ENERGYSOLUTIONS

CS-OP-PN-042

REMEDIAL WORK PLAN

WASTE EXCAVATION AND SITE RESTORATION FOR THE BRECKENRIDGE DISPOSAL SITE

Madison Road
St. Louis, Bethany Township, Michigan

Revision 0

Authored By:	Duane R. Quayle, Senior Health Physicist/Project Manager	1/27/2010 Date
Reviewed By:	Michael A. Carr, CHP, Radiological Engineer/Radiation Safety Officer	1-27-2010 Date
Approved By	Robert Hornbeck, Sr. Project Manager	1/27/2010 Date
Approved By	Arthur J. Palmer, CHP/PMP, Director, Health Physics & Radiological Engineering	1/27/2010 Date

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1.0 INTRODUCTION

The Breckenridge Disposal Site (Site) is located on Madison Road about 4 miles east of downtown St. Louis, Bethany Township, Michigan. The Breckenridge property is a narrow triangular-shaped parcel of land that is mostly flat and grassy with scattered large trees. The Site, bounded by Madison Road on the north, by Bush Creek on the east, and by farmland on the west, is about 5,100 square meters (m^2) in size. The nearest residence is located approximately 0.2 kilometers to the east across Bush Creek. A six-foot high chain-link fence controls access to the Site. Figures 1 and 2 provide the Site location and Site layout maps.

Between 1967 and 1970, the Site was used for the disposal of process wastes from an yttrium recovery operation managed by Michigan Chemical Corporation (MCC). These disposal activities were authorized under U.S. Atomic Energy Commission (AEC) License Number SMB-0833 and were performed in accordance with 10 CFR 20.304, "Disposal by Burial in Soil." The buried waste material is a solid waste byproduct known as filtercake, which originated from a rare-earth metal (yttrium) extraction process. Disposal records reported that the filtercake was typically a dense, clay-like material containing elevated levels of naturally occurring uranium and thorium. After site operations ceased, AEC License Number SMB-0833 was terminated.

Since elevated levels of surface radioactivity remained on the Site, the U.S. Nuclear Regulatory Commission (NRC) requested that the radiological conditions at the Site be evaluated. Several radiological evaluations have been performed at the Site in recent years as listed in References 1 through 9 from 1982 to 2007. Characterization efforts provided an estimate of the remaining radioactive waste source term on-site in terms of area, volume, and average thorium-232 and uranium-238 concentrations in the buried waste material. The average concentrations were determined to be 267 picocuries per gram (pCi/g) thorium-232+D and 170 pCi/g uranium-238+D (Reference 2) with "+D" indicating that the parent isotope is in equilibrium with its short lived decay daughters.

Waste inventory estimates were previously derived using two different methods (Reference 2) as follows. The first waste inventory estimate was derived based upon the confirmed waste areas and potential waste areas as identified during the 2001 Site characterization activities. Confirmed waste areas were (CWA) identified by the geophysical surveys and verified using core boring. The potential waste areas were identified as possible disposal areas by the geophysical surveys but they were not confirmed by core boring. The locations of the CWAs and associated remediation areas (RAs) are outlined in Figure 5.

The second waste inventory estimate used a filtercake volume estimate based on the average densities of the 19 waste-material samples collected and the historical estimate of 151 wet tons of total waste deposited at the Site, as provided in Reference 3. Surface scans of the Site also indicated that there are discrete locations of contamination in soils at and near the surface that should be considered in assessing the total activity at the Site. Reference 8 also provided a waste volume estimate. The volume of waste dispersed in the surface and near-surface soils is assumed to be very small in relation to the buried waste volume (i.e., less than 1%). Waste volume estimate ranges based on all characterization efforts are provided in Table 3.

