Office of Enterprise Assessments Review of the Hanford Site Sludge Treatment Project Engineered Container Retrieval and Transfer System Preliminary Documented Safety Analysis, Revision 00

April 2015

Office of Nuclear Safety and Environmental Assessments
Office of Environment, Safety and Health Assessments
Office of Enterprise Assessments
U.S. Department of Energy

Table of Contents

Acronyms iii

Executive Summary iv

1.0 Purpose 1

2.0 Scope 1

3.0 Background 1

4.0 Methodology 2

5.0 Results 3

6.0 Conclusions 5

7.0 Opportunities for Improvement 5

8.0 Items for Follow-up 8

Appendix A: Supplemental Information A-1

Appendix B: Documents Reviewed B-1

Appendix C: Observations on Sludge Treatment Project Engineered Container Retrieval and Transfer System Preliminary Documented Safety Analysis, Revision 0 .... C-1

Appendix D: Observations on Sludge Treatment Project Engineered Container Retrieval and Transfer System Preliminary Documented Safety Analysis, Revision 00 .... D-1

Appendix E: EA Comments on Historical Thermal and Structural Analyses Related to the STS Cask Transportation Fire Hypothetical Accident Conditions E-1

Acronyms

ARF	Airborne Release Fraction
ASME	American Society of Mechanical Engineers
BPVC	Boiler and Pressure Vessel Code
CFR	Code of Federal Regulations
CHPRC	CH2M Hill Plateau Remediation Company
CRAD	Criteria, Review and Approach Document
CSER	Criticality Safety Evaluation Report
ºC	Degrees Celsius
DBA	Design Basis Accident
DOE	U.S. Department of Energy
EA	Office of Enterprise Assessments
ECRTS	Engineered Container Retrieval and Transfer System
FHA	Fire Hazard Analysis
ºF	Degrees Fahrenheit
gpm	Gallons Per Minute
HAZOP	Hazard and Operability Analysis
HEPA	High Efficiency Particulate Air
HIH	Hose-In-Hose
HSS	Office of Health, Safety and Security
LFL	Lower Flammability Limit
KE	105-K East
KW	105-K West
MASF	Maintenance and Storage Facility
NFPA	National Fire Protection Association
NPE	Natural Phenomenon Event
OFI	Opportunity for Improvement
PacTec	Packaging Technology, Inc.
PC	Performance Category
PDSA	Preliminary Documented Safety Analysis
P&ID	Piping and Instrumentation Diagram
psi	Pounds per Square Inch
psia	Pounds per Square Inch – Absolute
psig	Pounds per Square Inch – Gauge
PSV	Pressure Safety Valve
RCR	Review Comment Record
rem	Roentgen equivalent man
Rev	Revision
RL	Richland Operations Office
SAC	Specific Administrative Control
SDC	Seismic Design Category
SS	Safety Significant
SSC	Structures, Systems, and Components
STP	Sludge Treatment Project
STS	Sludge Transport System
STSC	Sludge Transport and Storage Container
TLSB	Transfer Line Service Box
w.g.	Water Gauge
TRU	Transuranic
WIPP	Waste Isolation Pilot Plant

Executive Summary

The U.S. Department of Energy (DOE) Office of Enterprise Assessments (EA) conducted an independent assessment of the DOE Hanford Site Sludge Treatment Project Engineered Container Retrieval and Transfer System Preliminary Documented Safety Analysis (PDSA), Revision (Rev) 00. The purpose of this assessment was to evaluate the evolution of the safety bases, the design, and the associated technical supporting documents for the Sludge Treatment Project Engineered Container Retrieval and Transfer System PDSA, as well as to continue to evaluate the PDSA's compliance with applicable requirements and standards.

Hanford's Sludge Treatment Project is intended to remove the highly radioactive sludge that is currently stored in specially engineered containers at a facility at the Hanford Site. Removing the sludge from its current containers, repackaging it, and transporting it are very complex activities, requiring the development of new technologies, processes, procedures, and the associated new training. The PDSA is being developed by the Hanford Site environmental cleanup contractor, CH2M Hill Plateau Remediation Company (CHPRC), to demonstrate that the sludge removal activities can be performed safely.

CHPRC has made progress in resolving concerns raised by EA and other organizations with respect to the previous version of the PDSA (i.e., Rev 0). The revised PDSA, Rev 00, is generally more comprehensive, accurate, and understandable than the previous version, reflecting substantial increases in the levels of understanding and refinement of the facility designs and the PDSA's descriptions of those designs.

However, considerable additional design and analysis work is needed for some of the facility's safety systems, structures, and components and the corresponding PDSAs descriptions. The five most significant technical concern areas identified in EA's assessment of PDSA, Rev 0, were not adequately resolved in PDSA, Rev 00. These unresolved concern areas included: insufficient analysis of transportation fires, lack of overpressure protection, non-conservative failure analyses, non-conservative analysis of hydrogen buildup, and improper safety classification and qualification of some safety components. EA also identified four new technical concerns in PDSA, Rev 00, including one instance of an instrument not properly classified as safety significant, two instances of no provisions for leak testing valves or breakers, and one instance where the single failure criteria is not met.

These concern areas have profound safety implications and could also negatively impact project planning, cost, and/or the schedule of the resolutions. These concerns need to be addressed to provide reasonable assurance that the design of the Engineered Container Retrieval and Transfer System is technically valid; that it complies with applicable codes, standards, regulations, and orders; and that it is clearly, completely, and accurately described in the PDSA.

The PDSA is currently undergoing further review and revision. In future reviews, EA will evaluate the revisions to the PDSA.

Office of Enterprise Assessments of the
Hanford Site
Sludge Treatment Project Engineered Container Retrieval and Transfer System
Preliminary Documented Safety Analysis, Revision 00

1.0 PURPOSE

The Office of Nuclear Safety and Environmental Assessments, within the U.S. Department of Energy's (DOE) independent Office of Enterprise Assessments[1] (EA), conducted an independent assessment of the DOE Hanford Site Sludge Treatment Project (STP) Engineered Container Retrieval and Transfer System (ECRTS) Preliminary Documented Safety Analysis (PDSA), Revision (Rev) 00. The EA review is a follow-up to a previous EA assessment of PDSA, Rev 0, which was documented in an EA interim report dated January 2014. The previous and follow-up EA assessments were conducted during the period of July 12, 2013, through July 22, 2014. The purpose of this assessment was to evaluate the evolution of the safety bases, the design, and the associated technical supporting documents for the STP ECRTS PDSA, as well as to continue to evaluate the PDSA's compliance with applicable requirements and standards. The PDSA is being developed by the Hanford Site environmental cleanup contractor, CH2M Hill Plateau Remediation Company (CHPRC).

2.0 SCOPE

EA assessed the functionality, capabilities, capacities, and integration of safety structures, systems, and components (SSCs) identified in the PDSA, including associated supporting/interfacing SSCs. EA evaluated the accuracy, adequacy, completeness, consistency, understandability, etc., of the PDSA, the hazards analyses, and the other supporting analyses and calculations, as well as the actual detailed SSCs' designs and their compliance with applicable regulations, DOE orders, and industry codes, standards, and common practices. EA focused primarily on the status of the resolution of concerns identified in the January 2014 interim report.

3.0 BACKGROUND

The Hanford 105-K West (KW) Basin Sludge Treatment Program is intended to remove radioactive sludge that is currently stored in specially engineered containers within the KW Basin adjacent to the KW Reactor. This sludge, which resulted from deterioration of irradiated fuel rods that had been stored in these basins, is a gray, silty, highly radioactive substance composed of tiny fuel corrosion particles, fuel rod and metal fragments, and wind-blown soil and sand. The sludge must be removed to allow final demolition of the KW Reactor facilities.

The sludge inventory, approximately 27 cubic meters, originated from the cleanup of 105-K East (KE) and KW Basins and the Settler tanks in the KW Basin. This large volume of sludge will be ultimately disposed of at the Waste Isolation Pilot Plant (WIPP) in New Mexico, along with other remotely handled transuranic (TRU) wastes resulting from cleanup at Hanford. The sludge will initially be packaged during

[1] In May 2014, EA assumed the independent oversight function from the former Office of Health, Safety and Security (HSS). This report will use the current terminology except when citing document titles.

Phase 1 operations for storage[2] in the Hanford T-Plant until Phase 2 treatment capabilities can be provided to treat and package the material for disposal at WIPP.

The sludge material has a broad range of mechanical, chemical, and radiological characteristics and is currently stored in separate containers. The ECRTS will be used to remove the highly radioactive sludge from its current containers, repackage it, and transport it. These are very complex activities, requiring the development of new technologies, processes, and procedures, and the associated new training. To facilitate this development, the Maintenance and Storage Facility in Hanford's 400 Area, a multi-purpose, high bay facility originally used in support of the Fast Flux Test Facility, has been converted to house a mock-up fuel storage pool resembling the KW Basin. In this non-hazardous environment, Hanford is simulating the conditions (except for the radiation) that will exist during the actual process of moving the sludge out of the KW Basin to support ongoing research, development, fabrication, and creation of the necessary SSCs, processes, procedures, and associated training. These activities include the development of new sludge containers, new instrumentation, and new sludge removal and processing equipment, such as nozzles, pumps, and valves that can withstand the abrasive nature of much of the sludge.

The design life for ECRTS and required facility upgrades is 5 years with the exception of the Sludge Transport and Storage Containers (STSCs), which have a 30-year design life. The expected mission life for ECRTS is 1 year. The overall sludge removal process consists of a number of short duration transfers (about 10-15 minutes each) followed by a settling period and repeated transfers until the specified fill limit for each sludge container type is reached. Considering the total number of STSCs to be filled, the total actual transfer time from the ECRTS containers to the STSCs is about 9 hours over the 1-year mission life of the project. Hanford personnel factored the short mission life, in particular the short time period when the sludge is being retrieved and transferred to STSCs, into the selection of preventive and mitigative controls.

Each filled container will be transported from the KW Annex to a modified portion of Hanford's T-Plant, where it will be stored for an interim period until the final deposal phase is developed. As noted, this sludge will be treated, repackaged, and transported to the WIPP in New Mexico for permanent burial with other remote handled TRU wastes from Hanford Site cleanups.

In order to safely implement these activities, Hanford prepared a PDSA for the STP ECRTS final design. This PDSA and its associated engineering, analyses, SSCs, processes, and procedures were the primary focus of this EA assessment.

4.0 METHODOLOGY

This EA assessment was accomplished primarily through review of Revs 0 and 00 of the ECRTS PDSA and their supporting documents. EA conducted this assessment using methodology described in selected elements of several relevant criteria, review and approach documents (CRADs) and protocols, including:

- *Engineering Design and Safety Basis Inspection Criteria, Inspection Activities, and Lines of Inquiry*, HSS CRAD 64-19, dated December 22, 2009.

[2] Note: The DOE Richland Operations Office prefers to use the term "lag storage" to refer to "short term, contingency storage" instead of the term "interim storage," which implies an indefinite period that could last for many years. This definition of "lag storage" is based on a Record of Decision developed by the Environmental Protection Agency and DOE.

- *Safety Systems Inspection Criteria, Approach, and Lines of Inquiry*, dated December 17, 2012.

- *Essential Systems Functionality Inspection Criteria, Inspection Activities, and Lines of Inquiry*, HSS CRAD 64-11, dated October 16, 2008.

- *Criticality Safety Controls Implementation - Criteria and Review Approach Document*, HSS CRAD 45-18, dated May 31, 2013.

- *Office of Enforcement and Oversight Independent Oversight Program Appraisal Process Protocols*, dated November 2012.

- *Office of Safety and Emergency Management Evaluations Protocol for Small Team Oversight Activities*, dated June 2012.

Section 5 of this report summarizes the principal EA results. EA considered the safety analysis assumptions and supporting calculations, which establish the hazard controls and safety functions. EA also reviewed the safety basis, providing specific comments on the safety significant (SS) systems and important support systems. Section 6 summarizes EA's conclusions. Based on the results, EA identified opportunities for improvement (OFIs) in Section 7 that may assist line management in identifying options and potential solutions for various issues identified by EA. Section 8 identifies some of the more significant items for EA follow-up.

Supplemental information about the EA assessment is provided in Appendix A. Appendix B lists the documents reviewed by EA. Appendix C describes EA's interim report observations on the PDSA, Rev 0, in more detail, including the contractor's Review Comment Record responses and EA's discussion of those responses. Appendix D describes additional detailed observations from EA's assessment of the PDSA, Rev 00. Appendix E contains detailed EA observations on the thermal and structural analyses of the Sludge Transport System cask for a hypothetical transportation fire accident.

5.0 RESULTS

EA found that the PDSA, Rev 00, had better content and was more detailed than PDSA, Rev 0. CHPRC resolved many of the concerns that EA identified in its January 2014 interim report and made a number of improvements in other aspects of the PDSA. This EA report addresses two primary aspects of PDSA, Rev 00: (1) The status of resolution of concerns identified by EA in its January 2014 interim report, and (2) new concerns identified by EA in PDSA, Rev 00.

EA's January 2014 interim report identified 31 editorial and technical concerns (some with multiple examples) in PDSA, Rev 0. The editorial concerns consisted of significant omissions, ambiguities, or errors in the PSDA discussions with respect to DOE expectations described in DOE-STD-3009-94, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Documented Safety Analyses*, and other standards; the editorial concerns did not necessarily represent actual technical concerns. The technical concerns involved the technical adequacy of SSC designs; failure to meet applicable codes/standards/DOE orders; and non-conservative, non-enveloping hazards/accident analyses. Eight of EA's editorial concerns (Numbers 7, 9, 16, 18, 25, 26, 27, and 31) were fully resolved in PDSA, Rev 00, and one (Number 23) was not. Eight of the technical concerns (Numbers 12, 13, 14, 17, 19, 20, 21, and 30) were also fully resolved, and three others (Numbers 6, 15, and 24), though technically resolved, were left with unresolved editorial issues. Eleven of the technical concerns (Numbers 1, 2, 3, 4, 5, 8, 10, 11, 22, 28, and 29) were not resolved. The five most significant technical concern areas identified in EA's assessment of PDSA, Rev 0, were not among those resolved in PDSA, Rev 00. These unresolved concern areas are:

- **Transportation fire** (Concerns C3, C10, and C22): The designs and analyses of the STSC and the transportation cask, including associated hazards and accident analyses, do not adequately address a transportation fire, possibly the most significant accident that could be linked with this project.

