory for the PABC-2009 required by EPA as part of the 2009 compliance re-certification.

the inventories for only 16 radionuclides changed by less than a factor of 2, while 40 (or one-third) of the radionuclides had their inventories change by over a factor of 1,000. The results from Table 7-7 are applicable to both the radionuclide concentrations and the total inventory.

Table 7-7: Summary Comparison of 2004 and 2008 Baseline Inventories

Factor Difference	INV-PA-08 Lower	INV-PA-08 Higher	Total
Less Than 2	5	11	16
Between 2 and 5	9	9	18
Between 5 and 10	3	2	5
Between 10 and 100	14	17	31
Between 100 and 1,000	8	3	11
Greater Than 1,000	16	24	40
Total	55	66	121

Table 7-8 is similar to Table 7-7, except that it focuses on the 14 radionuclides analyzed by Crawford and Taggart (2007). Table 7-8 shows that for three of the analyzed radionuclides (Th-229, Cm-243, and Pu-238), the INV-PA-08 inventories are greater than the PABC-2004 inventories; five of the analyzed radionuclides (Pu-240, Pu-239, Cm-244, Am-241, and Cs-137) had their inventories reduced by less than an order of magnitude; and two of the analyzed radionuclides (Pu-243 and Cm-247) had their inventories reduced by more than eight orders of magnitude.

Table 7-8: Comparison of 2004 and 2008 Baseline Inventories by Radionuclide

Radionuclide	2004 Inventory / 2008 Inventory	Radionuclide	2004 Inventory / 2008 Inventory
Th-229	0.044	Cs-137	4.76
Cm-243	0.24	Eu-154	14.2
Pu-238	0.75	Co-60	150.3
Pu-240	1.57	Eu-152	875.3
Pu-239	1.80	Cs-134	967.7
Cm-244	2.44	Pu-243	3.0E+08
Am-241	3.16	Cm-247	3.0E+08

The FGR 12, Table A.1, ranking of the RH TRU radionuclides (see Section 7.3) was re-performed using the revised concentrations from INV-PA-08, Table A-1, with the results presented in Table 7-9.

Table 7-9 compares the original ranking by Crawford and Taggart (2007) [C&T] with the rankings done in this report (based on FGR 12), using both the 2004 and 2008 baseline inventories (BI). Table 7-9 shows that regardless of the ranking method, Cs-137 is always in the first position, and Co-60 and Am-241 are always in the top five. Four other radionuclides (Pu-238, Pu-240, Eu-152 and Eu-154) are ranked in the top 14 for all three rankings. Four radionuclides (Th-229, Pu-238, Pu-243, and Cm-243) are outside of the top 14 for the FGR 12 rankings using both the 2004 and 2008 baseline inventories. Finally, three radionuclides

(Cs-134, Cm 244, and Cm-244) move from being ranked in the top 14 when ranked using FGR 12 and the 2004 inventory to outside the top 14 using FGR 12 and the 2008 inventory.

Table 7-9: Significant RH TRU Radionuclides – 2008 Baseline Inventory

Nuclide	Curies per 200 mrem/hr‡	Energy Ranking		
		C&T 2007	2004 BI FRG 12	2008 BI FGR 12
Cs-137	2	1	1	1
Am-241	208855	2	5	2
Co-60	0.12	3	2	4
Pu-239	151240	4	97	129
Pu-238	2855368	5	10	7
Cs-134	0.64	6	6	25
Eu-154	0.29	7	4	3
Pu-240	9365050	8	11	13
Eu-152	0.39	9	3	10
Cm-244	697800	10	13	16
Cm-243	213	11	28	22
Cm-247	10	12	9	81
Pu-243	1396	13	17	85
Th-229	4435	14	39	21

‡ Source: Crawford and Taggart 2007, Table 3

Using Table 7-9, it is seen that 7 of the 14 C&T-selected radionuclides would also be selected as being significant, based on the 2008 inventory (INV-PA-08) and FGR 12 ranking. The remaining seven radionuclides selected by the 2008 BI/FGR 12 ranking, but not selected by C&T, were Bi-214(5), Tl-208(6), Pb-214(8), Bi-212(9), Pb-212(11), Sb-126m(12), and Eu-155(14).

Additional MicroShield runs were made for Bi-214, Tl-208, Pb-214, Bi-212, Pb-212, Sb-126m, and Eu-155 to determine their limiting drum activities. Likewise, in order to get an idea of the potential contribution of each significant radionuclide from INV-PA-08 to the RH TRU shielded container surface dose rate, the maximum activity loadings for all 21 C&T and 2008 BI/FGR 12 radionuclides were compared to the average 30-gallon drum loading, based on the volume-averaged RH TRU waste concentration given in INV-PA-08, Table A-1. As Table 7-10 shows, none of the average radionuclide activity loadings exceed their MicroShield calculated maximum activity loading (i.e., the RH average drum activity in column 5 divided by the limiting curies for a 200 mrem/hr dose in column 3 is less than unity, as shown in column 6).

Table 7-10: Surface Dose Limiting Versus Volume-Averaged Activity Loading – 2008 Baseline Inventory

Nuclide	2008 BI FGR 12 Rank	Limiting Curies per 200 mrem/hr‡		RH Conc (Ci/m^3)†	RH Drum Activity (Ci)††	Drum Activity/Limit
Cs-137	1	2	C&T	12.6	1.4238	0.7119
Am-241	2	208855	C&T	0.633	0.071529	3.42E-07
Eu-154	3	0.29	C&T	0.0113	0.001277	0.004403
Co-60	4	0.12	C&T	0.00173	0.000195	0.001629
Bi-214	5	1.48E-01	Added	0.00256	0.000289	0.001961
Tl-208	6	0.053905	Added	0.00102	0.000115	0.002138
Pu-238	7	2855368	C&T	0.722	0.081586	2.86E-08
Pb-214	8	3.62E+01	Added	0.00256	0.000289	7.99E-06
Bi-212	9	1.77E+00	Added	0.00285	0.000322	0.000182
Eu-152	10	0.39	C&T	0.000377	4.26E-05	0.000109
Pb-212	11	1.06E+06	Added	0.00284	0.000321	3.02E-10
Sb-126m	12	7.45E-01	Added	0.000233	2.63E-05	3.53E-05
Pu-240	13	9365050	C&T	0.14	0.01582	1.69E-09
Eu-155	14	6.5E+10	Added	0.00449	0.000507	7.81E-15
Cm-244	16	697800	C&T	1.23	0.13899	1.99E-07
Th-229	21	4435	C&T	0.00385	0.000435	9.81E-08
Cm-243	22	213	C&T	0.0271	0.003062	1.44E-05
Cs-134	25	0.64	C&T	0.132	0.014916	0.023306
Cm-247	81	10	C&T	1.99E-11	2.25E-12	2.25E-13
Pu-243	85	1396	C&T	1.97E-11	2.23E-12	1.59E-15
Pu-239	129	151240	C&T	1.1	0.1243	8.22E-07

‡ Source: C&T = Crawford and Taggart 2007, Table 3; Added = this analysis

† Source: INV-PA-08, Table A-1

††Product of RH Conc and volume of 30-gallon drum

Similar to the PABC-2004 inventory (DOE 2006c), even though the average Cs-137 concentration based on the INV-PA-08 inventory is below what would be allowable in a shielded container, the individual RH TRU waste stream Cs-137 concentrations range from <0.001 to 3,140 Ci/m^3 (INV-PA-08, Appendix B). Therefore, it is expected that the shielded container would not be usable for some fraction of the RH TRU waste. Figure 7-3 shows that about 97% of the final form waste volume from INV-PA-08, Appendix B, could be packaged in shielded containers, assuming Cs-137 is the limiting radionuclide.

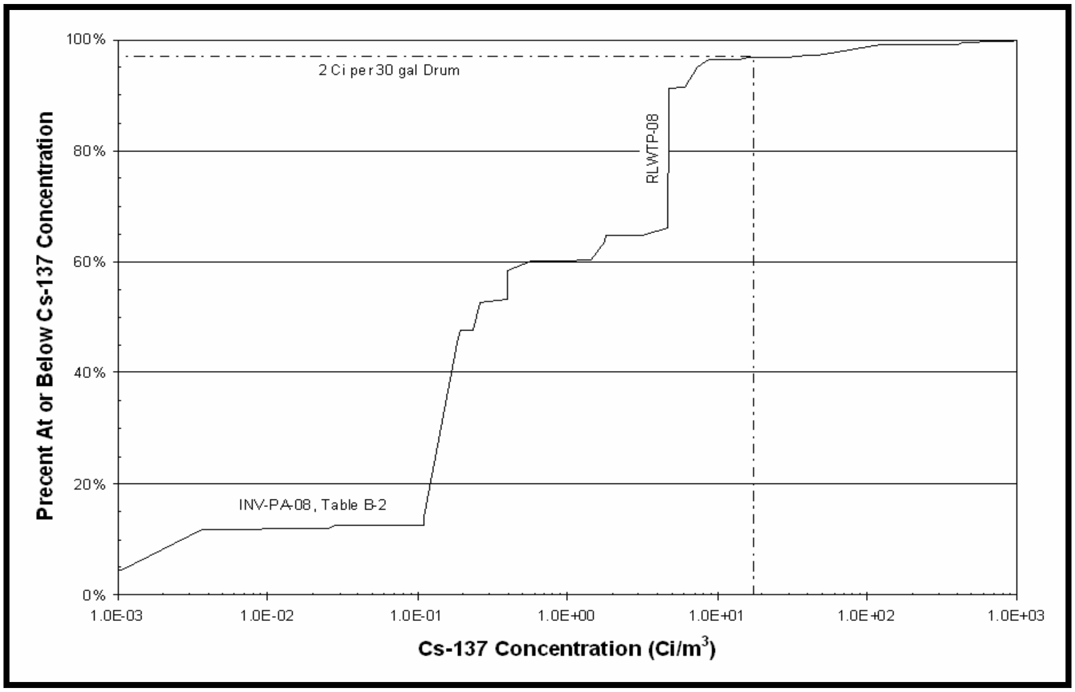

Figure 7-3: Fraction of RH TRU Waste Streams At or Below the Limiting Cs-137 Concentration Based on 2008 Inventory

Both PABC-2004 and INV-PA-08 assumed that the disposal volume of RH TRU is limited to 7,080 m^3, which is consistent with the DOE agreement with the state of New Mexico (DOE 1981).

