50-160

Georgia Institute of Technology

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February 26, 2001

Marvin M. Mendonca
USNRC
One White Flint North
Mail Stop 012-D3
11555 Rockville Pike
Rockville, MD 20852-2738

Dear Mr. Mendonca:

In accordance with your email request of November 30, 2000 find enclosed:

1) NRC approved Decommissioning Plan, May 1998(Book 1)
2) NRC approved Decommissioning Plan, June 1998
3) 10 CFR 50.59 Review Program for Changes/Tests
4) Site Characterization Reports, Parts A/B of Book 2
5) Site Characterization Reports, Parts A/B of Book 3
6) Final Survey Plan- Revision 3, January 2001

Duplicate copies of the above have also been sent to Craig Bassett, NRC and Timothy Vitkus, ORISE.

Thanks again.

Sincerely,

Nolan E. Hertel

Nolan E. Hertel
Director, NNRC

NEH/ars

Enclosures

A030

GEORGIA INSTITUTE OF TECHNOLOGY
RESEARCH REACTOR
DECOMMISSIONING PROJECT
RADIOLOGICAL CHARACTERIZATION REPORT

NES DOCUMENT NO. 82A9087

May, 1998

Prepared by:
NES, Inc.
44 Shelter Rock Road
Danbury, CT 06810

Book 1 of 3

GEORGIA INSTITUTE OF TECHNOLOGY RESEARCH REACTOR DECOMMISSIONING PLAN

May, 1998

Prepared by
NES, INC.
DANBURY, CT

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1.0 INTRODUCTION

The characterization of the Georgia Institute of Technology- Neely Nuclear Research Center (NNRC) was performed to collect sufficient information:

- to determine the nature and extent of radiological contamination;
- to plan the measures required to protect workers, the public and the environment during decommissioning;
- to estimate the amounts of waste that will be generated during the decontamination and decommissioning (D&D) activities; and
- to estimate the amount of material that might be salvaged or recycled.

The radiological and hazardous material characteristics of the NNRC must be known prior to preparing the Decommissioning Plan. This Characterization Report identifies the potential hazards present during the decommissioning operations. The extent of radiological and hazardous material contamination will be the determining factor for establishing specific safety requirements and decontamination techniques that will be used during decommissioning operations.

This document reports the results of the characterization activities that were performed by NES, Inc. (NES) according to the "Georgia Institute of Technology Research Reactor Decommissioning Project- Radiological Characterization Plan" (Reference 1) (herein referred to as the Characterization Plan). The activities described in the Characterization Plan were designed to collect information necessary to develop the Decommissioning Plan and prepare the Decommissioning Cost Estimate for the unrestricted release of the NNRC. The characterization survey was not intended to release the NNRC for unrestricted use.

2.0 FACILITY DESCRIPTION

The Georgia Tech Research Reactor (GTRR) began operations in 1964 as a research facility where engineering students would train to operate commercial nuclear power reactors and researchers would explore the frontiers of biology and material sciences. It is located on the 330-acre campus of the Georgia Institute of Technology.

The GTRR is a part of the Georgia Institute of Technology-Neely Nuclear Research Center (NNRC). The NNRC is comprised of the GTRR, the hot cell, support laboratories and offices. The reactor containment building is located approximately 200 feet north and northeast of the Electronics Research Building. The Physics building is approximately 700 feet south of the reactor building. The Baker building is about 200 feet to the west and the Office of Facilities is about 50 feet east and north east. The east boundary of the NNRC is Atlantic Drive, a street that carries moderate local traffic. The main laboratory and parking lot entrances are from Atlantic Drive. The land immediately adjacent to the reactor and the laboratory is surrounded by a personnel fence. There are two openings in this fence: a personnel gate and a truck gate at the rear of the laboratory, which is unlocked only for deliveries.

The point of closest public approach permitted by the fence is 45 feet from the containment building. The land slopes downward to the north and west around the laboratory building so that the elevation is at first floor height on the south and east, and ground floor level to the north and west.

Laboratories, office space and a variety of support facilities are housed in the three-story building adjoining the containment building. This building is approximately 90 feet by 130 feet and contains 24,000 square feet of floor space. In the southeast corner of the building is a high bay area which contains the hot cell, Radiochemistry Laboratory, Decontamination Room and storage facility. This high bay area is normally closed off from the remainder of the building, with routine access through the change room entrance. The handling of materials with potential contamination is restricted to this area. Only sealed sources or very small quantities of radioactive material are handled in the low-level laboratories. The major focus of the characterization effort was the reactor containment building.

More than 13,000 students are enrolled at Georgia Tech. The institute employs approximately 5,000 faculty and staff. The campus lies in a residential and commercial area just north of the center of downtown Atlanta. The city of Atlanta, according to the 1990 census, has a population of 394,017 and covers 323.4 square kilometers. Today, the metropolitan area is in excess of 3.1 million people. The eastern boundary of the Georgia Tech campus coincides with the combined leg of I-75 and I-85, the major traffic artery which runs north and south through Atlanta. The location of the Georgia Tech campus with respect to major traffic arteries is shown in Figure 2.1.

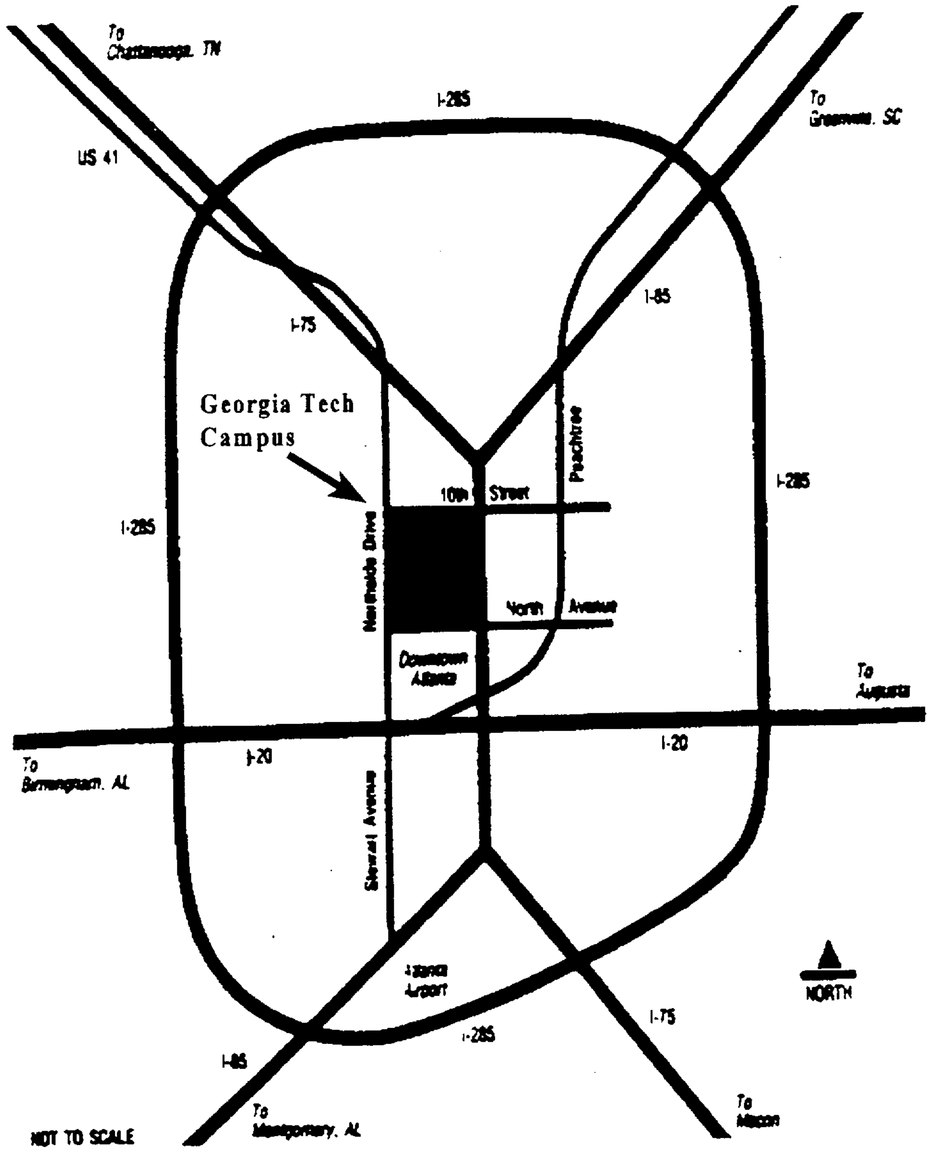


Figure 2.1 Map Indicating the Location of Georgia Tech with Respect to Major Traffic Arteries

2.1 OPERATING HISTORY

The GTRR was used for neutron activation analysis, operator training and material testing. The reactor operated almost daily for an average energy output of 1,297 MW-hrs. per year (Reference 2). The reactor is equipped with numerous horizontal and vertical experimental facilities to be used for the extraction of beams of fast and slow neutrons and for the performance of irradiations within the facilities.

The GTRR began operation in 1964 and continued operation until 1996 at power levels up to 5 MW. During its operating life, the reactor generated 40,204 MW-hrs. of thermal power (Reference 2). The fuel was removed from the NNRC complex on February 15, 1996. The decision to decommission the reactor was made on July 1, 1997.

3.0 SCHEDULE

NES, Inc. began mobilization on October 6, 1997. Health and safety training was performed on October 6, 1997. The characterization activities, including the material release survey and floor surface preparation, commenced on October 8, 1997 and were completed by October 22, 1997. Demobilization activities took place on October 23, 1997. Figure 3.1 summarizes the schedule for the on-site characterization activities.

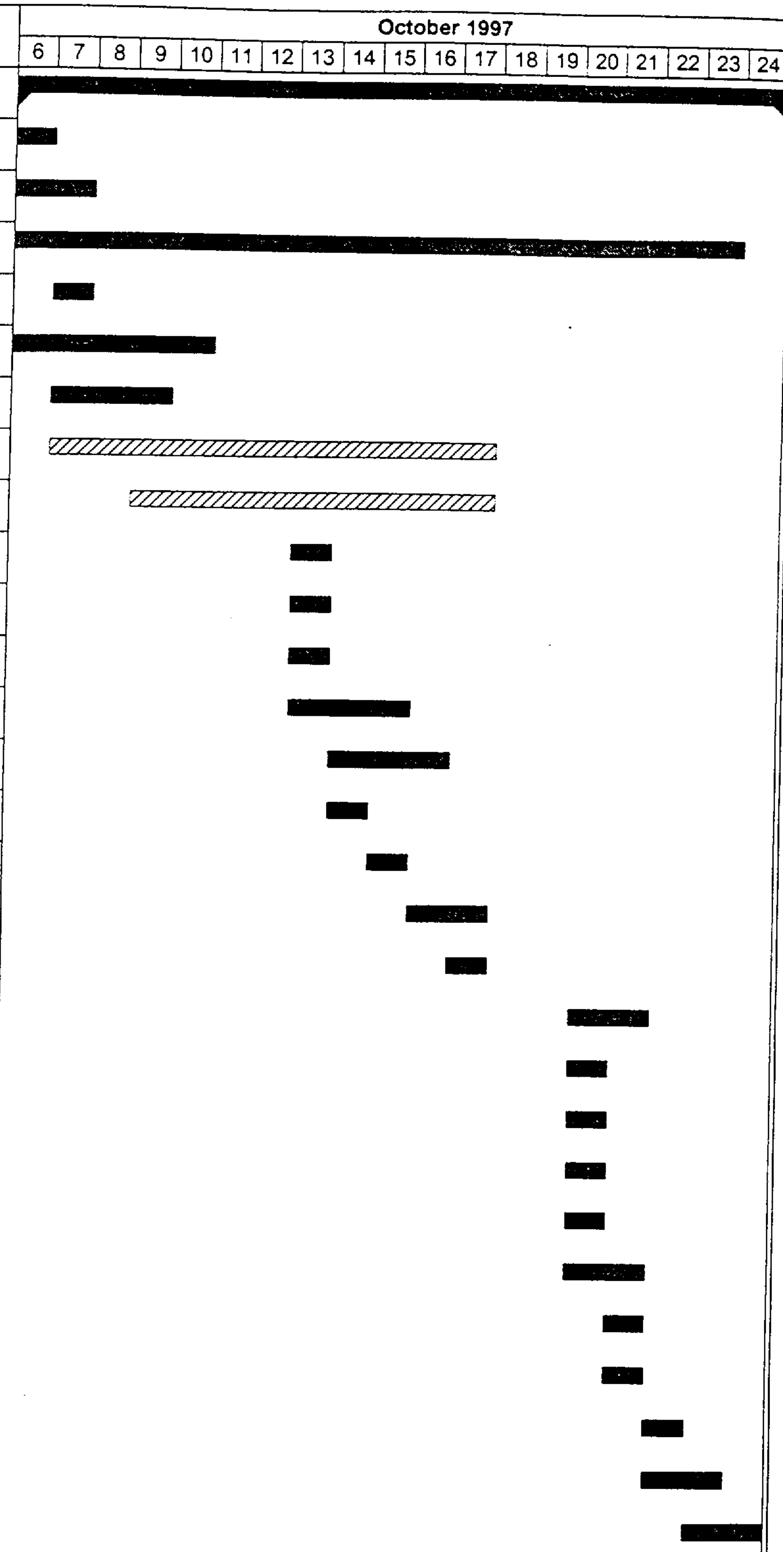
Before commencement of the characterization, personnel familiarized themselves with the project area by conducting a walk-through of the entire facility. All NES project personnel were trained in accordance with Georgia Tech and NES Training Procedures and received a general safety briefing.

Throughout the characterization activities and after project completion, a photographic survey was performed for historical purposes.

NES, Inc.

