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REPORT

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TITLE:
Radiological Survey of Building T005

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ABSTRACT

A radiological survey was performed at Building T005, located at Rockwell International's Santa Susana Field Laboratory (SSFL), to clarify and identify those areas needing further radiological inspection and requiring remedial action. Building T005, previously known as the Uranium Carbide Pilot Fuel Facility, was operated by North American Aviation and its successors for the Chicago Operations Office of the Atomic Energy Commission to cast and machine UC slugs (12.7 wt % enriched in U-235) during the mid to late 1960's. Following the completion of this program, equipment was removed and surfaces decontaminated to permit non-radiologic use of the facility as laboratory and office space.

Concurrent with and independent of the radiological survey, a few radioactive waste drain lines and exhaust vents were removed in an effort to prepare the facility for unrestricted use. These tasks were not completed. The majority of Building T005 is not contaminated with radioactive material. The survey results show that rooms 110 and 113, and the radioactive material exhaust ducts and filter plenums, are contaminated with enriched uranium, at levels above release limits prescribed in "Guidelines for Residual Radioactivity at FUSRAP and Remote SFMP Sites (March 5, 1985)." [1] Before the building can be released for unrestricted use, these rooms and equipment items must be decontaminated, or removed and disposed of as radioactive waste. Those areas and equipment items determined to be contaminated qualify for DOE assistance under the Surplus Facilities Management Program (SFMP).

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1.0 INTRODUCTION

Located in the Simi Hills of Ventura County, California, Building T005 was designed, constructed, and operated by North American Aviation and its successors for various research, development and production projects. The facility was contaminated with radioactive material (enriched uranium) during a Government-sponsored nuclear fuel fabrication program from 1966 to 1967. A comprehensive radiological survey has been performed to clarify the radiological conditions at T005 and to identify those areas which require remedial action under DOE's Surplus Facilities Management Program (SFMP). The survey was conducted as prescribed in Rockwell documents "Radiological Survey Plan for SSFL" and "Long-Range Plan for Decommissioning Surplus Facilities at the Santa Susana Field Laboratories," References 4 and 5, respectively.

Building T005 was built in the late 1950's for testing thermodynamic characteristics of proposed coolants for the Organic Moderated Reactor Experiment (OMRE) and Piqua reactors. These projects did not involve the use of radioactive materials. During the mid to late 1960's, the facility was converted to fabricate enriched uranium carbide fuel for the Atomic Energy Commission's (AEC's) Heavy-Water Organic-Cooled Reactor (HWOCR). Work was performed under the authority of AEC's Chicago Operations Office (COO). AEC/COO policy at that time exempted from licensure the activities conducted at Building T005. At the completion of this fuel fabrication program in 1967, equipment was removed and surfaces decontaminated to permit non-radiologic use of the building. Beginning in 1972, T005 was used as a molten salt test facility. Building T005 currently contains offices and the control rooms for the molten salt test bed and Process Demonstration Unit (PDU), both of which are operated in the equipment yard.

The only radioactive material handled at T005 was enriched uranium. During this fuel fabrication program from 1966 to 1967, T005 was designated as the Uranium Carbide Pilot Fuel Facility. About 700 UC_X

cylinders, 0.25-in.-diameter by 3-in.-long, were fabricated. Highly enriched uranium (93.1 wt %) metal was homogenized with 4.9 wt % enriched uranium by induction melting and casting to produce 12.7 wt % enriched uranium slugs. Uranium carbide slugs were then synthesized by reacting uranium metal with graphite in an arc furnace. Finally, cast UC slugs were machined into cylinders.

This uranium fuel fabrication process was completed in 1967, at which time all process equipment was removed. Since the cleanup completion in 1971-72, a few offices have been used in support of the molten salt test loop; the remaining areas serve as storage and tool cribs. No radioactive materials have been handled at T005 since 1967.

As part of the DOE Site Survey (Ref. 4), a radiation survey was performed on building floors and walls (up to 3 m in height) to measure the average, maximum, and removable alpha surface activity; average, maximum, and removable beta surface activity; and ambient gamma exposure rate. Similarly, ceilings were surveyed in rooms where accessible by use of an 8-ft ladder. Soil, sludge, sediment, dust, paint, and other miscellaneous samples were taken as appropriate and analyzed for radioactive material. Special structural features such as light fixtures and exhaust vents were surveyed when applicable. Outside concrete equipment pads and the gutter leading from Building T005 to south of G Street were also surveyed.

Survey areas were gridded in 3-m by 3-m sections. 1 m^2 out of each 9 m^2 section was analyzed for radioactive material. A specific 1 m^2 survey location within a 9 m^2 grid was biased toward an area suspect of potential residual contamination based on operational history and present appearance. Where radioactive contamination was found to exceed 50% of the acceptable contamination limit, a few additional locations in the immediate area were surveyed. Survey data were input into a PC computer graphics program which plots the radiation measurement value against cumulative probability for a given sample lot. The statistical tests applied to the survey, inspection by variables, are based on a consumer's risk of 0.1 at

10% probability, and assumes that the data follow a Gaussian probability distribution function.

The Department of Energy has adopted surface contamination limits established in "Guidelines for Residual Radioactivity at FUSRAP and Remote SFMP Sites," (Ref. 1). These limits are in agreement with U.S. NRC Regulatory Guide 1.86, the NRC "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material," and ANSI draft N13.12 as suitable for release of equipment and facilities for unrestricted use. These limits were used as nominal contamination values for the DOE Site Survey. Except for the radioactive material exhaust ducts, filter plenums, and drain lines (and rooms 110 and 113), the statistical analyses show no evidence that the building is not acceptably clean for release for unrestricted use. The equipment yard and surface and subsurface drainage gutter are also free of radioactive material contamination.

2.0 IDENTIFICATION OF FACILITY PREMISES

2.1 Location

Building T005 is located within Rockwell International's Santa Susana Field Laboratory (SSFL) in the Simi Hills of southeastern Ventura County, California, adjacent to the Los Angeles County line and approximately 29 miles northwest of downtown Los Angeles. The SSFL location relative to the Los Angeles area and surrounding vicinity is shown in Figure 2.1, Figure 2.2 is an enlarged map of neighboring SSFL communities, and Figure 2.3 is a map showing that part of SSFL which includes Building T005. Building T005 is not located on the Government-optioned land, but is Rockwell-owned.

2.2 Building Characteristics

As outlined in Rockwell International document 154SRR000001, "Radiological Survey Plan for SSFL," the premises surveyed consist of Building T005, equipment yards surrounding the building but within the facility exclusion area, the roadside gutter, which runs south of the building about 1000 ft to G Street, and the surface drainage area immediately south of G Street. Figure 2.4 is a site plan of the areas surveyed.

Building T005 is a tilt-up concrete structure with Butler aluminum siding and a few windows, 80 ft long (running north and south), and 60 ft wide. The building is divided into a small administration area, change rooms, chemistry and other service laboratories, and a large high-bay area. The architectural plan is shown in Figure 2.5. Figure 2.6 is a photo of Building T005 as viewed from the front (looking from 17th Street). During its use as the Uranium Carbide Pilot Fuel Facility, the building was a radiologically controlled access area. The administration area entry was uncontrolled. The remaining areas were controlled, of which one part required protective clothing. Figure 2.7 demonstrates this division.

Figure 2.1. Map of Los Angeles Area

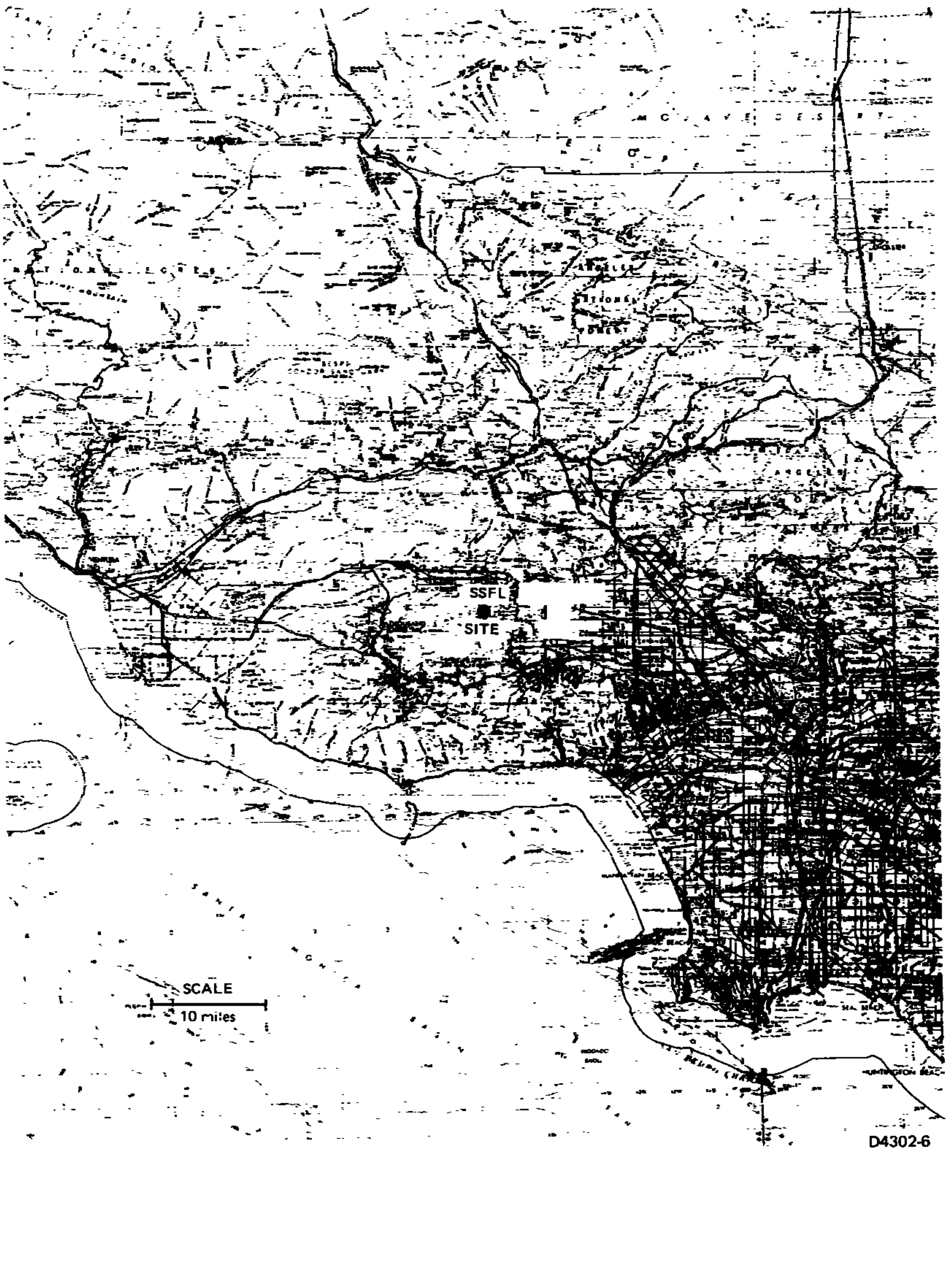
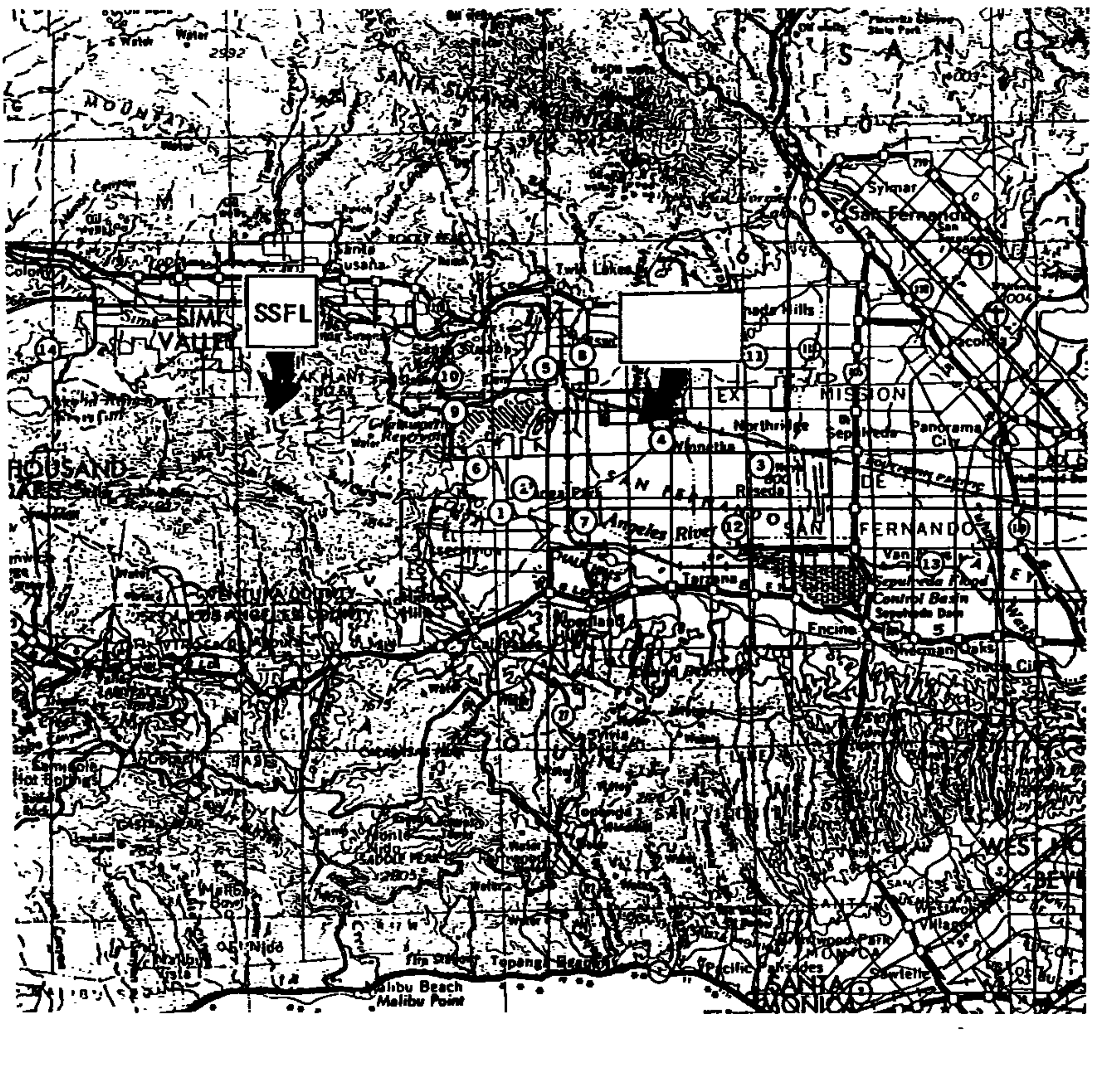


Figure 2.2. Map of Neighboring SSFL Communities



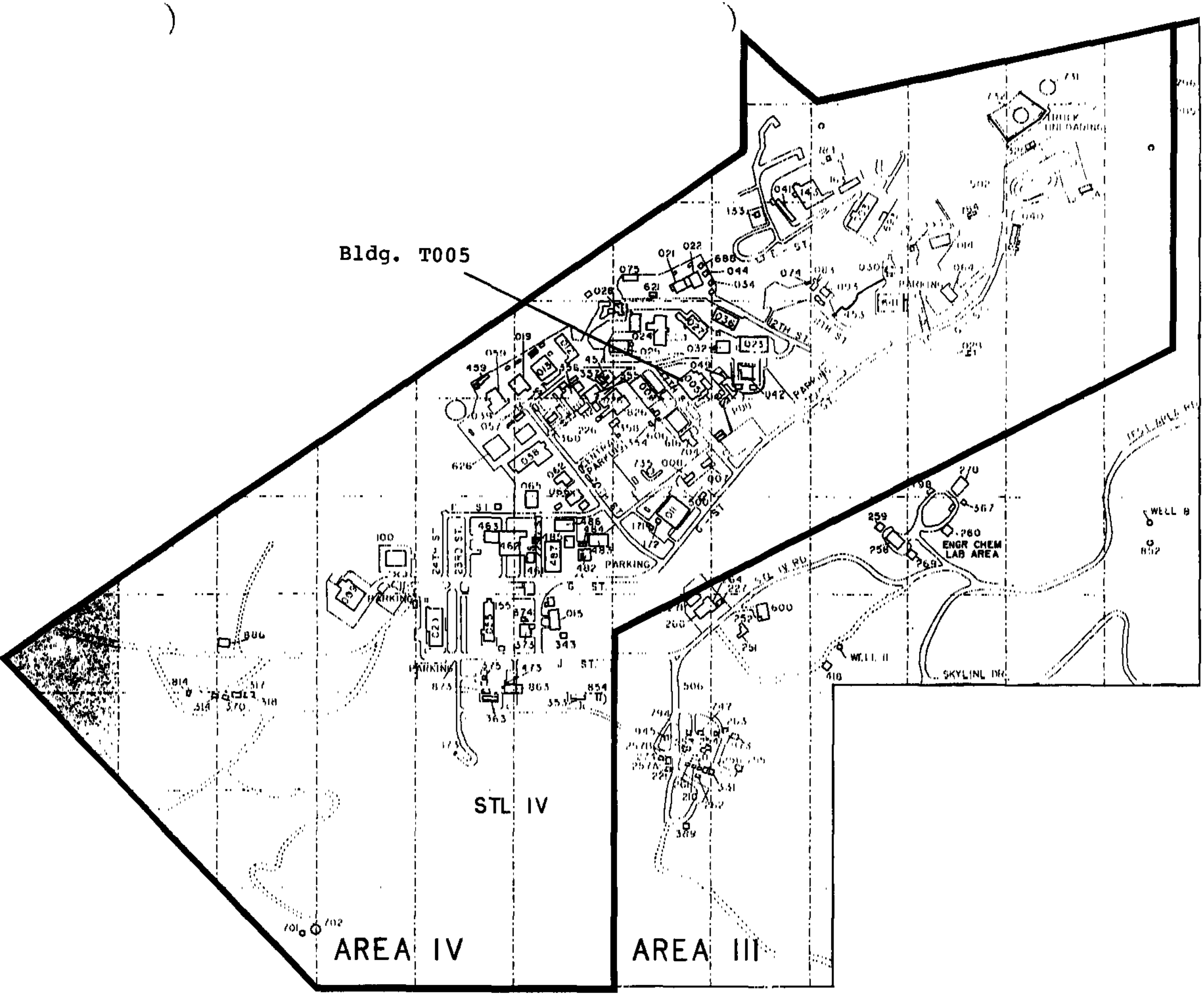


Figure 2.3. SSFL Layout

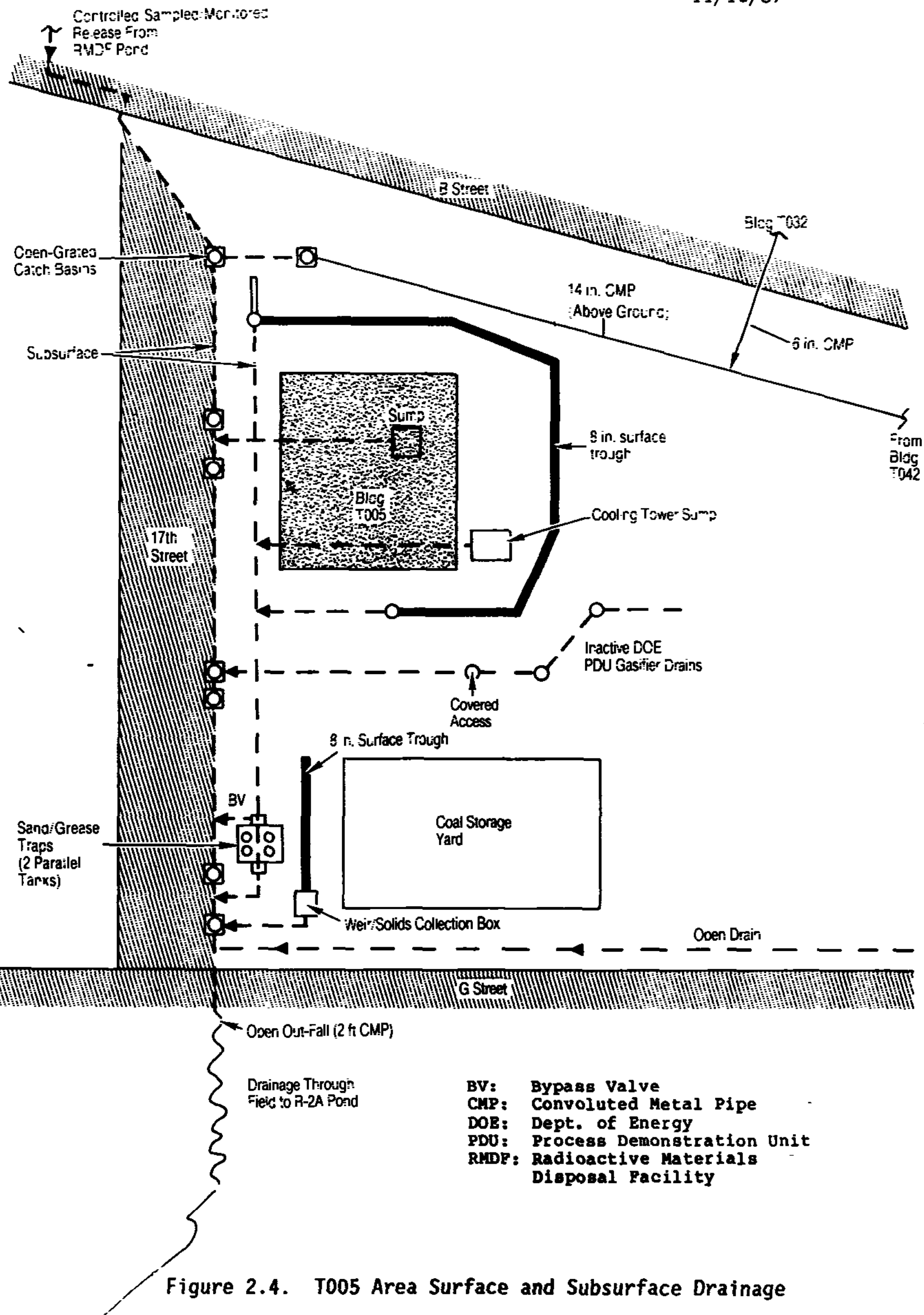
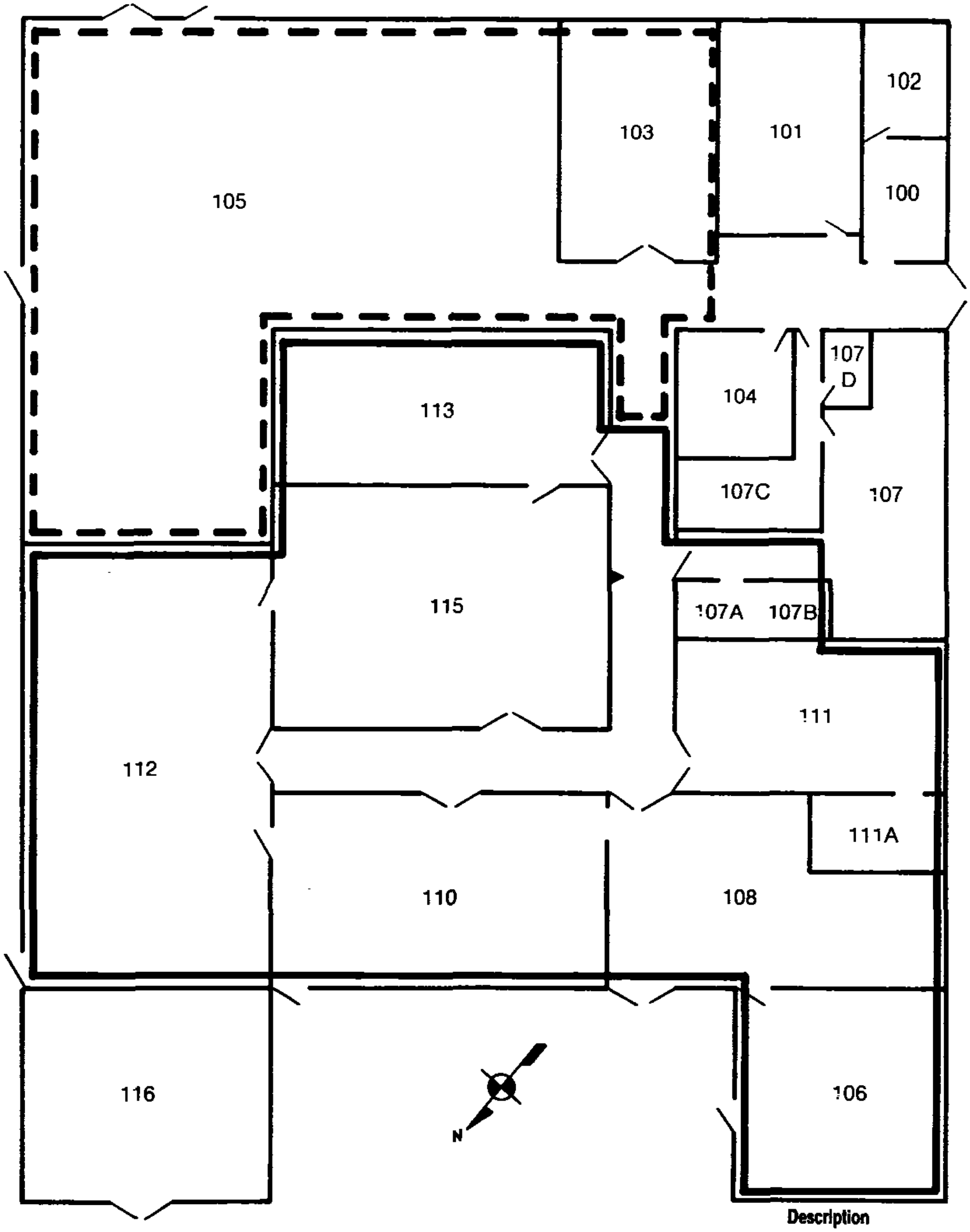


Figure 2.4. T005 Area Surface and Subsurface Drainage



Controlled Area Which Required the Use of Protective Clothing

Controlled Area Where Protective Clothing Was Not Required

Figure 2.5. T005 Floor Plan Showing Controlled Areas

Room	Description: When Used During UCPFF	Present
100	Secretary's Office	Office
101	Supervisor's Office	Office
102	Lunch and Conference Room	Same
103	Chemical Processing	Chem Lab
104	Health Physics	Women's Restroom
105	Cladding Machine, End Cap Weld, and Loading Areas	Storage
106	Storage	Storage
107	Restroom, Hot/Cold Change Rooms	Change and Restroom
108	Blending	Vacant
110	U02 to UC Conversion	Molten Salt Computer Control Center
111	Quality Control	Vacant
111A	Quality Control	Vacant
112	Slug Casting	Storage
113	Inspection Batching and Loading	Equipment Drib
115	Fuel Slug Machining	Refurbished (Vacant)
116	Silicon Rectifier	Tool Crib

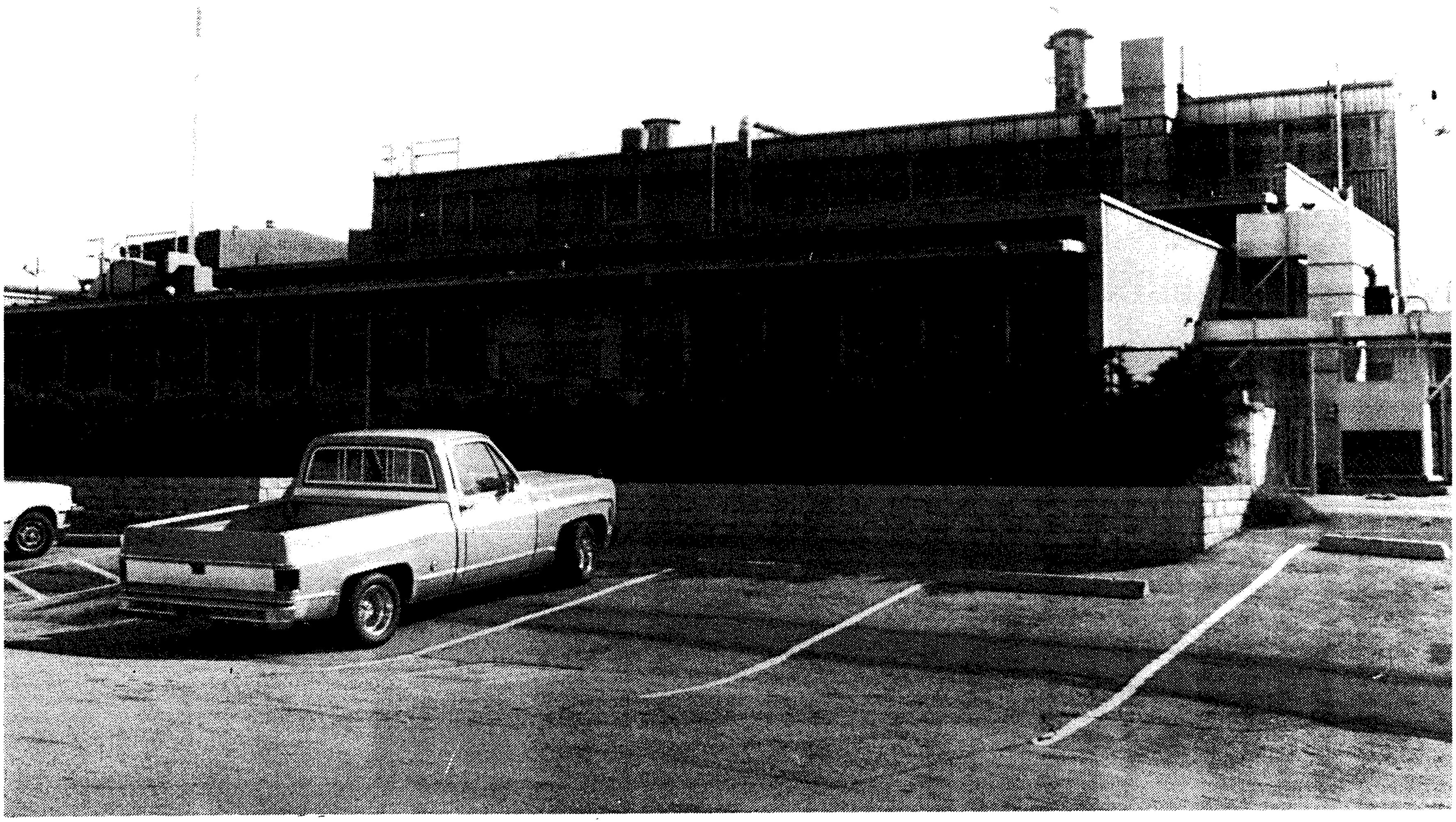
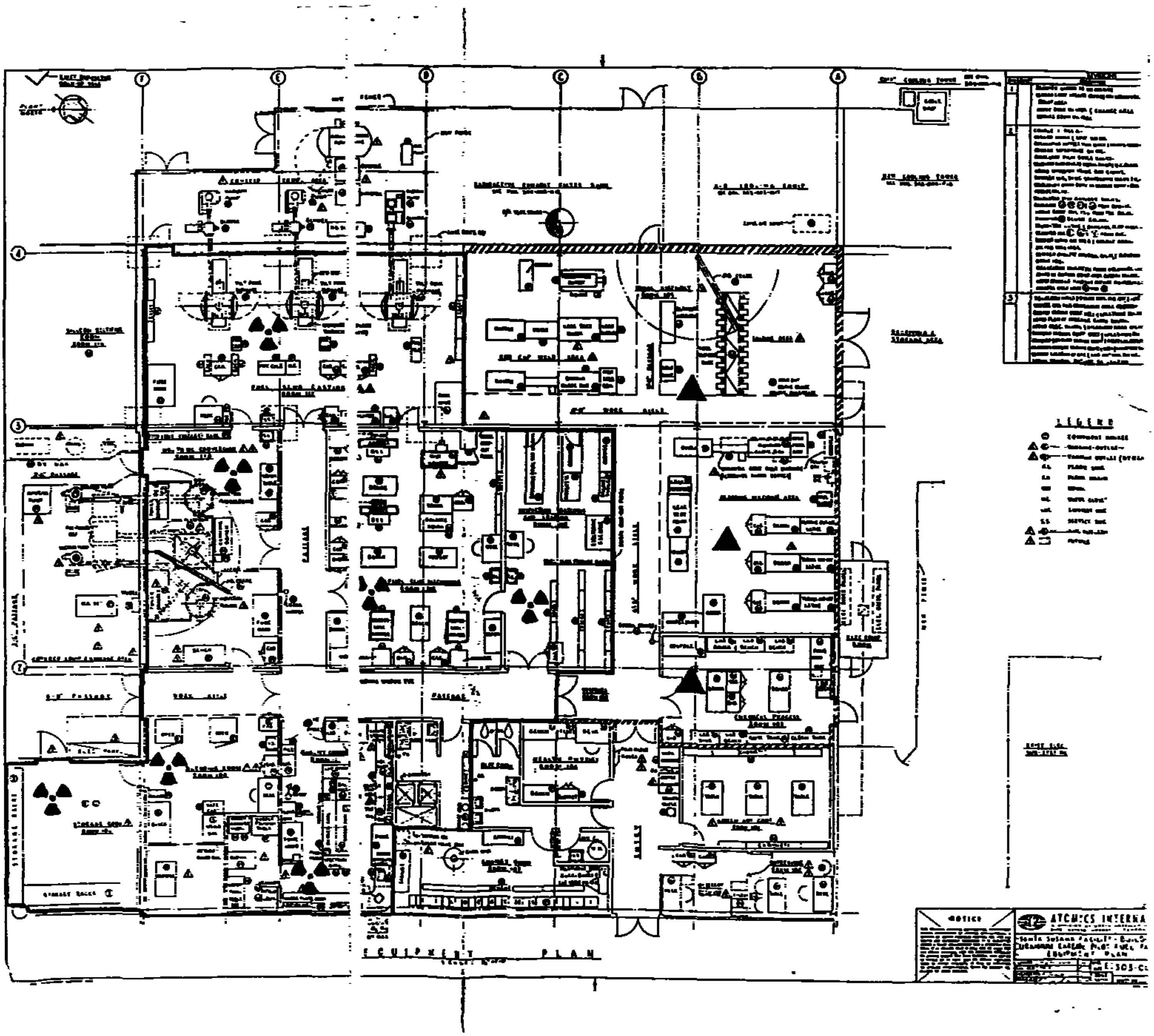


Figure 2.6. Front View of Building T005

Figure 2.7. Building T005 Equipment Plan
During Use as the Uranium Carbide Pilot Fuel Facility



The building is constructed of non-combustible materials on a concrete slab floor. During use, the floors had 9-in. polyvinyl tile covering. Following the completion of the fuel fabrication program in 1967, equipment was removed and surfaces decontaminated to permit non-radiologic use of the building. Decontamination and cleanup work are still in progress at a slow pace because the remaining equipment and building areas to be removed or decontaminated are not hazards to personnel currently working there. The only systems currently in place, which remain from the fuel fabrication operating era are: air conditioning and heating in a small portion of the building; radioactive exhaust system ducts in the attic and northern patio area; radioactive exhaust filter plenums; a few radioactive liquid waste drain lines; electrical power distribution; communication equipment; fire control; and plumbing. Section 3.0 provides information regarding the decontamination effort, equipment removed, radiological problems encountered, and final condition of the building prior to the radiological survey conducted August through October, 1987.

2.3 Radiological Condition

The only radioactive material handled in the facility was enriched uranium. Figure 2.7 shows a layout, room descriptions, and equipment used during the fuel fabrication effort. Most of the fuel machining and grinding was performed in room 115. Room 111A had a hooded grinding wheel, but the extent of its use is unknown. These two areas are the most highly suspect for residual contamination because of the operations conducted there. Ovens and furnaces for blending and casting were located in rooms 108, 110, and 112. These rooms are not highly suspect of containing loose contamination because of the operations conducted there. However, it is recognized that spreading of contamination from either room 115 or 111A was possible. Rooms 105, 106, 111, and 113 were used for packaging, storage, quality assurance, and inspection. No operations were conducted in these rooms are not suspected of leaving contamination. The administrative offices are not suspect of contamination. Certainly, all radioactive material drain lines, exhaust ducts, plenums, and filter banks are suspect of containing contamination.

Batches of 4.9 wt % U-235 and 93.1 wt % U-235 were blended to produce 12.7 wt % U-235 metal slugs, which were then cast into uranium carbide (UC) fuel slugs. Because U-234, a daughter product of U-238, behaves in the same physical manner as U-235 and because its half life is much shorter than either U-235 or U-238, its radioactivity concentration in 93.1 wt % enriched uranium is very large. The majority of radioactivity is enriched uranium is consequently due to U-234. The only radionuclides over handled were U-234, U-235, U-238, and their immediate daughters. By simplifying the complex decay schemes of U-234, U-235 and U-238, it is obvious that for every U-234 disintegration, one alpha particle is emitted directly. For every U-235 disintegration, one alpha and one beta particle are emitted. For every U-238 disintegration, one alpha and two beta particles are emitted. All three decay by alpha emissions with nominal energies of 4 MeV 100% of the time. No beta particles are emitted by U-234 or its daughters. Accompanying a U-235 disintegration, is one equivalent beta with various maximum energies ranging from 140 keV to 300 keV. This beta is actually emitted from Th-321 which is in equilibrium with U-235. Accompanying a U-238 disintegration is a Th-234 beta (Emax = 100 to 200 keV), and a Pa-234m beta (Emax = 2.29 MeV). This radioactivity is detectable by a competent survey.

Additionally, this contamination is qualifiable by gamma spectrometry techniques due to gamma emission accompanying U-235 and U-238 disintegrations. Detectable, resolvable 185 keV, and 205 keV gamma rays are emitted by U-235. A 93 keV gamma ray is emitted by U-238. Areas which are identified during the field survey as contaminated can be wiped and analyzed by gamma spectrometry for these gamma emissions.

During fuel production, the facility was operated in a radiologically controlled manner under the cognizance of an on-site health physicist. Considerable difficulty was experienced with the performance of the air exhaust system scrubbers and filters. This included a fire. Special care has been taken during the dismantling of all radioactive material exhaust ducts not to contaminate surrounding areas. This objective has been successful. The scope of this radiological survey included internal contamination measurements of exhaust ducts and filter plenums. Special treatment was given during the survey to other areas where contamination was suspected. No evidence suggests that the building ever experienced contamination incidents with consequences taking place outside of the exhaust ducts.

3.0 DECONTAMINATION EFFORTS

After the fuel fabrication program was completed in 1967, all rooms and areas needed to support the molten salt test facility were decontaminated to permit non-radiologic use of the facility. All associated project equipment was removed from the building. Since that time, the following decontamination efforts have been made:

* The wall between rooms 105 and 112 was removed to permit use of the entire high bay as a storage area as necessary.

* The floor tile in rooms 105, 112, 116, 111, 111A, 115, and 108 has been removed.

* Wall baseboards and coving have been removed.

* Radioactive material exhaust outlets have been cut and capped.

* Both radioactive liquid waste holdup tanks (500 gal each) have been removed.

* Some of the radioactive liquid drain lines have been removed.

All decontamination operations have been performed under the cognizance of Radiation & Nuclear Safety. Additionally, asbestos analysis was performed in suspect areas. No asbestos was found during tile removal, The underlying floor was surveyed to ensure that it was not contaminated. During the cutting and capping of exhaust ducts, surveys were conducted to ensure that the operation did not spread loose contamination to surrounding areas. During the late 1970s, both radioactive liquid holdup tanks were removed and disposed of. The excavation was backfilled and resurfaced with macadam blacktop. No personnel recollection or records show that the soil beneath the tanks was sampled and checked for radioactivity prior to backfilling. Radioactive liquid drain lines from T005 to the holdup tank location are still in place. Radioactive waste generated during all phases of the decontamination effort was disposed of according to appropriate regulations at the time.

4.0 SURVEY SCOPE

A sampling inspection plan using variables, discussed in section 5.0, was used to compare residual contamination quantities in the building against unrestricted-use acceptable contamination limits prescribed in DOE Guidelines (Ref. 1), Reg. Guide 1.86, and 10CFR20:

Table 4.1 Building T005 Maximum Acceptable Contamination Limits

Criteria	Alpha ($dpm/100cm^2$)	Beta ($dpm/100cm^2$)
Total, averaged over 1 m^2	5000	5000
Maximum in 1 m^2	15000	15000
Removable over 100 cm^2	1000	1000
Ambient Gamma Exposure Rate*	5uR/h above background	
Soil Activity Concentration**	56 pCi/g	100 pCi/g
Water Activity Concentration***	$1x10^{-4}$ uCi/ml	$1x10^{-5}$ uCi/ml

* Although DOE Guide (Ref. 1) recommends a value of 20 R/h above background for ambient gamma exposure rate, NRC has required 5 uR/h. For conservatism, 5 uR/h was used to compare survey results.

** Alpha activity concentration limits for enriched uranium is 30 pCi/g plus that contribution from naturally occurring activity, (about 26 pCi/g from Reference 15, p. 66). The total beta activity concentration limit is 100 pCi/g, including background (Ref. 13).

*** The most restrictive alpha/beta water radioactivity concentrations for restricted area taken from 10CFR20, Table 1, Column 2. Alpha corresponds to Pu-240, beta to Sr-90.

Three specific action levels were established for the survey:

1. Characterization Level - that level of radioactivity which is below 50% of the maximum acceptable limit. This level is

typical of natural background levels, or slightly above, and requires no further action.

2. Reinspection Level - that level of radioactivity which is above 50% of the maximum acceptable limit. A general resurvey of the area and a few additional samples are required in this case.

3. Investigation Level - that level of radioactivity which exceeds 90% of the maximum acceptable limit. Specific investigation of the occurrence is required in this case.

Direct radiation measurements were made for total average, and maximum, alpha/beta contamination, and ambient gamma exposure rate. Indirect radiation measurements were made for removable alpha/beta contamination. Soil, sediment, paint, and dust samples were analyzed by alpha/beta gas proportional methods and gamma spectrometry.

4.1 Sample Lots

For purposes of the radiological survey, the building and associated surrounding areas were divided into eight sampling lots. Each sampling lot was selected based on similar building characteristics, i.e. areas where radiological conditions and physical surroundings were thought to be similar were selected as a sampling lot. If upon further inspection a particular area within a sampling lot was determined not to be representative of the lot, that area was included in a more representative sampling lot. Distinguishable properties for selecting a sampling lot were: 1) areas thought to be clean of radiological contamination, 2) areas suspect of contamination, 3) areas of similar material or structure, and 4) soil.

The eight sampling lots were:

1. Non-suspect contamination area: Rooms 100, 101, 102, 103, 105, 112, 115, and 116.

2. Non-suspect contamination area: Rooms 104, 106, 107, 108, 110W, 111, 111A, hall1, and hall2.

3. Suspect contamination area: Rooms 110E, and 113.

4. Non-suspect contamination area: Concrete and black-top surfaces outside of building T005.

5. Non-suspect special structure: External removable contamination from pipes, I-beams, light fixtures, and ventilation and exhaust ducts.

6. Suspect special structures: Removable contamination inside radioactive material exhaust ducts.

7. Suspect special structures: Large radioactive material exhaust filter plenum.

8. Soil and sediment samples collected along water course from T005 to south of G Street.

Figure 4.1 shows the survey sampling lot plan. For each indoor sampling lot, a minimum of an 11% survey was conducted on every wall up to 10 feet in height, and an 11% survey of every floor. A 5.5% survey was conducted on all ceilings where the height was under 10 feet. Outdoor concrete pads and black-top were sampled with an 11% survey. The sampling inspection plan that was used is based upon a uniform 3-meter square grid ($9m^2$) superimposed on a uniform inspection area. A 3-meter square grid has been adopted to be consistent with NRC and State of California guidance for releasing a facility for unrestricted use. The grid was superimposed on walls, floors, and ceilings of each room. Each survey area was identified in matrix notation with codes indicating the surface (F = floor, C = ceiling, N, E, S, W = north, east, south, west, respectively) and a two figure Cartesian coordinate indicating the distance in meters from a local benchmark. The (1,1) position for the floor was benchmarked as the north-west corner of each room; an identical grid was reflected onto the ceiling. The (1,1) position of the walls was benchmarked as the top left hand corner of the wall as an observer would view it from the middle of the room. From each 3-square-meter grid ($9m^2$), a $1\text{-}m^2$ was surveyed; hence, an 11% survey. Each $1\text{-}m^2$ area was surveyed directly for alpha/beta/gamma contamination for 5 minutes. A 100 cm^2 wipe was taken in each selected $1\text{-}m^2$ for analysis of removable contamination.

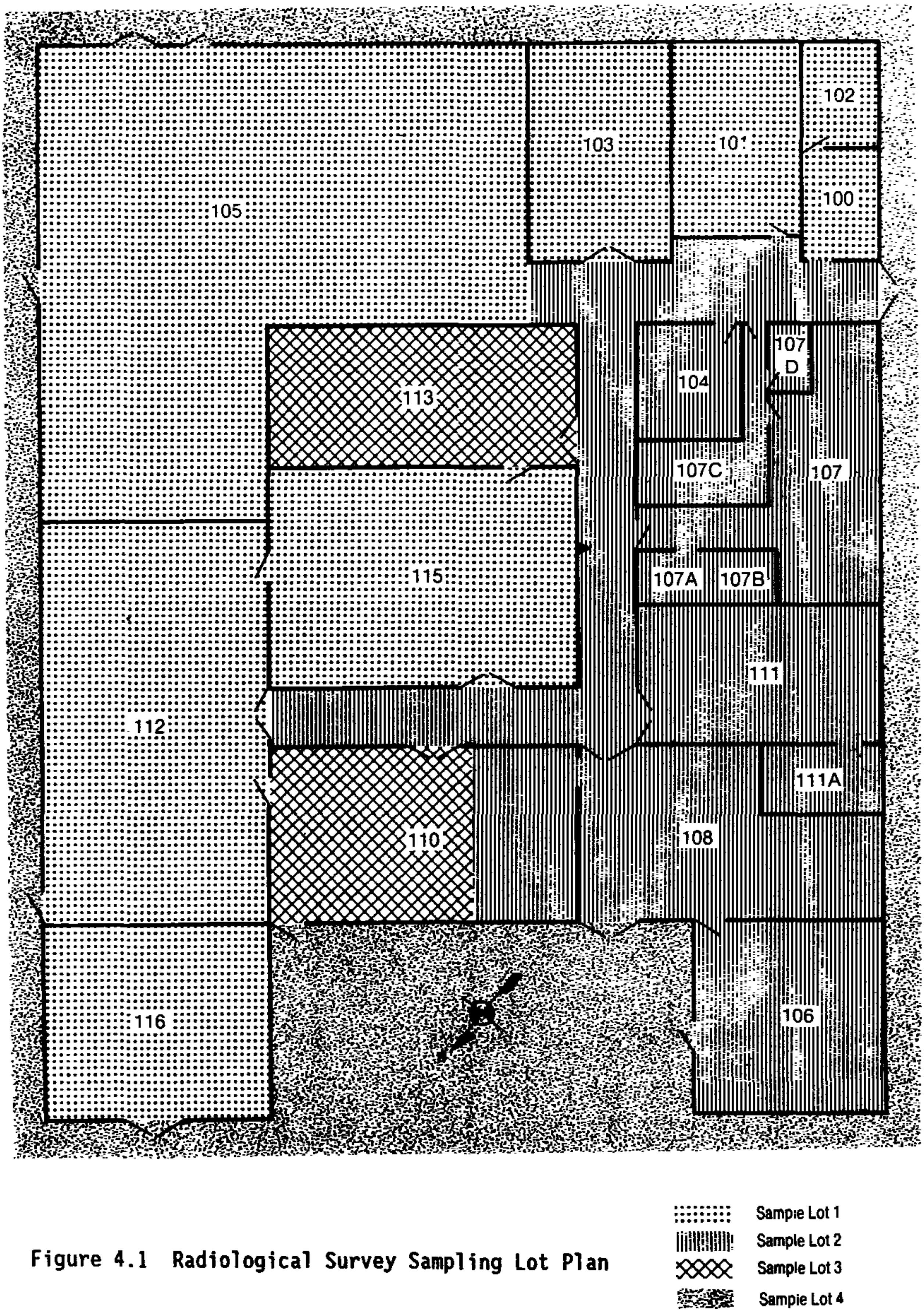


Figure 4.1 Radiological Survey Sampling Lot Plan

4.2 Data Acquisition

Within each 3-meter square grid, a single 1-m^2 area was surveyed. Each area was outlined by a marker, with its coordinates marked beside the area. The location of the 1-m^2 area was left to the surveyor's judgement: it was to be the area that, in his judgement, was most likely to have retained the most residual contamination of any similar area within the 3-m square grid. The surveyor was instructed to do this conscientiously to assure that any significant residual contamination would be detected. The use of a predetermined grid with discretion for the exact location provides a uniform survey biased towards the high end of the distribution. Selection of the 1-m^2 area out of the nine within each grid square provides an 11% sampling. If a particular surface of a room was smaller than 9-square-meters (3m x 3m), a minimum of 1-m^2 was surveyed for contamination.

