CDC
RAW Room #1
Decommissioning Plan

Prepared for:
U.S. Department of Health & Human Services
Centers for Disease Control & Prevention
4770 Buford Highway
Chamblee, GA 30341
Radioactive Materials License #10-06772-01

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ACRONYM LIST

ALARA	*As Low As Reasonably Achievable*
Bldg	*Building*
CDC	*United States Department of Health and Human Services Centers for Disease Control and Prevention*
CFR	*Code of Federal Regulations*
CPM	*Counts Per Minute*
CRSO	*Corporate Radiation Safety Officer*
D&D	*Decontamination and Decommissioning*
DP	*Decommissioning Plan*
$DCGL_W$	*Derived Concentration Guideline Level – Wilcoxon Rank Sum*
DQO	*Data Quality Objective*
DPM	*Disintegrations Per Minute*
FSS	*Final Status Survey*
FSSR	*Final Status Survey Report*
HSA	*Historical Site Assessment*
HPT	*Health Physics Technician*
LBGR	*Lower Bound of the Gray Region*
MARSSIM	*Multi-Agency Radiation Survey and Site Investigation Manual*
MDC	*Minimum Detectable Concentration*
NIST	*National Institute of Standards and Technology*
NMSS	*Nuclear Materials Safety and Safeguards*
NRC	*U.S. Nuclear Regulatory Commission*
NUREG	*Nuclear Regulatory Commission Guidance Document*
ESHCO	*Environment, Safety, and Health Compliance Office*
PM	*Project Manager*
PPE	*Personnel Protective Equipment*
RAM	*Radioactive Materials*
RAW Room #1	*Radioactive Waste Room #1*
RSO	*Radiation Safety Officer*
RWP	*Radiation Work Permit*
TEDE	*Total Effective Dose Equivalent*

GLOSSARY

ALARA. Acronym for "as low as is reasonably achievable," which means making every reasonable effort to maintain exposures to radiation as far below the dose limits as is practical, consistent with the purpose for which the licensed activity is undertaken, and taking into account the state of technology, the economics of improvements in relation to the state of technology, the economics of improvements in relation to the benefits to the public health and safety, and other societal and socioeconomic considerations, and in relation to utilization of nuclear energy and licensed materials in the public interest (see 10 CFR 20.1003).

Characterization survey. A type of survey that includes facility or site sampling, monitoring, and analysis activities to determine the extent and nature of residual radioactivity. Characterization surveys provide the basis for acquiring necessary technical information to develop, analyze, and select appropriate cleanup techniques

Decommission. To remove a facility or site safely from service and reduce residual radioactivity to a level that permits (1) release of the property for unrestricted use and termination of the license or (2) release of the property under restricted conditions and termination of the license (see 10 CFR 20.1003).

Decommissioning Plan (DP). A detailed description of the activities that the licensee intends to use to assess the radiological status of its facility, to remove radioactivity attributable to licensed operations at its facility to levels that permit release of the site in accordance with NRC's regulations and termination of the license, and to demonstrate that the facility meets NRC's requirements for release. A DP typically consists of several interrelated components, including (1) site characterization information; (2) a remediation plan that has several components, including a description of remediation tasks, a health and safety plan, and a quality assurance plan; (3) site-specific cost estimates for the decommissioning; and (4) a final status survey plan (see 10 CFR 30.36(g)(4).

Decontamination. The removal of undesired residual radioactivity from facilities, soils, or equipment prior to the release of a site or facility and termination of a license. Also known as remediation, remedial action, and cleanup.

Derived Concentration Guideline Levels (DCGLs). Radionuclide-specific concentration limits used by the licensee during decommissioning to achieve the regulatory dose standard that permits the release of the property and termination of the license. The DCGL applicable to the average concentration over a survey unit is called the DCGLW. The DCGL applicable to limited areas of elevated concentrations within a survey unit is called the DCGLEMC.

Dose (or radiation dose). A generic term that means absorbed dose, dose equivalent, effective dose equivalent, committed dose equivalent, committed effective dose equivalent, or total effective dose equivalent, as defined in other paragraphs of 10 CFR 20.1003 (see 10 CFR 20.1003). In this NUREG report, dose generally refers to total effective dose equivalent (TEDE).

Final Status Survey (FSS). Measurements and sampling to describe the radiological conditions of a site or facility, following completion of decontamination activities (if any) and in preparation for release of the site or facility.

Final Status Survey Plan (FSSP). The description of the final status survey design.

Final Status Survey Report (FSSR). The results of the final status survey conducted by a licensee to demonstrate the radiological status of its facility. The FSSR is submitted to NRC for review and approval.

Historical Site Assessment (HSA). The identification of potential, likely, or known sources of radioactive material and radioactive contamination based on existing or derived information for the purpose of classifying a facility or site, or parts thereof, as impacted or non-impacted (see 10 CFR 50.2).

Historical Site Assessment (HSA). The identification of potential, likely, or known sources of radioactive material and radioactive contamination based on existing or derived information for the purpose of classifying a facility or site, or parts thereof, as impacted or non-impacted (see 10 CFR 50.2).

Impact. The positive or negative effect of an action (past, present, or future) on the natural environment (land use, air quality, water resources, geological resources, ecological resources, aesthetic and scenic resources) and the human environment (infrastructure, economics, social, and cultural).

Impacted Areas. The areas with some reasonable potential for residual radioactivity in excess of natural background or fallout levels (see 10 CFR 50.2).

Leak Test. A test for leakage of radioactivity from sealed radioactive sources. These tests are made when the sealed source is received and on a regular schedule thereafter. The frequency is usually specified in the sealed source and device registration certificate and/or license.

MARSSIM. The Multi-Agency Radiation Site Survey and Investigation Manual (NUREG–1575) is a multi-agency consensus manual that provides information on planning, conducting, evaluating, and documenting building surface and surface soil final status radiological surveys for demonstrating compliance with dose- or risk-based regulations or standards.

Monitoring. Monitoring (radiation monitoring, radiation protection monitoring) is the measurement of radiation levels, concentrations, surface area concentrations, or quantities of radioactive material and the use of the results of these measurements to evaluate potential exposures and doses (see 10 CFR 20.1003).

Non-impacted Areas. The areas with no reasonable potential for residual radioactivity in excess of natural background or fallout levels (see 10 CFR 50.2).

Residual Radioactivity. Radioactivity in structures, materials, soils, ground water, and other media at a site resulting from activities under the licensee's control. This includes radioactivity from all licensed and unlicensed sources used by the licensee, but excludes background radiation. It also includes radioactive materials remaining at the site as a result of routine or accidental releases of radioactive material at the site and previous burials at the site, even if those burials were made in accordance with the provisions of 10 CFR Part 20 (see 10 CFR 20.1003).

RESRAD Code. A computer code developed by the U.S. Department of Energy and designed to estimate radiation doses and risks from RESidual RADioactive materials in soils.

RESRAD-BUILD Code. A computer code developed by the U.S. Department of Energy and designed to estimate radiation doses and risks from RESidual RADioactive materials in BUILDings.

Scoping Survey. A type of survey that is conducted to identify (1) radionuclide contaminants, (2) relative radionuclide ratios, and (3) general levels and extent of residual radioactivity.

Site Characterization. Studies that enable the licensee to sufficiently describe the conditions of the site, separate building, or outdoor area to evaluate the acceptability of the decommissioning plan.

Survey Unit. A geographical area consisting of structures or land areas of specified size and shape at a site for which a separate decision will be made as to whether or not the unit attains the site-specific reference-based cleanup standard for the designated pollution parameter. Survey units are established to facilitate the survey process and the statistical analysis of survey data.

1 EXECUTIVE SUMMARY

The U.S. Department of Health and Human Services Centers for Disease Control and Prevention (CDC) has decided to cease all operations and permanently decommission its Radioactive Waste facility located in Chamblee Building 1, radioactive water room #1 (RAW Room #1) at the CDC facility located at 4770 Buford Highway. RAW Room #1 had previously served as a facility for radioactive waste materials collection, storage, classification and packaging prior to shipment for disposal. As a result of the completion and occupation of Chamblee Building 164 (Chamblee Material Handling Facility), Philotechnics, Ltd. (Philotechnics) has been contracted to perform all decommissioning activities and attain release for unrestricted use of the facility. Upon release, the CDC will demolish RAW Room #1 and dispose of the contents in appropriate landfills. The room is vacated, and any unneeded and potentially contaminated items were surveyed. Surveyed items were found to be free of any residual contamination.

The CDC and Philotechnics conducted a Historical Site Assessment (HSA) documenting radiological operations from the beginning of licensed operations in RAW Room #1. A thorough review of the historical utilization of RAW Room #1 enclosure reveals that it was used to collect, store, classify, and process CDC licensed radioactive materials stored at the facility. Radioactive materials consisted of: H-3, C-14, P-32, P-33, S-35, Mn-54, Co-60, Zn-65, Sr-90, Y-90, I-125, Cs-137, Po-209, Po-210, U-235, U-238, Np-237, Pu-239, Pu-242, Am-241, natural thorium (Th-nat) and natural uranium (U-nat).

Philotechnics performed scoping surveys of RAW Room #1 on March 10 & 11, 2014. Scoping surveys included scans, direct/static measurements for total activity, and 100 cm^2 smears for removable activity.

Philotechnics will perform all decommissioning activities in accordance with this Decommissioning Plan (DP) developed following the requirements listed in Chapter 16 and 17 of NUREG 1757 Volume 1, Revision 2. The DP will ensure sufficient analytical data and pertinent information are acquired to provide for the radiological decommissioning. The DP will follow the guidance and recommendations provided in NUREG 1757, "*Consolidated NMSS Decommissioning Guidance*"; and NUREG 1575, "*Multi-Agency Radiation Survey and Site Investigation Manual*" (MARSSIM). This provides the approach, methods, and techniques for the radiological decommissioning of RAW Room #1. To demonstrate compliance with site-specific Derived Concentration Guideline Levels (DGGLs) generated using a RESRAD-BUILD dose model, final status surveys will implement the protocols and guidance provided in MARSSIM. This will ensure technically defensible data are generated to release the facility for unrestricted use in accordance with the criterion of 10 CFR 20.1402, *"A site will be considered acceptable for unrestricted use if the residual radioactivity that is distinguishable from background radiation results in a TEDE to an average member of the critical group that does not exceed 25 mrem (0.25 mSv) per year, including that from groundwater sources of drinking water, and that the residual radioactivity has been reduced to levels that are as low as reasonably achievable (ALARA).Determination of the levels which are ALARA must take into account consideration of any detriments, such as deaths from transportation accidents, expected to potentially result from decontamination and waste disposal"*

2 FACILITY OPERATING HISTORY

The decommissioning process evaluates a property's environmental status for release of impacted areas to allow unrestricted use by current or future tenants. Philotechnics and the CDC performed a Historical Site Assessment (HSA) to review facility operations as they pertain to radioactive materials (RAM) storage to identify potential residual radioactive contamination. This assessement was performed prior to commencing Scoping surveys. The purpose was to determine the status of the facility including potential, likely, or known sources of radioactive contamination by gathering data from various sources. This included physical characteristics of the site as well as information found in site operating records. Assessment activities related to the decommissioning of the facility included the following tasks:

- A visual survey of historic RAM storage areas in order to identify potential contamination and/or presence of radioactive materials;
- Interviews with client personnel regarding the historical use of RAM at the facility;
- Review of existing documentation, as provided, regarding prior inspections, investigations, events or conditions at the facility related to RAM use, including: Radioactive materials license, applications, amendment requests, incident reports, records of RAM delivered to and shipped from Building 1, RAM inventories and facility renovation records, radiological surveys of the facility and records of RAM shipments into an out of the facility, laboratories on the Chamblee campus and the RSO provided relevant records;
- Direct surveys of all impacted areas with the use of portable hand-held radiation detection equipment to identify the presence of radioactive materials; and
- Indirect surveys to test for removable contamination with the use of a scintillation counter and wipes taken throughout the impacted areas.
- Dose estimates for alpha sources using the entire on-hand quantities to determine if they can be excluded from consideration.

2.1 Licensed Operations

Mr. Dave Aguero, Philotechnics, interviewed Mr. Paul Simpson, the Radiation Safety Officer (RSO) at CDC. This interview and document reviews revealed RAW Room #1 was under the operation of the former CDC Office of Safety, Health and Environment (OSHE) from 1986 until May 2013 where it served as a storage facility for the collection, storage, classification, and packaging of radioactive waste materials prior to shipment. As part of a CDC program reorganization, OSHE became the Environment, Safety, and Health Compliance Office (ESHCO), and it maintains the same program and NRC license responsibilities over RAW Room #1. Since 1986, the enclosure has supported solid (dry), liquid, and scintillation vial waste retrieved from the radiation laboratories located on the Chamblee campus. No waste was received at RAW Room #1 prior to 1986. From the beginning, the focus at the facility was to follow strict procedures to maintain the space free of radiological contamination. During the first ten years of operation, pure beta emitters, H-3, C-14, S-35, P-32 and I-125, at millicuries levels or less were the primary radioisotopes in storage. Following this period, a wider range of radioisotopes, including the actinides in microcurie or smaller activity levels were stored in RAW ROOM #1 (limited by the sensitivities needed for R&D studies). Licensed radioactive materials consisted of: H-3, C-14, P-32, P-33, S-35, Mn-54, Co-60, Zn-65, Sr-90, Y-90, I-125, Cs-137, Po-209, Po-210, U-235, U-238, Np-237, Pu-239, Pu-242, Am-241, Th-nat, U-nat.

- The NRC added Th-nat and U-nat to the license in 2001 by amendment #39. The purpose was to meet requirements in 10 CFR Part 40 for the physical protection of import, export, and transient shipments of natural uranium that might "endanger life or property or the common defense and security."
- The NRC removed Th-nat and U-nat from the license in 2007 by amendment #44. The quantities of Th-nat and U-nat possessed by CDC were not an endangerment, and as such, were exemptable from licensing based on the specific exemptions in 10 CFR Part 40.14 (regarding no endangerment potential) and in 10 CFR Part 40.22 (which exemts the CDC possession quantities from specific licensing).
- The quantities of Th-nat and U-nat that CDC possessed were exemptable from specific licensing during 2001-2007. Therefore, it is considered that they are not relevant to this decommissioning and they should be excluded from the assessment of residual activity in Building 1 RAW Room #1; **however, for added conservatism in the survey design, they were included.**

Additionally, leak test records and historical radiological survey results indicated that radioactivity would be several orders of magnitude less than the DCGLs.

Radioactive wastes were only stored in RAW Room #1 as summarized in **Table 2-1 - Restricted Area Summary** below, and identified on the building diagrams in Appendix A. The Scoping survey was developed and implemented to detect all relevant radionuclides. Philotechnics performed 100% scan surveys of all accessible areas of RAW Room #1 during the initial Scoping. Radioactive waste was only stored in RAW Room #1, and no other area in Chamblee Building 1. Access to RAW Room #1 is only from the external dock, and no access exists from Chamblee Building 1 directly. Additionally, all waste packages were sealed and leak tested upon delivery and prior to shipment, and routine surveys of the room were performed each time waste was packaged into 55 gallon drums that were resident in the storage area. In addition, wipe surveys were conducted minimally once every four months. The results of the CDC radiation meter and wipe surveys indicated all items were free from any residual contamination and at natural background levels. Additionally, according to the CDC RSO, there were never any spills, leaks, container deterioration/breakage, or other contamination events in RAW Room #1; although contamination in RAW Room #1 was identified and remediated during Philotechnics Scoping surveys. A detailed description is provided in **Section 4.1 – Contaminated Structures** and **Section 9.2 – Decontamination/Dismantelment and Remedial Action Surveys.**

As part of their public health modernization program, the CDC relocated all radioactive waste activities in RAW Room #1 to a newly completed radioactive waste building on the Chamblee campus.

Table 2-1 - Restricted Area Summary

4770 Buford Highway		
Area	**Room**	**Historical Radionuclide Usage**
RAW Room	1	H-3, C-14, P-32, P-33, S-35, Cr-51, Mn-54, Co-57, Co-60, Zn-65, Y-88, Sr-90, Y-90, Cd-109, Sn-113, I-125, Cs-137, Ce-139, Po-209, Po-210, U-235, U-238, Np-237, Pu-239, Pu-242, Am-241, Th-nat, U-nat.