In INV-PA-08, waste stream RLWTP-08 at an estimated volume of 1,777.74 m^3 contributes 25% to the total RH waste volume, and with an estimated inventory of 8,430 Ci has a Cs-137 concentration of 4.7 Ci/m^3 or 0.54 Ci per 30-gallon drum, which is well below the drum limit of 2 Ci. Waste stream RLWTP-08 is not listed in PABC-2004; instead waste stream RP-W016 at an estimated 3,943.6 m^3 contributes 54% to the total waste volume, and has a Cs-137 concentration of 19.1 Ci.m^3 or 2.16 Ci per drum, which is above the drum limit for 200 mrem/hr. Waste stream RP-W016 is not listed in INV-PA-08, because the Hanford Office of River Protection tank wastes were reclassified as potential wastes. Waste stream RLWTP-08 was a newly identified waste stream at the time the year-end 2006 inventory was prepared. It projected RH debris from the not-yet-operational Hanford Waste Treatment Plant.

Clearly, the radionuclide inventory of RH TRU waste is very uncertain (as evident from the large changes shown by Table 7-7 and Table 7-8), which leads to uncertainty in the amount of RH TRU waste that can ultimately be packaged in shielded containers. Thus, the amount of RH TRU estimated to be disposed of at WIPP should be re-examined for each annual inventory estimate.

Nonetheless, we believe that the MicroShield analyses performed by DOE in support of the shielded container PCR are well-documented and appropriate for the intended use in scoping those RH TRU waste streams that are candidates for disposal in shielded containers.

8.0 IMPACT OF SHIELDED CONTAINERS ON REPOSITORY CHEMICAL CONDITIONS

The use of shielded containers for RH waste disposal in WIPP will significantly increase the quantities of steel and lead, and will slightly increase the amount of CPR in the repository (Section 6.0). The potential effects of these changes on repository chemistry were addressed by Dunagan et al. (2007). The chemistry-related processes that could be affected by use of the shielded containers in the repository include anoxic corrosion gas generation, gas generation caused by CPR degradation, redox conditions after repository closure, carbon dioxide (CO_2) consumption, and complexation of actinides by organic ligands. These processes are discussed in the following sections.

8.1 Anoxic Corrosion

Anoxic corrosion of iron-bearing alloys (i.e., steel) is expected to begin in the WIPP shortly after closure, following consumption of the limited amount of oxygen remaining in the repository by metal corrosion or aerobic degradation of CPR. Anoxic corrosion of iron in the waste containers and waste is expected to produce hydrogen gas (H_2) through the reactions:

$$Fe + (x+2)\ H_2O \rightarrow Fe(OH)_2 \bullet xH_2O + H_2$$
$$Fe + H_2S \rightarrow FeS + H_2$$

Corrosion of other metals in the repository could also generate H_2. For example, anoxic corrosion of lead could take place via the reactions (Wall and Enos 2006):

$$Pb + H_2O \rightarrow PbO + H_2$$
$$3\ PbO + H_2O \rightarrow Pb_3O_4 + H_2$$

Aluminum in the repository could also corrode and generate hydrogen via reactions such as:

$$Al + 3\ H_2O \rightarrow Al(OH)_3 + 1.5\ H_2$$

Anoxic corrosion of metals such as aluminum and lead in the waste were assumed to be insignificant for the CCA/PAVT and CRA-2004/PABC-2004 gas generation calculations, because of the small amounts of these metals in the repository compared to the large amounts of steel and iron in the waste and waste containers (Wilson et al. 1996).

Dunagan et al. (2007) considered the likely effects of the shielded containers on the amounts of steel and lead in the repository. However, the potential effects of these increased quantities of corrodible metal on anoxic corrosion gas generation rates were not addressed.

Although Dunagan et al. (2007) did not discuss the potential effects of increased quantities of corrodible metal on gas generation, this issue was considered during EPA's review of AMWTF compressed waste disposal (TEA 2004). At the request of EPA, Stein and Zelinski (2004) performed a series of BRAGFLO calculations to assess the effects of higher iron surface area and resulting anoxic corrosion rates on total gas generation, gas pressure, and brine saturation.

Calculations were carried out using the previously assumed 6 m^2 value for ASDRUM, as well as values 2.2 or 10 times this initial value. The higher assumed surface areas resulted in higher gas generation rates from anoxic corrosion and lower brine saturations. These lower brine saturations led to lower rates of gas generation from microbial degradation of CPR and anoxic corrosion. The increased values of ASDRUM resulted in slightly higher total gas generation and pressures during the first 1,500 to 2,500 years of the repository regulatory period, followed by slightly lower total gas generation and pressures. TEA (2004) agreed with the assessment by Stein and Zelinski (2004) that total repository releases are unlikely to significantly increase due to an increased iron surface area, and may decrease because of lower long-term pressures and brine saturations.

It should be noted that the Stein and Zelinski (2004) calculations were performed using microbial gas generation rates and probabilities assumed for the CCA PAVT. For the PABC-2004, long-term microbial degradation rates were assumed to be lower than in the CCA PAVT, and the probability of significant degradation of cellulosics in the repository was increased from 50% for the CCA PAVT to 100% for the PABC-2004. It is likely that the conclusions reached by Stein and Zelinski (2004) remain valid, despite these changes in microbial gas generation parameters for the PABC-2004. This assumption is reinforced by the fact that in none of the PABC-2004 realizations was all, or even most, of the iron consumed in 10,000 years (Nemer and Stein 2005, Figure 6-10). If increased iron in the repository from shielded containers resulted in increased iron surface area, the initial corrosion rate and gas generation rate would be higher. However, the process would be self-limiting, because increased repository pressure would prevent additional brine inflow and anoxic corrosion and microbial degradation would cease.

8.2 CPR Degradation

Use of shielded containers for RH waste disposal will not affect the CPR content of the waste, but will increase the amounts of emplacement cellulosics and plastics; emplacement of the shielded containers will increase the total CPR inventory by about 2% (Section 5.0). Dunagan et al. (2007) concluded that this relatively small increase in CPR inventory would not significantly affect the results of PA, based on previous PA calculations that investigated the effects of a 2.5-fold increase in CPR inventory (Dunagan et al. 2005). Although this PA (i.e., Dunagan et al. 2005) was carried out using the microbial degradation probability and rate parameters previously used for the PAVT and not the parameters used for the later PABC-2004, this conclusion should still be valid for the current, smaller microbial degradation rates. Consequently, because of the very small change in CPR inventory, use of the shielded containers is unlikely to have a significant effect on microbial gas generation in the repository.

8.3 Redox Conditions

Oxic corrosion of iron and aerobic microbial CPR degradation in the repository are expected to consume oxygen, creating relatively reducing repository conditions within 100 years of closure (DOE 2004a, Appendix PA Attachment SOTERM). Anoxic iron corrosion and anaerobic microbial degradation are expected to occur subsequently and maintain these reducing conditions. The assumption of reducing conditions in the repository is used qualitatively to constrain the oxidation states of the actinides in PA.

Because the lead inventory was relatively small compared to the large amounts of iron in the waste and waste containers, the potential effects of lead corrosion on repository chemistry were considered to be less important than iron for the PAVT and the PABC-2004. As summarized by Dunagan et al. (2007), Wall and Enos (2006) found that lead was likely to undergo anoxic corrosion, forming phases such as PbO and Pb_3O_4. Consequently, an increased amount of lead is likely to contribute to the maintenance of reducing conditions.

The presence of iron and lead in the repository is expected to have beneficial effects by contributing to the establishment and maintenance of reducing conditions. The additional iron and lead introduced by the shielded containers are not expected to have significant effects on redox conditions in the repository.

8.4 Carbon Dioxide Consumption

Additional CPR in the repository from the plastics and cellulosic materials used to emplace the shielded containers could result in the production of additional CO_2 by microbial degradation. The effects of CO_2 produced by microbial degradation of CPR are mitigated in the repository by the use of MgO backfill. The small increase in total CPR inventory will be addressed by the addition of sufficient MgO to maintain the required 1.2 Excess Factor, which is the moles of MgO in the repository divided by the moles of carbon in the CPR (Reyes 2008).

Iron and lead in the repository could react with CO_2, forming iron- and lead-carbonate phases (Dunagan et al. 2007):

$$Fe + H_2O + CO_2 \rightarrow FeCO_3 + H_2$$
$$Pb + H_2O + CO_2 \rightarrow PbCO_3 + H_2$$

However, iron sulfides and possibly lead sulfides are expected to be more stable than the carbonate phases under long-term repository conditions (Wall and Enos 2006). Consequently, DOE did not consider the consumption of CO_2 by iron or lead in the SCPA, which is appropriate given the uncertainties regarding the formation of iron- and lead-carbonate phases under repository conditions.

8.5 Complexation of Actinides by Organic Ligands

Dunagan et al. (2007) reviewed PABC-2004 calculations carried out to identify the likely effects of organic ligands on actinide solubilities and stated that these calculations showed that organic ligands will not form complexes with the +III and +IV actinides to a significant extent under expected WIPP conditions. Contrary to this statement, EPA (2006) previously reviewed the actinide solubility calculations cited by Dunagan et al. (2007) and concluded that the +III and +V actinide solubilities increased significantly with increased ligand concentrations because of complexation by EDTA and oxalate, respectively.

Competition of Fe^{2+} and Pb^{2+} with actinides for organic ligand binding sites may decrease the effects of organic ligands on actinide solubilities. However, increased inventories of iron and lead in the repository may not increase Fe^{2+} and Pb^{2+} concentrations in repository brines if the concentrations of these species are controlled by an equilibrium process. Regardless of the effects of metals inventories on dissolved metals concentrations, higher inventories of iron and lead resulting from the use of the shielded containers will not increase actinide solubilities.

8.6 Conclusions Regarding Repository Chemistry Effects of Shielded Containers

The principal effects of the shielded containers on WIPP repository chemistry would be expected to result from the relatively large increases in the inventories of iron and lead and relatively small increases in cellulosics and plastics from the emplacement materials. Processes such as gas generation from anoxic corrosion of metals and microbial degradation of CPR, establishment of reducing conditions by metals corrosion and CPR degradation, CO_2 consumption by reaction with iron and lead, and competition of aqueous iron and lead species for organic ligand binding sites with actinides were qualitatively evaluated. The results of this evaluation indicate that the increased iron, lead, and CPR inventories are not expected to have significant negative impacts on PA. In fact, previous PA calculations carried out to assess the effects of increased iron surface areas on repository releases indicate that higher iron surface areas would be expected to increase initial gas generation rates and brine consumption, thereby limiting total gas generation and repository releases because of lower brine saturation.

