the IT group℠

FINAL SURVEY PLAN

for the
Georgia Institute of Technology
Research Reactor Decommissioning Project

GSFIC Project I-47

Revision 3
January 2001

Client Approvals:

[signature] 2/06/01

Executive Engineer Date

[signature] 6 Feb 01

Technical Safety Review Committee Date

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A Member of The IT Group

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IT Project No. 784357

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Document Number GT-PL-FS-001
IT Project No. 784357

Revision 3
Date: January 2001

GTS Duratek / IT Corporation	January 2001
Prepared By	Date
	1/30/2001
Radiation Safety / ALARA	Date
	1/30/2001
Health & Safety	Date
	01/30/2001
Engineering	Date
	01/30/2001
Manager Approval	Date

FINAL SURVEY PLAN

for the

GEORGIA INSTITUTE OF TECHNOLOGY RESEARCH REACTOR FACILITY NRC LICENSE NUMBER R-97

REVISION 3
January 2001

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PREFACE

In July of 1999, GTS Duratek was awarded a subcontract to assist the IT Corporation (IT) to decontaminate and decommission (D&D) the Georgia Institute of Technology (Ga Tech) Research Reactor (GTRR). This assistance includes the design and performance of the Final Survey. This Final Survey Plan describes the methodology and instrumentation that will be used to perform Final Surveys at the GTRR facility.

IT Corporation is under contract to the Georgia State Financing and Investment Commission (GSFIC) for decommissioning of the GTRR facility and is the contracted organization responsible for performance of the Final Survey. GTS Duratek will execute the Final Survey under subcontract to IT with oversight and support from IT Corporation.

This Final Survey Plan includes the methods and requirements necessary to verify that decontamination efforts of the GTRR facility were successful and to document that residual contamination has been reduced to the extent necessary to achieve NRC license termination and unrestricted release of the facility.

As provided for in 10CFR20.1401 (b) (3), the Decommissioning Plan for the GTRR was submitted for NRC approval prior to August 20, 1998, and approved by the Nuclear Regulatory Commission (NRC) before August 20, 1999. This allows for Final Survey results to be compared to the criteria specified in NRC Regulatory Guide 1.86, *Termination of Operating Licenses for Nuclear Reactors*. Final Surveys will be conducted using guidance contained in NUREG/CR-5849, *Manual for Conducting Surveys in Support of License Termination,* and in accordance with the NRC approved Georgia Institute of Technology Research Reactor Decommissioning Plan.

Decommissioning Plans submitted after August 20, 1998 and approved after August 20, 1999 would be subject to the survey requirements contained in NUREG-1575, *Multi Agency Radiation Survey and Site Investigation Manual (MARSSIM),* and the dose based release criteria contained in 10CFR20 Subpart E (25 mrem/yr). The GTRR Decommissioning Plan fell within the transition period, and was "grandfathered" under the older guidelines contained in NUREG/CR-5849 and NRC Regulatory Guide 1.86.

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1.0 SITE HISTORY AND BACKGROUND

1.1 Introduction

The GTRR began operation in 1964 as a research facility where engineering students would train to operate commercial nuclear power plants 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 campus is located in a residential and commercial area just north of the center of downtown Atlanta.

The GTRR is a part of the Georgia Institute of Technology - Neely Nuclear Research Center (NNRC) in the city of Atlanta, Georgia. The NNRC is comprised of the GTRR located in the containment building, the hot cell, support laboratories and offices. The GTRR is operated under an NRC License, Number R-97. The remainder of the facilities operate under a State of Georgia Department of Natural Resources (DNR) License Number GA 147-1.

The GTRR decommissioning began in November, 1999. The end result of the decommissioning will be the termination of the NRC License Number R-97, and release for unrestricted use, in accordance with the NRC approved Georgia Institute of Technology Research Reactor Decommissioning Plan. This decommissioning involved the dismantlement and removal of all equipment and structural media within the NRC licensed area that exceeded the site specific release guidelines provided in Section 2.3. This Final Survey does not include those facilities operating under the State of Georgia DNR License.

1.2 Containment Building

The reactor containment building consists of three floors. The building is a cylindrical steel tank with a diameter of 82 feet and a 12-inch-thick concrete wall on the inside of the steel tank. The roof is a torispherical dome, approximately 50 feet above ground level, that provided a leak tight barrier to the escape of gas from the interior.

The Containment Building and Reactor Yard are shown in Figure 1.1. Immediately surrounding most of the containment building is the reactor yard. The reactor yard consists of both paved and open land areas.

1.2.1 Basement

The Basement of the containment building housed the reactor process equipment, the exhaust ventilation equipment and the electrical load center for the building. The Basement is accessed by elevator or stairways from the first floor of the containment building. The reactor process equipment was located within an area surrounded by 2-foot thick concrete walls. The filter bank and blower for the containment building exhaust are adjacent to this shielded area. A large holdup volume through which this exhaust must pass is cast into the Basement slab. Another large void in this concrete slab provides an expansion chamber which was connected to the helium/nitrogen space above the reactor core through a rupture disk.

The Basement also contained the service air compressor room, two rooms that contained the pneumatic tube support systems and an experimental area, and a cooling system for the bismuth blocks. The Basement plan is shown in Figure 1.2.

1.2.2 First Floor

The first floor was largely unoccupied except for the reactor near the center. Three removable floor patches permit large and heavy objects to be moved between the basement and the first floor using the cranes.

The first floor also contained a vertical water-filled fuel storage hole which was approximately 20 feet long extending to the Basement. Set into the outside wall of the containment building were 18 horizontal storage tubes.

The containment building is entered at the first floor level by one of three routes: the main air lock leading from the laboratory building; a smaller personnel air lock leading from the outside yard; and a truck door. A shielded irradiation room (the biomedical irradiation facility), is located on the first floor immediately adjacent to the reactor. Figure 1.3 shows the first floor plan.

1.2.3 Second Floor

The top of the reactor was 15 feet above the first floor. At this level, shown in Figure 1.4, there is a walkway which runs completely around the circumference of the building. At one point, this walkway is enlarged to form the second floor which connected with the top of the reactor. The reactor control room occupies much of the second floor. An Emergency Core Cooling System (ECCS) tank was also located on the second floor. Air was supplied to the containment building by an air-conditioning unit which was located on the roof of the control room. The second floor may be reached using the elevator or stairs.

