REVISED FINAL
DECOMMISSIONING PLAN

United States Department of Agriculture

Low Level Radioactive Burial Site
Beltsville Agricultural Research Center (BARC)
Beltsville, Maryland

NRC License Number 19-00915-03

January 2009

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ACRONYMS AND ABBREVIATIONS

AEC	Atomic Energy Commission
ALARA	As Low As Reasonably Achievable
ARAR	Applicable or Relevant and Appropriate Requirement
ARS	Agricultural Research Service
ASTM	American Society for Testing and Materials
BARC	Beltsville Agricultural Research Center
BWS	Beta Walkover Survey
^{14}**C**	Carbon-14
^{137}**Ce**	Cesium-137
^{36}**Cl**	Chlorine-36
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CFR	Code of Federal Regulations
Ci	Curie
CHP	Certified Health Physicist
COPC	Contaminants of potential concern
CWB	Certified Waste Broker
CZ	Construction Zone
CY	cubic yard
dBA	Decibels as measured on the A–weighted scale
DCGL	Derived Concentration Guideline Level
DOD	Department of Defense
DOE	Department of Energy
DOT	Department of Transportation
dpm	disintegrations per minute
EE/CA	Engineering Evaluation/Cost Analysis
EM	Electromagnetic
EPA	U.S. Environmental Protection Agency
FID	Flame Ionization Detector
FOM	Field Operations Manager
FQCR	Field Quality Control Representative
FSS	Final Status Survey
Ft	feet
GM	Geiger Muller
GPR	Ground Penetrating Radar
GPS	Global Positioning System
GWS	Gamma Walkover Survey
^{3}**H**	Tritium
HEPA	High Efficiency Particulate Air
HPT	Health Physics Technician
IDW	Investigation Derived Waste
^{125}**I**	Iodine-125
JMC	(US Army) Joint Munitions Command
K_d	Distribution Coefficient
LTR	License Termination Rule
LLRBS	Low Level Radioactive Burial Site
LLRW	Low level radioactive waste
MCL	Maximum contamination level
MDA	Minimum Detectable Activity
μCi/ml	micro-Curies per milliliter
mg/m^{3}	**milligrams per cubic meter**
MS/MSD	Matrix spike/matrix spike duplicate
NaI	Sodium Iodide
NAAQS	National Ambient Air Quality Standards
NAD	Normalized Absolute Difference
NCP	National Oil and Hazardous Substances Pollution Contingency Plan
NESHAP	National Emission Standards for Hazardous Air Pollutants
^{63}**Ni**	Nickel-63
NIST	National Institute of Standards and Testing
NRC	Nuclear Regulatory Commission
NSPS	New Source Performance Standards
OSHA	Occupational Safety and Health Administration
PA	Preliminary Assessment
PAH	Polynuclear aromatic hydrocarbon
^{210}**Pb**	Lead-210
PCB	Polychlorinated Biphenyls
^{32}**P**	Phosphorous-32
pCi/g	Pico-curies per gram
PID	Photo Ionization Detector
PM	Project Manager

PPE	Personal Protective Equipment
QA	Quality Assurance
QAPP	Quality Assurance Project Plan
QC	Quality Control
^{226}Ra	Radium-226
RBC	Risk Based Concentration
RCP	Radiological Controls Program
RCRA	Resource Conservation Recovery Act
RCZ	Radiologically controlled zone
ROC	Radionuclides of Concern
RI	Remedial Investigation
RSO	Radiation Safety Officer
RSP	Radiation Safety Procedures
RWP	Radiation Work Permit
SI	Site Inspection
SOP	Standard Operating Procedure
^{90}Sr	Strontium-90
SSHO	Site Safety and Health Officer
SSHP	Site Safety and Health Plan
SVOC	Semi Volatile Organic Compound
TBD	To Be Determined
TCLP	Toxicity Characteristic Leachability Procedure
TL	Technical Lead
TOX	Total Organic Halides
USDA	United States Department of Agriculture
VOC	Volatile Organic Compounds
WCS	Waste Characterization Survey

EXECUTIVE SUMMARY

The United States Department of Agriculture (USDA) has developed this Decommissioning Plan (DP) in support of site decommissioning activities and license termination. This plan specifically addresses the Low Level Radioactive Burial Site (LLRBS) at USDA's Agricultural Research Service (ARS), Henry A. Wallace Beltsville Agricultural Research Center (BARC), (Maryland). Figure 1 identifies the location of BARC. Hereafter, the BARC Low Level Radioactive Burial Site will be referred to as the LLRBS or the Site. The Site layout is identified on Figure 2. The Site is permitted under USDA's Nuclear Regulatory Commission (NRC) license No. 19-00915-03. The license is held by USDA's Radiation Safety Staff.

The LLRBS was established on June 23, 1949, and was used for the disposal of low-level radioactive waste (LLRW) until 1987. The LLRBS is permitted under a USDA-wide license originally issued by the Atomic Energy Commission (AEC), and later by the NRC. Records indicate the last liquid burial at the LLRBS was on September 17, 1984. From September 24, 1985 until disposal activities ended in 1987, all burials were dry solids packed in 55-gallon drums (ENTECH, 2000). It is currently fenced to restrict access, and encompasses approximately 60,000 square feet (sq ft; 1.38 acres). Table 1 provides a summary of the information that was compiled describing each of the burials. Table 2 provides a summary of the total inventory of specific radionuclides that were disposed in the LLRBS on an annual basis.

The BARC is owned and operated by the USDA. The LLRBS is located approximately one-quarter mile north of the Cherry Hill Road overpass of the Capital Beltway (I95/495) in Beltsville, Maryland, northeast of Washington D.C. Primary access is through BARC via US Highway 1/Baltimore Avenue. Secondary access to the LLRBS is a gravel road that leads from Cherry Hill Road to a cluster of BARC maintenance buildings and continues along the western side of the BARC. This access is fenced and locked. The address of the licensee is:

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Regionally, topography slopes to the east and southeast at 10 to 15 percent toward the nearest perennial stream, the Little Paint Branch, located approximately 2,000 ft east of the LLRBS boundary. Downstream, Little Paint Branch feeds into Paint Branch, 1.4 miles to the south, eventually draining into the Anacostia River. The LLRBS is located on a hillside with an access road located to the west and the surface slope to the east. Soils in the vicinity of LLRBS have been described as sand and gravel terrace deposits overlying bedrock, which occurs at depths exceeding 30 feet in the vicinity of the site (Apex, 1991). Depth to groundwater in the vicinity of LLRBS is approximately 25 ft below ground surface, and groundwater flow direction has been mapped as southeasterly.

The geology at BARC consists of Lower Cretaceous sediments of the Potomac Group, which consists of the Patuxent, the Arundel, and the Patapsco Formations, in decreasing age. The

Patuxent and Patapsco Formations are composed primarily of sand and gravel, and comprise the most prevalent water bearing aquifers in Prince George's County. The Arundel is mostly clay, and creates artesian conditions in the underlying Patuxent Formation in some locations. Recharge of the Patuxent Formation occurs where it outcrops in the western portions of BARC. This wedge of sediments made up of the Patuxent, Arundel, and Patapsco Formations dips to the southeast, parallel to the regional groundwater flow (Apex, 1991).

USDA has no current or future plans to release the BARC facility from ARS's current mission. The facility is best characterized as minimally developed, and is surrounded by land that is largely urbanized and densely populated. Inside the facility's boundaries, land use is agriculture, forest, and urban, with more than 800 buildings including laboratories, greenhouses, barns, office buildings, and some residences. A major portion of the facility is currently being used for crops, grazing livestock, and orchard research projects, primarily in the central and western portions of BARC. The central and eastern portions of the facility are primarily covered with mixed deciduous/evergreen forest. The urbanized portions of BARC are scattered throughout the property.

Land use outside of the facility boundaries is largely mixed urban and lightly developed, primarily forested parcels. There is widespread residential development along the western, southwestern, and northwestern boundaries of BARC. Commercial development is prevalent along U.S. Route 1 and the Beltsville Industrial Center, north of Sunnyside Avenue. Other major transportation routes that either border or pass through BARC are Interstate 95, Interstate 95/495, the Baltimore-Washington Parkway, and the B&O Railroad.

Historical data indicate the presence of subsurface radioactive and chemical contaminants in the LLRBS, an inactive disposal site for NRC-licensed low-level radioactive waste. ARS records indicate a total of 50 pits were designated as waste burial pits (Apex, 1993). Information derived from the Engineering Evaluation/Cost Analysis (EE/CA) conducted in 2000 and the waste characterization survey (WCS) performed in 2006 has led to the conclusion that only 46 out 50 potential burial pits were used for disposal of 33,000 cubic feet of waste (ENTECH 2000, Cabrera 2007). The Waste Characterization Survey activities conducted in 2007 resulted in remediation of four of 46 burial pits. In addition to soil contamination, the EE/CA report identified both tritium (^{3}H) and chloroform contamination in groundwater samples collected from wells downgradient from the site (ENTECH, 2000).

Burial of certain quantities of radioactive waste in soil by licensees without prior AEC (now NRC) approval was authorized on January 29, 1959 (22 FR 548). Originally, this authorization was codified in former 10 CFR 20.304 (NRC, 2003). The early LLRBS waste burials were performed in accordance with the former 10 CFR 20.304 regulation. On January 28, 1981, the Commission concluded that it was inappropriate to continue generic authorizations of burials pursuant to 10 CFR 20.304 without regard to factors such as location of burial, concentrations of radioactive material, form of packaging, and notification of NRC. Therefore, NRC rescinded 10 CFR 20.304 (45 FR 71761). As of January 28, 1981, licensees wishing to perform onsite disposals of the type previously authorized under 10 CFR 20.304 were required to obtain prior NRC approval in accordance with 10 CFR 20.302 (NRC, 2003). Subsequent LLRBS burials were performed with NRC concurrence through the approval process described above.

NRC's "Timeliness in Decommissioning of Material Facilities" established criteria for timely decommissioning upon termination of operations by amending 10 CFR Parts 2, 30, 40, 70, and

72. The "Timeliness Rule" establishes requirements for notifying NRC of pending decommissioning actions and cessations in licensee operations, establishes requirements for when decommissioning plans (DPs) need to be submitted, and establishes requirements for completing decommissioning activities (NRC, 2006). Licensed facilities convert from "active" status to "decommissioning" status when one of the following occurs:

- The license expires or is revoked by the NRC,
- The licensee decides to permanently cease operations with licensed material at the entire site or in any separate building or outdoor area that contains residual radioactivity, such that the area is unsuitable for release in accordance with NRC requirements,
- Twenty-four (24) months have elapsed since principal activities have been conducted under the license, or
- No principal activities have been conducted in a separate building or outdoor area for a period of 24 months, and residual radioactivity is present that would preclude its release in accordance with NRC requirements (NRC, 2006).

As discussed previously, BARC ceased use of the LLRBS in 1987 and as such has terminated licensed activities more than 24 months ago. There is not sufficient verifiable data to demonstrate that, in its current condition, the LLRBS is suitable for release for unrestricted use (i.e., meets the 10 CFR 20 Subpart E unrestricted Radiological Criteria for License Termination). Thus, based on the criteria above, USDA is required to initiate the NRC decommissioning process in accordance with 10 CFR 30.36(d).

The LLRBS EE/CA suggests that the preferred alternative is to perform a removal action that eliminates the source of contamination (i.e., material in the pits) (ENTECH, 2000). Based on NRC guidance, if removal actions are not currently authorized under an existing license, the licensee must develop a DP and submit a request for a license amendment. USDA's NRC radioactive materials license (19-00915-03) authorizes the following uses of radioactive material:

- *Research and development as defined in 10 CFR 30.4 including animal studies; in gauging and measuring devices and in field studies; and,*

- *Studies on human research subjects as approved by a Radioactive Drug Research Committee (RDRC) that has been approved by the Food and Drug Administration (FDA) (NRC, 2000).*

Regarding the need for development of a DP, 10 CFR 30.36(g) (1) states:

> *A decommissioning plan must be submitted if required by license condition or if the procedures and activities necessary to carry out decommissioning of the site or separate building or outdoor area have not been previously approved by the Commission and these procedures could increase potential health and safety impacts to workers or to the public, such as in any of the following cases: (i) Procedures would involve techniques not applied routinely during cleanup or maintenance operations; (ii) Workers would be entering areas not normally occupied where surface contamination and radiation levels are significantly higher than routinely encountered during operation; (iii) Procedures could result in significantly greater airborne concentrations of radioactive materials than are present during operation; or (iv) Procedures could result in significantly greater releases of radioactive material to the environment than those associated with operation.*

The USDA NRC license does not specifically authorize excavation, and packaging of buried radioactive wastes. The facility decommissioning has also been designated as a Group 5 decommissioning in accordance with the requirements of NUREG 1757 Vol. 1, Rev 1 due to existing groundwater contamination. Thus, the USDA must submit a DP to the NRC and apply for a license amendment to support planned removal actions.

The decommissioning process is to be completed within two years, unless an alternative schedule is approved (NRC, 2006). While the steps may vary for different sites, the basic decommissioning process is the same. The steps in the process (based on current regulations and guidance) are as follows:

1. Stop operations, either in a specific area or building or for the entire facility.
2. Notify NRC of the decision within 60 days.
3. Determine locations and concentrations of remaining radiological contamination.
4. If necessary, develop a Decommissioning Plan that includes all of the following:
    a) the current radiological contamination at the site;
    b) the criteria for the final condition of the site;
    c) the activities to remediate existing contamination that are not currently authorized by the license;
    d) procedures to protect workers;
    e) decommissioning cost estimates;
    f) the final survey method to demonstrate compliance with NRC criteria; and
    g) the schedule for remediation activities and license termination.
5. If necessary, provide environmental information for NEPA Compliance.
6. Clean up contamination, as needed.
7. Conduct Final Status Survey to show compliance with dose limits for license termination.
8. Request that NRC terminate the license (NRC, 2006).

This DP documents the plan of execution for decommissioning of the LLRBS. The development of the proposed tasks includes excavation and off site disposal of soils and waste materials from the 35 waste cells believed to remain in place. Excavated wastes will be classified and separated, and then disposed in appropriately offsite treatment or disposal facilities.

Activities described in this DP will be performed under a contractor NRC license and NRC-approved Radiation Safety Program (RSP) and under the guidance set forth in a site-specific safety and health plan (SSHP) for characterization activities at the LLRBS. Use of a contractor's NRC license is desirable because USDA's license does not authorize excavation of buried wastes and, as such, USDA programmatic elements do not address performance of such activities. The purpose of the RSP is to define program requirements and radiation protection standards in support of operations performed under the contractor's NRC license.

A site-specific risk assessment has been performed, and documented in this DP and is used to derive cleanup criteria for each of the potential radionuclide contaminants. Some important inputs to the risk assessment include: 1) the radionuclides of concern and their mobility; 2) the volume of contaminated soil and concentrations of contaminants (especially in the vadose zone, because a removal action should virtually eliminate the source term burials); 3) depth to groundwater; and 4) other physical site parameters. Potential risks associated with any chemical contaminants will be evaluated in a separate study, in accordance with applicable state and EPA guidance.

Derived concentration guideline levels (DCGLs), determined during the risk assessment are based on 10 CFR 20 Subpart E criteria for unrestricted release license termination. Depth to groundwater and other physical parameters are obtained from past site investigations, observations during the Characterization Survey, and published regional data.

It is assumed that a removal action will eliminate the source term burials. Three-dimensional models are used, in conjunction with the DCGLs established during the risk assessment, to design a remediation protocol that is protective of human health and the environment during and after implementation. It should be noted that implementation of the remedial design will be phased based on funding constraints.

Based on the current design, procedures have been developed to protect workers and the environment. In addition, gamma walkover surveys performed over the surface of the LLRBS support the initial assumption that all radioactive waste and contamination is subsurface.

In 2006 CABRERA conducted a waste characterization survey in accordance with "Characterization Survey Work Plan, Low Level Radioactive Burial Site Beltsville Agricultural Research Center (BARC)", (CABRERA, 2004) The Characterization Survey was implemented and involved the following tasks:

- Performance of gamma and beta walkover surveys over the surface of the LLRBS;
- Excavation and characterization of soils and waste materials of at least one and no more than four waste pits (the number of pits excavated and characterized will depend on the type and quantity of waste removed form the pits and funding available for disposal);
- Sampling and radiological/chemical analysis of soil at the base of each exhumed pit;
- Subsurface sampling and radiological/chemical analysis of vadose zone soils from the base of each exhumed pit to the underlying aquifer;
- Groundwater sampling in temporary wells set in the base of each exhumed pit;
- Backfill of each exhumed pit with low-permeability soil (to minimize potential for contamination of backfill materials).
- Offsite treatment/disposal of waste materials exhumed from each pit.

Based on data collected during the Characterization Survey, the DP has been updated. Characterization Survey activities were completed in March 2007. The Characterization Survey activities resulted in remediation of four of 46 potential burial pits. Remediation of the remaining 42 pits is planned to be executed under this DP. Current estimates indicate that it will cost approximately $7M to complete the decommissioning; this estimate is largely dependent on the amount of mixed waste in the burials. Based on this cost estimate and expected funding, the decommissioning will be executed / funded incrementally and occur over a five to nine year period. Thus, final decommissioning is expected to be completed as early as 2010.

USDA requests NRC to amend license No. 19-00915-03 to authorize activities presented in this DP. Implementation of this DP will result in removal of the radioactive source term at the LLRBS and final site conditions will meet the 10 CFR 20 Subpart E criteria for release for unrestricted use.

1.0 FACILITY OPERATING HISTORY

1.1 License Status and Authorized Activities

The BARC LLRBS was established in 1949 under a USDA/U.S. Atomic Energy Commission project (BARC, 1993). The LLRBS was used for the disposal of BARC's LLRW until May 28, 1987, the date of the last burial. The Site is permitted under USDA's NRC license No. 19-00915-03. The license is held by USDA's Radiation Safety Staff.

From its inception to September 17, 1984, various BARC research laboratories generated radioactive waste, which included dry solid materials such as gloves, paper, and syringes; other solid materials in the form of sealed sources and electron capture detectors; glass and plastic liquid scintillation vials containing scintillation liquids; aqueous and organic bulk liquids; plastic bags containing decomposed rat and chicken carcasses, bedding, and excreta; and other radioactively contaminated laboratory wastes.

The types of radioactive waste material at the LLRBS consisted of tritium (^{3}H), carbon-14 (^{14}C), cesium-137 (^{137}Cs), chlorine-36 (^{36}Cl), phosphorous-32 (^{32}P), nickel-63 (^{63}Ni), radium-226 (^{226}Ra), lead-210 (^{210}Pb), and other radionuclides. Some of the radiological materials deposited in the LLRBS are short-lived (i.e., have a short radiological half-life), including ^{32}P and ^{210}Po. Inventory records of burials from 1949 through 1960 could not be located. Table 1 provides a summary of the information that was compiled describing each of the burials. Table 2 provides a summary of the total inventory of specific radionuclides that were disposed in the LLRBS on an annual basis.

The LLRBS is permitted under a USDA-wide license originally issued by the Atomic Energy Commission (AEC), and later by the NRC. Records indicate the last liquid burial at the LLRBS was on September 17, 1984. From September 24, 1985 until disposal activities ended in 1987, all burials were dry solids packed in 55-gallon drums (ENTECH, 2000). It is currently fenced to restrict access, and encompasses approximately 60,000 square feet [(sq ft) (1.38 acres)].

The LLRBS is made up of a total of 50 designated waste burial pits. The Engineering Evaluation/Cost Analysis (EE/CA) conducted during 2000 and the waste characterization survey (WCS) performed during 2006 concluded that 46 out of 50 potential burial pits were used for disposal of waste (ENTECH 2000, Cabrera 2007). The pits are approximately 10 ft wide by 12 ft long by 10 ft deep and are separated approximately 6 feet horizontally from one another. Each pit was reportedly backfilled to surface grade with at least 5 ft of clean fill. Two contiguous fenced fields, the North Field and the South Field, make up the LLRBS, each of which is approximately 150 ft by 200 ft. Both the EE/CA and the WCS reported that the South Field was never used for disposal of any waste materials, although individual pits had been designated. The LLRBS is located on a hillside with the access road located to the west and the surface slope to the east. Soils in the vicinity of LLRBS have been described as sand and gravel terrace deposits overlying bedrock, which occurs at depths exceeding 30 feet in the vicinity of the site. Depth to groundwater in the vicinity of LLRBS is approximately 25 ft below ground surface, and groundwater flow direction has been mapped as southeasterly. Based on the estimated numbers and dimensions of each burial cell, the EE/CA estimated that 33,000 cubic ft of waste is buried at the site (ENTECH, 2000).

Buried materials include radioactive isotopes and scintillation fluids (isotopes and organic fluids); contaminated metal, glass, and plastic objects; contaminated animal carcasses; and animal wastes. USDA records do not reveal the types of organic liquids contained in the scintillation vials, nor is any indication of volume included. Typically, organic solvents associated with scintillation fluids include toluene and xylenes. Types of containers disposed, according to files, are cardboard boxes of 1 to 4 cubic ft, 1- to 5-gallon containers for liquids, plastic milk jugs, plastic carboys, solvent bottles, fiberboard drums, and 55-gallon drums. Liquid containers were placed in cardboard boxes, usually 4 to a box.

No known records exist listing the specific types of containers used for the early burials. Records from the 1980s indicate contaminated, non-flammable glass and plastic (vials, pipettes, needles, scalpels, etc.) were buried in cardboard boxes. Animal carcasses, bedding, and excreta were sealed in polyethylene bags and placed in boxes. Liquid wastes were packed in plastic containers and placed in cardboard boxes. Bulk liquid wastes were transferred into plastic or glass carboys and packaged in cardboard boxes with some cushioning and absorbent material. Liquid scintillation vials were either placed on vial trays and packaged in cardboard boxes for disposal, or placed loose in large plastic bags, then in cardboard boxes. Animal remains, generally contaminated with tritium (^{3}H) and carbon-14 (^{14}C), were routinely incinerated beginning in the early 1980s, in one of two incinerators located at the BARC. It is possible that incinerator ash, which tested positive for radioactivity, could have been sent to the LLRBS, however, no known records exist regarding ash disposal.

Another source of data is a draft summary of the LLRBS produced by an unknown author in the early 1990s and transmitted by ENTECH, Inc. to CABRERA in January 2003. The summary contains a schematic of the North Field depicting 50 burial sites. Also contained in the summary is a table that shows the radionuclides that were disposed in each trench, along with the activity in milliCuries, whether the disposed materials were solid, liquid, vials of liquid scintillation materials, and the date the pit was closed.

Depending on the anticipated amount of wastes generated, disposal pits were left uncovered until a sufficient volume of waste was contained. No animal carcasses or excreta were disposed in the pits until immediately before closure. After September 17, 1984, no liquids were disposed of at the LLRBS. After September 1985 only dry solid wastes and animal carcasses, packaged in 55-gallon steel drums, were disposed at the site. Radioactive waste shipment and disposal manifests were prepared and maintained for disposals after September 1985, thus providing supplementary waste information for these disposals. Four of the pits contain only dry solid or animal wastes, disposed after September 1985, packaged in steel drums.

In 2006, a WCS (CABRERA, 2007) was executed in four of the 46 burial pits, to determine nature and extent of waste emplaced in the characterized pits (Pits 1, 14, 26, and 34). Based on the results of this survey, it is anticipated that the nature and extent of the waste in the remaining pits will be highly variable. Waste materials encountered during excavation include laboratory trash (gloves, paper, metals, plastics, laboratory glassware, and other wastes generated during the process of performing laboratory analyses), liquid scintillation vials, radioactive sources, bulk soils containing small amounts of waste or debris not readily separable from soil, animal remains, and bulk liquids in their original containers. Disposed materials often contained both radionuclides and hazardous wastes at concentrations that exceed NRC and/or EPA screening criteria for groundwater and soil.

Radionuclide concentrations found in soil and water were generally within regulatory limits.

1.1.1 License Number/Status

The BARC LLRBS was established in 1949 under a USDA/U.S. Atomic Energy Commission project (BARC, 1993). The LLRBS was used for the disposal of BARC's LLRW until May 28, 1987, the date of the last burial. The site is permitted under USDA's NRC license No. 19-00915-03. The license is held by USDA's Radiation Safety Staff.

1.1.2 Authorized Activities

Subject to conditions 10 through 33 of the USDA Materials License 19-00915-03, authorized uses of licensed materials identified as A through K, the authorized activities are for "Research and development as defined in 10 CFR 30.4 including animal studies; in gauging and measuring and in field studies." Licensed material identified by the letter L of the materials license is approved for "Studies on human research subjects approved by a Radioactive Drug Research Committee (RDRC)that has been approved by the Food and Drug Administration (FDA).

Of particular note, license condition 29 states "*The licensee shall maintain control of each site where it disposed of radioactive material by burial and shall monitor the Beltsville burial site in accordance with letter dated July 2, 1992. No additional burials of radioactive material are authorized by this license."*

License condition 31 requires the licensee to submit to the NRC a draft work plan for remediation of the radioactive waste burial site.

1.2 License History

There are no previous licenses issued to this facility. There have been multiple revisions/amendments to the existing license.

1.3 Previous Decommissioning Activities

There have been no previous decommissioning activities at the Site.

1.4 Spills

As discussed in Section 1.1, bulk liquid wastes were transferred into plastic or glass carboys and packaged in cardboard boxes with some cushioning and absorbent material. Liquid scintillation vials were either placed on vial trays and packaged in cardboard boxes for disposal, or placed loose in large plastic bags, then in cardboard boxes. There are no written records of spills or uncontrolled releases of liquid wastes in the burial pits within the LLRBS.

1.5 Prior Onsite Burials

See Section 1.1 for detailed description of on-site burial activities. All onsite burials are limited to those in the LLRBS.

Table 1: Burial Pit Descriptions

Pit(s)	Date Closed	Nuclides	Activity (MilliCuries)	Solid	Liquid	LSV*	Notes
1	4/6/51		**0.063**[1]				Records not available
2	8/4/53						Records not available
3	5/13/59	^{14}C	2.00	X	X		
4	4/3/61	^{14}C ^{90}Sr ^{3}H	23.45 0.44 0.16	X X	X X		 Fertilizers
5	No date						Form of waste unknown
6	No date						“
7	No date						“
8	8/67	^{14}C ^{3}H ^{36}Cl ^{226}Ra ^{63}Ni	99.05 64.67 0.10 0.15 0.65	X			Pesticides
9	9/68	^{14}C ^{3}H ^{63}Ni	43.93 5.13 0.50	X X	X (10 g)		
10	5/70	^{14}C ^{3}H ^{63}Ni	9.72 9.71 0.30	X X			
11	11/70	^{14}C ^{3}H ^{63}Ni	8.82 2.18 0.20	X X			
12	4/71	^{14}C ^{3}H	4.25 0.04		X (2 L)		
13	11/71	^{14}C ^{3}H ^{63}Ni	9.62 1.78 0.30	X		100 vials	
14	7/72	^{14}C ^{3}H ^{63}Ni ^{203}Hg	13.65 0.86 0.15 0.01 **0.298**[1]	X X X	X X	1000 vials	
15	No date						Form of waste unknown
16	8/72	^{14}C ^{3}H ^{203}Hg	6.87 1.09				
17	11/72	^{14}C ^{3}H	5.53 0.32				
18	12/72	^{14}C ^{3}H	1.39 0.23				
19	3/73	No records					
20	2/73	^{14}C ^{3}H	1.17 0.59				
21	3/15/73	No records					
22	9/7/73	^{14}C	8.06				

Pit(s)	Date Closed	Nuclides	Activity (MilliCuries)	Solid	Liquid	LSV*	Notes
		^{3}H ^{137}Cs ^{210}Pb	0.91 3.50 3.00				
23	1974	^{14}C ^{3}H ^{63}Ni ^{63}Ni D ^{210}Pb	41.33 4.77 3.00 24.50 2.90				
24	1975	^{14}C ^{3}H ^{63}Ni ^{210}Pb ^{36}Cl ^{203}Hg	18.70 44.56 2.00 5.00 0.00 5.00				
25	1975	No records					
26	1976	^{14}C ^{3}H ^{63}Ni ^{210}Pb ^{203}Hg	27.00 26.97 157.11D 7.50 0.03 0.52 **1.92**[1]				
28/29	4/16/81	^{14}C ^{3}H ^{63}Ni	109.24 82.37 430.5D 10D	 X X			 Cinnamic acid Nicotine
30	1982	^{14}C ^{3}H ^{226}Ra	74.04 147.68 9D	X X			
31	5/23/83	^{14}C ^{3}H	13.80 16.74				
32	8/1/83	^{14}C ^{3}H	6.74 0.60				Arachidonic acid KetoprostglandinF1
33	5/28/87	^{14}C ^{3}H	0.03 7.96	X X			
34	12/16/83	^{14}C ^{3}H ^{210}Po	8.22 0.60 0.10 **5.26**[1]				
35	5/10/84	^{14}C ^{3}H	2.25 18.15	X X	X X	X	
36	9/20/84	^{14}C ^{3}H ^{210}Po ^{63}Ni	16.87 27.95 0.50 12D		X		
37	10/3/85	^{14}C ^{3}H	56.43 11.45	X X			

Pit(s)	Date Closed	Nuclides	Activity (MilliCuries)	Solid	Liquid	LSV*	Notes
		 ^{63}Ni ^{210}Po	829.84D 38.5D 1.00	X X X			
38	6/11/86	^{14}C ^{3}H	2.22 4.37	X X			
39	9/26/86	^{14}C ^{3}H	4.13 24.05	X X			
40/41	1977	^{14}C ^{3}H ^{210}Pb ^{90}Sr	15.04 51.13 103.89D 35.00 6.80				
42/43	1978	^{14}C ^{3}H ^{63}Ni	25.54 129.59 549.54D 24.00	X			
44/45	1979	^{14}C ^{3}H ^{137}Cs ^{63}Ni	82.04 857.65 1158.13D 2.96 88D				
46	1980	^{14}C ^{3}H ^{63}Ni	99.45 111.33 250D 15D				
		Various sources & IDW	14.84	X			From pits 1, 14, 26, & 34

Foot Notes:

* LSV = Liquid Scintillation vials
D = sealed source

[1] For Pits 1, 14, 26, and 34 the number in bold is the total activity of waste excavated from each pit during LLRBS Characterization activities in November 2005.

Source: Draft Memorandum in ENTECH files, USEPA Region III, from W.G. Horner, USDA

Table 2: Radioactive Materials Disposed at the Low level Radioactive Burial Site between 1960 and 1987 (Activity Decayed to 07/01/08, in milliCuries)

Year	^{3}H*	^{14}C	^{210}Pb*	^{90}Sr*	^{36}Cl	^{63}Ni	^{226}Ra	^{22}Na*	^{55}Fe*	^{137}Cs
1960	0.05	3.22	--	0.30	--	--	?	--	--	--
1961	0.02	10.21	--	--	--	--	--	--	--	--
1962	0.20	1.51	--	--	--	--	--	--	--	--
1963	0.08	6.63	--	--	--	--	--	--	--	--
1964	0.86	54.22	--	--	0.002	--	0.15	--	--	--
1965	0.01	1.59	--	--	--	--	--	--	--	--
1966	0.16	18.43	--	--	0.10	0.13	--	--	--	--
1967	25.88	6.61	--	--	--	0.45	--	--	--	--
1968	2.28	41.68	--	--	--	0.45	--	--	--	--
1969	3.5	34.88	--	--	--	0.27	--	--	--	--
1970	0.95	10.32	--	--	--	0.18	--	--	--	--
1971	0.69	12.83	--	--	--	0.27	--	--	--	--
1972	0.13	27.28	--	--	--	0.089	--	--	--	--
1973	0.67	17.50	0.091	--	--		--	--	--	0.46
1974	19.3	37.70	--	--	--	1.8 21.8**	--	--	--	--
1975	2.07	4.84	--	--	0.001	--	--	--	--	--
1976	11.3 65.8**	27.0	--	--	--	6.7	--	--	--	--
1977	21.4 43.5**	15.04	12.4	4.57	--	--	--	--	--	--
1978	54.3 230**	25.54	--	--	--	21.4	--	--	--	--
1979	359 485**	82.04	1.27	--	--	78.6**	--	--	--	2.07
1980	46.6 104**	99.45	--	--	--	13.4**	--	--	--	--
1981	30.7 173**	152.04	--	--	--	8.93**	9.0**	--	0.007	--
1982	16.8 243**	26.62	--	--	--	--	--	0.0002	--	--
1983	10.3	26.69	--	--	--	--	--	0.0004	--	--
1984	16.0	18.76	--	--	--		--	--	--	--
1985	4.79 347**	56.43	--	--	--	34.4**	--	--	0.005	--
1986	10.2	12.60	--	--	--	--	--	0.002		--
1987	3.33	0.03	--	--	--	--	--	--	0.0002	--
Total	**641 1692****	**831.69**	**14.6**	**4.87**	**0.103**	**31.6 156.7****	**0.15 9.0****	**0.003**	**0.012**	**2.53**

Foot Note

* - Decayed value: Na and ^{55}Fe $T_{1/2}$ = 2.6 yrs; ^{3}H $T_{1/2}$ = 12.3 yrs; ^{210}Pb $T_{1/2}$ = 20.4 yrs; and ^{90}Sr $T_{1/2}$ = 27 yrs
** - Solid form – sealed source or electron capture detector
Source: Memorandum to Mr. Nicholas DiNardo, USEPA Region III, from W.G. Horner, USDA

2.0 FACILITY DESCRIPTION

2.1 Site Location and Description

The BARC facility is situated in the Atlantic Coastal Plain Province, and in this area, it can be described as gently rolling with broad valleys. The elevation varies from about 60 ft above mean sea level (MSL) where Indian Creek flows beneath Interstate 95/495 to 268 ft MSL in the extreme western portion of the facility on Cherry Hill Road near LLRBS. Topography slopes to the east and southeast at 10 to 15 percent toward the nearest perennial stream, the Little Paint Branch, located approximately 2,000 ft east of the site boundary. Downstream, Little Paint Branch feeds into Paint Branch, 1.4 miles to the south, eventually draining into the Anacostia River. There are extensive wooded tracts in the central and eastern portions of BARC, while open agricultural fields are prevalent in the western section.

There are many perennial and intermittent streams, wetlands, and surface water bodies within BARC boundaries. Drainage features include Paint Branch and Little Paint Branch, which flow from north to south and are located in the western portion of the facility. Indian Creek also flows north to south parallel to Edmonston Road; and Beaver Dam Creek flows east to west in the south-central portion of BARC. All of these drainage features eventually flow southward into the Anacostia River (approximately 6 miles from the facility), which empties into the Potomac River at Washington, D.C.

There are no known wetlands in the immediate vicinity of the LLRBS.

2.2 Population Distribution

The Census 2000 population for Beltsville, Maryland is 15,690 (USCB 2003). The nearest resident is approximately 500 feet from the LLRBS (due West across Cherry Hill Road and upgradient).

2.3 Current/Future Land Use

USDA has no current or future plans to release the BARC facility from ARS's current mission. The facility is best characterized as minimally developed, and is surrounded by land that is largely urbanized and densely populated. Inside the facility's boundaries, land use is agriculture, forest, and urban, with more than 800 buildings including laboratories, greenhouses, barns, office buildings, and some residences. A major portion of the facility is currently being used for crops, grazing livestock, and orchard research projects, primarily in the central and western portions of the BARC. BARC property is designated as 'agricultural open space' by the State of Maryland. The central and eastern portions of the facility are primarily covered with mixed deciduous/evergreen forest. The urbanized portions of BARC are scattered throughout the property.