9.0 TEMPERATURE EFFECTS OF RH TRU IN SHIELDED CONTAINERS WASTE EMPLACEMENT

Elevated temperatures in the repository have been the focus of previous studies, because the of the potential for heat generation by the waste to impact repository closure rates. In the PABC-2004, temperature effects due to heat generation by the waste were screened out due to the low consequence demonstrated by these studies. Based on a steady-state heat transfer model, DOE demonstrated that when RH TRU waste was placed in boreholes in the disposal room walls, the temperature rise from a distance of about 30 m into the salt to the canister surface was 3°C, assuming that the canister was filled with RH TRU waste having a volumetric heat load of 71 W/m^3 (Sanchez and Trellue 1996, Table 12). Sanchez and Trellue (1996) used inverse shielding calculations to establish a conversion factor of 0.0037 W/Ci for RH TRU waste, based on the February 1995 WIPP Baseline Inventory Report. If these authors had used a volumetric heat load based on the volumetric activity limit of 23 Ci/L set by the WIPP LWA, the volumetric heat generation would have been 84 W/m^3 ($23\ Ci/L \times 1000\ L/m^3 \times 0.0037\ W/Ci$) and the temperature rise would have been about one-half degree higher.

The results of Sanchez and Trellue (1996) were updated by Djordjevic (2003) to reflect inventory information available in TWBID Revision 2.1, Version 3.12 (LANL 2003a, 2003b). Based on the revised information, the conversion factor adjusted for radioactive decay to 2001 was 0.00315 W/Ci (Djordjevic 2003, Table 5). Thus, the temperature rise from the salt to a canister in the wall would be about 3°C, based on the maximum allowable curie density of 23 Ci/L.

These small temperature changes will decrease rapidly with time as the energetic gamma radiation associated with shorter-lived radionuclides decays. For example, the radioactivity associated with gamma rays in RH TRU waste was 1.306×10^6 Ci in 2001, and this would fall to 5.587×10^4 Ci by 100 years after repository closure (2133) (Djordjevic 2003, Table 3). Based on the inventory used for PABC-2009 contained in PAIR 2008, the value at 2133 would be 3.18 $\times 10^4$ Ci, indicating an even more rapid temperature decay than that based on earlier inventories. The major sources of energetic gammas at closure are Ba-137m, Cs-137, Pu-241, Sr-90 and Y-90 (Djordjevic 2003, Table 3). One hundred years after closure, the major sources are Ba-137m, Cs-137, Sr-90 and Y-90 (Djordjevic 2003, Table 3). The contribution of Pu-241 with a half-life of 14.4 years becomes minor after 100 years.

In its PCR, DOE did not specifically address the temperature situation when shielded containers are placed on the floors of disposal rooms, but instead drew analogies to the minimal temperature impact and low consequence of emplacing canisters in the wall. Since stakeholders have expressed a concern that the lead liner in a shielded container might become overheated, a bounding calculation has been made here to show that this is not likely. For this calculation, it is assumed that a shielded container is located on the floor of a disposal room, and that steady-state heat transfer occurs from a uniform volumetric heat source through the walls of the container into the surrounding air.

In Table 9-1, the RH TRU curie contents for selected radionuclides taken from the 2007 Annual Transuranic Waste Inventory Report (ATWIR) (DOE 2007c) were converted to watts using conversion factors developed by Djordjevic (2003). The last four columns of Table 9-1 compare various heat generation criteria. The limiting criterion (i.e., the criterion that results in the lowest volumetric heat generation rate) is highlighted for each nuclide. From this summary, it can be determined that the highest possible heat generation rate is 184 W/m^3 (for Am-241, Cm-244, etc.). This value is used to calculate the temperatures in shielded containers.

Since the surface dose rate is limited to 200 mrem/hr for CH TRU waste, this imposes a limitation on the energetic gamma radiation in the shielded container. Crawford and Taggart (2007) determined the maximum quantity of radioactivity for dominant gamma emitters in RH TRU waste that could be emplaced in a shielded container and not exceed the 200 mrem/hr dose rate (see column 4 of Table 9-1). However, this may not establish the maximum heat generation rate. In addition, the waste must meet the 23 Ci/L (2.3×10^4 Ci/m^3) statutory requirement (see column 5 of Table 9-1). Consider, for example, Cs-137. The allowable Cs-137 content in a shielded container that meets the statutory 200 mrem/hr surface dose rate limit is 2 Ci and, since the volume of a 30-gallon drum is 0.113 m^3, the curie density is 18 Ci/m^3. This is substantially below the limit of 2.3×10^4 Ci/m^3. The waste must also meet decay heat limitations. The CH-TRAMPAC sets 40 watts (W) as the maximum decay heat load (DOE 2005, Table 5.2-1). If CH TRU waste with this decay heat limit were placed in a 55-gallon container, the volumetric heat generation rate would be 184 W/m^3 (see column 9 of Table 9-1). Still another limitation is the fact that the total radioactivity associated with several of the selected energetic gammas may be less than what would provide a surface dose rate of 200 mrem/hr. Using a bounding assumption that all the radioactivity associated with each selected radionuclide is placed in a 30-gallon drum, it can be seen that the volumetric heat generation is miniscule for radionuclides such as Cm-247 and Pu-243 (see column 6 of Table 9-1).

Table 9-1: Radioactivity and Heat Loading of Dominant Gamma Emitting Radionuclides in RH TRU Waste

Nuclide	RH TRU radioactivity (Ci)	Conversion factor (watts/Curie)	Max. activity for 200 mrem/hr (Ci)	30-gal drum curie density (Ci/m^3)	Heat gen. rate based on RH TRU Inventory (W/m^3)	Heat gen. rate based on 2.3E04 Ci/m^3 limit (W/m^3)	Heat gen. rate based on 200 mrem/hr (W/m^3)	Heat gen. rate based on decay heat limit of 40 watts (W/m^3)
Am-241	2.79E+04	0.0333	2.089E+05	1.8E+06	8220	766	61537	184
Cm-243	1.92E+02	0.0367	2.130E+02	1.9E+03	62	843	69	184
Cm-244	8.70E+03	0.0350	6.978E+05	6.2E+06	2696	806	216270	184
Cm-247	1.41E-07	0.0141	1.000E+01	8.8E+01	0.000000018	325	1	184
Co-60	1.09E+03	0.0153	1.200E-01	1.1E+00	148	353	0.02	184
Cs-134	9.33E+02	0.0102	6.400E-01	5.7E+00	84	234	0.1	184
Cs-137	3.39E+06	0.0011	2.000E+00	1.8E+01	33148	25	0.02	184
Eu-152	2.27E+04	0.0076	3.900E-01	3.5E+00	1520	174	0.03	184
Eu-154	5.30E+03	0.0090	2.900E-01	2.6E+00	420	206	0.02	184
Pu-238	2.25E+04	0.0332	2.855E+06	2.5E+07	6602	763	837858	184
Pu-239	3.18E+04	0.0308	1.512E+05	1.3E+06	8667	708	41222	184
Pu-240	2.12E+04	0.0311	9.365E+06	8.3E+07	5836	715	2578053	184
Pu-243	1.39E-07	0.0012	1.396E+03	1.2E+04	0.000000001	27	14	184
Th-229	2.73E+01	0.0306	4.435E+03	3.9E+04	7	704	1201	184

Data in column 2 are from ATWIR 2007 (DOE 2007c, Table 3-15, decayed through 2006). Data in column 3 are from Djordjevic 2003. Data in column 4 are from Crawford and Taggart 2007. Data in column 5 are obtained by dividing column 4 data by volume of 30-gallon drum (0.113 m^3). Data in column 6 are obtained by dividing the product of columns 2 and 3 by the volume of a 30-gallon drum. Data in column 7 are obtained by multiplying data in column 3 by 2.3×10^4 Ci/m^3. Data in column 8 are obtained by dividing the product of columns 3 and 4 by the volume of a 30-gallon drum. Data in column 9 are obtained by dividing decay heat limit of 40 watts by volume of 55-gallon drum (0.217 m^3). Limiting values for each radionuclide are highlighted.

As noted above, the shielded container concept involves placing a 30-gallon drum of RH TRU waste inside a shielded container consisting of a 3/16-in inner steel wall, a 1-in thick lead liner and a 1/8-in thick outer steel wall. The temperature change in the interior of the 30-gallon drum, assuming a uniform volumetric heat source, is given by the equation:

$$\Delta T = q/4\pi k_w$$

where k_w is the thermal conductivity of the waste and q is the linear power density; q is related to the volumetric heat generation rate, Q, by the equation $q = Q \times \pi r_w^2$, where r_w is the inner radius of the 30-gallon drum (0.231 m). A volumetric heat generation rate of 184 W/m^3 is equivalent to linear power density of 39 W/m.

The temperature drop associated with radial conductive heat transfer through each of several layers of the shielded container waste package can be expressed by the equation:

$$\Delta T = q/2\pi r_i \times (t_i/k_i)$$

where r_i is the inner radius of the *i*th container layer with thickness t_i and conductivity k_i.

There is an additional temperature drop that occurs from the outer surface of the shielded container to the bulk coolant in the disposal room (assumed here to be air[10]). The temperature drop associated with this surface film is given by the equation:

$$\Delta T = q/2\pi r_o h_s$$

where h_s is the surface heat transfer coefficient and r_o is the outer radius of the shielded container.

Using these equations and the parameters listed in Table 9-2, the temperature rise from the ambient repository temperature to the inner wall of the shielded container is about 2˚C.

Table 9-2: Parameters Used to Calculate Temperature Rise in Shielded Containers

Parameter	Value	Units
Volumetric heat generation rate	184	W/m^3
Linear power density	39	W/m
30-gallon drum inner radius	0.231	m
30-gallon drum wall thickness	0.00121	m
Air space between 30-gallon drum and SC	0.0257	m
SC inner wall thickness	0.00474	m
SC lead liner thickness	0.0253	m
SC outer wall thickness	0.00317	m
SC outer radius	0.291	m
Thermal conductivity – iron	80	W/m-K @ 300K

[10] At some point in time after repository closure, the oxygen will have been consumed by anoxic corrosion leaving a nitrogen-rich atmosphere, but the thermal properties of nitrogen and air are essentially the same.