4.2.1 Alpha and Beta Contamination

In order to determine the alpha/beta contamination level, four radiological characteristics were measured in each square meter; total average alpha surface activity, total average beta surface activity, removable alpha surface activity, and removable beta surface activity. An alpha probe and beta probe were each connected to a Ludlum Model 2220 portable scaler.

Measurements of the average alpha surface activity were made by use of a large-diameter (9.5 cm) alpha scintillation detector, sensitive only to alpha particles with energy exceeding about 1.5 MeV. This detector was calibrated by use of a Th-230 alpha source. The energy of the Th-230 alpha particles (4.6 MeV) is similar to that of the isotopes handled at T005; U-234, U-235, and U-238.

Measurements of the total average beta surface activity were made by use of a thin-window pancake Geiger-Mueller tube. While this detector is sensitive to alpha and beta particles and slightly sensitive to X- and gamma-rays, it is so predominately used to measure beta-activity that it is generally called a "beta-detector." This detector was calibrated by use of a Tc-99 beta source. The energy of the Tc-99 beta particles (maximum 0.3 MeV) is close to those from the U-238 daughters, Th-234 (maximum 0.2 MeV) and Pa-234m (maximum 2.29 MeV). The measurements were made over the same area as was used for each measurement of total average alpha surface activity.

Measurements of removable surface activity (alpha and beta) were made by wiping approximately 100 cm^2 of surface area, using a cloth disk (NPO cloth sampling smears 2 in. diameter). The activity on the disk was measured using a thin-window gas-flow proportional counter, calibrated with TH-230 and Tc-99 disk sources.

In order to ease the survey method, alpha and beta probes were connected by a face-plate such that the separation distance between probes was no greater than a couple of centimeters. Each square-meter was surveyed using the assembly for 5 minutes; this corresponds to a transit velocity of no greater than 3.3 cm/sec (ANSI draft standard N13.12). The standard states that the transit velocity (in cm/sec) when surveying for alpha contamination, shall not exceed one-third the numerical value of the detector window dimension (in cm) in the direction of the scan. The diameter of the Ludlum model 43-1 alpha probe is 10 cm. The number of counts registered by the instrument in a five minute scan was recorded by location. If a contaminated spot was detected during the course of the "average scan" survey, the location was identified; subsequently, a five minute stationary survey of that specific location was conducted. The average surface activity of the square meter, the maximum surface activity of one spot located within the square meter, and the removable surface activity of 100 cm^2 in the square meter were recorded.

Because the results must be reported in disintegrations per minute per 100 square-centimeters (dpm/100cm^2), conversion factors were applied as follows. First, "natural background" was determined by measurements made in an area of the building of similar construction known to be uncontaminated. Second, an efficiency factor of the survey instrument was calculated by comparing the number of counts recorded by the instrument to the number of disintegrations yielded by a calibration source. These determinations were made three times each day; first thing in the morning, at noon, and just before quitting time in the evening. Third, an area correction factor of the window was calculated in order to present results per 100 cm^2.

Thus, for the surface contamination measurements of alpha and beta activity, data included the sample location, the total counts recorded in the five minute scan, the maximum hot spot, natural background for five minutes, efficiency factor, and the area factor. The same data were recorded for removable contamination measurements except area factor, which is not applicable for the gas proportional detector since the measurement area refers to the area smeared.

4.2.2 Ambient Gamma Exposure Rate

Ambient gamma exposure rate measurements were made on each floor square-meter surveyed for alpha/beta contamination. The gamma radiations associated with U-238, U-235, and their daughters are not very energetic. Most of the predominant gamma radiations are less than 1 MeV. Consequently uranium contamination does not produce a very intense gamma field. The exposure rate measurements were taken to demonstrate compliance with the 5uR/h exposure rate limit currently imposed by NRC and to identify any trends or anomalies in the ambient gamma field. For example, exposure rate measurements may identify radioactive materials resulting from operations in nearby buildings.

Measurements of ambient gamma exposure rate were made by use of a NaI scintillation crystal coupled to a Ludlum Model 2220 portable scaler. This detector is calibrated quarterly by the calibration laboratory using Cs-137 as the calibration source. A detector efficiency plot is generated as a function of distance from the source. The efficiency in uR/h/cpm is fairly constant as a function of distance, with an average of .0068 uR/h/-cpm, (150 cpm/uR/h).

Instrument response was checked three times a day using a Ra-226 source. The source was placed 1 foot from the detector and counted for 5 min. If the scaler reading fell within ± 20% of the nominal value, then the instrument was qualified as operable for the day.

A tripod was built to support the detector at a distance of 1 meter from the floor surface. Gamma counts were collected for 5 min at each specified location and then converted to uR/h.

4.3 Data Reduction

The data were entered into VISICALC, a spread sheet software utility on the IBM PC. Columns were established to calculate the alpha/beta total-average, maximum, and removable contamination per 1-m^2 in dpm/100cm^2; and the floor ambient gamma exposure rate in uR/h. The standard deviation expected for each measurement was also calculated.

Data Input:

1. Room number
2. Grid location, ex. N(1,3).
3. Alpha total activity, averaged over 1-m^2 (counts in 5 min).
4. Alpha maximum activity for hot spot, if present (counts in 5 min).
5. Alpha removable activity from 100 cm^2 (counts in 5 min).
6. Beta total activity, averaged over 1-m^2 (counts in 5 min).
7. Beta maximum activity for hot spot, if present (counts in 5 min).
8. Beta removable activity from 100 cm^2 smear (counts in 5 min).
9. Ambient gamma exposure rate (counts in 5 min).

10. Alpha survey instrument background (5 min), efficiency factor (dpm/cpm), and area factor.
11. Alpha gas-proportional detector background (5 min) and efficiency factor (dpm/cpm).
12. Beta survey instrument background (5 min), efficiency factor (dpm/cpm), and area factor.
13. Beta gas-proportional detector background (5 min) and efficiency factor (dpm/cpm).
14. Gamma survey instrument background (5 min), and efficiency factor (uR/h/cpm).

Output:

1. Alpha total activity averaged over 1-m^2 with standard deviation (dpm/100cm^2).
2. Alpha maximum activity and standard deviation (dpm/100cm^2)
3. Alpha removable activity and standard deviation (dpm/100cm^2)
4. Beta total activity averaged over 1-m^2 with standard deviation (dpm/100cm^2).
5. Beta maximum activity and standard deviation (dpm/100cm^2)
6. Beta removable activity and standard deviation (dpm/100cm^2).
7. Ambient gamma exposure rate and standard deviation (uR/h).

4.3.1 Direct Alpha/Beta Contamination Measurement

The counts observed for the alpha and beta surface activity were converted to dpm/100cm^2 by:

$$SA = \frac{(C - B)}{5} E \frac{(100)}{A} \qquad \text{(Eq. 4.1)}$$

where: SA = surface activity (this is applied to either the average or maximum activity)

C = total counts in 5 min

5 = count time, min

B = background count in 5 min (generally 0-5 for alpha and about 440-460 for beta)

E = Efficiency factor, dpm/cpm (averages about 4.8 for alpha and about 3.7 for beta)

100 = 100 cm^2 standard area

A = probe sensitive area (71 cm^2 for Ludlum model 43-1 circular alpha scintillator; 20 cm^2 for Ludlum model 44-9 pancake G-M)

Note that the analysis is done using counts rather than count rates. The standard deviation of the measurement in dpm/100cm^2 is given by:

$$s = \frac{\sqrt{C + B}\,(100)(E)}{(5)(A)} \qquad \text{(Eq. 4.2)}$$

4.3.2 Removable Alpha/Beta Contamination Measurement

The results of smears counted by a gas-flow proportional counter for alpha and beta removable surface activity were converted to dpm/100 cm^2 by:

$$SA = \frac{(C - B)(E)}{5} \qquad \text{(Eq. 4.3)}$$

where the appropriate alpha and beta backgrounds and efficiency factors were used. Backgrounds are typically 0-2 counts for alpha and 40-50 counts for beta in a five minute time period. Efficiency factors are about 3.5 for alpha and 3.9 for beta.

The standard deviation of this measurement is:

$$s = \frac{\sqrt{C + B}\,(E)}{5} \qquad \text{(Eq. 4.4)}$$

4.3.3 Ambient Exposure Rate

Gamma scintillations recorded by a NaI instrument were converted from cpm to exposure rate (uR/h) by:

$$\dot{R} = \frac{(C - B)(E)}{5} \qquad \text{(Eq. 4.5)}$$

The efficiency factor is given in (uR/h/cpm) and is calculated quarterly by the instrument calibration laboratory using a Cs-137 source. The standard deviation of the measurement is given by equation 4.4, above.

4.3.4 Data Reduction Software Program

Software was developed to read output data from the VISICALC file into a graphics program which plots activity (dpm/100cm^2) against the Gaussian cumulative distribution function (cdf). For convenience, the distribution function, G(x) is plotted as the abscissa (probability grades), and x, the activity, is plotted as the ordinate (linear grades). The Gaussian plots take the shape of a straight line due to the orientation of the axes and the non-linear abcissa. Section 5.0 further discusses the statistical tests applied to survey data.

4.4 Data Analysis

An arithmetic mean of the contamination values is calculated for each data set. From the plot of activity vs. cumulative probability, the mean contamination value of the lot is the value on the ordinate axis where the distribution intersects the 50% cumulative probability. This type of analysis is described in section 5.0. The figures display the results on an expanded scale so that the variations in the data can be seen in detail. The distribution can be analyzed in terms of deviations from the mean, $\bar{x}$. The results of a sampling inspection ("inspection by variables") test follows the graphic results.

The test statistic, $\bar{x} + ks$, is compared to the test limit U, where:

$\bar{x}$ = average (arithmetic mean of measured values)
s = observed sample standard deviation
k = tolerance factor calculated from the number of samples to achieve desired sensitivity to the test
U = test limit.

Values of k for each sample size are calculated in accordance with the following equations:

$$k = \frac{K_2 + \sqrt{K_2^2 - ab}}{a}\ ;\ a = 1 - \frac{K_\beta}{2\ (n-1)}\ ;\ b = K_2^2 - \frac{K_\beta^2}{n} \qquad \text{(Eq. 4.6)}$$

where:

k = tolerance factor
K_2 = the normal deviate exceeded with probability of β, 0.10 (from tables, K_2 = 1.282)
K_β = The normal deviate exceeded with probability equal to the LTPD. 10.1 (from tables, K_β = 1.282)
n = number of samples

The State of California has stated that the consumer's risk of acceptance (β) at 10% defective (LTPD) must be 0.1. For these choices of β and LTPD, $K_\beta = K_2 = 1.282$.

The criteria for acceptance are presented as a plan of action. The plan of action is:

1) Acceptance: If the test statistic ($\bar{x}$+ks) is less than or equal to the limit (U), accept the region as clean. (Any single value, x, less than 50% of the limit is considered the <u>Characterization Level</u>, which requires no further action. If

any single measured value, x, exceeds 50% of the limit, reinspect that location and take a few additional samples in the immediate area for the analysis. This is the <u>Reinspection Level</u>. If any single measured value, x, exceeds 90% of the limit, investigate the source of occurrence. This is the <u>Investigation Level</u>.

2) Collect additional measurements: If the test statistic $(\bar{x}+ks)$ is greater than the limit (U), but $\bar{x}$ itself is less than U, independently resample and combine all measured values to determine if $\bar{x}+ks <= U$ for the combined set; if so, accept the region as clean. If not reject the region.

3) Rejection: If the test statistic $(\bar{x}+ks)$ is greater than the limit (U) and $\bar{x} >= U$, reject the region. Investigate the source of occurrence.

In addition to the computerized data reduction methods used for analysis of the survey grids, search and survey techniques were conducted throughout the building on special structural features and components where it is likely that contamination could have been deposited.

5.0 STATISTICS

5.1 Counting Statistics

The emission of atomic and nuclear radiation obeys the rules of quantum theory. As a result of this, only the probability that an emission will occur is determined. The absolute number of particles emitted by a radioactive source in a unit of time, is not constant in time; it has a statistical uncertainty because of the probabilistic nature of the phenomenon under study. The number of particles emitted per unit time is different for successive units of time. Therefore, only the average number of particles emitted per unit time and per unit area can be determined. Because of the probabilistic nature of particles emitted by radioactive elements, repeated measurements of the average number of emissions per unit time will show a distribution approximated by the Gaussian (or normal) probability density function (pdf); this is known as the central limit theorem. If measurements are made at many similar locations, these measurements will show a greater variability, but the distribution will remain adequately represented by a Gaussian function. Thus the number of occurrences of particular mean contamination values, g(x), shows a Gaussian pdf relative to the contamination value, and the data can be plotted accordingly. Subsequently, based on the results of the data analysis, a conclusion can be made regarding the level of residual contamination in the building.

The Gaussian distribution, g(x), is given by:

$$g(x)dx = \frac{1}{(\sqrt{2\pi})\sigma} \exp\left\{\frac{-(x-m)^2}{2\sigma^2}\right\} dx \qquad \text{(Eq. 5-1)}$$

where

g(x)dx = probability that the value of x, lies between x and x+dx

m = average, or mean of the distribution

σ = standard deviation of the distribution.

A graph of x vs. g(x) gives the following bell-shaped curve:

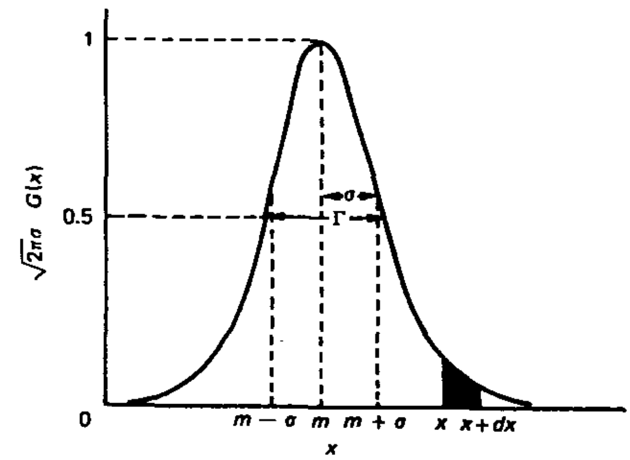


Figure 5.1 The Gaussian Probability Density Function

The cumulative distribution function (cdf), G(x), is equal to the integral of the pdf, for a continuous random variable, hence:

$$G(x) = \int_{-\infty}^{x} g(x)dx \qquad \text{(Eq. 5-2)}$$

$$= P(X <= x)$$

This function is commonly referred to as the error function, (erf). The graph of the Gaussian cdf is:

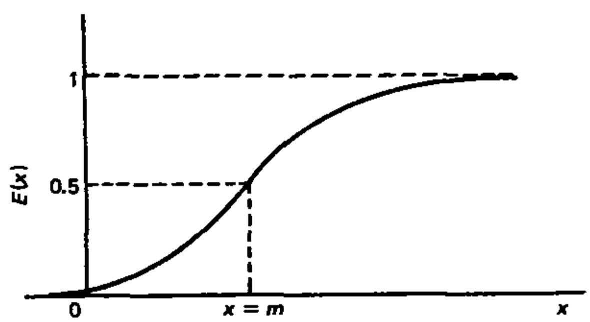


Figure 5.2 The Gaussian Cumulative Distribution Function

If x is the survey measurement (the number of counts) the standard deviation of the measurement is the square root of x. Background radiation must also be considered to calculate the net number of counts. Thus, the error, or standard deviation associated with the measurement becomes:

$$s = \sqrt{\frac{C + B}{T}} \qquad \text{(Eq. 5-3)}$$

where C = the number of counts recorded in time, T, of the sample
B = the number of counts recorded in time, T, of the background radiation environment

Equal values of the time, T, must be used for sample and background counts. Finally, corrections must be made for instrumentation parameters including detector geometry and efficiency.

This statistical summary presents the principles used to reduce and analyze data from contamination measurements made at T005.

5.2 Sampling Inspection by Variables

Acceptance inspection by variables is a method of judging whether a lot of items is of acceptable quality by examining a sample from the lot, or population. In the case of determining the extent of contamination at T005, it would be unacceptably time consuming and not cost effective to sample 100% of the building. However, by applying sampling inspection by variables methods, the accuracy of the conclusion made about the level of contamination is not sacrificed because of a decrease in number of sampling locations. We estimate the level of contamination in the building by sampling 11% of it.

By contrast, in acceptance inspection by attributes, the radiation measurement in a given area is recorded numerically and classified as either being defective or nondefective, according to the acceptance criteria. A defect means an instance of a failure to meet a requirement imposed on a unit with respect to a single quality characteristic. Second, a decision is made from the number of defective areas in the sample whether the percentage of defective areas in the lot is small enough for the lot to be considered acceptable.

On the other hand, in acceptance inspection by variables, the result is recorded numerically and is not treated simply as a boolean statistic, so fewer areas need to be inspected for a given degree of accuracy in judging a lot's acceptability.)

5.2.1 Calculated Statistics of the Gaussian Distribution

The test statistic for each sample area, $\bar{x} + ks$, is compared to the acceptance limit U, where:

$\bar{x}$ = average (arithmetic mean of measured values)
s = observed sample standard deviation (Eq. 4.4)
k = tolerance factor calculated from the number of samples to achieve the desired sensitivity for the test (Eq. 4.6)
U = acceptance limit.

For the sake of the T005 analysis, the building was divided into eight sample areas as described in section 4.1

The sample mean, standard deviation, and acceptance limit are easily calculable quantities; the value of k, the tolerance factor, bears further discussion. Of the various criteria for selecting plans for acceptance sampling by variables, the most appropriate is the method of Lot Tolerance Percent Defective (LTPD), also referred to as the Rejectable Quality Level (RQL). The LTPD is defined as the poorest quality in an individual lot that should be accepted. Associated with the LTPD is a parameter referred to as consumer's risk (β), the risk of accepting a lot of quality equal to the LTPD. It has been standard practice to assign a value of 0.10 for consumer's risk. Conventionally, the value assigned to the LTPD has been 10%. These a priori determinations are consistent with the literature and regulatory position, and are the same values used by the state of California. [2] Thus, based on sampling inspection, we are willing to accept the hypothesis that the probability of accepting a lot as not being contaminated which is in fact 10 percent defective (i.e. above the test limit, uc) is 0.10. The value of k, which is a function of the a priori determinations made for β and LTPD is given by equation 4.6 in the previous section.

Figure 5.3 demonstrates this principle. The operating characteristics curve of a Gaussian sample distribution shows the principles of consumer's and producer's risk, in addition to LTPD, acceptable quality level, (AQL), and rejectable quality level (RQL). The criteria for acceptance of a lot are presented in section 4.3.

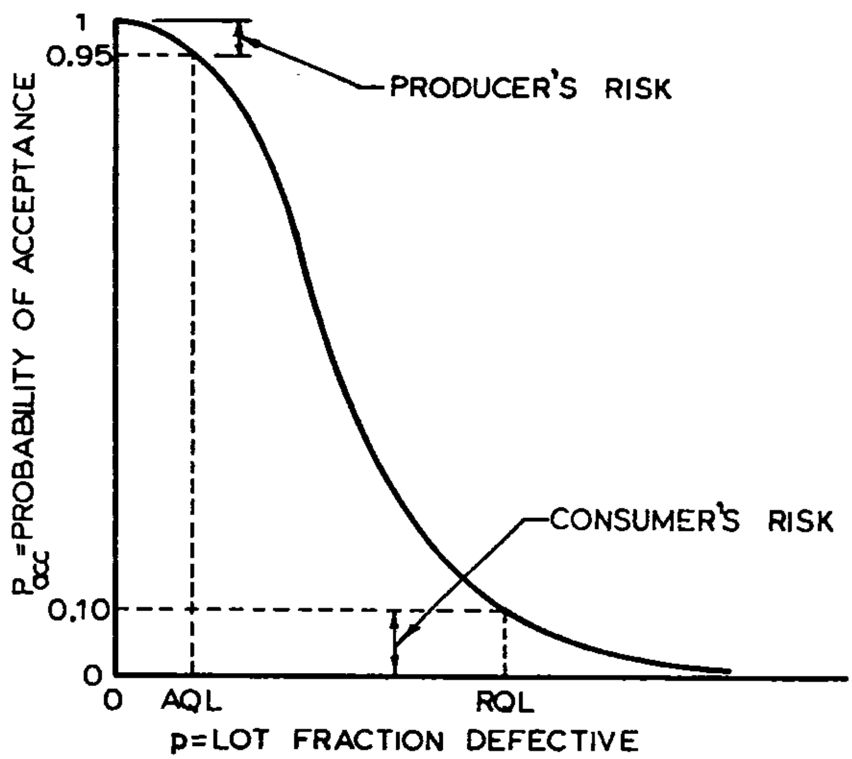


Figure 5.3 Operating Characteristics Curve

Simply by coincidence, the coefficients K_β and K_2 are equal because of the choice for the values of β and LTPD as 0.10. Refer to statistics handbooks listed in the reference section for additional understanding of this sampling principle. The a priori values chosen for the sampling coefficients are consistent with industrial sampling practice and regulatory guidance.

5.2.2 Graphical Display of Gaussian Distribution

When the cdf G(x), the integral of the Gaussian pdf, (Eq. 5.2), is plotted against x, the measurement value, a graph of the error function is generated (Fig. 5.2) on a linear-grade scale. For convenience of this survey and for readability, G(x) is plotted as the abscissa (x-axis) and the measurement value, x, is plotted as the ordinate (y-axis) on a probability-grade scale for the abscissa. G(x) values arranged in order of magnitude from left to right form a nearly straight line on probability-grade paper, when the sample lot contamination is normally distributed. Figure 5.4 shows this output.

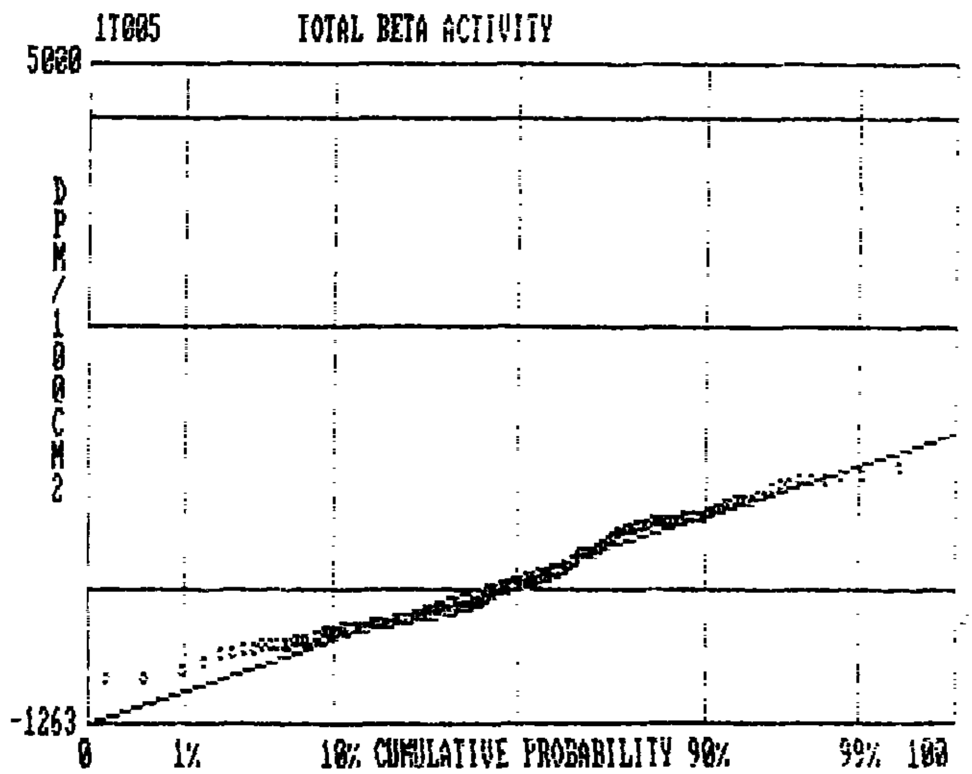


Figure 5.4 Gaussian cdf Plotted on Probability-Grade Paper

The power of this graphical display is that it permits identification of values with significantly greater contamination than expected for that lot. The calculated statistics numerically indicate the average and dispersion of the distribution, but are not effective for identifying trends or anomalies; for instance, identification of an isolated area in a sample lot which is contaminated at levels significantly greater than the fitted Gaussian line, and so are easily observable. Upon further inspection and analysis, these graphical displays are used to show contamination level differences between areas or structures in a sample lot. For example, variation in contamination levels between a floor and ceiling, or between a wall and light fixtures. The power of the fitted Gaussian graphical display is important in assessing significant variations in the contamination levels within sample lots.

6.0 PROCEDURES

The following radiological procedures were used in performing this survey.

6.1 Sample Selection Gridding

Superimpose 3-meter square grids on each surface to be radiologically characterized. If a surface is less than $9m^2$ in area, then grid the area by square meters as appropriate.

6.1.1 Floor

Select $1m^2$ out of each $9m^2$ on which to perform the survey. If a surface is less than $9m^2$ in area, then survey $1m^2$ as a minimum. Objects lying on the floor which are easily moveable should be moved to allow a complete floor survey. Survey around fixed objects.

6.1.2 Walls

Select $1m^2$ out of $9m^2$ on which to perform the survey for that part of the wall below 10 feet in height and which is readily accessible. Survey around cabinets, shelves, and equipment which cannot be easily relocated.

6.1.3 Ceiling

Select $1m^2$ out of $18m^2$ on which to perform the survey. Only survey those ceilings which are less than 10 feet above the floor.

6.1.4 Special Structural Features

Gridding is not necessary. Survey randomly for detectable alpha/-beta contamination. Smear the area for analysis of removable contamination.

6.2 Calibration and Instrument Checks

Instruments used for the final survey were calibrated and checked every morning, noon, and evening for the duration of the project as follows.

Portable Ludlum 2220 Survey Instruments:

1) Turn the instrument 'ON' and allow to warm up for 5 min.

2) Check high voltage (600-750V alpha, 800-950V beta).

3) Check threshold (140-190 alpha, 250-350 beta)

4) Window in/out switch is set to out.

5) Check battery (greater than 500).

6) Set range selector to 1, response to fast, and count time to 5 min.

7) Take and record a 5 min background count in an uncontaminated area which typifies the area to be surveyed.

8) Take and record a 5 min count of known alpha and beta standards; an electroplated Th-230 and electroplated Tc-99 source, respectively. The efficiency factor (dpm/cpm) is calculated as the ratio of 2 times the 2pi emission rate of the source (dpm) to the net count rate of the instrument. The radioactivity of the calibration sources is traceable to NBS. Use

a Ra-226 check source located 1 ft from the NaI detector to check the operability of the gamma instrument. The count rate should not vary by more than ± 20% from the initially established standard. The gamma calibration efficiency factor is calculated quarterly by the instrument shop.

9) Calculate the area of the alpha and beta end windows and record value. This is performed only once.

Gas-flow Proportional:

1) Equipment is to be left in the 'ON' position at all times.

2) Using uncontaminated planchets, take four 5 min background counts.

3) Take and record 5 min counts of known alpha and beta standards; 1 in. Th-230 and Tc-99 sources, respectively. Calculate efficiency factors accordingly.

Average the daily results:

Calculate the average background and efficiency factor of each instrument for morning and afternoon. The morning value should be the average of the 7:00 am and 11:30 am measurements; the afternoon value should be the average of the 11:30 am and 4:00 pm measurements.

6.3 Contamination Measurements

6.3.1 Total-Average Alpha/Beta Contamination Measurements

1) Identify 1-m^2 area to be measured; 1-m^2 per 9-m^2 surface

should be surveyed to be consistent with a minimum 11% sampling plan.

2) With portable scaler instrumentation (Ludlum 2220) set for 5-min count time, use an alpha probe (Ludlum Model 43-1) on one instrument and a beta probe (Ludlum 44-9) on another, uniformly scan the area. The probe transit velocity should be slow; less than one-third the numerical detector window diameter (in cm). This corresponds to a transit velocity not exceeding 2 cm/sec. The 5 min count time per square meter was adopted based on this transit velocity limit for alpha contamination. (Watch and listen for "hot spots" where radioactivity may exceed the average limit. These are to be resurveyed later).

3) Record the location, total count, background, efficiency factor, area factor, and date/time.

4) Enter the data into VISICALC spreadsheet.

6.3.2 Maximum Alpha/Beta Contamination Measurements

1) Return to any area identified as having a spot which measures considerably greater than the average contamination value for that area.

2) Repeat the scan of only the hot spot area, covering approximately 100 cm^2 with the probe.

3) Record the location, total count, background, efficiency factor, area factor, and date/time, as a maximum contamination value.

4) Enter the data into VISICALC spreadsheet.

6.3.3 Removable Alpha/Beta Contamination Measurements

1) Using an NPO 2" diameter - cloth swipe, wipe an "S" pattern, with legs approximately 6 in long, so as to sample removable contamination from an area of approximately 100 cm^2 within the 1-m^2 grids identified and sampled with the survey meters.

2) Place smear in envelope kit and record the location of the sample grid on the envelope. Save until ready for counting.

3) Count radioactivity using gas-flow proportional counter (Canberra Model 2201) for 5 min. (see Appendix A).

4) Record the location, total alpha and beta counts, background and efficiency factors for each.

5) Enter the data into VISICALC spreadsheet.

6.3.4 Ambient Gamma Exposure Rate Measurements

1) Mount the detector on a tripod which centers the detector 1 meter from the floor surface.

2) Set the count time to 5 min and take a measurement at each applicable location for that length of time.

3) Record the location, total counts, background, and efficiency factor (uR/h/cpm).

4) Enter the data into VISICALC spreadsheet.

6.3.5 Isotopic Qualification

1) As necessary, collect various samples of debris, dirt, and other which indicate detectable activity with the survey meter. Because U-238 and U-235 were the only radioactive materials handled at the facility, it is desireable to qualify the measurement by gamma spectrometry.

2) Place the sample in the calibrated high purity germanium (HPGE) detector chamber and use the multichannel analyzer to qualify the radioactive material.

6.3.6 Surveys of Special Structural Features and Components

1) Using a Ludlum Model 12 count rate meter in connection with a Ludlum Model 43-5 rectangular alpha scintillation probe, survey various building features and components which are suspect of containing residual alpha contamination.

2) Perform an instrument calibration check three times daily using the Th-230 source mentioned above.

3) Ensure that the transit velocity (in cm/s) does not exceed one-third the numerical value of the detector length or width (cm), in the direction of the scan. In this case, with an alpha window length of 18 cm, the transit velocity must not exceed 6 cm/s when the probe is moved lenthwise. If moved widthwise, the transit velocity must not exceed 1.3 cm/s.

4) Do the same for beta contamination using a Ludlum model 43-1 beta probe.

5) Record the gross count rate in a generalized manner as NDA (No Detectable Activity) or less than 20cpm, 30cpm, 100cpm, etc., as applicable.

6) Smear the special structural features and analyze for removable radioactivity. Follow the procedure in section 6.2.3.

7.0 SURVEY RESULTS

7.1 Statistical Results Format

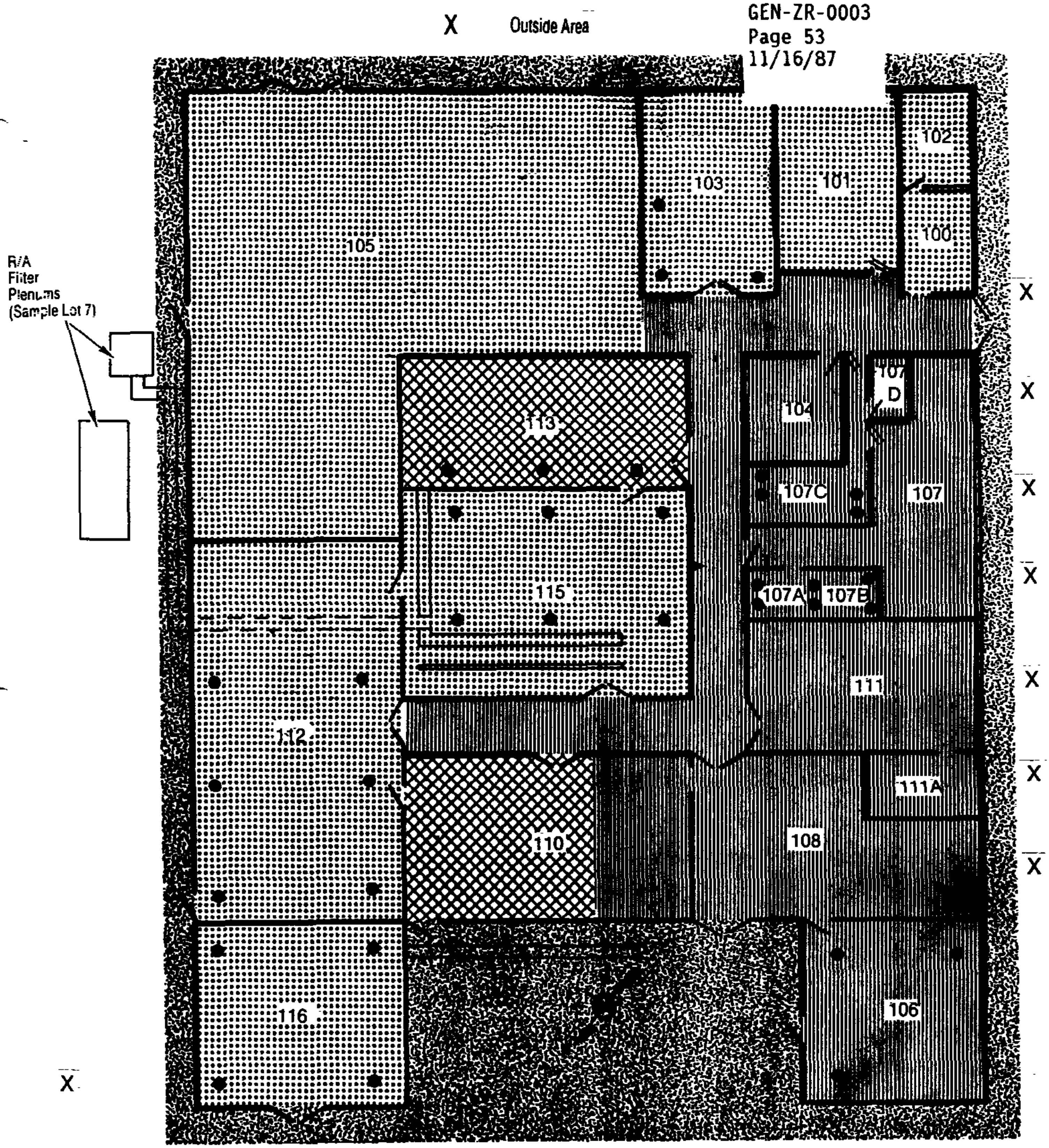
The T005 radiological survey was performed using the survey plan previously described. Eight sampling lots were established for statistical analysis, with a minimum of 30 sample locations or an 11% survey specified for each lot, whichever was greater. The lots are generally described in section 4.1. Figure 7.1 graphically shows each sampling inspection lot.

Because of the conservative action level maintained (50% of the acceptance level) during the survey, additional samples were taken in a few areas. Areas of increased inspection are discussed in each section, as appropriate. Various components and special features of the building were qualitatively surveyed to determine the presence of contamination in areas beyond the $1m^2$ grids surveyed for statistical analysis. These surveys were performed strictly for determining whether or not contamination was present. The conventions established for this survey were NDA, for no detectable activity (where the count rate meter did not increase noticeably above background); and < x cpm (less than x counts per minute) where the count rate noticeably increased. Special structural feature results are presented in section 7.6.

The format established for the results section follows.

7.1.1 Sample Lot General Description

A general description of the inspection sampling lot is presented first. Characteristics of the area beyond general building materials which may impact the conditions or results of the survey are described. Special structural features which are applicable to the survey are also mentioned. Discussion and results of radioactive material drain lines and exhaust ducts running through each sample area are presented in separate sections.



● Special Structural Features, (Sample Lot 5)

——— Exhaust Ducts, (Sample Lot 6)

X Soil and Sediment (Sample Lot 7)

Sample Lot 1
Sample Lot 2
Sample Lot 3
Sample Lot 4

Figure 7.1 Sampling Inspection Lots

7.1.2 Summary Tables

Statistical result summary Tables 7.1 through 7.10 presented for each sampling lot show the number of sampling locations, average and maximum contamination values of the sample lot, and the inspection test statistic ($\bar{x}$ + ks) corresponding to β = 10% and LTPD = 0.10. All values are reported as measurements above background radiation. Where a statistical analysis was inappropriate for the area or objects surveyed, a summary table shows the average contamination value as measured by a count rate meter, as in Tables 7.7 and 7.8.

7.1.3 Cumulative Probability Distributions

The survey data for each sampling lot are displayed as Gaussian cumulative distribution functions in Figures 7.2 through 7.42. Background radiation is subtracted from each distribution. These figures show each measurement value, arranged in order of magnitude from left to right, and a straight line representing the derived fitted-Gaussian distribution. In most cases the acceptance limit for unrestricted use is substantially greater than the mean and inspection test statistic of the distribution. In other cases, the test statistic exceeds the acceptance limit for unrestricted use, and the area must consequently be decontaminated. The graph is bounded in the positive y direction by either the acceptance limit, or the greatest measurement taken for that lot, whichever is greater.

The mean of each distribution is approximately that value on the ordinate which corresponds to a 50% cumulative probability on the abscissa. One, two, and three standard deviations above the mean corresponds to 84%, 97.7%, and 99.8% cumulative probability, respectively. The value of k used in the inspection test is very nearly 1.5 for each case; thus, the "k" line will run perpendicular to the abscissa corresponding to about a 93.3% cumulative probability. The Gaussian distribution line must pass below the intersection of the "k" line (about 93%) and the horizontal line showing the acceptance limit at that point in order to accept the lot as being non-contaminated. "k" increases as the number of samples in a lot decrease.

On those plots where the test limit is far greater than the cumulative probability distribution, e.g. alpha activity plots, the Gaussian fit "straight-line" is nearly buried in the abscissa. On other plots, e.g. beta and gamma, the number of counts collected by the instruments were sufficient to show a linear distribution, with slope greater than zero.

At the top left hand corner of the output is the file name of the data file for that sample lot. The maximum ordinate value in most cases is the test limit; otherwise, the greatest measurement value bounds the ordinate. The lower bound of the ordinate is either the smallest measured value (minus background) or the smallest value calculated for the Gaussian fit. This explains the negative numbers. Cumulative probability (abscissa) is plotted in probability grades, i.e. the distance between any two successive points increases as the distance from the 50% cumulative probability line increases. Four horizontal lines extending across each plot show from top to bottom, 100% of the test limit, 90% of the test limit (Investigation), 50% of the test limit (Reinspection), and zero.

7.2 Interior Area 1 (Non-Suspect)

7.2.1 General Description

Area 1 consists of rooms 100, 101, 102, 103, 105, 112, 115, and 116. Figure 7.1 graphically represents this sampling lot. The only areas where unencapsulated radioactive materials were handled were rooms 105, 112, and 115. However, those areas have since been extensively decontaminated and semi-renovated. Thus there was no evidence to believe that any part of the sampling lot was contaminated. All areas are of similar construction. Rooms 100, 101, and 102 are carpeted; the others are cement slab. Rooms 105, 112, and 116 have high ceilings, thus only the first 10 feet of wall was sampled by an 11% survey. The ceilings in these rooms (105, 112, 116) were not sampled.

Rooms 105 and 112 (the high-bay) are currently used as a temporary storage area. Most equipment was palletized so that during the survey pallets could be moved from one location to another to allow a 11% randomized floor survey.

7.2.2 Summary Table

Table 7.1 shows the results of area 1. In all cases the inspection test statistic ($\overline{x}$ + ks) is far below the acceptance limit, demonstrating that the area is acceptably clean for unrestricted use. This analysis also includes removable contamination measurements collected in the light fixtures and ventilation outlets in rooms 115 and 103.

Table 7.1. Summary of Survey Results

(Non-suspect Sample Lot #1: Rooms 100, 101, 102, 103, 105, 112, 115, 116)

Measurement	Number of Locations	Average Value	Maximum Value	Inspection Test Statistic	Limit
Average alpha (dpm/100 cm^2)	215	1.7	25	9.1	5,000
Maximum alpha (dpm/100 cm2)	0	-	-	-	15,000
Removable alpha (dpm/100 cm^2)	215	0.6	107	11.0	1,000
Average beta (dpm/100 cm^2)	215	102.8	1176	742.5	5,000
Maximum beta (dpm/100 cm^2)	0	-	-	-	15,000
Removable beta (dpm/100 cm^2)	215	5.6	425	66.3	1,000
Ambient exposure rate (uR/h)	95	0.0053	2.6	2.14	5

7.2.3 Cumulative Probability Distributions

Figures 7.2 through 7.6 show the statistical distributions of total average alpha, removable alpha, total-average beta, removable beta, and gamma exposure rates for area 1, respectively. No maximum "hot spot" measurements were observed in any of the square meters; consequently, "maximum" distributions are not applicable.

Figures 7.2 and 7.4 show Gaussian cumulative distributions for total contamination in area 1. There are no indications that there might exist a location which is contaminated above the unrestricted-use acceptance limits in either of the three cases. Figures 7.3 and 7.5, removable alpha and beta, respectively, show a few outliers from the Gaussian cumulative distribution. All outliers are below the 50% characterization level. These points, however, were subsequently determined to represent a separate lot: removable contamination inside the light fixtures in room 115. Upon further inspection, the light fixtures were checked for greater levels. The greatest removable contamination level determined for 5 light fixtures (interior) was 107 dpm/100cm^2-alpha and 425 dpm/100cm^2-beta, clearly below the 50% characterization level. The fixtures were thoroughly wiped clean after the survey as an additional ALARA measure.

Figure 7.6 shows the ambient radiation measurements recorded for Area 1. A model Gaussian cdf is shown in the figure, with no outliers or areas of increased ambient gamma radiation.

There are not any outliers in the distribution for average-total beta, but there is an area where average total beta contamination approaches 1200 dpm/100cm^2. Although well below the 50% characterization level, additional samples were taken. It was determined that the background radiation in this room was higher than the location where the background measurement was actually taken. No corrections were made to the data because the results are sufficiently low that this is not necessary.

The sampling lot is sufficiently clean that it may be released for unrestricted use.

