THE UNIVERSITY OF ARIZONA®

Nuclear Reactor Laboratory

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August 18, 2011

Mr. John B. Hickman, Project Manager 10 CFR 50.82
Office of Federal and State Materials and Environmental Management Programs
Division of Waste Management and Environmental Protection
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

RE: Changes to the RAIs and Final Status Survey Plan for University of Arizona Nuclear Reactor Laboratory, License R-52, Docket 50-113

Dear Mr. Hickman,

This letter amends our previously submitted request for additional information (RAI) response and provides our revised Final Status Survey Plan.

In our RAI response dated March 26, 2010, we stated: "Residual concrete and steel from the reactor tank will be sampled to ensure that all residual radioactivity associated with reactor operations has been removed to background levels." We intended to completely remove the bottom six feet of the reactor tank, specified in our design drawings as eight inches thick. We designed a tank support system to stabilize the structure from above to allow removing this activated concrete. After draining the reactor tank and coring at the reactor midplane we discovered the tank concrete was up to 24 inches thick, not the expected 8 inch thickness, and possesses low, but detectable, concentrations of activation products throughout core midplane region. We cannot adequately support the remaining tank if we entirely removed the lower six feet of concrete wall and floor.

Based upon the analytical results on the concrete core samples, we will apply the NRC's license termination screening criteria for soils to the residual concrete tank structure. This can be achieved by removing up to 4 inches of concrete in the core mid plane region and tank floor. Approximately 10 inches of concrete will be removed in the thermalizer region. This approach will enable safe remediation of the concrete to the above referenced criteria while maintaining the structural integrity of the concrete tank. Analytical results of the concrete core show the dominant isotope is Europium-152 (^{152}Eu). In addition to meeting the NRC screening levels, we will reduce the ^{152}Eu DCGL from 8.7 $^{pCi}/_{g}$ to 7.0 $^{pCi}/_{g}$ to meet the EPA/NRC Memorandum of Understanding consultation triggers for ^{152}Eu.

FSME20
A020
NRR

The first enclosure adjusts our answers to questions 6 and 15 in our original RAI response (ML100920089). The second attachment transmits Revision 1 to the Final Status Survey Plan which addresses NRC comments and changes our approach to surveying and releasing the concrete reactor tank.

Our Reactor Committee reviewed and approved this submission last week.

In advance, I appreciate your prompt review of this modification. We expect commencing the final status survey on August 23. Feel free to contact me at (520) 621-6205.

Cordially,

Robert A. Offerle

Robert A. Offerle, Acting Director
Nuclear Reactor Laboratory

Attachments (2)

Copy to:
U.S. NRC Document Control Desk
Office of the Mayor of Tucson
Arizona Radiation Regulatory Agency
Dr. Leslie Tolbert, Vice President for Research
Radiation Control Office
Test, Research and Training Reactor Newsletter
U.S. NRC Region IV
LVI Environmental Services Inc.
Enercon Services, Inc.

6. Release Criteria

Background:
Section 2.2.3 of the DP discusses release criteria for the remaining equipment and surfaces. Currently the licensee proposes to us the NRC screening values but discusses the possibility of using alternative site-specific release criteria developed using a dose modeling software code such as the RESRAD family of codes.

Information Required:
A license condition is needed to require NRC review and approval if the licensee decides to use alternative site-specific release criteria.

~~**University of Arizona response:**~~
~~The University will use the release criteria specified in the DP and as amended in the letter dated March 26, 2010. The University will submit a revised license condition for review and approval by the NRC if alternative release criteria are developed for release of the NRL.~~

The August 2011 University of Arizona response:
The University will use the release criteria specified in the DP and as amended in the Final Status Survey Plan, Revision 1, promulgated in this letter. The University will submit a revised license condition for review and approval by the NRC if alternative release criteria are developed for release of the NRL.

15. Calculate DCGLs for Activated Concrete or Volumetrically Contaminated Bulk Materials

Background:
There are NRC license termination screening values associated with surface and soil concentrations presented in the UARR DP, Section 2.2.2, page 19. However, Derived Concentration Guideline Levels (DCGLs) need to be determined for activated concrete or volumetrically contaminated bulk materials which will be encountered during the decommissioning process.

Section 2.2.3, page 19, of the DP states that "Residual concrete and steel from the reactor tank will also be sampled to ensure that residual radioactivity meets the following NRC screening values for soils".

Information Required:
The licensee should provide a description and calculation of the DCGLs for activated concrete or volumetrically contaminated bulk materials for NRC to review prior to performing decommissioning activities at the UARR. The licensee should refer to the applicable guidance documents (NUREG-1575, Sections 4.3 and NUREG-1757, Volume 2, Section 4.2) to provide the information that should be supplied by the licensee to allow NRC staff to verify that the licensee has adequately developed DCGLs for activated concrete or volumetrically contaminated bulk materials.

~~**University of Arizona response:**~~

~~The University revises the statement referenced above in Section 2.2.3, page 19, to read: "Residual concrete and steel from the reactor tank will be sampled to ensure that all residual radioactivity associated with reactor operations has been removed to background levels".~~

The August 2011 University of Arizona response:

The University revises the statement referenced above in Section 2.2.3, page 19, to read: "We will sample the remaining concrete and steel reactor tank to ensure that all residual radioactivity associated with reactor operations meets the NRC's license termination screening criteria for soils. Furthermore, we will reduce the ^{152}Eu DCGL from 8.7 $^{pCi}/_{g}$ to 7.0 $^{pCi}/_{g}$ to meet the EPA/NRC Memorandum of Understanding consultation triggers for ^{152}Eu."

FINAL STATUS SURVEY PLAN

University of Arizona

Nuclear Reactor Laboratory

Nuclear Regulatory Commission

Facility Operating License R-52

UA-MCP-FS-01
Rev 1 [redacted]
August 1, 2011

Prepared by
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And

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Prepared for
The University of Arizona
Decommissioning of the Nuclear Reactor Laboratory
LVI Project Number 341015

FINAL STATUS SURVEY PLAN
University of Arizona
Nuclear Reactor Laboratory
Nuclear Regulatory Commission
Facility Operating License R-52

UA-MCP-FS-01
Rev 1 [redacted]
August 1, 2011

Prepared by: Dustin G. Miller
Health Physicist

Reviewed by: Kevin E. Taylor
Certified Health Physicist

Approved by: Corey E. DeWitt
Project Manager

Approved by: Robert Offerle
Director, Nuclear Reactor Laboratory

Approved by: Daniel Silvain
Chairman, Reactor Committee

Summary of Changes

Revisions to this plan will be tracked when revisions are issued. Changed sections will be identified by special demarcation in the margin. A summary description of each revision will be noted in the following table.

Revision Number	Date	Description of Change
Rev 0	May 21, 2011	Initial Issue
Rev 1	August 1, 2011	• Modified expected survey approach due to change in remediation strategy. • Addressed comments by NRC. • Modified the list of radionuclides of concern for the reactor pit in response to additional characterization data collected from the reactor pit concrete cores. • Modified description of reactor tank based on as found conditions that were different from the design drawings.

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Acronyms

Acronym	Description
CFR	Code of Federal Regulations
COC	Chain Of Custody
$DCGL_{emc}$	Derived Concentration Guideline Level, Elevated Measurement Criteria
DCGL	Derived Concentration Guideline Limit
DPM	Disintegrations Per Minute
DQO	Data Quality Objectives
EMC	Elevated Measurement Criteria
ENERCON	ENERCON Services, Inc.
FSS	Final Status Survey
HSA	Historical Site Assessment
HTD	Hard To Detect
LBGR	Lower Boundary of the Gray Region
LSC	Liquid Scintillation Counter
LVI	LVI Environmental Services, Inc.
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
MDA	Minimum Detectable Activity
MDC	Minimum Detectable Concentration
MDC_{scan}	Minimum Detectable Concentration via Scan
NaI	Sodium Iodide
NIST	National Institute for Standards and Testing
NRL	Nuclear Reactor Laboratory
PM	LVI Project Manager
QA/QC	Quality Assurance/Quality Control
RCO	University of Arizona Radiation Control Office
TEDE	Total Effective Dose Equivalent
UA	University of Arizona
UARR	University of Arizona Research Reactor
UPM	University of Arizona Project Manager
USNRC	United States Nuclear Regulatory Commission
VSP	Visual Sample Plan©
WRS	Wilcoxon Rank Sum

1.0 INTRODUCTION

The University of Arizona (UA) is in the process of decommissioning its Nuclear Reactor Lab (NRL). This survey plan is associated with the control measures that are intended to enable the Final Status Survey (FSS) to be applied to the NRL building surfaces and any associated soils beneath or surrounding the facility. The Decommissioning Planning for the NRL utilizes NUREG-1575, Revision 1 *Multi-Agency Radiation Survey and Site Investigation Manual* (MARSSIM) and other guidance documents. Additional resource documents are listed in Section 6.0, References.

This document describes the Final Status Survey (FSS) protocols to be used in terms of the process that will be used. The goal of implementing the FSS Plan is to document the data collected by radiological surveys that establishes the facility's final radiological status in support of terminating the applicable radioactive materials license. These Final Status Surveys therefore provide the inputs for the MARSSIM statistical evaluation process.

1.1 Purpose

The purpose of the FSS Plan is to describe the FSS process that will be used to demonstrate that the NRL facility and site comply with the US Nuclear Regulatory Commission (USNRC) annual dose limit criteria of 25-mrem/yr Total Effective Dose Estimate (TEDE), by meeting the radiological criteria listed in the Decommissioning Plan (DP) approved for use by the USNRC in Technical Specifications Amendment 20 for the UA NRL Possession-Only License R-52. The University ceased operation of the facility on May 18, 2010 and the reactor fuel was removed by the Department of Energy on December 23, 2010. The ultimate goal of the FSS is to support license termination for the facility.

Decommissioning operations may require modifications to this FSS Plan which may include, but are not limited to: adjustments of the boundaries of a survey unit, changes in the locations of survey points, the addition of survey units, or substitution of survey instruments. Modifications to this plan altering the intent or purpose of the FSS or affecting the overall quality of survey data shall be documented in the FSS report. A license amendment is not required for issuing a revision to this FSS Plan.

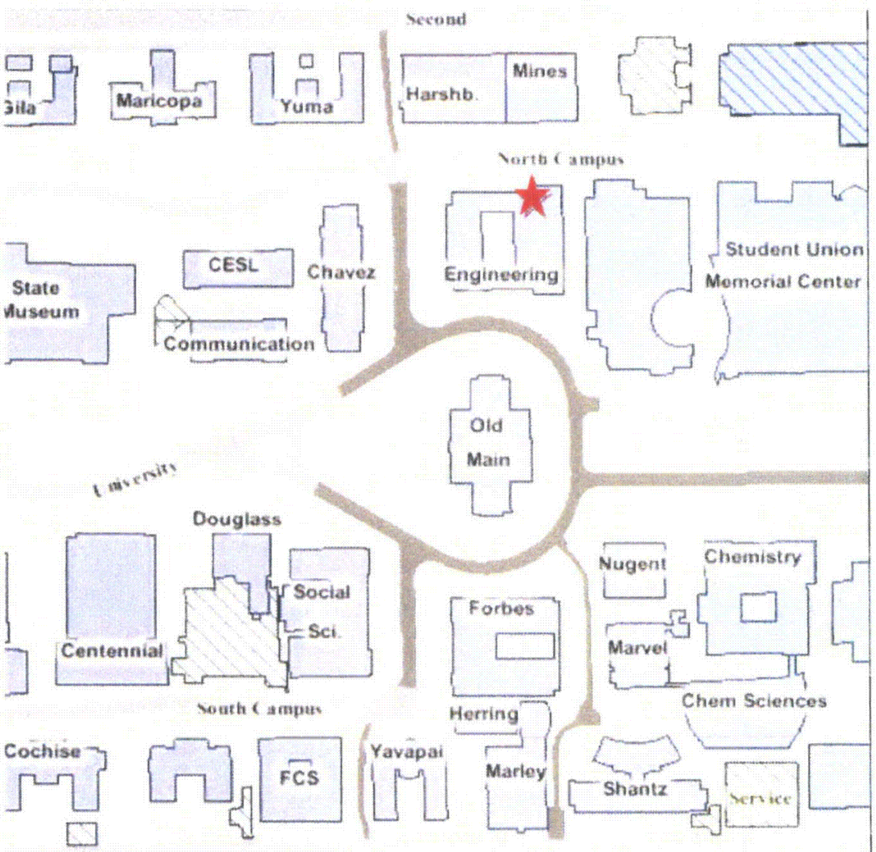
2.0 SITE DESCRIPTION

2.1 Facility Description

The University of Arizona Research Reactor (UARR) is a TRIGA pool-type reactor designed and constructed by General Atomic Division of General Dynamics Corporation. The reactor is located within the University of Arizona Nuclear Reactor Laboratory (NRL) on the 325 acre campus of the University of Arizona (University) in Pima County, Arizona in the city of Tucson. The University is about 65 miles north of the Mexican border at Nogales, AZ, 110 miles south east of Phoenix, AZ and 120 miles from the western border of New Mexico.

