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TITLE: "Radiological Survey of the Old Calibration Facility - Building T029"

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ABSTRACT

A radiological survey was performed at Building T029, located at Rockwell International's Santa Susana Field Laboratory (SSFL), to clarify and identify those areas needing further radiological inspection or requiring remedial action. Building T029, known previously as the Radiation Measurements Facility (RMF), was operated by North American Aviation and its successors in support of AEC, ERDA, and DOE nuclear related programs. T029 was used for storing and using sealed radioactive sources (Ra-226, Cs-137, Co-60, PoBe, and PuBe) to calibrate radiation detection instruments. All sources were handled in a totally encapsulated form. One incident occurred which resulted in the release of Ra-226 into its storage well. During recovery of the source, Ra-226 contamination spread in a limited area just around the opening of the well. T029 is currently used in a small capacity for storage of alkali metals. The purpose of this survey was to inspect the site for residual radioactive contamination.

The building interior, surrounding area, and entrance roadway were surveyed for gamma-emitting contamination. Ambient gamma exposure rate measurements were performed on a 6-m square lot plan. The Ra-226 source wells were surveyed for alpha contamination. An area south of T029 which was used in the early 1960s for storing barrels was also surveyed for residual radioactive material.

The results of this survey and analysis show that no residual radioactive contamination exists on the T029 floor or surrounding area. Survey of an area south of T029 which was used for storing barrels and drums in the early 60s shows no detectable activity. The Ra-226 source storage wells are still contaminated from the source rupture incident which took place in 1964. An alpha survey of a thimble raised from the bottom of the well showed 2800 α-dpm/100 cm^2. This contamination is not a radiological hazard in its present configuration, nor is this contamination spreading to the surrounding environment. Further investigation, decontamination, and disposition is required in these wells.

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1.0 INTRODUCTION

Building T029, formerly known as the Radiation Measurements Facility, was surveyed and analyzed for residual radioactive material. The facility and surrounding area were inspected for radioactive contamination to determine whether further investigation is required or remedial action is necessary. This radiological survey was conducted as prescribed in the "Radiological Survey Plan for SSFL," (Reference 4, section 5.4.3.1).

Building T029 is located in Area IV of Rockwell International's Santa Susana Field Laboratories (SSFL) in Ventura County, California. Designed and built in the late 1950's to support government nuclear-related programs, T029 was used for storing and using radioactive sources (Ra-226, Cs-137, Co-60, PoBe, and PuBe) to calibrate radiation detection instruments. These instruments were used at the Sodium Reactor Experiment (SRE), Organic Moderated Reactor (OMR), Uranium Carbide Pilot Fuel Facility (UCPFF), and other facilities where research was sponsored by the government. Ra-226, Cs-137 and Co-60 were used to calibrate gamma-sensitive instruments. PoBe and PuBe were used to calibrate neutron-sensitive instruments. These sources were always handled and stored in encapsulated form. Leak testing was performed every six months. All three Ra-226 sources were encapsulated in platinum iridium alloy. Each Ra-226 source was stored inside an individual "well" about 9.5 ft deep by 1 in diameter. The sources were raised and lowered using nylon string. The two smallest Ra-226 sources were inside plastic capsules, connected to nylon string. Each Ra-226 source could be raised about 3 ft above floor level. In March 1964, one of the Ra-226 sources fell to the well bottom, cracked, and released contamination. Because of this incident, these Ra-226 sources were removed, disposed of, and then replaced by two Cs-137 sources. The Co-60 source was used in a stationary configuration. A large concrete block above floor level housed the source. A 1-ft thick concrete rolling door on the top was rolled to one side to expose the source. The neutron sources (PuBe, PoBe) were stored in a pit and transferred to a graphite moderator block for use in calibrating instruments. The Co-60, PuBe, and PoBe sources were not used very long.

They were probably discontinued and removed in the early to mid 1960s. The Cs-137 sources were removed by April 29, 1974.

One incident occurred at T029 which resulted in a release of contamination to the building, but not to the surrounding environment. In 1964, the plastic capsule containing the smallest Ra-226 source became lodged and fractured, allowing the source to fall about 13 ft to the "well" bottom. The source capsule cracked on impact and released contamination inside the well. It was not realized that the capsule had ruptured until source recovery began the following day. Most contamination from the rupture and all recovery operations were concentrated in or around the well. Contamination was not spread. Shortly thereafter, the Ra-226 sources were replaced by a Cs-137 source. On November 20, 1970, the 4.6 Ci Cs-137 source dropped 10 ft to the bottom of the "well." No contamination release occurred. When all sources were removed from T029 in 1974, a radiation survey was performed to show that the facility was radiologically "clean." However, because of the one contamination incident and use of radioactive material at T029, a radiological survey was performed to document the facility's current radiological condition.

As part of the DOE SSFL Site Survey (Reference 4, section 5.4.3.1), a radiation survey was performed in the building and surrounding area 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 radiation emitted from the radioactive materials used at T029, except for the PuBe and PoBe sources. No problems were ever experienced with these sources. An alpha survey was performed on a source thimble raised from the storage well where Ra-226 contamination was released. Also included in this survey was an area just south of T029, which from an old photograph (Figure 2.5), shows that it was used for storing drums and barrels.

All 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. Because of the large variability observed for natural background at SSFL, ambient gamma measurements are first presented as gross data (without "background" subtraction) and compared against three independent "natural" background distributions. Then using an average natural ambient gamma exposure rate (i.e. "background") based on the three distributions, the measurement data is corrected for "background" and tested against the acceptance limit.

2.0 IDENTIFICATION OF FACILITY PREMISES

2.1 Location

Building T029 is located within Rockwell International's Santa Susana Field Laboratory (SSFL) in the Simi Hills of southeastern Ventura County, California, adjacent to the Los Angeles County line and approximately 29 miles northwest of downtown Los Angeles. The SSFL location relative to the Los Angeles area and surrounding vicinity is shown in Figure 2.1. Figure 2.2 is an enlarged map of neighboring SSFL communities. Figure 2.3 is a plot plan of the western portion of SSFL which includes Area IV where Building T029 is located. It is located within the 90.26-acre government-optioned area.

2.2 Building Characteristics and Site Topography

Building T029 was constructed in 1959 and was originally designated as the Radiation Measurements Facility (RMF). It was used until 1974 as a facility for calibrating radiation detection instruments.

Constructed as an open bay, T029 is a Butler-type building with a steel frame, and corrugated metal siding and roofing. The building is 20 ft x 40 ft with a 12-ft eave height. T029 has no restroom facilities, offices, or support laboratories; it is a single room. Ventilation is provided by an exhaust blower rated at 1540 ft^3/min. Facility air exhausts through two absolute filters. Air conditioners were never installed. The ceilings and walls are insulated with 1-in thick fiberglass mat. The floors were originally surfaced with asphalt tile. The floor now is a concrete slab.

Three calibrator locations were installed: 1) a graphite neutron exposure block, with a 3′ x 3′ x 2′-deep pit for storing a PoBe and PuBe source; 2) a concrete block with rolling door for storing Co-60; and 3) a 9.5-ft deep, 12-in diameter pipe with three 1-in diameter tubes for storing three Ra-226 sources, which were eventually replaced with a Cs-137 source.

Figure 2.1 Map of Los Angeles Area

Figure 2.2 Map of Neighboring SSFL Communities

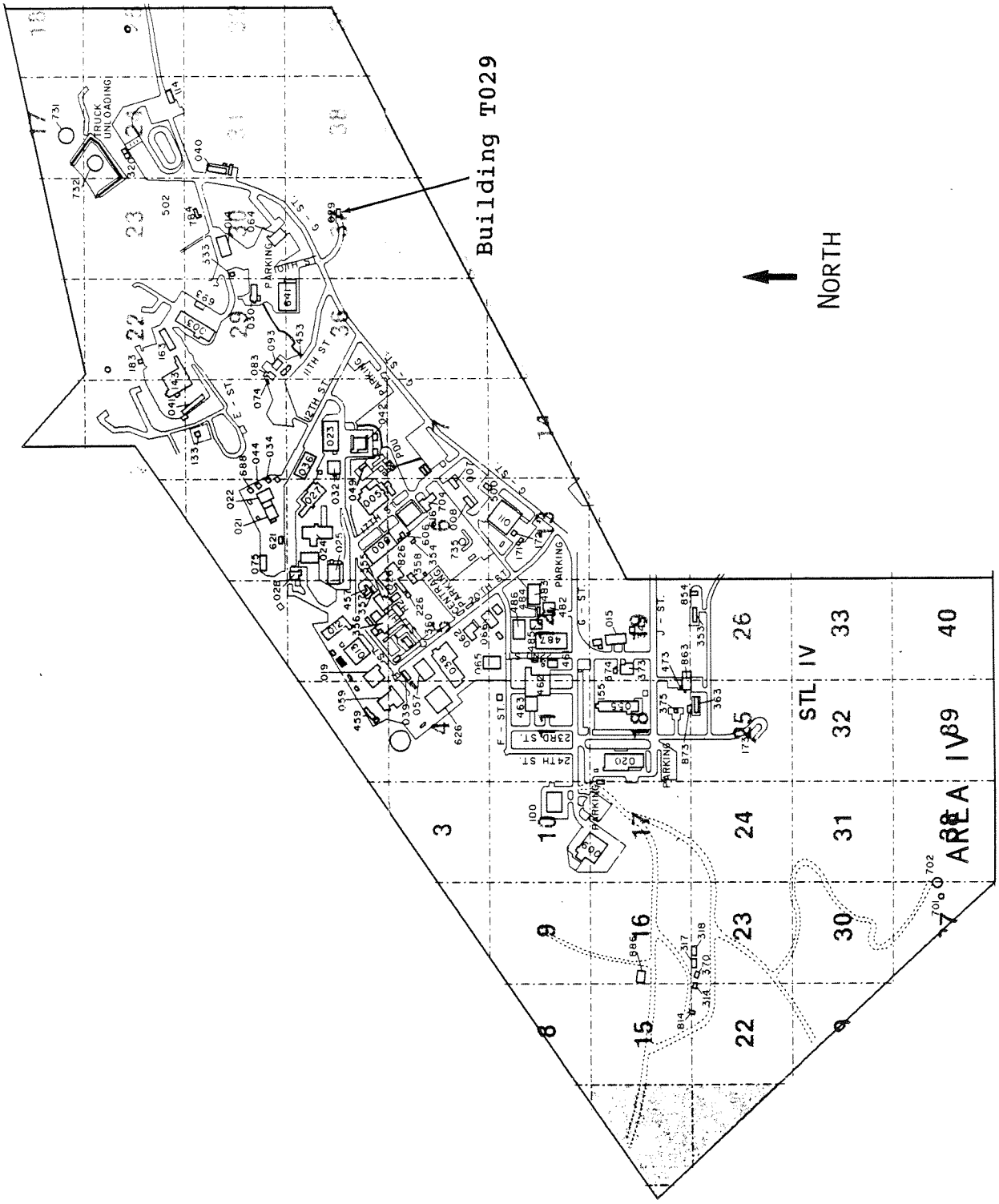


Figure 2.3 SSFL Layout, Showing the Location of Building T029

Figure 2.4 is a plot plan of building T029 showing the source storage locations and the calibrator control panel. Table 2.1 below lists the radioactive sources used at the facility.

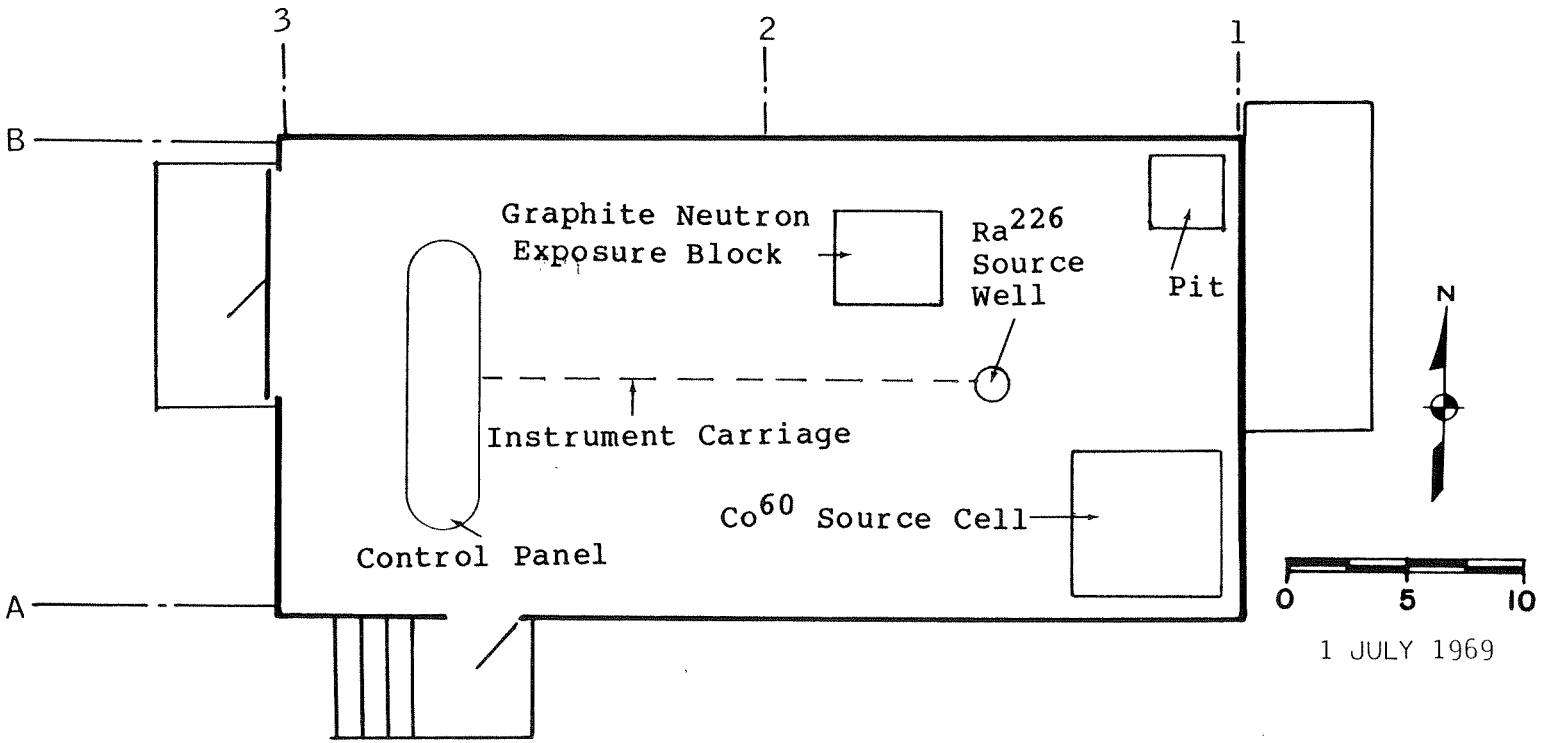
Table 2.1 Calibration Sources Used at T029

