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ABSTRACT

A radiological survey was performed at the Sodium Disposal Facility, T886, an area located on the far west end of Rockwell International's Santa Susana Field Laboratory (SSFL). This facility was used as a disposal site for sodium and sodium-potassium alloys, and combustible materials from DOE/AEC nuclear programs such as the SRE and SNAP. Some radioactive materials were found at the facility and immediately downslope. Consequently, radioactive contamination has been suspected in immediate areas surrounding the upper and lower open-field pits, where the disposal activities took place. The purpose of this survey was to identify those areas outside the open-field pits which need further radiological inspection and could require remedial action.

The scope of this survey specifically excluded the upper and lower open-field pits, because previous measurements show these two areas to be radiologically contaminated. Only outlying areas were suspect, and thereby characterized in accordance with the site radiological survey plan. Samples were not analyzed for chemical contaminants, (Reference 18).

Surface soil samples from areas surrounding the upper and lower open-field pits were collected and analyzed by gamma-ray spectrometry and gross alpha/beta techniques for potential radionuclides. About 250 soil samples were collected and analyzed. About 1400 ambient gamma exposure rate measurements were also made in these same areas, to identify slight surface and significant subsurface contamination.

The results of this survey and analysis show that no migration or deposition of radioactive contaminants has occurred from the upper and lower open-field pits to surrounding areas. The radioactivity concentrations measured previously in both open-field pits present no health hazards, but these pits should be further investigated to determine the extent of radioactive contaminants.

1.0 INTRODUCTION

The Sodium Disposal Facility, located at Rockwell International's SSFL in the Simi Hills of Ventura County, California, was surveyed and analyzed for residual radioactive material as part of the "Radiological Survey Plan for SSFL" (Reference 4). The purpose of this survey was to radiologically characterize surrounding areas suspect of being radioactively contaminated and determine whether further investigation is required or remedial action is necessary. The upper and lower open-field pits, known to be contaminated, were excluded from this survey. However, results from prior surveys which show specific contaminated locations are included in this report. The Burn Pit is not a radiological health hazard, and certainly does not meet any state or federal requirements for maintaining as a radiologically controlled area.

The Sodium Disposal Facility is commonly called the Old Sodium Burn Pit, but is referred to as the Burn Pit throughout this document. It was used in the 1960s and 1970s for disposal of combustible materials such as sodium, NaK, and kerosene used during government nuclear programs. "Santo-wax," used as a coolant for organic moderated reactors, was also burned in this area. These programs included the SRE (Sodium Reactor Experiment) and SNAP (Systems for Nuclear Auxiliary Power). Because these materials originated from nuclear facilities, it is possible that some could have been contaminated with radioactive material. In the late 1970s, a concerted effort to clean up the Burn Pit was launched. The gate was locked, and only documented items and materials (charge number and radiation survey) were admitted. Occasionally, however, material of unknown origin was deposited at the site gate.

Most disposal activities took place in a concrete pool and two open-field pits. The concrete pool has been decontaminated on the interior surfaces. However, contamination may exist in cracks in the structure. Previous radiological surveys show that the two open-field pits are contaminated with cesium-137, a fission product, and some zirconium hydride

contaminated with U-238. Further investigation and remedial action is required in both pits; however, neither pit is a radiological health hazard in current and near future uses.

Radioactive contamination is suspect in surrounding areas because of the potential for 1) radionuclide transport and migration from each pit, particularly in the direction of surface water runoff; and 2) dispersion and scattering of radioactive material during cleanup of the site. Contaminant mobility in this area was shown from previous chemical analysis to be very small. Water sampling down the drainage path has never shown chemical or radioactive contamination. Additionally, from all previous accounts of clean up efforts taking place, no residual debris is on the surface. All barrels, scrap, and miscellaneous junk have been removed and disposed of off-site. The open-field pits and surrounding area look like an ordinary field. It was found during the CERCLA Phase II program, (Reference 18), however, that significant amounts of debris are present several feet below the surface of each pit. The radiological extent of subsurface contaminants in these pits is not well known.

The extent of this survey was to radiologically characterize about 3 acres of land surrounding the two open-field pits. About 1400 ambient gamma exposure rate measurements were taken one meter above the ground. The distance from one measurement location to the next was sufficiently short so that any suspected gamma-ray emitter would have been detected as a perturbation of the measurements. On a six meter center-to-center spacing, two-pound surface soil samples were collected. This sampling frequency resulted in about 250 samples. Each sample was dried and ground, homogenized, then split into a 450-ml sample and a 2-g sample for gamma spectrometry analysis and gross alpha/beta counting, respectively.

Ambient gamma exposure rate measurements (in micro-roentgens/hour) were plotted as a cumulative distribution function and compared against measurements acquired from a similar geological area where absolutely no radioactive materials were ever used, stored, or handled.

Each 450-ml sample of soil was analyzed for gamma emitters, including naturally present thorium and uranium, and their decay daughters; U-235; mixed fission products; and activation products, including potassium 40, which although naturally occurring may be present in increased quantity because of NaK disposal from nuclear facilities. Activity concentrations of Th-232 to U-238 were plotted to demonstrate that naturally occurring amounts are present. Cs-137 and K-40 quantities were also plotted as cumulative distribution functions. This plotting technique will show any perturbations from a Gaussian probability function and thus allow the decision to be made whether further inspection is required or not.

The U-238 activity derived from the gamma spectrometry is based on Ra-226 and daughters. No satisfactory gamma-ray specific for U-238 could be utilized. Thus, the gross alpha and beta activity would be the best test for the presence of depleted or normal/low-enrichment uranium.

Each 2-g sample was analyzed in an gas-flow, 2 π proportional counter for gross alpha and beta activity. Alpha and beta activities were converted to picocuries/gram (pCi/g) and plotted as cumulative probability distributions.

The PC-based computer software and graphics utility used to plot radiation measurements as a cumulative probability 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 technique assumes the data follow a Gaussian probability distribution function.

The Department of Energy has adopted residual radioactivity limits 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 8), Regulatory Guide 1.86, and USNRC License SNM-21 (Reference 2). However, very little

guidance/agreement is found in the literature relative to soil sample analysis and allowable radioactivity concentrations for unrestricted use. We have used limits of 30 alpha-pCi/g above background and 100 beta-pCi/g total for soil at other Rockwell nuclear facilities. The alpha limit was published in "Disposal of Onsite Storage of Thorium or Uranium Wastes from Past Operations," Federal Register Vol. 46, No. 205 (Reference 13). Additionally, the NRC has adopted a limit of 5 μR/h ambient gamma exposure rate above background, where the U.S. DOE had adopted a value of 20 μR/h above background. Because of the large variability in natural background at the SSFL site, our ambient measurements were not corrected for background. Rather, an independent "natural" background distribution is presented as a comparison against the Burn Pit data.

Extensive sample collection and analysis has been performed to radiologically characterize the Burn Pit. Although the limits, by which one may choose to demonstrate the level of contamination, are not quite clear from the regulatory literature, the radioactivity concentrations found are very low, and do not present a radiological hazard, or require remedial action outside of the two open-field pits. Within the open-field pits, further radiological assessment is required to estimate the depth and heterogeneity of radioactive contaminants.

2.0 IDENTIFICATION OF FACILITY PREMISES

2.1 Location

The Burn Pit 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 neighboring communities is shown in Figures 2.1 and 2.2 respectively. Figure 2.3 is a map showing that part of SSFL which includes the Burn Pit. The Burn Pit is not an ETEC facility and is not on DOE-optioned land. Figure 2.4 is a photo showing the Burn Pit, with ETEC in the background. The entire area enclosed in white is the extent of the radiological survey. In the photo, taken in 1968, the concrete pool and both ponds (filled with water at that time) are distinguishable.

2.2 Site Profile and Topography

Located at the far west end of SSFL, beyond ETEC, the old active portion of the Burn Pit area covers a little over an acre (50,000 ft^2). This includes both open-field pits and the concrete pad/pool. However, the surrounding area which was surveyed under this plan covered almost 3 acres (130,000 ft^2). This included substantial areas to the east, west, north and south of the Burn Pit, proper.

A single-lane, bituminous asphalt road branches off of G Street and approaches the Burn Pit from the east. This road is several hundred yards in length, and slightly grades upward. The grade eventually plateaus on approach to the facility. The pavement ends at the facility and the road continues as dirt. The site grades upward to the south and downward to the north.

Figure 2.1 Map of Los Angeles Area

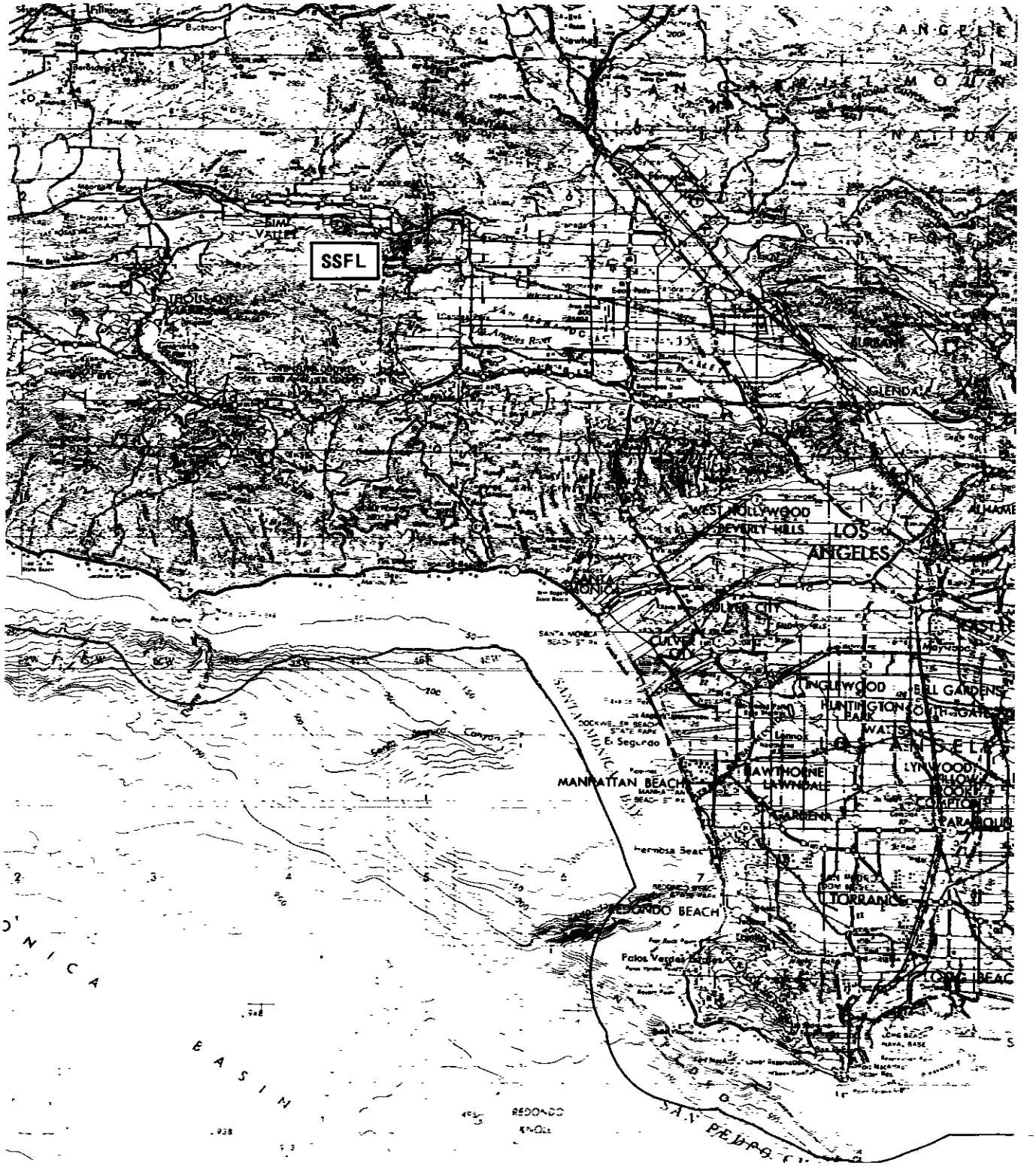


Figure 2.2 Map of Neighboring SSFL Communities

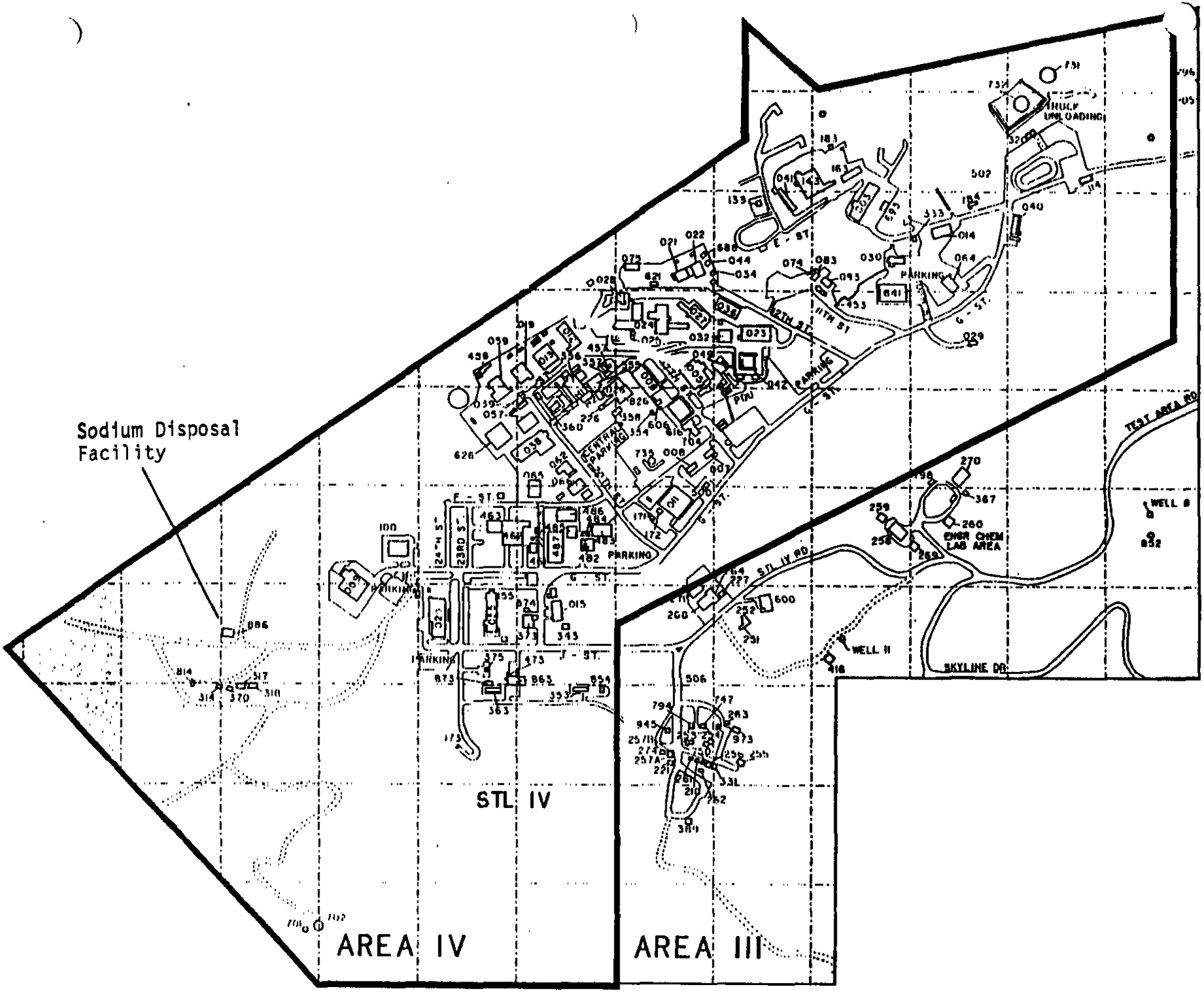


Figure 2.3 SSFL Layout

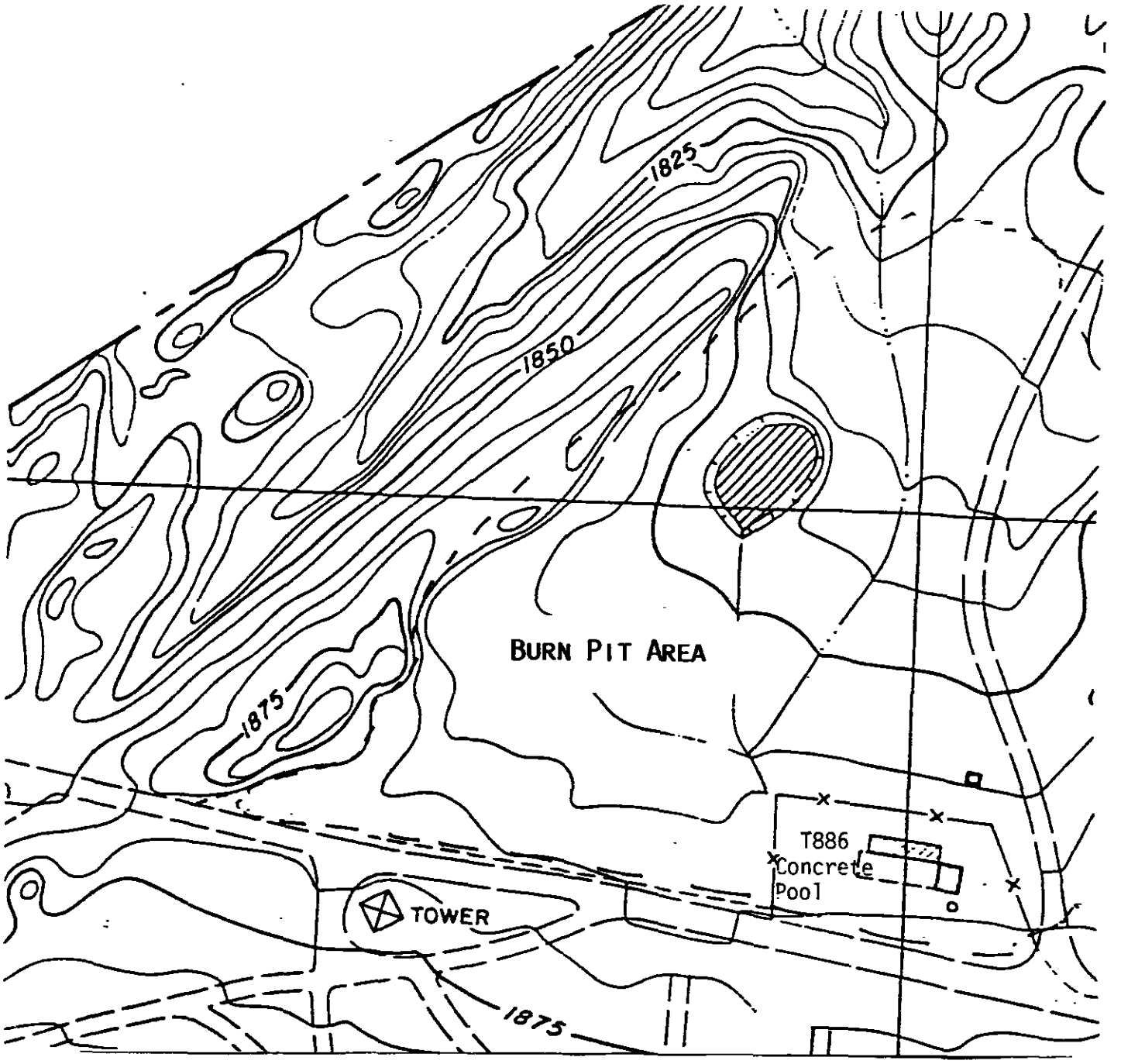
Figure 2.4 Photograph of Burn Pit

At the entrance gate is a large concrete pad with a deep, concrete pool about 10 ft wide and 40 ft long. Adjacent to the pool is a 40 ft by 16 ft concrete pad covered with iron sheets. Surrounding the conrete pad and pool is a paved area about 120 ft by 64 ft. Just north and further down grade is the upper open-field pit, about 100 ft by 100 ft. And still further north and downgrade, but adjacent to the upper pit, is the lower open-field pit also about 100 ft by 100 ft. The upper and lower open-field pits are incompletely bermed. An area adjacent to and west of the upper open-field pit was also used for storage.

The site is located on an irregular plateau in a mountain area of recent geological age sprinkled with outcroppings above the more level patches, with peripheral eroded gullies descending northerly to the Simi Valley. Running in a north/south direction west of the Burn Pit area is a large, continuous outcropping of Chico sandstone formation. Similarly, bordering on the east is a shorter formation. The elevation is about 1800 ft above sea level. Figure 2.5 shows the topography of the Burn Pit area. Figure 2.6 shows the general Burn Pit layout. The upper and lower open-field pits are designated as BPU and BPL. These pits were filled with water in the 1960s, and were referred to as ponds. Since early 1970, these "ponds" have been dry and are accordingly referred to as open-field pits.

The natural drainage from this area is north to Simi Valley. The contamination is not considered to be highly mobile since sampling of water down the drainage path has never shown chemical or radioactive contamination. Surface water flow to the north is via a dirt road east of the area and a gully on the west. The open-field pit areas are incompletely bermed, thus there is water runoff to adjacent areas. In March 1987, a trench was excavated in the west area to channel rain water around the open-field pits rather than over them. A more detailed presentation of geologic origin and likelihood for contaminant migration is given in the "CERCLA Program Phase II - Site Characterization" report (Reference 18). Refer to that report for more information about the soil porosity and permeability, and composition

Figure 2.5 Burn Pit Area Topography



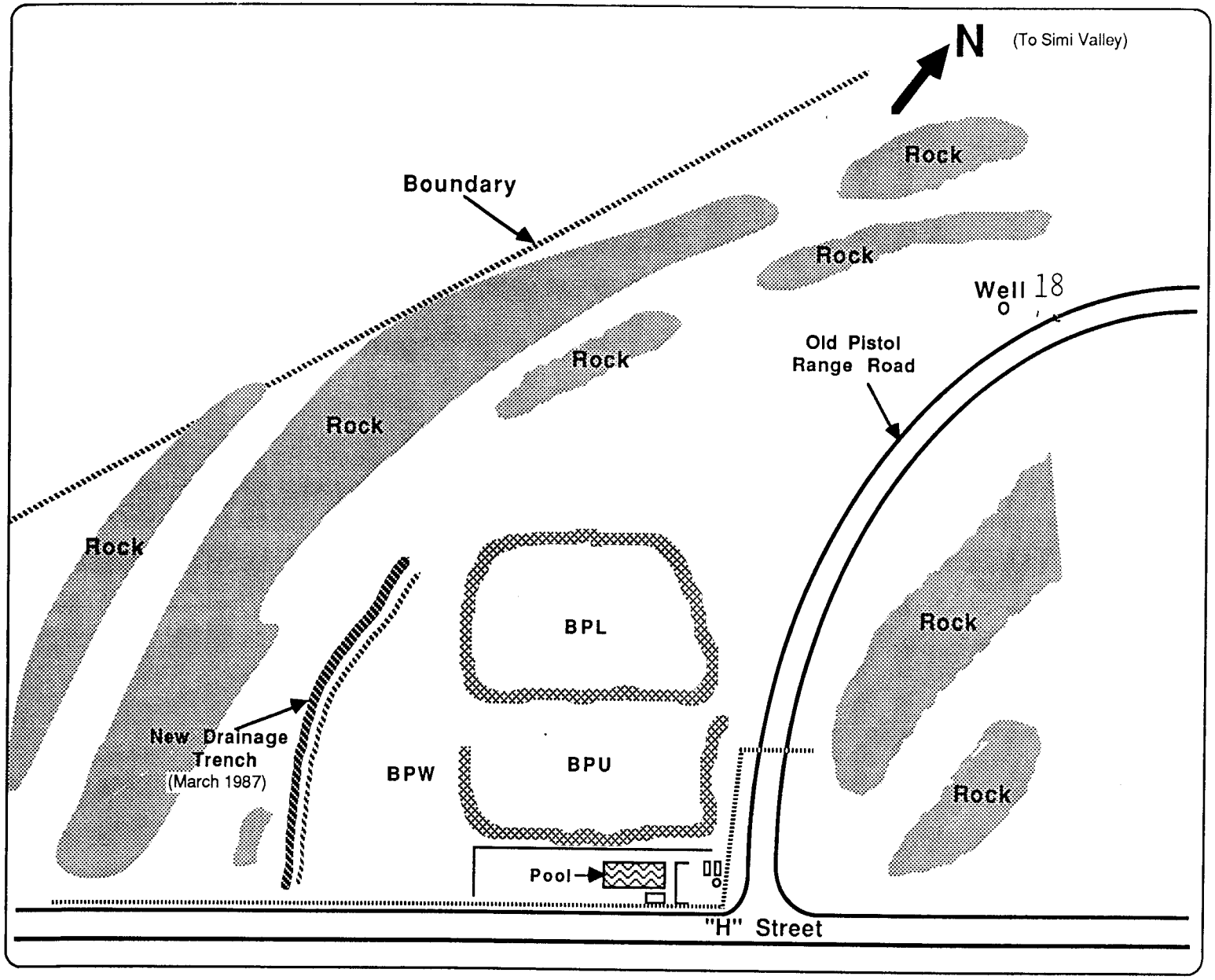


Figure 2.6. Burn Pit Area Layout

of surficial alluvium and the underlying zones of weathered sandstones and siltstone.

2.3 Historic Facility Use and Current Radiological Condition

The Burn Pit was used extensively during the 1960-1970 period for disposal of combustible materials such as sodium, NaK, and kerosene from the SRE, SNAP, and other nuclear program operations. Although the Burn Pit area is not an ETEC facility and not on DOE-optioned land, any hazardous, radioactive, or mixed wastes that may be buried in the Burn Pit Area resulted from activities of DOE's predecessor agencies.

