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FINAL REPORT

BUILDING 150/UNDERGROUND STORAGE TANK DECOMISSIONING PROJECT

BETHESDA NATIONAL NAVAL MEDICAL CENTER

Project No. USN 2000-035

Revision 3
January 6, 2004

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NEW WORLD TECHNOLOGY

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Approvals,

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TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

ACRONYMS & ABBREVIATIONS

AOC	Areas of concern
ALARA	As Low As Reasonably Achievable
Bkg	background
BRAC	Base Realignment and Closure
Cal	Calibration
CFR	Code of Federal Regulations
Ci	Curie
cm	Centimeter
cm^2	square centimeters
cpm	counts per minute
Co-60	cobalt-60
DAC	Derived Air Concentration
DCF	Duratek Consolidation Services Facility
DOT	Department Of Transportation
dpm	disintegrations per minute
$dpm/100cm^2$	disintegrations per minute per 100 square centimeters
eff	efficiency
FOP	Field Operating Procedures
g	gram
h	hour
H-3	hydrogen-3 (Tritium)
Inst	instrument
JMC	U.S. Army Joint Munitions Command
IAW	In Accordance with
LLD	Lower Level of Detection
LSC	Liquid Scintillation Counting
MARSSIM	Multi Agency Radiation Survey Site Investigation manual
MDA	Minimum Detectable Activity
MDC	Minimum Detectable Concentration
MDCR	Minimum Detectable Count Rate
mCi	millicurie
mm	millimeter
mrem	millirem
NIST	National Institute of Standards and Technology
NRC	Nuclear Regulatory Commission
NUREG	Nuclear Regulatory Guide
pCi	picocurie
ppm	parts per million
NNMC	Bethesda National Naval Medical Center
NWT	New World Technology

JMC	Joint Munitions Command
RASO	Naval Sea Systems Command Detachment, Radiological Affairs Support Office
RPO	Radiation Protection Officer
S/N	Serial Number
SOP	Standing Operating Procedure
USA	U.S. Army
USN	U.S. Navy
UST	Underground storage tank
μR/hr	microroentgen per hour
μCi	microcurie

1.0 EXECUTIVE SUMMARY

New World Technology (NWT) was contracted by the U.S. Army Joint Munitions Command (JMC) to perform the decommissioning of Building150, the removal of two underground storage tanks, decontamination of Building 150 interior concrete surfaces, the removal Building 150 contaminated sewer lines, and Final Status Surveys of the interior of Building 150 and land areas surrounding Building 150 following decontamination efforts.

Building 150 and the two underground storage tanks (UST's) were located at the Bethesda National Naval Medical Center (NNMC) facility located in Bethesda, Maryland. See Figure 1 for the location of Building 150 and the two UST's.

The facility used Building 150 back in the 1950s for conducting gamma exposure experiments on animals. The radionuclide of concern for Building 150 was Cobalt-60 (Co-60). Co-60 was also the radionuclide of concern for the Building 150 sewer lines and the land areas surrounding Building 150.

Tritium (H-3) was the radionuclide of concern for the two underground storage tanks (UST's) located between Buildings 17B and 155. The holding capacity of the two UST's was approximately one thousand gallons each. The UST removal and Final Status Survey effort included surveys for gross alpha, beta and gamma emissions.

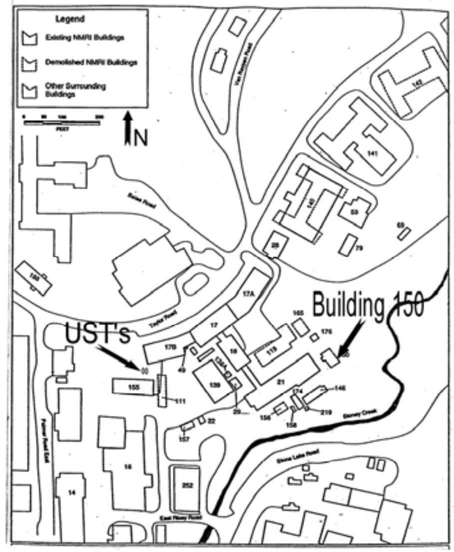
The Navy had previously confirmed the existence of Co-60 contamination in Building 150. Based upon scoping surveys previously performed it was assumed that not more than 5% of the interior building surfaces were contaminated above the release limit.

NWT conducted a scoping survey in March of 2002 for the UST's. It was determined the tanks residual contents in the tanks were contaminated with very low levels (~127 pCi/g) of Tritium (H-3).

A radiological final status survey was conducted in Building 150 and the land area outside of Building 150 to provide data for unrestricted release. Nuclear Regulatory Commission (NRC) guides, NUREG-1575, NUREG-1505, and NUREG-1507 were used as guidance in designing and conducting final status surveys of the building. The final status surveys included direct surface contamination surveys for beta radiation, direct exposure rate measurements, gamma scan surveys, and soil sample analysis. Survey results were statistically analyzed to determine if the residual radioactivity levels in the building and outside grounds (survey units) meet the release criterion or did not exceed natural background radiation levels.

Survey results were compared to a total effective dose equivalent (TEDE) not exceed 25 millirem per year (mrem/y) with the assurance that residual radioactivity has been reduced to levels that are “as low as is reasonably achievable” (ALARA). The final status survey found no evidence of small areas of elevated radioactivity and determined residual radioactivity in Building 150 and the land area outside of Building 150 to be less than the release criteria and similar to the background levels of radioactivity in a similarly constructed building and open land area (reference areas).

Figure 1. Building 150 and UST Location Map



2.0 PROJECT SCOPE

The scope of the project is summarized in Table 1 below.

Table 1. Project Summary Table

Area	Scope
Building 150 Interior	Decontaminate, Disposal, and Perform Final Status Surveys
Building 150 Floor Drains and Drain Lines	Remediate, Disposal, and Perform Final Status Surveys
Building 150 Outside Land Areas	Remediate, Disposal, and Perform Final Status Surveys
Underground Storage Tanks	Removal, Disposal, and Perform Final Status Surveys

3.0 FACILITY DESCRIPTION

3.1 Site Location and Description

Building 150 and the two underground storage tanks (UST's) were located at the Bethesda National Naval Medical Center (NNMC) grounds located in Bethesda, Maryland. See Figure 1 for the location of Building 150 and the two UST's.

3.2 Historical Use of the Building

Building 150 was constructed in the early 1950's to irradiate animals and determine the effects of ionizing radiation on the organ and cellular system. This was done as an economical alternative to transporting animals and attendants to the Bikini Islands atomic testing areas.

The radiation source was 2500 curies of Cobalt-60 in ceramic slugs arranged in tow circles and provided a dose rate of 200 rem per minute. There were several minor contamination incidents during routine maintenance between 1951 and 1962. These were probably due to cracks in the ceramic slugs. Workers were routinely monitored for internal deposition of radionuclides.

On April 18 1962 while undergoing routine internal monitoring, significant internal deposition was identified on HM2 W. Ross. Levels ranged from 0.06 to 0.12 mrem/h internal dose rate. An investigation revealed levels of 0.6 to 32 mrem/h inside of the building with widespread loose and fixed surface contamination. Decontamination was conducted and the Cobalt-60 source and contaminated materials were removed at a cost of $150,000.00. The building was then sealed until November of 1969.

The radionuclide of concern for Building 150 was Cobalt-60 (Co-60). Co-60 was also the radionuclide of concern for the Building 150 sewer lines and the grounds surrounding Building 150.

Tritium (H-3) was the primary radionuclide of concern for the two underground storage tanks (UST's) located between Buildings 17B and 155. The holding capacity of the two UST's was approximately one thousand gallons each. The UST removal and Final Status Survey effort included surveys for gross alpha, beta and gamma emissions. Anything distinguishable from background warranted further investigation.

The Navy had previously confirmed the existence of Co-60 contamination in Building 150. Based upon scoping surveys previously performed (information contained in References 19.1 and 19.2 of this report) it was assumed that not more than 5% of the interior building surfaces were contaminated above the release limit.

NWT conducted a scoping survey in March of 2002 for the UST's. It was determined that residual contents in the tanks were contaminated with very low levels (~127 pCi/g) of Tritium (H-3).

3.3 Current and Future Land Use

There are no current plans for the area that was previously occupied by the UST's. Building 150 is planned for demolition.

3.4 Previous Site Investigation

In March of 2002, NWT personnel performed a site investigation of the two UST's and Building 150 which included the following:

- Soil sampling around the exterior of the UST's
- Sampling of the residual material inside of the UST's
- Smoke testing of the UST's to identify line(s) leading to/from the UST's
- Performed vegetation removal of Building 150 exterior and site grounds
- Gamma scan walkover survey of the roof and exterior perimeter of Building 150
- Sampling of lead shot found on the roof of Building 150
- Soil sampling of the exterior perimeter of Building 150

The results and findings of that investigation are presented in Reference 19.4 of this report.

3.5 Site Conditions

It appeared as if the installation has arranged for decontamination efforts in the past based on information contained in Reference 19.2 of this report. The Navy has identified Co-60 contaminated areas in Building 150 based on information obtained from References 19.1 and 19.2 of this report. The floor drains in Building 150 were capped, and there was an elevated gamma dose (Navy confirmed it to be Co-60) from at least one of the drains based on information obtained from Reference 19.1 of this report.

The exterior of Building 150 and surrounding grounds were lightly overgrown with vegetation, as NWT cleared most of the vegetation in the site visit in March of 2002. The interior of Building 150 had debris and material covering the floor surfaces.

During an earlier site visit, NWT personnel made visual inspections of the interiors of the UST's through standpipes at the top of the tanks. The results of the visual inspection indicated that the tanks were empty excluding residual water and sludge (less than 10 total gallons) located in the bottom of the tanks.

4.0 RELEASE CRITERIA FOR UNRESTRICTED USE

4.1 Dose Based Release Limit

The Building 150 building interior surfaces and outside land areas are 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 millirem per year (mrem/y), and the residual radioactivity has been reduced to levels that are as low as is reasonably achievable (ALARA).

4.2 ALARA

Considering the conservatism that is factored into the default DandD scenario of “Building Occupancy,” and “Residential” the release criteria determined by this model does not need to demonstrate that these levels are ALARA. Sections 4.4 and 4.5 describes the dose model and “Building Occupancy” and ‘Residential” scenarios used to determine radionuclide specific release limits based on a TEDE of 25 mrem/y.

4.3 Radionuclides of Concern

The radionuclides of concern are Co-60 (Building 150 interior surfaces and outside land areas), and H-3 (underground storage tank area).

Table 2 lists the radionuclides of concern and the principal radiation (alpha, beta, and/or gamma) are identified for measurement purposes.

4.4 Dose Modeling for Radionuclides of Concern (Building Surfaces)

Radionuclide specific release criteria, derived concentration guideline levels (DCGLs), were derived using DandD Version 2.1 (NUREG/CR-5512, Vol. 2) based on the 25 mrem/y TEDE dose limit specified in Section 4.1. The model for the critical group was based on a default scenario of “Building Occupancy.” This scenario accounts for a worker in a commercial building to be exposed to residual fixed and removable surface contamination sources, on or within the structure following unrestricted release of the facility. This pre-defined model and generic screening parameters are used to calculate a conservative range of doses that the average worker could receive.

4.5 Dose Modeling for Radionuclides of Concern (Soil Concentrations)

Radionuclide specific release criteria, derived concentration guideline levels (DCGLs), were derived using DandD Version 2.1 (NUREG/CR-5512, Vol. 2) based on the 25 mrem/y TEDE dose limit specified in Section 4.1. The model for the critical group was based on a default scenario of "Residential." The residential scenario includes external and inhalation exposures and exposure from the following ingestion pathways: drinking water, food grown with contaminated irrigation water, food grown on contaminated soil, fish, and inadvertent ingestion of soil. Land-based foods considered are leafy vegetables, root vegetables, fruit, grain, beef, poultry, milk and eggs. Animal feeds include fresh forage, stored grain, and stored hay. This is the most conservative scenario as it assumes that persons living on the site can use the land for any purpose without land-use restrictions.

4.6 Release Criteria (DCGLs) for Radionuclides of Concern (Building Surfaces and Soil Concentrations)

The DCGLs for each radionuclide of concern are listed in Table 2 below. Each DCGL meets the release criteria for unrestricted use at 25 mrem/y. Conservatively, the soil and surface activity limits (DCGLs) are based on the upper end of the confidence interval for the resultant dose for each radionuclide of concern.

Table 2. Derived Concentration Guideline Limits (DCGL's) for Radionuclides of Concern

Radionuclide	Radiation	DCGL (dpm per 100 cm^2)	DCGL (pCi/g)	TEDE (mrem)
Co-60	Beta, Gamma	7,100	3.8	25
H-3	Beta	1,000 (removable)	110	25

5.0 RADIATION WORK PERMIT (RWP)

A Radiation Work Permit (RWP, NWT Form NWT-002) was prepared that specified the activities to be performed and all radiological safety requirements for the work. All personnel assigned to site work were required to read and understand the requirements prior to beginning work.

A copy of the RWP is presented in Appendix A.

6.0 BIOASSAY SAMPLING

Bioassay samples (urine) were collected (at the beginning and end of project) from NWT personnel to verify that no personnel exposure to radioactive materials had occurred. The samples were sent to NWT's laboratory in Livermore, CA for liquid scintillation analysis.

The results of the samples indicated that no internal exposure to radioactive material had occurred during the course of the project.

The laboratory results of the samples are presented in Appendix B.

7.0 UST REMOVAL ACTIVITIES

7.1 Pre Excavation Geoprobe Soil Sampling

Pre excavation geoprobe soil sampling of the soils around and above the two UST's was performed during NWT's previous site visit in March of 2002. The purpose of the soil sampling prior to removal of the two UST's was to provide data as to the physical condition of the two UST's and whether or not material inside of the UST's had leaked into the surrounding soils. Based upon the results of the soil sampling, the two UST's were intact and the contents of the

UST's had not leaked into the surrounding soils (see reference 16.4 of this report for the soil sampling results).

Figure 2 present a diagram of the pre excavation geoprobe sampling locations.

Figure 2. UST Geoprobe Sample Location Diagram

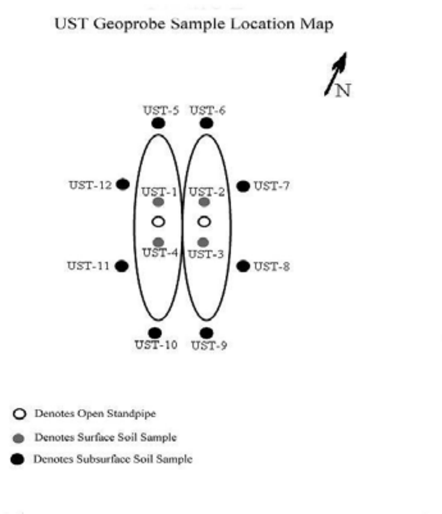


Table 3 and Table 4 present summaries of the geoprobe soil sample analysis results. The laboratory data for the soil samples is presented in this report in Appendix C.

