CDC
12 Labs
Final Status Survey Report

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*
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 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).

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) decided to cease all operations and permanently decommission 11 use labs located in buildings 15 (SB401, SSB401, SB101), 17 (4085, 5130), 18 (5-412, B703B.3) and 23 (10-624, 10-654, 10-439, 10-471) located on the CDC Roybal campus at 1600 Clifton Rd in Atlanta, Georgia; and one (1) use lab in building 110 (4207C) on the CDC Chamblee satellite campus at 4770 Buford Highway in Atlanta, Georgia. The labs were used for various research purposes. As a result of the completion of research within the labs requiring the use of radioactive materials (RAM), Philotechnics, Ltd. (Philotechnics) was contracted to perform all decontamination and decommissioning (D&D) activities to attain release for unrestricted use of the facility. Labs will continue research without radioactive material (RAM) upon release.

The CDC and Philotechnics conducted a Historical Site Assessment (HSA) documenting radiological operations from the beginning of operations in the labs. A thorough review of the historical utilization revealed they were used for research purposes. Radioactive materials consisted of: Carbon-14 (C-14), Tritium (H-3), Phosphorus-32 (P-32), Sulfur-35 (S-35), natural thorium (Th-nat) and natural uranium (U-nat).

Philotechnics performed all D&D activities in accordance with the CDC DP developed following the requirements and guidance provided in Chapter 16 and 17 of NUREG 1757 Volume 1, Revision 2. The CDC DP followed 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 provided the approach, methods, and techniques for the radiological D&D of the labs. To demonstrate compliance with site-specific release criteria for unrestricted use, Final Status Surveys (FSS) implemented the protocols and guidance provided in MARSSIM to ensure technically defensible data were 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 Total Effective Dose Equivalent (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 evaluated the property's environmental status for release of impacted areas to allow unrestricted use by current or future tenants. Philotechnics and the CDC performed a HSA to review facility operations as they pertained to RAM usage and storage to identify potential residual radioactive contamination. This assessement was performed prior to commencing field activities. 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 to identify potential contamination and/or presence of RAM;
- 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 (RML), 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 RAM;
- Indirect surveys to test for removable contamination with the use of a scintillation counter and wipes taken throughout the impacted areas; and
- Dose estimates for alpha sources using the entire on-hand quantities to determine if they could be excluded from consideration.

2.1 Licensed Operations

Mr. Ken Gavlik of Philotechnics, interviewed Mr. Narvaez Stinson, the Radiation Safety Officer (RSO) at CDC. The interview and document reviews revealed that the small microcurie quantities of radioactive material were used in these laboratories (limited by the quantities needed for R & D studies), along with the results of leak tests, monthly radiological surveys conducted by the laboratory staff, monthly laboratory inspections conducted by the Radiation Safety Team, and quarterly quality control surveys by the Radiaiton Safety Team, indicated that residual radioactivity would be several orders of magnitude less than the relevant derived concentration guideline (DCGL) levels. The interview with Mr. Stinson also indicated that there have not been any significant radioactive materials spills affecting any of the affected laboratories. Significant spills are defined as those spills that were not readily cleaned up by the researcher and/or caused contamination to be found during follow-up or routine contamination surveys in excess of regulatory limits. Monthly contamination surveys were included in the historical review of the license and there were no indications of contamination levels over the criteria for release affecting the areas included in this decommissioning survey.

Licensed RAM consisted of: H-3, C-14, P-32, S-35, Th-nat, and U-nat.

- The NRC added Th-nat and U-nat to the RML 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 RML 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 were not relevant to this decommissioning and were excluded from the assessment; **however, for added conservatism in the survey design, they were included.**
- Source Material uranium and thorium in the form of natural uranium and natural thorium were used in the CDC laboratory for the purpose of preparing calibration standards for the analysis of uranium and thorium in urine and tobacco using inductively coupled plasma-mass spectrometry. Standards were purchased in 1000 µg/mL or lower concentrations from the vendor, High Purity Standards, Charleston, SC. At these concentrations, the alpha emissions from natural uranium and natural thorium are very low. They are below levels requiring them to be shipped as radioactive materials by either DOT or NRC. The plastic bottles in which the solutions were received, and the aqueous solutions in which they were dissolved and further absorbed rendered detection of the very low alpha emissions. Upon receipt, these solutions were diluted to the ranges of uranium and thorium concentrations typically observed in human urine, between 0 and 500 ng/L for natural uranium and between 0 and 100 ng/L for natural thorium. Standards prepared in these low concentration ranges were further diluted 1/10 before analysis to correspond with dilution of urine in dilute acid solution for analysis. A 12 liter liquid waste container was calculated to contain approximately 6 x 10-12 mCi of uranium and/or thorium, so it becomes evident why it is literally impossible to detect the very low levels of these source materials that had been diluted to common urine concentrations. Even the entire waste container concentration is spilled on a 100 cm2 area, it would have been difficult to detect the low alpha emissions. However, since they were prepared on absorbent pads in spill trays, even if the levels had been measurable, spills that might have caused contamination were further prevented.

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

RAM on Roybal campus was only used or stored in the aforementioned labs, and no other area on Roybal campus, as well as the single lab on Chamblee campus as summarized in **Table 2-1 - Restricted Area Summary** below, and identified on the building diagrams in Appendix A.

Routine surveys were conducted by the CDC. Monthly laboratory surveys were conducted by the laboratory staff of each designated radiation laboratory. These monthly surveys employed the use of swipes that were taken at designated locations within each radiation laboratory. These swipes were then analyzed with the use of a calibrated liquid scintillation counter for detecting removable contamination. Additionally, the Radiation Safety Team conducted quarterly quality control surveys in which a minimum of 10 swipes were taken in each designate radiation laboratory. These wipes were analyzed with the use of a calibrated liquid scintillation counter to determine if any surfaces were contaminated. 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 the labs.

Table 2-1 - Restricted Area Summary

CDC 12 Laboratories Radioisotope Use Restricted Area Summary for Decommissioning		
Authorized User	**Laboratory**	**Radioisotopes**
Roybal Campus		
Brian Harcourt	Bldg. 15, Lab SB401	C-14, H-3, P-32, S-35
Brian Harcourt	Bldg. 15, Lab SSB401	C-14, H-3, P-32, S-35
Brian Harcourt	Bldg. 15, Lab SB101	C-14, H-3, P-32, S-35
James Posey	Bldg. 17, Lab 4085	P-32, S-35
P.I. James Stevens	Bldg. 17, Lab 5130	H-3
Paul Rota	Bldg. 18, Lab 5-412	H-3
Brian Harcourt	Bldg. 18, Lab B703B.3	C-14, H-3, P-32, S-35
John Barnwell (Retired)	Bldg. 23, Lab 10-624	H-3, P-32, S-35
John Barnwell (Retired)	Bldg. 23, Lab10-654	H-3, P-32, S-35
Evan Secor	Bldg. 23, Lab 10-439	H-3
Evan Secor	Bldg. 23, Lab10-471	H-3
Chamblee Campus		
Steve Pappas	Bldg. 110, Lab 4207C	Th-nat, U-nat – not licensed

2.2 License Number/Status/Authorized Activities

The CDC is currently authorized to possess the following radionuclides as summarized in **Table 2-2 - 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 Used and/or Stored within the 12 Labs

The following licensed radioactive materials were used and/or stored within the 12 labs:

Table 2-3 - Radionuclides Used and/or Stored within the 12 Labs

Nuclide	Half-life (years)	Half-Life >120 Days	Predominant Emissions
H-3	1.2E+01	YES	Beta
C-14	5.7E+03	YES	Beta
P-32	3.9E-02	NO	Beta
S-35	2.4E-01	NO	Beta

The radionuclides P-32 and S-35 were eliminated due to short half lives, and last use. T-nat and U-nat were eliminated as described in **Section 2.1 – Licensed Operations** above; **although they were included in FSS**.

