FINAL REPORT — INDEPENDENT CONFIRMATORY SURVEY RESULTS FOR SURVEY UNITS 1 AND 3 WITHIN THE PLANT 5 FOOTPRINT AT THE MALLINCKRODT INC. SITE ST. LOUIS, MISSOURI

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ABBREVIATIONS AND ACRONYMS

cm	centimeter
cpm	counts per minute
C-T	columbium and tantalum
$DCGL_W$	derived concentration guideline level
DP	decommissioning plan
EMC	elevated measurement comparison
FSS	final status survey
ft	feet
GPS	global positioning system
IEAV	Independent Environmental Assessment and Verification
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDC	minimum detectable concentration
MeV	million electron volts
NaI(Tl)	sodium iodide (thallium-activated)
NRC	U.S. Nuclear Regulatory Commission
ORAU	Oak Ridge Associated Universities
ORISE	Oak Ridge Institute for Science and Education
pCi/g	picocuries per gram
ROC	radionuclide of concern
SU	survey unit
TAP	total absorption peak

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1.0 INTRODUCTION

Mallinckrodt Inc. is a Delaware based company with its principal place of business located in St. Louis, Missouri. Mallinckrodt has held a Nuclear Regulatory Commission (NRC) License (STB-401) since 1961 for the extraction of columbium and tantalum (C-T) from natural and synthetic ores and slags. Prior to this, Mallinckrodt refined uranium ore and concentrate to produce uranium compounds which were used to develop atomic weapons under an Atomic Energy Commission source material license R-226 (Covidien 2008). Uranium, thorium, and rare earth elements were also extracted during this period. Mallinckrodt is currently seeking to terminate license STB-401 by remediation of radiological contamination associated with C-T production. C-T production occurred in an area designated as Plant 5 at the St. Louis Plant.

Historical information regarding radiological releases during the C-T extraction process was attained through interviews with past and present employees. The interviews revealed three different release events:

1) Raffinate tanks located north of buildings 246 and 247 overflowed on more than one occasion. Backup tanks were in place, but in some instances did not contain all spilled materials.
2) The entrained liquid from a high-pressure vacuum steam jet on the southwest roof of Building 238 occasionally sprayed into the air, which could have potentially contaminated roofs of surrounding buildings.
3) Minor spills occurred during various C-T raw material and residue handling activities outside of Plants 5, 6, and 7.

Specific information about the spills was not available. This historical information does not account for any accidental spills or releases of radioactive material from the time period in which the C-T extraction process was operational (Covidien 2008).

Decommissioning of the C-T extraction process is being performed in two phases (Covidien 2008). Phase I included decommissioning of the buildings and equipment to meet the NRC's guidelines for unrestricted release. Phase II includes remediation of the building slabs and foundations, paved surfaces, and all subsurface materials. Phase I was completed in December 2004 and Phase II began in July 2010.

2.0 SITE DESCRIPTION

The Mallinckrodt Inc. site is a 43-acre property occupying approximately 12 city blocks near the west bank of the Mississippi River in the northeastern section of St. Louis, Missouri. The Mallinckrodt property address is 3600 North Second Street, St. Louis, Missouri. Specifically, Plant 5 is located south of Destrehan Street and east of Second Street (Figure A-1).

3.0 OBJECTIVES

The objectives of the confirmatory activities were to provide independent contractor field data reviews and to generate independent radiological data for use by the NRC in evaluating the adequacy and accuracy of the contractor's procedures and final status survey (FSS) results.

4.0 DOCUMENT REVIEW

Prior to on-site activities, the Oak Ridge Institute for Science and Education (ORISE) reviewed the licensee's C-T Phase II Decommissioning Plan (DP). The DP was specifically reviewed for historical information, as well as to identify the radionuclides of concern (ROCs) and the applicable derived concentration guideline levels ($DCGL_W$). The purpose of this review was to ensure that the regulatory requirements were being met by the licensee and to develop the confirmatory survey plan. ORISE was to ensure that the current FSS activities within the Plant 5 area were adequate and appropriate, taking into account any supporting documentation and *Multi-Agency Radiation Survey and Site Investigation Manual* (MARSSIM) guidance (NRC 2000).

5.0 PROCEDURES

The ORISE survey team made two survey trips to the Mallinckrodt site to evaluate radionuclide concentrations in soils within the Plant 5 area. Survey efforts included performing visual inspections, measurements, and sampling activities. The first site visit occurred on April 28, 2011 and included

survey activities within Survey Unit (SU) 1 (Figure A-1). This SU was an approximate 100 feet (ft) by 60 ft soil area west of the C-T pad. Due to the licensee's schedule for backfilling the excavation, the contractor collected preliminary data and the results indicated that SU 1 was ready for FSS activities. Therefore, when ORISE arrived, the FSS activities by the contractor were in process.

ORISE began confirmatory surveys and identified several locations having elevated levels of radioactivity present within SU 1. ORISE also observed the contractor's procedure for collecting soil samples. The contractor sieved the soil samples resulting in the removal of the contaminated slag material (greater than the sieve size) and the excess material/slag was left behind in the surveyed area.

As a result of the findings from the first site visit, and at the request of the NRC, ORISE performed a follow up site visit to resurvey this area, during the period of June 1 through June 2, 2011. During the time between the two site visits, the FSS contractor indicated emergency sewer repairs were performed on an adjacent building which compromised the original SU 1 boundaries. The contractor redefined the original SU 1 boundary and performed additional soil removal. The contractor also had SU 3 ready for FSS at the time of the second visit (Figure A-2). SU 2 (not shown in the Figures), west of SUs 1 and 3, had been backfilled prior to the second visit. Additionally, soil to be used as backfill was also subject to confirmatory surveys and sampling. However, it was not available during either site visit.

ORISE survey activities were conducted in accordance with a project-specific confirmatory survey plan, the ORISE/Independent Environmental Assessment and Verification (IEAV) *Survey Procedures Manual*, and the Oak Ridge Associated Universities (ORAU) *Quality Program Manual* (ORISE 2011a, ORISE 2008, and ORAU 2011a). Questions and concerns were brought to the immediate attention of the NRC and are also noted in the Findings and Results section of this report.

The SUs were classified (by the contractor) based on contamination potential, as either Class 1, 2, or 3 in accordance with the MARSSIM (NRC 2000). A description of each class designation is as follows:

Class 1: Buildings or land areas that have a significant potential for radioactive contamination (based on site operating history) or known contamination (based on previous radiological surveys) that exceeds the expected site $DCGL_W$ value.

Class 2: Buildings or land areas, often contiguous to Class 1 areas that have a potential for radioactive contamination, but at levels less than the expected $DCGL_W$.

Class 3: Remaining impacted buildings and land areas that are not expected to contain residual contamination or are expected to contain levels of residual contamination at a small fraction of the $DCGL_W$.

5.1 REFERENCE SYSTEM

Survey scan data were referenced to global positioning system (GPS) coordinates. The Universal Transverse Mercator 15N North American Datum 1983 coordinate system was used for this confirmatory survey, with units represented in meters. Positional accuracy was within 0.5 meters at the 95th percentile.

5.2 GAMMA WALK-OVER SCANS

High density surface scans were performed within the SUs. Scans were performed using 2 inch × 2 inch sodium iodide, thallium-activated [NaI(Tl)] detectors coupled to ratemeter-scalers with audible indicators (Figures A-3 and A-4). ORISE field personnel relied on the audio output to identify any locations of elevated direct gamma radiation that might suggest the presence of residual contamination. Gamma detectors were also coupled to GPS systems that enable real-time gamma count rate and position data capture. Locations of elevated direct gamma radiation were marked with flags for further investigation during the sampling phase.

The contractor performed FSS scans alongside confirmatory activities during the first and second site visits. ORISE observed that the contractor technicians were not relying on the audible output of the instrument to pinpoint judgmental locations real time. Instead their process was to post process the data and go back and investigate suspect locations.

