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June 18, 2003 Docket 50-62

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Attention: Mr. Daniel E. Hughes, Project Manager
Operating Reactor Improvements Program

Subject: Transmittal of University of Virginia Reactor Decommissioning Project Groundwater Report and Final Survey Status Addenda: Underground Waste Tank Excavation; Reactor Facility Piping; Pond Sediments; Interior Structure Surfaces; Exterior Soil and Paved Surfaces; Exterior Structure Surfaces; and Special Soils Areas

References: 1. Amendment No. 26 to Amended Facility Operating License No. R-66 for the University of Virginia Research Reactor
2. Docket 50-62
3. Transmittal R. U. Mulder to D. E. Hughes, "Transmittal of the University of Virginia Reactor Decommissioning Project Master Final Status Plan, UVA-FS-002, Rev 0, March 2003" dated April 4, 2003

Dear Mr. Hughes,

The referenced amendment which approves the decommissioning plan for the University of Virginia Research Reactor calls for the licensee to submit a report of their investigation of groundwater conditions. We are pleased to transmit for your information three copies (enclosed) of the site investigative report on groundwater, prepared by the University of Virginia. As described in the referenced transmittal, eight Addenda to the Final Status Survey Plan are being submitted as additional information in support of that plan. We are pleased to transmit for your information three copies (enclosed) of the eight FSS Addenda, prepared for the University of Virginia by CH2M HILL and its subcontractor, Safety and Ecology Corporation. They are: Underground Waste Tank Excavation; Reactor Facility

A020

Piping; Pond Sediments; Interior Structure Surfaces; Exterior Soil and Paved Surfaces; Exterior Structure Surfaces; Special Soils Areas and Ventilation Systems. This completes the transmittal of the Addenda in Support of the Final Status Survey Plan.

If you have any questions regarding this transmittal, please contact me at (434) 982-5446.

Sincerely, I declare under penalty of perjury that the foregoing is true and correct.

Paul E Benneche for R.U. Mulder

Robert Mulder
Reactor Director
University of Virginia

Enclosures:

1. UVAP Final Status Survey Plan Addendum 001: Underground Waste Tank Excavation
2. UVAP Final Status Survey Plan Addendum 002: Reactor Facility Piping
3. UVAP Final Status Survey Plan Addendum 003: Pond Sediments
4. UVAP Final Status Survey Plan Addendum 004: Interior Structure Surfaces
5. UVAP Final Status Survey Plan Addendum 005: Exterior Soil and Paved Surfaces
6. UVAP Final Status Survey Plan Addendum 006: Exterior Structure Surfaces
7. UVAP Final Status Survey Plan Addendum 007: Special Soils Areas
8. UVAP Final Status Survey Plan Addendum 008: Ventilation Systems
9. UVAP Groundwater Characterization Report, University of Virginia Nuclear Reactor, Charlottesville, Virginia June 6, 2003

c: Ralph Allen, Chair Reactor Decommissioning Committee
Stephen Holmes, NRC

MASTER

Information Copy

Final Status Survey Plan

Addendum 001: Underground Waste Tank Excavation

Revision 0

Prepared for
University of Virginia
Reactor Facility Decommissioning Project

Prepared by

CH2MHILL

151 Lafayette Drive, Suite 110
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With assistance from
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2800 Solway Road
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June 2003

Final Status Survey Plan

Addendum 001: Underground Waste Tank Excavation

Revision 0

Prepared for
University of Virginia
Reactor Facility Decommissioning Project

June 2003

Client Approvals:

OEHS — Date: 6/12/2003

Technical Director — Date: 6/12/2003

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Final Status Survey Plan

Addendum 001: Underground Waste Tank Excavation

Revision 0

Prepared for
University of Virginia
Reactor Facility Decommissioning Project

June 2003

Certified Health Physicist — Date: 6/12/03

Project Manager — Date: 6-12-03

CH2MHILL

Contents

FIGURES

1. Introduction

The Master Final Status Survey Plan (UVA-FS-002) identifies the Data Quality Objectives for Final Status Survey (FSS) activities, together with the underlying technical assumptions, approaches, and methodologies for designing, implementing, and evaluating a FSS on each impacted area of the University of Virginia Research Reactor (UVAR) Facility. A separate survey area-specific addendum is prepared for each area or group of areas with common media, contaminants, and other characteristics, prior to beginning FSS; FSS for the specific area or group of areas will then be performed in accordance with that addendum. This addendum (Addendum 001) applies to the area of the underground waste tank excavation.

It should be noted that the field survey activities for the waste tank excavation described in this addendum were implemented during the week of January 20, 2003, to enable timely backfill of the excavation and to coordinate with the NRC for confirmatory scans and sampling.

2. Area Description

Two sets of underground metal tanks, located southeast of the UVAR facility adjacent to the pond, were used for collection/holdup of liquid wastes, which were potentially contaminated with low concentrations of radioactive materials (Figure 2-1). Two of these tanks serviced the hot cell facility and two were used for collection of demineralizer regeneration liquids from the 2-MW UVA Reactor. Both of these tank sets were initially equipped for environmental discharge to the pond, provided the liquid met appropriate release criteria following dilution with adjoining pond water. However, the demineralizer regeneration liquid tanks were later replumbed so that they discharged directly into the pond spillway. Both sets of tanks and associated piping, valves, pumps, etc., have been removed along with their concrete foundations. Small quantities of contaminated soil in the vicinity of the tanks and discharge lines were removed. The resulting excavation is approximately 175 m^2 in area and ranges up to approximately 3 m in depth; including the unexcavated soil edges. The area to be addressed by this survey is approximately 350 m^2 (Figure 2-2).

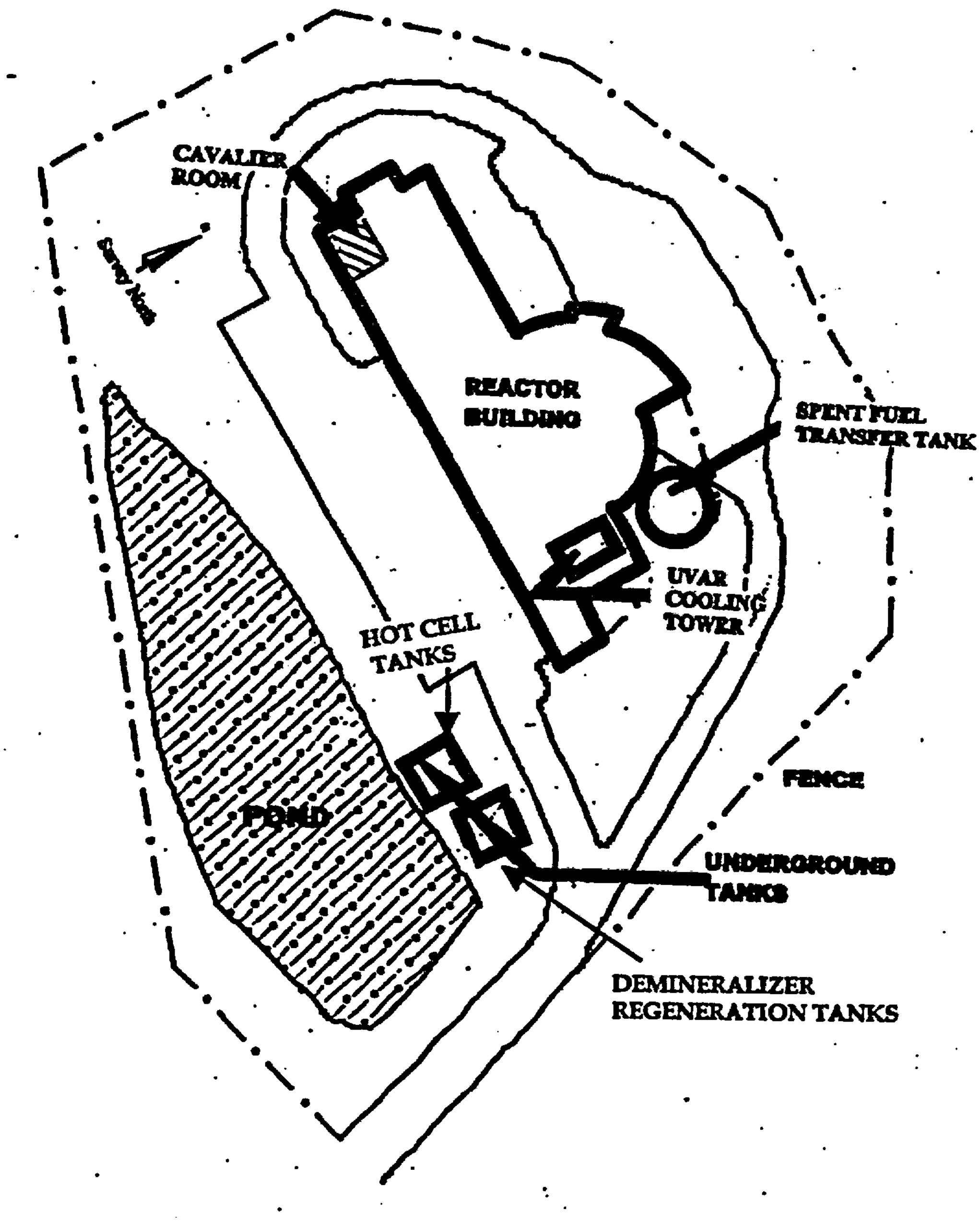

Figure 2-1 University of Virginia Reactor Facility and Environs

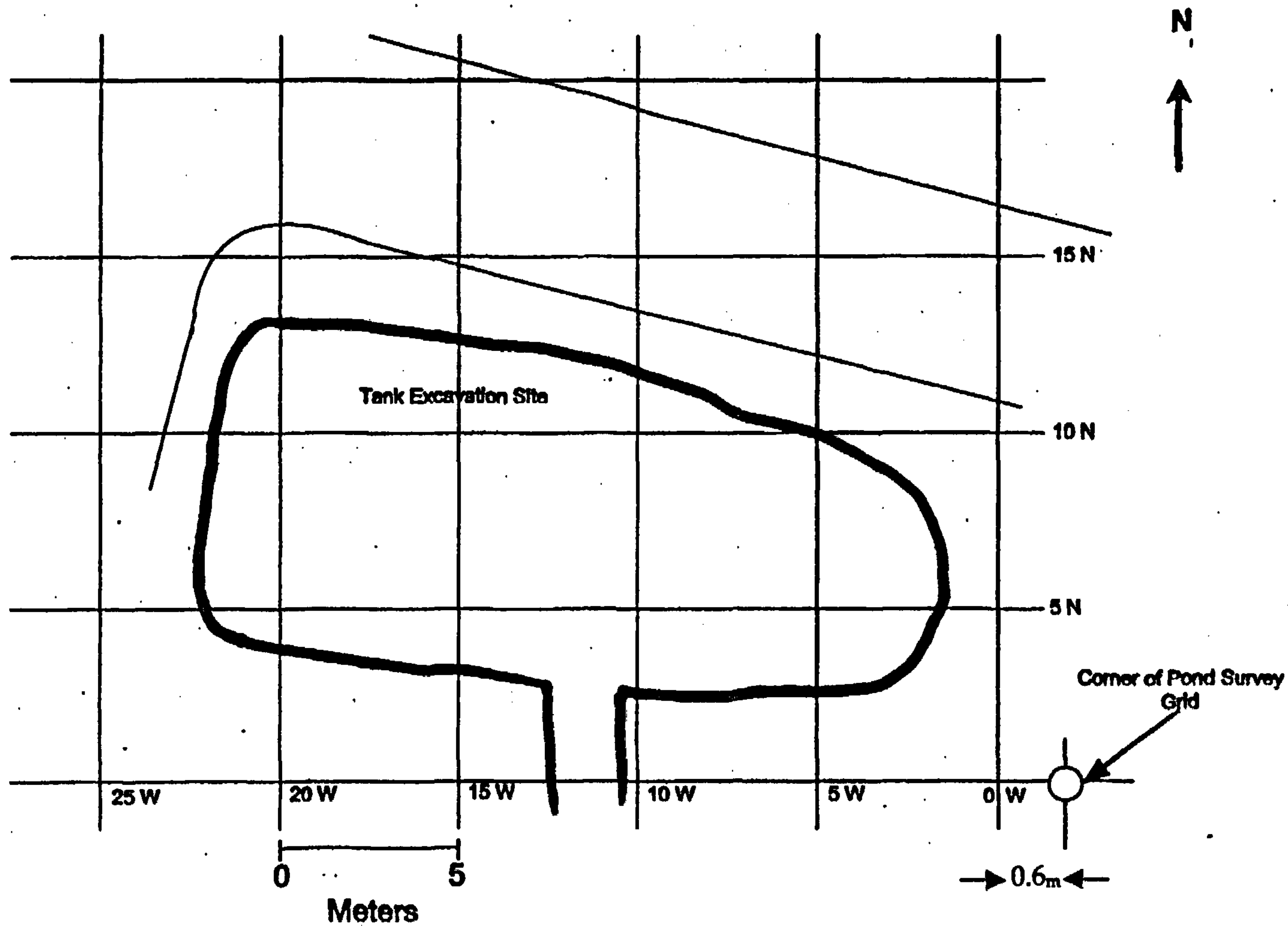

Figure 2-2 Waste Tank Excavation, Indicating the Grid System for Survey.

3. Contaminants of Concern and Guidelines

Contaminated soil was identified at the base of the demineralizer regeneration waste tank blockhouse. During excavation of the tanks samples of the soil were collected at depths down to approximately 3 m and analyzed by gamma spectrometry to determine the radiological nature of the contaminants. These analyses identified only Co-60 and Cs-137 at detectable concentrations. The two samples with detectable levels of these radionuclides contained: 1) 3.00 pCi/g of Co-60 and 4.26 pCi/g of Cs-137 and 2) 1.84 pCi/g of Co-60 and 1.74 pCi/g of Cs-137. During excavation, approximately 0.5 m of additional soil was removed from the surface where these samples were obtained. Hand augered samples from three other locations did not contain detectable concentrations of facility related gamma-emitting radionuclides. Approximately 40 additional samples of excavated (non-impacted) soil were collected and analyzed by gamma spectrometry. No gamma-emitting radioactive contaminants were identified at detectable levels in these samples. Therefore, the maximum reported Co-60 and Cs-137 concentrations in the excavated soils were the MDA's of <0.83 pCi/g and <0.87 pCi/g, respectively. A sample of waste tank sludge was collected and found to contain 255.3 pCi/g of Co-60 and 16.1 pCi/g of Cs-137. Based on these results and the history of reactor operations, the predominant radioactive contaminants in soils from the waste tank excavation are expected to be Co-60 and Cs-137.

NRC default screening criteria will be utilized as release criteria for the soil in the waste tank excavation area. Default screening criteria concentrations for Co-60 and Cs-137 are 3.8 pCi/g and 11.0 pCi/g, respectively. For the final status survey, soil samples from the excavation area will be analyzed for specific potential radionuclide contaminants and results must satisfy the Unity Rule for the sum of ratios of net radionuclide concentrations present to respective screening default guideline levels.

4. Survey Approach

4.1 Survey Reference System

A 5-meter grid was established over the excavation area and extended to unexcavated soil surrounding the excavation. This grid is an extension of the reference grid established for survey of the sediments in the adjacent pond, thus enabling the sampling locations to be related to the federal and/or state planar coordinate system. Figure 2-2 illustrates the reference grid system's relations ship to the excavation.

4.2 Survey Classification

Based on the facility use history and identification of contaminants of license origin in the soils in this area, the survey area is designated Class I for FSS planning and implementation purposes.

4.3 Survey Unit Identification

The area of the waste tank excavation and surrounding soil is approximately 350 m^2; this is within the MARSSIM-recommended area of 1000 m^2 for Class 1 open land survey units. Therefore, the area is a single survey unit.

4.4 Demonstration of Compliance with Release Guidelines

Compliance with decommissioning requirements will be demonstrated by comparing the results of FSS sample analyses with default screening criteria, using the sum of ratios; the sum of ratios must satisfy the Unity Rule. Because the radionuclides identified as potential contaminants are not present in background at concentrations, which are significant fractions of the release guidelines, correction of FSS sample data for background levels will not be required. Statistical testing of results will utilize the Sign Test to reject or accept the null hypothesis that the residual contamination exceeds the release criteria. Decision errors will be 0.05 (Type 1 and Type 2).

4.5 Number of Required Data Points

(Refer to Master FSS Plan, Section 7.8).
DCGL = 1, LBGR = 0.5 DCGL
Δ = DCGL - LBGR = 0.5
σ = 0.3 (based on maximum values for Co-60 and Cs-137 in post-excavation samples)
$\Delta/\sigma = 1.67$
From MARSSIM Table 5.5, N = 17; i.e., 17 samples required from survey unit for Sign test.

4.6 Sampling Pattern

A triangular pattern, based on 17 samples and 350 m^2 areas, was used to determine sampling locations. The distance (L) between samples is:

L = [350/(0.866 x 17)] 0.5 = 4.9 m (rounded to 5.0 m for ease of field implementation)

A random start point for the pattern is based on survey unit dimensions of 10 m N/S and 25 m E/W and random numbers from the MARSSIM random number table of 0.793416 and 0.448970. The resulting start point was 7.9 m N and 11.2 W.

Sampling locations are:

3.6 N, 1.3 W
3.6 N, 3.7 W
3.6 N, 8.7 W
3.6 N, 13.7 W
3.6 N, 18.7 W
3.6 N, 23.7W
7.9 N, 1.2 W
7.9 N, 6.2 W
7.9 N, 11.2 W
7.9 N, 16.2 W
7.9 N, 21.2 W
12.2 N, -1.3 W
12.2 N, 3.7 W
12.2N, 8.7 W
12.2 N, 13.7 W
12.2 N, 18.7 W
12.2N, 23.7 W

Because only 14 of these locations fell onto soil surfaces, an additional line of locations at -0.7 N was added.

-0.7 N, 1.2 W
-0.7 N, 6.2 W
-0.7 N, 11.2 W
-0.7 N, 16.2 W
-0.7 N, 21.2 W

The resulting total number of sampling locations is 19; Figure 4-1 indicates the sampling locations.

4.7 Survey Methods

Gamma walkover surface scans were performed using a 2"X 2" NaI detector (Ludlum Model 44-10) coupled with a Ludlum Model 2221 ratemeter/scaler. The detector was maintained within 5 to 10 cm of the soil surface and moved from side to side in a serpentine

pattern while noting any indication of elevated count rate, which might indicate the presence of radioactive contamination. Results (count rate) were documented on survey area maps. Locations of elevated response were noted for further investigation. Scanning coverage was 100% of the soil surface.

Surface (0 to15 cm) soil samples of at least 500 g were collected at the 19 discrete sampling locations identified above. If a sample could not be obtained from a pre-identified location one was obtained from the nearest soil location available. The survey/sampling record noted this deviation. The licensee and the NRC Inspector witnessed the soil sampling and selected samples for confirmatory purposes. Requested samples were homogenized and split. This process was used because the excavation required accelerated backfilling to maintain slope stability. Samples were assigned unique identification numbers and a chain of custody record and analytical request were prepared.

4.8 Sample Analyses

Samples were screened by on-site gamma spectrometry and then sent to an off-site commercial laboratory for individual gamma spectral analysis. A composite, consisting of an equal amount (about 10 grams) from each survey unit sampling location, was prepared for off-site analysis by gamma spectrometry and for hard-to-detect (10 CFR Part 61) radionuclides. Results of the full analyses of the composite sample will be used to develop fractional contributions from other radionuclides in survey unit samples. These fractional contributions will be used to adjust the results of individual sample gamma analyses for the presence of other contaminants. Gamma Analyses will be surrogates for calculating soil activity concentrations assuming the previously established mix of radionuclides remains proportional to the concentrations of principal gamma emitters.

4.9 Investigation

If surface gamma scans, sample analysis, or statistical data evaluation identify residual contamination exceeding release criteria, the source of the residual contamination will be determined and characterized. Remedial action will be conducted as required, and the FSS activities repeated utilizing a newly determined sampling pattern and random start point.

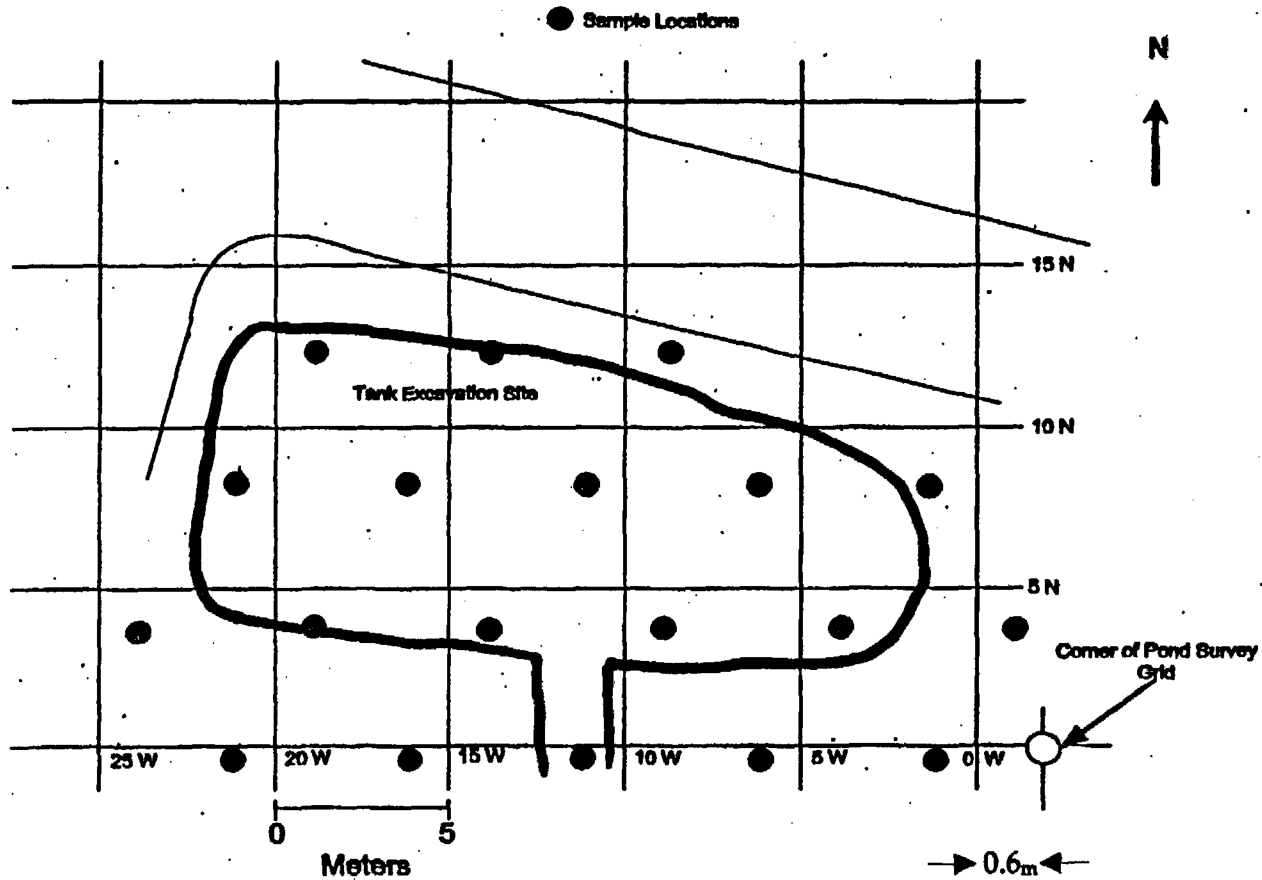

Figure 4-1 Waste Tank Survey Unit, Indicating FSS Soil Sampling Location

5. Data Evaluation

Sum of ratios will be used to determine if activity results meet release criteria for radionuclides that are of facility origin. The Sign Test will be performed and the results compared with the critical value for the appropriate number of samples and decision errors.

MASTER

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Final Status Survey Plan

Addendum 002: Reactor Facility Piping

Revision 0

Prepared for
University of Virginia
Reactor Facility Decommissioning Project

Prepared by

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OEHS — Date: 6/12/2003

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CH2MHILL

Contents

FIGURES

Attachment A

1. Introduction

The Master Final Status Survey Plan (UVA-FS-002) identifies the Data Quality Objectives for Final Status Survey (FSS) activities, together with the underlying technical assumptions, approaches, and methodologies for designing, implementing, and evaluating a FSS on each impacted area of the University of Virginia Research Reactor (UVAR) Facility. A separate survey area-specific addendum is prepared for each area or group of areas with common media, contaminants, and other characteristics, prior to beginning FSS; FSS for the specific area or group of areas will then be performed in accordance with that addendum. This addendum (Addendum 002) applies to the potentially impacted buried and embedded piping, associated with the reactor facility.

2. Description

The bulk of known potentially contaminated piping was removed from the UVAR Facility during remediation activities, but sections of radiologically impacted piping previously associated with the reactor coolant system and various drains from the reactor facility remain. This remaining piping is embedded in concrete or buried beneath concrete or asphalt paving and soil. The piping is generally of small diameter (2 in to 4 in ID); however there are several short sections of larger diameter. Remaining piping includes:

- Heat exchanger lines: Stainless Steel (SS), 6 in ID x 22 ft and 6 in ID x 32 ft.
- Reactor pool drains: SS, 2 in ID x 32 ft and 2 in ID x 36 ft.
- Reactor Room floor drains: Cast Iron (CI), 2 in ID x ~160 ft (multiple sections).
- Ground floor drain to Pond standpipe: CI, 2 in ID x 40 ft and 4 in ID x 140 ft.
- Reactor Demineralizer drain to outside underground collection tanks: CI, 2 in ID x 75 ft.
- Hot Cell drain to outside underground collection tanks: Duriron with PVC repair, 2 in ID x 55 ft.
- Ground floor Bulk Access Facility drains to Pond hillside: CI, 2 in ID x 40 ft and terra cotta, 4 in ID x 80 ft.
- Sanitary sewer from liquid release point to sewer manway: 4 in CI by 40 ft.
- Drain lines from CAVALIER facility to Pond hillside.

In addition, the facility and property are serviced by building and pool footing drains, storm drains and sanitary sewer drains, located beneath the paved area, south of the building. There is no history to suggest these systems may have become contaminated as a result of licensed facility operations.

Figures 2-1 through 2-4 illustrate the locations of the reactor facility piping.

Visual (boroscope) inspection of the internal surfaces of reactor room drain piping revealed breaks or blockages in the floor drain piping beneath the Reactor Room floor. Indications of potential breaks in the floor drain piping beneath the ground floor, and in the piping to the Hot Cell collection tanks under the roadway (tanks were removed during remediation). This inspection also identified accumulations of scale and loose debris, concentrated on the bottom surfaces of the piping. Visual inspection of storm and sanitary system piping was not conducted.

Broken or damaged areas of piping were accessed, and contaminated pieces of pipe and soil were identified and removed. The locations in the ground floor drain and hot cell piping

that indicated potential damage, were excavated, and determined to be intact. Hydrolazing of reactor piping internal surfaces was performed to remove scale and loose debris. Piping access points have been created to enable final status survey and NRC confirmatory activities.

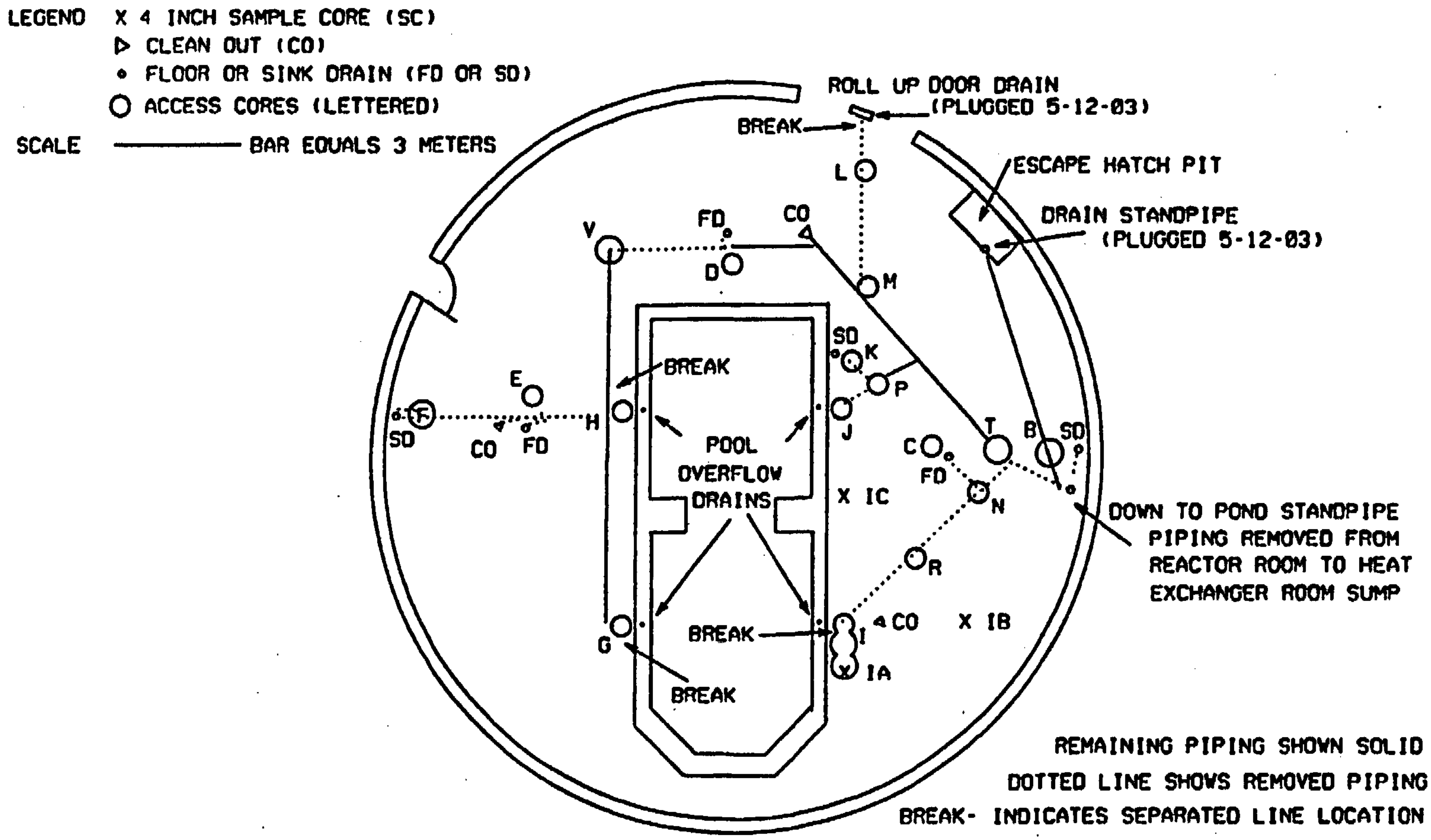

Figure 2-1 UVAR Reactor Room Floor Showing Remaining Drain Piping.

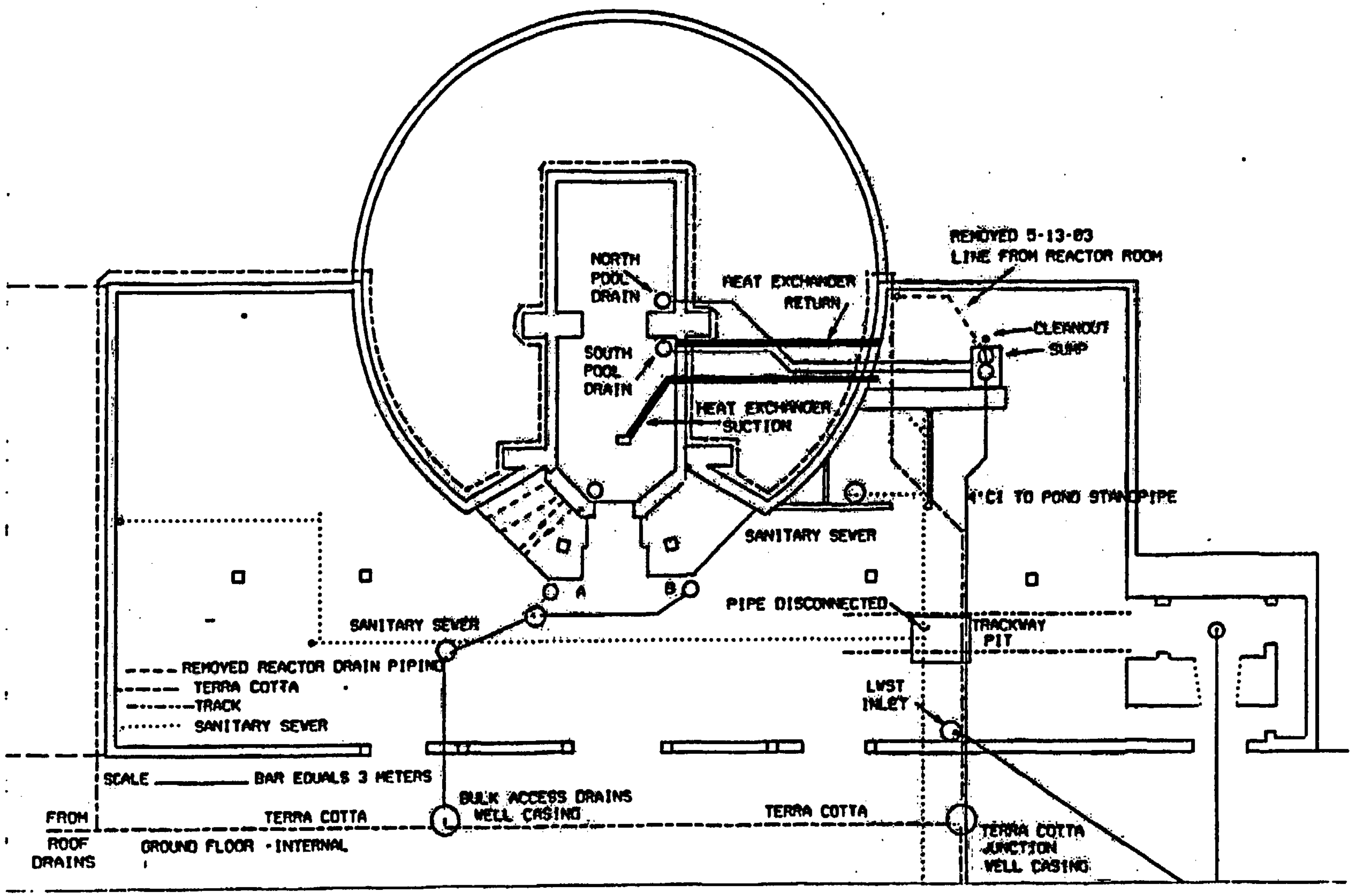

Figure 2-2 Ground Floor Indicating Remaining Drain Piping.

