RECEIVED

JAN 26 1998

OSTI

POEF-LMUS-111

LOCKHEED MARTIN

**PORTSMOUTH
GASEOUS
DIFFUSION
PLANT**

Final Technology Status Report

Gas-Phase Decontamination Demonstration on PORTS Cell X-25-4-2

By

Russell J. Riddle

MASTER

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

September 1997

NOTICE

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

This report has been reproduced directly from the best available copy.

Available to DOE and DOE contractors from the Office of Scientific and Technical Information, P. O. Box 62, Oak Ridge, TN 37831; prices available from (423) 576-8401.

Available to the public from the National Technical Information Service, U. S. Department of Commerce, 5285 Port Royal Rd., Springfield, VA 22161.

DISCLAIMER

**Portions of this document may be illegible
electronic image products. Images are
produced from the best available original
document.**

Final Technology Status Report
Gas-Phase Decontamination Demonstration
on PORTS Cell X-25-4-2

By
Russell J. Riddle

September 1997

LOCKHEED MARTIN UTILITY SERVICES, INC.
Portsmouth Gaseous Diffusion Plant
P. O. Box 628 Piketon, Ohio 45661

Under Contract USEC-96-C-0001
to the
UNITED STATES ENRICHMENT CORPORATION

TABLE OF CONTENTS

EXECUTIVE SUMMARY.....	1
BACKGROUND.....	2
INTRODUCTION.....	3
DISCUSSION.....	4
Diffusion Cell Selection and Description.....	4
Demonstration Test Conditions and Test Cell Preparation.....	4
Analytical Techniques and Instrumentation.....	6
NDA Measurements.....	10
Test Cell Metal Samples and Coupons.....	10
RESULTS.....	12
Composition of the Clean-Up Gases.....	12
Test Cell Temperatures and Pressures.....	17
NDA Findings.....	18
Radiochemistry Findings.....	23
ECONOMICS OF FULL-SCALE IMPLEMENTATION.....	31
Methodology.....	31
Manpower Cost Comparison.....	31
LTLT Cost Savings.....	33
PORTS and PGDP Site Savings.....	35
D&D Program Duration and Capital Equipment Costs.....	35
CONCLUSIONS.....	37
RECOMMENDATION.....	38
ACKNOWLEDGEMENTS.....	39
REFERENCES.....	40

TABLE OF CONTENTS
(Continued)

APPENDIX A: LTLT Test Sample Manifold.....	A-1
APPENDIX B: LTLT Test Coupon Fixture	
LTLT Test Fixture Assembly.....	B-1
LTLT Test Fixture Installation.....	B-2
APPENDIX C: Test Cell Internal Gamma Scans	
Before Clean-Up.....	C-1
Before and After Clean-Up.....	C-2
APPENDIX D: Neutron Measurements on the S/11 Centrifugal Compressor	
Memorandum, B. McGinnis to R. Riddle.....	D-1
NDA Measurements Worksheet.....	D-2
APPENDIX E: Radiochemistry Data Reduction	
Alpha Contamination - Uranium Isotopes.....	E-1
Alpha Contamination - Transuranics.....	E-4
Alpha Contamination - Thorium-232.....	E-7
Beta/Gamma Contamination.....	E-10
APPENDIX F: Comparison of Sample Uranium Alpha Contamination to Free Release Guidance in USAEC Regulatory Guide 1.86	
Copper Seal Exhaust from K-25.....	F-1
Barrier from Test Cell S/9 Converter.....	F-2
Wipe Samples from Test Cell S/9 Converter.....	F-3
Copper Instrument Line from Test Cell.....	F-4
Aluminum Cooler Fins from Failed Gas Cooler.....	F-5
Ni-Lined Steel from Test Cell S/9 Converter Shell.....	F-6
Bare Steel from X-705 Scrap.....	F-7
Aluminum Rotor Samples from Test Cell S/11 Compressor.....	F-8

EXECUTIVE SUMMARY

The Long-Term, Low Temperature (LTLT) process is a gas-phase in situ decontamination technique which has been tested by LMES/K-25 personnel on the laboratory scale with promising results. The purpose of the Gas-Phase Decontamination Demonstration at PORTS was to evaluate the LTLT process on an actual diffusion cascade cell at conditions similar to those used in the laboratory testing. The demonstration was conducted on PORTS diffusion cell X-25-4-2 which was one of the X-326 Building cells which was permanently shutdown as part of the Suspension of HEU Production at PORTS.

The demonstration full-scale test consisted of rendering the cell leak-tight through the installation of Dresser seals onto the process seals, exposing the cell to the oxidants ClF_3 and F_2 for a period of 105 days and evaluating the effect of the clean-up treatment on cell samples and coupons representing the major diffusion cascade materials of construction. The progress of the clean-up reactions was monitored using a Fourier Transform Infrared (FTIR) analyzer and a Corrosive Gas Chromatograph (CGC). Non-Destructive Assay (NDA) measurements were obtained before, during and after exposure to clean-up gases. Radiochemistry analyses were used to determine the pre and post-exposure levels of uranium isotopes and other nuclides present. The post-exposure contamination levels were then compared to acceptable surface contamination levels for release for unrestricted use contained in USAEC Regulatory Guide 1.86.

The results were extrapolated to determine the effectiveness of LTLT decontamination over the range of historical uranium isotope assays present in the diffusion complex. It was determined that acceptable surface contamination levels could be obtained in all of the equipment in the lower assay cascades which represents the bulk of the equipment contained in the diffusion complex.

An economic evaluation of a full-scale D&D program performed using LTLT decontamination compared to existing wet decontamination technology indicated that approximately \$73MM and \$79MM could be saved at PORTS and PGDP respectively using LTLT in situ technology. The evaluation also indicated that a D&D program using LTLT technology could be completed in a reasonable amount of time (2 to 4 years) with minimal initial capital expenditure.

BACKGROUND

The gaseous diffusion process for uranium enrichment uses isotopic separation barrier arranged in diffusion cascades to increase the assay of the fissionable isotope, U-235, to levels suitable for use in commercial nuclear power reactors. In this process, uranium is enriched in the form of UF_6 . The uranium enrichment complex consists of three plants; two operating plants in Portsmouth, Ohio (PORTS) and Paducah, Kentucky (PGDP) and a shutdown plant in Oak Ridge, Tennessee (K-25).

The enrichment process has been performed at subatmospheric pressures in most of the equipment of the diffusion plants. During the operation of the enrichment cascades, inleakage of moist ambient air occurred at various times and locations. This resulted in the formation of solid uranium deposition on the internal surfaces of diffusion cascade equipment through the reaction below:

While the solid uranium materials present are primarily in the form of UO_2F_2 , other uranium oxyfluorides of the form $U_nO_{2n-1}F_{2n+2}$ (where n is an integer) were formed in limited moisture conditions. In addition, the solid materials UF_4 and UF_5 were likely formed from the reaction of UF_6 with internal metal surfaces.

For over forty years, ClF_3 or mixtures of ClF_3 and F_2 have been used to remove solid uranium deposits from running cascade equipment. Such running treatments involve addition of the oxidant(s) to an evacuated running cell (the smallest portion of a diffusion cascade that can be isolated from onstream equipment) in the offstream mode. The cell compressors circulate the clean-up gases throughout the cell and uranium deposition as UO_2F_2 is removed via the following approximate reaction:

The duration of these treatments is limited by elevated gas temperatures resulting from the heat of compression and treatment of a previously shutdown cell may require considerable maintenance to prepare the cell for running treatment. In addition, a number of potential hazards associated with running treatments have been identified. These include oxidant/metal reactions due to localized depletion of ClF_3 and oxidant reactions with hydrocarbon oils present in compressor lubrication systems. While running oxidant treatments have been effective in removing large uranium deposits which result in operational or NCS concerns, the above factors make this technique unattractive as a decontamination alternative. Consequently, the existing technology for uranium contamination removal from diffusion cascade equipment consists of disassembly followed by wet decontamination.

INTRODUCTION

With the eventual phase-out of the gaseous diffusion process for uranium enrichment, diffusion cascade equipment at K-25, PGDP and PORTS will require extensive decommissioning in the future. Decontamination of this equipment must be carried out as a prerequisite to or concurrent with decommissioning to alleviate criticality, health physics, industrial hygiene, and environmental concerns. With these concerns in mind, it is obvious that in situ decontamination techniques, which can be carried out before equipment disassembly, have advantages over existing technologies which involve disassembly followed by wet decontamination.

In the existing wet decontamination technology, decontamination solutions must be stored, processed and ultimately calcined to uranium oxide. These are labor intensive and time consuming activities with numerous health physics, safety and environmental concerns. The uranium oxide produced must be stored indefinitely because no oxide conversion facility is currently available. However, gas-phase clean-up technologies are less labor intensive and convert uranium contaminants to UF_6 , which can be returned to an operating cascade for eventual withdrawal as LEU product.

The LTLT process is a gas-phase in situ decontamination technique which has been tested by K-25 personnel on the laboratory scale¹. The process employs the fluorinating agents ClF_3 and F_2 , which are currently used at the operating diffusion plants to remove solid uranium deposits from running diffusion cascade cells. However, the LTLT process is intended to be conducted on shutdown cascade equipment at much lower temperatures (ambient) and the equipment surfaces are exposed to higher concentrations of fluorinating agents for a much longer period of time compared to existing running cell clean-up procedures. The laboratory scale testing was performed on the major materials of construction found in the diffusion cascades with promising results. Surface contamination was reduced to below free release limits specified in USAEC Regulatory Guide 1.86² for most of the materials tested.

The purpose of the gas-phase decontamination demonstration at PORTS was to evaluate the LTLT process on an actual diffusion cascade cell at conditions which were similar to those used in the laboratory scale testing. The results of this evaluation were then used to determine the economics of full-scale implementation at the diffusion sites.

DISCUSSION

Diffusion Cell Selection and Description

A shutdown PORTS HEU cell (X-25-4-2) was selected for the demonstration. This X-25 size cell has twelve diffusion stages each consisting of a centrifugal compressor, a converter containing the diffusion barrier and a gas cooler. The volume of X-25 size cells is relatively small (1530 ft³). The cell piping is configured in what is known as the badger cluster arrangement in which the pressure of each of four clusters of three stages was controlled by a single control valve unlike other areas of the PORTS cascade in which each stage has an associated control valve (badger stage). A schematic of an X-25 size cell is shown in Figure 1 on page 5.

An HEU cell was chosen for a number of reasons. The production of HEU at PORTS has been suspended and the HEU cascade has been permanently shutdown making this equipment available for test purposes. The selected cell was fully intact allowing the demonstration to be performed with minimal preliminary maintenance. The shutdown HEU cells contain all of the major materials of construction found in the diffusion cascades. Therefore, decontamination effectiveness could be evaluated for a number of materials simultaneously. The HEU Clean-Up Program effort removed all large uranium deposits present in this equipment to less than NCS safe mass. Therefore, no large deposits which would rapidly consume the fluorinating agents were present in the selected cell. PORTS X-326 Building cell service equipment which remains in operation to support LEU production was available for use in disposing of reaction products from the demonstration testing. Recovered UF₆ was returned to the X-326 LEU cascade for withdrawal as LEU product. The volume of the small X-25 size cell selected was much less than that of other cells in the diffusion complex so the amount of fluorinating gases required to duplicate laboratory test conditions was reduced. The shutdown PORTS HEU cascade contains a higher concentration of non-uranium contaminants than the LEU cascades. These include Technetium-99, Thorium isotopes and transuranics which have their origins in reactor return materials which were side-fed into the PORTS cascade in the past. While the LTLT process was intended for removal of uranium contamination from cascade surfaces, selection of a shutdown HEU cell allowed the effects of the LTLT treatment on non-uranium contaminants to be evaluated.