Georgia Tech NNRC Characterization Schedule

ID	Task Name
1	NNRC Characterization
2	Arrival to Site
3	Training
4	Daily Briefings
5	Unpacked Equipment/ Instrument Calibrations
6	Performed Background Surveys
7	Performed Outdoor Surveys
8	Performed Indoor Surveys
9	Concrete Sample Collection
10	Surveyed Material in Front of Plug Storage Area
11	Insertion and Removal (24 hrs) of TLD Trees
12	Tank Surveys and Sample Collection
13	Lead Paint Inspections/ Asbestos Sample Collection
14	Commenced Process Equipment Room Survey
15	Commenced Vertical Beam Ports Survey
16	Plug Storage Area Survey
17	Commenced Survey of Areas Outside of Containment Area
18	Commenced Horizontal Beam Ports Survey
19	Finished Survey of Areas Outside of Containment Area
20	Tank Gamma Scans
21	Continued Horizontal Beam Ports Survey
22	Cutting of Horizontal Beam Port No. 12
23	Resurveyed Material in Front of Plug Storage Area
24	Miscellaneous Areas Surveys
25	Finished Vertical Beam Ports Survey
26	Storage Holes Survey
27	Finished Horizontal Beam Ports Survey
28	Clean-up Surveys
29	Demobilization



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Date: Wed 4/1/98

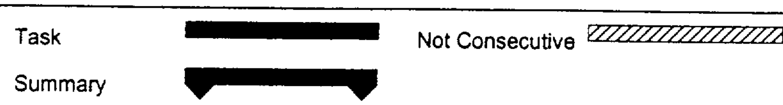


Figure 3.1

4.0 RADIOLOGICAL CONTROLS DURING CHARACTERIZATION

4.1 CONTAMINATION CONTROL

Radiologically Controlled Areas (RCAs) were established for the characterization activities that were considered to potentially generate airborne radioactivity. RCAs were isolated from the general work areas through the use of radiation barrier rope and warning signs. Prior to leaving an RCA, personnel monitored themselves for the presence of radioactive contamination.

Contamination control practices also included preventing survey instrumentation from coming into direct contact with surfaces potentially contaminated with loose activity, and ensuring that sampling equipment was cleaned between sample collections.

Contamination control precautions included the use of local contamination containments and HEPA ventilation units implemented during the opening of the Plug Storage Area and during the opening of the inner beam port plugs. Air sampling was periodically performed throughout the activities to ensure that there was no spread of airborne contamination.

Georgia Tech and NES surveyed and released all tools, equipment and material associated with the characterization activities. Final authorization for release of all equipment and materials was done by Georgia Tech Health Physicists.

A survey was performed to verify that contamination had not spread during characterization activities. These surveys aided in the substantiation that administrative and engineering controls implemented during characterization operations were adequate.

4.2 EXPOSURE CONTROL

Each radiation worker making an entry to the RCA was provided with a personal ionization chamber (PIC), thermoluminescent dosimeter (TLD) and finger ring. These personal monitoring devices were to be worn by all workers during characterization activities. In addition, pre- and post- urinalysis samples were collected and analyzed for personnel who worked in the RCAs. This helped monitor internal exposure.

Controls such as area radiation surveys, posting, shielding and control of work activities involving radioactive material were placed throughout the areas involved in the characterization effort to minimize external exposure. Monitoring airborne radioactivity and surface contamination helped to control internal exposure.

4.3 RADIATION WORK PERMITS

Characterization work within established RCAs was performed under the administration of a Radiation Work Permit (RWP) system. The RWPs provided the radiation workers with the radiological conditions under which work in the RCA was to be performed. The RWPs required the use of engineering controls and protective clothing, as necessary, to ensure that the work was accomplished in a radiologically safe manner while maintaining personnel radiation exposure as low as reasonably achievable (ALARA).

The RWPs were prepared by the NES Radiological Engineer based on expected radiological conditions. Georgia Tech reviewed and approved all RWPs.

All employees, workers and individuals associated with the characterization activities observed and complied with the provisions stated in the RWPs and those established by the NES radiation control procedures and the Georgia Tech Radiation Safety Manual. Georgia Tech procedures were implemented to minimize the possibility of contamination spread and personal injury.

5.0 RESPIRATORY PROTECTION

Baseline ambient air samples were performed inside the NNRC building prior to characterization activities. Air samples were also taken both on a routine basis and as specified in the RWPs. Routine air samples were taken during characterization activities that were expected to generate airborne radioactivity (i.e., opening of the plug storage holes). The air samples were used to verify that these characterization activities were not spreading contamination.

Respiratory protection was required when air concentrations were expected to exceed the most conservative values of 3×10^{-11} μCi/ml for alpha emitters and 9×10^{-9} μCi/ml for beta-gamma emitters. Airborne concentrations did not approach these levels.

6.0 QUALITY ASSURANCE

NES Procedure 82A9085, "Quality Assurance Program Plan for the Site Characterization of the Georgia Tech Neely Nuclear Research Center," (Reference 3) was implemented during the characterization project. The elements of the plan included daily instrument performance checks, data review of the surveys, review of radiation work permits, verification of the accuracy of devices used for radiological sampling, and the use of properly calibrated instrumentation. A log was maintained for operational checks. All radiation detection instruments were calibrated with NIST traceable radioactive sources. Instrument calibration certificates and response range checks are provided in Attachment A.

Other elements of the Quality Assurance Program included sample analysis duplication. Five (5) percent of the gamma spectroscopy samples were sent to an independent laboratory to verify on-site analysis results. Labels, sample collection forms and logs were used to assure that samples could be traced from collection to completion of analyses. See Attachment B for a comparison of the on-site and independent laboratory analysis results.

7.0 RELEASE CRITERIA AND GUIDELINE VALUES

7.1 RADIOLOGICAL RELEASE CRITERIA

The release criteria established for the NNRC decommissioning project include compliance with the surface contamination levels presented in the U.S. Nuclear Regulatory Commission's (USNRC) Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors" (Reference 4). Specifically, surface contamination on building surfaces, material and equipment equal to or above the USNRC guidance stated below was considered contaminated and cannot be released for unrestricted use.

For beta-gamma emitters except Sr-90:

> 5000 dpm/100 cm^2, average (over 1 m^2), fixed plus removable contamination;
> 1000 dpm/100 cm^2, removable contamination; and
> 15,000 dpm/100 cm^2, maximum (not more than 100 cm^2), fixed plus removable contamination.

U-nat., U-235, U-238, and associated decay products:

> 5000 dpm α/100 cm^2, average (over 1 m^2), fixed plus removable contamination;
> 1000 dpm α/100 cm^2, removable contamination; and
> 15,000 dpm α/100 cm^2, maximum (not more than 100 cm^2), fixed plus removable contamination.

For transuranic contamination (e.g., plutonium isotopes), Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, and I-129:

> 100 dpm/100 cm^2, average (over 1 m^2), fixed plus removable contamination;
> 20 dpm/100 cm^2, removable contamination; and
> 300 dpm/100 cm^2, maximum (not more than 100 cm^2), fixed plus removable contamination.

Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, and I-133:

> 1000 dpm/100 cm^2, average (over 1 m^2) fixed plus removable contamination;
> 200 dpm/100 cm^2, removable contamination; and
> 3000 dpm/100 cm^2, maximum (not more than 100 cm^2) fixed plus removable contamination.

Guidelines for release criteria for radioactive material concentrations in soil and material are taken from USNRC "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material" (Reference 5).

Radioactivity in soil - Activity concentration in soil is predominantly due to naturally occurring radionuclides in the uranium and thorium series. The radioactivity limit in soil is 10 pCi/g for naturally-occurring radioactive materials (NORM). Other radionuclides found may be attributed to the soil in that area.

Airborne radioactivity limits - The airborne radioactivity control limits are predicated upon maintaining exposure below those permitted by 10 CFR 20, Appendix B, Table 1, column 1. Beta-gamma ($\beta\gamma$) airborne particulate radioactivity limits are no greater than $9x10^{-9}$ μCi/ml above natural background. This limit is based on Co-60 which is the most limiting β or γ emitting isotope from the standpoint of airborne concentration. Alpha (α) airborne radioactivity will not be greater than $3x10^{-11}$ μCi/ml above normal background. This limit is based on insoluble Pu-238 which is the most limiting α emitting isotope from the standpoint of airborne concentration. However, the concentration of non-naturally occurring radionuclides is expected to be non detectable in non-radiological areas.

Retention tanks radioactivity release limits- Liquid samples were sampled and analyzed for alpha and beta radioactivity. Using the most conservative values, the release limits are 0.03 pCi/ml (the derived concentration for Pu-239) for alpha activity, and 1.0 pCi/ml (the derived concentration guide for Sr-90) for beta activity.

7.2 HAZARDOUS MATERIAL GUIDELINE CRITERIA

Hazardous material- a RCRA (Resource Conservation Recovery Act) hazardous waste is defined as waste which meets one of the following two criteria:

1. It exhibits the 40 CFR 261.20 specific properties of:
   - ignitability,
   - corrosivity,
   - reactivity,
   - toxicity, or
2. It is listed in 40 CFR 265 as a RCRA hazardous waste, and it exceeds the specified concentration limits.

Lead in paint - Paint is considered to be lead-containing if it has greater than 0.5% lead by weight.

Asbestos release criteria - A material is considered to be asbestos-containing material (ACM) if it contains greater than 1% asbestos by weight.

8.0 CHARACTERIZATION METHODOLOGY

8.1 OBJECTIVE

The objective of the characterization of the NNRC was to determine the extent and level of the radioactive and hazardous material contamination present in the reactor, containment building and surrounding exterior areas. Emphasis was placed on the characterization of the reactor core structure, distribution of neutron-induced activity of the surrounding concrete, the extent and level of residual contamination remaining in peripheral support systems, and outlying areas of the containment building and exterior areas. Characterization guidelines, quality assurance, sampling locations and documentation requirements were defined in the Characterization Plan.

8.2 INTERPRETATION OF SURVEY RESULTS

Direct contamination readings were converted to dpm/100 cm^2 using the following equation:

$$dpm\ /100cm^2 = \frac{Gross\ cpm\ -\ Background\ cpm}{(instrument\ efficiency)\ \left(\frac{probe\ area}{100\ cm^2}\right)} \qquad (1)$$

The average background count rate for the direct survey instrumentation was determined from a series of ten 1-minute counts. Each direct measurement of fixed contamination was 1 minute in duration.

8.2.1 Detection Sensitivity

The detection sensitivity of a measurement system refers to a radiation level or quantity of radioactive material that can be measured or detected with some known or estimated level of confidence. This quantity is a factor of both the instrumentation and the technique or procedure being used. The primary parameters that affect the detection capability of a radiation detector are the background count rate, the detection efficiency of the detector and the counting time interval.

8.2.1.1 Minimum Detectable Activities (MDAs)

The Minimum Detectable Activity (MDA) is defined as the smallest quantity of radioactivity that could be measured under specified conditions. The MDA depends on the Lower Limit of Detection (LLD) and on the counting efficiency of the counting system.

Calculation of the MDA determines the instrument detection sensitivity, which gives the statistically determined quantity of radiation that can be detected at a preselected confidence level. The MDA is a deductive estimate of the minimum activity level which is practically measurable with a specific instrument. MDAs were calculated for the radiological survey instrumentation using the following equation:

$$MDA = \frac{\frac{2.71}{T_s} + 3.29\sqrt{\frac{R_b}{T_b} + \frac{R_b}{T_s}}}{(efficiency)\left(\frac{probe\ area}{100\ cm^2}\right)}, \qquad (2)$$

where:

R_b = background counting rate (cpm)
T_b = background counting time (min), and
T_s = sample count time (min).

The MDA is the minimum number that the instrument can detect with 95% confidence.

For MDA results refer to characterization survey data sheets in Attachments D through G.

8.2.1.2 95% Confidence Level

Assuming that a system has a background response and that random and systematic uncertainties are accounted for separately, decision levels of the amount of actual activity present can be determined. This decision level considers only an alpha, or type 1, error. A type 1 (or "false positive") occurs when a detector response is considered to be above background when, in fact, only background radiation is present. That is, saying that there is activity in the sample when in fact there is none. The 95% confidence level means that false positives are present 5 times out of 100. A beta, or type 2, error occurs when a detector response is considered to be background when in fact radiation is present at levels above background. That is, this "false negative" shows no activity when there really is activity. In this project, the 95% confidence level was calculated for every survey location.

8.2.1.3 Lower Limit of Detection (LLD)

A Lower Limit of Detection (LLD) is based on consideration of both the alpha and beta errors. The LLD was calculated for each isotope of concern identified from spectral data during sample analysis using the Nucleus PCA II Multi-Channel Analyzer (MCA) with the EG&G Ortec Ge(Li) detector using the following equations (Reference 6):

$$LC = k[(N/2n) \ x \ 1 + (N/2n) \ x \ (B_1 + B_2) + I + (var^2)]^{1/2} \qquad (3)$$

where:

LC = critical level
I = interference level
var = variance
B_1 = sum of n channels below the peak
B_2 = sun of n channels above the peak
N = number of channels in peak
n = number of background channel on each side of peak
k= constant (1.645 for 95%)

It should be noted that because background data is determined at the same time as photopeak area, “I” (interference level) in the equation is equal to 0 (as is “var”) for the purposes of calculating the LLD. Therefore setting “I” to 0 and “var^2” to 0 reduces equation (2) to (Reference 7):

$$LC = k[(N/2n) \ x \ 1 + (N/2n) \ x \ (B_1 + B_2)]^{1/2} \qquad (4)$$

and,

$$LLD = LD \ (\ T \ x \ Y \ x \ E \ x \ V \ x \ 37000 \ ) \qquad (5)$$

where:

LLD = Lower Limit of Detection
LD = $k^2 + 2LC$
T = Data collection time in seconds
Y = Gamma ray yield in question
V = sample volume in grams

By using equations (2), (3), (4) and (5), and spectral data the LLDs can be calculated.

8.2.1.4 Instrument Efficiencies

National Institute of Science and Technology (NIST) traceable check sources were used to determine instrument detection response check for portable instrumentation (Refer to Attachment A). Efficiencies for laboratory instrumentation (i.e., Liquid Scintillation Counter) were determined by Georgia Tech procedures and are also included within the survey data sheets in Attachments D through G. Detection instruments used were certified as acceptable for detecting radiation levels appropriate to the NNRC. Records of calibration and periodic operational checks of fixed and portable laboratory radiation measuring equipment are also included in Appendix A. These records include frequencies, methods, dates, personnel training and traceability of calibration sources to the National Institute of Science and Technology.