The Burn Pit was created for the disposal of sodium and NaK by the exothermic reaction with water. After draining a system, small quantities of sodium or NaK were typically trapped in pipe elbows, valves, vessels, or insulation material. To remove the remaining reactive metal, the component was either tossed into the concrete pool or placed in the open-field pit and hosed down initially with a light spray of water and then a heavier spray as the reaction subsided. After the hose-down was complete and the reaction stopped, the items were inspected for residual material. Clean items were scrapped, and those with residual material were returned to the reaction pool. Occasionally, firearms were used on vessels to "safely" open containers to the atmosphere. Those items, after cleaning, were removed to a dumpster, usually for a scrap dealer. The facility was also made available for the open burning of any combustible material. This policy logically shifted to include just about anything that seemed undesirable for the regular trash, that would be rendered safe by burning. Terphenyl coolant for the organic-cooled reactor program, was one of these.

For the most part, residual debris was cleared and disposed of as scrap metal. Some large components and vessels were buried in place. A large batch of barrels and scrap was buried west of the area between two rock ridges, to the left of the rock ridge shown in Figure 2.6. A small

amount of material was dispersed onto surrounding terrain by explosions, even as far away as building T009.

In the late 1970s, a concerted effort to clean up the Burn Pit was launched. The gate was locked, and only documented items and materials (charge number and radiation survey) were admitted. However, occasionally material of unknown origin was deposited at the site gate.

With the construction of the new Sodium Burn Facility, T-133, and its continued operation, the accumulation of material at the Burn Pit subsided. All visible tanks were removed to the new facility for further disposition. The west burial site was excavated, hazardous materials removed, and trash hauled off. The pool was drained of water by a hazardous waste disposal company. The walls were found to be slightly contaminated with radioactive material and were scabbled clean. The open-field pits were surveyed, and the lower pit was found radioactively contaminated, and an effort to decontaminate some of the radiological "hot spots" was made. Later that year (1980), the dry lower pit was gridded, and a radiation survey was conducted. Cesium-137 was identified as the principal gamma-emitting constituent. The only other isotopes discovered at that time were primordial radionuclides. Section 7.2 presents the historical data from radiological surveys performed in each open-field pit. These pits were not currently surveyed because they are known to be contaminated from those previous surveys.

Water samples were and still are taken each rainy period, and only natural activity has been detected. After 1978, no further significant activities occurred until the March 31, 1987, CERCLA site chemical characterization study, except for periodic removal of "junk" that appeared.

During the clean-up phase from the early 1970s to mid 1980s, small pieces of debris such as pipes, elbows, machined metal parts, and tubes were dug, pushed over, and reburied by bulldozers performing clean-up tasks. This process only took place in both pits. Although still not well known,

the amount and depth of subsurface debris which exists in both pits was better characterized during the March 1987 CERCLA characterization study (Reference 18).

During this study, 23 trenches were excavated, ranging from 5 to 45 ft in length and 1 to 7 ft in depth. Figure 2.7 shows the locations of these trenches. The scope of this study was to collect and chemically analyze enough soil samples from various depths to better evaluate chemical hazards and migration patterns. To minimize the amount of radioactive material that might be sent to the SSFL Analytical Chemistry Laboratory all trenches were excavated in areas with natural background radiation levels; only samples which were not contaminated were sent to the lab. All samples were collected under the direction of a site health physicist.

As the trenches were excavated, the walls were surveyed with a portable thin-window pancake Geiger-Mueller probe and a micro-R meter to detect any radioactive contamination. Any areas with elevated readings of radioactivity were specifically not sampled. When a sample was chosen because its radioactivity levels were not detectable by a GM probe, it was analyzed by gamma spectrometry. Of the 92 samples collected, 19 samples tested positive for Cs-137 at levels greater than those expected from "natural" background. The greatest activity found in these samples of unknown weight was 200 pCi. This radiological characterization was for indication only so as to send only radiologically clean samples to the chemistry laboratory. Radiological quantification was not a goal of the CERCLA study. Contaminated areas were avoided.

While the trenches were excavated, various debris was clearly visible, along with color changes in the soil and pungent odors. Various radioactive debris was found during and previous to, trench excavation, including zirconium hydride reactor fuel end caps contaminated with U-238 and a thoriated oxygen sensor used for sodium loops. A description of what was found in each of the 23 trenches follows. Refer to Figure 2.7 for trench locations.

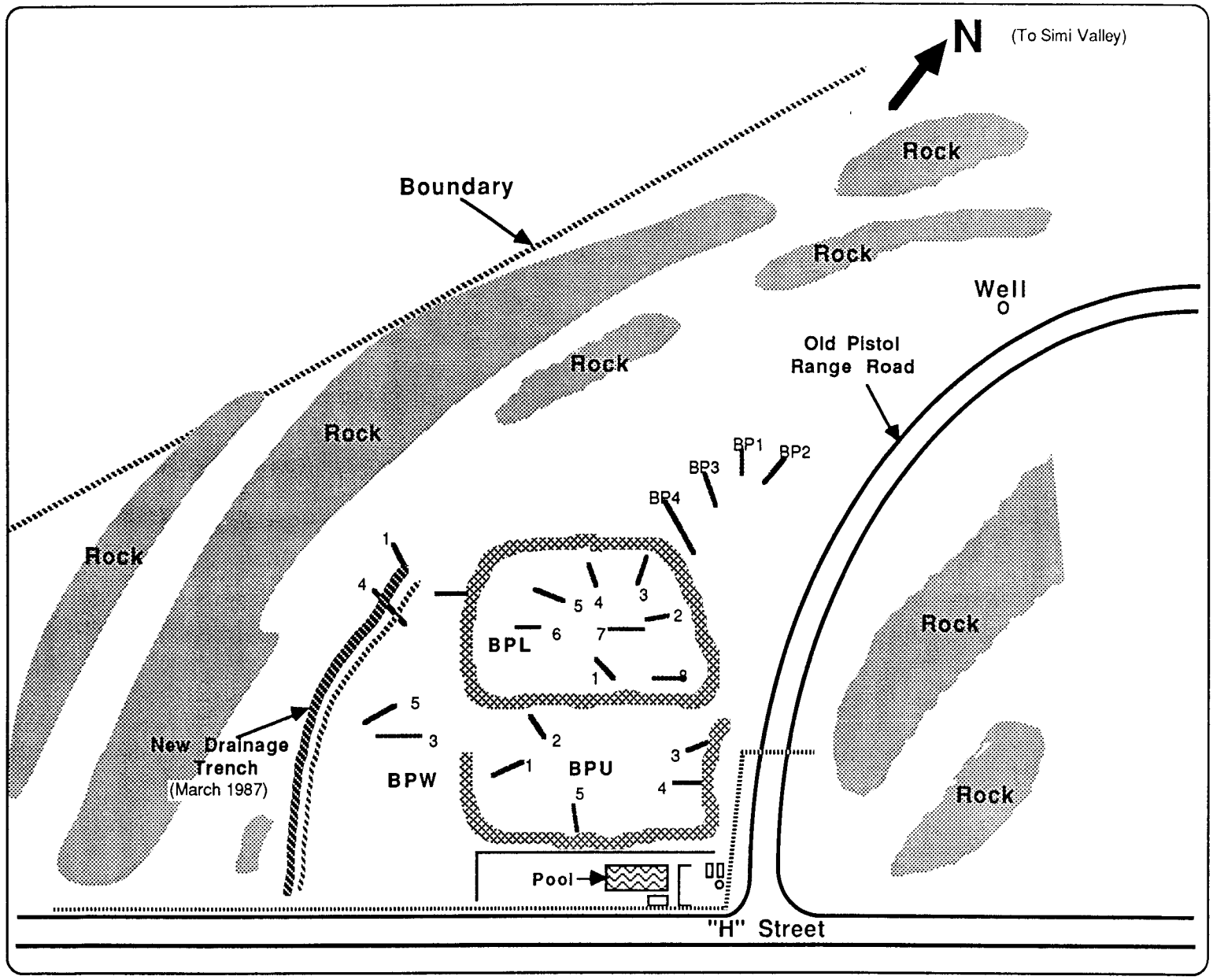


Figure 2.7. Sampling Locations for Chemical Contaminants (CERCLA Phase II March 1987)

Trench BP-1

The trench was 23.5 ft long and 6 ft deep. The soil was light brown silty clay, moderately cohesive, with pieces of concrete, pipe, and electrical wire. Five 4-oz samples were collected 11 ft from the stake, 6 ft down, (BP-1, 2, 3, 4 and 5). Chemical analysis was not performed. No radioactivity detected by portable probes.

Trench BP-2

The trench was approximately 30 ft long and 6.5 ft deep. The top 4 ft of soil was composed of light brown clayey silty sand. From 4 ft to total depth, the soil was dark brown silty clayey sand, obviously disturbed, probably bulldozed in from road construction. Samples were collected but not chemically analyzed because there were no debris and no unusual odors. No radioactivity detected by portable probes.

Trench BP-3

The trench was approximately 20 ft long and 6 ft deep. The soil was composed of medium brown cohesive sandy silty clay. At the bottom of the trench was Chatsworth Formation sandstone. The soil appeared undisturbed. No debris was found; samples were collected but not chemically analyzed. No radioactivity detected by portable probes.

Trench BP-4

The trench was approximately 45 ft long and 4 ft deep. The top 2 ft of soil was composed of medium brown silty clay with some sand. From 2 ft to total depth, the soil was light brown clayey silty sand, bottomed in Chatsworth Formation sandstone. The soil appeared undisturbed; samples were collected but not chemically analyzed. No radioactivity detected by portable probes.

Trench BPL-1 (Lower Cell)

The trench was approximately 8 ft long and 5 ft deep. The soil at the surface was stained dark and had radioactivity slightly higher than background levels. The top 1.5 ft of soil was light brown clayey sandy silt. From 1.5 ft to total depth, the soil was a slightly moist, dark brown clay. At the contact of the light brown soil and the dark brown clay was a lens of dark stained material with radioactivity higher than background levels. A 16-oz glass jar sample was collected at the surface from the dark stained material (BPL-1 surface). A 16-oz glass jar sample was collected from a depth of approximately 3 ft (BPL-1-3). A VOA sample was collected from 0.5 ft below the surface and another from approximately 3 ft below the surface (BPL-1 6"-12" and BPL-1 3'-3.5'). A lens contaminated with Cs-137 was found with total radioactivity of 200 pCi. The exposure rate in this area as 80 μR/h. A pancake GM read 1300 cpm.

Trench BPL-2

The trench was approximately 8 ft long and 6.5 ft deep. The soil was darkly stained at the surface. Glass jar and VOA samples were collected at the surface (PBL-2 surface and BPL-2 6"-12"). The soil was mottled medium to dark brown, very moist and cohesive silty clay. Metal components were found at all depths. A very strong organic odor was observed in the dark brown portions of the clay. VOA samples were collected at 1.5 ft below land surface and at 5.5 ft below land surface (PBL-2 1.5' and BPL-2 5.5'-6'). Another glass jar sample was also collected at 5.5 ft below land surface (BPL-2 5.5'). At this depth, the radiation was approximately equal to background levels. Cs-137 detected in some soil samples.

Trench BPL-3

The trench was approximately 8 ft long and 6 ft deep. VOA and 16 oz glass jar samples were collected at the surface (PBL-3 6-12", BPL-3 surface). The top 0.5 was light brown sandy clayey silt with radioactivity

reading above background levels. From 0.5 to 2.5 ft, the soil was medium brown silty clay. From 2.5 to 5.5 ft, the soil was dark brown cohesive silty clay. At 3.5 ft, VOA and glass jar samples were collected (BPL-3 3.5', BPL-3 3.5'). From 5.5 to 6 ft, the soil was light brown sandy silty clay, odorless, apparently undisturbed. Gamma radiation levels were 2 to 4 times background. Beta radiation levels were up to 14 times background at 1 foot. Cs-137 detected in some soil samples.

Trench BPL-4

The trench was approximately 8 ft long and 5 ft deep. VOA and glass jar samples were collected at the surface (BPL-4 6"-12", BPL-4 .5'-1'). From 0 to 1.5 ft, the soil was a dry, medium brown clayey silt. From 1.5 to 5 ft, the soil was a cohesive medium brown silty clay. VOA and glass jar samples were collected at 4.5 to 5 ft below land surface (BPL-4 4.5'-5', BPL-4 4.5'-5'). Gamma radiation levels twice background. Cs-137 detected in some soil samples.

Trench BPL-5

The trench was approximately 8 ft long and 5 ft deep. VOA and glass jar samples were collected at the surface (BPL-5 6"-12", BPL-5 surface). The soil in the top 1 ft was a cohesive medium brown sandy silty clay. From 1 to 5 ft, the soil was a medium reddish brown sandy silty clay. VOA and glass jar samples were collected at 4 ft below land surface (BPL-5 4.0', BPL-5 4'). No radioactivity detected by portable probes. Cs-137 detected in some soil samples.

Trench BPL-6

The trench was approximately 6 ft long and 3 ft deep. VOA and glass jar samples were collected at the surface (BPL-6 6"-12", BPL-6 surface). The soil was a slightly moist medium brown silty clay. No

components were unearthed, and no additional samples were taken. Slight beta radiation detected. Cs-137 detected in some soil samples.

Trench BPL-7

The trench was approximately 8 ft long and 6 ft deep. VOA and glass jar samples were collected from the surface (BPL-7 6"-12", BPL-7 surface). The soil in the top foot was dry light brown silty sand. From 1 to 3.6 ft, the soil was dark brown silty clay with metal components. At 3 ft, a black lens with metal components was observed, and VOA and glass samples were collected (BPL-7 3'-3.5', BPL-7 3'-3.5'). A light grey lens was observed from 4 to 5 ft. From 5 to 5.5 ft, the soil was light brown silty sandy clay, probably weathered Chatsworth Formation. From 5.5 to 6 ft, light brown silty sandstone (Chatsworth Formation) was encountered. Slight beta radiation detected. Cs-137 detected in some soil samples.

Trench BPL-8

The trench was approximately 8 ft long and 5 ft deep. VOA and glass jar samples were collected at the surface BPL-8 6"-12", BPL-8 surface). The top foot of soil was composed of very light brown silty clay. From 2.5 to 3.5 ft, the soil was a cohesive, moist, medium brown, silty clay. From 2.5 to 3.5 ft, the soil was a cohesive dark grey, sandy clay with slight hydrogen sulfide odor. VOA and glass jar samples were collected at 3 ft (BPL-8 3.0'-3.5', BPL-8 3.0-3.5'). From 3.5 to 4 ft, the soil was composed of a slightly moist, medium brown, silty sandy clay; from 4 to 5 ft was a light brown silty sand with Chatsworth Formation sandstone at the bottom. No radioactivity detected by portable probes. Cs-137 detected in some soil samples.

Trench BPU-1 (Upper Cell)

The trench was approximately 4 ft long and 1 ft deep. The very shallow soil was composed of medium brown silty sand with Chatsworth

Formation silty sandstone underneath. No samples were collected because of the shallowness of the soil. No radioactivity detected by portable probes.

Trench BPU-2

The trench was approximately 5 ft long and 1.5 ft deep. A glass jar sample was collected at the surface (BPU-2 surface), but not chemically analyzed. The shallow soil was composed of very light brown, dry, silty sand with Chatsworth Formation silty sandstone underneath. No radioactivity detected by portable probes.

Trench BPU-3

The trench was approximately 5 ft long and 3 ft deep. VOA and glass jar samples were collected from the surface (BPU-3 6"-12", BPU-3 .5'-1'). The soil in the top 0.5 ft was mottled, light to medium brown, silty sand with some dark stains on the surface. From 0.5 to 2 ft, the soil was a medium brown silty clay with metal components. From 2 to 2.5 ft, the soil was a light brown, silty clay. A glass jar sample was collected of a white, crystalline substance found at 2 ft (BPU-3 2'). At 2.5 ft, VOA and glass jar samples were collected (BPU-3 2.5', BPU-3 2.5'). From 2.5 to 3 ft, the soil was a medium brown clay. No radioactivity detected by portable probes. Cs-137 detected in one out of five samples.

Trench BPU-4

The trench was approximately 8 ft long and 3 ft deep. A glass jar sample was collected at the surface. The top 0.5 ft of soil was light brown, silty sand. From 0.5 to 3 ft, the soil was a medium brown, silty clay with some small areas of white crystalline powder. At 3 ft, Chatsworth Formation light brown silty sandstone was encountered. Glass jar and VOA samples were collected at 3 ft (BPU-4 3', BPU-4 3'). No radioactivity detected by portable probes. No Cs-137 in one of one sample.

Trench BPU-5

The trench was approximately 5 ft long and 3 ft deep. A glass jar sample was collected from the surface (BPU-5 surface). The top 1.5 ft of soil was a light brown, silty sand. From 1.6 ft to total depth was dark brown silty clay, underlain by the light brown silty sandstone of the Chatsworth Formation. No radioactivity detected by portable probes. No Cs-137 in two of two samples.

Trench BPU-6

The trench was approximately 7 ft long and 4.5 ft deep. The trench was cut into the berm between the upper and lower cells, and no components were unearthed. A glass jar sample was collected at the surface (BPU-6 surface). The soil was composed of a dark brown silty clay. No radioactivity detected by portable probes. No Cs-137 detected in one of one sample.

Trench BPW-1 (Western Cell)

The trench was approximately 35 ft long and from 1 to 2.5 ft deep. The soil was a medium brown, sandy silty clay underlain by weathered and unweathered Chatsworth Formation. The soil appeared undisturbed, and no samples were collected. No radioactivity detected.

Trench BPW-2

The trench was approximately 21 ft long and 5 ft deep. A glass jar sample was collected at the surface (BPW-2 surface). The soil from the surface to 4 ft below the surface was medium brown, dry cohesive silty clay with darker mottling and some components. At 1.5 ft below the surface, a whitish substance was observed. At 4 ft, VOA and glass jar samples were collected (BPW-2 4'). From 4 to 5 ft, the soil was dry medium reddish brown, silty clay with no mottling. No radioactivity detected.

Trench BPW-3

The trench was approximately 30 ft long and 5 ft deep. Metal components and barrels were exposed during excavation. The soil was very mottled and consisted of mixed grey clay, weathered Chatsworth Formation silty sand, and some medium brown silty clay. The soil was saturated with water in places. There was a natural organic odor. There was rust staining in the soil from numerous pipes and flattened barrels. VOA and glass jar samples were collected at 4.5 ft (BPW-3 4.5', BPW-3 4.5'). A glass jar surface sample was collected approximately 10 ft southeast of BPW-3 in some black, possible oil stained soil (BPW-3 surface). No radioactivity detected.

Trench BPW-4

The trench was approximately 35 ft long and from 1.5 to 3 ft deep. The soil appeared to be undisturbed, dry cohesive, medium brown silty clay underlain by light brown, silty sand Chatsworth Formation. No samples were collected. No radioactivity detected.

Trench BPW-5

The trench was approximately 20 ft long and 4 to 5 ft deep. The soil was a rust stained medium brown, silty clay. Large tanks and other components were exposed during excavation. VOA and glass jar samples were collected 3 ft below the surface (BPW-5 3', BPW-5 3'). No radioactivity detected.

The CERCLA phase II report (Reference 18) stated that the greatest ambient gamma exposure rate in the trenches was 80 μR/h in a background field of 15 μR/h, and that the greatest Cs-137 activity concentration in a sample was 200 pCi/g (Reference 18, p. 12). Because this chemical characterization study deliberately avoided sample collection if radiation levels were above background, this value of 200 pCi/g is lower than what may be

expected in the lower pit. From this CERCLA study, we know that chemical contaminants are present and that subsurface debris exists in significant quantities within both open-field pits. Although there is some junk and debris buried outside of the two pits, no radioactivity was detected on the western or northern sides (BP1-4, BPW1-5). We do not know accurately the extent of subsurface radionuclide contamination in each open-field pit because of the limitations imposed on the CERCLA characterization study. However, increased levels of ambient gamma exposure rates are observable at the surface of each pit. Results of previous radiologic measurements are presented in Section 7.2.

3.0 SURVEY SCOPE

Areas surrounding the upper and lower open-field pits were radiologically characterized by measuring ambient gamma exposure rates 1 meter above the surface, and by acquiring surface soil samples for analysis by gamma spectrometry and gross alpha/beta activity. About 1400 ambient gamma exposure rate measurements were made, each with a 1 min. counting time. About 250 surface soil samples were collected and split into a 450-ml sample for gamma spectrometry and a 2-g sample for alpha/beta counting. Ambient gamma exposure rates are reported in micro-roentgens per hour ($\mu R/h$). Radionuclide activity concentrations and gross alpha/beta activity are reported in picocuries per gram (pCi/g). Each sampling location was specified by grid notation.

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. 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 13). These appear to be the best, most appropriate and realistic limits, and compare quite favorably to DOE's "factor of 3 above background per $100m^2$ area" recommendation (Reference 1, Section C.1). Absolutely 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 the Burn Pit does not warrant this type of detailed analysis.

Table 3.1 Burn Pit Maximum Acceptable Contamination Limits

Criteria	Alpha	Beta
Ambient Gamma Exposure Rate*	5 μR/h above background	
Soil Activity Concentration**	46 pCi/g	100 pCi/g
Water Activity Concentration***	$1x10^{-4}$ μCi/ml	$1x10^{-5}$ μCi/ml

* Although DOE Guide (Ref. 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 plus that contribution from naturally occurring radioactivity, (about 26 pCi/g from Reference 15, p. 66, and determined to be 16 pCi/g from soil samples collected at the Burn Pit). We use the most conservative value. The total beta activity concentration limit is 100 pCi/g, including background (Ref. 13).

*** The most restrictive alpha/beta water radioactivity concentrations for restricted area taken from 10CFR20, Table 1, Column 2. Alpha corresponds to Pu-239, beta to Sr-90.

Three specific action levels were established during the survey. This is a proactive action level which is 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 Burn Pit radiological survey, it was sectioned into 4 areas: north, east, west, and south of the open-field pits. The north area comprised about 2280m^2; east, 2100m^2, west 3730m^2, and south 3880m^2. Total survey area amounts to about 3 acres. Figure 3.1 shows this sampling scheme. Because of the large amount of area to be surveyed, a surface soil sample was collected in each 6-meter-square area (36m^2). Gamma exposure rate measurements were made in each 3-meter-square area (9m^2). The west area, because of its operational history, was more suspect for containing residual radioactive material. The northern area, because it is downslope from the open-field pits is also more suspect.

Wood stakes were pounded in the ground every 3 meters, column wise and row wise. The resulting matrix-grid allowed each sample square to be identified in matrix notation with codes indicating the area (n = north, e = east, w = west, s = south) and a two figure Cartesian coordinate indicating the distance in meters from a local benchmark. Since one soil sample is collected in 1m^2 out of 36m^2, this corresponds to a 3% sampling frequency, minimum. Gamma measurements are made at an 11% sampling frequency. The sampling inspection plan that was used is based upon a uniform 6-meter-square grid superimposed on a uniform inspection area for soil samples, and a 3-meter-square grid for gamma measurements. Radiological conditions and physical surroundings were similar in all areas.

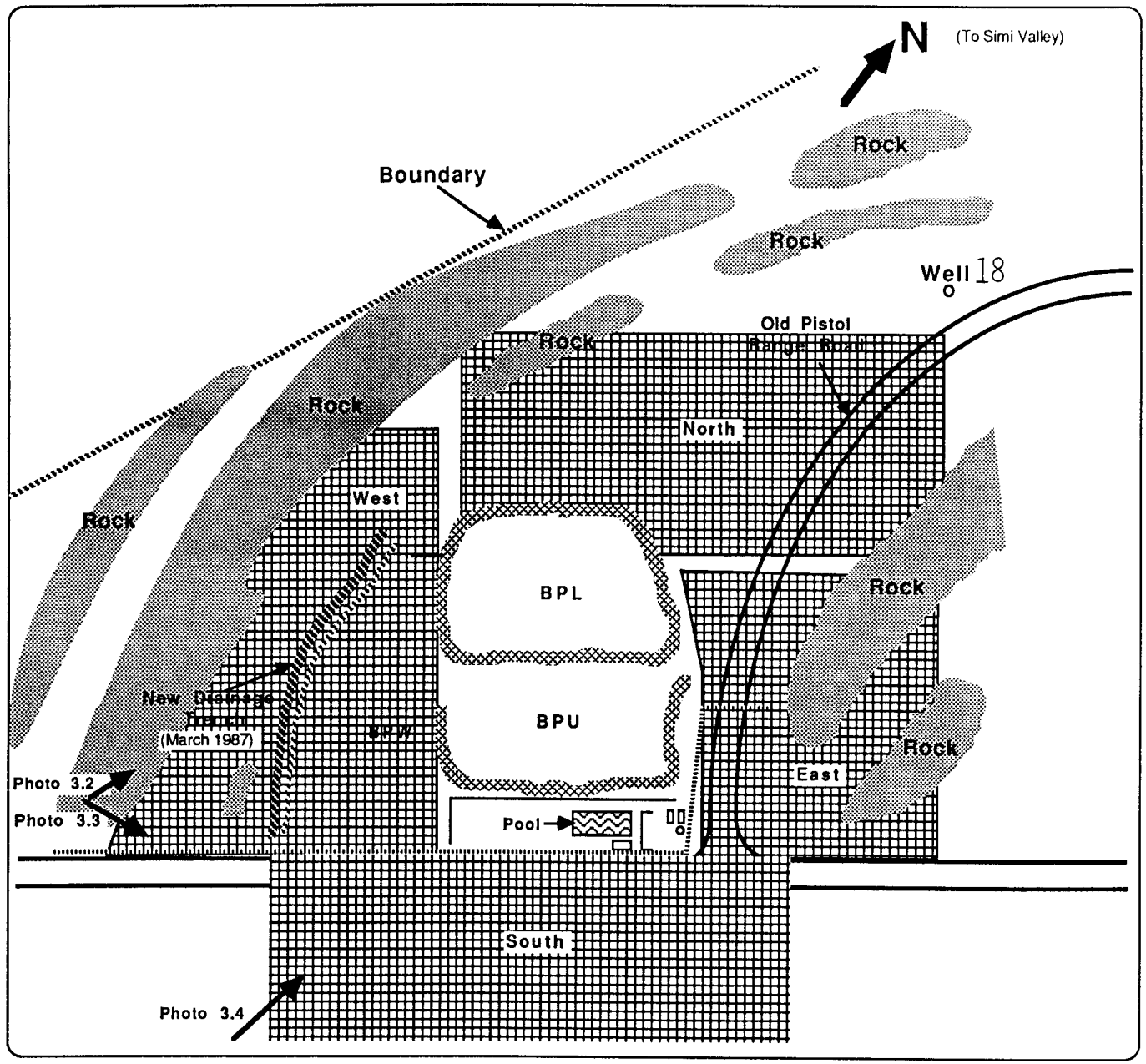


Figure 3.1. Burn Pit Area Sampling Lot

This gridding technique is pictured in Figures 3.2, 3.3, and 3.4 The location and direction in which each photograph was taken is identified in Figure 3.1. Figure 3.2 shows the Burn Pit area looking east-northeast from the top of the westernmost rock outcropping. From the bottom of the figure is the west area staked out on a 3-meter-square grid; followed by the lower and upper open-field pits (on the left and right, respectively); and the east area beyond that, fully gridded. ETEC is in the background. Figure 3.3 shows a portion of the west area and the entire south area on a slight embankment. The building in the background is not part of the Burn Pit. Figure 3.4 shows the entire area looking northerly. The south area is separated from the remaining area by a dirt road. The concrete pool and vacant storage building is on the right. The open-field pits are just below the parked trucks.

