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TITLE: "Radiological Survey of Building T009"

-APPROVALS-

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ABSTRACT

A radiological survey was performed at Building T009, located at Rockwell International's Santa Susana Field Laboratory (SSFL) Area IV, to clarify and identify those locations needing further radiological inspection or requiring remedial action. Building T009 was used in the 1960s and early 1970s for performing AEC-sponsored nuclear reactor critical experiments. Two reactor types were tested: the Sodium Graphite Reactor (SGR), and the Organic Moderated Reactor (OMR). SGR occupied the eastern high bay, OMR the western. Each high bay had its own set of designated support laboratories, change rooms, fuel storage vaults, and shops. The office area is common to both sides of the facility. No major contamination incidents are known to have occurred at either of these low power test reactors. Although some contamination incidents probably occurred, it has been common practice to decontaminate the affected area to its original "clean" condition. When these experimental reactor programs ended, all associated equipment was removed, radiation surveys were performed, and the facility was modified for other programs (some of which involved the use of radioactive material). The SGR side is currently used for In Service Inspection (ISI) work, and therefore was not covered by the scope of this survey. This radiological survey of the OMR side was performed to determine if any radioactive material has been accidentally left behind to such an extent that further surveying or decontamination is warranted.

The building interior (OMR side) was surveyed for total and removable alpha/beta radioactivity, and ambient gamma exposure rate. These interior measurements were made on 1 m^2 per 3-m square plot plan, consistent with an 11% sampling plan. Special building features were surveyed "for indication" of radioactivity. Additionally, samples of crud, grease, and sludge were collected from sink clean-outs, shower drains, machining equipment, and the SGR radioactive liquid hold-up tank and pit, and then analyzed for radioactivity by gamma spectrometry. Ambient gamma exposure rates were measured outside of T009, in an area northwest of the facility.

This area is partially paved and used as a storage yard, and partially natural-terrain in a location where the old sanitary leach field existed. Outdoor gamma measurements were made on a 6-m square plot plan.

Results of this survey, analysis, and interpretation show that a few locations inside the facility are slightly contaminated. The old fuel vault is contaminated with a maximum alpha activity of 92 dpm/100 cm^2 averaged over 1 m^2; the acceptable limit is 5000 dpm/100 cm^2. Corresponding increased beta measurements in those locations were not observed. Slight removable alpha activity was measured in those fuel vault locations and in a few cabinets in rooms 114 and 116, the counting room and laboratory, respectively. Maximum removable activity was 15 dpm/100 cm^2; the limit is 1000 dpm/100 cm^2. A sludge sample taken from inside the SGR radioactive liquid holdup tank was slightly contaminated with fission products, U-238, Th-232, and possibly U-235. No radioactive material was detected in other miscellaneous samples collected and this included those from all drain traps, and the pit which holds the SGR holdup tank. All alpha/beta surveys made on a uniform grid, and "for indication," and ambient gamma surveys pass acceptance criteria for unrestricted use by a wide margin. Statistical tests show that, based on this inspection plan, the likelihood of significant residual radioactivity existing in the fuel vault or in the cabinets is small. Further investigation is not required; the areas are far below any hazard level, and much less than unrestricted-use acceptance limits. Removal of the old (inactive) SGR radioactive material holdup tank, however, should be performed under the supervision of a health physicist. The tank will most likely be classified as low-level radioactive waste. At time of removal, drain lines should also be checked for radioactivity. The holdup tank is not leaking radioactive material to the surrounding pit and no hazard exists in its current configuration.

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1.0 INTRODUCTION

Building T009, located in Area IV of Rockwell International's Santa Susana Field Laboratories (SSFL) was inspected and analyzed for residual radioactive material. This facility supported AEC, ERDA, and DOE nuclear-related programs, including primarily the Sodium Graphite Reactor (SGR) and Organic Moderated Reactor (OMR). The eastern half of the building is commonly referred to as the SGR side, while the western half is referred to as the OMR side. Because the SGR side is currently in use for In Service Inspection (ISI) operations, it was not included with this survey. The OMR side was inspected for radioactivity to determine whether any radioactive material has been accidentally left behind and if further investigation is necessary or remedial action is required. This radiological survey was conducted as prescribed in the "Radiological Survey Plan for SSFL," (Reference 4, Section 5.4.20).

