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Disposal Options for Depleted Uranium Trioxide (DUO_3) Study

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EXECUTIVE SUMMARY

There exists at SRS 50 million pounds of depleted UO_3 (DUO) stored in 55-gallon drums stacked three high in several buildings. This storage configuration does not allow access to the individual drums for monitoring drum integrity and material accountability. A number of drums are known to be severely corroded, due to previous standing water in the buildings. SRTC has been asked by High Level Waste to prepare a scoping study of two disposal options: the feasibility of using the DUO as a constituent in grout for filling HLW storage tanks for closure, and disposing directly in E-Area.

Depleted UO_3 may be used as high-density inert filler in the production of grout for HLW Tank Closure. The DUO could serve as a stand-in for quartz sand, which is inert from the standpoint of environmental transport and contamination. As such, the DUO stored on site would be put to advantageous use, saving the cost of continued monitoring the stored DUO materials and the cost of the quartz sand grout constituent in filling the HLW storage tanks for closure.

Analysis of the radionuclide content of the depleted UO_3 versus the radionuclide acceptance limits for E Area shows that the UO_3 cannot now be disposed in E-Area. Laboratory studies of the leachability of radionuclides from the depleted uranium must be performed and the results incorporated into a Special Analysis if disposal in E-Area is to be pursued further. It is likely that the UO_3 will have to be stabilized in a material that retards technetium release in order for the waste to be acceptable for disposal in E-Area.

This report also discusses properties of the UO_3, precautions for safe handling, sampling needed for better characterization, physical integrity of the stored drums, and drum handling with a grout mixing process.

A recent study by MHF Logistics Solutions, Inc.[1] considered the logistical requirements to ship ~3,200 drums (~2,172 metric tons) of SRS DUO stored in buildings 728-F and 730-F to Envirocare of Utah and DOE's Nevada Test Site (NTS). The costs of disposal were determined based on the following options:

- Packaging the drums in groups of six drums into Lift Liners™ equipped with palet reinforced bottoms, placing the loaded Lift Liners™ into gondola rail cars, and shipping the material from Aiken, SC to Envirocare of Utah, Clive, UT. The cost per drum to ship to Envirocare in this manner is $355.40, and the total for the project is $1,134,790.79.

- Sorting the drums into groups of 34 drums, placing the drums into intermodal containers, transferring the intermodal container to ABC rail cars, and shipping the material from Aiken, SC to Envirocare of Utah, Clive, UT. The cost per drum to ship to Envirocare in this manner is $206.66, and the total for the project is $659,877.90.

- Sorting the drums into groups of six, packaging the groupings into TOC bins, placing the loaded TOC bins onto flat bed trailers, and shipping the material via highway

from Aiken, SC to Nevada Test Site, Mercury, NV. The cost per drum to ship to NTS in this manner is $592.46, and the total for the project is $1,891,715.29

- Sorting the drums into groups of 25 drums, placing the drums into Sealand containers, transferring the Sealand containers to ABC rail cars, shipping the material from Aiken, SC to Cisco, UT, transfer the Sealand containers to flat bed trucks, and highway transport to Nevada Test Site, Mercury, NV. The cost per drum to ship to NTS in this manner is $437.34, and the total for the project is $1,395,451.00.

This provides a baseline cost for disposing of the DUO with current available disposal options.

1. BACKGROUND

Uranium Trioxide UO_3 is a common intermediate compound encountered during the processing of uranium for beneficial uses. In the processing of uranium ore and the separation of uranium from other metallic compounds, UO_3 is formed. In the production of plutonium for both nuclear fuels and nuclear weapons, uranium has been utilized in pile reactions. Plutonium was first produced in 1940 by bombarding uranium with deuterons to form neptumium-238, which decays to plutonium-238. The separation of uranium from plutonium is necessary to recover the plutonium for nuclear applications. A variety of procedures have been developed for the recovery and decontamination of plutonium from uranium and fission products.[2,3,4,5]

The Savannah River Site has utilized the PUREX process for a number of years. This process is defined as a tributyl phosphate plutonium uranium reduction extraction process.[6] SRS converted from natural uranium to depleted uranium target assemblies in the plutonium production reactors in 1970. The separation of these target elements following reactor exposure utilized the PUREX process in 221F Canyon.[7] The resultant depleted uranyl nitrate solution was concentrated and de-nitrified in F-Area A-Line facility (Building 221-1F). The UO_3 products from the de-nitrification process were packed into plastic bags and placed in 55-gallon steel drums with a plastic liner for storage. Batch de-nitrators were utilized to form the UO_3 products from 1970 through 1989, producing a granular depleted uranium byproduct. After 1989 electrically heated continuous de-nitrators were utilized, which produced an finer powder depleted uranium byproduct. Any future plutonium needs will be met by recycling weapons-grade material, thereby eliminating the probability of future UO_3 generation.

Seven buildings across SRS were being used in 1995 to store 35,835 drums of DUO representing an approximate weight of 19,436,276 Kg.[8] The drums were stacked three high in all of the buildings. Of the seven buildings, only one had forced ventilation. Except for Building 105-R, the storage buildings can be generally described as structural steel framed metal buildings with concrete floors. The storage locations are inspected quarterly and an inspection report is prepared. Due to current stacking configuration, complete inspection of the drum integrity is not possible.

2. CHEMICAL AND PHYSICAL PROPERTIES OF URANIUM TRIOXIDE

The physical properties of the UO_3 will dictate the design of equipment used to move the material from existing drums to processing facilities. In characterizing the DUO at the Savannah River Site, chemical and physical information was obtained both from historical records and analysis of the material. [9] The data includes chemical composition and impurities, particle size, angular properties, radiological properties and leachability properties.

The drummed material can be described as a 200 mesh, 96.5% uranium trioxide, with trace impurities of aluminum, iron, phosphorous, sodium, silicon, chromium and nickel. The material has a bulk density of 200 pounds per cubic foot and an angle of repose of 34-40 degrees. The uranium-235 content is approximately 0.2 percent and the most recent analyses indicate the plutonium content is 20 parts per trillion. Gross gamma is 53,100 dpm per gram of uranium.

2.1 Chemical Analysis

A-Line finish product analytical reports for UO_3 drums numbered C9092 through C9460 provides the following information:

Chemical Component	Average	Range
Total, U, %	80.34	82.25-78.47
Aluminum, Al, ppm	68	>100 – 20
Calcium, Ca, ppm	6	10 - <5
Chromium, Cr, ppm	41	100 – 20
Copper, Cu, ppm	4	15 - <2
Iron, Fe, ppm	100	323 – 29
Molybdenum, Mo, ppm	10	<10
Sodium, Na, ppm	48	300 - <10
Nickel, Ni, ppm	21	50 - <10
Phosphorus, P, ppm	90	100 - <25
Lead, Pb, ppm	5	5 - <5
Silicon, Si, ppm	41	150 - <10
Plutonium, Pu, ppb	2.85	9.52 - .333

The evidence of trace chromium levels in the UO_3 leads to speculation that the material would be classified as mixed waste when the material is determined to have served its useful life. Samples of the UO_3 were analyzed for toxic metals in 1989 and 1993.[9] The 1989 tests utilized the EP Toxicity method of leaching and then later used the TCLP method. The average and maximum values for each of the metals are shown in Table 2.1.1.

Table 2.1.1 Leachability of Toxic Metals from Drummed UO_3 in PPM

Element	EPA Limit	See Note 1 1989 Average	1989 Maximum	See Note 2 1993 Average	1993 Maximum
Chromium	5	0.190	1.120	0.118	0.275
Arsenic	5	0.1600	0.7800	0.0001	0.0010
Selenium	1	0.030	0.030	0.033	0.075
Silver	5	0.1	0.1	NA	NA
Cadnium	1	0.18	0.53	NA	NA
Barium	100	0.780	2.560	0.205	0.360
Mercury	0.2	0.02	0.02	NA	NA
Lead	5	0.300	0.300	0.001	0.001

Note 1. Used EP-Toxicity Methodology – Average assumed minimum detectable level for analysis reported as less than. Eleven samples from the batch, MK-15 and continuous campaigns.

