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Boeing Michigan Aeronautical Research Center (BOMARC) Missile Shelters and Bunkers Scoping Survey Workplan

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14. ABSTRACT

The Weapons Safety Division of Headquarters, Air Force Safety Center (HQAFSC/SEW) and Radiation Surveillance Division, Air Force Institute for Operational Health (AFIOH/SDR jointly agreed to perform radiological scoping surveys of select structures on the Boeing Michigan Aeronautical Research Center (BOMARC) site. The work coincides with on-site oversight responsibilities of the two organizations during final status surveys and spot remediation under contract to Cabrera Services. The report details the strategy for conducting scoping surveys. In addition the report provides a comprehensive review of the risk analysis that was accomplished in the Remedial Investigation/Feasibility, that formed the basis for the Record of Decision (ROD). In particular, the discrete particle intake paradigm was discussed and other issues that raised recent concerns on the applicability of the risk analysis accomplished in the ROD. Criteria for acceptable surface contamination levels for shelter interiors were proposed. One for continued intermittent use with institutional controls inplace and the other under a shelter demolition/on-site burial assumption.

15. SUBJECT TERMS

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Acronyms

AEA	Atomic Energy Act
AEC	Atomic Energy Commission
AF	Air Force
AFI	Air Force Instruction
AFIOH	Air Force Institute for Operational Health
ALARA	as low as reasonably achievable
ARAR	applicable or relevant and appropriate requirements
Bkgd	Background
BOMARC	Boeing Michigan Aeronautical Research Center
Bq	Becquerel
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CI	confidence interval
Ci	Curie
cpm	counts per minute
DCF	dose conversion factor
DCGL	dose conversion guideline level
DEP	Department of Environmental Protection
DOE	Department of Energy
dpm	disintegration per minute
DTRA	Defense Threat Reduction Agency
DU	depleted uranium
EIS	environmental impact statement

EMC	elevated measurement comparison
EOD	explosive ordnance disposal
EPA	Environmental Protection Agency
FIDLER	field instrument for detection of low energy radiations
FRC	Federal Radiation Council
FS	feasibility study
FSS	final status survey
GI	gastrointestinal tract
HEU	highly enriched uranium
HpGe	hyperpure germaniun
HQ AFSC	Headquarters, Air Force Safety Center
ICRP	International Commission on Radiological Protection
keV	kilo electron Volts
LLD	lower limit of detection
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
μCi	microcurie
MDA	minimum detectable activity
MDC	minimum detectable concentration
MEI	maximally-exposed individual
MeV	Mega electron Volts
M&I	maintenance and inspection
$^{Total}U$	total uranium
nCi	nanocurie

NCRP	National Council on Radiation Protection and Measurements
NESHAPS	National Emission Standard for Hazardous Air Pollutants
NIST	National Institute of Standards and Technology
NTS	Nevada Test Site
NPL	National Priority List
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
pCi	picocurie
rad	radiation absorbed dose
RAM	radioactive material
rem	roentgen equivalent man
RI	remedial investigation
RIC	Radioisotope Committee
ROD	record of decision
RW	radiological waste
SDR	Radiation Surveillance Division
SDWA	Safe Drinking Water Act
SI	site investigation
TEDE	total effective dose equivalent
UNLV	University of Nevada Las Vegas
WGP	weapons grade plutonium

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Boeing Michigan Aeronautical Research Center (BOMARC)
Missile Shelters and Bunkers Scoping Survey Workplan

1.0 Introduction

1.1 Purpose and Objective.

The Radiation Surveillance Division of the Air Force Institute of Operational Health (AFIOH/SDR) and the Weapons Safety Division of Headquarters, Air Force Safety Center (HQ AFSC/SEW) jointly agreed to perform radiological scoping surveys of select structures on the Boeing Michigan Aeronautical Research Center (BOMARC), Fort Dix, New Jersey. Since the site was listed under the Air Force's Installation Restoration Program, it was designated RW-01. The surveys are planned to coincide with the initial stages of the "Final Status Survey and Spot Remediation" work contracted to Cabrera Services, Air Force Center for Environmental Excellence, Contract No. FA8903-04-D-8693, Task Order 0004 (Cabrera 2007). This allows staff from both organizations to be on-site for contract oversight and review of findings from the field. Information from the scoping surveys will assess radiological impacts to structures and the degree of impact. The relative effort of scoping survey among individual structures is based on historical potential for impact as recommended by the "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)" (NRC 1997), with those structures more likely having potential for impacts receiving greater survey effort than those with lesser potential. Dependent on the findings of investigation in target structures, the remainder of structures may be surveyed as part of the scope of this effort.

1.2 Site History.

On 7 June 1960, an explosion in a helium tank took place in Shelter 204 causing a fire in a liquid-fueled, nuclear-tipped BOMARC missile. The fire burned uninhibited for about 30 minutes. Fire-fighting activities, using water as a suppressant, were conducted for 15 hours. As a result, materials from the shelter flowed under the front shelter doors, down the asphalt apron and street between the row of shelters, and into the drainage ditch leading outside the site boundary fence. While a wind from the north to northeast was present during the initial stages of the fire, evidence from the Remedial Investigation/Feasibility Study (RI/FS) did not support any significant airborne transport (Earth Tech 1992). Though translocation of shelter contaminants was largely attributed to fire-fighting water (and later storm water run-offs), activities during accident response and routine activities post-accident (i.e., foot and vehicle traffic) were believed to be responsible for some cross-contamination of material (Earth Tech 1992). The site ceased operational use in 1972 and had a number of radiological investigations that were primarily accomplished to assess the integrity of asphalt and concrete engineering controls, placed to limit the movement of radiological contaminants.

The primary radiological contaminant released was weapons grade plutonium (WGP), with lesser activities of highly-enriched and depleted uranium. Post-accident recovery operations accomplished by Explosives Ordnance Disposal (EOD) and the Atomic Energy Commission (AEC) removed weapon debris that contained a significant fraction of the radiological material contained in the weapon, and shipped the materials to Medina Base, San Antonio TX. Though the original design

amounts of the weapon remain classified, an estimate of the material remaining on site after the initial removal was made by Department of Energy (DOE) and AF scientists, which placed an upper limit at 300 grams (Earth Tech 1992). A Record of Decision (ROD) was filed by the AF for the site, where the AF decided to pursue excavation and off-site disposal of contaminated waste, provided a disposal site is available and the absence of other events that would dramatically decrease the cost-effectiveness of the option (Vest 1992). The foundation of the ROD was the environmental impact statement (EIS) and remedial investigation/feasibility study (RI/FS) that were filed with the EPA by the Deputy Assistant Secretary of Air Force (Vest 1992). The risk-based criterion established in the RI/FS for unrestricted-release of soils was 8 picocuries/gram (pCi/g) ^{239}Pu, as modeled with the Residual Radiation (RESRAD) computer-based risk modeling code developed by Argonne Laboratory (Earth Tech 1992). These concentrations provide an annual dose of 4 millirem (mrem) to a maximally-exposed individual (MEI) and correspond to a lifetime excess cancer risk of 10^{-4} (70-year integrated exposure). A risk-based remediation goal was not established for structures; however, criteria in U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.86 were considered relevant and appropriate for an unrestricted release and recommended in the absence of a risk-based criterion (Earth Tech 1992).

Remediation of the Class 1 contaminated areas, based on MARSSIM terminology (NRC 1997) and results of the 1997 site characterization (OHM 1998), was conducted from March 2002 to June 2004 by Duratek Services, Inc, and Shaw Environmental and Infrastructure (E & I). A total waste volume of 21,998 cubic yards (yds^3) comprised of contaminated debris and soils were packaged, shipped, and disposed at Envirocare of Utah (Duratek 2006). Shelters 202 and 206 were removed as part of the demolition of shelter 204 due to the anticipation of interference with shoring requirements of the excavation. Twenty-two survey units were established for the final status survey with sizes between 124 and 2125 m^2, with the total area being 3.7 hectares (9.1 acres), and an average survey unit of 1674 m^2. Table A-1 contains a summary of the final status data from the Class 1 areas, to include data from applicable elevated measurement comparison (EMC) areas.

Since the remediation of the Class 1 areas, there have been four distinct activities accomplished on the site to assess the feasibility of completing the ROD-recommended action for areas outside areas remediated in 2002 – 2004. Cabrera Services conducted a historical site assessment (Cabrera 2006b), AFIOH/SDR conducted in-situ γ-spectroscopy scanning of potentially impacted areas of the site, National Security Technologies, LLC conducted similar surveys as AFIOH/SDR, and Cabrera Services conducted a discrete particle removal operation in August/September 2006. Each one of these activities had some impact on HQ AMC/A7V, AFIOH/SDR, and the 305 CES/CEV's confidence that the ROD-recommended action could be completed in a cost-effective manner. Specific details will be discussed, as appropriate, to the radiological scoping survey of structures.

2.0 Contaminant Characteristics

2.1 Radionuclides of Concern (ROC).

The primary ROC is WGP, with highly-enriched uranium (HEU) and depleted uranium (DU) lesser ROCs in rank order. Table A-2 provides mass fractions for the WGP at the site, estimated for 1958 from Rademacher (2001). Of the α-radiation emitting isotopes, ^{239}Pu and ^{240}Pu dominate

radiologically and have the same dose-conversion factors for internal dosimetry and dose-modeling applications. Further, because the two radionuclides have similar α-particle energies (Table A-3), discrimination is not possible through α-spectroscopy and analytical results will have them combined as $^{239+240}$Pu.

The HEU and DU are indiscriminant from another, as each has a varied activity fraction of ^{234}U, ^{235}U, and ^{238}U. Figure A-1 contains a plot of the ^{234}U to ^{238}U activity concentration ratio (Rademacher 1999b), based on data from the 1997 Characterization (OHM 1998). From the plot, it is clear that the HEU dominates DU in the overall isotopic characteristic. Among the three uranium isotopes, ^{234}U comprised about 90 % of the total activity, including naturally-occurring background sources. However, in comparison to the WGP, uranium isotopes provide negligible risk, where the $^{238+239+240}$Pu to $^{234+235+238}$U activity ratio was estimated at 469 in the waste-profiling process (Horton and Rademacher 1998). This process used α-spectroscopy radionuclide data from the 1997 Characterization (OHM 1998). Another important point gleaned from the 1997 Characterization data is that the two elements are spatially co-located, as evidenced from Figure A-2. Therefore, remediation of the WGP will effectively also remove the uranium co-contaminant.

2.1.1 $^{239+240}$Pu to ^{241}Am Activity Ratio. Direct assessment of $^{239+240}$Pu in soils at low activity concentration is difficult because both isotopes only have infrequent, low-energy photon emissions. As well, laboratory analyses of soils using high-resolution γ-spectroscopy is hampered by the same issue, leaving chemical dissolution, separation, and alpha spectroscopy as the most common direct assessment method. For soils containing heterogeneously distributed contaminants, large uncertainties can be observed in reported concentrations due to limited aliquot size for this method (Bernhardt 1976). A practical indirect alternative involves assessment of ^{241}Am, the decay daughter of ^{241}Pu:

$$^{241}\mathrm{Pu} \rightarrow {}^{241}\mathrm{Am} + {}^{0}_{-1}\beta ,$$

and calculation of the $^{239+240}$Pu through an established relationship between the two. HQ AFSC/SEW reviewed historical information on the $^{239+240}$Pu to ^{241}Am ratio and provided technical recommendations for future assessments (Rademacher 1999a). At that time, the best estimate was 5.4 $\pm$ 16 % [90 % confidence interval], based on α-spectroscopy data from the 1997 Characterization. During the 2002 – 2004 remediation, another $^{239+240}$Pu to ^{241}Am ratio data set was generated (Figures A-3 and A-4). Though the sample set was significantly smaller than that from the 1997 Characterization, the 90 % confidence interval of the estimate was over two-fold lower. The best point estimate from the two data sets is within 3% agreement, which is within the typical combined bias of Pu and Am chemical tracers used for these analyses. For its current work, Cabrera Services proposed α-spectroscopy analysis on a fraction of final status survey soils samples to verify the $^{239+240}$Pu to ^{241}Am ratio, however, this endeavor was deemed fruitless due to the increased variability observed in samples at or below the 8 pCi/g $^{239+240}$Pu criterion (Figure A-4). Also, over time the ratio decreases, making use of previously determined ratios more conservative (protective).

2.1.2 Discrete Particle Nature of Contaminant.

2.1.2.1 General. Plutonium dioxide (PuO_2) is the most stable of the oxides found in the environment and is formed under most conditions, especially when plutonium is ignited in air

(Burley 1990), as was the case of the BOMARC WGP. PuO_2 has a high melting point (2240 ^oC), has a high chemical stability, and is highly insoluble in water (Burley 1990). The behavior of plutonium in soils can vary depending on the local soil characteristics and the form the plutonium is in at the time of introduction. Four sites have been extensively monitored in the U.S. up to 30 years after introduction of the plutonium into the environment: the Nevada Test Site (NTS), Oak Ridge National Laboratory (ORNL) in Tennessee, Mound Laboratory in Ohio, and Rocky Flats in Colorado (Burley 1990). The source of plutonium is different for each site. At NTS, the plutonium is dispersed as an oxide as the result of safety research studies. At Oak Ridge, plutonium in a holdup pond was released when a dike broke. At Rocky Flats, cutting oil contaminated with metallic plutonium was released from leaking storage drums; while at the Mound Facility, a low-pH solution of plutonium leaked from a waste transfer line. For the soils studied at NTS and Rocky Flats, extraction of plutonium from soils was very low (10 – 15 %) as compared to 60 – 85 % extraction from the soils of the Mound Facility and ORNL (Burley 1990). Thus, if the plutonium was introduced into the environment as an oxide or metallic form, it exhibited low solubility; whereas if introduction was in the form of a soluble compound, much greater mobility was exhibited at a later time. Furthermore, autoradiographic comparisons of Rocky Flats and Mound Facility soils have indicated that the soils of the former exhibited discrete particles of plutonium, while the latter had a more homogeneous dispersion (Burley 1990).

2.1.2.2 RI/FS. In many locations of the RI/FS (Earth Tech 1992), the discrete particle nature of the BOMARC WGP contaminant was discussed:

2.1.2.2.1 Page 1-13, Section 1.5. The radioactive contamination is not distributed uniformly over the site, but occurs in discrete "hot spots," which in several instances have been found to be a single particle, presumably containing plutonium dioxide.

2.1.2.2.2 Appendix J, Page J-8, Section 2.0. Although high ^{239}Pu . . . some of these samples (e.g., 150k pCi/g), the high values are likely due to discrete particles . . .

2.1.2.2.3 Appendix J, Page J-6, Section 2.0. In order to assess the radiological impacts from the non-uniform contaminated soil at the BOMARC site, the RESRAD guidance relating to inhomogenous contamination was reviewed for the assessment.

2.1.2.2.4 Appendix I. Due to the non-uniform distribution of plutonium in discrete particles within site soils both the Air Force and EPA believed that it would be impossible to obtain split samples in the field with approximately equal concentrations of plutonium, due to the non-uniform distribution of plutonium in site soils. Both parties agreed that if a soil sample was to be split in the field, it was likely that one half of the sample would contain the bulk of the plutonium, due to the occurrence of plutonium in discrete particles.

The primary concern raised by the discrete particle issue was applicability of the RESRAD modeling, sample quantification, and split sample inter-comparison. In particular, the concern over high variability between split sample analyses was addressed for the 2002 - 2004 remediation and the current Cabrera effort. In both, split samples (between on-site and off-site laboratories) were recommended to have respective analyses conducted on an in-turn basis, where the same sample is analyzed by both laboratories and the counting geometry is identical.

2.1.2.3 AF Studies in Conjunction with Remediation. The AF completed two studies of heterogeneity with a primary focus on the effects on γ-spectroscopy analyses (Rademacher 1999 & 2001). In general, heterogeneity was observed to a degree that could influence overall uncertainties in laboratory analyses of samples. Conjugate measurement of individual samples in plane-symmetrical containers (e.g., right cylindrical containers, petri dishes, etc.) was effective at reducing uncertainty in sample activity. In addition to the effects on γ-spectroscopy analyses, the second study provided numerous examples of heterogeneity at the sample and sub-sample level (Figures A-5 to A-7). From Figure A-5, it is noteworthy that the second aliquot contains the vast majority of the total sample activity, and it appears, based on the conjugate count, that the activity in this aliquot is dominated by a single particle. Removal of this individual particle from this aliquot could reduce the total sample activity by 20 to 30-fold. Other examples are less drastic. For the sample in Figure A-6, the mean activity among the eight aliquots is near the criterion of 8 pCi/g, with aliquot 3 comprising about half of the total sample activity. For the example in Figure A-7, the contaminant appears relatively uniform within and among the aliquots. These studies prompted the use of conjugate counting to reduce the variability in reported results of final status soils samples.

2.1.2.4 Recent Studies.

Cabrera Services, in conjunction with the University of Nevada, Las Vegas, conducted studies on the contaminant as a follow-on to investigations conducted by AFIOH in 2005 (Cabrera 2006d). The work focused on an evaluation of a "less homogenous and more discrete contaminant distribution than previously encountered, and thus efforts were undertaken to confirm the risk assessment or regulatory assumptions set forth in the ROD." The field activities comprised locating AFIOH-targeted investigation areas with a field instrument for detection of low-energy radiation (FIDLER), in-situ γ-spectroscopy with a hyperpure germanium (HpGe) and Canberra ISOCS® software, with the principal activity being soil coring. Soil cores were evaluated for chemical and physical characteristics by UNLV. The general conclusion from this work was "discrete Pu particles are chemically and physically stable, and could remain in this form given normal environmental weathering conditions." Further, it was found through scanning electron microscopy (SEM) that many of the particles had physically large dimensions (100 – 500 μm), prohibiting respiratory tract intakes and air-suspension under typical site conditions. The samples were segmented for separate sub-sample level analysis, similar in some respects to the work accomplished in the AF study of heterogeneity. From this analysis, it was determined that the vast majority of activity is associated with discrete particles. However, since locations with very-high in-situ FIDLER response were targeted for sampling, very little discussion on discrete particle issues were provided for samples at or near the soils remediation criterion, 8 pCi/g $^{239+240}$Pu.

Table A-4 was developed to evaluate soil samples and sub-sample segments with respect to activity concentration. For the table, sample mass for individual segments was estimated based on segment volumes, an estimated soil density of 1.5 g/cm^2, and a $^{239+240}$Pu to ^{241}Am ratio of 5.4. For the vertical segment data summary (upper portion of Table A-4), significant variability exists within the data set. All of the samples, but one, have one vertical segment with mean activity concentration above 10^4 pCi/g. If all of the activity in one of these segments is due to a single particle, the minimum particle diameter ranges from 130 to 440 μm, assuming pure dioxide form, spherical, and $\rho = 11.5$ g/cm^3 (see Figure A-9). While the SEM images are not perfectly circular (i.e., allowing a spherical shape inference), the dimensions have some general consistency with the range of spherical

equivalent diameters. Among these samples, considerable variability exists in mean activity concentration of the other segments. For example, sample 15-D has three vertical segments that collectively comprise about 12 % of the total sample activity, with the one about 88 %, while 4-C, 5-F, 10-F, 11-F, and 12-F, have > 99 % of the total activity in a single segment.
The further division of samples into horizontal segments illustrates some interesting information on the distribution of activity. For example, in the case of sample 4-C, beyond the horizontal segment containing the highest activity (4-C-2-C), the other 11 segments had activity concentration ranging from 3.9 to 360 pCi/g (Table 4, lower portion). In some contrast to this is sample 5-F, where beyond segment 5-F-4-C, segments with reported activities ranged from 5.6 to 600 pCi/g, while five segments did not have reported activity. For these lower activity segments, some evidence exists for a more uniform distribution of activity, but it is clear that some discrete particle character exists for activity concentrations around the remediation criterion of 8 pCi/g $^{239+240}$Pu within some samples.