8.3 SURVEY PROCEDURES

The survey methodology for the characterization was divided into indoor and outdoor survey procedures. Radiation measurements and sampling consisted of direct measurements, removable surface contamination measurements, exposure rate measurements and air sampling. Systematic measurements of direct and removable surface contamination were performed to quantify the levels of residual activity.

MDAs were calculated for the direct and removable contamination survey instrumentation. The 95% confidence levels were also determined for each survey location. MDA calculations and 95% confidence level results are provided within the survey data sheets in Attachments D through G.

8.3.1 Survey Locations

8.3.1.1 Outdoor Survey

Gamma scans were performed around the NNRC and along adjacent roads, parking lots and walkways to identify any locations of elevated activity

levels. Locations of elevated radiation levels were noted for further investigation. Alpha and beta-gamma direct measurements and smears were performed on miscellaneous equipment, materials and debris located around the building. Exact locations were specified in the Characterization Plan. Exposure rates were obtained at representative locations around the NNRC and at locations of elevated radiation identified by gamma scans. Exposure rates were measured at 1 meter above the surface.

After scanning, systematic soil sampling was performed at approximately uniformly-spaced intervals. These included surface (0-15 cm) and subsurface (below 15 cm) soil samples, collected for analysis of radioactive material at specific locations identified in the Characterization Plan. Soil samples were taken in accordance with NES Procedure No. 82A8104, "Soil Sampling Procedure" (Reference 8).

8.3.1.2 Indoor Surveys - Areas Outside of the Reactor Building

Measurements of total alpha and beta-gamma contamination levels were performed at randomly selected locations on floors, walls, ventilation ducts, floor drains, heater coils, etc. Composition tile and paint were removed to determine the location and amount of contamination on the floor of the experiment area and corridors. Smears for the determination of removable contamination were taken at locations of direct measurements. Equipment and materials throughout the NNRC were also scanned. Limited measurements of total and fixed contamination levels were performed on the equipment and materials.

Building drains were scanned and smeared for elevated alpha and beta-gamma radiation levels. Measurements of total and removable alpha and beta-gamma contamination levels were performed on selected sink traps and drains in areas with a history of radioactive material use or potential contamination.

Samples of residue were obtained from areas identified in the Characterization Plan including floor drains, sink traps, and ventilation ductwork.

8.3.1.3 Indoor Survey - Restricted Areas

With the exception of the fuel assemblies, all reactor components and associated systems are still in place. Other items including source containers, shipping casks, miscellaneous equipment used during operation and materials stored in the plug storage vaults were also included in this survey.

Measurements of total alpha and beta-gamma contamination levels were performed at randomly selected locations on the reactor structure, walls, overhead crane, ventilation ducts, floor drains, and heater coils.

Concrete samples were taken to determine the location and amount of contamination on the floor of the reactor room and surrounding areas. Smears for the determination of removable contamination were taken at locations of direct measurements. Limited measurements of total and fixed contamination levels were performed.

Floor drains were scanned and smeared for elevated alpha (α) and beta-gamma ($\beta\gamma$) radiation levels. Measurements of total and removable alpha and beta-gamma contamination levels were performed on selected sink traps and drains in areas with a history of radioactive material use or potential contamination.

Samples of residue were obtained from areas identified in the Characterization Plan including floor drains, sink traps, and ventilation ductwork.

8.3.1.4 Pipes, Drain lines, Sump and Ductwork

Bulk samples were collected to determine the nature and extent of contamination present. The primary water system has been drained. Smear samples were obtained from inlet and outlet areas of these drain pipes and the accessible points of the exhaust system ductwork. The sump is a concrete pit 5-ft in diameter and 10-ft deep. At the bottom there is approximately one and a half inches of water, sludge and antifreeze. Samples of water and sludge were individually counted on a Ge(Li) gamma spectroscopy system and liquid scintillation counter.

8.3.2 Radiation Measurements and Sampling

The survey procedures contained herein are consistent with the recommendations of NUREG/CR-5849, "Manual for Conducting Radiological Surveys in Support of License Termination" (Reference 9) and were performed in accordance with the Characterization Plan.

8.3.2.1 Direct Measurements

One direct beta-gamma and one direct alpha surface contamination reading were taken at random locations identified in the Characterization Plan. Direct measurements were performed systematically and at randomly selected locations and at locations with elevated activity. A 1-minute direct beta-

gamma and a one 1-minute direct alpha surface contamination reading were taken at each designated location. The Ludlum models 2220 and 2221 meters with 44-9 GM pancake probes were used to perform the beta-gamma direct measurements, while the Ludlum model 2220 rate meters, with AC-3 and 43-65 alpha scintillation probes, were used to perform the alpha direct measurements (See Section 8.3 for survey instrument discussion).

8.3.2.2 Removable Contamination Measurements

One smear for removable contamination was collected at each location of a direct surface activity measurement. The smears were obtained by wiping an area of approximately 100 cm^2 using a dry filter paper. Smears were placed into envelopes to prevent cross-contamination.

Each smear was counted on a gas flow proportional automatic smear counter for measurements of alpha, beta and gamma removable surface contamination. Smears were also obtained at various locations and analyzed for tritium using liquid scintillation. The results of the removable surface contamination surveys were documented on the appropriate survey form.

8.3.2.3 Exposure Rate Measurements

Per the Characterization Plan, exposure rate measurements at 1 meter from the surface were obtained from the storage tanks throughout the NNRC. The Bicron Micro-Rem meter was used to obtain the exposure rate measurements.

8.3.2.4 Air Sampling

Air sampling was performed in selected indoor areas for protective measures and to detect releases of airborne contaminants. To provide an indicator of what kinds of airborne contamination might be expected during "routine phases" of decommissioning, air sampling was also performed during the most intrusive tasks of the characterization. These tasks involved the opening of the storage plugs, horizontal beam ports, the vertical beam ports, and during beam port plug sectioning.

8.3.3 Radiological Survey Instrumentation

Instrumentation was chosen to assure that the MDAs for the selected instrument/technique were less than the release criteria. Table 8.1 lists the instrumentation used during the characterization survey, its primary use, calibration due date and serial number. Instrument calibration data sheets and instrument efficiencies are included in Attachment A.

Table 8.1: Radiological Characterization Survey Instrumentation

Instrumentation	Serial Number Meter(s)/Probe(s)	Application	4π Efficiency (%)	Calibration Date(s)	Calibration Due Date(s)
Ludlum 2220 ratemeter/scaler w/ 43-65 probe	50061/063291 48409/62385	Alpha direct measurements and scan surveys	18 18	09/19/97 09/26/97	09/19/98 09/26/98
Ludlum 2220 ratemeter/scaler w/ AC-3 probe	50062/408951	Alpha direct measurements and scan surveys	7.7	09/19/97	09/19/98
Ludlum 2220 ratemeter/scaler w/ 44-9 probe	52823/11150	Beta-gamma direct measurements and scan surveys	21.87	09/16/97	09/16/98
Ludlum 2221 ratemeter/scaler w/ 44-9	68537/66762	Beta-gamma direct measurements and scan surveys	31.75	04/21/97	04/21/98
Eberline ESP-1 w/ HP-210	635	Beta-gamma direct measurements and scan surveys	11.1	09/22/97	09/22/98
Automatic Counting System	LB5100W	P-10 Gas Flow Proportional Counting System	α 28.35 β 43.45	As needed	As needed
Packard Model A4430 Liquid Scintillation Counter	13795	Liquid scintillation used to detect low energy betas	59.8	As needed	As needed
Bicron Micro-Rem	B171G B218L	Organic scintillator exposure rate surveys	Not applicable	04/29/97 07/28/97	04/29/98 07/28/98
Automess Teletector	37266	GM exposure rate surveys	Not applicable	9/23/97	9/23/98
Radeco High Volume Air Samplers	5389 5387	Job specific air sampling	Not applicable	08/06/97 08/06/97	08/06/98 08/06/98
EG&G Ortec MCA w/ Ge(Li) Detector	N/A	Gamma Spectroscopy Sample Analysis	Not applicable	As needed	As needed
Scitec Map-4 XRF Spectrum Analyzer	4-1412	Lead Inspections	Not applicable	As needed	As needed

8.4 HAZARDOUS MATERIAL SCREENING PROCEDURES

Hazardous material screening was performed in addition to the radiological sampling to provide a screening of the NNRC for the potential presence of asbestos containing materials (ACM) and also for lead in paint above the leachable limits specified in Title 40 of the Code of Federal Regulations, Part 261 (Reference 10).

8.4.1 Asbestos and Lead-Based Paint Screening Surveys

Screening for both ACM and lead-based paint was performed to determine the potential for handling radioactively contaminated ACM and the possibility for generating mixed wastes during the NNRC decommissioning.

On October 28, 1997, Enviromed Services, Inc., provided an asbestos analysis of samples collected in the NNRC using polarized light microscopy (PLM). This included analysis of suspect ACM from the roof of the containment area, in the pipe wrapping over the control room and in the hoods of the Radiochemistry Room and Decontamination Room. On October 13, 1997, NES conducted a lead-based paint screening survey of the NNRC containment area. The Scitec Map-4 XRF Spectrum Analyzer was used to determine if lead was present in the paint. The results of the XRF, in units of ppm of total lead, are intended to demonstrate the potential for the presence of leachable lead quantities above the limits specified in Title 40 of the Code of Federal Regulations, Part 261 (Reference 10).

The ACM and lead-based paint screening surveys were conducted to meet three objectives:

- Identify radioactively contaminated ACM and radioactively contaminated lead-based paint (LBP) prior to decommissioning and renovation of the facility.
- Determine if asbestos is present on the ceiling and if lead is present in the paint undercoats of the containment area, air-conditioning unit, and hoods in the Decontamination Room and Radiochemistry Room.
- Identify the presence of ACM and LBP prior to radiological sampling to ensure that the characterization process does not create an asbestos or lead exposure hazard or generate a mixed waste and/or regulated waste concern(s).

9.0 BACKGROUND DETERMINATIONS

To interpret quantitative results of the characterization, the survey results were compared to background values typically found in the natural environment, and in structures where no work has ever been performed with radioactive materials. Refer to Attachment C for a list of daily background data and survey locations.

9.1 OUTDOOR BACKGROUND MEASUREMENTS AND RESULTS

Background samples and measurements of open land areas were collected at locations unaffected by reactor building effluent releases. The open land areas were sampled (see the Characterization Plan for background soil sample and exposure rate measurement locations). The Techwood Avenue area was preferred because it has no history of licensed operations and is not affected or distributed by on-site activities including storm drains, railroad tracks, material handling areas and fill areas. The following measurements and soil samples were taken:

- Five (5) surface soil samples using NES Procedure 82A8102, "Soil Sampling Procedure" (Reference 8).
- Ten (10) exposure rate measurements using the Bicron MicroRem (one (1) meter from any surface).

Surface results show the presence of Kr-85 and naturally occurring radioactive material (NORM) indicated by small concentrations of Th-232 and U-238. No other isotopes were detected. Exposure rates in the Plant Operations Building ranged from 7 to 10 μR/hr (at 1 meter from any surface) with the average exposure rate of 9 μR/hr. Removable and direct contamination above release criteria was not found in the background survey locations.

9.2 INDOOR BACKGROUND MEASUREMENTS AND RESULTS

For direct measurements, probes were to move from point to point, and, as a result, it was expected that the background would vary significantly due to variations in background, source materials and changes in geometry and shielding. The background information, which differs for each floor and on each day the information, is provided in Attachment C.

To establish background radiation levels for the first floor of the containment building, direct measurements were obtained along the northeast wall. This area was preferred for background determination because of its positioning opposite the plug storage area, and away from the openings of the horizontal beam ports, which could produce radiation streaming. The average of the measurements taken at this area was used when determining background for the surveys on the first floor.

On the second floor, background determinations were obtained along the northeast wall of the outer area of the Control Room. This area was chosen to establish background radiation

levels since it was an adequate distance from the plug storage area. As with the first floor, the average of the measurements taken in this area was used when determining background for the surveys on the second floor.

Similar readings were obtained along the north wall of the ground floor. This area was chosen due to its distance from the Process Equipment Room and reactor faces.

The following measurements were taken within the mentioned areas:

- Thirty (30) direct $\beta\gamma$ measurements on walls, floors and ceilings.
- Thirty (30) direct α measurements on walls, floors and ceilings.

Additional measurements were taken at the Plant Operations Building, a structure on campus that is similar to the NNRC building. These measurements were used to show a comparison in radiation levels between the Reactor Building and other buildings built during the same time period. The Plant Operations Warehouse was preferred since it has no history of licensed radioactive operations and is of similar construction. The following measurements were taken within the Plant Operations Warehouse:

- Thirty (30) direct $\beta\gamma$ measurements on walls, floors and ceilings.
- Thirty (30) direct α measurements on walls, floors and ceilings.
- Ten (10) exposure rate measurements using the Bicron MicroRem (one (1) meter from any surface).
- Three 1 gram material samples of concrete. The samples were analyzed by gamma spectroscopy and liquid scintillation counting.

Refer to Attachment D for the Plant Operations Warehouse survey results.

10.0 CHARACTERIZATION RESULT SUMMARIES

At the request of the Georgia Institute of Technology, NES conducted a site characterization of the Neely Nuclear Research Center located on the campus in Atlanta, Georgia. Radiological characterization activities included direct and removable radiation measurements inside and outside the NNRC, exposure rate measurements, contamination levels on building surfaces and measurements of concentrations in soil, sediment and water. Asbestos and lead-based paint screening was also performed to identify areas in the restricted area that could pose potential contaminated asbestos or mixed waste hazards during decommissioning.