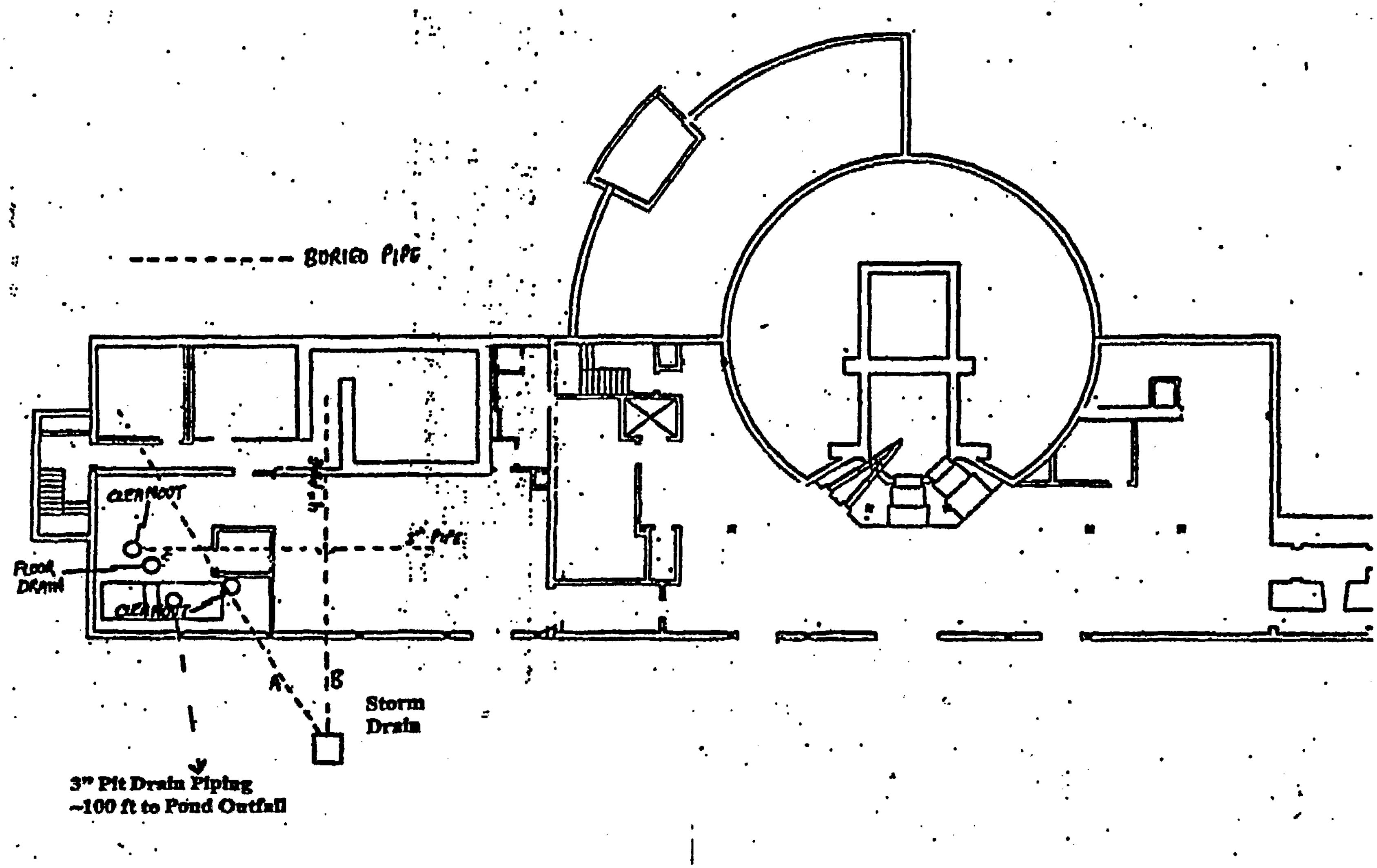

Figure 2-3 Drains Servicing the Former CAVALIER Facility.

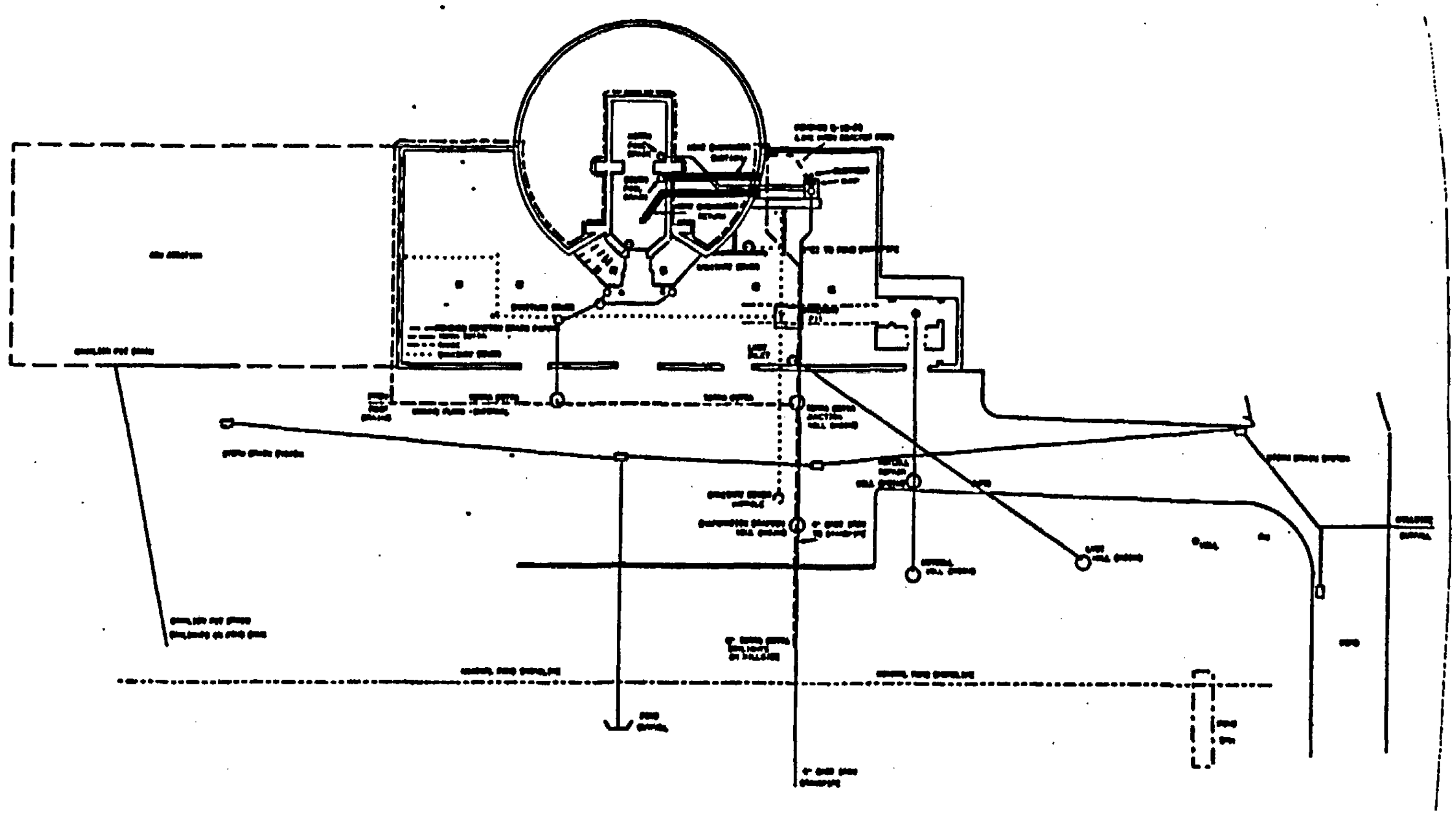

Figure 2-4 Sanitary and Storm Drains and Drain Outfalls for the UVAR Facility.

3. Contaminants of Concern and Guidelines

Soil removed during excavation of the underground waste tanks, soil from the vicinity of piping breaks, debris collected from piping, and pieces of removed piping were analyzed by gamma spectroscopy. These analyses identified Co-60 as the primary potential contaminant in most of the remaining piping. Cs-137 is the major potential contaminant associated with the Hot Cell drain and was also present in other piping systems, but at lower levels than Co-60. Even smaller quantities of Co-57, Eu-152 and Ag-108m were identified in several of the samples.

Direct and removable activity measurements were performed at exposed drain entries and on accessible internal piping surfaces, prior to remediation of piping breaks and blockage and hydrolazing to remove scale and loose debris. The large diameter heat exchanger piping was relatively accessible and was noted to have total surface contamination levels above the Co-60 criterion of 7100 dpm/100 cm^2. Small diameters, fittings, bends, and other internal obstructions limited access to most other piping and accumulations of scale and debris prevented reliable quantification of surface activity levels; however, direct measurements at drain openings and swabs indicated low levels of contamination, at levels below the Co-60 criterion. Based on use history, visual inspections, and results of the limited scoping surveys, all remaining reactor-associated piping is considered potentially contaminated. With exception of the Hot Cell drain, which is predominantly contaminated with Cs-137, the contaminant in reactor facility piping is assumed to be primarily Co-60; this is the most limiting of the identified potential contaminants and, in consideration of the limitations in performing such piping surveys, provides a level of conservative assurance that decommissioning is adequate. Criteria for FSS of reactor facility piping are 28,000 dpm/100 cm^2, and 7,100 dpm/100 cm^2 for Cs-137 and Co-60 contaminated piping, respectively; where mixtures of these contaminants are present, adjusted gross activity guidelines are applicable. Removable activity criteria are 10% of these total activity values.

4. Survey Approach

4.1 Survey Reference System

Locations of survey measurements and sampling will be referenced to piping access points and identified on facility drawings.

4.2 Survey Classification

Based on the facility use history and identification of contaminants of license origin in the remaining impacted piping, the reactor facility piping surfaces are designated Class 1 for FSS planning and implementation purposes. Storm drains, building and pool footing drains, the CAVALIER Facility drains, and the non-release path portion (west line) of the sanitary sewer system are designated Class 2.

4.3 Survey Unit Identification

Piping has been grouped into the following seven survey units:

- Reactor Room floor drain system.
- Heat Exchanger piping.
- Reactor pool drains.
- Hot Cell drain.
- Drain to liquid waste storage tanks.
- Reactor drains to pond.
- Sanitary sewer release path.
- CAVALIER Facility drains.
- Storm drains, building and pool footing drains and the non release path portion (west line) of the sanitary sewers.

4.4 Demonstrating Compliance with Release Guidelines

Compliance with decommissioning requirements will be demonstrated by comparing the results of FSS measurements with surface activity default screening criteria for Cs-137, Co-60 or a mixture determined in accordance with the Appendix A of the Master FSS Plan. Because instrument background may be a significant fraction (>10%) of the count rate representing the total activity criteria, background correction of the direct measurements will be performed. Average instrument background levels in non-impacted, buried metal,

PVC, and terra cotta piping will be determined. Sufficient measurements will be obtained to determine the average for each material type and instrument used to an accuracy of +/- 20% at the 95% confidence level (methodology of draft NUREG/CR-5849). Statistical testing of results will utilize the Sign Test to reject or accept the null hypothesis that the residual contamination exceeds the release criteria. Decision errors will be 0.05 (Type 1 and Type 2).

4.5 Number of Required Data Points

For Co-60 as the contaminant

DCGL = 7100, LBGR = 0.5 DCGL (Refer to Master FSS Plan, Section 7.8)
Δ = DCGL - LBGR = 3550
σ = 2300 (based on the MDA for the least sensitive measurement technique; refer to Attachment)
Δ/σ = 1.55
From MARSSIM Table 5.5, N = 18; i.e., 18 measurements required for each survey unit for Sign Test evaluation.

Although the relative shift (Δ/σ) would be higher and the number of data points required would be lower for Cs-137 as the controlling contaminant, for consistency the number of data points (i.e., 18) will remain the same for all piping survey units.

4.6 Sampling Pattern

Direct measurements will be obtained at equally spaced intervals along the piping to assure a minimum of 18 data points. For example, for the 55 ft (16.7 m) length of Hot Cell piping measurements will be obtained at 0.9 m intervals along the entire length of this piping.

4.7 Survey Methods

Scans and surface activity measurements of interior surfaces of 6 in (or larger) ID piping will be performed using Model 43-68 gas proportional or Model 44-9 pancake GM detectors, depending on accessibility with such detectors. Piping which is not accessible with those detectors will be surveyed using a Victoreen Model 491-30 GM detector. This latter detector has a 30 mg/cm² wall thickness and in an unshielded configuration has an effective field of view of slightly more than 100 cm² in a 2 in ID pipe. The overall diameter of the 491-30 detector assembly is approximately 2.9 cm, enabling access to most piping surfaces. Detector response to Co-60 in piping was determined by cross calibration, using a section of contaminated piping containing a measured activity level.

Following removal of contaminated piping and soil and hydrolazing to remove scale and loose contamination from internal surfaces, the interior surfaces will be scanned by passing the detector through the pipe. The rate of detector movement will be approximately 1 detector width/sec for the gas proportional and pancake GM detectors and 2.5 to 3.0 cm/sec for the 491-30 GM detector. Model 2221 scaler/ratemeters used with the detectors will be monitored for changes in audible signal and any indication of elevated count rate, suggesting possible presence of radioactive contamination, will be noted for further

investigation. Scan coverage will be 100% of the length of Class 1 piping and 25% of the length of Class 2 piping.

One-minute static counts will be performed at the designated systematic locations (see Sections 4.5 and 4.6) and at locations of elevated count rate identified by scans.

A swab will be passed through each pipe section to collect removable activity.

Where contaminated piping has been removed, gamma scans of the excavation soil surface will be performed to confirm the effectiveness of remediation. Samples of soil will be obtained from such excavations. Samples of gravel will be obtained from the French drain around the bottom of the reactor pool.

4.8 Sample Analyses

Swabs for removable activity will be scanned with a Model 43-68 detector to identify areas of elevated beta activity. This same detector will be used to perform a one-minute count to measure activity levels at each elevated beta activity location (if any) identified by the scans, or at a random location on the swab (if no areas of elevated activity are noted).

Samples of soil and gravel will be analyzed by a commercial laboratory by gamma spectrometry.

4.9 Investigation

If surface scans, direct measurements, swabs, samples, or statistical data evaluation identify residual contamination on Class 1 surfaces exceeding release criteria, remediation will be performed, followed by resurvey, as appropriate. Identification of contamination exceeding release criteria on Class 2 surfaces will require investigation, remediation, reclassification to Class 1, and resurvey in accordance with the higher rigor for Class 1 surfaces.

Following FSS activities, piping access points will be covered to prevent recontamination and to allow for future NRC confirmatory actions.

5. Data Evaluation

Total and removable net activity levels will be calculated. Data will be assessed for conformance with the FSS Plan and design DQOs. Additional data will be obtained, if required, and the assessment repeated. The Sign Test will be performed and results compared with critical values for the appropriate number of data points and decision errors.

Samples of soil and gravel will be compared directly with established criteria, using the Unity Rule.

ATTACHMENT A

To Addendum 002

Response and Field Use
of the
Victoreen Model 491-30 GM Detector
for
FSS of Reactor Facility Piping

Response and Field Use of the Victoreen Model 491-30 GM Detector for FSS of Reactor Facility Piping: Attachment A to Addendum 002 for the UVAR Facility Master FSS Plan

Introduction

Small-diameter piping and piping with internal obstructions is not adequately accessible by the 125 cm^2 gas proportional detector or the 15.5 cm^2 pancake GM detector, commonly used for final status survey (FSS) of other surfaces. Therefore a smaller detector is needed to perform scans and direct measurements of the internal surfaces of most of the impacted reactor piping remaining at the UVAR facility. A Victoreen Model 491-30 GM detector with a 30 mg/cm^2 window thickness, a tube length of 6.3 cm, and an effective field of view of approximately 100 cm^2 in the unshielded detector configuration was selected for such applications. The entire detector assembly, including the integral sliding shield, is approximately 2.9 cm in diameter. Piping with a diameter of 5.1 cm (2 in) or greater will therefore be accessible by this detector.

Detector response to Co-60 contamination in pipe scale on the interior of a 2 in ID pipe was determined as follows:

- A 20 to 25 cm section of contaminated cast iron pipe was obtained from beneath the Reactor Room floor.
- A gamma spectrum of this section was obtained and Co-60 was identified as the dominant (>80%) contaminant.
- The piping was cut lengthwise into 4 strips. These strips were arranged side-by-side to provide a "flat" contaminated surface source.
- The activity level on this surface was measured, using a 125 cm^2 Model 43-68 gas proportional detector; this gas proportional detector had a total efficiency factor of 0.10 for Tc-99, which has a beta energy spectrum, which is very similar to the energy spectrum of Co-60. The average source count was 358 c/m and the average detector background count was 208 c/m, resulting in a net count rate of 150 c/m. The resulting activity level determined for the piping surface was 1200 dpm/100 cm^2.
- The piping pieces were then reassembled into their original configuration, i.e., a 2 in diameter pipe with internal surface contamination; the pipe was positioned with the scale concentrated on the bottom of the pipe.
- The Victoreen detector (Serial No. 339) was covered with a thin plastic sleeve to prevent contamination and inserted into the pipe; the detector was in close contact with the contaminated scale on the bottom of the pipe, at the same location where the gas proportional detector measurement had been performed.

- Ten-minute counts were performed with the Victoreen 491-30 detector shielded (background) and unshielded (source response). The resulting count rates were: background, 36.1 c/m, and source, 52.3 c/m.

- The net response of the Victoreen detector was calculated to be (52.3 c/m – 36.1 c/m)/1200 dpm/100 cm² or (0.0135 c/m)/dpm/100 cm². This response can also be expressed as 74.1 dpm/100 cm² of Co-60 activity per 1 net c/m on the Victoreen 491-30 detector.

MDA's are estimated using the following relationships:

$$MDA_{\text{Static count}} = \frac{3 + 4.65[R_B \bullet t]^{0.5}}{t \bullet \epsilon_S}$$

Where R_B = Background count rate (C/min)
t = Counting time (min)
ϵ_S = Instrument response (c/m/dpm/100 cm²)

$$MDA_{\text{Scan}} = \frac{d'[R_B \bullet \frac{i}{60}]^{0.5}[\frac{60}{i}]}{\sqrt{p} \bullet \epsilon_S}$$

Where R_B = Background count rate (c/min)
i = interval in contact with source (sec)
p = Surveyor efficiency (0.5)
d' = index of sensitivity (1.38)
ϵ_S = Instrument response (c/m/dpm/100 cm²)

Based on the detector response and a background of 36.1 c/m, the estimated MDA for a 1-minute static measurement with the Victoreen 491-30 detector is:

$MDA_{\text{1-minute count}}$ = [3 + 4.65(36.1c/m)$^{0.5}$]/[0.0135 c/m/dpm/100 cm²]

= 2292 dpm/100 cm²

The MDA for a scan at the detector movement rate of 3 cm/sec (scan interval of 6.3 cm/3 cm/sec = 2.1 sec) is:

MDA_{scan} = (1.38(36.1/60)$^{0.5}$)(60/2.1)/ (0.5)$^{0.5}$(0.0135)

= 3204 dpm/100 cm²

Both the static count and scan MDA's are below the guideline values of 7100 dpm/100 cm² for Co-60.

For piping contaminated predominantly with Cs-137, use of the response factor developed for Co-60 will overestimate the contamination level, because of the higher energy of the Cs-137 beta particles and the resulting increases in detector efficiency and source efficiency.

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Final Status Survey Plan

Addendum 003: Pond Sediments

Revision 0

Prepared for
University of Virginia
Reactor Facility Decommissioning Project

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June 2003

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Contents

FIGURES

1. Introduction

The Master Final Status Survey Plan (UVA-FS-002) identifies the Data Quality Objectives for Final Status Survey (FSS) activities, together with the underlying technical assumptions, approaches, and methodologies for designing, implementing, and evaluating a FSS on each impacted area of the University of Virginia Research Reactor (UVAR) Facility. A separate survey area-specific addendum is prepared for each area or group of areas with common media, contaminants, and other characteristics, prior to beginning FSS; FSS for the specific area or group of areas will then be performed in accordance with that addendum. This addendum (Addendum 003) applies to the sediments in the small on-site pond to the south of the UVAR Facility.

It should be noted that a characterization survey of the pond sediments was conducted during late September 2002 to coordinate drainage of the pond with the time of the year when reduced precipitation is typically expected. The drainage and characterization were also timed to facilitate excavation of the adjacent underground waste tanks (refer to FSS Addendum 002). Design and implementation of the characterization survey of the pond sediments were based on providing adequate data to also enable evaluation as a final status survey, in case the results indicated the decommissioning criteria were satisfied, no remediation was required, and the sediment medium was not subject to potential future contamination during the decommissioning activities performed after the survey.

2. Description

Storm runoff from the adjacent land areas and overflow from the storm drain on the UVAR site are collected in a small pond, located to the south of the UVAR Building (see Figure 2-1). Some laboratory drains, floor drains, and other sources of non-sanitary wastewater with low potential for radiological or other hazardous constituents also routinely discharged to this pond. Two underground waste tanks serviced the Hot Cell, and two tanks were used for collection of demineralizer regeneration liquids from the reactor. Both of these sets of tanks were originally plumbed to allow the contents to be discharged to the pond, provided the liquid met appropriate release criteria following dilution with the pond water; the demineralizer regeneration tanks were later replumbed so they could be discharged directly into the pond spillway.

The pond covers a surface area of about 1450 m^2, and ranges in depth from approximately 2 to 4 m. The pond bottom is covered with sediments, ranging from a few cm to several m thick. Figure 2-2 is a map of the pond, indicating pertinent features.

During the late summer of 2002, the pond was drained. This allowed the sediments to dry and exposed the sediment and various piping surfaces to facilitate radiological monitoring and sampling.

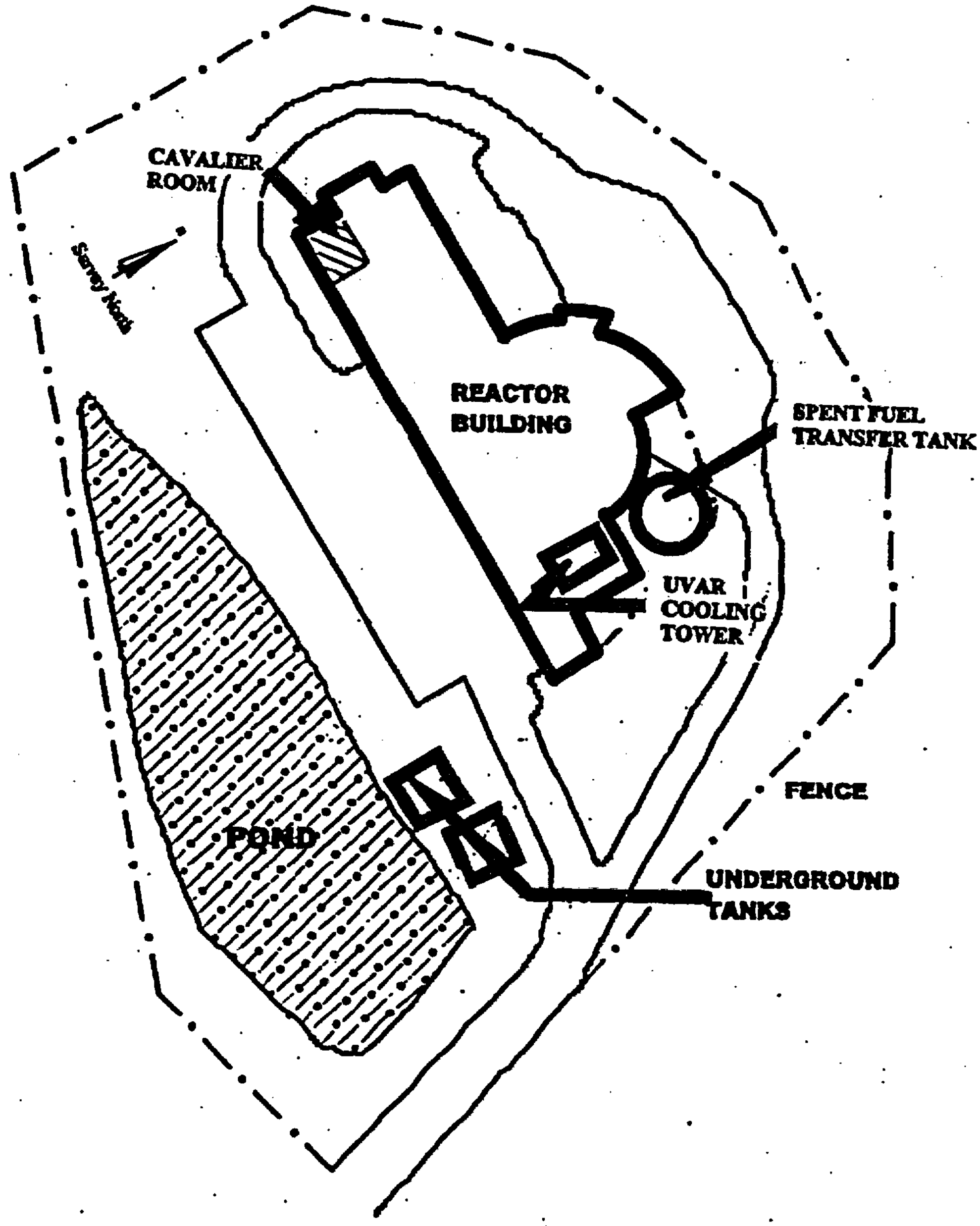

Figure 2-1 University of Virginia Reactor Facility and Environs

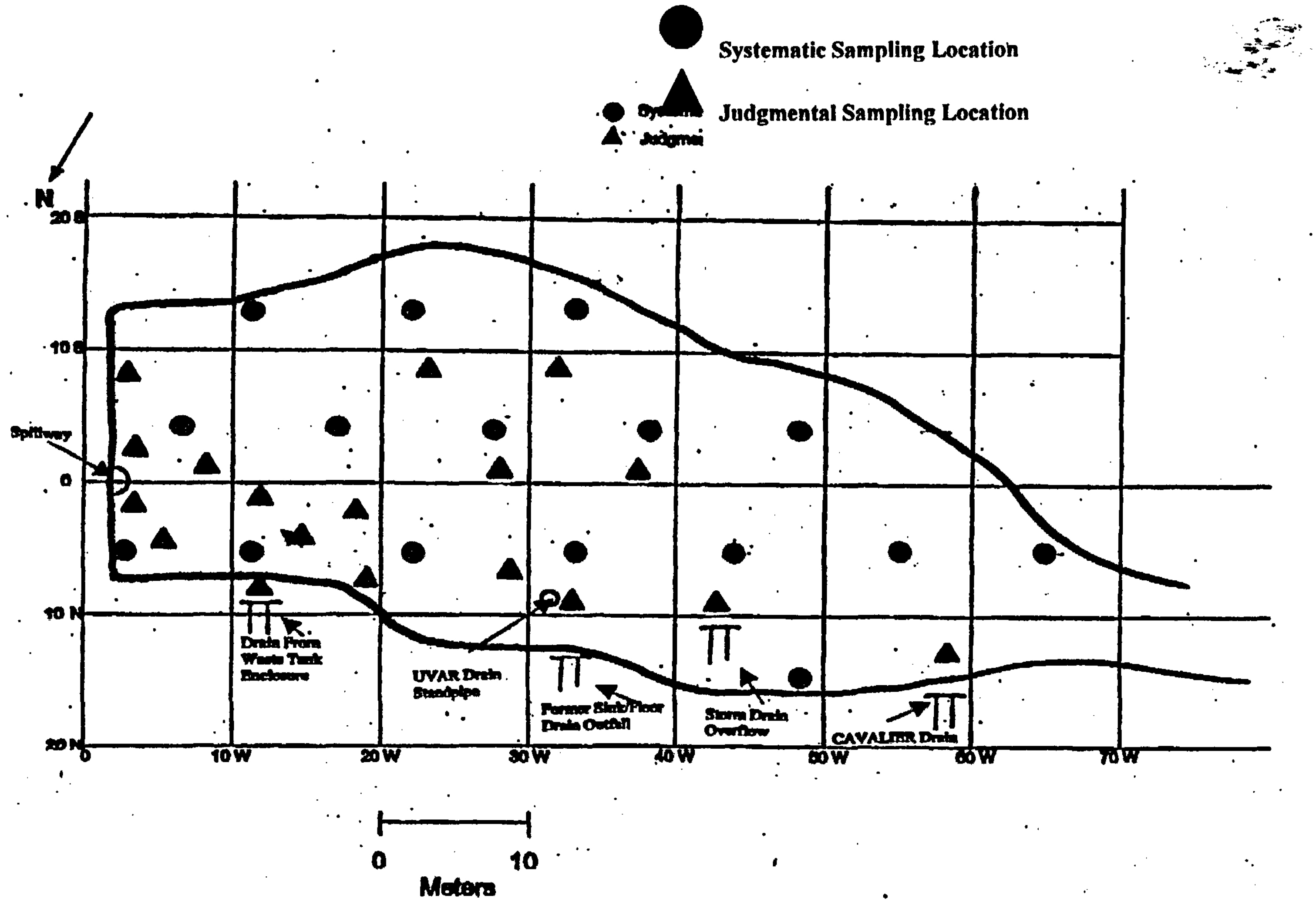

Figure 2-1 Figure 2-2 Plot Plan of Pond, Indicating Reference Grid and Sampling Location.

3. Contaminants of Concern and Guidelines

During facility operation, there were several intentional and unintentional discharges of low-level contaminated liquids to the pond ~~occurred~~. Two of these occurred in laboratories M005 and M008, and involved contamination by Tc-99 and Ni-63, respectively. Reactor Pool water discharges to the pond were made in the 1960's. A break in the piping from the demineralizer regeneration tanks resulted in release of low-level contaminated liquids, containing primarily Cs-137 and Co-60, onto the bank of the pond. Because of this history; there is a potential for the sediments to be contaminated with facility-derived radionuclides. Samples of pond sediments analyzed during the 1999 GTS Duratek characterization identified Cs-137, Co-60, Eu-152, and Pu-241; however, the concentrations in most samples were below the laboratory measurement sensitivities. Members of the natural uranium and thorium series were also noted in the sediments at concentrations ranging from 1 to 3 times typical background soil levels. The various members of these decay series appear to be in secular equilibrium. Based on this equilibrium state and the operating history of the site, the slightly elevated natural uranium and thorium concentrations are considered to be of natural origin and are not attributed to licensed activities.

Additional characterization was performed in September 2002, following draining of the pond. This characterization included surface gamma scans, sampling of sediments at depths ranging from the surface to undisturbed soil at systematic and judgmental locations, and gamma logging of augered sampling holes. This additional characterization is described further in Section 4 of this Addendum.

Release levels for the contaminants in pond sediment are the NRC default screening criteria; the default screening criteria concentrations for Co-60, Cs-137, Eu-152, and Pu-241 (contaminants identified by the GTS Duratek characterization) are 3.8 pCi/g, 11 pCi/g, 8.7 pCi/g, and 72 pCi/g, respectively. For the final status survey, sediments from the pond will be analyzed for specific potential radionuclide contaminants, and results must satisfy the Unity Rule for the sum of ratios of radionuclide concentrations present to respective screening default guideline levels.

4. Survey Approach

4.1 Survey Reference System

A 10-m grid was established over the pond area, extending onto the banks surrounding the pond. This grid will be referenced to the reference grid established for survey of the entire site land area, thus enabling the survey locations to be related to the federal and/or state planar coordinate system. Figure 2-2 indicates the reference grid system.

4.2 Survey Classification

Based on the facility use history and identification of contaminants of license origin in the pond sediments, the sediments are designated Class 1 for FSS planning and implementation purposes.

4.3 Survey Unit Identification

The sediments comprise one survey unit. For survey design purposes the planning area of the survey unit is 1450 m^2.

4.4 Demonstrating Compliance with Release Guidelines

Compliance with decommissioning requirements will be demonstrated by comparing the results of characterization/final status survey sample analyses with default screening criteria, using the sum of ratios; the sum of ratios must satisfy the Unity Rule. Because the radionuclides identified as potential contaminants are not present in background at concentrations, which are significant fractions of the release guidelines, correction of FSS sample data for background levels will not be required. Statistical testing of results will utilize the Sign Test to reject or accept the null hypothesis that the residual contamination exceeds the release criteria. Decision errors will be 0.05 (Type 1 and Type 2).

4.5 Number of Required Data Points

The following calculation is based on use of Unity Rule (i.e., DCGL = 1).

DCGL = 1, LBGR = 0.5 DCGL (Refer to Master FSS Plan, Section 7.8)
Δ = DCGL - LBGR = 0.5
σ = 0.25 (based on the sum of ratios of maximum levels of radionuclides detected in characterization samples to respective DCGL's)
$\Delta/\sigma = 2$
From MARSSIM Table 5.5, N = 15; i.e., 15 data points required in the survey unit for statistical evaluation using the Sign test.

4.6 Sampling Pattern

Sampling was performed at systematically spaced intervals on a triangular pattern throughout the pond. The spacing between data points was determined to be:

$L = [1450/(0.866 \times 15)]^{0.5} = 10.56$ m (A spacing of 10.5 m between samples and a spacing of 9.0 m between N-S lines of sampling points was used for ease of field implementation).

A random start point for the pattern was based on survey unit dimensions and random numbers from the MARSSIM random number table of 0.7337 and 0.4872. The resulting start point was 7.3 m W and 4.9 m S.

Systematic sampling locations were:

4.9 S, 7.3 W	4.1 N, 2.0.W
4.9 S, 17.8 W	4.1 N, 12.5 W
4.9 S, 28.3 W	4.1 N, 23.0 W
4.9 S, 38.8 W	4.1 N, 33.5 W
4.9 S, 49.3 W	4.1 N, 45.0 W
13.9 S, 12.5 W	4.1 N, 55.5 W
13.9 S, 23.0 W	4.1 N, 66.0 W
13.9 S, 33.5 W	13.1 N 49.3 W

In addition to the 16 systematic sample locations, samples were obtained at 18 judgmental ("biased") locations in the vicinity of inlet and outlet piping.

4.7 Survey Methods

Gamma walkover surface scans were performed using a 2"X 2" NaI detector (Ludlum Model 44-10) coupled with a Ludlum Model 2221 ratemeter/scaler. The detector was maintained within 5-10 cm of the sediment surface and moved from side to side in a serpentine pattern while noting any indication of elevated count rate, which might indicate the presence of radioactive contamination. Results (count rate) were documented on survey area maps. Locations of elevated response were noted for further investigation. Scanning coverage was 100% of the sediment surface.

Surface (0-15 cm) sediment samples of approximately 500 g were collected at the 16 systematic and 18 judgmental sampling locations identified above. In soft sediments, sediment columns were obtained by driving PVC pipe to refusal, capping and removing the pipe and extruding the core into a half pipe. In more resistant sediments, boreholes were augered through the sediments to the underlying soil using a 2-in diameter bucket auger.

Some locations required a combination of both methods. The resulting samples were obtained from depths of 15 to 45 cm, 45 to 75 cm, and 75 to 105 cm, where thickness of sediment allowed. If a sample could not be obtained from a pre-identified location, one was obtained from the nearest sediment location available; the survey/sampling record noted this situation. A total of 92 samples were obtained. Duplicate samples were collected at 4 locations. Samples were assigned unique identification numbers and a chain of custody record and analytical request were prepared.

Boreholes were gamma logged at 30 cm intervals from the surface to the bottom of the borehole; where necessary to maintain a borehole open, thin-walled PVC piping was inserted into the borehole as the auger was advanced.

4.8 Sample Analyses

Sample cores were scanned for gamma and beta activity. All samples were analyzed in the on-site laboratory by gamma spectrometry. Based on the results of surface scans, borehole logging, sample core scans, and on-site analyses, 6 samples were sent to an off-site commercial laboratory for gamma spectrometry and analysis for hard-to-detect (10 CFR Part 61) radionuclides. Results of these analyses will be used to develop fractional contributions of non-gamma emitting radionuclides in sediments. These contributions will be used to adjust the results of individual sample gamma analyses for the presence of other contaminants. Gamma analyses will be surrogates for calculating soil activity concentrations, assuming the previously analyzed mix of radionuclides remains proportional to the concentrations of principal gamma emitters. In accordance with NMSS Decommissioning Standard Review Plan (NUREG-1727, Appendix E, Section 11.1), samples from systematic locations will be homogenized over 1-m depths (where applicable) and analyzed by gamma spectrometry by the off-site laboratory. Results of these analyses will be used for final characterization and FSS, if appropriate.

4.9 Investigation

If sample analyses do not identify radionuclide concentrations in sediments exceeding release criteria, the characterization data will be regarded as FSS data and assessment and statistical evaluation will be performed (refer to Section 5). If sample analyses indicate or statistical data evaluation identifies residual radioactivity concentrations exceeding release criteria, contaminated sediments will be remediated and FSS activities repeated, utilizing a newly determined sampling pattern and random start point.

5. Data Evaluation

The sum of ratios will be calculated for radionuclides, which are of facility origin. The Sign Test will be performed and the results compared with the critical value for the appropriate number of samples and decision errors.

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Final Status Survey Plan

Addendum 004: Interior Structure Surfaces

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1. Introduction

The Master Final Status Survey Plan (UVA-FS-002) identifies the Data Quality Objectives for Final Status Survey (FSS) activities, together with the underlying technical assumptions, approaches, and methodologies for designing, implementing, and evaluating a FSS on each impacted area of the University of Virginia Research Reactor (UVAR) Facility. A separate survey area-specific addendum is prepared for each area or group of areas with common media, contaminants, and other characteristics, prior to beginning FSS; FSS for the specific area or group of areas will then be performed in accordance with that addendum. This addendum (Addendum 004) applies to the interior surfaces of the UVAR Facility building.

