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ABSTRACT

A radiological survey was performed at Buildings T030 and T641, located at Rockwell International's Santa Susana Field Laboratory (SSFL) Area IV, to clarify and identify those areas needing further radiological inspection or requiring remedial action. T030, formerly known as the Particle Accelerator Facility, and T641, formerly known as Shipping and Receiving, were operated by Rockwell International and its predecessors in support of AEC, ERDA, and DOE nuclear-related programs. T030 housed a Van deGraaf proton accelerator which used the $H^3(p,n)He^3$ reaction. T641 was a transfer point for all SSFL incoming and outgoing shipments, including radioactive materials. An area just south and west of T030 was used for storing palletized equipment items; some drums containing mixed fission products may have been stored there. No incidents which released radioactive material to the surrounding environment are known to have occurred. All operations were performed with sealed sources, or completely packaged radioactive material. No measurable neutron activation of building materials in T030 occurred. The purpose of this survey was to inspect these facilities and surrounding area to determine if any residual radioactive contamination was left behind from these operations to such an extent that further surveying or decontamination is warranted.

The interior of T030's accelerator room, T641's loading dock, and the surrounding area were surveyed for gamma-emitting contamination. Ambient gamma exposure rate measurements were performed on a 6-m square plot plan. The accelerator room and old pallet storage area outside of T030 were surveyed for beta contamination. Ten soil samples were collected outside of T030 and analyzed for tritium.

Results of this survey and analyses show that both facilities and surrounding area are "clean" of radioactive contaminants. Quantitative assessment and statistical analysis of gamma exposure rate measurements show that the area is equivalent to background. All beta measurements show no detectable activity. Tritium analysis of soil samples show in all cases, nothing statistically significant. No further investigation or remedial action is required in this area.

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1.0 INTRODUCTION

Building T030, formerly known as the Particle Accelerator Facility, and Building T641, formerly known as Shipping and Receiving, were surveyed and analyzed for residual radioactive material. Appropriate sections of each facility and surrounding area were inspected for radioactive contamination to determine whether insignificant quantities of contamination exist, further investigation is required, or remedial action is necessary. This radiological survey was conducted as prescribed in the "Radiological Survey Plan for SSFL," (Reference 4, section 5.4.3.2).

Buildings T030 and T641 are located in Area IV of Rockwell International's Santa Susana Field Laboratories (SSFL) in Ventura County, California. Building T030 was used from 1960 through 1964 as a Van deGraaf accelerator facility for performing activation experiments. Neutrons were produced by the $H^3(p,n)He^3$ nuclear reaction. Activation of building materials was negligible because drums of borated water were used around the target to thermalize and capture neutrons. A large concrete block built as a shield outside the north wall of T030 is not activated. The only suspect radionuclide is tritium, which was used as the target for producing neutrons (Reference 22). In 1965, T030 was converted to an office building, but the accelerator remained on-site in an unused condition until at least 1966. An asphalt area south of T030 was fenced-in and used in the early 1960s as a storage area for palletized items. Drums containing mixed fission products may have been stored in this area. Building T641 was the SSFL Shipping and Receiving building through 1985. All shipments made to or from SSFL were coordinated and handled through T641. Radioactive and nuclear material shipments were handled only on the outdoor dock; they were never stored in the warehouse. Shipments were always fully packaged and never opened at this location. No leaks or contamination incidents ever occurred. T641 is still used as a receiving and warehouse facility, but no radioactive or nuclear materials are shipped through this location. Because radioactive materials were handled in these facilities, a radiological survey was performed to document the current radiological condition.

As part of the DOE SSFL Site Survey (Reference 4, section 5.4.3.2), a radiation survey was performed in these buildings in locations where radioactive material was handled, and in surrounding areas to determine if any residual contamination exists on the site. Ambient gamma exposure rates were measured on a 6-m by 6-m grid. These measurements are sensitive to radiations emitted from radioactive materials handled at each facility: mixed fission products at T641 and at a storage yard south of T030; and activation products inside T030. Soil samples were collected near T030 and analyzed for tritium. A surface survey for beta-emitting contamination was performed outside the north and west walls of T030 and on interior coving and I-beams. These beta measurements were "for indication."

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

Radiation measurements are compared against DOE residual radioactivity limits specified in "Guidelines for Residual Radioactivity at FUSRAP and Remote SFMP Sites," (Reference 1). This guide generally agrees with previously published guides and standards, including ANSI Standard N13.12 (Reference 7), Regulatory Guide 1.86, and USNRC License SNM-21 (Reference 2). Limits for acceptable ambient gamma exposure rates differ between the DOE and NRC. DOE specifies 20 μR/h above background while NRC specifies 5 μR/h above background as acceptable gamma exposure rate limits. Natural "background" at SSFL is very difficult to determine because of a large observed variability in the measurements. Because of this large variation, total-gross gamma measurements are plotted and compared against

three independent "natural" background distributions. Then the average "background" exposure rate of the three "natural background" distributions is subtracted from the T030 and T641 data set to compare the results against the 5 μR/h above background criteria.

2.0 IDENTIFICATION OF FACILITY PREMISES

2.1 Location

Building T030 and T641 are located within Rockwell International's Santa Susana Field Laboratory (SSFL) in the Simi Hills of southeastern Ventura County, California, adjacent to the Los Angeles County line and approximately 29 miles northwest of downtown Los Angeles. The SSFL location relative to the Los Angeles area and surrounding vicinity is shown in Figure 2.1. An enlarged map of the neighboring SSFL communities is presented in Figure 2.2. A plot plan of the western portion of SSFL which includes Area IV, Figure 2.3, details the location of Buildings T030 and T641. The buildings are government owned and located off of the west side of "G" Street on 10th Street.

2.2 Building Characteristics and Site Topography

Building T030 was constructed in 1958 as a Particle Accelerator Facility. The building has a total enclosed area of 2,311 ft^2. The facility consists of two connecting sections, both with steel framing, siding, and roofs. The rear open (west) section was constructed at a right angle to the front office (east) section. The rear section was configured to accommodate a low-voltage particle accelerator used as a proton on tritium (P-T) neutron source. An outside concrete wall, north of the west section, provided shielding for the accelerator beam. Men's and women's restrooms were built into the facility so that the facility provided a complete self contained accelerator test installation.

The shipping and receiving building, T641, is adjacent to building T030 and was constructed in 1964. The building has 7,680 ft^2 which is composed of 7,440 ft^2 for storage and 240 ft^2 for office space. A loading dock is provided for receiving and shipping material and equipment.

Figure 2.1 Map of Los Angeles Area

Figure 2.2 Map of Neighboring SSFL Communities

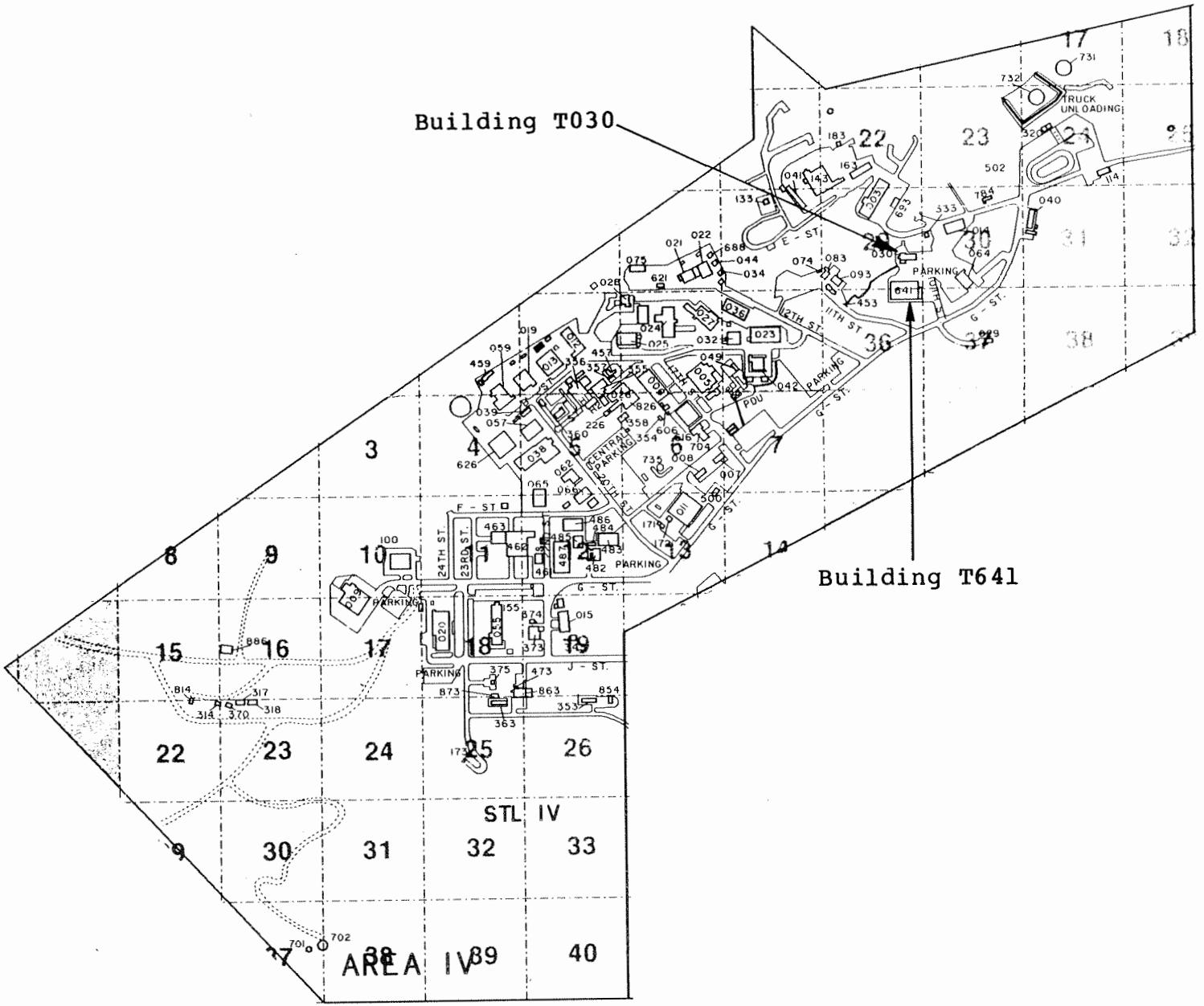


Figure 2.3 SSFL Area IV Layout, Showing Location of Buildings T030 and T641

An early aerial photo showing Building T030 and the site location of T641 is presented in Figure 2.4. Building T030 and T641 plot plans are shown in Figures 2.5 and 2.6, respectively.

These facilities are accessible off of "G" Street, on 10th Street. Between and east of these buildings is a large asphalt parking lot. A fenced-in area between T030 and T641 was previously used as a palletized material holding area. To the north of T030, south of T641, and west of both buildings are outcroppings of Chatsworth sandstone formation. This formation is only about 50 ft from the north and west sides of T030.

2.3 Building Utilization and Present Radiological Condition

Building T030 was designated Particle Accelerator Facility following construction, and a Van deGraaf accelerator was moved into the facility in 1960. The Van deGraaf could accelerate protons to tens of microamperes current of accelerated protons whose energy could be continuously varied from a few-hundred keV to the maximum energy of the machine, which was at least 1 MeV. The charged particle current was well focused, a few millimeters in diameter, with an energy spread of about 2 keV. Neutrons were produced from the Van deGraaf using the (p,n) reaction with tritium, $T(p,n)He^3$. Five-gallon cans of borated water were used for shielding around the machine. The accelerator was used through 1964 at which time it was mothballed. The accelerator remained at T030 for several years after 1964, even though it was not used. In 1966, a smear survey of the accelerator showed significant tritium contamination, (Reference 22), see Appendix E. Tritium contamination was localized to the accelerator. It is not believed that contamination was spread or released from the accelerator to surrounding areas. Following removal of the accelerator sometime during or after 1966, the facility was surveyed and cleared for other uses. The building has subsequently been used as an office building for purchasing and on-site traffic. An area just south of T030 was fenced-in and used for storing palletized containers and equipment. Mixed-fission-product contaminated pallets may have been stored there for short time

periods. This storage activity is not verifiable. No known contamination problems ever occurred at T030. A photograph of the building looking west is shown in Figure 2.7 and a view from the north shows the concrete wall used as a beam shield, Figure 2.8.

Following construction in 1964, Building T641 was designated Shipping and Receiving. The building and dock were used for both nuclear and non-nuclear materials shipping and receiving. Non-nuclear and non-radioactive materials were stored in the warehouse. Nuclear and radioactive materials were only handled on the outdoor loading dock. Individual gamma-graphic sources, radioactive laundry, and shipping casks were handled at this location. A radiation detector alarm system was installed in the dock area for radiation safety purposes. Radioactive materials being shipped from or received at SSFL were always completely containerized and packaged when received at T641. No containers were ever opened at this facility. In 1985, during the Atomics International/Rocketdyne reorganization, T641 was redesignated as an internal on-site moving and transport facility. Radio-active materials and contaminated equipment are no longer handled here. The radiation alarm system was removed. No contamination problems ever occurred at this facility. Photographs of Building T641 are shown in Figures 2.9 and 2.10.

