Draft Final

FINAL STATUS SURVEY PLAN
Building 7304 Vault
Fort Belvoir, Virginia

Contract Number

DAAA09-02-D-0024/003

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03-3040.03

March 2004

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GLOSSARY OF ACRONYMS AND ABBREVIATIONS

AFSC	United States Army Field Support Command
ALARA	As Low As Reasonably Achievable
Am-241	Americium-241
C-14	Carbon-14
CABRERA	Cabrera Services, Inc.
CAD	Computer-Aided Drawing
CFR	United States Code of Federal Regulations
cm	Centimeter
Cs-137	Cesium-137
DCGL	Derived Concentration Guideline Level
$DCGL_W$	Derived Concentration Guideline Level used for Non-Parametric Statistical Test
e.g.	For Example
EMC	Elevated Measurement Comparison
EPA	United States Environmental Protection Agency
FIDLER	Field Instrument for the Detection of Low-Energy Radiation
FSS	Final Status Survey
Ft.	Fort
ft	Feet

ft^2	Square Feet
GIS	Geographic Information System
GPS	Global Positioning System
GWS	Gamma Walkover Survey
H-3	Tritium
H_a	Alternative Hypothesis
H_o	Null Hypothesis
i.e.	That Is
LBGR	Lower Bound of the Gray Region
LTR	License Termination Rule
m	Meter
m^2	Square Meter
m/s	Meters per Second
MARSSIM	Multi-Agency Radiological Survey and Site Investigation Manual
MDC	Minimum Detectable Concentration
mrem/yr	Millirem per Year
MSL	Mean Sea Level
NaI	Sodium Iodide
Ni-63	Nickel-63
NIST	National Institute of Standards and Technology

NRC	United States Nuclear Regulatory Commission
PC	Personal Computer
pCi/g	PicoCuries per Gram
PE	Project Engineer
PM	Project Manager
Pm-147	Promethium-147
Pu-238	Plutonium-238
Pu-239/240	Plutonium-239/240
QA	Quality Assurance
QC	Quality Control
Ra-226	Radium-226
ROCs	Radionuclides of Concern
RSO	Radiation Safety Officer
SBCCOM	United States Army Soldier, Biological and Chemical Command
SOR	Sum of the Ratios
SU	Survey Unit
Tc-99	Technetium-99
TEDE	Total Effective Dose Equivalent
Th-232	Thorium-232
U-234	Uranium-234

U-235	Uranium-235
U-238	Uranium-238
U.S.	United States
USCG	United States Coast Guard
Vault	Building 7304 Vault
WRS	Wilcoxon Rank Sum

1.0 INTRODUCTION

Cabrera Services, Inc. (CABRERA) is under contract to the United States (U.S.) Army Field Support Command (AFSC), formerly known as the United States Army Joint Munitions Command. The Building 7304 Vault (the Vault) is located at Fort (Ft.) Belvoir in Fairfax County, Virginia and decommissioning activities are being performed to release the former Vault area for unrestricted use. CABRERA will be conducting a Final Status Survey (FSS) of the Vault excavation area following removal of the Vault and nearby impacted areas.

U.S. Army Soldier, Biological and Chemical Command (SBCCOM) is the license holder for this storage facility, U.S. Nuclear Regulatory Commission (NRC) license number (#) 45-00953-01 (NRC, 2003). The license is current and the Vault facility will be removed from the license following decommissioning and release for unrestricted use.

2.0 FACILITY AND SITE INFORMATION

2.1 Facility Location and Description

The Vault is a concrete construction, bunker-style building, enclosed within an earthen covering approximately three feet (ft) thick. The front of the building is concrete, with a walkway and shield wall. A security fence abuts the rear of the structure. The Vault building is 12 ft by 16 ft, equaling approximately 192 square feet (ft^2). The entire area to be remediated is expected to be approximately 500 ft^2. This encompasses an additional 4 ft on three sides of the structure to allow the excavation of the exterior buried drains, and approximately 60 ft^2 in the front of the building for the exterior walkway and shield wall.

The Vault is located on the western shore of the Potomac River within the Ft. Belvoir Military Reservation in Fairfax County, Virginia (See Figure 3-1). It is approximately 16 miles south by southwest of the center of Washington D.C. The nearest neighboring towns are Mount Vernon, three miles to the east, and Accotink one mile west. Four Miles away to the southeast is Marshall Hall, Maryland, across the Potomac River. U.S. Route 1 crosses the north boundary of the Ft. Belvoir Reservation, with Accotink Bay and Gunston Cove to the west and southwest. The Potomac River borders south and Dogue Creek the east.

The Vault lies within the Ft. Belvoir Military Reservation near Gunston Point at the southern end of a peninsula. The peninsula is relatively level across the center with gently contoured hills sloping toward the river and bays. The elevation within the Reservation ranges from 140 ft above mean sea level (MSL) to sea level at the shores. The elevation of the Vault site is approximately 52 ft above MSL. The area is wooded along the shores of the peninsula, and between the areas where the man-made structures are built. The peninsula is approximately 50 percent (%) wooded.

The man-made features within Ft. Belvoir consist of roads, buildings, docks, and rail spur(s). While most roads are paved, a relative few are gravel. The buildings range from research facilities, a nuclear reactor facility, administrative buildings and warehouses, to single and multiple family residences.

2.2 Previous Radiological Studies

This project has been scoped into phases. Phase I was completed in June 2002. It consisted of a site visit to inventory unwanted radioactive materials and to conduct limited sampling and analysis to support disposal decisions. Phase II included a characterization survey and the transportation and disposal of the unwanted radioactive materials to an offsite facility. The site work portion of Phase II was completed during the first quarter of 2003.

Results of the characterization survey radiological analyses indicated the presence of elevated tritium (H-3), Carbon-14 (C-14), Cesium-137 (Cs-137), Promethium-147 (Pm-147), Americium-241 (Am-241), and Thorium-232 (Th-232) in excess of release limits. Elevated levels of radioactivity were detected on the interior Vault floor, at wall storage vaults, at floor storage vaults, and in surface soils below the structure. The highest contamination exceedance of action levels was Cs-137 on the Vault floor and in the surface soil under the floor storage vaults and also H-3 inside the wall storage vaults. Contamination exceeding action levels outside the Vault is minimal and is concentrated on the north wall and floor just outside the Vault doorway.

3.0 FINAL STATUS SURVEY

Remediation activities will be performed to remove the Vault building and surrounding contaminated soils followed by FSS performance. Planned activities performed in support of the FSS will include gamma walkover surveys (GWS) and the collection and analysis of a discrete number of soil samples. These activities will be performed for the SU.

3.1 Soil Clean-Up Goals

3.1.1 Radionuclides of Concern

The primary radionuclides of concern (ROCs) for the decommissioning effort are those radionuclides which were identified in the sample results of the project Characterization Survey Report generated by CABRERA in 2003 as exceeding release limits. Certain ROCs previously included in the characterization survey, but not identified in characterization results are not included in the current list of ROCs. The ROCs for this decommissioning effort are H-3, C-14, Cs-137, Pm-147, Am-241, and Th-232.

3.1.2 Unrestricted Release

On June 21, 1997, the NRC published the final rule on "Radiological Criteria for License Termination", the License Termination Rule (LTR), as Subpart E to 10 United States Code of Federal Regulations (CFR) Part 20. The criteria for termination with unrestricted release is residual radioactivity, which is undistinguishable from background, and results in a total effective dose equivalent (TEDE) to an average member of the critical group that does not exceed 25 millirem per year (mrem/yr), including that from groundwater sources of drinking water, and the residual radioactivity has been reduced to levels that are as low as reasonably achievable (ALARA). Determination of the levels which are ALARA must take into account consideration of any detriments, such as deaths from transportation accidents, expected to potentially result from excavation and waste disposal activities. For the decommissioning of the Vault, a dose objective of 25 mrem/yr above background is applicable and is therefore used as the basis for demonstrating that the Vault should be released for unrestricted use. The method for evaluating the dose objective is provided below.

Supplemental information regarding the implementation of the LTR, including screening criteria for building surfaces and soil, was published by NRC in the Federal Register Volume 63, # 222, November 18, 1998 (NRC, 1998); the Federal Register Volume 64, # 234, December 7, 1999 (NRC, 1999); and also the Federal Register Volume 65, # 114, June 13, 2000 (NRC, 2000). Soil screening criteria required for Vault ROCs not presented in the preceding Federal Register documents have been referenced from Table 6.91 of (NUREG)/CR-5512, Volume 3, October 1999 (NRC, 1999b). These screening criteria have been used to establish Vault decommissioning Derived Concentration Guideline Levels (DCGLs) and to establish instrument/analysis sensitivity requirements for this survey.

