ENERGYSOLUTIONS CS-HP-PR-008

Characterization Report for the Aerotest Radiography & Research Reactor

[redacted], California

Project No. 313150 **Revision 0**

Authored By:	Steve Croslin, CHP, Principal Health Physicist	10/12/11 Date
Reviewed By:	Paul Ely, Senior Radiological Engineer	10/12/11 Date
Reviewed By:	Arthur J. Palmer, CHP, PMP, Director, Health Physics & Radiological Engineering	10/12/11 Date
Approved By:	Ben Sklar, Project Manager	10/12/11 Date

- [X] New
- [ ] Title Change
- [ ] Revision
- [ ] Rewrite
- [ ] Cancellation

Effective Date: ____________

TABLE OF CONTENTS

Section **Page**

ATTACHMENTS

APPENDICES

TABLE OF FIGURES

Figure | **Page**

TABLE OF TABLES

Table | **Page**

List of Acronyms and Abbreviations

ACM	Asbestos Containing Material
ALARA	As Low As Reasonably Achievable
Am	americium
ARRR	Aerotest Radiography and Research Reactor
C	carbon
Cf	californium
CFR	Code of Federal Regulations
Ci	curie
cm	centimeter
Co	cobalt
cpm	counts per minute
Cs	cesium
dpm	disintegrations per minute
dpm/100 cm^2	disintegrations per minute per 100 square centimeters
DQO	Data Quality Objective
Eu	europium
GM	Geiger-Mueller
H-3	tritium
HSA	Historical Site Assessment
kW(t)	kilowatt thermal
MARSSIM	Multi Agency Radiation Survey and Site Investigation Manual
MDA	Minimum Detectable Activity
MDC	Minimum Detectable Activity Concentration
mR/hr	milliRoentgen per hour
NaI	sodium iodide
NIST	National Institute for Standards and Technology
PCBs	polychlorinated biphenyls
pCi/g	picoCuries per gram
Pu	plutonium
QA	Quality Assurance
QC	Quality Control
RCRA	Resource Conservation and Recovery Act
TEDE	Total Effective Dose Equivalent
U	uranium
USNRC	United States Nuclear Regulatory Commission

EXECUTIVE SUMMARY

This Characterization Report provides the results of characterization and survey activities performed at the Aerotest Radiography and Research Reactor (ARRR) facility in [redacted], California. The primary purpose of the characterization report is to provide information for development of a Decommissioning Plan for the facility. The reactor facility is owned and licensed by Aerotest Operations, Inc., which awarded a contract to Energy*Solutions*, LLC to perform characterization activities and develop a Decommissioning Plan and associated cost estimate. The characterization activities were designed to define the nature, extent and location of residual radioactive material and other hazardous materials that remain in the facility. For reactor facilities such as the ARRR, characterization also includes determining the amount of neutron-activated materials in structural components and in the biological shield. This Characterization Report includes survey results for alpha, beta and gamma removable activity and fixed activity on surfaces such as floors, walls, equipment and in some areas, on ceilings. It also includes exposure rate measurements taken throughout the facility to determine general area gamma radiation levels and surveys of specific radioactive sources and radioactive waste currently present at the facility. In addition to radiological surveys, samples of soil, wood, spent resin, water and other known or potentially contaminated materials were collected and sent to an off-site laboratory for analysis.

The characterization process was performed in accordance with Energy*Solutions* plan CS-RS-PN-017, *Characterization and Survey Plan for the Aerotest Radiography & Research Reactor* to identify and document the radiological and other hazardous conditions at the ARRR facility. The information gathered will be used to develop the necessary decommissioning work evolutions, appropriate hazard controls, and estimated waste volumes for the facility Decommissioning Plan and Decommissioning Cost Estimate. Before performing characterization activities, a Historical Site Assessment (HSA) Report was generated using information gathered by Energy*Solutions* during an on-site visit that took place in March 2011. The HSA process included reviewing facility historical operations, obtaining reactor and facility design data, reviewing operational data and reports, conducting interviews with in-house personnel, and obtaining facility radiological survey data. This information was used to ensure that radiological surveys, sampling, and assessments performed during characterization activities were properly designed.

The HSA Report identifies certain systems, structures, and components that might contain asbestos-containing materials (ACM), mercury-containing equipment (switches, thermometers, light ballasts, etc.), polychlorinated biphenyls (PCBs) in electrical components and other hazardous materials such as lead. Historical facility experiments, past operations, spills and leaks or material storage could have possibly caused cross-contamination of hazardous materials with radioactive material, which results in mixed waste. As a result, characterization included analysis of materials and systems to determine the potential presence of mixed low-level radioactive material.

The data collected during the characterization will contribute to estimates of radiological and mixed waste volumes, methods for removal and packaging of waste materials, and disposal options for radioactive and mixed waste generated during decommissioning. This data will also

be incorporated into the decision-making process for development of a Final Status Survey (FSS) Plan for the eventual closeout and unrestricted release of the facility.

1. INTRODUCTION

The Aerotest Radiography and Research Reactor (ARRR) is located at [redacted] The ARRR is a TRIGA type reactor that reached initial criticality in 1965 and was operated for almost 45 years, before operations ceased in 2010. The reactor facility is owned and licensed by Aerotest Operations, Inc., a subsidiary of Autoliv ASP, Inc. The main use of the reactor was performing neutron radiography to determine structural integrity for aerospace, automotive, medical and various other components and materials.

Energy*Solutions*, LLC was selected by Aerotest Operations to develop a Decommissioning Plan and associated cost estimate for the facility. The steps involved in preparation of a Decommissioning Plan include gathering detailed facility information, developing a Historical Site Assessment, preparing a Characterization Plan, performing field surveys and sampling for characterization of the facility and preparation of a characterization report. This report documents the methods that were used to perform surveys and sampling of potentially radioactive components and systems at the facility, and the results of the characterization activities. In addition to survey and sample results, the characterization report also addresses the methods that were used for neutron activation analysis calculations, reactor and system modeling, and the resulting estimated activation products in facility structural materials.

The facility information needed for final decommissioning plans encompass the entire facility footprint, which means identification of all structural materials, components, and systems along with construction techniques and locations at the facility. The data collected during characterization surveys, plus estimates of activated materials, will help determine the various waste streams present, the total waste volumes expected, and the best dismantlement techniques for decommissioning of the facility.

The surveys conducted at the ARRR included measurements of removable contamination levels, total (fixed plus removable) surface activity levels, radiation exposure rates and sampling of various media for analysis at an off-site laboratory. The off-site analyses determined the specific radionuclides present, and in some cases, the volumetric concentration of radionuclides. Systematic measurements and samples were largely driven by the data needs identified in ASTM E 1892-97, *Standard Guide for Preparing Characterization Survey Plans for Decommissioning Nuclear Facilities* (Reference 7.0-1) and measurement protocols provided in *Multi-Agency Radiation Survey and Site Investigation Manual* (MARSSIM) (Reference 7.2-2). In addition, biased measurements were performed and biased samples were obtained at locations that were the most likely to exhibit elevated contamination or radiation levels, based on site operations.

The data quality objectives (DQOs) for the characterization work were defined in terms of six characteristics necessary for a high level of confidence: (1) precision, (2) accuracy, (3) representativeness, (4) completeness, (5) comparability, and (6) detection limits.

2. BACKGROUND AND FACILITY DESCRIPTION

2.1 Background Information

The Aerotest facility consists of a reactor building with a high bay section, control room, offices, a lunchroom/conference room, machine shop, laboratories and storage areas. The Reactor Building footprint is about 3,200 square feet and has two floor levels, and the total footprint for all buildings is 9,250 square feet. The reactor is a light water cooled and moderated TRIGA Conversion type reactor that had a steady state thermal power rating of 250 kilowatts (kW). The reactor is licensed pursuant to 10 CFR Part 50, *Domestic Licensing of Production and Utilization Facilities* (Reference 9.5) and is operated under NRC Facility License # R-98. The facility also maintains Radioactive Material License # 2010-07 with the state of California for possession and use of radioactive material that is independent of the reactor.

The TRIGA reactor was operational from 1965 until 2010, when Aerotest ceased operations to comply with NRC requirements concerning foreign ownership issues.

2.2 Facility Description

2.2.1. *Reactor Building and Reactor*

The reactor building dimensions are 40-ft by 80-ft by 22.5-ft tall, with corrugated steel siding and a steel roof over a concrete slab. Sheet rock covers some interior walls and other areas are only covered with insulation blankets. [redacted] can access most of the reactor building high bay area. Besides the reactor bay and control room, the Reactor Building contains several offices, a conference room/lunch room, bathrooms, change facilities, small laboratories, a waste storage room, and a machine shop. The ARRR building ventilation system was designed with two objectives: (1) provide normal heating, cooling, and ventilation functions for personnel comfort and equipment cooling; and, (2) control airflow to protect personnel from exposure to airborne radioactivity and prevent the spread of contamination. Airflow throughout the facility has been designed so that the control room, conference room/lunch room, offices, rest rooms, laboratories, waste storage room and the machine shop are at positive pressure with respect to the reactor high bay area.

The TRIGA reactor is a pool-type facility that is fueled with standard TRIGA fuel elements enriched to less than 20 % (weight percent) uranium-235. The reactor core forms a right circular cylinder and consists of a lattice of cylindrical fuel elements and graphite dummy elements immersed in water. The elements are spaced so that 33% of the core volume is occupied by water. The fuel elements are supported by upper and lower grid plates with 127 grid positions available for core components including fuel elements, graphite dummy elements, control rods, a neutron startup source

and a removable "glory hole" facility. The power level of the TRIGA reactor is controlled by the use of three boron carbide control rods.
The unique fuel elements consist of a solid homogeneous alloy of uranium fuel and zirconium-hydride moderator (U-ZrH). As of September 2011, there are 60 aluminum fuel elements in the core and 17 aluminum fuel elements in the pool, all of which have an enrichment of 8 weight percent (wt %) U-235. In addition, there are 26 stainless steel (SS) elements in the core and 12 unused SS elements in storage, all of which are 12 wt % U-235. There is also one single SS element in the core that is 8 wt % U-235.
The reactor fixed core rests on the bottom of a reactor tank, which has a diameter of 10-ft and is 23-ft deep. A 2.0 curie Am/Be neutron start up source is stored in an aluminum tube in the reactor tank. The reactor tank is embedded in a concrete floor, extends 22-ft below the surface of the floor and 1-ft above the floor, and there is a minimum of 1-ft of concrete at the bottom of the tank. The tank has an open top and ¼-inch aluminum walls, which are set in concrete.

