AMERICAN BOARD OF HEALTH PHYSICS

950 Herndon Parkway, Suite 450
Herndon, Virginia 20170

RADIATION PROTECTION REPORT COVER SHEET

Please complete this form and attach it to the report submitted with the Application for Certification.

Applicant's Name: Kaitlin Engel

Author: (Check all applicable, but at least one.)

- ☐ Report authored solely be the applicant.
- ☑ Applicant originated the first draft. (Pages 7-34 and 43-99 of this PDF package)
- ☐ Applicant solely responsible for major sections. (Mark those sections on the Report.)
- ☐ Applicant primarily responsible for the research and development behind the report and shared the writing effort.
- ☐ Describe manner in which this report reflects a "professional effort" by the applicant.

Subject Area of Report:

- ☐ Facility/Process Evaluation
- ☐ Protective Guidance Document
- ☑ Dose Assessment (Military Truck Salvage Yard Report)
- ☐ Retrospective/Prospective Radiation Protection Evaluations (e.g., accident evaluation, emergency planning)
- ☑ Other area in which ABHP tests and certifies expertise (specify): Independent assessment of land area for unconditional release. Collection and assessment of field data (Vallecitos Land Area Report).

Professional Element:

- ☑ Judgement (describe): Used professional judgmental to evaluate field data to assess doses to the public (Military Truck). Used professional judgment to evaluate three data sets to asses if the subject land area was consistent with other non-impacted land areas (Vallecitos).
- ☐ Non-regulatory guidance used (describe):

Signature of all authors (original signature in ink):

Kaitlin Engel (12/16/2019)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 31, 2018

Amerco Real Estate Company
2727 North Central Avenue
Phoenix, AZ 85004
Attn: Larry Hine,
Environmental Operations Manager

SUBJECT: MILITARY TRUCK SALVAGE YARD — RESULTS AND CONCLUSIONS OF THE U.S. NUCLEAR REGULATORY COMMISSION'S INITIAL SITE VISIT AND REQUEST FOR CONFIRMATION OF VOLUNTARY CONTROLS

Dear Mr. Hine:

I am writing to provide you with the results of the U.S. Nuclear Regulatory Commission (NRC) staff's initial site visit to the property at 5700 Boundary Avenue, Anchorage, Alaska, performed on July 18-19, 2017. The results are summarized below and are discussed in further detail in the enclosed report.

As described in the site summary that was attached to our letter dated October 6, 2016,[1] our records indicated that the property at 5700 Boundary Avenue, was the site of a former military truck salvage yard known as E.A. Patson Parts and Equipment, which collected military vehicles and "new old stock" (i.e., N.O.S) vehicle parts. The property is currently owned by the Amerco Real Estate Company and being leased by U-Haul Moving and Storage of North Anchorage (lessee). Based on the site summary, the staff suspected that radium-226 (Ra-226) was present in World War II-era vehicles and parts collected at the site by the former military truck salvage yard. Radium is a naturally occurring radioactive material. One use for radium was in fluorescent paint for luminous aircraft and vehicle instruments. The use of radium for this purpose generally ended several decades ago due to radiation safety considerations.

During the initial site visit in July 2017, the staff conducted radiation surveys over approximately 70 percent of the land areas and inside approximately 50 percent of the buildings located on the property. The staff did not survey under the current driveway or building foundations.

Based on the observations during the initial site visit and results of the radiation surveys performed, the staff confirmed the presence of Ra-226 in some of the military dials and gauges stored inside one of the buildings that had been associated with the military truck salvage yard. Further, based on observations of the dials and gauges containing Ra-226 that were readily visible, as well as the radiation survey results obtained from inside one of the buildings, it was estimated there were fewer than 100 dials and gauges containing Ra-226 stored in the building; therefore, you are considered a General Licensee in accordance with the NRC's rules and regulations.

[1] Agencywide Documents Access and Management System (ADAMS) Accession No. ML16277A282.

The extent of contamination from the military dials and gauges could not be determined as part of the initial site visit, due to portions of the buildings being inaccessible. However, the staff was able to perform an analysis on one of the gauges where higher dose rates were measured and concluded there was no removable contamination from that particular gauge because the gauge was intact. Please note that the survey performed by the staff did not fully characterize your property. More detailed radiation surveys would be necessary to assess any possible contamination of building surfaces, structural surfaces, or objects, and in surface soils.

The staff also observed that access to both of the buildings, where radiation above background was measured, was limited and that the lessee controlled the buildings by lock-and-key. The NRC regulates the dose limit to the public of 100 millirem per year (mrem/yr) from NRC regulated materials, as specified in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 20.1301, *Dose limits for individual members of the public*. Based on the radiation survey measurements, controls established by the lessee of the buildings and using conservative assumptions for time, distance, and shielding, the staff concluded that no member of the public was likely to receive a dose in excess of 100 mrem/yr due to the presence of the dials and gauges.

Up to 100 luminous products (e.g., gauges) may be possessed under a general license in accordance with 10 CFR 31.12(a)(4), *General license for certain items and self-luminous products containing radium-226*. Further, as a General Licensee, certain requirements must be followed under 10 CFR 31.12(c), which include:

1) You must notify the NRC should there be any indication of possible damage to the product so that it appears it could result in a loss of the radioactive materials. A report containing a brief description of the event, and the remedial action taken, must be sent to the Director of the Office of Nuclear Material Safety and Safeguards (NMSS), U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001 within 30 days.

2) You must not abandon products containing Ra-226. The product, and any radioactive material from the product, may only be disposed of in accordance with 10 CFR 20.2008, *Disposal of certain byproduct material*, or by transfer to a person authorized by a specific license to receive the Ra-226 in the product or as otherwise approved by the NRC.

3) You must not export (i.e., transfer to a person or an international organization in a foreign country) products containing Ra-226 except in accordance with 10 CFR Part 110, *Export and Import of Nuclear Equipment and Material*.

4) You must dispose of products containing Ra-226 only at a disposal facility authorized to dispose of radioactive material in accordance with any Federal or State solid or hazardous waste law, including the Solid Waste Disposal Act, as authorized under the Energy Policy Act of 2005, by transfer to a person authorized to receive Ra-226 by a specific license issued under 10 CFR Part 30, *Rules of General Applicability to Domestic Licensing of Byproduct Material*, or equivalent regulations of an Agreement State, or as otherwise approved by the NRC.

5) You must respond to written requests (including this one) from the NRC to provide information relating to the general license within 30 calendar days of the date of the request, or other time specified in the request. If you cannot provide the requested information within the allotted time, you must, within that same time period, request a longer period to supply the information. A written justification for the request must be provided to the Director of NMSS by means of an appropriate method listed in 10 CFR 30.6(a).

In accordance with 10 CFR 31.12(c)(5), the staff requests that you provide a response in writing to the following requests for information, within **120 days** from the date of this letter.

1) Please provide your intention for either continued possession of the military dials and gauges or disposition of the items.

2) Please provide information on the status regarding access to the buildings where military dials and gauges were being stored. Has anything changed since the initial site visit in July 2017? In addition, are there other controls in place to limit access to these buildings?

3) Please provide the inventory number of radium dials and gauges that are stored within each building. Please note that if you have more than 100 gauges, you may need a Specific License. If, while determining the inventory of the site, you discover any broken dials please note that a report to the NRC is required in accordance with 10 CFR 31.12(c)(1).

Please contact the NRC should you have any questions regarding the information requested above, or regarding the general license regulation and requirements.

As previously mentioned, additional work is necessary should you elect to determine the extent of contamination. As part of any voluntary cleanup effort, we suggest that you consider consulting an NRC or Agreement State specifically licensed service provider to ensure that there is limited potential for radiological contamination to be spread. Should you wish to dispose of the gauges and instruments, a licensed service provider should be utilized to conduct any packaging of radioactive waste for transport. Please be aware that any remediation activities pursued at your site may also have to meet any State of Alaska requirements and standards. As previously discussed, any voluntary site cleanup is the financial responsibility of the site owner.

As the NRC inspector discussed with the lessee at the end of the initial site visit, the staff recommends that you maintain limited access and continued control of the buildings that stored the military dials and gauges containing radium. Based on the restriction of access to the buildings as discussed above, the staff concludes that there are no immediate health and safety concerns at this site.

In accordance with 10 CFR 2.390 of the NRC's "Agency Rules of Practice and Procedure," a copy of this letter will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records component of NRC's ADAMS. ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html.

We will be contacting you in the near future to answer any questions you may have regarding this letter. Additionally, you may also contact Mr. Stephen Koenick, Chief, Materials Decommissioning Branch, Division of Decommissioning, Uranium Recovery and Waste Programs, Office of Nuclear Materials Safety and Safeguards, at (301) 415-6631, or Mr. Jeffrey Whited, Project Manager, at (301) 415-4090.

Sincerely,

/RA/

John R. Tappert, Director
Division of Decommissioning, Uranium Recovery and Waste Programs
Office of Nuclear Material Safety and Safeguards

Docket No.: 03038976

Enclosures:
1. Site Status Report for the Military Truck Salvage Yard (5700 Boundary Avenue)
2. Copy of Applicable NRC Regulations

cc w/ enclosures:
U-Haul Company of Alaska
4751 Old Seward Highway
Anchorage, Alaska 99503
Attn: John Norris

REGISTERED LETTER – RETURN RECEIPT REQUESTED

SUBJECT: MILITARY TRUCK SALVAGE YARD —RESULTS AND CONCLUSIONS OF THE U.S. NUCLEAR REGULATORY COMMISSION'S INITIAL SITE VISIT Dated May 31, 2018

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R. Kellar, RIV

ADAMS Accession No.: ML17214A755 ***via e-mail**

OFFICE	DUWP/MDB/PM	DUWP/LA	DUWP/MDB	DUWP/MDB
NAME	JWhited	CHolston	RNelson*	CGrossman*
DATE	08/03/2017	08/03/2017	08/09/2017	08/17/2017
OFFICE	RIV/DNMS/BC	DUWP/MDB/BC	OGC (NLO)	DUWP
NAME	RKellar*	SKoenick*	IIrvin*	JTappert
DATE	04/10/2018	04/16/2018	04/24/2018	05/31/2018

OFFICIAL RECORD COPY

Enclosure 1

OAK RIDGE ASSOCIATED UNIVERSITIES:

SITE STATUS REPORT FOR THE MILITARY TRUCK SALVAGE YARD AT 5700 BOUNDARY AVENUE, ANCHORAGE, ALASKA

May 31, 2018

EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) requested that Oak Ridge Associated Universities (ORAU) perform a radiation survey of the property at 5700 Boundary Avenue in Anchorage, Alaska. The property is currently a U-Haul franchise; however, the property was formerly E.A. Patson Parts and Equipment, a military truck salvage yard, which collected military vehicles and "new old stock" (i.e., N.O.S) vehicle parts. It is suspected that radium may have been present at the site in World War II-era vehicles and parts as luminous radium dials, gauges, and instruments. The objective of this survey was to locate possible discrete sources of radium, if any, that would be associated with the former salvage yard operations.

ORAU performed the radiation surveys of the accessible portions of the land area and buildings on July 18-19, 2017. Seven buildings remained on the property from the previous owner. During the survey, ORAU detected radiation levels through exterior walls of two of the buildings. Specifically ORAU measured radiation levels approximately three times the normal background levels outside of one of the buildings, and approximately two times the normal background levels outside of the second building.

Several gauges and dials containing radium paint were identified on the property, most likely remnants from when the site was a military truck salvage yard. The visual and radiation surveys were limited and did not cover the entire two buildings or the contents of the buildings. This was primarily due to the amount of materials and equipment on the floor in one of the buildings and because a key to the second building (with twice background radiation levels) was unavailable. The number of accessible dials and gauges containing radium and measured radiation levels (i.e., containing radium) was limited--fewer than 100 are estimated based on measured radiation levels.

Based on the radiation exposure rate measurements, the building access controls established by the lessee, and using conservative assumptions for time, distance, and shielding, the NRC concludes that no member of the public is likely to receive a dose in excess of 100 mrem/yr.

ORAU recommends that current building access controls (i.e., locked doors with a controlled key) be maintained until radium-containing items can be cataloged and safely stored or dispositioned, as appropriate, and building surfaces are surveyed for radium contamination.

SITE STATUS REPORT

Property:	Former Military Truck Salvage Yard E.A. Patson Parts and Equipment 5700 Boundary Avenue Anchorage, Alaska 99504
Docket Number:	03038976
Current Property Name:	U-Haul Moving and Storage of North Anchorage
Current Property Owner:	Amerco Real Estate Company
Inspection Dates:	July 18-19, 2017
Inspector(s):	Rachel Browder, CHP/U.S. Nuclear Regulatory Commission (NRC), supported by Kaitlin Engel/Oak Ridge Associated Universities (ORAU)

1.0 INTRODUCTION

The Energy Policy Act of 2005 amended section 11e.(3) of the Atomic Energy Act of 1954 to place discrete sources of radium-226 (Ra-226) under NRC regulatory authority as byproduct material. The NRC is evaluating properties where Oak Ridge National Laboratory's (ORNL's) review of historical information has identified Ra-226 use. Since it is possible that Ra-226 could be present in World War II-era vehicles and parts in the form of luminous radium dials, gauges, and instruments, ORNL included the property at 5700 Boundary Avenue in Anchorage, Alaska in its list of historical sites (ORNL 2015). The site was identified as the former military truck salvage yard known as E.A. Patson Parts and Equipment, which collected military vehicles and "new old stock" (i.e. N.O.S) vehicle parts. The objectives of the initial site visit were to determine if discrete sources of Ra-226 and/or distributed Ra-226 contamination were present, to identify the areas of highest contamination, to determine if there were any current health and safety concerns, and to determine if a scoping survey was needed. Surveys were performed as described within NRC's procedure, Temporary Instruction (TI) 2800/043, "Inspection of Facilities Potentially Contaminated with Discrete Radium-226 Sources" (NRC 2017).

Data collected during the July 18-19, 2017, initial site visit may generally be used to plan future actions that may be needed to reduce the radiation exposure of Ra-226 for current or future site occupants to levels that do not exceed the applicable regulatory requirement. It is important to note that destructive testing is not generally performed, as described within TI 2800/043.

2.0 PROPERTY DESCRIPTION AND INITIAL SITE VISIT CONSIDERATIONS

2.1 Property Description and History

E.A. Patson Parts and Equipment, the military truck salvage yard, was founded in 1954. The company collected old military vehicles and N.O.S. vehicle parts. The salvage yard closed in 2013 (ORNL 2015), and the property is currently owned by the Amerco Real Estate Company occupied by U-Haul Moving and Storage of North Anchorage (U-Haul, lessee).

Figure 1 depicts the 12,200 m^2 (3-acre) property in 2002 when occupied by the former military truck salvage yard. An on-line video from November 2013 provides a virtual tour of the salvage yard property (Alaska Trucker 2013). As shown in Figure 1 and in the video, trees, stock-piled materials (i.e., old vehicles, tires, etc.), and a variety of storage sheds and structures covered much of the property grounds. The storage sheds and structures contained the smaller vehicle parts (e.g., N.O.S.) and various other items.

Figure 2 and pictures in Appendix A illustrate the property in its current (2017) configuration. For the purposes of this report, the buildings have been labeled as "Main Building," U-Haul Storage Building, "Old Building," and Building 1 through Building 5. As shown in Figure 2, the site has been cleared and paved, though seven of the original structures remain. After clearing trees and stockpiled materials, gravel was used to level the property, and a large U-Haul Storage Building was built on the western side of the property. The northern most (Main) building is currently used by U-Haul as the office and showroom. The "Main Building" has new windows and brick trim on its northern and western sides. The other original buildings have roofs and new metal siding facades. Based on entry into one of the former buildings, the older structures are constructed of wood beams and wood supports, with chip board walls. Remaining original buildings are used to store leftover items from the former salvage yard and are locked to limit access. One of the original buildings (labeled as "Old Bldg." in Figure 2) was entered, though total access was limited because the floors were covered in old parts, broken glass, and fallen shelving (pictured in Appendix A). It is believed that the "Old Building" is shown in the video clip from minutes 5:50 to 8:00 (Alaska Trucker 2013).

An extensive internet search of public records did not reveal any information related to Ra-226 contamination or radiation exposure levels associated with the facility. The site summary included in the *Historical Non-Military Radium Sites Research Effort Addendum* report provides additional site details about the type, form, history, potential locations, and other information related to discrete sources of Ra-226 used at the site (ORNL 2015).

2.2 Initial Site Visit Considerations

Prior to commencing survey activities, the general layout of the property was examined for consistency with historical information and to identify impediments to conducting the survey and any potential health and safety considerations. The site had been paved over, and a new large U-Haul Storage Building was recently added along the western edge of the property. Only the hallways of the storage building were available for survey because individual storage units were locked. Multiple U-Haul trucks and trailers were parked on the property. Several standing buildings were present when E.A. Patson Parts and Equipment occupied the site. One original, the "Main Building," was maintained and used by U-Haul as the office and showroom. Buildings 1 through 5 were locked, so an assessment of those buildings was not performed. The ORAU representative and NRC inspector were granted access into the "Old Building" with the lessee and observed the floors covered in parts, broken glass, fallen shelving, etc., which hindered access throughout the "Old Building." As a result of the clutter and inaccessible areas of the building, the radiation surveys of the building and contents were limited (see Appendix A). In addition, the lessee suggested that the structural integrity of the "Old Building" was questionable (walls sometimes shift) and the long-term objective was to have the building razed.

Figure 1. Military Truck Salvage Yard Site Image from 2002 (Google Earth 2017)

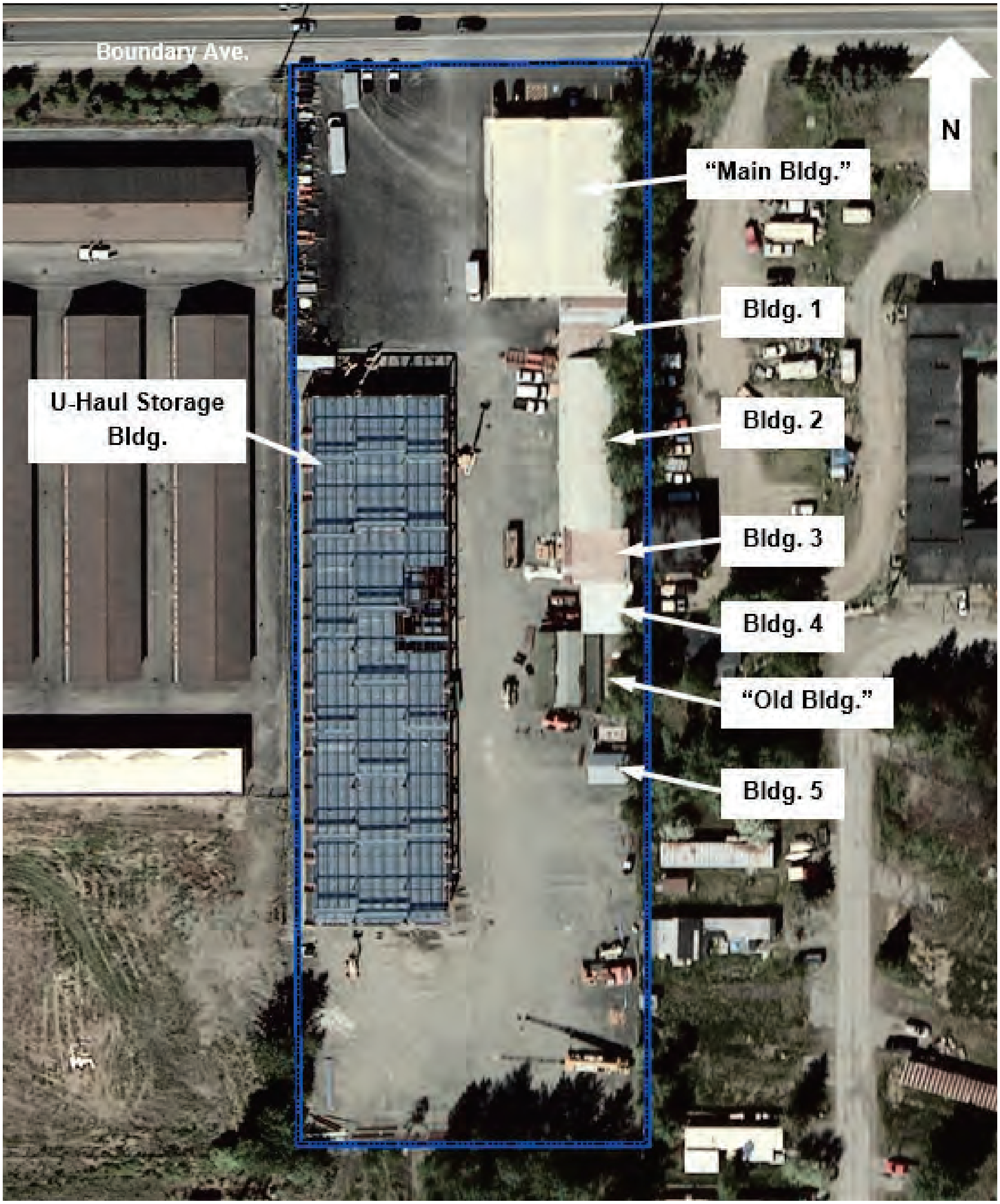

Figure 2. Military Truck Salvage Yard Site (ArcMap 2017)
Blue delineates the property boundary.