Figure 1.1
Reactor Yard

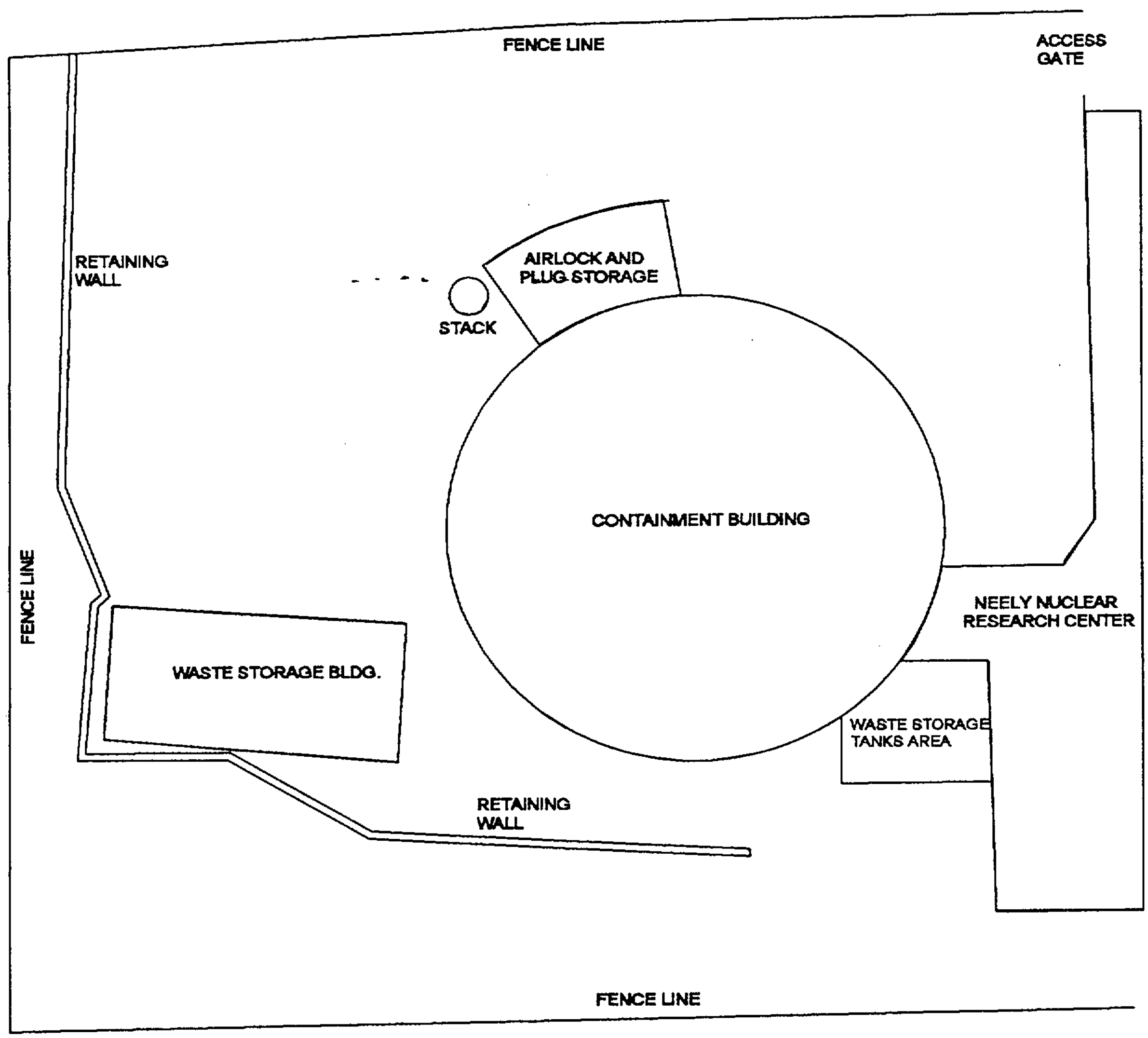


Figure 1.2
Containment Building Basement

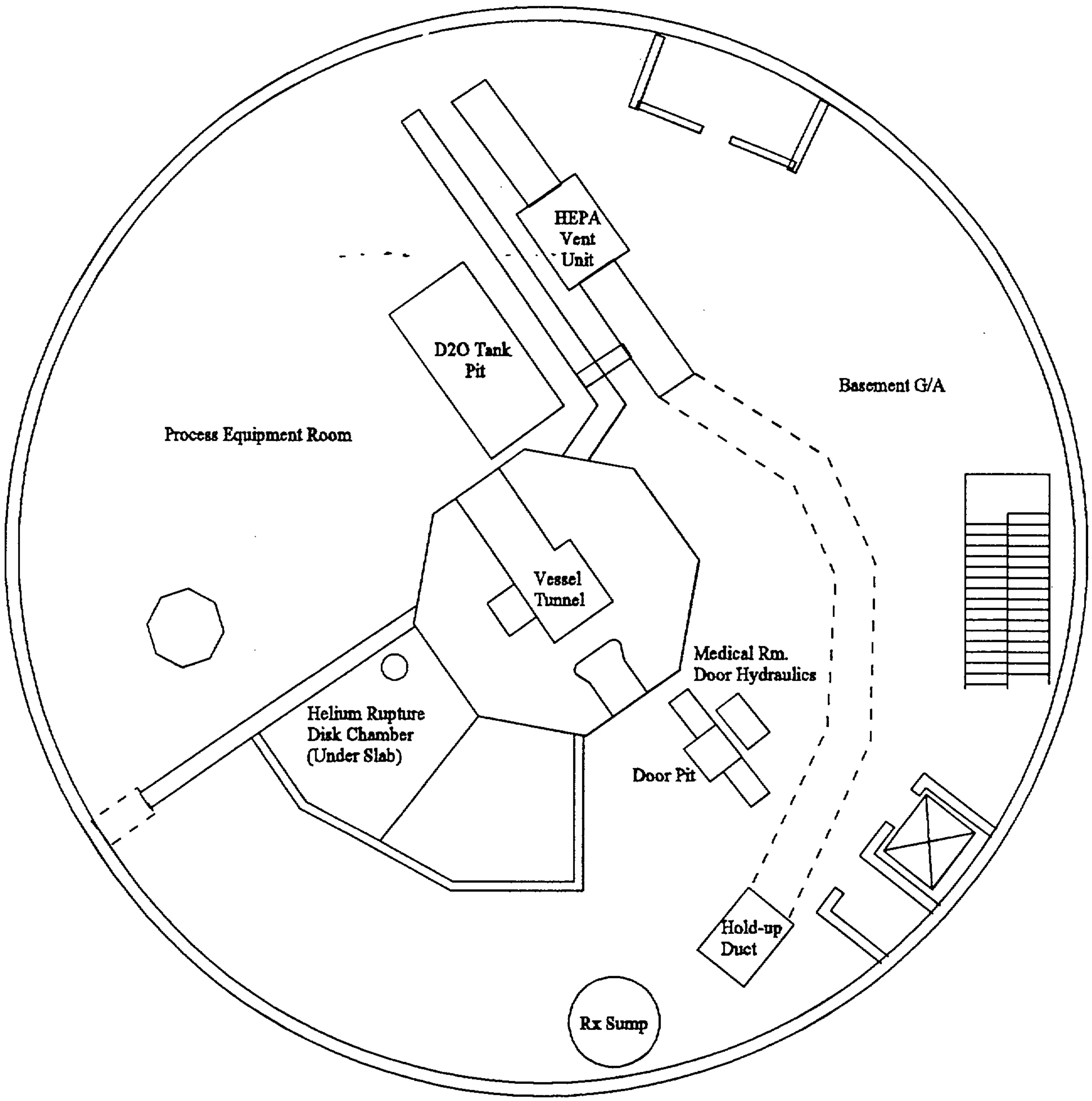


Figure 1.3
Containment Building First Floor

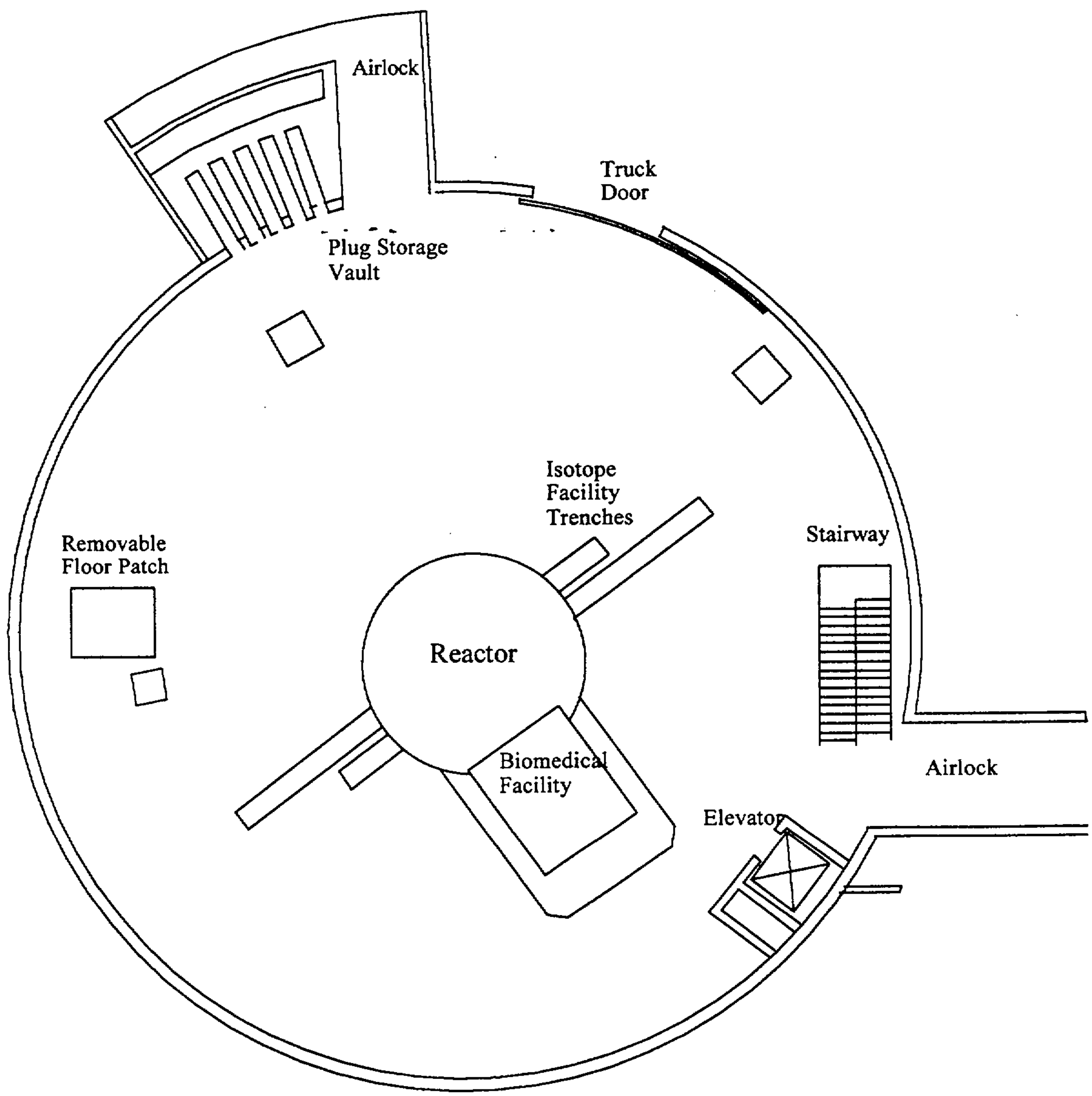
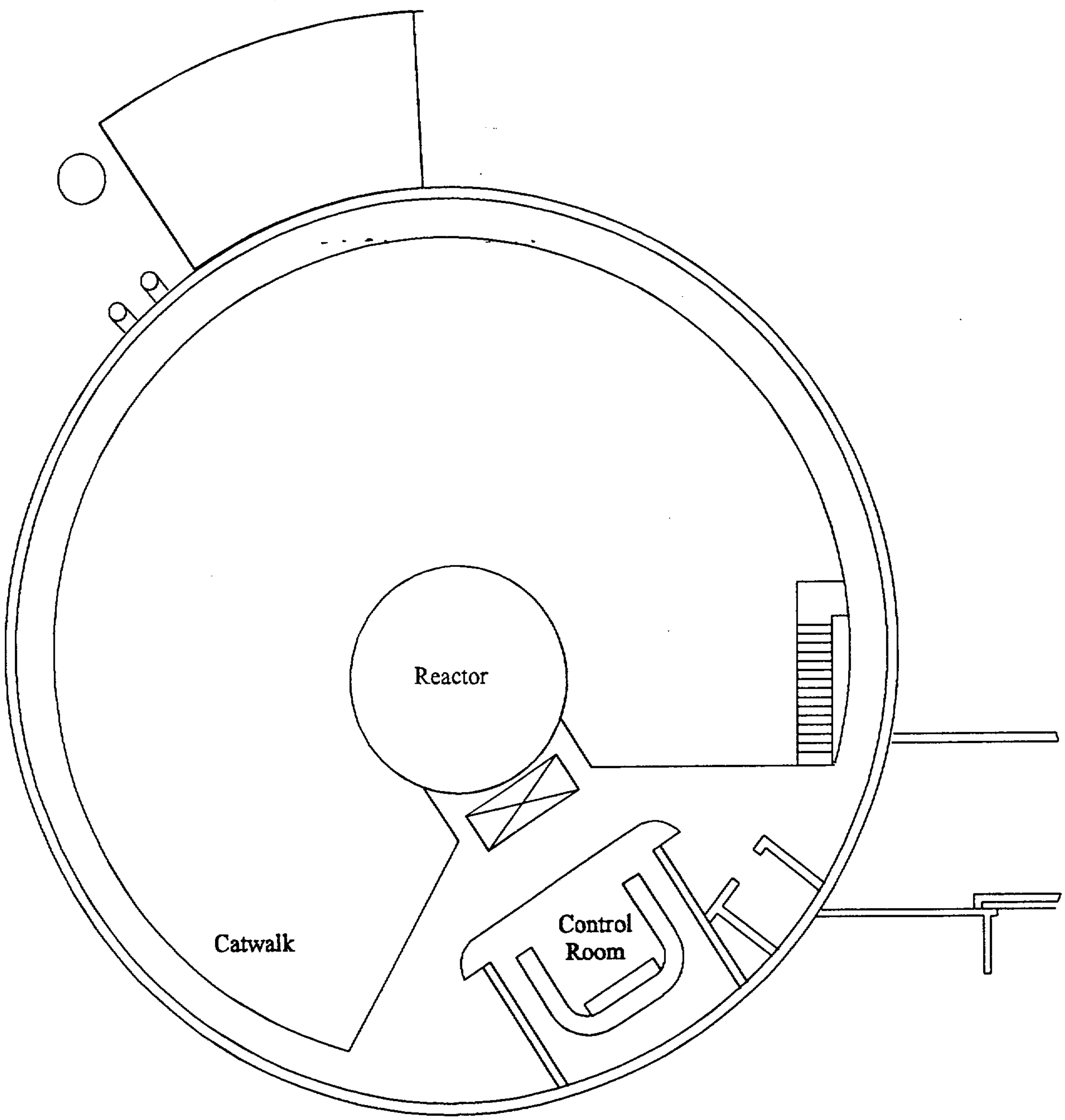


Figure 1.4
Containment Building Second Floor



2.0 SURVEY OVERVIEW

2.1 Survey Objective

The objective of the Final Survey is to verify that all remaining building surfaces, including permanent structures and any remaining equipment satisfy the criteria for release for unrestricted use.

This Final Survey is designed to verify that decontamination efforts were successful and document that residual contamination has been reduced to the extent necessary to achieve license termination and unrestricted release of the facility. Measuring and sampling techniques used for the Final Survey will be performed based on the guidance in NUREG/CR-5849.

2.2 Identity of Contaminants

Process history and characterization data, including sample analysis, demonstrated the presence of fission and activation products including Co-60, Zn-65, Cs-137, Eu-152, Eu-154, H-3, Fe-55, and C-14. Complete results of 10 CFR Part 61 analyses of samples from the GTRR, including a list of all identified fission and activation products, are provided in Attachment A.

2.3 Criteria for Release for Unrestricted Use for Building Surfaces

Site Specific Guideline Values, SPGLs, have been derived for use in demonstrating that the Georgia Tech Research Reactor meets the criteria for release for unrestricted use. These SPGLs are based on the radionuclide specific guideline values contained in Regulatory Guide 1.86 except for H-3 and Fe-55. The guideline values for H-3 and Fe-55 have been increased based on instructions from the Executive Engineer for this project (Reference 7.4). The increase in guideline values for H-3 and Fe-55 is consistent with previous NRC policy contained in NRC Policy Memorandum SECY-94-145. Although higher than the NRC Regulatory Guide 1.86 limits, these increased guideline values are orders of magnitude lower than the current guideline values for H-3 and Fe-55 specified in the NRC License Termination Rule. Table 2.1 lists the radionuclide specific guidelines used in deriving the SPGLs. With the exception of footnote "f", the footnotes to Table 2.1 are identical to those associated with Table 1 of Regulatory Guide 1.86.

