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TITLE:
"Radiological Survey of the Source and Special Nuclear Material Storage Vault-Bldg T64"

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ABSTRACT

A radiological survey was performed at Building T064, located at Rockwell International's Santa Susana Field Laboratory (SSFL), to clarify and identify those areas needing further radiological inspection or requiring remedial action. Building T064, known as the Source and Special Nuclear Material Storage facility, was operated by North American Aviation and its successors in support of AEC, ERDA, and DOE nuclear related programs. As the name implies, T064 was used for storage of packaged items of source material (normal uranium, depleted uranium, thorium) and special nuclear material (enriched uranium, plutonium, U-233). Miscellaneous radioactive wastes, irradiated fuel elements, and spent fuel casks have also been stored on the facility grounds. T064 is currently used in a small capacity for storage of depleted uranium. The purpose of this survey was to characterize the site for residual radioactive contamination.

The building interior was surveyed for fixed and removable alpha/beta contamination. Ambient gamma exposure rate measurements were performed within the fenced-in storage yard and surrounding 2-acre area. Soil samples, debris, and miscellaneous items were analyzed for radioactivity as appropriate.

The results of this survey and analysis show that a few miscellaneous items inside the facility and the filter plenums are slightly contaminated. Interior walls and floors are not contaminated. An area to the east of the facility fence measuring no more than 300 ft^2 in area is significantly contaminated with mixed fission products. Exposure rates in this area are 300 μR/h on contact. Cs-137 contamination exists in concentrations not exceeding 2700 pCi/g. From this isolated area (about 70 ft east of the fence) westward to the fence and eastward about 50 ft is a slightly contaminated area about 40 ft wide. Exposure rates here approach 40 μR/h at 1 meter. This 3700 ft^2 area may contain Cs-137 radioactivity at 500 pCi/g. Further investigation and remedial action is required in this area.

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1.0 INTRODUCTION

Building T064, known as the Source and Special Nuclear Material Storage facility, was surveyed and analyzed for residual radioactive material. The facility and surrounding 2-acre area were radiologically characterized to determine whether further investigation is required or remedial action is necessary. This radiological survey was conducted as prescribed in the "Radiological Survey Plan for SSFL," (Reference 4).

Building T064 is located in Area IV of Rockwell International's Santa Susana Field Laboratories (SSFL) in Ventura County, California. Designed and built in the late 1950's to support government nuclear-related programs, T064 was used for storing source and special nuclear material; repackaging source and enriched uranium powders; and sectioning and repackaging fresh fuel elements. Plutonium was always handled in a packaged form; never as a loose powder. The fenced-in storage yard was used on occasion for storing recoverable uranium scrap, irradiated fuel elements, and miscellaneous radioactive wastes. Spent fuel shipping casks and shipping trailers were stored on the T064 site, just outside the western fence-line. During performance of this survey, normal and depleted uranium metal was stored in the south vault, and an old Sodium Reactor Experiment (SRE) Moderator cask was parked on the west yard. Because of the storage and repackaging activities which took place here, a radiological survey was performed to document the facility's radiological condition.

Radioactive contamination was suspect in the storage vaults and less suspect inside the fence. Surrounding areas, particularly effluent pathways, were slightly suspect for residual contamination because of the potential for radionuclide transport and migration from the facility. Only one contamination incident has occurred. In the 1960s, a special drum (pig) containing an irradiated fuel element was stored in the east yard. Water, which occupied the void space in the drum, leaked out through a rusted drain plug in the bottom. Mixed fission product activity was released to the immediate area. A large area was excavated for burial. Slight contamination still exists in that area.

As part of the DOE Site Survey (Reference 4), a radiation survey was performed on building floors and walls (up to 3 meters in height) to measure the average, maximum, and removable alpha surface activity, and average, maximum and removable beta surface activity. Miscellaneous structures and items such as wall coving, wall-to-floor joints, I-beams, light fixtures, stored merchandise, fire extinguishers, fume hood, balances, doors, exhaust vents, and rain gutters were surveyed in the same manner. The fenced-in storage yard and surrounding 2-acre area were surveyed for gamma-emitting radionuclides. If contamination was indicated outside by gamma measurements, samples were collected and analyzed for radioactivity.

Interior areas were gridded in 3-m by 3-m sections. 1 m^2 out of each 9-m^2 section was analyzed for alpha/beta contamination. Each specific 1-m^2 location within a 9-m^2 grid was selected as an area suspect of potential residual contamination based on operational history or present appearance.

Exterior areas inside the fence were gridded in 3-m by 3-m sections. Those outside the fence in 6-m by 6-m sections. In both cases, one ambient gamma exposure rate measurement (in micro-roentgens/hour) was made in each grid; one per 9 m^2 and one per 36 m^2, respectively. Where radioactive contamination was found to exceed 50% of the acceptable contamination limit, a few additional locations in the immediate area were surveyed. In cases where sampling was required, each sample was analyzed for gross alpha/beta radioactivity and gamma-emitting radionuclides.

All survey data were input into a Personal Computer (PC) graphics program which plots the radiation measurement value against its cumulative probability. This software also calculates a test statistic using inspection by variables techniques. This test statistic is that value greater than the mean value of the distribution, which corresponds to a consumer's risk of acceptance of 10% probability with a Lot Tolerance Percent Defective (LTPD) of 0.10. This method assumes the data follow a Gaussian probability distribution function. Inspection by variables techniques allow a thorough, understandable, and conclusive study for assessing the facility contamination level.

Radiation measurements are compared against DOE residual radioactivity limits specified in "Guidelines for Residual Radioactivity at FUSRAP and Remote SFMP Sites," (Reference 1). This guide generally agrees with previously published guides and standards, including ANSI Standard N13.12 (Reference 7), Regulatory Guide 1.86, and USNRC License SNM-21 (Reference 2). The limits for acceptable ambient gamma exposure rates differ between the DOE and NRC. DOE specifies 20 μR/h above background while NRC specifies 5 μR/h above background as acceptable gamma exposure rate limits. Because of the large variability observed for natural background at SSFL, ambient gamma measurements were not corrected for background. Rather, independent "natural" background distributions are presented for comparison against T064 data.

2.0 IDENTIFICATION OF FACILITY PREMISES

2.1 Location

Building T064 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. Figure 2.3 is a plot plan of the western portion of SSFL which includes Area IV where Building T064 is located. It is located within the 90.26 acre government-optioned area.

2.2 Building Characteristics and Site Topography

Building T064 was designed and built as a special nuclear material and source radioactive material storage building. It was constructed in two phases. The first phase was constructed in 1958. This 2137 ft^2 portion, (now room 110), is a reinforced concrete structure with 11-in thick walls on a concrete slab. The building eave height is 16 ft, and the structure is open bay except for a 12 ft x 13 ft material handling area in the southeast corner of the building. A fume hood was installed in this small southeast corner, (room 104).

In 1963 the building was enlarged by adding a bay to the north (room 114) bringing the total square footage of the building to 4418 ft^2. This addition used 12-in concrete block construction with cores filled with concrete. Total square footage includes a small 150 ft^2 office (room 100) and a 50 ft^2 restroom (room 102), both located on the dock on the east side of the building. On the northwest corner is a small supply room, about 50 ft^2, (room 116).

Figure 2.1 Map of Los Angeles Area

Figure 2.2 Map of Neighboring SSFL Communities

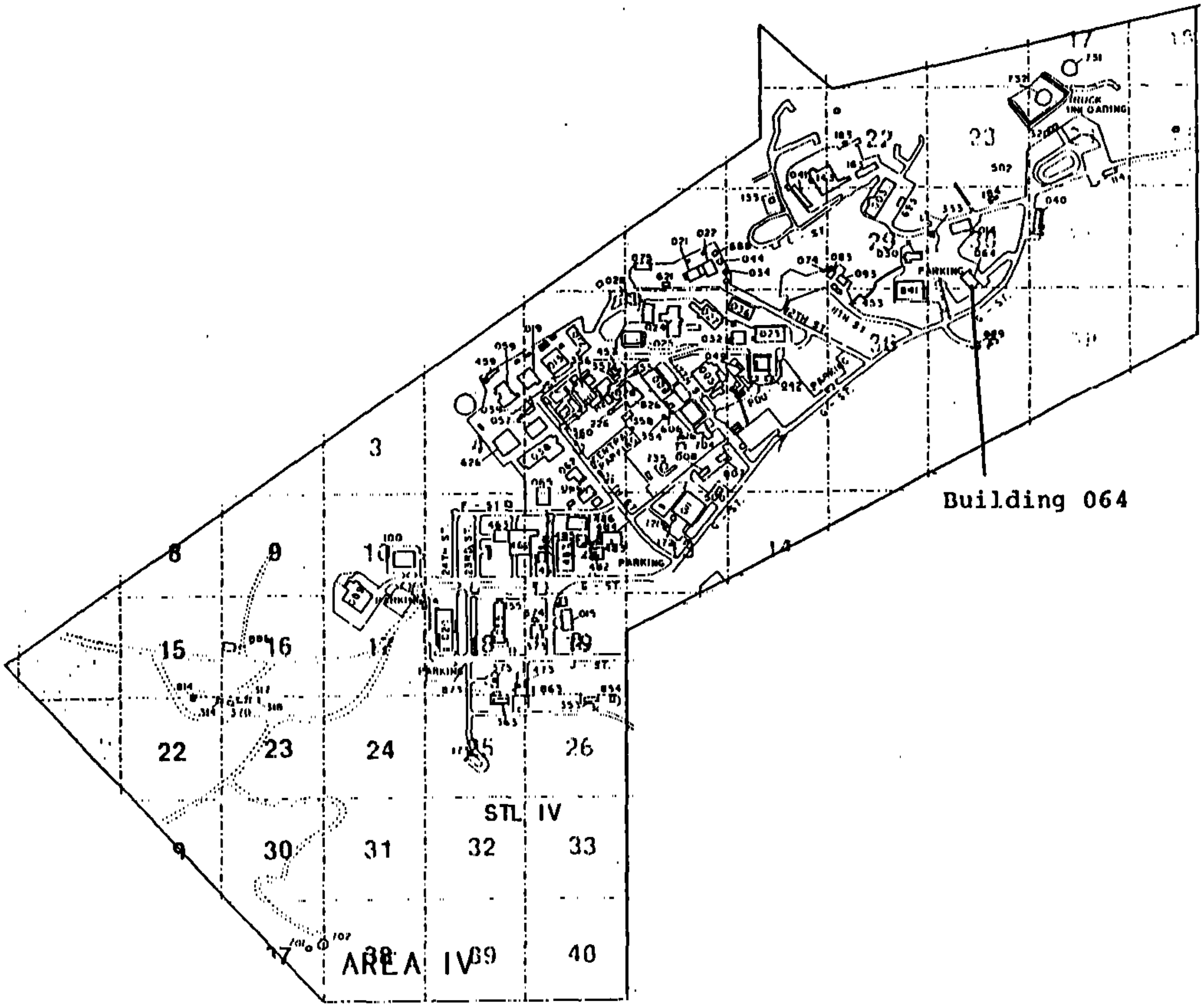


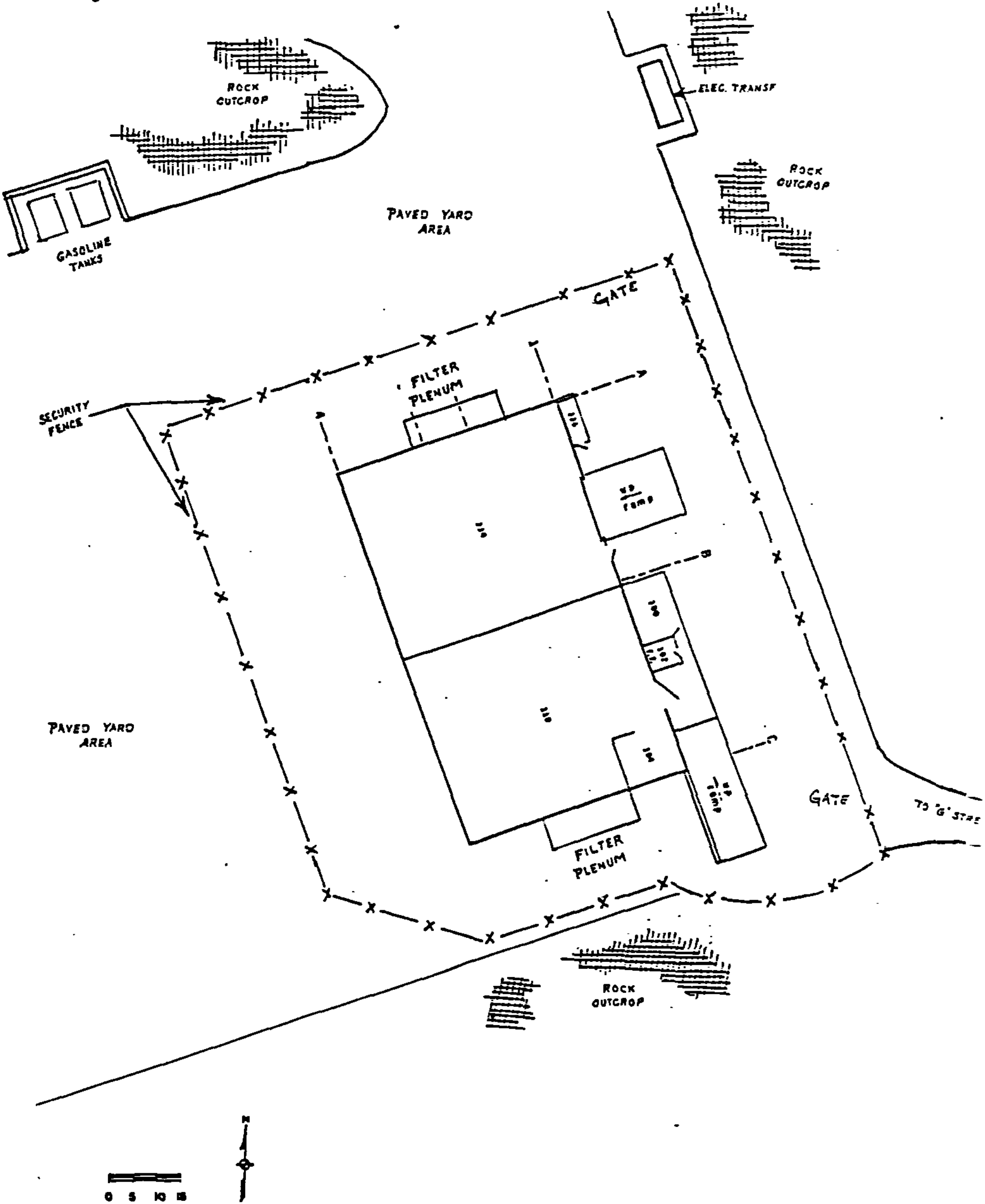
Figure 2.3 SSFL Layout Showing Location of T064

The concrete-slab floors are covered with 9-in square vinyl tiles. The concrete-block walls are painted. In 1980, the entire facility was re-roofed; interior wall surfaces were patched and painted; floor tile was removed and replaced; the restroom and office were restored; asphalt was patched; plumbing was repaired; heating and ventilation was repaired; and a window air-conditioner was installed in the office. Ten-ft long fluorescent lights are suspended from the 16-ft high ceiling. 14-ft high fuel storage racks were constructed to accommodate fuel. Room 114 is accessible from the east through a 20 ft x 15 ft electrically driven rollup door and a conventional hinged door. Room 110 is accessible from the east through a heavy secured door. These two rooms are extremely secure. Ramps leading to each room allow easy transport of materials via forklift.

Since nuclear material is only stored here, there is no processing equipment within the building. No sinks are installed in the storage areas. The only water supply is to the restroom (room 102); this water is released to the sewer. The facility is not air conditioned. Each vault is ventilated by dedicated blowers through a plenum containing pre-filters and HEPA filters. Room 104 has a fume hood which also exhausts through the south filter plenum.

Figure 2.4 is a plot plan of the building and immediate surrounding yard area. The facility sits atop a plateau about 25 ft above "G" Street and slightly above the 513 parking lot. Rock outcroppings exist upslope to the north-northeast and downslope in every other direction. Water runoff is primary due east at the southern end of the facility. A sanitary leach field existed several years ago just north of the access road to "G" Street on the southeast section of the property. About 2 acres of surrounding land was surveyed adjacent to the fenced-in facility. The building is surrounded by a chain link fence which is located from 20 to 30 ft from the exterior walls of the building. The area it encloses, including the building, is about 11,000 ft^2. The asphaltic concrete paving extends beyond the fence on the west side of the building, and to the north where it joins with parking lot 513. The west area has served as a parking area for trailers, shipping containers, and fuel casks.

Figure 2.4 Source and Special Nuclear Material Storage Building, T064



There are three points of access to the site location of Building T064. One access is directly from the north through the 513 parking area which is on the east side of 10th Street. A second point of access is directly off of 10th Street at the NW corner of the facility, and the third is a short paved roadway connecting the SE corner of the facility with "G" Street to the east. There are two gates for accessing the fenced-in storage yard. One from the northeast corner, off of the 513 parking lot. The other from the southeast corner, off of "G" Street. Figure 2.5 is an aerial photo of Building T064 as viewed from a location south of the facility, looking north. Figure 2.6 is a view of the east side of the facility including the dock, office, crane, and main entrance. Figure 2.7 is a photo of the north side, viewed from the northwest. It shows the northern filter plenum and stack which exhausts room 114.

2.3 Building Utilization and Current Radiological Condition

This building was used primarily for storage of packaged items of source material (normal uranium, depleted uranium, thorium) and special nuclear material (enriched uranium, plutonium, U-233) of various forms and configurations. Originally both the north (room 114) and the south (room 110) vaults contained steel racks for storing material. The south side was primarily used for storage of highly enriched uranium and plutonium bearing items; the north side primarily was used for source material and low enriched uranium storage.

Enriched uranium powders and source material powder packages were split into smaller units or combined into larger units in a glove box located in the small work area alcove (room 104) in the southeast corner of room 110. The glove box has since been removed from the building. Plutonium was handled only in packaged form; never in a loose form. No plutonium repackaging was done other than transferring sealed packages between containers. Transfers of solid metallic forms of material generally were always handled in the glove box; however, on occasion, larger pieces were transferred and repackaged within the vault area. During shutdown and termination of the SNAP program, excess Zr-U (enriched U) alloy product line material was sectioned into lengths suitable for packaging for shipment into

DOE (AEC) containers. This was done near the edge of the south side alcove in the vault. The floor was covered with plastic sheets before the Zr-U was "sectioned" using a common hack saw.

During the early 1960's, a special lead-pig cask containing irradiated "Seawolf" fuel elements was stored in the fenced-in area across from the crane on the east side. The irradiated fuel elements were probably transferred to the cask in a spent fuel pool. Before shipping to SSFL, the drain plug on the bottom of the cask should have been removed to drain the radioactive water, but was not. The cask was shipped and stored here while still containing water. The drain plug eventually rusted out, and water leaked out onto the T064 surface. The water contained mixed fission products which contaminated the area. A large area (6 ft across x 5 ft deep x 20 ft long) was excavated for burial to 0.3 mR/h and the section back-filled and repatched. Some slight Cs-137 contamination still exists outside the fence to the east. This area was thoroughly surveyed.

During the early 1960's, a changed storage configuration was required. The metal racks from the south half of room 110 was removed in order to store material in "birdcages" and drums. This storage included large quantities of special nuclear material recoverable scrap.

During this time, recoverable scrap space was at a premium. As a result, the entire yard area in front of the building (East), the side (North) and the back (West) was filled with 55 gallon drums of low enriched recoverable scrap. This material was shipped to various recovery sites in the mid-to late 1960's and early 1970's.

No plutonium or U-233 packages were ever opened in either vault. Any residual radioactive contamination is enriched uranium, normal uranium, depleted uranium, or thorium and generally could be expected to have come from "dust" from handling bare metallic pieces.

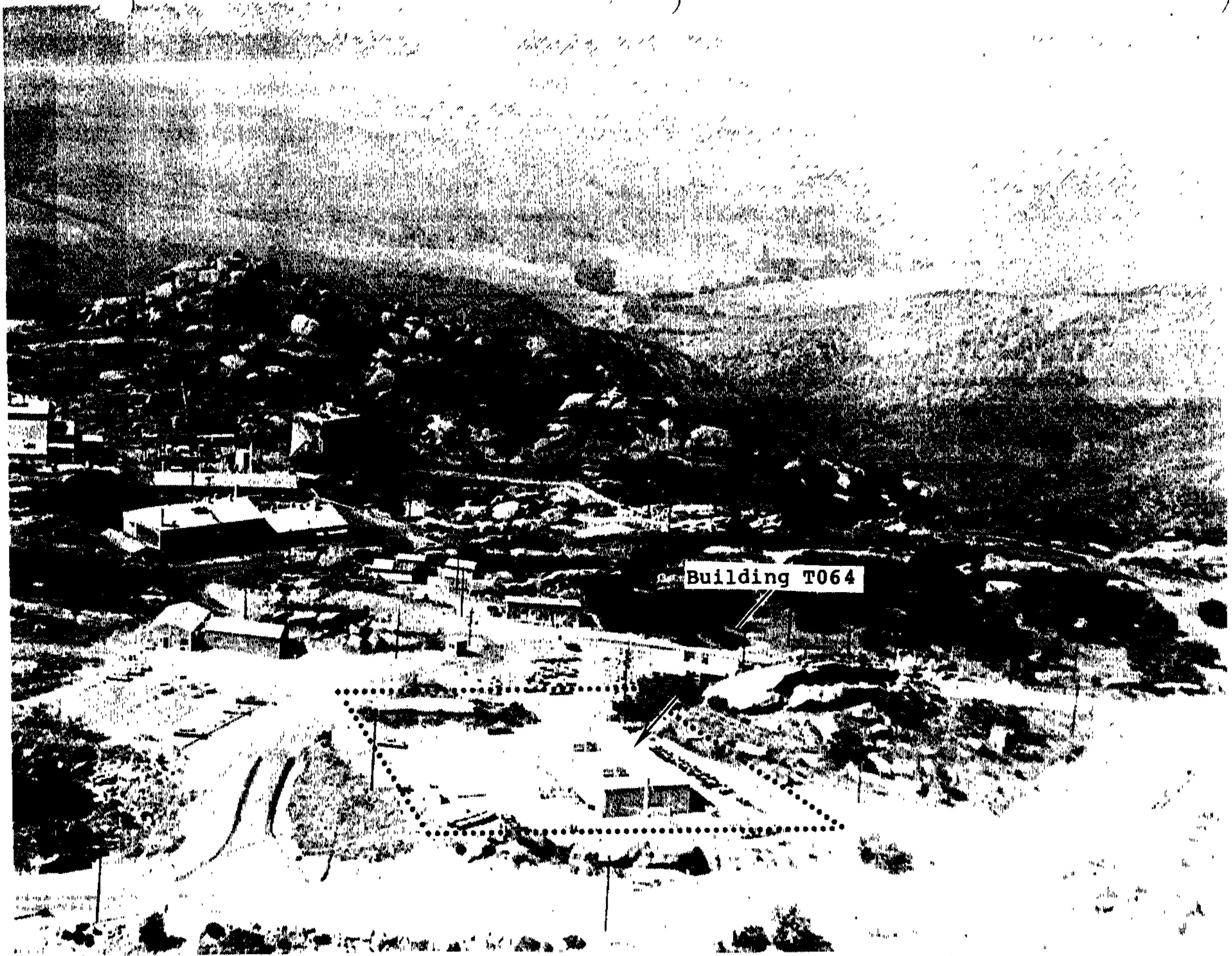


Figure 2.5 View of Building T064 From a Position South of the Complex (1963)

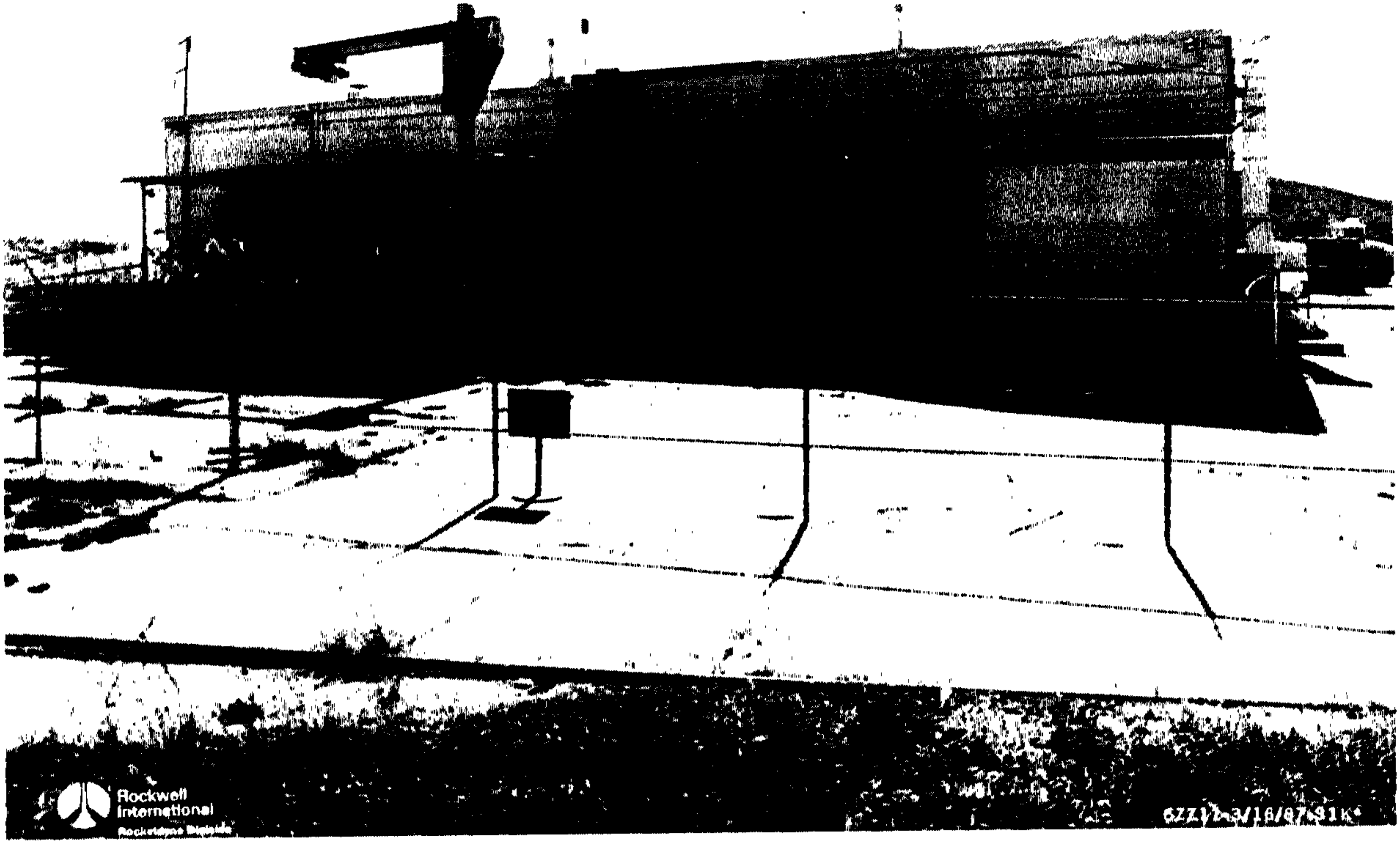


Figure 2.6 View From East of Building T064 (1987)

Figure 2.7 View from North of Building T064 (1987)

During the mid 1970's to early 1980's, most of the major DOE nuclear development and reactor contracts had ended. No special nuclear material powders were handled or repackaged after 1980. Most of the material had been sent to other DOE sites for recovery and use. Because of leaking, a new roof was installed in 1980. Shortly afterward, the walls were repainted and other repairs were made. The racks from the north vault were removed and the area converted to storage of non-nuclear DOE components. This area currently is being used to store supplies and equipment for Atomics International Operations functions (some of which may be packaged in a radioactive material condition).