3.3 Ambient Gamma Exposure Rate Measurements

In each $9m^2$ cell, 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 scaler, at 1 m from the ground. In each cell, a 1-min. count was collected and converted to $\mu R/h$. The measurement location and exposure rate were recorded in tabular form. About 1400 1-min. measurements were acquired.

3.4 Surface Soil Samples

A 2-lb surface soil sample (no greater than 3" deep) was collected from one spot per $36m^2$. The area of the scoop normally covered about $1m^2$. Each sample location was identified and marked on the 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

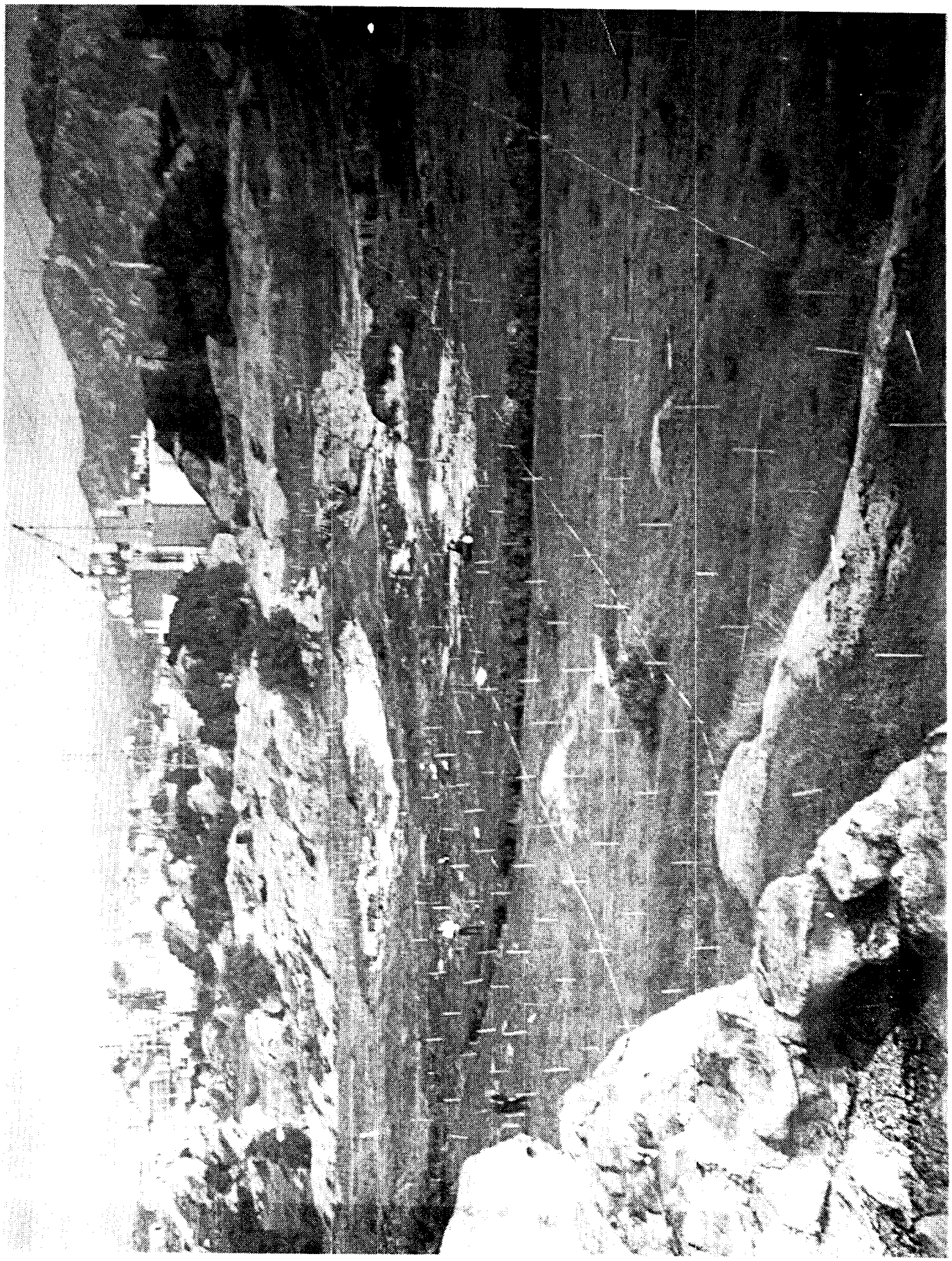


Figure 2-2 Burn Pit Area Photo Looking East

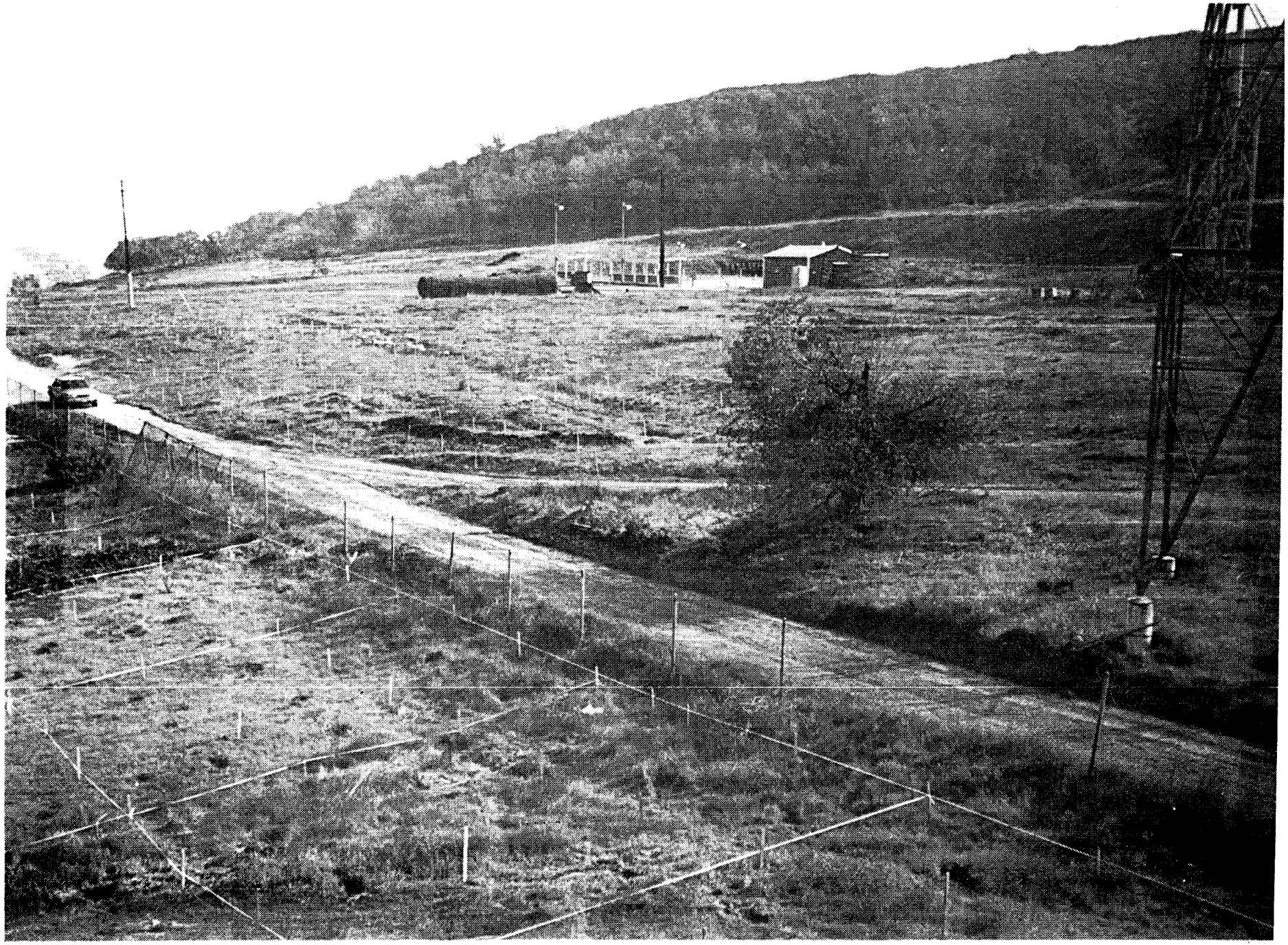


Figure 3.3 Burn Pit Photo Looking South-East

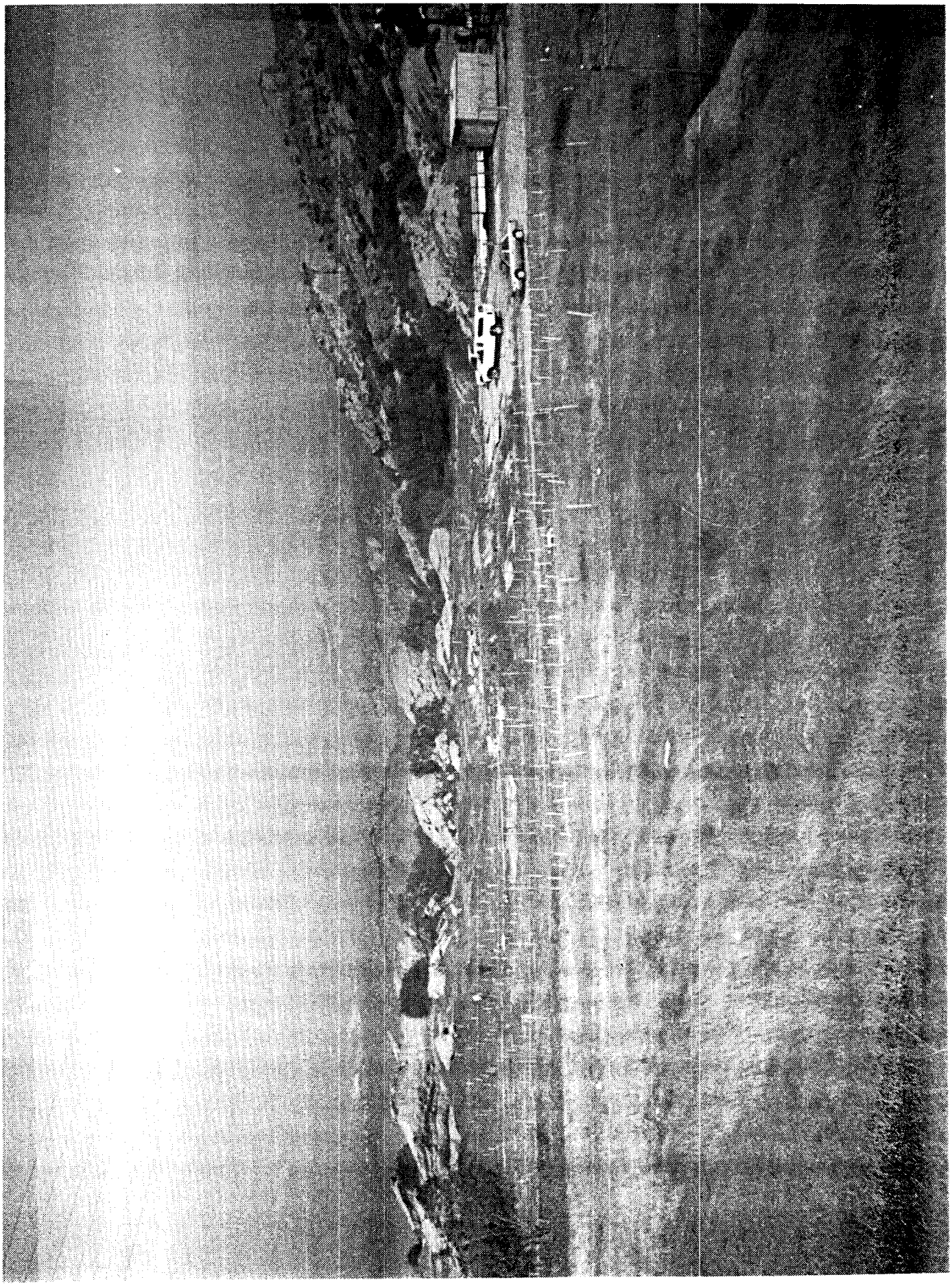


Figure 2-1 Plant Plot Photo Looking North

2" diameter aluminum planchet, and then counted for gross alpha/beta activity. About 250 surface soil samples were analyzed in this manner.

3.5 Goals and Limitations of Survey Scope

The goal of the Burn Pit radiological survey is to determine if radioactive contamination exists to such an extent that further surveying or remedial action is warranted. The survey scope is specifically limited to suspect areas surrounding the open-field pits and concrete pool, to determine if any radionuclide transport/migration has occurred on the surface or whether any gamma-emitting subsurface debris is present. Because surface transport of contaminants is the predominant process, surface sampling gives the best indication for subsequent spread of contamination. It is unlikely that subsurface transport is very significant. The amount of debris scattered beyond the two pits was minimal. No attempt to characterize subsurface contamination was made.

Because of the large area surveyed, a soil sample was collected in every $36m^2$. Although this may not appear to be a very thorough sampling plan, by applying Lot Tolerance Percent Defective techniques, we can determine with a statistical confidence of 0.90, that there is a probability of 90% that radioactive contamination does not exceed some predetermined acceptance limit. This determination varies inversely to the number of samples taken. This technique, along with the graphical representations of cumulative distribution functions will identify trends, anomalies, outliers, and perturbations in the radiation levels. We are able to conclude whether:

1. Any surface migration or dispersion of radioactive materials has occurred; and

2. Any relatively intense gamma-emitting debris is buried (see Section 5.4.4).

We can not conclude whether:

1. Any slight subsurface migration has occurred; or

2. Any buried debris with low intensity radiation is present.

The likelihood for occurrence of the above two conditions is small. First, migration periods of contaminants below the surface are typically very long. It is much easier for surface water flowing downslope to carry with it any contaminants. The settling out of these contaminants into the subsurface also takes a long time. Second, most burial activities took place in the open-field pits. Historically, any contaminated pieces of junk excavated from an area were detectable at the surface; the radiation was fairly intense. One final limitation is that we do not know the extent of contamination on the surface and subsurface of the two open-field pits; this characterization was beyond the scope of this survey.

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:

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

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:

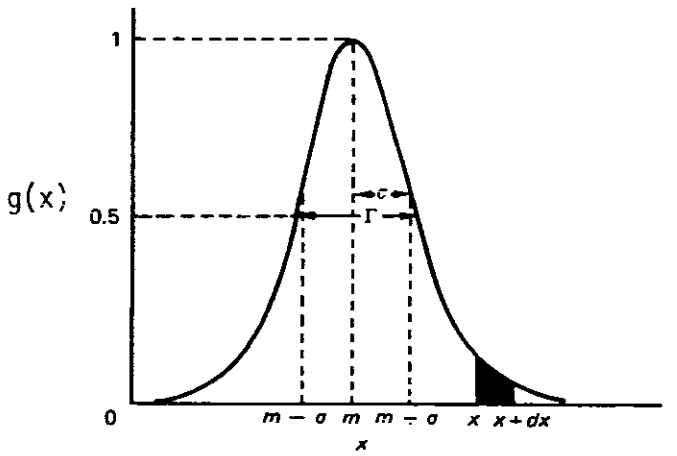


Figure 4.1 The Gaussian Probability Density Function

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:

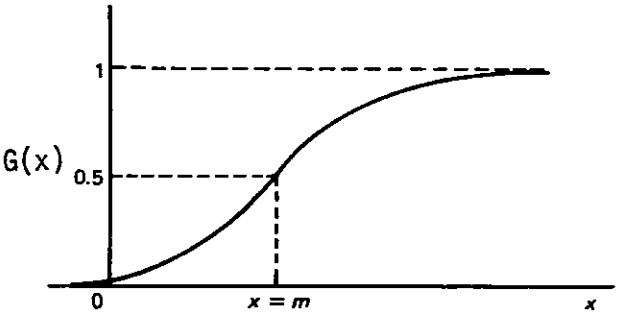


Figure 4.2 The Gaussian Cumulative Distribution Function

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 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 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.

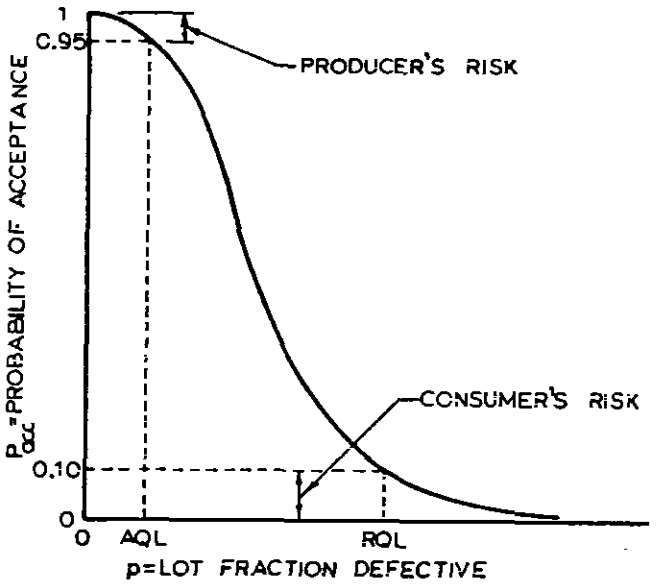


Figure 4.3 Operating Characteristics Curve

The value of k, and thus the value of $\bar{x}$ + 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 9):

$$k = \frac{K_2 + \sqrt{K_2^2 - ab}}{a}; \quad a = 1 - \frac{K_\beta^2}{2(n-1)}; \quad b = K_2^2 - \frac{K_\beta^2}{n} \qquad \text{(Eq. 4.8)}$$

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, $K_\beta = 1.282$)

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.

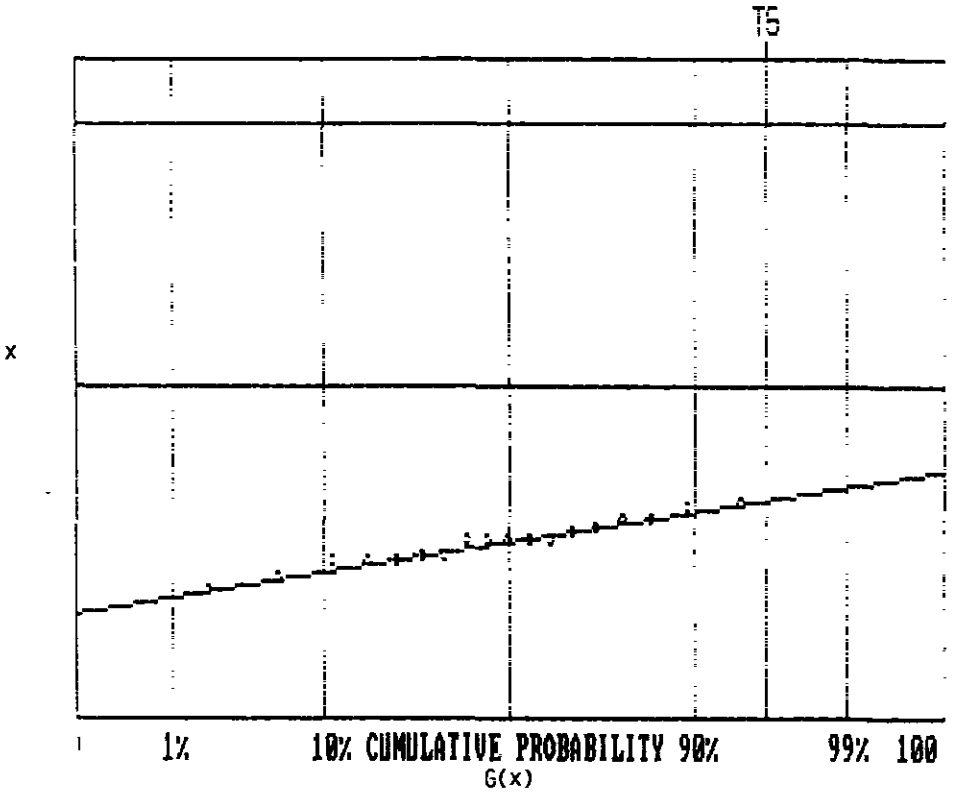


Figure 4.4 Gaussian cdf Plotted on Probability-Grade Paper

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 $\overline{x}$ + 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, $\overline{x}$ + 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, $\overline{x}$ + 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 ($\overline{x}$+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.

2) Collect additional measurements: If the test statistic ($\overline{x}$+ks) is greater than the limit (U), but $\overline{x}$ itself is less than U, independently resample and combine all measured

values to determine if $\overline{x}$+ks <= U for the combined set; if so, accept the region as clean. If not reject the region.

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

5.0 ANALYTICAL TECHNIQUES

The statistical methods presented in Section 4.0 were used to judge whether an area is slightly contaminated, contaminated above acceptance limits, or whether additional investigation is required. That decision is based on several radiological measurements:

1) Ambient gamma exposure rate;
2) Gross alpha/beta activity in soil samples; and
3) The presence and quantity of gamma emitters in soil samples (these include primordial, cosmogenic, fission, and activation products).

The analytical techniques used to acquire, evaluate, and interpret these radiological measurements are presented in detail in this section. This includes calibration corrections, 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 a measure of gamma exposure rate was made and a 2-lb soil sample was collected. Each square grid was outlined by wood stakes placed in the ground 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 soil discoloration, debris, crevices or cracks in the soil. The use of a predetermined grid with discretion for the exact location provides a uniform survey biased towards the high end of the distribution. Locations of noticeably greater exposure rates or radionuclide concentrations were always surveyed again.

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 exposure rates, and alpha/beta activity per gram. Software was developed in a program language called Quick Basic to read data from a SMART file into a graphics program which plots the radiological measurements against the 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).

The input for this data reduction was:

1) Grid location, ex. W (10,6) (W = West, E = East, N = North, S = South)
2) Ambient Gamma Exposure Rate (counts in 1 min.)
3) Alpha Counts in 2 g soil sample (counts in 30 min.)
4) Beta counts in 2 g soil sample (counts in 30 min.)
5) U-238 μCi per sample
6) Th-232 μCi per sample
7) K-40 μCi per sample
8) Cs-137 μCi per sample

Output for Gaussian plots:

1) Ambient gamma exposure rate and standard deviation (μR/h)
2) Alpha Activity in soil and standard deviation (pCi/g)
3) Beta Activity in soil and standard deviation (pCi/g)
4) U-238 Activity (pCi/g)
5) Th-232 Activity (pCi/g)
6) K-40 Activity (pCi/g)
7) Cs-137 Activity (pCi/g)

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 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 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.

5.4.1 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 a 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 $\pm$ 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.

5.4.2 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 5-1 and 5-2. Gamma scintillations produced by a NaI detector were converted from gross counts to exposure rate (μR/h) by:

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

where R = exposure rate (μR/h)
C = gross counts in 1 min
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 measureable natural background exposure rates approaches the NRC acceptance limit of 5 μR/h. Rather it was more meaningful to measure an area where no

cradioactive materials were ever handled, and then compare that gross distribution with the one under study.

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

$$s = \frac{\sqrt{C} * (EF)}{1 \text{ min}} \qquad \text{(Eq. 5-2)}$$

5.4.3 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. The test statistic, $\bar{x}$ + ks, was also calculated for the lot.

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.

In Section 7.3, where the ambient exposure rate results are presented, the Burn Pit distribution of measurements is compared against four 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 at the Burn Pit.

5.4.4 Sensitivity of Gamma Exposure Rate Measurements

The purpose of performing these measurements is to detect any significant quantity of gamma-emitting radionuclides. Operational history and surveys performed years ago show that the most significant radiological contaminant in the open-field pits is Cs-137. Therefore, this is the primary contaminant we would be looking for outside of the open-field pits. Since Cs-137 is a gamma emitter, it is detectable with the NaI detector.

The sensitivity of these measurements, or rather, the amount of contamination which could be there and which we wouldn't see, is based on two possibilities:

1) A uniformly contaminated region of soil; maybe a layer on the surface, or a layer several feet below the surface; or

2) A piece of contaminated debris located on the surface or buried several feet below.

Our acceptance criteria specify that no soil activity exceeding 100 pCi/g-beta is acceptable for unrestricted use. In comparison, 10 μCi of Cs-137, total, is the limit for exempt quantity according to 10CFR20, Appendix C.

The ambient gamma background radiation is about 10-15 μR/h at 1 meter from the ground, so the source material would have to produce an exposure rate of at least 5 to 10 μR/h in order to detect it. Table 5.1 shows theoretical exposure rates calculated for some uniformly contaminated soil and miscellaneous contaminated debris. The contaminant is assumed to be Cs-137. Condition (1) assumes a uniformly distributed layer of soil with

100 pCi/g Cs-137. Condition (2) assumes a point source of Cs-137 with total activity equal to 1 mCi.