Table 2.1
Radionuclide Specific Guideline Values

Nuclide[a]	Average[bc]	Maximum[bd]	Removable[be]
U-Nat, U-235, U-238, and associated decay products	5,000 dpmα/100 cm^2	15,000 dpmα/100 cm^2	1,000 dpmα/100 cm^2
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100 dpm/100 cm^2	300 dpm/100 cm^2	20 dpm/100 cm^2
Th-Nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1,000 dpm/100 cm^2	3,000 dpm/100 cm^2	200 dpm/100 cm^2
H-3 and Fe-55[f]	**200,000 dpm/100 cm^2**	**600,000 dpm/100 cm^2**	**10,000 dpm/100 cm^2**
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others mentioned above	5,000 dpmβγ/100 cm^2	15,000 dpmβγ/100 cm^2	1,000 dpmβγ/100 cm^2

[a] *Where surface contamination by both alpha and beta-gamma emitting nuclides exists, the limits established for alpha and beta-gamma emitting nuclides should apply independently.*
[b] *As used in this table, dpm (disintegrations per minute) means the rate of emissions by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.*
[c] *Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each object.*
[d] *The maximum contamination level applies to an area of not more than 100 cm^2.*
[e] *The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.*
[f] *Letter from Executive Engineer to IT Corporation dated 4/18/00 providing alternate H-3 and Fe-55 guidelines for the Ga Tech Research Reactor Decommissioning (Reference 7.4). Removable contamination to be assessed as described in Footnote e, except using a wet smear.*

2.3.1 Site Specific Guideline Values, SPGLs

The site specific guideline values for demonstrating that the Georgia Tech Research Reactor meets the criteria for release of unrestricted use were derived based on guidance contained in NUREG/CR-5849, the radionuclide specific guideline values listed in Table 2.1, and the actual radionuclide mix as determined by sample analysis. The SPGLs were calculated using the following formula:

$$SPGL = \frac{F}{\sum_{i=1}^{n} \frac{f_i}{GL_i}}$$

Where: F = the fraction of the assumed radionuclide mix that is considered detectable, unitless.

f_i = the relative fraction of radionuclide i, unitless.

GL_i = the radionuclide specific guideline value for radionuclide i taken from Table 2.1, dpm/100 cm^2.

n = the number of radionuclides present in the radionuclide mix.

Site specific guideline values for total beta activity and removable beta activity were calculated according to the above formula using the sample analysis results from three samples collected within the Georgia Tech Research Reactor Containment Building. Attachment B contains the spreadsheets used to perform these calculations. The results of these calculations are summarized in Table 2.2

Table 2.2
Site Specific Guideline Values
for Total Beta Activity and Removable Beta Activity

Sample Description	**SPGL (dpm/100 cm^2) Total Beta Activity**	**SPGL (dpm/100 cm^2) Removable Beta Activity**
GEL-1, Concrete Sample From Under Reactor Vessel	1,500	104
GEL-2, Concrete Sample From Pit Beneath Door To Biomedical Room	3,800	688
GEL-3, Primary Coolant Resin Sample	2,000	148
AVERAGE	**2,400**	**313**

In order to demonstrate compliance with the criteria for release for unrestricted use, the three sample SPGL's listed in Table 2.2 were averaged for a site SPGL . In order to validate the H-3 concentration, and its contribution to the radionuclide mix, a number of H-3 smears will be collected from each survey area, as described in Section 4.6.4.

2.4 Criteria for Release for Unrestricted Use of Open Land Areas

Under Subpart E, of 10 CFR 20.1402, NRC licenses may be terminated and a site released for unrestricted use if the residual radioactivity that is distinguishable from background results in a total effective dose equivalent, TEDE, that does not exceed 25 mrem/yr and the residual radioactivity has been reduced to levels that are as low as reasonably achievable, ALARA. NRC has published (Federal Register Vol. 64 page 68396, December 7, 1999) screening values for selected radionuclides in surface soils. These concentrations would be deemed in compliance with the 25 mrem/yr unrestricted release dose limit in 10 CFR 20.1402.

Table 2.3 lists screening values for the GTRR radionuclides of interest. These values were either published by the NRC or calculated using NRC methodology. These screening values will be used as the site specific guideline values for open land areas. Surface soil samples (0-6 inches) will be collected from open land areas within the reactor yard and compared to the criteria listed in Table 2.3 using the sum of fractions rule. These soil samples will be used to confirm the results of the characterization survey and demonstrate that the open land areas within the reactor yard fence meet the criteria for release for unrestricted use and that the levels are ALARA.

Table 2.3
Radionuclide Specific Soil GuidelineValues

Radionuclide	Guideline Values (pCi/g)
H-3	110
Fe-55	10,000
Pu-239/240	2.3
U-233/234	13.0
U-238	14.0
Ni-59	5,500
Cs-134	5.7
Cs-137	11.0
Co-60	3.8
Eu-152	8.7
Eu-154	8.0
Mn-54	15.0
Ag-110m	3.9
Zn-65	6.2
Sr-90	1.7
C-14	12.0
Ni-63	2,100
Tc-99	19.0

2.5 Organization and Responsibilities

The GTS Duratek Survey Supervisor will report to the IT Project Manager, who in turn reports to the Executive Engineer technically, and the Georgia State Financing and Investment Commission (GSFIC) contractually.

The on-site project survey team will be supported by the full resources of GTS Duratek's Corporate Office. Additional oversight and support from GTS Duratek's professional engineering and quality assurance staff will be provided as necessary to the GTS Duratek Survey Supervisor to ensure successful project execution and completion. Independent oversight will be provided by the IT Quality Assurance Manager in accordance with the Quality Assurance Project Plan & Quality Assurance Procedures for the Georgia Institute of Technology Research Reactor Decommissioning Project.

Survey teams will consist of personnel trained, qualified, and experienced in field radiological survey procedures.

2.5.1 Survey Supervisor

The Survey Supervisor will be responsible for overseeing the performance of the survey and sampling activities, and will be qualified in the use of all survey instrumentation and all aspects of surveying as described in NUREG/CR-5849 and this Final Survey Plan.

2.5.2 Health Physics (HP) Technician

The HP Technicians will be responsible for performing surveys and sampling. They will be qualified in the use of survey instrumentation and surveying in accordance with NUREG/CR-5849 and this Final Survey Plan.

3.0 SURVEY INSTRUMENTATION

Selection and use of instrumentation will ensure sensitivities are sufficient to detect the identified nuclides at the minimum detection requirements. A list of the instruments, types of radiation detected, and their calibration sources is provided in Table 3.1.

The Ludlum Model 2350 Data Logger and various detectors will be used for direct surface and exposure rate measurements. The data logger is a micro-processor computer based counting instrument designed for use with a wide range of detectors. NaI(Tl) scintillation, gas-flow proportional detectors, and GM detectors will be used to obtain field measurements for Final Surveys.

Detector selection will depend on the survey to be performed, surface contour and survey area size. For direct measurements, the 126 cm^2 gas-flow proportional detector will be normally used. A 1" x 1"-NaI(Tl) gamma scintillation detector will be used for exposure rate measurements.

Specialized gas-flow proportional and GM pipe detectors will be used in conjunction with the Model 2350 to survey the interior surfaces of drains, piping, and penetrations remaining in the GTRR. Piping and penetrations from 1" to 12" inside diameter can be accurately assessed using these detectors.

Smears for removable alpha and beta contamination will be analyzed using the Protean Low Background Alpha/Beta Counter or equivalent.

Smears for removable low energy beta emitters such as H-3 will be analyzed using a Beckman Liquid Scintillation System.

The EG&G Ortec Gamma Spectroscopy System, or equivalent, will be used for gamma isotopic identification and quantification.

Table 3.1
Final Survey Instrumentation

Instrument/Detector	Detector Type	Radiation Detected	Calibration Source	Use
Ludlum Model 2350/43-68	Gas-flow proportional ($126cm^2$)	Beta	^{99}Tc (β)	Direct measurements
Ludlum Model 2350/44-2	NaI scintillator	Gamma	^{137}Cs	Gamma exposure rate
Ludlum Model 2350/44-40	Shielded GM ($15.5cm^2$)	Beta	^{99}Tc (β)	Direct measurements
Ludlum Model 2350/43-94 and Ludlum Model 2350/43-98	Gas-flow proportional pipe detectors	Beta	^{99}Tc (β)	Direct measurements inside pipes
Protean Low Background Planchet Counter	Gas-Proportional	Alpha and Beta	^{230}Th (α) ^{99}Tc (β)	Smear counting
Beckman Liquid Scintillation System	Liquid Scintillation	Beta	^{3}H (β)	Tritium Analysis
Gamma Spectroscopy System	HPGe	Gamma	Mixed Gamma	Nuclide identification and quantification

3.1 Instrument Calibration

Data loggers and associated detectors are calibrated on a semi-annual basis using National Institute of Standards and Technology (NIST) traceable sources and calibration equipment.

The data logger calibration includes:

- High Voltage calibration
- Discriminator/threshold calibration
- Window calibration
- Alarm operation verification
- Scaler calibration verification

The detector calibration includes:

- Operating voltage determination
- Calibration constant determination
- Dead time correction determination

Calibration labels showing instrument identification number, calibration date and calibration due date are attached to all field instrumentation. The presence of a current calibration label will be verified by the user before each use.

3.2 Sources

All sources used for calibration or efficiency determinations for the survey will be selected to be representative of the instrument's response to the identified nuclides and will be traceable to NIST.

Radioactive sources used for instrument response checks and efficiency determination are controlled by health physics technicians. Sources will be stored securely and will be issued to survey technicians as needed in the field.

Georgia Tech Form C will be required for all sources that are stored on site. Georgia Tech Procedures 9501, *Control and Accountability of Radioactive Materials*, and 9312, *Sealed Sources Leak Test,* will be followed for all sources brought on campus.

3.3 Minimum Detectable Activity

Minimum Detectable Activity (MDA) is defined as the smallest amount or concentration of radioactive material in a sample that will yield a net positive count with a 5% probability of falsely interpreting background responses as true activity. The MDA value is dependent upon the counting time, geometry, sample size, detector efficiency and background count rate. The equation used for calculating the MDA for field instrumentation is:

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

Where:	MDA	=	Minimum Detectable Activity (dpm/100 cm^2)
	R_b	=	Background Count Rate (cpm)
	t_b	=	Background Count Time (min)
	t_s	=	Sample Count Time (min)
	E	=	Detector Efficiency (counts/disintegration)
	A	=	Detector Area (cm^2)

The *a priori* MDAs for the final release survey will be set at, or less than, 25% of the Site Specific Guideline Value. Table 3.2 provides the *a priori* MDA's that will be established for the GTRR Final Survey. Count times will be established up to a maximum of 1 minute, such that, in areas where background exposure rates will allow, these MDA's are achieved.

Table 3.2
Field and Laboratory Instrumentation MDA's

Survey/Analysis	MDA[1]
Direct Beta Surveys	600 dpm/100 cm^2
Removable Beta Surveys	75 dpm/100 cm^2
Removable Alpha Surveys	15 dpm/100 cm^2
Gamma Spec (Co-60, Cs-137)	0.1 pCi/g
Removable H-3 Surveys	200 dpm/100 cm^2

Note [1]: These MDA Values are instrument-specific. Actual values will vary based on count times, background, efficiency, etc.

4.0 SURVEY DESIGN

4.1 Introduction

The purpose of the Final Survey is to demonstrate that the facility meets the guideline values for unrestricted release. This requires the collection of accurate and reliable data to determine activity levels. The Final Survey will be performed in accordance with the GTRR project-specific radiological survey procedures and this Final Survey Plan.

Implementation of the Final Survey will include the following:

- GTS Duratek supervisory personnel will perform a preliminary inspection of the survey areas to identify survey requirements, including special survey situations, and to provide specific survey instructions.
- A survey package will be prepared for each survey area.
- Suspect Affected survey areas will be gridded to provide a systematic method for the collection of survey data.
- Unaffected and Non-Suspect Affected areas will have a sufficient number of measurements taken to accurately represent the surface being surveyed. The number of measurements will be determined based on the guidance in NUREG/CR-5849.
- Survey locations will be marked and mapped in all non-gridded areas.
- Survey measurements will be performed and samples analyzed using appropriate calibrated instrumentation. Daily source and background checks will be carried out before and after each day's work.
- Survey data collected during the final survey will be downloaded from the survey instrument into a database for storage and processing.
- The completed survey packages will be reviewed by supervisory personnel to ensure that all required surveys are performed and that the packages contain all necessary information.
- Survey results will be reviewed by supervisory personnel to verify that the criteria for release for unrestricted use are not exceeded. Areas of residual activity will be identified, evaluated, and reclassified as required.