Demonstration Test Conditions and Test Cell Preparation

The conditions of the full-scale demonstration of the LTLT process in cell X-25-4-2 duplicated the conditions of the laboratory scale testing at K-25 as much as was practicable. These conditions are

Figure 1. Typical X-25 size diffusion cell (PORTS X-326 Building).

shown compared to K-25 laboratory test conditions in Table 1 below. The greatest departure from laboratory conditions was in the addition of dry nitrogen to the test cell as a diluent and the higher target total pressure of 14.0 psia which was maintained throughout the exposure period. Use of a target test pressure close to atmospheric pressure minimized the chances of inleakage of ambient air or outgassing of clean-up gases during the exposure period. In addition, as an extra precaution, the 24 process seals of the test cell were partially disassembled and Dresser seals were installed. A Dresser seal is basically a flanged aluminum cap with a Teflon gasket which is attached to the external portion of the seal to eliminate leakage normally associated with shutdown process seals. Photographs of the Dresser seal installations on the test cell are shown in Figure 2 on page 7 and Figure 3 on page 8.

Analytical Techniques and Instrumentation

The test cell was instrumented with a Bomem Fourier Transform Infrared (FTIR) analyzer, an Applied Automation Corrosive Gas Chromatograph (CGC) and a Keithley data acquisition system. A photograph of the FTIR and CGC installed at the test cell cubicle is shown in Figure 4 on page 9.

Table 1. Comparison of PORTS Full-Scale Demonstration and K-25 Laboratory Test Conditions.

Condition	PORTS Demo	K-25 Laboratory*
Temperature	Ambient (70-85 deg. F) with no external sources of heat applied	3 of 5 reactors: 140 deg. F maximum maintained using heat tape and temp. controller. 2 of 5 reactors: no external sources of heat applied.
Oxidants	Mixture of chlorine trifluoride and fluorine.	Chlorine trifluoride only.
Diluent	Dry nitrogen.	No diluent used.
Total Pressure	Target of 14.0 psia.	1, 4 or 8 psia depending on the experiment.
Exposure Period	105 days.	35-105 days depending on the experiment.

* Based on work reported in K/TCD-1048.

Figure 2. Dresser seal installations on two centrifugal compressors, test cell X-25-4-2.

The FTIR and CGC were connected to the existing cell sample manifold using a special manifold which was constructed for the test (see drawing, Appendix A). A pump was installed on the manifold to provide a continuous flowing sample loop. The small sample flow was the only circulation provided for the clean-up gases. No other form of circulation was employed during the test. Early in the test exposure period, a KNF Neuberger diaphragm pump was employed but was later replaced with a B-4 piston pump because of failure of the Teflon coated plastic diaphragms. The FTIR provided in-place continuous analyses of all infrared absorbing compounds of interest including ClF_3 , UF_6 , ClO_2F and ClF . The CGC provided in-place continuous analyses of the diatomic compounds F_2 and Cl_2 . Chromatographic analyses were used in conjunction with infrared analyses to provide a more complete mass balance of the clean-up gases and to address concerns identified by preliminary laboratory investigations related to potential destructive reactions of chlorinating and fluorinating gases with aluminum

Figure 3. Close-up of Dresser seal installed on compressor seal.

contained in diffusion equipment. The potential for formation of the thermally unstable compound ClO_2 was also investigated. Free Cl_2 , which could react with aluminum with destructive results, cannot exist in significant concentration concurrent with the excess of oxidants that was maintained throughout the exposure period. In addition, an evaluation of laboratory test data by D.W. Simmons³ and kinetic and thermodynamic studies performed by L.D. Trowbridge⁴ indicated that ClO_2 also cannot exist in the presence of excess oxidant. However, as an extra precaution and as a provision of the Safety Evaluation of the Unreviewed Safety Question Determination (USQD) which was conducted for the demonstration⁵, ClO_2 and Cl_2 were continuously monitored using the FTIR and CGC respectively for the duration of the exposure period.

Each of the twelve stages of the cell were continuously monitored for two hours during each day of the exposure period. Process Operators manually changed the stage being sampled to the next

Figure 4. Analytical instruments installed at the test cell cubicle. The CGC is on the left and the FTIR and sample buggy is on the right.

higher stage at the cell sample manifold every two hours. In this way, any localized variations in clean-up gas concentrations could be observed.

The Keithley Data Acquisition System was connected to the existing pneumatic pressure instrumentation and temperature indicator at the rear of the test cell cubicle. This allowed the pressure and temperature of the clean-up gases to be automatically monitored and recorded using a PC.

In addition to the clean-up gas analysis data obtained using the in-place instruments, gas samples were also obtained from the test cell periodically for infrared and chromatographic analyses by the PORTS X-710 Laboratory. This data served as a check on the in-place instruments. Gas samples were obtained at a high frequency early in the exposure period at elapsed times of 0, 2, 4, 8, 24, 48 and 96 hours. Thereafter, two samples per week were obtained. Laboratory analyses were performed for UF_6 , ClF_3 , ClO_2F , F_2 and Cl_2 .

Gas samples were also obtained at the end of the exposure period from the X-326 Building surge drums which were used to remove the clean-up gases from the test cell. These were analyzed in a manner identical to the cell samples. In addition, a special gas sample was obtained at this location for determination of the U-235 assay of the recovered UF_6 .

NDA Measurements

Nondestructive Assay (NDA) measurements were performed on the test cell to estimate the quantity and location of uranium materials. Quantitative neutron measurements were obtained on each of the major components of the test cell (compressors, converters and gas coolers) before the start of the gas exposure period, after about 70 days exposure and after the exposure period duration of 105 days. Internal gamma scans (obtained inside the cell housing) of the cell components and piping were obtained before and after the exposure period to locate areas with high uranium materials concentrations.

Test Cell Metal Samples and Coupons

Before clean-up gases were added to the cell, samples of barrier from six different barrier tubes were removed from the stage 9 converter of the test cell by cutting a 6" x 10" access hole in the side of the converter. Four wipe samples of the internal nickel-lined steel surface of the converter were also obtained via the access hole. The removed portion of converter shell was replaced with a nickel-lined steel overlay patch which was welded in place after the samples were removed. Six 1" long specimens of barrier tube were sectioned from each of the samples and were submitted along with the wipe samples for radiochemistry analyses for all radionuclides present. After the exposure period was completed, the overlay patch was removed and six more barrier tubes were sampled and four follow-up wipe samples were obtained and submitted for radiochemistry analyses.

Twelve 1" x 1" nickel-lined steel specimens were prepared from the removed portion of the stage 9 converter shell using sawing techniques to minimize the amount of heat present and preserve sample integrity. Six of these specimens were submitted for radiochemistry analyses and the remaining six were placed on a nickel coupon fixture which was constructed for the demonstration. The coupon fixture allowed various metal coupons to be mounted inside the spool piece of the stage 8 compressor discharge expansion joint of the test cell for exposure to clean-up gases during the exposure period (see sketches, Appendix B). Other samples installed on the coupon fixture included 5/8" copper seal exhaust line specimens from the K-25 cascade, aluminum cooler fin

material from a failed PORTS X-326 process gas cooler previously located at X-27-3-13 stage 3 and contaminated bare steel obtained from the PORTS X-705 Decontamination Facility. For each of these materials, six specimens were placed on the fixture and six duplicates underwent baseline radiochemistry analyses. The seal exhaust line specimens obtained from K-25 personnel contained heavy UO_2F_2 deposition and were obtained from a section of seal exhaust piping adjacent to the location where samples were removed for laboratory-scale testing.

Post-exposure period samples were also obtained from the 1/4" copper instrument line of the stage 9 compressor discharge and the cast aluminum rotors of the stage 11 compressor of the test cell and analyzed for all radionuclides present. The stage 11 compressor was chosen because preliminary neutron measurements indicated that it contained more U-235 prior to exposure than the other eleven compressors of the cell. In addition, since this compressor was located at the middle of the top badger cluster of the cell, it is a two stage compressor (two inlets). Therefore, any variation in contamination level with compression stage could be noted.

Removal of the cast aluminum rotor samples from the stage 11 compressor required the compressor to be removed from the test cell and disassembled in the X-705 Decontamination Facility. As with the other samples and coupons, only sawing techniques were used to minimize the amount of heat present and preserve sample integrity. While the compressor was located in X-705, special neutron measurements were conducted to estimate the total U-235 present after exposure to clean-up gases.

RESULTS

Composition of the Clean-Up Gases

Trend plots of UF_6 and ClF_3 concentration in the test cell versus days of exposure as determined by the in-place FTIR and PORTS X-710 Laboratory analyses of gas samples are shown in Figure 5 on page 13 and Figure 6 on page 14 respectively.

Each of the data points shown in Figure 5 is the average of concentrations obtained by X-326 Process Operators every two hours over five days of exposure. Treatment of the data in this manner removed fluctuations introduced by random errors and depicts longer-term trends. It is clear that the rate of increase in UF_6 concentration and the rate of decrease in ClF_3 concentration were higher early in the exposure period and became linear after about 15 days of exposure.

Immediately after the clean-up gases were added to the cell, a UF_6 concentration of approximately 0.5 mole % was measured with the in-place FTIR. It is believed that this was due to residual UF_6 contained in the header which was used to add clean-up gases to the test cell from the previous treatment of cell X-27-2-18. This is supported by U-235 assay determinations which were performed on gas samples obtained on the second day of exposure and at the end of the 105 day exposure period. An assay of approximately 17% U-235 was determined from the day 2 sample which is close to the historical assay of cell X-27-2-18 (15% U-235). An assay of 49.5% U-235 was determined from the day 105 sample which is intermediate between the historical assays of cell X-27-2-18 and test cell X-25-4-2 (88% U-235).

The plot of laboratory analyses data in Figure 6 shows lower ClF_3 concentrations and a slightly higher rate of ClF_3 consumption compared to the FTIR data. The UF_6 plot shows higher UF_6 concentrations and a lower rate of UF_6 recovery. The difference between the ClF_3 concentrations observed may be due to a bias introduced into the in-place FTIR data as a result of using an adjacent minor absorbance peak of ClF_3 in the infrared spectrum rather than the fundamental peak located at 701 wavenumbers. This was necessary because the high partial pressure of ClF_3 present in the clean-up gases resulted in an off-scale peak at 701 wavenumbers which would introduce excessive relative error. This occurred even with a special short path length sample cell installed on the in-place FTIR. However, the lower total pressure of the gas samples permitted use of the fundamental absorbance peak in the laboratory analyses.

Figure 5. Test Cell UF6 & ClF3 Concen.
FTIR at Cell X-25-4-2

Figure 6. Test Cell UF6 & ClF3 Concen.
X-710 Lab Analyses of Can Samples

A clearer trend plot of UF_6 concentration determined by in-place FTIR versus days of exposure is shown in Figure 7 on page 16. The low-rate linear increase in UF_6 concentration is probably more indicative of the slow diffusion of UF_6 from reaction sites into the bulk gas during the exposure period than the actual rate of UF_6 recovery at the contaminated sites. This is supported by a comparison of UF_6 recovery estimates determined from analyses of gas samples taken from the test cell at the end of the exposure period and analyses of gas samples from the X-326 surge drums immediately after evacuating the clean-up gases from the test cell. Based on the test cell data, UF_6 recovery of 7.4 pounds is estimated. Based on the drum data, UF_6 recovery of 11.8 pounds is estimated. When the cell contents were removed to the surge drums using differential pressure, mixing occurred which resulted in a more representative measurement of the UF_6 content of the cell.

The test cell gas composition at the end of the exposure period based on data from both the in-place instruments and laboratory analyses is shown below in Table 2. F_2 concentration data obtained from the CGC is not shown because this would reveal the relative concentrations (recipe) of ClF_3 and F_2 . This recipe is the same as that currently used in running cell clean-up treatments and is classified as confidential-restricted data. Consumption of F_2 was observed during the exposure period. This, coupled with the absence of significant measured quantities of ClF , may indicate that F_2 may have played a role in the clean-up by regenerating ClF_3 via the recombination reaction below:

However, F_2 consumption may have occurred due to reaction with test cell metal surfaces (fluoride film growth) and/or conversion of UF_4 and UF_5 to UF_6 . Insufficient data was obtained to determine whether the recombination reaction above indeed occurred.