Note 2. Used TCLP Methodology – Average assumed minimum detectable level for analysis reported as less than. Four drum samples from each of the campaigns: Batch, continuous and MK-15.

2.2 Physical Properties

The physical properties of the UO_3 are of interest as one considers the handling of the material for repackaging or use as a grout constituent. The properties of primary interest are bulk density, particle size distribution, and angular properties.

The bulk density of the UO_3 may be measured by two different methods. A tamped density may be obtained by vibrating the UO_3 in a graduated cylinder to simulate the compaction of the material over time, or with the aid of vibrating compaction. The density was measured to be 3.5 gm/cc or 223 lb/ft^3 by Carolina Metals, Incorporated (CMI) as part of a study in 1995.[9] CMI also measured the bulk density by pouring material into the measuring cylinder without compaction assistance. This measurement method is useful in unvibrated drumming operations and storage hoppers handling. Bulk density samples taken from the top and bottom of a drum by CMI were 2.53 gm/cc (158 lb/ft^3) and 2.66 gm/cc (166 lb/ft^3), respectively.

The particle size distribution was determined by CMI on a sample taken from the top of a drum. Screens from 70 mesh to 325 mesh were used, with openings from 0.210 m to 0.044 mm. The distribution shown in Table 2.2.1 indicated an average size of 200 mesh or 0.074 mm. A wide distribution will impact the angular properties of the material. For comparison, the particle size of the material when it was produced is also given.[9]

Table 2.2.1 UO3 Particle Size Distribution

Analysis of UO3 Sample from Top of Drum C1068 Analysis performed at Carolina Metals, Inc. in 1994.	
Mesh Size	Percentage of Particles
• 70	78.4
• 100	66.2
• 140	56.1
• 200	48.4
• 270	32.9
• 325	27.8

Analysis of 38 Samples Taken From Drums C9092 through C9460 Analysis performed at Time of Production by Savannah River.	
Mesh Size	Percentage of Particles
• 40	97.6
• 80	81.6
• 325	12.8

The angular properties are considered essential in the design of hoppers and conveying equipment. For the UO_3 material analyzed by CMI, the angle of repose ranged between 34 and 40 degrees and the angle of internal friction ranged between 60 and 75 degrees. The angle of slide, measured on steel and on plastic, ranged from 30 to 35 degrees on steel and from 25 to 30 degrees on plastic.

3. PRECAUTIONS FOR SAFE HANDLING AND USE OF DEPLETED URANIUM TRIOXIDE

Depleted Uranium trioxide is practically insoluble in water, but soluble in acid or acidified water. All direct handling of the material should be conducted in a properly ventilated area according to industrial hygiene recommendations. Use proper protective equipment: disposable protective apparel, neoprene gloves, and full-face respiratory protection. Appropriate controls for dust must be utilized to maintain exposure below the Time Weighted Average of 0.2 mg/m^3. The material must be stored in a cool, dry place in tightly sealed containers. Good ventilation must be provided in the vicinity where material containers are opened, and careful handling of the containers must be practiced to prevent exposure.

3.1 Health Hazard Data

Depleted Uranium trioxide is highly toxic and slightly radioactive, targeting the lungs and kidneys of exposed individuals. The primary routes of entry are through direct skin and eye contact, ingestion and inhalation. Contact with the skin may cause irritation and

dermatitis. Contact with the eye results in severe eye damage, as well as systemic poisoning.

Insoluble uranium compounds are poorly absorbed by the gastrointestinal tract, if ingested. Uranium intoxication will cause kidney damage, which may become evident in as little as 24 hours following an acute exposure. Inhaled insoluble uranium compounds may remain in the lungs and tracheobronchial lymph nodes for many years, and may cause radiation damage.

Chronic effects of uranium are greater for soluble compounds. Soluble uranium compounds affect the blood vessels throughout the body. Capillary permeability and edema may increase, and clotting ability may decrease, because of the effects of uranium on blood chemistry. Uranium is known to induce minor damage in liver and muscle tissue, but is extremely toxic to kidneys. Damage to renal tubules resulting in nephritis will follow intake of any uranium compound.

Depleted uranium trioxide is a slightly radioactive alpha-emitter and may cause radiation damage. Prolonged exposure to uranium has caused fibrotic changes in animal lungs. Absorbed uranium deposited in bone creates a potential risk of radiation effects on bone marrow, but no hematologic effects were disclosed in clinical studies.

3.2 Exposure Controls

All direct handling of depleted uranium trioxide should be conducted in a properly ventilated area. This may include a fume hood of other appropriate controls to contain, and minimize exposure to, airborne dust. Exposure limits established by ACGIH, NIOSH and OSHA for uranium compounds are 0.2 mg/m^3.

4. STRATIFIED RANDOM SAMPLING FOR CHARACTERIZING DEPLETED URANIUM TRIOXIDE

4.1 Objectives & Recommendations

A statistical sampling approach is suggested for characterizing the approximately 35,000 depleted uranium trioxide drums located within seven storage buildings at SRS. Laboratory analyses will determine the concentrations for the metals of interest (arsenic, barium, cadmium, chromium, lead, mercury, selenium and silver) in the samples. The population of drums is split into two different types: approximately 3,300 post-1985 and 31,700 pre-1985 drums. We recommend that 20 drums be sampled for the post-1985 group and that 60 drums be sampled for the pre-1985 group. Thirty of these drums, ten from the post-1985 group and 20 from the pre-1985 group should be double sampled by obtaining one sample from the top and the other from the bottom of the drum. The analysis results will be statistically analyzed to determine the percentage of drums above the RCRA hazardous waste limits and to decide if additional sampling is necessary.

4.2 Details for Sampling Depleted Uranium Trioxide

We recommend that the pre-1985 and post-1985 depleted uranium trioxide drums be sampled by stratified random sampling. Stratified random sampling involves dividing the population into homogeneous subgroups and then taking a simple random sample in each subgroup. The population is already segregated into seven buildings at SRS and the number of drums within each building is already known. Ideally, a proportional random sample should be chosen in each building based on the number of drums in each building. To statistically determine the number of drums that should be sampled for each building, an estimate for the variances of the metals of interest needs to be available. However, prior data are not available for development of these estimates. It is assumed that the analytical procedures are highly reliable and that the variance contributed to the measurements is negligible with respect to the amount of metals present in the samples.

There are several major reasons why stratified sampling is preferred over simple random sampling. First, it assures us that we will be able to represent not only the overall population of drums, but also key subgroups of the population, especially buildings with smaller number of drums. Stratified sampling also assures us that we can effectively characterize these buildings. We may sample proportionally more from buildings with a smaller number of drums and later account for this when we desire the overall population estimate. Stratified random sampling generally has greater precision than simple random sampling when differences exist among the buildings.

The population of approximately 35,000 drums stored in seven buildings will be sampled in stages with the first stages of the sampling effort involving the post-1985 drums that are primarily stored in Building 728-F and 730-F. The drums have a volume of 55 gallons and are typically 66% full. The total number of drums in these two buildings is estimated at 3263 of which approximately 2096 are located in Building 728-F and approximately 1167 drums are located in Building 730-F. Ideally, a simple random sample of drums should be chosen within each building. However, this not possible because of the orientation and vertical stacking of the individual drums. Pragmatically, the drums to be sampled will be randomly selected over the visible and reachable area within the building.