Source	Source Strength (mCi)	Date
(1) Ra-226	24.8	1960
(2) Ra-226	132	1960
(3) Ra-226	930	1960
(4) Co-60	Unknown	
(5) PoBe	Unknown	
(6) PuBe	Unknown	
(7) Cs-137	5310	September, 1963
(8) Cs-137	5260	September, 1963

A PoBe and a PuBe source were stored in a small concrete pit in the northeast corner, no more than 3-ft x 3-ft wide and 2-ft deep. This pit was also used to store gamma sources in lead pigs when not in use. A graphite neutron exposure block was built for neutron moderation and measurement experiments. This neutron installation was rarely, if ever, used. The sources were fully encapsulated. No problems ever occurred. From all current personnel recollections, the neutron facility was dismantled in either 1964 or 1965.

A Co-60 calibrator was installed in the southeast corner. The source was stored in a 12-in diameter well (pipe) which extends 10 ft below floor elevation and 4 ft above floor elevation. Above grade, the pipe is enclosed with lead shielding and a 77-in square concrete rolling door. The concrete door was rolled to one side to expose the source. The encapsulated Co-60 source, was used in the early 1960s. No problems ever occurred. Because it was used on a limited basis, it was removed from the facility in about 1965. The well and concrete are still in place.

The primary radiation instrument calibrator used a 12-in diameter, schedule 40 galvanized pipe which extends 10 ft below floor elevation.



RADIATION MEASUREMENTS FACILITY

BLDG. NO. 029

Figure 2.4 Plot Plan of the Radiation Measurements Facility, Building T029

Within the 12-in pipe there was three, 1-in diameter pyrex tubes evenly spaced in a linear configuration and extending to a depth of 9 ft, 6 in. Three Ra-226 sources were each connected to nylon strings and placed at the bottom of the tubes. The void space between these tubes and the 12-in pipe was filled with concrete. To calibrate an instrument, a nylon string was pulled from the control panel, thus raising a source about 3 ft above floor elevation. This calibrator was a system of pulleys and cables. Ra-226 was used until the contamination incident occurred in 1964. Cs-137 was then used without a single contamination incident until the facility was closed in 1974. The control panel was removed but the 12-in diameter pipe with three 1-in tubes is still intact.

Access to T029 is from 10th Street, which intersects "G" Street just southwest of building T064. An asphalt concrete roadway (10th Street) runs right up to the facility. A portion of this roadway is fenced-in as part of the facility. A large outcropping of the Chatsworth sandstone formation is located adjacent to the east and north sides of the road, and north of the building. Figure 2.5 is an old photograph of T029 and the surrounding area looking south-southwest. This photo shows a small area in the field just south of T029 which was used for storing drums and barrels. The origin and contents of these containers is unknown. We doubt that radioactive material was ever placed there, but as part of this investigation, this area was surveyed for residual radioactive material. Figure 2.6 shows the facility from the south, including the fenced-in area and large sandstone outcropping. Figure 2.7 is a close-up of T029 from the south. Figure 2.8 shows the entrance gate on 10th Street and the west wall of T029.

2.3 Building Utilization and Present Radiological Condition

T029 was used as a radiation detection instrument calibration facility from 1956 to 1974. Primarily gamma-sensitive detectors were exposed to known radiation fields for determining instrument response. Neutron detectors were used for a short time period. All radioactive sources stored and handled here were fully encapsulated. All sources were leak tested at least every 6 months in compliance with California State

Radiation Control Regulations to ensure no leakage of radioactive material into the building.

The PoBe and PuBe neutron sources were stored and used on a limited basis, if at all. These sources were stored in large containers filled with paraffin inside the pit. Source handling was not performed with pulleys and cables. Most likely, sources were handled by a long pole. The neutron experimental apparatus was never completely installed and used. It was dismantled in about 1965. The sources were removed from the facility. No problems ever occurred. There is no reason to believe that loose alpha-emitting Pu is contaminating the facility. Po-210, even if it had leaked, with a 138 day half-life, has decayed away by now.

The Co-60 gamma source was also used very little. This calibrator was very awkward to use. An instrument would be placed on a tripod arrangement above the 12-in pipe. The heavy concrete rolling door was then rolled to expose the source to the detector. A mirror was installed to read the meter. This procedure was cumbersome and probably resulted in extra radiation exposure. The Co-60 source, fully encapsulated, was removed from T029 sometime after 1964. No problems ever occurred with this source.

T029 was equipped with three Ra-226 sources in a bromide salt form. The mass of each source was 24.78 mg, 132 mg and 930 mg. Each source was encapsulated in platinum-iridium alloy, 1 mm wall thickness, except for the 930 mg source which had a wall thickness of 1.5 mm. The two smallest sources were placed inside plastic capsules and attached to a nylon string. The large source was placed inside a metal source holder. Each source was raised and lowered in its 1-in pyrex tube, (10 ft long) from the control panel. The sources were raised during calibration about 3 ft above floor elevation.

In September 1961, the 132 mg Ra-226 source was dropped from the top of its source well. No radioactive material was released, (Reference 19, p. 4); see Appendix D.

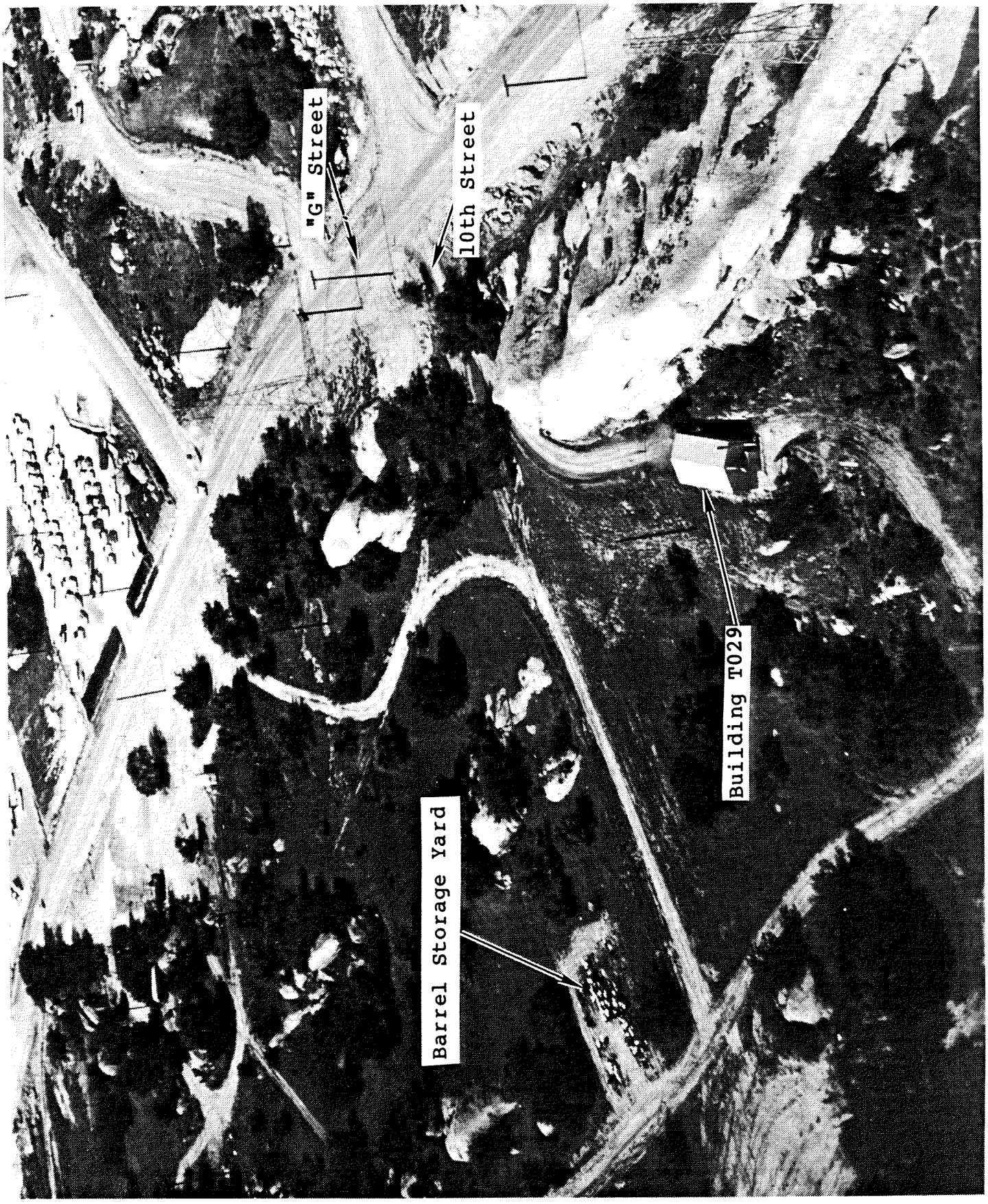


Figure 2.5 Photograph of T029 Looking South-Southwest Which Shows Collection of Drums South of the Facility