10.1 BUILDING SURVEY RESULTS

Each area surveyed included the floor, ceiling and wall areas including the suspect systems and components considered potentially contaminated. The background obtained at each floor was considerably high in certain areas and this may have led to some of the high readings. Refer to Attachment C for a list of survey locations by data point number. Field data is provided in Attachment N.

10.1.1 Outside of the Reactor Building

Areas surveyed outside of the reactor building in the NNRC were found to contain radioactive contamination above USNRC release criteria.

- **Chemical Addition Tanks** - Contamination was not detected by direct measurements on Tank No. 2. However, direct $\beta\gamma$ contamination up to 57, 365 dpm/100cm^2 were detected on Tank No. 1. Exposure rate measurements also show higher readings on Tank No. 1. The maximum exposure rate measurement on Tank No.2 was 88 μR/hr while the exposure rate measurement in Tank No.1 was 350 μR/hr. Removable contamination was not detected on the chemical addition tanks. Exposure rates on the other storage tanks ranged from 52 to 1,250 μR/hr. Refer to Attachment I for exposure rate measurements for the waste storage tanks.

- **Radiochemistry Room Hood-** The miscellaneous equipment in the hood in the Radiochemistry Room showed elevated activity with maximum $\beta\gamma$ direct activity up to 3,435,696 dpm/100cm^2. The lower portion of the hood had liquid on the bottom. This area indicated a direct $\beta\gamma$ activity of 31, 672 dpm/100cm^2. Removable α activity of 74 dpm/100cm^2 was found on the surface of a covered beaker in the upper portion of the hood.

- **Decontamination Room** - The Decontamination Room consists of a fume hood and an additional room used as a walk-in hood. The contamination identified in the Decontamination Room showed direct $\beta\gamma$ measurements at 17,386 dpm/100 cm^2. All of the stored materials and equipment in the walk-in hood had elevated direct $\beta\gamma$ activity with a maximum reading of

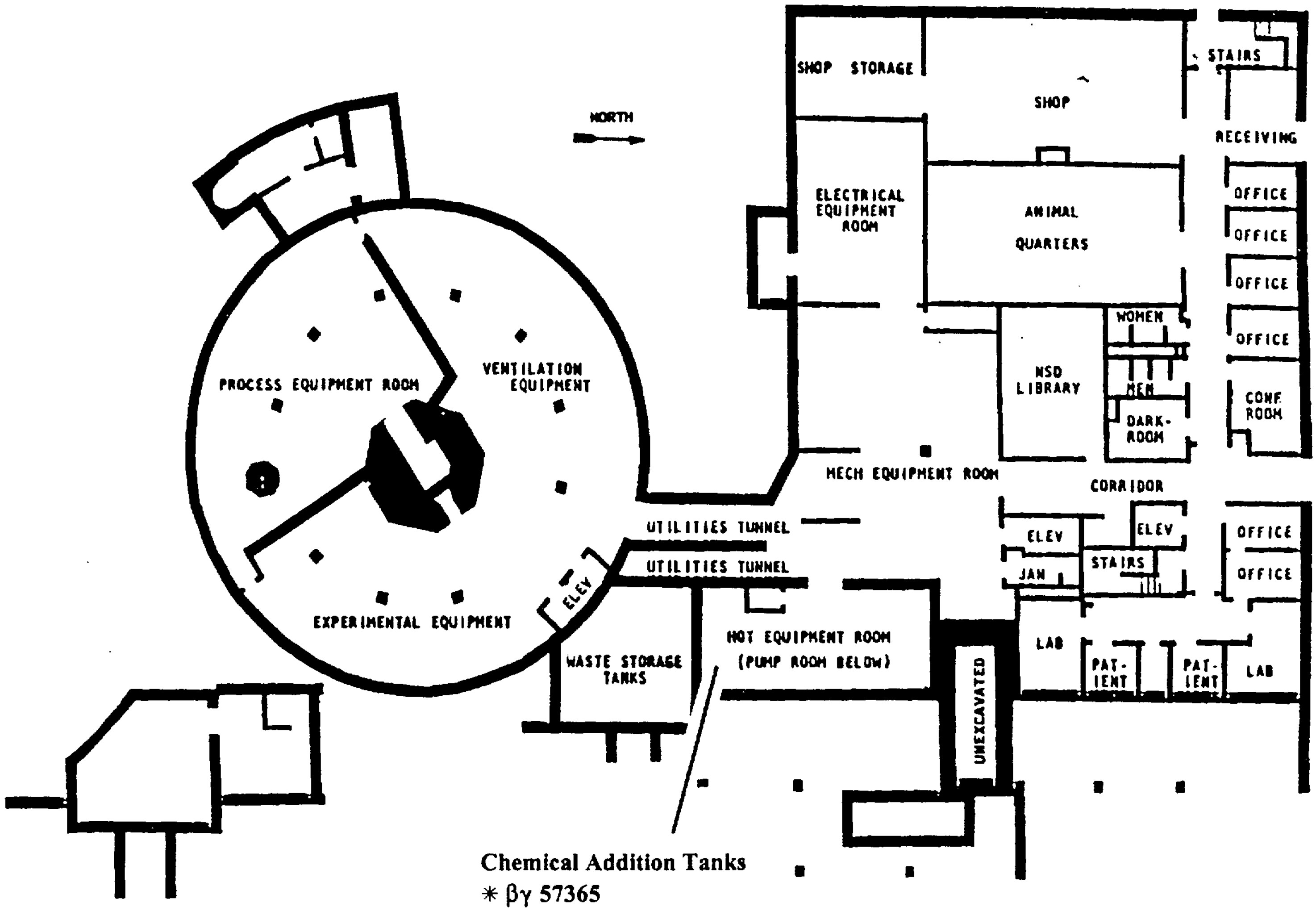
Table 10.1 shows the range of activity in the areas outside of the reactor building. Figure 10.1 and 10.2 show the maximum direct and removable activity per area. Attachment D provides more extensive data per area.

Table 10.1 Areas Outside of the Reactor Building Survey Data Summary

Survey Location	No. of Survey Points	Range of Activity (dpm/100 cm^2) Direct Alpha	Direct MDA	Direct Beta-Gamma	Direct MDA	Removable Alpha	Removable MDA	Removable Beta-Gamma	Removable MDA
Plant Operations Warehouse	30	0 to 53	24	1189 to 4298	1436	None Taken	-	None Taken	-
NES Field Office-Before Characterization	20	0 to 2	82	-3018 to -1859	1808	0 to 3	13	0 to 12	17
Chemical Addition Tanks	12	-47 to -9	123	-42 to 57365	1261	0 to 3	13	0 to 10	17
Radiochemistry Room Hood	13	-35 to 0	116	8349 to 343569	1808	0 to 74	13	0 to 988	17
Decontamination Room	31	0 to 4	123	-1428 to 17386	1245	0 to 7	13	0 to 102	17
Decontamination Room- Hood & Vent System	10	-47 to -9	123	-504 to 1092	1245	0 to 3	13	0 to 12	17
Decontamination Room-Walk-in Hood	40	-47 to 0	123	567 to 3931213	1245	0 to 7	13	0 to 46	17
HEPA Ventilation Unit	10	None Taken	-	None Taken	-	0 to 10	13	1 to 95	17
Scaffolding Equipment	16	None Taken	-	None Taken	-	0 to 7	13	0 to 42	17
Manlift	25	None Taken	-	None Taken	-	0 to 7	13	0 to 7	17
Clean-up Survey	115	None Taken	-	None Taken	-	0 to 10	13	0 to 111	17
NES Field Office-After Characterization	21	-26 to -18	95	-1311 to -152	1443	0 to 3	13	0 to 23	17

Shading represent areas where activities are above acceptable limits in
U.S. Nuclear Regulatory Guide 1.86, "Termination for Operating License for Nuclear Reactors," June 1974.

∗ Direct measurements in dpm/100 cm²
❍ Removable measurements in dpm/100 cm²
Bold numbers represent activities **above** U.S. NRC Regulatory Guide 1.86 limits.
All measurements were taken **INSIDE** the facility.



Chemical Addition Tanks
∗ βγ 57365
∗ α -9
❍ βγ 10
❍ α 3

Figure 10.1 Maximum Direct and Removable Activity in the Neely Nuclear Research Center Ground Floor

✳ Direct measurements in dpm/100 cm^2
❍ Removable measurements in dpm/100 cm^2
Bold numbers represent activities **above** U.S. NRC Regulatory Guide 1.86 limits.
All measurements were taken **INSIDE** the facility.

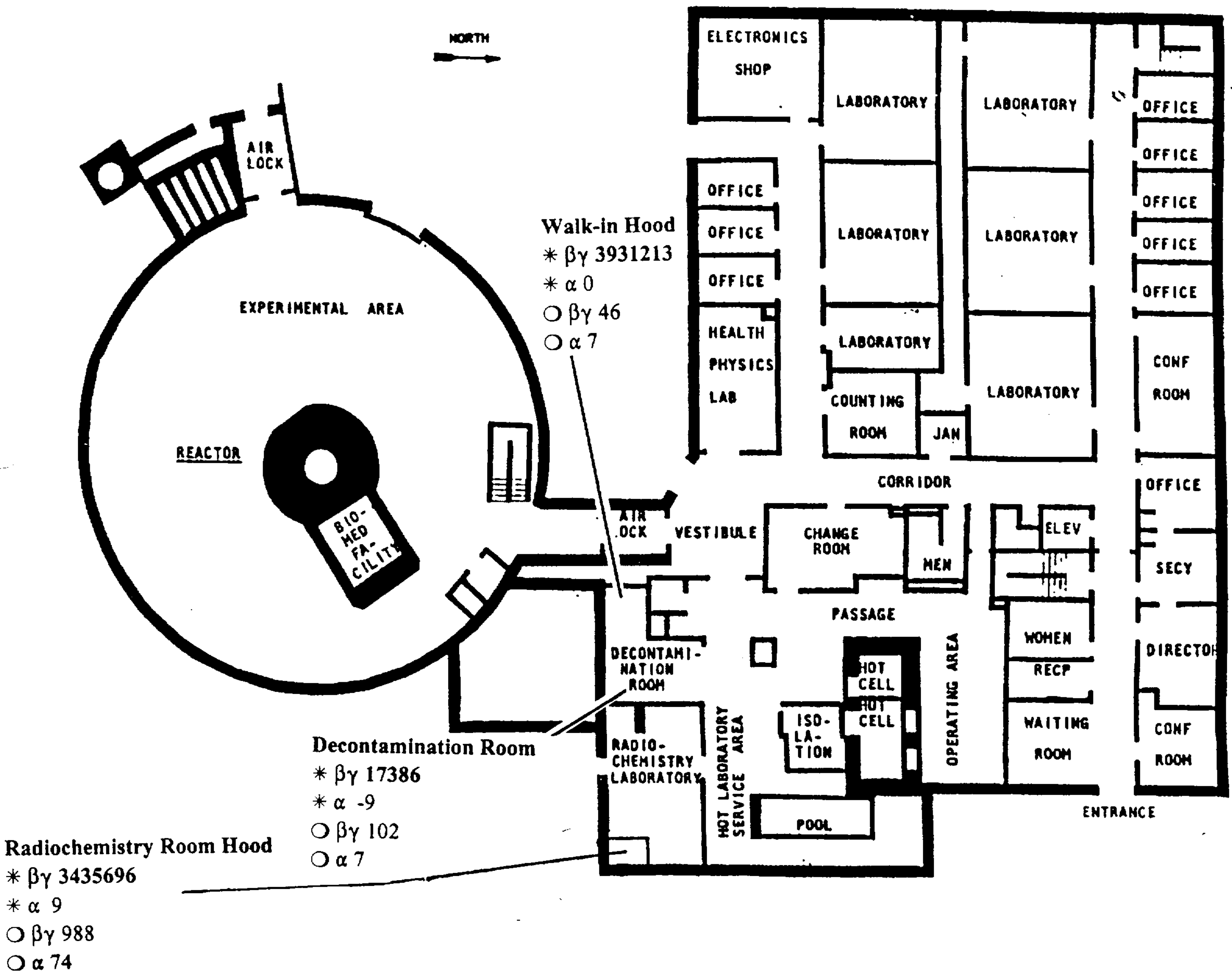


Figure 10.2 Maximum Direct and Removable Activity in the Neely Nuclear Research Center First Floor

10.1.2 Restricted Areas (excluding the Reactor)

Radioactive contamination (removable and/or fixed) was found throughout the entire facility. General area dose rates were low, ranging from 5.6 to 11 μrem/hr in the restricted area.

10.1.2.1 Ground Floor

- **Walls, Ceilings and Floors** - Direct and removable contamination was found throughout the ground floor walls, ceilings and floors. The north wall, especially, had both α and $\beta\gamma$ direct measurements above release criteria. Results show direct measurements up to 16,552 dpm/100cm^2 for $\beta\gamma$ and 110 for α. Other locations where contamination was found included the center columns and the west, northwest and southwest corners of the reactor building. Removable contamination above release criteria was not found on the walls, ceilings or floors of the ground floor.

- **Pump Room** - The contamination identified in the Pump Room showed direct α masurements of 62 dpm/100 cm^2 and direct $\beta\gamma$ measurments up to 10,547 dpm/100cm^2. Removable contamination above release criteria was not found in the Pump Room.

- **Experimental Rooms** - For reference purposes, the experimental rooms were designated as Rooms 1 and 2. Room 1 refers to the experimental room which contains the rabbit system, the other room is Room 2. The only location where activity was found was in Room 1. The vent to the left of the entrance door showed direct α measurement up to 220 dpm/100cm^2 and the rabbit system closest to the rear wall showed direct $\beta\gamma$ measurements up to 5,670 dpm/100cm^2 on the outer surface. Removable contamination above release criteria was not found throughout the experimental rooms. Removable and direct contamination above release criteria was not found in Experimental Room 2.