There is a 2-ft wide by 2-ft deep trench in the concrete surrounding the reactor tank, which is used for piping and control cables. A radial extension of the trench leads to the reactor control room, which is located to the west, and a second trench extends to the east for routing core water to the demineralizer system and cooling water loops. A third trench extends to the south and houses a fan that cools the control rod magnets. Any liquid in the trench drains into the Liquid Waste Storage tanks, located on the south side of the facility.

There are six (6) fuel storage pits, each 12-ft deep by 14-in diameter, [redacted] The storage pits were designed to store irradiated fuel, but all six are currently empty. Four (4) of these pits are sealed and covered by carpet and two are now enclosed within the [redacted], which was built at a later date. The reactor pool contains several specialized fuel handling tools that are designed for handling fuel elements or other unique components.

A 20-inch thick by 80-inch tall block wall made from high density concrete encloses the reactor area above the floor level and acts as a bioshield. The top of the bioshield consists of 11-inch thick wood (fir) beams for neutron attenuation and capture. The water within the reactor tank, plus the surrounding concrete and earth, provide the required shielding, so that no containment building was necessary. The primary cooling system includes a primary heat exchanger and a cooling tower. The demineralizer system is used to process the reactor cooling water to ensure proper purity (e.g., pH, conductivity).

The radiation and radioactive gaseous effluent monitoring systems consist of measurement devices and associated circuits, which automatically actuate visual and audible alarms when pre-set limits are exceeded for radiation levels and gaseous activity in the area above the reactor water tank.
The general floor plan of the Aerotest facility is shown on Figure 2-1.

2.2.2. Auxiliary Systems

Water, moving by natural convection, is employed as the primary reactor coolant. The ARRR also includes a flow path through a heat exchanger if forced convection cooling of the reactor is desired. The primary circulating system draws water from near the top of the pool and pumps it through a heat exchanger where a secondary loop removes heat from the primary coolant. The secondary water is pumped through a forced air cooling tower, which cools the secondary water by evaporation. The pool water is then returned to the bottom of the pool.

The Heat Exchanger Shed houses the primary coolant loop shell and tube heat exchanger. The primary coolant loop is composed of 3” diameter stainless steel piping and a 3 phase, 220-volt pump that routes the water from the reactor pool to the heat exchanger. The 5.0 horsepower (HP) primary pump has a capacity of 171 gallons per minute (gpm). The piping that returns the primary coolant to the reactor pool is 3” diameter stainless steel. The heat exchanger is approximately 1-ft diameter by 6-ft long.

The secondary coolant loop connects to a cooling tower located outside of the Reactor Building. The secondary coolant loop is almost identical to the primary loop, consisting of a 3 phase, 220 volt pump. The secondary pump is 5 HP and has a capacity of 175 gpm. Piping that carries the secondary coolant to the heat exchanger and to the cooling tower is 3” diameter stainless steel. There is a 3 phase, 200-volt, 2.0 HP fan on the cooling tower.

The reactor water treatment system is designed to remove impurities, including radioactive contamination, and maintain low water conductivity and optical clarity. The water treatment system consists of a suction line, a pump, filters, demineralizer, flow switches, and associated piping and controls.

2.2.3. Neutron Radiography and Experimental Facilities

The Neutron Radiography facility, the facility used for most of the neutron irradiation work, is located within the reactor complex and consists of a radiography set up room, a dark room and a cold work area. Neutron radiographs were in the Radiography Set-up Room and neutron exposures were performed in a room fed by a neutron guide tube originating in the reactor. The Vertical Beam Tube consists of a hollow sealed tube filled with helium. It is 23-ft tall and 8” by 10” at the base and it enlarges to 22” by 34” at the top. The bottom 48” of the tube is filled with graphite and the lower 84” is covered with lead, 3” thick on the reactor side and 1” thick on the other sides. A beam shutter offers five aperture settings and is located just above the graphite.

Small items were lowered directly into a void space (the glory hole) in the reactor core for irradiation. In-core irradiation could be performed using an aluminum tube of 1.5 inch outside diameter, which fits into any fuel element hole and extends from above the top of the wooden reactor shield to the lower grid plate. The tube is not filled with water, and it was used to lower material to be irradiated through the tube into the core region.

The only access for large components to the Neutron Radiography facility from within the reactor complex is through removable plugs in the ceiling of the facility. This design allows the Neutron Radiography facility to be serviced by the overhead crane within the reactor complex.

2.2.4. Offices and Laboratory Wing

The main entrance into the Aerotest Reactor facility is through the front door of the office and laboratory wing. The office and laboratory wing contains several offices, corridors, laboratories, reactor control room, counting room, utility rooms, storage rooms, restrooms, etc. Access to the control room is from the reactor complex or through the office and laboratory wing. A darkroom is used for developing radiographic film. There is also storage space for film, chemicals, supplies and miscellaneous equipment used for radiography.

The second floor includes two chemistry labs, the neutron gauge office, a sheet metal fabrication area, a calibration lab, an electronics lab and storage areas.

All major rooms and areas in the ARRR facility are identified on Figure 2-2.

[redacted]

Figure 2-1: Aerotest Facility General Floor Plan

[redacted]

3. SUMMARY OF TECHNICAL APPROACH AND ORGANIZATION

3.1 Technical Approach

The Characterization and Survey Plan (CS-RS-PN-017) provided the methods for collection of a combination of systematic and biased measurements and samples to ascertain the radiological condition of the Aerotest facility. The systematic measurements were largely be driven by the data needs identified within the ASTM E 1892-97 (Reference **0**) and the survey and sampling protocols in MARSSIM (Reference **2**). Energy*Solutions* systematically assessed the radiological conditions of the facility outside of the reactor and biological shield systems using guidance in the plan. Since systematic (random) measurements do not always reveal important radiological information about the facility, the survey process included some biased measurements and sampling. Biased measurements were performed at sumps, systems that handled primary coolant, the demineralizer system, and other locations that were more likely to exhibit contamination levels that could alter decommissioning efforts.

Through a standard process of fixed-point surveys, scanning surveys, removable contamination measurements, and gamma radiation surveys, the radiological conditions of the facility were determined and documented. A statistical basis was used for the survey design, and pre-defined data quality objectives (DQOs) were established to ensure the proper amount of data was collected and ensure the data was of high quality. The survey and sampling methods listed in Section 5.0 were used to implement the Characterization and Survey Plan.

An accurate estimate of the induced radioactivity of reactor components and the surrounding biological shield are needed to accurately estimate waste volumes for activated materials, determine the best dismantlement techniques, to estimate potential personnel doses and to plan the radiological controls necessary for decommissioning. Theoretical neutron activation calculations were performed by Energy*Solutions* to estimate the identity and quantity of activation products in reactor components and the bioshield (e.g., the reactor pool walls, floor and cover material). To supplement neutron activation calculations, radiological surveys were done inside the bioshield cover and samples of the wood covering the bioshield were obtained, as described in detail in Section 5.0.

3.2 Organization

The Energy*Solutions* on site project team consisted of a Project Manager (PM), a Health Physicist, two Health Physics technicians and subcontractor personnel for soil sampling. The Energy*Solutions* PM was responsible for managing the on-site project team, personnel safety and ensuring the successful completion of the characterization surveys and sampling. The Health Physicist assisted the PM in the areas of oversight of surveys and sampling, making necessary adjustments in the field, and review of instrument QC data and the characterization data. The Health Physics technicians performed most of the surveys and sample collection. A subcontractor (Strongarm Environmental Field Services) used a Geoprobe™ direct push system to obtain surface and deep soil samples.

Aerotest Operations personnel assisted with characterization activities by providing detailed information about the reactor and auxiliary systems, allowing Energy*Solutions* access to secured areas as needed, removing the reactor bioshield cover for surveys inside the bioshield, removing fuel and pool handling tools for surveys, and overseeing all survey and sampling efforts.

4. CHARACTERIZATION SURVEYS

4.1 Survey Design

Several different general types of radioactive material are normally present in a reactor facility, including discrete radiation sources, neutron activation products in components and structures, and contamination consisting of (normally) small amounts of fission and activation products that have migrated out of the reactor core and deposited into other systems, building surfaces or possibly adjacent soil.

The ARRR facility characterization included the following major tasks:

- Obtained background radiation measurements for each of the various types of survey instruments used, to ensure instruments were capable of detecting activity with a reasonable minimum detectable activity (MDA).
- Performed radiological characterization of outdoor areas adjacent to the facility, such as paved areas (parking lots, sidewalks, etc.) and soil areas. Activities included collection of surface soil samples around the perimeter of the facility, performing direct contamination surveys and collecting wipes (smears) for measurement of removable contamination.
- Performed radiological surveys of the main facility interior rooms and major structural components, which included direct contamination surveys and collecting wipes for measurement of removable contamination.
- Performed radiological surveys of the liquid waste storage tanks, the demineralizer system, the heat exchanger, main cooling tower, back-up cooling tower and other auxiliary systems.
- Surveyed sinks and drain lines in laboratories and other rooms that handled radioactive material.
- Collected samples of sub-surface soil at several locations adjacent to the main ARRR building, soil beneath the reactor high bay concrete pad, water from the sump adjacent to the liquid waste storage tanks, wood from the back-up cooling tower, spent resin that had been removed from the demineralizer system for two

different time periods, and samples of wood used for shielding located over the reactor bioshield (on top of the reactor).

- Performed a general hazardous material inventory for items such as asbestos, mercury (contained in switches, thermometers, light ballasts, etc.) and lead (discrete pieces of lead used for shielding, etc.).
- Prepared a list of major structural components and structural materials, including estimated volumes, linear feet of piping and ventilation ductwork, and an inventory of discrete radioactive sources and hazardous materials.

This Characterization Report documents the survey and sampling performed and presents the results of all surveys and sample analyses. The report includes descriptions of the characterization methods employed, the instruments used, a summary of neutron activation analysis and other information that may be pertinent to decommissioning.

The characterization surveys and sampling were designed using the classification approach described in MARSSIM, which distinguishes between non-impacted areas and impacted areas. Non-impacted areas are defined as areas that have no reasonable potential for residual radioactive contamination from facility operations. The determination of no impact is normally based on the location of historical operations, the area historical use, site discharge locations, and other site physical characteristics. Non-impacted areas usually include office and other routinely occupied areas and open land areas of the site, unless there has been a known or suspected leak or discharge that may have contaminated the rooms or areas. Impacted areas are defined as areas that may contain residual radioactivity from historical activities and reactor operations. The residual radioactivity might be from routine historical operations or from off-normal events such as a spill or leak that contaminated a normally unimpacted area.