2.2 License Number/Status/Authorized Activities

The CDC is currently authorized to possess the following radionuclides as summarized in **Table 2 - RAM License Possession Limits** below as referenced by amendment number 48 of Radioactive Materials License 10-06772-01:

Table 2-2 - RAM License Possession Limits

	Nuclide	Form	Possession Limit
A.	Any byproduct material with atomic numbers 1 through 83, except as specified below	Any	100 millicuries per radionuclide and 5 curies total
B.	Any byproduct material with atomic numbers 84 through 96, except as specified below	Any	2 millicuries per radionuclide and 25 millicuries total
C.	Hydrogen 3	Any	250 millicuries
D.	Phosphorus 32	Any	350 millicuries
E.	Sulfur 35	Any	350 millicuries
F.	Chromium 51	Any	350 millicuries
G.	Iodine 125	Any	220 millicuries
H.	Thorium 228	Any	1 millicurie
I.	Thorium 230	Any	1 millicurie
J.	Uranium 233	Any	1 millicurie
K.	Uranium 234	Any	1 millicurie
L.	Uranium 235	Any	0.7 millicurie
M.	Uranium 236	Any	1 millicurie
N.	Plutonium 238	Any	1 millicurie
O.	Plutonium 239	Any	1 millicurie
P.	Plutonium 240	Any	1 millicurie
Q.	Plutonium 242	Any	1 millicurie
R.	Californium 252	Any	1 millicurie
S.	Nickel 63	Foil or plated sources registered either with the U.S. NRC under 10 CFR 32.210 or with an Agreement State	400 millicuries
	Natural thorium (on NRC license in 2001-2007)	Any	0.151 millicurie
	Natural uranium (on NRC license in 2001-2007)	Any	0.453 millicurie

2.2.1 Authorized Use

A. through R. Research and development as defined in 10 CFR 30.4, and calibration and quality control standards for the licensee's instruments

S. To be used for sample analysis in compatible gas chromatography devices that has been registered with the U.S. Nuclear Regulatory Commission under 10 CFR 32.210 or with an Agreement State.

2.3 Licensed Radionuclides Stored at RAW Room #1

The following licensed radioactive materials were stored at RAW Room #1:

Table 2-3 - Nuclides Stored at the RAW Room #1

Nuclide	Half-life (years)	Half-Life >120 Days	Predominant Emissions
H-3	1.2E+01	YES	Beta
C-14		YES	Beta
Na-22	2.6E+00	YES	Beta
P-32	3.9E-02	NO	Beta
P-33	7.0E-02	NO	Beta
Cr-51	7.7E-02	NO	Beta
Mn-54	8.6E-01	YES	Gamma (ε)
Co-57	7.4E-01	YES	Beta
Co-60	5.3E+00	YES	Beta/Gamma
Zn-65	6.78E-01	YES	Gamma (ε)
Sr-85	1.8E-01	NO	Gamma (ε)
Sr-89	1.5e-01	NO	Beta
Sr-90	2.9E+01	YES	Beta
Y-88	3.0E-01	NO	Gamma (ε)
Cd-109	1.3E+00	YES	Beta
Sn-113	3.2E-01	NO	Gamma (ε)
I-125	1.6E-01	NO	Low E Beta
I-131	2.2E-02	NO	Beta
Cs-134	2.0E+00	YES	Beta
Cs-137	3.0E+01	YES	Beta/Gamma
Ba-133	1.0+01	YES	Gamma (ε)
Ce-139	3.8E-01	YES	Beta
Eu-152	1.3E+01	YES	Beta
Ir-192	2.0E-01	NO	Beta
Hg-203	1.3E-01	NO	Beta
Tl-204	3.8E+00	YES	Beta
Po-209	1.1E+02	YES	Alpha
Po-210	3.8E-01	YES	Alpha
Ra-226	1.6E+03	YES	Alpha
Th-232	1.4E+10	YES	Alpha
U-233	1.56E+05	YES	Alpha

Nuclide	Half-life (years)	Half-Life >120 Days	Predominant Emissions
U-235	7.1E+08	YES	Alpha
U-236	2.4E+07	YES	Alpha
U-238	4.5E+09	YES	Alpha
Pu-238	8.6E+01	YES	Alpha
Pu-239	2.4E+04	YES	Alpha
Pu-240	6.6E+03	YES	Alpha
Pu-242	3.8E+05	YES	Alpha
Am-241	4.3E+02	YES	Alpha
Am-243	7.9E+03	YES	Alpha

The radionuclides P-32, P-33, Cr-51, Sr-85, Sr-89, Y-90, Sn-113, I-125, I-131, Ir-192, and Hg-203 were eliminated as radionuiclides of concern due to short half lives.

2.4 Previous Decommissioning Activities

Based on interviews with the RSO and document reviews, there are no records of previous decommissioning activities performed at RAW Room #1.

2.5 Radioactive Materials Spills

One small localized area was discovered during Scoping scans. The small localized area was less than one square foot and indicated gross beta/gamma result of 7,323 dpm/100 cm^2, which may have exceeded the DCGL. The area was decontaminated four separate times; however, no additional activity was removed after the third remedial activity. Total activity post-remediation was 1,464 dpm/100 cm^2 and ALARA. Routine and task specific contamination surveys were included in the historical review of the license and there were no indications of contamination levels over the DCGLs for release of the impacted areas included in this decommissioning survey.

2.6 Prior On-site Burials

There is no record of any on-site burials at RAW Room #1.

3 FACILITY DESCRIPTION

Chamblee Building 1 was constructed in the 1940's as part of World War II mobilization. With the addition of the RAW Room #1 enclosure, the current total building footprint is 850 ft^2, with a covered loading dock adding an additional 275 ft^2.

3.1 Lower Level

At the time of its original construction, the site of Building 1 was partially excavated to create a 22' x 9' basement level, with a 10-ft high ceiling at the east end of the building Appendix A **Figure 6 - Chamblee Building 1 East Elevation**. However this Lower Level was left entirely unfinished, with no lighting and no ventilation. A variety of plumbing pipes were left exposed within the space. Initially it was used as a medical waste incinerator for the adjacent military hospital. The incinerator was never utilized to process radioactive materials or radioactive waste. In the early days, the Lower Level provided access to the incinerator pit and the ash dumps, which occupy most of this area. Today it is entirely abandoned due to its dilapidated condition. There is a paved court outside the door to the Lower Level, bounded on one side by the building and on two other sides by concrete retaining walls. The third side, facing the building to the east, is bounded by a later reinforced-earth retaining wall installed as part of a temporary access route for the construction of Building 107. On the north side of the building there is a narrow areaway open to the aforesaid court. In 1981, the upper level was transformed into a chemical waste storage facility and the lower level was abandoned.

3.2 Upper Level

The construction primarily consists of a concrete floor supported by reinforced concrete beams over a crawl space, exterior masonry walls, and a gable wooden roof on steel purlins, covered with asphalt shingles. The assemblies also include a metal enclosure Appendix A **Figure 4 - Chamblee Building 1 North Elevation**, and a small wood framed building extension Appendix A **Figure 5 - Chamblee Building 1 Office Space and Utility Closet viewed from the Southeast**. The exterior doors are hollow metal, except one wood door at the Lower Level. Most windows along the perimeter walls have been removed, and their openings filled solid with concrete masonry or with wood for what would be an interior wall, during building renovations in 1981. Today, only two windows remain, both of which are uninsulated and in disrepair. Several electric space heaters, mounted to the wall or ceiling, keptthe building interior warm during the winter months. A wall air-conditioning unit providedcool air to the chemical waste storage in the summer. However, there is a lack of cooling equipment inside the metal enclosure where the radioactive waste is stored. This room has no cooling or ventilation equipment, except 4 louvers on the perimeter walls – which are not insulated. Electrical services are being fed from a single phase, 120/240 V, 200 amp main breaker panelboard located inside Building 1, which is in turn fed from a transformer between Building 1 and Building 107. There is domestic water supply and sanitary drain serving both the building proper and the RAW Room #1 enclosure.

In 1986, a metal enclosure, RAW Room #1, was added to the northwest corner to provide a space for radioactive waste storage. RAW Room #1 is characterized by three metal walls with a fourth wall of concrete. The smooth concrete floor has dimensions of 8' x 17' = 136 ft^2, according to information provided by the CDC RSO. Descriptions and dimensions are provided in **Table 4- Building 1 RAW Room #1 Description** below. On the Upper Level, the Chemical Waste Storage room occupied approximately 480 ft^2 in the central portion of the building. On the south side of the Upper Level, there was a small office measuring roughly 7'x12', and a utility room of similar size. The office was generally occupied only when workers processed waste materials.

Table 3-1- Building 1 Room 1 Description

Radiation Officer	Building	Rm	H Area
Paul Simpson	Building 1	RAW Room #1	(8'x 15' = 120 ft^2)
Narvaez Stinson	Building 1	RAW Room #1	(8' x 17'= 136 ft^2)
N/A Actual Field Measurements	Building 1	RAW Room #1	(9'9"x19'10'=194 ft^2)

There are no chemical and/or radiological fume hoods in RAW Room #1. Disposal of liquid radioactive waste via the sewer was generally performed in other laboratories and was limited to small total activities or radionuclides that had decayed at least 10 half-lives. There are no in-house vacuum or ventilation systems.

3.3 Ownership

The facility is currently owned by the CDC. The facility will be demolished after unrestricted release.

3.4 Population Distribution

Not Applicable – all impacted areas indoors.

3.5 Current/Future Land Use

Not applicable – all impacted areas indoors.

3.6 Meteorology and Climatology

Not applicable – all impacted areas indoors.

3.7 Geology and Seismology

Not applicable – all impacted areas indoors.

3.8 Surface Water Hydrology

Not applicable – all impacted areas indoors.

3.9 Ground Water Hydrology

Not applicable – all impacted areas indoors.

3.10 Natural Resources

Not applicable – all impacted areas indoors.

4 RADIOLOGICAL STATUS OF THE FACILITY

RAW Room #1, a metal enclosure provided for radioactive waste storage, is located at the CDC Chamblee campus at 4770 Buford Highway, Atlanta, GA 30341. Prior to Philotechnics initial Scoping surveys of the facility on March 10 & 11, 2014, the room was vacated, any unneeded and potentially contaminated items were surveyed and all items were found to be free of any residual contamination. There are no chemical and/or radiological fume hoods, no in-house vacuum or ventilation systems in the enclosure, but there was a small shallow sink. The sink was removed and disposed of. Diposal of liquid radioactive waste via the sewer at RAW Room #1 may not have occurred, and on campus was generally limited to small total activities or radionuclides that had decayed at least 10 half-lives. Only one area of elevated actvity was identified as discussed in **Section 4.1 Contaminated Structures** below. The area identified was an approximate 1 ft^2 area on the concrete floor with gross total activity levels of 7,323 disintegrations per minute (dpm)/100 cm^2 beta/gamma contamination as indicated by direct/static measurement. **Scoping survey results are provided in Table 4-1 – RAW Room #1 Floors and Lower Walls Results and Table 4-2 – RAW Room #1 Ceiling and Upper Walls Results below, and survey unit location maps are provided in Figure 1 – RAW Room #1 Floor and Lower Walls and Figure 2 – RAW Room #1 Ceiling and Upper Walls below:**

Table 4-1 – RAW Room #1 Floors and Lower Walls Results

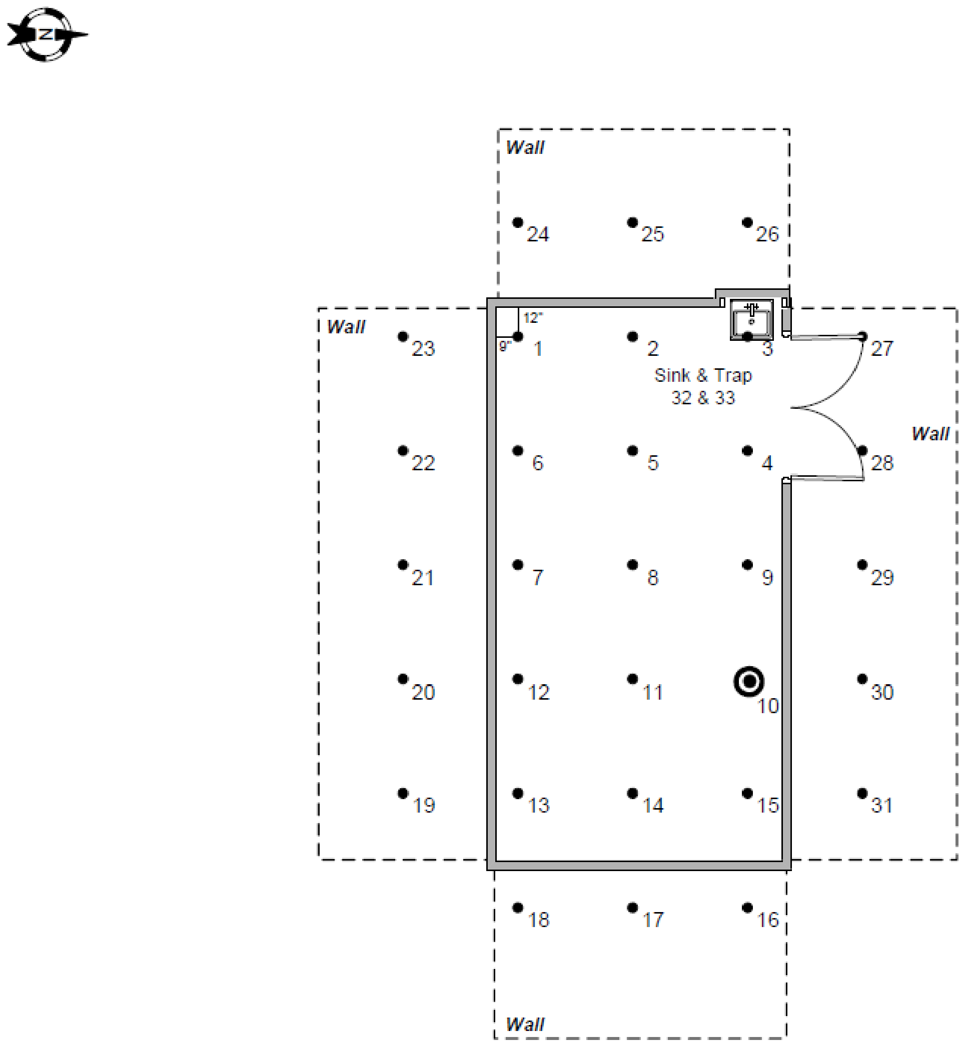
Result Description	Alpha dpm/100cm^2	Beta dpm/100cm^2	Gamma dpm/100cm^2
Average	24	1,720	757
Standard Deviation	19	1,765	625
Minimum	-14	-668	-36
Maximum	68	6,406	1,810

Table 4-2 – RAW Room #1 Ceiling and Upper Walls Results

Result Description	Alpha dpm/100cm^2	Beta dpm/100cm^2	Gamma dpm/100cm^2
Average	11	661	354
Standard Deviation	16	953	574
Minimum	-22	-316	-501
Maximum	35	3,331	1,616

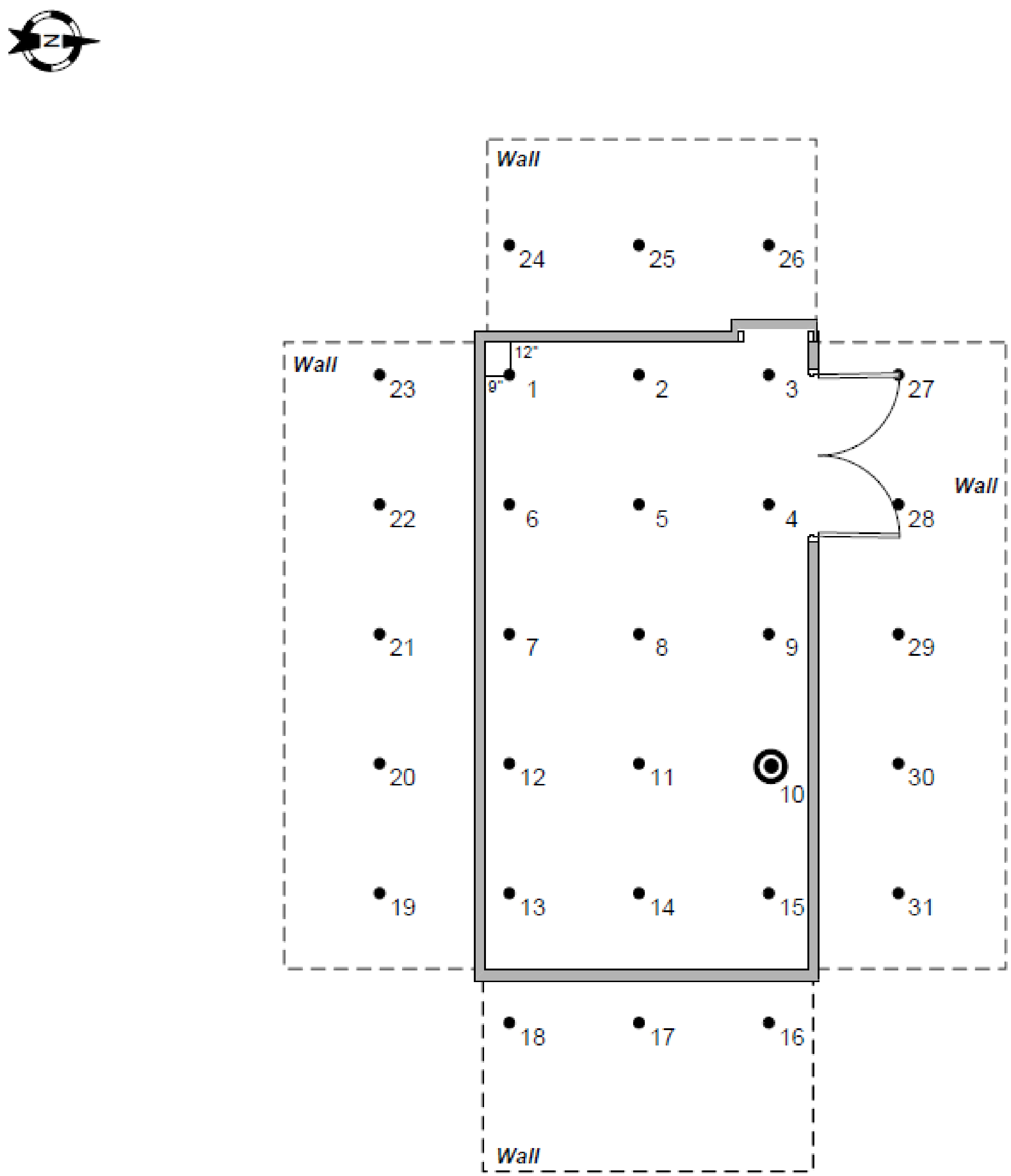
Note: Total instrument efficiency used for all alpha and beta measurements for floors, ceilings, lower and upper walls were performed in accordance with ISO-7503-1 using a surface efficeincy of 0.25 ($E_{Total} = E_{Instrument} * E_{Surface}$).