We recommend an approximate 0.57% sample from Building 728-F resulting in $n_{728\text{-}F}=12$ and an approximate 0.73% sample from Building 730-F resulting in $n_{730\text{-}F}=8$ sampled drums. Double samples should be randomly chosen from half of the selected drums in each building with the top and bottom portion of the drums sampled. This effort is preliminary and additional sampled drums may be necessary from these two buildings after the data are statistically analyzed. The samples should be labeled so that a particular drum can be resampled if necessary and the ordering of the samples should be randomized before being sent to the laboratory for analysis. The sample labels should not reveal the building, drum, and replicate status or location within a drum.

The statistical results will be examined and a decision will be made on any additional sampling from these two buildings to attain higher probability that the depleted uranium

oxide is not above the RCRA hazardous waste limits. Additional samples may be necessary if a significant percentage of data is below the detection limit for any of the measured metals. The statistical results and experiences with the data from sampling the drums in these two buildings should lead to greater sampling design efficiency for the remaining buildings in terms of the number of drums sampled and the number of drums with duplicate (top & bottom) samples.

The pre-1985 drums (essentially 1970's production) are located in five storage buildings across the site. We recommend that 12 drums be sampled from each building resulting in 60 sampled drums. Duplicate samples (top & bottom) should be randomly chosen from 20 of these drums. Proportional sampling would be implemented if vastly different number of drums exist over the five buildings.

4.3 Statistical Considerations

Statistical analysis of the analytical data for each population type will determine if there are differences between the drums in the storage buildings, differences between drums within a building and finally differences between the samples within drums. If no statistical differences exist, the data will be combined into a single population for distribution identification and estimation of tolerance intervals.

Suppose the results for each metal from the 20 post-1985 drums do not reveal any variation between Building 728-F and 730-F and that the variation within the drums is negligible. In this case, the analytical results will provide an estimate of the average and standard deviation. One-sided upper normal tolerance limits will be estimated as: average+k (standard deviation) where k is a tabulated value based on the number of sampled drums if the analytical results can be described by a normal distribution. In this case, the value k=2.396 will be used for the 20 drums to determine an upper 95% confidence limit so that 95% of the drums (coverage) will be below this estimated upper limit for any metal of interest. The estimated tolerance limit will be compared to the maximum allowable limit. If the estimated limit is greater than the maximum allowable limit, the corresponding coverage that will be less than 95% will be determined. Additional samples and analyses would have to be done if the data contain less than detection results or are not normal. For example, if we cannot assume a normal distribution, 60 drums would be necessary to have 95% confidence that 95% of the drums are less than an estimated non-parametric upper tolerance limit.

Differences between the pre-1985 drums in the five storage buildings as well as within drum differences will be evaluated. If no significant differences are found, the pre-1985 data will be considered as a single population for statistical analysis and determination of tolerance limits. Additional sampling would be necessary if differences between buildings or within drums are found.

5. DEPLETED URANIUM TRIOXIDE STORAGE DRUMS PHYSICAL INTEGRITY

5.1 Storage Configuration

The depleted uranium trioxide (DUO) powders have been stored on site in 55-gallon carbon steel drums since the 1970's. The drums are currently stacked in a three-tier matrix with each tier separated by wooden boards. The width and length of the matrices vary by building size. Exposure to rainwater in-leakage and possible run-off water has resulted in corrosion of the drums. The Materials Technology Section (MTS) of the Savannah River Technology Center reviewed the storage and physical conditions of the drums in 2001 by employing visual examination, ultrasonic thickness measurements, and calculations of minimum wall thickness for different modes of drum failure.[11] For drum integrity there must be no wall penetration that could allow DUO powders to escape the drum. If this situation occurs, the drum must be immediately placed into an over-pack.

Most of the drums inspected had typical atmospheric corrosion, which does not jeopardize the structural stability. The drums that had been exposed to standing water, i.e. the lower tier drums, showed accelerated corrosion of the drum wall near the bottom rim, although perforation of the drum was not evident. Leakage of DUO has not been detected, though small amounts of DUO could leak from small pinholes without detection. The potential for drum buckling may exist for severely corroded drums if the corrosion extends around the entire circumference of the drum.

5.2 Structural Condition Evaluation

An evaluation of twenty drums was conducted in 1994 to determine the condition of powders and the extent of degradation of the drums and the inner plastic bags.[9] Thickness measurements, including the most severely corroded areas, indicated that corrosion loss was less than 15% of the minimum gauge thickness for the sheet steel used in the drum construction. The inner plastic bags were damaged, and the securing masking tape showed evidence of embrittlement.

Prior to handling or moving the drums, each drum should undergo both visual and physical examination by trained personnel. These examinations should qualitatively assess the degree of degradation and the structural condition of each drum. If the drum is determined to be in good condition, it could be moved from the stack using standard drum handling equipment. Any drum considered suspect should have ultrasonic thickness measurements made of its wall. If acceptable, the drum could be moved using standard drum handling equipment. If unacceptable, the drum should be placed in an over-pack.

5.3 Determination of Minimum Wall Thickness For Handling

The drums are fabricated from carbon steel sheet of different gauges and subsequently painted. Minimum thickness values for the gauges are 0.053 in. for 16-gauge, 0.043 in.

for 18-gauge, and 0.032 in. for 20-gauge. The grade of steel is probably similar to AISI 1010 low carbon steel. The drums have the standard three rolled hoop construction with side and bottom welds.

Minimum wall thickness values may be calculated for three scenarios to determine structural soundness[12]. These scenarios are for (1) safe lifting of a drum, (2) buckling of a bottom row drum with two drums stacked upon it, and (3) buckling of a bottom row drum when 25% of the wall is corroded through, i.e. missing. A 25% loss is chosen as a reasonable estimate of perimeter loss for an aggressive pitting condition as exists for carbon steel in standing water. The results are shown in Table 5.3.1. Input parameters were used for a typical low carbon steel, such as ANSI 1010, and a standard 55-gallon drum. These parameters include material properties, such as yield and tensile strength, and drum dimensions. The buckling cases use axial stresses due to drums stacked three high. The lifting calculation used standard elastic shell stability equations. For all calculations, the allowable stress included a safety factor of 3, which is specified in the SRS Hoisting and Rigging Manual[13].

For lifting, the drum wall thickness can become quite thin before the bottom would separate or fall out. For the case of buckling, the effect of wall perforation dictates the wall thickness for support increase significantly. For example, in a drum made of 20-gauge sheet steel (0.033 in. minimum) with a 25% loss in wall perimeter, the minimum wall allows for essentially no corrosion on the remaining wall.

Table 5.3.1 Minimum Wall Calculations for DUO Drums

Scenario	Minimum Wall (inch)
Lifting	0.0024
Buckling	0.0054
Buckling with 25% loss	0.0296

6. E-AREA DISPOSAL AS LOW-LEVEL WASTE

6.1 Introduction

The E-Area Low-Level Waste Facility (ELLWF) has a number of types of disposal units for low-level solid waste (LLW). The units potentially suited to disposal of the depleted uranium drums are shown in Table 6.1.1.

Table 6.1.1 ELLWF Disposal Units Potentially Suited for Depleted Uranium Drums

Disposal Unit	Disposal Cost* $/m^3	Disposal Capacity ft^3	Number of Units to Accommodate 36,000 Drums
Intermediate Level Vault (ILV)	6,740	1.87×10^5	2.9
Low-Activity Waste Vault (LAWV)	2,519	1.08×10^6	0.5
Slit Trenches (set of 5)	417	1.015×10^6	0.5
Components-In-Grout (CIG) Trenches (set of 5)	3,726	1.015×10^6	0.5
Engineered Trench (ET)	935	1.6×10^6	0.4

* From System Plan for Solid Waste Management, WSRC-RP-99-01092, Rev. 4, 12, 2001.

All waste received for disposal at the ELLWF must be certified to be non-hazardous per the Resources Conservation and Recovery Act (RCRA) (i.e., contain no RCRA listed wastes and not be characteristically hazardous). Based on the data in Table 1, the depleted uranium drums are likely non-hazardous. However, assuming the drums are otherwise suitable for disposal in the ELLWF, they would have to be characterized per the requirements of the SRS 1S Manual.

