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June 6, 2000
Docket No. 50-62

Mr. Alexander Adams, Jr., Senior Project Manager
Non-power Reactors and Decommissioning Project Directorate
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission M.S. 0-11-D-19
Rockville, MD 20852-2738

Subject: Documentation in Support of the University of Virginia Reactor Decommissioning Plan
(Docket No. 50-62, License R-66).

Dear Mr. Adams,

In response to a phone conversation I had today with your associate Mr. Marvin Mendonca, we are at this time submitting the University of Virginia Reactor Facility Characterization Report in its entirety. Previously, we had submitted a summary of this report. Should the reviewers of the UVAR Facility Decommissioning Plan require any of the field data on which the characterization report is based, we will be happy to provide that also.

It is likely that you have been briefed about the news that South Carolina's governor has signed legislation for his state to join the Atlantic Compact in a waste disposal agreement. By joining this compact, South Carolina can accept shipments only from the other two states in the compact, New Jersey and Connecticut. It is still unclear whether South Carolina will abruptly stop receiving low level radioactive waste for burial at Barnwell, S.C. on July 1, or phase in a gradual stop to these waste shipments. In either case, the University of Virginia is concerned about the uncertainty created with regard to the orderly and timely decommissioning of its Reactor Facility, and again thanks the NRC for helping to expedite the review of the decommissioning plan.

Sincerely,

Robert U. Mulder, Director
UVa. Reactor Facility &
Assoc. Prof. of Nuclear Eng.

cc: Mr. Craig Basset, NRC Region II, Atlanta, Ga.
Document Control Desk, NRC, Washington

City/County of Albemarle
Commonwealth of Virginia

I hereby certify that the attached document is a true and exact copy of a letter, presented before
(type of document)

me this 8th day of June, 19 2000.
by Robert Mulder
(name of person seeking acknowledgement)

Wickie S. Thomas
Notary Public

My commission expires 2/28 19 2002

A020

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Date: Wed, 7 Jun 2000 19:20:16 PDT
From: AP <C-ap@clari.net>
Newsgroups: clari.tw.nuclear, clari.local.connecticut,
clari.local.south_carolina
Subject: SC Clear To Join Nuke Waste Compact

COLUMBIA, S.C. (AP) -- South Carolina no longer will be the final resting place for most of the nation's low-level nuclear waste, Gov. Jim Hodges said Wednesday as he signed legislation to join New Jersey and Connecticut in a disposal agreement.

The first-term Democrat had pledged to end what he said was the state's reputation as the ``nation's nuclear dumping ground.''

A landfill in Barnwell County now takes nuclear waste from throughout the nation.

South Carolina withdrew from the Southeast Compact Commission in 1995 after then-Gov. David Beasley said he was fed up with North Carolina's failure to build a replacement for the Barnwell landfill. South Carolina reopened Barnwell to all other states, intending to use the increased revenue for education, but the amount of money never reached projections.

New Jersey and Connecticut are expected to grant South Carolina's petition to join the Atlantic Compact, said John Clark, a senior aide to Hodges. The state would be admitted July 1.

By joining the compact, South Carolina can accept shipments from only those two states also in the compact.

With about 3 million cubic feet of storage remaining, state officials say the Barnwell site would fill in eight to 10 years if waste kept coming at current rates.

The new law limits New Jersey's and Connecticut's disposal space to a total of 800,000 cubic feet to ensure enough space for waste generated when South Carolina's nuclear power plants are decommissioned in about 30 years.

GTS DURATEK
CHARACTERIZATION SURVEY REPORT

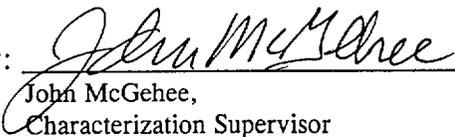
for the

University of Virginia
Reactor Facility

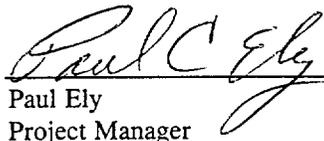
REVISION 0
MARCH 2000

Prepared by: GTS Duratek

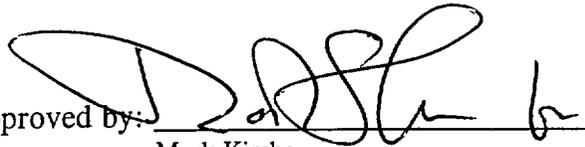
Date 3/10/2000

Reviewed by: 
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Characterization Supervisor

Date 3/10/2000

Reviewed by: 
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Date 3/10/2000

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Mark Kirshe
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Date 3/15/00

Reviewed and
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PREFACE

In June of 1999, GTS Duratek was awarded a contract to assist the University of Virginia (UVA) in its efforts to decontaminate and decommission (D&D) the University of Virginia Reactor (UVAR). The initial D&D activities requiring assistance included the development of a Decommissioning Plan and a Decommissioning Cost Estimate. To accurately assess the methods and costs associated with the decommissioning, it was important that the levels and characteristics of radioactive and non-radioactive hazardous constituents be known. In order to determine these levels and characteristics, GTS Duratek performed a characterization survey of the UVAR and the surrounding buildings and environs.

This document reports the results of the sampling and measurements collected by GTS Duratek during the UVAR facility characterization to provide survey data to support the development of a Decommissioning Plan.

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 Appendix C, UVA Reactor Facility Characterization Survey Packages

1.0 EXECUTIVE SUMMARY

The University of Virginia Reactor (UVAR) was permanently shutdown in June of 1998 following the decision to cease operations and to decontaminate and decommission (D&D) the facility. GTS Duratek (GTSD) was awarded a contract in June of 1999 to assist the University of Virginia (UVA) in its efforts to D&D the UVAR facility and site. GTSD was to assist UVA with development of a Decommissioning Plan and a Decommissioning Cost Estimate. This required performing a characterization survey of the UVAR and the surrounding buildings and environs to accurately assess the methods and costs associated with the decommissioning. The UVAR fuel was shipped offsite prior to beginning the site characterization July 1999.

The characterization entailed performing a site investigation process, preparing plans and performing a comprehensive radiation survey. The survey included sampling for potential hazardous materials at the facility and site following guidance provided in NUREG-1575, MARSSIM. The site investigation involved a site walk-down and historical site assessment, from which information such as the facility source term and operational history for the facility was gained. Based upon the information obtained from the site investigation, plans detailing the characterization survey were developed such as Characterization Survey Plan, Quality Assurance Plan, Site Health and Safety Plan and the Project Management Plan. The plans were developed by GTSD staff from the Radiological Engineering and Field Services office in Kingston Tennessee and reviewed and approved for use at the project site. Data quality objectives DQO's were stated in the plans and the characterization of the UVAR facility and site was performed according to these plans and the procedures referenced in the plans. The characterization survey achieved the planned DQO objectives.

Part of planning was to develop derived concentration guideline levels (DCGL's) for the radionuclides assumed to be present at the site for the purpose of instrument selection and to determine instrument operating parameters. These characterization DCGL's were developed to meet the intent of Subpart E, 10 CFR 20.1402, *Radiological Criteria for Unrestricted Use*. An estimate of actual site DCGL's was essential to develop parameters for instrument operation so that the level of sensitivity for radiation measurements taken during the survey would be less than 50% of the characterization DCGL's.

Once the planning phase was completed, the GTSD survey team mobilized to the site. At the site, GTSD and UVA personnel had a kick-off meeting and moved equipment into the accommodations made available by the UVA staff for the survey operation. All required training was completed for GTSD personnel for unescorted access to the UVAR facility. Next, an equipment set-up was completed. Over the course of eight and one-half weeks, eighteen survey packages describing rooms and areas of the UVAR facility and site surfaces, structures and environs were developed. Each survey package contained detailed instructions, drawings and location codes to perform 2,655 measurements and/or sample collections and analyze them for the characterization.

The measurements and samples collected for the radiological characterization survey consisted of direct alpha measurements, direct beta measurements, smear samples for removable alpha and beta contamination, smear samples for removable tritium contamination, exposure rate measurements and soil, sediment and water samples for gamma spectrum analysis. During the hazardous material assessment, survey package instructions were developed and measurements were performed and/or collected for lead and asbestos at the facility. Samples were analyzed for potential hazardous material constituents present at the site. The hazardous material assessment was partly for recognition of hazardous materials that may be present for the D&D phase of work, work safety considerations and also to meet requirements for waste shipments to Envirocare of Utah.

Survey package instructions were developed and measurements performed for background reference areas for surface, structures and the environs. Direct beta background values for naturally occurring radioactive material (NORM) in asphalt, brick, ceramic tile, cinder block and concrete were determined and applied to the direct beta measurements collected at the UVAR facility surfaces and structures to equate the reported beta activity to net results per 100 cm². Also, survey measurements and samples collected from the environs were analyzed to determine background values for NORM and the contribution from activity from weapons testing fallout in soil, sediment and water of the environs. These values are stated in the report but were not applied to results.

Survey package instructions were developed and samples collected and sent to offsite laboratories for 10 CFR Part 61 radionuclide analysis. One resin sample from the UVAR pool water clean up system and one sample of pond sediment where the facility drains enter the pond were taken. These sample results, in addition to other onsite characterization data, were used to determine UVAR site-specific DCGL for clean-up goals and for release of the facility and surrounding grounds for unrestricted use.

The findings of the site characterization were that the UVAR facility and site, for the most part, were radiologically below the characterization release criteria and mostly free of hazardous material contamination concerns. However, some areas and locations at the UVAR facility and site will require decontamination before the site is suitable for release from the NRC license currently in effect. These areas are described below.

Of the 2,655 total samples and measurements collected or performed for the survey, 1,142 of them were direct measurements collected for the radiological characterization. Twelve (12) direct beta measurement results were greater than the MDA goal of less than 50 % of the characterization DCGL (MDA \leq 5,000 dpm/100 cm²) and five (5) of these direct beta measurements were greater than the characterization DCGL of 9,250 dpm/100 cm². Of the 198 samples collected for gamma spectrum analysis, 23 sample results were greater than the characterization soil concentration DCGL's plus background of 1.91 pCi/g for Co-60 and/or 1.32 pCi/g for Cs-137.

The areas that will require remediation or further investigation and evaluation are presented in the discussions that follow. The areas, where the elevated measurements are located, are depicted in Figures 1-1, 1-2, 1-3 and 1-4 by number on each figure.

1. On Figure 1-1, the Reactor Confinement Interior, one measurement result was 12,593 dpm/100 cm² located on the East wall. However, this result may have been influenced by elevated background radiation levels due to radioactive materials stored in the area. The radioactive materials stored in the reactor confinement room during the period this characterization survey was performed may be removed and disposed of as radioactive waste prior to or as a part of the facility decommissioning. Measurement results from the three floor drains in the confinement room showed contamination levels ranging to 6,398 dpm/100 cm² in floor drain No.2 on the east side of the reactor pool.

2. On Figure 1-1, the Reactor Confinement Interior, the reactor pool will require remediation based upon the operational history, radioactive materials known to be present in the pool and activation products from reactor operations in the concrete walls and floor.

3. On Figure 1-1, the Reactor Confinement Interior, a composite sediment sample, collected from the exhaust at roof level of the confinement building stack during the building exterior surfaces survey, measured 0.8 pCi/g of Co-60 and 2.8 pCi/g of Cs-137 contamination.

4. On Figure 1-2, the Reactor Facility Mezzanine Level, in room M008 one direct measurement result for total beta activity was 26,365 dpm/100 cm² in a laboratory sink. The contaminant was suspected to be nickel-63.

5. On Figure 1-2, the Reactor Facility Mezzanine Level, in room M021A one direct measurement result for total beta activity was 8,318 dpm/100 cm² on the equipment surface of the reactor pool water clean up system. However, this surface contamination measurement result may have been influenced by elevated background radiation levels from the water clean up system's internal contamination. The water clean-up system tanks, pumps and piping and the prototype water clean up system tank and piping located in the adjacent room M021 will be removed and disposed of as radioactive waste as a part of the facility decommissioning once the reactor pool has been drained.

6. On Figure 1-3, the Reactor Facility Ground Level, an NRC approved decommissioning plan is in place for the CAVALIER Reactor and supporting systems located in room G007. The CAVALIER Reactor will be decommissioned in accordance with the approved plan. During the UVAR facility and site characterization, surveys were conducted for the surfaces in room G007 and G007A. The measurement results from the survey indicate that room G007 (excluding the CAVALIER Reactor, supporting systems, the sub-critical assembly and pit) was radiologically clean. Measurements performed on the sub-critical assembly and pit were inconclusive due to naturally occurring radon progeny contamination. The sub-critical facility will be shipped to DOE prior to the start of CAVALIER and UVAR decommissioning.

7. On Figure 1-3, the Reactor Facility Ground Level, the Hot Cell rooms G026 and G027 each had one direct measurement result for total beta activity greater than 9,250 dpm/100 cm². Results of measurements collected ranged to 63,661 dpm/100 cm² on the floor in room G026 and results from measurements collected ranged to 19,268 dpm/100 cm² on the floor in room G027. Results from a concrete core bore slice analyzed by gamma spectral analysis indicated Cs-137 contamination ranging to 7,650 pCi/g in the first 1/4" depth at the location in room G026.

8. On Figure 1-3, the Reactor Facility Ground Level, the sump in the heat exchanger room G024 had a gamma spectrum analysis sample result indicating facility-derived radionuclide activity above the characterization DCGL's for Co-60 and Cs-137. The sump in this room will require decontamination while the heat exchanger system, pumps and piping in the room G024 will be removed as radioactive waste as a part of the facility decommissioning once the reactor pool has been drained.

9. On Figure 1-3, the Reactor Facility Ground Level, in room G028 one direct measurement result for total beta activity was 8,246 dpm/100 cm² on the east wall half the way between rooms G024 and G025 and one direct measurement result for total beta activity was 7,522 dpm/100 cm² on the north wall outside of room G022. However, these surface contamination measurement results may have been influenced by elevated background radiation levels from radioactive materials stored in the area during the time of the survey. Rooms G007, G007A, G018, G019, G022 and G028 were used for and posted as radioactive material storage areas. The radioactive materials, stored in the rooms during the period this characterization survey was performed, may be removed and disposed of as radioactive waste prior to or as a part of the facility decommissioning.

10. On Figure 1-3, the Reactor Facility Ground Level, one direct measurement result for total beta activity was 5,758 dpm/100 cm² on the beam tube pipe end that was oriented toward the reactor on the movable shield plug for the Southeast Access Facility. However, this measurement result may have been influenced by elevated background radiation levels from radioactive materials stored in room G022 during the time of the survey. A gamma scan survey profiling the North and South Neutron Beam Ports in conjunction with knowledge of the operational history indicates the ports may be removed and disposed of as radioactive waste as a part of the facility decommissioning once the reactor pool is drained.

11. On Figure 1-4, the Reactor Facility Grounds, the Buried Liquid Waste Holding Tank bunker dirt floor and interior of the tanks had gamma spectrum analysis results above the characterization DCGL's for Co-60 and Cs-137. The radionuclide contamination was present in the dirt floor to the 30" depth sampled. Two direct measurement results for total beta activity, one each in each of the tanks, were 5,541 dpm/100 cm² in the east tank and 6,600 dpm/100 cm² in the west tank.

12. On Figure 1-4, the Reactor Facility Grounds, the Hot Cell Buried Waste Holding Tanks had direct measurements for total beta activity ranging to 16,907 dpm/100 cm² in the piping leading to the tanks. A sample of sediment collected from inside the pipe for qualitative

gamma spectral analysis indicated the presence of Cs-137. Five (5) water and sediment samples were collected from the contents in the tanks for gamma spectral analysis and one result indicated the presence of Cs-137 activity above the characterization DCGL.

13. On Figure 1-4, the Reactor Facility Grounds, a pond sediment sample collected and sent offsite for 10 CFR Part 61 analysis indicated the presence of Co-60 and Cs-137 activity above the characterization DCGL. The sample was a composite of surface sediment collected around the vertical pipe inlet to the pond from the facility.

14. On Figure 1-4, the Reactor Facility Grounds, in the outfall area between the buried liquid waste holding tank bunker and the pond, approximately half way up the bank under the drain pipe from the bunker, the 0-6" and 6-12" depth soil samples collected for gamma spectrum analysis indicated activity above the characterization DCGL's for Co-60 and Cs-137.

15. On Figure 1-4, the Reactor Facility Grounds, one sediment sample collected for gamma spectrum analysis from the storm drain in the parking lot area on the south side of the facility indicated activity above characterization DCGL's for Co-60 and Cs-137.