4.2 Area Classification

For the purpose of conducting the Final Survey at the GTRR facility, the survey areas will be classified into three different categories. First, the survey areas will be divided into Affected and Unaffected Areas based upon the initial characterization surveys, remedial actions, and historical information. Affected Areas will then be subdivided into Suspect and Non-Suspect Areas. Each area will be designated as a specific survey area and subdivided as necessary to facilitate the performance of Final Surveys. Large areas may be subdivided into several survey units, and small like areas located on the same floor may be combined into one survey unit. Equipment present in a survey area will normally be included within that area and be classified accordingly. However, each piece of equipment will be evaluated and appropriately classified as determined by the Survey Supervisor based on initial survey and post-remediation survey data.

4.2.1 Unaffected Areas

An Unaffected Area is defined as any area where there is no known history of the storage or use of unsealed licensed radioactive material. Survey results that indicate the presence of radioactive material or contamination would preclude an area from being classified as Unaffected.

4.2.2 Non-Suspect Affected Areas

A Non-Suspect Affected Area is normally defined as upper walls and overhead areas above two meters in an Affected area with minimal contamination potential based upon a review of the processes, history of the area, and characterization survey results. Survey results in excess of 50% of the site-specific guideline values (SPGLs) would preclude an area from being classified as Non-Suspect Affected.

4.2.3 Suspect Affected Areas

A Suspect Affected Area is defined as an area not meeting the requirements specified for Unaffected and Non-Suspect Affected areas. Upper walls and overhead surfaces above two meters with survey results in excess of 50% the SPGLs will be classified as Suspect Affected.

4.2.4 Survey Area Breakdown

The following is a preliminary Survey Area breakdown based on current information. This breakdown may be revised as necessary to facilitate Final Surveys:

- Unaffected Areas
  - Exterior surfaces of the Reactor Building
  - Reactor yard areas not classified as Affected

- Non-Suspect Affected Areas
  - Reactor building 2nd floor upper walls (above 2 meters) and dome
  - Reactor building 1st floor upper walls
  - Reactor building basement (outside the process equipment room) upper walls and ceiling, excluding the bismuth cooling system leak area.
  - Reactor building basement, process equipment room upper walls and ceiling.
  - Reactor building stack fan room upper walls and ceiling

- Suspect Affected Areas
  - Reactor building 2nd floor and lower walls (below 2 meters)
  - Reactor building 1st floor and lower walls
  - Reactor building basement (outside the process equipment room) floor and lower walls, including the reactor building sump, main HEPA ventilation equipment, the bismuth cooling leak area, and the biomedical door and sump.
  - Reactor building basement, process equipment room, floor and lower walls (includes the D2O tank pit and vessel pipe tunnel)
  - Reactor building ventilation hold-up duct
  - Reactor building helium rupture disc chamber
  - Reactor building stack fan room floor and lower walls
  - Reactor yard areas used for the storage of radioactive materials (includes the areas immediately outside the truck door, the secondary airlock door, and the Waste Storage Building door)

4.3 Area Re-Classification and Investigation

If during performance of the Final Survey, survey results indicate the presence of contamination in excess of 25% of the SPGL in an Unaffected area, or if survey results in excess of 50% of the SPGL on the containment ceiling insulation, or in excess of 25% of the SPGL in any additional Non-Suspect Affected areas, the area, or a part thereof, will be classified as a Suspect Affected area and an investigative survey performed. The investigative survey will consist of surveying the surrounding 9 square meters of the elevated survey pont as if it were a Suspect Affected area.

4.4 Survey Package Development

A package, or portfolio, will be developed for each survey area. A walk-down of the area will be performed, and a worksheet prepared containing a description of the survey area, historical information (as available), general survey instructions, location codes, specific survey instructions for any abnormal conditions within the survey area, and completion and review signature blocks.

During the survey, the package will be updated with direct and removable contamination survey data, exposure rate survey data, and results of any special surveys or sample analyses performed.

4.5 Survey Preparation

The GTRR dates back to the 1960's. Since the time of construction it is possible that original building surfaces may have been renovated or covered up. It may be necessary during the survey to determine on a case by case basis if contamination exists on surfaces that are now covered with later building materials, paint etc. It is not suspected that surfaces have been covered based upon initial characterization surveys and area inspections. However, if any covered areas are discovered during Final Surveys, then efforts will be made to properly expose these surfaces to be surveyed and assessed.

4.6 Survey Requirements

The Final Survey protocol is designed in part with NUREG/CR-5849 and other NRC approved Final Survey plans used at other Decommissioning and Decontamination projects as follows.

4.6.1 Suspect Affected Areas

- 100% scan of the affected surface for beta contamination to identify areas of elevated activity requiring further investigation.
- One direct measurement for each 1 m^2 of area for total beta contamination. The floors and lower walls will be gridded depending upon the number of interferences in the areas.
- One measurement for each 1 m^2 of area for removable beta and alpha contamination.
- One exposure rate measurement at 1 meter above the floor at each corresponding direct measurement location to identify any areas with elevated radiation levels relative to the normal background that will require additional investigation.

4.6.2 Non-Suspect Affected Areas

- Perform at least a 10% scan for beta contamination on building surfaces to identify areas of elevated activity requiring further investigation. The actual scanning coverage required for each "non-suspect affected" survey area (i.e. upper walls and ceilings) will be dependent upon the contamination potential for these surfaces and will be specified by the Survey Supervisor to the Survey Technician. The locations for scanning will be chosen by the technician performing the survey, and will be based on most probable areas for contamination.

- Minimum of one measurement for fixed beta, and removable alpha and beta contamination per 20 m^2, or 30 measurements, whichever is the greater for each survey area.

- Additional measurements for direct beta, and removable alpha and beta contamination on area equipment and structures if applicable.

4.6.3 Unaffected Areas

- 10% scan for beta contamination on building surfaces to identify areas of elevated activity requiring further investigation. The locations for scanning will be chosen by the technician performing the survey, and will be based on most probable areas for contamination.

- Minimum of one measurement for fixed beta, and removable alpha and beta contamination per 50 m^2, or 30 measurements, whichever is the greater for each survey area.

- Exposure rate measurements at 1 meter above the floor at each corresponding direct measurement location to identify any elevated areas requiring additional investigation.

- Random measurements for direct beta, and removable alpha and beta contamination on area equipment and structures, including measurements in locations where contamination could likely accumulate.

4.6.4 Additional Surveys for Removable H-3 Contamination

Since H-3 is expected to account for the majority of any potential residual radioactivity, in addition to the survey described above, measurements for removable H-3 will be performed. A minimum of 30 wet smears will be collected from each survey area and analyzed by liquid scintillation. The results of these smears will be used to validate the assumptions made during the preparation of SPGL's.

4.6.5 Exterior Building Surfaces

Building exterior surfaces will be classified as Unaffected and surveyed in accordance with the protocols specified above.

4.6.6 Reactor Yard

4.6.6.1 Affected Yard Areas

Affected paved areas, i.e., areas that have been used for the storage of radioactive materials, will be surveyed in the same manner as suspect affected structures.

4.6.6.2 Unaffected Yard Areas

- All unaffected paved areas in the yard will be surveyed in accordance with Section 4.6.3.

- All unaffected open land areas in the yard will receive a gamma scan covering at least 10% of the accessible areas. In addition, a minimum of 30 surface (0-6 inches) soil samples will be collected at biased locations. The RSO and EE will approve of the biased soil sample locations prior to collection of samples.

4.7 Gridding

This plan specifies a gridded survey method for affected area floors and lower walls. Area congestion in some areas may make gridding impractical. For affected areas where gridding cannot be performed, the areas will be measured and evenly spaced measurements taken and marked at a frequency of 1 measurement per square meter.

Unaffected and Non-Suspected Affected areas will not be gridded. Each survey point location will be marked or mapped identifying the approximate survey location.

4.8 Survey Records

Records of surveys will be maintained in a separate survey package for each survey area in accordance with project procedures. The specific records that will be compiled in a survey package are:

- Survey Package Worksheet giving the package identification, survey location information, historical information of area surveyed, general survey instructions and any specific survey instructions.
- Survey Comment Addendum provided for comments from the survey technician regarding any unusual situation that may be encountered while surveying as applicable.
- The Survey Unit Diagram of the area to be surveyed as available. Survey grids are represented on the drawing.
- Photographs of the survey area will be provided, as necessary, to show special or unique conditions.
- Printout of laboratory analysis results (if performed).
- Ludlum Model 2350 data files and Paradox® converted values for all radiation survey measurements.

Survey measurements will be taken using the Ludlum Model 2350 Data Logger system. Upon completion of an individual survey unit or combination of units, the contents of the data logger's memory will be downloaded to a Paradox® Database. The download process utilizes a proprietary program developed by Ludlum Measurements, Inc., and revised by GTS Duratek to fit specific survey needs.

A printed report, referred to as Survey Report, will then be generated for review. All raw data, converted data, and information by survey location code will be presented in the report. This report will be reviewed by the survey technician, survey supervisor and Certified Health Physicist for completeness, accuracy, any suspect entries and comparison to the site specific guideline value. Data will be made available to the RSO and the EE, after review by the survey supervisor and prior to submission of the report.

Any changes to the database tables such as detector efficiency, background, etc., that could affect survey results will require the survey supervisor approval. In addition, changes to data in the primary table will require a written explanation on a change request attached to the survey report and maintained as a permanent record.

Data and document control will include maintaining raw data files, translated data files (Paradox® data files) and corrected data files showing documentation of all corrections. Paradox® program scripts and related data and information, including modifications, used for the development of the report will be identified and controlled to ensure accurate identification. The databases will be backed-up daily, and archived on a weekly basis. Survey records will be turned over to Ga Tech with the issuance of the final report.

4.9 Data Analysis

Final survey data collected during the Final Survey of the Georgia Tech Research Reactor will be analyzed in accordance with the guidance contained in NUREG/CR-5849. Survey data, for measurements of total beta activity, removable beta activity, removable H-3 activity, and exposure rate will be grouped by survey unit or survey area as appropriate. Survey results will be corrected as appropriate to account for background, including naturally occurring radioactive material, present in building materials, and as a result of instrument noise.

4.9.1 Background

In order to analyze the survey results associated with total beta activity measurements and exposure rate measurements, the analyses will have to account for the naturally occurring radioactive background. Generally this requires a background study in an area representative of the areas surveyed during the final survey, using the same instrumentation and survey protocols that are used during the final survey. Since the Georgia Tech Research Reactor containment vessel is unique, identifying a background study area may be difficult. Of particular concern is the difficulty in identifying a background study area suitable for accessing exposure rates. An alternative means of establishing background exposure rates may be used upon the approval of the Georgia Tech RSO and the Executive Engineer.

4.9.2 Measurements of Total Beta Activity

Measurement results for total beta activity from each survey unit, once corrected for background, will be compared directly to the site specific guideline value for total beta activity. If any individual measurement result exceeds this value then the measurement result will be compared to the average and maximum criteria as defined in the footnotes to Table 2-1 as follows.

Individual measurement results less than 3 times the site specific guideline value for total beta activity will be accepted provided that the affected area is limited to less than approximately 100 cm^2.

NOTE: For the purpose of assessing the size of the contaminated area, an area of 126 cm^2 corresponding to the size of the open window on the Ludlum Model 43-68 gas flow proportional detector is acceptable.

If any individual measurement result, in any square meter, is between 1 and 3 times the site specific guideline value for total beta activity the average beta activity in the square meter must be evaluated. The average beta activity in the square meter, weighted by sizes of the areas of elevated activity, must be less than the site specific guideline value for total beta activity.

Once the individual measurements results for total beta activity from a given survey unit have been shown to meet the site specific guideline value or the average and maximum criteria as defined in the footnotes to Table 2.1 the 95 percent confidence level of the mean for the survey unit will be evaluated using the following equation:

$$u_x = \bar{x} + t_{(1-\alpha, df)} \frac{S_x}{\sqrt{n}}$$

Where: u_x = 95 percent confidence level of the mean, dpm/100 cm^2.

$t_{(1-\alpha, df)}$ = 95 percent confidence level for n-1 degrees of freedom, unitless.

Measurement results of total beta activity will be presented graphically by survey unit. Included with each graph will be the following statistics: the maximum result obtained in each survey unit, the mean (average) result, the standard deviation in the results, the number of measurements in the survey unit, and the 95 percent confidence level of the mean for the survey unit.

4.9.3 Measurements of Removable Beta Activity

Measurement results for removable beta activity from each survey unit will be compared directly with the site specific guideline value for removable beta activity. Measurement results that exceed the site specific guideline value for removable beta activity indicate the need for remedial action.