6.2 Analysis of Acceptability For Disposal in the ELLWF

To determine the potential acceptability of the drums for disposal at the ELLWF, the results of the radiological analysis of material from 22 of the drums (see the Appendix to this chapter) were used. These results are summarized in Table 6.2.1.

Two cases were considered. The average radionuclide content was used as one case and the average plus two standard deviations beyond the average (i.e. 95% confidence interval) was used for the other case, to judge how uncertainty in the radionuclide content might influence acceptability.

Waste to be disposed in the LLWF must conform to the radionuclide disposal limits established for each of the disposal units.[14] The limits are derived from a variety of perspectives. One is criticality safety, which results in a limit on the fissile gram equivalent (FGE) ^{235}U content. However, for depleted uranium, the FGE ^{235}U limit does not apply.

Table 6.2.1 Summary of UO_3 data

Radionuclide	Average Ci/ft^3	Average + 2 Standard Deviations Ci/ft^3
Sr-90	1.59E-05	4.18E-05
Tc-99	1.43E-02	3.36E-02
Cs-137	1.33E-04	4.27E-04
I-129	4.03E-09	9.30E-09
U-234	5.73E-03	1.44E-02
U-235	3.19E-04	5.03E-04
U-236	6.47E-04	1.17E-03
U-238	3.08E-02	3.08E-02
Np-237	1.73E-04	9.73E-04
Pu-238	1.09E-04	5.77E-04
Pu-239	1.93E-03	1.06E-02
Pu-240	1.30E-04	7.09E-04
Am-241	1.52E-01	6.38E-01

Other limits are derived from the radiological performance assessment (PA)[15] . Since the WAC will soon be updated by including PA limits recently updated, the updated PA limits, rather than those presently in the WAC, were used in this study.[16] The PA-derived radionuclide inventory limits for radionuclides in the depleted uranium are shown in Table 6.2.2.

Acceptance of waste, with respect to these limits, is based on two criteria. First, the concentration of radionuclides in a waste package may not generally exceed ten times the average concentration derived from the PA. Second, a running total of the inventory of radionuclides disposed in each unit is maintained so that the total radionuclide inventory limit, as shown in Table 6.2.2, is not exceeded. Since waste contains a mix of radionuclides, the running total is maintained on the sum of the fractions of each individual radionuclide limit; the sum of fractions must not exceed 1.
The radionuclide concentration limits are shown in Table 6.2.3.

Table 6.2.2 PA-Derived LLWF Radionuclide Inventory Limits

Radionuclide	LAWV, Ci	ILV, Ci	Slit Trench, Ci	ET, Ci	CIG, Ci
Sr-90	4.60E+18	1.80E+10	5.20E+02	5.20E+02	2.20E+05
Tc-99	6.00E+00	2.50E+01	6.10E-01	6.10E-01	3.50E-01
Cs-137	9.00E+03	6.50E+05	2.10E+04	2.10E+04	2.20E+06
I-129	1.20E-03	5.20E-04	1.00E-03	1.00E-03	4.20E-04
U-234	1.20E+02	1.50E+01	1.10E+01	1.10E+01	9.80E+00
U-235	2.60E+01	6.00E+00	8.00E+00	8.00E+00	2.30E+01
U-236	5.80E+02	3.10E+04	2.00E+00	2.00E+00	4.60E+02

Radionuclide	LAWV, Ci	ILV, Ci	Slit Trench, Ci	ET, Ci	CIG, Ci
U-238	1.30E+02	4.90E+01	7.40E+00	7.40E+00	1.20E+02
Np-237	4.00E+00	6.00E+00	9.10E-02	9.10E-02	9.40E+00*
Pu-238	1.50E+06	2.20E+10	8.80E+02	8.80E+02	1.40E+04
Pu-239	1.80E+02	2.90E+04	8.50E-01	8.50E-01	1.30E+02
Pu-240	2.60E+02	1.30E+05	1.10E+00	1.10E+00	1.30E+02
Am-241	2.10E+04	3.00E+04	4.50E+02	4.50E+02	2.70E+02

***** Limit derived for the Intimately-Mixed Cement-Stabilized Waste Trenches, the Components-in-Grout trenches were not analyzed for ^{237}Np because it was not in the waste forecast.

Table 6.2.3 PA-Derived LLWF Radionuclide Concentration Limits

Radionuclide	LAWV, Ci/ft^3	ILV, Ci/ft^3	Slit Trench, Ci/ft^3	ET, Ci/ft^3	CIG, Ci/ft^3
Sr-90	4.26E+13	9.63E+05	5.12E-03	3.25E-03	2.17E+00
Tc-99	5.56E-05	1.34E-03	6.01E-06	3.81E-06	3.45E-06
Cs-137	8.33E-02	3.48E+01	2.07E-01	1.31E-01	2.17E+01
I-129	1.11E-08	2.78E-08	9.85E-09	6.25E-09	4.14E-09
U-234	1.11E-03	8.02E-04	1.08E-04	6.88E-05	9.66E-05
U-235	2.41E-04	3.21E-04	7.88E-05	5.00E-05	2.27E-04
U-236	5.37E-03	1.66E+00	1.97E-05	1.25E-05	4.53E-03
U-238	1.20E-03	2.62E-03	7.29E-05	4.63E-05	1.18E-03
Np-237	3.70E-05	3.21E-04	8.97E-07	5.69E-07	9.26E-05*
Pu-238	1.39E+01	1.18E+06	8.67E-03	5.50E-03	1.38E-01
Pu-239	1.67E-03	1.55E+00	8.37E-06	5.31E-06	1.28E-03
Pu-240	2.41E-03	6.95E+00	1.08E-05	6.88E-06	1.28E-03
Am-241	1.94E-01	1.60E+00	4.43E-03	2.81E-03	2.66E-03

***** Limit derived from the Intimately-Mixed Cement-Stabilized Waste Trenches inventory limit, the Components-in-Grout trenches were not analyzed for ^{237}Np because it was not in the waste forecast.

To facilitate the comparison of the depleted uranium drums with the LLWF limits, a limit for ^{237}Np in the Components-In-Grout trenches was assumed, based on the PA results for the Intimately-Mixed Cement-Stabilized Waste Trenches, because ^{237}Np was not included in the CIG analysis.

Table 6.2.4 shows the ratio of the average drum concentration to the PA-derived concentration limits. Table 6.2.5 shows the ratio of the 95% confidence interval drum concentration (i.e., the average drum concentration plus two standard deviations) to the PA-derived concentration limits.

The ratios in Tables 6.2.6 and 6.2.7 show that, based on radionuclide concentrations, the depleted uranium drums are not generally acceptable for disposal in the LLWF. Table 6.5 shows that, based on the average UO_3 analysis, the ILV limits are restrictive for three radionuclides (i.e., ^{99}Tc, ^{234}U, and ^{238}U). If the 95% confidence interval concentrations

are used, the limits for two other radionuclides (i.e., ^{235}U, and ^{237}Np) are exceeded. For the other disposal units, more radionuclides exceed the concentration limits.

Waste acceptance procedures at the LLWF permit waste exceeding the concentration limits to be accepted, provided the waste does not exceed five percent of the respective inventory limits. Tables 6.2.6 and 6.2.7 show the number of drums required to equal five percent of the radionuclide inventory limits, based on individual radionuclides.