Table 9-2: Parameters Used to Calculate Temperature Rise in Shielded Containers

Parameter	Value	Units
Thermal conductivity – lead	35	W/m-K @ 300K
Thermal conductivity – air	0.0262	W/m-K @ 300K
Heat transfer coefficient – air	10	W/m^2-K (natural convection)

The preceding analysis was based on bounding calculations determined by various statutory and/or regulatory limits. For additional perspective, the list of 43 RH TRU waste streams that are candidates for disposal in shielded containers was examined to determine the waste stream with the highest Am-241 content (Crawford and Taggart 2007, Table 10). Hanford waste stream RL-W663 (S5420 heterogeneous waste) contains 42.5 Ci/m^3 of Am-241, as well as 5 Ci/m^3 of Cs-137. As shown in Table 9-3, this waste stream has a volumetric heat generation rate of 1.7 W/m^3, a value well below that used for the bounding calculations.

Table 9-3: Heat Generation in RH TRU Waste Stream with Highest Am-241 Content Being Considered for Shielded Container Disposal

Radionuclide	Waste Stream RL-W663		
	(Ci/m^3)	**Ci/drum**	**Watts/drum**
Am-241	4.25E+01	4.80E+00	1.60E-01
Cs-137	5.04E+00	5.70E-01	6.29E-04
Pu-238	6.78E+00	7.66E-01	2.54E-02
Pu-239	3.80E-01	4.29E-02	1.32E-03
Pu-240	4.90E-01	5.54E-02	1.72E-03
Pu-241	2.13E-03	2.41E-04	7.43E-09
Pu-242	2.44E-04	2.76E-05	8.19E-07
S-90	3.07E+00	3.47E-01	4.05E-04
Y-90	3.07E+00	3.47E-01	1.92E-03
Ba-137m	5.04E+00	5.70E-01	2.23E-03
U-235	6.61E-06	7.47E-07	1.96E-08
Total (watts/drum)			1.94E-01
Total (w/m^3)			1.71E+00

Note: Cs-137/Ba-137m and Sr-90/Y-90 assumed to be in secular equilibrium

10.0 SURFACE DOSE OF LESS THAN 200 MREM/HR AND MEASUREMENT UNCERTAINTY

This section discusses measurement procedures and their attendant uncertainties in demonstrating compliance with the surface dose rate limit of not greater than 200 mrem/hr for CH TRU waste established by WIPP LWA (PL 102-579).

10.1 Terminology Clarification

The parameter that classifies TRU waste as RH or CH, as required in the LWA, is the surface dose rate expressed in units of mrem per hour. This criterion is found in the subsequent DOE documents that flow down from the LWA, such as the RH Waste Characterization Program Implementation Plan (WCPIP) (DOE 2003)[11] and the WIPP waste acceptance criteria (DOE 2007a).

While a technical discussion of the interactions of radiation and matter is beyond the scope of this report, there are three units related to ionizing radiation that bear defining:

- *Dose rate* is a measure of the energy deposited per unit mass of absorbing material per unit time, and can include contributions from all types of ionizing radiations; alpha, beta, gamma and neutron. Dose rate is measured in units of rads or the SI equivalent Grays (Gys) per unit time.

- An external *exposure rate* is a measure of the charge created by the ionizing radiation within a given volume or air. Exposure is defined only for x-ray or gamma rays, by definition, and exposure rate is expressed in Roentgens per unit time (R/hr) or the SI equivalent coulombs per kilogram (C/kg).

- A *dose equivalent rate* is a measure of the probable biological effects of a given radiation exposure and can include contributions from all types of ionizing radiations—alpha, beta, gamma and neutron—and includes consideration of the radiation's biological effects. It is derived by multiplying the dose rate by a quality factor (QF) to adjust for the differences in energy transfer among the types of ionizing radiation. Dose equivalent rate is expressed in units of rem per unit time (e.g., rem/hr) or the SI equivalent sievert (Sv) per unit time.

Assuming that a waste container's contents are TRU, the sole determinant of the container's status as CH or RH is its "surface dose rate," defined in Section 2.0 Definitions of the LWA as being not greater than 200 mrem/hr. Guidelines for radiation exposure limits to personnel are typically quoted in units of dose equivalent (rem) in order to place exposures to different types and energies of radiation on a common basis. It is common usage to express the surface dose rate criterion in mrem/hr, even though it is more precise to refer to this as a surface dose equivalent rate.

Practically speaking, the measurable radiation associated with the bulk of packaged TRU wastes encountered throughout the DOE complex consists predominantly of gamma emissions, although a few waste types have large neutron components.[12] DOE has stated that the shielded containers will not be used for "neutron emitting RH TRU waste" (Moody 2008), and that main gamma-

---

[11] DOE/WIPP-02-3122 initially used similar language as the LWA and the CH-TRAMPAC, but this was revised in 2002 and Revision 1 correctly uses the term *dose equivalent rate* expressed in Rem (DOE 2007a).

[12] Due to the nature of beta radiation, it is largely absorbed by the waste matrix itself, and the waste container does not contribute to measurements at a container's surface. DOE has stated (Moody 2008) that the 1-in thick lead shield will ensure there is no beta component to the container's surface dose rate.

emitting radionuclide constituents of the wastes will be Cs-137 and minor amounts of Am-241 and Co-60. This is consistent with what EPA has observed during inspections of RH TRU generator sites to date. If a waste container exhibits only photon (gamma) radiation, it is appropriate to consider a measured exposure rate in mR/hr to be nearly the same as a dose equivalent rate in mrem/hr. The specific numerical adjustment is assigning a QF of 0.95 to convert the exposure in air (mR) to a tissue dose (equivalent) in mrem, as stated by DOE (Moody 2008).

Site Health Physics (HP) or radiation control (Rad Con) personnel routinely monitor waste containers during storage and movement using a variety of gamma and neutron-sensitive instruments. In practice, the units of exposure rate, dose equivalent rate, and dose rate are used interchangeably, a convention that is common throughout this industry. Due to the nature of the majority of TRU wastes, i.e., predominantly gamma-emitting materials, the terms do not differ significantly in magnitude.

10.2 Current Practices for TRU CH-RH Classification at DOE TRU Generator Sites

10.2.1 Surface Dose Rate Instrumentation

The instruments used to conduct surface dose rate surveys at CH and RH TRU generator sites are typical of the portable beta-, gamma-, and neutron-sensitive instruments in common use throughout the nuclear industry. They represent most manufacturers of radiation detection equipment (i.e., Ludlum, Eberline, ORTEC, Fluke), and both older analog and newer digital units are in widespread use.

Beta-gamma: Typical beta-gamma sensitive instruments are rate meters with Geiger-Mueller (GM) tubes or an equivalent probe that responds to beta particles above a specific energy and gamma radiation, and provide output in events per unit time, i.e., counts per minute (cpm), and/or mR/hr, depending on the calibration. As stated previously, beta particles are usually attenuated by the waste matrix/container and do not generally contribute to the waste container's surface dose rate. These meters are most often used for contamination control surveys.

Gamma: Typical gamma-sensitive instruments are rate meters with an interchangeable gamma-sensitive (sodium iodide [NaI]) probe, single-body units with an internal probe, and air ionization (ion) chambers. All of these units typically provide output in exposure (mR/hr), which is assumed to be equivalent to mrem/hr (for photons measured in air). These meters are best suited to measure the gamma contribution of a surface dose rate survey for TRU waste containers. The gamma instruments used for the Dose-to-Curie (DTC)[13] determination for RH TRU containers typically consist of a rate meter located outside of the shielded DTC enclosure or hot cell that is attached to a probe (ion chamber) held in a fixed geometry relative to the waste container being assayed within the DTC enclosure.

[13] Dose-to-Curie is a radiological characterization technique where the surface dose rate of a waste container is correlated to concentrations of specific gamma-emitting radionuclides. This technique is described in the procedure CCP-TP-504. DTC is currently used to characterize RH TRU wastes at INL, ANL and ORNL, BCLDP and GE VNC.

Neutron: Typical neutron detection units, commonly called Rem Balls, consist of a rate meter body attached to a circular boron trifluoride (BF_3) detector filled with tissue-equivalent material that, when calibrated, provides a reading in Rem. These tend to be the largest portable instruments in common use. For the Idaho National Laboratory Central Characterization Project (INL-CCP)[14] RH waste characterization program, site HP personnel operated a Rem Ball and gamma survey meter alongside the INL-CCP ion chambers within the DTC enclosure. The Argonne National Laboratory (ANL-CCP) and Oak Ridge National Laboratory (ORNL-CCP) RH waste characterization programs used neutron-sensitive instruments whose detectors were located within the DTC hot cell or assay enclosure and were cabled to digital display units outside the enclosure, allowing operators to easily read their output. These units provided the neutron component of the surface dose rate for the ANL-CCP and ORNL-CCP RH waste containers. The WIPP site uses Rem Balls to make hand measurements of surface dose rates for waste payloads (for example, 14 drum payloads) to assure adequate worker health and safety.

These same types of instruments would be used to survey shielded containers.

10.2.2 Current Surface Dose Rate Survey Practices

Procedures that control the surface dose rate determinations vary by site, although all site measurement and test equipment (M&TE) programs must be compliant with the CBFO QAPD (DOE 2004b). The QAPD identifies broad requirements and cites documents such as American Society for Testing and Materials (ASTM) methods and American National Standards Institute (ANSI) as appropriate guidance. A standard of particular interest is ANSI N323A, which is cited in the CCP DTC procedure (CCP 2008a) and the procedure for calibrating the RH gamma survey instruments used by INL-CCP (see Section 10.2.4). ANSI N323A specifies a variety of M&TE requirements related to survey meter calibration, surface dose rate operation, and documentation.

Regarding EPA's concern that all shielded containers have surface dose rates less than 200 mrem/hr, DOE has stated that they will "employ a system of controls for packaging and measuring that ensures a very high degree of confidence that this requirement is met" (Moody 2008). Differences have been observed among the surface dose rate practices at CH TRU generators. This may be due in part to the assumption that the likelihood of a container exceeding a surface dose rate of 200 mrem/hr after having been handled, staged, characterized, and repeatedly surveyed, is small, as discussed in Section 10.2.5. However, a container's contents may shift during handling, and a single or small number of "hot" items within a drum can produce an enhanced measurable surface dose rate by virtue of a positional change. Consistency among the performance of surface dose rate surveys is desirable in general and of greater importance for the use of shielded containers.

---

[14] The Central Characterization Project (CCP) operates CH and RH TRU waste characterization programs at several TRU generator sites. CCP currently operates all RH TRU waste characterization facilities, and these are referred to by the generator site abbreviation followed by CCP, e.g., INL-CCP, ORNL-CCP, ANL-CCP and LANL-CCP.