2.1.2.5 HQ AFSC.

One focus of the 2005 Cabrera effort was addressing reports that the contamination found in outlying areas was a "less homogenous and more discrete contaminant distribution than previously encountered." The report did not formally address this issue in its conclusions, which is logical since the report did not present any specific information on the discrete particle nature of contamination in the areas remediated 2002 – 2004. HQ AFSC does not believe there is a technical basis for a difference in contaminant characteristics in the areas remediated in 2002 – 2004 vs. outlying areas with contamination identified by AFIOH and National Security Technologies (2006). The RI/FS (Earth Tech 1992) and Historical Site Assessment (Cabrera 2006c) described the potential for inadvertent transport by vehicles and personnel during the accident response and subsequent site activities to include remedial activities. Vehicle- and personnel-supported translocations of radiological contamination from the primary contaminated area (i.e. shelter 204, area surrounding shelter 204, and water drainage ditch) to other areas are unlikely to comprise a particle distribution that is different than that in the primary contaminated zone. Although, significant differences in areal activity concentrations may exist among areas. Figures A-10 and A-11 help illustrate these concepts. Figures A-10a and A-11a illustrate initial contaminant depositions, with A-10a being a relatively large area (i.e., primary) and A-10b being small, indicative of isolated contamination (i.e., translocated to an outlying area). Over time, there will be a slow migration of material to adjacent soils, in both horizontal and vertical planes, caused by environmental effects (e.g., wind, surface water movement, water percolation) and routine site activities. For the primary contaminated areas, over time, there will be little noticeable change in the areal distribution of particles, as illustrated in the change from Figure A-10a to b, as horizontal migration will effectively be in a quasi-equilibrium condition (i.e., transport between adjacent areas is self-compensating). Naturally, for areas on the fringe of the primary contaminated area this is not the case, and there will be some dilution in the vertical plane. However, isolated contamination, as illustrated in Figure A-11, will not have an equilibrium condition, and over time, migration will cause an areal dilution in particle number concentration. This dilution gives the appearance that the distribution of particles activities (i.e., discrete vs. uniform) is different among areas evaluated, though there is no sound technical basis for this conclusion. The highlighted cells of Figure A-11 each contain the largest particle among the distribution of particles and there is no difference in particle number and size distribution among the two figures. However, in an analysis of sampled soils, similar to that conducted by UNLV (Cabrera 2006d), both highlighted cells are likely to provide a similar in-situ

FIDLER response (with minimal vertical migration) and overall total sample activity. But, sub-sample evaluation of the highlighted cells will be clearly different. The aged sample's activity will be comprised essentially of a single particle, while that of the initial deposition composed of multiple particles, and perhaps speculation that it possessed a diffuse component as well.

Historical information indicates that the radioactivity in the WGP contaminant is dominated by discrete particles. Due to the nature of the accident, these particles are expected to exits predominantly in a dioxide form. Laboratory analysis of discrete particles samples in 2005 indicate that these particles are physically stable and have not undergone observable physical changes since the accident. For soil samples with activity concentration near the remediation criterion, the effects of heterogeneity, 1) significant difference in activity among other subaliquots and 2) significant difference between conjugate count of the same sample (or aliquot), were not observed to the same degree in the studies conducted by the AF and Cabrera.

Final status survey (FSS) soil sampling completed in Class 1 areas by Duratek had a fairly uniform distribution of activity concentration among the systematic sampling locations. As well, elevated measurement comparison (EMC) had a fairly uniform distribution, but of a higher estimated mean among the sample group and a larger fraction of outlier (high concentration) samples. Figures A-12 and A-13 provide a summary of these data sets. From the two data sets, it is clear that a significant fraction of samples had activity concentration below the minimal detectable concentration (MDC) - 83 % for systematic and 42 % for EMC areas. Some caution should be made in inferring any conclusions on mean concentrations for Class 1 areas based on the relative number of systematic and EMC samples, as EMC comparison areas had a significantly higher areal sampling density than encompassed by systematic samples. While no direct particle size information exists for these samples, due to the relative uniformity, most samples must be comprised of a number of particles, rather than a single one. For the systematic samples, the highest measured activity concentration was 43 pCi/g, with total sample activity of 5,400 pCi (200 Bq). The two highest EMC samples had activity concentrations (and total sample activity) of 95 pCi/g (14 nCi) and 88 pCi/g (17 nCi). For these, if the entire activity existed in a single discrete particle, the respective volume equivalent diameters would be 33.4 and 35.8 μm, if the particles were pure PuO_2 (Figure A-8). These respective particle sizes will be discussed later.

2.2 Implications of Discrete Contaminant Nature.

2.2.1 Definitions.

The terms "discrete particle," "diffusely distributed," "heterogeneous," and "homogenous" have been used extensively throughout documents to describe the nature of the plutonium contaminant on the site, though no formal definition(s) have been provided. Other documents describing WGP plutonium have used similar terms. Defense Threat Reduction Agency (DTRA) in managing WGP on Johnston Atoll defined 1) "discrete, hot particles" as point sources with diameters greater than 45 μm with approximate activity of 135k pCi and 2) dispersed activity as particles with activity about 270 pCi and diameter about 10 μm (DTRA 2002).

The inconsistent use of the terms in the radiation protection field can lead to confusion. For example, the term "hot particle" in the radiation protection field is generally in reference to high,

specific-activity fission product particles that pose an exposure risk for highly-localized skin dose and can be readily identified by a Geiger counter. The highly-localized dose is predominantly due to the β-particle emissions, while associated γ-emissions deposit their energy over a significantly larger tissue volume. Hot particle exposures are of particular interest in the nuclear power industry and were an issue of fallout from the Chernobyl accident. In this context, discrete plutonium particles do not behave in a similar manner, with the only external dose from low-energy, low frequency emission x- and γ-radiations. For internal exposures through inhalation, plutonium particles can produce highly localized dose to adjacent tissue areas in the lung. In this case, the highly-localized energy deposition is from α-particles. However, in this exposure context, a DTRA-defined hot particle (i.e., diameters greater than 45 μm) cannot produce lung exposures since the particles are physically too large for respiratory intake under practical conditions. As such, use of the hot particle term has been discouraged for the WGP at the BOMARC site.

While PuO_2 is highly immobile and chemically very insoluble in most environments, theoretically, some fraction of the contamination will have been mobilized in an aqueous form. As disassociated ions, the plutonium can be incorporated into other chemical complexes (including dioxides) and subsequently have a more uniform distribution in the soil matrix. The degree that this condition is represented in soils at or below the $^{239+240}$Pu remediation criterion is difficult to assess because discrete plutonium particles (which potentially exist in large numbers per sample) are difficult to isolate as point sources in sample matrices. This is in contrast to samples that contain a single high activity discrete particle, orders of magnitude above the criterion. Therefore, the $^{239+240}$Pu in some samples may appear to be homogenously distributed, like that common to naturally occurring radionuclides in soils, but from an activity standpoint could be comprised primarily of numerous discrete particles distributed throughout the soil matrix. An important point regarding this issue is that high activity discrete particles are expected to be predominantly in a dioxide form, as they are chemically and physically stable (Cabrera 2006d). No physical mechanism exists for the natural environmental formation of high activity, relatively-pure plutonium particles of other chemical compounds (i.e., silicates, carbonates, nitrates). Rather, these compounds are more likely to exist in diffuse concentrations, as complexes with inert soil particles.

2.2.2 Issues.

2.2.2.1 General. The discrete nature of the WGP on the BOMARC has created some issues germane to remediation of the site, whether real or perceived. As noted, recent interest in this issue has generated discussion in technical reports supporting this remediation, and is worth some limited discussion here. Table 2.1 provides a brief summary of key issues of concern.

2.2.2.2 Laboratory Analysis.

Proper laboratory analysis of soils and other matrices supporting the remediation and FSS is critical to proper unrestricted radiological release of the site. The 1996 Characterization Study (OHM 1998) had significant variability in the $^{239+240}$Pu to ^{241}Am ratios distribution (Rademacher 1999b) that was attributed to heterogeneity. From that point forward, HQ AFSC only recommended α-spectroscopy for $^{239+240}$Pu to ^{241}Am ratio assessments, as the method was highly susceptible to variability when determining total activity. In 2000, the Department of Environmental Protection (DEP), State of New Jersey, requested that a fraction of FS soil samples be analyzed by α-spectroscopy. However,

in response to the request, the AF convinced the DEP that the method could lead to high variability and confounding results. The only α-spectroscopy analyses conducted on samples from the site recently have been for confirmation of the $^{239+240}$Pu to ^{241}Am ratio during the 2002 – 2004 remediation (Duratek 2006) and by U.S. Geological Services for water samples (Zapecza *et al* 2000).

HQ AFSC and AFIOH evaluated the effects of heterogeneity on γ-spectroscopy analysis of soil samples (Rademacher 1999a, 2001). Conjugate measurement of individual samples in plane-symmetrical containers (e.g., right cylindrical containers, petri dishes, etc.) was recommended to reduce uncertainty in sample activity. Problems with heterogeneity in assessments of FSS samples of the Class 1 area were limited during the 2002 – 2004 remediation. Cabrera Services has incorporated the method into FSS soil samples for their current scope of work (Cabrera 2007).

2.2.2.3 Sample Collection. Collection of soil samples representative to concentrations in soils in conjunction with in-situ γ-scanning are important to verifying potentially impacted areas have met risk-based criterion. Heterogeneity can impact the ability to collect samples representative of the true contamination levels. While this issue was apparent and important during evaluation of pre-remediated soils (Earth Tech 1992), the effect on soils at or below the remediation criterion of 8 pCi/g $^{239+240}$Pu is significantly lower. Clear from the 2002 – 2004 remediation FSS soil sampling results (Figures A-12 and A-13), the highest activity concentration observed was 95 pCi/g, only 12-times the criterion. To moderate potential effects of heterogeneity in EMC areas, multiple samples were collected and analyzed.

2.2.2.4 In-Situ γ-Scanning.

Most in-situ γ-screening surveys of soil model the instrument response to either a homogenous distribution or point source of radioactive material. For the FSS conducted in 2002 – 2004, the FIDLER response was calibrated to a semi-infinite plane of contaminated soil that was removed from the site (387 soil samples) and quantified thru laboratory analysis (Duratek 2006). Because this soil was blended and homogenized over an approximate 15 cm thickness, the calibration was conservative to actual site conditions where activity concentrations are greatest near the surface and decrease with depth. Duratek estimated the scanning MDC at about 40 pCi/g, but noted that a much lower MDC was demonstrated in the field.

The presence of a contaminant with a substantial fraction of total activity in discrete particles changes the dynamics of scanning. Radiological contamination isolated in point sources makes impacted areas much more readily identifiable than diffusely contaminated areas. Along this line, discrete particles evaluated from the site are chemically and physically stable, have had limited physical degradation since formation, and are likely to remain in this state under normal environmental weathering conditions (Cabrera 2006d). As such, high-activity, individual particles do not effectively become more diffuse over time, making them readily identified through scanning surveys. Naturally, due to this, areas with reasonable potential for impact should have carefully designed scanning surveys.

TABLE 2-1. Issues Related to Homogenous and Heterogeneous WGP Contaminant Distributions.

Context		Homogenous	Heterogeneous
Laboratory Analysis of Samples	α-Spectroscopy	Reasonable reproducible analytical results among sub-sample aliquots/split samples	Questionable reproducibility among sub-aliquots/split samples due to small aliquot sizes. Relationships between co-distributed contaminants reasonable if serial extraction method is used.
	γ-Spectroscopy (^{241}Am Target Analyte)	Reasonable reproducible analytical results among sub-sample aliquots/split samples.	Better reproducibility among sub-aliquots/split samples than α-spectroscopy, due to significantly larger aliquot size. Variability in reported activity concentration can be minimized with conjugate counting.
Sample Collection		No special provisions for sampling.	Variability in sampled activity is reduced for larger sample sizes.
In-situ γ-Scanning of Site		Good agreement to modeled instrument response.	Modeled response should account for homogenous and discrete particle response. High activity discrete particles in contaminant make identification of impacted areas easier than the case of a homogenous contaminant.
Exposure Routes [Dose equivalent fractions from RI/FS, Appendix J (Earth Tech 1992)]	Inhalation (76 %)	Standard dose model algorithms assume fairly uniform exposure to homogenous contaminant distributions. Models normally assume 100 % of contaminant is of respirable particle size.	Heterogeneous distributions could have widely varied modeled doses. Assumption of 100 % respirable contaminant is poor. Mean doses will be lower than modeled.
	Soil Ingestion (22 %)	Standard dose model algorithms assume fairly uniform exposure to contaminants, as the case for a homogenous contaminant.	Heterogeneous distributions could have widely varied modeled doses, dependent on the relative particle distribution.
	Plant (1 %)	Standard dose model algorithms assume fairly uniform aqueous partition of contaminant, which exists for homogenous cases.	Heterogeneous distributions should not have widely varied overall aqueous partition of contaminant. Aqueous phase should be lower for large activity, discrete particles than homogenous, due to lower surface to volume ratio.
Risk Modeling	Lung Intake	Primary source for risk modeling.	ICRP models are applicable to discrete particle exposures [(Harrison 2003); (Charles *et al* 2003)].
	GI Intake	Primary source for risk modeling.	PuO_2 has insignificant dose to GI tract in-transit. Transport to internal organs requires soluble form; discrete particles in bone and liver not applicable to GI intakes. Large activity, discrete particles are expected to have lower GI uptakes than homogenous form, due to lower surface to volume ratio.

2.2.2.5 Exposure Routes.

2.2.2.5.1 General. RESRAD Version 4.10 was used for pathway exposure analysis and dose modeling [RI/FS, Appendix J (Earth Tech 1992)]. Among the pathways considered in the resident-farmer exposure scenario, the most important exposure routes are: inhalation (76 %), ingestion (22 %), and consumption of plants (fruits and vegetables, 1 %), with ground shine and meat consumption combined providing less than 1 % of the projected effective dose equivalent (EDE). Table A-5 provides a list of dose conversion factors used in the RESRAD modeling and recent International Commission on Radiological Protection (ICRP). The dose conversion factors (DCFS) used were from DOE-EH-0070 (DOE 1988), where "aerosol class and gastrointestinal tract uptake fraction yielding the highest dose per unit were used" [RI/FS, Appendix J, page J-3 (Earth Tech 1992)]. This DOE-EH-0070 values were the same as those listed in Federal Guidance Report 11 (EPA 1988) and were for the most soluble forms of Pu. In contrast, for inhalation exposures, the DCF for PuO_2 is 40 % lower and for ingestion it is 68-fold lower.

2.2.2.5.2 Inhalation.

Inhalation is the most important exposure pathway for the resident-farmer scenario. Heterogeneously distributed activity among particles greatly reduces projected doses because a fraction of the contaminant is unavailable for air-suspension and unable to penetrate to deep portions of the lung (alveolar region) where the greatest lung retention times are realized. Figure A-14 was generated for PuO_2 particles, under the assumption of 100 % PuO_2 composition, spherical shape, and density 11.5 g/cm^3. Aerodynamic equivalence was calculated with the equation from McClellan and Henderson (1989), using slip correction factors from the EPA (2007) and listed in Table A-6. The plot contains two data curves. The black line relates $^{239+240}$Pu activity to aerodynamic equivalent diameter, while the gray line relates volume and aerodynamic equivalent diameters. For volume equivalent diameters, greater than 0.5 μm, slip corrections are small and the primary factor relating the two diameters is:

$$\sqrt{\frac{\rho_0}{\rho_\rho}},$$

where ρ_0 is unit density and ρ_p is the density of the particle in question. The plot contains indices important to inhalation exposures. The green line represents a practical threshold for particle suspension in the environment (30 μm, aerodynamic equivalent diameter). The two blue lines represent an "approximate" demarcation between alveolar and tracheobronchial deposition and the red a minimum particle size separating tracheobronchial and naso-oropharynheal deposition (McClellan and Henderson 1989). From the plot, a maximum single particle activity available for deposition in the alveoli is 1.8 pCi (0.067 Bq), which the bolded values in parentheses for particles of shape factor equal to 1.6. Based on ICRP Report 54, about 60 % of Class Y (inhalation class) radiological material deposited in the alveoli region has long-term retention (ICRP 1987), which is responsible for the greatest modeled cancer risk to the lung. In contrast, for deposition in other portions of the respiratory tract, retention times are significantly lower and very small fractions of deposited radioactivity are transferred to other target organs of the body.

The most recent ICRP lung model, ICRP 66 (ICRP 1995), is more sophisticated than the model of ICRP 30, which DCFs used in the RI/FS were derived. There has been a significant amount of computer-based modeling of variability and uncertainty associated with inhaled PuO_2, where there exists a stochastic intake paradigm of relatively small numbers of particles with high specific activity versus average intake (Aden and Scott 2003). In Aden and Scott's (2003) recent work using the ICRP 66 lung model, stochastic intakes of WGP were modeled, for single, 10, and 100 particles. The variability of deposited activity was highly dependant on the region of the respiratory tract of interest. The alveolar region had the least variability in deposited activity, with variability increasing for upper-more regions of the respiratory tract (Aden and Scott 2003). For the lower alveolar and tracheal/bronchial regions, under stochastic intake assumptions for WPG, the variability was low. However, for extrathoracic regions, much larger variability was observed and predominantly influenced by deposition of particles of large aerodynamic equivalent diameters. The implications of the paradigm were most marked for $^{238}PuO_2$ that is handled in the Department of Energy (DOE) complex, where the specific activity is about 240-fold higher than the WGP at the BOMARC site.

The RESRAD modeling conducted under ICRP 30 for the RI/FS assumes that the activity has a log-normal activity distribution with median aerodynamic diameter (AMAD) of 1 μm as a default, while 5 μm AMAD (log-normal) is the recommended default for occupational exposures under ICRP 66/68. In both cases, if the actual activity is distributed among particles of higher AMAD, DCF are lower. This is the expectation for the distribution of Pu activity on the BOMARC site and the case for relatively high activity discrete particles when stochastic intakes are of concern.

Application of the stochastic intake paradigm to WGP at the BOMARC site raises other paradigms. First, as already discussed, a significant fraction of the WGP left on-site after the accident was distributed in discrete particles, with aerodynamic equivalent diameters too large to afford respiratory intakes and for that matter airborne suspension. Even post remediation, some discrete particle characteristics may be present, but to a lesser degree than pre-remediation. And, for stochastic intake paradigm considerations of post-remediation soils, particle distributions will contain a reasonable fraction too large for suspension, making that fraction of activity unavailable for respiratory dose. Second, inhalation dose conversion factors used in the 1992 risk modeling (Earth Tech 1992), assumed a Class W material. As noted above, a Class Y DCF, as applicable to PuO_2, was 40 % lower than the Class W DCF under FGR 11. Under more recent ICRP guidance [Table A-5, (ICRP 1991, 1995)], a 30-year weighted PuO_2 DCF (termed "Type S") for the general public is about 10-fold lower than that used in the RI/FS risk modeling (Earth Tech 1992). As noted earlier, it is not plausible for high-activity $^{239+240}Pu$ particles to exist on the site in a non-dioxide chemical form. If high-activity discrete particles are of concern, use of a DCF applicable to a more soluble and mobile form of Pu is not appropriate. In principle, the two considerations are mutually exclusive.

2.2.2.5.3 Ingestion.

Ingestion is the second most important pathway for the resident-farmer scenario and comprised an overall fraction of modeled risk of 22 %, or 2.2×10^{-5} over 70-years. The impact of heterogeneity (discrete particles) on modeled ingestion doses has similar type of paradigms to that of the case of inhalation. First, the existence of high-activity discrete particles and the dose modeling used for the ingestion exposure route contradicts the conservative use of Class D gastrointestinal tract uptake

factor, f_1. Second, if one places great concern on potential ingestion of high-activity discrete particles, of necessity, an assumption must be made that a vast majority of residual activity in soils is comprised of such particles. Subsequently, inhalation risks would be virtually non-existent, as these particles would not be of proper aerodynamic equivalent diameter to allow air suspension.

To evaluate risks under the stochastic intake paradigm, data from the Class 1 FSS (Duratek 2006) were used to form a case study. This data set is most appropriate for review since it provides an extensive analysis of residuals from remediation, which comprise over a 1000 FSS samples. From Table A-1, survey unit 2 had the highest estimated mean residual concentration among systematic grid samples combined with area-weighting EMC concentrations among the survey units, with a mean of 3.24 pCi/g. Among the 154 samples, the highest total sample activity was 17 nCi, and is an estimate of the highest activity residual particle, assuming the entire sample activity is the result of a single particle. For a 2000 m^2 survey unit, monoactivity discrete particles are assumed to be dispersed in the top 15 cm of soil, of density 1.5 g/cm^3. Per the RESRAD modeling, 35 g of soil is ingested per year by a site resident [RI/FS, Appendix J (Earth Tech 1992)]. Table 2.2 contains

TABLE 2-2. Estimated Risks for Stochastic Intake Paradigm for Ingestion of 17 nCi (0.017 μCi) Monoactivity $^{239+240}$Pu Particles (Mean 70-year Probability of Single Particle Ingestion = 0.47).

DCF Source	Class	Particle Number	70-yr Ingestion Probability	Ingested Activity (μCi)	Dose Conversion Factor (mrem/μCi)	Effective Dose Equivalent (mrem)	70-year Risks (x 10^{-4})	
							Separate	Total
FGR 11 ICRP 26/30/48	D	1	0.293	0.017	$3.5 \times 10^{+3}$	60	0.062	0.099
		2	0.068	0.034		120	0.029	
		3	0.011	0.051		180	0.0068	
		4	0.0012	0.068		240	0.0011	
		5	0.00012	0.085		300	1.2×10^{-4}	
	Y	1	0.293	0.017	52	0.88	0.00092	1.5×10^{-3}
		2	0.068	0.034		1.8	0.00043	
		3	0.011	0.051		2.7	0.00010	
		4	0.0012	0.068		3.5	1.6×10^{-5}	
		5	0.00012	0.085		4.4	1.8×10^{-6}	
ICRP 60/72 (Public)	M	1	0.293	0.017	$1.5 \times 10^{+3}$	26	0.027	0.042
		2	0.068	0.034		51	0.012	
		3	0.011	0.051		77	0.00029	
		4	0.0012	0.068		102	4.5×10^{-4}	
		5	0.00012	0.085		128	5.3×10^{-5}	
ICRP 68/60, 78 (Workers)	S	1	0.293	0.017	33	0.56	0.00059	9.3×10^{-4}
		2	0.068	0.034		1.12	0.00027	
		3	0.011	0.051		1.68	6.3×10^{-5}	
		4	0.0012	0.068		2.24	9.9×10^{-6}	
		5	0.00012	0.085		2.81	1.2×10^{-6}	

summary calculations for the case study. Over a 70-year period, the mean probability of ingesting a single particle is 0.47. Figure 2.1 and Table 2.2 contains probabilities of ingesting various particle numbers based on a Poisson probability distribution. For various particle number and ingestion DCFs, 70-year cancer risks are listed under the assumption that 1 x 10^{-4} risk is equivalent to 280 mrem over a 70-year exposure time. Coincidently, 280 mrem is about the average background radiation Americans receive from natural background sources.