Measurement results of removable beta activity will be presented graphically by survey unit. Included with each graph will be the following statistics: the maximum result obtained in each survey unit, the mean (average) result, the standard deviation in the results, and the number of measurements in the survey unit.

Measurement results for removable H-3 activity will be used to validate the assumptions made concerning H-3 contribution to the nuclide mix during the preparation of Site Specific Guideline Values. For a given survey unit, the mean removable tritium activity will be compared to the mean removable beta activity. If the relative tritium fraction exceeds that assumed during the development of the site specific guideline value, the assumptions used to generate the site specific guideline values will be re-evaluated. Any such areas will be addressed on a case by case basis as they are identified by any party, and will be reviewed by the Executive Engineer and Georgia Tech RSO.

4.9.4 Exposure Rate Measurements

Exposure rate measurement results from each survey unit will be compared to the background exposure rate established for the survey unit. Measurement results that exceed the established background by more than 5 μR/hr will be investigated. Elevated exposure rates that cannot be attributed to variations in background may indicate the need for remedial action.

Exposure rate measurement results will be presented graphically by survey unit. Included with each graph will be the following statistics: The maximum result obtained in each survey unit, the mean (average) result, the standard deviation in the results, and the number of measurements in the survey unit.

4.9.5 Samples of Open Land Areas of Reactor Yard

Based on the fact that the open land areas of the reactor yard are classified as unaffected, the sample results will be compared to individual isotopic limits of Table 2.3. If any analysis result is found greater than 10% of the individual radionuclide specific concentration or the sum of fractions of the gamma emitting nuclides exceed 25%, then the sample with the highest total activity will be sent offsite for additional analyses. These additional analyses will include; isotopic uranium, Sr-90, C-14, and Tc-99.

5.0 QUALITY ASSURANCE AND QUALITY CONTROL

The GTRR project Quality Assurance Program is constructed to ensure that all quality and regulatory requirements are satisfied. All activities affecting quality are suitably controlled by project procedures and the Project Quality Assurance Plan. These procedures ensure that the appropriate equipment, environmental conditions, quality controls and prerequisites for any given activity are met. The following Quality Control measures will be implemented as an integral part of the final release survey process.

5.1 General Provisions

5.1.1 Selection of Personnel

Final Survey supervisory personnel are required to have extensive experience and familiarity with survey procedures and processes, including NUREG/CR-5849 and this Final Survey Plan. They must have prior experience with the radionuclide(s) of concern and a working knowledge of the instruments used to detect the radionuclides on site. The Survey Supervisor will have a minimum of five years of experience in performing and supervising final release surveys where NUREG/CR-5849 guidelines were implemented.

The selection of supervisory personnel to direct the survey will be based upon their experience and familiarity with the survey procedures and processes. Likewise, Health Physics Technicians who will perform the surveys will be selected based upon their qualifications and experience.

5.1.2 Written Procedures

All survey tasks which are essential to survey data quality will be controlled by project procedures. A list of procedures expected to be used during this survey are provided in Attachment 1 of the Initial Radiological Survey Plan (IT-GT-PL-SC-001).

5.1.3 Instrumentation Selection, Calibration and Operation

GTS Duratek has selected instruments proven to reliably detect the radionuclides present at the GTRR. Instruments will be calibrated off-site by GTS Duratek or qualified vendors under approved procedures using calibration sources traceable to the National Institute of Standards and Technology (NIST). All instruments brought to the GTRR project site will have current calibration certificates. A copy of these certificates will be maintained on-site in the project file.

All detectors are subject to daily response checks when in use. Measurements are performed using approved written procedures for each instrument. Issue, control and accountability of all survey instrumentation has been established by an instrumentation control procedure. Procedures for calibration, maintenance, accountability, operation and quality control of radiation detection instruments are written to implement the guidelines established in American National Standard Institute (ANSI) standard ANSI N323-1978 and ANSI N42.17A-1989.

5.1.4 Survey Documentation

The survey packages will be the primary method of controlling and tracking the hard copy records of survey results. Records of surveys will be documented and maintained in the survey package for each area. Each survey measurement will be identified by the date, technician, instrument type and serial number, detector type and serial number, location code, type of measurement, and mode of instrument operation.

5.1.5 Chain of Custody

Procedures establish responsibility for the custody of samples from the time of collection until results are obtained. Samples shipped off site for analysis will be accompanied by a chain-of-custody record to track each sample.

5.1.6 Records Management

Handling and storage of survey design and data packages will be controlled in accordance with the applicable project records management procedure.

5.1.7 Review of Survey Results

The survey package from each survey area will be reviewed by the Survey Supervisor to verify all documentation is complete and accurate and that release criteria have been met prior to input into the survey database.

5.2 Training

Prior to implementation of the Survey Plan, applicable project personnel involved in the survey shall receive all required project-specific training, including health and safety training, an overview of the Survey Plan, the objectives of the survey, and procedures governing the survey.

5.2.1 General

All project personnel will receive site specific training to identify the specific hazards present in the work and survey areas. Training will also include a briefing and review of the Final Survey Plan, applicable project procedures, and the Site Health & Safety Plan.

5.2.2 Emergency Procedures

During site orientation and training, survey personnel will become familiar with site emergency procedures. In the event of an emergency, personnel will act in accordance with all applicable site emergency procedures and the Site Health & Safety Plan.

6.0 FINAL SURVEY REPORT

GTS Duratek will begin preparation of the Final Survey Report while conducting the survey. General information can be drafted early in order to expedite report preparation when work is completed. The report will contain brief descriptions of the site and the survey performed, as well as photographs of the survey and sample locations. Survey results will be summarized, and presented in tabular and graphical form along with summary statistics for a Certified Health Physicist review prior to inclusion into the Final Survey Report. Raw survey data is normally presented as appendices to the report.

A draft Final Survey Report will be prepared and submitted to the Georgia Tech RSO and the Executive Engineer for comment upon completing site activities and/or the availability of all sample analyses. The Final Report will be submitted to the Georgia Tech RSO and the Executive Engineer following resolution of all comments. The Final Survey Report will provide adequate data and discussion of each topic to meet the intent of NUREG/CR-5849 and the NRC approved Georgia Institute of Technology Research Reactor Decommissioning Plan. The following is a typical Final Survey Report outline:

1.0 INTRODUCTION AND SCOPE
- 1.1 Introduction
- 1.2 Project Scope

2.0 BACKGROUND INFORMATION
- 2.1 Facility Description
- 2.2 Operating History
- 2.3 Management Approach

3.0 SURVEY
- 3.1 Introduction
- 3.2 Survey Guideline Values
- 3.3 Instrumentation
- 3.4 Survey Organization
- 3.5 Survey Design
- 3.6 Quality Assurance and Quality Control
- 3.7 Documentation
- 3.8 Data Reduction and Statistical Evaluation

4.0 SURVEY RESULTS
- 4.1 Survey Results Summary
- 4.2 Direct Surveys
- 4.3 Smear Surveys
- 4.4 Exposure Rate Surveys
- 4.5 Open Land Area Surveys

5.0 CONCLUSIONS AND RECOMMENDATIONS

6.0 REFERENCES

7.0 REFERENCES

7.1 NES Inc., *Georgia Institute of Technology Research Reactor Decommissioning Plan*, June 1998.

7.2 USNRC Regulatory Guide 1.86, *Termination of Operating Licenses for Nuclear Reactors.*

7.3 USNRC NUREG/CR-5849, *Manual for Conducting Radiological Surveys in Support of License Termination*, Draft, June, 1992

7.4 Letter from GSFIC Project I-47 Executive Engineer (CH2MHill) to IT Corporation, Response to letter dated April 5, 2000, Final Release Survey Plan - Request for Complete, Definite, and Clear Instructions (Article E-3), April 18, 2000.

7.5 10 CFR19, *Notices, Instructions, and Reports to Workers; Inspections.*

7.6 10 CFR20, *Standards for Protection Against Radiation.*

7.7 American National Standard Institute, ANSI N323-1978, *Radiation Protection Instrumentation Test and Calibration*

7.8 American National Standard Institute, ANSI N42.17A-1989, *Performance Specifications for Health Physics Instrumentation - Portable Instrumentation for Use in Normal Environmental Conditions*

7.9 The IT Group, *Quality Assurance Project Plan & Quality Assurance Procedures for the Georgia Institute of Technology Research Reactor Decommissioning Project*

8.0 ATTACHMENTS

Attachment A *Results of 10 CFR Part 61 Sample Analyses*

Attachment B *Worksheets for Calculation of Site Specific Guideline Values*

ATTACHMENT A
RESULTS OF 10 CFR PART 61
SAMPLE ANALYSES

General Engineering Laboratories, Inc.

COA Terms Summary:

The detail on the Certificate of Analysis includes the following:

Parameter:	Analyte or characteristic tested for in the sample
Qualifier:	Qualifier used for data interpretation
Result:	Final result of each parameter.
DL:	Method Detection Limit
RL:	Reporting Limit
Units:	Units of final result
DF:	Dilution factor
Analyst:	Initials of analyst who performed the test
Date:	Date of analysis
Time:	Time of analysis
Batch:	Analytical batch in which the sample was analyzed
Method:	Analytical method used for the analysis of the sample. Identified on the report numerically with a corresponding table.
Surrogate Recovery:	Provided for organics analysis only. Surrogate compound identified.
Test:	Analytical test associated with surrogate compound.
Percent:	Surrogate percent recovery
Acceptable Limits:	Limits established for surrogate recoveries based upon the method requirements.

The QC Summary Report contains the following headings:

Sample Parameter:	Analyte or characteristic tested for in the QC sample
Type:	Type of QC sample (i.e., blank, dup, LCS, LCS dup, MS, MSD)
Batch:	Analytical batch in which the QC sample was analyzed
NOM:	Nominal concentration of the spiking compound
Sample:	Amount of compound found in the sample associated with the QC sample.
QC:	Amount of compound found in the QC sample.
Units:	Units of final result
RPD%:	Relative percent difference between LCS/LCS dup, MS/MSD, and Sample/Sample duplicate
REC%:	Recovery for the control samples
Range:	Acceptance limits for control samples
Analyst:	Initials of analyst who performed the test
Date:	Date of analysis
Time:	Time of analysis

Types of QC samples that may be found on the QC Summary Report are:

Blank:	Results of the blank analysis for the sample batch
Dup:	Duplicate analysis of sample
LCS:	Lab control sample
LCSD:	Lab control sample duplicate
MS:	Matrix spike
MSD:	Matrix spike duplicate

The following are definitions of reporting limits used at General Engineering Laboratories:

DL — Detection Limit: The minimum level of an analyte that can be determined (identified not quantified) with 99% confidence. The values are normally achieved by preparing and analyzing seven aliquots of laboratory water spiked 1 to 5 times the estimated MDL, taking the standard deviation and multiplying it against the one-tailed t-statistic at 99%. This computed value is then verified for reasonableness by repeating the study using the concentration found in the initial study, calculating an F-ratio, and computing the final limit. Sample specific preparation and dilution factors are applied to these limits when they are reported.

The detection limit is the minimum concentration of a substance that can be identified, measured, and reported with 99% confidence that the analyte concentration is above zero. It answers the question "Is It Present".

QL — Quantitation Limit: The lowest concentration that can be reliably achieved within specified limits of precision and accuracy during routine laboratory operating conditions. The QL is generally 5 to 10 times the MDL. However, it may be nominally chosen within these guidelines to simplify data reporting. For many analytes the QL analyte concentration is selected as the lowest non-zero standard in the calibration curve. Sample QL's are highly matrix-dependent. Sample specific preparation and dilution factors are applied to these limits when they are reported.

The QL is always $\geq$ DL

RL — Reporting Limit: Is defined as the same as the QL except where driven by contract or client specifications. If the sample specific preparation and dilution factors cause the QL to be elevated above the RL, then the QL is used as the RL.