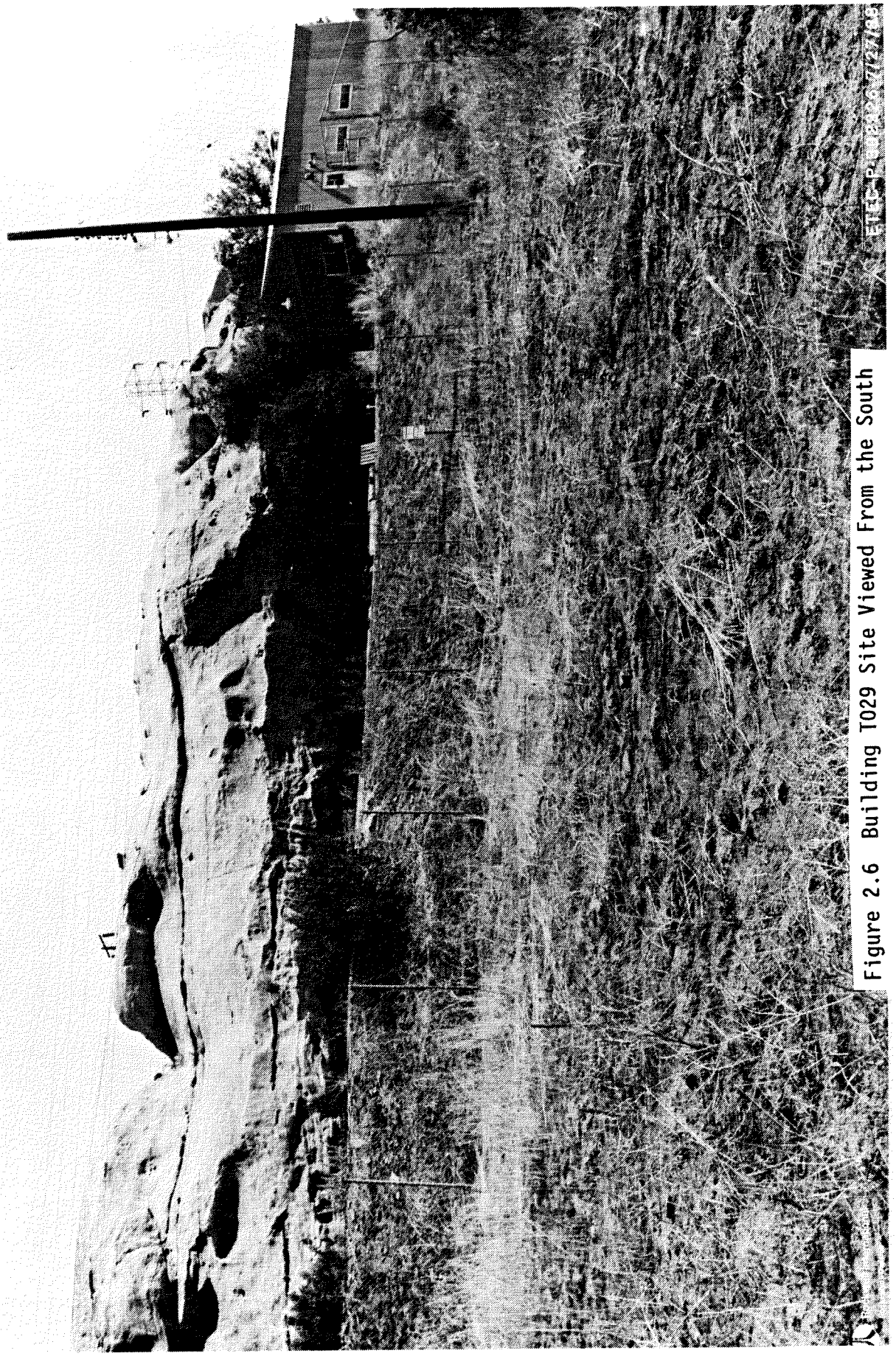


Figure 2.6 Building T029 Site Viewed From the South

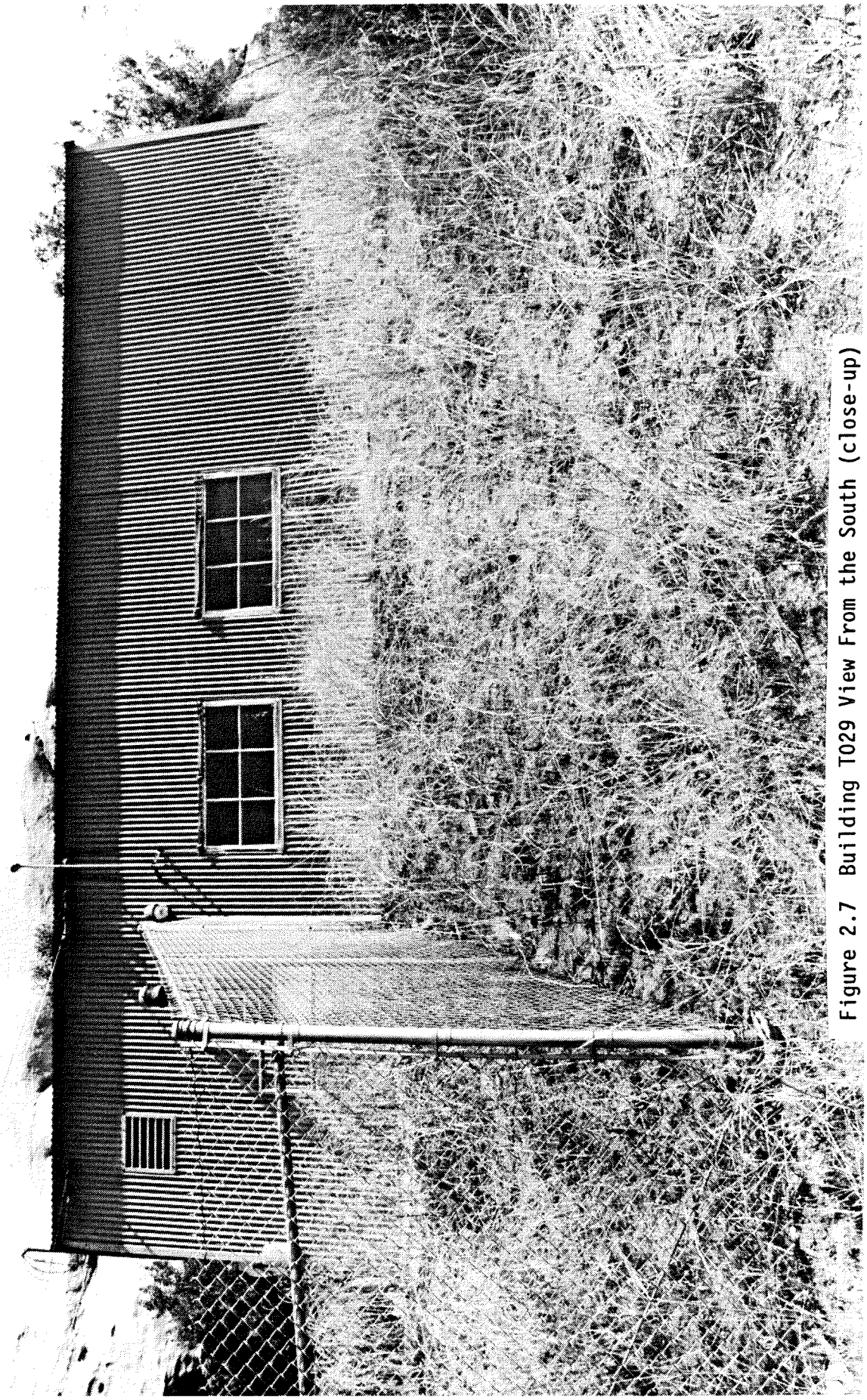


Figure 2.7 Building T029 View From the South (close-up)

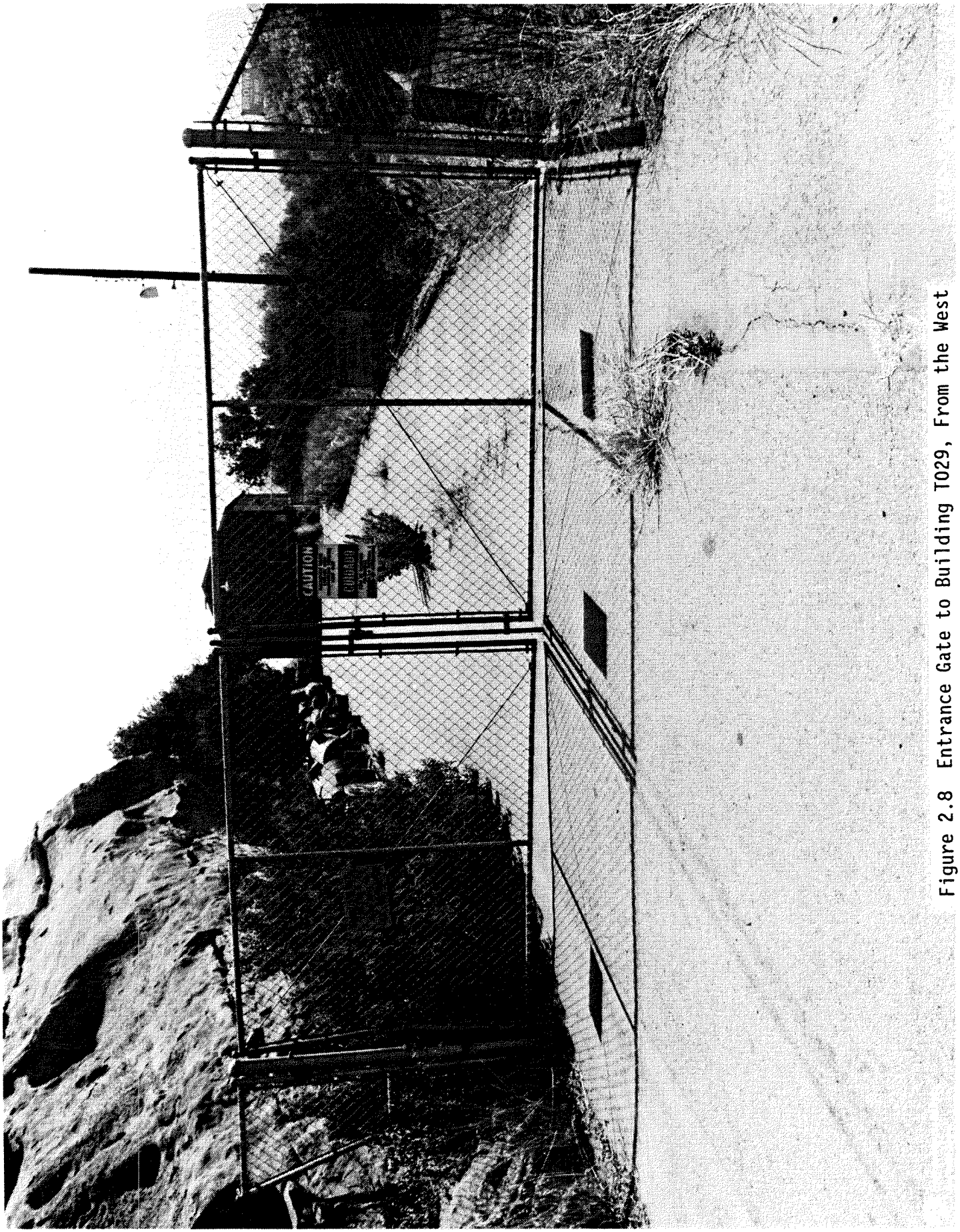


Figure 2.8 Entrance Gate to Building T029, From the West

On March 23, 1964, the plastic capsule which contained the 24.76 mg source, became lodged and fractured. The source and capsule fell about 13 ft to the well bottom. Not until the following day, when source recovery was attempted, was it realized that the source had leaked. Apparently, the source cracked on impact and released some loose Ra-226 into the well. High volume air samples were set up and smears were collected. Air sample results were at background levels. Contamination released was confined to the well and at floor level very near the penetration opening. Smears near the well showed 90 dpm/100 cm^2 (beta) and 120 dpm/100 cm^2 (alpha) - this was before source recovery. Once the source was recovered, it was sealed in a 3" long, 1/2" diameter stainless steel pipe to minimize Rn-222 emanation to the surrounding area. The area was decontaminated to background levels and a rubber stopper was placed in the 1-in pyrex tube, (Reference 19, pp. 1 through 4); see Appendix D. Although the area was deconned, contamination is still suspect in the wells.

Shortly after this contamination incident, the Ra-226 sources were dispositioned and replaced by two Cs-137 sources, rated at 5.31 curies and 5.26 curies (10/2/63). Each source consisted of a double-walled stainless steel inner capsule, encased in a heavy-walled 416 stainless steel outer capsule which was closed by brazing. These sources were raised and lowered within the pyrex tubes in a manner similar to that of the Ra-226 sources. On November 20, 1970, the 4.6 Ci Cs-137 source capsule in storage well 2 failed. The failure occurred at the brazed joint at the outer capsule bottom plate, releasing the double-walled inner source capsule into the 10-ft deep well. A smear survey of the outer container and an area adjacent to the wells indicated no release of radioactive contamination in excess of 50 dpm/100 cm^2, (Reference 20, pp. 1-2); see Appendix E. Once the source was recovered, a leak test was performed, indicating less than 0.005 μCi of removable contamination. Smears of the surrounding area showed removable contamination levels less than 50 dpm/100 cm^2. No contamination was released. The source was never used again. It is believed that the other Cs-137 source was used on a limited basis until the facility was closed on

April 29, 1974, at which time all radioactive sealed sources were removed and transferred to another facility.

When the sources were removed, all radiation and radioactive material warning signs were removed. The facility was surveyed and released for use by other programs. Subsequently, T029 was redesignated as a Hazardous Waste Storage Facility, and has been used for storage of excess alkali metals including Na, NaK, Li, and LiH_2. Alkali metal contaminated components have also been stored at this facility. The old Ra-226 storage wells continue to be a known contaminated area; however, the level is thought to be fairly low. A complete account of the contamination present 10-ft down is not achievable until the source wells are completely removed and an examination of surrounding soil can be made.

3.0 SURVEY SCOPE

Interior building areas and surrounding land were radiologically inspected by measuring ambient gamma exposure rates 1 meter above the surface. Further investigation would have commenced if this gamma measurement indicated contamination. Forty ambient gamma exposure rate measurements were made in the area. A survey "for indication only" was performed in a small area south of T029 where drums and barrels were once stored. The ambient gamma exposure rate data were analyzed statistically by sampling inspection by variables techniques against appropriate residual contamination acceptance limits.

3.1 Unrestricted-use Acceptable Contamination Limits

A sampling inspection plan using variables, discussed in Section 4.2, was used to compare radiological contamination quantities against unrestricted-use acceptable contamination limits prescribed in DOE guidelines (Reference 1), Regulatory Guide 1.86, NRC license SNM-21, and other references. The limits shown in Table 3.1 below have been adopted by Rocketdyne. Measurements of average surface alpha/beta contamination 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. Current guidance for acceptable soil radioactivity is nearly non-existent. The limits used here for alpha contamination are based on Ra-226 (Reference 1). No effort was made to sum the concentrations of individual radionuclides and calculate the dose for the mixture so as to show that it does not exceed the basic dose limit.

Table 3.1 Building T029 Maximum Acceptable Contamination Limits

Criteria	Alpha (dpm/100 cm^2)	Beta (dpm/100 cm^2)
Total Surface, averaged over 1 m^2	100	5000
Maximum Surface, in 1 m^2	300	15000
Removable Surface, over 100 cm^2	20	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.

** Alpha activity concentration limits 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.

*** 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.

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

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

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

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

3.2 Sample Lots

For purposes of the T029 radiological survey, the building and surrounding area was 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, and interior areas of T029 were inspected.

Because T029 is currently a storage building for alkali metals, about 50% of the floor space was occupied by palletized 55-gal drums. Six measurements were made in easily accessible locations. Outside the building, a 6-meter square grid was superimposed over the terrain and one ambient gamma exposure rate measurement made in each 36-m^2 area. Location (1,1) was the northwestern most grid on the site (near "G" Street). The barrel storage area south of T029 was surveyed for "indication only."