Table 2-4 - Radionuclides of Concern for 12 Labs

Nuclide	Half-life (years)	Half-Life >120 Days	Predominant Emissions
H-3	1.2E+01	YES	Beta
C-14	5.7E+03	YES	Beta

2.4 Previous Decommissioning Activities

Based on interviews with the RSO and document reviews, there are no records of previous decommissioning activities performed within the labs.

2.5 Radioactive Materials Spills

By completing a review of pertinent records and interviews, it was determined there have not been any significant radioactive material spills affecting the labs. Significant spills are defined as those spills that were not readily cleaned up by the researcher and/or caused contamination to be found during follow-up or routine contamination surveys in excess of regulatory limits. Routine contamination surveys were included in the historical review of the license and there were no indications of contamination levels in excess of the criteria for release.

2.6 Prior On-site Burials

There is no record of any on-site burials **or other disposals of radioactive materials anywhere on the Roybal of Chamblee campuses**.

3 FACILITY DESCRIPTION

3.1 Ownership

The facility is currently owned by the CDC. The facility will not be demolished after unrestricted release.

3.2 Population Distribution

Not Applicable – all impacted areas indoors.

3.3 Current/Future Land Use

Not applicable – all impacted areas indoors.

3.4 Meteorology and Climatology

Not applicable – all impacted areas indoors.

3.5 Geology and Seismology

Not applicable – all impacted areas indoors.

3.6 Surface Water Hydrology

Not applicable – all impacted areas indoors.

3.7 Ground Water Hydrology

Not applicable – all impacted areas indoors.

3.8 Natural Resources

Not applicable – all impacted areas indoors.

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4 RADIOLOGICAL STATUS OF THE FACILITY

The radiological status of the facility was determined by reviewing historical survey records, interviewing Radiation Safety personnel and performing closeout surveys. The facility has been surveyed on a routine basis, the results of the CDC radiation meter and wipe surveys indicated all items and areas were free from residual contamination and at natural background radiological levels for both total and removable surface activity. Based on the assessment, there have been no areas of elevated residual radioactivity identified by surveys in preparation for decommissioning. Routine periodic surveys were performed by researchers and Radiation Safety personnel. Laboratory closeout procedures were used when activities involving radioactive materials were completed. Based on a review of historical survey results and previous decommissioning surveys, it was expected that the facility meets the release criteria for unrestricted use, and will only require FSSs to confirm this assumption. Additionally, no areas of elevated activity were remediated.

4.1 Historical Routine Survey Results

During the historical assessment, document reviews and personnel interviews indicated routine removable contamination surveys were performed in areas where operations using licensed materials took place. Removable activity surveys were performed; results were recorded based on the current operations in the area surveyed. Routine surveys normally included working surfaces where materials were physically handled and the immediate surrounding areas. Removal contamination was measured by swipes counted on a liquid scintillation counter.

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 radiological release criteria of NRC lO 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 10CFR20.1402, "Radiological Criteria for Unrestricted Use." The criteria is 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)[1]. $DCGL_w$ 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. The building structural surfaces DCGLs for this project were developed utilizing NUREG 1757 Volume 1, *Table B.*

There are no impacted outdoor areas.

5.1 Dose Model

Dose modeling was performed during the decommissioning of Radioactive Waste Room Number 1 (RAW Room #1) at the CDC Chamblee campus and included in the *CDC RAW ROOM #1 Decommissioning Plan* dated April 11, 2016 and approved in letter from Dennis Lawyer, Commercial, Industrialm R&D and Academic Branch, Division of Nuclear Materials Safety, Region I, NRC on May 17, 2016. The model developed site specific DCGLs for unrestricted release of building structural surfaces for RAW Room #1 to include uranium, thorium, and special nuclear materials. Although the model was developed for additional radionuclides used and/or stored in RAW Room #1, it did include the alpha emiiters present from Chamblee campus, Building 110, Lab 4207C, authorized user Mr. Steve Pappas FSS Survey Unit 3. Because, as a conservative measure for alpha emitters, the purpose of the surveys was to release a single room, Building 110 Room 4207C FSS Survey Unit 3, from radiological controls, only residual surface radioactivity was considered and there are no impacted outdoor areas. However, Philotechnics did perform additional Scoping of any travel paths, and entrances/exits and areas surrounding impacted areas of concern 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.

[1] The CDC has established 1 mrem/yr as an ALARA goal to satisfy ALARA requirements. DCGLs will be used as maximum concentrations vs. average concentrations to simplify the survey design and to ensure the ALARA goal is met

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 did not include dose modeling for the alpha emitters present at the CDC.**

In order to develop site-specific DCGLs, a RESRAD-BUILD model was run. The following documents the process, the modeling and assumptions used, and the conclusion drawn.

Typically, RESRAD-BUILD is run 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. In addition to the initial model run for RAW Room #1,, and final RESRAD-BUILD Version 3.50 model was also run using the maximum total alpha activity contamination result with Total Effective Dose Equivalent results in mrem/yr provided in **Section 18.8 Determining Compliance**.

For this RESRAD-BUILD run, the relative abundances of radionuclides present 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 limit determined by RESRAD-BUILD to assign a DCGL. It is important to note, extremely conservative assumptions were used.

5.2 Determination of Radionuclides of Concern

The relative abundances of radionuclides present are not known. Radionuclides of concern (ROC) 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.
- 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 ROC in order to determine the limiting radionuclide for each decay mode: alpha, beta, gamma (electron capture), and low-energy beta. The limiting radionuclide was determined to be Th-232 plus decay products. 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.

The only ROCs remaining after sorting radionuclides used and stored in the facility were C-14 and H-3 as presented in **Table 2-4 - Radionuclides of Concern for 12 Labs**. **Table 5-1 – NRC NUREG 1757 Screening Values** below lists the radionuclide considered and the surface contamination level that produces a dose of 25 mRem/year.

Table 5-1 – NRC NUREG 1757 Screening Values[1]

Radionuclide	Primary Mode of Decay	Half Life	Screening Value NUREG 1757 (dpm/100cm²)
C-14	β	5730 years	3.7×10^6
H-3	Low Energy β	12.32 years	1.2×10^8

Using best industry practices, Philotechnics and CDC conservatively imposed an ALARA Goal level of 1 mrem/yr. The screening values from the table above for C-14 was converted from a Total Effective Dose Equivalent (TEDE) of 25 mrem/yr to 1 mrem/yr in disintegrations per minute (dpm) of 148,000 dpm/100 cm^2 for fixed beta-gamma contamination and a ten percent removable fraction for 14,800 dpm/100 cm^2 for removable beta-gamma contamination to determine if the impacted areas are suitable for unrestricted use. There were no surfaces or structures that exceeded these levels; no remediation and/or removal and disposal of low-level radioactive waste (LLRW) was performed.

In addition, as a conservative measure the following radionuclides were evaluated for Survey Unit 3, Room 4207C.

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 5-2 – RESRAD-BUILD Filtering Criterion and Equivalent Surface Contamination Limit

Radionuclide	Half-life (years)	Predominant Emissions	Equivalent Surface Contamination Limit [2] (dpm/100 cm^2)
Th-232	1.4E+10	Alpha	150
U-233	1.56E+05	Alpha	150
U-235	7.1E+08	Alpha	150
U-238	4.5E+09	Alpha	150

5.3 Default Screening Values

The default screening values for all radionuclides used in the decommissioning survey are listed in Table 5-2 below. The total activity values listed are maximum values and Philotechnics implemented practices to ensure all final readings were ALARA.