From Figure 2.1, the highest probable event is the ingestion of zero particles, 63 %, with the vast majority of total risk from single particle ingestion. Naturally, the probability function is highly dependent on the assumptions made in the model. A probability function of ingested particles derived from a broad distribution of particle activity's, would have a mean ingested particle number significantly higher than the case presented here, but much lower in mean particle activity and less variability in total ingested activity. However, the probability of ingestion of high-activity particles would be significantly lower than described here.

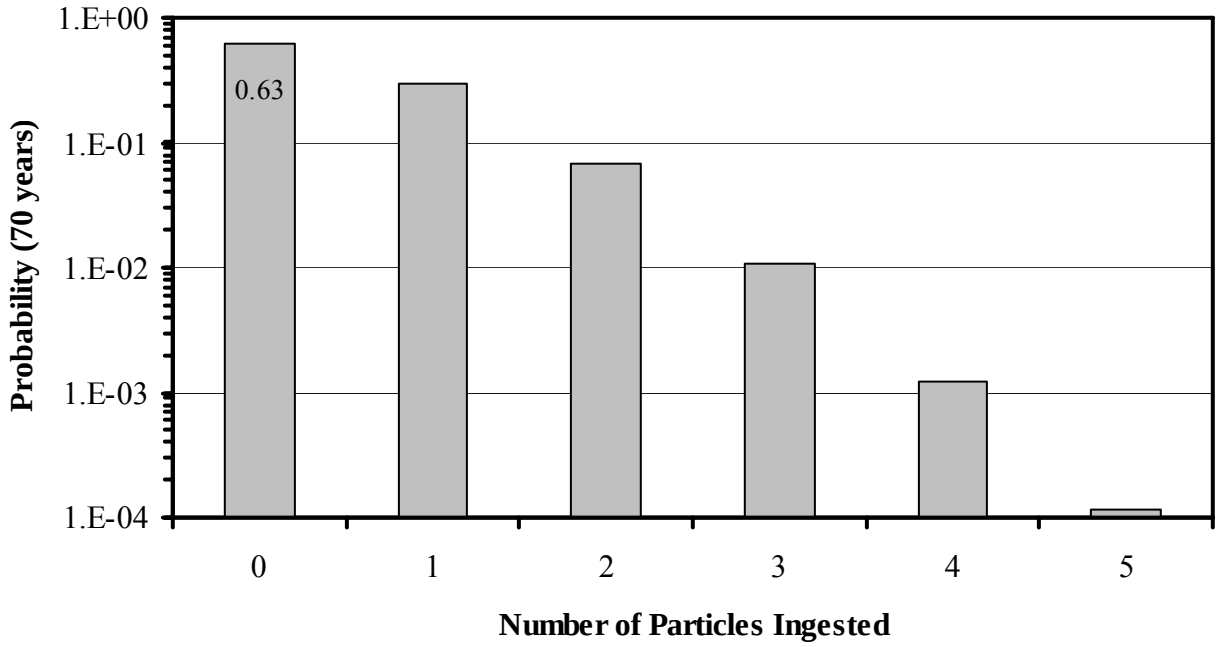

Figure 2-1. Seventy-Year Probability of Ingesting 17 nCi Particles.

Under ingestion Class D plutonium, FGR 11, a single 17 nCi particle provides an effective dose equivalent of 60 mrem, while two provides 120 mrem, which is a little over a single year acceptable general public exposure limit, 100 mrem. However, in the stochastic intake paradigm described here, the total risk, which includes the probability of intake and consequential risk of the intake is only a small fraction of this value, 0.099. If the average concentration in the survey unit was at the criterion, the total risk is 0.24 x 10^{-4}, which is very close to 22 % of the total risk projected under the deterministic model (see Table 2.1). Most important, however, in this case study, inhalation doses are not possible because the minimum diameter of pure 17 nCi $^{239+240}$Pu particles is 34 μm, which equates to an aerodynamic equivalent diameter of 115 μm (spherical). An even more compelling

paradigm for the stochastic intake of these high-activity discrete particles is use of Class D, DCF (FGR 11), which is 67-fold higher than a Class Y, DCF, which is appropriate for PuO_2. As discussed above for inhalation exposures, it is not plausible for high-activity $^{239+240}Pu$ particles to exist on the site in a non-dioxide chemical form.

The lower rows of Table 2.2 provide modeled ingestion risks based on more recent ICRP recommendations. The lowest row is applicable to ICRP 78, Class S, for workers, which is even lower than FGR 11, Class Y. With this DCF, ingestion of a 0.8 μCi particle provides an effective dose equivalent of 26.4 mrem, which is 9 % of a 70-year integrated dose criterion.

Figures A-15 and A-16 provide a summary of field findings from the Cabrera Services 2006 particle removal project (Cabrera 2006a). Figure A-15 is a scatterplot of estimated particle activity vs. depth. Evident from this project, the majority of high-activity particles removed had activities significantly higher than 1 μCi. Figure A-16 provides a scatterplot of in-situ FIDLER response and removed activity for locations with particle activity less than 6 μCi. From this plot, it is clear that particles with activities near 1 μCi are field-detectable with a FIDLER. Since this field effort concentrated on areas with the greatest residual activity, few particles with activity in this range were targeted for removal. A number of particles of this approximate diameter were located in-situ with FIDLERs, removed by coring, and isolated and analyzed by the University of Nevada, Las Vegas (Cabrera 2006d). Figure A-17 contain scanning electron microscopy (SEM) images of a 0.8 μCi $^{239+240}Pu$ particle evaluated in that effort. The minimum volume equivalent diameter of this particle is 122 μm, if pure plutonium. This particle typified the general UNLV finding, "chemically and physically stable, had limited [evidence of] physical degradation since formation."

Overall, while the stochastic intake paradigm has been discussed recently among some radiological specialists evaluating work on the BOMARC site, assumptions made in the RI/FS risk modeling are overly conservative, making the paradigm insignificant to ingestion exposures.

2.2.2.6 Risk Modeling.

There has been some general scientific debate on whether discrete particle uptake and tissue exposures pose a significantly higher cancer risk compared to the same uptake from a more uniformly dispersed contaminant. In regard to exposure from BOMARC WGP, the implication is only important for inhalation exposures, because modeled gastrointestinal (GI) tract uptakes to the blood stream require contaminants to be in soluble form and WGP provides negligible dose to the GI tract during transit. Further, for WGP inhalation exposures, only depositions in the alveolar region of the lung are "practically" subject to spatially non-uniform exposures. Depositions of WGP in upper portions of the respiratory system are rapidly cleared to either the circulatory system or GI tract.

The ICRP has recommended application of average tissue doses in estimation of risk, even in the case of non-uniform exposures (ICRP 1991). NCRP Report No. 46 provided theoretical evaluation, and review of pertinent observations in animal and humans exposed to WGP (NCRP 1975). NCRP noted theoretical considerations for an inverse dose effect for spatially non-uniform exposure to ^{239}Pu particles in the lung and confirmation of the effect in animal studies of lung exposures to ^{238}Pu and ^{239}Pu. A recent literature review on the topic was conducted by Charles *et al*. that included

Russian workers exposed to WGP (Charles *et al* 2003). They concluded that the ICRP dose-averaging is likely to provide a reasonable estimate of carcinogenic risk.

2.2.3 Summary. Historical site investigations noted observation of a discrete particle plutonium contaminant nature. WGP involved in high-temperature ignition events in air have a propensity to form PuO_2 chemical forms. Dioxide forms of plutonium are the most stable chemical form under most soil conditions. Application of water to suppress the fire aided rapid condensation of plutonium. Recent SEM images provide conclusive evidence of that this occurred in particles sampled from the site, and that the particles examined had limited surface degradation since formation. In this from, particles are chemically and physically stable, characteristic of a PuO_2 chemical form. Further evidence of this predominant chemical form was confirmed by strong acid challenges to contaminated soils. Use of the most conservative DCFs of FGR 11 provides an overestimate of risks to a hypothetical future site inhabitant under the resident-farmer scenario. Further, the concern of stochastic intake of discrete particles under the most conservative FGR 11 DCFs is contradictory to this assumption. Evaluated with DCFs appropriate to dioxide forms are significantly lower, rendering the low probability exposure scenario insignificant. Overall, regardless of chemical form, recent ICRP recommendations for plutonium are less conservative than those used in the risk modeling that formed the basis of the ROD.

2.3 Radiological Surveys in Structures.

2.3.1 Historical.

2.3.1.1 Shelter 204. Shelter 204 was the most severely impacted of the site shelters as it contained the weapon that was engulfed in fire. The fire burned uninhibited for about 30 minutes, with the Shelter 204 area sprayed with water for another 15 hours. In addition to the effects of the fire on the interior of Shelter 204, the floor and concrete were pitted by flying fragments of the helium and fuel tanks that exploded, and steel roof beams were deformed. A large fraction of shelter interior surfaces had measurable α-radiation contamination, though wall surfaces in the north to northwest part of the shelter were not impacted. This is reasonable, since wind during initial stages of the accident response were from this direction. Though significant volumes of water were sprayed on the shelter and decontamination operations were conducted on its interior the day after the accident, some floor areas had residual in-situ α-particle count rates in excess of two million counts per minute (CPM). [Note: It is assumed that the measurements were conducted with an Eberline PAC-1S that has a probe entrance window of 60 cm^2.] The high residual contamination retention was attributed to a thin layer of tar that had melted during the fire, and spread to and on the shelter floor. Four days after the accident, impacted areas on the interior of the shelter and asphalt areas on the outside were painted to immobilize the contamination. Alpha radiation measurements conducted post paint application were mostly zero, with a few fringe areas having readings between 50 and 500 cpm.

2.3.1.2 Other Shelters. It is unknown if surface contamination levels were assessed in other shelters shortly after the accident. There is no record of any mitigation of contamination applied to the other shelters, like paint, concrete overburden, etc.

2.3.2 RI/FS.

2.3.2.1 General. The evaluation of shelters accomplished in the RI/FS was the most extensive that has been documented. As well, the power and communication bunkers in the front of Shelter 204 were evaluated, since the openings to these bunkers were closed shortly after the accident. It is believed that some knowledge existed of impact to the interiors of the bunkers from plutonium contaminated water from fire-fighting activities. Shelter 204 received the greatest degree of survey effort, since it was the shelter that was most severely impacted. But, it was recognized in the RI/FS that Shelters 202, 205, and 209 were possibly contaminated as a result of the fire, fire-fighting, and subsequent decontamination of exterior locations. Also, shelter 210 was suspect, as it was used as a staging area for radiological sampling activities for many years.

2.3.2.2 Shelters.

Wipe samples were collected from Shelter 204 and 20 other shelters. Figure A-18 provides a modified version of Figure ES-4 from the RI/FS (Earth Tech 1992) that has color-coded summary of wipe results. Many of the shelters in the vicinity of Shelter 204 were sampled because of the potential for contamination from fire-fighting water and efforts conducted shortly after the accident to mitigate contamination in and around the shelter. Six shelters (101–106) were wiped to investigate the potential of an airborne transport mechanism, while four outlying ones (115, 127, 216, and 228) were investigated to assess impact that may have resulted from personnel-aided translocation of contamination. One-hundred wipes were collected from Shelter 204, while 25 were collected from the others. In addition to wipes, as practical, wipe locations were screened for fixed and total α-radiation levels with an Eberline PAC-4G gas-flow proportional α-counter.

A summary of the wipe results are in Figure A-18. Besides Shelter 204, the wipe results in the other shelters were less remarkable. The highest wipe result in Shelter 204 was 414.5 dpm/100 cm^2, while the highest in the other shelters was 233 dpm/100 cm^2 (Shelter 209). Other positive results (> 20 dpm/100 cm^2 per Reg. Guide 1.86) were 39 dpm/100 cm^2 (Shelter 205), 97.2 and 28.7 dpm/100 cm^2 (Shelter 210), and 37.4 dpm/100 cm^2 (Shelter 202). Table A-7 contains a listing of elevated PAC-4G survey readings. From the table, the highest measurements were observed in Shelter 204, with the maximum at 47,780 dpm/100 cm^2. Other shelters with measurable contamination in excess of 100 dpm/100 cm^2 were 127, 202, 206, and 216. No correlation existed between the elevated swipe results (removable) and in-situ measured fixed (and removable) contamination. The footnote to Table 4-13 of the RI/FS was interesting in the vastly different lower limit of detection values for Shelter 204 and the others, with the former being 2.25-fold lower, or nearly $\sqrt{5}$. A closer inspection of conversion factors in the calculation of surface contamination was performed. The RI/FS in description of data processing stated, "The results from background PAC-4G measurements collected between July 5 and August 17, 1990 identified average background activity levels at the site to be 13.2 $\pm$ 4.0 'clicks' per minute, which corresponds to 66 $\pm$ 20 counts per minute . . ." [RI/FS, Section 4.1.3.4.1, page 4-38 (Earth Tech 1992)]. The factor of five correlating 'clicks' with counts is not understood, nor was it discussed in any other portion of the document. One may suspect that the meter had an "audible divider" on it, where only a fraction of the interactions have an associated audible click or that a scale multiplier was used. However, meters historically attached to this probe had a log-scale meter movement, negating a need for a scale multiplier, and reportedly did not have an audible divider (Frame 2007). In review of data in Table 4-16 of the RI/FS (page 4-

65), raw PAC-4G count rates are listed for a few sampling locations that had isotopic plutonium analysis of wipe samples. Figure A-19 contains a scatterplot of the paired Table 4-13 and Table 4-16 data from the RI/FS. One data point, 202-WP-016-001, required correction in its Table 4-13 calculated total concentration, as it did not correspond with the net and background concentrations. Overall, the plot separates the data into two categories: a 24X or 4.8X factor correlating count rate and concentration, with a factor of five between the two. The 4.8X factor is clearly comprised of the quotient between the probe detector area, 59 cm^2, to 100 cm^2 correction factor (1.69), and the reported detection efficiency, 0.353. The 24X factor is for data points that additionally had the factor of five applied. The basis for this factor is not known, but believed to be a technical error, as the LLD of 192 dpm/100 cm^2 for α-radiation is abnormally high in an area that has a relatively low background. One data point fits neither line and is suspected to be a transcription error in one or both of the tables. HQ AFSC and AFIOH's assessment is that many of the reported results in Table 4-13 of the RI/FS are high by a factor of five. It is a moot issue for Shelter 204, which was completely dismantled in 2002, and everything but the concrete for Shelters 202 and 206, as the upper structural portions were dismantled in 2002.

2.3.2.3 Bunkers. Figure A-20 shows the spatial relationship of the power and communications bunkers, and connecting pipes to a respective shelter. Figure A-21 shows an elevation diagram of an individual shelter. During the RI/FS, wipe and in-situ measurements of surface α-particle emissions with an Eberline PAC-4G were collected at a number of upper locations in the power and communications bunkers. Sediment samples were collected in the communications bunker and rust samples were collected from the flange area of the each bunker's manhole covers. A summary of the results are in Table A-8. Good agreement existed between wipe and in-situ measurements. Rust samples from both shelters had quantifiable ^{241}Am, as screened on-site by a hyperpure germanium (HpGe) detection system. Both of these underground structures were demolished in 2002.

2.3.3 AFIOH. During evaluation of Class II and III areas in 2005 and 2006, AFIOH/SDR conducted α-radiation surveillance of personnel, survey instruments, stick mats placed on road surfaces adjacent to soil, and some shelter interior surfaces (Hensley 2007). Among the measurements, no remarkable α-radiation readings were recorded, even at a floor area of Shelter 213 that had an FIDLER reading, indicative of surface contamination.

3.0 Shelter and Bunker Survey Strategy

3.1 Contamination Transport Mechanism(s).

3.1.1 Water. Water transport was identified and confirmed by characterization surveys as the most important mechanism for transport of radiological contamination from Shelter 204 to other locations on- and off-site. For shelters in the vicinity of 204, it has been suggested in the RI/FS that these likely had some impact due to close proximity of fire-fighting activities. As well, the contamination impact to the Shelter 204's associated power and communication bunkers was most likely from fire-fighting water intrusion. This potential was further supported by results of the 1997 characterization, where contamination was detected to below grade depths of 16 feet, directly in front of Shelter 204 (OHM 1998). However, contamination was not identified at these depths in any

other contaminated areas investigated. It was logically concluded from this data that the contamination at these depths may have been transported through the concrete junctions in the Shelter 204 communication and power bunkers. Impact to other bunkers is unknown, but largely bounded by potential for water intrusions and contamination in the vicinity of the manhole covers. Bunkers associated with Shelters 202, 204, and 206 had the greatest potential for impact from fire-fighting water, due to proximity to the fire, but have already been removed. Bunkers associated with Shelters 208, 210, 212, and 214 have a much lower impact potential due to distance from Shelter 204 and elevation gradient. Other bunkers have even lower impact potential due to greater distances.

3.1.2 Mechanical. Mechanical transport by equipment or personnel during the accident response and recovery, or from translocations by the same means after the accident was identified in the RI/FS [Section 4.1.3.5, Page 4-73 (Earth Tech 1992)] as a possible mechanism. Based on the amount of residual activity and amounts found in outlying areas during the Cabrera Services particle removal conducted in 2006 (Cabrera 2006b) this is a distant secondary mechanism to water transport. All shelters have some potential for residual contamination due to normal operations post accident. Though most impacted areas on the site had some mitigation method to fix contamination in place, e.g. paint, concrete and asphalt overburdens, some soil areas in the vicinity of Shelter 204 and the drainage ditch had low levels of contamination in uncovered soils. Also, identified in 2005, an area on the southern portion of the site, near the dining facility and fire station, had apparently been used to washdown contaminated vehicles during the accident response. These unmitigated areas had the potential to support translocation of contamination to areas that were not previously impacted. However, the degree of impact to areas not directly contaminated during the accident or from response actions are expected to have significantly lower contamination concentrations and areal extent. As such, bunkers and shelters not subject to fire-fighting water have lower probability and expected degree of impact.

3.1.3 Storage and Use. Some shelters were used to store equipment or wastes supporting environmental surveillance and environmental remediation. Among shelters without impact potential from fire-fighting activities, these have a greater potential for impact due to the materials stored and increased foot and equipment traffic.

3.2 Contamination Retention. Contaminant retention characteristics in soils and the surfaces of materials is an important consideration for scoping survey work planned for shelters and bunkers. According to records, the only facility that had paint applied to mitigate contamination was Shelter 204. Surface coatings have not been observed in other shelters. Lubricating oils, transformer oils, hydraulic fluids, and solvents were used in many of the site facilities in addition to the shelters. Surface deposits of these fluids may have aided retention of contamination on concrete surfaces. Hydraulic oils were removed from 70 of the 83 shelters that had 200-gallon reservoirs, with pits in 65 of these cleaned with a high-pressure washer and steam cleaner, with the pits in the other shelters cleaned by hand (Cabrera 2006c). Hydraulic fluid was not removed in shelters 201 to 214 because of known and perceived concern for residual contamination. Pits in the shelters, other low elevation points on concrete surfaces, wall/floor interfaces, and locations with surface oil penetrations are key locations for retention. Retention in Shelter 204 surfaces was aided by a surface coating of tar and other processes that are related to the intense heat of the fire. These are not applicable to the other shelters or bunkers. Potential contamination in shelters and bunkers should retain the characteristic of heterogeneity that has been widely observed in impacted soils. As such, it is reasonable to

assume that if a shelter has a significant degree of impact, some particle(s) would exist in the contaminated area that would readily be detected by a FIDLER. Since water, personnel, and vehicle movements are the important tracking mechanisms, contamination deposition on floor surfaces, sumps, and missile shelter pits are most likely. Deposition of contamination on walls, roofs, missile launchers are must less probable and only suspect if contamination is identified on adjacent floors, pits, etc.