Figure 1 – RAW Room #1 Floor and Lower Walls



Typical Spacing = 1.22 m (4')
Area ~ 18 m² (194 ft²)
◉ - Random Starting Point

Figure 2 – RAW Room #1 Ceiling and Upper Walls



Typical Spacing = 1.22 m (4')
Area ~ 18 m² (194 ft²)
- Random Starting Point

SECTION 4.0 – RADIOLOGICAL STATUS OF THE FACILITY

4.1 Contaminated Structures

One contaminated area measuring approximately 1 ft^2 on the concrete floor was found to have gross total activity levels of 7,323 dpm/100 cm^2 beta/gamma as indicated by direct static measurement. Following decontamination of that area to a fixed level of 1,464 dpm/100 cm^2, the area was decontaminated 4 times, but no further activity was removed after the 3rd procedure (ALARA). Large area wipes were obtained in that area and the surrounding area. The residual fixed contamination was mwasured at 1,464 dpm/100cm2, as addressed above.

4.2 Contaminated Systems and Equipment

No contaminated systems or equipment were identified.

4.3 Surface Soil Contamination

Not applicable – all impacted areas indoors.

4.4 Subsurface Soil Contamination

Not applicable – all impacted areas indoors.

4.5 Surface Water

Not applicable – all impacted areas indoors.

4.6 Ground Water

Not applicable – all impacted areas indoors.

5 DERIVED CONCENTRATION GUIDELINE LEVEL DEVELOPMENT

The Derived Concentration Guideline Level (DCGL) is the radionuclide-specific surface area concentration that could result in a dose equal to the release criterion for unrestricted use specified in 10 CFR 20.1402. The building structural surfaces DCGLs for this project were developed utilizing a RESRAD-BUILD Dose Model. There are no impacted outdoor areas. Most default parameters were accepted; however, site specific parameter values were used for some critical parameters where compelling reasons existed.

5.1 Dose Model

Dose modeling was performed to develop site specific DCGLs for unrestricted release of building structural surfaces. Because the purpose of the surveys was to release a single room from radiological controls, only residual surface radioactivity was considered and there are no impacted outdoor areas. However, Philotechnics will perform additional Scoping of the travel path, loading dock and crawlspace area to verify this assumption. User's Manual for RESRAD-BUILD Version 3, Table 3.1, and NUREG/CR 6755, Table 4.1 were used where approropriate to assign site-specific building parameters. Resrad-BUILD was developed at Argonne National Laboratory and is recognized by the U.S. Nuclear Regulatory Commission as a tool for estimating annual doses to a member of the critical group.

The radiological release criteria of 10 CFR 20 Subpart E for unrestricted use are used for decommissioning this facility. Specifically, the facility will be surveyed in accordance with the guidance contained in MARSSIM to demonstrate compliance with the criteria of 10 CFR 20.1402, "Radiological Criteria for Unrestricted Use." The criteria are that residual radioactivity results in a TEDE to an average member of the critical group that does not exceed 25 mrem per year and that the residual radioactivity has been reduced to levels that are as low as reasonably achievable (ALARA).

A site specific dose model was used **primarily because DandD does not include dose modeling for some of the ROC present at the CDC.**

Scoping surveys were performed at the Chamblee, Georgia campus of the U.S. Centers for Disease Control in March 2014. The purpose of the surveys was to attempt to quantify and bound the site specific radiological status of the facility. Because the purpose of the surveys is to release a single room from radiological controls, only residual surface radioactivity was considered.

In order to develop site-specific DCGLs, a RESRAD-BUILD model was run after surveys were completed. This paper documents the process, the modeling and assumptions used, and the conclusion drawn.

Typically, RESRAD-BUILD is run after the Scoping of the site but before final decontamination and final status surveys. In such cases, the mixture and relative abundances of radionuclides present are known. All radionuclides, then, can be entered into a single model using the highest contamination levels. RESRAD-Build then calculates the expected dose to a member of the critical group at the present time and in the future.

Even though the RESRAD-BUILD was run after the surveys were completed, the relative abundances of radionuclides present in RAW Room #1 were not known. The model was run multiple times—once for each radionuclide present. The surface contamination level for each radionuclide to deliver a projected dose of 25 mrem was calculated. All alpha/beta/gamma activity measured was compared to the lowest alpha/beta/gamma limit determined by RESRAD-BUILD to assign a alpha/beta/gamma DCGL. It is important to note, these are conservative assumptions.

5.2 Determination of Nuclides of Concern

The relative abundances of radionuclides present in RAW Room #1 are not known. Nuclides of concern (NOC) and impacted rooms were determined by the following process (a brief overview is provided below, followed by a detailed description):

- CDC RSO reviews of limited nuclide receipt records for RAW Room #1.
- Exclude receipts of non-dispersible and gaseous forms.
- Decay-correct receipts.
- Determine the resulting surface activity concentration in dpm/100cm2.
- Determine site-specific DCGLs using RESRAD-BUILD version 3.5
- Multiple runs of the model

The model was run independently for each NOC in order to determine the limiting radionuclide for each decay mode: alpha, beta, gamma (electron capture), and low-energy beta. The limiting radionuclides were determined to be Th-232 plus decay products, Co-60, Mn-54, and tritium. After running uncertainty, the DCGL corresponding to 25 mrem/year was determined for each limiting radionuclide. Most default parameter values of the scenario were accepted. However, site-specific parameter values were used for some critical parameters where there are compelling reasons to justify a site-specific value. DCGLs were derived based on the highest 90th percentile dose from the probability distributions of each of the evaluation times.

5.2.1 Radionuclides evaluated:

Table 5-1 – RESRAD-BUILD Filtering Criterion below lists all the radionuclides considered and the surface contamination level of each that produces a dose of ≤ 25 mrem/year. Contamination limits shown are for the parent radionuclide only. Decay of parent and ingrowth of daughter activity is included in all dose calculations.

Table 5-1 – RESRAD-BUILD Filtering Criterion and Equivalent Surface Contamination Limit

Nuclide	Half-life (years)	Predominant Emissions	Equivalent Surface Contamination Limit [1] (dpm/100 cm^2)
H-3	1.2E+01	Beta	$1.0x10^6$
C-14	5.7E+.3	Beta	10,000
Na-22	2.6E+00	Beta	10,000
Mn-54	8.6E-01	Gamma (ε)	40,000
Co-57	7.4E-01	Beta	10,000
Co-60	5.3E+00	Beta/Gamma	10,000
Zn-65	6.78E-01	Gamma (ε)	40,000
Sr-90	2.9E+01	Beta	10,000
Cd-109	1.3E+00	Beta	10,000
Cs-134	2.0E+00	Beta	10,000
Cs-137	3.0E+01	Beta/Gamma	10,000
Ba-133	1.0+01	Gamma (ε)	40,000
Ce-139	3.8E-01	Beta	10,000
Eu-152	1.3E+01	Beta	10,000
Tl-204	3.8E+00	Beta	10,000
Po-209	1.1E+02	Alpha	150
Po-210	3.8E-01	Alpha	150
Ra-226	1.6E+03	Alpha	150
Th-232	1.4E+10	Alpha	150
U-233	1.56E+05	Alpha	150
U-235	7.1E+08	Alpha	150
U-236	2.4E+07	Alpha	150
U-238	4.5E+09	Alpha	150
Pu-238	8.6E+01	Alpha	150
Pu-239	2.4E+04	Alpha	150
Pu-240	6.6E+03	Alpha	150
Pu-242	3.8E+05	Alpha	150
Am-241	4.3E+02	Alpha	150
Am-243	7.9E+03	Alpha	150

[1] Equivalent Surface Contamination Limit is per radiation emission type individually and does not account for the use of Unity/Sum of Fractions. Limits were based on the use of most limiting radionuclide per emission type for alpha, beta, gamma, and low energy beta as specified in Table 5-7 RESRAD-BUILD Limiting Radionuclides. Justification is provided in Section 5.3 DCGL Development, and Table 5-2 RESRAD-BUILD Parameters and Table 5-3 RESRAD-BUILD Dose Details for Radionuclides.

5.3 DCGL Development

The DCGL is the radionuclide-specific surface activity concentration that could result in a dose equal to the release criterion. $DCGL_w$ is the concentration limit if the residual activity is evenly distributed over a large area. In the case of non uniform contamination, MARSSIM allows for evaluation of higher levels of activity over small areas using the $DCGL_{EMC}$. Due to the radiological cleanliness of the facility relative to the DCGLs, the desire to maintain simplicity of the FSS, and to assist in achieving ALARA goals, the $DCGL_w$ is used as a maximum value and small areas of elevated activity are not considered in this survey design. Those aras will be decontaminated to levels that are less than the DCGL and ALARA.

Site-specific dose modeling was performed, not because of the complexity of the site, but because nuclides were received that are not supported by the DandD dose model and because of excessive conservatism in the DandD model for some alpha emitters. As such, the building occupancy scenario was modeled using RESRAD BUILD, Version 3.4 to determine site-specific DCGLs. The goal was to develop a simple, conservative model for ease of review and implementation. Higher criteria could be obtained by refining critical parameters, but the effort required for justification would not be worthwhile. Some critical parameters have a significant amount of uncertainty. This uncertainty is offset by conservatism of the site conceptual model. Accepting extra conservatism has little impact on schedule or budget due to cleanliness of site. Conservatism is a common theme throughout selection of site-specific parameter values and development of DCGLs. This conservatism is used frequently to offset uncertainty such that qualitative statements may be used to justify site-specific parameter values.

5.3.1 RAW Room #1 Model Description

Room size is 18 m^2. Dimensions are 5.6 m x 3.2 m with a ceiling height of 4 m. Only natural ventilation is assumed.

5.3.1.1 Individual Radionuclide-Specific Trials

RESRAD-BUILD contains a number of default parameters, which are described in **Table 5-2 – RESRAD-BUILD Parameters** below. The user may accept default values or replace them with more realistic values to provide an accurate depiction of the building design characteristics, assumed future use and occupancy, radioactive contamination levels and behavior. Parameters that apply to all radionuclides are described below.

Table 5-2 – RESRAD-BUILD Parameters

PARAMETER DESCRIPTION	VALUE (s) Selected
Exposure Duration (days) – The period of time over which annual dose is integrated.	365
Indoor Fraction – The fraction of the receptor's time that is spent inside the room. This was conservatively assumed to be 2000 hours per year, such that the entire work year (40 hours/week for 50 weeks) is spent inside the room. A standard year is 8760 hours.	0.23
Number of Rooms	1
Deposition Velocity	Default value of 0.01 is used.
Receptor Time Fraction – The amount of time a receptor is in a given location within the room.	1
Receptor breathing rate	18 m^3d^{-1}
Receptor ingestion rate – Value taken from User Manual for Resrad-BUILD Version 3, Table 3.1.	1.12×10^{-4} m^2h^{-1}
Airborne Fraction – Value taken from User Manual for Resrad-BUILD Version 3, Table 3.1.	0.357
Direct Ingestion Rate – Value taken from User Manual for Resrad-BUILD Version 3, Table 3.1 and NUREG/CR 6755, Table 4.1.	3.06×10^{-6}
Source lifetime – For all radionuclides except H-3 (tritium), value taken from User Manual for Resrad-BUILD Version 3, Table 3.1. Tritium is assumed to have a lifetime of one year, and delivers all dose to the individual during that year.	10,000 days (365 days for tritium)
Resuspension Rate – Numerous publications estimate resuspension rate. The conservative value chosen is taken from User Manual for Resrad-BUILD Version 3, Table J-8.	Beta emitters: $1.1\text{x}10^{-5}$ s^{-1} Alpha emitters: $3.7\text{x}10^{-6}$ s^{-1}
Direct ingestion rate – Value taken from User Manual for Resrad-BUILD Version 3, Table 3.1.	3.06×10^{-6} s^{-1} (0 for tritium)
Removable Fraction – Value taken from User Manual for Resrad-BUILD Version 3, Table 3.1. Value is supported by scoping/Scoping survey results.	0.1 (For tritium, fraction is 1.)
Airborne Fraction – Value taken from User Manual for Resrad-BUILD Version 3, Table 3.1.	0.357 (For tritium, fraction is 1)

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Table 5-3 – RESRAD-BUILD Dose Details for Radionuclides

H-3						
Uniform contamination level is 1.8 x 10^8 dpm/100 cm^2.						
Time, years	0	1	3	10	30	100
Dose, mrem	24.9	0	0	0	0	0

C-14										
Uniform contamination level is 3.5 x 10^6 dpm/100 cm2.										
Time, years	0	1	3	10	30	100	300	1000	3000	10000
Dose, mrem	23.7	22.8	21.0	14.9	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1

Na-22						
Uniform contamination level is 2.0 x 10^4 dpm/100 cm^2.						
Time, years	0	1	3	10	30	100
Dose, mrem	23.9	18.2	10.6	1.6	<0.1	<0.1

Mn-54					
Uniform contamination level is 6.0 x 10^4 dpm/100 cm^2.					
Time, years	0	1	3	10	30
Dose, mrem	24.2	10.7	2.1	<0.1	<0.1

Co-57						
Uniform contamination level is 5.0 x 10^4 dpm/100 cm^2.						
Time, years	0	1	3	10	30	100
Dose, mrem	2.5	1.0	0.1	<0.1	<0.1	<0.1

Co-60						
Uniform contamination level is 1.5 x 10^4 dpm/100 cm^2.						
Time, years	0	1	3	10	30	100
Dose, mrem	24.5	21.3	16.2	6.22	0.404	<0.1

Zn-65					
Uniform contamination level is 8.0 x 10^4 dpm/100 cm^2.					
Time, years	0	1	3	10	30
Dose, mrem	11.6	4.08	0.503	<0.1	<0.1

Sr-90								
Uniform contamination level is 4.8 x 10^4 dpm/100 cm^2.								
Time, years	0	1	3	10	30	100	300	1000
Dose, mrem	24.7	23.3	20.6	12.7	<0.1	<0.1	<0.1	<0.1

Cd-109								
Uniform contamination level is 5.0 x 10^4 dpm/100 cm^2.								
Time, years	0	1	3	10	30	100	300	1000
Dose, mrem	2.7	1.5	0.5	<0.1	<0.1	<0.1	<0.1	<0.1

I-125				
Uniform contamination level is 4.2 x 10^5 dpm/100 cm^2.				
Time, years	0	1	3	10
Dose, mrem	22.9	0.336	<0.1	<0.1

Cs-134								
Uniform contamination level is 2.0 x 10^4 dpm/100 cm^2.								
Time, years	0	1	3	10	30	100	300	1000
Dose, mrem	19.8	14.0	7.0	0.6	<0.1	<0.1	<0.1	<0.1

Cs-137								
Uniform contamination level is 4.5 x 10^4 dpm/100 cm^2.								
Time, years	0	1	3	10	30	100	300	1000
Dose, mrem	24.6	23.8	22.1	16.9	7.88	1.56	<0.1	<0.1