The campus is centrally located in the city of Tucson, Arizona, and is roughly bounded by East Speedway Boulevard, North Campbell Avenue, East Sixth Street, and North Park Avenue. The reactor was operated by the Nuclear Reactor Laboratory (NRL) under operating license R-52 issued by the USNRC. The NRL is located on the Main Campus, in the Engineering Building, on the first floor of the north wing. The physical location of the Engineering Building is shown in Figure 2-1.

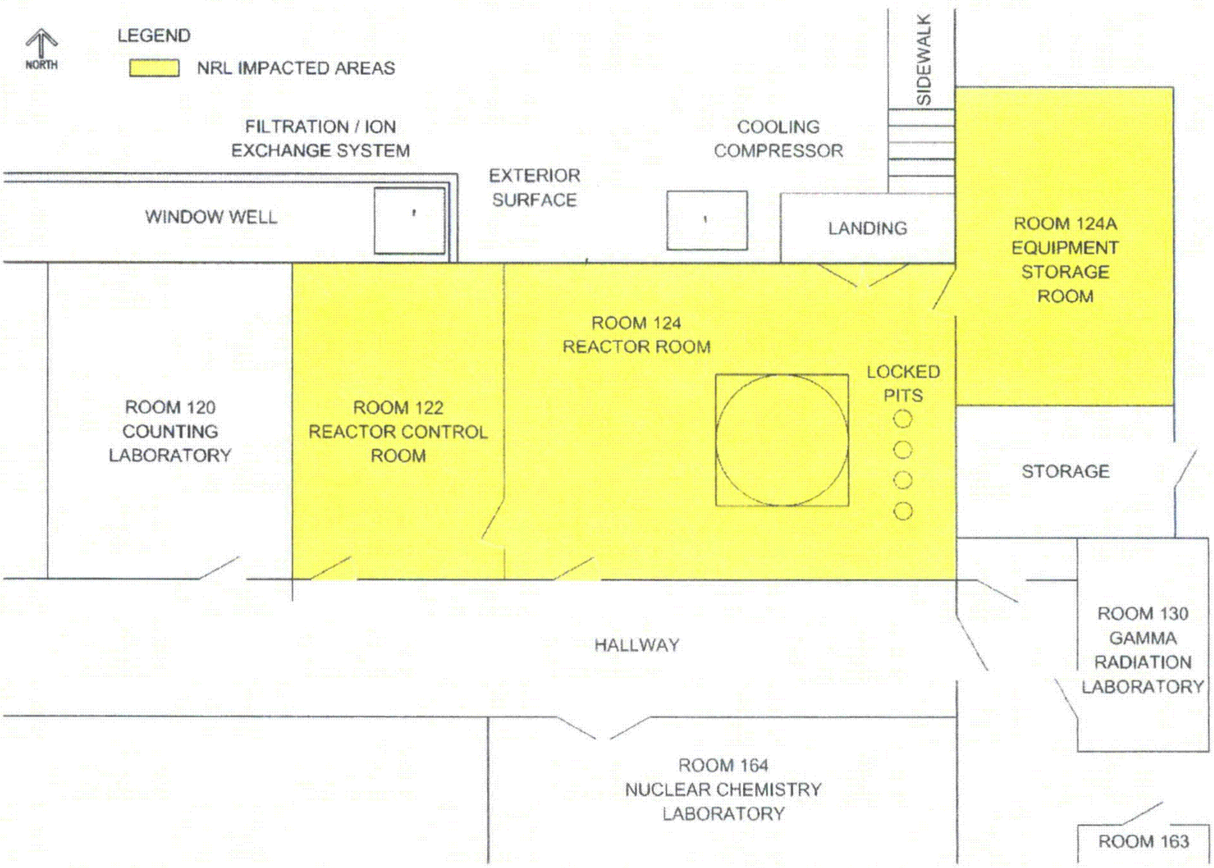
Figure 2-1: Location of Engineering Building on the UA Campus



Four adjacently located rooms in the Engineering Building are permanently established as the Nuclear

Reactor Laboratory and are designated a controlled access area. These are: 1) Room 122, the Control Room; 2) Room 124, the Reactor Room; 3) Room 216, and 4) Room 124A, the Equipment Storage and Experiment Setup Room. Rooms 122, 124, and 124A are shown in Figure 2-2. Room 216 is the room directly above the reactor room, which was originally designed to receive a beam of neutrons from the reactor. A 9-inch diameter hole penetrates the floor of this room directly above the center of the reactor core, and a 30 inch by 36 inch hatch to the roof above is directly over the hole. Little use was made of this beam capability, so during the refurbishment of the reactor in 1972, no provision was made in the new bridge for a hole to accommodate the beam tube. At this time the hole in the floor is capped and locked, and the room is used for storage of reactor supplies and departmental records.

Figure 2-2: Section of the Engineering Building North Wing Showing the Reactor Laboratory



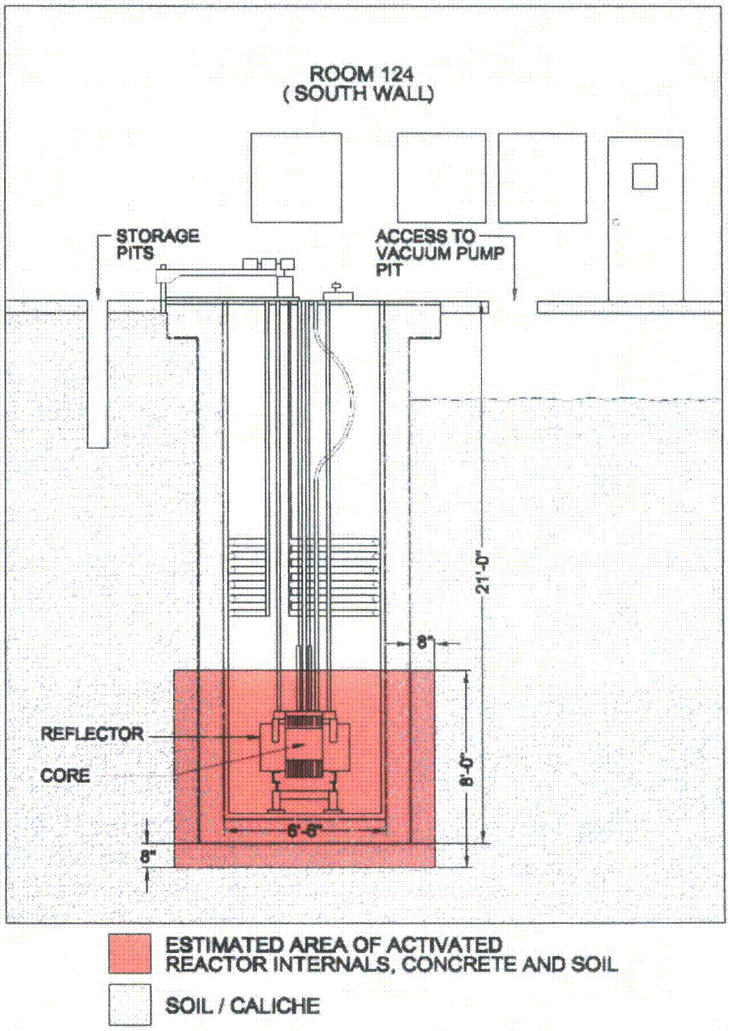
2.2 Reactor Description

The reactor is designated as a Mark I TRIGA reactor and operated at a maximum licensed steady state power of 110 kW (thermal), with a pulsing capability up to peak powers of approximately 650 MW. TRIGA stands for Test, Research, Isotope production, General Atomics.

The reactor core is located in a pool-type tank, which is 21 feet deep and 6.5 feet in diameter, located

below grade in Room 124, and shown in Figure 2-3. The pit contains a ¼ in. steel tank resting on a 1-ft-thick concrete slab. According to the design drawings, approximately 8 in. of poured concrete surrounds the outside of the tank. The steel tank served as the inner form for pouring the concrete and the outer form was a corrugated steel cylinder, which was left in place after pouring. The design drawings also indicated that the inside of the steel tank was covered on the sides by a layer of Gunite approximately 2 in. thick and on the bottom by a layer approximately 4 in. thick with the entire inner surface of the Gunite coated with Amercoat (an epoxy-base paint). After the core was removed, concrete and soil samples were collected at the core mid-plane in the tank wall and the core centerline in the tank floor. The actual wall thickness of the concrete outside the tank liner varies from 14 to 24 inches thick and there is no indication of an outer corrugated steel liner. Additionally, the gunite was 3-4 inches thick on the sides and up to 8 inches thick on the tank floor.

Figure 2-3: Elevation view of Reactor Tank



2.3 Radionuclides of Concern

During characterization of the site, locations measured less than MDC on the characterization measurements. As a conservative approach, typical Radionuclides of Concern (ROC) for a reactor with no fuel cladding failures were chosen for the building surfaces. The radionuclides of concern selected for soils were based on the history of the reactor and the radionuclides delineated in the WMG Component Activation Analysis Report that was submitted with the DP. During remediation of the reactor, additional characterization samples were collected in the concrete along the tank wall and the tank floor. These samples indicated that the only radionuclides of concern are Eu-152, Co-60, and tritium.

The following table provides the comprehensive list of the radionuclides of concern for the UA NRL and an updated matrix of the applicable area/media of where the radionuclide is of concern.

Table 2-1: Radionuclides of Concern

Radionuclide	Half-life	Building Surfaces	Soils	Activated Metal	Concrete
Tritium (H-3)	12.32 years	X	X	X	X
Carbon-14	5715 years	X		X	
Manganese-54	312.1 days	X		X	
Iron-55	2.73 years	X		X	
Cobalt-60	5.271 years	X		X	X
Nickel-63	100 years	X		X	
Technetium-99	2.13E+05 years	X		X	
Cesium-137	30.07 years	X			
Europium 152	13.54 years		X	X	X
Europium 154	8.593 years		X	X	X
Calcium-45	162.7 days			X	
Chromium-51	27.702 days			X	
Iron-59	44.51 days			X	
Cobalt-58	70.88 days			X	
Nickel-59	7.60E+04 years			X	
Zinc-65	243.8 days			X	
Niobium-94	2.00E+04 years			X	

3.0 FINAL STATUS SURVEY PROCESS

3.1 Overview

The FSS will include the four rooms of the NRL, all residual surfaces of the reactor tank, the concrete beneath the cooling system, the concrete beneath the Ion Exchange/filtration system, and the exposed soils that were potentially affected by the neutron flux of the reactor. The Activation Analysis and Component Characterization Report, Reference 2 of the DP, indicates that soils below the reactor pedestal and along the sides of the tank may be impacted from the neutron flux of the reactor.