5.3 SOIL SAMPLING

ORISE observed that slag material was scattered throughout the SUs. If this material were present at a specific sample location, it was collected as part of the sample. Surface soil samples, at 15 centimeter (cm) depths, were collected from judgmental locations that exhibited an elevated detector response. A static one minute gamma count was taken prior to sample collection and an approximate gamma count rate was recorded after the sample was collected. If the detector response significantly increased after collecting the initial sample, an additional sample was collected from the same location either at a greater depth or immediately adjacent to the initial sample. The sample material was also screened during sample collection to ensure the material exhibiting elevated radiation levels was obtained. Two additional samples were collected following the collection of sample S006. Sample locations from the two site visits are presented in Figures A-5 and A-6.

During the first site visit, 12 confirmatory soil samples were collected from eight locations. FSS samples were also being collected by the contractor during the ORISE confirmatory activities. ORISE observed that the greater than one-quarter inch (+¼ inch) sample material was sieved out of the sample, by the contractor, in the field and only the less than one-quarter inch (-¼ inch) material was retained as the sample. ORISE also observed that the larger +¼ inch portion of the sample was noted to have contained slag material.

During the second visit, 13 confirmatory soil samples were collected within SUs 1 and 3. SU 2 had already been backfilled at the time of the second site visit and was not originally planned for confirmatory activities; however, three samples (S026, S027 and S028) were collected at the request of the NRC. Similarly, the FSS contractor was still performing FSS scans within the SUs.

6.0 SAMPLE ANALYSIS AND DATA INTERPRETATION

Soil samples and data were returned to the ORISE facility in Oak Ridge, Tennessee for analysis and interpretation. All sample analyses were performed in accordance with the ORISE Laboratory Procedures Manual (ORISE 2011b). Soil samples were analyzed by solid-state gamma spectroscopy for gamma-emitting ROCs (Tables B-1 and B-2). Additionally, select samples from the second site visit were analyzed by alpha spectroscopy (Table B-3). Analytical results were reported in units of

picocuries per gram (pCi/g). The data generated were compared with the approved $DCGL_W$s established for the Mallinckrodt site (Table 1).

Due to the larger slag material observed in the field during sample collection, the ORISE Project Manager requested the laboratory staff first count the entire sample and then sieve each sample. Both the -¼ inch and +¼ inch sample material were then counted separately via gamma spectroscopy.

7.0 APPLICABLE SITE GUIDELINES

The ROCs from the C-T process and associated $DCGL_W$s are listed in Table 1 (Covidien 2008). Thorium, radium, and lead are separated from the uranium series because the dose factor for these radionuclides is much higher than the other daughters in the series.

TABLE 1: APPLICABLE DERIVED CONCENTRATION GUIDELINE LEVELS ($DCGL_W$s) FOR SOILS

RADIONUCLIDES OF CONCERN	$DCGL_W$ (pCi/g)
Th Series	23.9[a]
Natural Uranium	721[b]
Th-230 + Ra-226 + Pb-210	29.4[c]

[a]$DCGL_W$ of the thorium series is referenced to Th-232.
[b]$DCGL_W$ of natural uranium is referenced to U-238.
[c]$DCGL_W$ of Th-230, Ra-226 and Pb-210 is referenced to Ra-226.

8.0 FINDINGS AND RESULTS

The results for each of the confirmatory activities are discussed in the following subsections.

8.1 GAMMA WALK-OVER SCANS

Gamma walk-over scans were performed during both site visits. Numerous locations exhibiting elevated direct gamma radiation were identified during the scanning phase and marked with flags for further investigation during the sampling phase.

For the first site visit on April 28, 2011, the gamma walkover scan count rate ranged from less than 9,000 to 66,000 counts per minute (cpm) with a mean count rate of 12,600 cpm. Detector responses were the highest at the corners and western edge of the SU where count rates were in excess of 60,000 cpm (Figure A-3).

For the second site visit, during the period of June 1 through 2, 2011, the gamma walkover scan count rate ranged from less than 9,000 to 83,000 cpm with a slightly higher mean count rate of 14,000 cpm. However, there were fewer areas with significantly elevated count rates (Figure A-4). The additional remediation following the emergency sewer repair removed a significant portion of the contamination observed during the first site visit.

8.2 RADIONUCLIDE CONCENTRATIONS IN SOIL SAMPLES

Pre- and post-sample gamma count rates, which were measured at each sample location, are provided in Tables B-1 and B-2. These gamma count rates indicated that elevated activity remained beneath or immediately adjacent to the initial soil samples at several locations. Therefore, ORISE collected several additional soil samples (samples S005, S007, S008, S011, S014, and S027) in the immediate vicinity of these initial sample locations (S004, S006, S010, S013 and S026) to chase the contamination.

Table B-1 lists gamma spectroscopy results for each of the confirmatory soil samples collected during the first site visit. Gamma spectroscopy results confirmed that four (S002, S003, S005 and S011) of the 12 samples (Figure A-5) were clearly above the $DCGL_W$ for one or more of the ROCs. Laboratory analyses also indicated significant activity in the slag (+¼ inch sample material) in three of the 12 soil samples (S002, S005 and S011) with U-238 being present in the highest concentrations within the slag. The gamma spectroscopy results were provided to the NRC and the alpha spectroscopy analysis was deemed unnecessary. Based on the initial findings of the first site visit, the NRC requested that ORISE perform a second site visit after the issues determined during the first site visit were addressed by the contractor.

Table B-2 lists gamma spectroscopy results for each of the confirmatory soil samples collected during the second site visit. During the second site visit, 13 soil samples were taken from 12 locations within SUs 1 and 3 (Figure A-6). Also, as requested by the NRC, three samples (S026, S027, and S028) were collected from two locations within SU 2 for which the majority of the SU had already been backfilled. Due to an ORISE laboratory error the samples from the second site visit were not analyzed by the same methodology as the samples from the first site visit. Laboratory personnel sieved the samples and inadvertently discarded the +¼ inch sample material in a radioactive waste container (in accordance with the standard ORISE laboratory procedure for soil

samples). Therefore, only the -¼ inch material from the second set of samples was analyzed. Of the 16 samples, two samples (S026 and S027) from SU 2 exceeded the $DCLG_W$ for Ra-226. All of the second site visit samples were below the $DCGL_W$ for the Th and U series.

However, since the +¼ inch sample material was discarded, ORISE reviewed the gamma count rates recorded at each sample location as an indicator for samples from the second site visit that possibly could have exceeded the $DCGL_W$s. A review of sample S011, from the first set of data (Table B-1) indicates a significant decrease in the gamma count rate after the sample was collected. The sample analytical results for S011 also indicate that the $DCGL_W$ for Ra-226, Th-232 and U-238 were each exceeded. Sample S015 (Table B-2), collected during the second site visit, also indicated a significant gamma count rate decrease post sample collection. This indicates that the portion of this sample which contained the elevated radionuclide concentrations was inadvertently discarded prior to sample analysis. Sample S019 also indicated a post collection gamma count rate that increased and elevated radionuclide concentrations for Pb-210, Th-230, Ta-226, and U-238, although below respective $DCGL_W$ values.

Uranium isotopic concentrations were determined by alpha spectroscopy for selected samples (S015, S020, S025 and S028) from the second site visit using total sample dissolution and extraction chromatography (Table B-3). A small particle exhibiting elevated activity was identified and removed from sample S015, dissolved, diluted to 500mL, then a 10mL aliquot was analyzed. The uranium concentrations for the small particle isolated from sample S015, designated as S015X, greatly exceed the $DCGL_W$ via this analysis method. As stated in the preceding paragraph, this also indicates that radionuclide concentrations within the +¼ inch portion of sample S015 may have exceeded the $DCGL_W$ limits. The uranium isotopic analyses for these samples were in agreement with the gamma spectroscopy analyses.

9.0 SUMMARY

At the request of the NRC, ORISE conducted confirmatory survey activities within the Plant 5 area at the Mallinckrodt, Inc. site on April 28, 2011 and during the period of June 1 through June 2, 2011. The survey activities included visual inspections/assessments, gamma measurements, and soil sampling activities.