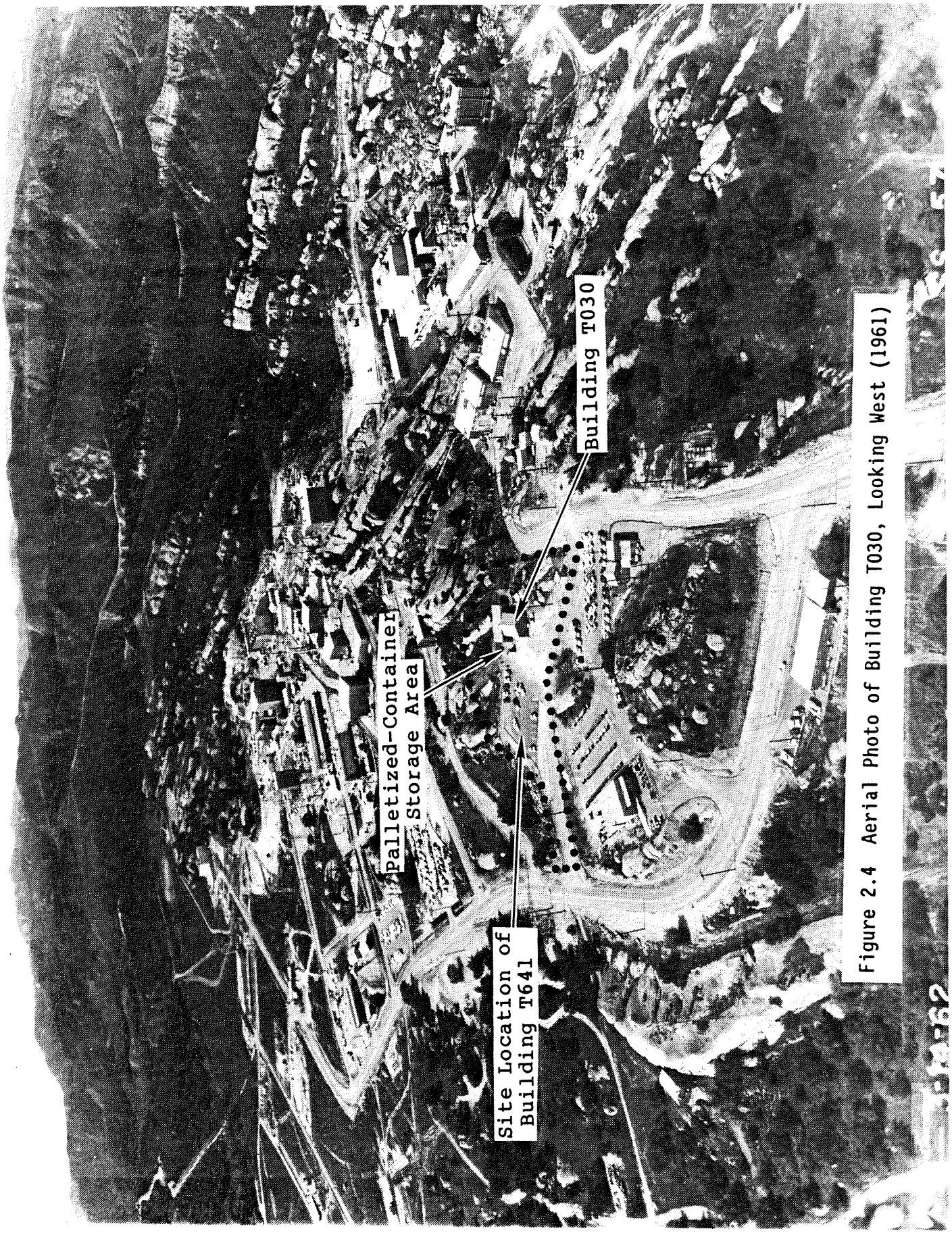


Figure 2.4 Aerial Photo of Building T030, Looking West (1961)