Following all reactor dismantlement, demolition, and remediation activities, an FSS will be prepared per the process described in this plan. The following text from Section 4.3.2 of the UA DP (ref 8) describes the potential survey unit classifications for the NRL FSS. Table 3-1 provides a summary of this text.

> Class 1 Areas – The reactor pit is the only Class 1 area in the NRL. Areas in the bottom of the reactor pit will be greater than the derived concentration guideline levels (DCGL).
>
> Class 2 Areas – The Class 2 areas of the NRL are limited to the floors and walls up to 2 meters of the Reactor Room and the Equipment Storage Room, Room 124 and Room 124A, respectively. Radioactive material and sources were handled in these areas, however, if residual contamination is found, it should be a small fraction of the DCGL.
>
> Class 3 Areas – The Class 3 areas of the NRL are the walls above 2-meters and ceilings in Rooms 124 and 124A; all surfaces in Room 122, the Reactor Control Room; and all surfaces in Room 216, the Second Floor Storage Room. The walls above 2-meters in Rooms 124 and 124A have very little potential to have any level of contamination, but there are no barriers to prevent possible contamination from being spread to these areas. There is also very little potential for any contamination in Rooms 122 and 216 and around the concrete pads of the cooling and filtration equipment, but these are a part of the NRL licensed area and are to be included in the FSS.

Table 3-1: NRL MARSSIM Classifications from UA DP

Survey Area		MARSSIM Classification
Reactor Pit	All surfaces	1
Room 124	Floor and Walls <2m	2
	Walls >2m and Ceiling	3
Room 124A	Floor and Walls <2m	2
	Walls >2m and Ceiling	3
Room 122	All surfaces	3
Room 216	All surfaces	3

This plan describes the following five (5) major steps in the FSS process:

1. Survey preparation
2. Survey design
3. Data collection
4. Data validation, assessment and evaluation
5. Documentation of survey results

3.1.1 Survey Preparation

Survey preparation is the first step in the FSS process and occurs after remediation is completed. Where remediation was required, a post-remediation survey is performed to confirm that remediation was successful prior to initiating final survey activities. Following the post-remediation survey, the FSS is performed.

The area to be surveyed is isolated and/or controlled to ensure that radioactive material is not reintroduced into the area from ongoing demolition or remediation activities nearby, and to maintain the final configuration of the area. Tools, equipment, and materials not needed to support survey activities are to be removed from the survey area. Routine access, material storage, and worker transit through the area are not allowed.

3.1.2 Survey Design

The survey design process establishes the methods and performance criteria used to conduct the survey. Survey design assumptions are documented in the FSS Report in accordance with this plan. Building surfaces are organized into survey areas and classified by contamination potential of the area. Survey unit size is based on the assumptions in the dose assessment models in accordance with the guidance provided in the MARSSIM. The percent coverage for scan surveys is discussed in Section 3.5.1. The number and location of structural surface measurements for total and removable contamination (and/or structural volumetric samples) and soil samples are established in accordance with Section 3.4.2. Investigation levels are also established in accordance with Section 3.7.1.

3.1.3 Data Collection

After preparation of the FSS design, the FSS data is collected. Trained and qualified personnel perform the necessary measurements using calibrated instruments in accordance with approved procedures and instructions contained in the survey package.

3.1.4 Data Assessment and Evaluation

Survey data assessment is performed to verify that the data are sufficient to demonstrate that the survey unit meets the unrestricted use criterion (i.e., the Null Hypothesis may be rejected). Statistical analyses are performed on the data and the data are compared to investigation levels. Depending on the results of an investigation, the survey unit may require further remediation, reclassification, and/or resurvey. Graphical representations of the data, such as posting plots or histograms, may be generated to provide qualitative information from the survey and to verify the assumptions in the statistical tests, such as spatial independence, symmetry, data variance and statistical power. Additional data needs, if required, are identified during this review.

3.1.5 Documentation of Survey Results

Survey results are documented by Survey Area in the FSS report. The FSS data is reviewed, analyzed, and processed, and the results documented in the FSS Report. The FSS Report provides the necessary information to support the decision to release the survey units for unrestricted use. The FSS Report provides the necessary data and analyses from the FSS and is submitted by the licensee to the USNRC along with a request for license termination.

3.2 Release Criteria

The release criteria for this FSS are listed in Section 2.2.3 of the UA Decommissioning Plan and approved by the NRC through Amendment 19 of the UA Technical Specifications. In the University's response (dated March 26, 2010) to a Request for Additional Information (RAI) from the NRC pertaining to the submitted DP, the University stated that the "residual concrete and steel from the reactor tank will be sampled to ensure that all residual radioactivity associated with reactor operations has been removed to background levels." The RAI response letter was incorporated by reference in the NRC's approval of Amendment 19. Following removal of reactor internals and pool water, concrete samples were collected from the sidewalls and the tank floor to bound the activated regions of the tank and surrounding soils. The cores revealed that on average,the concrete tank walls are two to three times thicker than indicated in the design drawings. The cores also revealed detectable levels of radioactive materials throughout a significant portion of the concrete in the lower section of the tank. The increase in volume of concrete raised concerns over the safety of activities required to remove all tank concrete to background levels. After discussions with the NRC, the University submitted a request for revision to the DP to apply the NRC Soil Screening criteria in NUREG 1757, Volume 1, Appendix B, Table B.2 to residual concrete in the reactor tank. Based upon the analytical results of tank concrete core samples, tank remediation can be conducted to meet these release criteria without compromising the structural integrity of the tank.

The surface release criteria discussed in Section 3.2.1 will be applied to the remaining portions of the reactor tank and NRL area in general. The soil release criteria discussed in Section 3.2.2 will be applied to the concrete activated by the neutron flux, i.e. the portion of the tank where the liner was removed .

The term "release criteria" is also known as the Derived Concentration Guideline Level (DCGL) in the MARSSIM. The terms "release criteria" and "DCGL" are used synonymously throughout this document.

3.2.1 Building Surfaces

The NRC screening values in NUREG 1757, Volume 2, Appendix B, Table B.1 for building surfaces and materials destined to remain in place have been approved as the release criteria for this FSS and are listed in Table 3-3.

Table 3-3: NRC License Termination Screening Levels for Surfaces

Radionuclide	Screening Level for Unrestricted Release (dpm/100cm^2)	Average Beta Energy (keV)	Detectable with Field Survey Instrument
Tritium (H-3)	1.2E+08	5.685	No
Carbon-14	3.7E+06	49.47	Yes
Manganese-54	3.2E+04	Electron Capture	No
Iron-55	4.5E+06	Electron Capture	No
Cobalt-60	7.1E+03	95.79	Yes
Nickel-63	1.8E+06	17.13	No
Technetium-99	1.3E+06	84.6	Yes
Cesium-137	2.8E+04	170.8	Yes

Because the most limiting release criteria for those isotopes detectable with field survey instruments is cobalt-60 (7100 dpm/100cm^2), it will be assumed that all measureable activity is from cobalt-60 and ENERCON will apply a single release criterion of 7100 for total measureable beta surface activity. Also, ENERCON will use a technetium-99 beta source to calculate the detection efficiency for survey instruments. Because the average beta decay energy for technetium-99 is less than the average beta decay energy for colbalt-60, this will add to the conservatism in each measurement.

The screening criteria in Table 3-3 are based on the assumption that the fraction of removable surface contamination is equal to 0.1. Therefore, removable surface contamination surveys must demonstrate that the removable activity is less than or equal to 10% of the Table 3-3 criteria. Because ENERCON is applying a total beta activity release criterion of 7100 dpm/100cm^2, the removable contamination release criterion will be 710 dpm/100cm^2.

3.2.2 Release Criteria for Soils

There has been no indication that the subsurface soils have been impacted by activities at the NRL, however, the WMG Activation Analysis did indicate that soil within the neutron flux field of the reactor could potentially contain Cobalt-60 from the activation of Cobalt-59 in the soil. Demolition of the activated portions of the reactor tank could introduce other radionuclides of concern in to the surface area of the soils, therefore, these radionuclide were included in Section 2.3. Should post remediation surveys indicate the presence of radionuclides of interest in the soils, they too will be remediated. Soils remaining in place will be shown to meet the NRC screening values presented in Table 3-4. The Eu-152 screening value is greater than the EPA and NRC Memorandum of Understanding (MOU) consultation trigger of 7.0 pCi/g. Therefore, the Eu-152 value has been reduced to the value contained in Table 1, *Consultation Triggers for Residential and Commercial/Industrial Soil Contamination*, of the EPA/NRC MOU.

Table 3-4. NRC License Termination Screening Levels for Soils

Radionuclide	Default DCGL (pCi/g)	Decay Mode	Analytical Method
Cobalt-60	3.8	Beta/gamma	Gamma Spectroscopy
Tritium (H-3)	110	Low-Energy Beta	Liquid Scintillation
Europium-152	7	Beta/Gamma	Gamma Spectroscopy
Carbon-14	12	Low-Energy Beta	Liquid Scintillation
Iron-55	10,000	Electron Capture	Liquid Scintillation
Nickel-63	2,100	Low-Energy Beta	Liquid Scintillation
Cesium-137	11	Beta/Gamma	Gamma Spectroscopy
Europium-154	8	Beta/Gamma	Gamma Spectroscopy
Calcium-45	57	Beta	Liquid Scintillation
Nickel-59	5,500	Electron Capture	Liquid Scintillation
Chromium-51	N/A (Half-Life 27.7 days)	Electron Capture	Liquid Scintillation
Iron-59	N/A (Half-Life 44.5 days)	Beta/Gamma	LS or Gamma Spec.
Cobalt-58	N/A (Half-Life 70.9 days)	Electron Capture	Gamma Spectroscopy
Zinc-65	Cobalt-60 Surrogate	Beta/Gamma	Gamma Spectroscopy
Niobium-94	Cobalt-60 Surrogate	Gamma	Gamma Spectroscopy

3.3 Area Classification

Area classification ensures that the number of measurements and the scan coverage is commensurate with the potential for residual contamination to exceed the approved DCGLs. Characterization of the radiological status and history of the site has been completed and initial classifications have been determined based on this characterization, operational radiological surveys, and the DP. The structures were classified following the guidance in Section 4.4 of the MARSSIM and reiterated in sections 3.3.1 and 3.3.2.

3.3.1 Non-Impacted Areas

Non-Impacted areas have no reasonable potential for residual contamination because there was no known impact from site operations. Non-impacted areas will not be required to be surveyed beyond what is completed as a part of site characterization to confirm the area's non-impacted classification.

3.3.2 Impacted Areas

Impacted areas may contain residual radioactivity from licensed activities. Based on the levels of residual radioactivity present, impacted areas are further divided into Class 1, Class 2 or Class 3 designations. Class 1 areas have the greatest potential for residual activity while Class 3 areas have the least potential for impacted areas. Each classification will typically be bounded by areas classified one step lower to provide a buffer zone around the higher class. Exceptions occur when an area is surrounded by a significant physical barrier that would make transport of residual activity unlikely from one area to the adjacent area. In such cases, each area will be classified solely on its own merit using the most reliable information available. The class definitions provided below are from Section 4.4 of the MARSSIM.