Ba-133								
Uniform contamination level is 2.0 x 10^4 dpm/100 cm^2.								
Time, years	0	1	3	10	30	100	300	1000
Dose, mrem	17.7	16.0	13.0	6.0	<0.1	<0.1	<0.1	<0.1

Ce-139								
Uniform contamination level is 5.0 x 10^4 dpm/100 cm^2.								
Time, years	0	1	3	10	30	100	300	1000
Dose, mrem	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1

Eu-152								
Uniform contamination level is 2.0 x 10^4 dpm/100 cm^2.								
Time, years	0	1	3	10	30	100	300	1000
Dose, mrem	14.1	13.3	11.9	8.0	2.6	<0.1	<0.1	<0.1

Tl-204								
Uniform contamination level is 5.0 x 10^4 dpm/100 cm^2.								
Time, years	0	1	3	10	30	100	300	1000
Dose, mrem	0.5	0.4	0.2	<0.1	<0.1	<0.1	<0.1	<0.1

Po-210				
Uniform contamination level is 8.0 x 10^3 dpm/100 cm^2.				
Time, years	0	1	3	10
Dose, mrem	23.0	3.6	<0.1	<0.1

Ra-226								
Uniform contamination level is 5.0 x 10^2 dpm/100 cm^2.								
Time, years	0	1	3	10	30	100	300	1000
Dose, mrem	2.8	3.0	3.1	3.3	0.6	0.6	0.5	0.4

Th-232 (Maximum dose of 25.3 mrem occurs in years 12, 13, 14, and 15.										
Uniform contamination level is 4.7 x 10^2 dpm/100 cm^2.										
Time, years	0	1	3	10	30	100	300	1000	3000	10000
Dose, mrem	20.2	20.6	21.6	24.1	3.33	3.45	3.45	3.45	3.45	3.45

U-233										
Uniform contamination level is 7.2 x 10^3 dpm/100 cm^2.										
Time, years	0	1	3	10	30	100	300	1000	3000	10000
Dose, mrem	24.2	24.0	23.7	22.3	<0.1	<0.1	<0.1	0.1	0.3	0.8

U-234										
Uniform contamination level is 5.0 x 10^3 dpm/100 cm^2.										
Time, years	0	1	3	10	30	100	300	1000	3000	10000
Dose, mrem	16.5	16.3	16.0	14.9	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1

U-235										
Uniform contamination level is 7.5 x 10^3 dpm/100 cm^2.										
Time, years	0	1	3	10	30	100	300	1000	3000	10000
Dose, mrem	23.9	23.7	23.3	21.8	0.7	0.7	0.8	0.8	0.9	1.1

U-236								
Uniform contamination level is 5.0 x 10^3 dpm/100 cm^2.								
Time, years	0	1	3	10	30	100	300	1000
Dose, mrem	15.6	15.4	15.2	14.1	<0.1	<0.1	<0.1	<0.1

U-238										
Uniform contamination level is 8.0 x 10^3 dpm/100 cm^2.										
Time, years	0	1	3	10	30	100	300	1000	3000	10000
Dose, mrem	24.1	23.8	23.4	21.7	0.2	0.2	0.2	0.2	0.2	0.2

Pu-238										
Uniform contamination level is 9.5 x 10^2 dpm/100 cm^2.										
Time, years	0	1	3	10	30	100	300	1000	3000	10000
Dose, mrem	16.6	16.1	15.3	12.1	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1

Pu-239										
Uniform contamination level is 1.1 x 10^3 dpm/100 cm^2.										
Time, years	0	1	3	10	30	100	300	1000	3000	10000
Dose, mrem	22.2	21.8	20.8	17.6	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1

Pu-240										
Uniform contamination level is 1.3 x 10^3 dpm/100 cm2.										
Time, years	0	1	3	10	30	100	300	1000	3000	10000
Dose, mrem	23.8	23.3	22.3	18.8	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1

Pu-242										
Uniform contamination level is 1.3 x 10^3 dpm/100 cm2.										
Time, years	0	1	3	10	30	100	300	1000	3000	10000
Dose, mrem	22.7	22.2	21.3	18.0	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1

Am-241										
Uniform contamination level is 1.1 x 10^3 dpm/100 cm^2.										
Time, years	0	1	3	10	30	100	300	1000	3000	10000
Dose, mrem	22.9	22.4	21.4	17.9	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1

Am-243										
Uniform contamination level is 1.1 x 10^3 dpm/100 cm^2.										
Time, years	0	1	3	10	30	100	300	1000	3000	10000
Dose, mrem	20.8	20.4	19.5	16.5	0.1	0.1	0.1	0.1	0.1	<0.1

Po-209 is not supported by either DandD or Resrad-BUILD, and was therefore not analyzed. A review of decay energy, half-life, and decay products indicated its dose potential would be much less than Th-232. Federal Guidance Reports 11 and 12 were consulted in an effort to compare Po-209 with other alpha-emitting radionuclides using dose tables. Th-232 remains the limiting alpha emitter.

5.3.1.2 Uncertainty

Th-232 was run using Deposition Velocity values of 1.0x10-2, 1.0x10-4, and 1.1x10-6. There was no effect on the final dose.

Th-232 was run again using Resuspension Rate values of 1.3x10-5, 3.7x10-6, 4.7x10-7, and 1.0x10-9. Again, there was no effect on the final dose.

5.3.2 Uncertainty and Selection of Final Values

Because deposition velocity and resuspension rate had little to no effect on the final outcome, a single uncertainty trial was run with uncertainty analyses on Breathing Rate and Receptor Ingestion Rate. It was thought the values selected in the First Trial would estimate higher doses at the 90th percentile, so the contamination levels were altered. Input values for each radionuclide are provided in **Table 5-5 – RESRAD-BUILD Input Values for Each Radionuclide** below:

Table 5-4 – RESRAD-BUILD Input Values for Each Radionuclide

Radionuclide	Value in First Model (dpm/100 cm^2)	Conservatively chosen Value (dpm/100 cm^2)	Input (dpm/m^2)
Th-232	$4.7x10^2$	$3.0x10^2$	$3.0x10^4$
Co-60	$1.5x10^4$	$1.0x10^4$	$1.0x10^6$
Mn-54	$6.0x10^4$	$4.0x10^4$	$4.0x10^6$
H-3	$1.5x10^7$	$1.0x10^7$	$1.0x10^9$

Doses from Co-60, Mn-54, and H-3 were well below 25 mrem, even at the 90th percentile. Th-232, however, produced 24 mrem at the 50th percentile and 29 mrem at the 90th percentile, as shown in **Table 5-5 – RESRAD-BUILD Percentile Output Doses** below:

Table 5-5 – RESRAD-BUILD 90th Percentile Output Doses (mrem/year)

	Th-232	Co-60	Mn-54	H-3
Time = 0	2.51E+01	1.64E+01	1.61E+01	2.04E+00
Time = 1 y	2.55E+01	1.43E+01	7.14E+00	1.85E+00
Time = 3 y	2.65E+01	1.08E+01	1.40E+00	1.54E+00
Time = 10y	2.94E+01	4.16E+00	0	0

A final Resrad-BUILD trial was run for each of the four radionuclide; the values shown above were used for all, except Th-232 was again reduced to 150 dpm/100 cm2. Results are displayed in **Appendix C and Table 5-6 – RESRAD-BUILD Dose Details for Limiting Radionuclides** below:

Table 5-6 – RESRAD-BUILD Dose Details for Limiting Radionuclides

Th-232							
Uniform contamination level is 1.5 x 10^2 dpm/100 cm^2.							
Time, years	0	1	3	10	12	15	30
Dose, mrem	6.5	6.6	6.9	7.8	7.9	8.0	1.3

Co-60							
Uniform contamination level is 1.0 x 10^4 dpm/100 cm^2							
Time, years	0	1	3	10	12	15	30
Dose, mrem	16.3	14.2	10.8	4.2	3.2	2.1	0.3

Mn-54							
Uniform contamination level is 4.0 x 10^4 dpm/100 cm^2							
Time, years	0	1	3	10	12	15	30
Dose, mrem	16.1	7.2	1.4	0	0	0	0

H-3							
Uniform contamination level is 1.0 x 10^7 dpm/100 cm^2							
Time, years	0	1	3	10	12	15	30
Dose, mrem	18.8	2.5	0.5	0	0	0	0

5.3.2.1 RESRAD-BUILD Limiting Nuclides

The limits for the FSS and for the decommissioning Project are provided in **Table 5-7 – RESRAD-BUILD Limiting Radionuclides** below:

Table 5-7 – RESRAD-BUILD Limiting Radionuclides

Type of	Limiting	Gross DCGL,	Removable

Emission	Radionuclide	dpm100 cm^2	Activity, dpm/100 cm^2
Alpha	Th-232	150	15
Beta	Co-60	10,000	1000
Gamma	Mn-54	40,000	4000
Low-E beta[1]	H-3	$1.0x10^7$	$1.0x10^6$

5.3.2.2 Hard to Detect Nuclides

Hard-to-detect nuclides (H-3) cannot be adequately surveyed using direct field measurements and are typically evaluated by removable activity only as analyzed by liquid scintillation counting (LSC). The Scoping survey indicated all tritium smears were less than minimum detectable activity (MDA). A liquid scintillation counter is not available on the CDC site. For these reasons, tritium is not considered in this survey design. To verify this assumption, smears will be evaluated on the Philotechnics LSC in Oak Ridge, TN upon return from the Project site. Those results will be included in the Final Status Survey Report.

5.3.3 Unity Calculations

Unity will be applied to each sample location using the following equation to determine compliance.

$$\frac{CAlpha}{DCGLAlpha} + \frac{CBeta}{DCGLBeta} + \frac{CGamma}{DCGLGamma} < 1$$

Where:

C_{Alpha} = Gross alpha result in dpm/100cm ^{2}
C_{Beta} = Gross beta result in dpm/100cm 2
C_{Gamma} = Gross gamma result in dpm/100cm 2
$DCGL_{Alpha}$ = Gross alpha result in dpm/100cm 2
$DCGL_{Beta}$ = Gross beta result in dpm/100cm 2
$DCGL_{Gamma}$ = Gross gamma result in dpm/100cm 2

This method ensures that, regardless of the radionuclide distribution in a particular location, **the dose limit of 25 mrem per year will not be exceeded** as long as the sum of fractions shown above is less than 1.

6 ENVIRONMENTAL INFORMATION

This project will not affect quality of the human environment, will not affect species listed in Section 7 of the Endangered Species Act, and will not affect historic properties.

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7 ALARA ANALYSIS

NUREG 1757, Volume 2, Appendix N states in part: "For ALARA during decommissioning, all licensees should use typical good-practice efforts such as floor and wall washing, removal of readily removable radioactivity in buildings or in soil areas, and other good housekeeping practices. In addition, licensees should provide a description in the Final Status Survey Report (FSSR) of how these practices were employed to achieve the final activity levels." Based on the levels indicated during the Scoping survey, a quantitative ALARA analysis is not expected to be required.

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8 SURVEY INSTRUMENTATION

8.1 Instrument Calibration

Laboratory and portable field instruments are calibrated at least annually with National Institute of Standards and Technology (NIST) traceable sources, where feasible, and to radiation emission types and energies that will provide detection capabilities and sensitivities required for the nuclides of concern. Records of instrument calibration shall be included with the final status report.

8.2 Functional Checks

Functional checks will be performed at least daily when in use. The background, source check, and field measurement count times for radiation detection instrumentation will be specified by procedure to ensure measurements are statistically valid. Reference background readings will be taken in an adjoining non impacted area as part of the daily instrument check and compared with the acceptance range for instrument and site conditions. If an instrument fails a functional check, all data obtained with the instrument since the last satisfactory check will be evaluated for usability by the PM or designee and unusable data discarded.

8.3 Determination of Counting Times and Minimum Detectable Concentrations

Minimum counting times for background determinations and measurement of total and removable contamination will be chosen to provide a Minimum Detectable Concentration (MDC) that meets the criteria specified in this DP. MARSSIM equations relative to building surfaces have been modified to convert to units of dpm/100cm^2. Count times and scanning rates are determined using the following equations:

8.3.1 Static Counting

Static counting MDC at a 95% confidence level is calculated using the following equation, which is an expansion of NUREG 1507, “Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions”, Table 3.1 (Strom & Stansbury, 1992): Examples **Figure 3 - Beta Total Activity Example, Figure 4 - Alpha Total Activity Example, and Figure 5 - Gamma Total Activity Example** were preparing using background count rates, background and sample count times, total detector efficiencies (including surface efficiencies guidance provided in ISO-7503-1), and detector probe areas utilized in the scoping surveys, to ensure they are consistent with actual field conditions at the project site.

$$MDC_{static} = \frac{3 + 3.29\sqrt{B_r \cdot t_s \cdot (1 + \frac{t_s}{t_b})}}{t_s \cdot E_{tot} \cdot \frac{A}{100cm^2}}$$

Where:

MDC_{static} = minimum detectable concentration level in dpm/100cm^2
B_r = background count rate in counts per minute
t_b = background count time in minutes
t_s = sample count time in minutes
E_{tot} = total detector efficiency for radionuclide emission of interest (includes combination of instrument efficiency and surface efficiency from ISO-7503-1 $E_{Total} = (E_{Instrument} * E_{Surface})$
A = detector probe area in cm^2

Figure 3 - Beta Total Activity Example

$$MDC_{static} = \frac{3 + 3.29\sqrt{256 \cdot 1 \cdot (1 + \frac{1}{1})}}{1 \cdot (.2731 * .25) \cdot \frac{100}{100cm^2}}$$

$$MDC_{static} = \frac{3 + 3.29\sqrt{512}}{.0683}$$

$$MDC_{static} = \frac{77.44}{.0683}$$

$$MDC_{static} = 1{,}134 \text{dpm/100cm}^2$$

Figure 4 - Alpha Total Activity Example

$$MDC_{static} = \frac{3 + 3.29\sqrt{3 \cdot 2 \cdot (1 + \frac{2}{2})}}{2 \cdot (.3884 * .25) \cdot \frac{100}{100cm^2}}$$

$$MDC_{static} = \frac{3 + 3.29\sqrt{12}}{.1942}$$

$$MDC_{static} = \frac{14.40}{.1942}$$

$$MDC_{static} = 74 \text{dpm/100cm}^2$$

Figure 5 - Gamma Total Activity Example

$$MDC_{static} = \frac{3 + 3.29\sqrt{1407 \cdot 1 \cdot (1 + \frac{1}{1})}}{1 \cdot (.2362 * .50) \cdot \frac{100}{100cm^2}}$$

$$MDC_{static} = \frac{3 + 3.29\sqrt{2{,}814}}{.1181}$$

$$MDC_{static} = \frac{177.53}{.1181}$$

$$MDC_{static} = 1{,}503 d\text{pm}/100\text{cm}^2$$

8.3.2 Beta/Gamma Ratemeter Scanning

Scan MDC is determined based on the guidance described in MARSSIM, Revision 1, Section 6.7.2 – *Scanning Sensitivity* and Decommissioning Health Physics, Second Edition, **Section 9.3 – Scan MDC**. Scanning is performed to identify areas of elevated activity in the survey unit. The scan MDC depends on many of the same factors that influence the detection of contamination under static conditions: the level of the background radiation levels; the nature (type and energy of emissions) and relative distribution of potential contamination (point versus distributed source and depth of contamination; the intrinsic characteristics of the detector (efficiency, physical probe area, etc.); the desired level of confidence (type I and type II); and the surveyor's skill in recognizing an increase in the audible or display output of an instrument. If one assumes constant parameters for each of the above variables, with the exception of the specific radionuclide of interest, the scan MDC may be reduced to a function of the radionuclide alone. These calculations are provided in Section 6 of MARSSIM.

The scan MDCs are determined based upon site-specific background data from the scoping surveys, using the equations below.

The number of source counts required for a specific time interval is calculated by MARSSIM Equation 6-8:

$$s_i = d'\sqrt{b_i}$$

Where:

d' = The performance factor based on required true and false positives rates. It can be assumed that at the first scanning stage a high rate (95%) of correct detections is required, and that a correspondingly rate of false positives (60%) will be tolerated. From MARSSIM Table 6.5, the value representing the performance goal is 1.38.

b_i = The number of background counts in the observation interval

Assuming that the source remains under the detector for 1.385 seconds (e.g., i=1.385) and the background count rate is the site-specific background of 256 cpm for beta and 1,407 cpm for gamma, the value for b_i is then calculated:

$$b_i \text{ (counts)} = (\text{Background (cpm)}) \times (i \text{ (sec.)}) \times (1 \text{ min}/60 \text{ sec})$$

The scan minimum detectable count rate (MDCR) is then calculated using the number of source counts required for a specific time interval is calculated by MARSSIM Equation 6-8:

$$s_i = d'\sqrt{b_i}$$

MARSSIM Equation 6-9:

$$\text{MDCR (cpm)} = (d') \times (\sqrt{b_i} \text{ (counts)}) \times (60 \text{ sec}/1\text{min})$$

The $MDCR_{surveyor}$ is calculated assuming a surveyor efficiency (p) of 0.5 (see MARSSIM page 6-45):

$$MDCR_{SURVEYOR} \text{ (cpm)} = \text{MDCR (cpm)}/(\sqrt{p})$$

Using the above input parameters, the scan MDC necessary to yield the MDC is calculated using MARSSIM Equation 6-10 for structures and surfaces.