ORISE is of the opinion, based on the independent measurement and sample results obtained during the first site visit, that the original SU was not ready for FSS activities or confirmatory activities. Concentrations of the ROCs were in excess of the $DCGL_W$s in four of the twelve confirmatory samples. During this site visit, larger slag material exhibiting elevated activity was observed in the field by ORISE (and noted to the NRC), yet the FSS contractor was sieving the +¼ inch sample material out of each sample in the field and leaving it behind in the excavation. The NRC took custody of ORISE sample S002 and requested that the site laboratory perform a preliminary analysis of the sample. The site lab's purpose is to screen samples prior to sending them to an offsite laboratory for analysis. Typically when the site lab receives a sample, only the -¼ inch material remains. The NRC requested a quick gamma spectroscopy count of the entire sample. The result slightly exceeded the $DCGL_W$ for Ra-226. Then the NRC requested the +¼ inch material be sieved and counted separately. This result greatly exceeded the $DCGL_W$ for Ra-226.

ORISE also observed that the FSS technicians were not relying on the audible output of the instrument to pinpoint judgmental locations real time. Their method was to post process the data in the office then go back and investigate suspect locations. However, listening to the audible output would enable the FSS technician to stop and pause at elevated locations allowing the radiological instrumentation to accumulate higher counts than would be observed when simply passing over it. If the FSS contractor had utilized the audible feature in combination with an in-field screening method during the sample sieving process, they may have also identified the +¼ inch slag material as a radiological contamination issue.

The independent measurement and sample results obtained during the second site visit identified two sample locations within Survey Unit 1 that, in all probability and based on the discarding of the +¼ inch sample material, exceeded the $DCGL_W$s. Sample S019 radiological analyses indicated the presence of elevated concentrations of Pb-210, Th-230 and Ra-226 and a particle isolated from sample S015 (designated as S015X) contained excessive uranium concentrations. Sample S020 also had elevated Th-230 concentration but Ra-226 and Pb-210 concentrations were not elevated as in sample S019. Two soil samples (S026 and S027), from one sample location within SU 2, exceeded the $DCGL_W$ for Ra-226.

Although ORISE identified a number of locations within Survey Units 1 and 3 which were in excess of the $DCGL_W$s, the licensee may use elevated measurement comparison (EMC) calculations to determine $DCGL_{EMC}$s for small areas of elevated activity. The calculated $DCGL_{EMC}$s should demonstrate that these small areas of localized contamination will meet the unrestricted release criteria established in the Decommissioning Plan.

10.0 REFERENCES

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U.S. Nuclear Regulatory Commission (NRC). *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM), NUREG-1575; Revision 1.* Washington, DC. August, 2000.