Based on the relative simplicity of the project, for the purpose of the FSS, the $DCGL_W$ was based on the published Default Screening Value (DSV) in NUREG 1757, Volume 1, *Table B.1 – Acceptable License Termination (Unrestricted Release) Screening Values of Common Radionulides for Building Surface Contamination*, determined from NRC Dand D Version 2.1 software and codes. Acceptable screening levels from NUREG 1757, Volume 1, Table B.1 are based on the release criterion for unrestricted use specified in 10 CFR 20.1402, *Radiological Criteria for Unrestricted Use - A site will be considered acceptable for unrestricted use if the residual radioactivity that is distinguishable from background radiation results in a TEDE to an average member of the critical group that does not exceed 25 mrem (0.25 mSv) per year, including that from groundwater sources of drinking water, and the residual radioactivity has been reduced to levels that are 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.* For the purpose of this FSS, acceptable screening levels were conservatively adjusted from 25 mrem/year to 1 mrem/year TEDE, resulting in a site-specific Building Surfaces DSV of 148,000 dpm/100cm^2 total activity. Removable activity is limited to <10% of the total activity, as indicated in the table below.

In addition, as previously explained, site-specific dose modeling was performed, not because of the complexity of the site, but because radionuclides 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

[2] 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 5-8 Default Screening Values for Limiting Radionuclides of Concern. Justification is provided in Section 5.3 DCGL Development, and Table 5-3 RESRAD-BUILD Parameters and Table 5-4 RESRAD-BUILD Dose Details for Radionuclides.

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 Average Laboratory Model Description

Based on the conservatism of the model and since the RAW Room #1 was smaller than Lab 4207C, the area of RAW Room #1 of 18 m^2 was maintained from the previous initial RESRAD-BUILD model. 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 5-3 – 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.	$1.12 \times 10^{-4}\ m^2h^{-1}$
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.	$3.06 \times 10^{-6}\ s^{-1}$ (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 5-4 – 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

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-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

Th-232 remained 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 5-5 – 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	$1.5x10^7$	$1.0x10^7$	$1.0x10^9$

Dose from H-3 was 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 5-6 – 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 5-7 – 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

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 5-8 – Default Screening Values for Limiting Radionuclides of Concern** below:

Table 5-8 – Default Screening Values for Limiting Radionuclides of Concern

Radionuclide	Limiting Radionuclide	Total Activity DCGLw (dpm/100cm^2)*	Removable Activity DCGLw (dpm/100cm^2)
Beta Emitters (C-14 and H-3)	**C-14**	**148,000**	**14,800**
Alpha Emitters	**Th-232**	**150**	**15**

* - Philotechnics implemented practices to ensure all final readings are ALARA.

To implement the ALARA principle, Philotechnics utilized industry-best practices to reduce residual radioactivity as ALARA goals for the FSS.

These DSVs also applied to internal surfaces of all mechanical systems.

Surveys were conducted to achieve MDCs based on the DSVs.

5.4 Hard to Detect Nuclides

Hard-to-detect nuclides (H-3) could not be adequately surveyed using direct field measurements and are typically evaluated by removable activity only as analyzed by liquid scintillation counting (LSC). As a conservative measure, Philotechnics verified the removable results by Philotechnics LSC in San Diego, CA upon return from the Project site. These results are included in Appendix G.

5.5 Unity

Unity was applied to each applicable sample location in Survey Unit 3 only, as it was also conservatively surveyed for alpha, to determine compliance.

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

6 ENVIRONMENTAL INFORMATION

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

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 was not required.

8 SURVEY INSTRUMENTATION

8.1 Instrument Calibration

Laboratory and portable field instruments were calibrated at least annually with National Institute of Standards and Technology (NIST) traceable sources, where feasible, and to radiation emission types and energies that provided detection capabilities and sensitivities required for the NOCs. Records of instrument calibration are included with this final status survey report and provided in Appendix C.

8.2 Functional Checks

Functional checks were performed at least daily when in use. The background, source check, and field measurement count times for radiation detection instrumentation were specified by procedure to ensure measurements were statistically valid. Reference background readings were 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 failed a functional check, all data obtained with the instrument since the last satisfactory check would be evaluated for usability by the PM or designee and unusable data discarded. All instrumentation passed its respective functional check.

8.3 Determination of Counting Times and Minimum Detectable Concentrations

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

8.3.1 Static Counting

Static counting MDC at a 95% confidence level was calculated using the following equation, which was 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 **Equation 8-1 – Beta Total Activity Example and Equation 8-2 – Alpha Total Activity Example** were prepared 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 FSS, to ensure they were 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

Equation 8-1 – Beta Total Activity Example

$$MDC_{static} = \frac{3 + 3.29\sqrt{476.6 \cdot 1 \cdot (1 + \frac{1}{1})}}{1 \cdot (.2379 * .25) \cdot \frac{100}{100cm^2}}$$

$$MDC_{static} = \frac{3 + 3.29\sqrt{953.2}}{.0595}$$

$$MDC_{static} = \frac{104.58}{.0595}$$

$$MDC_{static} = 1{,}758 \text{dpm/100cm}^2$$

Equation 8-2 – Alpha Total Activity Example

$$MDC_{static} = \frac{3 + 3.29\sqrt{3 \cdot 3 \cdot (1 + \frac{3}{3})}}{3 \cdot (.3004 * .25) \cdot \frac{100}{100cm^2}}$$

$$MDC_{static} = \frac{3 + 3.29\sqrt{18}}{.2253}$$

$$MDC_{static} = \frac{16.96}{.2253}$$

$$MDC_{static} = 75 \text{dpm/100cm}^2$$

8.3.2 Beta/Gamma Ratemeter Scanning

Scan MDC was determined based on the guidance described in MARSSIM Section 6.7.2 – *Scanning Sensitivity* and Decommissioning Health Physics, Second Edition, **Section 9.3 – Scan MDC**. Scanning was performed to identify areas of elevated activity in the survey unit. The scan MDC depended 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 was reduced to a function of the radionuclide alone. These calculations were provided in Section 6 of MARSSIM.

The scan MDCs were determined based upon site-specific background data from the FSS, using the equations below.

The number of source counts required for a specific time interval was 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 was assumed that at the first scanning stage a high rate (95%) of correct detections was required, and that a correspondingly rate of false positives (60%) would be tolerated. From MARSSIM Table 6.5, the value representing the performance goal was 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 was the site-specific background of 476.6 cpm, the value for b_i was 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) was then calculated using the number of source counts required for a specific time interval was 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):

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

Using the above input parameters, the scan MDC necessary to yield the MDC was 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 was calculated for Ludlum 2350-1 with a BP19DD or 43-93 probe for beta 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 remained under the detector for 1.385 seconds (e.g., i=1.385) and the background count rate was the site-specific ambient background rate of 476, the value for b_i was then calculated for beta as:

Equation 8-3 – Small Area Probe Beta Ratemeter Scanning Background Counts during Observation Interval Example

b_i (counts) = 476.6 cpm x 1.385 sec. x (1 min/60 sec)

b_i = 11.00 counts

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

Equation 8-4 – Small Area Probe Beta Ratemeter Scanning Specific Interval Example

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

Si = 1.38 x √11.00 counts

Si = 4.58 counts

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

Equation 8-5 – Small Area Probe Beta Ratemeter Scanning MDCR Example

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

MDCR = 4.58 counts x (60/1.385)

MDCR = 198.30 cpm

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

Equation 8-6 – Small Area Probe Beta Ratemeter Scanning $MDCR_{Surveyor}$ Example