The initial area classifications were based on a review of the site history as documented in the ARRR Safety Analysis Report (SAR), updates of the SAR and information gathered during the HSA process. In order to facilitate the characterization surveys, the areas to be surveyed were divided into survey units. Survey units are discrete areas of a specified size and shape for which separate decisions relative to its radiological condition will be made. Survey units were classified based on the history, potential for residual contamination, and physical characteristics. All impacted areas were divided into survey units.

Impacted survey units are further classified as Class 1, 2, or 3, in accordance with MARSSIM. Class 1 areas have or are expected to have concentrations of residual radioactivity for the radionuclides of concern that exceed the criteria for release for unrestricted use. Class 2 areas have a potential for residual radioactivity for the radionuclides of concern, but are not likely to have concentrations of the radionuclides of concern that exceed the criteria for release for unrestricted use.

Class 3 areas are not expected to contain any residual radioactivity or if residual radioactivity is present, it is at levels that are a small fraction of the criteria for release for unrestricted use. Since the criteria for release for unrestricted use has not yet been defined for the decommissioning of the ARRR facility, survey units were conservatively classified. Class 1 areas received the highest degree of survey effort as they have the greatest potential for contamination, followed by Class 2, then Class 3 areas.

4.2 Types of Surveys

The process of survey unit classification was explained in Section 4.1. The number of survey and sample locations chosen for a survey unit is a function of the area classification. Characterization surveys included alpha scans, beta scans, static measurements for total alpha and total beta activity, smears for determining the presence of removable alpha and removable beta activity and gamma exposure rate measurements. In addition, samples representative of various media (water, wood, soil, ion exchange resin) were collected for off-site analysis. Both biased and systematic measurement locations were defined for collection of data and for performing static (direct) surveys.

The criteria for unrestricted release of the ARRR have not been defined, which is normal for this early phase of decommissioning process. Therefore, the survey units were conservatively classified, based in part on the results of the Historical Site Assessment. Some survey units inside the ARRR building were classified as Class 1 or Class 2 based on historical operations that involved handling radioactive material in the areas. Certain areas where radioactive materials were not handled, such as offices, lunchroom/conference room, change areas, etc., were assigned as preliminary Class 3 areas. Soil and paved areas adjacent to the ARRR, auxiliary buildings, and the main building roof were also considered Class 3, since there was no history of any significant routine airborne releases or off-normal airborne or liquid releases of radioactivity from the facility.

Survey units were limited in size to ensure adequate survey coverage, and typically, a minimum number of measurement locations were identified for each survey unit. For scanning surveys, the percentage of the survey unit that was scanned (scan coverage) depended on the classification. The survey units established for characterization, the types of surveys performed and number and type of samples obtained are described in detail in Section 7.0.

4.3 Survey Package Development

A “survey package” was developed for each survey unit included in the characterization surveys. Survey packages contain specific survey instructions, floor plans or drawings, and work sheets for each survey unit. Survey packages include the survey and sampling requirements for the survey unit, plus instructions for performing specific tasks. They are the primary method for controlling and tracking survey and sample results.

The actual measurement locations, and in most cases direct survey and scan measurement results, were marked on a map or drawing of the area surveyed. All survey records, including sample analysis results, will be maintained with the survey packages.

4.4 Data Quality Objectives

The following data quality objectives were initially established and were accomplished for the characterization surveys at the ARRR facility:

- Identified acceptable decision errors and desired levels of confidence in the data collected for the surveys.
- Performed an adequate number of surveys and collected enough samples to ensure sufficient data to identify contaminated (impacted) areas and systems and to verify the radiological conditions of non-impacted areas.
- Collected representative samples of various media for off-site analysis to identify the radionuclides of concern that will be present during decommissioning activities. Sampling was designed to determine the extent and depth of contamination (for volumetric activity), where applicable.
- Collected sufficient data to determine the relative fractions of the radionuclides of interest and identify when fractions may be variable.
- Collected the type of data needed to estimate future radioactive waste volumes for decommissioning.

4.5 Radionuclides of Concern

The amount of solid radioactive waste generated at the reactor during the operational phase was minimal and consisted mostly of waste from experiments, the demineralizer system, used PPE, swipe samples and filters. At one time, waste was stored in steel drums for several years where most of the waste typically decayed to exempt levels, ultimately allowing most of the waste to be disposed of as non-radioactive waste. Radioactive waste that is now stored at the facility generally has more Cs-137 (and other long half-life fission products) than was present in waste from past years, and therefore, most of the waste will not decay to exempt levels for many years.

The list of radionuclides in Code of Federal Regulations, Title 10, Part 61, *Licensing Requirements for Land Disposal of Radioactive Waste* (10 CFR 61) (Reference 9.6), were analyzed for in one sample believed to be representative of the facility as a whole. The analysis of a sample of spent resin removed from the demineralizer in 2010 was performed to ensure that all potential radionuclides that must be accounted for before disposal of radioactive waste were detected. The complete analytical results are included in Appendix A, Sample Analysis Data. A summary of the results for the 2010 resin sample are shown in Table 4.1.

Table 4.1 Summary of Analytical Results for ARRR Spent Resin

Sample- ARRR Spent Resin 2010		
Isotope	**Result**	**Units**
C-14	2.73E+02	pCi/g
Na-22	1.47E+04	pCi/g
Fe-55	1.61E+04	pCi/g
Co-60	2.28E+03	pCi/g
Ni-63	2.39E+03	pCi/g
Sr-90	4.00E+05	pCi/g
Zr-95	1.06E+04	pCi/g
Nb-95	1.19E+04	pCi/g
Cd-109	1.04E+05	pCi/g
Cs-134	4.36E+03	pCi/g
Cs-137	2.49E+05	pCi/g
Ce-144	1.63E+05	pCi/g
Eu-152	7.26E+03	pCi/g
Eu-154	4.37E+04	pCi/g
Eu-155	1.45E+04	pCi/g
Pu-238	3.84E+02	pCi/g
Pu239/40	6.13E+02	pCi/g
Pu-241	1.17E+04	pCi/g
Am-241	3.17E+03	pCi/g
Cm-242	4.95E+02	pCi/g

Fission products are present due to slight leakage from the fuel elements. The most prominent fission products at the Aerotest facility identified by the off-site laboratory in the 2010 resin sample include Sr-90, Cs-137, Ce-144, Pu-241 and Am-241. The most prominent activation products found in the resin sample were Fe-55, Co-60, Ni-63, Zr-95, Nb-95, Cd-109, Eu-154, and Eu-155. Another radionuclide known to be present at the Aerotest facility is tritium in water.

Neutron activation analysis calculations were performed by modeling the ARRR to compute the flux in reactor components and in the bioshield using a representative fuel loading and the operating history for the facility. Trace elements in the materials exposed to the neutron fluxes often drive the induced radioactivity in reactor/bioshield components. The activation analysis resulted in an estimate of activation products that exist in the reactor components, reactor support structures and surrounding building structural material. The activation analysis also resulted in data to be used for waste classification of activated components, in accordance with the 10 CFR 61.

The predominant activation products with half-lives of at least one year identified by neutron activation analysis modeling include H-3, Fe-55, Co-60, Ni-63, Zn-65 and Sb-125.
Since all of the radionuclides of concern and their relative fractions were not known at the time characterization surveys were performed, both alpha and beta-gamma sensitive survey instruments were used. Alpha instrumentation was calibrated using Th-230 and beta instrumentation was calibrated using Tc-99. The use of Tc-99 and Th-230, based on their major radiations, including energies and intensities, was considered conservative for the radionuclides likely to be detected as contamination on surfaces at the ARRR facility.

4.6 Release Criteria

As mentioned earlier, the criteria for release of the ARRR facility will be determined at the time of decommissioning, using the applicable regulatory guidance. Under current NRC regulations, the site will be considered acceptable for unrestricted use if the residual radioactivity that is distinguishable from background radiation results in a Total Effective Dose Equivalent (TEDE) to an average member of the critical group that does not exceed 25 mrem per year, including from groundwater sources of drinking water (Reference 9.3). However, the state of California, which is the regulatory authority for the ARRR decommissioning, typically implements more restrictive release criteria than the NRC and may impose a lower annual dose limit than the NRC (reference 9.4). The release criteria are expressed in the form of Derived Concentration Guideline Levels (DCGLs) for surface and volumetric radioactivity.

Ultimately, survey and sample results from each survey unit (SU) will be tested against the final approved DCGLs. The final DCGLs will approved by the appropriate state of California regulatory agency based on radiation risk estimates. California may approve an administrative release criteria based on a TEDE to an average member of the critical group as low as 1 to 3 mrem per year. A risk assessment may be performed to justify higher limits if necessary and if justifiable.

Gross (total) activity measurements or concentrations are used to determine the net activity for surface and volumetric contamination, after accounting for natural background radiation levels. Each of the results within a SU is then compared to the DCGLs, and the average and standard deviation for the SU is calculated. MARSSIM provides a summary for interpreting the results of final status surveys.
Some of the surveys and sampling performed as part of the characterization effort may be suitable for use as final status (release) surveys. The results of all surveys and sample analyses, including tables showing the concentrations of residual radioactivity measured, are provided in this Characterization Survey Report.

4.7 Survey Instrumentation

The selection and use of survey instrumentation was based on the sensitivity of the instrument to detect the identified radionuclides at the minimum detection requirements. Instruments were provided by the Energy*Solutions* Instrumentation Group. All instruments were calibrated according to Energy*Solutions* procedure "*Calibration and Maintenance of Survey Instruments*" (See Table 6.1). Calibrations and efficiency determinations are traceable to the National Institute for Standards and Technology (NIST). Instruments were response checked at the beginning of each day to ensure proper operation and to confirm the quality control for measurements the previous day. Instrument control logs/charts were maintained and routinely reviewed for trends. The daily response checks included a background measurement and a source check. All instrument records, including dates of use, efficiencies, calibration due dates and source traceability were documented and were archived in accordance with the established Energy*Solutions* procedures. Key instrumentation quality control documentation is included in this report.

The following survey instruments were used for the ARRR characterization surveys:

- Ludlum Model 2350-1 Data Loggers were used in combination with large area gas flow proportional detectors to obtain measurements of total alpha and total beta activity on surfaces, and to perform beta scans on walls, floors and other surfaces. The Data Loggers are portable microprocessor computer based counting instruments designed to operate with a wide variety of detectors.
- Gas flow proportional detectors used were the Ludlum Model 43-68 hand-held detector for scanning walls and static (direct) surveys, and the large area Ludlum Model 239-1F cart-mounted detector for scanning floors.
- A sodium iodide detector (Ludlum Model 44-10) was used for gamma exposure rate measurements. The instrument has a lower scale that indicates exposure rates in micro-Roentgen per hour (μR/hr).
- Smears or wipes for removable alpha and beta activity were counted using a Ludlum Model 2929 with a 43-10-1 scintillation detector.