Class 1

Areas that have, or had prior to remediation, a potential for radioactive contamination (based on site operating history) or known contamination (based on previous radiological surveys). Examples of Class 1 areas include: 1) site areas previously subjected to remedial actions, 2) locations where leaks or spills are known to have occurred, 3) former burial or disposal sites, 4) waste storage sites, and 5) areas with contaminants in discrete solid pieces of material high specific activity. Note that areas containing contamination in excess of the DCGL prior to remediation should be classified as Class 1 areas.

Class 2

These areas have, or had prior to remediation, a potential for radioactive contamination or known contamination, but are not expected to exceed the DCGL. To justify changing an area's classification from Class 1 to Class 2, the existing data (from the HSA, scoping surveys, or characterization surveys) should provide a high degree of confidence that no individual measurement would exceed the DCGL. Other justifications for this change in an area's classification may be appropriate based on the outcome of the DQO process. Examples of areas that might be classified as Class 2 for the final status survey include: 1) locations where radioactive materials were present in an unsealed form (e.g., process facilities), 2) potentially contaminated transport routes, 3) areas downwind from stack release points, 4) upper walls and ceilings of some buildings or rooms subjected to airborne radioactivity, 5) areas where low concentrations of radioactive materials were handled, and 6) areas on the perimeter of former contamination control areas.

Class 3

Any impacted areas that are not expected to contain any residual radioactivity, or are expected to contain levels of residual radioactivity at a small fraction of the DCGL, based on site operating history and previous radiological surveys. Examples of areas that might be classified as Class 3 include buffer zones around Class 1 or Class 2 areas, and areas with very low potential for residual contamination but insufficient information to justify a non-impacted classification.

3.3.3 NRL Classifications

For this FSS, the Reactor Pit is considered Class 1. The Reactor Room and the Equipment Storage Room, Room 124 and Room 124A, all lower surfaces (i.e., floors and walls up to 2 meters) are considered Class 2 while all upper surfaces are Class 3. The Reactor Control Room, Room 122, and the Second Floor Storage Room, Room 216, are considered Class 3 for all surfaces. The building exterior and surrounding land area is non-impacted except for concrete surfaces beneath the filtration/ion exchange system and cooling compressor located outside and adjacent to the north wall of the building. These two exterior concrete pads will be surveyed in conjunction with the interior Class 2 surface areas. All areas are subject to increasing classification based on additional surveys.

3.3.4 Changes in Classification

Data from operational surveys performed in support of decommissioning, routine surveillance and any other applicable survey data may be used to change the initial classification of an area up to the time of FSS commencement as long as the classification reflects the levels of residual radioactivity that existed prior to remediation. After the FSS of a given survey unit begins, the basis for reclassification will be documented. If, during the conduct of an FSS, survey sufficient evidence is accumulated to warrant an investigation and reclassification of the survey unit, the survey may be terminated without completing the

survey unit package. A reclassification of a survey unit because of detected radioactivity may only result in a more stringent classification, unless demolition or remediation activities result in the entire survey unit being eliminated, such as a building being demolished, rather than surveyed for release.

3.4 Establishing Survey Units

Land areas, structures, and systems are made up of at least one smaller area defined as a survey unit. Each land area, structure, or system may have multiple survey units of differing classification since data acquisition, analysis, and reporting are done on a survey unit basis. The survey unit release records applicable to a larger survey area may be combined into one report for submission to USNRC.

3.4.1 Survey Unit

Survey areas may be divided into smaller survey units. Survey units are areas that have similar characteristics and contamination levels and will be assigned only one classification. Survey areas may include survey units of differing classifications since the site and facility are surveyed, evaluated, and released on a survey unit basis.

3.4.2 Survey Unit Size

Section 4.6 of the MARSSIM provides suggested sizes for survey units. However, as stated in the MARSSIM, the suggested survey unit sizes were based on a finding of reasonable sample density and consistency with commonly used dose modeling codes. Table 3-5 lists the recommended survey unit size for each applicable classification.

Table 3-5: Survey Unit Size

	Maximum[1]	
Classification	**Buildings**	**Open Land**
Class 1	100-m^2	2,000-m^2
Class 2	1,000-m^2	10,000-m^2
Class 3	No limit	No limit

[1] From MARSSIM Section 4.6

3.5 Survey Design

This section describes the methods and data required to determine the number and location of measurements or samples in each survey unit, the coverage for scan surveys, and requirements for measurements in background reference areas. The design activities described in this section are documented in a survey package for each survey unit. Survey design includes the following:

- Scan Survey Coverage
- Number of Locations

- Background Reference Areas as necessary
- Reference Grid and Sample Location

For the FSS of the UA NRL, the survey design to be implemented is expected to follow the design parameters shown in Table 3-6 provided that conditions are not altered during the dismantlement and demolition of the reactor tank. This plan describes what conditions may cause the design to be altered and the process required to alter the design.

Two removable contamination samples (wipe samples) will be collected at the location of each direct total surface contamination measurement. One sample will be analyzed for gross beta and gross alpha in a portable alpha/beta swipe counter. The second swipe will be analyzed for H-3 and other hard to detect beta emitting isotopes of concern using a liquid scintillation counter (LSC).

As shown in Table 3.6, characterization data shows that all wall and floor measurements were significantly less than half of the most restrictive release criteria shown in Section 3.3.

Table 3-6: Expected Survey Design

Survey Area		MARSSIM Classification	Area (m^2)	Characterization Data		Relative Shift	Number of Locations
				Mean Beta	Std Dev		
Reactor Pit	Lower Surfaces (concrete)	1	29	N/A	N/A	3	14
	Upper Surface (tank)	1	25	N/A	N/A	3	14
Room 124	Floor and Walls <2m Exterior concrete pads	2	102	253	1625	3	14
	Walls >2m and Ceiling	3	100	253	1625	3	14
Room 124A	Floor and Walls <2m	2	61	2293	742	3	14
	Walls >2m and Ceiling	3	45	2293	742	3	14
Room 122	All surfaces	3	107	320	438	3	14
Room 216	All surfaces	3	182	891	910	3	14

Note: Elevated readings in Room 124A were attributed to shine from four (4) tons of natural uranium contained in the Subcritical assembly.

3.5.1 Scan Survey Coverage

The area covered by scan measurements is based on the survey unit classification as described in Table 2 of the MARSSIM and is summarized in Table 3-7 below. A 100% accessible area scan of Class 1 survey units will be required. The emphasis is placed on scanning the higher risk areas of Class 2 survey units such as soils, floors and lower walls. Scanning in Class 3 survey units will focus on likely areas of contamination based on professional judgment and will generally consist of a minimum of the 1-m^2 area around the selected direct measurement location.

Table 3-7: Minimum Scan Coverage

Classification	Required Minimum Scan Coverage
Class 1	100% of accessible areas
Class 2	≥10% of accessible areas
Class 3	Judgmental, but generally a 1-m^2 area around each data location

3.5.2 Sample Size Determination

The MARSSIM describes the process for determining the number of survey measurements necessary to ensure a data set sufficient for statistical analysis. Sample size is based on the relative shift, the Type I and II errors, sigma, and the specific statistical test used to evaluate the data. These items are explained in greater detail in the following sections.

3.5.2.1 Statistical Test Determination

The Sign test will be the only statistical test implemented for this survey. Unity rules, surrogate methodologies, or combinations of unity rules and surrogate methodologies, as described in the MARSSIM and NUREG-1505 Chapters 11 and 12 will be used as necessary for this survey.

The Wilcoxon Rank Sum (WRS) test will not be implemented in this survey due to the insignificance of background radionuclides compared to the DCGL for structures. Beta scan measurements of the floors and walls during the Post Shutdown Characterization of the NRL were less than 10% of the most restrictive release criteria, i.e., Cobalt 60, shown in Table 3-3. It will not be necessary to subtract material background from the radiological measurements in order for the survey units to meet the release criteria for surface measurements.

3.5.2.2 Establish Decision Errors

The probability of making decision errors is controlled by hypothesis testing. The survey results will be used to select between one condition of the environment (the null hypothesis) and an alternate condition (the alternative hypothesis). These hypotheses, chosen from MARSSIM Scenario A, are defined as follows:

- Null Hypothesis (H_0): The survey unit does not meet the release criteria.
- Alternate Hypothesis (H_a): The survey unit does meet the release criteria.

A Type I decision error would result in the release of a survey unit containing residual radioactivity above the release criteria. It occurs when the null hypothesis is rejected, but in reality is true. The probability of making this error is designated as "α."

A Type II decision error would result in the failure to release a survey unit when the residual radioactivity

is below the release criteria. This occurs when the Null Hypothesis is accepted when it is not true. The probability of making this error is designated as "β."

Appendix E of NUREG-1727 recommends using a Type I error probability (α) of 0.05 and states that any value for the Type II error probability (β) is acceptable. Following the guidance, α will be set at 0.05. A β of 0.05 will initially be selected based on site-specific considerations. The β may be modified, as necessary, after weighing the resulting change in the number of required survey measurements against the risk of unnecessarily investigating and/or remediating survey units that are truly below the release criteria.

3.5.2.3 Relative Shift

The relative shift (Δ/σ) is a calculated value. Delta (Δ) is equal to the DCGL minus the Lower Boundary of the Gray Region (LBGR). The sigma used for the relative shift calculation may be recalculated based on the most current data obtained from post-remediation or post-demolition surveys; or from background reference areas, as appropriate. The LBGR may be adjusted to obtain an optimal value for the relative shift, normally between 1.0 and 3.0. Administratively, the relative shift will have a maximum value of 3.

3.5.2.4 Lower Bound of the Gray Region

The Lower Bound of the Gray Region (LBGR) is the point at which the Type II (β) error applies. The default value of the LBGR is initially set at the mean of the post-remediation survey results, if available, or at 0.5 times the DCGL, whichever is higher. If the relative shift is greater than 3.0, then the number of data points, N, listed for the relative shift values of 3.0 from Table 5.5 or Table 5.3 in the MARSSIM, will normally be used as the minimum sample size.

3.5.2.5 Sigma

Sigma values (the estimate of the standard deviation of the measured values in a survey unit, and/or reference area) will be initially calculated from characterization samples/survey data with activity at or below the approved DCGL since this value is what is expected to remain after necessary remediation has been completed. These sigma values can be used in FSS design or more current post-remediation sigma values can be used.

3.5.2.6 Sign Test Sample Size

The number of data points to be collected is calculated using VSP's Sampling Goals menu for fixed thresholds. This plan describes the parameters to be used to calculate the sample size for each survey unit.

3.5.3 Background Reference Areas

As stated in section 3.5.2.1, material specific background subtraction will not be used for this survey;

however, instrument background will be subtracted from measurements.

3.5.4 Reference Coordinate System

A reference coordinate system is used as an aid in the identification of survey locations within a survey unit. Visual Sample Plan© (VSP) or other such software is used to develop the actual survey locations to the nearest 0.1-m. The reference coordinate system is a Cartesian grid system (X-Y plot) of the surfaces to be surveyed that are within the survey unit. Survey areas within buildings will use a grid system based on the defined 'site north' beginning with the projected southwest corner of the interior space. This corner is known as the 'Point of Origin'. The grid patterns use a standard Cartesian grid system in the format of X-Y for each floor and wall surface. For floor surfaces, X represents the east-west coordinates and Y represents the north-south coordinates. The Point of Origin for floor surfaces is the southwest corner of the room. For walls, X represents the left-right coordinates and Y represents the up-down coordinates on the wall. The point of Origin for walls is the lower left hand corner.

Ceiling surfaces utilize a superimposition of the floor grid system to readily identify survey locations. For example, Grid Location A, 2 on the ceiling is directly above Grid Location A, 2 on the floor.