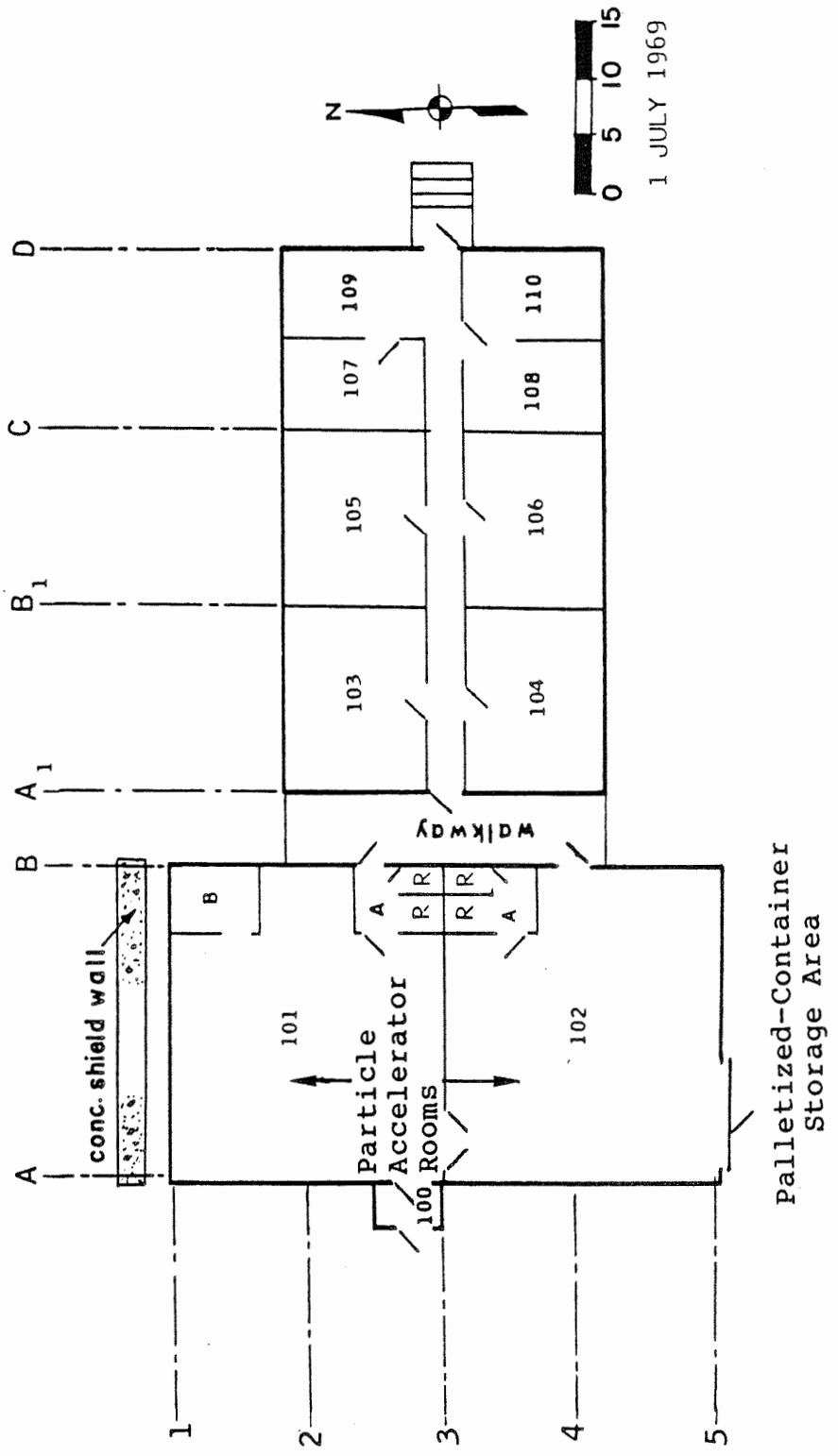


Figure 2.5 Plot Plan of the Particle Accelerator Facility, Building T030

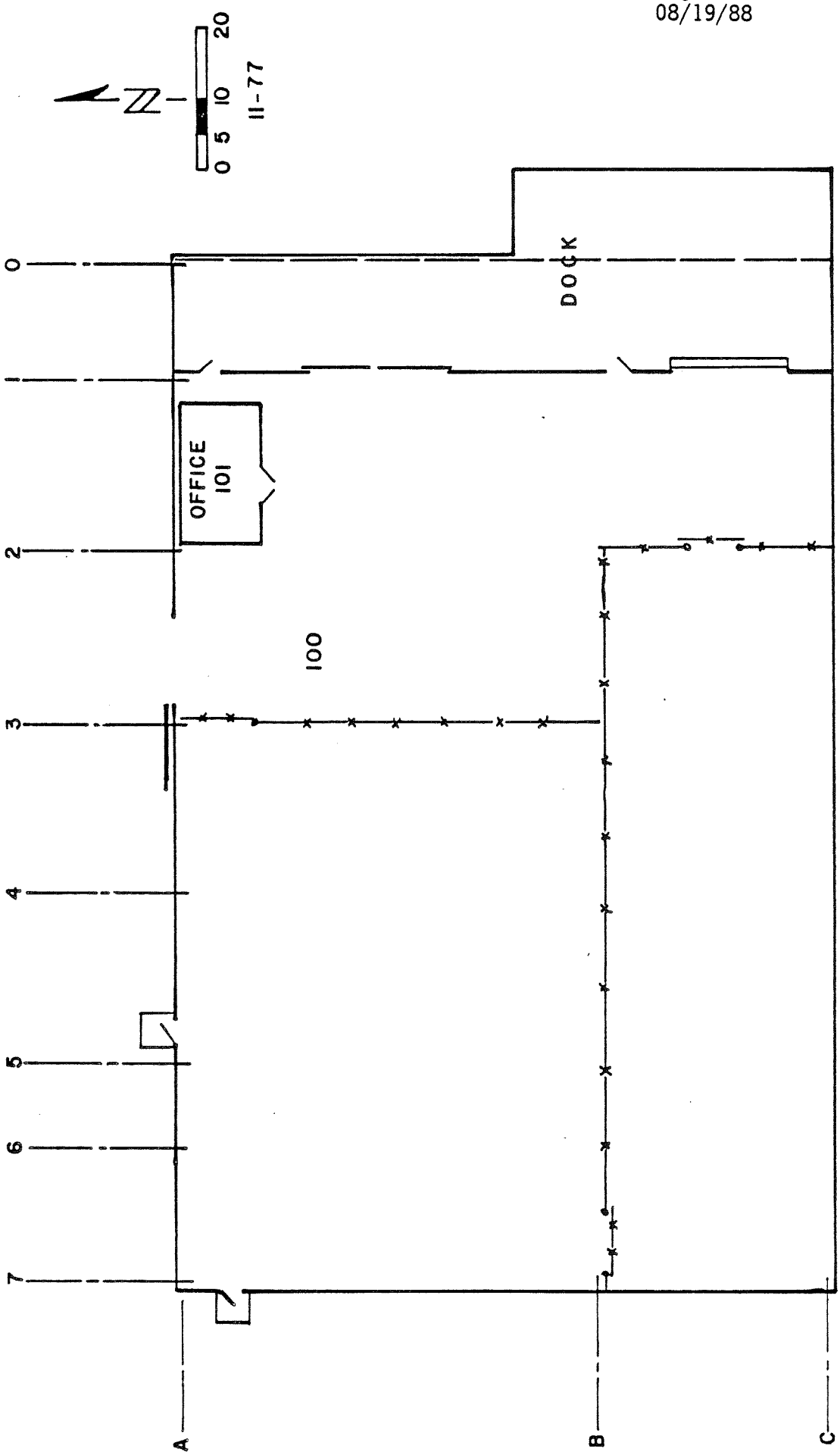


Figure 2.6 Plot Plan of the Receiving and Storage Building T641

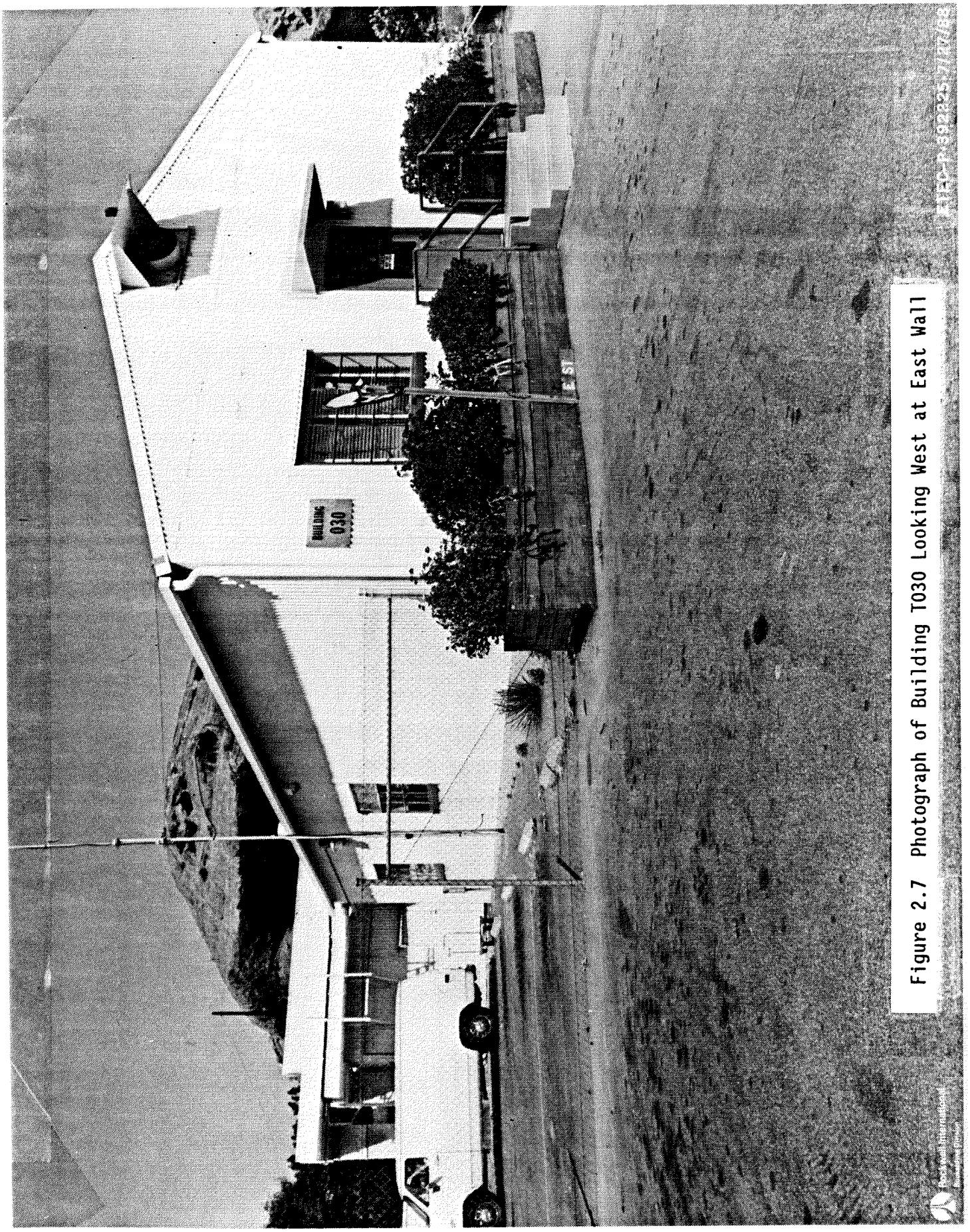


Figure 2.7 Photograph of Building T030 Looking West at East Wall

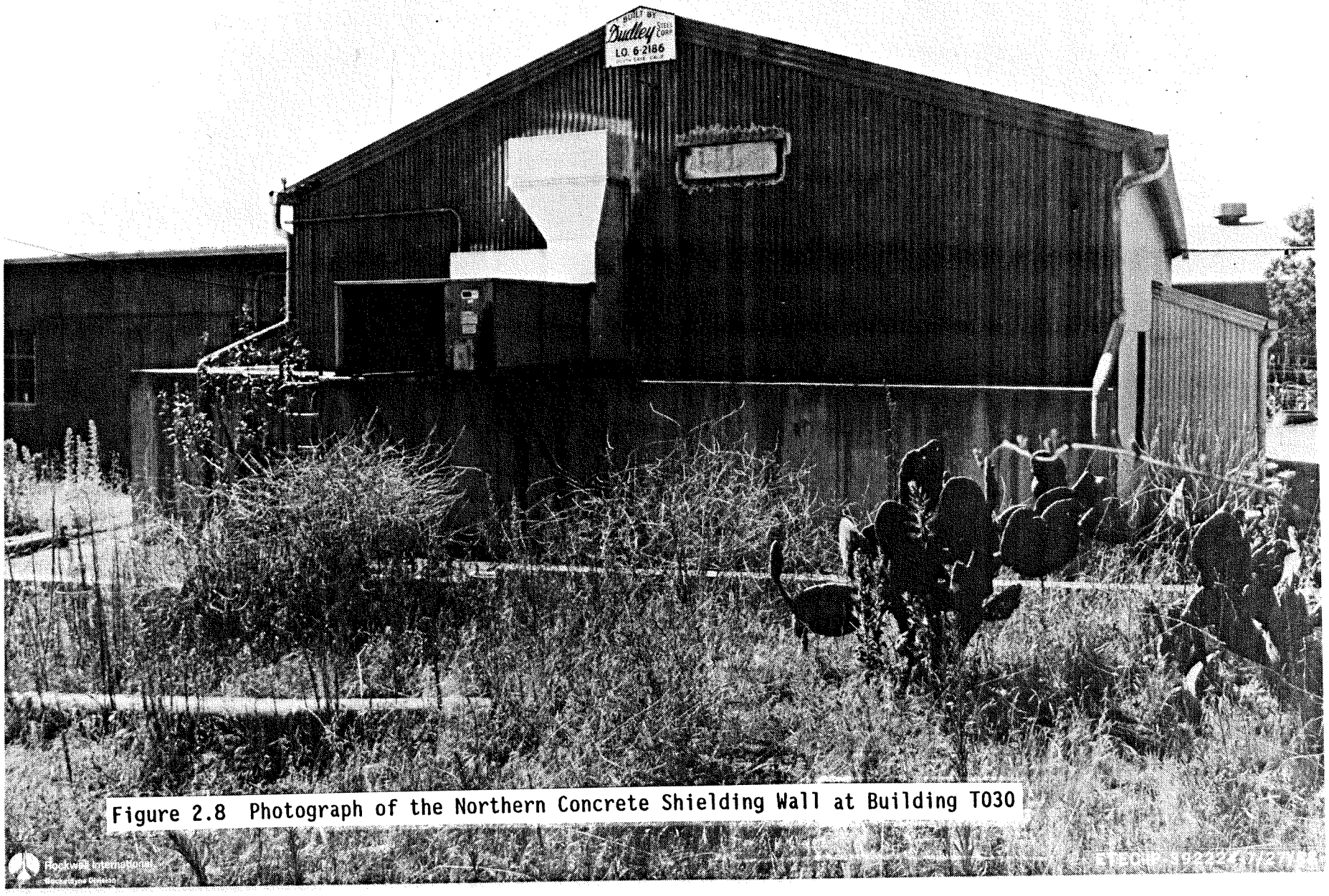


Figure 2.8 Photograph of the Northern Concrete Shielding Wall at Building T030

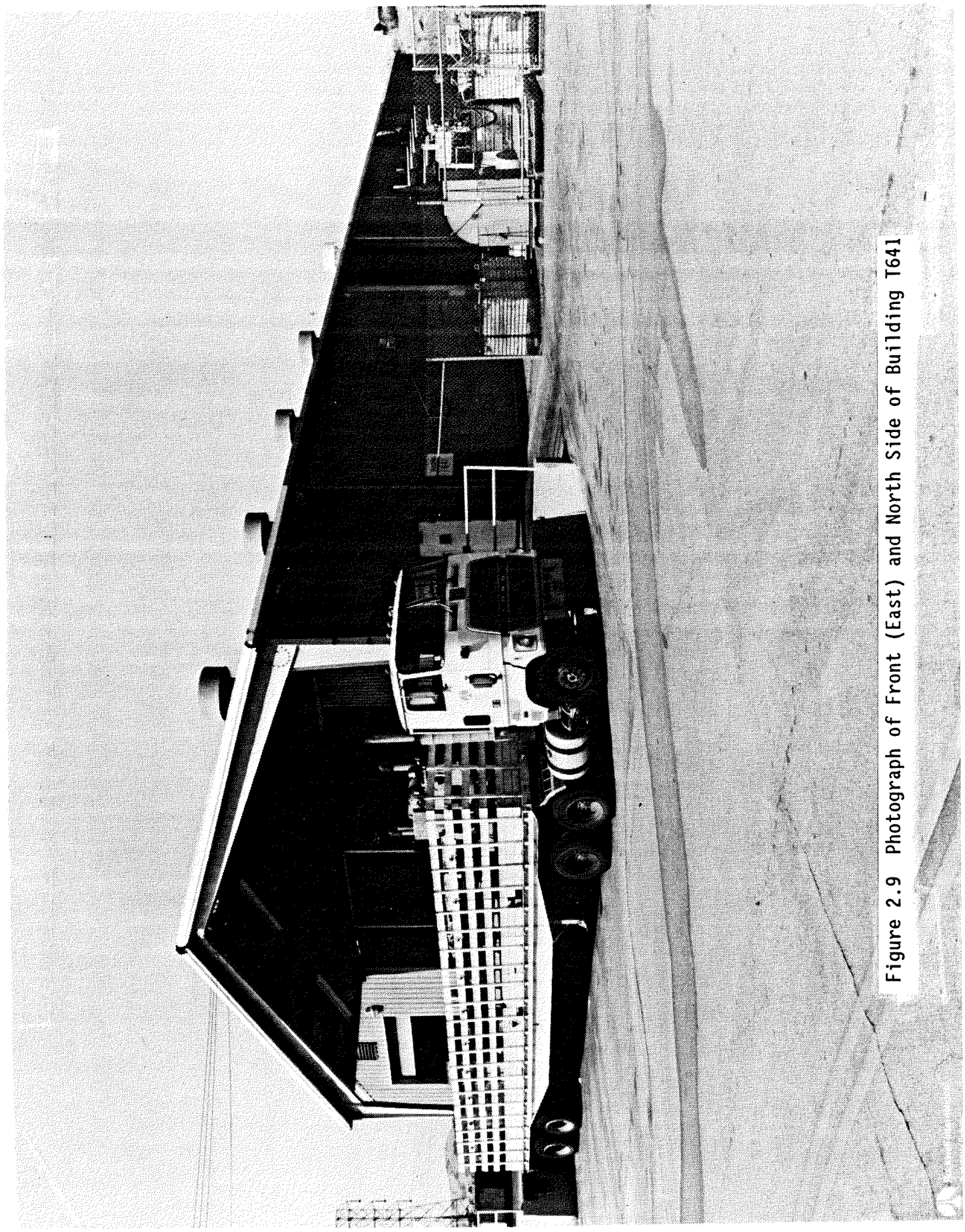


Figure 2.9 Photograph of Front (East) and North Side of Building T641

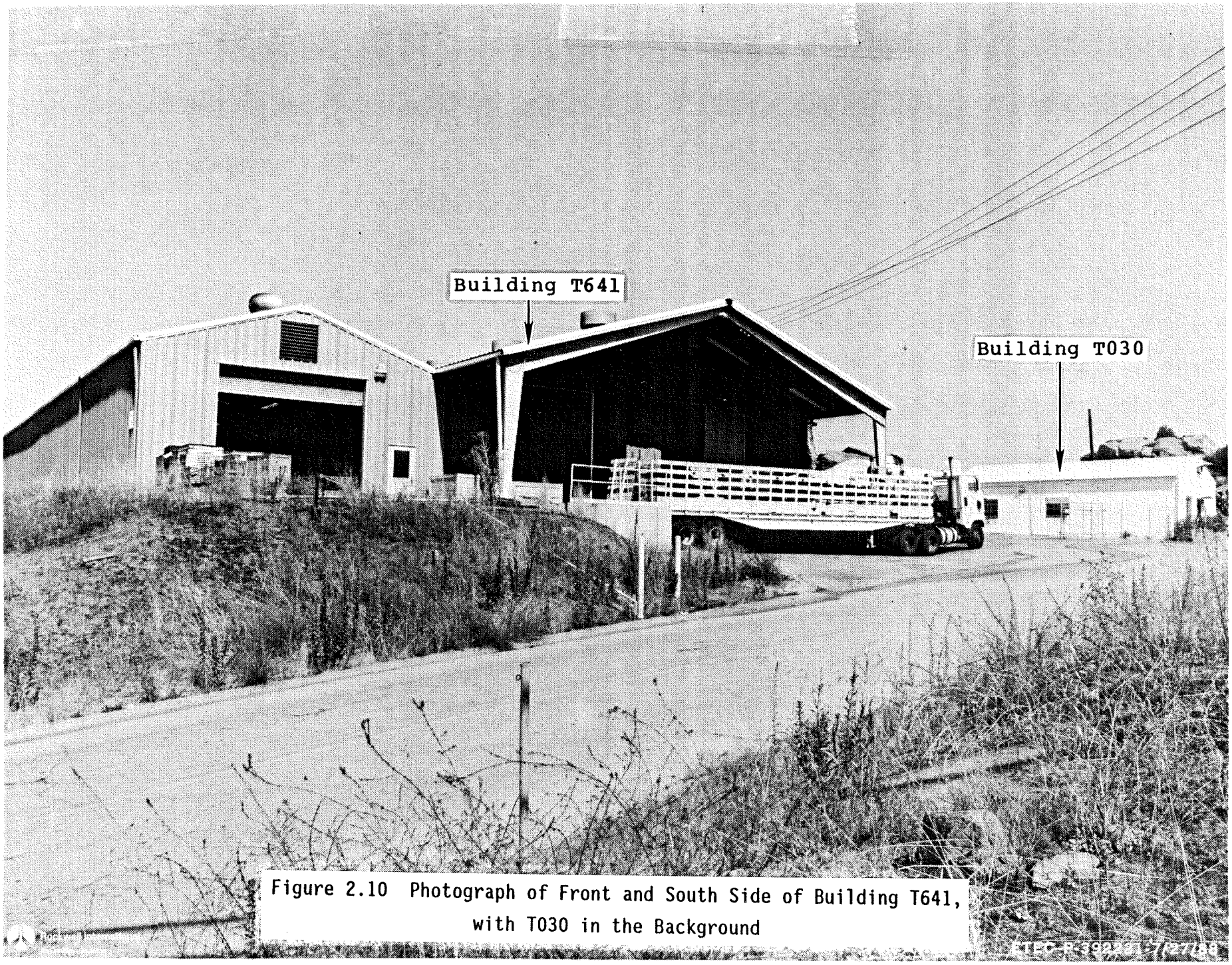


Figure 2.10 Photograph of Front and South Side of Building T641, with T030 in the Background

3.0 SURVEY SCOPE

The following areas were radiologically inspected by measuring ambient gamma exposure rates 1 meter above the surface:

1. Interior of T030 where the accelerator was used;
2. Loading dock of T641 where containerized packages of radioactive materials were received, handled, and shipped;
3. A small area south of T030 used for storing palletized items; and
4. Surrounding outdoor areas.

One hundred fourteen gamma exposure rate measurements were made in this area. Ten soil samples were collected near T030 for tritium analysis. A beta survey "for indication" was performed in the T030 accelerator room and palletized-container storage area south of T030. Gamma exposure rate data were analyzed statistically by sampling inspection by variables techniques against appropriate residual contamination acceptance limits.

The survey scope was set forth by the "Radiological Survey Plan for SSFL," (Reference 4, section 5.4.3.2):

1. Survey building surface (appropriate rooms) and grounds, especially pallet storage area. Survey for mixed fission products (MFP). Analyze soil north of T030 for tritium.

2. Survey surface pathway to the L-85 (T093).

3. Survey surface north (excluding sandstone areas) to ETB (T003) and T093 down to "G" Street.

These areas are shown within the boundary in Figure 2.4.

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. Although alpha and beta surface and soil contamination measurements were not required to be performed for this radiological inspection, these limits are presented for information. Measurements of average surface alpha/beta contamination are averaged over an area of not more than 1 m^2. The maximum allowable alpha/beta contamination level applies for a single area of not more than 100 cm^2 in that 1 m^2. Allowable removable alpha/beta contamination is based on a surface wipe with area equal to 100 cm^2.

Limits for soil and water radioactivity concentrations are also applicable on an as-required basis. The limits used here for alpha contamination, are based on Ra-226 contamination (Reference 1).

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

1. Characterization Level - that level of radioactivity which is below 50% of the maximum acceptable limit. This level is typical of natural background levels, or slightly above, and requires no further action.

Table 3.1 Buildings T030 and T641 Maximum Acceptable Contamination Limits

Criteria	Alpha (dpm/100 cm^2)	Beta (dpm/100 cm^2)
Total Surface, averaged over 1 m^2	5000	5000
Maximum Surface, in 1 m^2	15000	15000
Removable Surface, over 100 cm^2	1000	1000
Ambient Gamma Exposure Rate*	5 μR/h above background	
Soil Activity Concentration**	21 pCi/g 31 pCi/g	100 pCi/g
Water Activity Concentration***	$1x10^{-4}$ μCi/ml	$1x10^{-5}$ μCi/ml

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

** The alpha activity concentration limit for Ra-226 is 5 pCi/g (Reference 1) plus that contribution from naturally occurring radioactivity, (about 16 pCi/g from Reference 17, p. 93) averaged over the first 15 cm of soil below the surface. At a depth greater than 15 cm below the surface, 15 pCi/g averaged over 15-cm-thick layers of soil plus "background" is the limit. The total beta activity concentration limit is 100 pCi/g, including background which is about 24 pCi/g. 100 pCi/g applies for tritium concentrations at T030.

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

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 T030 and T641 radiological survey, both buildings and surrounding area were treated as a single sample lot for characterization and interpretation. Figure 3.1 shows the survey sampling lot plan. This figure shows that the roadway (10th Street), fenceline, interior area of T030, and loading dock of T641 were inspected.

6-m square grids were superimposed over the terrain and building areas. One ambient gamma exposure rate measurement was made in each 36-m^2 area. Location (1,1) was the northeastern-most grid on the site, near the T513 parking lot. The northwestern-most grid is conventionally used as location (1,1), but the surveyor's map, in this case, was accidentally rotated clock-wise 90° so that true north was west on his map.

3.3 Ambient Gamma Exposure Rate Measurements

In each 36-m^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 in x 1 in NaI crystal coupled to a photomultiplier tube and fed to a Ludlum 2220-ESG scaler, at 1 m from the ground. In each cell, a 1-min. count was collected and converted to $\mu R/h$. The measurement location and exposure rate were recorded in tabular form. About 114 1-min. measurements were acquired.