Table 4- provides a list of the instruments used in the characterization surveys, along with related details.

Table 4-2: ARRR Characterization Survey Instrumentation

Instrument/ Detector	Detector Type	Radiation Detected	Calibration Source	Primary Use
Ludlum 2350-1 Data Logger with a 43-68 detector	Gas-Flow proportional (126 cm^2)	Alpha & Beta	^{230}Th (α) ^{99}Tc (β)	Alpha & Beta; Scans and Fixed Point Surveys
Ludlum 2350-1 Data Logger with a 239-1F detector	Gas-Flow proportional (584 cm^2)	Alpha & Beta	^{230}Th (α) ^{99}Tc (β)	Alpha & Beta Scans of Floors
Ludlum Model 2929 with a 43-10-1 detector	Zinc Scintillator	Alpha & Beta	^{230}Th (α) ^{99}Tc (β)	Wipe Sample Counting - Alpha & Beta
Ludlum Model 44-10 detector	NaI (Tl) Scintillator	Gamma	^{137}Cs (ϒ)	Low Level Gamma Exposure Rates

4.8 Survey Instrument Calibration

All instrumentation used is calibrated on an annual basis using National Institute of Standards and Technology (NIST) traceable sources and calibration equipment. Procedures for calibration, maintenance, source checks, accountability, operation, and quality control of instrumentation are maintained by Energy*Solutions* to ensure that instrumentation functions properly.

Calibration labels showing the instrument identification number, calibration date, and calibration due date were attached to the instruments. All instruments were checked using an appropriate radiation source, each day before use and after certain maintenance activities. These daily source checks verified the calibration status and proper operation of the scaler and the detector. Source check criteria were established prior to the initial use of the instrument for comparison of instrument response over time and control charts were used to track instrument response.

4.9 Survey Minimum Detectable Activity

Minimum Detectable Activity (MDA) is defined as the smallest amount (activity) of radioactive material that will yield a 5% probability of falsely interpreting true activity as background. The MDA for measurements is dependent upon count time, geometry, sample size, detector efficiency, and background count rate. MDA is often provided in units of disintegrations per minute (dpm) or in dpm per 100 square centimeters (dpm/ 100 cm^2).

The MDAs for direct alpha and beta measurements of surface contamination and the MDA for the analyses of removable alpha and beta activity (on wipes) were determined for the instruments used for contamination surveys.

Alpha/beta scans were performed by positioning the detector an inch or less from the floor and scanning at a rate approximately equal to one detector width per second. Monitoring the audible output of the survey meter was done in order to spot areas of potentially elevated activity (“hot spots”). If an area of elevated activity was detected on scans, the area was included in direct (static) surveys. The actual net measured values and their associated errors are reported when measured values were greater than the MDA/MDC. For values lower than the MDA/MDC, the value is reported as less than the MDA/MDC, “< MDA or <MDC.” However, the actual measured values (including negative numbers) were documented and used in statistical evaluations.

4.10 MDA for Direct Measurements

The equation used for calculating the MDA for direct measurements is:

$$MDA = \frac{3 + 3.29\sqrt{R_b * t_s * (1 + \frac{t_s}{t_b})}}{t_s * \varepsilon_i * \varepsilon_s * (\frac{A}{100})}$$

The above equation, derived from basic counting statistics, is discussed in Reference 9.7.

Where: MDA = Minimum Detectable Activity (dpm/100 cm^2)

R_b = Background Count Rate (counts per minute or cpm)

t_b = Background Count Time (min)

t_s = Sample Count Time (min)

$\mathcal{E}_s$ = Surface Efficiency

$\mathcal{E}_i$ = Instrument (Detector) Efficiency (counts/disintegration)

A = Detector Area (cm^2)

The Static MDA calculations and the Static MDA values for the Ludlum 43-37 floor monitor detector and the 43-68 hand-held detector are provided in Attachment B.

4.11 MDC for Beta Scan Measurements

The Minimum Detectable Concentration (MDC) is defined as the *a priori* net activity level that an instrument can be expected to detect 95% of the time. Or stated another way, MDC is the smallest concentration of radioactive material that will yield a 5% probability of falsely interpreting true activity as background. The ability to identify a small area of elevated radioactivity during surface scanning is dependent upon the surveyor's skill in recognizing an increase in the audible or display output of an instrument. In addition to the factors stated above for determining the MDA, the scan rate and the efficiency of the surveyor are incorporated in the calculation of beta Scan MDC.

The beta Scan MDC for instruments used at Aerotest was calculated for scanning measurements in units of dpm/100 cm^2. The equation used for calculating the MDC for beta scans is:

$$MDC\ (dpm/100cm^2) = \frac{d' * \sqrt{b_i} * \frac{60}{i}}{\varepsilon_i * \varepsilon_s * \sqrt{p} * \frac{A}{100}}$$

The above equation was derived by combining equations 6-8 through 6-10 in MARSSIM (Reference 9.2).

Where: MDC = Minimum Detectable Concentration (dpm/100 cm^2)

d' = Decision error taken from Table 6-5 of MARSSIM

i = Observation counting interval (scan speed/detector width)

b_i = Background counts per observation interval (cts)

ε_i = Instrument (Detector) Efficiency (c/d)

ε_s = Surface Efficiency (typically about 50% for betas on hard surfaces)

p = Surveyor Efficiency (typically 50%)

A = Detector Area (cm^2)

The Scan MDC calculations and the beta Scan MDC values for the Ludlum 43-37 floor monitor detector and the 43-68 hand-held detector are provided in Attachment C.

4.12 Probability of Detection for Alpha Scan Measurements

Since the time a contaminated area is under the probe varies and the background count rate of some alpha instruments is less than 1 cpm, it is not practical to determine a fixed MDC for alpha scanning. Instead, it is more useful to determine the probability of detecting an area of alpha contamination at a predetermined concentration for given scan rates. For alpha survey instrumentation such as the ones used at Aerotest, with backgrounds up to a few cpm, a single count provides a surveyor sufficient cause to stop and investigate further. Assuming this is true, the probability of detecting given levels of alpha surface contamination can be calculated by use of Poisson summation statistics. The probability of detecting a "hot spot" of a given activity normally should be at least 70% for alpha scanning surveys.

The calculation used for determining the probability of detecting a given surface activity level is provided in equation 6-12 of MARSSIM (Reference 9.2). Given a known scan rate and a surface contamination level (hot spot) in dpm or dpm/100 cm^2, the probability of detecting a single count while passing over the contaminated area is:

$$P(n \geq 1) = 1 - e^{\frac{-1 * G * \varepsilon_i * \varepsilon_S * W_{\det}}{60 * Rate}}$$

Where:

G = Hot Spot Activity (dpm or dpm/100 cm^2)

ε_i = Instrument (Detector) Efficiency (c/d)

ε_s =S Efficiency (typically 25% or 0.25 for alphas on hard surfaces)

W_{det} = Detector Width in Direction of Scan (cm or inches)

Rate = Scan Rate (cm/s or in/s)

Once an alpha count is recorded, the surveyor will normally stop and perform a static alpha count for a pre-selected time such as one minute.

4.13 Samples of Off-Site Analyses

As part of the characterization of the ARRR facility, samples of various media, including ion exchange resin, water, wood, soil and removable contamination were sent to an off-site laboratory for analysis. The sample analysis results were used to identify and quantify radioactive material not contained within structural material or the reactor, and therefore compliment the estimated activity based on neutron activation analysis that was performed.

The highest activity media readily available for sampling was waste remaining from past operations of the demineralizer system, in the form of spent ion exchange resin. Two samples of ion exchange resin (one sample from resin taken out of service in 2003 and another sample from resin removed from service in 2007) were obtained and sent to the off-site laboratory for analysis. The analyses included gamma spectroscopy, total alpha activity and radionuclides specified in 10 CFR Part 61

(Reference 9.6). The resin sample results should be representative of the radionuclides that could be present and which must be accounted for in low-level radioactive waste sent to a licensed burial site, such as the Energy*Solutions* Clive, Utah site.

A sample of wood cover material above the reactor bioshield (pool) was collected for analysis of neutron activation products or other radionuclides. Fuel handling tools, used in the reactor pool, were surveyed for contamination/radiation levels.
Ten (10) surface and sub-surface soil samples were collected from the soil adjacent to the ARRR building and under the concrete floor slab in the reactor high bay area. A direct-push Geoprobe™ system was used to obtain five surface samples (0-2 feet) and five deep samples of soil (ranging from 14.5 feet to 24 feet below grade). A diamond drill bit was used to penetrate the concrete floor in the high bay area of the reactor. Prior to sampling, the sample areas were surveyed and underground utilities were marked. Soil samples were approximately 500 ml to 1-liter volume. All of the samples were field screened upon removal with a gamma survey meter. After sampling the soil, the sample holes were filled with a bentonite clay material.

All samples were uniquely numbered and tracked using chain of custody records, packing lists for transportation, laboratory verification of receipt, and laboratory tracking during analyses.

A summary of the samples collected for off-site analysis and the type of analysis performed are provided in Table 4-.

Table 4-3: ARRR Samples Collected and Analyzed at Off-Site Laboratory

Sample Media	Number of Samples	Sample Volume	Analysis Type/Radionuclides
Spent resin	1 (removed from service 2007)	2 ml	Analyses and radionuclides specified in 10 CFR Part 61
Spent resin	1 (removed from service 2003)	10 ml	Gamma Spectroscopy (fission & activation products), total alpha
Wood (Bioshield)	1	~ 30 g	Gamma Spectroscopy (fission & activation products),
Wood (Cooling Tower)	1	~ 30 g	Gamma Spectroscopy (fission & activation products, plus Ra-226)
Wipe	2	few grams	Gamma Spectroscopy (fission & activation products)
Wipe	1	few grams	Alpha Spec, total alpha
Soil	7	0.5-1 liter	Gamma Spectroscopy (fission & activation products)
Soil	3	0.5-1 liter	Gamma Spectroscopy (fission & activation products), total alpha
Water (sump)	1	~ 100 ml	Gamma Spectroscopy (fission & activation products), total alpha, H-3 & C-14

4.14 Theoretical Neutron Activation Analysis

Theoretical neutron activation calculations were performed by a subcontractor at the Georgia Institute of Technology (Georgia Tech) to identify and quantify activation products in the reactor pool, pool walls and the bioshield. Activation products are produced by neutron irradiation of materials used in and around the ARRR reactor due to trace quantities of impurities.