Table 2. Clean-Up Gas Composition (105 days exposure)

Compound	Concentration (Mole %)
ClF_3	11.3
UF_6	1.1
ClO_2F	1.1
Cl_2	0.5
ClF	ND
SiF_4	1.9
CF_4	0.7

Figure 7. Test Cell UF6 Concentration
FTIR at Cell X-25-4-2

The Cl_2 concentration measured with the in-place CGC during the exposure period was generally less than 1 mole %. The measured concentration intermittently rose to slightly higher than 1 mole % but a continuing upward trend was not noted at any time during the exposure period.

A mass balance of the clean-up gases indicated that the ratio of unaccounted fluorine to unaccounted chlorine near the end of the exposure period was approximately equal to three. This observation would tend to indicate that the fluorine and chlorine deficits were due to ClF_3 , which was not present in the bulk gas, possibly due to adsorption on test cell internal surfaces, particularly barrier.

The compounds SiF_4 and CF_4 were observed immediately after the clean-up gases were added to the test cell at the beginning of the exposure period. It is believed that these compounds were residual materials present in the new conditioning gas (oxidant) header which was used to add clean-up gases to the test cell. Test cell X-25-4-2 was the first cell to receive clean-up gases from the new header after it was placed in service. SiF_4 has been observed in the past when oxidants come into contact with fresh piping welds and CF_4 is a common reaction product observed when oxidants react with hydrocarbons. The belief that these compounds had their origins in the new header and not the test cell is supported by the fact that their concentrations remained unchanged from those observed initially for the duration of the exposure period.

Test Cell Temperatures and Pressures

The test cell temperatures observed with the Keithley data acquisition system generally tracked X-326 Building ambient temperatures and varied from about 70 to 85 degrees F. No indications of localized high temperature areas in the test cell were observed. Localized elevated temperatures are possible with the expected heat of reaction during clean-up if sufficient uranium contamination is present in a given area. Apparently, the quantity of uranium contamination was not high enough in any given area of the cell to result in elevated temperatures due to reaction with clean-up gases.

During the first five days of the exposure period, the test cell pressure slowly fell from the target pressure of 14.0 psia to about 13.3 psia. The cause of the pressure loss was identified as leakage through some of the individual stage seal exhaust isolation valves which were closed prior to admitting clean-up gases to the cell (the seal exhaust system is maintained at a pressure much less than the target test pressure). The problem was remedied by closing the main seal exhaust isolation valve for the cell. Thereafter, measured cell pressures fluctuated with ambient and cell temperature changes, primarily because of the effect of

temperature change on cell pressure measurement instrumentation. A slow upward trend was observed and average pressures near the end of the exposure period were close to the target of 14.0 psia. The upward trend may have been due to the stoichiometry of the clean-up reactions which was expected to increase the number of moles of gas present in the cell. It was not necessary to remove clean-up gases from the cell at any time during the exposure period to maintain test pressures below atmospheric pressure. However, if the leakage through the stage seal exhaust valves described above did not occur, it would have been necessary to remove approximately 0.3 psi from the cell to maintain pressures less than the high limit of 14.4 psia which was established for the test.

NDA Findings

Estimates of the quantities of U-235 contained in the test cell compressors, converters and gas coolers based on neutron measurements are given in Figure 8 on page 20, Figure 9 on page 21 and Figure 10 on page 22 respectively. These measurements were performed on the test cell before exposure to clean-up gases, after 70 days exposure and after 105 days exposure.

Before exposure, the test cell was estimated to contain a total of approximately 3.3 Kg U-235 (12.3 lbs. UF_6 recoverable) based on these measurements. At 70 days exposure, the cell contained the gas-phase UF_6 which had been recovered from cell surfaces plus solid uranium materials remaining on the surfaces so the estimate of cell total U-235 is close to the before exposure estimate (3.5 Kg U-235 or 12.9 lbs. as UF_6). At 105 days exposure, the clean-up gases containing the recovered UF_6 were removed from the cell and all that remained was the residual solid uranium materials on the internal cell surfaces. At this point, the neutron measurements indicated that the cell contained approximately 1.7 Kg U-235 (6.5 lbs. as UF_6) and that approximately 48% of the uranium material had been removed from the cell.

Figures 8, 9 and 10 indicate that generation of UF_6 in the test cell resulted in elevated converter mass estimates and reduced compressor and gas cooler estimates. Examining this shift in uranium mass and the test cell UF_6 concentration at 70 days exposure, it was determined that at least 1 Kg and as much as 1.6 Kg of U-235 had been converted from solid contamination to gaseous UF_6 by 70 days exposure. Therefore, most of the clean-up of cell surfaces occurred during the first 70 days of exposure. The economically optimum exposure period length will be dependent on equipment location in the diffusion cascades because of the relationship between historical uranium isotope assay and cascade location (see Economics of Full-Scale Implementation).

Internal gamma scans of the test components and piping obtained inside the test cell housing before and after exposure to clean-up gases are shown in Appendix C. The major components of the test cell showed a significant reduction in gamma activity due to the treatment. However, some inaccessible areas (vacuum breaks lines and expansion joints) remained close to pre-exposure levels.

Special neutron measurements were obtained on the stage 11 centrifugal compressor in the PORTS X-705 Decontamination Facility after removal of post-exposure period aluminum rotor samples (see Appendix D). The compressor was estimated to contain approximately 77 grams of U-235 which compares to 111 grams of U-235 estimated before removal from the test cell. The observed reduction in U-235 mass estimated is probably due to the fact that adjacent components and piping contributed to the in-cell measurements and disassembly and removal of the compressor seals was performed in X-705 prior to the special measurements.

Figure 8. Test Cell Compressors
Neutron Measurement Estimates

Figure 9. Test Cell Converters
Neutron Measurement Estimates

Figure 10. Test Cell Gas Coolers Neutron Measurement Estimates

Radiochemistry Findings

Radiochemistry analyses were performed on all before and after exposure samples and coupons by either leaching or dissolving each specimen in nitric acid to remove essentially all radiological material present. Since the specimen contaminated areas were small, less than the area of a typical detection instrument, they were considered as point sources and the total specimen activities were compared to free release guidance contained in USAEC Regulatory Guide 1.86.

Use of radiochemistry analyses in this manner for comparison to the acceptable surface contamination levels given in USAEC Guide 1.86 is a very conservative approach. Use of a detection instrument would result in much lower measured activities for a given specimen because of the effects of nuclide self-shielding and the porosity of the specimen surface. In addition, detection instruments obviously do not provide discrimination of individual nuclides which is needed for a thorough evaluation of LTLT process effectiveness and to provide data which can be extrapolated to apply to any diffusion cascade equipment location. In an actual D&D effort, surface detection instruments would be employed by health physics technicians for surface contamination measurement. However, such measurements provide insufficient information for purposes of this evaluation.

Reduction of radiochemistry results for comparison to free release guidance for each nuclide grouping is given in Appendix E. The observed nuclides were grouped according to information contained in USAEC Guide 1.86 as shown in Table 3 below. The measured picocuries per gram for each nuclide group for each specimen was converted to sample activity in disintegrations per minute. This sample activity was then compared to the acceptable surface contamination levels by determining the ratio of sample activity to

Table 3. Observed Nuclide Grouping for Comparison to USAEC Regulatory Guide 1.86

Nuclide Group	Radiation Type	Nuclides
Uranium Isotopes	Alpha	U-234, U-235, U-238
Transuranics	Alpha	Am-241, Np-237, Pu-238, Pu-239/240, Th-228, Th-230
Thorium	Alpha	Th-232
Beta-Gamma Emitters	Beta-Gamma	Tc-99, Th-231, Th-234

the average acceptable levels specified in the guide. The more rigorous average levels were used rather than the maximum levels specified in the guide because variation in contamination level over specimen surface was not determined.

Bar graphs depicting a comparison of sample alpha contamination from uranium isotopes to free release guidance for each material tested are given in Appendix F. As can be seen, the LTLT treatment resulted in a large reduction in uranium alpha contamination for all of the copper seal exhaust line specimens which contained a heavy UO_2F_2 deposit prior to exposure to clean-up gases. However, enough contamination remained that post-treatment sample activity ranged from slightly above to approximately twice acceptable levels. Barrier sample activity was reduced to below guidance levels for three of the six post-treatment samples. One post-treatment barrier sample (LTB-8) remained at close to pre-treatment levels. This sample is believed to have been removed from a barrier tube which was manually plugged based on leak test results. This was a common practice during converter assembly and preparation activities. This would have restricted the amount of clean-up gas present at the contaminated surface of the tube. Wipe samples obtained on the nickel-lined steel of the stage 9 converter shell were all well within removable contamination guidelines. These data are inconclusive since the areas above and below the converter access hole show an increase in activity after exposure and those left and right of the access hole show a decrease and essentially no change respectively. The copper instrument line samples which were removed after exposure were all well below guidance levels. While the pre-exposure aluminum cooler fin samples were well within acceptable levels, the post-exposure samples had slightly lower activities as a result of the clean-up. The cast aluminum rotor samples removed from the stage 11 compressor had activities ranging from about five to 14 times acceptable levels.

The pre-exposure nickel-lined steel samples from the stage 9 converter shell had activities well below guidance levels but curiously the post-exposure samples showed activities from slightly above to nearly three times acceptable levels. This anomaly is believed to be due to exposure of unlined (bare steel) sample surfaces to UF_6 contained in the clean-up gases. This is supported by an examination of K-25 data from laboratory testing of the LTLT process. These data, which indicated a decrease in both alpha and beta/gamma emissions as a result of LTLT treatment, were obtained using a detection instrument to measure activity on the contaminated surface of the samples only. The radiochemistry analyses used in the PORTS demonstration involved immersing the samples in nitric acid to remove essentially all contamination present on all surfaces of the specimens. While a significant reduction in activity was observed for the bare steel samples due to the LTLT treatment used in the PORTS demonstration, the post-exposure sample activities ranged from about two to ten times acceptable levels.

Average total uranium removal from the LTLT treatment was calculated for each material by converting uranium isotope activities before and after exposure to uranium weight in micrograms using known isotope specific activities. Percent uranium removal for each material is shown in Table 4 below. In order to determine the effectiveness that LTLT treatment would have on cascade equipment at various locations in the diffusion cascades, the average post-exposure uranium weight remaining for each material was extrapolated to determine alpha activity over the range of U-235 assays which were historically present in the diffusion cascades. Of the uranium isotopes observed, U-234 has by far the highest specific activity and is therefore responsible for most of the total uranium alpha activity observed on contaminated equipment surfaces. In the diffusion cascades, U-234 was enriched along with U-235 so higher concentrations of U-234 exist in uranium contamination in cascade equipment which historically contained high U-235 assays. The relationship between U-234 and U-235 assay based on representative gradient information for the PORTS cascade is shown in Figure 11 on page 26. The relationship between alpha contamination and diffusion cascade U-235 assay was derived as follows:

Total uranium alpha activity was assumed to be due to U-234, U-235 and U-238. Other uranium isotopes either have too low a specific activity or too low a concentration in the diffusion cascades to be of significance. Therefore:

$$d_{\text{tot}} = d_{234} + d_{235} + d_{238} \quad (4)$$

where d is disintegrations per minute (dpm)

Table 4. Total Uranium Removal with LTLT Treatment

Sample Material	Ave. Total U/sample		Ave. Removal (%)
	Before ug	After ug	
5/8" Copper Seal Exhaust	2007292	207	99.99
Bare Steel	7032	1033	85.31
Diffusion Barrier	318	58	81.76
Al Cooler Fins (Sheet)	26	23	11.54
Ni-Lined Steel	7	207	-2857.14
1/4" Copper Instr. Line	N/A	3	N/A
Al Compr. Rotor (Cast)	N/A	420	N/A

Figure 11. U-234 vs. U-235 Assay
Historical PORTS Diffusion Cascade

Based on representative gradient

Individual uranium isotope activity is as follows:

$$d_{234} = 13870 * W_{234} \quad (5)$$

$$d_{235} = 4.797 * W_{235} \quad (6)$$

$$d_{238} = 0.746 * W_{238} \quad (7)$$

where W is isotope weight (ug)

However, isotope weight can be expressed in terms of mole fraction of total uranium weight as follows:

$$W_{234} = M_{234} * W_u \quad (8)$$

$$W_{235} = M_{235} * W_u \quad (9)$$

$$W_{238} = M_{238} * W_u \quad (10)$$

where: M_n is the mole fraction of each isotope
($n=234, 235$ or 238)

W_u is the total uranium weight remaining
on the post-exposure samples

Substituting equations (5) through (10) into equation (4) gives:

$$d_{tot} = 13870 * M_{234} * W_u + 4.797 * M_{235} * W_u + 0.746 * M_{238} * W_u \quad (11)$$

The average total uranium weight remaining on the post-exposure samples and representative diffusion cascade multicomponent gradient data were used in the above equation to determine the relationship between uranium alpha contamination and diffusion cascade U-235 assay for each material. A family of curves was produced which depicts these relationships and is shown in Figure 12 on page 28.