3.5.5 Measurement Location

The development of measurement locations within a survey unit will be accomplished using the assistance of the most recent version of the VSP software, or another similar software package. VSP has been developed by Pacific Northwest National Laboratory to aid in the development, generation, and evaluation of sampling locations to meet various regulatory guidance documents and is regularly used by the NRC's independent Verification Contractor ORISE. One key aspect of VSP is the ability to import standard AutoCAD© drawings (DXF file extensions) or ESRI ArcView ShapeFiles (SHP file extensions) and output sample locations using the units within the drawing. The preferred format for this project will be AutoCAD© DXF files.

Measurement locations within the survey unit will be clearly identified and documented for purposes of reproducibility. Actual measurement locations will be identified by tags, labels, paint marks, or photographic record. An identification code and will match a survey location to a particular survey unit.

Sample points for Class 1 and Class 2 survey units will be based on a systematic pattern or grid throughout the survey unit by randomly selecting a start point coordinate. VSP uses a random number generator to determine the start point of the square grid pattern. Random measurement locations are used for Class 3 survey units. These sample location coordinates will be determined using VSP.

Measurement locations selected using either a random selection process or a random-start systematic pattern that do not fall within the survey unit or that cannot be surveyed due to site conditions will be

relocated to the nearest location that can be surveyed within the survey unit. The justification for the movement of a measurement location shall be described in the FSS Report.

3.6 Types and Methods of Surveys

Survey measurements and sample collection are performed by personnel trained and qualified in accordance with the applicable procedure. The techniques for performing survey measurements or collecting samples are specified in approved procedures. FSS measurements include surface scans, direct surface measurements, removable contamination measurements, and gamma spectroscopy of volumetric material samples. In-situ gamma spectroscopy or other methods not specifically described may also be used for FSS. If required, a technical basis document will be created and approved by the University. Upon the document's acceptance, UA will give the USNRC 30 days notice to review the technical basis document prior to implementation.

On-site and off-site lab facilities are used for gamma spectroscopy, liquid scintillation and gas proportional counting in accordance with applicable procedures. Regardless which facilities are used, analytical methods will have a minimum detectable concentration (MDC) of 50% the applicable DCGL value.

3.6.1 Scan Surveys

Scanning is performed in order to locate small areas of residual activity above the investigation level. Under certain conditions, in-situ gamma spectroscopy may be a reasonable alternate scan method. If in-situ gamma spectroscopy is used, a technical basis document will be developed demonstrating its suitability for final survey measurements and USNRC will be notified 30 days prior to its first use.

3.6.1.1 Two-Stage Scanning Method

The two-stage scanning method is one where a surveyor begins the scan at a pre-determined speed, e.g. 10-cm per second, until they detect an elevated count rate. At such time, they return to a location immediately before the elevated detection and repeat the scan at a slower rate to determine the maximum count rate in the area. When the count rate has returned to expected levels, the scan speed is returned to normal. This method relies on the ability of the surveyor to reliably detect elevated count rates. The 'Surveyor Efficiency' (ρ) is detailed in NUREG-1507. A variable accounting for this efficiency (a value from 0.5 to 0.75) is included in the formulae used in the MARSSIM. Lower values for ρ increase the MDC for scanning surveys (MDC_{scan}) indicating a smaller probability of detecting elevated count rates. The MDC_{scan} equations used throughout this document use a value for ρ of 0.5. Additional evaluations may be made on the effectiveness of eliminating ρ from the MDC_{scan} equation and utilizing the alarm

functions of the selected instrumentation. All scans performed in support of the FSS use the two-stage scan method to assure residual radioactive material is sufficiently quantified.

3.6.1.2 Beta-gamma Surface Scans

Surface scans for beta-gamma activity on structures and selected systems that are not potentially volumetrically contaminated will be performed at a scan rate capable of meeting a pre-determined MDC_{scan} applicable to the survey unit classification. The typical scan rate for beta-gamma activity is 1 probe width per second. Surface scans should have a probe to surface distance as close as practical, not to exceed 0.5-cm (~ 0.25 inch). Situations where the maximum detector to surface distance cannot be met may require an alternate scan method, a detector to surface distance correction factor, or justification for not completing the scan. For Class 1 areas, the MDC_{scan} will be no greater than the Elevated Measurement Comparison (EMC) DCGL ($DCGL_{emc}$) described in Section 3.7.3. Class 2 and Class 3 areas require a more stringent MDC_{scan} of no greater than the DCGL. Minimum scan coverage is detailed in Table 3-7 by classification.

The Investigation Level (IL) for the scan surveys will be equal to the gross minimum detectable count rate (MDCR) as defined in MARSSIM Section 6.7.2.1. If a surveyor observes a count rate greater than the IL, the surveyor will pause over the area or resurvey the area as described in the Two-Stage Scanning Method.

3.6.1.3 Gamma Surface Scans

Gamma surface scans will be performed on exposed concrete in the lower section of the reactor pit. Special considerations will be used during the scan because the Post-Shutdown Characterization Report indicates an average Potassium-40 (K-40) concentration of 22 pCi/g. The K-40 concentration will make it difficult to distinguish between the natural K-40 and reactor-related Co-60/Eu-152.

The scans will be performed using a Ludlum Model 2221 Single Channel Analyzer (SCA) with a lead collimated Ludlum Model 44-10 sodium iodide (NaI) 2"x2" detector calibrated to Co-60. The SCA will be set for the average Co-60 and the Eu-152 gamma energies in order to block out as much background radiation as possible. Co-60 has two gammas per decay, one at 1173 keV and the other at 1332 keV, with the average at 1253 keV. Eu-152 has a main gamma photon energy of 1408 keV. The SCA window should be set to encompass the two Co-60 gamma energies and the Eu-152 gamma plus 25 keV on each side of the peaks. Therefore the window would have a width of 235 keV (1149 keV to 1408 keV). FSS personnel should refer to the M2221 manual to ensure the proper setting of the SCA window.

Background measurements in a geometry equivalent to the reactor pit are not feasible for this project. Therefore, the only background subtraction will be the background reading from the daily source check

log. Scanning methodology will be detailed in the FSS Report to demonstrate the scans were capable of detecting residual activity below release criteria.

3.6.2 Direct Measurements

Direct measurements are performed to detect surface activity levels for measureable beta-emitting isotopes of concern. Direct measurements are conducted by placing the detector on or very near the surface to be measured and acquiring data over a pre-determined count time. A count time of one minute is typically used for surface measurements and generally provides detection levels well below the DCGL. (The count time may be varied provided the required detection level is achieved). The MDC for a static count may be calculated using the formula contained in Section 4.4.2.

Direct measurements will be collected at the location generated in section 3.4.4 and at locations where beta-gamma surface scans exceed the investigation level(s) described in Section 3.7.

Direct measurements will not be made for hard to detect isotopes with low average beta energy such as H-3 and Ni-63 and those isotopes that decay by electron capture (see Table 3-3).

3.6.3 Exposure Rate

Gamma exposure rate measurements will be collected for informational purposes only. They will be collected approximately one meter from the surface of each sample location and in the general areas of the UA NRL using a Ludlum Model 19 Micro-R meter. The background exposure rate will not be subtracted from these measurements.

3.6.4 Removable Activity

Removable contamination surveys will be collected to assess the removable activity fraction for selected structural and system surfaces. The surveys are required to demonstrate that the screening criteria, which assume only 10% removable contamination, are acceptable for use. Two wipe samples will be collected at each direct measurement location. When possible, a wipe sample will comprise 100-cm^2 of surface area. When a 100-cm^2 area is not obtainable, the wiped area will be documented and the analyzed result adjusted accordingly.

For each pair of wipe samples, one sample will be analyzed on-site utilizing a wipe sample counter. The second wipe will be analyzed in a LSC. Investigation levels will be 50% of the applicable removable contamination DCGL. The origin of removable activity in excess of this investigation level should be determined and the area should receive additional remediation and resurvey.

3.6.5 Volumetric Samples

Volumetric sampling of media will be implemented in the lower portion of the reactor pit where the

reactor tank has been demolished. Volumetric samples will be analyzed by gamma spectroscopy for gamma emitting radionuclides and liquid scintillation for tritium and other hard to detect isotopes.

The results will be evaluated by use of the unity rule and will require the use of a surrogate calculation to account for the radionuclides in the mixture that are not identified by gamma spectroscopy. This will be accomplished using the nuclide mixture established in the reactor activation analysis (ref 8).

Samples of approximately 1,500 grams are collected from the surface layer of the exposed concrete using procedure UA-MCP-RC-06, *Sample Collection Procedure*. Because the activation of the concrete is dependent on the horizontal distance from the reactor core, samples should not be greater than two (2) inches.

Sample preparation will be performed by the off-site laboratory. Separate containers will be used for each sample and each sample will be tracked through the analysis process using a chain-of-custody record. Samples will be split as directed in Section 5.3.

3.7 Investigation Levels and Elevated Areas Test

During survey unit measurements, levels of radioactivity may be identified by an increase in count rate or an elevated sample result which warrants investigation. Elevated measurements may result from discrete particles, a distributed source, or a change in background activity. In any case, investigative actions should be implemented.

Depending on the investigation results, the survey unit may require:

- No action
- Remediation
- Reclassification and resurvey

3.7.1 Investigation Levels

Table 5.8 in the MARSSIM provides guidance on investigation levels for scan surveys. In addition to investigation levels for scan surveys, direct measurement survey investigation levels may be used. These additional investigation levels include a conservative value for Class 3 survey units and are provided in Table 3-8.

Table 3-8: Measurement Result Investigation Levels

Classification	Investigation Level
Class 1	Result >$DCGL_{emc}$
Class 2	Result >DCGL
Class 3	Result >50% DCGL

3.7.2 Investigation Process

Technicians respond to all audibly detectable elevated count rates while surveying. Upon observing a count rate above the IL (MDCR), the technician stops and resurveys the suspect area to verify the count rate elevation and determine the areal extents of the elevated count rate. Technicians are cautioned, in training, about the importance of the elevated count rate and the verification survey. They are given specific direction regarding the extent and scan speed of the verification survey. If the elevated count rate is verified, the technician marks the area. Each marked area will receive an additional documented survey which requires a re-scan of the area and one or more direct measurements and removable contamination wipes. Results of each investigation are discussed and reported in the FSS Report.

The size and average activity level in the elevated area will be defined to determine compliance with the area factors. If any location in a Class 2 area exceeds the DCGL, scanning coverage is increased in order to determine the extent and level of the elevated reading(s). If any location in a Class 2 area exceeds the $DCGL_{emc}$, the area will be reclassified as a Class 1 area. If the elevated reading occurs in a Class 3 area, the scanning coverage is increased and the area may be reclassified, if necessary.

Investigations should consider:

- The assumptions made in the survey unit classification
- The most likely or known cause of the contamination
- The possibility that other areas within the survey unit may have elevated areas of activity that may have gone undetected

Depending on the results of the investigation, a portion of the survey unit may be reclassified if there is sufficient justification. The results of the investigation process are documented in the survey area Release Record. See Section 3.3.4 for additional discussion regarding potential reclassification of the survey unit.

3.7.3 Elevated Measurement Comparison DCGL

The $DCGL_{emc}$ is not expected to be used during this FSS based upon site characterization data. Should scan measurements of these soils exceed the investigation levels, use of the elevated measurement comparison $DCGL_{emc}$will be considered. The EMC DCGL ($DCGL_{emc}$) provides assurance that unusually large measurements receive the proper attention and that any area having the potential for significant dose contribution is identified and not averaged out over a large area. As stated in the MARSSIM, the EMC is intended to flag potential failures in the remediation process and should not be considered the primary means to identify whether or not a survey unit meets the release criterion.