Table 5.1 Exposure Rates of Cs-137 Contaminated Soil and Debris

(1) Contaminated Soil (100 pCi/g)	Exposure Rate (μR/h) 1 meter above surface	
Infinite Slab on the Surface		
0.3 meters thick	72	
1 meter thick	74	
Infinite Slab, 20 cm thick/10 cm thick		
at Surface	68	55
at 5 cm depth	32	25
at 10 cm depth	17	13
at 15 cm depth	9	7
at 30 cm depth	2	1
Rectangular Volume, 20 cm thick/10 cm thick		
1 square meter, surface	6.5	4.2
36 square meters, surface	47	34
(2) Contaminated Debris, (1 mCi total activity)		
at Surface	155	
at 15 cm depth	36	
at 30 cm depth	8	

For condition (1), 100 pCi/g Cs-137 layer of contaminated soil, these Burn Pit measurements would detect a surface layer greater than one cm thick, but would not detect a small thickness of soil (10 cm) buried more than a half of one foot from the surface. This is very good sensitivity, particularly since the likelihood of a narrow strata of contaminated soil located below the surface more than 6 in. is small. Contaminated debris, whose activity exceeded 1 mCi Cs-137 activity could be seen if it wasn't buried any deeper than a foot. 10 mCi could probably be seen down to 2 feet.

5.5 Surface Soil

A 2-lb surface soil sample was collected in each $36m^2$ area. The sample was 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.

5.5.1 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.

5.5.1.1 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.

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.

5.5.1.2 Data Reduction and Analysis

Gross alpha/beta counts were collected for each soil sample, 30 min each. This data with corresponding sampling location was input into SMART SPREADSHEET. Gross activities in pCi/g were calculated using the backgrounds and efficiency factors mentioned in Section 5.5.1.1. This radioactivity concentration calculation is given by the following expression:

$$A_C = \frac{C - B}{30\ \text{min}}\ EF \quad (\text{pCi/g}) \qquad \text{(Eq. 5-3)}$$

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:

$$s = \frac{\sqrt{C + B}}{30\ \text{min}} (EF) \qquad (pCi/g) \qquad (Eq.\ 5\text{-}4)$$

The sample activities were calculated and read by the computer graphics utility. The distribution was plotted and test statistics calculated to compare the results against previous annual reports and regulatory guides.

5.5.2 Gamma Spectrometry

Each 450-ml soil sample was placed in a Marinelli beaker and counted for 30 min on a Carberra 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.

5.5.2.1 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. The library of isotopes is presented in Appendix C.

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.

5.5.2.2 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.

A computer-based spreadsheet was established to calculate U-238 and Th-232 activity concentrations. This determination is made based on the activities of their daughter products. While it will not detect chemically purified U-238, due to the long half-life of its daughter U-234, it will detect chemically purified thorium. These decay schemes are shown in Figure 5.1. With each sample, the mass was entered, along with the calculated activities of the isotopes, and corresponding energies shown in Table 5.2.

Figure 5.1 Naturally Occurring Thorium and Uranium Decay Chains
(From "Radiological Health Handbook," Revised Edition
U.S. Department of Health, Education and Welfare, 1970)

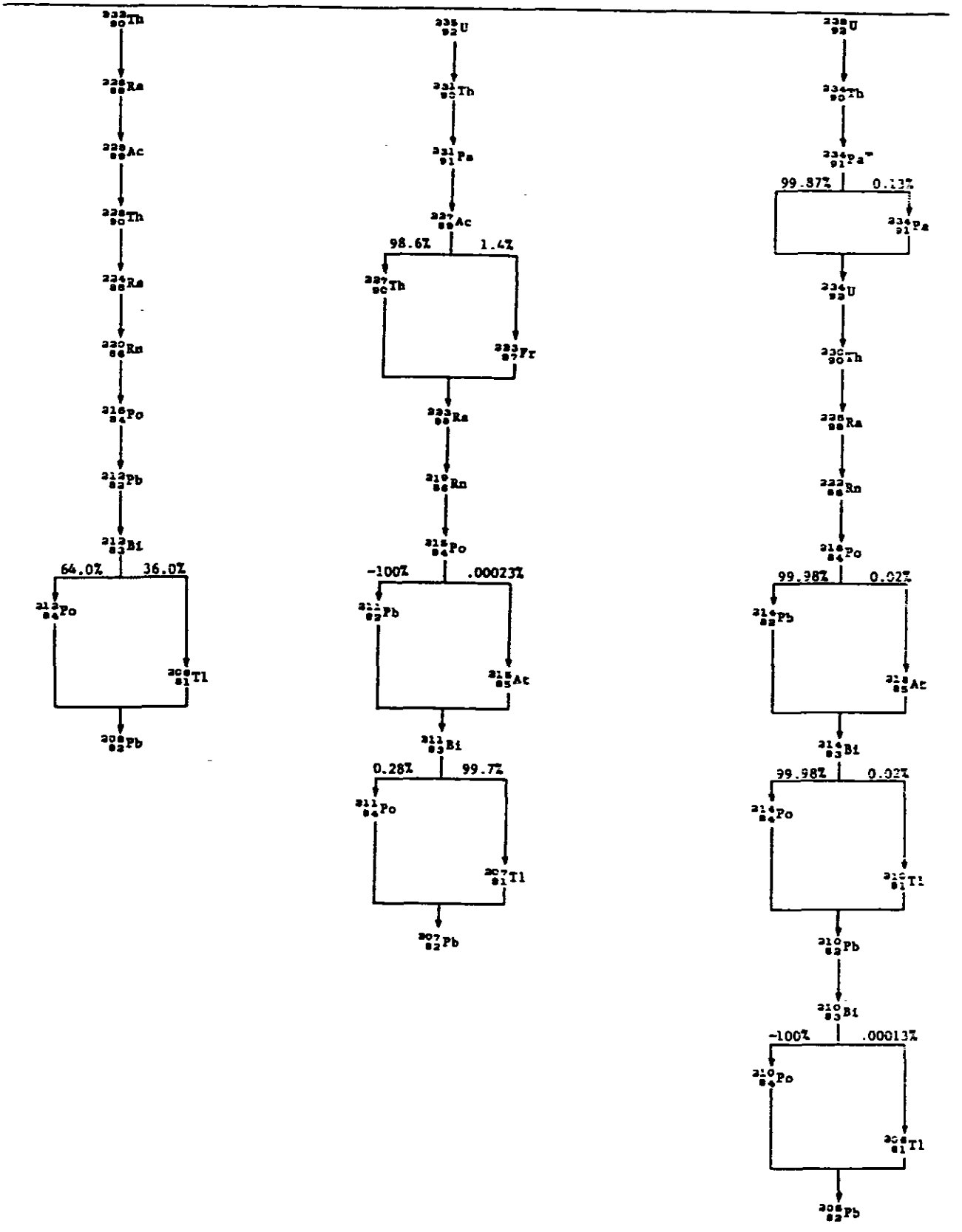


Table 5.2 Probable Gamma Energies for Determining Soil Radioactivity

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 (Cosmogenic)	
Be-7 (478 keV)****	

* 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:

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

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:

$$A_{Th\text{-}232} = \frac{\sum_{i=1}^{n} A_i}{n} + \frac{\sum_{1}^{3} A_{Tl\text{-}208}}{3 * 0.36} \quad (pCi/g) \qquad \text{(Eq. 5-6)}$$

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

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

Probability plots of U-238 and Th-232 activity (pCi/g) and a scatter plot of U-238 versus Th-232 activity will show whether or not any man-made contributions to naturally occurring radioactivity were present. The scatter plot is used to compare observed ratios of U-238 to Th-232 activity concentrations (pCi/g) 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. Probability plots show any deviations in activity; a determination can be made about the presence of depleted uranium or enriched uranium.

Cs-137 and K-40 are easily identifiable radionuclides by gamma spectrometry; they each emit a single characteristic gamma-ray, (see Table 5.2).

Probability plots of K-40 and Cs-137 will show elevated radioactivity areas within the sample lot. The software described previously is used to generate probability plots of these radionuclides.

6.0 PROCEDURES

The following radiological procedures were used in performing this survey.

6.1 Sample Location Selection

Superimpose 3-meter-square grids on each surface to be radiologically characterized.

6.1.1 West

Select $1m^2$ out of each $9m^2$ on which to perform an ambient gamma exposure rate measurement. Select $1m^2$ out of each $36m^2$ on which to obtain a surface soil sample.

6.1.2 South

Select $1m^2$ out of each $9m^2$ on which to perform an ambient gamma exposure rate measurement. South of the road from 36 m to 60 m, select $1m^2$ out of each $36m^2$ to perform an ambient gamma exposure rate measurement. Select $1m^2$ out of each $36m^2$ on which to obtain a surface soil sample.

6.1.3 East

Select $1m^2$ out of each $9m^2$ on which to perform an ambient gamma exposure rate measurement. Select $1m^2$ out of each $36m^2$ on which to obtain a surface soil sample.

6.1.4 North

Select $1m^2$ out of each $9m^2$ on which to perform an ambient gamma exposure rate measurement. Select $1m^2$ out of each $36m^2$ on which to obtain a surface soil sample.

6.2 Calibration and Instrument Checks

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

Portable Ludlum 2220 Survey Instrument coupled to 1" x 1" NaI crystal:

1) Take and record a 1-min background count in an uncontaminated area which typifies the area to be surveyed.

2) Use a Ra-226 check source located 1 ft from the NaI detector to check the operability of the gamma instrument. The count rate should not vary by more than ± 5% from the initially established standard. The gamma calibration efficiency factor is determined by comparison against a Reuter Stokes HPIC.

Gas-flow Proportional:

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

2) Before the analysis is run, using uncontaminated planchets, take ten 30-min background counts of processed sea sand. Take and record 30-min counts of known alpha and beta soil standards to obtain an efficiency calibration.

3) 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.

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 user's manual for proper calibration of device.

6.3 Radiological Measurements

6.3.1 Ambient Gamma Exposure Rate Measurements

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

2) Set the count time to 1 min and take a measurement at each applicable 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.

5) Take at least 30, 1-min counts in an area of similar topography where no radioactive materials were ever handled, stored, or used. This is the background distribution. Enter data in SMART SPREADSHEET.

6.3.2 Measurements of Gross Alpha/Beta 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 (alpha: 241120 ± 12056 dpm, beta: 59977 ± 2999 dpm).

6.3.3 Gamma Spectrometry Measurements

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 30min. 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. Enter data into SMART SPREADSHEET.

7.0 SURVEY RESULTS

The Burn Pit radiological survey was performed using the survey plan previously described. Radiological characterization of the two open-field pits was beyond the scope of this survey; however, Section 7.2 presents the results of previous surveys performed back to 1978. For designation purposes, four areas were identified as east, west, north, and south of the two open-field pits (Appendix E). Radiological data from these four areas were combined into one statistical sample lot to demonstrate the lot ambient gamma exposure rate; gross alpha/beta activity; and U-238, Th-232, Cs-137, and K-40 activity concentrations.

The format of data presentation established for this section follows.

7.1 Statistical Results Format

Historical survey data is mostly presented in tabular format, just as it was reported in the internal letters written at time of publication. Any measurement deviations or topographic/geographic changes which may impact the validity or uncertainty of a particular value is described accordingly. In sample cases where it was felt that the data would follow a Gaussian distribution, the distribution function is plotted. This is the case in Figures 7.1 through 7.3.

The radiological data collected during this survey are displayed as Gaussian cumulative distribution functions in Figures 7.4 through 7.15. 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. In all cases where an acceptance limit is applicable, the limit is substantially

greater than the mean and inspection test statistic (denoted "TS" on the graph) of the distribution.

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 "TS" line will run perpendicular to the abscissa corresponding to about a 93.3% cumulative probability. The Gaussian distribution line must pass below the intersection of the "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 the 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. 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 applicable, for example, U-238, Th-232, Cs-137, and K-40 activities, 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. In these cases, the mean is calculated and the shape of the distribution is observed to identify any areas of increased radioactivity.

7.2 Historical Radiologic Survey Data

The first documented report indicating radiological contamination at the Burn Pit was in October of 1978, when F. Badger surveyed some old sodium barrels and a pallet of various pipe, valves, and trash. The level of radioactivity of these items was about 1 mrad/h, or 2500cpm with a PUG1 instrument, (Reference 21).

R. J. Tuttle further investigated this occurrence and submitted a letter in November of 1978 to identify the probable sources of this radioactivity. The side walls of the concrete pool were contaminated to about 25,000 dpm/100cm^2 total beta contamination. Water in the pit showed approximately 8×10^{-7} μCi/ml Cs-137 and 8×10^{-8} μCi/ml Na-22. Sediment in the pit may have contained a significant amount of Sr-90. This radioactivity was attributed to the SRE. The pallet full of various debris which was found by F. Badger was delivered from Building 006. This debris was determined to be liquid metal oxygen sensors which contain thorium oxide, (Reference 22).

The concrete pool was subsequently decontaminated to acceptable levels. Scabbling of the walls and sludge removal was necessary. The pallet of items was also disposed of off-site.

Up to 1980, the only radiological findings were limited to the concrete pool facility. In December of 1980, a fairly comprehensive radiological survey was performed in both open-field pits and in a limited area surrounding the pits. A backhoe was used to collect soil samples at 1 and 2 foot depths. Figure 7.1 shows the sampling locations. Table 7.1 shows the contamination levels in beta-pCi/g and in μR/h. The average beta activity concentration measured in both open-field pits was 198 pCi/g. Analytical methods, equipment and calibration procedures have changed since this survey was performed; consequently, the values reported then may be different from those values determined today by our current standards. However, what is important in this case, is that any deviations from

"background" are observable. The data clearly show that significant amounts of contamination existed in the open-field pit areas. This contamination was observed in stratified layers of black tar type substance (8" below surface), darker soil layers, and miscellaneous buried debris, down to 2 feet. On the other hand, no radiological contamination was identified outside the ponds. This is demonstrated in Figure 7.2, where the beta soil activity follows a model Gaussian cdf, with a mean of 26.53 $\pm$ 3.53 pCi/g. No elevated readings were observed outside the pits. During soil excavation, it rained and the holes filled with water. Three water samples were collected and analyzed for radioactivity. Gross beta activity concentrations range between 1×10^{-8} and 2.6×10^{-7} μCi/ml. This radioactivity is well below allowable limits, (Reference 23).

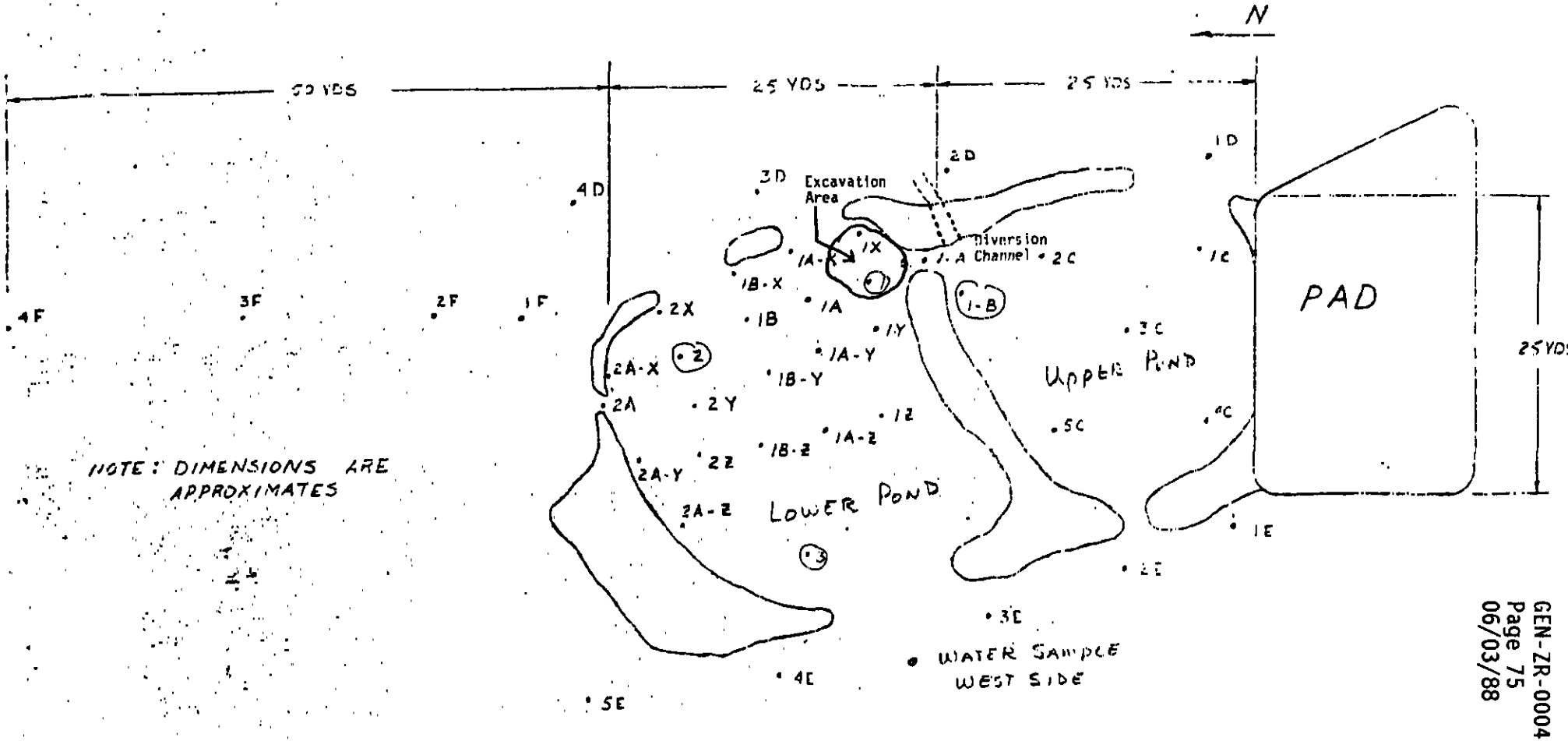


Figure 7.1 Radiological Survey Locations (December, 1980)

Table 7.1 Radiological Survey Results (December, 1980)

Location	Soil Sample Number	Sample Depth (in.)	Beta Activity (pCi/g)	Ambient Gamma Exposure Rate (μR/h)
Lower Pond	1	12	5391	3000
Lower Pond	2	12		100
Lower Pond	3	12	278	600
Upper Pond	1-B	12	22.80	16
(Near	1-B	24	26.28	
Diversion	1-A	12	22.99	18
Channel)	1-A	24	25.51	
Lower Pond	1A	12	166.96	26
(Near	1A	24	42.13	
Diversion	1B	12	32.27	14
Channel)	1B	24	28.21	
Lower Pond	2	12	51.98	100
	2	24	37.68	
	2A	12	25.31	35
	2A	24	29.95	
	1A-Y	20	23.77	50
	1B-Y	20	28.19	30
	1B-X	20	24.32	20
	2A-X	20	22.11	21
	2A-Y	18	44.22	300
	1A-Z	12	28.00	22
	1B-Z	13	25.79	45
	2A-Z	12	37.77	58
	1A-X	18	24.87	20
	2Z	13	45.32	80
	1Z	23	22.66	22
	2Y	24	25.98	115
	1Y	12	36.29	55
	1X	18	21.74	23
	2X	24	25.61	22
Upper Pond	1C	18	24.42	20
	2C	24	30.28	24
	3C	24	30.28	23
	4C	18	24.03	18
	5C	24	23.44	25
Average beta activity for ponds			198	

Table 7.1 Radiological Survey Results (December, 1980)(Cont.)

Location	Soil Sample Number	Sample Depth (in.)	Beta Activity (pCi/g)	Ambient Gamma Exposure Rate (μR/h)
East Area	1D	24	26.96	12
	2D	24	27.16	14
	3D	24	26.76	14
	4D	24	24.62	14
West Area	1E	24	26.96	14
	2E	24	23.44	12
	3E	24	21.49	14
	4E	24	28.33	14
	5E	24	28.52	16
North Area	1F	24	19.34	14
	2F	24	26.96	14
	3F	24	32.63	16
	4F	24	31.84	16

The radiological contamination identified in the pits during the December, 1980, survey was removed as best as reasonably achievable shortly thereafter. After the cleanup effort, another pit survey was performed by F. Badger in May of 1981 (Reference 24). This survey was limited to ambient gamma exposure rate measurements only. A Ludlum Model 12S micro-R meter was used for the survey. This model has an analog count rate meter. Measurements were made in every square meter of the lower open-field pit, at the surface and at 1 meter heights. 373 measurements were made at 1 meter from the ground, with an average reading of 13.7 $\pm$ 5.56 μR/h. 376 measurements were made at the surface, with an average reading of 19.4 $\pm$ 87 μR/h. The standard deviation of the first data set is 40% while the second is 450%. Inspection of the probability plots of these two sets of measurements (Figure 7.3 and 7.4, respectively), shows that the lower open-field pit is contaminated. In several areas, the levels are above acceptance limits. The deviation in the readings observed in Figure 7.4 do not appear to be that great; however, the upper limit has changed to 1600, thereby significantly changing the scale.

In August of 1981, shovels of soil and a few holes located in the lower open-field pit were surveyed. Exposure rates varied from 100 μR/h

shovel-full to 700 μR/h in a hole. Six soil samples were collected and analyzed for beta radioactivity. The beta radioactivity concentrations measured were 1280, 2587, 88, 131, 63, 68 pCi/g. The average of these values is 702 pCi/g. The particular locations sampled were from specific regions known to be significantly contaminated (Reference 25).

Figure 7.2 Soil Beta Activity Outside of Pits (December, 1980)

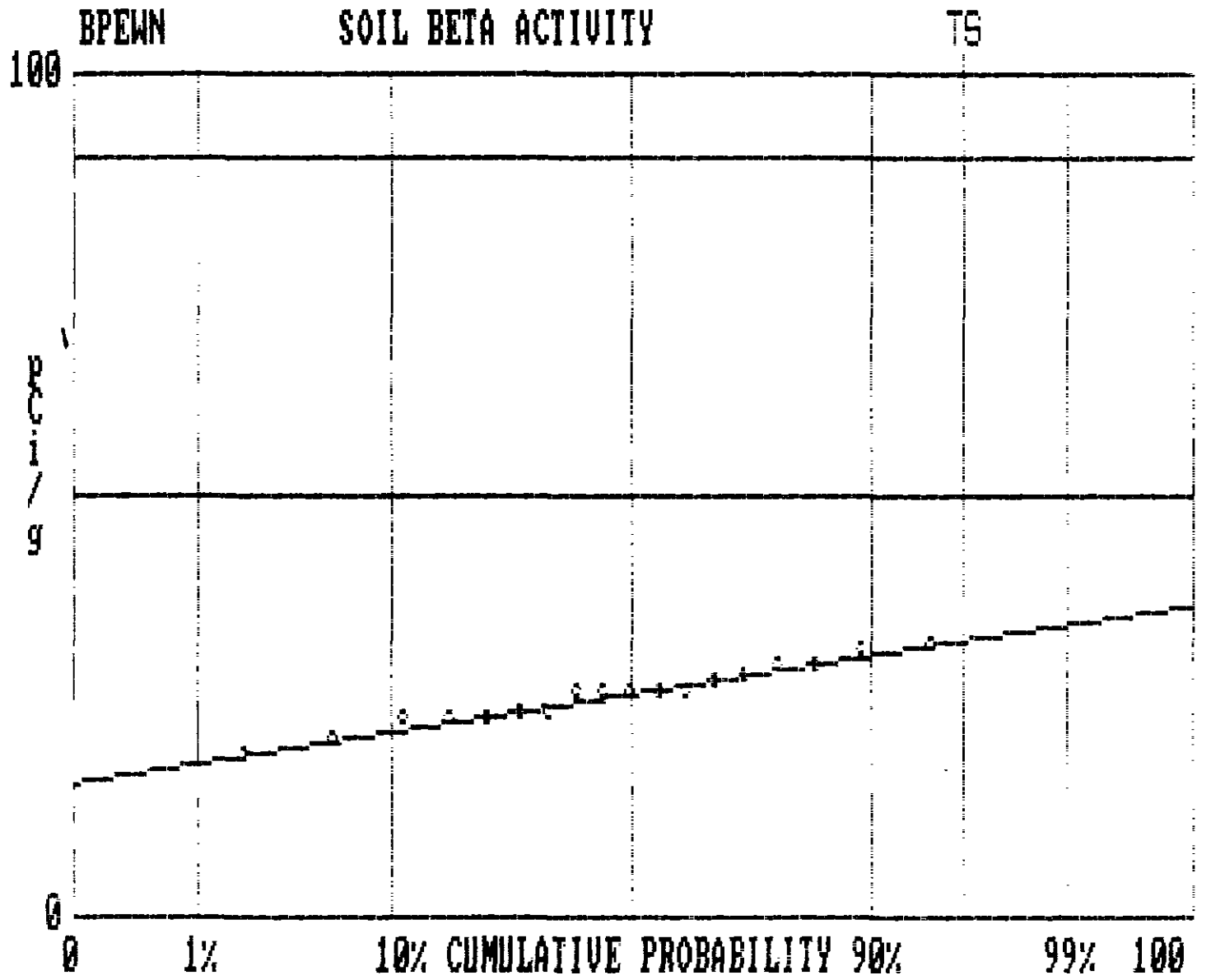


Figure 7.3 Ambient Gamma Exposure Rate Measurements
1 Meter from Surface at Lower Pond (May, 1981)