$MDCR_{Surveyor}$ (cpm) = 198.30/(√0.5)
$MDCR_{Surveyor}$ (cpm) = 280.44 cpm

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

Scan MDC = MDCR/(√p)*($\mathcal{E}_{tot}$)*(*probe area*/100cm^2)
Scan MDC = 198.30/(√0.5)x(23.79%*.25) x(100 cm^2/100cm^2)
Scan MDC = 4,715 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]
p	=	surveyor efficiency (0.5)

Equation 8-7 – Small Area Probe Beta Ratemeter Scanning MDC_{Scan} Example

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*(√11.00)*(60/1.385)/(√0.5)*(23.79%*.25)*(100 cm^2/100cm^2)
MDC_{scan} = 4,715 dpm/100 cm^2

8.3.2.2 Large Area Probe Beta Ratemeter Scanning MDC

Assuming that the source remained under the detector for 1.25 seconds (e.g., i=1.25) and the background count rate was the estimated typical site-specific ambient background rate of 1,191, the value for b_i was then calculated for beta as:

Equation 8-8 – Large Area Probe Beta Ratemeter Scanning Background Counts during Observation Interval Example

b_i (counts) = 1,191cpm x 1.25 sec. x (1 min/60 sec)
b_i = 24.81 counts

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

Equation 8-9 – Large Area Probe Beta Ratemeter Scanning Specific Interval Example

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

$$S_i = 1.38 \text{ x } \sqrt{24.81} \text{ counts}$$

$$S_i = 6.87 \text{ counts}$$

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

Equation 8-10 – Large Area Probe Beta Ratemeter Scanning MDCR Example

MDCR (cpm) = S_i x (60/i)
MDCR = 6.87 counts x (60/1.25)
MDCR = 330.00 cpm

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

Equation 8-11 – Large Area Probe Beta Ratemeter Scanning $MDCR_{Surveyor}$ Example

$MDCR_{Surveyor}$ (cpm) = 330.00/($\sqrt{0.5}$)
$MDCR_{Surveyor}$ (cpm) = 466.69 cpm

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

Scan MDC = MDCR/($\sqrt{p}$)*($\mathcal{E}_{tot}$)*(*probe area*/100cm^2)
Scan MDC = 330.00/($\sqrt{0.5}$)x(38.08%*.25) x(821 cm^2/100cm^2)
Scan MDC = 597 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]
p	=	surveyor efficiency (0.5)

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

Equation 8-12 – Large Area Probe Beta Ratemeter Scanning MDC_{Scan} Example

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

MDC_{scan} = 1.38*($\sqrt{24.81}$)*(60/1.25)/($\sqrt{0.5}$)*(38.08%*.25)*(821 cm^2/100cm^2)
MDC_{scan} =597 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 was very close to zero. The following covers scanning for alpha emitters. Since the time a contaminated area was under the probe varied and the background count rate of some alpha instruments was less than 1 cpm, it was not reasonable to determine a fixed MDC for scanning. Instead, it was 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 provided a surveyor sufficient cause to stop and investigate further and therefore the probability of detecting given levels of alpha surface contamination was calculated by use of Poisson summation statistics. MARSSIM, section 6.7.2.2 and Appendix J, contained the guidance for scanning for alpha emitters having low release limits. MARSSIM provided derivations, formulas and probability concepts in Appendix J.

8.3.3.1 Count Detection Probability 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 was given using the following equation in

Equation 8-13 – Count Detection Probability Single Count Equation below.

Equation 8-13 – Count Detection Probability Single Count Equation Example

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

Where:

P(n≥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 was recorded and the guideline level of contamination was present, the surveyor stopped and waited until the probability of getting another count was at least 90 percent. This time interval was calculated using the following equation in Equation 8-14 – Count Detection Probability Time Interval Equation below.

Equation 8-14 – Count Detection Probability Time Interval Equation Example

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

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 was not possible due to the limitations on probe size, background and detector efficiency; therefore, only the larger (821 cm^2) gas-proportional detectors was used for alpha scans.

8.3.3.2 Count Detection Probability 821 cm^2 Probe

The larger (821 cm^2) gas-proportional detectors had alpha background count rates on the order of 20 cpm, and a single count would not cause a surveyor to investigate further. A counting period long enough to establish that a single count indicated elevated contamination level would be prohivitively inefficient. For these types of instruments, the surveyor needed to get at least two counts while passing over the source area before stopping for further investigation, and therefore the probability of getting two or more counts was calculated using the following equation in

Equation 8-15 – Count Detection Probability Two or More Counts Equation below.

Equation 8-15 – Count Detection Probability Two or More Counts Equation Example

$$P(n \geq 2) = 1 - P(n = 0) - P(n = 1)$$
$$= 1 - \left(1 + \frac{(GE + B)d}{60v}\right)\left(e^{-\frac{(GE+B)d}{60v}}\right)$$
$$= 1 - \left(1 + \frac{(GE + B)t}{60}\right)\left(e^{-\frac{(GE+B)t}{60}}\right)$$

Where:

P(n≥2)	=	Probability of observing at least 2 counts
C	=	Contamination Guideline in dpm/100 cm^2,
A	=	Physical probe area (cm^2)
G	=	Contamination activity (dpm), 75 dpm/100 cm^2 (50% of DCGL to meet DQOs) adjusted for detector at 100 cm^2 from 821 cm^2 = 615.75
E	=	Detector efficiency (4π) including ISO 7503 surface efficency of 0.25 for alphas = 10.90%
B	=	Background in dpm/100cm^2 = 9.5
t	=	D/v, dwell time over the source(15.9 cm/ 4 cm/sec) = ~4 sec
v	=	Scan speed (cm/s) = 4 cm/sec or ~1.5 inches/sec
D	=	Width of detector in direction of scan (cm)

G = C*A/100

G = (75*821)/100

G = 615.75

$$P(\mathrm{n} \geq 2) = 1 - (1 + \frac{(615.75*.109+23)15.9}{60*4})(e^{-\frac{(615.75*.109+23)15.9}{60*4}})$$

$$P(\mathrm{n} \geq 2) = 1 - (1 + 5.970235)(e^{-5.970235})$$

$$P(\mathrm{n} \geq 2) = 1 - (6.970235)(.0025536)$$

$$P(\mathrm{n} \geq 2) = 1 - (.017799)$$

$$P\ (\mathrm{n} \geq 2) = 98\%$$

The scan rate to achieve a ≥ 95% probability of detection while passing over the contaminated area of 75 dpm/100 cm^2 was **1.5 inches/second**. If the surveyor detected two counts while performing the alpha scan surveys, the surveyor stopped, acquired a timed count, and investigated 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 was 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 scans and static measurements had an efficiency determined by a licensed calibration facility using NIST traceable sources. In addition, ISO 7503-1, 1988 methods were 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 used 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 the FSS, for both total and removable activity measurements. Radionuclides used for efficiency determination were:

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

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 operational parameters such as scan rate, count time, and the associated MDC.