- **STSC and cask overpressure protection** (Concerns C1, C2, C3, and C10): These vessels lack appropriate or adequate overpressure protection for all operational modes or accident conditions they may encounter (especially transportation fire).

- **Non-conservative STSC and cask failure analyses** (Concerns C3 and C22): These analyses do not consider the most potentially consequential failure mode of these vessels, a "zipper effect" failure of the vessel lid bolts, which could produce a steam-flashing explosive discharge of quantities of sludge orders of magnitude larger than what was analyzed. (The zipper effect is the failure of one bolt causing load transfer to adjacent bolts, causing their failure, and so on until all bolts have failed.)

- **Non-conservative analysis of hydrogen buildup in the STSC** (Concerns C2 and C10): This analysis, based on a temperature of 25 degrees Celsius (ºC), does not consider the effects of a transportation fire or other credible transportation conditions; therefore, the analysis is significantly non-conservative.

- **Improper safety classification and qualification of SSCs** (Concern C8): In multiple cases, SSCs have been improperly classified and qualified based on narrow interpretations of accident consequences; for example, some cases did not include all potential consequences to workers for accidents that could cause prompt worker fatalities or serious injuries caused by direct, non-radiological effects, such as from a hydrogen explosion.

All of these concern areas, which have profound safety implications, could also negatively impact project planning, cost, and/or schedule of the resolutions. The first four of these relate to weaknesses in the ECRTS design, analyses, and the overlap of the ECRTS PDSA[3] and the transportation documented safety analysis, which should – but do not – both seamlessly intersect in the ECRTS design, with no gaps.

In addition to the previously identified concerns, EA also identified the following four new technical concerns with PDSA, Rev 00 and the design of safety SSCs:

- **Unsuitable STS cask pressure instrument (Concern D5):** DOE-STD-1189-2008 requirement that SSCs that generate signals to prevent or mitigate an accident must be classified as either SC or SS, as appropriate. Contrary to this requirement, the instrument specified in the present design for monitoring STSC cask pressure, whenever the cask vent is required to be manually operated to prevent cask overpressure, is not classified SS. Additionally, its design is unsuitable for this purpose, since it has inadequate range, and because it is physically disconnected from the cask when it is sealed for transportation.

[3] Note: The designs of the STSC and the transportation cask are described in the ECRTS PDSA, since both devices have safety functions inside the Annex facility. However, after these devices' missions are completed in the Annex, and during their transportation to the T-Plant, these devices may be subjected to unique challenges that are not adequately addressed in their designs (and not adequately described in the ECRTS PDSA), the transportation fire event being most challenging. Additionally, the supporting analyses of the devices call into question the devices' related capabilities because of multiple significant non-conservatisms.

- **Isolation valves leak testing and the single failure criteria (Concern D6):** SS isolation check valves between each train of the auxiliary ventilation system and the GS Inert Gas System, intended to prevent backflow into Inert Gas System, are provided with no features to allow leakage testing. Without such testing to detect pre-existing undetected failures, this system cannot meet the single failure criteria, as required by the PDSA, Table 4-8. Additionally, no allowable leakage criteria are provided, and no leakage is accounted for in the analyses of the system's capacity to provide its safety function for 96 hours.

- **Vacuum breaker leak testing (Concern D7):** SS vacuum breakers in the Auxiliary Ventilation System are intended to close upon loss of normal STSC/cask ventilation to prevent backflow to the room air inlet. No criteria are provided for allowable leakage, and no leakage is accounted for in the analyses of the system's capacity to provide its safety function for 96 hours. Additionally, SS check valve ECRT-CV-605, which is forms part of the same pressure boundary and could also be subject to leakage, is not addressed in the PDSA; it has no leakage testing features, no specified allowable leakage, and no leakage allowance accounted for in the system's 96 hour design capacity requirement.

- **Isolation valve single failure criteria not met (Concern D8):** The interface of the SS Auxiliary Ventilation System with the non-safety Inert Gas System at check valve ECRT-CV-605 does not meet the single failure criterion, as required by the PDSA, Table 4-8, since there is only one valve is at this interface. Its failure to close on demand would cause the loss of both divisions of the Auxiliary Ventilation System.

These include one instrument not properly classified as safety significant, two instances of no provisions for leak testing valves or breakers, and one instance where the single failure criterion is not met. The nature of these four new technical concerns is similar to the previously identified technical concerns, and they have similar implications for safety and project planning, cost, and schedule. These new technical concerns are described in detail in Appendix D, along with four new editorial concerns.

6.0 CONCLUSIONS

Overall, CHPRC has made progress in resolving concerns raised by EA and other organizations with respect to PDSA, Rev 0. The PDSA, Rev 00, is generally more comprehensive, accurate, and understandable than PDSA, Rev 0, reflecting substantial increases in the understanding and refinement of the facility designs and in the PDSA's descriptions of those designs.

However, considerable additional design and analysis work is needed for some of the facility's safety SSCs and corresponding PDSA descriptions, as indicated by the number and technical significance of the unresolved EA comments on PDSA, Rev 0, regarding the design adequacy of these SSCs; supporting analysis discrepancies; the accuracy, clarity, and completeness of the PDSA; and the four new PDSA, Rev 00, technical concerns identified in Appendix D. (See **OFI 16-18**.)

7.0 OPPORTUNITIES FOR IMPROVEMENT

EA identified 18 OFIs addressing the significant technical concerns discussed in Appendices C and D. The first 15 address unresolved technical observations from the previous EA review, and the last 3 are the new observations. These potential enhancements are not intended to be prescriptive or mandatory. Rather, they are offered to the site to be reviewed and evaluated by responsible line management organizations and accepted, rejected, or modified as appropriate, in accordance with site-specific program objectives and priorities.

These technical observations represent concerns about the adequacy of SSC designs; failure to meet applicable regulations, DOE orders, codes, or standards; non-conservative, non-enveloping hazards/accident analyses; and significant omissions, ambiguities, inconsistencies, and/or errors in important PDSA discussions. Resolution of these OFIs will provide reasonable assurance that the design of the ECRTS is technically valid; that it complies with applicable codes, standards, regulations, and orders; and that it is clearly, completely, and accurately described in the PDSA. In addition, CHPRC should address the new editorial issues identified in observations D1 through D4 in Appendix D.

The order of the OFIs is consistent with the order in which they appear in Appendix C and Appendix D, which is generally sequential with respect to their PDSA discussion locations. OFI numbering is sequential, with the corresponding appendix observation number shown in parentheses. (For example, OFI-01 corresponds to the discussion in Appendix C, Observation C1, and OFI-09 corresponds to the discussion in Appendix C, Observation C11.) See the corresponding observations in Appendices C and D for more information on the technical concerns that led to the OFIs presented below:

OFI-01 (C1): Resolve ashfall timing inconsistencies and establish non-ambiguous ashfall design criteria for both SS systems and SSCs supporting SS functions when ashfall could degrade such safety functions.

OFI-02 (C2): Provide and document analyses of mechanical equipment room hazards in the PDSA; the analysis should address the Auxiliary Ventilation System components and any other vulnerable SS SSCs located inside and outside the loading bay and should identify controls as appropriate.

OFI-03 (C3): Modify the STSC transport cask design to provide engineered overpressure protection, modify the PDSA to describe the overpressure feature, and reference the analysis of the associated transportation fire event. Alternatively, explicitly describe the omission of such protection and associated analyses in the PDSA, and include the rationale for non-compliance with this American Society of Mechanical Engineers (ASME) code requirement and with regulatory requirements for worker protection, to ensure that DOE is fully informed of the concomitant risk when reviewing the PDSA. Ensure that the STSC and cask designs are compatible with the design safety requirements described in DOE standards and orders and in referenced industry codes for these containers' tenure in the Annex and for containment functions during transport to the T-Plant.

OFI-04 (C4): Modify the design to provide features to facilitate leak testing, as required by DOE Order 420.1B, of SS double isolation valves that prevent slurry backflow from contaminating various general service lines connected to the slurry transfer lines.

OFI-05 (C5): Ensure that fire suppression water discharge containment calculations fully and conservatively account for all relevant factors, such as conditions that could inhibit flow, water inputs from sources other than sprinklers, obstructions, and other factors, as described in Appendix C.

OFI-06 (C6): Ensure that the PDSA addresses the classifications, qualifications, and failure modes of the various electrical and control devices for the fire suppression water containment system, or the consequences of any such failure.

OFI-07 (C8): Ensure that SSCs are properly classified and qualified based on accident consequences (e.g., including non-radiological consequences, such as prompt worker fatalities or serious injuries from the direct effects of a hydrogen explosion). Reevaluate the examples cited in the Appendix C observation and change the design, classification, and qualification of affected SSCs as necessary to ensure conformance with the spectrum of accident consequences defined in regulations, codes, standards, and orders (e.g., 10 CFR 830, DOE Order 420.1B, DOE-STD-3009, DOE-STD-1189). Revise the PDSA to reflect such changes.

OFI-08 (C10): Perform analyses of hydrogen buildup that envelope all credible accident conditions, including a transportation fire, to which the STSC and cask could be subjected, and identify controls as appropriate.

OFI-09 (C11): Ensure that the PDSA analyzes the localized ground accelerations that could result from a seismic failure of the KW Reactor exhaust stack and its impact with the ground adjacent to the Annex, as well as the potential direct impacts on the ECRTS in the KW Basin.

OFI-10 (C15): Reevaluate the rupture disk setpoint modification, with consideration of lower-risk alternatives, such as visually and/or electrically monitored leakoff connections between the double isolation valves back to the pool, or overpressure protection devices, such as relief valves with a capacity appropriate for the systems in question. Change the PDSA to reflect any design modifications.

OFI-11 (C22): Perform a failure analysis of the STSC and cask that conservatively treats the airborne release fraction for credible cask failure modes, including loss of the closure lid due to bolt failure that results in sudden vessel depressurization.

OFI-12 (C23): Clarify wording in the slurry transfer line secondary confinement system evaluation to eliminate ambiguity.

OFI-13 (C24): Clarify the stress analyses of the ECRTS Transfer System transfer and decant boxes to account for nil-ductility transition temperature effects at extremely low temperatures.

OFI-14 (C28): Ensure that the cask pressure indicator range envelopes not only the normal operating range, as in the present design, but also the cask design pressure (80 pounds per square inch), plus a margin to allow determination of the level and rate of any challenge that exceeds this value and to allow monitoring of the venting progress, as required by DOE-STD-3009-94. Also ensure that this instrumentation is appropriately classified as SS, as required by DOE-STD-3009-94.

OFI-15 (C29): Classify and design as SS the components of the Inert Gas System that are required to support the function of the cask pressure monitoring instrumentation or that constitute a part of the SS cask pressure boundary, and describe them as such in the PDSA. As an alternative, provide other appropriately designed, qualified, and classified instrumentation and describe it in the PDSA.

OFI-16 (D5): Provide an appropriately ranged**,** located, classified, and qualified pressure instrument for monitoring the transportation cask pressure for conditions that may require venting to prevent an accident (as required by DOE-STD-1189-2008) when the cask vent tool has been replaced with a plug, such as when it is being transported.

OFI-17 (D6-D8): Establish maximum allowable leakage criteria for safety-to-non-safety boundary components of the Auxiliary Ventilation System. Verify the compatibility of such criteria with the SS performance requirement that the system provide 96 hours of auxiliary ventilation. Provide appropriate features in the system(s) design to facilitate testing for such leakage.

OFI-18 (D8): Revise the design of the SS Auxiliary Ventilation System at its interface with the non-safety Inert Gas System at check valve ECRT-CV-605 to meet the single failure criteria requirement of PDSA Table 4-8, *Auxiliary Ventilation System Functional Requirements and Performance Criteria*.

8.0 ITEMS FOR FOLLOW-UP

The PDSA is currently undergoing further review and revision. EA will continue to follow the development of the PDSA, particularly the five areas of most concern described previously: transportation fire, STSC and cask overpressure protection, non-conservative STSC and cask failure analysis, non-conservative analysis of hydrogen buildup in the STSC, and improper safety classification and qualification of SSCs. In future reviews, EA will evaluate the revisions to the PDSA.

Appendix A
Supplemental Information

Dates of Review

July 12, 2013 – July 22, 2014

Office of Enterprise Assessments Management

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Appendix B
Documents Reviewed

PRC-STP-00718, Revision 0, Preliminary Documented Safety Analysis for the Sludge Treatment Project Engineered Container Retrieval and Transfer System

PRC-STP-00697, Revision 2, Sludge Treatment Project Engineered Container Retrieval and Transfer System Final Design Hazards Analysis Supplement 1

PRC-STP-CN-N-00698, Revision 2, Sludge Treatment Project – Engineered Container Retrieval and Transfer System Preliminary Documented Safety Analysis Design Basis Accident Calculations

PRC-STP-00687, Revision 2, Sludge Treatment Project Engineered Container Retrieval and Transfer System Final Design Hazard and Operability Study

PRC-STP-00720, CSER-12-003: Criticality Safety Evaluation Report (CSER) for Retrieval of Sludge from Engineered Containers and the loading of Sludge into Sludge Transport and Storage Container

HNF-SD-SNF-TI-015, Revision 21, Volume 2, Spent Nuclear Fuel Project Technical Databook, Volume 2, Sludge

HNF-41051, Revision 11A, STP Container and Settler Sludge Process System Description and Material Balance

PRC-STP-00280, Revision 0, CSER 10-007: Criticality Safety Evaluation for the On-Site Transport for K Basin Container Sludge in the Sludge Transport System

CHPRC-01842, Revision 0, CSER 12-001 Criticality Safety Evaluation Report Sludge Transport and Storage Containers (STSC) at T Plant

EA also researched the following documents to address analyses that were not available in the above references, which were previously provided to EA:

SNF-18162, Revision 0, Thermal Analysis of Sludge Transport System for Argon Backfill and Extended Transport Window

SNF-13268, Revision 0, KE Basin Sludge Transportation System 100% Design Report - Project A.16

CHPRC-02095, Revision 0, STS Cask Structural Integration Analysis

CHPRC-02096, Revision 0, STS Cask Drop Analysis With SYSC

CHPRC-02097, Revision 0, STSC Type A Analysis

D9183766, FH-0205136, PacTec Calculation No. 12099-01, Rev1, STS Cask Structural Analysis

PacTec Calculation No. 12099-01, Rev 2, STS Cask Structural Analysis

FH-0205136 Attachment, Package Safety Analysis Assessment for the Sludge Transport System, SNF-10823

D8975014, FH-0200768, Attachment 1, Safety Basis Thermal Analysis of the Sludge Transport System

Appendix C
Observations on Sludge Treatment Project
Engineered Container Retrieval and Transfer System
Preliminary Documented Safety Analysis, Revision 0

C1. Ashfall event design criteria

Section 1.4.5, *Ashfall Events*, states that the arrival time of ashfall from a volcanic event could be as short as 1.5 hours, but that "The engineering design criteria used for the Annex and *process* [emphasis added] systems (PRC-PRO-EN-097) is 2 hours." This apparent non-conservative ambiguity is not addressed in the preliminary documented safety analysis (PDSA). Additionally, the ashfall time-related design criteria for supporting systems is not addressed.