The south vault currently is being used for storing only packaged solid depleted uranium pieces. This material is awaiting authorization for shipment to a user/disposal site. Normal uranium was shipped offsite shortly after performance of this survey.

3.0 SURVEY SCOPE

Interior building areas were radiologically characterized by measuring average, maximum, and removable alpha/beta contamination. The fenced-in storage yard and surrounding 2-acre area were radiologically characterized by measuring ambient gamma exposure rates 1 meter above the surface. If this gamma measurement indicated contamination, surface soil samples were acquired and analyzed by gamma spectrometry and for gross alpha/beta activity. About 101 m^2 were surveyed inside the building for alpha/beta contamination; this does not include miscellaneous items and building features. Each m^2 was surveyed uniformly for 5 min. The radioactive material exhaust vents and filter plenums were surveyed for residual contamination. About 58 and 168 ambient gamma exposure rate measurements were made inside the fenced-in yard and surrounding 2-acre area, respectively. These data were analyzed statistically by sampling inspection by variables techniques against appropriate residual contamination acceptance limits.

3.1 Unrestricted-use Acceptable Contamination Limits

A sampling inspection plan using variables, discussed in Section 4.2, was used to compare radiological contamination quantities against unrestricted-use acceptable contamination limits prescribed in DOE guidelines (Reference 1), Regulatory Guide 1.86, NRC license SNM-21, and other references. The limits shown in Table 3.1 below have been adopted by Rocketdyne. Measurements of average surface alpha/beta contamination should not be averaged over an area of more than 1 m^2. The maximum allowable alpha/beta contamination level applies for a single area of not more than 100 cm^2 in that 1 m^2. Allowable removable alpha/beta contamination is based on a surface wipe with area equal to 100 cm^2.

Table 3.1 Building T064 Maximum Acceptable Contamination Limits

Criteria	Alpha (dpm/100 cm^2)	Beta (dpm/100 cm^2)
Total Surface, averaged over 1 m^2	5000	5000
Maximum Surface, in 1 m^2	15000	15000
Removable Surface, over 100 cm^2	1000	1000
Ambient Gamma Exposure Rate*	5 μR/h above background	
Soil Activity Concentration**	46 pCi/g	100 pCi/g
Water Activity Concentration***	1×10^{-4} μCi/ml	1×10^{-5} μCi/ml

* Although DOE Guide (Reference 1) recommends a value of 20 μR/h above background for ambient gamma exposure rate, NRC has required 5 μR/h. For conservatism, we use 5 μR/h above background to compare survey results.

** Alpha activity concentration limits for enriched uranium is 30 pCi/g (Reference 12) plus that contribution from naturally occurring radioactivity, (about 16 pCi/g from Reference 18, p. 93). The total beta activity concentration limit is 100 pCi/g, including background.

*** The most restrictive alpha/beta water radioactivity concentrations for a restricted area taken from DOE Order 5480.1 Chapter XI, Table 1, Column 2. Alpha corresponds to Pu-239, beta to Sr-90.

Limits for soil and water radioactivity concentrations are also applicable on an as-required basis. Current guidance for acceptable soil radioactivity is nearly non-existent. The limits used here for alpha contamination, for example, are based on enriched uranium (Reference 12). These appear to be the best, most appropriate and realistic limits, and compare quite favorably to DOE's "factor of 3 above background per 100 m^2 area" recommendation (Reference 1, Section C.1). No effort was made to sum the concentrations of individual radionuclides and calculate the dose for the mixture so as to show that it does not exceed the basic dose limit. The level of contamination present at this facility does not warrant this type of detailed analysis.

Three specific action levels were established during the survey. These are proactive action levels initiated when the surveyor detects radiation according to the following criteria:

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.

3.2 Sample Lots

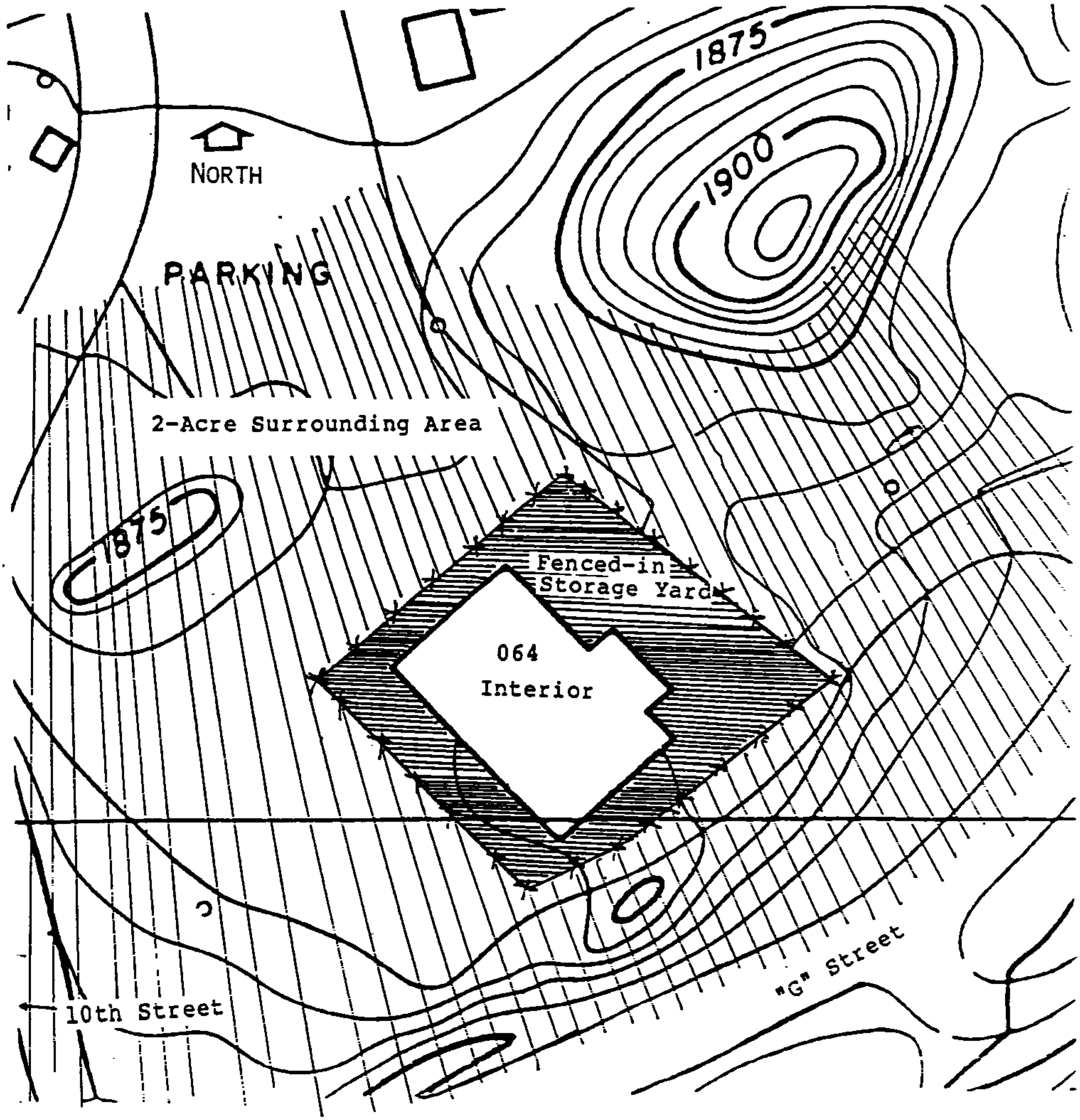
For purposes of the T064 radiological survey, the building, fenced-in storage yard, and surrounding 2-acre area were treated as separate sample lots for characterization and interpretation. Figure 3.1 shows the survey sampling lot plan.

For indoor areas and each loading ramp, a minimum of an 11% survey was performed on every wall up to 10 ft in height and on all floors. Because of the ceiling height (16 ft) and the low probability for residual contamination to exist there, the ceiling was not surveyed. Measurements for average, maximum, and removable alpha/beta contamination were made in this sample lot. The sampling inspection plan used is based on a uniform 3-meter square grid (9 m^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. A grid was superimposed on walls and floors. Each survey area was identified in matrix notation with codes indicating the surface (F = floor, N = north wall, E = east wall, S = south wall, W = west wall) and a two figure cartesian coordinate indicating the distance in meters from a local bench-mark. The (1,1) position for the floor was benchmarked as the northwest corner of each room. The (1,1) position on each wall was benchmarked as the bottom left hand corner of the wall as an observer would view it from the middle of the room. Position (3,1), (3,2), ...(3,n) was the highest row measured on each wall, the top of which equals about 10 ft. From each 3-m square grid (9 m^2), 1 m^2 was surveyed. Each 1-m^2 area was surveyed directly for alpha/beta contamination for 5 min. A 100 cm^2 wipe was taken in each selected 1 m^2 for analysis of removable contamination.

For the fenced-in storage yard, a 3-meter square grid was super-imposed on the area and one ambient gamma exposure rate measurement made in each 9 m^2 area. Location (1,1) was the northwestern most grid on the site.

For the surrounding 2-acre area, a 6-meter square grid was superimposed over the terrain and one ambient gamma exposure rate measure-ment made in each 36-m^2 area. Location (1,1) was the northwestern most grid on the site (near building 641).

Figure 3.1 Building T064 Sampling Lot Plan



3.3 Alpha and Beta Contamination Measurements

In order to determine alpha/beta contamination in each square meter surveyed per 9-m^2 area, four radiological characteristics were measured: total-average alpha surface activity, total-average beta surface activity, removable alpha surface activity, and removable beta surface activity. 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. An alpha probe and beta probe were each connected to a Ludlum Model 2220-ESG 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 using a Th-230 alpha source. The energy of Th-230 alpha particles (4.6 MeV) is similar to that of the isotopes handled at T064, U-235, U-234, and U-238.

Measurements of 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 U-238 daughters. The measurements were made over the same area as was used for each measurement of total average alpha surface activity.

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/100 cm^2), conversion factors were applied as follows. First, "background" radiation levels were determined for each 1 m^2 surveyed. A 1/4" piece of plywood was placed on the m^2 of interest to stop alpha and beta particles emitted from "the source." Instruments were placed on the plywood and a 1-min. background count collected. This task was performed because of high ambient radiation fluxes emitted from stored depleted and normal uranium in vault 110. Of course these gamma fluxes only increased the beta background. 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.

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. Detector "background" and efficiency was determined to convert the results to dpm/100 cm^2.

Thus, for surface contamination measurements of alpha and beta activity, data included sample location, total counts recorded in a five minute scan, maximum hot spot if present, natural background for one minute, efficiency factor, and 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.

Special structural features and miscellaneous items were surveyed in a similar manner for "indication only" of residual alpha/beta contamination.

3.4 Ambient Gamma Exposure Rate Measurements

In each 9-m^2 cell (in the fenced-in storage yard) and each 36-m^2 cell (in the surrounding area), a gamma exposure rate measurement was made 1 m from the surface. The particular location in each cell was chosen randomly, and identified on a map. A tripod was used to support a 1" x 1" NaI crystal coupled to a photomultiplier tube and fed to a Ludlum 2220-ESG scaler, at 1 m from the ground. In each cell, a 1-min. count was collected and converted to μR/h. The measurement location and exposure rate were recorded in tabular form. About 226 1-min. measurements were acquired.

3.5 Surface Soil Samples

If a gamma exposure rate measurement indicated radioactive contamination, a 2-lb surface soil sample (no greater than 3" deep) was collected from that spot. Sample locations were identified and marked on a sample bag. Each sample was transferred to a bread pan for drying in an oven. When dry, each sample was stirred, then split into a 450-ml sample and a 2-g sample. Each 450-ml sample was placed in a Marinelli beaker for counting by gamma spectrometry. Each 2-g sample was ground with a mortar and pestle, placed in a 2" diameter aluminum planchet, and then counted for gross alpha/beta activity.

4.0 STATISTICS

4.1 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 variability 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 or mass can be determined. The number of particles, x, emitted by a radiation source in time, T, obeys the Poisson distribution:

$$P_X = \frac{m^x e^{-m}}{x!} \qquad \text{(Eq. 4-1)}$$

where m is the average number of emissions in that time. x is what we measure each time an area or sample is surveyed. The standard deviation is the square root of the average squared deviation of x from its mean, m. For the Poisson distribution, the standard deviation is given by:

$$s = \sqrt{x}\ , \qquad \text{(Eq. 4-2)}$$

the square root of the counts observed, $(x = \overline{x} = m)$. Since background radiation is always inherent in a given sample measurement, propagation of errors tells us that the total standard deviation is:

$$s = \frac{\sqrt{C + B}}{T} \qquad \text{(Eq. 4-3)}$$

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 the sample and background counts. This Poisson distribution and standard deviation applies for single radiation measurements, of the discrete random variable, x, and is applicable only when the observation times are short compared with the half-life. This is the case for the site survey.

Because of the probabilistic nature of particles emitted by radioactive elements, repeated measurements of the average number of emissions per unit time shows a distribution approximated by the Gaussian (or normal) probability density function (pdf); this is known as the central limit theorem. This theorem holds for any random sample with finite standard deviation. 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. This Gaussian approximation is good when the number of samples collected is at least 30. Thus the number of occurrences of particular mean radiological 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 amount of radioactive material in an area, and any anomalous values can be identified.

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

$$g(x)dx = \frac{1}{(\sqrt{2\pi})\sigma} \exp\left(\frac{-(x-m)^2}{2\ \sigma^2}\right) dx \qquad \text{(Eq. 4-4)}$$

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

m = average, or mean of the population distribution

σ = standard deviation of the population distribution.

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

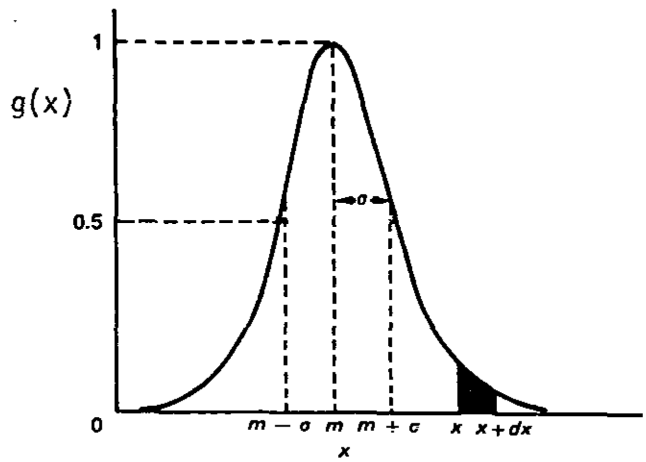


Figure 4.1 The Gaussian Probability Density Function

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

$$G(x) = \int_{\infty}^{X} g(x)dx \qquad \text{(Eq. 4-5)}$$

$$= P(x < X)$$

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

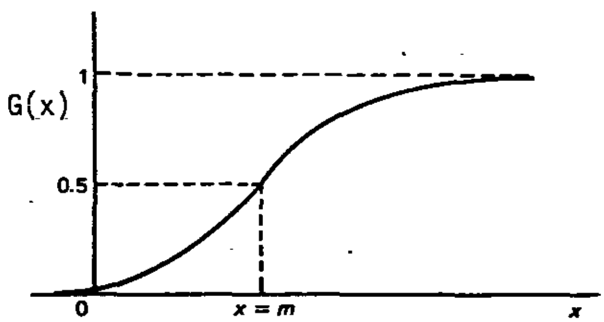


Figure 4.2 The Gaussian Cumulative Distribution Function

By plotting multiple measurements we make in the field; i.e. the average contamination values approximated by the Poisson distribution, as a cdf of the Gaussian distribution, we can identify whether the entire area is unacceptably contaminated, part of the area is contaminated more than the rest, or further radiological measurements are necessary. Furthermore, by making use of the Gaussian approximation, we can easily calculate the mean contamination value with its associated standard deviation, and apply inspection by variables techniques to either accept the area as clean or reject the area as contaminated.

This statistical summary presents fundamental principles used to reduce and analyze radiological measurement data from the site survey.

4.2 Sampling Inspection

4.2.1 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 in an area, it would be unacceptably time consuming and not cost effective to measure 100% of the population. 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 an area by making at least 30 measurements. This allows us to approximate a Gaussian distribution through the Central Limit Theorem. The entire area must have similar radiological characteristics and physical attributes. In acceptance inspection by variables, the result is recorded numerically and is not treated as a Boolean statistic, so fewer areas need to be inspected for a given degree of accuracy in judging a lot's acceptability.

4.2.2 By Attributes

By contrast, in acceptance inspection by attributes, the radiation measurement in a given area is recorded 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. More areas need to be inspected to obtain the same level of accuracy using this method. Consequently, we use inspection by variables.

4.3 Sampling Inspection by Variables

4.3.1 Calculated Statistics of the Gaussian Distribution

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

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

The sample mean is given by:

$$\overline{x} = \frac{\sum_{i=1}^{n} x_i}{n} \qquad \text{(Eq. 4-6)}$$

where: x_i = individual measurement values
n = number of measurement values

The standard deviation, s is given by:

$$s = \sqrt{\frac{\sum_{i=1}^{n} (x_i - \bar{x})^2}{n-1}} \qquad \text{(Eq. 4-7)}$$

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 some chosen limiting value of percent defective in a lot. Associated with the LTPD is a parameter referred to as consumer's risk (β), the risk or probability of accepting a lot with a percentage of defective items equal to the LTPD. It has been standard practice to assign a value of 0.10 for consumer's risk (β). 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 (Reference 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, U) 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 4-8.

Figure 4.3 demonstrates this principle. The operating characteristics curve of a Gaussian sample distribution shows the principles of consumer's and producer's risk, LTPD (or RQL), and acceptable quality level, (AQL). The criteria for acceptance of a lot are presented in section 4.3.3.

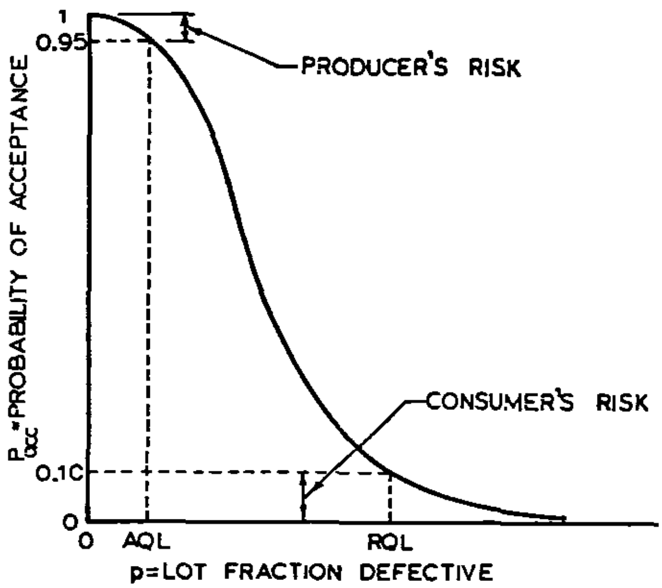


Figure 4.3 Operating Characteristics Curve

The value of k, and thus the value of x + ks, on which ultimately a decision is made whether the area is acceptably clean, is based on the conditions chosen for the test. k is calculated in accordance with the following equations, (Reference 8):

$$k = \frac{K_2 + \sqrt{K_2^2 - ab}}{a} \; ; \; a = 1 - \frac{K_\beta^2}{2(n-1)} \; ; \; b = K_2^2 - \frac{K_\beta^2}{n} \qquad \text{(Eq. 4-8)}$$

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. 0.10 (from tables, K_β = 1.282)

n = number of samples

As mentioned previously, 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, $K_\beta = K_2 = 1.282$.

Simply by coincidence, the coefficients $K\beta$ 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.

4.3.2 Graphical Display of Gaussian Distribution

When the cdf G(x), the integral of the Gaussian pdf, (Eq. 4-4), 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 straight line on probability-grade paper, when the sample lot contamination is normally distributed. Figure 4.4 shows this output.