Table 8-1 - Instrument Specifications

Detector Model	Detector Type	Detector Area (cm^2)	Meter Model	Total Efficiency (%)
43-93[3] Small Area Probe Alpha	Alpha Scintillation	100	Ludlum 2224-1	10.54
43-93[4] Small Area Probe Beta	Beta Scintillation	100	Ludlum 2224-1	5.95
BP19DD[2] Small Area Probe Beta	Beta Scintillation	100	Ludlum 2350-1	4.41
43-37-1 Large Area Probe Alpha	Gas Flow Proportional	821	Ludlum 2350-1	10.8
43-37-1 Large Area Probe Beta	Gas Flow Proportional	821	Ludlum 2350-1	9.5
Beckman LS6500 (or Equivalent)	Liquid Scintillation	N/A	N/A	65 (H-3) 95 (C-14)
Protean	ZnS+Dual Phosphor	N/A	N/A	8 (Th-230) 6 (Tc-99)

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

Table 8-2 - 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[4]	2.5	N/A	60	476	4,715 /148,000	3.2
Surface Scans Large Area Probe Alpha	43-37-1	1.5	N/A	60	23	40/150	26.7
Surface Scans Large Area Probe Beta	43-37-1	5	N/A	60	1,191	597/148,000	4.0
Total Surface Activity Small Area Probe Alpha	43-93[5]	N/A	60	60	1.1	74.7/150	50
Total Surface Activity Small Area Probe Beta	BP-19DD	N/A	60	60	476	1,758/148,000	1.2
Removable Activity	Beckman LS6500	N/A	60	60	18 (H-3) 8.4 (C-14)	35.6 (H-3)/148000 17.3 (C-14)/148000	<.01% For all channels
Gross Alpha Removable Activity	Protean	N/A	900	3,600	0.3	6.7/15	44.6
Gross Beta Removable Activity	Protean	N/A	900	3,600	44.70	68.8/150	45.8

8.6 Minimum Detectable Concentration (MDC) Calculations

Philotechnics analytical sheets are included as Appendix D, E, F, and G, 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, 43-93, or 43-68, with similar detector areas and efficiencies**

9 PLANNED DECOMMISSIONING ACTIVITIES

9.1 Radiological Scoping/Characterization Surveys

Radiological Scoping/Characterization was designed to identify areas of elevated activity that require remediation. Surveys consisted of scans surveys for building structural surface total activity, and smears for removable activity measurements. Surveys were designed to meet the same Data Quality Objectives (DQOs) as the FSS, such that Characterization data could be used as FSS data where possible.

9.1.1 Building Structural Surfaces

In order to identify locations of elevated activity, the building surfaces survey protocol consisted of performing scan surveys of 50% 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 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 was 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 and Remedial Action Surveys

Decontamination is the physical or chemical process of reducing and preventing the spread or potential exposure from contamination. No decontamination was required for this decommissioning project. Remedial action surveys are conducted in support of remediation activities, no remediation activities were performed.

10 PHILOTECHNICS MANAGEMENT ORGANIZATION

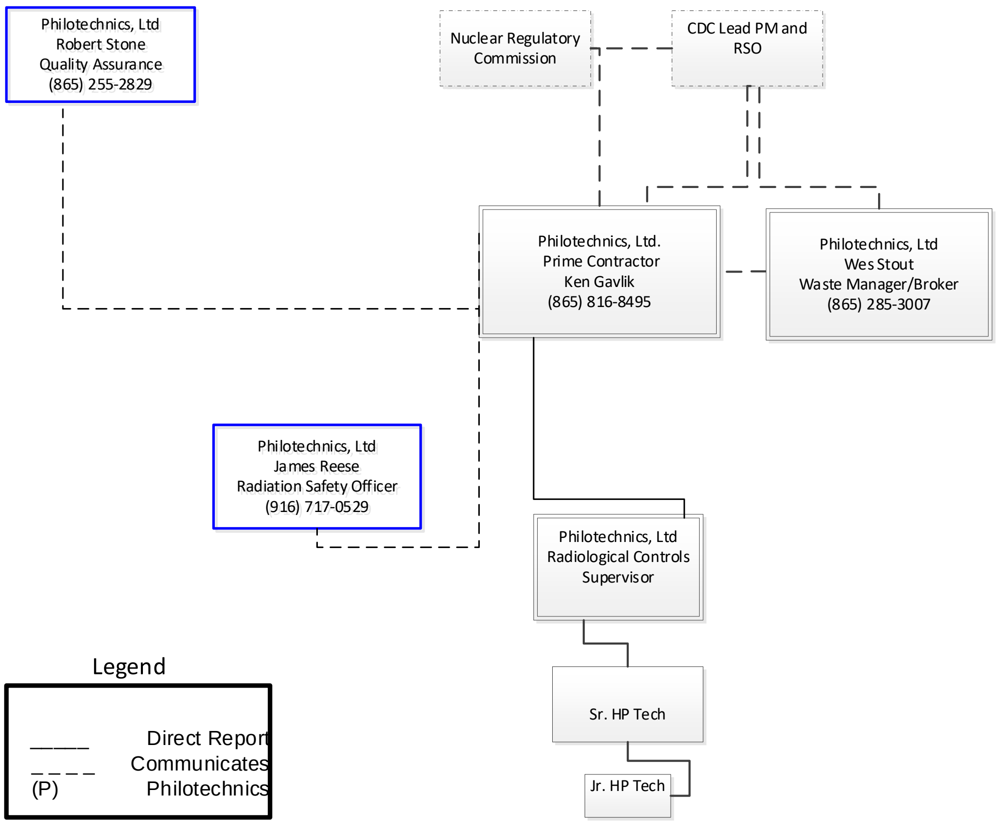
The following management structure was utilized for administration and implementation of the decommissioning.

Figure 10-1 – Team Experience on Similar Work

Name, Title / Role	Experience Highlights
Ken Gavlik • Project/Program Manager • Field Management • Radiological Controls Supervisor • Health Physics • Waste Disposition	• Nearly two decades of experience in radiation protection and radiological services, applied Health Physics, facility decommissioning, radiological and hazardous waste management. regulatory support and environmental compliance • BS with concentrations in Nuclear Engineering Technology and Radiation Protection • MBA • US Navy Submarine Veteran of the Navy Nuclear Power Program
Wesley Stout • Radiological Engineering • Waste Manager/Supervisor • Waste Disposition	• Over 25 years of experience as a project/waste manager with experiences in radiological D&D, industrial safety, and waste management. • Radiological and engineering lead for characterization of waste streams, identification of viable treatment/disposal alternatives and for federal client waste management technical support, including the first Cask disposal at WCS Federal facility. • BS • Developed entire waste management program for Tandem Van de Graaff accelerator project
Robert Stone • Quality Assurance/Site Safety Officer/Manager	• Ove 20 years safety experience • 10 years management experience • Safety oversight for DOE, DOD, USACOA, and Commercial • 36 Hour DoT Shipper • 40 Hour OSHA • RWI, RWII/GETII
William Button, • President • Support	• President Philotechnics • Over 40 years • Over 20 years management • BA • MS Nuclear Engineering • Post Graduate research Nuclear Chemistry
James Reese, CHP, RRPT • Program Manager • CHP • Support	• California License RSO • 35 years of experience in Health Physics; 5 years as Branch Chief for NRC Regions V and IV • Extensive experience in working and communicating with NRC, State, Local regulators for licensing and decommissioning activities • Extensive experience in radiological remediation and MARSSIM site closures at complex and sensitive sites. • President of the Decommissioning Section of the Health Physics Society • Extensive experience working the universities to remove historically sensitive buildings from radioactive material licenses through the implementation of MARSSIM • MARSSIM Certification

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
15 Radioactive Laboratories
Radiological Decommissioning/ License Termination Project - Organizational Chart



11 CDC DECOMMISSIONING TASK MANAGEMENT

Decommissioning was conducted in accordance with the CDC DP. All contractor activities were approved and overseen by the CDC to ensure compliance with the facility NRC RML. Decommissioning tasks were performed according to written plans and procedures to ensure they provided adequate worker protection and complied with the CDC NRC RML.

The following CDC management organization provided relevant data and support, and made final decisions for the decommissioning effort:

- CDC Radiation Safety Officer (RSO) – Narvaez Simpson is the OSSAM Senior Health Physicist and the current CDC RSO. Prior to Narvaez taking this position, Paul Simpson OSSAM Senior Health Physicist held this position as RSO at CDC from 1981 until 2016. The RSO 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 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.