Figure 7.2. Total-Average Alpha Activity in Area 1

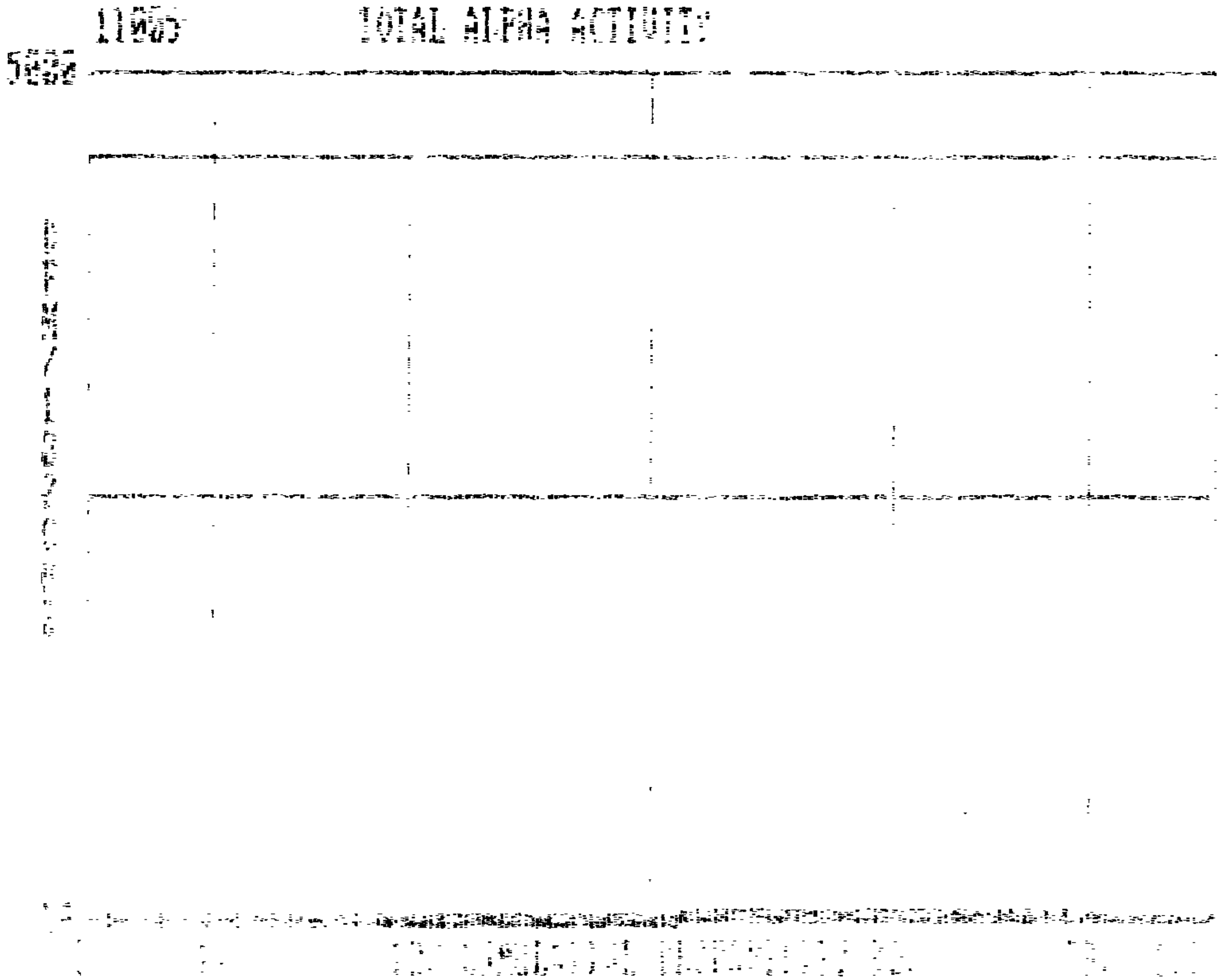


Figure 7.3. Removable Alpha Activity in Area 1

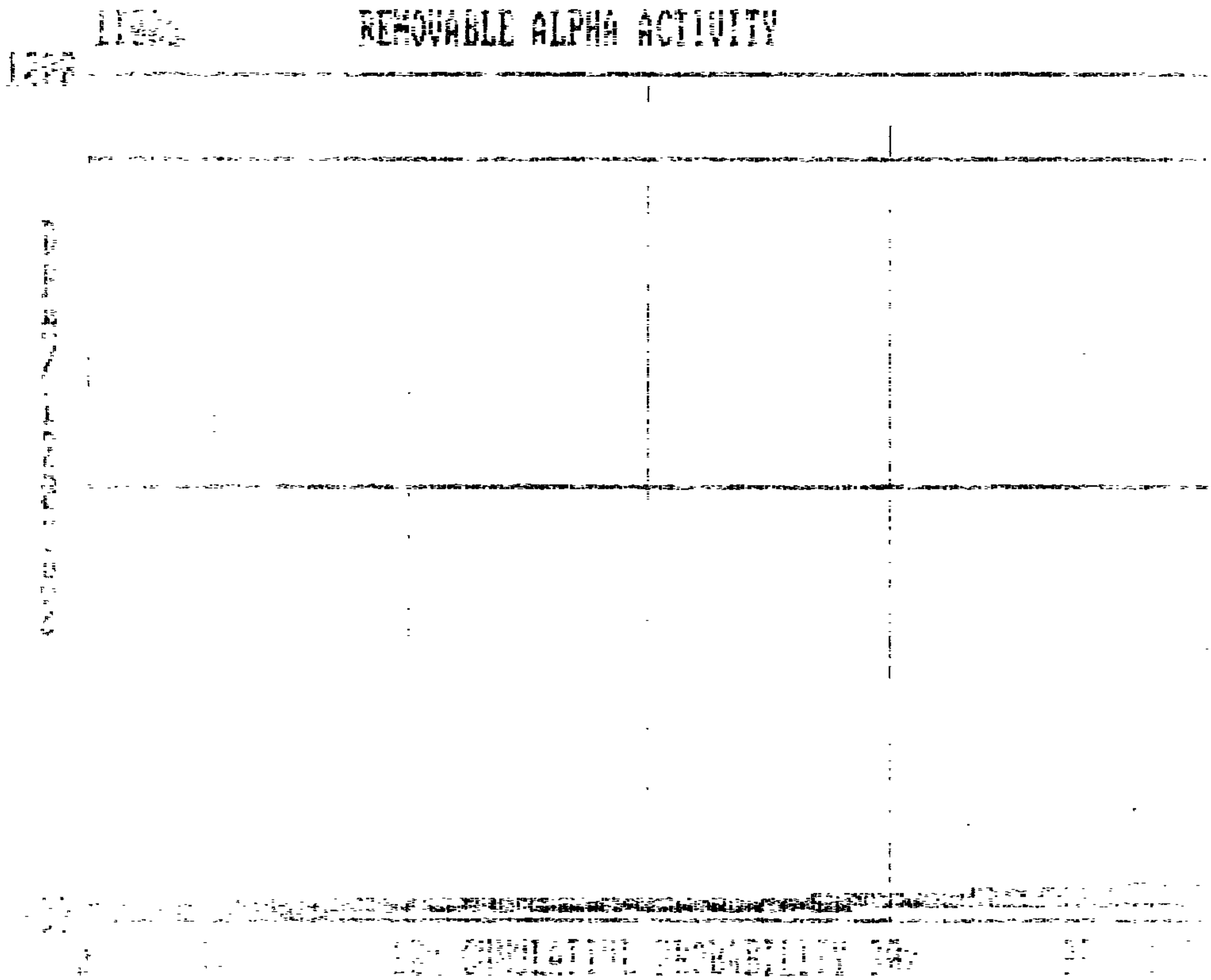


Figure 7.4. Total-Average Beta Activity in Area 1

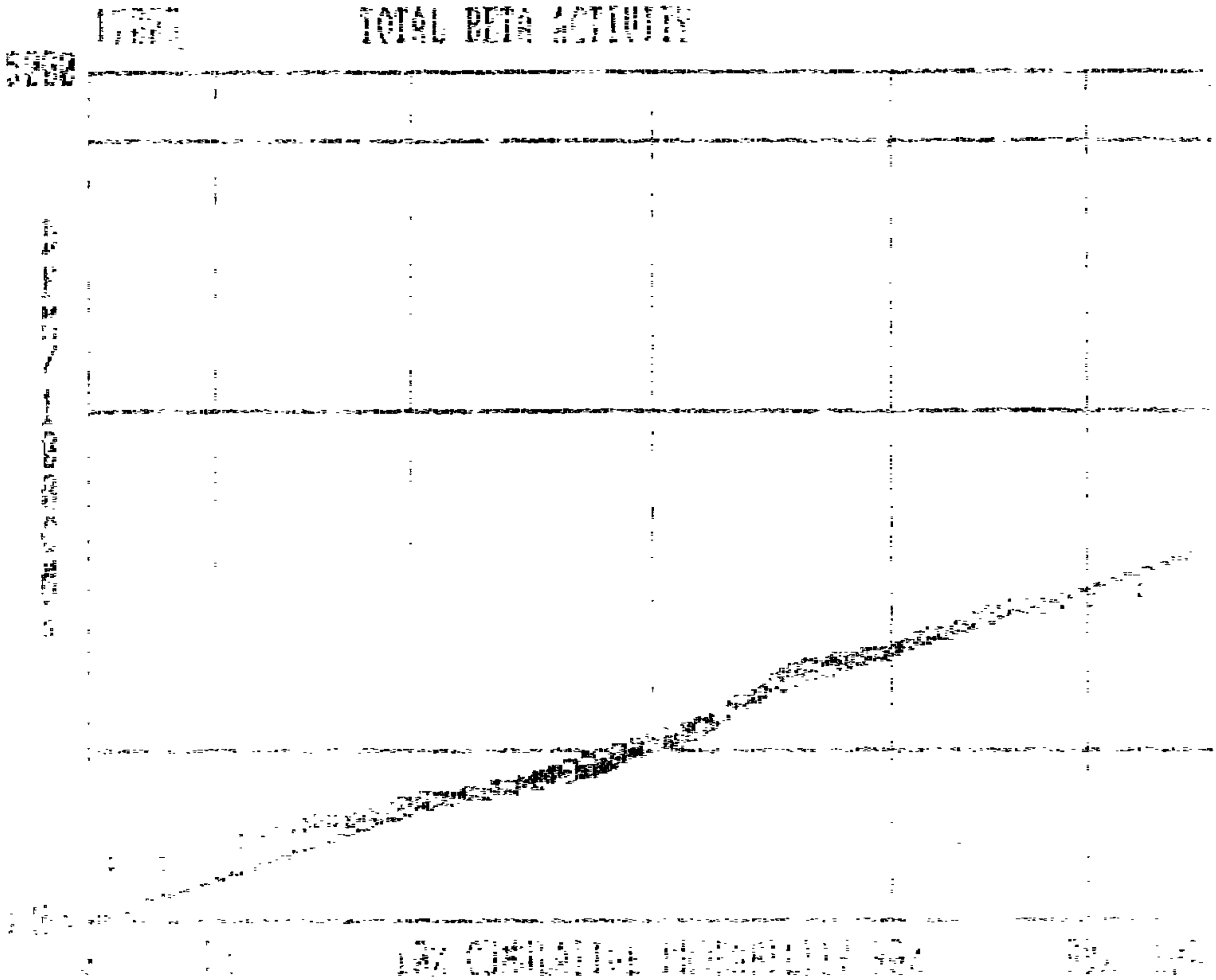


Figure 7.5. Removable Beta Activity in Area 1

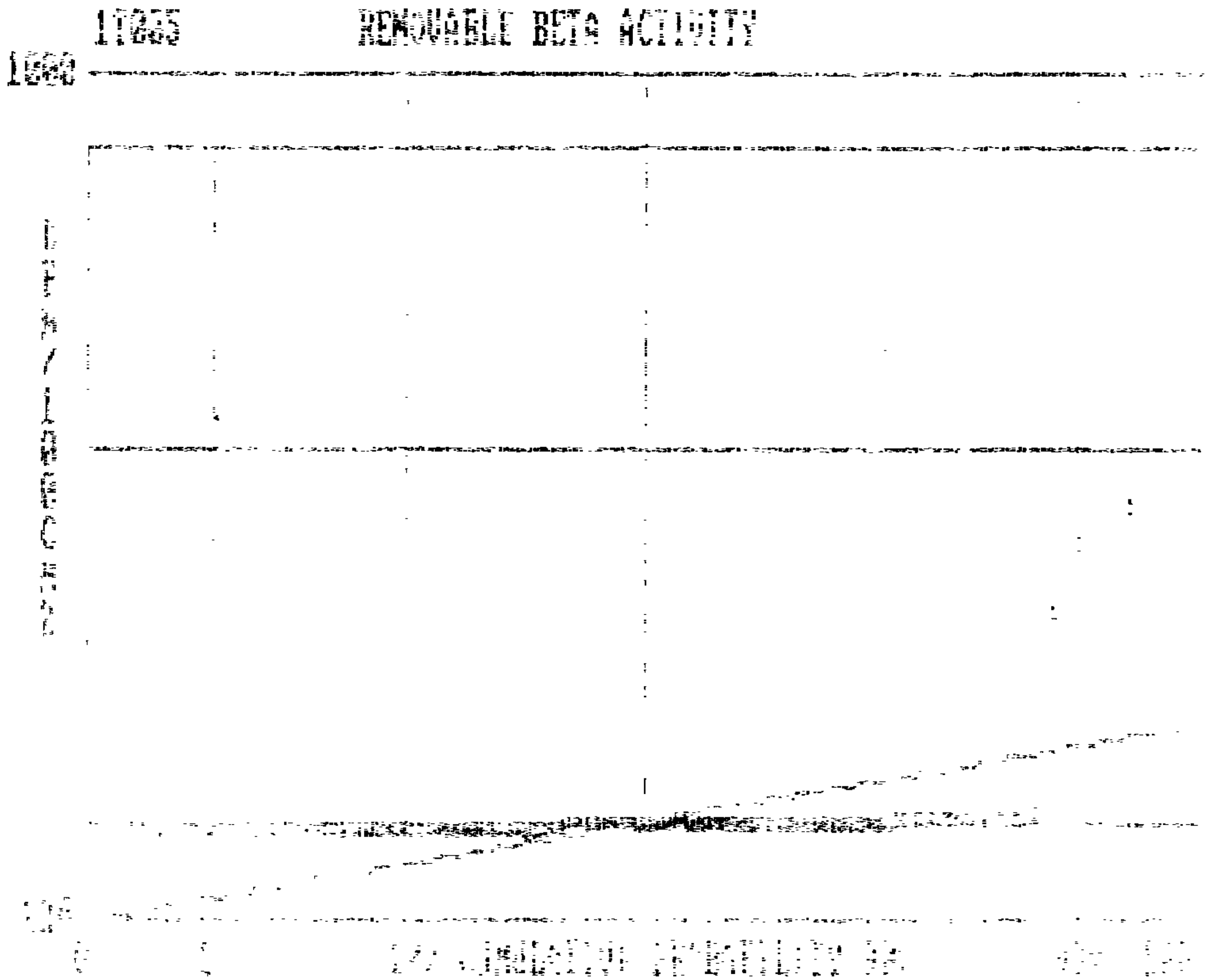
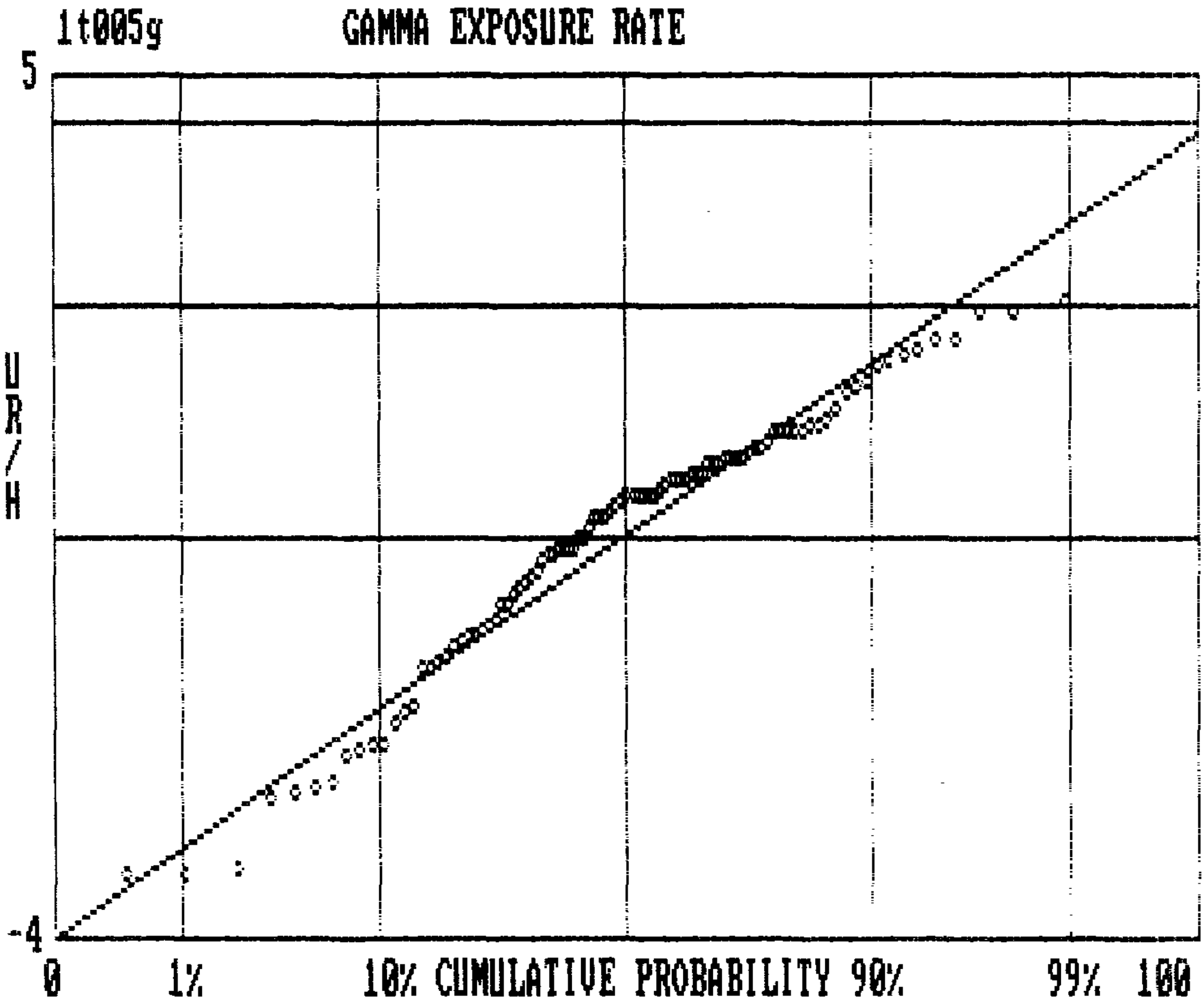


Figure 7.6. Ambient Gamma Exposure Rate in Area 1



7.3 Interior Area 2 (Non-Suspect)

7.3.1 General Description

Area 2 consists of rooms 104, 106, 107, 108, 110W, 111, 111A, and both hallways. Figure 7.1 graphically represents this sampling lot. Note that room 110 has been temporarily partitioned to house the computer center. In figure 7.1, 110E (east) is on the left and 110W (west) is on the right. 110E results are in section 7.4. The only areas where radioactive materials were handled were rooms 106, 108, 110W, 111, and 111A. However, those areas have since been extensively decontaminated and semi-renovated. Thus there was no reason to believe that any part of the sampling lot was contaminated. All areas are of similar construction. At the time of the survey, the floors of rooms 104, 106, and 107 were covered with 9in. vinyl tile. The other floor areas had been stripped of tile. All ceilings, except for 110W, are about 10 feet in height, allowing a thorough 11% survey on each wall and a 5.5% ceiling survey.

Except for room 106, which is used as a part storage area, and room 107, the men's rest room, all equipment had been removed, thereby allowing a complete unrestricted survey.

7.3.2 Summary Table

Table 7.2 shows the results of area 2. In all cases the inspection test statistic ($\overline{x}$ + ks) is far below the acceptance limit, demonstrating that the area is acceptably clean for unrestricted use. No maximum "hot spots" were detected.

Table 7.2. Summary of Survey Results
(Non-suspect Sample Lot #2: Rooms 104, 106, 107, 108, 110W, 111, 111A, hall1, hall2)

Measurement	Number of Locations	Average Value	Maximum Value	Inspection Test Statistic	Limit
Average alpha (dpm/100 cm^2)	199	1.7	32	7.9	5,000
Maximum alpha (dpm/100 cm^2)	0	-	-	-	15,000
Removable alpha (dpm/100 cm^2)	199	0.2	105	10.8	1,000
Average beta (dpm/100 cm^2)	199	21.9	1416	540.6	5,000
Maximum beta (dpm/100 cm^2)	0	-	-	-	15,000
Removable beta (dpm/100 cm2)	199	4.0	312	35.4	1,000
Ambient exposure rate (uR/h)	61	0.02	2.8	2.3	5

7.3.3 Cumulative Probability Distributions

Figures 7.7 through 7.11 show the statistical distributions of total-average alpha, removable alpha, total-average beta, removable beta, and gamma exposure rates for area 2, respectively. No maximum "hot spot" measurements were observed in any of the square meters. Figures 7.7, total-average alpha contamination, shows a model Gaussian cumulative distribution with no outliers. Figures 7.8 and 7.10 (removable alpha and beta) approximate a Gaussian; however, an outlier is present in each case. It corresponds to an isolated removable spot of contamination in the floor of room 108, location F5,8. Although both outliers are below the 50% characterization level, further inspection commenced. No isolated "hot spots" were found. Statistical evidence suggests that no additional locations would be contaminated above the test statistic 90% of the time.

The total-average beta distribution, Figure 7.9, suggests that two distributions are represented. Several floor measurements were in the 800 to 1000 dpm/100cm^2 range, with a couple in room 108 approaching 1200 dpm/100cm^2. This upper deviation from a Gaussian reflects what was observed to be differences in material background. Background measurements were taken in a room known to be noncontaminated (101), on a wooden table. The concrete slab poured in T005 had a greater beta background than the conference table on which the background measurements were made. No contamination measurement yielded a result greater than the 50% characterization level, so further investigation was unnecessary. Variations in background among different materials must always be investigated and accounted for.

The ambient gamma exposure rate distribution does not approximate a Gaussian very well. Quite a lot of variability is exhibited by the output. When the distribution was generated for the first time, the average value was less than zero. In order to keep the average exposure value positive, the background was uniformly decreased appropriately, a conservative modification. Two locations in room 111A slightly exceeded the 50% characterization level of 2.5 uR/h, but repeated measurements in that area showed exposure rates less than 2.5 uR/h above background.

This sampling lot is sufficiently clean that it may be released for unrestricted use.

Figure 7.7. Total-Average Alpha Activity in Area 2

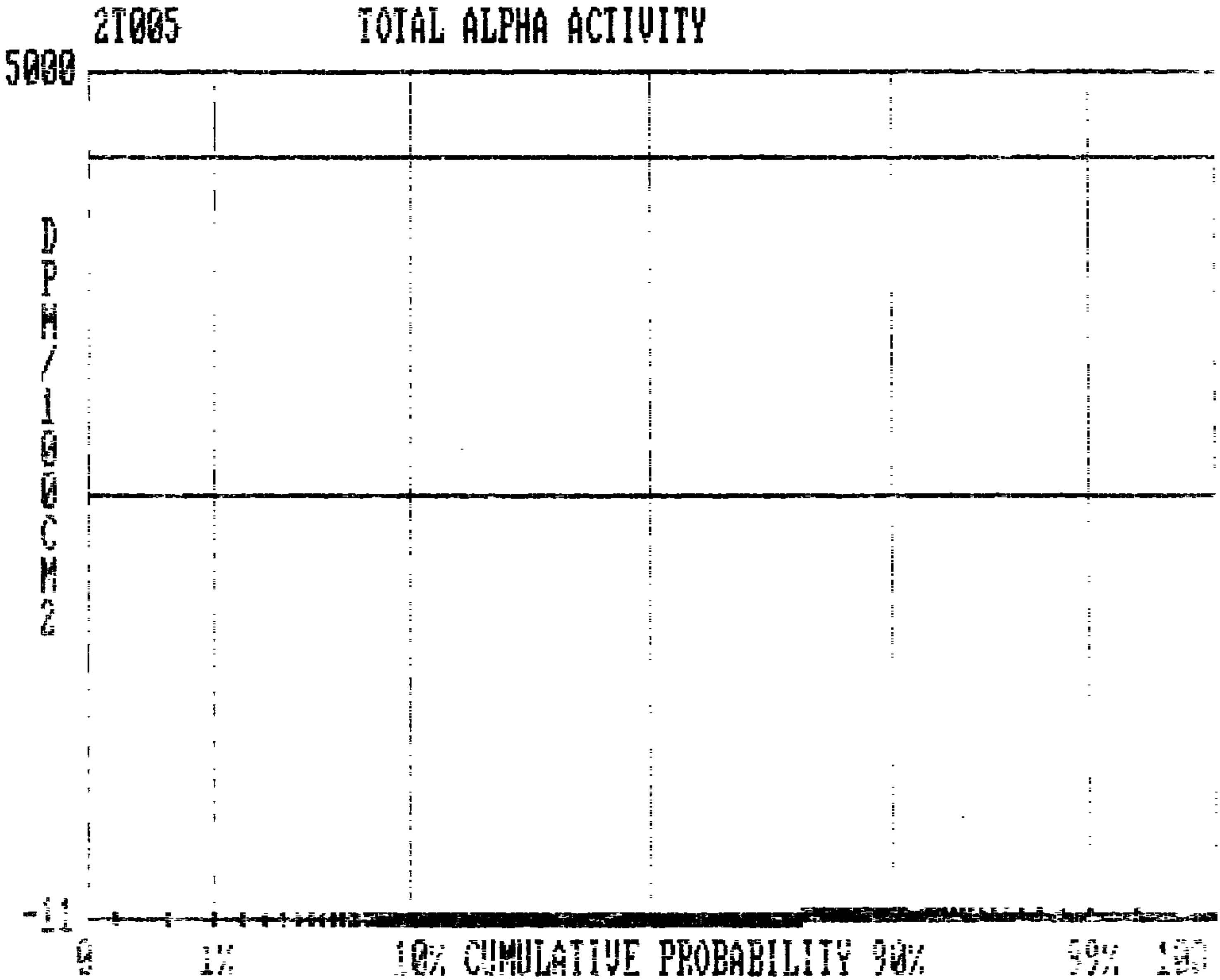


Figure 7.8. Removable Alpha Activity in Area 2

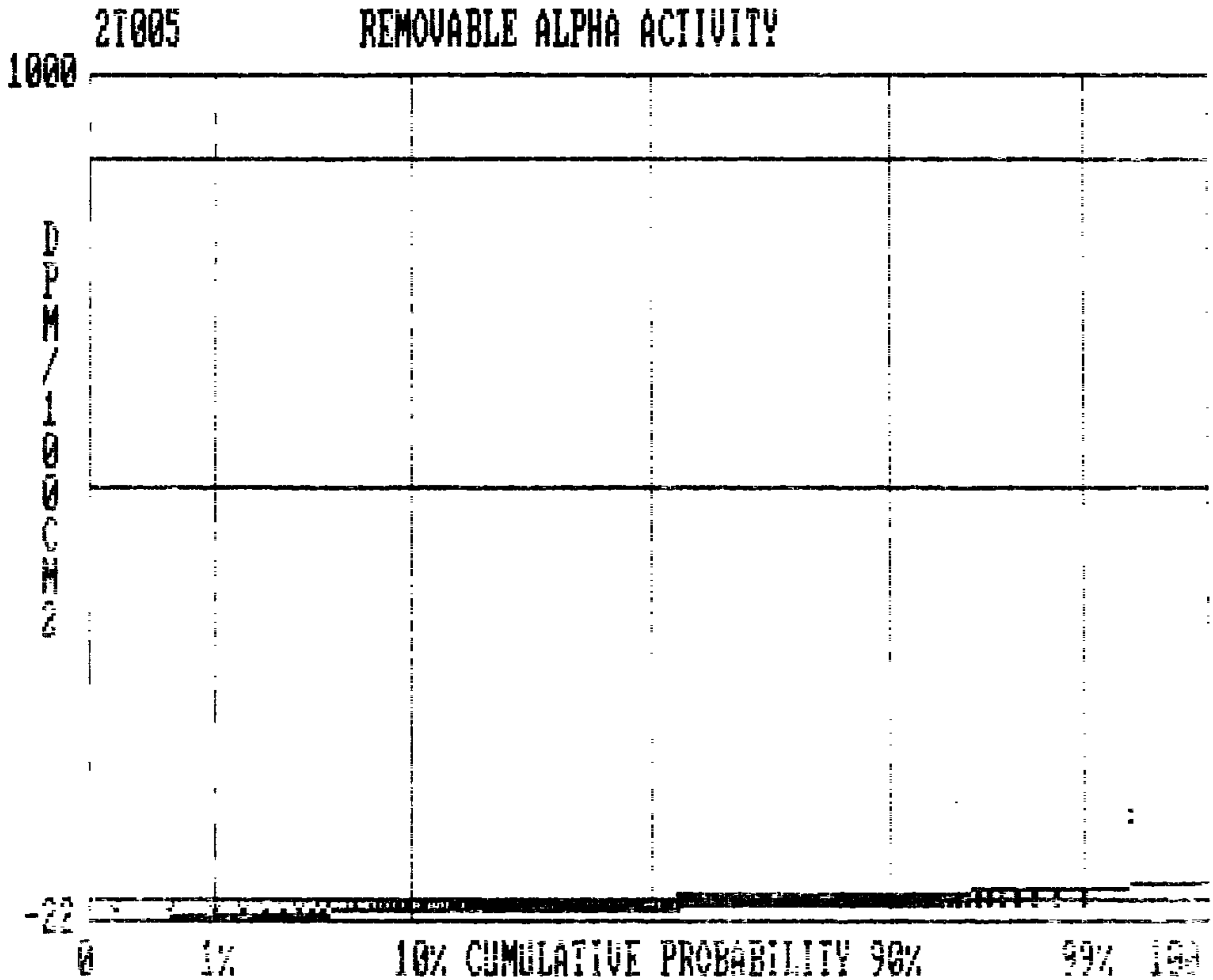


Figure 7.9. Total-Average Beta Activity in Area 2

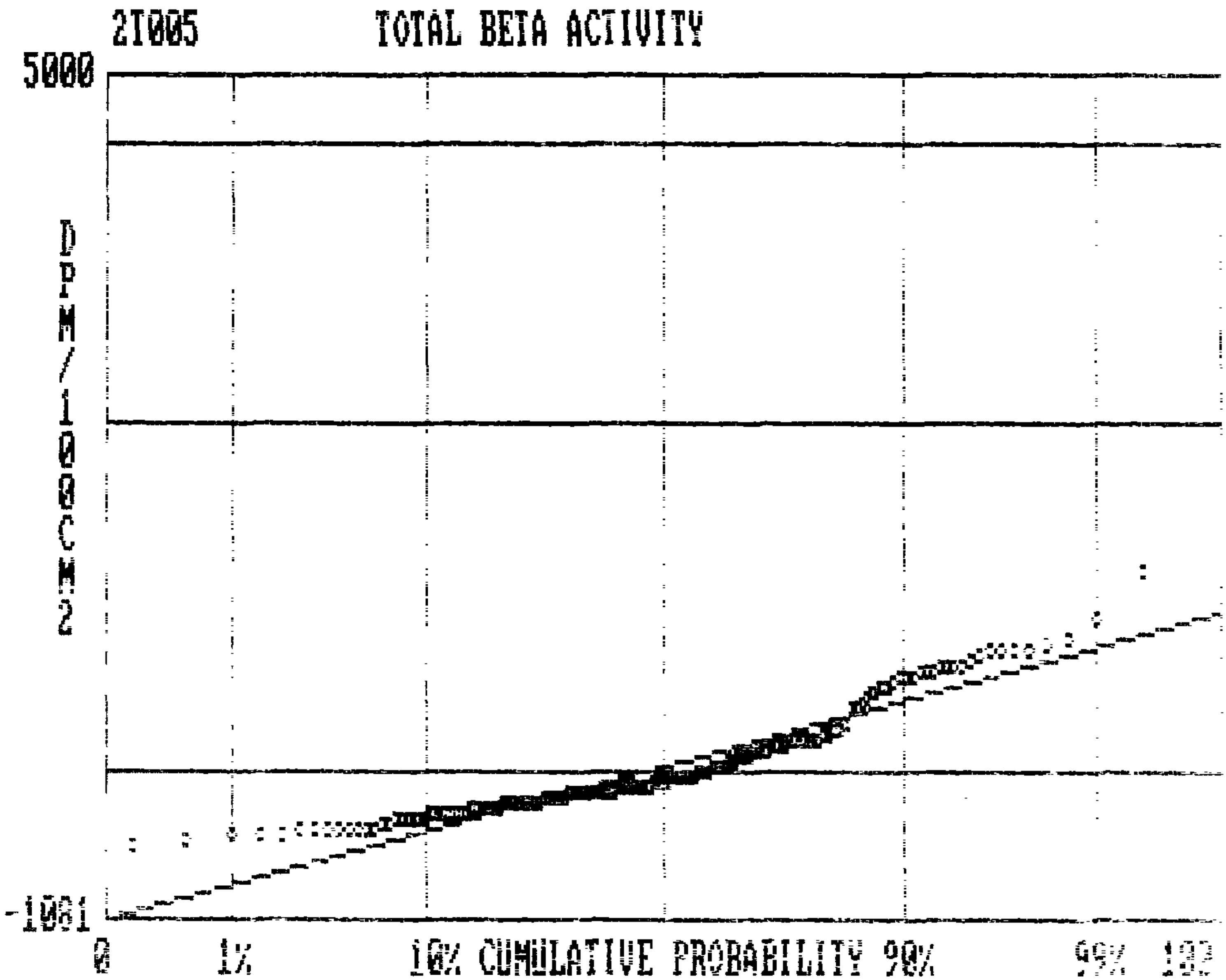


Figure 7.10. Removable Beta Activity in Area 2

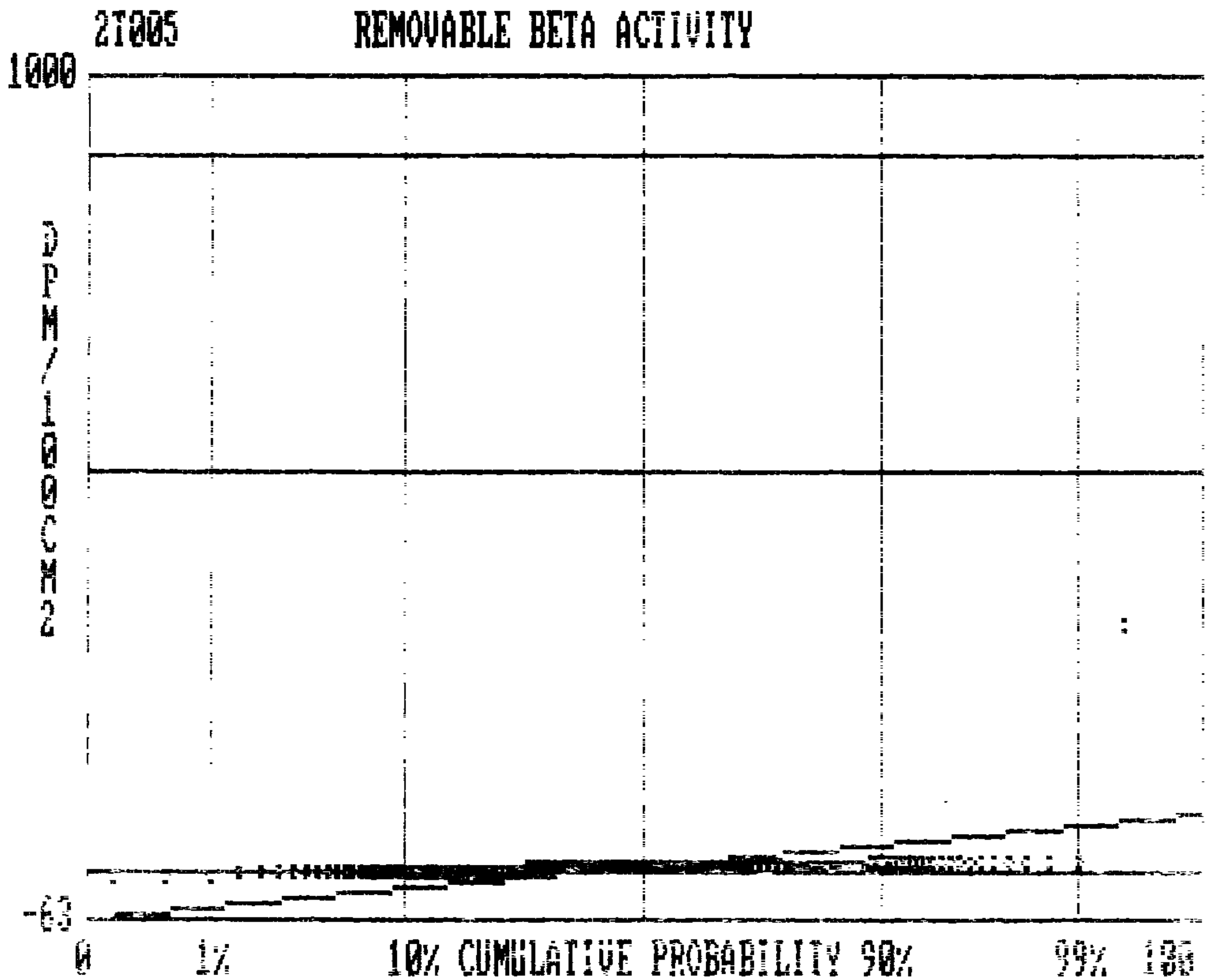
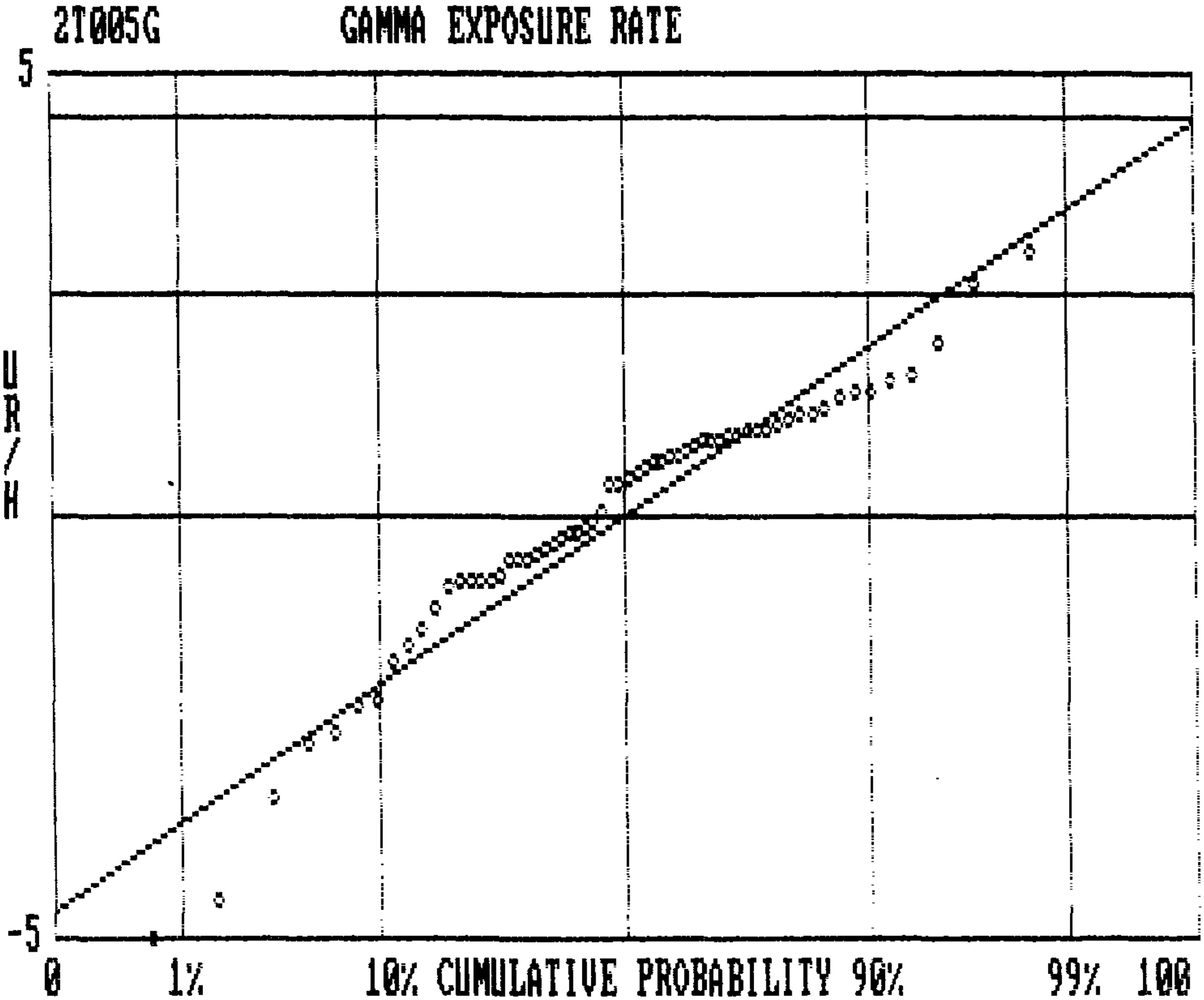


Figure 7.11. Ambient Gamma Exposure Rate in Area 2



7.4 Interior Area 3 (Suspect)

7.4.1 General Description

Area 3 consists of rooms 110E and 113. Figure 7.1 graphically shows this sampling lot. Radioactive materials were handled in both rooms. Neither room has since been decontaminated. Thus, both rooms are suspect of containing residual contamination. All areas are of similar construction. The floors are cement slab covered with 9 in. square vinyl floor tile. The ceiling in room 110E is much higher than 10 feet; therefore, the ceiling was not surveyed, and only the bottom portion of the wall was 11% surveyed. The ceiling in room 113 is about 10 feet high, so 100% of the walls qualified for the 11% survey and 5.5% of the ceiling surfaces were surveyed.

Room 110E currently houses the computer control center for the molten salt test facility. This heavy equipment was not relocated for the survey. Room 113 currently is used as a storage area for tools and parts. A major portion of the room is occupied by storage shelves. These shelves were not relocated during the survey, thus restricting the randomization of the floor and wall sampling selection. All accessible floor areas in both rooms were completely surveyed.

7.4.2 Summary Table

Table 7.3 shows the results of area 3. Maximum beta "hot spots" were detected between floor tiles in each room; therefore, maximum probability distributions were generated. Contamination above acceptance limits is present; however, it is not removable. The potential of spreading contamination from one room to another is minimal.

In all three alpha contamination measurements, the inspection test statistic ($\bar{x}$ + ks) is far below the acceptance limit. However, because beta particles were detected, the area is not necessarily clean of alpha contamination. Most likely, the radiological beta contamination detected between floor tile is from U-235 and U-238 which also emit alpha particles. The alpha particles do not penetrate through the tile to the detector, and consequently are not measurable under present survey conditions.

Beta contamination was detected between floor tiles. The average value for the maximum distribution is slightly greater than the acceptance limit; the inspection test statistic is far greater than the acceptance limit. Both rooms need to be decontaminated. This will involve tile removal and scrubbing of underlying cement.

Based on the output of the total-average beta distribution, it was evident that the floor distribution varied substantially from the wall distribution. For this reason, two additional rows of data are shown in Table 7.3, one for the wall distribution, the other for the floor. The walls are acceptably clean. The floor is contaminated slightly below the total-average beta acceptance limit for release.

Table 7.3. Summary of Survey Results
(Suspect Sample Lot #3: Rooms 110E, 113)

Measurement	Number of Locations	Average Value	Maximum Value	Inspection Test Statistic	Limit
Average alpha (dpm/100cm^2)	61	-2.0	16	5.2	5,000
Maximum alpha (dpm/100cm^2)	46	-1.9	15	4.8	15,000
Removable alpha (dpm/100cm^2)	107	-0.08	5	1.7	1,000
Average beta (dpm/100cm^2)	61	582	9118	2991	5,000
Average beta (dpm/100cm^2) (wall)	31	-233	386	158	5,000
Average beta (dpm/100cm^2) (floor)	30	1424	9118	4573	5,000
Maximum beta (dpm/100cm^2)	46	17688	107954	48637	15,000
Removable beta (dpm/100cm^2)	107	.8	13	8.2	1,000
Ambient Exposure Rate (uR/h)	30	.0007	1.5	2.5	5

7.4.3 Cumulative Probability Distributions

Figures 7.12 through 7.20 show the statistical distributions of total-average alpha, maximum alpha, removable alpha, total-average beta, maximum beta, removable beta, gamma exposure rates, total-average beta (walls), and total-average beta (floor) for area 3, respectively. Maximum beta "hot spot" measurements were observed between floor tiles. These distributions are included as Figures 7.13 and 7.16.

Figures 7.12 through 7.14 show model Gaussian cumulative distributions, with inspection test statistic values far less than the acceptance limits. Because beta contamination was detected between floor tiles, it is suspected that alpha contamination is present in proportional quantities. However, with current geometry and shielding, alpha contamination is not detectable.

The total-average beta distribution, Figure 7.15, suggests that because of the different slopes, two or more distributions are represented. For this reason, the analysis was divided into separate inspection lots; the walls and floor. As suspected, the floor distribution is quite different from the wall distribution; refer to Figures 7.19 and 7.20. Figure 7.19, total-average beta (wall), shows a model Gaussian distribution which indicates that the walls are acceptably clean. Figure 7.20, however, does not follow a Gaussian distribution, indicating that the floor is contaminated in varying amounts. The dispersion in the data is large.

The maximum beta distribution measurements were only taken on the floor. Figure 7.17 clearly shows that the area is contaminated above acceptable limits.

Figures 7.14 and 7.18 show that the contamination which is present is not removable.

The ambient gamma exposure rate distribution does not approximate a Gaussian very well. Quite a lot of variability is exhibited by the output. This variability is explained by what is thought to be variation observed in the ambient background. No corrections were made for this variation; background measurements made in an uncontaminated area were used for data reporting. When the distribution was generated for the first time, the average value was less than zero. In order to keep the average value positive, the background was uniformly decreased appropriately, a conservative modification. Gamma measurements made in room 110E approached, but did not exceed the 50% limit. No further inspection was initiated.

The area is contaminated above release limits.