10.2.3 Modifications to Surface Dose Rate Survey Procedures

Apart from DOE's statement in Moody 2008 that they will "ensure that all shielded containers exhibit a surface dose rate less than 200 mrem/hr when packaged for disposal," the specific surface dose rate expectations of shielded containers are unclear. Since a single design has been proposed, it will not be possible to increase a container's shielding to reduce the surface dose rate. Prospective control of each shielded container's surface dose rate consists of limiting the amounts of Cs-137, Co-60, Am-241 or other gamma contributors that are packaged in each shielded container.

EPA believes that the waste packaging and surface dose rate procedures at all DOE sites that plan to use shielded containers should undergo evaluation and revision to ensure a high degree of standardization. Specifically, each generator site using shielded containers should address or incorporate in its procedures the following:

- A minimum number of survey points on each waste container and the specific measurement geometry should be specified, including the container's top and bottom, for all surface dose rate procedures at all generator sites. Documentation for each survey should include a diagram of all survey points.
- A reproducible geometry should be employed. Using a device that consistently positions the detector the same way relative to the waste container would be helpful, i.e., a stand or jig that holds the detector and waste container in the same configuration for each measurement, much like is done currently for RH wastes at INL, ANL, ORNL and VNC-CCP.
- An explicit requirement stating that the highest observed reading must be used for the container's measurement of record must be established.
- The radionuclide (Cs-137, Co-60, Am-241, or other large gamma contributor) content of each container should be planned or adjusted during container loading to ensure that surface dose rates are less than 200 mrem/hr, based on container-specific data derived from DTC, other radiological characterization of the 30-gallon drum prior to placement within the shielded container or Acceptable Knowledge (AK). This approach is implied, but not explicit, in DOE's statement that it would "employ a system of controls for packaging and measuring that ensure a very high degree of confidence" that all shielded containers exhibit surface dose rates less than 200 mrem/hr (Moody 2008).
- A determination regarding a container's lack of neutron-emitting TRU waste based on measurement of the 30-gallon drum prior to placement within the shielded container should be made and documented with an appropriately calibrated and controlled neutron-sensitive instrument. This information should be included as part of each container's Batch Data Report (BDR) and be available for EPA inspection during routine and unscheduled site inspections.
- Surface dose rate survey records should be included as part of each container's BDR and be available for EPA inspection.

- The procedures should also address site calibration facilities and training of surface dose rate survey personnel specifically in handling of SCs.
- A formal CBFO procedure addressing these concerns should be initiated and implemented complex-wide as a replacement of, or supplement to, the current site-specific surface dose rate survey procedures.

10.2.4 CH TRU Wastes

The manner in which CH waste containers are stored and managed at DOE generator sites is based on historical information regarding their contents and/or generation processes (AK). Containers are stored, staged, assigned to waste streams, and ultimately assayed according to the available AK for those wastes. The management of CH containers involves survey measurements taken multiple times such as routine area or item monitoring; upon selection of containers for non-destructive assay (NDA), non destructive examination (NDE) or other characterization techniques; prior to any container movement and/or restacking; following movement to and/or staging at any characterization areas; routine monitoring of staged containers awaiting characterization; and post-characterization return to site storage for failed containers or promotion through characterization process for successful containers. Essentially, any time a CH drum is handled or moved, a site HP measurement is made and recorded. Any container that measures close to 200 mrem/hr would be easily identified and segregated as part of the site's routine as low as reasonably achievable (ALARA) practices.

EPA has observed the use of brightly colored markers in CH drum storage areas at LANL, Hanford, the Advanced Mixed Waste Treatment Project (AMWTP), INL and ORNL. A marker is placed on top of a drum to indicate a "hot spot," i.e., a surface dose rate above a site-specific administrative level, typically 50 mR/hr. This provides a clearly visual aid directing all personnel to avoid specific drums or areas, if possible, consistent with standard ALARA radiation protection practices. A measurement indicating a container with a surface dose rate above 200 mrem/hr would trigger at least one of several site HP controls, not to mention the problems it would pose for waste handling and measurement personnel at several points in the waste characterization process. It is difficult to imagine a realistic scenario in which such a container would go unnoticed at the TRU generator sites currently approved by EPA.

Consider all CH waste characterization sites operated by CCP.[15] According to the CCP Transuranic Waste Certification Plan (CCP 2007), the measurement of record by which the CH generator certifies that a container meets the CH Data Quality Objective (DQO) for a surface dose rate of less than 200 mrem/hr originates within each site's HP or Rad Con program. The CCP waste certification plan also stipulates that the "neutron contribution to the total dose equivalent rate" shall be reported (CCP 2007). Site HP personnel take the surface dose rate measurements when the payload is prepared for shipment and these values are recorded on the Payload Assembly Container Transportation Certification Document (PACTD), and are also reported separately using the WIPP Waste Data System (WDS).[16] These values are provided to the site's Waste Certification Official (WCO), who formally certifies that compliance with the

[15] This currently includes LANL, INL, ORNL, SRS, and Hanford, and in the past has included ANL, the Nevada Test Site (NTS) and Lawrence Livermore National Laboratory (LLNL).

[16] In 2009, the WDS replaced the WWIS.

CH surface dose rate criterion has been met. This certification is done for each individual drum within the payload and for the shipment as a whole. EPA has observed this process during CH inspections at several generator sites in the course of evaluating the WIPP Waste Information System (WWIS). An equivalent process to what is described at CCP sites occurs for the AMWTF CH waste characterization program using different operating procedures, but the same WDS WCO certification. Minor differences exist with respect to specific instruments, forms, and frequency of surface dose rate surveys; however, all CH TRU generator sites provide the same WCO certification within the WDS.

RH containers receive a greater emphasis than CH with respect to their status determination by virtue of their external dose rate and their potential for personnel exposure. As stated above, the majority of CH and RH containers have surface dose rates that are far from their classification limit, significantly less than or greater than 200 mrem/hr, respectively. For these containers, it is easy to ignore consideration of the uncertainty associated with surface dose rate determination and still ensure regulatory compliance with the CH-RH criterion. However, shielded containers may require a greater level of scrutiny and specific modifications of the existing site procedures to ensure that the highest measured values for all shielded containers are below 200 mrem/hr.

10.3 Uncertainty for Surface Dose Rate Measurements

Regulatory compliance must be demonstrated for both CH and RH waste containers, and the current practice is to report a single, measured value without consideration of the value's uncertainty. This uncertainty becomes relevant when a waste container is sufficiently close to a regulatory limit that it introduces questions regarding the container's classification, as discussed below.

According to the RH WCPIP, the "surface dose rate minimum and maximum limits for RH TRU waste are not established with an associated error or uncertainty" (DOE 2003). This is in contrast to other DQOs, such as radionuclide composition, Fissile Gram Equivalent (FGE), and Decay Heat, all of which must be reported in conjunction with an estimate of the parameter's uncertainty or Total Measurement Uncertainty (TMU). Also, radionuclide data derived from NDA (CH TRU) and radiological characterization (RH TRU) are presented with their uncertainties expressed as TMU. While TMU must be reported, there are no limits attached to it. EPA has observed CH TRU data with TMU values greater than 100% and RH TRU values approaching 10%, which have been acceptable.

TMU has not been addressed for surface dose rate measurements for several reasons, mainly because, as stated above, it is not required. Surface dose rate measurements are not controlled to the same extent as radionuclide-specific determinations, which generally involve greater attention to details like reproducible source-to-detector distance, longer counting times, isotopic distributions, and item/measurement geometry. In a technical sense, TMU is not directly applicable to surface dose rate measurements. For example, radionuclide-specific measurements[17] record the total number of events (counts) per unit time which, when sufficiently large, is assumed to follow a known distribution that allows the derivation of meaningful statistical parameters; acquisition (counting) times can be adjusted to accommodate low signal

[17] This includes radionuclide-specific measurements that are used to derive FGE, PE Ci and TAAC.

(activity) containers. Surface dose rate measurements look to simply record the highest response rate of a given item relative to the CH-RH criterion (200 mrem/hr), provided the instrument has been appropriately exposed to the source; increasing the counting time does not matter.

Positional effects must be considered. Typical CH and RH waste matrices, such as debris, may contain a small number or even a single item that is heavily contaminated within a matrix of much less contaminated debris. The position of such an item within a waste container can cause the surface dose rate readings to vary considerably from one side or end of the container to the other. Proper survey techniques help to minimize this effect, and CBFO has stated that (Moody 2008):

> *The surface dose rate measurement protocols that DOE uses at the various sites involve multiple measurements around each waste container, with the single highest measured value employed as the "dose rate of record" that governs how the container is classified.*

The use of multiple measurements around a waste container when performing a surface dose rate survey is the main way of addressing positional effects. Site procedures typically include this and, if properly executed, multiple measurements and attention to detail should identify any hot spots close to one side or end of a waste container. This assumes that for shielded containers, the single highest measured value would be the "dose rate of record," as CBFO has stated (Moody 2008). Although this is understood to be a common practice, it is not clear that all surface dose rate survey procedures currently in use at CH and RH generator sites include this as an explicit requirement.

As discussed previously, the CBFO QAPD requires that each site's M&TE program meet the requirements of ANSI-N323. This entails many general and specific requirements for site M&TE programs, i.e., calibration frequency, maintenance of instruments and standards, use of check sources, operational characteristics and documentation. The specified acceptance range for accuracy of neutron dose equivalent instruments is ± 20%;[18] the gamma acceptance range for accuracy is ± 10% to ± 15%[19] (ANSI 1997). This tolerance is assumed to encompass contributions from all sources of uncertainty involved in the calibration process, i.e., calibration source(s), positional effects, curve fitting, and differences between calibration source and actual wastes.

For example, assume that the results of surveying a container RH waste yield a surface dose rate of 210 mrem/hr. If the allowable calibration uncertainty of the survey instrument is 10% and this represents the results of a single gamma measurement, a simple restatement of these results is that the container's surface dose rate is between 189 and 231 mrem/hr. If the container's surface dose rate results of 210 mrem/hr were the composite of a gamma and a neutron measurement, and each measurement complied with the ANSI standard, the uncertainty calculation becomes more complex. Assuming the gamma and neutron measurements are independent, the 10% gamma uncertainty and the 20% neutron uncertainty would be added in quadrature, i.e., by

---

[18] The specific requirement is ±30% for 0–10 mrem/hr and ±20% for above 10 mrem/hr.