APPENDIX A
FIGURES

Figure A-1: Overview of the survey unit verified on April 28, 2011

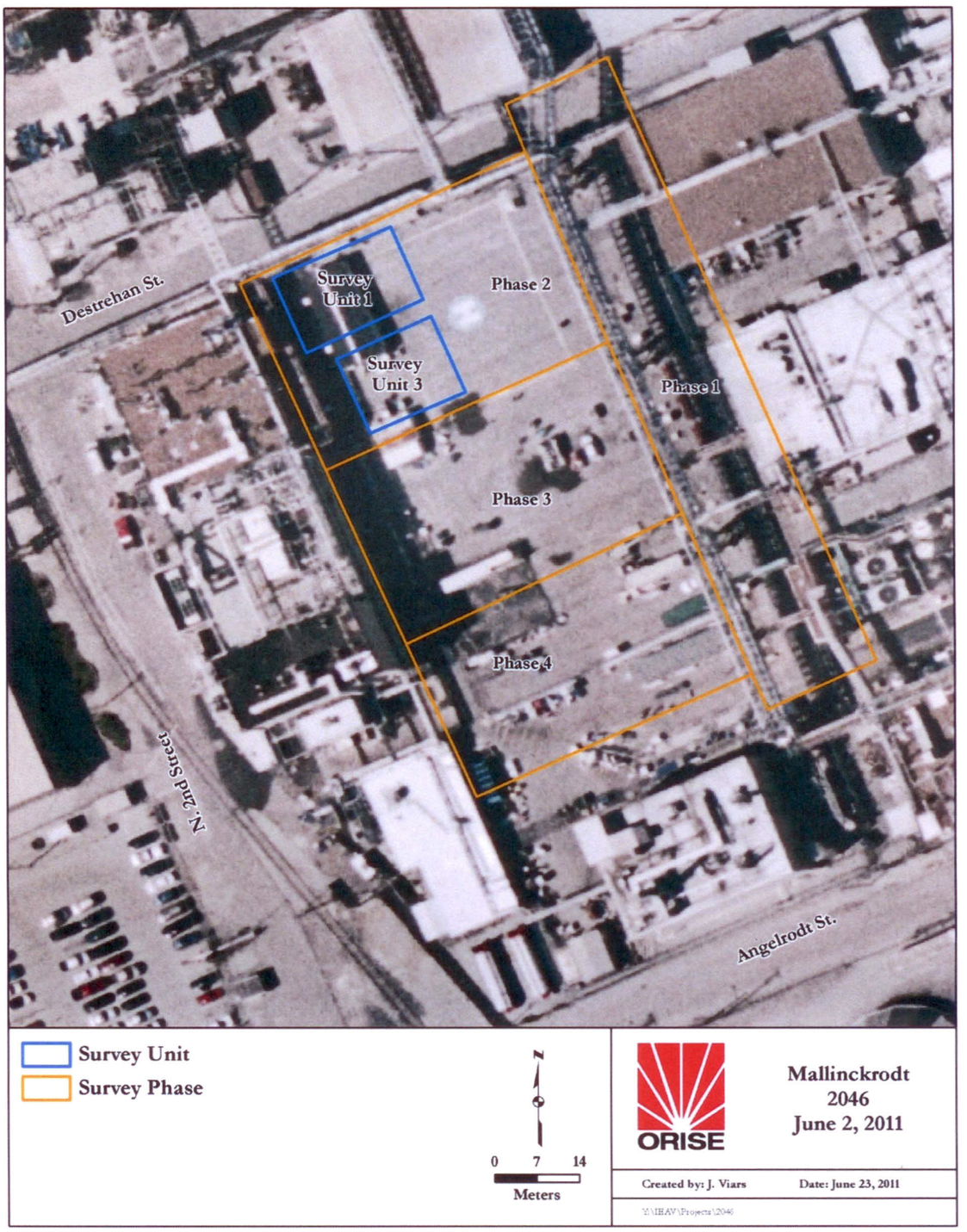

Figure A-2: Overview of the survey units verified during June 1 through 2, 2011

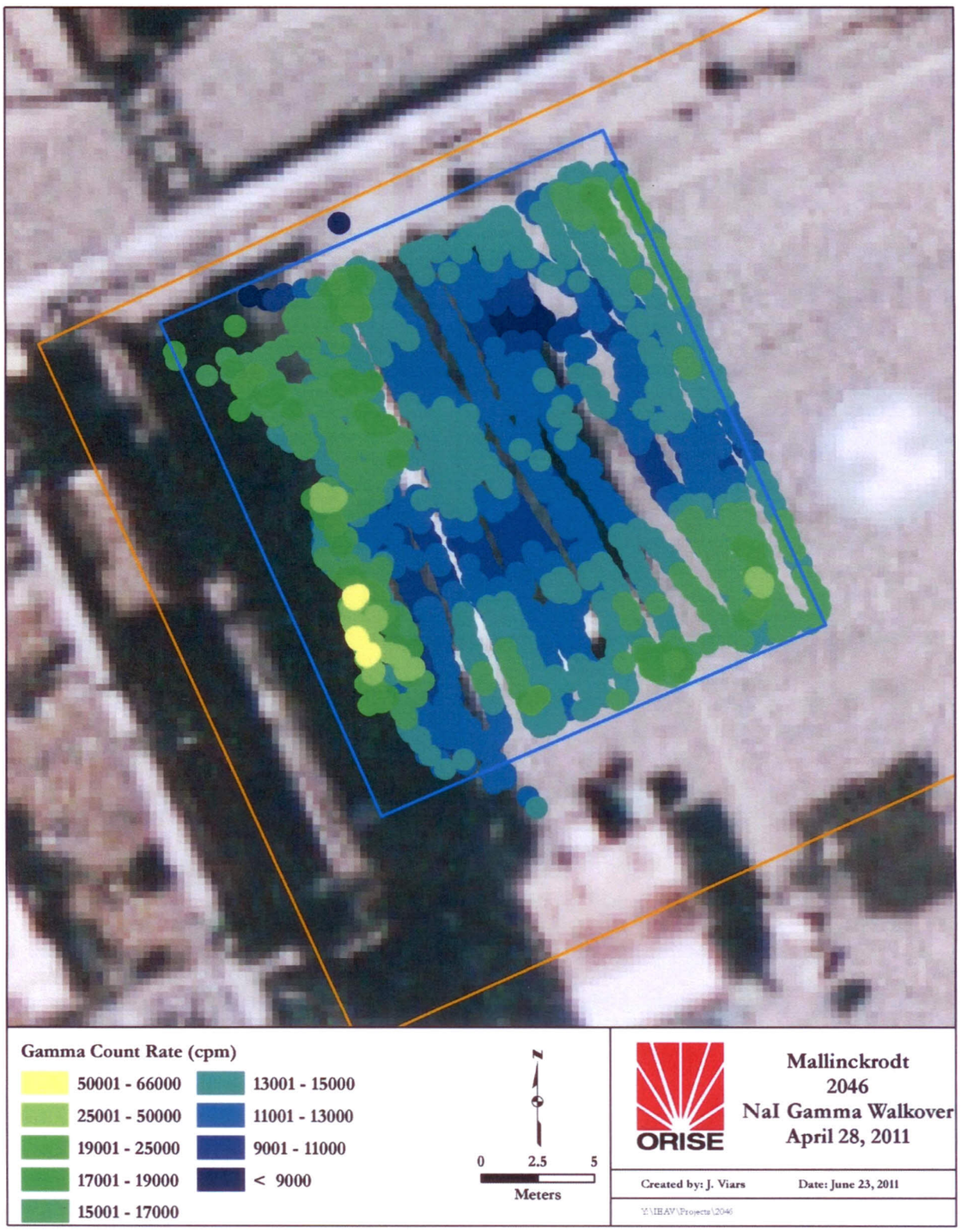

Figure A-3: NaI gamma scans performed on April 28, 2011

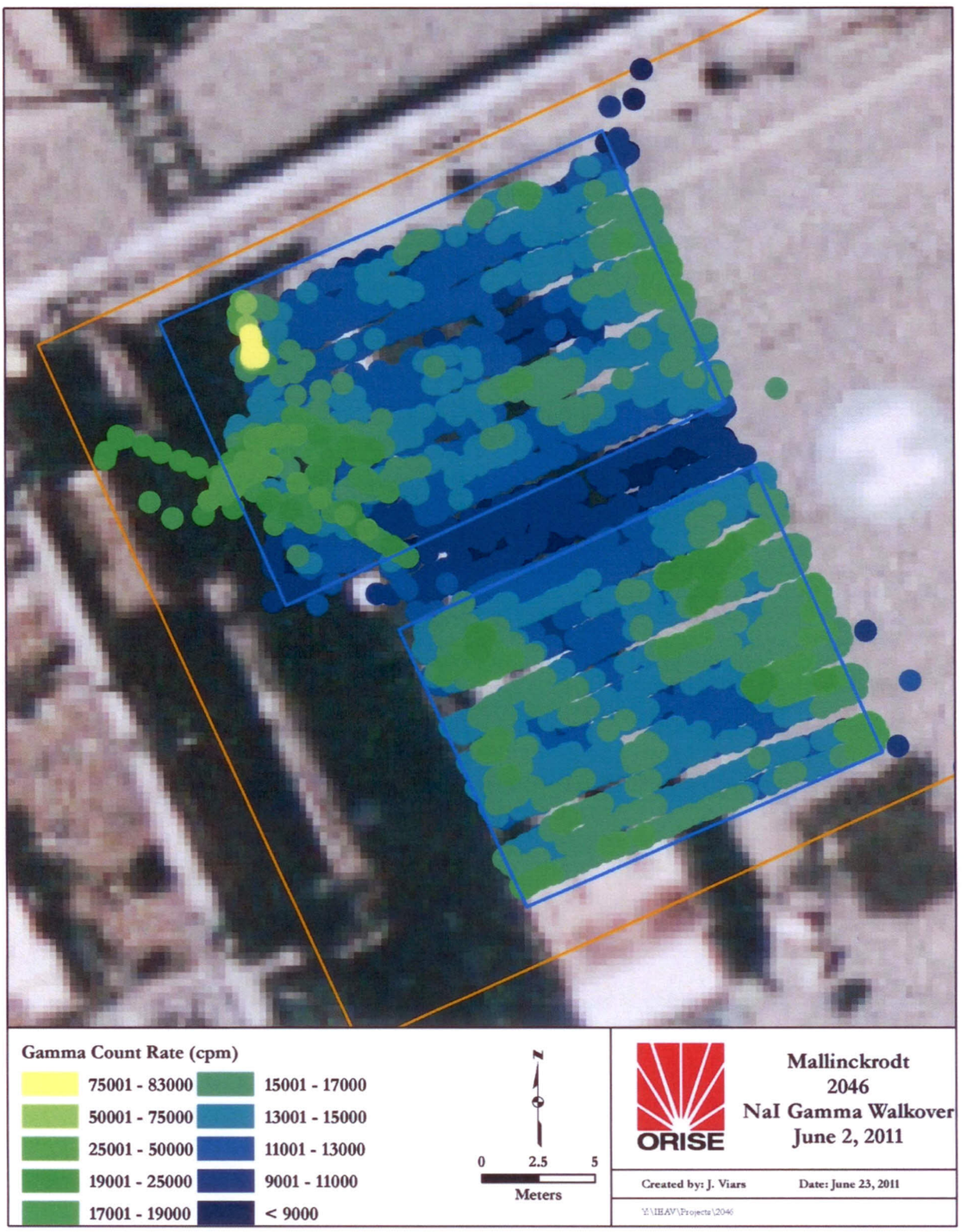

Figure A-4: NaI gamma scans performed June 1 through 2, 2011

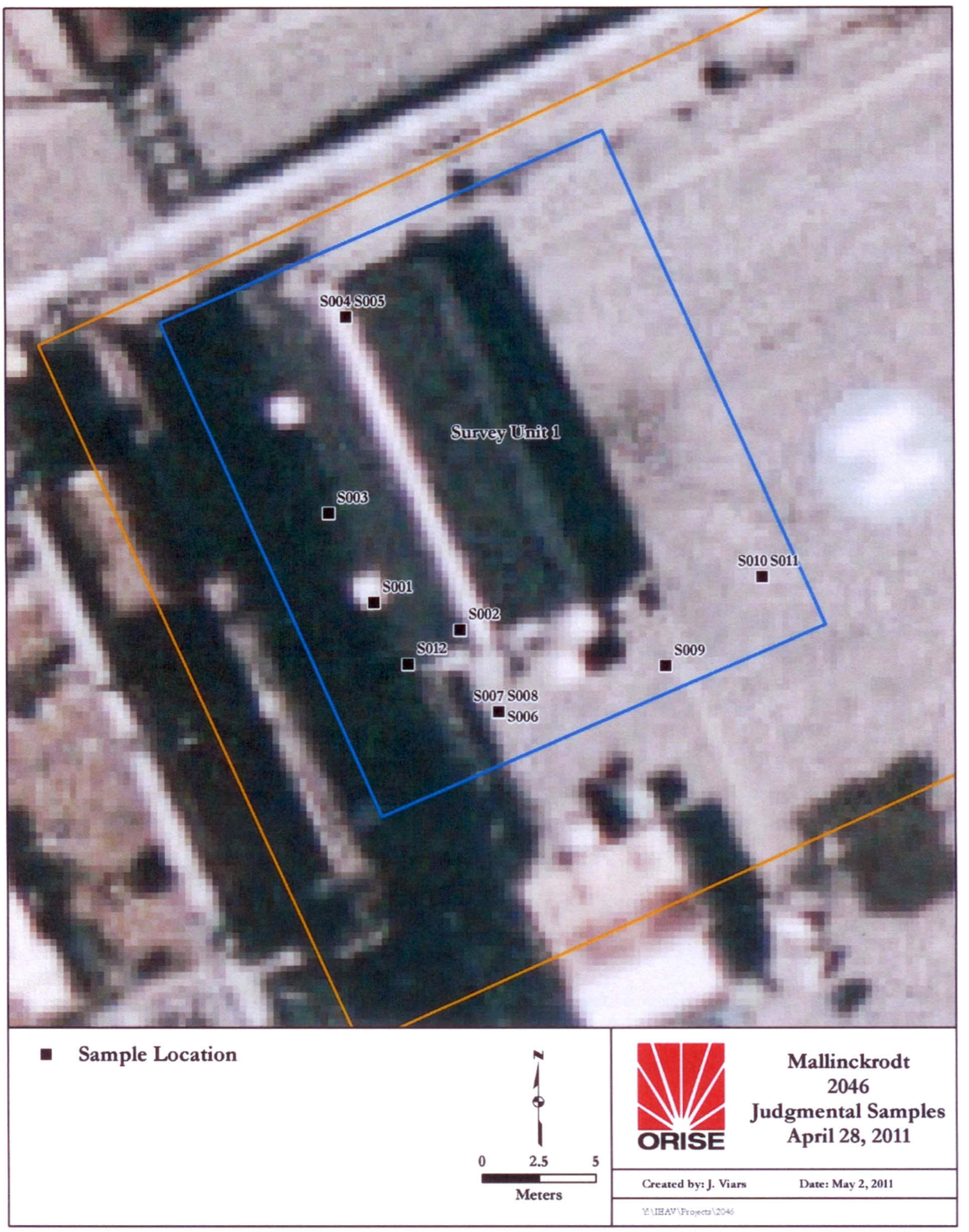

Figure A-5: Samples collected on April 28, 2011

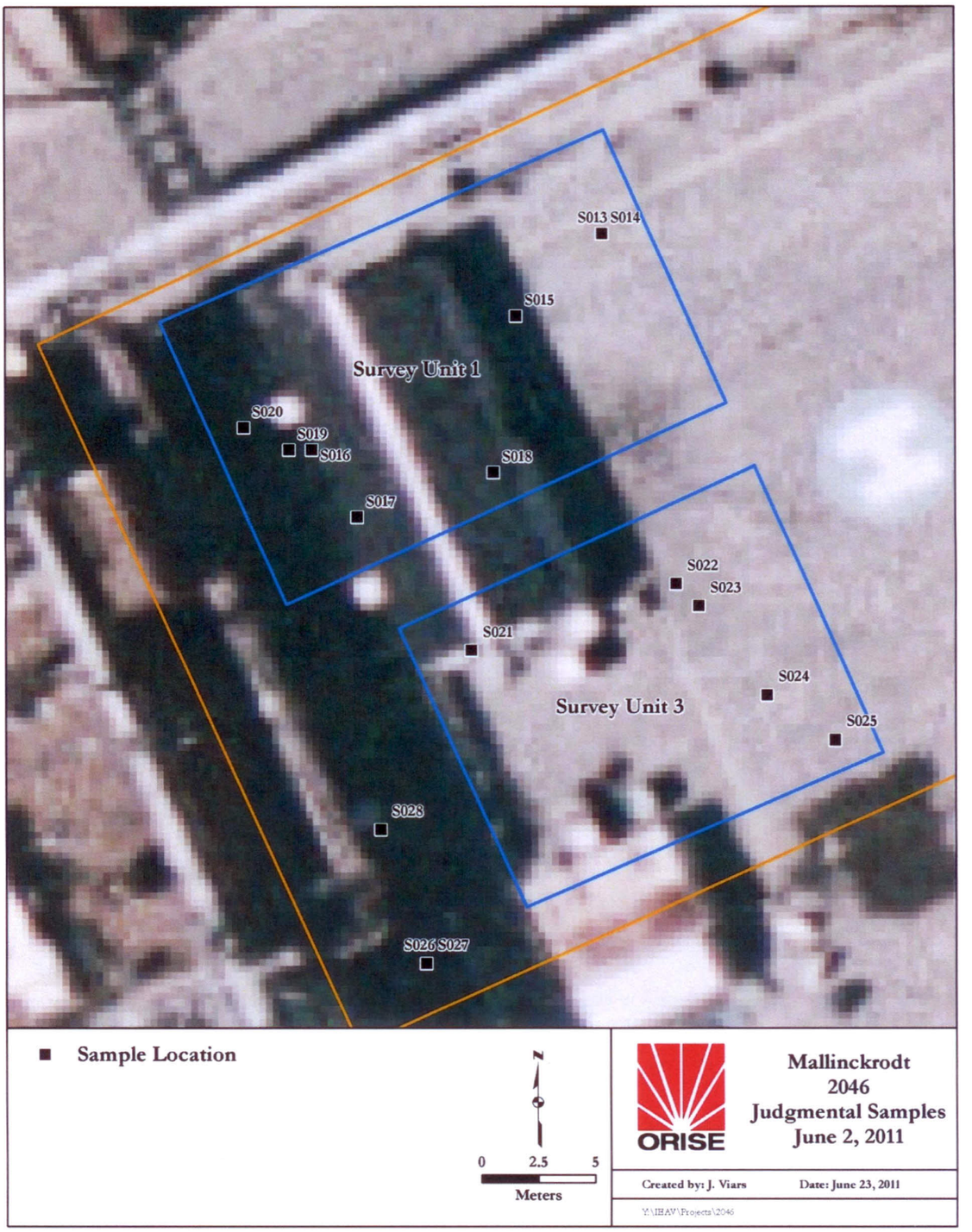

Figure A-6: Samples collected June 1 through 2, 2011

APPENDIX B
TABLES

TABLE B-1:
RADIONUCLIDE CONCENTRATIONS VIA GAMMA SPECTROSCOPY
FOR THE SITE VISIT ON 4/28/2011

ORISE Sample Location[a]	NaI Response (kcpm)		RADIONUCLIDE CONCENTRATIONS (pCi/g)								
	Pre-Sample	Post-Sample	Pb-210	Th-230	Ra-226[b]	Th-228	Th-232[b]	Total Th[c]	U-238[b]	U-235	Total U[d]
S001	52	45	11.2 ± 2.4[e]	280 ± 31	17.7 ± 1.1	0.40 ± 0.17	1.23 ± 0.31	1.63 ± 0.35	170 ± 20	11.49 ± 0.94	351 ± 20
S001A			6.6 ± 1.2	73 ± 11	6.75 ± 0.46	0.80 ± 0.10	1.01 ± 0.22	1.81 ± 0.24	86 ± 11	5.20 ± 0.47	177 ± 11
S001B			16.6 ± 2.2	407 ± 33	29.1 ± 1.8	-1.24[f] ± 0.17	1.25 ± 0.24	0.01 ± 0.29	260 ± 20	16.5 ± 1.1	537 ± 20
S002	160	NA	40 ± 10	2,470 ± 190	322 ± 18	-38.5 ± 3.0	-1.3 ± 3.3	-39.8 ± 4.5	1,410 ± 120	88.9 ± 5.8	2,909 ± 120
S002A			85 ± 14	4,100 ± 300	587 ± 32	-30.8 ± 2.7	0.4 ± 4.8	-30.4 ± 5.5	2,400 ± 190	152.7 ± 9.5	4,953 ± 190
S002B			20.2 ± 2.4	493 ± 37	41.4 ± 2.4	-1.95 ± 0.26	1.36 ± 0.24	-0.59 ± 0.35	225 ± 18	17.8 ± 1.2	468 ± 18
S003	31	23	38.9 ± 3.8	33 ± 11	32.3 ± 1.8	0.76 ± 0.10	1.01 ± 0.23	1.77 ± 0.25	17.2 ± 5.7	1.39 ± 0.34	35.8 ± 5.7
S003A			16.8 ± 1.8	-14 ± 16	13.65 ± 0.86	0.81 ± 0.09	0.93 ± 0.20	1.74 ± 0.22	-1 ± 11	0.52 ± 0.19	-1 ± 11