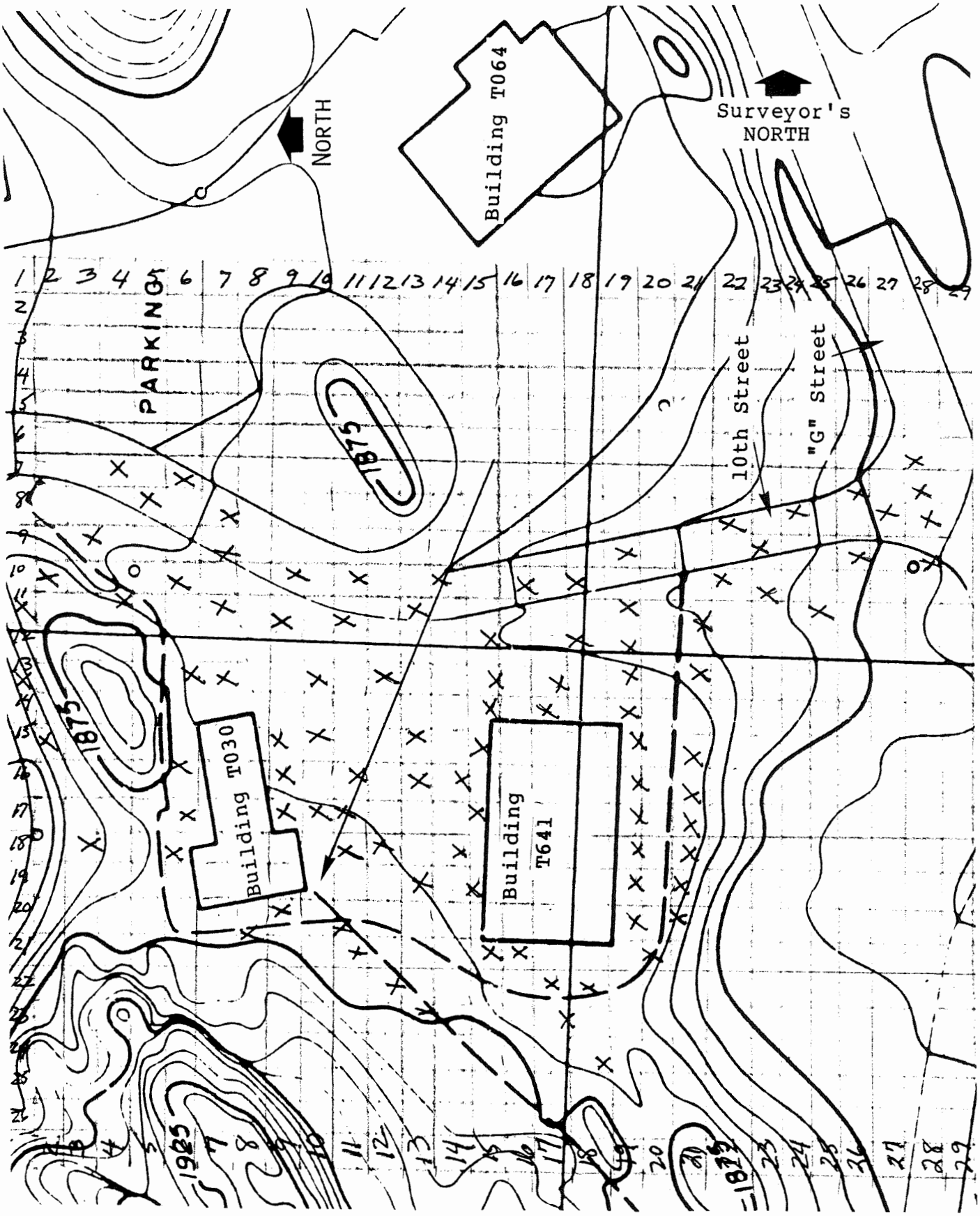


Figure 3.1 Survey Sampling Lot Plan for T030 and T641

3.4 Surface Soil Samples

Surface soil sampling was required by the site survey plan (Reference 4) near the west and north sides of T030. Figure 3.2 shows the soil sampling locations. 10 samples were collected randomly in this area and analyzed for tritium by U.S. Testing, Inc. U.S. Testing requires that a submitted soil sample contain at least 5 ml of water to achieve an analytical sensitivity of 500 pCi/l per sample. SSFL soil during the summer months has a very low moisture content, probably close to 5% with a 10% maximum. Each soil sample was a liter in volume to achieve the total volume of water required (5 ml).

3.5 Alpha and Beta Contamination Measurements

No gamma exposure rate measurement indicated a level exceeding the 50% characterization level; therefore, no further investigation was required based on that criteria. Measurements of beta activity "for indication" were made on the asphalt storage area surface south of T030, and inside the T030 accelerator room on wall coving and I-beams.

Beta activity measurements were made by using a Ludlum model 44-9 pancake Geiger-Mueller probe (active area = 20 cm^2) coupled to a Ludlum model 12 count rate meter. This detector was calibrated using a Tc-99 source.

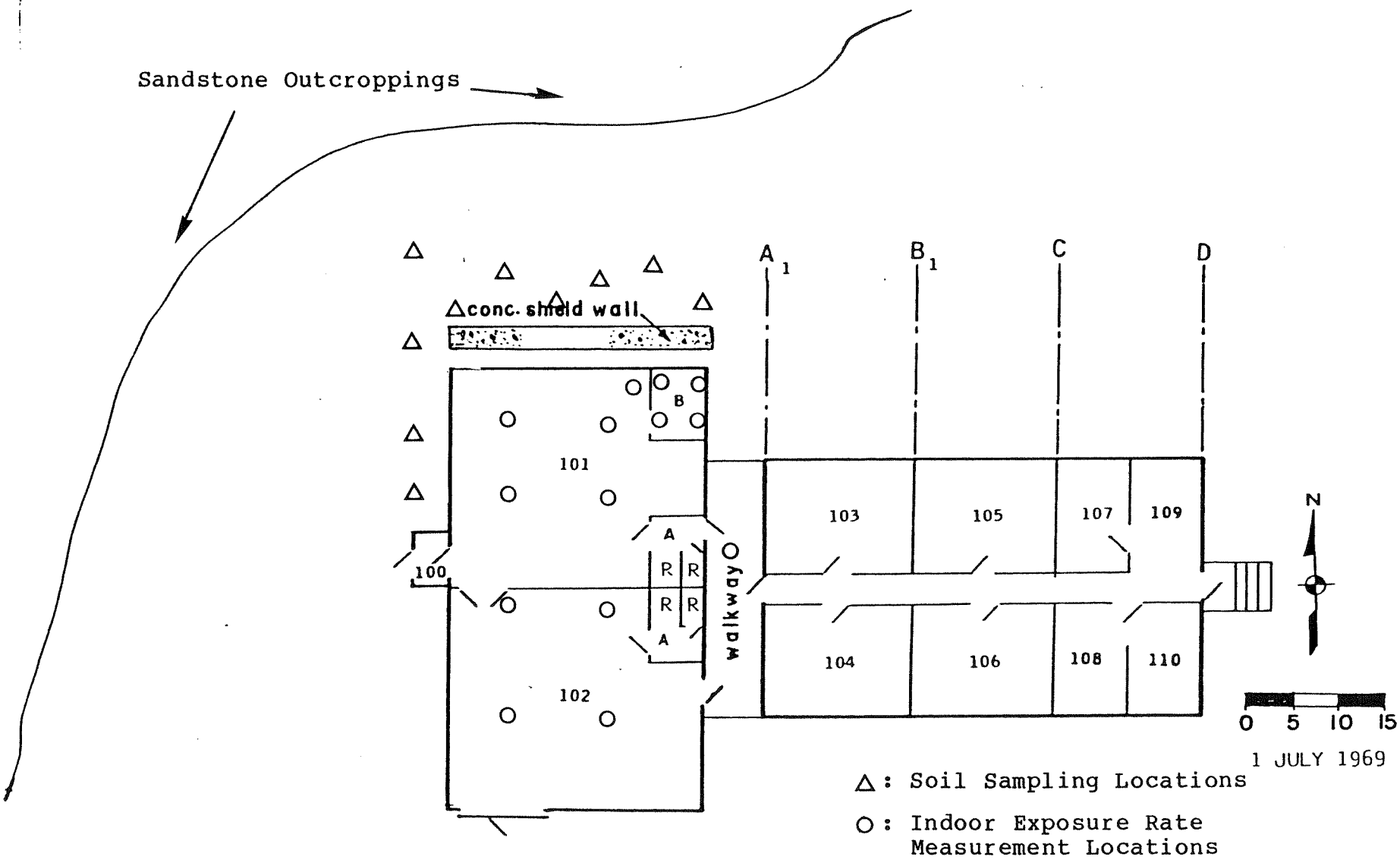


Figure 3.2 Soil Sampling Locations for Tritium Analysis

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 for equation 4-3 to apply. 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 probability density function, g(x), is given by:

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

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

m = average, or mean of the population distribution

σ = standard deviation of the population distribution.

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

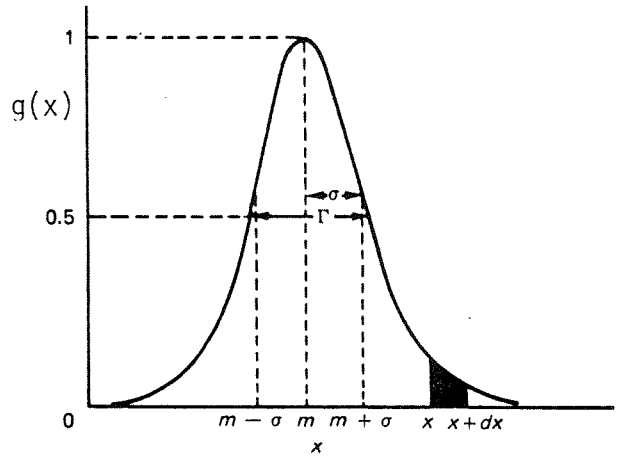


Figure 4.1 The Gaussian Probability Density Function

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

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

$$= P(x < X)$$

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

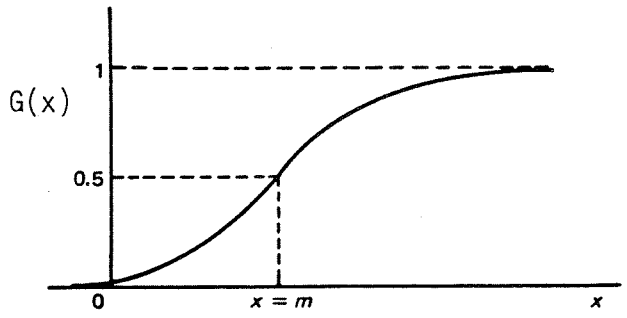


Figure 4.2 The Gaussian Cumulative Distribution Function

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

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

4.2 Sampling Inspection

4.2.1 By Variables

Acceptance inspection by variables is a method of judging whether a lot of items is of acceptable quality by examining a sample from the lot, or population. In the case of determining the extent of contamination in an area, it would be unacceptably time consuming and not cost effective to measure 100% of the population. However, by applying sampling inspection by variables methods, the accuracy of the conclusion made about the level of contamination is not sacrificed because of a decrease in number of sampling locations. We estimate the level of contamination in an area by making at least 30 measurements. This allows us to approximate a Gaussian distribution through the Central Limit Theorem. The entire area must have similar radiological characteristics and physical attributes. In acceptance inspection by variables, the result is recorded numerically and is not treated as a Boolean statistic, so fewer areas need to be inspected for a given degree of accuracy in judging a lot's acceptability.

4.2.2 By Attributes

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

4.3 Sampling Inspection by Variables

4.3.1 Calculated Statistics of the Gaussian Distribution

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

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

The sample mean is given by:

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

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

The standard deviation, s is given by:

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

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.

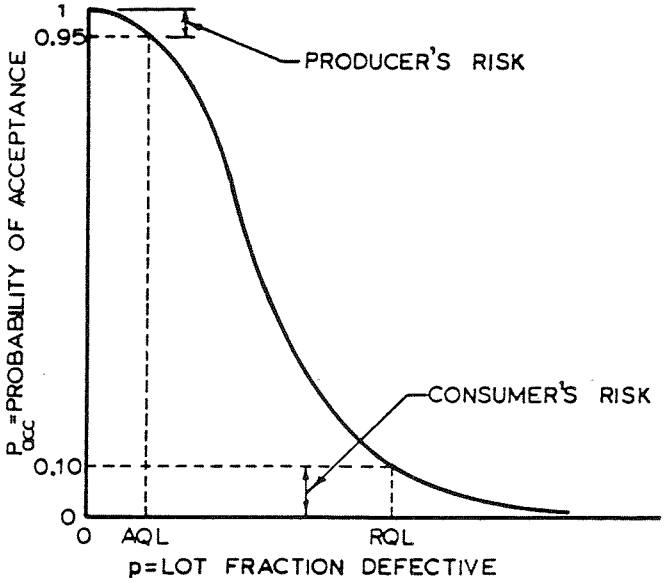


Figure 4.3 Operating Characteristics Curve

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

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

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, $K_\beta = K_2 = 1.282$.

The coefficients $K\beta$ and K_2 are equal because of the choice for the values of both β and LTPD as 0.10. Refer to statistics handbooks listed in the reference section for additional description of this sampling principle. The 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.

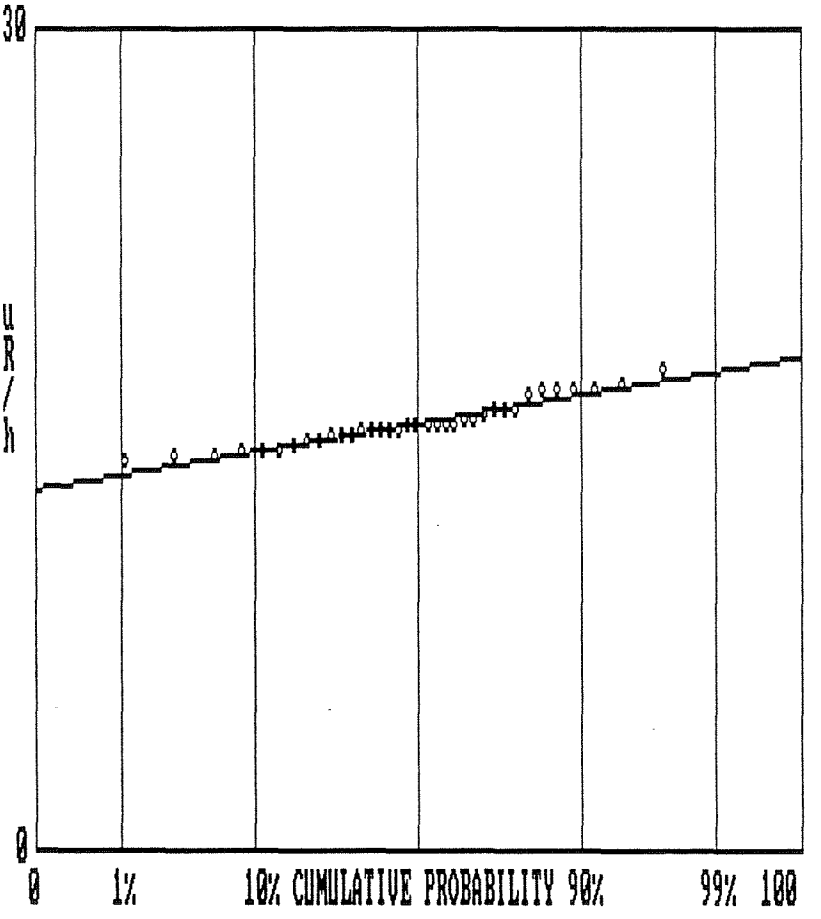


Figure 4.4 Gaussian cdf Plotted on Probability-Grade Paper

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. These proactive action levels were presented in section 3.1.)