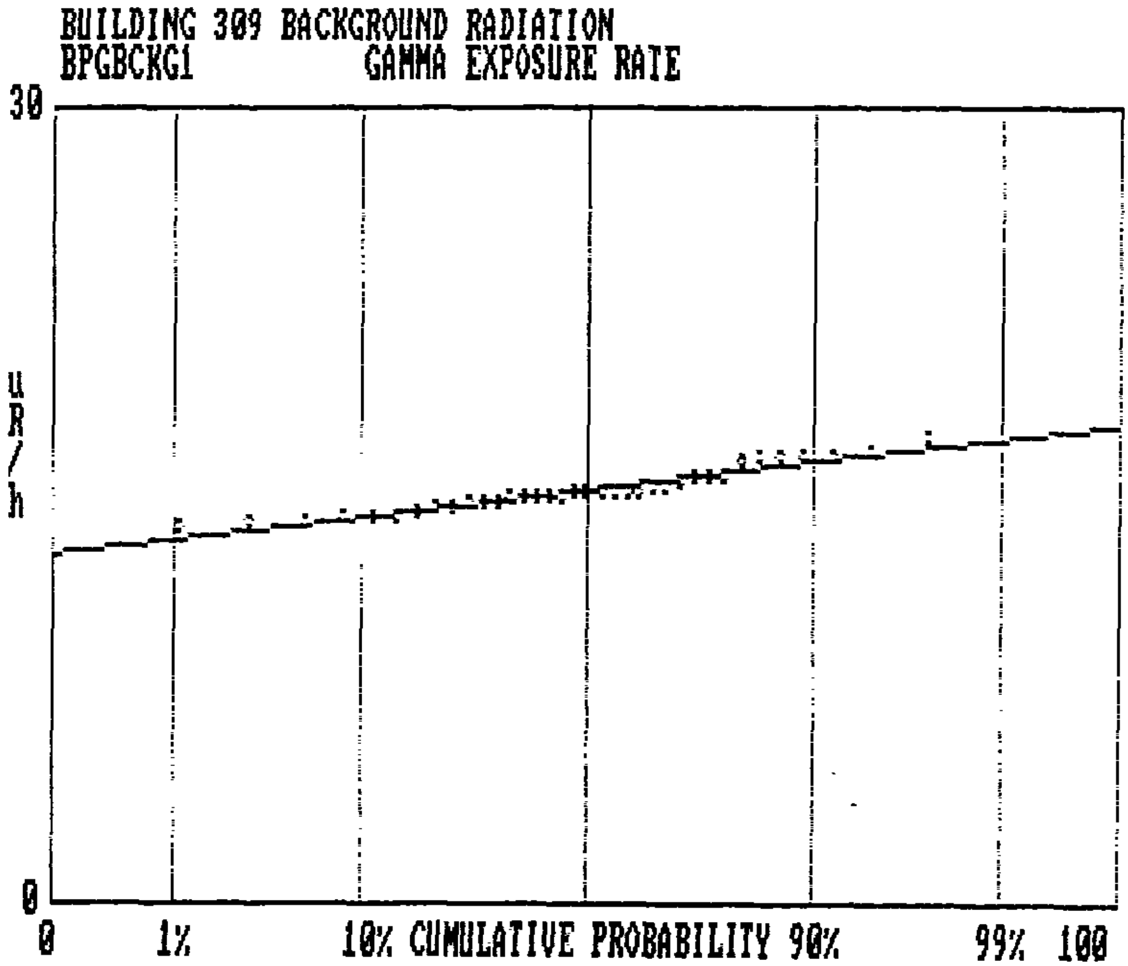


Figure 4.4 Gaussian cdf Plotted on Probability-Grade Paper

The power of this graphical display is that it permits identification of values with significantly greater contamination than expected for that lot. 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 are easily observable in the plot, but $\bar{x}$ + ks may still show acceptability. Upon further inspection and analysis, these graphical displays are used to show contamination level differences between areas or structures in a sample lot. The power of the fitted Gaussian graphical display is important in assessing significant variations in the contamination levels within sample lots.

4.3.3 Acceptance Criteria for an Uncontaminated Area

Once the test statistic, $\bar{x}$ + ks, is calculated and the Gaussian cdf probability plot is generated, a decision is made as to the extent of contamination in the area. Is the area clean? Is part of the area contaminated? Is the entire area contaminated? Are additional measurements necessary to make a determination?

First, the Gaussian distribution will identify significant variations in the radiological measurements. The sample output, if it represents the entire area well, should approximate a straight line. Measurements made which represent radiological conditions in a separate population from the one assumed, are easily observable as severe deviations in the straight line. The location of these anomalous measurements can be determined and subsequent follow-up is applied.

Second, the test statistic, $\bar{x}$ + ks, is calculated for the distribution. 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 Characterization Level, 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 Reinspection Level. If any single measured value, x, exceeds 90% of the limit, investigate the source of occurrence. This is the Investigation Level. This was presented in section 3.1.

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.

5.0 ANALYTICAL TECHNIQUES

Statistical methods presented in Section 4.0 were used to judge whether a sampling area is slightly contaminated, contaminated above acceptance limits, or whether additional investigation is required. That decision is based on several radiological measurements. For interior surfaces and selected special building features, these radiological measurements were:

1) Direct alpha and beta radiation; and
2) Removable alpha and beta contamination.

For exterior locations and surrounding areas, ambient gamma exposure rate measurements were made. Locations showing elevated gamma exposure rates were selectively sampled and analyzed for gross alpha/beta radioactivity and for qualification and quantification of detectable gamma-emitting radionuclides. Appendix C is a listing of gamma-emitting radionuclides.

Analytical techniques used to acquire, evaluate, and interpret these radiological measurements are presented in detail in this section. This includes systematic calibration corrections, background radiation determinations, alpha absorption corrections in soil samples, evaluation of computer-generated gamma spectrometry output, and computerized data analysis through inspection by variables.

5.1 Data Acquisition

In each designated square grid inside T064, total and removable alpha/beta contamination was measured. Each square grid was outlined and marked with its coordinates. The exact location within that square grid where the samples were collected was left to the surveyor's judgement: it was to be the area that, in his judgement, was most likely to have retained the greatest amount of contamination in that square grid. This decision is based on surface discoloration, debris, crevices or cracks. Use of a predetermined grid with discretion for exact location provides a uniform survey biased towards the high end of the distribution. Locations of noticeably greater radioactivity were always noted. Surrounding locations were then surveyed. In each designated square grid outside T064, ambient gamma exposure rate was measured.

5.2 Data Reduction Software Program

Each radiological measurement characteristic data value was input into SMART SPREADSHEET. This is an off-the-shelf computer software package which allows multiple computations to be performed on raw data values. Columns were established to calculate the alpha/beta total-average, maximum, and removable contamination per 1 m^2 in dpm/100 cm^2; and surface ambient gamma exposure rate in μR/h. The standard deviation of each measurement was also calculated. Software was developed in a program language called Quick Basic to read data from a SMART file into a graphics program which plots radiological measurements against a Gaussian cdf. For convenience, the distribution function, G(x) is plotted as the abscissa (probability grades), and x, the measurement value, is plotted as the ordinate (linear grades), see Figure 4.4.

Input for this data reduction was, for inside measurements:

1. Room number;
2. Grid location; ex. N(1,3), north wall, grid 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 smear (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. Alpha survey instrument background (5 min.), efficiency factor (dpm/cpm), and area factor;
10. Alpha gas-proportional detector background (5 min.) and efficiency factor (dpm/cpm);
11. Beta survey instrument background (5 min.), efficiency factor (dpm/cpm), and area factor;
12. Beta gas-proportional detector background (5 min.) and efficiency factor (dpm/cpm).

Output for Gaussian Plots of inside measurements:

1. Alpha total activity averaged over 1 m^2 with standard deviation (dpm/100 cm^2);
2. Alpha maximum activity and standard deviation (dpm/100 cm^2), only if observed;
3. Alpha removable activity and standard deviation (dpm/100 cm^2);
4. Beta total activity averaged over 1 m^2 with standard deviation (dpm/100 cm^2);
5. Beta maximum activity and standard deviation (dpm/100 cm^2), only if observed;
6. Beta removable activity and standard deviation (dpm/100 cm^2).

Input for data reduction of outside measurements was:

1. Ambient gamma exposure rate (counts in 1 min.; cpm);

2. Gamma survey instrument background (1 min.), and efficiency factor (μR/h/cpm).

Output for Gaussian plots of outside measurements:

1. Ambient gamma exposure rate and standard deviation (μR/h).

5.3 Data Analysis

An arithmetic mean and standard deviation of the radiological measurement values is calculated for each data set. The test statistic, x + ks, based on a consumer's risk of acceptance of 0.10 at 10% defective, is also calculated for each distribution. The acceptance criteria presented in Section 4.3.3 is applied to each sampling distribution.

From the plot of measurement values vs. cumulative probability, the mean radiological value of the lot is the point on the ordinate axis where the distribution intersects the 50% cumulative probability. In test cases where an acceptance limit has been established for acceptably clean, a vertical line is plotted corresponding to the test statistic x + ks. The figures display the results on an expanded scale so that the variations in the data can be seen in detail.

5.4 Direct Alpha/Beta Contamination Measurements

Direct alpha/beta contamination measurements were made by using Ludlum model 2220-ESG portable scalers to detect pulses from a Ludlum 43-1 alpha probe and Ludlum 44-9 beta probe, respectively.

5.4.1 Instrument Calibration

Each detector was calibrated three times daily. The alpha detector was calibrated with Th-230; the beta detector with Tc-99. Background levels were determined specifically for each survey grid. This technique corrected for major beta fluctuations in background due to storage of depleted and normal uranium.

5.4.2 Data Acquisition and Reduction

Each location where a measurement was made was identified on a map and in matrix notation. The gross number of alpha and beta counts recorded in 5 min. along with the matrix notation location was input into SMART SPREADSHEET. Columns were established to calculate total-average alpha and beta surface activity and the standard deviation (in dpm/100 cm^2) according to equations 5-1 and 5-2. Conversion from gross counts observed to dpm/100 cm^2 is given by:

$$SA = \frac{(C - B)}{5} \frac{(EF)(100)}{A} \qquad \text{(Eq. 5-1)}$$

where:

SA	=	surface 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)
EF	=	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/100 cm^2 is given by:

$$s = \frac{\sqrt{C + B}\ (100)(EF)}{(5)(A)} \qquad \text{(Eq. 5-2)}$$

5.4.3 Data Analysis

Total-average alpha/beta radioactivity in dpm/100 cm^2 per square meter were plotted, in order of magnitude from left to right, against cumulative probability, as in Figure 4.4. The test statistic, $\bar{x}$ + ks, was also calculated for the lot. $\bar{x}$ + ks is compared against the acceptance limits presented in section 3.1. Criteria for accepting the area as uncontaminated is presented in section 4.3.3.

If the measurements taken are represented by a Gaussian distribution, the data will be arranged in a straight line. If large breaks or changes in slope are observed in the distribution, then some specific area is contaminated to a greater level.

If large negative numbers exist, then the background radiation level is significantly greater than the area under study. The data point should be checked and the worth of keeping that measurement value should be assessed. If background levels are so high that contamination values less than a few hundred dpm/100 cm^2 are calculated, then the data point is meaningless.

5.5 Removable Alpha/Beta Contamination Measurements

A 100 cm^2 area of each square meter surveyed for fixed alpha/beta contamination was sampled for removable alpha/beta contamination. Each smear sample was placed in a gas-flow proportional counter for analysis.

5.5.1 Instrument Calibration

The Canberra Model 2201 gas-flow proportional counter was calibrated twice daily. Alpha efficiencies were determined by using a Th-230 calibration source. Beta efficiencies were determined by using a Tc-99 calibration source. A "clean" smear-paper was used to determine background radiation levels.

5.5.2 Data Acquisition and Reduction

Gross alpha and beta counts for each sample location were entered into SMART SPREADSHEET. Columns were established for input of instrument efficiency and background. Removable surface activity is converted to dpm/100 cm^2 by:

$$SA = \frac{(C - B)(EF)}{5} \quad (dpm/100\ cm^2) \quad \text{(Eq. 5-3)}$$

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

The standard deviation of this measurement is:

$$s = \frac{\sqrt{C + B}\ (EF)}{5} \quad (dpm/100\ cm^2) \quad \text{(Eq. 5-4)}$$

5.5.3 Data Analysis

Removable alpha/beta radioactivity in dpm/100 cm^2 per square meter were plotted, in order of magnitude from left to right, against cumulative probability, as in Figure 4.4. The same analytical criteria apply here as that presented in Section 5.4.3.

5.6 Ambient Gamma Exposure Rate

Measurements of ambient gamma exposure rate were made by use of a 1" x 1" NaI scintillation crystal coupled to a Ludlum Model 2220-ESG portable scaler, (Appendix A.3). This device was mounted on a tripod so that the sensitive crystal was 1 meter from the ground. The detector is nearly equally sensitive in all directions, i.e. 4 π geometry, and can detect variations in exposure rate down to one-one hundredth of a μR/h, using the digital scaler for a 1-min count time.

5.6.1 Instrument Calibration

This detector is calibrated quarterly by the calibration laboratory using Cs-137 as the calibration source. A voltage plateau is plotted and the voltage is set at a nominal 800 V. The detector is placed on a calibration range and readings taken at 5, 2, 1, 0.9, 0.5, 0.4, 0.3, and 0.2 mR/hr. A detector efficiency plot as a function of exposure rate is generated in this regard, (μR/h/cpm).

Because of an exposure rate-dependent effect and because our calibration range does not read less than 200 μR/h, this instrument was cross-calibrated against a Reuter Stokes High Pressure Ion Chamber (HPIC). Count rates were converted to exposure rates by the relationship that about 215 cpm = 1 μR/h, at background exposure rates. This calibration was performed several times.

Instrument response was checked three times a day using a Ra-226 source. The source was placed 1 ft from the detector and counted for 1 min. If the scaler reading fell within ± 5% of the nominal value, then the instrument was qualified as operable for the day, under the calibration conditions previously described. Recalibration was never necessary.

5.6.2 Data Acquisition and Reduction

Each location where a gamma measurement was made was identified on a map and in matrix notation. The gross number of counts recorded in 1 min. along with the matrix notation location was input into SMART SPREADSHEET. Columns were established to calculate the total exposure rate (μR/h) and its standard deviation according to the equations 5-5 and 5-6. Gamma scintillations produced by a NaI detector were converted from gross counts to exposure rate (μR/h) by:

$$R = \frac{(C) * (EF)}{1 \text{ min.}} \qquad \text{(Eq. 5-5)}$$

where R = exposure rate (μR/h)

C = gross counts in 1 min. (cpm)

EF = efficiency factor (0.0047 μR/h/cpm) based on cross calibration with HPIC.

Background was not subtracted in this case because the range of measurable natural background exposure rates approaches the NRC acceptance limit of 5 μR/h. Rather it was more meaningful to measure an area where no radioactive materials were ever handled, and then compare that gross distribution with the one under study.

The standard deviation, s, of a single measurement then becomes by Eq. 4-3:

$$s = \frac{\sqrt{C} \ *(EF)}{1 \text{ min.}} \qquad \text{(Eq. 5-6)}$$

5.6.3 Data Analysis

Total exposure rates in μR/h were plotted, in order of magnitude from left to right, against the cumulative probability, as in Figure 4.4.

Both the NRC and DOE criteria for acceptance as unrestricted use are given in μR/h above background, 5 and 20, respectively. During the survey we observed significant deviations in natural background radiation as a function of landscape geometry. For example, when the detector is placed near a large sandstone outcropping, the exposure rate may increase by almost 4 μR/h. This increase is due to primordial radionuclides in the sandstone, and because the source geometry has changed from 2 π to maybe, 3 π steradians.

The best solution for evaluating the potential or existence of residual contamination in an area where the radiation field varies naturally by swings as large as the acceptance limit, is to compare total exposure rates in different areas. The background, B, was not subtracted from any of the ambient gamma exposure rates.

T064 distributions of ambient exposure rate measurements are compared against three independent sampling areas of similar geologic characteristics. In these other areas, no radioactive materials were ever used, handled, stored, or disposed. These distributions represent natural ambient gamma radiation levels in this location. Measurements were taken on flat and rugged terrain, with Chico Formation sandstone, similar to conditions surrounding T064.

5.7 Surface Soil Samples

If radioactive contamination was detected from performance of ambient gamma exposure rate measurements, soil samples were collected from the general area to qualify and quantify the radioactivity. A soil sample was characterized for gross alpha/beta radioactivity and qualified by gamma spectrometry. A 2-lb surface soil sample was collected, and then dried in an oven after large chunks and rocks were removed. The sample was homogenized, then split into 450-ml and 2-g samples. The 2-g sample was crushed using a mortar and pestle, then placed in an aluminum planchet for alpha/-beta counting. The 450-ml sample was placed in a Marinelli beaker for gamma spectrometry.

5.7.1 Gross Alpha/Beta Analysis

Once the 2-g sample was finely ground and placed on a 2" aluminum planchet, it was placed on the sample loading magazine of the Canberra proportional alpha/beta counter, (Appendix A.2). Each sample was spread uniformly over the entire area of the planchet.

5.7.1.1 Instrument Calibration

When counting soil samples for radioactivity, it is very important that the geometry from sample to sample remain constant. Proper corrections must be made for detector background, and efficiency. Before any of the soil samples were analyzed, a precise determination was made of the background, the degree of alpha/beta absorption in soil, and the detector efficiency.

Detector background for "false positive" alpha/beta counts was determined by using processed sea sand. All primordial radioactive isotopes have been removed from this silica material. A 2-g sample was placed on a planchet and counted at least 10 times for 30 min. each to determine the alpha and beta background count rates. The average background determined for this instrument was 4.5 ± 1.8 alpha counts in 30 min. and 53.4 ± 11.1 beta counts in 30 min. for processed sea sand.

Alpha efficiency (detector plus self-absorption) was determined by using a 2-g soil sample spiked with 93% enriched uranium. The standard was spiked with 40 pCi/g-alpha activity. Natural primordial radioactivity in the standard contributed an additional 25.85 pCi/g alpha activity. The total alpha activity in the soil was therefore 65.85 pCi/g. By counting the standard several times for 30 min. each, an alpha efficiency factor of 32.45 pCi/g·cpm was calculated.

Beta efficiency was determined by using a 2-g KCl beta standard. At 0.00117% abundant, K-40 produces 1750 beta disintegrations per minute per 2-g sample of KCl. By counting the standard several times for 30 min. each, a beta efficiency factor of 1.44 pCi/g·cpm was calculated.

The efficiency factor calculations and background measurements were used throughout the duration of the analysis. An NBS traceable Th-230 calibration source was used twice daily as a check source. If, on a day to day basis, the check source alpha/beta count exceeded ± 5% of the nominal value, the instrument would be checked and recalibrated using the soil standards. This recalibration was never necessary.

5.7.1.2 Data Reduction and Analysis

Gross alpha/beta counts were collected for each soil sample, 30 min. each. Gross activities in pCi/g were calculated using the backgrounds and efficiency factors mentioned in Section 5.7.1.1. This radioactivity concentration calculation is given by the following expression:

$$A_C = \frac{(C - B)}{30 \text{ min.}} EF \qquad (\text{pCi/g}) \qquad (\text{Eq. 5-7})$$

where A_C = Activity Concentration (pCi/g)
C = Gross Counts (alpha or beta)
B = Background Counts (alpha or beta)
EF = Efficiency Factor (32.45 alpha - pCi/g·cpm)
(1.44 beta - pCi/g·cpm)
30 min. = Count Time

The standard deviation of this measurement is:

$$s = \frac{\sqrt{C + B}}{30 \text{ min.}} (EF) \qquad (\text{pCi/g}) \qquad (\text{Eq. 5-8})$$

Sample activities are presented in tabular form.

5.7.2 Gamma Spectrometry

Each 450-ml soil sample was placed in a Marinelli beaker and counted for 30 min. on a Canberra Series 80 gamma spectrometer, described in Appendix A.1. This analytical tool measures U-238, U-235, Th-232 and K-40 radioactivity, all of which are naturally occurring. It will also detect characteristic fission and activation products such as Cs-137, Co-60, and Eu-152. The nuclide library is shown in Appendix C.

5.7.2.1 Instrument Calibration

The instrument is calibrated routinely for energy and efficiency using a Marinelli Beaker Standard Source (MBSS), described in Appendix A.1. This calibration process is performed over a wide energy range: Cd-109 (88.03 keV), Co-57 (122.06 keV), Ce-139 (165,85 keV), Hg-203 (661.65 keV), Y-88 (898.02), Co-60 (1173.21 and 1332.47 keV), Y-88 (1836.04 keV). The multichannel analyzer automatically fits efficiency and energy-to-channel number curves for energies which are not included in the calibration spectrum. These calibrations are performed in accordance with the procedures prescribed by the Canberra Operator's Manual.

It is particularly important when performing gamma spectrometry analysis, that the sample geometry be identical to the standard geometry. Efficiency is a function of geometry, and varies significantly in this case.

5.7.2.2 Data Reduction and Analysis

The multi-channel analyzer is programmable; for any unknown sample, it will calculate the activity in μCi of any isotope it identifies corresponding to the signature library listed in Appendix C. The percent error in activity is also calculated based on the number of counts collected under the peak. Although the machine is quite good, a great deal of prudence must be used when evaluating the output.

Because soil samples were only acquired in areas of "greater than background" ambient radiation, extensive data reduction and interpretation techniques were not needed to satisfy this test. Any major gamma emitting isotopes are readily distinguishable by the MCA. If by observation, it is determined that uranium daughter products are present above natural background concentrations, further interpretation is necessary. In which case the isotopes presented in Table 5.2 are used to estimate U-238, U-235, and Th-232 activity.

Table 5.1 Probable Gamma Energies for Determining Soil Radioactivity

U-238 Chain (Primordial)	Th-232 Chain (Primordial)
Th-234 (93 keV)*	Ac-228 (908 keV)
Ra-226 (186 keV)**	Ac-228 (338 keV)
Pb-214 (295 keV)	Ac-228 (960 keV)
Pb-214 (352 keV)	Th-228 (84 keV)*
Bi-214 (609 keV)	Ra-224 (241 keV)***
Bi-214 (1120 keV)*	Pb-212 (239 keV)***
Bi-214 (1764 keV)*	Pb-212 (300 keV)*
	Bi-212 (727 keV)*
	Bi-212 (1620 keV)*
	Tl-208 (511 keV)*
	Tl-208 (583 keV)
	Tl-208 (860 keV)*
U-235 Chain (Primordial)	**Fission Products**
U-235 (93 keV)*	Cs-137 (661 keV)
U-235 (185.6 keV)**	
U-235 (205.2 keV)*	
K-40 (Primordial)	**Activation Products**
K-40 (1460 keV)	Eu-152 (several energies)
	Co-60 (1117 keV)
	(1332 keV)
Be-7 (Cosmogonic)	
Be-7 (478 keV)****	

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*	Not evident because of low gamma yield (rarely seen)
**	Peak overlaps from Ra-226 and U-235
***	Peak overlaps from Ra-224 and Pb-212
****	Formed in atmosphere - not normally found in soil

Estimates of radionuclide content in each sample were derived based on corrections for:

1) Multi-Channel Analyzer (MCA) output; and
2) Daughter Product decay for U-238, and Th-232.

Corrections to MCA calculated activities were made in two cases. First, because of peak overlap at 185-186 keV from Ra-226 and U-235, an estimate of each isotope had to be derived. Assuming that Ra-226 is in equilibrium with U-238 and that U-235 is 0.7% by weight of U-238, it can be shown that the true Ra-226 activity is equal to the Ra-226 MCA calculated activity multiplied by 0.5525. The true U-235 activity is then equal to the U-235 MCA calculated activity multiplied by 0.446. If enriched uranium is present in the sample, these corrected values will show up as large deviations.

Second, because of peak overlap at 239-240 keV from Ra-224 and Pb-212, estimates for true activity had to be derived. The true Pb-212 activity is equal to the MCA calculated activity multiplied by 0.91. Since Ra-224 and Pb-212 are in equilibrium, their activities are equal.

U-238 activity is calculated by:

$$A_{U-238} = \frac{\sum_{i=1}^{n} A_i}{n} \quad (pCi/g) \qquad \text{(Eq. 5-9)}$$

where A_i = all non-zero MCA calculated and corrected activities from U-238 daughter products listed in Table 5.2. (All daughters in equilibrium, branching ratios equal 100%)

n = number of non-zero activity values

pCi/g = appropriate conversion factors and sample mass used to obtain this unit.

Th-232 activity is calculated by:

$$A_{Th-232} = \frac{\sum_{i=1}^{n} A_i}{n} + \frac{\sum_{1}^{3} A_{Tl-208}}{3 * 0.36} \quad (pCi/g) \qquad \text{(Eq. 5-10)}$$

where A_i = all non-zero calculated and corrected activities from Th-232 daughter products listed in Table 5.2 (all daughters in equilibrium, branching ratios equal 100%)

n = number of non-zero activity values

A_{Tl-208} = Three identifiable gamma energies from Tl-208 (in equilibrium, branching ratio from Bi-212 is 36%).

Ratios of U-238 to Th-232 activity concentrations (pCi/g) can be compared against what would be expected in "natural" soil. From the CRC handbook, the activity concentration ratio U-238:Th-232 should be about 1.0. Cs-137, Co-60, and K-40 are easily identifiable radionuclides by gamma spectrometry; they each emit a single characteristic gamma-ray, (see Table 5.2).

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 9 m^2 in area, then grid the area by square meters as appropriate. Designate each square meter in matrix notation with floor location (1,1) being the northwestern most square in a room. Wall location (1,1) is the lower left square meter as the surveyor views the wall.