2. Description

The UVAR Facility is located on Old Reservoir Road, approximately 0.6 kilometers (km) west of the Charlottesville, VA city limits. The Facility includes the UVAR building, a small pond, and asphalt-paved roads, parking areas, and equipment/materials storage pads, situated on a land area of approximately 9500 m^2 (see Figure 2-1). The three-story building housed the UVA Research Reactor and the CAVALIER facility, as well as offices for the reactor staff and faculty and students of the Department of Nuclear Engineering, miscellaneous laboratories, and other support facilities for the reactors and Department of Nuclear Engineering.

Figures 2-2 through 2-4 show the three levels of the UVAR facility. The upper level has approximately 1190 m^2 of floor area. The Reactor Confinement Room (Rm 131), which housed the former UVA Research Reactor, is located on the upper floor (first floor). This room contained the 9.8 m long by 3.7 m wide by 8.2 m deep reactor pool, associated operating equipment and systems, the operating controls, and some research/experimental equipment. This room is circular and has an elevated (~10 m) ceiling. In addition, the Instrument Shop (Rm 128), Shipping Area (Rm 127), and multiple offices and other support facilities for staff and students are located on this building level.

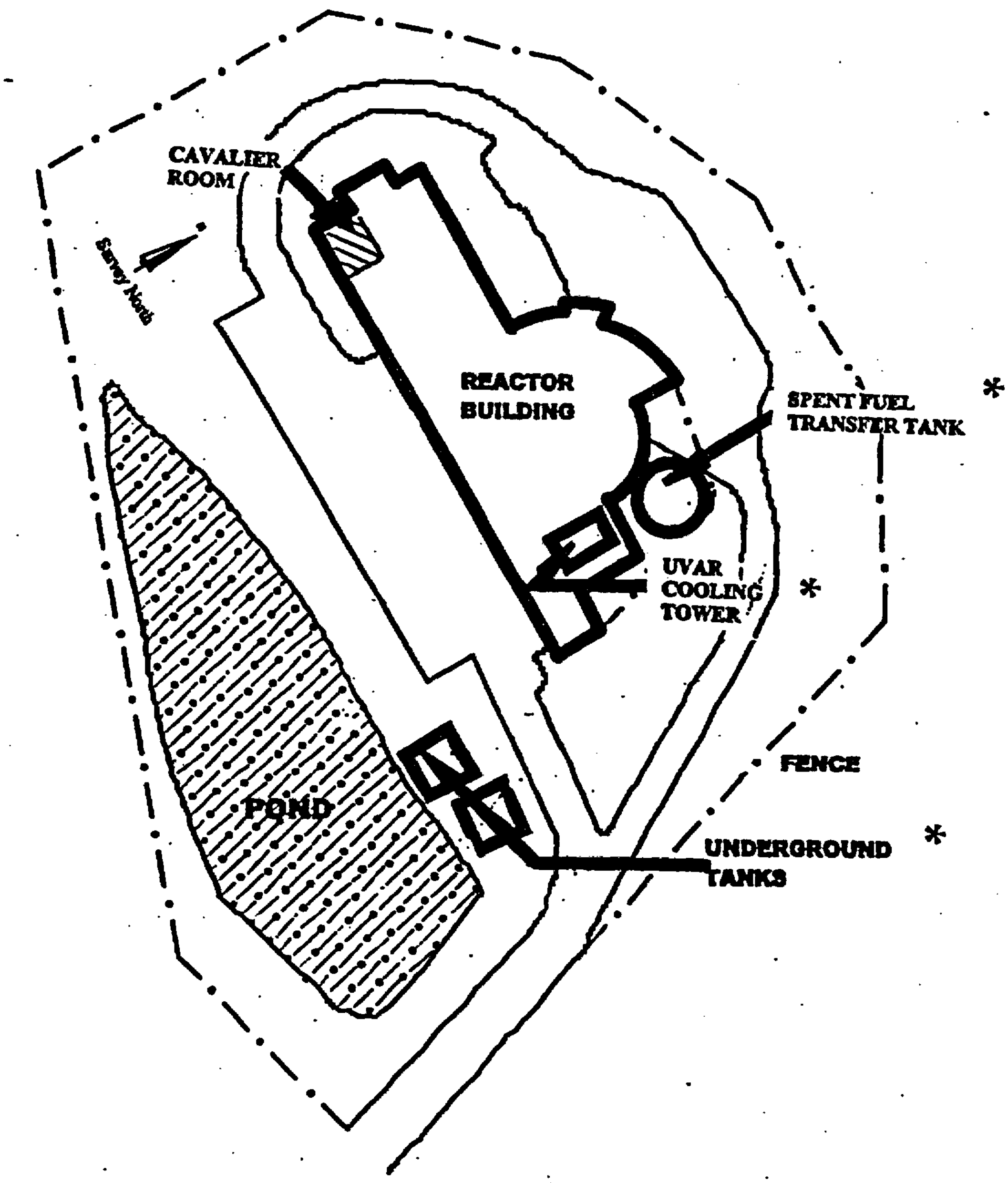
On the approximately 1100 m^2 Mezzanine level were located the Demineralizer (Rm M021), Mechanical Room (Rm M020), HP Laboratory (Rm M019), several partially contaminated laboratories (Rms M005 [Tc-99 contamination] and M008 [Ni-63 contamination]), and multiple offices and other support facilities for staff and students. A crawl space (MCS) is accessed from the stairwell on the Mezzanine level.

The 1210 m^2 ground floor contained the Heat Exchanger (Rm G024), Rabbit Room (Rm G005), Beamport/Experimental area (Rm G020), Hot Cell (Rms G025, G026, and G027), Counting Room (Rm G004), Woodworking and Machine Shop (Rm G008), Source Storage (Rms G022, G018, and G007A), the former CAVALIER facility (Rm G007), and miscellaneous support facilities and areas.

The UVAR building is of concrete block construction with brick veneer. Floors are concrete slab. Internal walls are block and drywall. Most offices, hallways, and small laboratories have a dropped ceiling of acoustical tile, and tile floors.

In preparation for implementing the Final Status Survey, impacted reactor and support systems and components were removed and disposed of as radioactive waste or surveyed and released for use without radiological restrictions. Contaminated facility surfaces and materials were removed or decontaminated.

Figure 2-1 University of Virginia Reactor Facility and Environs



*Removed during decommissioning

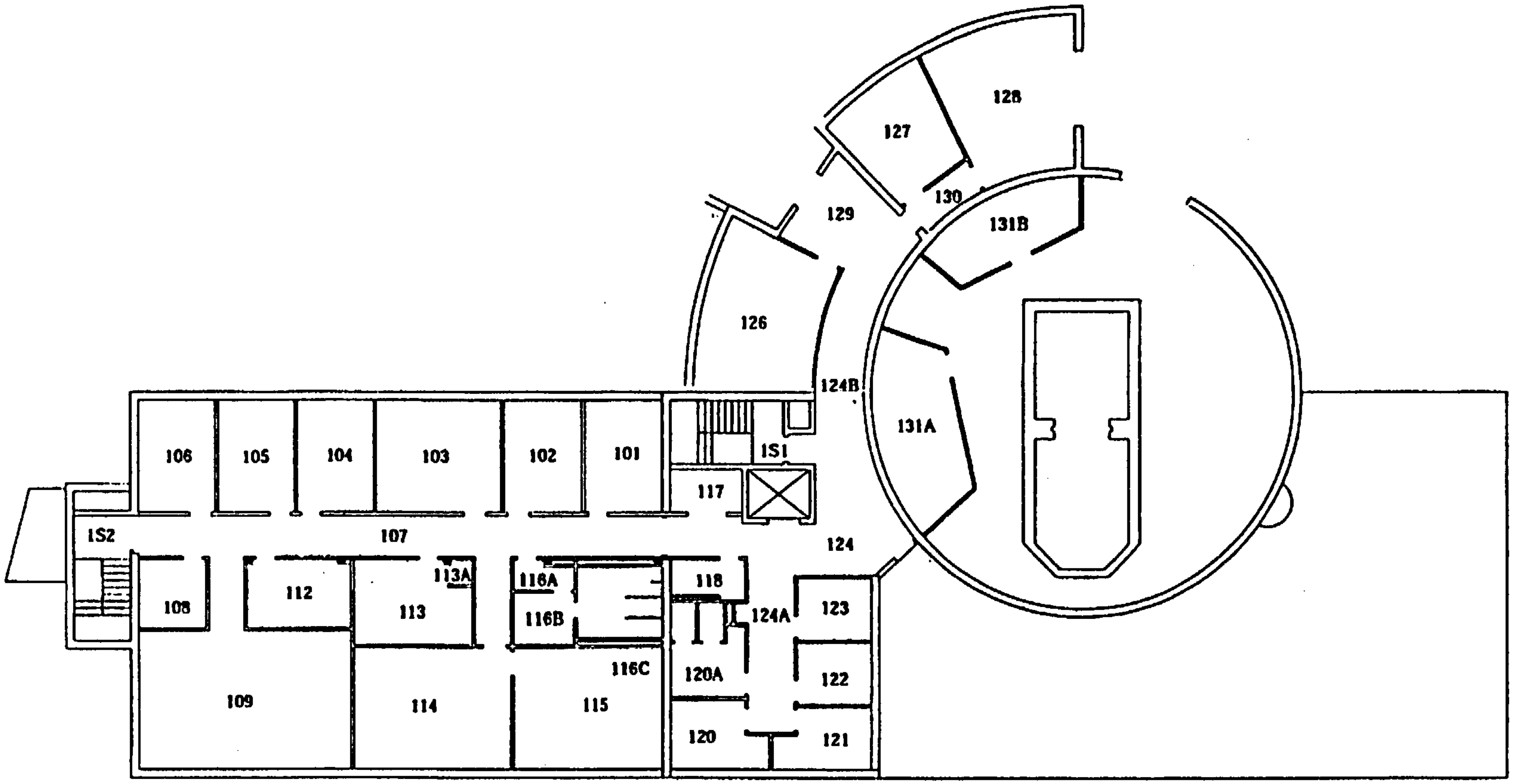


Figure 2-2 UVA Reactor First Floor Plan View

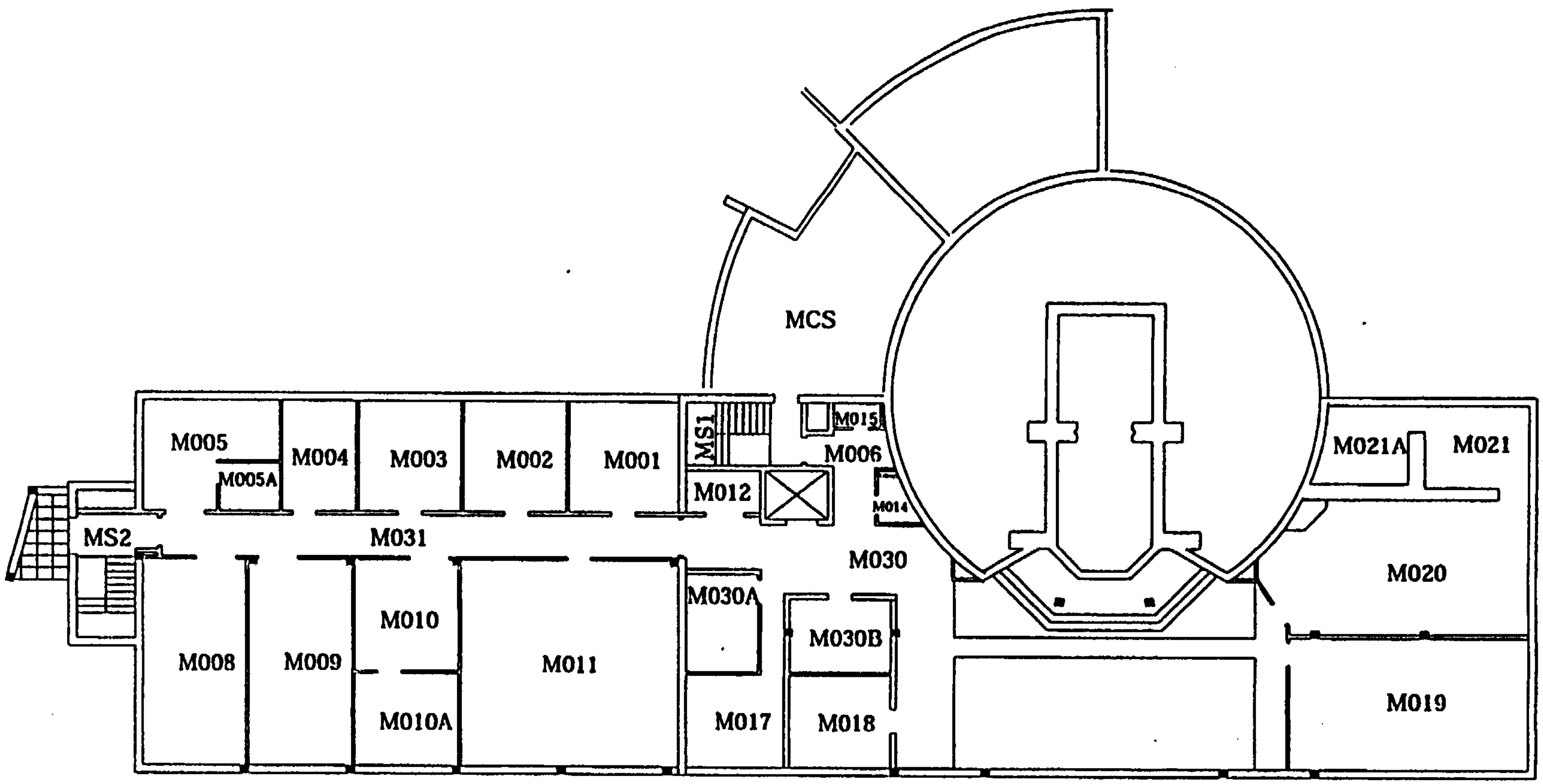


Figure 2-3 UVA Mezzanine Floor Plan View

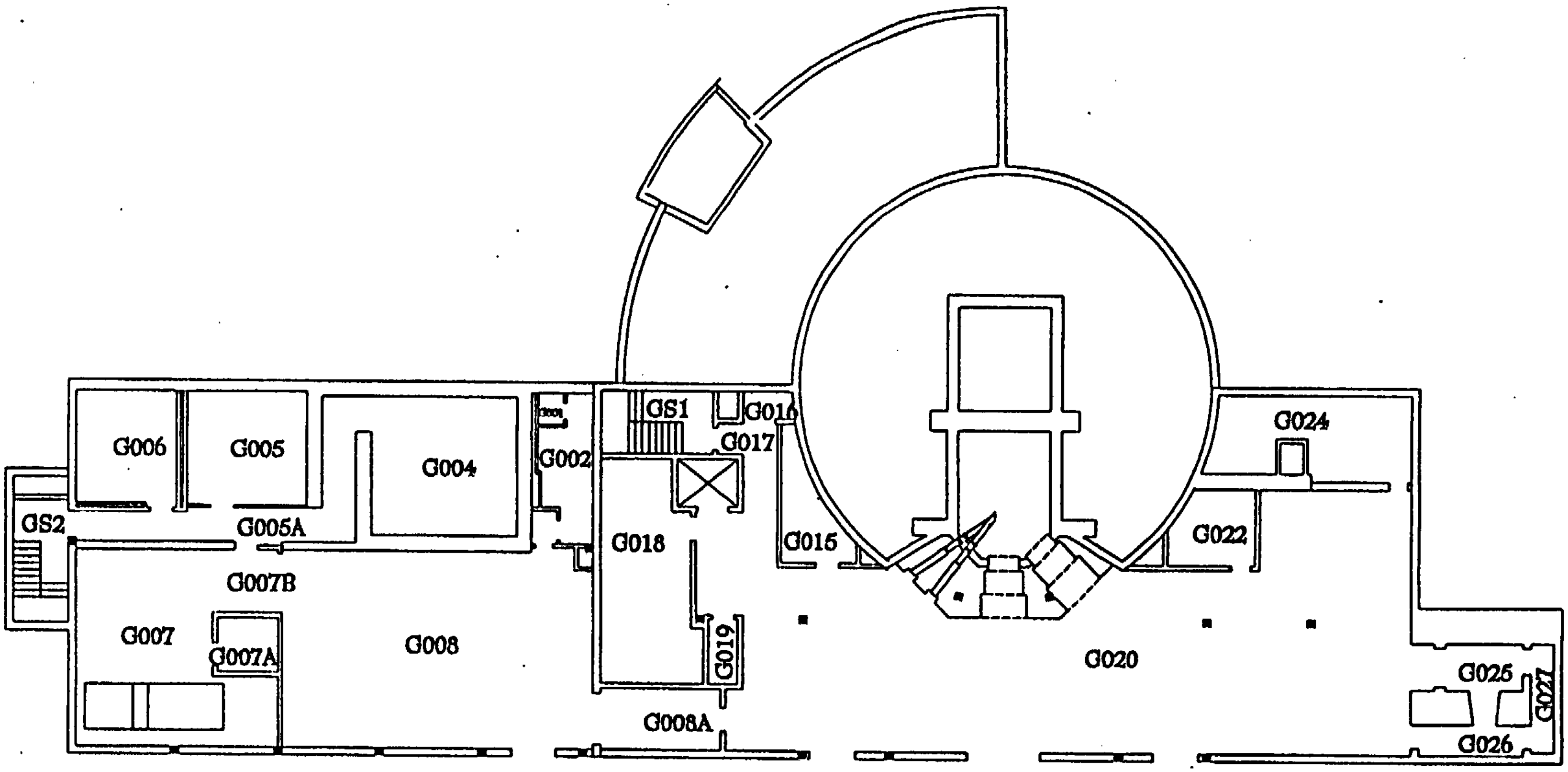


Figure 2-4 UVA Reactor Ground Floor Plan View

3. Contaminants of Concern and Guidelines

The GTS Duratek initial characterization and continuing characterization by the CH2M HILL team showed that radiological contamination was generally low level and was limited to a small portion of the structure interior. Major structural contamination was generally limited to surfaces exposed to or in contact with reactor coolant, reactor neutron fields, and materials containing high levels of activity (e.g., the Hot Cell). Depending on the mechanism of contamination and the medium, radionuclides and their relative ratios varied. The overall predominant radionuclide was Co-60; smaller activities of fission and activation products, namely Cs-137, C-14, Fe-55, and Eu-152 were identified in some media. Ni-63 and Tc-99 contaminants were present on facility surfaces from research projects in labs M008 and M005, respectively.

The Decommissioning Plan established the criteria for residual radioactive material contamination on UVAR facility surfaces. UVAR facility criteria also referred to as derived concentration guideline levels (DCGLs) are selected from the table of NRC default screening values in NUREG-1757. The screening DCGL of 7100 dpm/100 cm^2 for Co-60 is the most restrictive of the contaminants that have been identified as potential contaminants of concern. Unless there is specific evidence that contamination of a surface is comprised of radionuclides other than Co-60, the DCGL for Co-60 will be the basis for evaluating the final radiological status of the structure surfaces. Guidelines for removable structure contamination are 10% of the NRC screening default values for total surface activity, for Co-60 the removable activity limit is therefore 710 dpm/100 cm^2. This assures a conservative approach for satisfying the NRC dose-based criteria for future facility use. Appendix A of the Master FSS Plan describes the method for establishing guidelines for other radionuclides and combinations of radionuclides. If guidelines other than those for Co-60 are to be used, a justification will be prepared and included with the FSS documentation.

4. Survey Approach

4.1 Survey Reference System

A grid system will be established on surfaces to provide a means for referencing measurement and sampling locations. On Class 1 and 2 structure surfaces, a 1-m interval grid will be established; a 5-meter interval grid will be established on Class 3 structure surfaces. Upper surface (ceiling and overhead) locations may be referenced to the grid established for the floor beneath. Grid systems will originate at the southwest corner of the survey unit, except where specific survey unit characteristics necessitate alternate grid origins. Grids are assigned alphanumeric indicators to enable survey location identification and are referenced to building features. Maps and plot plans of survey areas will include the grid system identifications. Systems and surfaces of less than 20 m^2 will not be gridded, but survey locations will be referenced to prominent facility features.

4.2 Survey Classification

A listing of building interior surfaces and their MARSSIM classifications by contamination potential is contained in Table 4-1. Facility history (including the Historic Site Assessment) and radiological monitoring conducted during characterization and remedial activities are the bases for these classifications. Classification changes that indicate a lower potential for contaminated (and reduced FSS rigor) will require justification and concurrence by the NRC.

4.3 Survey Unit Identification

Table 4-1 lists the survey unit, based on MARSSIM classification and the recommended survey unit area limitations, suggested by MARSSIM. Contiguous structure surfaces will be grouped into survey units to satisfy the Class and area criteria. Classifications and survey unit boundaries may change, based on results as the FSS progresses; if classifications or boundaries change, the survey will be redesigned and the survey and data evaluation repeated.

Impacted structure surfaces of ≤ 10 m^2 and impacted land surfaces of ≤ 100 m^2 will not be designated as survey units. Instead, a minimum of 4 measurements (or samples) will be obtained from such areas, based on judgment, and compared individually with the $DCGL_W$.

4.4 Demonstrating Compliance with Release Guidelines

The Wilcoxon Rank Sum (WRS) test will be used for evaluating direct measurements of total surface activity, relative to the established criteria, where all survey unit measurements are on the same type of surface medium. Where multiple media are involved, the Sign test will be used. The selection of the test method will be survey unit-specific. The Null Hypothesis will be that activity levels in the survey unit exceed the release criteria. Rejection of the Null

Hypothesis will be required to demonstrate that the release criteria are satisfied. Decision errors will be 0.05 (Type 1 and Type 2).

Table 4-1 UVAR Building Interior Surface Survey Areas and Classifications

Room or Area	Surface	Class	Approximate Surface Area (m^2)	No. of Survey Units	Remarks
131 Reactor Room	Floor	1	130	2	
131 Reactor Room	Lower Walls	1	100	1	
Reactor Pool	Floor and Walls	1	150	2	
M005/005A	Floor and Lower Walls	1	45	1	
M008	Floor and Lower Walls	1	60	1	
M019	Floor and Lower Walls	1	80	1	
M020	Floor and Lower Walls	1	85	1	
M021/021A	Floor, Walls, and Ceiling	1	100	1	
Bio Shield Surfaces	Wall	1	100	1	
G005	Floor, Walls, and Ceiling	1	85	1	
G007/G007A	Floor, Pit and Lower Walls	1	100	1	
G018	Floor, Walls, and Ceiling	1	110	1	
G020	Floor and Lower Walls	1	300	3	
G022	Floor, Walls, and Ceiling	1	60	1	
G024	Floor, Walls, and Ceiling	1	100	1	
G025/G026/G027	Floor, Walls, and Ceiling	1	70	1	

Table 4-1 UVAR Building Interior Surface Survey Areas and Classifications (continued)

Room or Area	Surface	Class	Approximate Surface Area (m^2)	No. of Survey Units	Remarks
131 Reactor Room	Upper Walls and Ceiling	2	420	1	
127/128/130	Floor, Walls, and Ceiling	2	180	1	
107/124/124A/124B	Floor and Lower Walls	2	250	1	
M005/005A	Upper Walls and Ceiling	2	30	1	
M008	Upper Walls and Ceiling	2	40	1	
M019	Upper Walls and Ceiling	2	60	1	
M020	Upper Walls and Ceiling	2	65	1	
M006/M014/M015/M030/M031	Floor and Lower Walls	2	250	1	Includes catwalk over G020
MCS (crawl space)	Floor, Walls, and Ceiling	2	100	1	Soil samples per Addendum 006
G004/G005A	Floor and Lower Walls	2	100	1	
G006	Floor and Lower Walls	2	70	1	
G007B/G008/G008A/G016/G017/G019	Floor and Lower Walls	2	150	1	
Stairwell 1	Floor and Lower Walls	2	300	1	
Stairwell 2	Floor and Lower Walls	2	300	1	
Remainder of structure	Floors, walls, and Ceiling	3	4500	3	

NUREG-1505 (Ref. 5) contains details on data assessment/interpretation and selection and TBD = to be determined

4.5 Background Reference Areas and Materials

For applications of the WRS test, reference areas of the same material as the survey unit being evaluated, but without a history of potential contamination by licensed operations, will be identified. The number of reference data points will be the same (+/- 20%) as the number of data points required from the survey unit. A set of reference measurements will be obtained for each instrument being used for survey unit evaluation. For applications involving the Sign test, sufficient background determinations will be made for each media or surface material and with each instrument to provide an average background level that is accurate to within +/- 20%; this usually requires 8 to 10 measurements, which are then evaluated using the procedure described in draft NUREG/CR-5849 and additional data points obtained, as necessary. Reference area and background requirements will be survey-unit-specific.

4.6 Number of Required Data Points

The following calculation is based on Co-60 as the contaminant (i.e., DCGL = 7100).

DCGL = 7100, LBGR = 0.5 DCGL (Refer to Master FSS Plan, Section 7.8)
Δ = DCGL - LBGR = 3550
σ = 1200 (based on the MDA for the 43-68 gas flow proportional detector)
Δ/σ = 2.96
From MARSSIM Table 5.3, N/2 = 10; i.e., 10 data points each required for the survey unit and the reference area (WRS test). If the Sign test is to be used, 14 data points will be required for the survey unit.

4.7 Sampling Pattern

Sampling/measurements will be performed at systematically spaced intervals on triangular patterns throughout the soil and paved areas. The spacing between data points will be determined by the area of the survey unit and will therefore be survey unit-specific.

Random start points for the systematic sampling pattern will be determined for each survey unit, based on the overall survey unit dimensions and random numbers from the MARSSIM random number table.

In addition to the systematic locations, samples and measurements will be obtained at "biased" locations, identified by scanning or professional judgment of the FSS field supervisor as having the greatest potential for contamination. Examples of such locations include work area where radioactive materials were handled, high traffic areas, locations which required remediation to reduce or remove residual activity, and locations of elevated direct radiation, identified during the FSS.

4.8 Survey Methods

Gamma walkover surface scans will be performed using a 2"X 2" NaI detector (Ludlum Model 44-10) coupled with a Ludlum Model 2221 ratemeter/scaler. The detector will be maintained within 5-10 cm of the surface and moved from side to side in a serpentine pattern while noting any indication of audible elevated count rate, which might indicate the presence of radioactive contamination. Results (count rate) will be documented on survey area maps. Locations of elevated response will be noted for further investigation. Gamma scanning coverage will be 100% of Class 1 floor surfaces and a minimum of 25% for Class 2 and 10% for Class 3 floor surfaces. Where conditions allow, beta scans of structure floor surfaces will be performed using a large area (~580 cm²) gas proportional detector (Ludlum Model 43-37) coupled with a Ludlum Model 2221 ratemeter/scaler and Model 239-1 floor monitor; smaller Ludlum Model 43-68 gas proportional or Ludlum Model 44-9 pancake GM detectors coupled with Ludlum Model 2221 ratemeter/scalers will be used, as required by accessibility limitations. Ludlum Model 43-68 gas proportional or Ludlum Model 44-9 pancake GM detectors coupled with Ludlum Model 2221 ratemeter/scalers will be used to scan other structure surfaces. The detector will be maintained within ~1 cm of the surface while advancing the detector at a rate of approximately one detector width per second. Scan speed will be adjusted, as necessary, to assure detection sensitivities are less than 50% of the release criteria. Audible response will be monitored for indication of elevated count rate, which might indicate the presence of radioactive contamination. Results (count rate) will be documented on survey area maps. Locations of elevated response will be noted for further investigation. Beta scanning coverage for wall and ceiling surfaces will be 100% of Class 1 surfaces and a minimum of 25% for Class 2 and 10% for Class 3 surfaces.

Surface activity measurements will be performed at the systematic and judgmental locations (see Sections 4.6. and 4.7); 1-minute static measurements will be conducted using a Ludlum Model 43-68 gas proportional detector coupled with a Ludlum Model 2221 ratemeter/scaler.

Smears for removable activity will be performed at locations of direct activity measurements.

4.9 Sample Analyses

Smears will be analyzed for gross alpha and gross beta activity in the on-site counting facility.

4.10 Investigation

If measurements, sample analyses, or statistical data evaluation identifies residual radioactivity exceeding 50% of the release criteria, the source of the contamination will be investigated. Remediation will be performed, as necessary. The survey unit will be reclassified in accordance with the Master FSS Plan and FSS activities repeated, utilizing a newly determined sampling pattern and random start point.

5. Data Evaluation

Measurements will be compared with release criteria using the WRS test or Sign test, depending on the particular survey unit design. Results will be compared with the critical value for the appropriate number of samples and decision errors. Judgmental measurements and measurements from non-survey unit surfaces will be individually compared with DCGL.

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Final Status Survey Plan

Addendum 005: Exterior Soil and Paved Surfaces

Revision 0

Prepared for
University of Virginia
Reactor Facility Decommissioning Project

Prepared by

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June 2003

Final Status Survey Plan

Addendum 005: Exterior Soil and Paved Surfaces

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June 2003

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Project Manager — Date: 6-12-03

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Contents

FIGURES

1. Introduction

The Master Final Status Survey Plan (UVA-FS-002) identifies the Data Quality Objectives for Final Status Survey (FSS) activities, together with the underlying technical assumptions, approaches, and methodologies for designing, implementing, and evaluating a FSS on each impacted area of the University of Virginia Research Reactor (UVAR) Facility. A separate survey area-specific addendum is prepared for each area or group of areas with common media, contaminants, and other characteristics, prior to beginning FSS; FSS for the specific area or group of areas will then be performed in accordance with that addendum. This addendum (Addendum 005) applies to the open land and paved exterior surfaces surrounding the UVAR Facility building.

2. Description

The UVAR Facility is located on Old Reservoir Road, approximately 0.6 kilometers (km) west of the Charlottesville, VA city limits. The Facility includes the 1190 m^2 UVAR building, a small (~1450 m^2) pond, and asphalt-paved roads, parking areas, and equipment/materials storage pads, situated on a land area of approximately a 9500 m^2 (see Figure 2-1). The site terrain generally slopes from north to south. The east and south portions of the site are wooded; the northern portion of the site surface is dominated by rock outcroppings. A low (~1 m high) fence encompasses the site.

During Facility operations, there were two underground liquid-waste collection tanks, located southeast of the building, near the edge of the pond. These tanks have been removed and the excavation has been surveyed in accordance with FSS Addendum 001. The pond received site runoff and some facility liquid releases during operations. This pond has been drained and sediments were surveyed in accordance with FSS Addendum 003. The remainder of the exterior property is addressed by this Addendum.

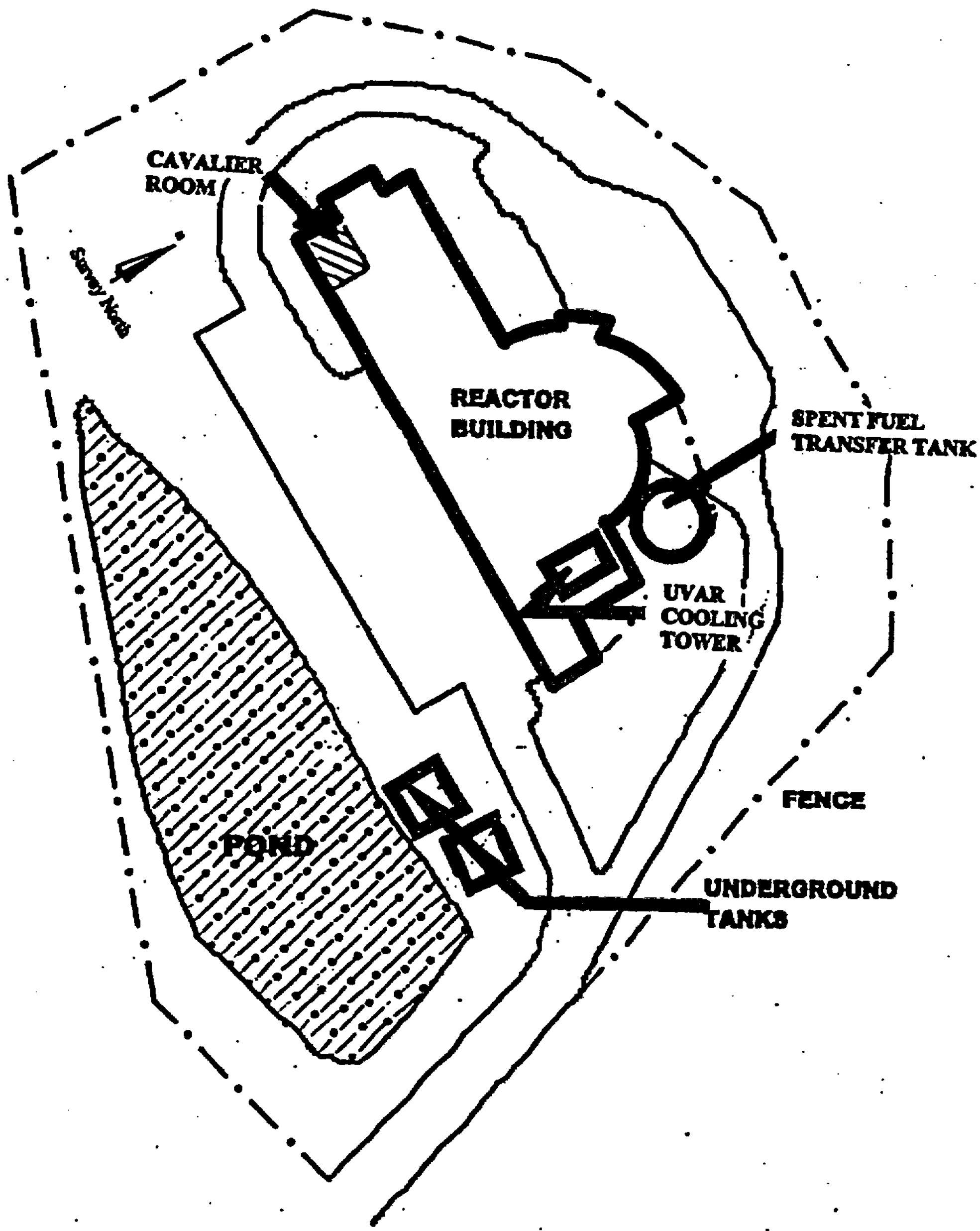

Figure 2-1 University of Virginia Reactor Facility and Environs

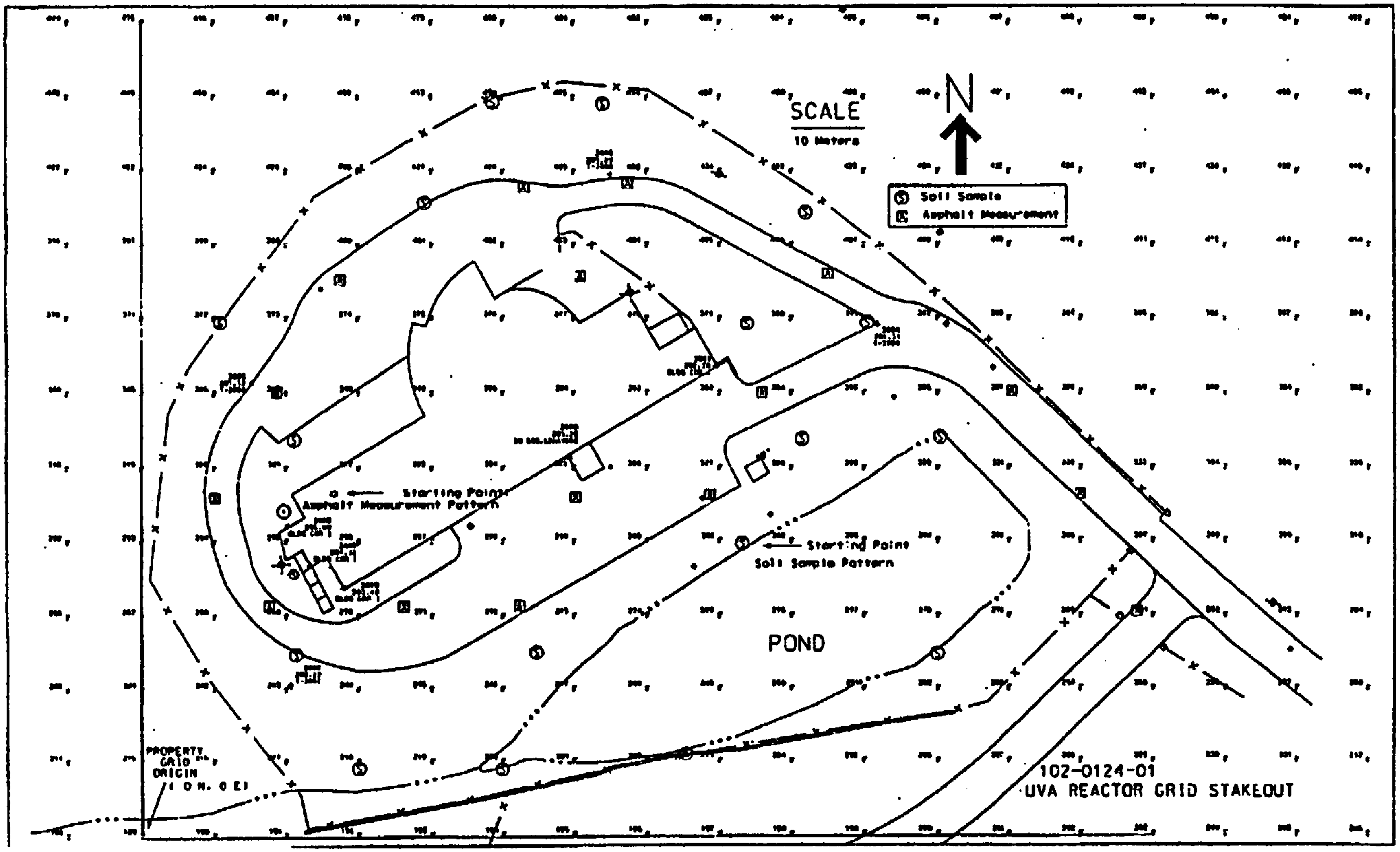

Figure 2-2 Plot of Site, Indicating Reference Grid System, and Measurement/ Sampling Locations.