Figure 7.12. Total-Average Alpha Activity in Area 3

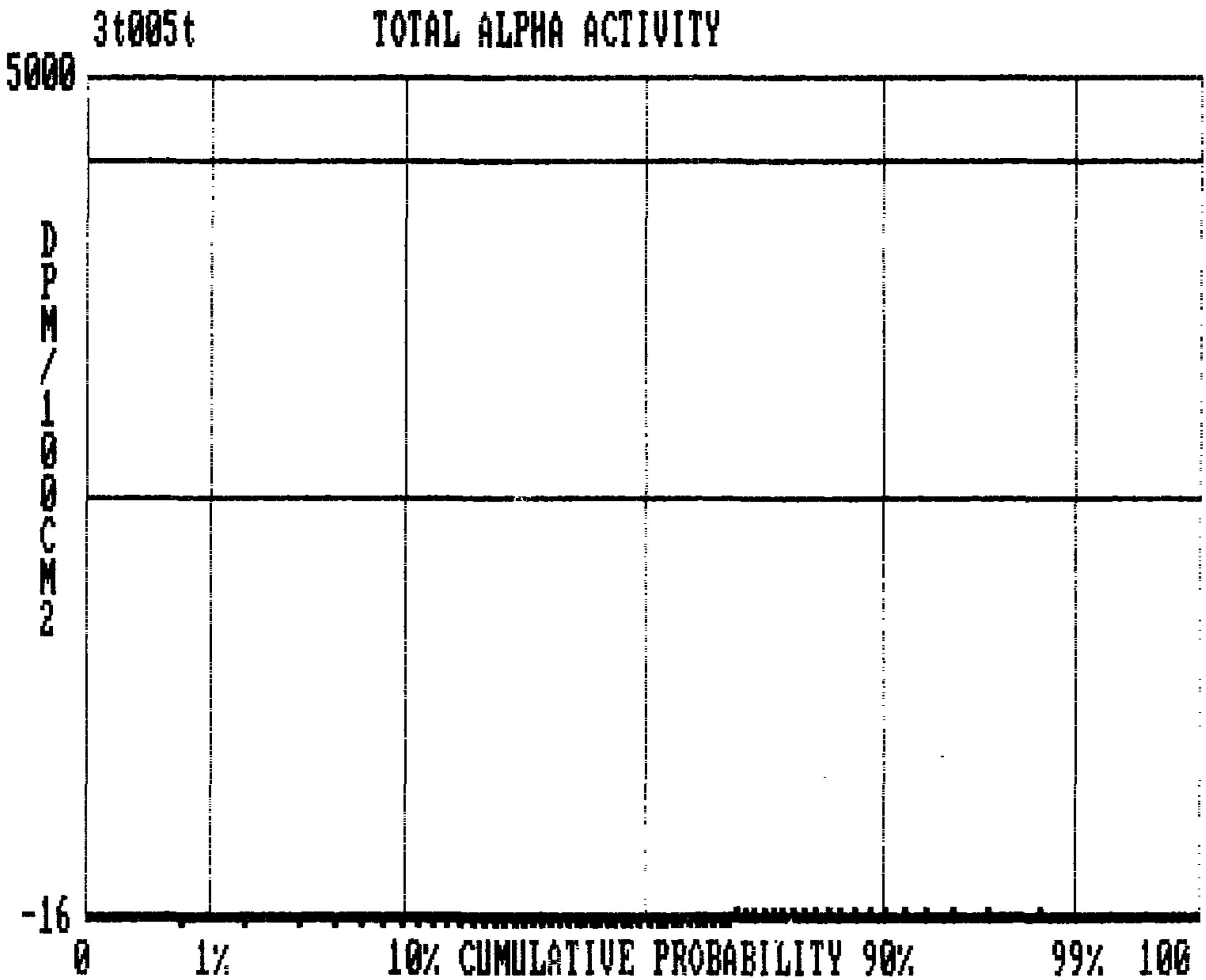


Figure 7.13. Maximum Alpha Activity in Area 3

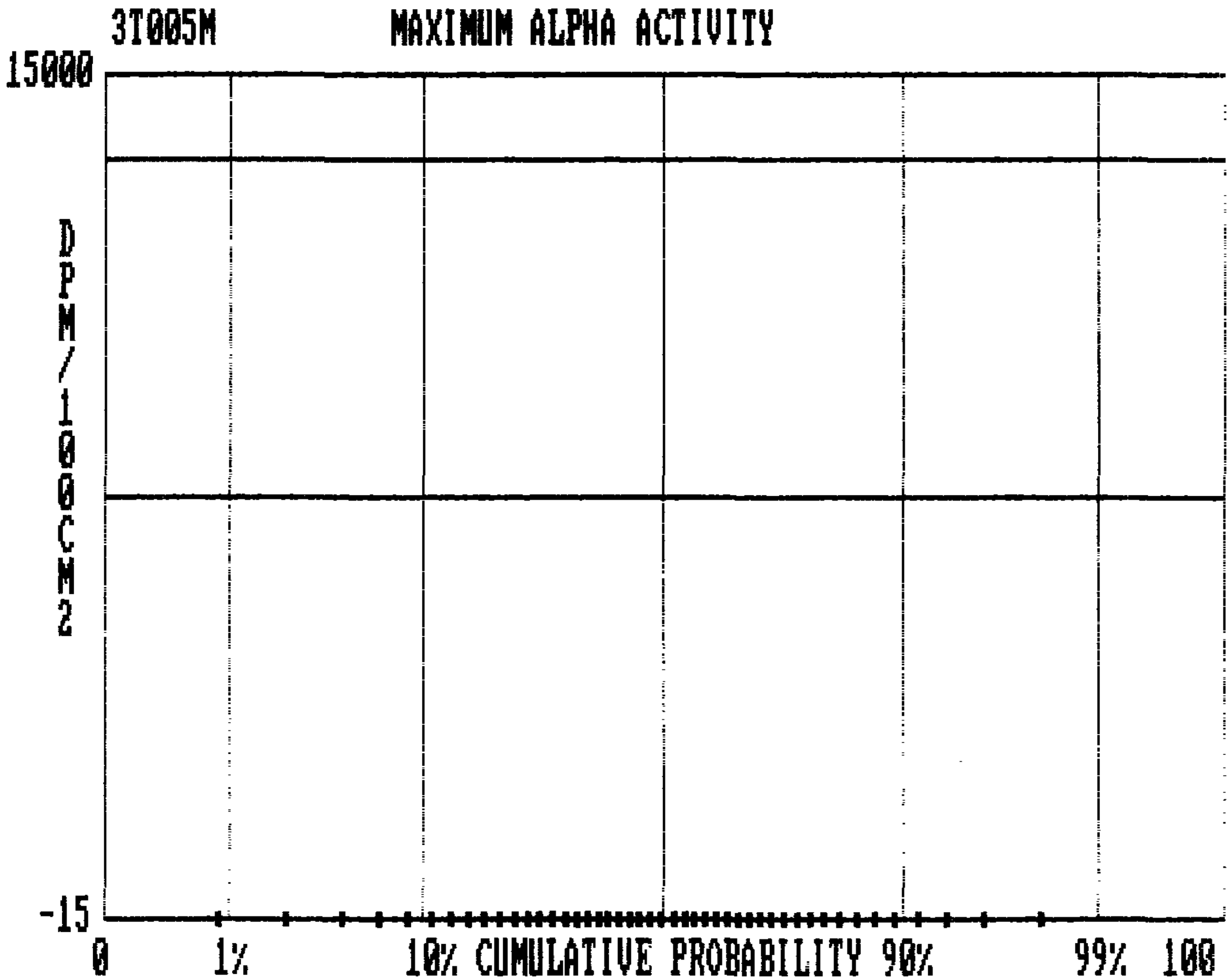


Figure 7.14. Removable Alpha Activity in Area 3

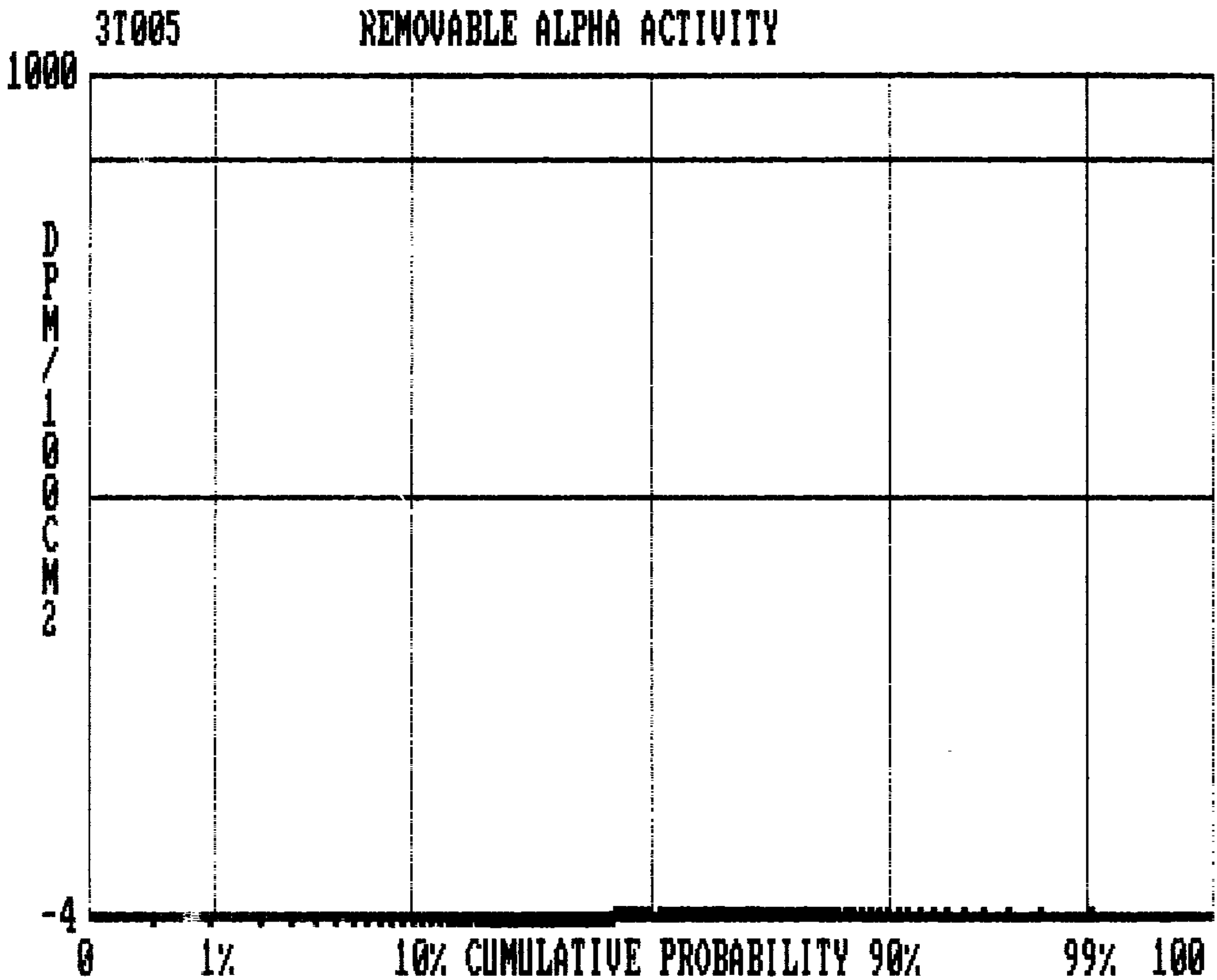


Figure 7.15. Total-Average Beta Activity in Area 3

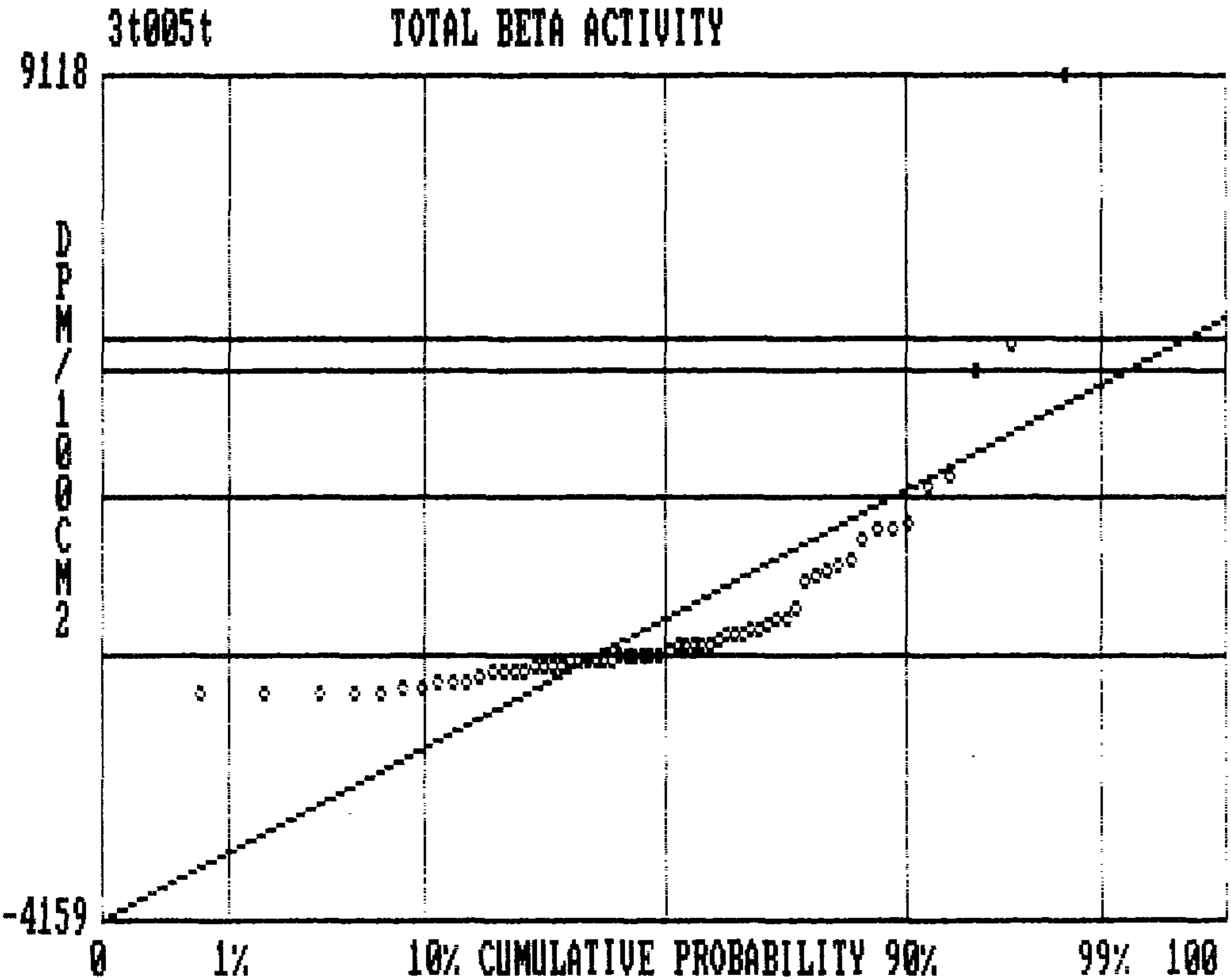


Figure 7.16. Maximum Beta Activity in Area 3

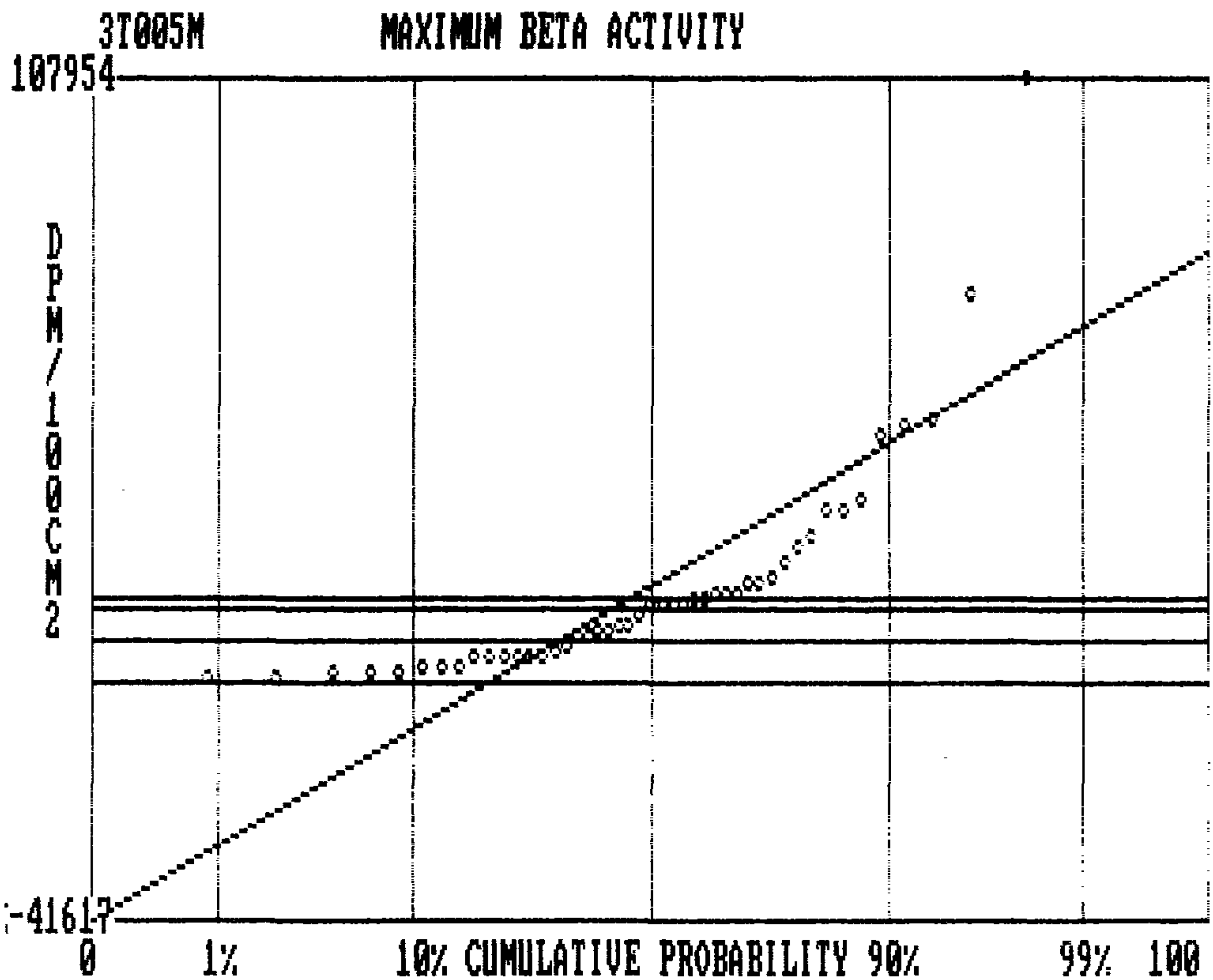


Figure 7.17. Removable Beta Activity in Area 3

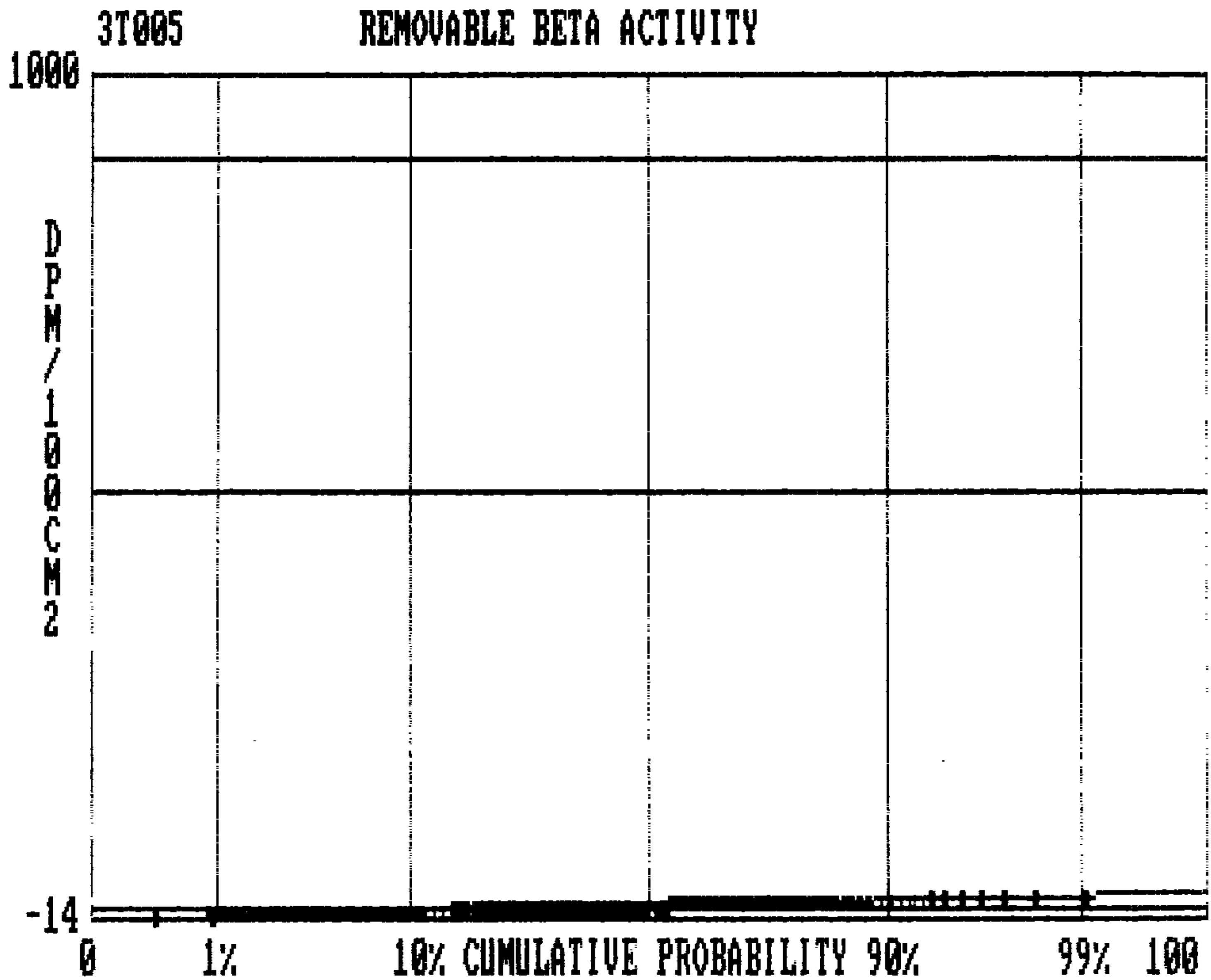


Figure 7.18. Ambient Gamma Exposure Rate in Area 3

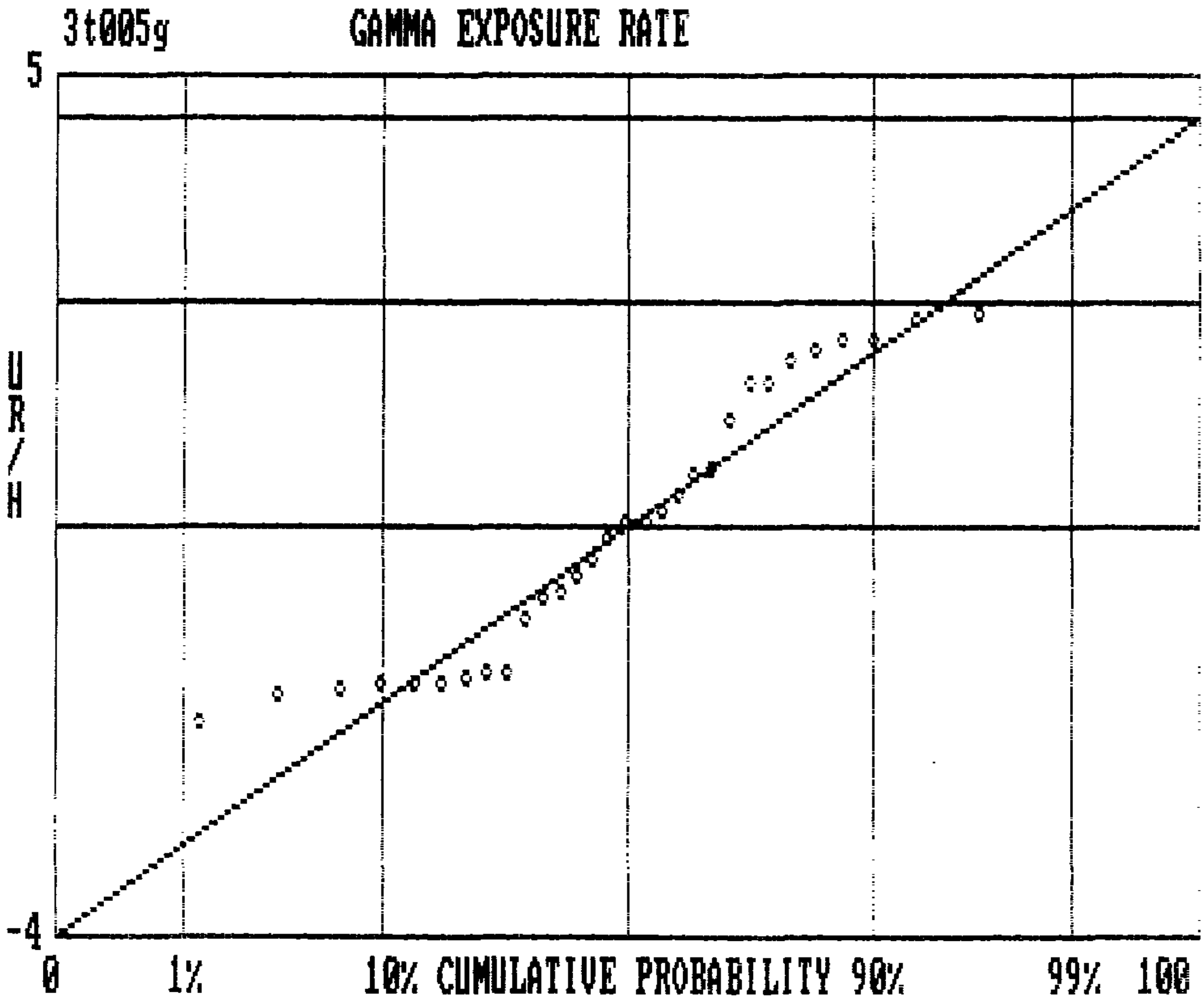


Figure 7.19. Total-Average Beta Activity on Walls of Area 3

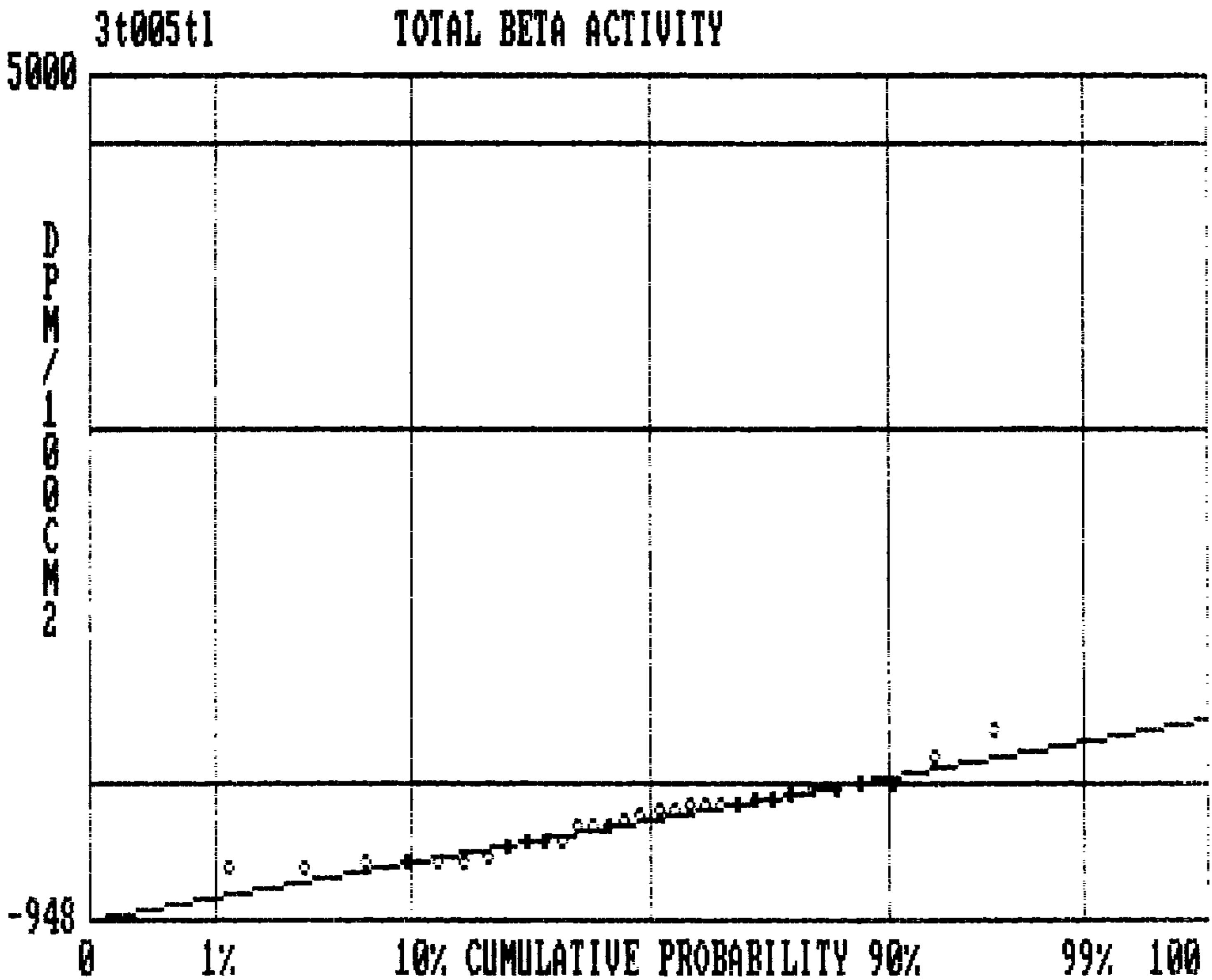
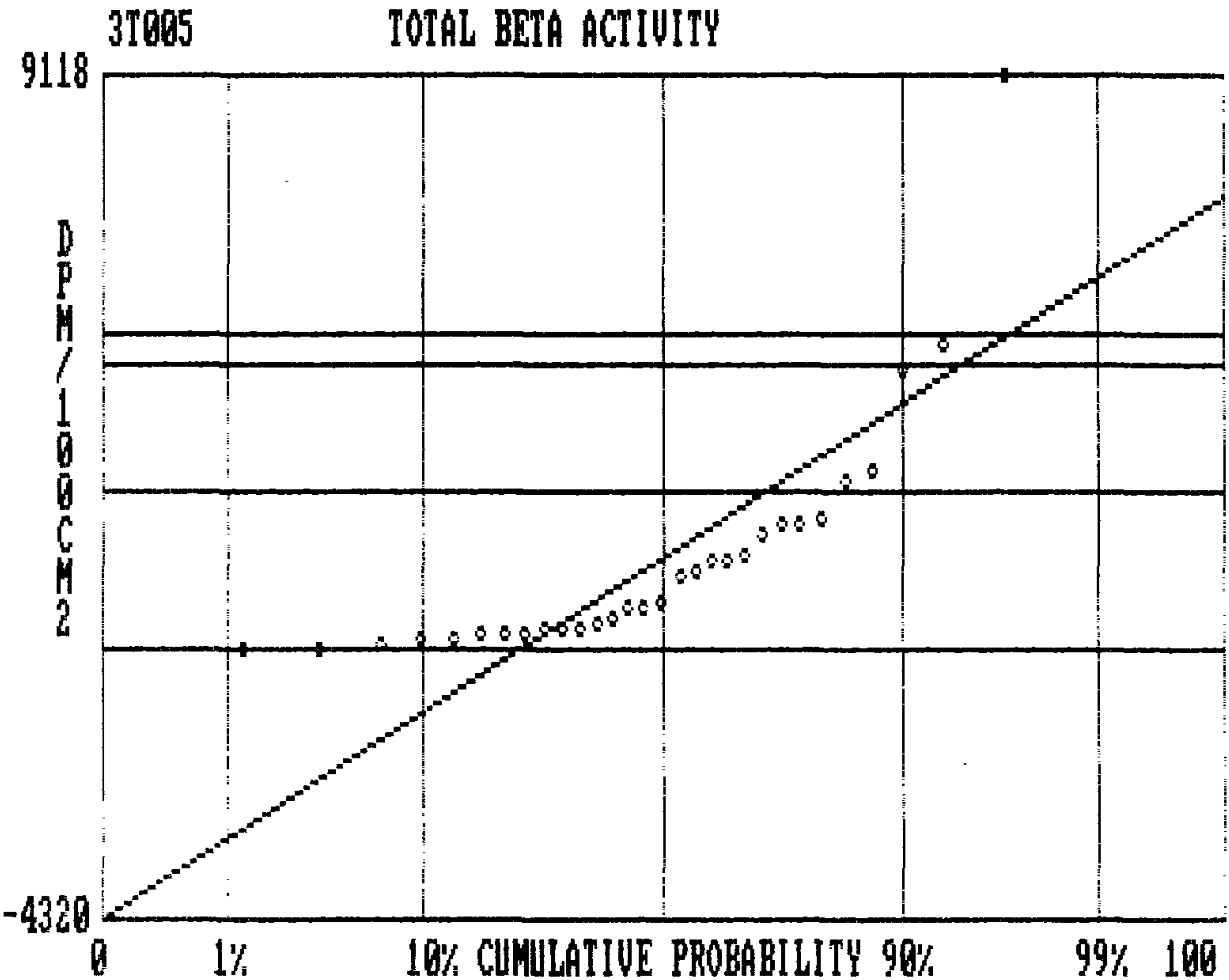


Figure 7.20. Total-Average Beta Activity on Floor of Area 3



7.5 Exterior Area (Non-Suspect)

7.5.1 General Description

Figure 7.21 graphically shows the areas surveyed outside building T005. Concrete slabs which support heavy equipment located to the north, east, and south of T005 were 11% surveyed. The large equipment items were randomly checked for contamination. Macadam blacktop surfaces surrounding the concrete slabs were also randomly surveyed. Macadam blacktop west of T005 to 17th Street and the gutter from T005 down 17th Street to G Street was 11% surveyed. The exterior survey was performed as thoroughly as the interior survey.

7.5.2 Summary Table

Table 7.4 shows the results of the exterior area. In all cases, except for gamma exposure rate, the inspection test statistic ($\bar{x}$ + ks) is far below the acceptance limit, demonstrating that the area is acceptably clean from an alpha/beta perspective. No maximum "hot spots" were detected.

The test statistic of the gamma distribution is greater than the NRC imposed acceptance limit of 5 uR/h. Either the true ambient gamma field increases in certain areas because of radiological operations conducted in nearby areas or significant contamination must be present within the surface materials. The former is suspected. Section 7.5.3 explains the reasoning for this judgement, and because increasing gamma exposure rates were observed, further inspection commenced; Figure 7.27 shows a gamma radiation map of the general area.

Table 7.4. Summary of Survey Results
(Non-suspect Sample Lot #4: Outside Areas)

Measurement	Number of Locations	Average Value	Maximum Value	Inspection Test Statistic	Limit
Average alpha (dpm/100 cm^2)	104	14.5	56	30.4	5,000
Maximum alpha (dpm/100 cm^2)	0	-	-	-	15,000
Removable alpha (dpm/100 cm^2)	104	0	1	1.0	1,000
Average beta (dpm/100 cm^2)	104	238.7	1657	1146.8	5,000
Maximum beta (dpm/100 cm^2)	0	-	-	-	15,000
Removable beta (dpm/100 cm^2)	104	5.3	14	10.5	1,000
Ambient exposure rate (uR/h)	104	3.4	14	8.6	5

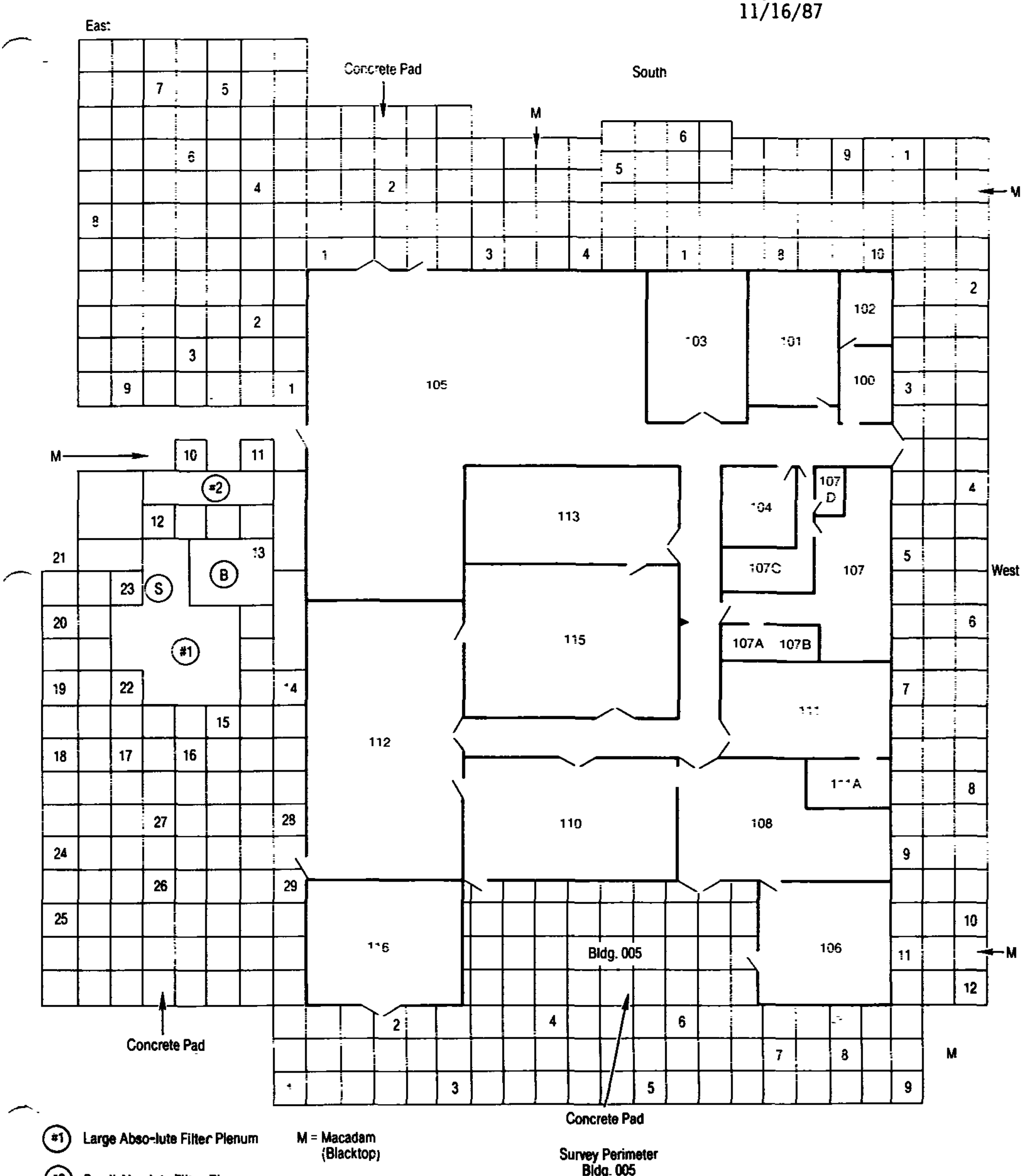


Figure 7.21. Building Exterior Sampling Inspection Plan

(#1) Large Absolute Filter Plenum

(#2) Small Absolute Filter Plenum

(#S) Large Stack

(B) Blower

7.5.3 Cumulative Probability Distributions

Figures 7.22 through 7.26 show the statistical distributions of total-average alpha, removable alpha, total-average beta, removable beta, and gamma exposure rates for the exterior area, respectively. No maximum "hot spot" measurements were observed in any of the square meters.

Figures 7.22, 7.23, and 7.25 show model Gaussian cumulative distributions with no outliers. The test statistic values are far below the acceptance test limits. Figure 7.24, total-average beta contamination, is a model Gaussian cumulative distribution with one outlier point at 1600 dpm/100cm^2, below the 50% characterization limit. The inspection test statistic ($\bar{x}$ +ks) of the beta distribution is 1147 dpm/100 cm^2, greater than the interior measurements. This is probably due to an increase in "beta" background. Further inspection was not necessary. No isolated alpha/beta "hot spots" were found.

The ambient gamma exposure rate distribution is not a Gaussian plot. The variability of data points about the mean of the distribution displays several independent Gaussian distributions. Because the average value of the distribution is above the 50% characterization level and the test statistic ($\bar{x}$ + ks) is greater than the acceptance limit, further investigation commenced according to Section 4.3. One of two hypotheses is assumed to be true:

1) The true ambient gamma exposure rate increases in certain areas closer to facilities which use or store strong radioactive sources which emit penetrating gamma rays;

2) Contamination is present in localized areas. The contamination does not emit alpha and/or beta particles which are detectable under current conditions.

All follow-up inspection data and analysis indicate that the first hypothesis is the most plausible. Because no contamination was detected on the smears of the exterior sampling distribution, hypothesis 2 is discounted. Furthermore, the soil samples taken from the T005 gutter, drainage sumps, and downstream ditches suggest that no strong gamma emitters are present in such concentrations that they would be detectable at the road surface. Refer to section 7.9 for soil and sediment results.

Upon further inspection using the gamma probe, it was discovered that the ambient gamma exposure rate increases upon approach to the Radioactive Materials Disposal Facility (RMDF), shown in Figure 7.27. Figure 7.27 was generated to demonstrate this feature. The amount of radioactive materials stored and handled at the RMDF changes on a frequent basis, thus the relative background at the fence line changes proportionally. The fence line dose rate never exceeds 2 mrem/h, thus all surrounding areas are operated within regulatory requirements. From Figure 7.27, it is clear that the decrease in exposure rate as a function of distance from the RMDF does not follow the inverse square law. This exposure rate departure from theory is observed for two reasons: the terrain and building structures prevent a planar non-obstructive $1/r^2$ extrapolation, and the skyshine contribution to exposure rate is uncertain, but is known to contribute to ambient gamma flux. The data and subsequent follow-up show that the ambient gamma exposure rate increases because of radiation related activities conducted at RMDF.

The 5 uR/h ambient gamma exposure rate acceptance limit imposed by the NRC is sufficiently small, that variations in background must be carefully evaluated. Because the test statistic ($\bar{x}$ + ks) exceeds the limit, and because the cumulative probability distribution is disjointed, further inspections and analyses show that these fluctuations in background should be expected in areas near the RMDF, within several hundred yards. No substantial evidence suggests that the pavement or underlying surfaces are contaminated.