Table 3. UST Geoprobe Soil Sample Liquid Scintillation Analysis Summary Table

Sample ID	Sample Depth	H-3 Results in pCi/g	MDC in pCi/g
UST-1A	Surface	0.070	0.21
UST-2A	Surface	0.006	0.11
UST-3A	Surface	0.045	0.14
UST-4A	Surface	-0.012	0.13
UST-5A	4-8'	-0.003	0.083
UST-6A	4-8'	0.004	0.075
UST-7A	4-8'	0.290	0.065
UST-8A	4-8'	0.052	0.069
UST-9A	4-8'	0.042	0.075
UST-10A	4-8'	0.047	0.076
UST-11A	4-8'	0.039	0.076
UST-12A	4-8'	0.005	0.077

Table 4. UST Geoprobe Soil Sample Gamma Spectroscopy Analysis Summary Table

SAMPLE		RADIONUCLIDE (Results in pCi/g)																						
ID	Depth	Ac-228	AM-241	Bi-212	Bi-214	Cd-109	Co-60	Cs-137	Eu-152	Eu-154	Eu-155	I-131	K-40	Mn-54	Na-22	Nb-95	Pa-234m	Pb-212	Pb-214	Th-227	Th-234	Tl-208	U-235	Zn-65
UST-1B	Surface	1.03	-0.28	1.7	0.54	1.9	-0.01	0.11	0.08	-0.02	0	0.03	15.1	-0.03	0.01	0.03	12	1.24	0.63	-0.23	1	0.32	-0.16	0.21
UST-6B	4-8'	1.33	-0.15	1.11	0.44	3.1	5.00 E-03	0.01	0.03	-0.05	0.14	-0.05	16.8	-0.02	-0.03	-0.02	7.1	1.41	0.62	-1.57	0.89	0.43	-0.09	0.02
UST-7B	4-8'	1.13	0.34	1	0.53	0.8	-0.11	-0.02	0.32	-0.06	0.07	5.00 E-03	20.4	-0.03	0.07	0.08	13	1.53	0.75	-0.24	0.5	0.38	-0.02	-0.24
UST-8B	4-8'	1.36	-0.07	1	0.59	0.92	-0.03	-0.01	-0.12	0	0.15	0.02	16	4.00 E-03	0.01	-1.00 E-03	0	1.35	0.72	0.1	0.9	0.37	0.05	0.01
UST-9B	4-8'	1.2	-0.16	2.43	0.5	1	6.00 E-03	-4.00 E-03	-0.02	-0.02	0.14	0.05	18.7	-0.03	-0.02	0.03	0.1	1.42	0.67	1.4	1	0.42	0.01	-0.11
UST-10B	4-8"	1. 9	0.09	13.7	0.73	1.4	4.00 E-03	0.06	-0.25	-0.11	0.01	-0.08	18	-0.1	-0.04	0.02	9	1.42	0.76	0.29	-0.3	0.59	-0.04	0.22
UST-11B	4-8'	1.49	0.04	2.1	0.79	1.6	0	-0.04	0.16	-0.26	0.06	0.01	19.7	0.03	-0.03	0.01	19	1.28	0.96	-0.1	1.3	0.44	-0.32	-0.11
UST-12B	4-8'	1.29	0.26	2.4	0.86	2.2	7.00 E-03	0.13	0.09	0.09	0.05	-0.05	17.5	-0.04	-0.12	7.00 E-03	-7	1.36	0.78	-0.23	1.06	0.45	0.04	-0.02
UST-2B	Surface	1.22	0.23	1.1	0.37	1.9	4.00 E-03	6.00 E-03	-0.02	-0.06	0.24	0.05	16.6	0.04	0.02	-0.02	7.7	1.38	0.76	-0.02	0.8	0.36	0.16	-0.03
UST-3B	Surface	1.26	-0.05	1.1	0.81	2.8	-0.05	0.17	0.17	-0.21	-0.12	-0.03	16.7	-0.02	0.06	0.02	-6	1.18	0.71	-0.13	-0.2	0.46	0.08	-0.33
UST-4B	Surface	1.07	-0.23	1.2	0.52	2	0.04	0.04	0.12	0.28	0.18	-0.02	14.3	0.02	-0.07	5.00 E-03	-7	1.09	0.62	0.1	1.9	0.41	0.01	-0.16
UST-5B	4-8'	1.17	0.05	1.6	0.52	3.2	-5.00 E-03	0.05	-0.05	-.033	0.06	-0.03	14.4	-1.00 E-03	0.01	0.06	10	1.22	0.67	-0.43	1.56	0.29	0.07	0.09
MDC		Ac-228	AM-241	Bi-212	Bi-214	Cd-109	Co-60	Cs-137	Eu-152	Eu-154	Eu-155	I-131	K-40	Mn-54	Na-22	Nb-95	Pa-234m	Pb-212	Pb-214	Th-227	Th-234	Tl-208	U-235	Zn-65
UST-1B		0.45	0.59	1.7	0.24	1.9	0.11	0.070	0.52	0.67	0.38	0.15	1.5	0.13	0.12	0.11	18	0.18	0.24	0.95	1.8	0.12	0.60	0.37
UST-6B		0.46	0.35	1.0	0.26	2.1	0.097	0.100	0.45	0.53	0.29	0.13	1.3	0.10	0.12	0.15	14	0.15	0.21	1.1	1.6	0.100	0.58	0.35

UST-7B		0.61	0.91	2.5	0.32	3.8	0.23	0.18	0.66	0.96	0.44	0.17	2.2	0.15	0.17	0.15	26	0.19	0.28	1.0	2.1	0.15	0.71	0.53
MDC		**Ac-228**	**AM-241**	**Bi-212**	**Bi-214**	**Cd-109**	**Co-60**	**Cs-137**	**Eu-152**	**Eu-154**	**Eu-155**	**I-131**	**K-40**	**Mn-54**	**Na-22**	**Nb-95**	**Pa-234m**	**Pb-212**	**Pb-214**	**Th-227**	**Th-234**	**Tl-208**	**U-235**	**Zn-65**
UST-8B		0.43	0.21	1.7	0.27	1.5	0.15	0.13	0.69	0.66	0.24	0.14	1.5	0.12	0.13	0.14	20	0.22	0.22	0.53	1.8	0.13	0.61	0.30
UST-9B		0.36	0.43	1.3	0.21	1.5	0.080	0.091	0.51	0.51	0.27	0.11	1.2	0.10	0.10	0.087	16	0.14	0.17	9.1	1.7	0.084	0.48	0.25
UST-10B		0.33	0.89	1.3	0.23	2.0	0.094	0.11	0.65	0.66	0.34	0.19	1.2	0.16	0.16	0.11	16	0.21	0.25	0.56	1.9	0.13	0.59	0.41
UST-11B		0.64	0.99	2.1	0.29	2.5	0.20	0.18	0.78	0.96	0.42	0.19	1.5	0.14	0.20	0.16	22	0.22	0.22	1.3	2.3	0.17	0.79	0.43
UST-12B		0.77	0.66	1.9	0.34	2.6	0.18	0.14	0.59	0.79	0.34	0.19	2.0	0.15	0.22	0.16	27	0.28	0.24	0.72	1.3	0.15	0.62	0.43
UST-2B		0.52	0.93	1.8	0.24	3.1	0.12	0.11	0.53	0.73	0.33	0.16	1.2	0.10	0.14	0.12	16	0.17	0.22	0.57	2.6	0.12	0.53	0.34
UST-3B		0.53	1.0	2.1	0.32	3.3	0.20	0.16	0.78	0.78	0.44	0.22	2.2	0.18	0.18	0.15	28	0.25	0.33	1.2	2.5	0.15	0.82	0.54
UST-4B		0.52	0.74	1.9	0.38	2.0	0.12	0.15	0.95	0.78	0.34	0.18	2.0	0.13	0.20	0.14	29	0.26	0.33	0.70	2.2	0.16	0.72	0.49
UST-5B		0.60	0.21	2.0	0.25	2.2	0.15	0.13	0.65	0.85	0.28	0.19	1.9	0.13	0.19	0.14	17	0.21	0.23	1.0	1.4	0.14	0.58	0.32

7.2 Removal of the UST's

a. The area directly above and adjacent to the two underground storage tanks was checked and marked for underground utilities (i.e. water, gas lines) prior to excavation beginning in the area.

b. A dig permit was issued by NNMC bas personnel prior to the start of excavation in the area. The dig permit is presented in Appendix D.

c. Barrier tape was placed around the area to prevent unnecessary personnel from entering the excavation site.

d. Soils were removed above and around the two underground storage tanks to expose the tanks prior to removing the tanks from the ground. This was accomplished using a backhoe.

e. Once the tanks were exposed. The inlet and outlet lines of the tanks were cut from the tanks. This was accomplished using the backhoe.

f. It was discovered during removal of the two UST's that the inlet and outlet valves to each tank were in the closed position. It appeared as if the inlet lines to the UST's originated from Building 17 and the outlet lines led to the oil/water separator located in the eastern portion of Building 155 (see Reference 19.4 of this report).

g. Water samples were previously obtained from the oil/water separator located inside of Building 155 during NWT site investigation in March of 2002. The results of the samples indicated very low tritium levels of 131 and 270 dpm.

h. The lines were broken upstream of the inlet lines, and downstream of the outlet lines in order to perform a radiological survey.

i. The inlet and outlet lines leading to and from the tanks were then surveyed for gross alpha, and beta emissions using appropriate radiological survey instrumentation. Swipes were taken and analyzed by Liquid Scintillation Counting (LSC) analysis for the presence of tritium. The results of the swipes and gross alpha, beta contamination surveys showed no detectable activity above background radiation levels. The results of the surveys are in Appendix E.

j. Since no contamination was detected inside of the lines, the lines leading to and from the tanks were then sealed with quick setting grout to prevent leakage of the lines residual contents into the environment.

k. Upon inspection of the tanks interior after removal from the ground it was found that there was no freestanding liquids inside of the tanks.

l. The tanks were then removed from the ground and the exterior of the tanks surveyed for gross alpha, and beta radiation. The results of the swipes, and gross alpha, beta contamination surveys showed no detectable activity above background radiation levels. The results of the surveys are in Appendix F.

7.3 Preparation of UST's For Transportation and Disposal

a. The UST's were then transported to a staging area behind Building 21.

b. A hole, approximately three foot by three foot in size was cut into the side of each of the tanks to gain access to the interior of the tanks using a chopsaw.

c. Residual material inside of the tanks was then vacuumed up using a HEPA filtered vacuum cleaner. The total amount of material vacuumed from both of the tanks was approximately fifteen gallons. The material removed from the tanks was then transferred from the vacuum cleaner to a B-12 box (strong type metal container ~ 44 ft^2 in size).

d. The interiors of the tanks were then surveyed for gross alpha, and beta emissions using appropriate radiological survey instrumentation. Swipes were taken and analyzed by Liquid Scintillation Counting (LSC) analysis for the presence of tritium. The results of the swipes and gross alpha, beta contamination surveys showed no detectable activity above background radiation levels. The results of the surveys are in Appendix G.

e. The tanks were then wrapped in plastic sheeting to await transportation to the disposal facility.

7.4 Final Status Surveys of UST Footprint Areas

a. The footprints of where the UST's were located was 100% gamma scan surveyed using a Ludlum Model 44-10 2" by 2" NaI detector coupled to a Ludlum Model 2350-1 Data Logger. No detectable activity above background radiation levels was detected during this survey.

b. Soil samples were obtained from the footprint areas where both tanks stood. The samples were sent to an offsite laboratory for liquid scintillation analysis for tritium. The results of the samples were well below the DCGL presented in Table 2.

c. Following the surveys and sampling, the hole left in the ground by the removal of the two UST's was covered with plastic sheeting to prevent water intrusion into the hole. The plastic was supported and the edges held in placed with tent stakes to help assist in water drainage. The hole was not backfilled until verbal permission to backfill the hole by RASO was given following review of survey and sample results of the open excavation.

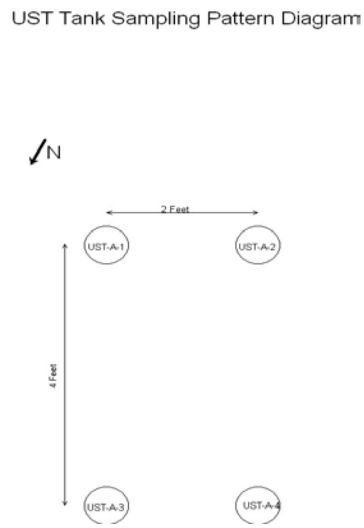
d. The laboratory data for the soil samples is presented in Appendix H.

e. Figure 3 and Figure 4 present diagrams showing the soil sample locations for the UST footprint samples.

f. Table 5 presents a summary of the UST footprint soil sample analysis.

g. Sample USTA-1 results (15.9 pCi/g) were above natural background levels but well below the DCGL. This sample was taken from an area where a small amount (~ 1 pint) of water leaked from the tank during removal.

Figure 3. UST-A



Footprint Sample Location Diagram

Figure 4. UST-B Footprint Sample Location Diagram

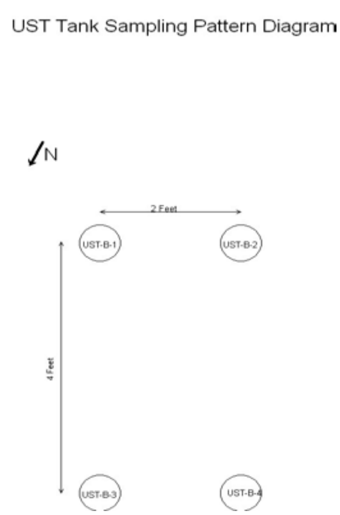


Table 5. UST Footprint Soil Sample Summary Table

Sample ID	H-3 Results in pCi/g	MDC in pCi/g
USTA-1	15.900	0.18
USTA-2	0.153	0.10
USTA-3	0.007	0.093
USTA-4	0.152	0.11
USTB-1	-0.015	0.12
USTB-2	0.028	0.14
USTB-3	-0.002	0.13
USTB-4	-0.025	0.14

7.5 Backfilling and Seeding of UST Footprint Areas

a. The footprint of where the UST's were located was backfilled with clean backfill soil provided from an off site source. A letter certifying that the backfill soil was free of hazardous constituents in presented in this report in Appendix I.

b. A 100% gamma scan survey was performed on the backfill soil prior to backfilling the UST footprint area. No detectable activity above background levels was detected. A copy of this survey is presented in this report in Appendix J.

c. Following backfill of the footprint area, a layer of topsoil was placed over the area and grass seed was planted over the area.

8.0 BUILDING 150 OUTSIDE LAND AREA REMEDIATION ACTIVITIES

8.1 Contaminated Soil Removal

a. During NWT's pervious site investigation in March of 2002, an area outside of Building 150 was identified as having soil concentrations above the DCGL.

b. This area was remediated by NWT personnel using hand tools.

c. The contaminated soil was transported and placed into the B-12 box using the bucket end of the backhoe. Approximately 40 cubic feet of soil was removed from this area.

8.2 Remediation Control Surveys

a. Following soil removal, was 100% gamma scan survey was made of the remediated area. A Ludlum Model 44-10 2" by 2" NaI detector coupled to a Ludlum Model 2350-1 Data Logger operating in single channel analyzer mode was used to conduct the survey. No detectable activity above the MDCR was detected during this survey.

8.3 Post Remediation Soil Samples

a. Two soil samples were collected from the area following the gamma scan survey. The samples were sent to NWT's laboratory in Livermore, CA for gamma spectroscopy analysis to quantify Co-60 concentrations. The results of the samples were below the DCGL.

b. The laboratory data for the soil samples is presented in this report in Appendix K.

c. Figure 5 presents a diagram showing the soil sample locations for the remediated area.

d. Table 6 presents a summary of the post remediation soil samples.