3.3 Priorities. Table A-9 contains a prioritized list of structures for scoping surveys. Structures highlighted in red have greatest priority, with lesser degree for orange, yellow, and green in order. Priority is based on likelihood of impact and the expected degree of impact. Naturally, among the shelters, 201, 203, and 205 are the highest priority because they had the greatest potential for contamination from fire-fighting activities that is considered a primary source. Other shelters have a lesser probability and expected degree of impact because the contamination would require translocation from a primary contaminated area. Logically, translocations would involve some dilution of the contaminant and encompass isolated areas like that of a shoe or tire tracks. The priority aids in setting the degree of scoping individual structures receive. Structures with highest priority will receive the greatest degree of survey effort, in line with strategy established in MARSSIM (NRC 1997). As well, consistent with MARRSIM, some change in priority may occur during the course of this work due to survey findings in a structure(s). Depending on accessibility, pit areas may not be surveyed as extensively as floor areas. From a risk standpoint, these areas present little potential for air suspension and ingestion of contamination because of the limited access. As well, since these areas had limited access after the accident, direct deposition of contamination from personnel would be significantly lower than for floor areas. Translocation of contamination from other areas is the only likely deposition method.

3.4 Overall Sampling Strategy.

3.4.1 In-Situ Low-Energy γ-Radiation Scanning. In-situ γ-radiation scanning with a FIDLER will be the first survey method for shelter interiors and serves two primary purposes. First, it will allow rapid identification of any locations with high-activity discrete particle locations. As the contaminant is dominated by discrete particles, this is perhaps the most important survey technique. Further, since the contaminant has limited solubility and penetrability in concrete, insignificant attenuation of the 59.5 keV photons from ^{241}Am will be observed for surface assessments, even with a thin overlying layer of dirt or paint. Second, identification of areas of high-activity particles allow isolation of these areas for more in-depth surveillance, particle isolation, spot particle removal, and application of greater personal protective measures. Also, because higher priority shelters are likely to have in-situ α-radiation screenings, surface dirt will be removed to lessen the potential for α-particle masking. Areas identified by the FIDLER screen as being impacted will require greater care in dirt removal than other areas.

3.4.2 In-Situ α-Radiation Scanning and Static Measurements. In-situ α-radiation scanning is anticipated for higher priority shelters and some interior surfaces of priority bunkers. Lower priority shelters may have some limited scanning, with the extent primarily based on potential for impact and the results of the FIDLER surveys. Both gas-flow proportional floor monitors and hand-held portable α-radiation scintillators will be used. Bunker interiors and areas of shelters with limited access will be assessed with hand-held instruments. Structures, like the fire-house, can be more

efficiently scanned with a floor monitor, with hand-held instrument use to more inaccessible locations of the structure. Walled surfaces will not be assessed unless in close proximity to an impacted floor location. Close attention will be made to floor surfaces with surface coatings like paint, epoxy, etc. that could reduce or completely mask α-particle detections. To address this potential, some surfaces may be gently abraded to remove surface material. In addition to scanning measurements, some locations may be assessed by static (fixed) measurements. This is likely for grid locations as MARSSIM-defined "direct measurements" in the event that a final status survey is performed in some high-priority shelters.

3.4.3 Wipe Samples. Contamination in a loose (removable) form is essential to significant exposure of individuals from the shelters, since radiation from fixed contamination provides insignificant external radiation exposure. Wipe samples provide an important measure of removable contamination and will be accomplished in significant number in high-priority shelters. All samples will be analyzed at AFIOH/SDRR for gross α-radiation, which will encompass the emissions from $^{238+239+240}$Pu, $^{234+235+238}$U, and ^{241}Am, all α-emitting radionuclides residuals from the accident, and naturally-occurring radionuclides, indigenous to the site. For structures with large numbers of wipe samples, after gross α-radiation analyses are accomplished, wipes will be composited by structure, ashed, and analyzed by isotopic plutonium. It is anticipated that for shelters with positively identified contamination, based on portable instrument response, a wipe will be collected in every survey grid, with additional samples being collected in grids with identified contamination.

3.4.4 Dirt/Sediment Samples. Dirt collected from preparation of floors for survey will, on a case-by-case basis, be collected and analyzed by high-resolution γ-spectroscopy for ^{241}Am content. These samples provide some information of potential for transference of contamination to loose surface material. As available, sediments from bunker sumps will be collected and analyzed by the same method.

3.4.5 Water Samples. Some of the shelters pit areas have standing water. Water samples will be collected in selected shelters. Priority in shelters selected for sampling will be based previous findings of contamination in a shelter from other survey methods described above.

4.0 Shelter Disposition Considerations and Residual Contamination Criteria

4.1 General.

Acceptable residual contamination criterion for the shelters and bunkers are predominantly based on future intended uses. The RI/FS and ROD were developed under CERCLA. The RI/FS predominantly focused on the extent of contamination in soils, current risks to the public, and risks to a future site inhabitant under the resident-farmer scenario. This unrestricted use scenario is generally considered a "worst-case" use because it encompasses the most significant potential exposure routes. While remediation of soils to this use scenario was selected, a future use has not been determined, and may not be for some time. The shelters provide little to no practical use for future military missions, and are not acceptable for unrestricted access to members of the public because of structural and other site hazards. As such, an unrestricted future use would likely incorporate demolition of the structures, with concrete and steel being recycled or disposed. Over

the past 35 years after the site was closed, a few shelters have been used to store investigation-derived waste and equipment during surveys and remediation. Similar uses are plausible in the future.

In general, CERCLA is not applicable to structures. The RI/FS did not establish a risk-based remediation goal for structures; however, criteria in U.S. Nuclear Regulatory Commission (NRC) Regulatory Guide 1.86 were considered relevant and appropriate for an unrestricted release and recommended in the absence of a risk-based criterion (Earth Tech 1992). Sophisticated computer risk modeling codes, like RESRAD for exterior soils, had not been developed at the time of the RI/FS for building interiors. In the mid-1990s, a RESRAD model for building interiors, RESRAD-Build, was developed; the latest is Version 3.3 (Yu *et al* 2003). RESRAD-Build is used here to model doses to site workers that may intermittently use a shelter for storage. In one case examined here, 10 % occupancy for a 2000 hour work year is assumed, with a 100 mrem annual exposure limit. If the shelters are demolished, the concrete could be buried on-site or at an off-site burial site. Under this condition, contaminated concrete can be treated with similarity to soil, in some respects. Buried concrete 1) does not provides a contaminated surface suitable for airborne release to receptors and 2) if ever regenerated for a future use, the original contaminated surface would be mixed with other materials, lending to a more appropriate volumetric risk modeling.

4.2 Regulatory Guide 1.86. Limits specified in Reg. Guide 1.86 are listed in Table B-1, with the row applicable to WGP and ^{241}Am highlighted in gray. Of the shelters evaluated in the RI/FS and still remaining, only three had removable contamination in excess of Reg. Guide 1.86 criteria, with two shelters in excess for in-situ measured locations. Reg. Guide 1.86 does not incorporate provisions for averaging concentration over an entire structure. Thus, if one area has been impacted with concentrations above the criterion, an acceptable release would require mitigation of these locations. In the case of RESRAD modeling, however, provision is made for small areas of contamination in excess of the average concentration criterion. In MARRSIM terms, these are referred to as elevated measurement comparison (EMC) areas.

4.3 RESRAD-Build Modeling.

The summary results of the RESRAD modeling are provided in Table B-2 for $^{239+240}$Pu, and ^{241}Am, and the total. In the modeling, it was assumed that the $^{239+240}$Pu to ^{241}Am ratio was 5.4. The modeling was conducted for various acceptable exposure levels. The most appropriate acceptable exposure level for a worker on-site is 100 mrem in a year, which is equivalent to the Nuclear Regulatory Commission (NRC) acceptable exposure limit for members of the public from licensed operations. This limit is broadly applicable to all ionizing radiation exposures in the Air Force - machine-generated or radioactive material (RAM), based on Air Force Instruction 48-148. In application of this limit to exposure of workers at the BOMARC site, it is important to understand that the exposure is under conditions of institutional control, where workers exposed below the criterion would be categorized as non-radiation workers and those with potential for exposures above as occupationally-exposed “radiation” workers.

The other cases are for a 4 mrem annual exposure, with the first limited to 4 mrem dose equivalent in the first year, and the second limited to a 30-year average of 4 mrem/yr. The distinction between the two is important for a WGP as a surface contaminant. The only appreciable exposure routes are

inhalation and ingestion (Table B-2), which assumes a fraction of the contaminant is in a loose (removable) form. Air suspension of the contaminant in an unclosed system, like a shelter, allows for a slow depletion of the source. With RESRAD-Build default parameters applied to this case, the modeled 30-year dose-equivalent is 2 % of that of the first year. Under a lifetime integrated risk approach of CERCLA, average risk is more important than a very low increment of annual risk. Overall, the difference between them is a factor of two (2).

Table B-3 contains the parameters used for the RESRAD-Build modeling. Many of them are default parameters used in the code; some are specific to structure like floor area and height, and others more specific to the expected future uses. For example, 192 h/y indoor fraction is deemed to be overly conservative for any conceivable future use of the shelters. As well, air exchange for the structure, 0.1/h, is on the minimum end for residential structures, with actual air exchanges much higher, as the structures were not designed to control air flow. The fraction released to the air, the air suspendable fraction, was assumed to be 0.1. This is overly conservative compared to the RESRAD-Build cited value for oxidized plutonium of 0.001 (Yu *et al* 2003). Another conservative assumption made in the model was the chemical form, FGR 11 Class W for inhalation and Class D for ingestion. As already discussed, due to the high temperature and highly oxidizing conditions existing during the accident, PuO_2 was the predominant form produced and still residual today.

Table B-4 contains area factors for various elevated measurement comparison area (EMC) sizes and associated total α-radiation surface concentrations. Since the modeling code assumes complete mixing of the contaminant, regardless of the source area, acceptable surface concentrations are inversely proportional to the area. This is generally the case for surface contaminants that provide the vast majority of dose through internal exposure routes rather than external. Figure B-2 contains an example grid system for use during shelter scoping surveys. The grid was designed around existing facility features, with individual grid areas about two square meters and over 60 in number.

4.4 Bulk Contamination. Figure B-1 contains a plot of $^{239+240}$Pu surface concentrations vs. the averaging thickness of concrete. Under this approach, an assumption is made that with the release of surface contamination from the concrete (as a buried material) there will be an accompanying degradation and a bulk release of concrete. An alternate approach in meeting RI/FS-specified average soil concentrations of 8 pCi/g $^{239+240}$Pu for buried concrete can be achieved by a mix with clean soil. The bulk thickness of clean mix is dictated by the surface concentration of Figure B-1. Under this approach, bulk release of surface concrete is not a required assumption. For burial of contaminated concrete on site, this criterion is expected to be easily met for surface contamination levels in remaining shelters, as documented in the RI/FS (summarized in Table A-7).

4.5 ALARA. The as low as reasonably achievable (ALARA) has been applied to the remediation of soils on the site, and for consistency should be applied to contamination identified in the shelters. Application of the principle should take into account the contaminated area(s), effort required for removal (i.e., light abrasives, needle guns), and level(s) of contamination. In this scoping effort, minor contamination may be removed.

4.6 Summary. Reg. Guide 1.86 was proposed in the RI/FS as a relevant and appropriate criterion for residual contamination in shelters and bunkers in lieu of health based standards that did not exist at the time. Two likely long-term shelter dispositions were considered: minor occupancy of workers

and burial of concrete materials on-site. Health-based criteria were developed for both scenarios. RESRAD-Build was used to model exposures for the minor occupancy scenario and mixing the contaminant with clean material was developed for the burial scenario. Other future scenarios are unlikely.

5.0 Materials and Methods

5.1 Instruments. Table 5-1 contains a listing of instruments planned for the survey work and associated details. Instruments will be provided and calibrated by AFIOH/SDR. The FIDLER will have additional daily calibrations in the field in an interior location that is expected to have background conditions similar to the shelters being investigated. A National Institute for Standards and Technology (NIST) traceable source will be used. The other instruments will have daily response checks and background radiation assessments.

5.2 Measurements. Scanning and static measurements will be collected in grid areas in accordance with the recommended approach in MARSSIM. For surfaces evaluated with wipe sampling, evaluated areas will be between 100 and 300 cm^2, dependent on the area being evaluated. In general, systematic wipes collected on floor surfaces will encompass 300 cm^2, while other test surfaces, like bunker entry flanges, equipment may be evaluated over lesser areas. It is important to note that maximum averaging areas for wipe samples under Reg. Guide 1.86 is 100 cm^2, but this has no practical health-based application to the RESRAD-Build modeling. Under RESRAD, the maximum averaging area is that within a grid.

TABLE 5-1. Instrumentation.

Description	Instrument	Use	Specifications
Portable α/β-Scintillator (α-Only Mode)	Ludlum Model 43-89 w/ Model 2360 Ratemeter/Scaler	Floor, Other Surface, & Personnel Frisking Surveys	Area: 125 cm^2 α-Efficiency: ~ 0.15
Gas-Flow Proportional Floor Monitor	Ludlum Model 43-37 w/ Model 2221 Meter	Floor Surveys	Area: 462 cm^2 α-Efficiency: ~ 0.18
Large Area Scintillator (FIDLER) or Other Large Area γ-Sensitive Scintillator	Bicron Model G5 w/ Ludlum Model 2221 Meter	Floor Surveys	Area: 12.7 cm diameter NaI(Tl) Thickness: 1.6 mm
			MDC (@ 30 cm)* Point: ~ 140 nCi (52 kBq) Area: ~ 360 nCi/m^2 (8 kdpm/100 cm^2)
			MDC (@ 10 cm)** Point: ~ 21 nCi (7.8 kBq)
			MDC (@ 5 cm)** Point: ~ 10 nCi (3600 Bq)

* From NAES Background Study (Rademacher *et al* 2002). ** Extrapolated from 30 cm height data.

5.3 Analytical Methods. Two laboratory methods are planned for sample analysis, as summarized in Table 5-2. Samples will be analyzed by AFIOH/SDR, with approximate minimal detectable activity (MDA) values listed in the table.

TABLE 5-2. Analytical Methods.

Sample Type	Method	Approximate Minimum Detectable Activity
Wipes	Gross-α/β Radiation	2 pCi (4.4 dpm)
	Isotopic Plutonium	70 fCi (0.16 dpm)
Dirt	High-Resolution γ-Spectroscopy	^{241}Am: 0.13 pCi/g (0.29 dpm/g) $^{239+240}$Pu: 0.7 pCi/g (w/ ^{241}Am Surrogate)
Water	Isotopic Plutonium	70 fCi (0.16 dpm)

5.4 Quality Assurance/Quality Control. Ten percent of the measured grids and swipe samples will have duplicate measurement or sample collection. Relative percent difference values will be calculated for paired measurement locations.

5.5 Chain of Custody. Chain of custody forms will be used for sample transfers to AFIOH/SDR.

6.0 Health and Safety

6.1 General. Radiological data from the RI/FS for shelter interiors indicate that the most highly impacted shelters have already been removed as part of the 2002 – 2004 remediation. Among the remaining shelters, impact from contamination is anticipated to be low based on previous measurements. FIDLER measurements will provide initial data on the potential for contamination that would warrant protective measures. Some structures have been posted with asbestos warnings and Shelters 201, 203, 205, and 207 – 214 have not had residual hydraulic fluids flushed from pit areas like that already accomplished for other shelters in the complex. Care will be taken in sample collection in shelter interiors to take note of possible areas containing hydraulic fluid and insulation. These areas will be avoided and noted on survey sheets.

6.2 Protective Measures. Boot coverings and latex gloves will be used to minimize cross-contamination potential. Floor contact surfaces like boots, knees, and hands will be frisked to evaluate contamination levels. Full-face air purifying respirators will be available, however, based on previous measurements their use is not deemed necessary for the survey measurement phase. If small scale removal operations are accomplished on contaminated areas, respirators will be used as a precaution.

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Appendix A

Historical Data

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Table A-1. Summary Data from Final Statuis Survey of Class 1 Areas, Compiled from Duratek (2006).

Survey Unit Number	Survey Unit Size (m^2)	Systematic-Grid Samples		Mean $^{239+240}Pu$ (pCi/g)	EMC 1		EMC 2		EMC 3		EMC 4	
		Number	# > DCGL		pCi/g	Area (m^2)	pCi/g	Area (m^2)	pCi/g	Area (m^2)	pCi/g	Area (m^2)
1	1925	45	2	2.1	14.91	18.2	4.09	10.7				
2	1925	45	4	2.92	13.32	36.97	5	22.75	7.28	10.7		
3	2000	40	0	1.17	23.42	29.8						
4	2018	44	0	1.24								
5	2125	43	2	1.97	6.48	85.3	4.32	2.7	2.82	85.3	6.37	113.7
6	2000	44	2	2.21	8.08	47.4	8.19	94.8				
7	2022	55	0	1.66								
8	1785	38	3	2.72	8.11	75.8	9.22	2.7	8.79	2.7		
9	2000	43	1	2.31	10.44	2.7						
10	1658	36	1	1.64	10.8	28.4	27.86	7.4				
11	833	19	0	1.55	7.3	42.9						
12	1800	39	0	1.5								
13	1959	42	1	2.3	3.93	2.7	8.7	2.7				
14	2031	45	1	2.18	4.36	338.9	3.82	2.7				
15	2143	48	0	1.26								
16	1654	36	0	1.05								
17	1304	27	0	1.56								
18	1406	31	2	2.11	6.16	10.7	8.12	10.7				
19	1835	40	0	1.59								
20	124	17	0	1.91								
21	1504	34	0	1.11	9.47	2.7	10.64	6.8				
22	766	18	0	1.41								
Totals	36817	829	Area-Weighted Mean	1.82								

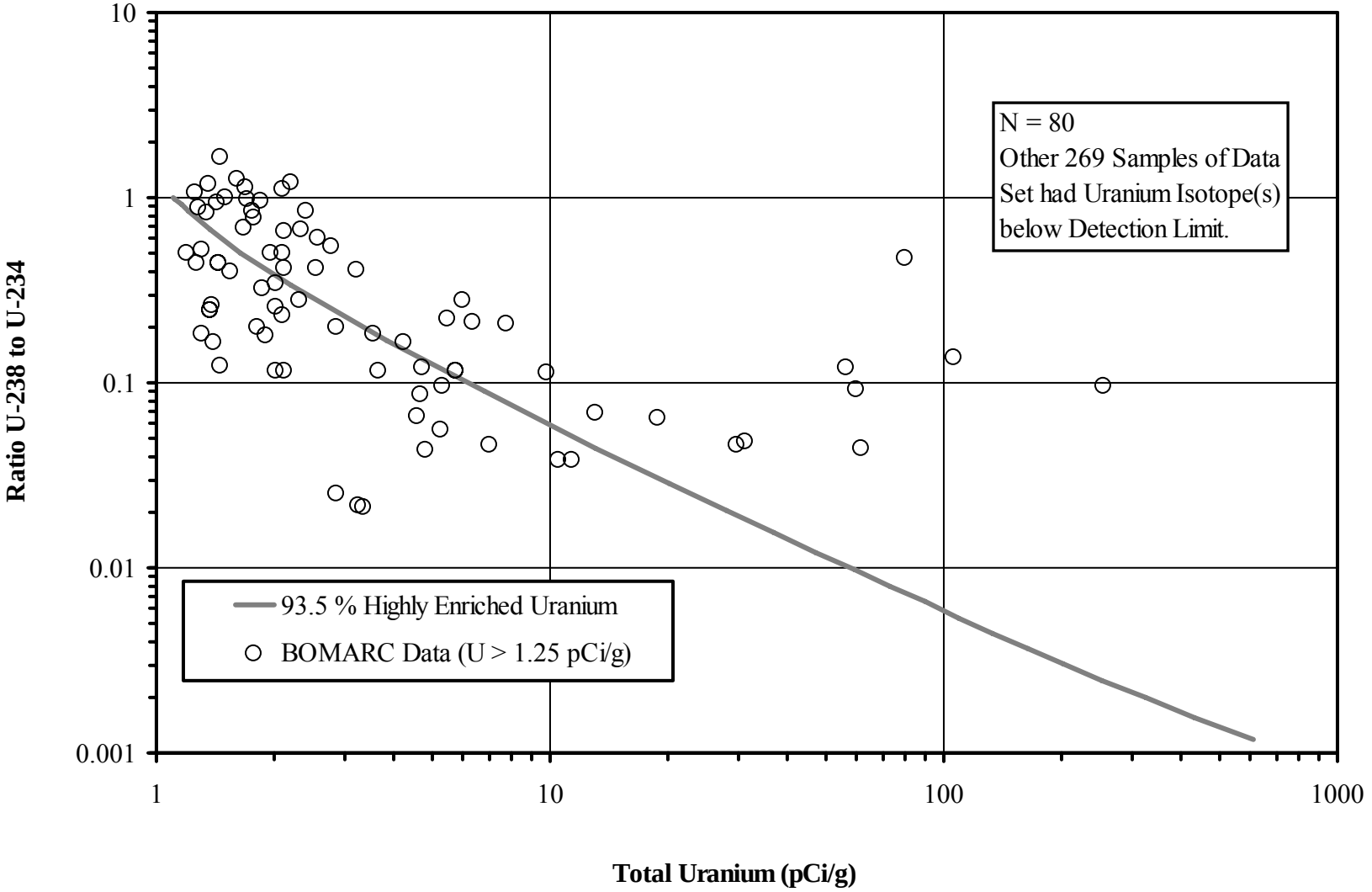


Figure A-1. ^{238}U to ^{234}U Ratios for Various Total Uranium Concentrations (Background = 1.1 pCi/g) Enriched Uranium Contaminant at BOMARC [Data from 1997 OHM Remediation] (Rademacher 2001).