S003B			43.9 ± 4.1	40 ± 10	41.9 ± 2.6	-0.70 ± 0.13	0.90 ± 0.19	0.20 ± 0.23	18.5 ± 4.9	1.08 ± 0.26	38.1 ± 4.9
S004[g]	23	35	5.17 ± 0.77	34.0 ± 6.0	4.51 ± 0.29	1.36 ± 0.11	1.37 ± 0.20	2.73 ± 0.23	25.2 ± 3.9	1.38 ± 0.16	51.8 ± 3.9
S004A			4.59 ± 0.78	16.5 ± 5.1	3.59 ± 0.25	1.55 ± 0.14	1.52 ± 0.23	3.07 ± 0.27	11.7 ± 3.3	0.76 ± 0.14	24.2 ± 3.3
S004B			6.9 ± 1.0	51.5 ± 7.7	6.04 ± 0.37	1.41 ± 0.12	1.58 ± 0.23	2.99 ± 0.26	32.6 ± 4.8	2.17 ± 0.23	67.4 ± 4.8
S005[g]	39	29	3.1 ± 1.1	41 ± 12	8.30 ± 0.49	1.05 ± 0.11	1.16 ± 0.19	2.21 ± 0.22	385 ± 27	6.50 ± 0.51	777 ± 27
S005A			6.0 ± 2.1	285 ± 31	35.6 ± 2.1	0.09 ± 0.19	1.10 ± 0.27	1.19 ± 0.33	1,860 ± 140	39.7 ± 2.7	3,760 ± 140
S005B			5.85 ± 0.93	48.9 ± 7.8	5.28 ± 0.33	1.44 ± 0.12	1.80 ± 0.25	3.24 ± 0.28	34.8 ± 5.0	2.17 ± 0.22	71.8 ± 5.0
S006[h]	21	52	2.66 ± 0.46	-1.6 ± 6.8	2.09 ± 0.14	0.71 ± 0.07	0.78 ± 0.13	1.49 ± 0.15	5.5 ± 1.7	0.31 ± 0.09	11.3 ± 1.7
S006A			2.7 ± 2.5	3.8 ± 9.1	2.18 ± 0.17	0.82 ± 0.09	0.87 ± 0.17	1.69 ± 0.19	6.9 ± 6.8	0.25 ± 0.11	14.1 ± 6.8
S006B			3.63 ± 0.64	-56 ± 13	2.69 ± 0.20	0.80 ± 0.08	0.93 ± 0.16	1.73 ± 0.18	4.7 ± 2.3	0.35 ± 0.11	9.8 ± 2.3
S007[h]	52	84	2.84 ± 0.66	-114 ± 19	2.50 ± 0.18	0.67 ± 0.08	0.90 ± 0.16	1.57 ± 0.18	6.5 ± 2.7	0.44 ± 0.13	13.4 ± 2.7
S007A			3.06 ± 0.57	-2.3 ± 7.9	2.75 ± 0.20	0.92 ± 0.09	0.98 ± 0.16	1.90 ± 0.18	5.0 ± 2.5	0.22 ± 0.09	10.2 ± 2.5
S007B			3.90 ± 0.56	15.1 ± 3.8	3.15 ± 0.20	0.80 ± 0.07	0.87 ± 0.13	1.67 ± 0.15	10.4 ± 2.3	0.65 ± 0.11	21.5 ± 2.3
S008[h]	84	65	3.31 ± 0.55	11.1 ± 3.8	3.07 ± 0.21	0.89 ± 0.08	0.79 ± 0.13	1.68 ± 0.15	6.3 ± 2.4	0.49 ± 0.10	13.1 ± 2.4

TABLE B-1:
RADIONUCLIDE CONCENTRATIONS VIA GAMMA SPECTROSCOPY
FOR THE SITE VISIT ON 4/28/2011

ORISE Sample Location[a]	NaI Response (kcpm)		RADIONUCLIDE CONCENTRATIONS (pCi/g)								
	Pre-Sample	Post-Sample	Pb-210	Th-230	Ra-226[b]	Th-228	Th-232[b]	Total Th[c]	U-238[b]	U-235	Total U[d]