6.1.1 Floor

Select 1 m^2 out of each 9 m^2 on which to perform the survey. If a surface is less than 9 m^2 in area, then survey 1 m^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 1 m^2 out of 9 m^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

Because the ceiling is greater than 10 feet in height and not readily accessible, do not survey.

6.1.4 Special Structural Features

Gridding is not necessary. Survey randomly for detectable alpha/-beta contamination. Smear the item for analysis of removable contamination. Special features include accessible coving, wall-to-floor joints, light fixtures, vertical I-beam supports, fire extinguishers, cracks, filters, and miscellaneous items.

6.1.5 Fenced-in Storage Yard

Select 1 m^2 out of each 9 m^2 on which to perform an ambient gamma exposure rate measurement. Collect soil samples in areas of increased gamma radiation and analyze by gamma spectrometry.

6.1.6 Surrounding 2-Acre Area

Select 1 m^2 out of each 36 m^2 on which to perform an ambient gamma exposure rate measurement. Collect soil samples in areas of increased gamma radiation and analyze by gamma spectrometry.

6.2 Calibration and Instrument Checks

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

Portable Ludlum 2220-ESG 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, 800V gamma).

3) Check threshold (140-190 alpha, 250-350 beta, 400 gamma).

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. for alpha and beta measurements. For ambient gamma exposure rate measurements, set time to 1 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 2π emission rate of the source (dpm) to the net count rate of the instrument. The radioactivity of the calibration sources is traceable to NBS. Similarly, use a Ra-226 check source located 1 ft from the NaI detector to check operability of the gamma instrument. The count rate should not vary by more than $\pm$ 5% from the initially established standard. The gamma calibration efficiency factor is determined by comparison against a Reuter Stokes High Pressure Ion Chamber (HPIC).

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 to determine the detector background for smear samples.

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 for smear samples.

4) Before a soil sample analysis is run, use uncontaminated planchets and take a couple 30 min. background counts of processed sea sand, 2 g each. Take and record 30-min counts of alpha and beta soil standards to determine an efficiency calibration. Alpha efficiency is determined by using a 2-g soil sample spiked with 93% enriched uranium, with total alpha emissions equivalent to 65.85 pCi/g. Beta efficiency is determined by using a 2-g KCl beta standard. A 2-g sample of KCl yields 1750 beta disintegrations per minute.

5) Use a Th-230 check source daily to ensure that the alpha/-beta count rates do not vary by more than ± 5% from the initially established standard.

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.

Gamma Spectrometer:

1) Check to make sure that the MCA has been calibrated for energy and efficiency.

2) If machine is not calibrated, refer to Canberra user's manual for proper calibration of device.

6.3 Radiological 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-ESG) set for a 5-min. count time, use an alpha probe (Ludlum Model 43-1) on one instrument and a beta probe (Ludlum 44-9) on another, then 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 3 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 spots are to be resurveyed later).

3) Because the beta background radiation level varies considerably from one location to another in T064 due to storage of depleted uranium, measure α/β background in each survey location. After a 1-m^2 grid has been surveyed, place a 1/4" piece of plywood on the floor. Then place the probes on top and collect a 1 min. count. This is the background radiation level for that area.

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

5) Enter the data into SMART 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) Determine background radiation in some manner as above, section 6.3.1 (3).

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

5) Enter the data into SMART 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 SMART spreadsheet.

6.3.4 Ambient Gamma Exposure Rate Measurements

1) Mount the detector on a tripod which supports the detector 1 meter from the ground.

2) Set the count time to 1 min. and take a measurement at each selected location for that length of time.

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

4) Enter the data into SMART spreadsheet.

6.3.5 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. Suspect areas include light fixtures, wall to floor joints, coving, beam supports, exhaust hood, horizontal surfaces, and miscellaneous equipment.

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 lengthwise.

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 20 cpm, 30 cpm, 100 cpm, etc., as applicable.

6) Smear the special structural features and analyze for removable radioactivity. Follow the procedure in section 6.3.3.

6.3.6 Measurements of Gross Alpha/Beta Soil Activity

1) After homogenizing a dried, 2-lb soil sample, take a few grams and place in a mortar. Using a pestle, grind the sample until a fine powder results. All big chunks should be removed, or broken down.

2) Take a 2" aluminum planchet, then place a 2-g soil sample evenly about its surface.

3) Place, in order of sampling location, each sample in the proportional counter sample magazine. Count each for 30 min.

4) Record the date, location and number of alpha and beta counts. Enter data with calibration numbers into SMART SPREADSHEET.

5) Count the Th-230 check source to ensure that the calibration efficiency and background factors are still applicable.

6.3.7 Gamma Spectrometry Measurements in Soil

1) After homogenizing a dried, 2-lb soil sample, take a 450-ml sample which has no large chunks, and place it in a Marinelli beaker. The soil should lay flat, 1 1/2" from the top of the beaker.

2) Place the beaker over the calibrated high purity germanium (HPGE) detector and collect counts for 30 min. Use the MCA to qualify and quantify radioactive material in the sample.

3) Evaluate and correct MCA calculated activities and reduce to units of pCi/g for each radionuclide of interest.

7.0 SURVEY RESULTS

The Building T064 radiological survey was performed using the survey plan previously described. Three sample lots were established to survey, analyze, and interpret radiological data: (1) the building interior; (2) fenced-in storage yard; and (3) surrounding two-acre area. Uniform 3-m square grids were set up inside to measure alpha/beta contamination. Uniform 3-m and 6-m square grids were set up outside to measure ambient gamma exposure rates. Radiological data for each area was statistically analyzed by sample lot. Analytical interpretation of each data set shows a few contaminated locations.

7.1 Statistical Results Format

Radiological data collected during this survey are displayed as Gaussian cumulative distribution functions in Figures 7.1 through 7.7. 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. Depending on the measurement type, an acceptance limit is used as the maximum ordinate value. In some cases, this convention is not applicable because an acceptance limit has not been set or is questionable.

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 for a one-sided test, respectively. The value of k used in the inspection test is very nearly 1.5 for each case; thus, the Test Statistic (TS) line will run perpendicular to the abscissa corresponding to about a 93.3% cumulative probability. The Test Statistic is $\bar{x} + ks$. The Gaussian distribution line must pass below the intersection of the "TS" line (about 93%) and the horizontal line showing the acceptance limit at that point in order to accept the lot as being noncontaminated.

"k" and thus the "TS" line increase as the number of samples in a lot decrease.

At the top left hand corner of the output is the file name of the data file for the 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, if applicable) or the smallest value calculated for a Gaussian fit. Negative numbers result when the measured value is less than background. 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. If an acceptance limit is applicable, 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.

In cases where an acceptance limit is not appropriate, for example, gamma exposure rate measurements, the four horizontal lines are not shown. Furthermore, a test statistic is not calculated because we were not testing the data against an acceptance limit. Since the variability in naturally-occurring ambient gamma exposure rates at SSFL is wide, background was not subtracted. In these cases, the mean is calculated and the shape of the distribution is observed to identify any areas of increased radioactivity.

7.2 Building Interior

Total-average alpha and beta measurements were made per square meter in a 3-m square grid. Similarly, removable alpha and beta measurements were made in each grid. 101 measurements of this type were made, 5 minutes each. These measurements were evaluated by analytical interpretation using Gaussian statistics. Miscellaneous features were surveyed for "indication only;" not a quantitative measurement.

7.2.1 Alpha/Beta Grid Measurements

Table 7.1 shows the results for 101 grid measurements taken of total-average alpha, removable alpha, total-average beta, and removable beta. No maximum "hot spots" were detected. The table shows three important statistics for each distribution: average value, maximum value, and the inspection test statistic (TS = $\overline{x}$ + ks). These results include 8 measurements taken on the floor of room 110 with the 9-in. vinyl tile removed. No variation in the results was observed between measurements made on tile vs no tile. Append D.1 lists the data used for this analysis, sorted on alpha activity then sorted on beta activity.

In Table 7.1 notice that there is a row for "revised" beta activity, and the number of measurements is 97 rather than 101. Four measurements were made in room 110, extremely close to the normal and depleted uranium. The beta background in this area was so high that the contamination measurement was useless; the background measured was much greater than the measurement itself. This resulted in four greatly negative numbers (on the order of -1000 dpm/100 cm^2). These points were deleted from the analysis to make the interpretation more meaningful. The average value, as expected, increased substantially while the inspection test statistic decreased because the data now follow a representative Gaussian distribution. Previously the distribution mean was offset by large negative numbers.

Figures 7.1 through 7.5 show the distributions for alpha/beta measurements made inside T064. Total-average alpha activity, Figure 7.1, is far below the acceptance limit of 5000 dpm/100 cm^2. Figure 7.2 is the same as Figure 7.1 except the ordinate scale has been expanded to show a variation in alpha activity observed on the ramps leading to rooms 110 and 114. Although far less than the acceptance limit, there is clearly two distinct Gaussian distributions. Appendix D.1 lists the data in order of alpha activities. The average alpha activity on those ramps is 51.1 $\pm$ 12.8 dpm/100 cm^2, much greater than the other measurements taken. A beta

Table 7.1 Survey Results for Building Interior

Measurement	Number of Locations	Average Value	Maximum Value	Inspection Test Statistic	Limit
Average alpha (dpm/100 cm^2)	101	10.5	74.7	38.8	5,000
Removable alpha (dpm/100 cm^2)	101	1.0	5.7	3.0	1,000
Average beta (dpm/100 cm^2)	101	167	1693	2351	5,000
Average beta (REVISED)* (dpm/100 cm^2)	97	388	1693	1138	5,000
Removable beta (dpm/100 cm^2)	101	4.4	12	8.6	1,000

* Corrected for large negative values due to large background in some areas.

correlation in this manner was not observed. This increased activity is probably attributed to naturally-occurring radionuclides concentrated in the concrete used for the ramps. Figure 7.3 shows that no removable alpha contamination is present.

Figure 7.4 shows the revised-corrected total-average beta activity distribution. Measurements follow a Gaussian distribution. No single measurement exceeded 50% of the acceptance limit. Figure 7.5 shows that no removable beta contamination is present.

Although some deviations were observed for total-average alpha activity, they occurred far below the acceptance limit. This activity is probably due to natural elements used in the concrete. The interior measurements indicate a "clean" area.

Figure 7.1 Total-Average Alpha Activity Inside T064

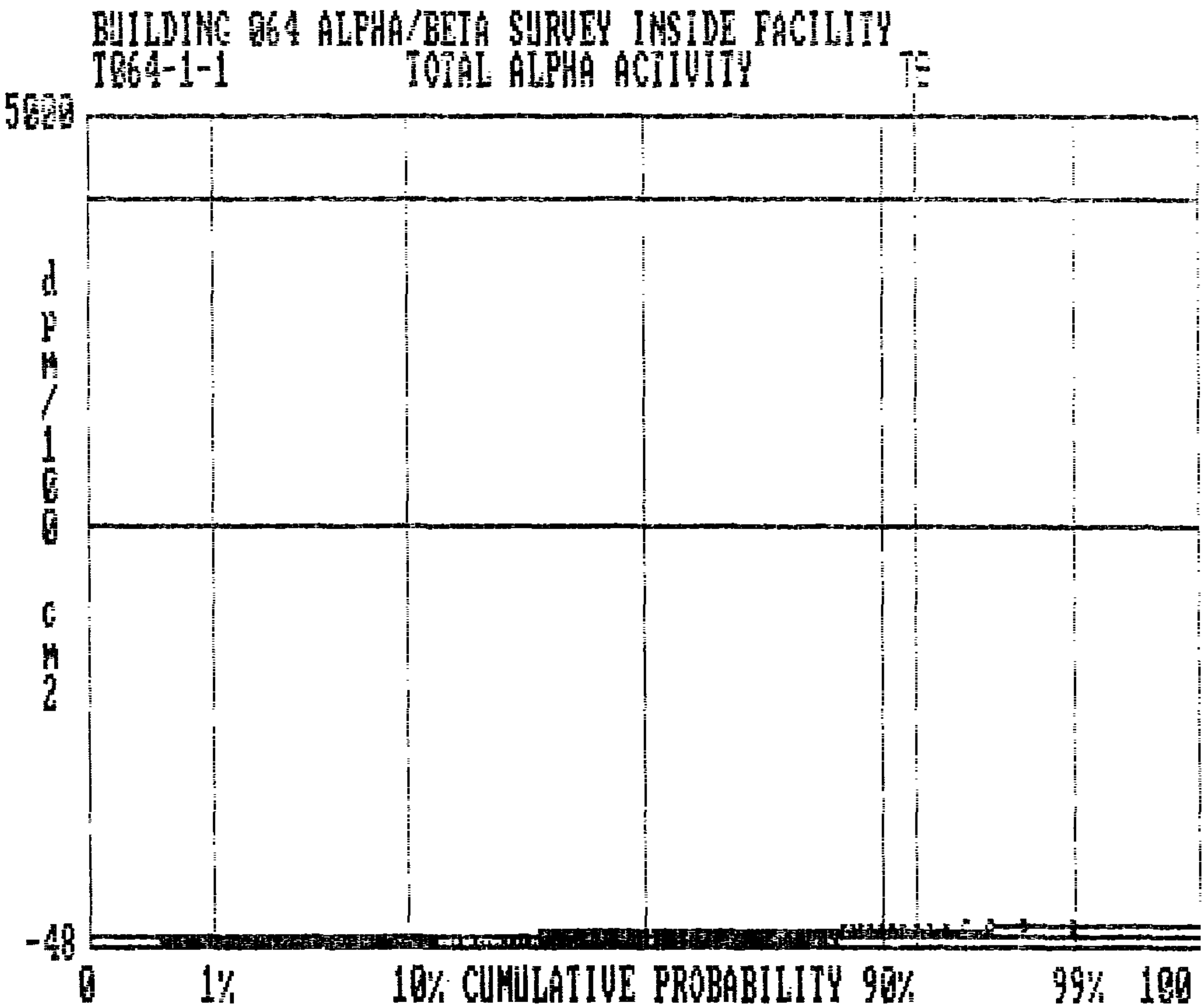


Figure 7.2 Total-Average Alpha Activity Inside T064 (Expanded Scale)

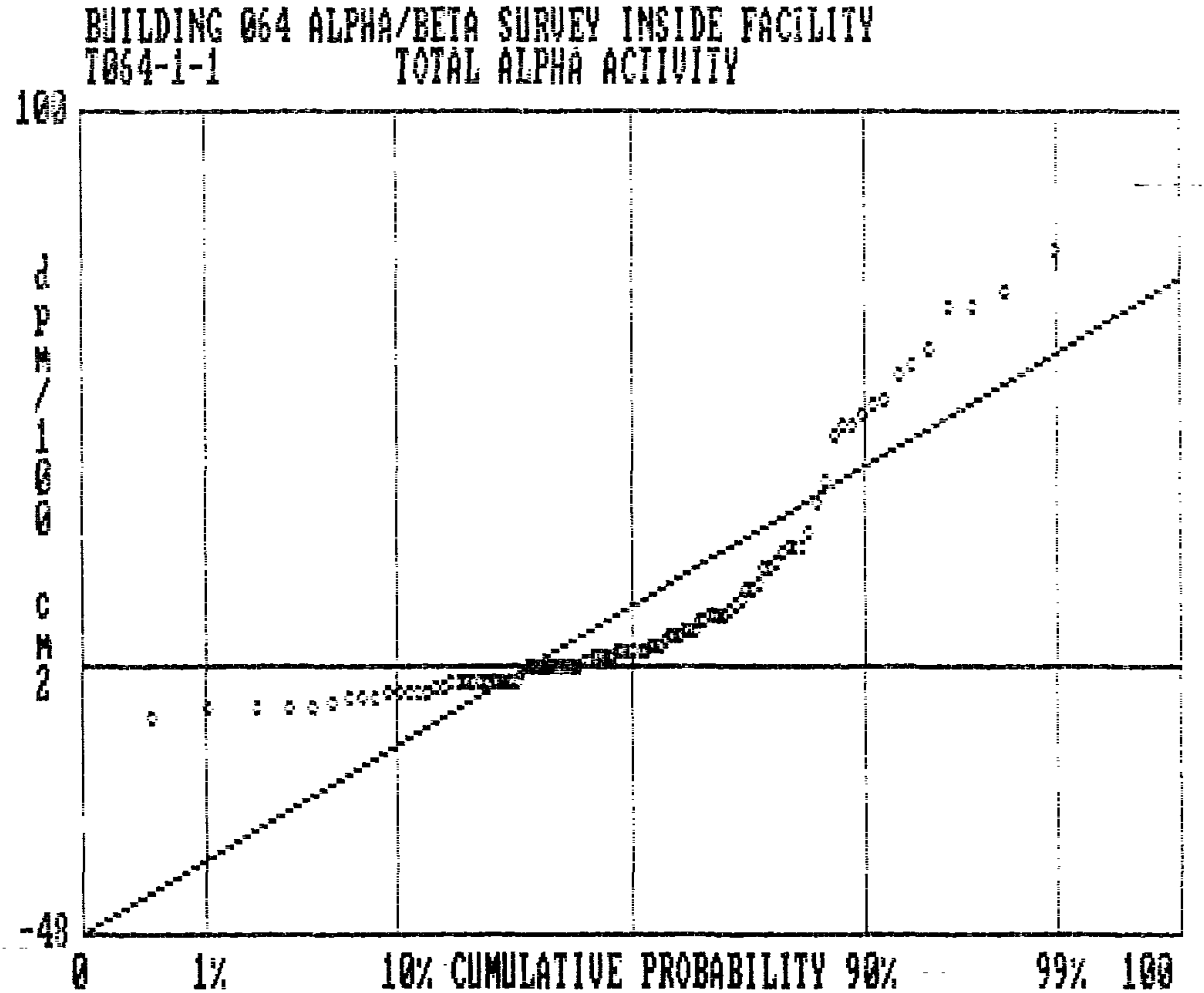


Figure 7.3 Removable Alpha Activity Inside T064

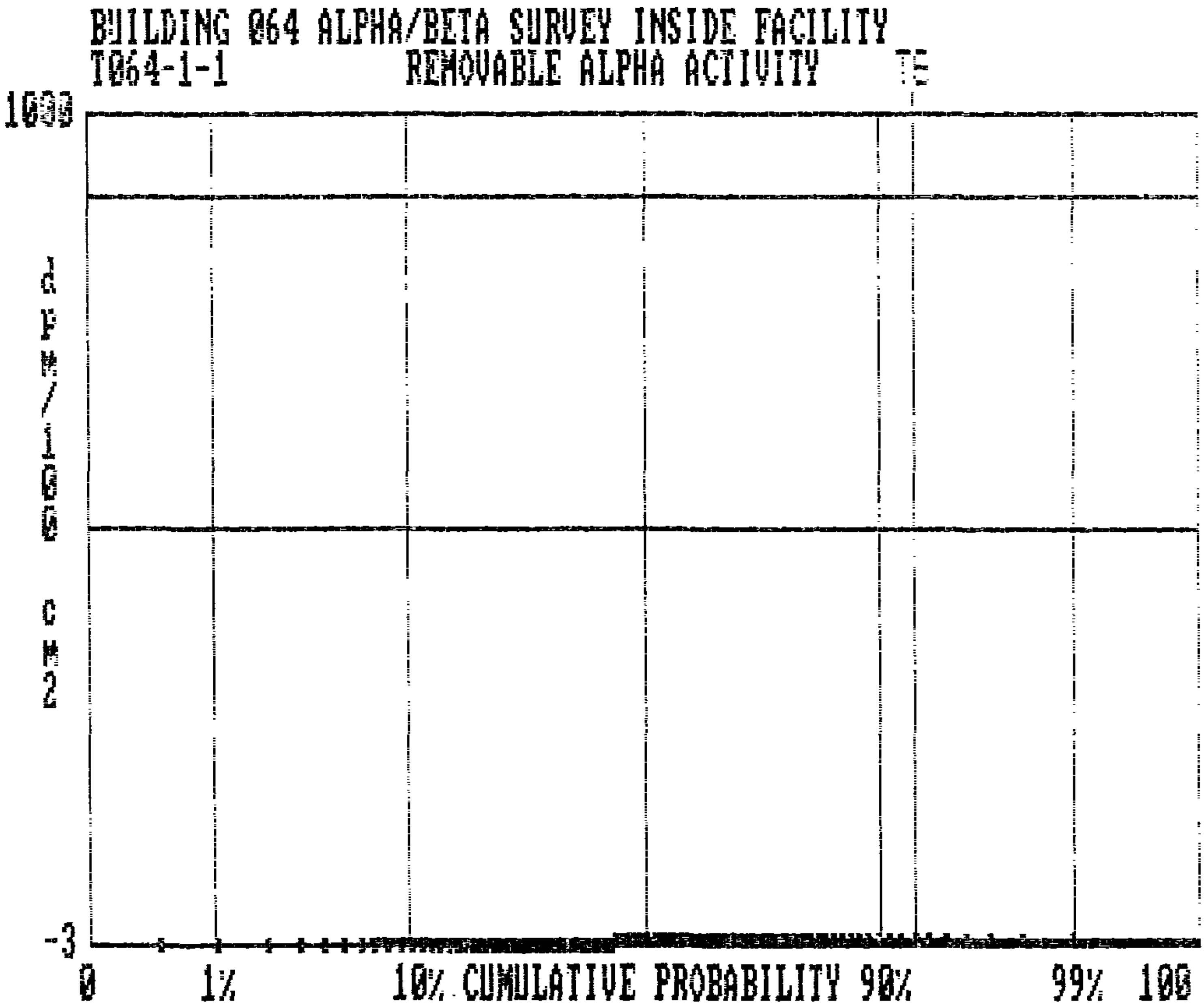
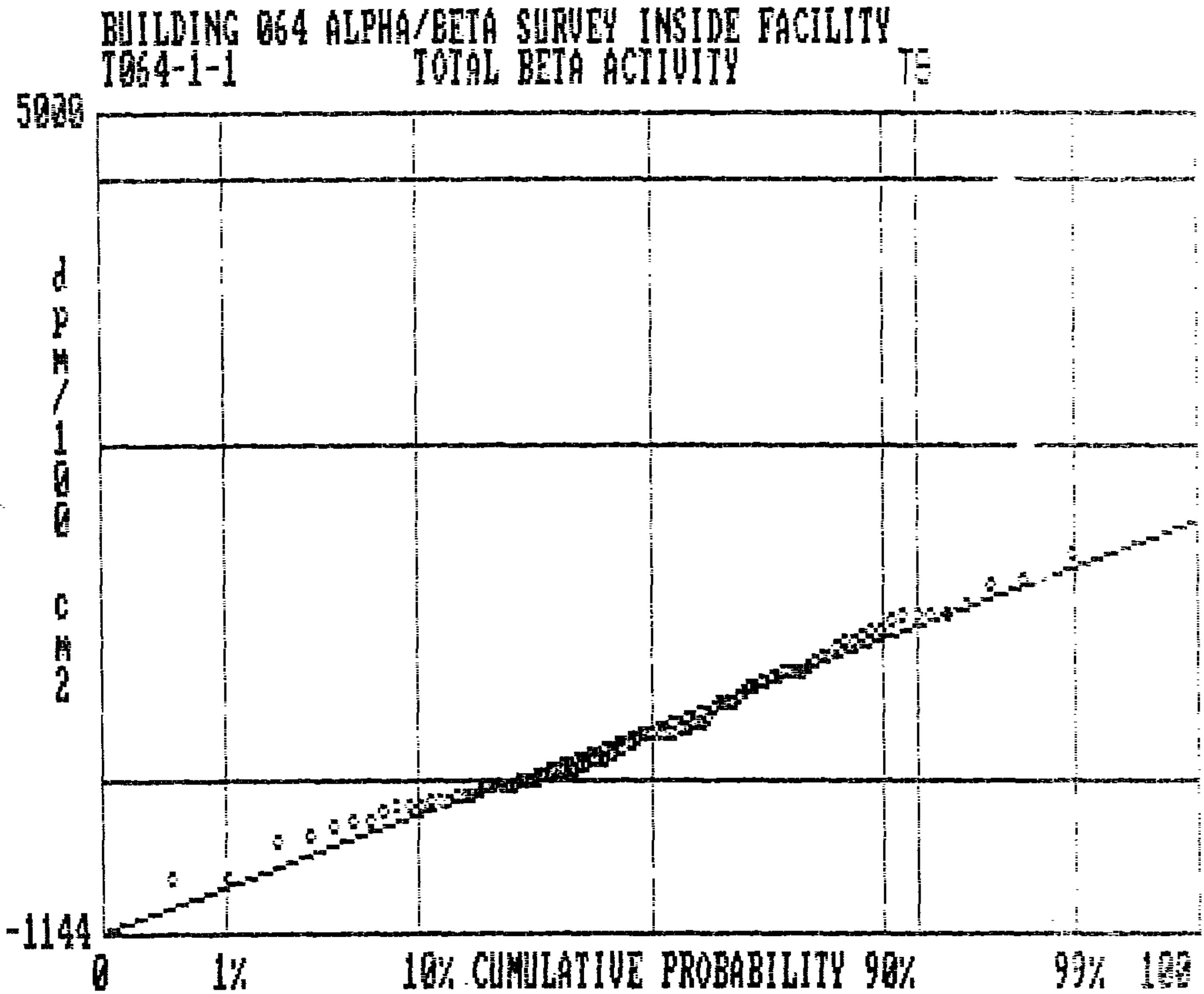