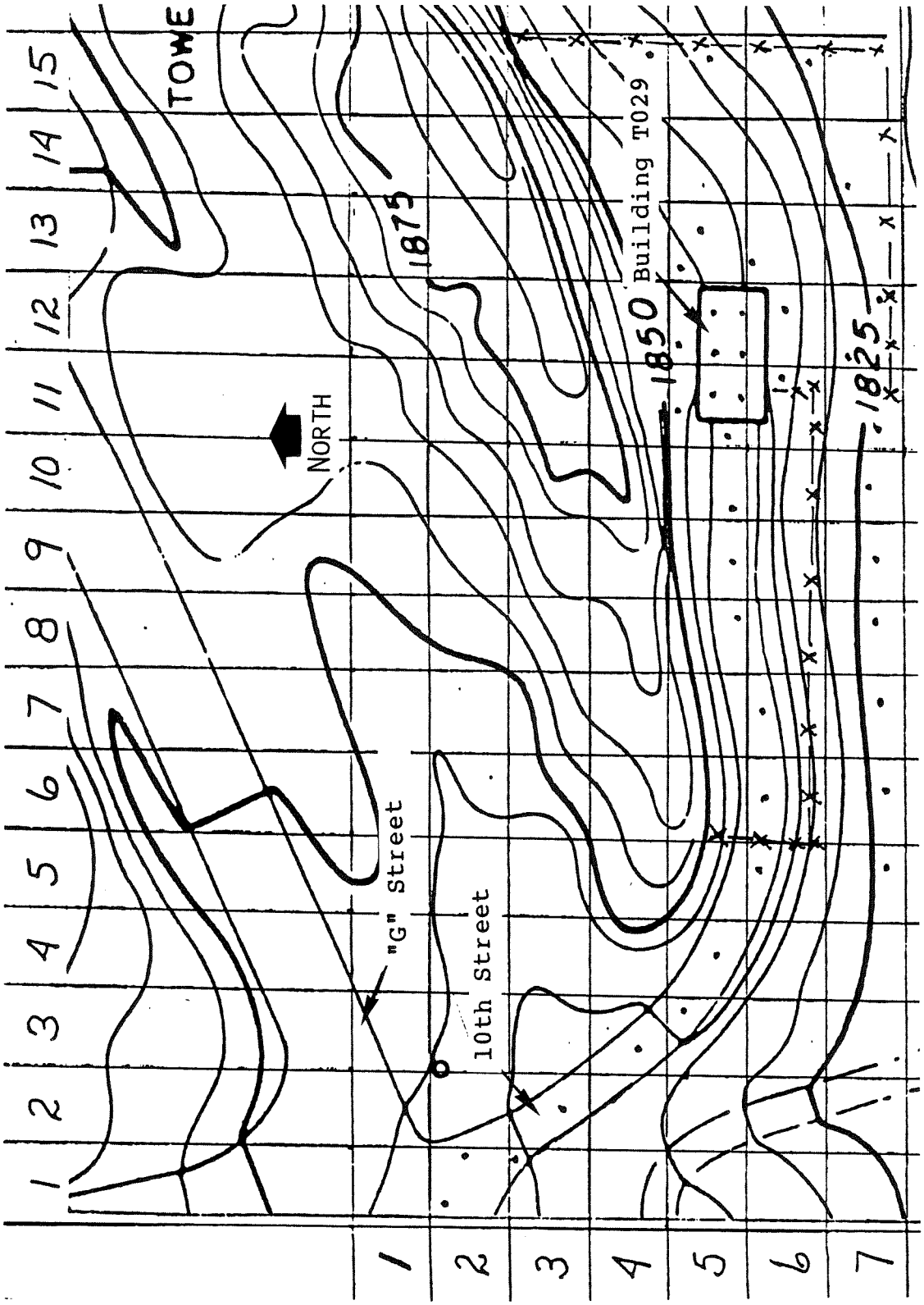


Figure 3.1 Building T029 Sampling Lot Plan

The survey scope was set forth by the "Radiological Survey Plan for SSFL," (Reference 4, Section 5.4.3.1):

1. "Survey floor of building - Ra-226 (investigate old records for disposition of storage holes in building);

2. Roadway and slope, surface around building (Ra-226)."

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" x 1" NaI crystal coupled to a photomultiplier tube and fed to a Ludlum 2220-ESG scaler, at 1 m from the ground. In each cell, a 1-min. count was collected and converted to μR/h. The measurement location and exposure rate were recorded in tabular form. About 40 1-min. measurements were acquired.

3.4 Surface Soil Samples

No gamma exposure rate measurements indicated a level exceeding the 50% characterization level; therefore, no soil samples were collected and analyzed.

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. Because of the contamination incident which occurred in 1964, the Ra-226 source wells were surveyed for alpha contamination.

Measurements of alpha activity were made by use of a Ludlum model 43-5 rectangular alpha scintillation probe (active area = 50 cm^2) coupled to

a Ludlum model 12 count rate meter. This detector was calibrated using a Th-230 source. The energy of Th-230 alpha particles (4.6 MeV) is similar to that of Ra-226 (4.78 MeV) and several of its daughters. A slight over-estimate of activity will result due to the higher energy alpha particles from some of the daughters.

Measurements of beta activity were not necessary and were not performed.

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:

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

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

m = average, or mean of the population distribution

σ = standard deviation of the population distribution.

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

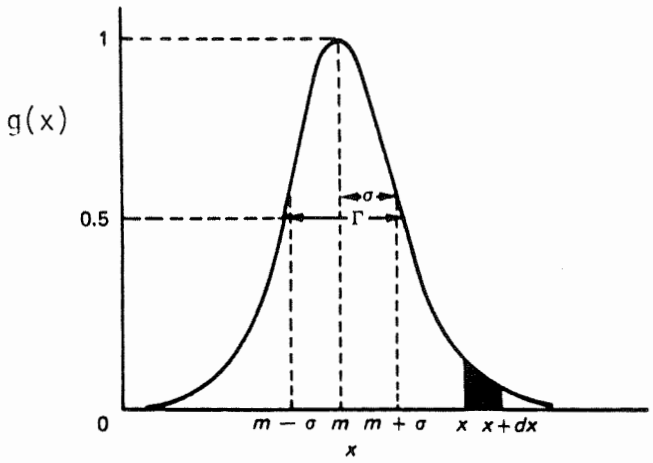


Figure 4.1 The Gaussian Probability Density Function

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

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

$$= P(x < X)$$

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

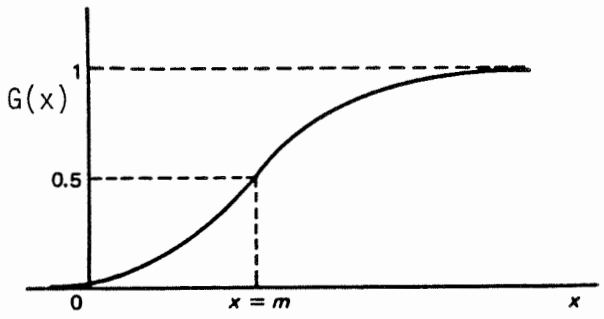


Figure 4.2 The Gaussian Cumulative Distribution Function

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

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

4.2 Sampling Inspection

4.2.1 By Variables

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

4.2.2 By Attributes

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

4.3 Sampling Inspection by Variables

4.3.1 Calculated Statistics of the Gaussian Distribution

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

$\overline{x}$ = average (arithmetic mean of measured values) of sample
s = observed sample 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:

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

The sample mean, standard deviation, and acceptance limit are easily calculable quantities; the value of k, the tolerance factor, bears further discussion. Of the various criteria for selecting plans for acceptance sampling by variables, the most appropriate is the method of Lot Tolerance Percent Defective (LTPD), also referred to as the Rejectable Quality Level (RQL). The LTPD is some chosen limiting value of percent defective in a lot. Associated with the LTPD is a parameter referred to as consumer's risk (β), the risk or probability of accepting a lot with a percentage of defective items equal to the LTPD. It has been standard practice to assign a value of 0.10 for consumer's risk (β). Conventionally, the value assigned to the LTPD has been 10%. These a priori determinations are consistent with the literature and regulatory position, and are the same values used by the state of California (Reference 2). Thus, based on sampling inspection, we are willing to accept the hypothesis that the probability of accepting a lot as not being contaminated which is in fact 10 percent defective (i.e. above the test limit, U) is 0.10. The value of k, which is a function of the a priori determinations made for β and LTPD is given by equation 4-8.

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

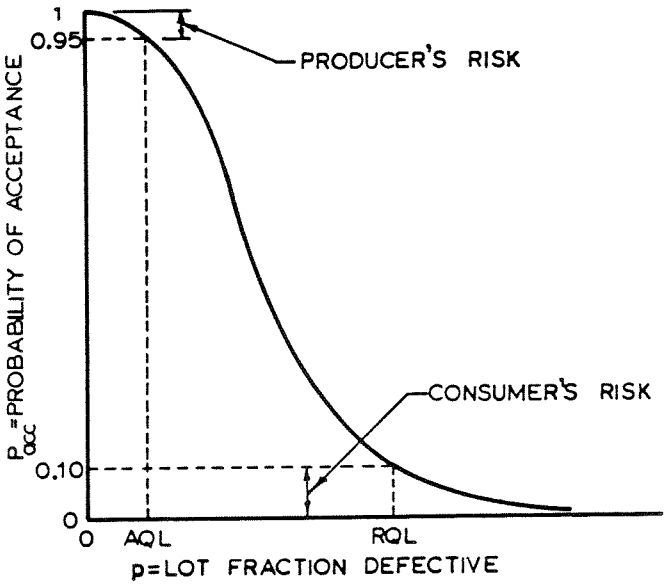


Figure 4.3 Operating Characteristics Curve

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

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

where:

k = tolerance factor

K_2 = the normal deviate exceeded with probability of β, 0.10 (from tables, K_2 = 1.282)

K_β = The normal deviate exceeded with probability equal to the LTPD. 0.10 (from tables, K_β = 1.282)

n = number of samples

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

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.

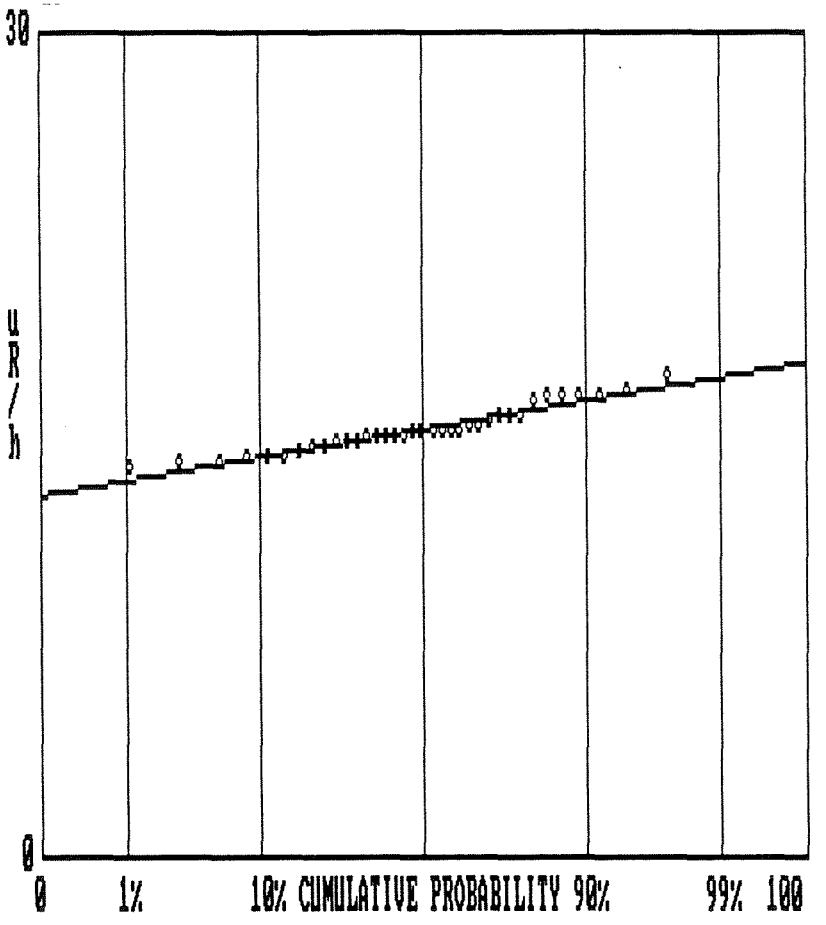


Figure 4.4 Gaussian cdf Plotted on Probability-Grade Paper

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. This was 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 a sampling area is slightly contaminated, contaminated above acceptance limits, or whether additional investigation is required. That decision is based on one type of radiological measurement: gamma exposure rate. This type of measurement is sensitive to Ra-226 radiations 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, follow-up investigation commenced. At TO29, further investigation was not required. Alpha and beta radioactivity measurements were made in the Ra-226 well and old barrel yard for "indication only," respectively.

Analytical techniques used to acquire, evaluate, and interpret these radiological measurements are presented in detail in this section. This includes 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.

5.2 Data Reduction Software Program

Each radiological measurement characteristic data value was input into SMART SPREADSHEET. This is an off-the-shelf computer software package which allows multiple computations to be performed on raw data values. Columns were established to calculate the 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 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 (μR/h/cpm).

Output for Gaussian plots of these measurements:

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

5.3 Data Analysis

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

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

5.4 Direct Alpha/Beta Contamination Measurements

Direct alpha/beta contamination measurements were made for "indication only" on an as-needed basis. Ludlum model 12 count-ratemeters were connected to Ludlum 43-5 alpha probes and 44-9 beta probes.

5.4.1 Instrument Calibration

Each detector was calibrated before use. The alpha detector was calibrated with Th-230; the beta detector with Tc-99. Background levels were determined in an area of similar characteristics, known to be uncontaminated.