An example of the surface and structures scanning MDC at a 95 percent confidence level is calculated for Ludlum 2350-1 with a BP19DD probe for beta and Ludlum 2350-1 with a GP-13A probe for gamma using the following equation, which is a combination of MARSSIM Equations 6-8, 6-9 and 6-10:

$$MDC_{scan} = \frac{d'\sqrt{b_i}\left(\frac{60}{i}\right)}{\sqrt{p} \cdot \varepsilon_{tot} \cdot \frac{A}{100cm^2}}$$

Where

d'	=	minimum detectable concentration level in dpm/100 cm^2
b_i	=	background counts during the observation interval
i	=	observation interval
p	=	surveyor efficiency (0.5)
ε_{tot}	=	total detection efficiency for radionuclide emission of interest (includes combination of instrument and surface efficiencies)
A	=	active area of the detector in cm^2

8.3.2.1 Small Area Probe Beta Ratemeter Scanning MDC

Assuming that the source remains under the detector for 1.385 seconds (e.g., i=1.385) and the background count rate is the site-specific ambient background rate of 256, the value for b_i is then calculated for beta as:

$$b_i \text{ (counts)} = 256\text{cpm} \times 1.385 \text{ sec.} \times (1 \text{ min}/60 \text{ sec})$$

$$b_i = 5.9 \text{ counts}$$

The scan minimum detectable count rate (MDCR) is then calculated using the number of source counts required for a specific time interval is calculated by MARSSIM Equation 6-8:

$$s_i = d'\sqrt{b_i}$$

$$S_i = 1.38 \text{ x } \sqrt{5.9}\text{counts}$$

$$S_i = 3.35 \text{ counts}$$

The scan minimum detectable count rate (MDCR) is then calculated using MARSSIM Equation 6-9:

$$\text{MDCR (cpm)} = S_i \text{ x } (60/i)$$

$$\text{MDCR} = 3.35 \text{ counts x } (60/1.385)$$

$$\text{MDCR} = 145.13 \text{ cpm}$$

The $MDCR_{surveyor}$ is calculated assuming a surveyor efficiency (p) of 0.5 (see MARSSIM page 6-45):

$$MDCR_{SURVEYOR} \text{ (cpm)} = 145.13/(\sqrt{0.5})$$

$$MDCR_{SURVEYOR} \text{ (cpm)} = 102.62 \text{ cpm}$$

The scan MDC is then calculated using MARSSIM Equation 6-10:

$$\textit{Scan MDC} = \text{MDCR}/(\sqrt{p})*(\mathcal{E}_{tot})*(\textit{probe area}/100\text{cm}^2)$$

$$\textit{Scan MDC} = 145.13/(\sqrt{0.5})\text{x}(27.31\%*.25) \text{ x}(100 \text{ cm}^2/100\text{cm}^2)$$

$$\textit{Scan MDC} = 3{,}006 \text{ dpm}/100 \text{ cm}^2$$

Where

MDCR	=	minimum detectable count rate
$\mathcal{E}_{tot}$	=	Instrument efficiency ($\mathcal{E}_i$) x surface efficiency ($\mathcal{E}_s$) Per ISO-7503-1 1988 $\mathcal{E}_s$ = 0.25 for [beta-emitters (0,15 MeV < EBmax < 0,4 MeV) and alpha-emitters]
p	=	surveyor efficiency (0.5)

Utilizing the combination of MARSSIM Equations 6-8, 6-9 and 6-10 from above:

$$MDC_{scan} = \frac{d'\sqrt{b_i}\left(\frac{60}{i}\right)}{\sqrt{p}\cdot\varepsilon_{tot}\cdot\frac{A}{100cm^2}}$$

$$MDC_{scan} = 1.38*(\sqrt{5.9})*(60/1.385)/(\sqrt{0.5})*(27.31\%*.25)*(100 \text{ cm}^2/100\text{cm}^2)$$

$$MDC_{scan} = 3{,}006 \text{ dpm}/100 \text{ cm}^2$$

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8.3.2.2 Large Area Probe Beta Ratemeter Scanning MDC

Assuming that the source remains under the detector for 1.25 seconds (e.g., i=1.25) and the background count rate is the estimated typical site-specific ambient background rate of 2,400, the value for b_i is then calculated for beta as:

$$b_i \text{ (counts)} = 2{,}400\text{cpm} \times 1.25 \text{ sec.} \times (1 \text{ min}/60 \text{ sec})$$

$$b_i = 49.998 \text{ counts}$$

The scan minimum detectable count rate (MDCR) is then calculated using the number of source counts required for a specific time interval is calculated by MARSSIM Equation 6-8:

$$s_i = d'\sqrt{b_i}$$

$$S_i = 1.38 \times \sqrt{49.998\text{counts}}$$

$$S_i = 9.76 \text{ counts}$$

The scan minimum detectable count rate (MDCR) is then calculated using MARSSIM Equation 6-9:

$$\text{MDCR (cpm)} = S_i \times (60/i)$$

$$\text{MDCR} = 9.76 \text{ counts} \times (60/1.25)$$

$$\text{MDCR} = 468.48 \text{ cpm}$$

The $\text{MDCR}_{surveyor}$ is calculated assuming a surveyor efficiency (p) of 0.5 (see MARSSIM page 6-45):

$$\text{MDCR}_{SURVEYOR} \text{ (cpm)} = 468.48/(\sqrt{0.5})$$

$$\text{MDCR}_{SURVEYOR} \text{ (cpm)} = 662.53 \text{ cpm}$$

The scan MDC is then calculated using MARSSIM Equation 6-10:

$$\textit{Scan MDC} = \text{MDCR}/(\sqrt{p})*(\mathcal{E}_{tot})*(\textit{probe area}/100\text{cm}^2)$$

$$\textit{Scan MDC} = 468.48/(\sqrt{0.5})\text{x}(38.47\%*.25)\text{ x}(821 \text{ cm}^2/100\text{cm}^2)$$

$$\textit{Scan MDC} = 839 \text{ dpm}/100 \text{ cm}^2$$

Where

MDCR	=	**minimum detectable count rate**
$\mathcal{E}_{tot}$	=	**Instrument efficiency ($\mathcal{E}_i$) x surface efficiency ($\mathcal{E}_s$)** **Per ISO-7503-1 1988 $\mathcal{E}_s$ = 0.25 for [beta-emitters (0,15 MeV < EBmax < 0,4 MeV) and alpha-emitters]**
p	=	**surveyor efficiency (0.5)**

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Utilizing the combination of MARSSIM Equations 6-8, 6-9 and 6-10 from above:

$$MDC_{scan} = \frac{d'\sqrt{b_i}\left(\frac{60}{i}\right)}{\sqrt{p}\cdot\varepsilon_{tot}\cdot\frac{A}{100cm^2}}$$

***MDC_{scan}* = 1.38*(√49.998)*(60/1.25)/(√0.5)*(38.47%*.25)*(821 cm^2/100cm^2)**

***MDC_{scan}* =839 dpm/100 cm^2**

8.3.2.3 Gamma Ratemeter Scanning MDC

Assuming that the source remains under the detector for 1.385 seconds (e.g., i=1.385) and the background count rate is the site-specific ambient background rate of 256, the value for b_i is then calculated for gamma as:

b_i (counts) = 1,407cpm x 1.385 sec. x (1 min/60 sec)

b_i = 32.47 counts

The scan minimum detectable count rate (MDCR) is then calculated using the number of source counts required for a specific time interval is calculated by MARSSIM Equation 6-8:

$$s_i = d'\sqrt{b_i}$$

S_i = 1.38 x √32.47counts

S_i = 7.86 counts

The scan minimum detectable count rate (MDCR) is then calculated using MARSSIM Equation 6-9:

MDCR (cpm) = S_i x (60/i)

MDCR = 7.86 counts x (60/1.385)

MDCR = 340.66 cpm

The $MDCR_{surveyor}$ is calculated assuming a surveyor efficiency (p) of 0.5 (see MARSSIM page 6-45):

$MDCR_{SURVEYOR}$ (cpm) = 340.66/(√0.5)

$MDCR_{SURVEYOR}$ (cpm) = 240.88 cpm

The scan MDC is then calculated using MARSSIM Equation 6-10:

Scan MDC = MDCR/(√p)*($\mathcal{E}_{tot}$)*(*probe area*/100cm^2)

Scan MDC = 340.66/(√0.5)x(23.62%*.5) x(100 cm^2/100cm^2)

Scan MDC = 4,080 dpm/100 cm^2

Where

MDCR	=	minimum detectable count rate
$\mathcal{E}_{tot}$	=	Instrument efficiency ($\mathcal{E}_i$) x surface efficiency ($\mathcal{E}_s$) Per ISO-7503-1 1988 $\mathcal{E}_s$ = 0.25 for [beta-emitters (0,15 MeV < EBmax < 0,4 MeV) and alpha-emitters and 0.5 for > 0.4 MeV]
p	=	surveyor efficiency (0.5)

Utilizing the combination of MARSSIM Equations 6-8, 6-9 and 6-10 from above:

$$MDC_{scan} = \frac{d'\sqrt{b_i}\left(\frac{60}{i}\right)}{\sqrt{p}\cdot\varepsilon_{tot}\cdot\frac{A}{100cm^2}}$$

$$MDC_{scan} = 1.38*(\sqrt{32.47})*(60/1.385)/(\sqrt{0.5})*(23.62\%*.5)*(100\ cm^2/100cm^2)$$

$$MDC_{scan} = 4{,}080\ dpm/100\ cm^2$$

8.3.3 Alpha Ratemeter Scanning

Scanning for alpha emitters differs significantly from scanning for beta and gamma emitters in that the expected background response of most alpha detectors is very close to zero. The following covers scanning for alpha emitters. Since the time a contaminated area is under the probe varies and the background count rate of some alpha instruments is less than 1 cpm, it is not reasonable to determine a fixed MDC for scanning. Instead, it is more practical to determine the probability of detecting an area of contamination at a predetermined DCGL for given scan rates. For alpha survey instrumentation with backgrounds ranging from less than 1 to 3 cpm, a single count provides a surveyor sufficient cause to stop and investigate further. Assuming this to be true, the probability of detecting given levels of alpha surface contamination can be calculated by use of Poisson summation statistics. MARSSIM, section 6.7.2.2 and Appendix J, contain the guidance for scanning for alpha emitters having low release limits. MARSSIM provides derivations, formulas and probability concepts for alpha scanning in Appendix J.

8.3.3.1 Count Detection Porbability 100 cm^2 Probe

Alpha scan rates were calculated using the Poisson summation statistics and selected from the probability charts in Appendix J to achieve a 95% probability Given a known scan rate and a surface contamination release limit, the probability of detecting a single count while passing over the contaminated area is given using the following equation in **Figure 7 – Count Detection Probability Single Count Equation** below.

Figure 6 – Count Detection Probability Single Count Equation

$$P(n \geq 1) = 1 - e^{Ged/60v}$$

Where:

$P(n \geq 2)$ = Probability of observing a single count
G = Contamination activity (dpm)
E = Detector efficiency (4π)
D = Width of detector in direction of scan (cm)
v = Scan speed (cm/s)

Once a count is recorded and the guideline level of contamination is present, the surveyor should stop and wait until the probability of getting another count is at least 90 percent. This time interval can be calculated using the following equation in Figure 8 – Count Detection Probability Time Interval Equation below.

Figure 7 – Count Detection Probability Time Interval Equation

$$t = 13{,}800/CAE$$

Where:

t = Time period for static count(s)
C = Contamination guideline (dpm/100 cm^2)
A = Physical probe area (cm^2)
E = Detector efficiency (4π)

The probability of detecting a single count while passing over the contaminated area for 100 cm^2 probe is not possible due to the limitations on probe size, background and detector efficiency; therefore, only the larger (821 cm^2) gas-proportional detectors will be used for alpha scans.

8.3.3.2 Count Detection Porbability 821 cm^2 Probe

The larger (821 cm^2) gas-proportional detectors have an alpha background count rates on the order of 5 to 10 cpm, and a single count will not cause a surveyor to investigate further. A counting period long enough to establish that a single count indicates elevated contamination level woould be prohivitively inefficient. For these types of instruments, the surveyor usually will need to get at least two counts while passing over the source area before stopping for further investigation.

Assuming this to be a valid assumption, the probability of getting two or more counts is calculated using the following equation in **Figure 9 – Count Detection Probability Two or More Counts Equation** below.

Figure 8 – Count Detection Probability Two or More Counts Equation

$$P(n \geq 2) = 1 - P(n = 0) - P(n = 1)$$

$$= 1 - \left(1 + \frac{(GE+B)\,d}{60v}\right)\left(e^{-\frac{(GE+B)\,d}{60v}}\right)$$

Where:

$P(n \geq 2)$ = Probability of observing at least 2 counts
G = Contamination activity (dpm)
E = Detector efficiency (4π)
D = Width of detector in direction of scan (cm)
B = Background in dpm/100cm^2
v = Scan speed (cm/s)

G = (75*821)/100

G = 615.75

$$\mathbf{P\,(n \geq 2)} \quad = 1 - \left(1 + \frac{(615.75 * .21 + 9.5)15.9}{60 * 6.35}\right)\left(e^{-\frac{(615.75 * .21 + 9.5)\,15.9}{60 * 6.35}}\right)$$

P (n ≥ 2) = 97.93%

The scan rate to achieve a ≥ 95% probability of detection while passing over the contaminated area of 75 dpm/100 cm^2 is 2.5 inches/second. If the surveyor detects two counts while performing the alpha scan surveys, the surveyor will stop, acquire a timed count, and investigate to determine if an area of elevated activity exists, or if the error was erroneous.

8.3.4 100 cm^2 Smear Counting

Smear counting Minimum Detectable Concentration at a 95% confidence level is calculated using the following equation, which is NUREG 1507, "Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions", Table 3.1 (Strom & Stansbury, 1992):

$$MDC_{smear} = \frac{3 + 3.29\sqrt{B_r \cdot t_s \cdot (1 + \frac{t_s}{t_b})}}{t_s \cdot E}$$

Where:

MDC_{smear} = minimum detectable concentration level in dpm/smear
B_r = background count rate in counts per minute
t_b = background count time in minutes
t_s = sample count time in minutes
E = instrument 4π efficiency for radionuclide emission of interest

8.4 Efficiency Determination

Field instruments for determination of total surface activity by scanning and static measurements will have an efficiency determined by a licensed calibration facility using NIST traceable sources. In addition, ISO 7503-1, 1988 methods shall be used to determine field concentrations for final status data and calculation of resultant doses from residual radioactivity from beta emitters greater than 0.15 MeV (which excludes H-3). ISO 7503-1 recommends that a conservative surface efficiency of 0.25 be used for beta particles in the energy range of 150 keV to 400 keV (0.15 MeV < $E_{\beta max}$ < 0.4 MeV)and alpha emitters, and a surface efficiency of 0.50 for all beta emitters greater than 400 keV ($E_{\beta max}$ > 0.4 MeV). Philotechnics will use the recommended ISO-7503 conservative surface efficiencies for all beta particles (as C-14 is a ROC), and all alpha emitters, within the purvue of this survey, for both total and removable activity measurements. Radionuclides used for efficiency determination are:

Beta: Tc-99 and/or C-14; Alpha: Th-230 and/or Pu-239; Gamma: I-129

8.5 Instrumentation Specifications

The instrumentation used for decommissioning surveys are summarized in the tables below. The first table lists the standard features of each instrument such as probe size and efficiency. The second table lists the typical operational parameters such as scan rate, count time, and the associated MDC. Alternate or additional instrumentation with similar detection capabilities may be utilized as needed for survey requirements with RSO approval.