TABLE A-2. Isotopic Composition of WGP in BOMARC Weapon Based on Los Alamos National Laboratory Estimates and Soil Analyses for 1958 (Rademacher 2001).

Isotope	**Mass Percent**	**α-Activity Percent**	**Radiological Half-life (y)**
Pu-238	0.0099	2.3	87.74
Pu-239	93.7	80.1	24,110
Pu-240	5.6	17.6	6,560
Pu-241	0.47	Not Applicable	14.35
Pu-242	Negligible	Negligible	376,000

TABLE A-3. Major Radiation Emissions of WGP Constituents (Scheien 1992).

Radionuclide	α-Particle Energies (MeV) & Frequency	β-Particle Energies (MeV) & Frequency	Photon Energies (MeV) & Frequency
Pu-239	5.155 (0.733) 5.143 (0.151) 5.105 (0.115)	None	0.113 (0.0005) 0.014 (0.044)
Pu-240	5.168 (0.735) 5.123 (0.264)	None	0.054 (0.0005) 0.014 (0.11)
Pu-241	None	0.021 (1.00)	None
Am-241	5.486 (0.852) 5.443 (0.128) 5.388 (0.014)	None	0.014 (0.427) 0.0595 (0.359) 0.026 (0.024)

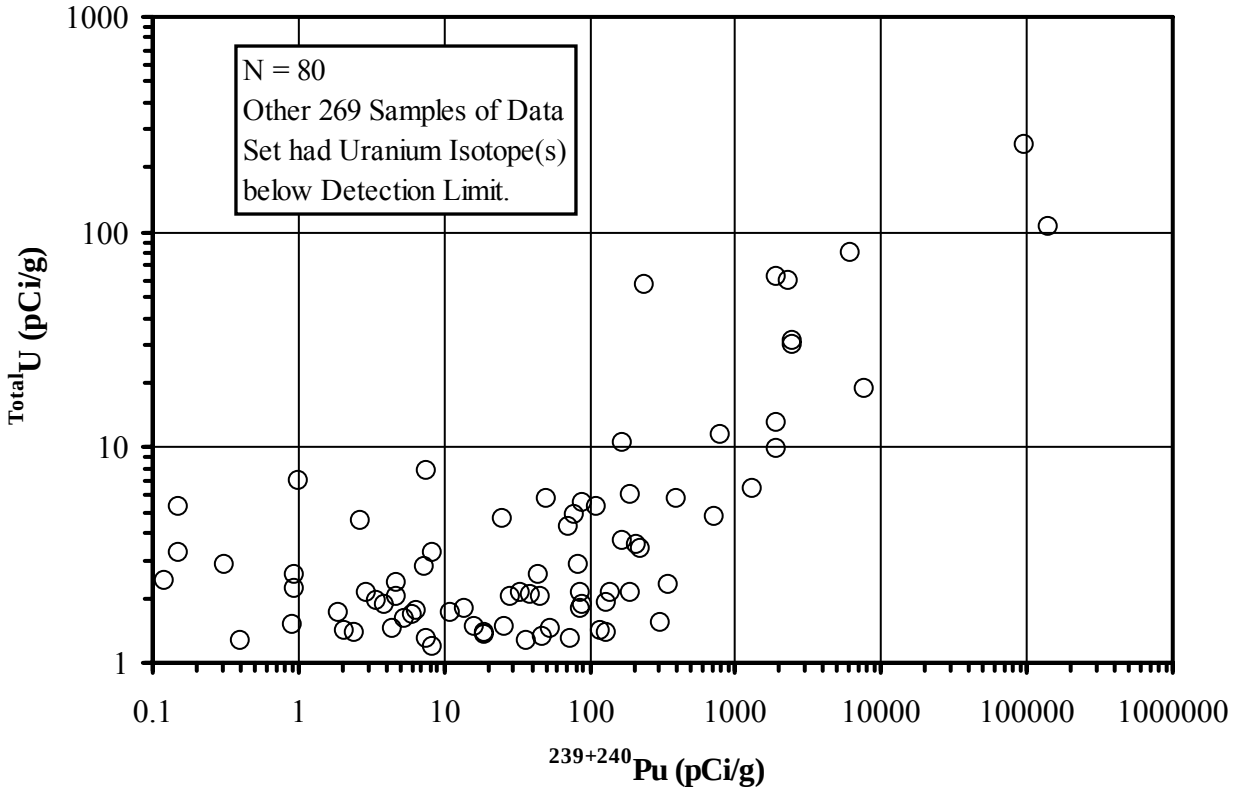


Figure A-2. Alpha Spectroscopy Data from 1997 Characterization (OHM 1998).

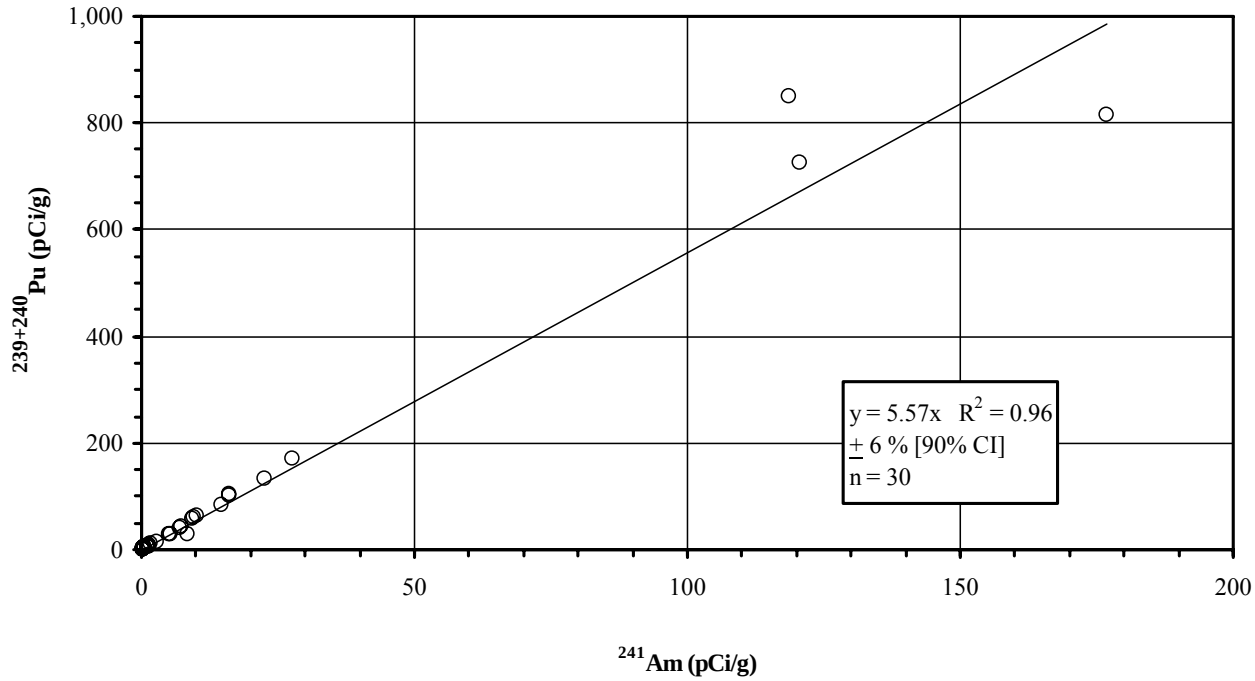


Figure A-3. $^{239+240}$Pu vs. ^{241}Am α-Spectroscopy [Data from Duratek (2006)].

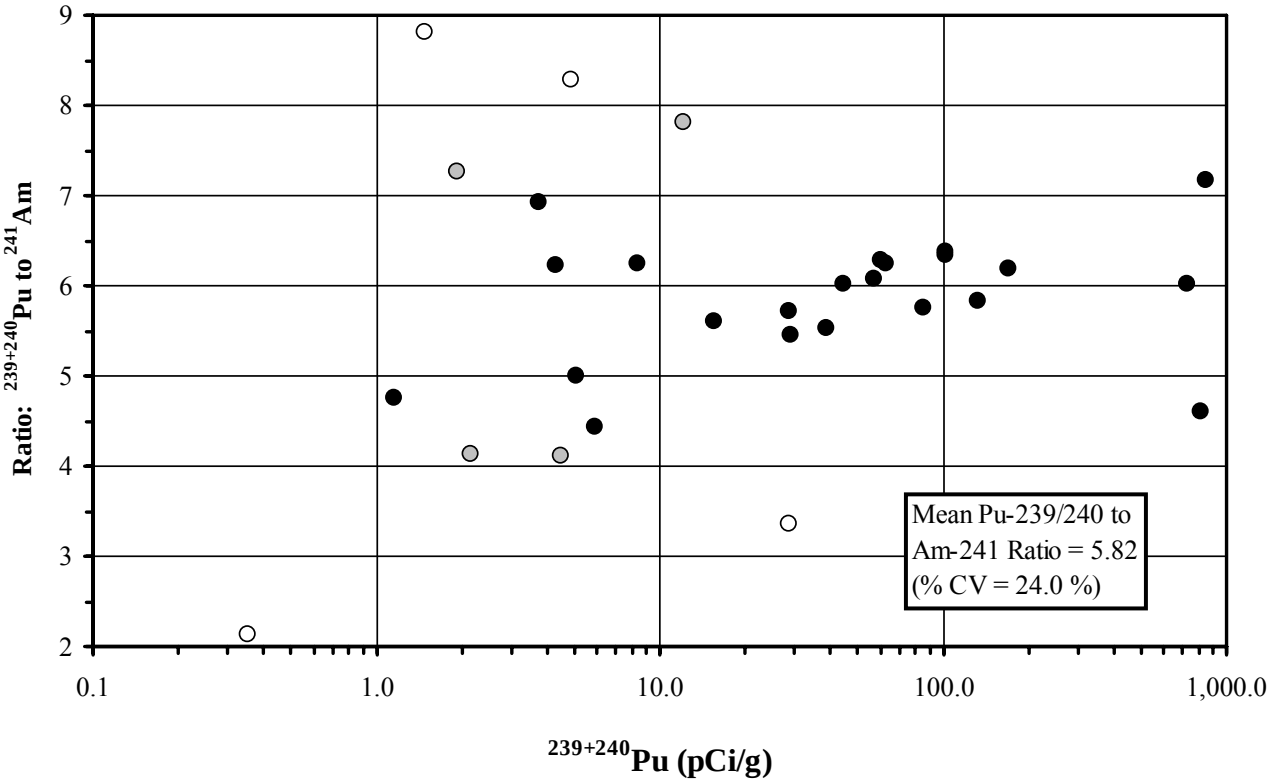


Figure A-4. $^{239+240}$Pu to ^{241}Am Ratio vs. $^{239+240}$Pu α-Spectroscopy [Data from Duratek (2006)].

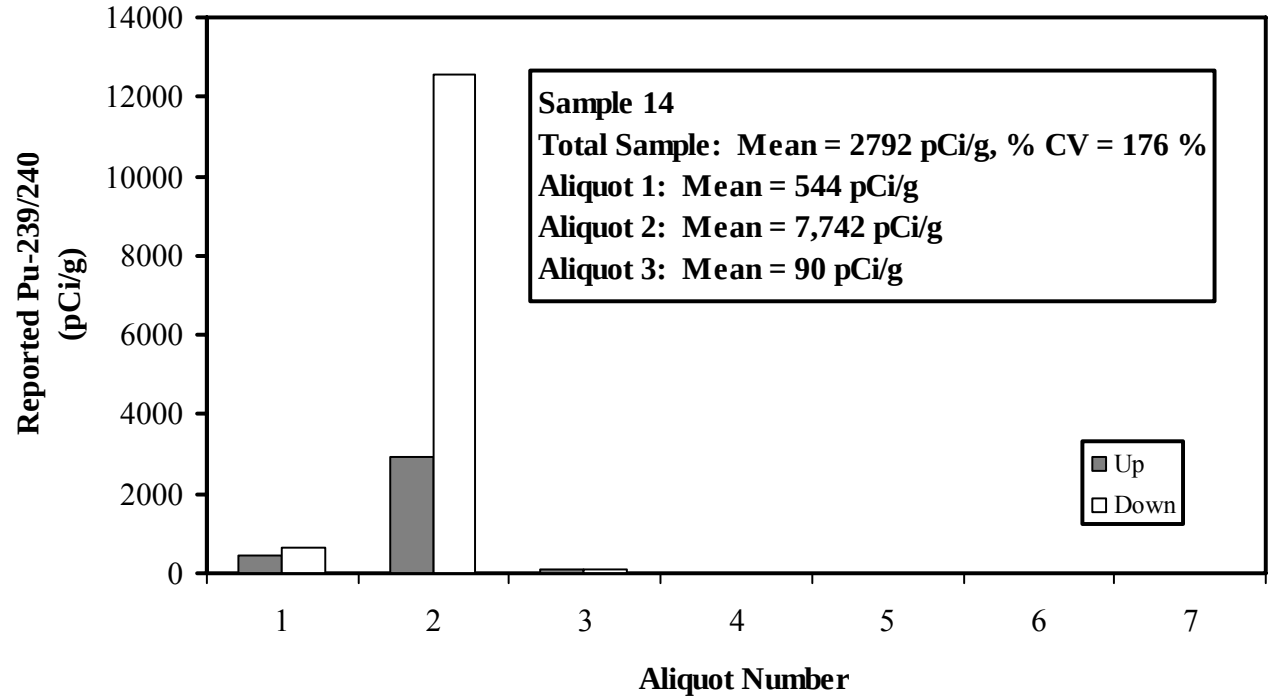


Figure A-5. $^{239+240}$Pu Activity Concentrations of Individual Aliquots (Rademacher 2001).

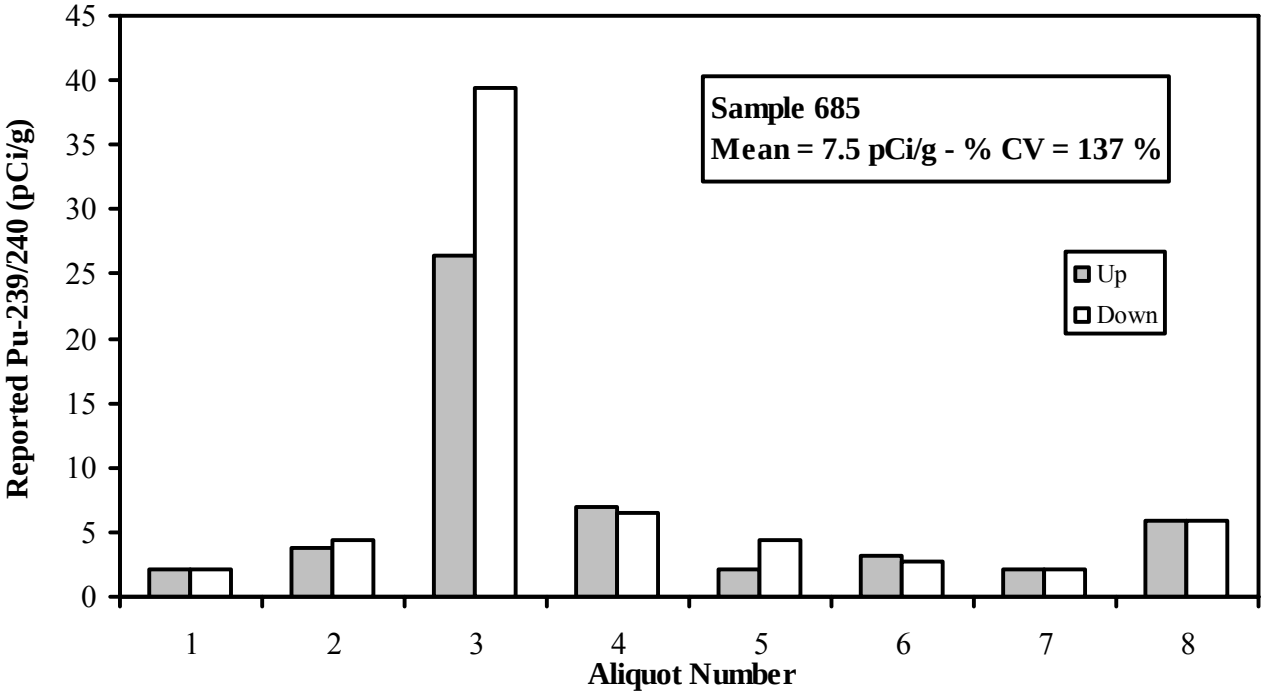


Figure A-6. $^{239+240}$Pu Activity Concentrations of Individual Aliquots (Rademacher 2001).

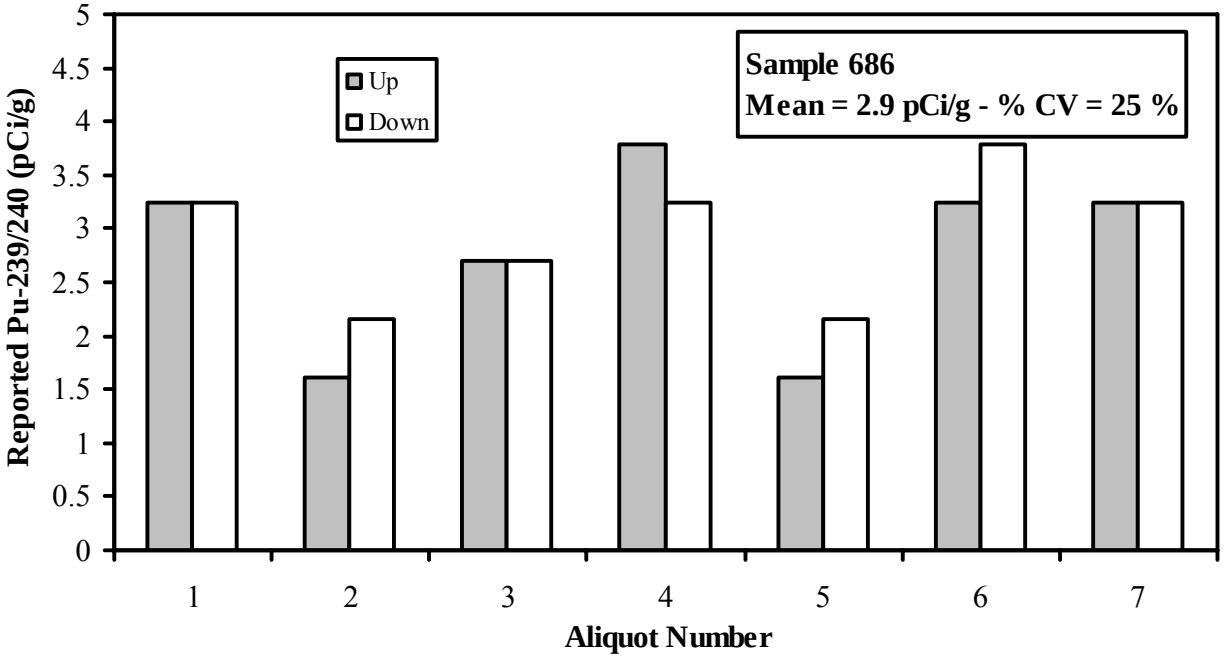


Figure A-7. $^{239+240}$Pu Activity Concentrations of Individual Aliquots (Rademacher 2001).

TABLE A-4. Estimated Activity Concentrations of Sample Segments from 5 cm Sample Cores, Assuming $^{239+240}$Pu:^{241}Am: 5.4, 38 g per Vertical Segments, and 13.3 g (A), 14.8 g (B), 6.2 g (C), and 4.8 g (D) per Horizontal Segments [Data from Cabrera (2006), Appendix E].