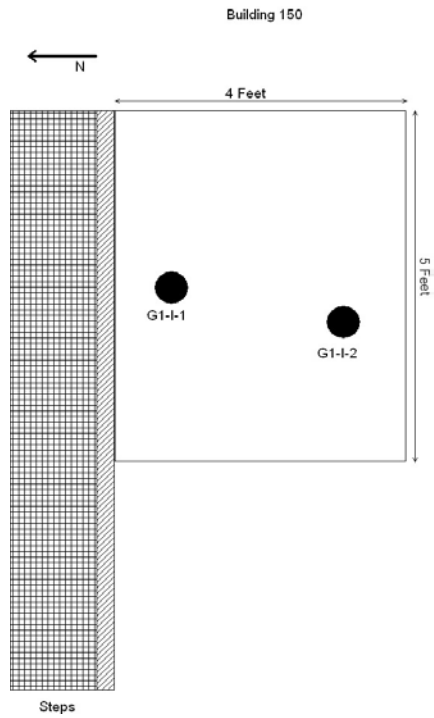


Figure 5. Post Remediation Soil Sample Location Diagram

Table 6. Post Remediation Soil Sample Summary Table

Sample ID	Co-60 Results in pCi/g	MDC in pCi/g
G1-I-1	1.383	0.020
G1-I-2	0.284	0.016

9.0 BUILDING 150 INTERIOR SURFACE DECONTAMINATION ACTIVITIES

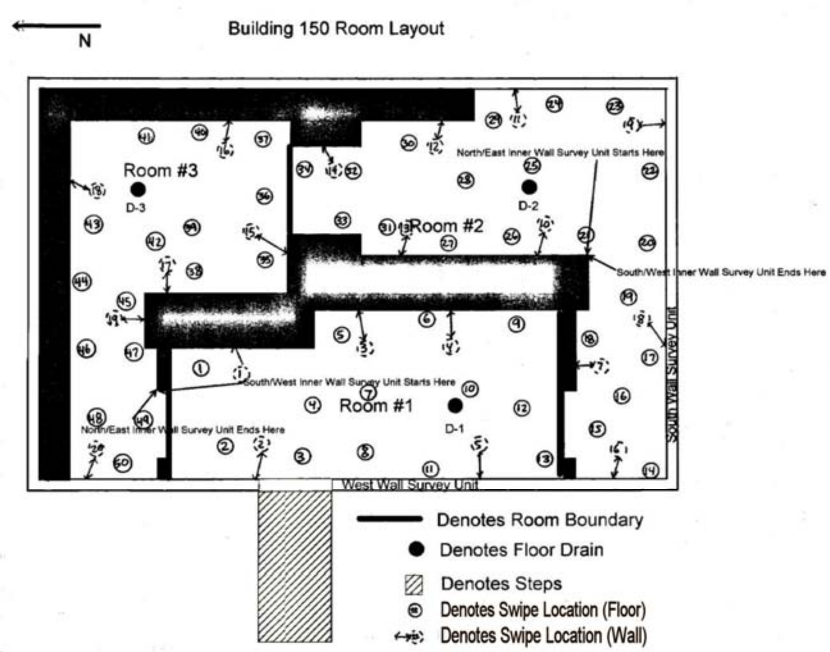
9.1 Building 150 Loose Surface Contamination Survey

a. A detailed loose surface contamination survey was performed on the surfaces and debris located inside of the building prior to work starting inside of the building.

b. Swipe samples were collected for analysis of removable alpha and beta contamination activity. The swipes were collected over an area of 100 cm^2. Dry cloth swipes were used for the swipe samples. Samples were analyzed using a Ludlum Model 2929 Dual Channel Scaler. The Ludlum Model 2929 Dual Channel Scaler is ZnS(Ag) phoswich scintillation alpha/beta radiation counter. This unit provides a full range of simultaneous alpha and beta counting at the levels required for environmental release surveys. Data is reported in units of dpm per 100 cm^2.

c. The locations of the swipe samples are presented in Figure 6 below.

Figure 6. Building 150 Swipe Location Map



d. Appendix L provides the swipe results. None of the swipe samples exceeded the MDA of 12 dpm/100cm ^{2} α and 228 dpm/100cm $^2\beta\gamma$. None of the swipe measurements exceeded 10% of the DCGL.

9.2 Building 150 Debris Surveys

a. Beta-gamma surface scan surveys were completed on 100 % of the accessible surfaces of all of the debris located inside of Building 150. The results of the surveys indicated no detectable beta-gamma activity above background levels.

b. The gross beta-gamma scan surveys were performed using a large area (126 cm^2) gas proportional detector system, a Ludlum Instruments Model 2224 rate/scaler coupled to a Ludlum Instruments Model 43-68 gas proportional detector.

c. The debris was removed from Building 150 and placed into a 30-cubic yard dumpster for disposal at a non-regulated landfill.

d. The results of the debris surveys are presented in Appendix M.

9.3 Scan Survey to Identify Areas Requiring Remediation

a. Beta-gamma surface scan surveys were completed inside of Building to determine where concrete removal was required inside of Building 150. Five areas above the DCGL of 7,100 dpm/100cm^2 beta-gamma were identified for remediation.

b. The gross beta-gamma scan surveys were performed on the walls using a large area (126 cm^2) gas proportional detector system, a Ludlum Instruments Model 2224 rate/scaler coupled to a Ludlum Instruments Model 43-68 gas proportional detector.

c. The gross beta-gamma scan surveys were performed on the floors using a large area (582 cm^2) gas proportional detector system, a Ludlum Instruments Model 2224 rate/scaler coupled to a Ludlum Instruments Model 43-37 gas proportional detector.

d. The areas that required remediation are presented in Figure 7 and Figure 8 below.

Figure 7. Building 150 Wall Areas Requiring Remediation

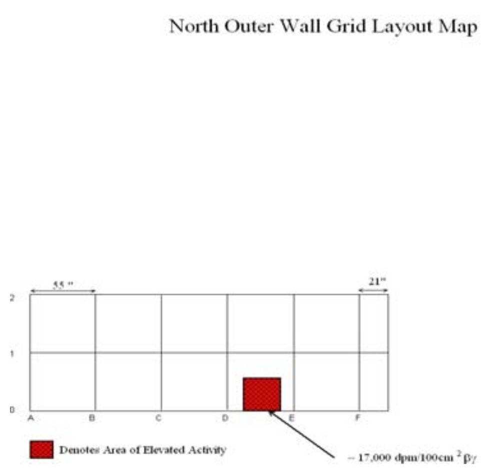
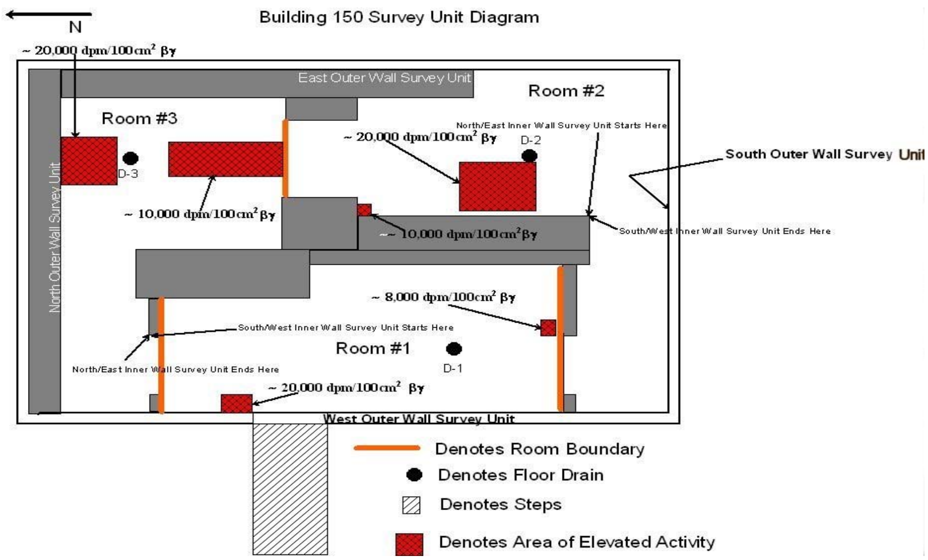


Figure 8. Building 150 Floor Areas Requiring Remediation



9.4 Contaminated Concrete Surface Removal

a. Any areas identified during the scan survey performed as outlined in Section 9.3 of this report that were above the DCGL, required remediation.

b. It was anticipated that the contamination was in the uppermost 1/2 inch (approximate) of the concrete. The areas identified during the scan survey as requiring remediation were remediated using a concrete scabbler. Removal of 1/2 inch of the concrete surface in the identified areas was sufficient.

c. Wetting of the surface, use of a HEPA filtered vacuum cleaner, and the use of a plastic containment enclosure was required to prevent the generation or airborne radioactive material during the concrete removal process.

d. Frequent monitoring, using high volume air samplers, with the sample obtained from the workers breathing zone, were utilized for evaluation of the potential airborne radioactive material hazard.

e. This data was also used to make decisions and evaluations for personnel protective equipment (PPE) requirements at the site. The results of the air samples indicated that there was no airborne radioactivity generated during the concrete surface removal activities and that airborne radioactivity levels were less than 10% of the DAC value for Co-60 (1.0 E-9 µCi/ml as found in Appendix B, Table 1, Column 3, of 10 CFR Part 20.

f. Gross alpha/beta counting of the air samples was performed using a Ludlum Model-2929 dual channel scaler. This counting system was used for gross alpha/beta counting of the air samples and swipes.

g. Equation 1 below was used when counting the air samples.

Equation 1. Air Sample Activity Calculation

$$\text{Air Sample Activity in } \mu\text{Ci/ml} = \frac{\text{Gross CPM - Background CPM}}{2.22\text{ E+6} * V * E*T * F* 2.832\text{ E+4}}$$

Where:

V= Sample Volume in Milliliters
T= Sample Count Time in Minutes
E= Instrument Efficiency Expressed as a Decimal
F= Filter Efficiency
2.834E+4 = Conversion from Cubic Feet Milliliters, if necessary
2.22E+6 = Conversion from dpm to microcuries

h. Equation 2 below was used to calculate the MDA of the air samples.

Equation 2. Air Sample MDA Calculation

$$\text{MDA in } \mu\text{Ci/ml} = \frac{2.71 + 4.65\sqrt{R_B/T}}{2.22\text{ E+6} * E * V}$$

Where:

V= Sample Volume in Milliliters
T= Sample Count Time in Minutes
E= Instrument Efficiency Expressed as a Decimal
R_B = Background Count Rate in CPM
2.22E+6 = Conversion from dpm to microcuries

i. In order to account for the presence of radon daughters the half-life of the air sample activity was calculated using Equation 3 below:

Equation 3. Air Sample Activity Half-Life Calculation

$$T_{1/2} = \frac{-0.693(t)}{\ln\left(\dfrac{\text{Final Activity in } \mu\text{Ci/ml}}{\text{Initial Activity in } \mu\text{Ci/ml}}\right)}$$

Where:

$T_{1/2}$= Sample Half Life in Minutes
(t) = Time in Minutes Between Initial Count and Final Count
ln = Natural Logarithm

j. The air sampling data is presented in Appendix N.

k. Once the top 1/2 inch was removed from the selected section, the removed concrete dust and material was then removed from the surface using an HEPA filtered vacuum cleaner, emptying the contents of the vacuum cleaner, as necessary, into the appropriate waste container.

l. A scan survey for contamination of the remaining concrete surface was then performed in accordance with Section 9.3 of this report. Again, identifying any areas exceeding the DCGL for surface contamination as defined in Section 4.6 of this report.

m. Once remediation in the area was completed, the work area was relocated to the next area to be remediated and the applicable steps outlined in this section were repeated until all identified contaminated concrete had been removed.

10.0 BUILDING INTERIOR 150 DRAIN LINE REMOVAL ACTIVITIES

10.1 Pre Concrete Removal Beta Scan Surveys

a. Beta surface scan surveys were completed where concrete removal was required inside of Building 150. The results of the surveys indicated that the concrete was below the DCGL of 7,100 dpm/100cm^2 beta.

10.2 Drain Line Samples

a. Sediment samples were collected inside the opening of the three drain lines located inside of Building 150 and from removed drain piping inside of Room #1 and Room #2. The samples were sent to an off site laboratory for gamma spectroscopy analysis to quantify Co-60 concentrations. The results of the samples were between 3.73 pCi/g and 71 pCi/g of Co-60. The laboratory data for the samples is presented in Appendix O. The results of the samples are summarized in Table 7 below.

Table 7. Building 150 Drain Line Sample Summary Table

Sample ID	Co-60 Results in pCi/g	MDC in pCi/g
D-1 (Opening at Floor)	7.50	0.37
D-2 (Opening at Floor)	3.73	0.34
D-3 (Opening at Floor)	12.50	0.54
Inside Drain Pipe #1	34.30	0.27
Inside Drain Pipe #2	71.00	0.46

10.3 Removal of Contaminated Drain Lines

a. Concrete floor surfaces were removed around the three drain lines to facilitate removal of the lines from inside of the building. This was accomplished using a jack hammer.

b. 100% gamma scan surveys were performed on the overburden soil around the drain lines. Surveys were conducted with with 2” by 2” NaI detectors coupled to data loggers operating in the single channel analyzer mode.

c. None of the overburden soil was found to have any anomalies above background radiation levels during the gamma scan survey.

d. The overburden soil was then removed using hand tools and staged adjacent to the trenches.

e. The drain lines were removed from the trenches using chains, slings, etc. The removed sections of the drain line were then sized so that they fit into a B-25 box. Precautions (i.e. sealing of the openings of the drain with plastic) were taken to prevent the contents of the drain line from escaping into the surrounding soils.

f. The removed drain lines were then staged on plastic sheeting inside of Building 150 and then transported and placed into the B-25 box.

g. Approximately 55 linear feet of drain piping was removed from inside of Building 150.

h. Following completion of concrete and drain removal actions, all equipment used during the effort underwent release surveys in accordance with the NWT release procedures. The release surveys are presented in Appendix P.

10.4 Gamma Scan Surveys/Soil Sampling of Trenches

a. The drain lines were removed from the trenches using hand tools and chain-falls.

b. A 100 % gamma scan survey was performed inside of the trenches. The survey was conducted with 2" by 2" NaI detectors coupled to data loggers operating in the single channel analyzer mode.

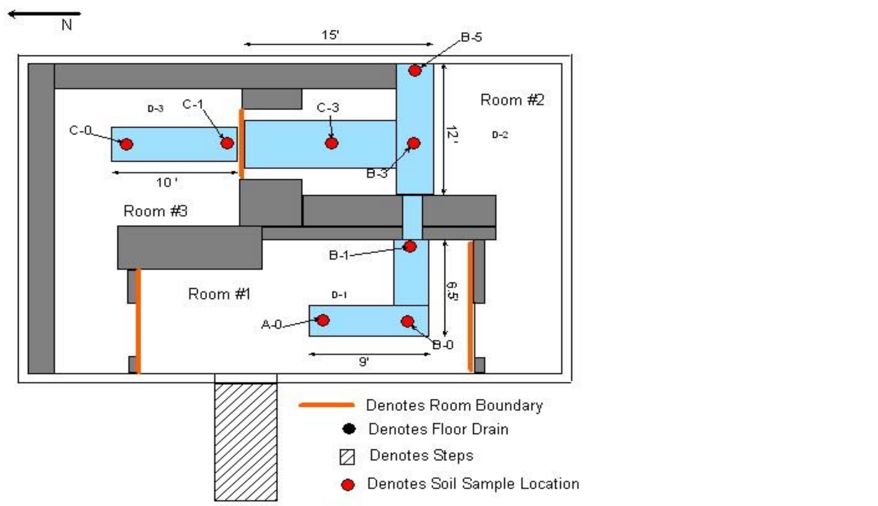
c. No detectable activity above background radiation levels was detected during these surveys.

d. Soil samples were collected at a maximum of every 10 linear feet of the trenches. The samples were sent to an offsite laboratory for gamma spectroscopy analysis to quantify Co-60 concentrations. None of the soil sample results exceeded the DCGL. The laboratory data for the soil samples is presented in Appendix Q.

e. Table 8 presents a summary of the soil sample results.

f. Figure 9 presents a map of the interior trenches and soil sample locations inside of Building 150.