- **Process Equipment Room** - The process equipment room was a prime suspect for contamination because of the piping and pumps handling primary cooling water. The heat exchangers are present with all of the piping, pumps and filters still connected. Direct and removable measurements taken in the process equipment room indicate direct $\beta\gamma$ measurements up to 8,048 dpm/100cm^2 on the outer wall, 6,677 dpm/100cm^2 on the ceiling and 22, 215 dpm/100cm^2 on the floor. The maximum $\beta\gamma$ measurement of 40,399 dpm/100cm^2 was found on the surface of the ISX-1 tank. Removable α contamination was found on Tank He-1 with a measurement of 28 dpm/100cm^2. Removable contamination above release criteria was not found anywhere else in the Process Equipment Room.

- **Elevator Shaft -** Removable contamination above release criteria was not found in Elevator Shaft.

- **Hydraulic Sump -**Removable contamination above release criteria was not found in the Hydraulic Sump.

- **Bismuth Shield Area -**Removable contamination above release criteria was not found in Bismuth Shield Area. Refer to "Miscellaneous Areas" survey data sheets for results in this area.

Table 10.2 shows the range of activity on the ground floor. Figure 10.3 shows the maximum direct and removable activity per area. Attachment E provides more extensive data per area.

Table 10.2 Ground Floor Survey Data Summary

Survey Location	No. of Survey Points	Range of Activity (dpm/100 cm^2) Direct Alpha	Direct MDA	Direct Beta-Gamma	Direct MDA	Removable Alpha	Removable MDA	Removable Beta-Gamma	Removable MDA
Center Columns	14	-88 to 110	289	-2713 to 3048	1860	0	13	0 to 44	17
North Wall of Reactor	20	-110 to 110	289	-2713 to 16522	1860	0 to 3	13	0 to 10	17
West Wall of Containment	7	-88 to 88	289	-3018 to 1799	1860	0 to 3	13	0 to 37	17
Northwest Wall of Containment	12	-110 to 22	289	-3018 to 5335	1860	0	13	0 to 23	17
Northwest Corner of Containment	10	-110 to 44	289	-2957 to 5792	1860	0	13	0 to 5	17
West Corner of Containment	7	-110 to 88	289	-1311 to 21795	1860	0 to 3	13	0 to 5	17
North Wall of Containment	12	-110 to 44	289	-2286 to 1616	1860	0	13	0 to 12	17
East Wall of Containment	6	-26 to 9	116	-3048 to -549	1854	0	13	0 to 7	17
Southeast Wall of Containment	14	-44 to 62	116	-3109 to 2195	1854	0	13	0 to 5	17
Southeast Corner of Containment	11	-26 to 441	116	-2622 to 14815	1854	0 to 3	13	0 to 12	17
Pump Room	22	-44 to 62	116	-2865 to 10242	1854	0 to 3	13	0 to 12	17
Pump Room, Ceiling	17	-44 to 9	116	-1036 to 10547	1854	0	13	0 to 10	17
East Wall, Outside of Experimental Rooms	10	-44 to 247	116	-2500 to 2408	1854	0 to 3	13	0 to 12	17
Experimental Room No. 1	41	-44 to 220	116	-2926 to 5670	1854	0 to 3	13	0 to 12	17
Experimental Room No. 2	26	-44 to 18	116	-2865 to 1402	1854	0	13	0 to 7	17
Experimental Room No. 2, Ceiling	15	-35 to 26	116	-2743 to -396	1854	0 to 3	13	0 to 5	17
Rabbit System (to include unrestricted areas)	13	-44 to 123	116	-2184 to 4283	1261	0 to 3	13	0 to 122	17
Process Equipment Room	27	-47 to 66	123	-1176 to 40399	1261	0 to 32	13	0 to 502	17
Process Equipment Room, Ceiling	8	-47 to -9	123	-1302 to 6677	1261	0 to 7	13	0 to 3	17
Outer Wall of Process Equipment Room	15	-47 to 28	123	-2439 to 8048	1854	0 to 3	13	0 to 12	17
Reactor Faces-Process Equipment Room	10	-47 to 28	123	-504 to 2520	1261	0	13	0 to 10	17
Elevator Shaft	5	None Taken	-	None Taken	-	0 to 17	13	0 to 175	17
Hydraulic Sump	4	None Taken	-	None Taken	-	0 to 7	13	0 to 173	17
Miscellaneous Areas	22	None Taken	-	None Taken	-	0 to 14	13	0 to 191	17

Shading represent areas where activities are above acceptable limits in U.S. Nuclear Regulatory Guide 1.86, "Termination for Operating License for Nuclear Reactors," June 1974.

* Direct measurements in dpm/100 cm^2
❍ Removable measurements in dpm/100 cm^2
Bold numbers represent activities **above** U.S. NRC Regulatory Guide 1.86 limits.
All measurements were taken **INSIDE** the facility.

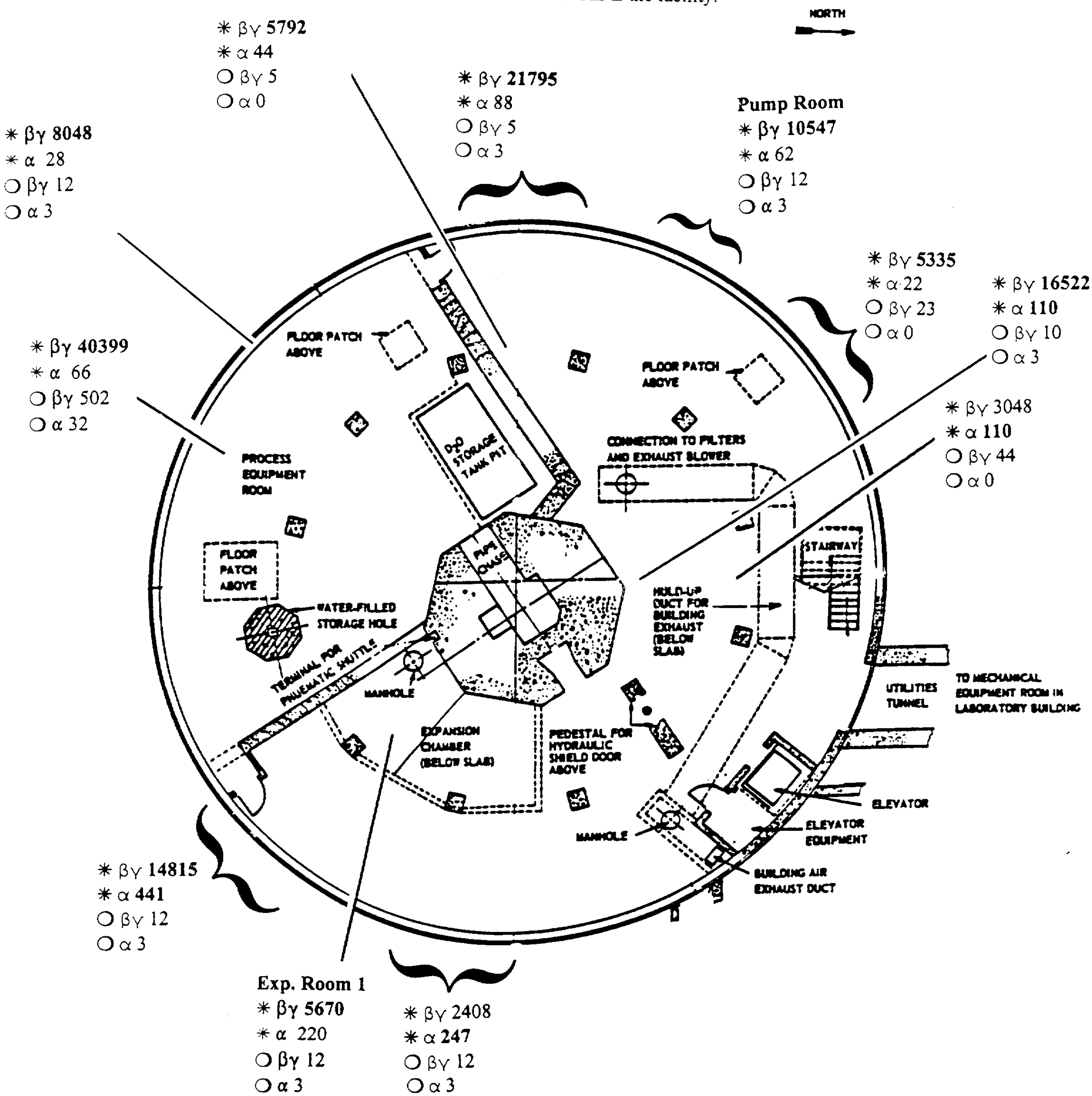


Figure 10.2 Maximum Direct and Removable Activity in the Reactor Containment Building Ground Floor

10.1.2.2 First Floor

- **Walls and Floors** - Removable and direct contamination was found throughout the first floor. The entrance door indicated direct α measurements up to 106 dpm/100cm^2 and $\beta\gamma$ measurements up to 9,907 dpm/100cm^2 on the floor in front of the entrance door. Removable contamination up to 7,468 dpm/100cm^2 was found on the wall behind the staircase. Up to 8,657 dpm/100cm^2 direct contamination was found on the lower portion of the catwalk. The wall over the emergency air lock showed direct $\beta\gamma$ measurements up to 14,266 dpm/100cm^2. The floor in front of the air lock indicated direct $\beta\gamma$ measurements up to 16,308 dpm/100cm^2. Several shipping casks placed in the first floor, southeast area, indicated direct $\beta\gamma$ contamination up to 1,421,978 dpm/100cm^2. One of these cask indicated a removable $\beta\gamma$ measurement of 1,724 dpm/100cm^2. The HEPA ventilation unit also demonstrated high activity up to 154 dpm/100cm^2 for direct α and 610,486 dpm/100cm^2 for direct $\beta\gamma$. Walls and floors throughout the first floor showed elevated activity.

- **Biomedical Irradiation Facility** - Direct $\beta\gamma$ contamination was found on the floor of the Biomedical Irradiation Facility with measurements up to 7,486 dpm/100cm^2. The ceiling indicated direct $\beta\gamma$ measurements up to 21,917 dpm/100cm^2. The sink had a direct $\beta\gamma$ measurement of 7,468 dpm/100cm^2 while the drain under the sink had a direct $\beta\gamma$ measurement of 6,920 dpm/100cm^2. Direct α contamination was found on the ceiling. Removable contamination above release criteria was not found in the Biomedical Irradiation Facility.

- **Stairs**- Direct measurements of the stairs indicate a $\beta\gamma$ activity of up to 7,926 dpm/100cm^2. Removable contamination above release criteria was not found on the stairs.

- **Irradiation Tunnels** - Direct measurements of the irradiation tunnels indicate a $\beta\gamma$ activity of up to 12,986 dpm/100cm^2. Removable contamination above the release criteria was not found on the irradiation tunnels.

- **Plug Storage Area** - The Plug Storage Area was a prime suspect for contamination because of the radioactive equipment that has been accumulated in the 18 storage plugs. Most of the storage plugs contain materials including graphite cylinders, metal rods, dummy elements, etc. Storage plugs 1, 11, 16, 17 and 18 did not have any materials inside. Attachment F provides a list of material in each storage plug.

  Removable contamination was found throughout the entire area with the α measurements reading up to 1,707 dpm/100cm^2 and the $\beta\gamma$ readings up to

87,610 dpm/100cm^2. Exposure rate measurements resulted in levels up to 235.2 mR/hr. This reading was taken in Storage Plug No. 5. Direct measurements were not taken, due to high background levels.

Removable contamination above release criteria was not found on the materials placed in the roped-off area in front of the Plug Storage Area. Due to high background levels, direct measurements were not taken.

- **Overhead Cranes-** Removable and direct contamination above release criteria was not found on the overhead cranes.

Table 10.3 shows the range of activity in the first floor. Figure 10.4 shows the maximum direct and removable activity per area. Attachment F provides more extensive data per area.

Table 10.3 First Floor Survey Data Summary

Survey Location	No. of Survey Points	Range of Activity (dpm/100 cm^2)							
		Direct				Removable			
		Alpha	MDA	Beta-Gamma	MDA	Alpha	MDA	Beta-Gamma	MDA
Entrance Door to Containment	12	-44 to 106	116	-2662 to 9907	2234	0 to 23	13	0 to 26	17
Northwest Wall of Containment	23	-44 to 9	116	-3384 to 7468	2234	0 to 3	13	0 to 7	17
West Wall of Containment	15	-44 to 44	116	-3353 to 8657	2234	0 to 3	13	0 to 14	17
Emergency Air Lock	26	-44 to 53	116	-4329 to 16308	2234	0 to 3	13	0 to 33	17
Southeast Wall of Containment	15	-44 to 35	116	-3414 to 1421978	2234	0 to 24	13	0 to 260	17
South Wall of Containment	16	-88 to 176	289	1341 to 25545	2243	0 to 113	13	0 to 1724	17
East Wall of Containment	27	-110 to 154	289	-3597 to 610486	2243	0 to 3	13	0 to 636	17
East Wall of Containment-B	16	-35 to 132	116	-5335 to 27435	2243	0 to 3	13	0 to 21	17
North Wall of Containment	12	-88 to 132	289	-4816 to 8352	2243	0 to 3	13	0 to 12	17
North Wall of Reactor	16	-66 to 352	289	-4024 to 154976	2243	0 to 3	13	0 to 97	17
Southeast Wall of Reactor	14	-88 to 616	289	-1402 to 32160	2234	0 to 3	13	0 to 21	17
South Wall of Reactor	13	-44 to 26	116	-3048 to 129858	2243	0 to 7	13	0 to 537	17
Southwest Wall of Reactor	14	-110 to 66	289	-4877 to -152	2243	0 to 3	13	0 to 14	17
East Wall of Reactor	4	-110 to 66	289	-4268 to -152	2243	0	13	0 to 5	17
Biomedical Irradiation Facility	38	-110 to 44	289	-5639 to 7468	2243	0 to 3	13	0 to 10	17
Biomedical Irradiation Facility, Ceiling	11	-88 to 110	289	-5030 to 21917	2243	0 to 3	13	0 to 26	17
Stairs from First Floor to Ground Floor	7	-44 to -18	116	-488 to 7926	1854	0 to 3	13	0 to 14	17
Irradiation Tunnels	10	-47 to -28	123	-1463 to 12986	2234	0 to 3	13	0 to 141	17
Irradiation Tunnels-B	11	-47 to 28	123	2743 to 10364	2234	0 to 3	13	0 to 63	17
Plug Storage Area	89	None Taken	-	None Taken	-	0 to 1707	13	0 to 87610	17
Equipment in Front of Plug Storage Area	42	None Taken	-	None Taken	-	0 to 14	13	0 to 309	17
Overhead Cranes	14	-44 to -26	116	-5213 to 1738		0 to 3	13	0 to 10	17

Shading represent areas where activities are above acceptable limits in
U.S. Nuclear Regulatory Guide 1.86, "Termination for Operating License for Nuclear Reactors," June 1974.