Land use outside of the facility boundaries is largely mixed urban and lightly developed, primarily forested parcels. There is widespread residential development along the western, southwestern, and northwestern boundaries of BARC. Commercial development is prevalent along U.S. Route 1 and the Beltsville Industrial Center north of Sunnyside Avenue. Other major transportation routes that either border or pass through BARC are Interstate 95, Interstate 95/495, the Baltimore-Washington Parkway, and the B&O Railroad.

2.4 Meteorology and Climatology

Meteorological data for the Beltsville area is derived from references to Washington DC area data and is summarized in Table 3.

Table 3: Average Meteorological Data for Washington, D.C. Area

Month	High Temperature (°F)	Low Temperature (°F)	Precipitation (inches)	Snow (inches)	Wind Speed miles per hour (mph)	Wind Direction
January	42	27	3	4	10	NW
February	46	29	3	4	11	NW
March	56	38	3	Trace	11	NW
April	67	46	3	-	11	S
May	76	57	4	-	10	S
June	85	67	3	-	9	S
July	89	71	4	-	9	S
August	87	70	4	-	9	S
September	80	62	3	-	9	S
October	69	50	3	-	9	S
November	58	41	3	Trace	10	S
December	47	32	3	1	10	NW

The humid continental climate for the area allows for a large variance in weather conditions and temperatures. There is a difference of 42.8 °F from the coldest month of the year (January) to the warmest month of the year (July). The area receives an average of 39.54 inches of rain per year, with an average of 12 inches of snow per year (WUI 2003). No wet and dry seasons exist since precipitation is well distributed throughout the year. The greatest rainfall recorded for a 24-hour period was 7.31 inches, on August 11-12, 1928. The greatest snowfall ever recorded in the area was 28 inches, which occurred in January of 1922. The average annual relative humidity for the area is 63% and the average annual wind speed is 9.4 mph (NWA 2002).

2.5 Geology and Seismology

The USDA Natural Resources Conservation Service (NRCS) soils maps for Prince George's County describes numerous soil associations and groups of soils within the facility. Many of these units are described as comprised of silty loam, loamy sand, and sandy loam of variable slope, drainage characteristics, and susceptibility to erosion. Surface soils are underlain by highly variable deposits ranging from gravels to clays, some as old as Cretaceous (SCS, 1967).

The geology at the BARC consists of Lower Cretaceous sediments of the Potomac Group, which consists of the Patuxent formation, the Arundel, and the Patapsco formation, the youngest of the three. The LLRBS lies on the Patuxent Formation. Soil textures beneath the site are well-sorted sand and gravel with minor clay lenses. This sand and gravel sequence overlies several feet of clay, below which are the igneous and metamorphic rocks of the Piedmont Province. The Patuxent and Patapsco formations are comprised primarily of sand and gravel, and comprise the most prevalent water bearing aquifers in Prince George's County. The Arundel is mostly clay, and creates artesian conditions in the underlying Patuxent formation in some locations. Recharge of the Patuxent formation occurs where it outcrops in the western portions of the

BARC. This wedge of sediments made up of the Patuxent, Arundel, and Patapsco formations dips to the southeast, parallel to the regional groundwater flow (Apex, 1991).

The history of this area shows a low probability of an earthquake of sufficient magnitude to cause damage to structures. Harford County, Maryland, experienced two to three earthquakes on March 11 and March 12, 1883. These earthquakes were felt as far away as Baltimore County. The intensity was in the IV - V range (USGS, 2008).

Another moderate earthquake occurred less than two years later, on January 2, 1885, in an area near the Frederick County, Maryland – Ludon County, Virginia, border. Clarke, Fairfax, Fauquier, and Shenandoah Counties, Virginia, also reported this earthquake. Since 1885, earth vibrations felt in Maryland have been associated with sources from adjacent states and points as far away as the St. Lawrence Valley and Timiskaming, Canada (USGS, 2005).

2.6 Surface Water Hydrology

There are many perennial and intermittent streams, wetlands, and surface water bodies within BARC boundaries. Drainage features include Paint Branch and Little Paint Branch, which flow from north to south and are located in the western portion of the facility. Indian Creek also flows north to south parallel to Edmonston Road; and Beaverdam Creek flows east to west in the south-central portion of BARC. All of these drainage features eventually flow southward into the Anacostia River (approximately 6 miles from the facility), which empties into the Potomac River at Washington, D.C. The distance from the Site's location on Little Paint Branch to the Potomac River is 13 stream miles downstream (Apex, 1993). The nearest surface water body is a branch of the Little Paint Creek, which lies 2,000 feet east of the site.

2.7 Groundwater Hydrology

The geology at BARC consists of Lower Cretaceous sediments of the Potomac Group, which consists of the Patuxent, the Arundel, and the Patapsco Formations, in decreasing age. The Patuxent and Patapsco Formations are composed primarily of sand and gravel, and comprise the most prevalent water bearing aquifers in Prince George's County. The Arundel is mostly clay, and creates artesian conditions in the underlying Patuxent Formation in some locations. Recharge of the Patuxent Formation occurs where it outcrops in the western portions of BARC. This wedge of sediments made up of the Patuxent, Arundel, and Patapsco Formations dips to the southeast, parallel to the regional groundwater flow (Apex, 1991).

The uppermost unit consists of the Terrace Deposits which are approximately 30 ft thick and unsaturated. The water-table aquifer coincides with the Patuxent Formation, and is approximately 20 ft thick. Beneath the Patuxent is the Wissahickon Gneiss. A saprolite layer of approximate 25 ft thickness creates an aquitard between the two formations. The nearest potable water well is 2.5 miles east of the LLRBS and is screened from 200 to 600 ft.

2.8 Natural Resources

There are no known significant mineral deposits, water resources, coal deposits or other natural resources which, if exploited would affect the licensee.

3.0 RADIOLOGICAL STATUS OF THE LLRBS

The LLRBS is made up of a total of 50 designated waste burial pits, of which, only 46 were reportedly used (see Table 1). The pits are approximately 10 ft wide by 12 ft long by 10 ft deep and are separated approximately 6 feet horizontally from one another. Each pit was reportedly backfilled to surface grade with at least 5 ft of clean fill. Two contiguous fenced fields, the North Field and the South Field, make up the LLRBS, each of which is approximately 150 ft by 200 ft. Figure 4 is a schematic of the North Field. The South Field was reportedly never used for disposal of any waste materials, although individual pits had been designated for such disposal (Apex 1993). The geophysical surveys performed during the WCS also confirmed with the historical data indicating that the South Field was not used for burials (Cabrera 2007).

The LLRBS was established on June 23, 1949, and was used for the disposal of low level radioactive waste until 1987. The LLRBS is permitted under a USDA nation-wide broad scope license originally issued by the AEC, and later by the NRC. Records indicate the last liquid burial at the LLRBS was on September 17, 1984. From September 24, 1985 until disposal activities ended in 1987, all burials were dry solids packed in 55-gallon drums (ENTECH, 2000).

The total volume of the waste buried at the LLRBS is not well documented. During the years of disposal activity, the NRC license required only the maintenance of radionuclide and activity records for wastes disposed as per NRC guidance presented in the Code of Federal Regulations (CFR) 10 CFR 20.304. ARS estimates indicate that as much as 33,000 cubic ft (1,222 cubic yards) of waste may have been buried at the site (Apex, 1993).

During the 2006 Characterization Survey, 4 of the 46 pits were excavated

Information obtained from USDA Radiation Safety Staff and the BARC Safety Occupational Health and Environmental Staff indicates the NRC license that covers the LLRBS was renewed through September 30, 2005 (ENTECH, 2000). The NRC license does not authorize additional burials of radioactive material, but requires the continued maintenance and monitoring of the LLRBS.

Buried materials include radioactive isotopes and scintillation fluids (isotopes and organic fluids); contaminated metal, glass, and plastic objects; contaminated animal carcasses; and animal wastes. ARS records do not reveal the types of organic liquids contained in the scintillation vials, nor is there any indication of volume, except as noted on Table 1. Typically, organic solvents associated with scintillation fluids include toluene and xylenes. Types of containers disposed, according to files, are cardboard boxes of 1 to 4 cubic ft, 1 to 5-gallon containers for liquids, plastic milk jugs, plastic carboys, solvent bottles, fiberboard drums, and 55-gallon drums. Liquid containers were placed in cardboard boxes, usually 4 to a box (Apex, 1993).

No known records exist listing the specific types of containers used for the early burials. Records from the 1980s indicate contaminated, non-flammable glass and plastic (vials, pipettes, needles, scalpels, etc.) were buried in cardboard boxes. Animal carcasses, bedding, and excreta were sealed in polyethylene bags and placed in boxes. Liquid wastes were packed in plastic containers and placed in cardboard boxes. Animal remains, generally contaminated with tritium (^{3}H) and carbon-14 (^{14}C), were routinely incinerated beginning in the early 1980s, in one of two incinerators located at BARC. It is possible that incinerator ash, which tested positive for radioactivity, could have been sent to the LLRBS (Apex, 1993), however, no known records exist regarding ash disposal.

Limited historical data is available regarding the layout of disposal trenches in the North Field. A summary of disposed isotopes and associated activities for materials disposed after 1960 is presented in the EE/CA (ENTECH, 2000).

A recently discovered source of historical data on actual pit boundaries is a letter from Radiation Service Organization, Inc., who was contracted by the ARS to conduct site clearing in 1988 (RSO, 1989). They produced a working drawing of the northern portion of the North Field, included as Figure 3. The clearing operations necessitated the destruction of the fence posts (they were cut off at ground level) separating the burial pits.

Another recently identified source of data is a draft summary of the LLRBS produced by Mr. Horner of USDA in the early 1990s. The summary contains a schematic of the North Field depicting 46 burial sites. A summary table (Table 1) includes radionuclides that were disposed in each trench, along with the activity in milliCuries, whether the disposed materials were solid, liquid, vials of liquid scintillation materials, other chemicals, and the date each pit was closed. This inventory was used as the basis for selecting pits to be exhumed for characterization purposes. Total annual inventories of radionuclide disposals are shown in Table 2.

From its inception to September 17, 1984, various BARC research laboratories generated radioactive waste, which included dry solid materials such as gloves, paper, and syringes; other solid materials in the form of sealed sources and electron capture detectors; glass and plastic liquid scintillation vials containing scintillation liquids; aqueous and organic bulk liquids; animal carcasses, bedding, and excreta; and other radioactively contaminated laboratory wastes.

The types of radioactive waste material at the LLRBS consisted of ^{3}H, ^{14}C, ^{137}Cs, ^{36}Cl, ^{32}P, ^{63}Ni, ^{226}Ra, ^{210}Pb, and other radionuclides. Some of the radiological materials deposited in the LLRBS are short-lived (i.e., have a short radiological half-life), including ^{32}P, ^{210}Po. Inventory records of burials from 1949 through 1960 could not be located.

The USDA's license, and then-current regulations, required only the maintenance of radionuclide and activity records for wastes disposed at the LLRBS. As a result, there is no record of waste container type, waste type, chemical form, or physical form. Interviews with employees present when the LLRBS was active have revealed that most of the waste materials were placed into cardboard boxes prior to disposal in the pits. Bulk liquid wastes were transferred into plastic or glass carboys and packaged in cardboard boxes with some cushioning and absorbent material. Liquid scintillation vials were either placed on vial trays and packaged in cardboard boxes for disposal, or placed loose in large plastic bags, then in cardboard boxes.

Depending on the anticipated amount of wastes generated, disposal pits were left uncovered until a sufficient volume of waste was contained. No animal carcasses or excreta were disposed in the pits until immediately before closure. After September 17, 1984, no liquids were disposed of at the LLRBS. After September 1985 only dry solid wastes and animal carcasses, packaged in 55-gallon steel drums, were disposed at the site. Radioactive waste shipment and disposal manifests were prepared and maintained for disposals after September 1985, thus providing supplementary waste information for these disposals. Four of the pits contain only dry solid or animal wastes, disposed after September 1985, packaged in steel drums.

3.1 Results of 2006 Waste Characterization Survey (WCS)

A WCS of LLRBS was conducted in 2006 to assess conditions in the waste pits and confirm existing assumptions about the nature of the disposals. The survey included the following tasks:

- Geophysical surveys in the North Field to delineate the burial cells, and in the South Field to confirm the assumption that no burials have taken place there,
- Gamma and beta walkover surveys to map potential near-surface radiological materials,
- Sampling groundwater from selected wells to assess migration of contaminants away from source materials,
- Excavation, radiological characterization, segregation, and packaging of waste soils and materials in 4 of the 46 documented waste cells,
- Sampling soil along the floor of each excavation,
- Installing temporary wells to sample groundwater from beneath each excavated pit,
- Backfilling and restoring all excavations with clean fill.

The geophysical surveys of the South Field further substantiated the assumption that the South Field was not used for disposals.

3.1.1 Impacted Soil and Groundwater

Approximately 4 of the 46 burial pits were excavated (Pits 1, 14, 26, and 34/34C). During excavation, the sidewall of Pit 34 failed, resulting in releasing leachate (perched water) at the adjacent area. It was decided to designate the adjacent area as Pit 34C in order to recover this waste. Based on the work completed thus far, it is anticipated that the nature and extent of the waste in the remaining pits will be highly variable. Waste materials encountered during excavation include laboratory trash (gloves, paper, metals, plastics, laboratory glassware, and other wastes generated during the process of performing laboratory analyses), liquid scintillation vials, radioactive sources, bulk soils containing small amounts of waste or debris not readily separable from soil, animal remains, and bulk liquids in their original containers. Disposed materials often contained both radionuclides and hazardous wastes at concentrations that exceed NRC and/or EPA screening criteria for groundwater and soil.

Radionuclide concentrations found in soil and water were generally within regulatory limits. In soil and debris the observed activity in composite samples ranged from zero to a maximum result of 223 pCi/g of ^{14}C.

3.1.2 Non-Radiological Contaminants

Non-radiological soil contaminants include the organic chemicals chloroform, benzene, bromodichloromethane, and trichloroethylene and the metals arsenic, chromium, and vanadium.

The maximum concentration of chloroform was 762 µg/kg collected from approximately one foot below Pit 26. The USEPA Region 3 Risk Based Concentration (RBC) for chloroform is 1E7 µg/kg. The maximum concentration of benzene was 625 µg/kg from 2 feet below Pit 34. The RBC for benzene is 52,000 µg/kg. The maximum concentration of bromodichloromethane was 379 µg/kg, collected from 2 feet below Pit 34. The RBC for bromodichloromethane is 46,155 µg/kg. The maximum concentration of trichloroethylene was 474 µg/kg, collected from 2 feet below Pit 34. The RBC for trichloroethylene is 1,600 µg/kg.

Arsenic was the most common non-radiological contaminant, often exceeding RBC concentrations, even in the Reference Area. The maximum arsenic result in soil was 59.6 mg/kg in a sample from approximately 18 feet beneath Pit 34. The RBC for arsenic is 1.91 mg/kg. The maximum concentration of chromium in soil was 68 mg/kg, compared to an RBC of 3066 mg/kg. The maximum concentration for vanadium was 90 mg/kg in a sample from Pit 34. The RBC for vanadium is 307 mg/kg.

Groundwater that was sampled from beneath the pits had levels of benzene, trichloroethylene, and chloroform that were slightly greater than the RBCs. The maximum aqueous benzene concentration was 1.6 µg/L, which was beneath Pit 34C. The RBC for benzene is 0.34 µg/L. The maximum trichloroethylene concentration was 0.487 µg/L, compared to the RBC of 0.026 µg/L. The maximum aqueous chloroform concentration measured was 10.6 µg/L, compared to the RBC of 0.15 µg/L.

Carcasses of rats and chickens with associated decomposed tissue were encountered, disposed in polyethylene bags, and some bones were encountered in fiberboard boxes.. The volume was less than 0.01% of wastes.

3.1.3 Radiological Contaminants

Radiological contaminants ^{3}H and ^{14}C are found in low concentrations throughout the waste where liquid scintillation vials were present, with ^{90}Sr, ^{36}Cl, and ^{226}Ra occurring less frequently.

The maximum ^{3}H concentration in subsurface soils was 140 pCi/g, collected from Pit 14, which is slightly higher than the NRC guideline screening level of 110 pCi/g. All other results for ^{3}H in soil were 30 pCi/g or less.

The maximum concentration of ^{226}Ra was 7.53 pCi/g, collected from the base of Pit 34C. Seven of 13 soil samples collected from Pit 34C exceeded the NRC guideline level of 0.7 pCi/g. Two of 8 samples collected from the reference area also exceeded the NRC guideline level for ^{226}Ra.

The maximum ^{14}C concentration in soil samples was 210 pCi/g, collected from the bottom of Pit 26. Two other samples for ^{14}C, another one in Pit 26 and on in Pit 14 had ^{14}C concentrations of approximately 15 pCi/g. These three results exceed the NRC guideline level of 12 pCi/g, while all other results were below that level.

The maximum ^{90}Sr concentration in soils was 0.278 pCi/g, compared to the NRC guideline level of 1.7 pCi/g. The maximum concentration of ^{36}Cl in soil was 0.11 pCi/g compared to a guideline level 0.36 pCi/g.

The radionuclides that exceeded RBCs in water samples were ^{228}Ra, ^{230}Th, and ^{3}H. The maximum ^{228}Ra concentration in unfiltered groundwater was 29.7 pCi/L, compared to a field-filtered aliquot of the same sample which contained 13.9 pCi/L of ^{228}Ra. The MCL for combined $^{228}Ra + ^{226}Ra$ is 5 pCi/L. ^{230}Th was encountered in groundwater beneath Pits 1, 14, and 34 at concentrations ranging from 18 pCi/L to 28 pCi/L, compared to an MCL concentration of 15 pCi/L.

The maximum ^{3}H concentration in groundwater beneath the pits was 1,030 pCi/L, compared to the NRC Derived Concentration guideline level of 20,000 pCi/L.

3.1.4 Summary of Treatment and Disposal Activities

A total of less than 25 mCi of activity has been removed from the excavated pits (Pits 1, 14, 26, and 34/34C). This activity did not exceed the DandD limit and was small compared to the total activity buried at the site shown in Tables 1 and 2. Pits 1 and 14 were dry. During the characterization survey, a total of 825 gallons of leachate was collected from two pits: Pit 26 and Pit 34/34C. Pit 26 contained roughly 660 gallons, 34 and 34C contained the remaining 165 gallons. More than 75% of the activity is due to ^{226}Ra and ^{3}H. The ^{3}H activity was largely in perched water pumped from Pits 26 and 34 during dewatering operations. Another 20% of the activity is from ^{14}C and ^{63}Ni, each with about 10% of the total. The remaining 5% is from ^{210}Pb, ^{36}Cl, and ^{90}Sr.

To date 6.5 cubic yards (CY) [28 drums] of liquid scintillation vials, 0.5 CY [2 drums] of bulked liquids, and 3 radioactive sources/devices have been shipped from the site for processing and disposal. Table 4 summarizes the volumes of wastes recovered. Represented as percentages, the recovered wastes were composed of 60% contaminated soil, 21% debris, 10% LSC vials, 8% dry active waste, 1% bulk liquids. The liquid scintillation vials and the bulked mixed waste were shipped to the Permafix facility in Gainesville, FL for treatment and disposal. Two ^{226}Ra sources and a ^{63}Ni electron capture device were shipped to Alaron Corporation and were received under their source recycling license.

Table 4 Summary of Wastes Excavated During Characterization Survey

Pit #	**Soil**	**Debris**	**Bones**	**Liquid Scintillation Vials**	**Dry Active Waste**	**Bulked Liquids**	**Leachate**	**Radioactive Sources**
Units	CY		fiber boxes	CY				
Pit 1	10.9	0.5						
Pit 14	10.6		4					
Pit 26	5.2	4.4		1.9				
Pit 34	9.2	8.2		2.7				
Pit 34C	6.8	2.7		1.9				
Misc.					6.3	0.5	4	3
TOTAL	**42.7**	**15.8**	**4**	**6.5**	**6.3**	**0.5**	**4**	**3**

3.2 Contaminated Structures

There are no known structures within the LLRBS where licensed activities occurred; therefore, no structural decommissioning activities are warranted.

3.3 Contaminated Systems and Equipment

There are no known systems or equipment within the LLRBS; therefore, no system or equipment decommissioning activities are warranted.

3.4 Surface Soil Contamination

Based on historical records provided by the BARC and the 2006 Characterization survey, the top five feet of overburden associated with each burial pit is considered uncontaminated soil. There is no known surface contamination.

3.5 Subsurface Soil Contamination

Due to previous burial activities within the LLRBS, subsurface soils associated with each burial pit have the potential for significant radiological and non-radiological contamination.

3.6 Surface Water

There are no known surface water features within the LLRBS and; therefore, are not addressed in this decommissioning plan.

3.7 Groundwater

Previous investigations suggest radionuclide and chemical contaminants may have migrated from the burial pits due to infiltration of surface water and vertical migration to the water table. Such migration could result in contamination of the vadose zone and aquifer. Both ^{3}H and chloroform plumes have been identified in the underlying aquifer. Figure 5 identifies sampling wells and results of groundwater sample analyses.

4.0 DOSE MODELING EVALUATIONS

On June 21, 1997, the NRC published the final rule on "Radiological Criteria for License Termination", the License Termination Rule (LTR), as Subpart E to 10 CFR Part 20. The criteria for termination with unrestricted release are

1) Residual radioactivity that is distinguishable from background, and results in a total effective dose equivalent (TEDE) to an average member of the critical group that does not exceed 25 mrem/yr, including that from groundwater sources of drinking water; and

2) Residual radioactivity has been reduced to levels that are ALARA.

Determination of the levels, which are ALARA, must take into account consideration of any detriments, such as deaths from transportation accidents expected to potentially result from excavation and waste disposal activities. For the decommissioning of the LLRBS, a dose objective of 25 mrem/yr above background is applicable and is therefore used as the basis for demonstrating that the LLRBS should be released for unrestricted use. The method for evaluating the dose objective is provided below.

Unrestricted release of a site requires an evaluation of the potential to produce a radiation dose to individuals that might be exposed to future soil, water, and foodstuffs derived from the site. The radiation dose provides a measure of risk due to the residual radioactivity remaining in the soil after site remediation has occurred. The residual radioactivity is that amount of radioactivity in the soil that is in excess of the naturally occurring background radioactivity in the surrounding areas.

The potential risk, in turn, may be translated into a soil concentration remediation guideline or Derived Concentration Guideline Level (DCGL). The DCGL is a site-specific soil concentration determined to be protective of the health of individuals who may become exposed to the residual radioactive materials remaining at the site. The DCGL establishes a soil concentration guideline level consistent with the exposure scenario and dose pathways determined for the site to ensure any dose or risk remains protective of the health of individuals and is less than regulatory guidelines. The guideline is conservatively developed to address the risk to an average member of the critical exposure group.

The development of DCGLs for the site include:

- Determination of radiation exposure scenarios and pathways consistent with likely, as well as possible future uses, of the site.
- Development and inclusion of site-specific information from the "Characterization Survey Work Plan, Low Level Radioactive Burial Site, Beltsville Agricultural Research Center (BARC), Beltsville, Maryland" (CABRERA 2004).
- Consideration and use of regulatory guidance document values for physical, behavioral, and metabolic factors shown in NUREG/CR-5512 (NRC 1999).

The DCGLs are based on conservative assumptions with a dose criteria limit of 25 mrem/yr peak annual TEDE over a 1,000 year time period as provided by 10 CFR Part 20 Subpart E.

4.1 Evaluation of DCGL Determination Options

The site-specific soil DCGL or soil screening DCGL may be determined by one of two methods. The first option may include utilizing a conservative generic soil screening radioactivity level in the soil as described by the NRC screening levels listed in 64 FR 68395 and further detailed in NUREG 1757 (NRC, 2006). Alternatively, the derivation of site-specific soil concentrations based on the health hazard (dose) posed by residual radioactive material remaining in the soil on the site may be used.

The soil guideline level may also be tied to restricted or unrestricted use. In the restricted use case, the residual levels of radioactivity may require occupancy and/or use restrictions along with third party custodial care, control, and maintenance of the site. In the unrestricted use situation, no restrictions are attached to the site and no special custodial care, control, or maintenance is required.

Consideration and inclusion of exposure scenarios and pathway inputs may be varied so a conservative bounding approach is assumed. This bounding scenario assumes the resident farmer scenario whereby a member of the critical exposure group is expected to live on the site and utilize water, food stuffs, meat, milk, and fish derived from the Site. This generic scenario was selected and utilized for the evaluation.

4.1.1 Potential Application of Screening Levels

This option provides a generic soil guideline level based on limiting and conservative assumptions with respect to occupancy time, water consumption, food consumption, and default site-specific factors. This method may result in decisions resulting in the removal of larger than needed land areas due to conservatisms embodied within the generic screening level DCGL derivations. Additional detriments include a higher degree of excavation, truck traffic associated with a higher level of removal operations, and disposal of increased amounts of soil waste.

As described in the guidance presented in Federal Register Volume 65, June 13, 2000, the use of the soil screening values presented in Federal Register Volume 64, December 7, 1999 may be used to demonstrate compliance for soils under specific guidelines. The four guidelines by which soils may be deemed acceptable for release for unrestricted use are as follows:

1) The residual radioactivity has been reduced to levels that are ALARA
2) The residual radioactivity is contained in the top layer of the surface soil (i.e., within approximately 30 centimeters [cm] from surface in accordance with NUREG-1757 Volume 2 [NRC, 2006])
3) The unsaturated zone and the groundwater are initially free of radiological contamination; and
4) The vertical saturated hydraulic conductivity at the specific site is greater than the infiltration rate.

The Site does not meet the criteria for use of generic screening criteria, because there is existing groundwater contamination. Thus, these criteria are not proposed for decommissioning of the LLRBS.

4.1.2 Application of Site-Specific Soil Guidelines

This option provides for a property-specific guideline that utilizes site-specific parameters along with exposure scenarios and pathways best suited to the site. Likely future uses of the site and dose pathways result in a corresponding acceptable and safe level of potential risk to an average member of the general public, without applying the generic default guidelines utilized by the screening level method. This approach minimizes the potential for significant soil excavation and the accompanying damage to the ecosystem surrounding the Site, as well as reducing truck traffic and excavated soil disposal volumes. This site-specific option has been selected and utilized for the evaluation presented herein in Section 4.2. Certain input parameter values will remain as the computer code default values.

4.2 Unrestricted Release Using Site-Specific Information

RESRAD (Residual Radioactivity) is a computer code developed by Argonne National Laboratory (ANL). The code was developed for the DOE to calculate site-specific cleanup criteria for radioactive materials in soils. The code has been benchmarked and validated and is widely used by the DOE, NRC, EPA, USACE, industrial firms, universities, and foreign governments. This code was used to estimate potential dose to the average member of the critical group.

The code provides radiation dose assessment and guideline calculations based on internal models and database information. The code calculates direct dose, inhalation dose, and ingestion doses from air, water, produce, meat, milk, fish and aquatic foodstuffs, and incidental ingestion of contaminated soils. The code utilizes the EPA Federal Guidance Report No. 11 and Federal Guidance Report No. 12 for inhalation and ingestion dose and for direct external dose conversion factors, respectively.

The DCGL analysis provided herein was produced with the aid of the latest version of RESRAD available at the time of analysis, RESRAD 6.3, August 2005.

4.2.1 Definition of the Critical Group

For the purpose of conservatism, the critical group has been defined as a resident farmer. Applicable exposure pathways for the resident farmer are as follows:

- External exposure to penetrating radiation from volume soil sources while outdoors.
- External exposure to penetrating radiation from volume soil sources while indoors.
- Inhalation exposure to resuspended contaminated soil while outdoors.
- Inhalation exposure to resuspended surface sources of contaminated soil tracked indoors.
- Ingestion of drinking water from a groundwater source.
- Ingestion of plant products grown in contaminated soil.
- Ingestion of plant products irrigated with contaminated groundwater.

- Ingestion of animal products grown onsite (i.e., after animals ingest contaminated drinking water, plant products, and soil).
- Ingestion of fish and aquatic foods from a contaminated surface water source.
- Direct ingestion of contaminated soil.

4.2.2 RESRAD Parameter Definitions

The model utilizes numerous parameters that are input to calculate dose. These parameters can be broadly grouped into physical parameters, behavioral parameters, and metabolic parameters.

Default physical parameters incorporated in RESRAD may be used to generate potential doses. These input parameters are defined to encompass the variability expected across a large spectrum of conditions in all regions of the country and have been benchmarked by NRC (NRC 1999b). These parameters may depend on physical features of the site may vary based on local geological and meteorological characteristics.

Modifications to these parameters were made based on a combination of site-specific values based on (CABRERA 2004).

The default behavioral and metabolic parameters represent the average member of the critical exposure group. These parameters are based on the variability between individuals in the critical exposure group. They were defined by the development of distributions representing the exposure group, and then selecting the mean of the distribution to represent the average member of the group. These mean values are not expected to change based on site-specific information. However, as with the default physical parameters, the behavioral and metabolic parameters may utilize site-specific parameters to better define expected food consumption rates associated with the Site. In addition, dose conversion factors associated with elemental plant, meat, milk, and fish transfer factors and bioaccumulation as provided by NRC (NRC 1999b) were used. In certain cases a weighted average value was utilized.

Appendix A, Attachments A and B provide a summary of RESRAD model input parameters and the data source. Highlighted areas are other than RESRAD 6.3 default values. Appendix A, Attachment C provides information with respect to bioaccumulation factors and determination technique for certain bioaccumulation factors to accommodate differences between the DandD and RESRAD codes.

4.2.3 Summary of Calculated DCGLs

The applicable resident farmer scenario pathway is selected for analysis by the computer code (e.g., RESRAD) based on the assumed critical exposure group. The input default parameters or modified site-specific parameters are entered. The computer code is set to provide dose evaluations for a time period of 0-1,000 years. The dose is ultimately expressed as a soil cleanup level or DCGL concentration in terms of pCi/g in the soil.

The summary DCGL results from the RESRAD computer runs are provided in Appendix A, Attachments D and E. These reports show the dose conversion factors, a summary of the site-specific parameters, summary of the pathway selections, summary of the contaminated zone and total dose, total dose components by time, the single radionuclide soil guideline DCGL, the dose by nuclide summed over all pathways and the soil concentration by nuclide.

The individual DCGL for each isotope varies due to the various pathways considered. Table 5 provides a summary of the individual DCGL values using NRC soil screening values and using the Beltsville site-specific values generated with RESRAD 6.3.

In two cases the site-specific DCGL is greater than the NRC soil screening value. This is due to differences between the DandD and RESRAD models and how they handle certain soil-to-plant bioaccumulation factors and associated animal and human food consumption factors for those plant foodstuffs.

Table 5: Site-Specific DCGLs and Screening Criteria

Isotope	NRC Screening Value	Beltsville Site Specific DCGL
^{14}C	1.16E+01	1.62E+02
^{36}Cl	3.62E-01	7.10E+00
^{137}Cs	1.10E+01	1. 68E+01
^{55}Fe	1.03E+04	1.07E+06
^{3}H	1.08E+02	2.64E+02
^{22}Na	4.25E+00	4.87E+00
^{63}Ni	2.11E+03	2.06E+04
^{210}Pb	8.46E-01	1.62E+00
^{226}Ra	6.94E-01	1.34E+00
^{90}Sr	1.72E+00	3.88E+00

4.2.4 DCGL Sum of the Fractions Rule

When there are multiple radionuclides present in the soil, the allowed soil concentration levels must follow the sum of the fractions rule. This will ensure that the sum of the individual fractions for each isotope to its individual DCGL fraction does not exceed unity and enables field measurement of a gross activity DCGL. The gross activity DCGL considering the isotopes of concern is described by:

$$\text{Gross Activity DCGL} = \frac{1}{\left(\frac{f_1}{DCGL_1} + \frac{f_2}{DCGL_2} + \ldots + \frac{f_n}{DCGL_n}\right)}$$

where,

f_1 = Fraction of 1st isotope in the soil

$DCGL_1$ = DCGL for the 1st isotope, pCi/g

f_2	=	Fraction of 2nd isotope in the soil
$DCGL_2$	=	DCGL for the 2^{nd} isotope, pCi/g
f_n	=	Fraction of n^{th} isotope in the soil and
$DCGL_n$	=	DCGL for the n^{th} isotope, pCi/g

5.0 PLANNED DECOMMISSIONING ACTIVITIES

The scope of decommissioning activities on the site will include the excavation and removal of the waste materials and soils impacted above the radiological DCGLs. The scope of decommissioning activities also includes the proper packaging and transportation of LLRW in IP-1 or IP-2 containers and non-radioactive industrial waste to licensed burial and industrial landfill facilities, respectively. LLRW will be shipped to Energy Solutions of Utah for disposal (or other suitable licensed disposal facility). Radiological FSSs will be performed using MARSSIM (NRC 2000b) guidance following decommissioning activities.

Results of the 2006 WCS were used to estimate the volume of wastes remaining at the LLRBS. The footprint of the rectangle composing the North Field of the LLRBS is 200 ft long and 150 ft wide. A total of 46 waste cells are in this area, and each is approximately 12 ft long and 10 ft wide. Each cell is separated from neighboring cells by sidewalls that are 5 feet thick. There is a 5 ft thick layer of clean overburden over each cell, beneath which is a 5 ft thick layer of waste. The ex-situ volume of waste is estimated to be approximately 1040 CY, comprised of approximately 600-660 CY of waste soil, 200-240 CY of contaminated debris, 80-100 CY of LSC vials, 80-100 CY of dry active waste, and 10-12 CY of bulk liquids. It may be technically impracticable to segregate and clear the interior sidewalls and separate the waste cells from each other. Adding the volume of the soil in the sidewalls to the volume of the waste increases the total ex-situ volume of waste to approximately 6600-7300 CY.

Decommissioning activities are planned to be executed in a single phase, but based on funding constraints more than one phase may be required. Likely and potential mixed wastes in the burials are expected to result in substantial waste management costs. If more than one field event is required, final status survey samples will be collected in each excavated pit and low-permeability fill, and clean cover soil, may be used to backfill the excavation. Following excavation and backfill of all pits, a surface MARSSIM final status survey will be performed.

Decommissioning activities will be performed by a contractor who possesses an NRC decommissioning radioactive materials license. The contractor's license safety procedures will serve as the primary radiation protection protocols. All onsite decommissioning activities will be governed under the contractor's license. A written agreement will be executed between USDA and the contractor prior to initiation of decommissioning. This agreement will unambiguously describe which the license responsibilities of USDA and the contractor. USDA recognizes that this approach in no way affects its responsibility to decommission the LLRBS in a timely manner.

Several decommissioning tasks will be performed under this decommissioning plan. The tasks are listed below in the general order of performance, however some tasks may be performed concurrently:

1. Excavate the non-impacted earthworks that serve as a covering for the entire north field (shown in Figure 2).
2. Excavate the waste from the cells. Use a power screen or other mechanical separation as appropriate to size and segregate debris, LSC vials, and soil. Precautions will be taken to prevent free liquids being sent to a rad-waste landfill.
3. Place the excavated material in a temporary storage area separate from LLRW for future reuse or packaging and shipment. Separated waste streams will include;

   a. intact LSC vials,
   b. and waste soil/debris,
   c. Bulk liquids.

4. Package wastes, prepare shipping papers, and transport the removed LLRW to the licensed disposal facility.
5. The soil will be surveyed as the area is excavated. During excavation, the licensee will conduct surveys of sufficient quality to serve as portions of the FSS. The South Field does not contain waste cells, however, since packaged wastes will be temporarily stored there, a release survey will be conducted following removal of waste. The temporary storage area in the South Field will be sampled and surveyed prior to use in order to insure that it uncontaminated. The boundary of the South Field may need to be expanded if necessary to manage material.
6. Perform a FSS of the LLRBS site to support release for unrestricted use.