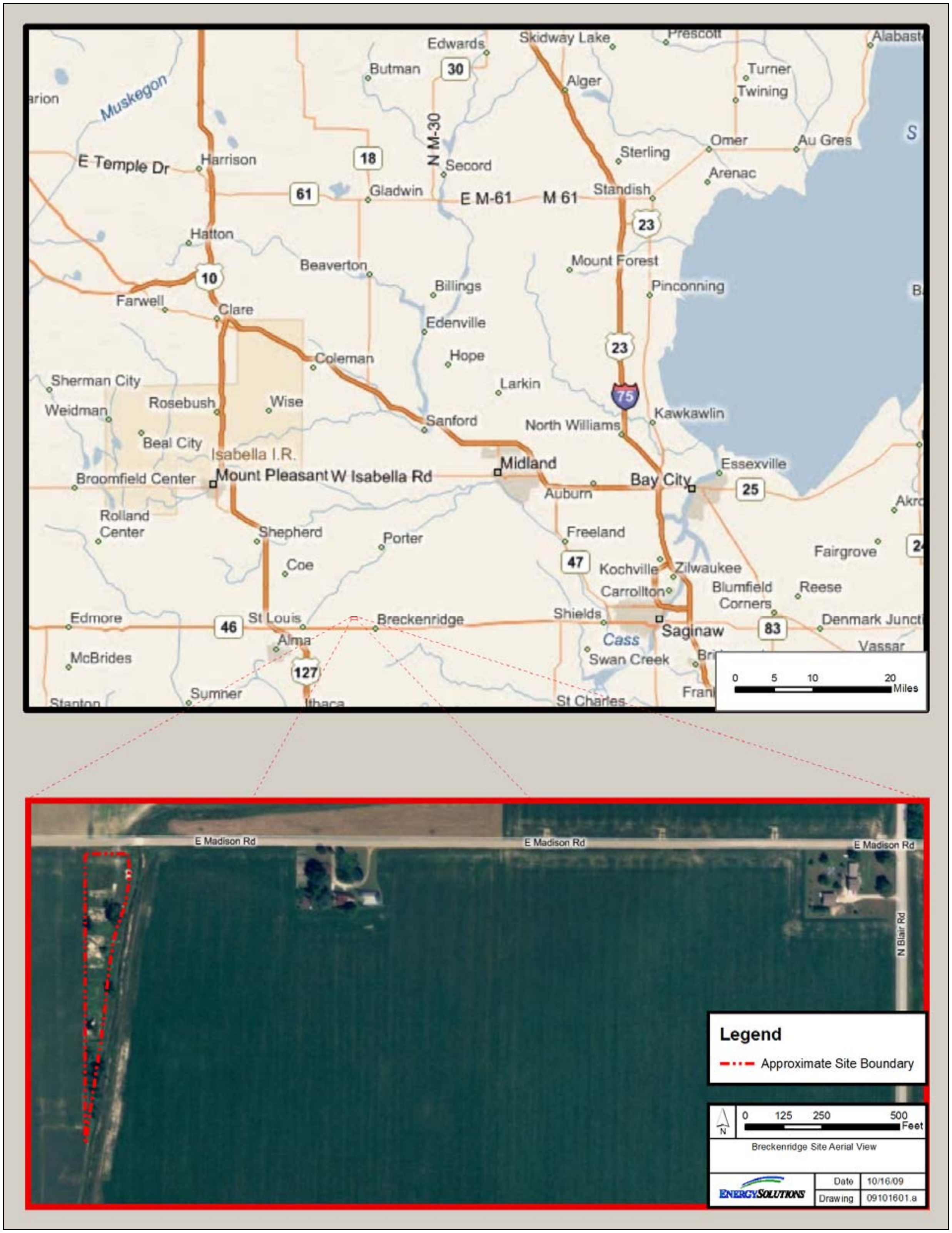

Figure 1: Breckenridge Site Location

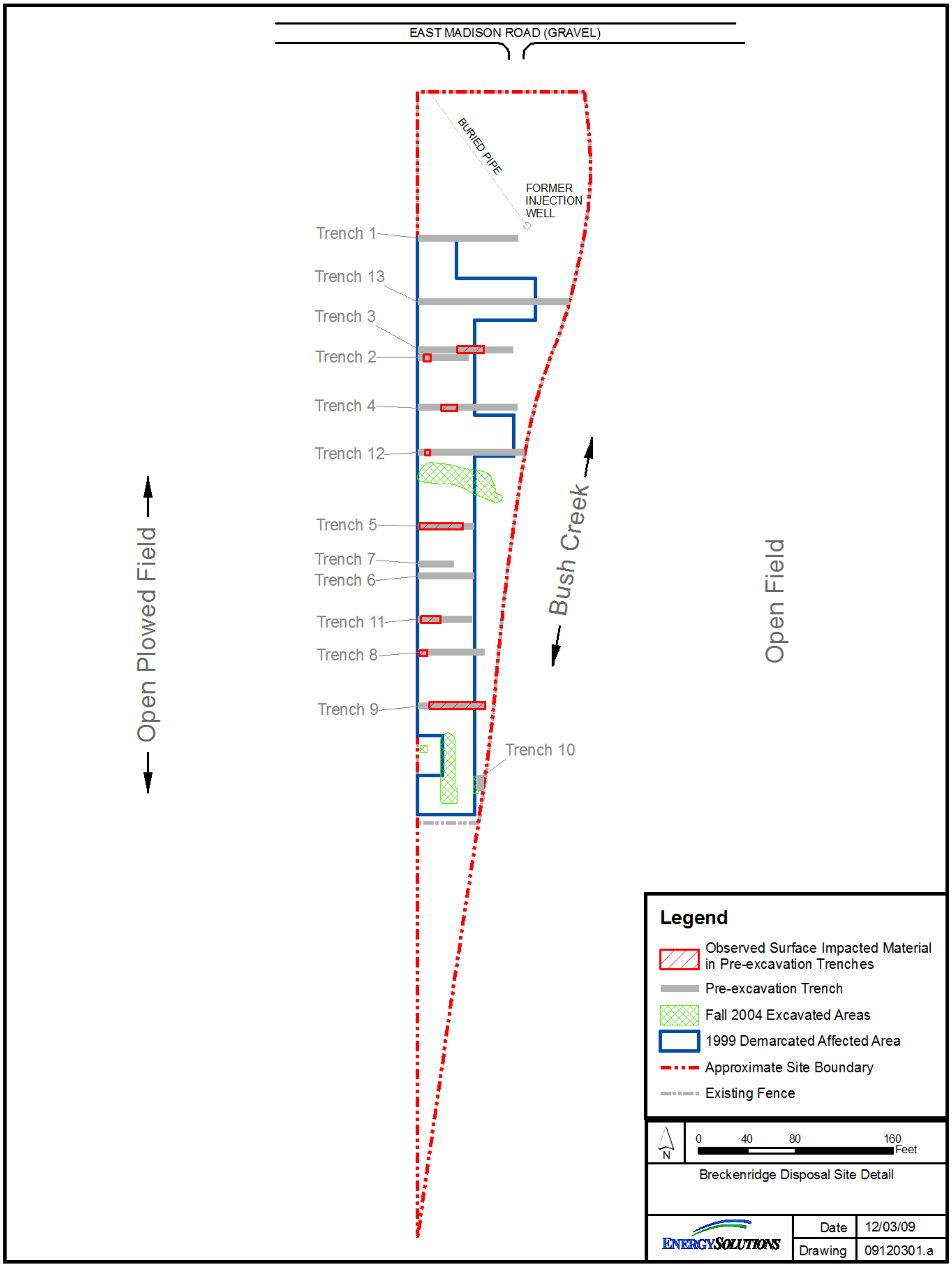

Figure 2: Breckenridge Site - Detail

2.0 OBJECTIVES

The primary objective of the Site remediation project is to remove radioactive waste filtercake and/or contaminated soils and release the Site for unrestricted use. The supporting objectives will consist of a 100% gamma radiation survey of the Site for identification of surface and near-surface radiation, the excavation of identified surface and near-surface radiation, the excavation of contaminated soils within the confirmed and potential locations of slit trenches, and the performance of a final status survey.

The Site remediation activities are expected to begin in January 2010 with the completion of all activities by April 2010, contingent on obtaining all necessary regulatory approvals and permits. Figure 3 provides a projected schedule of the major project tasks. These tasks include the following:

- <u>Mobilization</u> – A field team consisting of the project manager (PM), project health physicist (PHP), radiological engineer (RE), senior health physics technicians (HPTs), and equipment operators (EO) will travel to the Site. All necessary equipment and supplies will be delivered. A field office trailer will be set up for Site use to include on-site gamma spectroscopy analysis, gross alpha and beta counting, and air sample analysis. The PM and PHP will provide all required training, to include radiation awareness training, to the project team members.
- <u>Site Preparations</u> – Site preparation activities are included as part of the mobilization and include cutting the grass/weeds and removing small trees and shrubs (clearing and grubbing). Upon completion of clearing and grubbing, a walkover gamma radiation survey coupled to GPS will be performed of the surface soils, the elevated areas and known waste areas and will be flagged and/or marked by either surveyors' paint or surveyors' flags or a combination of both.
- <u>Remediation</u> – Remediation activities will include the initial excavation of surface and near-surface contamination based on the gamma radiation survey(s). In addition, the contaminated soil areas will be excavated based on remediation support surveys and prior characterization data.
- <u>Waste Shipments</u> – All excavated contaminated soils and/or debris will be loaded into bags, transloaded to the rail spur, and shipped to the Energy*Solutions* disposal site in Clive, Utah. The project team will use a certified waste broker with experience in shipping radioactive waste to provide transportation and disposal support services.
- <u>Final Status Surveys</u> – The HPTs will conduct the final status survey according to this Work Plan (WP) and as directed by the RE and/or PHP.

- Site Restoration – Site restoration activities will include backfilling excavated areas with clean fill, grading the site and seeding disturbed areas. Backfill used for the Site restoration shall include a combination of soils that have been cleared for use as backfill originating on-site, as well as soils originating from an off-site borrow area. Soils originating on-site will be surveyed, sampled and stockpiled accordingly for reuse as backfill. These will typically consist of surface soils that have been surveyed and sampled to the surface DCGLs as well as "clean" soils that have been removed to slope the excavation to ensure a safe working environment. All on-site soils removed and stockpiled for reuse will be placed on a barrier such as a geotextile or poly liner to prevent any potential cross-contamination. Any soils used for backfilling the top 1.5 meters of the Site will meet the surface DCGLs while soils used for backfilling areas deeper than 1.5 meters shall meet the subsurface DCGLs.

  On-site soils that have been removed shall be sampled either *in-situ* prior to removal and/or composite sampled after removal such that the sampling frequency follows the MARSSIM sampling protocols prior to reuse. Off-site soils shall be sampled to the surface soil DCGLs at a frequency of 1 composite sample per 100 cubic yards of fill for radiological sampling. Additionally, there will be one overall composite sample for all off-site borrow used for non-radiological analysis to ensure no hazards are introduced to the Site upon site restoration. This approach is compliant with Michigan Department of Environmental Quality (MDEQ) Part 201.

- Demobilization – The field team will package and ship all equipment and supplies with arrangements made to remove the field office trailer. The project team will then travel back to their respective home offices. After site demobilization copies of all regulatory and applicable project documentation pertaining to the safe decommissioning of the site will be provided to the Trustee.

- Final Status Survey Report – Upon completion of the Final Status Survey Report copies of said report will be forwarded to the Trustee, NRC and MDEQ for review and comment. Approval of the Final Status Survey Report will be by the NRC.

Task Name	Duration	Start	Finish
Breckenridge	**271 days**	**Wed 9/30/09**	**Tue 10/26/10**
Contract Award Date - NTP	0 days	Wed 9/30/09	Wed 9/30/09
Pre-Mobilization	**78 days**	**Wed 9/30/09**	**Thu 1/28/10**
Work Plan Development	73 days	Wed 9/30/09	Thu 1/21/10
HASP Plan Development	73 days	Wed 9/30/09	Thu 1/21/10
Complete Required Permits	73 days	Wed 9/30/09	Thu 1/21/10
NRC Licenses Approval	73 days	Wed 9/30/09	Thu 1/21/10
Finalize & NRC Approve All Plans & Permits	5 days	Fri 1/22/10	Thu 1/28/10
Work Plan Approval Milestone	**0 days**	**Thu 1/28/10**	**Thu 1/28/10**
Field Activities	**52 days**	**Mon 4/5/10**	**Tue 6/15/10**
Site Mobilization	5 days	Mon 4/5/10	Fri 4/9/10
Site Mobilization Complete Milestone	**0 days**	**Fri 4/9/10**	**Fri 4/9/10**
Scoping Surveys	5 days	Mon 4/12/10	Fri 4/16/10
Excavation & Remediation	30 days	Mon 4/19/10	Fri 5/28/10
Excavation Complete Milestone	**0 days**	**Fri 5/28/10**	**Fri 5/28/10**
Perform Final Status Surveys (FSS)	30 days	Mon 4/26/10	Fri 6/4/10
FSS Complete Milestone	**0 days**	**Fri 6/4/10**	**Fri 6/4/10**
Waste Packaging, Transport, and Shipment	30 days	Wed 4/28/10	Tue 6/8/10
Radiological Waste Shipped Milestone	**0 days**	**Tue 6/8/10**	**Tue 6/8/10**
Demobilzation	5 days	Wed 6/9/10	Tue 6/15/10
Final Reports & Analytical	**95 days**	**Wed 6/16/10**	**Tue 10/26/10**
Off-Site Analytical & Data Validation	35 days	Wed 6/16/10	Tue 8/3/10
Final Status Survey Report (FSSR) Preparation	45 days	Wed 7/7/10	Tue 9/7/10
Closure Report Preparation	45 days	Wed 7/7/10	Tue 9/7/10
Internal Review of FSSR and Closure Report	5 days	Wed 9/8/10	Tue 9/14/10
Submit FSSR and Closure Report	0 days	Tue 9/14/10	Tue 9/14/10
FSSR Report Submitted to NRC Milestone	**0 days**	**Tue 9/14/10**	**Tue 9/14/10**
Site Released Milestone	**0 days**	**Tue 10/26/10**	**Tue 10/26/10**