[19] The specific requirement is that the instrument be tested near the end points of the range, i.e., 20% and 80% of full scale, and the acceptance ranges are ±15% for the low point and ±10% for the upper point.

taking the square root of the sum of the squares for both measurements. This yields an uncertainty of 22%, allowing a restatement of the 210 mrem/hr value as between 163 and 256 mrem/hr. However, if the gamma and neutron measurements are not considered to be independent, their uncertainties would be additive (summed), increasing the uncertainty relative to the 210 mrem/hr measured value. The same approach is applicable to a hypothetical CH waste container with a measured surface dose rate of 190 mrem/hr, which could be stated as a surface dose rate between 171 and 209 mrem/hr based on a single gamma measurement, and between 148 and 231 mrem/hr based on two independent measurements, one gamma and one neutron. This hypothetical drum of PFP waste would be classified as CH TRU. Because there is no consideration for uncertainty or error, one could reasonably question the validity of the classifications for these containers as CH or RH.

10.4 Summary of Measurement Issues

DOE surface dose rate measurement procedures for photons and neutrons should specifically address uncertainties to ensure that the legal limit of 200 mrem/hr is not exceeded. In addition, DOE should develop site-wide procedures to ensure that surface dose rate measurements are conducted consistently at all TRU waste generator sites. EPA recognizes that these issues are not relevant to the SC design; rather they are implementation issues. However, they must be addressed as part of any waste disposal program utilizing shielded containers.

11.0 SUMMARY AND CONCLUSIONS

SC&A has reviewed the DOE PCR regarding the use of shielded containers for the disposal of selected RH TRU waste streams on the floor of the WIPP disposal rooms. In the review, particular attention was directed to *Analysis Report for Shielded Container Performance Assessment* (Dunagan et al. 2007) and *Analysis of RH TRU Wastes for Containment in Lead Shielded Containers* (Crawford and Taggart 2007), since these two documents summarize DOE's technical approach to PA. A number of relevant supporting documents were also reviewed. SC&A demonstrated that the maximum temperature increase in the wall of a shielded container from photon interactions was only a few degrees. SC&A examined the MicroShield quality assurance documentation and found it to be acceptable. SC&A ran the MicroShield code duplicating LANL results. SC&A evaluated the impact of additional radionuclides that are strong photon emitters, in addition to those selected by LANL.

The impact of the PCR on inventory changes was reviewed and the DOE approach was validated. Possible changes to the repository chemical conditions were examined and it was concluded that impacts from the PCR on repository performance were negligible.

DOE modeling showed that the shielded container approach resulted in barely perceptible changes in the PA mean CCDFs, even for the bounding case where all of the RH TRU waste was processed like CH TRU waste and placed in shielded containers on the floors of the disposal rooms. The calculational approach used in the SCPA was reviewed and found to be acceptable and properly implemented.

Based on its review of the shielded container safety analysis, SC&A found that the shielded container created no accident risks to the MOI or nearby worker that were greater than for CH

TRU in 55-gallon drums. In fact, due to the robust container design, the risks were lower for a variety of postulated accidents.

Important implementation questions must be addressed before disposal in shielded containers can be implemented. As described in Section 10, consideration should be given to adopting a DOE complex-wide procedure or requirements for characterizing the surface dose rate for shielded containers, and the procedure or requirements should account for measurement uncertainty to ensure that the 200 mrem/hr surface dose rate limit for CH TRU is not exceeded. This would include procedures to ensure that the radiation source does not shift within the container during handling and transportation.

12.0 REFERENCES

ANSI 1997. *American National Standard Radiation Protection Instrumentation Test and Calibration, Portable Survey Instruments*. ANSI N323A-1997. American National Standards Institute.

ATWIR 2007. *Annual Transuranic Waste Inventory Report – 2007.* DOE/TRU-2008-3379, Rev. 1. U.S. Department of Energy, Carlsbad Field Office, Carlsbad, New Mexico.

ATWIR 2008. *Annual Transuranic Waste Inventory Report – 2008.* DOE/TRU-2008-3425, Rev. 0. U.S. Department of Energy, Carlsbad Field Office. October 2008.

Bradley, E.W., C.F. Wu, and T.E. Goff. 1993. *Technical Basis for External Dosimetry at the Waste Isolation Pilot Plant (WIPP)*, Science Applications International Corp. and Westinghouse Electric Corp. DOE/WIPP-93-068, December 31, 1993.

Camphouse, C. 2010. *Analysis Package for CCDFGF: CRA-2009 Performance Assessment Baseline Calculation,* Rev. 0. Sandia National Laboratories. ERMS 553027.

CCP (Central Characterization Project) 2007. *CCP Transuranic Waste Certification Plan*. CCP-PO-002, Rev. 20.

CCP (Central Characterization Project) 2008a. *CCP Dose-to-Curie Survey Procedure for Remote-Handled Transuranic Waste*. CCP-TP-504, Rev. 6.

Chavez, M. 2007a. Document Review and Comment Form. *Technical review by Tom Kirchner for new parameters for AP-135*. ERMS 546767.

Chavez, M. 2007b. *Parameter Approval for new parameters for AP-135*. ERMS 546784.

Clayton, D. 2008. Document Review and Comment Form, *Technical Review by Daniel J Clayton for SNL Response to First Completeness Comments on the Department of Energy's (DOE) Submittal for its Remote Handled (RH) Shielded Container Planned Change Request*. ERMS 548626.

Crawford, B.A. 2005. *Tables Containing Lead Data from TWBID Rev. 2-1 D.4.15*. Memorandum to J. Trone, October 20, 2005. Carlsbad, New Mexico: Los Alamos National Laboratory. ERMS 541735.

Crawford, B. and D. Taggart. 2007. *Analysis of RH TRU Wastes for Containment in Lead Shielded Containers.* Los Alamos National Laboratory Carlsbad Operations, Simple Analysis Report INV-SAR-08, Rev. 0.

Day, Brad. 2008, "Design Overview of Shielded Containers," Washington TRU Solutions, Presentation to the Nuclear Regulatory Commission, July 8.

Djordjevic, S. 2003. *Estimation of Maximum RH TRU Thermal Heat Load for WIPP for Compliance Recertification Application*, Rev. 1. Sandia National Laboratories, Carlsbad, New Mexico. ERMS 531593.

DOE (U.S. Department of Energy) 1981. *Agreement for Consultation and Cooperation Between the Department of Energy and the State of New Mexico on WIPP*. U.S. Department of Energy and State of New Mexico, July 1, 1981, as modified (updated through April 18, 1988).

DOE 1996, *Test and Evaluation Document for DOT Specification 7A Type A Packaging*, DOE/RL-96-57, Rev. 0-F, Volume 1, http://rampac.energy.gov/NonCert/dot7a/rl96-57/ptoc.htm, Accessed: August 19, 2008.

DOE 2003. *Remote-Handled Waste Characterization Program Implementation Plan*, Rev. 0. DOE/WIPP-02-3214. October 30, 2003.

DOE 2004a. *Title 40 CFR 191 Parts B and C Compliance Recertification Application*. U.S. Department of Energy Field Office, March 2004. Docket A-98-49 Category II-B2.

DOE 2004b. *Quality Assurance Program Document*, Rev. 6. DOE/CBFO-94-1012. September 2004.

DOE 2005. *Contact Handled Transuranic Waste Authorized Methods of Payload Control (CH-TRAMPAC)*, Rev. 2, May 2005.

DOE 2006a. *Remote Handled Transuranic Waste Authorized Methods of Payload Control (RH-TRAMPAC),* Rev. 0. DOE/WIPP 90-045 Rev. 7. RH-TRAMPAC Document Rev. 0, June 2006. U.S. Department of Energy, Carlsbad Field Office, Carlsbad, New Mexico.

DOE 2006b. *Waste Isolation Pilot Plant Remote Handled (RH) Waste Documented Safety Analysis*, U.S. Department of Energy, DOE/WIPP-06-3174, Rev. 0, March 2006.

DOE 2006c. *Transuranic Waste Baseline Inventory Report 2004,* Rev. 0. U.S. Department of Energy, Carlsbad Field Office, DOE/TRU-2006-3344.

DOE 2006d. *RH TRU 72-B, Safety Analysis Report*, U.S. Department of Energy, Rev. 4, June 2006.

DOE 2006e. *Preparation Guide for U.S Department of Energy Nonreactor Nuclear Facility Documented Safety Analyses,* U.S. Depatment of Energy DOE-STD-3009-94, Rev. 3, March 2006.DOE 2007a. *Transuranic Waste Acceptance Criteria for the Waste Isolation Pilot Plant*. DOE/WIPP-02-3122, Rev. 6.1. U.S. Department of Energy, Carlsbad Field Office. October 29, 2007.

DOE 2007b. *WIPP Technical Fact Sheet: Planned Change Request for Shielded Containers*. Attachment to Moody 2007.

DOE 2007c. *Annual Transuranic Waste Inventopry Report – 2007*. DOE/TRU-2008-3379, Rev. 1. U.S. Department of Energy, Carlsbad Field Office.

DOE-CBFO 2008. *HalfPACT Safety Analysis Report*, Rev. 5. U.S. Department of Energy, Carlsbad Field Office, Carlsbad, New Mexico.

DOE 2009a. Letter from D. Moody, Manager, Carlsbad Field Office, DOE to Jonathan Edwards, Acting Director, EPA Radiation Protection Division dated June 11, 2009. *Subject: Transmittal of NRC Certificate of Compliance.*

DOE 2009b. Transuranic Waste Acceptance Criteria for the Waste Isolation Pilot Plant, Rev. 6.4. DOE/WIPP-02-3122. U.S. Department of Energy, Carlsbad Field Office. December 10, 2009.

DOE 2009c. Letter from D. Moody, Manager, Carlsbad Field Office, DOE, to Jonathan Edwards, Acting Director, EPA Radiation Protection Division dated January 21, 2009. *Subject: Self-Certification of the Shielded Container to Department of Transportation (DOT) Requirements.*

DOE 2009d. *Title 40 CFR Part 191 Subparts B and C Compliance Recertification Application for the Waste Isolation Pilot Plant, Appendix PA-2009 Performance Assessment*. DOE/WIPP-09-3424. U.S. Department of Energy, Carlsbad Field Office.

DOE 2010a. Letter from D. Moody, Manager, Carlsbad Field Office, DOE, to Jonathan Edwards, Acting Director, EPA Radiation Protection Division dated March 4, 2010. *Subject: Submittal of Safety Analyses in Support of the Planned Change Request for Lead Shielded Containers.*

DOE 2010b. Letter from Susan McCauslin, DOE/CBFO to Jonathan Walsh, USEPA/ORIA: *Summary of a Revised Safety Impact Analysis for the Lead Shielded Container Assembly*. December 3, 2010.

Dunagan, S.C., C.W. Hansen, and W.P. Zelinski. 2005. *Effect of Increasing Cellulosics, Plastics, and Rubbers on WIPP Performance Assessment, Rev. 1*. Sandia National Laboratories, Carlsbad, New Mexico. ERMS 538445.