The activities that carried out as part of the activation analysis included the following:

4.14.1 Neutron Flux Calculations

- Reactor models were constructed to compute the neutron flux in the reactor components and shielding using a representative fuel loading for the facility.
- The models included sufficient detail of the affected systems to make reasonable engineering estimates of the neutron fluxes.
- The regional neutron flux was computed using a state-of-the-art neutron transport code (MCNP5).
- A 3D flux plot of the reactor and bioshield were generated to locate areas of particular interest for the characterization study.

4.14.2 Activation Calculations

- The activation calculations were performed with the SCALE version of the well-known isotope generation and depletion code ORIGEN to generate estimations of activation products in the reactor and shielding at representative locations.
- The activation calculations were based on the known operating power history of the reactor.
- The trace elements in the materials exposed to the neutron fluxes often drive the induced radioactivity in reactor/bioshield components. The trace elements for standard compositions of the materials in the ARRR structural materials were obtained from descriptions available in ARRR facility documents (steel specifications, concrete compositions, etc.). Based on previous similar work, other trace elements of importance were added to the material to reflect an average reference composition and a maximum composition in terms of activation levels. The maximum compositions should ensure that the actual radioactivity levels are bounded.
- The Neutron Activation Analysis Report was generated, which includes the following sections:

  - An Executive Summary
  - A summary table of activities
  - An Introduction
  - Methods of analysis
  - Assumptions and input data
  - Computed neutron fluxes and computed activities
  - Activities and threshold levels

4.14.3 Neutron Activation Analysis Results

Activation products were calculated for the reactor core and fuel, core shroud and associated structures, reactor pool and support structures, instrument guide tubes, thermal column and neutron radiography assembly, the bioshield, miscellaneous hardware and the soil surrounding the reactor tank. Neutron activation products were calculated for the following specific components and systems:

- Aluminum fuel elements
- Stainless steel fuel element
- Graphite element
- Control rods
- Bottom Grid Plate
- Top Grid Plate
- Core Shroud
- Core Support Shroud
- Neutron Radiography Facility
- Graphite Thermal Column
- Beam Port
- Reactor pool

The Neutron Activation Analysis Report is attached to this report as Appendix B. The predominant activation products with half-lives of at least one year identified by neutron activation analysis modeling include H-3, Mn-54, Fe-55, Co-60, Ni-63, Zn-65 and Sb-125. In addition, significant concentrations of Eu-152 and Eu-154 were calculated to be in concrete and/or the thermal column and Neutron Radiography Assembly.

Since no chemical analyses of ARRR components or materials used in the construction were available, the results in the report can only be considered to be estimates made using representative impurity levels in the components of the reactor. However, for the purposes of estimating decommissioning waste the neutron activation analysis results should provide a conservative starting point.

4.15 MDA/MDC Values for Off-Site Laboratory Analyses

The off-site laboratory used to analyze samples collected during characterization surveys was TestAmerica, Inc., located in Richland, Washington. All samples were analyzed using gamma spectroscopy in order to identify and quantify gamma-emitting radionuclides. Selected samples were also analyzed for total alpha activity, H-3, C-14, isotopic thorium, Pu-238, Pu-239, Pu-240, and/or isotopic uranium. One spent resin sample, which should be representative of activity from reactor operations, was analyzed for the full suite of radionuclides required for waste classification under 10 CFR 61. A 30-day turnaround time (TAT) was requested to ensure analysis results were available in a timely manner, but remained cost-effective.

The laboratory analytical data provided in Appendix A, Sample Analytical Data, shows the radionuclide specific MDAs that were required of the offsite laboratory for the characterization surveys.

5. QUALITY ASSURANCE AND QUALITY CONTROL

The Energy*Solutions* Quality Assurance/Quality Control (QA/QC) Program was utilized to ensure that all quality and regulatory requirements were satisfied. All activities affecting quality were controlled by following written plans and procedures. The QA/QC implementation included the following QC measures and was an integral part of the characterization survey.

5.1 Selection of Personnel

Management and supervisory personnel were experienced in performing characterization surveys of reactor facilities, and in implementing NUREG-1575 (MARSSIM). Project management, along with the Health Physics Technicians supporting the characterization survey, were familiar with the requirements of this characterization survey plan, the instrumentation used and all implementing procedures.

5.2 Training

All project personnel had current Radiation Worker training or refresher training within the last year. In addition to Radiation Worker training, project personnel received site-specific training, as required by Aerotest Operations and Energy*Solutions*. The training included information about site-specific hazards present, emergency response procedures and other facility access requirements. All Energy*Solutions* personnel and subcontractors performed the work in accordance with an approved Job Hazards Analysis (JHA), Energy*Solutions* safety guidelines and Aerotest Operations safety requirements.

5.3 Plans and Procedures

All activities affecting quality were controlled by approved plans and procedures. Table 5.1 provides a list of the Energy*Solutions* approved procedures that were used for the characterization activities.

Table 5-1: EnergySolutions Implementing Procedures

Document Number	Energy*Solutions* Program/Procedure Title
CS-RS-PG-001	Commercial Services Radiation Protection Program
CS-AD-PR-002	Commercial Services Project Records
CS-FO-PR-001	Performance of Radiological Surveys
CS-FO-PR-002	Calibration and Maintenance of Radiological Survey Instruments
CS-FO-PR-003	Soil Surveys; Collection of Water, Sediment, Vegetation and Soil Samples; and Chain-of-Custody
CS-FO-PR-004	QA/QC of Portable Radiological Survey Instruments
CS-FO-PR-005	General Operation of Radiological Survey Instruments
CS-RS-PR-001	Selection and Use of Radiological Protective Clothing
CS-RS-PR-002	Personnel Survey and Decontamination
CS-RS-PR-006	Unconditional Release of Tools, Equipment, and Waste Materials from Projects

5.4 Survey Documentation

Hard copies of all surveys and all sample analysis results are maintained by Energy*Solutions* and electronic copies were generated for documentation. A separate survey package has been established for each survey unit. Radiological measurements and sample results are identified by date, technician, instrument type and serial number, detector type and serial number, location code, etc. All completed survey packages were reviewed by the Project Manager or designee, to ensure the surveys were complete and results were adequate to characterize the survey unit.

5.5 Chain of Custody

Energy*Solutions* procedure CS-FO-PR-003, *Soil Surveys; Collection of Water, Sediment, Vegetation and Soil Samples; and Chain-of-Custody* establishes responsibility for the custody of samples from the time they are collected until results are obtained. All samples sent off site for analysis were accompanied by a chain of custody record to track the location of the sample and ensure that each sample received the appropriate analyses.

6. CHARACTERIZATION SURVEY RESULTS

This section contains the survey measurement locations and results of surveys performed as part of the Aerotest facility characterization. In addition, the locations of samples taken and the results for samples analyzed at the off-site laboratory are summarized and included.

To be consistent with the survey packages, the results are presented by survey unit number. The original completed survey packages containing all of the data and notes from the field, plus signatures, are provided in Attachment A, ARRR Characterization Survey Packages. Table 6-1 provides an overview of the survey units, the types of measurements performed in each one and samples obtained for characterization.

Table 6-1: ARRR Characterization Survey Units

Class 1 Area	Survey Unit #	Description	Surveys Performed	Number of Locations	Sample(s)
Reactor Bldg: Top of Reactor & Inside Bioshield	C-001	Top of bioshield: floor & walls; Inside bioshield: floor & walls	Direct alpha, gamma exposure, wipes	12	Wood from underside of bioshield
Reactor Bldg: Radioactive Material Storage Room	C-002	Floor & walls	Direct alpha, gamma exposure, wipes	10	Wipes (2)
Maintenance Off, Demineralizer Bldg & Heat Exchanger Bldg	C-003	Floors, walls & ceilings	Direct alpha, gamma exposure, wipes	17	Spent resin (2003)
Waste Storage Tank Sump & Concrete Pads	C-004	Sump interior walls & concrete pads	Direct alpha, gamma exposure, wipes	7	Water from sump
Waste Storage Tanks	C-017	Accessible exterior & interior of tanks	Direct surveys & scans -alpha, beta & gamma; wipes	7	N/A
Class 2 Area	**Survey Unit #**	**Description**	**Surveys Performed**	**Number of Locations**	**Sample(s)**
Bldg Addition 1 - Counting Rm. Reactor Bldg - Conf. Room, Control Room, Employee Locker Rooms, GM Off, Machine Shop, Off. Supply Room, South End Radiography	C-005	Floor, walls & ceilings	Direct surveys & scans alpha, beta & gamma; wipes	50	Wipe
Tagging Bldg - Safe	C-006	Floor, walls & ceiling	Direct surveys & scans alpha, beta & gamma; wipes	6	N/A
Reactor Bldg Mezzanine - Prep Lab, Chemistry Lab, Inst Cal Tm, Electronics Lab, Stairway, N-Ray Gauge Off, Sheet Metal Fabrication Area, Storage Area	C-007	Floor, walls & ceilings	Direct surveys & scans alpha, beta & gamma; wipes	31	N/A

Table 6-1: ARRR Characterization Survey Units (Continued)

Class 3 Area	Survey Unit #	Description	Surveys Performed	Number of Locations	Sample(s)
Building Addition 1- Office, Viewing Area, QC Room, Dark Room, Hallway, Safe, Film Storage, Shipping & Receiving, N-Ray Setup Room	C-008	Floors & walls	Direct surveys & scans alpha, beta & gamma; wipes	30	N/A
Reactor Building- Business Off, Accounting Off, Ladies Restroom, Men's Restroom	C-009	Floors & walls	Direct surveys & scans alpha, beta & gamma; wipes	30	N/A
Reactor Bldg- Entry vestibule, Tagging Area, Tagging Back Room, Storage Bldg, Compressor Bldg & Chemical Shed	C-010	Floors & walls	Direct surveys & scans alpha, beta & gamma; wipes	30	N/A
Reactor Bldg, Bldg Addition 1, Tagging Bldg, Storage Bldg	C-011	Exterior walls	Direct surveys- alpha, beta & gamma; wipes	30	N/A
Demineralizer Bldg, Heat Exchanger Bldg, Compressor Bldg, Maintenance Off, Storage Shed	C-012	Exterior walls	Direct surveys- alpha, beta & gamma; wipes	20	Spent resin (2010)
Public Parking Lot	C-013	Paved surfaces	Direct surveys- alpha, beta & gamma	20	N/A
Parking & Paved Areas Inside Fence	C-014	Paved surfaces	Direct surveys for alpha, beta & gamma	20	N/A
Soil Areas	C-015	Soil surface	Direct surveys- alpha, beta & gamma	10	N/A
Cooling Towers	C-016	Exterior and interior walls, floors	Direct surveys- alpha, beta & gamma; wipes	5	Wood Back-up Cooling Tower
Soil Samples	C-018	Shallow & deep soil samples	GeoProbe samples	5	Soil (10)

6.1 Instrument MDA and MDC Values

The instrument static MDC values for alphas and betas, the scan MDC values for beta scans, the probability of detecting an elevated area ("hot spot") for alphas and the MDC for the smear counter were calculated in accordance with Energy*Solutions* procedure CS-FO-PR-001, *Performance of Radiological Surveys.* The calculations for the Ludlum 43-68 hand-held detector used and the Ludlum 43-37 floor monitor detector used are shown in Attachment C, *Field Survey Instrument MDA/MDC Values*. Each instrument is identified by serial number. The alpha and beta MDC values for the Ludlum 2929 with the 43-10-1 detector used for counting smears are documented on the smear counting reports, since daily changes in the background count rate resulted in small changes to the MDC values.