3.0 SITE OBSERVATIONS AND FINDINGS

3.1 Summary of Activities

The ORAU representative and NRC inspector conducted an initial site visit at the 5700 Boundary Avenue property on July 18-19, 2017. The site contact (lessee) was unavailable on July 18, 2017, for the pre-inspection meeting but gave permission for surveys to commence. However, the lessee was available on July 19, 2017, to provide access into the "Old Building."

Radiation surveys, performed by the ORAU representative and NRC inspector, consisted of gamma radiation scans using a Ludlum model 44-10 2-inch by 2-inch (2×2) sodium iodide detector connected to a Ludlum model 2221 ratemeter/scaler, alpha-plus-beta radiation direct measurements using a Ludlum model 44-142 plastic scintillator connected to a Ludlum model 2221 ratemeter/scaler, and radiation exposure rate measurements using a Ludlum model 192 sodium iodide-based microRoentgen (μR) ratemeter.[1] Field gamma spectrum measurements were made with a SAM-940 gamma spectrum analyzer. Table 1 presents the specific instruments used during the site visit. In addition, one smear sample was also collected from one of the gauges selected from the "Old Building" to quantify the removable surface activity levels.

Table 1. Military Truck Salvage Yard Survey Instruments

Radiation Type (units)	Detector Type	Detector Model (Number)	Ratemeter (Number)
Alpha-plus-beta (cpm)	Plastic Scintillator	44-142 (689)	2221 (505)
Gross gamma (cpm)	Sodium Iodide	44-10 (1152)	2221 (403)
Gross gamma (μR/h)	Sodium Iodide	192 (1127)	Exposure Meter
Gamma Spectrum Analyzer (SAM-940)	Lanthanum Bromide	940 (40272)[a]	N/A

N/A = not applicable
Number = ORAU equipment barcode
cpm = counts per minute
μR/h = microRoentgen per hour
[a]Device performs automatic calibration upon startup and is source checked before use.

Summary of Daily Activities – July 18, 2017:

The ORAU representative and NRC inspector arrived at the site at 9:45 a.m. Surveys began on the land area, including the paved lot and graveled surfaces surrounding the perimeter of the property. The team used a 2×2 sodium iodide detector connected to global positioning system equipment and model 192 exposure ratemeter to measure gamma radiation levels. Approximately 70 percent of the total land area was surveyed, including all accessible areas not covered by vehicles or trailers. No discrete areas exhibiting elevated radiation were identified over the land area.

[1] NOTE: Roentgen is a unit of exposure (energy absorbed in air), whereas a rem is a unit of dose delivered to a person (resulting from the radiation energy absorbed in that person). While Roentgen and rem are related, these are different units. Because they are similar for gamma ray energies from Ra-226, NRC makes the simplifying assumption in this case that these units are equivalent (1 Roentgen = 1 rem).

The ORAU representative and NRC inspector continued to perform surveys inside the “Main Building” using a 2×2 sodium iodide detector and model 192 exposure ratemeter. Approximately 50 percent of the “Main Building” was surveyed, including items on display from the former military truck salvage yard such as an old gasoline pump, a moped, helmet, warning light, etc. No discrete areas exhibiting radiation levels above background were identified inside the “Main Building.”

The ORAU representative and NRC inspector then surveyed around the accessible, outside perimeter of the remaining original buildings using a 2×2 sodium iodide detector and model 192 exposure ratemeter. The west sides of the original buildings were accessible and were surveyed. The east sides of the buildings were close to the property fence and inaccessible and, therefore, were not surveyed. The south sides of the “Old Building” and Building 5 were accessible and were surveyed. Based on 2×2 sodium iodide detector and exposure ratemeter responses, small increases in radiation levels were measured through the exterior walls of two of the original buildings; Building 4 and the “Old Building” in Figure 2. One location of slightly increased radiation levels was identified near the middle of Building 4 on the west side. The other locations were on the northwest corner, southwest corner, south side, and southeast corner of the “Old Building.” Both contact and 1-meter exposure rate measurements were collected for each location. Direct measurements and smears were not collected on building exteriors, as it was suspected that the radiation was coming from items located inside the buildings and not from contamination on the exterior walls, based on the historical information of the facility and the recently added exterior siding on the buildings. Both buildings were locked and could not be entered.

Finally, the ORAU representative and NRC inspector surveyed inside of the newly built U-Haul Storage Building using a 2×2 sodium iodide detector and model 192 exposure ratemeter to measure gamma radiation levels. No discrete areas exhibiting elevated radiation were identified. Approximately 10 percent of the first floor of the Storage Building was surveyed. The inspection team departed the site at 7:00 p.m.

Summary of Daily Activities – July 19, 2017:

The ORAU representative and NRC inspector arrived at the site at 10:00 a.m. and met with the site contact (lessee) who provided access into the “Old Building.” Limited surveys (approximately five percent of the building) were performed using a 2×2 sodium iodide detector and model 192 exposure ratemeter. Several instrument panels containing dials and gauges were identified in the southwest corner, immediately inside the building. Based on survey measurements, some of the intact dials and gauges exhibited gamma radiation levels indicative of dials and gauges that contained radium. Other dials and gauges were identified on shelves inside the building, but these dials and gauges did not exhibit elevated radiation levels indicative of radium contamination. In addition, it appeared the majority of the dials and gauges observed were intact. There were two gauges observed on the shelf that did not have a faceplate; however, it was not confirmed that the two gauges contained radium. One gauge was selected that exhibited higher radiation levels than other ones on the shelf. This particular gauge was removed from the building for further direct measurements and smears. The survey results for this one gauge are provided in Section 3.2. Once this gauge was removed from the building, elevated radiation levels remained the same within the building, suggesting there were multiple discrete sources of Ra-226 in the “Old Building.” Other interior portions of the building were not accessible and therefore there was no additional survey measurements performed to correlate with the radiation measurements recorded on the outside of the building (southeast corner).

Though it was estimated there were fewer than 100 dials and gauges containing radium paint based on visual observations and exposure rate measurements, the building conditions limited accessibility and precluded a thorough investigation.

The site contact stated that he believed that all gauges from the military truck salvage yard were located in the “Old Building.” However, he was unable to provide access into Building 4; therefore, the cause of the small increase in radiation levels along the exterior wall of the building was not determined. The interior of Buildings 1 through 3, and Building 5 were not surveyed because they were locked. Additionally, NRC determined that there was no need to survey these buildings because there were no elevated levels of radiation measured outside the buildings. A post-inspection meeting was held with the lessee to discuss the results of the survey and recommended controls. The inspection team departed the site at 12:00 p.m.

3.2 Summary of Results

Select pictures are presented in Appendix A. Appendix B presents maps and tabulated results from the initial site visit conducted on July 18-19, 2017. Figures B-1 through B-5 are maps presenting gamma radiation survey data. Tables B-1 through B-4 present total and removable alpha-plus-beta surface activity results in units of disintegrations per minute per 100 cm^2 (dpm/100 cm^2), 2×2 sodium iodide gross responses in counts per minute (cpm), and gross exposure rates in μR/h for contact and at 1 meter, as applicable.

The alpha-plus-beta direct measurements and the field count of the smear for removable surface activity in cpm were converted to total surface activity units of dpm/100 cm^2 using the equation below:

$$dpm/100\ cm^2 = \frac{C - B}{\varepsilon_{tot} \times G}$$

Where:
C = measured count rate (cpm)
B = background count rate (cpm)
G = geometry factor (unitless) = $\frac{Physical\ Detector\ Area\ (cm^2)}{100\ cm^2}$ = 1.0
ε_{tot} = total weighted efficiency (unitless) = 1.6

Due to the number of emissions from Ra-226 and its associated progeny, multiple radiation particles are counted during the surface activity measurement. Therefore, a total weighted efficiency for Ra-226 and its associated progeny was calculated by:

$$\varepsilon_{tot} = \sum_{n} F_n \times \varepsilon_{i,n} \times \varepsilon_{s,n}$$

Where:
F_n = fractional abundance of n^{th} emission
$\varepsilon_{i,n}$ = instrument efficiency for n^{th} emission
$\varepsilon_{s,n}$ = surface efficiency (0.25 for alpha and low-energy beta particles, 0.5 for high-energy beta particles) for n^{th} emission

A summary of the survey results are presented in Table 2, below. Gamma radiation levels varied based on proximity with materials known to contain naturally occurring radioactive material (NORM), i.e., concrete, such as on the west side of the U-Haul Storage Building, and tile in the bathrooms located in the Main Building.

Table 2. Summary of Survey Results for Military Truck Salvage Yard		
Floor/Area	**2×2 Sodium Iodide Gross Response (cpm)**	**Gross Exposure Rate (μR/h at 1 meter)**
Land [a]	2,500 to 22,000	2 to 14
"Main Building"	2,500 to 6,800	3 to 7
U-Haul Storage Building	3,300 to 4,500	3 to 4
Inside "Old Building" [b]	11,000 to 500,000	7 to 45
Background	ORAU Insert cpm	3 to 4

[a]Land area also includes the survey results around the exterior of Building 4 and "Old Building."
[b]Maximum values recorded for the "Old Building" were on items stored inside.

Field count results from the one gauge that was removed from the "Old Building" for a direct measurement, are presented in Table 3, below. The gauge is pictured in Figure A-9. Further analysis of the smear sample by an independent laboratory did not identify any removable contamination.

Table 3. Data Collected from a Gauge at Military Truck Salvage Yard						
Smear ID	**2×2 Sodium Iodide Gross Response (cpm)**	**Gross Exposure Rate (μR/h on contact)**	**Gross Exposure Rate (μR/h at 1 m)**	**Total Surface Activity, Alpha-plus-Beta (dpm/100 cm^2)**	**Removable Surface Activity, Alpha-plus-Beta (dpm/100 cm^2)[a]**	**Size (m^2)**
5307R0003	500,000	1,300	7	46,000	43	0.01

[a]Based on field count with same detector used for direct measurement.

3.3 Summary of Dose Assessment Results

A site-specific dose assessment has not been performed for the former military truck salvage yard site based on the radiation levels measured during the initial site visit and the controls and limited access to the buildings, are described below. The TI 2800/043 presents two action levels (ALs) that correlate to a public dose estimate of 100 mrem/yr; for an industrial building occupant after 2300 hours of exposure per year (1 meter measurement of 40 μR/h above background), or for a residential building occupant after 6800 hours of exposure per year (1 meter measurement of 15 μR/h above background). These two ALs are based on gamma exposure rate and the time an individual is present in either an industrial building or residential building. These ALs may be used to quickly identify radiation levels that could conservatively produce a dose above the public dose criterion under Title 10 of the *Code of Federal Regulations* Section 20.1301, *Dose limits for individual members of the public*.

Background exposure rate levels vary based on the proximity to NORM-containing materials, such as concrete and tile. The background measurements in accessible site buildings on the property generally ranged from 3 to 4 μR/h, with a maximum of 7 μR/h, which was attributable to tile in the bathrooms. Based on an average of 4 μR/h exposure rate for the property and buildings where customers and U-Haul employees occupied, no member of the general public is likely to receive a dose in excess of 100 mrem/yr.

In addition, the gross exposure rate values recorded for the "Old Building" as documented in Table B-4, were based on items stored inside the building and not for the general area exposure rates throughout the building. The buildings were not occupied in support of the lessee's site operations, and access to the buildings was limited through the lessee's control of the locks and keys. Therefore, because there is limited occupancy and the lessee maintained control of the buildings such that the 2300 hour occupancy limit would not be exceeded, it is expected that no member of the general public is likely to receive a dose in excess of 100 mrem/yr.

It should be noted that the entirety of the "Old Building" was not explored and all possible sources of elevated gamma radiation (presumably from discrete sources of Ra-226) could not be identified during the initial site visit. Additionally, the condition of all gauges remaining in the "Old Building" is unknown (i.e., intact or broken). Therefore, it is possible that higher levels of contamination may be present inside the buildings; however, based on the survey measurements around the accessible portions of the perimeter of the building, no member of the general public is likely to receive a dose in excess of 100 mrem/yr.

An evaluation of any contribution to the dose assessment based on the one gauge that was analyzed for alpha-plus-beta total activity was considered. The contact measurement result was 46,000 dpm using the 100 cm^2 44-142 detector, as documented in Table 3. A field count of the removable surface activity based on a smear of the gauge face showed 43 dpm/100 cm^2, which is acceptable error propagation of the instrument, and further laboratory analysis of the smear did not indicate any removable radiation activity. The lack of removable activity suggests that external gamma pathway doses would be the most significant (for an intact gauge) and actual reliance on gamma exposure data was acceptable. Therefore, the gamma exposure rate measurements (μR/h) collected during the site visit are sufficient for demonstrating compliance with the 100 mrem/yr public dose criterion.

Finally, Building 4 could not be entered because the key was not available at the time of the site visit. A small but notable increase in radiation levels of approximately 5-7 μR/hr was observed in several locations along the exterior walls of the building, though the source of the increase in radiation levels could not be explored during the initial site visit. However, based on the survey measurements around the accessible portions of the perimeter of Building 4, no member of the general public is likely to receive a dose in excess of 100 mrem/yr.

4.0 OBSERVATIONS AND RECOMMENDATIONS

Based on the data collected, the former military truck salvage yard property located at 5700 Boundary Avenue does contain discrete sources of Ra-226. However, the extent of the items that contain Ra-226 and their integrity is not fully known.

ORAU made the following observations:

- Gamma radiation levels, with a maximum net exposure rate in excess of 40 μR/h, was identified on gauges in the "Old Building." Access into the "Old Building" was limited due to the structure's integrity and the materials and equipment on the floor of the building, which hindered access; therefore, the number and condition of discrete sources of Ra-226 is unknown. Radiation levels of (46,000 dpm using the 100 cm^2 44-142 detector) for alpha-plus-beta was identified on at least one gauge. Because access to the building was limited, it is possible that other dials and gauges would produce similar results.

- Gamma radiation levels of approximately 5-7 μR/h around outside of Building 4 suggest there are items inside that may also contain Ra-226. The presence, number, and condition of items inside Building 4 is unknown due to lack of access. However, Building 4 was maintained locked and controlled by the lessee, and did not have any windows. Buildings 1, 2, 3 and 5 were not entered; therefore, the presence or absence of discrete sources of Ra-226 could not be absolutely determined. These buildings were not entered because there were no increased levels of radiation measured outside of these buildings.

The NRC inspector communicated to the lessee that doses above 100 mrem/yr are unlikely based on the current limited usage of the buildings and that the controls in place to limit access to the original buildings (i.e., locked doors with a controlled key) should be maintained. In addition, the NRC inspector discouraged anyone from entering the building, since the radiation conditions were not fully quantified and the levels of contamination were unknown.

The current state of the original buildings, located at the former military truck salvage yard site, limits access, thus the total number and condition of discrete sources of Ra-226 is unknown. If the buildings were to collapse (e.g., after a weather event), the Ra-226-containing items could become damaged and/or released to the surrounding environment. It is recommended that contents inside of the “Old Building” and possibly inside Building 4 be removed, inventoried, placed in a stable storage, or dispositioned by a licensed entity. Prior to razing the “Old Building” and Building 4, shelving, flooring, etc., should be surveyed and dispositioned, accordingly. Other original buildings that were inaccessible during the initial site visit should also be surveyed for discrete sources of Ra-226, when available. The owner should control and mitigate risks from exposure to discrete sources of Ra-226 at the military truck salvage yard facility, especially as related to broken items with removable activity, and buildings with areas exhibiting elevated activity.

5.0 REFERENCES

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APPENDIX A

PHOTOS FROM THE MILITARY TRUCK SALVAGE YARD SITE VISIT

A-1. Original Bldgs. Looking Northeast

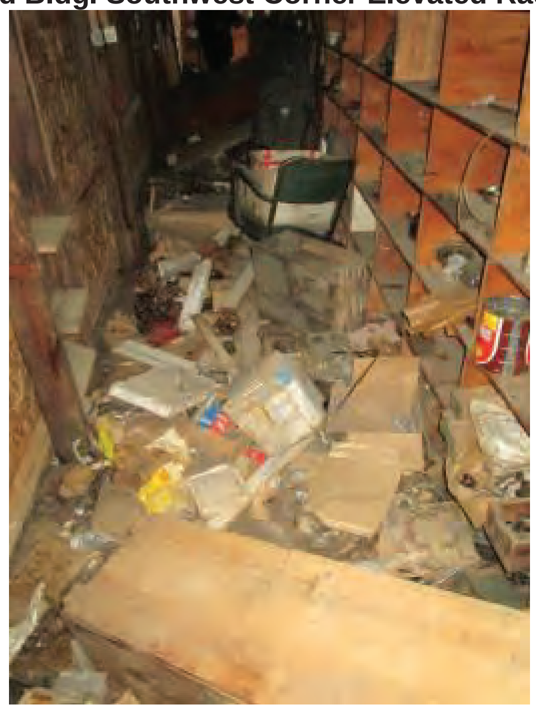
A-2. Old Bldg. West Side Looking East

A-3. Old Bldg. South Side Looking East

A-4. Inside Old Bldg.


A-5. Inside Old Bldg. Southwest Corner Elevated Radiation Levels


A-6. Shelving Inside Old Bldg.

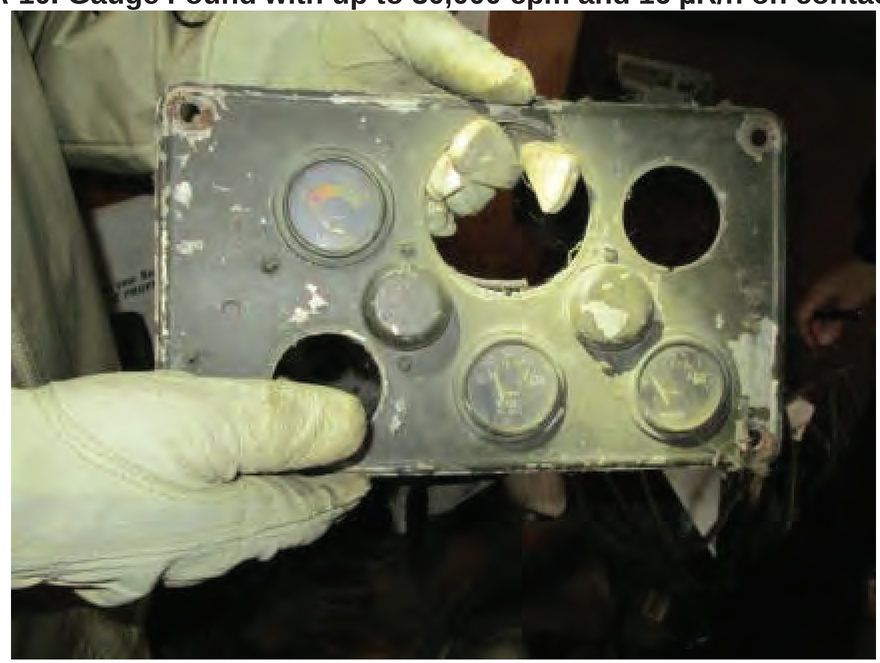

A-7. Floor Inside Old Bldg.