Figure 1-1
UVA Reactor First Floor Plan View

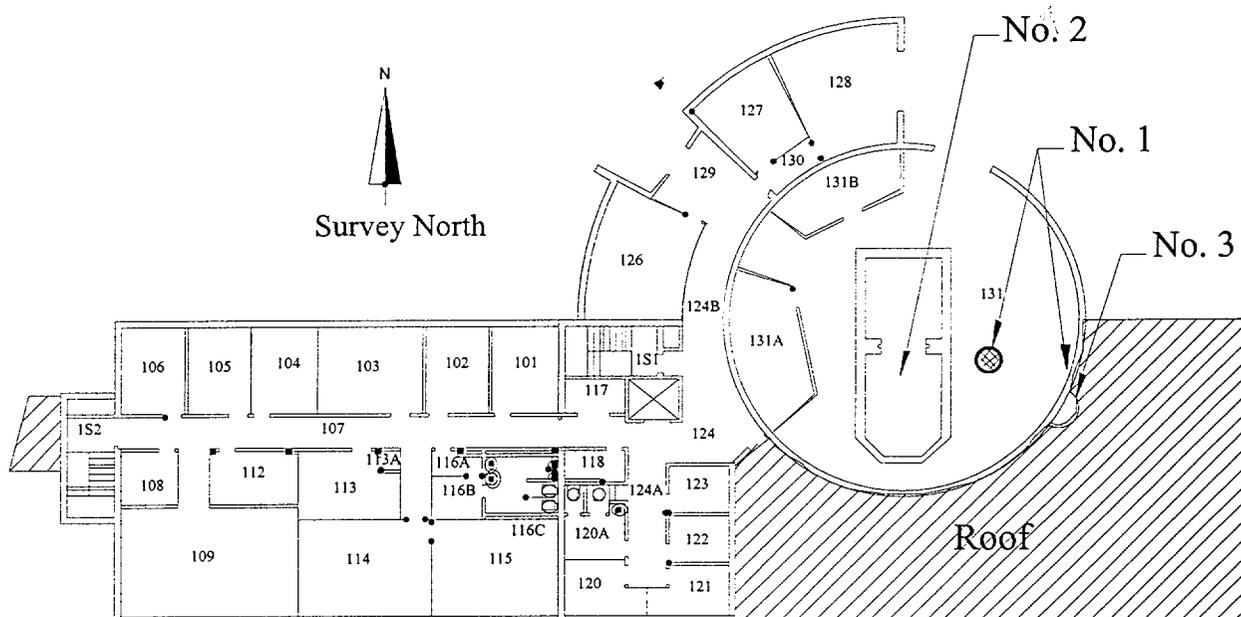


Figure 1-2
UVA Reactor Mezzanine Floor Plan View

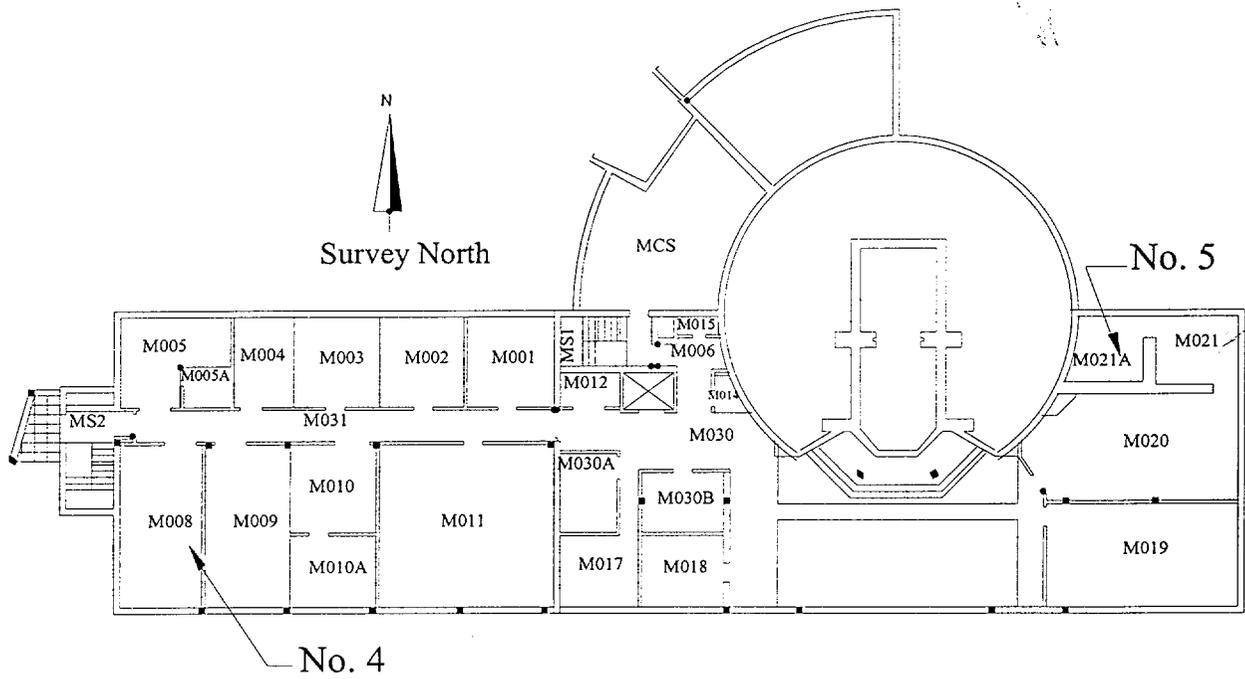


Figure 1-3
UVA Reactor Ground Floor Plan View

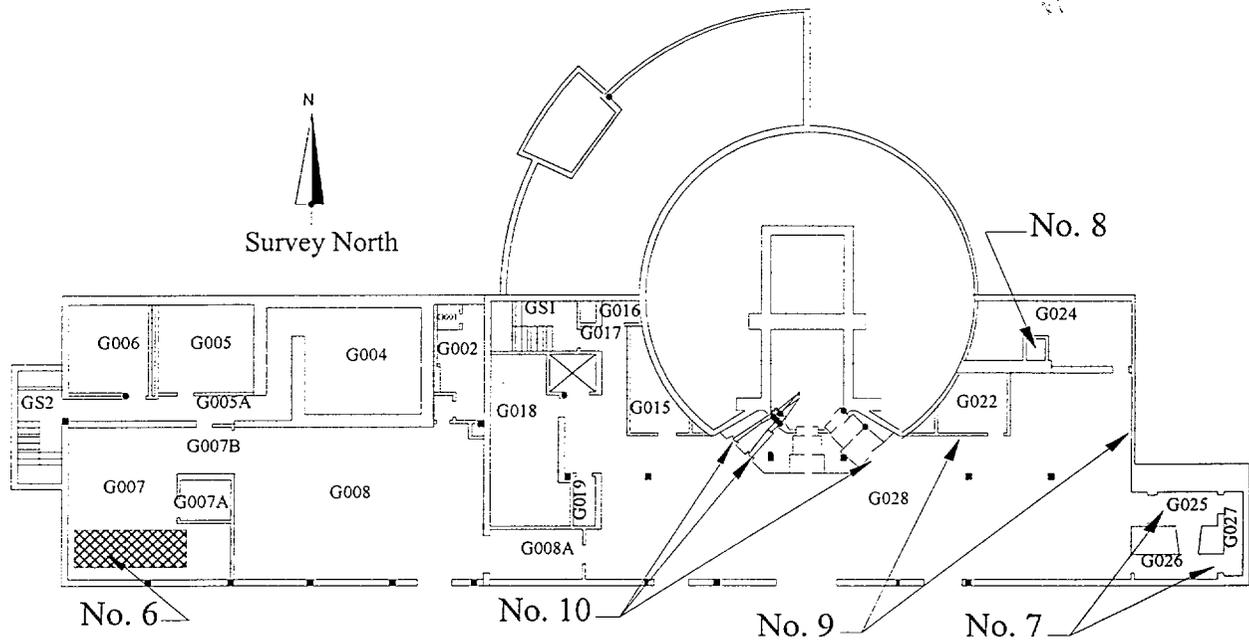
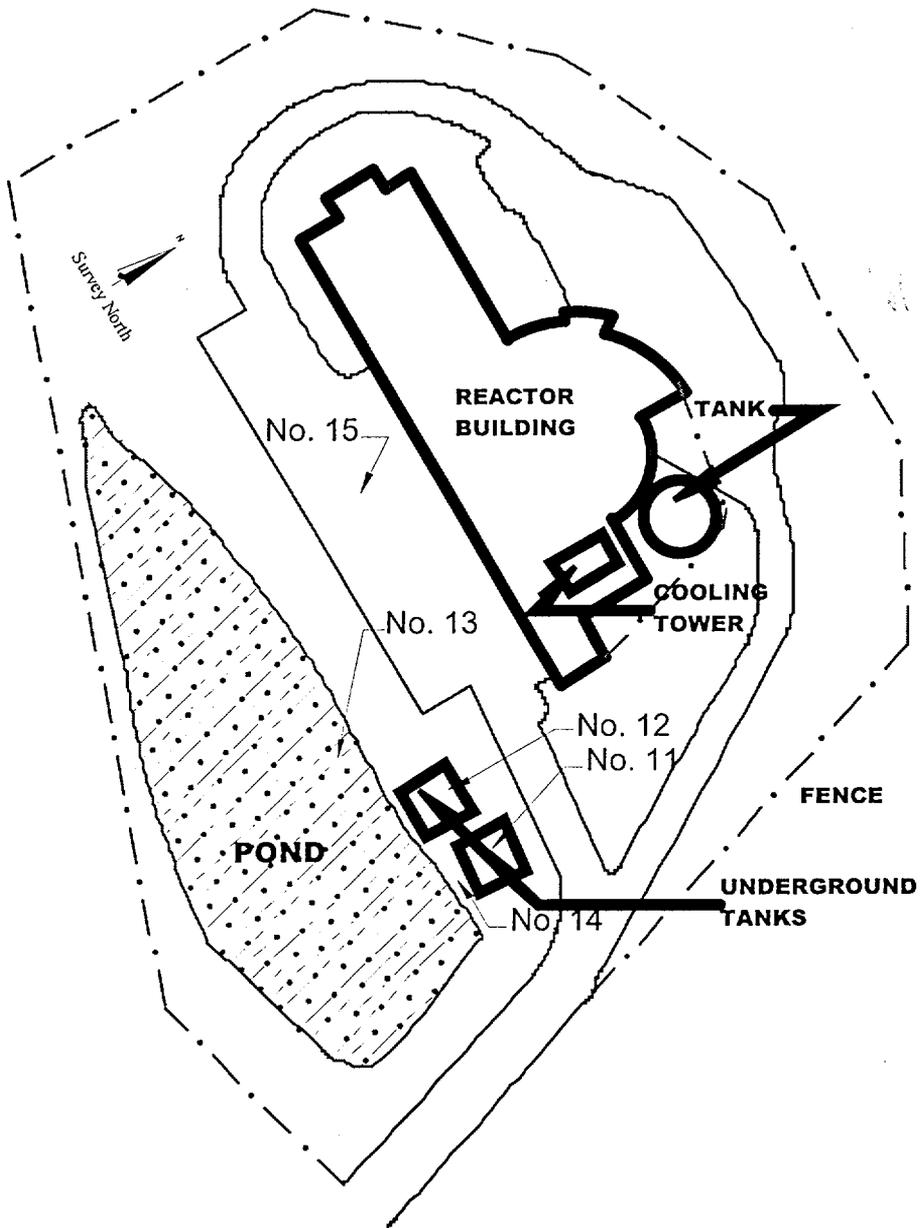


Figure 1-4
UVA Reactor Facility Grounds



2.0 SITE INFORMATION

2.1 Site Description

The Reactor Facility at the University of Virginia was part of the Department of Mechanical, Aerospace, and Nuclear Engineering. The University of Virginia operated a nuclear research reactor (UVAR), a light-water cooled, moderated, and shielded "pool" reactor which was commissioned in 1960 at a power level of 1 Megawatt (MW) thermal, utilizing Highly Enriched Uranium (HEU) fuel elements. The power level was later upgraded to 2 MW in 1971. In the spring of 1994 the reactor was converted to Low Enriched Uranium (LEU) fuel. The reactor was permanently shutdown in June of 1998 and the fuel shipped offsite prior to July 1999. The reactor is contained in a large pool of water that cools and radiologically shields the reactor core. The UVAR pool is 32 feet long, 12 feet wide and 27 feet deep. It is built into the side of Mount Jefferson, just west of the main grounds of the University. Three sides of the pool are recessed into the hillside. The exposed end is shielded by 7 feet of concrete, through which there are several ports for conducting experiments. The reactor was used for radiation research, service work in the areas of activation analysis and isotope production, neutron radiography radiation damage studies, silicon doping, BNCT studies, public tours and the training of Nuclear Engineering students.

The UVAR Facility is located approximately 2,000 feet west of the city limits of Charlottesville, Virginia. The site is located at a former city water reservoir, further referenced as the pond throughout this document, 200 feet up a ridge that runs between Mt. Jefferson and Lewis Mountain. The pond is located in a draw which begins at the top of the ridge. The reactor facility is located on a side of this draw, approximately 50 feet above the water level of the pond. The UVAR facility building has approximately 22,000 ft² of floor space on three levels consisting of the main reactor room, radiation laboratory area, and supporting laboratory, shop, hot cell and office space. The UVAR facility building is shown in Figures 1-1, 1-2 and 1-3 of the previous section. The building is surrounded by a 102,000 ft² fenced yard which defines the UVAR site as shown in Figure 1-4 of the previous section.

2.2 Identity of Contaminants

The radionuclides of concern identified at the facility (produced by reactor operations) were mixed fission and activation products such as cesium-137 (Cs-137), strontium-90 (Sr-90) and cobalt-60 (Co-60). The identity of the radionuclides was based on review of process history and previous survey data, including sample analyses. From interviews with current UVAR personnel, other radionuclides were identified that were not produced by reactor operations. These radionuclides were used in various laboratories at the facility and were identified as nickel-63 (Ni-63), technetium-99 (Tc-99), lead-210 (Pb-210), radium-226 (Ra-226), americium-241 (Am-241) and natural uranium oxide powder (UO₂).

Radionuclides identified during the characterization study in samples analyzed onsite by gamma spectrum analysis were manganese-54 (Mn-54), cobalt-57 (Co-57), Co-60, zinc-65 (Zn-65), antimony-125 (Sb-125), Cs-137 and europium-154 (Eu-154). These radionuclides were present in sediment from inside the buried liquid waste storage tank; all but Eu-154 were present in resin samples from the currently used UVAR pool water processing system and the prototype water processing system that is no longer used; Mn-54, Co-60 and Cs-137 were present in sediment from the sump in the heat exchanger room; Co-57, Co-60 and Cs-137 were present in surface soil from the floor of the buried liquid waste storage tank bunker; Zn-65 and Co-60 were present in floor sweepings from the south access facility; and Cs-137 was present in the concrete floor of the Hot Cell.

Four locations in the facility were selected for collection of concrete cores. The concrete cores were sliced, direct beta measurements performed on the slices and five of the slices were sent to Duke Engineering and Services Environmental Laboratory (DESEL) for further radionuclide analysis. The analysis results indicated the presence of Co-60, Cs-137 in significant concentrations and Sb-125 in low concentration. The analysis results are presented for Survey Area A0800, Concrete Core Samples in Section 7.3 of this report.

A resin sample from the UVAR pool water processing system was sent offsite for 10 CFR Part 61 radionuclide analysis. The sample was sent to DESEL and analysis results indicated the presence of Mn-54, Fe-55, Co-60, Ni-63 and Zn-65. The laboratory's analysis report for the resin sample is provided in Appendix A, UVA Reactor Facility Characterization Survey Area Reports, Section C0100. The analysis results for the resin sample are presented in the summary for 10 CFR Part 61 Radionuclide Analysis Results in Section 7.5 of this report.

A pond sediment sample was sent offsite for 10 CFR Part 61 radionuclide analysis, and for the Envirocare Suite of Hazardous Material analyses. The sample was sent to Barringer Laboratories and analysis results from the 10 CFR Part 61 analysis indicated the presence of Co-60, Cs-137, Eu-152, U-233/234, U-238 and Pu-241. The results from the Envirocare Suite of Hazardous Material analyses indicated no hazardous materials present in the pond sediment sample greater than 15% of the U.S. Environmental Protection Agency Region III Risk-Based Concentration Table values with residential criteria applied. The laboratory's analysis report for the pond sediment sample is provided in Appendix A, UVA Reactor Facility Characterization Survey Area Reports, Section H0100. The analysis results for the pond sediment sample are presented in the summary for Environmental Hazardous Material Survey in Section 7.4 of this report.

The hazardous materials of concern for UVAR facility surfaces were lead in paint and asbestos in insulation, paneling and floor tiles. During the radiological characterization, a representative from UVA's Environmental Health and Safety department performed a lead and asbestos survey for areas within the UVAR

facility considered to be radiologically impacted. The results for the survey are presented in the summaries for Surfaces and Structures Hazardous Material in section 7.4 of this report. The survey report issued by UVA Environmental Health and Safety department for the hazardous material survey is provided in Appendix A, UVA Reactor Facility Characterization Survey Area Reports, Section G0100.

3.0 SURVEY OVERVIEW

3.1 Survey Data Quality Objectives

The survey was performed to achieve the following Data Quality Objectives:

- Verify the historical characterization data obtained by interviews, document reviews and facility walk-downs.
- Perform an accurate assessment of the radioactive contamination and activation levels, including depth of penetration, in the impacted areas of the facility.
- Show that most areas previously considered non-impacted by radioactive materials were truly not impacted.
- Collect sufficient isotopic data such that accurate Derived Concentration Guideline Values (DCGL's) could be determined.
- Provide information to be used in preparing the site Decommissioning Plan and Decommissioning Cost Estimate.

The characterization surveys included measurements and sampling in impacted areas known to require remediation. The survey showed that not all impacted areas actually require remediation. The surveys also examined the surrounding areas, structures and environs believed to be non-impacted.

3.2 Characterization Survey Guideline Values

To determine appropriate instrument sensitivities for the characterization survey for both building surfaces and soil, initial site data was used to develop assumed Derived Concentration Guideline Levels (DCGL_a) for 25 mrem/yr total effective dose equivalent or TEDE. The radionuclides of concern were assumed to be Co-60, Sr-90 and Cs-137, distributed in the radioactive mix in the proportions of 50%, 25% and 25%, respectively. The radionuclides of concern were chosen based upon information provided by UVA personnel during the initial site walk-down. The radionuclide distribution was conservatively determined by the survey team based on previous experience. The assumed radionuclide mix was input into spreadsheets to determine the limit of each radionuclide according to the guidance provided in NUREG-1575 and DG-4006. The limit input for each radionuclide for surface contamination came from the Federal Register, Volume 63, No. 222, Wednesday, November 18, 1998 Notices. The limit input to the spreadsheet calculations for each radionuclide for soil concentration limits were calculated using RESRAD, Version 5.82 for the same mix. The spreadsheet calculations are provided as Attachment 2, "Assumed Surface Contamination DCGLs" and Attachment 3, "Assumed Soil Concentration DCGLs". The assumed DCGLs are provided in Table

3-1, Characterization Survey Assumed DCGLs for Surfaces and Soil that follows.

Table 3.1
Characterization Survey
Assumed DCGLs for Surfaces and Soil

Radionuclide	Contribution to Mix (%)	Surface Contamination DCGL in dpm/100cm ²	Soil Concentration DCGL in (pCi/g)
Co-60	50	4,630	1.91
Sr-90*	25	2,310	0.95
Cs-137	25	2,310	0.95
Total	100%	9,250	3.81

*Note: Sr-90 is not detectable by gamma spectrum analysis. The value was provided for total or gross activity DCGL calculation only.