The gamma background rate varied considerably across the site, based on the proximity to systems or areas with elevated radiation levels. The demineralizer system, the reactor bioshield, and the Radioactive Waste Storage Room all had sources that caused elevated gamma radiation levels that interfered with beta readings and in some cases, even contributed to elevated alpha readings due to the crossover effect. Due to the variable background, the gross alpha and gross beta readings for scans and for static readings were recorded instead of net readings above background. In some cases, it was noted that elevated gamma background was causing elevated beta or alpha readings.

Attachment C, *Field Instrument Daily QA/QC System Checks*, provides the results of QC checks for the detectors used for the surveys, including the 43-68 hand-held probe, the 43-37 floor monitor and the 44-10 NaI detector.

6.2 Summary of Results for Survey Units

This section contains a description of each survey unit and a summary of the results of the surveys and sampling performed for each survey unit. The actual Survey Packages are included in Attachment A, Characterization Survey Packages.

All smear results are shown in Attachment D, *Removable Contamination Results*. The smears are organized by date. The removable contamination results include 2929 smear counter alpha and beta efficiencies, surface efficiencies, background values, count times, MDC values, the gross counts, the net counts, the dpm/100 cm^2 and the date. The actual net measured values for smear counting results are included in this report where measured values are greater than the MDA/MDC value. The associated errors are also reported. For smears, the actual measured values (including negative numbers) were documented and were used in statistical evaluations.

Table 6-2 shows a summary of the results of analyses performed by the off-site laboratory on various samples of media, including soil, wood, spent resin, and water. All off-site laboratory analytical results for samples are provided in Appendix A, Sample Analytical Data.

Table 6-2 Summary of ARRR Sample Analysis Results

Sample Type/#	Radionuclide	Results (pCi/g)
Surface Soil/#6	Cd-109	1.31E+00
Subsurface Soil/#7	Cd-109	1.45E+00
Surface Soil/#8	ND	ND
Subsurface Soil/#9	Cd-109	1.87E+00
Surface Soil/#10	ND	ND
Subsurface Soil/#11	Cd-109	2.09E+00
Surface Soil/#12	ND	ND
Subsurface Soil/#13	Cd-109	1.39E+00
Surface Soil/#14	ND	ND
Subsurface Soil/#15	ND	ND
Smear Waste Rm/#16	Co-60	4.43E+01
	Cs-137	4.09E+02
	Eu-152	1.06E+01
	Na-22	3.80E+00
Smear NaI Cave/Tray Rm/#17	Cd-109	1.06E+02
	Co-60	9.77E+02
	Cs-137	4.65E+03
	Eu-152	1.58E+02
	Eu-154	1.06E+02
Sump Pit/#03	H-3	7.01E+03
	C-14	1.35E+00
	Cs-137	1.36E+03
Wood Back-up Cooling Tower/#01	ND	ND
Wood from Reactor Cover/#02	Cs-137	3.99E+01
Spent Resin Removed 2003/#04	Co-60	3.40E+02
	Cs-137	1.56E+05

6.2.1. Survey Unit ARRR C-001

This survey unit consists of the Reactor Building: Inside the Bioshield and the Area on Top of the Bioshield Cover.

Alpha and Beta Direct Surveys
Due to high gamma background levels, only alpha scans and fixed readings were performed on 50% of the floors and 20% of the walls on top of the bioshield cover. Floor scans were performed using the Ludlum 239-1F floor monitor with a 43-37 detector. Fixed alpha readings and smears were obtained for 12 locations on the top (walking surface) of the bioshield cover and the walls surrounding the bioshield cover. The maximum alpha fixed readings using the 43-68 detector were 22 cpm on the top of the bioshield and 28 dpm inside the bioshield.

High background levels caused interference with both beta and alpha readings inside the bioshield, which was accessed by removal of a section of the wood covering. A beta scan was attempted on the section of wood covering after it was lowered to the reactor high bay floor level, but background was still too high to perform an effective beta survey.

Removable Contamination
All alpha removable activity was less than 20 dpm/100 cm^2. Beta removable activity ranged from background to 2,197 dpm/100 cm^2 on the floor inside the bioshield. The inside of the bioshield was decontaminated by Aerotest personnel and the removable beta activity dropped to a maximum of 227 dpm/100 cm^2.

Gamma Surveys
Gamma fixed measurements were made at 14 locations at 1 centimeter and 1 meter from surfaces using the 44-10 NaI detector. Gamma readings ranged from 45 µR/hr to 160µR/hr at 1 meter from the floors and walls.

Samples
A sample of the wood covering from inside the bioshield (Sample #ARRR-02) was obtained for analysis at the off-site lab by gamma spectroscopy.

6.2.2. Survey Unit ARRR C-002

This survey unit consists of the Reactor Building: Radioactive Material Storage Room Floors, Walls and Ceiling.

Alpha and Beta Direct Surveys
Due to high gamma background levels, only alpha scans and fixed readings were performed on 50% of the floors and 20% of the walls. Floor scans were performed using the Ludlum 239-1F floor monitor with a 43-37

detector.

Fixed alpha readings and smears were obtained for 10 locations on the walls, floor and the ceiling. The maximum alpha fixed reading with the 43-68 detector was 238 cpm. However, it appears that beta-gamma crossover into the alpha channel may have caused the elevated reading, since placing a piece of paper over the floor resulted in a slightly higher reading (alpha would be blocked by a sheet of paper). The high background was from various items of waste stored in the Radioactive Material Storage Room.

Removable Contamination
All alpha removable activity was less than 20 dpm/100 cm^2. Beta removable activity ranged from background to 3,388 dpm/100 cm^2 on the floor of the Radwaste Room. The floor was decontaminated by Aerotest personnel and the removable beta activity dropped to a maximum of 441 dpm/100 cm^2.

Gamma Surveys
Gamma scans were performed and gamma fixed measurements were made at 10 locations at 1 centimeter and 1 meter from surfaces using the 44-10 NaI detector. Gamma readings ranged from 213 µR/hr to 600 µR/hr at 1 meter from surfaces. As mentioned above, the general area background rate was elevated due to waste stored in the Radioactive Material Storage Room.

Samples
Two smear samples were taken in the Radioactive Waste Storage Room (Sample #s ARRR-16 and ARRR-18). Both samples were sent to the off-site lab. Gamma spectroscopy analysis was requested for sample ARRR-16. Analysis for total alpha activity and alpha spectroscopy for transuranic radionuclides were requested for sample ARRR-18.

6.2.3. Survey Unit ARRR C-003

This survey unit consists of the Maintenance Office, Demineralizer Building and Heat Exchanger Building: Floors, Walls and Ceiling.

Alpha and Beta Direct Surveys
Due to high gamma background levels, only alpha scans and fixed readings were performed on 20% of the floors and 20% of the walls. Floor scans were performed using the Ludlum 239-1F floor monitor with a 43-37 detector. Fixed alpha readings and smears were obtained for 17 locations on floors, walls and ceilings. The ceilings in the Demineralizer Building were too high for access.

The maximum alpha fixed reading with the 43-68 detector was 9774 cpm. However, it is believed that beta-gamma crossover into the alpha channel may have caused the elevated reading, since placing a piece of paper over the floor resulted in a higher reading (alpha would be blocked by a sheet of paper). The high gamma background was due to the ion exchange resin column located in the Demineralizer Building.

Removable Contamination
All alpha removable activity was less than 20 dpm/100 cm^2. Beta removable activity ranged from background to 15,850 dpm/100 cm^2 on the floor. The floor was decontaminated by Aerotest personnel and the removable beta activity dropped to a maximum of 386 dpm/100 cm^2.

Gamma Surveys
Gamma scans were performed and gamma fixed measurements were made at 17 locations at 1 centimeter and 1 meter from surfaces using the 44-10 NaI detector. Gamma readings ranged from about 500 to 3,300 µR/hr at 1 meter from surfaces in the Maintenance Office, from 100 to 400 µR/hr at 1 meter from surfaces in the Heat Exchanger Building, and from 2,500 to 8,000 µR/hr general area in the Demineralizer Building.

As mentioned above, elevated gamma background was present in all of the areas due to the ion exchange resin column located in the Demineralizer Building.

Samples
A sample was taken of the spent resin in a used demineralizer column stored in the Demineralizer Building. The resin sample was taken out of service in December of 2010. The sample (#ARRR-05) was sent to the off-site lab for gamma spectroscopy and 10 CFR Part 61 radionuclide analyses.

6.2.4. Survey Unit ARRR C-004

This survey unit consists of the Waste Storage Tank Area: Sump Walls and Cover, Concrete Pad and Soil in the Area.

Alpha and Beta Direct Surveys
Beta and alpha scans and fixed readings were performed on 50% of the concrete pads, using a Ludlum 43-68 detector. Surveys of the interior walls of the sump pit were planned, but due to interfering pipes, the alpha/beta detector could not be lowered into the sump pit.

Fixed beta and alpha readings and smears were obtained at 7 locations on the concrete pads and surrounding gravel surfaces. Using a 43-68 detector the maximum beta fixed reading was 818 cpm and the maximum alpha fixed reading was 22 cpm. There were slightly gamma background levels at several locations, most likely due to proximity to the Radioactive Material

Storage Room. The elevated gamma background probably contributed to the beta readings.