✳ Direct measurements in dpm/100 cm^2
❍ Removable measurements in dpm/100 cm^2
Bold numbers represent activities **above** U.S. NRC Regulatory Guide 1.86 limits.
All measurements were taken **INSIDE** the facility.

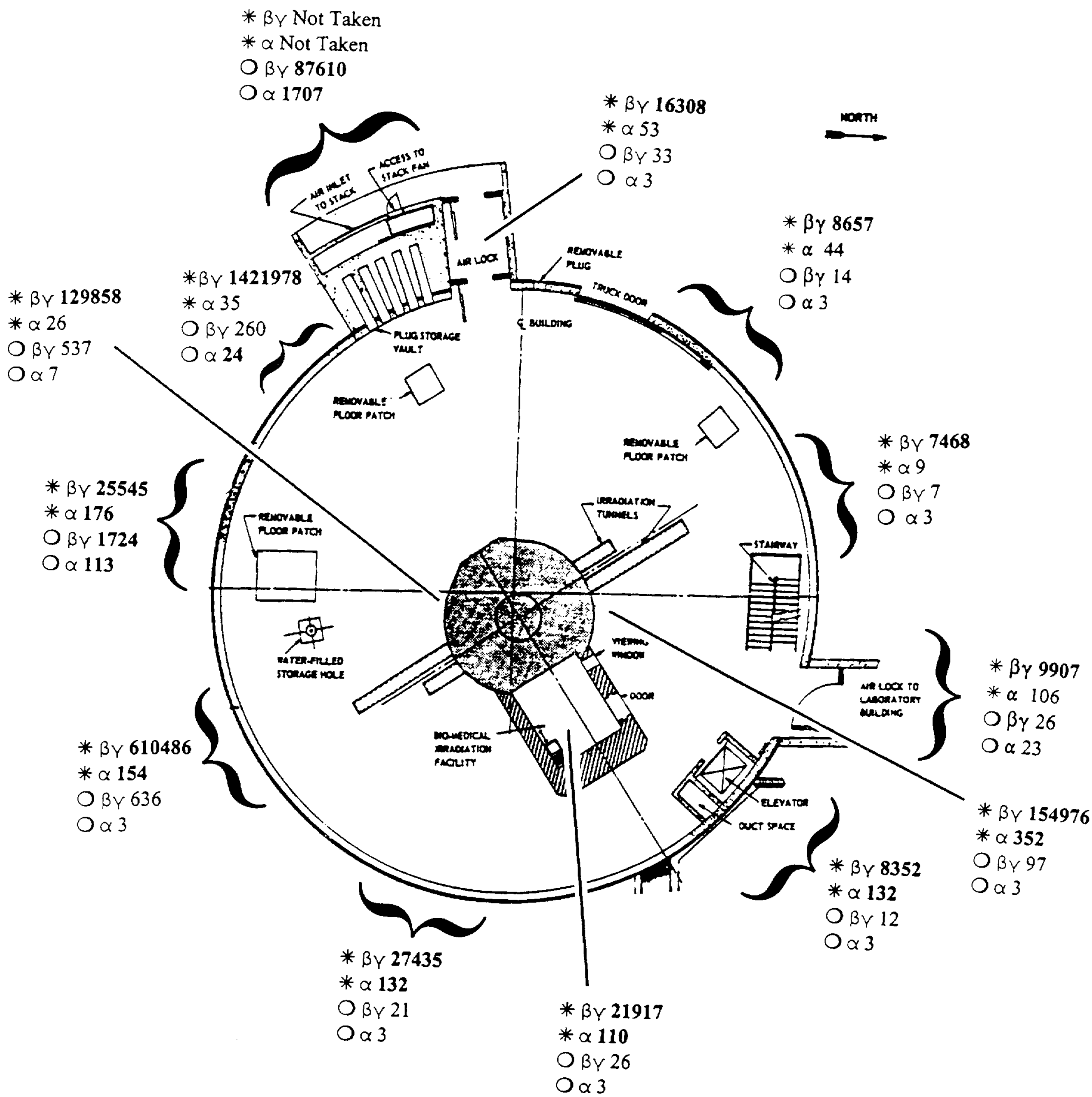


Figure 10.2 Maximum Direct and Removable Activity in the Reactor Containment Building First Floor

10.1.2.3 Second Floor

- **Control Room -** The control room is divided into two areas- the inner area where the control panel is included, and the outer area which shows the rear of the control panel. Direct α contamination was found on the ceiling of the inner area with measurements of 286 dpm/100cm^2. Direct contamination above the release criteria was not found in the inner area of the Control Room.

  The outer area of the Control Room showed direct α contamination up to 286 dpm/100cm^2 and direct $\beta\gamma$ contamination up to 5,914 dpm/100cm^2. Removable contamination above the release criteria was not found in the outer area of the Control Room.

- **Walls and Floors -** Removable and direct contamination above release criteria was not found on the walls and floor areas surrounding the Control Room.

- **Catwalk -** Direct contamination was found on the catwalk with α measurements of up to 220 dpm/100cm^2 and $\beta\gamma$ measurements up to 6,462 dpm/100cm^2. Removable contamination above the release criteria was not found on the catwalk.

- **Air-conditioning Unit -** Although direct contamination was not found in the air-conditioning unit or associated piping , direct α contamination was found on the floor next to the air-conditioning unit. This area had direct α contamination up to 638 dpm/100cm^2. Removable contamination above the release criteria was not found in the air-conditioning unit or on the areas surrounding the unit.

- **Crane Bridge -** Removable contamination above the release criteria was not found on the crane bridge.

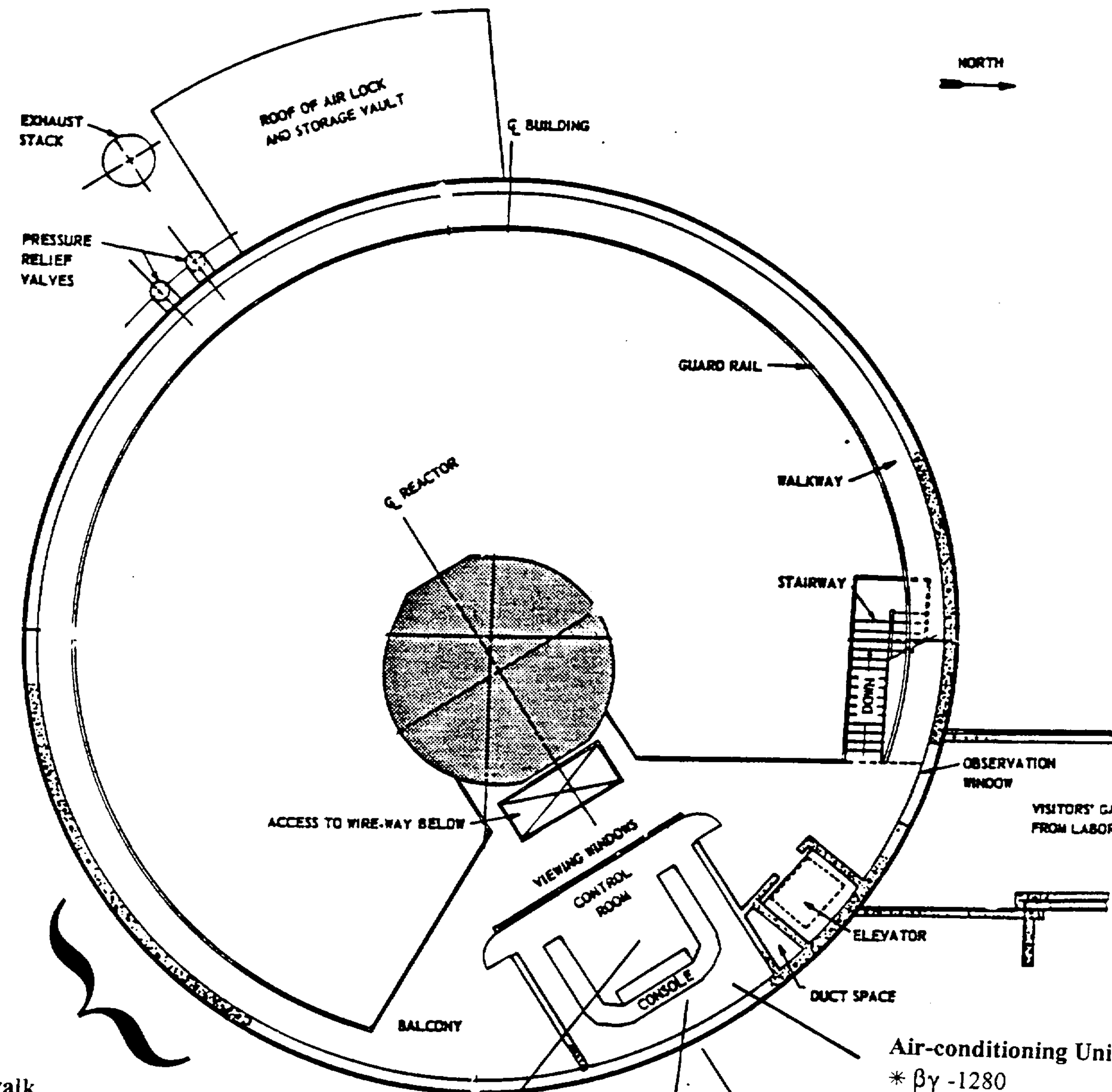
Table 10.4 shows the range of activity in the second floor. Figure 10.5 shows the maximum direct and removable activity per area. Attachment G provides more extensive data per area.

Table 10.4 Second Floor Survey Data Summary

Survey Location	No. of Survey Points	Range of Activity (dpm/100 cm^2)							
		Direct				Removable			
		Alpha	MDA	Beta-Gamma	MDA	Alpha	MDA	Beta-Gamma	MDA
Inner Area of Control Room	10	-88 to 22	289	-2439 to 1646	2663	0 to 3	13	0 to 7	17
NE Wall of Inner Area of Control Room	9	-22 to 44	289	-3018 to 1006	2663	0 to 3	13	0 to 7	17
NW Wall of Inner Area of Control Room	5	-88 to 66	289	-2225 to 732	2663	0 to 3	13	0 to 12	17
SE Wall of Inner Area of Control Room	11	-110 to 66	289	-3323 to 183	2663	0	13	0 to 10	17
Control Room, Ceiling	9	**-110 to 286**	289	-2317 to 2134	2663	0 to 3	13	0 to 7	17
Windows of Control Room	7	-88 to 22	289	-1981 to 3018	2663	0	13	0 to 7	17
Outer Area of Control Room	62	**-110 to 286**	289	**-4115 to 5914**	2663	0 to 3	13	0 to 14	17
Area West of Control Room	10	-88 to 44	289	-5274 to 4237	2663	0 to 3	13	0 to 14	17
Area East of Control Room	9	-110 to 44	289	-6920 to 2195	2663	0 to 3	13	0 to 10	17
Area East of Reactor Top	4	-66 to -44	289	-6310 to -488	2663	0	13	0 to 14	17
Catwalk	61	**-110 to 220**	289	-7865 to 6462	2663	0 to 7	13	0 to 21	17
Air-Conditioning Unit	48	**-110 to 638**	289	-6798 to -1280	2663	0 to 3	13	0 to 10	17
Top of Crane Bridge	9	None Taken	-	None Taken	-	0 to 3	13	1 to 26	17
Northeast Wall of Containment	7	-66 to -22	289	-6219 to -2743	2663	0	13	0 to 7	17

Shading represent areas where activities are above acceptable limits in
U.S. Nuclear Regulatory Guide 1.86, "Termination for Operating License for Nuclear Reactors," June 1974.

✳ Direct measurements in dpm/100 cm^2
❍ Removable measurements in dpm/100 cm^2
Bold numbers represent activities **above** U.S. NRC Regulatory Guide 1.86 limits.
All measurements were taken **INSIDE** the facility.



Catwalk
✳ $\beta\gamma$ **6462**
✳ α 220
❍ $\beta\gamma$ 21
❍ α 7

Inner Area
✳ $\beta\gamma$ 2134
✳ α **286**
❍ $\beta\gamma$ 12
❍ α 3

Outer Area
✳ $\beta\gamma$ **5914**
✳ α **286**
❍ $\beta\gamma$ 14
❍ α 3

Air-conditioning Unit
✳ $\beta\gamma$ -1280
✳ α **638**
❍ $\beta\gamma$ 10
❍ α 3

Figure 10.2 Maximum Direct and Removable Activity in the Reactor Containment Building Second Floor

10.2 REACTOR BLOCK SURVEYS AND ANALYSIS

Several survey locations of the reactor were found to have elevated activity. These areas include the reactor surface, the storage holes and the miscellaneous equipment on the reactor top. Internal reactor areas were found to contain radioactive contamination. The principal contaminants were H-3, Co-60 and Zn-65.

10.2.1 Radioactivity Results

- **Horizontal Beam Ports -** To obtain a representative sample, horizontal beam ports No. 1,2,5,8,10,11 (at both ends), 12 and 14 were surveyed. As expected, removable contamination was found throughout the inside of the horizontal beam ports. Removable measurements were up to 3,429 dpm/100cm^2 for removable α contamination and 10,614 dpm/100cm^2 for $\beta\gamma$ contamination. Exposure rate measurements taken inside the horizontal beam ports, after the removal of the plugs, were up to 9.5 R/hr.