Figure 7.22. Total-Average Alpha Activity in Exterior Area

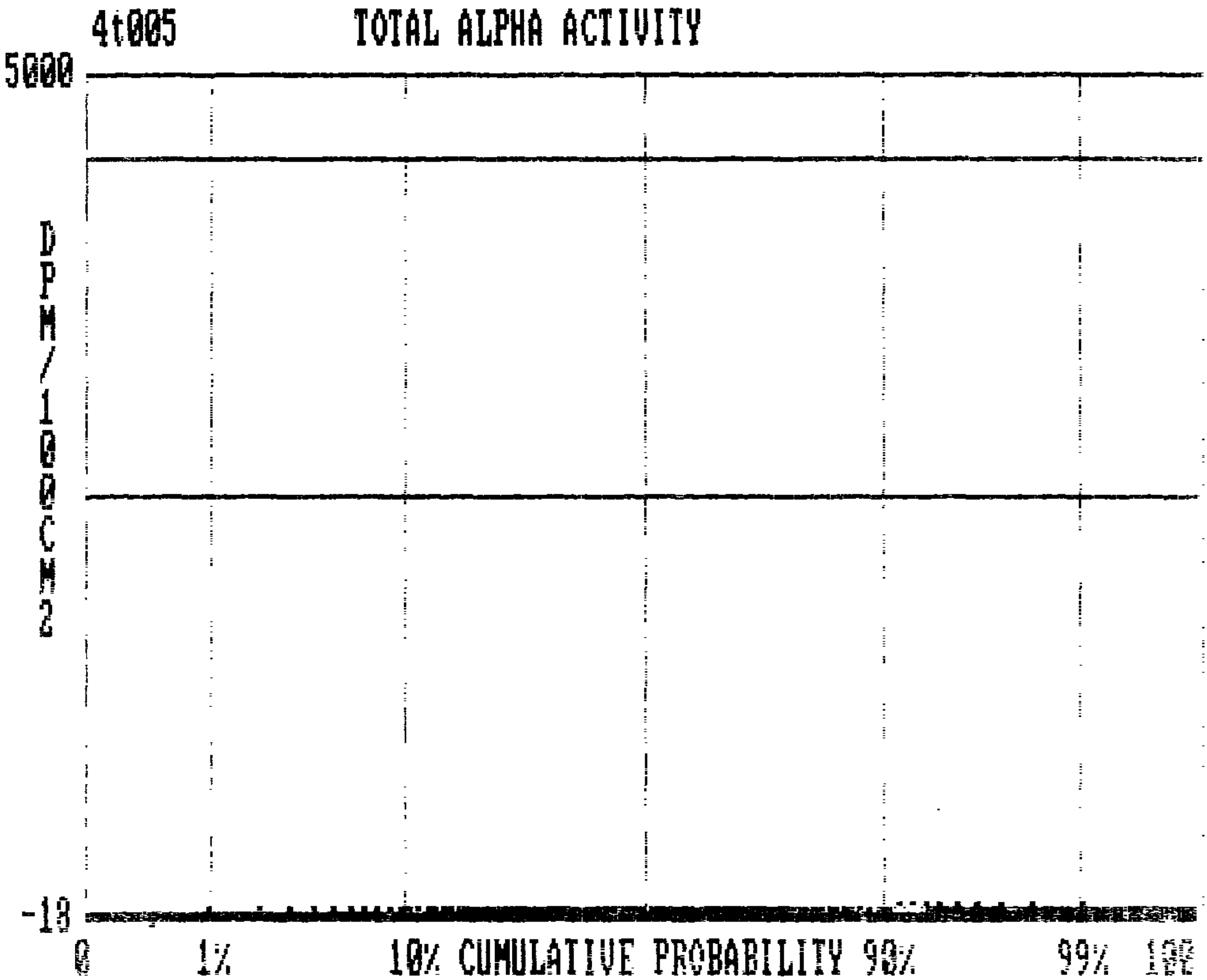


Figure 7.23. Removable Alpha Activity in Exterior Area

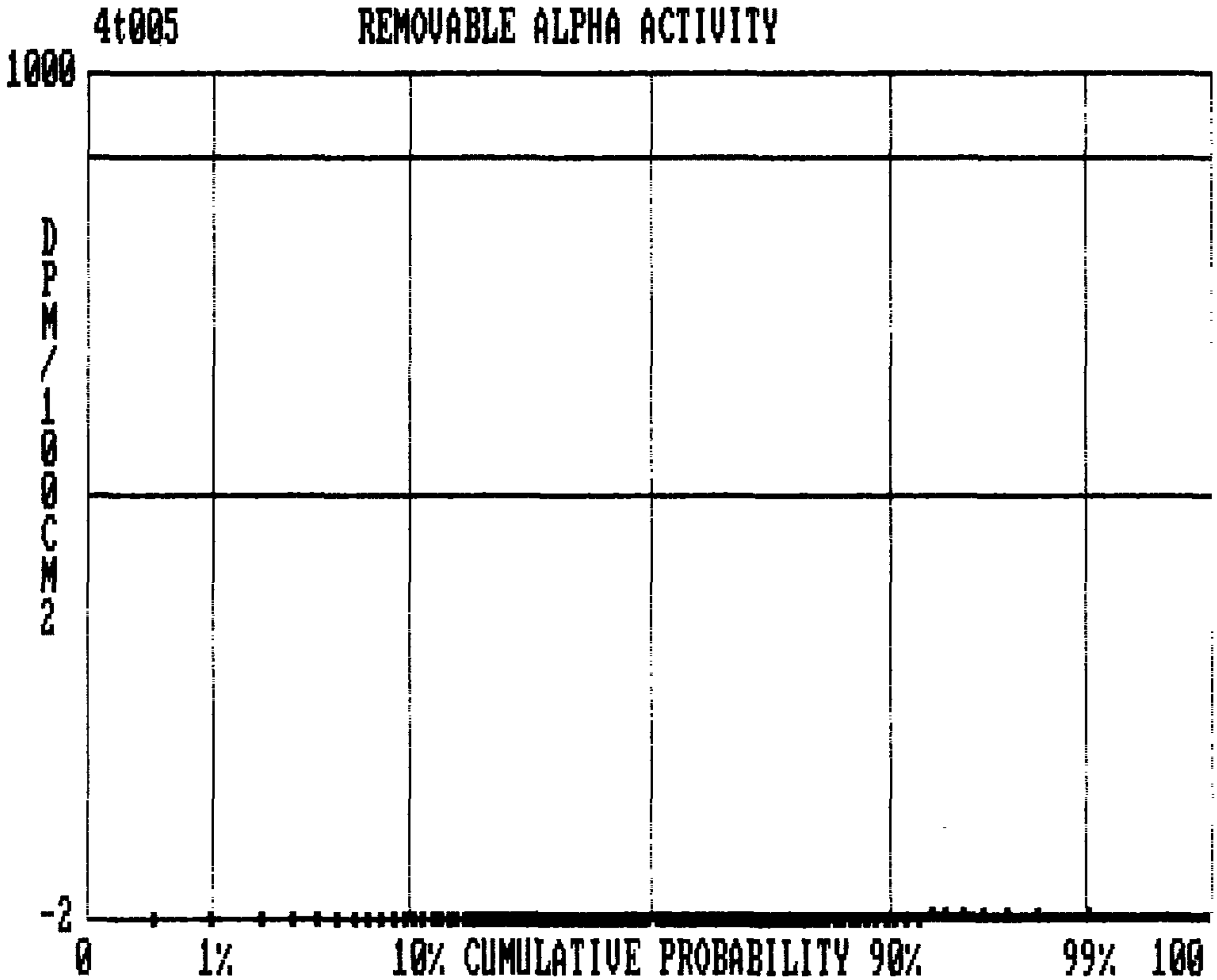


Figure 7.24. Total-Average Beta Activity in Exterior Area

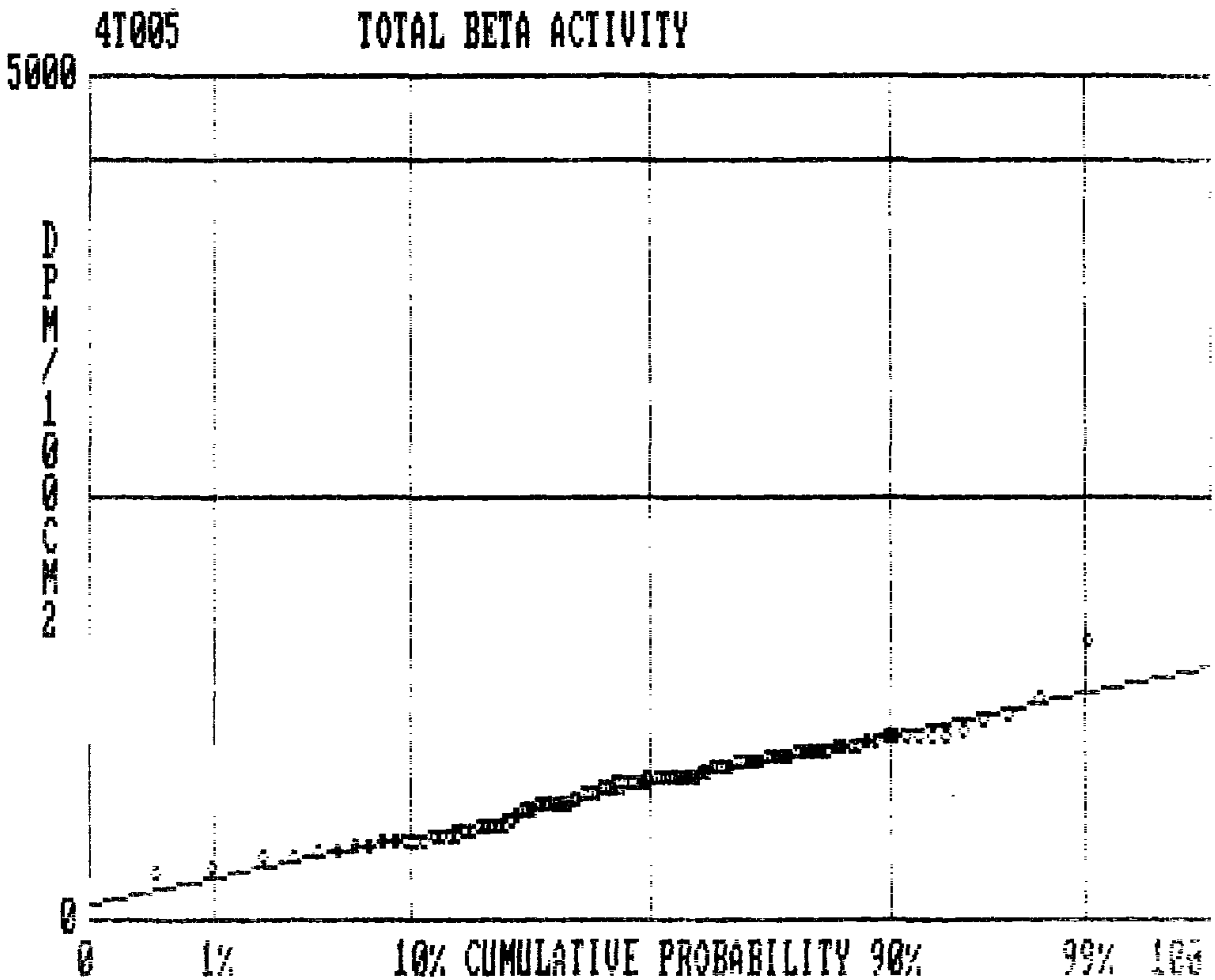


Figure 7.25. Removable Beta Activity in Exterior Area

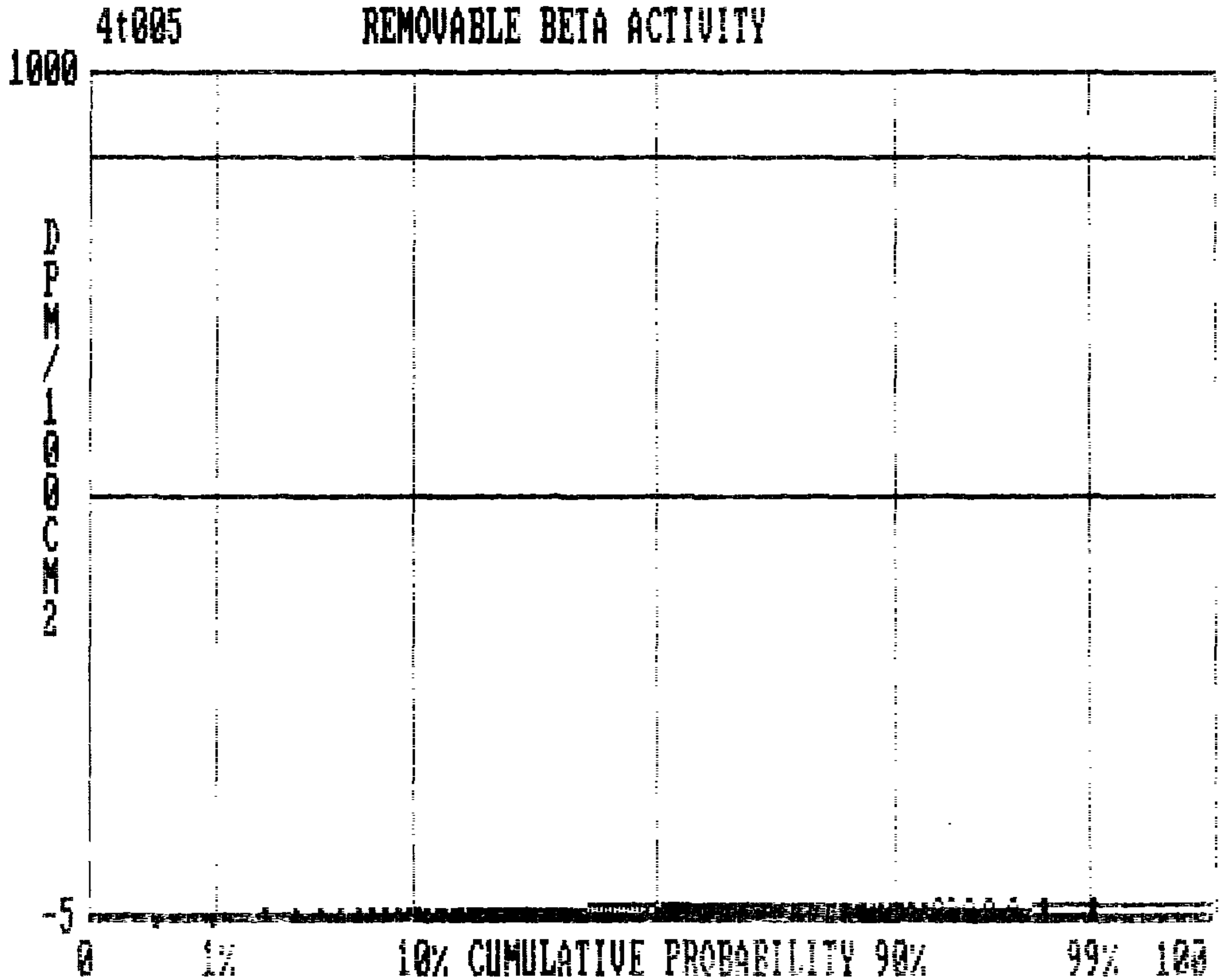


Figure 7.26. Ambient Gamma Exposure Rate in Exterior Area

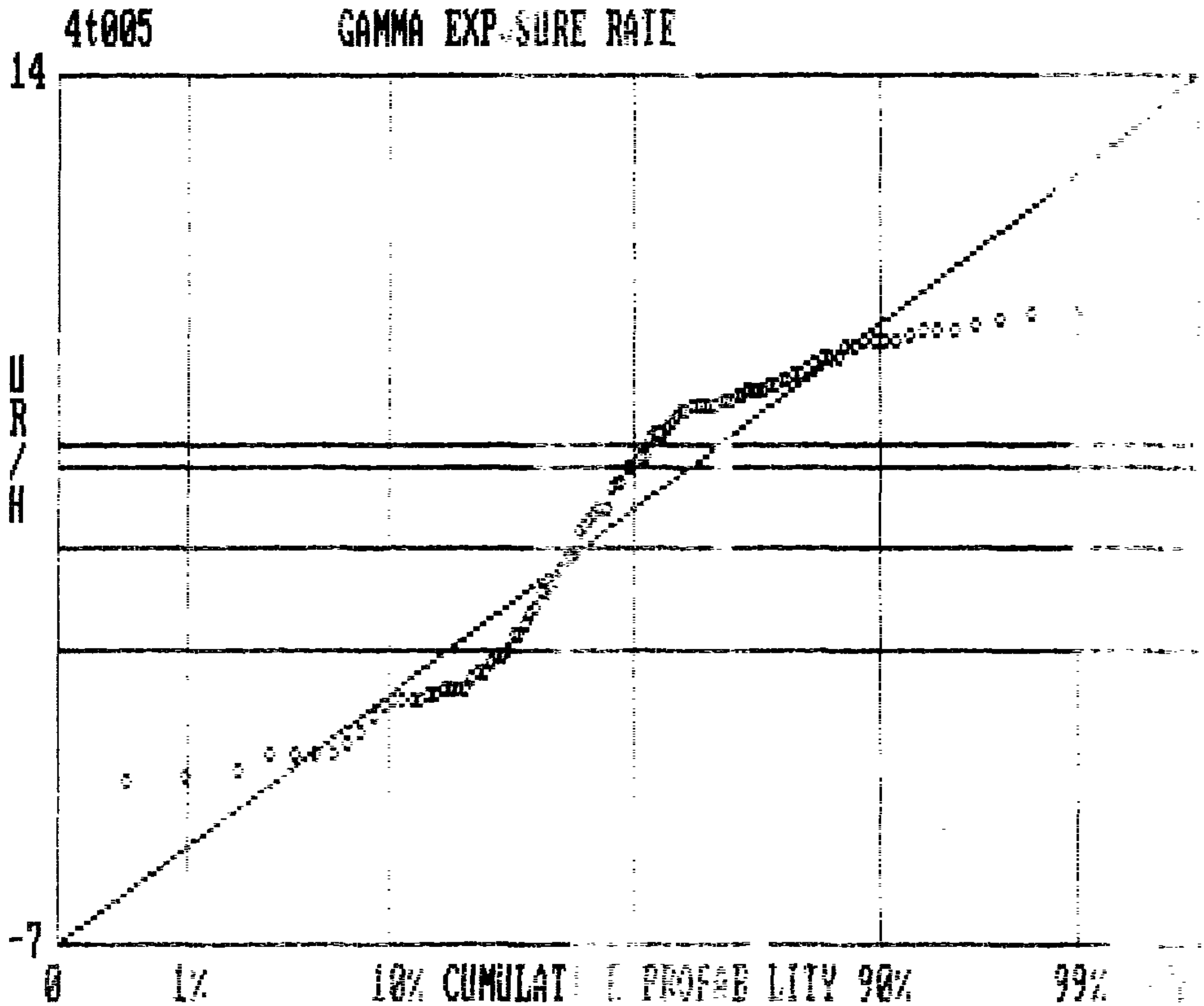
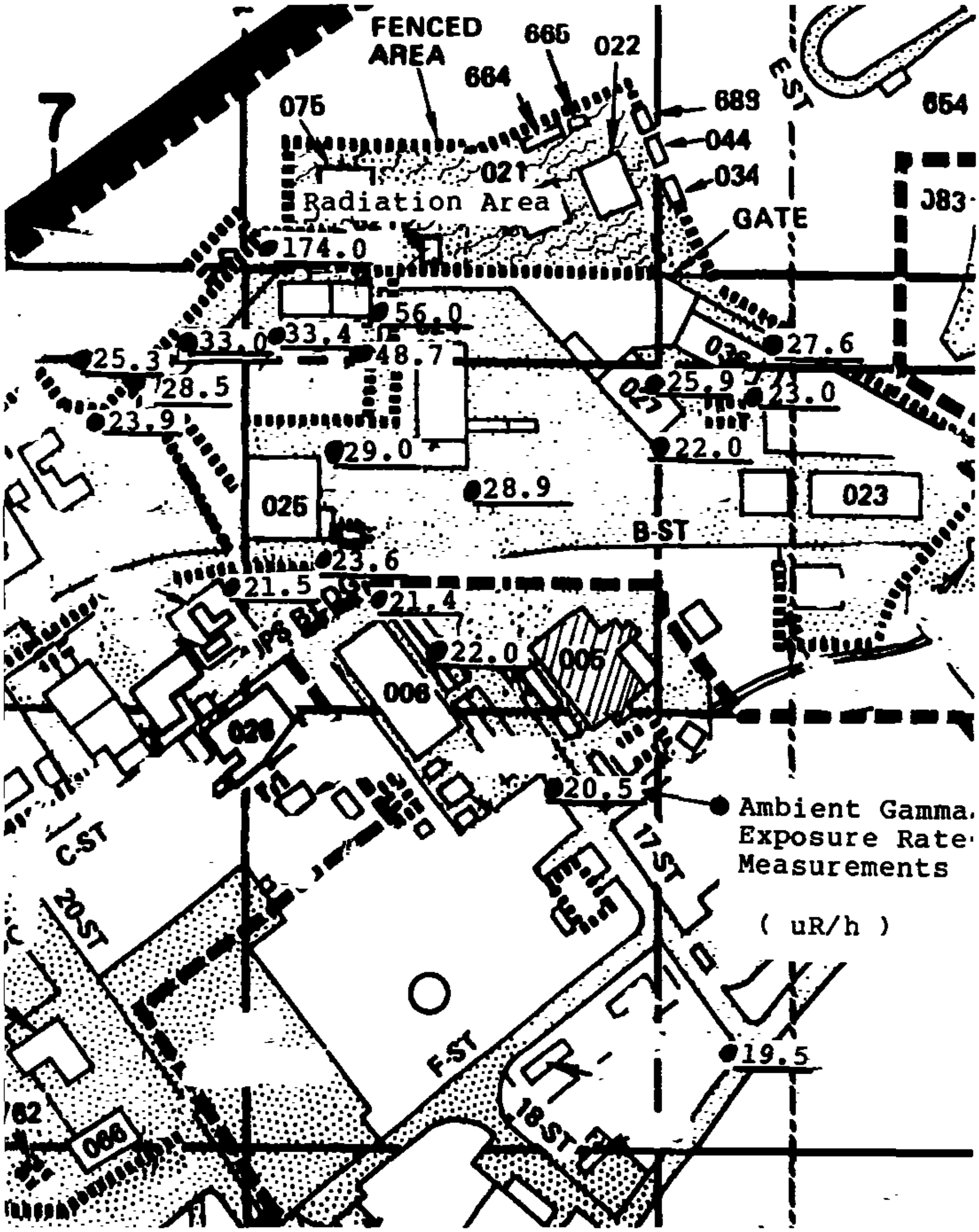


Figure 7.27 Ambient Gamma Exposure Rate Map of T005 and Surrounding Area



7.6 Special Structural Features (Non-Suspect)

7.6.1 General Description

Many structural features were surveyed for radioactive contamination. These features include exterior surveys of air conditioning ducts; radioactive material exhaust ducts and entrance vents; pipes; support I-beams; light fixtures; and other ancillary structures. Most measurements were made in the attic above rooms 115 and 113. Each surface was smeared for removable contamination and randomly surveyed for the existence of total contamination. Because of the geometrical nature of these special features, application of efficiency factors and area factors are not appropriate for the total survey. The objective was to determine whether any contamination was present or not. Contamination quantification is difficult under these conditions.

7.6.2 Summary Table

Table 7.5 shows contamination results of the special structural features. Each surface was "frisked" with alpha and beta probes coupled to count rate meters. In all cases, there was no detectable activity. Ambient gamma exposure rate measurements were not applicable to these features. From the statistical analysis output, it is shown that the inspection test statistic for removable contamination is far below the acceptance limit. No contamination was found in the attic, external to the radioactive material exhaust ducts. Internal measurements of the R/A exhaust ducts were performed; these results are presented in Section 7.7.

Table 7.5. Summary of Survey Results
(Special Structural Features, Pipes, I-beams,
Light Fixtures, External Duct Smears)

Measurement	Number of Locations	Average Value	Maximum Value	Inspection Test Statistic	Limit
Average alpha (dpm/100 cm^2)	200	No detectable activity			5,000
Maximum alpha (dpm/100 cm^2)	0	No detectable activity			15,000
Removable alpha (dpm/100 cm^2)	200	1.8	-	14.9	1,000
Average beta (dpm/100 cm^2)	200	No detectable activity			5,000
Maximum beta (dpm/100 cm^2)	0	No detectable activity			15,000
Removable beta (dpm/100 cm^2)	200	4.2	-	37.3	1,000
Ambient exposure rate (UR/h)	0	Not applicable		-	5

7.6.3 Cumulative Probability Distributions

Figures 7.28 and 7.29 show the statistical distributions of removable alpha, and removable beta for special structural features, respectively.

Figure 7.28, removable alpha contamination, shows a Gaussian cumulative distribution with an outlier from the distribution with a value of 150dpm/100cm^2, far below the 50% characterization level. Figures 7.29, removable beta contamination, shows a distribution with a few outliers approaching 250 dpm/100cm^2. These alpha/beta positive results were from inside the light fixtures of room 115. Although the maximum readings taken were far below the 50% characterization level, a few more light fixture smears were collected and analyzed in both distributions. None of the special structural features surveyed are contaminated above the 50% characterization level.

Figure 7.28. Removable Alpha Activity on Special Features

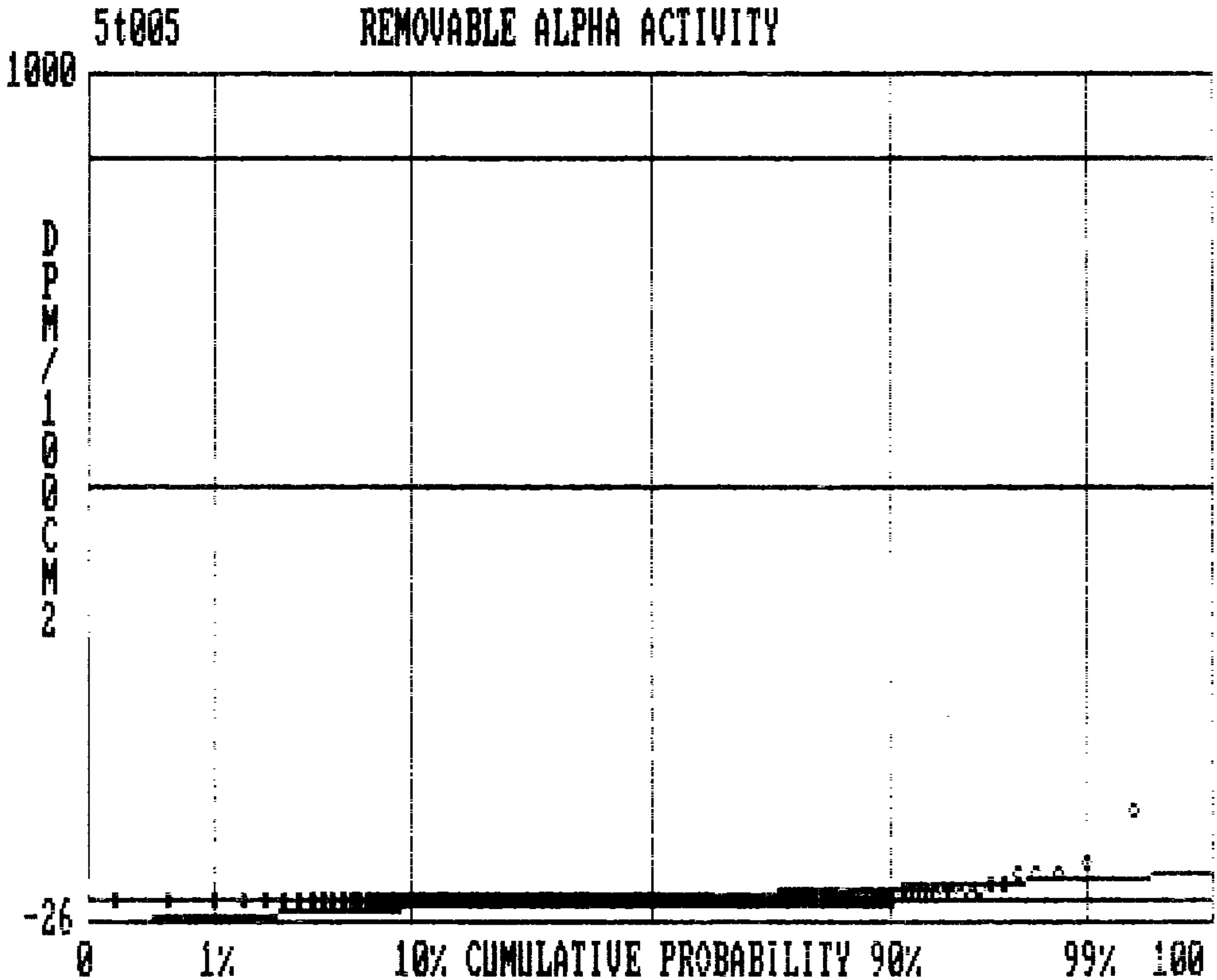
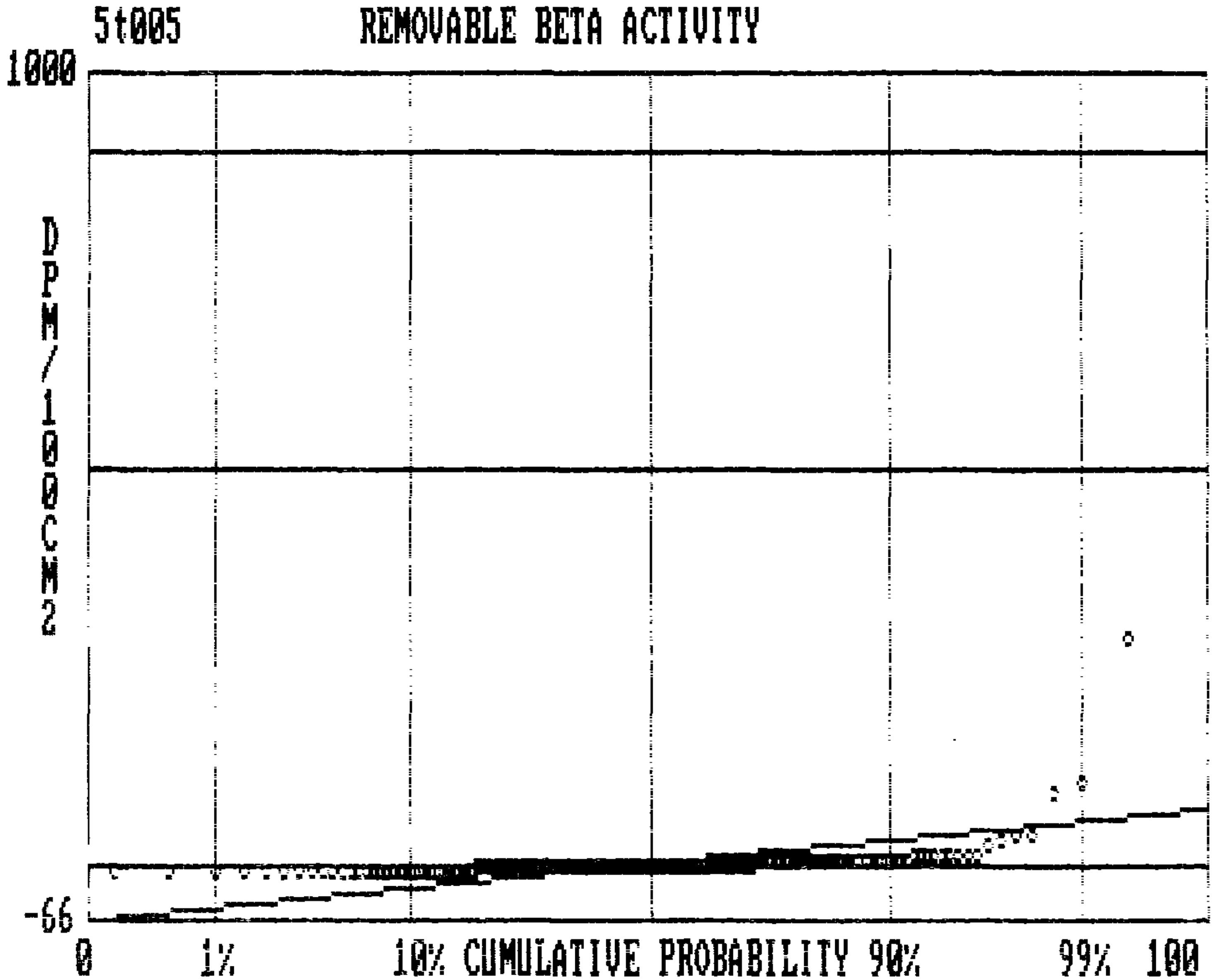


Figure 7.29. Removable Beta Activity on Special Features



7.7 Radioactive Material Exhaust Ducts (Suspect)

7.7.1 General Description

Figure 7.30 shows the locations of the remaining radioactive material exhaust ducts in T005. The duct diameters vary from 12 in. to 30 in. Radiation surveys were performed at all accessible view ports, capped access penetrations, and entrance vents. Where possible, alpha and beta probes were placed inside ducts to obtain a total contamination measurement. However, in several cases, because of potential for contaminating the probes, these surveys were limited. Smears for removable contamination were taken for alpha/beta analysis. A few crud and dust samples were taken from ducts for gamma spectrometry analysis. Figure 7.31 shows a section of ductwork remaining in the attic above room 115.

7.7.2 Summary Table

Tables 7.6 and 7.7 show the results for removable contamination and total contamination, respectively. Table 7.6 shows that only 10 removable samples were taken from R/A ducts. The access points were very limited. However, the survey did not require a large sampling inspection to demonstrate that significant amounts of contamination remain inside each duct, as expected. The standard deviation is presented below the average value in Table 7.6. The average value in this case is not a very meaningful statistic when the standard deviation is large and the sample size is small. The inspection test statistic accounts for small sample sizes and large standard deviations. In both cases, the inspection test statistic exceeds the removable contamination acceptance limit of 1000dpm/100cm^2.

Table 7.6. Summary of Survey Results
(Removable Radioactivity from R/A Exhaust Ducts)

Measurement	Number of Locations	Average Value	Maximum Value	Inspection Test Statistic	Limit
Removable alpha (dpm/100 cm^2)	10	417.3 $\pm$683	2467	1792.3	1,000
Removable beta (dpm/100 cm^2)	10	1053.2 $\pm$1750	6302	4574.6	1,000

Total contamination was measured directly in only a few locations because of portal limitations relative to detector size. Table 7.7 shows the results and clearly demonstrates that the ducts are contaminated. As described for the special structural feature survey, the probes in this case were coupled to count rate meters, rather than scalers. The results are consequently reported as less than so many alpha or beta particle disintegrations per minute.

Table 7.7. Summary of Survey Results
(Total Radioactivity from Inside R/A Exhaust Ducts)

Location	Average alpha-dpm	Maximum Value beta-dpm
1, Backside of Plexiglass Window	98,000	104,000
2, Bottom of Duct Surface	7,000	31,200
3, Bottom of Duct Surface	18,200	15,600
4, Bottom of Duct Surface	31,500	52,000
5, Backside of Plexiglass Window	135	520
6, Bottom of Duct Surface	135	730
7, 8, 9, Bottom	420	520
10, Bottom of Duct Surface	70	50
11, Gasket (outside)	NDA	2460
Gasket (inside lip)	700	2900
Bottom of Duct Surface	*	6960
Bottom of Duct Surface	*	23,200
Gamma Exposure Rate Below Duct at Location 11	30uR/h	

* Alpha detector would not fit into access way; no data available

NOTE: Location of survey shown in Figure 7.30.

Figure 7.30. Radioactive Material Exhaust Ducts

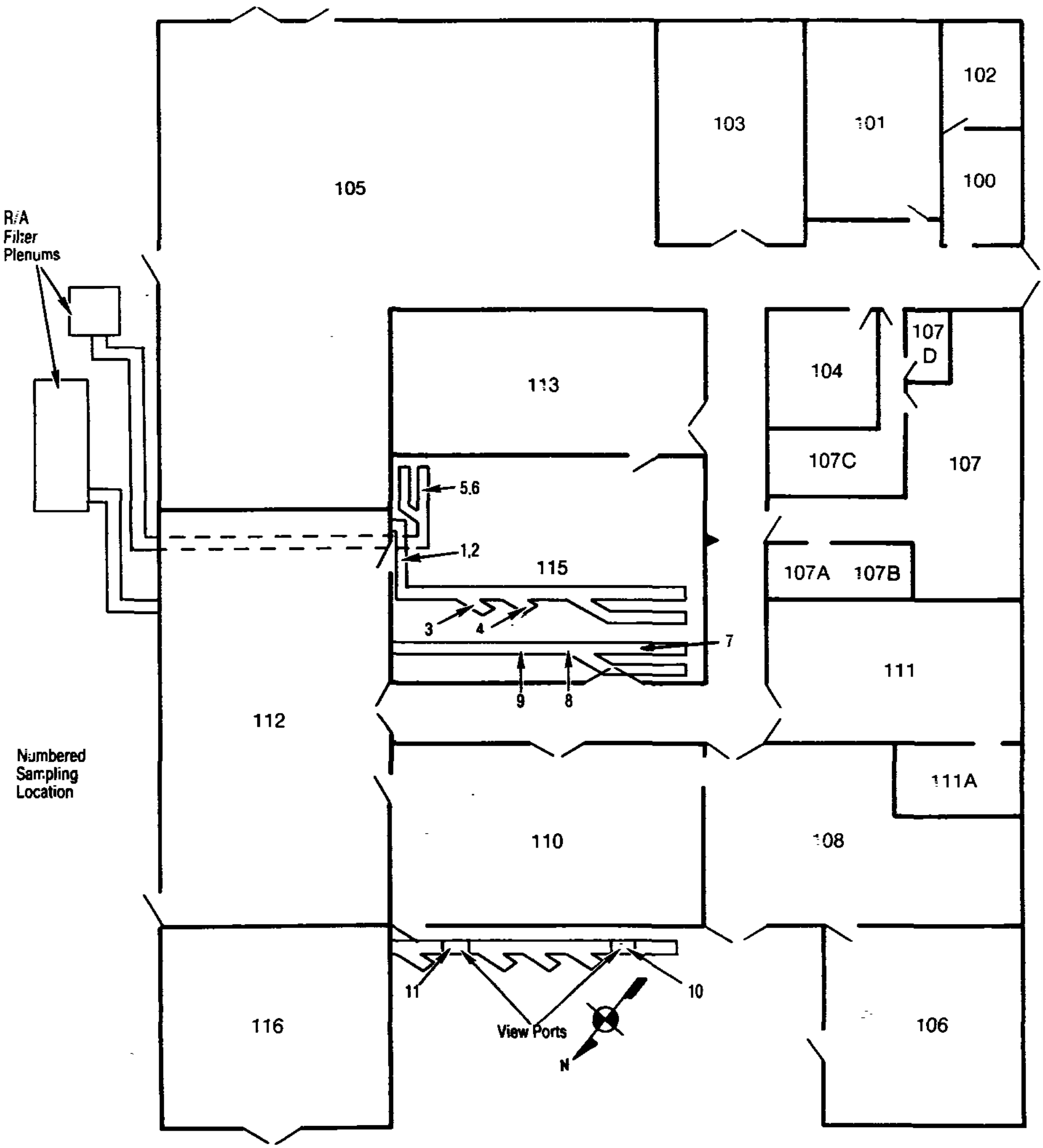
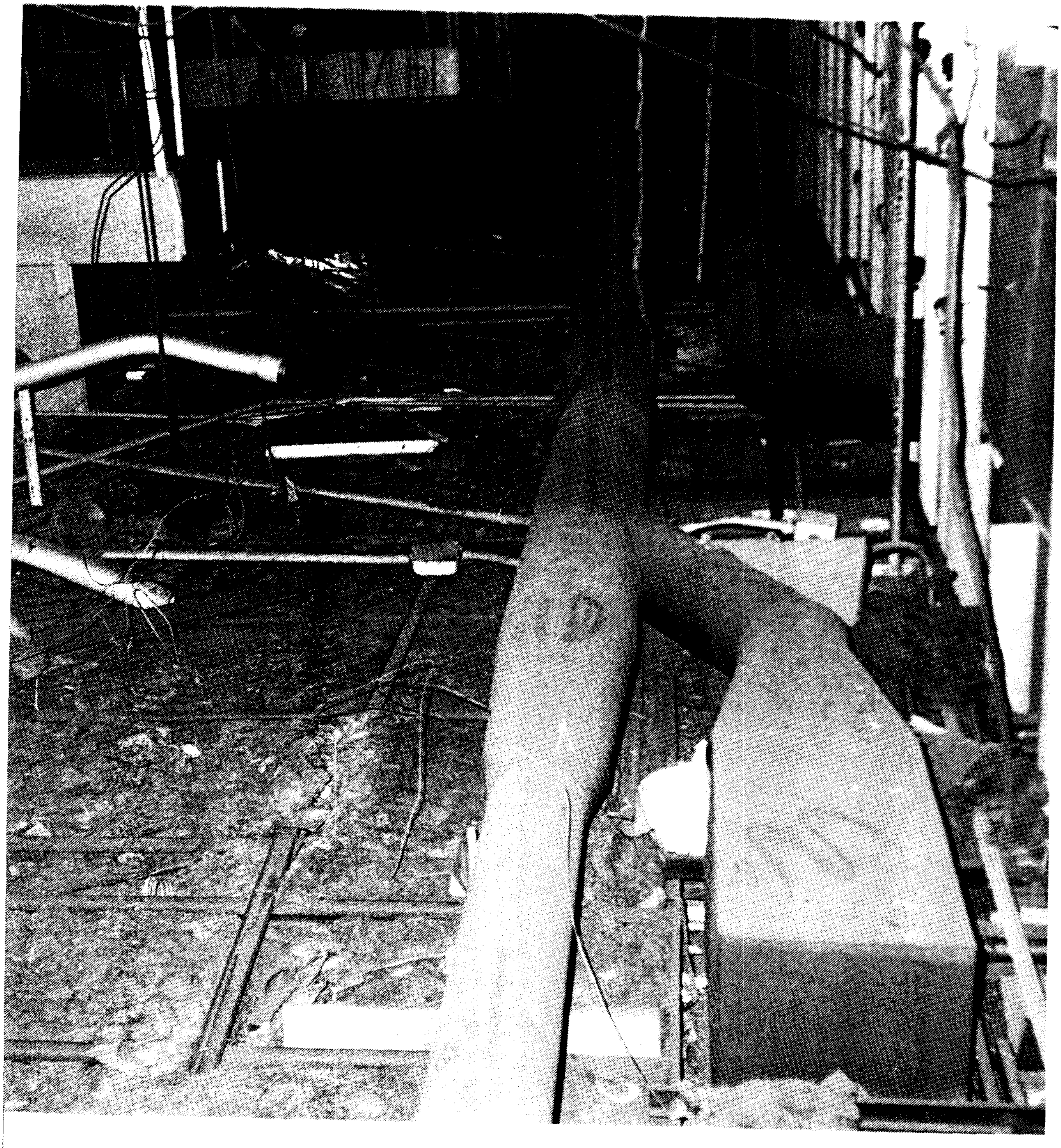


Figure 7.31. Photo of Radioactive Material Exhaust Ductwork in Attic



7.7.3 Cumulative Probability Distributions

Figures 7.32 and 7.33 show the statistical distributions of removable alpha, and removable beta from inside the radioactive material exhaust ducts, respectively. Both distributions have very similar shapes with large deviations in the measurements. The beta to alpha activity ratio is very consistent (2.47 ± 0.28). Theoretically, for natural uranium, this ratio would be about 2, and for 10% enriched uranium, this ratio would be about 1.9. The experimental value reported here is a little high because the alpha efficiency is actually lower than reported because of self absorption in the filter media. Neither distribution approximates a Gaussian cumulative distribution function, indicating that the sampling area is contaminated. The degree of contamination is large based on the inspection test statistic value reported in Table 7.6.

All radioactive material exhaust ducts must be removed before the building can be cleared for unrestricted use.

Figure 7.32. Removable Alpha Activity Inside R/A Exhaust Ducts

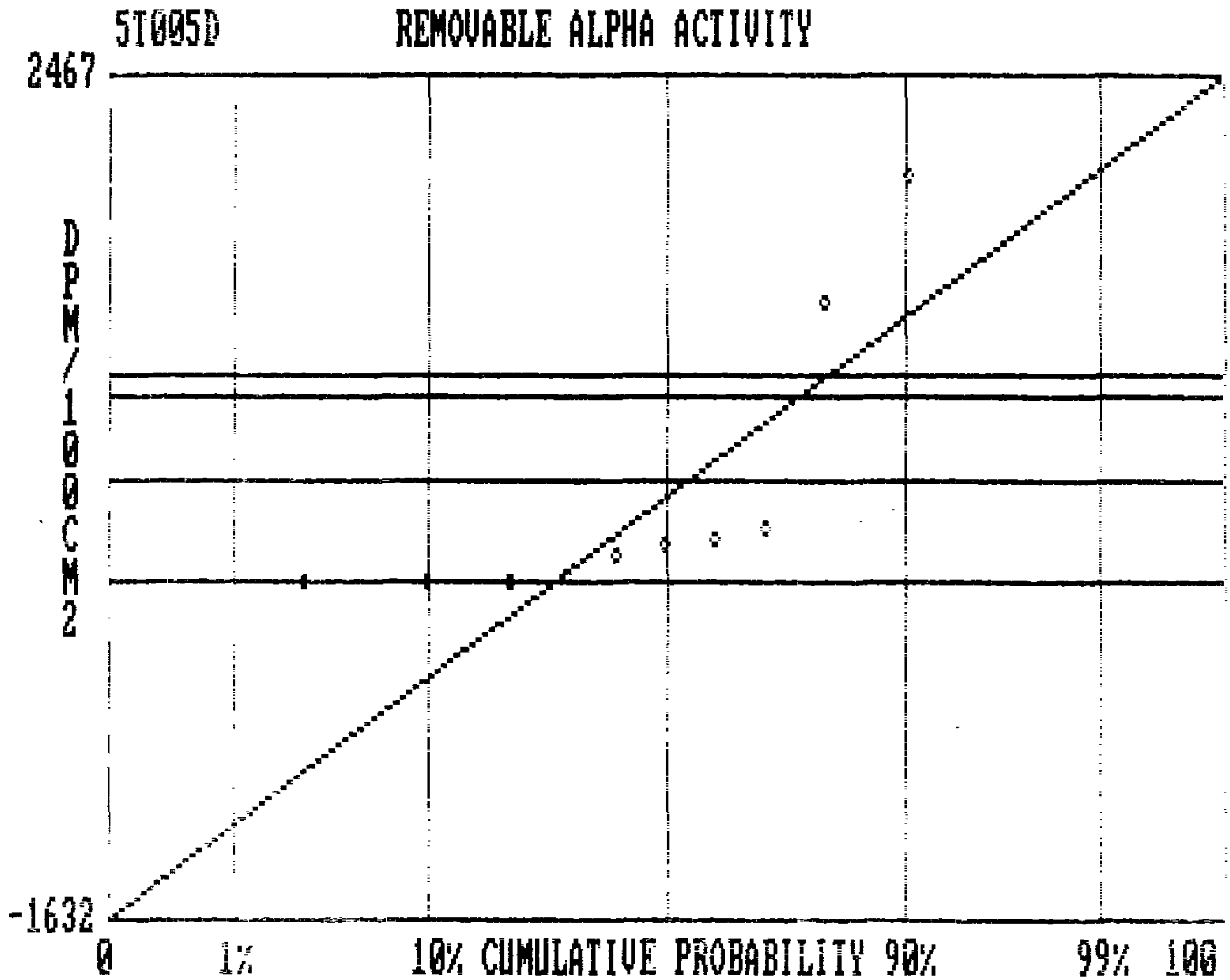
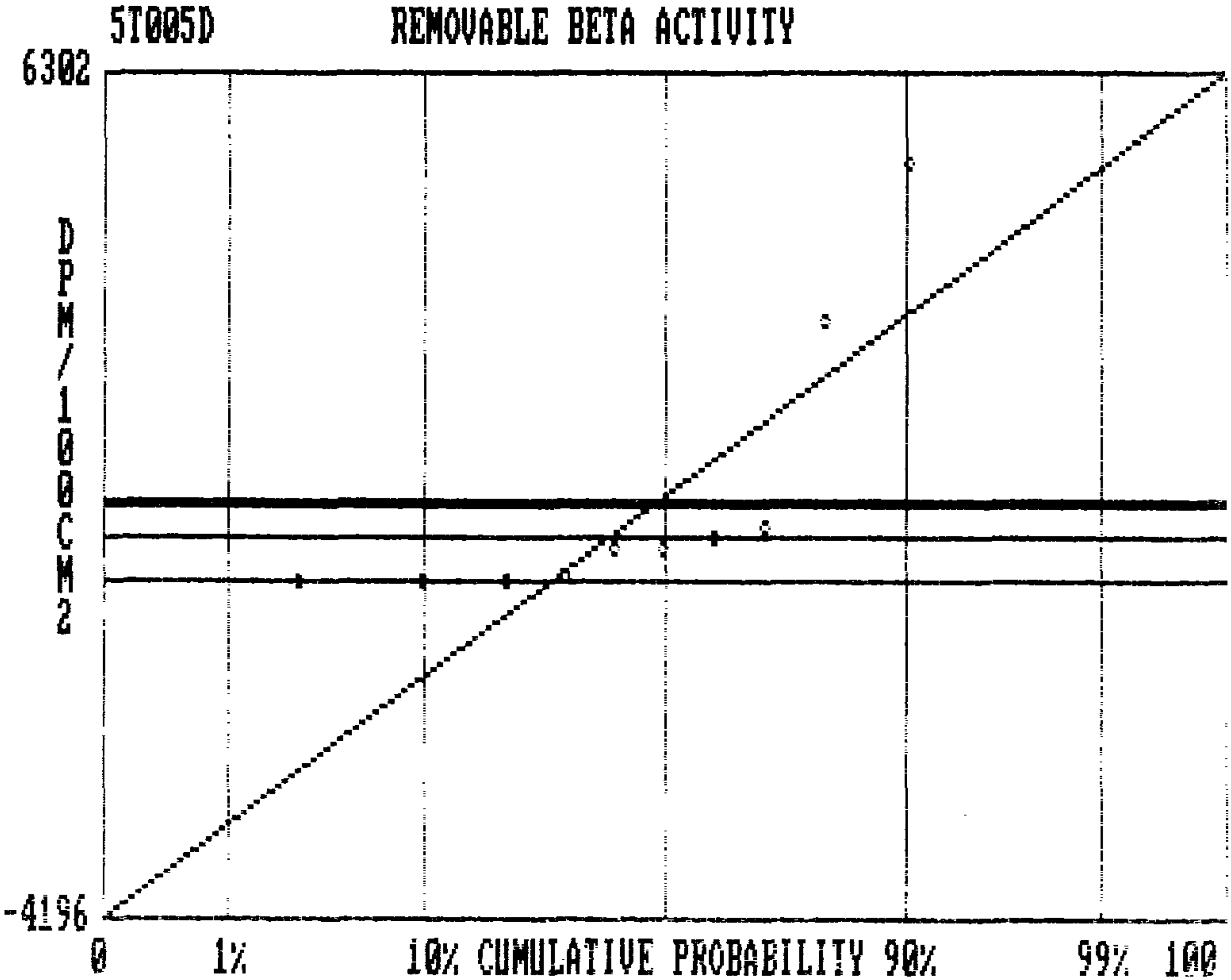


Figure 7.33. Removable Beta Activity Inside R/A Exhaust Ducts



7.8 Radioactive Material Exhaust Filter Plenums

7.8.1 General Description

During fuel production, all controlled areas within T005 were ventilated and then exhausted through filter plenums before discharge through the stack. These filter plenums contained both pre-filters and absolute filters which are still in place. All of the process equipment, with the exception of the grinders, were exhausted to the main radioactive material filtering system. The machining equipment located in room 115 was exhausted through a separate small plenum (12' x 5' x 4') prior to discharge through the stack. This small plenum contains four pre-filters and four absolutes. Each induction furnace in room 110, and each tilt pour furnace in room 112 exhausted through individual filter banks containing four pre-filters, prior to exhausting to the main filter plenum.

Following the use of T005 as a controlled radioactive material handling and processing facility, the radioactive material exhaust system was mothballed. All of the R/A exhaust outlets from each room have since been cut and bagged in the attic. Selected ducts have also been removed. All filters are still in place.

At the time of the radiological survey, the R/A exhaust system consisted of:

1. an exhaust stack (3' diameter x 20' tall)
2. main radioactive material filter plenum (20'x12'x8')
3. process equipment R/A filter plenum (12'x5'x4')
4. R/A ducting from room 112 to each plenum
5. three R/A ducts in the attic above room 115; and
6. an R/A duct in the north outside patio area.

Other than cutting exhaust outlets and removing a few sections of ducting, the R/A exhaust system has not been modified, cleaned, or used since operation in the late 1960s.

Both radioactive material exhaust filter plenums located on the equipment pads east of T005 were surveyed internally for contamination. Figure 7.34 is a photo of the large walk-in filter plenum with three doors for entry. The left hand door accesses the fan impeller and the clean side of the absolute filters. The second door from the left accesses the hot side of the absolutes and cold side of the pre-filters. The third door accesses the hot side of the pre-filters and the plenum inlet. Figure 7.35 shows the relative interior size of the large filter plenum as it was surveyed in the left hand chamber. Four rows, eight columns each, of absolute filters were surveyed in the large plenum. The small plenum is not pictured; each chamber would accommodate space for one man, hunched over.

Because the plenums were highly suspect and later proven to be significantly contaminated, an 11% survey was not necessary. In each case the absolute and pre-filters were surveyed. The objective was not to determine the most accurate value of contamination per square meter, but to obtain a general idea of contamination present. Alpha and beta probes were coupled to count rate meters to determine contamination values.

The large walk-in plenum survey involved a complete suit-up (with respirator) radiological survey. The small plenum was surveyed using a probe connected to an extension pole so that the surveyor did not have to enter the confined plenum. Because the small plenum was contaminated to such a level that personnel exposure must be carefully controlled, only a few quick measurements were taken.

7.8.2 <u>Summary Table</u>

Table 7.8 shows the results for the large radioactive material filter plenum. Most contamination was embedded in the filters, thus alpha particles were not readily detectable. For the same reason, contamination was not readily removable, as seen by the low inspection test statistics for

Table 7.8 Summary of Survey Results
(Large Radioactive Material Filter Plenum)

Measurement	Number of Locations	Average Value	Maximum Value	Inspection Test Statistic	Limit
Average alpha ($dpm/100cm^2$)	Did not Perform				5000
Maximum alpha ($dpm/100cm^2$)	Did not Perform				15000
Removable alpha ($dpm/100cm^2$)	40	7.6	78	35.4	1000
Average beta ($dpm/100cm^2$)	75	10091	154000	45787	5000
Maximum beta ($dpm/100cm^2$)	75	12940	154000	47995	15000
Removable beta ($dpm/100cm^2$)	40	18.3	101	65.3	1000
Ambient Exposure Rate (uR/h)			26		5

alpha and beta removable contamination. The large value for average total beta contamination demonstrates that the large filter plenum is significantly contaminated above release limits.

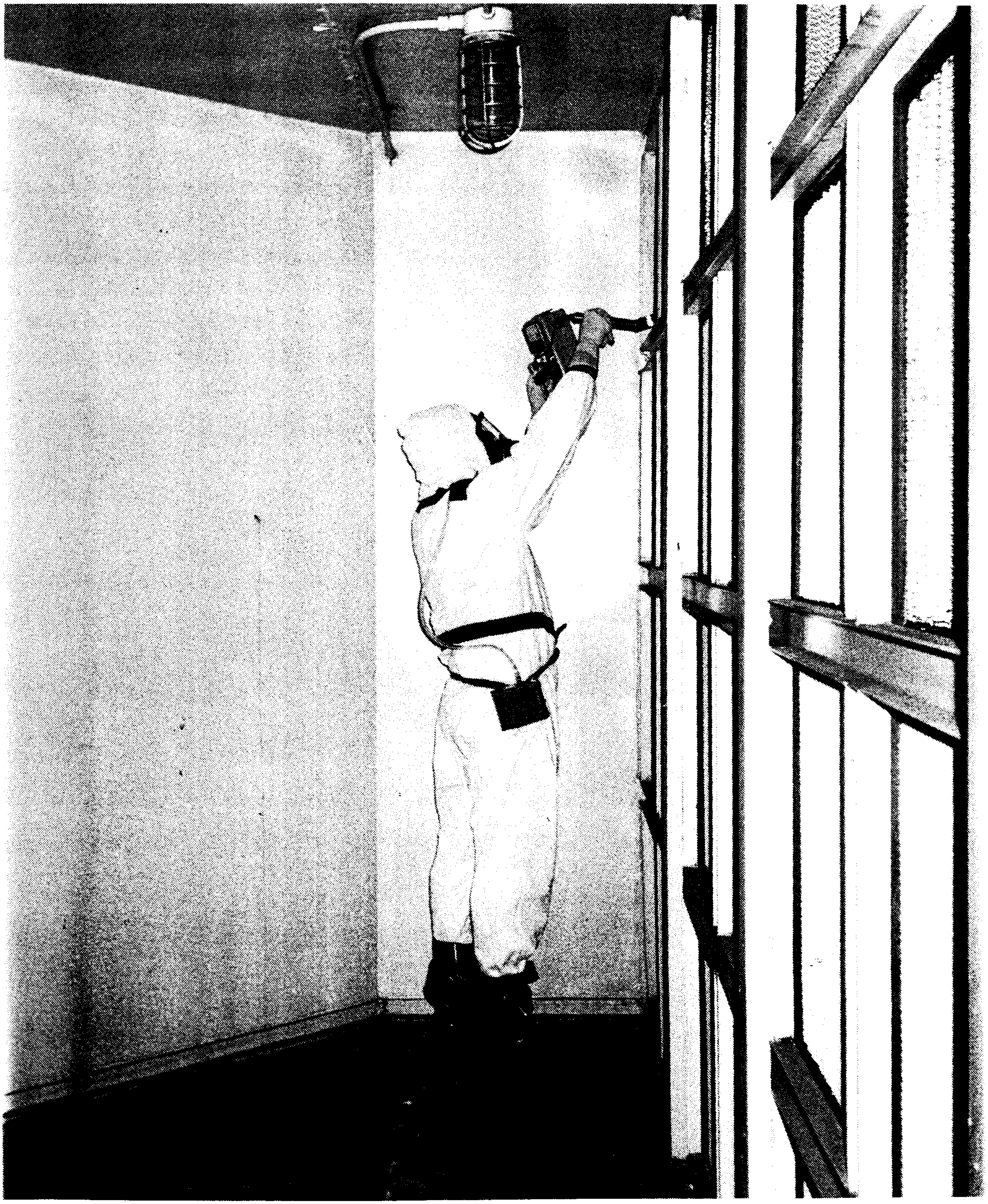
Table 7.9 shows the results for the small radioactive material filter plenum. Because of the severity of contamination, only a few measurements were taken. Count rates were converted to alpha and beta disintegration rates (alpha-dpm and beta-dpm) by multiplying the count rate by an appropriate efficiency factor for alpha and beta particles from Th-230 and Tc-99 sources, respectively.