Located in Ventura County, California, Area IV of Rockwell International's SSFL has been used to develop and test nuclear reactors; fabricate nuclear reactor fuels; and disassemble irradiated nuclear fuel elements. Building T009 was designated the Critical Experiment Building; critical assemblies were made and tested for the Hallam Nuclear Power Facility (HNPF) and the Piqua Nuclear Power Facility (PNPF). In the early 1970s, these programs ended and the SGR and OMR were removed from the facility. The only systems currently still in place which supported these tests are two independent ventilation/exhaust/filter systems, and the SGR (east side) radioactive liquid holdup tank. These systems have not been in use since the early 1970s. The SGR ventilation system is known to be uncontaminated. Previous samples collected from the SGR tank showed radioactive material. Both the OMR ventilation system and SGR holdup tank were surveyed and analyzed for radioactivity. Except for sampling the SGR holdup tank and pit, and the sink trap in room 121 which leads to it, only the OMR side of the facility was inspected.

Enriched uranium metal was stored in the OMR fuel vault during its operation, but no loose radioactive material is known to have been used or stored on the OMR side. Possible residual contamination could result from an undetected release in the high bay test-area, fuel contamination in the vault, machining/milling operations performed on contaminated items in the machine shop, held-up sludge/debris in drain lines of sinks and showers, or residual material in the counting laboratory. The likelihood of significant contamination existing at T009 is small.

No known significant incidents occurred at T009 which would have released contamination to the inspected areas. Although some minor radiological contamination occurred, it was common practice to decontaminate and return an affected location to its natural condition. The purpose of this survey was to detect any radioactive material accidentally left behind from these operations.

As part of the DOE SSFL Site Survey (Reference 4) a radiation survey was performed in the OMR side and the outside northwestern area to determine if any residual contamination exists. Total and removable alpha/beta activity measurements and ambient gamma exposure rate measurements were made indoors on a 3-m by 3-m grid. Outdoor gamma exposure rate measurements were made on a 6-m by 6-m grid. If radioactive contamination was indicated during performance of the gamma measurements, samples were to be collected and analyzed for radioactivity, and beta surface activity measurements were to be performed. Sample collection because of increased gamma measurements was not required for this particular survey. Special facility features, including wall coving, miscellaneous horizontal surfaces, cracks, crevices, cabinets, sinks, drains, showers, and the OMR exhaust system and filter plenum were inspected for radioactivity. Smears were collected and analyzed for removable alpha/beta activity, and surveys "for indication" of total alpha/beta activity were performed. Additionally, samples were collected from sink traps, shower drains, and the SGR holdup tank for analysis of radioactivity by gamma spectrometry.

All total and removable alpha/beta contamination data and ambient gamma exposure rate data, were input into a Personal Computer (PC) graphics program which plots the radiation measurement value against its cumulative probability. The 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 density function. Inspection by variables techniques allows a thorough, understandable, and conclusive study for assessing the contamination level in an area. Samples collected for radioactivity analysis were analyzed "for indication." Miscellaneous alpha/beta surveys were also performed "for indication."

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 total and removable alpha/beta activity agree between standards. Limits for acceptable ambient gamma exposure rates differ between the DOE and NRC. DOE specifies 20 μR/h above background while NRC specifies 5 μR/h above background as acceptable gamma exposure rate limits. "Natural background" at SSFL is very difficult to determine because of a large observed variability in the measurements. Because of this large variation, total-gross gamma measurements made in a survey area are plotted and compared against three independent "natural" background distributions. If the average "background" exposure rate of the three "natural background" distributions (which is 15 μR/h) represents similar ambient conditions in a test-area, then this value is used to correct gross test-area values for background. If "ambient" conditions are dissimilar, then the best estimate for "ambient background" is the median value of gross-total measurements made in a sample lot. "Ambient" conditions are assessed on a sample lot by sample lot basis in order to compare gamma measurements against acceptance limits "above background." Two sample lots were established for analyzing the results of this survey -- an indoor lot and an outdoor lot. "Ambient background" varies between these sample lots.

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2.0 IDENTIFICATION OF FACILITY PREMISES

2.1 Location

Building T009 is located within Rockwell International's Santa Susana Field Laboratory (SSFL) in the Simi Hills of southeastern Ventura County, California. The site is adjacent to the Los Angeles County line, and approximately 29 miles northwest of downtown Los Angeles. The SSFL location relative to the Los Angeles area and surrounding vicinity is shown in Figure 2.1. An enlarged map of neighboring SSFL communities is presented in Figure 2.2. Building T009 is located in the western portion of SSFL, which is referred to as Area IV. Figure 2.3 is a plot plan of Area IV showing the location of Building T009. The building is owned by Rockwell and is located at the end of "G" Street just beyond 24th Street in the far western area of SSFL. The nearest occupied structure is Building T100, about 200 ft northeast. Building T009 is not within the 90.26-acre Government-Optioned Area. Photographs of T009 are shown in Figures 2.4 and 2.5; Building T100 is indicated in the photos for reference only.