Table 6.2.4 Ratio of Average Drum Concentration to Concentration Limit*

Radionuclide	LAWV	ILV	Slit Trench	ET	CIG
Sr-90	3.73E-19	1.65E-11	3.10E-03	4.89E-03	7.33E-06
Tc-99	2.57E+02	1.07E+01	2.37E+03	3.74E+03	4.14E+03
Cs-137	1.59E-03	3.81E-06	6.41E-04	1.01E-03	6.12E-06
I-129	3.63E-01	1.45E-01	4.09E-01	6.45E-01	9.74E-01
U-234	5.16E+00	7.14E+00	5.29E+01	8.33E+01	5.93E+01
U-235	1.33E+00	9.94E-01	4.05E+00	6.38E+00	1.41E+00
U-236	1.20E-01	3.90E-04	3.28E+01	5.17E+01	1.43E-01
U-238	2.56E+01	1.17E+01	4.22E+02	6.65E+02	2.60E+01
Np-237	4.66E+00	5.38E-01	1.92E+02	3.03E+02	1.86E+00
Pu-238	7.86E-06	9.28E-11	1.26E-02	1.99E-02	7.92E-04
Pu-239	1.16E+00	1.25E-03	2.31E+02	3.63E+02	1.51E+00
Pu-240	5.38E-02	1.86E-05	1.19E+01	1.88E+01	1.01E-01
Am-241	7.82E-01	9.48E-02	3.43E+01	5.41E+01	5.72E+01

* Ratios greater than one (i.e., drum concentration exceeds limit) are shaded

Using the average radionuclide content of the drums, 2.9 drums would equal 5% of the ^{99}Tc inventory limit for the LAWV, 11 drums would equal 5% of the ^{238}U limit for the ILV; 0.3 drums would equal 5% of the ^{99}Tc limit for the slit trenches and the ET; and 0.2 drums would equal 5% of the ^{99}Tc inventory limit for CIG. Using the 95% confidence interval number of drums, 1.2 drums would equal 5% of the ^{99}Tc inventory limit for the LAWV; 5 drums would equal 5% of the ^{99}Tc inventory limit for the ILV; 0.1 drum would equal 5% of the ^{99}Tc limit for the slit trenches and the ET; and 0.07 drums would equal 5% of the ^{99}Tc limit for CIG.

Table 6.2.5 Ratio of 95% Confidence Interval (i.e., Average + 2 Std Dev) Drum Concentration to Concentration Limit*

Radionuclide	LAWV	ILV	Slit Trench	ET	CIG
Sr-90	9.82E-19	4.34E-11	8.16E-03	1.29E-02	1.93E-05
Tc-99	6.04E+02	2.51E+01	5.59E+03	8.81E+03	9.74E+03
Cs-137	5.12E-03	1.23E-05	2.06E-03	3.25E-03	1.97E-05
I-129	8.37E-01	3.35E-01	9.44E-01	1.49E+00	2.25E+00
U-234	1.30E+01	1.79E+01	1.33E+02	2.09E+02	1.49E+02

Radionuclide	LAWV	ILV	Slit Trench	ET	CIG
U-235	2.09E+00	1.57E+00	6.38E+00	1.01E+01	2.22E+00
U-236	2.18E-01	7.06E-04	5.94E+01	9.37E+01	2.58E-01
U-238	2.56E+01	1.18E+01	4.22E+02	6.66E+02	2.60E+01
Np-237	2.63E+01	3.03E+00	1.08E+03	1.71E+03	1.05E+01
Pu-238	4.15E-05	4.90E-10	6.65E-02	1.05E-01	4.18E-03
Pu-239	6.36E+00	6.83E-03	1.26E+03	1.99E+03	8.27E+00
Pu-240	2.94E-01	1.02E-04	6.54E+01	1.03E+02	5.53E-01
Am-241	3.28E+00	3.98E-01	1.44E+02	2.27E+02	2.40E+02

* Ratios greater than one (i.e., drum concentration exceeds limit) are shaded

Table 6.2.6 Number of Drums (Average) to Equal 5% of Inventory Limit*

Radionuclide	LAWV	ILV	Slit Trench	ET	CIG
Sr-90	1.97E+21	7.71E+12	2.23E+05	2.23E+05	9.42E+07
Tc-99	2.86E+00	1.19E+01	2.91E-01	2.91E-01	1.67E-01
Cs-137	4.62E+05	3.33E+07	1.08E+06	1.08E+06	1.13E+08
I-129	2.02E+03	8.77E+02	1.69E+03	1.69E+03	7.09E+02
U-234	1.42E+02	1.78E+01	1.31E+01	1.31E+01	1.16E+01
U-235	5.54E+02	1.28E+02	1.71E+02	1.71E+02	4.90E+02
U-236	6.10E+03	3.26E+05	2.10E+01	2.10E+01	4.84E+03
U-238	2.87E+01	1.08E+01	1.64E+00	1.64E+00	2.65E+01
Np-237	1.58E+02	2.36E+02	3.59E+00	3.59E+00	3.70E+02
Radionuclide	LAWV	ILV	Slit Trench	ET	CIG
Pu-238	9.34E+07	1.37E+12	5.48E+04	5.48E+04	8.72E+05
Pu-239	6.34E+02	1.02E+05	2.99E+00	2.99E+00	4.58E+02
Pu-240	1.37E+04	6.83E+06	5.78E+01	5.78E+01	6.83E+03
Am-241	9.39E+02	1.34E+03	2.01E+01	2.01E+01	1.21E+01
Sum Of Inventory Fractions**	5.76E-02	1.33E-01	7.07E-02	7.07E-02	5.18E-02

* The least number of drums for each unit is shaded

** The sum of the fractions of each radionuclide inventory limit, assuming the least number of drums for each unit

Tables 6.2.6 and 6.2.7 also show the sum of the fractions of the inventory limits, assuming the least number of drums for each unit is disposed. The sum of fractions is up to 3 times higher than the five- percent criterion. Therefore, the number of drums that could be disposed in each unit would have to be reduced accordingly.

Table 6.2.7 Number of Drums (95% Confidence Interval) to Equal 5% of Inventory Limit*

Radionuclide	LAWV	ILV	Slit Trench	ET	CIG
Sr-90	7.48E+20	2.93E+12	8.46E+04	8.46E+04	3.58E+07
Tc-99	1.22E+00	5.06E+00	1.24E-01	1.24E-01	7.09E-02
Cs-137	1.43E+05	1.04E+07	3.35E+05	3.35E+05	3.51E+07
I-129	8.77E+02	3.80E+02	7.31E+02	7.31E+02	3.07E+02
U-234	5.67E+01	7.09E+00	5.20E+00	5.20E+00	4.63E+00
U-235	3.52E+02	8.12E+01	1.08E+02	1.08E+02	3.11E+02
U-236	3.37E+03	1.80E+05	1.16E+01	1.16E+01	2.67E+03
U-238	2.87E+01	1.08E+01	1.63E+00	1.63E+00	2.65E+01
Np-237	2.80E+01	4.20E+01	6.36E-01	6.36E-01	6.57E+01
Pu-238	1.77E+07	2.59E+11	1.04E+04	1.04E+04	1.65E+05
Pu-239	1.16E+02	1.86E+04	5.46E-01	5.46E-01	8.35E+01
Pu-240	2.49E+03	1.25E+06	1.06E+01	1.06E+01	1.25E+03
Am-241	2.24E+02	3.20E+02	4.79E+00	4.79E+00	2.88E+00
Sum Of Inventory Fractions**	5.64E-02	1.20E-01	7.85E-02	7.85E-02	5.23E-02

* The least number of drums for each unit is shaded

** The sum of the fractions of each radionuclide inventory limit, assuming the least number of drums for each unit

These results show that, given the current PA-derived radionuclide inventory limits, disposal of the depleted uranium drums in the ELLWF is not practical.

6.3 Reducing Conservatism in ELLWF PA Limits

The PA analyses are designed to be conservative (i.e., over estimate impacts). The PA process is an iterative one. Initially, results of analyses (i.e., radionuclide limits) are compared with projected waste disposals. If the projected waste can be readily accommodated within the calculated limits, the analyses need not be updated. However, if new waste, such as the depleted uranium, is forecast and the new waste cannot be accommodated within the PA limits, analyses can be revised, as appropriate to accommodate the new waste.