5.4.2 Data Acquisition and Reduction

The gross number of alpha and beta counts recorded by the count-ratemeter were converted to dpm/100 cm^2 by equation 5-1:

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

where:
- SA = surface activity
- C = gross count rate as displayed on analog meter (cpm)
- B = background count rate as displayed on analog meter (generally 0-5 cpm for alpha and about 50-75 cpm for beta)
- EF = Efficiency factor, dpm/cpm (averages between 6.5-7.0 for alpha and about 5.0-6.0 for beta)
- 100 = 100 cm^2 standard area
- A = probe sensitive area (50 cm^2 for Ludlum model 43-5 rectangular alpha scintillator; 20 cm^2 for Ludlum model 44-9 pancake G-M)

The standard deviation of the measurement in dpm/100 cm^2 is given by:

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

C and B are both measured in units of counts per minute (cpm).

5.4.3 Data Analysis

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

5.5 Removable Alpha/Beta 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" x 1" 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 detect 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 does not read less than 200 μR/h, this instrument was cross-calibrated against a Reuter Stokes High Pressure Ion Chamber (HPIC). Count rates were converted to exposure rates by the relationship that about 215 cpm = 1 μR/h, at background exposure rates. This calibration was performed several times.

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

5.6.2 Data Acquisition and Reduction

Each location where a gamma measurement was made was identified on a map and in matrix notation. The gross number of counts recorded in 1 min. along with the matrix notation location was input into SMART SPREADSHEET. Columns were established to calculate the total exposure rate (μR/h) and its standard deviation according to the equations 5-3 and 5-4. Gamma scintillations produced by a NaI detector were converted from gross counts to exposure rate (μR/h) by:

$$R = \frac{(C) * (EF)}{1 \text{ min.}} \qquad \text{(Eq. 5-3)}$$

where R = exposure rate (μR/h)
C = gross counts in 1 min. (cpm)
EF = efficiency factor (0.0047 μR/h/cpm) based on cross calibration with HPIC.

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

$$s = \frac{\sqrt{C} * (EF)}{1 \text{ min.}} \qquad \text{(Eq. 5-4)}$$

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

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

The T029 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 location. Measurements were taken on flat and rugged terrain, with Chico Formation sandstone, similar to conditions surrounding T029.

These distributions make no corrections for "background"; the total-gross gamma exposure rate is considered. Then, the average of the "true background" areas is used as our best estimate for background radiation at SSFL. Using this value, we correct the T029 data and compare the resulting distribution against acceptance limits.

5.7 Surface Soil Samples

If radioactive contamination was detected from performance of ambient gamma exposure rate measurements, soil samples were collected from the general area to qualify and quantify the radioactivity. At T029, soil sample analysis was not necessary and was not performed.

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 T029, (1,1) is the western-most location on 10th Street, adjacent to "G" Street. Measurement locations should be marked off every 6 meters down (east) 10th Street to T029. Mark grids inside T029 and in the area immediately surrounding it.

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) Window in/out switch is set to out.

5) Check battery (greater than 500).

6) Set range selector to 1, response to fast, and count time 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 43-5 rectangular alpha scintillation probe, survey various building features and components which are

suspect of containing residual alpha contamination. The only suspect area is the Ra-226 source well.

2) Perform an instrument calibration check using a Th-230 source.

3) If necessary, do the same for beta contamination using a Ludlum model 43-1 beta probe.

4) 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 Building T029 radiological survey 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 and around T029 to measure ambient gamma exposure rates. Radiological data for this lot was statistically analyzed. Analytical interpretation of this data set shows an uncontaminated facility and grounds. The Ra-226 storage wells are known, previous to this survey, to be contaminated.

7.1 Statistical Results Format

Gamma exposure rate data collected during this survey of T029 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 T029 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 noncontaminated. "k" and thus the "TS" line increase as the number of samples in a lot decrease.

At the top left hand corner of the output is the file name of the data file for the sample lot. 30 μR/h is normally used for convenience, as the maximum ordinate value. If measurements exceed 30 μR/h, then the greatest measurement forms 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.

7.2 Ambient Gamma Exposure Rates

Forty ambient gamma exposure rate measurements were made in and near building T029. Six measurements were made inside, which exceeded our criteria that a minimum 5.5% survey be performed (i.e. 1 m^2 surveyed per 36

m^2). Appendix C shows the data set sorted in order of decreasing exposure rate. Notice that the smallest exposure rates were measured inside. Table 7.1 shows the statistics for the T029 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) the "background" gamma-radiation environment is quite variable at SSFL; and 2) 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 last two 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 clearly the difference between indoor and outdoor measurements. The smallest 6 points correspond to indoor measurements. The slight deviations observed on the high end of the distribution 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 slope equal to zero. 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 T029 Measurements

Location	No. of Measurements	Average Exposure Rate (μR/h)	Standard Deviation μR/h	Range μR/h
T029 Entire Data Set	40	14.4	1.55	6.05
T029 Indoor Data Set	6	11.4	0.94	1.32
T029 Outdoor Data Set	34	14.9	0.87	3.35
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 approaches the NRC limit of 5 μR/h.

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. The deviations observed in the T029 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

Figure 7.1 Ambient Gamma Radiation at Building T029 and Surrounding Area

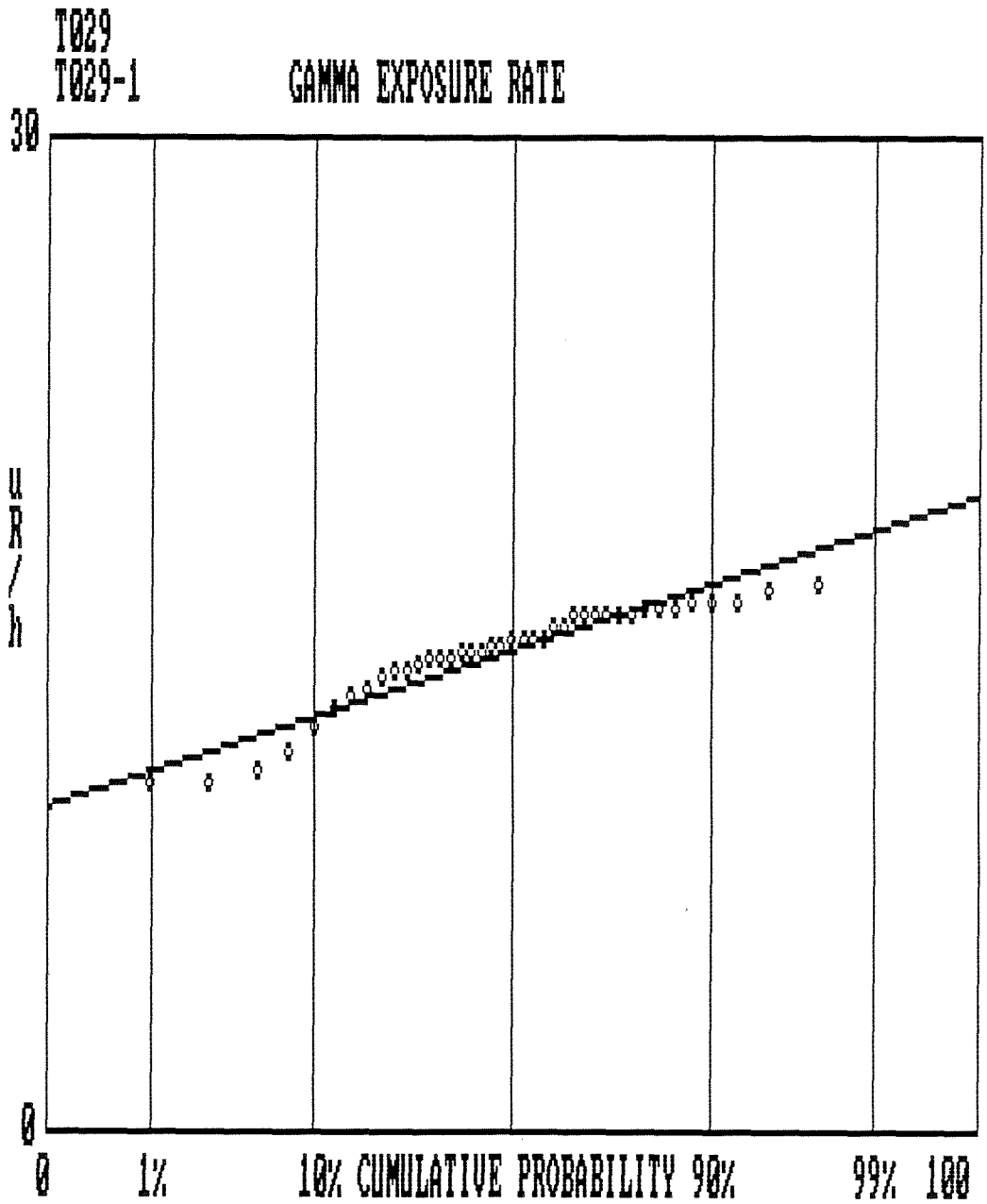


Figure 7.2 Ambient Gamma Radiation at Area Surrounding Building 309
(Background Distribution)

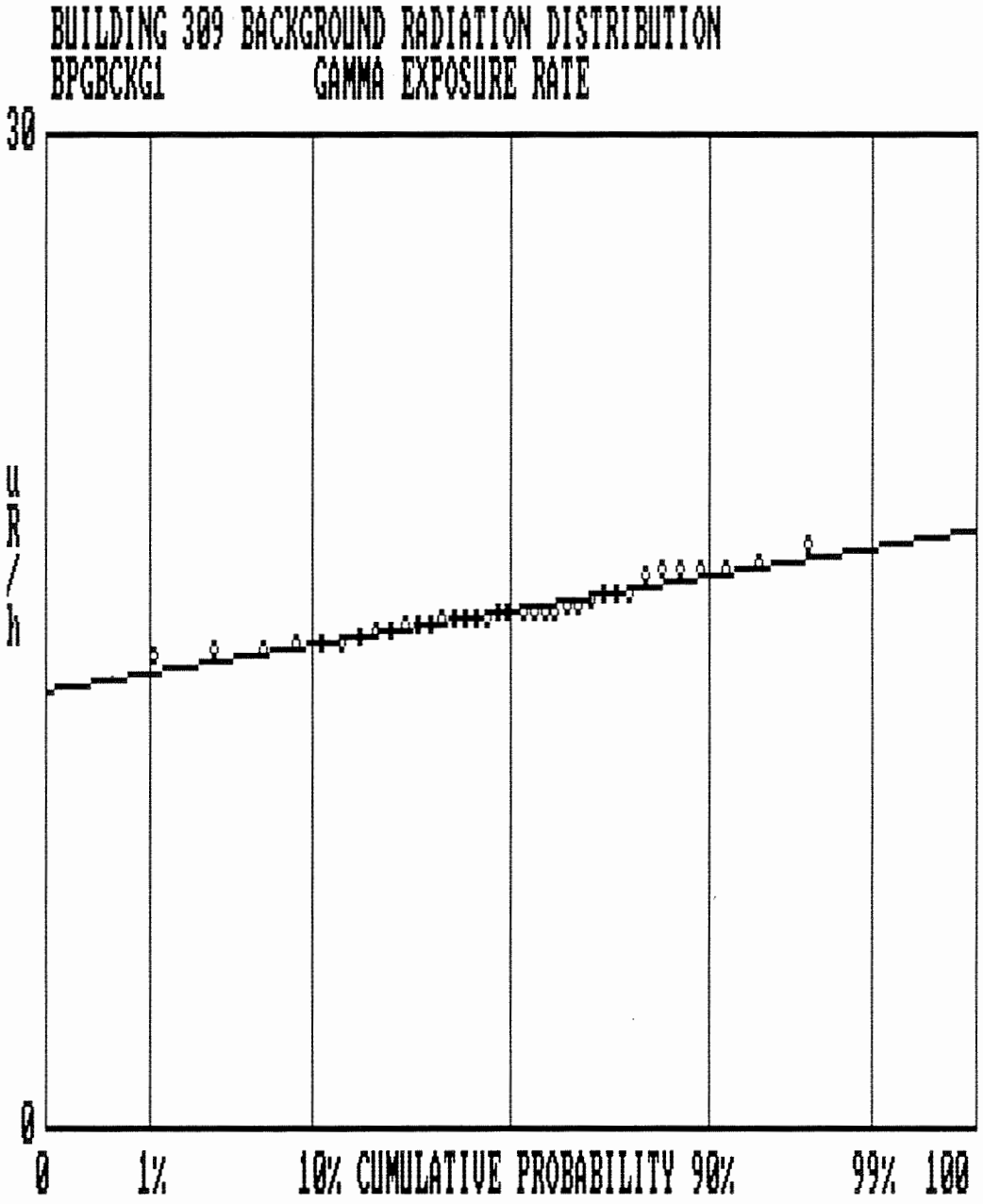


Figure 7.3 Ambient Gamma Radiation at Area Well #13 Road
(Background Distribution)