Results from two samples collected for 10 CFR Part 61 analysis are presented in Section 7.5, 10 CFR Part 61 Radionuclide Analysis Results of this report. The 10 CFR Part 61 analysis data in conjunction with data obtained from the UVAR facility characterization was used to determine post-characterization release criteria, DCGLs. Section 7.6, Post Characterization DCGL Determination, discusses the development of DCGL for surfaces in dpm/100cm² and DCGL for radionuclides in pCi/g for soil according to the current dose specific regulations for release for unrestricted use.

3.3 Organization and Responsibilities

GTS Duratek implemented an integrated management approach that included project management oversight and technical support. The resources of GTS Duratek's Bear Creek and Gallaher Road offices, including professional engineering and quality assurance staff, supported the Project Manager and the onsite team to ensure the project success in execution and completion.

The onsite survey and sampling team, consisting of an engineer/supervisor and Radiation Protection technicians were trained, qualified, and experienced in field radiological survey procedures. All project personnel received annual radiological worker training.

3.3.1 Project Manager

The Project Manager was the primary point of contact and interface with the University staff. The minimum qualification requirements for the Project Manager were ten years of decommissioning and cost estimating experience including prior management experience.

3.3.2 Survey Supervisor

The Survey Supervisor was responsible for the performance of the survey and sampling activities and was onsite during the project. The minimum qualification requirements for the Survey Supervisor were five years of health physics experience including supervision of survey teams on similar projects. The individual was qualified in the use of all survey instruments and all aspects of surveying as described in this Survey and Sampling Plan.

3.3.3 Radiation Protection (RP) Technician

The RP Technician was responsible for performing surveys and collecting samples, and was qualified in the use of the survey instruments and surveying in accordance with the Survey and Sampling Plan.

4.0 SURVEY INSTRUMENTATION

Careful selection and use of instruments ensured sensitivities sufficient to detect the identified primary radionuclides at the minimum detection requirements. Table 4-1 provides a list of the instruments, types of radiation detected and calibration sources used during the characterization survey. The hard-to-detect radionuclides, which include low-energy beta emitters, cannot be measured using these field instruments. These radioisotopes were quantified through offsite laboratory analysis on an as-needed basis.

GTS Duratek used the Ludlum Model 2350-1 Data Logger with a variety of detectors for direct measurements of total alpha and beta surface activity as well as exposure rate measurements. The Data Logger is a portable micro-processor computer based counting instrument capable of operation with NaI(Tl) gamma scintillation, gas-flow proportional, GM and ZnS scintillation detectors. The Data Logger is capable of retaining in memory the survey results and instrument/detector parameters for up to 1000 measurements. This data is then downloaded to a personal computer for subsequent reporting and analysis.

Detector selection depended on the survey to be performed, surface contour and survey area size. The project team normally used the 126 cm² gas-flow proportional detector for direct alpha and beta measurements and a 1" x 1" Sodium Iodide (NaI) gamma scintillation detector for exposure rate measurements.

In addition to the standard detector systems, GTS Duratek has developed a series of GM and gas-proportional detectors which are used for direct surface measurements of the interiors of system piping. These detectors, coupled to the Model 2350-1 Data Logger, were used in performance of the systems surveys during the UVAR facility characterization.

Smears for removable alpha and beta contamination measurements were collected over a 100cm² area and analyzed using UVA's Tennelec, Model LB5100W, Low Background Alpha/Beta Planchet Counter.

Isotopic quantification and identification (qualification) was performed on soil, water, sediment and debris samples using an EG&G Instruments NOMAD High Purity Germanium (HPGe) Gamma Spectroscopy System.

Underwater surveys of components, tools and other equipment in the reactor pool were performed using an Eberline Ion Chamber, Model RO-7 with a mid-range detector.

**Table 4-1
Survey Instrumentation**

Instrument/Detector	Detector Type	Radiation Detected	Calibration Source	Use
Ludlum Model 2350/43-68 4	Gas-flow proportional (126 cm ²)	Alpha or beta	⁹⁹ Tc (β) ²³⁰ Th (α)	Direct measurements
Ludlum Model 2350/43-94 or 43-98	Gas-flow proportional Pipe Detector	Alpha or beta	⁹⁹ Tc (β) ²³⁰ Th (α)	Direct measurements
Ludlum Model 2350/ SP-113-3m or SP-175-3m	GM Pipe Detector	Alpha or beta	⁹⁹ Tc (β) ²³⁰ Th (α)	Direct measurements
Ludlum Model 2350/44-2	1" x 1" NaI scintillator	Gamma	¹³⁷ Cs (γ)	Gamma exposure rate
Eberline Low to High Ion Chamber Model RO-7	Ion Chamber	Gamma	⁶⁰ Co (γ)	Gamma exposure rate
Tennelec LB5100W Alpha/Beta Planchet Counter	Gas-flow proportional	Alpha and Beta	²³⁰ Th (α) ⁹⁹ Tc (β)	Smear counting
EG&G NOMAD Gamma Spectroscopy System	High Purity Germanium	Gamma	Mixed Gamma	Nuclide identification and quantification

4.1 Instrument Calibration

GTS Duratek calibrated the data loggers and associated detectors on a semi-annual basis using National Institute of Standards and Technology (NIST) traceable sources and calibration equipment. The instrument calibration included:

- high voltage calibration,
- discriminator/threshold calibration,
- window calibration,
- alarm operation verification, and
- scaler calibration verification.

The detector calibration included:

- operating voltage determination,
- calibration constant determination, and
- dead time correction determination.

Calibration labels showing the instrument identification number, calibration date and calibration due date were attached to all portable field instruments and were checked before each use. Calibration certifications for each instrument used for collecting characterization measurements are contained in Appendix B, UVA Reactor Facility Characterization QA/QC Data.

4.2 Sources

All radiation sources used for calibration or efficiency determinations for the survey were representative of the instrument's response to the identified nuclides and were traceable to NIST. Calibration certifications for each source used are contained in Appendix B, UVA Reactor Facility Characterization QA/QC Data.

Radiation Protection Technicians controlled radioactive sources used for instrument response checks and efficiency determination. Sources were stored securely and routinely inventoried by survey technicians. After demobilization from the site, all sources used for the project were accounted for and returned to GTS Duratek Instrument Facility.

4.3 Instrument Response Test

All instrumentation used for characterization survey measurements were response tested prior to use according to the respective operating procedure for the instrument. The data loggers and associated detectors were response tested prior to and upon completion of the survey using NIST traceable sources. During the "pre-survey" response test, the detector efficiency and background were observed and recorded on download records and were reviewed for acceptance. After the survey was completed, the "post-survey" response test was performed in the same manner as the pre-survey response test, the results were reviewed for acceptance and the instrument was downloaded to a computer download station. Once downloaded, the measurement results were printed and reviewed for acceptance. Instrument pre-survey and post-survey response test records and download reports from each survey are provided in Appendix C, UVA Reactor Facility Characterization Survey Packages.

The low background alpha/beta counter and gamma spectroscopy system were response tested prior to use for each day used. The response test consisted of counting a sample blank for a daily background indication and counting NIST traceable sources to test the accuracy of each instrument. The response tests were reviewed for acceptability prior to counting or analyzing characterization samples. The instruments' daily background measurement result, from counting the sample blank, was compared to the initial and previous days background indication. The pre-use accuracy response test for the alpha/beta counter was performed by counting a calibrated or known quantity alpha source and then counting a calibrated or known quantity beta source. The results from counting the alpha source and the beta source in disintegrations per minute (dpm) per source were plotted on QC Control Charts. For the gamma spectroscopy system, the results from counting a known quantity or calibrated mixed gamma standard were plotted on QC Control Charts for Am-241 and for Co-60 activity in micro curies (uCi) total activity, energy of the peak channel for Am-241 and for Co-60 and for the full width half maximum for the Co-60 1,332 keV peak. The QC Control Charts for the low

background alpha/beta counter and gamma spectroscopy system instruments used for counting and/or analyzing characterization survey samples are provided in Appendix B, UVA Reactor Facility Characterization QA/QC Data.

4.4 Characterization Procedures

All instrumentation used for characterization survey measurements were operated according to an operating procedure. A list of procedures used for the characterization work are provided in Attachment 1, UVAR Facility Characterization Procedures.

4.5 Minimum Detectable Activity

Minimum Detectable Activity (MDA) is defined as the smallest amount or concentration of radioactive material in a sample that will yield a net positive count with only a 5% probability of falsely interpreting background responses as true activity. The MDAs were dependent upon the counting time, geometry, sample size, detector efficiency and background count rate. As a data quality objective, measurement MDAs were set to be equal to, or less than, 50% of the assumed DCGLs developed for the survey for direct total contamination measurements and set for less than 10% for the removable component. Most direct measurements collected in low or normal background areas achieved measurement MDAs of less than 10% of the assumed DCGLs.

In a few areas where background radiation levels were elevated, the MDA for the measurement exceeded 50% of the assumed DCGL. The locations exhibiting high direct beta MDAs also exhibited elevated exposure rate measurements in the areas. The direct beta measurements with elevated MDAs and exposure rate measurements were due to radioactive material storage areas in rooms G007, G018, G022, G028, the water processing system in room M021A and the heat exchange system in room G024. The direct beta measurements with MDAs greater than 50% of the assumed DCGL, and the elevated exposure rate measurement results, may be viewed in Appendix A, UVA Reactor Facility Characterization Survey Area Reports, Survey Area and/or Section A0200.

The equation used for calculating the MDA for field instrumentation and the smear counter was:

$$MDA = \frac{\frac{2.71}{t_s} + 3.29 \sqrt{\frac{R_b}{t_s} + \frac{R_b}{t_b}}}{E \left(\frac{A}{100} \right)}$$

Where: MDA	=	Minimum Detectable Activity (dpm/100 cm ²)
R _b	=	Background Count Rate (cpm)
t _b	=	Background Count Time (min)
t _s	=	Sample Count Time (min)
A	=	Detector Area (cm ²)
E	=	Detector Efficiency (c/d)

Instrument operating parameters for measurements of potential contamination from alpha emitting radionuclides and tritium were conservatively set based upon on the instrument and measurement capability. A DCGL basis for measurement MDAs or sensitivities was not developed for contamination measurements of alpha emitting radionuclides or tritium since they were not included in the assumed radionuclide mix, expected to be or found to be present at the facility.

A priori MDAs were established and verified each day for the gamma spectroscopy equipment by counting an empty cave, analyzing the spectrum with the 250 ml soil geometry calibration and reviewing the analysis report to ensure the adequate MDAs were met. This also served as an initial and daily background check for the system.

Table 4.2, Characterization Survey Measurement MDA and Methods, shows all the measurement types used during the UVAR facility characterization, the assumed DCGL, if applicable, and the associated MDA or sensitivity. The "Actual-Measurement MDAs," presented in the table, represent a goal of the Data Quality Objectives. The MDA goal was achieved but mostly exceeded for all measurement types where the measurement results were not influenced by background radiation fields as previously discussed in this section. In this regard, the value presented for the MDA is preceded by a "less than or equal to" symbol (\leq) to represent the actual measurement MDAs. Measurement results, with their associated MDA, are provided in Appendix A, UVA Reactor Facility Characterization Survey Area Reports.

Table 4.2
Characterization Survey
MDAs and Methods

Measurement Type	Instrument Used	Method Used	Assumed DCGL	Actual Measurement MDA
Direct total beta	M2350	Gas flow Prop. or GM	9,250 dpm/100cm ²	≤ 5,000 dpm/100cm ²
Direct total alpha	M2350	Gas flow Prop.	Not established*	≤ 100 dpm/100cm ²
Smears for removable alpha contamination	Tennelec LB5100W	Gas flow Prop.	Not established*	≤ 20 dpm/100cm ²
Smears for removable beta contamination	Tennelec LB5100W	Gas flow Prop.	1,000 dpm/100cm ²	≤ 100 dpm/100cm ²
Smears for removable tritium contamination	Offsite Analysis at GTSD	Liquid Scin. Counter	Not established*	≤ 20 dpm/100cm ²
20 ml liquid Tritium samples	Offsite Analysis at GTSD	Liquid Scin. Counter	Not established*	≤ 20 dpm/ml
250 ml soil and sediment samples	EG&G NOMAD	Gamma Spec Co-60 Cs-137	1.91pCi/g 0.95pCi/g	≤ 0.1 pCi/g ≤ 0.1 pCi/g
1 liter water samples pCi/g (ml)	EG&G NOMAD	Gamma Spec Co-60 Cs-137	Not established*	≤ 0.03 pCi/g ≤ 0.03 pCi/g
Qualitative Samples, soil, sediment or debris samples < 250 ml or < 1g/cc	EG&G NOMAD	Gamma Spec	Not applicable	"Present" or "Not- present" indication.

*Note: The need to develop DCGLs for the emissions from radionuclides that are not associated with the radionuclides of concern was not necessary and, therefore, DCGLs were not established.

5.0 CHARACTERIZATION DATA REVIEW AND REPORTING

5.1 Characterization Data Review

GTS Duratek personnel verified the following for each survey package, as a minimum:

- The survey technician initialed and dated the survey package instructions.
- The survey technician collected the number of measurements and samples specified in the survey package in the appropriate locations.
- The sensitivity of the measurements met designated criteria (i.e., count times, background, and detection efficiency were appropriate to meet MDAs).
- Properly calibrated/response checked instruments were used to perform measurements/analysis.
- Quality control checks were performed as designated in the survey package.
- Chain of custody forms, data download reports and laboratory analysis results were complete and in the survey package.
- Any outlying data in the survey/sample results that differed significantly from those expected were evaluated for need for follow-up samples/measurements.

The Characterization Supervisor reviewed onsite and offsite laboratory analysis results for accuracy and signed the Survey Package Cover Sheet signifying survey package completion.

5.2 Characterization Survey Data Reports

GTS Duratek used a proprietary computer program to perform data analysis calculations and generate the characterization survey data reports. The program compiles data from survey units or survey packages and generates a statistical summary report for each survey area or unit (structure, system or open land area).

The compilation of statistical summaries, area-specific historical information, applicable maps and drawings, and conclusions constitutes the characterization survey report for each survey package. The characterization survey data reports are contained in Appendix A, UVA Reactor Facility Characterization Survey Area Reports. The content of these characterization survey data reports is summarized in Table 5-1, Characterization Survey Area Report Contents, that follows.

Table 5-1
Characterization Survey Area Report Contents

Form Title	Contents
Characterization Summary	<ul style="list-style-type: none"> • General historical information on the survey area, including equipment operating history, construction and coating materials, and past contamination incidents • Summary and results of characterization activities • References (e.g., documents, interviews)
Summary of Survey Units	<ul style="list-style-type: none"> • Package description (e.g., building and elevation) • List of units within each survey area (i.e., area, room, or equipment identification), and surfaces (i.e., ceiling, equipment, floor, walls) within each unit, if applicable • List of materials in each survey area (e.g., concrete, metal) and the assigned beta levels for each taken from the background survey results
Survey Map or Drawing	<ul style="list-style-type: none"> • Representative diagram of the survey area and units • Location of survey measurements.
Statistical Summary and Graph	Statistics and a graph of results for each type of measurement (direct measurements for total beta activity, removable alpha and beta activity, and gamma exposure rate at 1 meter) performed for the survey area. Statistics typically include: mean, maximum, minimum, standard deviation, minimum detectable activity, samples reported and samples prescribed.
Results Listing Report	Corrected data and location information for each applicable measurement type.
Download File & Survey Instrumentation Calibration Summary	A summary of survey date, file number (Download #- Station #), detector model number, instrument and detector serial numbers and calibration due date(s), detector efficiency and Technician's ID number for each instrument and detector combination used to collect data.

5.3 Characterization Data Analysis Equations

The following sections describe the equations used for characterization data analysis in the report.

5.3.1 Direct Measurements for Total Surface Activity

Direct measurements for total alpha and beta surface activity per unit area were calculated for counts from the Ludlum Model 2350-1 Data Logger by the equation that follows.

$$A = \frac{\left(C * \frac{1}{t_s} \right) - \left[R_b + \left(A_m * E * \frac{a}{100} \right) \right]}{E * \left(\frac{a}{100} \right)}$$

Where:

A	=	net surface activity (dpm/100 cm ²),
C	=	integrated gross counts (counts),
t _s	=	sample count time (min),
R _b	=	background count rate (cpm),
A _m	=	naturally-occurring material-specific surface activity (dpm/100 cm ²),
E	=	efficiency of the survey instrument (cpd), and
a	=	detector area (cm ²).

5.3.2 Measurements for Removable Surface Contamination

For the measurements of removable alpha and beta surface contamination, survey technicians analyzed smear samples taken from an approximate 100 cm² area. The following equation was used to convert gross counts into net activity per unit area.

$$A = \frac{\left(C * \frac{1}{t_s} \right) - R_b}{E * \left(\frac{a}{100} \right)}$$

Where:

A	=	net removable activity (dpm/100 cm ²),
C	=	integrated gross counts (counts),
t _s	=	sample count time (min),
R _b	=	background count rate (cpm),
E	=	efficiency of the survey instrument (cpd), and
a	=	area of the sample (cm ²).

5.3.3 Mean and Standard Deviation

The following equations were used to calculate the mean and standard deviation of characterization measurements in a survey package data set:

$$\bar{x} = \frac{1}{n} \sum x_i$$

Where: \bar{x} = the mean of the measurements
 x_i = the individual measurement
 n = the number of sample measurements

$$s = \sqrt{\frac{\sum(\bar{x} - x_i)^2}{n-1}}$$

Where: s = the standard deviation of the population
 \bar{x} = the mean of the measurements
 x_i = the individual measurement
 n = the number of sample measurements

6.0 SURVEY DESIGN

The purpose of the characterization survey was to provide sufficient, accurate data to develop a credible decommissioning cost estimate for a safe and effective decommissioning plan. The layout of the UVAR Site is presented in Figures 5-1 and the three floor levels of the UVAR Facility are presented in Figures 5-2 to 5-4 in the follow pages. The project team performed surveys according to GTS Duratek procedures and the Characterization Survey Plan. UVA personnel performed sampling and analysis for asbestos and lead in paint. The procedures identified survey instrument requirements, and methods for measurement, sample collection, data reduction and evaluation, while the plan identified the survey protocols.