Given the conservative nature of comparing radiochemistry analyses data to acceptable surface contamination levels that was explained earlier, these curves provide equipment assays below which acceptable surface contamination levels can be obtained for each material using an LTLT treatment with the same conditions as this evaluation. These assays, which are at the intersections of the material curves and the USAEC guidance line are summarized in Table 5 on page 29.

Bare steel was the most difficult material to clean to below acceptable surface contamination levels. However, bare steel is not a normal diffusion cascade material of construction but results from failure of nickel-lined steel surfaces. The failed areas are

Figure 12. Alpha Contamination vs. Assay
Extrapolation of Post-Exposure Data

Based on representative gradient

small and isolated compared to the huge surface area of nickel-lined steel with good integrity present in the diffusion cascades. For these reasons, exposed bare steel areas could be handled on a case-by-case basis in a full-scale D&D effort. Cast aluminum was the second most difficult material to clean. Major components of diffusion cascade compressors including rotors in the centrifugal equipment and rotors, stators and blades in the axial equipment are constructed of cast aluminum. However, the data extrapolation indicates that acceptable surface contamination levels can be obtained below about 9% U-235 assay. The bulk of the equipment in the diffusion complex including all of the PGDP cascade and the lower PORTS and K-25 cascades have historical assays less than 9%. Since acceptable surface contamination levels can be obtained for the other materials tested at even higher historical assays, it can be concluded that an LTLT treatment similar to the one evaluated here would be an effective technique for in situ decontamination of the bulk of the equipment in the diffusion complex with the exceptions of isolated areas in the low assay cascades and the PORTS and K-25 high assay cascades.

D&D of the high assay cascades may also present special problems associated with other radiological contaminants. Other alpha emitting nuclides observed in the radiochemistry analyses were the transuranics Am-241, Np-237, Pu-238, Pu-239/240, Th-228 and Th-230 and Th-232. However, the uranium isotopes were responsible for the bulk of the alpha contamination observed on the post-exposure samples and the other nuclides contributed only a small percentage as shown in Table 6 on page 30. In addition, the LTLT treatment resulted in a significant reduction in transuranic alpha activity on the copper seal exhaust and barrier samples even though

Table 5. Diffusion Cascade Assays Below Which
Free Release is Obtained.

Sample Material	Intersection Assays (% U-235)
Bare Steel	3.5
Al Compr. Rotor (Cast)	9.3
Ni-Lined Steel	20.9
5/8" Seal Exhaust from K-25	20.8
Barrier	76.1
Al Cooler Fin (Sheet)	97+
1/4" Copper Instr. Line	97+

transuranic materials would not be expected to readily form volatile fluorides at the conditions employed. However, it is difficult to predict the location and concentration of these nuclides and the effect of LTLT treatment on their removal.

Beta-gamma emitting nuclides included in the radiochemistry analyses were Tc-99, Th-231 and Th-234. In general, an increase in beta activity was observed in post-exposure samples which did not originate from the test cell due to the presence of Tc-99 in the test cell. Some decrease in Tc-99 activity occurred on the barrier samples which were taken from the stage 9 converter of the test cell. However, nickel-lined steel removed from the stage 9 converter shell showed a significant increase in Tc-99 activity. Analogous to the observed increase in uranium alpha activity, this is believed to have been due to exposure of the unlined (bare steel) surfaces of the coupons to clean-up gases containing Tc-99. As with the non-uranium alpha emitting nuclides above, the location and concentration of Tc-99 in the high assay cascades and the effects of LTLT treatment, which was not intended for its removal, are difficult to predict.

Th-231 and Th-234 emissions were calculated from the U-235 and U-238 emissions respectively. The metastable Pa-234 daughter of U-238, which would have emissions approximately equivalent to Th-234, was not included in the beta/gamma grouping. Tc-99 accounted for greater than 90% of the beta/gamma activity in most of the sample matrices analyzed with the U-238 daughters usually playing a much smaller role. This was expected since most of the sample matrices had high U-235 assay uranium contamination and exhibited significant Tc-99 contamination. However, U-238 daughters will be present in significant quantities in older aged deposits in the low assay cascades and in UF₆ cylinder heels.

Table 6. Contribution of Non-Uranium Nuclides to Total Alpha Activity of Post-Exposure Samples

Material	Uranium Alpha (%)	Other Alpha (%)
5/8" Seal Exhaust from K-25	99.5	0.5
Barrier from S/9 Converter	97.5	2.5
Wipes from S/9 Converter Shell	99.0	1.0
1/4" Copper Instr. Line	78.8	21.2
Aluminum Cooler Fin	95.7	4.3
Ni-Lined Steel, S/9 Converter	99.3	0.7
Bare Steel, X-705 Scrap	99.9	0.1
Aluminum Rotor, S/11 Compressor	98.3	1.7

ECONOMICS OF FULL-SCALE IMPLEMENTATION

Methodology

As discussed earlier, the existing technology for uranium removal from diffusion cascade equipment consists of disassembly followed by wet decontamination. For this reason, costs associated with full-scale D&D of PORTS and PGDP using the existing technology and LTLT in situ decontamination followed by equipment disassembly are estimated and compared. The cost estimates are based on decontamination experience at PORTS and are then extrapolated to PGDP. K-25 has no operable cell service capability (evacuation booster stations, surge drum systems, etc.) and the cascade equipment at this site has been segmented by the removal of components for use at the operating sites. Therefore, the cost of a D&D effort at K-25 is not comparable to D&D of the operating sites and K-25 was excluded from the cost analyses presented here. While it is clear that a D&D effort at PORTS and PGDP using the existing technology would require that existing wet decontamination facilities be upgraded and that new facilities be constructed, no attempt was made to address the capital costs associated with such an effort.

Manpower Cost Comparison

Prior to implementation of a D&D program for diffusion cascade equipment using either existing or LTLT technology, it is assumed that a gross deposit removal program would be pursued to remove large solid uranium deposits from the equipment for NCS purposes. For disassembly followed by wet decontamination, this would permit safe disassembly of the equipment. For LTLT decontamination, this would prevent rapid depletion of clean-up gases by reaction with large deposits. At the currently operating diffusion sites (PORTS and PGDP) this could be accomplished using existing running cell treatment procedures. Since similar gross deposit removal programs would be conducted prior to any D&D effort, the costs associated with such an effort would be comparable regardless of the D&D technology chosen.

The radiochemistry findings indicated that acceptable surface contamination levels could be obtained for all major materials of construction in the low assay cascades of the diffusion complex. At PORTS and PGDP, this equipment would include all of the 33 ("000"), 31 ("00") and 29 ("0") size equipment. Also included would be the 2X purge cascade equipment at PGDP which employs "0" size converters and centrifugal compressors. Manhour costs associated with performing D&D using the existing technology based on experience at PORTS is shown in Table 7 on page 32. D&D of a diffusion cell using the existing technology would consist of field disassembly of the cell in the process buildings followed by

Table 7. Estimated manhour cost of diffusion cell D&D using existing wet decontamination technology.

Equipment Type	Manhours			Total Manhours	Total Cost* (\$K/cell)
	Field Disassembly	Component Disassembly	Wet Decon.		
29 ("0")	2454	1289	5156	8899	667
31 ("00")	2496	1310	5241	9047	679
33 ("000")	2070	1060	4240	7370	553

* Assuming \$75/hr.

individual component disassembly in a decontamination facility and finally wet decontamination. This cost estimate assumed an hourly rate of \$75/hr.

A similar estimate of manhour costs was prepared for D&D of each of the cell types using LTLT decontamination rather than wet decontamination. The D&D sequence would then be to render the cell leak-tight (Dresser seals), perform LTLT in situ decontamination followed by field disassembly and finally disassembly of the cell components. Since decontamination is performed prior to disassembly in this sequence, it was assumed that field and component disassembly manhours would be reduced by 25%. The manhour cost estimate is shown below in Table 8. As with the existing technology estimate, an hourly rate of \$75/hr was assumed to determine the manhour costs. The NDA measurements performed on the test cell indicated that at least 60% of the uranium removal occurred in the first 70 days of exposure. Therefore, the manhours for LTLT decontamination of 31 and 33 size cells, which have lower historical assays than 29 size cells, were based on assuming an LTLT treatment with a 70 day duration. A 105 day duration was assumed for the higher assay 29 size cells.

Table 8. Estimated manhour cost of diffusion cell D&D using LTLT in situ decontamination

Equipment Type	Manhours				Total Manhours	Total Cost* (\$K/cell)
	Dresser Seal Installation	LTLT Decon.	Field Disassembly	Component Disassembly		
29 ("0")	108	635	1840	967	3550	266
31 ("00")	108	416	1872	982	3378	253
33 ("000")	88	416	1553	795	2852	214

*Assuming \$75/hr.

LTLT Cost Savings

The LTLT process converts solid uranium contamination to UF_6 gas which can be returned to an operating diffusion cascade for eventual withdrawal as LEU product. Wet decontamination has uranium oxide as an end product. Since no oxide conversion facility is currently in operation, this material must be stored indefinitely. Oxide material has some potential value to fuel fabricators and while a few vendors have expressed an interest in purchasing oxide, no market currently exists. For these reasons, there is a cost credit associated with the value of the recovered UF_6 from the LTLT process. In addition, storage costs associated with uranium oxide production can be avoided. These factors must be considered in addition to D&D cost in determining the cost savings of LTLT decontamination versus conventional wet decontamination.

It was assumed that the gross deposit removal effort conducted prior to D&D would result in averages of 10, 15 and 30 Kgs of uranium remaining in 29, 31 and 33 size diffusion cells respectively. It was also conservatively assumed that LTLT treatment would result in 70% removal of this material. The value of enriched uranium varies with U-235 assay as shown in Figure 13 on page 34. Therefore, average historical assays of 3.1, 1.5 and 1.1 were assumed for 29, 31 and 33 size cells respectively. These were used in conjunction with the assumed uranium recoveries, the value of the uranium shown in Figure 13 and the value of the normal material (unenriched feed stock) used to produce the enriched uranium to determine the cost credit associated with the recovered UF_6 . An estimate of the total cost savings of D&D using LTLT rather than wet decontamination for each cell type is shown in Table 9 below.

Table 9. Estimated cost savings per cell of D&D using LTLT compared to wet decontamination.

Equipment Type	D&D Savings (\$K/cell)	UF6 Value (\$K/cell)	Total Savings (\$K/cell)
29 ("0")	401	2.9	404
31 ("00")	426	1.6	428
33 ("000")	339	1.9	341

Figure 13. Value of Recovered Uranium
\$/KgU vs. Assay

* Assumes a value of \$50/SWU

Costs associated with low assay uranium oxide storage at the PORTS site include the manhour cost of maintaining two full-time Uranium Material Handlers (UMH) and costs associated with the storage facility. The DOE currently bears the annual facility cost which is estimated at \$109K/yr. USEC provides the UMH personnel at an annual manhour cost of approximately \$200K/yr. Therefore, the current total low assay oxide storage costs at PORTS are in excess of \$300K annually. These cost would increase significantly if a full-scale D&D effort using existing technology were pursued resulting in a large increase in the amount of oxide to be stored (assuming that no market existed for oxide at the time).