Removable Contamination
All alpha removable activity was less than 20 dpm/100 cm^2 and all beta removable activity was less than 200 dpm/100 cm^2 for smears taken on the concrete pads and inside the sump pit.

Gamma Surveys
Gamma scans were performed and gamma fixed measurements were made at 7 locations at 1 centimeter and 1 meter from surfaces using the 44-10 NaI detector. Gamma readings ranged from 20 to 110 µR/hr at 1 meter from surfaces. There was high gamma background in the area, most likely due to proximity to the Radioactive Waste Storage Room inside the south end of the ARRR Building.

Samples
A water sample was taken of water in the sump pit (Sample #ARRR-03). The sample was sent to the off-site lab for analysis by gamma spectroscopy, total alpha activity, H-3 and C-14.

6.2.5. Survey Unit ARRR C-005

This survey unit consists of the Building Addition 1: Counting Room and Reactor Building: Conf Room, Control Room, Employee Locker Room, GM Office, Machine Shop, Office Supply Room and the South End Radiography Area.

Alpha and Beta Direct Surveys
Beta and alpha scans and fixed readings were performed on 50% of the floors and 20% of the walls. Floor scans were performed using the Ludlum 239-1F floor monitor with a 43-37 detector. Fixed beta and alpha readings and smears were obtained using a 43-68 detector for 50 locations on walls, floors and ceilings, spread proportionately among the eight areas. The ceilings were too high for access in several areas where no ceiling surveys were performed.

The maximum beta fixed reading found in all areas was 68,044 cpm and the maximum alpha fixed reading was 40 cpm at Location # 27 in the South End Radiography Room. The elevated readings at Location # 27 and at several other locations were likely due to high background in the area caused by radioactive material stored in the Radioactive Material Storage Room.

Removable Contamination
All alpha removable activity was less than 20 dpm/100 cm^2. Beta removable activity ranged from background to 1,559 dpm/100 cm^2 on the shielded "cave" used for counting samples with the gamma spectroscopy system. The "cave" was decontaminated by Aerotest personnel and a subsequent smear showed removable beta was less than 200 dpm/100 cm^2.

Gamma Surveys
Gamma scans were performed and gamma fixed measurements were made at 10 locations, at 1 centimeter and 1 meter from surfaces, using the 44-10 NaI detector. Gamma readings ranged from 22 to 310 µR/hr at 1 meter from surfaces. There was elevated gamma background in a several areas, including the area with the highest gamma readings, most likely due to proximity to the Radioactive Waste Storage Room

Samples
The original smear taken from the sample "cave" in the Counting Room (Sample #ARRR-16), was sent to the off-site lab for analysis by gamma spectroscopy.

6.2.6. Survey Unit ARRR C-006

This survey unit consists of the Tagging Building Safe.

Alpha and Beta Direct Surveys
Beta and alpha scans and fixed readings were performed on 20% of the inside surfaces on the safe, all using a Ludlum 43-68 hand-held detector. Fixed beta and alpha readings and smears were obtained for 6 locations. Using the 43-68 detector, the maximum beta fixed reading was 1,050 cpm and the maximum alpha fixed reading was 39 cpm. The readings were most likely elevated due to close proximity to the uranium fuel in the safe, which created elevated gamma levels.

Removable Contamination
All alpha removable activity was less than 20 dpm/100 cm^2. Beta removable activity ranged from background to 208 dpm/100 cm^2.

Gamma Surveys
Gamma fixed measurements were made at 6 locations, at 1 centimeter and 1 meter from surfaces, using the 44-10 NaI detector. Gamma readings ranged from 125 to 2,000 µR/hr general area. The elevated readings were due to the fuel stored in the safe.

Samples
No samples were taken in this survey unit.

6.2.7. Survey Unit ARRR C-007

This survey unit consists of the Reactor Building Mezzanine: Prep Lab, Chemistry Lab, Inst Cal Room, Electronics Lab, Stairway, N-Ray Gauge Office, Sheet Metal Fabrication Area, and Storage Area.

Alpha and Beta Direct Surveys
Due to high gamma background levels, only alpha scans and fixed readings were performed on 50% of the floors and 20% of the walls. The elevated gamma background was most likely from the Demineralizer Building or from radioactive material stored on the mezzanine. Floor scans were performed using the Ludlum 239-1F floor monitor with a 43-37 detector. Fixed alpha readings and smears were obtained for 31 locations on walls, floors and ceilings. The ceilings were too high to access in most areas.

Using the 43-68 detector the maximum beta fixed reading was 1,953 cpm and the maximum alpha fixed reading with was 21 cpm, both in the Mezzanine Hallway.

Removable Contamination
All alpha removable activity on floors, walls and ceilings was less than 20 dpm/100 cm^2. Beta removable activity on floors, walls and ceilings ranged from background to 203 dpm/100 cm^2 on the floor. Smears were also taken on several fuel handling tools and an N-Ray tube cutter. The removable alpha was less than 20 dpm/100 cm^2 and the beta removable activity was up to 2,341 dpm/100 cm^2 on fuel handling tools and 384 dpm/100 cm^2 on the N-Ray tube cutter.

Gamma Surveys
Gamma fixed measurements were made at 31 locations at 1 centimeter and 1 meter from surfaces or sources using the 44-10 NaI detector. Gamma readings ranged from near background up to 2,700 µR/hr at 1 meter from the floor in the Prep Lab. The Prep Lab is the closest area in the survey unit to the reactor core and bioshield, which probably contributed to the elevated readings.

Gamma readings were also taken on several radioactive items stored inside shields on the Mezzanine. A tub with metal bearings read 5,000 µR/hr at 1 inch, a shutter mechanism read 5,600 µR/hr at 1 inch and an N-Gauge tube read 7,500 µR/hr at 1 inch, all unshielded. The shutter mechanism inside the shield read 1,400 µR/hr at 1 meter and the shielded N-Gauge tube read 70 µR/hr at 1 meter.

Samples
No samples were taken in this survey unit.

6.2.8. Survey Unit ARRR C-008

This survey unit consists of the Building Addition 1: Office, Viewing Area, QC Room, Dark Room, Hallway, Safe, Film Storage, Shipping & Receiving and N-Ray Set-up (Rooms 1-8).

Alpha and Beta Direct Surveys
Beta and alpha scans and fixed readings were performed on 20% of the floors and 20% of the walls. Floor scans were performed using the Ludlum 239-1F floor monitor with a 43-37 detector. Fixed alpha readings and smears were obtained for 30 locations on floors and walls. Using the 43-68 detector the maximum beta fixed reading was 2,600 cpm and the maximum alpha fixed reading was 12 cpm. Several rooms had elevated gamma background, which affected the beta survey readings.

Removable Contamination
All alpha removable activity was less than 20 dpm/100 cm^2 and all beta removable activity was less than 200 dpm/100 cm^2.

Gamma Surveys
Gamma fixed measurements were made at 30 locations at 1 centimeter and 1 meter from surfaces using the 44-10 NaI detector. Gamma readings ranged from background up to 65µR/hr at 1 meter from surfaces. There was elevated gamma background in several areas, most likely due to the proximity to the reactor.

Samples
No samples were taken in this survey unit.

6.2.9. Survey Unit ARRR C-009

This survey unit consists of the Reactor Building: Men's Room, Ladies Room, Business Office and Accounting Office (Rooms 14, 15, 18 & 19).

Alpha and Beta Direct Surveys
Beta and alpha scans and fixed readings were performed on 20% of the floors and 20% of the walls. Floor scans were performed using the Ludlum 239-1F floor monitor with a 43-37 detector. Fixed beta and fixed alpha readings and smears were obtained for 30 locations.

Using the 43-68 detector, the maximum beta fixed reading was 36,000 cpm and the maximum alpha fixed reading was 7 cpm in the Business Office. The elevated beta readings in the Business Office (and several other rooms) are most likely due to the nearby Radioactive Material Storage Room.

Removable Contamination
All alpha removable activity was less than 20 dpm/100 cm^2 and all beta removable activity was less than 200 dpm/100 cm^2.

Gamma Surveys
Gamma fixed measurements were made at 30 locations at 1 centimeter and 1 meter from surfaces using the 44-10 NaI detector. Gamma readings ranged from about 30 up to 2,000 μR/hr at 1 meter from a wall in the Business Office. The elevated readings in the Business Office and several other rooms are likely due to the nearby Radioactive Material Storage Room.

Samples
No samples were taken in this survey unit.

6.2.10. Survey Unit ARRR C-010

This survey unit consists of the Reactor Building: Entry Vestibule, Tagging Area, Tagging Back Room, Storage Building, Compressor Building and Chemical Shed.

Alpha and Beta Direct Surveys
Beta and alpha scans and fixed readings were performed on 20% of the floors and 20% of the walls. Floor scans were performed using the Ludlum 239-1F floor monitor with a 43-37 detector. Fixed beta and alpha readings and smears were obtained for 30 locations.

Using a 43-68 detector the maximum beta fixed reading was 32,000 cpm and the maximum alpha fixed reading was 110 cpm in the Tagging Area Entry Vestibule. There were elevated gamma background levels in the Tagging Area Entry Vestibule and in the Tagging Area, most likely due to proximity to the Radioactive Material Storage Room. The elevated gamma background affected the beta and alpha readings.

Removable Contamination
All alpha removable activity was less than 20 dpm/100 cm^2 and all beta removable activity was less than 200 dpm/100 cm^2.

Gamma Surveys
Gamma fixed measurements were made at 30 locations at 1 centimeter and 1 meter from surfaces using the 44-10 NaI detector. Gamma readings ranged from up to 800 μR/hr at 1 meter from the walls in the Tagging Area. There were elevated gamma background levels in the Tagging Area and in the Tagging Area Entry Vestibule, most likely due to proximity to the Radioactive Material Storage Room.

Samples
No samples were taken in this survey unit.

6.2.11. Survey Unit ARRR C-011

This survey unit consists of the -Exterior Walls of the Reactor Building, Building Addition 1, Tagging Building, and Storage Building.

Alpha and Beta Direct Surveys
Scan surveys were not performed on exterior walls of the buildings. Fixed alpha and beta readings and smears were obtained for 20 locations on the walls. Using the 43-68 detector, the maximum beta fixed reading was 4,614 cpm and the maximum alpha fixed reading was 30 cpm on the south wall of the Reactor Building. The Reactor Building south wall (and other walls on each building) are in elevated gamma background areas due to proximity to the Radioactive Material Storage Room. In addition, the Reactor Building east wall has elevated gamma background from either the Reactor or the Demineralizer Building.

Removable Contamination
All alpha removable activity was less than 20 dpm/100 cm^2 and all beta removable activity was less than 200 dpm/100 cm^2.