Sample Number	Estimated Vertical Segment $^{239+240}$Pu Activity Concentration (pCi/g)							
	1	2	3	4	5	6	7	8
4-C	1.2E+01	8.3E+05	6.1E+02	9.6E+00	7.9E+00			
5-F	0.0E+00	8.2E-01	6.4E-01	1.2E+05	3.1E+01	3.2E+00	0.0E+00	4.6E+01
9-F	0.0E+00	2.4E-01	3.2E-01	2.7E-01	4.7E-01	3.5E-01	3.6E+03	1.4E+01
10-F	8.6E+00	1.6E+00	3.0E+00	2.9E+00	2.3E+00	3.3E+00	2.1E+04	1.3E+01
11-F	7.3E-01	2.4E-01	2.0E-01	3.6E-01	1.5E+00	1.1E+01	4.5E+04	1.7E+00
12-F	6.2E-01	1.3E+00	2.1E+00	2.7E+00	1.4E+01	3.2E+04	1.2E+02	3.4E+01
15-D	0.0E+00	8.8E+02	2.9E+03	6.6E+03	7.3E+04			
16-D	6.5E+01	8.9E+01	5.3E+02	1.1E+04	1.5E+05			

Sample Number	Estimated Horizontal Segment $^{239+240}$Pu Activity Concentration (pCi/g)				Sample Number	Estimated Horizontal Segment $^{239+240}$Pu Activity Concentration (pCi/g)			
	A	B	C	D		A	B	C	D
4-C-1	3.9E+00	6.4E+00	4.7E+01	8.2E+00	10-F-8	1.1E+00	5.9E+00	1.1E+01	8.4E+01
4-C-2	1.8E+01	3.6E+02	5.1E+06	4.1E+01	11-F-6	2.0E+01	6.2E+00	4.8E+00	5.1E+00
4-C-3	5.4E+01	8.2E+01	5.1E+01	2.0E+01	11-F-7	7.6E+01	1.2E+05	3.5E+00	4.9E+00
5-F-3	0.0E+00	0.0E+00	0.0E+00	6.5E+00	12-F-5	2.8E+00	2.6E+00	4.5E+01	5.1E+01
5-F-4	0.0E+00	5.6E+00	7.2E+05	6.0E+02	12-F-6	1.3E+01	2.0E+01	1.9E+05	2.5E+03
5-F-5	0.0E+00	3.5E+01	3.0E+01	1.3E+02	12-F-7	6.6E+00	2.4E+01	5.0E+02	3.0E+02
9-F-6	0.0E+00	0.0E+00	1.6E+00	8.6E-01	15-D-4	4.0E+02	9.1E+02	2.2E+04	2.6E+04
9-F-7	9.6E-01	1.2E+00	4.0E+00	3.6E+04	15-D-5	2.4E+02	2.4E+02	4.4E+05	9.3E+02
9-F-8	5.3E+00	1.2E+01	1.6E+01	5.4E+01	16-D-4	1.5E+03	1.1E+03	1.3E+03	9.5E+04
10-F-6	1.2E+00	1.4E+00	3.3E+00	1.8E+01	16-D-5	1.0E+04	1.0E+02	9.1E+05	3.3E+02
10-F-7	5.5E+00	1.4E+01	1.7E+01	2.2E+05					

0 – **8**	**8** – 80	80 – 800	800 – 8k	8k – 80k	> 80k	pCi/g

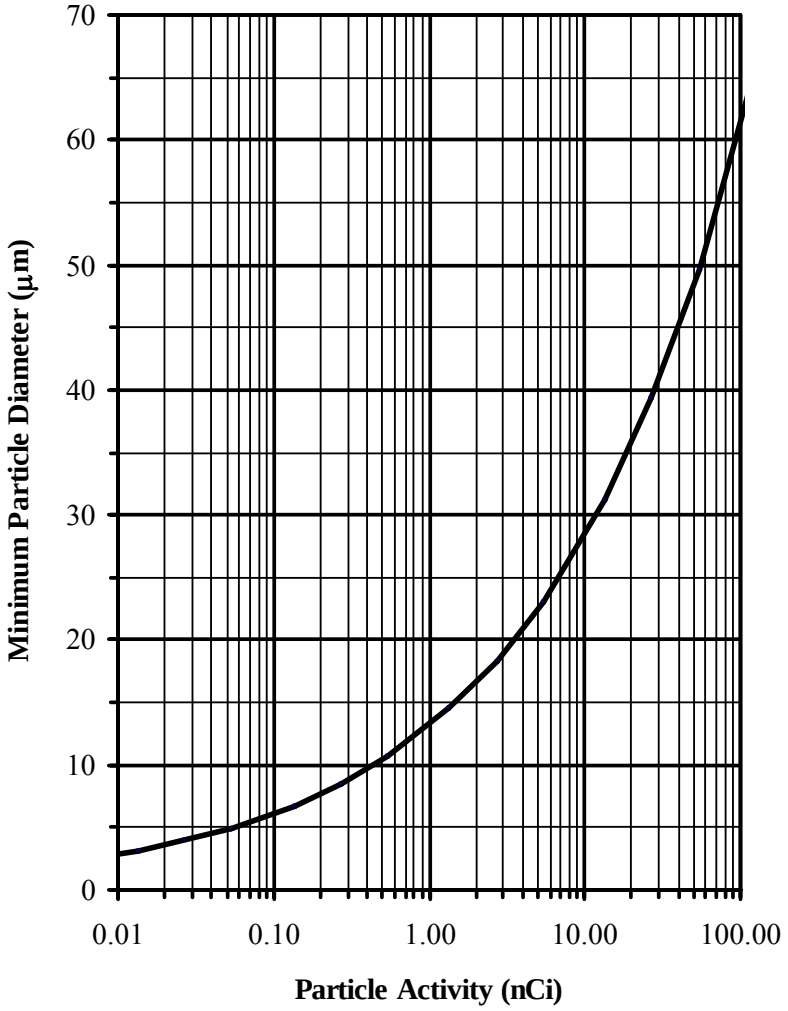


Figure A-8. Minimum $^{239+240}PuO_2$ Particle Diameter, Assuming Spherical, and ρ = 11.5 g/cm^3 (Low Range).

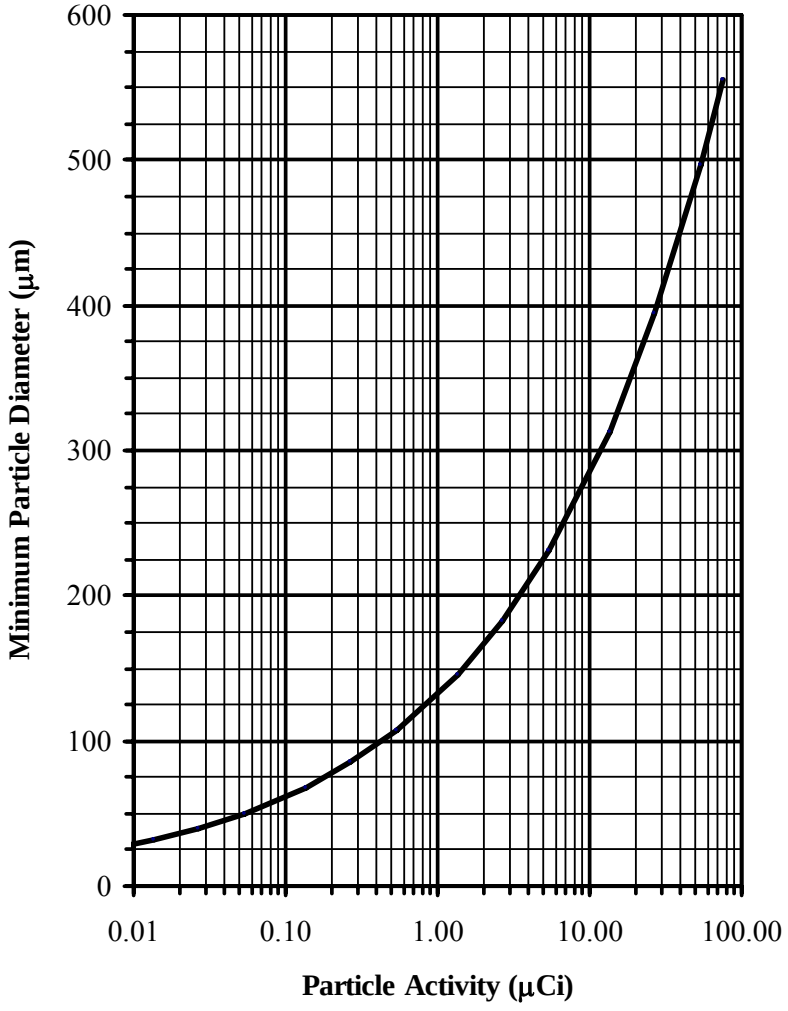
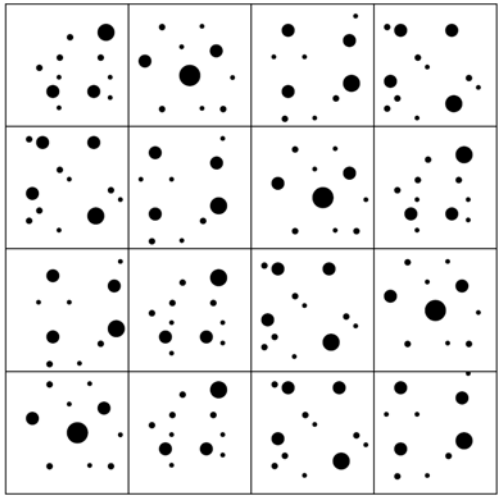
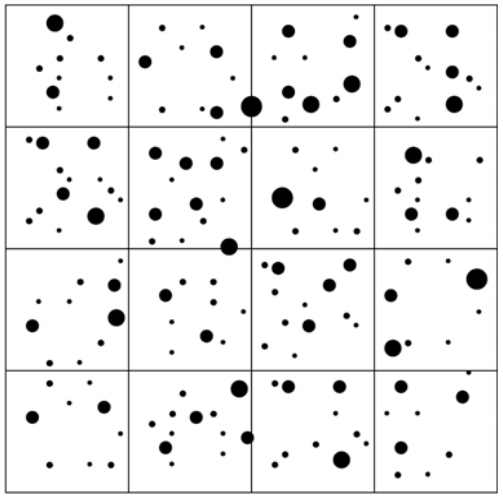


Figure A-9. Minimum $^{239+240}PuO_2$ Particle Diameter, Assuming Spherical, and ρ = 11.5 g/cm^3 (High Range).

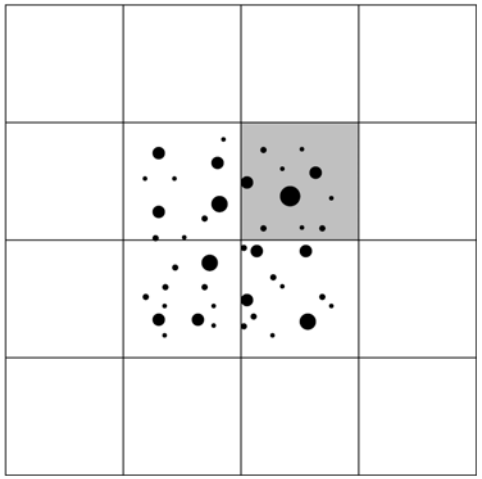


a. Contaminated Area (Initial Deposition).

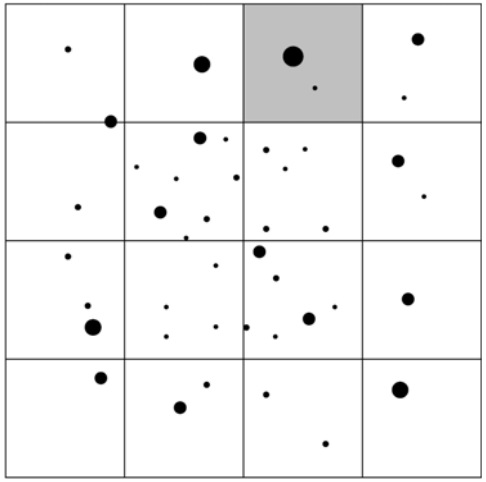


b. Contaminated Area (Significant Time After Deposition).

Figure A-10. Large Contaminated Area Conceptual Diagram.



a. Contaminated Area (Initial Deposition).



b. Contaminated Area (Significant Time After Deposition).

Figure A-11. Small Contaminated Area Conceptual Diagram.

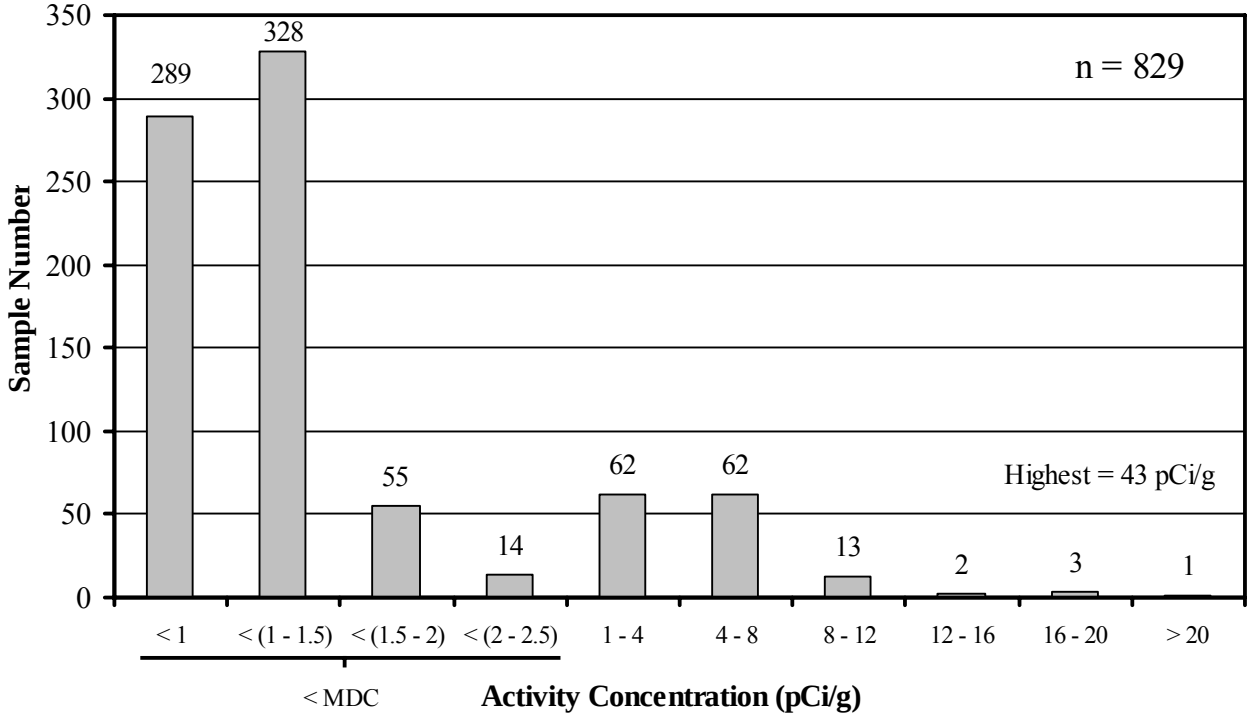


Figure A-12. Systematic Final Status Survey $^{239+240}$Pu Activity Concentration Histogram [Data from Duratek (2006)].

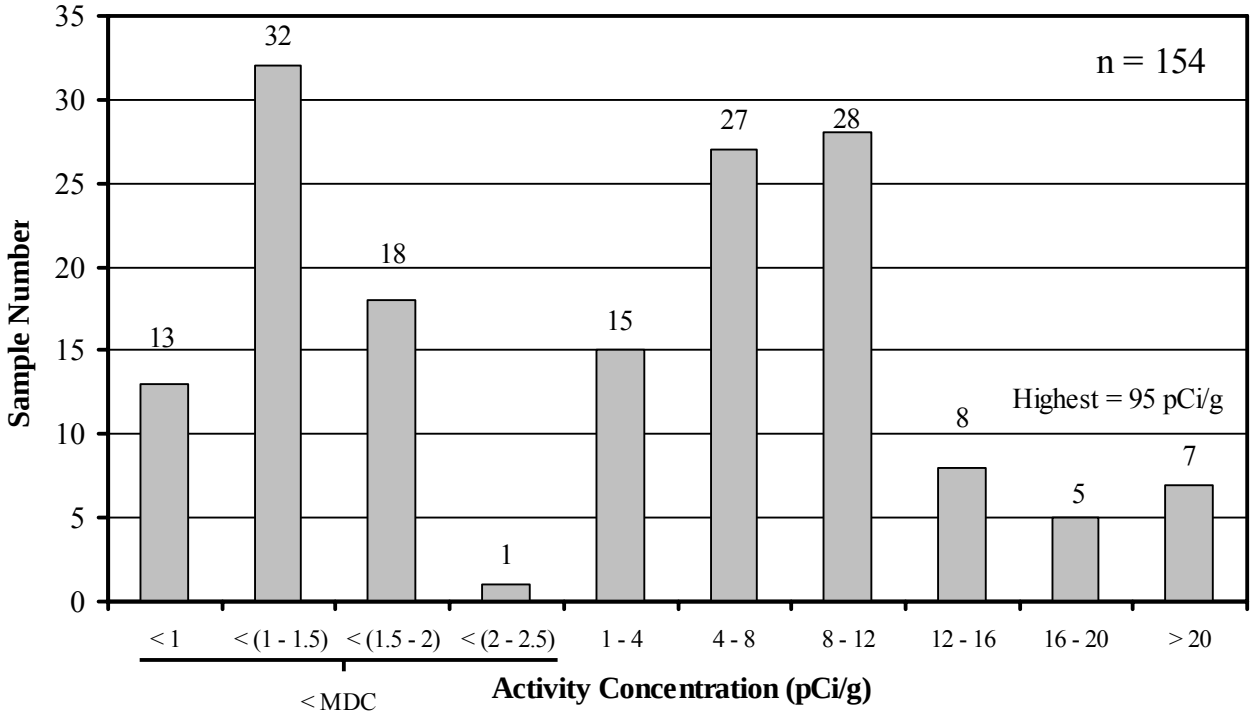


Figure A-13. Elevated Measurement Comparison Final Status Survey $^{239+240}$Pu Activity Concentration Histogram [Data from Duratek (2006)].

TABLE A-5. Comparison of RI/FS and ICRP Dose Conversion Coefficients for $^{239+240}$Pu.		Inhalation Factors		Ingestion Factors		
		DCF (EDE) (mrem/μCi)	f_1	DCF (EDE) (mrem/μCi)	f_1	
	RI/FS (DOE-EH-0071, 1988)	5.1E+05		4.3E+03		Compounds
Workers	ICRP 26/30/48 [FGR 11] (Class D)	NA	NA	3.5E+03	1.0E-03	
	ICRP 26/30/48 [FGR 11] (Class W)	4.3E+05	1.0E-03	3.7E+02	1.0E-04	All but PuO_2
	ICRP 26/30/48 [FGR 11] (Class Y)	3.1E+05	1.0E-05	5.2E+01	1.0E-05	PuO_2
	ICRP 60/30/48 (Class D)	NA	NA	2.1E+03	1.0E-03	
	ICRP 60/30/48 (Class W)	2.5E+05	1.0E-03	2.3E+02	1.0E-04	All but PuO_2
	ICRP 60/30/48 (Class Y)	2.4E+05	1.0E-05	4.5E+01	1.0E-05	PuO_2
	ICRP 68/60	NA	NA	9.3E+02	5.0E-04	Unspecified
	ICRP 68/60 (Type M, Unspecified Compounds)	1.2E+05	5.0E-04	2.0E+02	1.0E-04	Nitrates
	ICRP 60/68 (Type S, Insoluble Oxides)	3.1E+04	1.0E-05	3.3E+01	1.0E-05	Insoluble Oxides
	ICRP 78	NA	NA	9.3E+02	5.0E-04	Unspecified
	ICRP 78 (Type M, Unspecified Compunds)	1.1E+05	5.0E-04	2.0E+02	1.0E-04	Nitrates
	ICRP 78 (Type S, Insoluble Oxides)	4.1E+04	1.0E-05	3.3E+01	1.0E-05	Insoluble Oxides
General Public	ICRP 60/72 (Type F, 3 months old)	7.8E+05	5.0E-04			
	ICRP 60/72 (Type F, 1 year old)	7.4E+05	5.0E-04			
	ICRP 60/72 (Type F, 5 year old)	5.6E+05	5.0E-04			
	ICRP 60/72 (Type F, 10 year old)	4.4E+05	5.0E-04			
	ICRP 60/72 (Type F, 15 year old)	4.1E+05	5.0E-04			
	ICRP 60/72 (Type F, 25 year old)	4.4E+05	5.0E-04			
	ICRP 60/72 (Type M, 3 months old)	3.0E+05	5.0E-04	1.6E+04	5.0E-04	30-yr Weighted Mean = 1.5E+03
	ICRP 60/72 (Type M, 1 year old)	2.9E+05	5.0E-04	1.6E+03	5.0E-04	
	ICRP 60/72 (Type M, 5 year old)	2.2E+05	5.0E-04	1.2E+03	5.0E-04	
	ICRP 60/72 (Type M, 10 year old)	1.8E+05	5.0E-04	1.0E+03	5.0E-04	
	ICRP 60/72 (Type M, 15 year old)	1.7E+05	5.0E-04	9.1E+02	5.0E-04	
	ICRP 60/72 (Type M, 25 year old)	1.9E+05	5.0E-04	9.3E+02	5.0E-04	
	ICRP 60/72 (Type S, 3 months old)	1.6E+05	1.0E-05	30-yr Weighted Mean = 4.8E+04		
	ICRP 60/72 (Type S, 1 year old)	1.4E+05	1.0E-05			
	ICRP 60/72 (Type S, 5 year old)	1.0E+04	1.0E-05			Used for ROD
	ICRP 60/72 (Type S, 10 year old)	7.0E+04	1.0E-05			
	ICRP 60/72 (Type S, 15 year old)	6.3E+03	1.0E-05			Most applicable of respective models
	ICRP 60/72 (Type S, 25 year old)	5.9E+04	1.0E-05			