5.1 Contaminated Structures

There are no known structures within the LLRBS; therefore, no structural decommissioning activities are planned.

5.2 Contaminated Systems and Equipment

There are no known systems or equipment within the LLRBS; therefore, no decommissioning activities associated with systems and equipment are planned.

5.3 Surface and Groundwater

The licensee expects that no surface water exists within the LLRBS site and it is expected that no groundwater will be encountered during excavation activities. However, during the WCS, perched water (or leachate) was encountered in two excavations (Pits 26 and 34). Steps will be taken during the excavation of the site. Accordingly, any water accumulated during future decommissioning activities will be handled, stored, and dispositioned in accordance with appropriate procedures and regulatory requirements. A Surface Erosion Control Plan will be implemented to mitigate any inflow or runoff due to precipitation.

5.4 Schedule

The proposed schedule to accomplish the decommissioning activities, as outlined in this plan, is as follows:

- Revise DP and Resubmit – February 2009
- Perform 1st round of decommissioning – July 2009
- Perform potential 2nd round of decommissioning – March 2010
- Perform Final Status Survey of LLRBS surface soils – April 2011 (or earlier if only a single round of decommissioning is necessary)

Submit Final Status Survey Report – June 2011.(or earlier if only a single round of decommissioning is necessary).

6.0 PROJECT MANAGEMENT AND ORGANIZATION

This project has been initiated by, and all decommissioning activities will be performed under, the cognizance of the authorized representative of the license holder for this Site:

Mr. John Jensen – Radiation Safety Officer
United State Department of Agriculture
G.W. Carver Center
5601 Sunnyside Avenue, Mail Stop 5510
Beltsville, MD 20705-5500
(301) 504-2440 Fax:(301) 504-2450
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6.1 Decommissioning Management Organization

This decommissioning project will be performed within the management and organizational structure described in this section. USDA will retain overall responsibility for management and execution of the DP. USDA will contract an NRC licensed decommissioning company to execute the DP under their direction. Responsibilities of key individuals are described in the following subsections. Additional staff, along with applicable subcontractors may be utilized as appropriate.

6.2 Decommissioning Task Management

Safety guidelines for work in radiologically controlled areas have been established in the RSP. These guidelines will be amended by RWPs, as necessary, to provide administrative control of all activity within areas that may have radiological hazards, in order to maintain exposure ALARA. The radiation work described in this decommissioning plan will be considered when specific procedures are developed for work in radiologically controlled areas. RWPs will address the need for the following radiation safety precautions, at a minimum:

- External Dosimetry
- Contamination Control, including boundaries and protective clothing
- Access Controls and sign-in requirements
- Internal Dosimetry
- Air sampling and respiratory protection requirements

These RWPs will be reviewed and approved by the contractor's SRSO and prior to implementation. The SRSO will ensure that ambient radiation, surface radioactivity, and airborne radioactivity surveys are performed, as required to define and document the radiological conditions for each job. RWPs will describe the job to be performed, outline tasks with elevated dose potentials and significant radiological hazards, define protective clothing and equipment to be used, and identify personnel monitoring requirements. RWPs will also specify any special instructions or precautions pertinent to radiation hazards in the area, including listing the radiological hazards present; the area dose rates, and the presence and intensity of hot spots; removable surface radioactivity; and other hazards as appropriate.

The subsections below describe the contractor's oversight of the decommissioning operations. This arrangement serves to minimize administrative functions, keep overhead costs to a practical minimum, and provide maximum flexibility for resource allocation.

6.3 Decommissioning Management Positions and Qualifications

Responsibilities of key individuals are summarized in the following subsections. Additional staff, along with applicable subcontractors may be utilized as appropriate.

6.3.1 USDA Radiation Safety Officer

The USDA Radiation Safety Officer, Mr. John Jensen, and his management, have overall responsibility for decommissioning of the LLRBS. Mr. Jensen is responsible for:

- Identifying funding needs to USDA upper management;
- Ensuring activities conducted under the USDA license are in compliance with applicable standards and requirements;
- Coordinating communication among the USDA and contractor(s);
- Coordination with the NRC;
- Overseeing contractor activities.

6.3.2 Contractor Project Manager

The Contractor Project Manager (PM) is responsible for ensuring that all necessary resources are provided to the project for its successful completion. The PM is the main point of contact for all questions, requests, and other information requested of the contractor. The PM reports to the contractor's higher management level. The PM is responsible for:

- Reviewing and approving project documents;
- Reviewing and approving schedules and work activities;
- Coordinating communication among the contractor and USDA;
- Reviewing proposed project methodologies to ensure they serve the data quality needs of the project;
- Serving as official contact for quality assurance (QA) and quality control (QC) matters pertaining to the project;
- Obtaining approval for proposed major changes to the Work Plan and other critical project documents;
- Establishing a project record system.

6.3.3 Site Supervisor

The Site Supervisor is responsible for the day-to-day operations for all construction work activities. Should the PM determine that additional construction management personnel are required, a field operations leader (FOL) will be assigned to the site. The FOL will take direction from the Site Supervisor and act as the line manager in the supervision of field personnel.

The Site Supervisor reports to the PM and is responsible for:

- Ensuring that all work is conducted in accordance with the SSHP,
- Reviewing on a daily basis all QC documents generated from field activities;
- Transmitting data generated to the Project Technical Support Team on a daily basis;
- Preparing and reviewing project documents;
- Conducting daily tailgate safety meetings and maintaining attendance logs and records;
- Assigning duties to project staff and orienting the staff to the needs and requirements of this survey;
- Supervising project team performance and day-to-day field operations;
- Reviewing major project deliverables for technical accuracy and completeness prior to their release;
- Ensuring field personnel receive necessary training on the requirements of the Work Plan, SSHP, and other project documents, as well as applicable regulatory issues;
- Routinely communicating project status, progress, and/or problems to the PM;
- Proactively identifying and responding to QA/QC needs.

6.3.4 Certified Waste Broker, TBD

The FSC-Certified Waste Broker (CWB) reports directly to the PM and is responsible for:

- Developing the waste profiles for the disposal site(s) of choice, based on sample data generated at the site;
- Ensuring all generated waste materials are inspected and properly prepared for shipment as per applicable regulations;
- Manifesting the wastes;
- Obtaining generator certification from proper BARC/ARS personnel;
- Procuring all necessary permits;
- Ensuring that the waste hauler is properly certified;
- Witnessing all waste shipments from the site; and
- Acting as support oversight for waste disposal activities.

6.3.5 Field Technicians (Equipment Operator, Groundwater Sampling Lead, and Health Physics Technician)

The contractor will provide qualified, cross-trained field technicians to conduct the decommissioning activities at the LLRBS. Contractor technical staff will be experienced professionals possessing the expertise and technical competence to perform field work effectively and efficiently.

6.4 Contractor Radiation Safety Officer

The Contractor Radiation Safety Officer (CRSO) reports directly to the PM and is responsible for ensuring that radiation health and safety procedures designed to protect personnel and the public is maintained throughout the project. The CRSO establishes radiological areas, monitors radiation exposure levels, and inspects all material/equipment entering or leaving the project for compliance with the SSHP and other applicable regulations. The CRSO has authority to direct such activities, to stop and restart work if necessary, and to take appropriate actions, as required, in emergency situations. For this project the SSHO may also serve as the CRSO. The CRSO is responsible for the execution of the routine on-site duties for health and safety that include:

- Conducting periodic safety reviews of the project site and project documentation;
- Performing regular and frequent site inspections to identify hazards and observe employees at work;
- Ensuring radiological instruments and test equipment are properly calibrated and response checked.
- Determining emergency evacuation routes, establishing and posting local emergency telephone numbers, and arranging emergency transportation;
- Ensuring that all site personnel and visitors have received the proper training and medical clearance prior to entering the site;
- Establishing any necessary controlled work areas;
- Assisting during daily tailgate safety meetings;
- Discussing potential health and safety hazards with the Site Supervisor and the PM;
- Implementing air monitoring in the event that it becomes necessary and forwarding all employee exposure monitoring information to the contractor corporate office to enable exposure notification;
- Implementing the field elements of the contractor Respiratory Protection Program, if required;
- Maintaining decontamination procedures that meet established criteria.

6.5 Training

Personnel working on LLRBS decommissioning activities with unescorted access to the facility will be trained in regard to the type and magnitude of the radiological, chemical, and physical hazards they might face (all visitors to the site will be escorted). The following subsections briefly describe the various training programs that will be implemented by the CRSO or his/her designee as part of this Decommissioning Plan.

6.5.1 Visitor Training

Visitors to the work zone will be escorted while in the LLRBS work area. Visitors shall not be allowed to enter airborne areas where exposure to internal dosage may occur without proper training.

6.5.2 General Employee Training

General Employee Training (GET) in radiation protection will be administered to project personnel who will provide support during implementation of this Plan. GET, to be provided at the start of fieldwork, will consist of an oral presentation by the CRSO, handout materials, and completion of a form acknowledging receipt of training. GET will address the following topics:

- The type and form of radioactive material present at the facility.
- The location of the contractor’s radiation protection policies and procedures.
- Employee and management responsibilities for radiation safety.
- Identification of radiation postings and barriers.
- Protective equipment and procedures.
- Work zone setup and decontamination procedures.
- Emergency procedures.
- How to contact contractor representatives and project radiation safety staff.

6.5.3 Hazardous Waste Operations Training

Project personnel who work in controlled areas with the potential for exposure to hazardous materials are required to undergo Hazardous Waste Site Operations (HAZWOPER) training in accordance with 10 CFR 1910.120 and contractor requirements.

Although not planned, it may be necessary for workers to enter an excavated pit. Therefore, site workers will be properly trained according to the Permit-Required Confined Spaces standard, 29 CFR 1910.146.

Additional training for site workers will include: Hazard communication (1910.1200), Personal Protective Equipment (29 CFR 1910.132), and Bloodborne Pathogens (29 CFR 1910.1030).

6.5.4 Radiation Worker Training

Project personnel who will work in radiologically controlled areas are required to undergo Radiation Worker Training (RWT) prior to arrival at the site. The contractor RWT will include the following topics at a minimum:

- Radioactivity and radioactive decay,
- Characteristics of ionizing radiation,
- Man made radiation sources,
- Effects of exposure to radiation,

- Risks associated with occupational radiation exposures,
- Special considerations with respect to exposure of women of reproductive age,
- Dose-equivalent limits,
- Modes of exposure (internal and external),
- Dose-equivalent determinations,
- Basic protective measures (time, distance, and shielding),
- Specific procedures for maintaining exposures ALARA,
- Radiation survey instrumentation (calibration, use, and limitations),
- Radiation monitoring programs and procedures,
- Contamination control, including protective clothing, equipment, and workplace design,
- Personnel decontamination,
- Emergency procedures,
- Warning signs, labels, and alarms, and
- Responsibilities of employees and management.

RWT will consist of a classroom lecture and procedure review, a 2-hour practical demonstration, a question/answer period, and a handout. The duration of training is approximately 4 hours. A graded exam to test employee proficiency in the class subject matter shall be administered. A passing score of 80 percent is required. A challenge test may be administered in lieu of a formal classroom training session for individuals previously trained by the contractor or demonstrating knowledge regarding radiological hazards expected to be present on-site.

6.5.5 Tailgate Safety Training

A tailgate safety meeting will be conducted at the beginning of each work shift, whenever significant changes in job scope are made, whenever significant changes in site conditions (physical or radiological) occur, or whenever new personnel arrive at the job site. Health and safety procedures and issues for the day, any unique hazards associated with an activity, and a review of any significant topics from previous activities will be presented at this meeting. The information discussed will be recorded, which will serve as confirmation that the information was presented to those persons whose signatures are on the form. There will be at least one signed form for each work shift. Tailgate safety training forms will be incorporated into the decommissioning records.

6.5.6 Training Records

A form will be developed to demonstrate that training commitments have been met. This form will include the following information: the facility name; the date; the time; the task number; the type of work; the hazardous/radioactive materials used; the protective clothing/equipment used; the chemical, radiological, and physical hazards; emergency procedures; the hospital's/clinic's telephone numbers; the paramedic's telephone number; the hospital's address; any special equipment needed; and any other safety topics that may be relevant. All training records will be incorporated into the decommissioning records. A contractor Notice to Employees, a copy of the

NRC Form 3 Notice to Employees, and a copy of the USDA NRC Materials License will be posted in plain view at the worksite.

An NRC Form 4 will be requested for all individuals requiring radiation-monitoring dosimetry.

6.6 Contractor Support

Contractor efforts will be focused on nuclear, health and safety, regulatory compliance, and project management matters. Specialty services necessary to complete certain aspects of the Plan (e.g., disposal, treatment, transportation, and laboratory analyses) may be subcontracted to firms with the appropriate skills and experience.

Each subcontractor will designate a Task Manager and, as necessary, a health and safety and/or quality control contact, who will report to the Task Manager. At all times, however, the contractor will remain responsible for the scope, quality, and timeliness of services provided by all subcontractors. The CRSO will verify that the subcontractor personnel are adequately informed of the hazards, the preventive measures, and the procedures associated with performing each decommissioning task. The CRSO will verify that subcontractor personnel perform decommissioning activities in accordance with all license commitments and NRC requirements.

7.0 HEALTH AND SAFETY PROGRAM DURING DECOMMISSIONING

Decommissioning activities will be completed in a manner that is protective to workers, the environment, and the public. It is USDA's policy to maintain minimal human and environmental exposure to known or suspected radioactive and/or hazardous material. The contractor will accomplish this by following the guidance of standard operating procedures (SOPs), encompassing both administrative procedures (APs) and operating procedures (OPs) presented in the contractor's NRC approved RSP. Administrative procedures will address, at a minimum:

- Record Retention,
- Radiological Conditions Awareness Reports,
- Radiological Compliance Audits,
- ALARA,
- Bioassay,
- Dosimetry Program, and
- Radiological Training.

The RSP may also include additional administrative and operational procedures that will be used by contractor management to guide the conduct of all relevant decommissioning activities. Copies of procedures pertinent to decommissioning activities will be maintained onsite for reference and regulatory review. Deviations from procedures must have prior approval of the CRSO.

The contractor will provide a workplace in which employees, visitors, and contractors are adequately protected from hazards, including the hazards associated with exposure to radiation and radioactive material. While the expected exposures associated with the planned decommissioning operations are low, all exposures are assumed to entail some risk to employees, visitors, and contractors. The ALARA requirement will be communicated to all subcontractors at the outset of this project. All individuals must understand their responsibilities to reduce their radiation exposure. Methods to be used to reduce exposure will be reviewed during initial site-specific training and tailgate meetings. Monitoring and surveillance information will be summarized and reviewed by the workforce on a planned and periodic basis.

A site-specific HASP has been developed to describe the practices to reduce employee exposure to potentially present radioactive materials and hazardous chemicals, as well as construction safety concerns. The HASP will remain in effect during all on-site decommissioning activities. The contractor will maintain documentation sufficient to demonstrate the effectiveness of the health and safety program. The Health and Safety Officer (HSO), or designee, operating under the direction of the CRSO, and pursuant to the requirements of the license, will monitor on-site health and safety. As necessary, the CRSO, or designee, will conduct tailgate safety training, implement the surveillance and individual monitoring programs, perform release surveys for personnel and equipment during decommissioning operations, and maintain all health and safety records generated during the decommissioning efforts.

7.1 Radiation Safety Controls and Monitoring for Workers

The contractor's RSP shall be implemented to control exposure to ionizing radiation through approved SOPs. The SOPs reference and provide instructions to ensure compliance with applicable federal regulatory documents including 10 CFR Parts 19 and 20, and ensure that no occupational exposure limits set forth in 10 CFR Part 20 are exceeded. Radioactive materials and sources of radiation will be controlled in such a manner that radiation exposures to workers do not exceed limits specified in 10 CFR Part 20, Subpart C.

Radiation safety personnel assigned to the project will assess the effectiveness of posted warning signs during the conduct of these surveys. Surveys will be conducted using survey instrumentation and equipment suitable for the nature and range of hazards anticipated. Equipment and instrumentation will be calibrated, and where applicable, operationally tested prior to use in accordance with procedural requirements. Routine surveys will be conducted at a specified frequency to ensure that contamination and radiation levels in unrestricted areas do not exceed license, or federal, state, or site limits. The CRSO or designee will also perform surveys during decommissioning whenever work activities create a potential to impact radiological conditions.

As required in 10 CFR 20.1502, the need for individual monitoring for internal and external exposures will be determined and documented prior to the start of work based on existing data. Potential exposures to personnel working at the site during decommissioning include direct contact (e.g., ingestion exposure pathway) and airborne dusts that may be contaminated (inhalation exposure pathway). Personnel will perform routine monitoring for radioactive contamination to minimize the spread of contamination and consequently the ingestion pathway.

Radiation monitoring (external and/or internal) shall be conducted continuously when it is likely that any individual will exceed 10 percent of the annual limit. If air samples detect the presence of radiation in excess of 10 percent of the derived air concentration (DAC), the CRSO will evaluate the need for a bioassay program. Occupationally exposed workers who have received radiation exposure prior to employment with the contractor are required to provide their radiation exposure history records or names and addresses of previous employers and locations where they have received exposures. Copies of this letter will be sent to the individual, and maintained in the individual's personnel radiological exposure file.

7.2 Air Sampling Program

Radiological air sampling surveys and monitoring will be performed in accordance with written procedures, and in compliance with 10 CFR Parts 20.1204 and 20.1501(a)-(b). Air samples will be collected under known physical conditions (e.g., sample time, flow rate). The flow meters of air samplers will be calibrated prior to each mobilization and following repair and/or modification.

Both breathing zone and general area air samples will be collected from areas where there is the potential for generation of airborne radioactive material. Breathing zone air samples will be the primary method of monitoring the worker's intake of radioactive material and will be collected from the workers' breathing zones at work locations most expected to be known or suspected release points. General area air samples will also be collected from general and localized areas, especially concentrated on areas downwind from excavation and other areas with the greatest likelihood of the presence of airborne dust. Appropriate air sampling equipment will be selected.

The type of sampling that is desired will determine the appropriate collection media required to collect the contaminant. The frequency at which air filters will be changed will be determined based on the radiological and physical condition of the work location, worker stay times, and the type of air sampling performed.

Air sampling will be performed prior to initiating construction activities in order to document background radioactive airborne particulate activities. Air sampling will be performed to monitor airborne particulate activity when excavation activities commence, routinely during decommissioning activities, and after any significant changes in operating conditions. Sampling durations will be determined prior to the commencement of sample collection based on required action levels, counting instrument sensitivities, and other conditions as warranted.

Following air sample collection, the filter media will be stored for at least 24 hours in order for short-lived radon progeny to be allowed to decay. Air samples will then be counted with sufficient time to achieve required minimum detectable concentration (MDC) goals for each specific radionuclide. Air sample analysis results will be compared with the DAC for radionuclides. Breathing zone air samples are collected using personal lapel (or equivalent) air samplers or grab samplers. If the breathing zone concentration exceeds 10% of DAC values, the CRSO should be notified so appropriate actions can be taken and exposures received by workers evaluated and included in their personal exposure file.

If gross alpha activity significantly in excess of (i.e., three times greater than) background is identified, then the air samples will be shipped to an accredited offsite analytical laboratory for determination of the presence of thorium and uranium isotopes.

7.3 Respiratory Protection Program

Respiratory protection shall be maintained by the application of practicable engineering controls such as process, containment, and ventilation equipment and the concurrent monitoring of airborne dusts. Maximum dust loading will be assumed for ^{3}H and other low-energy beta emitting radionuclides.

Emergency conditions are unplanned events characterized by the need for rapid and aggressive actions to prevent or mitigate the effects of rapidly deteriorating conditions. The use of respirators during such is often a reasonable substitute for engineering controls that must be assumed to be nonfunctional or ineffective. There exists an extremely low probability of the occurrence of an emergency event of this nature during LLRBS decommissioning activities.

7.4 Internal Exposure Determination

Internal exposure determination will be determined through analysis of breathing zone air samples in compliance with written procedure(s) and, as necessary, bioassay results. The assessment of a workers' Committed Effective Dose Equivalent (CEDE) will be limited to less than 10% of the allowable limit on intake (ALI) as specified in Table I, Columns 1 and 2, of Appendix B of 10 CFR Part 20, providing the total effective dose to the individual is maintained ALARA. The CRSO will determine the validity of bioassay and air monitoring results prior to their inclusion in the internal dose assessment process.

7.5 External Exposure Determination

External exposure potential will be routinely monitored through the use of microR, or equivalent meters to assess the level of external exposures to ionizing radiation. If it is determined that personnel may likely receive within one year, a dose in excess of 10% of the applicable limits from radiation sources external to the body, will be monitored by personnel dosimetry such as themoluminescent dosimeters. The personnel dosimetry devices will indicate the amount of ionizing radiation to which the wearer was exposed. The personnel dosimeter will normally be worn on the upper front torso. Personnel are responsible to wear dosimetry as directed by the CRSO. If a personnel dosimeter is lost, misplaced, or indicates an off-scale reading, the employee is required to notify their supervisor, health physics and/or the CRSO immediately.

All reasonable efforts will be made to keep ionizing radiation exposure to the unborn child to the lowest practical level, as prescribed in 10 CFR 20.1208. Once a female employee determines that she is pregnant, she is encouraged to notify in writing of her pregnancy. The contractor will then institute radiation control measures that will limit radiation exposure to the unborn fetus to less than 500 millirem (mrem) for the term of the pregnancy and below 50 mrem per month in any month after declaration.

7.6 Summation of Internal and External Exposures

If the CRSO makes a determination in accordance with 10 CFR 1201(a) that external dosimetry is not required (i.e., personnel are not likely to receive within one year, a dose in excess of 10 percent of the applicable limits from radiation sources external to the body), then the external dose need not be added to the internal dose.

If the CRSO makes a determination in accordance with Appendix B, 10 CFR Part 20 that internal dosimetry is not required (i.e., personnel are not likely to receive within one year a dose in excess of 10 percent of the applicable ALI as specified in Table I, Columns 1 and 2, of Appendix B of 10 CFR Part 20), then the internal dose need not be added to the external dose.

7.7 Contamination Control Program

Radioactive material will be controlled as specified in the project HASP and in such a manner that the surface contamination does not exceed the levels specified in NRC guidelines for the decontamination of facilities and equipment prior to release for unrestricted use as presented in the NRC’s Policy and Guidance Directive FC 83-23 (NRC, 1983). These limits are presented in Table 6.

Routine surveys will be performed throughout the decommissioning process, with each survey being planned in advance with regard to the specific radiation type, the predetermined radiation levels, the location where radiation is expected, and any special condition warranting a survey. The initial level of protection for the intrusive tasks of this decommissioning operation (i.e., where residual radioactivity may be encountered) will be Level D modified PPE, including hard hats, Tyvek© coveralls, safety glasses with side shields, steel-toed boots, and gloves. Upgrading or downgrading the level of protection will be based on ambient conditions as work proceeds. The CRSO will determine if it is necessary to upgrade to a higher level of protection.

To ensure that radioactive materials remain under the control of the contractor, each worker involved in this decommissioning effort and working in a contaminated area will be frisked using calibrated, hand-held instruments prior to leaving the contaminated work area. Equipment and materials will be frisked and decontaminated, as necessary, prior to exiting the controlled area. Records of release surveys will be maintained on standardized forms and maps and will be placed in the decommissioning records. Release criteria will be consistent with those contained in the RSP. In the event that a sealed radioactive source is used at the site, the CRSO will verify the conditions of the license, which regulates the use of the sealed source. This will include verifying the training of the operators and the frequency of wipe tests.

Table 6 : Acceptable Surface Contamination Levels

Radionuclide[a]	**Acceptable Surface Contamination Levels ($dpm/100\ cm^2$)**		
	Removable [b,c]	**Average** [d]	**Maximum** [e]
Transuranics, ^{226}Ra, ^{228}Ra, ^{230}Th, ^{228}Th, ^{231}Pa, ^{227}Ac, ^{125}I, ^{129}I	20	100	300
Th-natural, ^{232}Th, ^{90}Sr, ^{223}Ra, ^{224}Ra, ^{232}U, ^{126}I, ^{131}I, ^{133}I	200	1,000	3,000
U-natural, ^{235}U, ^{238}U, and associated decay products	1,000 α	5,000 α	15,000 α
Beta-gamma emitters (radionuclides with decay modes other than alpha emission or spontaneous fission) except ^{90}Sr and others noted above.	1,000 β-γ	5,000 β-γ	15,000 β-γ

Footnote

[a] Where surface contamination by both alpha- and beta-gamma emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides should apply independently Note

[b] As used in this table, dpm (disintegrations per minute) 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] 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 paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionately and the entire surface should be wiped.

[d] Measurements of average contaminant should not be averaged over more than 1 m^2. For objects of less surface area, the average should be derived for each object.

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

(A) Exposure Control:

Application of engineering, administrative, and personnel protection provisions will control personnel exposure to radioactive material. The priority of application will be descending with respect to their order of description below.

1. Engineering - Engineering controls will be used, as practicable, to minimize or prevent the presence of uncontained radioactive material. Engineering controls will predominantly be comprised of containment, isolation, ventilation, and decontamination.

2. Administrative - Administrative controls will be used to control work conditions and work practices. Administrative controls will predominantly be comprised of the following:

    - Access Control: Routine access to work areas will be limited to personnel necessary to accomplish tasks or activities. Access will also be controlled with respect to training and use of specified personnel protection equipment.
    - Postings and Barriers: Postings will be used to inform personnel of relevant hazards or conditions and associated access requirements. Barriers may be used to prevent unauthorized access.
    - Procedures: Written procedures may be used to describe specific radiation protection requirements necessary for tasks that involve radioactive material.
    - Radiation Work Permits: RWPs will be used to describe specific or special worker protection requirements for activities involving radioactive material and not covered by a procedure. RWPs may also be used in conjunction with a procedure.
    - Contamination Control: Action levels and limits for radiation surveys, described later in this section, will be used to control the levels of radioactivity on equipment and in areas.

3. Personal Protective Equipment - PPE will be used to control personnel exposure to radioactive material when administrative controls are not sufficient and engineering controls are not practicable. PPE may include head covering, eye protection, respiratory protection, impervious outerwear, gloves, and/or protective shoes or shoe covers.

(B) **Radiation Surveys:**

Radiation surveys will be performed to describe the radiation types and levels in an area or during a task, to identify or quantify radioactive material, and to evaluate potential and known radiological hazards.

The types of radiation surveys and their frequency are described in the following subsections.

1. Contamination Measurements - Measurements will be made of removable alpha, beta, and beta-gamma radiation, as applicable. The measurements will be made by wiping an area with cloth, paper, or tape. The radiation levels will be measured on the wipe. Contamination surveys shall be performed at the end of each workday where invasive demolition of contaminated surfaces was performed.

2. Radiation - Exposure rate measurements will be performed using an ion chamber or equivalent. Measurements will be made at approximately 30 cm. Measurements may also be made at contact.

3. Personnel - Personnel will be frisked prior to leaving access-controlled areas.

4. Action Levels - Action levels are established to inform facility personnel when a situation needs to be evaluated so that corrective actions can be taken. Action levels are set so that corrective actions can be made before a regulatory limit is exceeded. Based on

information collected during waste characterization survey (CABRERA 2007), action levels as outlined in regulatory requirements (e.g. 10 CFR 20) will be sufficient to ensure worker protection.

In cases where an action level is exceeded, an investigation will be conducted that includes evaluation of preventative and/or corrective action. The investigation, and documentation of such, is completed commensurate with the significance of the condition.

7.8 Instrumentation Program

Instruments used for radiation detection and measurement will be operated in compliance with contractor RSP SOPs. SOPs contain instructions on the proper use of the instrument, as well as calibration instructions for those instruments, which are calibrated by a certified calibration facility. Radiation detection instruments are calibrated in manner and frequency as per license and manufacturer requirements and after each repair that would affect the accuracy of the instrument. Only personnel trained in accordance with the procedures will use radiation detection instruments. A calibration sticker shall be attached to the instrument to allow the operator to verify the instrument is within current calibration prior to use. Health physics instruments shall be visually inspected, battery checked, and source checked prior to use. Radiation survey equipment and instrumentation suitable for detecting and quantifying the radiological hazards to workers and the public will be present on-site throughout decommissioning activities. The selection of equipment and instrumentation to be utilized will be based upon knowledge of the radiological contaminants; concentrations, chemical forms, and chemical behaviors that are expected to exist as demonstrated during radiological characterization activities. Equipment and instrumentation selection will also take into account the working conditions, contamination levels, and source terms encountered during the performance of decommissioning work, as presented in this decommissioning plan. In all cases, the program will be consistent with the requirements set forth in the contractor RSP.

7.9 Nuclear Criticality Safety

Available historical records show that radioactive materials buried within the LLRBS are not expected to meet the definition of Special Nuclear Material (SNM) per 10 CFR 70.4. Therefore, nuclear criticality safety measures will not be necessary.

7.10 Health Physics Audits, Inspections, and Recordkeeping Program

The RSP shall be subject to an annual audit and periodic inspections. Each are performed to determine if radiological operations are being conducted in accordance with regulations, license conditions, and written procedures.

An audit of the program shall be conducted annually. The audit shall be conducted by the CRSO or designee, but shall not be a member of the contractor organization. The audit will consider the basic functional areas of the program; e.g., RWPs, Radiation Protection Procedures, radiological surveys and air monitoring, ALARA program, individual and area monitoring results, access controls, respiratory protection program, and training.

The audit shall be conducted in accordance with a specific audit plan developed by the auditor. A written report describing the results shall be generated upon completion of the audit. The report shall be distributed to site management. As necessary, a written corrective action plan shall be prepared to address non-compliance issues. All corrective actions shall be tracked to completion. Once corrective actions have been completed, a written closure report shall be distributed to management documenting the completion of corrective actions.

The Health and Safety staff shall conduct the periodic inspections. These inspections shall be routine reviews performed of operations and activities. The inspections shall normally be completed against a pre-established checklist. Checklists may be developed independently for differing periods; e.g., daily, weekly, monthly, etc. The checklist items shall usually be comprised of routine procedural requirements. Any findings discovered during the routine inspection shall be recorded on a tracking log. The log shall be maintained by the CRSO. The log shall include a description of planned corrective action and date of completion of corrective action.

7.10.1 Personnel Records

A personnel file is maintained for each employee assigned work duties involving radioactive materials. The content of these files include:

- A record of radiation exposure received by the individual during previous employment is maintained by requesting personal exposure information from previous employers where the individual worked with radioactive materials.
- A record of personnel dosimeter measurements is recorded in the personnel file to provide a permanent record of radiation exposure received during the course of work assignments.
- If a personal dosimeter is lost or damaged, an exposure investigation will be performed and an exposure will be assigned for the monitoring period. A report detailing the exposure estimate will be included in the personnel record.
- If the air concentration in the work area exceeds 10% of DAC values, air samples and bioassay samples will be used to estimate internal exposures received by the worker and included into their personal exposure file.
- If a worker finds contamination on their person above the limits specified in Table I, a report of the incident will be placed in the personnel file to determine exposure from the incident.

- The personnel records will be maintained indefinitely and personnel may review their file or request copies of information within their files. The licensee for which work is performed will be provided individual exposure information as required by their license or applicable regulations.

7.10.2 Radiation and Contamination Records

Radiation and contamination survey records collected during site surveys, remediation/decontamination activities, and other decommissioning activities shall be stored in project-specific files at the contractor office. Duplicate copies of the records are also supplied to the licensee where the work was performed.

7.10.3 Records of Waste Disposal

Radiation Survey Records, contamination survey records, shipping manifests, and certifications generated for a licensee's shipment of radioactive materials to a licensed disposal site shall be stored in specific shipment files in contractor office. Duplicate copies of the records are supplied to the licensee for the work performed.

8.0 ENVIRONMENTAL MONITORING AND CONTROL PROGRAM

USDA management and personnel are committed to maintaining exposure of ionizing radiation to workers, the public, and to the environment at ALARA levels and will strive to conduct decommissioning activities in a manner that supports this commitment. The contractor will ensure the minimization of the impact of ionizing radiation to human health and the environment through the use of NRC approved RSP SOPs. Environmental monitoring and control activities performed during decommissioning activities will comply with 10 CFR Part 20 regulatory requirements.

8.1 Effluent Monitoring Program

Concentrations of radionuclides in site effluents are not expected to change as a result of decommissioning activities.

The primary routes of contaminant transport during the on-site decommissioning activities are anticipated to be airborne dust from the excavation of the site, handling of the waste, covering operations, and from the movement of vehicles and equipment. Area air samples will be collected in locations that present the possibility of airborne effluent releases. In addition, samplers will be positioned downwind of work locations to ensure that the samples collected within the immediate work area are representative of actual releases. The positions of the air samplers will be evaluated frequently by the CRSO to take into account any shifts in prevailing wind direction and any movement in the locations of dust-generating operations. Worker lapel and general area air monitoring will be performed daily, following SOPs, and air samples will be analyzed in order to estimate inhalation dose to the public and workers by airborne radioactive material. Consideration will be given to more frequent filter change-outs during high dust conditions, as determined using best professional judgment. Air samples will be analyzed onsite for gross alpha and gross beta emitters and sent to an accredited offsite laboratory to be analyzed for low-energy beta emitters. Releases will be maintained ALARA and below the limits in Table 2 of Appendix B to 10 CFR Part 20. Dust monitoring will be performed along with air sampling to provide immediate results for the airborne activity of low-energy beta emitters through the assumption of maximum dust loading.

Background samples will also be collected prior to the commencement of site activities in order to establish baseline background radionuclide concentrations.

Significant amounts of liquid effluents are not expected to be encountered or generated during decommissioning activities.

8.2 Effluent Control Program

Based on the results of the characterization survey, decommissioning activities are not expected to generate significant levels of airborne particulate contamination. If significant dust is generated during decommissioning activities, controls will be implemented to moisten excavation areas as necessary in order to reduce the potential for generating airborne contamination. Any soil or similar material that is staged in piles, containers, or vehicles will be covered as practical to prevent dispersion by wind and precipitation.

If radiological air monitoring results indicate the presence of airborne contaminants exceeding project action levels, then work will be stopped, proper personnel including the CRSO will be informed, corrective measures will be implemented, and dose evaluations performed, as necessary.

During performance of previous waste characterization activities (CABRERA 2007) real time particulate air monitoring did not exceed the action level of 0.3 milligrams per cubic meter (mg/m^3).

During characterization survey activities, leachate was encountered in the excavated burial pits. This water was containerized and stored in three 275-gallon tanks. If potentially contaminated liquid effluents are encountered, runoff will be controlled through the use of berms, silt fencing, absorbent materials, solidifying agents, or by other means as necessary. Any captured liquids will be collected and stored for disposition in accordance with applicable regulations. A Sediment and Erosion Control Plan will be used to plan for the prevention of runoff.

9.0 RADIOACTIVE WASTE MANAGEMENT PROGRAM

9.1 Solid Radioactive Waste

Materials requiring disposal will include miscellaneous items, i.e., glass, plastic, syringes, decomposed animal carcasses, debris, and soil. The estimated volume of material requiring disposal is approximately 33,000 cubic feet.

Waste materials exhumed from the pits will be transferred to a designated storage area in the South field of the LLRBS to be managed and sampled prior to disposition. In this area, materials will be segregated by media type and sampled for waste profiling purposes. Once sampling results are available, materials will be packaged into DOT compliant containers for transport to the appropriate treatment and/or disposal facilities. Packaged waste will be staged while awaiting transport unless bulk containers, such as IP1/IP2 intermodals are used.