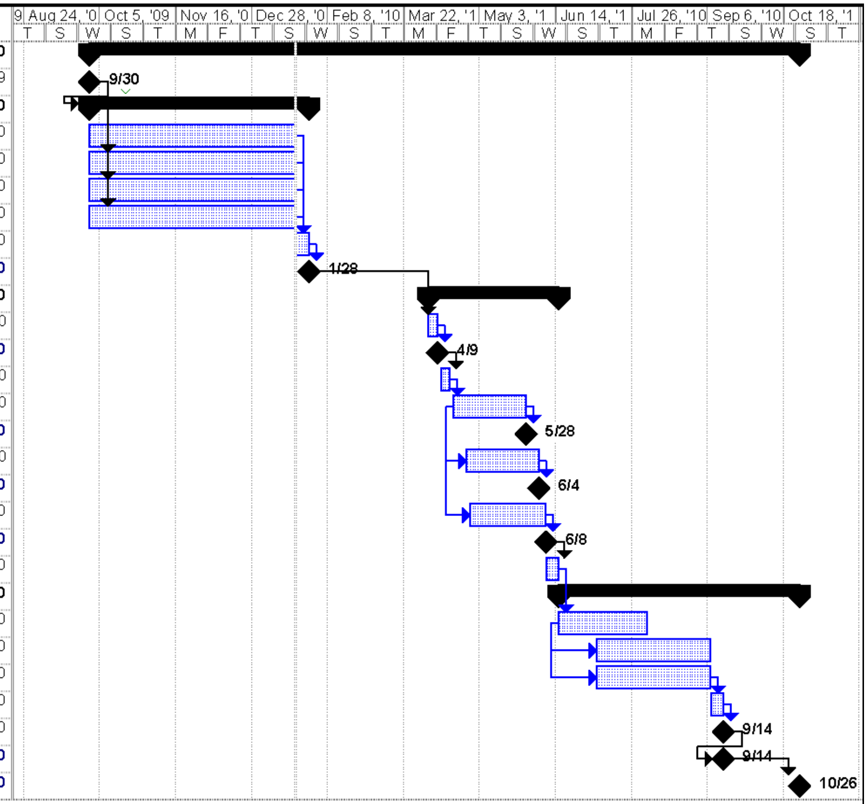

Figure 3: Breckenridge Remediation Schedule

NOTE: This schedule is an approximation of anticipated field activities and deliverables—the actual schedule will be driven by weather, permit requirements and approval, etc.

3.0 PROJECT ORGANIZATION

The remediation project is being managed by Energy*Solutions*, an international nuclear services company with operations throughout the United States and around the world. With over 5,500 world-class professionals, Energy*Solutions* is a world leader in the safe recycling, processing and disposal of nuclear material. Energy*Solutions* provides integrated services and solutions to the nuclear industry, the United States Government, the Government of the United Kingdom, hospitals and research facilities.

The project organizational chart (Figure 4) is listed below. The PM has the overall responsibility of the project and will control the project budget, set the project schedule, and coordinate the field activities. Based on the level of effort for the project and minimal complexity, the PM will also serve as the site supervisor (SS)—responsible for providing training to the on-site team and ensuring that the appropriate areas are being excavated to support remediation efforts and unrestricted site release. The PHP will serve as the on-site radiation safety officer and the site health and safety officer (SHSO) and is responsible for the setup, use, and calibration of the radiological and health and safety instrumentation and ensuring the Energy*Solutions* radiation protection and health and safety programs are implemented. All Site personnel have full stop work authority to ensure safe operations. The HPTs and subcontractors on-site will work under the direction of the PM/SS and follow the health and safety requirements of the PHP/SHSO.

Energy*Solutions* is providing a certified waste broker (CWB) to transport waste to the Clive, Utah licensed disposal facility. The CWB, in concert with the PM/SS, will schedule the waste transport and rail shipments. The CWB will maintain records of waste shipments until the end of the project. Upon demobilization, records will be sent to the PM/SS for archival.

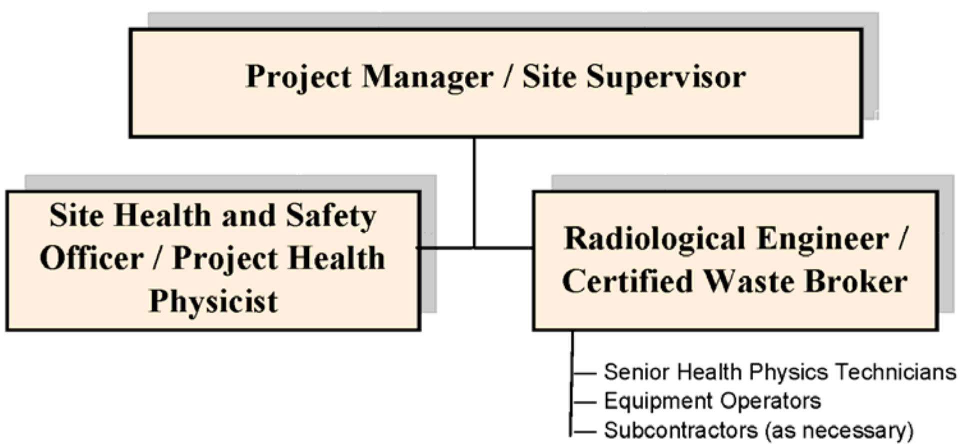

Figure 4: Project Organizational Chart

Upon or shortly after demobilization, Energy*Solutions* will provide the documents identified in the Contract, Exhibit B, (e.g. Final Closure Report), as well as any other documents required by the Energy*Solutions* USNRC Radioactive Materials License.

4.0 DERIVED CONCENTRATION GUIDELINE LEVELS

The source term for the purpose of determining the acceptable derived concentration guideline levels (DCGLs) is the residual concentrations of radioactive material that will be allowed to remain on-site after remediation and license termination for unrestricted use. That concentration is bounded by an upper limit radiation dose or total effective dose equivalent (TEDE) of 25 millirem per year (mrem/yr). This upper limit radiation dose is based upon NRC regulatory compliance per 10 CFR 20.1402 "*Radiological Criteria for Unrestricted Use*" which states:

> *A site will be considered acceptable for unrestricted use if the residual radioactivity that is distinguishable from background radiation results in a TEDE to an average member of the critical group that does not exceed 25 mrem (0.25 mSv) per year, including that from groundwater sources of drinking water, and the residual radioactivity has been reduced to levels that are as low as reasonably achievable (ALARA). Determination of the levels which are ALARA must take into account consideration of any detriments, such as deaths from transportation accidents, expected to potentially result from decontamination and waste disposal.*

The radionuclides of concern (ROC) at the Site include processed uranium and thorium. The DCGLs were developed in units of picocuries per gram (pCi/g) to facilitate analytical assessment using standard field and laboratory protocols (standard industry practices). For the purpose of this work plan and the Final Status Survey (FSS), the DCGL applicable to the average concentration over a survey unit is called the $DCGL_W$. The DCGL applicable to limited areas of elevated concentrations within a survey unit is called the $DCGL_{EMC}$; however, since area factors were not developed with the DCGLs, the $DCGL_{EMC}$ will not be used.

The filtercake that was disposed at the Site was processed such that the uranium and thorium were not in equilibrium with their decay progeny, including radium. For the thorium decay series, over 90 percent of the secular equilibrium (activity parent equals the activity of the progeny) is achieved in 20 years since processing. Since the material has been in the soils for over 35 years, it is technically acceptable to state that the thorium decay series is in secular equilibrium with the parent, thorium-232 (Th-232). For the uranium decay series, the time required to meet secular equilibrium is over 75,000 years, thus uranium-238 (U-238) and its short lived progeny are modeled and analyzed independently, as well as radium-226 (Ra-226). Table 1 provides the specific ROCs, their associated surrogate or measured radionuclide for quantification (gamma emitting progeny in secular equilibrium), and the associated surface and subsurface DCGLs per the 2007 Velsicol Supplemental Characterization Report (Reference 8).
The DCGLs and Site background activity concentrations are presented in Table 1. However, during mobilization or shortly thereafter, the PHP may evaluate an additional background reference area. Lastly, no area factors were provided in the DCGL document, thus all survey units will be remediated to the $DCGL_W$.