The survey protocols consisted of a mix of surface activity measurements and sampling. Based upon sample analytical results, selected samples were shipped offsite for additional analyses such as alpha spectroscopy and/or low-energy gamma and beta analysis (hard-to-detect radionuclides). These were limited to one resin sample from the UVAR pool water processing system that showed mixed fission product activity, and one sample of pond sediment from the location where the facility drains enter the pond. Both samples were analyzed such that the relative abundances of the radioisotopes would be known and could be used to develop DCGLs for the UVAR facility surfaces and environs for site release of unrestricted use. The pond sample was also analyzed for hazardous materials that may be present in the pond sediment and environs.

6.1 Survey Design

The guidance contained in NUREG 1575, Multi Agency Radiation Survey and Site Investigation Manual (MARSSIM), provided for the classification of the site into either Impacted or Non-Impacted Areas. Areas with some potential for contamination were classified as impacted. Areas that had no reasonable potential for contamination were classified as non-impacted.

The impacted areas were further divided into one of three classifications:

- Class 1 Areas: Areas that have, or had prior to remediation, a potential for radioactive contamination (based on operating history) or known contamination (based on previous surveys) above the DCGL.
- Class 2 Areas: Areas that have, or had prior to remediation, a potential for radioactive contamination, but are not expected to exceed the DCGL.
- Class 3 Areas: Any impacted area that is not expected to contain any residual radioactivity, or is expected to contain levels of residual radioactivity at a small fraction of the DCGL.

For the purpose of this survey, the site was pre-classified based on initial walk-downs by GTS Duratek engineering staff, and from previous experience with other similar sites. This classification did not affect the level or intensity of the survey, but rather the types of measurements and samples that were performed. For example, it may be reasonable to perform direct measurements on the interior of secondary system piping for surface activity, however, a similar type of survey would be impractical in primary coolant piping. In the latter case exposure rate measurements were more useful.

6.2 Survey Group Designations

A walk-down of the site was conducted to assist in the survey design. Survey area classifications were established and measurement frequency and type determined for each survey unit within a survey area. The Survey Group designations for each classification were as described below in Table 6-1, Survey Group Designations.

Table 6-1
Survey Group Designations

Survey Group Designation	Survey Description - Survey Area Type
A	Radiological - Impacted Plant Surfaces & Structures
B	Radiological - Non-Impacted Plant Surfaces & Structures
C	Radiological - Impacted Plant Systems
D	Radiological - Non-Impacted Plant Systems
E	Radiological - Impacted Environmental Areas
F	Radiological - Non-Impacted Environmental Areas
G	Hazardous Material Characterization of Surfaces & Structures
H	Hazardous Material Characterization of Environmental Areas

6.3 Characterization Survey Packages

Survey packages were developed for each of the survey areas for the UVAR facility and site. These survey packages provided the survey technicians with specific sampling and measurement instructions. The survey packages developed for the characterization were as presented in Table 6-2, UVAR Characterization Survey Packages, that follows.

Table 6.2
Characterization Survey Packages

Survey Package No.	Survey Area Description
A0100	Reactor Confinement Structure - Interior
A0200	Reactor Facility Structure - Interior
A0300	Reactor Pool
A0400	Buried Waste Holding Tanks
A0500	Building Exterior Surfaces
A0600	Bioshield Beam Ports, Facility Ports and Room G022
A0700	Cavalier Reactor Room
A0800	Concrete Core Samples
B0100	Facility Office Structure
B0200	Background Reference Area - Structures
C0100	Reactor Systems
D0100	Facility Systems
D0200	Hot Cell Buried Waste Holding Tanks
E0100	Environs Inside Restricted Area
F0100	Environs Outside Restricted Area
F0200	Background Reference Area - Environs
G0100	Hazardous Material Surfaces and Structures
H0100	Hazardous Material Environmental

6.4 Radiological Characterization Survey Measurements

The project team performed 1,515 alpha, beta and exposure rate survey measurements and collected 1,140 smear, liquid, sediment and soil samples for analysis as defined in the survey packages to complete the UVAR facility and site characterization. The totals exclude measurements collected for the background reference areas. Measurements were performed using appropriate calibrated instruments and with daily instrument quality control (QC) checks performed before and after each day's work. Table 6-3, Total Number of Survey Measurements Performed, summarizes the number types of measurements and samples performed or collected for the survey for each survey group.

**Table 6-3,
Number of Survey Measurements Performed**

Survey Measurement Type	Number of Measurements and Samples						Total
	Survey Group A	Survey Group B	Survey Group C	Survey Group D	Survey Group E	Survey Group F	
Direct Beta	838	215	13	46	30	0	1142
Removable Alpha/Beta	671	215	2	28	0	0	916
Direct Alpha	29	0	0	0	0	0	29
Removable Tritium	0	0	1	25	0	0	26
Exposure Rate	200	62	10	0	56	16	344
Gamma Spec Samples	28	0	2	19	117	32	198
Total	1766	492	28	118	203	48	2655

The project team marked the survey measurement or sample locations at the location and on drawings depicting the survey area or survey unit as applicable. Survey data collected during the project was downloaded from survey instruments into a database for storage, analysis, and reporting. Supervisory personnel reviewed the completed survey packages to ensure that all required surveys were performed and that the completed survey packages contained all necessary information.

Figure 6-1
UVAR Facility Site

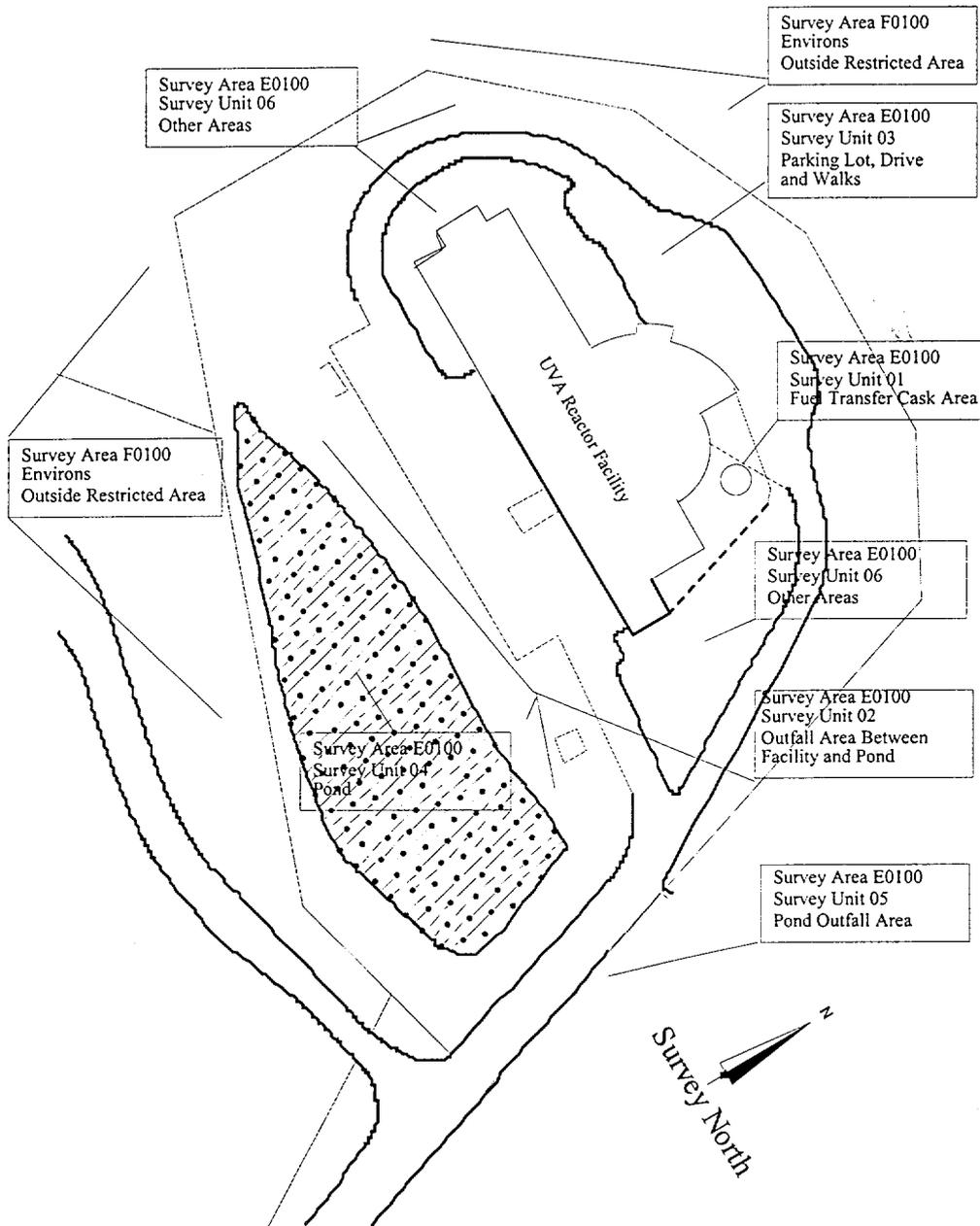


Figure 6-2
UVAR Facility First Floor Plan

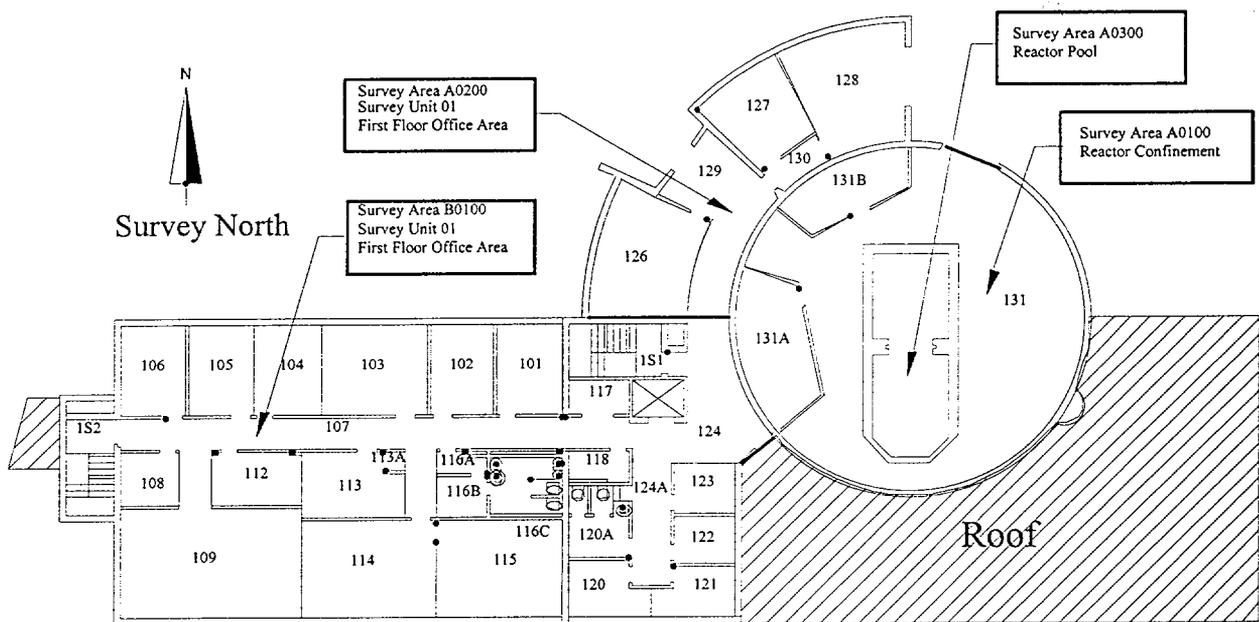


Figure 6-3
UVAR Facility Mezzanine Floor Plan

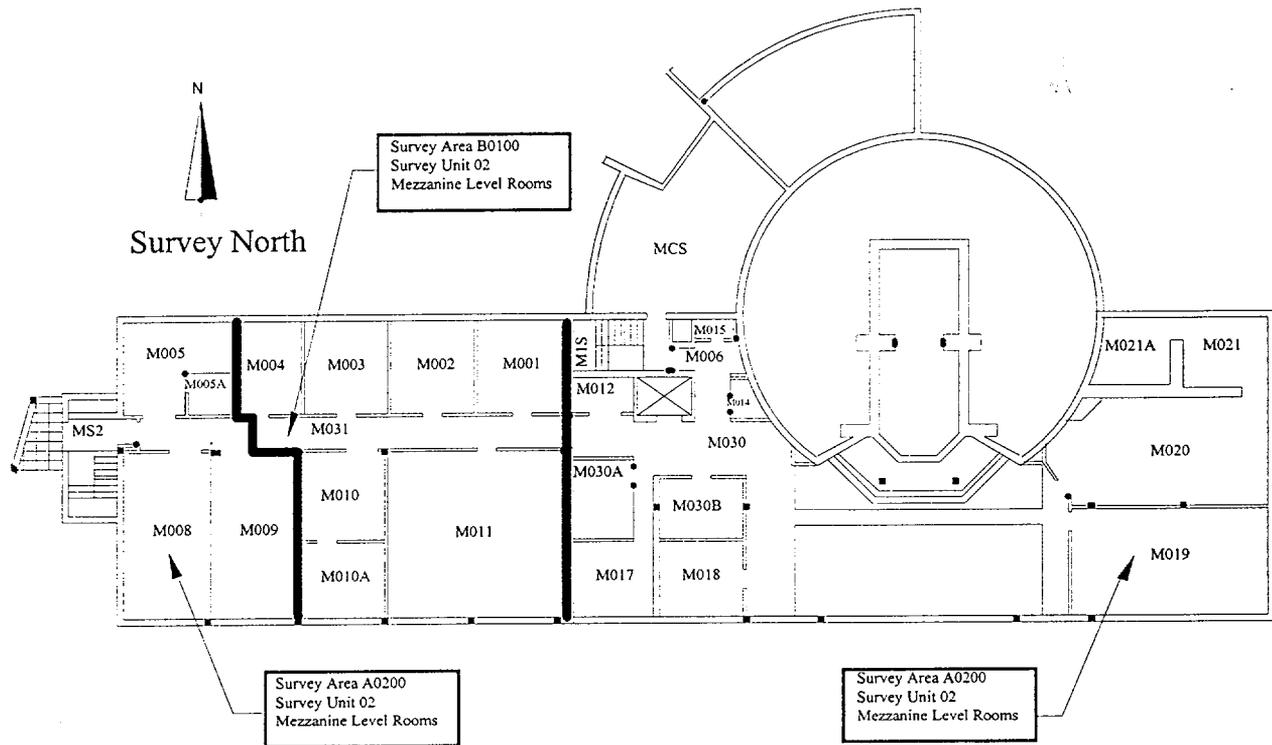
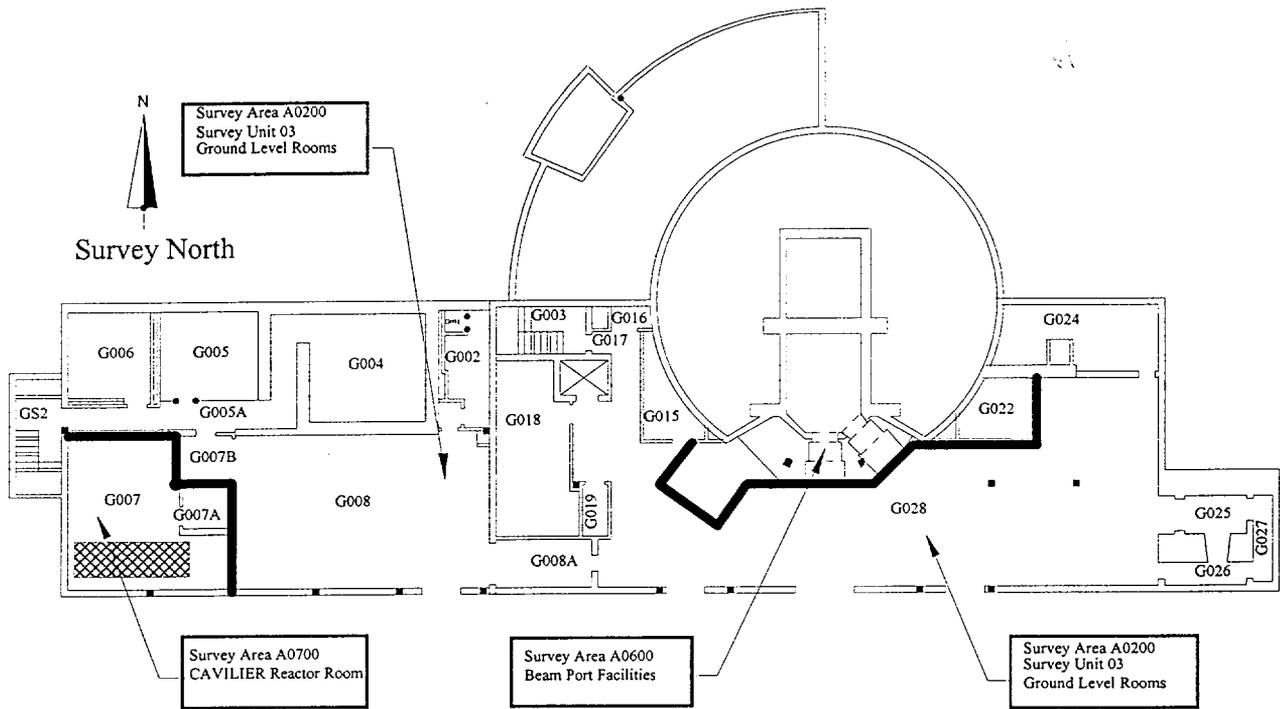


Figure 6-4
UVAR Facility Ground Floor Plan



6.5 Survey Area Historical Summaries

The operational history and use of each survey area was considered for classification and appropriate survey. The sections that follow discuss the operational history and use of each survey area.