Table 8-1 - Instrument Specifications

Detector Model	Detector Type	Detector Area (cm^2)	Meter Model	Typical Total Efficiency (%)
43-93[2] Small Area Probe Alpha	Alpha Scintillation	100	Ludlum 2350-1	**9.7**
BP-19DD Small Area Probe Beta	Beta Scintillation	100	Ludlum 2350-1	**6.8**
43-37-1 Large Area Probe Alpha	Gas Flow Proportional	821	Ludlum 2350-1	**10.9**[3]**/20**[4]
43-37-1 Large Area Probe Beta	Gas Flow Proportional	821	Ludlum 2350-1	**9.6**
GP-13A Small Area Probe Gamma	Gamma Scintillation	100	Ludlum 2350-1	23 (I-125)
2200CA TriCarb (or Equivalent)	Liquid Scintillation	N/A	N/A	61 (H-3) 96 (C-14) Open
Protean	ZnS+Dual Phosphor	N/A	N/A	8 (Th-230) 6 (Tc-99)

[2] **Or equivalent, to include 43-89 or 43-68, with similar detector areas and efficiencies**.
[3] **Total Efficiency for Alpha Statics using 2π instrument efficiency (43.48%) and a 25% surface efficiency per ISO-7503-1**
[4] **Total Efficiency for Alpha Scans using 4π instrument efficiency per NUREG 1575, MARSSIM, Appendix J.**

Table 8-2 - Typical Operating Parameters and Sensitivities

Measurement Type	Detector Model	Scan Rate (in/s)	Count Time (s)	Bkg. Time (s)	Bkg. (cpm)	MDC/DCGL (dpm/100cm^2)	MDC Percent DCGL (%)
Surface Scans Small Area Probe Beta	BP-19DD	2.5	N/A	60	256	3,006 /10,000	30
Surface Scans Small Area Probe Gamma	GP-13A	2.5	N/A	60	1,407	4,080/40,000	10.20
Surface Scans Large Area Probe Alpha	43-37-1	2.5	N/A	60	9.5	75/150	50
Surface Scans Large Area Probe Beta	43-37-1	5	N/A	60	2,400	839/10,000	8.39
Total Surface Activity Small Area Probe Alpha	**43-93**[5]	N/A	120	120	3	74/150	49.33
Total Surface Activity Small Area Probe Beta	BP-19DD	N/A	60	60	256	1,134/10,000	11.34
Total Surface Activity Small Area Probe Gamma	GP-13A	N/A	60	60	1,407	1,503/40,000	3.75
Removable Activity	2200CA TriCarb	N/A	60	60	25 (H-3) 15 (C-14) Open	44 (H-3)/ 10000000 26 (C-14)/10000 Open/10000	<.01% For all three channels
Gross Alpha Removable Activity	Protean	N/A	900	3,600	0.1	6.3/15	42
Gross Beta Removable Activity	Protean	N/A	120	3,600	70	346/1,000	34.6

8.6 Minimum Detectable Concentration (MDC) Calculations

Philotechnics analytical sheets are included as Appendix D, which show calculations for the static MDC for the scintillation counter, static MDC and scanning MDC for hand-held instruments. The MDC's were calculated using the most conservative background values. These calculations follow the guidance in NUREG-1575 and NUREG-1507 and the information is used to verify the effectiveness of the instrumentation used in units of dpm/100 cm^2.

[5] **Or equivalent, to include 43-89 or 43-68, with similar detector areas and efficiencies**.

9 PLANNED DECOMMISSIONING ACTIVITIES

9.1 Radiological Scoping Surveys

On March 10 & March 11, 2014, Philotechnics completed a comprehensive wipe and meter survey in specified impacted areas, which included walls, floors and sinks. Survey maps depicting these areas are included as Appendix B.

Radiological Scoping was designed to identify areas of elevated activity that require remediation. Scoping consisted of scans and smears for building structural surfaces and smears for removable activity measurements on drain internal surfaces. Scoping surveys were designed to meet the same Data Quality Objectives (DQOs) as the Final Status Surveys (FSS) such that Scoping data can be used as final status data where possible.

Scoping included:

- 100% scans for alpha and beta emitters of all accessible areas, with some instrument detector overlap, of RAW Room #1 including heaters, ceiling, louvers and peripheral areas. Although scans for gamma emitters were also performed, due to insufficient instrument source check data, gamma scans will be repeated in their entirety.
- Total activity scans of the structural surfaces of the entire room based on the conservative initial survey unit classifications of Class 1 and required percentages
- Removable activity measurements of the structural surfaces of the entire building based on the conservative initial survey unit classifications of Class 1 and required percentages
- Removable activity measurements in the drain system, and
- Static measurements and smears at areas of elevated activity

9.1.1 Building Structural Surfaces

In order to identify locations of elevated activity, the building surfaces Scoping survey protocol consisted of performing scan surveys of 100% of all accessible surfaces, with judgmental smears and static measurements on areas with the highest probability areas for residual radioactivity.

9.1.2 Building Systems

The building systems Scoping survey protocol consisted of removable contamination measurements of internal surfaces of the drain system. 100% of accessible openings in the drain system were surveyed. Geometric configuration made direct measurements impossible. Philotechnics used convenient locations to obtain measurements where there is the highest probability of residual radioactivity, such as low flow areas and elbows where impingement of particulates could occur.

9.2 Decontamination/Dismantlement and Remedial Action Surveys

9.2.1 Decontamination/Dismantlement

Decontamination is the physical or chemical process of reducing and preventing the spread or potential exposure from contamination. Decontamination options include the use of commercially available materials and/or equipment that will effectively remove radioactive materials from surface areas so the contamination can be collected and properly disposed. Decontamination was required on a 1 ft^2 area on the floor with gross beta/gamma total activity of 7,323 dpm/100 cm^2 as indicated by static/direct measurement. The area was remediated by CDC personnel and the affected area was scanned again. Post-decontamination results indicated gross beta/gamma total activity <DCGL of 1,464 dpm/100 cm^2 as indicated by static/direct measurement and LAWs at or near background levels. These levels were ALARA, additionally; four separate remedial events were conducted. After three, no change in total activity was removed, so remediation was terminated. The survey results did not indicate the presence of any other levels of radioactive materials that would require decontamination. .

9.2.2 Remedial Action Surveys

Remedial action surveys consisted of scan surveys and direct measurements. These were conducted following remediation activities to establish the success or failure of the efforts to decontaminate the applicable area. Results of the survey were the decision basis for continued remediation. Remedial action surveys were designed to meet the objectives of the final status surveys.

10 PHILOTECHNICS MANAGEMENT ORGANIZATION

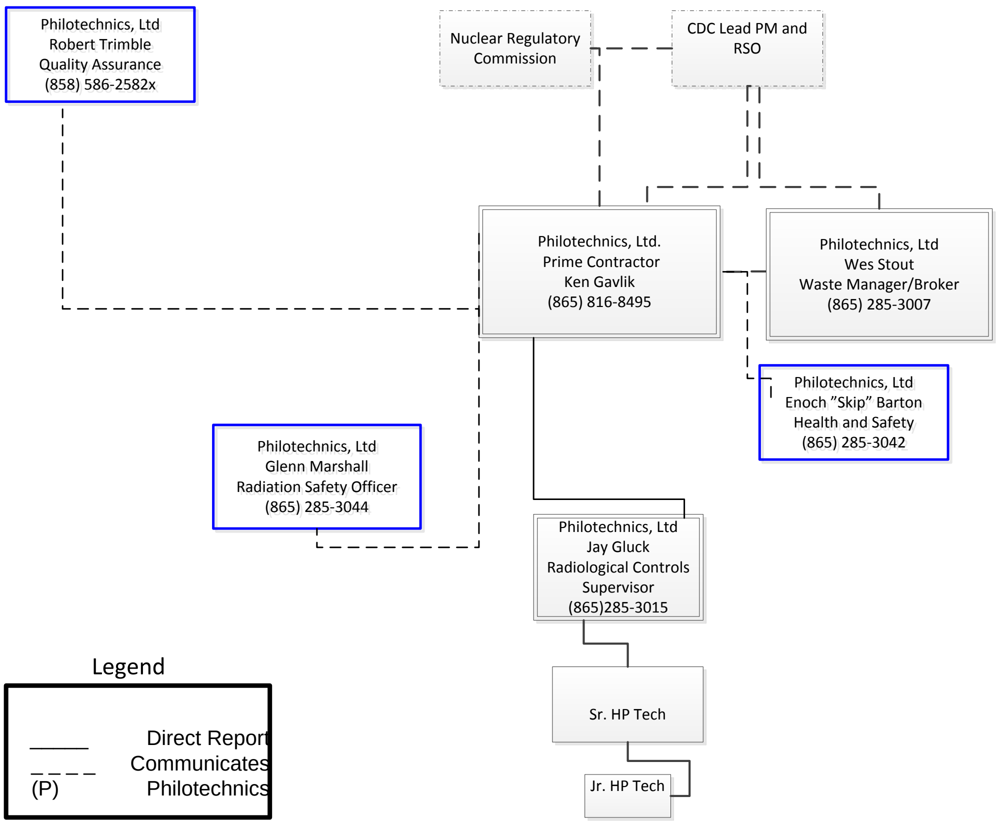
The following management structure will be utilized for administration and implementation of this plan.

Figure 10-1 – Team Experience on Similar Work

Name, Corporate Position Qualifications and Relevant Experience
Project Manager - Ken Gavlik, VP Radiological Services and RSO for State of North Dakota RML: A veteran of the U.S. Navy Nuclear Power Program with a B.S. in Nuclear Engineering Technology and Radiation Protection, an MBA, is MARSSIM certified, and is a member of the Conference of Radiation Control Program Directors and the NYC Radiological Advisory Committee. More than 18 years' experience in radiation protection and radiological services. Personally designed, planned and managed over 50 radiological services projects, including facility release for unrestricted use of facilities within the State of Pennsylvania, with many of the facilities released without question or comment from various regulatory authorities, Agreement States and the NRC. Projects include License Termination and release for unrestricted use of two of the largest R&D facilities in the US, containing in excess of two million square feet of impacted areas each.
Radiation Safety Officer - Glenn Marshall, CHP, RRPT, Corporate RSO: An experienced RSO for NRC and Agreement States. Over 30 years of experience in supervisory and management in applied health physics, licensing, procedure/program development, sealed source encapsulation, radioactive materials waste management, ALARA and dosimetry. A CHP, who has been certified by the ABHP since 2003, and the NRRPT since 1994, including serving on the panel of examiners. He was the Corporate RSO for over a decade of MARSSIM facility release projects.
Quality Assurance - Robert Trimble, Director of West Coast Operations and RSO of Philotechnics State of California RML: A B.S. in Physics and an M.S. in Radiological Health Physics. Over 19 years of practical experience in the comprehensive practice of health physics. Mr. Trimble has provided radiological oversight and engineering expertise for decommissioning projects in California in accordance with MARSSIM and CDPH criteria, releasing over 50 facilities for unrestricted use in the last 5 years. A comprehensive knowledge of NRC regulations and regulatory requirements, and has provided radiological oversight and engineering expertise for dozens of radiological services and waste management in accordance with EPA, DOT, NRC or Agreement State regulations. Provides health physics and environmental consultation services to clients including annual program reviews, dose assessments, chemical audits, and general support of their safety programs.
Waste Management - Wesley Stout, Director of Radiological Engineering and Waste Brokerage: A veteran of the U.S. Navy Nuclear Power Program with a B.S. Degree. Over 25 years of experience as a project manager with experiences in radiological D&D, industrial safety, and waste management. He is the Radiological lead for characterization of waste streams, identification of viable treatment/disposal alternatives and for federal client waste management technical support.

Additionally, the CDC management organization described in Section 11.0, provided relevant data and support, and made final decisions for the decommissioning effort:

Center for Disease Control and Prevention
RAW Room #1
Radiological Decommissioning/ License Termination
Project - Organizational Chart



11 CDC DECOMMISSIONING TASK MANAGEMENT

Decommissioning will be conducted in accordance with this DP. All contractor activities will be approved and overseen by the CDC to ensure compliance with the facility radioactive materials license. Decommissioning tasks will be performed according to written plans and procedures to ensure they provide adequate worker protection and comply with the radioactive materials license.

The following CDC management organization will provide relevant data and support, and make final decisions for the decommissioning effort:

- CDC Radiation Safety Officer (RSO) – Paul Simpson is the OSSAM Senior Health Physicist who has been the CDC RSO since 1981. He keeps and provides access to records relevant to this decommissioning effort, and is the final decision maker for releasing Building 1, RAW Room #1 for unrestricted use.
- CDC Project Manager – Kenneth Bryson is an architect who represents the CDC Projects and Construction Management Services Office (PCMSO) as project manager for this decommissioning.
- CDC MARSSIM consultant – Sam Keith is a Certified Health Physicist and an author of the NUREG 1575 MARSSIM Manual. He has conducted several MARSSIM decommissionings of CDC facilities on the Chamblee and Roybal Campuses, and is a consultant to the CDC RSO for this decommissioning effort.

Alhtough not expected to be used, Radiation Work Permits (RWP)s will be used to accomplish remediation activities. The RWP contains the location and description of the task to be performed, expected contamination and radiation levels, posting requirements, radiological monitoring requirements, Personnel Protective Equipment (PPE) requirements, and special work instructions necessary to complete the work in a safe and compliant manner.

Survey packages will be developed for each survey unit that contains specific survey instructions. Survey package preparation and completion will be approved by the PM to ensure all survey requirements and Data Quality Objectives (DQOs) are met.

12 PROJECT TRAINING REQUIREMENTS

The CDC will provide personnel with site specific Contractor Orientation Training.

12.1 Radiological Training

Basic Radiation Worker training will be completed and documented. The PM will maintain a copy of each individual's certification on site in the project file.

12.2 Project Specific Training

Prior to project start-up, personnel will attend an initial project-specific training session conducted by the PM. The training session will include the following items:

- Review of the DP
- Project security control and operational work zones
- Emergency response and site evacuation procedures
- Project communications
- General safe work practices
- Data quality and chain of custody procedures, and
- Review of applicable regulatory standards as applied to project operations

12.3 General Safety Briefings

General safety meetings will be held at the beginning of each work shift, if the project encompasses more than a single work shift. The purpose of this meeting will be to discuss project status, potential problem areas, general safety concerns, and to reiterate DP requirements.

12.4 Visitor Orientation

All non-essential personnel and visitors will be briefed on the DP requirements. Visitors will be escorted at all times and receive visitor training. Visitor training shall be administered to all personnel, contractors, and visitors requiring access to restricted areas. The scope of orientation shall be commensurate with the activities being performed and the risks involved. The orientation shall consist of the following:

- Project-specific health and safety orientation
- The location of restricted areas and escort requirements
- Posting and labeling identification of radiological areas and packages
- Requirement for PPE and dosimetry
- Escort requirements
- Review of Regulatory Guide 8.13 "Instructions Concerning Prenatal Radiation Exposure," Appendix B (required for female contractors or visitors), and

Visitor training shall be valid only for the particular project at which it is administered. Escorts shall have a minimum of Basic Radiation Worker training. Additionally, all visitors must receive training and/or briefings in accordance with CDC policies prior to entering restricted areas of the facility.

12.5 Transportation Training

Persons who prepare hazardous materials for transportation or are otherwise responsible for safely transporting hazardous material will be trained in accordance with the requirements of 49 CFR 172, subpart H.

13 RADIATION SAFETY AND HEALTH AND SAFETY PROJECT PLANS

A site-specifc Radiation Protection Plan (RPP) and Health and Safety Project Plan (HASP) will be prepared for all on-site activities.

Philotechnics, Ltd.
201 Renovare Blvd. Oak Ridge, TN 37830
Phone: 865.285.3027 • Fax: 865.220.0686

14 ENVIRONMENTAL MONITORING AND CONTROL

All project activities will be performed indoors, under stirct controls, and in a manner thar does not present an elevated risk of environmental releases above normal operations.

Philotechnics, Ltd.
201 Renovare Blvd. Oak Ridge, TN 37830
Phone: 865.285.3027 • Fax: 865.220.0686

15 RADIOACTIVE WASTE MANAGEMENT PLAN

Although no waste is expected to be generated a site specific Waste Management Plan will be prepared for all on-site activities.

16 QUALITY ASSURANCE PROJECT PLAN

A site specific Quality Assurance Project Plan (QAPP) will be prepared for all on-site activities.

17 FINAL STATUS SURVEYS

Final status surveys (FSS) are performed to demonstrate that residual radioactivity in each survey unit satisfies the predetermined criteria for release for unrestricted use. FSS are conducted by performing the appropriate combination of scan surveys, total activity measurements and removable activity measurements as discussed further in this section. Scoping and remedial action survey data will be used as final status survey data to the maximum extent possible in order to minimize overall project costs.

17.1 Background Determination

Reference background areas are available and have been selected for this survey design. Ambient background has been and will be determined for each survey to calculate the actual survey MDCs and associated counting errors.

For total surface activity measurements, ambient background levels are generally determined by performing a sixty-second timed count with the probe at waist level and away from survey unit surfaces. Reference background is subtracted from each total activity gross measurement. Material background, the contribution from naturally-occurring radioactivity in building structural materials, is part of the ambient background in the matched reference background areas and survey units.

Background corrections are performed for removable activity measurements. The liquid scintillation counter was set up to report results in net dpm in each channel, and all removable activity results are reported in net dpm/100cm^2.