3. Contaminants of Concern and Guidelines

During facility operation, several small spills of contaminated liquids occurred in the vicinity of the waste collection systems. Equipment, materials, and wastes with a potential for low-level contamination were stored on surfaces south of the building during facility operations and in connection with the facility remediation. In addition, several liquid discharge points from the building to the pond terminate on the hillside north of the pond. Initial characterization by GTS Duratek and follow-on monitoring during the decommissioning actions included scanning and sampling of potentially affected surfaces. Cs-137 was identified in surface soil, but at concentrations typical of background soil. Surveys of pond sediments and waste tank excavations also identified Cs-137 as the dominant contaminant from facility operations; smaller quantities of Co-60 were present in a few of the samples. Significant levels of other site-related radionuclides were not identified by this monitoring.

Release levels for site are the NRC default screening criteria; the default screening criteria concentrations for Cs-137 and Co-60 are 11 pCi/g and 3.8 pCi/g, respectively. Default screening criteria surface activity is 28,000 dpm/100 cm^2 for Cs-137 and 7100 dpm/100 cm^2 for Co-60. For the final status survey, soils will be analyzed for specific potential radionuclide contaminants, and results must satisfy the Unity Rule for the sum of ratios of radionuclide concentrations present to respective screening default guideline levels. The more restrictive Co-60 criterion will be used to demonstrate compliance for contamination on paved surfaces.

4. Survey Approach

4.1 Survey Reference System

A 10-meter grid has been established over the entire site. This grid has been referenced to the federal planar coordinate system. Figure 2-2 indicates the reference grid system. Further grid identification (e.g., northing and easting from a southwest origin point) will be assigned to each node to facilitate location of sampling/measurement points.

4.2 Survey Classification

Based on the facility use history and characterization and remediation control monitoring, the exterior soil and paved surfaces sediments are designated Class 3 for FSS planning and implementation purposes.

4.3 Survey Unit Identification

For survey design purposes the planning area of the total site (excluding the pond and building footprint) is 6860 m^2. The site is comprised of two survey units; one is the paved surfaces of approximately 2500 m^2, and the other is the soil surfaces of approximately 4360 m^2.

4.4 Demonstrating Compliance with Release Guidelines

Compliance with decommissioning requirements will be demonstrated by comparing the results of final status survey measurements and sample analyses with default screening criteria. For soil samples, the sum of ratios of identified radionuclides will be used; the sum of ratios must satisfy the Unity Rule. Because the radionuclides identified as potential contaminants are not present in background at concentrations, which are significant fractions of the release guidelines, correction of FSS sample data for background levels will not be required. Statistical testing of results will utilize the Sign Test to reject or accept the null hypothesis that the residual contamination exceeds the release criteria. Decision errors will be 0.05 (Type 1 and Type 2).

For activity measurements on paved surfaces, for planning purposes Co-60 is assumed to be the contaminant of interest, because it is the most restrictive of anticipated site contaminants. Measurements will be compared with the Co-60 default screening criteria of 7100 dpm/100 cm^2. Because the direct measurement procedure includes inherent detector background contributions, correction of FSS measurement data for background levels will be required, through use of an appropriate reference area of similar paving material in a non-impacted location. Statistical testing of results will utilize the WRS Test to reject or accept the null hypothesis that the residual contamination exceeds the release criteria. Decision errors will be 0.05 (Type 1 and Type 2).

4.5 Number of Required Data Points

Soil (Samples)

The following calculation is based on use of Unity Rule (i.e., DCGL = 1).

DCGL = 1, LBGR = 0.5 DCGL (Refer to Master FSS Plan, Section 7.8)
Δ = DCGL - LBGR = 0.5
σ = 0.25 (based on the sum of ratios of maximum levels of radionuclides detected in characterization samples to respective DCGL's)
$\Delta/\sigma = 2$
From MARSSIM Table 5.5, N = 15; i.e., 15 data points required for the survey unit for Sign Test.

Paved Surfaces (Activity Measurements)

The following calculation is based on Co-60 as the contaminant (i.e., DCGL = 7100).

DCGL = 7100, LBGR = 0.5 DCGL (Refer to Master FSS Plan, Section 7.8)
Δ = DCGL - LBGR = 3550
σ = 1200 (based on the MDA for the 43-68 gas flow proportional detector)
$\Delta/\sigma = 2.96$
From MARSSIM Table 5.3, N/2 = 10; i.e., 10 data points each required for the survey unit and the reference area for WRS Test.

4.6 Sampling Pattern

Sampling/measurements will be performed at systematically spaced intervals on triangular patterns throughout the soil and paved areas. The spacing between data points for soil is:

$L = [4360/(0.866 \times 15)]^{0.5} = 18.3$ m (A spacing of 18 m between samples and 15 m between E/W sampling lines will be used for ease of field implementation).

The spacing between data points for paved surfaces is:

$L = [2500/(0.866 \times 10]^{0.5} = 16.99$ m (A spacing of 17 m between samples and 14 m between E/W sampling lines will be used for ease of field implementation).

Random start points for the systematic sampling patterns have been selected, using overall survey unit dimensions of 100 m x 140 m and random numbers from the MARSSIM random number table. The resulting start point for soil area survey is 39 N, 84 E; the start point for paved surface survey is 46 N, 27 E (refer to Figure 2-2).

Based on these parameters the following systematic sampling/measurement data points have been identified (refer to Figure 2-2):

Soil Surfaces	Paved Surfaces
9 N, 30 E	32 N, 18.5 E
9 N, 48 E	32 N, 35.5 E
9 N, 66 E	32 N, 52.5 E
24 N, 21 E	32 N, 137.5 E
24 N, 57 E	46 N, 10 E
24 N, 111 E	46 N, 61 E
39 N, 84 E	46 N, 78 E
54 N, 21 E	46 N, 129 E
54 N, 93 E	60 N, 18.5 E
54 N, 111 E	60 N, 86.5 E
69 N, 12 E	60 N, 120.5 E
69 N, 84 E	74 N, 27 E
69 N, 102 E	74 N, 61 E
84 N, 39 E	74 N, 95 E
84 N, 93 E	88 N, 52.5 E
99 N, 48 E	88 N, 69.5 E
99 N, 66 E	

In addition to the systematic sample/measurement locations, samples and measurements will be obtained at "biased" locations, identified by scanning or professional judgment of the FSS field supervisor as having the greatest potential for contamination. Examples of such locations include egress points from the reactor room on the upper level, egress points from the beamport/experimental facilities on the ground level, collection areas for natural drainage pathways, and drainage outfalls to the pond hillside.

4.7 Survey Methods

Gamma walkover surface scans will be performed using a 2"X 2" NaI detector (Ludlum Model 44-10) coupled with a Ludlum Model 2221 ratemeter/scaler. The detector will be maintained within 5-10 cm of the surface and moved from side to side in a serpentine pattern while noting any indication of audible elevated count rate, which might indicate the presence of radioactive contamination. Results (count rate) will be documented on survey area maps. Locations of elevated response will be noted for further investigation. Gamma

scanning coverage will be a minimum of 50% of the soil and paved surfaces. Beta scans of paved surfaces will be performed using a large area (~580 cm^2) gas proportional detector (Ludlum Model 43-37) coupled with a Ludlum Model 2221 ratemeter/scaler and Model 239-1 floor monitor. The detector will be maintained within ~1 cm of the surface while advancing the detector at a rate of approximately one detector width per second. Audible response will be monitored for indication of elevated count rate, which might indicate the presence of radioactive contamination. Results (count rate) will be documented on survey area maps. Locations of elevated response will be noted for further investigation. Beta scanning coverage will be a minimum of 50% of the paved surfaces.

Surface (0 to 15 cm) soil samples of approximately 500 g will be collected at the systematic and 4 to 6 judgmental sampling locations (see Sections 4.5. and 4.6). Surface activity measurements will be performed at the systematic and 4-6 judgmental locations (see Sections 4.5. and 4.6); 1-minute static measurements will be conducted using a Ludlum Model 43-68 gas proportional detector coupled with a Ludlum Model 2221 ratemeter/scaler. Samples will be assigned unique identification numbers and a chain of custody record and analytical request will be prepared.

4.8 Sample Analyses

Soil samples will be analyzed by an off-site commercial laboratory for gamma emitters (gamma spectrometry) spectrometry and for hard-to-detect (10 CFR Part 61) radionuclides.

4.9 Investigation

If measurements, sample analyses, or statistical data evaluation identifies residual radioactivity concentrations exceeding 50% of the release criteria, the source of the contamination will be investigated. If results are confirmed, remediation will be performed, as necessary, the impacted areas reclassified, and FSS activities repeated, utilizing a newly determined sampling pattern and random start point.

5. Data Evaluation

For soil samples, the sum of ratios will be calculated for radionuclides, which are of facility origin. The Sign Test will be performed for systematic samples and the results compared with the critical value for the appropriate number of samples and decision errors. Judgmental sample results will be individually compared with DCGL's using the Unity Rule.

Systematic paved surface measurements will be compared with the Co-60 DCGL of 7100 dpm/100 cm^2, using the WRS Test and results will be compared with the critical value for the appropriate number of samples and decision errors. Judgmental measurement results will be individually compared with the DCGL.

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Final Status Survey Plan

Addendum 006: Exterior Structure Surfaces

Prepared for
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Addendum 006: Exterior Structure Surfaces

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Contents

FIGURES

1. Introduction

The Master Final Status Survey Plan (UVA-FS-002) identifies the Data Quality Objectives for Final Status Survey (FSS) activities, together with the underlying technical assumptions, approaches, and methodologies for designing, implementing, and evaluating a FSS on each impacted area of the University of Virginia Research Reactor (UVAR) Facility. A separate survey area-specific addendum is prepared for each area or group of areas with common media, contaminants, and other characteristics, prior to beginning FSS; FSS for the specific area or group of areas will then be performed in accordance with that addendum. This addendum (Addendum 006) applies to the exterior surfaces of the UVAR Facility building. Other related Addenda include 004 (Interior Structure Surfaces) and 005 (Exterior Soil and Paved Areas).

2. Description

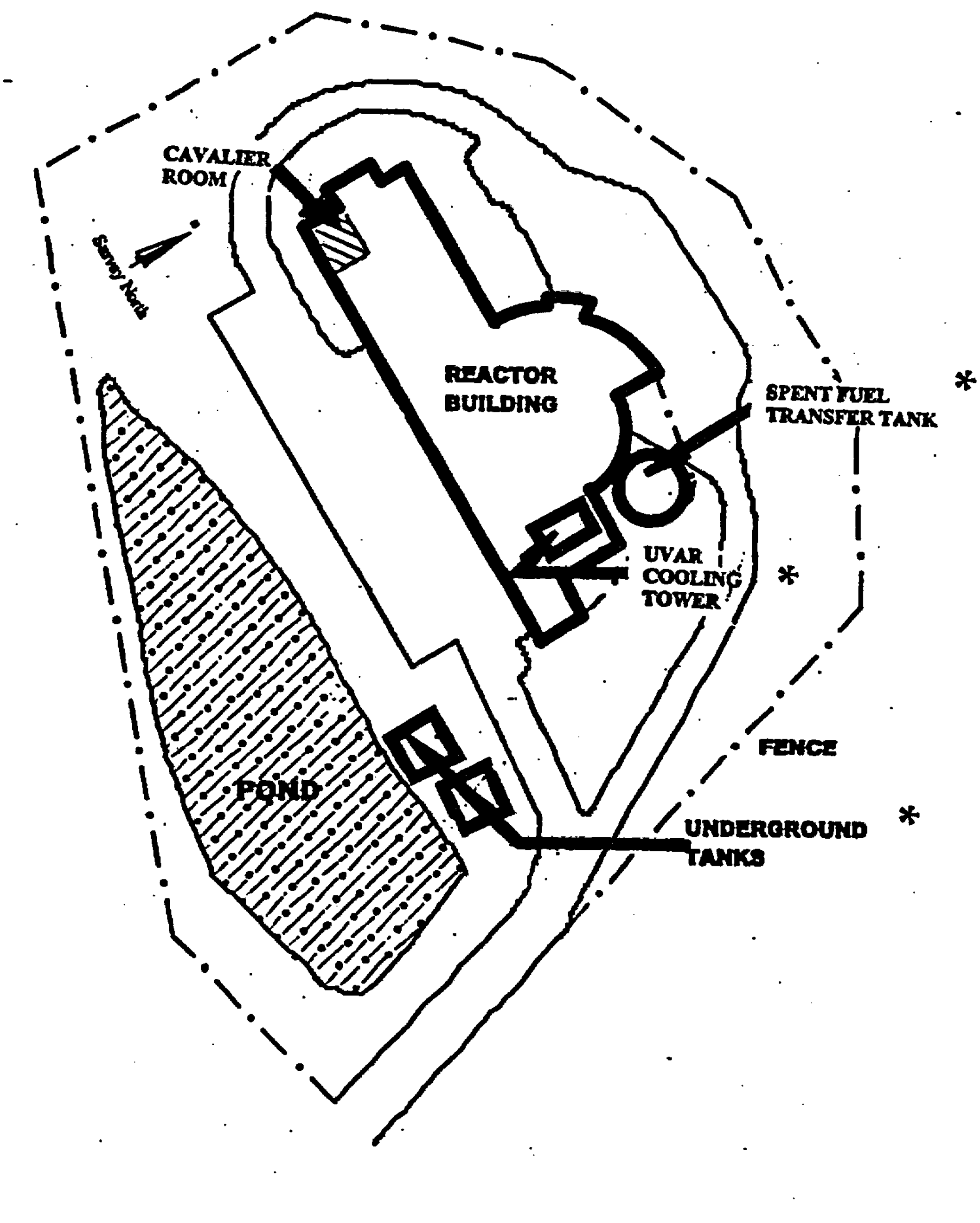
The UVAR Facility is located on Old Reservoir Road, approximately 0.6 kilometers (km) west of the Charlottesville, VA city limits. The Facility includes the UVAR building, a small pond, and paved roads, parking areas, and equipment/materials storage pads, situated on a land area of approximately 9500 m^2 (see Figure 2-1). The three-story building housed the UVA Research Reactor and the CAVALIER facility, as well as offices for the reactor staff and faculty and students of the Department of Nuclear Engineering, miscellaneous laboratories, and other support facilities for the reactors and Department of Nuclear Engineering.

Figure 2-2 is a plot plan of the UVAR building. The UVAR building is of concrete block construction with brick veneer. Floors are concrete slab. There is approximately 1190 m^2 of roof area, at two elevations; one covers the Reactor Confinement structure – a surface area of approximately 175 m^2, and the other (approximately 1015 m^2) covers the remainder of the structure. During operation there was a cooling tower on the roof to the southeast of the Reactor Room; this structure was removed during decommissioning. Roofs are of tar-and-gravel composition. The roofs are essentially clear of obstructions such as items of HVAC equipment. There are multiple sewer line vents and rainwater drains on the roofs.

Other exterior building surfaces of concern include discharge grills and stacks servicing small laboratory exhaust ventilation systems; some of these, e.g., those from rooms M005 and M008, were known to have at one time been contaminated. Doors at exits from areas handling radioactive and/or potentially contaminated materials are also surfaces of interest. These exterior locations are identified on figures 2-3 to 2-5.

In preparation for implementing the Final Status Survey, impacted reactor and support systems and components were removed and disposed of as radioactive waste or surveyed and released for use without radiological restrictions. Additional characterization surveys were performed to identify potentially contaminated surfaces; any such surfaces were removed or decontaminated.

Figure 2-1 University of Virginia Reactor Facility and Environs



*Removed during decommissioning

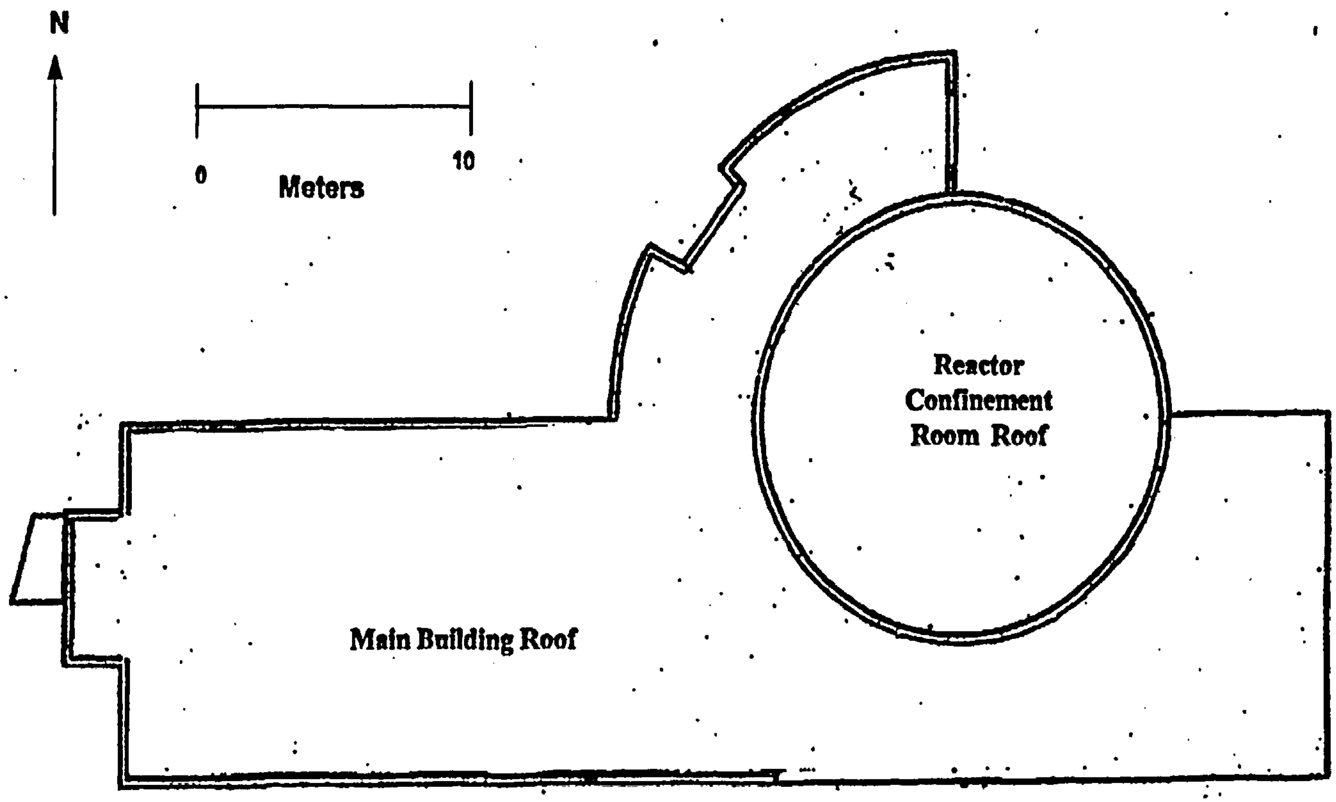
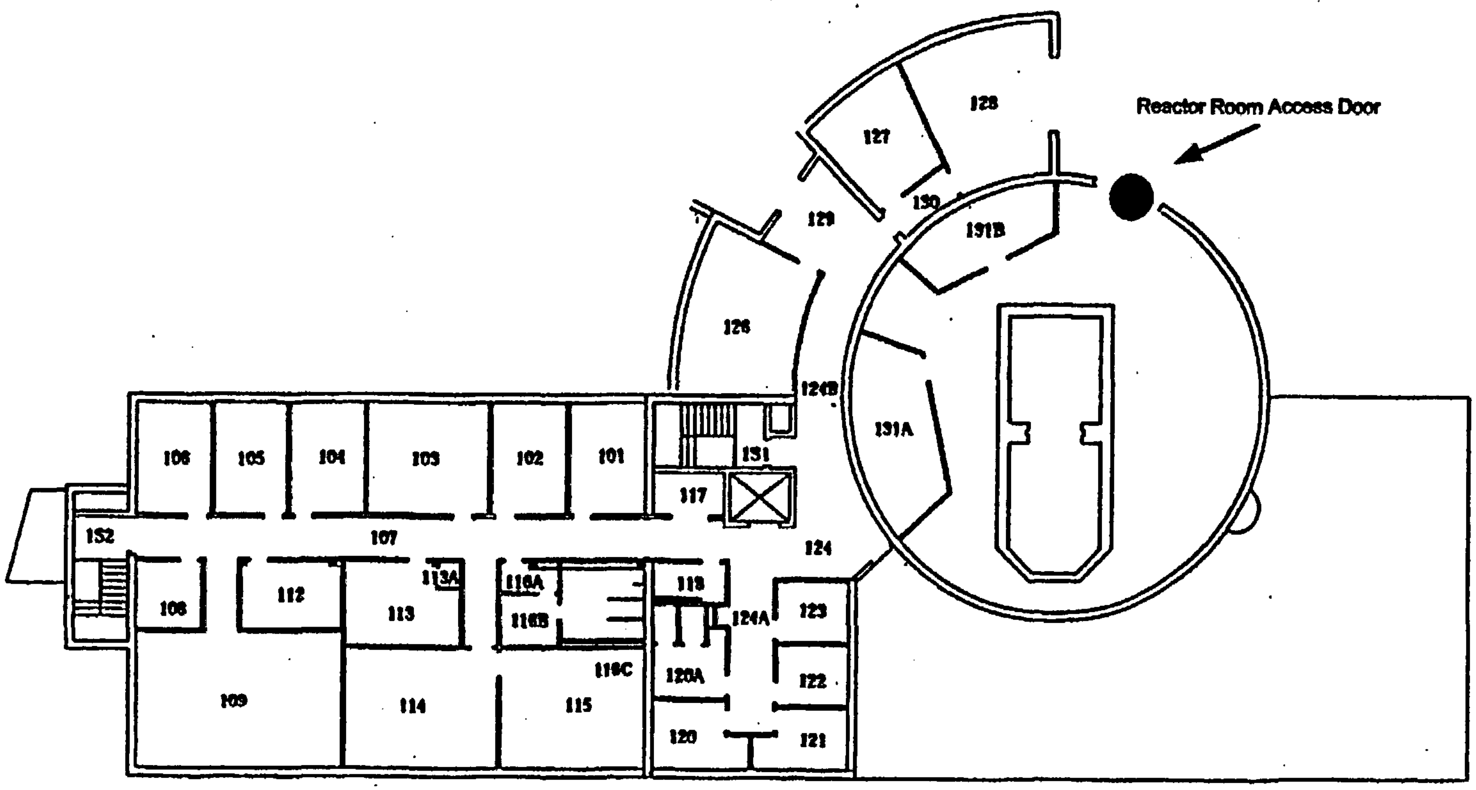


Figure 2-2 UVA Reactor Floor Plan View Indicating Roof Areas.



Figures 2-3 UVA Reactor First Floor Plan View Indicating Survey Locations on Exterior Building Surfaces.

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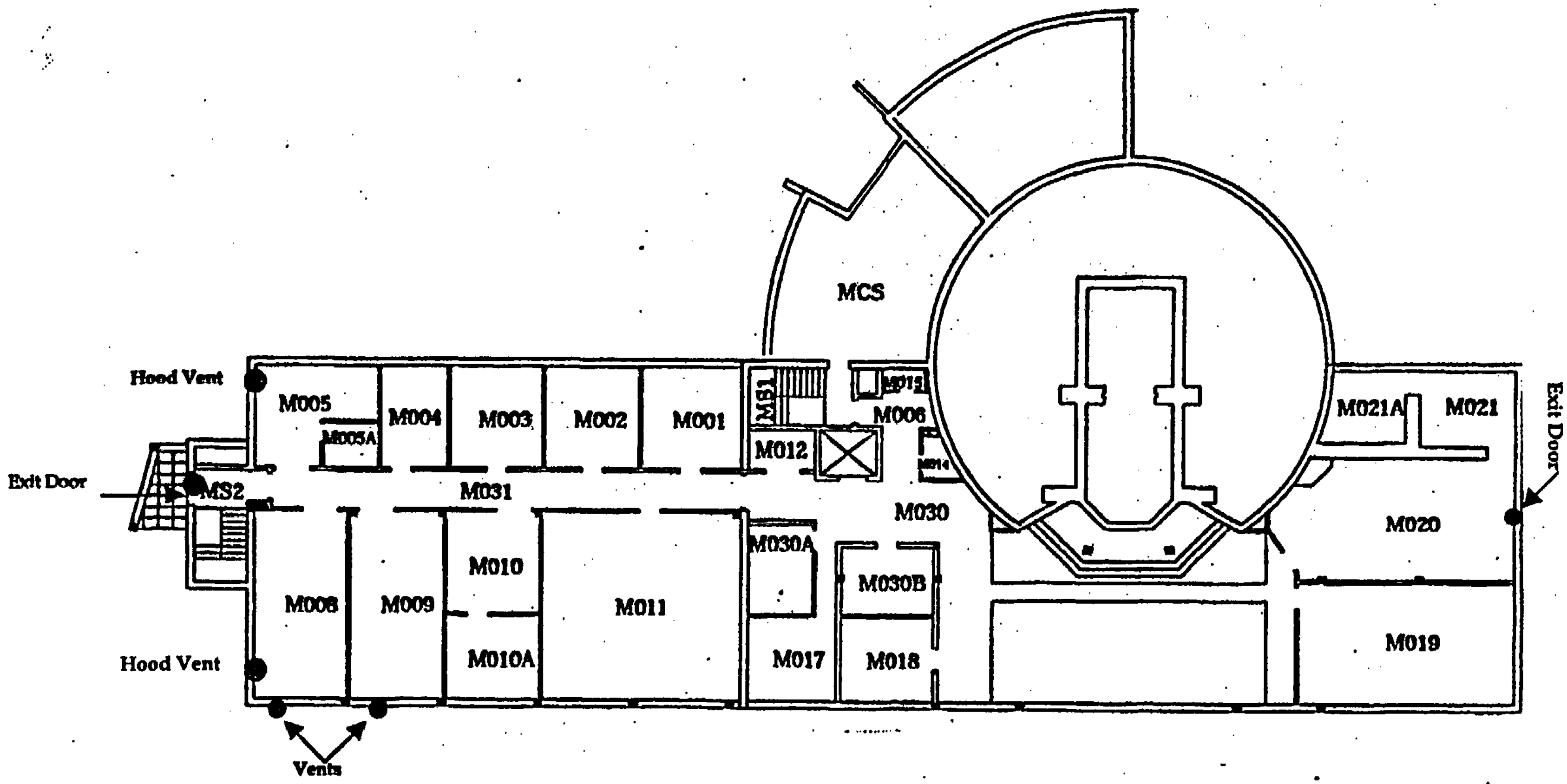

Figure 2-4 UVA Mezzanine Floor Plan View Indicating Survey Locations on Exterior Building Surfaces.

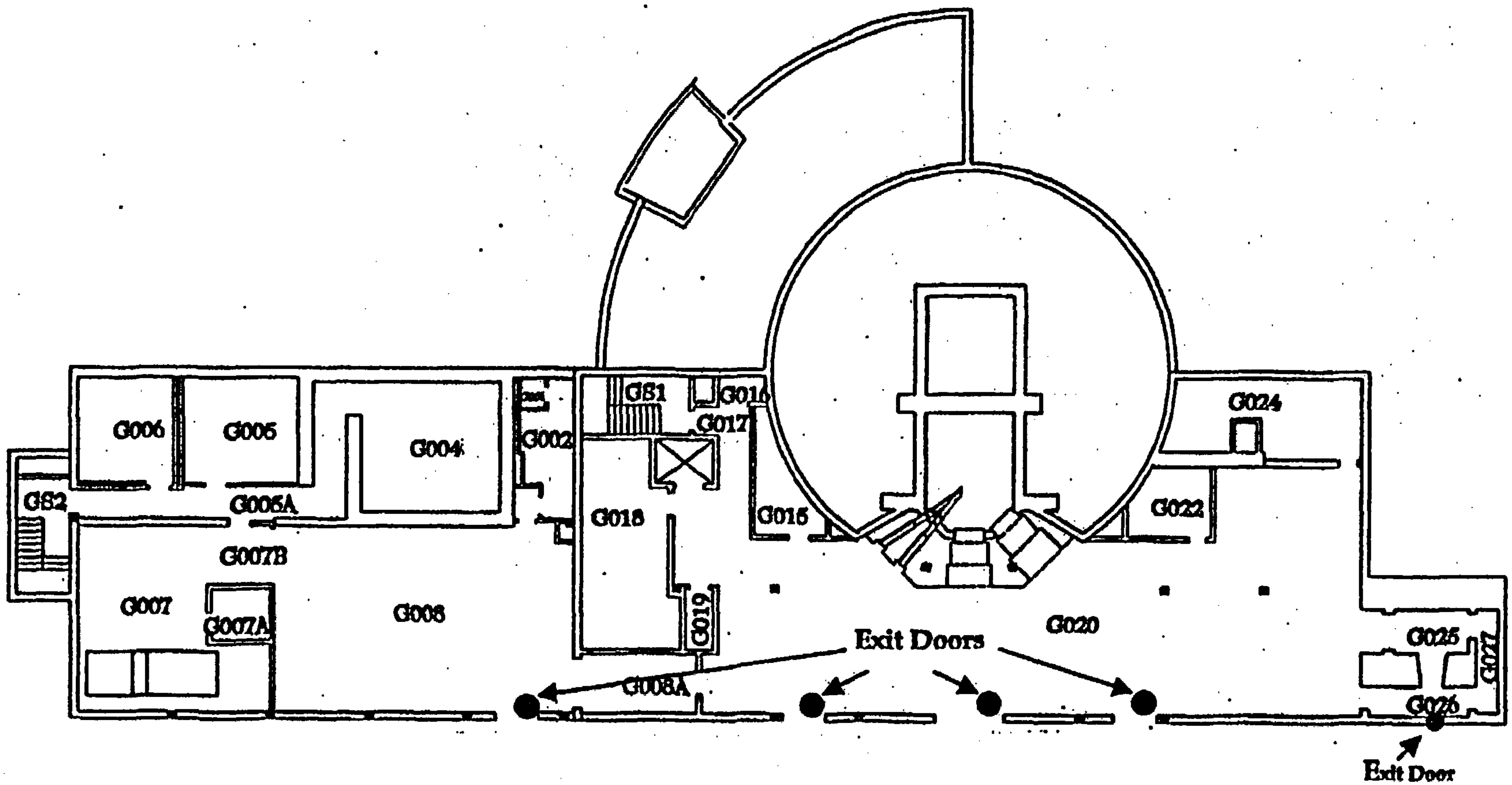

Figure 2-5 UVA Reactor Ground Floor Plan View Indicating Survey Locations on Exterior Building Surfaces.

3. Contaminants of Concern and Guidelines

The GTS Duratek initial characterization and continuing characterization by the CH2M HILL team showed that radiological contamination was generally low level and was limited to a small portion of the Building interior. The overall predominant radionuclides were Co-60 and Cs-137; smaller activities of fission and activation products, namely C-14, Fe-55, and Eu-152 were identified in some media. Ni-63 and Tc-99 contaminants were present on facility surfaces from research projects in labs M008 and M005, respectively. No significant levels of contamination have been identified on exterior building surfaces.

The Decommissioning Plan established the criteria for residual radioactive material contamination on UVAR facility surfaces. UVAR facility criteria also referred to as derived concentration guideline levels (DCGLs) are selected from the table of NRC default screening values in NUREG-1757. The screening DCGL of 7100 dpm/100 cm^2 for Co-60 is the most restrictive of the contaminants that have been identified as potential contaminants of concern. Unless there is specific evidence that contamination of a surface is comprised of radionuclides other than Co-60, the DCGL for Co-60 will be the basis for evaluating the final radiological status of the structure surfaces. Guidelines for removable structure contamination are 10% of the NRC screening default values for total surface activity, for Co-60 the removable activity limit is therefore 710 dpm/100 cm^2. This assures a conservative approach for satisfying the NRC dose-based criteria for future facility use. Appendix A of the Master FSS Plan describes the method for establishing guidelines for other radionuclides and combinations of radionuclides. If guidelines other than those for Co-60 are to be used, a justification will be prepared and included with the FSS documentation.

4. Survey Approach

4.1 Survey Reference System

A grid system will be established on roof surfaces to provide a means for referencing measurement and sampling locations. On Class 1 and 2 structure surfaces, a 1-meter interval grid will be established; a 5-meter interval grid will be established on Class 3 structure surfaces. Vertical and overhead surfaces may be referenced to the grid established for the area beneath. Grid systems will originate at the southwest corner of the survey unit, except where specific survey unit characteristics necessitate alternate grid origins. Grids are assigned alphanumeric indicators to enable survey location identification and are referenced to building features. Maps and plot plans of survey areas will include the grid system identifications. Systems and surfaces of less than 20 m^2 will not be gridded, but survey locations will be referenced to prominent facility features.

4.2 Survey Area Classification

The roofs (main building and Reactor Confinement Room) are designated MARSSIM Class 2 surfaces; other exterior surfaces are designated Class 3. Facility history (including the Historic Site Assessment) and radiological monitoring conducted during characterization and remedial activities are the bases for these classifications.

4.3 Survey Unit Identification

Based on the MARSSIM classification by contamination potential and the survey unit area limitations, suggested by MARSSIM, the following survey units have been identified:

- Reactor Confinement Room roof
- Main building roof

Impacted structure surfaces of ≤ 10 m^2 and impacted land surfaces of ≤ 100 m^2 will not be designated as survey units. Instead, a minimum of 4 measurements will be obtained from such areas, based on judgment, for comparison individually with the DCGLs. Such surfaces include exterior surfaces of vents, stacks, and doors, exiting from areas of former radioactive materials use and facilities requiring remedial action during this decommissioning project.

Classifications and survey unit boundaries may change, based on results as the FSS progresses; if classifications or boundaries change, the survey will be redesigned and the survey and data evaluation repeated.

4.4 Demonstrating Compliance with Release Guidelines

The Wilcoxon Rank Sum (WRS) test will be used for evaluating direct measurements of total surface activity, relative to the established criteria. The Null Hypothesis will be that activity levels in the survey unit exceed the release criteria. Rejection of the Null Hypothesis will be required to demonstrate that the release criteria are satisfied. Decision errors will be 0.05 (Type 1 and Type 2).

4.5 Background Reference Areas and Materials

For applications of the WRS test, a reference area of the same material as the surfaces being evaluated, but without a history of potential contamination by licensed operations, will be identified. The number of reference data points will be the same (+/- 20%) as the number of data points required from the survey unit. A set of reference measurements will be obtained for each instrument being used for survey area evaluation. Reference area and background requirements will be survey-area specific.

4.6 Number of Required Data Points

The following calculation is based on Co-60 as the contaminant (i.e., DCGL = 7100).