2.2 Local Area Topography and Facility Characteristics

Building T009 sits on an irregular plateau in a mountainous area of recent geological age sprinkled with outcroppings above the more level patches. Outcroppings of Chico sandstone formation are numerous behind (northwest of) the facility. Surrounding the facility about 100 ft in all directions is asphalt concrete paving. This surrounding pavement is used for storing components and as access to the facility from the northeast (off "G" Street). The minimum distance from T009 to the SSFL boundary is approximately 650 ft. This boundary lies in a northwesterly direction (Simi Valley direction). Grade floor elevations are approximately 1835 ft above sea level, (Reference 25).

The general slope of Area IV, including these facilities and surrounding areas, is in an easterly direction. Water runoff is directed to

Figure 2.1 Map of Los Angeles Area

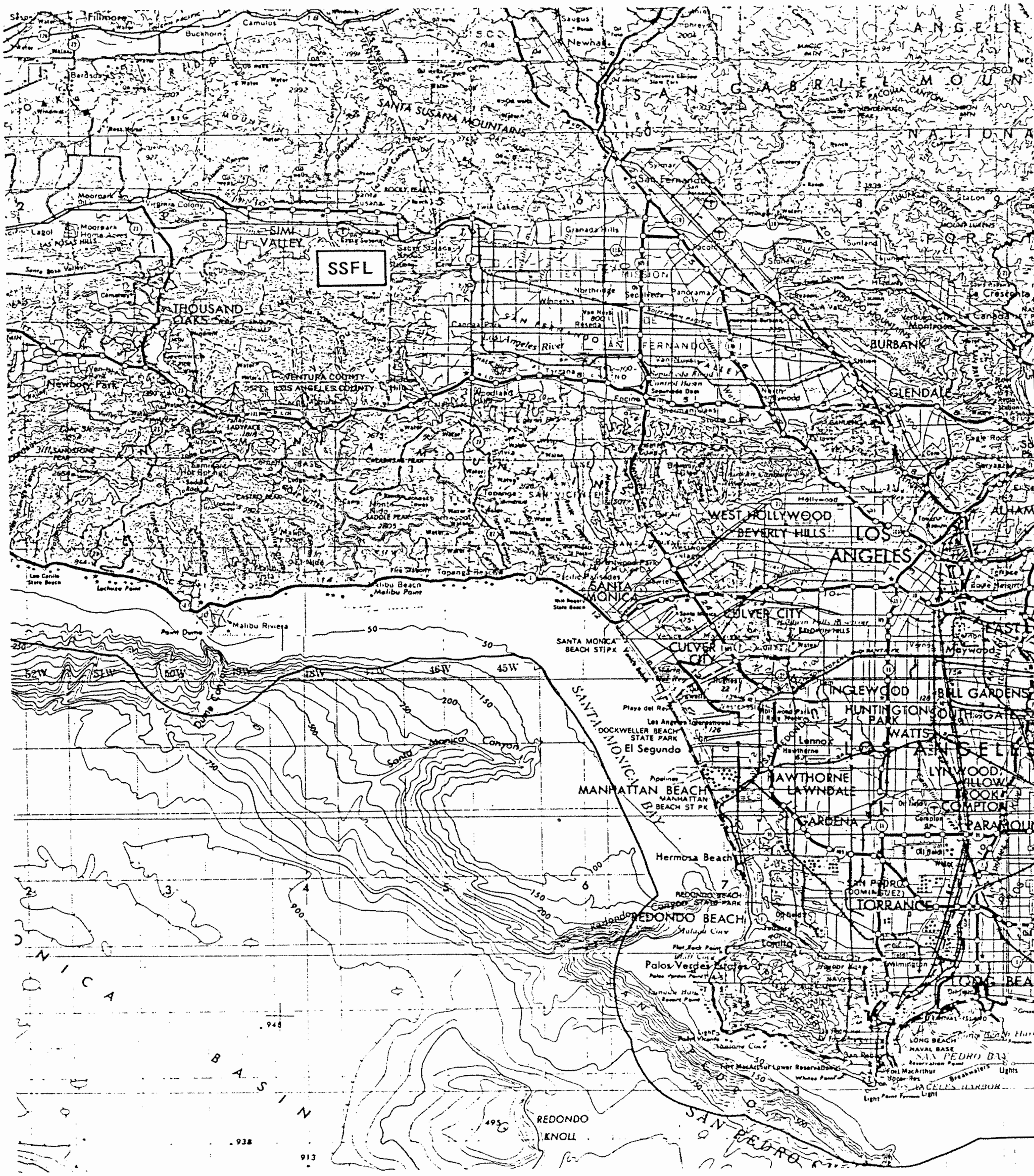


Figure 2.2 Map of Neighboring SSFL Communities

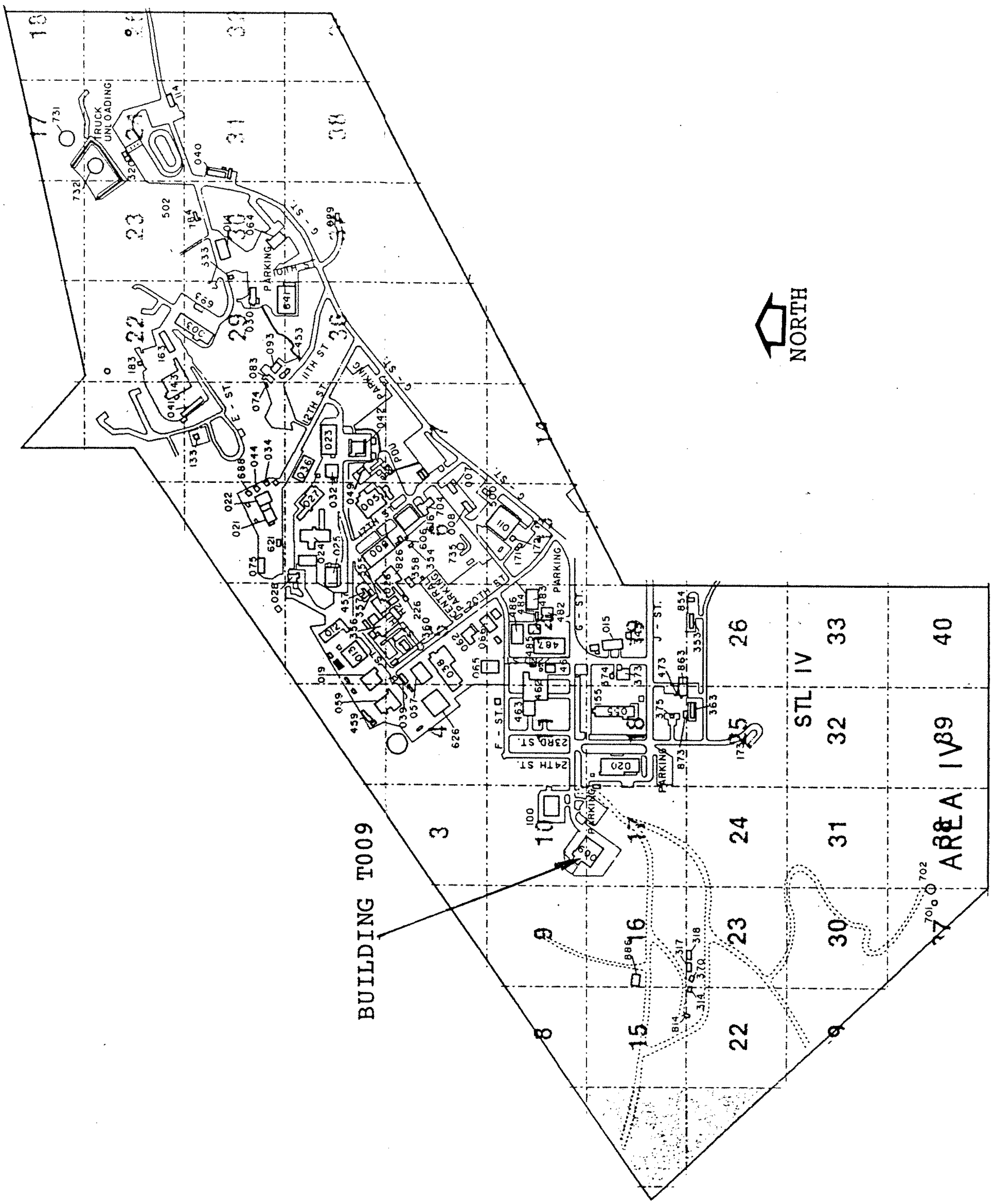