As shown in Table 6.2.6, concentration limits for ^{99}Tc, ^{234}U and ^{238}U limit the acceptability of the depleted uranium for the ILV. The 99Tc limit is derived from the groundwater pathway analysis. To increase the limit, waste form specific data on the leachability of the technetium from the UO_3 would have to be obtained. Then, the groundwater model would have to be exercised to determine the impact on the limit. Unless the UO_3 is stabilized with some material to precipitate the technetium (e.g., blast

furnace slag), the technetium leachability is not likely to be low enough to significantly increase the limit.

The ^{234}U limit derives from the emanation of radon. The PA radon analysis is very conservative for the disposal units that surround the waste with concrete (i.e., LAWV, ILV, and CIG). A Special Analysis could be done to refine the radon analysis, which would likely increase the ^{234}U limits for those units. However, the ^{234}U limit for the ILV, derived from the intruder resident scenario at 10,000 years is only 25 curies. Thus, a significant increase in the ^{234}U limit, derived from the radon analysis, for the ILV will result in the resident scenario limit becoming the controlling limit and the ILV ^{234}U limit would increase only from 15 curies to 25 curies. If conservatism could be significantly reduced in the resident scenario, the ILV ^{234}U limit could be increased to about 8,300 curies (i.e., the groundwater limit). Reducing the conservatism in the resident scenario would also increase the ^{238}U limit.

6.4 Disposal of UO3 in Saltstone

The Saltstone facility at SRS is a LLW disposal facility for low-level salt waste from the SRS high-level waste tanks. The salt waste is piped to the Saltstone treatment facility in Z-Area where the waste is mixed with cementitious solids. The resulting grout is then pumped to large concrete vaults for disposal. A PA has been performed for Saltstone disposal[16] A total of 10,032 55-gallon drums of grouted waste from the SRS Naval Fuels facility have also been emplaced in the Saltstone disposal facility. The drummed waste contains a total of 30 kilograms of uranium.

The Saltstone PA was conducted to validate the disposal of the entire projected inventory of decontaminated salt solution arising from the In-Tank Precipitation process. Therefore, disposal limits were not derived from the PA. Recently, efforts began to dispose of salt waste that has not been decontaminated (i.e., low-curie salt). Part of that work is to revise the Saltstone PA analyses to provide radionuclide disposal limits. When that work is complete (i.e., about October, 2002), the radionuclide content of the depleted uranium could be compared with the Saltstone disposal limits to assess the acceptability of the depleted uranium for disposal in Saltstone.

6.5 Conclusions and Recommendations

Analysis of the radionuclide content of the depleted UO_3 versus the radionuclide acceptance limits for the ELLWF shows that the UO_3 cannot now be disposed in E-Area. If disposal in E-Area needs to be pursued further, laboratory studies of the leachability of radionuclides from the depleted uranium must be done and the results incorporated into a Special Analysis. New samples of the drummed UO_3 should be collected and analyzed to confirm the current levels of radionuclides, with particular interest in Tc-99, to assess whether the material meets the radiological performance assessment limits. It is likely that the UO_3 will have to be stabilized in a material that retards technetium release in order for the waste to be acceptable for disposal in E-Area.

Table 6.6.1 Summary of UO3 Data

Radio-nuclide	Average ppm	Std Dev ppm	Specific Activity, Ci/g	Average Ci/g UO_3	Std Dev Ci/g UO_3	Average Ci/ft^{3*}	Std Dev Ci/ft^{3*}	Avg + 2 Std Dev Ci/ft^{3*}
K-40	2.87E+03	1.40E+03	6.99E-06	2.01E-08	9.79E-09	1.84E-03	8.98E-04	3.64E-03
Sr-90	1.25E-06	1.02E-06	139.031	1.73E-10	1.41E-10	1.59E-05	1.30E-05	4.18E-05
Tc-99	9.07E+00	6.14E+00	0.017146	1.56E-07	1.05E-07	1.43E-02	9.65E-03	3.36E-02
Cs-137	1.67E-05	1.85E-05	86.70568	1.45E-09	1.60E-09	1.33E-04	1.47E-04	4.27E-04
I-129	2.73E-04	1.79E-04	0.000161	4.40E-14	2.88E-14	4.03E-09	2.64E-09	9.30E-09
U-234	1.00E+01	7.56E+00	0.006247	6.25E-08	4.72E-08	5.73E-03	4.33E-03	1.44E-02
U-235	1.61E+03	4.63E+02	2.16E-06	3.48E-09	1.00E-09	3.19E-04	9.18E-05	5.03E-04
U-236	1.09E+02	4.42E+01	6.47E-05	7.05E-09	2.86E-09	6.47E-04	2.62E-04	1.17E-03
U-238	9.98E+05	5.04E+02	3.36E-07	3.35E-07	1.69E-10	3.08E-02	1.55E-05	3.08E-02
Np-237	2.67E+00	6.19E+00	0.000705	1.88E-09	4.36E-09	1.73E-04	4.00E-04	9.73E-04
Pu-238	6.96E-05	1.49E-04	17.11435	1.19E-09	2.55E-09	1.09E-04	2.34E-04	5.77E-04
Pu-239	3.40E-01	7.62E-01	0.061974	2.11E-08	4.72E-08	1.93E-03	4.33E-03	1.06E-02
Pu-240	6.23E-03	1.39E-02	0.226674	1.41E-09	3.16E-09	1.30E-04	2.90E-04	7.09E-04
Am-241	4.83E-01	7.73E-01	3.431395	1.66E-06	2.65E-06	1.52E-01	2.43E-01	6.38E-01