Figure 9. Building 150 Interior Trench/Sample Location Map



11.0 BUILDING EXTERIOR 150 DRAIN LINE REMOVAL ACTIVITIES

11.1 Removal of Contaminated Drain Lines

a. A dig permit was issued by NNMC bas personnel prior to the start of excavation in the areas outside of Building 150. The dig permit is presented in Appendix D.

b. The drain lines leading from the three rooms of Building 150 exited the building at the southeast corner area of the building.

c. At the point of exit from the building the lines tied into one drain line.

d. 100% gamma scan surveys were performed on the overburden soil around the drain lines. The survey was conducted with 2" by 2" NaI detectors coupled to data loggers operating in the single channel analyzer mode.

e. All of the overburden soil was found to have no anomalies above background radiation levels during the gamma scan survey.

f. The overburden soil was removed using a backhoe and then staged adjacent to the trenches.

g. The drain line was removed from the trenches using chains, slings, etc. The removed sections of the drain line were then sized so that they fit into a B-25 box. Precautions (i.e. sealing of the openings of the drain with plastic) were taken to prevent the contents of the drain line from escaping into the surrounding soils.

h. The removed drain line was then staged on plastic sheeting and then transported and placed into the B-25 box.

i. Approximately 20 linear feet of drain piping was removed outside of Building 150. At the 20 foot mark there was evidence that the rest of the piping had been removed in the past.

j. Following completion of removal actions, all equipment used during the remediation effort underwent release surveys in accordance with the NWT release procedures. The release surveys are presented in Appendix P.

11.2 Gamma Scan Surveys/Soil Sampling of Trenches

a. A 100 % gamma scan survey was performed inside of the trenches. The survey was conducted with 2" by 2" NaI detectors coupled to data loggers operating in the single channel analyzer mode.

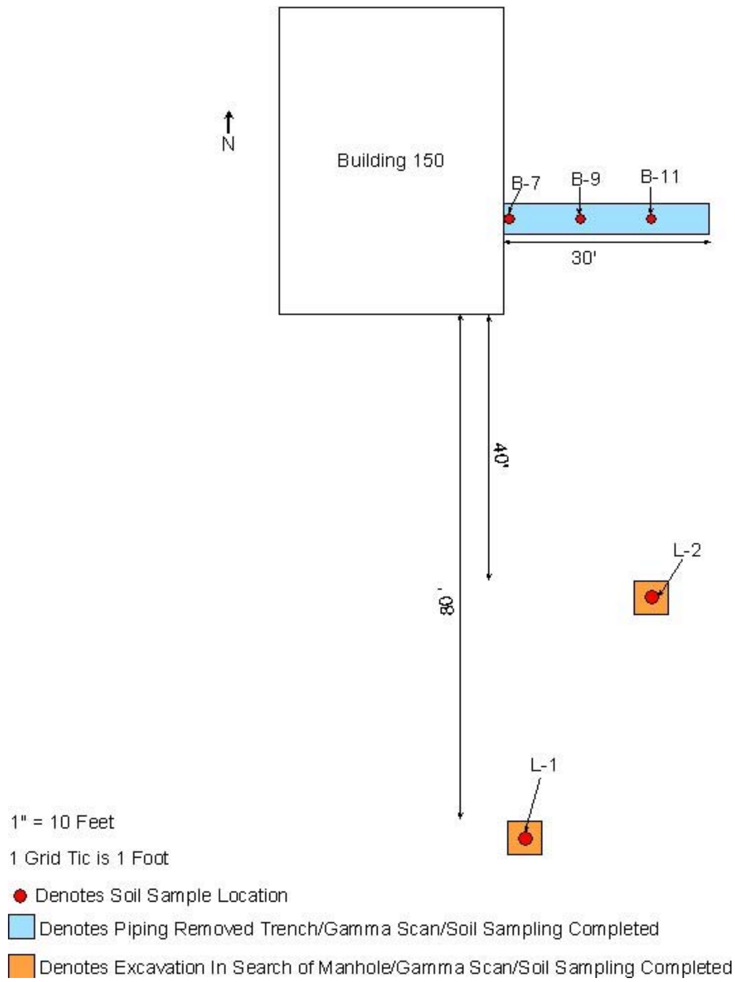
b. No detectable activity above background radiation levels was detected during these surveys.

c. Soil samples were collected at a maximum of every 10 linear feet of the trenches. The samples were sent to an offsite laboratory for gamma spectroscopy analysis to quantify Co-60 concentrations. None of the soil sample results exceeded the DCGL. The laboratory data for the soil samples is presented in Appendix R.

d. Table 8 presents a summary of the soil sample results.

e. Figure 10 presents a map of the exterior trenches and soil sample locations outside of Building 150.

Figure 10. Building 150 Exterior Trench/Sample Location Map



11.3 Manhole Location Investigation

a. An attempt to locate manhole 047M was made by excavating test holes in the locations where site engineering drawings indicated the manhole was located.

b. Figure 11 presents a map of the possible location of manhole 047M.

c. The test holes were approximately 5 feet in diameter and 8 foot in depth. No indication of the manhole was identified inside of the test holes.

d. A 100 % gamma scan survey was performed inside of the test holes. The survey was conducted with 2" by 2" NaI detectors coupled to data loggers operating in the single channel analyzer mode.

e. No detectable activity above background radiation levels was detected during these surveys.

f. Soil samples were collected in the bottom of each of the test holes. The samples were sent to an offsite laboratory for gamma spectroscopy analysis to quantify Co-60 concentrations. None of the soil sample results exceeded the DCGL. The laboratory data for the soil samples is presented in Appendix S.

g. Table 8 presents a summary of the soil sample results.

h. Figure 10 presents a map of the manhole investigation trenches and soil sample locations outside of Building 150.

Figure 11. Manhole 047M Location Map

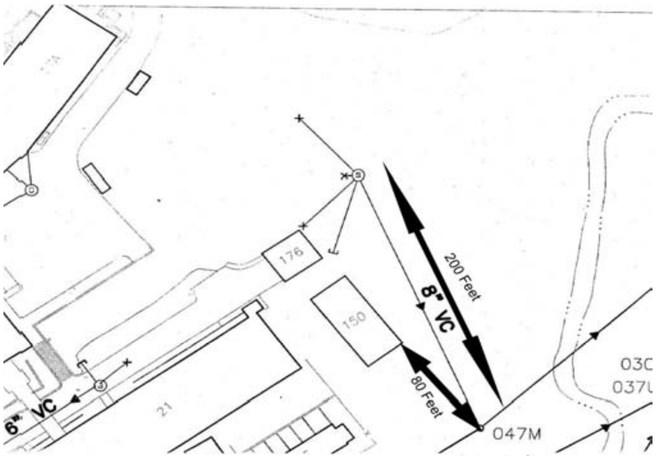


Table 8. Drain Line Trench Soil Sample Summary Table

Sample ID	Co-60 Results in pCi/g	MDC in pCi/g
Room #1 Grid A-0	0.03	0.12
Room #1 Grid B-0	0.02	0.09
Room #1 Grid B-1	0.07	0.11
Room #3 Grid C-0	0.03	0.12
Room #3 Grid C-1	-0.05	0.14
Room #2 Grid C-3	0.02	0.13
Room #2 Grid B-3	0.14	0.12
Room #2 Grid B-5	-0.02	0.12
Trench Grid B-7	0.01	0.12
Trench Grid B-9	0.03	0.12
Trench Grid B-11	0.09	0.08
L#1	-0.03	0.17
L#2	0.00	0.19

12.0 WASTE PACKAGING AND DISPOSAL

12.1 UST Disposal

a. The UST's were loaded onto a flatbed truck and transported to U.S. Ecology's facility located in Robstown, TX for disposal.

b. The waste manifests and appropriate documents are presented in Appendix T.

12.2 Building 150 Decontamination Waste Disposal

a. The B-12 and B-25 boxes were inspected for container integrity, surveyed, weighed and all required labeling performed.

b. The B-12 box and B-25 box were shipped by an enclosed van to the licensed and permitted disposal site, Duratek's Consolidation Service Facility located in Barnwell, SC for processing and disposal.

c. The NWT Broker (JMC qualified) prepared all necessary Department of Transportation (DOT) and procedurally required forms (Radioactive Shipment Record, Bill of Lading, etc.). All DOT shipping and labeling requirements were met prior to release of the closed van truck from the site.

d. The total volume of waste shipped to the DCF was 140 cubic feet.

e. The waste manifests and appropriate documents are presented in Appendix T.

13.0 FINAL STATUS SURVEYS

The Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) NUREG-1575, NUREG-1505, and NUREG-1507 were used as guidance in designing and conducting final status surveys.

13.1 Objective of Final Status Surveys

The objective of a final status survey is to demonstrate that residual radioactivity levels meet the release criterion. In demonstrating the objective is met, the null hypothesis (H_o) that residual contamination exceeds the release criterion is tested; the alternative hypothesis (H_a) is that residual contamination meets the release criterion.

13.2 Scoping Survey

A scoping survey was performed inside of Building 150 to collect data to help in determining the number of data points required for the Sign Test.

Thirty random measurement locations were chosen throughout the building (15 on wall surfaces, and 15 on floor surfaces). Figure 12 presents a diagram that shows the thirty random direct measurement points.

Direct surface contamination surveys for alpha and beta radiation were conducted at each of the thirty data points with Ludlum Model 43-68 alpha-beta gas proportional probes and

Ludlum Model 2224 alpha/beta ratemeters/scalers. The probe has a 0.8 mg/cm^2 thick Mylar windows. Direct measurements were conducted with the detector on contact with the surface for a period of 1 minute.

The data obtained from the scoping surveys is presented in Appendix U.

Table 9 presents a summary of the results of the scoping survey.

Figure 12. Scoping Survey Map

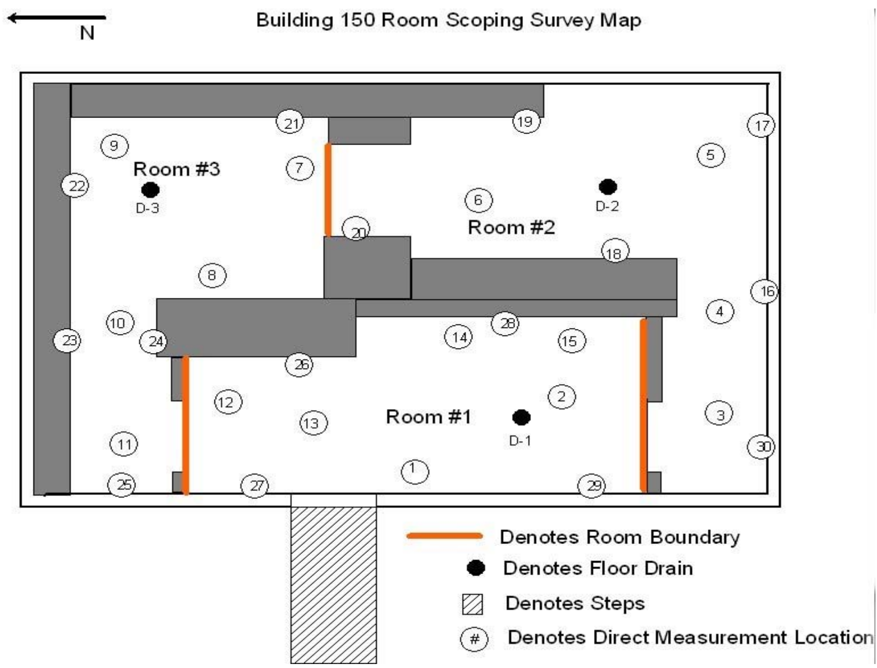


Table 9. Scoping Survey Summary Table

Data Point	Instrument # 118242	
	α Reading in CPM	β Reading in CPM
1	2	176
2	3	145
3	0	153
4	2	139
5	0	175
6	1	162
7	1	198
8	2	222
9	0	221
10	2	163
11	1	152
12	2	146
13	0	157
14	1	146
15	1	180
16	2	194
17	1	187
18	1	167
19	3	150
20	0	168
21	1	156
22	2	144
23	1	144
24	0	179
25	2	146
26	1	228
27	0	175
28	0	173
29	0	172
30	1	168
Average Activity Level in dpm/100cm^2	**12**	**1,583**
Activity Level Standard Deviation	**10.1**	**224.5**

13.3 Survey Units, Building 150 Interior and Roof

The Building 150 interior area was divided into three different rooms. Each room was then divided into survey units based upon contamination potential. Room # 1,Room # 2, and Room # 3 floors and ceiling areas were separated into two survey units each (one floor, one ceiling). The inner walls were separated into two separate survey units, and the outer walls were separated into four separate survey units. There were a total of 12 survey units inside of Building 150. The roof of Building 150 was divided into two separate survey units. The elevator shaft room was divided into one survey unit.

Figure 13 is a diagram of the Building 150 interior areas, presenting a layout of the different survey units.

Figure 14 is a diagram of the elevator shaft room survey unit.

Figure 15 is a diagram of the Building 150 roof area, presenting a layout of the different survey units.