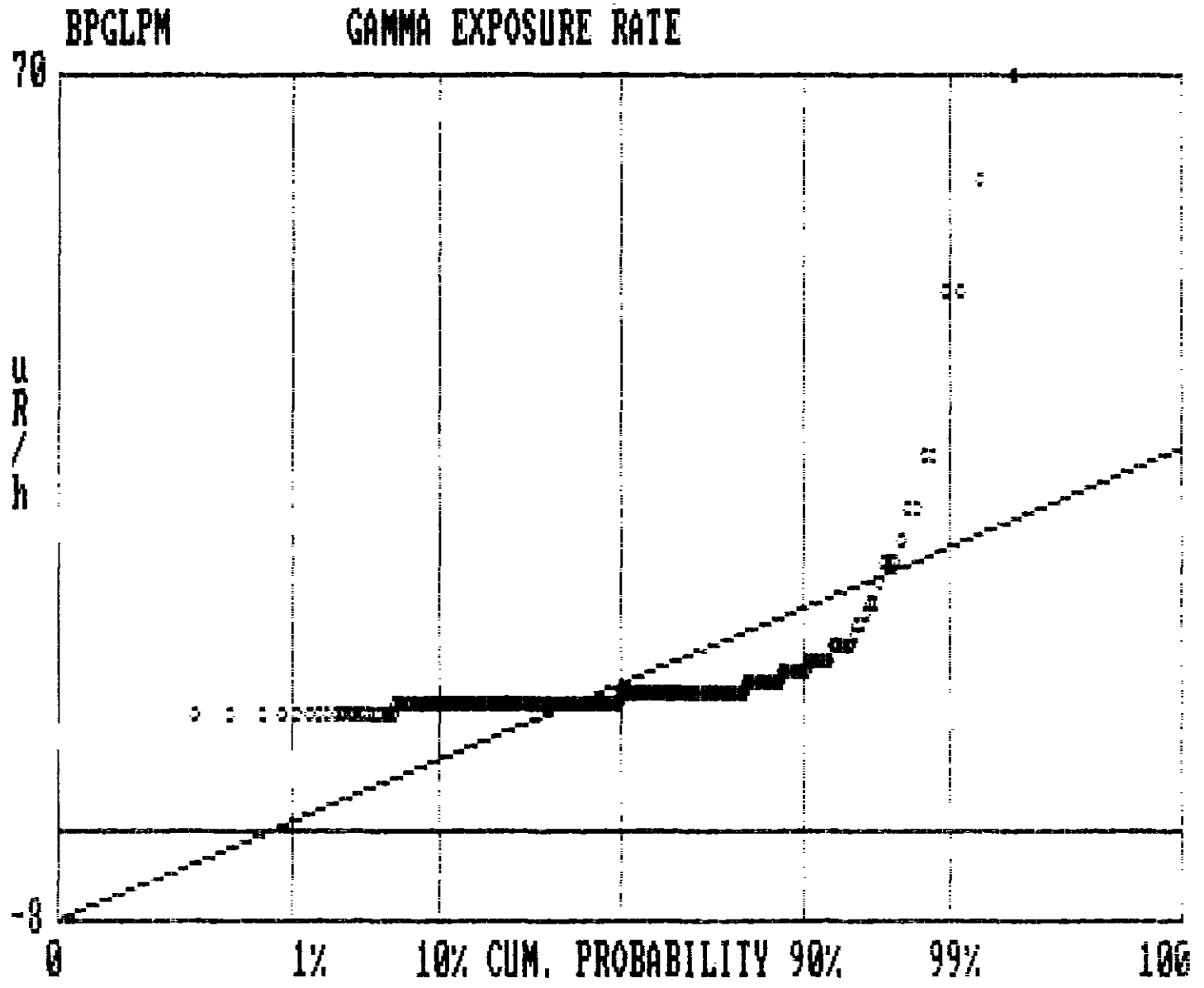
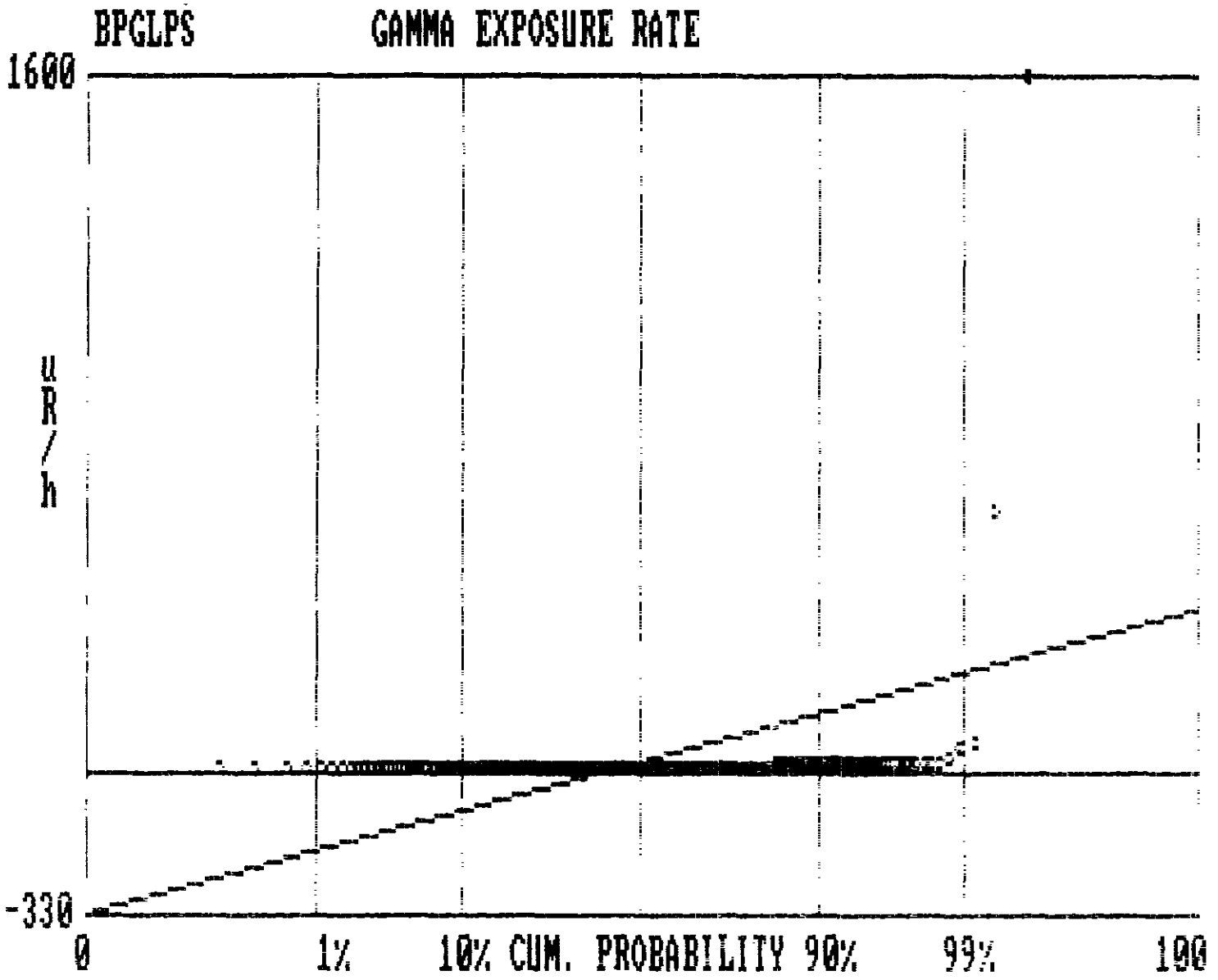


Figure 7.4 Ambient Gamma Exposure Rate Measurements at Surface of Lower Pond (May, 1981)



The next documented Burn Pit survey was performed in September, 1983, when eight soil samples were analyzed by gamma spectrometry, gross alpha/beta activity, and gamma exposure rate. One-pound soil samples were collected in the upper and lower field pits, and one, 50 ft from the SSFL northern boundary (Reference 26). The results are presented below in Table 7.2

Table 7.2 Burn Pit Radiological Survey Results
from September, 1983 (Reference 26)

Sample Location	Gamma Exposure Rate (μR/h) (1)	Soil Gross Activity (pCi/g) Alpha (2)	Beta	Cs-137 (3)
(1) Dry Channel, 50' from No. Boundary	<15	11	22	
(2) SE Lower Pond	<15	14	26	.83
(3) S-Center Lower Pond	180	10	48	6.97
(4) N-Center Lower Pond	120	6	171	129.66
(5) NE Lower Pond	60	16	25	.54
(6) NW Lower Pond	120	63	38	70.14
(7) E Upper Pond	<50	13	25	.45
(8) W Upper Pond	<50	9	24	.10
(9) S-Center Upper Pond	<50	9	20	N/D
(10) N-Center Upper Pond	<50	12	29	N/D

(1) Measurements made with Ludlum micro-R meter (Bckg = 11 μR/h)
(2) Alpha/Beta measurements made with proportional counter
(3) Cs-137 was the only significant radionuclide identified by gamma-spec
N/D Not determined.

On October 27, 1983, ten more soil samples were analyzed by gamma spectrometry. Six of these samples were taken from areas around the open-field pits. Four were taken from the upper and lower pits. None of the samples showed radioactivity concentrations or radiation levels greater than background. Cs-137 was identified in only 3 samples (concentrations less than 0.6 pCi/g). A water sample was also collected and found to have a beta radioactivity concentration of $2.61 \pm 0.45 \times 10^{-8}$ μCi/ml; no alpha radiation was detected, (Reference 27).

Following the two radiological surveys performed at the end of 1983, the Burn Pit remained inactive. Just before the CERCLA Phase II site characterization for chemical contaminants began in March of 1987, four soil samples were collected and analyzed for radioactivity. Samples were collected at (1) the north boundary line in a gooey, slimy sump hole; (2) 20 ft south of the north boundary line; (3) a squirrel mound in that same location; and (4) the center of the upper pit. Each sample was split into 5 smaller samples for independent analysis. Direct gamma measurements were also made in these areas. Background radiation levels were found. Cs-137 activity was less than 0.2 pCi/g, and gross alpha/beta activity was at background levels. Gross alpha/beta activities measured for each area were: (1) N. boundary sump = 10.2 $\pm$ 0.7 alpha-pCi/g; 22.8 $\pm$ 1.15 beta-pCi/g; (2) background 20 ft south of sump = 13.0 $\pm$ 0.4 alpha-pCi/g; 24.1 $\pm$ 0.6 beta-pCi/g; (3) squirrel mound = 13.2 $\pm$ 0.8 alpha-pCi/g; 23.4 $\pm$ 0.6 beta-pCi/g; and (4) upper pit = 9.4 $\pm$ 0.5 alpha-pCi/g; 19.1 $\pm$ 0.4 beta-pCi/g. Only one sample (upper pit) showed elevated concentrations of U-238 daughters; about 5 times normal. Pb-214 at 295.1 keV calculated 33 pCi/g, and Pb-214 at 352.0 keV calculated 35 pCi/g, (Reference 28).

The chemical characterization study performed in March of 1987 for CERCLA Phase II was conducted under the cognizance of a health physicist. Section 2.3 described the sampling locations and the radiation levels found. Keep in mind that every effort was made to avoid any radioactively contaminated areas to avoid giving radioactive material to the chemistry lab. Hence, the results presented are biased to the low side. By conversation with the HP in charge, it was found that in several cases the beta radiation readings using a PUG probe were significantly high at various trench depths. These areas were specifically avoided for sampling. These areas were not analyzed by gamma spectrometry. The gamma spec results presented in that study are only from those samples which had background radiation levels as measured by a PUG, (Reference 29).

Although the CERCLA chemical characterization effort provided opportunity to study radiological contamination in each open-field pit, this

was not utilized. The evolution of radiological surveys performed to-date and presented in this section, show that both open-field pits are contaminated with radioactive material. The extent and degree of contamination is not very well known, particularly depthwise. Further characterization is needed in this regard.

As far as historical surveys are concerned in areas surrounding each pit, no evidence suggests that contamination has migrated beyond the berms of each pit. Water samples are collected after most rain storms in areas downslope of the pits; no sample has ever showed radioactivity above background. Independent chemical analysis of water has shown no chemical contaminants downslope. Soil samples taken previously also show that no contamination spread has occurred. However, because these periodic surveys were never detailed and always limited in scope, this current project was funded to further investigate radionuclide migration. The results of this comprehensive survey are presented in the following section.

7.3 Ambient Gamma Exposure Rate Measurements

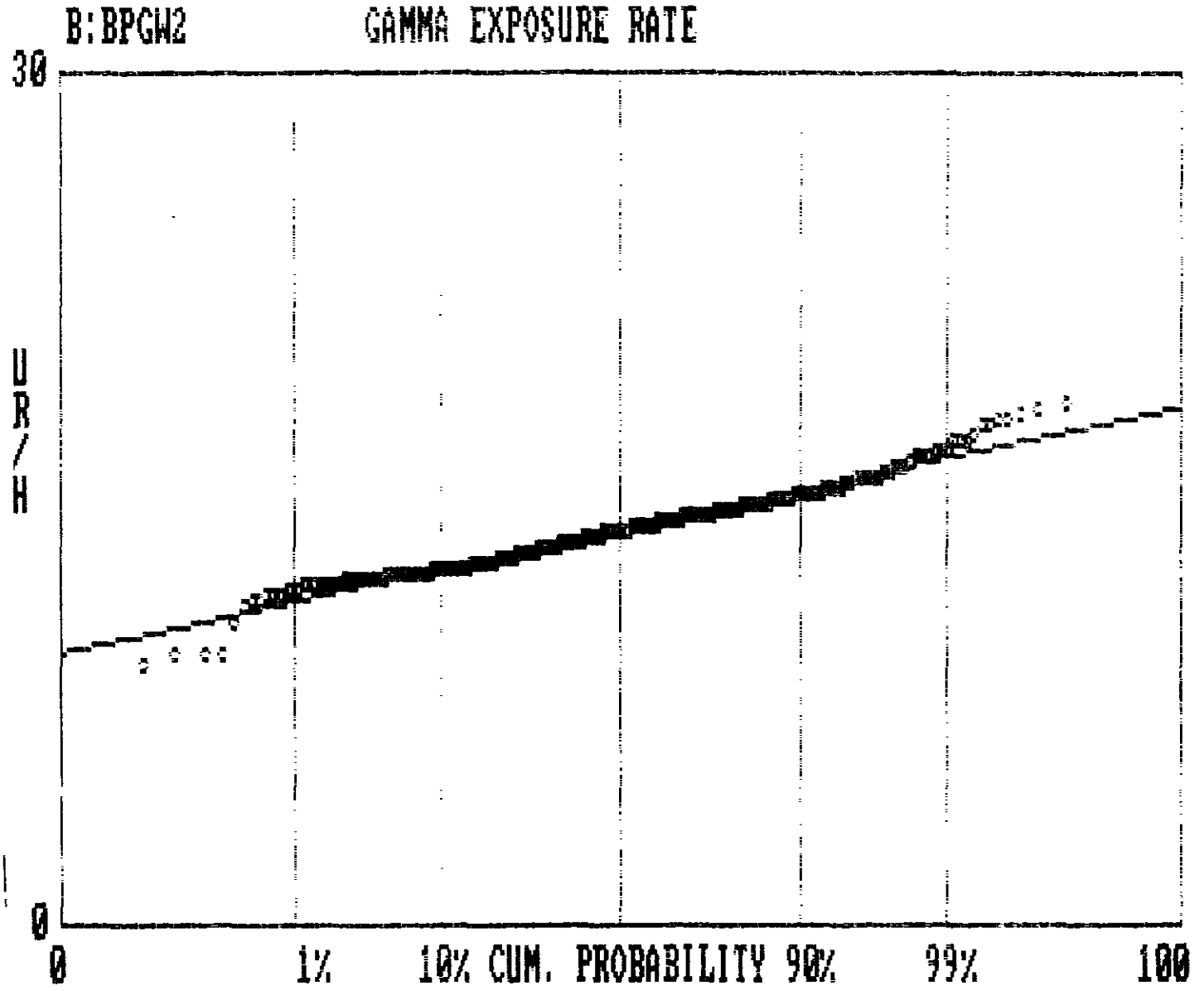
7.3.1 Burn Pit

Ambient gamma exposure rate measurements were made 1 meter above the ground in each $9m^2$ grid. This survey plan resulted in 1398 measurements made in an entire area surrounding the open-field pits. The average total (uncorrected for background*) exposure rate for the area was

*Rather than attempt to derive an appropriate exposure rate background which accounts for time and spatial dependencies of each measurement, total (gross) exposure rates are reported. Any significant deviations due to contamination would be identifiable on the probability plot. In order to observe the deviations in natural background at SSFL, several background measurements were made and plotted as Gaussian cdfs in the next section.

13.9 ± 1.07 μR/h. A Gaussian cdf plot of the data shows no regions of significantly elevated exposure rates (Figure 7.5). Variations of 4 to 6 μR/h were observed between measurements made in the middle of an open-field and those made near a sandstone outcropping. This slight variation is noticeable at both ends of the cdf plot. This spatial dependency of the gamma measurement is significant enough that the results of a measurement may be identified as unacceptably contaminated above NRC limits of 5 μR/h above background. For this reason, background was not subtracted for these results. There are slight deviations from the expected Gaussian distribution at both the high and low extremes. These are not considered to be significant. No gamma emitting contamination was found by this survey. This claim, of course, is restricted to the sensitivities of the device as described in Section 5.4.4.

Figure 7.5 Ambient Gamma Radiation Surrounding Open-Field Pits
(December, 1987)



7.3.2 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 Burn Pit measurements, four independent areas were surveyed, all in locations where no radioactive material was ever handled, used, stored, or disposed. Three of the four 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. The fourth area is located at ETEC's Amazement Park, a picnic area. Measurements were made in this area on a daily basis (morning, evening) for the duration of the Burn Pit survey to identify daily changes in gamma background. Table 7.3 shows the results of these measurements.

Table 7.3 Ambient Gamma Radiation at SSFL

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
Amazement Park (over time)	36	12.3	0.3	1.2
Burn Pit Area	1398	13.8	1.07	8.7

Measurements from the area surrounding building 309 show the most variability of all four background areas. This is attributed to large sandstone outcroppings in the area; the spatial dependency of each measurement is observable in this case. The topography of the first three locations is very similar to that of the Burn Pit. 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.

The time dependent behavior of background radiation is what we would expect - a Poisson distribution, unless some technologically enhanced radiation source affected the ambient radiation environment, for example, a weapons test or a nuclear accident. The average value for this distribution was 12.3 $\pm$ 0.29 μR/h. If one takes the square root of the count rate for this value and converts to μR/h, it should correlate with the distribution standard deviation, (for a truly Poisson distribution). This is the case; $\sqrt{2674}$ cpm x .0046 = 0.24 μR/h, very nearly 0.29, the distribution standard deviation, showing that the major component in the variability is the expected counting statistics.

To assist in observing the variability of ambient gamma radiation levels, we have plotted the probability function for each area in Figures 7.6 through 7.9. 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).

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. If we assume the background is either of the three Rocketdyne areas, we end up with a negative exposure rate. It is quite clear from the data, however, that the Burn Pit areas surveyed in this project are not contaminated. Compare these results to the open-field pit measurements presented in Figures 7.3 and 7.4.

Figure 7.6 Ambient Gamma Radiation at Area Surrounding Building 309
(January 19, 1988)

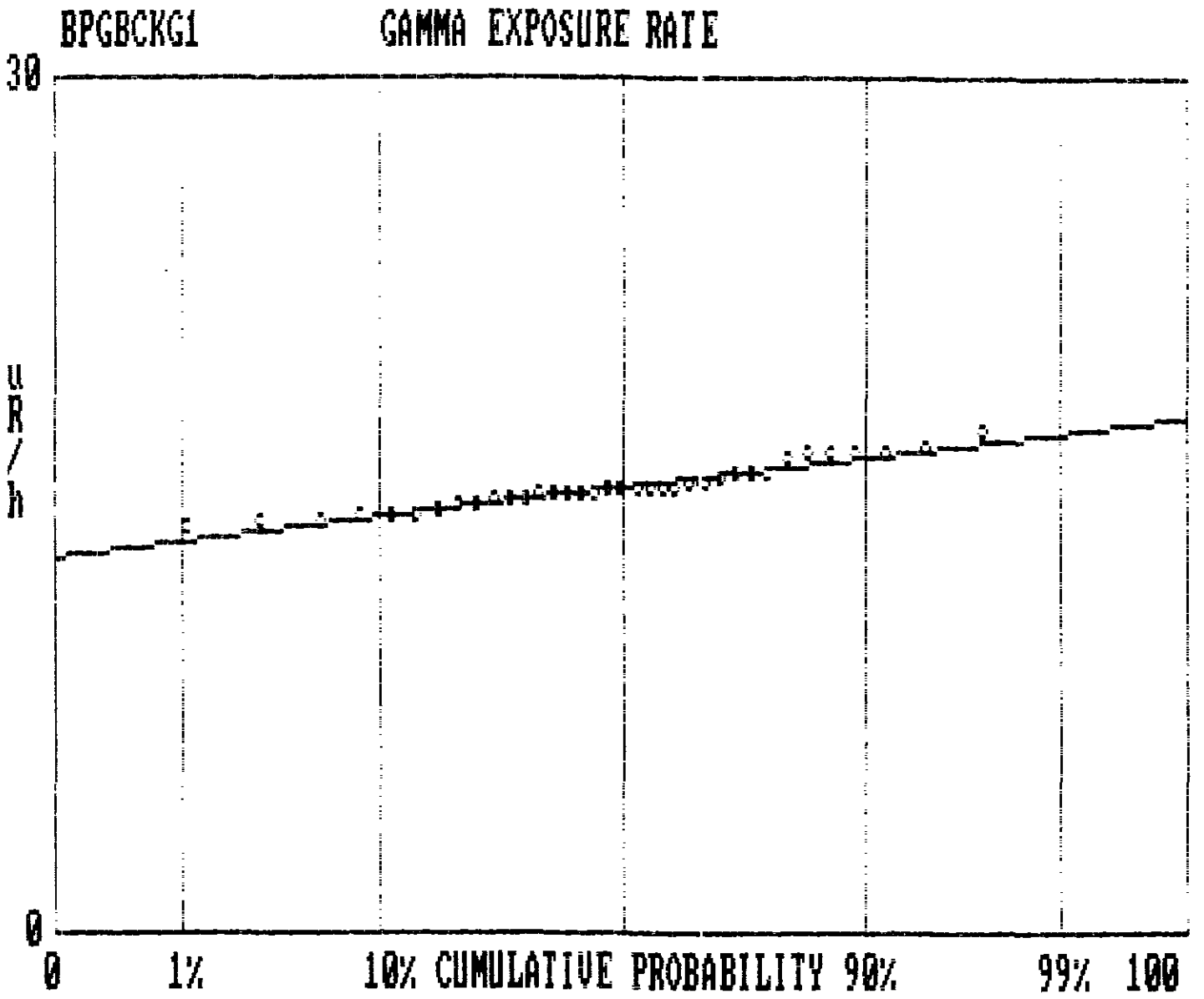


Figure 7.7 Ambient Gamma Radiation at Area Well #13 Road
(April 29, 1988)

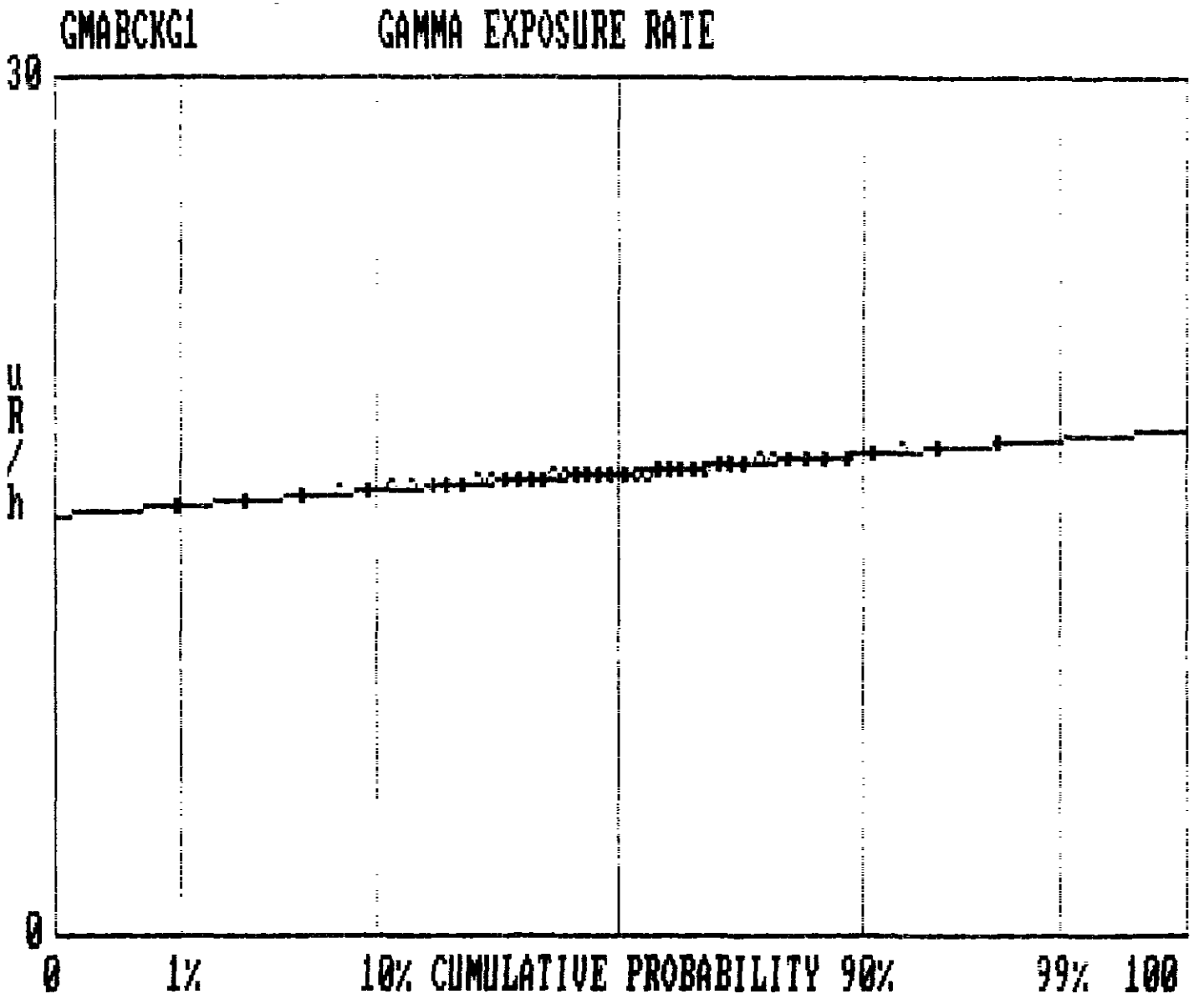


Figure 7.8 Ambient Gamma Radiation at Incinerator Road
(April 29, 1988)