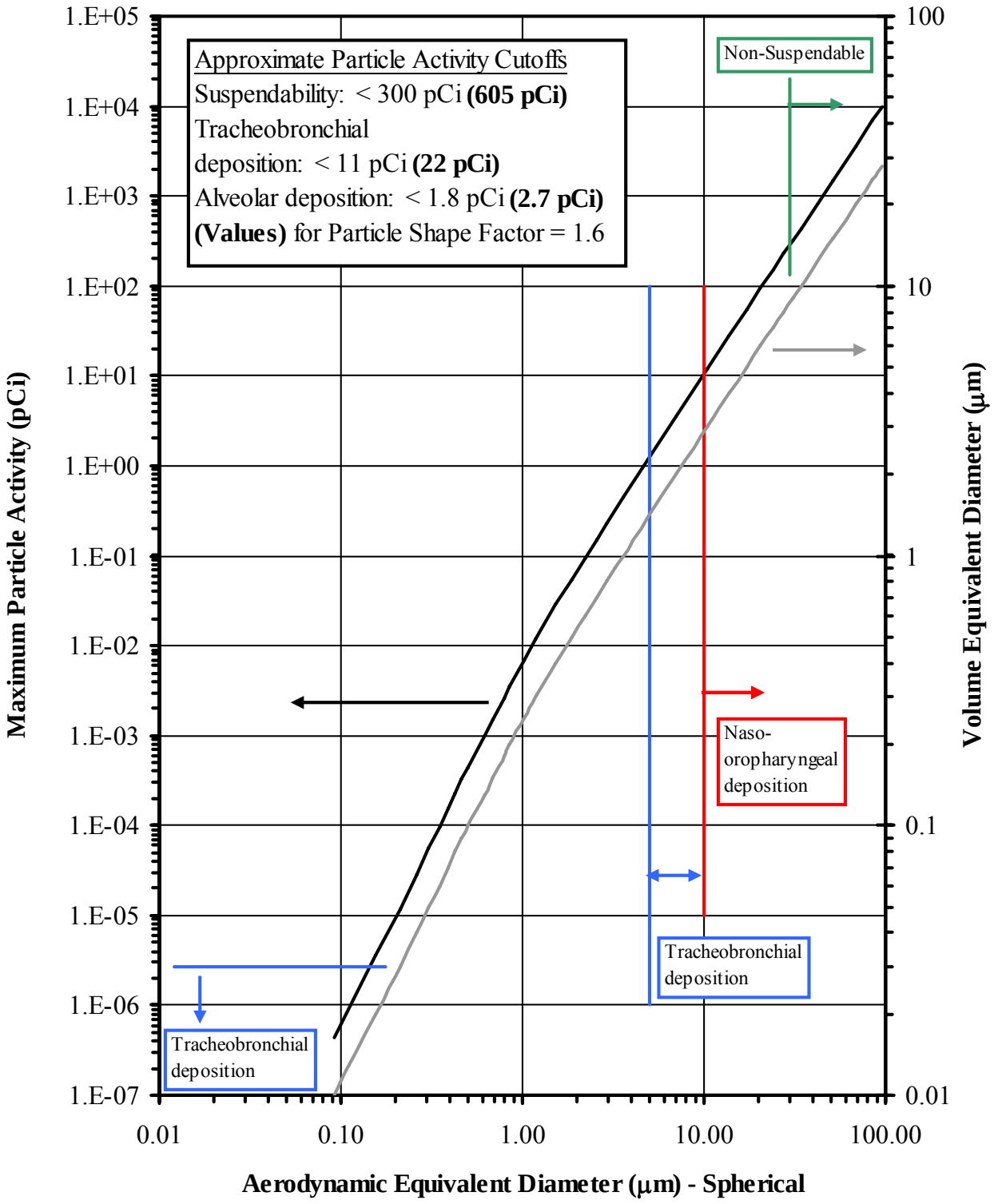


Figure A-14. Maximum Particle Activities and Volume Equivalent Diameters for Aerodynamic Equivalent Diameter (Spherical) PuO_2 Particles.

TABLE A-6. Cunningham Slip Correction Factors for Air [298 ^{o}K, 1.0 atm] (EPA 2007).

D_{pa} (μm)	C_c	d_{pa} (μm)	C_c	d_{pa} (μm)	C_c
0.001	221.6	0.05	5.060	0.8	1.210
0.002	111.1	0.06	4.337	0.9	1.186
0.003	74.25	0.07	3.823	1.0	1.168
0.004	55.83	0.08	3.441	2.0	1.084
0.005	44.78	0.09	3.145	3.0	1.056
0.006	37.41	0.1	2.911	4.0	1.042
0.007	32.15	0.2	1.890	5.0	1.034
0.008	28.20	0.3	1.574	6.0	1.028
0.009	25.14	0.4	1.424	7.0	1.024
0.01	22.68	0.5	1.337	8.0	1.021
0.02	11.65	0.6	1.280	9.0	1.019
0.03	7.978	0.7	1.240	10.0	1.017
0.04	6.151				

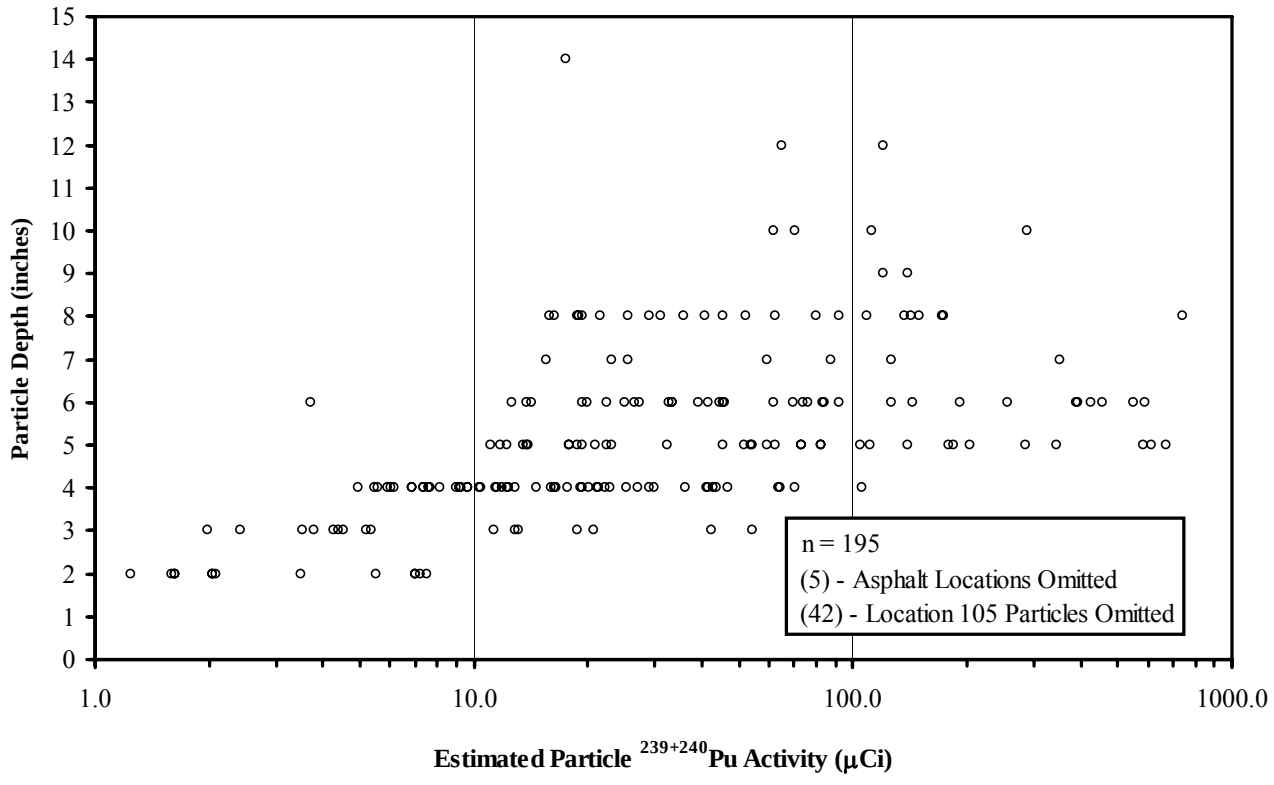


Figure A-15. Estimated Particle $^{239+240}$Pu Activities vs. Depth [Cabrera Services Particle Removal (Cabrera 2006a)].

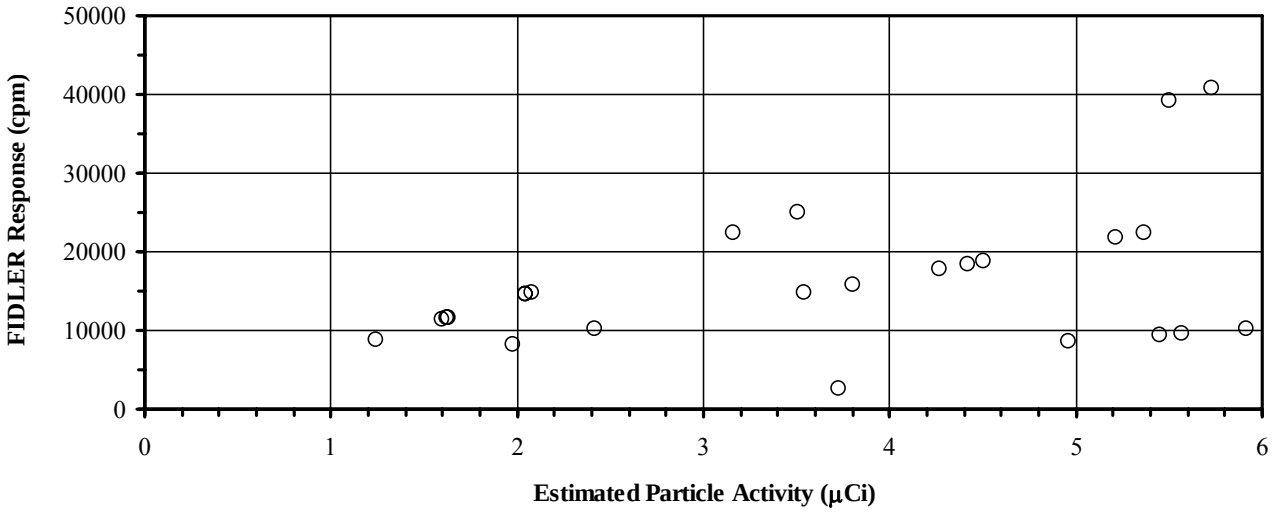
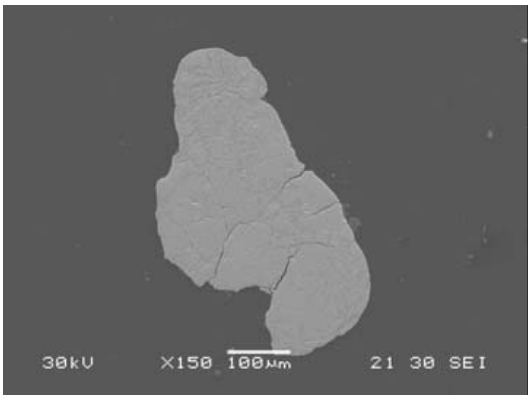
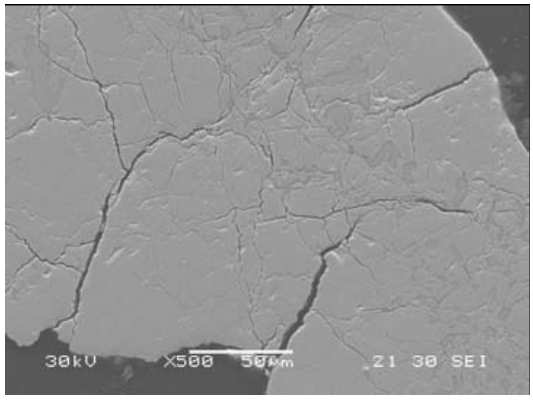


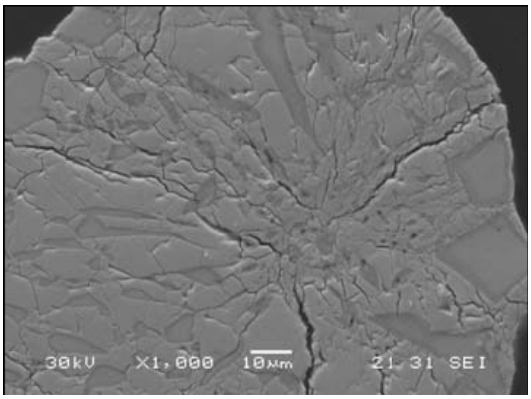
Figure A-16. FIDLER Response vs. Estimated $^{239+240}$Pu Activity [Appendix C, (Cabrera 2006a)].



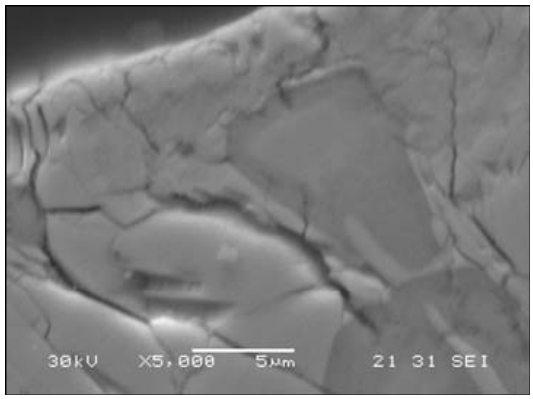
a. SEI x150



b. SEI x500



c. SEI x1000



d. SEI x5000

Figure A-17. Scanning Electron Microscopy (SEM) Image, 0.8 µCi $^{239+240}$Pu Particle Evaluated by Radiochemistry Research Group, Harry Reid Center and Department of Chemistry, University of Nevada, Las Vegas, NV [Figure 13, Appendix E, (Cabrera 2006d)].

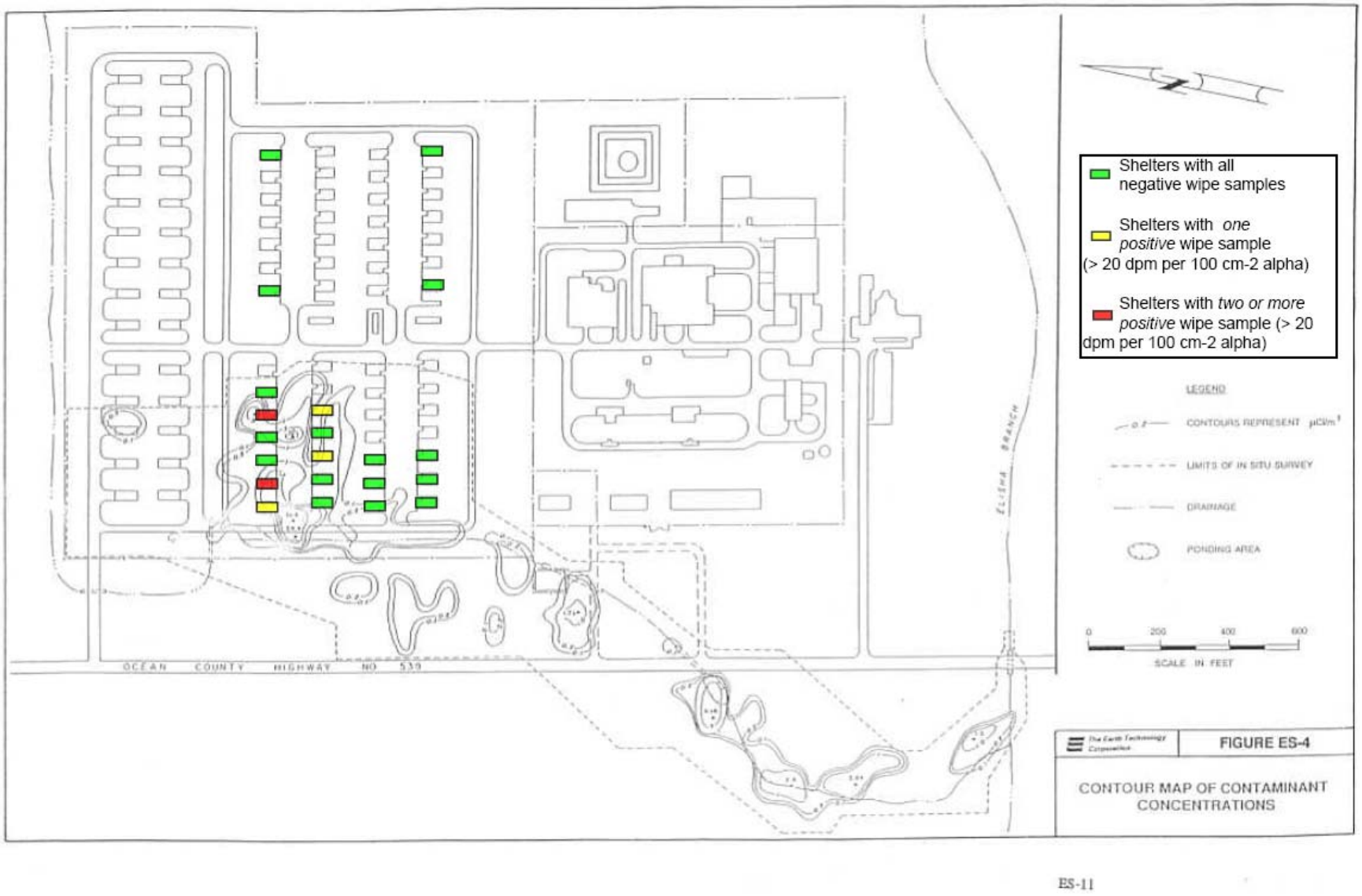


Figure A-18. Modified Figure ES-4 from RI/FS (Earth Tech 1992).

TABLE A-7. Eberline PAC-4G Elevated Results for Shelter Interior Surfaces (Fixed+Removable), [From Table 4-13, RI/FS (Earth Tech 1992)].

Shelter Number	Highest Net $^{239+240}$Pu Results (dpm/100 cm^2)									
	1	2	3	4	5	6	7	8	9	10
101	68	44								
102	68	44								
103	68	44	20							
104	92	68	44	20	20					
105	68	20								
106	140	68	68	44	44	20				
127	283	188	164	116	116	116	92	92	68	44
201	116	116	92	68	44	44	44	20		
202	267	212	164	164	140	92	92	92	92	68
203	68	20	20							
204	47,780	2,106	2,011	718	412	407	407	383	311	215
	119	110	96	72	72	48	48	24		
205	116	92	68	44	20	20				
206	212	164	164	92	68	68	44	44	20	
216	331	236	188	92	92	92	92	92	68	68
Lower Limit of Detection (LLD) = 192 dpm/100 cm^2 (All but Shelter 204)										
Lower Limit of Detection (LLD) = 85 dpm/100 cm^2 (Shelter 204)										
	Values above 100 dpm/100 cm^2 and LLD									

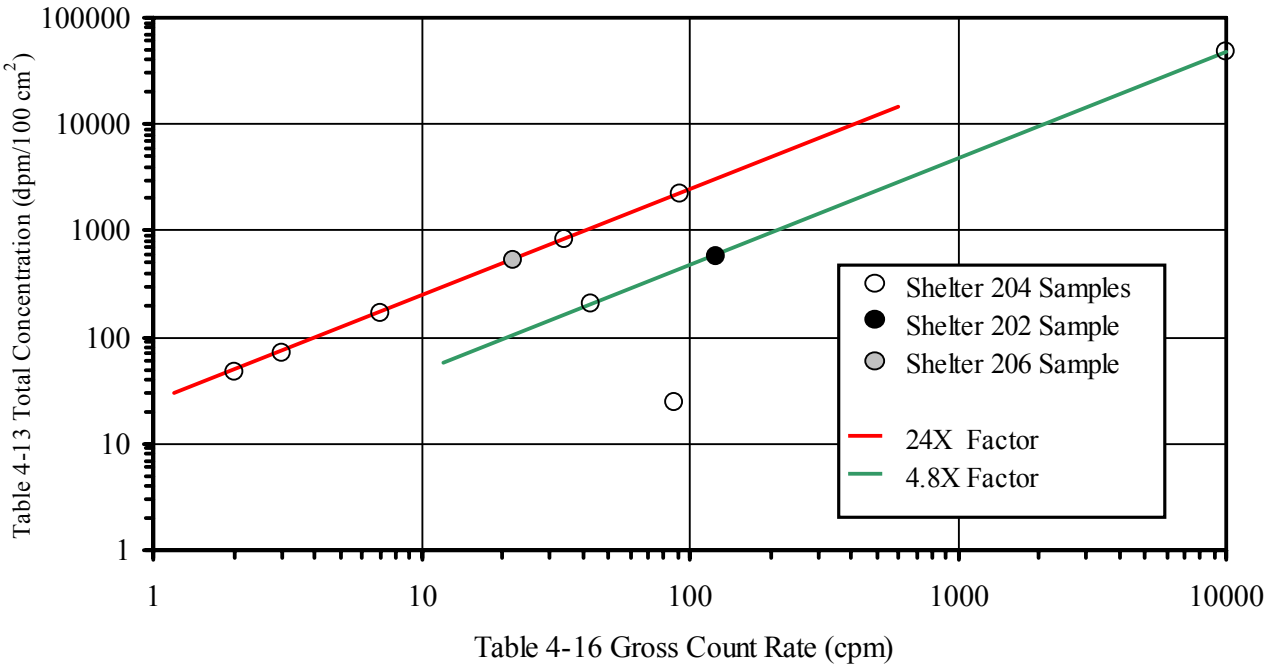


Figure A-19. Scatterplot of Calculated Gross Contamination Concentration (Table 4-13) vs. Reported Instrument Count Rate (Table 4-16) from RI/FS (Earth Tech 1992).