Figure 7.4 Total-Average Beta Activity Inside T064



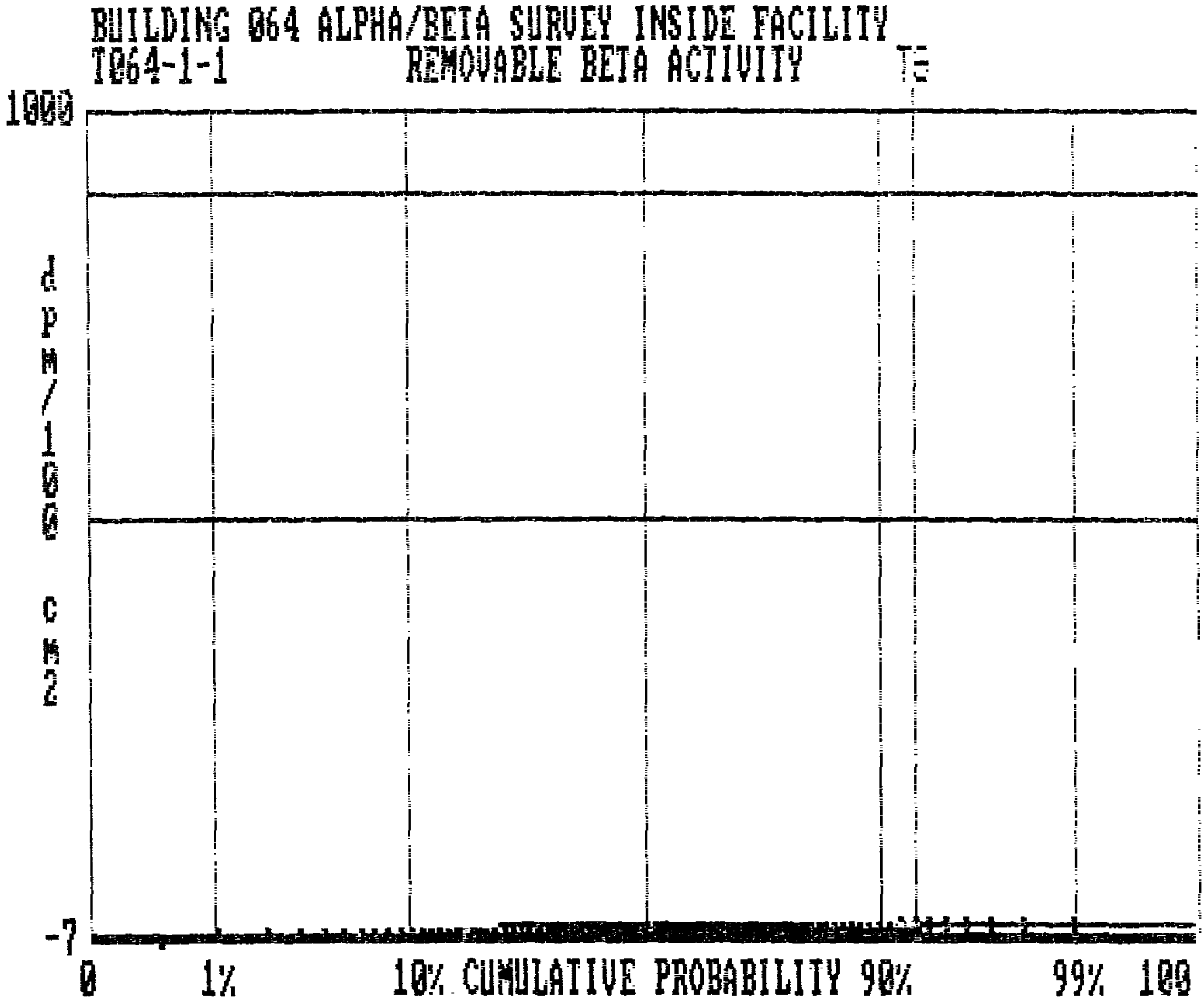


Figure 7.5 Removable Beta Activity Inside T064

7.2.2 Miscellaneous Features

Section 6.3.5 describes the procedure used to survey miscellaneous building features. These surveys were for "indication only." All of the accessible wall coving, wall-to-floor joints, fire extinguishers, and miscellaneous stored items surveyed were not contaminated. These competent surveys indicate No Detectable Activity (NDA). Table 7.2 lists those items which were discovered contaminated or known to be contaminated.

Table 7.2 Contamination on Miscellaneous Items

Item		Average Level Fixed Contamination		Average Level Removable Contamination	
		Alpha (dpm/100 cm^2)*	Beta (dpm/100 cm^2)*	Alpha (dpm/100 cm^2)	Beta (dpm/100 cm^2)
(1) Voland & Son 25 kg capacity balance		Not Measured		37	NDA**
(2) Top of Light Fixtures, Rm 110	#1	320	NDA	NDA	NDA
	#2	390-450	NDA	NDA	NDA
	#3	320	NDA	NDA	NDA
(3) Top of Light Fixtures, Rm 114	#1 to #5	NDA	NDA	NDA	NDA
(4) Floor Mop in Rm 110		Not Detectable	6750	Not Measured	
(5) Carey Scale		NDA	6500-10000	NDA	NDA
(6) Fume Hood (Rm 104)		NOT SURVEYED - KNOWN TO BE CONTAMINATED			

* Total contamination is measured and presented here in dpm/100 cm^2, based on an "indication only" reading on an analog scale (ratemeter) instrument.
**NDA: No Detectable Activity

(1) 19 smears were taken inside the large cabinet containing the balance. Direct-fixed readings were not easily accessible; they weren't performed. Of the 20 smears analyzed, 11 were NDA-alpha; all 20 were NDA-beta. Of the 8 positive-alpha smears, the average is 61 ± 30.3 dpm/100 cm^2, with a greatest value of 114 dpm/100 cm^2 The balance interior is slightly contaminated.

(2) The tops of 3 light fixtures in room 110 were surveyed. Since fixed alpha contamination was detected on all 3 at the same level, (about 500 dpm/100 cm^2) additional light fixtures were not surveyed. These fixtures need to be decontaminated or disposed of properly before final decommissioning. The 3 fixtures were directly in front of the door on the east side.

(3) 5 light fixtures were surveyed in Room 114. All were NDA.

(4) Floor mop found slightly contaminated. Alpha radiations are not detectable because of geometry limitations and absorption.

(5) Large industrial-strength scale found contaminated on its weighing platform.

7.3 Fenced-in Storage Yard

Fifty-eight ambient gamma exposure rate measurements were made in this outside area. As mentioned in Section 5.6.2, "background" was not subtracted from these measurements because of the strong dependency of the measurement on location. The variability of natural background at SSFL is demonstrated in Figures 7.8, 7.9, and 7.10. These are measurement values from areas where absolutely no radioactive materials were ever handled; e.g., "background" measurements.

Figure 7.6 shows the statistical distribution of gamma measurements in the fenced-in storage yard. The average value of this distribution of measurements is 20 ± 14.3 μR/h, with two maximum values equal to 76 μR/h. By observation of the distribution, there is obviously a "clean" Gaussian distributed area, and a contaminated area. This is distinguishable from the large break, or change in slope, of the distribution. The Gaussian "fitted" line is meaningless here; the points are not normally distributed. Further investigation of the occurrence of these elevated readings shows a

Figure 7.6 Ambient Gamma Radiation in Fenced-in Storage Yard

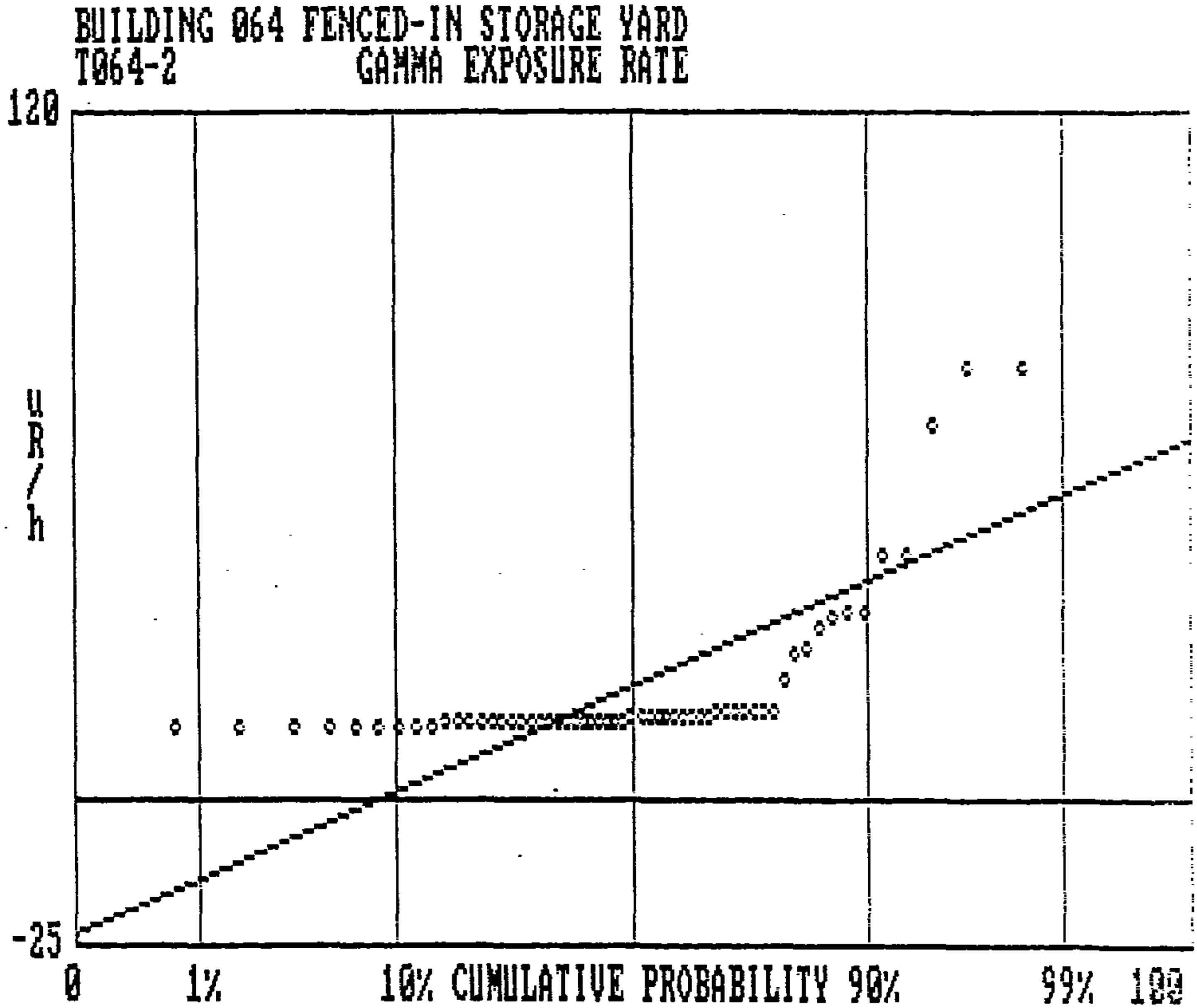
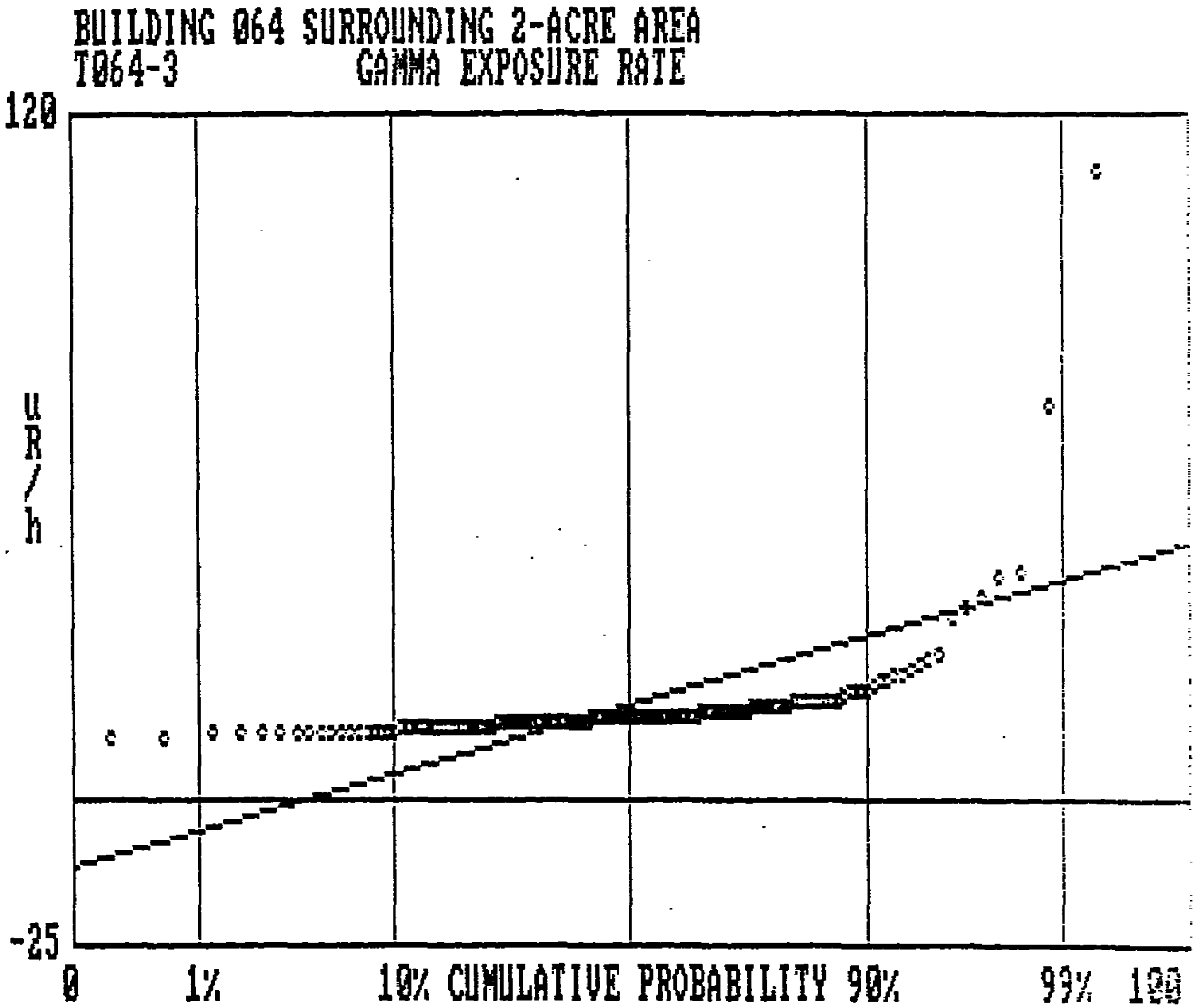


Figure 7.7 Ambient Gamma Radiation in Surrounding 2-Acre Area



contaminated area along the eastern fence-line, outside the fence boundary. Survey and analysis of the outside area (Section 7.4) show Cs-137 contamination. Some higher readings were also attributed to the south side of the building, where the depleted and normal uranium was stored in room 110.

The photon flux was of course attenuated through the building materials. Appendix D.2 shows the gamma measurement data and a map showing measurement locations.

7.3.1 Filter Plenums

Each filter plenum, (one on the south and one on the north), was entered and surveyed for radioactive contamination. The north plenum had 2 entrance doors; one to the HEPA filter side, the other to the pre-filter side. The south plenum had only one door. Table 7.3 shows the results of this survey. The same survey criteria apply here as in Section 7.2.2.

Table 7.3 Contamination Survey of T064 Filter Plenums

Filter Plenum Location	Average Level Fixed Contamination		Average Level Removable Contamination	
	Alpha ($dpm/100\ cm^2$)	Beta ($dpm/100\ cm^2$)	Alpha ($dpm/100\ cm^2$)	Beta ($dpm/100\ cm^2$)
South (single door)				
Floor	NDA	NDA	NDA	NDA
HEPA filter	NDA	NDA	NDA	NDA
Pre-filter (cool)	NDA	2000	16	60
Pre-filter (hot)	NDA	4500	44	51
Gamma exposure rate at background levels (12 μR/h)				
North (HEPA-filter side)				
Baffle	NDA	750	15	53
Pre-filter	NDA	750	NDA	NDA
(Pre-filter side)	NDA	2500	NDA	NDA
Gamma exposure rate at background levels (12 μR/h)				

Maslin wipes from each filter plenum show in all four samples about 0.1 nCi of Cs-137 activity. This is simply naturally occurring Cs-137 from bomb blast fallout. These analyses show no presence of gamma-emitting radionuclides from operations performed at T064.

The filter plenums are slightly contaminated, but not hazardous. Care should be used during dismantling to ensure that radioactive material perhaps trapped in corners is not released uncontrolled.

7.3.2 Miscellaneous Samples

Many smears were collected form the roof gutters, and exhaust stacks. All levels of radioactivity were NDA. Soil samples were collected below the gutter downspouts on the south side of the building. Each sample had background amounts of radioactivity: 17pCi/g - alpha; 25 pCi/g - beta; 1 pCi/g of U-238 and Th-238; and .9 pCi/g Cs-137. We did not expect to find amounts greater than background levels.

7.4 Surrounding 2-Acre Area

168 ambient gamma exposure rate measurements were made in this surrounding area. "Background" was not subtracted from these measurements. These measurements should be compared to Figures 7.8, 7.9 and 7.10, which show areas of "natural background" radiation at SSFL.

Figure 7.7 shows the statistical distribution of gamma measurements. The average value of this distribution of measurements is 16.6 $\pm$ 9.4 μR/h, with the maximum value equal to 109 μR/h. The large standard deviation and change in slope of the plotted points suggest a contaminated area. The area east of the eastern fence-line is contaminated with mixed fission products. We measured Cs-137 contamination. Figure 7.13 shows the location of this contamination. Appendix D.3 lists the measurement data and a map of measurement locations.

An average of the greatest 8 values (corresponding to the distribution break) is 48 ± 28 μR/h, clearly above ambient background. A location near the utility pole 70 ft east of the fence was the most radioactive. Surface exposure rate was 300 μR/h, about 18 times background. Two soil samples taken in that location shows Cs-137 radioactivity at 2500 and 2700 pCi/g. Natural Cs-137 activity in soil is normally between 0.1 to 1.0 pCi/g. Alpha analysis results show alpha activity concentrations at 31.4 and 23.8 pCi/g, slightly greater than an average background of about 16 pCi/g. Beta analysis results show beta activity concentrations at 1153 and 1187 pCi/g, much greater than an average background of 24 pCi/g. Acceptance criteria for soil beta activity is 100 pCi/g. The area is contaminated. Further investigation is required to determine the depth and extent of contamination, and the Sr-90 contribution. Remedial action is necessary.

7.4.1 Survey of SRE Moderator Cask

Outside the fenced-in storage yard, on the west side of the facility, is an area commonly used for storing shipping trailers and casks. An SRE moderator cask currently parked there is known to be internally contaminated. Thirty smear samples were taken externally on the front, side, rear, and bottom. Results were all NDA. Exposure rate measurements were also taken in these locations. The majority were equal to "background" levels. Internal contamination near the front lid, however, is emitting enough gamma radiation to produce exposure rates of 50 to 80 μR/h on contact.

The cask needs to be decontaminated inside. Contamination is not sweating out to the environment. The cask is not a radiological hazard in its current condition.

7.5 Ambient Gamma Exposure Rates in Non-Radiological Areas

Because the background gamma-radiation environment is quite variable at SSFL and because the limits for unrestricted use are based on limits above background, further demonstration of this variability is necessary. For comparison against the T064 measurements, three independent areas were surveyed, all in locations where no radioactive material was ever handled, used, stored, or disposed. All three areas are located on the eastern side of SSFL: (1) Area surrounding building 309 on Area I Road; (2) well #13 Road; and (3) Incinerator Road. At least 30 measurements were made in each area on the same day. Table 7.4 shows the results of these measurements.

Table 7.4 Ambient Gamma Radiation at SSFL Compared to T064 Measurements

Location	No. of Measurements	Average Exposure Rate (μR/h)	Standard Deviation μR/h	Range μR/h
Bldg. 309 Area (1/19/88)	36	15.6	0.8	3.4
Well #13 Road (Dirt) (4/29/88)	43	16.2	0.5	2.2
Incinerator Road (Dirt) (4/29/88)	35	14.0	0.4	1.4
T064 Fenced-In Storage Yard	58	20.1	14.3*	63*
T064 Surrounding 2-Acre Area	168	16.6	9.4*	98*

* Indicate potential contaminated area. See probability plots.

Measurements from the area surrounding building 309 show the most variability of all three background areas. This is attributed to large sandstone outcroppings in the area; the spatial dependency of the measurements is observable in this case. Otherwise, the topography of each location is similar. The variability of each distribution depends on the number of measurements made directly against the rock versus the number made many feet from the rock. Also of importance here is the range of measurement values with a maximum of 3.4 μR/h. The background variability approaches the NRC limit of 5 μR/h.