Gamma Surveys
Gamma fixed measurements were made at 20 locations at 1 centimeter and 1 meter from wall surfaces using the 44-10 NaI detector. Gamma readings ranged from background to 850 μR/hr (at 1 meter from the east wall of the Reactor Building). As mentioned above, several exterior walls on each building are in elevated gamma background areas due to their proximity to the Radioactive Material Storage Room, the reactor core or the Demineralizer Building.

Samples
No samples were taken in this survey unit.

6.2.12. Survey Unit ARRR C-012

This survey unit consists of the Exterior Walls of the Demineralizer Building, Heat Exchanger Building, Compressor Building, Maintenance Office, and Chemical Storage Shed.

Alpha and Beta Direct Surveys
Scan surveys were not performed on exterior walls of the buildings. Fixed alpha readings and smears were obtained for 20 locations on the exterior

walls of the five buildings. Only 10 beta fixed measurements were made, due to high background gamma levels in several areas. Using the 43-68 detector, the maximum beta fixed reading was 446 cpm on the west wall of the Compressor Building and the maximum alpha fixed reading was 1,419 cpm on the east wall of the Demineralizer Building. The beta and alpha readings were likely elevated due to high gamma background in the area of the surveys.

The walls on all of the buildings except the Chemical Storage Shed have elevated gamma background due to proximity to the ion exchange material inside the Demineralizer Building. In addition, the Reactor core may contribute to the elevated gamma readings on the west side of the Reactor Building.

Removable Contamination
All alpha removable activity was less than 20 dpm/100 cm^2. Beta removable activity ranged from background to 203 dpm/100 cm^2.

Gamma Surveys
Gamma fixed measurements were made at 20 locations at 1 centimeter and 1 meter from wall surfaces using the 44-10 NaI detector. Gamma readings ranged from near background on the Chemical Storage Shed walls to 4,400 µR/hr at 1 meter from the east wall of the Demineralizer Building. As mentioned above, walls on all of the buildings except the Chemical Storage Shed had elevated gamma background due to proximity to the ion exchange material inside the Demineralizer Building.

Samples
No samples were obtained of the building exterior walls for this survey unit.

6.2.13. Survey Unit ARRR C-013

This survey unit consists of the Public Parking Lot.

Alpha and Beta Direct Surveys
Beta and alpha scans and fixed readings were performed on 10% of the paved area in the parking lot. Pavement scans were performed using the Ludlum 239-1F floor monitor with a 43-37 detector. Fixed beta and alpha readings were made at 10 locations and smears were obtained at 20 locations. Using the 43-68 detector, the maximum beta fixed reading was 325 cpm and the maximum alpha fixed reading was 8 cpm.

Removable Contamination
All alpha removable activity was less than 20 dpm/100 cm^{2}and all beta removable activity was less than 200 dpm/100 cm^2 on the pavement.

Gamma Surveys
Gamma fixed measurements were made at 20 locations at 1 centimeter and 1 meter from the paved surfaces using the 44-10 NaI detector. Gamma readings ranged from background to 8.7 µR/hr at 1 meter from paved surfaces.

Samples
No samples were taken in this survey unit.

6.2.14. Survey Unit ARRR C-014

This survey unit consists of the Parking Lot and Paved Areas inside the Fence.

Alpha and Beta Direct Surveys
Beta and alpha scans were performed on 10% of the paved surfaces. Scans were performed using the Ludlum 239-1F floor monitor with a 43-37 detector. Fixed alpha and beta measurements and smears were obtained at 10 locations. Using the 43-68 detector, the maximum beta fixed reading was 786 cpm and the maximum alpha fixed reading was 14 cpm. Some areas had elevated gamma background due to proximity to the Reactor, the Demineralizer Building, or the Radioactive Material Storage Room.

Removable Contamination
All alpha removable activity was less than 20 dpm/100 cm^2 and all beta removable activity was less than 200 dpm/100 cm^2.

Gamma Surveys
Gamma fixed measurements were made at 20 locations at 1 centimeter and 1 meter from surfaces or sources using the 44-10 NaI detector. Gamma readings ranged from background to 97 µR/hr at 1 meter from the pavement. As mentioned above, some areas had elevated gamma background due to proximity to the Reactor, the Demineralizer Building, or the Radioactive Material Storage Room.

Samples
No samples were taken in this survey unit.

6.2.15. Survey Unit ARRR C-015

This survey unit consists of the Soil Areas inside the Fence.

Alpha and Beta Direct Surveys
Beta and alpha scans were performed on 20% of soil surfaces, which are located on the south and west sides of the ARRR facility. Fixed alpha

readings were obtained for 10 locations. Using the 43-68 detector, the maximum beta fixed reading was 589 cpm and the maximum alpha fixed reading was 15 cpm. It appeared that some areas had elevated gamma background due to proximity to the reactor or the Radioactive Material Storage Room.

Removable Contamination
No smears were performed on soil.

Gamma Surveys
Gamma fixed measurements were made at 10 locations at 1 centimeter and 1 meter from the soil surface using the 44-10 NaI detector. Gamma readings ranged from background to 42 µR/hr at 1 meter from soil surfaces. As mentioned above, it appeared that some areas had elevated gamma background due to proximity to the reactor or the Radioactive Material Storage Room.

Samples
No samples were taken in this survey unit.

6.2.16. Survey Unit ARRR C-016

This survey unit consists of the Main and Back-up Cooling Towers.

Alpha and Beta Direct Surveys
Beta and alpha scans were performed on 20% of the floors and 20% of the walls. Fixed beta and alpha readings and smears were obtained at 5 locations. Water in the bottom of the Back-up cooling tower prevented smears or direct surveys of the floor.

Using the 43-68 detector, the maximum beta fixed reading was 4,700 cpm and the maximum alpha fixed reading was 41 cpm, both on the Main Cooling Tower. The gamma background was elevated due to the proximity of both cooling towers to the Demineralizer Building.

Removable Contamination
All alpha removable activity was less than 20 dpm/100 cm^2 and all beta removable activity was less than 200 dpm/100 cm^2.

Gamma Surveys
Gamma scans were performed on 20% of surfaces and gamma fixed measurements were made at 5 locations at 1 centimeter and 1 meter from surfaces or sources using the 44-10 NaI detector. Gamma readings ranged from 26 to 210 µR/hr at 1 meter from surfaces. The highest gamma reading was at the Main Cooling Tower. As mentioned above, the gamma

background was elevated due to the proximity of both cooling towers to the Demineralizer Building.

Samples
A sample of the redwood slats in the water cascade was taken from the Back-up Cooling Tower. The sample (ARRR-01) was sent to the off-site lab for analysis by gamma spectroscopy. The redwood slats could have possibly concentrated naturally occurring radioactive material (NORM), such as Ra-226. This is not uncommon in older cooling towers. If NORM is present in the wood, the activity levels will need to be compared to acceptance criteria for the landfill (or other site) planned for disposal of the wood.

6.2.17. Survey Unit ARRR C-017

This survey unit consists of the Waste Storage Tanks.

Alpha and Beta Direct Surveys
Beta and alpha scans were performed on 20% of the accessible interior and exterior surfaces of the waste tanks. Fixed beta and alpha readings and smears were obtained at only 7 locations instead of the planned 10 locations, due to high gamma background in some areas from the Radioactive Material Storage Room located in the south end of the ARRR facility.

Using the 43-68 detector, the maximum beta fixed reading was 1,569 cpm and the maximum alpha fixed reading was 21 cpm.

Removable Contamination
All alpha removable activity was less than 20 dpm/100 cm^2 and all beta removable activity was less than 200 dpm/100 cm^2.

Gamma Surveys
Gamma scans were performed on 20% of storage tank exterior surfaces and gamma fixed measurements were made at 7 locations at 1 centimeter and 1 meter from the tank surface using the 44-10 NaI detector. Gamma readings ranged from 48 to 100 µR/hr at 1 meter from surfaces. As mentioned above, the gamma background was elevated due to the proximity to the Radioactive Material Storage Room.

Samples
No samples were taken in this survey unit.

6.2.18. Survey Unit ARRR C-018

This survey unit consists of Soil Samples. There were no alpha or beta scans and no removable contamination smears for this survey unit.

Samples
Ten (10) soil samples were obtained from soil areas surrounding the main ARRR facility, the public parking lot and beneath the reactor high bay area concrete floor. A direct-push Geoprobe™ system was used to obtain five (5) surface samples (0-2 feet) and five (5) deep samples of soil (ranging from 14.5 feet to 24 feet below grade). A diamond drill bit was used to penetrate the concrete floor in the high bay area of the reactor. The exact depth of each sample is provided as part of the survey package.

The soil samples (sample #s ARRR-06 through ARRR-15) were sent to the off-site lab for analysis by gamma spectroscopy. In addition to gamma spectroscopy, total alpha activity was requested for samples ARRR-11 and ARRR-12, and total alpha activity and alpha spectroscopy analysis for transuranic radionuclides were requested for sample ARRR-15.

Gamma readings were taken on contact with each sample extracted, using the 44-10 NaI detector. All gamma readings were less than or equal to background.

7. REFERENCES

1. ASTM E 1982-27, *Standard Guide for Preparing Characterization Survey Plans for Decommissioning Nuclear Facilities*, American Society for Testing and Material, 1997.

2. NUREG-1575 Rev. 1, *Multi-Agency Radiation Survey and Site Investigation Manual* (MARSSIM), U. S. Nuclear Regulatory Commission, 2000.

3. 10 CFR, Part 20, Subpart E, *Radiological Criteria for License Termination,* U. S. Nuclear Regulatory Commission, 2007.

4. Title 17, California Code of Regulations, Division 1, Chapter 5, Subchapter 4, Section 30256, *Standards for Protection Against Radiation*, State of California, 2010.

5. 10 CFR Part 50, *Domestic Licensing of Production and Utilization Facilities*, U. S. Nuclear Regulatory Commission, 2010.

6. Code of Federal Regulations, Title 10, Part 61, *Licensing Requirements for Land Disposal of Radioactive Waste*, U. S. Nuclear Regulatory Commission, 2008.

7. *Minimum Detectible Activity When Background is Counted Longer than the Sample,* Daniel J. Strom and Paul S. Stansbury, 1992.

Attachment A

ARRR Characterization Survey Packages

Attachment B

Field Survey Instrument MDA/MDC Values

Attachment C

Field Instrument Daily QA/QC System Checks

Attachment D

Removable Contamination Results

Appendix A

Sample Analytical Data

Appendix B

Neutron Activation Analysis Report