PORTS and PGDP Site Savings

The LTLT cost savings per cell for each equipment type which were based on decontamination experience at PORTS were used to determine the potential cost savings for the PORTS and PGDP sites. Because of similarities with 29 size equipment, PGDP 2X cells were assumed to have the same cost savings per cell as 29 size cells. As mentioned earlier, no attempt was made to determine costs associated with construction of new wet decontamination facilities and upgrading of existing facilities which would clearly be required for a full-scale D&D effort to be conducted. The estimated cost savings for D&D efforts at PORTS and PGDP using LTLT technology rather than existing technology are shown in Table 10 below. Savings of approximately \$73MM and \$79MM were estimated for the PORTS and PGDP sites respectively.

D&D Program Duration and Capital Equipment Costs

The existing PORTS X-705 Decontamination Facility has limited capacity and component disassembly and wet decontamination could be completed on only approximately four diffusion cells per year. This would result in an unreasonable D&D program duration and clearly indicates the need for upgraded or new decontamination facilities at significant capital expenditure if the existing technology is chosen for a full-scale D&D effort.

Table 10. Estimated cost savings for D&D of the low assay cascades at PORTS and PGDP.

Site	Equipment Type	No. Cells	Cost Savings (\$K/cell)	Cost Savings (\$K/equip. type)	Site Savings (\$K/site)
PORTS	29 ("0")	60	404	24,240	72,920
	31 ("00")	50	428	21,400	
	33 ("000")	80	341	27,280	
PGDP	2X (purge)	10	404	4,040	79,200
	31 ("00")	80	428	34,240	
	33 ("000")	120	341	40,920	

The initial capital cost associated with a D&D effort using the LTLT process would be dependent on the desired program duration. A shorter duration program would require a large number of diffusion cells to be decontaminated simultaneously and therefore a greater quantity of cell preparation and analytical equipment (FTIRs and accessories, Dresser seals) would need to be procured. A longer duration program would allow less cells to be decontaminated simultaneously and a smaller initial capital investment. As an example, D&D of the low assay cascade at PORTS could be completed in an approximately two year program with a staff of 27 Process Operators and an initial capital expenditure of approximately \$2MM. An approximately four year program could be conducted by a staff of 16 Process Operators with an initial capital expenditure of approximately \$1MM. A similar program could be planned for PGDP with comparable initial capital costs.

CONCLUSIONS

1. The LTLT decontamination process can be conducted safely on diffusion cascade cells that do not contain large gross uranium deposits without generation of significant quantities of free Cl_2 , other unwanted compounds including thermally unstable ClO_2 , and without localized high temperature areas.
2. Installation of Dresser seals is an effective means of rendering diffusion cells leak-tight in preparation for LTLT treatment.
3. Test cell NDA measurements indicate that the majority of the uranium contamination was removed in the first 70 days of exposure to clean-up gases.
4. Analyses performed on gas samples removed from the PORTS X-326 Building surge drums after evacuating the clean-up gases from the test cell at the end of the exposure period indicate that approximately 11.8 pounds of UF_6 was recovered (or 96% of the uranium material originally contained in the cell based on NDA measurements) by the LTLT treatment.
5. Radiochemistry analyses of the metal samples and coupons indicate that significant removal of uranium contamination was achieved for copper, bare steel, diffusion barrier and sheet aluminum and that the most difficult materials to clean to below acceptable surface contamination levels were bare steel and cast aluminum.
6. An extrapolation of the radiochemistry data indicates that acceptable surface contamination levels can be obtained on all materials tested (with the exception of bare steel) in diffusion equipment with historical assays less than approximately 9% U-235. This includes the bulk of the equipment in the diffusion complex.
7. Significant cost savings over the existing wet decontamination technology (in excess of \$150MM) could be realized in a full-scale D&D program of the low assay cascades of PORTS and PGDP.
8. A full-scale D&D program using LTLT technology could be completed in a reasonable amount of time with minimal capital expenditure whereas the time required to complete a similar D&D program using existing wet decontamination technology would be prohibitive without significant capital expenditure for upgrade of existing and construction of new wet decontamination facilities.

RECOMMENDATION

Because of the effectiveness, favorable economics and simplicity of the LTLT decontamination technology compared to the existing wet decontamination technology, LTLT technology should be given consideration in DOE planning for implementation as part of future D&D programs at the gaseous diffusion sites.

ACKNOWLEDGEMENTS

The author wishes to thank Clare Welch, formerly of LMES/PORTS for her extensive efforts which included seeking and obtaining funding for the project, preparation of the first Technical Task Plan and numerous project management activities. Without her devoted efforts, this demonstration would not have taken place.

Special thanks are also due to Dick Schwab of LMUS/PORTS for his tireless efforts in providing technical expertise for the operation of test cell analytical instruments and interpretation of test data, Dave Netzer of LMUS/PORTS who, through his insight and wisdom, conceived the use of the shutdown X-326 cascade as a test bed for gas-phase D&D, Jim Morgan, LMUS/PORTS acting General Manager, who as LMUS/PORTS Enrichment Plant Manager recognized the value of this work and supported it financially and administratively as well as morally, Keith Kibbe, formerly D&D Integrated Demonstration Coordinator with MMES/K-25, whose interest, support and administrative action was instrumental in making this work possible and Jerry Harness of DOE-ORO for his loyalty, assistance and continued commitment to supporting this work through many difficult times.

Thanks are also due to Mike Orlett of LMUS/PORTS for his work in preparing analytical instrumentation, Bob Konanz of LMUS/PORTS for providing assistance and technical expertise to initiating the full-scale test, Toby Knox of LMUS/PORTS for providing manhour estimates, Dr. Tony Saraceno and Dr. David Manuta of LMUS/PORTS for their efforts on the preliminary laboratory investigation study and for providing input on the demonstration proposal, Chuck Good of LMUS/PORTS for providing his expertise to the interpretation of radiochemistry data, Brent McGinnis, Dewey Cordle and other members of the LMUS/PORTS Applied Nuclear Technology staff for obtaining, analyzing and interpreting NDA data, Ken Whittle, Jack Tully and other members of the PORTS X-326 Operations staff for performing the full-scale test in a safe and professional manner, Jim Folsom formerly of PORTS X-326 Integrated Scheduling for coordinating planning and scheduling activities in a highly professional manner, Jeff Woodard, Mitch Helms and other members of the PORTS X-705 Operations staff, John Eckstein of X-705 Maintenance Planning and Max Potter and other members of the X-705 Mechanical Maintenance staff for handling the difficult issues related to disassembly and sampling of the test compressor, Jan Pruett formerly of LMES/K-25 OETP for highly professional administrative support and for acting as an effective liaison between LMUS and DOE, Robin O'Hatnick of LMER/ORNL OETP for providing invaluable technical and administrative assistance to project tracking, Earl Munday of LMES/K-25 for his input and insight on the LTLT process and Dr. Lee Trowbridge of LMER/ORNL, D.W. Simmons of LMES/K-25 and the late Dr. John Barber for their work on the preliminary laboratory investigations and input on the demonstration proposal.

REFERENCES

1. E.B. Munday and D.W. Simmons, "Feasibility of Gas-Phase Decontamination of Gaseous Diffusion Equipment," K/TCD-1048, Martin Marietta Energy Systems, Inc., Oak Ridge, TN, February, 1993.
2. "Termination of Operating Licenses for Nuclear Reactors," Regulatory Guide 1.86, USAEC Directorate of Regulatory Standards, June, 1974.
3. D.W. Simmons, "The Stability of ClO_2 as a Product of Gas-Phase Decontamination Treatments," K/TCD-1116, Martin Marietta Energy Systems, Inc., Oak Ridge, TN, September, 1994.
4. L.D. Trowbridge, " $\text{ClF}_3 + \text{H}_2\text{O}$ Thermodynamics and Kinetics," letter to R.T. Shelton, Martin Marietta Energy Systems, Inc., Oak Ridge, TN, July, 1993.
5. W.P. Holmes, "Unreviewed Safety Question Determination; X-326 Long-Term, Low-Temperature Treatment Testing," POEF-530-94-1089, Lockheed Martin Utility Services, Inc., Piketon, OH, February, 1996.

APPENDIX A

LTLT Test Sample Manifold

MATERIAL LIST			
ITEM	QUAN	DESCRIPTION	PART NO.
1	7	VALVE, 1/4" NPT, MAKE 455104H	77-862-3390
2	8	BODY, FLANGE & MOUNTING, 1/4"	77-892-3267
3	1	GLAND, BLIND, 1/4"	07-428-0038
4	8	NUT, GLAND, 1/4"	06-651-0238
5	6	TEE, 1/4", WROT COPPER	06-651-0248
6	AR	ALUMINUM PLATE, 1/8"	06-651-0270
7	AS	TUBING, COPPER, 1/4" OD x 0.030" WALL	07-244-1120
8	2	COMPRESSOR GAUGE, ASHROFT

DETAIL

**DALTON FITTING MOUNTING
BRACKET PROFILE**
SCALE 1" = 1"

DETAIL 2

DALTON FITTING
MOUNTING BRACKET
SCALE 1" = 1"

DRILLING ARRG'T 'A'

DRILLING ARRG'T 'B'

[illegible]

APPENDIX B

LTLT Test Coupon Fixture

TYPICAL CUPON
ASSEMBLY

TACK WELD

4"

TYPICAL ASSY. 8-PL.

B-1

1/4-20 UNC HEX NUT
MONEL
64 REQUIRED

TACK WELD

3 1/2"

3 1/2"

LTLT Test Fixture Assembly

EXISTING EXPANSION JOINT
@ STAGE 8" DISCHARGE PIPE
EXPANSION JOINT # X-10A

TACK WELD
TYP. 4-PL.

TACK WELD
TYP. 4-PL.

LTLT Test Fixture Installation

APPENDIX C

Test Cell Internal Gamma Scans

Analysts: McGinnis, Cordle

Cell X-25-4-2 Internal Gamma Scan

Date: 8/3/95

0.3 Gram U-235 Foil Check Source:

698K
709K
704K

Bkg: 6.4K

Bkg: 7K

Purge Valves:
Left Valve 32K
Center Valve 285K
Right Valve 140K

Bkg: 7K

Bkg: 6K

Analysts: McGinnis, Cordle

Cell X-25-4-2 Internal Gamma Scan

9.23.0
Date: 8/3/91

690K	698K
704K	709K
699K	704K

Bkg: 6.4K

BG: 5K

Bkg: 7K

BE: 6K

Purge Valves
Left Valve 32K 30K
Center Valve 285K 200K
Right Valve 140K 110K

Bkg: 7K

Bkg: 6K

BG: 4K

BG: 5K

APPENDIX D

Neutron Measurements on the Stage 11 Centrifugal Compressor

Memorandum

Date: July 9, 1997
POEF-38-344-97-334

To: R. Riddle, MS-2213

From: B. McGinnis, X-710, MS-2214, (5978) *BM*

Subject: NDA Measurements of X-25-4-2 S/11 Compressor and Associated Containers

Nondestructive assay (NDA) measurements were performed on the stage 11 compressor of cell X-25-4-2 while it was located in the X-705 building South Annex. A total of four independent measurements were performed using a neutron slab detector. The neutron measurements performed estimate that the compressor contains 77 ± 40 grams of ^{235}U or 0.37 ± 0.19 lbs. of UO_2F_2 . This estimate is based on the following:

- 1) a measured ^{234}U concentration of $1.23 \pm 0.74\%$ by weight.
- 2) the chemical form of the deposit is assumed to be UO_2F_2 .
- 3) the weight percent ^{235}U was measured at $59 \pm 3.6\%$.

The ^{234}U concentration and weight percent ^{235}U were determined by performing an NDA measurement of a sample of material removed from the compressor. A U-Pu Inspector spectroscopy system was used to perform the measurement.