17.2 Data Quality Objectives (DQO)

The following is a list of the major DQOs for the survey design described in this report:

- Static measurements will be taken to achieve an MDC_{static} of less than 25% of the DCGL or 2,500 dpm/100cm^2 Beta, 25% of the DCGL or 10,000 dpm/100cm^2 Gamma, and 50% of the DCGL or 75 dpm/100cm^2 Alpha.
- Scanning will be conducted at a rate to achieve an MDC_{scan} of less than 25% of the DCGL or 2,500 dpm/100cm^2 Beta, 25% of the DCGL or 10,000 dpm/100cm^2 Gamma, and 50% of the DCGL or 75 dpm/100cm^2 Alpha.
- Smear counting will be conducted to achieve an MDC of less than 2,50 dpm/100cm^2 Beta, 1,000 dpm/100cm^2 Gamma, and 7.5 dpm/100cm^2 Alpha.
- Individual measurements will be made to a 95% confidence interval.
- Decision error probability rates will initially be set at 0.05 for both α and β.
- The null hypothesis (H_0) and alternate null hypothesis (H_A) are that of NUREG 1505 scenario A:
- H_0 is that the survey unit does not meet the release criteria
- H_A is that the survey unit meets the release criteria
- Scoping and remedial action support surveys will be conducted under the same quality assurance criteria as final status surveys such that the data may be used as final status survey data to the maximum extent possible.
- Quality Assurance Surveys will be conducted at a rate of 5%.

17.3 Area Classifications

Based on the results of the historical site assessment, facility areas were classified as impacted or non-impacted areas. Non-impacted areas are areas with no potential residual radioactivity from licensed activities. Impacted areas are those areas that may have some level of potential residual radioactivity from licensed activities.

Impacted areas are typically divided into Class 1, 2, or 3 areas. Class 1 areas have the greatest potential for contamination and therefore receive the highest degree of survey effort for the final status survey, followed by Class 2 and then by Class 3. **Table 17-1 - Recommended Maximum Survey Unit Size Limits** below lists the recommended maximum survey unit sizes based on floor area. It should be noted that these limits are recommended and are not absolute.

17.3.1 Class 1 Areas

Areas with the highest potential for contamination, and meet the following criteria: (1) impacted; (2) potential for delivering a dose above the release criterion; (3) potential for small areas of elevated activity; and (4) insufficient evidence to support classification as Class 2 or Class 3.

- For conservatism, the CDC chose to classify RAW Room #1, including the ceiling, the heater and ventilation louvers as Class 1

17.3.2 Class 2 Areas

Areas that meet the following criterion: (1) impacted; (2) low potential for delivering a dose above the release criterion; and (3) little or no potential for small areas of elevated activity.

- For conservatism, the CDC chose to classify the loading dock, the parking pad and the walk ramp as Class 2

17.3.3 Class 3 Areas

Areas that meet the following criterion: (1) impacted; (2) little or no potential for delivering a dose above the release criterion; and (3) little or no potential for small areas of elevated activity.

- For conservatism, the CDC chose to classify the crawlspace area under RAW Room #1 as Class 3

17.4 Non-impacted

Building exterior, outside grounds, indoor areas other than those identified as restricted areas by the licensee, and the roof.

Table 17-1 - Recommended Maximum Survey Unit Size Limits

Survey Unit	Class 1	Class 2	Class 3
Structures	Up to 100 m^2	100 m^2 to 1,000 m^2	No limit
Land	Up to 2,000 m^2	2,000 m^2 to 10,000 m^2	No limit

Table 17-2 - Classification below lists the survey units and their final classification. During the survey none of the data collected during the scans, static or removable measurements warranted re-classifying any of the survey units. Each previously impacted area in the building was made its own survey unit.

Table 17-2 - Classification

4770 Buford Highway	Survey Unit	Initial Classification
RAW Room #1 Lower Walls and Floor	1	Class 1
RAW Room #1 Upper Walls, Ceiling, heater and ventilation louvers	2	Class 1
RAW Room #1 Loading Dock, Parking Pad and Walkway	3	Class 2
RAW Room #1 Crawlspace	4	Class 3

17.5 Survey Methodology

Determination of Class 1 survey unit sample locations is accomplished by first determining sample spacing and then systematically plotting the sample locations from a randomly generated start location. The random starting point of the grid provides an unbiased method for obtaining measurement locations to be used in the statistical tests. Class 1 survey units have the highest potential for small areas of elevated activity so the areas between measurement locations may be adjusted to ensure that these areas can be detected by scanning techniques. All of RAW Room #1 was classified as Class 1 for conservatism and the potential for radioactive contamination although it was not expected to exceed the $DCGL_w$. Philotechnics utilized a square grid system for the Class 1 area. Judgmental sample locations were taken. For FSS, the starting point will be determined using a random number generator.

For FSS, similar systematic spacing methods are used for Class 2 survey units because there is an increased probability of small areas of elevated activity. The use of a systematic grid allows the decision-maker to draw conclusions about the size of the potential areas of elevated activity based on the area between measurement locations. The loading dock, the parking pad and the walkway were classified as Class 2 for conservatism due to the potential for leaks during RAM transport.

For Class 3 survey units MARSSIM guidance recommends simple random measurement patterns to ensure the measurements are independent and support the assumptions of the statistical tests. Fopr conservatism, eventhough 6 mil plastic lined the RAW Room #1 floor and the CDC could find no evidence of spills, the crawlspace under RAW Room #1 was classified as Class 3 for conservatism due to the potential for leaks during RAM transport.

17.6 Background Determination

A suitable reference background area is available and selected for determining ambient background for the radiological surveys of RAW Room #1. This decision is based on the guidance provided in Section 12 of NUREG-1505, "A Nonparametric Statistical Methodology for the Design and Analysis of Final Decommissioning Surveys." This section states that "*better precision is possible if the average of the measurements made on the reference material is subtracted from each measurement made on that material*". This section states that better precision is possible if the average of the measurements made on the reference material is subtracted from each measurement made on that material.

However, in the case of this survey, based on the results of the scoping surveys, the use of reference background measurements or paired background is not necessary, as material and ambient background levels are not present in significant levels in comparison to the DCGLs. Therefore, for conservatism in the survey design, ambient background levels determined by taking ten (10) one-minuted timed counts for beta and gamma, and ten (10) two-minute counts for alpha, and calculating the mean of the ten (10) timed counts to provide an ambient background level for each radiation type. The mean ambient background will be determined by taking the requisite timed counts for each radiation type in the center of a non-impacted area of the facility at waist level. The mean ambient background will be subtracted from gross measurement count rates (in cpm) to determine the net measurement count rate. The mean ambient background will also be used to calculate the actual survey MDCs and the associated count errors. The number of measurements required for each material type will be calculated for the Sign test.

17.7 Surface Scans

Scanning is used to identify locations within the survey unit that exceed the DCGL. These locations are marked and receive additional investigations to determine the concentration, area, and extent of the contamination. For Class 1 areas, scanning surveys are designed to detect small areas of elevated activity that are not detected by the measurements using the systematic pattern. The percentage of actual accessible building structural surfaces to be scanned compared to MARSSIM recommendations are presented in **Table 17-3 - Scan Survey Coverage**.

Table 17-3 - Scan Survey Coverage

Classification	Percentage of Surface Area Requiring Scan Coverage (MARSSIM)	CDC's Surface Area Scan Coverage
1	100%	100% of all accessible areas (holders/casing for the instrument detectors normally prevent direct scans along the intersection of walls, floors and ceiling)
2	10 – 100% (Judgmental)	50% of all accessible areas
3	Judgmental	25% of all accessible areas

The scan survey percentage was chosen in order to provide a comprehensive survey of the impacted areas and provide confidence there is no contamination present above the DCGLs. In the event of any elevated activity noted from the survey, the location will be marked, additional measurements will be taken to quantify the activity, and any decontamination determined to be appropriate will be conducted prior to a resurvey. The probe is maintained at a constant distance of approximately 1/8-1/4" (ensuring < 1 cm or 0.4 inches) above the surface using moving at a scan rate of 2.5 in/sec for large area probe alpha scans and 5 in/sec for large area probe beta/gamma scans. Survey instrumentation detectors, both small and large area probes are designed to float across all surfaces (floors, walls, structures) on state of the art Ultra-Wear-Resistant PTFE-Filled Delrin® Acetal Resin Teflon slides to maintain a constant 1/8-1/4" (ensuring < 1 cm or 0.4 inches) detector distance, as the detector is independent of the normal cart system associated with large area probe monitoring sytsems, which by design encompasses a fulcrum point, causing fluctuations in distance of the detector. The design is also not dependent on the technician attempting to hold the detector at a predetermined distance, while cautiously ensuring they do not damage the sensitive mylar by allowing the detector to creep to close to the surface or an uneven surface.

In addition total activity measurements will be collected in a random-systematic grid in accordance with the MARSSIM approach. Removable contamination measurements will be performed at each total activity measurement location. The scan ranges, square footage and sample spacing of each survey unit from the Scoping survey are presented in **Table 17-4 – Scoping Survey Area, Spacing, Scan Data**.

Table 17-4 – Scoping Survey Area, Spacing, Scan Data

Survey Unit	Room	Area (sq. ft.)	Sample Spacing	BP-19DD Range in CPM/100cm²	BP-19DD Average Static Count in DPM/100cm²	GP-13A Average Scan Range in CPM/100cm²	GP-13A Static Range in DPM/100cm²
1	Radioactive Waste Room	194	4 ft.	0-700	219	1300-1800	757
2	Radioactive Waste Room	194	4 ft.	0-500	130	1200-1800	354

The floor, the louvers and the ventialtion of the room and all other surfaces and structures were scanned using a Ludlum 2350-1 (serial# 186180) with a GP13A (100 cm^2 Gamma probe) and a Ludlum 2224 (serial# 187286) with a 43-93 (100 cm^{2}Alpha/Beta probe). Our data shows that ***all scan surveys are expected to be below the established DCGL$_W$.***

Based on the Scoping data, the minimum number of samples for FSS is calculated below.

17.8 Total Activity Direct or Static Measurements

Static measurements for total surface activity will be completed using a timed count on the surface to be measured at each specified sample location. A systematic grid with a random starting point will be used to determine the survey locations in the Class 1 areas. The probe will be held as close to the surface as practicable to determine a count rate in counts per minute. Scaler count times will be determined to achieve the detection sensitivities stated in the DQOs. Gross alpha and gross beta field measurements are converted to activity concentrations using the following equation:

$$\text{Activity (dpm/100cm}^2\text{)} = \frac{cpm_{sample} - cpm_{background}}{E_{total} \cdot \dfrac{A}{100cm^2}}$$

Where:

cpm_{sample} = sample count rate in counts per minute

$cpm_{background}$ = background count rate in counts per minute

E_{tot} = total detector efficiency for radionuclide emission of interest (includes combination of instrument efficiency and surface efficiency)

A = active area of detector

17.8.1 Determining the Minimum Number of Samples

In accordance with Section 5 of MARSSIM, the minimum number of samples required for the Sign Test was calculated using the following equations. The maximum alpha and beta/gamma standard deviations of total surface activity from the Scoping data will be used for calculations. The LBGR was set at 50% of the DCGL and then adjusted to provide a relative shift between one and three as described in Section 5.5.2 of MARSSIM. The calculation performed to determine the required number of samples is provided below.

17.8.2 Determination of the Relative Shift

The number of required samples depends on the ratio of the activity level to be measured relative to the variability in the concentration. This ratio is called the Relative Shift, $\Delta/\sigma S$ and is defined in MARSSIM as:

$$\Delta / \sigma_S = \frac{DCGL - LBGR}{\sigma_S}$$

Where:

DCGL	=	Derived Concentration Guideline Level for each specific radiation type, 150 dpm/100 cm^2 alpha, 10,000 dpm/100 cm^2 beta, 40,000 dpm/100 cm^2 gamma
LBGR	=	Concentration at the lower bound of the gray region. The LBGR is the average concentration to which the survey unit should be cleaned in order to have an acceptable probability of passing the test Appendix N, the Roadmap for Survey Design and Section 2.3.1 – Planning Effective Surveys of MARSSIM recommend the LBGR is initially set arbitrarily to one half (1/2) the $DCGL_W$. Therefore LBGR was initially set to 75 dpm/100 cm^2 alpha, 5,000 dpm/100 cm^2 beta, 20,000 dpm/100 cm^2 gamma , and will be verified and validated or adjusted during the Data Qaulity Assessment and Interpretation of Results phase.
σ_S	=	an estimate of the standard deviation of the residual radioactivity in the survey unit Arbitrarily set to 19 dpm/100 cm^2 alpha, 882 dpm/100 cm^2 beta, 625 dpm/100 cm^2 gamma, for conservatism based on the largest standard deviation of each radiation type from the Scoping Surveys

Utilizing the inputs from above, example Relative Shifts for each radiation type are provided in the figures below using the largest standard deviation, between Floor and Lower Wall, and Ceiling and Upper Wall Scoping Survey results, and a .25 surface efficiency for all alpha and beta measurement reults, for added conservatism in the survey design. The largest standard deviations were:

- Alpha: 19 dpm/100 cm^2 Floors and Lower Walls
- Beta: 1,765 dpm/100 cm^2 Floors and Lower Walls
- Gamma: 625 dpm/100 cm^2 Floors and Lower Walls

Figure 2 – Relative Shift Alpha

$$\Delta / \sigma_S = \frac{150 - 75}{19}$$

$$\Delta / \sigma_S = 3.95$$

Figure 3 – Relative Shift Beta

$$\Delta / \sigma_S = \frac{10{,}000 - 5{,}000}{1{,}765}$$

$$\Delta / \sigma_S = 2.83$$

Figure 4 – Relative Shift Gamma

$$\Delta / \sigma_S = \frac{40{,}000 - 20{,}000}{625}$$

$$\Delta / \sigma_S = 32$$

The most conservative value for Relative Shift, using the most conservative inputs from the Scoping Survey, was 2.83 for the Beta Relative Shift value. The value for Relative Shift is between one (1) and three (3), therefore no adjustments were required.

17.8.3 Determination of Acceptable Decision Errors

A decision error is the probability of making an error in the decision on a survey unit by failing a unit that should pass (β decision error) or passing a unit that should fail (α decision error). MARSSIM uses the terminology α and β decision errors; this is the same as the more common terminology of Type I and Type II errors, respectively.

The applicable decision errors (Type I Type II errors) were selected in accordance with the established Data Quality Objectives.

17.8.4 Determination of Number of Data Points

For the purposes of the final status survey it is assumed that the contaminant is not present in background at significant levels compared to the DCGLs. Therefore, material-specific background is ignored and is not subtracted from the total surface activity measurements. Using this methodology, the Sign Test was chosen for the statistical evaluation of survey data.

The number of direct measurements for a survey unit, employing the Sign Test, is determined from MARSSIM Table 5.5, which is based on the following equation (MARSSIM equation 5-2):

$$N = \frac{\left(Z_{1-\alpha} + Z_{1-\beta}\right)^2}{4\left(SignP - 0.5\right)^2}$$

Where:

N	=	number of samples needed in the survey unit
$Z_{1-\alpha}$	=	percentile represented by the decision error α
$Z_{1-\beta}$	=	percentile represented by the decision error β
SignP	=	estimated probability that a random measurement will be less than the DCGL when the survey unit median is actually at the LBGR

Utilizing the inputs from above, the calculation for Number of Samples is as follows in **Figure 13 – Number of Samples Required per Survey Unit** below.

Figure 5 – Number of Samples Required per Survey Unit

$$N = \frac{(1.645 + 1.645)^2}{4(0.993790 - 0.5)^2}$$

$$N = 11.10$$

Note: Percentiles $Z_{1-\alpha}$ and $Z_{1-\beta}$ are determined from MARSSIM Table 5.2. SignP is determined from MARSSIM Table 5.4 using the most coservative Relative Shift noted above, and rounding down for conservatism to a Relative Shift of 2.5.

MARSSIM recommends increasing the calculated number of measurements by 20% to ensure sufficient power of the statistical tests and to allow for possible data losses. Therefore, the number of samples needed for the structural surfaces of the survey for planning purposes is 14 using the calculation above, and 15 using MARSSIM Table 5.5. So to ensure the conservatism of the survey design, the number of samples the survey design will require is 15 sample locations per surevy unit, and the spatial independence of the sample distribution wil include floor area only, and not walls, thereby increasing the number of samples in the survey unit. This will in turn increase the number of samples on the areas with the highest probability of contamination, horizontal surfaces. The scoping surveys included a total of 31 total samples.