S008A			2.67 ± 0.71	-14 ± 11	2.49 ± 0.21	0.89 ± 0.10	1.00 ± 0.21	1.89 ± 0.23	2.6 ± 7.4	0.34 ± 0.26	5.5 ± 7.4
S008B			3.2 ± 1.1	17.9 ± 7.8	3.54 ± 0.31	0.61 ± 0.10	0.77 ± 0.23	1.38 ± 0.25	14.3 ± 5.6	0.74 ± 0.23	29.3 ± 5.6
S009	18	20	8.5 ± 1.1	-17 ± 14	5.26 ± 0.34	1.09 ± 0.11	1.36 ± 0.22	2.45 ± 0.25	9.2 ± 3.2	0.55 ± 0.15	19.0 ± 3.2
S009A			8.2 ± 1.2	13 ± 12	5.62 ± 0.39	1.34 ± 0.14	1.09 ± 0.24	2.43 ± 0.28	6.7 ± 8.5	0.44 ± 0.17	13.8 ± 8.5
S009B			6.07 ± 0.79	13.6 ± 4.9	5.08 ± 0.33	1.08 ± 0.10	1.30 ± 0.20	2.38 ± 0.22	8.4 ± 6.2	0.39 ± 0.12	17.2 ± 6.2
S010[i]	53	110	4.78 ± 0.67	-2.8 ± 8.7	3.24 ± 0.21	1.68 ± 0.13	1.93 ± 0.25	3.61 ± 0.28	6.5 ± 2.0	0.55 ± 0.11	13.6 ± 2.0
S010A			4.09 ± 0.72	-54 ± 13	2.73 ± 0.21	1.26 ± 0.12	1.63 ± 0.23	2.89 ± 0.26	6.6 ± 2.7	0.32 ± 0.12	13.5 ± 2.7
S010B			5.12 ± 0.67	7.4 ± 5.4	4.19 ± 0.26	2.16 ± 0.17	2.34 ± 0.28	4.50 ± 0.33	10.9 ± 2.4	0.68 ± 0.12	22.5 ± 2.4
S011[i]	120	19	16.3 ± 3.4	-91 ± 35	330 ± 20	15.1 ± 1.2	42.8 ± 4.1	57.9 ± 4.3	426 ± 33	8.4 ± 1.0	860 ± 33
S011A			167 ± 16	-950 ± 120	1,530 ± 87	121.7 ± 8.8	179 ± 17	301 ± 19	1,810 ± 120	41.2 ± 3.2	3,661 ± 120
S011B			7.0 ± 1.1	9 ± 14	7.02 ± 0.45	2.69 ± 0.22	2.45 ± 0.33	5.14 ± 0.40	14.0 ± 4.0	0.96 ± 0.19	29.0 ± 4.0
S012	25	24	5.91 ± 0.84	37.9 ± 6.1	4.80 ± 0.30	0.83 ± 0.08	1.18 ± 0.18	2.01 ± 0.20	27.6 ± 4.1	1.63 ± 0.18	56.8 ± 4.1
S012A			3.44 ± 0.68	19.9 ± 5.4	3.14 ± 0.23	0.78 ± 0.08	0.75 ± 0.16	1.53 ± 0.18	17.0 ± 4.0	1.02 ± 0.16	35.0 ± 4.0
S012B			6.46 ± 0.99	43.0 ± 7.5	7.02 ± 0.47	0.82 ± 0.09	1.01 ± 0.18	1.83 ± 0.20	32.6 ± 5.0	2.18 ± 0.24	67.4 ± 5.0

[a]Refer to Figure A-5. Sample locations without an extension were counted without sieving. The "A" extension denotes the fraction of sample that did not pass through a 1/4" sieve. The "B" extension denotes the fraction of sample that did pass through a 1/4" sieve.