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Certificate of Analysis

Company : GTS Duratek
Address : 628 Gallaher Road
Kingston, TN 37763

Report Date: February 15, 2000

Contact: Mr. Phil Mann
Project: Routine Analytical

Client Sample ID: UNDER VESSEL
Sample ID: 20577001
Matrix: Misc Solid
Collect Date: 12-JAN-00
Receive Date: 13-JAN-00
Collector: Client

Project: GTSD00799
Client ID: GTSD002

Parameter	Qualifier	Result		DL	RL	Units	DF	Analyst	Date	Time	Batch	Method
Rad Alpha Spec												
Alphaspec Am241, Cm, solid												
Americium-241	U	ND	+/-0.556	1.36	0.1	pCi/g	1	AMH	02/02/00	1724	8941	
Curium-242	U	ND	+/-0.464	1.24	0.1	pCi/g	1					
Curium-243/244	U	ND	+/-0.785	1.77	0.1	pCi/g	1					
Curium-245/246	U	ND	+/-0.235	0.693	0.1	pCi/g	1					
Alphaspec Pu, solid												
Plutonium-238	U	ND	+/-0.409	0.962	0.1	pCi/g	1	AMH	02/02/00	1740	8932	
Plutonium-239/240	U	ND	+/-0.128	0.41	0.1	pCi/g	1					
Plutonium-244	U	ND	+/-0.134	0.478	0.1	pCi/g	1					
Alphaspec U, solid												
Uranium-233/234		1.71	+/-0.444	0.326	1	pCi/g	1	AMH	01/28/00	2211	8762	
Uranium-235/236	U	ND	+/-0.0925	0.148	1	pCi/g	1					
Uranium-238		1.65	+/-0.417	0.182	1	pCi/g	1					
Liquid Scint Pu241, solid												
Plutonium-241	U	ND	+/-32.6	57.5	5	pCi/g	1	JAL	02/08/00	1427	8734	
Rad Gamma Spec												
Gamma Fe55, solid												
Iron-55	U	167	+/-168	45.9	10	pCi/g	1	SRB	02/07/00	1635	8337	
Gamma I129, solid												
Iodine-129	U	ND	+/-1.13	0.784	1	pCi/g	1	LGJ	01/28/00	1031	7217	
Gamma Ni59, solid												
Nickel-59	U	ND	+/-17.3	19.3	10	pCi/g	1	SRB	02/04/00	1057	8336	
Gammaspec, Gamma, solid												
Cesium-134		0.756	+/-0.186	0.18	0.05	pCi/g	1	SRB	01/23/00	1454	8340	
Cesium-137		1.21	+/-0.337	0.223	0.05	pCi/g	1					
Cobalt-57	U	ND	+/-0.35	0.116	0.05	pCi/g	1					
Cobalt-58	U	ND	+/-0.192	0.306	0.05	pCi/g	1					
Cobalt-60		153	+/-15.2	0.204	0.05	pCi/g	1					
Europium-152		7.04	+/-0.813	0.377	0.1	pCi/g	1					
Europium-154		1.41	+/-0.654	0.653	0.5	pCi/g	1					
Europium-155	U	ND	+/-0.244	0.218	0.5	pCi/g	1					
Manganese-54	U	ND	+/-0.183	0.289	1000	pCi/g	1					
Niobium-94	U	ND	+/-0.138	0.209	1000	pCi/g	1					

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GENERAL ENGINEERING LABORATORIES

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Certificate of Analysis

Company : GTS Duratek
Address : 628 Gallaher Road
Kingston, TN 37763

Report Date: February 15, 2000

Contact: Mr. Phil Mann
Project: Routine Analytical

Client Sample ID: UNDER VESSEL
Sample ID: 20577001

Project: GTSD00799
Client ID: GTSD002

Parameter	Qualifier	Result		DL	RL	Units	DF	Analyst	Date	Time	Batch	Method
Rad Gamma Spec												
Gammaspec, Gamma, solid												
Silver-110m	U	ND	+/-0.155	0.209	0.05	pCi/g	1					
Zinc-65	U	ND	+/-0.578	0.796	0.1	pCi/g	1					
Rad Gas Flow												
GFPC, Sr89&Sr90, Solid												
Strontium-89	U	ND	+/-0.154	0.734	0.5	pCi/g	1	HGA	02/02/00	0047	8514	
Strontium-90		0.735	+/-0.349	0.563	0.5	pCi/g	1					
Rad Liquid Scint												
LSC, Tritium Dist, solid												
Tritium		31700	+/-610	56.4	.8	pCi/g	1	FDP	02/10/00	1637	9926	
Liquid Scint C14, solid												
Carbon-14	U	211	+/-6.72	4.59	1	pCi/g	1	KDA	02/05/00	0357	10391	
Liquid Scint Ni63, solid												
Nickel-63		69.3	+/-9.64	23.4	10	pCi/g	1	AEA	01/29/00	1157	8487	
Liquid Scint Tc99, solid												
Technetium-99	J	1.91	+/-0.56	1.03	10	pCi/g	1	MTB	01/26/00	0810	7846	

The following Prep Methods were performed

Method	Description	Analyst	Date	Time	Prep Batch
Ash Soil Prep	Ash Soil Prep EPI A-021,A-021B,A-026	CRS	01/21/00	0811	8155
Dry Soil Prep	Dry Soil Prep EPI A-021,A-021B,A-026	CRS	01/20/00	1249	7536

The following Analytical Methods were performed

Method	Description
1	DOE EML HASL 300
2	DOE EML HASL 300
3	DOE EML HASL 300
4	DOE EML HASL 300
5	DOE RESL Fe-1
6	LANL EM-9
7	DOE RESL Ni-1
8	DOE EML HASL 300
9	EPA 905.0
10	EPA 906.0

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Certificate of Analysis

Company : GTS Duratek
Address : 628 Gallaher Road
Kingston, TN 37763

Report Date: February 15, 2000

Contact: Mr. Phil Mann
Project: Routine Analytical

Client Sample ID: UNDER VESSEL
Sample ID: 20577001

Project: GTSD00799
Client ID: GTSD002

Parameter	Qualifier	Result	DL	RL	Units	DF	AnalystDate	Time	Batch	Method
11	EPA 504.1									
12	DOE RESL Ni-1									
13	DOE EML HASL 300									

Notes:
The Qualifiers in this report are defined as follows :

* Indicates that a quality control analyte recovery is outside of specified acceptance criteria.
** Indicates the analyte is a surrogate compound.
J Indicates an estimated value. The result was greater than the detection limit, but less than the reporting limit.
U Indicates the compound was analyzed for but not detected above the detection limit

The above sample is reported on an "as received" basis.

This data report has been prepared and reviewed in accordance with General Engineering Laboratories, Inc. standard operating procedures. Please direct any questions to your Project Manager, Pete Wilber at 843-556-8171 Ext. 4411.

Reviewed by

GENERAL ENGINEERING LABORATORIES

Meeting today's needs with a vision for tomorrow.

Certificate of Analysis

Company : GTS Duratek
Address : 628 Gallaher Road
Kingston, TN 37763

Report Date: February 15, 2000

Contact: Mr. Phil Mann
Project: Routine Analytical

Client Sample ID: DOOR PIT
Sample ID: 20577002
Matrix: Misc Solid
Collect Date: 12-JAN-00
Receive Date: 13-JAN-00
Collector: Client

Proiect: GTSD00799
Client ID: GTSD002

Parameter	Qualifier	Result		DL	RL	Units	DF	Analyst	Date	Time	Batch	Method
Rad Alpha Spec												
Alphaspec Am241, Cm, solid												
Americium-241	U	ND	+/-0.544	0.872	0.1	pCi/g	1	AMH	02/02/00	1730	8941	
Curium-242	U	ND	+/-0.179	0.639	0.1	pCi/g	1					
Curium-243/244	U	ND	+/-0.542	1.2	0.1	pCi/g	1					
Curium-245/246	U	ND	+/-0.181	0.58	0.1	pCi/g	1					
Alphaspec Pu, solid												
Plutonium-238	U	ND	+/-0.351	0.59	0.1	pCi/g	1	AMH	02/02/00	1745	8932	
utonium-239/240	U	ND	+/-0.243	0.363	0.1	pCi/g	1					
utonium-244	U	ND	+/-0.142	0.363	0.1	pCi/g	1					
Alphaspec U, solid												
Uranium-233/234	J	0.952	+/-0.316	0.19	1	pCi/g	1	AMH	01/28/00	2211	8762	
Uranium-235/236	U	ND	+/-0.085	0.155	1	pCi/g	1					
Uranium-238	J	0.685	+/-0.259	0.0685	1	pCi/g	1					
Liquid Scint Pu241, solid												
Plutonium-241	U	ND	+/-39.9	70.3	5	pCi/g	1	JAL	02/08/00	1529	8734	
Rad Gamma Spec												
Gamma Fe55, solid												
Iron-55		468	+/-82.7	44.5	10	pCi/g	1	SRB	02/07/00	1532	8337	
Gamma I129, solid												
Iodine-129	U	ND	+/-0.636	0.825	1	pCi/g	1	LGJ	01/28/00	1337	7217	
Gamma Ni59, solid												
Nickel-59	U	ND	+/-23	24.1	10	pCi/g	1	SRB	02/04/00	1207	8336	
Gammaspec, Gamma, solid												
Cesium-134	U	ND	+/-0.149	0.205	0.05	pCi/g	1	SRB	01/23/00	1454	8340	
Cesium-137		2.46	+/-0.426	0.257	0.05	pCi/g	1					
Cobalt-57	U	ND	+/-0.186	0.0685	0.05	pCi/g	1					
Cobalt-58	U	ND	+/-0.225	0.347	0.05	pCi/g	1					
Cobalt-60		234	+/-26.8	0.221	0.05	pCi/g	1					
Europium-152		1.46	+/-0.457	0.436	0.1	pCi/g	1					
Europium-154	U	ND	+/-0.381	0.615	0.5	pCi/g	1					
Europium-155	U	ND	+/-0.165	0.28	0.5	pCi/g	1					
Manganese-54	U	ND	+/-0.221	0.337	1000	pCi/g	1					
Niobium-94	U	ND	+/-0.154	0.239	1000	pCi/g	1					
Silver-110m	U	ND	+/-0.177	0.243	0.05	pCi/g	1					

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Certificate of Analysis

Company : GTS Duratek
Address : 628 Gallaher Road
Kingston, TN 37763

Report Date: February 15, 2000

Contact: Mr. Phil Mann
Project: Routine Analytical

Client Sample ID: DOOR PIT
Sample ID: 20577002

Project: GTSD00799
Client ID: GTSD002

Parameter	Qualifier	Result		DL	RL	Units	DF	Analyst	Date	Time	Batch	Method
Rad Gamma Spec												
Gammaspec, Gamma, solid												
Zinc-65	U	ND	+/-0.554	0.899	0.1	pCi/g	1					
Rad Gas Flow												
GFPC, Sr89&Sr90, Solid												
Strontium-89	U	ND	+/-0.159	0.729	0.5	pCi/g	1	HGA	02/02/00	0020	8514	
Strontium-90	U	ND	+/-0.338	0.582	0.5	pCi/g	1.					
Rad Liquid Scint												
LSC, Tritium Dist, solid												
Tritium		139	+/-7.74	6.83	8	pCi/g	1	FDP	02/10/00	1709	9926	
uid Scint C14, solid												
urbon-14		77.4	+/-4.59	4.59	1	pCi/g	1	KDA	02/05/00	0429	10391	
Liquid Scint Ni63, solid												
Nickel-63		82	+/-13.1	31.8	10	pCi/g	1	AEA	01/29/00	1228	8487	
Liquid Scint Tc99, solid												
Technetium-99	U	ND	+/-0.813	1.73	10	pCi/g	1	MTB	01/26/00	0913	7846	

The following Prep Methods were performed

Method	Description	Analyst	Date	Time	Prep Batch
Ash Soil Prep	Ash Soil Prep EPI A-021,A-021B,A-026	CRS	01/21/00	0811	8155
Dry Soil Prep	Dry Soil Prep EPI A-021,A-021B,A-026	CRS	01/20/00	1249	7536

The following Analytical Methods were performed

Method	Description
1	DOE EML HASL 300
2	DOE EML HASL 300
3	DOE EML HASL 300
4	DOE EML HASL 300
5	DOE RESL Fe-1
6	LANL EM-9
7	DOE RESL Ni-1
8	DOE EML HASL 300
9	EPA 905.0
10	EPA 906.0
11	EPA 504.1

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Project: Routine Analytical

Client Sample ID: DOOR PIT
Sample ID: 20577002

Project: GTSD00799
Client ID: GTSD002

Parameter	Qualifier	Result	DL	RL	Units	DF	AnalystDate	Time	Batch	Method
12	DOE RESL Ni-1									
13	DOE EML HASL 300									

Notes:
The Qualifiers in this report are defined as follows :

* Indicates that a quality control analyte recovery is outside of specified acceptance criteria.
** Indicates the analyte is a surrogate compound.
J Indicates an estimated value. The result was greater than the detection limit, but less than the reporting limit.
U Indicates the compound was analyzed for but not detected above the detection limit

:e above sample is reported on an "as received" basis.