DCGL = 7100, LBGR = 0.5 DCGL (Refer to Master FSS Plan, Section 7.8)
Δ = DCGL - LBGR = 3550
σ = 1200 (based on the MDA for the 43-68 gas flow proportional detector)
Δ/σ = 2.96
From MARSSIM Table 5.3, N/2 = 10; i.e., 10 data points each required for the survey unit and the reference area (WRS test).

4.7 Sampling Pattern

Measurements will be performed at systematically spaced intervals on triangular patterns throughout the roof surfaces. The spacing between data points for the main building roof is:

$L = [1015/(0.866 \times 10)]^{0.5} = 10.8$ m (A spacing of 10 m between samples will be used for ease of field implementation).

The spacing between data points for the Reactor Confinement Room roof is:

$L = [175/(0.866 \times 10]^{0.5} = 4.5$ m (A spacing of 4 m between samples will be used for ease of field implementation).

Random start points for the systematic sampling patterns on the roofs have been selected, using survey unit dimensions and random numbers (N-0.501578, E-0.204221 and N-0.644294, E-0.821341) from the MARSSIM random number table.

In addition to the systematic locations, measurements will be obtained at "biased" roof locations, identified by scanning or professional judgment of the FSS field supervisor as having the greatest potential for contamination. On other building exterior surfaces,

measurements will be obtained at "biased" locations, identified by scanning or professional judgment of the FSS field supervisor as having the greatest potential for contamination.

4.8 Survey Methods

Gamma walkover surface scans will be performed using a 2"X 2" NaI detector (Ludlum Model 44-10) coupled with a Ludlum Model 2221 ratemeter/scaler. The detector will be maintained within 5 to 10 cm of the surface and moved from side to side in a serpentine pattern while noting any indication of audible elevated count rate, which might indicate the presence of radioactive contamination. Results (count rate) will be documented on survey area maps. Locations of elevated response will be noted for further investigation. Gamma scanning coverage will be 100% of Class 1 surfaces and a minimum of 25% for Class 2 and 10% for Class 3 floor surfaces. Where conditions allow, beta scans of roof surfaces will be performed using a large area (~580 cm^2) gas proportional detector (Ludlum Model 43-37) coupled with a Ludlum Model 2221 ratemeter/scaler and Model 239-1 floor monitor; smaller Ludlum Model 43-68 gas proportional or Ludlum Model 44-9 pancake GM detectors coupled with Ludlum Model 2221 ratemeter/scalers will be used, as required by accessibility limitations. Ludlum Model 43-68 gas proportional or Ludlum Model 44-9 pancake GM detectors coupled with Ludlum Model 2221 ratemeter/scalers will be used to scan other exterior structure surfaces. The detector will be maintained within ~1 cm of the surface while advancing the detector at a rate of approximately one detector width per second. Scan speed will be adjusted, as necessary, to assure detection sensitivities are less than 50% of the release criteria. Audible response will be monitored for indication of elevated count rate, which might indicate the presence of radioactive contamination. Results (count rate) will be documented on survey area maps. Locations of elevated response will be noted for further investigation. Beta scanning coverage for wall surfaces will be 100% of Class 1 surfaces and a minimum of 25% for Class 2 and 10% for Class 3 surfaces.

Surface activity measurements will be performed at the systematic and judgmental locations (see Sections 4.6. and 4.7); 1-minute static measurements will be conducted using a Ludlum Model 43-68 gas proportional detector coupled with a Ludlum Model 2221 ratemeter/scaler.

Smears for removable activity will be performed at locations of direct activity measurements.

4.9 Sample Analyses

Smears will be analyzed for gross alpha and gross beta activity in the on-site counting facility.

4.10 Investigation

If measurements, sample analyses, or statistical data evaluation identifies residual radioactivity exceeding 50% of the release criteria, the source of the contamination will be investigated. Remediation will be performed, as necessary. The survey unit will be reclassified in accordance with the Master FSS Plan and FSS activities repeated, utilizing a newly determined sampling pattern and random start point.

5. Data Evaluation

Measurements will be compared with release criteria using the WRS test. Results will be compared with the critical value for the appropriate number of samples and decision errors. Judgmental measurements and measurements from non-survey unit surfaces will be individually compared with DCGL.

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Final Status Survey Plan

Addendum 007: Special Soils Areas

Revision 0

Prepared for
University of Virginia
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Non-soil structure surfaces in the crawl space will be surveyed in the same manner as other building interior surfaces. Addendum 004: *Interior Structure Surfaces* describes the process for FSS of those surfaces.

2. Description

The UVAR Facility is located on Old Reservoir Road, approximately 0.6 kilometers (km) west of the Charlottesville, VA city limits. The Facility includes the UVAR building, a small pond, and roads, parking areas, and equipment/materials storage pads, situated on a land area of approximately 9500 m^2 (see Figure 2-1). The three-story UVAR building housed the UVA Research Reactor and the CAVALIER facility, as well as offices for the reactor staff and faculty and students of the Department of Nuclear Engineering, miscellaneous laboratories, and other support facilities for the reactors and Department of Nuclear Engineering.

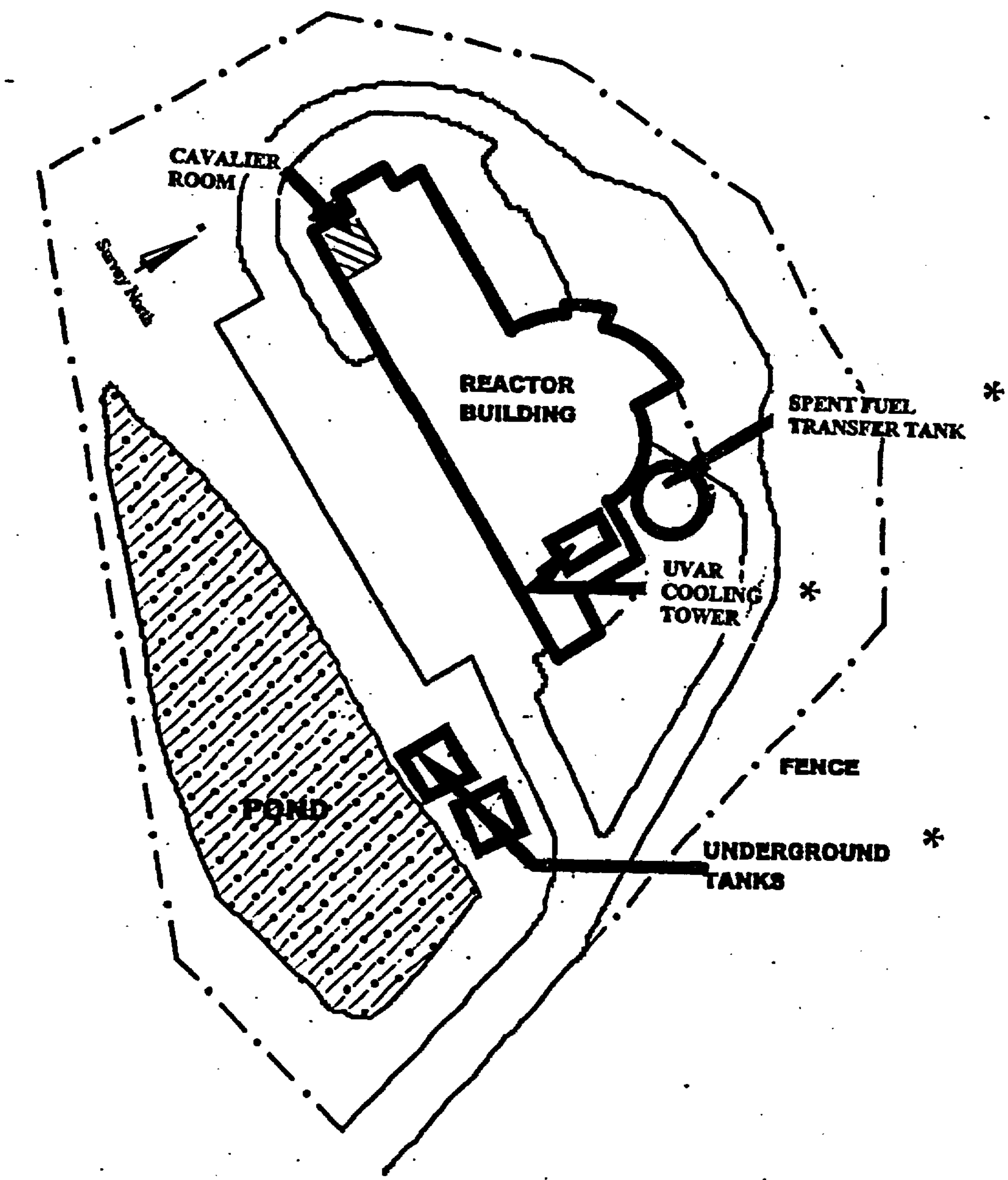
The UVAR building is of concrete block construction with brick veneer. Floors are concrete slab; the floors of facilities with heavy equipment use and/or higher potential for radioactive contamination, including the Reactor Room, ground level experimental (beam port) area, maintenance facilities, and workshops are painted or bare concrete. Internal walls are block and drywall. Most offices, hallways, and small laboratories have a dropped ceiling of acoustical tile, and tiled floors. There are also several soils areas inside the building, which have a potential for radioactive contamination, based on the operating history of the facility. One of these is a small crawl space adjacent to the Reactor Confinement Room. This space, located between the first and Mezzanine levels, is accessed from the stairwell between these two floors. The crawl space is of masonry construction with a dirt (soil) floor covering an area of approximately 50 m^2. This crawl space was used for storage of equipment, materials, and supplies, including some radioactive sources and potentially contaminated components and miscellaneous materials. Characterization surveys of this crawl space identified slightly elevated direct radiation levels, due to the masonry construction and the presence of elevated radon/thoron progeny, which is believed to originate from naturally occurring radionuclides in the soil floor and accumulate in this unventilated space.

The soil surrounding the reactor pool is another area of potential soil contamination. The reactor pool is approximately 10 m x 3.6 m and extends approximately 7.5 m below the reactor room floor level. The reactor pool is located inside the circular Reactor Confinement structure, which has a diameter of approximately 16 m. The space between the outer pool walls and the Confinement structure contains soil fill. Since the base of the Confinement structure does not incorporate a floor, the pool therefore is underlain with soil and bedrock. During reactor operations, small losses of pool water were a common occurrence. Specific locations of any pool leakage have not been identified; however, such leakage potentially could have resulted in contamination of soils around and beneath the pool. Breaks in piping beneath the Reactor Room floor were identified during facility remediation. Leakage of contaminated liquids from floor, sink, and pool overflow drains could have contaminated surface soils in the vicinity of these breaks. Characterization of surface and subsurface soils beneath the Reactor Room floor identified small, localized areas of contaminated surface soil; these areas were remediated. Characterization of the fill around the pool and in the soil, bedrock, and groundwater beneath the pool did not identify contamination of these media requiring remediation.

Figures 2-2 to 2-4 indicate the locations of these soils areas inside the UVAR building.

In preparation for implementing the Final Status Surveys, materials and equipment were removed from the crawl space and piping and other potentially contaminated items and components were removed from the fill area beneath the Reactor Room floor and around the reactor pool.

Figure 2-1 University of Virginia Reactor Facility and Environs



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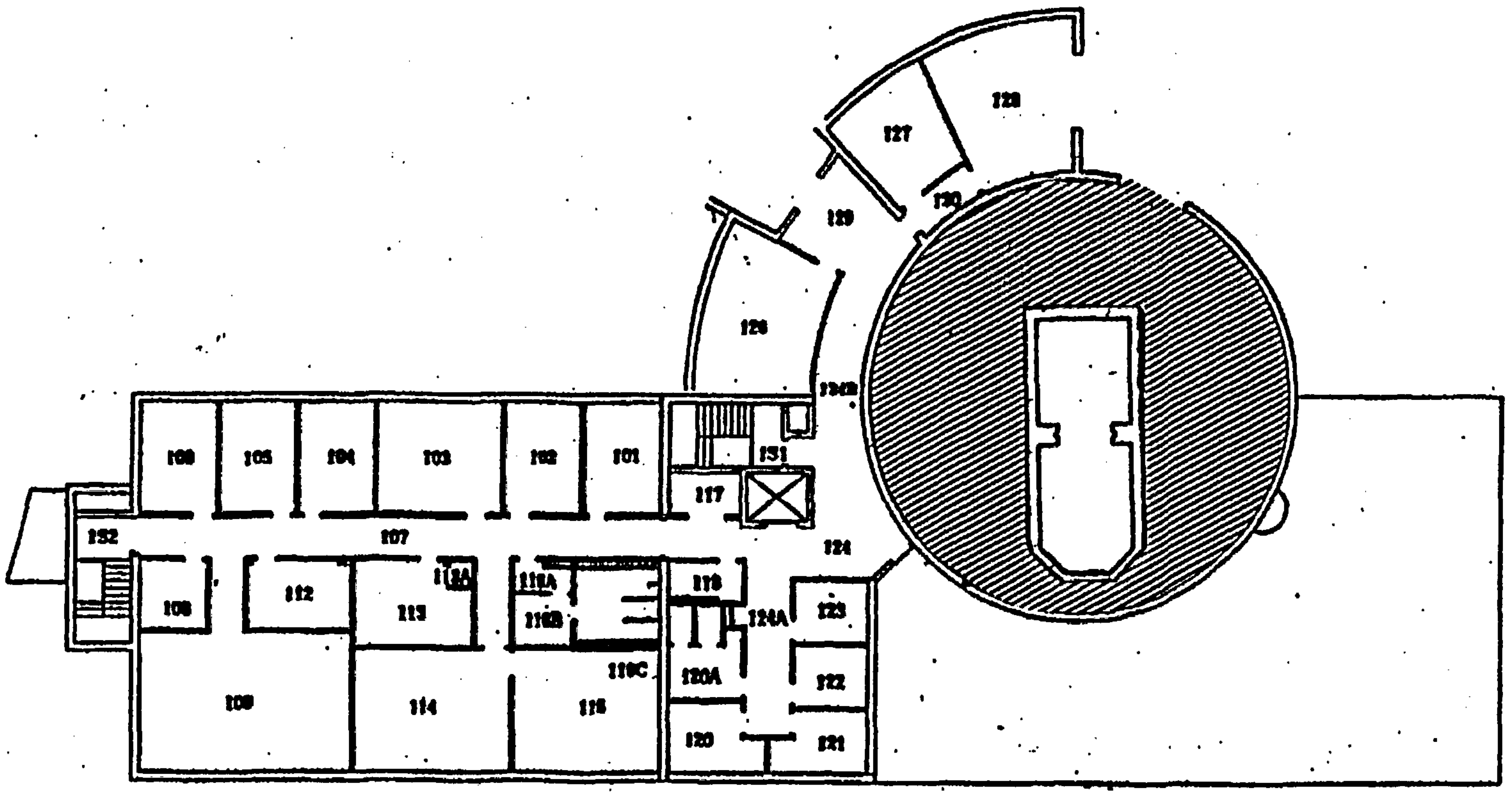

Figure 2-2 UVA Reactor First Floor Plan View Indicating the Location of Soil Fill Around Reactor Pool.

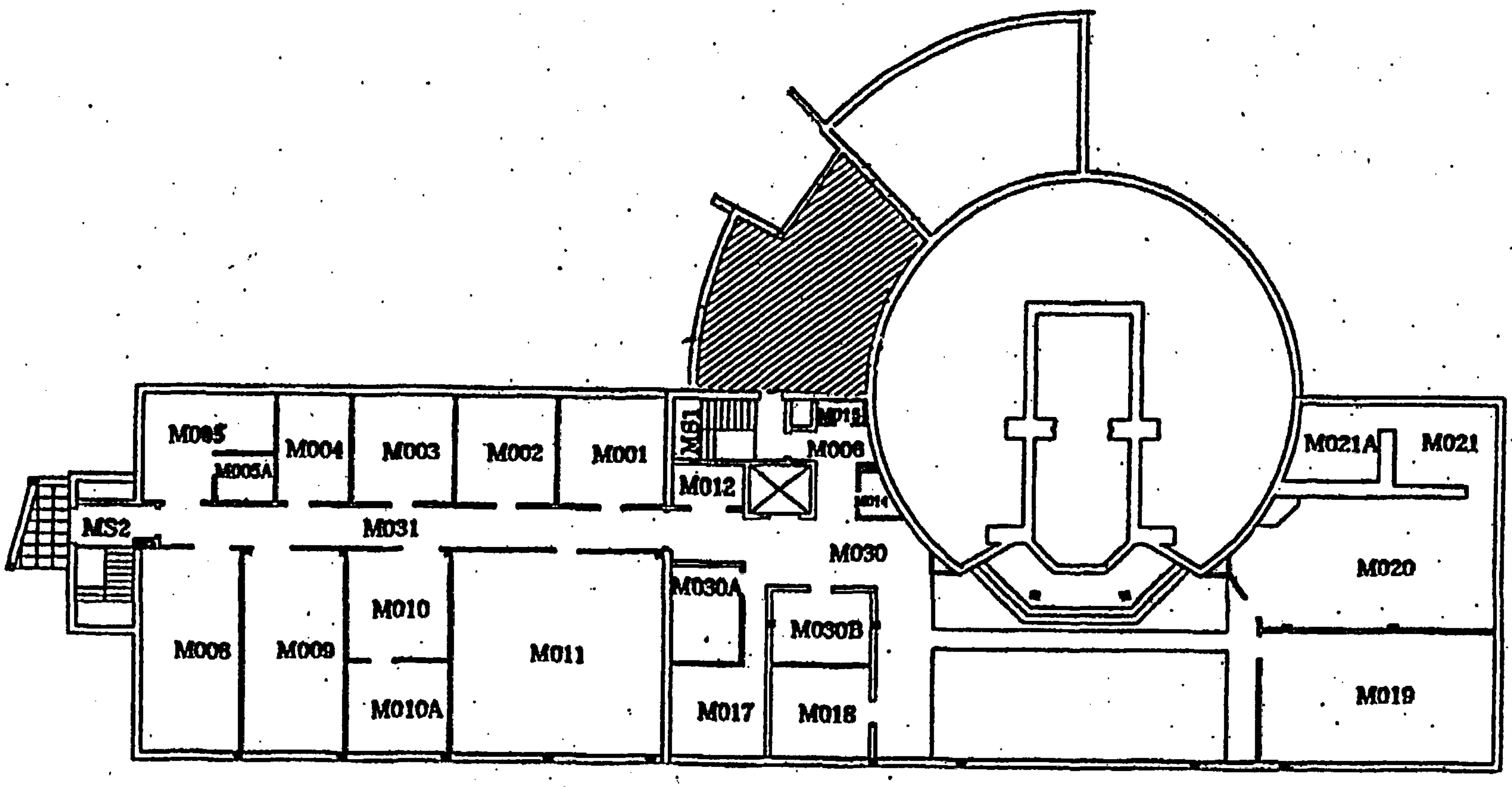

Figure 2-3 UVA Mezzanine Floor Plan View Indicating the Location of the Mezzanine Crawl Space.

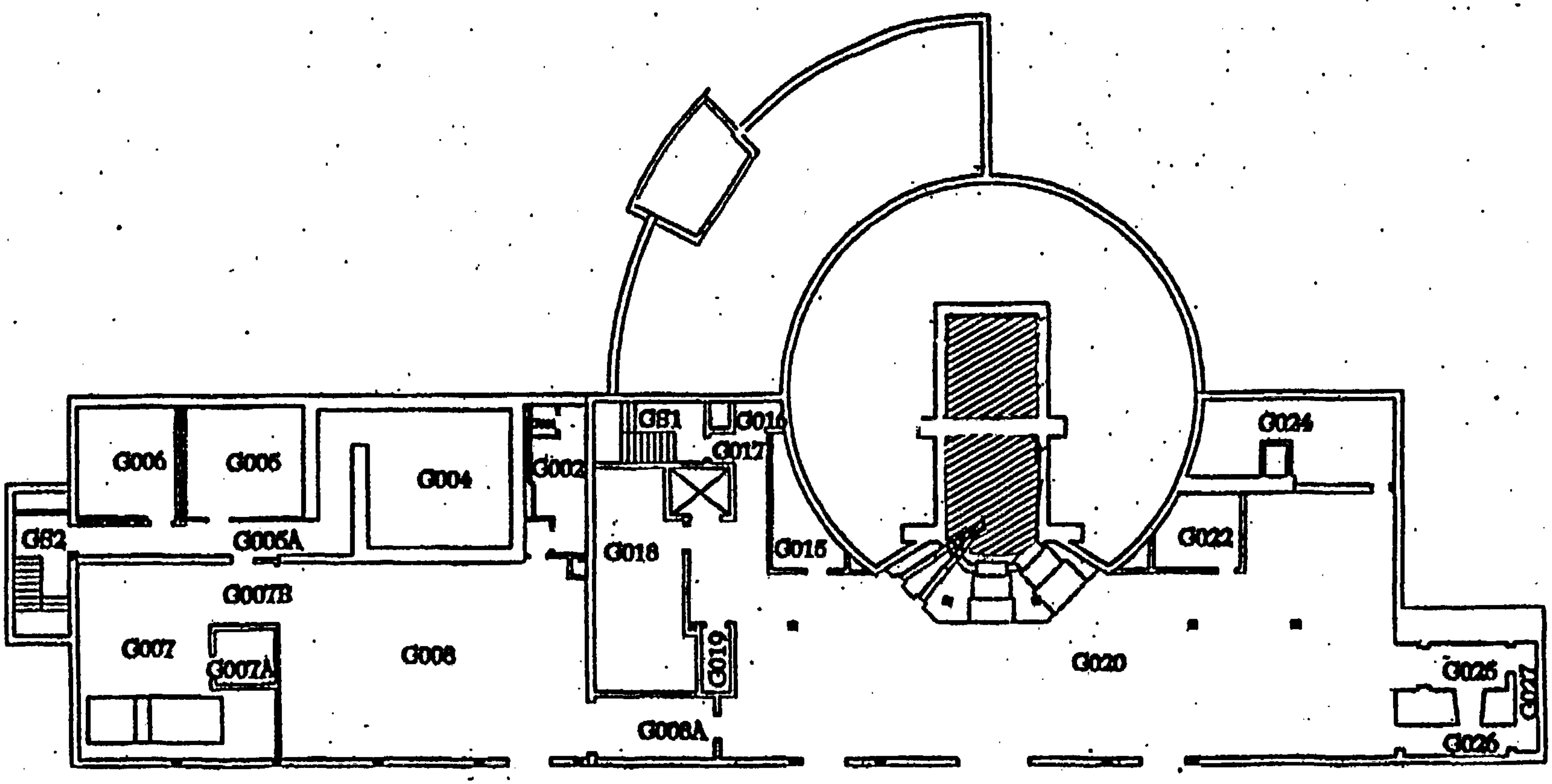

Figure 2-4 UVA Reactor Ground Floor Plan View Indicating the Location of the Soils Beneath the Reactor Pool.

3. Contaminants of Concern and Guidelines

The GTS Duratek initial characterization and continuing characterization by the CH2M HILL team showed that radiological contamination was generally low level and was limited to a small portion of the structure interior. Major structural contamination was generally limited to surfaces exposed to or in contact with reactor coolant, reactor neutron fields, and materials containing high levels of activity (e.g., the Hot Cell). Depending on the mechanism of contamination and the medium, radionuclides and their relative ratios varied. The overall predominant radionuclides were Co-60 and Cs-137; smaller activities of fission and activation products, namely C-14, Fe-55, and Eu-152 were identified in some media.

The Decommissioning Plan established the criteria for residual radioactive material contamination on UVAR facility surfaces. UVAR facility criteria, also referred to as derived concentration guideline levels (DCGLs), are selected from the table of NRC default screening values in NUREG-1757. The screening DCGLs for the predominant contaminants are 3.8 pCi/g for Co-60 and 11 pCi/g for Cs-137. These DCGLs will be applicable for all soil media, including surface, fill, and subsurface soils. For the final status survey, soils from the facility will be analyzed for specific potential radionuclide contaminants, and results must satisfy the Unity Rule for the sum of ratios of radionuclide concentrations present to respective screening default guideline levels.

4. Survey Approach

4.1 Survey Reference System

A 1-meter interval grid system will be established on surfaces to provide a means for referencing measurement and sampling locations. Grid systems will originate at the southwest corner of the survey unit, except where specific survey unit characteristics necessitate alternate grid origins. Grids are assigned alphanumeric indicators to enable survey location identification and are referenced to building features. Maps and plot plans of survey areas will include the grid system identifications.

4.2 Survey Area Classification

Based on facility operating history, characterization survey results, and findings during remediation, the crawl space is designated MARSSIM Class 2 contamination potential, and the soil areas around and beneath the reactor pool is designated Class 1 for survey planning purposes.

4.3 Survey Unit Identification

Based on MARSSIM classification (Class 1 and Class 2) and the recommended survey unit area limitations, suggested by MARSSIM, the interior soils would be surveyed as three separate areas. These are:

1) Mezzanine crawl space floor
2) fill around the reactor pool
3) soil beneath the reactor pool

Because of their small surface areas and location (inside the building), and inclusion of subsurface material, the FSS will not follow traditional MARSSIM approaches; however, the survey frequency and data analyses and evaluation will be consistent with the intent of MARSSIM.

4.4 Demonstrating Compliance with Release Guidelines

Compliance with decommissioning requirements will be demonstrated by comparing the results of final status survey sample analyses with default screening criteria, using the sum of ratios; the sum of ratios must satisfy the Unity Rule. Because the radionuclides identified as potential contaminants are not present in background at concentrations, which are significant fractions of the release guidelines, correction of FSS sample data for background levels will not be required. Statistical testing of results will utilize the Sign Test to reject or

accept the null hypothesis that the residual contamination exceeds the release criteria. Decision errors will be 0.05 (Type 1 and Type 2).

4.5 Number of Required Data Points

The following calculation is based on use of Unity Rule (i.e., DCGL = 1).

DCGL = 1, LBGR = 0.5 DCGL (Refer to Master FSS Plan, Section 7.8)
Δ = DCGL - LBGR = 0.5
σ = 0.25 (based on the sum of ratios of maximum levels of radionuclides detected in characterization samples to respective DCGL's)
$\Delta/\sigma = 2$
From MARSSIM Table 5.5, N = 15; i.e., 15 data points required for statistical evaluation of a survey unit using the Sign Test.

4.6 Sampling Pattern

Sampling/measurements will be performed at systematically spaced intervals throughout the soil areas or volumes of interest. The spacing between data points will be determined by the surface area or volume.

For the small crawl space area, samples of surface soil will be obtained on the same pattern and at the same intervals (about 3.5 m) as the surface activity measurement data points on the non-soil surfaces of this area. Thus 5 samples are expected from this soils area. Each sample will be individually compared (Unity Rule) with the release criteria.

The sampling pattern for the fill and soils around and beneath the reactor pool will be determined on the basis of soil surface area and volume. The surface area of soil beneath the Reactor Room floor is approximately 140 m^2, and the pool fill volume is assumed to occupy the entire space between the pool and outer room walls to the depth of the pool; the resulting volume is approximately 1000 m^3. Fifteen surface samples from the top of the fill under the reactor room floor will be obtained on a triangular (as conditions and accessibility permit) pattern at 3 m intervals. In addition, 15 subsurface samples (1 at each surface sampling location, at a randomly selected depth of 1 m to 7 m) will be obtained.

Sample borings will be performed through the pool bottom at a minimum of 4 existing locations, into the soils beneath the reactor pool. Four samples will be obtained from each of the access locations at depths extending to bedrock "refusal" for a total of 16 samples. Depths at each location will be obtained at the surface immediately beneath the pool, at the maximum depth obtained, and at two equally spaced depths between the pool and the maximum level.

In addition to the systematic locations, samples will be obtained at "biased" locations, identified by scanning or professional judgment of the FSS field supervisor as having the greatest potential for contamination. Examples of such locations include locations, which required remediation to reduce or remove residual activity, and locations of elevated direct radiation, identified during the FSS.

4.7 Survey Methods

Gamma scans of accessible surfaces will be performed using a 2"X 2" NaI detector (Ludlum Model 44-10) coupled with a Ludlum Model 2221 ratemeter/scaler. The detector will be maintained within 5 to 10 cm of the surface and moved across the surface while noting any indication of audible elevated count rate, which might indicate the presence of radioactive contamination. Results (count rate) will be documented on survey area maps. Locations of elevated response will be noted for further investigation. Gamma scanning coverage will be 100% of accessible surface soil surfaces. Gamma logs of boreholes for subsurface sampling will be conducted using the NaI detector (Ludlum Model 44-10 or Ludlum Model 44-2) coupled with a Ludlum Model 2221 ratemeter/scaler. Gamma levels (c/1 min) at 1-m depth intervals will be obtained throughout the length of the borehole. Audible response will be monitored during detector movement for indication of elevated count rate, which might indicate the presence of radioactive contamination. Gamma scan and logging results will be documented on survey maps. Locations of elevated response will be noted for further investigation.

Soil samples of approximately 500 g each will be collected at systematic and judgmental sampling locations. Surface samples will be obtained from the upper 15 cm soil layer, using trowels or bucket augers. Subsurface samples will be obtained using bucket augers, split spoon samplers, or other methods consistent with the drilling technique and equipment.

Samples will be assigned unique identification numbers and a chain of custody record and analytical request will be prepared.

4.8 Sample Analyses

Samples will be sent to an off-site commercial laboratory for gamma spectrometry analysis. Composite samples, consisting of equal amounts from individual samples representing each of the survey areas, will be prepared and analyzed for hard-to-detect (10 CFR Part 61) radionuclides. Results of these analyses will be used to develop fractional contributions of non-gamma emitting radionuclides in sediments. These contributions will be used to adjust the results of individual sample gamma analyses for the presence of other contaminants. Gamma analyses will be surrogates for calculating soil activity concentrations, assuming the previously analyzed mix of radionuclides remains proportional to the concentrations of principal gamma emitters.

4.10 Investigation

If measurements, sample analyses, or statistical data evaluation identifies residual radioactivity exceeding the release criteria, the source of the contamination will be investigated. Remediation will be performed, as necessary, and FSS activities repeated, utilizing a newly determined sampling pattern.

5. Data Evaluation

The sum of ratios will be calculated for radionuclides, which are of facility origin. The Sign Test will be performed for systematic samples and the results compared with the critical value for the appropriate number of samples and decision errors. The results of crawl space samples and judgmental samples will be compared individually with the release criteria.

Final Status Survey Plan

Addendum 008: Ventilation Systems

Revision 0

Prepared for
University of Virginia
Reactor Facility Decommissioning Project

Prepared by

CH2MHILL

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June 2003

Final Status Survey Plan

Addendum 008: Ventilation Systems

Revision 0

Prepared for
University of Virginia
Reactor Facility Decommissioning Project

June 2003

Client Approvals:

Ralph Allen | 6/23/3
OEHS | Date

| 6/20/2003
Technical Director | Date

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Final Status Survey Plan

Addendum 008: Ventilation Systems

Revision 0

Prepared for
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Reactor Facility Decommissioning Project

June 2003

Certified Health Physicist — Date: 6/18/03

Project Manager — Date: 6/18/03

CH2MHILL

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1. Introduction

The Master Final Status Survey Plan (UVA-FS-002) identifies the Data Quality Objectives for Final Status Survey (FSS) activities, together with the underlying technical assumptions, approaches, and methodologies for designing, implementing, and evaluating a FSS on each impacted area of the University of Virginia Research Reactor (UVAR) Facility. A separate survey area-specific addendum is prepared for each area or group of areas with common media, contaminants, and other characteristics, prior to beginning FSS; FSS for the specific area or group of areas will then be performed in accordance with that addendum. This addendum (Addendum 008) applies to the interior surfaces of the potentially impacted ventilation systems in the UVAR Facility building.

2. Description

The UVAR Facility is located on Old Reservoir Road, approximately 0.6 kilometers (km) west of the Charlottesville, VA city limits. The Facility includes the UVAR building, a small pond, and asphalt-paved roads, parking areas, and equipment/materials storage pads, situated on a land area of approximately 9500 m^2 (see Figure 2-1). The three-story building housed the UVA Research Reactor and the CAVALIER facility, as well as offices for the reactor staff and faculty and students of the Department of Nuclear Engineering miscellaneous laboratories, and other support facilities for the reactors and Department of Nuclear Engineering.

Several systems provided ventilation for facilities having a potential for airborne radioactivity. The remaining systems/components, which are potentially radiologically impacted, are:

- Exhaust for fume hood in Room M005.
- Exhaust for fume hood in Room M008.
- Exhaust for fume hoods (2) in Room M019.
- Exhaust for source storage Room G022.
- Hot Cell exhaust.
- Reactor Room recirculation and exhaust.

Because the exhaust ventilation systems in laboratories M005 and M008 had become contaminated with Tc-99 and Ni-63, respectively, during research projects in those facilities, new fume hoods and ductwork between the hoods and the exhaust fans were installed in these rooms a short time before the reactor decommissioning activities began. The blower assembly was removed from Room M-008 during D&D operations; the original squirrel-cage blower for the M005 exhaust system remains, along with the ductwork downstream of both fan units. During facility operation, these systems exhausted through the outside laboratory walls and into vertical ducts on the building exterior; the vertical ducts discharged above the roof level through rain-cap covered stacks. The remaining exhaust ventilation systems in laboratories M005 and M008 are potentially still impacted and will be included in this survey. Because the new hoods and ductwork were never used for contaminated operations, the potential for contamination of those surfaces is considered negligible.

Fume hoods in Room M019 became contaminated with Tc-99. Hood baffles were removed and cleaned. Ductwork from the rear of the hood was removed up to and including the HEPA filter and housing. A short section of ductwork, which connected the exhausts from this facility to the former exhaust ventilation from the Hot Cell, remains. The Hot Cell exhaust duct from inside the Hot Cell to the blower in Room M020, remains; the HEPA filter box has been removed from the point where the ductwork joins the blower. The combined

Hot Cell and M019 fume hood exhausts pass through a duct inside the Reactor Stack and discharge into the suction plenum of the Reactor Room exhaust fan.

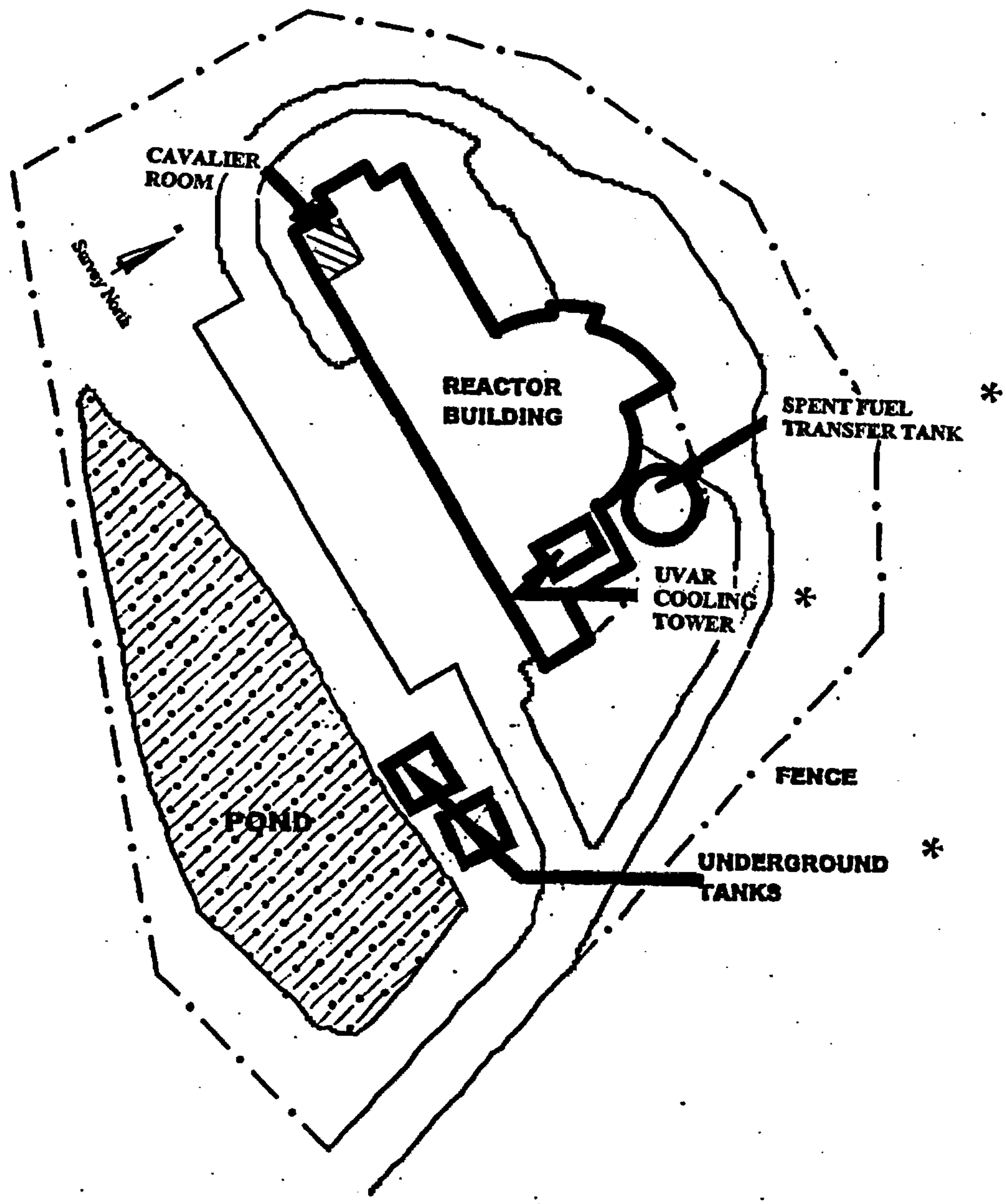
Reactor Room air is exhausted through a duct near the ceiling of the Reactor Room into the suction plenum of the Reactor Room exhaust fan at the top of the Reactor Stack. At this location the duct from the Hot Cell/M019 hood and the Reactor Room are combined and exhausted through the plenum vertically on the roof of the Reactor Room.