Table 7.9 Summary of Survey Results
(Small Radioactive Material Filter Plenum)

Location	Disintegration Rate alpha-dpm	Average Value beta-dpm
Absolute Filter Bank	32,000	10,000 1,200,000 max
Gamma Exposure Rate in Absolute Plenum Chamber	120uR/h	

Figure 7.34 Photo of Large Radioactive Material
Exhaust Filter Plenum

Figure 7.35 Relative Size of Large Radioactive Material
(Exhaust Filter Plenum {Interior})



7.8.3 Cumulative Probability Distributions

Figures 7.36 through 7.39 show the statistical distributions of removable alpha, total-average beta, maximum beta, and removable beta for the large radioactive material filter plenum, respectively. Statistical analyses for alpha in the large plenum and for all parameters in the small plenum were not performed because of lack of data. The figures show that the contamination is relatively fixed to plenum surfaces and filters. The total-average and maximum beta distributions show that the data does not approximate a Gaussian distribution. This demonstrates that the area is contaminated to some degree. The test statistics reported in Table 7.7 show that the plenum is contaminated above release limits.

Although no statistical plots were generated for the small plenum, it is clear from the few measurements that it is significantly contaminated. Both plenums must be removed and disposed of appropriately before the building can be released for unrestricted use.

Figure 7.36. Removable Alpha Activity in Large Filter Plenum

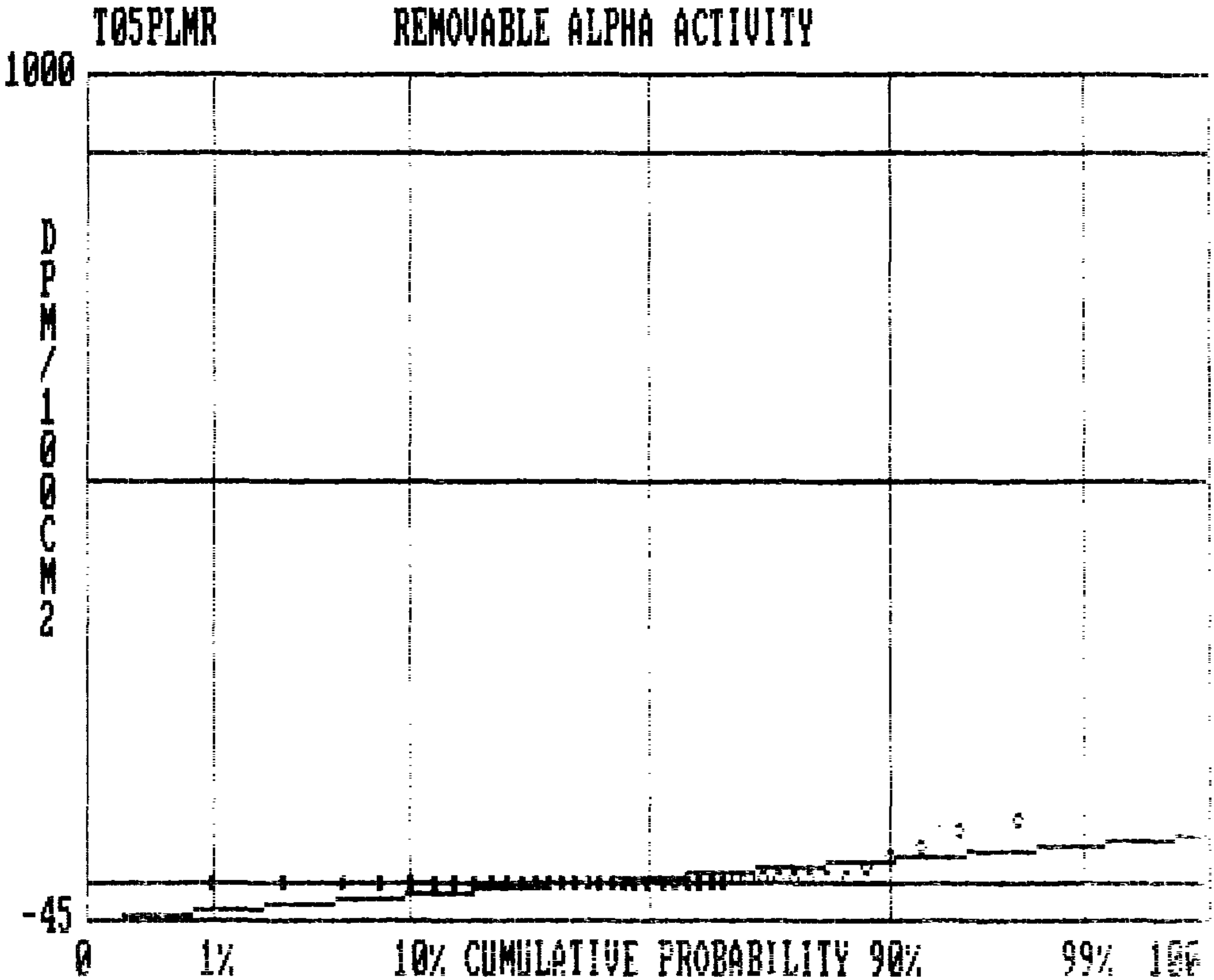


Figure 7.37. Total-average Beta Activity in Large Filter Plenum

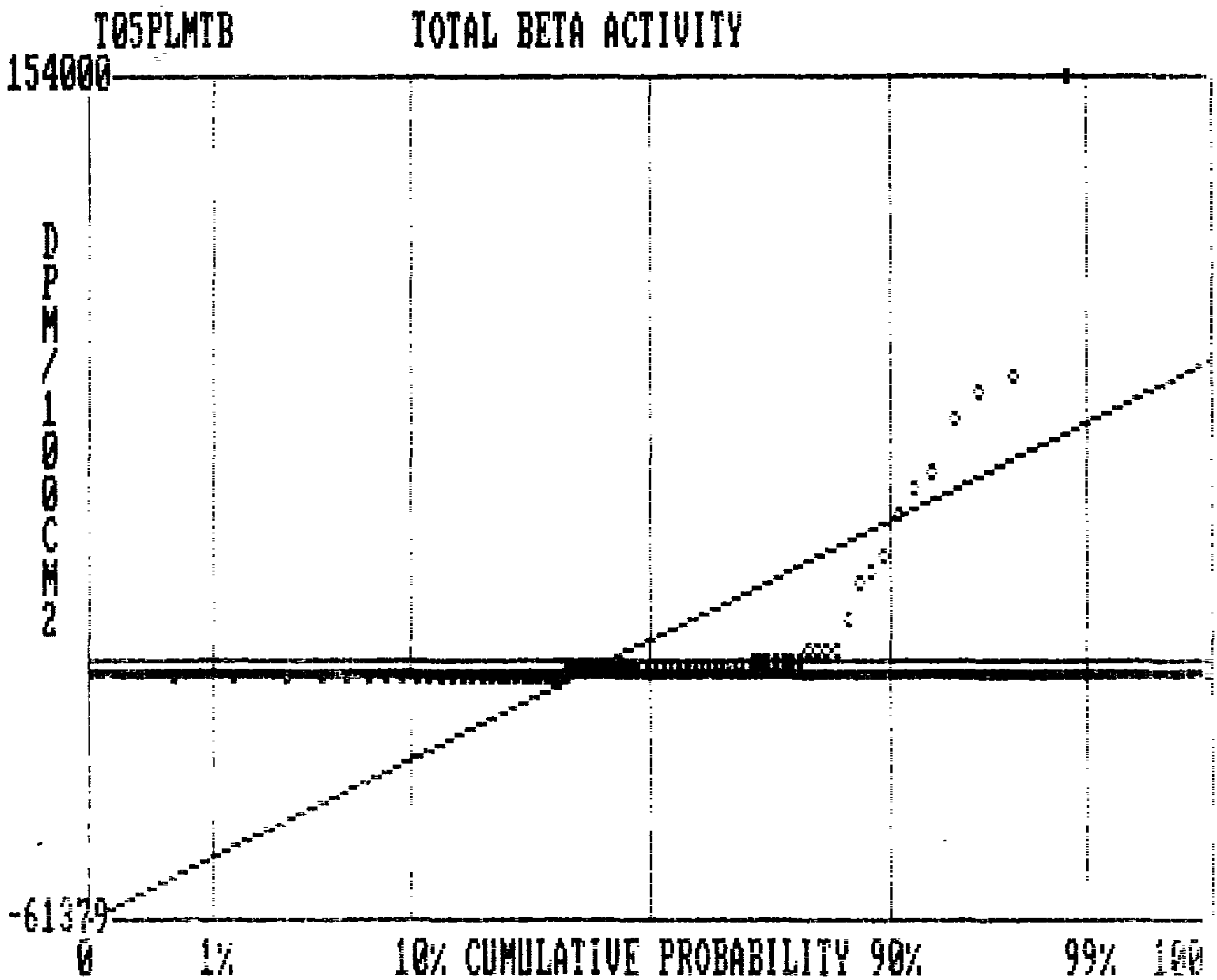


Figure 7.38. Maximum Beta Activity in Large Filter Plenum

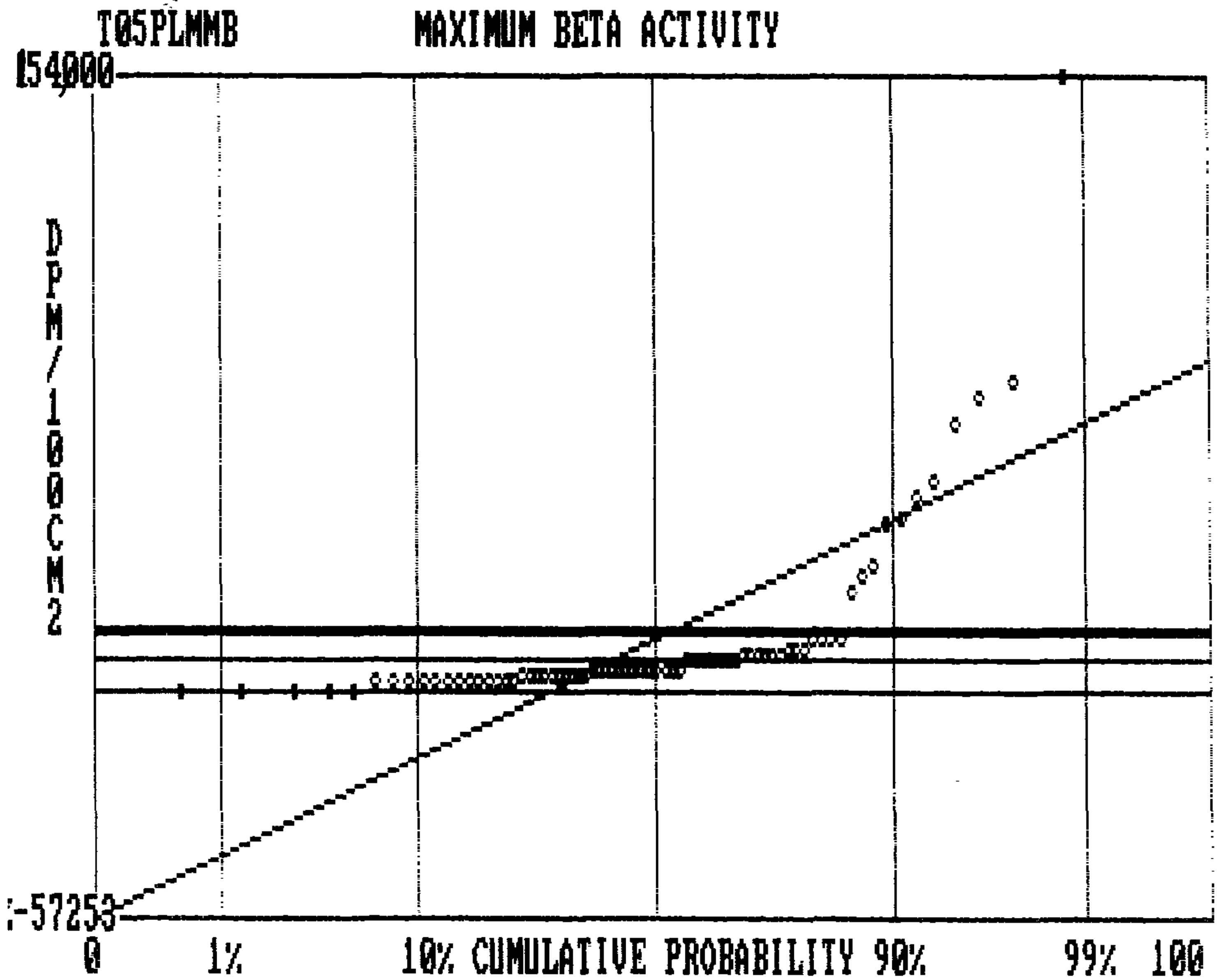
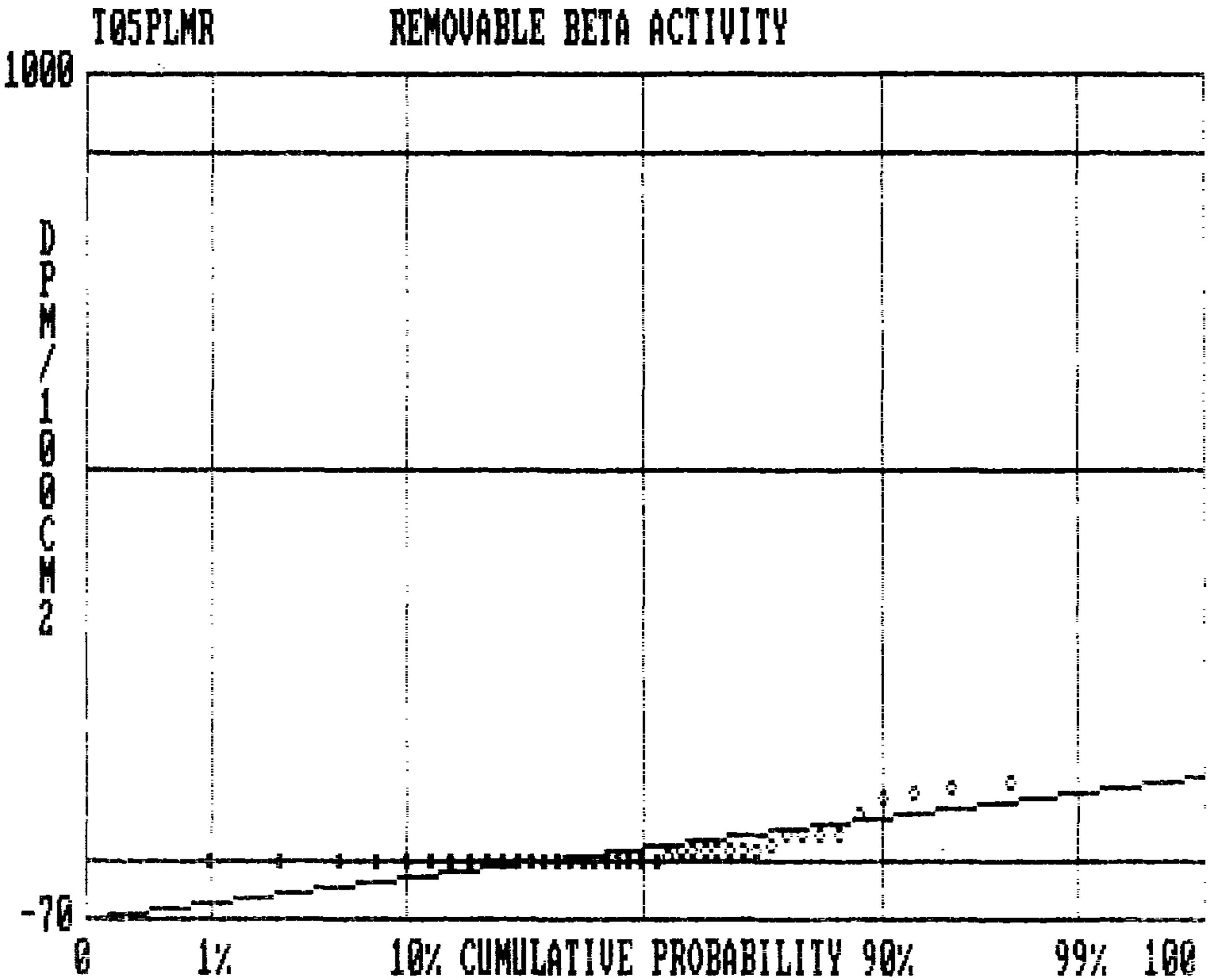


Figure 7.39. Removable Beta Activity in Large Filter Plenum



7.9 Soil and Sediment From T005 Water Course

7.9.1 General Description

Figure 7.40 shows the soil and sediment sampling locations around T005. Water drainage is from the north of T005, south. A subsurface drainage conduit with four catch basins runs down the east side of 17th Street directly in front of T005. Once the drainage conduit passes south and below G Street, water flows on the ground to the R-2A pond, located several hundred yards downstream. Figure 7.47 in section 7.11.1 shows the water drainage continuation to R-2A pond, including shallow ground-water wells. A small perimeter open drainage trough surrounds T005 from the north, around the east, and culminates on the south side. This small, cemented trough (8 in x 8 in), enclosed on the top by a grating, drains surface water from the T005 equipment yard to the south side of the yard, where it follows the natural surface drainage down 17th Street.

Soil and sediment samples were collected in both planters west of T005; the small open trough surrounding T005; the 17th Street subsurface drainage system and catch basins; and the open culvert south of G Street. In the open culvert south of G Street, 8 locations were sampled at the surface, and at 1 foot and 2 feet below the surface.

Two radiological analyses were performed on each of the 49 soil samples collected. Each sample (about 2 pounds) was dried in a furnace and then split into a 2 g sample and a 450 ml sample. The 2 gram sample was finely ground using a mortar and pestle, and then analyzed for gross alpha/beta activity; the 450 ml sample was placed in a Marinelli beaker and analyzed by gamma spectrometry.

The purpose of alpha/beta analysis using a proportional counter is to compare alpha and beta soil activity concentrations (pCi/g) against regulatory limits and against previously reported soil concentrations in

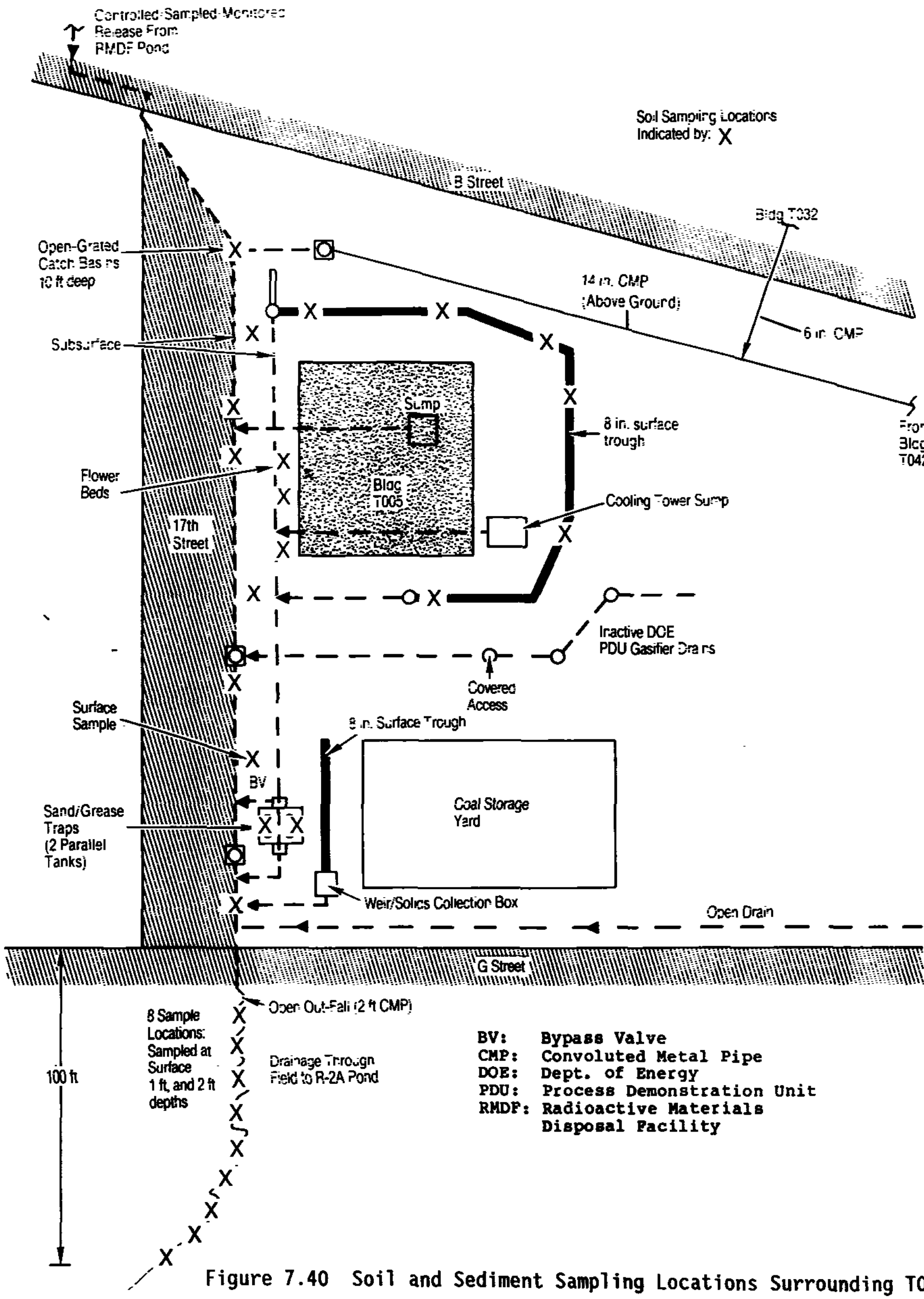


Figure 7.40 Soil and Sediment Sampling Locations Surrounding T005

Rocketdyne's Annual Environmental Monitoring report. Gamma spectrometry allows qualification and limited quantification of gamma emitting isotopes, including enriched uranium, fission products, activation products, and naturally occurring U-238, Th-232, and K-40. Gamma spectrometry is used here to identify the presence of nonsuspected gamma emitters and to show that the concentrations of U-238 and Th-232 in each soil sample compares with what would be expected from natural "uncontaminated" soil.

Results for gross alpha/beta activity are reported as alpha-pCi/g, and beta-pCi/g, respectively. Results of gamma spectrometry are reported as total pCi/g of U-238 and Th-232.

7.9.2 <u>Summary Table</u>

Table 7.10 shows the soil and sediment sample results. Analysis included gross alpha/beta activity per 2 gram sample, and gamma spectrometry per 450 ml sample contained in a Marinelli beaker. The acceptance limits applied for gross alpha/beta activity are based on contamination limits for enriched uranium, (Ref. 13).

The analytical results for gross alpha/beta activity can be compared to two benchmarks: 1) the NRC recommended acceptance limit (Ref. 13) which is 30 alpha pCi/g above natural "uncontaminated" soil, and 100 beta pCi/g total; and 2) the soil radioactivity data reported in the 1986 Facility Effluent Annual Report, (Ref. 15). The inspection test statistic in each case is less than the acceptance limit. Results for alpha/beta radioactivity reported in Reference 15 are 26.7 alpha pCi/g and 26.1 beta pCi/g. Results reported here for alpha contamination are much less than the values given in Reference 15, but this variation is acceptable and should be expected under the conditions of the analysis. Beta activity results compare quite favorably.

The difference between alpha activity concentration reported here and in the annual report is attributed to differences in soil sample preparation methods. Each sample prepared for the annual report is placed in a ball mill, which crushes the sample into very fine powder. Because these T005 samples were crushed in a mortar and pestle, some comparatively large chunks of soil existed in each sample. The enriched uranium standard for calculating an alpha absorption correction factor in sample soil was also crushed into very fine powder with a ball mill. Because alpha particle penetration through a sample media is strongly dependent on media composition and geometry, variation is expected between the results reported in the annual report and this current analysis. A fewer number of alpha particles would be expected to penetrate through the larger chunks of soil in this analysis than for the source standard or the annual report analysis. A correction factor for this absorption is not available; thus, the alpha concentration results are comparatively smaller.

The results reported from gamma spectrometry analysis show that U-238 and Th-232 activity concentrations represent what would be expected from natural "uncontaminated" soil in Southern California. Seventeen of the 49 samples tested positive for Cs-137, all concentrations less than 1 pCi/g. About four samples were identified as containing Eu-152, an activation product, in quantities no greater than 6 pCi/g. No trends or problem areas were identified.

Table 7.10 Summary of Survey Results
(Soil Samples from T005 Water Course)

Measurement	Number of Locations	Average Value	Maximum Value	Inspection Test Statistic	Limit
Gross Alpha Activity Concentration (pCi/g)	49*	10.3	40.5	31.8	50
Gross Beta Activity Concentration (pCi/g)	49*	29.2	38.1	37.8	100
U-238 Activity Concentration (pCi/g)	49*	0.97	2.5	2	N/A
Th-232 Activity Concentration (pCi/g)	49*	1.04	4.0	1.8	N/A

* Includes 5 background soil samples collected at various locations in SSFL.

** Note: The activity concentration ratio of U-238: Th-232 in natural "uncontaminated" soil is 1. By comparison, the ratio of U-238:Th-232 in these 49 samples is 0.97:1.043 = 0.93.

7.9.3 Cumulative Probability Distributions

Figures 7.41 and 7.42 show the statistical distributions of alpha and beta soil radioactivity concentrations, respectively. Each figure represents a model Gaussian Cumulative Distribution Function, with no outliers or trends indicating that the soil is contaminated. The inspection test statistic ($\bar{x}$ + ks) for each distribution is less than the NRC imposed acceptance limit. These distributions were generated from alpha/beta soil analyses using a gas-proportional detector. Correction factors were derived for alpha particle shielding within the sample matrix by use of an enriched uranium soil standard, traceable to NBS. The alpha concentration reported here is less than values reported previously because of unavailable corrections for alpha absorption in soil chunks. Beta concentrations are in agreement with previously reported results.

As mentioned in the previous section, only trace amounts of Cs-137 and Eu-152 were identified in a few samples (and not the same samples) by gamma spectrometry. The calculated error in most cases was quite high. The only identifiable and reproducible gamma peaks throughout the samples collected were those attributable to naturally-occurring U-238 and Th-232 decay chains, and K-40. Subsequently, calculated activity concentrations were used as a comparison against published ratios of U-238 to Th-232 activities.

Figure 7.43 shows a scatter plot of each sample, U-238 concentration against Th-232 concentration. A bilateral fit of the data is also generated. The samples are quite consistent and the ratios are very near what would be expected from natural "uncontaminated" soil. One sample had a typical U-238 concentration value of 1.2 pCi/g but an exceptionally high Th-232 value of 4 pCi/g. The value was treated as an outlier and excluded from further analysis - the point does not appear on the plot.

Figure 7.44 and 7.45 are Gaussian cdfs of U-238 and Th-232 concentrations (pCi/g) respectively. The ratios of U-238 to Th-232 were reported earlier as being very close to 1; this would be expected in natural "uncontaminated" soil. U-235 was not identified in any of the samples.

The U-238 distribution deviates from a model Gaussian cumulative probability plot, which shows the variability in naturally occurring U-238 concentration.

The Th-232 distribution follows a Gaussian distribution with no outliers. Evidently, Th-232 concentration in soil at SSFL is more consistent than U-238 concentration.

Figure 7.41 Alpha Radioactivity Concentration in Soil (pCi/g)

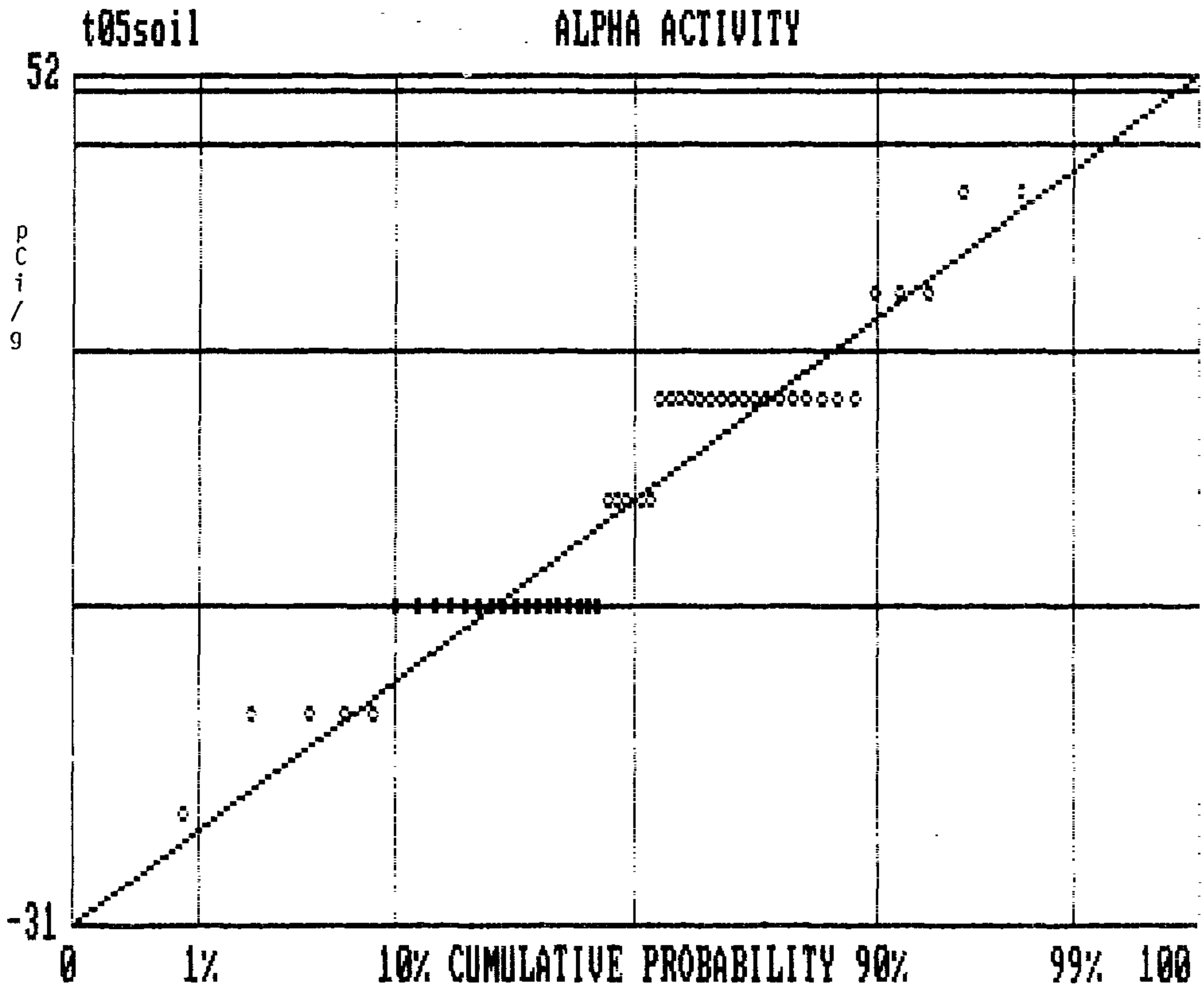


Figure 7.42. Beta Radioactivity Concentration in Soil (pCi/g)

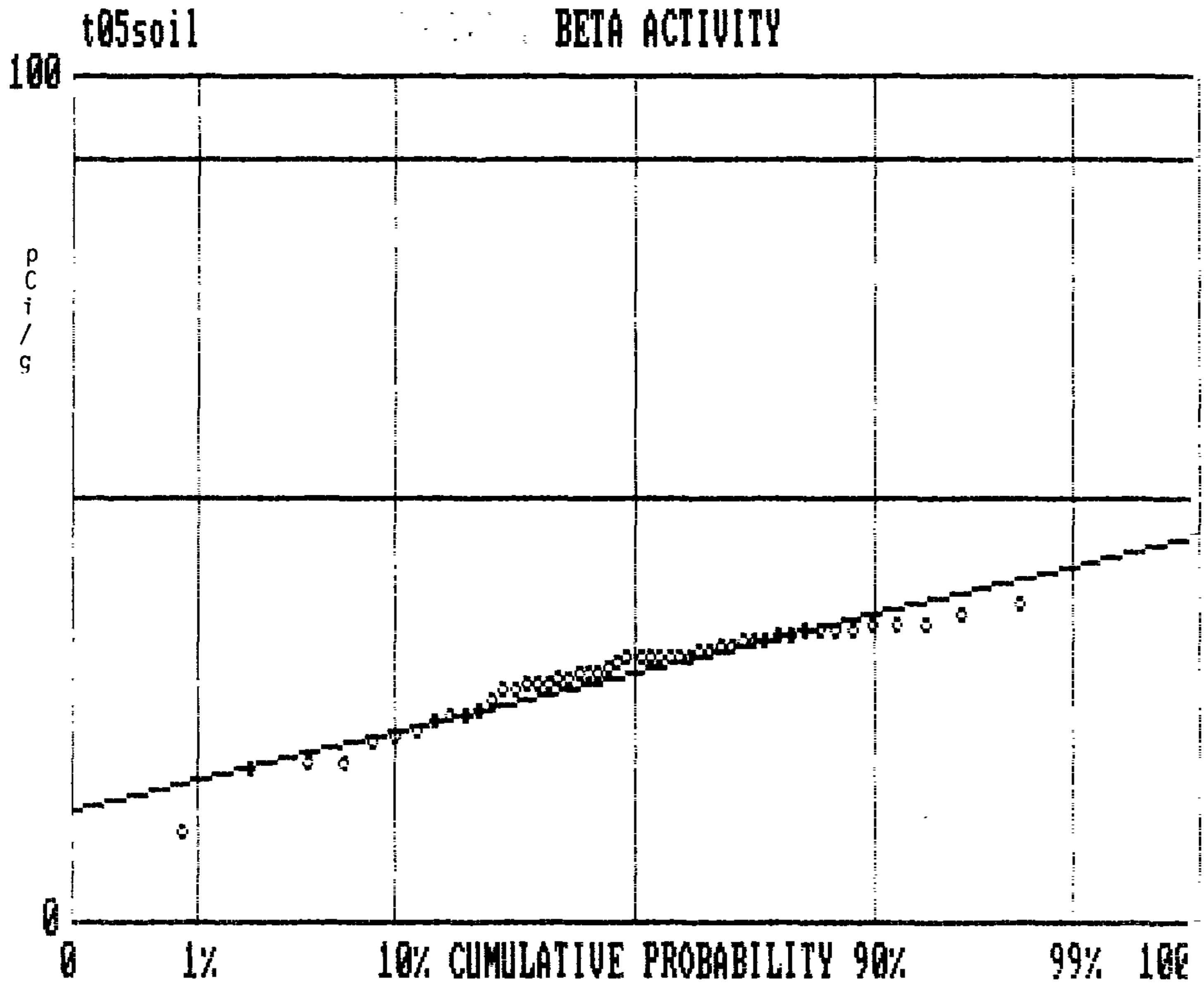


Figure 7.43 Scatter Plot of U-238 vs Th-232 Activity Concentration in Soil Samples Collected Near T005

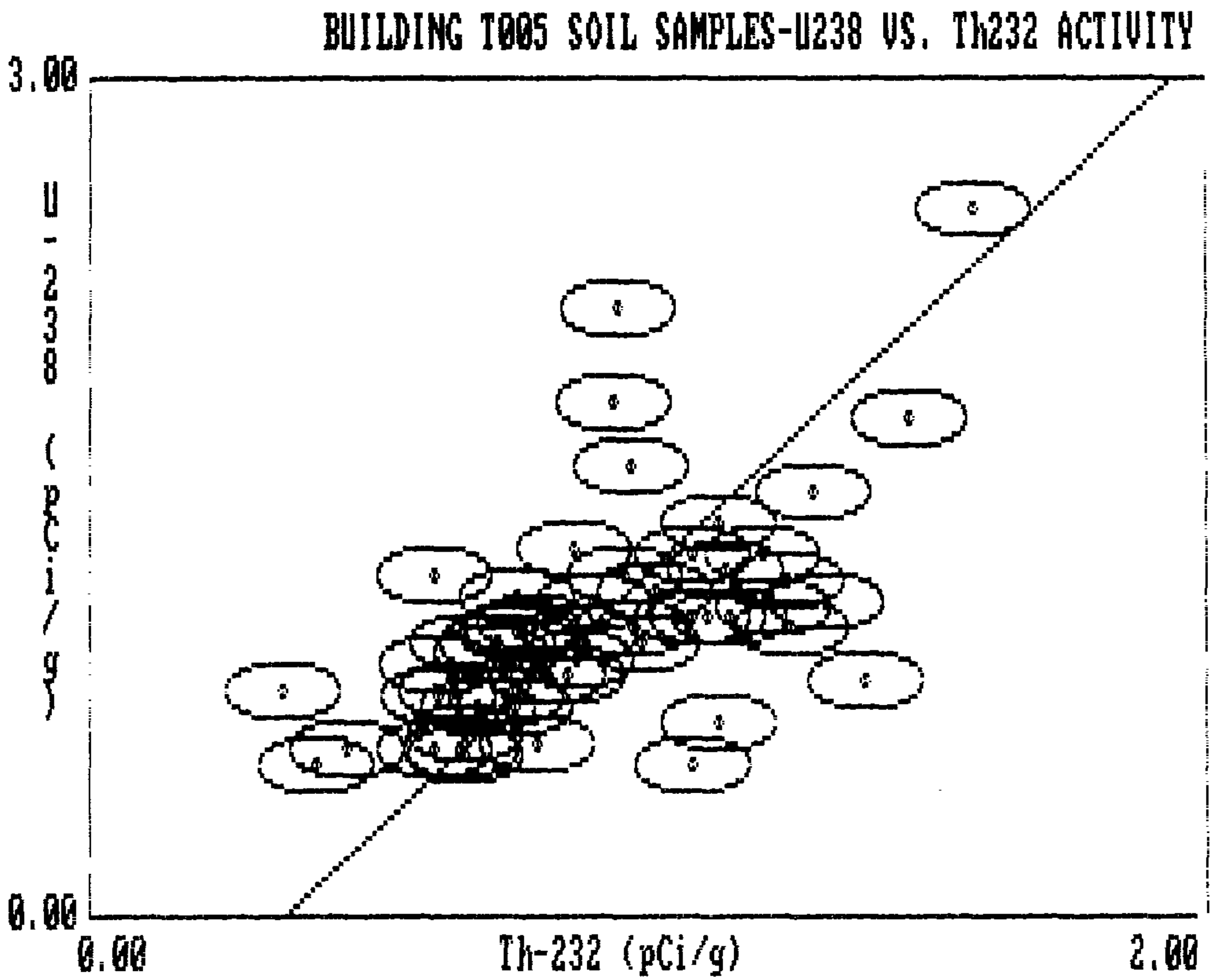


Figure 7.44 U-238 Activity Concentration in T005 Soil Samples

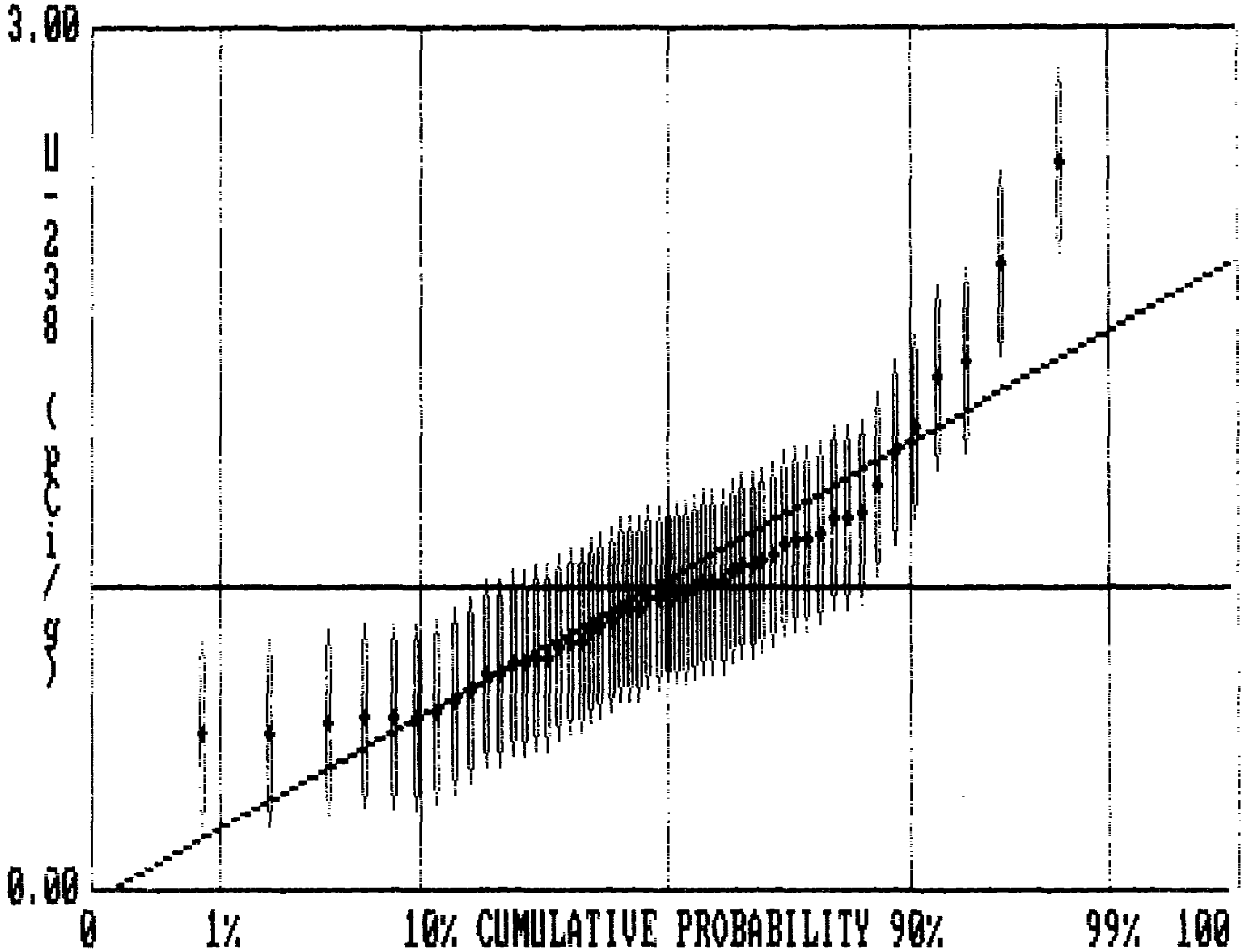
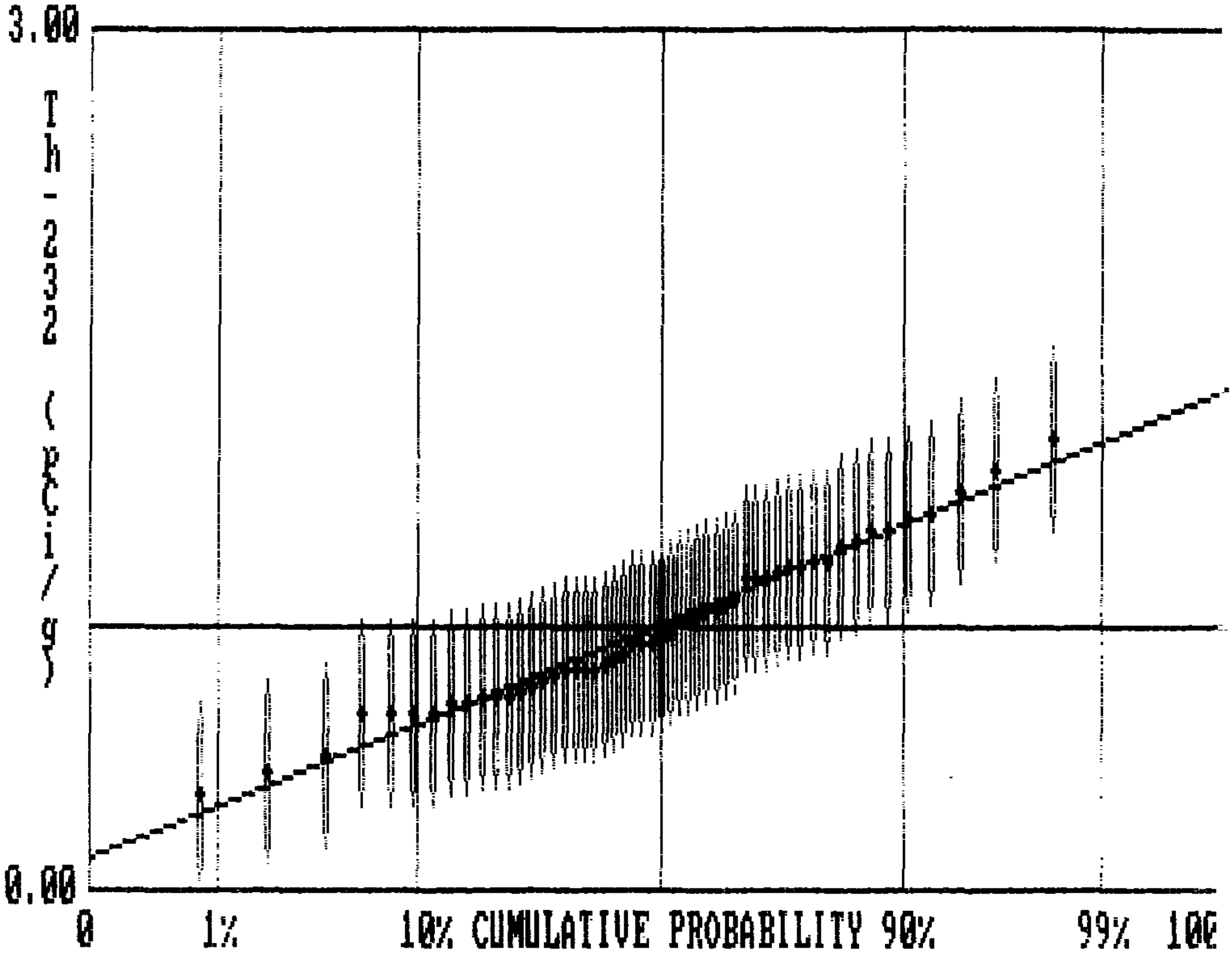


Figure 7.45 Th-232 Activity Concentration in T005 Soil

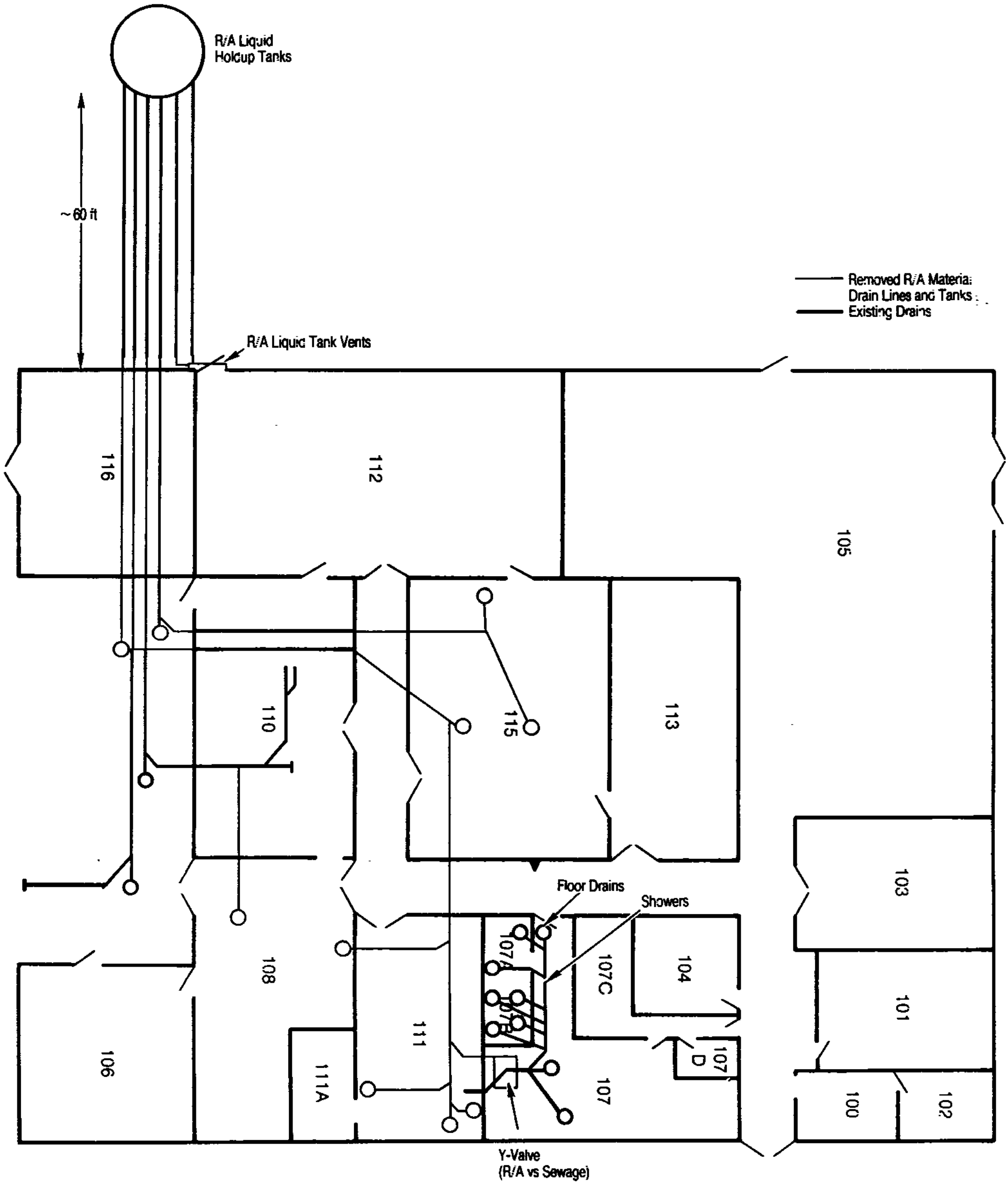


7.10 Radioactive Material Drain Lines and Holdup Tank

Figure 7.46 shows the radioactive liquid drain system and holdup tanks, as installed and used during facility operation. The holdup tanks were removed several years ago. Some drain-line sections have been removed in recent months as part of the T005 clean-up effort. During the drain-line removal process, samples of scabbled concrete and soil beneath the drain lines were surveyed for gross alpha/beta contamination and analyzed by gamma spectrometry. Over 200 soil and concrete samples were analyzed. In all cases, no radioactive materials in concentrations above background were identified.