6.5.1 Survey Area A0100, Reactor Confinement Structure - Interior

The Confinement Structure interior contains the UVAR (2.0 MW Pool Reactor) with associated equipment and systems. The area is posted and contains radiation areas from operations and from radioactive material storage areas. The Confinement Structure interior is known to be radiologically impacted based upon operations conducted in the area, current survey data and spills that are known to have occurred and subsequent sampling has verified the presence of radioactive material that may require disposal as low-level waste.

6.5.2 Survey Area A0200, Reactor Facility Structure - Interior

These are rooms outside the confinement structure areas in which the need for remediation is unknown and radioactive material may be present in low concentrations. The objective of sampling in Class 2 and 3 areas was to confirm with 95 percent confidence the presence or absence of radioactive material in excess of DCGLs. According to interviews with current employees, the hot cell had never been used for work involving radioactive materials. The area may have been used to store radioactive or contaminated materials.

6.5.3 Survey Area A0300, Reactor Pool

The UVA Reactor Pool area was known to be radioactively contaminated based upon UVAR operations and current survey data. The survey was to take place in two phases. Phase 1 was the survey of items in the pool and was performed once Cobalt-60 source material was removed. On 9/14/1999, after the Phase 1 dose rate survey was completed, the UVA Decommissioning Committee decided to not drain the reactor pool due to ALARA concerns from high dose rate items in the pool. The pool will be maintained filled with water until the items in the pool are dispositioned, therefore, Phase 2 of the survey has been postponed.

Phase 2 of the survey was to entail performing direct total beta measurements, collecting smears samples for removable alpha and beta contamination and collecting debris samples and/or scrapings, and core samples from the sides and bottom of the pool to determine the depth and

extent of contamination and activation. This survey may be performed following removal of high dose rate items/material and the pool water.

6.5.4 Survey Area A0400, Buried Waste Holding Tanks

Bunker for Reactor Buried Waste Holding Tanks: These are areas which are known to be radioactively contaminated based upon operations conducted in the area, current survey data or releases that are known to have occurred. Subsequent sampling has verified the presence of radioactive material that may require disposal as low-level waste. Two 5,000 gallon underground storage tanks received effluent from demineralizer regeneration prior to discharge to the pond. These tanks are no longer in service, and have been emptied. Past spillage from the tanks resulted in low-level contamination of the dirt floored access chamber.

Bunker for Hot Cell Liquid Waste Holding Tanks: Two additional 250 gallon tanks, located in a separate underground enclosure, were designed to collect waste from the hot cell. According to interviews with current employees, the hot cell had never been used for work involving radioactive materials. The area may have been used to store radioactive or contaminated materials. Direct and removable contamination measurements were performed in the hot cell and adjacent corridor.

6.5.5 Survey Area A0500, Building Exterior Surfaces

The building exterior surfaces were areas in which the need for remediation was unknown. The purpose of the exterior surfaces survey was to show that radioactive material was not present at these areas of the facility.

6.5.6 Survey Area A0600, Bioshield Beam Ports, Facility Ports and Room G022

These are areas which had the potential to be radiologically contaminated. The contamination was from pool water leaks or from activation during reactor operations and was based upon surveys conducted in the area and current survey data that verified the presence of radioactive material that may require disposal as low-level waste. This portion of the survey required radiation sources stored at the South end of the reactor pool to be moved toward the North end of the pool. Direct beta measurements, smears for removable alpha and beta contamination and exposure rate measurements were performed on the accessible surfaces of the three access areas of the bioshield walls and bottom, although the general area dose rates were elevated.

Three locations within the concrete bioshield provide access for irradiation experiments. Two 8 inch beam tubes occupy the Southwest section of the

bioshield for the Neutron Beam Port. A large gated and locked room was constructed adjacent to the beam port area for shielding and for an access corridor during operations. The room is constructed of loose concrete block and is not a permanent part of the reactor bioshield.

Two large access facilities, South Access Facility and the Southeast Access Facility, penetrate the bioshield on the South and Southeast side of the pool bioshield, each used for shielding experiments. The South Access Facility also contained a tangential beam port for capture gamma-ray experiments and the Southeast Access Facility contained the Southeast Neutron Beam Port. The South and the Southeast Access Facility each have an external movable shield plug and layered stacks of concrete block and lead brick shielding materials, which were removed to facilitate the survey. When the shielding was removed, this exposed an aluminum plate pool water seal. Minor and temporary reactor coolant water leaks have been observed at various locations and time periods in the concrete bioshield.

Room G022 located East of the Southeast Access Facility currently is being used to store highly activated radioactive materials. The room entrance is locked and posted as a High Radiation Area. Direct measurements were not performed in this room due to the high background. Smears for removable alpha and beta contamination were collected.

6.5.7 Survey Area A0700, Cavalier Reactor Room

The Cooperatively Assembled Virginia Low Intensity Educational Reactor (CAVALIER) Room is known to be radiologically impacted based upon operations conducted in the area and data. Besides the CAVALIER Reactor, the room G007 contains the CAVALIER water demineralizer system and tank, a sub-critical assembly in a pit adjacent to the CAVALIER separated by a wall of loose stacked concrete blocks and a source storage room (G007A). The area is a posted Radiation Area due to several tubes containing natural uranium and 9-12 packages of radioactive waste temporarily stored in the room. The natural uranium and 9-12 packages of radioactive waste stored in the room were moved to another location prior to the characterization survey.

6.5.8 Survey Area A0800, Concrete Core Samples

Four 3" diameter by 6" depth concrete core samples were collected from the locations that follow for direct beta measurements and subsequent gamma spectroscopy analysis.

Reactor Confinement Floor - Contamination events occurred in Reactor Confinement that were remediated and painted over.

Bioshield wall on Mezzanine Level - reactor coolant water leaks have been observed at various times and locations in the concrete bioshield.

South Access Facility Floor - reactor coolant water leaks have been observed at various locations and times in the South Access Facility.

Hot Cell Room G026 Floor - Radionuclide contamination was identified at several locations on the Hot Cell floor during the radiological characterization.

6.5.9 Survey Area B0100, Facility Office Structure

This is an area where the presence of radioactive material is not expected. The number of measurements were the minimum number sufficient to provide statistically sound survey results. If radioactive materials had been encountered during the survey, the area would have been re-classified, and a survey plan for determining the lateral and vertical extent of radioactive material developed in consultation with UVA.

6.5.10 Survey Area B0200, Background Reference Area - Structures

UVA Zehmer Hall was selected as the structures background reference area due to similarities in the construction period, around 1959, and materials of construction. The building and grounds have not been affected by any operations that included the use, storage or handling of licensed radioactive materials. The building is located approximately ½ mile North of the UVA Reactor Facility (UVAR) and is at approximately 1 mile driving distance. Direct beta measurements from the background reference area were collected on construction materials which typically contain naturally occurring radioactive materials which are also present in the UVAR facility construction. The background reference data was used to adjust for the naturally occurring radioactivity in various materials of construction for evaluation of the UVAR radiological survey data.

6.5.11 Survey Area C0100, Reactor Systems

On 9/14/1999 the UVA Decommissioning Committee decided to not drain the reactor pool due to ALARA concerns from high dose rate items in the pool. The pool is to be maintained full of water until the items in the pool are dispositioned. The Reactor Systems are known to be radiologically contaminated, based upon operations conducted in the area and current survey data verifying the presence of radioactive material that may require disposal as low-level waste.

The Reactor Heat Exchange System: Direct and removable measurements

were performed on internal surfaces of the secondary coolant side of the heat exchanger. This involved the temporary removal of flange(s) and/or pipe sections for access to this system. The primary side of the heat exchanger was inaccessible due to the reactor still filled with pool water.

Water Processing Systems: The reactor pool clean-up system, located above the heat exchanger room on the mezzanine level, consisted of a carbon filter, a mixed-bed ion exchange demineralizer and associated piping. The room also contains a prototype ion exchange demineralizer that was used only once. Exposure rate measurements were performed on all equipment and piping. A sample of resin was collected for gamma isotopic analysis. The sample was sent offsite for analysis for hard to detect nuclides and nuclide distribution.

6.5.12 Survey Area D0100, Facility Systems

Class 2 and 3 Systems are systems in which the need for remediation is unknown and radioactive material may be present in low concentrations. The systems consist of floor drains and sink drains of Laboratory, Office, Shop areas inside the UVAR building, the inlet to Hot Cell Buried Waste Holding Tanks and the outlet drains to the pond from Reactor Buried Waste Holding Tanks and CAVALIER Reactor Tank.

Subsequently, contamination was detected in the Hot Cell and in the drain line to the Hot Cell Buried Waste Holding Tanks. See Survey Package D0200 for further information.

6.5.13 Survey Area D0200, Hot Cell Buried Waste Holding Tanks

Two 250 gallon tanks, located in a separate underground enclosure, were designed to collect waste from the hot cell. The North tank is closest to the facility and the South Tank closest to the pond. Contamination was detected at locations on the floor of the Hot Cell while conducting characterization surveys for surfaces and structures survey package A0200. Contamination was also detected in measurements and samples collected from the drain line from the Hot Cell at the inlet to the storage tanks inside the bunker while conducting characterization surveys for facility systems survey package D0100.

Further interviews with former employees indicated the hot cell had been used at one time in the early 1960's to store plasma thermocouples that had ruptured in the pool and another time to store fission plates that had leaked. A floor drain located in the Hot Cell drains directly to the Hot Cell Storage Tanks. At the time of the survey the North Tank was full of water and the

South Tank was less than half full of water.

6.5.14 Survey Area E0100, Environs Inside Restricted Area

These areas, inside the fence, were potentially radiologically contaminated based upon operations conducted in the area, current survey data or releases that are known to have occurred and subsequent sampling has verified the presence of radioactive material that may require disposal as low-level waste.

Survey Unit 01, Fuel Transfer Cask Area: A large, water-filled tank, located outside, and east of the reactor building, was used for the transfer of fuel and other high activity sources from the transfer cask to a shipping cask. This area had in the past been subject to leaks and potential contamination.

Survey Unit 02, Outfall Area Between Facility and Pond: This area is down-gradient from the reactor building, and any run-off from the building (from rain, snow melt, etc.), will pass over this area prior to entering the pond. Two 5,000 gallon underground storage tanks received effluent from the regeneration of the demineralizer. These tanks are no longer in service and have been emptied. Spills and leaking into the dirt floor of the bunker for the tanks occurred during operations.

Survey Unit 03, Parking Lot, Drive and Walks: These are the paved areas inside the fence.

Survey Unit 04, Pond: Laboratory sinks, floor drains and storm water run-off drain to the pond. Early in its operating history, reactor pool water was intentionally discharged to the pond. For the survey, the pond was drained to the extent needed to allow relatively easy access to subsurface sediments from a small boat. Sufficient water was left in the pond such that the resident wildlife was not affected.

Survey Unit 05, Pond Outfall: Monitored releases from the pond were made to a creek in this area.

Survey Unit 06, Other Areas: These are areas inside the fence that border the areas mentioned above.

6.5.15 Survey Area F0100, Environs Outside Restricted Area

These are the areas outside the fence surrounding the UVAR facility. The presence of radioactive material is not expected in this area. The number of samples were the minimum number sufficient to provide statistically sound survey results.

6.5.16 Survey Area F0200, Background Reference Area - Environs

The Ragged Mountain Reservoir was selected as a background reference area for soils, sediments, and surface water. The reservoir has been in existence for approximately 100 years and is currently used and controlled by the regional Rivanna Water & Sewer Authority. The reservoir stores water that is processed and used for drinking water. The reservoir and grounds have not been affected by any operations that included the use, storage or handling of licensed radioactive materials. The reservoir is located approximately 3 miles Southwest of the UVAR and is approximately 4 miles driving distance. Exposure rate measurements and samples from the background reference area were taken from the area environs which contain typical naturally occurring radioactive materials also present in the environs surrounding the UVAR. The background reference data will be used for evaluation of the UVAR radiological survey data.

6.6 Survey Area Characterization Activities

The operational history, use and classification of each survey area was considered during development of the survey package instructions. A detailed account of the survey, including survey instructions, measurement results and drawings of the survey areas annotated with survey measurement location numbers, are presented in Appendix A, UVA Reactor Facility Characterization Survey Area Reports. The Survey Area Reports discuss the measurements that were performed and samples that were collected to complete the survey in each area.

During the survey of the facility surfaces, survey measurement locations were marked on drawings depicting the survey area and on the surface being surveyed with a ½ to 1" orange sticker specifying the measurement location number. Wall measurements included the walls above and below 2 meters, unless otherwise stated.

All direct survey measurements were performed using Ludlum M2350 datalogger instrument and detector combination and logging or storing the measurement and location information in the instrument memory. A minimum of three pre-survey and three post-survey 60 second shielded local area background measurements were collected for the direct beta survey of each area and for each download performed. Some survey areas required more than one download of the instruments' stored memory to complete the prescribed survey.

Single background direct beta measurements, each with a shielded local area background measurement, were required in areas of elevated background count rate. The measurement results for single background direct beta measurements are presented in separate survey reports from the direct beta measurement results with average beta background value applied which were collected in areas with low observed background count rates. The survey reports in Appendix A that were

impacted by the single background beta results were the reports for A0100, A0200 and A0500. The single background direct beta measurements were required during surveys in areas where radioactive materials were stored or present and the radiation from the areas contributed to, or caused, elevated background interferences.

During the survey of the environs, survey measurement location numbers were marked on drawings and with a wood stake or white paint at the measurement locations. All soil and sediment samples were separated by depth, dried and sifted prior to analysis.

At completion of the survey, an error was identified with regard to room number 106. The confinement building interior is actually room number 131 and not room 106. Room 106 is an office located on the first floor of the facility and was surveyed as part of Survey Area B0100. For the purpose of this report only, the interior of the confinement building as a part of Survey Area A0100 will remain as room 106.

6.7 Hazardous Material Characterization Survey Areas and Activities

Sampling locations for asbestos containing materials and lead based paint were identified during the radiological surveys. The areas were selected by GTS Duratek personnel and are depicted in drawings located in Appendix A, UVA Reactor Facility Characterization Survey Area Reports, Section G0100 and H0100 along with details concerning the survey and the survey results, summaries follow.

6.7.1 Survey Area G0100, Hazardous Material Surfaces and Structures

One survey unit was established for the area. The purpose of this survey package was to provide information regarding the presence or absence of hazardous materials such as lead and Asbestos in or on the surfaces and structures of the UVAR facility.

UVA Environmental Health and Safety personnel performed a hazardous material survey consisting of *insitu* measurements for lead using a portable, hand held x-ray fluorescent based spectrometer and visual inspection with sample collection and analysis of material samples for asbestos. UVA Environmental Health and Safety personnel also reviewed survey results from Hall-Kimbrall HAZMAT characterization report from 1988 for asbestos at this reactor facility.

6.7.2 Survey Area H0100, Hazardous Material Environmental

One survey unit was established for the area. The purpose of this survey was to provide information regarding the presence or absence of hazardous materials in the environs surrounding the UVAR facility.

UVA Environmental Health and Safety personnel collected one *insitu* surface soil measurement for lead analysis.

One sample of pond sediment from the location where several facility drains enter the pond was collected by GTSD and sent to Barringer Laboratories. The sample was analyzed such that the relative abundances of any detected radioisotopes or hazardous materials may be known and could be used to develop DCGLs or remedial action clean-up goals, respectively, for the UVAR facility environs.

7.0 SURVEY RESULTS

Summaries of the characterization survey results for the UVAR and surrounding structures and environs are presented in the sections that follow. Detailed data reports for the characterization survey performed for each survey area are contained in Appendix A, UVA Reactor Facility Characterization Survey Area Reports. The summaries that follow begin with environmental background study results and structure surfaces background study results. Summaries of survey areas where elevated results were identified greater than assumed DCGLs for surface contamination and for soil concentration follow. The last section contains all characterization survey area summary results.

7.1 Environmental Background Study

GTSD conducted a background study to determine environmental nuclide concentrations and direct measurement levels from areas similar but not associated with the operation of the UVAR. The local background determination serves as the baseline for decommissioning criteria. The background study was used in the evaluation of the onsite survey data and, as such, measurements, sampling, sample preparation and sample analysis were the same as those used for the onsite measurements.

The background study was performed on soils, sediments and surface water which have common characteristics with the samples and measurements to be collected onsite, but which are unaffected by UVAR operations. The Ragged Mountain Reservoir was selected as a background reference area for soils, sediments, and waters due to similar geology and characteristics as the environs at the UVAR. The reservoir has been in existence for approximately 100 years and is currently used and controlled by the regional Rivanna Water & Sewer Authority. The reservoir stores water that is processed and used for drinking water. Details of the environmental background study are contained in Appendix A, Tab F0200, Background Reference Area - Environs. The detailed report includes drawings depicting the area with survey measurement and sampling locations and measurement and sample analysis results. The findings of the survey are presented in Table 7.1, Environmental Background Study Results for UVAR Characterization Survey that follows. The values are an average of each measurement type.

Table 7.1
Environmental Background Study Results
for UVAR Characterization Survey

Radionuclide	Surface Soil 0-6" Depth Ave. Conc.	Subsurface Soil 6-12" Depth Ave. Conc.	Sediment 0-6" Depth Ave. Conc.	Water Concentrations
H-3	NR ¹	NR ¹	NR ¹	<9 dpm/ml
Co-60	<0.1 pCi/g	<0.1 pCi/g	<0.1 pCi/g	<0.03 pCi/g
Cs-137	0.37 ±0.1 pCi/g	0.05 ±.01 pCi/g	0.33 ±.01 pCi/g	<0.03 pCi/g

Note 1, NR = Not Required

Note 2, The "<" symbol followed by a value indicates the measurement MDA.

The average and maximum one meter exposure rate measurement results were 6 μ R/hr and 7.1 μ R/hr respectively.

The number of measurements and samples performed or collected for the environs background survey were as indicated below.