Table 1: Breckenridge DCGLs

Isotope	Measured Radionuclide (γ energy, keV)	Background (pCi/g)		Background Adjusted $DCGL_W$ (pCi/g) [a,b,c]			
				Ra-226 & U-238 <50% Equilibrium		Ra-226 & U-238 ≥50% Equilibrium	
		Surface	Subsurface	Surface	Subsurface	Surface	Subsurface
Th-232	Ac-228 (911)	0.5	0.4	4.4	34.4	4.4	34.4
U-238	Th-234 (63/93)	1.9	4.0	4.4	25.5	3.2	14.5
Ra-226	Pb-214 (352)	0.3	0.5	1.6	11.0	1.6	11.0

[a] Includes the assumed mixture of radionuclides in the $DCGL_W$ (e.g., sum-of-fraction (SOF) is accounted for in the $DCGL_W$).
[b] All DCGLs include background levels.
[c] The DCGLs were developed in the 2002 Scientech Dose Assessment Report, but later modified and described in the 2007 Velsicol Supplemental Characterization Report.

5.0 EXCAVATION ACTIVITIES

The information obtained from the November 2001 site characterization (Reference 2) estimates that approximately 150 wet tons (i.e., approximately 100 to 125 m^3) of slit trench waste material with varying concentrations of thorium and uranium are buried at the Site with these areas of excavation being well defined. Surface and near surface contamination on the Site resulted from either unrefined handing processes during disposals or spread of waste material during previous investigations.

In order to release the Site for unrestricted use, the project team will coordinate the excavation process in two distinct phases. The first phase will consist of a 100% gamma radiation walkover scan of the entire Site and the associated remediation of the elevated areas of surface and near surface contamination inside the Site fence-line. The contaminated soils will be placed in waste liners and staged for off-site disposal.

Upon completion of the remediation of the surface and near-surface soils, a MARSSIM Class 1 final status survey (FSS) will be conducted on the surface soils. Upon review and acceptance of the surface soil surveys, the RE and/or PHP will initiate excavation of the "overburden" soils in one foot lifts. Each one foot lift will be stockpiled onto a geotextile fabric or equivalent to prevent cross contamination, and upon meeting the surface DCGLs will be used for backfill on-site. Composite samples will also be collected upon lift removal and stockpiling of the soil for reuse as backfill. These samples will be collected to confirm the *in-situ* sampling results and to support the conclusion that the material meets the requirements for reuse as backfill per the restoration of the Site as discussed previously.

Prior to the excavation of each subsequent one foot lift, a gamma radiation walkover survey will be performed using a NaI detector and the exposed soil visually inspected for the presence of any filter cake material to ensure that there is no intrusion into the slit trenches. If the gamma radiation walkover survey indicates the presence of soil contamination per the action levels provide in Table 2 below or the in-situ sample results exceed the applicable release limits, the RE and/or PHP will consult with the PM and determine if the slit trenches have been breached in order to begin phase 2 of the excavation.

Table 2: Scanning Action Levels

Applicable Model	Action Level	
	6-Inch scan height	12-Inch scan height
Surface Soil	3,000 cpm above Background	1,500 cpm above Background
Subsurface Soil	23,000 cpm above Background	11,000 cpm above Background

These scanning action levels presented above were developed using dose modeling and the guidance as provided in NUREG-1507 and MARSSIM for the instruments that will be used. These action levels shall be confirmed through actual Site survey and sampling to further document the gamma radiation surface scan sensitivities of the survey instruments and to provide further guidance for remediation surveys.

If filtercake is not identified, elevated measurements are not detected and all in-situ sample results are less than the applicable release criteria, the current lift of cover material will be treated as “potentially clean”, the lift removed and composite samples collected to further quantify the activity concentration and verify that it meets the applicable release criteria for re-use as backfill. All on-site soil samples will be analyzed using a portable gamma spectroscopy high purity germanium (HPGe) detector, with 5 percent of the samples sent to an off-site analytical laboratory for cross comparison quality control.

Once the slit trenches are exposed, the second phase of remediation will consist of the precision excavation of the buried slit trench waste and impacted soils. The contaminated soils will be excavated, placed into waste liners, and staged for off-site disposal. The excavation of the known contaminated waste areas (CWA) will continue based on the characterization and Site process knowledge. Upon completion of all the excavation for the waste trenches, each open trench will be sloped as necessary per the MIOHSA requirements for personnel entry for post-remediation surveys and FSS. If any areas of excavation approach the Site boundary and based on gamma scans or sampling, remediation is necessary, the location will be marked by a surveyors’ flag as well as documented with GPS to ensure location ID. Based upon the existing characterization results, Energy*Solutions* does not anticipate to excavate any areas near the boundary to a depth of greater than 4 to 6 feet, which would not require sloping. In addition, Energy*Solutions* does not anticipate on excavating any soil, for remediation or sloping purposes, beyond the Site boundary; however, as necessary, shoring may be used near site boundaries to facilitate the excavation of subsurface contamination or sloping of adjacent properties performed provided permission is obtained from the applicable land owner. All sloping soils that are not expected to contain contamination will be staged on-site on a geotextile material or equivalent for use as backfill, and composite samples collected following the direction of the RE and/or PHP to verify that the soils meet the applicable release criteria for use as backfill.

All soils that are excavated in order to achieve safety compliance for personnel entry (sloping soils) as well as the overburdened soils (cover soils) are anticipated to be at or near background radiation levels and are to be used as backfill. In-situ surveys and composite samples will be collected to ensure the materials may be stockpiled and used as backfill; however, Energy*Solutions* will also periodically scan excavator buckets as these soils are removed and collect up to 20 random samples (1 per 20 cubic yards of removed material) to be analyzed and the results compared directly to the surface DCGLs. This sampling frequency is based on a MARSSIM Class 3 sampling protocol and the sampling frequencies as prescribed in Section 6.3.3.

Based on the characterization depth profiles and verification from visual inspection and Remedial Action Support Surveys (RASS) that all of the waste has been removed (i.e., in-situ soils are less than the DCGL based on gamma radiation scans), the HPTs will perform an FSS in the survey units (Figure 6) per the MARSSIM protocols of a Class 1 survey unit and as detailed in Section 6.0. All soils will be surveyed to the applicable DCGLs based on relative depth with respect to the dose model. If the soil depth is less than 1.5 m from ground surface, the survey will be compared directly to the surface DCGLs and if it is greater than or equal to 1.5 m deep it will be compared directly to the subsurface DCGLs.

All *in-situ* sidewall soils will be surveyed to the applicable DCGLs based on relative depth with respect to the dose model. If the *in-situ* sidewall is less than 1.5 m in depth from ground surface, the survey will be compared directly to the surface DCGLs and if it is greater than or equal to 1.5 m deep it will be compared directly to the subsurface DCGLs. At no point will the contamination zone be allowed to be greater than 2 feet deep in order to ensure compliance with the dose model.

To demonstrate compliance with the dose models, at the direction of the RE and/or PHP, subsurface samples will be collected via geoprobe at each FSS location in each survey unit at excavation depths greater than 1.5 meters, to ensure that the as left condition meets the dose model and that any subsurface contaminated area does not exceed 2 feet thick. In addition, the RE and/or PHP may direct random subsurface samples via geoprobe in areas excavated to less than 1.5 meters deep in order to support the conclusion that no additional buried materials exceeding the applicable DCGL require excavation.