Remediation activities will be performed to remove the Vault building and surrounding contaminated soils. Remaining soils expected to contain very small levels of residual radioactivity will be the area of study for unrestricted release surveys. As described in the guidance presented in Federal Register Volume 65, June 13, 2000, the use of the soil screening values presented in Federal Register Volume 64, December 7, 1999 may be used to demonstrate compliance with the LTR for soils under specific guidelines. The four guidelines by which soils may be deemed acceptable for release for unrestricted use are as follows:

1) The residual radioactivity has been reduced to levels that are ALARA

2) The residual radioactivity is contained in the top layer of the surface soil (that is [i.e.], within approximately 15 centimeters [cm] of the surface)

3) The unsaturated zone and the groundwater are initially free of radiological contamination; and

4) The vertical saturated hydraulic conductivity at the specific site is greater than the infiltration rate.

Under the guidance presented in the above mentioned Federal Register documents and also in NUREG-1757 Volume 1 (NRC, 2003b), the soil screening criteria presented in the December 7, 1999 Federal Register document have been used to develop the DCGLs for the Vault decommissioning, as presented in Table 3-1 to show compliance with the 25 mrem/yr dose criteria.

Table 3-1

Building 7304 Decommissioning DCGLs

Nuclide	DCGL (picoCuries per gram [pCi/g])
H-3	110
C-14	12.0
Pm-147	8.2E3
Cs-137	11.0
Am-241	2.10
Th-232	1.10

3.2 Area Classification Based on Contamination Potential

As discussed in Multi-Agency Radiological Survey and Site Investigation Manual (MARSSIM) (NRC, 2000b), areas of sites undergoing an FSS should be classified into SU according to their potential for residual radioactivity. Section 2.2 of MARSSIM provides the following definitions for classifying areas (herein identified as survey units):

Non-impacted Areas: Areas that have no reasonable potential for residual contamination.

Impacted Areas: Any area not classified as non-impacted. Areas with a possibility of containing residual radioactivity in excess of natural background or fallout levels.

Class 1 Areas: Impacted areas that have, or had prior to remediation, a potential for contamination (based on site operating history) or known contamination (based on previous radiological surveys) above the Derived Concentration Guideline Level used for Non-Parametric Statistical Test ($DCGL_w$).

Class 2 Areas: Impacted areas that, prior to remediation, are not likely to have concentrations of residual radioactivity that exceed the $DCGL_w$.

Class 3 Areas: Impacted areas that have a low probability of containing residual radioactivity.

Following remediation activities, the entire excavated area will considered one Class 1 SU following the MARSSIM guidance for selecting a Class 1 SU, as presented above. The MARSSIM suggested limit in area for a Class 1 outdoor SU is 2,000 square meters (m^2) (21,528 ft^2). The former Vault location to be final status surveyed will encompass less area than 2,000 m^2. Class 1 SUs require a scan survey across 100% of the accessible SU surface as per MARSSIM guidance.

3.3 Survey Reference Coordinate System

A FSS reference coordinate system will be developed and installed early in the FSS process. Coordinates will be referenced to the State Plane Coordinate System. At a minimum, the boundaries of the SU will be identified and clearly marked. Additionally, to facilitate the GWS, intermediate markings may be installed using pin flags to mark the start and end points of planned survey lines. The use of a global positioning system (GPS) obviates the need for marking small grid intervals.

3.4 Number of Sample Locations for Survey Units and Reference Area

MARSSIM discusses a method to determine the number of sample locations required in a given SU. A minimum number of sample locations are required in the SU to obtain sufficient statistical confidence that the conclusions drawn from the measurements are correct. For the purpose of this FSS, the minimum required number of measurements is based on expected radionuclide concentrations near or at background in site areas that may be suitable for release for unrestricted use. The following sections describe the bases for and derivation of the minimum required measurement locations per SU.

3.4.1 Estimation of Relative Shift

The minimum number of sample locations required is dependent on the distribution of site residual radionuclide concentrations relative to the DCGL and acceptable decision error limits (α and β) established in Section 4.6. When multiple contaminants are present on a site, such as the Vault site, radiological conditions are evaluated using the sum of the ratios (SOR) and a $DCGL_w$ of SOR =1.0. The SOR is calculated as follows:

$$SOR = \frac{C_1}{\mathrm{DCGL}_1} + \frac{C_2}{\mathrm{DCGL}_2} + \frac{C_3}{\mathrm{DCGL}_3} + \ldots \frac{C_n}{DCGL_n}$$

Where: C_n.= Measured activity concentration for a given nuclide

$\mathrm{DCGL}_n = \mathrm{DCGL}_w$ pCi/g for the given nuclide

The relative shift describes the relationship of site residual radionuclide concentrations to the DCGL and is calculated using the following equation, found in Section 5.5.2.3 of MARSSIM. Based on available site data and the assumptions described below, the relative shift used to determine the minimum number of samples is 1.0.

$$\Delta/\sigma = \frac{\mathrm{DCGL}_w \text{ - LBGR}}{\sigma}$$

Where: DCGL_w = DCGL [i.e., release limit]

LBGR = Concentration at the lower bound of the gray region (LBGR). The LBGR is the concentration to which the SU must be remediated to have an acceptable probability of passing the statistical tests. The LBGR effectively becomes the FSS action level

σ = An estimate of the standard deviation of the residual radioactivity SORs (which includes real spatial variability in the concentration as well as the precision of the measurement system)

(1) DCGL_w

As described previously, the DCGL_w is equal to an SOR of 1.0.

(2) LBGR

The LBGR is typically used as a clean-up guideline (or action level), as discussed above. This application of the LBGR is not directly applicable to the survey design, because the remediation goal is to effectively remove all residual radioactivity. MARSSIM suggest using one-half of the DCGL_w for the LBGR. The LBGR for an SOR of 1 is then 0.5.

(3) Sigma (σ)

Considering the decommissioning action goal is the removal of the Vault and to ensure no residual radioactivity above background exists following excavation, the standard deviation of SU activity concentration estimates is assumed to be approximately equal to background. The assumed standard deviation must also consider the expected variability in the measurement techniques. For conservatism, the standard deviation SOR value will be assumed for this study to be 0.25.

The resulting relative shift is 2.0 after subtracting the LBGR from the $DCGL_w$ and dividing by 0.25. This value was re-calculated with 20% added to the average standard deviation SOR for further conservatism (0.30) and the resulting relative shift is calculated to be 1.6.

3.4.2 Determination of Number of Required Sample Locations

The Wilcoxon Rank Sum (WRS) statistical test will be used to determine whether portions of the site are suitable for release for unrestricted use. The minimum number of systematic measurement locations required in the SU for the WRS statistical test is determined using the calculated relative shift presented in the previous section and MARSSIM Table 5.3.

Section 4.6 establishes the acceptable decision errors to be α=0.05 and β=0.05. Based on these acceptable decision errors, and the relative shift of 1.6 established in the previous section, the minimum number of measurement locations in the SU, the number of required sample locations (N), is interpolated to be 16. MARSSIM includes 20% additional samples in the sample number values presented in Table 5.3 to protect against the possibility of lost or unusable data. Therefore, this value of 16 samples includes the 20% increase in the number of measurement locations per SU.

3.4.3 Elevated Measurement Comparison Criterion

MARSSIM states that a dose area factor must also be used to evaluate the magnitude by which the concentration within a small area of elevated activity can exceed the $DCGL_w$ while maintaining compliance with the release criterion. The following formula is listed in section 5.5.2.4 of MARSSIM for determining the necessary scan sensitivity when incorporating the area factor:

Scan Minimum Detectable Concentration (required) = ($DCGL_w$) x (Area Factor)

If the actual scan MDC is greater than the required scan MDC, additional samples are required to ensure that the dose-based criterion is satisfied. The calculated scan MDCs for the FIDLER presented in Section 5.1.1 are all less than the respective DCGL for each radionuclide. Incorporating area factors using the formula above could only increase the necessary scan sensitivity.

3.4.4 Biased Sample Measurements

If areas of elevated radioactivity are identified during the GWS, biased samples will be collected to facilitate evaluation of elevated area radionuclide concentrations against MARSSIM EMC criteria. At a minimum, one biased soil sample will be collected in the SU at the location of the highest gamma walkover reading. Biased samples may also be collected at locations where GWS data is greater than three standard deviations from the average (each "Z" score is one standard deviation from the average).

3.5 Establishing Sample Locations

3.5.1 Establishing Sample Locations in the Class 1 SU

Systematic sample locations in The SU will be established and marked using survey flags, or equivalent, prior to sample collection. A triangular sampling grid will be established for the SU based on its area and the number of sampling locations. The grid spacing for the SU will be determined based on the measured area of the SU, using the following equation (Equation 5-7 from MARSSIM).

$$L = \sqrt{\frac{A}{0.866(N)}}$$

Where: L = triangular grid spacing for SU

A = area of SU

N = number of sample locations

Assuming that the Class 1 SU area is approximately 500 m^2, the calculated spacing between sample locations in the triangular grid will be 6 m.

If the gamma walkover survey identifies areas of elevated radioactivity, biased samples will be collected to evaluate elevated area radionuclide concentrations against MARSSIM EMC criteria.