  A graphite cylinder was found past the shutter in horizontal beam port No. 2. It had an exposure rate of 368 mR/hr while inside the beam port. After removal, the graphite cylinder had an exposure rate of 20 mR/hr. A second graphite cylinder was found with an exposure rate of 768 mR/hr. This second graphite cylinder could not be removed.

- **Miscellaneous Equipment on Reactor Top -** Removable and direct contamination was found on the miscellaneous equipment on the reactor top. Measurement show direct $\beta\gamma$ contamination up to 28,410 dpm/100cm^2 and removable α contamination up to 70 dpm/100cm^2.

- **Storage Holes -** Direct $\beta\gamma$ contamination was found in storage holes No. 1 and 3. Direct $\beta\gamma$ measurements were up to 13,942 dpm/100cm^2 in storage hole No.1 and 116,766 dpm/100cm^2 in storage hole No. 3. Removable α contamination, up to 46 dpm/100cm^2, was found in No. 2.

- **Vertical Beam Ports -** Removable contamination was found in the vertical beam ports up to 765 dpm/100cm^2 for α contamination and 17,675 dpm/100cm^2 for $\beta\gamma$ contamination. Exposure rate measurements taken inside the vertical beam ports, after the removal of the plugs, were up to 4.7 R/hr. As expected, the exposure rate increased as each measurement was taken every 12 inches into the port.

Table 10.5 shows the range of activity on the reactor block. Figure 10.6 shows the maximum direct and removable activity per area. Attachment H provides more extensive data per area.

Table 10.5 Reactor Block Survey Data Summary

Survey Location	No. of Survey Points	Direct Alpha	Direct MDA	Direct Beta-Gamma	Direct MDA	Removable Alpha	Removable MDA	Removable Beta-Gamma	Removable MDA
		Range of Activity (dpm/100 cm^2)							
Horizontal Beam Ports	41	None Taken	-	None Taken	-	0 to 3429	13	1 to 10614	17
Miscellaneous Eqpmt-Reactor Top	30	-44 to 26	116	-8169 to 28410	2628	0 to 70	13	0 to 461	17
Storage Holes on Reactor Top	15	-47 to 28	123	-5417 to 116766	1805	0 to 46	13	0 to 456	17
Vertical Beam Ports	43	None Taken	-	None Taken	-	0 to 765	13	0 to 17675	17

Shading represent areas where activities are above acceptable limits in
U.S. Nuclear Regulatory Guide 1.86, "Termination for Operating License for Nuclear Reactors," June 1974.

∗ Direct measurements in dpm/100 cm^2
❍ Removable measurements in dpm/100 cm^2
Bold numbers represent activities **above** U.S. NRC Regulatory Guide 1.86 limits.
All measurements were taken **INSIDE** the facility.

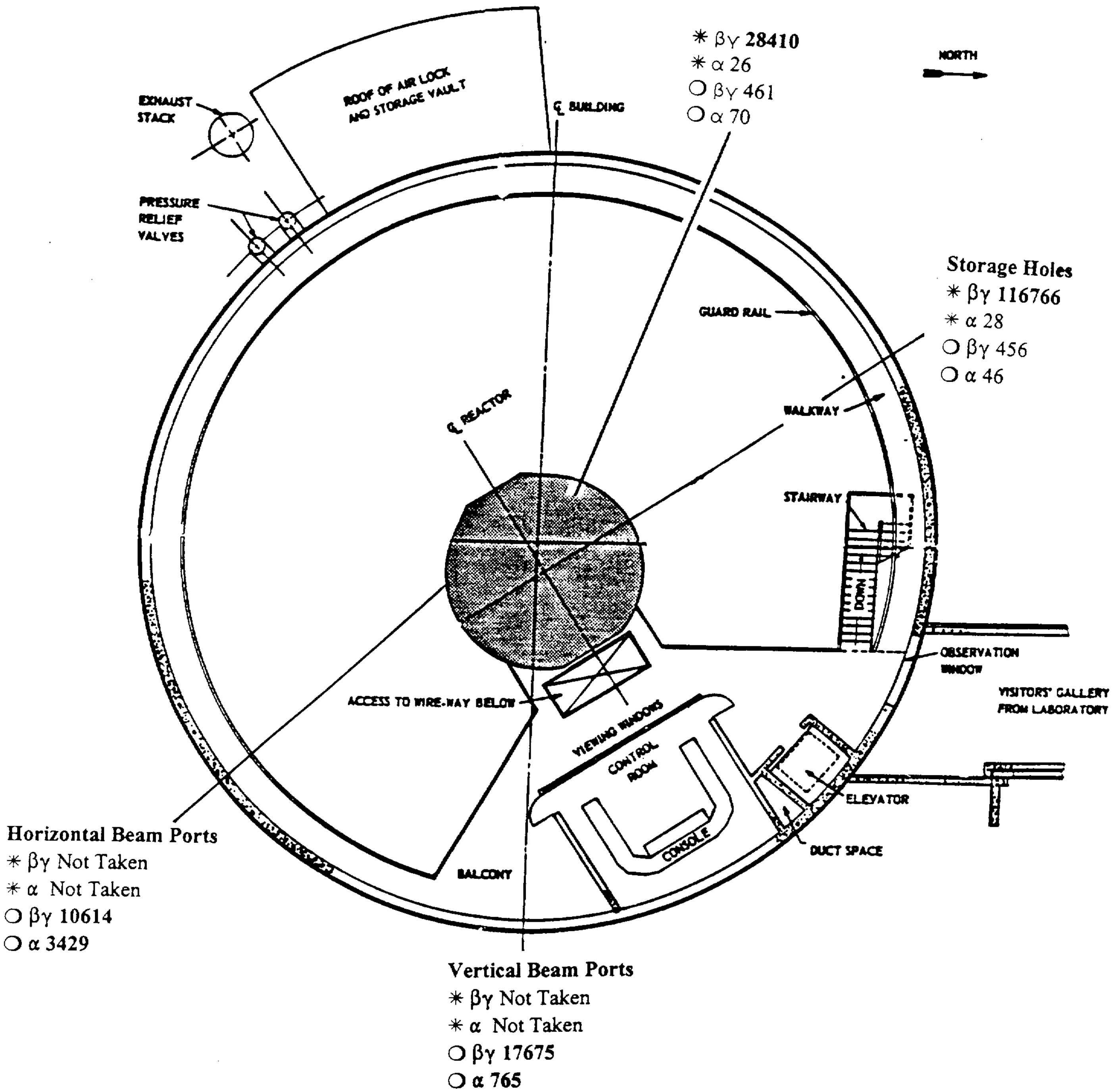


Figure 10.2 Maximum Direct and Removable Activity on the Reactor

10.2.2 Assessment of Radioactivity in the Reactor Core and Biological Shield

For these assessments, information was obtained from "Activation Products in the Biological Shield of the Georgia Tech Research Reactor" (Reference 2). Dose rate measurements were obtained using calibrated pocket ionization chambers at the center of the core tank. The average measurement was 6.7 R/hr. A photon spectrum was developed by obtaining a measurement over an open vertical port at the center of the core tank using a portable GeLi scintillation detector. The principal isotopes identified were Co-60 and Zn-65.

To quantify the amount, and assess the inventory and location of residual radioactivity in the reactor internals, and the biological shield, a three step series of computer codes, Scale 4.3 (Reference 2), was used. First, models were developed for the radial direction, top portion and bottom portion of the reactor. From these models, three energy groups of neutron flux spectra through the shield were calculated using the CSASI and XSDRNPM modules of the Scale 4.3 computer code. Second, from this flux spectra and operational history of the reactor, activation products for each layer of the biological shield were calculated using the COUPLE and ORIGEN-S modules. Finally, a dose rate at the center of the core tank was calculated from the photon spectrum of the activation products, using the QADS module.

Specific material compositions were not documented for the biological shield, therefore, solving for activation products proceeded as an iterative process. This process was accomplished by adjusting the input material compositions and comparing the calculated dose rates to the measured dose rates at the center of the core tank.

Results of the iterations revealed the two predominate isotopes located in the biological shield were Co-60 and Zn-65 which agree with the preliminary photon spectrum measured. Using the maximum results from the computer codes, which agree with the measured data, as a worst case for radioactivity present, 6.10 Curies of Co-60 and 7.03 Curies of Zn-65 were calculated for the biological shield. Most of the activity is located in the core tank, graphite retaining sleeve and stainless steel/ boron layers of the shield. The remainder, approximately 100 microcuries of Co-60, is located in the steel and concrete regions.

10.2.3 Neutron Activation Products Calculations-Spherical Shell Model

It was assumed that a spherical shell model would adequately describe the neutron activation concentrations present in the biological shield and embedments of the GTRR. This assumption was based on data from the decommissioning of the University of California, Los Angeles (UCLA) ARGONAUT Reactor and the University of Washington Research Reactor (Reference 1).

A sample of concrete was obtained to determine the radiological activity present in the concrete of the biological shield. The samples was obtained using an abrasive saw which sectioned the horizontal beam port plug H-12. The sample was analyzed by gamma spectroscopy. Efforts to correlate results of the activity present in the GTRR using the

spherical shell model failed. Based on results of the UCLA and University of Washington characterizations and a comparison of their operational histories to the GTRR, the radial boundary of the activated layer of the biological shield should have been less than 60 inches from the center of the core. However, the GTRR was constructed without inner horizontal shield plugs which enabled sampling and provided measured data. The previous studies were used to establish the boundary of activation only (Reference 2).

10.2.4 Neutron Activation Product Calculations-TLDs

A set of direct measurements was made using thermoluminescent dosimeters (TLDs). TLDs were placed in the vertical access beam ports from the top of the reactor. Three "TLD trees" with eight TLDs each, at 12-inch intervals, were lowered into the core and exposed for approximately 24 hours (see Figure 10.7). The results are presented in Table 10.6.

The average dose rate at the core centerline was 4,670 mrem/hr. At a radius of 2 feet, the average dose rate was 5,134 mrem/hr. The average dose rate at a radius of 4 feet was 315 mrem/hr. Upon further investigation of the "Safety Analysis Report of the Georgia Tech Research Reactor" (Reference 2), it was found that the bottom of vertical port V-43 coincided with the radial core centerline instead of the bottom of the reactor core.

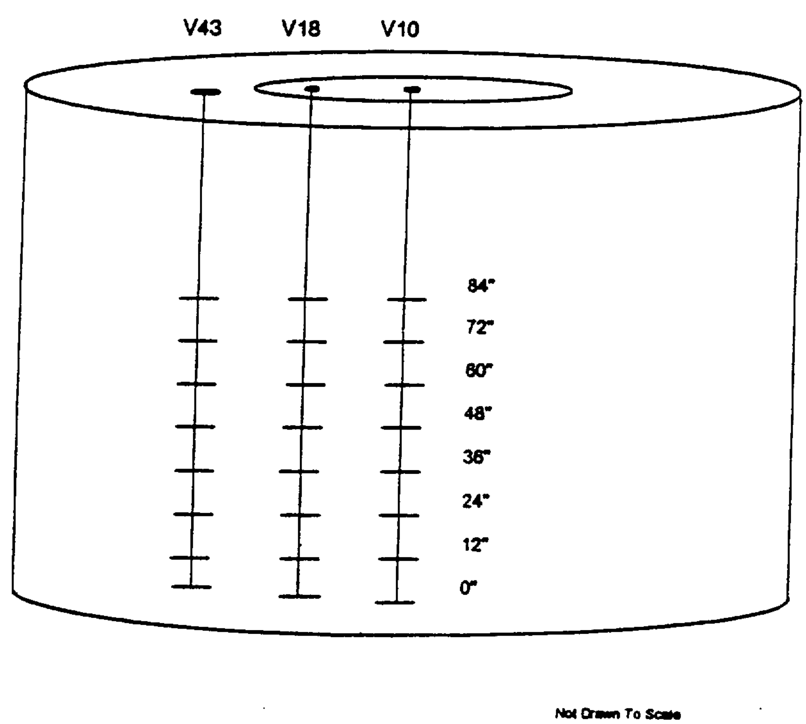
To estimate the boundary of activated concrete in the biological shield of the GTRR, the TLD dose rates were incorporated into two simple models of the reactor internals (see Attachment J). The distance was calculated from the center of the core to the point where the measured dose rate was below 200 mrem/hr in port V-43. This resulted in an activation boundary radius of approximately 68 inches from the center of the core. An activated concrete volume was calculated by subtracting the core volume from the volume of activation. Assuming the core was a sphere of radius 60 inches resulted in a volume of 235 ft^3 of activated concrete. To confirm this volume, the core was modeled as a cylinder which calculated a volume of 229 ft^3 of activated concrete.

For clarity, the 224 mrem/hr at port V-43 is simply used as a factor to complete the calculation. It does not represent the residual activity of the concrete to be removed from the biological shield during decommissioning. Hence, the calculated volume of 235 ft^3 of activated concrete is expected and will be profiled and disposed of as low-level radioactive waste.

Table 10.6 Dose Equivalent of TLDs Placed in Vertical Beam Ports

	R=0'	R=2'	R=4'
	Center of Reactor		Fuel Storage Hole
	V10	V18	V43
Distance From Bottom of Reactor Core	**Dose Equivalent (mRem/hr)**	**Dose Equivalent (mRem/hr)**	**Dose Equivalent (mRem/hr)**
84"	3632	5879	4
72"	3835	3913	9
60"	3305	3012	19
48"	3389	3499	190.
36"	3785	3773	224
24"	5757	5266	365
12"	5218	7268	579
0"	8438	8459	1131
Average	4670	5134	315

Figure 10.7 Representation of TLD Placement in Reactor Core



10.3 SOIL SAMPLE RESULTS

~~Background soil~~ samples from the vacant lot behind 763 Techwood Drive showed traces of K-40, Kr-85, Th-232 and U-238. Soil samples taken in the facility yard were analyzed for radioactivity concentrations. Table 10.7 and 10.8 shows the radioactive concentrations in the background soil samples and in the facility yard soil samples. Attachment N-Soil contains the results of the soil sample analysis.