There was a small exhaust from the source storage room (Room G022). The blower has been removed, but the ductwork which discharges at the Mezzanine level on the east end of the building remains.

Reactor Room air is replenished by a recirculating system. This system draws fresh air from the outside and combines it with room air. This stream is heated or cooled as needed and then discharged back into the Reactor Room through 12 vents, located at the base of the Reactor Room wall.

Figures 2-2 to 2-4 indicate the locations of the remaining potentially impacted ventilation system surfaces. Except for portions of the recirculating air vents, which are encased in concrete, there is access to interior surfaces of components of these ventilation systems to conduct surface activity scans and measurements. It is anticipated that access will be adequate to demonstrate that radiological conditions satisfy decommissioning criteria.

Figure 2-1 University of Virginia Reactor Facility and Environs



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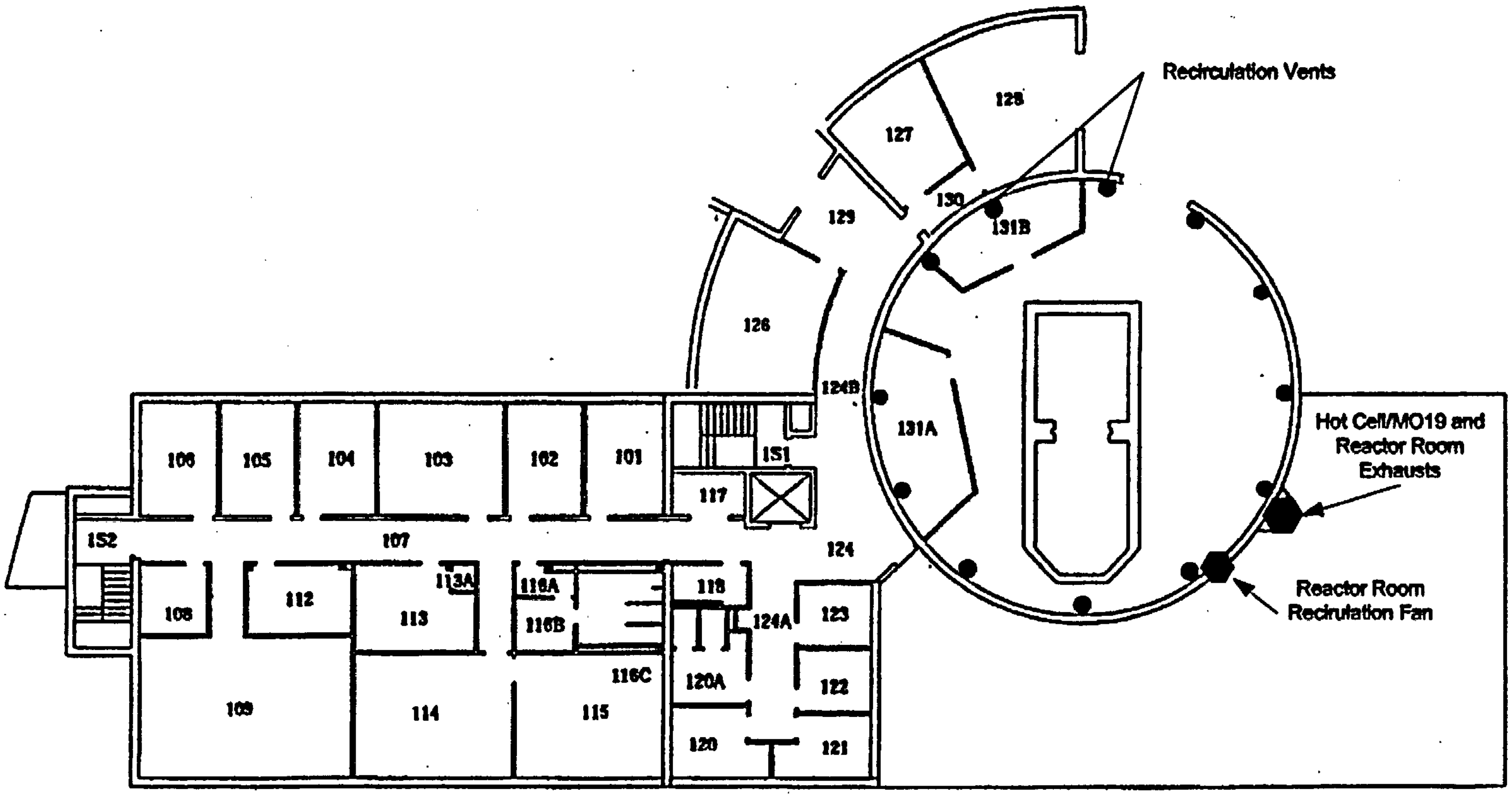

Figure 2-2 UVA Reactor First Floor Indicating Potentially Impacted Ventilation Systems.

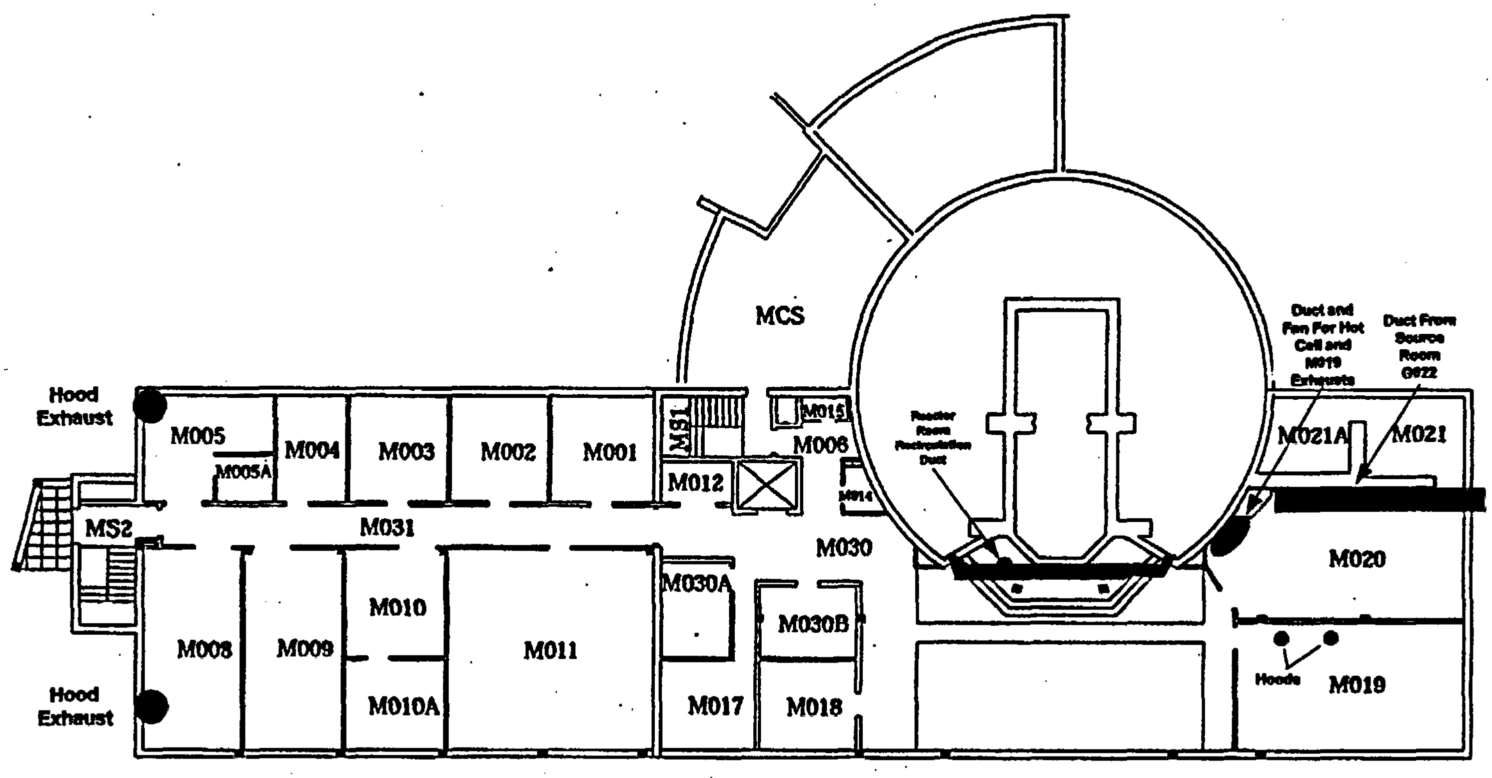

Figure 2-3 UVA Mezzanine Floor Indicating Potentially Impacted Ventilation Systems.

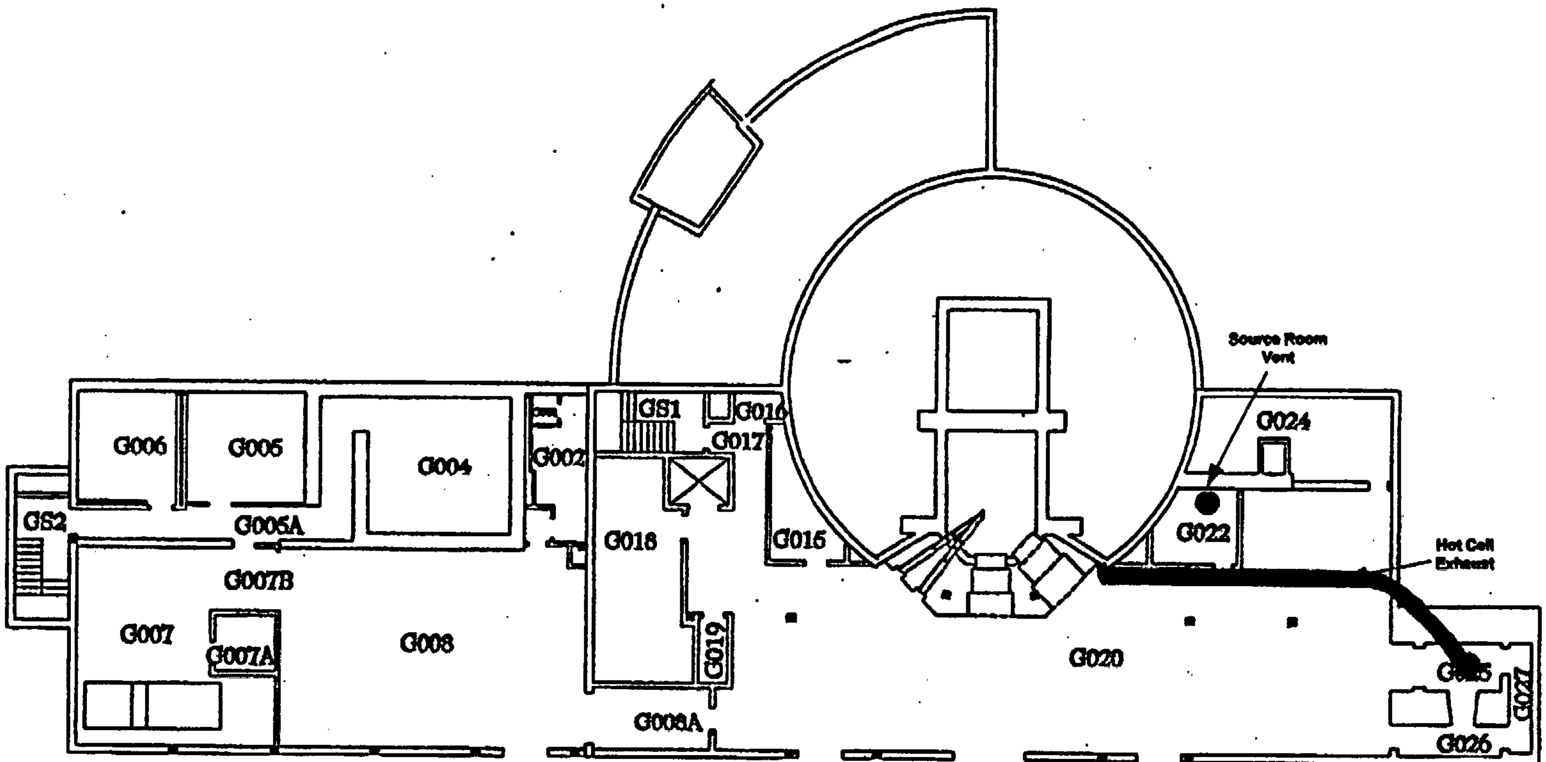

Figure 2-4 UVA Reactor Ground Floor Indicting Potentially Impacted Ventilation Systems.

3. Contaminants of Concern and Guidelines

The GTS Duratek initial characterization and continuing characterization by the CH2M HILL team showed that radiological contamination was generally low level and was limited to a small portion of the structure interior. The overall predominant radionuclides were Co-60 and Cs-137; smaller activities of fission and activation products, namely C-14, Fe-55, and Eu-152 were identified in some media. Ni-63 and Tc-99 contaminants were present on facility surfaces from research projects in labs M008 and M005, respectively.

The Decommissioning Plan established the criteria for residual radioactive material contamination on UVAR facility surfaces. UVAR facility criteria, also referred to as derived concentration guideline levels (DCGLs), are selected from the table of NRC default screening values in NUREG-1757. The screening DCGLs of 1.8 E+6 dpm/100 cm^2 for Tc-99 will be used for Room M005, and the screening DCGL of 1.3 E+6 dpm/100 cm^2 for Ni-63 will be used for Room M008. For the remainder of the facility ventilation systems, the screening DCGL of 7100 dpm/100 cm^2 for Co-60 - the most restrictive of the contaminants - will be considered the potential contaminant of concern. Unless there is specific evidence that contamination of a surface is comprised of radionuclides other than Co-60, the DCGL for Co-60 will be the basis for evaluating the final radiological status of the structure surfaces. Guidelines for removable structure contamination are 10% of the NRC screening default values for total surface activity, for Co-60 the removable activity limit is therefore 710 dpm/100 cm^2. This assures a conservative approach for satisfying the NRC dose-based criteria for future facility use. Appendix A of the Master FSS Plan describes the method for establishing guidelines for other radionuclides and combinations of radionuclides. If guidelines other than those indicated here are to be used, a justification will be prepared and included with the FSS documentation.

4. Survey Approach

4.1 Survey Reference System

Because of the nature and limited area of the ventilation system components, reference grid systems will not be established on surfaces to provide a means for referencing measurement and sampling locations. Instead, survey locations will be referenced to prominent facility features.

4.2 Survey Classification

Based on facility history (including the Historic Site Assessment), radiological monitoring conducted during characterization, and remedial activities, all potentially impacted exhaust ventilation systems are Class 1.

4.3 Survey Unit Identification

Impacted structure surfaces of $\leq 10\ m^2$ are not designated as survey units. Instead, a minimum of 1 measurement per m^2 or a total of 4 measurements, whichever is greater, will be obtained from such areas and compared individually with the $DCGL_W$.

4.4 Demonstrating Compliance with Release Guidelines

Total and removable surface activity measurements from ventilation system surfaces will be individually compared directly with the applicable DCGL. Because of the small surfaces and possible small number of data points, statistical tests will not be performed.

4.5 Background Reference Areas and Materials

If ambient direct radiation levels necessitate an unshielded/unshielded measurements approach, a material or instrument background value will not be required for correction of surface activity data. Otherwise, an instrument and/or material background will be determined on a similar material as the surface being surveyed, but without a history of potential contamination by licensed operations. If required, sufficient background determinations will be made for each instrument to provide an average background level that is accurate to within +/- 20%; this usually requires 8 to 10 measurements, which are then evaluated using the procedure described in draft NUREG/CR-5849 and additional data points obtained, as necessary.

4.6 Number of Required Data Points

Impacted structure surfaces of ≤ 10 m^2 are not designated as survey units. Instead, a minimum of 1 measurement per m^2 or a total of 4 measurements, whichever is greater, will be obtained from such areas and compared individually with the $DCGL_W$.

4.7 Sampling Pattern

To the extent that the surfaces are accessible, measurements will be performed at locations uniformly spaced throughout the ventilation system surfaces. Measurements will be biased to locations which, based on scanning results and professional judgment of the FSS field supervisor, have the greatest potential for contamination. Examples of such locations include duct inlet and discharge points, bends, seams, and locations of discoloration and accumulations of dirt and debris.

4.8 Survey Methods

Where conditions allow, beta scans of interior system surfaces will be performed using Ludlum Model 43-68 gas proportional or Ludlum Model 44-9 pancake GM detectors coupled with Ludlum Model 2221 ratemeter/scalers. The detector will be maintained within ~1 cm of the surface while advancing the detector at a rate of approximately one detector width per second. Scan speed will be adjusted, as necessary, to assure detection sensitivities are less than 50% of the release criteria. Audible response will be monitored for indication of elevated count rate, which might indicate the presence of radioactive contamination. Results (count rate) will be documented on survey area maps. Locations of elevated response will be noted for further investigation. Beta scanning coverage will be 100% of accessible surfaces.

Surface activity measurements will be performed at uniformly distributed and judgmental locations; 1-minute static measurements will be conducted using a Ludlum Model 43-68 gas proportional detector coupled with a Ludlum Model 2221 ratemeter/scaler.

Smears for removable activity will be performed at locations of direct activity measurements.

4.9 Sample Analyses

Smears will be analyzed for gross alpha and gross beta activity in the on-site counting facility.

4.10 Investigation

If measurements, sample analyses, or statistical data evaluation identifies residual radioactivity exceeding 50% of the release criteria, the source of the contamination will be investigated. Remediation will be performed, as necessary, and FSS activities repeated, utilizing a newly determined sampling pattern and random start point.

5. Data Evaluation

Measurements will be individually compared directly with release criteria. Only if all measurements are less than the criteria, will the surface satisfy the requirements for release.

GROUNDWATER CHARACTERIZATION REPORT
UNIVERSITY OF VIRGINIA NUCLEAR REACTOR
CHARLOTTESVILLE, VIRGINIA

JUNE 6, 2003

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GROUNDWATER CHARACTERIZATION REPORT
UNIVERSITY OF VIRGINIA NUCLEAR REACTOR

1.0 INTRODUCTION

As part of the decommissioning of the University of Virginia (UVA) research nuclear reactor, a groundwater characterization investigation was undertaken to describe the hydrogeologic conditions at the site and assess the impacts the operation of the reactor has had on local groundwater. It has been documented that leakage of water from the reactor pool was occurring prior to the draining of the pool in October 2002. According to the reactor staff, leakage rates of up to 900 gallons per day were recorded with the average leak rate being closer to 400 gallons per day. One of the goals of this investigation was to determine the likely pathway(s) taken by this released water.

2.0 PRIOR INVESTIGATIONS

In July 1996, the UVA Department of Environmental Sciences was asked to conduct an investigation to characterize groundwater conditions at the facility. Following the installation of two monitoring wells immediately adjacent to the pond just downhill of the reactor building, work was halted because of difficulties encountered in installing additional monitoring wells between the pond and the reactor building. In 1997, the UVA Office of Environmental Health and Safety (OEHS) was tasked to complete the groundwater characterization study at the reactor. OEHS hired a private drilling contractor to install additional wells at the site. A third downgradient monitoring well was installed immediately adjacent to the reactor building in the Spring of 1997. On October 24, 1997, the final report titled "Groundwater Monitoring System and Analytical Results University of Virginia Nuclear Reactor" was submitted to Dr. Robert Mulder, Director of the facility. That report concluded that groundwater was flowing downhill from the reactor building to the pond and that the monitoring wells at the facility were in the correct positions to monitor that groundwater. The report further concluded that the water releases from the reactor had not significantly impacted groundwater quality at the site. With the commencement of the reactor decommissioning expected to begin late in 2001, the NRC requested in initial reviews that a more complete characterization of the groundwater hydrology

of the reactor site be completed. In the Fall of 2001, the UVA Office of Environmental Health and Safety was tasked with this objective. This report incorporates all groundwater data gathered to date for this facility including that submitted in the October 1997 report.

3.0 FACILITY LOCATION AND SITE DESCRIPTION

The University of Virginia Reactor is located on the southeastern side of Jefferson Mountain on the western edge of the Grounds of the University in Charlottesville, Virginia (Figures 1, 2 and 3). Access to the Site is via Old Reservoir Road that originates at McCormick Road. The facility is located within a fenced area and consists of a single building that formerly housed two small research nuclear reactors, offices, laboratories, and support services; a parking area around the building; and a surrounding wooded area. Within the fenced in area and directly downhill of the reactor building is a pond that was a water supply reservoir for the university and Charlottesville in the late 1800's to early 1900's. The reservoir (hereafter pond) is considered part of the reactor facility. During the period of time over which the reactor operated, the pond water was monitored for water quality before water was released to Meadow Creek downhill of the dam.

4.0 TECHNICAL APPROACH

The goal of this project is to characterize the hydrogeology in the area of the reactor and to assess whether there has been a detectible release of radionuclides to the groundwater. To achieve this goal, this investigation included the following:

- A review of available geologic literature and previous studies
- Field examination of geologic conditions,
- Examination of facility plans and infrastructure,
- Installation of two additional monitoring wells,
- Collection of groundwater samples for radionuclide analysis,
- Performance of an aquifer test to determine aquifer properties,
- Performance of a tracer study to determine groundwater flow paths and rates, and
- Monitoring of water table elevations for use in developing a water table contour map.

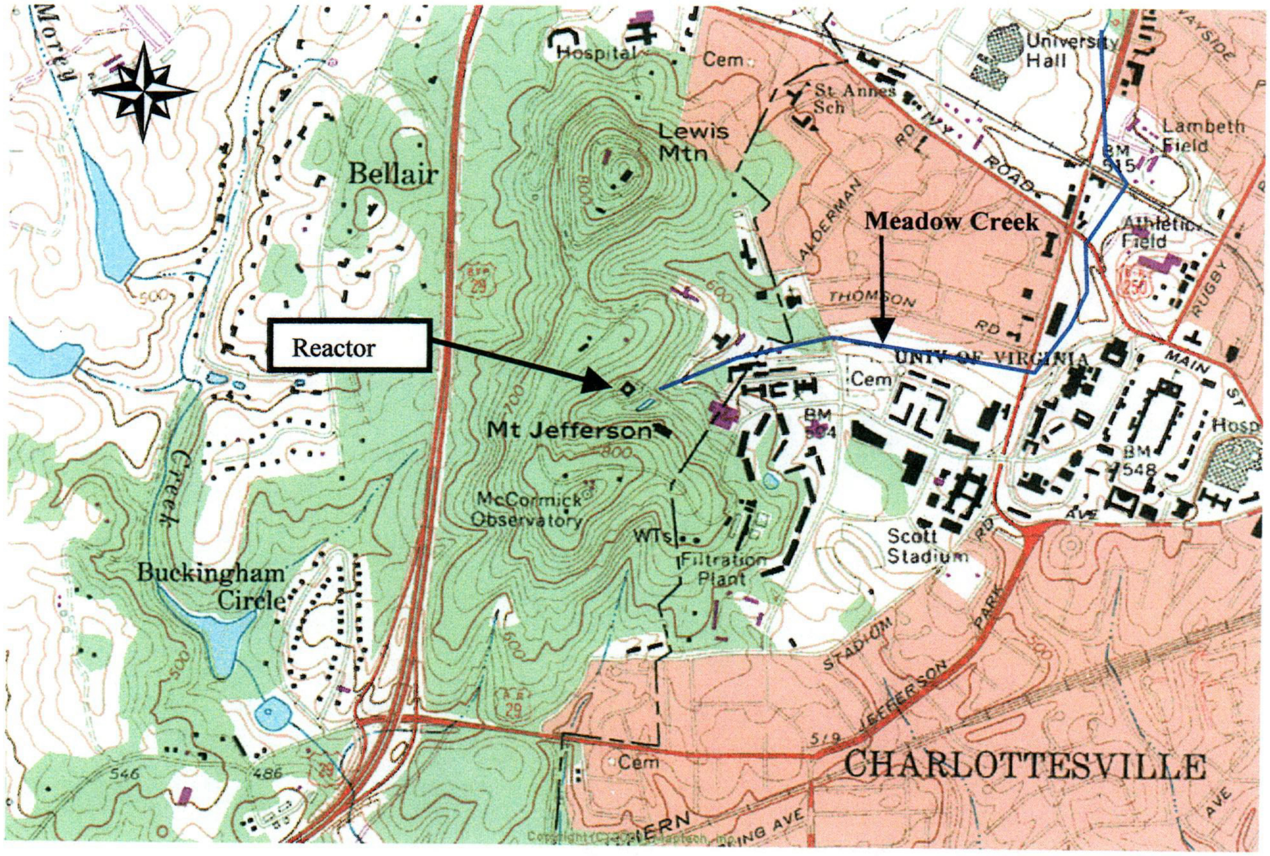

Figure 1. Site Location map from USGS Charlottesville West 7.5 Minute Quadrangle, PR 1987.

4.1 Monitoring Well Installations

In 1997, prior to OEHS taking on this project, graduate students from the Department of Environmental Sciences at the University installed two shallow monitoring wells immediately upslope from the reactor facility pond using a small power auger (Figure 4). This auger was used to bore an approximate three-inch diameter borehole to at least five feet below the water table. Following the removal of the auger flights, 2-inch ID schedule 40 PVC well casing and screen were installed in the borehole establishing two shallow monitoring wells. The wells designated as MW-1 and MW-2 are located on the steep slope approximately five feet vertically uphill of the reactor facility pond. The well logs are included in Appendix A. Attempts to install

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wells further upslope using this equipment were thwarted by large rocks in the overburden fill material that was emplaced during the construction of the parking lot.

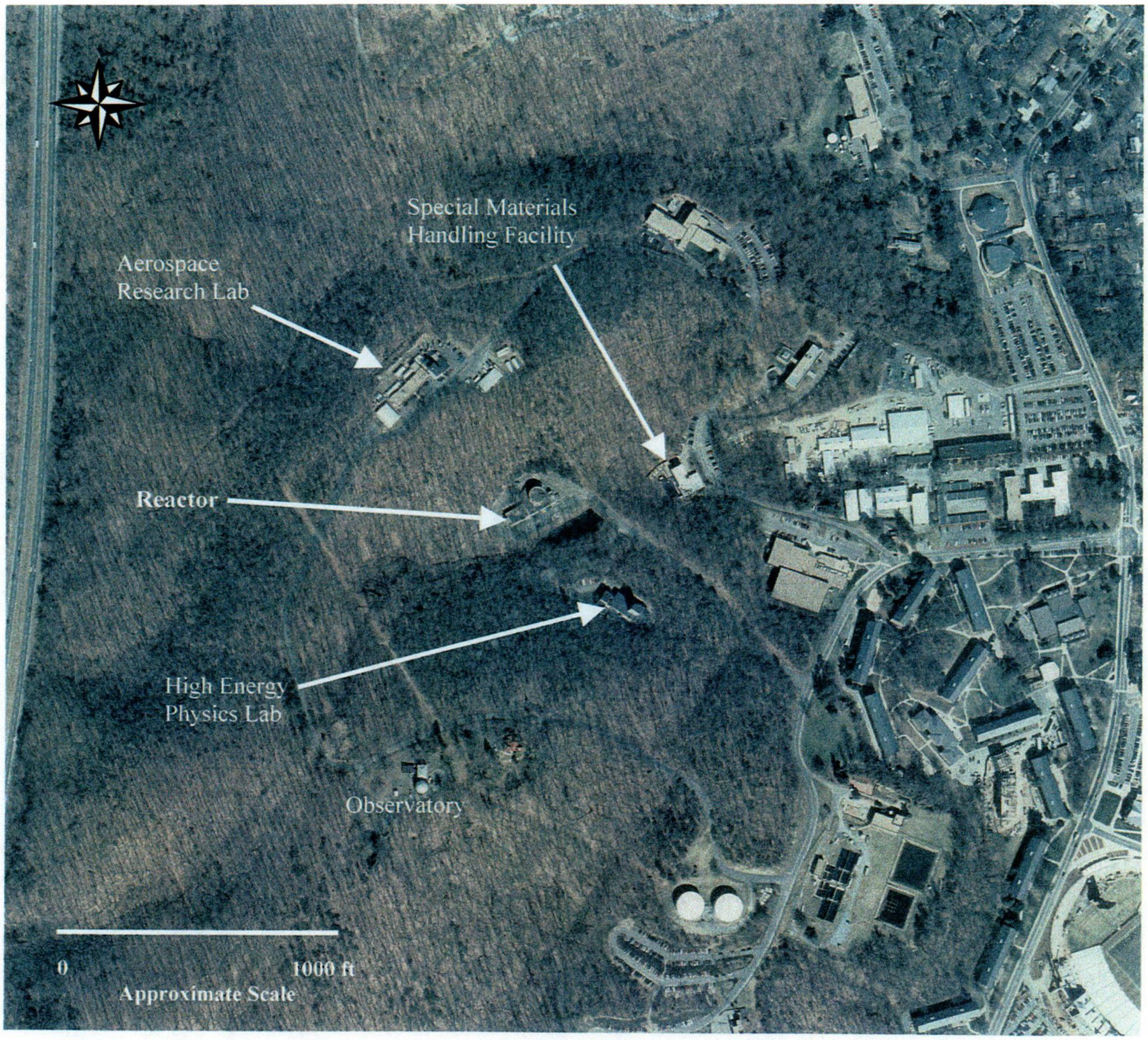

Figure 2. Year 2000 aerial photograph of the reactor site and surrounding area.

OEHS contracted with Certified Environmental Drilling of Charlottesville to install two borings and monitoring wells in the parking lot area between the Reactor building and the slope to the pond. The borings were installed using a truck-mounted hollow-stem augering rig. The borings were emplaced by Certified Environmental Drilling using hollow stem auger drilling methods.

C02

The first boring was located on the edge of the lower parking lot just above the pond. The second boring was installed in the lower parking lot about 15 feet from the back of the reactor building and directly in line between the reactor and the pond. Split-spoon soil samples were obtained at five-foot intervals within the borings in accordance with ASTM D1586-84 Standard Penetration Test methods. Prior to sampling, the split-spoon sampler was decontaminated using detergent and tap water followed by a deionized water rinse. The soil samples collected from the split-spoons were geologically logged by Jeffrey Sitler of OEHS. Soil samples were also collected by the reactor staff for radiological analysis. The results of the analysis are included in Appendix B, Table 1.

Figure 3. Year 2000 aerial photograph of the reactor facility.

C03

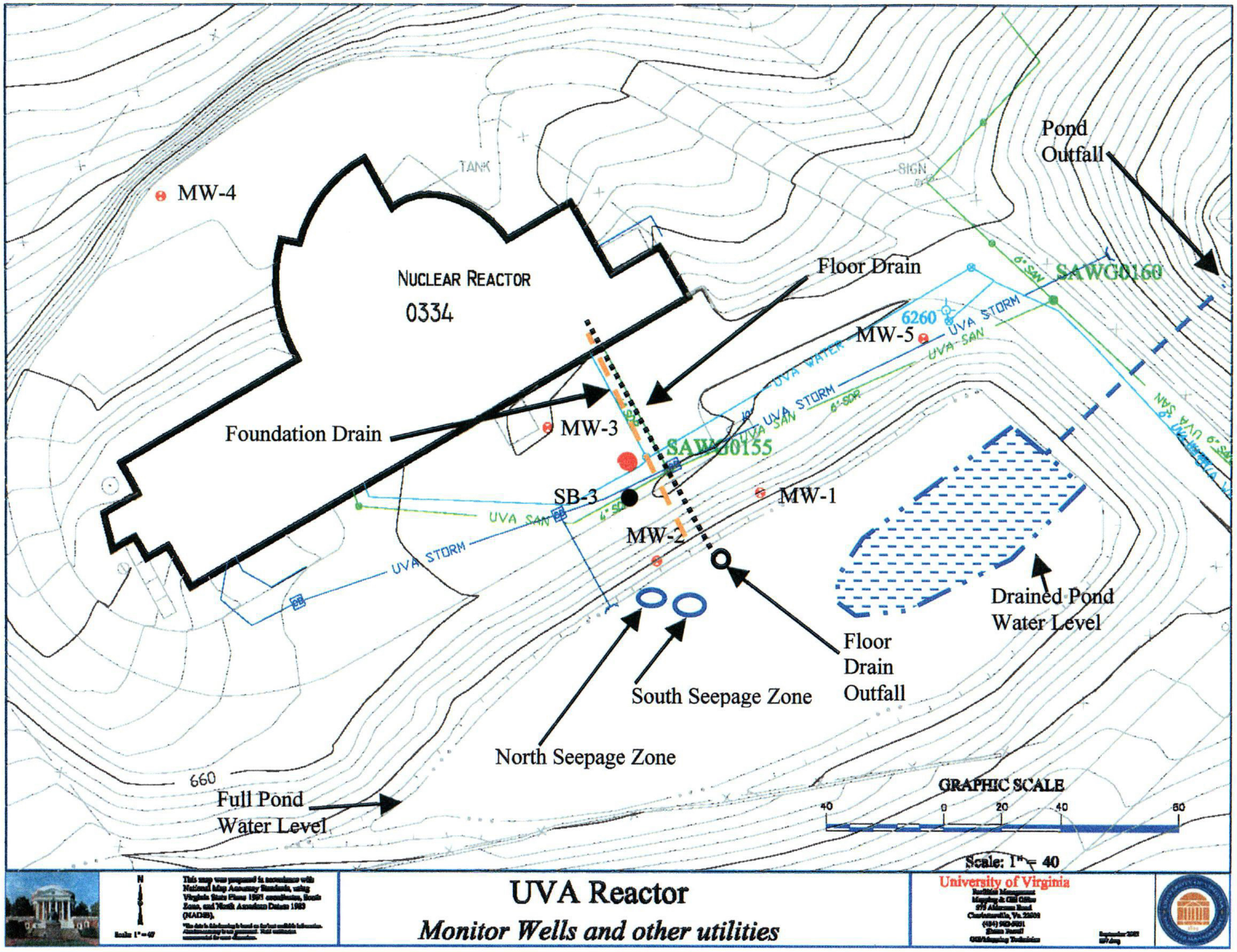

Figure 4. Detailed map of reactor area showing monitoring wells, underground utilities, and tracer study monitoring points

C04

Because auger refusal was encountered above the water table in the first boring at the edge of the parking lot, a well was not constructed at that location. Following the completion of the second boring, a two-inch monitoring well designated as MW-3 was constructed in the borehole. The well was constructed concurrent with the withdrawal of the augers to permit the accurate placement of the well screen, filter pack, bentonite seal, riser and grout.

In the review of the decommissioning plans, the NRC staff indicated that an upgradient well was desired. In addition, the OEHS and reactor staff decided that a downgradient monitoring in the vicinity of the pond dam would also be advantageous in describing the groundwater conditions at the site. On September 7, 2001, OEHS again contracted with Certified Environmental Drilling of Charlottesville to install two additional monitoring wells. Because of the potential presence of shallow bedrock, OEHS decided to install these wells using air rotary methods. A T450 Rotadrill manufactured by Schramm was used.