Figure 2.3 SSFL Layout, Showing Location of Building T009

Figure 2.4 Aerial Photo of SSFL Area IV Showing Building T009

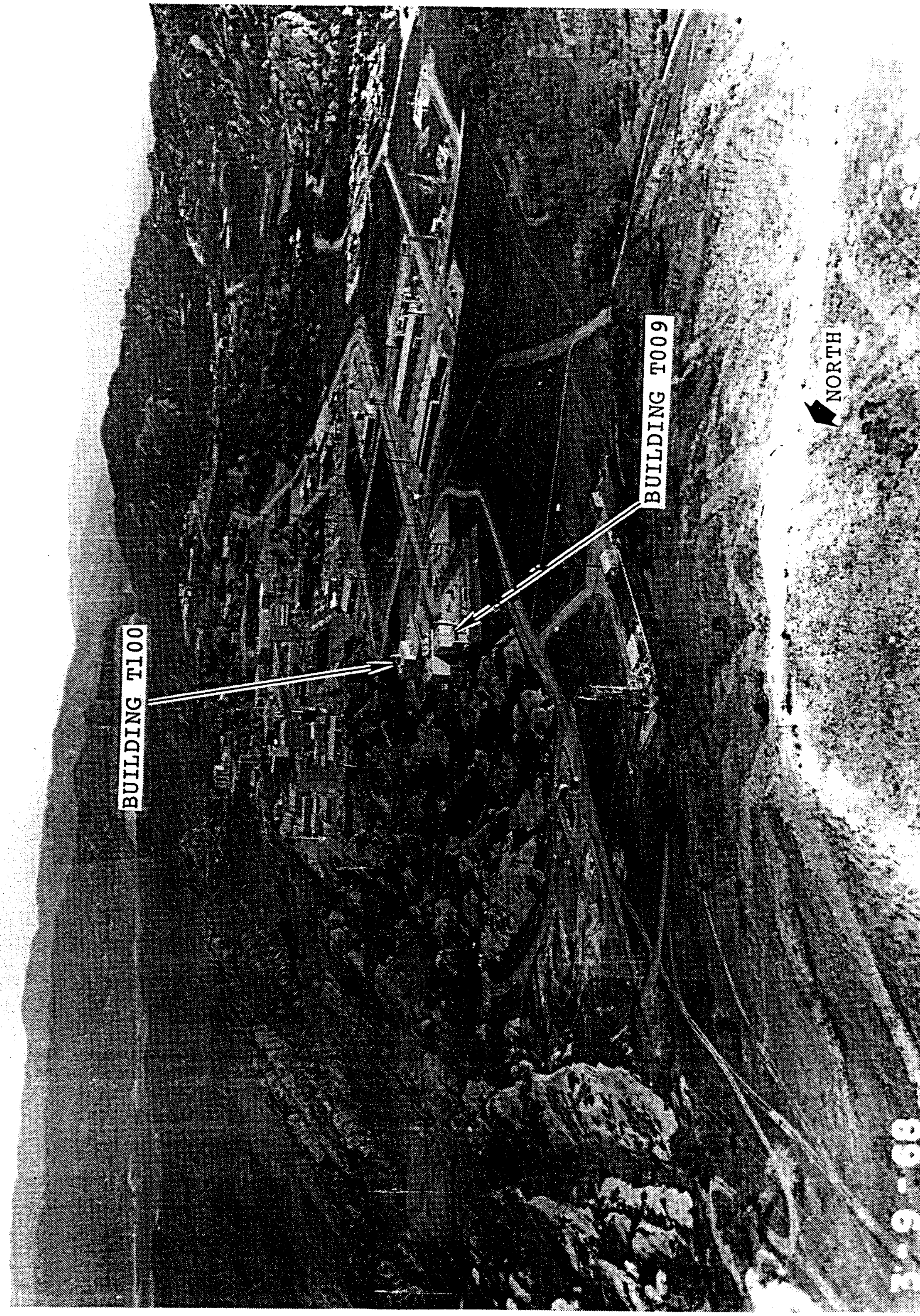


Figure 2.5 Photograph of Building T009 Looking Northeast

the retention reservoirs which are part of the SSFL industrial effluent control system. Liquid effluent discharge from the final retention pond into the Bell Canyon drainage occurs only after controlled effluent hold-up and sampling. Figure 2.6 is a topographic map of T009 and the surrounding area.