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

Statistical methods presented in Section 4.0 were used to judge whether the sampling area is not contaminated, slightly contaminated, contaminated above acceptance limits, or whether additional investigation is required. For this particular survey, that judgement is based on one type of radiological measurement: gamma exposure rate. This measurement is sensitive to radiations emitted from mixed fission products and activation products, and is suitable for achieving the objectives specified in the Site Survey Plan, (Reference 4). If these gamma measurements were to show elevated areas of contamination (section 3.1), follow-up investigation commenced. During this survey, further investigation was not required. Beta radioactivity measurements were made inside T030's old accelerator room and in an area just south and west of T030 "for indication." Soil samples were collected for tritium analysis.

Analytical techniques used to acquire, evaluate, and interpret these radiological measurements are presented in detail in this section. These techniques include instrument calibration, background radiation determinations, and computerized data analysis through inspection by variables.

5.1 Data Acquisition

In each designated 6-m square grid, ambient gamma exposure rate was measured. Areas not easily accessible because of rock outcroppings and other objects were surveyed as best as possible. Wall coving and I-beams in T030 were surveyed for beta radioactivity. Results were reported "for indication." Ten 1-liter soil samples were collected in designated locations near T030.

5.2 Data Reduction Software Program

Each gamma exposure rate measurement value was input into SMART SPREADSHEET. This is an off-the-shelf computer software package which allows multiple computations to be performed on raw data values. Columns

were established to calculate the surface ambient gamma exposure rate in $\mu R/h$. The standard deviation of each measurement was also calculated. Software was developed in a program language called Quick Basic by Microsoft to read data from a SMART file into a graphics program which plots radiological measurements against a Gaussian cdf. For convenience, the distribution function, G(x) is plotted as the abscissa (probability grades), and x, the measurement value, is plotted as the ordinate (linear grades), see Figure 4.4.

Input for data reduction of these measurements was:

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

2. Gamma survey instrument background (1 min.), and efficiency factor ($\mu R/h/cpm$).

Output for Gaussian plots of these measurements:

1. Ambient gamma exposure rate and standard deviation ($\mu R/h$).

5.3 Data Analysis

An arithmetic mean and standard deviation of the radiological measurement values is calculated for each data set. The test statistic, $\bar{x} + ks$, based on a consumer's risk of acceptance of 0.10 at 10% defective, is calculated for each distribution being tested against acceptance limits. The acceptance criteria presented in Section 4.3.3 is applied to each sampling distribution as appropriate.

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 Gaussian "fitted" 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 $\overline{x}$ + ks. The figures display the results on an expanded scale so that the variations in the data can be seen in detail. When an acceptance limit is applied to a test case, horizontal lines are displayed on the graph at 0 (zero), 50% of the acceptance limit (Reinspection), 90% of the acceptance limit (Investigation), and at the acceptance limit.

5.4 Direct Beta Contamination Measurements

Direct beta contamination measurements were made "for indication" on an as-needed basis. Primarily, on the wall coving and I-beams of the accelerator room at T030, and on the pavement just west and south of T030. Pavement cracks and fissures were investigated for residual beta radioactivity. These measurements were made with Ludlum model 44-9 beta probes coupled to Ludlum model 12 count-ratemeters.

5.4.1 Instrument Calibration

Each beta detector was calibrated before use with Tc-99. Background levels were determined in an area of similar characteristics, known to be uncontaminated.

5.4.2 Data Acquisition and Reduction

Radioactivity measurements made "for indication" are reported as No Detectable Activity (NDA), or less than 20, 30, 40 cpm above background, etc. A positive indication of beta contamination is a steady count rate above background (50-75 cpm). In cases where the count rate does not change from background, NDA is reported.

5.4.3 Data Analysis

Data analysis is not applicable to measurements made "for indication."

5.5 Removable Contamination Measurements

Removable contamination measurements were not required and were not performed.

5.6 Ambient Gamma Exposure Rate

Measurements of ambient gamma exposure rate were made by use of a 1 in x 1 in NaI scintillation crystal coupled to a Ludlum Model 2220-ESG portable scaler, (Appendix A). 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 show variations in exposure rate down to one-half of a μR/h, using the digital scaler for a 1-min count time.

5.6.1 Instrument Calibration

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

Because of an exposure rate-dependent effect and because our calibration range is not set up to read less than 200 μR/h, this instrument was cross-calibrated against a Reuter Stokes High Pressure Ion Chamber (HPIC) at ambient "background" levels. 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, on different days during performance of the site survey.

Instrument response was checked three times a day using a Ra-226 source. The source was placed 1 ft from the detector and counted for 1 min.

If the scaler reading fell within ± 5% of the nominal value, then the instrument was qualified as operable for the day, under the calibration conditions previously described. Recalibration was never necessary.

5.6.2 Data Acquisition and Reduction

Each location where a gamma measurement was made was identified on a map and in matrix notation. The gross number of counts recorded in 1 min. along with the matrix notation location was input into SMART SPREADSHEET. Columns were established to calculate the total exposure rate (μR/h) and its standard deviation according to 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. (counts)
EF = efficiency factor (0.0047 μR/h/cpm) based on cross calibration with HPIC.

The standard deviation, s, of a single radiation measurement not corrected for background then becomes by Eq. 4-3:

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

5.6.3 Data Analysis

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

Both the NRC and DOE criteria for acceptance of an area for 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 a change in source geometry from a planar 2 π-steradian to a rocky, 3 π-steradian surface.

The best solution for evaluating the potential or existence of residual contamination in an area where the radiation field varies naturally by a range as large as the acceptance limit, is to compare total-gross exposure rates in different areas.

The T030 and T641 distribution of ambient exposure rate measurements is compared against three independent sampling areas of similar geologic characteristics. In these other areas, no radioactive materials were ever used, handled, stored, or disposed. These distributions represent "natural ambient gamma radiation" levels in this SSFL location. Measurements were taken on flat and rugged terrain, with Chico Formation sandstone, similar to conditions surrounding these facilities.

These distributions make no corrections for "background"; the total-gross gamma exposure rate is considered. Then in order to compare the T030 and T641 data against acceptance limits "above background," the average of the "true background" areas is used as our best estimate for background radiation at SSFL. Using this value, we correct the T030 and T641 data and compare the resulting "background corrected" distribution against the acceptance limit of 5 μR/h above background.

5.7 Surface Soil Samples

As described in the Site Survey Plan, (Reference 4), surface soil sampling was required near T030. 10 1-liter surface soil samples (no deeper than 6 in) were taken and placed in 1-liter plastic bottles. The location from where each sample was taken was numbered and marked on a map. Samples

were sent to U.S. Testing Company of Richland, WA for tritium (H^3) analysis. The sensitivity of these tests was 500 pCi/l. Since these samples and analyses were performed "for indication," data analysis is not applicable; the results reported by U.S. Testing are presented in tabular format.

6.0 PROCEDURES

The following radiological procedures were used in performing this survey.

6.1 Sample Selection Gridding

Superimpose 6-meter square grids on each surface to be radiologically characterized. Designate each square meter in matrix notation with location (1,1) being the northwestern most square in a sample lot. For T030 and T641, (1,1) should be just north of T030, near Building T003. From this northwestern-most location, mark a location off every 6 meters east to the T513 parking lot, and south down 10th Street to "G" Street. Where it is not convenient to make a measurement because of rock outcroppings, step to the nearest clear area.

6.2 Calibration and Instrument Checks

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

Portable Ludlum 2220-ESG Survey Instruments:

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

2) Check high voltage (800V gamma).

3) Check threshold (400 gamma).

4) Set window in/out switch to "out."

5) Check battery (greater than 500).

6) Set range selector to 1, response to fast, and count time for ambient gamma exposure rate measurements to 1 min.

7) Take and record a 1 min. background count in an uncontaminated area which typifies the area to be surveyed. Verify that ambient background falls within ± 20% of daily-averaged background measurements.

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

6.3 Radiological Measurements

6.3.1 Ambient Gamma Exposure Rate Measurements

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

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

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

4) Enter the data into SMART spreadsheet.

6.3.2 Surveys of Special Structural Features and Components

1) Using a Ludlum Model 12 count rate meter in connection with a Ludlum Model 44-9 pancake GM beta probe, survey various

building features as appropriate The only areas requiring this type of investigation are: wall coving and I-beams in the accelerator room; and the pavement where the old pallet storage area was located.

2) Perform an instrument calibration check using a Tc-99 source.

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

7.0 SURVEY RESULTS

The radiological survey of buildings T030 and T641 was performed using the survey plan previously described. A single sample lot was established to survey, analyze, and interpret radiological data. Uniform 6-m square grids were set up inside T030 and around both facilities to measure ambient gamma exposure rates. Radiological data for this lot was statistically analyzed. Analytical interpretation of this data set shows that both facilities and grounds are uncontaminated. Tritium analysis of 10 soil samples, and beta surveys of the accelerator room and outdoor storage area show no detectable activity.

7.1 Statistical Results Format

Gamma exposure rate data collected during this survey are displayed as a Gaussian cumulative distribution function in Figure 7.1. Figures 7.2 through 7.4 are distributions of gamma exposure rate measurements made at 3 independent SSFL locations to demonstrate the variability of "natural" background. Figure 7.5 is a distribution of the same T030/T641 data corrected for "natural" background based on the average of the results presented in Figures 7.2 through 7.4. 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.

The mean of each distribution is approximately that value on the ordinate which corresponds to a 50% cumulative probability on the abscissa. One, two, and three standard deviations above the mean corresponds to 84%, 97.7%, and 99.8% cumulative probability for a one-sided test, respectively. The value of k used in the inspection test is very nearly 1.5 for each case; thus, the Test Statistic (TS) line ($\overline{x}$ + ks) 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 uncontaminated. "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. 30 μR/h is normally used for convenience, as the maximum ordinate value. If measurements exceed 30 μR/h, then the greatest measurement value is the upper bound of the ordinate axis. In cases where the measurements have been corrected for "background," 5 μR/h (the NRC acceptance limit) is used as the maximum ordinate value. The lower bound of the ordinate is either the smallest measured value (minus background, if applicable) or the smallest value calculated for a Gaussian fit. Negative numbers result when the measured value is less than background. Cumulative probability (abscissa) is plotted in probability grades, i.e. the distance between any two successive points increases as the distance from the 50% cumulative probability line increases. If an acceptance limit is applicable, four horizontal lines extending across each plot show, from top to bottom, 100% of the test limit, 90% of the test limit (Investigation), 50% of the test limit (Reinspection), and zero.

In cases where an acceptance limit is not appropriate, for example, gamma exposure rate measurements not corrected for background, the four horizontal lines are not shown. Furthermore, a test statistic is not calculated because we were not testing the data against an acceptance limit. Since the variability in naturally-occurring ambient gamma exposure rates at SSFL is wide, background was not subtracted at first. In these cases, the mean is calculated and the shape of the distribution is observed to identify any areas of increased radioactivity. Then the shape of the curve is compared against three "background" distributions. Finally, "background" is subtracted and inspection by variables techniques are applied to prove or disprove the hypothesis that the area is not contaminated. For soil sample tritium-analysis results, probability plots are not applicable. These results are displayed in tabular format, Table 7-2.

7.2 Ambient Gamma Exposure Rates

One hundred fourteen ambient gamma exposure rate measurements were made in and near both facilities. Fourteen measurements were made inside the accelerator room at T030. Appendix C shows the data set. Notice that the smallest exposure rates were measured inside. Table 7.1 shows the statistics for the T030/T641 data set compared against data from three independent areas where no radioactive material was ever handled, used, or stored. This type of comparison is necessary for two reasons: 1) to demonstrate the variability of "background" gamma-radiation at SSFL; and 2) to estimate "true" background at SSFL because the limits for unrestricted use by which we use to demonstrate an "acceptable" area are based on above "background" criteria. So, unless we confidently know what "background" is, the area under study may be found incorrectly acceptable if the background used was too high, or incorrectly unacceptable if the background used was too low. Because of the difference observed between indoor and outdoor measurements, these indoor/outdoor statistics were also calculated independently. They appear as the second and third rows in Table 7.1. It would have been more appropriate to compare the indoor measurements against a "clean" facility, but a facility of similar construction was not available.

Figure 7.1 shows the statistical distribution of gamma measurements plotted against cumulative probability. The distribution of points deviates from a model Gaussian distribution and shows the difference between indoor and outdoor measurements. The smallest 14 points correspond to indoor measurements. The slight deviations observed throughout the distribution (more pronounced in Figure 7.5) are attributed to changes in topography and the presence of Chico sandstone formation, which we have observed to produce a greater "natural" exposure rate.

Figures 7.2 through 7.4 are probability plots of the three independent "background" areas surveyed. All three areas are located on the eastern side of SSFL: (1) Area surrounding building 309 on Area I Road; (2) well #13 Road; and (3) Incinerator Road. At least 30 measurements were made

in each area on the same day. In the plots, a uniform background rate would appear as a straight line with a minimal slope. That slope would be fit such that 1 standard deviation from the mean of values would be equivalent to the mean-value standard deviation. All three distributions show Gaussian functions; however, the variability is greatest in the area near building 309 (slope is greatest).