A computer-aided drawing (CAD) program will be used to lay a triangular grid with proper length spacing over the SU. A random start point for the grid will be established using a computer-generated random coordinate set. The number of sample locations corresponding to the random grid will be determined using CAD. Location coordinates will then be located in the FSS unit using a GPS and measuring tape, as necessary.

3.6 Survey Instrumentation and Survey Techniques

3.6.1 Gamma Walkover Survey

A 100% gamma walkover survey will be performed over accessible areas within the SU, in accordance with MARSSIM guidance for Class 1 SU. The purpose of the gamma walkover survey is to identify areas of elevated radioactivity. Equipment required for performing the GWS includes the following:

- GPS Base Station: Trimble Universal Reference Station (or equivalent, as necessary).
- GPS Rover: Trimble Pathfinder Pro - XR (or equivalent)
- FIDLER NaI detector, or equivalent NaI detector and associated rate-meter/scalar

- Hardware: IBM-compatible Pentium (minimum) personal computer (PC), color printer, large capacity data storage device (e.g., zip drive), modem, large format plotter, (note that some hardware may not be site-based).

- Software: Trimble Pathfinder, AutoCAD (or equivalent CAD software) with coordinate geometry capability, ARCInfo (or equivalent geographic information system [GIS] software), access to GIS spatial information data files (via the World Wide Web).

The survey will be performed following MARSSIM protocol by walking straight parallel lines over an area while moving the detector in a serpentine motion, 2 inches to 4 inches above the ground surface. Survey passes will be approximately one meter apart. Data from the ratemeter/scaler will be automatically logged into the GPS unit every 1 second. After completion of the survey, the data will be downloaded from the GPS into a PC file and entered into a geospatial software program. After completion of data processing, a PC file with the contoured results of the survey will be presented to the CABRERA Project Engineer (PE) for evaluation.

3.6.2 Soil Sampling

Surface soil samples will be collected from 0 to 12 inches below ground surface at each sample location to facilitate statistical evaluation of site radionuclide concentrations relative to the $DCGL_W$, in accordance with NUREG-1757 Volume 2 (NRC, 2003b). Soil samples will also be collected from 12 to 24 inches below ground surface and analyzed only if the surface soil samples identify ROC activity significantly greater than background. Soil samples will be collected in accordance with CABRERA Standard Operating Procedure OP-005, Volumetric and Material Sampling and will be shipped by CABRERA to a certified analytical laboratory for radionuclide analysis. Soil samples will be collected by using a hand auger.

4.0 DATA QUALITY OBJECTIVES

4.1 Step 1: Sate the Problem

4.1.1 Problem Description

The objective of FSS activities is to obtain data of sufficient quality and quantity to support unrestricted release of the Vault area following decommissioning activities.

4.1.2 Planning Team Members

FSS planning is being performed by a team of CABRERA personnel, with input and direction from AFSC, SBCCOM, and the NRC.

4.1.3 Primary Decision Maker

The ultimate decision regarding site disposition will rest with the NRC. SBCCOM will work with the NRC in support of activities required to accomplish the Vault decommissioning and release for unrestricted use.

4.1.4 Available Resources

Sufficient resources are available through the combined staff of SBCCOM, AFSC, CABRERA, and CABRERA subcontractors, to perform and complete all work required to achieve FSS objectives.

4.2 Step 2: Identify the Decision

4.2.1 Principal Study Question

Do concentrations of ROCs at the site exceed background concentrations by more than the $DCGL_w$ (SOR of 1.0) following remedial activities and, if so, where are elevated concentrations located?

4.2.2 Decision Statement

The following statements assume that ROC concentrations in the Class 1 SU will be found to exceed an SOR of 1.0, following remedial activities. Decision statements should be evaluated sequentially, as shown below.

(A) Determine whether SU SORs exceed background SORs by more than the $DCGL_w$

(B) Based on sample results, if SU SORs exceed background SORs by more than the $DCGL_w$, recommend whether further excavation is required within the SU.

4.3 Step 3: Identify Inputs to the Decision

A variety of data are required to resolve the decision statements listed above. This section lists data needs, describes the sources of that data, and discusses the means of obtaining the required data points.

4.3.1 Information Inputs:

The following site characteristics must be determined to resolve applicable decision statements:

(A) Concentrations of residual radioactive material in the SU:

This information will allow determination as to whether or not a SU is likely to be suitable for release for unrestricted use. Obtaining this data will facilitate cost effective decision-making regarding the project's direction and duration. This data will be used to calculate SU SORs for comparison against the $DCGL_W$.

(B) Information Sources:

The GWS and volumetric sample analysis data will provide sufficient information to enable estimation of SU radionuclide concentrations and to identify areas of elevated gamma fluence.

4.4 Step 4: Define the Study Boundaries

4.4.1 Population of Interest Defining Characteristics:

The population of interest for the site is the concentration of ROCs and their associated SORs in surface and shallow subsurface soils.

4.4.2 Spatial Boundaries of the Decision Statement:

The population of interest is horizontally limited to land areas located within the Vault excavation area and surrounding soils (Class 1 SU). The vertical study area extends from the land surface to the depth of up to six inches below ground surface.

4.4.3 Temporal Boundaries of the Decision Statement

(A) Time frame to which the decision applies:

$DCGL_W$ values are based on risks to an average member of the Critical Group over a 1,000-year period following the study.

(B) Time for data collection:

Data collection and analysis should be performed as soon as practical, as timely completion of the site restoration is contingent upon the results of the FSS.

4.4.4 Scale of Decision Making:

Decisions will be made for small areas in the Class 1 SU that may exhibit elevated levels of radioactivity, then for the entire SU regarding whether or not it meets the criteria for unrestricted release.

4.4.5 Constraints on Data Collection:

Data collection activities can be constrained due to excessive moisture or rain, which can have an adverse effect on field instrumentation.

4.5 Step 5: State the Decision Rules

4.5.1 Parameter of Interest

Parameters of interest are the mean, median, and standard deviation of data collected during the study. Based on the data distribution characteristics resulting from FSS data collection, the preceding parameters may be transformed to equivalent descriptive measures (e.g., logarithms, etc.) to allow more representative statistical testing. By using a graded approach to data testing as discussed below, decisions will be made according to the decision rule stated at the end of this section.

4.5.2 Scale of Decision Making

Decisions are made on two fundamental scales, for the SU and for smaller localized areas of elevated activity. Localized areas of elevated radiation levels are evaluated on an ongoing basis throughout the field effort. In cases where clear indications of elevated measurements are observed, decisions on remediation, SU subdivision, etc., may be taken as appropriate. On a larger scale, and as a final determination, data will be evaluated on a SU-specific basis.

4.5.3 Action Level

Decisions on a SU's acceptability for release are based on comparison of the $DCGL_w$ to the difference between measured residual radioactivity concentrations in SU and measured radioactivity in the reference area, subject to applicable statistical analyses specified in MARSSIM. Inputs to this decision will be based on a graded approach to data analysis intended to avoid unnecessary analytical and/or remediation efforts, while also ensuring that project data quality objectives are met.

4.5.4 Decision Inputs

Geospatial modeling of position-correlated GWS data will provide a graphical view of surface gamma radiation levels and will be updated as the FSS progresses. These data will serve as the primary decision inputs during performance of the fieldwork because data will be reduced soon after collection, in comparison to the longer turn around time associated with laboratory sample analyses.

Assessment of soil sample data will be as simple as visually inspecting data to identify obvious indicators that the action level has or has not been met. If all sample SORs are below the $DCGL_w$ of SOR = 1.0, the SU will meet the release criteria. If not, the WRS test will be applied to the data.

(A) Field Measurements of Survey Unit Dimensions

The dimensions of the SU will be measured using a measuring tape, GPS, or other accurate means of measurement, once the boundaries are established. At a minimum, the corners of the SU will be logged using the GPS system. The area of the SU will then be calculated in units of m^2; GPS data will be entered into a CAD program to support SU area calculations. These data will be used to determine sampling locations.

(B) Gamma Walkover Survey in Reference Area

Soon after the background (reference) area is established, a GWS will be performed. These data will be used, in part, to make decisions regarding the radiological status of the Vault area SU. Reference Area GWS data will be reduced and evaluated as follows:

- The average and standard deviation of the GWS data will be calculated. This will be used to evaluate SU data
- The measurement results will be plotted and color-coded for visual review and evaluation
- The Z score for each data point will be plotted and color-coded for visual review and evaluation
- These data will be reviewed for obvious anomalies to determine if the chosen reference area is acceptable

(C) Gamma Walkover Survey in Survey Units

GWS of the SU will begin after completion of the GWS on the reference area, calculation of the mean and standard deviation, and determination that the reference area is acceptable. SU GWS data will be reduced and evaluated as follows:

- The measurements will be plotted and color-coded for visual review and evaluation. The average and standard deviation of the SU will also be calculated. The coordinates of the highest measurement will be identified.
- The Z score for each data point will be plotted and color-coded for visual review and evaluation. All areas exceeding three standard deviations above the average (i.e., Z score greater than or equal to 3.0) will be identified. The frequency of these occurrences and the maximum measurement in these areas will be compared to the Reference Area. The geospatial plot will also be visually inspected to identify anomalies in the distribution of measurement data.
- These data will be reviewed to make a preliminary determination of whether a SU has a low probability of passing the MARSSIM statistical tests.