Dunagan, S.C. 2007. *New Parameters for AP-135*. Memorandum August 27, 2007. Sandia National Laboratories, Carlsbad, New Mexico. ERMS 546765.

Dunagan, S.C., G.T. Roselle, E.D. Vugrin, and J.T. Long. 2007. *Analysis Report for Shielded Container Performance Assessment*. Sandia National Laboratories, Carlsbad, New Mexico, ERMS 547197. October 31, 2007.

Dunagan, S.C. and E.D. Vugrin. 2007. *Analysis Plan to Assess the Impact of Shielded Container Emplacement in the Waste Isolation Pilot Plant*, AP-135. Sandia National Laboratories, Carlsbad, New Mexico. ERMS 546628.

Eckerman, K.F., and J.C. Ryman. 1993. *External Exposure to Radionuclides in Air, Water, and Soil*, Oak Ridge National Laboratory, Prepared for U.S. Environmental Protection Agency, Office of Radiation and Indoor Air, Federal Guidance Report No. 12.

EPA (Environmental Protection Agency) 2006. *Technical Support Document for Section 194.24: Evaluation of the Compliance Recertification Actinide Source Term and Culebra Dolomite Distribution Coefficient Values*. Office of Radiation and Indoor Air, Docket No. A-98-49, Item II-B1-16, March 2006.

EPA (Environmental Protection Agency) 2007. *Waste Characterization Inspection Report: EPA Baseline Inspection No. EPA-INL-CCP-RH-6.06-8 of the Central Characterization Project Remote-Handled Transuranic Waste Characterization Program At The Idaho National Laboratory*, June 12–16, and August 9 and 29, 2006.

Fox, B. 2005. *Analysis Package for EPA Unit Loading Calculations*. CRA-2004 Performance Assessment Baseline Calculation, Rev. 0. Sandia National Laboratories, Carlsbad, New Mexico. ERMS 540378.

Fox, B. 2007. Document Review and Comment (DRC), *Technical Review by Brian Fox for Letter to Russ Paterson from Sean Dunagan Entitled Sandia's Inventory Data Needs for Shielded Container Performance Assessment.* ERMS 546360.

Garner, J.W. and C.D. Leigh. 2005. *Analysis Package for PANEL*. CRA-2004 Performance Assessment Baseline Calculation, Rev. 0. Sandia National Laboratories, Carlsbad, New Mexico. ERMS 540572.

Hansen, F.D. and J.S. Stein. 2005. *WIPP Room Evolution and Performance Assessment Implications.* Milestone Report, February 26, 2005. Sandia National Laboratories, Carlsbad, New Mexico. ERMS 538870.

Hansen, C.W., L.H. Brush, M.B. Gross, F.D. Hansen, B.Y. Park, J.S. Stein, and T.W. Thompson. 2004. Effects of Supercompacted Waste and Heterogeneous Waste Emplacement on Repository Performance, Rev. 2 (January 19). ERMS 533551. Carlsbad, NM: Sandia National Laboratories.

Helton, J. 1996. WIPP Data Entry Form: 464, Area for RH Waste Disposal in CCDFGF Model. Sandia National Laboratories, Carlsbad, New Mexico. ERMS 548735.

Kirchner, T.B. 2005. *Sensitivity of Performance Assessment Baseline Calculation Releases to Parameters*. Sandia National Laboratories, Carlsbad, New Mexico. ERMS 540767.

Kirchner, T.B. 2007a. *Parameter Problem Report 2007-001: REFCON:FVW. Sandia National Laboratories*, Carlsbad, New Mexico. ERMS 545481.

Kirchner, T.B. 2007b. Document Review and Comment Form. *Technical review by Tom Kirchner for new parameters for AP-135*. ERMS 546766.

Kirchner T.B., 2007c. Document Review and Comment (DRC), *Technical Review by Thomas Kirchner for Analysis Report for the Shielded Container Performance Assessment*. ERMS 547198.

LANL (Los Alamos National Laboratory) 2003a. *Response to Request for Radionuclide Activities in TRU Waste Streams from TWBID Revision 2.1, Version 3.12 Data Version D.4.08*. Los Alamos National Laboratory, Carlsbad, New Mexico. ERMS 530918.

LANL 2003b. *Response to the Request for Waste Material and Container Material Densities from TWBID, Revision 2.1, Version 3.12 Data Version D.4.08*. Correspondence. Los Alamos National Laboratory, Carlsbad, New Mexico. ERMS 530767.

LBNL 2008. *LBNL Isotopes Project Nuclear Data Dissemination Home Page*. Lawrence Berkeley National Laboratory, http://ie.lbl.gov/toi.html, Accessed: August 14, 2008

Leigh, C.D., J.R. Trone, and B. Fox. 2005. *TRU Waste Inventory for the 2004 Compliance Recertification Application Performance Assessment Baseline Calculation*. Sandia National Laboratories, Carlsbad, New Mexico. ERMS 541118.

Long, J. 2007. Parameter Data Entry Form. ERMS 546783.

Lowry, T.S. 2005. *Analysis Package for Salado Transport Calculations*. CRA-2004 Performance Assessment Baseline Calculation, Rev. 0. Sandia National Laboratories, Carlsbad, New Mexico. ERMS 541084.

Lowry, T.S., and J. Kanney. 2005. *Analysis Report for the CRA-2004 PABC Culebra Flow and Transport Calculations.* Sandia National Laboratories, Carlsbad, New Mexico. ERMS 541508.

Moody 2007. Letter from David C. Moody, DOE/CBFO to Juan Reyes, USEPA. Subject: *Transmittal of Planned Change Request for Shielded Containers*. November 15, 2007.

Moody 2008. Letter from David C. Moody, DOE/CBFO to Juan Reyes, USEPA. Subject: *Shielded Container Planned Change Request*. April 30, 2008.

Moody 2010. Letter from David C. Moody, DOE/CBFO to Mike Flynn USEPA/ORIA. Subject: *Shielded Containers*. September 8, 2010.

Moo Lee. 2007a. Document Review and Comment Form. *Technical Review by Tom Kirchner for New Pparameters for AP-135.* ERMS 546768.

Moo Lee. 2007b. Document Review and Comment Form, *Management Review by Moo Lee for Sandia's Inventory Data Needs for Shielded Container Performance Assessment*. ERMS 546362.

Nemer, M. and J. Stein. 2005. *Analysis Package for BRAGFLO: 2004 Compliance Recertification Application Performance Assessment Baseline Calculation*. Sandia National Laboratories, Carlsbad, New Mexico. ERMS 540527.

NRC (U.S. Nuclear Regulatory Commission) 2007. *ASME Code Cases Not Approved for Use*. Regulatory Guide 1.193, Rev. 2. October 2007.

PAIR 2008. *Performance Assessment Inventory –2008*. INV-PA-08, Rev. 0. Los Alamos National Laboratory – Carlsbad Operations, Carlsbad, New Mexico. LA-UR-09-02260, April 23, 2009.

Park, B.Y., and F.D. Hansen. 2003. Analysis Report for Determination of the Porosity Surfaces of the Disposal Room Containing Various Waste Inventories for WIPP PA (Revision 0). ERMS 533216. Albuquerque: Sandia National Laboratories.

Patterson 2010. Letter from Russell Patterson, DOE/CBFO to Jonathan Walsh USEPA/ORIA. Subject: *Response to Environmental Protection Agency Regarding the Type A Testing of Shielded Containers*. November 10, 2010.

Reyes, J. 2007. Letter from Juan Reyes, USEPA to David C. Moody, DOE/CBFO. *Comments of the Department of Energy's (DOE) Remote Handled (RH) Shielded Container Planned Change Request*. December 7, 2007.

Reyes, J. 2008. U.S. Environmental Protection Agency, Office of Air and Radiation, Washington, D.C., Letter to D.C. Moody U.S. Department of Energy, Carlsbad Field Office, Carlsbad, New Mexico. February 11, 2008.

Sanchez, L.C., and H.R. Trellue. 1996. *Estimation of Maximum RH TRU Thermal Heat Load for WIPP.* Memo to T. Hicks (Galson Sciences Ltd.), January 17, 1996. Sandia National Laboratories, Albuquerque, New Mexico. ERMS 231165.

Shleien, Bernard. 1992. *The Health Physics and Radiological Health Handbook*, Scinta, Inc.

Sellmer, T. 2007. *Final Container Design, Shielded Container*. Washington TRU Solutions, Carlsbad, New Mexico. ERMS 547052.

Stein, J.S., and W.P. Zelinski. 2004. *Effect of Increased Iron Surface Area from Supercompacted AMWTP Waste on Long-Term Repository Conditions as Modeled in Performance Assessment of the WIPP.* Sandia National Laboratories, Carlsbad, New Mexico.

Stein, J.S., M.B. Nemer, and J.R. Trone. 2005. *Analysis Package for Direct Brine Releases, Compliance Recertification Application, 2004 PABC, Rev. 0.* Sandia National Laboratories, Carlsbad, New Mexico. ERMS 540633.

TEA (Trinity Engineering Associates) 2004. *Review of Effects of Supercompacted Waste and Heterogeneous Waste Emplacement on WIPP Repository Performance*. Prepared for the U.S. Environmental Protection Agency, Office of Radiation and Indoor Air, Washington, DC, March 17, 2004.

TEA 2005. *Review of WIPP Performance Assessment Computer Code Migration Activities, Version 3.* Prepared for the U.S. Environmental Protection Agency, Office of Radiation and Indoor Air, Washington, DC, December 31, 2005.

Trone, J. 2007. Document Review and Comment Form, *Technical Review by Joan Trone Letter to Russ Paterson from Sean Dunagan Entitled Sandia's Inventory Data Needs for Shielded Container Performance Assessment*. ERMS 546361.

Vugrin, E.D. 2005a. *Analysis Package for DRSPALL, CRA 2004 Performance Assessment Baseline Calculation.* Sandia National Laboratories, Carlsbad, New Mexico. ERMS 540415.

Vugrin, E.C. 2005b. *Analysis Package for CUTTINGS_S, CRA-2004 Performance Assessment Baseline Calculation.* Sandia National Laboratories, Carlsbad, New Mexico ERMS 540468.

Vugrin, E.C. and S. Dunagan. 2005. *Analysis Package for CCDFGF: CRA-2004 Performance Assessment Baseline Calculation*. Sandia National Laboratories, Carlsbad, New Mexico ERMS 540771.