Table 7.1 Ambient Gamma Radiation at SSFL Compared to T030/T641

Location	No. of Measurements	Average Exposure Rate (μR/h)	Standard Deviation μR/h	Range μR/h
T030/T641 Entire Data Set	114	12.7	1.81	8.56
T030 Indoor Data Set	14	9.4	0.69	2.54
T030/T461 Outdoor Data Set	100	13.2	1.40	6.76
Bldg. 309 Area (1/19/88)	36	15.6	0.82	3.4
Well #13 Road (Dirt) (4/29/88)	43	16.2	0.49	2.2
Incinerator Road (Dirt) (4/29/88)	35	14.0	0.36	1.4

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

Figure 7.1 Ambient Gamma Radiation at Buildings T030/T641 and Surrounding Area (Entire Data Set)

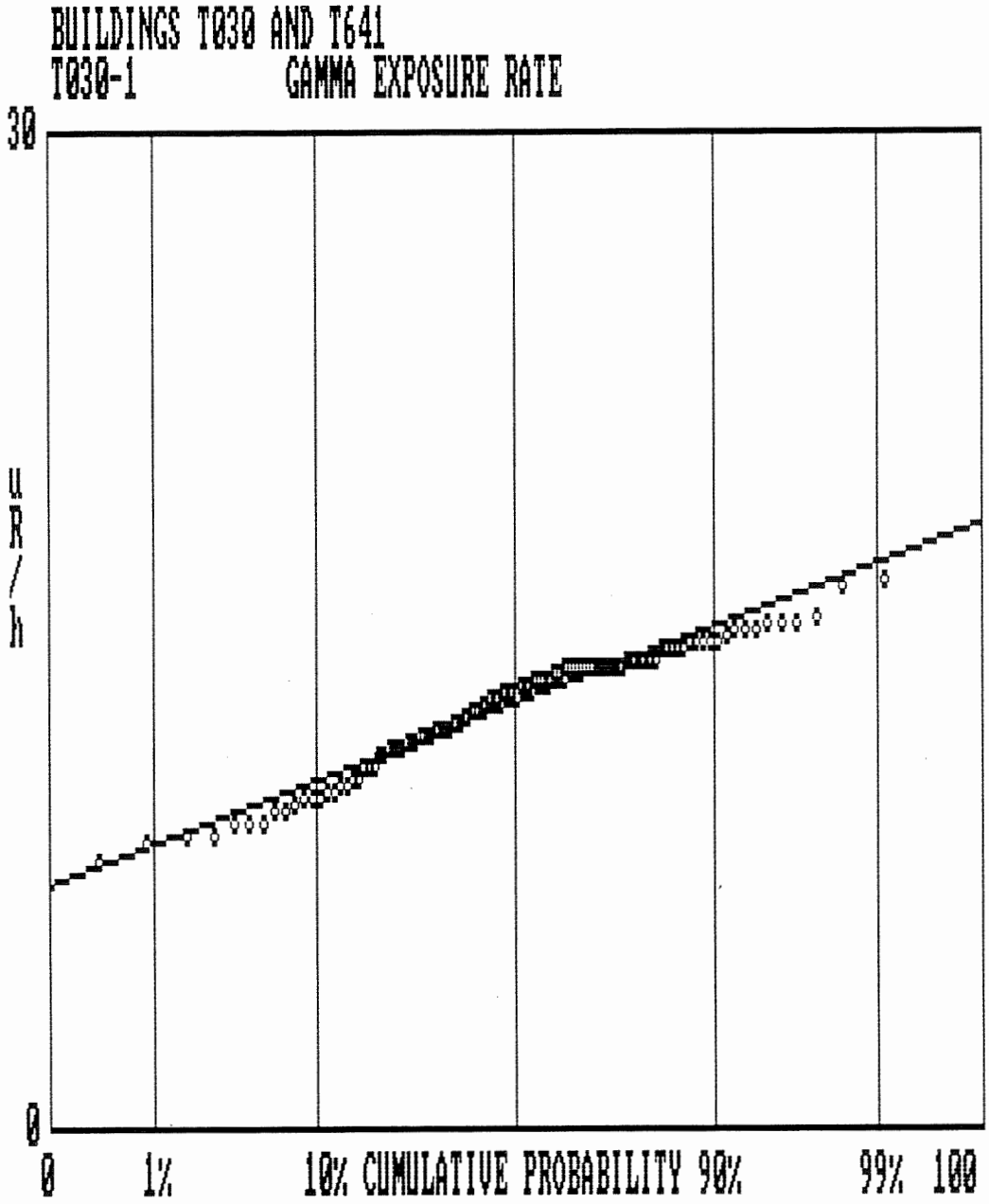


Figure 7.2 Ambient Gamma Radiation at Area Surrounding Building 309
(Background Distribution)

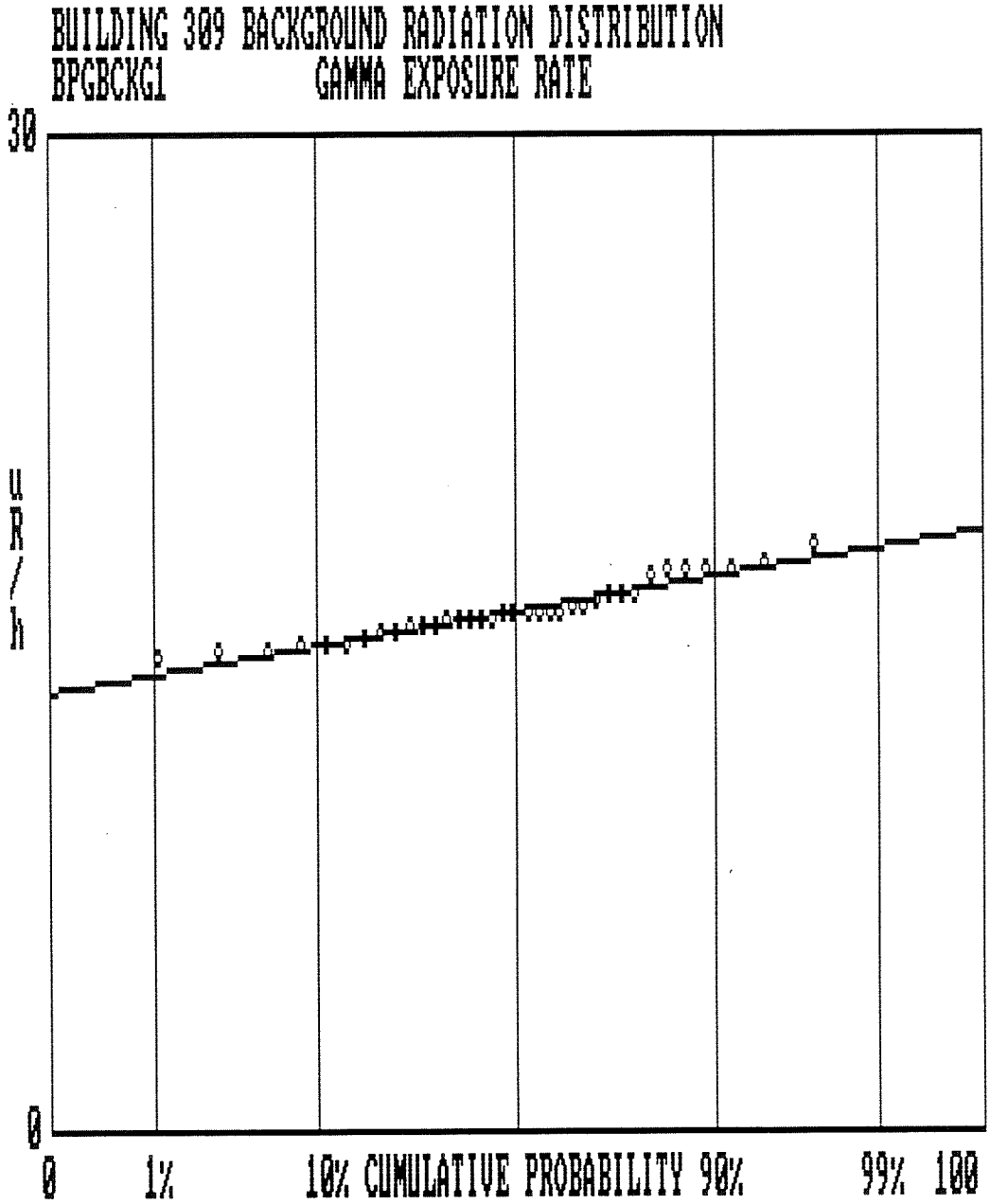


Figure 7.3 Ambient Gamma Radiation at Area Well #13 Road
(Background Distribution)

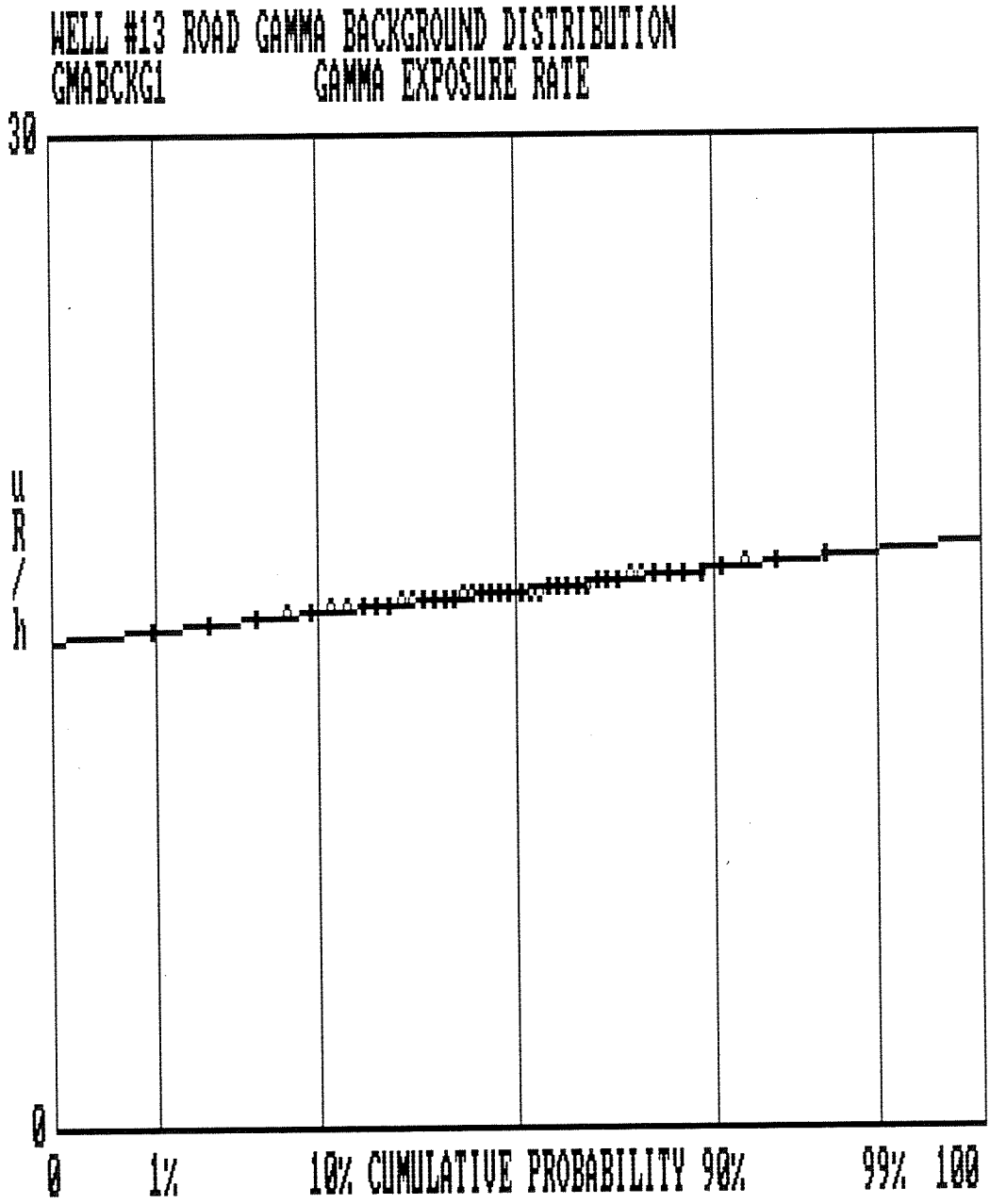
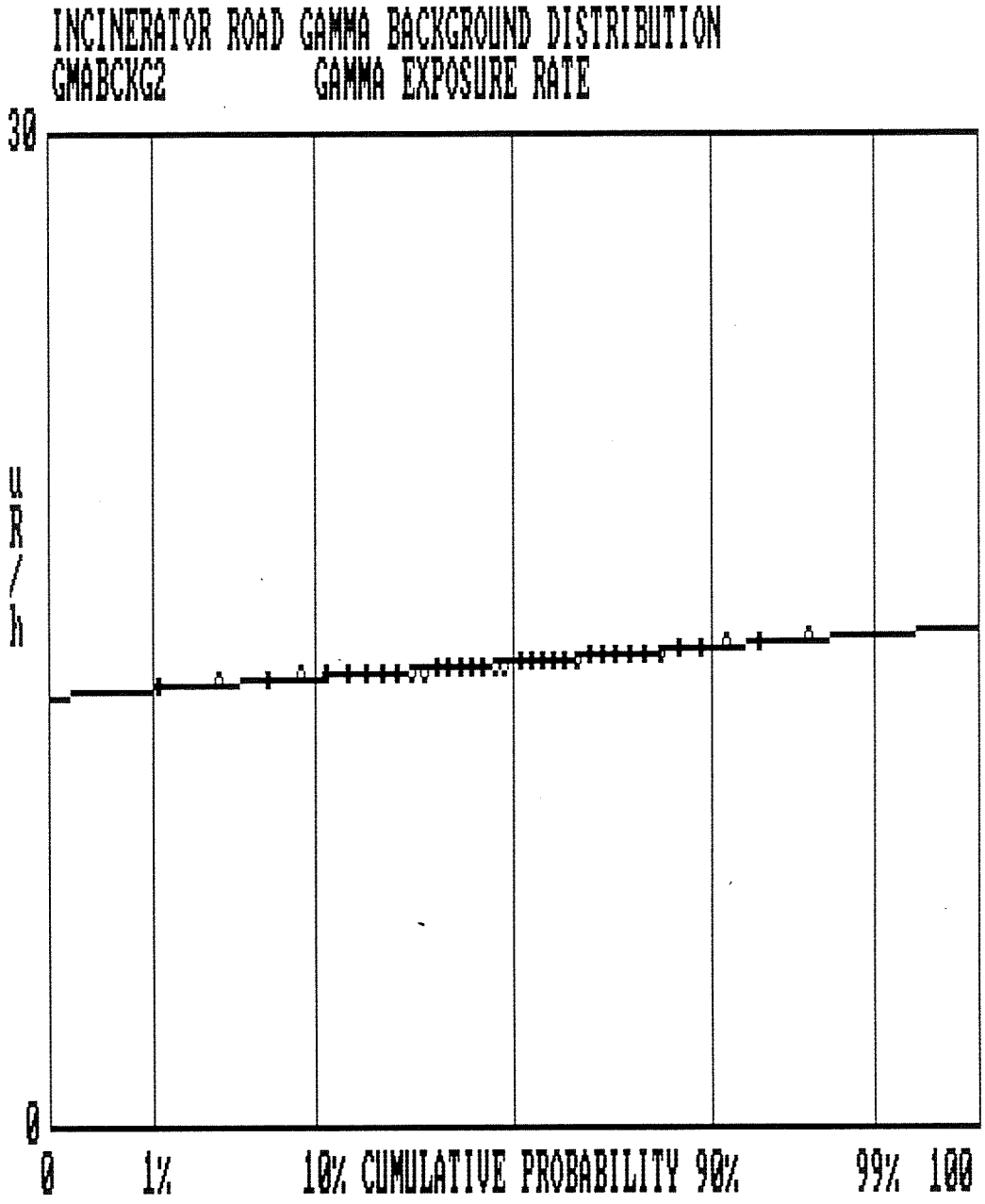


Figure 7.4 Ambient Gamma Radiation at Incinerator Road
(Background Distribution)



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. Deviations observed in the T030/T641 data are attributed to naturally-occurring causes; the building shields against ambient radiation, and sandstone produces a greater exposure rate. If the data points are corrected for background based on an average of the three "background" areas, and plotted against cumulative probability with the maximum ordinate value equal to the NRC acceptance limit (5 μR/h), Figure 7.5 is produced. Again, the indoor measurements affect the Gaussian fit. Deviations in the plotted points are enlarged in this "corrected for background" distribution because of variations observed in "natural" background. All data points were treated for background equally; "background" was treated uniformly. Fortunately, the test statistic accounts for this observed dispersion. The average value of this "background" corrected distribution is -0.84 ± 1.55 μR/h. Ambient background measurements are significantly greater than the measurements made near T030 and T641. The Test Statistic ($x + ks$) is 0.084, which is less than our 50% characterization level. The area is found acceptably "clean" by this inspection measurement.