(D) Volumetric Results: Wilcoxon Rank Sum Statistical Test

Comparison of reference area (background) radionuclide concentrations with SU concentrations will be performed using the two-sample WRS statistical test. This test is selected because certain ROCs are present in natural background. The two-sample WRS test assumes the reference area and SU data distributions are similar except for a possible shift in the medians. When the data are severely skewed, the value for the mean difference between SU measurements and reference measurements may be above the $DCGL_W$, while the median difference is below the $DCGL_W$. In such cases, the SU does not meet the release criterion regardless of the result of the statistical test. On the other hand, if the difference between the largest SU measurement and the smallest reference area measurement is less than the $DCGL_W$, the WRS test will always show that the SU meets the release criterion.

In using this test, the hypotheses being tested are:

> *Null Hypothesis (H_o)*: The median concentration in the SU exceeds that in the reference area by more than the $DCGL_W$.
>
> versus the alternative:
>
> *Alternative Hypothesis (H_a)*: The median concentration in the SU exceeds that in the reference area by less than the $DCGL_W$.

The WRS should be applied to sample data via the following sequential steps:

1) Reduce Reference Area and SU isotopic data to SORs using the equation presented in Section 2.
2) Add the $DCGL_W$ value (i.e., 1.0) to each reference area SOR value, Xi, to obtain the adjusted reference area SORs, Zi. (Zi = Xi + 1.0).
3) The *m* adjusted SORs, Z_i, from the reference area and the *n* SORs, Y_i, from the SU are pooled and assigned a rank in order of increasing measurement value from *I* to *N*, where $N = m + n$.
4) If several SORs are tied (have the same value), they are all assigned the average rank of that group of tied measurements.
5) Sum the ranks of the adjusted SORs from the reference area, W_r. Since the sum of the first *N* integers is $N (N+1)/ 2$, one can equivalently sum the ranks of the SORs from the SU, W_s, and calculate $W_r = (N (N+1)/ 2)-W_s$.
6) Compare W_r with the critical value given in Table I.4 in MARSSIM for the approximate values of *n*, *m*, and α. If W_r is greater than the tabulated value, reject the H_0 that the SU exceeds the release criterion.

4.5.5 Decision Rules

(A) Gamma Walkover Survey in Reference Area

If review of reference area GWS data indicates that the chosen area exhibits excessive variance or appears to be impacted by radiological or non-radiological activities (e.g., phosphate fertilizer, fossil fuel wastes, etc.), biased sample measurements will be performed to support the area's non-impacted designation.

(B) Gamma Walkover Survey in Survey Units

- A biased soil sample will be collected at the location where the highest GWS data point is observed.
- If areas exceeding three standard deviations above the average are observed (i.e., a Z score greater than or equal to 3.0), additional biased soil samples may be collected at the discretion of the CABRERA Project Manager (PM), or designee.

(C) Volumetric Results: Wilcoxon Rank Sum Statistical Test

- If all sample results for the SU have associated SORs that are less than the $DCGL_w$ (i.e., SOR = 1.0), the SU is deemed to meet the release criterion.
- If any of the SORs for the SU exceeds the $DCGL_w$, perform the WRS test:
- If W_r, the sum of the adjusted reference area ranks from the WRS test, is greater than the applicable critical value, then the median value for residual radioactivity in the SU is less than the $DCGL_w$ to the specified confidence level. In this case, the H_o is rejected and the SU meets the release criterion. If W_r is less than the critical value, the H_o is accepted and the SU does not meet the release criteria.

4.6 Step 6: Define Acceptable Decision Errors

NRC guidance (NRC, 1998b) provides a discussion regarding decision errors. This discussion includes the concept that acceptable error rates, which balance the need to make appropriate decisions with the financial costs of achieving high degrees of certainty, must be specified.

> *Errors can be made when making site remediation decisions. The use of statistical methods allows for controlling the probability of making decision errors. When designing a statistical test, acceptable error rates for incorrectly determining that a site meets or does not meet the applicable decommissioning criteria must be specified. In determining these error rates, consideration should be given to the number of sample data points that are necessary to achieve them. Lower error rates require more measurements, but result in statistical tests of greater power and higher levels of confidence in the decisions. In setting error rates, it is important to balance the consequences of making a decision error against the cost of achieving greater certainty.*

Acceptability decisions are often made based on acceptance criteria. If the mean and median concentrations of a contaminant are less than the associated acceptance criteria, for example, the results can usually be accepted. In cases where data results are not so clear, statistically based decisions are necessary. Statistical acceptability decisions, however, are always subject to error. Two possible error types are associated with such decisions.

The first type of decision error, called a Type I error, occurs when the H_o is rejected when it is actually true. A Type I error is sometimes called a "false positive." The probability of a Type I error is usually denoted by α. Considered in light of H_o used for this site (discussed above), this error could result in higher potential doses to future site occupants than prescribed by the dose-based criterion.

The second type of decision error, called a Type II error, occurs when the H_o is not rejected when it is actually false. A Type II error is sometimes called a "false negative." The probability of a Type II error is usually denoted by β. The power of a statistical test is defined as the probability of rejecting the null hypotheses when it is false. It is numerically equal to 1-β where β is the Type II error rate. Consequences of Type II errors include unnecessary remediation expense and project delays.

For the purposes of the FSS, the acceptable error rate for both Type I and Type II errors is five percent (i.e., $\alpha = \beta = 0.05$).

5.0 METHODOLOGY AND APPROACH TO PERFORMING SURVEYS

5.1 Gamma Walkover Survey Utilizing Global Positioning System

5.1.1 Estimated Scan Sensitivity

A gamma walkover scan survey will be performed following the remediation of the Vault in order to identify areas exhibiting elevated gamma fluence. The GWS will be performed as described in Section 3.6. MARSSIM Section 6.7.2.1 describes the methodology used to calculate scan MDCs for land areas that are delineated in MARSSIM Table 6.7.

Minimum detectable concentrations (MDC) have been calculated for each gamma emitting ROC, assuming an initial scan speed of 0.5 meters per second (m/s) and a minimum contaminated area 56 cm in diameter and 15 cm in depth. The scan MDC calculations are presented as Attachment 1. The ROC scan MDCs calculated for a Field Instrument for the Detection of Low-Energy Radiation (FIDLER) sodium iodide (NaI) are as follows:

- For Am-241 (in equilibrium with decay progeny), the scan MDC is 1.5 pCi/g.
- For Cs-137 (in equilibrium with decay progeny), the scan MDC is 3.3 pCi/g.
- For Th-232 (in equilibrium with decay progeny), the scan MDC is 0.46 pCi/g.

The MARSSIM MDC analysis is based on detecting a contaminated area that is 56 cm in diameter. The number of side-to-side detector passes performed per meter must be sufficient to detect this area. Three detector passes per meter, evenly spaced (approximately every 33 cm) will ensure that the detector crosses a minimum distance of 45 cm across the contaminated area. This provides a minimum observation interval of 0.9 seconds (i.e., 0.45 m ÷ 0.5 m/s) compared to the 1 second used in MARSSIM Section 6.7.2.1. The difference in MDC between a 1 second and 0.9 second observation interval is readily determined as 1 over the square root of 0.9 (i.e., 1 ÷ 0.987) or 1.053, from equations 1 and 2 of MARSSIM Section 6.7.2.1. This demonstrates that the MDC for the minimum 45 cm pass across the contaminated area will be 105.3% of the MDC calculated for the 56 cm pass.

5.1.2 Global Positioning System Setup

The GPS system will provide high quality, precision geospatial positioning data to support data verification, and remediation. The rate-meter/scalars used for this work plan will be configured to output directly to the GPS unit. The GPS unit will perform data logging functions.

In order for the GPS unit to achieve sub-meter accuracy, differential position correction is necessary. Two methods for doing this are: (1) post-processing differential correction and (2) real-time differential correction. Essentially, the difference between the two is the time during which the corrections are made. Real-time differential correction is possible if the site being considered is located within range of a United States Coast Guard (USCG) GPS differential correction beacon. These beacons are located in coastal areas of the U.S. Real-time correction does not require a GPS base station. Based upon manufacturers' recommendations, the Property is likely to be within range of a USCG beacon, allowing for real-time differential correction. However, if for some reason the USCG beacon cannot be received by the GPS system, post-processing differential correction will be used.

5.1.3 Survey Limitations

Although the GPS unit identifies its position using the signals from several satellites, GPS positioning may be affected by overhead obstructions. A loss of satellite signal due to these obstructions may prevent collection of location data, depending on the severity of the loss and the positional filter settings in use in the GPS unit. If this occurs, data collection will not resume until satellite lock is regained (usually by moving past the obstruction) and the positional filter requirements are satisfied. If the signal is lost during a survey, the operator shall continue to walk at constant velocity in a straight line until satellite lock has been reestablished or until a boundary is reached. In such cases, due to positional filter settings in the GPS unit, no gamma logging occurs. In that event, gamma readings must be taken by hand. The surveyor will need to inform the data processing specialist if the gamma count rates between pairs of GPS positional data changes considerably. Such information will be logged in project logs as appropriate.