A-8. Shelving Inside Old Bldg.

A-9. Gauge from which Additional Data was Collected (5307R0003)


A-10. Gauge Found with up to 30,000 cpm and 16 μR/h on contact


A-11. Gauges Found with Measurable Exposure Rates up to 100 μR/h on contact


A-12. Gauges Found with Measurable Exposure Rates up to 100 μR/h on contact

APPENDIX B

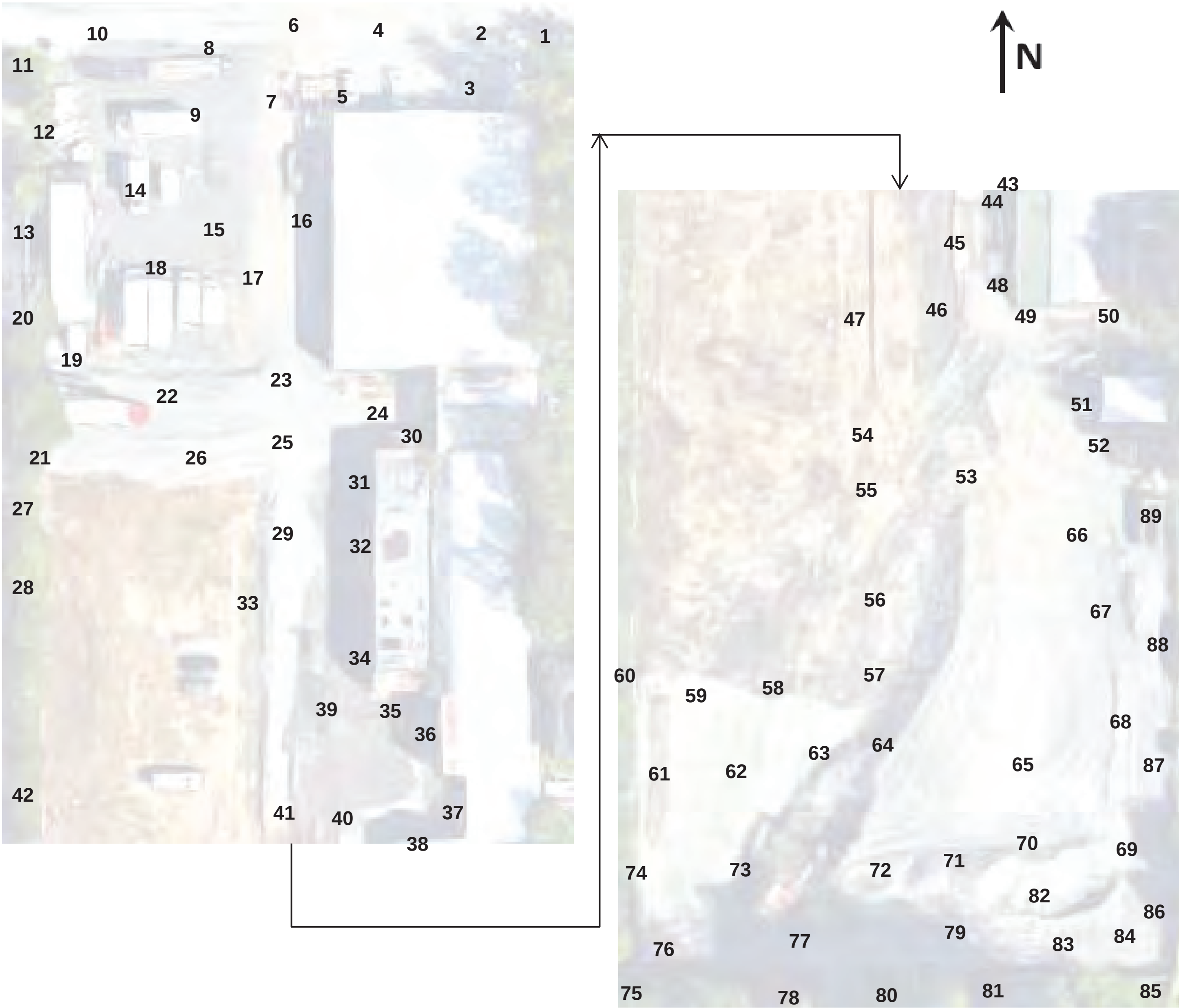
SURVEY RESULTS FROM THE MILITARY TRUCK SALVAGE YARD SITE VISIT

Figure B-1. Gamma Walkover Survey Data for the Land Area

Site: Military Truck Salvage Yard	Area: Land	Date(s): 07/18/2017	Time: 1020-1815
Surveyor(s): KME		Purpose: Site Visit	

Radiation Type	Instrument	Detector	Background
Gamma	192 No. 1127	NA	2-5 μR/h[a]
Gamma	NA	NA	NA
NA	NA	NA	NA

[a]Background varied depending on naturally occurring radioactive material in the area.



= General area measurements provided in attached table.

Figure B-2. Land Area Survey Map

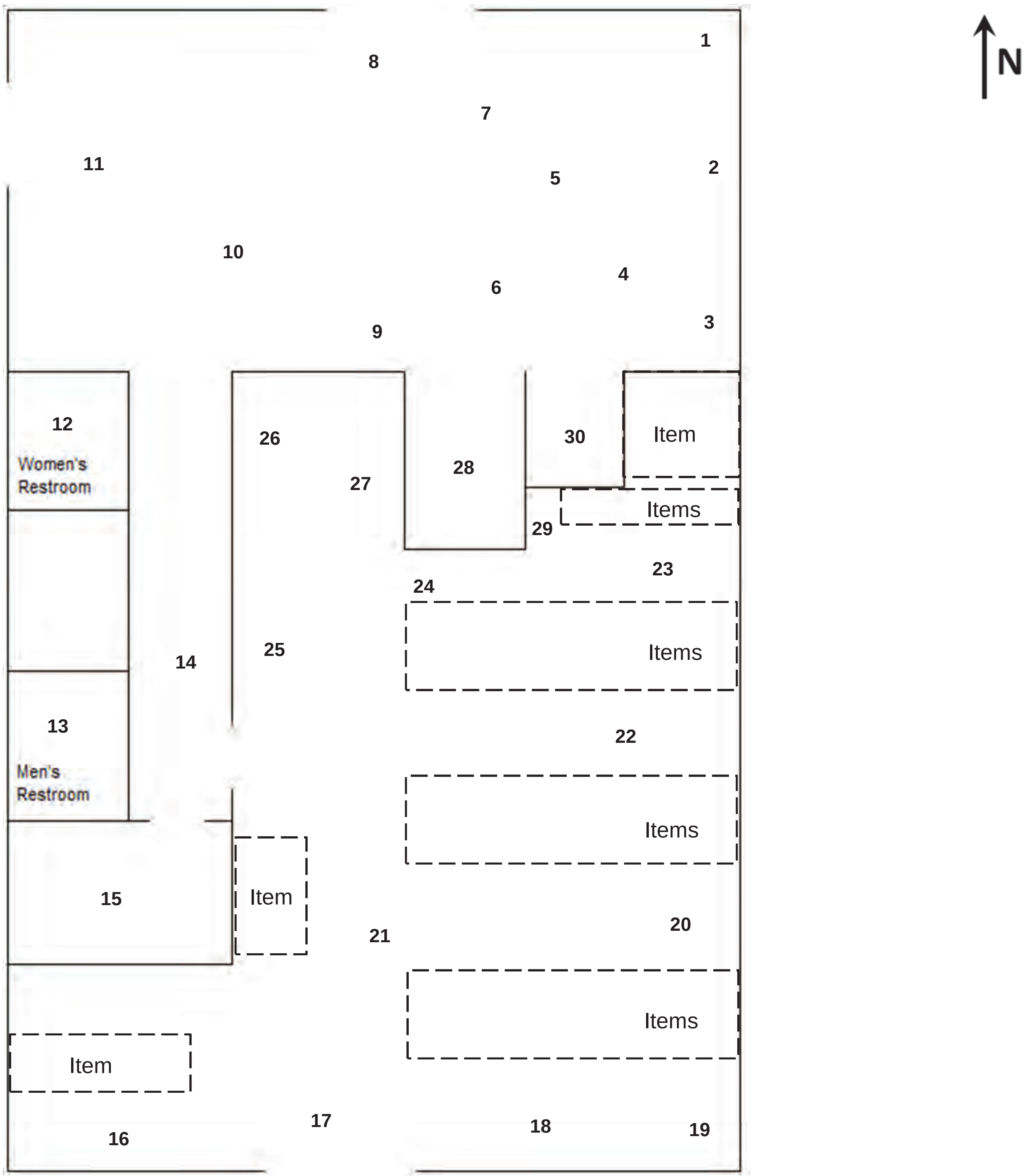
Table B-1. Military Truck Salvage Yard Survey Results - Land Area		
Location No.	Gamma[a] μR/h at 1 meter	Comments
1	3	Gravel
2	3	
3	3	
4	3	
5	4	
6	3	
7	3	
8	3	
9	3	
10	3	
11	4	Gravel
12	3	
13	4	Gravel
14	3	
15	3	
16	3	
17	3	
18	3	
19	3	
20	4	Gravel
21	4	
22	3	
23	3	
24	3	
25	3	
26	3	
27	4	Gravel
28	5	Gravel
29	3	
30	3	
31	3	
32	3	
33	3	
34	2	
35	3	
36	3	
37	5	Elevated readings from inside building. Measurement 1 m from building at 1 m above the ground
38	3	
39	3	
40	3	
41	3	
42	5	Gravel
43	7	On Contact, unable to take 1 m reading due to trailer
44	7	On Contact, unable to take 1 m reading due to trailer
45	3	
46	3	
47	3	
48	14	1 m from building at 1 m above the ground

Table B-1. Military Truck Salvage Yard Survey Results - Land Area		
Location No.	Gamma[a]	Comments
	µR/h at 1 meter	
49	13	1 m from building at 1 m above the ground
50	13	1 m from building at 1 m above the ground
51	3	
52	4	
53	4	
54	4	
55	4	
56	4	
57	4	
58	3	
59	3	
60	5	Gravel
61	4	
62	3	
63	4	
64	4	
65	4	
66	4	
67	4	
68	4	
69	4	
70	3	
71	4	
72	4	
73	4	
74	4	Gravel
75	3	Gravel
76	3	
77	4	
78	3	Gravel
79	3	
80	3	Gravel
81	3	Gravel
82	4	
83	3	
84	4	
85	3	Gravel
86	3	Gravel
87	4	Gravel
88	3	Gravel
89	5	Gravel
a) Ludlum 44-10 NaI with Ludlum 2221 rate meter; Ludlum 192 NaI		

Site: Military Truck Salvage Yard	Area: Inside Main Building	Date(s): 07/18/2017	Time: 1430/1510
Surveyor(s): KME		Purpose: Site Visit	

Radiation Type	Instrument	Detector	Background
Gamma	2221 No. 403	44-10 No. 1152	2.5-6.8 kcpm[a]
Gamma	192 No. 1127	NA	3-7 μR/h[a]

[a]Background varied depending on naturally occurring radioactive material in the area.



= Measurements provided in attached table.

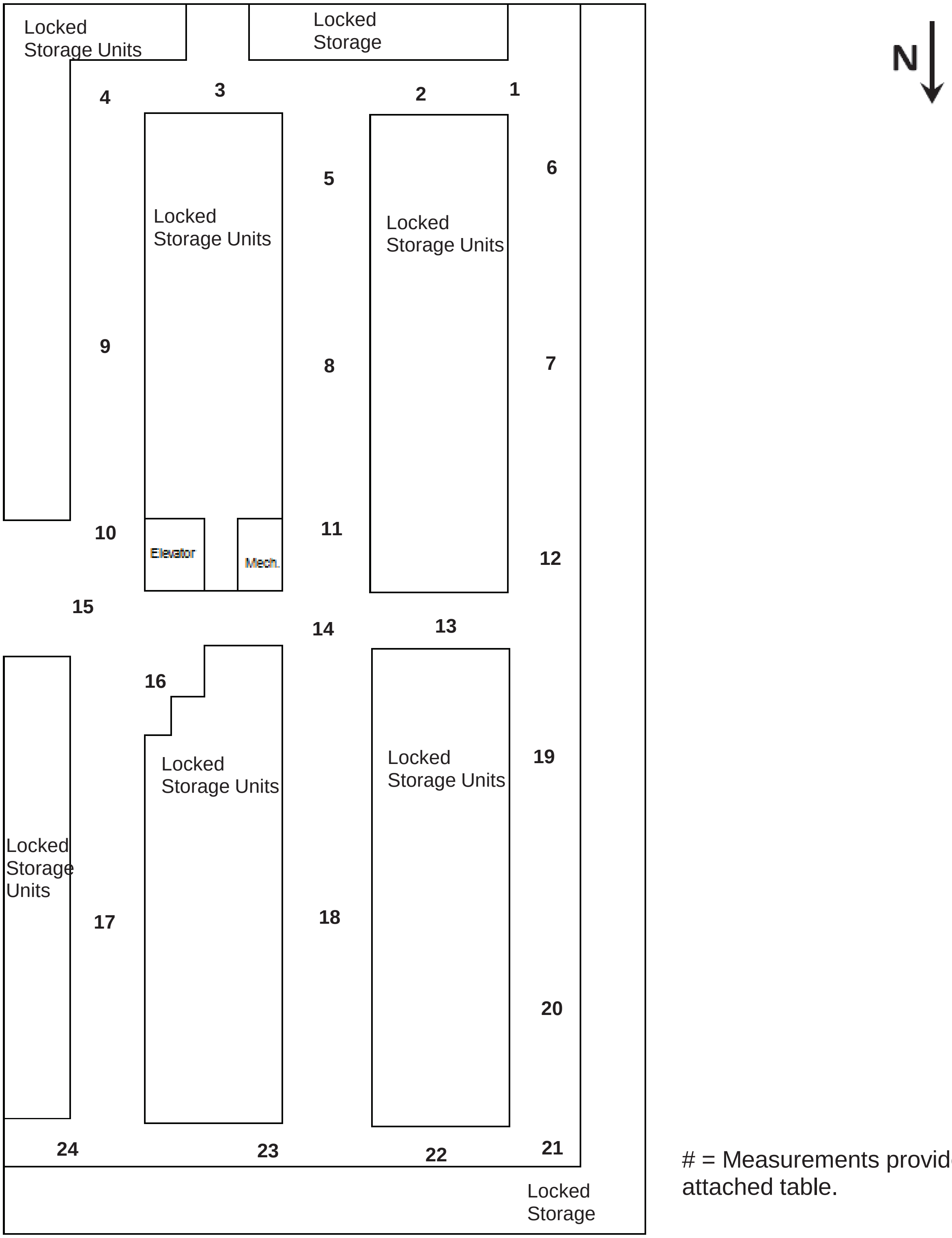
Figure B-3. "Main Bldg." Survey Map

Table B-2. Military Truck Salvage Yard Survey Results - "Main Bldg."			
Location No.	Gamma[c]		Comments
	Contact cpm	1 m µR/hr	
1	3,800	4	
2	3,700	3	
3	4,400	4	
4	4,100	3	
5	4,400	4	
6	4,400	3	
7	4,500	4	
8	4,000	4	
9	4,300	3	
10	4,400	3	
11	4,300	4	
12	6,800	7	New tile flooring
13	6,400	4	New tile flooring
14	4,300	3	
15	3,800	3	
16	4,000	4	
17	2,500	3	
18	3,800	3	
19	4,100	3	
20	4,500	3	
21	3,800	3	
22	4,400	3	
23	4,600	3	
24	4,100	3	
25	4,400	3	
26	3,900	3	
27	4,200	3	
28	4,100	3	
29	4,200	3	
30	4,200	4	
a) Ludlum 44-10 NaI with Ludlum 2221 rate meter; Ludlum 192 NaI			

Site: Military Truck Salvage Yard	Area: U-Haul Storage Bldg.	Date(s): 07/18/2017	Time: 1815/1840
Surveyor(s): KME		Purpose: Site Visit	

Radiation Type	Instrument	Detector	Background
Gamma	2221 No. 403	44-10 No. 1152	3.3-4.5 kcpm[a]
Gamma	192 No. 1127	NA	3-4 μR/h[a]

[a]Background varied depending on naturally occurring radioactive material in the area.



= Measurements provided in attached table.

Figure B-4. U-Haul Storage Bldg. Survey Map

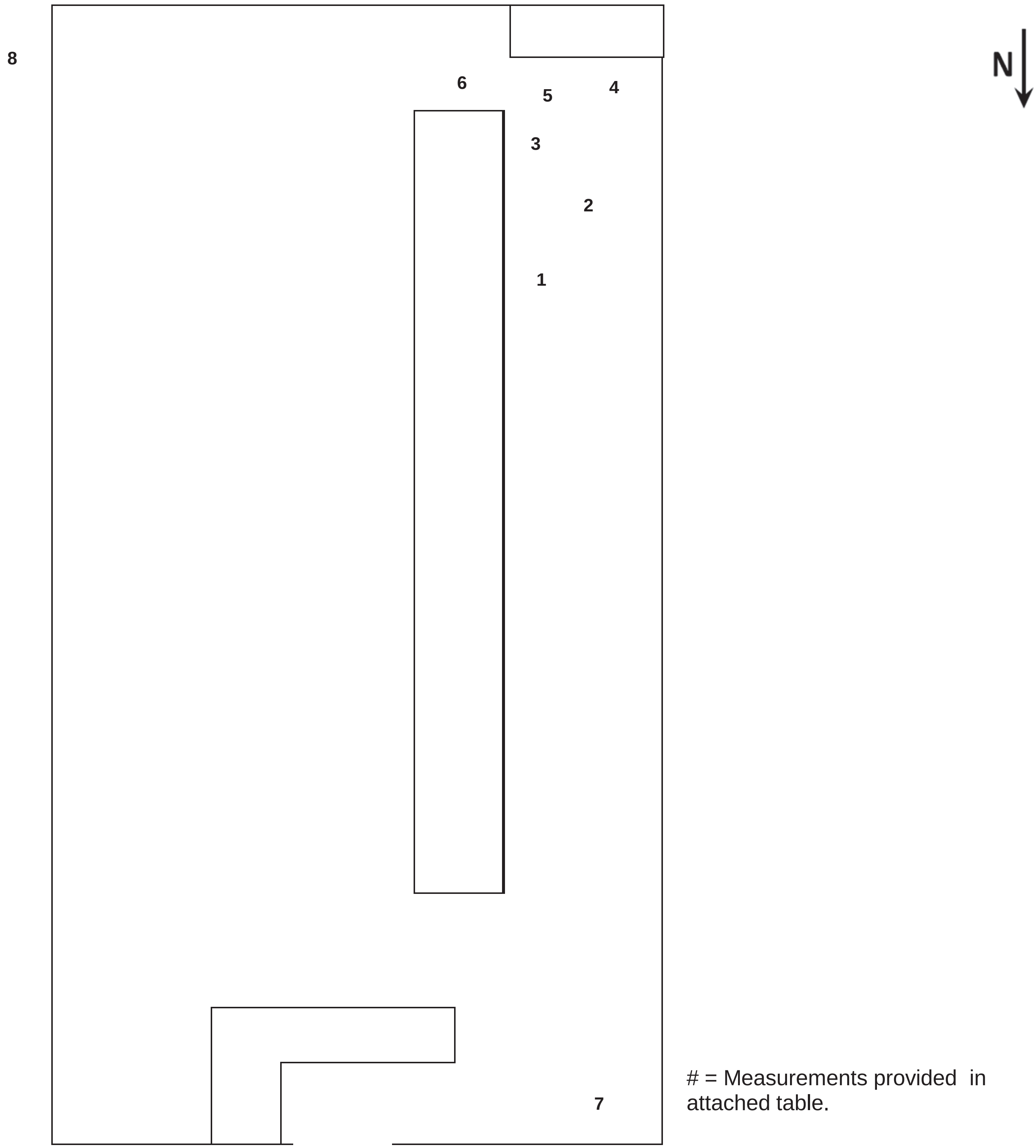
Table B-3. Military Truck Salvage Yard Survey Results - U-Haul Storage Bldg.			
Location No.	Gamma[c]		Comments
	Contact	1 m	
	cpm	µR/hr	
1	4,200	3	
2	3,900	3	
3	4,300	3	
4	4,100	3	
5	4,100	4	
6	3,800	3	
7	4,000	3	
8	3,900	3	
9	4,200	4	
10	3,800	3	
11	4,500	4	
12	4,100	3	
13	4,100	3	
14	3,700	4	
15	4,200	3	
16	4,200	4	
17	4,400	4	
18	3,300	3	
19	4,300	4	
20	3,600	4	
21	3,800	3	
22	4,000	3	
23	4,000	3	
24	3,900	3	

a) Ludlum 44-10 NaI with Ludlum 2221 rate meter; Ludlum 192 NaI

Site: Military Truck Salvage Yard	**Area:** Old Building	**Date(s):** 07/19/2017	**Time:** 1000/1200
Surveyor(s): KME		**Purpose:** Site Visit	

Radiation Type	Instrument	Detector	Background
Gamma	2221 No. 403	44-10 No. 1152	3.4-5 kcpm[a]
Gamma	192 No. 1127	NA	3-4 μR/h[a]
Alpha-plus-Beta	2221 No. 505	44-142 No. 689	267 cpm

[a]As determined from land area surveys.