Review Comment Record (RCR) response: "The referenced sentence is in error and will be deleted. There is no design criteria related to ash arrival time."

Office of Enterprise Assessments (EA) response to RCR response: Design criteria are needed because the timing of the ashfall could have a profoundly negative impact on equipment operability, depending on the phase of the sludge removal process at the time of a volcanic event. For example, the arrival time of the ashfall could relate directly to the analyses of pressure buildup time in the Sludge Transport and Storage Container (STSC) and the Sludge Transport System (STS) cask after they are closed and the related need for appropriate pressure relief devices in their designs, as described in later EA observations. The ashfall timing could be of particular concern during the transportation-to-T-Plant phase of operations, when ashfall could negatively affect the transport vehicle's ability to complete its mission within the time needed to prevent exceeding design pressures. Although removing the entire phrase "engineering design criteria" eliminated the ambiguity in the draft PDSA, Revision (Rev) 00, the concern was not resolved because the necessary criteria for the supporting systems or the process systems were also removed. **Not resolved.** (See **OFI-01**.)

C2. Unanalyzed mechanical equipment room hazards

Section 2.4.2.2, *Mechanical Equipment Room*, in the second paragraph, states that the room contains an "electric hot water boiler" and three compressed air receiver tanks. However, no steam or air explosions, respectively, are considered in the hazards analysis or otherwise described in the PDSA, especially (but not exclusively) in the context of a potential fire in this room; a fire could raise the pressure in the vessels, reduce their structural integrity with regard to allowable stresses, and inhibit pressure relief functions. Therefore, the PDSA considers no potential negative effects of such an explosion on safety structures, systems, and components (SSCs) or the appropriate safety significant (SS) controls to protect safety SSCs from such events. (See Sections 2.8.1, *Heating, Ventilation, and Air Conditioning System*, and 3.6.4, *Fire*.)

RCR response: "There are no safety SSCs in the Mechanical Room. Loss of services and utilities was evaluated in the hazards analyses relative to impacts to ECRTS [Engineered Container Retrieval and Transfer System] processes. Failure of either the water heater or instrument air receiver tanks due to overpressurization was judged to be a standard industrial hazard. This judgment was based on (1) that the heater and receiver tanks are located in the Mechanical Equipment Room which is separated from the Loading Bay by a 15-in concrete wall such that their failure is not a contributor to a significant uncontrolled release of hazardous material, and (2) water heaters and receiver tanks are routinely encountered in general industry and are covered by national consensus codes and standards. The hazard analyses will be revised to clarify this point."

EA response to RCR response: This condition, which could be a "standard industrial hazard," may relieve the project from addressing direct risk to workers, but it does not relieve the project from considering risk to the SS SSCs (e.g., fire is also a standard industrial hazard that must be considered as a threat to SS SSCs). The RCR response did not address the threat to the SS Auxiliary Ventilation System with main components located just outside the Mechanical Equipment Room at the northwest side of and outside of the 15-inch-walled protection of the loading bay referred to in the RCR response. Contrary to the RCR response, these main components are separated from the Mechanical Equipment Room by sheet metal siding only. **Not resolved.** (See **OFI-02**.)

C3. **STS cask pressure relief device**

Section 2.6.17, *STSC Transport Vent Assembly and Cask Transport Vent Port Tool*, states, "In preparation for transport to T Plant, the STS cask lid is bolted into place. If transport of the cask is delayed for a long period of time, the cask could overpressurize, potentially *resulting in facility worker serious injury or death* [emphasis added]. The cask transport vent port tool is used to vent the cask, if necessary, thereby preventing such over-pressurization." However, Section 4.3.11.4, *System Evaluation*, in the subsection entitled *Specific Criteria*, states, "The STS cask has been pressured [*sic*] tested to 150 percent of the cask design pressure per the requirements of 10 CFR 71.85, 'Preliminary Determinations,' *and the design requirements for a BPVC* [Boiler and Pressure Vessel Code], *Section III, Subsection NB Service Level A vessel* [emphasis added]." This design code, Article NB-7110(b), states that "Pressure relief devices are required when the operating conditions considered in the Overpressure Protection Report would cause the Service Limits specified in the Design Specification to be exceeded." It further states that, "A *pressure relief device…*may be a pressure relief valve or a nonreclosing pressure relief device [rupture disk]." This subsection also requires that such valves "shall open *automatically* [emphasis added] by direct action of the fluid pressure." Contrary to this requirement, no such pressure relief device has been incorporated into the STS cask design. Although a vent port tool is provided for this purpose, it is not an "automatic" pressure relief device, since it requires operator action to effect pressure relief. This is significant for five reasons:

- Per DOE-STD-3009, DOE Order 420.1B, and DOE-STD-1189, engineered controls, such as a relief valve, are preferred over administrative controls, such as operator action.

- In preparing a loaded cask for transportation, per PDSA Section 4.3.11.4, *System Evaluation*, "The STS cask is inerted using the vent port and drain port tools. Once an inert atmosphere is established and the STS cask is pressurized, *the tools are removed, and the pressure boundary is established by the vent port and drain port **plugs***…[emphasis added]." (This discussion is also reflected in Section 3.6.13.6, *Control Selection and Classification–Over-Pressurization Release.*) These actions would leave the cask with no overpressure protection, even manual pressure relief, until it reached the T-Plant and the vent tool was reinstalled.

- The most likely severe challenge to the pressure integrity of the cask (and possibly also the STSC) is a transportation fire. The severity of this challenge would far exceed the effects from the sources of internal pressure currently analyzed, but in a much shorter time and with increased risk for prompt worker fatalities or serious injuries. Additionally, for a transportation fire, even if the vent port tool were installed (which it is not, since it is replaced with a plug), it would not be accessible for re-installation and manual operation during a fire.

- Per Table 4-11, *Sludge Transport System Cask Pressure Boundary Functional Requirements and Performance Criteria*, Item 4, "The cask vent tool shall be shown by analysis or testing to be capable of flowing greater than 0.016 scfm [standard cubic feet per minute] with a differential pressure of 1 atmosphere (14.7 psi [pounds per square inch]) or less." With such a low limit on relief flow capacity, even if the vent tool were installed and accessible it might not be capable of providing the needed relief flow capacity for such an event.

- Hazard analysis documents (PRC-STP-00687 and PRC-STP-00697) and Section 3.2.1 of the PDSA, *Hazard Analysis Methods*, which list the various accident/event types considered in the PDSA, do not include a transportation fire or other transportation accidents that may mandate this and other cask design features for its functions outside the K-Basin Annex. Even if such scenarios are addressed by other design and safety basis documents, they should be provided for in the cask design and addressed in this PDSA, which addresses only the cask's critical design safety features for other-than transportation events. Note: This absence of engineered controls for events outside the Annex is inconsistent with Section 3.6.4.4, *Control Selection and Classification–Fire in the Annex*, which describes preventing a fire that could threaten the cask inside the Annex by applying several engineered preventive and mitigative controls, including concrete stops that prevent the tractor fuel tanks from entering the Annex when the trailer is being moved.

RCR response: "ACCEPT: The STS Cask is not a stamped, Section III vessel and is not described as such in the PDSA. To avoid confusion, the sentence in Chapter 4 will be revised to deleted [*sic*] reference to the BVPC [*sic*] and thus will reference only 10 CFR 71 relative to testing to 150% of the design pressure. The STS Cask will be transported to T Plant under the F-SPA (Fuel-Special Package Authorization), which ensures the shipment meets the requirements of a Risk Based packaged under DOE/RL-2001-36, Hanford Sitewide Transportation Safety Document."

EA response to RCR response: Contrary to the RCR response, PDSA, Rev 0, did describe the cask as being per "the design requirements for a BPVC, Section III, Subsection NB Service Level A vessel." The RCR response stated that this description was removed from PDSA, Rev 00. This removal did not resolve the concern, since it did not remove the 10 CFR 830 and DOE-STD-3009 requirement that the design of safety SSCs, such as the cask, be such that the public, workers, and the environment are protected. The current cask design, without an appropriate pressure relief device, does not provide such required protection. Note: PDSA, Rev 00, still contains the BPVC reference.

The PDSA clearly recognizes that transport vehicle fire in the Annex entails a level of risk requiring an SS engineered control. The probability factor of this risk, because the tractor's fuel tanks could be located in the Annex at the same time as a loaded cask, presented an unacceptable fire condition that could cause failure of the STSC and/or the cask and prompt worker fatalities or serious injuries. Therefore, the Annex truck stop was provided in the facility design as an engineered control to prevent the tractor fuel tanks from being inside the Annex when a loaded STSC on the trailer was positioned there.

The RCR response states that "a Risk Based packaged [*sic*] under DOE/RL-2001-36, Hanford Sitewide Transportation Safety Document" will address this concern, implying that this document will demonstrate acceptable risk during transportation of a loaded cask for the design without an equivalent engineered control. Risk is the product of two factors, probability and consequences. For the Annex fire, the truck stop virtually eliminated the possibility of an engulfing fire, and thereby eliminated the consequences. The "Risk Based package" would need to demonstrate a commensurate reduction in these factors in order to demonstrate a commensurate reduction in risk during transit.

However, during transit, the *probability* of an engulfing fire from the tractor's volatile fuel is relatively high (compared to when the cask is in the Annex) for two reasons: (1) the immediate proximity of the fuel to the cask, and (2) the vehicle's motion energy and thus its potential for energetic interaction with other objects or conditions with the resultant potential to rupture the tanks, as well as to provide an ignition source.

Additionally, with the current design, the probability of a cask failure without relief protection with such a fire, and therefore the probability of release of materials from the cask, is much higher than in the Annex.

Furthermore, the radiological consequences of such a cask failure have not been analyzed and would likely be significantly higher than current analyses indicate for cask failure inside the Annex. Such a failure could be much more energetic, since the current analysis only considered pressurization caused by the internal heat source from uranium corrosion and the resultant hydrogen generation. The transit fire would, in addition, entail the fire's external heat source, as well as the increased hydrogen generation effects of the exponentially accelerated uranium corrosion rate caused by the increasing internal temperature. The consequences could also be significantly higher than currently analyzed because of a cask failure mode that is not currently recognized and analyzed: failure of the cask lid bolts. Such a failure could be virtually instantaneous due to the "zipper effect" (the failure of one bolt causing load transfer to adjacent bolts, causing their failure, and so on until all bolts have failed). This type of failure would result in rapid cask and STSC depressurization and resultant steam flashing of the liquid in the STSC, which in turn would result in a higher airborne release factor than is currently analyzed for a "surface phenomenon" release.

Therefore, the overall risk (product of probability and consequences) during STSC transport to the T-Plant could be significantly higher than any event currently analyzed for the Annex. However, no engineered SS control to prevent cask overpressure, such as a pressure relief device, is provided.

The RCR response implies that the risk-based transportation document may deal with the transportation risks using administrative controls. However, DOE standards and orders recognize that administrative controls alone are inherently less effective and less reliable than engineered controls, and therefore should not be used unless engineered controls are not practicable. Since incorporating a pressure relief device in the cask design is practicable, reliance on administrative controls to address the increased risks associated with transportation of the STSCs would be inconsistent with these DOE standards and orders and with fundamental design principles.

To understand the challenges to the STS cask from a transportation fire, and therefore the need for engineered pressure relief protection, EA reviewed four related analyses: (1) SNF-13268, *KE Basin Sludge Transportation System 100% Design Report*, Rev 0, dated December 6, 2002, which contains a thermal analysis of the cask for this event; (2) Attachment 8 to that design report, Packaging Technology, Inc., (PacTec) Submittal 12329/STS 22, *Final Design Report*, which partially addresses the structural analysis of the cask for this event; (3) SNF-18162, *Thermal Analysis of Sludge Transport System for Argon Backfill and Extended Transport Window*, Revision 1, dated October 2, 2003, which is a subsequent, updated thermal analysis; and (4) PacTec Calculation 12099, *STS Cask Structural Analysis*, Revision 2, dated November 1, 2002, which is a subsequent, complete structural analysis. These last two documents appear to be the latest available documents addressing their respective subjects, the cask thermal analyses and the corresponding structural analyses. These analyses concluded that the cask structures were not challenged beyond their code-allowable limits. However, EA's assessment identified several significant non-conservatisms in these latest analyses (described in detail in Appendix E) that together have the potential to render this conclusion invalid. **Not resolved.** (See **OFI-03**.)

C4. **Double-valve isolation testing provisions**

PDSA Section 2.6.3, *Double-Valve Isolation*, discusses the SS function of double-valve isolation, which is provided in the design to ensure that no slurry backflow contaminates various general service lines that connect to the slurry transfer lines. However, the design has no indication of features for leak testing, such as test connections, and no measures are described for leak testing these SS components – particularly the check valves, which are typically not very efficient or reliable as leak-tight barriers. It should be noted that during the Technical Readiness Level-6 system testing at the Maintenance and Storage Facility (MASF), the piping system ball valves were leak checked after all operational testing evolutions had been completed. According to the test engineers, the leak test results showed zero leakage across the ball valve seats. The duration of the operational test runs far exceeded any anticipated actual sludge transfer run time. (See PDSA Section 4.3.3, *Double-Valve Isolation.*)

RCR response: "Double-valve isolation will not be tested inservice. As discussed in PDSA Section 4.3.3.4, 'System Evaluation,' full-scale testing using sludge simulants and replicating the system design has been performed. As documented in PRC-STP-TR-00664, STP-ECRTS Test Report for Valve Cycling and Leak Rate Determination, the valves passed the leak tests after 3-times the mission volume of slurry was run through the test set-up. As part of the test, the ball valves were cycled over 1000 times (2 times the number of expected production cycles). The valves have a 5-year service life. The expected mission life is 1-yr and, based on process flowsheets values, the total slurry transfer time is approximately 9 hours."