- (30) Surface (0-6") Soil Samples
- (30) Shallow Subsurface (6-12") Soil Samples
- (2) Water Tritium Samples
- (3) Surface Water Samples
- (3) Sediment Samples
- (30) Micro-R Measurements (1 at each soil sample location)

The environmental background reference data may be used to evaluate the radioactivity found in comparable sample results from the environs at the UVAR facility site.

7.2 Structures Background Study

UVA Zehmer Hall was selected as the structures background reference area due to similarities in the period constructed, around 1959, and materials of construction. The building and grounds were not affected by any operations that included the use, storage or handling of licensed radioactive materials. The building was located approximately 1/2 mile north of the UVAR.

Direct beta measurements from the background reference area were collected on construction materials which typically contain naturally occurring radioactive materials which are also present in the UVAR facility construction. The data collected during the background study was reviewed for trends, and statistics were performed on the various data sets. The details of the structures background study are contained in Appendix A, Tab B0200, Background Reference Area - Structures and include measurement results. The findings of the survey are presented in Table 7.2, Structures Background Study Results for UVAR Characterization Survey.

Table 7.2
Structures Background Study Results
for UVAR Characterization Survey

Material Surface	Material Code	Number of Measurements	Background dpm/100cm ²
Asphalt	B0016	30	755.3
Brick	B0010	30	765.6
Ceramic tile	B0007	30	1,034.8
Cinder block	B0003	30	799.4
Concrete	B0001	30	588.4

The background reference data was used to adjust for the naturally occurring radioactivity in various materials of construction. The UVAR facility structure radiological survey data could then be evaluated. When measurements on materials as shown in Table 4.4 were collected, the associated background value was applied to account for material background in the reported net measurement result.

7.3 Radiological Characterization Survey Summary Results

Details of the of the radiological characterization survey, including measurement results and drawings of survey areas annotated with survey measurement location numbers, are presented in Appendix A, UVA Reactor Facility Characterization Survey Area Reports. The Survey Area Reports present the results of the measurements that were performed and samples that were collected to complete the survey for each survey area. Table 7-3, Summary of Survey Measurement Results, presents the number and type of measurement or sample results obtained from the survey and the number of elevated results.

**Table 7-3,
Summary of Survey Measurement Results**

Survey Measurement Type	Total Measurements	Results >50% of DCGL	Results >DCGL
Direct Beta	1142	12	5
Direct Alpha	29	0	0
Removable Alpha/Beta	916	0	0
Removable Tritium	26	1	0
Exposure Rate	344	NA ⁽¹⁾	NA ⁽¹⁾
Gamma Spec Samples	198	28 ⁽²⁾	23 ⁽³⁾
Total	2655	41	28

Note (1): NA = Not Applicable

Note (2): Analysis results ≥ 0.96 pCi/g for Co-60 and/or ≥ 0.85 pCi/g for Cs-137.

Note (3): Analysis results ≥ 1.91 pCi/g for Co-60 and/or ≥ 1.32 pCi/g for Cs-137.

The sections that follow discuss summary results for each measurement type, including measurement result averages and minimum and maximum result range for the survey. Refer to Appendix A, UVA Reactor Facility Characterization Survey Area Reports, for detailed information regarding each Survey Area.

7.3.1 Direct Beta Measurement Results

The combined total of samples collected and measurements performed for the radiological characterization was 2,655. Of the measurements, 1,142 were direct beta measurements performed at survey measurement locations on facility surfaces, in and on systems and on walks, drives and surfaces outdoors. Twelve (12) direct beta measurement results were greater than the MDA goal of less than 50% of the characterization DCGL (MDA $\leq 5,000$ dpm/100 cm²) and five (5) of these direct beta measurement results were greater than the characterization DCGL of 9,250 dpm/100 cm². The maximum direct beta measurement result was 63,661 dpm/100cm². The location of the measurements and direct beta measurement results that were greater than 5,000 dpm/100 cm² follow:

1. From the survey of the reactor confinement interior surfaces, Survey Area A0100, Survey Unit 01, one direct beta measurement result was 12,593 dpm/100 cm² on the east wall. However, this result may have been influenced by elevated background radiation levels due to radioactive materials stored in the area.

2. From the survey of the facility floor drains Survey Area D0100, Survey Unit 01, direct beta measurement results from the three floor drains in the confinement room showed contamination levels ranging to 6,398 dpm/100 cm² in floor drain No.2 on the east side of the reactor pool.
3. From the survey of the reactor facility mezzanine level surfaces, Survey Area A0200, Survey Unit 02 in room M008, one direct beta measurement result was 26,365 dpm/100 cm² in laboratory sink. The contaminant is suspected to be Nickel-63.
4. From the survey of the reactor facility mezzanine level surfaces, Survey Area A0200, Survey Unit 02 in room M021A, one direct beta measurement result was 8,318 dpm/100 cm² on the equipment surface of the reactor pool water clean up system. However, this surface contamination measurement result may have been influenced by elevated background radiation levels from the water clean up system internal contamination.
5. From the survey of the reactor facility ground level surfaces, Survey Area A0200, Survey Unit 03 in Hot Cell room G026, one direct beta measurement result was 63,661 dpm/100 cm² on the floor, and in room G027 one direct beta measurement result was 19,268 dpm/100 cm² on the floor.
6. From the survey of the reactor facility ground level surfaces, Survey Area A0200, Survey Unit 03 in room G028, one direct beta measurement result was 8,246 dpm/100 cm² on the east wall, half the way between rooms G024 and G025, and one direct beta measurement result was 7,522 dpm/100 cm² on the north wall outside of room G022. However, these surface contamination measurement results may have been influenced by elevated background radiation levels from radioactive materials stored in the area during the time of the survey.
7. From the survey of the Bioshield Beam Port, Access Facilities (South and Southeast Access Facilities) and Room G022, Survey Area A0600, Survey Unit 03 Southeast Facility Port, one direct beta measurement result was 5,758 dpm/100 cm² on the beam tube pipe end that was oriented toward the reactor on the movable shield plug for the southeast port facility. However, this measurement result may have been influenced by elevated background radiation levels from radioactive materials stored in room G022 during the time of the survey.

8. From the survey of reactor facility surfaces, Survey Area A0400, Survey Unit 01 the Buried Liquid Waste Holding Tanks, one direct beta measurement result was 5,541 dpm/100 cm² in the east tank and one direct beta measurement result was 6,600 dpm/100 cm² in the west tank.
9. From the survey of reactor facility system drains, Survey Area D0200, Survey Unit 01 the Hot Cell Buried Waste Holding Tanks, one direct beta measurement result was 16,907 dpm/100 cm² in the piping leading to the tanks.

7.3.2 Direct Alpha Measurement Results

Direct alpha measurements were performed at 29 survey measurement locations during the characterization survey of facility surfaces. Out of the 29 measurements performed, no measurement result exceeded the maximum MDA of 60 dpm/100 cm².

7.3.3 Removable Alpha Measurement Results

Smear samples were collected and analyzed for removable alpha and beta activity for 916 survey measurement locations on facility surfaces in, and on, systems. Out of the 916 measurements performed, no measurement results for removable alpha activity exceeded the maximum MDA of 14.7 dpm/100cm² that could be attributed to licensed radionuclide contamination. However, elevated removable alpha activity was detected on smear samples from surveys in the subcritical assembly pit (next to the CAVALIER tank), Survey Area A0700, Survey Unit 02, and HVAC duct in the hallway between rooms G004 and G005, Survey Area A0200, Survey Unit 03. The elevated results were attributed to radon progeny or daughter products contamination. The radon determination was made by recounting the same smear sample after more than a 24 hour period from the initial count.

7.3.4 Removable Beta Measurement Results

Smear samples were collected and analyzed for removable alpha and beta activity for 916 survey measurement locations on facility surfaces in, and on, systems. Out of the 916 measurements performed, there was one (1) measurement result for removable beta activity that could be attributed to licensed radionuclide contamination above the maximum MDA of 24.3 dpm/100cm². There were no measurement results for removable beta activity above the characterization DCGL of 100 dpm/100cm². The one removable beta activity measurement result was 74.7 dpm/100cm². The smear sample was collected during the systems survey for Survey Area D0100, Survey Unit 02 from the sink on the west side of room M005. Other elevated

removable beta activity was detected on smear samples from surveys in the subcritical assembly pit, Survey Area A0700, Survey Unit 02, and HVAC duct in the hallway between rooms G004 and G005. The elevated results were attributed to radon progeny contamination. The radon determination was made by recounting the same smear sample after more than a 24 hour period had elapsed from initial count.

7.3.5 Removable Tritium Measurement Results

Smear samples were collected and analyzed for removable tritium activity for 26 survey measurement locations during the characterization survey of facility systems Survey Areas C0100 and D0100. Out of the 26 measurements performed, there was one measurement result for removable tritium activity above the maximum MDA of 18 dpm/100cm². The removable tritium activity measurement result was 75 dpm/100cm² and the smear sample was collected during the systems survey for Survey Area D0100, Survey Unit 02 from the sink drain on the east side of room M008. The contaminant is suspected to be Nickel-63.

7.3.6 Exposure Rate Measurement Results for Surfaces and Structures

During the characterization survey of the facility for Survey Areas A0100, A0200 and A0600, 176 exposure rate measurements were collected at survey measurement locations on the floor at one meter distance from the surface. The average of the exposure rate measurement results was 62.2 uR/hr. The exposure rate measurement results ranged from 6.6 µR/hr to 899 uR/hr. Exposure rate measurement results greater than 20 uR/hr in the confinement room at the time of this survey were due to radiation from a posted radioactive material storage area along the east wall. Rooms G007, G007A, G018, G019, G022 and G028 of the facility were posted as radioactive material storage areas during the time this characterization survey was performed. Elevated exposure rate measurements also were due to radiation from the water processing system in room M021A and the heat exchanger system in room G024. Radiation from the materials stored in these rooms, or from the systems, was the primary contributor to exposure rate measurement results greater than 20 uR/hr at other locations within the facility.

During the characterization survey of the facility for Survey Area A0100, Survey Unit 01, exposure rate measurements were collected at 10 survey measurement locations on equipment. For each of the locations, a measurement was collected at 15 cm distance from the surface. The average of the exposure rate measurement results was 79.2 uR/hr. The exposure rate measurement results ranged from 33.7 µR/hr to 0.2 mR/hr.

A gamma survey was performed to profile the North and South Neutron Beam Ports as a part of the survey conducted for Survey Area A0600, Survey Unit 01. The detector was placed inside the each of the beam ports and a measurement was collected beginning with the outer most edge of the beam port tube and at each 1 ft. interval until refusal was met. Prior to the survey, items stored in the south end of the reactor pool were moved approximately 10 ft. north in the pool, including the reactor bridge and attached reactor assembly. For the North Beam Port, 7 exposure rate measurements were collected. The average measurement result was 0.9 mR/hr and the results ranged from 23.8 uR/hr to 5.5 mR/hr. For the South Beam Port, 7 exposure rate measurements were collected. The average measurement result was 0.2 mR/hr and the results ranged from 14.5 uR/hr to 0.9 mR/hr. The elevated exposure rate measurement results may have been due to activation of the port material, activation of materials of construction of the pool and/or contribution from the UVAR pool.

During the characterization survey of Group B Survey Areas of the facility surfaces, 62 exposure rate measurements were collected at survey measurement locations on the floor at one meter distance from the surface. The average exposure rate measurement result was 9.3 uR/hr. The exposure rate measurement results ranged from 5.8 μ R/hr to 17 uR/hr.

7.3.7 Exposure Rate Measurement Results for Systems

During the characterization survey for Survey Area C0100, Reactor Systems, exposure rate measurements were collected at 5 survey measurement locations on equipment. For each of the 5 locations, a measurement was collected at 15 cm from the surface and at one meter distance from the surface for a total of 10 measurements. The average of the exposure rate measurement results was 0.4 mR/hr. The exposure rate measurement results ranged from 39.3 μ R/hr to 0.6 mR/hr. The heat exchanger, reactor pool clean-up system tanks, pump and piping and the prototype ion exchange demineralizer tank were surveyed.

7.3.8 Dose Rate Survey of Items in Reactor Pool

During the survey of the UVAR Pool, Survey Area A0300, 51 items in the pool were inventoried and contact dose rate measurements collected. The UVA Decommissioning Committee decided not to have the reactor pool drained due to ALARA concerns from high dose rate items in the pool. The pool will be maintained full of water until the items in the pool are dispositioned. Many high dose rate items remain in the UVAR Pool. See Appendix A, A0300 for a detailed listing of the items with dose rate readings and a drawing depicting the location of the items in the pool.

7.3.9 Exposure Rate Measurement Results for Environs

During the characterization survey of the site environmental areas inside the fence for Survey Area E0100, 56 exposure rate measurements were collected at soil sample locations at one meter distance from the soil surface. The measurements were performed prior to sample collection. The average of the exposure rate measurement results was 14 uR/hr. The exposure rate measurement results ranged from 7.9 μ R/hr to 22 uR/hr.

During the characterization survey of the site environmental areas outside the fence for Survey Area F0100, 16 exposure rate measurements were collected at soil sample locations at one meter distance from the soil surface. The measurements were performed prior to sample collection. The average of the exposure rate measurement results was 16.1 uR/hr. The exposure rate measurement results ranged from 12.1 μ R/hr to 18.9 uR/hr.

7.3.10 Samples Analyzed by Gamma Spectroscopy

Of the 198 samples collected and analyzed by gamma spectroscopy for the characterization survey, twenty-eight (28) sample analysis results were greater than 50% of the characterization soil concentration DCGLs plus background of 0.95 pCi/g for Co-60 and/or 0.85 pCi/g for Cs-137. Twenty-three (23) sample analysis results were greater than the characterization soil concentration DCGLs plus background of 1.91 pCi/g for Co-60 and/or 1.32 pCi/g for Cs-137. The samples collected were sediment, soils and material samples from the facility surfaces and sediment, soils, material and water samples from the environs surrounding the facility. A summary of the sample and sample locations with analysis results greater than 50% of the DCGLs plus background follow:

1. During the survey of the reactor facility ground level, Survey Area A0200, Survey Unit 03, the heat exchanger room G024, one sediment sample result analyzed by gamma spectroscopy showed Co-60 and Cs-137 concentrations greater than the assumed DCGLs. In addition to Co-60 and Cs-137, Mn-54 was present in the sample. The sample was obtained from the sump in the heat exchange room.
2. During the survey of the reactor facility ground level, Survey Area A0200, Survey Unit 03, the Hot Cell room G026, one qualitative material sample result analyzed by gamma spectroscopy showed elevated Cs-137 contamination. The sample was obtained the floor of the Hot Cell room G026. The location was re-sampled for Survey Area A0800 and the concentration of Cs-137 contamination was quantified.

3. During the facility surfaces survey for Survey Area A0400, Survey Unit 01, Buried Waste Holding Tank Bunker, twelve (12) sample results analyzed by gamma spectroscopy showed Co-60 and Cs-137 concentrations greater than the assumed DCGLs. The samples were collected from the buried waste tanks, one from each tank, and the dirt bunker floor. The radionuclide contamination was present in samples down to the 30" depth. In addition to Co-60 and Cs-137, Mn-54, Co-57, Zn-65, Sb-125 and Eu-154 were present in the sample.
4. During the facility systems survey, Survey Area D0100, Survey Unit 04, Hot Cell Storage Tank Inlet, one qualitative sediment sample result analyzed by gamma spectroscopy showed elevated Cs-137 contamination. The sample was collected from cutting the inlet piping inside the storage tank bunker. Subsequently, an additional survey was conducted for Survey Area D0200, Hot Cell Buried Waste Holding Tanks and water and sediment samples were collected from each of the tanks. Of the 5 water samples collected for gamma spectral analysis, one sample result analyzed by gamma spectroscopy showed Cs-137 concentration greater than the assumed DCGLs. The sample was from water and sediment pumped from the bottom of the South tank.
5. During the survey of the reactor facility exterior surfaces, Survey Area A0500, Survey Unit 01, the reactor confinement building, one composite sediment sample result analyzed by gamma spectroscopy showed Co-60 and Cs-137 concentrations greater than the assumed DCGLs. The sample was obtained from the top of the brick stack inside the accessible enclosure.
6. During the survey of Environs Inside Restricted Area, Survey Area E0100, in Survey Unit 02, Outfall Area Between Facility and Pond, two sample results analyzed by gamma spectroscopy for location No.11 showed Co-60 and Cs-137 concentrations greater than the assumed DCGLs in the 0-6" surface and subsurface 6-12" depth soil samples. The sample location was beneath the tile drain from the buried waste storage tank bunker, between the bunker and the pond.
7. During the survey of Environs Inside Restricted Area, Survey Area E0100, in Survey Unit 03, Parking Lot, Drive and Walks one sediment sample results analyzed by gamma spectroscopy from the storm drain location No.2 showed Co-60 at greater than 50% of the assumed Co-60 DCGL and Cs-137 at greater than the assumed Cs-137 DCGL.

8. During the survey of the reactor systems, Survey Area C0100, Survey Unit 02, Reactor Pool Clean-up System Tanks, one sample of resin for qualitative gamma spectrum analysis showed Co-60 at 6,780 pCi/g. The samples was collected from the resin tank currently in use. In addition to Co-60, Mn-54, Fe-55, Ni-63 and Zn-65 were present in the sample. Subsequently, a portion of the sample was sent offsite for 10CFR Part 61 analysis. The analysis results are presented in Appendix A for Survey Area C0100.
9. During the survey of the reactor systems, Survey Area C0100, Survey Unit 03, Prototype Ion Exchange Demineralizer, one sample of resin for qualitative gamma spectrum analysis showed Co-60 at 60.6 pCi/g. In addition to Co-60, Mn-54, Co-57 and Sb-125 were present in the sample.
10. During the survey of Hazardous Material Environmental, Survey Area H0100, one composite sediment sample result, sent offsite for 10CFR Part 61 analysis, showed Co-60 and Cs-137 concentrations greater than the assumed DCGLs. The sample was collected from 0-6" depth sediment surrounding the vertical inlet drain pipe from the facility to the pond. In addition to Co-60 and Cs-137, Eu-152, U-233/234, U-238 and Pu-241 were present in the sample.