This data report has been prepared and reviewed in accordance with General Engineering Laboratories, Inc. standard operating procedures. Please direct any questions to your Project Manager, Pete Wilber at 843-556-8171 Ext. 4411.

Reviewed by

GENERAL ENGINEERING LABORATORIES

Meeting today's needs with a vision for tomorrow.

Certificate of Analysis

Company : GTS Duratek
Address : 628 Gallaher Road
Kingston, TN 37763

Report Date: February 15, 2000

Contact: Mr. Phil Mann
Project: Routine Analytical

Client Sample ID: PRIMARY SYSTEM
Sample ID: 20577003
Matrix: Misc Solid
Collect Date: 12-JAN-00
Receive Date: 13-JAN-00
Collector: Client

Project: GTSD00799
Client ID: GTSD002

Parameter	Qualifier	Result		DL	RL	Units	DF	Analyst	Date	Time	Batch	Method
Rad Alpha Spec												
Alphaspec Am241, Cm, solid												
Americium-241	U	ND	+/-0.428	1.15	0.1	pCi/g	1	AMH	02/02/00	1730	8941	
Curium-242	U	ND	+/-0.341	1.09	0.1	pCi/g	1					
Curium-243/244	U	ND	+/-0.843	1.81	0.1	pCi/g	1					
Curium-245/246	U	ND	+/-0.403	0.782	0.1	pCi/g	1					
Alphaspec Pu, solid												
Plutonium-238	U	ND	+/-0.415	0.614	0.1	pCi/g	1	AMH	02/02/00	1745	8932	
ιtonium-239/240		0.946	+/-0.49	0.38	0.1	pCi/g	1					
ιutonium-244	U	ND	+/-0.121	0.414	0.1	pCi/g	1					
Alphaspec U, solid												
Uranium-233/234	J	0.686	+/-0.284	0.214	1	pCi/g	1	AMH	01/28/00	2211	8762	
Uranium-235/236	U	ND	+/-0.0956	0.174	1	pCi/g	1					
Uranium-238	U	ND	+/-0.0802	0.174	1	pCi/g	1					
Liquid Scint Pu241, solid												
Plutonium-241	U	ND	+/-21.3	37.4	5	pCi/g	1	JAL	02/08/00	1631	8734	
Rad Gamma Spec												
Gamma Fe55, solid												
Iron-55	U	2020	+/-471	23.7	10	pCi/g	1	SRB	02/07/00	1422	8337	
Gamma I129, solid												
Iodine-129	U	ND	+/-11.8	7.81	1	pCi/g	1	LGJ	01/28/00	1509	7217	
Gamma Ni59, solid												
Nickel-59		26.9	+/-16.5	11.8	10	pCi/g	1	SRB	02/04/00	1336	8336	
Gammaspec, Gamma, solid												
Cesium-134		17	+/-2.37	1.42	0.05	pCi/g	1	SRB	01/23/00	1455	8340	
Cesium-137		4220	+/-512	1.8	0.05	pCi/g	1					
Cobalt-57	U	ND	+/-0.439	0.503	0.05	pCi/g	1					
Cobalt-58	U	ND	+/-1.48	2.35	0.05	pCi/g	1					
Cobalt-60		12200	+/-1320	1.76	0.05	pCi/g	1					
Europium-152		13.1	+/-2.82	3.27	0.1	pCi/g	1					
Europium-154	U	ND	+/-4.03	4.86	0.5	pCi/g	1					
Europium-155	U	ND	+/-1.81	1.93	0.5	pCi/g	1					
Manganese-54	J	53.3	+/-7.65	2.3	1000	pCi/g	1					
Niobium-94	U	ND	+/-1.24	1.62	1000	pCi/g	1					
Silver-110m		122	+/-15.1	1.66	0.05	pCi/g	1					

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Address : 628 Gallaher Road
Kingston, TN 37763

Report Date: February 15, 2000

Contact: Mr. Phil Mann
Project: Routine Analytical

Client Sample ID: PRIMARY SYSTEM   Project: GTSD00799
Sample ID: 20577003   Client ID: GTSD002

Parameter	Qualifier	Result		DL	RL	Units	DF	Analyst	Date	Time	Batch	Method
Rad Gamma Spec												
Gammaspec, Gamma, solid												
Zinc-65		150	+/-18	5.92	0.1	pCi/g	1					
Rad Gas Flow												
GFPC, Sr89&Sr90, Solid												
Strontium-89	U	ND	+/-6.9	0.935	0.5	pCi/g	1	HGA	02/02/00	0020	8514	
Strontium-90		6570	+/-13.1	0.687	0.5	pCi/g	1					
Rad Liquid Scint												
LSC, Tritium Dist, solid												
Tritium		4.880E+07	+/-2.660E+05	2240	8	pCi/g	1	FDP	02/10/00	1710	9926	
-uid Scint C14, solid												
-rbon-14		8.320E+05	+/-9410	182	1	pCi/g	1	KDA	02/05/00	0430	10391	
Liquid Scint Ni63, solid												
Nickel-63		4800	+/-96.2	230	10	pCi/g	1	AEA	01/29/00	1253	8487	
Liquid Scint Tc99, solid												
Technetium-99	U	114	+/-9.06	9.61	10	pCi/g	1	MTB	02/08/00	1325	7846	

The following Prep Methods were performed

Method	Description	Analyst	Date	Time	Prep Batch
Ash Soil Prep	Ash Soil Prep EPI A-021,A-021B,A-026	CRS	01/21/00	0811	8155
Dry Soil Prep	Dry Soil Prep EPI A-021,A-021B,A-026	CRS	01/20/00	1249	7536

The following Analytical Methods were performed

Method	Description
1	DOE EML HASL 300
2	DOE EML HASL 300
3	DOE EML HASL 300
4	DOE EML HASL 300
5	DOE RESL Fe-1
6	LANL EM-9
7	DOE RESL Ni-1
8	DOE EML HASL 300
9	EPA 905.0
10	EPA 906.0
11	EPA 504.1

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Company : GTS Duratek
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Kingston, TN 37763

Report Date: February 15, 2000

Contact: Mr. Phil Mann

Project: Routine Analytical

Client Sample ID: PRIMARY SYSTEM
Sample ID: 20577003

Project: GTSD00799
Client ID: GTSD002

Parameter	Qualifier	Result	DL	RL	Units	DF	AnalystDate	Time	Batch	Method
12	DOE RESL Ni-1									
13	DOE EML HASL 300									

Notes:
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This data report has been prepared and reviewed in accordance with General Engineering Laboratories, Inc. standard operating procedures. Please direct any questions to your Project Manager, Pete Wilber at 843-556-8171 Ext. 4411.

Reviewed by

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ATTACHMENT B
WORKSHEETS FOR CALCULATION OF SITE-SPECIFIC GUIDELINE VALUES

ATTACHMENT B

Calculation of Site-Specific Guideline Values (SPGLs)

The equation used to calculate the site-specific guideline value for each of the three samples is:

$$SPGL = \frac{F}{\sum_{i=1}^{n} \frac{f_i}{GL_i}}$$

Where: F = The fraction of the radionuclide mix that is considered detectable, unitless.

f_i = The relative fraction of radionuclide i in the mix, unitless.

GL_i = The radionuclide specific guideline, dpm/100 cm^2 from Table 2.1

Using this equation, a site-specific guideline value (SPGL) was calculated for both average total activity and removable activity for each of three samples selected for 10 CFR Part 61 analysis. Average SPGLs were then calculated based on the SPGLs obtained for each sample. The calculated SPGLs are summarized in the table below.

Calculation of Site-Specific Guideline Values

	Sample GEL-1 - Concrete under Reactor Vessel	Sample GEL-2 - Concrete in Biomedical Door Pit	Sample GEL-3 - Primary Resin Sample	Average of Samples GEL-1, 2, & 3
Description	Activity (dpm/100cm^2)	Activity (dpm/100cm^2)	Activity (dpm/100cm^2)	Activity (dpm/100cm^2)
Calculated guideline value for average total activity	1,500	3,800	2,000	2,400
Calculated guideline value for removable activity	**104**	**688**	**148**	**313**

SITE SPECIFIC GUIDELINE, SPGL, BASED ON GEL-1(TOTAL ACTIVITY)

RADIONUCLIDE SPECIFIC GUIDELINE FOR H-3 AND Fe-55 RAISED TO 200,000 dpm/100 cm^2

SAMPLE GEL-1 COLLECTED UNDER VESSEL ON 1/12/00

NUCLIDE	Act pCi/g	HALF-LIFE years	Detectable Y/N	GL	f	F	f/GL
H-3	3.170E+04	1.228E+01	N	2.000E+05	9.808E-01		4.904E-06
Fe-55	1.670E+02	2.700E+00	N	2.000E+05	5.167E-03		2.584E-08
Pu-239/240	<	2.413E+04	N	1.000E+02	0.000E+00		0.000E+00
U-233/234	1.710E+00	1.592E+05	N	5.000E+03	5.291E-05		1.058E-08
U-238	1.650E+00	4.460E+09	Y	5.000E+03	5.105E-05	5.106E-05	1.021E-08
Ni-59	<	7.500E+04	N	5.000E+03	0.000E+00		0.000E+00
Cs-134	7.560E-01	2.062E+00	Y	5.000E+03	2.339E-05	2.339E-05	4.678E-09
Cs-137	1.210E+00	3.017E+01	Y	5.000E+03	3.744E-05	3.744E-05	7.488E-09
Co-60	1.530E+02	5.271E+00	Y	5.000E+03	4.734E-03	4.734E-03	9.468E-07
Eu-152	7.040E+00	1.360E+01	Y	5.000E+03	2.178E-04	2.178E-04	4.356E-08
Eu-154	1.410E+00	8.800E+00	Y	5.000E+03	4.363E-05	4.363E-05	8.725E-09
Mn-54	<	8.600E-01	N	5.000E+03	0.000E+00		0.000E+00
Ag-110m	<	6.800E-01	Y	5.000E+03	0.000E+00	0.000E+00	0.000E+00
Zn-65	<	6.700E-01	N	5.000E+03	0.000E+00		0.000E+00
Sr-90	7.350E-01	2.860E+01	Y	1.000E+03	2.274E-05	2.274E-05	2.274E-08
C-14	2.110E+02	5.730E+03	Y	5.000E+03	6.528E-03	6.529E-03	1.306E-06
Ni-63	6.930E+01	1.001E+02	N	5.000E+03	2.144E-03		4.288E-07
Tc-99	1.910E+00	2.130E+05	Y	5.000E+03	5.910E-05	5.910E-05	1.182E-08
	3.232E+04				9.999E-01	1.172E-02	7.731E-06

SPGL 1.516E+03

SITE SPECIFIC GUIDELINE, SPGL, BASED ON GEL-2 (TOTAL [illegible]IVITY)