EA response to RCR response: DOE Order 420.1 (in all versions since its inception) requires (in Section 3.b.(4)(b) of the most current version) that "Facilities shall be designed to facilitate inspections, *testing* [emphasis added], maintenance, and repair and replacement of safety SSCs as part of an overall reliability, availability, and maintainability program." The RCR response appears to imply that, considering the conceptual qualification testing performed, no subsequent in-service leakage testing would be performed. Considering the short mission life required for these valves, the very short total slurry transfer time, the successful ball valve leakage testing under simulated sludge conditions, and the inherent reliability of ball valves, the described rationale in the response is reasonable for the ball valves used in this application. However, given the above-quoted DOE order requirement, the intention to not perform such testing and the supporting rationale should be provided in the PDSA. Additionally, the other valve types, such as check valves, relied upon for this function are not addressed in the RCR response. Unless these other valve types were similarly qualification tested – particularly the check valves, which typically have very poor reliability compared to ball valves (especially in the very challenging fluid conditions involving sludge) – the rationale in the RCR response for not performing surveillance testing would not be valid for these other valve types. **Not resolved.** (See **OFI-04**.)

C5. **Fire suppression water confinement analyses**

Section 2.8.8, *Fire Suppression Water Discharge Containment*, describes facility floors that are sloped to drains in each room, and then to an exterior sump, to prevent the release of potentially contaminated fire water to the environment. However, no analyses are described or identified that demonstrate sufficient flow area in the floor drains and no design features, such as curbs with sufficient heights to produce the necessary drainage head, that would provide drain flow rates greater than or equal to the maximum sprinkler discharge rates. Such flow rates are necessary to prevent uncontrolled releases to the environment.

RCR response: "44577-C-CALC-001, Fire Water Floor Drain Capacity, demonstrates that maximum possible fire sprinkler flows established in calculation 44577-F-CALC-007 can be expected to be contained in the building and enter the fire drainage sumps. This calculation is evidence that floor drains, floor configuration and room configuration meet requirements of specification section 3.10.2.1 of PRC-STP-00329 Rev 4."

EA response to RCR response: Although this concern was explicitly addressed in the revised PDSA, with two applicable calculations identified in the RCR, EA identified several concerns with calculation 44577-C-CALC-001 as follows:

- Per Section 3.0, *Scope*, this calculation only accounts for the drain flow entrance resistance of the floor drain grating and weirs into the room sump, but not for the drain piping or the drain tank conditions that may inhibit flow, such as tank level or any other potential flow resistance factors. Therefore, this analysis appears to be very non-conservative. Additionally, this analysis does not address the adequacy of the drain sump and tank capacities versus the required capacity for the design basis fire.

- The only water flow inputs to the rooms used in this calculation are from the sprinkler flows from Calculation 44577-F-CALC-007, which do not include the hose stream flows, for which the National Fire Protection Association (NFPA) 13 Code (Section 11.2.1.1 or 11.2.3.1.1, depending on whether the sprinkler system is a "pipe schedule" or "hydraulic" method design, respectively) requires sprinkler systems to be designed. No rationale is provided for this exclusion. Note: The NFPA 13 Code does not address area drainage directly, but if drainage is critical, as in this case, then the maximum potential flow to the rooms would include the hose streams.

- The calculation states in Section 8.0, Item 1.a, that the fire water flow in the Loading Bay is based on the combined sprinkler flow for both levels (the main level and the mezzanine level) of 1,023 gallons per minute (gpm). However, Section 10.0, *Results*, Item 1.a, states that the "fire flow" (presumably meaning the drain capacity) is only 986 gpm, and the table at the end of this section inexplicably lists this as "OK."

- Section 6.0, *Assumptions*, Assumption 4 states, "Doors and openings are closed and all seals are in place." However, the PDSA provides no indications that doors and openings are designed to withstand such conditions (e.g., with appropriate opening directions; suitable latches, hinges, and seals; and adequate strength to withstand the static water pressure) and does not describe any administrative controls to ensure that such doors and openings are in the required states, i.e., that they are maintained closed (or open, if so required) and that the seals are properly maintained and tested.

- Per Section 10.0, *Results*, Item 3, for Room 103, the High Efficiency Particulate Air (HEPA) Room, credit is taken for flow into the adjacent room, presumably through an open door or under a door. However, the PDSA lists no controls to ensure that such a pathway is adequate and kept unobstructed. Additionally, if the door is a fire barrier, keeping it open for drainage is inconsistent with its fire barrier function.

- The calculation is based an unstated assumption that the drain gratings would not be obstructed by any debris, such as paper or rags, that could be washed into them. However, the PDSA does not describe any controls to ensure that this assumption is protected.

Not resolved. (See **OFI-05**.)

C6. **Fire suppression water confinement qualification**

Nothing in the PDSA, including Section 2.8.8, *Fire Suppression Water Discharge Containment*, addresses the classifications, qualifications, and failure modes of the power supplies and control circuitry of the floor drain system solenoid valves, all of which must operate in response to a fire alarm. The PDSA does not address the pumps that must operate in order for this system to function or the bounding consequences if any of these SSCs fail to function as intended. All of these aspects of the system should be addressed in the PDSA with regard to the environmental protection mandates of the regulations, codes, and standards.

RCR response: "The Fire Suppression Water Discharge Containment is a general service system. Section 2.8.8 will be revised to include the information on page 25 of the Fire Hazard Analysis (PRC-STP-00499) addressing reasonable assurance of operation. The bounding consequences due to resuspension from a liquid pool do not exceed evaluation guidelines. As documented in PRC-STP-00698, the dose consequences from a liquid spill are 7.5 (rem) to the collocated worker and 12 rem to the facility worker. Approximately 97% of this dose is from the initial spill with the remainder resulting from pool resuspension over an 8-hr period."

EA response to RCR response: The first sentence about inserting words addressing "reasonable assurance of operation" of the system is reasonable. However, PDSA, Rev 00, does not include such words, or any words about not exceeding the evaluation guideline. **Not resolved.** (See **OFI-06**.)

C7. **Incorrect double check valve description**

Section 2.8.11, *Modified KW Basin Annex Instrument Air System*, next to last sentence, states, "Double check valves prevent loss of air in the event of line breaks in the system." It is not clear where in the system such check valves are located in relation to the air sources so that they can perform this function.

RCR response: "Based on the comment, Section 2.8.11 and the piping and instrumentation diagrams (P&IDs) were reviewed by Nuclear Safety and the process Design Authority. The review concluded that the sentence is incorrect and will be deleted."

EA response to RCR response: Resolved.

C8. **Improper SSC classification and qualification**

The following items illustrate a concern about improper classification and qualification of SSCs:

- Section 3.2.4.3, *Support Systems and Interface Design Criteria*, describes three classification criteria for supporting SSCs. The third criterion states, "Support SSCs to SS SSCs that mitigate or prevent accidents with the potential for *significant localized consequences* [emphasis added] need not be classified as SS." This statement is non-conservative with respect to the requirements of DOE-STD-3009-94, 2006, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Documented Safety Analyses*, which makes no classification exception for "support" SSCs with respect to "significant localized consequences." Although this statement is consistent with the previous DOE Guide 420.1-1, it is not consistent with the current version, 1A, dated December 4, 2012, which correctly eliminated this classification criterion. (Although the PDSA correctly references this latest version of DOE Guide 420.1-1, it fails to follow the guidance regarding this criterion.) Further, in at least the following two specific cases the PDSA indicated support SSCs not classified as SS based on this non-existent criterion: for electrical power for the oxygen analyzer and electrical power for the Transfer Line Service Box (TLSB) leak detection alarm. Considering these examples, other support SSCs should be verified not to have been incorrectly exempted per this criterion.

RCR response: "The DOE-approved code-of-record (HNF-44226, Rev 2) under which the STP [Sludge Treatment Project] is executed specifies DOE G[uide] 420.1-1, therefore the criterion is applicable. The reference to DOE G[uide] 420.1-1A is in error and will be corrected. It is noted that, relative to TLSB leak detector, the indicator light (ECRT-IL-116) is fail-safe on loss of power. Under normal operating condition, all indicator lights on Safety Control Panel ECRT-PNL-103 are energized and are 'green.' If ECRT-IL-116 is de-energized for any reason, operators will respond in accordance with the TLSB Leak Detector Alarm Response Specific Administrative Control. The PDSA will be revised to provide this information. Relative to the oxygen analyzer, the associated LCO [limiting condition of operation] will require verifying operability prior to use. If, during the inerting process, power was lost, the analyzer would be declared inoperable and the specified corrected actions would be taken."

EA response to RCR response: With respect to the first sentence, although DOE may have approved a code-of-record that specified DOE Guide 420.1-1, that approval does not diminish the fact that the code-of-record revision was non-conservative and in error with respect to the requirements of DOE-STD-3009-94 concerning the classification of support SSCs. DOE Headquarters subsequently recognized this error and corrected it in the current revision of DOE Guide 420.1-1. However, regardless of this current revision, compliance with DOE-STD-3009-94 requires that support SSCs for SS SSCs must also be classified as SS, without exempting support SSCs associated with events of "significant localized consequences." If DOE accepts the PDSA's current non-compliance with DOE-STD-3009-94 for this project, this acceptance and the rationale should be explicitly described in the PDSA.

Although the subsequent sentences in this RCR example's response are accepted as factually correct and relate positively to the intended safe operation of this equipment, they are not relevant to the point of the concern; the important point is that these support SSCs are non-compliant with DOE-STD-3009-94 with respect to their non-classification as SS. **Not resolved.**

Section 3.6.6.4, *Control Selection and Classification–High Winds*, describes one of the SS engineered controls as the Auxiliary Ventilation System, which prevents a hydrogen explosion in the STSC by providing nitrogen purge gas flow to the STSC in order to maintain the hydrogen level at less than 25 percent of the lower flammability limit (LFL). Section 3.6.6.4 states, "High winds could initiate a hydrogen explosion in an STSC if sludge was present in the STSC *and the high winds damaged the Annex Ventilation System or resulted in a loss of power to the Ventil*ation System *or damaged the Auxiliary Ventilation System* [emphasis added]." Section 1.4.3, *Extreme Winds*, states that the wind-driven missile criteria are "a nominal 2-by-4-in. [inches] timber plank weighing 15 lb [pounds]…and a maximum speed of 50 mph." Although Table 1-3, *Wind Load and Wind-Driven Missile Design Criteria*, and Table 3-61, *High Wind Initiated Hydrogen Explosion Summary*, imply that the "Auxiliary Ventilation System bottle rack [is] protected from [a] design basis wind event," it is implied elsewhere, in Section 3.6.6, *Natural Phenomenon–High Winds*, that the SS components of the Auxiliary Ventilation System are not protected from wind-driven missiles. (See Table 4-8, *Auxiliary Ventilation System Functional Requirements and Performance Criteria*, Item 8 and Section 4.3.8.4, *System Evaluation*, subsection labeled *Environmental Design*, third paragraph.) Failure to protect against wind-driven missiles is unacceptable for SS controls credited with preventing an accident, in this case a hydrogen explosion in the STSC. It is possible that such protection was not provided because the wind design criteria was designated Performance Category (PC)-2, simply because the radiological consequences from such an explosion were below the PC-3 threshold for missile protection. If this was the PDSA rationale, then the PDSA did not consider the complete spectrum of consequences; in this case, the consequences should include prompt worker fatalities or serious injuries from the non-radiological direct effects of a hydrogen explosion. The PDSA should be very clear about the protection of SSCs from wind-driven missiles and from other natural phenomena in the context of all the credible worker consequences. If this protection is not provided, the rationale for such design should be fully provided in the PDSA.

RCR response: "There is no evaluation guideline that specifies the application of PC-3 design criteria based on facility worker consequences. The April 15th, 2009, Owendoff memorandum titled, *Implementation of DOE-STD-1189, Integration of Safety into the Design Process for Environmental Management Activities*, specifies application of PC-3 to non-seismic NPH [natural phenomena hazards] based on the off-site and collocated worker radiological and toxicological consequences. The DOE-approved code-of record specifies DOE Guide 420.1-2, which states that 'PC-2 SSCs are meant to…prevent physical injury to in-facility workers.'"

EA response to RCR response: EA's key concern is whether the SS Auxiliary Ventilation System should be protected from wind-driven missiles from a design basis wind event. If it is not protected, many of the system's key components located outside the Annex are subject to failure from a missile strike. Such a failure could result in a hydrogen explosion of an STSC or cask, which could result in prompt worker fatalities or serious injuries and the release of radioactive materials from the facility. Per the PDSA, if the system is designated as PC-2, then missile protection is not required, but if it were designated as PC-3, such protection would be required. The system is currently designated as PC-2 and unprotected.

The applicability of the quotation cited in the RCR response for this system must be considered in the context of the document from which it was extracted, DOE Guide 420.1-2. Although DOE Guide 420.1-2 does not describe any specific system to which the quotation cited in the RCR response is applicable, it provides general criteria, with examples, against which SSCs should be evaluated to determine the most appropriate PC to apply.

The PC-2 criterion states, "PC-2 SSCs are meant to ensure the operability of essential facilities (e.g., fire house, emergency response centers, and hospitals) or to prevent physical injury to in-facility workers. When safety analyses determine that local and limited confinement of low-hazard materials is required for worker safety, PC-2 designation should be used for the SSCs involved. In these cases, PC-2 designation may apply to SSCs, such as drums, packaging, gloveboxes; local HEPA filters; air flow control systems (ventilation and dampers); and room air monitors, alarms, corridors, stairways and doors, pager systems, and emergency lighting important to evacuation. Design of PC-2 SSCs should result in limited structural damage from design basis natural phenomena events to ensure minimal interruption to facility operation and repair following the event."

This PC-2 criterion cites "low hazard" materials and facilities, such as fire houses and hospitals, as examples of the types of materials and facilities that should prevent injury to in-facility workers. The materials in the Annex, on the other hand, are not low hazard; the Annex is unrelated to the types of facilities described; and the consequences that must be protected against are not just injury, but rather prompt worker fatalities or serious injuries as a result of a hydrogen explosion. The PC-2 criterion also cites preventing "interruption of facility operation, and minimal repair following the event," which is a considerably lower level of event concern than applies to the Auxiliary Ventilation System. All applicability elements of the PC-2 criterion are substantially below the characteristics of the Auxiliary Ventilation System, and therefore, PC-2 is not the appropriate PC for this system.