Extrapolation of gamma data positions beyond good GPS locations requires additional post-processing programs or hand editing of data. It is desirable, therefore, to begin and end a survey path with good GPS positions. The survey crew shall extend the beginning or end of a survey path (in a straight line) beyond a designated boundary in order to obtain satellite lock, if necessary. On occasion, it may not be possible to get a good satellite lock because of satellite positions in the sky or technical problems with the satellite system. In this case, a short wait (e.g., one-half hour) is usually sufficient to regain satellite lock. If necessary, survey paths without good satellite locks will be repeated and/or hand surveyed and located.

5.2 Data Processing

5.2.1 Field Records

Project data will be recorded in a Project Log Book. Field Log Book records will be sufficient to allow data transactions to be reconstructed after the project is completed. One designated Project Log Book will be used during the field effort. The PE is responsible to ensure logbook entries are made as necessary and appropriate. The PE will review the Project Log Book at least daily and will report significant issues to CABRERA's PM, or designee.

Field Log Books are used for each survey team and/or equipment used during the field effort. Multiple field data logbooks are acceptable to use, as long as they are assigned to individual FSS teams and/or equipment. The PE is responsible to ensure logbook entries are made as necessary and appropriate. The PE will review Field Log Book(s) at least daily and will report significant issues to CABRERA's PM, or designee.

Field Log Books are considered legal records. Log Books will be permanently bound and the pages will be numbered. Pages may not be removed from Log Books under any circumstances. Entries will be legible, factual, detailed, and complete and will be signed and dated by the individual(s) making the entries. If a mistake is made, the error shall be denoted by placing a single line through the erroneous entry and initialing the deletion. Under no circumstances will any previously entered information be completely obliterated. Use of whiteout in Log Books is not permitted for any reason.

5.2.2 Electronic Data

Electronic data collected during the day will be backed-up at the end of the same day in the field (e.g., to tape or zip drive) and before processing or editing. This is an archive of the raw data and, once created, will not be altered. More than one day's data may go on a single tape or zip disk. Field computer(s) used to store GPS data will be backed up weekly. Raw archived data will be stored in a different location from weekly backups. Electronic GPS data will be provided daily to off-site data processing specialists. The time and date that data files are transmitted will be recorded in the data logbook. File names will be verified by comparison with field notes and corrected if necessary, following approval by the CABRERA Radiation Safety Officer (RSO).

5.2.3 Post Processing

Post-processing specialists will convert daily GWS/GPS data to state plane coordinates, as necessary, and review the data for errors to fluctuations/interferences in the GPS signal. Post-processing specialists will inform the CABRERA RSO, or designee of any identified deficiencies and will make corrections as directed. All conversions, errors, corrections, and/or adjustments to project data shall be documented in the data logbook.

6.0 SURVEY QUALITY ASSURANCE/QUALITY CONTROL

Activities associated with this work plan shall be performed in accordance with written procedures and/or protocols in order to ensure consistent, repeatable results. Topics covered in project procedures and protocols may include proper use of instrumentation, Quality Control (QC) requirements, equipment limitation, etc. Implementations of Quality Assurance (QA) measures for this work plan are described herein.

6.1 Instrumentation Requirements

The PE is responsible for determining the instrumentation required to complete the requirements of this work plan. Only instrumentation approved by the PE will be used to collect radiological data. The PE is responsible for ensuring individuals are appropriately trained to use project instrumentation and other equipment, and that instrumentation meets the required detection sensitivities. Instrumentation shall be operated in accordance with either a written procedure or manufacturers' manual, as determined by the PE. The procedure and/or manual will provide guidance to field personnel on the proper use and limitations of the instrument.

6.1.1 Calibration Requirements

Instruments used during the FSS shall have current calibration/maintenance records kept on site for review and inspection. The records will include, at a minimum, the following:

- name of the equipment
- equipment identification (model and serial number)
- manufacturer
- date of calibration
- calibration due date

Instrumentation shall be maintained and calibrated to manufacturers' specifications to ensure that required traceability, sensitivity, accuracy and precision of the equipment/instruments are maintained. Instruments will be calibrated at a facility possessing appropriate NRC and/or Agreement State licenses for performing calibrations using National Institutes of Standard Technology (NIST) traceable sources.

6.1.2 Sodium Iodide Quality Control Source and Background Checks

Prior to and after daily use, instruments will be QC checked by comparing the instruments' response to a designated gamma radiation source and to ambient background. Prior to the commencement of field operations, a site reference location shall be selected for performance of these checks. QC source checks will consist of a one-minute integrated or scaler count with the designated source positioned in a reproducible geometry, performed at the reference location. Background checks will be performed in an identical fashion with the source removed. The results of the background and QC checks will be recorded in a field logbook. At the start of the field activities, this procedure will be repeated at least three times to establish average instrument response.

Instrument response to the designated QC check source will be evaluated against the average established at the start of the field activities. A performance criterion of ± 20% of this average will be used as an investigation action level. The PE will investigate results exceeding this criterion and will make appropriate corrections if response is deviated by factors beyond personnel control, such as large humidity or temperature changes. The PE has authority to decide whether or not the instrument is acceptable for use or must be removed from service.

Instrument response to ambient background will be used to establish a mean background response for each instrument, to monitor gross fluctuations in background activity (e.g., from changes in barometric pressure and other, non-contaminant related causes), and to evaluate detector response. The background measurements are made solely for the purpose of normalizing each day's survey results and eliminating bias introduced by natural fluctuations in site radiological conditions, if necessary. Given the qualitative or semi-quantitative nature of this survey, no attempt will be made to subtract naturally occurring radioactivity from survey data to derive net activity.

During QC checks, instruments used to obtain radiological data should be inspected for physical damage, current calibration and erroneous readings in accordance with applicable procedures and/or protocols. The individual performing these tasks shall document the results in accordance with the associated instrument procedure and/or protocols. Instrumentation that does not meet the specified requirements of calibration, inspection, or response check will be removed from operation. If the instrument fails the QC response check, any data obtained to the point, but after the last successful QC check will be considered invalid due to faulty instrumentation.

6.2 Global Positioning System Requirements

6.2.1 Daily Field Checks

Section 6.1.2 describes performance of sodium iodide source and ambient background response checks. These checks are always performed at the same location and are logged in the field logbook. A reference location will also be established for the GPS system. At the start of the field effort, when average sodium iodide source and ambient background response is established, the average easting and northing GPS position data will be established for this reference location. A minimum of three measurements will be used to establish the average response of the GPS system. During subsequent routine checks, GPS position data will be compared to the established averages and recorded in the field logbook. Measurements differing by more than one meter from this average will be investigated and necessary corrective actions will be implemented.

6.2.2 Quality Control of the Field Survey

Data quality control will be accomplished with mapping control points, viewing plotted survey data, and keeping detailed field notes. Mapping control points (a discrete point at a known location such as in the corner of a base map building) will ensure that the area surveyed will overlay with existing maps. Gamma surveys, when plotted, should exhibit the same configuration as shown in annotated field sketches and field notes. Any anomalies observed by the data processing specialist and/or technicians performing field surveys shall be brought to the attention of the PE.

6.3 Soil Sampling

6.3.1 Duplicate Samples

Soil samples will be sent to a certified analytical laboratory for gamma spectroscopic analysis. Duplicate samples will be required for 10% of samples sent for analysis. When duplicate analysis is required, two samples will be obtained from the same homogenous mixture. The samples will be numbered using unique identifiers and will be sent to the laboratory for analysis. Additionally, the analytical laboratory should perform duplicate laboratory analyses on selected samples as specified in their quality assurance procedures. Analyses of field and laboratory duplicates will be compared to the initial analytical results by determining a Z score value for each data set by the following equation:

$$Z = \frac{|S - D|}{\sqrt{\sigma_S^2 + \sigma_D^2}}$$

Where: S, D, $\equiv$ value of (S)ample and (D)uplicate measurements; and,
σ $\equiv$ one sigma error associated with (S)ample and (D)uplicate measurements.