7.3 Tritium Activity in Soil Samples

As expected, no statistically significant tritium activity was present in the ten soil samples analyzed by U.S. Testing. All results, presented below in Table 7.2, are less than the overall error, specified as 2σ. As statistically expected, the standard deviation of the 10 values is very close to being equal to each individual measurement error, about 370 pCi/l. Figure 3.2 shows the locations where samples were collected. A representative sampling of the area was achieved. Residual tritium is non-existent.

Table 7.2 Results of Tritium Analysis in Soil Samples Collected Near T030*

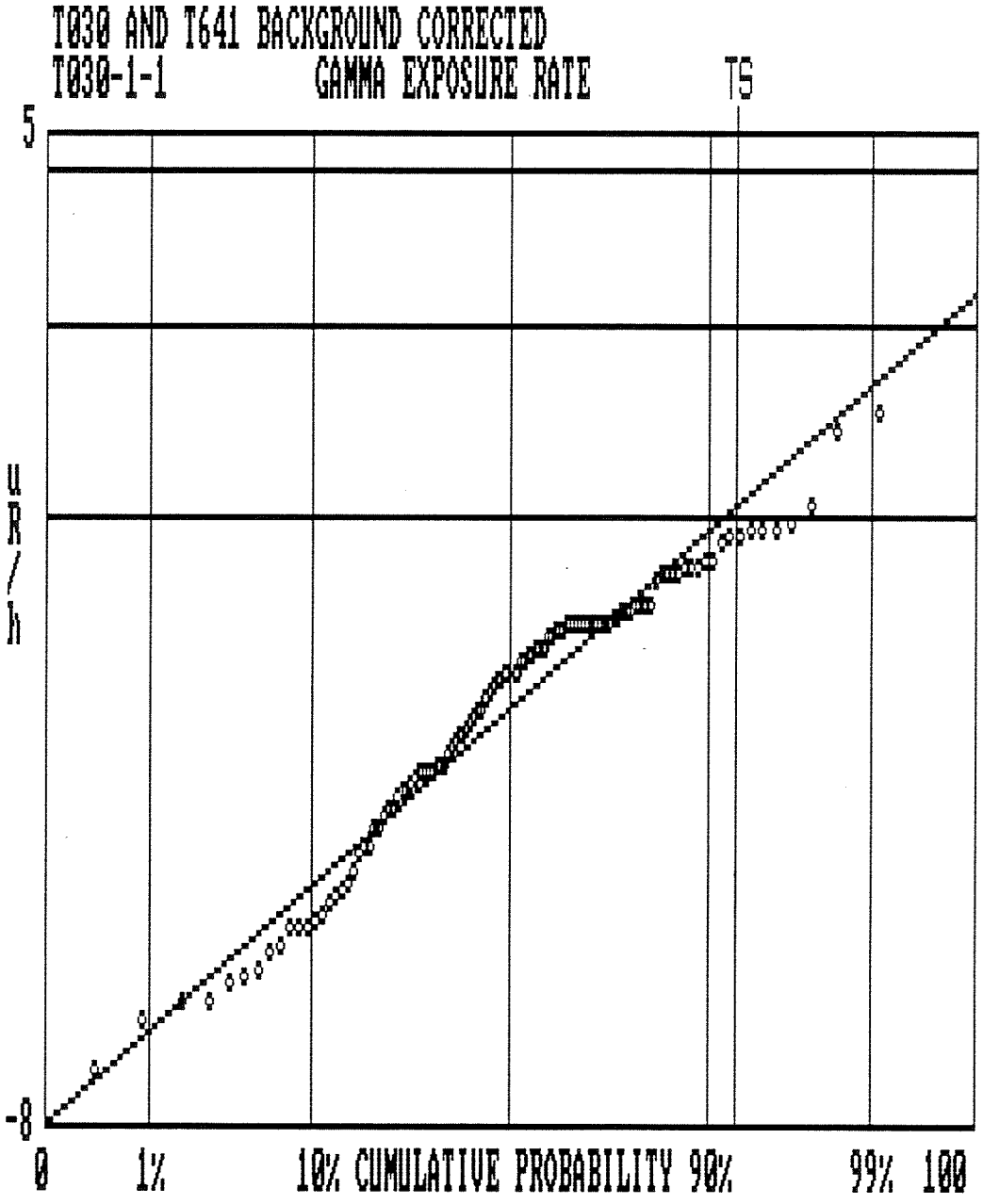
Sample #	Weight (g)	Tritium Activity Concentration (pCi/l)**	Counting Error (pCi/l)	Overall Error (pCi/l)
1	1120.90	-155	189	364
2	1003.62	-148	191	365
3	1047.18	- 50.8	192	369
4	1156.11	- 57.3	192	368
5	1152.04	27.0	195	373
6	1232.03	121	199	378
7	1176.53	366	208	391
8	1198.06	17.3	195	372
9	1201.90	- 72.5	192	368
10	1263.48	5.41	195	372
	Average	5.31	N/A	303

*As reported by United States Testing Company, Inc. of Richland, WA (Reference 21)
** All results are less than the overall error, specified as 2σ.

7.4 Beta Radiation Measurement

Beta radiation measurements made "for indication" show no detectable activity. The wall-coving and I-beams in rooms 101 and 102 of T030 (Old Accelerator Room) were thoroughly surveyed. The outside pavement area used for storing palletized-containers (shown in Figure 3.2) was also surveyed, particularly in cracks, ridges, and fissures.

Figure 7.5 Ambient Gamma Radiation at Buildings T030/T641
(Corrected for Background)



8.0 CONCLUSIONS

Buildings T030 and T641, and the surrounding area were inspected for radioactive contaminants. Gamma exposure rate measurements show that no residual radioactive contamination exists in T030's accelerator room; T030's palletized container storage area; T641's shipping dock; or in the nearby area. Gaussian probability plots of these data and of "background" areas show the great difficulty in assessing the radiological condition of a facility based on a government acceptance requirement above "background" conditions. Variability of gamma exposure rates is quite large and depends on whether the measurement was made indoors, outdoors, or near a large sandstone outcropping. Accounting for these variations and deviations, and subtracting a value that we believe represents "natural" background gamma radiation at SSFL, we conclude through inspection by variables, that the area is clean of any residual radioactive contamination, with a consumer's risk of acceptance of 0.1 at an LTPD of 10%.

Ten surface soil samples collected randomly in locations near T030 all show tritium (H^3) concentrations less than the overall error reported by the analytical laboratory. No statistically significant tritium activity was found.

Further radiological investigation of the T030 accelerator room and palletized-container storage area using a beta probe shows in all cases, No Detectable Activity. Within the limits prescribed by the Site Survey Plan, this area is clean of radioactive contaminants. Further radiological investigation and remedial action is not required.

9.0 REFERENCES

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

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

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

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

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

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

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

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

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

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

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

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

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

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

15. "Lower Limit of Detection and Statistically Significant Activity for Radiologic Measurements," IL from R. J. Tuttle to Radiation and Nuclear Safety, RI, June 24, 1986.

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

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

18. "Radiological Survey of the Source and Special Nuclear Material Storage Vault - Building T064", J. A. Chapman, Rocketdyne/Rockwell International, August, 1988.

19. "Radiological Survey of the old Calibration Facility - Building T029", J. A. Chapman, Rocketdyne/Rockwell International, August, 1988

20. IL from J. A. Chapman to Those Listed, "Radiological Survey Plan for Building T030", Rocketdyne/Rockwell International, March 30, 1988

21. Letter from R. G. Swoboda of U.S. Testing Company, Inc. to J. D. Moore of Rocketdyne, "Results of Ten Soil Samples Analyzed for Tritium, P.O. #R84-PJZ-88410488", June 21, 1988

22. IL from A. R. Mooers to W. F. Heine, "Tritium Smear Survey, Building 030 Van deGraaf Accelerator", Atomics International, March 29, 1966

APPENDIX A. DESCRIPTION OF NUCLEAR INSTRUMENTATION

During the radiological survey, direct radiation measurements were made by using portable instruments. Because sample collection was not necessary, analytical laboratory equipment was not required.

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

A Ludlum model 12 count-ratemeter was coupled to a Ludlum model 44-9 pancake G-M beta probe to measure beta contamination. The probe active area is 20 cm^2. Instrument calibration is performed using Tc-99. This instrument is best suited "for indication" determinations.

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

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

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

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

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

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

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

B. BASIC DOSE LIMITS

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

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.

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

I. REFERENCES

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

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

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

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

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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. BUILDINGS T030 and T641 RADIOLOGICAL SURVEY DATA

Sorted by Location

T030-1.WS T030/T641 DATA SORTED BY LOCATION

ROOM NUMBER	GRID NAME	uR/h TOTAL	uR/h STD DEV
	1-4	16.40	0.28
	1-16	14.10	0.26
	1-19	14.50	0.26
	2-12	11.17	0.23
	2-17	16.58	0.28
	3-9	15.08	0.26
	3-18	14.05	0.26
	4-9	15.10	0.26
	4-16	13.38	0.25
	4-20	13.80	0.25
	5-19	13.81	0.25
	6-9	14.56	0.26
	6-10	13.81	0.25
	6-14	14.52	0.26
	6-16	13.22	0.25
	6-20	13.89	0.25
	7-5	14.51	0.26
	7-14	13.75	0.25
	7-16	12.63	0.24
	7-18	13.85	0.25
	7-19	13.21	0.25
	8-6	14.55	0.26
	8-17	13.84	0.25
	9-7	15.14	0.26
	9-10	9.86	0.21
	9-11	10.35	0.22
	9-12	11.44	0.23
	9-16	11.37	0.23
	9-17	13.80	0.25
	10-10	10.90	0.22
	10-12	10.87	0.22
	10-14	11.35	0.23
	11-6	14.94	0.26
	11-9	12.90	0.24
	11-10	10.90	0.22
	11-11	13.09	0.25
	11-16	12.27	0.24
	11-17	13.80	0.25
	12-5	15.10	0.26
	12-9	13.09	0.25
	12-14	11.99	0.24
	13-4	14.58	0.26
	13-8	14.01	0.25
	13-11	11.66	0.23
	13-12	11.53	0.23
	13-16	13.97	0.25
	14-9	11.69	0.23
	15-6	14.63	0.26
	15-8	13.97	0.25
	15-11	12.41	0.24
	15-12	12.20	0.24
	15-13	11.87	0.23
	15-16	13.82	0.25
	16-6	14.69	0.26
	16-17	13.85	0.25

T030-1.WS T030/T641 DATA SORTED BY LOCATION

ROOM NUMBER	GRID NAME	uR/h TOTAL	uR/h STD DEV
	17-4	14.87	0.26
	17-5	13.49	0.25
	17-13	11.89	0.23
	17-14	11.95	0.24
	18-3	15.41	0.27
	18-5	14.53	0.26
	18-	12.93	0.24
	18-17	13.80	0.25
	19-7	11.90	0.23
	19-8	11.70	0.23
	19-9	11.91	0.23
	19-10	11.99	0.24
	19-11	11.91	0.23
	19-12	12.16	0.24
	19-13	11.81	0.23
	19-14	12.69	0.24
	19-15	12.88	0.24
	19-16	13.68	0.25
	19-18	13.53	0.25
	20-6	13.84	0.25
	21-7	13.85	0.25
	21-8	10.15	0.22
	21-9	13.02	0.25
	21-10	12.49	0.24
	21-11	12.52	0.24
	21-12	12.69	0.24
	22-17	13.84	0.25
	22-19	13.85	0.25
	23-17	14.05	0.26
	23-18	13.53	0.25
	24-19	13.91	0.25
	25-16	13.36	0.25
	26-18	14.41	0.26
	26-20	13.43	0.25
	27-19	13.21	0.25
	28-18	13.54	0.25
	28-19	13.47	0.25
	28-20	13.22	0.25
	28-21	13.70	0.25
OUT-WALL	1-1	15.01	0.26
OUT-WALL	1-2	14.10	0.26
OUT-WALL	2-1	11.18	0.23
OUT-WALL	2-2	10.47	0.22
OUT-WALL	2-3	12.40	0.24
OUT-WALL	2-4	9.82	0.21
IN-T030	3-1	8.90	0.20
IN-T030	3-2	9.57	0.21
IN-T030	3-3	8.02	0.19
IN-T030	3-4	10.26	0.22
IN-T030	3-5	10.56	0.22
IN-T030	4-1	9.94	0.21
IN-T030	4-2	9.59	0.21
IN-T030	4-3	8.68	0.20
IN-T030	4-4	8.86	0.20
IN-T030	5-1	9.28	0.21