On the other hand, the PC-3 criterion states, “PC-3 SSCs are those for which failure to perform their safety function could pose a potential hazard to public health, safety, and the environment because radioactive or toxic materials are present and could be released from the facility as a result of that failure. PC-3 SSCs would prevent or mitigate criticality accidents, *chemical explosions* [such as a hydrogen explosion – emphasis added], and events with the potential to release hazardous materials outside the facility. Design considerations for these categories are to limit facility damage as a result of design basis natural phenomena events so that hazardous materials can be controlled and confined, occupants are protected, and the functioning of the facility is not interrupted.” The first sentence cites SSCs whose failure “could pose a potential hazard to public health, safety, and the environment because radioactive or toxic materials are present and could be released from the facility as a result of that failure.” It cites no specific exposure thresholds, just a radioactive or toxic hazard that could be released. These elements of the criterion are met by the Auxiliary Ventilation System. The second sentence cites chemical explosions with the potential to release such hazardous materials outside the facility, without qualifying the degree of such release. The Auxiliary Ventilation System also meets this element of the criterion. The third sentence cites design for control and confinement of hazardous materials and protection of facility occupants, which is also met for the Auxiliary Ventilation System.

It is clear that Auxiliary Ventilation System closely fits the PC-3 criterion elements cited in DOE Guide 420.1-2, but is not enveloped by the PC-2 criterion elements. The Auxiliary Ventilation System should be designated as PC-3 and provided with wind-driven missile protection, as required by PDSA Table 1-3. **Not resolved.**

Section 3.6.7.4, *Control Selection and Classification–Snow or Ashfall*, states, “The collocated worker consequence resulting from a snow and ashfall induced hydrogen explosion is significantly below 100 rem. Therefore, vulnerable portions of the Auxiliary Ventilation System (i.e., those components exposed to snow and ashfall loads) shall be designed to PC-2 snow and ashfall loads in accordance with PRC-PRO-EN-097.” The possibility of prompt worker fatalities or serious injuries from these events, including any non-radiological consequences, should be considered for this and all other areas in the facility design when these situations exist.

The RCR did not provide a response for this example; however, the RCR response is similar to the previous example, and EA’s response is identical to that provided in the previous example (i.e., the Auxiliary Ventilation System should be designated as PC-3). **Not resolved.**

Table 4-8, *Auxiliary Ventilation System Functional Requirements and Performance Criteria*, Item 7 states, “The Auxiliary Ventilation System shall meet SDC [Seismic Design Category]-1 requirements. Note: The Auxiliary Ventilation System is SS for facility worker protection, whereas the SDC-1 designation is based on the unmitigated dose to the collocated worker.” Also, Section 4.3.8.4, *System Evaluation*, in the subsection titled “Specific Criteria,” last paragraph, states, “The Auxiliary Ventilation System is also credited with performing its safety functions during and after an SDC-1 seismic event.” As with the windstorm event discussed above, despite the fact that the collocated worker’s unmitigated radiation exposure is below the consequences threshold for SDC-3, the direct, non-radiological consequences of this system’s failure to provide the described prevention of an explosion could be serious injury or prompt worker fatalities or serious injuries. Therefore, the Auxiliary Ventilation System should be seismically qualified as SDC-3.

RCR response: “There is no evaluation guideline that specifies the application of SDC-3 design criteria based on facility worker consequences. SDC is based on the collocated worker and public unmitigated consequences per Table A-1 of DOE-STD-1189.”

EA response to RCR response: Resolved.

Overall EA response to C8 RCR example responses: Only one of the above examples was resolved. **Not resolved.** (See **OFI-07**.)

C9. **Wording error**

Section 3.6, *Results of Analysis of Accidents*, states in mid-paragraph, "As discussed in Section 3.4, 'Hazard Analysis Results,' *three* DBAs [design basis accidents] were identified [emphasis added]." The next sentence states, "*both* [emphasis added] of these DBAs can be initiated by operational events." The described number of DBAs is inconsistent between these two sentences.

RCR response: "The word 'both' was changed to 'each.'"

EA response to RCR response: Resolved.

C10. **Non-conservative hydrogen generation analysis temperature**

Section 3.6.2, *Explosion – Hydrogen Deflagration or Detonation*, discusses this event in the STSC or the cask. Page 3-80, second paragraph, states, "PRC-STP-CN-CH-00804, *STP ECRTS Hydrogen in STS Cask Prior to Inerting*, calculates that if the lid is placed on the cask upon completion of STSC inerting, it takes approximately 26 hours to reach 25 percent of the LFL in the cask headspace." This statement's supporting analysis was performed at 25 $^{\circ}$C. However, for the transportation fire scenario previously discussed (see Observation C3, last bullet), the hydrogen generation rate (an exponential, positive feedback function with respect to temperature) could be significantly accelerated because of the increased temperature of the uranium-water reaction. Consequently, the time to reach 25 percent LFL could be significantly less than stated. Additionally, for the cask overpressure failure due to a transportation fire scenario previously discussed (see Observation C3, last bullet), the instantaneous release of accumulated hydrogen from the STSC into the flames could result in a virtually simultaneous hydrogen explosion. This scenario has not been analyzed.

RCR response: "The transportation fire occurs in the presence of the tractor and trailer, and the tractor is not present in the KW Basin Annex except when leaving the trailer/cask/STSC to be loaded, and after the STSC/STS is inerted and ready for transportation. Further, the loading bay design explicitly precludes the tractor fuel load being within the Modified Annex, and the ramp external to the roll up door is sloped away from the building to prevent a fuel spill from draining into the structure. The transportation fire scenario has historically shown that the inerted load survives the fully engulfing transportation design basis event without failing the cask. However, the transportation fire is not a specific design basis event for the Modified Annex. Rather, it is a transportation scenario which is not analyzed in the PDSA. However, the controls provided for the hydrogen explosion include provisions described above to keep the tractor fuel load out of the Modified Annex. A range of fires are considered in the FHA [fire hazard analysis] and in the PDSA, and the potential for a fire to contribute to an explosion is considered. It is worth noting that the historic fully engulfing transportation fire analysis found that there was very modest increase in sludge temperature due to the cask and load design including very large thermal inertia. In addition, the above mixes a cask overpressure event with a hydrogen explosion event. In any configuration which would support a cask overpressure, the cask would be closed and would respond slowly to a fire (as in the transportation fire), the STSC has been inerted, and long time periods would be required to reach explosive conditions or overpressure, and the radiological source material for a radiological release would be within the inert STSC within the STS. Explosion of gases released from the

STS in an overpressure event would not influence the radiological material within the STSC. The analyzed hydrogen explosion in the STSC event (without the protection of the STS) would clearly be bounding for considering this event as stated in sludge temperature due to the cask and load design including very large thermal inertia."

EA response to RCR response: The first two sentences of the RCR response address a fire in the Annex, which is adequately covered in the PDSA and irrelevant to this EA concern. The third sentence states, "The transportation fire scenario has historically shown that the inerted load survives the fully engulfing transportation design basis event without failing the cask." Although this statement is accurate, the "historical" analyses may be invalid. As addressed in Observation C3 and Appendix E, the historical cask thermal and structural analyses for the transportation fire (calculations SNF-18162, *Thermal Analysis of Sludge Transport System for Argon Backfill and Extended Transport Window*, Revision 1, dated October 2, 2003, and PacTec Calculation 12099, *STS Cask Structural Analysis*, Revision 2, dated November 1, 2002, respectively) contain significant non-conservatisms, which together would challenge the validity of this statement.

The eighth sentence states, "It is worth noting that the historic fully engulfing transportation fire analysis found that there was very modest increase in sludge temperature due to the cask and load design including very large thermal inertia." However, contrary to this response, historical calculation SNF-18162, Table 8-3, *HAC Results*, shows that the maximum bulk sludge temperature for this event would be 159 degrees Fahrenheit (°F) (71 °C), which is significantly more than a "modest" temperature increase, even without reconsidering the non-conservatisms identified in Appendix E by EA. **Not resolved.** (See **OFI-08**.)

C11. **Exhaust stack ground impact accelerations**

Section 3.6.5.4, *Control Selection and Classification–Seismic Event*, starting on page 3-108, discusses the seismic failure of the 175 foot tall KW Reactor exhaust stack and its impact on the Annex. Section ES.6, *Safety-in-Design Conclusions*, states, "To mitigate the risk associated with potential changes to the seismic hazard curves, the seismic design input has been conservatively increased as documented in PRC-STP-00586." However, the PDSA, PRC-STP-00586, and other documents do not consider the possible effects on the Annex of the additional local ground accelerations that would occur due to the impact of the stack on the ground, which could be significant if addressed alone or in combination with the seismic accelerations. (Additionally, the considerations and effects of PRC-STP-00586 on the Annex design are not discussed anywhere in the main body of the PDSA, only in the executive summary.)

RCR response: "The shock wave generation from the stack falling as a result of a greater than 0.2g earthquake and response of the KW Basin Modified Annex to that shock wave in combination with the design basis earthquake (DBE) has not been explicitly analyzed. The KW stack [a different stack from the Annex stack] has been analyzed to survive a 0.2g earthquake, and further to likely fail in buckling if it does fail (note that the stack was historically shortened to increase the level of earthquake that it would survive). Thus laying down the full length to be near the modified annex and falling in the direction of the annex combine to be very low probability, and the failure would only occur in the presence of a substantial earthquake. Similar stacks have been dropped as part of decontamination and decommissioning at Hanford without observation of appreciable shock waves. The attached explains the shock wave effect of a large structure being explosively lowered. It is a relatively minor effect given the substantial earthquake that would be the simultaneous impact to the Modified Annex structure. It is judged that the conservatism in the seismic analysis methodology would envelope the effects of this relatively minor additional acceleration. In addition, seismic switches are installed in the ECRTS design which interlock to shutdown transfers (prevent seismic induced spray release) at levels of earthquake less than that would result in stack failure. http://www.usgs.gov/blogs/ features/ usgs_top_story/buildings-demolition-gives-rise-to-seismicknowledge/"

EA response to RCR response: The response makes a probability statement, which is not enveloping, to dismiss this concern. The RCR response states, "Thus laying down the full length [of the stack] to be near the modified annex and falling in the direction of the annex combine to be very low probability." However, since the stack is only 175 feet tall, the maximum distance of the center of impact could only be about 200 to 300 feet from the Annex, which could still have the potential to produce substantial additional ground acceleration at the Annex. It also states, "Similar stacks have been dropped as part of decontamination and decommissioning at Hanford without observation of appreciable shock waves." EA has three misgivings about this statement: (1) For safety reasons, it is unlikely that observers were within 200 to 300 feet of the location of the stacks' bases, and therefore, would not have experienced the intensity of a shock wave of concern; (2) even if observers were located within that range, it is unlikely that they were focused on comparatively evaluating the ground acceleration produced against a previously observed 0.2g earthquake; and (3) it is unlikely that any acceleration instrumentation was positioned close enough to the drop zone to record the event. Therefore, this questionable anecdotal statement in RCR's response (not empirical data or directly applicable analysis) is an invalid seismic design basis element for the Annex. The RCR response also states, "In addition, seismic switches are installed in the ECRTS design which interlock to shutdown transfers (prevent seismic induced spray release) at levels of earthquake less than that would result in stack failure." Although this is an accurate statement, it is not necessarily relevant to determining the maximum ground acceleration that could occur at the Annex, which is the subject of EA's concern. Finally, the RCR response provided an Internet link to a United States Geological Survey (USGS) article related to the implosion demolition of a building and the resulting ground accelerations. The referenced article stated that the ground accelerations at the sensors were expected to be similar to a "magnitude 2 earthquake." Although that may have produced "relatively mild" accelerations at the sensors, the article also stated that the sensors were placed at a two-mile radius from the building. Since the energy from the drop would be dissipated exponentially with the distance from the impact point, it would be reasonable to conclude that the ground accelerations within 200 to 300 feet from the building being demolished would have been significantly higher than a "magnitude 2 earthquake." Rather than supporting the RCR response, this article reinforces EA's concern.

In a similar situation several decades ago, the Nuclear Regulatory Commission discovered that a design basis maximum loss-of-coolant-accident in boiling-water-reactor commercial power facilities could cause a water lift phenomenon in their pressure suppression pools. When the water fell back to its normal position, the resultant ground accelerations actually exceeded the design basis seismic accelerations in many of the plants. The resultant loads (deemed hydrodynamic loads) exceeded the design bases for many of the plants' SSCs, which in many cases required redesign to reinforce and/or replace some of the structures and systems. This experience illustrates EA's concern that the ground impacts of nearby heavy objects, such as the stack, is a valid consideration that should not be dismissed without a proper basis. **Not resolved.** (See **OFI-09**.)

C12. **Wording error**

Table 3-42, *Hydrogen Explosion in the STS Summary,* first column, seventh bullet, lists pressure safety valve (PSV) ECRT-PSV-601 as a potential preventive design feature. It is not clear whether this valve would be located on the cask itself or on some attachment. It is also not clear how a PSV would prevent a hydrogen explosion. If this feature was mistakenly placed in the wrong column and was intended instead to describe the mitigative effects of a hydrogen explosion in the cask by relieving the resultant pressure, then this is not a valid design, since a conventional PSV would likely not be capable of providing such protection, since its reaction time would likely be inadequate to address the pressure pulse of a hydrogen explosion.

RCR response: "ECRT-PSV-601 is part of the Inert Gas System and is located on the nitrogen supply line where the system pressure is controlled to 200 pounds per square inch psig. It was identified as a preventive control in the HAZOP [hazard and operability analysis] for Inert Gas System process deviations associated with high pressure causing failures of the supply line and thus an inability to properly inert and pressurize the cask."

EA response to RCR response: Resolved.