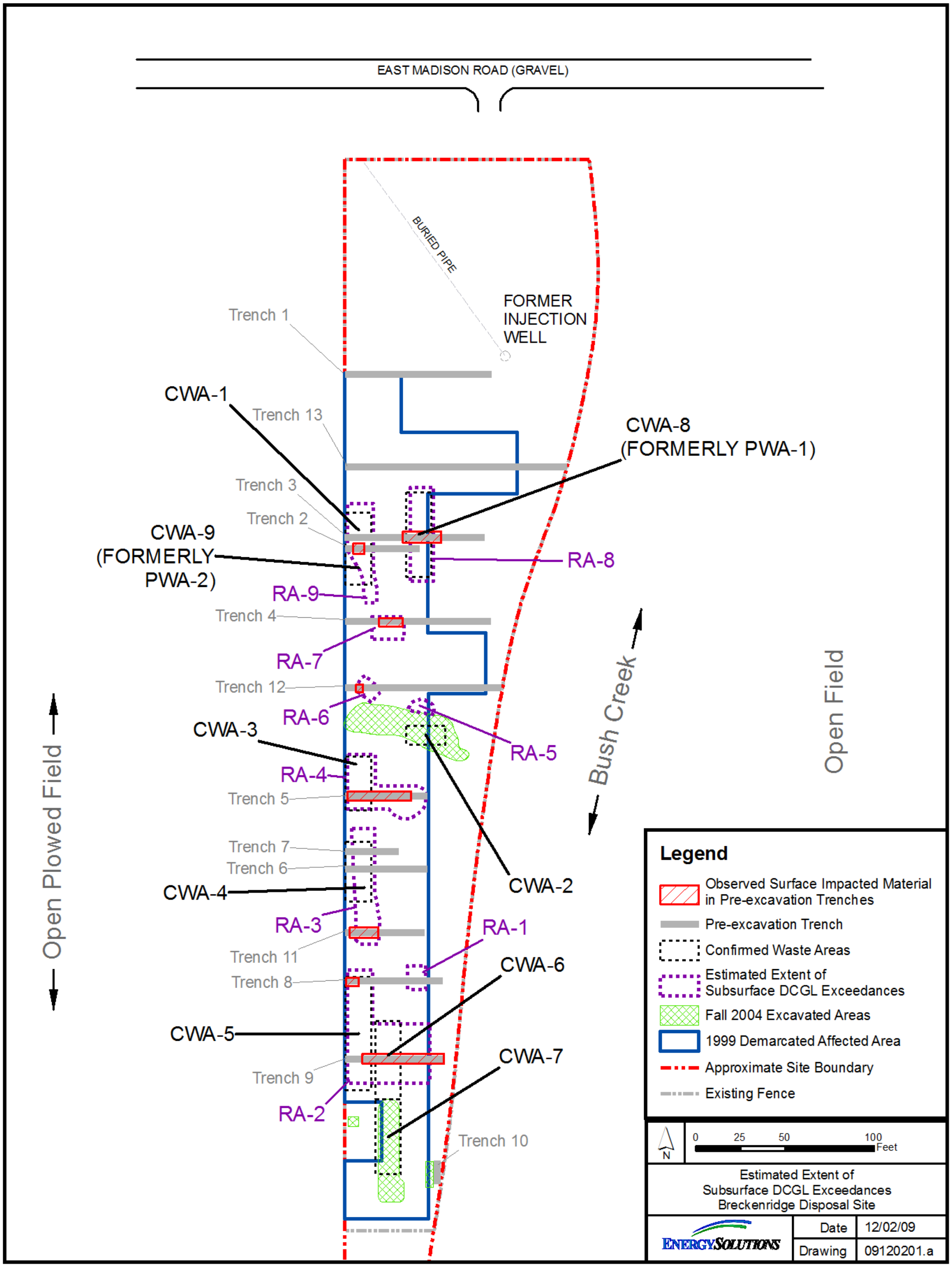

Figure 5: Breckenridge Disposal Site - CWAs and RAs

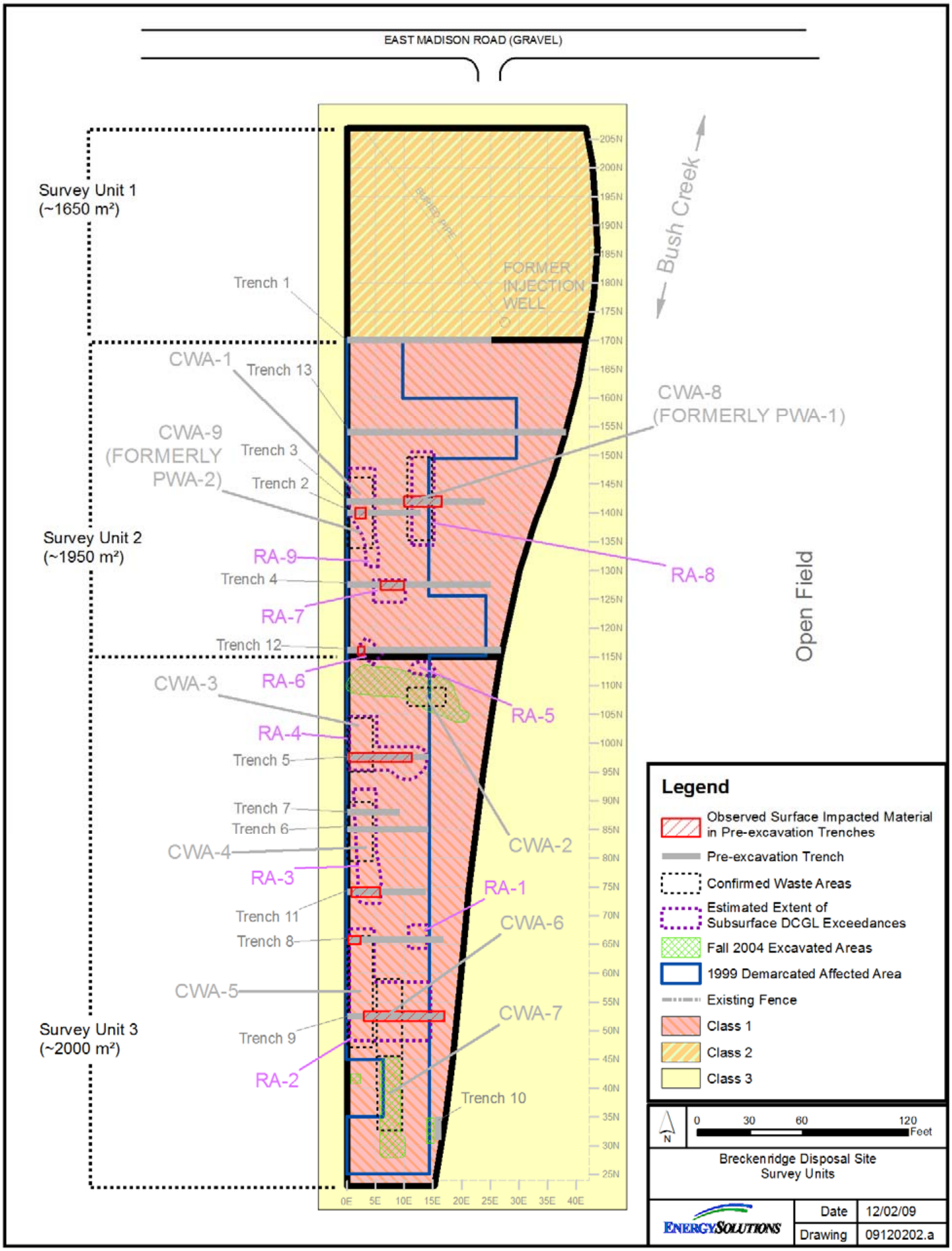

Figure 6: Breckenridge Disposal Site - Survey Units

5.1 WASTE AREAS, DEPTHS, AND VOLUMES

Table 3 provides the approximate size of each CWA, and the expected volume and potential waste areas defined in the various Site assessments and characterization efforts from 1982 through 2007 (References 1 though 9). Figure 5 outlines the locations of these CWAs in relationship to other features of the Site. The buried slit trench waste is expected to be approximately 1.2 meters (m) thick beginning at a depth of approximately 1.5 to 1.8 m below the ground surface (beneath the overburden soils).

Table 3: Estimated Slit Trench Waste Volumes

Waste Area	Approximate Size (m^2)	Estimated Volume (yd^3)[a]
CWA – 1 and 2	85	26 – 89
CWA – 3	60	18 – 43
CWA – 4	50	15 – 175
CWA – 5	75	3 – 22
CWA – 6	75	11 – 22
CWA – 7	75	22 – 91
CWA – 8 (PWA – 1)	75	22 – 129
CWA – 9 (PWA – 2)	25	2 – 8
TOTALS	**410**	**119 – 579**

[a] It was assumed in Reference 3 that the waste was buried at a rate of one 55-gallon drum per square meter (rounded up to the nearest integer) (55 gallons = 0.21 m^3 per drum).

On-site excavations are expected to remove, at a minimum, the total volume of waste presented in Table 3. However, based on process knowledge, the filtercake waste is believed to have been originally buried in fiberboard drums that have degraded over time. As such, some of the soil surrounding the trench waste will also be excavated and packaged with the filtercake for off-site disposal.

5.2 EQUIPMENT/MATERIAL AND PERSONNEL

An excavator capable of soil removal to depths of at least 3.8 m, approximately 1 m below the maximum expected waste depth, will be used for the on-site excavations. The excavator will be rented from a local firm and delivered to the Site prior to the end of mobilization. Equipment operators will be hired to operate the equipment and will receive radiation awareness and all required site-specific training as directed by the PHP/SHSO.