= Measurements provided in attached table.

Figure B-5. "Old Bldg." Survey Map

Table B-4. Military Truck Salvage Yard Survey Results - "Old Bldg."

Location No.	Removable[a]		Alpha-plus-Beta[b]		Gamma[c]			Comments
	Smear No.	(dpm/100 cm^2)	Gross	Total	Contact		1 m	
		Alpha-plus-Beta	cpm	dpm/100 cm^2	cpm	μR/hr	μR/hr	
1	--	--	--	--	11,000	--	--	
2	--	--	--	--	57,000	--	--	
3	5307R0003	43	69,302	46,000	500,000	1,300	7	Gauge 8340566, model SW503-E
4	--	--	--	--	--	--	34	
5	--	--	--	--	--	--	45	
6	--	--	--	--	--	390	40	Near shelves with gauges
7	--	--	--	--	--	100	--	Small gauges
8	--	--	--	--	200,000	--	--	

a) Smear field counted with Ludlum 44-142 plastic scintillator with Ludlum 2221 rate meter
b) Ludlum 44-142 plastic scintillator with Ludlum 2221 rate meter
c) Ludlum 44-10 NaI with Ludlum 2221 rate meter; Ludlum 192 NaI
— indicates measurement not collected at this location

Enclosure 2

U.S. NUCLEAR REGULATORY COMMISSION

APPLICABLE REGULATIONS FROM

TITLE 10 OF THE *CODE OF FEDERAL REGULATIONS*

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

§ 20.2008 Disposal of certain byproduct material.

(a) Licensed material as defined in paragraphs (3) and (4) of the definition of *Byproduct material* set forth in §20.1003 may be disposed of in accordance with part 61 of this chapter, even though it is not defined as low-level radioactive waste. Therefore, any licensed byproduct material being disposed of at a facility, or transferred for ultimate disposal at a facility licensed under part 61 of this chapter, must meet the requirements of § 20.2006.

(b) A licensee may dispose of byproduct material, as defined in paragraphs (3) and (4) of the definition of *Byproduct material* set forth in § 20.1003, at a disposal facility authorized to dispose of such material in accordance with any Federal or State solid or hazardous waste law, including the Solid Waste Disposal Act, as authorized under the Energy Policy Act of 2005.

[72 FR 55922, Oct. 1, 2007]

Page Last Reviewed/Updated Tuesday, August 29, 2017

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

Home > NRC Library > Document Collections > NRC Regulations (10 CFR) > Part Index > § 31.12 General license for certain items and self-luminous products containing radium-226

§ 31.12 General license for certain items and self-luminous products containing radium-226

(a) A general license is hereby issued to any person to acquire, receive, possess, use, or transfer, in accordance with the provisions of paragraphs (b), (c), and (d) of this section, radium-226 contained in the following products manufactured prior to November 30, 2007.

(1) Antiquities originally intended for use by the general public. For the purposes of this paragraph, antiquities mean products originally intended for use by the general public and distributed in the late 19th and early 20th centuries, such as radium emanator jars, revigators, radium water jars, radon generators, refrigerator cards, radium bath salts, and healing pads.

(2) Intact timepieces containing greater than 0.037 megabecquerel (1 microcurie), nonintact timepieces, and timepiece hands and dials no longer installed in timepieces.

(3) Luminous items installed in air, marine, or land vehicles.

(4) All other luminous products, provided that no more than 100 items are used or stored at the same location at any one time.

(5) Small radium sources containing no more than 0.037 megabecquerel (1 microcurie) of radium-226. For the purposes of this paragraph, "small radium sources" means discrete survey instrument check sources, sources contained in radiation measuring instruments, sources used in educational demonstrations (such as cloud chambers and spinthariscopes), electron tubes, lightning rods, ionization sources, static eliminators, or as designated by the NRC.

(b) Persons who acquire, receive, possess, use, or transfer byproduct material under the general license issued in paragraph (a) of this section are exempt from the provisions of 10 CFR parts 19, 20, and 21, and § 30.50 and 30.51 of this chapter, to the extent that the receipt, possession, use, or transfer of byproduct material is within the terms of the general license; provided, however, that this exemption shall not be deemed to apply to any such person specifically licensed under this chapter.

(c) Any person who acquires, receives, possesses, uses, or transfers byproduct material in accordance with the general license in paragraph (a) of this section:

(1) Shall notify the NRC should there be any indication of possible damage to the product so that it appears it could result in a loss of the radioactive material. A report containing a brief description of the event, and the remedial action taken, must be furnished to the Director of the Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001 within 30 days.

(2) Shall not abandon products containing radium-226. The product, and any radioactive material from the product, may only be disposed of according to § 20.2008 of this chapter or by transfer to a person authorized by a specific license to receive the radium-226 in the product or as otherwise approved by the NRC.

(3) Shall not export products containing radium-226 except in accordance with part 110 of this chapter.

(4) Shall dispose of products containing radium-226 at a disposal facility authorized to dispose of radioactive material in accordance with any Federal or State solid or hazardous waste law, including the Solid Waste Disposal Act, as authorized under the Energy Policy Act of 2005, by transfer to a person authorized to receive radium-226 by a specific license issued under part 30 of this chapter, or equivalent regulations of an Agreement State, or as otherwise approved by the NRC.

(5) Shall respond to written requests from the NRC to provide information relating to the general license within 30 calendar days of the date of the request, or other time specified in the request. If the general licensee cannot provide the requested information within the allotted time, it shall, within that same time period, request a longer period to supply the information by providing the Director of the Office of Nuclear Material Safety and Safeguards, by an appropriate method listed in § 30.6(a) of this chapter, a written justification for the request.

(d) The general license in paragraph (a) of this section does not authorize the manufacture, assembly, disassembly, repair, or import of products containing radium-226, except that timepieces may be disassembled and repaired.

[53 FR 19246, May 27, 1988; 72 FR 55927 Oct. 1, 2007; 79 FR 75739, Dec. 19, 2014]

Page Last Reviewed/Updated Tuesday, August 29, 2017

United States Nuclear Regulatory Commission
Protecting People and the Environment

§ 30.6 Communications.

(a) Unless otherwise specified or covered under the regional licensing program as provided in paragraph (b) of this section, any communication or report concerning the regulations in parts 30 through 37 and 39 of this chapter and any application filed under these regulations may be submitted to the Commission as follows:

(1) By mail addressed: ATTN: Document Control Desk, Director, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001.

(2) By hand delivery to the NRC's offices at 11555 Rockville Pike, Rockville, Maryland.

(3) Where practicable, by electronic submission, for example, via Electronic Information Exchange, or CD–ROM. Electronic submissions must be made in a manner that enables the NRC to receive, read, authenticate, distribute, and archive the submission, and process and retrieve it a single page at a time. Detailed guidance on making electronic submissions can be obtained by visiting the NRC 's Web site at *http://www.nrc.gov/site-help/e-submittals.html;* by e-mail to *MSHD.Resource@nrc.gov;* or by writing the Office of the Chief Information Officer, U.S. Nuclear Regulatory Commission, Washington, DC 20555–0001. The guidance discusses, among other topics, the formats the NRC can accept, the use of electronic signatures, and the treatment of nonpublic information.

(b) The Commission has delegated to the four Regional Administrators licensing authority for selected parts of its decentralized licensing program for nuclear materials as described in paragraph (b)(1) of this section. Any communication, report, or application covered under this licensing program must be submitted to the appropriate Regional Administrator. The Administrators' jurisdictions and mailing addresses are listed in paragraph (b)(2) of this section.

(1) The delegated licensing program includes authority to issue, renew, amend, cancel, modify, suspend, or revoke licenses for nuclear materials issued pursuant to 10 CFR parts 30 through 36, 39, 40, and 70 to all persons for academic, medical, and industrial uses, with the following exceptions:

(i) Activities in the fuel cycle and special nuclear material in quantities sufficient to constitute a critical mass in any room or area. This exception does not apply to license modifications relating to termination of special nuclear material licenses that authorize possession of larger quantities when the case is referred for action from NRC's Headquarters to the Regional Administrators.

(ii) Health and safety design review of sealed sources and devices and approval, for licensing purposes, of sealed sources and devices.

(iii) Processing of source material for extracting of metallic compounds (including Zirconium, Hafnium, Tantalum, Titanium, Niobium, etc.).

(iv) Distribution of products containing radioactive material under §§ 32.11 through 32.30 and 40.52 of this chapter to persons exempt from licensing requirements.

(v) New uses or techniques for use of byproducts, source, or special nuclear material.

(2) *Submissions*. (i) *Region I*. The regional licensing program involves all Federal facilities in the region and non-Federal licensees in the following Region I non-Agreement States and the District of Columbia: Connecticut, Delaware, and Vermont. All mailed or hand-delivered inquiries, communications, and applications for a new license or an amendment, renewal, or termination request of an existing license specified in paragraph (b)(1) of this section must use the following address: U.S. Nuclear Regulatory Commission, Region I, Nuclear Material Section B, Region I, 2100 Renaissance Boulevard, Suite 100, King of Prussia, PA 19406–2713; where email is appropriate it should be addressed to *RidsRgn1MailCenter.Resource@nrc.gov*.

(ii) *Region II.* The regional licensing program involves all Federal facilities in the region and non-Federal licensees in the following Region II non-Agreement States and territories: West Virginia, Puerto Rico, and the Virgin Islands. All mailed or hand-delivered inquiries, communications, and applications for a new license or an amendment, renewal, or termination request of an existing license specified in paragraph (b)(1) of this section must use the following address: U.S. Nuclear Regulatory Commission, Region I, Nuclear Material Section B, Region I, 2100 Renaissance Boulevard, Suite 100, King of Prussia, PA 19406–2713; where email is appropriate it should be addressed to *RidsRgn1MailCenter.Resource@nrc.gov*.

(iii) *Region III.* (A) The regional licensing program for mining and milling involves all Federal facilities in the region, and non-Federal licensees in the Region III non-Agreement States of Indiana, Michigan, Missouri and the Region III Agreement States of Minnesota, Wisconsin, and Iowa. All mailed or hand-delivered inquiries, communications, and applications for a new license or an amendment, renewal, or termination request of an existing license specified in paragraph (b)(1) of this section must use the following address: U.S. Nuclear Regulatory Commission, Region III, Material Licensing Section, 2443 Warrenville Road, Suite 210, Lisle, IL 60532 –4352; where e-mail is appropriate it should be addressed to *RidsRgn3MailCenter.Resource@nrc.gov*.

(B) Otherwise, the regional licensing program involves all Federal facilities in the region and non-Federal licensees in the Region III non-Agreement States of Indiana, Michigan, and Missouri. All mailed or hand-delivered inquiries, communications, and applications for a new license or an amendment, renewal, or termination request of an existing license specified in paragraph (b)(1) of this section must use the following address: U.S. Nuclear Regulatory Commission, Region III, Material Licensing Section, 2443 Warrenville Road, Suite 210, Lisle, IL 60532–4352; where e-mail is appropriate it should be addressed to *RidsRgn3MailCenter.Resource@nrc.gov*.

(iv) *Region IV.* (A) The regional licensing program for mining and milling involves all Federal facilities in the region, and non-Federal licensees in the Region IV non-Agreement States and territory of Alaska, Hawaii, Idaho, Montana, South Dakota, Wyoming and Guam and Region IV Agreement States of Oregon, California, Nevada, New Mexico, Louisiana, Mississippi, Arkansas, Oklahoma, Kansas, Nebraska, and North Dakota. All mailed or hand-delivered inquiries, communications, and applications for a new license or an amendment, renewal, or termination request of an existing license specified in paragraph (b)(1) of this section must use the following address: U.S. Nuclear Regulatory Commission, Region IV, Division of Nuclear Materials Safety, 1600 E. Lamar Blvd., Arlington, TX 76011–4511; where email is appropriate, it should be addressed to *RidsRgn4MailCenter.Resource@nrc.gov*.

(B) Otherwise, the regional licensing program involves all Federal facilities in the region and non-Federal licensees in the following Region IV non-Agreement States and territory: Alaska, Hawaii, Idaho, Montana, South Dakota, Wyoming, and Guam. All mailed or hand-delivered inquiries, communications, and applications for a new license or an amendment, renewal, or termination request of an existing license specified in paragraph (b)(1) of this section must use the following address: U.S. Nuclear Regulatory Commission, Region IV, Division of Nuclear Materials Safety, 1600 E. Lamar Blvd., Arlington, TX 76011–4511; where email is appropriate, it should be addressed to *RidsRgn4MailCenter.Resource@nrc.gov*.

[48 FR 16031, Apr. 14, 1983, as amended at 49 FR 19630, May 9, 1984; 49 FR 47824, Dec. 7, 1984; 50 FR 14693, Apr. 11, 1985; 51 FR 36000, Oct. 8, 1986; 52 FR 8241, Mar. 17, 1987; 52 FR 38392, Oct. 16, 1987; 52 FR 48093, Dec. 18, 1987; 53 FR 3862, Feb. 10, 1988; 53 FR 43420, Oct. 27, 1988; 58 FR 7736, Feb. 9, 1993; 58 FR 64111, Dec. 6, 1993; 59 FR 17465, Apr. 13, 1994; 60 FR 24551, May 9, 1995; 62 FR 22880, Apr. 28, 1997; 68 FR 58803, Oct. 10, 2003; 70 FR 69421, Nov. 16, 2005; 71 FR 15007, Mar. 27, 2006; 72 FR 33386, Jun. 18, 2007; 73 FR 5717, Jan. 31, 2008; 74 FR 62681, Dec. 1, 2009; 75 FR 21980, Apr. 27, 2010; 75 FR 73942, Nov. 30, 2010; 76 FR 72085, Nov. 22, 2011; 77 FR 39905, Jul. 6, 2012; 77 FR 43689, Jul. 25, 2012; 78 FR 17006, Mar. 19, 2013; 78 FR 32338, May 29, 2013; 79 FR 75739, Dec. 19, 2014; 80 FR 74979, Dec. 1, 2015]

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August 13, 2019

Mr. Jack Parrott, Sr. Project Manager
U.S. Nuclear Regulatory Commission
Office of Nuclear Material and Safeguards
Division of Decommissioning, Uranium Recovery, and Waste Programs
Reactor Decommissioning Branch
TWFN Mail Stop: T-8F5
Rockville, MD 20852

SUBJECT: DOE Contract No. DE-SC0014664
INDEPENDENT CONFIRMATORY SURVEY SUMMARY AND RESULTS FOR THE CALIFORNIA STATE ROUTE 84 FRONTAGE ASSOCIATED WITH THE GE HITACHI VALLECITOS NUCLEAR CENTER; SUNOL, CALIFORNIA
DOCKET NO. 05000018 AND 05000183; RFTA NO. 19-003;
DCN 5334-SR-01-1

Dear Mr. Parrott:

The Oak Ridge Institute for Science and Education (ORISE) is pleased to provide the enclosed revised final report, which describes the procedures and results of the California State Route 84 frontage independent confirmatory survey that ORISE performed during the period of February 5–6, 2019 at the GE Hitachi Vallecitos Nuclear Center in Sunol, California. Additional U.S. Nuclear Regulatory Commission's (NRC's) comments were addressed in this revised version.

You may contact me at 865.576.6659 or Kaitlin Engel at 865.574.7008 if you have any questions or require additional information.

Sincerely,

Erika N. Bailey
Survey and Technical Projects Group Manager
ORISE

KME:jc

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INDEPENDENT CONFIRMATORY SURVEY SUMMARY AND RESULTS FOR THE CALIFORNIA STATE ROUTE 84 FRONTAGE ASSOCIATED WITH THE GE HITACHI VALLECITOS NUCLEAR CENTER SUNOL, CALIFORNIA

K. M. Engel
ORISE

FINAL REPORT
Revision 1

Prepared for the
U.S. Nuclear Regulatory Commission

AUGUST 2019

Further dissemination authorized to the NRC only; other requests shall be approved by the originating facility or higher NRC programmatic authority.

INDEPENDENT CONFIRMATORY SURVEY SUMMARY AND RESULTS FOR THE CALIFORNIA STATE ROUTE 84 FRONTAGE ASSOCIATED WITH THE GE HITACHI VALLECITOS NUCLEAR CENTER SUNOL, CALIFORNIA

Prepared by

K. M. Engel
ORISE

AUGUST 2019

FINAL REPORT
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This document was prepared for the U.S. Nuclear Regulatory Commission (NRC) by the Oak Ridge Institute for Science and Education (ORISE) through interagency agreement number 31310018N0014 with the U.S. Department of Energy (DOE). ORISE is managed by Oak Ridge Associated Universities under DOE contract number DE-SC0014664.