The floor drains and traps leading from the men's change room were thoroughly smeared and analyzed for radioactivity. Sludge samples were scraped from all available cleanouts and analyzed by gamma spectrometry. No detectable activity was indicated in all cases. The drains and pipes in the men's change room up to the gate valve at the intersection of rooms 107 and 111 do not have to be removed. The removal of drain pipes located in rooms 110 and 110A, and from the northeast corner of T005 out to where the holdup tank once existed is planned. This should be performed before the building is released for unrestricted use.

Figure 7.46 Radioactive Liquid Drain System As Used During Facility Operation



7.11 Nearby SSFL Ground-Water Wells

7.11.1 General Description

Figure 7.47 shows the water gutter from T005 to the final holdup pond (R-2A), and the shallow ground water wells in the surrounding area. Groundwater Resources Consultants, Inc. obtains water samples from the SSFL wells on a periodic basis. A split sample is given to the Rocketdyne Radiation & Nuclear Safety Department for radioactivity analysis. The shallow ground water wells applicable to T005 runoff are RS-15, RS-9, and RS-11.

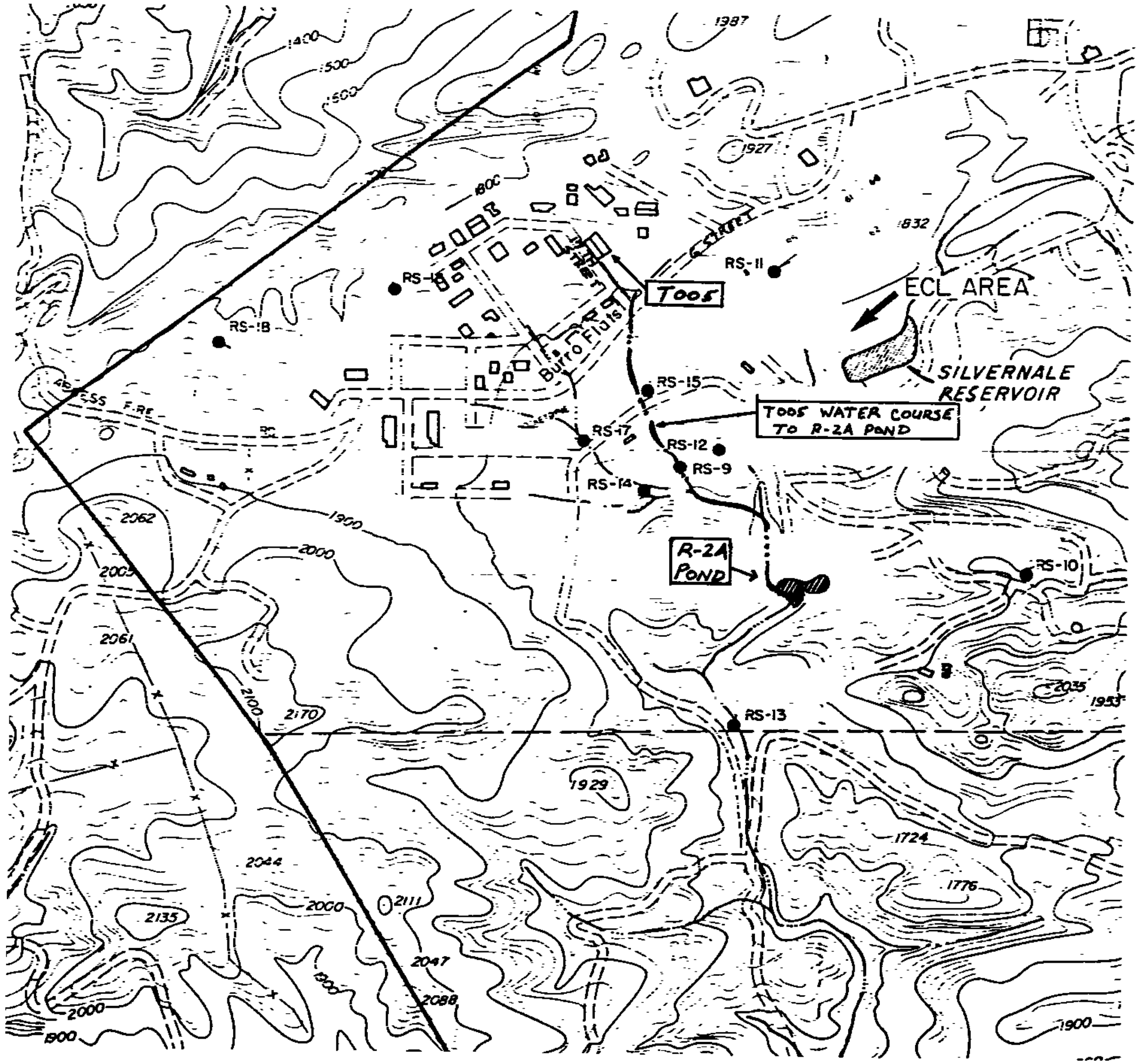
7.11.2 Summary Table

Table 7.11 shows the 1987 well water sample results (in pCi/l) for shallow wells RS-15, RS-9, and RS-11. Only two samples have been taken from RS-9 this year. One sample each has been taken from wells RS-15 and RS-11.

Table 7.11 Summary of Survey Results
(T005 Local Area Shallow Wells)

Well Number	Month (1987)	Activity Concentration (pCi/l) alpha	beta
RS-9	June	0.300	2.276
	September	0.582	1.877
RS-11	September	0.328	0.93
RS-15	September	0.303	3.707

Figure 7.47 T005 Water Drainage Showing Shallow Wells



8.0 CONCLUSIONS

A comprehensive radiological survey was performed at Building T005 to identify areas and equipment requiring further radiological inspection and/or remedial action. Measured values of residual radioactive contamination show that four areas are contaminated at levels above release limits adopted by DOE: 1) room 113; 2) room 110E; 3) remaining R/A exhaust ducts (four total); and 4) both R/A exhaust filter plenums. Analysis of the data according to the sampling plan shows, in all other areas, that the Inspection Test Statistic ($\bar{x}$ +ks) is below acceptance limits. This method of analysis shows that any other similar set of radiological measurements should be found acceptable also, and further, that all locations within those areas determined to be "uncontaminated" have residual radioactivity below the limits. Analyses show that no unacceptable releases to the environment via the liquid effluent pathway occurred during facility operation.

With the exception of the four "contaminated" areas mentioned above, the results of this survey show essentially no residual contamination and demonstrate a negligible risk of there being any undetected contamination exceeding acceptance limits. Those "contaminated" items must be decontaminated, or removed and disposed of as radioactive waste before the facility can be released for unrestricted use. It is recommended that a decontamination plan be prepared under SFMP, in order that the facility can be released for unrestricted use.

9.0 REFERENCES

1. "Guidelines for Residual Radioactivity at FUSRAP and Remote SFMP Sites," U.S. DOE, March 5, 1985.

2. "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material," Annex B, USNRC License SNM-21, Docket 70- 25, Issued to Energy Systems Group of Rockwell International, last revision June 5, 1984.

3. "State of California Guidelines for Decontaminating Facilities and Equipment Prior to Release for Unrestricted Use," DECON-1, Revised March 24, 1983.

4. "Radiological Survey Plan for SSFL," 154SRR000001, F. H. Badger and R. J. Tuttle, Rockwell International, September 25, 1985.

5. "Long Range Plan for Decommissioning Surplus Facilities at the Santa Susana Field Laboratories," N001TI0000200, W.D. Kittinger, Rockwell International, September 30, 1983.

6. "Nuclear Safety Analysis," number 201, J.H. Walter, Atomics International, May 17, 1967.

7. "Final Radiation Survey of the NMDF," N704SRR990027, J. A. Chapman, Rockwell International, December 19, 1986.

8. "Draft American National Standard Control of Radioactive Surface Contamination on Materials, Equipment, and Facilities to be Released for Uncontrolled Use," ANSI N13.12, August 1978, American National Standards Institute, Inc.

9. "Selected Techniques of Statistical Analysis," Statistical Research Group, Columbia University, McGraw-Hill Book Co., Inc., 1947.

10. "Some Theory of Sampling," W. E. Deming, Dover Publications, Inc., New York, 1950.

11. "Statistics in Research," B. Ostle and R. Mensing, The Iowa State University Press, 1979.

12. "Measurement and Detection of Radiation," N. Tsoulfanidis, Hemisphere Publishing Corp., Washington D.C., 1983.

13. "Disposal or Onsite Storage of Thorium or Uranium Wastes from Past Operations," Federal Register Vol. 46, No. 205, October 31, 1981.

14. "Standards for Protection Against Radiation," Title 10 Part 20, Code of Federal Regulations, January 1, 1985.

15. "Rocketdyne Division Environmental Monitoring and Facility Effluent Annual report Desoto and Santa Susana Field Laboratories Sites 1986," RI/RD87-133, J. D. Moore, Rockwell International, March 1987.

APPENDIX A. DESCRIPTION OF NUCLEAR INSTRUMENTATION

Smear test wipes from structural surfaces, and soil samples collected from sub-floor and other areas during the radiological survey of Building 005 were analyzed for radioactivity level and in some cases for specific radionuclide identification by one or more of the following nuclear instrumentation systems.

A.1 Gamma Spectrometry Analyzer

Gamma spectrometry of selected samples including the soil samples was performed with a Canberra Industries, Inc. Series 80 Multichannel Analyzer (MCA). The MCA is coupled to a high planar purity germanium (HPGe) radiation detector having about a 10% relative sensitivity (relative to the sensitivity of a 3" x 3" NaI detector for cesium-137 gamma radiation), and a photopeak resolution capability of about 2.5 keV (FWHM) for the higher energy line of cobalt-60. The Series 80 MCA used for soil analyses has a 8192 channel memory capacity with a 1E+06 counts per channel capacity. Functional operation options include integral, net area, strip, and energy calibration, all used for spectrum analysis. The Series 80 was calibrated both for gamma energy and for nuclide quantification with a Marinelli Beaker Standard Source (MBSS) as specified in document ANSI/IEEE Std 680-1978, "IEEE Standard Techniques for Determination of Germanium Semiconductor Detector Gamma-Ray Efficiency Using a Standard Marinelli (Reentrant) Beaker Geometry." All soil samples analyzed by gamma spectroscopy were presented to the detector with the same geometric configuration as the MBSS.

A.2 Gross Alpha/Beta Automatic Proportional Counter

Smear wipe test samples were analyzed for gross alpha and gross beta radioactivity with a Canberra Industries Model 2201 Ultra Low Level Counting System. Model 2201 consists of a highly efficient gas flow sample detector operating in the proportional gas amplification region and a cosmic detector for coincidence event cancellation. The two detectors operate in an anticoincidence mode to reduce the count rate due to cosmic events. When cosmic or background events occur, the input circuit to the count integrator is gated off and the simultaneous event is discarded. Thus, only true alpha and/or beta radiation events are recorded. The detectors are coupled through dual Model 2006A preamplifiers to a Model 2015A system amplifier then through a Model 2209A coincidence analyzer to the alpha or beta event scaling unit. The Series 2201 has a sample capacity of 99 samples contained in a magazine designed to accept sample planchets having a 2-inch diameter. Calibration of the sample detector for alpha and for beta radiation is done with NBS traceable certified thorium-230 (alpha) and technicium-99 (beta) radiation sources having a configuration essentially equivalent to that of the smear wipes.

A.3 Portable Instruments

Ludlum model 2220 portable scalar/ratemeters coupled to alpha, beta, and gamma probes were used during the course of the survey. The 2220 has a six decade LCD readout; combination four decade linear and log rate meter; adjustable HV threshold, and window positions, with readouts on digital display; audio provided by unimorph speaker with pitch change in relation to count rate; and preset electronic timer. Three 2200s were connected to separate probes, alpha, beta, gamma.

A Ludlum model 43-1 alpha scintillation detector was coupled to one 2220 for alpha contamination measurements. The scintillator is ZnS(Ag). The window ($0.8 mg/cm^2$) is aluminized mylar with an active area of about $75 cm^2$. Background for this probe is less than 2 counts per 5 minutes. Efficiency for Pu-239 or Th-230 alpha particles is between 25% and 30%.

A Ludlum model 44-9 pancake Geiger-Mueller detector was coupled to another 2220 for beta contamination measurements. The window ($1.7 mg/cm^2$) is mica with an nominal active area of $20 cm^2$. Background for this probe is about 80 to 100cpm. Efficiency for Tc-99 beta particles is between 25% and 20%.

A Ludlum model 44-10 NaI gamma scintillator was coupled to the third 2220 for ambient gamma measurements. The NaI(Tl) crystal is extremely sensitive to changes in gamma flux. The efficiency of the probe coupled to the 2220 for Cs-137 gamma rays is about 150cpm/uR/h.

APPENDIX B. COPY OF DOE REPORT, "GUIDELINES FOR RESIDUAL RADIOACTIVITY AT FUSRAP AND REMOTE SFMP SITES," March, 1985

Department of Energy
Richland Operations Office
P.O. Box 550
Richland, Washington 99352

MAR 0 5 1985

CORRESPONDENCE

Addressees

GUIDELINES FOR RESIDUAL RADIOACTIVITY AT FUSRAP AND REMOTE SFMP SITES

The attached guidelines, "U.S. Department of Energy Guidelines for Residual Radioactivity at Formerly Utilized Sites Remedial Action Program and Remote Surplus Facilities Management Program Sites," (January 1985) have been issued by the Division of Remedial Action Projects for implementation by FUSRAP and SFMP in order to establish authorized limits for remedial actions. While these Guidelines are specifically intended for "remote" SFMP sites (those located outside a major DOE R&D or production site), they should be taken into consideration when developing authorized limits for remedial actions on major DOE reservations. The guidelines provide specific authorized limits for residual radium and thorium radioisotopes in soil, for airborne radon decay products, for external gamma radiation, and for residual surface contamination levels on materials to be released for unrestricted use. These guidelines will be supplemented in the near future by a document providing the methodology and guidance to establish authorized limits for residual radioisotopes other than radium and thorium in soil at sites to be certified for unrestricted use. The supplement will provide further guidance on the philosophies, scenarios, and pathways to derive appropriate authorized limits for residual radionuclides and mixtures in soil. These guidelines are based on the International Commission on Radiation Protection (ICRP) philosophies and dose limits in ICRP reports 26 and 30 as interpreted in the draft revised DOE Order 5480.1A. These dose limits are 500 mrem/yr for an individual member of the public over a short period of time and an average of 100 mrem/yr over a lifetime.

The approval of authorized limits differing from the guidelines is described in Section D, last sentence of the attached document. If the urgency of field activity makes DRAP concurrence not cost effective, a copy of the approval and backup analysis should be furnished to DRAP as soon as possible, although not necessarily prior to beginning field activities. This does not remove the requirement for approval by SFMPO.

As a result of a recent court decision, the Environmental Protection Agency (EPA) has issued <u>airborne</u> radiation standards applicable to DOE facilities. These final standards, issued as revisions to 40 CFR 61, are:

02067RL

2

- 25 mrem/yr-whole body
- 75 mrem/yr-organ
- waiver of these standards will be granted if DOE demonstrates that no individual would receive 100 mrem/yr continuous exposure whole body dose equivalent from all sources within 10 km radius, excluding natural background and medical procedures
- radon and radon daughters are excluded (these standards are covered in 40 CFR 192)

The attached guidelines were written to be consistent with the revision of the DOE Order 5480.1A now in draft at Headquarters and have received the concurrence of the Public Safety Division, Office of Operational Safety. The guidelines will be included in the SFMP Program Plan beginning with the next revision (for FY 1986-1990).

Please refer any questions to Paul F. X. Dunigan, Jr. (FTS 444-6667), of my staff.

Clarence E. Miller, Jr.

Clarence E. Miller, Jr., Director
Surplus Facilities Management
Program Office

SFMPO:PFXD

Attachment:
As stated

cc: R. N. Coy, UNC
E. G. DeLaney, NE-24, HQ

3

Addressees - Memorandum dated MAR 05 1965

K. J. Peterson, AL
R. J. Grandfield, DAO
F. F. Gorup, CH
E. K. Hunter, ID
M. K. Tucker, GJPO
L. F. Campbell, OR
D. R. Brown, OR
J. H. Slaughter, RL
S. L. Samuelson, SAN
M. G. O'Rear, SR
J. J. Schreiber, RL-SSDPO
R. Garde, LANL
W. P. Davis, Mound
W. H. Kline, ANL
R. H. Meservey, EG&G-ID
I. N. Abramiuk, BPEC
G. Coxon, BNI
T. E. Myrick, ORNL
C. C. Conners, SSFL
K. S. Kotti, SRL

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U.S. DEPARTMENT OF ENERGY GUIDELINES FOR RESIDUAL RADIOACTIVITY AT FORMERLY UTILIZED SITES REMEDIAL ACTION PROGRAM AND REMOTE SURPLUS FACILITIES MANAGEMENT PROGRAM SITES

(February 1985)

A. INTRODUCTION

This document presents U.S. Department of Energy (DOE) radiological protection guidelines for cleanup of residual radioactive materials and management of the resulting wastes and residues. It is applicable to sites identified by the Formerly Utilized Sites Remedial Action Program (FUSRAP) and remote sites identified by the Surplus Facilities Management Program (SFMP).* The topics covered are basic dose limits, guidelines and authorized limits for allowable levels of residual radioactivity, and requirements for control of the radioactive wastes and residues.

Protocols for identification, characterization, and designation of FUSRAP sites for remedial action; for implementation of the remedial action; and for certification of a FUSRAP site for release for unrestricted use are given in a separate document (U.S. Dept. Energy 1984). More detailed information on applications of the guidelines presented herein, including procedures for deriving site-specific guidelines for allowable levels of residual radioactivity from basic dose limits, is contained in a supplementary document--referred to herein as the "supplement" (U.S. Dept. Energy 1985).

"Residual radioactivity" includes: (1) residual concentrations of radionuclides in soil material,** (2) concentrations of airborne radon decay products, (3) external gamma radiation level, and (4) surface contamination. A "basic dose limit" is a prescribed standard from which limits for quantities that can be monitored and controlled are derived; it is specified in terms of the effective dose equivalent as defined by the International Commission on Radiological Protection (ICRP 1977, 1978). Basic dose limits are used explicitly for deriving guidelines for residual concentrations of radionuclides in soil material, except for thorium and radium. Guidelines for

*A remote SFMP site is one that is excess to DOE programmatic needs and is located outside a major operating DOE research and development or production area.

**The term "soil material" refers to all material below grade level after remedial action is completed.

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residual concentrations of thorium and radium and for the other three quantities (airborne radon decay products, external gamma radiation level, and surface contamination) are based on existing radiological protection standards (U.S. Environ. Prot. Agency 1983; U.S. Nucl. Reg. Comm. 1982). These standards are assumed to be consistent with basic dose limits within the uncertainty of derivations of levels of residual radioactivity from basic limits.

A "guideline" for residual radioactivity is a level of residual radioactivity that is acceptable if the use of the site is to be unrestricted. Guidelines for residual radioactivity presented herein are of two kinds: (1) generic, site-independent guidelines taken from existing radiation protection standards, and (2) site-specific guidelines derived from basic dose limits using site-specific models and data. Generic guideline values are presented in this document. Procedures and data for deriving site-specific guideline values are given in the supplement.

An "authorized limit" is a level of residual radioactivity that must not be exceeded if the remedial action is to be considered completed. Under normal circumstances, expected to occur at most sites, authorized limits are set equal to guideline values for residual radioactivity that are acceptable if use of the site is not be restricted. If the authorized limit is set higher than the guideline, restrictions and controls must be established for use of the site. Exceptional circumstances for which authorized limits might differ from guideline values are specified in Sections D and F. The restrictions and controls that must be placed on the site if authorized limits are set higher than guidelines are described in Section E.

DOE policy requires that all exposures to radiation be limited to levels that are as low as reasonably achievable (ALARA). Implementation of ALARA policy is specified as procedures to be applied after authorized limits have been set. For sites to be released for unrestricted use, the intent is to reduce residual radioactivity to levels that are as far below authorized limits as reasonable considering technical, economic, and social factors. At sites where the residual radioactivity is not reduced to levels that permit release for unrestricted use, ALARA policy is implemented by establishing controls to reduce exposure to ALARA levels. Procedures for implementing ALARA policy are described in the supplement. ALARA policies, procedures, and actions must be documented and filed as a permanent record upon completion of remedial action at a site.

B. BASIC DOSE LIMITS

The basic limit for the annual radiation dose received by an individual member of the general public is 500 mrem/yr for a period of exposure not to exceed 5 years and an average of 100 mrem/yr over a lifetime. The committed effective dose equivalent, as defined in ICRP Publication 26 (ICRP 1977) and calculated by dosimetry models described in ICRP Publication 30 (ICRP 1978), shall be used for determining the dose.

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C. GUIDELINES FOR RESIDUAL RADIOACTIVITY

C.1 Residual Radionuclides in Soil Material

Residual concentrations of radionuclides in soil material shall be specified as above-background concentrations averaged over an area of 100 m^2. If the concentration in any area is found to exceed the average by a factor greater than 3, guidelines for local concentrations shall also be applicable. These "hot spot" guidelines depend on the extent of the elevated local concentrations and are given in the supplement.

The generic guidelines specified below are for concentrations of individual radionuclides occurring alone. If mixtures of radionuclides are present, the concentrations of individual radionuclides shall be reduced so that the dose for the mixture would not exceed the basic dose limit. Explicit formulas for calculating residual concentration guidelines for mixtures are given in the supplement.

The generic guidelines for residual concentrations of Th-232, Th-230, Ra-228, and Ra-226 are:

- 5 pCi/g, averaged over the first 15 cm of soil below the surface

- 15 pCi/g, averaged over 15-cm-thick layers of soil more than 15 cm below the surface

The guidelines for residual concentrations in soil material of all other radionuclides shall be derived from basic dose limits by means of an environmental pathway analysis using site-specific data. Procedures for deriving these guidelines are given in the supplement.

C.2 Airborne Radon Decay Products

Generic guidelines for concentrations of airborne radon decay products shall apply to existing occupied or habitable structures on private property that are intended for unrestricted use; structures that will be demolished or buried are excluded. The applicable generic guideline (40 CFR 192) is: In any occupied or habitable building, the objective of remedial action shall be, and reasonable effort shall be made to achieve, an annual average (or equivalent) radon decay product concentration (including background) not to exceed 0.02 WL.* In any case, the radon decay product concentration (including background) shall not exceed 0.03 WL. Remedial actions are not required in order to comply with this guideline when there is reasonable assurance that residual radioactive materials are not the cause.

C.3. External Gamma Radiation

The level of gamma radiation at any location on a site to be released for unrestricted use, whether inside an occupied building or habitable structure or outdoors, shall not exceed the background level by more than 20 μR/h.

*A working level (WL) is any combination of short-lived radon decay products in one liter of air that will result in the ultimate emission of 1.3×10^5 MeV of potential alpha energy.

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C.4 Surface Contamination

The following generic guidelines, adapted from standards of the U.S. Nuclear Regulatory Commission (1982), are applicable only to existing structures and equipment that will not be demolished and buried. They apply to both interior and exterior surfaces. If a building is demolished and buried, the guidelines in Section C.1 are applicable to the resulting contamination in the ground.

	Allowable Total Residual Surface Contamination (dpm/100 cm^2)†[1]		
Radionuclides†[2]	Average†[3],†[4]	Maximum†[4],†[5]	Removable†[6]
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100	300	20
Th-Natural, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1,000	3,000	200
U-Natural, U-235, U-238, and associated decay products	5,000α	15,000α	1,000α
Beta-gamma emitters (radionuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above	5,000β-γ	15,000β-γ	1,000β-γ

†[1] As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute measured by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

†[2] Where surface contamination by both alpha- and beta-gamma-emitting radionuclides exists, the limits established for alpha- and beta-gamma-emitting radionuclides should apply independently.

†[3] Measurements of average contamination should not be averaged over an area of more than 1 m^2. For objects of less surface area, the average should be derived for each such object.

†[4] The average and maximum dose rates associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/h and 1.0 mrad/h, respectively, at 1 cm.

†[5] The maximum contamination level applies to an area of not more than 100 cm^2.

†[6] 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 measuring the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of surface area less than 100 cm^2 is determined, the activity per unit area should be based on the actual area and the entire surface should be wiped. The numbers in this column are maximum amounts.

D. AUTHORIZED LIMITS FOR RESIDUAL RADIOACTIVITY

The remedial action shall not be considered complete unless the residual radioactivity is below authorized limits. Authorized limits shall be set equal to guidelines for residual radioactivity unless: (1) exceptions specified in Section F of this document are applicable, in which case an authorized limit may be set above the guideline value for the specific location or condition to which the exception is applicable; or (2) on the basis of site-specific data not used in establishing the guidelines, it can be clearly established that limits below the guidelines are reasonable and can be achieved without appreciable increase in cost of the remedial action. Authorized limits that differ from guidelines must be justified and established on a site-specific basis, with documentation that must be filed as a permanent record upon completion of remedial action at a site. Authorized limits differing from the guidelines must be approved by the Director, Oak Ridge Technical Services Division, for FUSRAP and by the Director, Richland Surplus Facilities Management Program Office, for remote SFMP--with concurrence by the Director of Remedial Action Projects for both programs.

E. CONTROL OF RESIDUAL RADIOACTIVITY AT FUSRAP AND REMOTE SFMP SITES

Residual radioactivity above the guidelines at FUSRAP and remote SFMP sites must be managed in accordance with applicable DOE Orders. The DOE Order 5480.1A requires compliance with applicable federal, state, and local environmental protection standards.

The operational and control requirements specified in the following DOE Orders shall apply to both interim storage and long-term management.

a. 5440.1B, Implementation of the National Environmental Policy Act

b. 5480.1A, Environmental Protection, Safety, and Health Protection Program for DOE Operations

c. 5480.2, Hazardous and Radioactive Mixed Waste Management

d. 5480.4, Environmental Protection, Safety, and Health Protection Standards

e. 5482.1A, Environmental, Safety, and Health Appraisal Program

f. 5483.1, Occupational Safety and Health Program for Government-Owned Contractor-Operated Facilities

g. 5484.1, Environmental Protection, Safety, and Health Protection Information Reporting Requirements

h. 5484.2, Unusual Occurrence Reporting System

i. 5820.2, Radioactive Waste Management

E.1 Interim Storage

a. Control and stabilization features shall be designed to ensure, to the extent reasonably achievable, an effective life of 50 years and, in any case, at least 25 years.

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b. Above-background Rn-222 concentrations in the atmosphere above facility surfaces or openings shall not exceed: (1) 100 pCi/L at any given point, (2) an annual average concentration of 30 pCi/L over the facility site, and (3) an annual average concentration of 3 pCi/L at or above any location outside the facility site (DOE Order 5480.1A, Attachment XI-1).

c. Concentrations of radionuclides in the groundwater or quantities of residual radioactive materials shall not exceed existing federal, state, or local standards.

d. Access to a site should be controlled and misuse of onsite material contaminated by residual radioactivity should be prevented through appropriate administrative controls and physical barriers--active and passive controls as described by the U.S. Environmental Protection Agency (1983--p. 595). These control features should be designed to ensure, to the extent reasonable, an effective life of at least 25 years. The federal government shall have title to the property.

E.2 <u>Long-Term Management</u>

a. Control and stabilization features shall be designed to ensure, to the extent reasonably achievable, an effective life of 1,000 years and, in any case, at least 200 years.

b. Control and stabilization features shall be designed to ensure that Rn-222 emanation to the atmosphere from the waste shall not: (1) exceed an annual average release rate of 20 $pCi/m^2/s$, and (2) increase the annual average Rn-222 concentration at or above any location outside the boundary of the contaminated area by more than 0.5 pCi/L. Field verification of emanation rates is not required.

c. Prior to placement of any potentially biodegradable contaminated wastes in a long-term management facility, such wastes shall be properly conditioned to ensure that (1) the generation and escape of biogenic gases will not cause the requirement in paragraph b of this section (E.2) to be exceeded, and (2) biodegradation within the facility will not result in premature structural failure in violation of the requirements in paragraph a of this section (E.2).

d. Groundwater shall be protected in accordance with 40 CFR 192.20(a)(2) and 192.20(a)(3), as applicable to FUSRAP and remote SFMP sites.

e. Access to a site should be controlled and misuse of onsite material contaminated by residual radioactivity should be prevented through appropriate administrative controls and physical barriers--active and passive controls as described by the U.S. Environmental Protection Agency (1983--p. 595). These controls should be designed to be effective to the extent reasonable for at least 200 years. The federal government shall have title to the property.

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F. EXCEPTIONS

Exceptions to the requirement that authorized limits be set equal to the guidelines may be made on the basis of an analysis of site-specific aspects of a designated site that were not taken into account in deriving the guidelines. Exceptions require approvals as stated in Section D. Specific situations that warrant exceptions are:

a. Where remedial actions would pose a clear and present risk of injury to workers or members of the general public, notwithstanding reasonable measures to avoid or reduce risk.

b. Where remedial actions--even after all reasonable mitigative measures have been taken--would produce environmental harm that is clearly excessive compared to the health benefits to persons living on or near affected sites, now or in the future. A clear excess of environmental harm is harm that is long-term, manifest, and grossly disproportionate to health benefits that may reasonably be anticipated.

c. Where the cost of remedial actions for contaminated soil is unreasonably high relative to long-term benefits and where the residual radioactive materials do not pose a clear present or future risk after taking necessary control measures. The likelihood that buildings will be erected or that people will spend long periods of time at such a site should be considered in evaluating this risk. Remedial actions will generally not be necessary where only minor quantities of residual radioactive materials are involved or where residual radioactive materials occur in an inaccessible location at which site-specific factors limit their hazard and from which they are costly or difficult to remove. Examples are residual radioactive materials under hard-surface public roads and sidewalks, around public sewer lines, or in fence-post foundations. In order to invoke this exception, a site-specific analysis must be provided to establish that it would not cause an individual to receive a radiation dose in excess of the basic dose limits stated in Section B, and a statement specifying the residual radioactivity must be included in the appropriate state and local records.

d. Where the cost of cleanup of a contaminated building is clearly unreasonably high relative to the benefits. Factors that shall be included in this judgment are the anticipated period of occupancy, the incremental radiation level that would be effected by remedial action, the residual useful lifetime of the building, the potential for future construction at the site, and the applicability of remedial actions that would be less costly than removal of the residual radioactive materials. A statement specifying the residual radioactivity must be included in the appropriate state and local records.

e. Where there is no feasible remedial action.

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G. SOURCES

Limit or Guideline	Source
Basic Dose Limits	
Dosimetry Model and Dose Limits	International Commission on Radiological Protection (1977, 1978)
Guidelines for Residual Radioactivity	
Residual Radionuclides in Soil Material	40 CFR 192
Airborne Radon Decay Products	40 CFR 192
External Gamma Radiation	40 CFR 192
Surface Contamination	U.S. Nuclear Regulatory Commission (1982)
Control of Radioactive Wastes and Residues	
Interim Storage	DOE Order 5480.1A
Long-Term Management	DOE Order 5480.1A; 40 CFR 192

H. REFERENCES

International Commission on Radiological Protection. 1977. Recommendations of the International Commission on Radiological Protection (Adopted January 17, 1977). ICRP Publication 26. Pergamon Press, Oxford. [As modified by "Statement from the 1978 Stockholm Meeting of the ICRP." Annals of the ICRP, Vol. 2, No. 1, 1978.]

International Commission on Radiological Protection. 1978. Limits for Intakes of Radionuclides by Workers. A Report of Committee 2 of the International Commission on Radiological Protection. Adopted by the Commission in July 1978. ICRP Publication 30. Part 1 (and Supplement), Part 2 (and Supplement), Part 3 (and Supplements A and B), and Index. Pergamon Press, Oxford.

U.S. Environmental Protection Agency. 1983. Standards for Remedial Actions at Inactive Uranium Processing Sites; Final Rule (40 CFR Part 192). Fed. Regist. 48(3):590-604 (January 5, 1983).

U.S. Department of Energy. 1984. Formerly Utilized Sites Remedial Action Program. Summary Protocol: Identification - Characterization - Designation - Remedial Action - Certification. Office of Nuclear Energy, Office of Terminal Waste Disposal and Remedial Action, Division of Remedial Action Projects. April 1984.

12

U.S. Department of Energy. 1985. Supplement to U.S. Department of Energy Guidelines for Residual Radioactivity at Formerly Utilized Sites Remedial Action Program and Remote Surplus Facilities Management Program Sites. A Manual for Implementing Residual Radioactivity Guidelines. Prepared by Argonne National Laboratory, Los Alamos National Laboratory, Oak Ridge National Laboratory, and Pacific Northwest Laboratory for the U.S. Department of Energy. (In preparation.)

U.S. Nuclear Regulatory Commission. 1982. Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material. Division of Fuel Cycle and Material Safety, Washington, DC. July 1982. [See also: U.S. Atomic Energy Commission. 1974. Regulatory Guide 1.86. Termination of Operating Licenses for Nuclear Reactors. Table I.]

13

APPENDIX C. RADIOLOGICAL INSPECTION INTERNAL LETTERS DURING THE DECONTAMINATION OF T005 FROM JUNE TO DECEMBER, 1987

1/18/88 3T00

F. G. Schmidt
635, 055-T006

R. J. Tuttle
641, 055-T100

4436

Radiological Decontamination Acceptance Surveys in Rooms 108, 111, and 111A at T005

Rooms 108, 111, and 111A at building T005 have been surveyed by Radiation and Nuclear Safety and found to satisfy the requirements needed to be acceptable for release for unrestricted use. A minimum 11% sample survey was performed on the floor and walls (5.5% on the ceiling) to measure the average, maximum, and removable alpha and beta surface activity. Exposure rate measurements were also taken 1m above the floor at 12 locations.

The results and statistical distributions are published in ETEC report GEN-ZR-0003, "Radiological Survey of Building T005." In all cases, the measurements show that the tests are satisfactorily passed and the rooms are acceptably clean. The rooms may be refurbished.

R. J. Tuttle, Manager
Radiation & Nuclear Safety

cc: C. J. Rozas CB01
M. E. Remley LA06
W. R. McCurnin T020
E. L. Babcock T020
Authorization File T100

1/18/88 2t005.

F. G. Schmidt
635, 055-T006

R. J. Tuttle
641, 055-T100

4436

Review of T005 Soil Radioactivity Data and Approval to Backfill and Repair Trenches in Rooms 110W, 116, and NOA

Soil exposed and removed during excavation and removal of drain lines from Rooms 110W(west), 116, and the North Outside Area (NOA) in building T005 was analyzed for radioactivity by gamma spectrometry. Approximately 60 samples were analyzed from this area. The attached figure shows the excavations. Soil samples were obtained below pipes, joints, and elbows. Water samples were obtained from drain lines when accessible. In one case, backhoe operations caused breakage of a 2" diameter, suspect R/A drain line. Water and soil samples were taken from beneath the broken drain line.

Each soil sample was analyzed by gamma spectrometry. The results show in every case, that either "No Detectable Activity" (NDA) or "Natural Radioactivity" is present. NDA means radioactivity content below the Minimum Detectable Activity of the spectrometer. "Natural Radioactivity" refers to naturally occurring radionuclides: K-40; and U-238, Th-232, and their daughter products.

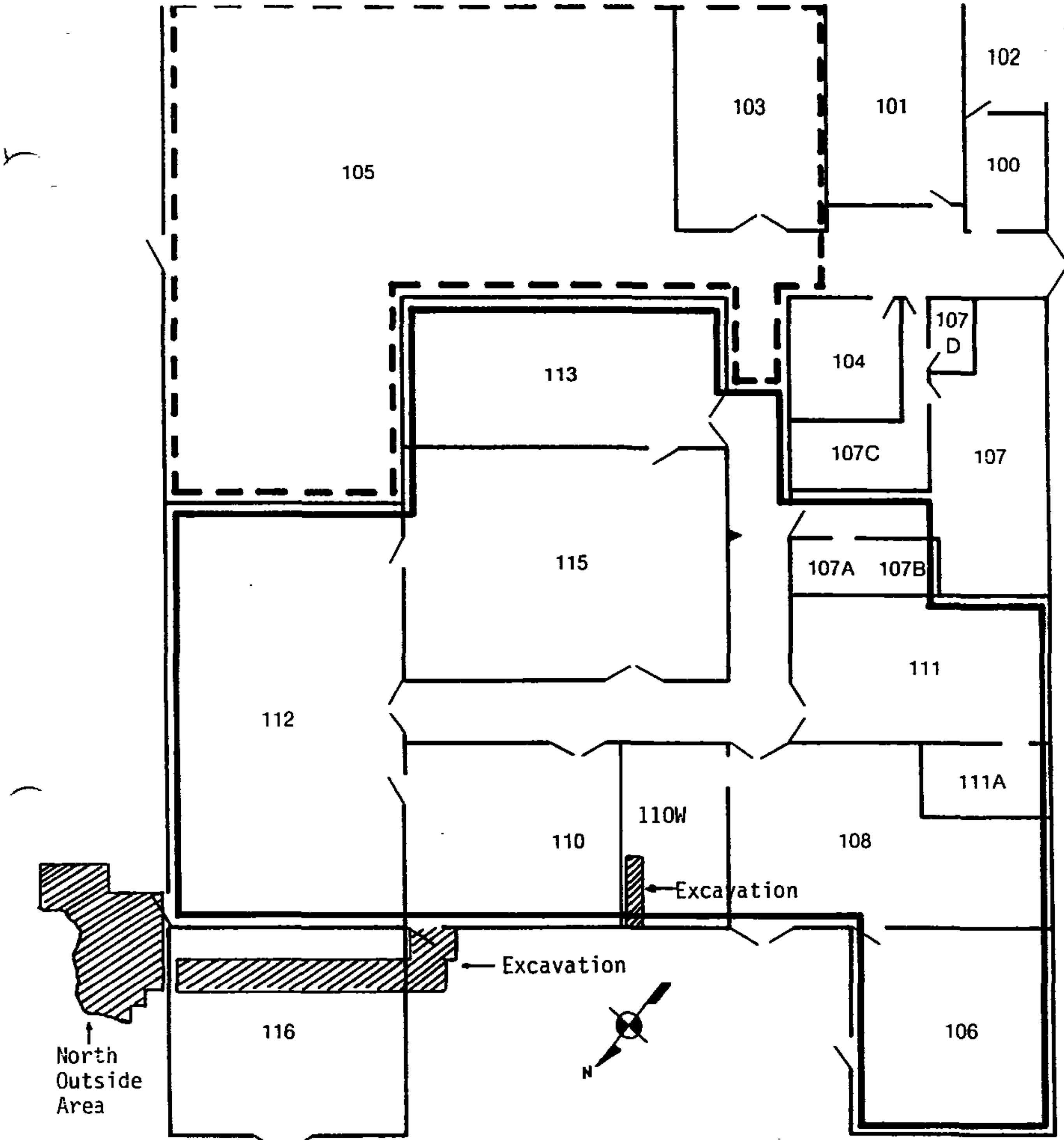
Therefore, the soil may be used as backfill in the trenches or disposed of in any other way.

RJTuttle

R. J. Tuttle, Manager
Radiation & Nuclear Safety

Attachment: Figure of Drain Line Excavations

cc: C. J. Rozas CB01
M. E. Remley LA06
W. R. McCurnin T020
E. L. Babcock T020
Authorization File T100
J. A. Chapman T100



Controlled Area Which Required the Use of Protective Clothing

Controlled Area Where Protective Clothing Was Not Required

Room	Description When Used During UCPFF	Present
100	Secretary's Office	Office
101	Supervisor's Office	Office
102	Lunch and Conference Room	Same
103	Chemical Processing	Chem Lab
104	Health Physics	Women's Restroom
105	Cladding Machine, End Cap Weld, and Loading Areas	Storage
106	Storage	Storage
107	Restroom, Hot/Cold Change Rooms	Change and Restroom
108	Blending	Vacant
110	U02 to UC Conversion	Molten Salt Computer Control Center
111	Quality Control	Vacant
111A	Quality Control	Vacant
112	Slug Casting	Storage
113	Inspection Batching and Loading	Equipment Drib
115	Fuel Slug Machining	Refurbished (Vacant)
116	Silicon Rectifier	Tool Crib

Date: 10/29/87 No.: 312il.rjt

To: Those Listed

From: R. J. Tuttle
641, 055-T100
4439

Subject: Sanitary Leach Field at T005

At a recent review meeting for the DOE Site Survey, the question was raised regarding testing of the T005 sanitary leach field for radioactivity. I do not believe this is warranted, for the following reasons:

1. Following discovery of radioactive contamination in the RMDF leach field, a review was made (by A. M. Stelle) of other leach fields. It is my recollection (I was not able to find his report) that no other leach fields were identified as suspect. (The RMDF leach field was a problem because it was connected to the liquid waste hold up tank. I think this was not true at T005. It was strictly a sanitary waste sewer system.)

2. All buildings in Area IV (the AI area) were connected to the SSFL sewer treatment plant in 1960-61. Most, or all, the radioactive work at T005 was done after this connection.

3. No radioactivity was found in the existing sanitary drain lines during the current survey.

For these reasons, I do not think a survey of the sanitary leach field is appropriate.

RJ Tuttle

R. J. Tuttle, Manager
Radiation and Nuclear Safety

dist:	K. T. Stafford	T038
	J. A. Chapman	T100
	K. L. Adler	T036
	W. R. McCurnin	T020
	M. E. Remley	LA06
	F. H. Badger	T020

10/6/87

R. J. Tuttle
641, 055-T100

F. H. Badger
641, 055-T020

5216

Occurrence of Radioactivity In Raccoon Fecal Samples

The site survey program required the radiological survey of the attic in T-005. The building had been used for the fabrication of configured uranium metal and the R/A exhaust duct was still present in the attic. The duct was found to be open to the environment and contaminated therein up to 80,000 dpm per 20 square centimeters beta-gamma activity and 60,000 dpm per 20 square centimeters alpha activity. Located in the same area was gross evidence of a family of raccoons (procyon lotor) living in the attic. Suspecting the animals may have entered the R/A exhaust duct and incidentally ingested some of the uranium dust in the duct, samples of the piles of fecal samples were carefully collected for spectographic analysis.

The fecal samples were dry and contained, in addition to normal material, plastic, similar to that used in sandwich wrap. The samples were size reduced and weighed (97 grams). They were placed in a Marinelli beaker and scanned for characteristic gamma peaks. The 16-hour scan with a high purity germainium detector coupled to a series 85 Canberra revealed the presence of cesium-137 but not uranium. The cesium was present at concentrations of 1.63xE-5 uCi/gram dry weight plus or minus 5% (16pCi/g).

In order to explain the presents of Cs-137 in the fecal sample one must understand the behavior patterns of the raccoon. The raccoon is a nocturnal carnivore and feeds on insect, reptiles, birds, eggs, fish and small rodents. They occasionally eat carrion and vegetable matter. They may travel up to 5 miles in an evening in search of food but are generally not far from water.

The RMDF, RIHL, and other facilities that have handled unencapsulated radioactive material have been aware of the insect and rodent problem in their controlled areas. Mice (peromyscuo boylei) have been trapped and found to be contaminated. If these mice, lizards, crickets etc. were consumed by carnivores such as raccoons, bobcats (lynx rufus), and coyote (canis latrans) their fecal samples(No activity detected in fecal samples to date) would logically contain radioactive material. It is fairly easy to restrict access of an animal the size of a raccoon but may prove impracticable to prevent mice, lizards and crickets which are prevalent throughout this area, from entering radiologically controlled areas with minor quantities of loose radioactive material present.

CONCLUSION:

The problem of intrusion of wild life into radiologically controlled areas must continually be addressed. While it is felt that the quantities detected in this case do not present a biological problem to the animal or the environs, continuing efforts should be made to restrict access of vermin to areas of unencapsulated radioactive material.

F. H. Badger
Radiation and Nuclear Safety

c.c.