7.3.11 Environmental Core Bore Drill Samples

During the survey of the environmental areas inside the fence surrounding the facility, Survey Area E0100, five locations were selected for core bore drill sampling. One location was sampled as a part of Survey Unit 01 and four locations were sampled as a part of Survey Unit 02. At each core bore sample location, each two foot depth obtained was collected from split spoon sampler equipment as a sample for gamma spectrum analysis. The core bore drill spoils that were deposited outside of the bore hole were composited for an additional sample from each of the locations. The depth of the sample drilling was to twenty feet (20') depth or to the depth of refusal. The samples collected and depth were as follows:

1. From Survey Unit 01, Fuel Transfer Cask Area at location No.1, seven (7) subsurface core bore soil samples and 1 composite soil sample was collected for gamma spectral analysis. A depth of 17' was achieved at the location and all gamma spectrum analysis results less than 50% of the DCGL for Co-60 and Cs-137. One result showed Co-60 present at 0.3 pCi/g at the 3- 5' depth.

2. From Survey Unit 02, Outfall Area Between Facility and Pond at location No.1, five (5) subsurface core bore soil samples and 1 composite soil sample was collected for gamma spectral analysis. A depth of 11' was achieved at the location and all gamma spectrum analysis results were less than 50% of the DCGL for Co-60 and Cs-137.
3. From Survey Unit 02, Outfall Area Between Facility and Pond at location No.4, ten (10) subsurface core bore soil samples and 1 composite soil sample was collected for gamma spectral analysis. A depth of 21' was achieved at the location and all gamma spectrum analysis results were less than 50% of the DCGL for Co-60 and Cs-137. One result showed Co-60 present at 0.5 pCi/g at the 13- 15' depth.
4. From Survey Unit 02, Outfall Area Between Facility and Pond at location No.7, five (5) subsurface core bore soil samples and 1 composite soil sample was collected for gamma spectral analysis. A depth of 11' was achieved at the location and all gamma spectrum analysis results were less than 50% of the DCGL for Co-60 and Cs-137.
5. From Survey Unit 02, Outfall Area Between Facility and Pond at location No.10, ten (10) subsurface core bore soil samples and 1 composite soil sample was collected for gamma spectral analysis. A depth of 21' was achieved at the location and all gamma spectrum analysis results were less than 50% of the DCGL for Co-60 and Cs-137.

7.3.12 Concrete Core Bore Sample Results

During the survey of the facility surfaces, Survey Area A0800, four locations were selected for concrete core bore drill sampling. The sample objective was to determine if contamination was present in the surface of the concrete core, find the depth of contamination by slicing the concrete cores into approximate 1/4" thick slices and then identify and quantify any contamination detected by gamma spectroscopy or through other analyses. A 3" diameter by approximately 6" depth concrete core sample was obtained at the designated sample locations. Each concrete core was marked with regard to floor surface orientation, up to six (6) approximate 1/4" thick slices were cut from the floor surface end of the concrete core, a direct beta measurement was collected from the top and bottom of each slice and remainder of the concrete core, and the top slice and other slices showing elevated beta activity were sent offsite for further analysis. The samples collected and depth were as follows:

1. For Survey Unit 01, the reactor confinement floor, a concrete core bore sample was obtained at a location where a contaminating event had occurred and the area had been remediated and painted over. Six (6) slices were cut from the concrete core, direct beta measurements were collected on the top and bottom of each slice and the top and bottom of the remaining core. The direct beta measurement results showed no total beta activity above the individual measurements' MDA. Slice No.1 was sent offsite for gamma spectrum analysis. The analysis result indicated the presence of Co-60 at 0.05 pCi/g and Cs-137 at 0.05 pCi/g, or less than a tenth of the respective DCGL for Co-60 and Cs-137.
2. For Survey Unit 02, the bioshield wall on the mezzanine level, a concrete core bore sample was obtained at a location where reactor pool water leaks had occurred and contamination may have been present. Four (4) slices were cut from the concrete core, direct beta measurements were collected on the top and bottom of each slice and the top and bottom of the remaining core. The direct beta measurement results showed no total beta activity above the individual measurements' MDA. Slice No.1 was sent offsite for gamma spectrum analysis. The analysis result indicated the presence of Co-60 at 0.07 pCi/g, Sb-125 at 0.12 pCi/g and Cs-137 at 0.04 pCi/g.
3. For Survey Unit 03, the South Access Facility floor, a concrete core bore sample was obtained at a location where reactor pool water leaks had occurred and contamination may have been present. Five (5) slices were cut from the concrete core, direct beta measurements were collected on the top and bottom of each slice and the top and bottom of the remaining core. The direct beta measurement results showed no total beta activity above the individual measurements' MDA. Slice No.1 was sent offsite for gamma spectrum analysis. The analysis result indicated the presence of Co-60 at 0.07 pCi/g and Cs-137 at 0.07 pCi/g, or less than a tenth of the respective DCGL for Co-60 and Cs-137.
4. For Survey Unit 04, Hot Cell room G026 Floor, a concrete core bore sample was obtained at a location where in Survey Area A0200, Survey Unit 03 in the Hot Cell room G026, the one direct beta measurement result was 63,661 dpm/100 cm² on the floor. Four (4) slices were cut from the concrete core, direct beta measurements were collected on the top and bottom of each slice and the top and bottom of the remaining core. There were 3 direct beta measurement results greater than 5000 dpm/45cm². The maximum measurement result was 38,233.2 dpm/45cm². The measurement #4 on the bottom of the

2nd slice at 0.625" depth was less than MDA. Slice No.1 was sent offsite for gamma spectrum analysis and for H-3, C-14 and Sr-90 analyses. The analysis result indicated the presence of Cs-137 at 7,650 pCi/g. H-3, C-14, Co-60 and Sr-90 were reported to be all less than measurement MDA. Slice No.3 was also sent offsite for gamma spectrum analysis. The analysis result indicated the presence of Cs-137 at 1.2 pCi/g.

7.4 Hazardous Material Survey Results

7.4.1 Survey Area G0100, Hazardous Material Surfaces and Structures

UVA Environmental Health and Safety personnel reviewed a report issued in 1988 from an asbestos survey conducted at the Reactor Facility by the Hall-Kimbrall firm. The report, with regard to the UVA Reactor Facility, indicated that floor tiles, sprayed acoustical material in rooms 120 to 129 and some pipe insulation and fittings at various locations in the facility contained asbestos. The report also indicated that ceiling tiles throughout the facility did not contain asbestos. A visual inspection during the pre-survey walk down identified two sheets of asbestos insulation between the UVAR heat exchanger and its supports.

The result of the lead survey was that no lead containing paints were identified with lead concentrations above the EPA action level for lead of 1.0 mg/cm². This included a direct surface soil measurement at a lead storage location behind the facility. There were numerous areas inside and behind the facility where bare lead metal was stored. The lead was mostly in brick form but was also in sheets and various other shapes. Most of the surfaces of the lead metal were bare and many were oxidized. The hazardous material survey results are provided in a report issued by UVA Environmental Health and Safety Department presented in Appendix A, UVA Reactor Facility Characterization Survey Area Reports, for Survey Area G0100.

7.4.2 Survey Area H0100, Hazardous Material Environmental

The characterization for hazardous materials in the environs at the facility was achieved by collecting a composite sediment sample from the pond at the vertical drain pipe position in pond, and where several facility drains enter the pond. The pond sediment sample was sent offsite for 10 CFR Part 61 radionuclide analysis, and for the Envirocare Suite of Hazardous Material analyses. The hazardous material assessment for the UVAR facility and site was accomplished by sending one composite pond sediment sample to Barringer Laboratories, a State of Utah certified laboratory, for the Envirocare Suite of hazardous materials analyses.

The purpose of performing the Envirocare Suite of analysis was to begin the qualification process for sending low-level radionuclide contaminated soils from the UVAR facility and site to Envirocare for disposal. The analyses required by Envirocare for evaluation with regard to their waste acceptance criteria consists of the following:

- TCLPs for 8 RCRA metals plus copper and zinc,
- TCLP for 32 organics and PCBs,
- Totals for metals and organics,
- Asbestos,
- Total Organic Halides (TOX),
- Hydrogen sulfide,
- Hydrogen cyanide,
- Soil pH and Paint Filter Liquids Test,
- Pesticides,
- Standard Proctor Test, and
- Radionuclide analysis (covered by 10 CFR Part 61 sample analysis).

The hazardous material sampling was performed according to GTS Duratek procedures, the characterization plan and survey work package instructions. Instructions were written in the survey work package to collect one composite sediment sample for 10CFR Part 61 radionuclide analysis, and for Envirocare Profile HazMat analysis, to support the UVAR site characterization.

Characterization survey data was reviewed for the location appropriate for the Envirocare profile samples and 0-6" depth composite sample was collected in the vicinity of the vertical end of a drain pipe which runs from the facility to the pond. The sample collection process for the hazardous material characterization survey involved performing the steps that follow:

- Ensured all sampling tools and containers were thoroughly cleaned prior to sample collection,
- Located the selected survey measurement location,
- Ensured an adequate volume of material was recovered at the sample location,
- Deposited the composite into a stainless steel mixing bowl,
- Homogenized the sample,
- Placed the sample in sample containers provided by Barringer Laboratories,
- Marked and sealed the containers, and
- Stored the samples inside a secure refrigeration unit to maintain sample at or below 4°C (39°F) to preserve the sample prior to shipment.

Once all containers with the composited samples were prepared the steps that follow were performed to ship the sample:

- Prepared a Barringer Laboratory chain of custody form,
- Placed samples in a cooler provided by Barringer Laboratory with ice packs,
- Prepared the shipping paper work,
- Notified Barringer Laboratory of the sample shipment and expected delivery date,
- Shipped the samples by Federal Express priority overnight, and
- Observed sample hold time requirements.

The shipment of the samples to Barringer Laboratory was a non-radioactive shipment as regards U.S. DOT regulations, based upon representative characterization data.

The hazardous material sample analysis results are provided in a final report issued by Barringer Laboratory for the pond sediment composite sample No. H0100-01EP1-1. The report is presented in Appendix A, UVA Reactor Facility Characterization Survey Area Reports, for Survey Area H0100. All Envirocare Suite analytes for the hazardous material characterization were analyzed for and reported. The results of the hazardous material analysis was that all analytes analyzed for were below the U.S. Environmental Protection Agency's (EPA) Risk Based Concentration (RBC) values from the EPA Region III RCB table dated October 7, 1999, for soil using the residential criteria. The hazardous materials of concern within the UVAR facility and site were lead and asbestos. The lead and asbestos and were not detected above their respective minimum detectable concentrations (MDC) in the pond sediment sample. Of other hazardous materials analyzed for, the PCB Aroclor 1254 was detected with a result of 48 ug/Kg. The result for PCB Aroclor 1254 is approximately 15% of the EPA Region III RCB table value for soil of 0.32 mg/Kg with the residential criteria and the result is considered to be at a level below regulatory concern.

7.5 10 CFR Part 61 Radionuclide Analysis Results

Two samples were collected and sent to offsite laboratories for 10 CFR Part 61 radionuclide analyses. One sample was a portion of a composite pond sediment sample sent to Barringer Laboratories for the Envirocare Suite of hazardous material analyses and 10 CFR Part 61 radionuclide analyses. The other sample was a resin sample from the reactor pool water clean-up system, and it was sent to Duke Engineering and Services Environmental Laboratory (DESEL) for 10 CFR Part 61 analysis.

The purpose of the sample analysis for the 10 CFR Part 61 radionuclides is to analyze for all radionuclides that may be present at the facility and site, including hard-to-detect radionuclides (HTDR). The HTDRs are radionuclides that would require measurement by means other than direct measurements utilizing portable instrumentation or sampling and measurement by gamma spectroscopy. Typically, HTDRs measurements require special sampling and sample preparation methods and require counting and analysis by liquid scintillation counting, alpha spectroscopy or x-ray spectroscopy.

For the 10 CFR Part 61 analysis, the samples were analyzed such that the relative abundances of any detected radioisotopes would be known and could be used to develop DCGLs and/or remedial action clean-up goals for the UVAR facility surfaces and soils of the environs, based upon known radionuclide distribution.

7.5.1 Pond Sediment Sample 10 CFR Part 61 Analysis Results

During the survey for hazardous materials in the environs, Survey Area H0100, one composite pond sediment sample was collected by GTSD from the location where several facility drains enter the pond. The sample was sent to Barringer Laboratories for 10 CFR part 61 radionuclide analysis. This was a portion of the same sample that was collected and analyzed for the Envirocare Suite of hazardous material analyses as discussed in section 7.4.2 above, and the sample handling, collection and shipping was as stated.

The pond sediment sample was sent to Barringer Laboratory for the 10 CFR Part 61 radionuclide analysis for all radionuclides that could be present, including HTDRs. The results of the analysis are presented in Table 7.4, Pond Sediment Sample 10 CFR Part 61 Analysis Results, that follows and shows the analyses requested, analysis sensitivity or MDA and resulting concentration. The radionuclides detected in the pond sediment sample were Co-60, Cs-137, Eu-152, U-233/234, U-238 and Pu-241. The analysis report from Barringer Laboratory for sample No. H0100-02EP1-1 is located in Appendix A, Survey Area H0100.

Table 7.4
Pond Sediment Sample
10CFR Part 61 Analyses Results

Radionuclide	Type of Analysis	MDA (pCi/g)	Result (pCi/g)
H-3	Liq Scin	0.8	< 0.8
C-14	Liq Scin	20	< 20
Fe-55	X-ray Spec	7	< 7
Co-60	Gamma Spec	0.2	9.5
Ni-63	Liq Scin	8	< 8
Tc-99	Beta Spec	4	< 4
Cs-137	Gamma Spec	0.2	3.7
U-233/234	Alpha Spec	0.8	1.4
U-238	Alpha Spec	0.8	1.0
Pu-238	Alpha Spec	0.2	< 0.2
Pu-239/240	Alpha Spec	0.2	< 0.2
Pu-241	Liq Scin	0.6	3.1
Am-241	Alpha Spec	0.07	< 0.07
Cm-242	Alpha Spec	0.07	< 0.07
Cm-243/244	Alpha Spec	0.06	< 0.06

7.5.2 Resin Sample 10 CFR Part 61 Analysis Results

During the survey of the reactor systems, Survey Area C0100, Survey Unit 02, a resin sample was collected from the reactor pool water clean-up system for gamma isotopic analysis and subsequent 10 CFR part 61 radionuclide analysis. UVAR personnel aided GTSD during the sample collection process. The sample was collected and placed into a 250 ml Marinelli beaker, then analyzed on the GTSD gamma spectroscopy system for qualitative results. The analysis results were considered to be qualitative because the gamma spectroscopy systems' 250 ml Marinelli calibration was for soil samples with a density of approximately 1.7 g/cc, whereas the resin sampled was approximately 1 g/cc in density, and the sample contained water. However, a comparison was made between the onsite analysis and the subsequent offsite 10 CFR Part 61 analysis results. As expected, the onsite results showed that the 250 ml soil calibration yielded approximately 25% higher concentrations for the detected gamma emitting radionuclides present in the sample than did

the offsite analysis results.

The onsite characterization sample analysis results were used only for the purpose of preparing shipping papers to send the sample offsite. Comparing the sample results was not intended due to the calibration considerations of the onsite measurement. A more important aspect of the sampling was the preservation of the material in the "as sampled" condition. Once the resin sample was analyzed and the radionuclide activity was approximated, Duke Engineering and Services Environmental Laboratory (DESEL) was contacted and consulted with regard to analyzing the resin. DESEL prescribed sending a smaller quantity of the resin in a glass vial for the requested analyses. As prescribed by DESEL, 15.5 grams of the resin was collected from the 250 ml Marinelli beaker of resin and placed into a liquid scintillation vial. The remaining resin was placed back into the tank where it had originated. A sample chain of custody form provided by DESEL was completed for the resin sample. A separate chain of custody form from DESEL was completed for concrete core bore samples from Survey Area A0800, but all samples were packaged and shipped by Federal Express together in the same shipment.

The resin sample was sent to DESEL for the 10 CFR Part 61 radionuclide analysis for all radionuclides that may be present, including HTDRs. The results of the analysis are presented in Table 7.5, Resin Sample 10 CFR Part 61 Analysis Results. Shown are the analyses requested, analysis sensitivity or MDA and the resulting concentration. The radionuclides present in the resin sample were Mn-54, Fe-55, Co-60, Ni-63 and Zn-65. The analysis report from DESEL for sample No. C0100-02EQ1-1 is presented in Appendix A, Survey Area C0100.

Table 7.5
Resin Sample
10CFR Part 61 Analyses Results

Radionuclide	Type of Analysis	MDA (pCi/g)	Result (pCi/g)
H-3	Liq Scin	810	< 810
C-14	Liq Scin	790	< 790
Mn-54	Gamma Spec	200	3,350
Fe-55	X-ray Spec	250	1,290
Co-57	Gamma Spec	190	< 190
Co-58	Gamma Spec	320	< 320
Co-60	Gamma Spec	160	6,780
Ni-59	X-ray Spec	180,000	< 180,000
Ni-63	Liq Scin	320	350
Zn-65	Gamma Spec	510	1,590
Sr-90	Gas Flow Prop	290	< 290
Cs-134	Gamma Spec	320	< 320
Cs-137	Gamma Spec	260	< 260
Pu-238	Alpha Spec	3.4	< 3.4
Pu-239/240	Alpha Spec	3.3	< 3.3
Pu-241	Liq Scin	500	< 500
Am-241	Alpha Spec	7.8	< 7.8
Cm-242	Alpha Spec	3.4	< 3.4
Cm-243/244	Alpha Spec	2.8	< 2.8

7.6 Post Characterization DCGL Determination

In order to demonstrate compliance with the 25 mrem/yr criteria for release for unrestricted use, site specific derived concentration guideline levels (DCGLs) should be calculated separately for building surfaces and open land areas. These site specific DCGLs will be calculated based on the relative fraction of each radionuclide in the expected radionuclide mix and radionuclide specific guideline values. The radionuclide specific guideline values relate activity concentrations, in terms of dpm/100 cm² for surfaces or pCi/g for soil, to the criteria for release for unrestricted use.