C13. **Control SSCs' classification and qualification**

Table 3-54, *Seismic Engineered Controls*, Item No.1, lists the seismic shutdown switches and Nuclear Safety Interlock I-1 as SDC-3 SS controls. Per Table 3-56, *Seismic Initiated Spray Release Summary*, these controls are to "Prevent a seismic induced spray release of slurry by terminating slurry transfers during sludge retrieval and transfer upon detection of seismic motion." These devices are intended to accomplish this function by terminating the power and air supplies to booster pump ECRT-P-101 and overfill recovery pump ECRT-P-301, respectively, thereby removing the motive pressure necessary to cause a spray. In order for this function to be accomplished, the final devices that supply power to these pumps, such as the power breaker for the booster pumps, must perform the active safety function of changing state. These components should have the same safety classification (i.e., SS) and seismic qualification (i.e., SDC-3) as the seismic shutdown switches and Nuclear Safety Interlock I-1, as well as any other intermediate devices whose active or passive failure could prevent the final safety functions from being carried out. The safety classification (but not the seismic qualification) of the contactors is addressed in Section 4.3.7.2, *System Description*, Item 1, which states, "The rack on which the panel is mounted also supports the *SS contactors* [emphasis added] (ECRT-CNTAC-101, and ECRT-CNTAC-102) to interrupt power to booster pump ECRT-P-101." However, this concern is not addressed for the air supply components. (See 4.3.6.2, [Nuclear Safety Shutdown Interlock I-1] *System Description.*)

RCR response: "The release safety function specifies terminating slurry transfers…during sludge retrieval and transfer. Terminating slurry transfers during sludge retrieval and transfer is accomplished by removing power to pump ERCT-P-101. There are no air supply components associated with removing power to this pump. The second paragraph on page 3-106 of the PDSA describes how Interlock I-1was designed to remove the air supply to overfill recovery pump ECRT-P-301, but that this is no longer a release safety-significant function because the associated dose is now calculated to be < 100 rem to the collocated worker. Although a spray release during overfill recovery does not require safety-significant controls, a spill within the TLSB during overfill recovery could result in a hydrogen explosion. For seismic events, the credited control is safety-significant transfer line piping and hose within the TLSB designed to SDC-2 as shown in Table 4-10, Item #3. (Note that the SDC-2 is the bounding SDC based on a spray release during sludge retrieval and transfer, for hydrogen explosions, the associated SDC is SDC-1)."

EA response to RCR response: Resolved. (Note: Section 4.3.7.2, the second paragraph in PDSA, Rev 00, was previously Item 1 in the list of panels in PDSA, Rev 0, and its number was apparently inadvertently deleted in PDSA, Rev 00.)

C14. **Isolation valves' support SSCs**

Similar to the above item, Section 4.3.3, *Double-Valve Isolation (includes Handswitch ECRT-HS-123)*, describes the SS double-valve isolation between the ECRTS and connecting SSCs, including the following: "With the exception of the manual ball valve ECRT-V-301 and the check valves, these valves are operated remotely, either from a control panel in the Annex sludge loading bay, or from a control panel in the KW Basin administrative area. These valves are specified to fail in a closed position, making this the failure mode on loss of compressed air, loss of electrical power, or loss of signal. Closed is the safe position for their double isolation safety function." It further states, "**Support Systems—**The double-valve isolation systems do not rely on support SSCs to perform their safety function to [close]. Although . . . AOV [air operated valve] operation depends on the availability of electrical power, instrument air, and the signal line to the valve, the design specifies that these valves fail closed should any of these support systems be interrupted." It is unclear from these descriptions whether any of these valves have non-safety SSCs in their motive energy supplies (e.g., electricity or compressed air) that must change state upon demand for the SS isolation valves to close or, upon loss of their motive energy, in order for the SS valves to close. If they do contain such SSCs, those SSCs should also be classified and qualified consistent with the valves.

RCR response: "There are no non-safety SSCs whose failure could prevent the SS isolation valves from performing their safety function."

EA response to RCR response: Resolved.

C15. **Slurry transfer line pressure relief**

Section 4.3.2, *Slurry Transfer Line Rupture Disk*, describes the rupture disk that performs the safety function of preventing the spray release of slurry by preventing overpressurization of slurry transfer lines during sludge retrieval and transfer. Section 4.3.2 describes the setpoint of the rupture disk as "220 psi, ± 5%." However, DCN-STP-ECRTS-129 changed the nominal setpoint to 115 psig to provide pressure protection for the Decant/Filter System, which is branched into the Transfer System, in case the double isolation valves between the two systems leak. Although this change is conservative with respect to the integrity of the Decant/Filter System, it is not conservative with respect to the probability of unintentional slurry release into the pool. Such an event may produce consequences within SS allowable limits (although the radiological consequences of this event are not addressed), but because this setpoint is close to the nominal operating pressure of the Transfer System (which does not consider credible normal transients), this change would appear to significantly increase the probability of such an event, which could also have substantial, unnecessary consequences for radiation control, operations, and facility cleanup.

RCR response: "The change in setpoint was mandated during final design review for enhanced compliance with B31.3, 'Process Piping,' requirements. It has been recognized since conceptual design that opening of the rupture disk will result in some quantity of slurry being released underwater in the basin. To minimize operational impacts, the rupture disk is instrumented to alert operators to take corrective action should it open. Relocating the rupture disk further away from the booster pump skid to minimize pressure pulses (and thus spurious failures) is being evaluated as part of ongoing, full-scale testing at MASF."

EA response to RCR response: Although the RCR response appears to be technically valid, the PDSA description is still inconsistent with the current design. The PDSA indicates in several places that the rupture disk setpoint is 220 psig, whereas the current design setpoint is 115 psig. The PDSA should be revised to reflect the appropriate setpoint changes. **Not resolved.** (See **OFI-10**.)

C16. **Wording error**

In Section 3.6.6.2, *Radiological Source Term–High Winds*, the first sentence states, "The bounding accidents associated with a *seismic event* [emphasis added] are the operational spray release and the hydrogen explosion." Since this section's subject is high winds, not a seismic event, this sentence, in this context, does not make sense. It should refer instead to a high wind event.

RCR response: "Wording was changed to 'high winds.'"

EA response to RCR response: Resolved.

C17. **Lightning protection**

Section 3.6.8, *Natural Phenomenon–Lightning Strike*, appears to discuss all relevant aspects of this event, but there is no discussion of any lightning protection features for the Annex. Regardless of nuclear consequences and probabilities, lightning protection is typical industrial practice for such a building to protect personnel and property, since the building will exist and be occupied for quite some time. Also, it is unclear why probabilistic reasons are given for not providing such engineered controls, when such reasons are not considered valid for other natural phenomenon events (NPEs). Additionally, other administrative controls, such as stopping operations that could produce a spray event, could reasonably be applied for an approaching thunderstorm (or any other NPE with readily observable precursors, such as snowfall or ashfall).

RCR response: "ACCEPT: The text will be revised to clarify that lightning protection is not required by NFPA 780, Standard for Installation of Lightning Protection Systems, based on the lightning strike frequency. This is currently described in Section 16.3, 'Lightning,' of the ECRTS FHA. A similar argument was not made for seismic events because the analogous frequency is an order of magnitude higher (i.e., 1E-6/yr)."

EA response to RCR response: Resolved.

C18. **Wording error**

In Section 3.6.9.4, *Control Selection and Classification–Low Temperatures*, first paragraph, last sentence, make the word "Table" plural, and add "3-72" after "and."

RCR response: "Accept – will revise per comment."

EA response to RCR response: Resolved.

C19. **Low temperature hose-in-hose qualification**

Section 3.6.9.4, *Control Selection and Classification – Low Temperatures*, second paragraph, last sentence, states, "The SS above-water slurry transfer line within the outdoor horizontal shielded hose chase is a single length of HIH [hose-in-hose] that is rated for the design basis low ambient temperature of -27 °F." Although the design temperature rating is an important factor for the hose, it is unclear whether this envelopes the possibility that a hose filled with water might freeze; freezing could produce stresses exceeding the hose's strength properties at that temperature. (See Table 4-4, *Slurry Transfer Line Secondary Confinement System Functional Requirements and Performance Criteria*, Item 5.) Additionally, Section 4.3.4.4, *[Slurry Transfer Line Secondary Confinement System] System Evaluation*, in the section labeled *Environmental Design*, states, "The HIH specification requires the vendor to demonstrate that the inner hose will withstand freezing." This statement does not explicitly require the hose to withstand the forces generated by water freezing in the hose and does not account for the altered material properties, such as tensile strength and susceptibility to brittle fracture, of the hose at the minimum temperature, for both the EPDM (a synthetic rubber) and the metal components.

RCR response: "The hose-in-hose vendor has performed design qualification testing, subjecting the hose to repeated freeze/thaw cycles followed by a burst test at room temperature. The hose assembly did not rupture with frozen water, and there was no degradation of the rated hose burst pressure afterwards."

EA response to RCR response: Section 3.6.9.4 addresses this concern by adding the following statement in Section 3.6.9.4: "The HIH vendor will provide evidence that the hose assemblies have been tested for limited impact if contained liquids were to freeze." **Resolved.**

C20. **Low temperature specific administrative control (SAC)**

Section 3.6.9.4, *Control Selection and Classification – Low Temperatures*, second paragraph, also states, "If liquid in the above-water slurry transfer lines were to freeze and cause a failure, then a spray release could occur when the liquids thawed and the line was subsequently pressurized. The principal control strategy is to prevent a low temperature induced spray release by crediting an environmental control SAC." The PDSA does not specify the details of this SAC or how they are to be implemented.

RCR response: "ACCEPT: An environmental control SAC was not selected for the outdoor portion of the horizontal shielded hose chase because vendor testing will show that a HIH containing water within the chase can withstand freezing temperatures without failure. The PDSA text in Sections 3.6.9.4 and 4.3.1.4 will be revised to clarify this point."

EA response to RCR response: Resolved.

C21. **HIH low temperature protection**

Tables 3-71 and 3-72 raise the following concerns: Contrary to the statement cited in the previous item, no positive freeze preventive control is in place for all of the HIH, so flush water in the unprotected "Doghouse" portion of the inner hose could freeze. Additionally, the PDSA describes no controls for this scenario that would ensure that a hose rupture due to freezing would be detected before the system is started. Note: Proper design of heat trace was a major issue during the previous HIH transfer line design and operation.

RCR response: "General-service freeze prevention is provided the KW Basin, the horizontal shielded hose chase, and the Modified KW Basin Annex. As stated in Section 4.4.9, 'Environmental Control,' if temperatures outside the specified range are measured (40-100°F) in the KW Basin or the Modified KW Basin Annex, the facility will be placed in a safe condition and a recovery plan developed. The recovery plan would logically include an evaluation of potential damage to above-water slurry transfer lines and, based on the results of the evaluation, might require line integrity testing. In contrast to the hose-in-hose previously used for the 105 KE East-to-KW sludge transfer where the heat trace was directly attached to hose-in-hose; for ECRTS the heat trace is run within the horizontal shielded hose chase (PDSA, Figure 2-11). With this design, the heat trace functions as a 'space heater' and provides protection for all of the hoses within the chase. Calculations are documented in 44577-MCALC-101, *KW Annex Trace Heat Sizing Calculation*. The heat trace terminates where the horizontal shielded hose chase interfaces with the KW basin shielded hose chase (aka, the 'Doghouse'). The Doghouse does not contain any heat trace. Because the majority of the Doghouse is within the KW Basin, the air temperature within the Doghouse will be primarily dictated by the air temperature in the KW Basin, which is environmentally controlled. Approximately 3-ft of the Doghouse extends outside of the KW Basin where it interfaces with the horizontal shielded hose chase. Because the doghouse and the horizontal shielded hose chase share a common atmosphere, the air temperature in the Doghouse will be heated to some degree by operation of the heat trace. The concrete shielding associated with the horizontal shielded hose chase and Doghouse provides a significant thermal mass that will function to minimize rapid air temperature fluctuations."

EA response to RCR response: Resolved.

C22. **STSC failure mode**

In Section 3.6.13.2, *Radiological Source Term–Over-Pressurization Release*, the section addressing STSC failure states, "A breach of the STSC at the bottom or side is less likely with the cask capturing most of the released material through impingement." This statement is questionable for several reasons:

- First, what is the basis for assuming that a breach is less likely to occur at the bottom or sides than at the top?

- Second, if the first assumption is true, why are the effects of the breach at the lower locations addressed in the second phrase ("with the cask capturing most of the released material through impingement")?

- Third, the point of the sentence's second phrase appears to be that a breach at the sides or the bottom of the cask, with the maximized impingement, would be less enveloping than a breach at the top, as indicated by the PDSA's subsequent discussion. The breach-at-the-top scenario that is subsequently analyzed is described as a "surface phenomena" event with an airborne release fraction (ARF) of 1E-3. However, that ARF does not appear to be enveloping using that scenario, since it does not consider a loss of the STSC head due to a "zipper effect" failure of the head flange bolts because of high pressure and temperature in the vessel, either from the uranium metal-water reaction, or from a transportation fire (see Observation C3), or from a combination thereof. This chain of events could result in a steam-flashing, shotgun-type blast upward out of the STSC that would produce a much higher ARF than was analyzed.

RCR response: "The sentence will be revised for clarity to read: 'All the penetrations of the STSC occur at the top. With the most likely point of failure at the penetration welds and the not the STSC vessel body, the top of the STSC is considered to be the most likely location for a failure. The consequences of breach through the bottom or side of the STSC would be bounded by the release from the top because of the restraining effect of the cask."

EA response to RCR response: The revised wording is clearer. However, it still does not address the last and most significant point of EA's concern: the ARF used in the release analysis. Two credible failure modes should be considered: (1) a gradual, progressive failure at a penetration nozzle weld that would allow a relatively slow pressure release and that would not likely progress to a large opening due to the force/stress relief as the pressure subsided; and (2) a relatively instantaneous failure producing a large opening that would result in an almost instantaneous pressure loss. Although the first failure mode would be more likely for a penetration nozzle weld (assuming a sound weld) due to the metal's ductility and would be consistent with the consequence analysis that was performed (analysis assumed a "surface phenomena," and the corresponding ARF that was selected is low), an additional credible failure mode of the second type could occur but was not examined: loss of the vessel lid due to lid bolt failure. Such a failure could occur when one of the lid bolts fails and its load is transferred to the two adjacent bolts, which could also fail due to overload, and thus the failure could rapidly progress around the bolt circle until the lid is blown open to the extent that the pressure is rapidly released (this failure mode is termed a "zipper effect"). If the pressure source was caused by uranium corrosion heatup alone or in combination with an external heat source, such as in a transportation fire, the liquid could be superheated, in which case the instantaneous pressure release would cause instantaneous steam flashing. For this scenario, surface phenomena would no longer appropriately describe the conditions inside the vessels. Rather, a shotgun-like blast of materials from the opening would likely more appropriately describe the phenomena, in which case the 1E-3 ARF used in the existing analysis would be substantially non-conservative. Factors that should be considered in analyzing the bolts would include the reduction in the allowable bolt load and the higher applied loads, both due to the higher temperatures that could be produced by a transportation fire. **Not resolved.** (See **OFI-11** and Appendix E.)