- Assuming each 55-gallon drum contains 1500 pounds of UO_3

Appendix
Results of Analysis of UO_3

	Am-241	Cs-137	K-40	Np-237	Pu-238	Pu-239	Pu-240	Tc-99	I-129	Sr-90	Uranium Isotopics (wt %)			
	ppm	ppm	ppm	ppm	ppm	ppm	ppm	ppm	ppm	ppm	U-234	U-235	U-236	U-238
Average	4.83E-01	1.67E-05	2.87E+03	2.67E+00	6.96E-05	3.40E-01	6.23E-03	9.07E+00	2.73E-04	1.25E-06	1.00E-03	1.61E-01	1.09E-02	9.98E+01
Std Dev	7.73E-01	1.85E-05	1.40E+03	6.19E+00	1.49E-04	7.62E-01	1.39E-02	6.14E+00	1.79E-04	1.02E-06	7.56E-04	4.63E-02	4.42E-03	5.04E-02
LIMS #														
124454	3.255	7.6E-06	8.5E+02	0.783	8.24E-05	0.25	4.71E-03	34.7	1.40E-04	1.94E-06	0.001	0.137	0.009	99.853
124512	2.305	3.1E-05	4.6E+03	29.679	5.24E-05	0.33	5.98E-03	13.7	3.70E-04	1.46E-06	0.000	0.123	0.007	99.870
124513	0.287		3.4E+03	0.776	3.75E-05	0.19	3.42E-03	6.0	2.72E-04	1.75E-06	0.001	0.139	0.009	99.852
124514	0.237		4.2E+03	3.494	7.99E-06	0.03	6.08E-04	9.0	3.04E-04	2.61E-06	0.002	0.282	0.020	99.696
124515	0.150			0.794	1.62E-05	0.07	1.24E-03	5.9	6.96E-05	1.93E-06	0.001	0.146	0.009	99.844
124516	0.443	6.2E-05		0.769	1.01E-05	0.06	1.16E-03	7.7	9.11E-05	1.74E-06	0.001	0.137	0.009	99.854
124517	0.409		3.3E+03	6.299	1.55E-05	0.10	1.77E-03	9.0	5.88E-04	2.09E-06	0.001	0.122	0.008	99.869
124518	0.327		3.9E+02	0.764	8.99E-05	0.45	8.21E-03	8.1	1.66E-04	1.89E-06	0.001	0.185	0.012	99.802
124519	0.064	9.2E-06	1.0E+03	0.782	3.16E-05	0.10	1.81E-03	5.9	3.26E-05	1.77E-06	0.000	0.145	0.010	99.845
124520	0.233	7.1E-06		0.782	1.63E-05	0.07	1.24E-03	8.5	4.61E-05	2.15E-06	0.002	0.253	0.017	99.729
124521	0.414		3.3E+03	0.777	2.15E-05	0.14	2.46E-03	7.8	8.81E-05	1.83E-06	0.000	0.137	0.008	99.856
124522	0.112	8.4E-06		0.783	0.00E+00	0.00	0.00E+00	4.3	2.08E-04	2.59E-07	0.001	0.124	0.007	99.868
124523	0.473		2.1E+03	0.768	1.56E-05	0.10	1.78E-03	8.7	3.42E-04	2.47E-07	0.000	0.122	0.006	99.872
124524	0.235			2.979	1.45E-05	0.06	1.10E-03	6.8	3.87E-04	2.54E-07	0.000	0.152	0.011	99.837
124525	0.145	8.7E-06		0.785	2.57E-05	0.11	1.96E-03	7.9	2.95E-04	2.57E-07	0.001	0.191	0.008	99.799
124526	0.144	7.1E-06		3.090	1.14E-05	0.06	1.04E-03	6.9	1.11E-04	2.55E-07	0.002	0.150	0.011	99.837
124527	0.231		3.0E+03	0.756	3.25E-05	0.13	2.47E-03	6.7	4.97E-04	2.59E-07	0.000	0.140	0.011	99.849
124528	0.163	9.1E-06	3.7E+03	0.799	0.00E+00	0.00	0.00E+00	12.3	4.06E-04	2.58E-07	0.001	0.142	0.010	99.848
124529	0.091		4.3E+03	0.777	8.99E-05	0.24	4.56E-03	8.4	3.87E-04	2.53E-07	0.002	0.256	0.017	99.725
124530	0.084			0.773	2.24E-04	1.41	2.56E-02	6.2	6.68E-04	2.61E-07	0.001	0.133	0.010	99.856
124531	0.680		4.2E+03	0.769	3.67E-05	0.11	2.09E-03	9.9	1.51E-04	3.69E-06	0.002	0.139	0.008	99.852
124532	0.149		1.8E+03	0.765	7.00E-04	3.49	6.39E-02	4.9	3.95E-04	2.50E-07	0.002	0.187	0.023	99.789

For neptunium results, shaded data represents the limit of detection

7. DISPOSAL OF UO3 IN HLW TANKS

A review of the modeling work performed in support of the Tank 17 Closure Module reveals an inventory of Tc-99 of 3.58 Ci. The output tables of the model shows a dose at the seepline from the Barnwell-McBean aquifer as being the highest measured at 0.060 mrem/yr. Since this model reflects a direct proportion between inventory and dose, a ratio can be established relating the maximum inventory, the inventory used, the groundwater protection limit (4 mrem/yr) and the calculated dose:

X Ci Tc-99 / 3.58 Ci Tc-99 = 4 mrem/yr / 0.060 mrem/yr

Solving this equation gives the limiting inventory of 239 Ci of Tc-99. The methods utilized in modeling the HLW tank farm reflect the total limit for all of the F-Area tanks contributing to the F-Area GTS to be approximately 239 Ci of Tc-99. The limit of 239 Ci of Tc-99 is considerably less than the projected inventory of Tc-99 in the depleted uranium of over 2,000 Ci.

8. DEPLETED URANIUM TRIOXIDE AS GROUT CONSTITUENT FOR HLW TANKS CLOSURE

8.1 Acceptability of DUO as a Grout Constituent

Depleted uranium trioxide may be used as high-density inert filler in the production of grout. The DUO could serve as a stand-in for quartz sand, which is inert from the standpoint of environmental transport and contamination. As such, the DUO stored on site would be put to advantageous use, saving the cost of continued monitoring the stored DUO materials and the cost of the quartz sand grout constituent.

The chemical analyses cation constituents identified in Section 2.1 will not present problems in processing grout suitable for High Level Waste tank closure. However, an anion analysis should be conducted to confirm the soluble salts concentrations are below 1 to 2 wt%, as higher concentrations may present setting problems, i.e. flash setting.

The DUO material is a fine powder similar to Portland cement in particle size distribution. Section 2.2 indicates about 81.6% of the DUO powder is smaller than 80 mesh, or about 180 micrometers in grain size. This is slightly courser than Portland cement, but is not a direct replacement for quartz sand that is typically about 5 micrometers in size. As such, proportioning of materials and material handling will have to be adjusted to use DUO as a replacement for quartz sand. Care must be taken to avoid clumping of damp material during grout production using DUO. Replacing all of the quartz sand with DUO powder will produce a fill material that resembles thick slurry rather than the typical grout. Grout made with quartz sand consists of fine aggregate in a paste/slurry carrier, while grout made with DUO in place of quartz sand will consist of particles having about the same size. Testing should be performed to evaluate the flow and setting behavior of grout containing DUO.

8.2 Scoping Studies for DUO Grout Development

The property requirements and test methods for a bulk tank fill material are shown in Table 8.2.1. The objective of the tests is to develop a flowable, self-leveling, zero bleed water fill that can be pumped into the empty HLW tanks. The cured function of the fill grout is to support the weight of the tank overburden and prevent future subsidence. Chemical stabilization of incidental waste is not a requirement of the bulk fill material.

Scoping studies using DUO as a replacement for quartz sand in producing grout should be performed at SRTC in a ventilation hood specified by radcon personnel. Radiological and other hazardous characteristics (such as dusting) of the DUO powder must be considered in determining the exact location where the scoping tests would be performed. A technical task plan (TTP) should be prepared for the task, which would identify the exact location for performance of the tests, as well as other precautions and safety requirements that need to be followed in handling the DUO powder.

Table 8.2.1. Test methods and acceptance criteria for HLW bulk tank fill material.

PROPERTY		ASTM REFERENCE	ACCEPTANCE CRITERIA
Flow		D-6103	10 inches (minimum)
Bleed Water		C-232	Zero vol. % after 24 hr.
Air Content		C-231	N/A
Unit Weight		C-138	80 lbs/ft^3 (minimum)
Compressive Strength	7 days	C-39	50 psi
	28 days		100 psi (minimum)
	56 days		500 psi (minimum)
Setting Time		C-403	30 hr. (maximum)

Approximately 1 kilogram of DUO powder would be required to perform the scoping studies, to be supplied to SRTC by HLW or other designated organization. The initial test would involve simply mixing the DUO powder with cement and water to obtain a pourable, flowable mixture. The resultant mix would then be observed for settling of the DUO powder, due to its higher density property.

In order to prepare 0.8 cubic foot batches of DUO grout for scale-up testing, a mixer and compressive strength instrumentation would have to be purchased by SRTC. A ventilation hood or containment hut would have to be constructed for working with the larger volume of radioactive grout material. Provisions would have to be made for clean-up of the mixing and testing equipment used with the larger DUO grout batches. The TTP must include requirements to develop procedures and identify facilities for

conducting the scale-up scoping tests. Off-site options for conducting experiments with the actual DUO powder should be considered.

8.3 DUO Drum Handling and Grout Mixing Process

The process for mixing DUO and grout will require measuring the various batch ingredients and DUO powder in the correct proportions, conveying the constituents into a mixer (continuous or batch), and mixing the ingredients (including slag, fly ash etc.) to form a slurry. The resulting slurry will then require pumping into the HLW waste tanks. Key issues to be addressed include DUO storage drum handling and unloading, proportioning of constituents, mixing, and pumping. Safety and industrial hygiene considerations must be addressed. A team should be assembled including radcon personnel, T&T personnel, BSRI engineers and construction (who are supporting Tank 19 closure) and SRTC to address these topics. A simplified schematic of the grouting operation is described below.