Locations identified by scan with levels of residual radioactivity which exceed the $DCGL_{emc}$ or static measurements with levels of residual radioactivity which exceed the $DCGL_{emc}$ are subject to additional surveys to determine compliance with the EMC. The size of the area containing the elevated residual

radioactivity and the average level of residual activity within the survey unit are determined. The initial $DCGL_{emc}$ is established during the survey design and is calculated as follows:

$DCGL_{emc}$ determination: $$DCGL_{emc} = AF * DCGL$$

Where:

AF = Area Factor corresponding to the size of the elevated area
DCGL = Derived Concentration Guideline Limit

The area factor is a multiple of the DCGL that is permitted for the area of elevated residual radioactivity without remediation. The area factor is related to the size of the area over which the elevated activity is distributed. That area is generally bordered by levels of residual radioactivity below the DCGL and is determined by the investigation process.

The area assumed for the NRC screening values is unlimited based on the setting for the DandD code. To calculated the $DCGL_{emc}$, the DandD code was run for the radionuclides at the default values except the area of contamination was changed to 1 square meter. For small areas of radioactivity, the DCGL values calculated from the modified DandD run (listed in Table 3-9 below) were used to calculate the area factors and then the most conservative factor (i.e., 5 for tritium) was selected to be used for all radionuclides.

Table 3-9: $DCGL_{emc}$ Values

Radionuclide	Default DCGL (pCi/g) Unlimited Area	$DCGL_{emc}$ (pCi/g) 1 m^2	Area Factor
Cobalt 60	3.80	554	5
Tritium	110	27	7
Europium 152	7.0	62	8

The actual area of elevated activity is determined by investigation surveys and the area factor is adjusted for the actual area of elevated activity. The product of the adjusted area factor and the DCGL determines the actual $DCGL_{emc}$. If the $DCGL_{emc}$ is exceeded, the area is remediated and resurveyed. Should the EMC be used during this FSS, investigation surveys and the EMC evaluation shall be described in detail in the FSS report.

The results of the elevated area investigations in a given survey unit that are below the $DCGL_{emc}$ limit are evaluated using the equation below. If more than one elevated area is identified in a given survey unit,

the unity rule can be used to determine compliance. If the formula result is less than unity, no further elevated area testing is required and the EMC test is satisfied.

Elevated area evaluation: $$\frac{\delta}{DCGL_w} + \frac{C_{avg} - \delta}{(AF)(DCGL_w)} < 1$$

Where:

δ = average residual activity in the survey unit
C_{avg} = average concentration of the elevated area
AF = Area Factor corresponding to the size of the elevated area

When calculating δ for use in this inequality, measurements falling within the elevated area may be excluded provided the overall average in the survey unit is less than the DCGL.

Compliance with the soil $DCGL_{emc}$ is determined using gamma spectroscopy results and a unity rule approach. These general methods are also applied to other materials where sample gamma spectroscopy is used for FSS. The application of the unity rule to the EMC requires that area factors and a corresponding $DCGL_{emc}$ be calculated separately for any gamma emitters of reactor origin identified during FSS.

3.7.4 Remediation and Reclassification

Areas of elevated residual activity above the $DCGL_{emc}$ within any classification are remediated to reduce the residual radioactivity to acceptable levels. Whenever an investigation confirms activity above an action level applicable to the classification, an evaluation of the operational history, design information, and sample results is performed to assure the area was classified properly. The evaluation considers:

- The elevated area location, dimensions, and sample results.
- An explanation of the potential cause and extent of the elevated area in the survey unit.
- The recommended extent of reclassification, if considered appropriate.
- Any other required actions.

Areas that are reclassified as Class 1 will typically be bounded by a Class 2 buffer zone to provide further assurance that the reclassified area completely bounds the elevated area. This process is established to avoid the unwarranted reclassification of an entire survey unit (which can be quite large) while at the same time prescribing an assessment of the extent and reasons for the elevated area.

If an individual scan or static location measurement within a survey unit exceeds the applicable investigation level listed in Table3-8, the survey unit or a portion of it may be reclassified and the survey redesigned and re-performed accordingly. Instrument performance, background fluctuation, surveyor performance, ambient radiological conditions, and other variables should be considered to avoid unnecessary reclassification.

3.7.5 Reclassification and Resurvey

Following an investigation, if a survey unit is reclassified or if remediation activities occur, a resurvey will be performed. If the average value of Class 2 direct survey measurements was less than the DCGL, the MDC_{scan} was sensitive enough to detect the $DCGL_{emc}$ and there were no areas greater than the $DCGL_{emc}$, the survey redesign may be limited to obtaining a 100% scan without having to re-perform the direct measurements. This condition assumes that the sample density meets the requirements for a Class 1 area. If the Class 2 area had contamination greater than the DCGL, but the MDC_{scan} was not sensitive enough to detect the $DCGL_{emc}$, the affected area is reclassified as Class 1 and resurveyed with the sample density determined for the new classification. Class 3 areas are treated in a similar manner, using 50% DCGL as the investigation limit. If a Class 3 area had activity in excess of 50% DCGL, but less than the DCGL and the MDC_{scan} was sensitive enough to detect the $DCGL_{emc}$, then the expansion of scan survey coverage to 100% will be sufficient. If activity is detected above the DCGL, or the MDC_{scan} was not sensitive enough, the area is increased to the appropriate classification as determined by the activity detected and the survey redesigned. Reclassification of a survey area will be detailed in the FSS Report.

3.8 Data Collection and Processing

3.8.1 Sample Handling and Record Keeping

A chain-of-custody (COC) record will accompany each media or materials sample from the collection point through obtaining the final results to ensure the validity of the sample data (exclusive of wipe samples). COC records are controlled and maintained in accordance with applicable procedures.

Each survey unit has an associated document package which covers the design and field implementation of the survey requirements. Survey unit records are considered quality records.

3.8.2 Data Management

Document Control procedures establish requirements for record keeping. Measurement records include, at a minimum, the surveyor's name, the location of the measurement, the instrument used, measurement results, the date and time of the measurement and any surveyor comments.

3.8.3 Data Verification and Validation

The FSS data will be reviewed prior to data assessment to ensure that they are complete, fully documented and technically acceptable. The review criteria for data acceptability will include, at a minimum, the following items:

- The instrumentation MDC for fixed or volumetric measurements was below the DCGL or if not, it was below the $DCGL_{emc}$ for Class 1, below the DCGL for Class 2 and below 50% DCGL for Class 3 survey units.

- The instrument calibration was current and traceable to National Institute for Standards and Testing (NIST) standards.
- The field instruments were source checked with satisfactory results before and after use each day data were collected or data were evaluated if instruments did not pass a source check in accordance with Section 4.3.3.
- The MDCs and assumptions used to develop them were appropriate for the instruments and techniques used to perform the survey.
- The survey methods used were proper for the types of radiation involved and for the media being surveyed.
- The COC was tracked from the point of sample collection to the point of obtaining results.
- The data set is comprised of qualified measurement results collected in accordance with the survey design which accurately reflects the radiological status of the area.
- The data have been properly recorded.

3.9 Data Assessment and Compliance

An assessment is performed on the FSS data to ensure that they are adequate to support the determination to release the survey unit. Simple assessment methods such as comparing the survey data to the DCGL or comparing the mean value to the DCGL are first performed. The statistical tests may then be applied to the final data set and conclusions are made as to whether the survey unit meets the site release criterion.

3.9.1 Data Evaluation

The results of the survey measurements are evaluated to determine whether the survey unit meets the release criteria described in Section 3.3. In some cases, the determination can be made without performing complex, statistical analyses.

An assessment of the measurement results is used to quickly determine whether the survey unit passes or fails the release criterion or whether statistical analyses must be performed.

If all concentrations within the survey unit are less than the DCGL, the unit meets the criterion and no statistical tests are necessary. If the average concentration is greater than the DCGL, the survey unit does not meet the release criterion and additional remediation may be necessary. If the average concentration is less than the DCGL, but one or more individual measurements exceed the DCGL, the sign test should be conducted to determine the disposition of the survey unit.

In addition, survey data are evaluated against the EMC criteria as previously described in Section 3.7.3. The statistical test is based on the null hypothesis (H_o) that the residual radioactivity in the survey unit exceeds the DCGL. There must be sufficient survey data at or below the DCGL to reject the null hypothesis and conclude the survey unit meets the site release criterion for dose. Statistical analyses are performed using a specially designed software package, spreadsheet, or, if necessary, hand calculations.

3.9.1.1 Sign Test

The sign test and sign test unity rule are one-sample statistical tests used for situations in which the radionuclide of concern is not present in background, or is present at acceptably low fractions compared to the DCGL. If present in background, the measurement is assumed to be entirely from NRL activities, i.e., no contribution from natural material specific radionuclides. This option is used when it can be reasonably expected that including the background concentration will not affect the outcome of the sign test. The advantage of using the sign test is that a background reference area is not needed.

The sign test is conducted as follows:

- The survey unit measurement values, X_i, where i = 1, 2, 3, …, N; and N = the number of measurements; are listed.
- X_i is subtracted from the DCGL to obtain the difference D_i = DCGL - X_i, i = 1, 2, 3... N.
- Differences where the value is exactly zero are discarded and N is reduced by the number of such zero measurements.
- The number of positive differences is counted. The result is the test statistic S+. Note that a positive difference corresponds to a measurement below the DCGL and contributes evidence that the survey unit meets the site release criterion.
- The value of S+ is compared to the critical value given in Table I.3 of the MARSSIM. The table contains critical values for given values of N and α. The value of α is set at 0.05 during survey design. If S+ is greater than the critical value given in the table, the survey unit meets the site release criterion. If S+ is less than or equal to the critical value, the survey unit fails to meet the release criterion.

3.9.1.2 Unity Rule

The radionuclides of concern ratio will vary in the final survey soil samples, and this will be accounted for using the unity rule approach as described in NUREG-1505 Chapter 11. Unity values, also called the sum of fractions (SOF), will be calculated as shown in the following equation.

SOF calculation: $$SOF = \frac{C_1}{DCGL_1} + \frac{C_2}{DCGL_2} + \ldots + \frac{C_n}{DCGL_n}$$

Where:

C_x = radionuclide concentration

The SOF calculation results are used to demonstrate compliance by defining the DCGL as 1.0 and using the decision criteria listed in Section 3.9.1.

The unity rule method described above will be applied as necessary when multiple radionuclides or emission types are being evaluated as opposed to a single radionuclide, a surrogate nuclide that incorporates a nuclide fraction, gross alpha measurements, or gross beta-gamma measurements.

3.10 Statistical Conclusions

The results of the statistical tests, including application of the EMC, allow one of two conclusions to be made. The first conclusion is that the survey unit meets the site release dose criterion. The data provide statistically significant evidence that the level of residual radioactivity in the survey unit does not exceed the release criterion. The decision to release the survey unit is made with sufficient confidence and without further analysis.

The second possible conclusion is that the survey unit fails to meet the release criterion. The data are not conclusive in showing that the residual radioactivity is less than the release criterion. In this case, the data are analyzed further to determine the reason for the failure.

Possible reasons are:

- The average residual radioactivity exceeds the DCGL.
- The test did not have sufficient power to reject the null hypothesis (i.e., the result is due to random statistical fluctuation).

The power of the statistical test is a function of the number of collected measurements and the standard deviation of the measurement data. The power is determined from $1-\beta$ where β is the value for Type II errors. A retrospective power analysis may be performed using the methods described in Appendices I.9 and I.10 of the MARSSIM. If the power of the test is insufficient due to the number of measurements, additional samples may be collected and the data re-evaluated. Additional measurements increase the probability of passing if the survey unit actually meets the release criterion. If failure was due to the presence of residual radioactivity in excess of the release criterion, the survey unit must be remediated and resurveyed.

3.10.1 Compliance

The FSS is designed to demonstrate that licensed radioactive materials have been removed from the UA facilities and property to the extent that residual levels of NRL radioactive materials are below the radiological release criteria defined in Section 3.2.