C23. **Wording error**

Section 4.3.4.4, *[Slurry Transfer Line Secondary Confinement System] System Evaluation*, in the section labeled *Conservative Design Features*, second paragraph, states, "The TLSB is rated at -15 in. w.g. [water gauge.] To *realize* [emphasis added] the -15 in. w.g. negative pressure, all components except for the TLSB would need to be isolated from the general service Ventilation System and the exhaust fan would need to speed up after a failure of the variable frequency drive to its maximum dead head capability against the TLSB." This statement needs further elaboration for the reader to reach the apparently intended conclusion – i.e., that the -15 in. w.g. condition could not be exceeded. Also, simply *realizing* the -15 in. w.g. pressure, as stated, rather than *exceeding* it, is not an unacceptable condition, since the TLSB is rated at this vacuum.

RCR response: "Accept – 'realize' will be changed to 'equal or exceed' to clarify the intent of the paragraph that equal or exceeding the rating is unacceptable and that multiple failures would be required for this to occur."

EA response to RCR response: Changing the wording from "realize" to "equal or exceed" does not fully resolve the concern, since the term "equal" is essentially synonymous with "realize," and equaling the rated pressure is not an unacceptable condition (only exceeding the rated pressure is unacceptable). The words "equal or" should be dropped to leave only the term "exceed" in order to eliminate this ambiguity. **Not resolved.** (See **OFI-12**.)

C24. **Nil-ductility transition temperature concern**

Section 4.3.4.4, in the section labeled *Environmental Design*, third paragraph, states, "The allowable stresses used in calculations 44577-M-CALC-030 and PRC-STP-CN-C-00434, *STP ECRTS – Transfer System, Transfer & Decant Boxes Assemblies, Frames, Lifting Lugs & Seismic Tie-Downs*, were computed accounting for these temperatures extremes." However, it is unclear whether these calculations accounted for the nil-ductility transition temperature effects at extremely low temperatures. This consideration should be described as clearly as it is for the stainless steel STSC in Section 4.3.10.4, which says the STSC is "not subject to brittle fracture at low temperatures."

RCR response: "The calculations did not account for the nil-ductility transition temperature effects at low extreme temperature. At the SAC-controlled temperature range of 40-100F, the piping system materials can be considered fully ductile. The basic allowable stress from ASME [American Society of Mechanical Engineers] B13.3 [*sic*]– 2008 Appendix A for ASTM A312 stainless steel pipe is 16.7 ksi [thousand pounds per square inch] from -425F to 300F. Per Paragraph 323.3 and Table 323.2.2, impact testing is not required, nor are strength reductions, for austenitic stainless steels down to -20F."

EA response to RCR response: The response does not address the intent of EA's concern that if the materials used were stainless steel, its "Environment Design" tolerance for low temperatures should be consistently described as it was in Section 4.3.10.4 for the STSC, "not subject to brittle fracture at low temperatures" because it is stainless steel. Although no technical concern remains, the PDSA wording is still inconsistent with the descriptions of the stainless steel materials used that make this piping tolerant of low temperatures. **Not resolved.** (See **OFI-13**.)

C25. **Wording error**

In Section 4.3.4.4, in the section labeled *Interface Design*, page 4-37, last sentence, the word "in" should be "is."

RCR response: "Accept – will revise per comment."

EA response to RCR response: Resolved.

C26. **Wording error**

Section 4.3.8.2, *System Description*, first paragraph, states, "The system uses nitrogen gas stored in high-pressure cylinders to provide *passive* [emphasis added] purge flow through the STSC." The word "passive" is unnecessary to convey the intended thought and implies an invalid characteristic for the system (i.e., that this purge flow is a passive function versus an active function). Since the system actively changes state to perform its safety function (the nitrogen purge flow through the STSC), this is an active function. Furthermore, since fluid flow is an active function, this is, in every sense, an active, not a "passive," function.

RCR response: "Accept – will delete the word 'passive.'"

EA response to RCR response: Resolved.

C27. **Wording error**

Section 4.3.8.2, *System Description*, third paragraph, states, "Each [nitrogen bottle] station contains 16 cylinders with a minimum void space 'water capacity' of 50 L (1.76 ft^3)." This statement is ambiguous in at least five ways:

- First, it is not clear what the term "water capacity" means. If it means that the maximum amount of water volume that would be allowed in the bottles to ensure sufficient available nitrogen volume, then this should be spelled out.

- Second, if this is the intended meaning, then the word “minimum” should be “maximum.”
- Third, the term “*water* [emphasis added] capacity” is ambiguous, seeming to indicate the capability to hold water, which appears to be irrelevant to the discussion.
- Fourth, the word “void” appears to be inappropriate, since “void” is generally understood to mean empty, and this space, if it contains water and/or nitrogen, is not empty.
- Fifth, if this is the limiting allowable amount of condensate, it is unclear whether the 50 L limit is per cylinder or per station.

RCR response: “ACCEPT: Sentence will be revised to read, ‘Each station contains 16 cylinders, each of which has a water capacity of 50 L [liter] (1.76 ft3).’ Water capacity is a term commonly used in the compressed gas cylinder industry and is typically provided in vendor catalogs, webpages, etc. It is defined as the volume of water which could be contained by a cylinder; and is provided to differentiate between the physical volume of cylinder and the compressed gas capacity.”

EA response to RCR response: Resolved.

C28. **STS cask pressure indicator range**

Table 4-12, *Sludge Transport System Cask Pressure Indicator Functional Requirements and Performance Criteria*, states, “The specified indicator must be shown to be capable of measuring the cask pressure over the range of 0 psi to 15 psi.” However, the instrument’s range should, at least, envelope the design pressure of the cask, i.e., 80 psi, plus some margin above this value, so that under conditions that would challenge the intended maximum pressure, operators can determine the degree of challenge in order to know when to effect venting and subsequently to monitor the progress of venting. Expanding this range would also provide consistency with Section 4.4.13.2, [SAC] *Description*, which states, “Operators will vent the STS cask prior to its reaching 80 psig with an appropriate margin of safety.” Additionally, DOE-STD-3009-94, page 7, addresses this concept of instrument range under the heading of Defense in Depth, where it states, “In the event that the inner layer of defense in depth is compromised from either equipment malfunction (from whatever cause) or operator error and there is a progression from the normal to an abnormal range of operation, the next layer of defense in depth is relied upon. It can consist of: (1) automatic systems; or (2) *means to alert the operator to take action or manually activate systems that correct the abnormal situation and halt the progression of events toward a serious accident* [emphasis added].”

RCR response: “Pressure indicator ECRT-PI-760-606 does not have a cask overpressure prevention function. During cask pressurization, over-pressure protection is provided by pressure safety valve ECRT-PSV-602 which is set at 50 psig (versus the 80 psig rating of the vessel). Overpressurization of the cask during pressurization using nitrogen from the Inert Gas System was judged in the HAZOP (PRC-STP-00687) to be a standard industrial hazard addressed by consensus codes and standards, i.e., B31.3, ‘Process Piping.’ The venting of the cask by operators quoted is based on elapsed time, using analysis to protect the design pressure with broad margin.”

EA response to RCR response: Using analyses to determine the actual cask pressure is an invalid pressure indication, since analyses only indicate the *expected* pressure for planned evolutions, not the actual pressure. Having no fully functional pressure indication is particularly unsuitable for unplanned conditions that could be encountered, including cask overpressurization. (If analyses alone were a valid indication for both planned and unplanned conditions, there would be no need for *any* indicator instrumentation in the first place.) The pressure indicator should be designed and intended to allow monitoring of cask pressure in order to ensure that the pressure is as intended and, if it exceeds the intended pressure, to allow monitoring of the extent of exceedance and the effectiveness of corrective actions. For this case, knowledge of the cask pressure is required in order to know whether the cask exceeded its intended pressure not only during pressurization from the Inert Gas System (as implied by the RCR response), but also at any other time when cask pressure integrity could be challenged. This SS function is attributed to the cask vent tool in a least 13 places in the PDSA, either in general terms, such as to "Prevent STS cask over-pressurization by venting pressure" from Table 30-84, *STSC and STS Cask Over-Pressurization Engineered Controls*, or in more explicit terms, such as "The cask vent tool is manually installed and operated, if required…, the STSC/STS shipment preparation SAC will require that the vent tool be used to vent the cask at a pressure less than 80 psig" from Section 4.3.11.5, *[Sludge Transport System Cask Pressure Boundary] Controls (Technical Safety Requirements).* If, as stated in the RCR response, pressure instrument ECRT-PI-760-606 is not the SS instrument to be used to monitor the SS function of preventing exceeding its 80 psig design pressure from any pressure source, then there is no other pressure instrument, SS or otherwise, shown on the system P&IDs or described in the PDSA that could satisfactorily perform this essential SS function (the instrument addressed in the RCR response has a range of only 0-30 psig, and the only other pressure instrument connected to the cask, ECRT-PI-605, has a range of only 0-15 psig). It should also be noted that if there were any challenge to the cask's pressure integrity, both of these instruments would be seriously over ranged, which would likely render them dangerously unreliable or failed as a pressure boundary. Additionally, any time the Inert Gas System is connected to the cask, these instruments and their attendant piping are extensions of the cask's SS pressure boundary and thus should also be classified as SS. Further, a cask failure caused by overpressure constitutes an accident with consequences well beyond what could be considered a "standard industrial hazard" because this type of failure involves the potential release of significant quantities of radioactive materials. **Not resolved.** (See **OFI-14**.)

C29. **STS cask pressure instrument support system classification**

Section 4.3.12.4, *[Sludge Transport System Cask Pressure Indicator] System Evaluation*, subsection entitled "Interface Design," states, "The STS cask pressure indicator interfaces with the STS cask and *is a component of the Inert Gas System* [emphasis added]." However, the subsection entitled "Support Systems" states, "the performance of the Inert Gas System for pressurizing the STS cask is not SS." Section 5.4.4.3, *Limiting Conditions for Operations: Pressure Indicator (PI-760-606)*, states, "Pressure indicator ECRT-PI-760-606, a component of the Inert Gas System, is used to verify the cask pressure; therefore; is SS." This last statement is inconsistent with the previous quote. Venting the cask to prevent its failure due to overpressurization is a safety function that would require the use of this instrument. A fundamental principle of nuclear safety design, as discussed in DOE Order 3009-94, Section 4.4, *SAFETY-SIGNIFICANT STRUCTURES, SYSTEMS, AND COMPONENTS* and elsewhere, is that instrumentation required for the performance of SS functions must also be classified as SS. Therefore, all portions of the Inert Gas System required for performing this SS function, including the pressure instrument, should be classified as SS and described as such in this section.

RCR response: "ACCEPT: Text will be revised to clarify which components of the Inert Gas System are safety significant and which are general service, and their associated safety functions. The statement, '…the performance of the Inert Gas System for pressuring the STS cask is not SS,' is correct. Note that, as stated in the response to comment #28, pressure indicator ECRT-PI-760-606 does not have a cask overpressure prevention function. During cask pressurization, overpressure protection is provided by pressure safety valve ECRT-PSV-602 which is set at 50 psig (versus the 80 psig rating of the vessel). Overpressurization of the cask during pressurization was judged to be a standard industrial hazard because (1) overpressurization would not a contributor to a significant uncontrolled release of hazardous material, and (2) vessel overpressurization hazards are routinely encountered in general industry and are covered by national consensus codes and standards, in this case B31.3 process piping."

EA response to RCR response: See EA's response to RCR response to Observation C28 above. Additionally, although Section 5.4.4.3 of the previous PDSA was eliminated in PDSA, Rev 00, the statement quoted above was simply moved to Section 5.4.5, *Oxygen Analyzer and Pressure Indicator*. The response states, "During cask pressurization, overpressure protection is provided by pressure safety valve ECRT-PSV-602 which is set at 50 psig (versus the 80 psig rating of the vessel)." However, per the P&IDs provided to EA from the project, the only PSV downstream of the pressure regulator in the Inert Gas System line supplying the cask is ECRT-PSV-604, which is set at 80 psig, not 50 psig as stated in the response (safety valve ECRT-PSV-602 could not be found on these P&IDs). Therefore, this response statement appears to be incorrect. Additionally, since there are several potentially-intervening manual valves between this safety valve and the cask, this safety valve cannot be Code-credited for protection of the cask from other pressure sources, such as hydrogen production from uranium corrosion. Further, as in Observation C28, a cask failure due to overpressure constitutes an accident with consequences well beyond what could be considered a "standard industrial hazard," since, as analyzed in the PDSA, it involves the potential release of significant quantities of radioactive materials. **Not resolved.** (See **OFI-15**.)

C30. Wording error

Section 4.3.14, *Sludge Quantity Instrumentation*, states: "The safety function for sludge quantity instrumentation is to protect initial conditions assumed in the STSC thermal and gas analyses regarding the quantity of sludge present in an STSC during interim storage." Section 4.3.14.1, *Safety Function*, states, "These general service uses of the sludge quantity instrumentation are not credited for any safety function." The first quote states that this instrumentation has a safety function, and the second states that it is not credited with a safety function – an obvious contradiction. Since protection of the analysis assumptions is vital to ensuring no explosion in an STSC "during interim storage" in the T-Plant, as well as before the STSC reaches interim storage (which is inappropriately not addressed in the PDSA), the instrumentation does perform a safety function. Therefore, it should be classified as SS, not general service, as stated in the second quote.

RCR response: "ACCEPT: Text will be revised to add clarification. The instrumentation is SS, not general service. The safety-significant measurements are an initial weight/level measurement, and a final weight/ level measurement. The text 'These general service uses of the sludge quantity instrumentation are not credited for any safety function,' refers to the previous two sentences in the PDSA which describes that the level and weight are continuously measured and transmitted to the control room and displayed as an operator aid. It was judged to be important to make this point in order to properly define the safety significant system boundaries. As discussed in the 'Interface Design' subsection on page 4-82, signal isolators separate the SS and general service data signals.

"Wording change was made to differentiate between the non-safety control room instrumentation readouts and the SS readouts at instrument panel ECRT-PNL-401."

EA response to RCR response: Resolved.

C31. **Wording error**

Section 4.3.14.2, *[Sludge Quantity Instrumentation] System Description*, second paragraph, states, "All four load cells are inputs to a summation box (smart section controller, WX-740-401) that *equalizes* [emphasis added] all four load cell signals…and outputs a single truck weight signal." The term "equalizes" does not appear to correctly describe what the summation box actually does; the correct word is "sums" or "averages," depending on what the individual load cell signals actually represent.

RCR response: "ACCEPT: Text will be revised to say that the summation box 'sums' all four load cell signals." The word "equalizes" was changed to "sums."

EA response to RCR response: Resolved.