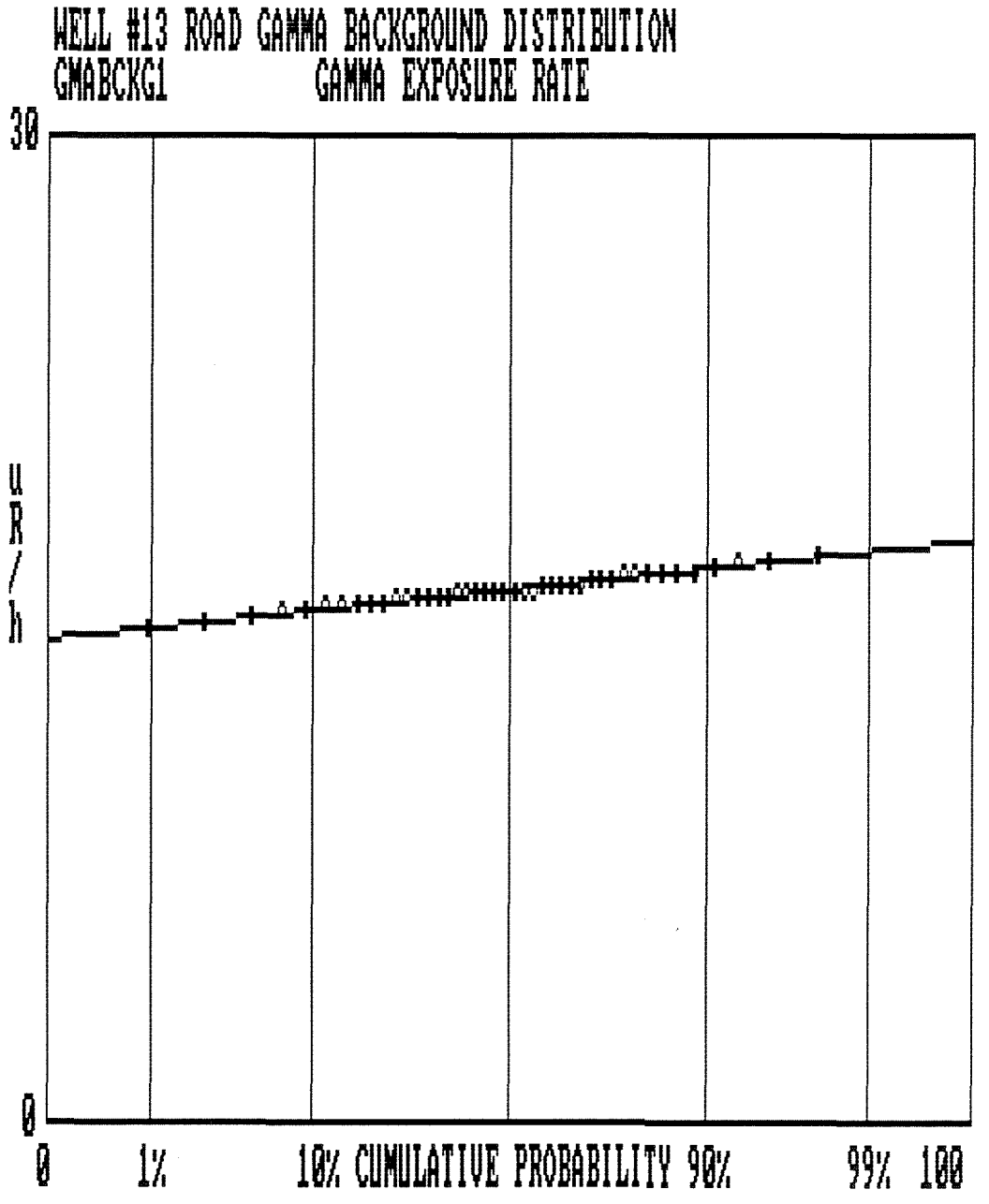


Figure 7.4 Ambient Gamma Radiation at Incinerator Road
(Background Distribution)

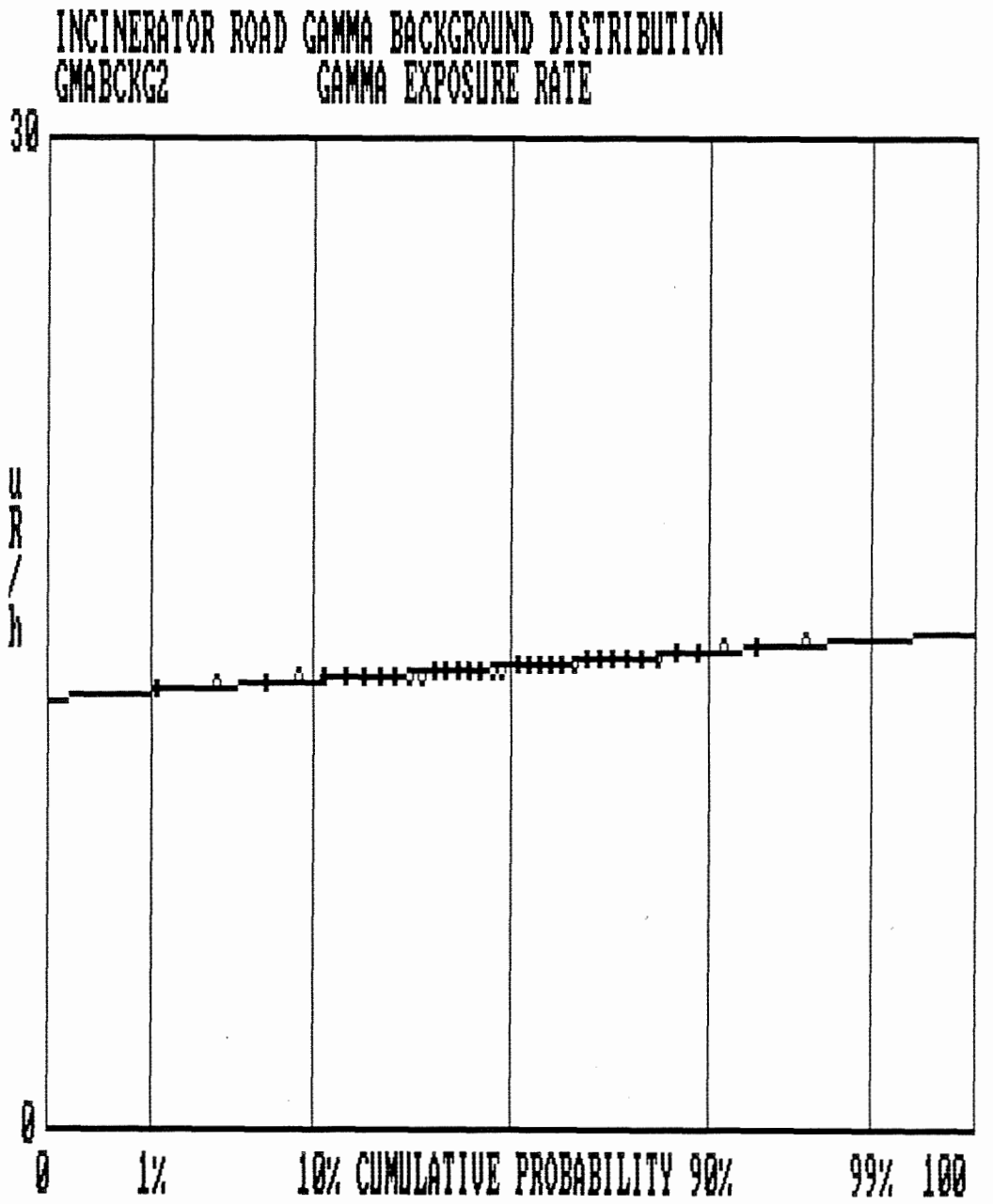
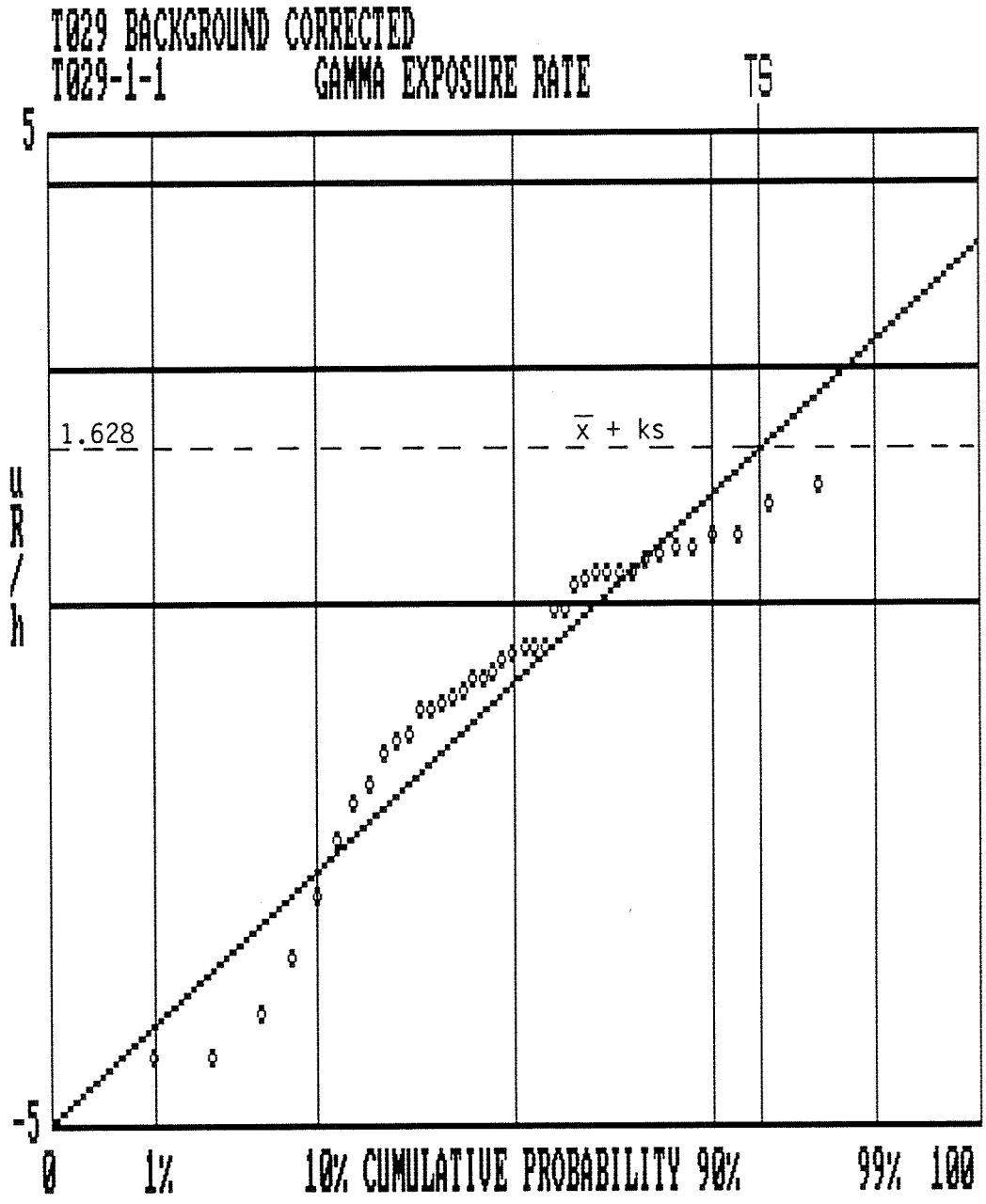


Figure 7.5 Ambient Gamma Radiation at Building T029
(Corrected for Background)



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 equivocally; "background" was treated uniformly. Fortunately, the test statistic accounts for this observed dispersion. The average value of this "background" corrected distribution is -0.84 $\pm$ 1.55 μR/h. The Test Statistic ($\bar{x}$ + ks) is 1.628, which is less than our 50% characterization level. The area is found acceptably "clean" by this inspection measurement.

7.3 Radiation Measurements of Ra-226 Storage Wells

The 12-in diameter pipe with three 1-in diameter tubes are still in place. The tubes produce no measurable gamma exposure rate. A thimble was raised from a tube and surveyed with an alpha probe. 200 cpm was registered by the meter. This corresponds to about 2800 α-dpm/100 cm^2. At the discretion of the health physicist, the thimble was lowered back in position and the survey terminated. It is known that the source wells are still contaminated. An adequate, thorough investigation will be possible when the wells are removed and dispositioned. This work will require supervision by a health physicist and controls to minimize spread of contamination during this operation.

7.4 Radiation Measurements of Old Barrel Storage Area (Figure 2.5)

Gamma exposure rate measurements and beta countrate measurements were made "for indication only" in a small area where barrels were once stored temporarily in the early 1960s (see Figure 2.5). We do not know what was stored in them; we suspect non-radioactive material. All measurements show no detectable activity.

8.0 CONCLUSIONS

Building T029 and its surrounding area were inspected for radioactive contaminants. Gamma exposure rate measurements show that no residual contamination exists on the facility floor or in the surrounding area. The Gaussian plots of T029 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. 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 subtracting a value best representing natural ambient gamma radiation in this area, we conclude through inspection by variables, that the area is clean of any residual radioactive contamination.

The Ra-226 storage wells inside the facility are still contaminated. An alpha survey of a thimble raised from the bottom of a well showed 2800 α-dpm/100 cm^2 total contamination. Further investigation, decontamination, and disposition is required in these wells. Controls should be applied to minimize contamination spread during this procedure.

An area south of T029 where, from an old photo, was shown to be used as some type of barrel storage yard, was also surveyed. All measurements show No Detectable Activity.

9.0 REFERENCES

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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.

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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. IL from D. D. Busick and W. D. Hanson to R. M. Hill, "Report of Radioactive Contamination Incident of the Radiation Measurements Facility - Building 029 - March 24, 1964", North American Aviation, Inc., April 10, 1964

20. IL from R. K. Owen and J. D. Moore to W. F. Heine, "Incident Report - Sealed Source Capsule Failure at T029", North American Rockwell, December 2, 1970

21. IL from J. D. Moore to W. F. Heine, "Transfer of Radioactive Sources from T029", Rockwell International, May 1, 1974

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 43-5 alpha scintillation probe to measure alpha contamination. The probe active area is 50 cm^2. Instrument calibration is performed using Th-230. This instrument is best suited for "indication only" determinations.

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 only" 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.

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.

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

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

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

E.2 Long-Term Management

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

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

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

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

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

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F. EXCEPTIONS

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

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

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

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

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

e. Where there is no feasible remedial action.