If the measurement results pass the requirements of Section 3.10, and the elevated areas evaluated per Section 3.7.3 pass the EMC, then the survey unit is determined to meet the criteria for license termination.

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3.11 Report Format

Survey results are documented in the FSS Report. Survey results will be described in written reports to the USNRC.

The FSS report provides a summary of the survey results and the overall conclusions which demonstrate that the UA facility and site meet the radiological criteria for release. Information such as the number and type of measurements, basic statistical quantities, and statistical analysis results are included in the report. The level of detail is sufficient to clearly describe the FSS program and to certify the results. The basic outline of the final reports will be:

1.0 Overview of the Results
2.0 Discussion of Changes to FSS
3.0 Final Status Survey Methodology

- Survey unit sample size
- Justification for sample size
- Survey Instrument MDCs

4.0 Final Status Survey Results

- Number of measurements taken
- Survey maps
- Sample concentrations
- Statistical evaluations, including power curves
- Judgmental and miscellaneous data sets
- Elevated Measurement Comparisons (if used)

5.0 Anomalous Data
6.0 Conclusion for each survey unit

4.0 SURVEY INSTRUMENTATION

Radiation detection and measurement instrumentation for the FSS are selected to provide both reliable operation and adequate detection sensitivity of the radionuclides of concern as identified during the characterization evolution. Radiological survey instruments are made up of a meter and a detector. Collectively, this combination is referred to as an instrument. Pairings of a meter and a detector are controlled by calibration procedures. Detector selection is based on detection sensitivity, operating characteristics and expected performance in the field. The instrumentation, to the extent practicable, is capable of data logging operations.

Commercially available portable and laboratory instruments and detectors are typically used to perform the five basic survey measurements:

1. Surface scanning
2. Direct surface contamination measurements
3. Removable surface contamination measurements
4. Removable surface contamination measurements – hard to detect isotopes
5. Spectroscopy of soil and other bulk materials, such as concrete

This section discusses common radiological survey instrument selection, their capabilities, general operation, and calibration. The purpose of this section is to:

- Present a selection of instruments that may be used in support of the MARSSIM activities at the NRL
- Provide methods for evaluation of the suitability of various instruments for their intended roles
- Discuss statistical formulae demonstrating detection efficiencies, minimum detectable concentrations, and expected surface efficiencies
- Establish general calibration requirements
- Set Quality Assurance/Quality Control (QA/QC) requirements for the instrument program

4.1 Units for Activity Concentration

Measurement data for direct and surface scan measurements are recorded as counts per unit time per detector area. These are normally converted into standardized units for evaluation purposes which eliminate instrument specific parameters. Common standardized units within the United States are disintegrations per minute per 100-cm^2 (DPM/100cm^2) which will be used in this document.

Volumetric analyses are typically reported in units of radioactivity per unit mass. PicoCuries per gram is the most common US standard unit for volumetric measurements for decommissioning activities and is used in this document.

4.2 Selection

Radiation instruments and detectors are selected based on the type and quantity of radiation to be measured. The instruments used for direct measurements are capable of detecting the radiation of concern to an MDC of less than 50% of the Co-60 DCGL of 7100 dpm/100cm^2. The use of 50% of the DCGL is an administrative limit only. Any value below the DCGL is acceptable in impacted survey units. MDCs of less than 50% of the DCGL allow detection of residual activity in Class 3 survey units at an investigation level of 0.5 times the DCGL. Instruments used for scan measurements in Class 1 areas are required to be capable of detecting radioactive material at less than or equal to the $DCGL_{emc}$.

Specific instrument selection and performance capabilities (e.g. detection efficiency, MDCs, and data logging) are discussed in Section 4.3. UA will generally follow the instrument manufacturer's recommendations and/or supporting basis documents for considerations such as temperature dependency and other operational parameters.

As the project proceeds, other measurement instruments or technologies, such as in-situ gamma spectroscopy or continuous data collection scan devices, may be found to be more efficient than the survey instruments currently under consideration. The acceptability of such an instrument or technology for use in the FSS program would be justified in a technical basis document. The technical basis document would include, among other things, the following:

- A description of the conditions under which the method would be used.
- A description of the measurement method, instrumentation, and criteria.
- Justification that the technique would provide equivalent scan coverage for the given survey unit classification and that the MDC_{scan} is adequate when compared to the $DCGL_{emc}$.

A demonstration that the method provides data that has a Type 1 error (falsely concluding that the survey unit is acceptable) equivalent to 5% or less and provides sufficient confidence that the $DCGL_{emc}$ criterion is satisfied.

The list of instrumentation currently being considered for use in support of the FSS Plan is provided in

Table 4-1. Equivalent instrumentation may be selected as necessary. All instrumentation used for the FSS will be described in the FSS report.

Table 4-1: FSS Plan Instrumentation

Measurement Type	Detector Type	Detector Area & Density	Manufacturer and Model	Output Units
Surface alpha/beta-gamma (Scan and direct)	Gas flow proportional	126-cm^2 0.8 mg/cm^2	Ludlum Model 43-68	CPM
Surface alpha/beta-gamma (Scan and direct)	Large area gas flow proportional	584- or 821-cm^2 0.8 mg/cm^2	Ludlum Model 43-37 or 43-37-1	CPM
Gamma exposure rate	NaI	1"x1"	Ludlum Model 19	μR/hr
Gamma scan – reactor pit concrete	NaI	2"x2"	Ludlum Model 44-10	CPM
Surface alpha/beta-gamma (Removable)	Alpha/Beta phoswich	20-cm^2 0.8 mg/cm^2	Ludlum Model 2929 or 3030E	CPM
Gamma spectroscopy	Provided by independent radioanalytical laboratory			
Liquid beta scintillation	Provided by University or independent radioanalytical laboratory			

4.2.1 Instrumentation Descriptions

This section will present several instrument and detector brands and models currently being evaluated for use as FSS data collection devices. An instrument is a combination of a meter and a detector. The meter houses the power supply and electronics which record the count data. Meters may have specialized abilities, such as data logging, multiple detectors, digital display, or have multiple modes such as scaler or ratemeter.

4.2.1.1 Meters

Ludlum Model 2360

The Ludlum Model 2360 (M2360) is a digital/analog display instrument capable of scaler and ratemeter/ scan operations with one or two detection channels (dual discriminator). The M2360 has data logging capabilities and may be calibrated with one detector type at a time without adjusting operating parameters (voltage, threshold). Common applications for the M2360 are to match it with a Ludlum Model 43-44 air proportional or Ludlum Model 43-68 gas proportional detector and calibrate it for alpha and/or beta emitters. In dual discriminator mode, it has the ability to distinguish between alpha and beta on separate channels. The drawback to dual discriminator mode is lower detection efficiencies when compared to single channel calibration.

Ludlum Model 2221

The Ludlum Model 2221 (M2221) is a single-channel scaler/ratemeter with digital and analog display. The M2221 may be fitted with a GPS connection so that textual output of the count data may be sent to an external GPS unit for synchronization to location data. This meter may also be used as a single-channel analyzer with the proper setup of operating parameters (i.e., voltage, threshold, and window) which allows it to be used as a very capable screening tool for known isotopes. Common detector pairings are with a L43-68 for alpha or beta detection or a Ludlum Model 44-10 sodium iodide scintillator for gross gamma capabilities.

Ludlum Model 2241-2

The Ludlum Model 2241-2 (M2241) is digital display data logging instrument with the ability to store and use two detector calibrations. The M2241 is more user friendly than the M2350 but maintains much of the flexibility. The ability to use different calibration units allows this meter to be calibrated for surface activity as well as exposure rates depending on the attached detector.

Ludlum Model 19

The Ludlum Model 19 (M19) is a MicroR ratemeter with a 1"x1" NaI crystal. The M19 has an analog display and is capable of detection ranges up to 5,000 μR/hr. The low display range and steady response make this a good instrument for ambient exposure rate surveys. While the detection response may vary by isotope, as a hand-held exposure rate meter it is quite capable. More accurate measurements may be obtained with an instrument such as one of the Reuter-Stokes Pressurized Ion Chamber models, but for portable operational needs, the M19 is sufficient.

4.2.1.2 Detectors

Ludlum Model 43-68

The Ludlum Model 43-68 (M43-68) is a gas proportional (P-10 gas) alpha and beta detector with a Mylar window. The probe active surface area is 126-cm^2. Typical 2π detection efficiency ε_i is ≈20%-30% depending on the meter voltage and other calibration parameters. This detector is most commonly used on hard, relatively smooth surfaces, such as concrete. Care must be taken in areas where sharp edges are present to avoid puncturing the Mylar window or breaking an anode wire. The detector is best used with a constant gas flow, but may be used without for a limited time period if there is a sufficiently tight seal.

Ludlum Model 43-89

The Ludlum 43-89 (M43-89) is a 125-cm^2 (active) Zinc Sulfide (ZnS) alpha and beta scintillation detector. Typical 2π efficiencies are 6-33% depending on the isotope and other calibration parameters. As

is common with all ZnS detectors, the M43-89 is sensitive to light leaks in the housing or through the Mylar window so care must be taken when it is used near sharp edges.

Ludlum Model 44-9

The Ludlum 44-9 (M44-9) Geiger-Mueller tube detector is useful for alpha, beta, and gamma detection for small, discreet areas. It is ideally suited for very small areas where relatively high levels of residual contamination are possible. While it has fairly good typical detection efficiencies from 5-32% (4π), the small detection area severely limits its suitability for FSS.

Ludlum Model 44-10

The Ludlum Model 44-10 (M44-10) detector is a gamma scintillator with a 2" x 2" Sodium Iodide (NaI) crystal. It is commonly used for open area gross gamma surveys. Typical count response is approximately 900 counts per minute per microR per hour for Cs^{137} energy (662 keV).

4.3 Quality Control

4.3.1 Calibration and Maintenance

Instrumentation used for FSS is calibrated and maintained in accordance with each instrument's operating manual. Instruments will be calibrated at least annually, unless the manufacturer recommends a shorter calibration cycle. Radioactive sources used for calibration will be traceable to the National Institute of Standards and Technology (NIST).

4.3.2 Sources

Instruments and detectors are calibrated for the radiation types and energies of interest at the site. The calibration sources for beta survey instruments typically are Tc^{99}, Cs^{137}, or Co^{60}. Gamma scintillation detectors are generally calibrated using Cs^{137}. Calibration sources, other than those listed, may be used provided they demonstrate appropriate detection efficiency for the radionuclides of interest. In all cases, the surface efficiency as determined appropriate for the weighted mean energy of the radionuclides of concern will be utilized regardless of the energy of the calibration source.

4.3.3 Source and Response Checks

Instrument response checks are conducted to assure proper instrument response and operation using ENERCON procedure UA-MCP-RC-04, *Radiological Surveys Procedure*. An acceptable response for field instrumentation is an instrument reading within ±10% of the check source value established during or immediately after calibration. Laboratory instrumentation standards are within ±3 sigma as documented on a control chart. Response checks are performed daily before instrument use and again at the end of use. Check sources are appropriate for the type of radiation as that being measured in the field

and are, to the extent practical, held in fixed geometry jigs for reproducibility.

4.4 Instrument Statistics

Radiological instrumentation for the FSS have been selected to provide both reliable operation and adequate detection sensitivity of the final list of the radionuclides of concern as identified for the DCGL.