Appendix D
Observations on Sludge Treatment Project
Engineered Container Retrieval and Transfer System
Preliminary Documented Safety Analysis, Revision 00

In addition to the open concerns with the Preliminary Documented Safety Analysis (PDSA), Revision (Rev) 0, the independent Office of Enterprise Assessments (EA) identified the following new concerns with Rev 00 of the PDSA.

D1. **Spelling error**

Section 2.5.5, *Sludge Retrieval*, second paragraph, next to last sentence, the word "emonstrated" should be "demonstrated."

D2. **Page numbering error**

The page numbering of this document is inconsistent with the tables of contents; all page numbers are preceded with "A-", but the tables of contents do not contain these characters.

D3. **Control valve setpoint error**

In Section 2.8.4, *Auxiliary Ventilation System*, the first paragraph states that the Auxiliary Ventilation System's pressure control valve is "set at 40 to 65 psig (pounds per square inch – gauge]" and that "the system's gas supply is not consumed unless the inert gas supply pressure falls below 150 psi." These setpoint statements are inconsistent with Section 2.8.3, *Inert Gas System*, which states that the Inert Gas System's pressure control valve is set at 65 psi.

D4. **Wording error**

In Section 4.3.4.4, in the "Environmental Design" subsection, the phrase "these *temperatures* [emphasis added] extremes," should read "these temperature extremes," with no "s" on "temperatures."

D5. **STS Cask pressure instrument**

Section 3.2.4.4, *Protection of Assumptions*, quotes a statement from DOE-STD-1189-2008, "As discussed in DOE-STD-1189-2008, Appendix D, SSCs [structures, systems, and components] that function to monitor initial conditions assumed in accident analyses are not required to be safety classified based on the monitoring function if…They do not generate a signal (indication, alarm, or interlock function) that causes an action (operator action or change of state) that is required to prevent or mitigate an accident." In other words, this standard requires that if SSCs function to monitor initial conditions assumed in accident analyses and generate an indication signal that causes an operator action that is required to prevent an accident, then those SSCs must be safety classified. The PDSA states in numerous places that the safety significant (SS) cask vent is required to perform the SS manual (operator) function of being opened to relieve pressure, thereby preventing overpressure failure of the cask. Therefore, the standard requires an SS monitoring instrument for this function. The monitoring instrument (pressure gauge) specified for the present design is not only not classified as SS, but also is not suitable for this application because (1) it has inadequate range, and (2) it is disconnected from the cask after it is sealed for transportation.. Therefore, the PDSA-quoted requirement of the standard is not met. This observation reinforces EA's Observations C28 and C29 in Appendix C of this report and is contrary to the site's corresponding Review Comment Record (RCR) responses. (See **OFI-16**.)

D6. Isolation valves single failure criteria and leak testing

Section 4.3.8.2, *Auxiliary Ventilation System, System Description*, states, "Backflow from the SS auxiliary ventilation supply into the general service portions of the Inert Gas System is prevented by SS check valves for each auxiliary ventilation train (ECRT-CV-612 and ECRT-CV-622)." However, no design features are provided to allow leakage testing of these valves, no criteria are provided for allowable leakage, and there are no indications that the design allowable leakage rate is accounted for in the available system capacity to provide the design auxiliary ventilation function for 96 hours. Such testing and provisions would be necessary to ensure that the system can meet the single failure criteria, as indicated by PDSA Table 4-8, *Auxiliary Ventilation System Functional Requirements and Performance Criteria*, which requires that pre-existing undetectable failures must be assumed. Such undetectable failures could exist without such surveillance testing. (See **OFI-17**.)

D7. Vacuum breaker leak testing

Section 4.3.8.4, *[Auxiliary Ventilation System] System Evaluation*, states, "Components of the room air inlet line upstream of the tee are designated SS up to and including the fail-closed vacuum breakers ECRT-PCV-632 and ECRT-PCV-633. In the event of loss of sludge transport and storage container negative pressure (due to loss of electrical power or other problems with the normal Ventilation System) the spring loaded vacuum breakers ECRT-PCV-632 and ECRT-PCV-633 will close. By design, whether open or closed these vacuum breakers will not permit backflow from the Auxiliary Ventilation System to the room air inlet." However, as with the check valves described in Observation D6 above, no criteria are provided for allowable leakage, and there are no indications that the design allowable leakage rate is accounted for in the available system capacity to provide the design auxiliary ventilation function for 96 hours. Additionally, SS check valve ECRT-CV-605, which forms part of the same pressure boundary and could also be subject to back leakage, is not addressed in the PDSA; it has no design features for leakage testing and no specified allowable leakage; and there are no indications that a design allowable leakage is considered in the system's 96 hour design capacity requirement. Such testing and provisions would be necessary to ensure that the system can meet the single failure criteria, as indicated by PDSA Table 4-8, *Auxiliary Ventilation System Functional Requirements and Performance Criteria*, which requires that pre-existing undetectable failures must be assumed. Such undetectable failures could exist without such surveillance testing. (See **OFI-17**.)

D8. Isolation valve single failure criteria not met

PDSA Table 4-8, *Auxiliary Ventilation System Functional Requirements and Performance Criteria*, requires that "SS components (e.g., *check valves* [emphasis added], signal isolators) shall protect interfaces with non-safety systems" and that such "Active components shall meet the single failure criterion." Contrary to these requirements, the interface of the SS Auxiliary Ventilation System with the non-safety Inert Gas System at check valve ECRT-CV-605 does not meet the single failure criterion, since only one valve is at this interface. This valve's failure to close on demand could cause the loss of both divisions of the Auxiliary Ventilation Systems. (See **OFI-17** and **OFI-18**.)

Appendix E
Office of Enterprise Assessments Comments on Historical Thermal and Structural Analyses Related to the Sludge Transport System Cask Transportation Fire Hypothetical Accident Conditions

The independent Office of Enterprise Assessments (EA) reviewed multiple documents, one of which was provided by the U.S. Department of Energy Richland Operations Office, related to the Sludge Transport System (STS) cask transportation fire hypothetical accident conditions. These reviews were aimed at gaining a better understanding of the cask's thermal and structural analyses for this event, in order to verify the adequacy of its design, as described in the Preliminary Documented Safety Analysis (PDSA), Revision (Rev) 00. EA reviewed two historical documents that appeared to be the latest analyses on the thermal and structural analysis and provided the following comments.

Calculation SNF-18162, *Thermal Analysis of Sludge Transport System for Argon Backfill and Extended Transport Window,* Revision 1, dated October 2, 2003

- **Uranium particle shape assumption.** Section 3.1, *Sludge Thermal Properties*, states, "The relationship between the mass of the metallic uranium and the reaction surface areas is provided by the assumption that the uranum metal exists in the form of uniform spherical particles with a diameter of 500 microns." Section 3.4, *Thermal Heat Load*, states, "The heat generation resulting from thermal reations within the sludge container is a function of the temperature and the *reacting surface area* [emphasis added]." The first quote's assumption of "spherical" shaped uranium particles is a geometry that would represent the absolute *minimum* "reacting surface area" per unit mass; therefore, this assumption is non-conservative with respect to heat and hydrogen generation rates.

- **Cask maximum temperature.** Section 8.2, *Hypothetical Accident Conditions Of Transportation*, page 81, second paragraph, states, "Although a peak cask temperature of 1,133 degrees Farenheit (ºF) is seen at the end of the fire, the Figure 8-34 temperature distribution clearly illustrates that this temperature level is only attained at the corners of the cask lid flange and the cask base where the exposed surface area per unit mass is the greatest." This statement implies that these locations on the cask shell are not subject to concern, because they do not experience high stress levels. However, the cask lid flange location is EA's primary area of concern with respect to cask integrity, because it relates to the loading of the lid hold-down bolts, which are in that area and are subject to high stress levels; these are addressed in the comments on Packaging Technology, Inc. (PacTec) Calculation 12099, *STS Cask Structural Analysis*, later in this appendix.

- **Homogeneous uranium mixture in the sludge.** Section 7.3, *STS Transportation Cask And Sludge Payload Modeling of HAC*, fourth paragraph, item 5, states an underlying assumption: "a drop event with sufficient energy to upset the cask and cause the LDC [Large Diameter Container] to crack will also act to disturb and mix the sludge. As such, it is assumed that any layering existing within the sludge prior to the fire event will be lost and the sludge can be treated as a homogeneous mixture." Although sludge layers would likely mix, the mixing would not likely be "homogeneous." For non-homogeneous mixing, the higher concentration areas would experience more heating, and therefore higher temperatures, accelerating the uranium/water reaction process; uranium heating produces a positive feedback mechanism with respect to temperature, causing the heat rate to increase exponentially. Consequently, this assumption is non-conservative.

- **Cask pressure bleed-down assumption.** Section 8.1.3, *Extended NCT [Normal Conditions of Transport] Transportation With Argon Backfill*, fourth paragraph, the next to last sentence states, "As demonstrated by the change in the rate of hydrogen gas generation illustrated in Figure 8-13 and the associated gas pressure transient in Figure 8-14, heat up of the sludge and cask above the initial conditions assumed by this analysis will shorten the available operational time before the cask must again be bled down to approximately 94 hours." It is unclear from the current draft PDSA how transportation personnel are expected to monitor the cask pressure and safely effect such a bleed down from 80 psig (pounds per square inch – gauge) to 15 psig, since no installed pressure instruments or venting or pressure relief devices are installed on the cask during this period of concern. (See Appendix C, Observations C28 and C29.) Since the cask vent tool is replaced with a plug in preparation for transportation, it is unclear from the PDSA how, when the cask has become pressurized during transportation, the plug can be removed and replaced with the vent tool without an uncontrolled release of the pressure and some of the radioactive/explosive contents from the cask.

- **Invalid cask material properties statements.** Section 6., *Acceptance Criteria*, ninth paragraph, states, "As shown in the ASME [American Society of Mechanical Engineers] Code, the strength properties of steels do not change due to short-term exposure up to 1,000 ºF and the original material structural properties will be recovered when the temperature decreases. Therefore, short-term exposure to the temperatures of this magnitude does not have any significant effect on mechanical properties of the materials." These statements are ambiguous and non-conservative, as follows. The first statement of the first sentence (i.e., "strength properties of steels do not change due to short-term exposure up to 1,000 ºF") is ambiguous with respect to the second statement (i.e., "original material properties will be recovered when the temperature decreases"). In other words, it is not consistent for the strength properties to remain unchanged when the material is heated and for the strength properties to be "recovered when the temperature decreases." The first statement is also incorrect; the material properties of steels are, in fact, significantly degraded from 70 ºF to 1,000 ºF; for example, the ultimate strength of 304 stainless steel (i.e., the cask shell material) decreases from 84,000 pounds per square inch (psi) to 70,000 psi (approximately 17%). (The premise of these statements is incorrect but does not appear to have been applied in this calculation.)

- **Uranium loading per container.** Table 3-1, *Homogeneous Sludge Parameters w/o Gas Retention*, indicates that the concentration of metallic uranium in the sludge per container used in the calculation (the primary source of hydrogen gas and chemical reactive heat) was 63.8 kilograms per cubic meters. For the calculation's assumed sludge loading of 2 cubic meters per container, the total metallic uranium would be 127.6 kg. Although the corresponding PDSA, Rev 00, concentration from Table 3-25 is higher, at 163 kilograms per cubic meter, the total metallic uranium per container is lower than what was analyzed in this calculation (81.5 kilograms per container), since the PDSA, Rev 00, maximum allowed container sludge loading is only 0.5 cubic meters. Therefore, the calculation's assumed metallic uranium loading per container is approximately 57% higher than the PDSA limits and thus is conservative.

PacTec Calculation 12099, *STS Cask Structural Analysis,* Revision 2, dated November 1, 2002

- **Non-conservative transportation fire accident temperature.** Page 30, Table 10-1, *Material Property Summary for Evaluation of Bolts*, provides the key material properties (i.e., yield strength, ultimate strength, design stress intensity, elastic modulus, and coefficient of thermal expansion) for the Type 304 stainless steel material for the cask, and the ASME SA564, Grade 630 (H1100) material for the 24 head hold-down bolts (1½" nominal diameter - 6UNC-2A). These properties are provided for the temperature range of -40 ºF to 800 ºF. The 800 ºF temperature is reflected in the analysis as the maximum transportation fire event evaluation temperature for the bolts and for other components (e.g., in Section 10.3, *Main Seal Evaluation*, and Table 10.3-1, *Helicoflex Seal Evaluation – Input*). However, the 800 °F temperature is substantially less than the 1,133 ºF transportation fire peak cask temperature determined in calculation SNF-18162, Rev 0, October 2, 2003, *Thermal Analysis of*

Sludge Transport System For Argon Backfill And Extended Transport Window, for the edge of the cask lid flange, which is the approximate location of the bolts. This higher temperature at this location substantially degrades the material properties of the bolts, and also substantially increases the bolts' loading because of increased differential thermal expansion of the stainless steel head versus the carbon steel bolts (the stainless steel head's thermal expansion coefficient is approximately twice that of the carbon steel bolts), and also because of the additional differential thermal expansion due to the flange being slightly hotter than the bolts.

- **Non-conservative peak pressure.** Section 10.3, *Main Seal Evaluation*, second sentence, states, "For a pressure of 123 psia [pounds per square inch – absolute] (108.3 psig), and a temperature of 661 ºF [even less than the 800 °F value discussed above], the applied load [of the head bolts] is greater than the 'load to be applied.'" This statement indicates that the bolts' preload force is greater than the cask head liftoff force resulting from the internal pressure. However, Calculation SNF-18162, Rev 0, October 2, 2003, *Thermal Analysis of Sludge Transport System For Argon Backfill And Extended Transport Window*, Section 8.2, *Hypothetical Accident Conditions of Transportation*, page 81, states that the peak pressure of the transportation fire is 153.4 psia (138.7 psig), which is 25% higher than the above-described pressure used in the bolt analysis. This higher pressure would require the bolts' preload (and their stress) to be greater than what is determined by this calculation.

- **Non-conservative stress intensity ratio.** In Table 2.9-10 (Table 10-10 of Calculation 12099-01), *Summary of Stress Intensity Ratios*, for Case 7, "NCT (Normal Conditions of Transport) Hot Impact (End)," the calculated stress intensity ratio (ratio of applied-to-allowable stress intensity) is 0.9992 (acceptable is < 1.0). Given the variations of material properties, the less than absolute precision of the calculation formulae, and the multiple variables of this calculation, the minuscule margin of this result would appear to be non-conservative, even without considering that this result is based on the non-conservative temperatures and pressures described in the previous comments.