Initially the DCGLs will be set equal to the radionuclide specific screening values provided by the NRC. These screening values relate residual radioactivity to the criteria for release for unrestricted use based on a set of conservative assumptions. If the calculated site specific DCGLs prove unacceptable low, a site specific dose assessment may be performed to calculate more realistic radionuclide specific guideline values. However, based on the expected radionuclide mixes to be encountered at UVA, a site specific dose assessment is not anticipated. Table 7.6, NRC Screening Values for Building Surfaces, list some of the NRC screening values for building surfaces and Table 7.7, NRC Screening Values for Open Land Areas, list some of the NRC screening values for open land areas. These screening values have either been published in the NRC in the Federal Register, or were calculated using the NRC's DandD Code, Version 1, as indicated in the source column of the tables. If additional radionuclides are identified, the NRC's DandD Code can also be used to calculate additional screening values.

TABLE 7.6
NRC Screening Values for Building Surfaces

Radionuclide	Source*	dpm/100 cm ²
Mn-54	Fed. Reg.	32,000
Fe-55	Fed. Reg.	4,500,000
Co-57	DandD	210,000
Co-60	Fed. Reg.	7,100
Ni-63	Fed. Reg.	1,800,000
Zn-65	DandD	48,000
Sb-125	DandD	44,000
Cs-137	Fed. Reg.	28,000
Eu-152	DandD	13,000
Eu-154	DandD	11,000
U-233	DandD	88
U-234	DandD	90
U-235	Fed. Reg.	97
U-238	Fed. Reg.	100
Pu-241	DandD	1,400

*Note: Fed. Reg. = Federal Register

TABLE 7.7
NRC Screening Values for Open Land Areas

Radionuclide	Source*	Soil pCi/g
Mn-54	Fed. Reg.	15
Fe-55	Fed. Reg.	10,000
Co-57	Fed. Reg.	150
Co-60	Fed. Reg.	3.8
Ni-63	Fed. Reg.	2100
Zn-65	DandD	6.3
Sb-125	DandD	26
Cs-137	Fed. Reg.	11
Eu-152	Fed. Reg.	8.7
Eu-154	Fed. Reg.	8.0
U-233	DandD	8.9
U-234	Fed. Reg.	13
U-235	Fed. Reg.	8
U-238	Fed. Reg.	14
Pu-241	Fed. Reg.	72

*Note: Fed. Reg. = Federal Register

For building surfaces it is assumed that the final status surveys used to demonstrate compliance with the criteria for release for unrestricted use will be performed with large area gas flow proportional counters, or equivalent, calibrated using a NIST traceable source. Such surveys will include scans as well as fixed point measurements. If only one radionuclide is present, then the site specific DCGL is the appropriate screening value from Table 7.6. Since the survey technique cannot provide radionuclide specific information, if more than one radionuclide is present, the site specific DCGL can be calculated using the equation that follows:

$$DCGL = \frac{F}{\sum \frac{f_i}{GL_i}}$$

Where:

- DCGL = The site specific DCGL for building surfaces, dpm/100 cm².
- F = The fraction of the radionuclide mix which is considered detectable based on the radionuclide beta yields and end-point energies relative to the radionuclide used to calibrate the detector (unit less).
- f_i = The relative fraction of radionuclide i in the mix (unitless).
- GL_i = The radionuclide specific guideline value (NRC screening values) for radionuclide i, dpm/100 cm².

If more than one radionuclide mix is identified, site specific DCGLs can be calculated for each radionuclide mix. In such cases a decision will have to be made as to applying the DCGLs separately, calculating an average DCGL to apply all cases, or applying the most conservative DCGL in all cases.

For open land areas it is assumed that the final status survey will include gamma scans and samples collected for gamma spectral analysis. Additional analyses may prove useful, at least on a limited number of samples, if no gamma emitting radionuclides were present. If only one radionuclide was present, then the site specific DCGL is the appropriate screening value from Table 7.7.

For open land areas, where more than one radionuclide is present the concept of a site specific DCGL does not necessarily apply since the analysis results are expected to be isotope specific. In such cases the "sum of the fraction rule" applies and is calculated as follows:

$$\sum \frac{\text{Conc}_i}{\text{GL}_i} \leq 1$$

Where:

Conc_i = Activity concentration of radionuclide i, pCi/g.

GL_i = The radionuclide specific guideline value (NRC screening values) for radionuclide i, pCi/g.

If, based on analysis results, activity concentrations are known for each radionuclide that may be present in the mix the above equation may be evaluated directly. For radionuclide mixes, for which the activity concentration of all radionuclides that may be present are not known, the activity concentrations of the unknown radionuclides must first be estimated by the application of previously defined scaling factors.

7.7 Survey Records

The project team maintained records of surveys in the survey packages for each area according to project procedures. The completed survey packages with the information listed below is located in this report in Appendix C, UVA Reactor Facility Characterization Survey Packages. The survey package included the following records:

- Survey Package Worksheet giving the package identification, survey location information, general survey instructions and any specific survey instructions.
- Survey Comment Addendum containing comments from the survey technician regarding any unusual situation encountered while surveying.
- The Survey Unit Drawing of the area to be surveyed as available.
- Printout of laboratory analysis results.
- Ludlum Model 2350-1 data files and Paradox[®] converted values for all radiation survey measurements.

The survey team took direct measurements for total alpha and beta surface activity using the Ludlum Model 2350-1 Data Logger system. Upon completion of a survey, the contents of the Data Logger's memory was downloaded to a database. GTS Duratek used a proprietary computer program to download and generate a survey report that presents all raw data, converted data and information by survey location. The survey technician and supervisor reviewed these reports for completeness, accuracy, suspect entries and compared the data to the guideline values.

Any changes to the database tables such as detector efficiency and background, that could affect survey results required supervisor approval. In addition, changes to data in the primary table required a written explanation on a change request. There were no changes made to the database tables that would affect survey results. However, changes were made to correct location code entry errors identified while reviewing the download report generated at the completion of each survey. These changes were made by the technician downloading the instrument with the approval of the characterization supervisor. The change reports are attached to the survey download report and maintained as a permanent record. The Model 2350 download reports are presented in Appendix C, UVA Reactor Facility Characterization Survey Packages.

Data and document control included the maintenance of the raw data files, translated data files (Paradox[®] database files) and documentation of all corrections made to the data. The databases were backed up on a daily basis. GTS Duratek then turned all original survey records over to UVA with the final report.

8.0 QUALITY ASSURANCE AND QUALITY CONTROL

GTS Duratek's Quality Assurance/Quality Control Programs ensured that all quality and regulatory requirements were satisfied. All activities affecting quality were controlled by procedures and the Quality Assurance/Quality Control Plan. These documents include the following Quality Control measures as an integral part of the survey process. The QA/QC information listed below is located in this report in the Appendix B, UVA Reactor Facility Characterization Survey QA/QC Data.

8.1 General Provisions

8.1.1 Selection of Personnel

Project management and supervisory personnel were required to have extensive experience with GTS Duratek procedures and the QA/QC plan and be familiar with the requirements of the Survey and Sampling Plan. Management had prior experience with the radionuclide(s) of concern and a working knowledge of the instruments used to detect the radionuclides on site.

GTS Duratek selected supervisory personnel to direct the survey based upon their experience and familiarity with the survey procedures and processes. Likewise, Health Physics technicians who performed the surveys were selected based upon their qualifications and experience.

8.1.2 Written Procedures

All survey tasks which were essential to survey data quality were controlled by procedures and the survey plan. A list of plans and procedures is provided in the Project Management Plan (Ref. 7.1).

8.1.3 Instrumentation Selection, Calibration and Operation

GTS Duratek selected instruments proven to reliably detect the radionuclides present at the UVA facility. Instruments were calibrated by GTS Duratek or qualified vendors under approved procedures using calibration sources traceable to the National Institute of Standards and Technology (NIST). All detectors were subject to daily response checks when in use.

Procedures for calibration, maintenance, accountability, operation and quality control of radiation detection instruments implement the guidelines established in American National Standard Institute (ANSI) standard ANSI N323-1978 and ANSI N42.17A-1989.

8.1.4 Survey Documentation

The survey packages were the primary method of controlling and tracking the hard copy records of survey results. Records of surveys were documented and maintained in the survey package for each area according to GTS procedures. Each survey measurement was identified by the date, technician, instrument type and serial number, detector type and serial number, location code, type of measurement, mode of instrument operation, and Quality Control (QC) sample number, as applicable.

8.1.5 Chain of Custody

Procedures established responsibility for the custody of samples from the time of collection until results were obtained. All samples shipped off site for analysis were accompanied by a chain-of-custody record to track each sample.

8.1.6 Records Management

Generation, handling and storage of survey data packages were controlled by an approved procedure.

8.1.7 Independent Review of Survey Results

The survey package and survey data from each area received an independent review to verify all documentation is complete and accurate.

8.2 Training

All project personnel received site specific training to identify the specific hazards present in the work and survey areas. Training also included a briefing and review of the Survey Plan, GTS procedures, the QA Plan(Ref. 7.2) and the Site Safety and Health Plan (Ref. 7.3). Copies of all training records were maintained on site through the duration of the onsite activities. During site orientation and training, survey personnel became familiar with site emergency procedures.

8.3 Sample Analysis

GTS Duratek performed quality assurance checks on a minimum of 5% of all samples analyzed by gamma spectroscopy. This consisted of the analysis of split, duplicate, blank and spike samples. Split samples were analyzed if an ample amount of material was collected in a sample. The sample was homogenized and split into two separate samples for analysis. Duplicate analysis consisted of the analysis of the same sample twice at different times to check the quality of the analyses. The daily background checks also met the purpose of QC sample blanks. The daily

background checks, or QC sample blanks, showed the gamma spectroscopy system was free of radionuclide contamination during operations and provided a record that the *a priori* MDA goals were met for each radionuclide listed in the systems' analysis library. The daily QC source check spectrums also met the purpose of QC sample spikes. The results of the daily QC source checks, or sample spikes, were plotted on QC control charts to provide a daily record of the systems' performance. The system performance was evaluated for each days use for low and high energy activity, low and high energy peak channel and high energy peak FWHM (Full Width Half Maximum) system parameters.

GTS Duratek returned all samples to UVA for UVA QA verification of results at their discretion and for archiving. QC data and calibration documentation for the instruments used for the characterization and the results of the gamma spectroscopy duplicate, split, blank and spike samples are presented in Appendix B, UVA Reactor Facility Characterization QA/QC Data. The daily QC checks performed for the Ludlum Model 2350 Datalogger and associated detectors are presented with the survey packages, Model 2350 downloads, Tenelec data reports and the gamma spectroscopy sample analysis reports in Appendix C, UVA Reactor Facility Characterization Survey Packages.

9.0 REFERENCES

- 9.1 *Project Management Plan for the University of Virginia Reactor Facility*, Rev 0, July 1999, GTS Duratek.
- 9.2 *Quality Assurance Project Plan for the UVA Characterization and Decommissioning Plan Support Project*, UVA-QAP, Rev 0, July 1999, GTS Duratek.
- 9.3 *Site Safety and Health Plan for the UVA Reactor Characterization*, Rev 0, July 1999, GTS Duratek.
- 9.4 Draft NUREG-1549, *Using Decision Methods for Dose Assessment to Comply with Radiological Criteria for License Termination*, February 3, 1998.
- 9.5 Regulatory Guide DG-4006, *Demonstrating Compliance with the Radiological Criteria for License Termination*, March 13, 1998.
- 9.6 NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)*, December 1997.
- 9.7 10 CFR20, *Standards for Protection Against Radiation*.
- 9.8 American National Standard Institute (ANSI) Standard ANSI N323-1978
- 9.9 ANSI Standard ANSI N42.17A-1989.
- 9.10 Federal Register, Volume 63, No. 222, Wednesday, November 18, 1998 Notices.
- 9.11 Federal Register, Volume 64, No. 234, Tuesday, December 7, 1999 Notices.
- 9.12 RESRAD, Version 5.82

10.0 ATTACHMENTS

Attachment 1, UVAR Facility Characterization Procedures
Attachment 2, Characterization Assumed Surface Contamination DCGL
Attachment 3, Characterization Assumed Soil Concentration DCGL

11.0 APPENDICES

Appendix A, UVA Reactor Facility Characterization Survey Area Reports
Appendix B, UVA Reactor Facility Characterization QA/QC Data
Appendix C, UVA Reactor Facility Characterization Survey Packages

Attachment 1
UVAR Facility Characterization Procedures

	PROC. NO.	TITLE	REVISION
1.	REDS-CHM-101	Sample Identification and Chain-of-Custody	2
2.	REDS-CHM-108	Data Quality Management Guidelines for Offsite Laboratories	0
3.	REDS-CHM-205	Sample Preparation for Gross Alpha and Gross Beta Analysis	0
4.	REDS-CHM-206	Sample Preparation for Gamma Spectral Analysis	0
5.	REDS-CHM-216	Operation of Denver Instrument Model XE3000 Analytical Balance	1
6.	REDS-CHM-217	Preparation of Smears for Dual Labeled Analysis of C-14 and H-3.	0
7.	REDS-CHR-100	Site Characterization Plan	1
8.	REDS-CHR-101	Characterization of Structures	0
9.	REDS-CHR-102	Characterization of Systems	0
10.	REDS-CHR-106	Surface Soil Sampling	0
11.	REDS-CHR-107	Subsurface Soil Sampling	0
12.	REDS-CHR-108	Surface Soil Surveys and the Collection of Water, Sediment, Vegetation and Surface Soil Samples	2
13.	REDS-CSA-203	Ludlum Model 2350 Series Data Logger Download Operation	0
14.	REDS-HS-106	Operation of the Industrial Scientific TMX410 Multi-Gas Monitor	1
15.	REDS-INST-100	Radiation Protection Instrumentation Program	2
16.	REDS-INST-101	Issue, Control and Accountability of Radiation Protection Instrumentation	2
17.	REDS-INST-102	Quality Control of Counting Systems and Portable Counters	2
18.	REDS-INST-104	Calibration and Test Requirements for Radiation Protection Instrumentation	0
19.	REDS-INST-105	Quality Control for Gamma Spec Systems	0
20.	REDS-INST-201	Operation of Ludlum Model 2350 Data Logger	4
21.	REDS-INST-204	Operation of Eberline RO-7 High Range Ion Chamber	0
22.	REDS-INST-206	Operation of the Ludlum Model 19 Micro-R Meter	0
23.	REDS-INST-207	Operation of Eberline Ion Chamber Model RO-2/RO-2A	2
24.	REDS-INST-209	Operation of Ludlum Model 177 Portable Frisker	1
25.	REDS-INST-213	Operation of Tennelec LB5100/W Alpha/Beta Counter	1

	PROC. NO.	TITLE	REVISION
26.	REDS-INST-247	Operation of the EG&G Nomad Portable Gamma Spectroscopy System with Gamma VisionTM-32 Software	0
27.	REDS-INST-249	Operation of the DCA Model 3090 Monitor	0
28.	REDS-INST-313	Calibration of Tennelec LB5100/W Alpha Beta Counter	1
29.	REDS-INST-314	Calibration of EG&G NOMAD Portable Gamma Spectroscopy System	1
30.	REDS-INST-330	Pipe Monitor Efficiency and Response Test	0
31.	REDS-OPS-301	Performance of Surveys	5
32.	REDS-OPS-302	Survey Documentation and Review	4
33.	REDS-PDC-102	Document Control	1
34.	REDS-PDC-103	Preparation, Review and Approval of Documents	1
35.	REDS-PDC-104	Acknowledgment of Document Understanding	1
36.	REFS-QA-2.1	Personnel Training and Indoctrination	1
37.	REFS-QA-12.1	Control of Measuring and Test Equipment	1
38.	REFS-QA-15.1/16.1	Nonconformance Reporting and Corrective Action	0
39.	REFS-QA-17.1	Quality Records Management	2
40.	REFS-QA-18.1	Audit Program	0

Attachment 2
 Characterization Assumed Surface Contamination DCGL

Table A1

Assumed UVA Radionuclide Mix
Assumed Derived Concentration Guideline Level (DCGL) for MDA Determination for Direct Beta Measurements

		Est.		G		Detectable
Radionuclide	Halflife [Yr]	Activity(uCi) [8/1/1999]	f	dpm/100cm2 25 mrem/yr	f/G	Amount dpm/100cm2
Co-60	5.27e+00	2.00e+00	5.00e-01	7.10e+03	7.04e-05	4.63e+03
Sr-90	2.86e+01	1.00e+00	2.50e-01	8.70e+03	2.87e-05	2.31e+03
Cs-137	3.02e+01	1.00e+00	2.50e-01	2.80e+04	8.93e-06	2.31e+03
	Totals	4.00e+00	1.00e+00			9.25e+03
			Gross Activity	DCGL =	9.25e+03	dpm/100cm2

*Note: Values for "G" from Federal Register / Vol. 63, No. 222 / Wednesday, November 18, 1998 / Notices, Table 1.

**Attachment 3
Characterization Assumed Soil Concentration DCGL**

Table A2							
Assumed UVA Radionuclide Mix							
Derived Concentration Guideline Level [DCGL] for MDA Determination for Gamma Spectrum Analysis							
		Est.		RESRAD	G		
Radionuclide	Half-life	Activity(uCi)	f	mrem/yr	pCi/g per	f/G	Limit
	[Yr]	[8/1/1999]		per pCi/g	25 mrem/yr		(pCi/g)
Co-60	5.27e+00	2.00e+00	5.00e-01	9.45e+00	2.65e+00	1.89e-01	1.91e+00
Sr-90	2.86e+01	1.00e+00	2.50e-01	5.06e+00	4.94e+00	5.06e-02	9.53e-01
Cs-137	3.02e+01	1.00e+00	2.50e-01	2.29e+00	1.09e+01	2.29e-02	9.53e-01
	Totals	4.00e+00	1.00e+00				3.81e+00
				Gross Activity	DCGL =	3.81e+00	(pCi/g)