The HPTs will support remediation activities by performing remedial action support surveys (RASS) and FSS for each phase as directed by the PHP. The RASS and FSS will be conducted using properly calibrated scaler/rate meters coupled to 2-inch × 2-inch sodium iodide (2×2 NaI) detectors. The 2×2 NaI detectors will be used to scan surface soils, field-screen soil samples, and survey large volumes of soils in waste liners and/or containers.

Following appropriate decontamination, equipment that is to be released from the Site must be surveyed. This equipment may include, but is not limited to, the excavator, hand tools, survey instruments, buckets, personnel protective clothing, etc. These surveys will consist of scanning with a Geiger Mueller (GM) β-γ "pancake" detector (Eberline HP-260 or equivalent) coupled to a Ludlum 2221 ratemeter/scaler or equivalent. Removable surface contamination samples (smears) will also be collected and analyzed using a Ludlum Model 2929 or equivalent to assess the level of removable contamination on equipment, and reported in disintegrations per 100 square centimeters (dpm/100 cm^2).

The applicable surface contamination limits for personnel and equipment are presented in Table 4. These values are standard industry limits presented in the NRC Regulatory Guide 1.86 (Reference 10). Since the natural thorium limit is more restrictive than the natural uranium limit and field-screening techniques cannot distinguish between the two, for conservative measures, the natural thorium limit will be used.

Table 4: Regulatory Guide 1.86 – Equipment/Material Release Criteria

Nuclide	**Material Release Criteria (dpm/100 cm^2)**		
	Average[a]	**Maximum**	**Removable**
Natural Thorium	1,000	3,000	200

[a] When averaged over a maximum area of 1 m^2.

Energy*Solutions* will also perform supplemental radiological surveys of the accessible portions of the former injection well casing. The deep injection well has been previously plugged and abandoned. The survey will consist of a radiological survey for fixed and removable alpha and beta surface contamination per the limits of Table 4 above on the accessible portions of the well head. Any remedial action required for the well is specifically excluded from the scope of work.

6.0 SAMPLING AND ANALYSIS

During remediation activities and prior to FSS, the HPTs will conduct gamma radiation surveys and collect representative soil samples in support of the remediation efforts. These surveys are performed to evaluate the presence of radioactive materials to help guide and terminate the excavation efforts. Upon the completion of remediation efforts and RASS surveys, the HPTs will perform an FSS of the specific survey units as defined in Figure 6. Additionally, once the excavation have been demonstrated to meet the release criteria and the areas backfilled and graded, a walk-over survey will be performed of the entire site to ensure the surface soils have not been impacted by the excavation activities and that the site still meets the requirements for release.

6.1 SURVEY INSTRUMENTATION

Radiation detection and measurement instrumentation for the FSS will be selected to provide both reliable operation and with the best possible sensitivity to detect the ROCs. When possible, instrumentation selection will be made to identify the ROCs at levels sufficiently below the $DCGL_W$. Detector selection will be based upon detection sensitivity, operating characteristics, and expected performance in the field. The instrumentation will, to the extent practicable, use data logging to automatically record measurements to minimize any transcription errors.

6.1.1 Instrument Selection

Radiation detection and measurement instrumentation will be selected based on the type and quantity of radiation to be measured. The instruments used for direct measurements will be capable of detecting the radiation or ROCs to a MDC between 10% and 50% of the applicable $DCGL_W$, to the maximum extent practical. The use of 10% to 50% of the $DCGL_W$ is an administrative limit only. Any value below the $DCGL_W$ is acceptable.

6.1.2 Calibration and Maintenance

Instruments and detectors will be calibrated for the radiation types and energies of interest or to a conservative energy source. Instrument calibrations will be documented with calibration certificates and/or forms and maintained with the project records. Calibration labels will also be attached to all portable survey instruments. Prior to using any survey instrument, the current calibration will be verified and all operational checks will be performed.

Instrumentation used for FSS will be calibrated and maintained in accordance with approved calibration procedures. Radioactive sources used for calibration will be traceable to the National Institute of Standards and Technology (NIST) and have been obtained in standard geometries to match the type of samples being counted. When a characterized high-purity germanium (HPGe) detector is used, suitable NIST-traceable sources will be used for calibration, and the software set up appropriately for the desired geometry.

6.1.3 Scan Minimum Detectable Concentration (MDC)

Per Table 6.4 of NUREG-1507 (Reference 12), a Scan MDC of 1.8 pCi/g Th-232 for the 2×2 NaI detector will be used. Since the dose assessment was developed to determine the DCGLs with fixed independent ratios between the ROCs, and since 1.8 pCi/g can detect the presence of Th-232, then all other ROCs will be detected below their respective DCGLs.

6.1.4 Response Checks

Prior to use on-site, all project instrument calibrations will be verified and initial response data collected. These initial measurements will be used to establish performance standards (response ranges) which the instruments will be tested against on a daily basis when in use. An acceptable response for field instrumentation is an instrument reading within ±20% of the established check source value. Laboratory instrumentation standards will be within ±3-sigma as documented on a control chart with warning limits set at ±2-sigma.

The daily response tests results will be documented and compared to these operating parameters and ranges to ensure that the instrumentation is functioning properly. When an instrument fails a response test, the results will be investigated to determine the cause of failure. In the event that the instrument is not functioning properly, the instrument will be removed from service for repair and re-calibration.

6.2 REMEDIAL ACTION SUPPORT SURVEYS (RASS)

As previously stated, HPTs will perform gamma radiation surveys and collect soil samples during the excavation activities, to assist with guiding and terminating the remediation for compliance with the DCGLs. Routine RASS will be conducted to delineate waste or contaminated soil from soil that is less than DCGL, based on instrument response, visual inspection, and sample analytical results. Contaminated waste material will be loaded directly into waste liners and surveyed to provide the documentation necessary to meet the waste acceptance criteria (WAC) for the Energy*Solutions* Clive, Utah licensed disposal facility.

6.3 FINAL STATUS SURVEYS (FSS)

Since the DCGLs were derived by incorporating the unity rule into each individual DCGL and isotopic fraction in the mixture, the unity rule is inherently adhered to. Therefore, all soil concentrations will be directly compared to their respective $DCGL_W$ per Table 1. In addition, since no area factors were developed with the DCGLs, Energy*Solutions* will remediate the Site to the individual $DCGL_W$ values as provided in Table 1.

6.3.1 Survey Design

As discussed previously, radiation surveys will be performed with portable survey instruments sensitive to gamma radiation. Scanning will generally be conducted by moving the detector in a serpentine pattern over the surface at a rate that does not exceed 0.5 meters per second. Both random and biased surveys will be performed. Biased surveys will be based on results of historical surveys, walk-downs, historical use of the area, areas remediated, and professional judgment.

The average net count rate corresponding to the $DCGL_w$ (or some fraction of the $DCGL_w$) will be determined and used to guide the remediation. Once the gamma radiation surface scans indicate levels below the $DCGL_w$, samples may be collected to confirm the scan results. If the scan MDC is greater than the $DCGL_w$, scanning will still likely be initially used to guide remediation, but additional soil samples may be needed as the area approaches the levels that can be released for unrestricted use. Suspect contaminated soil will be sampled and analyzed to determine if the levels are below the $DCGL_w$.

As soil is remediated, operational gamma scans will be used to guide the remediation and segregation of soil. When gamma scans indicate that the remaining soil in the excavation area is below the remediation criteria, or samples indicate activity concentration below the $DCGL_w$, the area will be turned over for FSS.

The objective of the surveys during removal of overburdened soils intended to be used as backfill, is to perform the necessary scans prior to removal and/or during removal such that the requirements of the final status survey for the detection of “hot spots” are satisfied prior to stockpiling. The collection of a composite sample and/or performing in-situ sampling or gamma spectroscopy counting will be used to determine the average activity concentration.

6.3.2 Survey Units

Areas that have been classified based upon the contamination potential were divided into survey units as shown in Figure 6. An FSS will be performed in each survey unit and the data evaluated to demonstrate compliance with the unrestricted release criterion.

With respect to Bush Creek, as presented in Figure 6, Energy*Solutions* will perform Class 3-like FSS of Bush Creek up to 300 yards adjacent to the eastern boundary of the Site. This investigation is limited to scanning the area of the banks of Bush Creek and the perimeter of the Site with a 2×2 NaI detector using a global positioning system (GPS) and collection and analysis of up to 20 soil sediment samples for gamma emitting radionuclides, uranium and thorium by gamma spectroscopy. While this survey is anticipated to be performed consistent with the MARSSIM Class 3 protocols applicable to other portions of the Site, it is not a part of the FSS per contractual agreements and any additional effort, including remediation, is not within the scope of this work plan.