The entire drum and DU powder-handling operation should be installed in a container trailer. In placing the system within a trailer, the transportable grouting operation (TGO) drum handling and grouting equipment could be delivered to the tank to be grouted, and then moved to another tank upon completion. One self-contained system could be used for all the grouting, easily moved between tanks. The need for cleanup of equipment between grouting operations would be minimized.

With this system, the drums of DUO would be placed into an airlock system. The outside airlock door would close, the inside door would open, and the drum would be transferred into a drum dumping station. Automated drum handling equipment would load the drum into the drum dumping station. The design of the drum dumping station will be dependent on the flowability of the powder in the drums and the condition of the drums. Ideally the drums would be mated to a seal to minimize dusting, then the drums would be inverted. The opposite end of this drum seal will have a pipe and a valve that mate to an auger. The valve and pipe will have to be large enough to insure the powder cannot bridge. The valve would be opened and the DUO powder would flow into the auger.

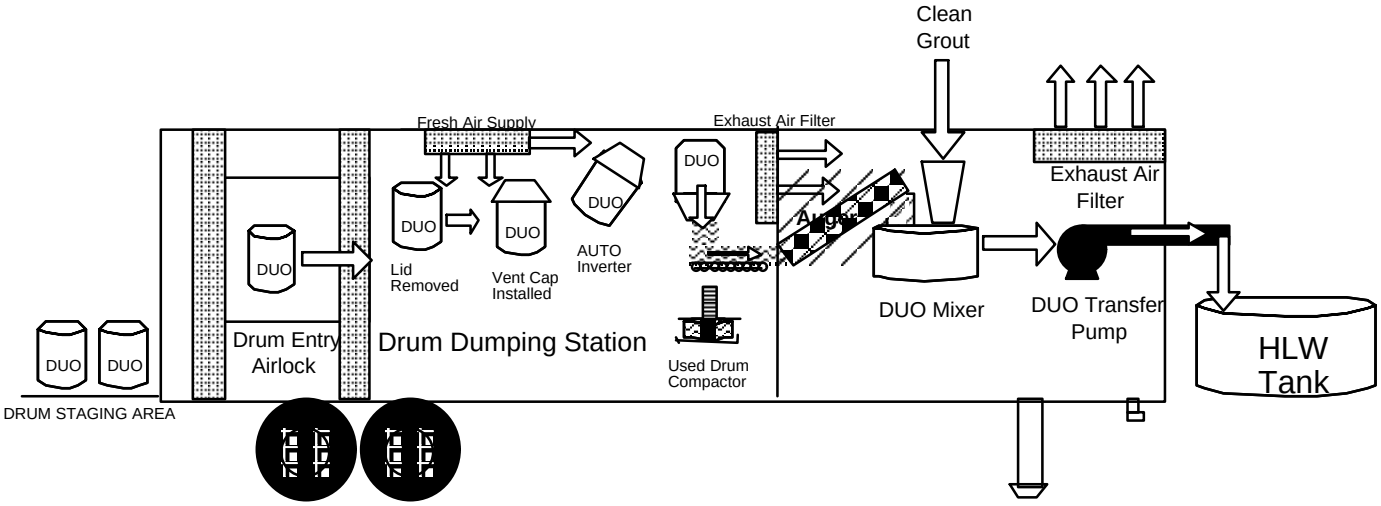
It is possible to use misting systems to minimize dusting, but this could create flow problems and/or clumping. Internal filtered vacuum systems can also be used to control dusting. Since this would all be contained within the trailer, environmental and industrial hygiene concerns would be controlled. The internal filtration system would be used only to filter the air near the dusty operations to keep the dust from building up within the trailer. The filtration system would also reduce the amount of dust building up on equipment and cameras, prolonging equipment life and maintaining camera functionality.

Brush, vibration, or impact systems could be used to remove residual powder from the drums. The drums would then go to a crusher and be transferred through a bagless transfer port to a larger 85-gallon drum for eventual disposal. Several designs of these ports already exist in the complex. These used drums would then be handled as low level waste.

The auger would feed the DUO powder into an in-line mixer. This auger would be a commercially available, rate-controlled auger. This line from the auger to the mixer would be sealed to prevent additional dusting. Grout from another source would also be pumped into the trailer and into the mixer. The mixer would have to be sized such that the correct amount of grout and DUO are combined and properly mixed in a single pass through operation. The amount of clean grout from the external source would be metered using the clean grout pump. This information would be fed to the DUO auger, which would take the input from the clean grout pump and meter out the correct amount of DUO to properly combine with the clean grout.

Depending on the location of the grouting system, the mixed DOU grout would then feed into another grout pump that would transfer the mixture into the HLW tank. To allow for flexibility in locating the grouting trailer, it is recommended that the DUO grout transfer pump also be located in the trailer. The DUO grout transfer pump will be part of the contaminated equipment.

Sketch 8.3.1 Simplified Transportable Grouting Operation



8.4 Costs Benefits from Grouting with DUO

The purpose of including depleted UO_3 in tank fill grout is to use it as a high-density inert filler. Such filler would take the place of quartz sand, which is inert from the standpoint of environmental transport and contamination. Because the site already has DUO powder in storage and available for use, the cost of the quartz sand typically used in the grouting process that will be replaced by the DUO powder could be saved.

The cost of quartz sand is about $10/2000 lbs. The amount of sand in the bulk tank fill is 2300 pounds per cubic yard. Since the specific gravity of the depleted UO_3 powder is

greater than that of the sand, a greater amount of UO_3 by weight may be included in the grout mix. If twice the weight is used, 4700 lbs. of UO_3 may be included in a cubic yard of tank fill. This represents \$10/yd^3 (or \$82,000 per HLW tank @ 8,200 yd^3/tank) savings in quartz sand cost, as the proportioning is by a volumetric basis. Studies conducted in the fall of 2001 indicate 16 HLW tanks would be required to dispose of the UO_3 powder involved, representing a cost savings of \$82K x 16 = \$1,312K. This estimated cost savings would be offset by the additional costs for DUO handling and mixing, which is yet to be determined. The exact cost savings should be confirmed by experiments, as the amount of UO_3 calculated here to be included in a unit volume of HLW tank fill material is only estimated.

8.5 Conclusions and Recommendations

The cured function of the fill grout in HLW tank closure is to support the weight of the tank overburden and prevent future subsidence. Chemical stabilization of incidental waste is not a requirement of the bulk fill material. Depleted uranium trioxide may be used as high-density inert filler in the production of grout, serving as a stand-in for quartz sand. Proportioning of materials and material handling would have to be adjusted to use DUO as a replacement for quartz sand, and care must be exercised to avoid clumping of damp material during grout production. Replacing all of the quartz sand with DUO powder will produce a fill material that resembles thick slurry rather than the typical grout.

A technical task plan (TTP) should be prepared identifying scoping studies using DUO as a replacement for quartz sand in producing tank closure grout. The TTP would identify the location for the performance of the tests, as well as other precautions and safety requirements that should be followed in handling the DUO powder. Anion analyses should be conducted with the DUO powder to confirm the soluble salts concentrations are low enough (below 1 to 2 wt%) so as to not to present setting problems. Testing should evaluate the flow and setting behavior of grout containing DUO.

Approximately 4,700 lbs. of DUO may be included in a cubic yard of tank fill grout. This represents \$10/yd^3 (or \$82,000 per HLW tank @ 8,200 yd^3/tank) savings in quartz sand cost. 16 HLW tanks would be required to dispose of the DUO powder, representing a cost savings of \$82K x 16 = \$1,312K in grout components cost. Some of this savings would be offset by additional costs incurred for special DUO handling and mixing requirements.

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