INDEPENDENT CONFIRMATORY SURVEY SUMMARY AND RESULTS FOR THE CALIFORNIA STATE ROUTE 84 FRONTAGE ASSOCIATED WITH THE GE HITACHI VALLECITOS NUCLEAR CENTER SUNOL, CALINFORNIA

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FINAL REPORT
Revision 1

AUGUST 2019

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ACRONYMS

AA	alternative action
CFR	Code of Federal Regulation
cm	centimeter
cpm	counts per minute
CU	confirmatory unit
DOE	U.S. Department of Energy
DQO	data quality objective
DS	decision statement
GEH	GE Hitachi
GPS	global positioning system
LLNL	Lawrence Livermore National Laboratory
HTD	hard-to-detect
MDC	minimum detectable concentration
NaI	sodium iodide
NIST	National Institute of Standards and Technology
NORM	naturally occurring radioactive material
NRC	U.S. Nuclear Regulatory Commission
ORAU	Oak Ridge Associated Universities
ORISE	Oak Ridge Institute for Science and Education
pCi/g	picocurie per gram
PSQ	principal study question
Q-Q	quantile-quantile
TAP	total absorption peak
ROC	radionuclide of concern
UCL	upper confidence level
VNC	Vallecitos Nuclear Center

OAK RIDGE INSTITUTE FOR SCIENCE AND EDUCATION

INDEPENDENT CONFIRMATORY SURVEY SUMMARY AND RESULTS FOR THE CALIFORNIA STATE ROUTE 84 FRONTAGE ASSOCIATED WITH THE GE HITACHI VALLECITOS NUCLEAR CENTER SUNOL, CALIFORNIA

EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) requested that the Oak Ridge Institute for Science and Education (ORISE) perform an independent confirmatory survey of the California State Route 84, also known as Vallecitos Road, frontage of the GE Hitachi (GEH) Vallecitos Nuclear Center (VNC) in Sunol, California. In December 2018, GEH submitted a formal request to the NRC for approval to release for unrestricted use approximately 2.8 hectares of VNC property within a construction easement along California State Route 84. This land will be made available to Alameda County Transportation Commission to support road development and widening (GEH 2018).

ORISE performed independent assessment activities during the period of February 5–6, 2019. Confirmatory survey activities included gamma walkover scanning, gamma direct measurements, and soil sampling in the applicable land area.

Elevated direct gamma radiation levels above background were identified in the landscaped area near the road leading into the site. The elevated counts were attributed to naturally occurring radioactive material (NORM) in the lava rocks used in the landscaping. A total of 20 soil samples were collected throughout the land area: 13 random samples, one judgmental sample, and six additional confirmatory samples, as requested by the NRC.

Radionuclide concentrations in soil samples from the California State Route 84 frontage were evaluated for the presence of gamma-emitting mixed activation and fission products, with particular emphasis on Cs-137, which is a VNC radionuclide of concern (ROC). Samples also were analyzed for gross alpha and beta concentrations.

Cs-137 is ubiquitous in the environment from global atmospheric fallout from weapons testing and the Chernobyl and Fukushima nuclear releases. Therefore, Cs-137 concentrations in the California State Route 84 frontage soil samples were compared with soil samples collected from off-site background and other VNC non-impacted populations. The concentrations observed were compared to the Lawrence Livermore National Laboratory’s (LLNL's) environmental surveillance

program results and to C1/C2 non-impacted land area samples collected during a previous confirmatory survey in 2015. In addition to Cs-137, concentrations of naturally occurring radionuclides (K-40, Ra-226, Th-232, and U-238) also were examined between the LLNL, C1/C2, and California State Route 84 frontage land areas.

The concentration of Cs-137 in soil samples collected from the California State Route 84 frontage were within the same range as the 2017 LLNL off-site background environmental monitoring results for Cs-137. Furthermore, the Cs-137 concentrations in the California State Route 84 frontage were statistically demonstrated to be lower than the Cs-137 concentrations found in the C1/C2 area. The lower concentrations were likely due to routine tilling of the road frontage for a fire break. The statistical assessment objectively failed to reject the null hypothesis, thereby concluding that the California State Route 84 frontage land area Cs-137 concentrations are less than or equal to the non-impacted C1/C2 land area concentrations. Comparison of naturally occurring radionuclide concentrations showed that the California State Route 84 frontage soils shared similar natural radiological conditions as the land area outside of LLNL and the C1/C2 land area. Therefore, all results were consistent with the non-impacted determination.

INDEPENDENT CONFIRMATORY SURVEY SUMMARY AND RESULTS FOR THE CALIFORNIA STATE ROUTE 84 FRONTAGE ASSOCIATED WITH THE GE HITACHI VALLECITOS NUCLEAR CENTER SUNOL, CALIFORNIA

1. INTRODUCTION

The GE Hitachi (GEH) Vallecitos Nuclear Center (VNC) in Sunol, California has been engaged in reactor research, development, testing operations, and post-irradiation examination of reactor fuel since the 1950s (GEH 2018). Much of the reactor-related activities have ceased. One test reactor remains operational while three are in a SAFeSTORage condition. VNC is currently licensed under 10 Code of Federal Regulation (CFR) 50 and 70 as well as a State of California radioactive materials license. In December 2018, GEH submitted a formal request to the U.S. Nuclear Regulatory Commission (NRC) for approval to unconditionally release for unrestricted use a construction easement along California State Route 84, also known as Vallecitos Road. This area is referred to as the California Route 84 frontage in this report. This land will be made available to Alameda County Transportation Commission to support road development and widening (GEH 2018).

The licensee categorized the 2.8 hectare (7 acre) area as non-impacted and, as such, plant-derived radionuclides in concentrations exceeding background should not be present. In support of the release request, GEH conducted an environmental assessment, which included a small-scale sampling campaign and a review of site operating history. The NRC requested that the Oak Ridge Institute for Science and Education (ORISE) perform confirmatory survey activities within the 2.8 hectare area that GEH is requesting for unconditional release. ORISE performed the confirmatory survey on February 5–6, 2019. A previous non-impacted site release request and associated ORISE confirmatory surveys involved Areas “C1” and “C2” (C1/C2) at the VNC site (ORISE 2015). The results from the survey of the C1/C2 area were used for comparison to the current confirmatory survey results. Results from Lawrence Livermore National Laboratory’s (LLNL's) annual environmental surveillance program were used for comparison to the current confirmatory survey results as well.

2. SITE DESCRIPTION

The VNC consists of approximately 650 hectare (1,600 acres) and is located at 6705 Vallecitos Road, Sunol, California. The site is approximately 56 kilometers (35 miles) east-southeast of San Francisco, California, in the Pleasanton quadrangle of Alameda County. The site is situated in a primarily agricultural setting with a small residential population existing west of the site. The nearest sizable town is Pleasanton, California, located 6.4 kilometers (4.0 miles) north-northwest of the site. The site's boundaries have not changed since the property was purchased in 1956 and are delineated with fencing and "No Trespassing" signs (GEH 2018).

Only approximately 55 hectares of the 650 hectare VNC site were ever used for conducting licensed activities whereas the balance of the site is mostly undeveloped grasslands with hills ranging from 120 to 370 meters (400 to 1,200 feet) above sea level. Developed industrialized areas of the site exist at elevations between 120 and 180 meters (400 to 590 feet) and slope to the southwest. The property is primarily drained by ditches flowing into Vallecitos Creek.

Figure 2.1 presents the VNC land area delineations. The California State Route 84 frontage is located along the southern-most boundary of Areas "A" and "B" as shown in Figure 2.1. The 2.8 hectare VNC property area that is the focus of this survey effort is within an irregular construction easement along California State Route 84 frontage as shown on Figure 2.2.

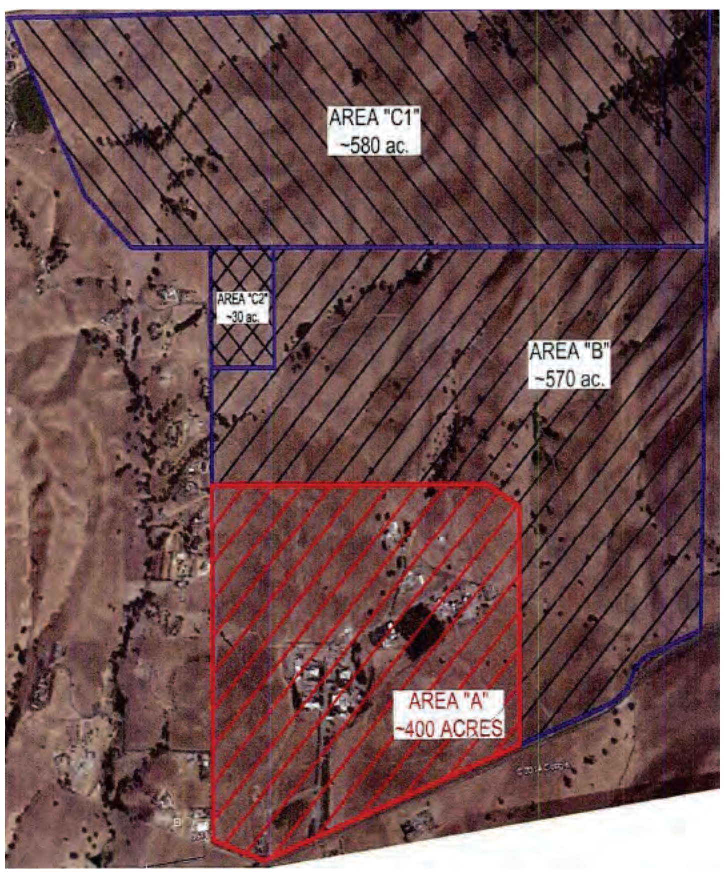

Figure 2.1. VNC Land Area Map

Figure 2.2. California State Route 84 Frontage (Blue Outline)

3. DATA QUALITY OBJECTIVES

The data quality objectives (DQOs) described herein are consistent with the *Guidance on Systematic Planning Using the Data Quality Objectives Process* (EPA 2006) and provide a formalized method for planning radiation surveys, improving survey efficiency and effectiveness, and ensuring that the type, quality, and quantity of data collected are adequate for the intended decision applications. The seven steps in the DQO process are as follows:

1. State the problem
2. Identify the decision
3. Identify inputs to the decision
4. Define the study boundaries
5. Develop a decision rule
6. Specify limits on decision errors
7. Optimize the design for obtaining data

3.1 STATE THE PROBLEM

The first step in the DQO process defines the problem that necessitates the study, identifies the planning team, and examines the project budget and schedule. The licensee requested approval from the NRC to remove the 2.8 hectare, non-impacted survey area from its 10 CFR Part 50 license. The NRC requested that ORISE perform independent contractor document and field reviews and to conduct confirmatory surveys to generate radiological data to assist the NRC in evaluating the licensee's request for partial site release. The NRC uses these data to assess the site radionuclides of concern (ROCs) and to determine the analytical suite for the samples collected from the non-impacted survey area. Therefore, the problem statement was formulated as follows:

> Confirmatory surveys must be performed to generate independent radiological data to assist the NRC with their assessment of the non-impacted classification of the 2.8 hectare land area included in the licensee's request for partial site release and assess the ratios between identified ROCs in the soil.

3.2 IDENTIFY THE DECISION

The second step in the DQO process identifies the principal study questions (PSQs) and alternative actions (AAs), develops decision statements (DSs), and organizes multiple decisions, as appropriate. This was done by specifying AAs that could result from a "Yes" response to the PSQs and combining the PSQs and AAs into DSs. PSQs, AAs, and combined DSs are organized based on the survey unit type (i.e., the associated final status survey methodology) and are presented in Table 3.1

Table 3.1. VNC Confirmatory Survey Decision Process	
Principal Study Questions	**Alternative Actions**
PSQ1: Are radionuclide concentrations in the non-impacted land area consistent with available regional background data and ROC statistical parameters of the confirmatory survey for the C1/C2 non-impacted area?	**Yes:** Compile confirmatory data and report results to the NRC for their decision making. Provide independent interpretation that confirmatory field surveys did not identify anomalous areas of residual radioactivity and quantitative laboratory data are consistent with prior background and non-impacted area values, and/or that statistical sample population examination/assessment conditions were met. **No:** Compile confirmatory data and report results to the NRC for their decision making. Provide independent interpretation of confirmatory survey results identifying any anomalous field or laboratory data and/or when statistical sample population examination/assessment conditions were not satisfied for the NRC's determination of the adequacy of the GEH survey.
PSQ2: Are gamma-emitting ROCs present within collected samples, and/or non-gamma emitting hard-to-detect (HTD)ROCs, if applicable?	**Yes:** Provide analytical results to the NRC that include identified radionuclides and, if applicable, ratios of HTDs to gamma-emitting ROCs. **No:** Provide analytical minimum detectable concentraions (MDCs) and the less-than-MDC results to the NRC.
Decision Statement	
Confirmatory survey results did/did not identify anomalous results or other conditions that refute the non-impacted classification of the subject land area. Independent confirmatory survey results did/did not identify gamma-emitting ROCs in samples and include/do not include additional HTD ROCs for confirmatory samples collected from the non-impacted land area (California State Route 84 Frontage).	

3.3 Identify Inputs to the Decision

The third step in the DQO process identifies both the information needed and the sources of this information, determines the basis for action levels, and identifies sampling and analytical methods to meet data requirements. For this effort, information inputs included the following:

- GEH background assessment and soil sample analytical results collected along California State Route 84
- LLNL background datasets
- Area C1/C2 confirmatory survey analytical results
- ORISE gamma walkover surveys
- ORISE volumetric sample analysis results for soil
- Applicable instrumentation and survey and sampling procedures, method procedures, and data management procedures (ORAU 2016a)
- The *Oak Ridge Associated Universities (ORAU) Environmental Services and Radiation Training Quality Program Manual* (ORAU 2018)
- Applicable laboratory equipment and procedures (ORAU 2017)

3.4 DEFINE THE STUDY BOUNDARIES

The fourth step in the DQO process defines target populations and spatial boundaries, determines the timeframe for collecting data and making decisions, addresses practical constraints, and determines the smallest subpopulations, area, volume, and time for which separate decisions must be made. The study boundary is based on the land area identified in the licensee's request for approval of the partial site release and the supplemental information the site provided in response to an NRC request for additional information (GEH 2018 and 2019). This area, the 2.8 hectare California State Route 84 frontage, constituted the confirmatory survey decision boundary as a single confirmatory unit (CU). Temporal boundaries to complete this survey were limited to two 10-hour days on-site on February 5–6, 2019. The majority of the California State Route 84 frontage land area was accessible; inaccessible areas were due to muddy conditions.

3.5 DEVELOP A DECISION RULE

The fifth step in the DQO process specifies appropriate parameters (e.g., mean, median), confirms action levels were above detection limits, and develops an "if…then…" decision rule statement. For this survey effort, the parameter of interest was the Cs-137 concentration in the California State Route 84 frontage. If non-impacted by site operations, Cs-137 concentrations for a representative sample population from the California State Route 84 frontage should be comparable to the concentrations attributable to atmospheric fallout that have been observed for local background or other non-impacted areas. As such, the California State Route 84 frontage Cs-137 concentrations were directly compared to the LLNL environmental monitoring background concentrations and the previously investigated C1/C2 concentrations. In the event that results were "too close to objectively call," hypothesis testing was planned with the previously collected data for the non-impacted C1/C2. Hypothesis testing adopts a scientific approach where the survey data are used to select between the baseline condition (the null hypothesis, H_0) and an alternative condition.

The null and alternative hypotheses were stated as:

> H_0: California State Route 84 frontage Cs-137 concentration population mean (μ_{CU}) is less than or equal to the C1/C2 mean concentration. Mathematically, the null hypothesis is stated as $\mu_{CU} \leq$ C1/C2.
>
> H_A: California State Route 84 frontage Cs-137 concentration population mean is greater than the C1/C2 mean concentration. Mathematically, the alternative hypothesis is stated as $\mu_{CU} >$ C1/C2.

Identical H_0 and H_A statements also could be made to evaluate the California State Route 84 frontage mean concentration population parameter with the LLNL off-site background environmental monitoring data. However, ORISE did not specifically plan to include this additional two-sample statistical test because the pedigree of the LLNL off-site background data representing independent, random samples from the population—a necessary condition of the statistical test—was not available. Rather these data served the purpose of direct comparison of the relative data dispersion (ranges).

The complete decision rule was stated as follows:

> Compare data with the general distribution, including the maximum observed LLNL background data. Additionally, if the null hypothesis is not rejected and there are no outliers identified indicative of elevated Cs-137 concentrations for individual sample results, then conclude the California State Route 84 frontage Cs-137 concentrations are consistent with previous non-impacted VNC lands that have been released from the site license. Otherwise, perform further evaluation(s). For any noted statistical hypothesis rejections or

concentration anomalies, provide technical comments/recommendations to the NRC for their evaluation and decision making.

3.6 Specify Limits on Decision Errors

The sixth step in the DQO process specifies the decision maker's limits on decision errors, which are then used to establish performance goals for the survey. There were two types of decision errors to consider: Type I (typically designated as alpha or α) and Type II (typically designated as beta or β). A Type I error occurs when the null hypothesis is rejected when it should not be, also known as a false positive, and reflects the confidence level in the decision. A Type II error is incorrectly failing to reject the null hypothesis when the alternative hypothesis is true. It also is known as a false negative. The ability to reject the null hypothesis when it is false is known as the power of the test (power is defined as 1-β).

Two orders of control were implemented to minimize decision errors regarding the DSs introduced in Table 3.1. The first order of control was to select decision error rates that were conservative yet still allowed for the project to be completed within the study boundaries. The Type I error rate was set to 0.05, that is, there is a 5% chance of concluding the CU is greater than the C1/C2 concentrations when it actually is not. The Type II error rate and subsequent power achieved was dependent on the number of samples collected and the concentration variability in the sample set. The number of samples required was based on estimating the CU mean at the 95% confidence level within 0.025 picocuries per gram (pCi/g) above/below the true mean (i.e., a two-sided confidence interval). Based on the assumption that the California State Route 84 frontage was non-impacted, the radiological survey data collected during this survey should be similar to data previously collected in the non-impacted C1/C2 where the mean, upper confidence level (UCL), standard deviation, and maximum Cs-137 levels were 0.134, 0.156, 0.041, and 0.201 pCi/g, respectively (ORISE 2015). The difference between the C1/C2 mean and UCL provided the 0.025 confidence level width planning input discussed above together with the previously determined C1/C2 variability for sample size determination. Both inputs were adjusted for rounding and sample size optimization. For this investigation, 13 random sample locations were planned.

The second order of control was to optimize the confirmatory field measurement and laboratory analytical MDCs. Field scanning MDCs were minimized by following survey procedures for scan speeds and liberal pausing in response to gamma radiation count rates distinguishable from background.

3.7 Optimize the Design for Obtaining Data

The seventh step in the DQO process reviews the DQO outputs, develops data collection design alternatives, formulates mathematical expressions for each design, selects the sample size to satisfy DQOs, decides on the most resource-effective design of agreed alternatives, and documents requisite details. Survey design and laboratory analyses were optimized by implementing the procedures presented in Sections 5 and 6.

4. APPLICABLE SITE GUIDELINES

The primary ROCs identified for the VNC were beta-gamma emitters—fission and activation products—resulting from reactor operation. Previous review of the facility operating history, historical events, and the results of radiological surveys have been completed. As a result of the review, GEH defined the subject land area as non-impacted, meaning that any of the plant-derived ROCs should not be present in excess of background. The only plant-derived, gamma-emitting, fission product that GEH detected, Cs-137, is also detectable in background. If a land area has not been impacted by site activities, Cs-137 should only be present at concentrations attributable to global fallout (GEH 2018).

5. PROCEDURES

The confirmatory survey activities, conducted during the period of February 5–6, 2019, were in accordance with the project-specific confirmatory survey plan, the *ORAU Radiological and Environmental Survey Procedure Manual,* and the *ORAU Environmental Services and Radiation Training Quality Program Manual* (ORISE 2019, ORAU 2016a, 2018). Appendices B and C provide additional information regarding survey instrumentation and related processes discussed within this section.

5.1 Reference System

ORISE referenced confirmatory measurement/sampling locations to global positioning system (GPS) coordinates, specifically NAD 1983 (CORS96) State Plane California. Other prominent site features also were referenced. Measurement and sampling locations were documented on detailed survey maps.