- **Radioactive Waste, sources.** Segregation operations uncovered three radioactive sources/devices, including two ^{226}Ra sources and a ^{63}Ni electron capture device source.
- **Radioactive waste, soil and debris.** The investigation generated more than 200 containers (drums and boxes) of soil and debris. The majority of the wastes generated were either mixed wastes or hazardous waste due to organic contaminants found throughout the soil column. Their presence was detected in all the soil and debris wastes removed from disposal cells that contained vials or laboratory reagent wastes.

9.2 Liquid Radioactive Waste

During the WCS, liquid wastes were encountered intact in their disposal containers. Such wastes were disposed of in 1- to 5-gallon containers, solvent bottles (1 liter), and scintillation vials (sometimes packed in 55-gallon drums). Liquids were composited, and packaged for offsite treatment/disposal in accordance with DOT requirements.

It is probable that additional liquids will be discovered during decommissioning activities. These liquids will be composited, and packaged for offsite treatment/disposal in accordance with DOT requirements. Packaged waste will be staged indoors while awaiting transport. Appropriate containment devices will be used to ensure potential spills are contained.

No surface water exists within the LLRBS site and it is expected that no groundwater will be encountered during excavation activities. However, during the WCS, leachate was encountered in one excavation (Pit 34). Steps will be taken during the excavation of the site (e.g. covering the excavated areas with tarps to keep rain water from collecting). Accordingly, any water accumulated during future decommissioning activities will be handled, stored, and dispositioned in accordance with appropriate procedures and regulatory requirements.

9.3 Mixed Waste

Decommissioning activities may result in the generation of mixed waste and subsequent need for treatment prior to disposal. Mixed waste meeting the proper acceptance criteria will be transferred to appropriate facilities for treatment and disposal. Packaged waste will be staged indoors while awaiting transport.

- **Mixed Waste, Liquid Scintillation vials (LSVs).** LSVs are commonly generated in research. There are established treatment and disposal outlets for them. The investigation generated 28 drums of LSVs.
- **Mixed Waste, Liquids bulked from various containers.** Excavations uncovered many laboratory reagent bottles usually in the form of one gallon or one liter amber bottles. Contents were tested for compatibility and bulked into plastic 55-gallon drums. The investigation generated two drums of bulked liquids that were shipped offsite for treatment and disposal as mixed waste.
- **Mixed Waste, F039 liquid (multisource leachate, water) pumped from the excavations.** Removal of water from the excavations generated three 275-gallon containers of F039 waste. The liquid also exhibits low levels of radioactive material contamination and is therefore classified as a mixed waste.
- **Mixed Waste, solid, both debris and soil-like material.**

10.0 QUALITY ASSURANCE PROGRAM

10.1 Organization

Only qualified and trained personnel will operate the equipment and instrumentation used in the field activities specified in this decommissioning plan. Personnel will be trained in the technical, quality control, and health and safety aspects of the project, as well as in the calibration, maintenance, and SOPs for their assigned equipment.

Daily tailgate safety meetings will provide supplemental training and ensure that personnel are given clear direction and the proper tools for performing their respective tasks. These meetings will also provide a forum for the field personnel to relate any potential safety or quality concerns that may require attention from the CRSO or PM. Tailgate meeting notes and attendance sheets will be maintained onsite and included in the project file.

Persons responsible for ensuring that the QA Program has been established and for verifying that activities affecting quality are being correctly performed will have sufficient authority, access to work areas, and organizational freedom to accomplish the following:

- Identify quality concerns
- Ensure that further decommissioning activities are controlled until proper resolution of a non-conformance or deficiency has occurred
- Initiate, recommend, or provide solutions to quality problems through designated channels
- Verify implementation of solutions

The Project Engineer will serve as QA Officer for the project and will have direct access to responsible management at a level at which appropriate corrective actions can be implemented, as necessary. Therefore, the onsite quality control representative will report to the PM, or designee, to ensure that required authority and organizational freedom are provided. The quality control representative may authorize others to implement specific elements of the QA Program.

10.2 Quality Assurance Program

Activities associated with this decommissioning plan shall be performed in accordance with written procedures and/or protocols in order to ensure consistent, repeatable results. Topics covered in project procedures and protocols may include proper use of instrumentation, quality control (QC) requirements, equipment limitation, etc. Implementation of quality assurance (QA) measures for this decommissioning plan is described in the sections below.

10.3 Document Control

Data will be recorded and documented in a data management system. The radiation survey maps will designate the location being surveyed, as well as the name of the surveyor. To the extent practical, state plane coordinates will be used to define the location of a soil sample. If not available, site-specific references will be used to locate a sample. Data management personnel will ensure that chain-of-custody (COC) and data management procedures are followed for samples related to the FSS. Procedures to properly handle, ship, and store samples after they are collected will follow established protocols.

Both direct radiation measurements and analytical results will be documented. The results-for each survey measurement and/or each sample will be listed in tabular form along with the corresponding grid block location or coordinate. Radiation survey data will be recorded in a verifiable manner and reviewed for accuracy and consistency. Each of the major phases of the decommissioning process will be documented in a manner that is suitable for audits or assessments.

Changes to the Decommissioning Plan and the proposed QAPP will be submitted to the NRC for review and approval before they are implemented. The records discussed in the preceding paragraphs will be maintained until the license termination.

10.4 Control of Measuring and Test Equipment

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

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

- Name of the equipment
- Equipment identification (model and serial number)
- Manufacturer
- Date of calibration
- Calibration due date

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

Prior to daily use, project instrumentation will be QC checked by comparing instrument response to a benchmark response. Prior to the commencement of field operations, site reference locations shall be selected for performance of these checks; subsequent QC checks will be performed at these locations. QC source checks will consist of a one-minute integrated count, or other count time designated by the Project HP, or designee, with the designated source positioned in a reproducible geometry, performed at the reference location. Prior to the start of initial surveys, this procedure will be repeated at least ten times to establish average instrument response. Equipment should also be inspected for physical damage, current calibration and erroneous readings in accordance with applicable procedures and/or protocols. Instrumentation that does not meet the specified requirements of calibration, inspection, or response check will be removed from operation.

10.5 Corrective Action

The QA Officer has overall responsibility for reporting all procedure and contract violations found. The CRSO will determine if the deficiency requires work to be stopped or if notification is required to the NRC.

A deficiency or nonconforming condition is documented on a Corrective Action Request (CAR) Form. The form is completed by the individual who reported the nonconformance and submitted to the QA Officer who will review the CAR for completeness. The completed form will provide a detailed description of the nonconforming condition and reference the affected documents that apply.

The QA Officer will review the response and verify that the actions address the original concern and provide effective preventive actions. If satisfactory, the QA Officer will accept the response and close the CAR. The person writing the CAR will sign the document. The QA Officer will review the form and maintain a log of all CARs, indicting the current status of the CAR. After corrective action has been completed and verified by the QA Officer, the closed CAR (original) will be filed. If needed, a new CAR will be issued to address additional required corrective action.

10.6 Quality Assurance Records

QA records will be monitored by the contractor. Data reduction, QC review, and reporting will be the responsibility of the analytical laboratory. Data reduction includes all automated and manual processes for reducing or organizing raw data generated by the laboratory. The laboratory will provide a data package for each set of analyses that will include a copy of the raw data in electronic format, and any other information needed to check and recalculate the analytical results. Once a data package is received from the laboratory, the analytical results and pertinent QC data will be entered into a computer database. The data packages will serve as basic reference sheets for data validation, as well as for project data use.

The generation, handling, computations, evaluation, and reporting of final radiological survey data will be as specified in written procedures. Included in these procedures will be a system for data review and validation to ensure consistency, thoroughness, and acceptability of the data. Some data points will be chosen for evaluation will be examined to determine compliance with QA requirements and other factors that determine the quality of the data. Any rejected sample data or data omissions identified during the data validation will be evaluated to determine their impact on the project. Other corrective action may include re-sampling and reanalyzing, evaluating and amending sampling and analytical procedures, and accepting data acknowledging the level of uncertainty.

One of the most important aspects of sample management is to ensure that the integrity of the sample is maintained, that is, that there is an accurate record of sample collection, transport, analysis, and disposal. This ensures that samples are neither lost nor tampered with, and that the sample analyzed in the laboratory is actually and verifiably the sample taken from a specific location in the field. The individual(s) responsible for sample collection will initiate a COC record using a standard form provided by the contractor. A copy of this form will accompany the samples throughout transportation and analyses; and any breach in custody or evidence of tampering will be documented.

10.7 Audits and Surveillances

Periodic audits will be performed by the Decommissioning PM, the CRSO, the HSO, and/or persons so designated to verify that decommissioning activities comply with established procedures and other aspects of the QAPP (such as scope, status, adequacy, and compliance) and to evaluate the overall effectiveness of the QA Program. The PM and QA Officer will verify that qualified personnel are employed to conduct audits to ensure that the applicable procedures are being properly implemented. The audits will be conducted on at least a quarterly basis, in accordance with written guidelines or checklists. Health and safety personnel will also conduct semiannual audits in their area of concern. External program audits may also be used at the discretion of the contractor.

Audit results will be reported to USDA in writing, and actions to resolve identified deficiencies will be tracked and appropriately documented. The audit information will become part of the decommissioning record for the site.

11.0 FACILITY RADIATION SURVEYS

11.1 Release Criteria

The radiological release criteria are discussed in Section 4.2.3.

11.2 Characterization Survey

A characterization survey was performed to quantify and confirm the presence of potentially elevated levels of ^{3}H, ^{14}C, ^{137}Cs, ^{36}Cl, ^{32}P, ^{63}N, ^{226}Ra, ^{210}Pb, ^{90}Sr, and other radionuclides at the LLRBS. A summary of these results are presented in section 3 of this document. Detailed results are contained in the "Low Level Radioactive Burial Site Characterization Survey" (CABRERA, 2007)

11.3 In-Process Surveys

Radiological support surveys will be performed during decommissioning activities to assist and guide excavation activities. In addition, surveys of waste material removed from the burial pits will be conducted.

11.4 Final Status Survey Design

The FSS was designed in accordance with MARSSIM guidance and applicable Federal and Maryland regulations and guidance, and is being performed to support the release of the LLRBS for unrestricted use. A background reference area will be established in a non-impacted area. The design for the FSS and establishment of radiological release criteria are presented in the Final Status Survey Plan (FSSP), presented as Appendix B.

11.5 Final Status Report

The Final Status Report will be prepared following decommissioning and FSS activities.

12.0 FINANCIAL ASSURANCE

12.1 Cost Estimate

Table 7 provides a detailed cost estimate for implementation of this DP. The estimated total cost is approximately $7 million. Costs are based on current estimates, and resulting contracts, for implementation of the Characterization Plan, which essentially involves remediation of one to four pits.

It should be noted that the cost estimate is largely dependent on the assumed mixture of LLRW and mixed waste. Mixed waste treatment/disposal costs are substantially higher than LLRW disposal costs.

12.2 Certification Statement/Financial Mechanism

As a federal agency, USDA will issue to the NRC a statement of intent to fund the required decommissioning actions.

Table 7: Summary of Estimated Decommissioning Costs

Task Description	Labor	Travel	Subcontractors	Other Direct Costs	Total
Mobilization	$ 14,600	$ 17,292	$ -	$ 13,600	$ 45,492
Pit Investigation/ Remediation/Survey	$ 363,000	$ 397,017	$ 119,680	$ 155,225	$ 1,034,922
Waste Segregation	$ 457,650	$ 352,501	$ -	$ 122,500	$ 932,651
Waste Packaging/ Transportation	$ 15,400	$ -	$ -	$ 843,500	$ 858,900
Backfill-Demobilization	$ 15,400	$ 11,337	$ 800	$ 92,000	$ 119,537
Waste Disposal- Mixed Waste- Not Treated	$ -	$ -	$ -	$ -	$ -
Waste Disposal- LSCs [treated]	$ 9,400	$ 4,064	$ 14,900	$ 790,200	$ 818,564
Waste Disposal - LLRW	$ 9,400	$ -	$ -	$ 3,209,500	$ 3,218,900
Final Status Survey	$ 21,600	$ 17,844	$ -	$ -	$ 39,444
Totals:	$ 906,450	$ 800,056	$ 135,380	$ 5,226,525	$ 7,068,411

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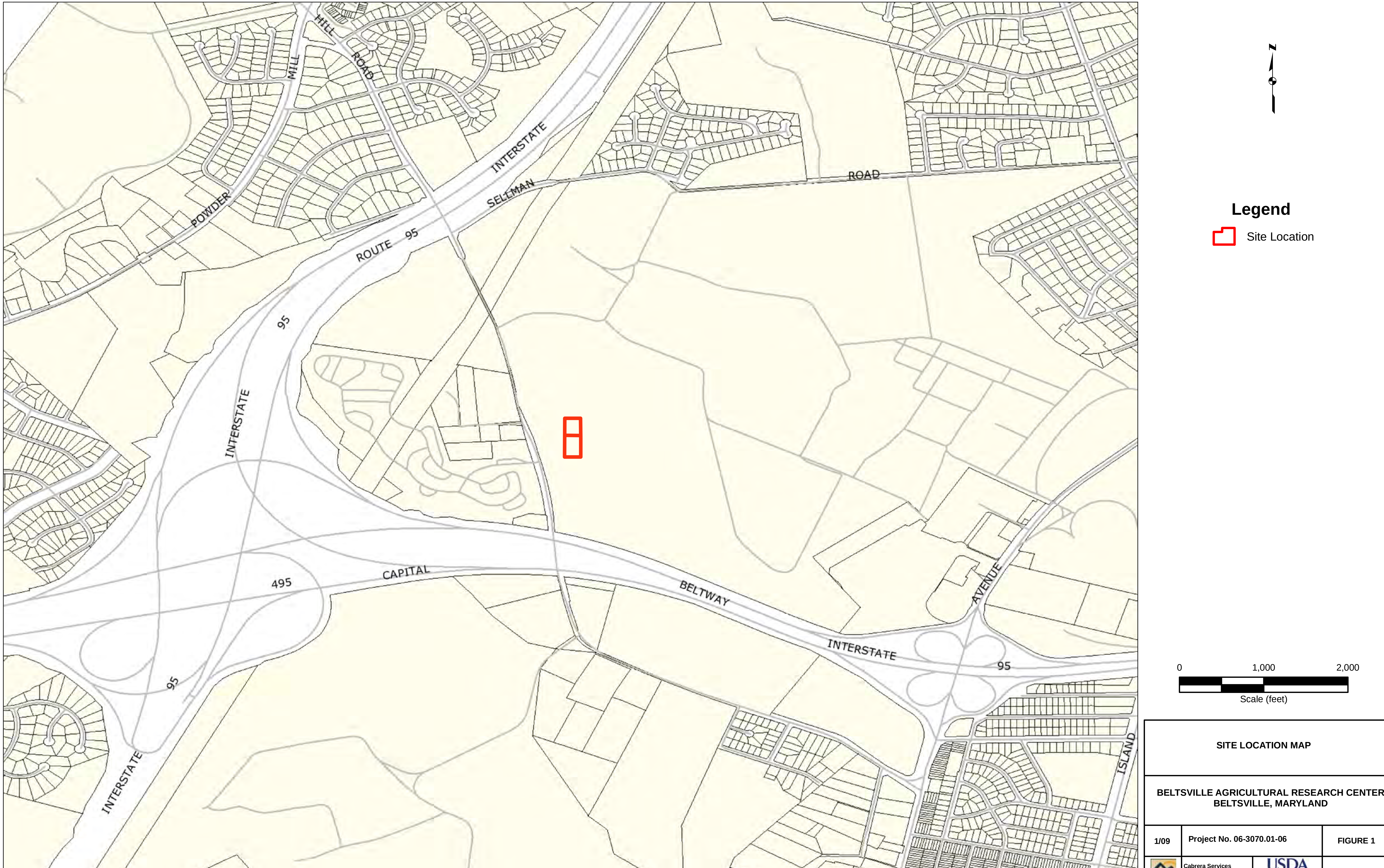
FIGURES



SITE LOCATION MAP

BELTSVILLE AGRICULTURAL RESEARCH CENTER
BELTSVILLE, MARYLAND

1/09	Project No. 06-3070.01-06	FIGURE 1

Cabrera Services
103 E. Mount Royal Ave.
Baltimore, MD 21202

USDA

N

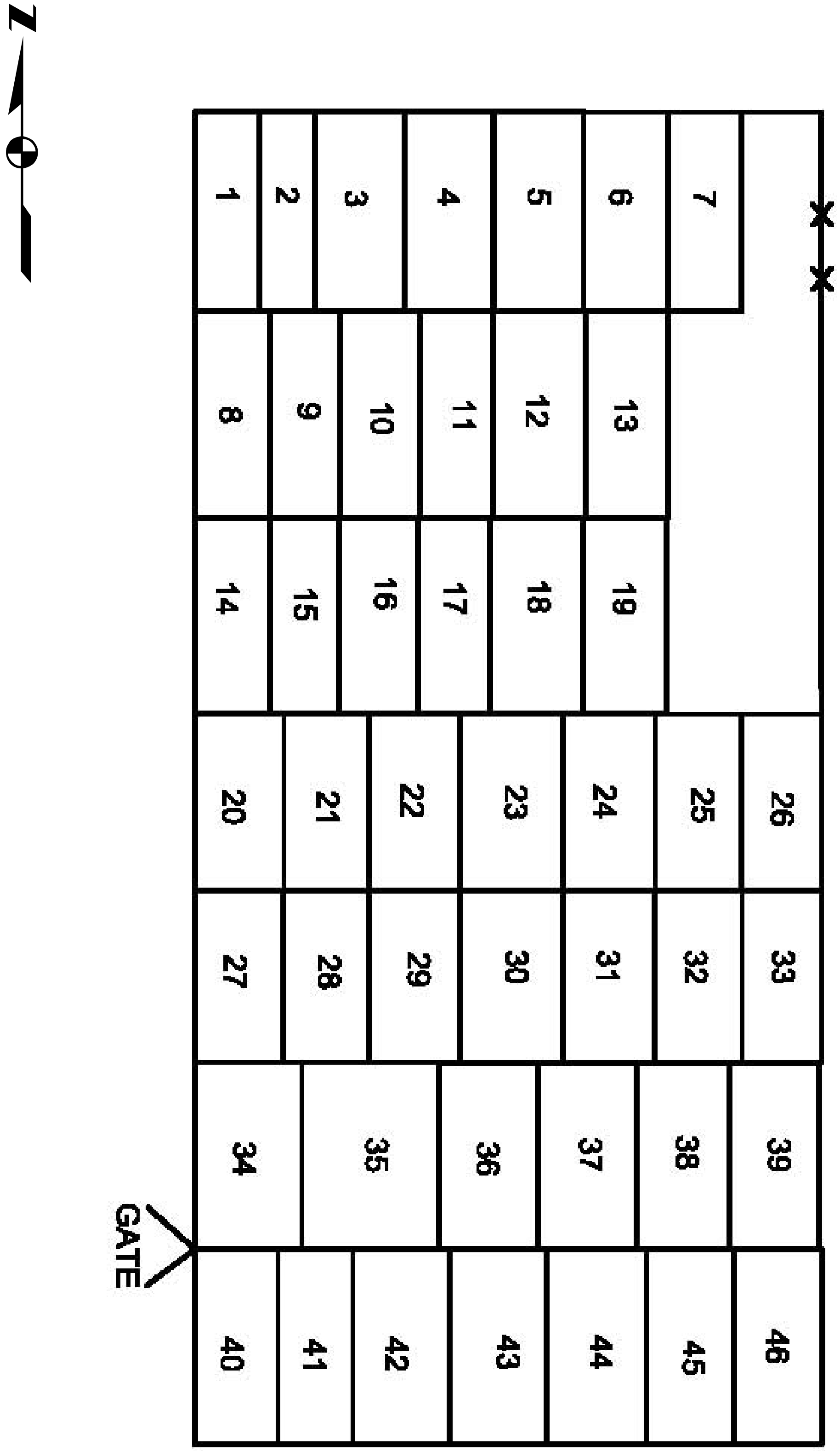
Legend

Fence Line

0 60 120

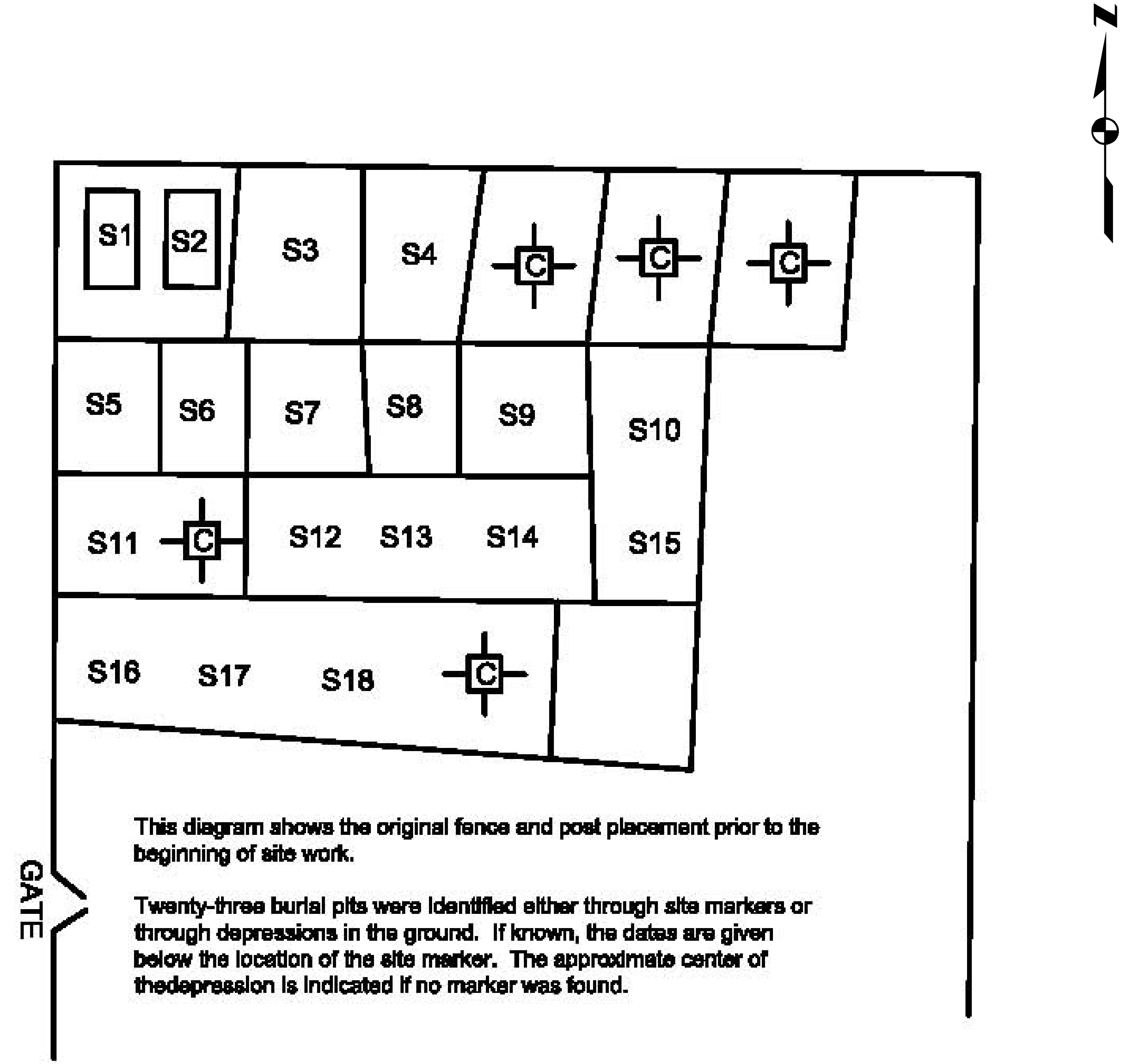
Scale (feet)

SITE LAYOUT		
BELTSVILLE AGRICULTURAL RESEARCH CENTER BELTSVILLE, MARYLAND		
1/09	Project No. 06-3070.01-06	FIGURE 2
Cabrera Services 103 E. Mount Royal Ave. Baltimore, MD 21202	USDA	



Note: Not to scale. Entech provided
to Cabrera via fax on 1/10/03

S1 = 4/6/51
S2 = 8/4/53
S3 = 5/13/59
S4 = 4/3/61
S5 = 8/67
S6 = 9/68
S7 = 5/70
S8 = 11/70
S9 = 4/71
S10 = 11/71
S11 = 7/72
S12 = 8/72
S13 = 11/72
S14 = 12/72
S15 = 3/73
S16 = 2/73
S17 = 3/15/73
S18 = 9/7/73

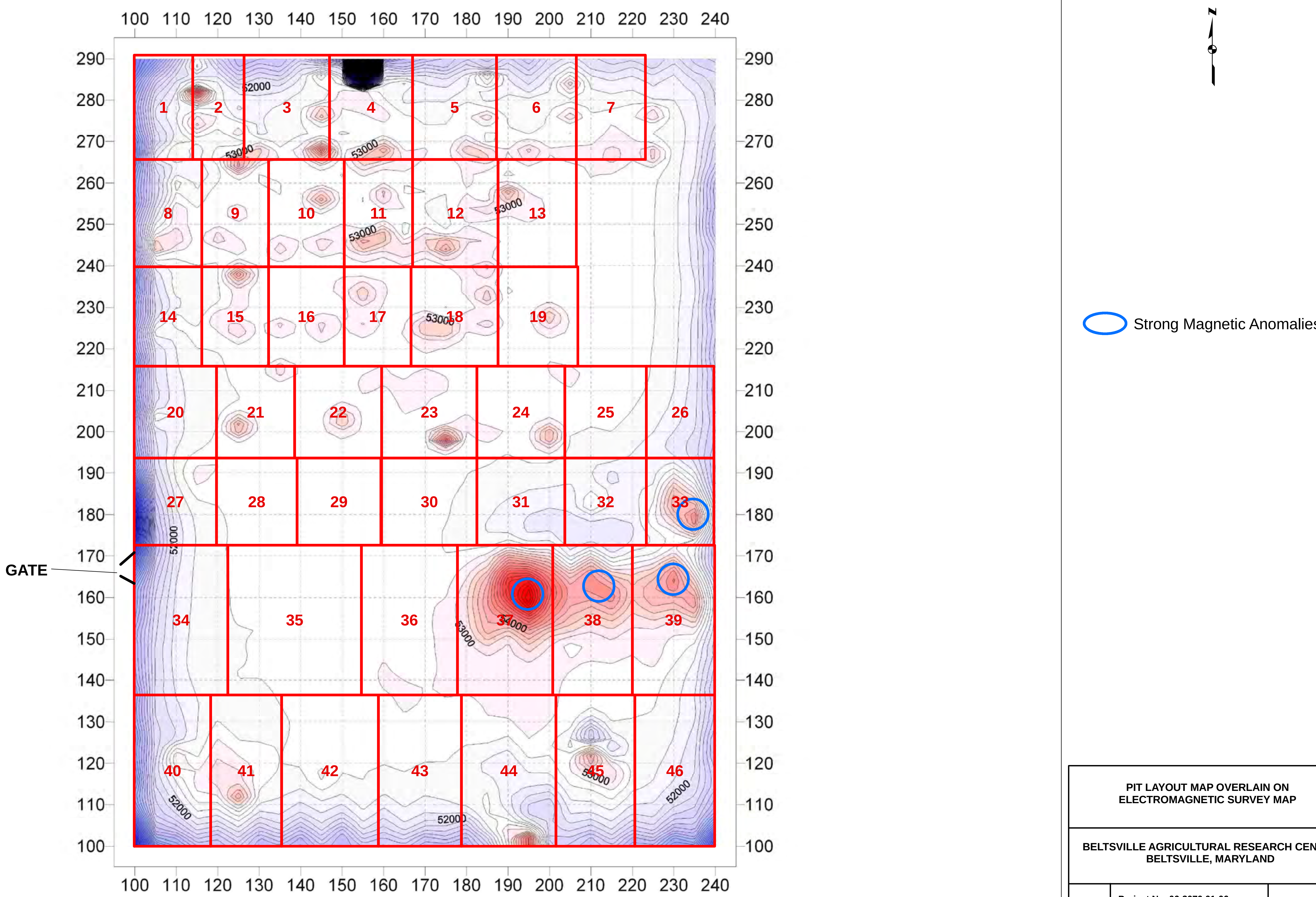


This diagram shows the original fence and post placement prior to the beginning of site work.

Twenty-three burial pits were identified either through site markers or through depressions in the ground. If known, the dates are given below the location of the site marker. The approximate center of thedepression is indicated if no marker was found.

Source: Letter dated 8/11/1989 from J.F. Patterson of RSO, Inc. to B. Flook, Agricultural Research Service

REV	DATE	DESCRIPTION	BY
CABRERA SERVICES 103 E. Mount Royal Ave Office 2B Baltimore, MD 21202		USDA Henery A. Wallace Beltsville Agriculture Reasearch Center Beltsville, Maryland Prepared for: U.S. Army Field Support Command Rock Island, Illinois	
PREPARED BY: JDM		BELTSVILLE	MARYLAND
REVIEWED BY:		HISTORICAL PIT LOCATIONS IN THE NORTH FIELD	
		Contract No. W52P1J-05-D-0043/0001 Goverment Project AG 2002-001	DRAWING # 3
		SCALE:	DATE: MAY 2007



PIT LAYOUT MAP OVERLAIN ON ELECTROMAGNETIC SURVEY MAP

BELTSVILLE AGRICULTURAL RESEARCH CENTER
BELTSVILLE, MARYLAND

1/09	Project No. 06-3070.01-06	FIGURE 4

Cabrera Services
103 E. Mount Royal Ave.
Baltimore, MD 21202

USDA

Topography slopes downward from west to east.

Waste pits are reportedly 10 ft wide by 12 ft long by 10 ft deep with 5 feet of cover and separated horizontally by 6 ft.

Native soils consist of approximately 50 feet of fine sand and gravelly sand overlying bedrock. The depth to groundwater beneath LLRBS is approximately 25 feet.

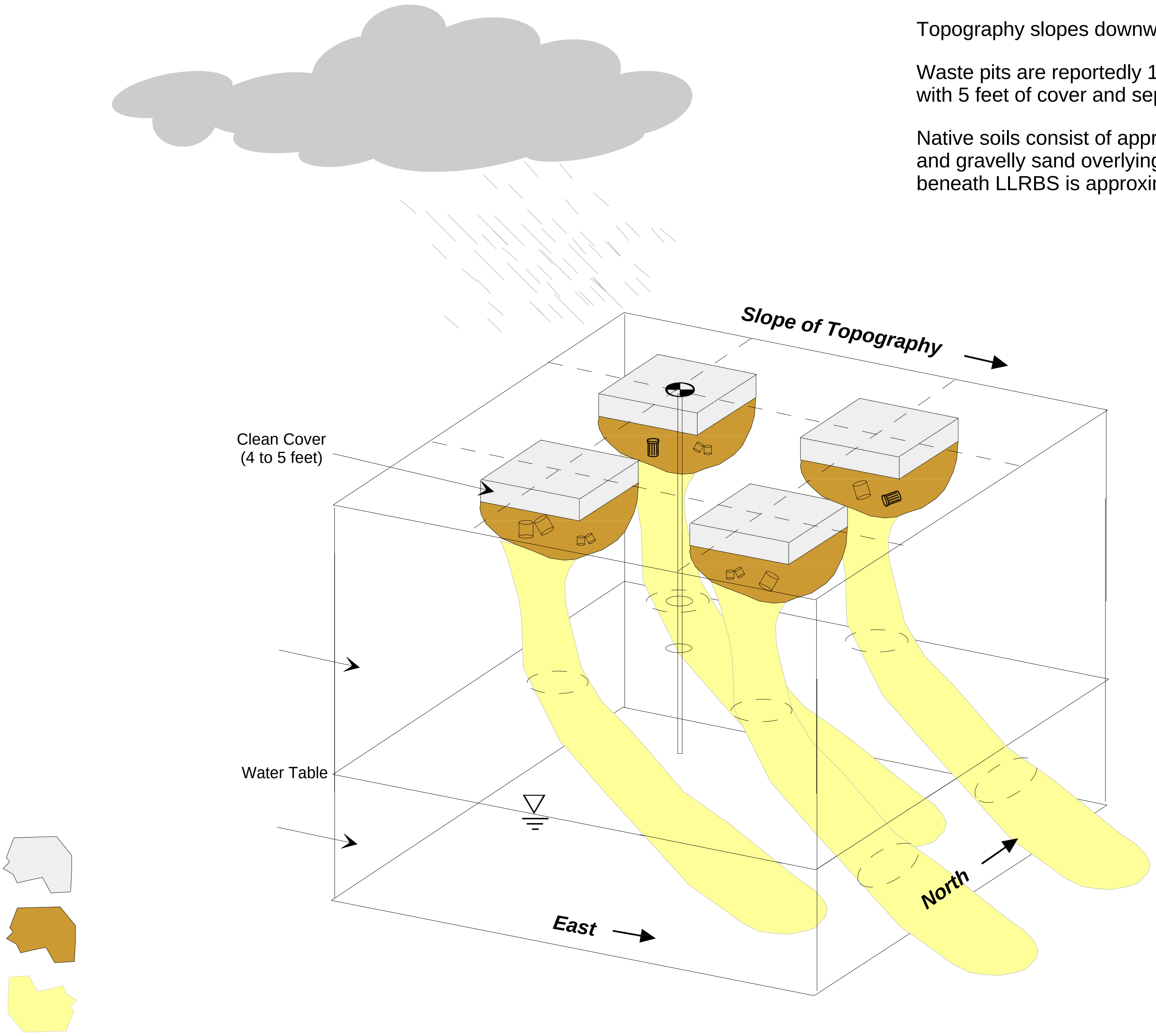


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Figure 5
Existing Monitoring Wells and
Contaminants of Concern in Groundwater

APPENDIX A

RESRAD Dose Modeling Supporting Information

BELTSVILLE AGRICULRAL RESEARCH CENTER, BELTSVILLE, MARYLAND
DEFAULT AND RECOMMENDED VALUES FOR RESRAD INPUT PARAMETERS

RESRAD Version 6.3					Parameter Justification	
Parameter	**Code**	**Default Value**	**User Input Value**	**Units**	**Comments**	**Reference**
PATHWAY SELECTIONS						
External Gamma	N/A	Active	Active	N/A	N/A	N/A
Inhalation (without radon)	N/A	Active	Active	N/A	N/A	N/A
Plant Ingestion	N/A	Active	Active	N/A	N/A	N/A
Meat Ingestion	N/A	Active	Active	N/A	N/A	N/A
Milk Ingestion	N/A	Active	Active	N/A	N/A	N/A
Aquatic Foods	N/A	Active	Active	N/A	N/A	N/A
Drinking Water	N/A	Active	Active	N/A	N/A	N/A
Soil Ingestion	N/A	Active	Active	N/A	N/A	N/A
Radon	N/A	Inactive	Inactive	N/A	Not applicable per Federal Register, 1994, p. 43210	NRC 1994
CONTAMINATED ZONE PARAMETERS						
Area of contaminated zone	AREA	10,000	513	m^2	Based on the dimensions of each pit, site-specific value was calculated.	ANL 1993 (Section 30)
Thickness of contaminated zone	THICK0	2	**3.048**	m	As a conservative approach, the contamination was assumed to be uniformly distributed from the bottom of the pit to the depth of groundwater (10’ = 3.048 meter).	ANL 1993 (Section 39)
Length parallel to the aquifer	LCZPAQ	100	22.7	m	For all cases, the length parallel to the aquifer was calculated as the square root of the contaminated zone area.	ANL 1993 (Section 16)
Times for calculations	TI	1, 3, 10, 30, 100, 300, 1000	1, 3, 10, 30, 100, 300, 1000	yr	RESRAD defaults for calculation times.	Yu 2001
COVER AND CONTAMINATED ZONE HYDROLOGICAL DATA						
Cover depth	COVER)	0	0	m	As a conservative approach for dose modeling, no cover depth was assumed.	ANL 1993 (Section 31)
Density of cover material	DENSCV	1.5	**N/A**	g/cm^3	Lack of cover depth precludes an assigned value for this parameter.	ANL 1993 (Section 2)
Cover erosion rate	VCV	0.001	**N/A**	m/yr	Lack of cover depth precludes an assigned value for this parameter.	ANL 1993