Radiation Work Permits (RWP)s were used to accomplish remediation activities. The RWP contained 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 were developed for each survey unit that contained specific survey instructions. Survey package preparation and completion were approved by the PM and RSO to ensure all survey requirements and DQOs were met.

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12 PROJECT TRAINING REQUIREMENTS

The CDC provided personnel with site specific Contractor Orientation Training.

12.1 Radiological Training

Basic Radiation Worker training was completed and documented prior to arrival on site for Philotechnics personnel. The PM maintained a copy of each individual's certification on site in the project file.

12.2 Project Specific Training

Prior to project start-up, personnel attended an initial project-specific training session conducted by the PM. The training session included 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 were held at the beginning of the work shift. The purpose of this meeting was to discuss project status, potential problem areas, general safety concerns, and to reiterate CDC DP requirements.

12.4 Visitor Orientation

The project had no visitors.

12.5 Transportation Training

The project had no transportation required.

13 RADIATION SAFETY AND HEALTH AND SAFETY PROJECT PLANS

Site-specifc Radiation Protection Plan (RPP) and ssHASP were prepared and implemented for all on-site activities.

14 ENVIRONMENTAL MONITORING AND CONTROL

All licensed project activities were performed indoors, under stirct controls, and in a manner that did not present an elevated risk of environmental releases above normal operations.

15 RADIOACTIVE WASTE MANAGEMENT PLAN

Although no waste was generated, as a conservative measure a site specific Waste Management Plan was prepared for all on-site activities.

16 QUALITY ASSURANCE PROJECT PLAN

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

17 FINAL STATUS SURVEYS

FSS were performed to demonstrate that residual radioactivity in each survey unit satisfied the predetermined criteria for release for unrestricted use. FSS were 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 was used as FSS data to the maximum extent possible in order to minimize overall project costs.

17.1 Background Determination

A suitable reference background area was available and selected for determining ambient background for the radiological surveys. Additional material specific background measurements were aqcuired; however, in maintaining with the conservativeness of this FSS were not used to calculate sample location results. This decision was based on the guidance provided in NUREG-1505, “A Nonparametric Statistical Methodology for the Design and Analysis of Final Decommissioning Surveys.”

For this FSS, the use of reference background measurements or paired background was not necessary, as material and ambient background levels were not present in significant levels in comparison to the release criteria. Therefore, for conservatism in the survey design, ambient background levels were determined by taking ten (10) one-minuted timed counts for beta 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 was 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 was subtracted from gross measurement count rates (in cpm) to determine the net measurement count rate. The mean ambient background was also used to calculate the actual survey MDCs and the associated count errors. The number of measurements required for each material type was calculated for the Sign test.

For total surface activity measurements, ambient background levels were generally determined by performing a sixty-second (60) timed count with the probe at waist level and away from survey unit surfaces. Reference background was subtracted from each total activity gross measurement. Material background, the contribution from naturally-occurring radioactivity in building structural materials, was part of the ambient background in the matched reference background areas and survey units.

Background corrections were 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 were 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 were taken to achieve an MDC_{static} of less than 50% of the release criteria 148,000 dpm/100cm^2 Beta and 150 dpm/100cm^2 Alpha.
- Scanning was conducted at a rate to achieve an MDC_{scan} of less than 50% of the release criteria 148,000 dpm/100cm^2 Beta and 150 dpm/100cm^2 Alpha..
- Smear counting was conducted to achieve an MDC of less than 14,800 dpm/100cm^2 Beta and 15 dpm/100cm^2 Alpha.
- Individual measurements were made to a 95% confidence interval.
- Decision error probability rates were 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 were 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 were conducted at a rate of 5%.

17.3 Area Classifications

Based on the results of the HSA, 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 FSS, 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 were not absolute.

17.3.1 Class 1 Areas

Areas with the highest potential for contamination and met 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.

- There were no Class 1 Survey Units

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 all labs 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.

- There were no Class 3 Survey Units

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 2
Structures	100 m^2 to 1,000 m^2
Land	2,000 m^2 to 10,000 m^2

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

Location	Survey Unit	Initial Classification
Bldg 17, Lab 4085	1	Class 2
Bldg 23, Lab 10-471	2	Class 2
Bldg 110, Lab 4207C	3	Class 2
Bldg 15, Lab SB401	4	Class 2
Bldg 15, Lab SB101	5	Class 2
Bldg 15, SSB401	6	Class 2
Bldg 23, Lab 10-624	7	Class 2
Bldg 23, Lab 10-654	8	Class 2
Bldg 23, Lab 10-439	9	Class 2
Bldg 18, Lab B703B.3	10	Class 2
Bldg 18, Lab 5-412	11	Class 2
Bldg 17, Lab 5130	12	Class 2

17.5 Survey Methodology

Determination of Class 2 survey unit sample locations was 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 provided an unbiased method for obtaining measurement locations used in the statistical tests. Class 2 survey units had the highest potential for small areas of elevated activity so the areas between measurement locations was adjusted to ensure that these areas could be detected by scanning techniques.

All use labs were classified as Class 2 for conservatisms. Philotechnics utilized a square grid system for the Class 2 area. Judgmental sample locations were taken. For FSS, the starting point was determined using a random number generator.

17.6 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 2 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
2	50%	50% of all accessible areas (holders/casing for the instrument detectors normally prevent direct scans along the intersection of walls, floors and ceiling)

The scan survey percentage was chosen in order to provide a comprehensive survey of the impacted areas and provided confidence there was no contamination present above the DCGLs. In the event of any elevated activity noted from the survey, the location would have been marked, additional measurements taken to quantify the activity, and any decontamination determined to be appropriate conducted prior to a re-survey. The probe was 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 were 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 was 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 was also not dependent on the technician attempting to hold the detector at a predetermined distance, while cautiously ensuring they did 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 were collected in a random-systematic grid in accordance with the MARSSIM approach. Removable contamination measurements were performed at each total activity measurement location.

Minimum number of samples for FSS is calculated below.

17.7 Total Activity Direct or Static Measurements

Static measurements for total surface activity were completed using a timed count on the surface to be measured at each specified sample location. A systematic grid with a random starting point was used to determine the survey locations in the Class 1 areas. The probe was held as close to the surface as practicable to determine a count rate in counts per minute. Scaler count times were 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:

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

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

17.7.1 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 were 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.

17.7.2 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, Δ/σS 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 148,000 dpm/100 cm^2 beta-gamma

LBGR = Lower bound of the gray region. MARSSIM recommended the LBGR was initially set arbitrarily to 1/2 the $DCGL_W$. Therefore LBGR was initially set to 74,000 dpm/100 cm^2 beta-gamma .

σ_S = The largest standard deviation of the residual radioactivity in the survey unit set to 2,015 dpm/100 cm^2 beta-gamma.

Utilizing the inputs from above, Relative Shift for beta-gamma is provided in the figure below using the largest standard deviation, between Floor and Lower Wall, and Scoping Survey results, and a .25 surface efficiency for beta-gamma measurement results, for added conservatism in the survey design. The largest standard deviation was:

- Beta-Gamma: 2,015 dpm/100 cm^2 Floors and Lower Walls

Equation 17-1 – Relative Shift Beta-Gamma

$$\Delta/\sigma_S = \frac{148{,}000 - 74{,}000}{2{,}229}$$

$$\Delta/\sigma_S = 33.20$$

The most conservative value for Relative Shift, using the most conservative inputs from the FSS, was 33.20. The value for Relative Shift was not between one (1) and three (3), therefore relative shift was adjusted to three (3).