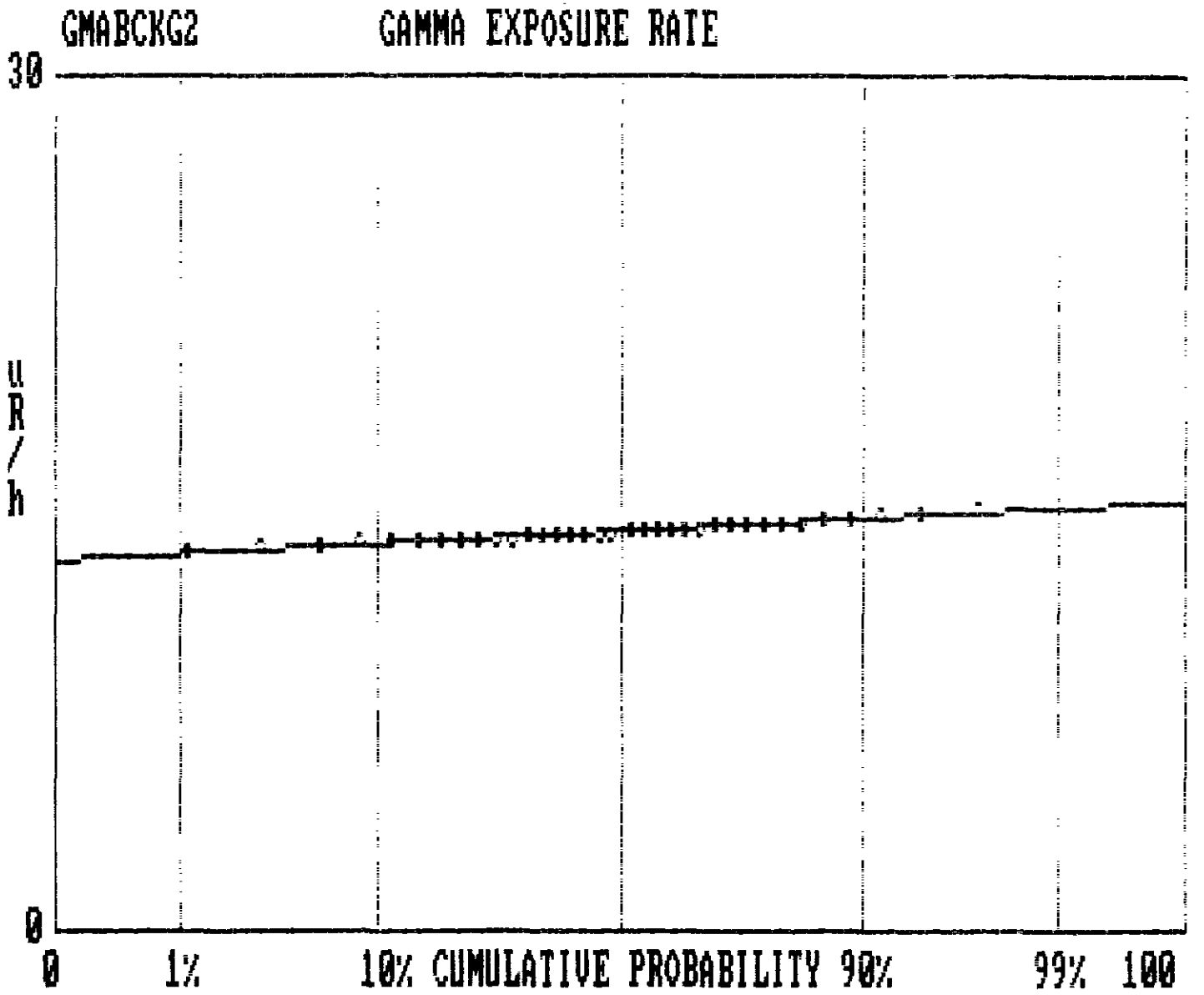
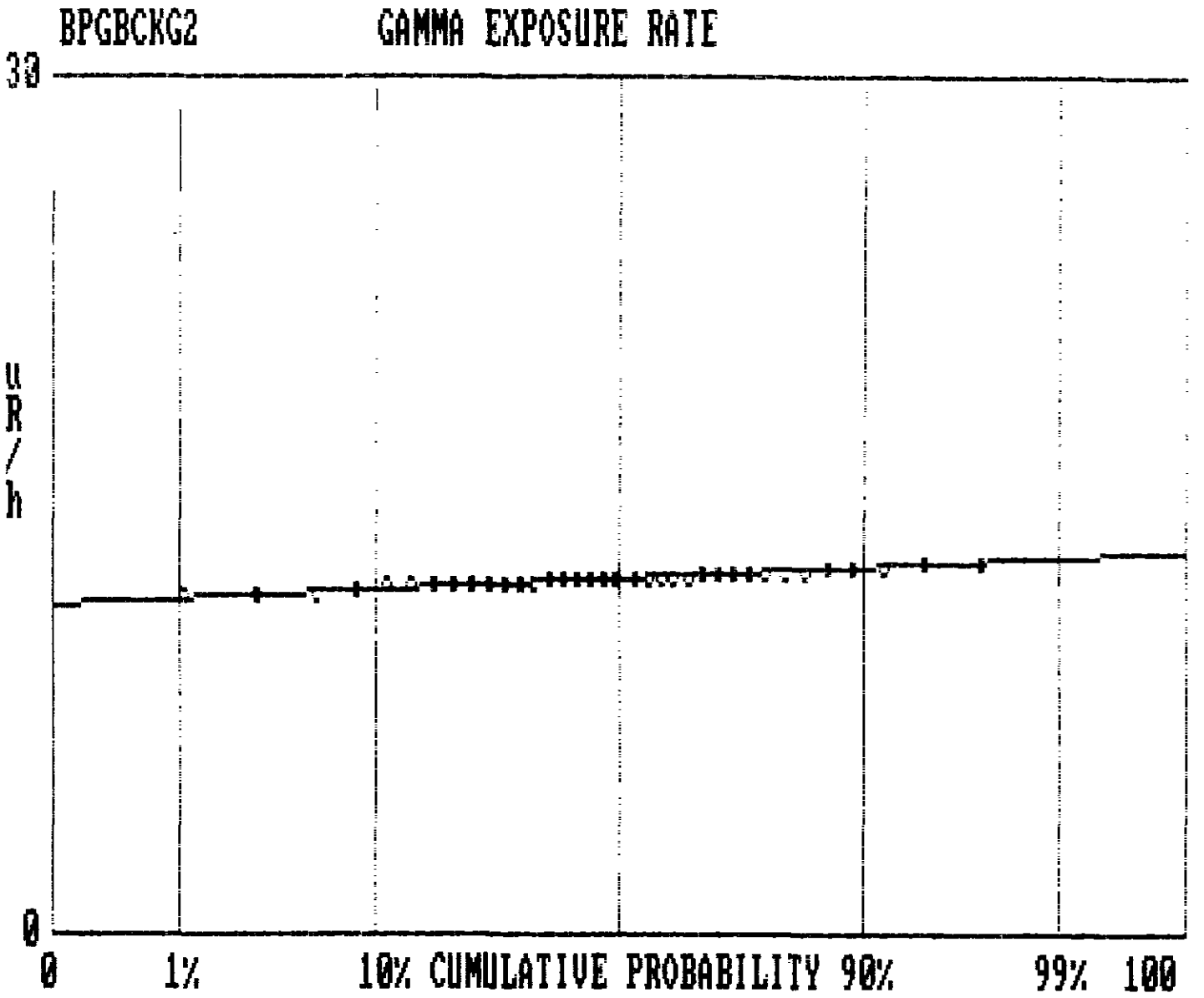


Figure 7.9 Daily Changes in Ambient Gamma Radiation at ETEC's Amazement Park (December, 1987)



7.4 Gross Alpha/Beta Surface Soil Radioactivity

In each 36 square-meter area, a surface soil sample was collected and analyzed for gross alpha/beta activity. This sampling frequency resulted in 253 samples. Since acceptance limits for release for unrestricted use were previously established as 30 pCi/g above background (alpha) and 100 pCi/g total beta activity, these measurement values were statistically evaluated to show compliance with these limits. The results are shown in Table 7.4.

Table 7.4 Gross Alpha/Beta Radioactivity at Burn Pit Area

	Alpha (pCi/g)	Beta (pCi/g)
Average	15.8	23.5
Standard Deviation	5.7	2.5
$\bar{x}$ + ks	23.7	27.0
Acceptance Limit	46 *	100

*Alpha acceptance limit is 30 pCi/g above background. Using a background of 15.8 pCi/g results in a limit of 46 pCi/g. The average soil alpha activity concentration reported in Reference 15, our Annual Environmental Report, was 26 pCi/g. Alpha activity concentrations at the Burn Pit site are significantly lower than background samples measured off site.

The distribution functions plotted in Figures 7.10 and 7.11 show the alpha and beta activity concentrations, respectively. The plots indicate that no areas exist with greater than background levels of radioactivity. Furthermore, and more importantly, there is no additional Gaussian distribution from which we only sampled a few areas. If this were the case, a significant lip at the end of the distribution would be evident. The inspection test statistic (x + ks) is less than the acceptance limit in both cases.

Figure 7.10 Surface Soil Alpha Radioactivity Concentration in Burn Pit Area

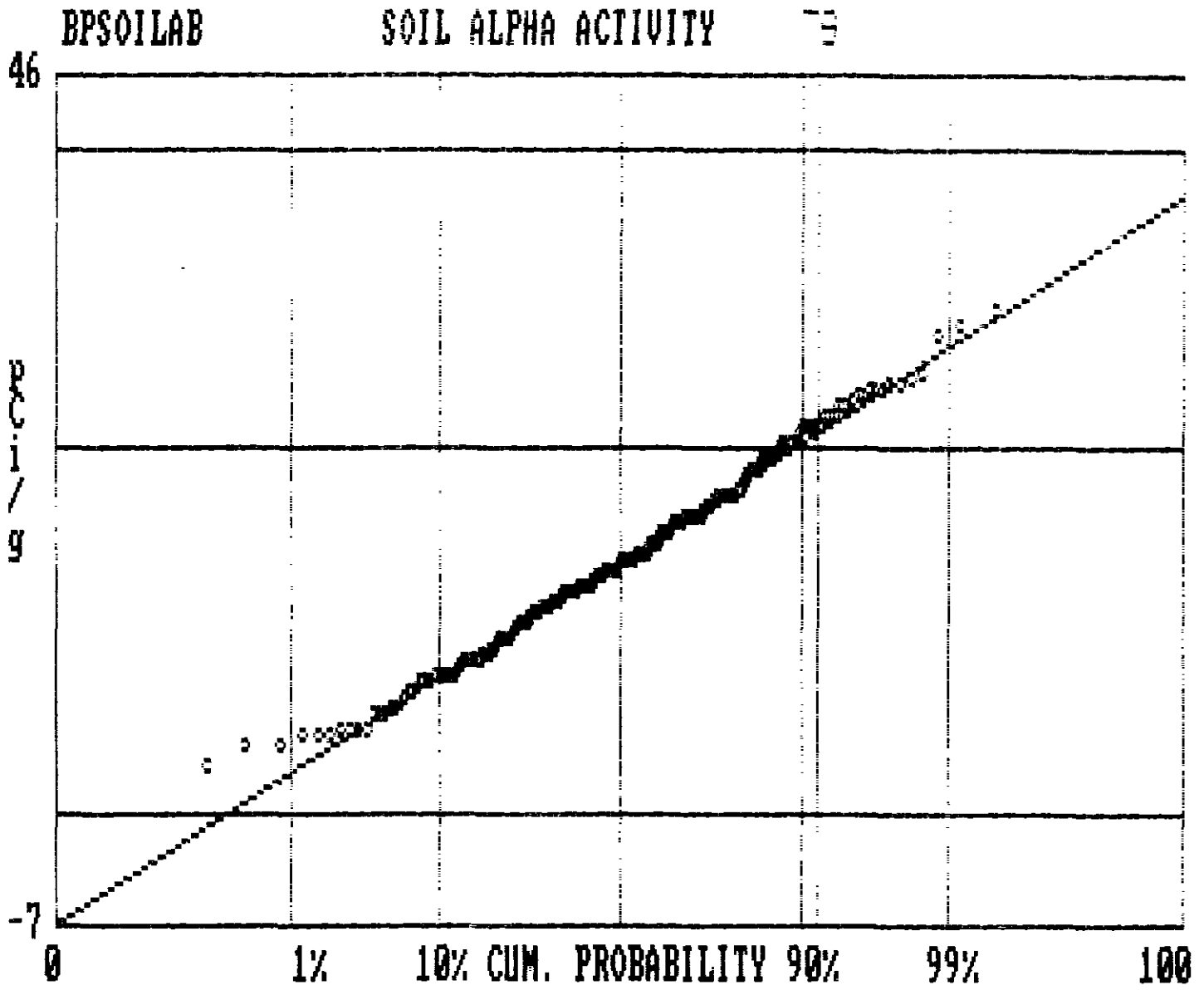
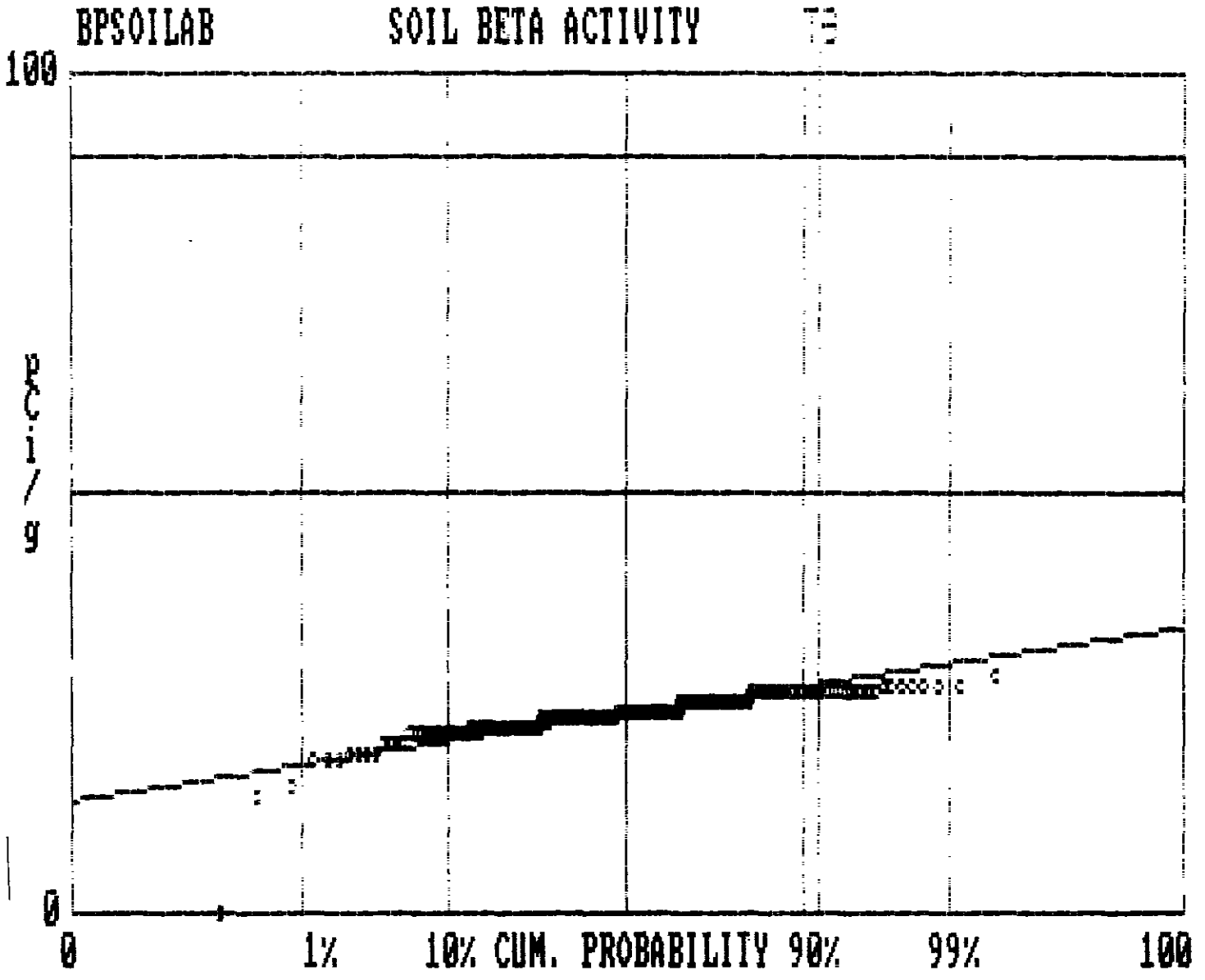


Figure 7.11 Surface Soil Beta Radioactivity Concentration in Burn Pit Area



7.5 Gamma Emitting Radionuclides Identified in Surface Soil Samples

From each $36m^2$, a surface soil sample was collected and homogenized. A 450ml portion of that sample was then analyzed for gamma emitters by gamma spectrometry using the methods and system described in Section 5.3.3 and Appendix A. The radionuclide library is listed in Appendix C. 280 samples were analyzed.

Daughter products of U-238 and Th-232 (see Table 5.2) were identified in almost every sample. No conclusive evidence for U-235 activity was identified. Naturally occurring K-40 was also found in almost every sample. Cs-137, a fission product, was only identified in 64 samples, all at environmental background levels (note: Cs-137 activity has been distributed globally from weapons testing, and in lesser quantities from the Chernobyl reactor incident). No activation products were identified. The following sections show the radioactivity concentrations of U-238, Th-232, Cs-137, and K-40. The raw data are listed in Appendix D.

7.5.1 Radioactivity Concentrations of U-238 and Th-232

U-238 and Th-232 are primordial, naturally occurring radionuclides; however, their radioactivity concentrations in soil vary markedly around the world, depending on the soil composition. Their radioactive decay chains are quite complex, emitting many different energy alpha and beta particles, and gamma rays. We observed the discrete energy gamma emissions from the radioactive daughter products to estimate the parent activities. Both radionuclides are also used in man-made devices. The results of this analysis would also show the presence of enriched uranium or thoriated devices mentioned in the historical literature. U-235 activity was also calculated if the proper peaks were identified; otherwise it was assumed to be naturally occurring at 0.7 weight percent. No specific U-235 photo-peaks were identified.

As a benchmark as to what naturally occurring activity might be, from the CRC handbook, we know that the average composition of the earth's crust is 4g Uranium/1000 kg and 12g Thorium/1000 kg. These values are probably good to ± 100% around the world, except for rich mineral areas. In this case, the ratio of activity concentrations (pCi/g) of U-238: Th-232 is very nearly one. If the measured ratio is less than one, we would suspect enriched uranium or thorium; if it is greater, we would conclude that the ratio of natural uranium to thorium in this area is greater than the value reported in the literature.

Table 7.5 shows the average radioactivity concentrations of U-238 and Th-232. The ratio of U-238 to Th-232 is about 0.8 for the average of the two distributions, a little less than 1.0, but certainly acceptable as natural primordial activity. Table 7.5 also shows good agreement with the alpha activity detected and reported in Table 7.4. U-238 emits 8 alpha particles through its decay chain, and Th-232, 6 alphas. This would yield an alpha activity of 15.34 pCi/g, based on the gamma spec average results. Alpha analysis reported 15.8 in Table 7.4. Beta agreement is not as good because of the beta emitters in the soil besides U-238 and Th-232.

Table 7.5 Gamma Emitting Radionuclide Activity Concentrations in 280 Surface Soil Samples

Radionuclide	# of Samples with Positive Identification	% Detected	Average pf Positive Values* (pCi/g)	Conclusion
U-238	275	98%	0.98 ± 0.18	Natural
Th-232	275	98%	1.25 ± 0.20	Natural
Cs-137	64	23%	0.34 ± 0.25	Natural
K-40	262	94%	20.7 ± 2.89	Natural

*Samples not showing radionuclide of interest were not averaged.

To show the fairly uniform distribution of U-238 to Th-232 activity, Figure 7.12 is a linear scatter plot. The line passing through the data points is a bilateral fit. The plotted data show natural radioactivity with no elevated outliers. Figures 7.13 and 7.14 are plots of the Gaussian cdf for U-238 and Th-232 activities, respectively. The horizontal line intersecting the ordinate corresponds to the average value listed in Table 7.5. Although a few points (1 for U-238, 5 for Th-232) are elevated above the Gaussian fit, they are only a few percent high and do not indicate any significant contamination level in an area. The Federal Register (Reference 13), indicates acceptable soil contamination limits for unrestricted use of 35 pCi/g depleted uranium, and 10 pCi/g thorium. Our gamma spectrometry output did not show in any case, U-238 as a pure chemical element without its daughters past U-234. Our average U-238 activity was far below this limit, even though it's really not applicable here. Furthermore, the expected alpha activity correlated with gross alpha activity. As for thorium, the greatest activity measured was 2 pCi/g. The data show a clean area with naturally occurring quantities of U-238 and Th-232.

7.5.2 Radioactivity Concentration of Cs-137

Table 7.5 shows that Cs-137 was only detected in 64 of the 280 samples (just 23%). Over the years, we have typically observed Cs-137 activity in soil anywhere from 0.1 to 1.0 pCi/g. The finding in this survey is that the average value fits within this "background" window at 0.34 $\pm$ 0.25 pCi/g. Figure 7.15 shows a probability plot of the positively identified Cs-137 soil samples. Three points are greater than the Gaussian fit; 1.17, 1.26, and 1.36 pCi/g. Adjacent areas to these randomly distributed spots were sampled and nothing was found. Gamma exposure rate measurements taken in these grids are not above normal.

Figure 7.12 Surface Soil Th-232 vs. U-238 Activity Concentrations in Burn Pit Area