BELTSVILLE AGRICULRAL RESEARCH CENTER, BELTSVILLE, MARYLAND
DEFAULT AND RECOMMENDED VALUES FOR RESRAD INPUT PARAMETERS

RESRAD Version 6.3					Parameter Justification	
Parameter	**Code**	**Default Value**	**User Input Value**	**Units**	**Comments**	**Reference**
						(Section 14)
Density of contaminated zone	DENSCZ	1.5	1.5	g/cm^3	Due to complex soil composition, RESRAD default value was assumed.	ANL 1993 (Section 2)
Contaminated zone erosion rate	VCZ	0.001	0.0006	m/yr	No site-specific data, NRC and EPA value for this parameter could be located. Assuming 2% slope and significant farming and gardening activities at the site, 0.0006 m/yr was assigned for this parameter.	ANL, 1993
Contaminated zone total porosity	TPCZ	0.4	0.4	unitless	Due to complex soil composition, RESRAD default value was assumed.	ANL 1993 (Section 3)
Contaminated zone field capacity	FCCZ	0.2	0.2	unitless	Due to complex soil composition, RESRAD default value was assumed.	Yu 2001
Contaminated zone hydraulic conductivity	HCCZ	10	10	m/yr	Due to complex soil composition, RESRAD default value was assumed.	ANL 1993 (Section 5)
Contaminated zone b parameter	BCZ	5.3	5.3	unitless	Due to complex soil composition, RESRAD default value was assumed.	ANL 1993 (Section 13)
Humidity in air	HUMID	8	8	g/m^3	No site-specific data available. RESRAD default was used.	Yu 2001
Evapotranspiration coefficient	EVAPTR	0.5	0.5	unitless	No site-specific data available. RESRAD default was used.	ANL 1993 (Section 12)
Wind speed	WIND	2	**4.2**	m/sec	Site-specific value based on average annual wind speed of 9.4 mph.	ANL 1993 (Section 21) Cabrera 2007
Precipitation	PRECIP	1	**1**	m/yr	Site-specific value based on reported average annual rainfall, 39.54 in (100.4 cm) over the year.	ANL 1993 (Section 9) Cabrera 2007
Irrigation	RI	0.2	0.2	m/yr	No site-specific data available. RESRAD default used.	ANL 1993 (Section 11)
Irrigation mode	IDITCH	Overhead	Overhead	unitless	The “Overhead” and “Ditch” designations are independent of the depth of contaminated zone and have no significant impact on the RESRAD evaluation. The RESRAD default designation was selected.	Yu 2001
Runoff coefficient	RUNOFF	0.2	0.2	unitless	The RESRAD default value was selected.	ANL 1993

BELTSVILLE AGRICULRAL RESEARCH CENTER, BELTSVILLE, MARYLAND
DEFAULT AND RECOMMENDED VALUES FOR RESRAD INPUT PARAMETERS

RESRAD Version 6.3					Parameter Justification	
Parameter	**Code**	**Default Value**	**User Input Value**	**Units**	**Comments**	**Reference**
						(Section 10)
Watershed area for nearby stream or pond	WAREA	1.00E6	1.00E6	m^2	RESRAD default was used.	ANL 1993 (Section 17)
Accuracy for water/soil computations	EPS	0.001	0.001	unitless	RESRAD default was used.	Yu 2001
SATURATED ZONE HYDROLOGICAL DATA						
Density of saturated zone	DENSAQ	1.5	1.5	g/cm^3	Due to complex soil composition, RESRAD default value was assumed.	USDA 2006b ANL 1993 (Section 2)
Saturated zone total porosity	TPSZ	0.4	0.4	unitless	Due to complex soil composition, RESRAD default value was assumed.	ANL 1993 (Section 3)
Saturated zone effective porosity	EPSZ	0.2	0.2	unitless	Due to complex soil composition, RESRAD default value was assumed.	ANL 1993 (Section 4)
Saturated zone field capacity	FCSZ	0.2	0.2	unitless	Due to complex soil composition, RESRAD default value was assumed.	Yu 2001
Saturated zone hydraulic conductivity	HCSZ	100	100	m/yr	Due to complex soil composition, RESRAD default value was assumed.	ANL 1993 (Section 5)
Saturated zone hydraulic gradient	HGWT	0.02	0.02	unitless	Due to complex soil composition, RESRAD default value was assumed.	ANL 1993 (Section 15)
Saturated zone b parameter	BSZ	5.3	5.3	unitless	Due to complex soil composition, RESRAD default value was assumed.	ANL 1993 (Section 13)
Water table drop rate	VWT	0.001	0.001	m/yr	Due to complex soil composition, RESRAD default value was assumed.	ANL 1993 (Section 18)
Well pump intake depth (meters below water table)	DWIBWT	10	10	m	Due to complex soil composition, RESRAD default value was assumed.	ANL 1993 (Section 19)
Model for Water Transport Parameters [Non-dispersion (ND) or Mass-Balance (MB)]	MODEL	ND	**MB**	unitless	Per NRC guidance, the MB model is an acceptable approach and provides a potentially more conservative dose estimate relative to the ND model. The MB model assumes a well is located at the center of the site rather than on the down gradient side of the Site boundary.	NUREG-1757, Vo. 2, App. I, page I-40 NRC 1999b

BELTSVILLE AGRICULRAL RESEARCH CENTER, BELTSVILLE, MARYLAND
DEFAULT AND RECOMMENDED VALUES FOR RESRAD INPUT PARAMETERS

RESRAD Version 6.3					Parameter Justification	
Parameter	**Code**	**Default Value**	**User Input Value**	**Units**	**Comments**	**Reference**
Well pumping rate	UW	250	250	m^3/yr	RESRAD default was used.	Yu 2001
UNCONTAMINATED UNSATURATED ZONE PARAMETERS						
Number of unsaturated zone strata	NS	1	1	unitless	RESRAD default value was assumed.	ANL 1993 (Section 25)
Unsaturated zone thickness	H(1)	4	**0**	m	The unsaturated zone is assumed to be contaminated; hence no unsaturated zone was assumed for this site.	ANL 1993 (Section 25)
Unsaturated zone soil density	DENSUZ(1)	1.5	1.5	g/cm^3	RESRAD default value was assumed.	ANL 1993 (Section 2)
Unsaturated zone total porosity	TPUZ(1)	0.4	0.4	unitless	RESRAD default value was assumed.	ANL 1993 (Section 3)
Unsaturated zone effective porosity	EPSZ(1)	0.2	0.2	unitless	RESRAD default value was assumed.	ANL 1993 (Section 4)
Unsaturated zone field capacity	FCSZ(1)	0.2	0.2	unitless	RESRAD default value was assumed.	Yu 2001
Unsaturated zone hydraulic conductivity	HCSZ(1)	100	100	m/yr	RESRAD default value was assumed.	ANL 1993 (Section 5)
Unsaturated zone b parameter	BSZ	5.3	5.3	unitless	RESRAD default value was assumed.	ANL 1993 (Section 13)
ELEMENTAL DISTRIBUTION (PARTITION) COEFFICIENTS AND LEACH RATES: CARBON-14						
Contaminated zone	DCNUCC(2 & 3)	0	6.7	cm^3/g	NUREG/CR 5512 Volume 1 Table 6.7	NUREG/CR 5512 Vol.1 ANL 1993 (Section 32)
Unsaturated zone	DCNUCU(2 & 3,1)	0	6.7	cm^3/g	NUREG/CR 5512 Volume 1 Table 6.7	
Saturated zone	DCNUCS(2 & 3)	0	0	cm^3/g	RESRAD default was used.	
Leach rate	ALEACH(2 & 3)	0	0	y^{-1}	RESRAD default was used.	Yu 2001
Solubility constant	SOLUBK(2 & 3)	0	0	unitless	RESRAD default was used.	Yu 2001

BELTSVILLE AGRICULRAL RESEARCH CENTER, BELTSVILLE, MARYLAND
DEFAULT AND RECOMMENDED VALUES FOR RESRAD INPUT PARAMETERS

RESRAD Version 6.3					Parameter Justification	
Parameter	**Code**	**Default Value**	**User Input Value**	**Units**	**Comments**	**Reference**
ELEMENTAL DISTRIBUTION (PARTITION) COEFFICIENTS AND LEACH RATES: CHLORINE-36						
Contaminated zone	DCNUCC(1)	0.1	**1.7**	cm^3/g	NUREG/CR 5512 Volume 1 Table 6.7	NUREG/CR 5512 Vol.1 ANL 1993 (Section 32)
Unsaturated zone	DCNUCU(1,1)	0.1	**1.7**	cm^3/g	NUREG/CR 5512 Volume 1 Table 6.7	
Saturated zone	DCNUCS(1)	0.1	0.1	cm^3/g	RESRAD default was used.	
Leach rate	ALEACH(1)	0	0	y^{-1}	RESRAD default was used.	Yu 2001
Solubility constant	SOLUBK(1)	0	0	unitless	RESRAD default was used.	Yu 2001
ELEMENTAL DISTRIBUTION (PARTITION) COEFFICIENTS AND LEACH RATES: CESIUM-137						
Contaminated zone	DCNUCC(1)	4600	**270**	cm^3/g	NUREG/CR 5512 Volume 1 Table 6.7	NUREG/CR 5512 Vol.1 ANL 1993 (Section 32)
Unsaturated zone	DCNUCU(1,1)	4600	**270**	cm^3/g	NUREG/CR 5512 Volume 1 Table 6.7	
Saturated zone	DCNUCS(1)	4600	4,600	cm^3/g	RESRAD default was used.	
Leach rate	ALEACH(1)	0	0	y^{-1}	RESRAD default was used.	Yu 2001
Solubility constant	SOLUBK(1)	0	0	unitless	RESRAD default was used.	Yu 2001
ELEMENTAL DISTRIBUTION (PARTITION) COEFFICIENTS AND LEACH RATES: IRON-55						
Contaminated zone	DCNUCC(1)	1,000	**160**	cm^3/g	NUREG/CR 5512 Volume 1 Table 6.7	NUREG/CR 5512 Vol.1 ANL 1993 (Section 32)
Unsaturated zone	DCNUCU(1,1)	1,000	**160**	cm^3/g	NUREG/CR 5512 Volume 1 Table 6.7	
Saturated zone	DCNUCS(1)	1,000	1,000	cm^3/g	RESRAD default was used.	
Leach rate	ALEACH(1)	0	0	y^{-1}	RESRAD default was used.	Yu 2001

BELTSVILLE AGRICULRAL RESEARCH CENTER, BELTSVILLE, MARYLAND
DEFAULT AND RECOMMENDED VALUES FOR RESRAD INPUT PARAMETERS

RESRAD Version 6.3					Parameter Justification	
Parameter	**Code**	**Default Value**	**User Input Value**	**Units**	**Comments**	**Reference**
Solubility constant	SOLUBK(1)	0	0	unitless	RESRAD default was used.	Yu 2001
ELEMENTAL DISTRIBUTION (PARTITION) COEFFICIENTS AND LEACH RATES: SODIUM-22						
Contaminated zone	DCNUCC(1)	10	**76**	cm^3/g	NUREG/CR 5512 Volume 1 Table 6.7	NUREG/CR 5512 Vol.1 ANL 1993 (Section 32)
Unsaturated zone	DCNUCU(1,1)	10	**76**	cm^3/g	NUREG/CR 5512 Volume 1 Table 6.7	
Saturated zone	DCNUCS(1)	10	10	cm^3/g	RESRAD default was used.	
Leach rate	ALEACH(1)	0	0	y^{-1}	RESRAD default was used.	Yu 2001
Solubility constant	SOLUBK(1)	0	0	unitless	RESRAD default was used.	Yu 2001
ELEMENTAL DISTRIBUTION (PARTITION) COEFFICIENTS AND LEACH RATES: NICKEL-63						
Contaminated zone	DCNUCC(1)	1,000	**400**	cm^3/g	NUREG/CR 5512 Volume 1 Table 6.7	NUREG/CR 5512 Vol.1 ANL 1993 (Section 32)
Unsaturated zone	DCNUCU(1,1)	1,000	**400**	cm^3/g	NUREG/CR 5512 Volume 1 Table 6.7	
Saturated zone	DCNUCS(1)	1,000	1,000	cm^3/g	RESRAD default was used.	
Leach rate	ALEACH(1)	0	0	y^{-1}	RESRAD default was used.	Yu 2001
Solubility constant	SOLUBK(1)	0	0	unitless	RESRAD default was used.	Yu 2001
ELEMENTAL DISTRIBUTION (PARTITION) COEFFICIENTS AND LEACH RATES: LEAD-210						
Contaminated zone	DCNUCC(1)	100	**270**	cm^3/g	NUREG/CR 5512 Volume 1 Table 6.7	NUREG/CR 5512 Vol.1 ANL 1993 (Section 32)
Unsaturated zone	DCNUCU(1,1)	100	**270**	cm^3/g	NUREG/CR 5512 Volume 1 Table 6.7	
Saturated zone	DCNUCS(1)	100	100	cm^3/g	RESRAD default was used.	

BELTSVILLE AGRICULRAL RESEARCH CENTER, BELTSVILLE, MARYLAND
DEFAULT AND RECOMMENDED VALUES FOR RESRAD INPUT PARAMETERS

RESRAD Version 6.3					Parameter Justification	
Parameter	**Code**	**Default Value**	**User Input Value**	**Units**	**Comments**	**Reference**
Leach rate	ALEACH(1)	0	0	y^{-1}	RESRAD default was used.	Yu 2001
Solubility constant	SOLUBK(1)	0	0	unitless	RESRAD default was used.	Yu 2001
ELEMENTAL DISTRIBUTION (PARTITION) COEFFICIENTS AND LEACH RATES: RADIUM-226						
Contaminated zone	DCNUCC(1)	70	**500**	cm^3/g	NUREG/CR 5512 Volume 1 Table 6.7	NUREG/CR 5512 Vol.1 ANL 1993 (Section 32)
Unsaturated zone	DCNUCU(1,1)	70	**500**	cm^3/g	NUREG/CR 5512 Volume 1 Table 6.7	
Saturated zone	DCNUCS(1)	70	70	cm^3/g	RESRAD default was used.	
Leach rate	ALEACH(1)	0	0	y^{-1}	RESRAD default was used.	Yu 2001
Solubility constant	SOLUBK(1)	0	0	unitless	RESRAD default was used.	Yu 2001
ELEMENTAL DISTRIBUTION (PARTITION) COEFFICIENTS AND LEACH RATES: STRONIUM-90						
Contaminated zone	DCNUCC(1)	30	**15**	cm^3/g	NUREG/CR 5512 Volume 1 Table 6.7	NUREG/CR 5512 Vol.1 ANL 1993 (Section 32)
Unsaturated zone	DCNUCU(1,1)	30	**15**	cm^3/g	NUREG/CR 5512 Volume 1 Table 6.7	
Saturated zone	DCNUCS(1)	30	30	cm^3/g	RESRAD default was used.	
Leach rate	ALEACH(1)	0	0	y^{-1}	RESRAD default was used.	Yu 2001
Solubility constant	SOLUBK(1)	0	0	unitless	RESRAD default was used.	Yu 2001
OCCUPANCY, INHALATION AND EXTERNAL GAMMA DATA						
Inhalation rate	INHALR	8,400	8,400	m^3/y	RESRAD default used.	ANL 1993 (Section 43)
Mass loading for inhalation	MLINH	0.0001	0.0004	g/m^3	NUREG/CR 5512 Vol 4 Table 4	NUREG/CR 5512 Vol 4 ANL 1993

BELTSVILLE AGRICULRAL RESEARCH CENTER, BELTSVILLE, MARYLAND
DEFAULT AND RECOMMENDED VALUES FOR RESRAD INPUT PARAMETERS

RESRAD Version 6.3					Parameter Justification	
Parameter	**Code**	**Default Value**	**User Input Value**	**Units**	**Comments**	**Reference**
						(Section 35)
Exposure duration	ED	30	30	yr	RESRAD default used. DCGL calculations not influenced by exposure duration.	Yu 2001
Inhalation shielding factor	SHF3	0.4	0.4	unitless	RESRAD default used.	ANL 1993 (Section 36)
External gamma shielding factor	SHF1	0.7	**0.5512**	unitless	NUREG/CR 5512 Vol 4 Table 4	NUREG/CR 5512 (Vol. 4, Table 4)
Indoor time fraction	FIND	0.5	**0.6571**	unitless	NUREG/CR 5512 Vol 4 Table 4	NUREG/CR 5512 (Vol. 4, Table 4)
Outdoor time fraction	FOTD	0.25	**0.1181**	unitless	NUREG/CR 5512 Vol 4 Table 4	NUREG/CR 5512 (Vol. 4, Table 4)
Shape of the contaminated zone (circular or non-circular)	FS	Circular	Circular	unitless	RESRAD default used.	ANL 1993 (Section 50)
INGESTION PATHWAY (DIETARY DATA)						
Fruits, vegetables and grain consumption	DIET(1)	160	**112**	kg/yr	No site-specific value is available. Hence, NRC value was assigned.	ANL 1993 (Section 42) NUREG/CR 5512 (Vol. 4, p. 3-6)
Leafy vegetable consumption	DIET(2)	14	**21.4**	kg/yr	No site-specific value is available. Hence, NRC value was assigned.	ANL 1993 (Section 44) NUREG/CR 5512 (Vol. 4, p. 3-6)
Milk consumption	DIET(3)	92	**233**	L/yr	No site-specific value is available. Hence, NRC value was assigned.	ANL 1993 (Section 47) NUREG/CR 5512 (Vol. 4, p. 3-6)
Meat and poultry consumption	DIET(4)	63	**65.1**	kg/yr	No site-specific value is available. Hence, NRC value was assigned.	ANL 1993 (Section 46) NUREG/CR 5512 (Vol. 4, p. 3-6)

BELTSVILLE AGRICULRAL RESEARCH CENTER, BELTSVILLE, MARYLAND
DEFAULT AND RECOMMENDED VALUES FOR RESRAD INPUT PARAMETERS

RESRAD Version 6.3					Parameter Justification	
Parameter	**Code**	**Default Value**	**User Input Value**	**Units**	**Comments**	**Reference**
Fish consumption	DIET(5)	5.4	**20.6**	kg/yr	No site-specific value is available. Hence, NRC value was assigned.	ANL 1993 (Section 46) NUREG/CR 5512 (Vol. 4, p. 3-6)
Other seafood consumption	DIET(6)	0.9	0.9	kg/yr	No site-specific or NRC value is available. Hence, RESRAD default value was assigned.	ANL 1993 (Section 46) NUREG/CR 5512 (Vol. 4, p. 3-6)
Soil ingestion rate	SOIL	36.5	18.3	g/yr	No site-specific value is available. Hence, NRC value was assigned.	ANL 1993 (Section 46) NUREG/CR 5512 (Vol. 4, p. 3-6)
Drinking water intake	DWI	510	**478.5**	L/yr	No site-specific value is available. Hence, NRC value was assigned.	ANL 1993 (Section 46) NUREG/CR 5512 (Vol. 4, p. 3-6)
Contamination fraction of drinking water	FDW	1	1	unitless	Maximum NRC value used.	NRC 1999b Yu 2001
Contamination fraction of household water	FHHW	1	**NA**	unitless	Radon pathway is not selected; hence this parameter is not applicable	N/A
Contamination fraction of livestock water	FLW	1	1	unitless	Maximum NRC value used.	N/A
Contamination fraction of irrigation water	FIRW	1	1	unitless	Maximum NRC value used.	NRC 1999b
Contamination fraction of aquatic food	FR9	0.5	0.5	unitless	No site-specific or NRC value is available. Hence, RESRAD default value was assigned.	N/A
Contaminated fraction of plant food	FPLANT	-1	-1	unitless	Setting the parameter to -1 allows RESRAD to determine appropriate value based on area factor.	ANL 1993
Contaminated fraction of meat	FMEAT	-1	-1	unitless	Setting the parameter to -1 allows RESRAD to determine appropriate value based on area factor.	N/A
Contaminated fraction of milk	FMILK	-1	-1	unitless	Setting the parameter to -1 allows RESRAD to determine appropriate value based on area factor.	N/A
INGESTION PATHWAY (NON-DIETARY DATA)						

BELTSVILLE AGRICULRAL RESEARCH CENTER, BELTSVILLE, MARYLAND
DEFAULT AND RECOMMENDED VALUES FOR RESRAD INPUT PARAMETERS

RESRAD Version 6.3					Parameter Justification	
Parameter	**Code**	**Default Value**	**User Input Value**	**Units**	**Comments**	**Reference**
Livestock fodder intake for meat	LP15	68	**26.85**	kg/day	No site-specific value was available. NUREG/CR-6697 provides a range of values (13.4 to 53.6 kg/day); however, a specific justification for selecting a particular value within the range provided could not be identified. Therefore, the NRC value was selected.	NUREG/CR-5512, Vol 4 Table 3
Livestock fodder intake for milk	LP16	55	**63.25**	kg/day	No site-specific value was available. NUREG/CR-6697 provides a range of values (31.6 to 126 kg/day); however, a specific justification for selecting a particular value within the range provided could not be identified. Therefore, the default value was selected.	NUREG/CR-5512, Vol 4 Table 3
Livestock water intake for meat	LW15	50	50	L/day	No site-specific value was available. NUREG/CR-6697 provides a range of values (25 to 100 L/day). The "base" or middle value (50 L/day) is in agreement with the RESRAD default value was selected.	NUREG/CR-6697, Table 3-1, p. 158) ANL 1993 (Section 45)
Livestock water intake for milk	LW15	160	**60**	L/day	No site-specific value was available. NUREG/CR-5512 Vol 4 Table 3, milk cow water intake of 60 L/day was used.	NUREG/CR-5512, Vol 4 Table 3 ANL 1993 (Section 45)
Livestock intake of soil	LS1	0.5	**0.02**	kg/day	No site-specific value was available. NUREG/CR-5512 Vol 4 Table 3, milk cow soil intake of 0.02 kg/day was used.	NUREG/CR-5512, Vol 4 Table 3
Mass loading for foliar deposition	MLFD	0.0001	0.0001	g/m^3	No site-specific value was available. NUREG/CR-6697 provides a range of values (1E-7 to 7E-4 g/m^3); however, a specific justification for selecting a particular value within the range provided could not be identified. Therefore, the default value was selected.	NUREG/CR-6697, Table 3-1, p. 158)
Depth of soil mixing layer	DM	0.15	0.15	m	No site-specific value was available. NUREG/CR-6697 provides a range of values (0.075 to 0.3 m). The "base" or middle value (0.15 m) is in agreement with the RESRAD default value was selected.	NUREG/CR-6697, Table 3-1, p. 158) ANL 1993 (Section 35)
Depth of roots	DROOT	0.9	0.9	m	No site-specific value was available. NUREG/CR-6697 provides a range of values (0.3 to 3 m). The "base" or middle value (0.9 m) is in agreement with the RESRAD default value was selected.	NUREG/CR-6697, Table 3-1, p. 159) ANL 1993 (Section 37)
Groundwater fractional usage: Drinking water	FGWDW	1	1	unitless	No site-specific value was identified; the RESRAD and NRC parameter values are identical and were selected.	NUREG/CR-6697, Table 3-1, p. 3-7)
Groundwater fractional usage: Household water	FGWHH	1	**N/A**	unitless	Radon pathway not active	N/A

BELTSVILLE AGRICULRAL RESEARCH CENTER, BELTSVILLE, MARYLAND
DEFAULT AND RECOMMENDED VALUES FOR RESRAD INPUT PARAMETERS

RESRAD Version 6.3					Parameter Justification	
Parameter	**Code**	**Default Value**	**User Input Value**	**Units**	**Comments**	**Reference**
Groundwater fractional usage: Livestock water	FGWLW	1	1	unitless	No site-specific value was identified; the RESRAD and NRC parameter values are identical and were selected.	NUREG/CR-6697, Table 3-1, p. 3-7)
Groundwater fractional usage: Irrigation water	FGWIR	1	1	unitless	No site-specific value was identified; the RESRAD and NRC parameter values are identical and were selected.	NUREG/CR-6697, Table 3-1, p. 3-7)
PLANT TRANSPORT FACTORS						
Wet weight crop yield: non-leafy vegetables	YV(1)	0.7	**4**	kg/m^2	No site-specific value was available. NUREG/CR-5512 Vol 1 Table 6.14 value was selected.	NUREG/CR-6697, Table 3-1, p. 159)
Wet weight crop yield: leafy vegetables	YV(2)	1.5	**2**	kg/m^2	No site-specific value was available. NUREG/CR-5512 Vol 1 Table 6.14 value was selected.	NUREG/CR-6697, Table 3-1, p. 159)
Wet weight crop yield: fodder	YV(3)	1.1	**1.5**	kg/m^2	No site-specific value was available. NUREG/CR-5512 Vol 1 Table 6.14 value was selected.	NUREG/CR-6697, Table 3-1, p. 159)
Length of growing season: non-leafy vegetables	TE(1)	0.17	0.17	years	No site-specific value was available. NUREG/CR-6697 provides a range of values (0.085 to 0.4932 years); however, a specific justification for selecting a particular value within the range provided could not be identified. Therefore, the default value was selected.	NUREG/CR-6697, Table 3-1, p. 159)
Length of growing season: leafy vegetables	TE(2)	0.25	0.25	years	No site-specific value was available. NUREG/CR-6697 provides a range of values (0.062 to 0.246 years); however, a specific justification for selecting a particular value within the range provided could not be identified. Therefore, the default value was selected.	NUREG/CR-6697, Table 3-1, p. 159)
Length of growing season: fodder	TE(3)	0.08	0.08	years	No site-specific value was available. NUREG/CR-6697 provides a range of values (0.04 to 0.16 years). The “base” or middle value (0.08) is in agreement with the RESRAD default value was selected.	NUREG/CR-6697, Table 3-1, p. 160)
Translocation factor: non-leafy vegetables	TIV(1)	0.1	0.1	unitless	No site-specific value was available. NUREG/CR-6697 provides a range of values (0.06 to 0.2). The “base” or middle value (0.1) is in agreement with the RESRAD default value was selected.	NUREG/CR-6697, Table 3-1, p. 160)
Translocation factor: leafy vegetables	TIV(2)	1	1	unitless	No site-specific value was available. NUREG/CR-6697 provides a range of values (0.5 to 1). The maximum value (1) is in agreement with the RESRAD default value was selected.	NUREG/CR-6697, Table 3-1, p. 160)

BELTSVILLE AGRICULRAL RESEARCH CENTER, BELTSVILLE, MARYLAND
DEFAULT AND RECOMMENDED VALUES FOR RESRAD INPUT PARAMETERS

RESRAD Version 6.3					Parameter Justification	
Parameter	**Code**	**Default Value**	**User Input Value**	**Units**	**Comments**	**Reference**
Translocation factor: fodder	TIV(3)	1	1	unitless	No site-specific value was available. NUREG/CR-6697 provides a range of values (0.5 to 1). The maximum value (1) is in agreement with the RESRAD default value was selected.	NUREG/CR-6697, Table 3-1, p. 160)
Weathering removal constant	WLAM	20	20	y^{-1}	No site-specific value was available. NUREG/CR-6697 provides a range of values (10 to 40). The “base” or middle value (20) is in agreement with the RESRAD default value was selected.	NUREG/CR-6697, Table 3-1, p. 160)
Wet foliar interception fraction: non-leafy vegetables	RWET(1)	0.25	0.25	unitless	No site-specific value was available. NUREG/CR-6697 provides a range of values; however, a specific justification for selecting a particular value within the range provided could not be identified. Therefore, the default value was selected.	NUREG/CR-6697, Table 3-1, p. 160)
Wet foliar interception fraction: leafy vegetables	RWET(2)	0.25	0.25	unitless	No site-specific value was available. NUREG/CR-6697 provides a range of values; however, a specific justification for selecting a particular value within the range provided could not be identified. Therefore, the default value was selected.	NUREG/CR-6697, Table 3-1, p. 161)
Wet foliar interception fraction: fodder	RWET(3)	0.25	0.25	unitless	No site-specific value was available. NUREG/CR-6697 provides a range of values; however, a specific justification for selecting a particular value within the range provided could not be identified. Therefore, the default value was selected.	NUREG/CR-6697, Table 3-1, p. 161)
Dry foliar interception fraction: non-leafy vegetables	RDRY(1)	0.25	0.25	unitless	No site-specific value was available. NUREG/CR-6697 provides a range of values; however, a specific justification for selecting a particular value within the range provided could not be identified. Therefore, the default value was selected.	NUREG/CR-6697, Table 3-1, p. 160)
Dry foliar interception fraction: leafy vegetables	RDRY(2)	0.25	0.25	unitless	No site-specific value was available. NUREG/CR-6697 provides a range of values; however, a specific justification for selecting a particular value within the range provided could not be identified. Therefore, the default value was selected.	NUREG/CR-6697, Table 3-1, p. 160)
Dry foliar interception fraction: fodder	RDRY(3)	0.25	0.25	unitless	No site-specific value was available. NUREG/CR-6697 provides a range of values; however, a specific justification for selecting a particular value within the range provided could not be identified. Therefore, the default value was selected.	NUREG/CR-6697, Table 3-1, p. 160)
STORAGE TIMES BEFORE USE						
Fruits, non-leafy vegetables and grain	STOR_T(1)	14	14	days	No site-specific value was identified; the RESRAD and NRC parameter values are identical and were selected.	NUREG/CR-6697, Table 3-1, p. 3-7)

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DEFAULT AND RECOMMENDED VALUES FOR RESRAD INPUT PARAMETERS

RESRAD Version 6.3					Parameter Justification	
Parameter	**Code**	**Default Value**	**User Input Value**	**Units**	**Comments**	**Reference**
Leafy vegetables	STOR_T(2)	1	1	days	No site-specific value was identified; the RESRAD and NRC parameter values are identical and were selected.	NUREG/CR-6697, Table 3-1, p. 3-7)
Milk	STOR_T(3)	1	1	days	No site-specific value was identified; the RESRAD and NRC parameter values are identical and were selected.	NUREG/CR-6697, Table 3-1, p. 3-7)
Meat	STOR_T(4)	20	20	days	No site-specific value was identified; the RESRAD and NRC parameter values are identical and were selected.	NUREG/CR-6697, Table 3-1, p. 3-7)
Fish	STOR_T(5)	7	7	days	No site-specific value was identified; the RESRAD and NRC parameter values are identical and were selected.	NUREG/CR-6697, Table 3-1, p. 3-7)
Crustacea and mollusks	STOR_T(6)	7	7	days	No site-specific value was identified; the RESRAD and NRC parameter values are identical and were selected.	NUREG/CR-6697, Table 3-1, p. 3-7)
Well water	STOR_T(7)	1	1	days	No site-specific value was identified; the RESRAD and NRC parameter values are identical and were selected.	NUREG/CR-6697, Table 3-1, p. 3-7)
Surface water	STOR_T(8)	1	1	days	No site-specific value was identified; the RESRAD and NRC parameter values are identical and were selected.	NUREG/CR-6697, Table 3-1, p. 3-7)
Livestock fodder	STOR_T(9)	45	45	days	No site-specific value was identified; the RESRAD and NRC parameter values are identical and were selected.	NUREG/CR-6697, Table 3-1, p. 3-7)

APPENDIX B

Final Status Survey Plan

LOW LEVEL RADIATION BURIAL SITE
FINAL STATUS SURVEY DESIGN PLAN

United States Department of Agriculture

Henry A. Wallace
Beltsville Agricultural Research Center (BARC)
Beltsville, Maryland

Prepared for:

U.S. Army Field Support Command
Rock Island, Illinois

Prepared by:

Cabrera Services, Inc.
Baltimore, Maryland

Contract No. W52P1J-05-D-0043/0001
Government Project AG 2002-001

August 2007

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GLOSSARY OF ACRONYMS AND ABBREVIATIONS

α	Type I error
AEC	Atomic Energy Commission
AFSC	U.S. Army Field Support Command
ARS	Agricultural Research Service
β	Type II error
BARC	Henry A. Wallace Beltsville Agricultural Research Center
^{14}C	carbon-14
CABRERA	Cabrera Services, Inc.
CCOC	chemical contaminants of concern
CFR	Code of Federal Regulations
^{36}Cl	chlorine-36
^{137}Cs	cesium-137
CSM	conceptual site model
Δ	the width of the gray region
DCGL	Derived Concentration Guideline Level
$DCGL_{EMC}$	Used when small areas of elevated radioactivity exist within larger areas ("$_{EMC}$" stands for elevated measurement comparison).
$DCGL_w$	Reference criterion, or radioactivity level for residual radioactivity evenly distributed over a wide area.
DGPS	Differentially-Corrected Global Positioning System
DoD	U.S. Department of Defense
DOE	U.S. Department of Energy
DP	Decommissioning Plan
DQOs	Data Quality Objectives
DSR	dose-to-source ratio
EE/CA	Engineering Evaluation/Cost Analysis
EMC	Elevated Measurement Comparison
EPA	U.S. Environmental Protection Agency
FDA	Food and Drug Administration
^{55}Fe	iron-55
FIDLER	Field Instrument for the Detection of Low Energy Radiation
FSS	Final Status Survey
σ	estimate of the standard deviation
GIS	Geographic Information System
GM	cluster pancake Geiger-Mueller detector
GWS	Gamma Walkover Survey
^{3}H	hydrogen-3/tritium
H_a	alternative hypothesis
H_0	null hypothesis

IDW	investigation-derived waste
LBGR	lower bound of the gray region
LLRBS	Low Level Radiation Burial Site
LLRW	low-level radioactive waste
µg/kg	micrograms per kilogram
µg/L	micrograms per liter
m^2	square meters
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MCL	Maximum Contaminant Level
MDCs	Minimum Detectable Concentrations
mg/kg	milligrams per kilogram
mrem/yr	millirem per year
MSL	mean sea level
^{22}Na	sodium-22
NaI	sodium iodide
^{63}Ni	nickel-63
NIST	National Institute of Standards and Technology
NRC	U.S. Nuclear Regulatory Commission
^{32}P	phosphorus-32
PA/SI	Preliminary Assessment/Site Investigation
^{210}Pb	lead-210
pCi/g	picoCuries per gram
pCi/L	picoCuries per liter
QA	quality assurance
QC	quality control
^{226}Ra	radium-226
^{228}Ra	radium-228
RBCs	Risk Based Concentrations
RDRC	Radioactive Drug Research Committee
RESRAD	Residual Radioactivity
RI	Remedial Site Investigation
ROCs	radionuclides of concern
RSO	Radiation Safety Officer
SOR	sum of the ratios
^{90}Sr	strontium-90
SU	survey unit
SVOC	semi-volatile organic compound
TEDE	total effective dose equivalent
^{230}Th	thorium-230
USDA	U.S. Department of Agriculture

VOC	volatile organic compound
WRS	Wilcoxon Rank Sum

1.0 INTRODUCTION

The United States Army Field Support Command (AFSC) has contracted Cabrera Services, Inc. (CABRERA) to prepare a Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) (U.S. Department of Defense [DoD], U.S. Department of Energy [DOE], U.S. Environmental Protection Agency [EPA], and U.S. Nuclear Regulatory Commission [NRC] 2000) compliant Final Status Survey (FSS) Design Plan in support of site decommissioning activities of a Low Level Radiation Burial Site (LLRBS). The work is being conducted for the U.S. Department of Agriculture (USDA), Agricultural Research Service (ARS), Henry A. Wallace Beltsville Agricultural Research Center (BARC), in Beltsville, Maryland. Hereafter, the BARC Low Level Radiation Burial Site will be referred to as the LLRBS or the Site (Figure 1).