Additional NDA measurements were also performed on two GP Polybottles located in the South Annex. The results are listed in the following table:

GP Container No.	^{235}U (Grams)	Enrichment
GP009997	1.8 ± 1.1	$43.1 \pm 7.8\%$
GP009998	0.6 ± 0.4	$59.0 \pm 3.6\%$
Sample Coupons	1 ± 1	$59.0 \pm 3.6\%$

BRM

cc:	R. Coriell	MS-4025	P. Potter	MS-9030
	S. Brown	MS-1106	D. Shisler	MS-1112
	M. Helms	MS-3020	J. Tully	MS-4004
	M. Kelly	MS-1232	J. Woodard	MS-3020
	D/38-344 File - RC			

Special Equipment NDA Measurements Worksheet

Equipment Type:	Centrifugal Compressor	Measured by:	TT, JGR
Equipment Name:	Type 9B	Measurement Date:	7/9/97
Serial or ID #:	9B-0354L	Detector Type(s):	Neutron Slab
Origin:	X-25-4-2 S/11		

Check Source Measurements				
Det ID	Count Time	Initial	Final	Source ID
252	60 sec.	4300	4300	T-536
252	60 sec.	51/53	62/65	446-089

Neutron or Gamma Quantification Information			
Chemical Form:	UO ₂ F ₂	²³⁵ U/ ²³⁴ U Ratio:	50
Geometry:	Centrifugal Compressor	²³⁵ U (grams):	77 ± 40
Estimated Enrichment:	59 ± 3.6%	UO ₂ F ₂ (pounds):	0.37 ± 0.19
Method to determine Enr.	U-Pu Inspector	Safe mass (grams ²³⁵ U):	377/g or 1.82 lbs. UO ₂ F ₂
Peer reviewed by:	<i>Tom D. Pulk</i> 7-9-97	Supervisory review by:	<i>[Signature]</i> 7-9-97

Check the appropriate classification box: PEH ☐ UH ☒

Sketch of item being measured and detector location.

Background = 982

Flange View = 1280

Motor Shaft View = 1450

Thrust Bearing View = 1160

Side View = 1300

All measurements are at 36" offset from casing of compressor for 900 sec.

Comments:

APPENDIX E

Radiochemistry Data Reduction

ALPHA CONTAMINATION - Uranium Isotopes

5/8" Copper Seal Exhaust from K-25 (even samp. #s - before, odd samp. #s - after)								
Sample Number	U-234 pCi/g	U-235 pCi/g	U-238 pCi/g	Total U pCi/g	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTA-2	23325.0	1102.0	37725.0	62152.0	16.780	1,042,911	2,315,261	463.05
LTA-4	28550.0	2105.0	44610.0	75265.0	16.212	1,220,196	2,708,836	541.77
LTA-6	26150.0	1596.0	42395.0	70141.0	15.728	1,103,178	2,449,054	489.81
LTA-8	25530.0	1936.0	40990.0	68456.0	14.764	1,010,684	2,243,719	448.74
LTA-10	23860.0	1426.0	38830.0	64116.0	16.516	1,058,940	2,350,846	470.17
LTA-12	23785.0	1580.0	36040.0	61405.0	20.716	1,272,066	2,823,986	564.80
LTA-1	325.0	13.0	4.5	342.5	12.514	4,286	9,515	1.90
LTA-3	259.0	10.0	3.4	272.4	11.021	3,002	6,665	1.33
LTA-5	437.0	18.0	5.7	460.7	10.429	4,805	10,666	2.13
LTA-7	227.0	12.0	4.1	243.1	12.339	3,000	6,659	1.33
LTA-9	264.0	11.0	4.0	279.0	12.162	3,393	7,533	1.51
LTA-11	196.0	7.9	3.0	206.9	11.335	2,345	5,206	1.04

Barrier from X-25-4-2.9 Converter (samp. #s 1 to 6 - before, samp. #s 7 to 12 - after)								
Sample Number	U-234 pCi/g	U-235 pCi/g	U-238 pCi/g	Total U pCi/g	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTB-1	5843.0	258.0	44.0	6145.0	1.709	10,502	23,314	4.66
LTB-2	5021.0	186.0	25.0	5232.0	1.588	8,308	18,445	3.69
LTB-3	5880.0	219.0	35.0	6134.0	1.722	10,563	23,449	4.69
LTB-4	5761.0	206.0	31.0	5998.0	1.666	9,993	22,184	4.44
LTB-5	4056.0	155.0	28.0	4239.0	1.661	7,041	15,631	3.13
LTB-6	5127.0	195.0	29.0	5351.0	1.636	8,754	19,434	3.89
LTB-7	503.0	24.0	2.7	529.7	1.585	840	1,864	0.37
LTB-8	7543.0	248.0	33.0	7824.0	1.499	11,728	26,037	5.21
LTB-9	528.0	24.0	3.0	555.0	1.924	1,068	2,371	0.47
LTB-10	739.0	33.0	3.5	775.5	2.098	1,627	3,612	0.72
LTB-11	1338.0	50.0	5.9	1393.9	2.106	2,936	6,517	1.30
LTB-12	1204.0	51.0	6.1	1261.1	1.939	2,445	5,429	1.09

Wipes of X-25-4-2.9 Converter Shell (samp. #s 1 to 4 - before, samp. #s 5 to 8 - after)								
Sample Number	U-234 pCi	U-235 pCi	U-238 pCi	Total U pCi	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTC-1	79.0	2.8	0.3	82.1	wipe	82	182	0.18
LTC-2	149.0	3.9	0.9	153.8	wipe	154	341	0.34
LTC-3	224.0	7.7	1.4	233.1	wipe	233	517	0.52
LTC-4	38.0	1.7	0.3	40.0	wipe	40	89	0.09
LTC-5	286.0	9.6	0.6	296.2	wipe	296	657	0.66
LTC-6	249.0	9.0	0.8	258.8	wipe	259	575	0.57
LTC-7	132.0	4.7	1.0	137.7	wipe	138	306	0.31
LTC-8	45.0	0.9	0.2	46.2	wipe	46	102	0.10

ALPHA CONTAMINATION - Uranium Isotopes (continued)

1/4" Copper Instrument Line from S/9 Hiside (after exposure)

Sample Number	U-234 pCi/g	U-235 pCi/g	U-238 pCi/g	Total U pCi/g	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTD-5	15.0	0.4	0.6	16.0	2.365	16	35	0.01
LTD-6	13.0	0.6	1.6	15.2	2.350	36	79	0.02
LTD-7	12.0	0.5	0.7	13.1	2.483	33	72	0.01
LTD-8	11.0	0.5	0.5	12.1	2.484	30	67	0.01

Aluminum Fins from X-27-3-13.3 Cooler (even samp. #s - before, odd samp. #s - after)

Sample Number	U-234 pCi/g	U-235 pCi/g	U-238 pCi/g	Total U pCi/g	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTE-2	307.0	13.0	4.7	324.7	1.431	465	1,032	0.21
LTE-4	268.0	15.0	5.5	288.5	1.475	426	945	0.19
LTE-6	282.0	12.0	4.8	298.8	1.530	457	1,015	0.20
LTE-8	287.0	12.0	4.0	303.0	1.135	344	763	0.15
LTE-10	300.0	13.0	4.1	317.1	1.281	406	902	0.18
LTE-12	246.0	10.0	4.1	260.1	1.289	335	744	0.15
LTE-1	170.0	7.4	2.6	180.0	1.348	243	539	0.11
LTE-3	210.0	8.9	3.8	222.7	1.480	330	732	0.15
LTE-5	250.0	9.8	3.1	262.9	1.435	377	838	0.17
LTE-7	150.0	7.7	3.0	160.7	1.400	225	499	0.10
LTE-9	260.0	9.0	4.7	273.7	1.285	352	781	0.16
LTE-11	420.0	18.0	6.5	444.5	1.443	641	1,424	0.28

Ni Lined Steel from X-25-4-2.9 Converter Shell (even samp. #s - before, odd samp. #s - after)

Sample Number	U-234 pCi/g	U-235 pCi/g	U-238 pCi/g	Total U pCi/g	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTF-2	1.7	0.1	0.0	1.8	59.958	109	241	0.05
LTF-4	1.6	0.1	0.0	1.7	55.374	94	208	0.04
LTF-6	1.5	0.1	0.0	1.6	82.547	130	290	0.06
LTF-8	1.6	0.1	0.0	1.7	60.158	101	224	0.04
LTF-10	1.2	0.0	0.0	1.3	55.879	71	158	0.03
LTF-12	1.2	0.0	0.0	1.3	62.619	78	174	0.03
LTF-1	64.0	2.5	0.7	67.2	53.802	3,615	8,026	1.61
LTF-3	78.0	3.0	0.9	81.9	65.375	5,353	11,883	2.38
LTF-5	54.0	2.2	0.6	56.8	52.736	2,994	6,647	1.33
LTF-7	98.0	3.7	1.1	102.8	60.014	6,169	13,696	2.74
LTF-9	40.0	1.6	0.4	42.0	56.034	2,355	5,227	1.05
LTF-11	71.0	2.9	0.9	74.8	62.176	4,648	10,319	2.06

ALPHA CONTAMINATION - Uranium Isotopes (continued)

Bare Steel from X-705 Contaminated Scrap (even samp. #s - before, odd samp. #s - after)								
Sample Number	U-234 pCi/g	U-235 pCi/g	U-238 pCi/g	Total U pCi/g	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTG-2	1088.0	29.9	29.5	1147.4	70.196	80,543	178,805	35.76
LTG-4	540.0	16.6	17.2	573.8	76.506	43,899	97,456	19.49
LTG-6	1017.0	29.1	31.1	1077.2	66.009	71,105	157,853	31.57
LTG-8	874.0	26.2	26.5	926.7	81.477	75,505	167,621	33.52
LTG-10	1380.0	40.0	41.9	1461.9	74.975	109,606	243,325	48.67
LTG-12	722.0	20.9	21.9	764.8	70.609	54,002	119,884	23.98
LTG-1	244.0	7.7	5.7	257.4	73.724	18,977	42,128	8.43
LTG-3	63.0	2.1	1.4	66.5	76.946	5,117	11,360	2.27
LTG-5	87.0	2.9	2.0	91.9	71.953	6,612	14,680	2.94
LTG-7	267.0	10.0	7.0	284.0	79.728	22,643	50,267	10.05
LTG-9	200.0	6.0	4.7	210.7	64.311	13,550	30,082	6.02
LTG-11	103.0	3.3	2.5	108.8	71.389	7,767	17,243	3.45

Aluminum Rotor Samples from X-25-4-2.11 Compressor (after exposure)								
Sample Number	U-234 pCi/g	U-235 pCi/g	U-238 pCi/g	Total U pCi/g	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTH-1	1102.0	49.0	2.2	1153.2	12.184	14,051	31,192	6.24
LTH-2	2021.0	82.0	3.0	2106.0	15.436	32,508	72,168	14.43
LTH-3	1480.0	60.0	5.0	1545.0	10.485	16,199	35,963	7.19
LTH-4	1138.0	53.0	2.7	1193.7	9.647	11,516	25,565	5.11
LTH-5	1343.0	68.0	3.0	1414.0	10.061	14,226	31,582	6.32
LTH-6	1120.0	57.0	3.7	1180.7	6.774	7,998	17,756	3.55

ALPHA CONTAMINATION - Transuranics

ND - less than detection limit. NA - analysis not performed.