17.7.3 Determination of Acceptable Decision Errors

A decision error was 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 DQOs.

17.7.4 Determination of Number of Data Points

For the purposes of FSS, it was assumed that the contaminant was not present in background at significant levels compared to the release criteria. Therefore, material-specific background was ignored and was 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, was determined from MARSSIM Table 5.5, based 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 was as follows in **Equation 17-1 – Number of Samples Required per Survey Unit** below.

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Equation 17-1 – Number of Samples Required per Survey Unit

$$N = \frac{(1.645 + 1.645)^2}{4(0.998650 - 0.5)^2}$$

$$N = 10.88$$

Note: Percentiles $Z_{1-\alpha}$ and $Z_{1-\beta}$ were determined from MARSSIM Table 5.2. SignP was determined from MARSSIM Table 5.4 using the most coservative Relative Shift noted above, and rounding down for conservatism to a Relative Shift of 3.0.

MARSSIM recommended 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 was 14 using the calculation above, and 14 using MARSSIM Table 5.5. So to ensure the conservatism of the survey design, the number of samples the survey design required was 14 sample locations per surevy unit, and the spatial independence of the sample distribution included floor area only for Class 2 areas, and not walls, thereby increasing the number of samples and sample density in the survey unit. This, in turn, increased the number of samples on the areas with the highest probability of contamination - horizontal surfaces. The Class 2 areas included a minimum of 19 total samples, and increase of 90% of the number of sample locations required.

17.7.5 Determination of Sample Locations

Determination of Class 2 survey unit sample locations was 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 provided an unbiased method for obtaining measurement locations to be used in the statistical tests. Random starting location was 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 generated was multiplied by maximum “x” and maximum “Y” coordinates from survey location maps to provide the “x” and “y” coordinates for the random start location.

Class 2 survey units had potential for small areas of elevated activity, so the areas between measurement locations could be adjusted to ensure that these areas can be detected by scanning techniques.The use of a systematic grid allowed the decision-maker to draw conclusions about the size of the potential areas of elevated activity based on the area between measurement locations.

Survey protocols for all areas are summarized in the table below.

Table 17-4 - Survey Sample Placement Overview

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

17.7.5.1 Determining Class 1 and Class 2 Sample Locations

In Class 2 survey units, the sampling locations were 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 was determined from MARSSIM equation 5-8:

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

Where:

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

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

Equation 17-2 – Example Sample Spacing Interval for a typical Lab Survey Unit

$$L = \sqrt{\frac{300}{15}} \text{ for a square grid}$$

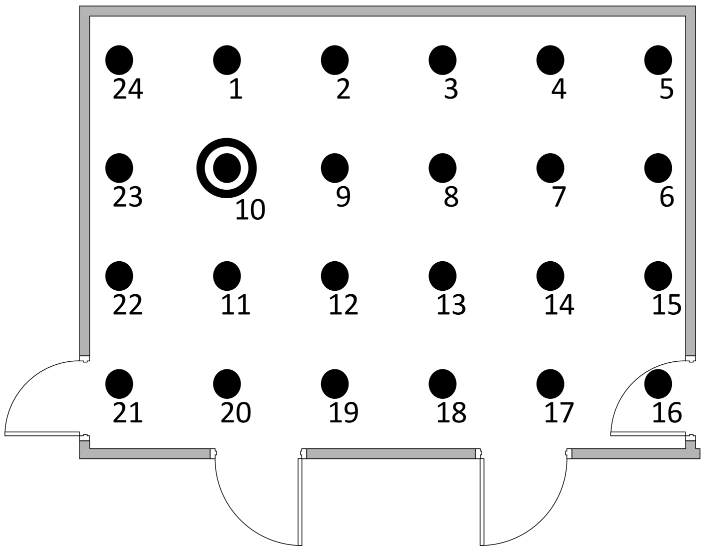
$$L = \sqrt{20} \text{ for a square grid}$$

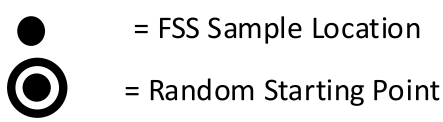
$$L = 4.5 \text{ for a square grid}$$

-or 4’6” spacing for square grid

Note: For conservatism, to increase sample distribution, only floor area square footage was used to calculate sample spacing intervals for 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 Error! Reference source not found. below.

Lab 4085
EXAMPLE





17.8 Removable Measurements Building Structures and Systems

Removable contamination measurements (smears) were collected on building structural surfaces at each sample location. Each smear encompassed an area of approximately $100cm^2$. If an area of less than $100cm^2$ was 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 was 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 were from "clean" surfaces due to Philotechnics pre-survey cleaning. Per McFarland's data for filter paper, alpha particle counting efficiency was 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 was 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)

Smear samples taken at the CDC were 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 and C-14 in Channel B. Scintillation standards were used to determine if the scintillaiton counter was operating within normal parameters. The efficiencies for the scintillation counter were 64.0% for H-3 and 95.4% for C-14 for the scoping survey, and efficiencies current at the time of the FSS were used.

17.8.1 Survey of Building Mechanical System Internals

Survey design for systems was out of the scope of MARSSIM; however, for added conservatism systems were included in FSS. According to interviews with the RSO, no RAM was released to the sanitary sewer system in the impacted labs, and sanitary sewer disposal on campus was generally limited to small total activities or radionuclides that had decayed at least 10 half-lives..

17.9 Survey Investigation Levels

Investigation levels were used to flag locations that required special attention and further investigation to ensure areas were properly classified and adequate surveys were performed. No locations were identified. The survey investigation level for each type of measurement is listed by classification in **Table 17-5 - Survey Investigation Levels** below.

Table 17-5 - 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

17.10 Unity Calculations

Unity was applied to each sample location in Survey Unit 3 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 ensured that, regardless of the radionuclide distribution in a particular location, the dose limit of 25 mrem per year would not be exceeded as long as the sum of fractions shown above was less than 1.

18 DATA QUALITY ASSESSMENT AND INTERPRETATION OF RESULTS

The statistical guidance contained in Section 8 of MARSSIM was used to determine if areas were below cleanup criteria or additional surveys or sample measurements were required.

18.1 Preliminary Data Review

A preliminary data review was performed for each survey unit to identify any patterns, relationships, or anomalies. Additionally, measurement data was reviewed and compared with the DCGLs and investigation levels to confirm the correct classification of survey units. All calculations of means, standard deviations, minimum and maximum values, and comparisons between survey data and investigation levels are presented in

Table 18-1 – Calculated Values for FSS Mean, Standard Deviation, Min and Max.

18.2 Survey Documentation

Each survey unit was surveyed under survey instructions from the PM which specified the survey protocol to be followed. The survey instructions ensured the DQOs were met:

- Survey protocol instructions such as the number of samples, sample spacing, sample locations, areas to be scanned, etc.
- General survey requirements
- Random number generations to determine survey locations
- Instrumentation to be used and associated MDCs, count times, and scan rates
- Scaled survey unit maps detailing survey locations and placement methodology
- Recommended survey sequence
- Survey data sheets

To ensure proper data management and organization, each static and removable contamination measurement location was 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. Breakdowns of the location code and specific code components are provided below. Each sample will be identified as follows:

WWW: ***Up to 3-character designation of the facility (for example, "CDC")***

XX: ***Up to 2-character designation of the survey unit (for example, "01")***

YYYY: ***Up to 4-character designation of the surface type (for example, "CON" represents concrete, "TER" for terracotta tile/fire brick, etc.)***

ZZZ: ***3-character designation of consecutive sample number (for example, 004)***

For example, in the sample identification number CDC-01-CON-004, "CDC" represents the facility (Centers for Disease Control and Prevention), "01" represents the survey unit,

“CON” represents concrete material and “004” represents the sample collected at location 4.