G. SOURCES

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

H. REFERENCES

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

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

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

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. BUILDING T029 RADIOLOGICAL SURVEY DATA

Data Sorted by Location

T029-1.WS

ROOM NUMBER	GRID NAME	uR/h TOTAL	uR/h STD DEV
	2-1	14.48	0.26
	2-1	13.85	0.25
	3-1	14.30	0.26
	3-2	14.14	0.26
	4-3	14.54	0.26
	5-3	13.89	0.25
	5-4	14.20	0.26
	6-5	14.78	0.26
	6-6	14.83	0.26
	6-7	14.79	0.26
	5-8	14.50	0.26
	5-9	14.70	0.26
	5-10	14.37	0.26
	5-11	13.39	0.25
	5-11	15.21	0.27
	5-12	14.19	0.26
	5-13	15.23	0.27
	5-13	16.50	0.28
	4-15	16.00	0.27
	5-15	15.71	0.27
	6-15	15.50	0.27
	7-14	15.60	0.27
	7-12	15.61	0.27
	7-12	15.59	0.27
	7-11	15.50	0.27
	7-10	15.85	0.27
	7-9	15.60	0.27
	7-8	15.84	0.27
	7-7	16.32	0.27
	7-6	15.80	0.27
	7-5	16.02	0.27
	6-11	13.15	0.25
	6-12	13.68	0.25
	6-13	14.81	0.26
INSIDE	5-11	12.17	0.24
INSIDE	5-12	10.51	0.22
INSIDE	5-12	10.95	0.23
INSIDE	5-12	12.77	0.24
INSIDE	5-12	10.45	0.22
INSIDE	5-11	11.52	0.23

Data Sorted by Exposure Rate

T029-1.WS

ROOM NUMBER	GRID NAME	uR/h TOTAL	STD DEV
	5-13	16.50	0.28
	7-7	16.32	0.27
	7-5	16.02	0.27
	4-15	16.00	0.27
	7-10	15.85	0.27
	7-8	15.84	0.27
	7-6	15.80	0.27
	5-15	15.71	0.27
	7-12	15.61	0.27
	7-9	15.60	0.27
	7-14	15.60	0.27
	7-12	15.59	0.27
	7-11	15.50	0.27
	6-15	15.50	0.27
	5-13	15.23	0.27
	5-11	15.21	0.27
	6-6	14.83	0.26
	6-13	14.81	0.26
	6-7	14.79	0.26
	6-5	14.78	0.26
	5-9	14.70	0.26
	4-3	14.54	0.26
	5-8	14.50	0.26
	2-1	14.48	0.26
	5-10	14.37	0.26
	3-1	14.30	0.26
	5-4	14.20	0.26
	5-12	14.19	0.26
	3-2	14.14	0.26
	5-3	13.89	0.25
	2-1	13.85	0.25
	6-12	13.68	0.25
	5-11	13.39	0.25
	6-11	13.15	0.25
INSIDE	5-12	12.77	0.24
INSIDE	5-11	12.17	0.24
INSIDE	5-11	11.52	0.23
INSIDE	5-12	10.95	0.23
INSIDE	5-12	10.51	0.22
INSIDE	5-12	10.45	0.22

APPENDIX D COPY OF INTERNAL LETTER

"Report of Radioactive Contamination Incident of the Radiation Measurements Facility Building 029 - March 24, 1964"

INTERNAL LETTER

NORTH AMERICAN AVIATION, INC.

DATE April 10, 1964

TO E. M. Hill
ADDRESS 779-21
Building 040,
Santa Susana

FROM D. D. Busick and W. D. Hanson
ADDRESS 779-21
Building 040,
Santa Susana
PHONE 6301

SUBJECT Report of Radioactive Contamination Incident of the Radiation Measurements Facility - Building 029 - March 24, 1964

Summary

At 1045 hours, March 24, 1964, a call was received at Building 040, (Health and Safety) from personnel in Building 029, reporting a probable rupture of one of the Ra^{226} calibration sources. This source is one of three containing 24.78 mgm of Ra^{226} as a bromide salt. The other sources contain 132 and 930 mgms of Ra^{226} also in the bromide chemical form. All three sources are incapsulated in platinum iridium alloy of 1.0 mm wall thickness except for the 930 mgm source which has a wall thickness of 1.5 mm of platinum alloy.

Follow-up investigation revealed that three Instrumentation Applications unit personnel were contaminated with alpha activity, primarily on their hands - 1000 c/m (2500 d/m) maximum as determined by a portable alpha survey meter. The personnel were evacuated to Building 020 hot change room for decontamination. Cleaning was easily accomplished after the first soap and water treatment.

Bioassay specimens were collected, submitted and proved negative for radium content.

The facility was surveyed thoroughly for removable and fixed radioactive contamination. What was found, was largely confined to a small area surrounding the source storage wells, along with some low level removable alpha contamination elsewhere in the building. Decontamination of the facility reduced fixed and removable contamination below detectible levels with one exception, the source storage well at the floor level was (6 d/m/100cm² est).

The offending source was believed to be at the bottom of its storage well. Efforts to locate it elsewhere in the facility were negative. The α activity at the top of the storage well was approximately 1.5 α/hr, indicating that the source was in fact, at the bottom. (North well). A high volume air sample showed normal background alpha activity, 2.5×10^{-13} uc/cm³.

Building 382, the instrument shop, was also surveyed and found to be free of removable and fixed alpha and beta-gamma contamination.

To: R. M. Hill
From: D. D. Busick and W. D. Hanson
Subject: Report of Radioactive Contamination Incident of the Radiation Measurements Facility - Building 029 - March 24, 1964

On March 25, 1964, the offending source was recovered from its source well, apparently intact, by Health and Safety personnel. The source was placed in its lead shipping container to await disposition after first being placed in a sealed stainless steel tube to contain Rn^{222} generated by the decay of Ra^{226}.

The source evidently is not leaking badly since smears are reasonably free of removable alpha contamination of Rn^{222} and other short lived daughters of Ra^{226}. For this reason, the magnitude of the incident was limited. Surface and personnel contamination (internal and external) would have been several orders of magnitude greater had the source ruptured to the extent that the $RaBr_2$ had been released.

Sequence of Events

Monday afternoon, March 23, 1964, D. E. Van Dyke, Department 744-42, Clock #53 and J. W. Dodd, Jr., Department 744-42, Clock #9, were in the process of calibrating radiation survey instruments. This work was being accomplished at the Radiation Measurements Facility (RMF), Building 029, Santa Susana. At the present time, the facility is equipped with three radium sources, 24.78, 132, and 930 mgms, which are stored inside individual "wells". Each well is approximately 9.5 feet deep by 1 inch diameter. The sources are raised and lowered using nylon string. Over each well is pyrex tube 10 feet long, which is used to prevent lateral motion of the individual source during calibration. The two smaller sources are inside plastic capsules which are attached to the nylon string. The larger source is in a metal source holder.

At approximately 1500, March 23, 1964, the plastic capsule which contained the 24.78 mg source, became lodged and fractured allowing the source to fall an estimated 13 feet to the bottom of the well. Since the primary container for the source is made of platinum, Van Dyke and Dodd did not feel a hazard existed, and did not notify either Health and Safety nor the Control Center. No attempt was made at that time to recover the source.

The next day, March 24, 1964, K. W. Sloscomb, Department 744-42, Clock #47 and D. J. Dunlavy, Department 744-42, Clock #10, accompanied Van Dyke to the facility to assist in the recovery of the source. A 12 foot long piece of copper tubing, with a piece of tuck tape on the end, was inserted inside the well in an attempt to retrieve the source. After several futile attempts one of the employees discovered that the tape on the end of the tubing indicated 1.5 mrad/hr according to a Nuclear Chicago 27 survey meter which they were using to monitor the operation. At that time they became concerned that the source had ruptured and notified Health and Safety at Building 040 at 1045.

W. D. Hanson from Health and Safety was instructed to evaluate the problem and monitor the personnel involved. Another Nuclear Chicago 27 survey instrument was used to monitor the employees indicating only

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background radiation. Smears were taken inside to determine the magnitude and extent of the R/A contamination. The smear nearest the well in question showed 600 d/m $\beta\gamma$ and 200 d/m α. An Eberline portable monitor PAC-3G was borrowed from RMDF to determine if additional alpha contamination was present. All three employees showed up to 1000 c/m (2500 d/m α) on their hands. Shoes were contaminated up to 100 c/m (250 d/m α). D. D. Busick of Health and Safety was notified of the personnel contamination and was requested to send protective clothing for the three employees, sufficient to allow transportation to another area for decontamination and more extensive monitoring. Due to the implications of a ruptured radium source, Mr. Busick classed this condition as an emergency and requested assistance from Traffic and Industrial Security. The personnel in question were taken to the CDHC for decontamination. Nasal smears were taken on the suspects indicating $<$2d/m alpha and $<$30d/m $\beta\gamma$. The personnel were thoroughly monitored, and successfully decontaminated including all personal effects. They were then requested to submit a 24 hour bioassay with one sample to be submitted for immediate analysis. J. W. Dodd was also included in the bioassay since he was present when the source was dropped. He was also monitored for clothing, skin, and nasal contamination. Nothing above background was detected.

While the decontamination of personnel was progressing, other members of the Health and Safety Emergency Team were making a more extensive survey of Building 029. An air sample taken showed 2.5×10^{-11} uc/cc alpha which is normal for this site. Smear results of Building 029 showed almost all of the contamination to be concentrated around the well with a maximum of 90 d/m/100cm^2 $\beta\gamma$ and 120 d/m/100cm^2 α. A smear survey of Building 383 and also the pickup truck used by Department 744 showed nothing above background.

A portion of the broken plastic capsule and a piece of tuck tape that had been used in attempt to retrieve the source were taken to Building 028 for analysis on the multi-channel analyzer. The gamma scan demonstrated the presence of the daughters of Ra^{226}.

Decontamination of the floor of Building 029 was accomplished by Maintenance, and restrictions for entry were lifted at 1545, March 24, 1964. The well was plugged with a rubber stopper. A 2500 cc ion chamber was used to withdraw a sample of atmosphere from the well after a period of 24 hours. The sample was taken inside the well 5 feet from the top. Analysis showed 6.7×10^{-4} uc/cc of Rn^{222} as determined by Health and Safety Laboratory Unit. Rn^{222} has an eight hour MPC of 1×10^{-8} uc/cc, a half-life of 3.8 days and is the first radioactive daughter of Ra^{226}. The total volume of Building 029 is $\sim 3 \times 10^8$ cm^3. The concentration of radon in the building from the accumulated radon in the source well will raise the airborn alpha activity from $\sim 2 \times 10^{-11}$ uc/cm^3 to 1.7×10^{-9} uc/cm^3.

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The exhaust system for Building 029 is rated at 1540 ft^3/min producing 8.5 air-changes-per hour in the building. The atmosphere is exhausted from the building through two absolute filters and dumped immediately outside. This many air-changes-per hour will clear the building of radon in the above stated concentration in approximately 40 minutes.

The recovery of the source was accomplished at 1600 on March 26, 1964. This was done after a light was lowered into the well and a visual observation thru a telescope proved the source to be in one piece. The source removal was accomplished by dropping a small weight with a piece of tuck tape attached to one end.

The source was placed inside a lead shielded container. The equipment used was assumed to be contaminated, and after decontamination, ~50 smears taken showed no removable contamination above background. The offending source has been sealed in a 3 inch length of stainless steel pipe 1/2" in diameter to prevent the continuous release of Rn^{222} from contaminating the immediate environment.

Discussion and Recommendations

On or about September 2, 1961, the intermediate range radium source (132 mgm) plastic source holder was dropped in a similar manner from the top of its source well. This incident did not result in a release of radioactive material. However, like the incident of March 25, 1964, an immediate report was not made to Health and Safety, Industrial Security or to facility supervision. The situation was discovered 10 days following the 1961 incident. While the most recent incident was reported within eighteen hours after the source was dropped, an immediate report would have limited the emergency response. Also, we do not know if the source was leaking before or as the result of the initial recovery attempts.

The total body burden for Ra^{226} is 0.1 ug. The 24.78 mg source represents ~250,000 body burdens. The ingestion and subsequent tissue fixation of 4×10^{-4}% of the radium content of this capsule could result in an employee sustaining a single body burden of radium. Bone tumors have been reported to develop with 0.5 ug of radium fixed in bony tissue.

Radium has chemical properties very similar to barium. Halogen salts of barium and radium are extremely soluble in body fluids. The solubilities of $BaBr_2$ and $BaCO_4$ are 104 and 2.4×10^{-4} gm/100 gm of H_2O. Radium salts having similar chemical properties also will exhibit similar solubility properties. The halogen salts of radium then are more hazardous by many orders of magnitude than the sulphate radium compounds. Therefore, a very large portion of $RaBr_2$ ingested by personnel would be fixed in body tissue.

The personnel involved treated this incident with a great deal of casualness. Also, since a similar type of incident occurred on at least

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one other occasion by different personnel of the same unit, a greater supervisory effort by Department 744 seems in order. Although similar radium sources are handled safely every day in many places by many different people, unusual or atypical situations must be reported to operations supervision and Health and Safety immediately. The consequences of a single mistake resulting in the release of radium to uncontrolled environments are both costly and extremely hazardous to personnel.

The need for additional and continuing training of both supervisory and non-supervisory personnel is clear. These people are well indoctrinated in the control of external radiation problems. However, an understanding of the more serious problem of ingestion seems to be lacking.

Written procedures for use of this facility has been submitted to Health and Safety for approval. No recommendations pending the review of these procedures will be made at this time.