6.3.3 Sample Size Determination

Section 5.5 of Reference 13 and Appendix A of Reference 14 both describe the process for determining the number of sampling and measurement locations (sample size) necessary to ensure an adequate set of data that are sufficient for statistical analysis such that there is reasonable assurance that a survey unit will pass the requirements for unrestricted release. The number of sampling and measurement locations is dependent upon the anticipated statistical variation of the final data set, such as the standard deviation, the decision errors, and a function of the gray region as well as the statistical tests to be applied.

For the Site, the characterization results identified surface and near-surface radioactivity, as well as subsurface radioactivity that will require remediation. Because of this, residual surface and subsurface DCGLs were developed as listed in Table 1 of this Work Plan.

For clean overburden, such as the off-site borrow area, composite sampling shall be performed at a frequency of one sample for every 100 cubic yards of material. For soils excavated for reuse from a Class 1 area on the Site, composite sampling shall be performed at a frequency of at least one sample for every 20 cubic yards of material. This equates to approximately 20 samples spread out over a 2,000 m^2 survey area with a depth of approximately 0.15 meters. It should be noted that *in-situ* sampling may be performed in lieu of composite sampling prior to lift removal. However, actual sampling will consist of a combination of both *in-situ* sampling and composite sampling during excavation. This is consistent with the MARSSIM protocols for Site release.

6.3.4 Decision Errors

The probability of making decision errors is part of the decision process in establishing the performance goals for the data collection design and can be controlled by adopting a scientific approach through hypothesis testing. In this approach, the survey results will be used to select between the null hypothesis or the alternate condition (the alternative hypothesis) as defined and shown below.

- Null Hypothesis (H_0) – The survey unit does not meet the release criterion and
- Alternate Hypothesis (Ha) – The survey unit does meet the release criterion.

A Type I decision error would result in the release of a survey unit containing residual radioactivity above the release criterion, or false negative. This occurs when the null hypothesis is rejected, when in fact it is true. The probability of making this error is designated as "α".

A Type II decision error would result in the failure to release a survey unit when the residual radioactivity is below the release criterion, or false positive. This occurs when the Null Hypothesis is accepted, when it is in fact not true. The probability of making this error is designated as "β".

Appendix E of Reference 13 recommends using a Type I error probability (α) of 0.05 and states that any value for the Type II error probability (β) is acceptable. Following the guidance in Reference 13, α will be set at 0.05. A β value of 0.10 will initially be selected. The β value may be modified, as necessary, after weighing the resulting change in the number of required sampling and measurement locations against the risk of unnecessarily investigating and/or the remediation of survey units that are truly below the unrestricted release criterion.

6.3.5 Gray Region

The gray region is defined as the range of values for the specified parameter of interest for the survey unit in which the consequences of making a decision error is relatively minor. This can be explained as the range of values for which there is a potential of making a decision error; however, there is reasonable assurance that the parameters will meet the specified criteria for the rejection of the null hypothesis.

The gray region is established by setting an upper and lower boundary. Values for the specified parameter above and below these boundaries usually result in a “black and white” or “go no go” decision. Values between the upper and lower boundary are within the “gray region” where decision errors apply most. By establishing the decision errors as specified above based on acceptable risk, the number of sampling and measurement locations may be controlled within reason.

6.3.6 Upper Boundary of the Gray Region

For the purposes of the FSS; release parameters at or near the release guidelines will typically result in a relative definite decision that the survey unit will not meet the requirements for unrestricted release. As a result, the upper boundary of the gray region is typically set to the $DCGL_W$.

6.3.7 Lower Boundary of the Gray Region

The lower boundary of the gray region (LBGR) is the point at which the Type II error (β), or false positive, applies. The LBGR will initially be set at the level of residual contamination in the survey unit; however, this value may be adjusted as necessary and may be set as low as the MDC for the specific analytical technique. This will help in maximizing the relative shift and effectively reduce the number of required sampling and measurement locations based upon acceptable risks and decision errors.

6.3.8 Relative Shift

The relative shift (Δ/σ) for the survey unit data set will be calculated and is a function of the gray region and the sample statistics for the survey area. Delta (Δ) is defined as the upper boundary of the gray region, or in the case of this Site, the $DCGL_W$ minus the LBGR, while sigma (σ) is defined as the standard deviation of the data set. For survey design purposes, sigma values in a survey unit and/or reference area may initially be calculated from preliminary survey and/or investigation data to assess the readiness of a survey area for FSS. Standard deviation values as determined from the characterization data are generally not recommended for Class 1 areas as this will typically contain values in excess of the guidelines and have excessive variability which will not be representative of the conditions at the time of the FSS. The standard deviation at the time of the FSS should be approximated as best as possible to ensure the FSS requirements are not too restrictive. Optimal values for the relative shift range between 1 and 3.

6.3.9 Determining Which Test Will Be Used

Appropriate tests will be used for the statistical evaluation of the survey data. Tests such as the Sign test and Wilcoxon Rank Sum (WRS) test are implemented using the unity rule, surrogate methodologies, or combinations thereof.

If background is a significant fraction of the $DCGL_W$, the WRS test is typically used. If the contaminant is not in the background or constitutes a small fraction of the $DCGL_W$, the Sign test may be used. The specific test to be used will be predetermined by the PHP.

6.3.10 WRS Test Sample Size

The number of sampling and measurement locations, N/2, that will be collected from the reference area and survey unit will be determined by establishing the acceptable decision errors, calculating the relative shift, and using Table 5-3 of Reference 13. The shift (Δ) is the $DCGL_W$ minus the LBGR. In other words, the shift is the width of the gray region.

$$\Delta = DCGL_w - LBGR \qquad \textbf{(Equation 1)}$$

The LBGR is a site-specific variable. The standard approach is to initially set the LBGR at the anticipated mean activity of the FSS data set. The relative shift must then be calculated whether the WRS Test or the Sign Test will be performed.

$$\text{Relative Shift} = \frac{\Delta}{\sigma} \qquad \textbf{(Equation 2)}$$

The value used for σ will be an estimate of the standard deviation expected for the measurements in the survey unit or reference area, whichever is greater. Desirable values for the relative shift are from 1 to 3, noting that smaller values substantially increase the number of required sampling and measurement locations, while larger values do little to reduce the required number.

By reading the relative shift from the left side of Table 5-3 of Reference 13 and cross referencing the specified decision errors, the number of sampling and measurement locations can be determined. The specified number within the table includes a recommended 20% adjustment or increase to ensure an adequate set of data is collected for statistical purposes. Equation 5-1 of Reference 13 may alternatively be used to calculate the number of sampling and measurement locations. The result will be rounded up by 20%. Note that N/2 locations will be identified in both the survey unit and reference area. The sample size calculations may be performed using a specially designed software package such as COMPASS or, as necessary, using hand calculations and/or spreadsheets per the direction of the RE and/or PHP.

6.3.11 Sign Test Sample Size

For the Sign test, the number of sampling and measurement locations that will be required is determined from Table 5-5 of Reference 13 in a similar manner as described previously for the WRS test, except that a reference area is not used. The specified values within the table also include the recommended 20% adjustment or increase in samples to ensure an adequate set of data is collected for statistical purposes. Equation 5-2 of Reference 13 may alternatively be used to calculate the number of sampling and measurement locations. The result will be rounded up by 20%.

6.3.12 Reference Grid and Sampling and Measurement Locations

The survey sampling and measurement locations are a function of the sample size and the survey unit size. The current strategy is to utilize GPS based off of the Michigan State coordinate system.

6.3.13 Systematic Sampling and Measurement Locations

Systematic sampling and measurement locations for Class 1 and Class 2 survey units will be located in a systematic pattern or grid. The grid spacing, *L*, will be determined using Equation 3 below (form of MARSSIM Equation 5-5) based upon the survey unit size and the minimum number of sampling or measurement locations determined.

$$L = \sqrt{\frac{A}{0.866 \times n}} \qquad \textbf{(Equation 3)}$$

where: A = Area of the survey unit, and
n = Number of sampling and measurement locations.

A random number generator will be used to determine the starting coordinate from a reference point of a triangular grid pattern. At the starting coordinates, a row of points is identified parallel to the x-axis using the calculated grid spacing, *L*. A second row of points is developed, parallel to the first row, at a distance (along the y-axis) of $0.866 \times L$ from the first row, with each sample point midway between the points of the first row (along the x-axis). This process is repeated throughout the survey unit.

The grid spacing may be rounded down for ease of locating sampling and measurement locations on the reference grid. The number of sampling and measurements locations identified will be counted to ensure the appropriate number of locations has been identified. Depending upon the configuration and layout of the survey unit and the starting grid location, the minimum number of sampling and measurement locations may not be identified. In this event, either a new random starting location will be specified or the grid spacing adjusted downward until the appropriate number of locations is reached.