5.2 SURFACE SCANS

Surface scans of the CU land areas were performed with Ludlum Model 44-10 5 centimeter (cm) by 5 cm sodium iodide (NaI) scintillation detectors coupled to Ludlum Model 2221 ratemeter-scalers with audible indicators. Detectors also were coupled to GPS data logging systems that enabled real-time gamma count rate and spatial data capture. Medium-density surface scans were performed within the survey area as time and access permitted. Total scan coverage was dependent on accessibility of the CU. Areas of mud and water limited accessibility to some areas of the CU. Overall scan coverage was 50% to 75%. It was noted during the survey that the California State Route 84 frontage had been routinely tilled to create a fire break.

5.3 GAMMA RADIATION MEASUREMENTS AND SOIL SAMPLING

In total, 20 soil samples were collected from the CU: 13 random locations and seven judgmental locations of which one was based on gamma scan results and six corresponded with licensee sample locations. Site sampling locations were initially approximated using figures and later by using GPS coordinates after conversion to NAD 1983 (CORS96) State Plane California (hence, two samples at site sample location #2).

Samples were collected at a depth of 0 to 15 cm from the surface of the native soil using hand trowels. Sampling equipment was decontaminated in the field after each sample to minimize the potential for cross-contamination. Gamma measurements were performed prior to sample collection. Additional gamma measurements were made at the 15-cm depth after sample collection.

6. SAMPLE ANALYSIS AND DATA INTERPRETATION

Samples and data collected on site were transferred to the ORISE facility for analysis and interpretation. Sample custody was transferred to the Radiological and Environmental Analytical Laboratory in Oak Ridge, Tennessee. Sample analyses were performed in accordance with the *ORAU Radiological and Environmental Analytical Laboratory Procedures Manual* (ORAU 2017). Soil samples were crushed and homogenized and analyzed by gamma spectrometry for gamma-emitting fission and activation products and for gross alpha and beta concentrations using a low-background proportional counter. The gamma spectra also were reviewed for other identifiable photopeaks. With NRC concurrence, no further analysis for HTD radionuclides was performed based on the radionuclide-specific results of the gamma spectroscopy and gross alpha and beta activity from low background proportional counting. Analytical results are reported as gross concentrations in units of pCi/g. Gamma radiation scan and static measurements are presented as gross counts per minute (cpm).

Scan data sets and radionuclide concentrations were graphed in quantile-quantile (Q-Q) plots, strip charts, and/or box plots for assessment. The Q-Q plot is a graphical tool for assessing the statistical distribution of a data set. For the scan data, the Y-axis represents gross gamma radiation levels in units of cpm. For the soil samples, the Y-axis represents the radionuclide concentration in units of pCi/g. The X-axis represents the data quantiles about the median value. Values less than the median are represented in the negative quantiles; values greater than the median are represented in the positive quantiles. A normal distribution that is not skewed by outliers will appear as a straight line, with the slope of the line subject to the degree of variability among the data population. More than one distribution, such as background plus contamination or other outliers, will appear as a step function. Section 7 provides specific analytical and scan data results and discussions.

7. FINDINGS AND RESULTS

The results of the confirmatory survey are discussed in the following subsections.

Appendices A and B provide the survey data for the CU investigated. Appendices C and D provide additional details regarding field and laboratory instrumentation as well as additional information on calibration, quality assurance, survey and analytical procedures, and detection sensitivities.

7.1 SURFACE SCANS

Table 7.1 provides a summary of the confirmatory gamma radiation scanning survey data.

Table 7.1. Summary of Scanning Results			
Area	**NaI Scan Range (cpm)**		
California State Route 84 frontage	3,600	to	11,000

Figures A.1 through A.9 in Appendix A provide the gamma walkover survey maps for the CU. Elevated direct gamma radiation levels above background were identified in the landscaped area near the road leading into the site (Figure A.2). Figure 7.1 provides the Q-Q plot for the CU's data set and shows a step function indicating more than one data population. The elevated counts in the plot are attributed to naturally occurring radioactive material (NORM) in the lava rocks used in the landscaping. The NRC staff flagged one area along the fence line for judgmental sampling based on scan results.

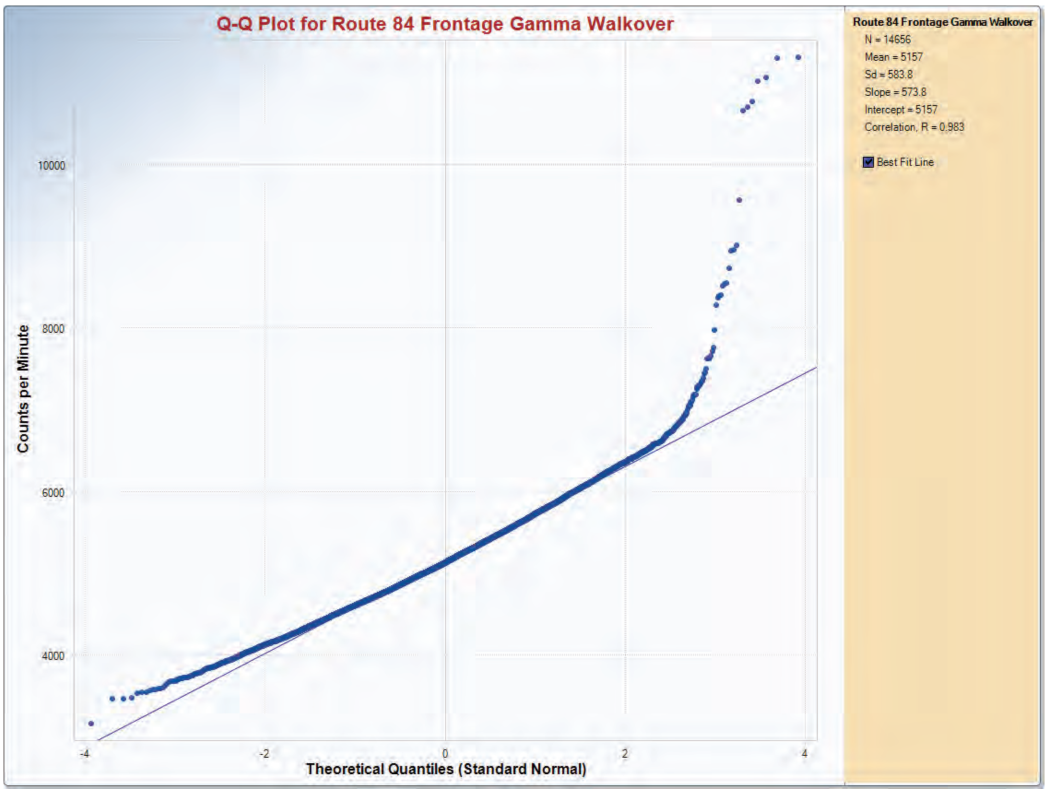

Figure 7.1. Q-Q Plot for Gamma Walkover Survey Data

7.2 GAMMA RADIATION MEASUREMENTS AND RADIONUCLIDE CONCENTRATIONS IN SOIL

Figures A.10 through A.14 in Appendix A provide all soil sampling locations where gamma measurements and samples were collected. Table 7.2 provides a summary of the NaI direct measurements collected pre- and post-sampling.

Table 7.2. Summary of Soil Sampling Direct Measurements

Measurement Type	No. of Samples	NaI Measurement (cpm)	
		Pre-Sample	Post-Sample
Random	13	4,700 to 6,200	4,800 to 7,500
Judgmental	7	4,400 to 5,300	4,800 to 5,700

The gamma direct measurements corresponded to the levels observed during surface scans, with no elevated count rates noted at sampling locations. Additionally, the post-sample measurements did not identify any subsurface anomalies.

Table 7.3 provides a summary of the Cs-137, gross alpha and gross beta concentrations. Review of the gamma spectra did not identify any additional fission/activation products or gamma-emitting transuranics. The only other radionuclides identified were NORM. Summary data for the primary NORM radionuclides also are provided in Table 7.3. Appendix B provides the individual sample data.

Table 7.3. Summary of Radionuclide Concentrations (pCi/g)

Measurement Type		Cs-137	Gross Alpha	Gross Beta	K-40	Ra-226 (by Pb-214)	U-238 (by Th-234)	Th-232 (by Ac-228)
Random Samples	Min	0.006	3.2	6.8	4.33	0.322	0.39	0.435
	Max	0.103	8.5	13.9	10.35	0.520	1.18	0.699
Judgmental	Min	0.010	3.2	9.8	6.57	0.361	0.32	0.411
	Max	0.071	10.2	13.6	10.19	0.510	0.80	0.687

Figure 7.2 provides strip charts comparing the Cs-137, potassium-40 (K-40), thorium-232 (Th-232), and uranium-238 (U-238) concentrations for the California State Route 84 frontage, the last 9 years of LLNL data, and the C1/C2. Ra-226 was not provided in the LLNL dataset; therefore, results were not depicted in Figure 7.2. The LLNL data used in the comparison represent the 2009 to 2017 environmental surveillance program results for soil samples collected from off-site areas that are known to be non-impacted by LLNL. (Gallegos 2010, Jones 2011-2015, Rosene 2016-2018).

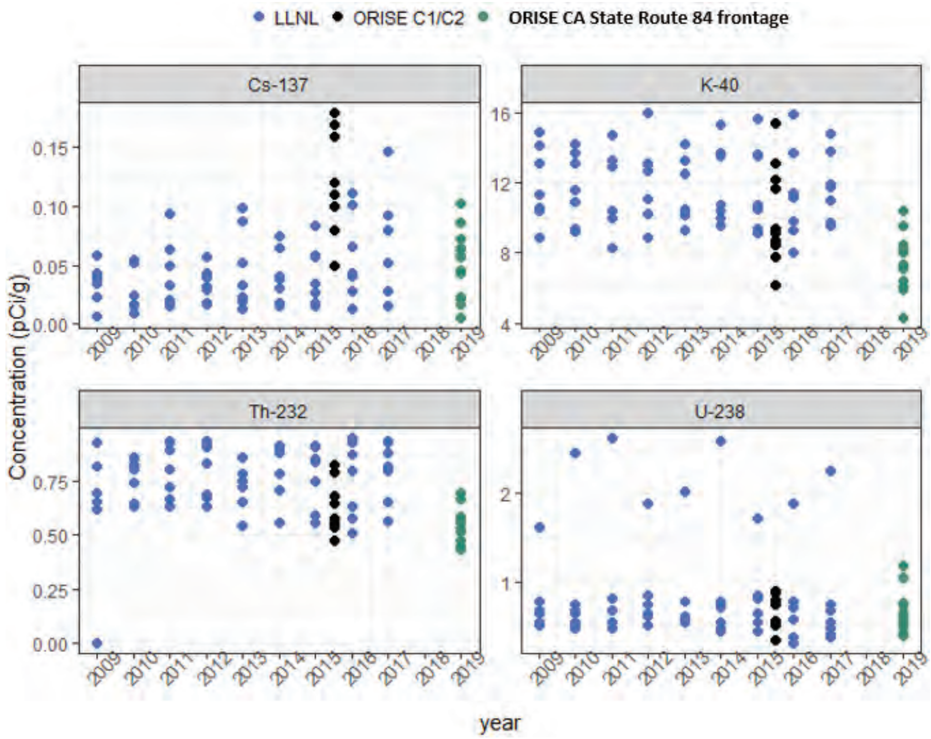

Figure 7.2. Comparison of Cs-137, K-40, Th-232, and U-238 Concentrations

Cs-137 is present in soil as the result of atmospheric fallout deposition and is typically present within the first few centimeters of soil. Therefore, comparison of the three populations is necessary to evaluate whether the Cs-137 identified in the California State Route 84 frontage samples is due to site operations. The California State Route 84 frontage results were compared with the LLNL population. Figure 7.2 also shows that the 2019 California State Route 84 frontage Cs-137 data are within range of the LLNL data from the previous years. It is noted that that the C1/C2 Cs-137 concentration upper bound is slightly higher than both the LLNL and California State Route 84 frontage Cs-137 data ranges. The reason for this can be attributed to differences between disturbed and undisturbed soils. Routine tilling that was performed along California State Route 84 frontage will result in blending surface soil with deeper strata (i.e., below 15 cm), thereby reducing the Cs-137 concentration within the 15-cm depth increment that the samples represent. Similarly, a number of LLNL off-site sample locations appear to be in developed areas where the soil is likely to have been disturbed, based on aerial photographs. In contrast, undisturbed surface soils are expected to exhibit higher concentrations of surface deposited Cs-137, such as in the C1/C2.

The second population comparison was between the California State Route 84 frontage and the C1/C2. Evaluation of Figure 7.2 illustrates that the California State Route 84 frontage random sample Cs-137 concentration sample distribution shifted downward relative to the C1/C2 distribution. Overall, the California State Route 84 frontage population mean concentration is 2.5 times less than the C1/C2 population, with means of 0.05 pCi/g and 0.13 pCi/g, respectively. Although unnecessary based on the above evidence, a hypothesis test was performed on the data. The Student (pooled) t-test was used to compare the difference between the means of the two sample populations. This test is appropriate for small sets of data collected independently from one another, assumes equal variances between the data sets, and when both populations represent normal distributions. The null and alternative hypotheses, H_0 and H_A, respectively, are stated in Section 3.5. The test was performed using ProUCL 5.1.002 and is presented in Appendix A. Because the test statistic (t-test value) is less than the critical value (-5.774 and 1.717, respectively) and the p-value (1) is greater than 0.05, there is not enough evidence to reject H_0 (null hypothesis); therefore, conclude that the CU1 mean concentration is less than or equal to the C1/C2 area mean.

Additional analysis of the California State Route 84 frontage data included comparisons of Cs-137 concentrations in random and judgmental samples for evidence of outliers. The Figure 7.3 box plot shows that judgmental samples were within the random sample population parameters.

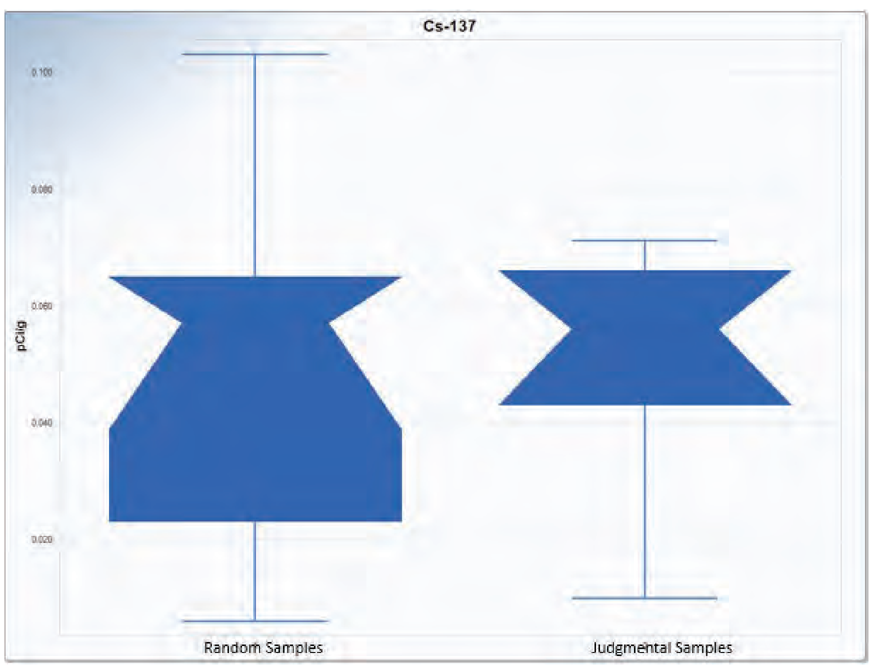

Figure 7.3. Comparison of Random and Judgmental Sample Populations

In addition to Cs-137, the NORM concentrations were compared between the California State Route 84 frontage, the LLNL, and the C1/C2 sample populations. Figure 7.2 also provides the strip charts for the three populations for K-40, Th-232, and U-238. With the exception of K-40 and Th-232, the comparisons show the California State Route 84 frontage, LLNL, and C1/C2 have similar naturally occurring radiological conditions. For K-40 and Th-232, the California State Route 84 frontage concentrations are slightly lower than the C1/C2 concentrations, which is commonly observed for different soil types and/or due to spatial variability in natural background concentrations. Changes in NORM concentrations from tilling are expected to be minimal, as NORM is distributed throughout the soil column rather than only as a result of surface deposition.

The LLNL dataset did not provide gross alpha and beta results for the samples collected outside of LLNL boundaries. Therefore, only the gross alpha and beta results from GEH's investigation were compared to the ORISE California State Route 84 frontage results (GEH 2018).

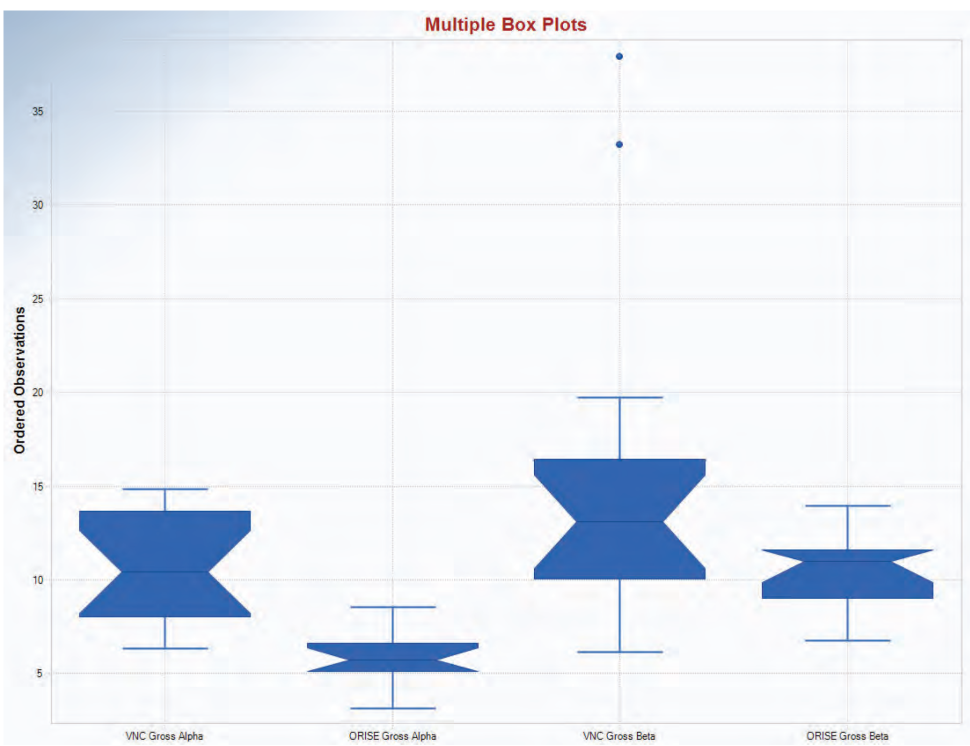

Figure 7.4. Comparison of Gross Alpha and Beta Results for California State Route 84 Frontage

Figure 7.4 provides box plots of the gross alpha and beta results for both the licensee's and ORISE's samples. Appendix B provides the datasets used for comparison. The ORISE gross alpha and beta results were generally lower than the licensee's results. The observed differences are unremarkable. Moreover, the gross activity data only serve as a qualitative screening tool that the site used to select a sample for more rigorous analyses. The licensee's systematic higher results would result in a greater probability that the gross activity would exceed the investigation level that the site uses to require gamma spectroscopy. The differences are possibly a result of systematic bias between the analytical processes of the two laboratories.

8. SUMMARY

At the NRC's request, ORISE conducted confirmatory survey activities of the California State Route 84 frontage at the GEH Vallecitos Nuclear Center in Sunol, California during the period of February 5–6, 2019. The survey activities included gamma scans, gamma direct measurements, and soil sampling.

Elevated direct gamma radiation levels above background were identified in the landscaped area near the road leading into the site. The elevated counts are attributed to NORM in the lava rocks used in the landscaping. Overall gamma scans ranged from 3,600 cpm up to 11,000 cpm. Twenty soil samples were collected throughout the CU: 13 random and seven judgmental sampling locations.