The calculated Z score results will be compared to a performance criteria of less than or equal to 2.57. The value of 2.57 corresponds to a 99% confidence level, or, 99% of the Z score values will be below 2.57, and only 1% of the values will be above this acceptance criteria, if the sample and the duplicate are truly of the same distribution. Calculated Z score values less than 2.57 will be considered acceptable and values greater than 2.57 will be investigated for possible discrepancies in analytical precision, or for sources of disagreement with the following assumptions of the test:

- the sample measurement and duplicate or replicate measurement are of the same normally distributed population
- the standard deviations, σ_S and σ_D, represent the true standard deviation of the measured population

6.3.2 Laboratory Spike Analyses

Spike analyses may be performed by the laboratory and used to estimate the extent of bias in the analytical measurements. The analytical laboratory adds a known quantity of radioactive material, or analyte, to representative media, and analyzes the spiked media. Bias in the results will be quantified by determining percent difference values for spike analyses data provided by the laboratory. Percent difference values will be determined by the following equation:

$$\% \text{ Difference} = \frac{(C_S - C_M)}{C_S} * 100$$

Where: $C_S \equiv$ Concentration of spike analyte added to sample

$C_M \equiv$ Measured concentration of analyte in sample

Percent differences will be compared to a performance criteria of 20%. Percent differences greater than 20% will be investigated for possible discrepancies in measurement bias. The error associated with the measured values should be a consideration when evaluating percent differences and qualifying data which do not meet these performance criteria.

6.3.3 Laboratory Blanks

The analytical laboratory performs blank analyses to test analytical accuracy and to estimate the extent of bias in the measurements. Laboratory blanks are also used to demonstrate that laboratory contamination is not the cause of reported analytical results. A blank sample is prepared and analyzed by the analytical laboratory and typically consists of a sample of similar media, free of radiological contamination, which remains with the field sample throughout the entire analytical procedure and analyzed to determine the concentration of the radionuclide of concern. Blank analyses should be performed in accordance with the laboratory's quality assurance procedures. If blank results reported by the laboratory indicate the presence of contamination above the detection limit, or results are not qualified, data should not be used.

7.0 REFERENCES

(NRC, 1998) Federal Register Notice Volume 63, No. 222, U.S. Nuclear Regulatory Commission, dated November 18, 1998

(NRC, 1998b) NUREG-1505, Rev.1, *A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys*, U.S. Nuclear Regulatory Commission, dated June, 1998.

(NRC, 1999) Federal Register Notice Volume 64, No. 234, U.S. Nuclear Regulatory Commission, dated December 7, 1999

(NRC, 1999b) NUREG/CR-5512, Volume 3, U.S. Nuclear Regulatory Commission, dated October 1999.

(NRC, 2000) Federal Register Volume 65, # 114, June 13, 2000

(NRC, 2000b) Multi-Agency Radiological Site and Survey Investigation Manual (MARSSIM). NUREG-1575. EPA 402-R-97-016). August, 2000.

(NRC, 2003) U.S. Nuclear Regulatory Commission Materials License No. 45-00953-01, Docket No. 030-06511, Amendment 42, Department of the Army, U.S. Army Soldier and Biological Chemical Command, dated April 17, 2003.

(NRC, 2003b) NUREG-1757, Volumes 1 and 2, *Consolidated NMSS Decommissioning Guidance*, U.S. Nuclear Regulatory Commission, dated September, 2003.

Final Status Survey Plan

Attachments

Fidler Scan for Am-241 @ 1pCi/g, No Soil Cover, 15 cm thick x 28 cm RADIUS
Fluence rate to exposure rate (FRER, no units) = ~ (1 uR/h)/(Eγ)(u_{en}/ρ)air

TABLE 1

Energy$_\gamma$, keV	$(u_{en}/\rho)_{air}$, cm^2/g	FRER
15	1.29	0.0517
20	0.516	0.0969
30	0.147	0.2268
40	0.064	0.3906
50	0.0384	0.5208
60	0.0292	0.5708
80	0.0236	0.5297
100	0.0231	0.4329
150	0.0251	0.2656
200	0.0268	0.1866
300	0.0288	0.1157
400	0.0296	0.0845
500	0.0297	0.0673
600	0.0296	0.0563
800	0.0289	0.0433
1,000	0.0280	0.0357
1,500	0.0255	0.0261
2,000	0.0234	0.0214

Probability of interaction (P) through end of detector for given energy is

Probability = 1-e$^{-(\mu/\rho)\text{NaI}\ (x)\ (\rho\text{NaI})}$

TABLE 2

Energy$_\gamma$, keV	$(\mu/\rho)_{NaI}$, cm^2/g	P
15	47.4	1.00
20	22.3	1.00
30	7.45	0.99
40	19.3	1.00
50	10.7	1.00
60	6.62	0.98
80	3.12	0.84
100	1.72	0.64
150	0.625	0.31
200	0.334	0.18
300	0.167	0.09
400	0.117	0.07
500	0.0955	0.05
600	0.0826	0.05
800	0.0676	0.04
1,000	0.0586	0.03
1,500	0.0469	0.03
2,000	0.0413	0.02

for Fidler 12.7cm dia x 0.16 cm thick NaI crystal

x = 0.16 cm

ρ = 3.67 g/cm^3

beryllium window per Fidler catalog 0.010 inch

Relative Detector Response (RDR) = relative fluence-to-exposure rate (FRER) times probability (P) of interaction

TABLE 3

Energy$_\gamma$, keV	FRER	P	RDR
15	0.0517	1.00	0.0517
20	0.0969	1.00	0.0969
30	0.2268	0.99	0.2239
40	0.3906	1.00	0.3906
50	0.5208	1.00	0.5199
60	0.5708	0.98	0.5591
80	0.5297	0.84	0.4449
100	0.4329	0.64	0.2752
150	0.2656	0.31	0.0816
200	0.1866	0.18	0.0332
300	0.1157	0.09	0.0108
400	0.0845	0.07	0.0056
500	0.0673	0.05	0.0037

600	0.0563	0.05	0.0027
800	0.0433	0.04	0.0017
1,000	0.0357	0.03	0.0012
1,500	0.0261	0.03	0.0007
2,000	0.0214	0.02	0.0005

Estimated Fidler 12.7cm dia x 0.16cm thick NaI response for Cs-137 is 1287 cpm/uR/hr

based on Cabrera 12/3-4/01; 20 measurements @ 5.3 urem/hr bkg

Use same methodology and interpolating for Cs-137 response have:

Energy$_\gamma$, keV	$(u_{en}/\rho)_{air}$, cm^2/g		
662	0.0294	FRER ~	0.0514

Energy$_\gamma$, keV	$(\mu/\rho)_{NaI}$, cm^2/g		
662	0.0780	Probability =	0.04
		RDR =	0.0023

For this detector the response to another energy is based on the ratio of the relative detector response, RDR to the Cs-137 energy

cpm/μR/h, $E_i = (cpm_{cs\text{-}137})*(RDR_{Ei})/(RDR_{Cs\text{-}137})$

TABLE 4

Energy$_\gamma$, keV	RDR_{Ei}	Fidler NaI Detector, E_i, cpm per μR/hr
15	0.0517	28934
20	0.0969	54250
30	0.2239	125355
40	0.3906	218695
50	0.5199	291052
60	0.5591	313006
80	0.4449	249068
100	0.2752	154090
150	0.0816	45680
200	0.0332	18602
300	0.0108	6053
400	0.0056	3140
500	0.0037	2056
600	0.0027	1493
662	0.0023	1287
800	0.0017	942
1,000	0.0012	676
1,500	0.0007	398
2,000	0.0005	287

MDC for Cs-137 energy

Assume 10 μR/hr bkg then have 12,870 cpm

b_i =	214.5	counts
MDCR =	1212.673773	cpm
$MDCR_{surveyor}$ =	1715	cpm

minimum detectable exposure rate = 1.33 μR/hr

Table 5

keV	MicroShield Exposure Rate, μR/hr (with buildup)	cpm/μR/hr	cpm/μR/hr (weighted)	Percent of NaI detector response
15	0.000E+00	28934	0	0.0%
20	0.000E+00	54250	0	0.0%
30	8.059E-05	125355	2788	0.9%
40	0.000E+00	218695	0	0.0%
50	0.000E+00	291052	0	0.0%
60	3.519E-03	313006	304008	98.6%
80	2.357E-05	249068	1620	0.5%
100	0.000E+00	154090	0	0.0%
150	0.000E+00	45680	0	0.0%
200	0.000E+00	18602	0	0.0%

300	0.000E+00	6053	0	0.0%
400	0.000E+00	3140	0	0.0%
500	0.000E+00	2056	0	0.0%
600	0.000E+00	1493	0	0.0%
800	0.000E+00	942	0	0.0%
1000	0.000E+00	676	0	0.0%
1500	0.000E+00	398	0	0.0%
2000	0.000E+00	287	0	0.0%
Total	3.623E-03		308416	100%

Minimum Detectable Exposure Rate =

MDCR surveyor/(cpm/μr/hr)
0.0056 μr/hr

and MDC for total uranium and 50-year equilibrium progeny based on a normalized 1 pCi/g total uranium

Scan MDC = (Assumed MDC Am241Conc) x (Exposure Rate $MDCR_{Surveyor}$)/(Exposure $Rate_{assumed\ Am241Conc}$)

Scan MDC = 1.53 pCi/g
56.8 Bq/kg

Fidler Scan for Cs-137 @ 1pCi/g, No Soil Cover, 15 cm thick x 28 cm RADIUS
Fluence rate to exposure rate (FRER, no units) = ~ (1 uR/h)/(Eγ)(u_{en}/ρ)air