Vugrin, E.D., and M.B. Nemer. 2007. *Analysis Plan for the 2009 Compliance Recertification Application Performance Assessment.* Sandia National Laboratories, Carlsbad, New Mexico. ERMS 545496.

Wall, N.A. and D. Enos. 2006. *Iron and Lead Corrosion in WIPP-Relevant Conditions*. TP 06-02, Rev. 1. Sandia National Laboratories, Carlsbad, New Mexico, ERMS 543238.

Wilson, C., D. Porter, J. Gibbons, E. Oswald, G. Sjoblom, and F. Caporuscio. 1996. *Conceptual Models Peer Review Report*, Prepared for the U.S. Department of Energy, Carlsbad, New Mexico, July 1996, Docket No. A-93-02 Item II-G-1.

WTS 2008. Shielded Container Type A Evaluation Report, ECO No. 11834, Rev. 0, Washington TRU Solutions LLC, Carlsbad, New Mexico, June 2008.

APPENDIX A

RESOLUTION OF CORRECT THICKNESS FOR INNER AND OUTER STEEL SHELLS FOR SHIELDED CONTAINERS

EPA questioned DOE about the use of different thicknesses for the inner and outer shells of the shielded containers reported in various DOE documents. EPA's specific comment, DOE's response, and EPA's conclusion are presented in this Appendix.

EPA Comment

During our review of various DOE documents related to shielded containers, we have noted several inconsistencies regarding the thickness of the inner and outer steel shells of the container:

- Dunagan et al. 2007 (Section 2.2) lists the thicknesses of the outer and inner walls as 1/8 in (0.125) and 3/16 in (0.1875), respectively. Elsewhere (i.e., Section 3.1.6.2), the same authors list the combined thickness of the inner and outer steel shells as 0.3125 in.
- According to DOE 2010a, Enclosure 1, the inner and outer steel walls are 7 gauge (0.144 in) and 11 gauge (0.091 in), respectively.
- DOE states that the inner wall thickness is 0.1793 in (DOE 2009 Enclosure, Section 4.2.3).

DOE should indicate which dimensions are the correct ones and what effects, if any, the corrected dimensions have on the various analyses that DOE has performed.

EPA References

Dunagan, S.C., G.T. Roselle, E.D. Vugrin, and J.T. Long, 2007. *Analysis Report for Shielded Container Performance Assessment*. Sandia National Laboratories, Carlsbad, New Mexico, ERMS 547197. October 31, 2007.

DOE 2010a. Letter from D. Moody, Manager, Carlsbad Field Office, DOE, to Jonathan Edwards, Acting Director, EPA Radiation Protection Division, dated March 4, 2010. *Subject: Submittal of Safety Analyses in Support of the Planned Change Request for Lead Shielded Containers*. (Enclosure 1 is "Summary of the Safety Impact Analysis for the Lead Shielded Container.")

DOE 2009. Letter from D. Moody, Manager, Carlsbad Field Office, DOE, to Jonathan Edwards, Acting Director, EPA Radiation Protection Division, dated January 21, 2009. *Subject: Self-Certification of the Shielded Container to Department of Transportation (DOT) Requirements.* (Enclosure is "Shielded Container Type A Evaluation Report.")

DOE Response

The correct dimensions for the thickness of outer shell of the shielded container is 0.120 in (+/- 0.008 in), 11 gauge (GA), and the inner shell is 0.179 in (+/- 0.008 in), 7 GA (Washington TRU Solutions 2007).

Concerning the Shielded Container Performance Assessment Document:

Since the analysis in Dunagan et al. (2007) was performed, the inner and outer thicknesses of the shielded container have changed to 0.120 ± 0.008 in (11 GA) for the outer wall and 0.179 ± 0.008 in (7 GA) for the inner wall. Though these values are less than the values in Dunagan et al. (2007), the conclusions formed based on the thickness of the walls remain unchanged. The small reduction of the wall thicknesses will have no impact on the results of the analysis performed in Dunagan et al. (2007).

Concerning the Safety Impact Analysis Document:

The dimensions stated in "Summary of the Safety Impact Analysis for the Lead Shielded Container" were incorrect. However, the Safety Impact Analysis was performed using the correct dimensions stated above and does not change the results and conclusions in the report, i.e., consequences to the public (i.e., the maximally exposed offsite-individual) and to the onsite (facility) worker will not increase with the use of the shielded containers.

The following paragraph, page 1 in "Summary of the Safety Impact Analysis for the Lead Shielded Container," should be replaced with the following paragraph.

> The shielded container has approximately the same exterior dimensions as a 55-gallon drum and holds a single 30-gallon drum that will contain the RH TRU waste (See Figure 1). The cylindrical sidewall of the shielded container has nominal 1-in thick lead shielding sandwiched within a double-walled steel shell. The external wall is 11 gauge (0.120 in ± 0.008 in) steel and the internal wall is 7 gauge (0.179 in ± 0.008 in) steel. The lid and the bottom of the container are made of carbon steel and are approximately 3 in thick. The 30-gallon inner container has a gross internal volume of 4.0 ft^3 (0.11 m^3) and a maximum loaded weight of 2,260 pounds. The empty weight of the shielded container is 1,726 lbs.

This will be noted as errata to the "Summary of the Safety Impact Analysis for the Lead Shielded Container" document.

Concerning the Shielded Container Type A Evaluation Report:

The analyses, tests, and evaluations performed on the shielded container to demonstrate compliance of the packaging design for use as a standalone DOT 7A Type A packaging were performed using the correct dimensions for the inner and outer shells as stated above. The results and conclusions in "Shielded Container Type A Evaluation Report" for use of the shielded container as a DOT 7A Type A packaging remain valid.

DOE Reference

Shielded Container Assembly, Drawing No. 165-F-026-W5, Rev. B, 2007, Washington TRU Solutions LLC, Carlsbad, New Mexico.

EPA Conclusion

Based on this information, the Agency has concluded that thickness of the inner and outer steel shells of the shielded containers has been properly addressed by DOE.

APPENDIX B

CONSEQUENCES OF FIRE DAMAGE TO SHIELDED CONTAINERS

To better understand why the consequences of a fire that damages a shielded container containing RH TRU waste are no greater than the consequences of a fire that damages a standard 55-gallon drum of CH TRU waste, a brief description of the Accident Analysis (AA) used in the WIPP Documented Safety Analysis (DSA) is provided here (WIPP 2010a).

In the AA, the total effective doses (TED) to both the maximally exposed offsite individual (MOI) and an onsite worker located 100 m from the accident are calculated. The source term (ST) for the calculation is the amount of Pu-239 equivalent curies (PE-Ci) released to the atmosphere from the fire accident. The source term is multiplied by a dose conversion factor (TED_i - rem/PE-Ci) to obtain the TED incurred by the *i*th receptor (D_i - rem):

$$D_i = ST \times TED_i$$

The source term for a given accident scenario is a function of five factors:

- MAR – material at risk (PE-Ci): the amount of material available to be acted upon by the accident (fire)
- DR – damage ratio: fraction of material actually involved in the accident
- ARF – airborne release fraction
- RF – respirable fraction
- LPF – leak path factor: fraction of radionuclides that get filtered out or deposited by natural processes within the facility

Accordingly, the equation for the source term is:

$$ST = MAR \times DR \times ARF \times RF \times LPF$$

For the shielded container AA, the material at risk is assumed to be the number of containers involved in the accident times the PE-Ci per container.

The concept of Pu-239 equivalent curies is designed to normalize risks from other TRU radionuclides to Pu-239 as described in Appendix B of the WIPP Waste Acceptance Criteria (DOE 2010). The Pu-239 equivalent activity for the *i*th radionuclide (AM_i) is defined as A_i/WF_i, where Ai is the activity of the *i*th radionuclide and WF_i is the PE-Ci weighting factor. WF_i is, in turn, defined as E_0/E_i, where E_0 is the 50-year whole-body committed dose for inhalation of Pu-239 particles with a 1 micron AMAD and a weekly lung clearance class. E_i is the 50-year committed whole-body dose for the *i*th radionuclide with a one micron AMAD and a lung clearance class that results in the highest value of E*i*. Values of the 50-year commited dose are obtained from DOE 1988. The dose conversion factor for Pu-239 is 5.1E+02 rem/µCi, while the comparable value for Cs-137 (a major source of RH TRU activity) is 3.2E-02 rem/µCi. Thus, any releases of Cs-137 from a SC would involve insignificant incremental inhalation exposures.

DOE equates the SC to a pipe overpack container (POC) in terms of general physical characteristics, i.e., they both involve a container within a container (WIPP 2010b). Testing of POCs during fires is described in (DOE-STD-5506-2007):

> *Four POCs were subjected to Type B protocol thermal tests as summarized in Appendix C. The associated 150 MW fuel pool fire caused the one outer 55-gallon drum of a POC package with a metal filter to experience lid loss. This occurred within the first three minutes of the fire. Post-fire inspection showed the pipe component seal and filter gasket to be damaged. Associated leak rate testing of this POC showed a total leak rate of 24 cm^3/s at a differential pressure of 87 kPa.*

Based on this information, the product of ARF × RF was estimated to be 8.1E-08 (WIPP-2010b, Section 7.5). The comparable values for ARF × RF for 55-gallon drums ranged from 1E-04 to 1E-02 for confined burning and unconfined burning, respectively (WIPP 2010a, Table 3-1). On a per-container basis, the MAR is the same for both a standard 55-gallon drum and an SD, with the limit specified as ≤ 80 PE-Ci. Thus, the ST is smaller for the SC than for a 55-gallon drum. This result is based on the assumption that the inner liner of the SC assembly and the internal 30-gallon drum retain their integrity during the fire except for minimal leakage. This assumption is consistent with that made for POCs. It should also be noted that 55-gallon drums are typically shipped and emplaced as seven-packs, while SCs will be emplaced as three-packs. Thus, the total MAR for the SC waste is less than for CH TRU waste.

Appendix B References

DOE 1988. *Internal Dose Conversion Factors for Calculation of Dose to the Public*. DOE/EH-0071.

DOE 2007. *Preparation of Safety Basis Documents for Transuranic (TRU) Waste Facilities*. DOE-STD-5506-2007. April 2007.

DOE 2010. *Transuranic Waste Acceptance Criteria for the Waste Isolation Pilot Plant*, Revision 6.5. DOE/WIPP-02-3112. June 30, 2010.

WIPP 2010a. *WIPP Accident Analysis Calculations for Events Involving Released from the Gamma Shielded Container*, WIPP-031.

WIPP 2010b. *Fire Analysis of the Shielded Container for the Waste Isolation Pilot Plant, Carlsbad, New Mexico*, WIPP-032.