RADIONUCLIDE SPECIFIC GUIDELINE FOR H-3 AND Fe-55 RAISED TO 200,000 dpm/100 cm^2

SAMPLE GEL-2 COLLECTED AT DOOR TO PIT ON 1/12/00

NUCLIDE	Act pCi/g	HALF-LIFE years	Detectable Y/N	GL	f	F	f/GL
H-3	1.390E+02	1.228E+01	N	2.000E+05	1.382E-01		6.909E-07
Fe-55	4.680E+02	2.700E+00	N	2.000E+05	4.652E-01		2.326E-06
Pu-239/240	<	2.413E+04	N	1.000E+02	0.000E+00		0.000E+00
U-233/234	9.520E-01	1.592E+05	N	5.000E+03	9.463E-04		1.893E-07
U-238	6.850E-01	4.460E+09	Y	5.000E+03	6.809E-04	6.809E-04	1.362E-07
Ni-59	<	7.500E+04	N	5.000E+03	0.000E+00		0.000E+00
Cs-134	<	2.062E+00	Y	5.000E+03	0.000E+00	0.000E+00	0.000E+00
Cs-137	2.460E+00	3.017E+01	Y	5.000E+03	2.445E-03	2.445E-03	4.891E-07
Co-60	2.340E+02	5.271E+00	Y	5.000E+03	2.326E-01	2.326E-01	4.652E-05
Eu-152	1.460E+00	1.360E+01	Y	5.000E+03	1.451E-03	1.451E-03	2.903E-07
Eu-154	<	8.800E+00	Y	5.000E+03	0.000E+00	0.000E+00	0.000E+00
Mn-54	<	8.600E-01	N	5.000E+03	0.000E+00		0.000E+00
Ag-110m	<	6.800E-01	Y	5.000E+03	0.000E+00	0.000E+00	0.000E+00
Zn-65	<	6.700E-01	N	5.000E+03	0.000E+00		0.000E+00
Sr-90	<	2.860E+01	Y	1.000E+03	0.000E+00	0.000E+00	0.000E+00
C-14	7.740E+01	5.730E+03	Y	5.000E+03	7.694E-02	7.694E-02	1.539E-05
Ni-63	8.200E+01	1.001E+02	N	5.000E+03	8.151E-02		1.630E-05
Tc-99	<	2.130E+05	Y	5.000E+03	0.000E+00	0.000E+00	0.000E+00
	1.006E+03				1.000E+00	3.141E-01	8.233E-05
			SPGL	3.815E+03			

SITE SPECIFIC GUIDELINE, SPGL, BASED ON GEL-3 (TOTAL ACTIVITY)

RADIONUCLIDE SPECIFIC GUIDELINE FOR H-3 AND Fe-55 RAISED TO 200,000 dpm/100 cm^2

SAMPLE GEL-3 RESIN SAMPLE COLLECTED 1/12/00

NUCLIDE	Act pCi/g	HALF-LIFE years	Detectable Y/N	GL	f	F	f/GL
H-3	4.880E+07	1.228E+01	N	2.000E+05	9.827E-01		4.913E-06
Fe-55	2.020E+03	2.700E+00	N	2.000E+05	4.068E-05		2.034E-10
Pu-239/240	9.460E-01	2.413E+04	N	1.000E+02	1.905E-08		1.905E-10
U-233/234	6.860E-01	1.592E+05	N	5.000E+03	1.381E-08		2.763E-12
U-238	<	4.460E+09	Y	5.000E+03	0.000E+00	0.000E+00	0.000E+00
Ni-59	2.690E+01	7.500E+04	N	5.000E+03	5.417E-07		1.083E-10
Cs-134	1.700E+01	2.062E+00	Y	5.000E+03	3.423E-07	3.423E-07	6.847E-11
Cs-137	4.220E+03	3.017E+01	Y	5.000E+03	8.498E-05	8.498E-05	1.700E-08
Co-60	1.220E+04	5.271E+00	Y	5.000E+03	2.457E-04	2.457E-04	4.913E-08
Eu-152	1.310E+01	1.360E+01	Y	5.000E+03	2.638E-07	2.638E-07	5.276E-11
Eu-154	<	8.800E+00	Y	5.000E+03	0.000E+00	0.000E+00	0.000E+00
Mn-54	5.330E+01	8.600E-01	N	5.000E+03	1.073E-06		2.147E-10
Ag-110m	1.220E+02	6.800E-01	Y	5.000E+03	2.457E-06	2.457E-06	4.913E-10
Zn-65	1.500E+02	6.700E-01	N	5.000E+03	3.021E-06		6.041E-10
Sr-90	6.570E+03	2.860E+01	Y	1.000E+03	1.323E-04	1.323E-04	1.323E-07
C-14	8.320E+05	5.730E+03	Y	5.000E+03	1.675E-02	1.675E-02	3.351E-06
Ni-63	4.800E+03	1.001E+02	N	5.000E+03	9.666E-05		1.933E-08
Tc-99	1.140E+02	2.130E+05	Y	5.000E+03	2.296E-06	2.296E-06	4.591E-10
	4.966E+07				1.000E+00	1.722E-02	8.484E-06

SPGL 2.030E+03

SITE SPECIFIC GUIDELINE, SPGL, BASED ON GEL-1(REMOVABLE ACTIVITY)

RADIONUCLIDE SPECIFIC GUIDELINE FOR H-3 AND Fe-55 RAISED TO 10,000 dpm/100 cm^2

SAMPLE GEL-1 COLLECTED UNDER VESSEL ON 1/12/00

NUCLIDE	Act pCi/g	HALF-LIFE years	Detectable Y/N	GL	f	F	f/GL
H-3	3.170E+04	1.228E+01	N	1.000E+04	9.808E-01		9.808E-05
Fe-55	1.670E+02	2.700E+00	N	1.000E+04	5.167E-03		5.167E-07
Pu-239/240	<	2.413E+04	N	2.000E+01	0.000E+00		0.000E+00
U-233/234	1.710E+00	1.592E+05	N	1.000E+03	5.291E-05		5.291E-08
U-238	1.650E+00	4.460E+09	Y	1.000E+03	5.105E-05	5.105E-05	5.105E-08
Ni-59	<	7.500E+04	N	1.000E+03	0.000E+00		0.000E+00
Cs-134	7.560E-01	2.062E+00	Y	1.000E+03	2.339E-05	2.339E-05	2.339E-08
Cs-137	1.210E+00	3.017E+01	Y	1.000E+03	3.744E-05	3.744E-05	3.744E-08
Co-60	1.530E+02	5.271E+00	Y	1.000E+03	4.734E-03	4.734E-03	4.734E-06
Eu-152	7.040E+00	1.360E+01	Y	1.000E+03	2.178E-04	2.178E-04	2.178E-07
Eu-154	1.410E+00	8.800E+00	Y	1.000E+03	4.363E-05	4.363E-05	4.363E-08
Mn-54	<	8.600E-01	N	1.000E+03	0.000E+00		0.000E+00
Ag-110m	<	6.800E-01	Y	1.000E+03	0.000E+00	0.000E+00	0.000E+00
Zn-65	<	6.700E-01	N	1.000E+03	0.000E+00		0.000E+00
Sr-90	7.350E-01	2.860E+01	Y	2.000E+02	2.274E-05	2.274E-05	1.137E-07
C-14	2.110E+02	5.730E+03	Y	1.000E+03	6.528E-03	6.528E-03	6.528E-06
Ni-63	6.930E+01	1.001E+02	N	1.000E+03	2.144E-03		2.144E-06
Tc-99	1.910E+00	2.130E+05	Y	1.000E+03	5.910E-05	5.910E-05	5.910E-08
	3.232E+04				9.999E-01	1.172E-02	1.126E-04
			SPGL	1.041E+02			

SITE SPECIFIC GUIDELINE, SPGL, BASED ON GEL-2 (REMOV. ..VLE ACTIVITY)

RADIONUCLIDE SPECIFIC GUIDELINE FOR H-3 AND Fe-55 RAISED TO 10,000 dpm/100 cm^2

SAMPLE GEL-2 COLLECTED AT DOOR TO PIT ON 1/12/00

NUCLIDE	Act pCi/g	HALF-LIFE years	Detectable Y/N	GL	f	F	f/GL
H-3	1.390E+02	1.228E+01	N	1.000E+04	1.382E-01		1.382E-05
Fe-55	4.680E+02	2.700E+00	N	1.000E+04	4.652E-01		4.652E-05
Pu-239/240	<	2.413E+04	N	2.000E+01	0.000E+00		0.000E+00
U-233/234	9.520E-01	1.592E+05	N	1.000E+03	9.463E-04		9.463E-07
U-238	6.850E-01	4.460E+09	Y	1.000E+03	6.809E-04	6.809E-04	6.809E-07
Ni-59	<	7.500E+04	N	1.000E+03	0.000E+00		0.000E+00
Cs-134	<	2.062E+00	Y	1.000E+03	0.000E+00	0.000E+00	0.000E+00
Cs-137	2.460E+00	3.017E+01	Y	1.000E+03	2.445E-03	2.445E-03	2.445E-06
Co-60	2.340E+02	5.271E+00	Y	1.000E+03	2.326E-01	2.326E-01	2.326E-04
Eu-152	1.460E+00	1.360E+01	Y	1.000E+03	1.451E-03	1.451E-03	1.451E-06
Eu-154	<	8.800E+00	Y	1.000E+03	0.000E+00	0.000E+00	0.000E+00
Mn-54	<	8.600E-01	N	1.000E+03	0.000E+00		0.000E+00
Ag-110m	<	6.800E-01	Y	1.000E+03	0.000E+00	0.000E+00	0.000E+00
Zn-65	<	6.700E-01	N	1.000E+03	0.000E+00		0.000E+00
Sr-90	<	2.860E+01	Y	2.000E+02	0.000E+00	0.000E+00	0.000E+00
C-14	7.740E+01	5.730E+03	Y	1.000E+03	7.694E-02	7.694E-02	7.694E-05
Ni-63	8.200E+01	1.001E+02	N	1.000E+03	8.151E-02		8.151E-05
Tc-99	<	2.130E+05	Y	1.000E+03	0.000E+00	0.000E+00	0.000E+00
	1.006E+03				1.000E+00	3.141E-01	4.569E-04

SPGL 6.875E+02

SITE SPECIFIC GUIDELINE, SPGL, BASED ON GEL-3 (REMOVABLE ACTIVITY)

RADIONUCLIDE SPECIFIC GUIDELINE FOR H-3 AND Fe-55 RAISED TO 10,000 dpm/100 cm^2

SAMPLE GEL-3 RESIN SAMPLE COLLECTED 1/12/00

NUCLIDE	Act pCi/g	HALF-LIFE years	Detectable Y/N	GL	f	F	f/GL
H-3	4.880E+07	1.228E+01	N	1.000E+04	9.827E-01		9.827E-05
Fe-55	2.020E+03	2.700E+00	N	1.000E+04	4.068E-05		4.068E-09
Pu-239/240	9.460E-01	2.413E+04	N	2.000E+01	1.905E-08		9.525E-10
U-233/234	6.860E-01	1.592E+05	N	1.000E+03	1.381E-08		1.381E-11
U-238	<	4.460E+09	Y	1.000E+03	0.000E+00	0.000E+00	0.000E+00
Ni-59	2.690E+01	7.500E+04	N	1.000E+03	5.417E-07		5.417E-10
Cs-134	1.700E+01	2.062E+00	Y	1.000E+03	3.423E-07	3.423E-07	3.423E-10
Cs-137	4.220E+03	3.017E+01	Y	1.000E+03	8.498E-05	8.498E-05	8.498E-08
Co-60	1.220E+04	5.271E+00	Y	1.000E+03	2.457E-04	2.457E-04	2.457E-07
Eu-152	1.310E+01	1.360E+01	Y	1.000E+03	2.638E-07	2.638E-07	2.638E-10
Eu-154	<	8.800E+00	Y	1.000E+03	0.000E+00	0.000E+00	0.000E+00
Mn-54	5.330E+01	8.600E-01	N	1.000E+03	1.073E-06		1.073E-09
Ag-110m	1.220E+02	6.800E-01	Y	1.000E+03	2.457E-06	2.457E-06	2.457E-09
Zn-65	1.500E+02	6.700E-01	N	1.000E+03	3.021E-06		3.021E-09
Sr-90	6.570E+03	2.860E+01	Y	2.000E+02	1.323E-04	1.323E-04	6.615E-07
C-14	8.320E+05	5.730E+03	Y	1.000E+03	1.675E-02	1.675E-02	1.675E-05
Ni-63	4.800E+03	1.001E+02	N	1.000E+03	9.666E-05		9.666E-08
Tc-99	1.140E+02	2.130E+05	Y	1.000E+03	2.296E-06	2.296E-06	2.296E-09
				1.000E+03			
	4.966E+07				1.000E+00	1.722E-02	1.161E-04

SPGL 1.483E+02