Using a nominal 6-inch diameter air rotary hammer, the boreholes were advanced at ten-foot increments with the OEHS staff geologist logging the cuttings blown from the hole. At the dam location, bedrock was encountered at approximated 12 feet in depth. After advancing ten feet, the drilling was stopped and the borehole was checked for the presence of water after letting it set for ten minutes. This process was repeated for each ten-foot interval until adequate ground water was encountered. Adequate water was encountered at 50 feet in depth. After pulling the drill stems from the borehole, a two-inch monitoring well (MW-5) was constructed in the borehole.

The upgradient well (MW-4) was installed in the upper parking lot against the bedrock cliff face formed by the bench cut excavated when the facility was built. The actual well location was chosen for three reasons. First, it is uphill and upgradient of the entire reactor facility. Second, it is directly uphill and upgradient of the reactor containment. Third, it is located in a near vertical bedrock unit that is deeply weathered and in line with the reactor containment and MW-3, thus would likely be in the flow path of groundwater moving downhill and downgradient. Based on the drilling conditions encountered, it appears that MW-3 and MW-4 were installed in this unit.

As with drilling the previous well, air rotary drilling was used to install MW-4. Solid bedrock was not encountered during the drilling operation confirming the location in the weathered bedrock unit. The first show of water occurred at 42 feet but adequate water was not observed until around 57 feet. The well was set at 60 feet and constructed of 2-inch PVC screen and riser. As with all of the wells, a sand pack was used opposite the screen, followed by a two foot Bentonite seal, followed by a cement grout and a bolt-down, water tight manhole.

Following the installation of the monitoring wells, each was developed by pumping with a 12-volt submersible pump until the water from each well ran clear and between two and ten-well volumes had been removed. The construction details for all of the site wells are included with the boring logs in Appendix A and are summarized in Table 1.

Table 1 Monitoring Well Detail

Data	MW-1	MW-2	MW-3	MW-4	MW-5
Date Installed	7/2/1996	7/2/1996	4/25/1997	9/7/2001	9/7/2001
Top of casing elevation (feet AMSL (1)	652.76	651.22	659.76	682.57	658.36
Casing Stick up (Feet above ground surface)	1.78	2.14	-.038	-0.50	-0.55
Initial Water Level from Top of Casing	8.37	6.79	5.20	19.93	15.48
Water table elevation (ft AMSL)	644.49	644.43	654.56	662.64	642.88
Total Well Depth below surface	9.56	6.40	30	50	40
Screened interval length (ft)	1.25	1.25	15	40	30
Standing water in well (ft)	3.13	2.06	24.50	30.07	24.52

(1) AMSL = above mean sea level

4.2 Groundwater Sampling and Radiological Analysis

Quarterly groundwater samples were collected from the monitoring wells on fifteen occasions for radiological analysis. Before each sampling event, a water level measurement was taken using an electronic water level probe. Monitoring Wells MW-1 and MW-2 were sampled using a clean decontaminated stainless steel bailer for each well. Wells MW-1 and MW-2 were bailed dry before three well volumes could be purged. In these cases, the purged wells were allowed to recover for several minutes and then the groundwater sample was taken. Because of the larger volumes of standing water in Monitoring Wells MW-3, MW-4 and MW-5, they were purged and sampled using a 12-volt submersible pump and polyethylene tubing. The pump was set at 28 feet below the surface, well within the screened interval. The wells were purged of at least three volumes before the groundwater samples were collected. Typically between 10 and 20 gallons of water were purged from the wells.

The water samples were collected for radiological analysis and were collected directly into clean new polyethylene two-liter containers by EHS staff. Some of the water samples were analyzed in house by EHS and Reactor staff while the remainder of the samples were sent to contract laboratories including Teledyne, Framatome, and Duke. The results are presented and discussed later in this report.

Because the water levels in Monitoring Wells MW-1 and MW-2 dropped below the well bottoms after the pond was drained, it was not possible to sample these wells after July 2002.

4.3 Tracer Study

To assist in determining the path(s) that water leaking from the reactor pool was taking and predict travel times, a tracer was added to the reactor pool water following the removal of all reactor equipment from the pool during the decommissioning. Initially, it was decided that either a chloride or bromide tracer would be used because at even moderate concentrations neither of these would present an environmental concern when released to the environment. To determine which would have the least interference from background concentrations, initial water samples were taken from the reactor pool, the pond discharge outfall pipe, MW-3, and two seepage zones

on the banks of the pond. By this time, the pond had been drained so that pond sediments could be collected during the decommissioning.

The water samples were submitted to UVA Department of Environmental Sciences aquatic chemistry laboratory operated by Dr. Jim Galoway. This laboratory was established to analyze atmospheric precipitation for very low concentrations of contaminants; therefore, they were well equipped to analyze the tracer water samples to a detection limit of 0.02 mg/l. The water samples were analyzed by ion chromatography using a Dionex IC unit following Dionex Application Note 135 that incorporates EPA method 300. This procedure uses the AS-9HC separator column with 9mM Na2CO3 and ASRS-Ultra suppressor. To increase the sensitivity of the method, 1000ul of sample is injected. With each group of samples submitted to the laboratory, one QA/QC sample was included. The QA/QC sample was either a field blank or duplicate.

As the water chemistry results in Table 2 show, chlorides were present in all of the water sampled while bromide was absent in all. Based on this result, it was decided that bromide would be used as the tracer and would be introduced into the system as sodium bromide.

On August 30, 2002, 6 Kg of reagent grade sodium bromide salt (99+% purity) were added to the reactor pool. Based on the current water level in the reactor pool, the reactor staff estimated that the pool contained approximately 72,000 gallons of water, thus, adding 6 Kg grams of NaBr to the water should produce a solution with a bromide concentration of 17 mg/l, almost 1000 times the method detection limit for bromide. The NaBr was added as a granular salt through 1.5-inch diameter PVC pipes extending to the bottom of the pool. This was done to initially create a layer of very high bromide concentration at the base of the pool where the leakage was likely occurring. Measurements of specific conductivity after the addition of the bromide confirmed that that the bromide was largely staying in a layer about six inches thick over the bottom of the reactor pool. The bottom layer of water had a specific conductivity of 3600 μmhos/cm while above this the conductivity dropped to 32 μmhos/cm. After a day, circulating pumps were turned on to mix the bromide throughout the reactor pool. During the first sampling event, the reactor pool water was sampled and analyzed for bromide. The pool water was found

to have a concentration of 16.92 mg/l. Water samples were taken from the six downgradient locations mentioned above nine times over the period ending on February 4, 2003. The results are discussed below in the Findings Section.

Water remained in the reactor pool from August 30, 2002 to October 20, 2002. On October 21, 2002 the pool was drained to allow continuation of the decommissioning. Over this period, it is estimated that the reactor leaked approximately 15,000 gallons of water into the subsurface. This is based on the drop of the water level in the pool and on previous measured leak rates, taking into account evaporation.

4.4 Examination of Facility

As part of the decommissioning, the reactor building and infrastructure were investigated in great detail to identify contaminated materials that required removal. Part of this examination included investigating the probable flow paths of water discharging from the facility, specifically identifying the likely paths the known leakage from the reactor pool was taking. This work included finding and exposing underground piping in numerous locations, tracing and mapping piping using cameras and other means, exposing the foundations of the reactor pool, and numerous boring through the concrete of the reactor pool to assess its integrity.

5.0 FINDINGS

5.1 Hydrogeology

The hydrogeology of the Site was characterized through published literature, site observations, and field investigations including installation of monitoring wells, aquifer slug tests, and tracer studies.

5.1.1 General Hydrogeology

The reactor is located in the Piedmont Physiographic Province. According to the Virginia Geologic Map and the Geologic Map of Albemarle County, the site is underlain by the Precambrian-age rocks of the Lynchburg Group (Johnson, Stanley, 1993). Specifically the Site is underlain by the Rockfish Conglomerate. The Rockfish Conglomerate consists of a 1,200-foot

thick metamorphosed sandstone specifically characterized as a metagraywacke and quartzose schist that has a 100 foot basal conglomerate (Figure 5)

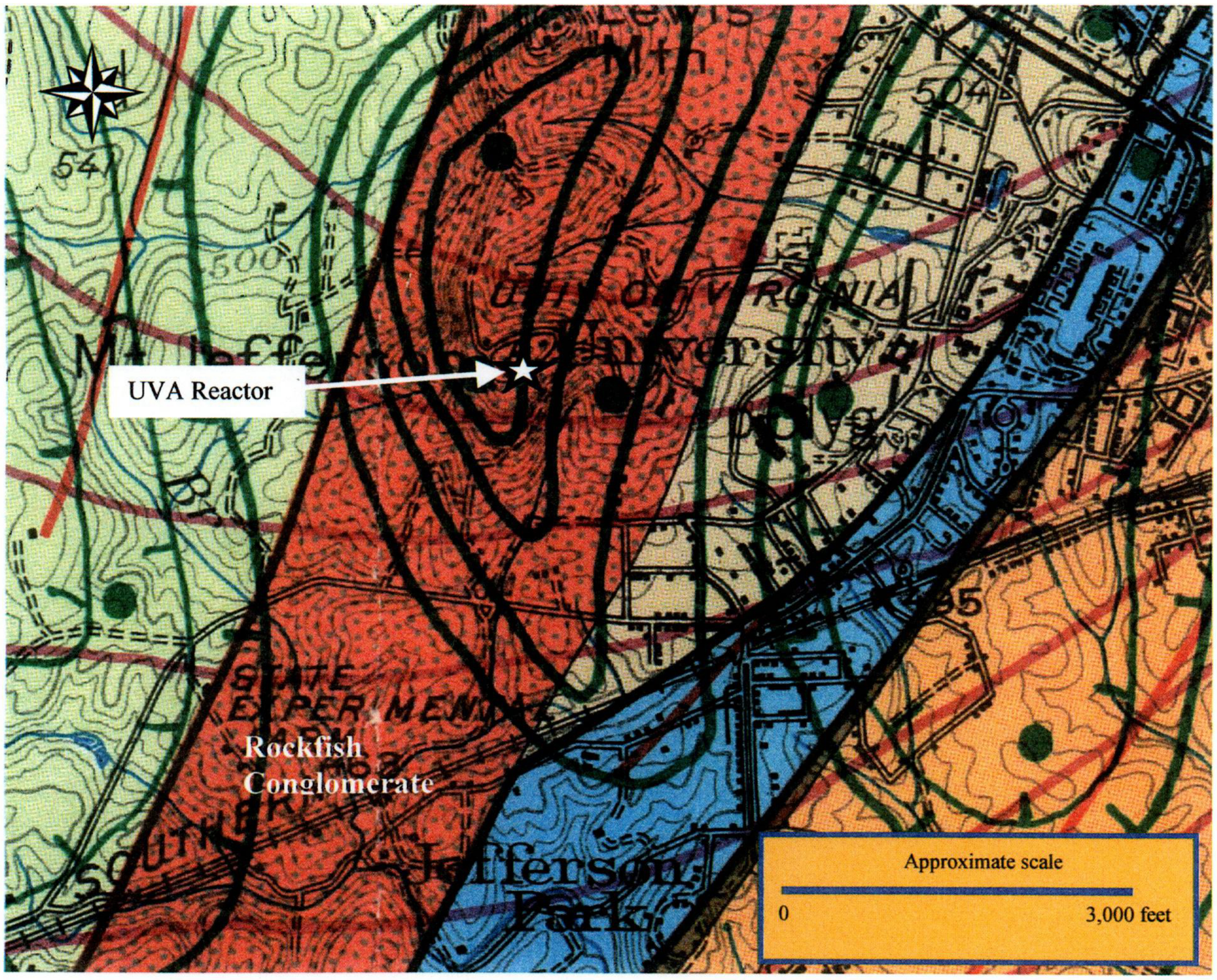

Figure 5. Geologic Map of Region with the reactor site marked. This base map also contained gravity and magnetic field data represented as contours. ***(Johnson, Stanley S. and Palmer C. Sweet; Magnetic and Gravity Surveys of Albemarle and Fluvanna Counties, Virginia; Virginia Division of Mineral Resources, Charlottesville, VA; 1969.)***

Typically overlying the bedrock in this area are residual soils or saprolites formed by the in-place weathering of the underlying bedrock. The saprolite retains the structure of the underlying bedrock and grades from competent bedrock to surficial soils within a few feet of the ground surface. The saprolite encountered during this investigation consisted of micaceous sandy silt and varied in thickness from ten feet to over 30 feet. Competent bedrock was not encountered by

C05

30 feet in MW-3 and MW-4 while competent bedrock was encountered at 21 feet below the surface at the lower edge of the parking lot and competent bedrock outcrops just above the upper parking lot. MW-1 and MW-4 are located in a narrow, 30 feet wide, bedrock unit that exhibits very deep weathering. From this data, it is clear that the competent bedrock surface is not planar in nature.

Above the saprolite are the clay-rich surficial soils and in much of the area of the parking lot, fill material used to build the parking lot. Along the back edge of the parking lot, ten feet of fill material was penetrated before the original land surface was encountered.

Structurally, the bedrock of the Piedmont and adjacent Blue Ridge are strongly folded and in some places faulted. Figure 6 is a photograph of the bedrock outcrop on the upper side of the reactor. In the area of the site, the units of the bedrock are nearly vertical and strike in a southwest to northeast direction parallel to the Blue Ridge Mountains and perpendicular to the tectonic forces that caused the deformation.

Figure 6. Photograph of outcrop on the upper side of the reactor facility showing the near vertical layering of the units.

Groundwater storage and movement occur in both the saprolite and underlying bedrock as a single water table aquifer. Recharge to groundwater is derived from precipitation and occurs over broad areas as a result of the infiltration of precipitation into the saprolite. The Virginia Department of Mines, Minerals, and Energy estimates that 15% of all precipitation infiltrates as recharge to groundwater. Using an average rainfall of 45 inches, 6.75 inches of rainfall is recharged to the aquifer in an average year.

The saprolite has a high porosity and stores a vast quantity of water; however, because of its fine-grained texture groundwater movement within the unit is very slow. Groundwater in the bedrock is recharged by the saprolite. Overall, the igneous and metamorphic bedrock types found in the Piedmont are considered to be fair to poor aquifers because of the lack of primary porosity and permeability. Groundwater movement and storage takes place mainly in fractures and other secondary features of bedrock that typically do not yield large quantities of water in this geographic area. Groundwater flow and yield are dependent on the frequency and interconnection of the fractures. Local wells are predominantly bedrock wells between 60 and 400 feet deep. The depth to groundwater, based on OEHS's experience in this region, can be expected to range from a few feet to 50 feet below the surface. At this Site, the depth to the water table ranges from at the surface at the pond to as much as twenty feet in the upper parking lot.

Actual water table elevations measured in the site wells varied considerably over the length of this study as a result of a three-year drought that ended in the Fall of 2002. As an example, the static water level in MW-4 was at it lowest, 25.76 feet below the surface, in September 2002 while in May 2003 it was measured at 5.85 feet below the surface, the highest level measured over the course of this study (Figure 8 and Appendix A). At this elevation, the surface of the water table would be at or near the lower level of the reactor building. This was confirmed by the water that was observed in the boreholes placed through the reactor floor in several locations in April and May of 2003.

The opposite was observed with Monitoring Wells MW-1 and MW-2. When the pond was drained during the summer of 2002, the water table dropped below the bottom of the wells and remained there through out this study.

The water table in the Piedmont Physiographic Province generally mimics the topographic surface with the water table being closer to the surface at valley bottoms than at ridge tops. The resulting groundwater flow is from ridge tops to valley bottoms where groundwater discharges to streams and rivers providing the base flow that maintains stream flow when surface runoff from precipitation events is not occurring. Based on the water levels measured in the wells installed at the Site, groundwater is flowing to the southeast from the reactor building to the former reservoir. Figure 7 is a water table contour map constructed using the April 15, 2003 water level data. (Appendix A). Under these conditions, releases to groundwater in the area of the reactor move with groundwater and discharge to the pond or the Meadow Creek tributary on which the pond is situated.

This groundwater configuration remained constant throughout the drought of 2002 to the current time. At no time did the groundwater levels decline or rise in such a manner that the groundwater flow direction reversed on changed direction dramatically. As is shown in Figure 8, the measured water table elevations in the wells varied in a consistent manner with the upgradient wells remaining upgradient and the downgradient well remaining downgradient.

5.1.2 Observed Groundwater Flow Rates and Pathways

Specific measurement of the hydraulic properties of the geologic materials at the site was performed for this investigation. In addition, the tracer study highlighted the likely flow paths water leaking from the reactor facility would follow.

5.1.2.1 Flow Rates

The hydraulic conductivity of the native materials encountered in MW-3 at the Site was determined to be 1.5 x 10^{-4} feet/minute based on a slug test conducted on October 10, 1997. The slug test data are included in Appendix C. The slug test was conducted by the insertion of a solid slug into the well. The change in water level was measured using a pressure transducer attached

to a Campbell Scientific data logger. The data was then downloaded to a computer and analyzed to determine the hydraulic conductivity. Based on the water level measurements in all of the wells at the Site on February 4, 2003, a hydraulic gradient of 0.14 ft/ft was calculated. Using an effective porosity of 40% for the saprolite (mixture of silt and clay), groundwater at the Site was calculated to be moving at a rate of 7.6×10^{-2} feet/day or 27 ft/yr. Given this velocity, it would take a conservative contaminant approximately one year to move from the reactor to MW-3 though the native materials at the Site.

The groundwater flow rate to MW-3 was also determined using the bromide tracer as described previously. The tracer was added to the reactor pool on August 30, 2002 and was first detected in MW-3 on February 4, 2003, approximately five months travel time (Table 2). Given the expected heterogeneities and "man made short circuits" in the flow system, this is in relatively close agreement with the slug test estimated groundwater flow velocity.

It should also be noted that bromide was detected in two other location further downgradient site before it was detected in MW-3. These locations included the seepage zone on the edge of the pond, the floor drainpipe at the pond, and the pond discharge pipe (Figure 4). These faster flow rates are likely directly attributable to man made conditions at the site.

5.1.2.2 Released Water Flow Paths

Through visual examinations of the facility construction and from data gathered during the tracer study, it is apparent that leakage from the reactor had multiple paths to follow in the subsurface. In addition, the detailed examination of the reactor pool did not reveal the presence of a specific leak points rather it appears likely that the leakage was diffuse occurring via many very small openings and cracks in the concrete. As stated previously, the pool was leaking at rates that varied from 400 gallons per day to 900 gallons per day, which translates to only 0.3 gallons per minute to 0.6 gallons per minute. The flow pathways included the natural groundwater flow paths, pipes, and subsurface interfaces between natural geologic materials and fill.

The obvious pathway would be for the water to infiltrate into the subsurface and enter the natural groundwater flow. In this case, the flow is as shown in Figure 7. This pathway is further

enhanced by the geologic condition beneath the reactor. As discovered during the installation of MW-3 and MW-4 and the examination of the bedrock outcrops above the reactor, the reactor pool actually rests on a near vertical geologic unit that has been weathered to the point that it is unconsolidated. As a result, the units can be characterized as a micaceous sandy silt and as such is relatively permeable and would provide a good flow path for groundwater. There is indirect evidence that this is where much of the leaking water ended up. As can be seen in the water level data in MW-3 just down gradient of the reactor, the water table was in a steady decline, likely because of the drought that actually began in the Summer of 1999 and ended in the Fall of 2002. It is interesting to note however, that the water level in MW-3 did not rebound as quickly in the Fall of 2002 as it did in MW-4 and MW-5 when the drought ended. This delayed rebound is potentially the result of the water table still responding to the draining of the reactor pool, which stopped the recharge to the groundwater that was occurring from the leaking pool.

Another pathway was observed through investigations carried out by the reactor decommissioning contractor. A French drain system and a floor drain system were identified beneath the reactor facility. The French drain consisted of approximate six-inch diameter terracotta pipe bedded in gravel and surrounding the reactor pool at a level just above the foundation. Additional lines of the French drain were found to be extending off to other areas of the building foundation. Because the pipe was located six inches or more above the base of the gravel bedding, water could only flow through pipe when the water level in the gravel bed reached the level of the pipe whereby water would flow into the drain pipe and flow out of the building to its outfall located about half way down the hillslope to the pond.

Water leaking from the reactor pool would likely have been collected by the gravel, however, it would only flow out the pipe if the water level in the gravel reached the pipe, otherwise, the water would find another course by infiltrating deeper into the subsurface as groundwater recharge to the underlying soil and bedrock as described previously. During the initial examination of the French Drain in February and March, 2003, it was found to be dry. As described previously, in April and May of 2003, the water table at the facility rose significantly resulting from the above average precipitation amounts over Winter and Spring. By the end of

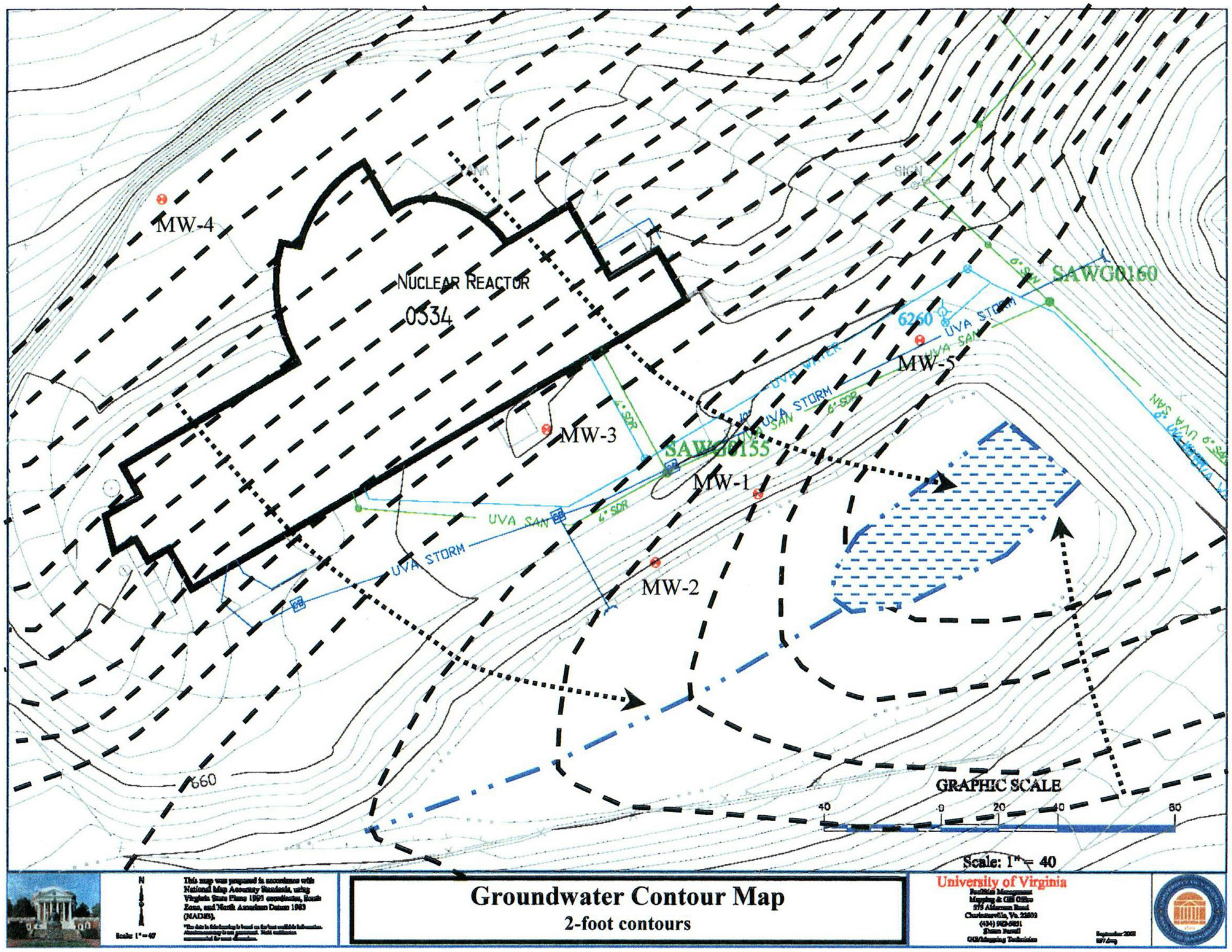

Figure 7. Ground Water Contour Map with arrows showing the direction of groundwater flow. Based on water level measurements made on April 15, 2003.

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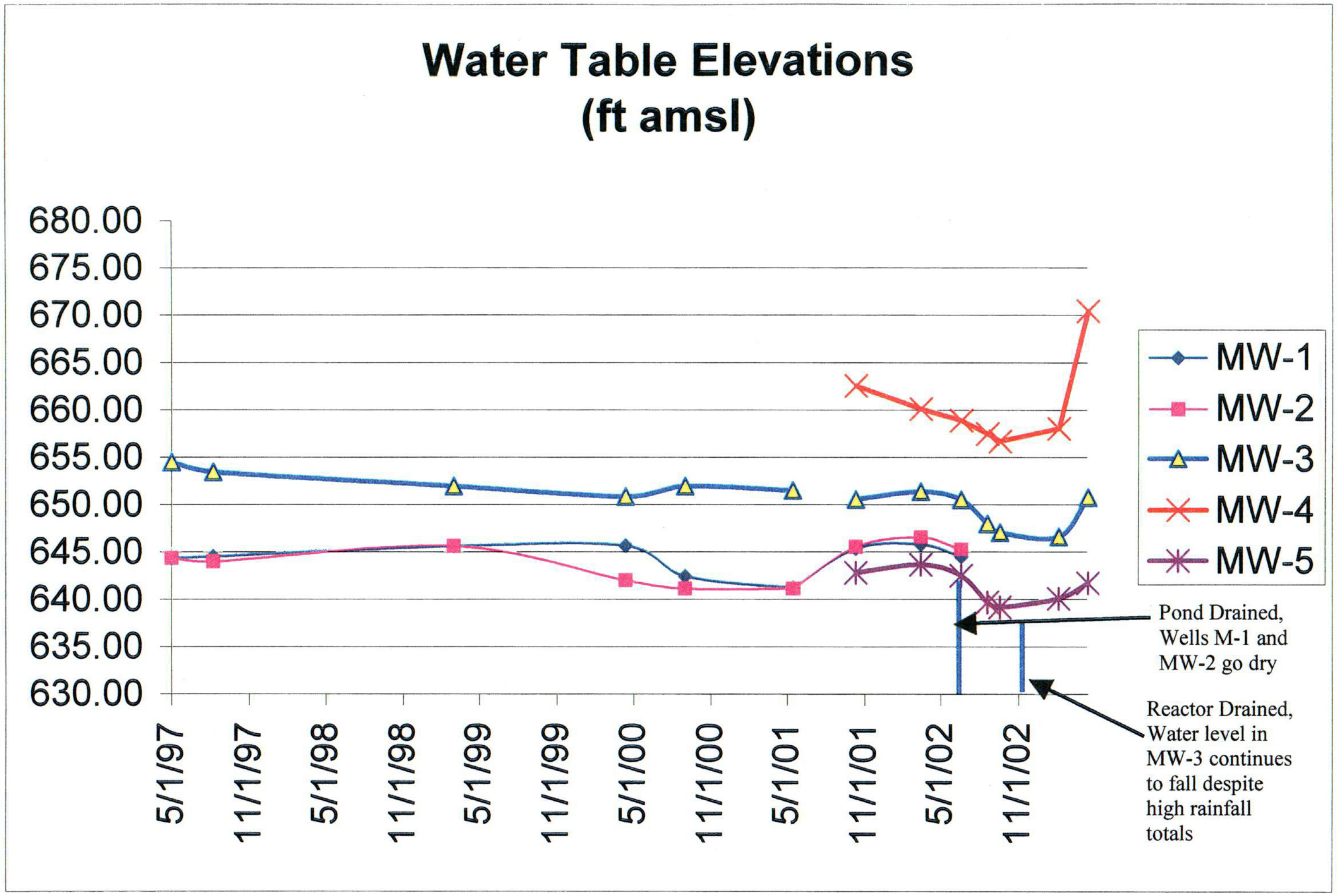

Figure 8. Water table elevation data for each monitoring well

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Table 2. Groundwater Tracer Study Results (1)

Date	8/21/02	8/21/02		9/6/02	9/12/02	9/18/02	9/24/02	10/2/02	10/17/02		11/8/02	12/4/02	2/4/03	4/15/03	5/15/03
Sampling Location	Chloride (mg/l)	Bromide (mg/l)	Bromide tracer added to the reactor pool water 8/30/02	Bromide (mg/l)	Bromide (mg/l)	Bromide (mg/l)	Bromide (mg/l)	Bromide (mg/l)	Bromide (mg/l)	Reactor pool Drained 10/21/02	Bromide (mg/l)	Bromide (mg/l)	Bromide (mg/l)	Bromide (mg/l)	Bromide (mg/l)
Reactor Pool	0.21	< 0.02		16.96	NA	NA	NA	NA	NA		dry	dry	dry	dry	Dry
Seepage zone north	7.95	< 0.02		< 0.02	< 0.02	< 0.02	< 0.02	0.031	< 0.02		<0.020	0.096	<0.02	<0.02	NA
Seepage zone south	1.7	< 0.02		< 0.02	< 0.02	< 0.02	< 0.02	< 0.02	0.027		<0.020	<0.02	<0.02	NA(1)	NA
MW-3	5.6	< 0.02		<0.02	<0.02	< 0.02	< 0.02	< 0.02	< 0.02		NA	NA	0.02	0.09	<0.02
Pond outfall pipe	5.37	< 0.02		0.022	0.022	0.036	< 0.02	< 0.02	< 0.02		<0.020	0.067	<0.02	<0.02	<0.02
Floor Drain Pipe	NA	NA		NA	0.245	0.182	0.146	0.170	0.203		0.383	0.322	<0.02	dry (2)	NA
MW-4	NA	NA		NA	NA	<0.02	NA	NA	NA		NA	NA	<0.02	0.03	<0.02
MW-5	NA	NA		NA	NA	<0.02	NA	NA	NA		NA	NA	<0.02	<0.02	<0.02
French Drain (3)	NA	NA		NA	NA	NA	NA	NA	NA		NA	NA	NA	NA	0.02

Notes:

1. Monitoring Wells MW-1 and MW-1 were dry throughout this study because of their shallow construction and the pond had been drained.
2. Due to decommissioning work, the floor drainpipe was no longer a viable sampling point
3. Groundwater entered the French drain system beneath the building floor. Sampled from sump.

May 2003, the rainfall total was 24.8 inches or 5.54 inches above normal for the period from January through May 2003 (National Weather Service Forecast Office). Observation and testing holes placed though the facility floor in several locations were observed to collect water from the subsurface and water was observed flowing within the French drain system. This clearly demonstrates that during periods of high water table conditions, water would drain from the facility via the French drain system.

One of the pathways highlighted by the tracer study appears to follow the interface between the original ground surface and the fill added to produce the level parking area between the building and the pond. It is well documented in engineering and scientific literature that the interface between soil fills and native soils can become a preferred ground water flow path. This was evidenced here by the two seepage zones observed coming from the banks of the pond when the pond was drained (Figure 4). This area would be close to the outcrop of the interface between the fill and the native ground surface. Bromide was detected here early in the tracer study indicating a rapid flow path to these seepage zones. Seepage in this area appears to have decreased with the draining of the reactor pool even though rainfall levels were above normal.

Lastly, the floor drains in the lower area of the reactor building were tied into the reactor pool water-cooling system. A single valve separated the piping runs. During the conduct of the tracer study, bromide was very quickly detected in the pond discharge. The bromide was traced back to the outfall of the floor drain in the pond (Figure 4). Flow measurement of the pipe discharge indicated a flow rate of 20 gallons per day. This is not enough flow to account for the leakage rates observed, however, we do not have historical flow rates from this pipe. In any event, it is likely that this leakage was a contributing factor.

5.2 Soil Sample Radiological Analysis Results

Table 2 is a summary of the radionuclide analyses performed by the reactor staff of the soil samples taken during the installation of Boring SB-3 and SB-4/MW-3. With the exception of two samples from Soil Boring SB-3, these analyses revealed the presence of only naturally occurring radionuclides in the soil samples. The presence of Zr-97 is questionable due to the

short half-life of this isotope (17 h) and is likely a result of peak misidentification. Zn-65 is an activation product that has historically been detected in small concentrations in pond sediment; however, the amount reported in SB-3 was less than the MDA of 1.8 X 10^{-7} uCi/g for Zn-65.

Table 3 - Radiological Analysis Results on Soil

Well #	Core Depth	Results		Well #	Core Depth	Results
SB-3	Spoon sample	Zn-65: 9.2E-8		MW-3	Cuttings	K-40: 3.5E-5
	9-11 ft.	K-40: 3.6E-5			From	Ra-226: 1.3E-6
		Ra-226: 1.9E-6			Depth of	
		Pb-214: 2.2E-4			About 1 ft.	
		Bi-214: 1.2E-3				
SB-3	Spoon sample	K-40: 3.9E-5		MW-3	Spoon sample	K-40: 4.2E-5
	14-16 ft.	Ra-226: 8.6E-7			4-6 ft.	Ra-226: 1.4E-6
SB-3	Spoon sample	Zr-97: 6.1E-6		MW-3	Cuttings	K-40: 4.5E-5
	19-20 ft.	K-40: 4.1E-5			From	Ra-226: 1.4E-6
		Ra-226: 1.2E-6			depth of	U-235: 3.2E-7
					about 8 ft.	
SB-3	Cuttings	K-40: 4.0E-5		MW-3	Spoon sample	K-40: 4.8E-5
	From	Ra-226: 1.2E-6			9-11 ft.	Ra-226: 1.0E-6
	About 21 ft.					
				MW-3	Cuttings	K-40: 4.7E-5
					From a	Ra-226: 1.2E-6
					Depth of	
					About 13 ft.	
				MW-3	Spoon sample	K-40: 4.6E-5
					14-16 ft.	Ra-226: 1.0E-6
				MW-3	Cuttings	K-40: 5.1E-5
					From	Ra-226: 1.3E-6
					About 17 ft.	
				MW-3	Spoon sample	K-40: 4.7E-5
					19-21 ft.	Ra-226: 1.5E-6
				MW-3	Cuttings	K-40: 4.1E-5
					From	Ra-226: 1.2E-6
					About 28 ft.	

Note: E-n notation is equivalent to x 10^{-n}

5.3 Groundwater Radiological Analytical Results

Groundwater samples were collected for analysis from the monitoring wells on twelve occasions over the period beginning on May 1, 1997 and ending on February 4, 2003. The analysis results are presented in Appendix C. The only radionuclides detected in the water samples were cesium (Cs-137) and tritium (H-3). The Cs-137 activity in these samples was just slightly above the method detection concentration (MDC). Errors associated with individual results indicate the

presence of Cs-137 activity above the MDC is questionable. No gamma emitters other than naturally occurring K-40 have been detected in any water samples collected after 5/16/01. The presence of H-3 in groundwater downgradient of the reactor does however indicate that water released from the reactor was entering the local groundwater system.

Tritium, a known contaminant in the reactor pool water, has been detected in groundwater samples collected from wells MW-2 and MW-3. A typical H-3 concentration in pool water during operation was 3.5×10^{-4} uCi/ml. After shutdown, H-3 concentrations in the pool decreased steadily as a result of decay and dilution. Only well MW-3 has consistently yielded samples with H-3 activity above the method detection concentration (MDC). Activities detected have been very low; the highest at 9.6×10^{-6} uCi/ml on March 14, 2002, with an MDC of 1.2×10^{-6} uCi/ml. On average, the H-3 concentrations were three orders of magnitude below the release limit of 1×10^{-3} mCi/ml for this facility.

In summary, the only reactor-produced radionuclide consistently detected at levels above background has been H-3. The last analysis result of groundwater from well MW-3 in 2/2003 was 2.5×10^{-6} uCi/ml. This is an order of magnitude below the EPA MCL of 20,000 pCi/l (2×10^{-5} uCi/ml) for H-3 in drinking water. From a risk standpoint, this data shows that there is little evidence that the water that leaked from the reactor pool through leakage has adversely impacted the groundwater quality in the area surrounding the reactor building.