18.3 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

Additionally, all FSS data was entered into the FSS data sheets. This provided the means to sort survey data, verify activity calculations, and to compute MDC and counting errors.

18.4 Nuclide Verification

As an additional check on assumptions made during the planning phase, removable samples were specifically analyzed for energies of Carbon-14 and H-3. Added to these was a conservative qualitative check for the presence of any other nuclides of interest. Essentially a third channel of the LSC, which encompassed energies from 0-2000 keV would be used for any elevated removable or total activity measurement. The area of interest for the third channel included the peak energies of all nuclides above the 156keV. Increased count values in this range could indicate the presence of additional nuclides. For purposes of the FSS, no removable activity measurement result for H-3, C-14, or beta or alpha emitters from the planchet counter exceeded the DCGL and no removable contamination was detected inside the facility.

18.5 Determining Compliance for Surfaces and Structure Surveys

Scan surveys were completed for all survey units at the prescribed coverage. Removable activity measurements were compared directly to the applicable investigation levels and DCGLs to determine if an area required further actions or surveys. All removable activity measurements collected during the FSSs were less than the applicable investigation levels and significantly less than the established DCGL. Elevated activity detected during characterization surveys was remediated as discussed in **Section 9.2 “Decontamination/Dismantlement”.** These locations were not included in the FSS unless a random or systematic location fell on these locations.

All total surface activity measurements were compared directly to the DCGLs and investigation levels to determine if an area required further surveillance. All total surface activity measurements collected during FSSs were less than the DCGLs for total surface activity. No FSS measurements exceeded the investigation level for the applicable DCGLs. Due to the use of ambient backgrounds for all FSS results for conservatism, and not utilizing materials specic backgrounds, results were artificially elevated.

The table below details the calculated values for the mean, standard deviation, minimum, and maximum values for the surface and structures survey units.

Table 18-1 – Calculated Values for FSS Mean, Standard Deviation, Min and Max Ambient Background ONLY

Survey Unit	Alpha dpm/100cm^2				Beta dpm/100cm^2			
	Mean	Std Dev	Min	Max	Mean	Std. Dev	Min.	Max.[6]
1	N/A	N/A	N/A	N/A	2,932	908	831	3,958
2	N/A	N/A	N/A	N/A	467	280	-256	821
3	-1	9	-10	18	169	221	-239	624
4	N/A	N/A	N/A	N/A	**6,418**	1,704	2,649	**8,642**
5	N/A	N/A	N/A	N/A	5,387	**2,229**	831	7,455
6	N/A	N/A	N/A	N/A	5,211	2,076	2,041	7,926
7	N/A	N/A	N/A	N/A	-271	640	**-1,708**	814
8	N/A	N/A	N/A	N/A	-330	611	-1,675	713
9	N/A	N/A	N/A	N/A	-565	501	-1,574	427
10	N/A	N/A	N/A	N/A	4,967	1,131	2,579	6,379
11	N/A	N/A	N/A	N/A	1,335	1,031	-262	4,832
12	N/A	N/A	N/A	N/A	3,521	1,232	612	5,219

[6] Maximum values were determined using conservative ambient background, not material specific backgournds.

Table 18-2 – Calculated Values for FSS Mean, Standard Deviation, Min and Max Material Background INCLUDED

Survey Unit	Alpha dpm/100cm^2				Beta dpm/100cm^2			
	Mean	Std Dev	Min	Max	Mean	Std. Dev	Min.	Max. [7]
1	N/A	N/A	N/A	N/A	2,674	1,016	343	3,958
2	N/A	N/A	N/A	N/A	467	280	-256	821
3	-1	9	-10	18	169	221	-239	624
4	N/A	N/A	N/A	N/A	293	1,881	-1,342	4,602
5	N/A	N/A	N/A	N/A	839	1,207	-1,091	3,369
6	N/A	N/A	N/A	N/A	1,935	1,355	-789	4,261
7	N/A	N/A	N/A	N/A	-271	640	-1,708	814
8	N/A	N/A	N/A	N/A	-330	611	-1,675	713
9	N/A	N/A	N/A	N/A	-565	501	-1,574	427
10	N/A	N/A	N/A	N/A	-301	**2,222**	**-2,050**	**5,169**
11	N/A	N/A	N/A	N/A	1,335	1,031	-262	4,832
12	N/A	N/A	N/A	N/A	**3,521**	1,232	612	5,219

18.6 Verification of Number of Samples for Surface & Structures

A minimum number of samples were needed to obtain sufficient statistical confidence that the conclusions drawn from the samples were correct. The number of samples depended on the relative shift (the ratio of the concentration to be measured relative to the statistical variability of the contaminant concentration). The minimum number of samples is obtained from MARSSIM tables or calculated using equations in Section 5 of MARSSIM. For this project, we used the data from Philotechnics scoping and characterization surveys to estimate the relative shift. To calculate the actual relative shift, data from the FSS was used. Once the actual relative shift was calculated, the number of samples required by MARSSIM was compared to the actual number of samples collected.

[7] Maximum values were determined using conservative ambient background, not material specific backgournds.

As an additional conservative measuure, the number of samples required was greatly increased by CDC and Philotechnics as an added ALARA measure.

18.7 Assessment and Interpretation of Survey Results

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

18.7.1 Preliminary Data Review

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

The following preliminary data reviews were performed:

- Calculations of the survey unit mean, median, maximum, minimum, and standard deviation for each type of reading and for unity.
- 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.

18.8 Determining Compliance

For Class 1, 2, and 3 areas, 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 were less than the DCGL, performance of the Sign test was not necessary because the survey unit will pass the Sign test.

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

Additionally, to demonstrate compliance, the maximum total activity concentration result in dpm/100 cm^2 for the most limiting ROC (C-14 for the highest TEDE per dpm/100 cm^2 of surface contamination), using the most conservative ambient background, was converted to a TEDE in mrem/year utilizing NRC DandD Version 2.4.0, and a comaprison of DVS in dpm/100 cm^2 to TEDE of 25 mrem/year provided in NUREG 1757 Volume 1, Appendix B, Table B-1, "*Acceptable License Termination Screening Values of Common Radionuclides for Building-Surface Contamination*" for the most conservative ROC, which equates to 3.7E+06 dpm/100 cm^2.

The maximum FSS total activity surface contamination result of **8,642 dpm/100 cm^2** results in a maximum as left radionuclide concentrations equal to a peak **TEDE associated with each survey unit of:**

- **6.40E-02 mrem/year** calculated using NRC DandD Version 2.4.0
- **5.84E-02 mrem/year** calculated using NUREG 1757, Volume 1, Appendix B, Table B-1

In addition, to demonstrate compliance with alpha emitters, the maximum total activity result in dpm/100 cm^2, using the most conservative ambient background, was converted to a Total Effective Dose Equivalent (TEDE) in mrem/year. This was done by utilizing an RESRAD-BUILD Version 3.50 software model. The model was run utilizing conservative parameters and the maximum FSS total activity result as the input. The maximum FSS total alpha net activity result of **18 dpm/100 cm^2** results in a maximum as left radionuclide concentrations equal to a peak **TEDE was 2.10E-03 mrem/year at year five. M**odel output provided in **Appendix I** associated with Survey Unit 3.

18.9 Mechanical System Survey Data Analysis

Survey design for systems was out of the scope of MARSSIM; howevcr for added conservatism, systems were surveyed as part of the FSS.

19 FINANCIAL ASSURANCE

Decommissioning was completed within 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

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- CDC Radioactive Materials License Number #10-06772-01