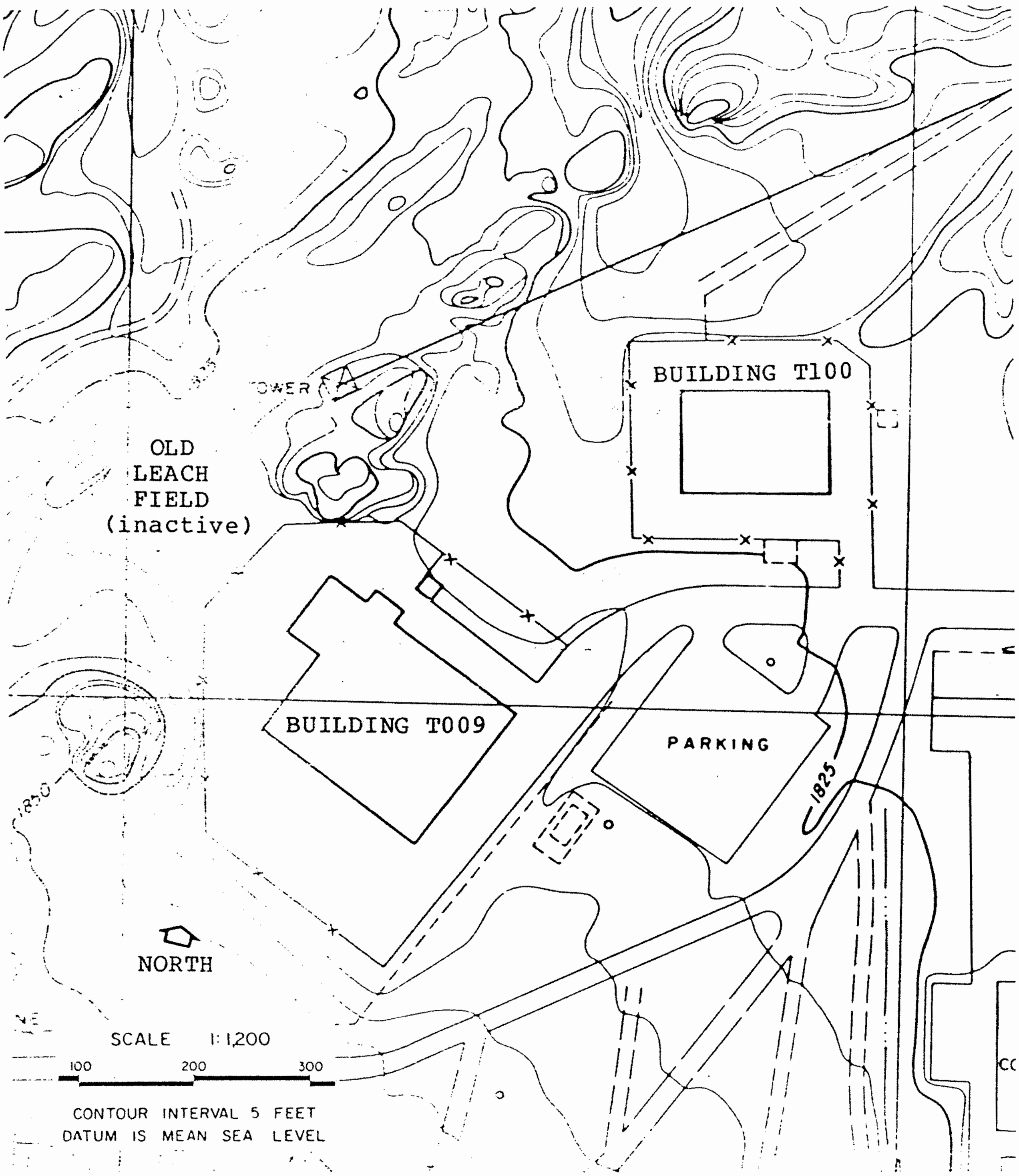
Because the facility was built to accommodate two independent critical-assembly test laboratories, facility characteristics are described separately for each of the two areas: The Sodium Graphite Reactor (SGR) side and the Organic Moderated Reactor (OMR) side. As shown in Figures 2.7 and 2.8, SGR occupied the east side of the facility, OMR the west side, (Reference 25).

2.2.1 Facility Characteristics of the SGR Side

Although the SGR experiment has ended and the reactor has been removed, most of the facility remains essentially unchanged. This facility description is from Reference 25, which was written in 1962. The SGR side of T009 (see Figure 2.8) consists of a high-bay building, which housed the critical assembly cell and a fuel-and-graphite-storage area, and an adjoining low-bay area which housed the control room, offices, and miscellaneous supporting laboratories. The high bay is a concrete structure approximately 70 ft long by 40 ft wide, with a 4-in. thick reinforced concrete roof deck on steel framing, with an eave height of 39 ft. A concrete block penthouse, which housed the critical assembly control rod drive mechanisms, is located on the roof over the critical assembly cell. A 5-ton capacity overhead crane runs north and south over the entire high-bay area, to service both the critical cell and storage area.

The critical assembly cell has dimensions of 36 by 36 ft, and a floor-to-ceiling height of 46 ft, with the floor 10 ft below grade level. A 10-ft deep hexagonal pit, 14 ft across the flats, is located in the center of the critical cell floor and provided access to the underside of the critical assembly. A manhole and passageway leads to the pit. The three