For Class 3 survey units, each sampling and measurement location will be randomly selected using a random number generator.

6.3.14 Biased Sampling and Measurement Locations

In addition to the systematic sampling, biased samples will be collected at elevated areas identified during the walk-over gamma scans as directed by the RE and/or PHP. This will be performed to investigate any areas of potential concern and to validate the scan sensitivities of the field instruments. As a minimum, biased sampling will be performed at a frequency of 1 sample location for every 10 linear feet along the bottom of each slit trench (centerline).

6.3.15 Geo-probe sampling

Geoprobe sampling shall be performed at each final status survey location within areas that have been excavated to a depth of greater than 1.5 meters but less then 3 meters. This includes all biased sampling locations, specifically those along the centerline of each slit trench as identified in the field. Additional samples may be collected as necessary based upon the direction of the RE and/or PHP.

The purpose of greoprobe sampling is to provide additional reassurance no further subsurface contamination exists and to demonstrate that any residual subsurface contamination does not exceed 2 feet thick per the dose model. Each core sample will be scanned with gamma detection field instrumentation along its length and the core sample segregated into specific sample depths as necessary. Provided no elevated measurements are identified from the scan of the core, the top 6-inches will be sampled and the remaining core will be composited as directed by the RE and/or PHP. If elevated scans are identified along the core, the elevated area(s) will be measured (depth) and that 1-foot section (6 inches above and below the elevated location) of the core will be sampled..

6.3.16 Data Conclusions

The results of the statistical testing allow one of two conclusions to be made. The first conclusion is that the survey unit meets the unrestricted release criterion through the rejection of the null hypothesis. The data provides statistically significant evidence that the level of residual radioactivity within the survey unit does not exceed the unrestricted release criteria. The decision to release the survey unit will then be made with sufficient confidence and without any further analyses.

The second conclusion that can be made is that the survey unit fails to meet the unrestricted release criteria. The data may not be conclusive in showing that the residual radioactivity is less than the unrestricted release criteria. As a result, the data will be analyzed further to determine the reason for failure. Potential reasons may include:

- The average residual radioactivity exceeds the $DCGL_W$;
- The survey design or implementation was insufficient to demonstrate compliance for unrestricted release, (i.e., an adequate number of measurements was not performed); or,
- The test did not have sufficient power to reject the null hypothesis (i.e., the result is due to random statistical fluctuation).

The power of the statistical test is a function of the number of measurements made and the standard deviation of the measurement data. The power is determined from $1-\beta$ where β is the value for Type II errors. A retrospective power analysis may be performed using the methods as described in Sections I.9 and I.10 of Reference 13.

If failure was due to the presence of residual radioactivity in excess of the unrestricted release criteria, the survey unit shall be remediated. Survey unit failure due to inadequate design or implementation shall require additional investigation and the re-performance of the FSS process. As a result, it is imperative that the FSS be properly designed.

6.3.17 Final Status Survey Report

The FSS planning, data, and assessment information will be compiled for each survey unit. The included documentation will provide a complete and unambiguous record of the radiological status of the survey unit relative to the established $DCGL_W$. The information provided will also allow for an independent evaluation of the survey results at a later time, including a repeat survey, commonly referred to as a confirmatory survey.

The following list provides a summary of the information that will be provided in the FSS report as given in Appendix D of Reference 14.

- Overview of the results of the FSS;
- Discussion of changes that were made in the FSS from what was proposed in this WP;
- Description of the method by which the number of samples was determined for each survey unit;
- Number of measurements/samples performed/collected in the survey unit;
- Description of the survey unit, including maps of measurement and sampling locations showing random start systematic locations for Class 1 and 2 survey units and random locations for Class 3 survey units;
- Discussion of remedial actions and unique features;
- Measured sample concentrations in units that are comparable to the $DCGL_w$;
- Statistical evaluation of the measured concentrations;
- Judgmental and miscellaneous sample data sets reported separately from systematic data;
- Discussion of anomalous data, including areas of elevated direct radiation detected during scanning that exceeded investigation levels or measurement locations in excess of the $DCGL_w$;
- A statement that the survey unit satisfied the $DCGL_w$;
- Description of any changes in the initial survey unit assumptions relative to the extent of residual radioactivity;
- Description of how ALARA practices were employed to achieve final activity levels; and
- If a survey unit fails, a description of the investigation process and a discussion of the impact of the failure on other survey units and the Site in general.

Actual measurement locations will be identified by GPS location, photographic record, or equivalent.

6.4 GAMMA SPECTROSCOPY ANALYSIS

Energy*Solutions* will use an on-site portable gamma spectroscopy system that is equipped with a high purity germanium (HPGe) detector with gamma spectrum analysis software. This combination of hardware and software will be used to conduct near real-time analysis of soil samples or containers for the ROCs.

The gamma spectrometer system will be calibrated to the various soil sample geometries that will be analyzed such as the 250 or 500 mL Marinelli beaker. The systems will be calibrated using NIST-traceable mixed gamma sources or intrinsic calibration routines. The counting system will have software-calculated MDC values that are less than or equal to the $DCGL_W$ for the ROCs measured radionuclide as identified in Table 1, with a range of 10 to 50% of the $DCGL_W$ being preferable.

For quality assurance measures, approximately 5 percent of all soil samples analyzed on-site will be sent to an independent laboratory for analysis. The on-site and off-site results will be measured for accuracy using the absolute value of the relative percent difference and referred to simply as RPD. For the purpose of these analytical results, the RPD should not exceed ± 30% for the on-site and off-site analytical sample activity concentrations when the results for the samples exceed 5x the MDC. For samples in which the analytical results are less than 5x the MDC, the RPD control limit will be within ± 50%. The RPD is calculated using equation 4 below.

$$\%RPD = \frac{|S_1 - S_2|}{\bar{S}} \times 100 \qquad \textbf{(Equation 4)}$$

Where:

S_1 = the value for the off-site sample result (pCi/g), and

S_2 = the value for the on-site sample result (pCi/g).

7.0 WASTE MANAGEMENT

Initially, excavated materials will be sorted into two separate volumes: contaminated and potentially clean. The material known to be contaminated (from visual inspections and gross gamma measurements) will be loaded directly into large-volume waste liners such as soil sacks. Potentially clean material will also be segregated. After thorough scanning with a NaI detector and visual inspection for filtercake, the potentially clean overburden soils will be sampled and analyzed using gamma spectroscopy to determine if the mean activity concentrations of uranium and thorium meet the $DCGL_W$. If the concentrations are less than the $DCGL_W$, the material will be declared as backfill for on-site reuse.

Waste shipments will comply with applicable U.S. Department of Transportation (DOT) regulations. According to 49 CFR 173, unlimited quantities of natural thorium and uranium activity are allowed in bulk waste containers. Therefore, if the following criteria are met the container is exempt from the requirements of 49 CFR 173, Subpart I, Class 7 (Radioactive) Materials and may be shipped on a generic bill of lading (BOL) as a limited quantity shipment:

- Radiation level at all points on the exterior surface of each bulk container is less than 0.5 millirem per hour (mrem/hr);
- The removable contamination limits do not exceed 22 dpm/100 cm^2 for natural uranium and natural thorium (total alpha and beta);
- The shipping container bears the marking “Radioactive”;
- The package contains less than 15 grams of uranium-235; and,
- The BOL is marked accordingly with the limited quantity statement or proper shipping name – *Radioactive Materials, excepted package – limited quantity of material, 7, UN2910*.

If these criteria cannot be met, the regulations regarding shipping papers, marking, and labeling in 49 CFR 173, Subpart H, will apply.

8.0 SITE-SPECIFIC HEALTH AND SAFETY PLAN

The site-specific Health and Safety Plan was developed as a stand-alone separate plan (SHASP, Reference 15) which addresses all radiological and non-radiological hazards that may be encountered at the Site. Among other things, the SHASP defines the levels of personnel protective equipment and provides emergency medical information. The SHASP must be read and acknowledged by everyone coming on site during potentially hazardous operations.

In regard to airborne radioactivity, Energy*Solutions* is not expecting to generate measurable airborne concentrations based on Site operations; however, general area and perimeter air sampling will be performed to monitor both the project employees and the general public. All air sample filters will be analyzed for gross alpha and beta activity over a series of days to ensure any radon and associated daughter products have decayed. General area air samples will be compared to the occupational air derived air concentration (DAC) limits and the perimeter monitors will be compared to the air effluent limits for the isotopes of concern, as listed in 10 CFR 20, Appendix B, Tables 1 and 2 respectively. As needed additional analysis may be requested, depending upon the gross alpha and beta analyses and may include gamma spectroscopy analysis.

9.0 REFERENCES

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