5.4. Local Groundwater Usage

One of the tasks in this study was to determine the uses of groundwater in the local area around the reactor facility. This task was accomplished by placing a 2,000-foot radius around the facility and identifying all water supply wells and springs within or near the delineated area (Figure 9). The 2,000 foot radius was chosen because it is the furthest distance that groundwater originating from the reactor area would likely travel before it surfaced and discharged to local streams. In most cases, the flow path is much shorter.

The University of Virginia and the Charlottesville area have been provided with public water for over 100 years. In fact, the first surface reservoir in the area was the pond at the UVA reactor. Currently the Rivanna Water and Sewer Authority supplies the region with potable water from three surface reservoirs including South Fork Rivanna River, Ragged Mountain, and Sugar Hollow. The reactor is not in the watershed of any of these reservoirs. Surface impoundments have been the public water source of choice because the local geology will not support large production wells. Smaller yielding individual wells are used for homes and businesses outside of the authority's service area. There are no water supply wells within 2,000 feet of the reactor facility. The closest well is a water supply backup well for Gilmer Hall at the university (Figure 9). This well system is designed to supply water to water-dependent laboratories in the event of a disruption in the public distribution system.

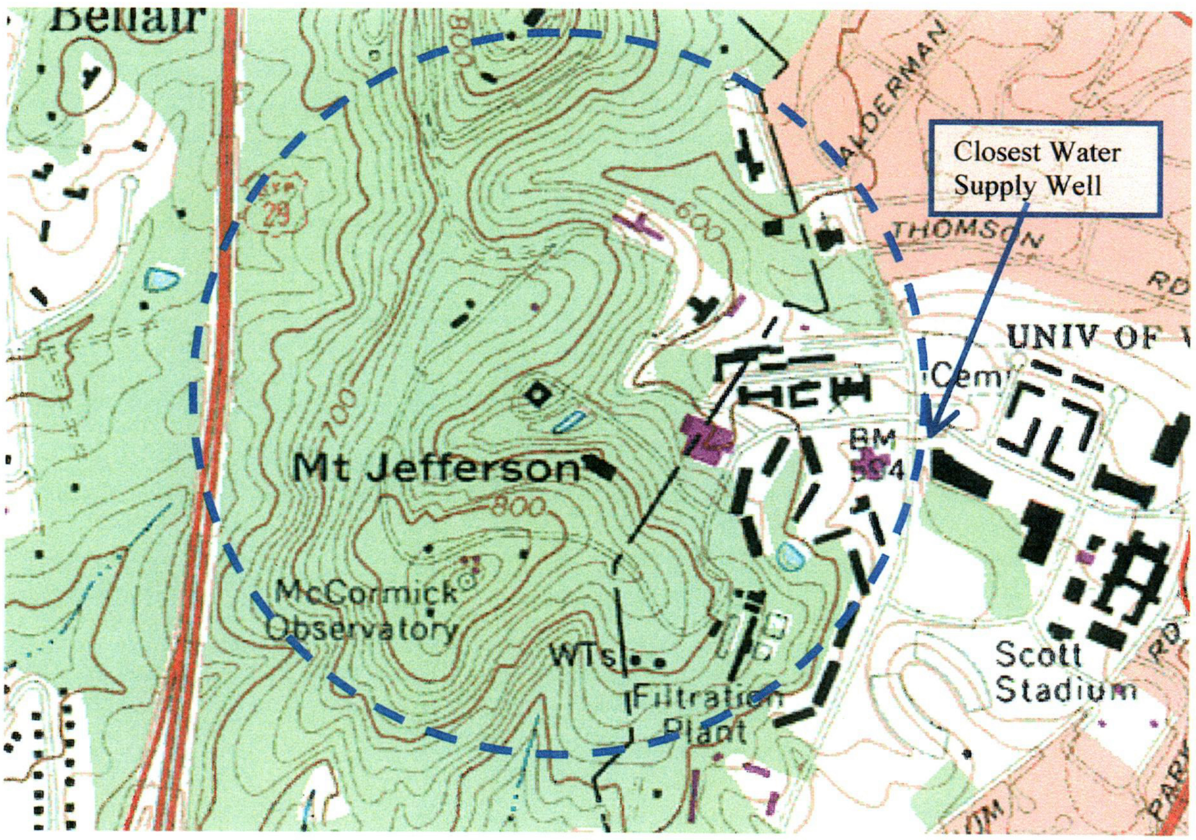


Figure 9. Water Supply Map showing 2,000-foot search radius around reactor facility.

C08

6.0 CONCLUSIONS

The data gathered during this investigation confirms that shallow groundwater is moving from the area of the reactor to the pond (former reservoir) where it discharges to the surface, therefore, leakage from the reactor ultimately discharged to the pond just below the facility. This investigation also showed that the leakage likely followed multiple flow paths from the reactor to the pond.

The water quality data clearly show that groundwater quality has not been impacted to a level of concern. On average the radiological analysis of the numerous groundwater samples show that the contaminant concentrations remained at least an order of magnitude below the EPA MCLs. Given this information, it can be stated that the releases to groundwater from this reactor represent a minimal to nonexistent threat to human health and the environment.

APPENDIX A
Soil Boring/Well Logs
And
Water Level Measurement Data

Groundwater Elevation Data

	MW-1	MW-2	MW-3	MW-4	MW-5	MW-1	MW-2	MW-3	MW-4	MW-5
Location	East well at pond	West well at pond	Back door in parking lot	Upgradient above facility	Fire hydrant next to dam	East well at pond	West well at pond	Back door in parking lot	Upgradient above facility	Fire hydrant next to dam
Date Installed	7/2/96	7/2/96	4/25/97	9/7/01	9/7/01	7/2/96	7/2/96	4/25/97	9/7/01	9/7/01
Casing Elevation (ft amsl)	652.76	651.22	659.76	682.57	658.36	652.76	651.22	659.76	682.57	658.36
Date of Measurement	Feet Below Top of Casing					Water Table Elevation (feet amsl)				
	MW-1	MW-2	MW-3	MW-4	MW-5	MW-1	MW-2	MW-3	MW-4	MW-5
5/1/97	8.37	6.79	5.22			644.39	644.43	654.54		
8/7/97	8.17	7.14	6.25			644.59	644.08	653.51		
3/1/99	7.07	5.51	7.74			645.69	645.71	652.02		
4/12/00	7.04	9.12	8.85			645.72	642.10	650.91		
9/1/00	10.27	10.00	7.73			642.49	641.22	652.03		
5/14/01	11.50	10.00	8.21			641.26	641.22	651.55		
5/16/01	11.50	10.00				641.26	641.22			
10/11/01	7.35	5.60	9.16	19.93	15.48	645.41	645.62	650.60	662.64	642.88
3/14/02	6.90	4.60	8.32	22.35	14.62	645.86	646.62	651.44	660.22	643.74
6/18/02	8.20	5.90	9.17	23.57	15.79	644.56	645.32	650.59	659.00	642.57
8/19/02	11.50	10.00	11.72	24.97	18.62	641.26	641.22	648.04	657.60	639.74
9/18/02	dry	dry	12.64	25.76	19.12			647.12	656.81	639.24
2/4/03	dry	dry	13.09	24.44	18.21			646.67	658.13	640.15
4/15/03	dry	dry	8.93	12.03	16.58			650.83	670.54	641.78
5/15/03	dry	dry	7.37	5.85	16.91			652.39	676.72	641.45

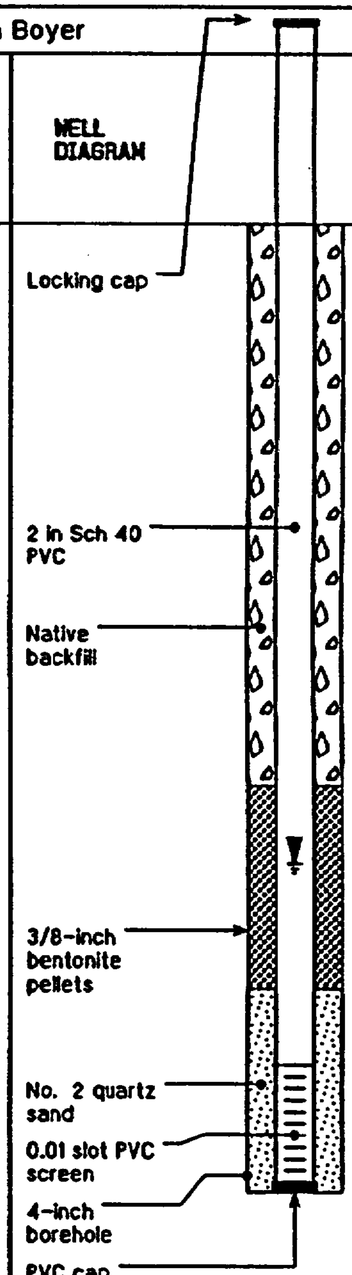
APPENDIX B
Slug Test Data

APPENDIX C
Radiological Analysis Results on Groundwater

UVA - ENVIRONMENTAL HEALTH AND SAFETY	LOG OF MW-1	Page 1 of 1

PROJECT: Groundwater Monitoring	**LOCATION:** University of Virginia
DATE DRILLED: 7/2/96	**TOP OF CASING ELEVATION:** 600 Feet
DRILLING METHOD: Small Rotary Auger	**TOTAL DEPTH:** 9.56 Feet
DRILLING COMPANY: Department of Env. Sciences	**GEOLOGIST:** Beth Boyer

DEPTH feet	SAMPLE NUMBER	BLOWS	OVA (ppm) VALUES	OVA (ppm) PROFILE	GRAPHIC LOG	GEOLOGIC DESCRIPTION	WELL DIAGRAM
5							
10							

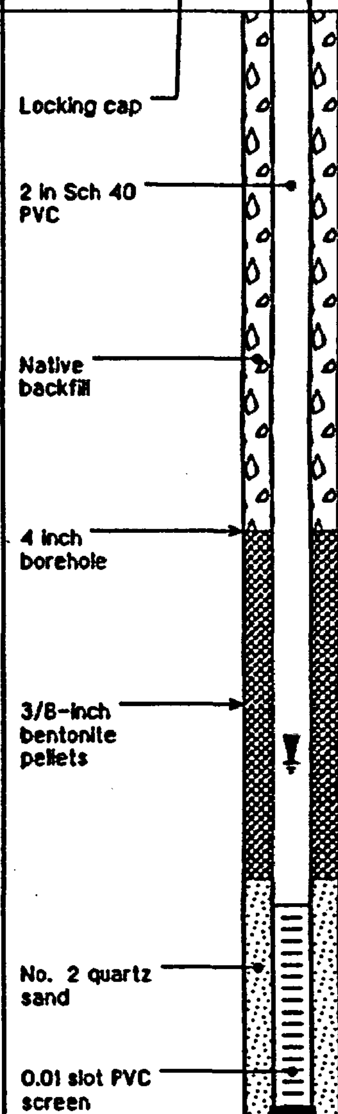


JOB NUMBER: UVA Reactor

UVA - ENVIRONMENTAL HEALTH AND SAFETY	LOG OF MW-2	Page 1 of 1

PROJECT: Groundwater Monitoring	**LOCATION:** University of Virginia
DATE DRILLED: 7/2/96	**TOP OF CASING ELEVATION:** 598.42 Feet
DRILLING METHOD: Small Rotary Auger	**TOTAL DEPTH:** 6.4 Feet
DRILLING COMPANY: Department of Env. Sciences	**GEOLOGIST:** Beth Boyer

DEPTH feet	SAMPLE NUMBER	BLOWS	OVA (ppm) VALUES	OVA (ppm) PROFILE	GRAPHIC LOG	GEOLOGIC DESCRIPTION	WELL DIAGRAM
5							



JOB NUMBER: UVA Reactor

UVA – ENVIRONMENTAL HEALTH AND SAFETY

LOG OF SB-3

PROJECT: Ground Water Monitoring	**LOCATION:** University of Virginia
DATE DRILLED: 4/25/97	**TOP OF CASING ELEVATION:** 659 Feet
DRILLING METHOD: Hollow-Stem/split spoon	**TOTAL DEPTH:** 21 Feet
DRILLING COMPANY: Certified Environmental Drilling	**GEOLOGIST:** Jeffrey A. Sitler

DEPTH feet	SAMPLE NUMBER	BLOWS	OVA (ppm) VALUES	OVA (ppm) PROFILE	GRAPHIC LOG	GEOLOGIC DESCRIPTION	WELL DIAGRAM
5	SS-1	2,3, 5,9				**FILL.** Gray micaceous fill, sandy silt with upto 5% gravel, very friable to granular, moist	
10	SS-2	2,2 2,1				**FILL and Native Soil.** Gray micaceous fill in top half of sample and original top soil horizon in bottom half (black loam layer, followed by a brown clayey silt, moist, very friable)	
15	SS-3	45, Refusal				**SAPROLITE.** Very hard saprolite or soft bedrock, moist, gray-brown silt, firm to friable, spoon refusal after six inches	
20	SS-4	50, Refusal				**BEDROCK.** Spoon refusal at 19 feet and auger refusal at 21 feet	
						Bedrock refusal before water was encountered, boring checked for water later and found to be dry. Well was not constructed.	
25							

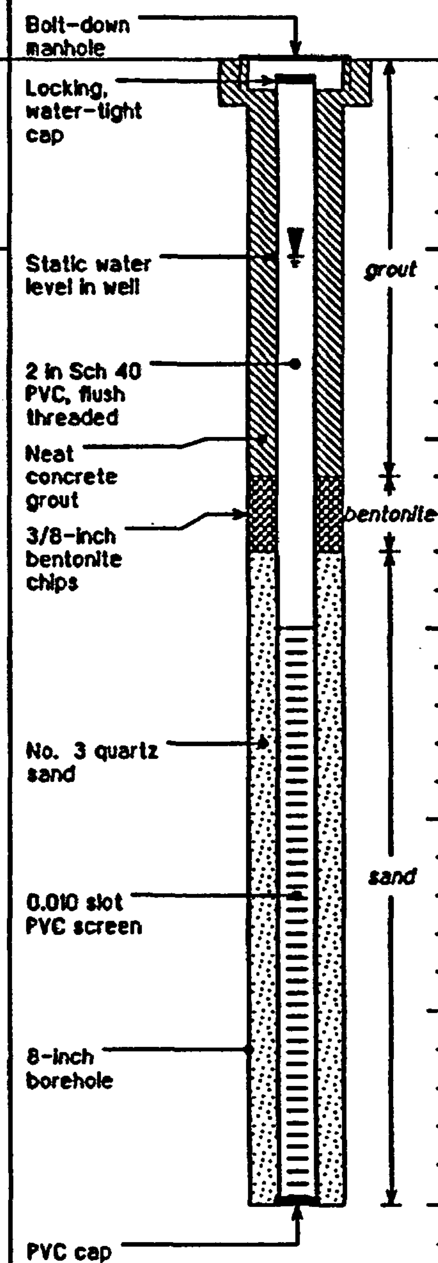
JOB NUMBER: UVA Reactor

UVA - ENVIRONMENTAL HEALTH AND SAFETY	LOG OF SB-4, MW-3	Page 1 of 1

PROJECT: Ground Water Monitoring	**LOCATION:** University of Virginia
DATE DRILLED: 4/25/97	**TOP OF CASING ELEVATION:** 660 Feet
DRILLING METHOD: Hollow-Stem/split spoon	**TOTAL DEPTH:** 30 Feet
DRILLING COMPANY: Certified Environmental Drilling	**GEOLOGIST:** Jeffrey A. Sitler

DEPTH feet	SAMPLE NUMBER	BLOWS	OVA (ppm) VALUES	OVA (ppm) PROFILE	GRAPHIC LOG	GEOLOGIC DESCRIPTION
0–5						FILL, Lt. orange-brown clayey silt and gravel, fill mateiral under asphalt
5	SS-1					
						SAPROLITE, Gray micaceous clayey silt saprolite, moist, friable
10	SS-2					SAPROLITE, Gray micaceous clayey silt saprolite, moist, very friable
15	SS-3					SAPROLITE, Gray and brown micaceous sandy silt saprolite, wet, very friable
20	SS-4					SAPROLITE, Gray and brown micaceous sandy silt saprolite, iron staining on partings and joints
25						Boring terminated at 30 feet and well constructed. Bedrock was not encountered
30						

WELL DIAGRAM



JOB NUMBER: UVA Reactor

UNIVERSITY of VIRGINIA

Office of Environmental Health and Safety
University of Virginia
P.O. Box 400322
Charlottesville, VA 22904-4322

Log of Borehole and Well: MW-4

Project No.: UVA Reactor Decommissioning

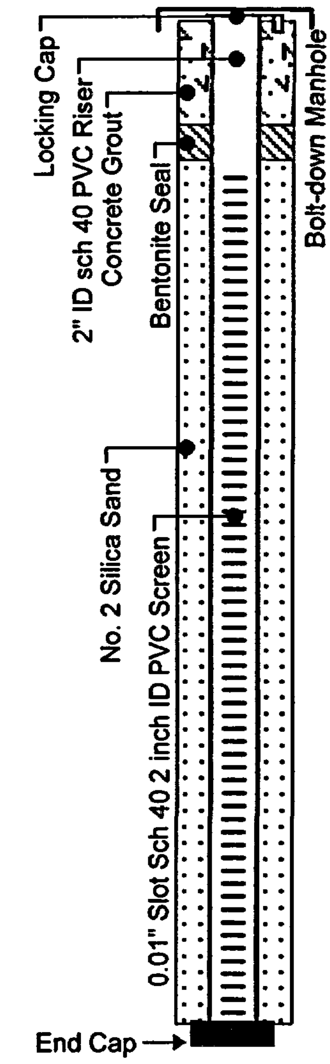
Project: Ground Water Characterization

Client: Reactor ***Project Manager:*** Jeffrey Sitler

Location: Upper Parking Lot at Outcrop (Site upgradient well)

SUBSURFACE PROFILE				SAMPLE				
Depth (ft / m)	Symbol	Description	Depth/Elev.	Number	Type	Recovery	Vapour	Well Completion Details
		Ground Surface	683.1					
0		***Saprolite*** Tan to gray, micaceous silt, dry to damp to 57 feet. Very easy drilling.	0.0					
20		***Static Water Level*** *Water level on 10-11-2001 - 19.93 feet*	663.1 / 19.9					
55		***Water*** *First show of water at 57 feet.*	628.1 / 55.0					
60		*Drilling stopped at 60 feet. Bedrock not encountered.*	623.1 / 60.0					

Depth scale: ft 0–64; m 0–18



Drilled By: Certified Environmental Drilling ***Hole Size:*** 6 inch

Drill Method: Air Rotary ***Datum:*** NAVD 88

Drill Date: 9/7/01 ***Sheet:*** 1 of 1

UNIVERSITY of VIRGINIA

Office of Environmental Health and Safety
University of Virginia
P.O. Box 400322
Charlottesville, VA 22904-4322

Log of Borehole and Well: MW-5

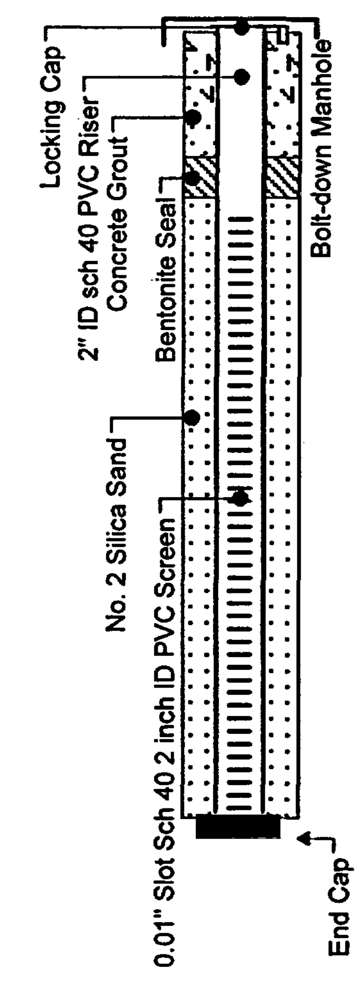
Project No.: UVA Reactor Decommissioning

Project: Ground Water Characterization

Client: Reactor ***Project Manager:*** Jeffrey Sitler

Location: North West of Dam (site downgradient well)

SUBSURFACE PROFILE				SAMPLE				Well Completion Details
Depth (ft / m)	Symbol	Description	Depth/Elev.	Number	Type	Recovery	Vapour	
		Ground Surface	658.9					
0		***Saprolite and Fill*** Tan to gray, micaceous silt, dry to damp to mixture of saprolite and fill. Very easy drilling.	0.0					
12		***Bedrock*** Rockfish conglomerate - meta sandstone - tan, micaceous.	646.9 / 12.0					
14.5		***Static Water Level*** *Static water level of 15.48 feet on 10-11-2001*	644.4 / 14.5					
40		***Caving borehole*** *Borehole collapsed to 40 feet when drill stem withdrawn. Well casing set at 40 feet. First show of water at 42 feet*	618.9 / 40.0					
50			608.9 / 50.0					



Drilled By: Certified Environmental Drilling

Drill Method: Air Rotary

Drill Date: 9/7/01

Hole Size: 6 inch

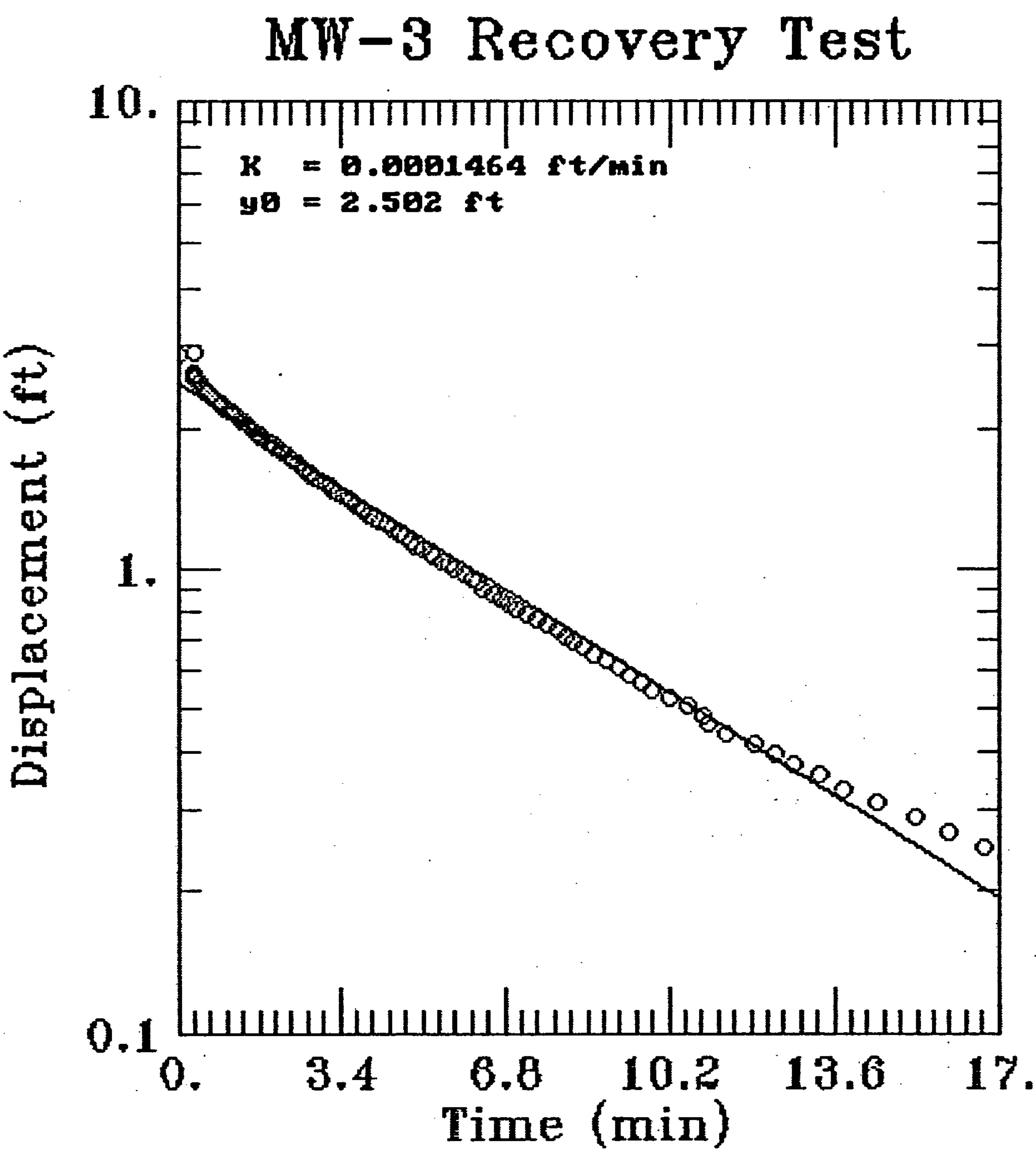
Datum: NAVD 88

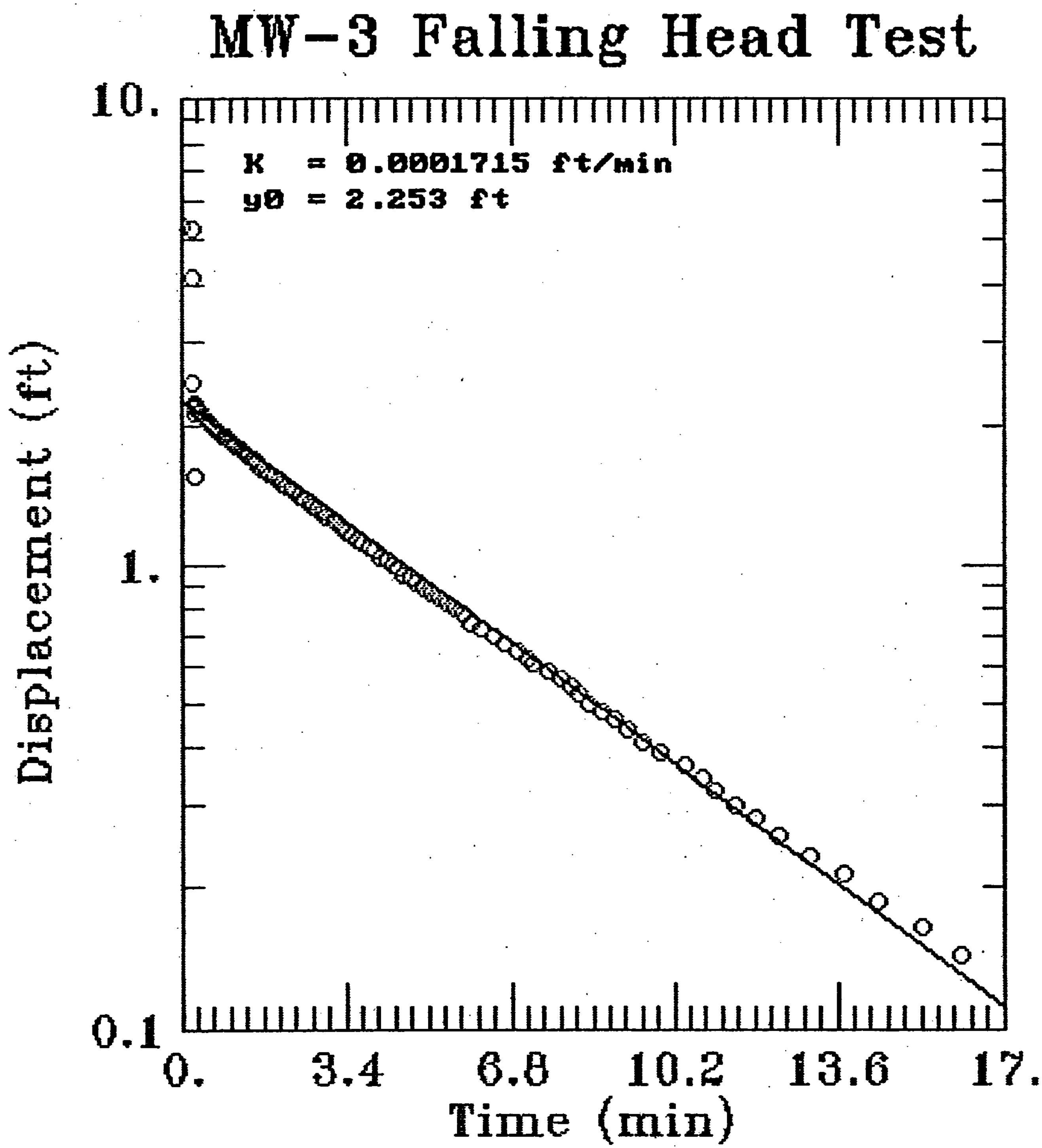
Sheet: 1 of 1

Groundwater Elevation Data

	MW-1	MW-2	MW-3	MW-4	MW-5	MW-1	MW-2	MW-3	MW-4	MW-5
Location	East well at pond	West well at pond	Back door in parking lot	Upgradient above facility	Fire hydrant next to dam	East well at pond	West well at pond	Back door in parking lot	Upgradient above facility	Fire hydrant next to dam
Date Installed	7/2/96	7/2/96	4/25/97	9/7/01	9/7/01	7/2/96	7/2/96	4/25/97	9/7/01	9/7/01
Casing Elevation (ft amsl)	652.76	651.22	659.76	682.57	658.36	652.76	651.22	659.76	682.57	658.36
Date of Measurement	Feet Below Top of Casing					Water Table Elevation (feet amsl)				
	MW-1	MW-2	MW-3	MW-4	MW-5	MW-1	MW-2	MW-3	MW-4	MW-5
5/1/97	8.37	6.79	5.22			644.39	644.43	654.54		
8/7/97	8.17	7.14	6.25			644.59	644.08	653.51		
3/1/99	7.07	5.51	7.74			645.69	645.71	652.02		
4/12/00	7.04	9.12	8.85			645.72	642.10	650.91		
9/1/00	10.27	10.00	7.73			642.49	641.22	652.03		
5/14/01	11.50	10.00	8.21			641.26	641.22	651.55		
5/16/01	11.50	10.00				641.26	641.22			
10/11/01	7.35	5.60	9.16	19.93	15.48	645.41	645.62	650.60	662.64	642.88
3/14/02	6.90	4.60	8.32	22.35	14.62	645.86	646.62	651.44	660.22	643.74
6/18/02	8.20	5.90	9.17	23.57	15.79	644.56	645.32	650.59	659.00	642.57
8/19/02	11.50	10.00	11.72	24.97	18.62	641.26	641.22	648.04	657.60	639.74
9/18/02	dry	Dry	12.64	25.76	19.12			647.12	656.81	639.24
2/4/03	dry	Dry	13.09	24.44	18.21			646.67	658.13	640.15
4/15/03	dry	Dry	8.93	12.03	16.58			650.83	670.54	641.78
5/15/03	dry	Dry	7.37	5.85	16.91			652.39	676.72	641.45

APPENDIX B
Slug Test Data


MW-3 Recovery Test
K = 0.0001464 ft/min
y0 = 2.502 ft
10.
1.
0.1
Displacement (ft)
0.
3.4
6.8
10.2
13.6
17.
Time (min)



MW-3 Falling Head Test
K = 0.0001715 ft/min
y0 = 2.253 ft
Displacement (ft)
10.
1.
0.1
0.
3.4
6.8
10.2
13.6
17.
Time (min)

APPENDIX C
Radiological Analysis Results on Groundwater

Radiological Analysis Results for Groundwater at the UVA Research Reactor

Sample Date	Well #	Gamma K-40 µCi/ml	Ra-226 µCi/ml		µCi/ml	H-3 µCi/ml	Analyzed By	Comments
5/1/97	MW-1	No Data				< 7 E-7	UVAR/peb	8 hr counts
	MW-2	No Data				< 7 E-7	UVAR/peb	
	MW-3	1.7 E-5	5.5 E-7			< 7 E-7	UVAR/peb	
5/1/97	MW-1	1.7 E-5	3.1 E-7	Cs-137	2.3 E-8		UVAR/peb	
re-analysis	MW-2	1.7 E-5	2.7 E-7				UVAR/peb	
	MW-3	1.7 E-5	3.4 E-7				UVAR/peb	
8/7/97	MW-1	1.7 E-5	3.0 E-7			< 7 E-7	EHS	gamma by peb
	MW-2	1.7 E-5	4.0 E-7	(Cd-109) 1	6.1 E-7	8.7 E-7	EHS	see note
	MW-3	1.7 E-5	2.4 E-7			< 7 E-7	EHS	
8/7/97	BKG.	1.8 E-5	2.30E-07			1.07 E-6	EHS	distilled H20
2/26/98	MW-1	< 6 E-8	< 1.0 E-7			< 3 E-7	Teledyne	gamma analysis
	MW-2	< 8 E-8	< 8 E-8	Cs-137	8.41 E-9	6.6 E-7	Teledyne	by Teledyne
	MW-3	< 1 E-7	< 8 E-8			< 3 E-7	Teledyne	
3/8/99	MW-1	< 8 E-8				< 7 E-7	EHS	
	MW-2	< 5 E-8				5.7 E-7 3.4 E-7	EHS Teledyne	
	MW-3	< 1 E-7				< 7 E-7	EHS	
7/30/99	MW-1	Dry/No Data						
	MW-2	Dry/No Data						
	MW-3	< 8 E-8	< 6 E-8	GrossBeta	2.3 E-8			
4/12/00	MW-1	< 4 E-8	< 7 E-8			< 7 E-7 < 2 E-7	EHS Teledyne	gamma analyses
	MW-2	3.76 E-8	< 6 E-8			< 7E-7	EHS	by Teledyne
	MW-3	< 7 E-7	< 6 E-7			7.1 E-6	EHS	
9/1/00	MW-1	< 7.8 E-8	< 8.8 E-9			< 7 E-7 < 2 E-6	EHS Teledyne	gamma analyses
	MW-2	dry						by Teledyne
	MW-3	< 6.6 E-8	< 8.9 E-9			8.3 E-6 7.1 E-6	EHS Teledyne	

Sample Date	Well #	Gamma K-40 µCi/ml	Ra-226 µCi/ml		µCi/ml	H-3 µCi/ml	Analyzed By	Comments
5/16/01	MW-1		3.1 E-7	Cs-137	1.1 E-8	4.5 E-6	EHS/PEB	
	MW-2		3.7 E-7			6.5 E-6	EHS	
	MW-3		2.85 E-7	Cs-137	1.3 E-8	3.9 E-6	EHS	
10/11/01	MW-1	2 E-8				4.7 E-6 < 8.6 E-7	EHS Duke	gamma analyses
	MW-2	< 6.6 E-8				5.4 E-6 1.04 E-6	EHS Duke	by Duke Engin.
	MW-3	< 8.8 E-8				1.1 E-5 5.28 E-6	EHS Duke	
	MW-4	4.2 E-8				4.4 E-6 < 8.6 E-7	EHS Duke	new well first sample
	MW-5	2.8 E-8				4.8 E-6 < 8.6 E-7	EHS Duke	new well first sample
3/14/02	MW-1	2.5 E-8				4.9 E-7	Framatome	
	MW-2	< 4 E-8				6.3 E-7	Framatome	
	MW-3	< 5.5 E-8				9.6 E-6	Framatome	
	MW-4	< 4 E-8				2.4 E-7	Framatome	
	MW-5	< 4 E-8				3.6 E-7	Framatome	
6/18/02	MW-1	3.4 E-8		Ac/Th-228	' - 2.2 E-7	- 6 E-8	Framatome	
	MW-2	1 E-9				4.1 E-7	Framatome	
	MW-3	1.01 E-7				7.24 E-6	Framatome	
	MW-4	8.7 E-8				-1.6 E-7	Framatome	
	MW-5	2.7 E-8				6 E-8	Framatome	
9/18/02	MW-1	no sample						pond drained/dry
	MW-2	no sample						pond drained/dry
	MW-3	3.1 E-8				3.1 E-6	Framatome	
	MW-4	3.9 E-8				2.6 E-7	Framatome	
	MW-5	< 6 E-8				1.9 E-7	Framatome	
2/4/03	MW-1	no sample						pond drained/dry
	MW-2	no sample						pond drained/dry
	MW-3	- 2.6 E-8				4.6 E-6 2.5 E-6	EHS Framatome	
	MW-4	2.3 E-8				1.4 E-6 - 2.2 E-7	EHS Framatome	
	MW-5					1.2 E-6		

Notes: 1) review of this data indicates that the data analysis program incorrectly identifed this as Cd when in fact if was most likely Pb-214/212