5/8" Copper Seal Exhaust from K-25 (even samp. #s - before, odd samp. #s - after)											
Sample Number	Am-241 pCi/g	Np-237 pCi/g	Pu-238 pCi/g	Pu-239/240 pCi/g	Th-228 pCi/g	Th-230 pCi/g	Total TRU pCi/g	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTA-2	ND	ND	ND	ND	ND	5.8	5.80	16.780	97	216	2.16
LTA-4	ND	ND	ND	ND	0.230	6.4	6.63	16.212	107	239	2.39
LTA-6	ND	ND	ND	ND	0.210	6.0	6.21	15.728	98	217	2.17
LTA-8	ND	ND	ND	ND	ND	5.5	5.50	14.764	81	180	1.80
LTA-10	ND	ND	ND	ND	0.220	5.5	5.72	16.516	94	210	2.10
LTA-12	ND	ND	ND	ND	ND	6.2	6.20	20.716	128	285	2.85
LTA-1	NA	NA	NA	NA	0.061	1.4	1.46	12.514	18	41	0.41
LTA-3	NA	NA	NA	NA	0.050	1.5	1.55	11.021	17	38	0.38
LTA-5	NA	NA	NA	NA	0.140	1.7	1.84	10.429	19	43	0.43
LTA-7	NA	NA	NA	NA	0.066	1.3	1.37	12.339	17	37	0.37
LTA-9	NA	NA	NA	NA	0.068	1.2	1.27	12.162	15	34	0.34
LTA-11	NA	NA	NA	NA	0.074	1.5	1.57	11.335	18	40	0.40

Barrier from X-25-4-2.9 Converter (samp. #s 1 to 6 - before, samp. #s 7-12 - after)											
Sample Number	Am-241 pCi/g	Np-237 pCi/g	Pu-238 pCi/g	Pu-239/240 pCi/g	Th-228 pCi/g	Th-230 pCi/g	Total TRU pCi/g	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTB-1	NA	27	0.58	4.0	3.60	109	144.18	1.709	246	547	5.47
LTB-2	NA	27	0.48	2.5	3.50	95	128.48	1.588	204	453	4.53
LTB-3	NA	29	0.85	3.1	2.80	93	128.75	1.722	222	492	4.92
LTB-4	NA	36	0.47	4.0	2.70	93	136.17	1.666	227	504	5.04
LTB-5	NA	21	0.30	2.0	1.40	59	83.70	1.661	139	309	3.09
LTB-6	NA	23	0.44	2.4	2.60	77	105.44	1.636	172	383	3.83
LTB-7	0.11	0.89	ND	ND	ND	11	12.00	1.585	19	42	0.42
LTB-8	0.34	8.9	0.59	2.9	0.98	66	79.71	1.499	119	265	2.65
LTB-9	ND	1.2	ND	0.2	ND	10	11.38	1.924	22	49	0.49
LTB-10	ND	1.5	ND	ND	ND	13	14.50	2.098	30	68	0.68
LTB-11	ND	2.7	ND	1.0	ND	22	25.66	2.106	54	120	1.20
LTB-12	0.14	2.4	ND	0.4	ND	14	16.97	1.939	33	73	0.73

Wipes of X-25-4-2.9 Converter Shell (samp. #s 1 to 4 - before, samp. #s 5 to 8 - after)											
Sample Number	Am-241 pCi	Np-237 pCi	Pu-238 pCi	Pu-239/240 pCi	Th-228 pCi	Th-230 pCi	Total TRU pCi	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTC-1	NA	ND	ND	ND	ND	5.20	5.20	wipe	5	12	0.58
LTC-2	NA	ND	ND	ND	ND	1.50	1.50	wipe	2	3	0.17
LTC-3	NA	ND	ND	ND	ND	ND	ND	wipe	ND	ND	NA
LTC-4	NA	ND	ND	ND	ND	ND	ND	wipe	ND	ND	NA
LTC-5	NA	NA	NA	NA	ND	3.50	3.50	wipe	4	8	0.39
LTC-6	NA	NA	NA	NA	ND	0.85	0.85	wipe	1	2	0.09
LTC-7	NA	NA	NA	NA	ND	1.60	1.60	wipe	2	4	0.18
LTC-8	NA	NA	NA	NA	ND	0.79	0.79	wipe	1	2	0.09

ALPHA CONTAMINATION - Transuranics (continued)

ND - less than detection limit. NA - analysis not performed.

1/4" Copper Instrument Line from S/9 Hiside (after exposure)											
Sample Number	Am-241 pCi/g	Np-237 pCi/g	Pu-238 pCi/g	Pu-239/240 pCi/g	Th-228 pCi/g	Th-230 pCi/g	Total TRU pCi/g	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTD-5	NA	ND	ND	ND	ND	2.9	2.90	2.365	7	15	0.15
LTD-6	NA	ND	ND	ND	ND	2.4	2.40	2.350	6	13	0.13
LTD-7	NA	0.072	ND	ND	ND	3.7	3.77	2.483	9	21	0.21
LTD-8	NA	0.280	ND	ND	ND	3.1	3.38	2.484	8	19	0.19

Aluminum Fins from X-27-3-13.3 Cooler (even samp. #s - before, odd samp. #s - after)											
Sample Number	Am-241 pCi/g	Np-237 pCi/g	Pu-238 pCi/g	Pu-239/240 pCi/g	Th-228 pCi/g	Th-230 pCi/g	Total TRU pCi/g	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTE-2	NA	ND	ND	ND	ND	12	12.00	1.431	17	38	0.38
LTE-4	NA	ND	ND	ND	ND	12	12.00	1.475	18	39	0.39
LTE-6	NA	ND	ND	ND	ND	11	11.00	1.530	17	37	0.37
LTE-8	NA	ND	ND	ND	ND	11	11.00	1.135	12	28	0.28
LTE-10	NA	ND	ND	ND	ND	10	10.00	1.281	13	28	0.28
LTE-12	NA	ND	ND	ND	0.64	12	12.64	1.289	16	36	0.36
LTE-1	NA	NA	NA	NA	ND	12	12.00	1.348	16	36	0.36
LTE-3	NA	NA	NA	NA	0.27	11	11.27	1.480	17	37	0.37
LTE-5	NA	NA	NA	NA	0.53	12	12.53	1.435	18	40	0.40
LTE-7	NA	NA	NA	NA	ND	11	11.00	1.400	15	34	0.34
LTE-9	NA	NA	NA	NA	0.38	11	11.38	1.285	15	32	0.32
LTE-11	NA	NA	NA	NA	0.37	12	12.37	1.443	18	40	0.40

Ni-Lined Steel from X-25-4-2.9 Converter Shell (before exposure)											
Sample Number	Am-241 pCi/g	Np-237 pCi/g	Pu-238 pCi/g	Pu-239/240 pCi/g	Th-228 pCi/g	Th-230 pCi/g	Total TRU pCi/g	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTF-2	NA	ND	ND	ND	ND	0.24	0.24	59.958	14	32	0.32
LTF-4	NA	ND	ND	ND	0.02	ND	0.02	55.374	1	2	0.02
LTF-6	NA	ND	ND	ND	ND	0.21	0.21	82.547	17	38	0.38
LTF-8	NA	ND	ND	ND	ND	0.23	0.23	60.158	14	31	0.31
LTF-10	NA	ND	ND	0.05	ND	0.22	0.27	55.879	15	33	0.33
LTF-12	NA	ND	ND	ND	ND	1.79	1.79	62.619	112	249	2.49

Bare Steel from X-705 Contaminated Scrap (before exposure)											
Sample Number	Am-241 pCi/g	Np-237 pCi/g	Pu-238 pCi/g	Pu-239/240 pCi/g	Th-228 pCi/g	Th-230 pCi/g	Total TRU pCi/g	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTG-2	NA	ND	ND	ND	ND	0.41	0.41	70.196	29	64	0.64
LTG-4	NA	ND	ND	ND	ND	0.04	0.04	76.506	3	7	0.07
LTG-6	NA	ND	ND	ND	ND	0.14	0.14	66.009	9	21	0.21
LTG-8	NA	ND	ND	ND	ND	0.13	0.13	81.477	11	24	0.24
LTG-10	NA	ND	ND	ND	ND	0.05	0.05	74.975	4	8	0.08
LTG-12	NA	ND	ND	ND	ND	0.14	0.14	70.609	10	22	0.22

ALPHA CONTAMINATION - Transuranics (continued)

ND - less than detection limit. NA - analysis not performed.

Aluminum Rotor Samples from X-25-4-2.11 Compressor (after exposure)											
Sample Number	Am-241 pCi/g	Np-237 pCi/g	Pu-238 pCi/g	Pu-239/240 pCi/g	Th-228 pCi/g	Th-230 pCi/g	Total TRU pCi/g	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTH-1	ND	0.68	0.036	0.31	0.14	9.1	10.27	12.184	125	278	2.78
LTH-2	0.071	1.40	0.061	0.71	0.25	15.0	17.49	15.436	270	599	5.99
LTH-3	ND	0.87	0.041	0.37	0.21	10.0	11.49	10.485	120	267	2.67
LTH-4	0.260	0.58	0.520	3.10	0.35	35.0	39.81	9.647	384	853	8.53
LTH-5	0.360	0.62	0.440	3.00	0.45	44.0	48.87	10.061	492	1092	10.92
LTH-6	0.210	0.50	0.310	2.50	0.42	41.0	44.94	6.774	304	676	6.76

ALPHA CONTAMINATION - Thorium-232

ND - less than detection limit. NA - analysis not performed

5/8" Copper Seal Exhaust from K-25 (even samp. #s - before, odd samp. #s - after)					
Sample Number	Th-232 pCi/g	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTA-2	ND	16.780	ND	ND	NA
LTA-4	ND	16.212	ND	ND	NA
LTA-6	ND	15.728	ND	ND	NA
LTA-8	ND	14.764	ND	ND	NA
LTA-10	0.084	16.516	1.39	3.08	0.003
LTA-12	ND	20.716	ND	ND	NA
LTA-1	ND	12.514	ND	ND	NA
LTA-3	ND	11.021	ND	ND	NA
LTA-5	ND	10.429	ND	ND	NA
LTA-7	ND	12.339	ND	ND	NA
LTA-9	ND	12.162	ND	ND	NA
LTA-11	ND	11.335	ND	ND	NA

Barrier from X-25-4-2.9 Converter (samp. #s 1 to 6 - before, samp. #s 7 to 12 - after)					
Sample Number	Th-232 pCi/g	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTB-1	0.32	1.709	0.55	1.21	0.001
LTB-2	ND	1.588	ND	ND	NA
LTB-3	ND	1.722	ND	ND	NA
LTB-4	ND	1.666	ND	ND	NA
LTB-5	ND	1.661	ND	ND	NA
LTB-6	0.15	1.636	0.25	0.54	0.001
LTB-7	ND	1.585	ND	ND	NA
LTB-8	ND	1.499	ND	ND	NA
LTB-9	ND	1.924	ND	ND	NA
LTB-10	ND	2.098	ND	ND	NA
LTB-11	ND	2.106	ND	ND	NA
LTB-12	ND	1.939	ND	ND	NA

Wipes of X-25-4-2.9 Converter Shell (samp. #s 1 to 4 - before, samp. #s 5 to 8 - after)					
Sample Number	Th-232 pCi	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTC-1	ND	wipe	ND	ND	NA
LTC-2	ND	wipe	ND	ND	NA
LTC-3	ND	wipe	ND	ND	NA
LTC-4	ND	wipe	ND	ND	NA
LTC-5	ND	wipe	ND	ND	NA
LTC-6	ND	wipe	ND	ND	NA
LTC-7	ND	wipe	ND	ND	NA
LTC-8	ND	wipe	ND	ND	NA

ALPHA CONTAMINATION - Thorium-232 (continued)

ND - less than detection limit. NA - analysis not performed

1/4" Copper Instrument Line from S/9 Hiside (after exposure)					
Sample Number	Th-232 pCi/g	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTD-5	ND	2.365	ND	ND	NA
LTD-6	ND	2.350	ND	ND	NA
LTD-7	ND	2.483	ND	ND	NA
LTD-8	ND	2.484	ND	ND	NA

Aluminum Fins from X-27-3-13.3 Cooler (even samp. #s - before, odd samp. #s - after)					
Sample Number	Th-232 pCi/g	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTE-2	ND	1.431	ND	ND	NA
LTE-4	ND	1.475	ND	ND	NA
LTE-6	ND	1.530	ND	ND	NA
LTE-8	ND	1.135	ND	ND	NA
LTE-10	ND	1.281	ND	ND	NA
LTE-12	ND	1.289	ND	ND	NA
LTE-1	ND	1.348	ND	ND	NA
LTE-3	ND	1.480	ND	ND	NA
LTE-5	ND	1.435	ND	ND	NA
LTE-7	ND	1.400	ND	ND	NA
LTE-9	ND	1.285	ND	ND	NA
LTE-11	ND	1.443	ND	ND	NA

Ni-Lined Steel from X-25-4-2.9 Converter Shell (before exposure)					
Sample Number	Th-232 pCi/g	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTF-2	0.05	59.958	3	7	0.007
LTF-4	ND	55.374	ND	ND	NA
LTF-6	ND	82.547	ND	ND	NA
LTF-8	ND	60.158	ND	ND	NA
LTF-10	ND	55.879	ND	ND	NA
LTF-12	ND	62.619	ND	ND	NA

Bare Steel from X-705 Contaminated Scrap (before exposure)					
Sample Number	Th-232 pCi/g	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTG-2	ND	70.196	ND	ND	NA
LTG-4	ND	76.506	ND	ND	NA
LTG-6	ND	66.009	ND	ND	NA
LTG-8	ND	81.477	ND	ND	NA
LTG-10	0.02	74.975	1	3	0.003
LTG-12	ND	70.069	ND	ND	NA

ALPHA CONTAMINATION – Thorium-232 (continued)

ND – less than detection limit. NA – analysis not performed

Aluminum Rotor Samples from X-25-4-2.11 Compressor (after exposure)					
Sample Number	Th-232 pCi/g	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTH-1	0.030	12.184	0.4	0.8	0.0008
LTH-2	0.020	15.436	0.3	0.7	0.0007
LTH-3	0.019	10.485	0.2	0.4	0.0004
LTH-4	0.056	9.647	0.5	1.2	0.0012
LTH-5	0.095	10.061	1.0	2.1	0.0021
LTH-6	0.100	6.774	0.7	1.5	0.0015

BETA/GAMMA CONTAMINATION

ND - less than detection limit. NA - analysis not performed.