10.4 LIQUID SAMPLE RESULTS

Liquid samples were obtained throughout the facility. Table 10.9 shows the results of the liquid samples taken around the reactor building. Attachment N-Liquid contains the results of the liquid sample analysis.

10.5 CONCRETE SAMPLE RESULTS

Results of the concrete samples taken throughout the facility show the presence of Ag-112, K-40 and Co-60 in the pipe chase area. No other radionuclides were found in the laboratory analysis. Table 10.10 Lists the results of this analysis. Attachment N-Concrete contains the results of the concrete sample analysis.

10.6 MATERIALS SAMPLE RESULTS

Table 10.11 lists the results of the sampling analyses for material samples. The results of the radioactivity concentrations of the graphite block samples and the stack are also listed in Table 10.11. Table 10.12 also includes radionuclide concentrations of locations throughout the facility which showed elevated removable contamination. Attachment N-Material contains the results of the material sample analysis. Attachment N-Elevated Smears contains the results of analysis of locations where removable contamination was found.

Attachments K and L show the results of the ACM and lead in paint screening results, respectively.

Table 10.7: Radioactivity Concentrations in Background Soil Samples

Sample ID	Location	Radionuclides	Activity
			(in pCi/g)
BG-1	Vacant Lot Behind 763 Techwood Drive	K-40	1.56×10^{4}
		Th-232	1.10×10^{3}
		U-238	7.07×10^{2}
BG-2	Vacant Lot Behind 763 Techwood Drive	K-40	9.32×10^{3}
		Th-232	5.11×10^{2}
		U-238	4.47×10^{2}
BG-3	Vacant Lot Behind 763 Techwood Drive	K-40	2.24×10^{1}
		Kr-85	8.21×10^{1}
		Th-232	1.32×10^{0}
		U-238	9.58×10^{-1}
BG-4	Vacant Lot Behind 763 Techwood Drive	K-40	2.19×10^{1}
		Th-232	1.39×10^{0}
		U-238	<BKG
BG-5	Vacant Lot Behind 763 Techwood Drive	K-40	1.9×10^{1}
		Th-232	1.08×10^{0}
		U-238	<BKG
BG1-5Comp	Vacant Lot Behind 763 Techwood Drive	K-40	1.98×10^{1}
		Th-232	1.08×10^{0}
		U-238	8.93×10^{-1}
BG-6A	Vacant Lot Behind 763 Techwood Drive	K-40	2.42×10^{1}
		Th-232	1.48×10^{0}
		U-238	1.39×10^{0}

Table 10.8: Radioactivity Concentrations in Outdoor Soil Samples

Sample ID	Location	Radionuclides	Activity
			(in pCi/g)
Run-off 1	Facility Yard	Th-232	1.18×10^{0}
		Cs-137	2.08×10^{-1}
		K-40	2.16×10^{1}
		U-238	<BKG
Run-off 2	Facility Yard	K-40	2.77×10^{1}
		U-238	<BKG
		Th-232	<BKG
SC-1	Facility Yard	Th-232	9.79×10^{-1}
		U-238	<BKG
SC-2	Facility Yard	Th-232	1.03×10^{0}
		K-40	2.33×10^{1}
		U-238	<BKG
SC-3	Facility Yard	K-40	1.81×10^{1}
		U-238	<BKG
		Th-232	<BKG
SC-4	Facility Yard	Th-232	1.29×10^{0}
		K-40	2.61×10^{1}
		Cs-137	2.43×10^{-1}
		U-238	<BKG
SC-5	Facility Yard	K-40	2.76×10^{1}
		U-238	<BKG
		Th-232	<BKG
SC-6	Facility Yard	U-238	1.09×10^{0}
		Th-232	1.15×10^{0}
		Cs-137	2.0×10^{-1}
		K-40	1.89×10^{1}
SC-7	Facility Yard	U-238	1.88×10^{0}
		Th-232	1.81×10^{0}
		K-40	2.26×10^{1}
SC-8	Facility Yard	Th-232	1.33×10^{0}
		K-40	2.48×10^{1}
		U-238	<BKG
SC-9	Facility Yard	U-238	2.92×10^{0}
		Th-232	1.82×10^{0}
		K-40	2.83×10^{1}

Table 10.8: Radioactivity Concentrations in Outdoor Soil Samples (continued)

Sample ID	Location	Radionuclides	Activity (in pCi/g)
SC-10	Facility Yard	Th-232	1.2×10^{0}
		K-40	2.34×10^{1}
		U-238	<BKG
SC-11	Facility Yard	U-238	<BKG
		Th-232	<BKG
		Cs-137	4.72×10^{-1}
		K-40	1.51×10^{1}
SC-12	Facility Yard	Th-232	1.69×10^{0}
		Cs-137	1.74×10^{-1}
		U-238	<BKG
SC-13	Facility Yard	Th-232	1.46×10^{0}
		K-40	2.87×10^{1}
		U-238	<BKG
SC-14	Facility Yard	U-238	7.17×10^{-1}
		K-40	3.14×10^{1}
		Th-232	1.91×10^{0}
SC-15	Facility Yard	Th-232	1.91×10^{0}
		K-40	2.57×10^{1}
		U-238	<BKG
SC-16	Facility Yard	U-238	1.64×10^{0}
		Th-232	1.33×10^{0}
		K-40	2.62×10^{1}

Table 10.9: Radioactivity Concentrations in Liquid Samples

Sample ID	Location	Radionuclides	Activity
			(in pCi/l)
LIQ-1	Low-Level Waste Tank # 1	K-40	1.02×10^{1}
LIQ-2	Low-Level Waste Tank # 2	K-40	1.02×10^{1}
LIQ-3	Low-Level Waste Tank # 1 Sludge	Cs-137	5.07×10^{-1}
		Co-60	9.32×10^{-1}
		K-40	1.00×10^{1}
LIQ-4	Low-Level Waste Tank # 2 Sludge	Cs-137	2.29×10^{-1}
		Co-60	5.74×10^{-1}
		K-40	9.58×10^{0}
LIQ-5	Suspect Waste Tank	K-40	1.10×10^{1}
LIQ-6	Suspect Waste Tank Sludge	Cs-137	5.55×10^{0}
		Co-60	1.25×10^{2}
		K-40	6.74×10^{0}
LIQ-7	Ground Floor of Containment Building	K-40	1.18×10^{1}
LIQ-8	Bismuth Tank TB-1	K-40	1.28×10^{1}
		Co-60	4.72×10^{0}
LIQ-9	HXD-1 Tank- Process Equipment Room	K-40	1.11×10^{1}
	(Secondary Side 1)		
LIQ-10	HXD-2 Tank- Process Equipment Room	K-40	1.12×10^{1}
	(Secondary Side 2)		
LIQ-11	Compressor Oil	K-40	1.25×10^{1}
LIQ-12	Bismuth Shield Block Coolant	Co-60	9.29×10^{0}
	Liquid in Plastic Sheet Overhang	K-40	2.28×10^{1}
LIQ-13	ZnBr-Biomedical Irradiation Facility	K-40	5.57×10^{0}
LIQ-14	Water Filled Storage Hole -First Floor	Cs-137	1.40×10^{4}
		Co-60	1.32×10^{4}
		K-40	1.31×10^{4}
LIQ-15	Oil fron MD-2A Tank- Process Equipment Ro	Kr-85	7.83×10^{1}
		Co-60	9.33×10^{0}
		K-40	1.38×10^{1}
		Th-232	<BKG
LIQ-16	TD-1 Tank- Process Equipment Room	Co-60	8.94×10^{2}
		K-40	1.16×10^{4}
		Th-232	<BKG
LIQ-17	Liquid Drain Basin Outside of SE	K-40	1.39×10^{1}
	Corner of Containment Building		
LIQ-18	Dalney Street Sewer	K-40	1.23×10^{1}
		Th-232	<BKG

Table 10.10: Radioactivity Concentrations in Concrete Samples

Sample ID	Location	Radionuclides	Activity (in pCi/g)
PT-1	Pipe Chase Tunnel	ND	-
PT-2	Pipe Chase Tunnel	Co-60	2.40×10^2
PT-3	Pipe Chase Tunnel	ND	-
BS-1	Bismuth Shield Leak Area	ND	-
BS-2	Bismuth Shield Leak Area	ND	-
BS-3	Bismuth Shield Leak Area	ND	-
BS-4	Bismuth Shield Leak Area	ND	-
BS-5	Bismuth Shield Leak Area	ND	-
BS-6	Bismuth Shield Leak Area	K-40	9.93×10^2
BS-7	Bismuth Shield Leak Area	ND	-
BS-8	Bismuth Shield Leak Area	ND	-

ND= Radionuclides in Library Were Not Detected

Table 10.11 Radioactivity Concentrations in Material Samples

Sample ID	Location	Radionuclides	Activity
			(in pCi/g)
MAT-1	Graphite from Horizontal Beam Port 2 (H-2)	Eu-152	1.19×10^{-1}
		Co-60	3.49×10^{0}
		K-40	1.22×10^{1}
MAT-2	Metal and Cement Stampings from H-12	Sn-126	2.73×10^{1}
		Eu-152	7.13×10^{2}
		Co-60	8.65×10^{3}
		K-40	4.65×10^{1}
MAT-3	Dust and Sediment from H-12	Co-60	1.2×10^{4}
MAT-4	Dust and Sediment from H-1	ND	
MAT-5	Stack Residue	K-40	1.26×10^{3}
MAT-6	Scale from Vertical Beam Port No. 43 (V-43)	Co-60	1.72×10^{3}
MAT-7	Chemical Addition Tank No. 1	K-40	9.94×10^{2}
MAT-8	Chemical Addition Tank No. 2	Co-60	1.69×10^{4}
MAT-9	Oil from MH-3	ND	
MAT-10	Drain Stop in Biomedical Irradiation Room	ND	

ND= Radionuclides in Library Were Not Detected

Table 10.12 Radioactivity Concentrations in Locations with Elevated Removable Activity

Sample ID	Location	Radionuclides	Activity (in pCi/g)
911	Graphite cylinder in Plug Storage No. 2	Co-60	2.18×10^2
		K-40	9.94×10^2
912	Inside Plug Storage Hole No. 2	Co-60	2.34×10^3
921	Graphite cylinder in Plug Storage No. 4	Co-60	1.32×10^3
923	Metal Rod in Plug Storage Hole No. 4	Co-60	3.64×10^2
		K-40	9.6×10^2
988	Graphite cylinder in Plug Storage No. 6	Co-60	3.37×10^2
1012	Graphite cylinder in Plug Storage No. 8	Co-60	3.67×10^2
1302	Inside Horizontal Port No. 12 (NW side)	Co-60	6.88×10^2
1309	Inside Horizontal Port No. 11 (NW side)	Co-60	2.26×10^2
1454	Inside Vertical Port No. 40	K-40	1.06×10^3
1504	Inside Horizontal Port No. 8	Co-60	1.08×10^3
		K-40	1.16×10^3
1516	Inside Horizontal Port No. 11 (E side)	Zn-65	2.44×10^2
		Co-60	5.15×10^2
		K-40	1.09×10^3

10.7 AIR SAMPLE RESULTS

Air sampling was performed prior to characterization activities. Results of these air samples indicate the presence of Rn-222. Samples were recounted approximately 24 hours later to verify Rn-222. Air sampling was also performed during the opening of the storage plugs, horizontal beam ports, the vertical beam ports, and during beam port plug sectioning. Air samples taken during characterization activities that were expected to generate airborne radioactivity were all less than the allowable maximum permissible levels of 3×10^{-11} μCi/ml for alpha emitters and 9×10^{-9} μCi/ml for beta-gamma emitters. Air sample results are found in Attachment M.

10.8 TRITIUM ANALYSIS RESULTS

Tritium analysis performed throughout the facility showed few areas where tritium contamination was present. The horizontal and vertical beam ports had contamination up to 2,512 dpm/100cm^2. Tank He-1 in the Process Equipment Room showed contamination on its surface up to 5,446 dpm/100cm^2. Outside of the restricted areas tritium contamination above release criteria was not found.

Tables 10.13 and 10.14 show survey locations and samples where tritium contamination above release criteria was found. Tritium results are found in Attachment N-Tritium.

Table 10.13 Tritium Survey Results

Action level (DPM): 1000 dpm/100 cm^2

Tritium Sample Number	Corresponding Smear Number	Location	Tritium Activity (dpm/100 cm^2)
6	1117	Process Equipment Room- Tank He-1	5446
7	1118	Process Equipment Room- Tank He-1	2215
89	214	Horizontal Beam Port No.12 (North)	5282
203	1302	Inside Horizontal Beam Port No.12 (North)	1842
204	1309	Inside Horizontal Beam Port No.11 (North)	1732
205	1310	Inside Horizontal Beam Port No.11 (North)	1062
329	1516	Inside Horizontal Beam Port No.11 (South)	2152

Table 10.14 Tritium Survey Analysis of Material and Liquid Samples

Tritium Sample Number	Tritium Activity (dpm)	Tritium Concentration (pCi/l or pCi/g)
LIQ-1	56	2,293
LIQ-2	33	1,351
LIQ-3	58	2,375
LIQ-4	33	1,351
LIQ-5	38	1,556
LIQ-6	33	1,351
LIQ-7	45	1,843
LIQ-8	2,311	94,636
LIQ-9	1	41
LIQ-10	2	82
LIQ-11	1,455	59,582
LIQ-12	6,057	248,034
LIQ-13	72	2,948
LIQ-14	456,012	18,673,710
LIQ-15	514	21,048
LIQ-16	22,332	914,496
LIQ-17	3	123
LIQ-18	<LLD	-
Biomedical Irradiation Facility	112	4,586
PT-1	1,664	68,141
PT-2	766	31,368
PT-3	67	2,744
BS-1	48	1,966
BS-2	39	1,597
BS-3	90	3,686
BS-4	77	3,153
BS-5	44	1,802
BS-6	74	3,030
BS-7	79	3,235
BS-8	45	1,843

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