Tasks defined in this FSS Design Plan include the following:

- Beta and gamma walkover surveys to map potential near-surface radiological materials
- Excavation, radiological characterization, segregation, and packaging of soils and waste materials in the remaining waste cells or disposal pits
- Sampling surface and subsurface soil along the floor of each excavation (from 0 to 2 feet below the excavation floor), and installing a temporary well in each disposal pit to sample groundwater
- Backfilling all excavations with clean fill and restoring the site to pre-disposal conditions

This FSS Design Plan addresses post-removal action surveys for radiological materials at the Site. Cleanup activities will be accomplished in accordance with a Removal Action Plan and associated documents. The removal action will be performed concurrently with final status surveys. The expected outcome of the selected remedy is that the Site will no longer present an unacceptable risk to adjacent residents via exposure to contaminated soil and sediment and will be suitable for a resident farmer.

The objective of this FSS Design Plan is to provide a consistent approach for planning, performing, and assessing radionuclides of concern (ROCs) present in site surface and subsurface soils through final status surveys in order to demonstrate compliance with established dose and risk-based criteria Derived Concentration Guideline Level (DCGLs). This FSS Design Plan does not address chemical contaminants of concern.

2.0 SITE DESCRIPTION

2.1 Project Location

The BARC Superfund site comprises 6,600-acre parcel of property in northwestern Prince George's County near Beltsville, Maryland (Figure 1). In 1910, the USDA purchased a 475-acre farm in order to conduct agricultural research. The facility expanded to a maximum of 12,000 acres and is now at its present size of 6,600 acres. Research at BARC involves soil, water, and air conservation, plant sciences, animal sciences, commodity conversion and delivery, and human nutrition. In addition, research is done on pesticides, herbicides, insecticides, and fungicides. On-site laboratories are equipped with numerous chemicals, solvents, cleaners, and low-level radioactive chemicals for laboratory studies. Solid wastes generated at BARC included manure, waste bedding, animal carcasses, vegetative cuttings, wood, paper, scrap metal, laboratory waste, construction debris, and pesticide-, herbicide-, insecticide-, and fungicide-derived wastes.

The LLRBS is located approximately one-quarter mile north of the Cherry Hill Road overpass of the Capital Beltway (I95/495) in Beltsville, Maryland (Figure 1), northeast of Washington D.C. Primary access is through BARC via US Route 1/Baltimore Avenue. Secondary access to the LLRBS is a gravel road that leads from Cherry Hill Road to a cluster of BARC maintenance buildings and continues along the western side of the BARC. This access is fenced and locked.

2.2 Disposal Site Background

The LLRBS was established on June 23, 1949, and was used for the disposal of low-level radioactive waste (LLRW) until 1987. The LLRBS is permitted under a USDA-wide license originally issued by the Atomic Energy Commission (AEC), and later by the NRC. Records indicate the last liquid burial at the LLRBS was on September 17, 1984. From September 24, 1985 until disposal activities ended in 1987, all burials were dry solids packed in 55-gallon drums (Entech, 2000).

The LLRBS is made up of a total of 50 designated waste disposal pits, of which, only 39 were reportedly used. Figure 2 presents two conflicting historical maps depicting the disposal pit locations, indicating why there is uncertainty in both the location and contents of the disposal pits. The disposal pits are reported to be approximately 10 feet wide by 12 feet long by 10 feet deep and are separated approximately 6 feet horizontally from one another. Each disposal pit was reportedly backfilled to surface grade with at least 5 feet of clean fill. Two contiguous fenced fields, the North Field and the South Field, make up the LLRBS, each of which is approximately 150 feet by 200 feet (Figure 3). The South Field was reportedly never used for disposal of any waste materials, although individual disposal pits had been designated (Apex, 1993). Buried materials within North Field disposal pits include radioactive isotopes and scintillation fluids (isotopes and organic fluids); contaminated metal, glass, and plastic objects; contaminated animal carcasses; and animal wastes. USDA records do not reveal the types of organic liquids contained in the scintillation vials, nor is any indication of volume included. Typically, organic liquids associated with scintillation fluids include the aromatic solvents; toluene and xylenes. Types of containers disposed, according to files, are cardboard boxes of 1 to 4 cubic feet, 1- to 5-gallon containers for liquids, plastic milk jugs, plastic carboys, solvent bottles, fiberboard drums, and 55-gallon drums. Liquid containers were placed in cardboard boxes, usually four to a box.

No known records exist listing the specific types of containers used for the early burials. Records from the 1980s indicate contaminated, non-flammable glass and plastic (vials, pipettes, needles, scalpels, etc.) were buried in cardboard boxes. Animal carcasses, bedding, and excreta were sealed in polyethylene bags and placed in boxes. Liquid wastes were packed in plastic containers and placed in cardboard boxes. Animal remains, generally contaminated with hydrogen-3/tritium (^{3}H) and carbon-14 (^{14}C), were routinely incinerated beginning in the early 1980s, in one of two incinerators located at the BARC. It is possible that incinerator ash, which tested positive for radioactivity, could have been sent to the LLRBS; however, no known records exist regarding ash disposal.

2.3 Physical Setting

The BARC facility is situated in the Atlantic Coastal Plain Province, and in this area, it can be described as gently rolling hills with broad valleys. The elevation varies from about 60 feet above mean sea level (MSL) where Indian Creek flows beneath Interstate 95/495 to 268 feet MSL in the extreme western portion of the facility on Cherry Hill Road near the LLRBS. Topography slopes to the east and southeast at 10 to 15 percent toward the nearest perennial stream, the Little Paint Branch, located approximately 2,000 feet east of the site boundary. Downstream, Little Paint Branch feeds into Paint Branch, 1.4 miles to the south, eventually draining into the Anacostia River. There are extensive wooded tracts in the central and eastern portions of BARC, while open agricultural fields are prevalent in the western section.

There are many perennial and intermittent streams, wetlands, and surface water bodies within BARC boundaries. Drainage features include Paint Branch and Little Paint Branch, which flow from north to south and are located in the western portion of the facility. Indian Creek also flows north to south parallel to Edmonston Road; and Beaver Dam Creek flows east to west in the south-central portion of BARC. All of these drainage features eventually flow southward into the Anacostia River (approximately 6 miles from the facility), which empties into the Potomac River at Washington, D.C.

There are no wetlands adjacent to LLRBS. The nearest wetlands are approximately 2,300 feet to the east, on the banks of the Little Paint Branch.

2.4 Geology

The USDA Natural Resources Conservation Service soils maps for Prince George’s County describe numerous soil associations and groups of soils within the facility. Many of these units are described as comprising silty loam, loamy sand, and sandy loam of variable slope, drainage characteristics, and susceptibility to erosion. Surface soils are underlain by highly variable deposits ranging from gravels to clays, some as old as Cretaceous.

The geology at BARC consists of Lower Cretaceous sediments of the Potomac Group, which consists of the Patuxent, the Arundel, and the Patapsco Formations, in decreasing age. The Patuxent and Patapsco Formations are composed primarily of sand and gravel, and comprise the most prevalent water bearing aquifers in Prince George's County. The Arundel is mostly clay, and creates artesian conditions in the underlying Patuxent Formation in some locations. Recharge of the Patuxent Formation occurs where it outcrops in the western portions of BARC. This wedge of sediments made up of the Patuxent, Arundel, and Patapsco Formations dips to the southeast, parallel to the regional groundwater flow (Apex, 1991).

The LLRBS lies on the Patuxent Formation. Soil textures beneath the site are well-sorted sand and gravel with minor clay lenses. This sand and gravel sequence overlies several feet of clay, below which are the igneous and metamorphic rocks of the Piedmont Province.

2.5 Hydrogeology

The site lies within the outcrop area of the Patuxent Formation, a part of the Potomac Group. The depth to groundwater is approximately 25 feet and the depth to bedrock is approximately 55 feet. The predominant soil texture is fine sand. Based on the results of three monitor wells, the mean hydraulic conductivity is estimated to be 6.5E-4 centimeters/second (Apex, 1993). Using the estimated average aquifer thickness of 30 feet and effective porosity of 30 percent, the average transmissivity beneath the site is estimated to be 1.78 square centimeters/second. Using the average gradient of 0.021 feet/foot, the average groundwater flow velocity is estimated to be 5.4 meters/year.

The Patuxent Formation is used as a drinking water supply in Prince George's county. Nine water supply wells on BARC property tap the Patuxent Aquifer (Apex, 1993).

2.6 Land Use

The BARC facility is best characterized as minimally developed, and is surrounded by land that is largely urbanized and densely populated. Inside the facility's boundaries, land use is agriculture, forest, and urban, with more than 800 buildings including laboratories, greenhouses, barns, office buildings, and some residences. A major portion of the facility is currently being used for crops, grazing livestock, and orchard research projects, primarily in the central and western portions of BARC. The central and eastern portions of the facility are primarily covered with mixed deciduous/evergreen forest. The urbanized portions of BARC are scattered throughout the property.

Land use outside of the facility boundaries is largely mixed urban and lightly developed, primarily forested parcels. There is widespread residential development along the western, southwestern, and northwestern boundaries of BARC. Commercial development is prevalent along U.S. Route 1 and the Beltsville Industrial Center, north of Sunnyside Avenue. Other major transportation routes that either border or pass through BARC are Interstate 95, Interstate 95/495, the Baltimore-Washington Parkway, and the B&O Railroad.

2.7 Previous Investigations

There have been a series of environmental investigations at BARC that were not specific to, but included investigation of the LLRBS starting with a Preliminary Assessment/Site Inspection (PA/SI) in 1991 (Apex, 1993), and continuing through a Remedial Site Investigation (RI) being conducted by Entech, Inc., that began in July 1997, and continues with ongoing groundwater monitoring. The results of these investigations are described in more detail in the Engineering Evaluation/Cost Analysis (EE/CA), and the relevant information is summarized below:

- PA/SI, May 1991; 44 potentially contaminated sites were identified (Apex, 1991), including the Low Level Radioisotope Burial Site; the LLRBS was not included in the SI, and so no analytical data resulted from this effort;

- Hydrologic Characterization and Monitoring of the LLRBS (Apex, 1993); study included 8 soil borings to bedrock with split spoon samples taken every 5 feet (for lithology, radiological and organic contaminants); 3 monitoring wells were installed for potentiometric groundwater data;

- Environmental Monitoring Summary Report of the LLRBS, March 1994; 5 additional monitoring wells were installed to complement the initial 3 wells, forming 3 clusters (shallow/deep) and two additional side-gradient shallow wells;

- EPA Region III Technical Assistance Team Sampling, February 1996; sampling of the 8 monitoring wells (analysis for volatile organic compounds [VOCs], semi volatile organic compounds [SVOCs], metals, gross alpha, gross beta, and gamma isotopes, and selected radionuclides);

- Aerial Photographic Site Analysis: BARC, January 1997;

- A Streamlined Risk Evaluation (Entech, 1998).

Conclusions from these investigations are summarized below:

- The stratigraphy at the LLRBS consists of silty sand and gravel terrace deposits overlying well-sorted sand and gravel of the Patuxent Formation; the hydrology can be described as a single, unconfined aquifer over bedrock varying in thickness from 20 to 30 feet, with southeasterly groundwater flow along topography.

- There are no radiological or chemical constituents in soils upgradient or downgradient of the LLRBS above background levels;

- There is a chloroform groundwater plume downgradient from the LLRBS;

- The EPA Technical Assistance Team concluded that there is a tritium source zone that likely originates from the LLRBS (contradicting the March 1994 Apex Environmental Report);

- The northern field was active from 1952 to 1987 (based on 11 sets of aerial photos taken from 1937 to 1993);

- There is no evidence that the southern field was ever used for disposal, although it was used for drum storage of non-hazardous investigation-derived waste (IDW);

- Residual radioactivity modeling performed as part of the Streamlined Risk Evaluation (Entech, 1998) indicates that the best long-term solution for the LLRBS will include removal of the source material.

2.8 Results of 2006 Characterization Survey

A characterization survey of LLRBS was conducted in 2006 to assess conditions in the disposal pits and confirm existing assumptions about the nature of the disposals. The survey included the following tasks:

- Geophysical surveys in the North Field to delineate the burial cells, and in the South Field to confirm the assumption that no burials have taken place there,
- Gamma and beta walkover surveys to map potential near-surface radiological materials,
- Sampling groundwater from selected wells to assess migration of contaminants away from source materials,
- Excavation, radiological characterization, segregation, and packaging of waste soils and materials in 4 of the 50 documented waste cells,
- Sampling soil along the floor of each excavation,
- Installing temporary wells to sample groundwater from beneath each excavated disposal pit,
- Backfilling and restoring all excavations with clean fill.

The geophysical surveys of the South Field further substantiated the assumption that the South Field was not used for disposals.

2.8.1 Summary of Treatment and Disposal Activities

A total of less than 25 milliCuries of activity has been removed from the excavated. More than 75% of the activity is due to radium-226 (^{226}Ra) and ^{3}H. The ^{3}H activity was largely in groundwater pumped from Disposal Pits 26 and 34 during dewatering operations. Another 20% of the activity is from ^{14}C and nickel-63 (^{63}Ni), each with about 10% of the total. The remaining 5% is from lead-210 (^{210}Pb), chlorine-36 (^{36}Cl), and strontium-90 (^{90}Sr).

To date, 28 drums of liquid scintillation vials, 2 drums of bulked liquids, and 3 radioactive sources/devices have been shipped from the site for processing and disposal. The liquid scintillation vials and the bulked mixed waste were shipped to the Permafix facility in Gainesville, Florida for treatment and disposal. Two ^{226}Ra sources and a ^{63}Ni electron capture device were shipped to Alaron Corporation and were received under their source recycling license.

This characterization survey excavated approximately 10% (Disposal Pits 1, 14, 26, and 34) of the 50 known disposal pits at the LLRBS as a part of characterization survey for the site. Based on the work completed thus far, it is anticipated that the nature and extent of the waste in the remaining disposal pits will be highly variable. Waste materials encountered during excavation include laboratory trash (gloves, paper, metals, plastics, laboratory glassware, and other wastes generated during the process of performing laboratory analyses), liquid scintillation vials, radioactive sources, bulk soils containing small amounts of waste or debris not readily separable from soil, animal remains, and bulk liquids in their original containers (CABRERA, 2007a).

Disposed materials within the disposal pits often contained both radionuclides and hazardous wastes at concentrations that exceed NRC and/or EPA screening criteria for groundwater and soil. Radionuclide concentrations found in soil and water adjacent to the disposed materials were generally within regulatory limits.

2.9 Radionuclides of Concern

The types of radioactive waste material at the LLRBS consisted of ^{3}H, ^{14}C, ^{36}Cl, ^{63}Ni, ^{90}Sr, ^{210}Pb, ^{226}Ra, sodium-22 (^{22}Na), phosphorus-32 (^{32}P), iron-55 (^{55}Fe), cesium-137 (^{137}Cs), and other radionuclides. Some of the radiological materials deposited in the LLRBS are short-lived (i.e., have a short radiological half-life), including ^{22}Na, ^{32}P, and ^{210}Po. Inventory records of burials from 1949 through 1960 could not be located.

Radiological contaminants ^{3}H and ^{14}C are found in low concentrations throughout the waste where liquid scintillation vials were present, with ^{90}Sr, ^{36}Cl, and ^{226}Ra occurring less frequently.

The maximum ^{3}H concentration in subsurface soils was 140 picoCuries per gram (pCi/g), collected from Disposal Pit 14, which is slightly higher than the NRC guideline screening level of 110 pCi/g. All other results for ^{3}H in soil were 30 pCi/g or less.

The maximum concentration of ^{226}Ra was 7.53 pCi/g, collected from the base of Disposal Pit 34C. Seven of 13 soil samples collected from Disposal Pit 34C exceeded the NRC guideline level of 0.7 pCi/g. Two of 8 samples collected from the reference area also exceeded the NRC guideline level for ^{226}Ra.

The maximum ^{14}C concentration in soil samples was 210 pCi/g, collected from the bottom of Disposal Pit 26. Two other samples for ^{14}C, one collected from Disposal Pit 26 and another collected from Disposal Pit 14 had ^{14}C concentrations of approximately 15 pCi/g. These three results exceed the NRC guideline level of 12 pCi/g, while all other results were below that level.

The maximum ^{90}Sr concentration in soils was 0.278 pCi/g, compared to the NRC guideline level of 1.7 pCi/g. The maximum concentration of ^{36}Cl in soil was 0.11 pCi/g compared to a guideline level 0.36 pCi/g.

The only radionuclides that exceeded RBCs in water samples were radium-228 (^{228}Ra) and thorium-230 (^{230}Th). The maximum ^{228}Ra concentration in unfiltered groundwater was 29.7 picoCuries per liter (pCi/L), compared to a field-filtered aliquot of the same sample, which contained 13.9 pCi/L of ^{228}Ra. The EPA Maximum Contaminant Level (MCL) for ^{228}Ra is 5 pCi/L. ^{230}Th was encountered in groundwater beneath Disposal Pits 1, 14, and 34 at concentrations ranging from 18 pCi/L to 28 pCi/L, compared to an MCL concentration of 15 pCi/L.

The maximum ^{3}H concentration in groundwater beneath the disposal pits was 1,030 pCi/L, compared to the NRC Derived Concentration guideline level of 20,000 pCi/L.

2.9.1 Non-Radiological Contaminants

Non-radiological soil contaminants include the organic chemicals chloroform, benzene, bromodichloromethane, and trichloroethylene and the metals arsenic, chromium, and vanadium.

The maximum concentration of chloroform was 762 micrograms per kilogram (µg/kg) collected from approximately one foot below the ground surface of Disposal Pit 26. The EPA Region 3 Risk Based Concentration (RBC) for chloroform is 1E7 µg/kg. The maximum concentration of benzene was 625 µg/kg from 2 feet below the ground surface of Disposal Pit 34. The RBC for benzene is 52,000 µg/kg. The maximum concentration of bromodichloromethane was 379 µg/kg, collected from 2 feet below the ground surface of Disposal Pit 34. The RBC for bromodichloromethane is 46,155 µg/kg. The maximum concentration of trichloroethylene was 474 µg/kg, collected from 2 feet below the ground surface of Disposal Pit 34. The RBC for trichloroethylene is 7,150 µg/kg.

Arsenic was the most common non-radiological contaminant, often exceeding RBC concentrations, even in the Reference Area. The maximum arsenic result in soil was 59.6 milligrams per kilogram (mg/kg) in a sample from approximately 18 feet beneath the ground surface of Disposal Pit 34. The RBC for arsenic is 1.91 mg/kg. The maximum concentration of chromium in soil was 68 mg/kg, compared to an RBC of 3,066 mg/kg. The maximum concentration for vanadium was 90 mg/kg in a sample from Disposal Pit 34. The RBC for vanadium is 307 mg/kg.

Groundwater that was sampled from beneath the disposal pits had levels of benzene, trichloroethylene, and chloroform that were slightly greater than the RBCs. The maximum aqueous benzene concentration was 1.6 micrograms per liter (µg/L), which was within Disposal Pit 34C. The RBC for benzene is 0.34 µg/L. The maximum trichloroethylene concentration was 0.487 µg/L, compared to the RBC of 0.026 µg/L. The maximum aqueous chloroform concentration measured was 10.6 µg/L, compared to the RBC of 0.15 µg/L.

2.10 Conceptual Site Model

The conceptual site model (CSM) identifies the relationship between the sources of contamination, source areas, transport mechanisms, exposure routes, and the receptor. The CSM provides a description of how contaminants enter into the environment, how they are transported within the environment, and the routes of exposures to humans (CABRERA, 2007b).

Based on historical records provided by the BARC, the top five feet of overburden associated with each disposal pit is considered uncontaminated soil. There is no known surface contamination. Due to previous burial activities within the LLRBS, subsurface soils associated with each disposal pit have the potential for significant radiological and non-radiological contamination (Figure 4). As a part of the decommissioning activities, the upper fifteen (15) feet of clean fill, waste, and contaminated soil present within each disposal pit will be excavated and shipped offsite for disposal from the site. The residual radioactive material in this CSM is defined as contaminated soil present below the bottom of the disposal pit to the depth of groundwater. The thickness of the contaminated zone is assumed to be 10 feet (3.048 meters). Environmental pathways include external gamma radiation, inhalation of suspended dust, ingestion of impacted fruits and vegetables, ingestion of impacted fish, ingestion of impacted groundwater, and ingestion of contaminated soil. Note that RESRAD models migration of contaminants into groundwater and subsequent dispersal though water dependent pathways. This modeling component is included in DCGL calculations. The critical receptor is a resident farmer.

2.11 Decommissioning

As discussed previously, BARC ceased use of the LLRBS in 1987 and as such has terminated licensed activities more than 24 months ago. There is not sufficient verifiable data to demonstrate that, in its current condition, the LLRBS is suitable for release for unrestricted use (i.e., meets the 10 Code of Federal Regulations [CFR] 20 Subpart E unrestricted Radiological Criteria for License Termination). Thus, based on the criteria above, USDA is required to initiate the NRC decommissioning process in accordance with 10 CFR 30.36(d).

The LLRBS EE/CA suggests that the preferred alternative is to perform a removal action that eliminates the source of contamination (i.e., material in the disposal pits) (Entech, 2000). Based on NRC guidance, if removal actions are not currently authorized under an existing license, the licensee must develop a Decommissioning Plan (DP) and submit a request for a license amendment. USDA's NRC radioactive materials license (19-00915-03) authorizes the following uses of radioactive material:

- *Research and development as defined in 10 CFR 30.4 including animal studies; in gauging and measuring devices and in field studies; and,*

- *Studies on human research subjects as approved by a Radioactive Drug Research Committee (RDRC) that has been approved by the Food and Drug Administration (FDA)*

Regarding the need for development of a DP, 10 CFR 30.36(g) (1) states:

> *A decommissioning plan must be submitted if required by license condition or if the procedures and activities necessary to carry out decommissioning of the site or separate building or outdoor area have not been previously approved by the Commission and these procedures could increase potential health and safety impacts to workers or to the public, such as in any of the following cases: (i) Procedures would involve techniques not applied routinely during cleanup or maintenance operations; (ii) Workers would be entering areas not normally occupied where surface contamination and radiation levels are significantly higher than routinely encountered during operation; (iii) Procedures could result in significantly greater airborne concentrations of radioactive materials than are present during operation; or (iv) Procedures could result in significantly greater releases of radioactive material to the environment than those associated with operation.*

The USDA NRC license does not specifically authorize excavation, and packaging of buried radioactive wastes. The facility decommissioning has also been designated as a Group 5 decommissioning in accordance with the requirements of NUREG-1757 Vol. 1, Rev. 1 due to existing groundwater contamination. Thus, the USDA must submit a DP to the NRC and apply for a license amendment to support planned removal actions.

Decommissioning actions will occur concurrent with implementation of this FSS Degisn Plan. The FSS involves:

- Performance of gamma and beta walkover surveys over the surface of the LLRBS;
- Sampling and radiological/chemical analysis of soil at the base of each exhumed disposal pit;
- Subsurface sampling and radiological/chemical analysis of vadose zone soils from the base of each exhumed disposal pit to the underlying aquifer;
- Groundwater sampling in temporary wells set in the base of each exhumed disposal pit;
- Backfill of each exhumed disposal pit with low-permeability soil (to minimize potential for contamination of backfill materials).
- Offsite treatment/disposal of waste materials exhumed from each disposal pit.

Based on data collected during the Characterization Survey, the need to update and resubmit this DP will be evaluated. Remediation of the remainder of the North Field disposal pits will be executed concurrently with implementation of this FSS Design Plan.

3.0 FINAL STATUS SURVEY DESIGN

The purpose of this section is to describe the technical approach to FSS that will be implemented at the Site. Site-specific soil cleanup criteria must be met before the Site may be released for restricted use in accordance with the DCGLs established in the Decommissioning Plan. Federal guidance listed below provides acceptable methodology to demonstrate compliance with project cleanup goals:

- MARSSIM (NRC, 2000)
- NUREG-1505, A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys (NRC, 1998a)

A major component of a survey design is the efficient use of laboratory sampling at distinct locations combined with scan surveys to accurately determine the final status of a given survey unit (SU). The statistical procedures described in this section are used to establish the number of samples taken at distinct locations needed to determine if the median concentration in a given SU exceeds the regulatory limit, with a specified degree of precision. Thus, these statistical procedures are essential in the planning and design of the FSS and the analysis and interpretation of the resulting data.

The survey and sampling approach for the Site described below encompass both sampling at distinct locations, and scanning of the excavations and the undisturbed surface areas. In this manner, both the average concentration and elevated areas of residual radioactive material exceeding the cleanup criteria are addressed.

3.1 Detection Methods

The following radiation detection methods will be used during the radiological surveys:

- Gross beta radioactivity count rate measurements
- Gross gamma fluence (count rate) measurements
- Distinct locations (systematic) soil sampling and off-site laboratory analysis
- Biased location soil sampling and off-site laboratory analysis

Field survey methodology, techniques, and terminology are based on guidance contained in MARSSIM (NRC, 2000). Beta scan surveys do not have calculated MDCs because these surveys are for qualitative use only; beta scan survey data is intended solely for the purpose of locating areas of elevated radioactivity to direct biased sample collection.

3.2 Derived Concentration Guideline Levels

A site-specific risk assessment has been performed and is used to derive cleanup criteria for each of the potential radionuclide contaminants. Some important inputs to the risk assessment include: 1) the radionuclides of concern and their mobility; 2) the volume of contaminated soil and concentrations of contaminants (especially in the vadose zone, because a removal action should virtually eliminate the source term burials); 3) depth to groundwater; and 4) other physical site parameters. Potential risks associated with any chemical contaminants will be evaluated in a separate study, in accordance with applicable state and EPA guidance.

The NRC is the regulatory licensing authority for the site. The NRC has promulgated a primary limit of 25 millirem total effective dose equivalent (TEDE) in any one year, in excess of natural background, for releasing a radiologically contaminated site. This radiological criterion was used in the derivation of soil DCGLs.

DCGLs were derived using dose modeling and the RESRAD code Version 6.3. In the modeling, a residential farmer was considered as the critical receptor for the site. The resident farmer is assumed to move onto the site after the site was released for unrestricted use. The resident farmer builds a home and raises crops on the property for consumption. As a result, the resident farmer will be exposed to the residual radioactive contamination.

The site has both soil and groundwater contamination. RESRAD model determines dose contribution from future groundwater contamination (soil to groundwater), and does not determine doses for existing groundwater contamination. Hence, EPA's standard Risk Assessment Guidance Standard equations and the dose conversion factors present in Federal Guidance Report No 11 were used to determine the dose. Due to presence of limited groundwater concentrations, recent maximum concentrations for ^{14}C and ^{3}H were utilized during the determination of dose for direct ingestion of groundwater. However, the resulted dose was less than 0.2 millirem per year (mrem/yr). Due to the insignificant dose potential caused by the existing groundwater contamination, the soil DCGLs were determined based on the RESRAD model and full 25 mrem/yr dose limit.

In the derivation of the soil DCGLs, soil dose assessments were performed by using a unit concentration of one picocurie per gram (1 pCi/g) for each of the radionuclides of concern, individually. RESRAD version 6.3 model was used for the dose assessments. The output of each RESRAD model run could then be interpreted as an estimate of the dose per unit activity (in mrem/year per pCi/g, or mrem). This is also called a dose-to-source ratio (DSR) that the receptor could receive in a single year from 1 pCi/g of the radionuclide in soil. The primary dose limit was divided by the DSR to yield a DCGL for that radionuclide, in units of pCi/g.

Table 1 presents the proposed site-specific soil single DCGLs for the radionuclides of concern at the Site. Each DCGL represents the concentration (based on the presented model) that would produce 25 mrem/yr.

Table 1: Site-Specific DCGLs

RCOC	Site-Specific DCGL (pCi/g)
Carbon-14	1.62E+02
Chlorine-36	7.10E+00

Cesium-137	1.68E+01
Iron-55	1.07E+06
Tritium	2.64E+02
Sodium-22	4.87E+00
Nickel-63	2.06E+04
Lead-210	1.62E+00
Radium-226	1.34E+00
Strontium-90	3.88E+00

It is assumed that a removal action will eliminate the source term burials. Three-dimensional models are used, in conjunction with the DCGLs established during the risk assessment, to design a remediation protocol that is protective of human health and the environment during and after implementation. As stated previously, data collected during the Characterization Survey exhumation and evaluation of disposal pit contents may be used to revise the remedial design, DCGLs, and cost estimates for decommissioning. It should be noted that implementation of the remedial design will be phased based on funding constraints.

Based on the current design, procedures have been developed to protect workers and the environment. In addition, gamma walkover surveys performed over the surface of the LLRBS support the initial assumption that all radioactive waste and contamination is subsurface.

3.2.1 Sum of the Ratios

When there are multiple radionuclides present in the soil, the allowed soil concentration levels may be evaluated by employing sum of the ratios (SOR). This will ensure that the sum of the individual fractions for each isotope to its individual DCGL fraction does not exceed unity and enables field measurement of a gross activity DCGL. The gross activity DCGL considering the isotopes of concern is described by:

$$\text{Gross Activity DCGL} = \frac{1}{\left(\frac{f_1}{DCGL_1} + \frac{f_2}{DCGL_2} + \ldots + \frac{f_n}{DCGL_n}\right)}$$

where,

f_1 = Fraction of 1st isotope in the soil

$DCGL_1$ = DCGL for the 1st isotope, pCi/g

f_2 = Fraction of 2nd isotope in the soil

$DCGL_2$ = DCGL for the 2nd isotope, pCi/g

f_n = Fraction of nth isotope in the soil

$DCGL_n$ = DCGL for the nth isotope, pCi/g

3.2.2 Elevated Measurement Comparison

For individual measurement values that exceed the appropriate action level within a survey unit, a $DCGL_{EMC}$ (elevated measurement comparison) is calculated as a release criteria that may be utilized for discrete areas that is less stringent but still protective of human health. The $DCGL_{EMC}$ is the $DCGL_{w}$ (DCGL for residual radioactivity evenly distributed over a wide area.) modified using a correction factor (i.e., an area dose factor) to account for the difference in area and the change in dose or risk. An area dose factor "is the magnitude by which the concentration within the small area of elevated activity can exceed $DCGL_{w}$ while maintaining compliance with the release criterion" (NRC, 2000). For example, the action level for ^{226}Ra may be modified for a one square meter outdoor area using the area dose factor of 54.8 listed in MARSSIM Table 5.7, yielding a $DCGL_{EMC}$ of 7.34 E+01 pCi/g (i.e., [1.34E+00 pCi/g] × [54.8]).

The RESRAD computer code may be used to calculate the maximum allowable elevated concentrations pertaining to ROCs not listed in MARSSIM Table 5.6 (Illustrative Examples of Outdoor Area Dose Factors) and Table 5.7 (Illustrative Examples of Indoor Area Dose Factors) based on survey unit size for small areas of elevated activity in buildings/indoor areas or outdoor areas on an as-needed basis.

3.3 Representative Reference (Background) Area

A background reference area is a geographical area from which representative samples of background conditions are selected for comparison with samples collected in specific survey units at the remediated site (NRC, 1998b). The background reference area has similar physical, chemical, radiological, and biological characteristics to the site being remediated, but is not contaminated by site activities (NRC, 1998b). The distribution of measurements in the reference area should be similar to the distribution of measurements in the SUs.

A reference area was established durinwest of the South Field in an area deemed to be free of radiological contaminants during the Characterization Survey. A gamma walkover survey was conducted there to establish background radiological activities. Soil samples were collected from six locations for comparison with samples to be collected in excavations of disposal pits.

3.4 Survey Unit Classification

As discussed in MARSSIM, site areas being final status surveyed should be classified according to their potential for residual radioactivity. This classification process is discussed in detail in MARSSIM Sections 2.2, 4.4, 5.5.2, and 5.5.3.

3.4.1 Non-Impacted versus Impacted

- **Non-Impacted** – Non-impacted areas are those areas identified through knowledge of site history or previous survey information where there is no reasonable possibility (extremely low probability) for residual radioactive contamination.

- **Impacted** – According to MARSSIM, impacted areas have a potential for radioactive contamination based on historical data or contain known radioactive contamination based on past or preliminary radiological surveillance. This includes areas where radioactive materials were used and stored; records of spills, discharges, or other unusual occurrences resulting in the spread of contamination; and areas where radioactive materials were buried or disposed. Areas immediately surrounding or adjacent to these locations are included in this classification due to the potential for the inadvertent spread of contamination.

For the purposes of this FSS, the entire Site is considered impacted because all areas within the North Field area have the potential for residual contamination. Impacted areas are further divided into one of three groups (Class 1, Class 2, and Class 3), as defined by MARSSIM (NRC, 2000).

3.4.2 Class 1, Class 2, and Class 3 Survey Units

- **Class 1** – Class 1 areas are areas that have, or had prior to remediation, known radioactive contamination that exceeds the $DCGL_w$ or a potential for such contamination. For the purposes of this FSS, soil residing at the bottom of the disposal pits from the limit of remedial excavation to the groundwater interface, and interstitial soil residing below a height of six feet below original surface grade between individual disposal pits throughout the North Field area are considered to require a Class 1 FSS.

- **Class 2** – Class 2 areas are areas that have known radioactive contamination or a potential for such contamination, but are not expected to exceed the $DCGL_w$. For the purposes of the FSS, clean cover fill material utilized in disposal pits from five feet below surface grade to six feet below surface grade or approximately the soil from five to six feet below original surface grade throughout the North Field area are considered to require a Class 2 FSS. This classification is warranted because this material may coincide with the top layers of waste material in the disposal pits, which presents an increased potential for radiological contamination from soils within this interval.

- **Class 3** – Class 3 areas are impacted areas that are not expected to contain any residual radioactivity, or are expected to contain levels of residual radioactivity at a small fraction of the $DCGL_w$, based on site operating history and previous radiological surveys. For the purposes of this FSS, clean cover fill material utilized in disposal pits from original surface grade to five feet below original surface grade or approximately the top five feet of soil throughout the North Field area are considered to require a Class 3 FSS.

The suggested maximum survey unit size for the variously classified areas is given in Table 2. These areas are suggested because they give a reasonable sampling density. However, the size and shape of a particular survey unit may be adjusted to conform to the actual shape and size of the excavations. It should be noted that several separate excavations may be combined to comprise a single Class 1 survey unit as long as the 2,000 square meters (m^2) size limit is maintained.

Table 2: Suggested Land Area Survey Unit Size

Classification of Area	Suggested Survey Unit Size

Class 1	Up to 2,000 m^2
Class 2	2,000 to 10,000 m^2
Class 3	No limit

3.5 Establish the Survey Reference Coordinate System

The contractor will establish a 20-meter square reference grid tied to the Maryland State Plane coordinate system. The corners of SUs may be marked and identified to facilitate the beta and gamma walkover survey process. The use of a differentially-corrected global positioning system (DGPS) obviates the need for marking smaller grid intervals.