5/8" Copper Seal Exhaust from K-25 (even samp. #s - before, odd samp. #s - after)								
Sample Number	Tc-99 pCi/g	Th-231 pCi/g	Th-234 pCi/g	Total pCi/g	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTA-2	22.7	1102.0	37725.0	38849.7	16.780	651898	1447213	289.44
LTA-4	37.2	2105.0	44610.0	46752.2	16.212	757947	1682642	336.53
LTA-6	34.6	1596.0	42395.0	44025.6	15.728	692435	1537205	307.44
LTA-8	36.4	1936.0	40990.0	42962.4	14.764	634297	1408139	281.63
LTA-10	36.6	1426.0	38830.0	40292.6	16.516	665473	1477349	295.47
LTA-12	37.5	1580.0	36040.0	37657.5	20.716	780113	1731850	346.37
LTA-1	65440.0	13.0	4.5	65457.5	12.514	819135	1818480	363.70
LTA-3	62520.0	10.0	3.4	62533.4	11.021	689181	1529981	306.00
LTA-5	71110.0	18.0	5.7	71133.7	10.429	741853	1646914	329.38
LTA-7	60050.0	12.0	4.1	60066.1	12.339	741156	1645365	329.07
LTA-9	67800.0	11.0	4.0	67815.0	12.162	824766	1830981	366.20
LTA-11	74730.0	7.9	3.0	74740.9	11.335	847188	1880758	376.15

Barrier from X-25-4-2.9 Converter (samp. #s 1 to 6 - before, samp. #s 7 to 12 - after)								
Sample Number	Tc-99 pCi/g	Th-231 pCi/g	Th-234 pCi/g	Total pCi/g	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTB-1	181000	258	44.0	181302.0	1.709	309845	687856	137.57
LTB-2	178000	186	25.0	178211.0	1.588	282999	628258	125.65
LTB-3	203000	219	35.0	203254.0	1.722	350003	777008	155.40
LTB-4	179000	206	31.0	179237.0	1.666	298609	662912	132.58
LTB-5	149000	155	28.0	149183.0	1.661	247793	550100	110.02
LTB-6	216000	195	29.0	216224.0	1.636	353742	785308	157.06
LTB-7	118913	24	2.7	118939.7	1.585	188519	418513	83.70
LTB-8	273549	248	33.0	273830.0	1.499	410471	911246	182.25
LTB-9	78623	24	3.0	78650.0	1.924	151323	335936	67.19
LTB-10	125534	33	3.5	125570.5	2.098	263447	584852	116.97
LTB-11	175242	50	5.9	175297.9	2.106	369177	819574	163.91
LTB-12	123469	51	6.1	123526.1	1.939	239517	531728	106.35

Wipes of X-25-4-2.9 Converter Shell (samp. #s 1 to 4 - before, samp. #s 5 to 8 - after)								
Sample Number	Tc-99 pCi	Th-231 pCi	Th-234 pCi	Total pCi	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTC-1	10043	2.80	0.33	10046.13	wipe	10046	22302	22.30
LTC-2	5636	3.90	0.86	5640.76	wipe	5641	12522	12.52
LTC-3	5055	7.70	1.40	5064.10	wipe	5064	11242	11.24
LTC-4	4260	1.70	0.33	4262.03	wipe	4262	9462	9.46
LTC-5	6980	9.60	0.55	6990.15	wipe	6990	15518	15.52
LTC-6	3390	9.00	0.80	3399.80	wipe	3400	7548	7.55
LTC-7	2750	4.70	0.95	2755.65	wipe	2756	6118	6.12
LTC-8	1000	0.92	ND	1000.92	wipe	1001	2222	2.22

BETA/GAMMA CONTAMINATION (continued)

ND - less than detection limit. NA - analysis not performed.

1/4" Copper Instrument Line from S/9 Hiside (after exposure)								
Sample Number	Tc-99 pCi/g	Th-231 pCi/g	Th-234 pCi/g	Total pCi/g	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTD-5	ND	0.41	0.58	0.99	2.365	2	5	0.001
LTD-6	ND	0.57	1.60	2.17	2.350	5	11	0.002
LTD-7	ND	0.47	0.65	1.12	2.483	3	6	0.001
LTD-8	ND	0.53	0.53	1.06	2.484	3	6	0.001

Aluminum Fins from X-27-3-13.3 Cooler (even samp. #s - before, odd samp. #s - after)								
Sample Number	Tc-99 pCi/g	Th-231 pCi/g	Th-234 pCi/g	Total pCi/g	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTE-2	4412	13.0	4.7	4429.7	1.431	6339	14072	2.81
LTE-4	4056	15.0	5.5	4076.5	1.475	6013	13348	2.67
LTE-6	4076	12.0	4.8	4092.8	1.530	6262	13902	2.78
LTE-8	3240	12.0	4.0	3256.0	1.135	3696	8204	1.64
LTE-10	3349	13.0	4.1	3366.1	1.281	4312	9573	1.91
LTE-12	2957	10.0	4.1	2971.1	1.289	3830	8502	1.70
LTE-1	25180	7.4	2.6	25190.0	1.348	33956	75383	15.08
LTE-3	46060	8.9	3.8	46072.7	1.480	68188	151376	30.28
LTE-5	41630	9.8	3.1	41642.9	1.435	59758	132662	26.53
LTE-7	31780	7.7	3.0	31790.7	1.400	44507	98805	19.76
LTE-9	74670	9.0	4.7	74683.7	1.285	95969	213050	42.61
LTE-11	35860	18.0	6.5	35884.5	1.443	51781	114955	22.99

Ni-Lined Steel from X-25-4-2.9 Converter Shell (even samp. #s - before, odd samp. #s - after)								
Sample Number	Tc-99 pCi/g	Th-231 pCi/g	Th-234 pCi/g	Total pCi/g	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTF-2	15	0.09	1.70	16.79	59.958	1007	2235	0.45
LTF-4	20	0.06	1.60	21.66	55.374	1199	2663	0.53
LTF-6	9.7	0.05	0.04	9.79	82.547	808	1794	0.36
LTF-8	21	0.10	ND	21.10	60.158	1269	2818	0.56
LTF-10	18	0.03	ND	18.03	55.879	1007	2237	0.45
LTF-12	17	0.02	ND	17.02	62.619	1066	2366	0.47
LTF-1	2265	NA	NA	2265.00	53.802	121862	270533	54.11
LTF-3	3049	NA	NA	3049.00	65.375	199328	442509	88.50
LTF-5	6588	NA	NA	6588.00	52.736	347425	771283	154.26
LTF-7	3349	NA	NA	3349.00	60.014	200987	446191	89.24
LTF-9	2025	NA	NA	2025.00	56.034	113469	251901	50.38
LTF-11	3994	NA	NA	3994.00	62.176	248331	551295	110.26

BETA/GAMMA CONTAMINATION (continued)

ND - less than detection limit. NA - analysis not performed.

Bare Steel from X-705 Contaminated Scrap (even samp. #s - before, odd samp. #s - after)								
Sample Number	Tc-99 pCi/g	Th-231 pCi/g	Th-234 pCi/g	Total pCi/g	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTG-2	56	29.9	29.5	115.4	70.196	8101	17983	3.60
LTG-4	62	16.6	17.2	95.8	76.506	7329	16271	3.25
LTG-6	43	29.1	31.1	103.2	66.009	6812	15123	3.02
LTG-8	70	26.2	26.5	122.7	81.477	9997	22194	4.44
LTG-10	84	40.0	41.9	165.9	74.975	12438	27613	5.52
LTG-12	76	20.9	21.9	118.8	70.609	8388	18622	3.72
LTG-1	35465	NA	NA	35465.0	73.724	2614622	5804460	1160.89
LTG-3	22358	NA	NA	22358.0	76.946	1720359	3819196	763.84
LTG-5	24531	NA	NA	24531.0	71.953	1765079	3918475	783.70
LTG-7	33574	NA	NA	33574.0	79.728	2676788	5942469	1188.49
LTG-9	34030	NA	NA	34030.0	64.311	2188503	4858477	971.70
LTG-11	30057	NA	NA	30057.0	71.389	2145739	4763541	952.71

Aluminum Rotor Samples from X-25-4-2.11 Compressor (after exposure)								
Sample Number	Tc-99 pCi/g	Th-231 pCi/g	Th-234 pCi/g	Total pCi/g	Samp. Wt. g	Samp. Act. pCi	Samp. Act. dpm	Meas./Guide
LTH-1	7009	49	2.2	7060.2	12.184	86021	190968	38.19
LTH-2	7536	82	3.0	7621.0	15.436	117638	261156	52.23
LTH-3	7839	60	5.0	7904.0	10.485	82873	183979	36.80
LTH-4	15618	53	2.7	15673.7	9.647	151204	335673	67.13
LTH-5	18765	68	3.0	18836.0	10.061	189509	420710	84.14
LTH-6	16582	57	3.7	16642.7	6.774	112738	250278	50.06

APPENDIX F

**Comparison of Sample Uranium Alpha Contamination
to Free Release Guidance in USAEC Regulatory Guide 1.86**

Alpha Contamination Uranium Isotopes

LTA Series Sample No. (5/8" Copper)

Alpha Contamination Uranium Isotopes

Removable Alpha Contamination Uranium Isotopes

Alpha Contamination After Exposure Uranium Isotopes

LTD Series Sample No. (1/4" Copper)

Alpha Contamination Uranium Isotopes

Alpha Contamination Uranium Isotopes

LTF Series Sample No. (Ni-Lined Steel)

Alpha Contamination Uranium Isotopes

LTG Series Sample No. (Bare Steel)

Alpha Contamination After Exposure Uranium Isotopes

LTH Series Sample No. (S/11 Compressor)

DISTRIBUTION

LMUS/PORTS

D. A. Fosson
C. Good
T. C. Houk
R. B. Kefgen
R. R. Konanz
D. M. Manuta
B. R. McGinnis
J. B. Morgan
W. D. Netzer
M. J. Orlett
G. S. Price
A. J. Saraceno
R. J. Schwab
B. W. Short
H. H. Thomas
D. A. Towne
J. E. Tully
R. C. Warren
J. J. Williams
J. L. Woodard
Central Files (2)
Technical Information (2)
Technical Library (2)

LMUS/PGDP

R. B. Gross
R. L. Jones
Technical Library (2)

LMES/PORTS

J. R. Goodbred
T. C. Houk
T. D. Taylor

LMES/K-25

R. L. Fellows
R. L. Higgins
E. B. Munday
D. W. Simmons

LMER/ORNL

A. P. Malinauskas
T. Myrick
L. D. Trowbridge

DOE/PORTS

E. W. Gillespie

DOE/ORO

J. L. Harness
D. L. Mason
J. O. Moore

DOE/HQ

J. M. Hyde

DOE/FETC

S. Bossart
J. Duda
P. Hart
H. Shoemaker