The radionuclide concentrations in the soil samples from the California State Route 84 frontage were compared to the radionuclide concentrations in soil samples collected as part of LLNL's environmental surveillance programs and from the C1/C2 area collected during a previous confirmatory survey. In addition to Cs-137, other background radionuclide concentrations were examined: K-40, Ra-226, Th-232, and U-238. The California State Route 84 frontage Cs-137 concentrations were within the range of the LLNL off-site environmental monitoring Cs-137 concentrations. The Cs-137 concentrations in the California State Route 84 frontage were lower than the Cs-137 concentrations found in the C1/C2. Thus, the null hypothesis was not rejected, concluding that the California State Route 84 frontage sample population was less than or equal to the C1/C2 sample population. Comparison of the NORM concentrations showed that the California State Route 84 frontage had similar radiological conditions as the LLNL off-site locations and the C1/C2. Therefore, all results were consistent with the non-impacted determination.

9. REFERENCES

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APPENDIX A: FIGURES

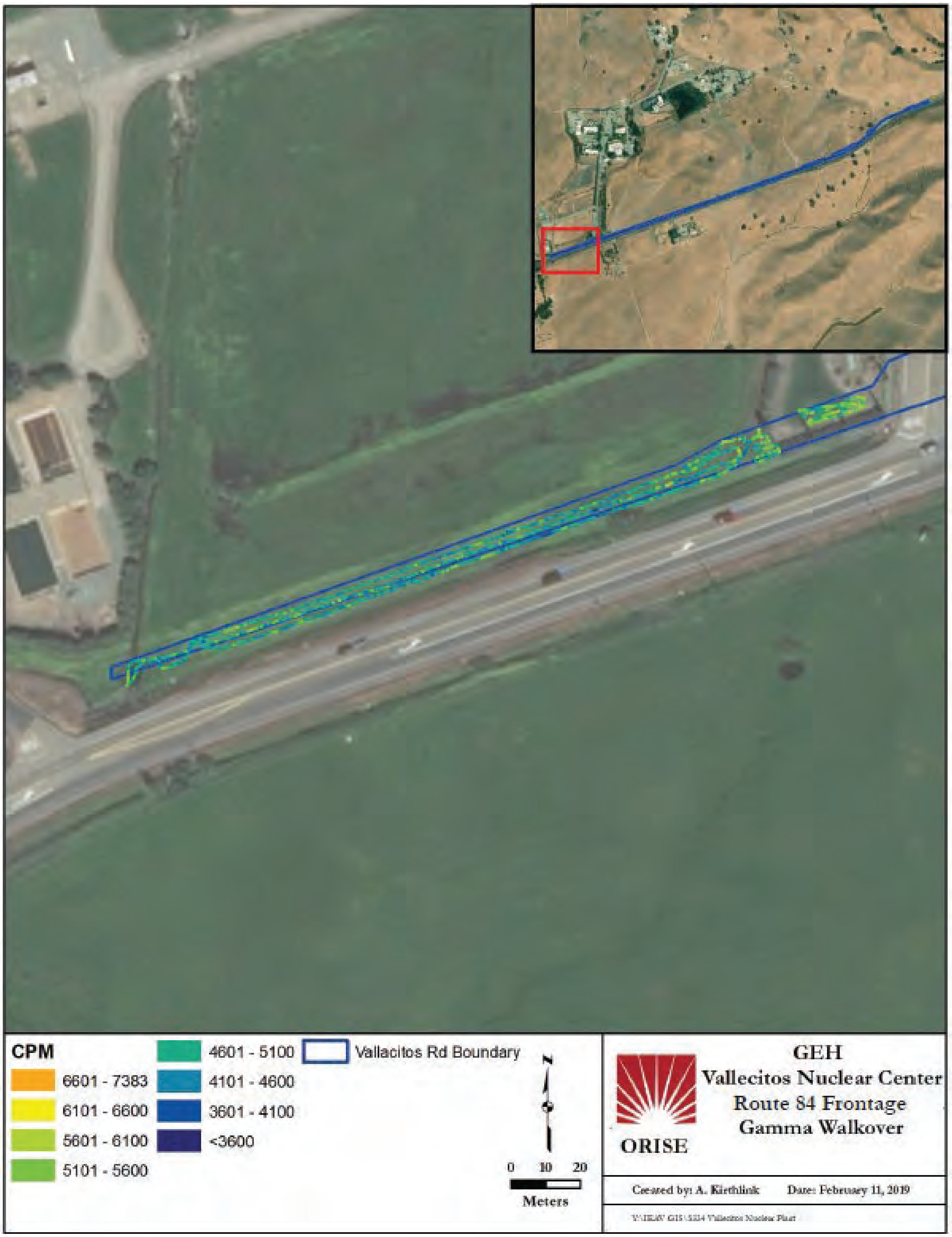

Figure A.1. Gamma Walkover Data (1 of 9)

Figure A.2. Gamma Walkover Data (2 of 9) with Elevated Radiation Levels in the Landscaping

Figure A.3. Gamma Walkover Data (3 of 9)

Figure A.4. Gamma Walkover Data (4 of 9)

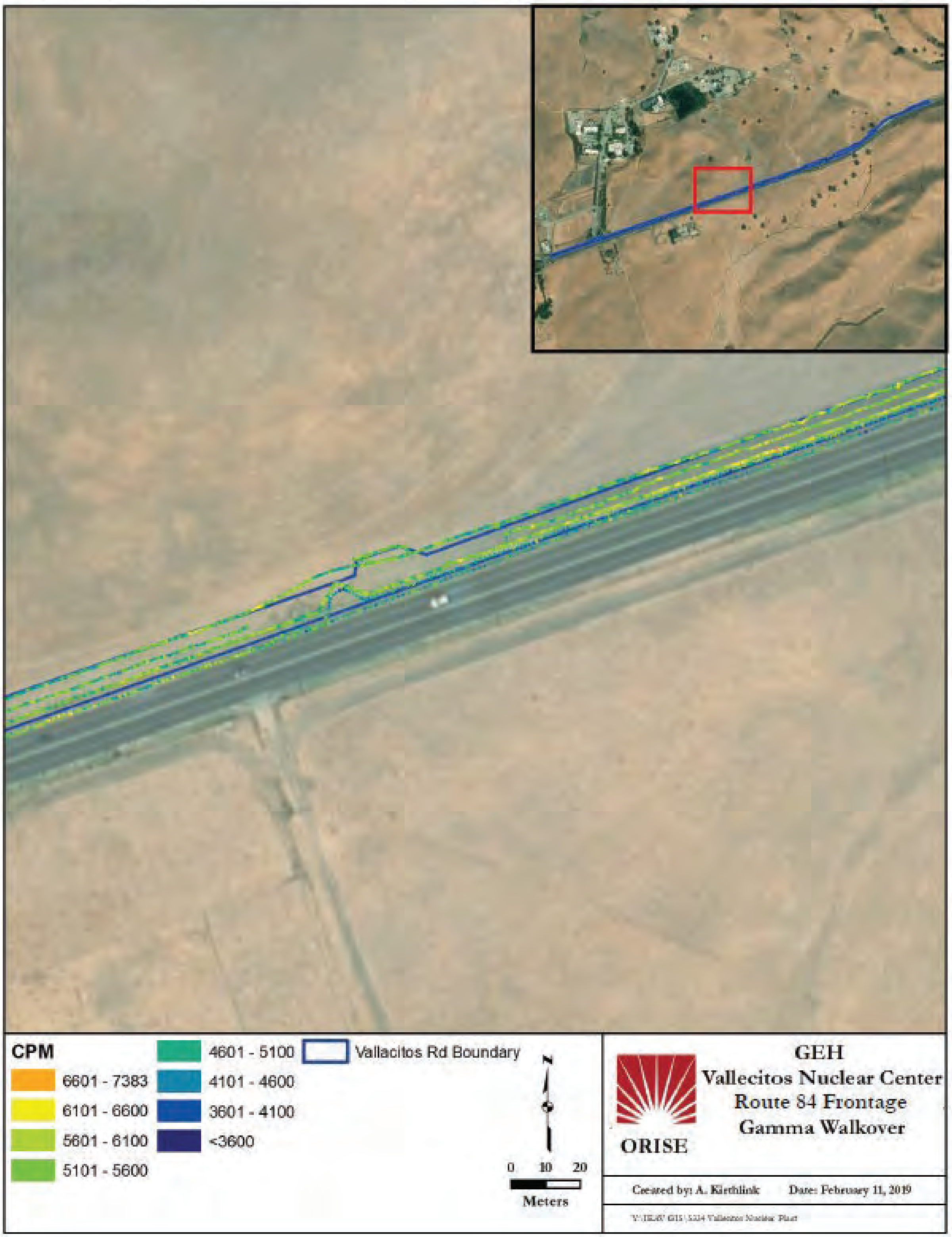

Figure A.5. Gamma Walkover Data (5 of 9)

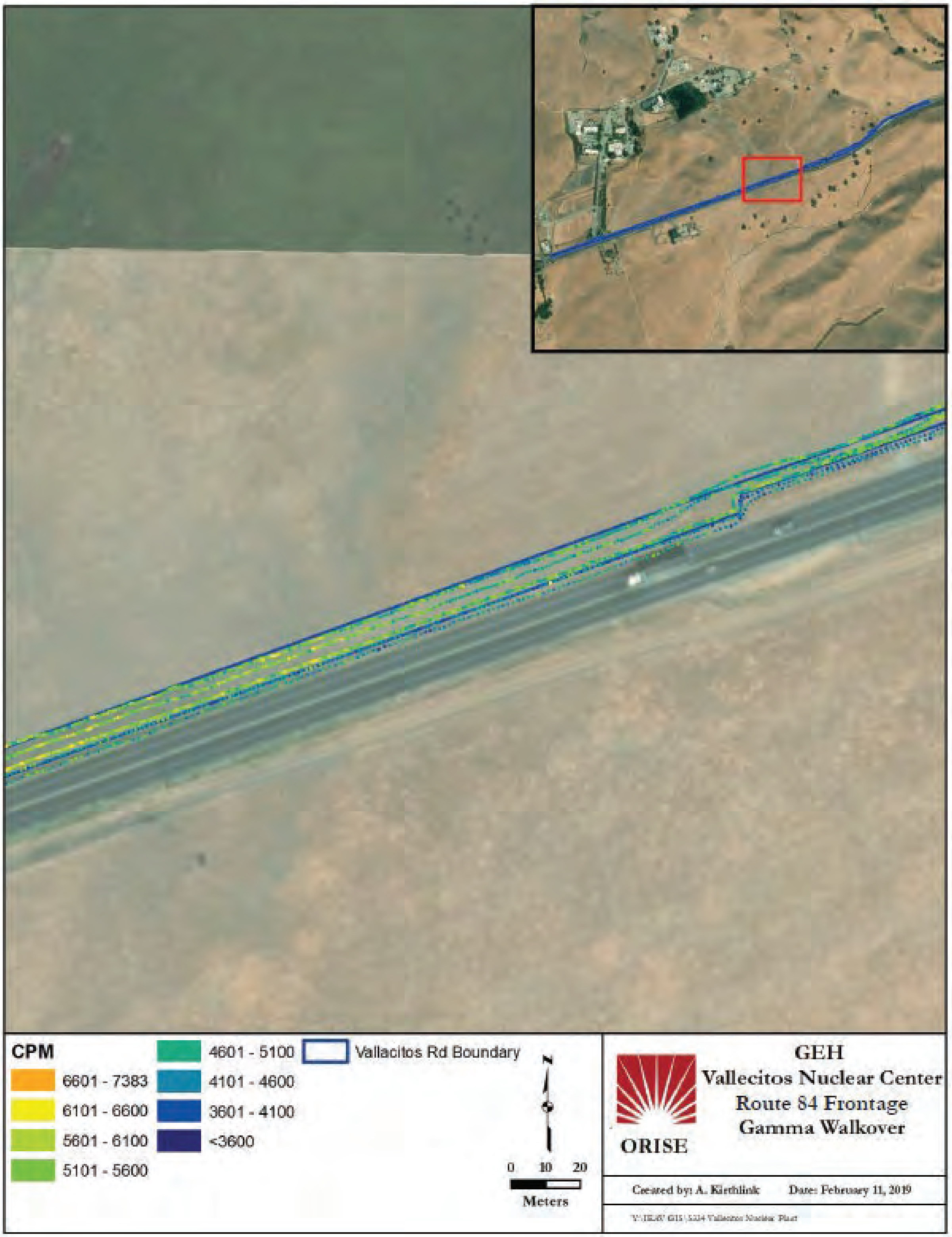

Figure A.6. Gamma Walkover Data (6 of 9)

Figure A.7. Gamma Walkover Data (7 of 9)

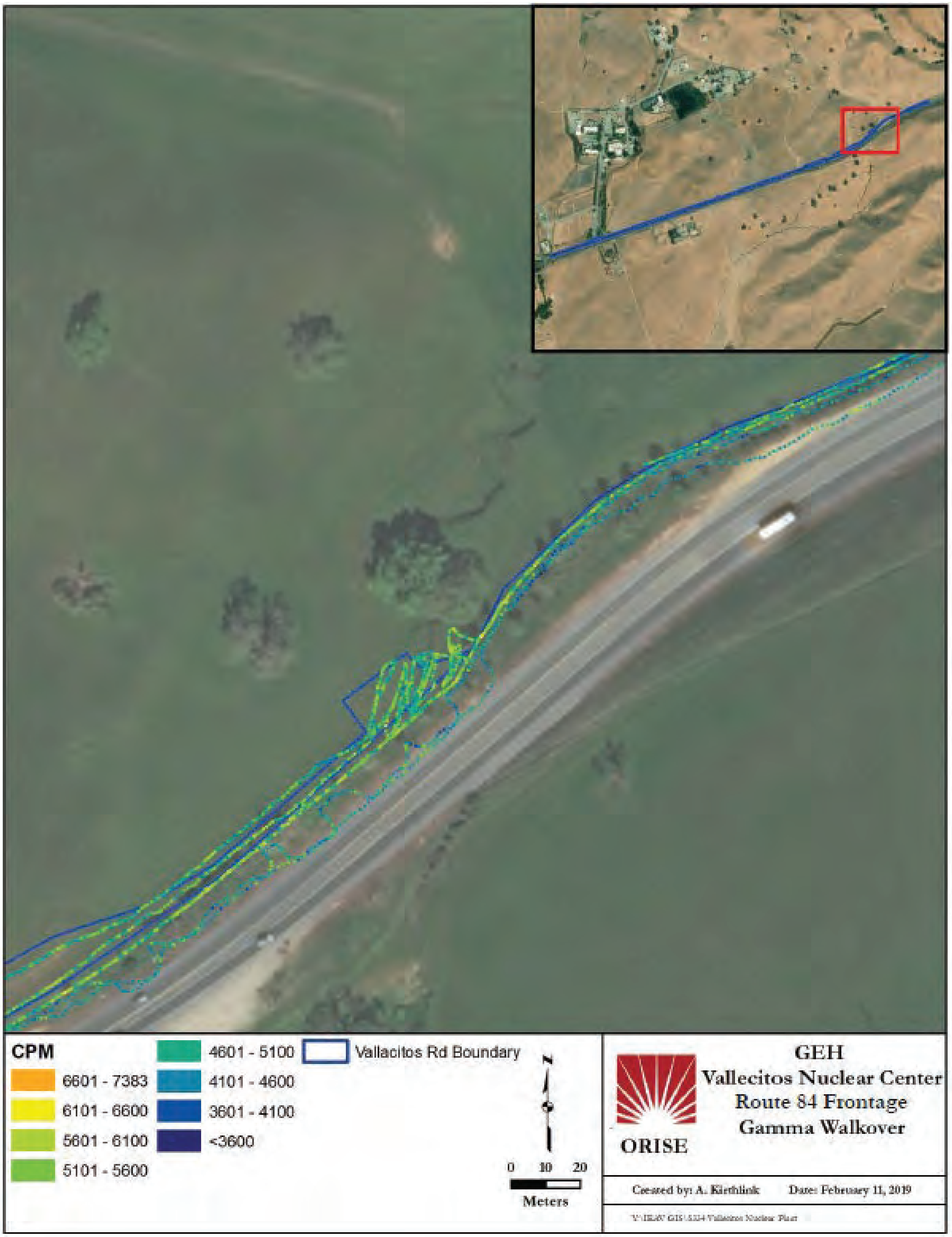

Figure A.8. Gamma Walkover Data (8 of 9)

Figure A.9. Gamma Walkover Data (9 of 9)

Figure A.10. Soil Sampling Locations (1 of 5)

Figure A.11. Soil Sampling Locations (2 of 5)

Figure A.12. Soil Sampling Locations (3 of 5)

Figure A.13. Soil Sampling Locations (4 of 5)

Figure A.14. Soil Sampling Locations (5 of 5)

APPENDIX B: DATA TABLE

Table B.1. Soil Sampling Measurement Locations						
Coordinates (ft)		Measurement Type	Gamma Count (cpm)		Soil Sample Number	Notes
X (Easting)	Y (Northing)		Pre-Sample	Post-Sample		
2044959	6172291	Random	5,500	6,100	5334S0001	
2045913	6175097	Random	5,400	5,900	5334S0002	
2045792	6174717	Random	5,400	6,200	5334S0003	
2045453	6173792	Random	5,700	6,300	5334S0004	
2047297	6177847	Random	5,200	5,600	5334S0005	
2045472	6173831	Random	5,500	6,300	5334S0006	
2045295	6173299	Random	6,200	7,500	5334S0007	
2046184	6175801	Random	4,900	5,700	5334S0008	
2045249	6173128	Random	5,800	6,600	5334S0009	
2046292	6176076	Random	5,200	5,900	5334S0010	
2046506	6176583	Random	4,700	4,800	5334S0011	
2046771	6177039	Random	5,000	5,500	5334S0012	
2046351	6176266	Random	4,800	5,300	5334S0013	
Mean			5,300	6,000	--	
Minimum			4,700	4,800	--	
Maximum			6,200	7,500	--	
6173364	2045927	Judgmental	5,300	5,700	5334S0014	Flagged by NRC
6173141	2043220	Judgmental	4,900	5,400	5334S0015	Site Loc. 2, Approx. Location
6172340	2044969	Judgmental	4,800	5,500	5334S0016	Site Loc. 10, using GPS coords.
6174783	2045803	Judgmental	5,000	5,400	5334S0019	Site Loc. 8, using GPS coords.
6177110	2046829	Judgmental	4,500	4,800	5334S0020	Site Loc. 9, using GPS coords.
6173133	2045219	Judgmental	4,900	5,400	5334S0021	Site Loc. 2, using GPS coords.
6172618	2045149	Judgmental	4,400	5,000	5334S0022	Site Loc. 7, using GPS coords.
Mean			4,800	5,300	--	
Minimum			4,400	4,800	--	
Maximum			5,300	5,700	--	

Table B.2. Radionuclide Concentrations in Soil Samples

Sample	Measurement Type	Cs-137 (pCi/g)				Gross Alpha (pCi/g)				Gross Beta (pCi/g)			
		Conc.		TPU[a]	MDC[b]	Conc.		TPU	MDC	Conc.		TPU	MDC
5334S0001	Random	0.023	±	0.014	0.029	**5.4** [c]	±	2.7	3.5	**11.6**	±	2.8	3.5
5334S0002	Random	**0.063**	±	0.019	0.034	**6.7**	±	2.8	3.4	**11.5**	±	2.8	3.4
5334S0003	Random	**0.046**	±	0.013	0.031	**4.0**	±	2.4	3.4	**12.9**	±	2.9	3.3
5334S0004	Random	0.006	±	0.028	0.062	3.2	±	2.4	3.5	**6.8**	±	2.4	3.4
5334S0005	Random	**0.073**	±	0.020	0.034	**6.8**	±	2.9	3.4	**9.4**	±	2.6	3.5
5334S0006	Random	0.023	±	0.009	0.025	**8.5**	±	3.1	3.4	**11.1**	±	2.7	3.2
5334S0007	Random	**0.043**	±	0.016	0.031	**6.3**	±	2.8	3.4	**13.9**	±	3.0	3.5
5334S0008	Random	**0.103**	±	0.021	0.031	**5.4**	±	2.7	3.4	**7.2**	±	2.4	3.3
5334S0009	Random	**0.065**	±	0.016	0.026	**4.8**	±	2.8	3.9	**13.9**	±	3.0	3.2
5334S0010	Random	**0.059**	±	0.014	0.029	**5.7**	±	2.8	3.7	**11.0**	±	2.7	3.3
5334S0011	Random	**0.086**	±	0.018	0.025	**6.6**	±	2.9	3.6	**7.6**	±	2.3	3.1
5334S0012	Random	**0.057**	±	0.019	0.034	**6.6**	±	2.9	3.6	**9.0**	±	2.5	3.1
5334S0013	Random	0.017	±	0.010	0.021	**5.1**	±	2.7	3.6	**9.9**	±	2.6	3.3
Mean		0.051			--	5.8			--	10.4			--
Minimum		0.006			--	3.2			--	6.8			--
Maximum		0.103			--	8.5			--	13.9			--
5334S0014	Judgmental	**0.066**	±	0.016	0.032	**10.0**	±	3.4	3.7	**13.6**	±	3.0	3.4
5334S0015	Requested by NRC	**0.049**	±	0.016	0.035	**7.1**	±	3.0	3.6	**11.9**	±	2.7	3.1
5334S0016	Requested by NRC	**0.056**	±	0.015	0.025	3.2	±	2.4	3.6	**10.2**	±	2.6	3.2
5334S0019	Requested by NRC	**0.065**	±	0.017	0.030	**10.2**	±	3.5	3.8	**13.1**	±	2.9	3.3
5334S0020	Requested by NRC	0.010	±	0.012	0.029	**4.5**	±	2.7	3.8	**11.4**	±	2.7	3.3
5334S0021	Requested by NRC	**0.043**	±	0.014	0.027	**8.1**	±	3.2	3.9	**10.0**	±	2.6	3.3
5334S0022	Requested by NRC	**0.071**	±	0.016	0.024	**4.3**	±	2.6	3.8	**9.8**	±	2.5	3.2
Minimum		0.010			--	3.2			--	9.8			--
Maximum		0.071			--	10.2			--	13.6			--

[a] Uncertainties are based on total propagated uncertainties at the 95% confidence level.
[b] MDC = minimum detectable concentrations.
[c] Results greater than MDC are bolded.