Figure 13. Survey Units-Building 150

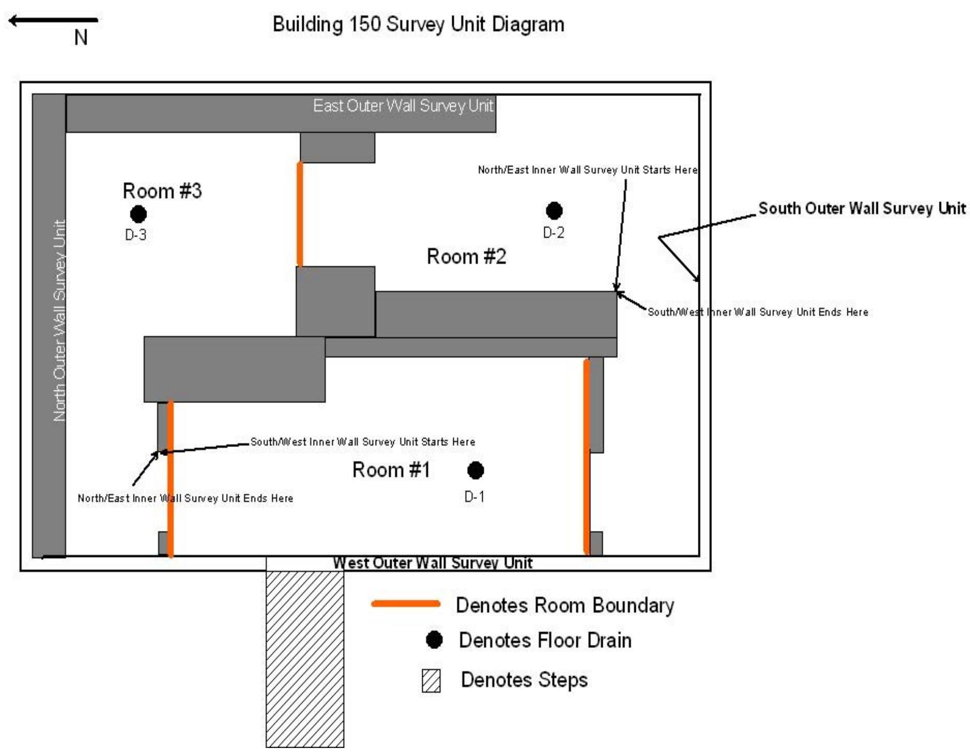


Figure 14. Elevator Shaft Room Survey Unit Diagram

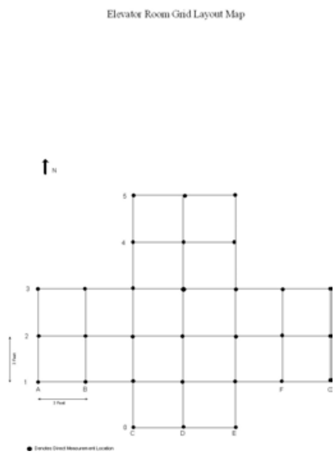
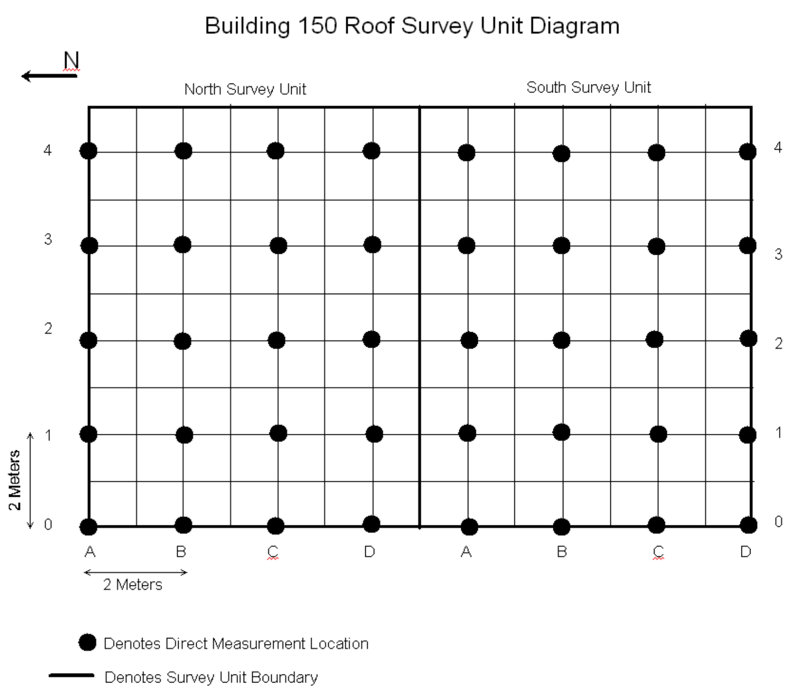


Figure 15. Building 150 Roof Survey Unit Diagram

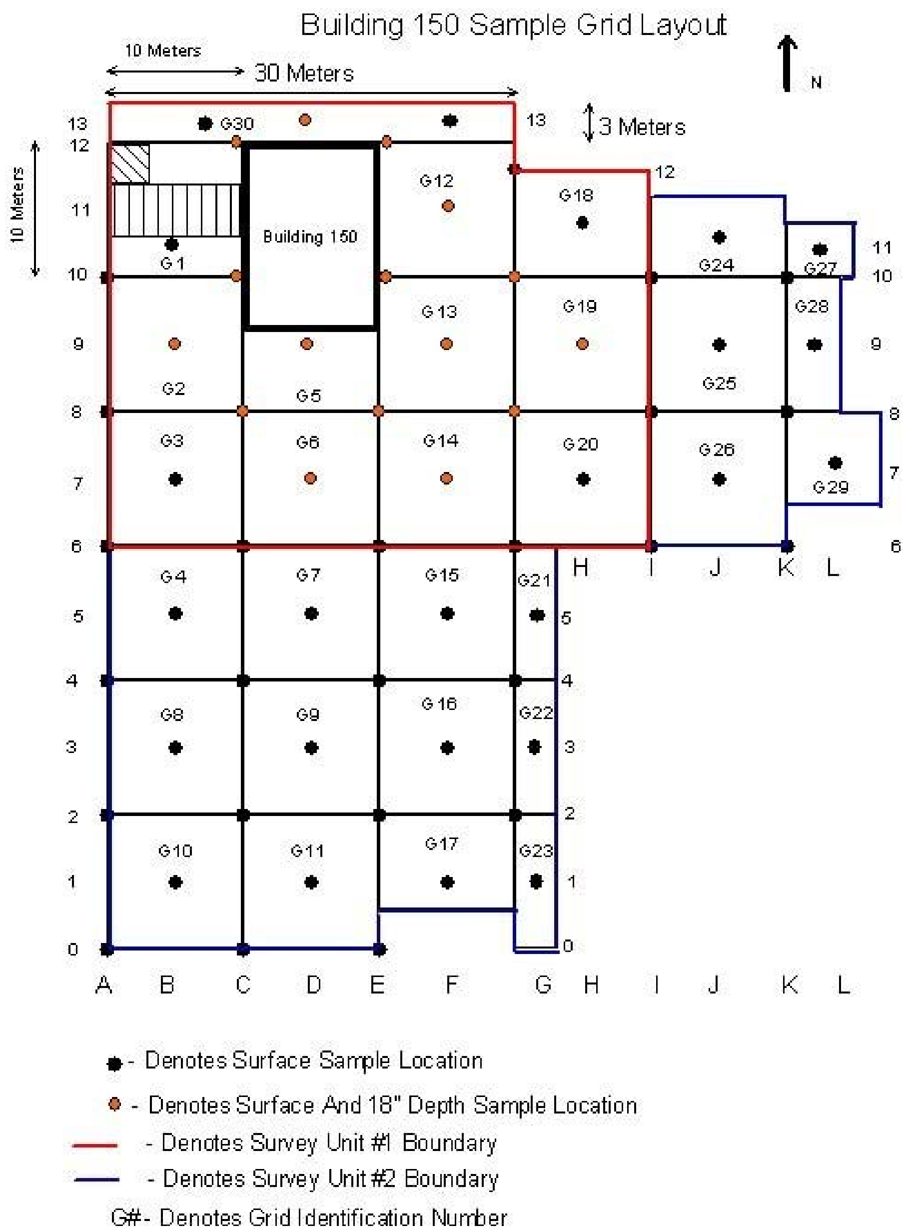


13.4 Survey Units, Building Outdoor Open Land Areas

The open land area outside of Building 150 was divided into two survey units based upon contamination potential. Whenever possible, ten meter by ten meter grids were setup in each of the survey units.

Figure 16 presents a diagram showing the survey units outside of Building 150.

Figure 16. Building 150 Outside Open Land Area Survey Unit Layout Diagram



13.5 Survey Unit Classification

Based on historical documents and previous radiological surveys, all of the survey units were classified as Class 1 survey units. Table 10 presents a summary of the various survey units.

Table 10. Survey Unit Summary Table

Area/Location	MARSSIM Classification	Total Surface Area (Square Meters)	Number of Survey Units
Room # 1 Floor	Class 1	~ 27	1
Room #1 Ceiling	Class 1	~ 27	1
Room # 2 Floor	Class 1	~ 34	1
Room #2 Ceiling	Class 1	~ 34	1
Room # 3 Floor	Class 1	~ 24	1
Room #3 Ceiling	Class 1	~ 24	1
North Outer Wall	Class 1	~ 23	1
East Outer Wall	Class 1	~ 39	1
South Outer Wall	Class 1	~ 23	1
West Outer Wall	Class 1	~ 39	1
North/East Inner Wall	Class 1	~ 38	1
South/West Inner Wall	Class 1	~ 47	1
Elevator Shaft Room	Class 1	~ 18	1
Roof Of Building 150	Class 1	~ 126 (64 each unit)	2
Building 150 Outdoor Open Land areas	Class 1	~ 1130 (Survey Unit #1) ~ 1320 (Survey Unit #2)	2

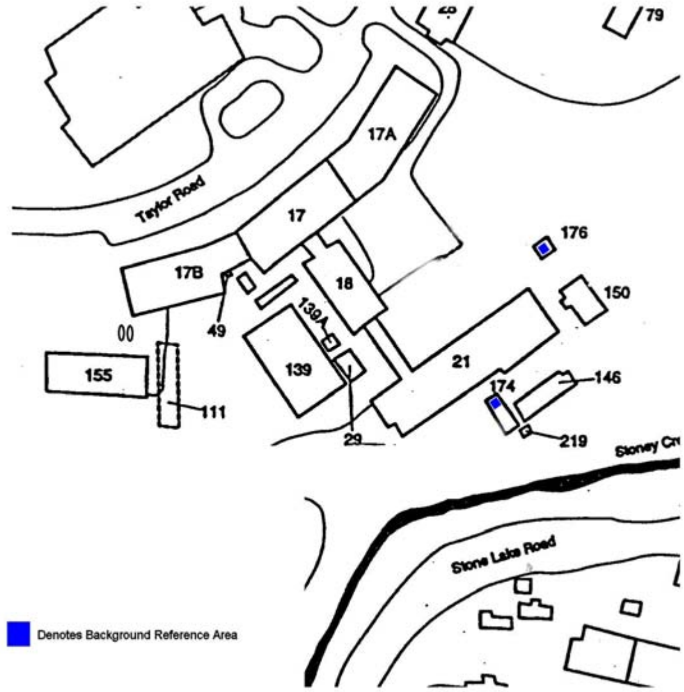
13.6 Reference (Background) Areas for Building Surfaces

The interior of Buildings 176 and 174 were selected as the reference (background) areas. These areas were selected because they were non-impacted from Building 150 operations and have the same building structure characteristics of Building 150. Figure 17 provides a diagram of the reference areas where radiation measurements were taken.

A total of thirty alpha/beta direct measurements were recorded with a Ludlum Model 2224 ratemeter/scaler coupled to Ludlum Model 43-37 and 43-68 large area gas proportional detectors for a period of 1-minute in contact with the surface.

The results of the building surface background reference area surveys are presented in Appendix V.

Figure 17. Background Reference Areas-Building Surfaces



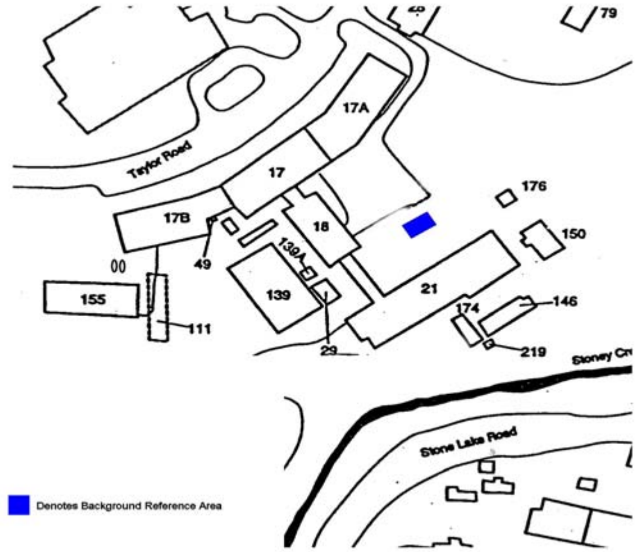
13.7 Reference (Background) Area for Outside Land Areas

An open land area north east of Building 21 was selected as the reference (background) areas for the open land areas outside of Building 150. This area was selected because it was non-impacted from Building 150 operations and has the same chemical, physical, biological, and geological characteristics as the exterior open land areas outside of Building 150. Figure 18 provides a diagram of the reference area showing the location where radiation measurements were taken.

A total of one hundred gamma direct measurements were recorded with a Ludlum Model 2350-1 Data Logger coupled to a Ludlum Model 44-10 NaI detector operating in the single channel mode. The detector was held at approximately 2-inches from the surface.

The results of the outside land area background reference area surveys are presented in Appendix W.

Figure 18. Background Reference Area-Outside Land Areas



13.8 Reference Grids

A reference coordinate system was laid out for each of the survey units. A square grid system was used for the Final Status Surveys. The length, L, of a side of the square grid was determined by the total number of samples or measurements to be taken. The length of the square determined the distance between direct measurement/soil sample location points (NUREG-1505). The length or spacing of the grids was calculated for each of the survey units using the following equation:

$$L = \sqrt{\frac{A}{N}}$$

Where,

L = length of squares grids (m);

A = surface area of the survey unit (m^2); and

N = statistically calculated number of samples.

The length of the measurement/sampling intervals for each of the survey units is presented in Table 11 below.

Table 11. Survey Unit Data Table

Survey Area/Location/Unit	Survey Unit Size in Square Meters	Number of Required Direct Measurement Locations (N_R)	Number of Actual Direct Measurement Locations (N_A)	Lengh of Grid Pattern in Meters	Lengh of Grid Pattern in Inches
Room # 1 Floor	~ 27	16	29	1.3	51
Room #1 Ceiling	~ 27	16	29	1.3	51
Room # 2 Floor	~ 34	16	27	1.4	55
Room #2 Ceiling	~ 34	16	27	1.4	55
Room # 3 Floor	~ 24	16	29	1.2	47
Room #3 Ceiling	~ 24	16	29	1.2	47
North Outer Wall	~ 23	16	18	1.4	55
East Outer Wall	~ 39	16	30	1.5	59
South Outer Wall	~ 23	16	21	1.5	59
West Outer Wall	~ 39	16	27	1.4	55
North/East Inner Wall	~ 38	16	30	1.5	59
South/West Inner Wall	~ 47	16	30	1.7	67
Elevator Shaft Room	~ 18	16	30	1.0	39
Building 150 Roof (2 Survey Units)	~ 64	16	20	2.0	78
Building 150 Outside Outdoor Land Area Survey Unit #1	~ 1130	16	39	5.0	N/A
Building 150 Outside Outdoor Land Area Survey Unit #2	~ 1320	16	41	5.0	N/A

Maps presenting the locations of the direct measurement/sampling points for each of the survey units is presented in Appendix X.

13.9 Statistical Tests

MARSSIM (NUREG 1575) recommends using the Sign test to evaluate survey results when the contaminant is not present in background; i.e., the radionuclides of concern (Co-60 and tritium) are not normally present in background. Conservatively, the Sign test will be used instead of the Wilcoxon Rank Sum (WRS) Test (used to evaluate survey results when the radionuclides of concern are present in background) to evaluate the beta-gamma measurements from the final status surveys.

13.10 Determining the Numbers of Data Points for the Sign Test

The number of data points, N, to be obtained for the Sign test can be obtained from table 5.5 in MARSSIM or calculated using Equation 4 below.

Equation 4

$$N = \left(\frac{(Z_{1-\alpha} + Z_{1-\beta})^2}{4(Sign\ p - 0.5)^2} \right)(1.2)$$

where,

$Z_{1-\alpha}$ = type I decision error level
$Z_{1-\beta}$ = type II decision error level
Sign p = random measurement probability, which is base on relative shift discussed below in Section 13.10.1

The second term in the equation increases the number of data points by 20%. The value of 20% is selected to account for a reasonable amount of uncertainty in the parameters used to calculate N and still allow flexibility to account for some lost or unusable data.