[b]$DCGL_w$s for Ra-226, Th-232, and U-238 are 29.4, 23.9, and 721 pCi/g, respectively. Samples S002, S003, S005, and S011 exceed these $DCGL_w$ values.

[c]Total Th is the sum of Th-228 and Th-232.

[d]Total U is calculated by the formula U-235 + (2 * U-238).

[e]Uncertainties represent the 95% confidence level, based on total propagated uncertainties.

[f]Statistically negative concentrations are due to the force fit calculation in the gamma counting software.

[g]Samples S004 and S005 were at the same location.

[h]Samples S006, S006 and S008 were at the same location.

[i]Samples S010 and S011 were at the same location.

TABLE B-2:
RADIONUCLIDE CONCENTRATIONS VIA GAMMA SPECTROSCOPY
FOR THE SITE VISIT ON 6/1-2/2011

ORISE Sample Location[a]	NaI Response (kcpm)		RADIONUCLIDE CONCENTRATIONS (pCi/g)								
	Pre-Sample	Post-Sample	Pb-210	Th-230	Ra-226[b]	Th-228	Th-232	Total Th[b,c]	U-238	U-235	Total U[d]
Survey Unit 1											
S013[e]	18	23	4.76 ± 0.69[f]	9.8 ± 4.2	4.78 ± 0.31	1.36 ± 0.12	1.56 ± 0.23	2.92 ± 0.26	10.83 ± 0.92	0.74 ± 0.13	22.4 ± 1.8
S014[e]	22	24	3.47 ± 0.63	-127[g] ± 17	3.66 ± 0.26	1.19 ± 0.11	1.43 ± 0.20	2.62 ± 0.23	13.0 ± 1.1	0.67 ± 0.13	26.7 ± 2.2
S015	92	12	2.11 ± 0.38	2.0 ± 5.8	1.77 ± 0.12	0.74 ± 0.06	0.81 ± 0.12	1.55 ± 0.13	8.06 ± 0.66	0.40 ± 0.08	16.5 ± 1.3
S016	22	25	6.92 ± 0.79	2.4 ± 8.1	3.98 ± 0.26	1.10 ± 0.09	1.24 ± 0.17	2.34 ± 0.19	17.4 ± 1.3	1.02 ± 0.12	35.8 ± 2.6
S017	24	27	4.64 ± 0.71	-20 ± 10	3.77 ± 0.26	1.08 ± 0.10	1.17 ± 0.20	2.25 ± 0.22	6.18 ± 0.72	0.31 ± 0.12	12.7 ± 1.4
S018	19	20	0.97 ± 0.41	-36.4 ± 9.2	1.59 ± 0.13	0.67 ± 0.07	0.73 ± 0.13	1.40 ± 0.15	2.37 ± 0.40	0.20 ± 0.09	4.94 ± 0.81
S019	47	63	12.4 ± 1.3	11.7 ± 6.9	13.02 ± 0.76	1.34 ± 0.11	1.34 ± 0.20	2.68 ± 0.23	17.1 ± 1.4	0.85 ± 0.16	35.1 ± 2.8
S020	21	23	2.68 ± 0.47	12.5 ± 4.4	2.31 ± 0.16	1.58 ± 0.13	1.57 ± 0.20	3.15 ± 0.24	6.78 ± 0.63	0.53 ± 0.09	14.1 ± 1.3
Survey Unit 3											
S021	19	20	6.34 ± 0.87	8.8 ± 4.7	5.42 ± 0.36	1.32 ± 0.12	1.50 ± 0.24	2.82 ± 0.27	7.91 ± 0.83	0.59 ± 0.14	16.4 ± 1.7
S022	20	22	4.67 ± 0.85	-96 ± 18	4.11 ± 0.30	1.06 ± 0.11	1.17 ± 0.20	2.23 ± 0.23	9.56 ± 0.99	0.70 ± 0.15	19.8 ± 2.0
S023	22	23	5.68 ± 0.73	6.4 ± 8.3	4.68 ± 0.29	1.15 ± 0.10	1.18 ± 0.17	2.33 ± 0.20	7.63 ± 0.72	0.34 ± 0.10	15.6 ± 1.4
S024	20	22	7.48 ± 0.92	5.3 ± 6.3	6.15 ± 0.39	1.94 ± 0.16	1.74 ± 0.25	3.68 ± 0.30	7.72 ± 0.77	0.62 ± 0.13	16.1 ± 1.5
S025	19	21	6.06 ± 0.87	-27 ± 12	4.19 ± 0.29	1.51 ± 0.14	1.57 ± 0.25	3.08 ± 0.29	10.9 ± 1.0	0.61 ± 0.15	22.4 ± 2.0
Survey Unit 2											
S026[h]	31	40	39.6 ± 3.8	53 ± 12	37.1 ± 2.3	1.98 ± 0.19	3.47 ± 0.43	5.45 ± 0.47	30.5 ± 2.5	3.47 ± 0.40	64.5 ± 5.0
S027[h]	40	45	42.3 ± 3.9	104 ± 13	35.1 ± 2.0	2.58 ± 0.21	3.34 ± 0.41	5.92 ± 0.46	40.6 ± 2.9	6.32 ± 0.49	87.5 ± 5.8
S028	19	19	8.4 ± 1.1	15.4 ± 5.7	6.56 ± 0.42	1.25 ± 0.12	1.07 ± 0.19	2.32 ± 0.22	15.8 ± 1.3	0.96 ± 0.17	32.6 ± 2.6

[a]Refer to Figure A-6.

[b]$DCGL_w$s for Ra-226, Th-232, and U-238 are 29.4, 23.9, and 721 pCi/g, respectively. Samples S002, S003, S005, and S011 exceed these $DCGL_w$ values.

[c]Total Th is the sum of Th-228 and Th-232.

[d]Total U is calculated by the formula U-235 + (2 * U-238).

[e]Samples S013 and S014 were at the same location with S013 at 0 to 15 cm and S014 at 15 to 30 cm.

[f]Uncertainties represent the 95% confidence level, based on total propagated uncertainties.

[g]Statistically negative concentrations are due to the force fit calculation in the gamma counting software.

[h]Samples S026 and S027 were at the same location with S026 at 0 to 15 cm and S027 at 15 to 30 cm.

TABLE B-3:
URANIUM CONCENTRATIONS VIA ALPHA SPECTROSCOPY
FOR SELECT SAMPLES
FOR THE SITE VISIT ON 6/1-2/2011

Sample Location[a]	URANIUM ISOTOPIC CONCENTRATIONS (pCi/g)								
	U-234			U-235			U-238		
S015	12.30	±	0.91[b]	0.50	±	0.07	10.79	±	0.80
S015X[c]	10,090	±	770	489	±	69	9,790	±	740
S020	7.69	±	0.60	0.37	±	0.06	7.66	±	0.59
S025	13.8	±	1.0	0.68	±	0.09	14.3	±	1.1
S028	18.3	±	1.4	0.84	±	0.11	18.5	±	1.4

[a]Refer to Figure A-6.

[b]Uncertainties represent the 95% confidence level, based on total propagated uncertainties (TPUs).