TABLE 1

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15	1.29	0.0517
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80	0.0236	0.5297
100	0.0231	0.4329
150	0.0251	0.2656
200	0.0268	0.1866
300	0.0288	0.1157
400	0.0296	0.0845
500	0.0297	0.0673
600	0.0296	0.0563
800	0.0289	0.0433
1,000	0.0280	0.0357
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Probability of interaction (P) through end of detector for given energy is

Probability = $1\text{-}e^{-(\mu/\rho)NaI\ (x)\ (\rho NaI)}$

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200	0.1866	0.18	0.0332
300	0.1157	0.09	0.0108
400	0.0845	0.07	0.0056
500	0.0673	0.05	0.0037
600	0.0563	0.05	0.0027
800	0.0433	0.04	0.0017
1,000	0.0357	0.03	0.0012

1,500	0.0261	0.03	0.0007
2,000	0.0214	0.02	0.0005

Estimated Fidler 12.7cm dia x 0.16cm thick NaI response for Cs-137 is 1287 cpm/uR/hr
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662	0.0294	FRER ~	0.0514

Energy_γ, keV	$(\mu/\rho)_{NaI}$, cm^2/g		
662	0.0780	Probability =	0.04
		RDR =	0.0023

For this detector the response to another energy is based on the ratio of the relative detector response, RDR to the Cs-137 energy
cpm/μR/h, E_i = $(cpm_{cs\text{-}137})*(RDR_{Ei})/(RDR_{Cs\text{-}137})$

TABLE 4

Energy_γ, keV	RDR_{Ei}	Fidler NaI Detector, E_i, cpm per μR/hr
15	0.0517	28934
20	0.0969	54250
30	0.2239	125355
40	0.3906	218695
50	0.5199	291052
60	0.5591	313006
80	0.4449	249068
100	0.2752	154090
150	0.0816	45680
200	0.0332	18602
300	0.0108	6053
400	0.0056	3140
500	0.0037	2056
600	0.0027	1493
662	0.0023	1287
800	0.0017	942
1,000	0.0012	676
1,500	0.0007	398
2,000	0.0005	287

MDC for Cs-137 energy

Assume 10 μR/hr bkg then have 12,870 cpm

b_i =	214.5	counts
MDCR =	1212.673773	cpm
$MDCR_{surveyor}$ =	1715	cpm

minimum detectable exposure rate = 1.33 μR/hr

Table 5

keV	MicroShield Exposure Rate, μR/hr (with buildup)	cpm/μR/hr	cpm/μR/hr (weighted)	Percent of NaI detector response
15	0.000E+00	28934	0	0.0%
20	0.000E+00	54250	0	0.0%
30	2.178E-04	125355	113	6.8%
40	5.958E-05	218695	54	3.2%
50	0.000E+00	291052	0	0.0%
60	0.000E+00	313006	0	0.0%
80	0.000E+00	249068	0	0.0%
100	0.000E+00	154090	0	0.0%
150	0.000E+00	45680	0	0.0%
200	0.000E+00	18602	0	0.0%
300	0.000E+00	6053	0	0.0%
400	0.000E+00	3140	0	0.0%
500	0.000E+00	2056	0	0.0%
600	2.421E-01	1493	1491	90.0%
800	0.000E+00	942	0	0.0%
1000	0.000E+00	676	0	0.0%

1500	0.000E+00	398	0	0.0%
2000	0.000E+00	287	0	0.0%
Total	2.424E-01		1657	100%

Minimum Detectable Exposure Rate =

MDCR surveyor/(cpm/μr/hr)
1.0349 μr/hr

and MDC for Cs-137 based on a normalized 1 pCi/g

Scan MDC = (Assumed MDC Cs137 Conc) x (Exposure Rate $MDCR_{Surveyor}$)/(Exposure $Rate_{assumed\ Cs137Conc}$)

Scan MDC = 4.27 pCi/g
158.0 Bq/kg

Fidler Scan for Th232 Nat @ 1pCi/g, NO SOIL COVER 15 cm thick x 28 cm RADIUS
Fluence rate to exposure rate (FRER, no units) = ~ (1 uR/h)/(Eγ)(u_{en}/ρ)air

TABLE 1

Energy$_γ$, keV	(u_{en}/ρ)$_{air}$, cm^2/g	FRER
40	0.064	0.3906
60	0.0292	0.5708
80	0.0236	0.5297
100	0.0231	0.4329
150	0.0251	0.2656
200	0.0268	0.1866
300	0.0288	0.1157
400	0.0296	0.0845
500	0.0297	0.0673
600	0.0296	0.0563
800	0.0289	0.0433
1,000	0.0280	0.0357
1,500	0.0255	0.0261
2,000	0.0234	0.0214
3,000	0.0205	0.0163

Probability of interaction (P) through end of detector for given energy is

Probability = $1-e^{-(\mu/\rho)NaI\ (x)\ (\rho NaI)}$

TABLE 2

Energy$_γ$, keV	(μ/ρ)$_{NaI}$, cm^2/g	P
40	18.3	1.00
60	6.23	0.97
80	2.86	0.81
100	1.58	0.60
150	0.566	0.28
200	0.302	0.16
300	0.153	0.09
400	0.11	0.06
500	0.0904	0.05
600	0.079	0.05
800	0.0657	0.04
1,000	0.0576	0.03
1,500	0.0464	0.03
2,000	0.0412	0.02
3,000	0.0367	0.02

for Fidler 12.7cm dia x 0.16 cm thick NaI crystal
x = 0.16 cm
ρ = 3.67 g/cm^3

Relative Detector Response (RDR) = relative fluence-to-exposure rate (FRER) times probability (P) of interaction

TABLE 3

Energy$_γ$, keV	FRER	P	RDR
40	0.3906	1.00	0.3906
60	0.5708	0.97	0.5561
80	0.5297	0.81	0.4309
100	0.4329	0.60	0.2617
150	0.2656	0.28	0.0751
200	0.1866	0.16	0.0303
300	0.1157	0.09	0.0099
400	0.0845	0.06	0.0053
500	0.0673	0.05	0.0035
600	0.0563	0.05	0.0026
800	0.0433	0.04	0.0016
1,000	0.0357	0.03	0.0012
1,500	0.0214	0.03	0.0006
2,000	0.0214	0.02	0.0005
3,000	0.0163	0.02	0.0003

Estimated Fidler 12.7cm dia x 0.16cm thick NaI response for Cs-137 is 1287 cpm/uR/hr
based on Cabrera 12/3-4/01; 20 measurements @ 5.3 urem/hr bkg

Use same methodology and interpolating for Cs-137 response have:

Energy$_\gamma$, keV	$(u_{en}/\rho)_{air}$, cm^2/g		
662	0.0294	FRER ~	0.0514
Energy$_\gamma$, keV	$(\mu/\rho)_{NaI}$, cm^2/g		
662	0.0749	Probability =	0.04
		RDR =	0.0022

For this detector the response to another energy is based on the ratio of the relative detector response, RDR to the Cs-137 energy
cpm/μR/h, E_i = $(cpm_{cs\text{-}137})*(RDR_{Ei})/(RDR_{Cs\text{-}137})$

TABLE 4

Energy$_\gamma$, keV	RDR_{Ei}	Fidler NaI Detector, E_i, cpm per μR/hr
40	0.3906	227464
60	0.5561	323807
80	0.4309	250914
100	0.2617	152403
150	0.0751	43735
200	0.0303	17654
300	0.0099	5791
400	0.0053	3076
500	0.0035	2027
600	0.0026	1486
662	0.0022	1287
800	0.0016	953
1,000	0.0012	692
1,500	0.0006	334
2,000	0.0005	297
3,000	0.0003	202

MDC for Cs-137 energy

Assume 10 μR/hr bkg then have 12,870 cpm

b_i =	214.5	counts
MDCR =	1212.673773	cpm
$MDCR_{surveyor}$ =	1715	cpm

minimum detectable exposure rate = 1.33 μR/hr

Table 5

keV	MicroShield Exposure Rate, μR/hr (with buildup)	cpm/μR/hr	cpm/μR/hr (weighted)	Percent of NaI detector response
40	4.808E-05	227464	11	0.3%
60	6.721E-05	323807	22	0.6%
80	7.360E-03	250914	1906	49.3%
100	1.860E-03	152403	293	7.6%
150	2.169E-03	43735	98	2.5%
200	4.200E-02	17654	765	19.8%
300	3.301E-02	5791	197	5.1%
400	4.090E-03	3076	13	0.3%
500	3.014E-02	2027	63	1.6%
600	8.205E-02	1486	126	3.3%
800	1.070E-01	953	105	2.7%
1000	2.384E-01	692	170	4.4%
1500	7.630E-02	334	26	0.7%
2000	2.154E-03	297	1	0.0%
3000	3.422E-01	202	71	1.8%
Total	9.688E-01		3869	100%