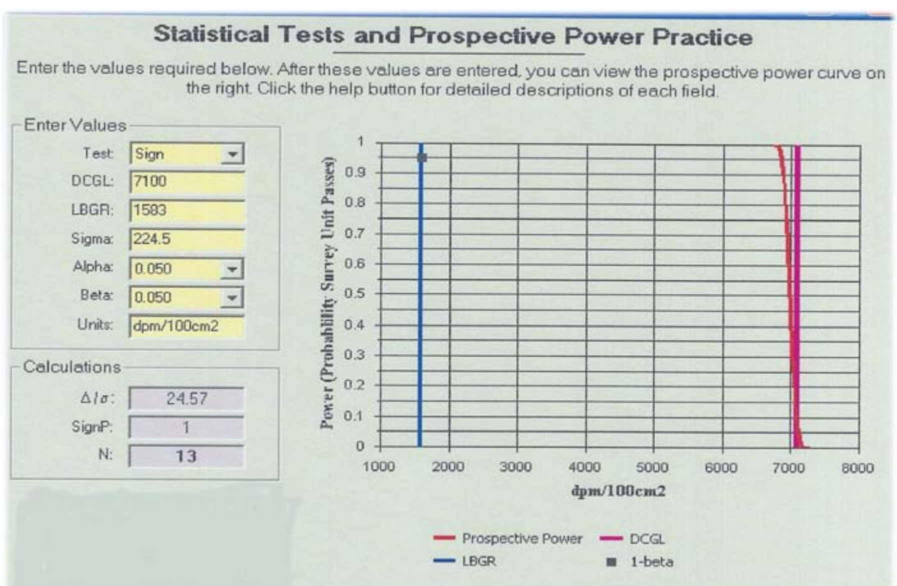
13.10.1 The Relative Shift

Sign p in Equation 4 above is based on the relative shift. The relative shift is equal to Δ/σ, where Δ is equal to (DCGL-LBGR) and σ is an estimate of the standard deviation of the measured values in the survey unit. The LBGR (Lower Bound of the Grey Region) is the net median concentration of the contaminant in the survey unit. This value (1,583 dpm/100cm^2 β) was calculated using the data obtained from the scoping survey. Likewise, the standard deviation (224.5) of the contaminant in the survey unit was calculated from the data obtained from the scoping survey.

Using the data collected during the scoping surveys, the number of data points required was calculated using Equation 4. The calculated number of data points was 13.

As a conservative measure, 16 data points were selected as the required number of data points. Figure 19 presents a graphical illustration of the prospective power curve, calculated relative shift and Sign p values that were generated when calculating the required number of data points.

Figure 19. Statistical Test and Power Curve



13.10.2 Determining Data Points for Small Areas of Elevated Activity

The statistical tests described above evaluate whether or not the residual radioactivity in an area exceeds the $DCGL_W$ for contamination conditions that are approximately uniform across the survey unit. In addition, there should be a reasonable level of assurance that any small areas of elevated residual radioactivity that could be significant relative to the $DCGL_{EMC}$ are not missed during the final status survey. The statistical tests introduced in the previous sections may not successfully detect small areas of elevated contamination. Instead, systematic direct measurements and sampling, in conjunction with surface scanning, are used to obtain adequate assurance that small areas of elevated radioactivity will still satisfy the release criterion or the $DCGL_{EMC}$. The procedure is applicable for all radionuclides, regardless of whether or not they are present in background, and is implemented for Class 1 survey units.

One method for determining values for the $DCGL_{EMC}$ is to modify the $DCGL_W$ using a correction factor that accounts for the difference in area and the resulting change in dose or risk. The area factor is the magnitude by which the concentration within the small area of elevated activity can exceed $DCGL_W$ while maintaining compliance with the release criterion.

Table 8.2 of NUREG-1505 provide examples of indoor area factors generated using exposure pathway models.

The minimum detectable concentration (MDC) of the scan procedure needed to detect an area of elevated activity at the limit determined by the area factor is calculated as follows:

Scan MDC (required) = ($DCGL_W$) × (Area Factor)

Using the calculated scan MDC value for beta-gamma emitting radionuclides the required scan MDC is calculated using the above equation.

The calculated scan MDCs were determined for the instrumentation in accordance with Section 15.0 of this report. The actual MDC of the selected scanning technique was then compared to the required scan MDC.

Calculations in Section 13.10.1 specify that the number of data points needed for statistical testing is 16 for the Class 1 survey units. The largest distance between measurement locations in any of the Class 1 survey units for this number of data points is 2.0 m. The area encompassed by a square sampling pattern of 2.0 m is 4.0 m^2. From Table 8.2 of NUREG-1505 and using Co-60 as the radionuclide of concern, an area factor of 5.5 is determined.

The required scan MDC is then calculated using the equation below:

Scan MDC (required) = ($DCGL_W$) × (Area Factor)

39,050 = 7100 x 5.5

The acceptable concentration in a 4.0 m^2 area is therefore 39,050 dpm/100cm^2 $\beta\gamma$. Since the calculated scan MDC to be used is less than the $DCGL_W$ times the area factor, the data points calculated in Section 13.10.1 are sufficient to show compliance using the elevated measurement comparison criteria.

13.10.3 Fixed Beta Measurements (Building 150 Interior)

A random measurement pattern was used for both the reference area and survey units. All of the measurement locations were randomly selected. A few of the measurements obtained in the survey units were biased. The biased measurement locations were selected based on locations where the highest scan readings were obtained, on the history of prior use of the facility, or on the observations and experience of the surveyor (i.e., low-lying areas, drains or unusual concrete patterns).

13.10.4 Soil Samples (Building 150 Outside Land Areas)

Sixty-four surface, and sixteen subsurface (2 foot depth) soil samples were collected and sent to an off site laboratory for gamma spectroscopy analysis. The laboratory data for the samples is presented in Appendix Y.

13.11 Scanning Surveys Gross Beta

Gross alpha/beta scans were conducted over 100% of the surface of each of the Class 1 survey units.

13.12 Gamma Scan Surveys of Open Land Areas Outside of Building 150

Gamma scan surveys were conducted over 100% of the surface of each of the two Class 1 survey units.

Figure 20 and Figure 21 present diagrams of the surveyor's direction of travel during the performance of the gamma scan surveys.

NOTE: All the grids were surveyed with the same starting point and direction of travel with the exception of Grid #30.

Figure 20. Surveyor Direction of Travel Diagram

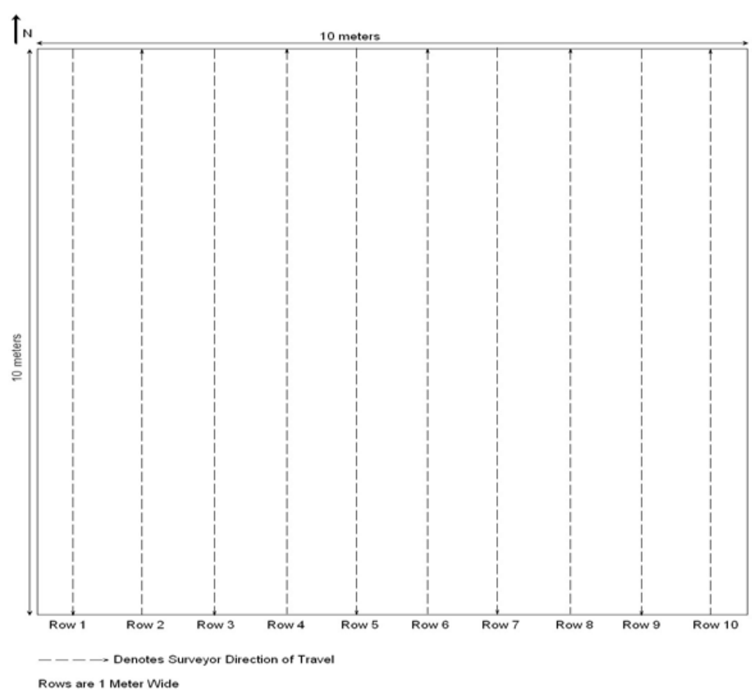
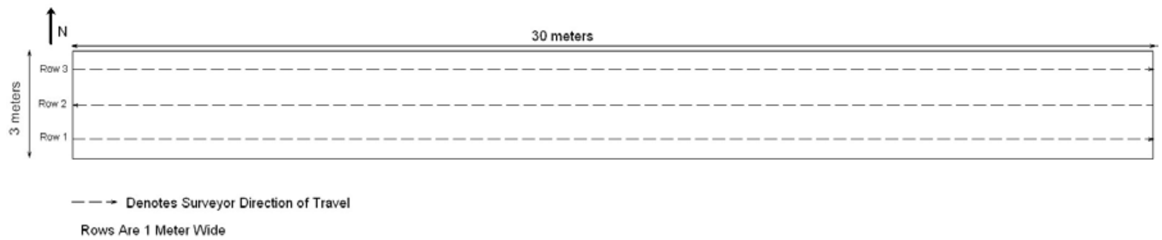


Figure 21. Grid #30 Surveyor Direction of Travel Diagram



14.0 SURVEY INSTRUMENTATION

14.1 Instrumentation Selection

Instruments were selected that were suitable for the physical and environmental conditions at the site as detailed in the Project Work Plan. The instruments and measurement methods selected were able to detect the radionuclide of concern or radiation types of interest, and are, in relation to the survey or analytical technique, capable of measuring levels that are equal to or less than the DCGL.

Several radiation detection methods were used during the radiological surveys: gamma detector response rate (scan) measurements, beta/gamma detector response rate (scan) measurements, beta/gamma detector integrated (1-minute) measurements, and soil sampling and analysis. Field survey methodology, techniques, and terminology was in accordance with the Federal guidance document MARSSIM (Rev. 1, August 2000). Chapters 5.3 and 5.5 of MARSSIM provides specific details as to how the surveys will be performed.

Appendix Z provides a detailed description of the instruments used to conduct surveys.

14.2 Instrument Calibration and Quality Assurance (QA) Procedures

All survey and laboratory instruments were calibrated with National Institute of Standards and Technology (NIST) traceable standards prior to the start of the project. Survey instruments were source checked each day prior to the start of the survey activities and again at the end of the day. If, during field operations, it became necessary to replace the Mylar window on the alpha-beta gas proportional probes, the probe-instrument package was source recalibrated on-site, using a NIST traceable source.

Appendix AA provides the instrument calibration procedure, calibration certificates, and daily instrument source check logs.

14.3 Instruments for the Scan Surveys for Beta Surface Activity

Surface scan surveys for beta radiation were conducted with Ludlum Model 43-37 alpha-beta gas proportional probes and Ludlum Model 2224 alpha/beta ratemeters/scalers and Ludlum Model 43-68 alpha-beta gas proportional detectors. The probes have 0.8 mg/cm^2 thick Mylar windows. The detector was moved over the surface being surveyed at a rate of one inch per second. The detector was held within ¼" of the surface being surveyed. Audible indicators were used during the surveys

Table 12 provides a summary of the instrumentation used for the beta surface scan surveys.

14.4 Instruments for the Direct Measurements for Beta Surface Activity

Direct surface contamination surveys for beta radiation were conducted with Ludlum Model 43-68 alpha-beta gas proportional probes and Ludlum Model 2224 alpha/beta ratemeters/scalers. The probes have 0.8 mg/cm^2 thick Mylar windows. Direct measurements were conducted with the detector on contact with the surface for a period of 1 minute.

Separate alpha-only and beta-only measurements were performed because it was not possible to establish defensible ratios of alpha and beta radiations for measurement purposes.

14.5 Instruments for the Gamma Scan Surveys of Open Land Areas

Gamma count rate responses were used to determine whether specific areas exhibit activity levels that are significantly above site-specific background. Gross gamma count rates were measured using a 2" by 2" sodium iodide (NaI) gamma scintillation detector system (Ludlum Instruments Model 2350-1 data logger coupled to a Ludlum Instruments Model 44-10 or the equivalent). Using the system in a single channel mode, this radiation detection system measures energies in the range of about 1,000 to 1,400-kilo electron Volt (keV). This energy range includes the gamma rays emitted by Cobalt-60. Using the system in a single channel mode effectively eliminated most of the background count rate on the instrument.

Sensitivities for scanning techniques were based upon movement of the detector over the surface in a serpentine pattern at a rate of approximately 1-foot per second and use of audible indicators to detect changes in instrument count rate. The detector was held within approximately four inches or less from the surface being surveyed.

Table 12 provides a summary of the instrumentation used for the gamma scan surveys of open land areas.

Table 12. Instrumentation Used for Beta Surveys of Building Surfaces/Gamma Scan Surveys of Open Land Areas

	Gamma Scans		Alpha Measurements		Beta Measurements	
Instrument Model	Ludlum 2350-1	Ludlum 2350-1	Ludlum 2224	Ludlum 2224	Ludlum 2224	Ludlum 2224
Model Serial No.	142489	134735	118242	146713	118242	146713
Instrument Detector	2" x 2" NaI 44-10	2" x 2" NaI 44-10	Gas Proportional 43-37/43-68	Gas Proportional 43-37/43-68	Gas Proportional 43-37/43-68	Gas Proportional 43-37/43-68
Probe Serial No.	168673	170813	148504/147403	147964/160688	148504/147403	147964/160688
Calibration Source	Co-60	Co-60	Th-230	Th-230	Tc-99	Tc-99
Probe Window Thickness mg/cm^2	N/A	N/A	0.8/0.8	0.8/0.8	0.8/0.8	0.8/0.8
Probe Size in cm^2	N/A	N/A	582/126	582/126	582/126	582/126
Instrument Efficiency (ε_i)	N/A	N/A	0.30/0.29	0.27/0.29	0.34/0.34	0.36/0.32
Surface Efficiency (ε_s)	N/A	N/A	0.25	0.25	0.25	0.25

Total Efficiency (4Π)	N/A	N/A	0.07/0.07	0.06/0.07	0.08/0.08	0.09/0.08
Radionuclides of Concern	Co-60	Co-60	N/A	N/A	Co-60	Co-60
Static MDC in dpm/100cm^2	N/A	N/A	N/A	N/A	141/682	137/775
Scan MDC in dpm/100cm^2	1.4 pCi/g	1.4 pCi/g	N/A	N/A	740/3577	720/4078
Gross DCGL	3.8 pCi/g	3.8 pCi/g	N/A	N/A	7100	7100
Data Points Required	16	16	N/A	N/A	16	16
Data Points Survey	Various > 16	Various > 16	Various > 16	Various > 16	Various > 16	Various > 16
Stat Test	Sign	Sign	N/A	N/A	Sign	Sign

14.5.1 Determination of Instrument Efficiency (ε_i) for Beta Surface Activity Measurements

The instrument efficiency (ε_i) is determined during calibration and is defined as the ratio between the net count rate (in counts per minute (cpm)) of the instrument and the surface emission rate of the calibration source for a specified geometry. The surface emission rate is the 2Π particle fluence that is affected by both the attenuation and backscatter of the radiation emitted from the calibration source. Equation 5 is used to calculate the instrument efficiency in counts per particle, although efficiency is typically reported as having no units or unitless).

Equation 5

$$\varepsilon_i = \frac{R_{S+B} + R_B}{q_{2\Pi}\left(W_A / S_A \right)}$$

where,

R_{S+B} = the gross count rate of the calibration measurement (cpm)
R_B = the background count rate in cpm
$q_{2\Pi}$ = surface emission rate of the calibration source (NIST traceable)
W_A = Active Area of the detector window (cm^2)
S_A = Area of the source (cm^2)

Note: This equation assumes that the dimensions of the calibration source be sufficient to cover the window of the instrument detector. If the dimensions of the calibration source are smaller than the detector's window, set W_A equal to the dimensions of the calibration source, i.e., set the quotient of W_A and S_A equal to 1.

The instrument efficiency is determined during calibration by obtaining static counts with the detector over a calibration source that has a National Institute of Standards and Technology (NIST) traceable surface emission rate. The 2Π particle fluence rate is corrected for decay, attenuation and scatter, then; the surface emission rate of the source must be corrected for the area subtended by the probe. Factors that can also affect the instruments efficiency are discussed below:

- Calibration Sources. The calibration sources selected emit alpha or beta radiation with energies similar to those expected from the contaminant in the field, i.e., similar to the expected radionuclide(s) of concern.
- Source Geometry Factors. The instrument efficiency is determined with a calibration source equal to or greater than the area of the probe.
- Source-to-Detector Distance. The detector is calibrated at a source-to-detector distance that is the same as the detector-to-surface distance used in the field.
- Window Density Thickness. The detector is calibrated with a probe window density thickness that is the same as the probe window density thickness used in the field.
- Detector-Related Factors - Ambient Conditions. If ambient conditions such as the temperature, pressure, and humidity vary significantly, during calibration and during field use, corrections to the detector's response will be considered.