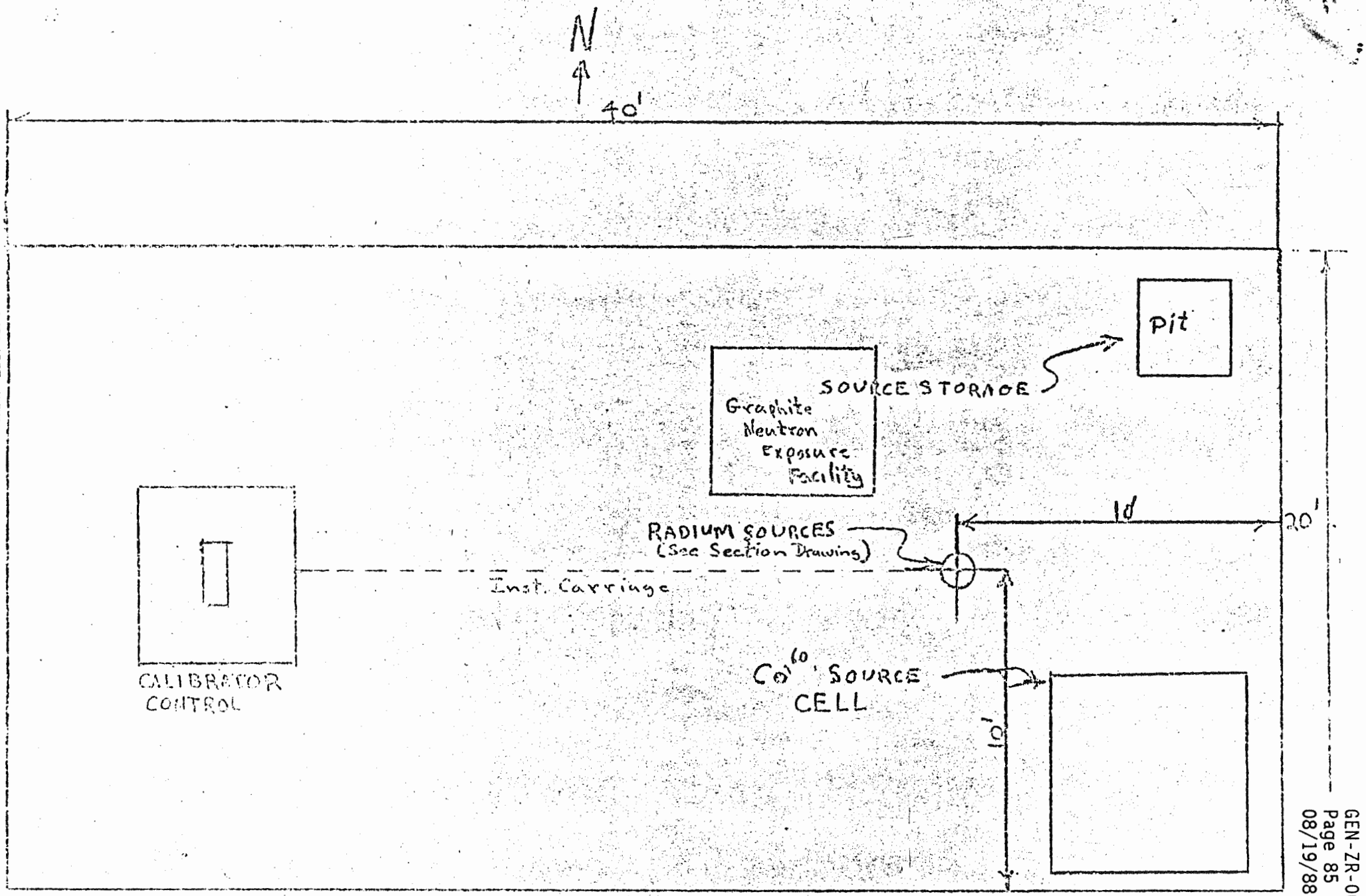
Finally we would recommend that the radium sources be replaced with Co-60 or Cs-137. The maximum permissible body burdens are 100 and 300 times that of Ra-226. Encapsulation problems are also minimal and have a long history of use under severe conditions with a correspondingly low incident of failure.

D. D. Busick
Health and Safety Assistant
Santa Susana Operations
Health and Safety Section

W. D. Hanson
Associate - Health Physics
Santa Susana Operations
Health and Safety Section

DDB:WDH:ap

cc: J. C. Lang
M. E. Healey
D. K. Rhoemler
E. O. Anderson
S. L. Jones



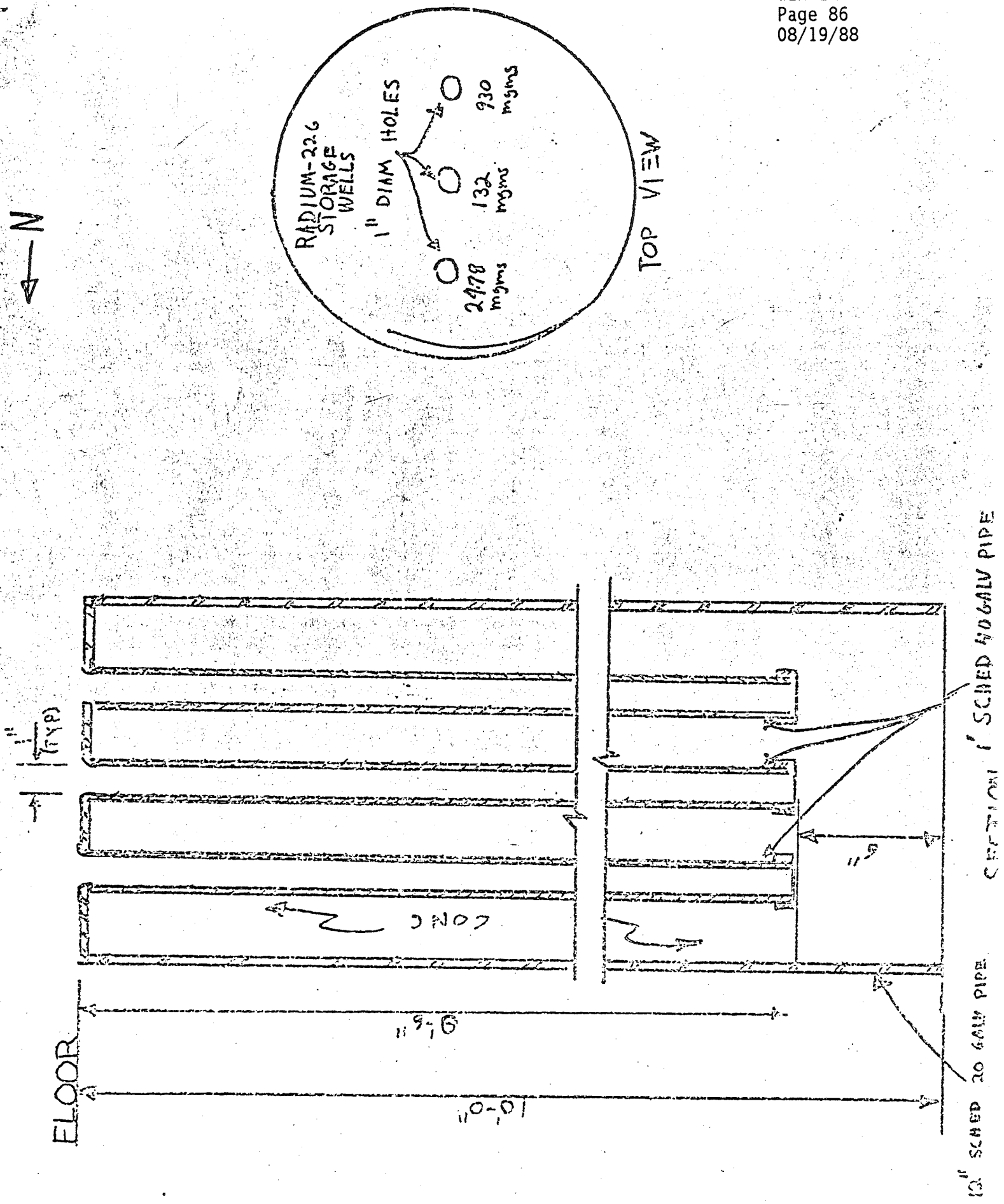
B1029

FLOOR PLAN

1/4" = 1'-0"

5

Personnel
Door


N
RADIUM-226
STORAGE
WELLS
1" DIAM HOLES
930 mgms
132 mgms
2478 mgms
TOP VIEW
CONC
FLOOR
11'-9"
11'-0"
SCHED 40 6" PIPE
SCHED 40 6" PIPE

APPENDIX E COPY OF INTERNAL LETTER

"Incident Report - Sealed Source Capsule Failure at T029, November 20, 1970"

Internal Letter — **North American Rockwell**

Date	December 2, 1970	**No.**	
TO	W. F. Heine	**FROM**	R. K. Owen/J. D. Moore
Address	779-021 NB13	**Address**	779-021 T059 / 779-021 NB13
		Phone	6107 / 1329

Subject Incident Report - Sealed Source Capsule Failure at T029

On November 20, 1970, at approximately 1330, a triply encapsulated sealed source containing 4.6 Ci of Cs^{137}, as a Cs-salt, which is used at T029 for calibration of radiation survey meters, failed during withdrawal of the source from storage well #2. The failure occurred at the brazed joint at the tertiary capsule bottom plate, releasing the double-walled inner source capsule into the 10-ft deep source well. The failure occurred as the ascending source capsule entered a glass guide tube positioned over the storage well. A slight misalignment of the guide tube over the storage well may have prevented a smooth transition of the capsule into the tube, resulting in mechanical shock to the source capsule. The capsule bottom closure braze was evidently badly deteriorated and tertiary capsule failure was probably imminent regardless of any stress from use of the source.

At the time of the incident, the calibration facility and the Cs^{137} source were being used in accordance with the requirements of Instrument Applications Unit Procedure 15-085, Revision B, which prescribes the procedures for use of the facility. Personnel present at the facility when the source failure occurred were Q. W. Koon, D/737, S/N 351008, who is custodian of the source, and R. K. Owen, D/779, S/N 198641.

An unsuccessful effort was made to retrieve the source by placing the adhesive side of green tape outward on a length of copper tube and inserting the tube into the storage well. A smear survey of the tertiary container, which remained attached to the source withdrawal mechanism, and the area adjacent to the storage wells indicated no radioactive contamination in excess of 50 d/m/100 cm^2.

The source, which was purchased from Western Radiation Laboratory, was delivered to AI on October 23, 1963. A source description and calibration certificate delivered therewith states that the source consists of an ORNL, double-walled stainless steel inner capsule, encased in a heavy-walled 416 stainless steel outer capsule which is closed by brazing. The latter containment and brazing were performed at Western Radiation Laboratories. The vendor source leak test certificate and the subsequent semi-annual leak tests performed at AI have all shown less than 0.005 uCi of removable contamination. It should be noted that a second Cs^{137} source, identical to the subject source, and received on the same date, is currently in storage in HQ-131 and is available for use at

T029 as a replacement for the failed source. This source should be inspected for containment integrity prior to use.

Possible source recovery methods now being studied include use of tygon tube connected to a vacuum source, and use of an adhesive at the end of a rod. It is recommended that the source be inspected "in situ" with a borescope prior to further recovery attempts to determine its position and to detect any possible capsule damage. This will also help to select the best recovery method.

Health physics aspects of the source recovery, assuming no gross primary capsule failure with widespread contamination, are limited to the source external radiation level, which is estimated at approximately 16 R/hour at 1 foot. The use of remote handling equipment and short exposure times should minimize personnel radiation exposures. The original source safe is available in the facility to receive the source when recovered.

It is recommended that the source be disposed of since it is at least 16 years old, may have been damaged, and probably cannot be safely again encapsulated in a tertiary container.

R. K. Owen
Associate Health Physics
Operational Safety Unit

J. D. Moore
Health & Safety Representative
Operational Safety Unit

cc:			
	J. H. Ecroyd	052	SS12
	E. L. Gardner	737-073	NB14
	R. V. Inman	737-073	T027
	M. E. Remley	779-020	NA06
	V. J. Schaubert	785	KB09
	J. F. Trevillyan	051	LA24

Internal Letter

North American Rockwell

Date December 9, 1970

No.

TO W. F. Heine

Address 779-021
NB13

FROM J. D. Moore

Address 779-021
NB13

Phone 1329

Subject Recovery of Sealed Source Capsule at T029

REF: IL - Incident Report - Sealed Source Capsule Failure at T029, dated December 2, 1970

On December 4, 1970, the 4.6 Ci Cs^{137} source which had failed as reported in the referenced document, was recovered from well #2 by means of adhesive tape attached to the end of a rod.

The source was recovered intact with no apparent damage to the secondary containment. A leak test performed immediately following recovery indicated less than 0.005 uCi of removable contamination. The capsule was placed in a suitable cask for interim storage pending a decision as to its final disposition. No personnel involved in the recovery effort received significant radiation exposure. A smear survey of the area surrounding the source storage well indicated surface contamination levels of less than 50 dpm/100 cm^2.

J. D. Moore
Health & Safety Representative
Operational Safety Unit

cc:

J. H. Ecroyd	052	SS12
E. L. Gardner	737-073	NB14
R. V. Inman	737-073	T027
B. I. Johnson	779-021	NB13
J. P. Klostermann	779-021	T020
Q. W. Koon	737-073	NB14
M. E. Remley	779-020	NA06
V. J. Schaubert	785	KB09
J. F. Trevillyan	797	LA24