As the project proceeds, other measurement instruments or technologies, such as in-situ gamma spectroscopy or continuous data collection scan devices, may be found to be more efficient than the survey instruments currently under consideration. The acceptability of such an instrument or technology for use in the final survey program may be justified in a technical basis document. The technical basis document would include, among other things, the following:

- A description of the conditions under which the method or equipment would be used
- A description of the measurement method, instrumentation, and criteria
- Justification that the technique would provide equivalent scan coverage for the given survey unit classification and that the MDC_{scan} is adequate when compared to the $DCGL_{emc}$
- A demonstration that the method provides data that has a Type 1 error (falsely concluding that the survey unit is acceptable) equivalent to 5% or less and provides sufficient confidence that the $DCGL_{emc}$ criterion is satisfied

4.4.1 Instrument Efficiency

4.4.1.1 Total Instrument Efficiency

The total instrument efficiency (ε_t) is a calculation of the percentage of activity present in or on a surface that an instrument detects. The ε_t is a product of two components, the instrument efficiency (ε_i) and the surface efficiency (ε_s) as shown in Equation 4.1 as shown below:

Equation 4.1 – Total Detection Efficiency

$$\varepsilon_t = (\varepsilon_i) * (\varepsilon_s)$$

4.4.1.2 Instrument Efficiency

The instrument efficiency (ε_i) is a measurement of the surface emissions that interact with the detector elements with an instrument compared to the activity of the source. To determine this, the instrument detector is exposed to a source of a known emission rate for a specified time period and the number of interactions is recorded. The efficiency percent is then calculated by Equation 4.2 as shown below:

Equation 4.2 – Instrument Efficiency

$$\varepsilon_i = \frac{(C_s - C_b)}{S}$$

Where:

C_s = Measured interaction count per one minute
C_b = Measured background interaction count per one minute
S = known source value in DPM (2π)

The known source value should be that of the hemispherical area (2π) exposed to the detector as opposed to the total emission of the sphere around the source (4π). The surface efficiency (ε_s), discussed below, accounts for the remaining half of the emission sphere.

4.4.1.3 Surface Efficiency

Surface efficiency (ε_s) is an estimation of the affect the media surface has on the interaction of residual radioactive material with a detector and is a function of the surface condition, i.e. smoothness, and the relative emission energy of the radionuclide. The ε_s for potentially contaminated structures and systems will follow recommendations contained in NUREG-1507 for the energies applicable to the radionuclides of concern developed in the characterization study. In general, it is expected that beta-gamma detection instruments will use an ε_s of 0.5 and alpha detection instruments will use an ε_s of 0.25.

The methods for determining efficiency in NUREG-1507 were specifically developed to address situations when the source, in this case concrete, affects radiation emission rate due to self-attenuation, backscatter, and thin coverings.

Media-specific ε_s may be developed as necessary. These new surface efficiencies will subsequently override NUREG-1507 recommendations upon acceptance.

The condition of the surface being measured has an effect on the ε_s as well. For direct surface and scan measurements, the surface area beneath the detector should not have variability of depth greater than 0.5 cm more than the source to detector distance used for the instrument calibration. According to NUREG-1507, instrument efficiency drops considerable when the source to detector distance increases more than

0.5 cm.

4.4.2 Minimum Detectable Concentration

MDC and Minimum Detectable Activity (MDA) are estimations of the lowest level of concentration or activity the subject instrument can detect 95% of the time. For purposes of this document MDC and MDA will be used interchangeably. Factors that directly affect the MDC are the total instrument efficiency (discussed in Section 4.4.1), background rate, and count duration. The MDC is calculated individually for direct (static) measurements, ratemeter count rates, and scans.

4.4.2.1 Minimum Detectable Concentration for Direct Measurements

Direct, or static, measurements are measurements where the detector is placed into a fixed position and the instrument records individual unique counts for a specified period of time. Measurement data for direct surface measurements are recorded as counts per unit time per detector area prior to conversion to standardized units.

For static (direct) surface measurements, with conventional detectors, the MDC will be calculated using Equation 4.3 (Formula 3-10 in NUREG-1507) shown on the next page:

Equation 4.3 – Minimum Detectable Concentration

Direct measurement MDC:

$$MDC = \left[\frac{3 + 3.29\sqrt{(R_b)(T_s)(1 + T_s/T_b)}}{(T_s)(\varepsilon)(P_r)}\right]$$

Where:

MDC = Minimum detectable concentration (DPM/100-cm^2)
R_b = Background count rate (CPM)
T_b = Background count time (minute)
T_s = Sample count time (minute)
ε = Counting system efficiency (decimal)
P_r = Probe Ratio (probe area in cm^2 divided by 100cm^2)

Direct measurements require an MDC less than the DCGL, but an administrative limit of ≤50% DCGL will be used to assure adequate sensitivity for the investigation levels applicable to lower unit classifications (Class 2 and Class 3).

4.4.2.2 Scan Measurement MDC

Scan measurements are measurements taken with the detector in a steady motion over an area larger than the surface area of the detector and at a predetermined surface to detector distance

Scan measurement MDC (MDC$_{scan}$) calculations depend on the emissions of the radionuclides of concern.

Beta-gamma MDC_{scan} is discussed in Section 6.7.2.1 of the MARSSIM. The desired MDC_{scan} for an area is a function of its classification. Because of the lower activity anticipated in lower class areas (Classes 2 and 3), a more stringent MDC_{scan} is recommended, as shown in Table 4–2 below.

Table 4-2: MDC_{scan} Levels

Classification	MDC_{scan} Level
Class 1	$\leq DCGL_{emc}$
Class 2	$\leq$DCGL
Class 3	$\leq$DCGL

4.4.2.2.1 <u>Beta-Gamma Scan MDC</u>

The MDC_{scan} for beta-gamma measurements may be calculated by first determining the Minimum Detectable Count Rate (MDCR). The MDCR is calculated by first defining the minimum detectable net source counts (s_i) using Equation 4.4 (Formula 6–8 from the MARSSIM) as shown below.

<u>Equation 4.4 – Minimum Detectable Source Counts</u>

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

Where:

d' = value taken from Table 6.5 in the MARSSIM for applicable true and false positive rates

b_i = Number of background counts in a given time interval

The MDCR is then calculated using Equation 4.5 (Formula 6-9 in the MARSSIM) as shown below:

<u>Equation 4.5 – Minimum Detectable Count Rate</u>

$$MDCR = s_i * \frac{60}{i}$$

Where:

i = Observed time interval in seconds

Finally, applying the detection efficiency correction results in an MDC_{scan} in standardized units (DPM/100-cm^2) using Equation 4.6 (Formula 6–9 in NUREG-1507) as shown below:

<u>Equation 4.6 – Scan MDC</u>

$$MDC_{scan} = \frac{MDCR}{\sqrt{\rho} * \varepsilon_i * \varepsilon_s * \frac{probearea}{100cm^2}}$$

Where:

ρ = Surveyor efficiency (value from a range between 0.5 and 0.75)
ε_i = Instrument efficiency
ε_s = Surface efficiency

The value for ρ has been developed in Draft NUREG/CR-6364 and NUREG-1507 and is a percentage estimate of the likelihood a surveyor will reliably detect an elevated count rate.

4.4.2.2.2 Alpha Scan Probability of Detection

Alpha emitting radionuclides are not listed as a Radionuclide of Concern in the Release Criteria Section of the DP, Section 2.2.3. Therefore, the probability of detection for alpha scans is not applicable to the FSS of the UA NRL.

5.0 QUALITY ASSURANCE

It is important to maintain the integrity of the data collected during the final status survey.

5.1 Quality Objectives and Measurement Criteria

Type I errors are established at 0.05 unless another value is authorized by the USNRC. Type II errors will be set at 0.05.

5.1.1 Training and Qualification

Personnel performing FSS measurements will be trained and qualified. At a minimum, training will include the following topics:

- Procedures governing handling FSS data such as, but not limited to, document control, records retention, and chain of custody.
- Operating field and laboratory instrumentation used for FSS.
- Performing FSS measurements and collecting samples.

The extent of training and qualification will be commensurate with the education, experience and proficiency of the individual and the scope, complexity and nature of the activity. Training records will be maintained as quality records.

5.1.2 Survey Documentation

Each FSS measurement will be identified by date, instrument, location, type of measurement, and mode of operation. Generation, handling and storage of the original FSS design and data packages will be controlled. The FSS records have been designated as quality documents and will be maintained in accordance with document control procedures.

5.2 Measurement/Data Acquisition

5.2.1 Survey Design

The site will be divided into survey areas. Each survey area will contain one or more survey units. A survey package specifies the type and number of measurements required for a survey unit based on the classification and known characterization data results. Each survey area will have one or more survey packages.

5.2.2 Written Procedures

Sampling and survey tasks must be performed properly and consistently in order to assure the quality of the FSS results. The measurements are performed in accordance with approved, written procedures. Approved procedures describe the methods and techniques used for the FSS measurements. Each procedure written for the purpose of directing FSS data collection or evaluation shall include a section

describing the QA/QC goals and methods specific to that procedure.

5.2.3 Sampling Methods

Samples are collected and placed into new containers using either new tools or tools that have been thoroughly decontaminated and double-rinsed with clean water from two sources, i.e., two separate dip tanks. Surface abrasion of the tools may be necessary to dislodge adhered media from previous samples. This may entail using a stiff-bristled brush in the first dip tank. Tools will be air- or towel-dried prior to reuse.

5.2.4 Chain of Custody

Responsibility for custody of samples from the point of collection through the determination of the final survey results is established by procedure UA-MCP-RC-06, *Sample Collection Procedure.* When custody is transferred, a COC will accompany the sample for tracking purposes. Secure storage is provided for archived samples until such time it is determined to no longer be necessary, i.e. license termination for samples utilized for FSS.

5.3 Volumetric Analyses

For soil samples, Quality Control will consist of requiring the vendor analytical laboratory to be NVLAP accredited. However, as an additional quality measure, randomly selected samples are subject to blank sample, blind duplicate, split, recount, or third party analyses. The acceptance criterion for blank samples is that no plant-derived radionuclides are detected to the required MDA. Some sample media, such as asphalt, will only be subject to third party analyses or a sample recount due to the lack of homogeneity. The criterion for blind duplicates, split, recount, and third party analyses is that the two measurements are within ±20% of each other.

5.4 Instrument Selection, Calibration and Operation

Proper selection and use of instrumentation ensure sensitivities are sufficient to detect radionuclides at specified MDCs. An overview of the required capabilities is presented in Section 4.0. These requirements help assure the validity of the collected survey data. Instrument calibrations are performed with sources traceable to the National Institute for Standards and Testing (NIST) using approved procedures.

5.4.1 Data Management

Survey data control from the time of collection through evaluation is specified by procedure. All survey and data analysis records pertaining to the final radiological status of UA NRL are considered quality records and are maintained in accordance with applicable document control procedures.

5.5 Data Validation

Survey data are reviewed prior to evaluation or analysis for completeness and for the presence of outliers. Comparisons to investigation levels are made and measurements exceeding the investigation levels are evaluated.

5.6 Confirmatory Measurements

It is anticipated that the USNRC and other regulatory agencies will choose to conduct confirmatory measurements in accordance with applicable laws and regulations. The USNRC may take confirmatory measurements to make a determination in accordance with 10 CFR 50.82(a)(11) that the final radiation survey and associated documentation demonstrate that the facility and site are suitable for release in accordance with the criteria for decommissioning in 10 CFR Part 20, Subpart E. Confirmatory measurements taken by the USNRC and other regulatory agencies are based on the same DCGLs. Timely and frequent communications with these agencies ensure that they are afforded sufficient opportunity for these confirmatory measurements prior to any difficult to reverse decommissioning actions.

6.0 REFERENCES

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2. 10CFR50.82, *Termination of License*
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5. NUREG-1505, *A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys*, Rev.1, June 1998 draft
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7. University of Arizona Nuclear Reactor Laboratory Decommissioning Plan, May 21, 2009
8. WMG Report 08-125D-RE-122, *University of Arizona Activation Analysis and Component Characterization*, April 2009,