[c]This sample was a small particle taken from sample S015 with elevated activity. The particle was dissolved, diluted to 500mL, then a 10mL aliquot was analyzed.

APPENDIX C
MAJOR INSTRUMENTATION

The display of a specific product is not to be construed as an endorsement of the product or its manufacturer by the author or her employer.

C.1 Scanning and Measurement Instrument/Detector Combinations

C.1.1 Gamma

Ludlum NaI Scintillation Detector Model 44-10, Crystal: 5.1 cm x 5.1 cm
(Ludlum Measurements, Inc., Sweetwater, TX)
coupled to:
Ludlum Ratemeter-scaler Model 2221
(Ludlum Measurements, Inc., Sweetwater, TX)
coupled to:
Trimble GeoXH Receiver and Data Logger (Trimble Navigation Limited, Sunnyvale, CA)

C.2 Laboratory Analytical Instrumentation

High-Purity, Extended Range Intrinsic Detector
CANBERRA/Tennelec Model No: ERVDS30-25195
(Canberra, Meriden, CT)
Used in conjunction with:
Lead Shield Model G-11
(Nuclear Lead, Oak Ridge, TN) and
Multichannel Analyzer
Canberra's Apex Gamma Software
Dell Workstation
(Canberra, Meriden, CT)

High-Purity, Extended Range Intrinsic Detector
Model No. GMX-45200-5
(AMETEK/ORTEC, Oak Ridge, TN)
used in conjunction with:
Lead Shield Model SPG-16-K8
(Nuclear Data)
Multichannel Analyzer
Canberra's Apex Gamma Software
Dell Workstation
(Canberra, Meriden, CT)

LABORATORY ANALYTICAL INSTRUMENTATION (CONTINUED)

High-Purity Germanium Detector
Model GMX-30-P4, 30% Eff.
(AMETEK/ORTEC, Oak Ridge, TN)
Used in conjunction with:
Lead Shield Model G-16
(Gamma Products, Palos Hills, IL) and
Multichannel Analyzer
Canberra's Apex Gamma Software
Dell Workstation
(Canberra, Meriden, CT)

Alpha Spectrometry System
Tennelec Model 256
(Canberra, Meriden, CT)
Used in conjunction with:
Ion Implanted Detectors and
Multichannel Analyzer
Canberra Apex Alpha Software
Dell Workstation
(Canberra, Meriden,CT)

Alpha Spectrometry System
Canberra Model 7401VR
(Canberra, Meriden, CT)
Used in conjunction with:
Ion Implanted Detectors and
Multichannel Analyzer
Canberra Apex Alpha Software
Dell Workstation
(Canberra, Meriden, CT)

APPENDIX D
SURVEY AND ANALYTICAL PROCEDURES

D.1 PROJECT HEALTH AND SAFETY

The proposed survey and sampling procedures were evaluated to ensure that any hazards inherent to the procedures themselves were addressed in current job hazard analyses. Prior to on-site activities, a pre-job integrated safety management checklist was completed and discussed with field personnel. Additionally, upon arrival at the site, contractor representatives provided the ORISE with general safety information within the project area. The planned activities were thoroughly discussed with site personnel prior to implementation to identify hazards present. ORISE also had a site escort at all times due to the various alarms and notifications associated with an active industrial facility. All survey and laboratory activities were conducted in accordance with Oak Ridge Associated Universities (ORAU) health and safety and radiation protection procedures (ORAU 2011b and c).

D.2 CALIBRATION AND QUALITY ASSURANCE

Calibration of all field and laboratory instrumentation was based on standards/sources, traceable to National Institute of Standards and Technology (NIST).

Analytical and field survey activities were conducted in accordance with procedures from the following documents of the Independent Environmental Assessment and Verification (IEAV) Program:

- Survey Procedures Manual (ORISE 2008)
- Laboratory Procedures Manual (ORISE 2011b)
- Quality Program Manual (ORAU 2011a)

The procedures contained in these manuals were developed to meet the requirements of DOE Order 414.1C and the U.S. Nuclear Regulatory Commission (NRC) *Quality Assurance Manual for the Office of Nuclear Material Safety and Safeguards* and contain measures to assess processes during their performance.

Quality control procedures include:

- Daily instrument background and check-source measurements to confirm that equipment operation is within acceptable statistical fluctuations.

- Participation in Mixed Analyte Performance Evaluation Program (MAPEP), NIST Radiochemistry Intercomparison Program (NRIP), and Intercomparison Testing Program (ITP) Laboratory Quality Assurance Programs.

- Training and certification of all individuals performing procedures.

- Periodic internal and external audits.

D.3 SURVEY PROCEDURES

D.3.1 Surface Scans

Scans for elevated gamma radiation were performed by passing the detector slowly over the surface. The distance between the detector and surface was maintained at a minimum. The 2x2 NaI(Tl) scintillation detectors were coupled to global positioning system (GPS) units with data loggers enabling real-time recording in one-second intervals of both geographic position and the gamma count rate. Position and gamma count rate data files were transferred to a computer system, positions differentially corrected, and the results plotted on geo-referenced aerial photographs or plot plans of the facility. Positional accuracy was within 0.5 meters at the 95th percentile.

Specific scan MDCs for the detectors were not determined as the instruments were used solely as a qualitative means to identify elevated gamma radiation levels in excess of background. The identification of elevated radiation levels that could exceed the site criteria were determined based on an increase in the audible signal from the indicating instrument.

D.3.2 Soil Sampling

Approximately 0.5 kilogram of soil was collected at each sample location. Additional samples were collected at several locations as a result of the count rate increasing significantly after collecting the first sample. ORISE noted there was a significant amount of +¼ inch slag material in many of the samples. Samples were placed in a plastic bag, sealed, and labeled in accordance with ORISE survey procedures.

D.4 RADIOLOGICAL ANALYSIS

D.4.1 Gamma Spectroscopy

Samples were dried, mixed, crushed, and/or homogenized as necessary, and a portion sealed in a 0.5-liter Marinelli beaker or other appropriate container. The quantity placed in the beaker was chosen to reproduce the calibrated counting geometry. Net material weights were determined and the samples counted using intrinsic germanium detectors coupled to a pulse height analyzer system. Background and Compton stripping, peak search, peak identification, and concentration calculations were performed using the computer capabilities inherent in the analyzer system. All total absorption peaks (TAP) associated with the ROCs were reviewed for consistency of activity. TAPs used for determining the activities of ROCs and the typical associated MDCs for a one-hour count time were:

TABLE D-1: MDC DERIVED FROM TOTAL ABSORPTION PEAK

Radionuclide	TAP (MeV)	MDC (pCi/g)
U-235	0.063	0.24
U-238	0.144	0.75

Spectra were also reviewed for other identifiable TAPs.

D.4.2 Alpha Spectroscopy

Samples were homogenized and dissolved by a combination of potassium hydrogen fluoride and pyrosulfate fusions. The fusion cakes were dissolved and all alpha emitters are co-precipitated on barium sulfate. The barium sulfate is re-dissolved and the contaminant of concern was separated from the other actinides by extraction chromatography utilizing Eichrom Technologies LLT resins, co-precipitated with cerium fluoride, and analyzed using ion implanted detectors, alpha spectrometers, and multichannel analyzers.

The alpha spectroscopy detector system calculates a MDC for each individual isotope per sample based on the detector background, counting efficiency, chemical yield, and sample quantity. TAPs used for determining the activities of ROCs and the typical associated MDCs for a one-hour count time were:

TABLE D-2: MDC DERIVED FROM TOTAL ABSORPTION PEAK		
Radionuclide	**TAP (MeV)**	**MDC (pCi/g)**
U-234	5.275	0.02
U-235	5.499	0.01
U-238	5.155	0.01

D.4.6 Detection Limits

Detection limits, referred to as minimum detectable concentrations (MDCs), were based on 95% confidence level via NUREG 1507 method. Because of variations in background levels, measurement efficiencies, and contributions from other radionuclides in samples, the detection limits differ from sample to sample and instrument to instrument.