W. R. McCurnin	T-020
C. J. Rozas	CB01
J. A. Chapman	T-100
J. D. Moore	T-100
W. A. McCollum Jr.	T-065
R. C. Shepard	T-065
M. S. Maseda	T-065
K. L. Adler	T-036
K. T. Stafford	T-038
P. S. Olsen	T-486

Internal Letter

Rockwell

10/6/87

joe2

TO: E. L. Babcock
635, 055-T020

FROM: R. J. Tuttle
641, 055-T100

4439

Disposal of Concrete From T005

Concrete slabs from the north outside area were surveyed before and after removal.

No radioactivity above natural background was detected. Therefore, this material may be treated as conventional trash and disposed of in any suitable manner.

R. J. Tuttle, Manager
Radiation & Nuclear Safety

cc:	J. Chapman	T100
	R. Gay	T006
	F. Schmidt	T006
	F. Schrag	T020
	D. Trippeda	T020
	J. Wallace	T034
	C. Rozas	CB01

9/23/87

1T005.ws

F. G. Schmidt
635, 055-T006

J. A. Chapman
641, 055-T100

5766

Decontamination Surveys in Room 115 at T005.

Room 115 at building T005 has been surveyed by the radiological survey inspection team and found acceptable for release for unrestricted use. A minimum 11% sample survey was performed on the floor and walls (5.5% on the ceiling) to measure the average, maximum, and removable alpha surface activity; and average, maximum, and removable beta surface activity. No maximum "hot spots" were found. Exposure rate measurements were also taken 1m above the floor at 12 locations. In all cases, the measurements show that the test is satisfactorily passed and the room is acceptably clean. The room may be refurbished.

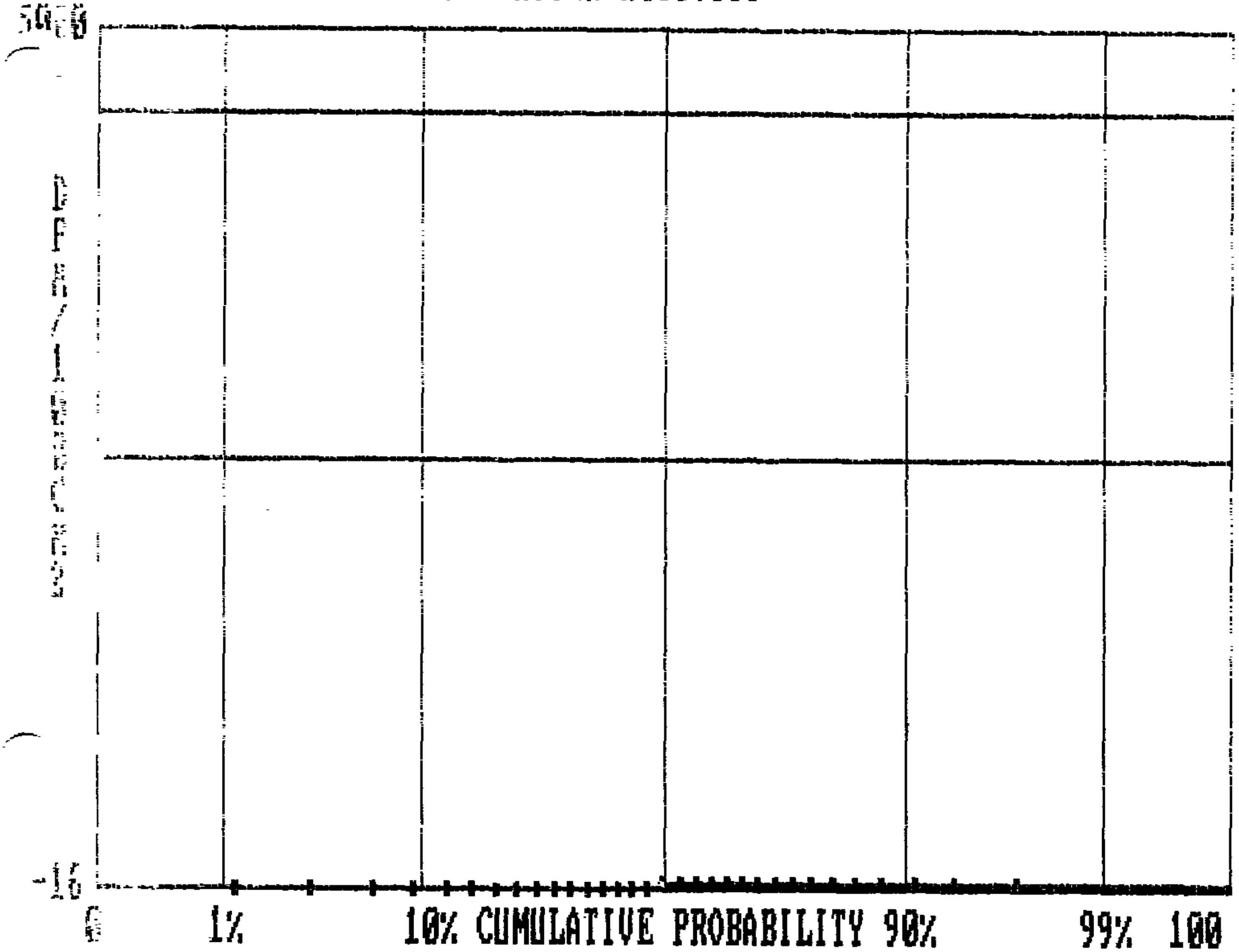
J. A. Chapman

J. A. Chapman
Radiation & Nuclear Safety

Attachments: Statistical Analysis Results

dist: W. McCurnin T020
E. Babcock T020
F. Schrag T020
P. Horton T034
Radiation & Nuclear Safety
AWC Inc.

151005F TOTAL ALPHA ACTIVITY



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[illegible] = [illegible]
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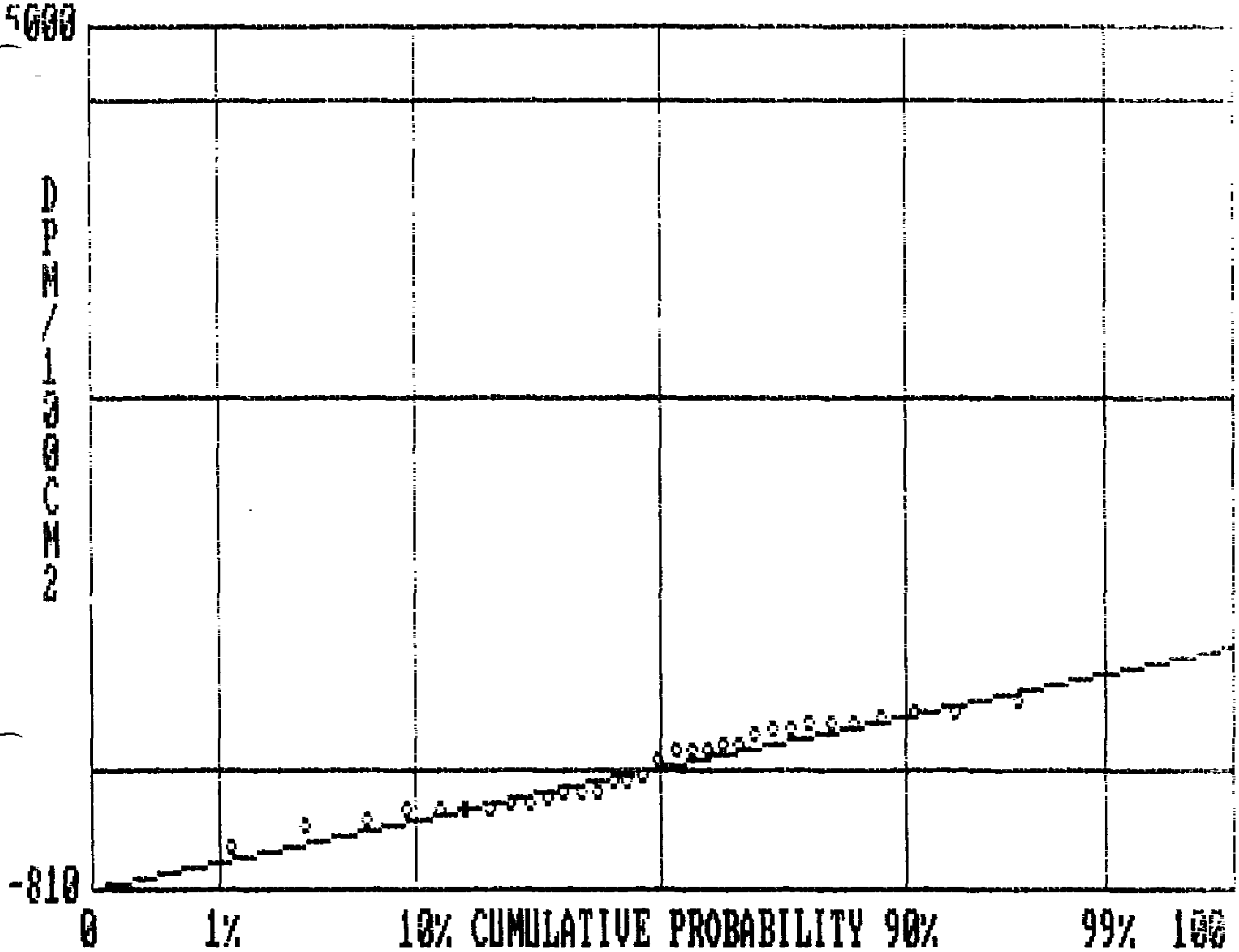
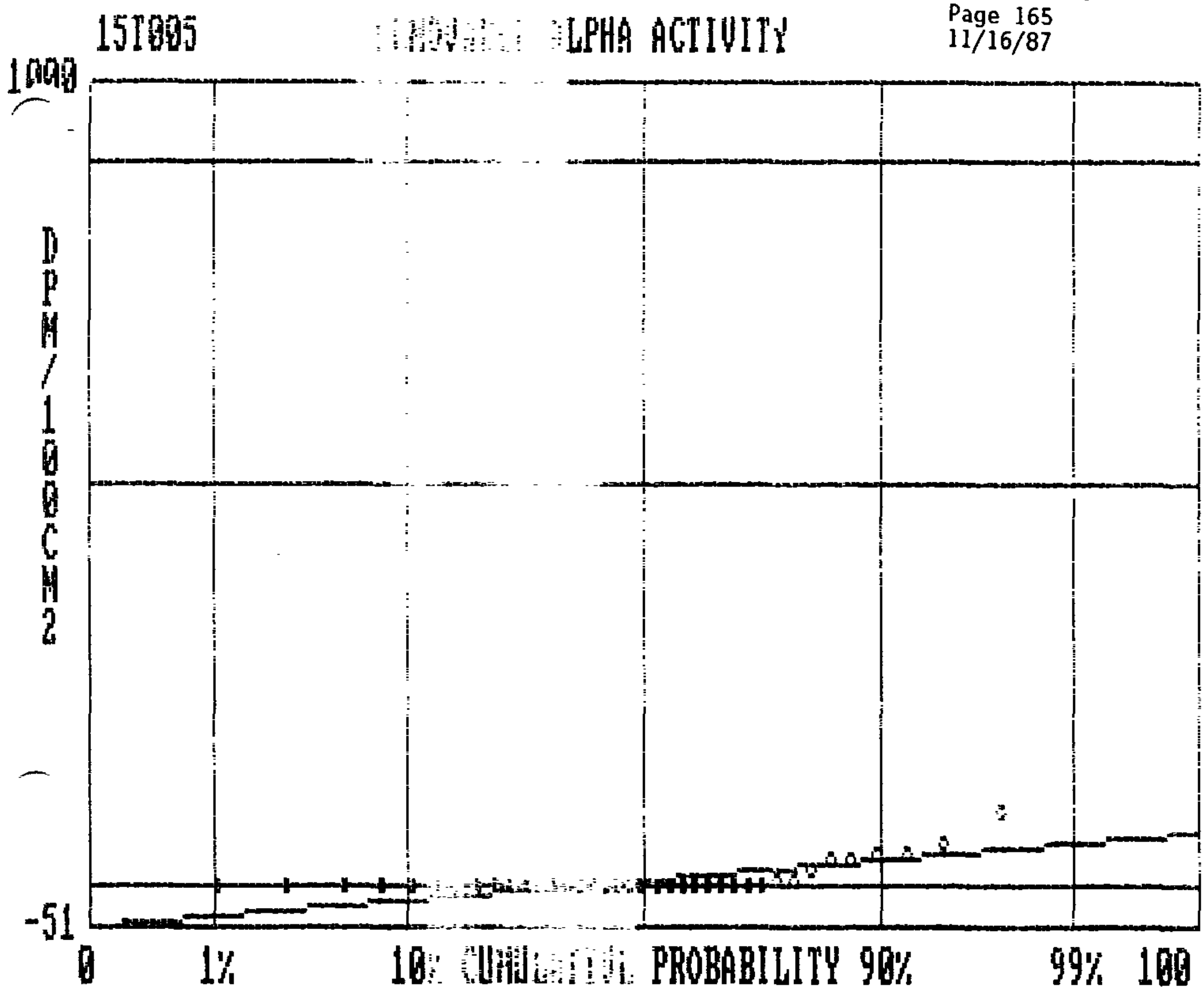
[illegible]
[illegible]
[illegible]

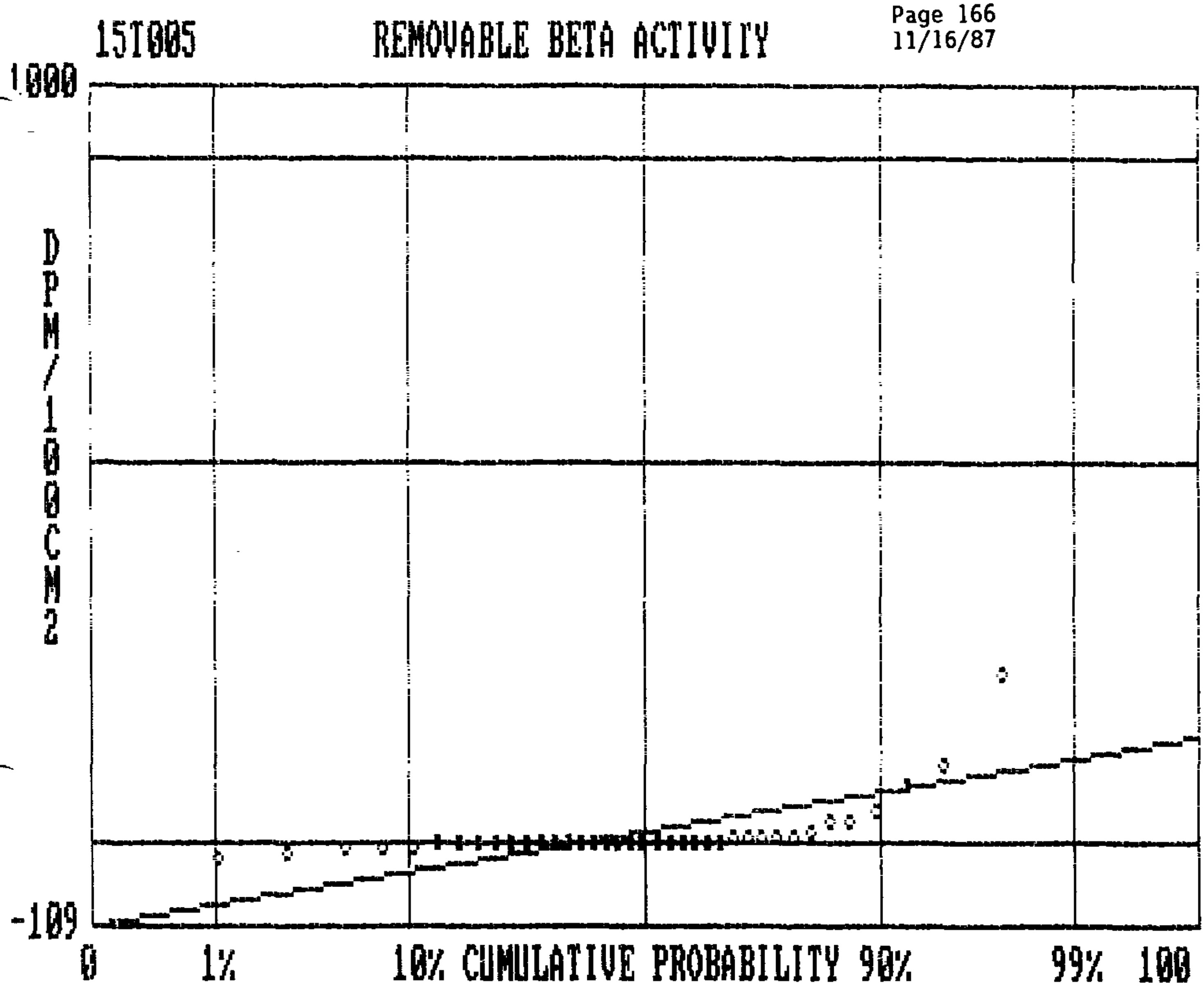
15T005F TOTAL BETA ACTIVITY



15T005F 09-23-1987
THE SURVEY CHARACTERISTIC IS: TOTAL BETA ACTIVITY
NUMBER OF POINTS = 72
AVERAGE VALUE = 13.11375
STANDARD DEVIATION = 274.3912
THE TEST STATISTIC (FOR CONSUMERS' RISK OF 0.1 AT 10% PROBABILITY) IS:
460.6297
THE ASSIGNED ACCEPTANCE LIMIT IS: 5000 DPM/100CM2

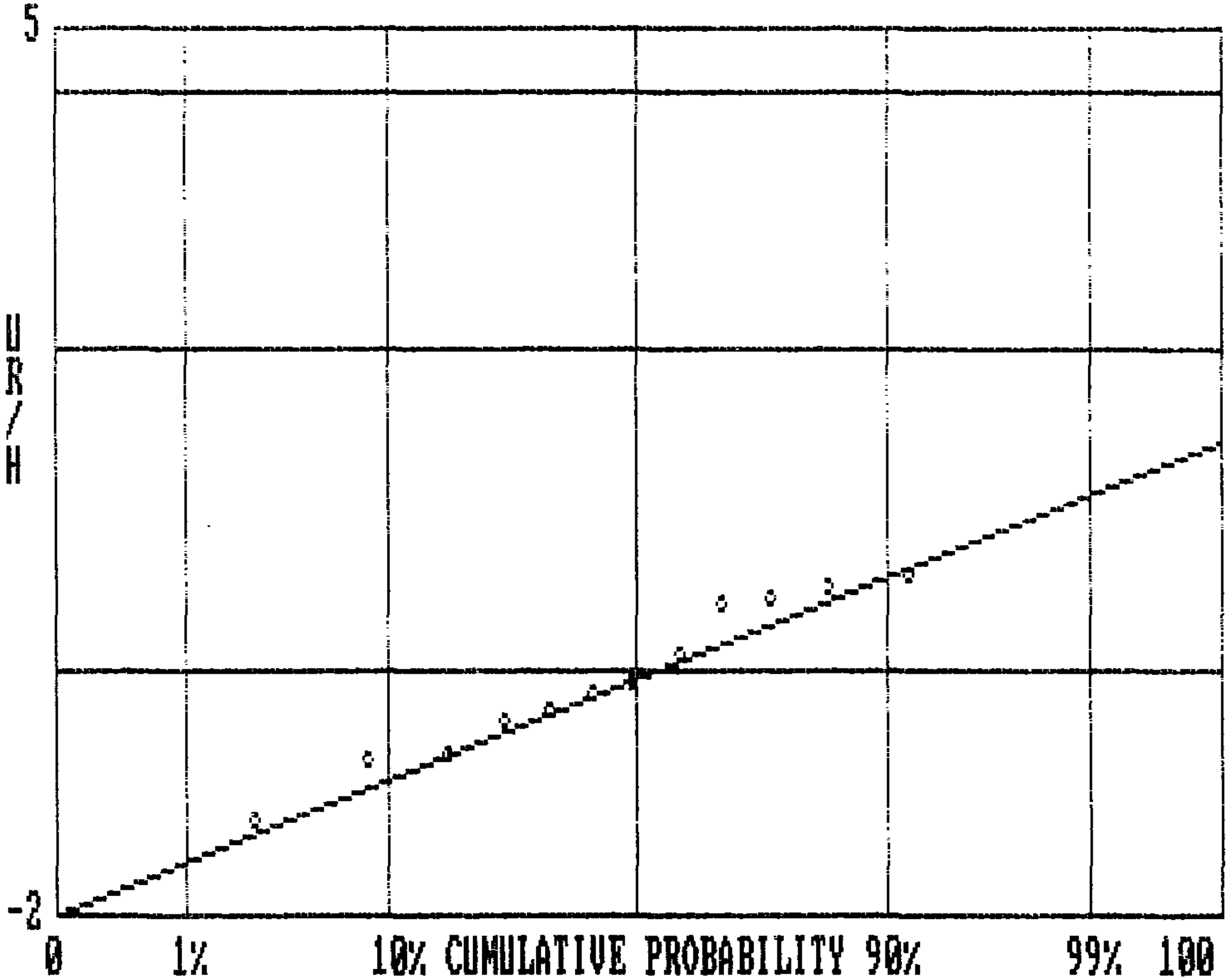


15T005 09-23-1987
THE SURVEY CHARACTERISTIC IS: REMOVABLE ALPHA ACTIVITY
NUMBER OF POINTS = 38
AVERAGE VALUE = 9.348632
STANDARD DEVIATION = 29.16274
THE TEST STATISTIC (FOR CONSUMERS' RISK OF 0.1 AT 10% PROBABILITY) IS:
41.56926
THE ASSIGNED ACCEPTANCE LIMIT IS: 1000 DPM/100CM2



15T005 09-23-1987
THE SURVEY CHARACTERISTIC IS: REMOVABLE BETA ACTIVITY
NUMBER OF POINTS = 38
AVERAGE VALUE = 14.42958
STANDARD DEVIATION = 41.2987
THE TEST STATISTIC (FOR CONSUMERS' RISK OF 0.1 AT 10% PROBABILITY) IS:
60.42607
THE ASSIGNED ACCEPTANCE LIMIT IS: 1000 DPM/100CM2

15T005G GAMMA EXPOSURE RATE



15T005G 09-23-1987
THE SURVEY CHARACTERISTIC IS: GAMMA EXPOSURE RATE
NUMBER OF POINTS = 12
AVERAGE VALUE = -5.287668E-02
STANDARD DEVIATION = .6072928
THE TEST STATISTIC (FOR CONSUMERS' RISK OF 0.1 AT 10% PROBABILITY) IS:
1.116143
THE ASSIGNED ACCEPTANCE LIMIT IS: 5 UR/H

9/22/87 joel

E. L. Babcock
635, 055-T020

R. J. Tuttle
641, 055-T100

4439

Disposal Of Concrete From T005

Concrete floor slabs from room 116 equipment room were surveyed before and after removal. No radioactivity above natural background was detected. Therefore, this material may be treated as conventional trash and disposed of in any suitable manner.

R. J. Tuttle, Manager
Radiation & Nuclear Safety

cc:	J. Chapman	T100
	R. Gay	T006
	F. Schmidt	T006
	F. Schrag	T020
	D. Trippeda	T020
	J. Wallace	T034
	C. Rozas	CB01

9/9/87 296il.rjt

R. L. Gay
635, 055-T006

R. J. Tuttle
641, 055-T100

4439

T005, Room 107

Ref: IL from R. J. Tuttle to F. G. Schmidt, dated 7/31/87.

Room 107, the men's shower room survey has been completed with the following results.

A. All available cleanout sludge samples were analyzed by gamma-spectrometry, specifically for U-235 which indicated no detectable activity.

B. Twelve 9"x9" floor grids were selected for tile removal and survey, all floor tile and floor grids indicated no detectable activity.

Room 107 is now released for unrestricted use.

Jim Rowles for RJT
R. J. Tuttle, Manager
Radiation & Nuclear Safety

dist:

E. Babcock	T020
J. Chapman	T100
F. Schmidt	T006
F. Schrag	T020
D. Trippeda	T020
J. Wallace	T034
C. Rozas	CB01

8/27/87

290il.rjt

E. L. Babcock
635, 055-T020

R. J. Tuttle
641, 055-T100

4439

Disposal/Interim Storage of Concrete from North Outside Area at T005

Concrete in the north outside area at T005 was surveyed. No radioactivity above natural background was detected. Therefore, if this material is removed, it may be treated as conventional trash and disposed of in any suitable manner. It may be stored in any location not subject to contamination.

RJ Tuttle

R. J. Tuttle, Manager
Radiation & Nuclear Safety

cc: Radiation and Nuclear Safety
AWC

F. Schrag	T020
P. Horton	T034
D. Trippeda	T020

8/21/87

290il.rjt

E. L. Babcock
635, 055-T020

R. J. Tuttle
641, 055-T100

4439

Disposal/Interim Storage of Concrete Rubble from Room 110A/111 at T005

Concrete removed from the floors in Rooms 110A(lunch room)/111 was surveyed (after selective scabbling) before and during removal. No radioactivity above natural background was detected. Therefore, this material may be treated as conventional trash and disposed of in any suitable manner. It may be stored in any location not subject to contamination.

RJ Tuttle

R. J. Tuttle, Manager
Radiation & Nuclear Safety

cc: Radiation and Nuclear Safety
AWC
F. Schrag T020
P. Horton T034
D. Trippeda T020

8/10/87
"REVISED"

290il.rjt

E. L. Babcock
635, 055-T020

R. J. Tuttle
641, 055-T100

4439

Disposal/Interim Storage of Concrete Rubble from Room 108/110 at T005

Concrete removed from the floors in Rooms 108/110 at T005 was surveyed (after selective scabbling) before and during removal. No radioactivity above natural background was detected. Therefore, this material may be treated as conventional trash and disposed of in any suitable manner. It may be stored in any location not subject to contamination.

R. J. Tuttle, Manager
Radiation & Nuclear Safety

cc: Radiation and Nuclear Safety
AWC
F. Schrag T020
P. Horton T034
D. Trippeda T020

7/31/87

288il.rjt

F. G. Schmidt
635, 055-T006

R. J. Tuttle
641, 055-T006

4439

Decontamination Surveys in Room 107 at T005.

Room 107 is the men's shower room at T005. Considering the past use of this building, it is very unlikely that the showers and associated drains were ever contaminated. However, the drains must be inspected to determine acceptability for release for unrestricted use. The following steps should be taken:

1. Mark the connected urinal "Not for Use" before and during the survey.

2. Flush the urinal wall and pour 1 gallon chlorine bleach into it. Let soak for 1 hour and flush several times.

3. Open all available cleanouts and scrape sludge samples separately from each. Collect as much as possible up to about 100 ml.

4. Analyze the samples collectively by gamma-spectrometry, specifically for U-235. If any is detected, the individual samples should be analyzed to identify the contaminated cleanouts.

5. If no activity is detected, the drains may be considered releasable. If activity is detected, further sampling/analysis and or cleaning may be required.

We do not expect the floor to be contaminated, either. This will be verified by removing a square set of 2x2 tiles in at least 10 locations determined by the HP surveyor. Survey the underside of the tiles, if possible, and the floor surface, for alpha and beta activity. If no activity is detected, all material may be disposed of as conventional waste.

If measurable activity less than 5000 cpm/100 cm^2 is found, it may be appropriate to remove all the floor tile and do a complete floor scan. If activity exceeding 5000 dpm/100 cm^2 is found, all floor tile should be removed and the floor should be cleaned.

No material with detectable activity may be disposed of as conventional waste.

RJ Tuttle

R. J. Tuttle, Manager
Radiation & Nuclear Safety

dist: W. McCurnin T020
E. Babcock T020
F. Schrag T020
P. Horton T034
Radiation & Nuclear Safety
AWC Inc.

7/22/87 282il.rjt

F. G. Schmidt
635, 055-T006

R. J. Tuttle
641, 055-T100

4439

Analysis of T005 Soil and Approval to Backfill and Repair Trenches

The soil exposed and removed during the excavation and removal of drain lines from Room 115 in T005 was analyzed for radioactivity by counting 2-gram samples in an NMC gas-flow thin-window proportional counter. For the purpose of qualifying the soil as having "no detectable activity above background", direct comparison was made with the radioactivity in outside soil taken nearby. The resultant values were reported as differences from background activity, proportional to radioactivity concentration in pCi/g. The alpha counts were not corrected for sample thickness.

The radioactivity values were then plotted as a linearized Gaussian cumulative distribution function. (Values from a Gaussian distribution lie along a straight line in this sort of plot.) These plots are attached to this IL.

These plots show that the values are well represented by a single Gaussian distribution for each type (alpha and beta). Since the majority of the soil would be expected to show no artificial radioactivity, and the average shows little difference from the natural soil activity, this shows that none of the soil has detectable radioactivity.

Therefore, the soil may be used as backfill in the trenches or disposed of in any other way. The trenches should be filled and the floor repaired before the final acceptance survey is started.

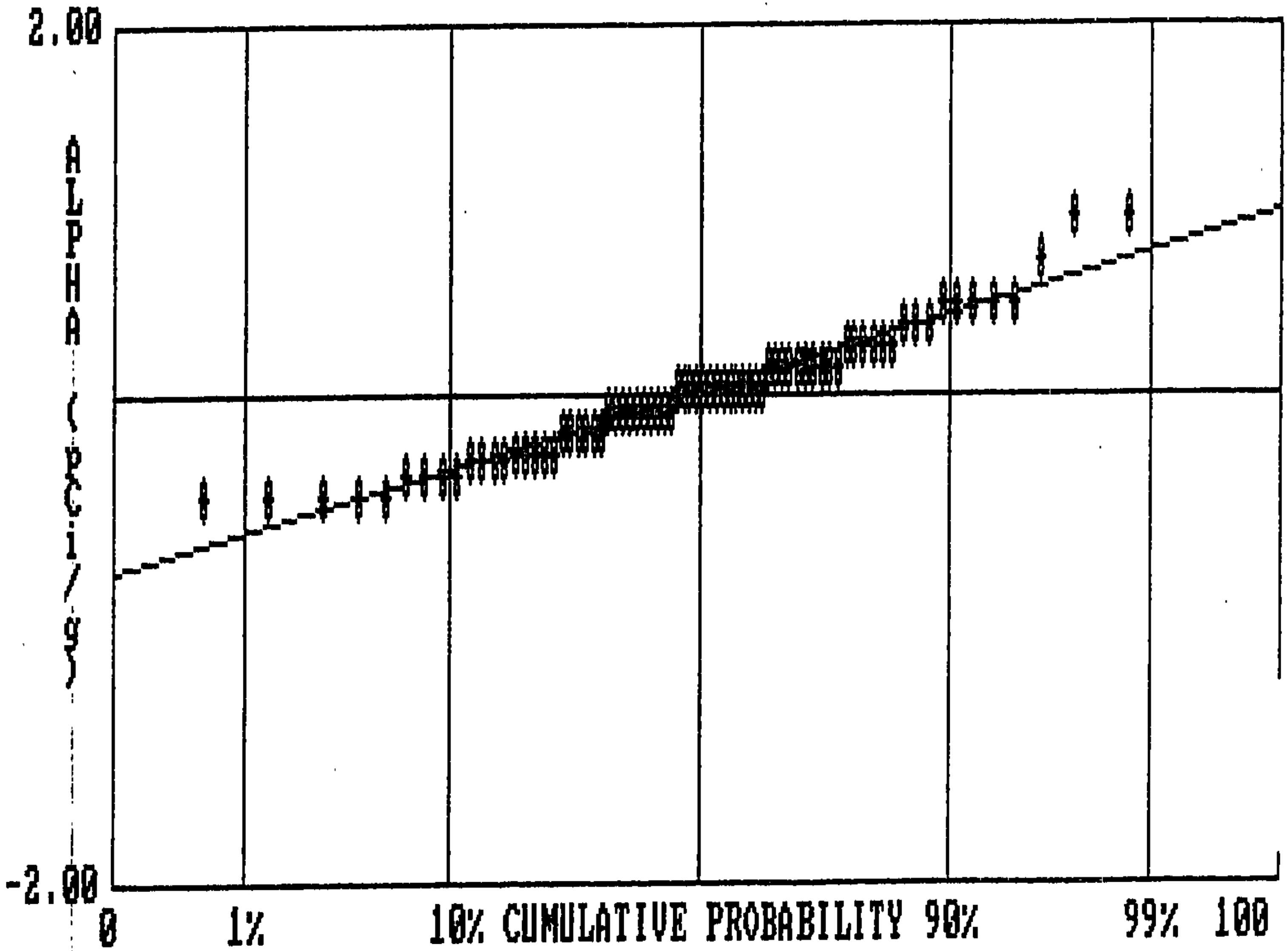
RJTuttle

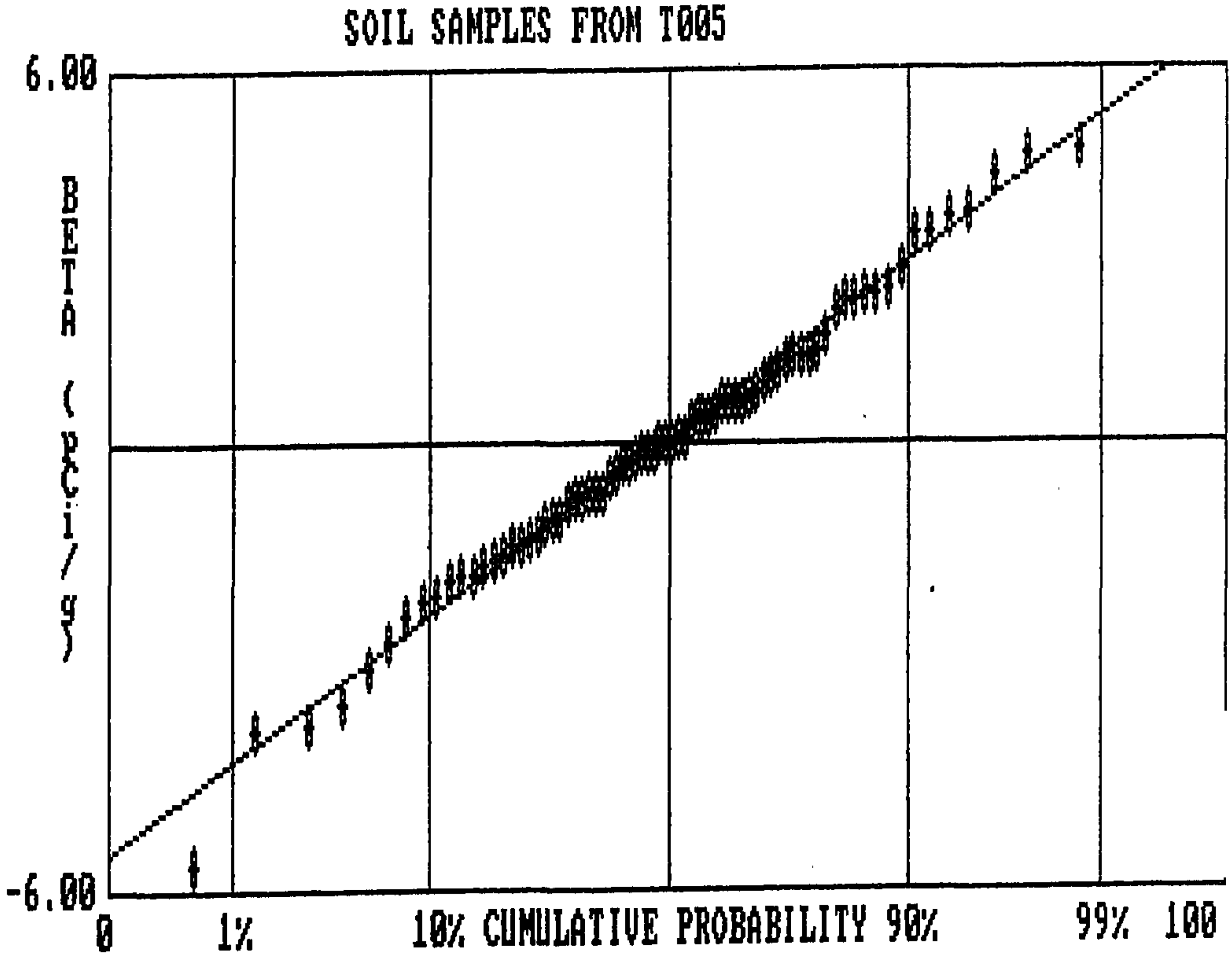
R. J. Tuttle, Manager
Radiation & Nuclear Safety

Attachments

cc: C. J. Rozas CB01
M. E. Remley LA06
W. R. McCurnin T020
E. L. Babcock T020
Authorization File T100
J. A. Chapman T100

SOIL SAMPLES FROM T005





GEN-ZR-0003
Page 177
11/16/87

Internal Letter

7/13/87 ray5

To: Those Listed

From: E. R. McGinnis
641, 055-T100
(AWC Inc.)
4439

Subject: Status of Wall Braces in Room 115, T005.

Ref: IL dated 6/17/87 "Contamination Under Wall Braces in Room 115, T005".

The baseboard was removed in Room 115 and a 100% scan was performed with a beta-gamma frisker. A 6" piece of the wall brace was removed at ten selected locations. Survey of the area underneath showed that the contamination was not liquid seepage that could have gone under the brace as was first thought. All of the activity is contained within the mastic which just goes to the edge of the brace. It was found to be easily removable by scraping, and this operation can be performed without removing the wall brace.

Comparison of the readings obtained with the detector cocked sideways at the edge of the brace and flat on the floor after brace removal showed up to seven times higher with the latter method. This means, when surveying at the edge of the wall brace, levels approaching the Minimum Detectable Activity Value of the survey instrument could be above the limit. All areas showing detectable activity will be decontaminated. A copy of all of the survey data involved in the test is attached for your information.

E R McGinnis

E. R. McGinnis
AWC Inc.

cc: F. Schmidt T006
E. Babcock T020
W. McCurnin T020
R. Tuttle T100
F. Badger T020
J. Bowman SS11
D. Sheptanko SS11
D. Hunter SS11
D. Trippeda T006

Internal Letter — Rockwell

6/25/87 — 272il.

TO: Those Listed

FROM: R. J. Tuttle
641, 055-T100

4439

Minimization of Radioactive Waste in Decon Jobs

In the current decontamination work at T005 an unnecessarily excessive amount of radioactive waste was produced because of inadequate awareness of the current regulatory policy that no radioactive material may be disposed of as conventional waste. This resulted from saw cutting and jack hammering of a radioactively contaminated concrete floor where the concrete could have been easily removed by selective scabbling. As a result, we have approximately 11 drums of radioactive waste rather than a bucket of scabbling dust for burial and a large amount of non-radioactive, uncontaminated conventional concrete rubble.

In all future work and planning, please take explicit notice of the requirement that no radioactive material, even material that passes the limits for release in unrestricted use, may be disposed of as conventional waste.

RJ Tuttle

R. J. Tuttle, Manager
Radiation & Nuclear Safety

cc: Radiation & Nuclear Safety
AWC Inc.

W. McCurnin	T020
E. Babcock	T020
D. Elliott	T020
R. Meyer	T009
G. Poucher	T009
P. Horton	T034
F. Seward	T034
F. Schmidt	T006
D. Parker	T020
R. Cutting	T020
D. Tripeda	T020
S. Pendleberry	LB26
F. Schrag	T020
W. Dennison	LB01
R. Frazier	T009
D. Harrison	T020
W. Nagel	LB03

Date 6/17/87 rayl

To From

F. Schmidt
635, 055-T006

E. McGinnis
641, 055-T100
(AWC Inc.)
4439

Subject Contamination Under Wall Braces in Room 115, T005.

Preliminary survey of the baseboard area of the walls in Room 115, T005, showed the presence of fixed beta-gamma contamination. At some of these locations the wall was cut out and surveys showed that contamination seeped through the wall and under the metal wall braces inside. The entire baseboard area of the walls will be cut out and a 100% beta-gamma scan will be performed along the wall brace. At ten known contaminated spots, a small piece of the wall brace will be removed. A survey of the floor underneath will be performed to see if here is any correlation between the surveys of the brace and the area underneath.

This information will be forwarded to Operations and Radiation and Nuclear Safety management so that a decision can be made on whether to remove the entire brace or leave it with documented contamination.

E.R. McGinnis

E. R. McGinnis
Health Physicist
AWC Inc.

cc: R. J. Tuttle T100
W. R. McCurnin T020
F. H. Badger T020
R. L. Gay T006
J. Bowman SS11
D. Sheptenko SS11
D. Hunter SS11

Internal Letter   Rockwell International

Date 6/2/87   No. 263il.rjt

TO: Those Listed

FROM: R. J. Tuttle
641, 055-T100

4439

Subject: Radiological Survey Procedures for Partial Decontamination of Building T005

Parts of Building T005 at SSFL are being decontaminated in preparation for moving in some laboratories from DeSoto 104. This IL specifies the radiological surveys to be performed.

1. Wall and ceiling paint:

   Scrape approximately 10 grams of paint from 5 locations in each 3-m x 3-m square. Gamma spectometry for U-235. Homogenize and sample 2 grams, count for gross beta and alpha.

2. Conventional waste:

   Survey for detectable activity. No material with detectable radioactivity may be disposed of as conventional waste.

3. Uncovered floor:

   100% scan with pancake GM. Then measure at least 30 locations in 3-m x 3-m grid for alpha and beta (total and removable) using scanning scaler and 100 cm^2 smear.

4. Walls:

   11% locations in 3-m x 3-m grid for alpha and beta, as for floors.

5. Ceiling:

   5.5% locations in 3-m x 3-m for alpha and beta, as for walls.

6. Special structural features:

   Total survey for alpha and beta (total and removable) using countrate meter and 100 cm^2 smears.

7. Soil in drain line trenches.

   Sample (10g) at least every meter of length, count (2g) for gross alpha and beta. Take additonal samples as indicated by condition of pipe.

The PC program developed for the NMDF survey will be used for analysis and interpretation. All surveys are be fully documented.

R. J. Tuttle, Manager
Radiation & Nuclear Safety

cc:	F. Schmidt	T006
	V. Swanson	T006
	W. McCurnin	T020
	P. Horton	T034
	M. Remley	LA06
	C. Rozas	CB01
	Radiation & Nuclear Safety	
	AWC Inc.	

APPENDIX D. GAMMA SPECTROMETRY RADIONUCLIDE GAMMA-SIGNATURE LIBRARY

1.	ZR-95	64.40 D								
	724.0	44%	756.6	55%						
2.	NB-95	35.15 D								
	765.7	99%								
3.	RU-103	39.35 D								
	497.0	86%	610.0	5%						
4.	SB-125	0.1011E04 D								
	176.2	6%	428.0	29%	463.5	10%	606.7	5%	636.1	11%
5.	I-131	8.04 D								
	284.2	6%	364.5	81%	636.9	7%				
6.	CS-134	752.63 D								
	563.2	8%	569.2	15%	604.6	98%	795.7	85%	801.7	9%
7.	CS-136	12.98 D								
	66.8	12%	86.2	6%	153.1	7%	176.5	14%	273.5	13%
	340.5	47%	818.5	100%	1048.0	80%	1235.2	20%		
8.	CS-137	0.1095E05 D								
	661.6	85%								
9.	BA-140	12.80 D								
	162.5	5%	537.3	20%						
10.	LA-140	1.68 D								
	328.7	18%	487.0	43%	815.7	22%	867.8	5%	925.0	6%
	1596.0	95%								
11.	CE-141	32.50 D								
	36.0	8%	145.1	48%						
12.	CE-144	284.19 D								
	133.5	11%								
13.	CR-51	27.70 D								
	320.0	9%								
14.	MN-54	312.19 D								
	834.7	100%								
15.	FE-59	45.10 D								
	1099.1	56%	1291.5	43%						
16.	CO-58	70.78 D								
	511.0	30%	810.7	99%						

17.	CO-60	0.1924E04 D									
	1173.1	100%	1332.5	100%							
18.	ZN-65	243.80 D									
	511.0	3%	1115.5	51%							
19.	RH-102	0.1054E04 D									
	418.2	10%	475.0	93%	628.0	6%	631.0	56%	697.0	45%	
	766.7	33%	1046.5	33%	1112.6	17%					
20.	RH-102M	206.00D									
	475.0	44%	511.0	23%							
21.	SB-124	60.20 D									
	602.6	98%	645.7	7%	722.7	12%	1691.0	50%	2091.1	6%	
22.	BE-07	53.40 D									
	477.5	10%									
23.	NA-22	949.00 D									
	511.0	180%	1274.5	100%							
24.	K-040	0.46E12 D									
	1460.7	11%									
25.	RA-226	0.584E06 D									
	186.0	3%									
26.	PB-214	0.02 D									
	74.7 6%	77.0 11%	241.8	7%	295.1	19%	352.0	37%			
27.	BI-214	0.01 D									
	609.2	46%	1120.2	15%	1238.0	6%	1764.5	15%			
28.	RA-224	3.66 D									
	241.0	4%									
29.	PB-212	0.44 D									
	74.7	9%	77.0	18%	87.1	6%	238.5	43%			
30.	BI-212	0.04 D									
	727.1	12%	1620.5	3%							
31.	TL-208	0.00 D									
	277.3	6%	510.6	22%	583.0	86%	860.5	12%			
32.	AC-228	0.25 D									
	338.3	12%	911.0	29%	964.5	5%	968.8	17%			

33.	TH-234	24.10 D								
	63.2	4%	92.3	2%	92.7	3%				
34.	U-232	0.263E05 D								
	269.0	4%								
35.	U-235	0.26E12 D								
	93.3	2%	143.7	11%	163.3	5%	185.6	54%	205.2	5%
36.	AM-241	0.158E06 D								
	59.5	36%								
37.	NP-237	0.7817E09 D								
	29.0	9%	86.1	13%						
38.	PU-242	0.1409E09 D								
	44.5	3%								
39.	AM-243	0.2699E07 D								
	74.6	66%								
40.	NP-239	2.35 D								
	99.5	15%	103.6	24%	106.0	23%	117.6	8%	228.1	11%
	277.5	14%								
41.	AL-26	0.2612E10 D								
	511.0	164%	1808.6	100%						
42.	NB-94	0.7409E07 D								
	702.5	100%	871.0	100%						
43.	AG-108M	0.4635E05 D								
	79.5	7%	433.6	90%	614.3	90%	722.9	90%		
44.	CD-109	453.00 D								
	88.0	3%								
45.	BA-133	0.3906E04 D								
	81.0	33%	276.2	7%	302.6	19%	355.8	62%	383.6	9%
46.	EU-148	54.00 D								
	413.8	11%	414.0	7%	550.1	99%	553.1	17%	571.8	9%
	611.2	19%	629.8	71%	725.6	12%	1034.0	8%		
47.	EU-152	0.4636E04 D								
	121.7	29%	244.6	8%	344.2	27%	778.8	13%	964.0	14%
	1085.7	10%	1112.0	13%	1408.0	21%				

48.	EU-154	0.3102E04 D										
	123.0	40%	248.0	7%	723.2	20%	873.1	11%	996.2	11%		
	1004.7	18%	1274.7	35%								
49.	EU-155	0.181E04 D										
	86.3	33%	105.2	22%								
50.	TB-158	0.5475E05 D										
	79.5	11%	181.8	9%	780.1	9%	944.1	43%	962.1	20%		
51.	PT-193	0.1825E05 D										
	63.2	24%	64.8	44%	73.5	15%						
52.	CO-57	270.00 D										
	122.0	86%	136.3	11%								
53.	SR-85	64.73 D										
	513.9	99%										
54.	Y-88	106.60 D										
	898.0	94%	1836.0	99%								
55.	SN-113	115.10 D										
	391.6	64%										
56.	CE-139	137.50 D										
	165.7	80%										
57.	HG-203	46.59 D										
	72.8	6%	279.1	81%								
58.	TA-182	115.00 D										
	67.7	41%	100.1	14%	152.4	7%	222.0	7%	1121.2	35%		
	1189.0	16%	1221.4	27%	1230.9	11%						