17.8.5 Determination of Sample Locations

Determination of Class 1 survey unit sample locations is accomplished by first determining sample spacing and then systematically plotting the sample locations from a randomly generated start location. The random starting point of the grid provides an unbiased method for obtaining measurement locations to be used in the statistical tests. Random starting location is accomplished by utilizing maximum "x" and and maximum "Y" coordinates from survey location maps. Using the random number generator function in Excel, the random number generatedis multiplied by maximum "x" and maximum "Y" coordinates from survey location maps to provide the "x" and "y" coordinates for the random start location. The Excel spreadsheets used to determine random start locations will be included in the FSSR.

Class 1 survey units have the highest potential for small areas of elevated activity, so the areas between measurement locations may be adjusted to ensure that these areas can be detected by scanning techniques.

Similar systematic spacing methods are used for Class 2 survey units because there is an increased probability of small areas of elevated activity. The use of a systematic grid allows the decision-maker to draw conclusions about the size of the potential areas of elevated activity based on the area between measurement locations.
Class 3 survey locations are determined from computer selected randomly generated x and y coordinates. The crawl space is the only Class 3 survey unit in this survey.

Survey protocols for all areas are summarized in **Table 17-5 - Survey Sample Placement Overview** below.

Table 17-5 - Survey Sample Placement Overview

Survey Unit Classification		$DCGL_W$ Comparison	Elevated Measurement Comparison	Measurement Locations
Impacted	Class 1	Yes	N/A	Systematic Random
	Class 2	Yes	N/A	Systematic Random
	Class 3	Yes	N/A	Random
Non-Impacted		None	None	None

17.8.5.1 Determining Class 1 and Class 2 Sample Locations

In Class 1 survey units, the sampling locations are established in a unique pattern beginning with the random start location and the determined sample spacing. After determining the number of samples needed in the survey unit, sample spacing is determined from MARSSIM equation 5-8:

$$L = \sqrt{\frac{A}{N}} \quad \text{for a square grid}$$

Where:

L = sample spacing interval
A = the survey unit area
N = number of samples needed in the survey unit

Maps of the survey unit will be generated and a random starting point will be determined on the floor using computer-generated random numbers coinciding with the x and y coordinates of the survey unit. A grid will be plotted across the survey unit surfaces based on the random start point and the determined sample spacing. A measurement location will be plotted at each intersection of the grid plot.

Figure 6 – Sample Spacing Interval per Survey Unit

$$L = \sqrt{\frac{194}{15}} \text{ for a square grid}$$

$$L = \sqrt{12.93} \text{ for a square grid}$$

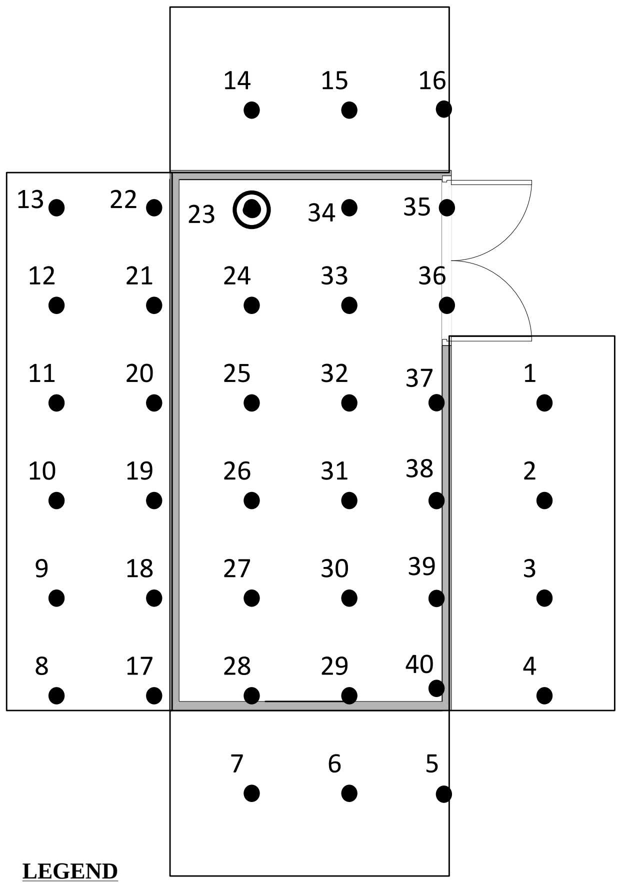
$$L = 3.6 \text{ for a square grid}$$

-or 3’7” spacing for square grid

Note: For conservatism, to increase sample distribution, only floor area square footage will be used to calculate sample spacing intervals for Class 1 and Class 2 areas in the survey design, thereby increasing the number of sample locations in excess of 100%. See example of Class 1 Survey Unit 1 sample locations provided in **Figure 15 – Example Survey Unit 1 Floor and Lower Walls Location Map** below.

Figure 7 – Example Survey Unit 1 Floor and Lower Walls Location Map

CDC
RAW ROOM #1 - First Floor
Example
Survey Unit Location Map
Survey Unit 1
Floor and Lower Walls

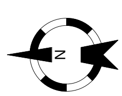


LEGEND

- Random Starting Point from Random Number Generation
X-Axis Start Location 3'
Y-Axis Start Location 18'

Typical Spacing = 1.10 m (3'7")

Area ~ 18.02 m² (194 ft²)



17.8.5.2 Determining Class 3 Sample Locations

For the only Class 3 area (the crawl space), a map will be generated of the survey unit's surfaces included in the statistical tests and will show the room's footprint and support pilingsfolded out in a 2-dimensional view. Sample locations are determined using computer generated random **x** and y coordinates for each sample location. Each location is plotted on the applicable survey map.

17.9 Removable Measurements Building Structures and Systems

Removable contamination measurements (smears) will be collected on building structural surfaces at each sample location. Each smear encompassed an area of approximately 100cm^2. If an area of less than 100cm^2 is wiped, a comment is added to the survey data sheet estimating the surface area wiped to allow for area correction of the results. The total efficiency is determined from the reported emission rate on the calibration trace form for the source and the surface efficiency set to approximate dirt loading on the smear paper. Most smears will be from "clean" surfaces due to Philotechnics pre-survey cleaning. Per McFarland's data for filter paper, alpha particle counting efficiency is lowered by approximately 15% from dirt loading of 5 mg on filter paper. "Clean" surfaces typically contain 1-3 mg of dirt. However, ISO 7503-1 recommends that a conservative surface efficiency of 0.25 be used for beta particles in the energy range of 150 keV to 400 keV and alpha emitters. Therefore, the ISO 7503-1 efficiency is used.

$$\text{Activity}\,(\text{dpm}/100\text{cm}^2) = \frac{cpm_{sample} - cpm_{background}}{E_{total}}$$

Where:

cpm_{sample} = sample count rate in counts per minute

$cpm_{background}$ = background count rate in counts per minute

E_{tot} = total detector efficiency for radionuclide emission of interest (includes combination of instrument efficiency and surface efficiency)

All of the smear samples taken at the CDC are counted on a Tri-Carb Liquid Scintillation Counter (LSC) for one minute and a Protean Gross Alpha/Beta Counter. The channels for the LSC were set up so H-3 would be detected in Channel A, C-14 in Channel B and a wide open Channel C. Scintillation standards were used to determine if the scintillaiton counter was operating within normal parameters. The efficiencies for the scintillation counter were 61% for H-3 and 96% for C-14 for the scoping survey, and efficiencies current at the time of the FSS will be used. For Channel C (wide open) we reported the data in cpm/100cm^2 to show that no other radioisotopes of concern were detected.

17.9.1 Survey of Building Mechanical System Internals

Survey design for systems is out of the scope of MARSSIM; however for the purposes of identifyng potential residual contamination a removable contamination measurement will be collected at all system internal locations. Swabs will be used when system or component access points are not large enough to allow for a wipe of a $100cm^2$ surface area. According to interviews with the RSO, no radioactive material was released to the sanitary sewer system at RAW Room #1. Although there was a sink in RAW Room #1, sanitary sewer disposal on campus was generally limited to small total activities or radionuclides that had decayed at least 10 half-lives

17.10 Survey Investigation Levels

Investigation levels are used to flag locations that require special attention and further investigation to ensure areas are properly classified and adequate surveys are performed. These locations are marked and receive additional investigations to determine the concentration, area, and extent of the contamination. Investigations will include alpha activity measurements as appropriate. The survey investigation level for each type of measurement is listed by classification in **Table 17-6 - Survey Investigation Levels** below.

Table 17-6 - Survey Investigation Levels

Survey Unit Classification	Flag Direct Measurement Result When:	Flag Scanning Measurement Result When:	Flag Removable Measurement Result When:
1	>50% of DCGL	>MDC	> 50% of DCGL
2	>25% of DCGL	>MDC	> 25% of DCGL
3	>MDC	>MDC	>MDC

17.11 Unity Calculations

Unity will be applied to each sample location using the following equation to determine compliance.

$$\frac{CAlpha}{DCGLAlpha} + \frac{CBeta}{DCGLBeta} + \frac{CGamma}{DCGLGamma} < 1$$

Where:

C_{Alpha} = Gross alpha result in $dpm/100cm^2$
C_{Beta} = Gross beta result in dpm/100cm 2
C_{Gamma} = Gross gamma result in dpm/100cm 2
$DCGL_{Alpha}$ = Gross alpha result in dpm/100cm 2
$DCGL_{Beta}$ = Gross beta result in dpm/100cm 2
$DCGL_{Gamma}$ = Gross gamma result in dpm/100cm 2

This method ensures that, regardless of the radionuclide distribution in a particular location, the dose limit of 25 mrem per year will not be exceeded as long as the sum of fractions shown above is less than 1.

18 SURVEY DOCUMENTATION AND DATA ASSESSMENT

Each survey unit will be surveyed under survey instructions from the PM which will specify the survey protocol to be followed. The survey instructions are to ensure the Data Quality Objectives (DQOs) are met:

- Survey protocol instructions such as the number of samples, sample spacing, sample locations, areas to be scanned, etc.
- Random number generations to determine survey locations
- Instrumentation to be used
- Scan rates, static count times, and/or minimum sample volumes
- Scaled survey unit maps
- Recommended survey sequence

Each static and removable contamination measurement location is assigned a unique alpha-numeric location code consisting of a sequence of identifiers to indicate specific information about its location, such as the building, survey unit, structural surface (floor, wall, benchtop, etc.), structural material (concrete, cinderblock, sheetrock, etc.) and a numerically sequenced location number within the survey unit.

18.1 Data Validation

Field data will be reviewed and validated to ensure:

- Completeness of forms
- Proper types of surveys were performed
- The MDCs for measurements met the established data quality objectives
- Independent calculations were performed on a representative sample of data sheets
- Satisfactory instrument calibrations and daily functionality checks were performed as required

18.2 Data Quality Assessment (DQA) and Interpretation of Survey Results

The statistical guidance contained in Section 8 of MARSSIM will be used to determine if areas were acceptable for unrestricted release, and whether additional surveys or sample measurements were needed.

18.2.1 Preliminary Data Review

A preliminary data review will be performed to identify any patterns, relationships or potential anomalies. Additionally, measurement data will be reviewed and compared with the DCGLs and investigation levels to identify areas of elevated activity.

The following preliminary data reviews will be performed:

- Calculations of the survey unit mean, median, maximum, minimum, and standard deviation for each type of reading.
- Comparison of the actual standard deviation to the assumed standard deviation used for calculating the number of measurements
- Comparison of survey data with applicable Investigation Levels.

18.3 Determining Compliance

For Class 1areas, if it is determined that all total activity results are less than the applicable DCGL, then no further statistical tests are required. If any of the total activity measurements are greater than the $DCGL_W$, then the survey unit fails and the null hypothesis is not rejected.

The Sign test is used to determine the minimum number of sample locations. However, the Sign test is not performed in this survey design because the total activity DCGL is used as a maximum. If all measurements are less than the DCGL, performance of the Sign test is not necessary because the survey unit will pass the Sign test.

Data from Class 2 and 3 areas are initially compared to the investigation levels to ensure the survey units have been properly classified. If all results are less than the investigation levels, the survey unit meets the release criterion. If not, then Philotechnics will perform an investigation to verify all initial assumptions.

Removable contamination measurements will be compared directly to the applicable DCGL. No contingency is established for elevated removable contamination. Therefore, if any removable contamination is detected which exceeds the removable contamination DCGL, the survey unit is determined not to meet the release criterion. However, if all removable contamination measurements are less than the removable contamination DCGL, then compliance shall be determined based on total activity measurements.

18.4 Mechanical System Survey Data Analysis

Results of mechanical system surveys will be compared directly with the DCGL. This comparison will consider the applicable DCGL as a maximum value, rather than an average. If any measurement exceeds the applicable DCGL, then the survey unit does not meet the release criterion and is considered contaminated. Remediation or removal of the affected system components may be required. If all measurements are less than the applicable DCGL, then the system meets the release criterion and is considered releasable.

18.5 Final Status Survey Report

At the completion of final status surveys, a final status survey report (FSSR) will be developed. The FSSR shall be reviewed for technical content by Philotechnics Radiological Services Management personnel, a Certified Health Physicist, and submitted to the CDC RSO for review and a final decision on the decommissioning effort. The following content must be included in the FSSR:

- An overview of the results of the final status survey
- A discussion of any changes that were made in the final status survey from what was proposed in the DP or other prior submittals
- A description of the method by which the number of samples was determined for each survey unit
- A summary of the values used to determine the number of samples and a justification for these values
- The survey results for each survey unit include:
    - The number of samples taken for the survey unit
    - A description of the survey unit, including (a) a map or drawing of the survey unit showing the reference system and random start systematic sample locations for Class 1 and 2 survey units and random locations shown for Class 3 survey units and background areas, and (b) a discussion of remedial actions and unique features
    - The measured sample concentrations in units that are comparable to the DCGL
    - The statistical evaluation of the measured concentrations
    - Judgmental and miscellaneous sample data sets reported separately from those samples collected for performing the statistical evaluation
    - A discussion of anomalous data, including any areas of elevated direct radiation detected during scanning that exceeded the investigation level or measurement locations in excess of $DCGL_W$, and
- A description of any changes in initial survey unit assumptions relative to the extent of residual radioactivity (e.g., material not accounted in this DP)
- A description of how ALARA practices were employed to achieve final activity levels
- If a survey unit fails, a description of the investigation conducted to ascertain the reason for the failure and a discussion of the impact that the failure has on the conclusion that the facility is ready for final radiological surveys and that it satisfies the release criteria
- If a survey unit fails, a discussion of the impact that the failure has on other survey unit information

19 FINANCIAL ASSURANCE

The expected cost of decommissioning is expected to be low (~$9,000) relative to the CDC's operating budget.

19.1 Cost Estimate

Not applicable.

19.2 Certification Statement

Not applicable.

19.3 Financial Mechanism

Not applicable.

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20 RESTRICTED USE/ALTERNATE CRITERIA

Not applicable.

21 REFERENCES

- NUREG-1575, “Multi-Agency Radiation Survey and Site Investigation Manual” (MARSSIM)Revision 1, August 2000 – including June 2001 updates
- NUREG-1505, Revision 1, “A Nonparametric Statistical Methodology for the Design and Analysis of Final Decommissioning Surveys," June 1998
- NUREG-1507, “Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions,” June 1998
- NUREG-1757, Volume 1, Rev. 1 “Consolidated NMSS Decommissioning Guidance: Decommissioning Process for Materials Licensees,” 2006
- NUREG-1757, Volume 2 “Consolidated NMSS Decommissioning Guidance: Scoping, Survey, and Determination of Radiological Criteria,” 2006
- NUREG-1757, Volume 3 “Consolidated NMSS Decommissioning Guidance: Financial Assurance, Recordkeeping, and Timeliness,” 2012
- NUREG/CR-5512, “Residual Radioactivity from Decommissioning: Parameter Analysis,” August 1999.
- ISO-7503-1, “Evaluation of Surface Contamination -Part 1: Beta Emitters (Maximum Beta Energy Greater Than 0.15 MeV) and Alpha Emitters”, First Edition 1988-08-01.
- ISO-7503-2, “Evaluation of Surface Contamination -Part 2: Tritium Surface Contamination”, First Edition 1988-08-01.
- ANL/EAD/03-1 “User’s Manual for RESRAD-BUILD Version 3,” June 2003
- “Decommissioning Health Physics, A Handbook for MARSSIM Users,” 2001
- “Handbook of Health Physics and Radiological Health”, 3rd Edition, 1998
- Federal Guidance Report No. 11 (EPA-5201/1-88-020), “Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,” 1988
- Federal Guidance Report No. 12 (EPA-402-R-93-081), “External Exposure to Radionuclides in Air, Water, and Soil,” September 1993
- Federal Guidance Report No. 13 (EPA-402-R-99-001), “Cancer Risk Coefficients for Environmental Exposure to Radionuclides,” September 1999
- Regulatory Guide 1.86, “Termination of Operating License for Nuclear Reactors”, U.S. Nuclear Regulatory Commission, Washington, DC, June, 1974
- CDC Radioactive Materials License Number #10-06772-01