Figure 7.8 Ambient Gamma Radiation at Area Surrounding Building 309
(Background Distribution)

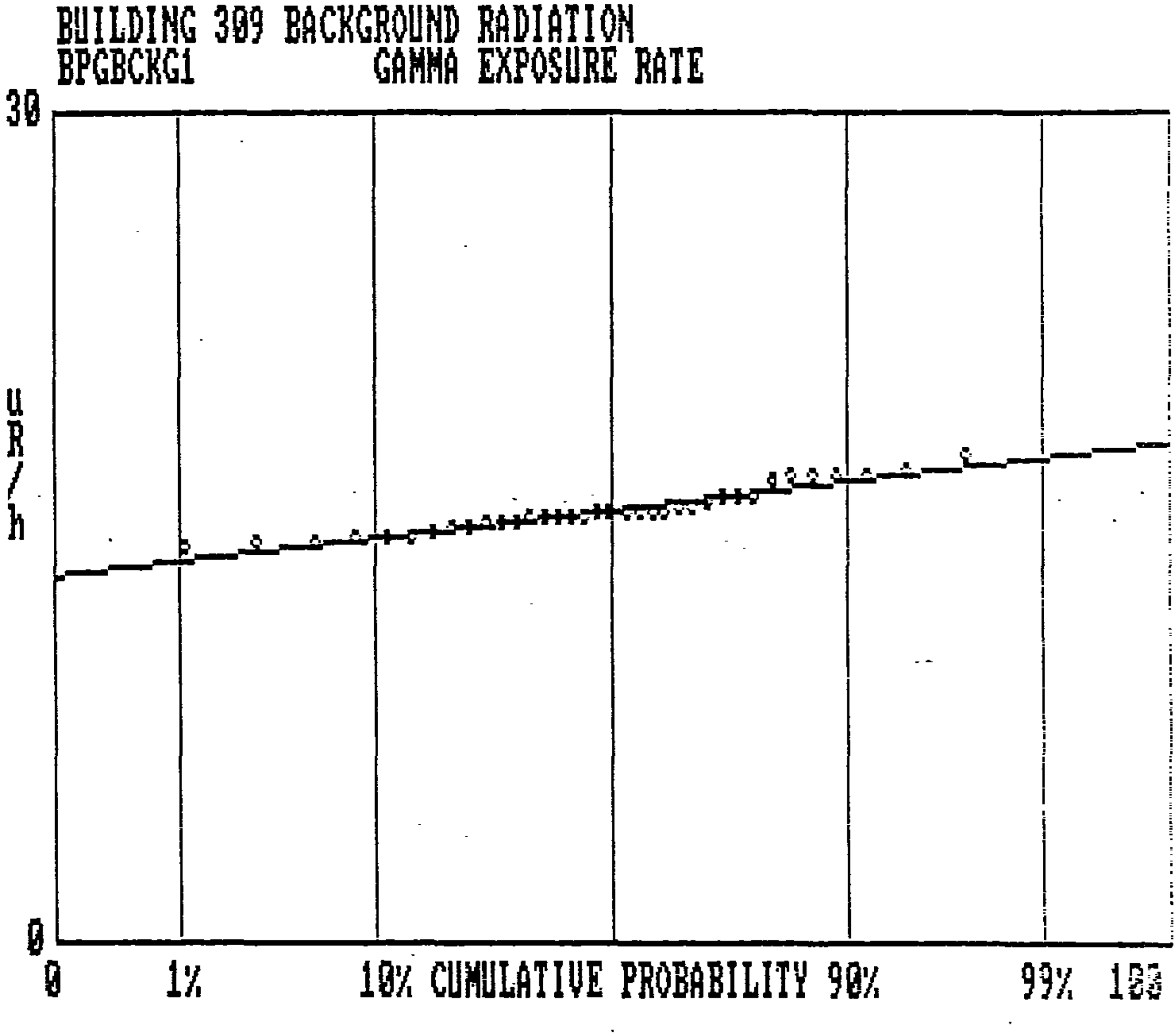


Figure 7.9 Ambient Gamma Radiation at Area Well #13 Road
(Background Distribution)

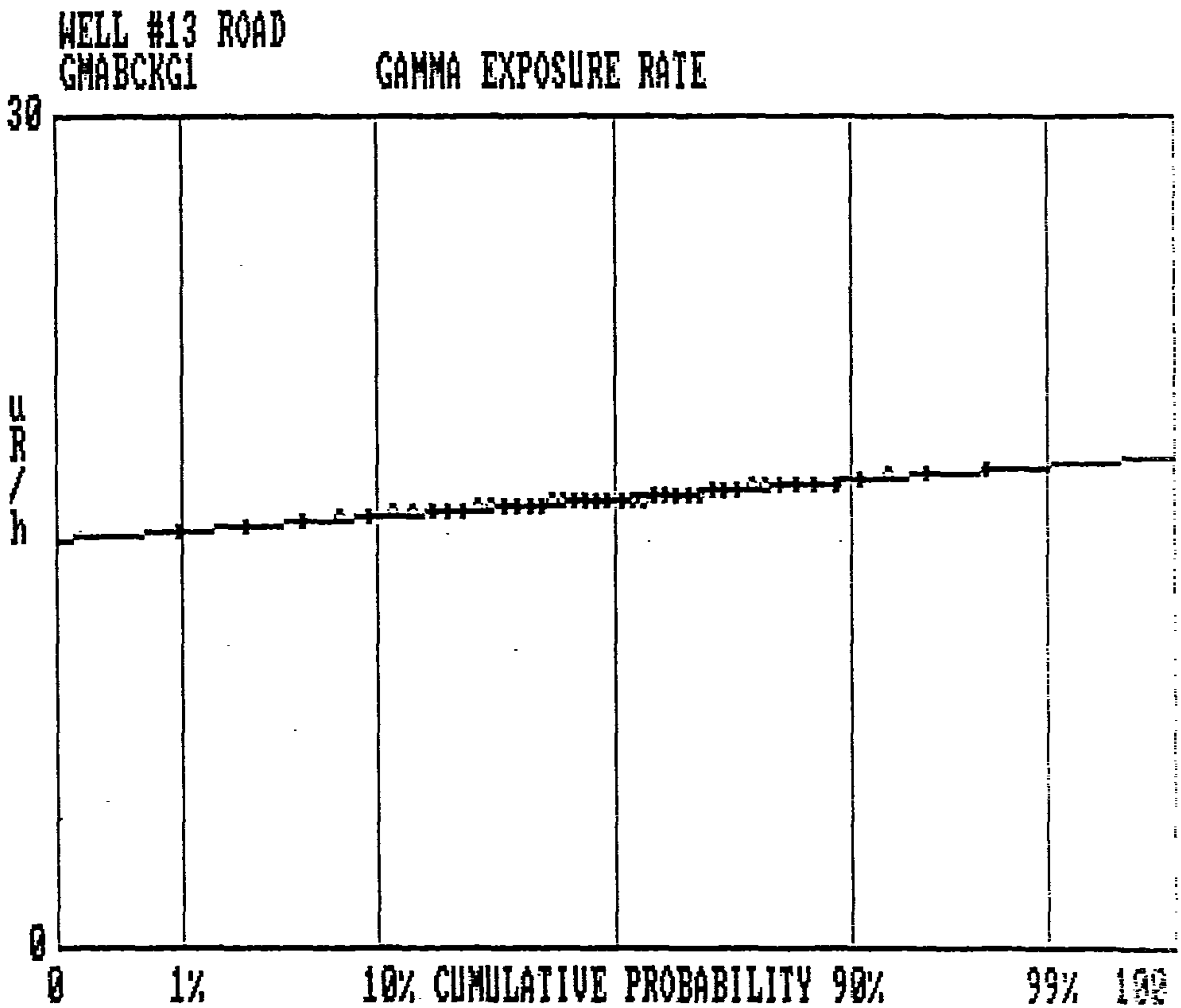
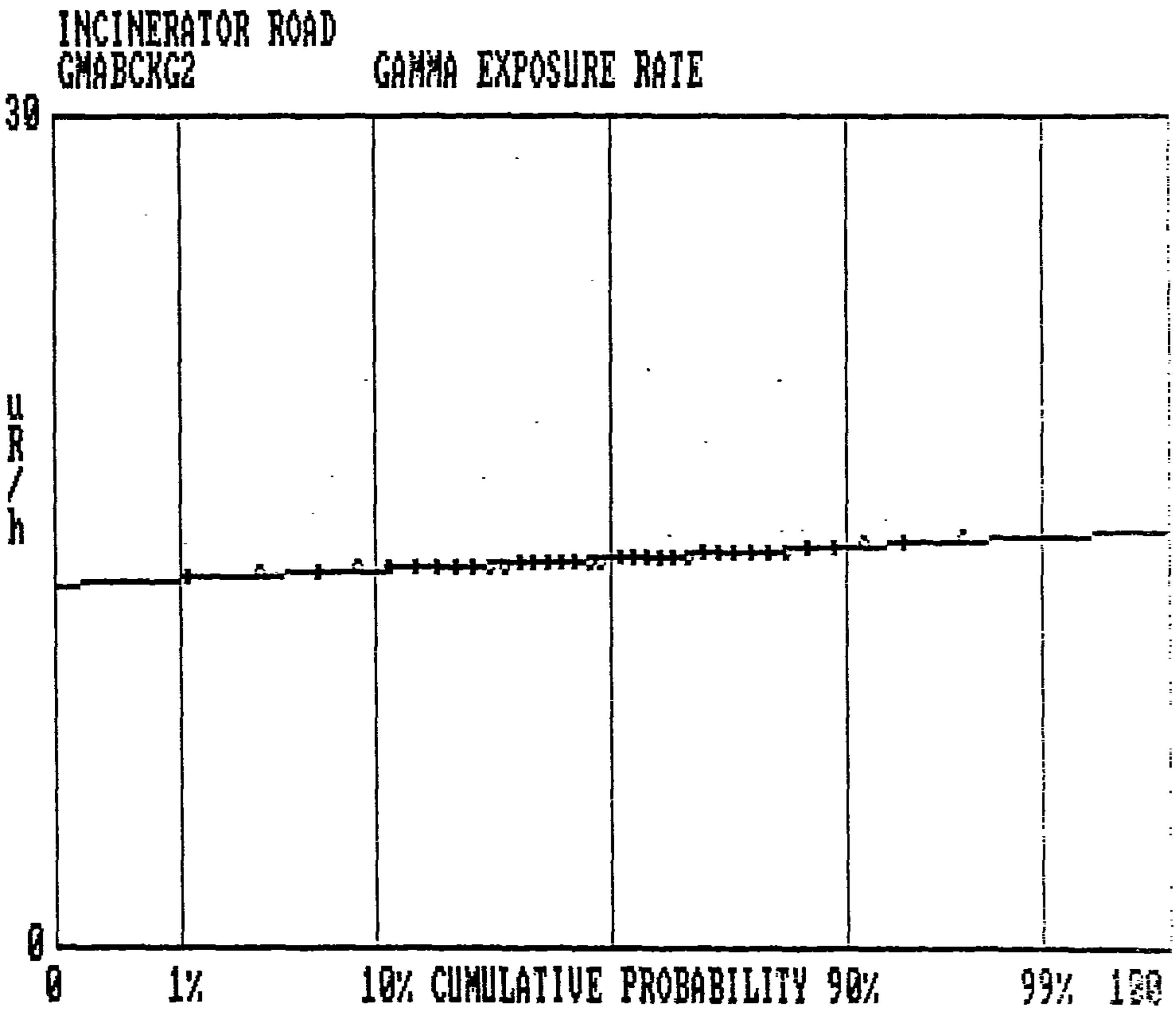


Figure 7.10 Ambient Gamma Radiation at Incinerator Road
(Background Distribution)



To assist in observing the variability of ambient gamma radiation levels, we have plotted the probability function for each "background" area in Figures 7.8 through 7.10. A uniform background rate would appear as a straight line with slope equal to zero. All four distributions show model Gaussian functions; however, the deviation is greatest in the area near building 309 (slope is the greatest). For comparison, the T064 plots are displayed again in Figures 7.11 and 7.12.

This analysis shows the great difficulty in assessing whether an area is contaminated based on the NRC limit of 5 μR/h above background. The DOE limit of 20 μR/h is more reasonable. It is quite clear from the data, however, that there is a contaminated area at T064. Although both distributions of outdoor gamma exposure rate measurements show contamination, only an area *outside the fence-line is contaminated. Measurements made inside* the fence are sensitive enough to detect that contamination on the other side of the fence.

7.6 Assessment of Radiological Condition

Results of this survey show that a few items inside building T064 are slightly contaminated below release limits. Concrete used for both ramps is "naturally" slightly radioactive. The filter plenums are slightly contaminated, but certainly not hazardous. An area bordering and outside of the eastern fence is contaminated with mixed fission products. This area covers up to approximately 4000 ft^2. An area covering about 300 ft^2 of that is significantly contaminated with consistent exposure rate readings from 50-100 μR/h at 1 meter. This location is where 2 soil samples were taken to show Cs-137 activity of 2500 pCi/g. Figure 7.13 shows the general vicinity of this contamination. Further investigation and remedial action is required in this area.

Figure 7.11 Ambient Gamma Radiation in Fenced-In Storage Yard

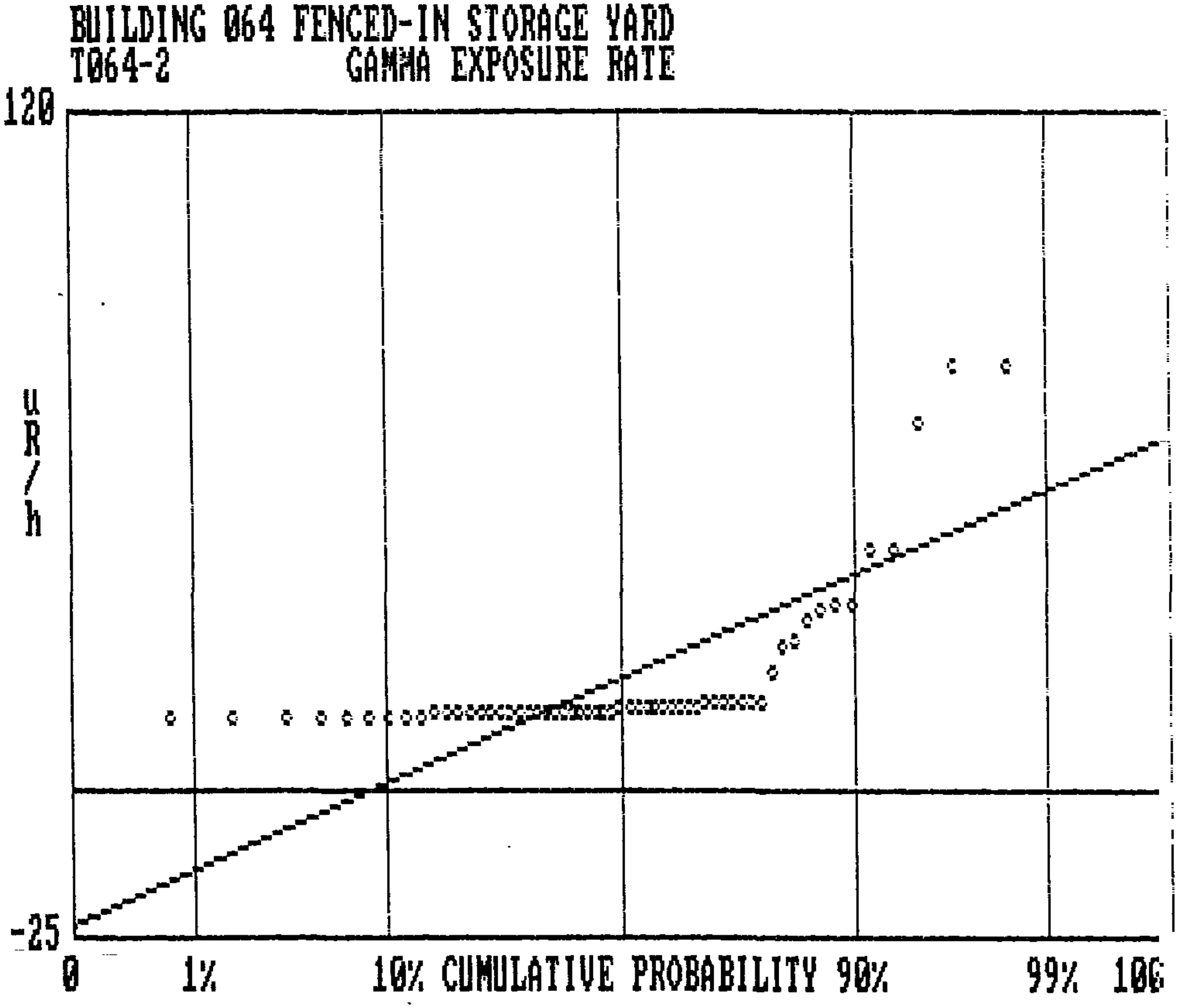


Figure 7.12 Ambient Gamma Radiation in Surrounding 2-Acre Area

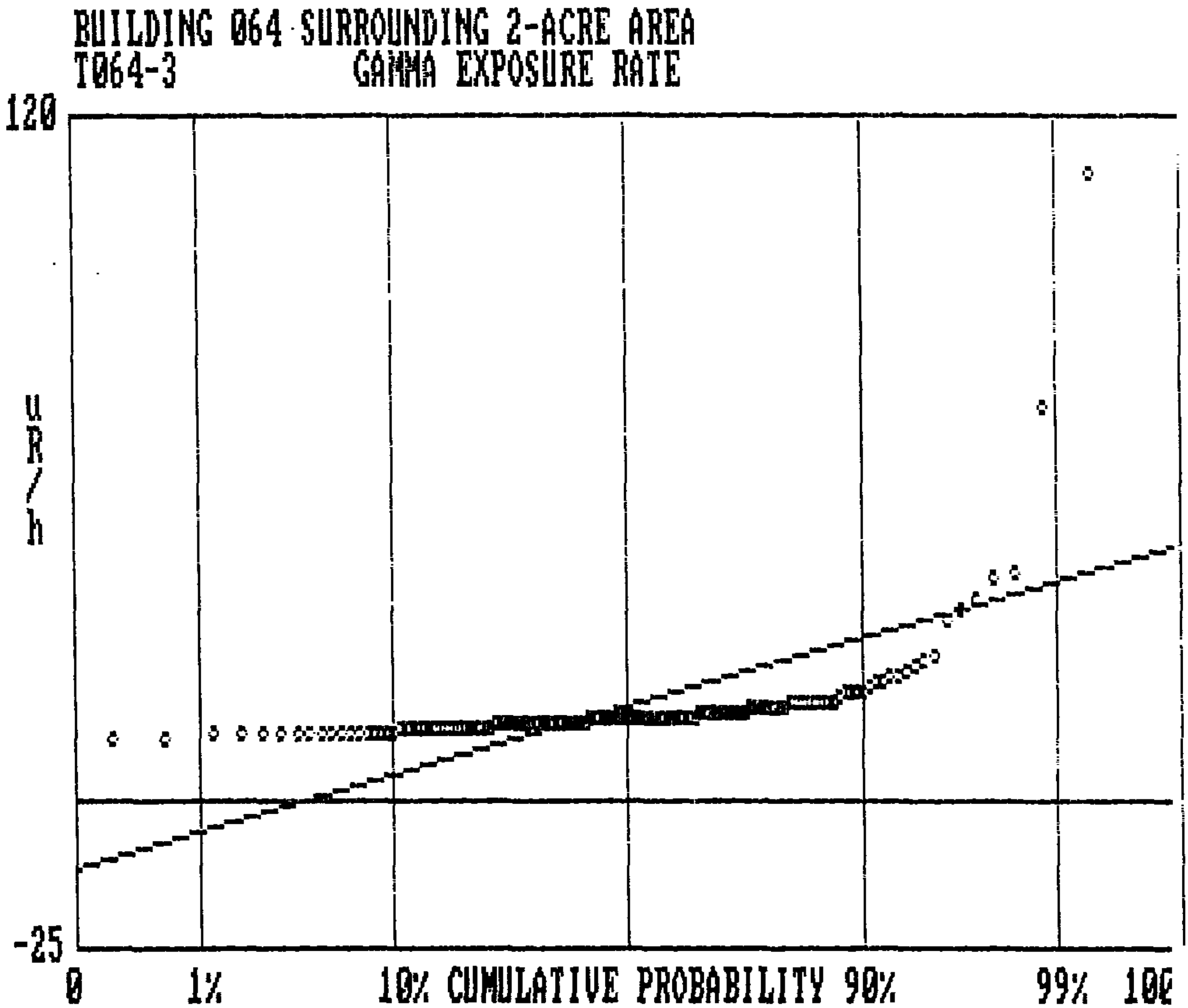
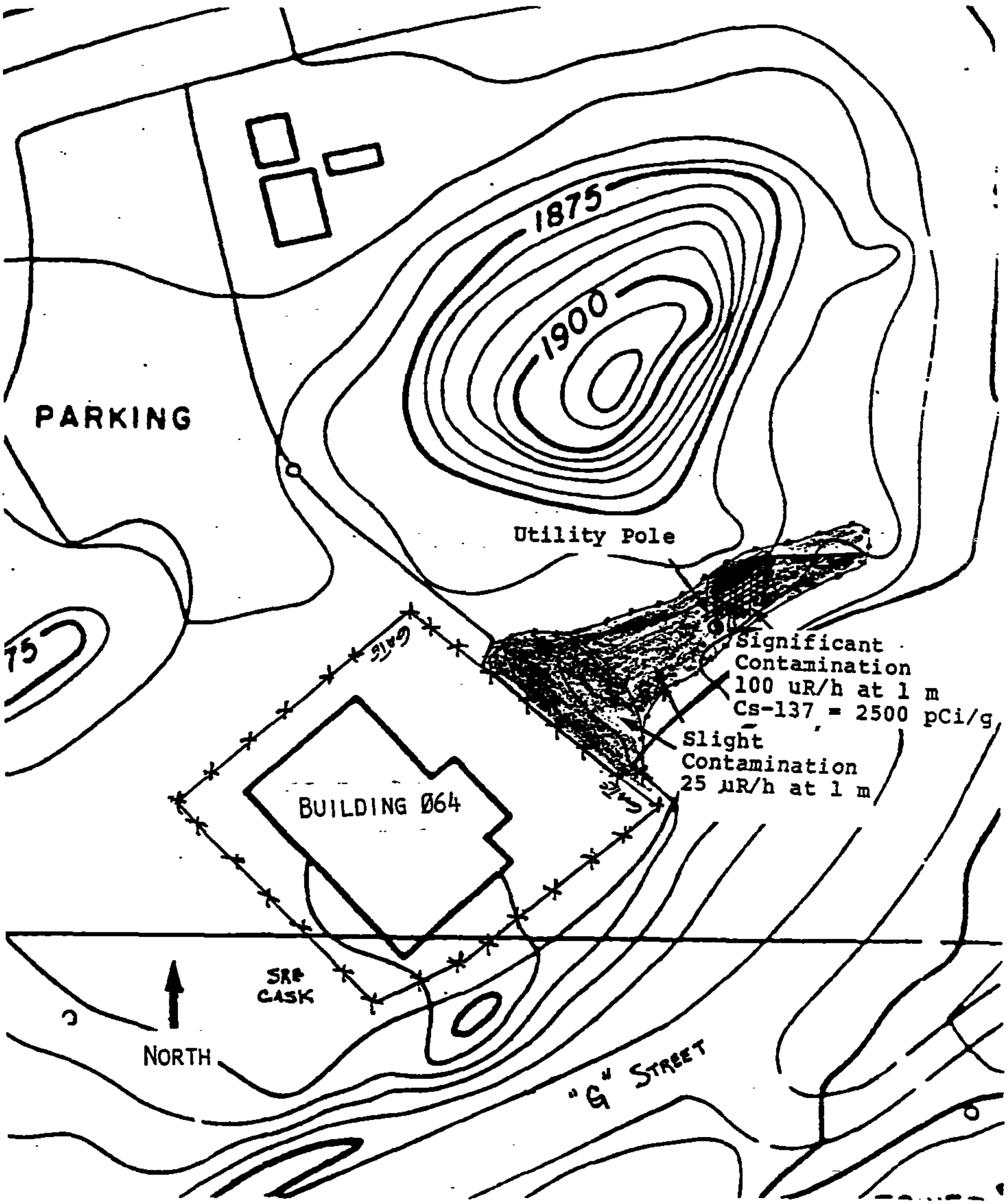


Figure 7.13 Topography Map of T064 Showing Contaminated Soil Area



8.0 CONCLUSIONS

The building T064 interior, filter plenums, fenced-in storage yard, and 2-acre surrounding area were inspected for radioactive contaminants. All direct and removable measurements made for alpha/beta radioactivity on inside walls and floors show that contamination is far below acceptance limits for release for unrestricted use. The effect of wall repainting in 1980 on these survey results is not known. The concrete ramps leading to rooms 110 and 114 are radioactive at a fairly uniform rate, with alpha activity at 51 $\pm$ 13 dpm/100 cm^2, much greater than all other areas surveyed. Figure 7.2 shows this deviation. This alpha activity is from natural elements in the concrete, and is not removable. Contamination was not detected underneath floor tile. A few miscellaneous items inside are slightly contaminated, much less than acceptance limits. These items include a 25 kg capacity balance, light fixtures, floor mop, scale, and the fume hood. Each filter plenum is slightly contaminated, primarily in nooks and crannies, and on the pre-filter side of each HEPA filter.

Ambient gamma exposure rate measurements made within the fenced-in storage yard show a contaminated area bordering and outside of the eastern fence. Figure 7.6 shows this deviation. The average value of these measurements is 20 $\pm$ 14 μR/h. A significantly large standard deviation demonstrates a wide variation in the measurements. Soil samples taken in the yard contain primordial and atmospheric radionuclides in "background" quantities. No contamination was detected inside the fenced-in storage yard.

Ambient gamma exposure rate measurements made outside the fence in a 2-acre area show an area of no more than 4000 ft^2 contaminated with Cs-137. This area is shown in Figure 7.13. Figure 7.7 shows the deviation observed in the measurements. Two soil samples taken from an area (up to 300 ft^2) of greatest exposure rate (100 μR/h @ 1 m, 300 μR/h @ contact) shows Cs-137 radioactivity concentrations of 2500 pCi/g. This contamination is 2500 times normal background concentrations. The remaining surface soil, 3700 ft^2, (exposure rate ≈ 40 μR/h on contact) might be contaminated to about 500 pCi/g. Beta radioactivity in the area of greatest contamination was measured at 1200 pCi/g. Sr-90, which normally accompanies Cs-137, is probably present here, but not specifically measured and analyzed. The area is not hazardous. Further investigation is required to measure specifically the extent of contamination area-wise and depth-wise. Assuming as an upper limit a 300 ft^2 section contaminated 1 ft deep at 2000 pCi/g and a 3700 ft^2 section at 500 pCi/g, total Cs-137 in the area may approach 100 mCi (25 mCi + 75 mCi, respectively). Remedial action is required.