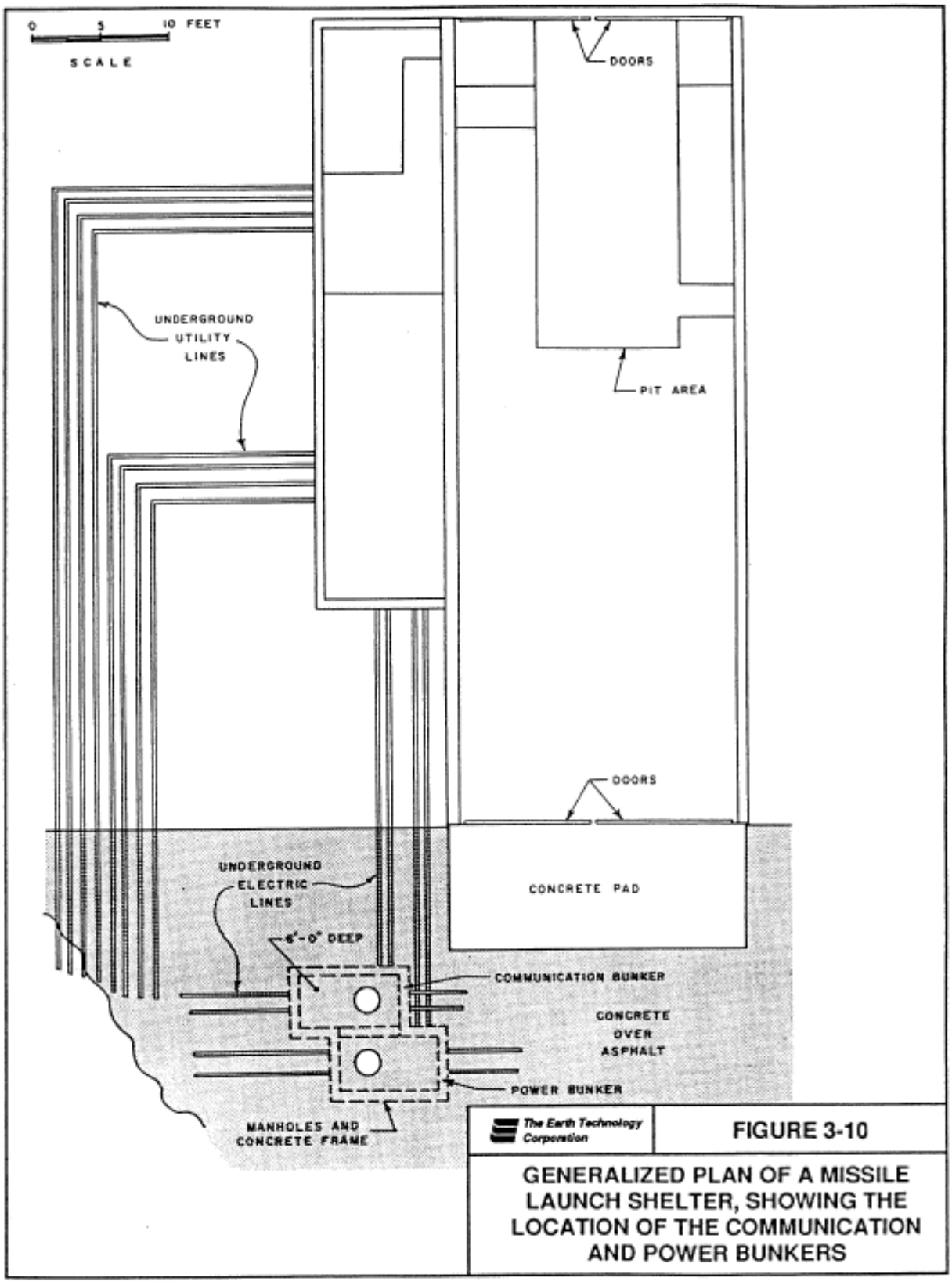


Figure A-20. Figure 3-10 from RI/FS (Earth Tech 1992).

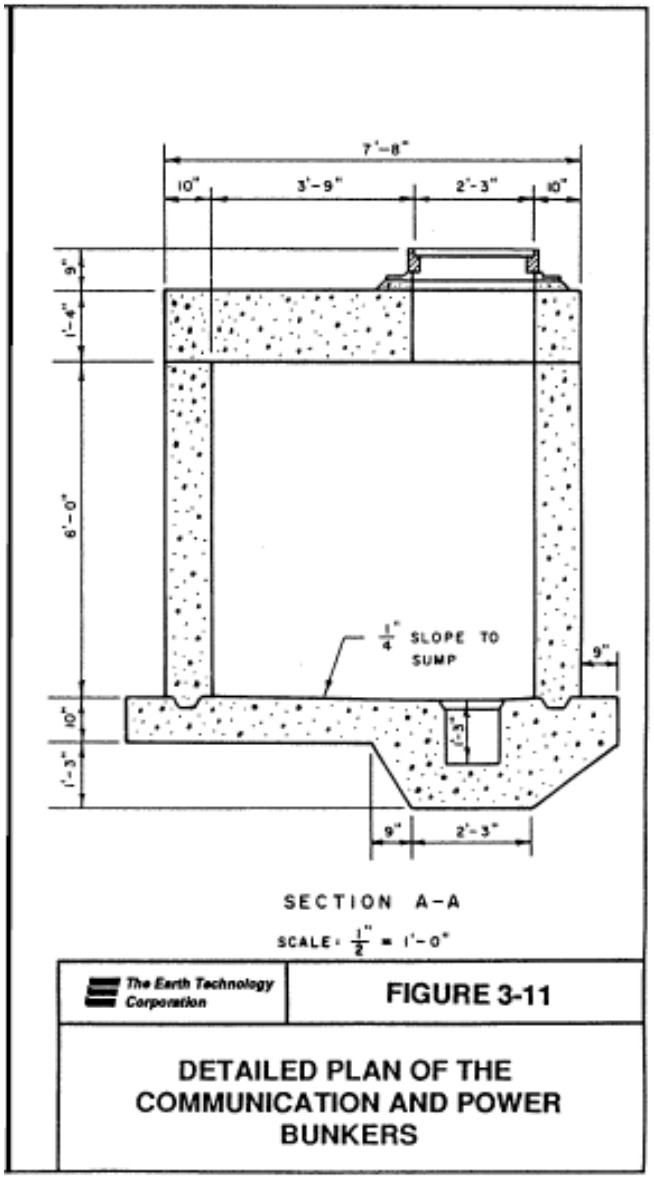


Figure A-21. Truncated Portion of Figure 3-11 from RS/FS (Earth Tech 1992).

TABLE A-8. Wipe and In-Situ α-Radiation Measurements of Communication Bunker [Data from Table 4-18, RI/FS, (Earth Tech 1992)].

Sample Number	PAC-4G Count Rate (cpm)	Ludlum Model 2000 Count Rate on Wipe Sample (cpm)	Removable Fraction*
204-WP-C58-001	7,000	167	0.024
204-WP-C59-001	4,000	165	0.041
204-WP-C60-001	80,000	2,329	0.029
204-WP-C61-001	80,000	1,958	0.025
204-WP-C62-001	1,500	184	0.012
Weighted Mean (PAC-4G Count Rate)			0.028

*Assumes assessed areas and calibration factors are roughly equivalent.

TABLE A-9. Summary of Expected Radiological Impact and Potential for BOMARC Structures Based on Historical Data.

Facility Number	Structure Type	Potential Transport		Expected Degree of Impact (If Impacted)	Document Noting Site
		Mechanism	Liklihood		
201	Missile Shelter	Fire-fighting water	Possible	Low	RI/FS (1992)
203	Missile Shelter	Fire-fighting water	Possible	Low	RI/FS (1992)
205	Missile Shelter	Fire-fighting water	Possible	Low	RI/FS (1992)
207	Missile Shelter	Site activities post accident period	Unlikely	Very Low	RI/FS (1992)
208	Missile Shelter	Adjacent to site remediation	Unlikely	Very Low	RI/FS (1992)
209	Missile Shelter	Site activities post accident period	Unlikely	Very Low	RI/FS (1992)
210	Missile Shelter	Waste or investigation equipment storage	Possible	Low	RI/FS (1992)
212	Missile Shelter	Waste or investigation equipment storage	Possible	Low	RI/FS (1992)
214	Missile Shelter	Waste or investigation equipment storage	Possible	Low	RI/FS (1992)
216	Missile Shelter	Site activities post accident period	Possible	Low	RI/FS (1992)
213	Missile Shelter	Waste or investigation equipment storage	Possible	Low	RI/FS (1992)
118	Missile Shelter	Waste or investigation equipment storage	Possible	Very Low	Cabrera (2005)
158	Restroom	Potential use during accident response	Possible	Low	Cabrera (2005)
27	Security Building	Use during accident response	Possible	Low	Cabrera (2005)
28	Firestation	Use during accident response	Possible	Low	NST (2006)
T40	Used 2002-2004	Use during 2002-2004 Remediation	Possible	Very Low	Duratek (2005)
201 – 208	Power/Comm. Bunkers	Fire-fighting water	Possible	Low	NST (2006)
209 – 214	Power/Comm. Bunkers	Fire-fighting water	Very Unlikely	Very Low	NST (2006)
158	Septic Tank	Use during accident response	Unlikely	Low	Cabrera (2005)
28	Septic Tank/Drain	Use during accident response	Unlikely	Low	NST (2006)
Others	Power/Comm. Bunkers	Site activities post accident period	Highly Unlikely	Extremely Low	Cabrera (2005)

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Appendix B

Risk-Modeling Data and Survey Support

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TABLE B-1. NRC Reg. Guide 1.86, Excerpted from (Rademacher 2005). [Bold for WGP + ^{241}Am].

Nuclide[a]	disintegrations/minute/100 square-centimeters (dpm/100 cm^2)		
	Average[b c f]	Maximum[b d f]	Removable[b e]
U-nat, ^{235}U, ^{238}U & associated decay products	5,000 (α)	15,000 (α)	1,000 (α)
Transuranics, ^{226}Ra, ^{228}Ra, ^{230}Th, ^{228}Th, ^{231}Pa, ^{227}Ac, ^{125}I, ^{129}I	100	300	20
Th-nat, ^{232}Th, ^{90}Sr, ^{223}Ra, ^{224}Ra, ^{232}U, ^{126}I, ^{131}I, ^{133}I	1,000	3,000	200
β–γ emitters (nuclides with decay modes other than α-emission or SF) except ^{90}Sr and others noted above	5,000 (β–γ)	15,000 (β–γ)	1,000 (β–γ)

Notes:

[a]Where surface contamination by both α- and β–γ-emitting nuclides exists, the limits established for α- and β–γ-emitting nuclides should apply independently. [The values apply to radioactive contamination deposited on, but not incorporated into the interior of, the contaminated item.]

[b]As used in this table, dpm means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[c]Measurements of average contamination should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

[d]The maximum contamination level applies to an area of not more than 100 cm^2.

[e]The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent material, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. [The use of dry material may not be appropriate for tritium.] When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire area should be wiped. [Except for transuranics and ^{228}Ra, ^{227}Ac, ^{228}Th, ^{230}Th, ^{231}Pa, and α-emitters, it is not necessary to use wiping techniques to measure removable contamination levels if direct scan surveys indicate that the total residual surface contamination (i.e. removable and fixed) are within the limits for removable contamination.]

[[f]The average and maximum radiation levels associated with surface contamination resulting from β–γ-emitting nuclides should not exceed 0.2 mrad/hr @ 1 cm and 1.0 mrad/hr @ 1 cm, respectively, measured through 7 milligrams per square centimeter (mg/cm^2) of total absorber.]

TABLE B-2. Summary of RESRAD-Build Dose Calculations for Contaminated Floor Surface of BOMARC Shelter.

Annual Dose Equivalent (mrem)	α-Radiation (dpm/100 cm^2)		
	$^{239+240}$Pu	^{241}Am	Total
100 (t = 0)	10,700	2,000	12,700
4 (t = 0)	430	80	510
4 (30-year mean)	850	160	1,010
Route	Inhalation	Ingestion	External
Dose Percents	97.7	2.1	0.07

TABLE B-3. RESRAD-Build Parameters Used for Evaluations.

Parameter	Value	Parameter	Value
Total Time	10,950 d (30 y)	Inhalation Rate	18 m^3/d
Time Inside	192 h/y	Ingestion Dust	1 x 10^{-4} m^2/h
Fraction Inside	0.022	Deposition Velocity	1 x 10^{-2} m/s
Time to Remove Contaminant	10,950 d (30 y)	Resuspension Rate	5 x 10^{-7} 1/s
Building Area	130 m^2	Fraction Released to Air	0.1
Building Height	4.1 m	Removable Fraction	0.1
Building Floor Dimensions	7 m x 18.5 m	$^{239+240}$Pu to ^{241}Am Ratio	5.4
Air Exchange Rate	53.3 m^3/h	Inhalation DCF (mrem/μCi)	4.3 x 10^5 (W)
Air Exchanges per Hour	0.1	Ingestion DCF (mrem/μCi)	3.5 x 10^3 (D)

TABLE B-4. Summary of RESRAD-Build Area Factors for Values of Table B-2 and Main Shelter Area.

Categories	Area Factors for Various EMC-Sizes					
Geometry (m)	3.0 x 3.0	2.5 x 2.5	2.0 x 2.0	1.5 x 1.5	1.0 x 1.0	0.5 x 0.5
Area (m^2)	9.0	6.25	4.9	2.25	1.0	0.25
Area Factor	14	20	32	56	127	506
Acceptable [α] Concentration (cpm) for 100 mrem/yr	177k	254k	401k	715k	1,610k	6,430k

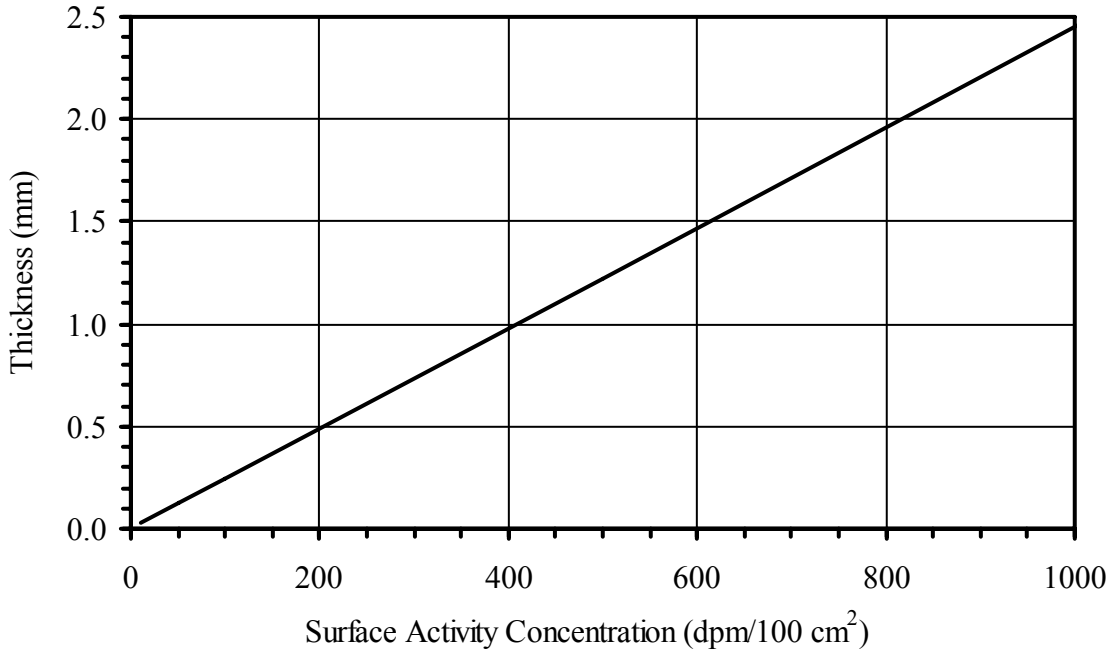

Figure B-1. Averaging Thickness of Concrete vs. Mean Surface Concentrations of $^{239+240}$Pu at 8 pCi/g.

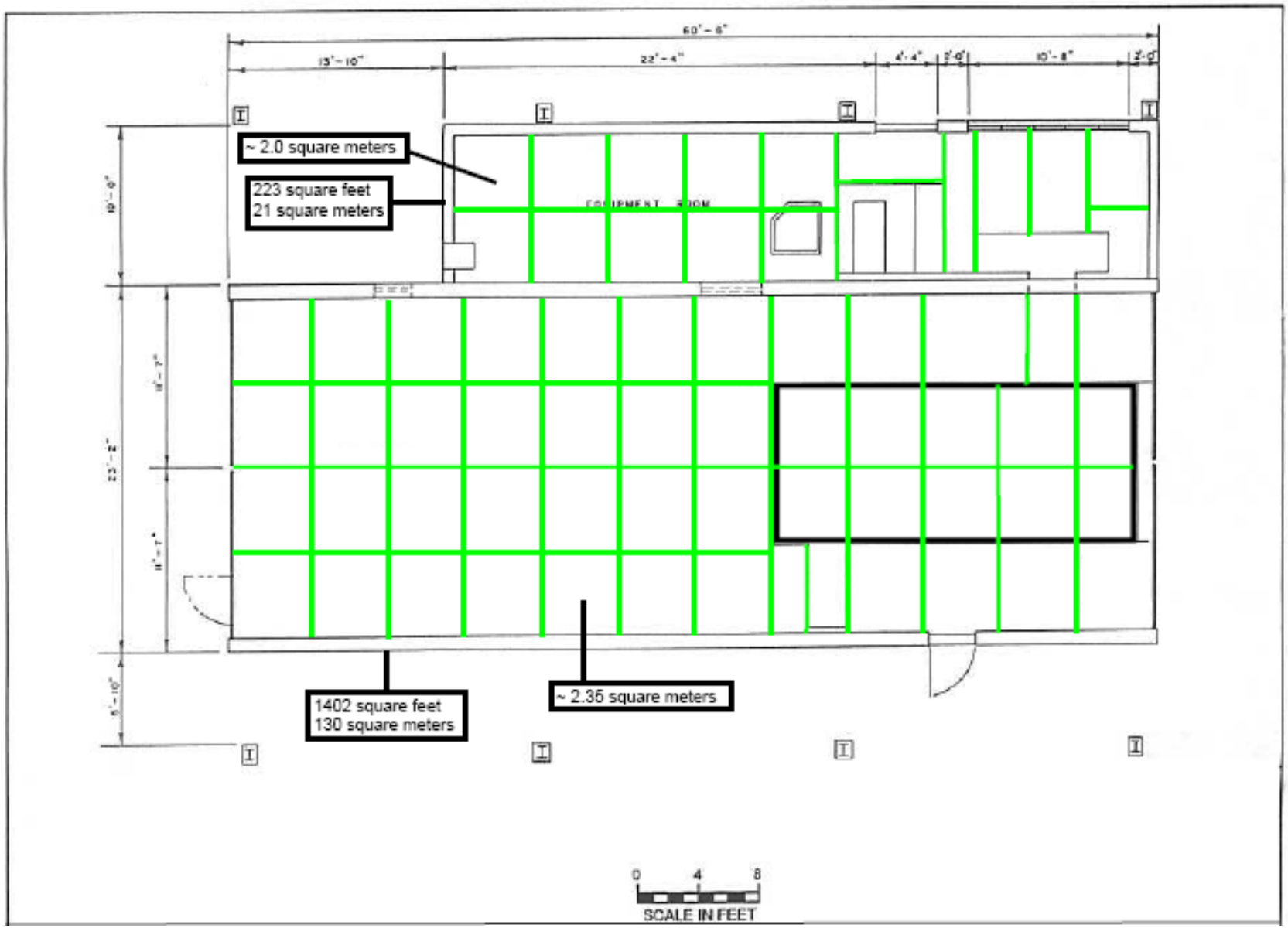

Figure B-2. Example Grid System for Shelter Floor and Pit Surveys.

AFIOH/DOPB (STINFO)
2513 KENNEDY CIRCLE
BROOKS CITY-BASE TX 78235-5116

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