3.6 Identify Survey Units

The Site has three types of survey units:

- Class 1 (soil from disposal pit bottoms to 15 feet below original surface grade and interstitial soil residing below a height of six feet below original surface grade between individual disposal pits throughout the North Field area)
- Class 2 (soil from five to six feet below original surface grade)
- Class 3 (soil from original surface grade to five feet below original surface grade)

Since the limits of the final disposal pit excavations are unknown prior to remediation activities, each excavation area will be evaluated for size and subdivided into suitable MARSSIM-compliant Class 1 survey units before initiating the final status surveys. Several areas of excavation together totaling 2,000 m^2 will be combined to comprise a single Class 1 survey unit.

3.7 Number of Sample Locations and Survey Coverage

MARSSIM discusses a method to determine the number of data points required in a given SU. A minimum number of measurement locations are required in each SU to obtain sufficient statistical confidence that the conclusions drawn from the measurements are correct. The following subsections describe the basis for and derivation of the minimum required measurement locations per SU.

3.7.1 Estimation of Relative Shift

The minimum number of measurements required is dependent on the distribution of site residual radionuclide concentrations relative to the $DCGL_w$ and acceptable decision error limits (α and β) (established in Section 4.6).

The relative shift (Δ/σ) describes the relationship of site residual radionuclide concentrations to the $DCGL_w$ and is calculated using the following equation, from Section 5.5.2.3 of the MARSSIM.

$$\Delta/\sigma = \frac{DCGL_w - LBGR}{\sigma}$$

Where: $DCGL_w$ = the derived concentration guideline level (i.e., release limit)

LBGR = concentration at the lower bound of the gray region (LBGR) (the LBGR is the concentration to which the SU must be remediated in order to have an acceptable probability of passing the statistical tests; the LBGR effectively becomes the survey's action level); for conservatism, the LBGR will be set to 0.5 times the $DCGL_w$ for this FSS

Δ = The width of the gray region, i.e., $DCGL_w$-LBGR

σ = estimate of the standard deviation of the concentration of residual radioactivity in the SU (which includes real spatial variability in the concentration as well as the precision of the measurement system); σ is estimated as 0.3 times the $DCGL_w$

Using a $DCGL_w$ equal to an SOR of 1, an LBGR of 0.5 (half the $DCGL_w$), and a sigma equal to 0.3, the relative shift is calculated to be 1.67.

3.7.2 Determination of N (Number of Required Measurement Locations)

The Wilcoxon Rank Sum (WRS) statistical test will be used to determine when survey unit s are suitable for release for unrestricted use, according to the DCGLs. The minimum number of systematic measurement locations required in each survey unit for the WRS statistical test can be determined using Table 5.3 in MARSSIM (NRC, 2000). Section 4.6 establishes the acceptable decision errors for the survey units as $\alpha = \beta = 0.05$. Based on the relative shift established above in Section 3.7.1 and these decision errors, the estimated minimum number of required measurement locations is 16. MARSSIM recommends an additional 20% to protect against lost or unusable data. Therefore, a minimum of 20 sample locations is required in each Class 1 survey unit (disposal pit bottoms and interstitial soil), each Class 2 survey unit (soil from five to six feet below original surface grade), each Class 3 survey unit (soil from original surface grade to five feet below original surface grade), and the reference area.

3.7.3 Additional Samples to Meet Elevated Measurement Comparison Criteria

Since the ROCs include ^{3}H and ^{14}C which cannot be readily identified using scan survey techniques, the sample density will be increased for MARSSIM Class 1 areas. The appropriate number of samples could become extremely large under this scenario. Therefore, to adequately characterize concentrations of ^{3}H and ^{14}C, a more practical but still revealing sample density has been applied. It has been determined that approximately 100 sample locations distributed throughout the North Field excavations would provide adequate sample density.

3.7.4 Biased Samples

Beta and gamma walkover survey data will be used to identify areas of elevated radioactivity. Biased soil samples will be collected at locations where beta and gamma walkover survey data z scores (i.e., number of standard deviations from the average; one z score equals one standard deviation) exceed 3.0 and will be analyzed in an off-site laboratory . These samples will be evaluated using DCGLs and the SOR calculation. However, these samples are not assessed using the WRS statistical test since they violate the random criteria used to establish systematic sampling locations.

Although it is difficult to estimate the total number of biased samples that will be required, it is assumed that approximately two biased samples will be collected and analyzed as part of the FSS from each SU. At a minimum, one biased sample will be collected in each SU at the location of the highest beta and at the location of the highest gamma walkover measurement.

3.7.5 Specify Sampling Locations

Field personnel will mark the perimeter of each SU using DGPS. Actual SU dimensions will be measured in the field and are contingent upon the extent of excavations performed in the course of remediation. This data will then be transferred into the geographic information system (GIS) so that a triangular sampling grid can be established. A random start point will be generated and systematic sample locations will be calculated in an equilateral triangular grid pattern using the spacing given by the equation shown on the following page (Equation 5-5 from MARSSIM).

$$L = \sqrt{\frac{A}{0.866 \times N}}$$

Where:	L	=	triangular grid spacing for SU
	A	=	area of SU
	N	=	number of sample locations

After the systematic sample locations have been established, soil sample locations will be marked in the field using a DGPS, prior to sample collection.

Measurement locations in Class 3 survey units have been established using a randomization process to arbitrarily assign locations in a survey unit based on the total area. Random measurement patterns are used for Class 3 survey units to ensure that the measurements are independent and support the assumptions of the statistical tests (NRC, 2000).

3.8 Survey Instrumentation

Equipment required for performing the beta and gamma walkover survey includes the following:

- DGPS Rover: Trimble Pathfinder Pro-XRS (or equivalent)

- Ludlum 44-89 square cluster pancake Geiger-Mueller detector (GM) and associated ratemeter/scaler, equipped with RS-232 download port

- Bicron G-5 five-inch by 0.063-inch sodium iodide (NaI) scintillation detector (field instrument for the detection of low energy radiation [FIDLER]) and associated ratemeter/scaler, equipped with RS-232 download port

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

- Software: Trimble Pathfinder Office, GIS software with coordinate geometry capability

The beta and gamma walkover surveys will be performed following MARSSIM protocol by walking straight parallel lines over an area at a rate of approximately 0.5 meters per second while moving the GM/FIDLER in a side to side, serpentine motion, 0.05 to 0.10 meters (two to four inches) above the ground surface. Survey passes will be approximately 1 meter apart. Data from the ratemeter/scaler will be automatically logged into the DGPS unit at the rate of once per second.

After completion of the survey, the raw data will be downloaded from the DGPS and processed for export into a geospatial software program. The processed data with the contoured results of the survey will then be evaluated.

3.9 Soil Sampling

Soil samples will be collected using hand augers or equivalent equipment and will be collected from depths of zero to 0.15 meters (zero to six inches) in all SUs. The following equipment (or equivalent) will be required for this task.

- Hand auger, hand trowel, or shovel

- Large stainless steel mixing bowl

- Stainless steel utensil for removal of soil core from hand auger after sample is retrieved and for mixing and packaging samples in containers

- Sample containers and chain of custody forms/seals

Soil samples will be extruded from the ground and transferred into a stainless steel bowl where they will be thoroughly mixed or homogenized. Visually identifiable non-soil components such as stones, twigs, and foreign objects will be manually separated in the field and excluded from the laboratory samples to avoid biasing results low. Samples will not be preserved in the field, as there are no preservation requirements for the radiological analyses.

Hand augers/hand trowels/shovels, mixing utensils, and homogenizing bowls will be decontaminated between samples to avoid cross-contamination. Decontamination will be performed by rinsing with clean potable water and returning the rinsate to the ground surface in the location where the sample was collected. After tools and equipment are dried, they will be frisked for contamination before being used to collect additional samples.

Systematic, Bias, and Duplicate soil samples will be numbered, logged, and transferred, under applicable chain of custody procedures to an off-site laboratory for analyses. Field duplicate samples will be collected at the frequency of 5%, as specified in Section 6.3. Soil samples will be submitted to a quality assurance (QA) laboratory. All soil samples will be analyzed for ^{14}C (method EERF C-01 mod), ^{36}Cl (method GL-Rad-A033), ^{63}Ni (method RESL Ni-1 mod), ^{90}Sr (method 905.0 mod), ^{3}H (method 906.0 mod), and gamma spectroscopy scan (method 901.1 mod/HASL 300) for ^{226}Ra, ^{210}Pb, ^{137}Cs, and other gamma emitting isotopes.

3.10 Geoprobe™ Direct Push Sampling

To characterize the deeper subsurface vadose zone soils and groundwater below each disposal pit, a Geoprobe™ direct push sampling system (or equivalent) will be lowered into the disposal pit and core samples will be collected from the center of the disposal pit in four-foot lengths until groundwater is reached. (Alternatively, an angled Geoprobe™ boring will be advanced from an area outside the LLRBS so that the screened interval is beneath each of the exhumed disposal pits).

3.11 Final Status Survey Methodology Summary

The FSS consists of the following steps:

- The North Field area will be surveyed via beta and gamma walkover survey utilizing the beta and gamma radioactivity detection equipment described above and a MARSSIM Class 3 level of survey effort.

- Soil will then be excavated from the top five feet of clean cover fill material in one foot lifts. As each one-foot-lift is excavated, the newly-exposed ground surface will be surveyed via beta and gamma walkover survey. Beta and gamma walkover surveys will be performed at the undisturbed, original ground surface, then at one foot below original surface grade, then at two feet below original surface grade, all the way down to five feet below original surface grade.

- Systematic soil samples will be collected from each Class 3 soil lift at a rate of four samples per lift (20 samples per survey unit divided by five one-foot lifts). MARSSIM does not address depth and sampling per unit volume; given that there is no size limitation for land area Class 3 survey units, the total amount of Class 3 soils will comprise a single Class 3 survey unit. The volume of soil that comprises this survey unit is estimated at 5,555 cubic yards (the North Field area is approximately 150 × 200 feet = 30,000 square feet or 3,333 square yards × 1.67 yards depth [i.e., 5 feet depth] = 5,555 cubic yards of soil).

- Once the fifth one-foot lift is removed, the level of survey effort will be increased from MARSSIM Class 3 to MARSSIM Class 2 from the interval of five to six feet below original surface grade. This classification is warranted because this material may coincide with the top layers of waste material in the disposal pits, which presents an increased potential for radiological contamination from soils within this interval.

- Systematic soil samples will be collected at a rate of 20 samples per Class 2 soil lift (i.e., 20 samples per survey unit).

- The exposed disposal pits will then be excavated and the wastes removed. The removal and disposal of these wastes is outside of the scope of this document.

- Once the waste materials have been removed from a given disposal pit, the interstitial soil residing below a height of six feet below original surface grade between individual disposal pits will be excavated and transported to a laydown area where it will be laid out in six-inch lifts and be surveyed via beta and gamma walkover survey with a MARSSIM Class 1 level of survey effort.

- Systematic soil samples will be collected from Class 1 interstitial soils at a rate of one sample for every 398 cubic yards of soil; given that MARSSIM does not address depth and sampling per unit volume, this volumetric sampling rate is approximated by multiplying the six-inch thickness of the Class 1 lifts by the MARSSIM recommended size limitation for a Class 1 area (2,000 m^2 or 2,392 square yards × 0.167 yards depth [i.e., six inches depth] = 398 cubic yards of soil).

- Once all interstitial soils have been excavated and transported to the laydown area, the excavated areas within the North Field area will represent a single, large excavation. Any material remaining of the upper fifteen (15) feet of material below original surface grade (consisting of clean fill, waste, contaminated soil, and interstitial soil present within each disposal pit) will be excavated and removed.

- This entire excavated area will be surveyed via beta and gamma walkover survey with MARSSIM Class 1 level of survey effort.

- Systematic soil samples will be collected from the entire floor of the excavated area at the rate of approximately one soil sample per 100 m^2.

- The perimeter of the excavation will be surveyed with a DGPS to verify the real extent of the excavation. If it is much different than originally planned, then the location of the systematic soil samples need to be re-determined.

- Consider a Class 1 SU complete when at least 20 samples have been collected within a contiguous series of excavations and the total area does not exceed 2,000 m^2.

- Conduct the beta and gamma walkover survey in accordance with MARSSIM protocol: straight parallel lines approximately 1 meter apart while moving the GM/FIDLER in a side to side, serpentine motion, 0.05 to 0.10 meters above the ground surface or water surface.

- Process beta and gamma walkover survey data, evaluate data, and select biased sample locations, as appropriate.

- Collect biased soil samples and submit to laboratory for analysis.

- Collect systematic soil samples and submit to laboratory for analysis.

- To characterize the deeper subsurface vadose zone soils and groundwater below each disposal pit, a Geoprobe™ direct push sampling system (or equivalent) will be lowered into the disposal pit and core samples will be collected from the center of the disposal pit in four-foot lengths until groundwater is reached. (Alternatively, an angled Geoprobe™ boring will be advanced from an area outside the LLRBS so that the screened interval is beneath each of the exhumed disposal pits).

- Four-foot Geoprobe™ core sections will be advanced until the groundwater interface is reached. Each four-foot Geoprobe™ core section will be surveyed for beta and gamma radiation and one soil sample will be collected from the location exhibiting the most elevated activity from the four soil cores. A composite sample will be collected from the top foot of the first soil core, comprising the floor of the disposal pit. Another sample will be collected at the soil interface with groundwater. Therefore it is anticipated that three subsurface soil samples from each disposal pit will be selected and submitted for laboratory analysis. It is anticipated that the water table will be encountered approximately 25 feet below original surface grade in the vicinity of LLRBS and that the floor of each disposal pit will be encountered approximately 10 feet below original surface grade.

- Filtered and unfiltered groundwater samples will be collected from a temporary Geoprobe™ well.

- When analytical results are received from laboratory, evaluate and interpret data.

4.0 DATA QUALITY OBJECTIVES

Data Quality Objectives (DQOs) are qualitative and quantitative statements that establish a systematic procedure for defining the criteria by which data collection design is satisfied in order to make determinations regarding non-excavated soil following remedial activities. The DQOs at the Site include:

- Clarifying the project problem
- Defining the data necessary for achieving the end use decisions
- Determining the appropriate method of data collection
- Specifying the level of decision errors acceptable for establishing the quantity and quality of data needed to support the project decisions

The overall QA objective for this project is to develop and implement procedures for obtaining and evaluating data that meet the DQOs to ensure that the required remediation is accomplished. Specifically, radionuclide data will be generated to demonstrate that the remedial effort has achieved the DCGLs. QA procedures are established to ensure that field measurements, sampling methods, and analytical data provide information that is comparable and representative of actual field conditions, and that the data generated are technically defensible.

To determine the project DQOs, a series of planning steps are used, as specified in the *EPA Guidance for Data Quality Objective Process QA/G-4* (EPA, 2000), to identify the data needed to support project decisions and develop a data collection program. The process is intended to be iterative, optimizing data collection to meet the applicable decision criteria. The six steps, as applied to the Site, are detailed in Sections 4.1 through 4.6.

4.1 Step 1: State the Problem

(A) Problem Description

The problem is the presence of radioactive material due to disposal activities between 1949 and 1987. The Site ROCs are ^{3}H, ^{14}C, ^{22}Na, ^{32}P, ^{36}Cl, ^{55}Fe, ^{63}Ni, ^{90}Sr, ^{137}Cs, ^{210}Pb, ^{226}Ra, and potentially other radionuclides.

(B) Primary Decision Maker

The ultimate decision regarding disposition of the Site rests with the USDA Beltsville Agricultural Research Center, in consultation with NRC.

4.2 Step 2: Identify the Decision

(A) Principal Study Question

Do ROC concentrations at the Site exceed the DCGLs?

(B) Decision Statement

Determine whether ROC concentrations at the Site exceed the DCGLs and whether the ROC concentrations in excess of background meet the SOR criteria of less than or equal to 1.0 following remedial activities.

4.3 Step 3: Identify Inputs to the Decision

The objective of the survey and sampling activities is to demonstrate that residual radioactivity in each survey unit, following remediation, satisfies the predetermined DCGLs. This section lists the data needs, describes the sources of that data, and discusses the means of obtaining the required data to resolve the decision statement listed in Section 4.2.

(A) Information Inputs

The required information is the concentration of residual radioactive material in the sidewalls and bottoms of excavated areas, and in the disposal pit covering materials removed prior to excavation of the disposal pits. This information will allow determination as to whether or not the survey units are suitable for unrestricted release in accordance with the DCGLs. Obtaining this data will facilitate cost effective decision-making about the project's direction and duration.

(B) Information Sources for Above Listed Items

Decisions will be based on the data received from a combination of scan survey and soil sampling events including beta and gamma walkover surveys and off-site laboratory analytical results.

4.4 Step 4: Define the Study Boundaries

(A) Population of Interest Defining Characteristics

The populations of interest for the Site are the concentration of ROCs and their associated SORs in Class 1 areas (soil from disposal pit bottoms to 15 feet below original surface grade and interstitial soil from between individual disposal pits), Class 2 areas (soil from five to six feet below original surface grade), and Class 3 areas (soil from original surface grade to five feet below original surface grade). The population will be subdivided, as necessary, through the use of survey units.

(B) Spatial Boundaries of the Decision Statement

The spatial boundaries of this project are horizontally limited to the land area within the North Field of the Site. Vertically, the boundaries are limited to the bottoms of the exhumed disposal pits, 15 feet below original ground surface, and the groundwater interface (approximately 25 feet below original ground surface).

(C) Temporal Boundaries of the Decision Statement

(1) Timeframe to which the decision applies

DCGL values are based on dose to an average member of the critical group over a 1,000-year period following the study. The critical group is defined as a group of individuals expected to receive the greatest exposure to residual radioactivity for any applicable set of circumstances (NRC, 2000).

(D) Scale of Decision Making

Decisions will be made on a survey unit basis as to whether or not the concentrations of ROCs and their associated SORs are less than 1.0.

(E) Constraints on Data Collection

Data collection activities can be constrained due to excessive moisture or rain, which can have an adverse effect on field instrumentation and soil sample collection. Extreme cold, extreme heat, or dramatic temperature change within a single day may also inhibit data collection due to its effects on both equipment and project personnel. Additional constraints include the remediation schedule, the roughness of the fill materials in sidewalls and bottoms of the excavations, non-uniformity and layering of fill material, and the orientation of survey instruments within the excavation.

4.5 Step 5: State the Decision Rules

(A) Parameters of Interest

Parameters of interest are the mean, median, and standard deviation of ROC results collected during the FSS. Decisions will be made according to the decision rules stated in (E) of this section.

(B) Scale of Decision Making

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

(C) Action Levels

Decisions on a SU’s acceptability for release according to the DCGLs are based on two primary criteria: results of the beta and gamma walkover surveys and off-site laboratory results of soil samples. Inputs to this decision are intended to avoid unnecessary analytical and/or remediation efforts, while also ensuring that project DQOs are met.

(D) Decision Inputs

Geospatial modeling of position-correlated beta and gamma walkover survey data will provide a graphical view of surface gamma radiation levels and will be updated as the survey progresses. This will serve as the primary decision input during performance of the fieldwork for identifying areas of elevated activity and selecting locations for collecting biased soil samples. Off-site laboratory soil sample results will be used for statistical testing.

(1) Field Measurements of Survey Unit Dimensions

The dimensions of the SUs will be determined using DGPS data, downloaded and interpreted in the GIS. At a minimum, the corners of the SUs will be logged using the DGPS system. The area of each SU will then be calculated in units of m^2. DGPS/GIS will be used to determine grid spacing and sample locations (see Section 3.7.5), and ensure that SUs do not exceed the maximum sizes recommended by MARSSIM.

(2) Beta and Gamma Walkover Survey in the Reference Area

Soon after the reference area is established, a beta and gamma walkover survey will be performed. These data will be used, in part, to evaluate and make decisions regarding the radiological status of the Site SU data. Reference area beta and gamma walkover survey data will be reduced and evaluated as follows:

- The average and standard deviation of the beta and gamma walkover survey data will be calculated.
- The measurement results will be plotted and color-coded for visual review and evaluation.
- The "z score" for each data point (i.e., number of standard deviations from the average; one z score equals one standard deviation) will be plotted and color-coded for visual review and evaluation.
- These data will be reviewed for obvious anomalies to determine if the chosen reference area is acceptable (i.e., non-impacted).

(3) Beta and Gamma Walkover Survey in Site Survey Units

The beta and gamma walkover survey of SUs will begin following the beta and gamma walkover survey in the reference area, determination that the reference area is acceptable, and calculation of the mean and standard deviation. The beta and gamma walkover survey data will be reduced and evaluated as follows:

- The measurements will be plotted and color-coded for visual review and evaluation. The average and standard deviation of each SU will also be calculated. The coordinates of the highest measurement will be identified.
- The z score for each data point will be plotted and color-coded for visual review and evaluation. A z score tells how many standard deviations above the mean (positive) or below the mean (negative) a number falls. All areas exceeding three standard deviations above the average (i.e., the z score is equal to or greater than 3.0) will be identified. The frequency of these occurrences and the maximum measurement in these areas will be compared to the reference area. The

geospatial plot will also be visually inspected to identify anomalies in the distribution of measurement data.

- Z score results will be used for the selection of biased soil sample locations.

(4) Off-site Laboratory Sample Results

Statistical testing of off-site laboratory sample results includes a series of calculations. First the gross SOR is calculated. If the sample fails this, then background is subtracted from the SOR to produce a Net SOR. If the sample still fails the Net SOR, then the $DCGL_{EMC}$ is calculated based on an area factor. The Net SOR_{EMC} is calculated by subtracting the background from the $DCGL_{EMC}$, if it fails. Finally, if the sample fails all of the above calculations, then a W_r is calculated. The following pages discuss these calculations in greater detail.

Sum of the Ratios

Typically, each radionuclide $DCGL_w$ corresponds to the release criterion (e.g., regulatory limit in terms of dose or risk). However, in the presence of multiple radionuclides, the total of the $DCGL_w$s for all radionuclides could exceed the release criterion. In this case, the individual $DCGL_w$s need to be adjusted to account for the presence of multiple radionuclides contributing to the total dose. While there are several methods used for adjusting the $DCGL_w$s, the one used for this final status survey is the unity rule or SOR. The SOR is satisfied when radionuclide mixtures yield a combined fractional concentration limit that is less than or equal to one.

When multiple contaminants are present on a site, site radiological conditions are evaluated using the SOR = 1.0. The SOR is calculated as follows:

$$SOR = \frac{^{226}Ra_{Conc}}{^{226}Ra_{DCGLw}} + \frac{^{3}H_{Conc}}{^{3}H_{DCGLw}} + \frac{^{14}C_{Conc}}{^{14}C_{DCGLw}} + \frac{^{137}Cs_{Conc}}{^{137}Cs_{DCGLw}} + \ldots$$

Where:

$^{226}Ra_{Conc}$ = Measured activity concentration for ^{226}Ra

$^{3}H_{Conc}$ = Measured activity concentration for ^{3}H

$^{14}C_{Conc}$ = Measured activity concentration for ^{14}C

$^{137}Cs_{Conc}$ = Measured activity concentration for ^{137}Cs

$^{226}Ra_{DCGLw}$ = Site-Specific $DCGL_w$ for ^{226}Ra

$^{3}H_{DCGLw}$ = Site-Specific $DCGL_w$ for ^{3}H

$^{14}C_{DCGLw}$ = Site-Specific $DCGL_w$ for ^{14}C

$^{137}Cs_{DCGLw}$ = Site-Specific $DCGL_w$ for ^{137}Cs

Net SOR

For every sample location within a SU where the SOR is greater than 1, the Net SOR for that sample location will be calculated. Net SOR considers the concentration of each radionuclide less its respective background concentration (as established by the reference area soil sample results) divided by the $DCGL_w$ established for the radionuclide, as follows:

$$\text{Net SOR} = \sum_i \left[\frac{(\text{Conc}_i - \text{B}_i)}{DCGL_{wi}} \right]$$

Where:

Conc_i = concentration of ROC "i" present in the soil sample

B_i = average concentration of ROC "i" present in the reference area

$DCGL_{wi}$ = $DCGL_w$ of ROC "i"

Elevated Measurement Comparison -- $DCGL_{EMC}$ and Net SOR_{EMC}

For every sample location within a SU where the Net SOR is greater than 1, an elevated measurement comparison is made.

The $DCGL_{EMC}$ is calculated as follows:

$$DCGL_{EMC} = DCGL_w \times \text{Area Factor}$$

Net SOR_{EMC} is calculated using the $DCGL_{EMC}$ as follows:

$$\text{Net SOR}_{EMC} = \sum_i \left[\frac{(\text{Conc}_i - \text{B}_i)}{DCGL_{EMC}} \right]$$

W_r

Comparison of reference area (background) radionuclide concentrations with SU concentrations will be performed using the two-sample WRS statistical test. This test is selected because some ROCs are present in natural background. The two-sample WRS statistical test assumes the reference area and SU data distributions are similar except for a possible shift in the medians.

When the data are severely skewed, the value for the mean difference between SU measurements and reference measurements may be above the $DCGL_w$, while the median difference is below the $DCGL_w$. In such cases, the SU does not meet the release criterion regardless of the result of the statistical test. On the other hand, if the difference between the largest SU measurement and the smallest reference area measurement is less than the

$DCGL_w$, then the WRS statistical test will always show that the SU meets the release criterion.

In using this test, the hypotheses being tested are:

Null Hypothesis (H_0): The median concentration in the survey unit exceeds that in the reference area by more than the DCGL

versus the alternative

Alternative Hypothesis (H_a): The median concentration in the survey unit exceeds that in the reference area by less than the DCGL

The WRS statistical test will be applied to the laboratory sample data via the following sequential steps:

a. Reduce reference area and SU isotopic data to SOR using the equation presented in Section 3.2.1.

b. Add the $DCGL_w$ value (i.e., 1.0) to each reference area SOR value, X_i, to obtain the adjusted reference area SOR, Z_i, where

$$Z_i = X_i + 1.0$$

c. The *m*-adjusted SOR, Z_i, from the reference area and the *n* SOR, Y_i, from the SU are pooled and assigned a rank in order of increasing measurement value from *I* to *N*, where

$$N = m + n$$

d. If several SORs are equal (e.g., have the same value), then they are all assigned the average rank of that group of tied measurements.

e. Sum the ranks of the adjusted SOR from the reference area, W_r. Since the sum of the first *N* integers is $N(N + 1)/2$, one can equivalently sum the ranks of the SOR from the SU, W_s, and calculate

$$W_r = \frac{N(N+1)}{2} - W_s$$

f. Compare W_r with the critical value given in Table 1.4 in MARSSIM for the approximate values of *n*, *m*, and α. If W_r is greater than the tabulated value, then reject the null hypothesis that the SU exceeds the release criterion.

(E) Decision Rules

(1) Field Measurements of SU Dimensions

If the measured dimensions of a Class 1 SU exceed the 2,000 m^2 maximum recommended by MARSSIM, then the SU boundaries will be adjusted accordingly. If the measured dimensions of a Class 2 SU exceed the 10,000 m^2 maximum, then the SU boundaries will be adjusted accordingly. MARSSIM has no recommended size limitation for Class 3 survey units.

(2) Beta and Gamma Walkover Survey in the Reference Area

If review of the reference area beta and gamma walkover survey data indicates that the chosen area exhibits excessive variance or appears to be impacted by radiological or non-radiological activities, then biased soil samples will be collected and analyzed to support the area's non-impacted designation.

(3) Gamma Walkover Survey in the Site SUs

a. A biased soil sample will be collected at the location where the highest beta and gamma walkover survey data point is observed.

b. If areas exceeding three standard deviations above the average are observed (i.e., the z score is equal to or greater than 3.0), additional biased soil samples may be collected at the discretion of the Project Manager or designee.

(4) Laboratory Sample Results

The following decision rules for off-site laboratory sample results are adapted from MARSSIM Table 8.2 "Summary of Statistical Tests" for radionuclides in background (NRC, 2000).

a. If the difference between the largest SU SOR and the smallest reference area SOR is less than one, then the SU meets the DCGLs, or:

$$\max SOR_{SU} < 1 + \min SOR_{RefArea}$$

b. If the difference between the average SU SOR and the average reference area SOR is greater than one, then the SU fails, or:

$$\text{avg } SOR_{SU} > 1 + \text{avg } SOR_{RefArea}$$

c. If the difference between the maximum SU SOR and the minimum reference area SOR is greater than one and the difference between the averages is less than one, then the following steps shall be taken. Net SOR will be calculated. If the Net SORs are less than one, then the WRS test will be utilized to determine if the SU meets the DCGLs. If any Net SORs are greater than one, then the SOR_{EMC} will be calculated and the area will be evaluated against EMC criteria.

The purpose of the statistical testing is to determine if the null hypothesis should be accepted or rejected. Rejection of the null hypothesis leads to the determination that the SU meets the release criteria; however, in the case where isolated areas of elevated activity trigger elevated measurement comparisons, a final check will be used to ensure that the total dose is within the release criteria by utilizing MARSSIM equation 8-2 (NRC, 2000):

$$\left(\frac{{}^{226}Ra_{Ave} - {}^{226}Ra_{REFArea}}{DCGL_{W,Ra226}}\right) + \left(\frac{{}^{3}H_{Ave}}{DCGL_{W,H3}}\right) + \left(\frac{{}^{14}C_{Ave} - {}^{14}C_{REFArea}}{DCGL_{W,C14}}\right) + \left(\frac{{}^{137}Cs_{Ave} - {}^{137}Cs_{REFArea}}{DCGL_{W,Cs137}}\right) + \ldots$$

$$+$$

$$\left(\frac{{}^{226}Ra_{EMC} - {}^{226}Ra_{Ave} - {}^{226}Ra_{REFArea}}{(AreaFactor)(DCGL_{W,Ra226})}\right) + \left(\frac{{}^{3}H_{EMC} - {}^{3}H_{Ave}}{(AreaFactor)(DCGL_{W,H3})}\right) + \left(\frac{{}^{14}C_{EMC} - {}^{14}C_{Ave} - {}^{14}C_{REFArea}}{(AreaFactor)(DCGL_{W,C14})}\right) + \left(\frac{{}^{137}Cs_{EMC} - {}^{137}Cs_{Ave} - {}^{137}Cs_{REFArea}}{(AreaFactor)(DCGL_{W,Cs137})}\right) + \ldots$$

4.6 Step 6: Define Acceptable Decision Errors

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

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

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

The first type of decision error, called a Type I error, occurs when the null hypothesis is rejected when it is actually true. A Type I error is sometimes called a "false positive." The probability of a Type I error is usually denoted by α. Considered in light of H_0 used for this site (discussed above), this error could result in higher potential doses to future site occupants than prescribed by the dose-based criterion.

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

For the purposes of this FSS, the acceptable error rate for Class 1, Class 2, and Class 3 survey units for both Type I and Type II errors is five percent ($\alpha = \beta = 0.05$). The classification system used in MARSSIM surveys is described in Section 3.4.2.

5.0 METHODOLOGY AND APPROACH TO PERFORMING SURVEYS

5.1 Estimated Scan Sensitivity of the Gamma Walkover Survey

MARSSIM Section 6.7.2.1 describes the methodology used to calculate the scan MDCs for land areas that are delineated in MARSSIM Table 6.7. The approximate detection sensitivity of the gamma walkover survey is 4.3 pCi/g for ^{137}Cs and 1.7 pCi/g for ^{226}Ra. The detection sensitivities for ^{226}Ra is modeled with 40 years in-growth of associated progeny. This methodology is based on a scan speed of 0.5 meters per second and a minimum contaminated area 56 centimeters in diameter, 15 centimeters in depth, and assumes saturated soil. The gross gamma walkover scan survey in this plan was designed using these parameters (NRC, 1997). Appendix A provides details regarding the calculations used to determine the scan MDCs for ^{137}Cs and ^{226}Ra.

5.2 DGPS Unit and Data File Setup

A DGPS will provide high quality, precision geospatial positioning data to support the final status survey data verification and remediation. The DGPS unit will perform data logging functions and will be configured to record the data output of the ratemeter/scalers at least every two seconds. This beta walkover survey/gamma walkover survey/data logging protocol will provide a minimum data density of one measurement per square meter of ground surface.

In order for the DGPS unit to achieve sub-meter accuracy, differential position correction is necessary. Each technician will carry his/her own rover unit operating both the detector and DGPS (making entries into the DGPS and checking detector responses).

Each survey will be designed to optimize the data collection procedure, taking into account the SU's configuration, hazards, and other obstructions. Copies of the base map on which temporary structures, roads, or other major features have been located will be available on-site. Technicians will annotate copies of the base map with information relevant to the survey, as appropriate. Each survey will be assigned a SU number and date of collection.

5.3 Survey Limitations of DGPS

Although the DGPS unit identifies its position using the signals from several satellites, DGPS positioning may be affected by overhead obstructions during the course of survey. A loss of satellite signal due to these obstructions may prevent collection of location data, depending on the severity of the loss and the positional filter settings in use in the DGPS unit. If this occurs, data collection will not resume until satellite lock is regained (usually by moving past the obstruction) and the positional filter requirements are satisfied. If the signal is lost during a survey, the operator shall continue to walk at constant velocity in a straight line until satellite lock has been reestablished or until a boundary is reached. In such cases, due to positional filter settings in the DGPS unit, no gamma logging occurs. At the discretion of the Project Manager or designee, logging filters may be temporarily overridden to enable gamma logging without DGPS lock for brief periods. In such cases, it is especially important to proceed at a constant velocity so that locations for data collected in this manner can be interpolated or extrapolated. The surveyor will need to inform the data processing specialist if the gamma count rates between pairs of DGPS positional data changes considerably. Such information will be logged in project logs, as appropriate.

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

5.4 Scan Survey Coverage for Class 1, Class 2, and Class 3 Survey Units

Section 5.5.3 in MARSSIM discusses the recommended scanning survey coverage for land areas. The objective of a scanning survey is to identify locations within the SU that exceed the cleanup level. These areas of elevated readings are marked for additional investigation. For Class 1 areas, scanning surveys are designed to detect small areas of elevated activity that are not detected by the soil samples collected using the systematic pattern. To achieve this goal, a 100% beta and gamma walkover survey will be performed over all Class 1 soils. Class 2 SUs have a lower probability for areas of elevated activity than Class 1 SUs, but some portions of the SU may have a higher potential than others. Judgmental scanning coverage will be utilized in Class 2 areas to focus on the portions of the SU with the highest probability for areas of elevated activity and provide beta and gamma walkover survey coverage between 10% and 100%. If the entire SU has an equal probability for areas of elevated activity, or the judgmental scans don't cover at least 10% of the area, systematic scans along transects of the SU or scanning surveys of randomly selected grid blocks are performed. Class 3 survey units have the lowest potential for areas of elevated activity, and scan surveys are typically conducted in areas with the highest potential for contamination (e.g., corners, ditches, drains) based on professional judgment. Class 3 survey units will maintain a minimum of 10% beta and gamma walkover survey scan coverage.

5.5 Final Status Survey Data Evaluation

All data collected as part of the final status survey of the Site will be evaluated in accordance with MARSSIM protocol. To validate the adequacy and completeness of soil removal from Class 1 SUs, confirmatory soil sample results will be evaluated using the SOR methodology (Section 3.2.1). The SOR will be compared to the DCGLs.