Minimum Detectable Exposure Rate =

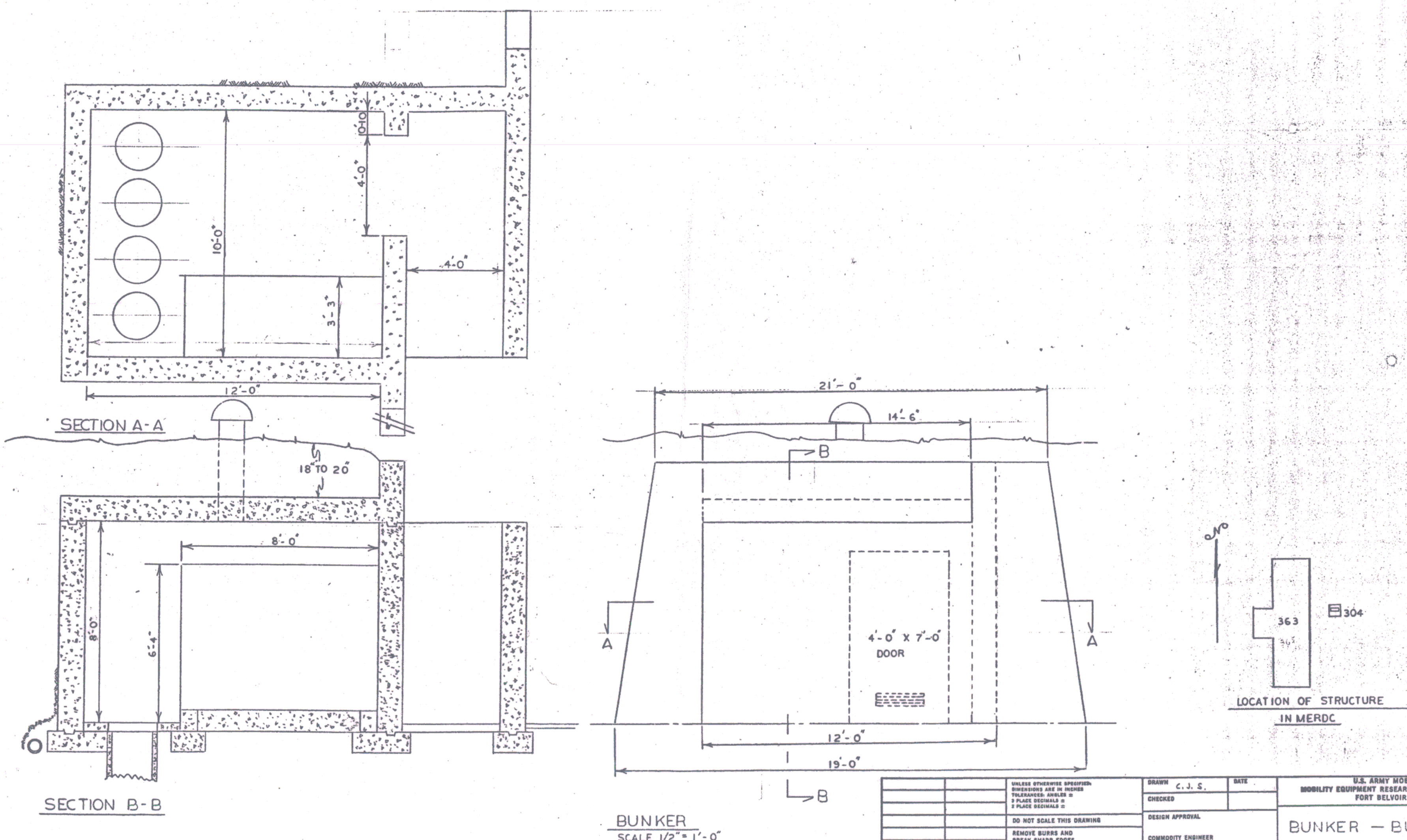
MDCR surveyor/(cpm/μr/hr)

0.4433 μr/hr

and MDC for Th232 and 50-year equilibrium progeny based on a normalized 1 pCi/g Th232

Scan MDC = (Assumed $MDC_{Th232\ Conc}$) x (Exposure Rate $MDCR_{Surveyor}$)/(Exposure $Rate_{assumed\ Th\ Conc}$)

Scan MDC = 0.46 pCi/g



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TOLERANCES: ANGLES ±
3 PLACE DECIMALS ±
2 PLACE DECIMALS ±

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SHARP EDGES TO

FILLET RADII TO

MATERIAL

NEXT ASSY | USED ON

APPLICATION

FOR INTERPRETATION OF DIMENSIONING AND TOLERANCING, SEE USAS Y14.5

DRAWN C. J. S. DATE

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DESIGN APPROVAL

COMMODITY ENGINEER

APPROVED FOR PRODUCTION

CHIEF, PRODUCTION ENGINEERING

RELEASED FOR PROCUREMENT

CHIEF, ENGINEERING DEPT.

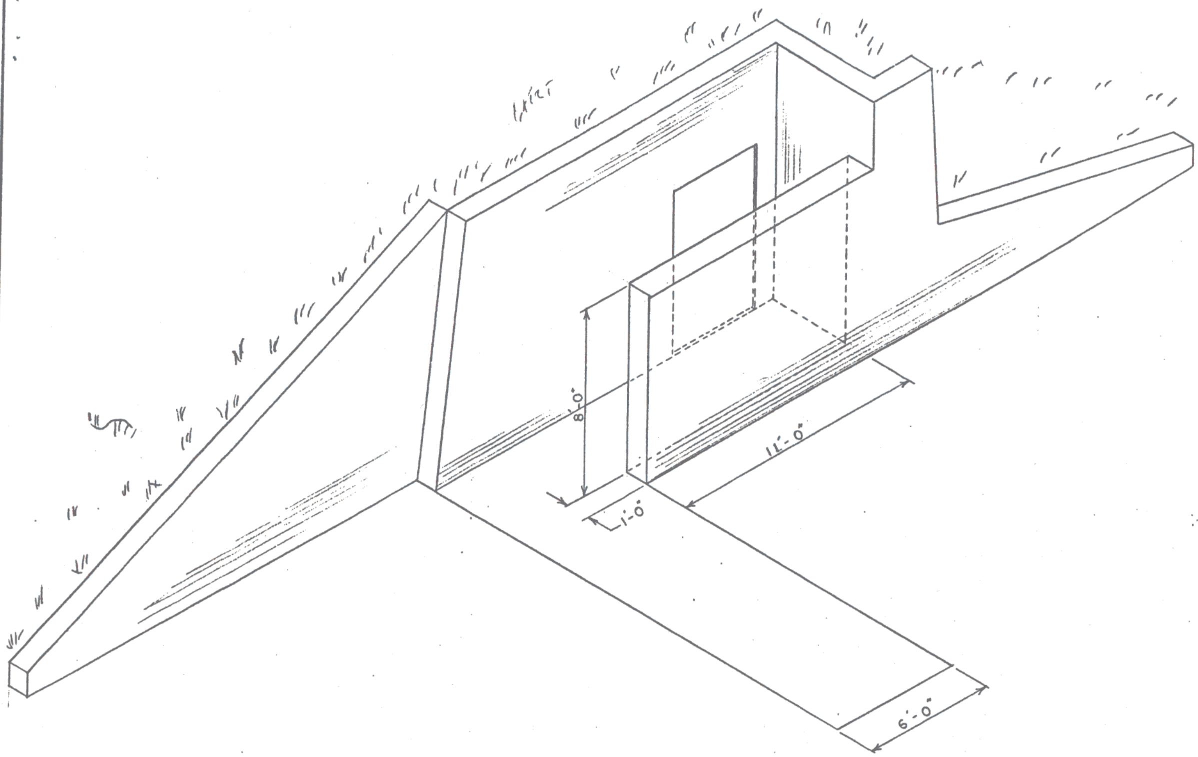
DATE

U.S. ARMY MOBILITY COMMAND
MOBILITY EQUIPMENT RESEARCH AND DEVELOPMENT CENTER
FORT BELVOIR, VIRGINIA 22060

BUNKER — BUILDING 304

SIZE D | CODE IDENT NO. 97403 | 35-06-418

SCALE | SHEET 1-4



EXISTING STRUCTURE

BUNKER OF CONCRETE & EARTHEN CONSTRUCTED

REVISIONS			
ZONE	LTR	DESCRIPTION	DATE

FIND NO.	CODE IDENT.	DWG. SIZE	PART OR IDENTIFYING NO.	QTY. REQD.	NOMENCLATURE OR DESCRIPTION	SPECIFICATION	MATER

LIST OF MATERIAL

NEXT ASSY	USED ON
APPLICATION	

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TOLERANCES: ANGLES ±
3 PLACE DECIMALS ±
2 PLACE DECIMALS ±

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REMOVE BURRS AND BREAK SHARP EDGES

SHARP EDGES TO

FILLET RADII TO

MATERIAL

DRAWN CJS	DATE
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DESIGN APPROVAL COMMODITY ENGINEER	
APPROVED FOR PRODUCTION CHIEF, PRODUCTION ENGINEERING	
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