A description of the calibration sources, procedures, and determination of instrument efficiencies for each instrument used to measure surface activity is provided in Appendix AA.

14.6 Sample Collection

The sample was sent to an off site laboratory for gamma spectroscopy analysis for radionuclide identification and quantification of activity levels.

15.0 DETECTION SENSSITIVITY-STATIC AND SCAN MINIMUM DETECTABLE CONCENTRATION (MDC)

15.1 Static MDC (Building Surfaces)

The static MDC is the level of radioactivity, on a surface, that is practically achievable by the overall measurement process. The conventional equation, Equation 6 is used to calculate instrument MDCs in dpm per 100 cm^2 when the background and sample are counted for the same time intervals.

Equation 6

$$MDC = \frac{3 + 4.65\sqrt{C_B * \mathrm{T_B}}}{\varepsilon_i \varepsilon_s \frac{W_A}{100\ cm^2} \mathrm{T_B}}$$

where,

C_B = Background count rate (cpm)
T_B = Background counting time (min)
ε_ι = the instrument efficiency (count per particle)
ε_s = the contaminated surface efficiency (particle per disintegration)
W_A = the area of the detector window (cm^2)

If the background and sample are counted for different time intervals, Equation 7 is used to calculate the MDC in dpm per 100 cm^2.

Equation 7

$$MDC = \frac{3 + 3.29\sqrt{R_B T_{S+B}\left(1 + \frac{T_{S+B}}{T_B}\right)}}{\varepsilon_i \varepsilon_s \frac{W_A}{100\ cm^2} T_{S+B}}$$

where,

R_B = background count rate (cpm)
T_B = background counting time (min)
T_{S+B} = sample counting time (min)
ε_i = the instrument efficiency (count per particle)
ε_s = the contaminated surface efficiency (particle per disintegration)
W_A = the area of the detector window (cm^2)

15.2 Scan MDC (Building Surfaces)

The scan MDC is determined from the minimum detectable count rate (MDCR) by applying conversion factors that account for detector and surface characteristics and surveyor efficiency. As discussed below, the MDCR accounts for the background level, performance criteria (*d'*), and observation interval. The observation interval during scanning is the actual time that the detector can respond to the contamination source. This interval depends on the scan speed, detector size in the direction of the scan, and area of elevated activity.

The scan MDC for structure surfaces is calculated using Equation 8.

Equation 8

$$Scan\ MDC = \frac{MDCR}{\sqrt{p}\ \varepsilon_i \varepsilon_s \frac{W_A}{100\ cm^2}}$$

where,

MDCR = discussed in Section 15.2.1
p = surveyor efficiency factor
ε_i = the instrument efficiency (count per particle)
ε_s = the contaminated surface efficiency (particles per disintegration)
W_A = the area of the detector window (cm^2)

15.2.1 Scanning Minimal Detectable Count Rate, (MDCR)

The minimum detectable number of net source counts in the scan interval, for an ideal observer, can be arrived at by multiplying the square root of the number of background counts (in the scan interval) by the detectability value associated with the desired performance (as reflected in *d'*) as shown in Equation 9.

Equation 9

$$MDCR = d'\sqrt{b_i} \times {}^{60}\!/_{i}$$

where,

d'	=	index of sensitivity (α and β error) – MARSSIM Table 6.5
b_i	=	number of background counts in scan time interval (count)
i	=	scan or observation interval (s)

Therefore, for an ideal observer, the number of source counts required for a specified level of performance can be arrived at by multiplying the square root of the number of background counts (determined to be ~ 270 cpm when using the detection system in the single channel mode) by the detectability value associated with the desired performance (as reflected in d') as shown in the equation below:

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

Where:

$$d' = \text{index of sensitivity } (\alpha \text{ and } \beta \text{ error})$$
$$b_i = \text{number of background counts in scan time interval}$$
$$d' = 3.28$$
$$b_i = 270x(1/60)$$
$$b_i = 4.5 \text{ counts}$$

Therefore:

$$S_i = 3.28\sqrt{4.5}$$
$$S_i = 6.9$$

The MDCR is then calculated using the formula below:

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

Where:

$i =$ scan time interval

Therefore:

$$\text{MDCR} = 6.9 x (60/1)$$

$$\text{MDCR} = 414 \text{ cpm}$$

The $MDCR_{surveyor}$ may then be calculated assuming a surveyor efficiency (p) of 0.5 as follows:

$$MDCR_{surveyor} = 414/\sqrt{0.5}$$

$$MDCR_{surveyor} = 585 \text{ cpm}$$

For example, the determined background count rate at the Building 150 outside ground areas when operating in the single channel mode site is approximately 270 cpm. The instrumentation uses a one second scan interval. Using an index of sensitivity of 3.28 (95% true positive rate and 5% false positive rate); the $MDCR_{surveyor}$ is 585 cpm (or 855 cpm-gross).

15.3 Scan MDC (Outdoor Land Areas)

In addition to the MDCR and detector characteristics, the scan MDC (in pCi/g) for land areas is based on the area of elevated activity, depth of contamination, and the radionuclide (*i.e.*, energy and yield of gamma emissions). If one assumes constant parameters for each of the above variables, with the exception of the specific radionuclide in question, the scan MDC may be reduced to a function of the radionuclide alone.

The corresponding minimum detectable exposure rate is determined for this detector and radionuclide. The manufacturer of this particular 2" by 2". NaI(Tl) scintillation detector quotes a count rate to exposure rate ratio for Co-60 of 430 cpm per μR/h. The minimum detectable exposure rate is calculated by dividing the count rate (585 cpm) by the count rate to exposure rate ratio for the radionuclide of interest (430 cpm per μR/h). The minimum detectable exposure rate for this example is 1.4 μR/h.

Modeling (using Microshield ™ Version 5.05) was used to determine the net exposure rate produced by 3.8 pCi/g of Co-60 contaminated soil.

The factors considered in the modeling included:

The dose point of 4 inches above the soil was used.

The density of 1.6 g/cm^3 was used for soil.

The depth of the area of elevated activity was 15 cm.

The areal dimension of the cylindrical area of elevated activity was .25 m^2.

The corresponding minimum detectable exposure rate was then determined for a 2" by 2" NaI(Tl).

The modeling code performed the appropriate calculations and determined an exposure rate of 3.94 μR/hr for Co-60 (which accounts for buildup).

A count rate due to the presence of Co-60 in soil at a concentration of 3.8 pCi/g can then be calculated by the following:

$$\frac{430 \text{ cpm}}{\mu\text{R/hr}}\, x \; 3.94 \; \mu\text{R/hr} = 1694 \text{ cpm}$$

The radionuclide concentration of Co-60 (scan MDC) necessary to yield the minimum detectable exposure rate (1.9 μR/hr) may be calculated using the following formula.

$$\text{ScanMDC} = \frac{3.8 \text{ pCi/g } (1.4 \; \mu\text{R/hr})}{3.94 \; \mu\text{R/hr}} = 1.4 \text{ pCi/g}$$

16.0 SURVEY PROCEDURES AND MEASUREMENT DATA INTERPRETATION

16.1 Surface Activity Measurements

Measurements to quantify surface activity levels represent the fundamental compliance measurements for buildings and structures. ISO-7503, NUREG-1507, and ASTM were used as technical guidance to ensure the accurate measurement of surface activity.

Equation 10 is used to calculate the surface activity in dpm per 100 cm^2.

Equation 10

$$A_S = \frac{R_{S+B} - R_B}{\varepsilon_i \varepsilon_s \dfrac{W_A}{100\, cm^2}}$$

where,

A_S = total surface activity (dpm/100 cm^2)
R_{S+B} = the gross count rate of the measurement in cpm,
R_B = the background count rate in cpm
ε_i = the instrument efficiency (counts per particle)
ε_s = the contaminated surface efficiency (particles per disintegration)
W_A = the area of the detector window (cm^2)

This equation has two efficiency terms, which account for differences between the conditions under which the detector is calibrated, and conditions under which the detector is used in the field. The instrument efficiency (ε_i) is discussed in Section 14.5.1, and is determined under ideal conditions in the laboratory. The surface efficiency, discussed below, is used to determine the 4Π total efficiency for a particular surface and condition.

16.1.1 Surface Efficiency (ε_s) for Surface Activity Measurements

The surface efficiency term in Equation 10 is used to determine the 4Π total efficiency for a particular surface and condition. Suitable values are based on the radiation and radiation energy, and are primarily impacted by the backscatter and self-absorption characteristics of the surface on which the contamination exists in the field. Backscatter is most affected by the energy of the radiation and the density of the surface material. Self-absorption characteristics or attenuation are also a function of the radiation's energy and surface condition. Surfaces typically encountered in the field include concrete, wood, dry wall, plaster, carpet, and metal. Surface conditions include both physical effects, such as scabbled concrete, and the effect of surface coatings, i.e., dust, paint, rust, water, and oil.

In the absence of experimentally determined surface efficiencies, ISO-7503-1 and NUREG 1507, provide conservative recommendations for surface efficiencies. ISO-7503-1, recommends a surface efficiency of 0.5 for maximum beta energies exceeding 0.5 MeV, and to use a surface efficiency of 0.25 for beta energies between 0.15 and 0.4 MeV and for alpha emitters. NUREG-1507 provides surface efficiencies based on studies performed primarily at ORISE. In general, NUREG-1507 indicates that the ISO rule-of-thumb for surface efficiencies are conservative, particularly for beta-emitting radionuclides with end-point energies between 0.25 MeV and 0.4 MeV.

The surface condition in Building 150 is concrete that was slightly covered with dust. Some of the radionuclides of concern occur naturally in concrete and produce a wide range of beta energies. The surface efficiency used in accordance with ISO-7503-1 is 0.25.

16.1.2 Probe Area Correction Factor for Surface Activity Measurements

In Equation 10, W_A is the size of the "active" area of the detector window. If the area of the detector window (cm^2) does not equal 100 cm^2, it is necessary to convert the detector response to units of dpm per 100 cm^2.

17.0 ANALYSIS AND RESULTS

17.1 Beta Scan Surveys of Survey Units, Analysis and Results

No elevated areas of activity distinguishable above background levels were detected during the scan surveys.

17.2 Fixed Beta Measurements of Survey Units, Analysis and Results

A final status survey was implemented per MARSSIM. One set of beta measurements were conducted in the appropriate reference areas and each of the survey units, each with a different but comparable survey instrument. The beta survey data was converted to units of dpm per 100 cm^2 so the results could be compared directly to the DCGL. The static MDCs for the instruments used are 136, 171, and 142 dpm per 100 cm^2, respectively. Table 12 gives a summary of the instruments used to evaluate beta surface activities and the efficiencies used to convert instrument readings to DCGL units. Table 12 also provides a summary of instrument sensitivities, the release criteria, and the number of samples taken in the survey units. The DCGL for the radionuclide of concern (Co-60) is 7,100 dpm per 100 cm^2. The sign test was used, since the radionuclide of concern is not present in background, to evaluate if the residual radioactivity levels in the survey units meet the release criterion (DCGL) and to test the null hypothesis (H_o) or alternative hypothesis (H_a) (see Section 13.0).

Beta results are provided in Appendix BB. The mean beta levels for all of the survey units are less than the DCGL. The Sign test also demonstrates that the concentration of beta radioactivity is less than the DCGL in all survey units. None of the beta measurements taken in any of the survey units exceeded the DCGL. Scatter plots of the beta measurement data are graphed paying particular attention to the elevated measurements (if any) and their spatial relationship. The elevated readings are within the normal range of their sampling distribution and the scatter plots show no systematic spatial trends for these values.

The H_o was rejected for all of the survey units; the survey units means are less than the beta DCGL, and there is no evidence of small areas of elevated activity.

17.3 Gamma Scan Surveys of Open Land Areas, Analysis and Results

A 100% walkover gamma scan survey was performed in all of the grids in the open land area surrounding Building 150. No elevated gamma readings were detected during the survey and none of the gamma scan survey results were above the MDCR of the counting system. The gamma scan survey data is presented in Appendix CC.

17.4 Soil Sample Analysis and Results

Surface and depth (2 feet below grade) soil samples were collected from the two survey units outside of Building 150. The sampling pattern was designed at 5- meter intervals. 41 samples were collected from Survey Unit #1 and 39 samples were collected form survey Unit #2. The samples were sent to an offsite laboratory for gamma spectroscopy analysis to quantify Co-60 concentrations. All of the results were below the $DCGL_W$. Table 13 and Table 14 present a summary of the results. The soil sample laboratory data is presented in Appendix Y.

Table 13. Building 150 Open Land Area Soil Sample Summary Table, Survey Unit #1

Sample ID	Sample Depth	Co-60 Results in pCi/g
B-7	Surface	1.85
D-7	Surface	0.045
D-7-D	2 Feet	-0.035
F-7	Surface	-0.027
F-7-D	2 Feet	-0.005
H-7	Surface	-0.018
A-8	Surface	0.017
C-8	Surface	0.11
C-8-D	2 Feet	0.149
E-8	Surface	-0.066
E-8-D	2 Feet	-0.021
G-8	Surface	0
G-8-D	2 Feet	-0.002
B-9	Surface	-0.038
B-9-D	2 Feet	0.021
D-9	Surface	0.017
D-9-D	2 Feet	0.021
F-9	Surface	0.069
F-9-D	2 Feet	-0.02

H-9	Surface	0.045
H-9-D	2 Feet	0.02
A-10	Surface	-0.06
C-10	Surface	0.37
C-10-D	2 Feet	0.041
E-10	Surface	0.25
E-10-D	2 Feet	-0.012
G-10	Surface	0.019
G-10-D	2 Feet	0.6
B-11	Surface	-0.014
F-11	Surface	0.14
F-11-D	2 Feet	0.11
H-11	Surface	0
C-12	Surface	-0.05
C-12-D	2 Feet	0.01
E-12	Surface	0.056
E-12-D	2 Feet	0.062
G-12	Surface	0.48
B-13	Surface	0.32
D-13	Surface	0.15
D-13-D	2 Feet	0.094

Table 14. Building 150 Open Land Area Soil Sample Summary Table, Survey Unit #2

Sample ID	Sample Depth	Co-60 Results in pCi/g
A-0	Surface	-0.024
C-0	Surface	0.002
E-0	Surface	-0.05
B-1	Surface	0.039
D-1	Surface	-0.016
F-1	Surface	-0.012
G-1	Surface	-0.064
A-2	Surface	0.04
C-2	Surface	0
E-2	Surface	-0.03
G-2	Surface	-0.011
B-3	Surface	0.002
D-3	Surface	0.001
F-3	Surface	-0.023
G-3	Surface	0.043
A-4	Surface	0.11
C-4	Surface	-0.009
E-4	Surface	-0.054
G-4	Surface	0
B-5	Surface	0.013

D-5	Surface	0.017
F-5	Surface	0.038
G-5	Surface	0.047
A-6	Surface	-0.044
C-6	Surface	0.031
E-6	Surface	0.037
G-6	Surface	-0.018
I-6	Surface	-0.038
K-6	Surface	0.175
I-8	Surface	0.06
I-10	Surface	-0.028
J-7	Surface	0.03
J-9	Surface	-0.04
J-11	Surface	0.022
K-8	Surface	0.013
K-10	Surface	0.013
L-7	Surface	-0.06
L-9	Surface	0.029

17.5 Evaluating Data from Final Status Surveys - Statistical Tests

Two statistical tests are used to evaluate fixed measurement data from final status surveys. For contaminants that are not present in background like Co-60 and H-3, the Sign test is used. To determine data needs for these tests, the acceptable probability of making Type I decision errors (0.05) and Type II decision errors (0.05) is established using the MARSSIM Data Quality Objectives (DQO) Process.