T030-1.WS T030/T641 DATA SORTED BY LOCATION

ROOM NUMBER	GRID NAME	uR/h TOTAL	uR/h STD DEV
IN-T030	5-2	9.87	0.21
IN-T030	5-3	9.13	0.21
IN-T030	6-1	10.04	0.22
IN-T030	6-2	9.25	0.21

Sorted by Exposure Rate

T030-1.WS T030/T641 DATA SORTED BY EXPOSURE RATE

ROOM NUMBER	GRID NAME	uR/h TOTAL	uR/h STD DEV
	2-17	16.58	0.28
	1-4	16.40	0.28
	18-3	15.41	0.27
	9-7	15.14	0.26
	12-5	15.10	0.26
	4-9	15.10	0.26
	3-9	15.08	0.26
OUT-WALL	1-1	15.01	0.26
	11-6	14.94	0.26
	17-4	14.87	0.26
	16-6	14.69	0.26
	15-6	14.63	0.26
	13-4	14.58	0.26
	6-9	14.56	0.26
	8-6	14.55	0.26
	18-5	14.53	0.26
	6-14	14.52	0.26
	7-5	14.51	0.26
	1-19	14.50	0.26
	26-18	14.41	0.26
OUT-WALL	1-2	14.10	0.26
	1-16	14.10	0.26
	23-17	14.05	0.26
	3-18	14.05	0.26
	13-8	14.01	0.25
	15-8	13.97	0.25
	13-16	13.97	0.25
	24-19	13.91	0.25
	6-20	13.89	0.25
	16-17	13.85	0.25
	7-18	13.85	0.25
	21-7	13.85	0.25
	22-19	13.85	0.25
	20-6	13.84	0.25
	22-17	13.84	0.25
	8-17	13.84	0.25
	15-16	13.82	0.25
	6-10	13.81	0.25
	5-19	13.81	0.25
	4-20	13.80	0.25
	18-17	13.80	0.25
	11-17	13.80	0.25
	9-17	13.80	0.25
	7-14	13.75	0.25
	28-21	13.70	0.25
	19-16	13.68	0.25
	28-18	13.54	0.25
	23-18	13.53	0.25
	19-18	13.53	0.25
	17-5	13.49	0.25
	28-19	13.47	0.25
	26-20	13.43	0.25
	4-16	13.38	0.25
	25-16	13.36	0.25
	6-16	13.22	0.25

T030-1.WS T030/T641 DATA SORTED BY EXPOSURE RATE

ROOM NUMBER	GRID NAME	uR/h TOTAL	uR/h STD DEV
	28-20	13.22	0.25
	7-19	13.21	0.25
	27-19	13.21	0.25
	11-11	13.09	0.25
	12-9	13.09	0.25
	21-9	13.02	0.25
	18-	12.93	0.24
	11-9	12.90	0.24
	19-15	12.88	0.24
	19-14	12.69	0.24
	21-12	12.69	0.24
	7-16	12.63	0.24
	21-11	12.52	0.24
	21-10	12.49	0.24
	15-11	12.41	0.24
OUT-WALL	2-3	12.40	0.24
	11-16	12.27	0.24
	15-12	12.20	0.24
	19-12	12.16	0.24
	12-14	11.99	0.24
	19-10	11.99	0.24
	17-14	11.95	0.24
	19-9	11.91	0.23
	19-11	11.91	0.23
	19-7	11.90	0.23
	17-13	11.89	0.23
	15-13	11.87	0.23
	19-13	11.81	0.23
	19-8	11.70	0.23
	14-9	11.69	0.23
	13-11	11.66	0.23
	13-12	11.53	0.23
	9-12	11.44	0.23
	9-16	11.37	0.23
	10-14	11.35	0.23
OUT-WALL	2-1	11.18	0.23
	2-12	11.17	0.23
	10-10	10.90	0.22
	11-10	10.90	0.22
	10-12	10.87	0.22
IN-T030	3-5	10.56	0.22
OUT-WALL	2-2	10.47	0.22
	9-11	10.35	0.22
IN-T030	3-4	10.26	0.22
	21-8	10.15	0.22
IN-T030	6-1	10.04	0.22
IN-T030	4-1	9.94	0.21
IN-T030	5-2	9.87	0.21
	9-10	9.86	0.21
OUT-WALL	2-4	9.82	0.21
IN-T030	4-2	9.59	0.21
IN-T030	3-2	9.57	0.21
IN-T030	5-1	9.28	0.21
IN-T030	6-2	9.25	0.21
IN-T030	5-3	9.13	0.21

T030-1.WS T030/T641 DATA SORTED BY EXPOSURE RATE

ROOM NUMBER	GRID NAME	uR/h TOTAL	uR/h STD DEV
IN-T030	3-1	8.90	0.20
IN-T030	4-4	8.86	0.20
IN-T030	4-3	8.68	0.20
IN-T030	3-3	8.02	0.19

APPENDIX D COPY OF INTERNAL LETTER

"Radiological Survey Plan for Building T030 - March 30, 1988"

Date: 3/30/88

To: Those Listed

File: rdsrvy19.ws

From: J. A. Chapman
641, 055-T100

5766

Subject: Radiological Survey Plan for Building T030

Building T030 housed a deuterium/tritium 14Mev neutron generator. Additionally, pallets contaminated with mixed fission products may have been stored there. Ambient gamma exposure rates have already been measured in the building and surrounding area. We need to analyze the soil for tritium and survey the pallet storage area for beta contamination. Refer to plot plan (attachment 1), for sampling locations.

Soil Analysis for Tritium

Collect 10 soil samples on the embankment west and north of T030, as indicated on the map. Dig no further than 1foot deep to collect the sample. Fill completely a one-liter poly-bottle (used for John's bioassay program) for each sample. Mark sample location on outside of bottle. Back fill the holes as best as possible. All 10 samples will be sent to US Testing for tritium analysis. See attachment 3 for tritium analysis sensitivity and cost.

Beta Outside-Surface Survey

Using a beta pancake probe coupled to a Ludlum count rate meter, survey the pallet storage area west of the building and the pavement surrounding the concrete shield on the north (see map). Gridding is not necessary, but try to survey at least 10% of the area. Particular areas to survey will be cracks, fissures, low spots, surface drainage areas, and building-to earth joints. Report as NDA (No Detectable Activity) or less than xx cpm, e.g. less than 200cpm (counts per minute).

Beta Inside Building Joint Survey

Using the same beta probe, survey the building grade beams and coving around the interior of rooms 101 and 102, including the small office in the northeast corner of room 101.

This effort should take at most 1 man day.

J. A. Chapman

J. A. Chapman
Radiation & Nuclear Safety

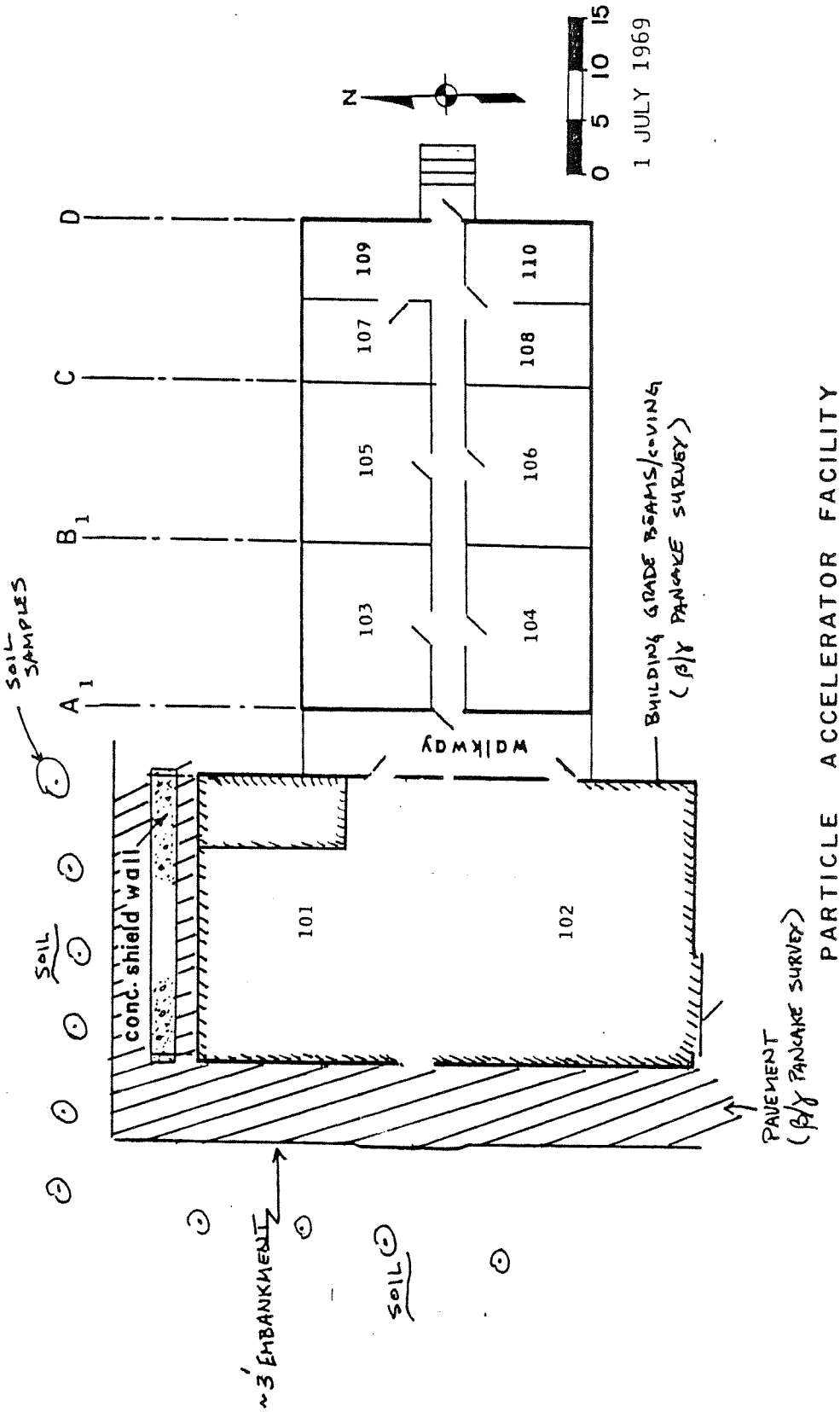
page 2

attachments: 1) T030 Plot-Survey Plan
2) SSFL Radiological Survey Plan-T030
3) Soil Sample Analysis Requirements and Cost Estimate

Those Listed:

K. L. Adler	T036
F. H. Badger	T020
W. A. McCollum	T065
J. D. Moore	T100
P. S. Olson	T487
R. C. Sheppard	T065
K. T. Stafford	T038
R. J. Tuttle	T100

ATTACHMENT 1 : TΦ3Φ PLOT. SURVEY PLAN



PARTICLE ACCELERATOR FACILITY

BUILDING No. 030

ATTACHMENT 2: From Site Survey Plan

5.4.3.2 T-030 and 641 - Shipping Receiving - Old Accelerator Area

Project Name: Van de Graaff accelerator

Isotopes: Tritium targets

Physical and Chemical Form of Radioisotopes:

Process Conducted: Van de Graaff generator activation experiments

Known Problems: None

Suspected Problem Areas: Tritium analysis poor in those days. MFP-contaminated pallets may have been stored here as well as leaking shipments.

1. Surface survey building and grounds, especially pallet storage area. MFP - Tritium north of T-030, in soil

2. Survey surface pathway to L-85. MFP

3. Survey surface north (excluding sandstone areas) to ETB (T003) and L-85 (T-093) down to "G" Street. MFP.

Attachment 3: Soil Sample Analysis Requirements and Cost Estimate

US Testing Inc. tests soil for tritium content. The sensitivity of their analysis is dependent on the amount of water in each soil sample. The following table shows the tritium analysis sensitivity and cost for two different quantities of water per soil sample.

Water Quantity (ml)	Sensitivity (pCi/l)	Cost (ea.)
5	500	$64.00
150-200	10	$160.00

A 500 pCi/l sensitivity is good enough for our survey. Ten samples will cost $640.00 plus shipping. Assuming the moisture content of our soil is 10%max, a one liter soil sample will contain about 100ml of water. Turn-around time for tritium soil analysis is about 30 days.

APPENDIX E. COPY OF INTERNAL LETTER "Tritium Smear Survey, Building 030 Van deGraaf Accelerator - March 29, 1966"

INTERNAL LETTER
ATOMICS INTERNATIONAL
A Division of North American Aviation, Inc.

DATE March 29, 1966

TO W. F. Heine
ADDRESS 779 S040

FROM A. R. Mooers
ADDRESS 779 S040

PHONE 6301

SUBJECT Tritium Smear Survey, Building 030 Van de Graaff Accelerator

Tritium contamination levels on the building 030 Van de Graaff accelerator and its associated equipment have been assessed by means of a swipe sample survey. Five of the samples taken in the survey indicated severe tritium contamination. Table I describes the location and magnitude of this contamination.

TABLE I

Bld. 030 Tritium Contamination Levels

LOCATION	ACTIVITY (d/m-100 cm^2)
Plate below target holder	$\sim 1 \times 10^5$
Tritium target storage can	$\sim 1 \times 10^5$
Lower cooling hose	$\sim 1 \times 10^5$
Target holder (sides & front)	$\sim 3.6 \times 10^6$
Target holder (rear)	$\sim 3.6 \times 10^6$

Two of the samples contained sufficient contamination to saturate the gas proportional counter utilized in measuring the tritium activity. The activity of these samples was estimated by means of the following technique: Each sample was covered with a clean piece of sample paper with a 1/8" diameter hole at its center. Thus, the 18 kev tritium beta particles from the sample area outside the 1/8" diameter hole were shielded and only a 1/8" diameter area of the sample was counted. Counting in this manner produced a count rate of approximately

IL To: W. F. Heine
From: A. R. Mooers
Subject: Tritium Smear Survey. Building 030 Van de Graaff Accelerator

March 29, 1966

75,000 d/m. Assuming equal distribution of the contamination over the total area of the sample, a ratio of count rate to area was established and the total sample count rate was calculated by extrapolating to the total area of the 1" diameter sample.

It is understood that the accelerator is to be placed in operation in the near future. Prior to that time the areas of severe contamination should be decontaminated to an acceptable level. Further, it appears that the criteria in Radiation Safety Analysis #SSA-2 should be applied to this device, and that a safety procedure similar to that now being written for the 14 Mev neutron generator located in the SRE hot cell should be prepared for the Van de Graaff operations.

A. R. Mooers
Health Physics
Radiation Safety Unit

ARM/bam

cc: Alexander, R.E. 779 C004
Clow, H.E. 779 B040
Hillig, O.R. 736 B009
Reiter, M. 736 B035
Remley, M.E. 779 C004
Central File