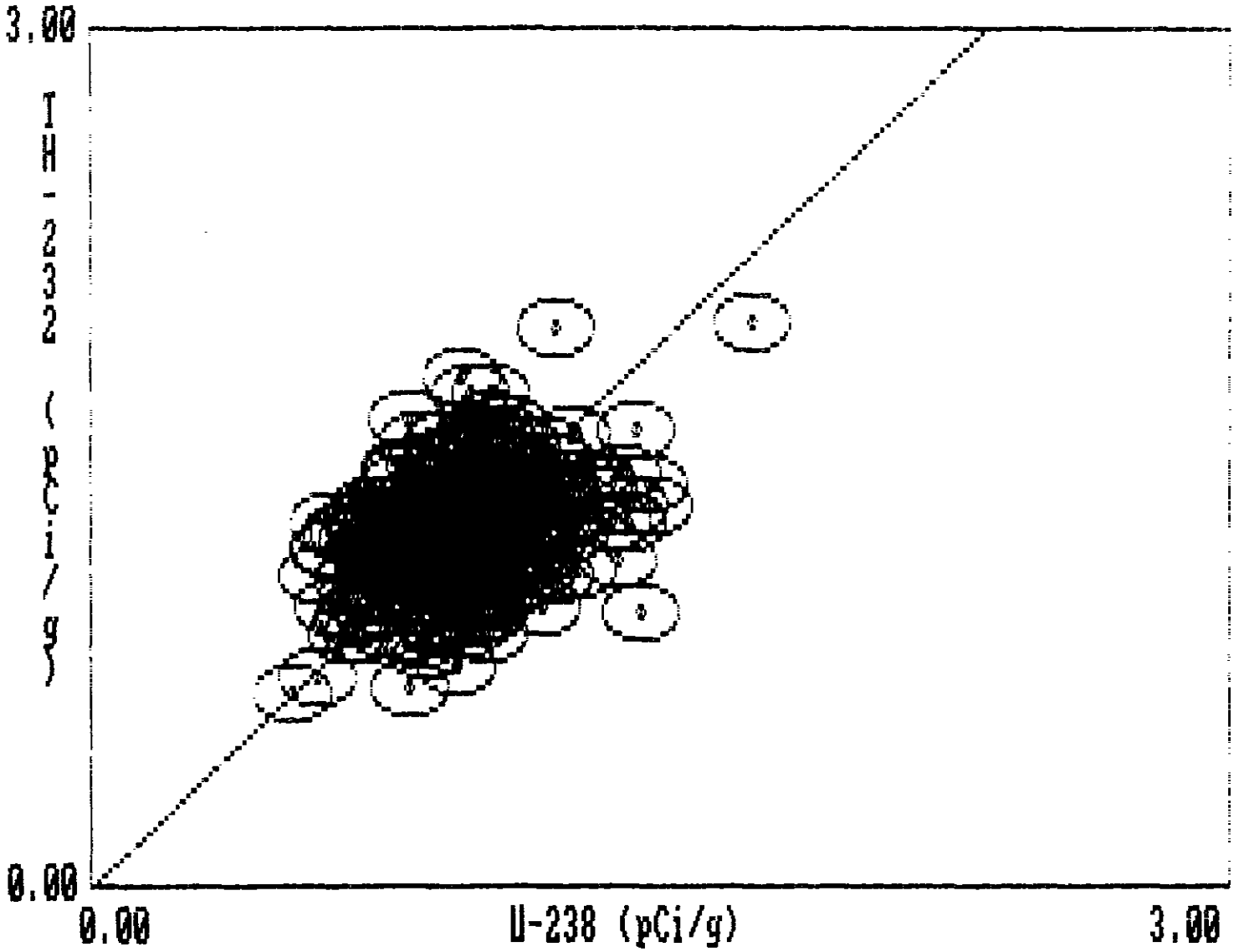


Figure 7.13 Surface Soil U-238 Activity Concentration in Burn Pit Area

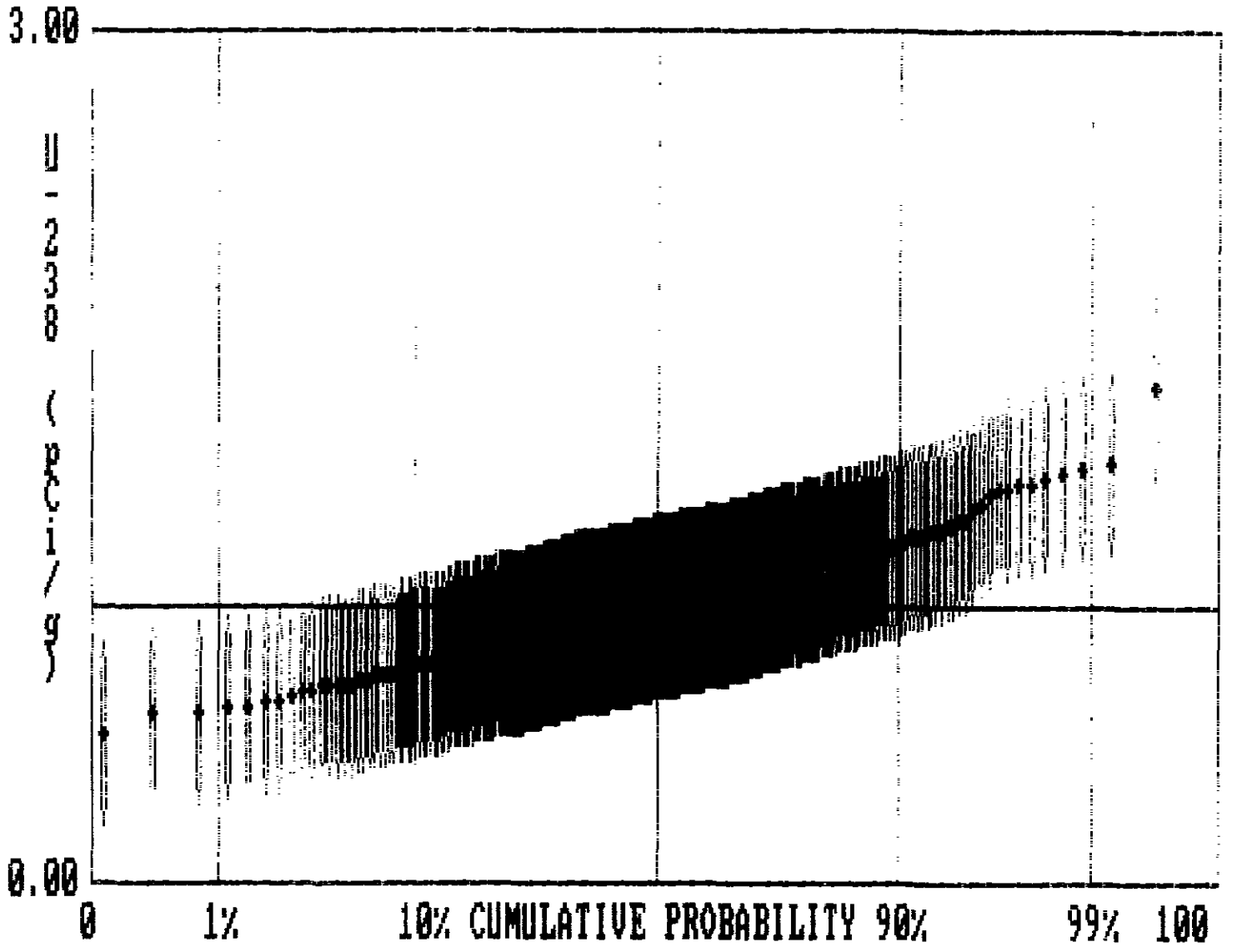


Figure 7.14 Surface Soil Th-232 Activity Concentration in Burn Pit Area

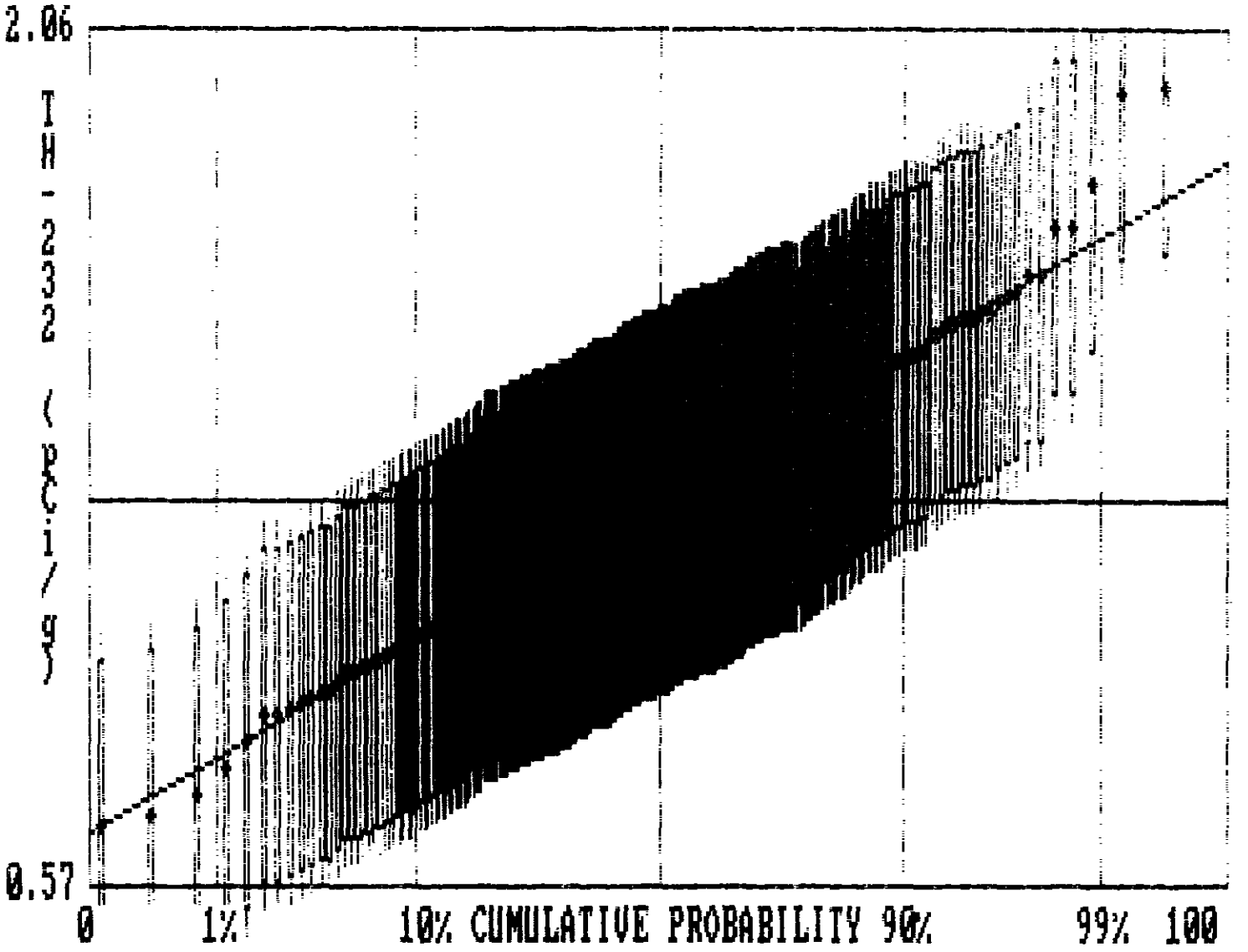
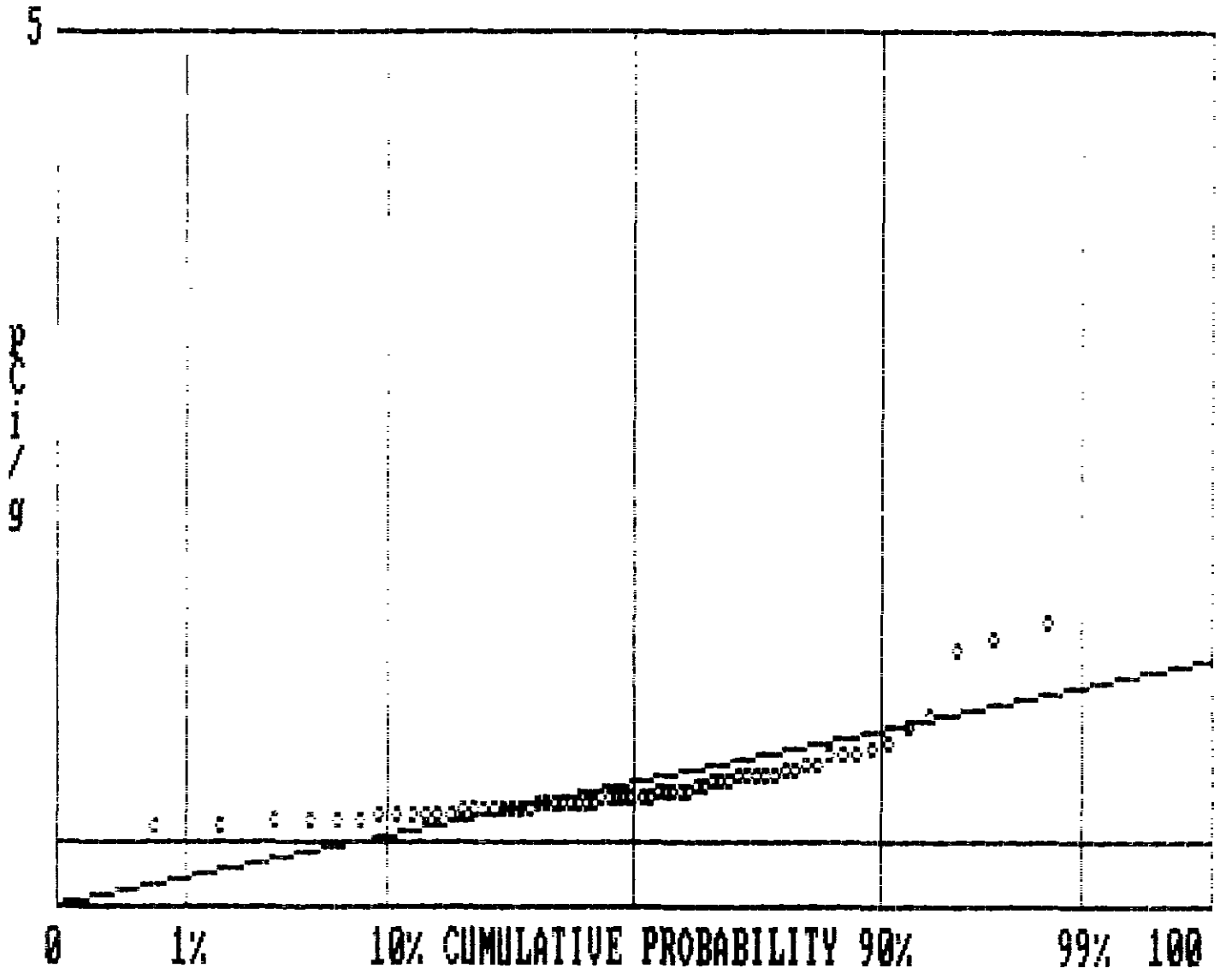


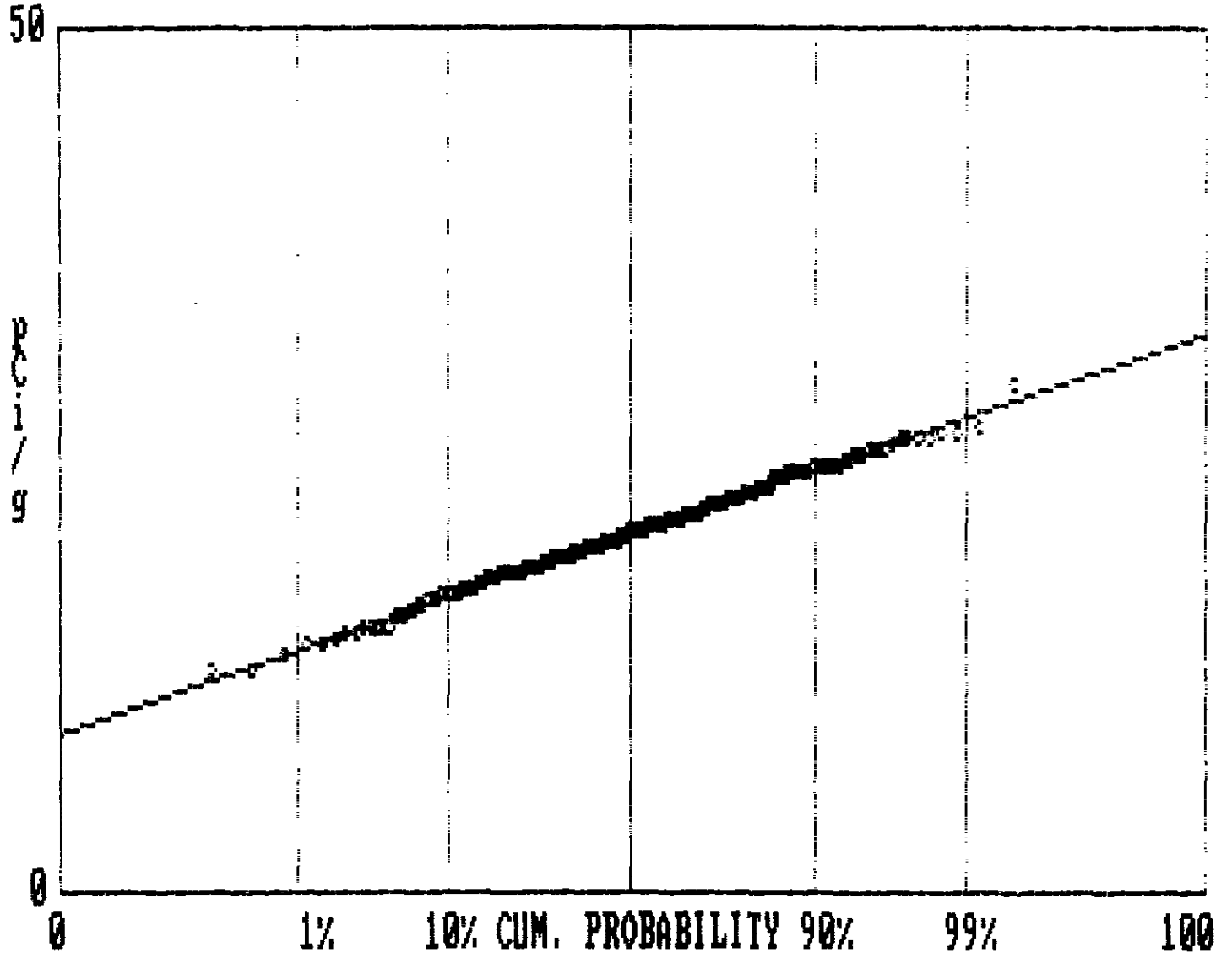
Figure 7.15 Surface Soil Cs-137 Activity Concentration in Burn Pit Area



7.5.3 Radioactivity Concentration of K-40

Table 7.5 shows that K-40 was detected in 262 of the 280 soil samples. The average K-40 radioactivity concentration is 20.7 $\pm$ 2.9 pCi/g. This value is in good agreement with previously determined K-40 activities in soil at the SSFL site; this is natural background levels. The Gaussian cdf of these values shown in Figure 7.16 shows a model distribution with no outliers. Any NaK which was disposed of in that area has since been removed, or never migrated to the surrounding area.

Figure 7.16 Surface Soil K-40 Activity Concentration in Burn Pit Area



8.0 CONCLUSIONS

A 3-acre area surrounding the open-field pits at the Sodium Disposal Facility was characterized for radioactive contaminants. All direct measurements and soil sample analyses show the surrounding area to be free of radioactive contaminants. Previous measurements show that the two open-field pits are contaminated with radioactive material. Further investigation is necessary in this regard.

The average ambient gamma exposure rate for the surrounding area is 13.9 $\pm$ 1.07 μR/h; this compares closely with four other "background" areas measured. The average alpha and beta activity concentrations are 15.8 $\pm$ 5.7 pCi/g and 23.5 $\pm$ 2.5 pCi/g, respectively. With a consumer's risk of acceptance at 10% probability with LTPD of 0.10, we accept the area as radiologically clean. U-238, Th-232, Cs-137, and K-40 are present in environmentally normal concentrations: 0.98 pCi/g, 1.25 pCi/g, 0.34 pCi/g, and 20.7 pCi/g, respectively. No significant deviations from expected Gaussian distributions were observed in any case. No surface migration of radioactive contaminants has occurred. From operational history, we know that the likelihood of buried radioactive debris outside of each open-field pit is acceptably small and that no subsurface measurements are necessary.

Within the two open-field pits, however, radioactive contamination was detected in the past. From the chemical characterization performed for CERCLA Phase II in March 1987, (Reference 18), it has been estimated that 37,000 ft^3 of soil and debris will be excavated from both open-field pits and disposed of appropriately. Assuming a 200 pCi/g (Reference 23, 29, Table 7.1) average uniform concentration of Cs-137, or a weighted average concentration of 273 pCi/g Cs-137, (Reference 23, 25, 29, Table 7.1) an upper range of total Cs-137 activity is 300 mCi to 410 mCi. This upper range is from averages of soil samples from locations known to be contaminated. Further radiological characterization is warranted in each open-field pit. No radiological health hazard exists in the entire Burn Pit area.

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29. "Radiological Information on Old Sodium Disposal Area," IL from F. H. Badger to R. J. Tuttle, RI, April 23, 1987.

APPENDIX A. DESCRIPTION OF NUCLEAR INSTRUMENTATION

During the radiological survey, soil samples and miscellaneous crud items were analyzed for radioactivity content by one or more of the following nuclear instrumentation systems. These systems, when calibrated and used properly, give results which are as good as is reasonably achievable.

A.1 Gamma Spectrometry Analyzer

Gamma spectrometry of selected samples including all soil 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 wipe test samples, where appropriate, 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

A Ludlum model 2220 portable scalar/ratemeter coupled to a gamma probe was used during the course of this survey. The 2220 has a six decade LCD readout; combination four decade linear and log rate meter; adjustable HV threshold, and window positions, with readouts on digital display; audio provided by unimorph speaker with pitch change in relation to count rate; and preset electronic timer.

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 efficiency of the probe coupled to the 2220 for Cs-137 gamma rays is about 215 cpm/μR/h. Because of limitations with the Cs-137 calibration range and because of fluctuations in ambient background radiation, this instrument was calibrated 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

MAR 05 1985

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:

02067RL

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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

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.

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.

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.

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

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

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

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

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

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

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

D. AUTHORIZED LIMITS FOR RESIDUAL RADIOACTIVITY

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

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

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

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

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

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

c. 5480.2, Hazardous and Radioactive Mixed Waste Management

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

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

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

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

h. 5484.2, Unusual Occurrence Reporting System

i. 5820.2, Radioactive Waste Management

E.1 Interim Storage

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

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

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

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

E.2 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

H. REFERENCES

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

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

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

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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.]

APPENDIX C. GAMMA SPECTROMETRY RADIONUCLIDE GAMMA-SIGNATURE 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 Energy (keV)	Half-Life % 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. GAMMA SPECTROMETRY DATA FOR BURN PIT SOIL SAMPLES

This appendix lists the radioactivity concentrations of gamma emitting radionuclides identified in each sample by gamma spectrometry. Blank entries mean that that radionuclide was not identified in the sample. Sample numbers are designated with B = Background, E = East Burn Area, N = North Area, S = South Area, and W = West Area. The Cartesian coordinate identifies the specific sampling location in the grid. Sample mass corresponds to the 450 ml volume of a Marinelli beaker. Since none of the samples showed conclusive evidence of U-235 activity, the amount shown in column 5 is derived from the U-238 activity at 0.7% by weight.

SAMPLE NUMBER	MASS gms	RADIONUCLIDE ACTIVITY (pCi/g) U-238	TH-232	U-235	K-40	CS-137
E 1-6	605.98	.90	1.13	.04	18.70	
E 1-9	584.36	.94	1.10	.04	18.33	
E 2-1	556.56	1.48	1.33	.07	20.68	.77
E 2-7	556.34	1.28	1.11	.06	19.09	
E 3-5	612.16	.96	1.12	.04	15.42	
E 3-8	509.36	.96	1.15	.04	18.62	
E 4-3	689.36	1.38	1.14	.06	19.64	.29
E 4-8	523.54	1.03	1.30	.05	21.39	.69
E 5-10	595.49	1.06	1.37	.05	22.44	.27
E 5-4	667.6	.84	1.10	.04	18.23	
E 5-8	627.65	.83	1.63	.04	20.23	.21
E 6-7	614.67	1.08	1.49	.05		
E 6-9	620.08	1.20	1.34	.05	15.22	
E 7-10	671.92	.85	1.35	.04	17.52	
E 7-6	609.87	1.11	1.41	.05	18.79	
E 8-9	634.43	.88	1.27	.04	19.32	
N 1-15	551.89	.97	1.79	.04	18.12	.60
N 1-3	673.45	.94	1.14	.04	14.85	
N 1-8	557.79	.97	1.23	.04	17.93	.22
N 11-8	609.43	1.02	1.26	.05	17.21	.25
N 12-11	731.13	1.07	1.56	.05	13.68	
N 13-13	582.79	1.43	1.60	.06	17.16	
N 15-12	645.62	1.14	1.37	.05	21.10	.55
N 15-17	567.01	.86	1.56	.04	20.14	.18
N 2-6	637.98	.77	.90	.03	20.38	.39
N 3-15	620.75	1.16	1.18	.05	20.94	.25
N 3-9	689.08	.83	.69	.04	19.65	
N 4-11	691.94	.88	1.26	.04	14.45	
N 4-17	665.62	.74	.95	.03	15.02	
N 4-7	547.02	.98	.99	.04	18.28	.39
N 5-4	609.26	.91	1.12	.04	24.62	
N 6-2	627.35	.93	1.11	.04	21.58	
N 6-8	642.37	1.00	1.21	.04	15.38	
N 7-15	505.83	1.19	1.54	.05	19.77	
N 7-6	668.58	1.19	1.17	.05	21.24	.38
N 8-1D	607.11	1.07	1.13	.05	22.30	.10
N 8-11	631.87	1.44	.96	.06	20.80	
N 8-17	560.71	.88	1.13	.04	20.99	.44
N 8-5	563.42	.91	.91	.04	21.83	1.26
N 9-3	628.03	.82	1.23	.04	19.94	
N 9-8	700.91	.91	.92	.04	14.27	.30
RS 1' W	569	1.23	1.37	.06	24.25	

SAMPLE NUMBER	MASS gms	RADIONUCLIDE ACTIVITY (pCi/g) U-238	TH-232	U-235	K-40	CS-137
W 10-4	604.68	.92	1.19	.04	15.39	
W 10-8	560.24	.85	1.28	.04	21.38	
W 11-16	510.6	1.04	1.23	.05	21.58	
W 11-18	585.04	1.04	.96	.05	20.02	
W 11-18	511.25	1.08	1.24	.05	20.19	
W 11-24	588.68	.98	.89	.04	19.98	.18
W 11-6	676.81	.80	1.01	.04	15.37	
W 12-12	622.17	1.01	1.22	.05	20.27	
W 12-12	622.17	.84	1.23	.04	18.36	
W 12-14	689.25	1.12	1.36	.05	20.21	
W 12-20	537.14	1.00	1.01	.04	19.21	
W 12-22	509.83	1.41	1.27	.06	22.89	
W 12-4	723.64	1.14	1.27	.05	19.19	
W 12-4	723.64	1.16	1.28	.05		
W 12-8	525.83	1.12	1.35	.05	16.69	
W 12-8	525.83	1.05	1.18	.05	19.28	
W 15-10	612.17	.76	1.10	.03		
W 15-11	410.16	1.33	1.41	.06		
W 15-11	410.16	1.11	1.18	.05		
W 16-14	518.93	1.26	1.58	.06	21.12	
W 16-8	553.11	1.00	1.27	.04	22.56	
W 18-11	566.46	.76	1.15	.03	19.22	
W 18-8?	569	1.09	1.20	.05	18.37	
W 18-9	622.81	1.21	1.27	.05	19.89	
W 2-1	687.36	.69	.94	.03	16.69	
W 2-11	491.59	1.16	1.36	.05	20.44	
W 2-14	425.14	.69	1.07	.03	23.52	
W 2-17	498.39	.89	1.27	.04	20.10	
W 2-17	498.39	.70	1.01	.03	19.47	
W 2-24	381.86	.99	1.28	.04	18.82	
W 2-29	608.67	1.01	1.22	.05	19.42	
W 2-29 (1)	608.67	1.04	1.28	.05	22.74	
W 2-5	392.84	.91	1.07	.04	22.51	
W 20-10	599.87	1.08	1.38	.05	19.92	
W 22-11?	569	.90	1.26	.04	18.37	
W 3-18	380.92	1.74	1.96	.08	24.76	
W 3-18	380.92	1.22	1.95	.05	25.84	
W 3-21	569.57	.94	1.11	.04	17.56	
W 3-7	426.22					
W 3-7	426.22	.98	1.28	.04	18.60	
W 4-12	569	.85	1.23	.04	18.21	
W 4-13	673.84	.92	1.49	.04	17.66	
W 4-18	569	.94	1.06	.04	18.37	
W 4-20	569	.90	1.19	.04	20.58	
W 4-22	709.15	1.02	1.53	.05	19.09	
W 4-28	587.11	.84	1.19	.04	21.17	.22
W 5-14	616.42	.82	1.21	.04		
W 5-19	569	.85	1.14	.04	18.21	
W 5-22	641.71	1.13	1.13	.05	21.44	
W 5-6	654.77	.88	1.01	.04	21.02	