17.6 Sign Test

The sign test is a one-sample test that compares the distribution of a set of measurements in a survey unit to that of a set of measurements in a reference area. The test is performed by comparing the survey unit data directly to the $DCGL_{W.}$ The test is outlined below.

- Subtract each measurement, *Xi* , from the DCGLW to obtain the differences:
- Discard each difference that is exactly zero and reduce the sample size, *N*, by the number of such zero measurements.
- Count the number of positive differences. The result is the test statistic S+. Note that a positive difference corresponds to a measurement below the $DCGL_W$ and contributes evidence that the survey unit meets the release criterion.
- Large values of S+ indicate that the null hypothesis (that the survey unit exceeds the release criterion) is false. The value of S+ is compared to the critical values in presented in Table I.3 of MARSSIM. If S+ is greater than the critical value, *k*, in that table, the null hypothesis is rejected.

Table 15, Table 16, and Table 17 below present the statistical sign tests results for the Building 150 open land areas, and the former UST area. The sign test data for the Building 150 interior surveys unit are presented with the survey data for each of the survey units in Appendix BB.

Table 15. UST Area Sign Test Data

DCGL = **110** **pCi/g**

Sample ID #	Data pCi/g	$DCGL_W$-Data pCi/g	Sign Test Positive Difference
UST-1A	0.07	109.93	1
UST-2A	0.01	109.99	1
UST-3A	0.05	109.96	1
UST-4A	-0.01	110.01	1
UST-5A	0.00	110.00	1
UST-6A	0.00	110.00	1
UST-7A	0.29	109.71	1
UST-8A	0.05	109.95	1
UST-9A	0.04	109.96	1
UST-10A	0.05	109.95	1
UST-11A	0.04	109.96	1
UST-12A	0.01	110.00	1
USTA-1	15.90	94.10	1
USTA-2	0.15	109.85	1
USTA-3	0.01	109.99	1
USTA-4	0.15	109.85	1
USTB-1	-0.02	110.02	1
USTB-2	0.03	109.97	1
USTB-3	0.00	110.00	1
USTB-4	-0.03	110.03	1
Maximum =	**15.900**	**N =**	**20**
Average =	**0.839**	**S+ =**	**20**
Standard Deviation =	**3.546**	**Critical Value =**	**14**
Median =	**0.0335**		

Reject null hypothesis: The survey unit mean is less than the DCGL(w) - the survey unit may be released

Table 16. Building 150 Open Land Area Survey Unit #1 Sign Test Data

DCGL = 3.8 pCi/g

Sample ID #	Data pCi/g	$DCGL_W$-Data pCi/g	Sign Test Positive Difference
B-7	1.85	1.95	1
D-7	0.05	3.76	1
D-7-D	-0.04	3.84	1
F-7	-0.03	3.83	1
F-7-D	-0.01	3.81	1
H-7	-0.02	3.82	1
A-8	0.02	3.78	1
C-8	0.11	3.69	1
C-8-D	0.15	3.65	1
E-8	-0.07	3.87	1
E-8-D	-0.02	3.82	1
G-8	0.00	3.80	1
G-8-D	0.00	3.80	1
B-9	-0.04	3.84	1
B-9-D	0.02	3.78	1
D-9	0.02	3.78	1
D-9-D	0.02	3.78	1
F-9	0.07	3.73	1
F-9-D	-0.02	3.82	1
H-9	0.05	3.76	1
H-9-D	0.02	3.78	1
A-10	-0.06	3.86	1
C-10	0.37	3.43	1
C-10-D	0.04	3.76	1
E-10	0.25	3.55	1
E-10-D	-0.01	3.81	1
G-10	0.02	3.78	1
G-10-D	0.60	3.20	1
B-11	-0.01	3.81	1
F-11	0.14	3.66	1
F-11-D	0.11	3.69	1
H-11	0.00	3.80	1
C-12	-0.05	3.85	1
C-12-D	0.01	3.79	1
E-12	0.06	3.74	1
E-12-D	0.06	3.74	1
G-12	0.48	3.32	1
B-13	0.32	3.48	1
D-13	0.15	3.65	1
D-13-D	0.09	3.71	1
F-13	0.03	3.77	1
Maximum =	**1.85**	**N =**	**41**
Average =	**0.115366**	**S+ =**	**41**
Standard Deviation =	**0.312352**	**Critical Value =**	**26**
Median =	**0.021**		

Reject null hypothesis: The survey unit mean is less than the DCGL(w) - the survey unit may be released

Table 17. Building 150 Open Land Area Survey Unit #2 Sign Test Data

DCGL = **3.8** **pCi/g**

Sample ID #	Data pCi/g	$DCGL_W$-Data pCi/g	Sign Test Positive Difference
A-0	-0.02	3.82	1
C-0	0.00	3.80	1
E-0	-0.05	3.85	1
B-1	0.04	3.76	1
D-1	-0.02	3.82	1
F-1	-0.01	3.81	1
G-1	-0.06	3.86	1
A-2	0.04	3.76	1
C-2	0.00	3.80	1
E-2	-0.03	3.83	1
G-2	-0.01	3.81	1
B-3	0.00	3.80	1
D-3	0.00	3.80	1
F-3	-0.02	3.82	1
G-3	0.04	3.76	1
A-4	0.11	3.69	1
C-4	-0.01	3.81	1
E-4	-0.05	3.85	1
G-4	0.00	3.80	1
B-5	0.01	3.79	1
D-5	0.02	3.78	1
F-5	0.04	3.76	1
G-5	0.05	3.75	1
A-6	-0.04	3.84	1
C-6	0.03	3.77	1
E-6	0.04	3.76	1
G-6	-0.02	3.82	1
I-6	-0.04	3.84	1
K-6	0.18	3.63	1
I-8	0.06	3.74	1
I-10	-0.03	3.83	1
J-7	0.03	3.77	1
J-9	-0.04	3.84	1
J-11	0.02	3.78	1
K-8	0.01	3.79	1
K-10	0.01	3.79	1
L-7	-0.06	3.86	1
L-9	0.03	3.77	1
L-11	0.00	3.80	1
Maximum =	**0.175**	**N =**	**39**
Average =	**0.006179**	**S+ =**	**39**
Standard Deviation =	**0.046062**	**Critical Value =**	**25**
Median =	**0.001**		

Reject null hypothesis: The survey unit mean is less than the DCGL(w) - the survey unit may be released

18.0 CONCLUSION

Statistical tests were used to determine if the residual radioactivity levels in the survey units of the interior of Building 150, outside land areas around Building 150, and the UST area met the release criterion or did not exceed natural background radiation levels. The survey and sampling data show that the residual radioactively in Building 150, the land areas outside of Building 150, and the former UST area is less than the release criteria, and is similar to the background levels of radioactivity for a similarly constructed building, and geologically similar open land area, and that there is no evidence of small areas of elevated activity.

19.0 REFERENCES

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19.14 U.S. Nuclear Regulatory Commission, Draft Regulatory Guide DG-4006, August 1998, *Demonstrating Compliance With The Radiological Criteria For License Termination*

19.15 New World Technology, *Field Operations Procedures*

APPENDIX A

Radiation Work Permit (RWP)

APPENDIX B

Bioassay Sample Results

APPENDIX C

UST Geoprobe Sample Laboratory Data

Gamma Spectroscopy Analysis

Liquid Scintillation Analysis

APPENDIX D

NNMC Dig Permits

APPENDIX E

UST Survey Data

APPENDIX F

UST Exterior Survey Data

APPENDIX G

UST Interior Survey Data

APPENDIX H

UST Footprint Sample Laboratory Data

APPENDIX I
Import Backfill Soil Certification Letter

APPENDIX J
Import Backfill Soil Gamma Scan Survey Data

APPENDIX K

Post Remediation Soil Sample Laboratory Data

APPENDIX L

Building 150 Loose Surface Contamination Survey Data

APPENDIX M

Building 150 Debris Survey Data

APPENDIX N

Air Sampling Data

APPENDIX O

Building 150 Drain Sample Laboratory Data

APPENDIX P

Equipment Release Survey Data

APPENDIX Q

Building 150 Interior Trench Soil Sample Laboratory Data

APPENDIX R

Building 150 Exterior Trench Soil Sample Laboratory Data

APPENDIX S

Manhole Investigation Soil Sample Laboratory Data

APPENDIX T

Waste Manifests

UST's

Building 150

APPENDIX U

Building 150 Scoping Survey Data

APPENDIX V

Building Surface Background Reference Area Survey Data

APPENDIX W

Open Land Area Background Reference Area Survey Data

APPENDIX X

Building 150 Survey Unit Diagrams

APPENDIX Y

Building 150 Open Land Area Soil Sample Laboratory Data

Survey Unit #1

Survey Unit #2

APPENDIX Z

Radiological Survey Instrument Descriptions

Appendix AA

<u>Instrument Calibration Data</u>

1. Calibration Procedure

Each Model 43-47 probe was divided into eight overlapping regions over the entire face of the detector. Figure 1 below presents a diagram of the probe. Figure 2 presents a diagram of the eight probe regions.

Using Th-230 (Alpha), and Tc-99 (Beta) NIST traceable 4" by 4" plate sources, a series of five, one minute instrument responses were obtained over each of the eight detector regions and recorded for a total of forty readings. The source was placed at a distance of approximately ¼" from the detector. The total counts of the forty readings were summed and averaged to obtain an instrument efficiency for the entire active window of the detector which is 582 cm^2.

Diagram 1. Model 43-37 Detector Dimension Diagram

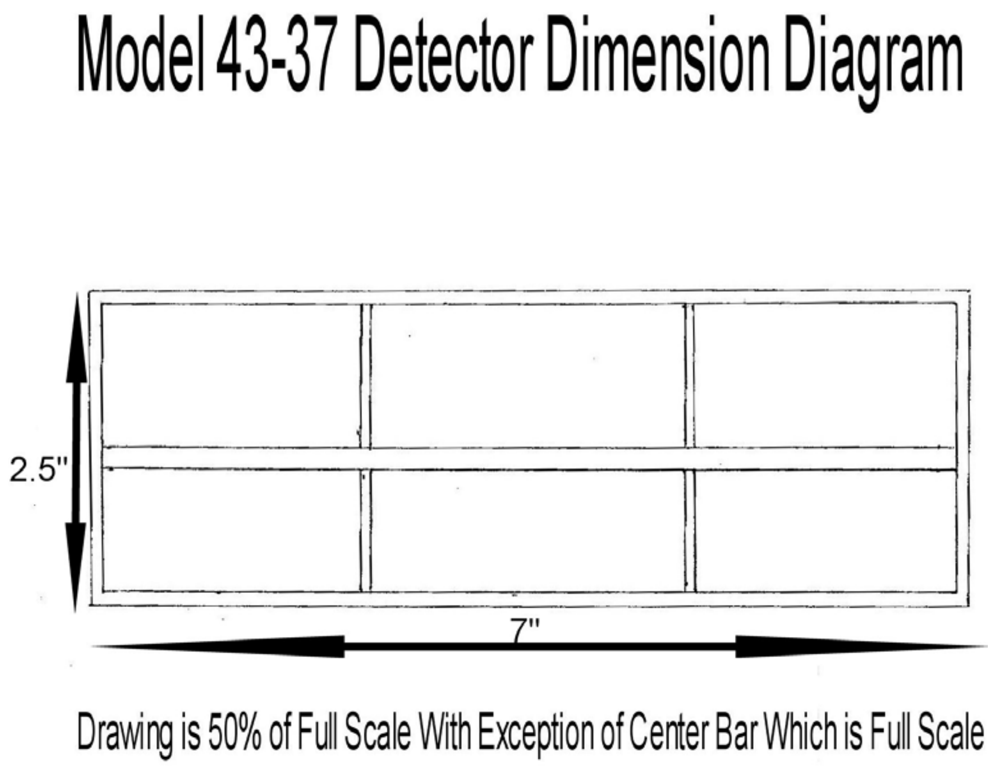
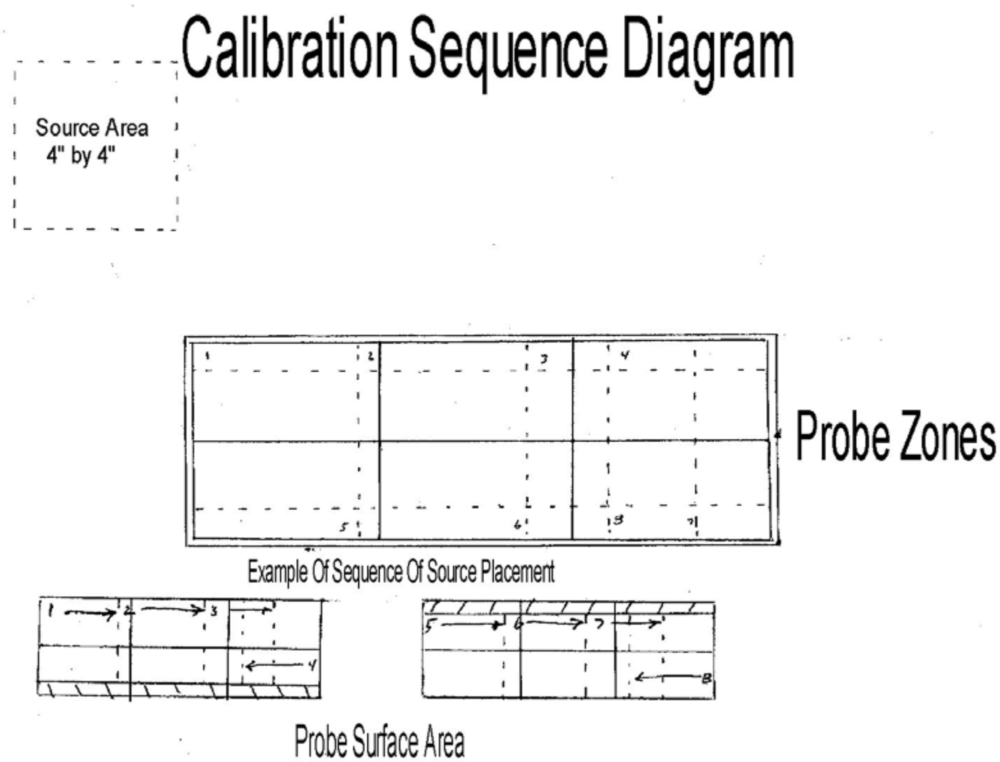


Diagram 2. Model 43-37 Detector Region Diagram



Calibration Data

Chi-Square Test Data Sheets

Instrument Response Check Logs

Appendix BB

Building 150 Beta Direct Measurement Data

Elevator Shaft Room

Room #1 Floor

Room #1 Ceiling

Room #2 Floor

Room #2 Ceiling

Room #3 Floor

Room #3 Ceiling

North Outer Wall

East Outer Wall

South Outer Wall

West Outer Wall

North/East Inner Wall

South/West Inner Wall

Roof Northern Survey Unit

Roof Southern Survey Unit

Appendix CC

Building 150 Open Land Area Gamma Scan Survey Data

Survey Unit #1

Survey Unit #2