As previously discussed in Section 3.3, comparison of reference area (background) radionuclide concentrations with SU concentrations will be performed using the two-sample WRS statistical test. This test is selected because the ROCs also occur naturally (i.e., background). The two-sample WRS statistical test assumes the reference area and SU data distributions are similar except for a possible shift in the medians.

As discussed in Section 0, biased soil sampling will be conducted where indicated by beta and gamma walkover survey results.

5.6 Field Records

For surveys of all types, it is essential that significant events be documented and retained for future reference. While some types of project events have specific forms on which they are documented, many events occur on a routine basis during survey field activities that must be documented as they occur. To provide a practical means of capturing this information, project logbooks will be used.

5.6.1 Instrument Quality Control Logbook

A logbook dedicated to recording daily quality control (QC) of instrumentation shall be maintained by field personnel. Site conditions that could potentially impact data collection and instrumentation performance shall also be recorded.

The instrument QC logbook is considered a legal record. Logbooks will be permanently bound and the pages will be pre-numbered. Pages may not be removed from the logbook under any circumstances. Entries shall be legible, factual, detailed, and complete and shall be signed and dated by the individual(s) making the entries. If a mistake is made, the individual making the entry shall place a single line through the erroneous entry and shall initial and date the deletion. Under no circumstances shall any previously-entered information be completely obliterated. Use of whiteout in the Project Data Logbook is not permitted for any reason. Only one Project Data Logbook will be maintained. If a Project Data Logbook is completely filled, another volume shall be initiated. In this case, each volume shall be sequentially numbered.

5.6.2 Field Survey Logbooks

Field survey logbooks will be carried by each survey team during project operations. If multiple volumes are needed to support multiple field survey teams, each volume will be clearly identified with a unique designation approved by the field operations lead. Any survey teams collecting project data shall carry a field survey logbook.

Like the instrument QC logbook, field survey logbooks are considered legal records and will be subject to the same provisions described in Section 4.6.1, above.

5.6.3 Other Logbooks

The Project Manager may initiate additional logbooks as deemed necessary to ensure project activities are adequately documented. Additional logbooks will be considered legal records and will be subject to the same provisions described above in Section 5.6.1.

5.7 Project Electronic Data

5.7.1 Data Backup

Electronic data collected during the day will be backed-up at the end of the same day on which it was collected (e.g., to CD, or equivalent), before processing or editing. This is an archive of the raw data and, once created, it shall not be altered. More than one day’s data may be written to a CD. Field computer(s) used to store project data will be backed up weekly, at a minimum. Raw archived data will be stored in a different location from weekly backups. Electronic data will be provided daily to data processing specialists. The time and date that data files are transmitted will be recorded in the data logbook. File names will be verified by comparison with field notes and corrected if necessary, following approval by the Project Manager or designee.

5.7.2 Data Processing

Beta walkover survey/gamma walkover survey/DGPS data will be processed daily, as necessary, and review the data for errors caused by fluctuations/interferences in the DGPS signal. Data processing specialists will inform the Project Manager or designee of any identified deficiencies and will make corrections as directed. Conversions, errors, corrections, and/or adjustments to project data shall be documented in the data logbook.

6.0 SURVEY QUALITY ASSURANCE AND QUALITY CONTROL

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

6.1 Instrumentation Requirements

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

(A) Calibration

Equipment used during the FSS will be maintained and calibrated to manufacturer's specifications. Current calibration and/or maintenance records for instruments used during the survey will be maintained on-site for review and inspection. The records will include, at a minimum, the following:

- Name of the equipment
- Equipment identification (model and serial number)
- Manufacturer
- Data of Calibration
- Calibration Due Date

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

Field instruments will be source-checked at the beginning and end of each workday periodically. Written records of daily checks will be maintained and filed in the project file.

(B) Source and Background Checks

Prior to and after daily use, instruments will be QC checked by comparing the instrument's response to ambient background and to a designated gamma radiation source. The results of the ambient background and source checks will be recorded in the instrument QC logbook.

Instrument response to ambient background will be used to establish a mean background response for each instrument, following the system source check but prior to the commencement of a beta walkover survey/gamma walkover survey. Background readings shall be conducted at the beginning of each day prior to collecting data in the field. Results from these surveys will be used to monitor gross fluctuations in background gamma fluence (e.g., from changes due to barometric pressure and other, non-contaminant-related causes), and to check detector response. Please note that the background measurements are made solely for the purpose of normalizing each day's survey results and eliminating bias introduced by natural fluctuations in site radiological conditions.

Source checks will consist of one-minute integrated counts with the designated source positioned in a reproducible geometry, performed at the designated location. Instrument response to the designated QC check source will be plotted on control charts and evaluated against the average established at the start of the field activities. A performance criterion of ±20% of this average will be used as an investigation action level. The radiation safety officer (RSO) will investigate results exceeding this criterion and will make appropriate corrections to instrument readings if response is deviated by factors beyond personnel control, such as large humidity or temperature changes. The RSO has authority to decide whether or not the instrument is acceptable to use or must be removed from service.

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

6.2 DGPS Requirements and Quality Control

DGPS QC will be accomplished with calibration points, viewing plotted survey data, and keeping detailed field notes. A calibration point is a location with known horizontal and vertical coordinates (e.g., a benchmark) that can be used to check the accuracy of DGPS data. Calibration points will ensure that the differential position corrections are being calculated properly, and that equipment is performing to manufacturer's specifications. DGPS calibration points shall be set in convenient locations near the areas to be surveyed. Existing unique features such as manholes, fire hydrants, or other permanent features may serve as calibration points. Calibration points shall be set in areas clear of overhead obstructions.

One or more DGPS calibration points will be established prior to beginning this FSS. At each calibration point, ten initial DGPS position readings will be collected, each having a one-minute duration or more. Each set of ten readings will be used to develop the average position of the applicable calibration point. Prior to beginning and following completion of a survey, technicians shall collect position data at one of the calibration points. Data may also be collected at a calibration point at any point in a survey if anomalous readings or other indications of potential DGPS data quality problems are observed. Data shall be collected at the calibration point at least two times for each day's survey. Each time calibration point data are collected, the result shall be compared to the average location of that point, as calculated above. Measurements differing from the average by more than one meter will be investigated and corrective measures will be implemented as appropriate.

To collect a calibration point, the following steps shall be completed:

- Position the DGPS unit precisely over the desired calibration point location
- Ensure the radiation detector is turned off
- Open the DGPS data file for the designated beta walkover survey/gamma walkover survey
- Select "Calibration Point" or equivalent in the data dictionary of the rover computer
- Record the calibration point number (there may be more than one calibration point) in the data file
- Collect DGPS (position) data until the computer indicates the measurement is complete

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

6.3 Soil Sampling Quality Assurance

Soil samples will be sent to a laboratory chosen to perform QA analyses on samples collected for ^{14}C (method EERF C-01 mod), ^{36}Cl (method GL-Rad-A033), ^{63}Ni (method RESL Ni-1 mod), ^{90}Sr (method 905.0 mod), ^{3}H (method 906.0 mod), and gamma spectroscopy scan (method 901.1 mod/HASL 300) for ^{226}Ra, ^{210}Pb, ^{137}Cs, and other gamma emitting isotopes. Duplicate samples will be scheduled at a rate of 5% of samples collected. The samples will be numbered using a unique identifier and will be sent to the laboratory for analysis. Additionally, the analytical laboratory should perform duplicate analyses on selected samples as specified in their QA procedures. Duplicate analyses performed by the laboratory will be compared to the initial analytical results by determining a z score value for each data set by the following equation:

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

Where: S, D, $\equiv$ value of (S)ample and (D)uplicate measurements

σ $\equiv$ one sigma error associated with (S)ample and (D)uplicate measurements

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

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

7.0 REFERENCES

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(Entech, 1998) RESRAD Modeling in Support of BARC LLRBS Decommissioning, prepared for U.S. Department of Agriculture, Agricultural Research Service, Beltsville, MD, April 1998.

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(NRC, 1998a) A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys, (NUREG-1505, Rev.1). United States Nuclear Regulatory Commission. Washington, D.C.: U.S. Government Printing Office, June 1998.

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Office, August 2000.

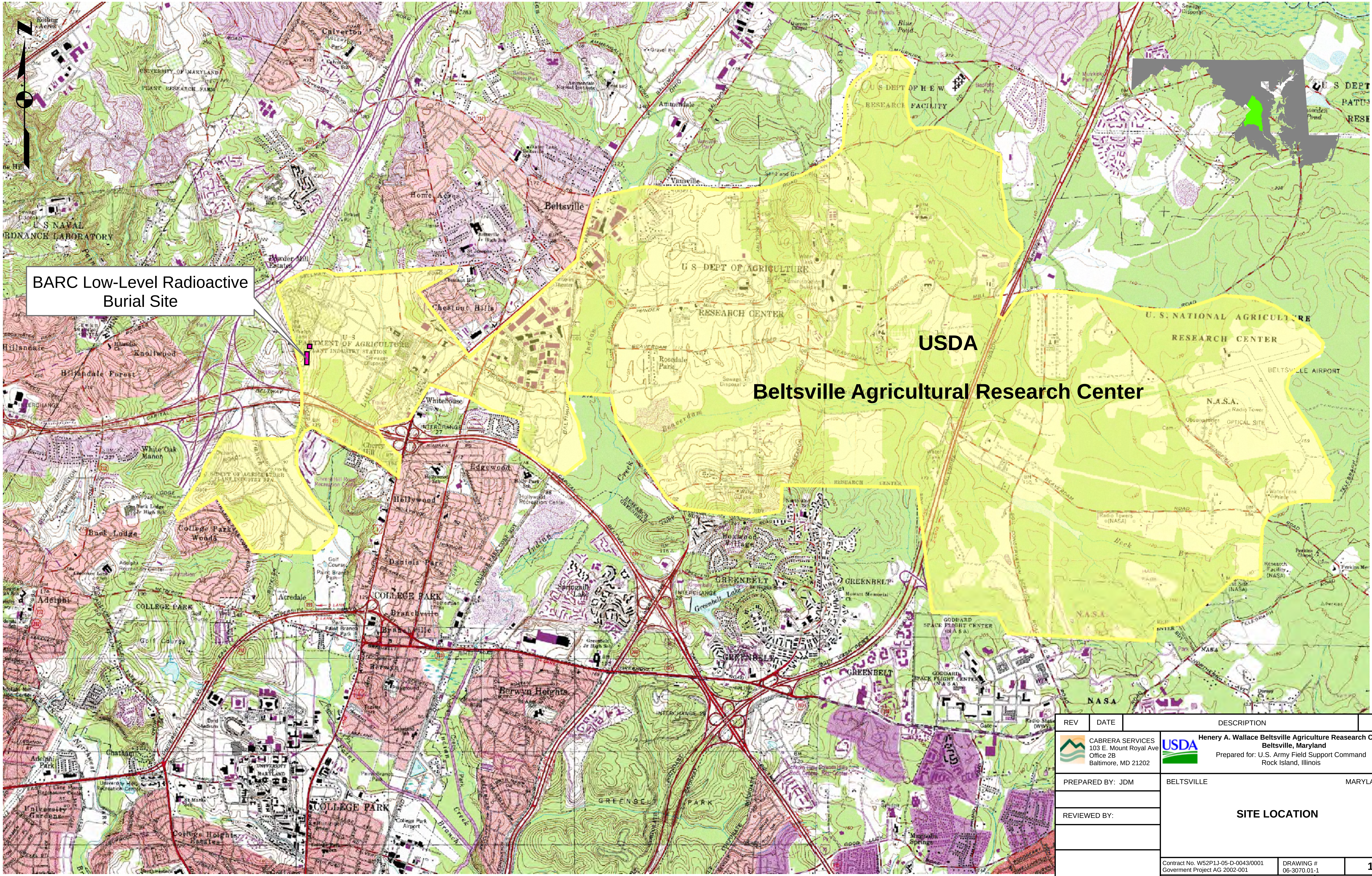
Figures

Figure 1. Site Location

Figure 2. Historical Pit Locations in the North Field

Figure 3. BARC Low-Level Radioactive Burial Site North & South Fields

Figure 4. Site Conceptual Model



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Prepared for: U.S. Army Field Support Command
Rock Island, Illinois

PREPARED BY: JDM

BELTSVILLE MARYLAND

REVIEWED BY:

SITE LOCATION

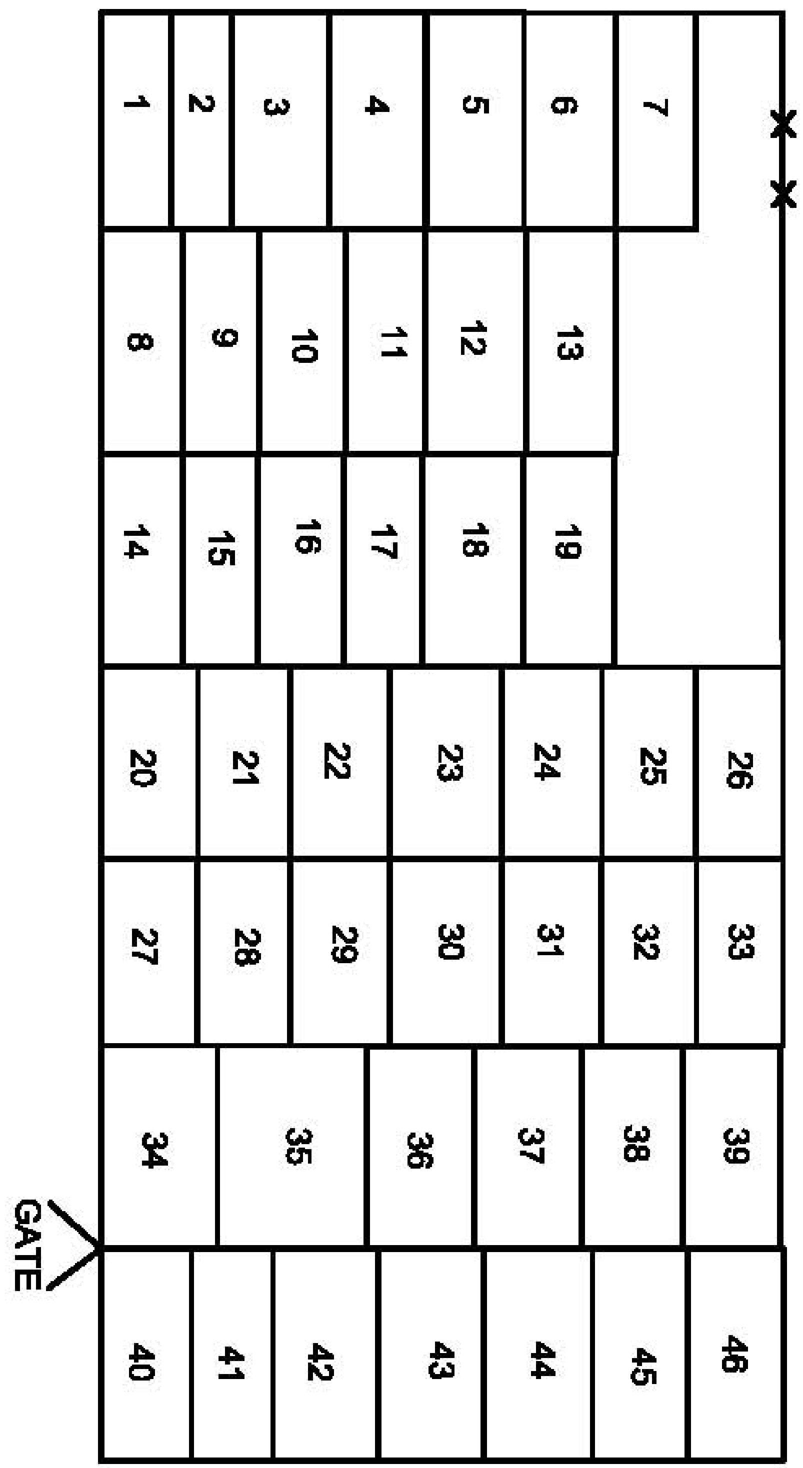
Contract No. W52P1J-05-D-0043/0001
Goverment Project AG 2002-001

DRAWING # 06-3070.01-1

1

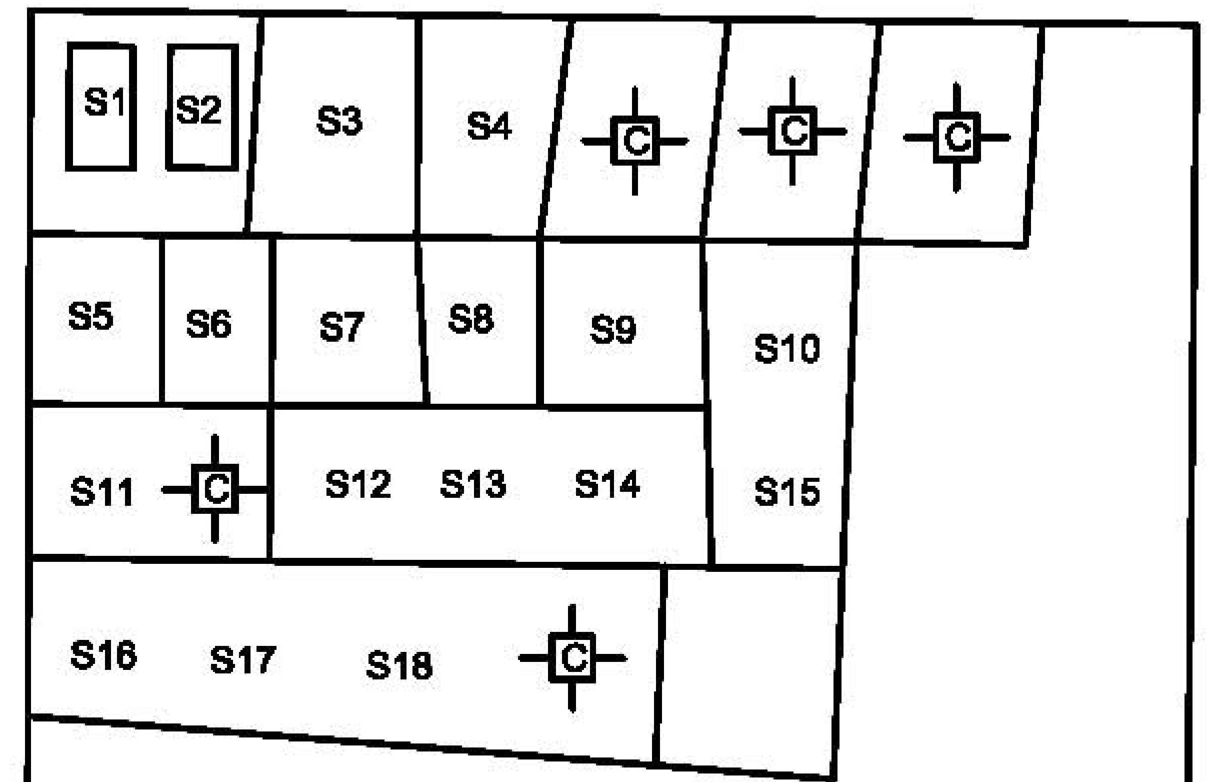
SCALE: 0 1,000 2,000 Feet

DATE: MAY 2007



Note: Not to scale. Entech provided to Cabrera via fax on 1/10/03

S1 = 4/6/51
S2 = 8/4/53
S3 = 5/13/59
S4 = 4/3/61
S5 = 8/67
S6 = 9/68
S7 = 5/70
S8 = 11/70
S9 = 4/71
S10 = 11/71
S11 = 7/72
S12 = 8/72
S13 = 11/72
S14 = 12/72
S15 = 3/73
S16 = 2/73
S17 = 3/15/73
S18 = 9/7/73



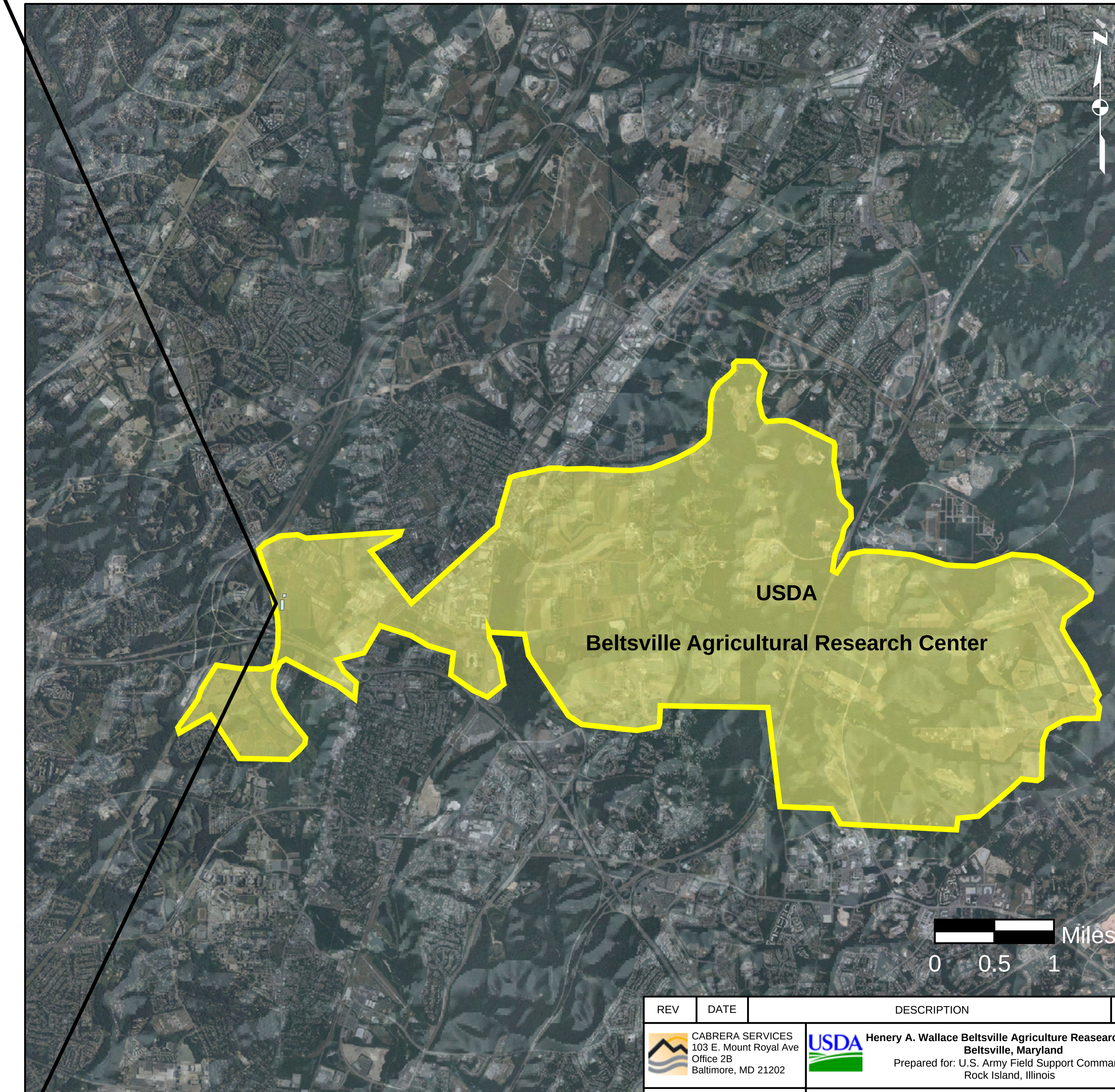
GATE

This diagram shows the original fence and post placement prior to the beginning of site work.

Twenty-three burial pits were identified either through site markers or through depressions in the ground. If known, the dates are given below the location of the site marker. The approximate center of thedepression is indicated if no marker was found.

Source: Letter dated 8/11/1989 from J.F. Patterson of RSO, Inc. to B. Flook, Agricultural Research Service

REV	DATE	DESCRIPTION	BY
CABRERA SERVICES 103 E. Mount Royal Ave Office 2B Baltimore, MD 21202		USDA Henery A. Wallace Beltsville Agriculture Reasearch Center Beltsville, Maryland Prepared for: U.S. Army Field Support Command Rock Island, Illinois	
PREPARED BY: JDM		BELTSVILLE	MARYLAND
REVIEWED BY:		HISTORICAL PIT LOCATIONS IN THE NORTH FIELD	
		Contract No. W52P1J-05-D-0043/0001 Goverment Project AG 2002-001	DRAWING # 06-3070.01-2 — 2
		SCALE:	DATE: MAY 2007



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Prepared for: U.S. Army Field Support Command
Rock Island, Illinois

PREPARED BY: JDM

REVIEWED BY:

BELTSVILLE MARYLAND

BARC LOW-LEVEL RADIOACTIVE BURIAL SITE
NORTH & SOUTH FIELDS

Contract No. W52P1J-05-D-0043/0001
Goverment Project AG 2002-001

DRAWING #
06-3070.01-3

3

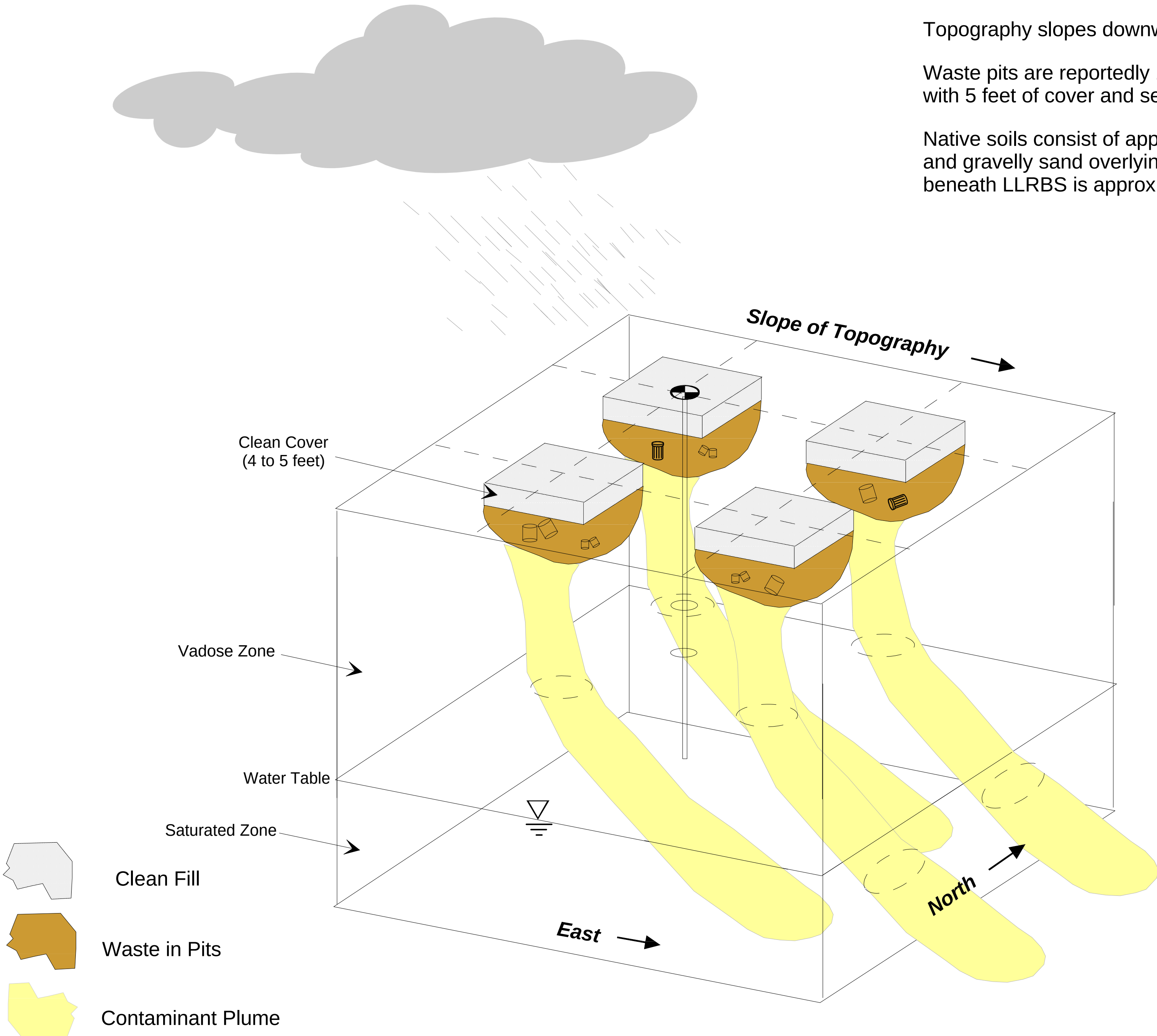
SCALE:

DATE: MAY 2007

Topography slopes downward from west to east.

Waste pits are reportedly 10 ft wide by 12 ft long by 10 ft deep with 5 feet of cover and separated horizontally by 6 ft.

Native soils consist of approximately 50 feet of fine sand and gravelly sand overlying bedrock. The depth to groundwater beneath LLRBS is approximately 25 feet.



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	Henry A. Wallace Beltsville Agricultural Research Center (BARC) Beltsville, Maryland	Figure 4 Site Conceptual Model

Appendix A

Estimation of Minimum Detectable Concentrations (MDC)

Shpack Fidler Scan for Ra226 @ 3.1 pCi/g NO SOIL COVER 15 cm thick x 28 cm RADIUS 40 yr decay
15% background variability
Fluence rate to exposure rate (FRER, no units) = ~ (1 uR/h)/(Eγ)(u_{en}/ρ)air

TABLE 1

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

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

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

TABLE 2

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

800	0.0676	0.04
1,000	0.0586	0.03
1,500	0.0469	0.03
2,000	0.0413	0.02

for Fidler G-5 12.7cm dia x 0.16 cm thick NaI crystal

$x = 0.16$ cm

$\rho = 3.67$ g/cm^3

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

TABLE 3

Energy$_\gamma$, keV	FRER	P	RDR
15	0.0517	1.00	0.0517
20	0.0969	1.00	0.0969
30	0.2268	0.99	0.2239
40	0.3906	1.00	0.3906
50	0.5208	1.00	0.5199
60	0.5708	0.98	0.5591
80	0.5297	0.84	0.4449
100	0.4329	0.64	0.2752
150	0.2656	0.31	0.0816
200	0.1866	0.18	0.0332
300	0.1157	0.09	0.0108
400	0.0845	0.07	0.0056
500	0.0673	0.05	0.0037
600	0.0563	0.05	0.0027
800	0.0433	0.04	0.0017
1,000	0.0357	0.03	0.0012
1,500	0.0261	0.03	0.0007
2,000	0.0214	0.02	0.0005

Estimated Fidler G-5 12.7cm dia x 0.16cm thick NaI response for Cs-137 is 1287 cpm/uR/hr

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

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

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

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

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

TABLE 4

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

MDC for Cs-137 energy

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

b_i =	214.5	counts
MDCR =	2664.09	cpm
$MDCR_{surveyor}$ =	3768	cpm

Utilize a 15% bkg variability instead of SQRT of 214.5 counts in 1 sec interval for MDCR; more realistic

minimum detectable exposure rate = 2.93 μR/hr

Table 5

keV	MicroShield Exposure Rate, μR/hr (with buildup)	cpm/μR/hr	cpm/μR/hr (weighted)	Percent of NaI detector response
15	0.000E+00	28934	0	0.0%
20	0.000E+00	54250	0	0.0%
30	0.000E+00	125355	0	0.0%
40	0.000E+00	218695	0	0.0%
50	2.946E-04	291052	120	3.9%
60	0.000E+00	313006	0	0.0%

80	4.490E-03	249068	1560	51.2%
100	4.048E-05	154090	9	0.3%
150	0.000E+00	45680	0	0.0%
200	9.007E-03	18602	234	7.7%
300	2.763E-02	6053	233	7.7%
400	6.944E-02	3140	304	10.0%
500	4.051E-03	2056	12	0.4%
600	1.295E-01	1493	270	8.8%
800	3.294E-02	942	43	1.4%
1000	1.324E-01	676	125	4.1%
1500	1.118E-01	398	62	2.0%
2000	1.955E-01	287	78	2.6%
Total	7.171E-01		3048	100%

Minimum Detectable Exposure Rate =

MDCR surveyor/(cpm/μr/hr)
1.236 μr/hr

and MDC for radium 226 and 40-year equilibrium progeny based on a soil concentration of 3.1 pCi/g Ra-226

Scan MDC = (Assumed MDC $Ra226_{Conc}$) x (Exposure Rate $MDCR_{Surveyor}$)/(Exposure $Rate_{assumed\ Ra226\ Conc}$)

Scan MDC = 1.72 pCi/g

Fidler Scan for Cs-137 @ 1pCi/g, No Soil Cover, 15 cm thick x 28 cm RADIUS

Fluence rate to exposure rate (FRER, no units) = ~ (1 uR/h)/(Eγ)(u_{en}/ρ)air

TABLE 1

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

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

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

TABLE 2

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

1,000	0.0586	0.03
1,500	0.0469	0.03
2,000	0.0413	0.02

for Fidler G-5 12.7cm dia x 0.16 cm thick NaI crystal

beryllium window per Fidler catalog 0.010 inch

$x = 0.16$ cm

$\rho = 3.67$ g/cm^3

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

TABLE 3

Energy$_\gamma$, keV	FRER	P	RDR
15	0.0517	1.00	0.0517
20	0.0969	1.00	0.0969
30	0.2268	0.99	0.2239
40	0.3906	1.00	0.3906
50	0.5208	1.00	0.5199
60	0.5708	0.98	0.5591
80	0.5297	0.84	0.4449
100	0.4329	0.64	0.2752
150	0.2656	0.31	0.0816
200	0.1866	0.18	0.0332
300	0.1157	0.09	0.0108
400	0.0845	0.07	0.0056
500	0.0673	0.05	0.0037
600	0.0563	0.05	0.0027
800	0.0433	0.04	0.0017
1,000	0.0357	0.03	0.0012
1,500	0.0261	0.03	0.0007
2,000	0.0214	0.02	0.0005

Estimated Fidler G-5 12.7cm dia x 0.16cm thick NaI response for Cs-137 is 1287 cpm/uR/hr based on Cab

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

Energy$_\gamma$, keV	$(u_{en}/\rho)_{air}$, cm^2/g		
662	0.0294	FRER ~	0.0514
Energy$_\gamma$, keV	$(\mu/\rho)_{NaI}$, cm^2/g		
662	0.0780	Probability =	0.04
		RDR =	0.0023

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

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

TABLE 4

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

MDC for Cs-137 energy

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

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

minimum detectable exposure rate = 1.33 μR/hr

Table 5

keV	MicroShield Exposure Rate, μR/hr (with buildup)	cpm/μR/hr	cpm/μR/hr (weighted)	Percent of NaI detector response
15	0.000E+00	28934	0	0.0%
20	0.000E+00	54250	0	0.0%
30	2.178E-04	125355	113	6.8%
40	5.958E-05	218695	54	3.2%
50	0.000E+00	291052	0	0.0%
60	0.000E+00	313006	0	0.0%
80	0.000E+00	249068	0	0.0%
100	0.000E+00	154090	0	0.0%

150	0.000E+00	45680	0	0.0%
200	0.000E+00	18602	0	0.0%
300	0.000E+00	6053	0	0.0%
400	0.000E+00	3140	0	0.0%
500	0.000E+00	2056	0	0.0%
600	2.421E-01	1493	1491	90.0%
800	0.000E+00	942	0	0.0%
1000	0.000E+00	676	0	0.0%
1500	0.000E+00	398	0	0.0%
2000	0.000E+00	287	0	0.0%
Total	2.424E-01		1657	100%

Minimum Detectable Exposure Rate =

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

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

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

Scan MDC = 4.27 pCi/g
158.0 Bq/kg