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," N001TI000200, W.D. Kittinger, Rockwell International, September 30, 1983.

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

7. "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.

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

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

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

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

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

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

14. "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.

15. "Sampling Procedures and Tables for Inspection by Variables for Percent Defective," MIL-STD-414, June 11, 1957.

16. "Lower Limit of Detection and Statistically Significant Activity for Radiologic Measurements," IL from R. J. Tuttle to Radiation and Nuclear Safety, Rocketdyne/Rockwell International, June 24, 1986.

17. "Radiological Survey of Building T005," GEN-ZR-0003, J. A. Chapman, Rocketdyne/Rockwell International, February 1, 1988.

18. "Radiological Survey of the Sodium Disposal Facility - Building T886," GEN-ZR-0004, J. A. Chapman, Rocketdyne/Rockwell International, June 3, 1988.

APPENDIX A. DESCRIPTION OF NUCLEAR INSTRUMENTATION

During the radiological survey, smear-test wipes from interior surfaces, and soil samples were analyzed for radioactivity content by one or more of the following nuclear instrumentation systems. Direct radiation measurements were made by using portable instruments.

A.1 Gamma Spectrometry Analyzer

Gamma spectrometry of selected samples, was performed with a Canberra Industries, Inc. Series 80 Multichannel Analyzer (MCA). The MCA is coupled to a planar high 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 spectrometry were presented to the detector with the same geometric configuration as the MBSS.

A.2 Gross Alpha/Beta Automatic Proportional Counter

Soil samples and smear test wipes 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. The system detects radiation in a 2π geometry using P-10 gas (90% methane, 10% argon). A cosmic-ray detector provides coincidence event cancellation to reduce instrument background. The two detectors operate in an anticoincidence mode to reduce the count rate due to cosmic-ray events. When cosmic-ray 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 on smear-wipes 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. Calibration for soil counting involves the use of an NBS traceable U-235 spiked soil standard for alpha radiation; KCl for beta radiation; and nutrient-depleted sea sand for detector background measurements.

A.3 Portable Instruments

Ludlum model 2220-ESG portable scaler/ratemeters coupled to alpha, beta, and gamma probes were used during the course of this survey. The 2220-ESG 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 2220-ESGs were connected to separate probes; alpha, beta, and 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 72 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 a nominal active area of 20 cm^2. Background for this probe is about 80 to 100 cpm. Efficiency for Tc-99 beta particles is between 25% and 20%.

A Ludlum model 44-10 NaI gamma scintillator was used for detecting gamma radiation. The NaI(Tl) crystal is extremely sensitive to changes in gamma flux. The probe efficiency varies with exposure rate. At background ambient gamma exposure rates, the efficiency is about 215 cpm/(μR/h). This determination was made by calibrating the 2220-ESG against a Reuter Stokes High-Pressure Ion Chamber (HPIC). The HPIC displays a digital readout every 3 to 4 seconds in μR/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

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RECEIVED
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 airborne radiation standards applicable to DOE facilities. These final standards, issued as revisions to 40 CFR 61, are:

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- 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

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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.

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*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.

Y

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 Long-Term Management

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.

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

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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.

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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.]

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APPENDIX C. GAMMA SPECTROMETRY RADIONUCLIDE

GAMMA-RAY ENERGY AND YIELD LIBRARY

	Isotope Energy (keV)	Half-Life % Yield								
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%						

Isotope Half-Life
Energy (keV) % Yield

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%

	Isotope Energy (keV)	Half-Life % Yield								
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%

	Isotope Energy (keV)	Half-Life % Yield								
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%				

APPENDIX D

BUILDING T064 SURVEY DATA

D.1 Building Interior: Sorted by Room

T064-1.WS

ROOM NUMBER	GRID NAME	ALPHA DPM/100CM2 TOTAL	STD DEV	MAX	STD DEV	REM	STD DEV	BETA DPM/100CM2 TOTAL	STD DEV	MAX	STD DEV	REM	STD DEV
116	F1,3	3.1	6.5			2.3	0.6	377	120			5	2
116	W2,2	1.5	6.4			-0.1	0.3	1229	123			3	2
116	E2,3	0.0	6.2			0.0	0.3	392	132			2	2
116	W1,3	-3.1	5.8			1.0	0.4	893	133			5	2
114	F12,15	1.5	6.4			0.8	0.4	-90	119			2	2
114	F9,15	0.0	6.2			0.2	0.3	-41	120			0	1
114	F7,15	0.0	6.2			0.0	0.3	-15	120			4	2
114	F1,15	0.0	6.2			0.7	0.4	-26	123			4	2
114	F4.14	-7.7	5.1			0.2	0.3	-128	123			5	2
114	F10,14	6.2	6.9			0.2	0.3	-230	123			3	2
114	F9,13	-3.1	5.8			0.6	0.4	226	123			4	2
114	F9,9	3.1	6.5			-0.3	0.2	-94	124			1	2
114	F3.2	0.0	4.6			0.0	0.3	79	126			4	2
114	F3.10	-5.3	3.8			0.1	0.3	-307	120			8	2
114	F5.10	-2.7	4.2			0.1	0.3	-7	121			7	2
114	F6,6	-4.0	4.0			0.1	0.3	-464	116			4	2
114	F6,9	-2.7	4.2			0.3	0.3	311	114			9	2
114	F6,12	1.3	4.8			0.0	0.3	-132	116			3	2
114	F7.11	-2.7	4.2			0.3	0.3	536	116			7	2
114	F7,5	-4.0	4.0			0.0	0.3	89	115			6	2
114	F9,5	-5.3	3.8			0.5	0.4	164	116			5	2
114	F12,4	-2.7	4.2			0.2	0.3	-25	116			5	2
114	F11,14	-5.3	3.8			0.7	0.4	161	111			5	2
114	F8,12	-2.7	4.2			-0.1	0.3	100	114			4	2
114	F10,9	-4.6	5.6			0.1	0.3	192	121			4	2
114	N2,1	-4.6	5.6			0.0	0.3	-57	135			5	2
114	N2,6	-3.1	5.8			0.2	0.3	690	147			3	2
114	N1,10	-7.7	5.1			-0.2	0.2	927	122			4	2
114	E1.1	-4.6	5.6			-0.1	0.3	826	127			5	2
114	E2,10	-3.1	5.8			0.0	0.3	-4	130			10	2
114	W1,7	-3.1	5.8			-0.1	0.3	1229	125			5	2
114	W2,5	-6.2	5.3			0.0	0.3	746	125			7	2
114	S1,13	-9.3	4.9			0.1	0.3	347	118			5	2
114	S2,11	-4.6	5.6			0.0	0.3	0	0			7	2
114	S2,2	-7.7	5.1			0.1	0.3	347	122			3	2
110	F8,1	18.7	6.0			1.6	0.5	-34	190			5	2
110	F4,2	9.3	4.8			0.9	0.4	151	162			2	2
110	F7,3	21.4	6.3			2.8	0.6	-397	236			7	2
110	F10,2	9.3	4.8			1.7	0.5	-293	320			4	2
110	F9,4	13.4	5.3			3.2	0.7	-1013	332			10	2
110	F4,5	6.7	4.4			1.0	0.4	112	165			5	2
110	F8,6	20.0	6.1			1.6	0.5	-715	254			4	2
110	F9,7	30.7	7.2			4.7	0.8	-1228	261			10	2
110	F6,10	17.4	5.8			3.1	0.7	625	170			5	2
110	F8,9	14.7	5.5			2.4	0.6	272	203			7	2
110	F10,9	12.0	5.2			3.9	0.7	457	274			9	2
110	F11,10	24.0	6.5			3.2	0.7	-707	225			5	2
110	F12,11	21.4	6.3			3.5	0.7	125	165			11	2
110	F8,12	9.3	4.8			4.5	0.8	642	182			12	2
110	F7,11	9.3	4.8			2.4	0.6	99	198			7	2
110	F6,14	2.7	3.8			3.2	0.7	-6784	332			8	2
110	F7,14	21.4	6.3			1.8	0.5	776	208			4	2
110	F8,15	17.4	5.8			5.7	0.9	-121	162			9	2
110	F9,14	13.4	5.3			2.7	0.6	-134	151			6	2

110	F10,13	13.4	5.3	2.1	0.6	228	149	9	2
110	F11,14	5.3	4.2	2.7	0.6	345	145	6	2
110	S3,1	4.0	4.0	0.7	0.4	397	143	3	2
110	X-1	2.7	3.8	1.2	0.5	-190	143	6	2
110	S1,3	4.0	4.0	1.4	0.5	599	142	5	2
110	E2,11	10.7	5.0	0.5	0.4	849	174	4	2
110	S2,5	2.7	3.8	0.1	0.3	99	175	3	2
110	S2,10	4.0	4.0	0.7	0.4	-11689	673	3	2
110	W1,3	-2.7	2.7	0.1	0.3	43	195	3	2
110	N3,6	4.0	4.0	0.0	0.3	203	165	2	2
110	N2,15	1.3	3.5	0.2	0.3	-177	144	2	2
110	E1,8	6.7	4.4	0.0	0.3	-177	136	2	2
110	N2,2	-7.7	5.1	3.8	0.7	366	120	10	2
110	N1,10	-4.6	5.6	0.2	0.3	415	121	5	2
110	E2,4	0.0	6.2	0.6	0.4	1082	131	1	2
110	W2,11	0.0	6.2	0.5	0.4	358	120	1	2
110	W2,6	4.6	6.7	0.5	0.4	275	174	3	2
100	F2,1	0.0	2.1	0.3	0.3	366	106	6	2
100	F2,4	1.5	2.5	0.9	0.4	-339	110	3	2
100	F3,3	0.0	2.1	1.6	0.5	430	106	4	2
100	E2,1	5.9	3.6	0.0	0.3	24	94	1	1
100	N2,1	2.9	2.9	0.2	0.3	40	100	3	2
100	W2,1	5.9	3.6	0.3	0.3	571	93	5	2
100	S3,1	10.3	4.4	0.2	0.3	726	102	3	2
RAMP110	F3,1	33.7	7.3	1.6	0.5	823	119	4	2
RAMP110	F1,4	45.4	8.4	2.9	0.6	991	122	7	2
RAMP110	F3,5	52.7	9.0	1.2	0.5	706	119	3	2
RAMP110	F2,6	67.4	10.1	0.5	0.4	571	119	4	2
RAMP110	F1,8	74.7	10.7	0.3	0.3	1028	116	5	2
RAMP110	F3,9	57.1	9.4	0.3	0.3	1055	114	3	2
RAMP110	F2,11	46.9	8.5	0.7	0.4	433	123	3	2
RAMP110	F3,14	54.2	9.1	-0.1	0.3	1230	116	5	2
RAMP110	F3,5	64.4	9.9	1.2	0.5	601	122	3	2
RAMP114	NP1	29.2	6.6	0.5	0.3	1194	138	4	2
RAMP114	NP2	48.2	8.2	0.5	0.3	1134	132	1	2
RAMP114	NP3	41.9	7.7	-0.2	0.2	1308	135	-1	2
RAMP114	NP4	43.1	7.8	1.9	0.5	1115	134	2	2
RAMP114	NP5	43.1	7.8	0.5	0.3	926	130	3	2
RAMP114	NP6	64.7	9.4	0.5	0.3	1489	140	2	2
110TR	F7,1	0.0	3.5	0.5	0.3	775	164	1	2
110TR	F6,14	8.5	5.3	0.5	0.3	828	261	-7	2
110TR	F11,10	-1.4	4.3	1.2	0.5	1240	219	-1	2
110TR	F8,15	1.4	6.5	3.4	0.7	832	150	-1	2
110TR	F11,15	8.5	5.3	4.1	0.7	170	135	1	2
110TR	F4,5	1.4	5.8	1.2	0.5	363	166	5	2
110TR	F10,2	2.8	4.5	1.4	0.5	1467	285	7	2
110TR	F9,4	0.0	4.5	1.0	0.4	1693	293	2	2

Sorted by Alpha Measurements

T064-1.WS		ALPHA DPM/100CM2						BETA DPM/100CM2					
ROOM NUMBER	GRID NAME	TOTAL	STD DEV	MAX	STD DEV	REM	STD DEV	TOTAL	STD DEV	MAX	STD DEV	REM	STD DEV
RAMP110	F1,8	74.7	10.7			0.3	0.3	1028	116			5	2
RAMP110	F2,6	67.4	10.1			0.5	0.4	571	119			4	2
RAMP114	NP6	64.7	9.4			0.5	0.3	1489	140			2	2
RAMP110	F3,5	64.4	9.9			1.2	0.5	601	122			3	2
RAMP110	F3,9	57.1	9.4			0.3	0.3	1055	114			3	2
RAMP110	F3,14	54.2	9.1			-0.1	0.3	1230	116			5	2
RAMP110	F3,5	52.7	9.0			1.2	0.5	706	119			3	2
RAMP114	NP2	48.2	8.2			0.5	0.3	1134	132			1	2
RAMP110	F2,11	46.9	8.5			0.7	0.4	433	123			3	2
RAMP110	F1,4	45.4	8.4			2.9	0.6	991	122			7	2
RAMP114	NP4	43.1	7.8			1.9	0.5	1115	134			2	2
RAMP114	NP5	43.1	7.8			0.5	0.3	926	130			3	2
RAMP114	NP3	41.9	7.7			-0.2	0.2	1308	135			-1	2
RAMP110	F3,1	33.7	7.3			1.6	0.5	823	119			4	2
110	F9,7	30.7	7.2			4.7	0.8	-1228	261			10	2
RAMP114	NP1	29.2	6.6			0.5	0.3	1194	138			4	2
110	F11,10	24.0	6.5			3.2	0.7	-707	225			5	2
110	F7,3	21.4	6.3			2.8	0.6	-397	236			7	2
110	F12,11	21.4	6.3			3.5	0.7	125	165			11	2
110	F7,14	21.4	6.3			1.8	0.5	776	208			4	2
110	F8,6	20.0	6.1			1.6	0.5	-715	254			4	2
110	F8,1	18.7	6.0			1.6	0.5	-34	190			5	2
110	F6,10	17.4	5.8			3.1	0.7	625	170			5	2
110	F8,15	17.4	5.8			5.7	0.9	-121	162			9	2
110	F8,9	14.7	5.5			2.4	0.6	272	203			7	2
110	F9,4	13.4	5.3			3.2	0.7	-1013	332			10	2
110	F9,14	13.4	5.3			2.7	0.6	-134	151			6	2
110	F10,13	13.4	5.3			2.1	0.6	228	149			9	2
110	F10,9	12.0	5.2			3.9	0.7	457	274			9	2
110	E2,11	10.7	5.0			0.5	0.4	849	174			4	2
100	S3,1	10.3	4.4			0.2	0.3	726	102			3	2
110	F4,2	9.3	4.8			0.9	0.4	151	162			2	2
110	F10,2	9.3	4.8			1.7	0.5	-293	320			4	2
110	F8,12	9.3	4.8			4.5	0.8	642	182			12	2
110	F7,11	9.3	4.8			2.4	0.6	99	198			7	2
110TR	F6,14	8.5	5.3			0.5	0.3	828	261			-7	2
110TR	F11,15	8.5	5.3			4.1	0.7	170	135			1	2
110	F4,5	6.7	4.4			1.0	0.4	112	165			5	2
110	E1,8	6.7	4.4			0.0	0.3	-177	136			2	2
114	F10,14	6.2	6.9			0.2	0.3	-230	123			3	2
100	E2,1	5.9	3.6			0.0	0.3	24	94			1	1
100	W2,1	5.9	3.6			0.3	0.3	571	93			5	2
110	F11,14	5.3	4.2			2.7	0.6	345	145			6	2
110	W2,6	4.6	6.7			0.5	0.4	275	174			3	2
110	S3,1	4.0	4.0			0.7	0.4	397	143			3	2
110	S1,3	4.0	4.0			1.4	0.5	599	142			5	2
110	S2,10	4.0	4.0			0.7	0.4	-11699	673			3	2
110	N3,6	4.0	4.0			0.0	0.3	203	165			2	2
116	F1,3	3.1	6.5			2.3	0.6	377	120			5	2
114	F9,9	3.1	6.5			-0.3	0.2	-94	124			1	2
100	N2,1	2.9	2.9			0.2	0.3	40	100			3	2
110TR	F10,2	2.8	4.5			1.4	0.5	1467	285			7	2
110	X-1	2.7	3.8			1.2	0.5	-190	143			6	2
110	S2,5	2.7	3.8			0.1	0.3	99	175			3	2

110	F6,14	2.7	3.8	3.2	0.7	-6784	332	8	2
116	W2,2	1.5	6.4	-0.1	0.3	1229	123	3	2
114	F12,15	1.5	6.4	0.8	0.4	-90	119	2	2
100	F2,4	1.5	2.5	0.9	0.4	-339	110	3	2
110TR	F8,15	1.4	6.5	3.4	0.7	832	150	-1	2
110TR	F4,5	1.4	5.8	1.2	0.5	363	166	5	2
110	N2,15	1.3	3.5	0.2	0.3	-177	144	2	2
114	F6,12	1.3	4.8	0.0	0.3	-132	116	3	2
110	E2,4	0.0	6.2	0.6	0.4	1082	131	1	2
110	W2,11	0.0	6.2	0.5	0.4	358	120	1	2
100	F2.1	0.0	2.1	0.3	0.3	366	106	6	2
100	F3,3	0.0	2.1	1.6	0.5	430	106	4	2
114	F7,15	0.0	6.2	0.0	0.3	-15	120	4	2
114	F9,15	0.0	6.2	0.2	0.3	-41	120	0	1
114	F1,15	0.0	6.2	0.7	0.4	-26	123	4	2
114	F3,2	0.0	4.6	0.0	0.3	79	126	4	2
116	E2,3	0.0	6.2	0.0	0.3	392	132	2	2
110TR	F7,1	0.0	3.5	0.5	0.3	775	164	1	2
110TR	F9,4	0.0	4.5	1.0	0.4	1693	293	2	2
110TR	F11,10	-1.4	4.3	1.2	0.5	1240	219	-1	2
114	F6,9	-2.7	4.2	0.3	0.3	311	114	9	2
114	F5,10	-2.7	4.2	0.1	0.3	-7	121	7	2
114	F7,11	-2.7	4.2	0.3	0.3	536	116	7	2
114	F12,4	-2.7	4.2	0.2	0.3	-25	116	5	2
114	F8,12	-2.7	4.2	-0.1	0.3	100	114	4	2
110	W1,3	-2.7	2.7	0.1	0.3	43	195	3	2
116	W1,3	-3.1	5.8	1.0	0.4	893	133	5	2
114	F9,13	-3.1	5.8	0.6	0.4	226	123	4	2
114	N2,6	-3.1	5.8	0.2	0.3	690	147	3	2
114	W1,7	-3.1	5.8	-0.1	0.3	1229	125	5	2
114	E2,10	-3.1	5.8	0.0	0.3	-4	130	10	2
114	F6,6	-4.0	4.0	0.1	0.3	-464	116	4	2
114	F7,5	-4.0	4.0	0.0	0.3	89	115	6	2
110	N1,10	-4.6	5.6	0.2	0.3	415	121	5	2
114	N2,1	-4.6	5.6	0.0	0.3	-57	135	5	2
114	S2,11	-4.6	5.6	0.0	0.3	0	0	7	2
114	F10,9	-4.6	5.6	0.1	0.3	192	121	4	2
114	E1,1	-4.6	5.6	-0.1	0.3	826	127	5	2
114	F11,14	-5.3	3.8	0.7	0.4	161	111	5	2
114	F9,5	-5.3	3.8	0.5	0.4	164	116	5	2
114	F3,10	-5.3	3.8	0.1	0.3	-307	120	8	2
114	W2,5	-6.2	5.3	0.0	0.3	746	125	7	2
114	S2,2	-7.7	5.1	0.1	0.3	347	122	3	2
114	F4,14	-7.7	5.1	0.2	0.3	-128	123	5	2
110	N2,2	-7.7	5.1	3.8	0.7	366	120	10	2
114	N1,10	-7.7	5.1	-0.2	0.2	927	122	4	2
114	S1,13	-9.3	4.9	0.1	0.3	347	118	5	2

Sorted by Beta Measurements

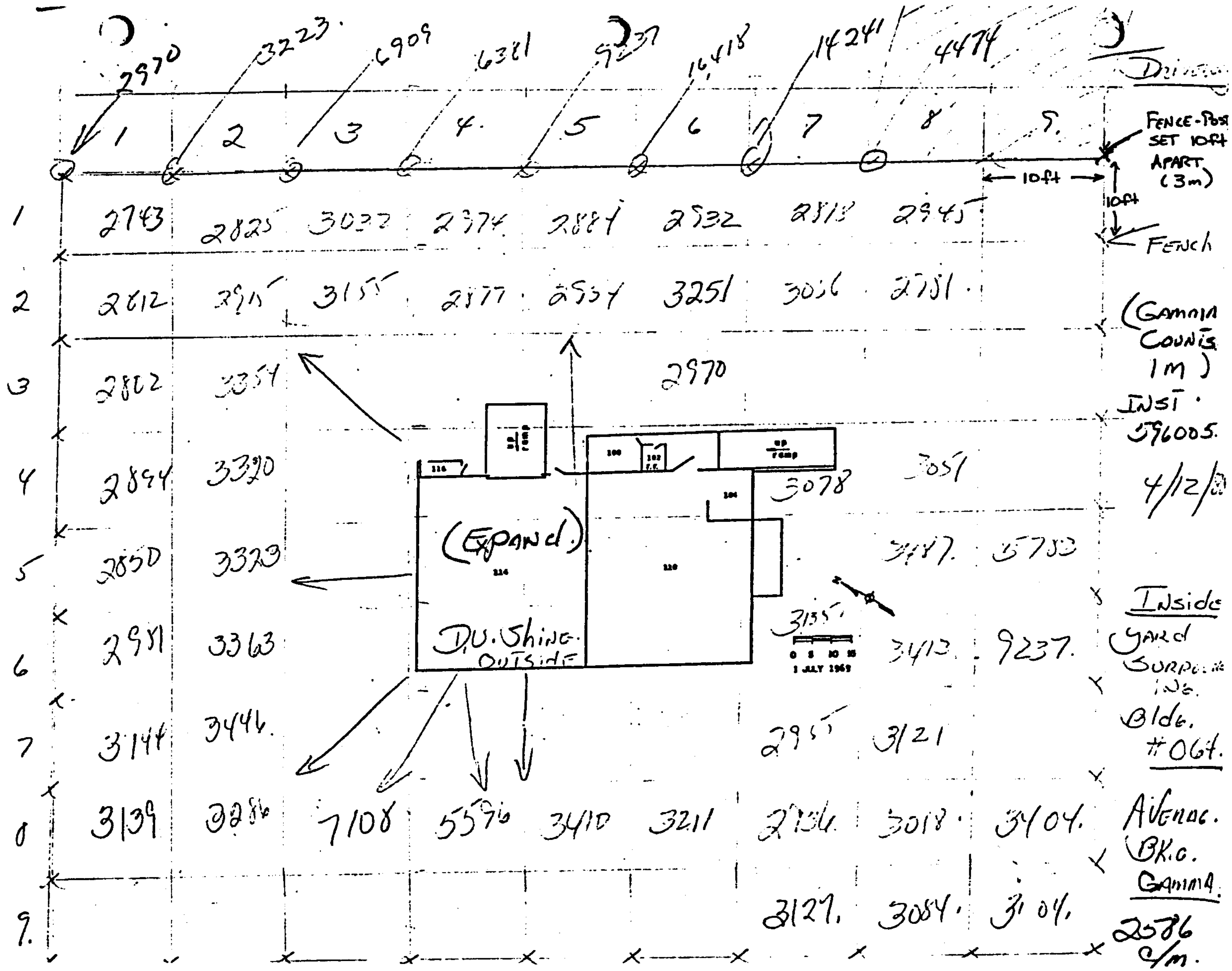
T064-1.WS		<------ ALPHA					------>	<------ BETA					------
ROOM	GRID	DPM/100CM2						DPM/100CM2					
NUMBER	NAME	TOTAL	STD DEV	MAX	STD DEV	REM	STD DEV	TOTAL	STD DEV	MAX	STD DEV	REM	STD DEV
110TR	F9,4	0.0	4.5			1.0	0.4	1693	293			2	2
RAMP114	NP6	64.7	9.4			0.5	0.3	1489	140			2	2
110TR	F10,2	2.8	4.5			1.4	0.5	1467	285			7	2
RAMP114	NP3	41.9	7.7			-0.2	0.2	1308	135			-1	2
110TR	F11,10	-1.4	4.3			1.2	0.5	1240	219			-1	2
RAMP110	F3,14	54.2	9.1			-0.1	0.3	1230	116			5	2
116	W2,2	1.5	6.4			-0.1	0.3	1229	123			3	2
114	W1,7	-3.1	5.8			-0.1	0.3	1229	125			5	2
RAMP114	NP1	29.2	6.6			0.5	0.3	1194	138			4	2
RAMP114	NP2	48.2	8.2			0.5	0.3	1134	132			1	2
RAMP114	NP4	43.1	7.8			1.9	0.5	1115	134			2	2
110	E2,4	0.0	6.2			0.6	0.4	1082	131			1	2
RAMP110	F3,9	57.1	9.4			0.3	0.3	1055	114			3	2
RAMP110	F1,8	74.7	10.7			0.3	0.3	1028	116			5	2
RAMP110	F1,4	45.4	8.4			2.9	0.6	991	122			7	2
114	N1,10	-7.7	5.1			-0.2	0.2	927	122			4	2
RAMP114	NP5	43.1	7.8			0.5	0.3	926	130			3	2
116	W1,3	-3.1	5.8			1.0	0.4	893	133			5	2
110	E2,11	10.7	5.0			0.5	0.4	849	174			4	2
110TR	F8,15	1.4	6.5			3.4	0.7	832	150			-1	2
110TR	F6,14	8.5	5.3			0.5	0.3	828	261			-7	2
114	E1,1	-4.6	5.6			-0.1	0.3	826	127			5	2
RAMP110	F3,1	33.7	7.3			1.6	0.5	823	119			4	2
110	F7,14	21.4	6.3			1.8	0.5	776	208			4	2
110TR	F7,1	0.0	3.5			0.5	0.3	775	164			1	2
114	W2,5	-6.2	5.3			0.0	0.3	746	125			7	2
100	S3,1	10.3	4.4			0.2	0.3	726	102			3	2
RAMP110	F3,5	52.7	9.0			1.2	0.5	706	119			3	2
114	N2,6	-3.1	5.8			0.2	0.3	690	147			3	2
110	F8,12	9.3	4.8			4.5	0.8	642	182			12	2
110	F6,10	17.4	5.8			3.1	0.7	625	170			5	2
RAMP110	F3,5	64.4	9.9			1.2	0.5	601	122			3	2
110	S1,3	4.0	4.0			1.4	0.5	599	142			5	2
100	W2,1	5.9	3.6			0.3	0.3	571	93			5	2
RAMP110	F2,6	67.4	10.1			0.5	0.4	571	119			4	2
114	F7,11	-2.7	4.2			0.3	0.3	536	116			7	2
110	F10,9	12.0	5.2			3.9	0.7	457	274			9	2
RAMP110	F2,11	46.9	8.5			0.7	0.4	433	123			3	2
100	F3,3	0.0	2.1			1.6	0.5	430	106			4	2
110	N1,10	-4.6	5.6			0.2	0.3	415	121			5	2
110	S3,1	4.0	4.0			0.7	0.4	397	143			3	2
116	E2,3	0.0	6.2			0.0	0.3	392	132			2	2
116	F1,3	3.1	6.5			2.3	0.6	377	120			5	2
100	F2,1	0.0	2.1			0.3	0.3	366	106			6	2
110	N2,2	-7.7	5.1			3.8	0.7	366	120			10	2
110TR	F4,5	1.4	5.8			1.2	0.5	363	166			5	2
110	W2,11	0.0	6.2			0.5	0.4	358	120			1	2
114	S2,2	-7.7	5.1			0.1	0.3	347	122			3	2
114	S1,13	-9.3	4.9			0.1	0.3	347	118			5	2
110	F11,14	5.3	4.2			2.7	0.6	345	145			6	2
114	F6,9	-2.7	4.2			0.3	0.3	311	114			9	2
110	W2,6	4.6	6.7			0.5	0.4	275	174			3	2
110	F8,9	14.7	5.5			2.4	0.6	272	203			7	2
110	F10,13	13.4	5.3			2.1	0.6	228	149			9	2