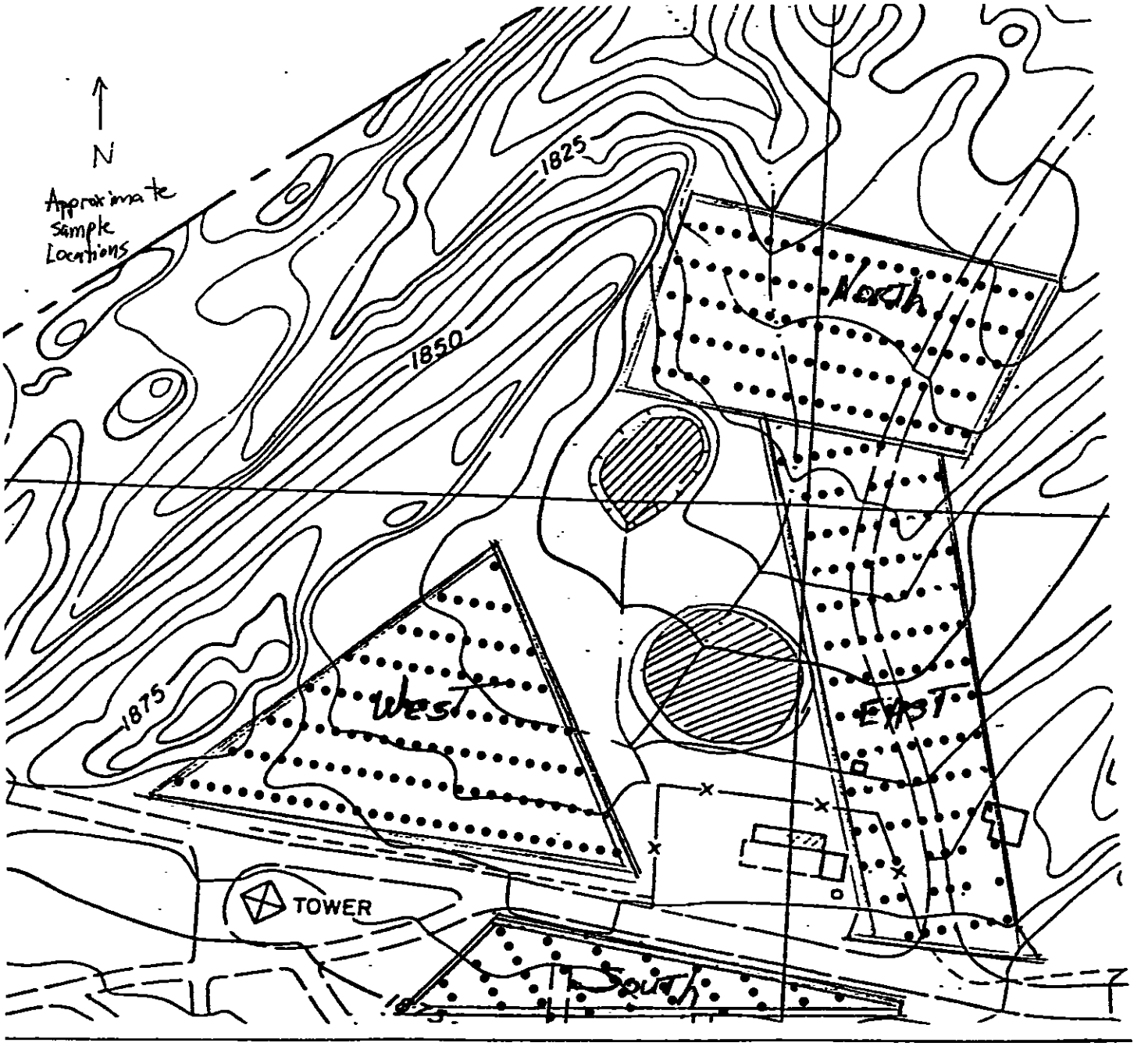
SAMPLE NUMBER	MASS gms	RADIONUCLIDE ACTIVITY (pCi/g) U-238	TH-232	U-235	K-40	CS-137
W 5-7	441.75	1.09	1.36	.05	23.27	
W 5-7	441.75	1.02	1.49	.05	24.65	
W 6-1	651.03	.67	1.23	.03	18.56	
W 6-11	455.69	1.10	1.61	.05	26.38	
W 6-11	455.69	1.00	1.18	.04		
W 6-16	433.08	1.16	1.45	.05	25.05	
W 6-16	433.08	1.07	1.09	.05	18.74	
W 6-18	403	1.32	1.38	.06	24.43	
W 6-18	403	1.03	1.14	.05		
W 6-27	577.76	.62	1.27	.03	17.31	
W 6-29	597.66	1.03	.99	.05	19.33	.35
W 6-4	621.48	.84	1.05	.04	18.30	
W 7-11	515.68	1.05	.99	.05	21.37	
W 7-17	616.47	.91	1.25	.04	20.46	
W 7-6	533.37	.93	1.30	.04	18.60	
W 7-6	533.37	.85	1.17	.04	20.59	
W 9-1	530.41	1.11	1.06	.05	16.92	
W 9-10	593.88	.96	1.53	.04	21.77	
W 9-17	559.43	1.17	1.16	.05	22.81	
W 9-19	521.49	1.22	1.43	.06	19.52	.14
W 9-5	536.33	1.10	1.48	.05	17.46	
e 10-10	550.54	.97	1.27	.04	22.46	
e 11-7	621.65	.94	1.25	.04	25.37	
e 11-9	569.73	.93	1.43	.04	24.59	
e 12-10	570.85	.81	1.47	.04	21.80	.38
e 12-20	569.6	.94	1.35	.04	26.38	
e 13-19	555.54	1.06	1.34	.05	21.98	
e 13-7	597	1.04	1.63	.05	26.94	
e 13-9	747.2	.77	1.12	.03	20.79	
e 14-10	723.63	.98	1.34	.04	23.09	.19
e 14-11	653.59	1.04	1.24	.05	22.57	.52
e 14-20	602.38	.92	1.15	.04	22.15	.33
e 14-8	691.63	.73	1.33	.03	23.54	
e 15-10	594.68	.96	1.35	.04	21.72	
e 15-19	563.8	.82	1.19	.04	19.71	.46
e 16-10	574.56	.96	1.24	.04	20.71	
e 16-17	541.24	1.08	1.57	.05	21.11	.53
e 16-20	502.43	.93	1.55	.04	24.69	.53
e 17-14	734.29	.80	1.41	.04	19.40	.20
e 17-17	591.68	.92	1.04	.04	24.54	.23
e 17-20	638.9	.96	.78	.04	18.87	1.36
e 18-14	732.16	.87	1.21	.04	20.95	
e 18-16	714.15	.76	.97	.03	20.22	
e 18-17	748.44	.98	1.21	.04	23.16	
e 19-14	635.9	.93	1.31	.04	22.28	.10
e 19-16	596.3	.96	1.04	.04	21.46	
e 20-16	753.3	.88	.82	.04	19.22	
e 22-14	688.61	1.03	1.42	.05	24.61	
e 22-17	646.33	.93	1.56	.04	21.62	
e 23-14	645.14	1.46	1.40	.07	22.81	

SAMPLE NUMBER	MASS gms	RADIONUCLIDE ACTIVITY (pCi/g) U-238	TH-232	U-235	K-40	CS-137
e 24-13	687.27	.86	1.26	.04	21.01	
e 24-16	656.22	.80	1.26	.04	21.95	.47
e 5-4	667.6	1.01	1.35	.05	21.40	
e 9-10	579.14	.87	1.31	.04	25.41	.13
e 9-7	724.06	.92	1.35	.04	20.11	
n 1-1	584	.97	1.46	.04	24.19	
n 10-14	583.99	.83	1.21	.04	23.92	.36
n 10-5	588	.97	1.28	.04	22.50	
n 10-9	676.01	1.01	1.39	.05	21.83	
n 11-12	597.05	.93	1.14	.04	21.50	
n 12-17	528.02	1.41	1.33	.06	22.90	.25
n 14-15	603.73	.92	1.37	.04	20.48	.31
n 2-12	668.46	1.01	1.40	.05	21.60	
n 3-4	677.82	.59	.72	.03	22.81	1.17
n 4-2	628.25	.76	1.33	.03	23.98	.23
n 5-9	741.86	.95	1.04	.04	22.37	
n 6-14	544.07	.96	1.50	.04	22.80	.30
n 7-9	591.63	1.39	1.45	.06	23.09	.14
n 8-1	784.35	.71	1.27	.03	19.71	
n 9-13	584.84	.98	1.41	.04	18.54	
s 1-12	389.09	.99	1.43	.04	22.03	
s 1-18	444.54	.89	1.40	.04	19.72	.28
s 1-24	545.17	1.10	1.32	.05	21.25	.30
s 1-6	452.98	.98	1.46	.04	22.46	.22
s 12-1	605	.94	1.13	.04	22.45	
s 12-11	689.2	1.00	1.03	.05	21.69	
s 12-17	623.3	.85	1.22	.04	23.11	.44
s 12-22	597.25	.69	.95	.03	20.12	
s 12-25	527.76	.78	1.17	.04	20.02	.20
s 12-5	723.05	.80	1.29	.04	19.54	
s 2-15	446.09	.87	1.49	.04	19.48	
s 2-21	400.23	1.24	1.35	.06	21.90	.15
s 2-27	401.98	.99	1.72	.04	22.68	.35
s 2-3	434.3	.99	1.34	.04	21.54	
s 2-9	471.93	1.03	1.32	.05	19.57	
s 3-11	531.78	.81	1.12	.04	19.36	.21
s 3-17	551.55	.84	1.31	.04	17.74	.25
s 3-5	527.66	1.13	1.54	.05	22.18	
s 6-14	597	1.02	1.13	.05	16.78	
s 6-20	513.99	1.09	1.15	.05	22.08	
s 7-1	576.63	.72	1.29	.03	21.58	
s 7-7	588.32	.74	1.07	.03	17.23	
w 11-8	582.32	1.06	1.19	.05	24.13	
w 13-10	627.04	1.07	1.34	.05	19.29	
w 13-15	601.01	1.12	1.07	.05	24.42	
w 13-18	689.17	.99	1.24	.04	23.57	
w 13-2	645.66	1.09	1.38	.05	21.28	.39
w 13-6	637.59	.85	1.09	.04	21.36	
w 14-12	649.23	.81	1.29	.04	19.65	.25
w 14-14	621.48	.96	1.10	.04	23.55	

SAMPLE NUMBER	MASS gms	RADIONUCLIDE ACTIVITY (pCi/g) U-238	TH-232	U-235	K-40	CS-137
w 14-19	601.72	1.15	1.28	.05	24.52	.20
w 14-4	666.56	.97	1.17	.04	23.25	
w 14-8	566	1.21	1.09	.05	21.37	
w 16-8	553.11	1.04	1.02	.05	21.51	
w 17-13	592.19	1.00	.95	.05	26.10	
w 19-12	593.86	1.24	1.29	.06	25.57	.31
w 2-19	565.46	1.27	1.43	.06	24.36	
w 3-13	724.24	1.01	1.37	.05	25.51	
w 3-15	349.31	1.14	1.39	.05	24.42	
w 3-18	380.92	1.18	1.39	.05	20.86	
w 3-23	551.04	1.22	1.48	.06	26.70	
w 3-24	489.47	1.37	1.28	.06	26.71	
w 3-4	488.63	1.22	1.48	.05	29.31	
w 4-10	447.5	1.16	1.29	.05	24.40	
w 4-17	435.15	.94	1.35	.04	23.56	
w 4-19	619.19	1.15	1.29	.05	25.60	
w 4-2	603.79	.64	.97	.03	17.37	
w 4-21	575.9	1.25	1.24	.06	24.19	
w 4-24	644.07	1.17	1.24	.05	24.67	.13
w 4-5	487.21	.91	1.05	.04	24.34	
w 6-24	703.25	.95	1.37	.04	24.10	
w 6-9	455.89	1.07	1.14	.05	21.72	
w 7-14	461.88	1.00	1.57	.04	24.23	
w 7-2	737.79	.77	1.03	.03	19.95	
w 7-24	605.13	1.06	1.50	.05	25.09	
w 7-27	594.36	1.11	1.11	.05	26.40	.17
w 8-10	549.39	.93	1.13	.04	18.95	
w 8-16	615.16	.99	1.34	.04	23.54	
w 8-20	799.48	1.11	1.29	.05	21.78	
w 8-22	639.84	.91	1.00	.04	24.71	
w 8-25	745.02	.98	1.24	.04	25.48	
w 8-3	737.13	1.00	1.17	.04	21.24	
(w 8-7	642.73	1.01	1.24	.05	20.82	

APPENDIX E. RADIOLOGICAL SURVEYOR'S SAMPLING GRID FOR ALL FOUR AREAS

Figure E.1 Sampling Areas



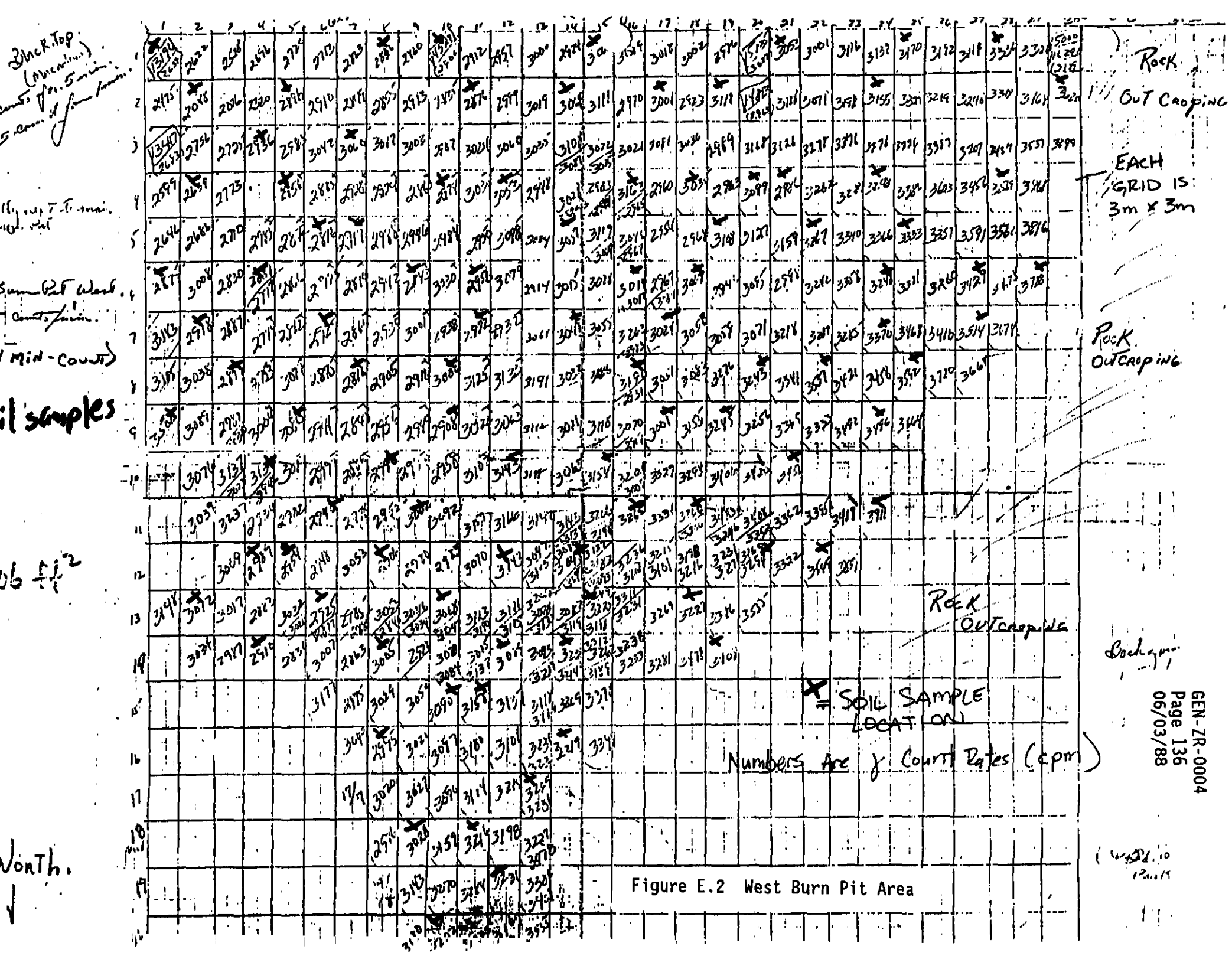
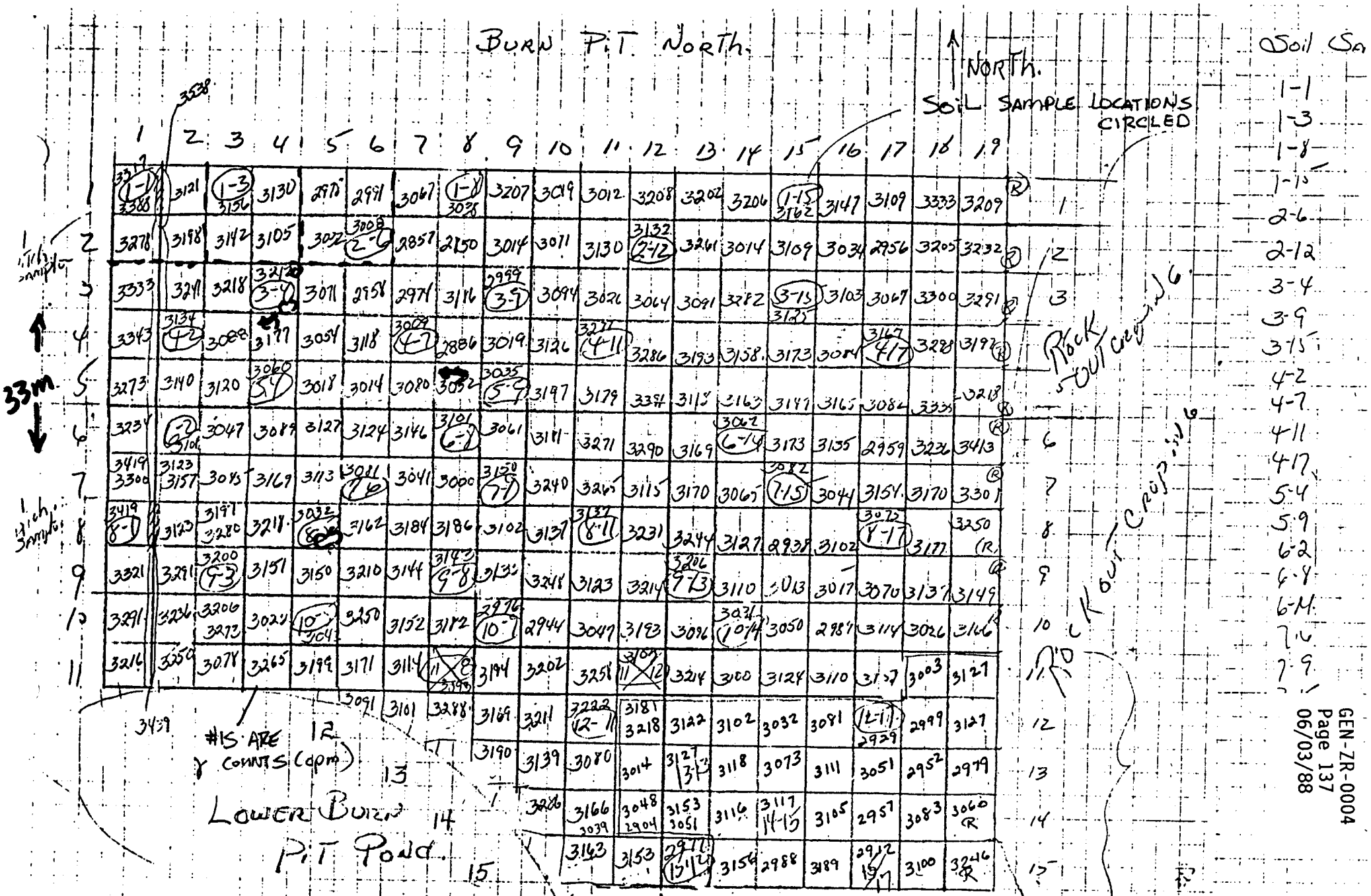


Figure E.2 West Burn Pit Area



Each grid is 3m x 3m

$227\ m^2 = 24,509\ ft^2$

Figure E.3 North Burn Pit Area

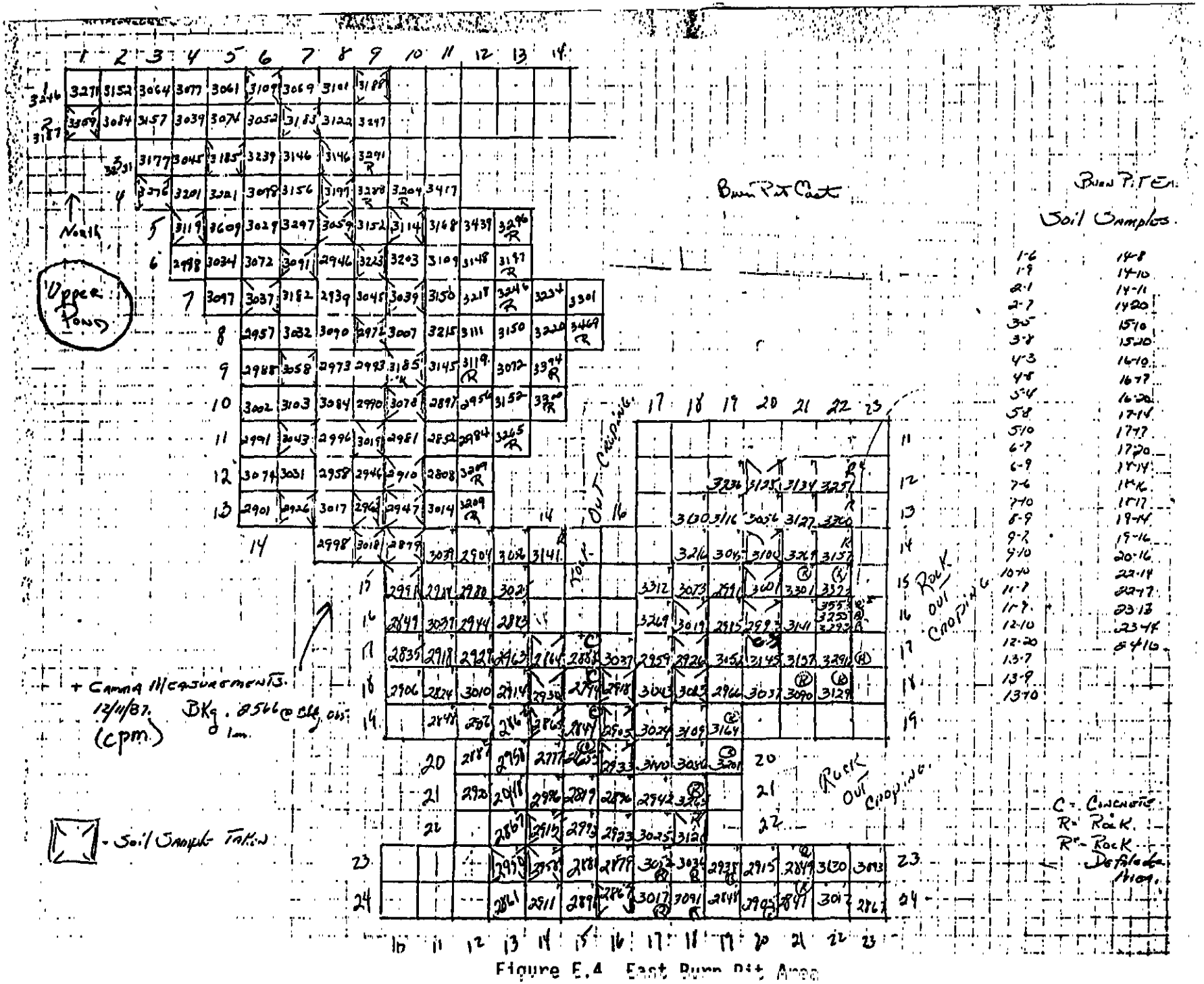


Figure E.4 East Burn Pit Area

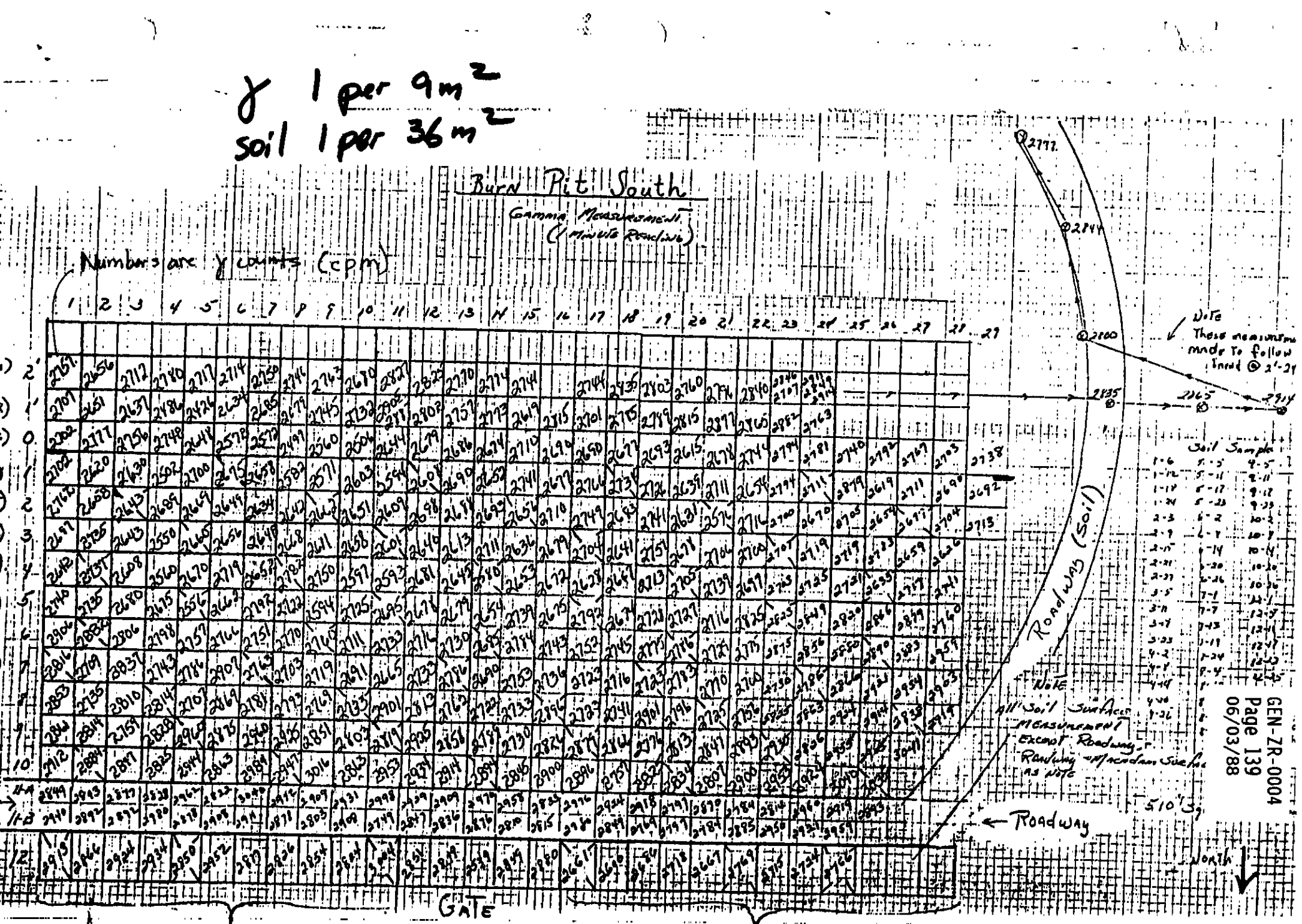


Figure E.5 South Burn Pit Area