Table B.3. NORM Radionuclide Concentrations in Soil Samples

Sample	Measurement Type	K-40 (pCi/g)				Ra-226 (pCi/g)				U-238 (pCi/g)				Th-232 (pCi/g)			
		Conc.		TPU[a]	MDC[b]	Conc.		TPU	MDC	Conc.		TPU	MDC	Conc.		TPU	MDC
5334S0001	Random	**8.41** [c]	±	0.86	0.44	**0.506**	±	0.054	0.049	0.49	±	0.31	0.67	**0.56**	±	0.10	0.14
5334S0002	Random	**8.28**	±	0.80	0.49	**0.460**	±	0.049	0.049	**0.76**	±	0.29	0.48	**0.67**	±	0.10	0.12
5334S0003	Random	**7.18**	±	0.74	0.61	**0.439**	±	0.050	0.060	0.72	±	0.55	1.21	**0.584**	±	0.097	0.119
5334S0004	Random	**6.43**	±	0.83	0.83	**0.322**	±	0.048	0.060	0.42	±	0.27	0.59	**0.45**	±	0.11	0.17
5334S0005	Random	**7.12**	±	0.76	0.49	**0.432**	±	0.050	0.046	0.59	±	0.30	0.61	**0.442**	±	0.094	0.125
5334S0006	Random	**8.06**	±	0.73	0.44	**0.449**	±	0.047	0.057	**0.64**	±	0.24	0.40	**0.567**	±	0.085	0.088
5334S0007	Random	**10.35**	±	0.98	0.66	**0.520**	±	0.057	0.067	1.04	±	0.63	1.33	**0.67**	±	0.11	0.12
5334S0008	Random	**5.94**	±	0.68	0.44	**0.382**	±	0.049	0.052	**0.58**	±	0.28	0.54	**0.514**	±	0.098	0.110
5334S0009	Random	**9.57**	±	0.84	0.45	**0.495**	±	0.047	0.044	**0.73**	±	0.26	0.43	**0.699**	±	0.099	0.107
5334S0010	Random	**7.34**	±	0.74	0.59	**0.512**	±	0.052	0.055	**1.18**	±	0.59	1.16	**0.517**	±	0.090	0.115
5334S0011	Random	**4.33**	±	0.53	0.57	**0.385**	±	0.046	0.056	0.54	±	0.49	1.09	**0.475**	±	0.083	0.099
5334S0012	Random	**7.30**	±	0.77	0.43	**0.444**	±	0.051	0.049	0.39	±	0.26	0.57	**0.53**	±	0.10	0.13
5334S0013	Random	**6.12**	±	0.59	0.37	**0.357**	±	0.037	0.040	**0.60**	±	0.22	0.36	**0.435**	±	0.071	0.077
Mean		7.42			--	0.439			--	0.67			--	0.547			--
Minimum		4.33			--	0.322			--	0.39			--	0.435			--
Maximum		10.35			--	0.520			--	1.18			--	0.699			--
5334S0014	Judgmental	**10.19**	±	0.96	0.66	**0.510**	±	0.056	0.062	0.46	±	0.58	1.34	**0.66**	±	0.11	0.13
5334S0015	Requested by NRC	**7.49**	±	0.80	0.46	**0.482**	±	0.056	0.057	**0.77**	±	0.32	0.58	**0.59**	±	0.11	0.14
5334S0016	Requested by NRC	**6.61**	±	0.68	0.37	**0.418**	±	0.045	0.045	**0.77**	±	0.29	0.50	**0.518**	±	0.091	0.113
5334S0019	Requested by NRC	**7.42**	±	0.71	0.68	**0.450**	±	0.050	0.068	0.36	±	0.45	1.04	**0.625**	±	0.093	0.115
5334S0020	Requested by NRC	**6.57**	±	0.64	0.61	**0.361**	±	0.042	0.058	0.32	±	0.42	0.98	**0.411**	±	0.075	0.110
5334S0021	Requested by NRC	**8.64**	±	0.76	0.41	**0.469**	±	0.044	0.039	**0.73**	±	0.25	0.41	**0.625**	±	0.089	0.092
5334S0022	Requested by NRC	**7.28**	±	0.69	0.46	**0.393**	±	0.048	0.071	**0.80**	±	0.30	0.51	**0.687**	±	0.097	0.093
Minimum		6.57			--	0.361			--	0.32			--	0.411			--
Maximum		10.19			--	0.510			--	0.80			--	0.687			--

[a] Uncertainties are based on total propagated uncertainties at the 95% confidence level.
[b] MDC = minimum detectable concentrations.
[c] Results greater than MDC are bolded.

Table B.4. C1/C2 Cs-137 and NORM Concentrations (pCi/g)						
C1/C2 Sample ID	**Year**	**Cs-137**	**K-40**	**Ra-226**	**U-238**	**Th-232**
5273S0001	2015	0.058	12.2	0.5	0.51	0.65
5273S0002	2015	0.186	9.4	0.452	0.54	0.65
5273S0003	2015	0.128	6.15	0.421	0.53	0.536
5273S0004	2015	0.093	13.1	0.424	0.56	0.79
5273S0005	2015	0.179	8.74	0.503	0.74	0.554
5273S0006	2015	0.129	11.7	0.508	0.8	0.68
5273S0007	2015	0.130	9.13	0.331	0.53	0.549
5273S0008	2015	0.129	15.4	0.554	0.87	0.83
5273S0009	2015	0.201	8.62	0.428	0.9	0.58
5273S0010	2015	0.123	8.43	0.366	0.34	0.479
5273S0011	2015	0.116	7.74	0.391	0.5	0.566

Table B.5. VNC Gross Alpha and Gross Beta Concentrations (pCi/g)			
VNC Sample ID	**Year**	**Gross Alpha**	**Gross Beta**
S01	2018	10.5	16.2
S02	2018	14.8	12.5
S03	2018	10.3	8.60
S04	2018	8.00	8.13
S05	2018	8.95	13.3
S06	2018	7.34	10.8
S07	2018	14.3	13.7
S08	2018	14.4	16.7
S09	2018	8.06	33.2
S10	2018	11.9	37.9
S11	2018	7.12	6.19
S12	2018	6.38	9.33
S13	2018	9.5	19.7
S14	2018	14	12.3
S15	2018	12.2	12.9
S16	2018	13.3	15.3

Table B.6. 2009-2013 LLNL Concentrations (pCi/g)[a]									
LLNL Sample ID	**2009**	**2010**	**2011**	**2012**	**2013**	**2014**	**2015**	**2016**	**2017**
Cs-137 Concentrations Decay-Corrected to 2019									
L-AMON-SO	0.0365	0.017	0.0630	0.018	0.087	0.0747	0.0567	0.101	0.0929
L-CHUR-SO	0.0580	0.0527	0.094	0.0575	0.099	0.0651	0.0838	0.0656	0.0800
L-FCC-SO	0.0408	0.017	0.016	0.02992	0.0235	0.0386	0.0271	0.0277	0.0284
L-HOSP-SO	0.0344	0.0242	0.0337	0.0391	0.0330	0.0313	0.0345	0.111	0.147
L-PATT-SO	0.0236	0.015	0.021	0.0322	0.020	0.018	0.018	0.0404	0.0284
L-TANK-SO	0.0430	0.0549	0.0495	0.0437	0.0518	0.0410	0.0592	0.0429	0.0516
L-ZON7-SO	0.0073	0.009	0.017	0.016	0.014	0.016	0.015	0.013	0.015
K-40 Concentrations									
L-AMON-SO	14.11	13.70	14.70	13.11	14.19	15.30	15.59	15.89	14.81
L-CHUR-SO	13.11	13.11	13.30	13.00	13.30	13.51	13.51	13.70	13.81
L-FCC-SO	11.30	9.378	10.00	10.19	10.41	10.70	9.351	9.2703	9.703
L-HOSP-SO	10.41	11.59	9.9459	12.70	10.486	10.405	10.486	11.189	11.89
L-PATT-SO	14.89	14.19	12.89	16.00	12.49	13.70	13.59	11.41	11.70
L-TANK-SO	8.8378	9.2432	8.270	8.8378	9.297	9.5135	9.081	8.0000	9.568
L-ZON7-SO	10.59	10.89	10.41	11.11	10.108	10.00	10.70	9.784	11.00
U-238 Concentrations									
L-AMON-SO	0.673	0.74	0.67	0.74	0.57	0.74	0.84	0.71	0.67
L-CHUR-SO	0.77	0.50	0.81	0.84	0.61	0.77	0.54	0.77	0.74
L-FCC-SO	0.77	0.47	0.54	0.50	0.57	0.437	0.44	0.30	0.370
L-HOSP-SO	0.64	0.639	0.47	0.606	0.606	0.471	0.81	0.370	0.47
L-PATT-SO	0.50	0.67	0.54	0.639	0.77	0.71	0.54	0.57	0.404
L-TANK-SO	0.54	0.538	0.538	0.505	0.54	0.538	0.64	0.370	0.54
L-ZON7-SO	1.6	2.5	2.6	1.9	2.0	2.6	1.7	1.9	2.3
Th-232 Concentrations									
L-AMON-SO	0.929	0.831	0.929	0.831	0.754	0.918	0.918	0.951	0.885
L-CHUR-SO	0.820	0.809	0.940	0.907	0.863	0.907	0.853	0.929	0.929
L-FCC-SO	0.656	0.634	0.721	0.689	0.656	0.710	0.590	0.579	0.656
L-HOSP-SO	0.623	0.645	0.634	0.678	0.546	0.557	0.557	0.514	0.568
L-PATT-SO	0.820	0.853	0.809	0.940	0.721	0.885	0.842	0.874	0.798
L-TANK-SO	0.700	0.743	0.667	0.634	0.754	0.787	0.754	0.634	0.820
L-ZON7-SO	0.831	0.863	0.896	0.929	0.787	0.896	0.907	0.798	0.940

[a] Ra-226 concentrations not available.

Table B.7. Results of the Student (Pooled) t-Test					
Date/Time of Computation		ProUCL 5.1 7/2/2019 12:47:43 PM			
From File		WorkSheet.xls			
Full Precision		OFF			
Confidence Coefficient		95%			
Substantial Difference (S)		0			
Selected Null Hypothesis		Sample 1 Mean <= Sample 2 Mean (Form 1)			
Alternative Hypothesis		Sample 1 Mean > the Sample 2 Mean			
Sample 1 Data: ORISE					
Sample 2 Data: C1/C2					
Sample 1 vs Sample 2 Two-Sample t-Test					
H_0: Mean of Sample 1 - Mean of Sample 2 <= 0					
Method		DF	t-Test Value	Critical t (0.05)	P-Value
Pooled (Equal Variance)		22	-5.774	1.717	1
Pooled SD 0.035					
Conclusion with Alpha = 0.050					
Student t (Pooled) Test: Do Not Reject H_0 Conclude Sample 1 <= Sample 2					
Test of Equality of Variances					
Variance of Sample 1			8.09E-04		
Variance of Sample 2			0.00172		
Numerator DF	Denominator DF	F-Test Value	P-Value		
10	12	2.128	0.216		
Conclusion with Alpha = 0.05					
Two variances appear to be equal					

APPENDIX C: SURVEY AND ANALYTICAL PROCEDURES

C.1. PROJECT HEALTH AND SAFETY

ORISE performed all survey activities in accordance with the *ORAU Radiation Protection Manual*, the *ORAU Health and Safety Manual*, and the *ORAU Radiological and Environmental Survey Procedures Manual* (ORAU 2014, ORAU 2016b, and ORAU 2016a). Prior to on-site activities, a Work-Specific Hazard Checklist was completed for the project and discussed with field personnel. The planned activities were thoroughly discussed with site personnel prior to implementation to identify hazards present. Additionally, prior to performing work, a pre-job briefing and walk down of the survey areas were completed with field personnel to identify hazards present and discuss safety concerns. Should ORISE have identified a hazard not covered in the *ORAU Radiological and Environmental Survey Procedures Manual* (ORAU 2016a) or the project's Work-Specific Hazard Checklist for the planned survey and sampling procedures, work would not have been initiated or continued until the hazard was addressed by an appropriate job hazard analysis and hazard controls.

C.2. CALIBRATION AND QUALITY ASSURANCE

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

Field survey activities were conducted in accordance with procedures from the following documents:

- *ORAU Radiological and Environmental Survey Procedures Manual* (ORAU 2016a)
- *ORAU Radiological and Environmental Analytical Laboratory Procedures Manual* (ORAU 2017)
- *ORAU Environmental Services and Radiation Training Quality Program Manual* (ORAU 2018)

The procedures contained in these manuals were developed to meet the requirements of U.S. Department of Energy (DOE) Order 414.1D and the 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 and Intercomparison Testing Program laboratory quality assurance programs
- Training and certification of all individuals performing procedures
- Periodic internal and external audits

C.3. SURVEY PROCEDURES

C.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. Specific scan minimum detectable concentrations (MDCs) for the sodium iodide (NaI) scintillation detectors were not determined because the instruments were used solely as a qualitative means to identify elevated gamma radiation levels in excess of background. Identifications of elevated radiation levels that could exceed the site criteria were determined based on an increase in the audible signal from the indicating instrument.

C.3.2 Soil Sampling

Soil samples (approximately 0.5 kilogram each) were collected by ORISE personnel using a clean garden trowel to transfer soil into a new sample container. The container was then labeled and security sealed in accordance with ORISE procedures. ORISE shipped samples under chain-of-custody to the ORISE laboratory for analysis.

C.4. RADIOLOGICAL ANALYSIS

C.4.1 Gamma Spectroscopy

Samples were analyzed as received, mixed, crushed, and/or homogenized, as necessary, and a portion sealed into an appropriate volume Marinelli beaker or 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, high-purity, germanium detectors coupled to a pulse height analyzer system. Background and Compton stripping, peak search, peak identification, and concentration calculations were performed using computer capabilities inherent in the analyzer system. All total absorption peaks (TAPs) associated with the ROCs were reviewed for consistency of activity. Spectra also were reviewed for other identifiable TAPs. TAPs used for determining the activities of ROCs and the typical MDCs for a 1-hour count time for ROCs are presented in Table C.1.

Table C.1. Typical MDCs Total Absorption Peak for Gamma Emitters

Radionuclide	TAP (keV)[a]	MDC (pCi/g)
Cs-137	661.66	0.05
U-238 by Th-234	63.29	0.75
Ra-226 by Pb-214	351.93	0.08
K-40	1,460.82	0.5
Th-232 by Ac-228	911.20	0.14

[a]kilo electron volt

C.4.2 Low Background Proportional Counter

Samples were dried and processed to provide homogeneity, and a known quantity was transferred to a planchet and counted in a low-background proportional counter. The activity determined by this method is not indicative of any specific nuclide, but, instead, gross alpha and gross beta. Samples were counted for 200 minutes. Typical MDCs are 3.8 pCi/g for alpha and 3.3 pCi/g for beta.

C.4.3 Detection Limits

Detection limits, referred to as MDCs, were based on a 95% confidence level. 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.

APPENDIX D: 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 his employer.

D.1. SCANNING AND MEASUREMENT INSTRUMENT/ DETECTOR COMBINATIONS

D.1.1 GAMMA

Ludlum NaI Scintillation Detector Model 44-10, Crystal: 5.1 cm × 5.1 cm
(Ludlum Measurements, Inc., Sweetwater, Texas)
coupled to: Ludlum Ratemeter-scaler Model 2221
(Ludlum Measurements, Inc., Sweetwater, Texas)
coupled to: Trimble Geo 7X
(Trimble Navigation Limited, Sunnyvale, CA)

D.2. LABORATORY ANALYTICAL INSTRUMENTATION

Low-Background Gas Proportional Counter
Series 5 XLB
(Canberra, Meriden, Connecticut)
Used in conjunction with:
Eclipse Software
Dell Workstation
(Canberra, Meriden, Connecticut)

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

High-Purity, Intrinsic Detector
EG&G ORTEC Model No. GMX-45200-5
Canberra Lynx ® Multichannel Analyzer
Canberra Gamma-Apex Software
(Canberra, Meriden, Connecticut)
Used in conjunction with:
Lead Shield Model G-11
(Nuclear Lead, Oak Ridge, Tennessee) and
Dell Workstation
(Canberra, Meriden, Connecticut)

High-Purity, Intrinsic Detector
EG&G ORTEC Model No. GMX-30P4
Canberra Lynx ® Multichannel Analyzer
Canberra Gamma-Apex Software
(Canberra, Meriden, Connecticut)
Used in conjunction with:
Lead Shield Model G-11
(Nuclear Lead, Oak Ridge, Tennessee) and
Dell Workstation
(Canberra, Meriden, Connecticut)

High-Purity, Intrinsic Detector
EG&G ORTEC Model No. CDG-SV-76/GEM-MX5970-S
Canberra Lynx ® Multichannel Analyzer
Canberra Gamma-Apex Software
(Canberra, Meriden, Connecticut)
Used in conjunction with:
Lead Shield Model G-11
(Nuclear Lead, Oak Ridge, Tennessee) and
Dell Workstation
(Canberra, Meriden, Connecticut)