	114 F9,13	-3.1	5.8	0.6	0.4	226	123	4	2
	110 N3,6	4.0	4.0	0.0	0.3	203	165	2	2
	114 F10,9	-4.6	5.6	0.1	0.3	192	121	4	2
110TR	F11,15	8.5	5.3	4.1	0.7	170	135	1	2
	114 F9,5	-5.3	3.8	0.5	0.4	164	116	5	2
	114 F11,14	-5.3	3.8	0.7	0.4	161	111	5	2
	110 F4,2	9.3	4.8	0.9	0.4	151	162	2	2
	110 F12,11	21.4	6.3	3.5	0.7	125	165	11	2
	110 F4,5	6.7	4.4	1.0	0.4	112	165	5	2
	114 F8,12	-2.7	4.2	-0.1	0.3	100	114	4	2
	110 F7,11	9.3	4.8	2.4	0.6	99	198	7	2
	110 S2,5	2.7	3.8	0.1	0.3	99	175	3	2
	114 F7,5	-4.0	4.0	0.0	0.3	89	115	6	2
	114 F3,2	0.0	4.6	0.0	0.3	79	126	4	2
	110 W1,3	-2.7	2.7	0.1	0.3	43	195	3	2
	100 N2,1	2.9	2.9	0.2	0.3	40	100	3	2
	100 E2,1	5.9	3.6	0.0	0.3	24	94	1	1
	114 S2,11	-4.6	5.6	0.0	0.3	0	0	7	2
	114 E2,10	-3.1	5.8	0.0	0.3	-4	130	10	2
	114 F5,10	-2.7	4.2	0.1	0.3	-7	121	7	2
	114 F7,15	0.0	6.2	0.0	0.3	-15	120	4	2
	114 F12,4	-2.7	4.2	0.2	0.3	-25	116	5	2
	114 F1,15	0.0	6.2	0.7	0.4	-26	123	4	2
	110 F8,1	18.7	6.0	1.6	0.5	-34	190	5	2
	114 F9,15	0.0	6.2	0.2	0.3	-41	120	0	1
	114 N2,1	-4.6	5.6	0.0	0.3	-57	135	5	2
	114 F12,15	1.5	6.4	0.8	0.4	-90	119	2	2
	114 F9,9	3.1	6.5	-0.3	0.2	-94	124	1	2
	110 F8,15	17.4	5.8	5.7	0.9	-121	162	9	2
	114 F4,14	-7.7	5.1	0.2	0.3	-128	123	5	2
	114 F6,12	1.3	4.8	0.0	0.3	-132	116	3	2
	110 F9,14	13.4	5.3	2.7	0.6	-134	151	6	2
	110 N2,15	1.3	3.5	0.2	0.3	-177	144	2	2
	110 E1,8	6.7	4.4	0.0	0.3	-177	136	2	2
	110 X-1	2.7	3.8	1.2	0.5	-190	143	6	2
	114 F10,14	6.2	6.9	0.2	0.3	-230	123	3	2
	110 F10,2	9.3	4.8	1.7	0.5	-293	320	4	2
	114 F3,10	-5.3	3.8	0.1	0.3	-307	120	8	2
	100 F2,4	1.5	2.5	0.9	0.4	-339	110	3	2
	110 F7,3	21.4	6.3	2.8	0.6	-397	236	7	2
	114 F6,6	-4.0	4.0	0.1	0.3	-464	116	4	2
	110 F11,10	24.0	6.5	3.2	0.7	-707	225	5	2
	110 F8,6	20.0	6.1	1.6	0.5	-715	254	4	2
	110 F9,4	13.4	5.3	3.2	0.7	-1013	332	10	2
	110 F9,7	30.7	7.2	4.7	0.8	-1228	261	10	2
	110 F6,14	2.7	3.8	3.2	0.7	-6784	332	8	2
	110 S2,10	4.0	4.0	0.7	0.4	-11689	673	3	2

D.2 Fenced-In Storage Yard: Sorted by Gamma-Ray Exposure Rate

T064-2.WS

ROOM NUMBER	GRID NAME	uR/h TOTAL	uR/h STD DEV
	7,9	76	0.6
fen6,1	c	76	0.6
fen7,1	b	66	0.6
	6,9	43	0.4
fen5,1	d	43	0.4
	8,3	33	0.4
	2,7	33	0.4
fen3,1	f	32	0.4
fen4,1	e	30	0.4
	5,9	27	0.4
	8,4	26	0.3
fen8,1	a	21	0.3
	5,8	16	0.3
	7,2	16	0.3
	6,8	16	0.3
	8,5	16	0.3
	8,9	16	0.3
	6,2	16	0.3
	3,2	16	0.3
	5,2	15	0.3
	4,2	15	0.3
	8,2	15	0.3
	2,6	15	0.3
fen2,1	g	15	0.3
	8,6	15	0.3
	2,3	15	0.3
	7,1	15	0.3
	8,1	15	0.3
	6,7	15	0.3
	9,7	14	0.3
	7,8	14	0.3
	9,9	14	0.3
	9,8	14	0.3
	4,7	14	0.3
	4,8	14	0.3
	1,3	14	0.3
	8,8	14	0.3
	6,1	14	0.3
	2,8	14	0.3
	1,4	14	0.3
	5,7	14	0.3
	3,6	14	0.3
fen1,1	h	14	0.3
	7,7	14	0.3
	2,5	14	0.3
	1,8	14	0.3
	8,7	14	0.3
	1,6	14	0.3
	2,2	14	0.3
	4,1	13	0.2
	1,5	13	0.2
	2,4	13	0.2
	5,1	13	0.2
	1,2	13	0.2
	1,7	13	0.2
	2,1	13	0.2
	3,1	13	0.2
	1,1	13	0.2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3223
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4474
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3
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5
6
7
8
9
FENCE-POST SET 10ft APART (3m)
10ft
10ft
FENCh
2743
2825
3032
2374
2884
2532
2813
2945
2612
3155
2877
2534
3251
3056
2751
(GAMMA COUNTS 1m)
2802
2970
INST · 596005.
4/12/
2894
3320
3078
3051
(Expand)
2850
3329
3487
5783
Inside
Yard
Survey
Bldg. #064.
2981
3363
DU. Shine OUTSIDE
3135
3412
9237
3144
3446
2955
3121
3139
3286
7108
5596
3410
3211
2134
3018
3404
Avenac. BKG. GAMMA.
3127
3084
3104
2586 c/m.
1 JULY 1969

D.3 Surrounding 2-Acre Area: Sorted by Location

T064-3.WS

ROOM NUMBER	GRID NAME	uR/h TOTAL	STD DEV
T064	1,21	14.89	0.26
SURROUND	1,23	14.41	0.26
AREA	1,24	15.12	0.26
	2,21	14.10	0.26
	2,22	14.87	0.26
	2,23	15.22	0.27
	3,22	15.24	0.27
	3,23	15.52	0.27
	4,22	15.47	0.27
	4,23	15.03	0.26
	5,21	15.99	0.27
	5,23	15.49	0.27
	6,19	16.50	0.28
	6,20	16.05	0.27
	6,23	15.70	0.27
	7-19	17.07	0.28
	7-23	15.66	0.27
	8-18	16.93	0.28
	8-24	16.51	0.28
	8-23	15.21	0.27
	9-60	14.02	0.25
	9-13	17.98	0.29
	9-10	17.12	0.28
	9-19	16.28	0.27
	9-20	14.67	0.26
	9-21	15.54	0.27
	10-10	14.02	0.25
	10-13	17.34	0.28
	10-14	17.98	0.29
	10-15	17.64	0.29
	10-19	69.58	0.57
	10-23	15.10	0.26
	11-5	13.67	0.25
	11-8	13.71	0.25
	11-9	13.15	0.25
	11-10	13.29	0.25
	11-11	14.19	0.26
	11-12	14.64	0.26
	11-13	18.31	0.29
	11-14	20.77	0.31
	11-15	17.56	0.29
	11-16	17.62	0.29
	11-18	23.73	0.33
	11-18	109.96	0.71
	11-18	34.07	0.40
	11-20	13.55	0.25
	11-21	12.84	0.24
	12-3	14.46	0.26
	12-7	13.91	0.25
	12-8	13.43	0.25
	12-9	12.87	0.24
	12-12	14.54	0.26
	12-12	23.00	0.33
	12-13	39.35	0.43

12-14	21.22	0.31
12-14	19.48	0.30
12-15	18.98	0.30
12-16	17.44	0.28
12-17	17.81	0.29
12-18	16.33	0.28
12-18	17.83	0.29
13-2	13.98	0.25
13-7	13.51	0.25
13-8	12.95	0.24
13-13	35.35	0.40
13-14	40.13	0.43
13-14	24.49	0.34
13-14	31.91	0.38
13-15	25.47	0.34
13-15	20.46	0.31
13-16	19.49	0.30
13-17	16.99	0.28
13-17	14.38	0.26
13-18	14.28	0.26
13-20	14.31	0.26
13-21	14.53	0.26
14-1	14.16	0.26
14-2	12.63	0.24
14-3	12.77	0.24
14-4	12.87	0.24
14-6	12.32	0.24
14-7	12.68	0.24
14-15	21.75	0.32
14-16	21.79	0.32
14-17	14.06	0.26
14-17	13.81	0.25
14-17	14.29	0.26
14-18	14.74	0.26
14-19	15.38	0.27
14-21	15.51	0.27
15-2	12.39	0.24
15-4	13.16	0.25
15-5	13.16	0.25
15-5	12.92	0.24
15-6	12.97	0.25
15-6	13.57	0.25
15-7	12.22	0.24
15-16	14.40	0.26
15-17	13.72	0.25
15-18	15.26	0.27
15-20	15.20	0.27
16-2	15.15	0.26
16-3	14.41	0.26
16-4	13.57	0.25
16-5	12.41	0.24
16-6	12.41	0.24
16-6	16.22	0.27
16-7	17.68	0.29
16-8	12.16	0.24
16-15	14.18	0.26
16-16	16.48	0.28

17-2	14.43	0.26
17-3	13.57	0.25
17-5	11.58	0.23
17-6	12.08	0.24
17-7	12.34	0.24
17-8	12.15	0.24
17-9	13.27	0.25
17-15	15.14	0.26
17-17	16.76	0.28
17-19	14.94	0.26
18-4	14.11	0.26
18-5	12.26	0.24
18-6	12.96	0.25
18-7	13.00	0.25
18-8	11.75	0.23
18-10	12.59	0.24
18-13	19.08	0.30
18-14	15.32	0.27
18-18	14.80	0.26
19-2	14.20	0.26
19-3	14.28	0.26
19-6	11.85	0.23
19-7	12.22	0.24
19-8	12.75	0.24
19-9	11.81	0.23
19-10	13.25	0.25
19-11	13.29	0.25
19-12	13.81	0.25
19-13	15.28	0.27
19-16	16.15	0.27
20-3	14.79	0.26
20-4	13.90	0.25
20-6	15.26	0.27
20-7	15.11	0.26
20-8	12.60	0.24
20-8	11.28	0.23
20-9	13.71	0.25
20-10	13.47	0.25
21-15	16.32	0.27
21-17	15.12	0.26
21-4	15.13	0.26
21-5	15.74	0.27
21-6	15.40	0.27
21-7	14.81	0.26
21-8	13.78	0.25
21-15	16.03	0.27
22-3	14.56	0.26
22-7	17.59	0.29
22-8	14.79	0.26
22-11	14.99	0.26
22-12	14.93	0.26
23-4	15.06	0.26
23-5	15.61	0.27
23-8	14.66	0.26
23-9	15.05	0.26
23-10	15.00	0.26
23-11	15.06	0.26

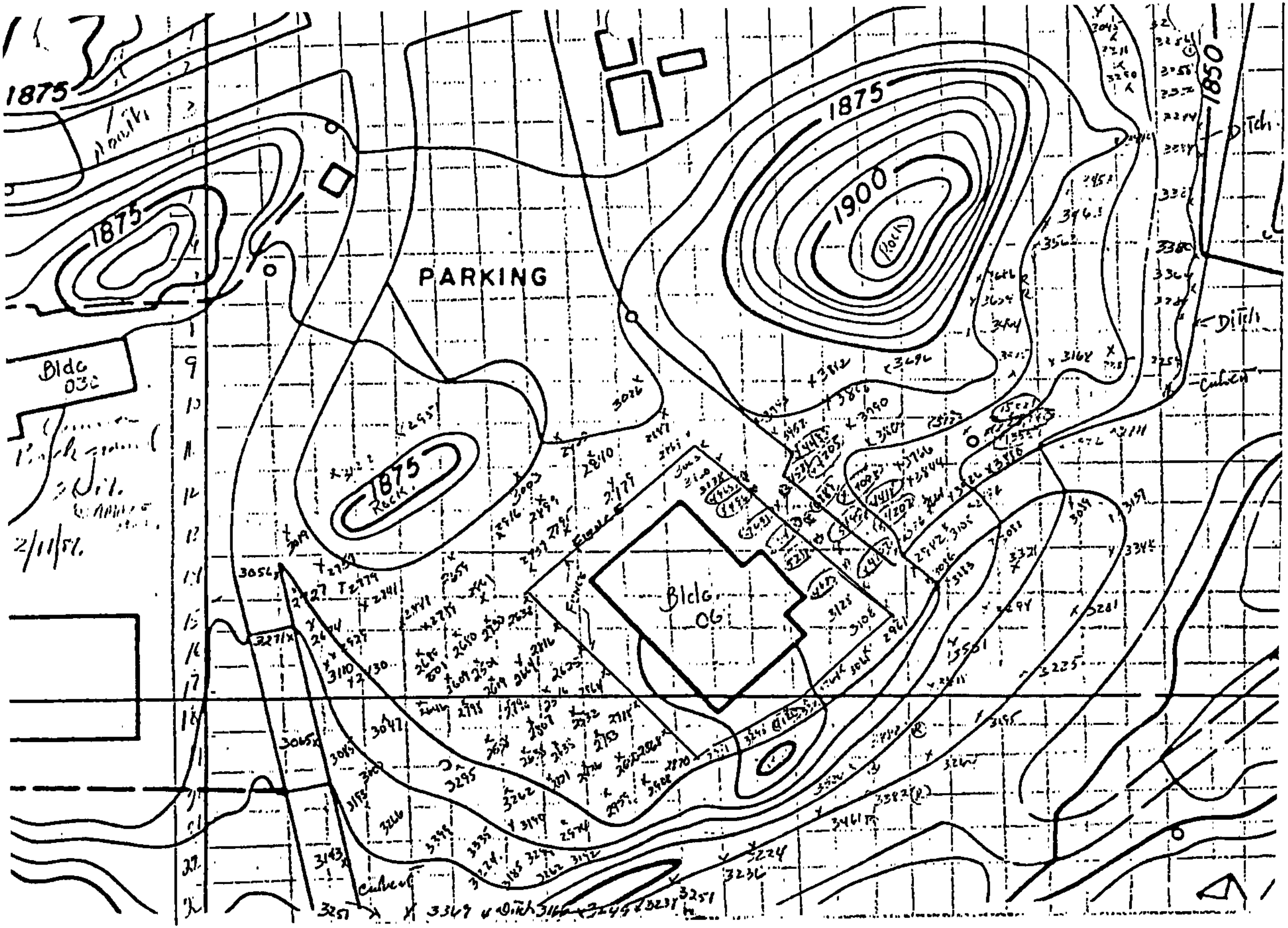
Sorted by Gamma-Ray Exposure Rate

T064-3.WS

ROOM NUMBER	GRID NAME	uR/h TOTAL	STD DEV
	11-18	109.96	0.71
	10-19	69.58	0.57
	13-14	40.13	0.43
	12-13	39.35	0.43
	13-13	35.35	0.40
	11-18	34.07	0.40
	13-14	31.91	0.38
	13-15	25.47	0.34
	13-14	24.49	0.34
	11-18	23.73	0.33
	12-12	23.00	0.33
	14-16	21.79	0.32
	14-15	21.75	0.32
	12-14	21.22	0.31
	11-14	20.77	0.31
	13-15	20.46	0.31
	13-16	19.49	0.30
	12-14	19.48	0.30
	18-13	19.08	0.30
	12-15	18.98	0.30
	11-13	18.31	0.29
	9-13	17.98	0.29
	10-14	17.98	0.29
	12-18	17.83	0.29
	12-17	17.81	0.29
	16-7	17.68	0.29
	10-15	17.64	0.29
	11-16	17.62	0.29
	22-7	17.59	0.29
	11-15	17.56	0.29
	12-16	17.44	0.28
	10-13	17.34	0.28
	9-10	17.12	0.28
	7-19	17.07	0.28
	13-17	16.99	0.28
	8-18	16.93	0.28
	17-17	16.76	0.28
	8-24	16.51	0.28
	6,19	16.50	0.28
	16-16	16.48	0.28
	12-18	16.33	0.28
	21-15	16.32	0.27
	9-19	16.28	0.27
	16-6	16.22	0.27
	19-16	16.15	0.27
	6,20	16.05	0.27
	21-15	16.03	0.27
	5,21	15.99	0.27
	21-5	15.74	0.27
	6,23	15.70	0.27
	7-23	15.66	0.27
	23-5	15.61	0.27
	9-21	15.54	0.27
	3,23	15.52	0.27

14-21	15.51	0.27
5,23	15.49	0.27
4,22	15.47	0.27
21-6	15.40	0.27
14-19	15.38	0.27
18-14	15.32	0.27
19-13	15.28	0.27
20-6	15.26	0.27
15-18	15.26	0.27
3,22	15.24	0.27
2,23	15.22	0.27
8-23	15.21	0.27
15-20	15.20	0.27
16-2	15.15	0.26
17-15	15.14	0.26
21-4	15.13	0.26
1,24	15.12	0.26
21-17	15.12	0.26
20-7	15.11	0.26
10-23	15.10	0.26
23-4	15.06	0.26
23-11	15.06	0.26
23-9	15.05	0.26
4,23	15.03	0.26
23-10	15.00	0.26
22-11	14.99	0.26
17-19	14.94	0.26
22-12	14.93	0.26
1,21	14.89	0.26
2,22	14.87	0.26
21-7	14.81	0.26
18-18	14.80	0.26
20-3	14.79	0.26
22-8	14.79	0.26
14-18	14.74	0.26
9-20	14.67	0.26
23-8	14.66	0.26
11-12	14.64	0.26
22-3	14.56	0.26
12-12	14.54	0.26
13-21	14.53	0.26
12-3	14.46	0.26
17-2	14.43	0.26
16-3	14.41	0.26
1,23	14.41	0.26
15-16	14.40	0.26
13-17	14.38	0.26
13-20	14.31	0.26
14-17	14.29	0.26
19-3	14.28	0.26
13-18	14.28	0.26
19-2	14.20	0.26
11-11	14.19	0.26
16-15	14.18	0.26
14-1	14.16	0.26
18-4	14.11	0.26
2,21	14.10	0.26

14-17	14.06	0.26
9-60	14.02	0.25
10-10	14.02	0.25
13-2	13.98	0.25
12-7	13.91	0.25
20-4	13.90	0.25
14-17	13.81	0.25
19-12	13.81	0.25
21-8	13.78	0.25
15-17	13.72	0.25
11-8	13.71	0.25
20-9	13.71	0.25
11-5	13.67	0.25
15-6	13.57	0.25
17-3	13.57	0.25
16-4	13.57	0.25
11-20	13.55	0.25
13-7	13.51	0.25
20-10	13.47	0.25
12-8	13.43	0.25
19-11	13.29	0.25
11-10	13.29	0.25
17-9	13.27	0.25
19-10	13.25	0.25
15-5	13.16	0.25
15-4	13.16	0.25
11-9	13.15	0.25
18-7	13.00	0.25
15-6	12.97	0.25
18-6	12.96	0.25
13-8	12.95	0.24
15-5	12.92	0.24
12-9	12.87	0.24
14-4	12.87	0.24
11-21	12.84	0.24
14-3	12.77	0.24
19-8	12.75	0.24
14-7	12.68	0.24
14-2	12.63	0.24
20-8	12.60	0.24
18-10	12.59	0.24
16-6	12.41	0.24
16-5	12.41	0.24
15-2	12.39	0.24
17-7	12.34	0.24
14-6	12.32	0.24
18-5	12.26	0.24
19-7	12.22	0.24
15-7	12.22	0.24
16-8	12.16	0.24
17-8	12.15	0.24
17-6	12.08	0.24
19-6	11.85	0.23
19-9	11.81	0.23
18-8	11.75	0.23
17-5	11.58	0.23
20-8	11.28	0.23


PARKING
1900
1875
1850
Rock
Bldg 06
Bldg 030
Ditch
Culvert
Fence