Figure 2.6 Topographic Map of Building T009 and Surrounding Area



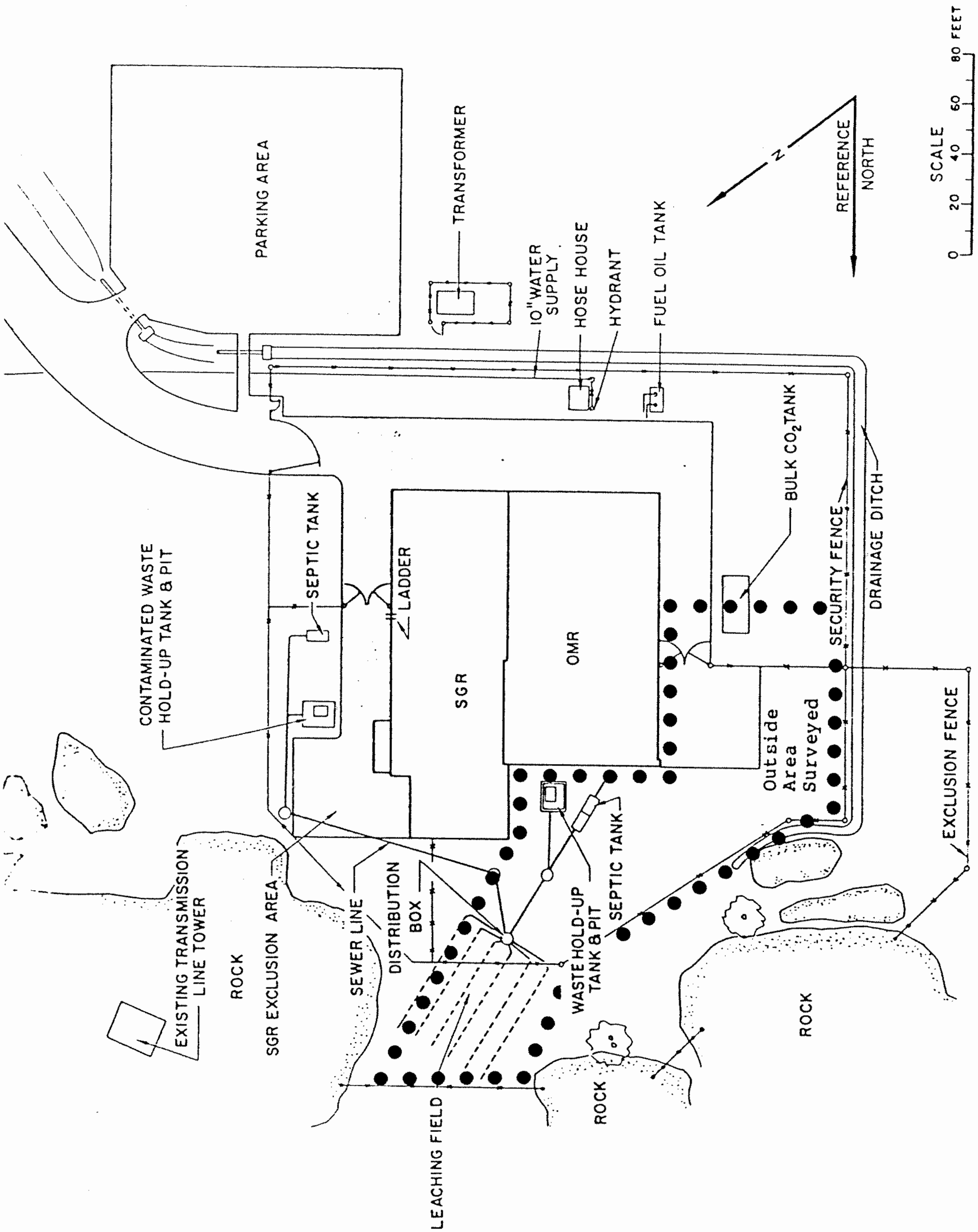


Figure 2.7 Plot Plan of Building T009, Showing SGR and OMR Test Rooms

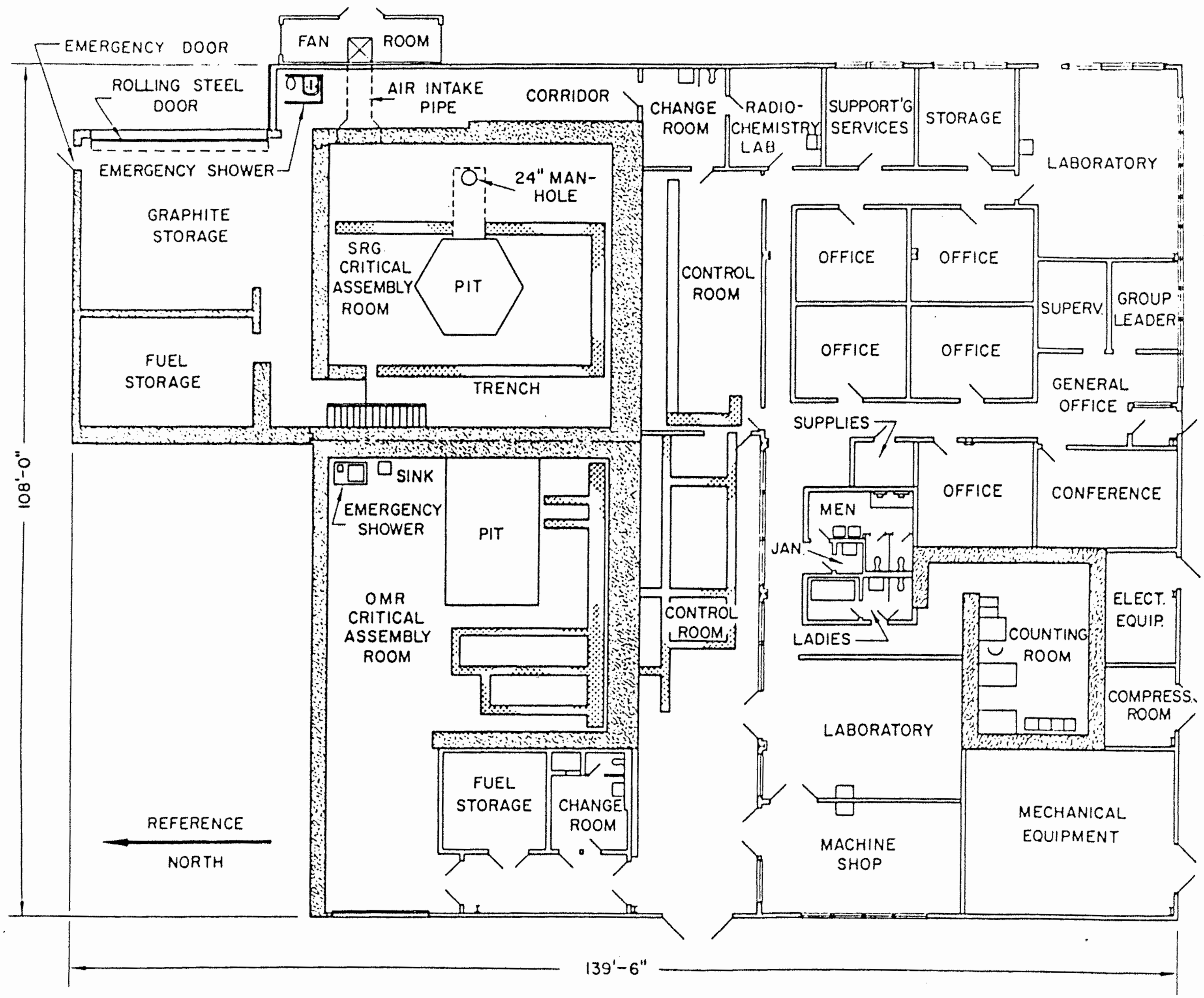


Figure 2.8 Floor Plan of Building T009

sides of the critical cell away from the storage area are shielded by poured-concrete walls extending to the roof. A service door is provided on the northwest corner for truck entry to the graphite storage area. The fuel storage area was 14 by 22 ft, and the graphite storage area was 29 by 24 ft. These areas were separated by 2-ft thick concrete walls 18 feet high. A section of the 2-ft thick wall between the critical cell and storage areas consisted of removable concrete tiers to facilitate handling of critical assembly materials. A 10-ft clearance between the top of this wall and the ceiling was provided for crane travel. The floors of these areas are at grade level and the ceiling height is 35 ft. The OMR Critical Facility assembly room is adjacent to the west wall of the SGR Critical Facility assembly room (common shield wall, 4 ft thick).

The low-bay building is a single-story structure attached to the south side of the SGR and OMR high-bay areas. The building is a conventional steel frame structure with insulated sheet metal siding and interior walls of metal-lath and plaster. The roof consists of gravel and tar on rigid insulation over metal decking. The entire building is 108 ft long and 70 ft wide, with the west half occupied and used by the OMR project and the east half by the SGR project. The portion that was used by the SGR operations personnel contained a change room, control room, radiochemistry laboratory, general laboratory area, offices, and miscellaneous storage facility support areas. The change room was located at the entrance of a corridor leading to the critical assembly room and was adjacent to the radiochemistry lab. The control room was located next to the south shield wall of the critical cell. Surveillance of cell operations was provided by nuclear instrumentation and a closed-circuit television channel.

The ventilation system was designed so that air flowed from non-radioactive areas to the critical assembly areas. The low-bay area is serviced by conventional heating and ventilating equipment which maintains a positive pressure in the area. The radiochemistry laboratory area was maintained slightly negative with respect to surrounding areas, by exhausting air through a fume hood and HEPA filters. The SGR and OMR critical assembly areas have separate ventilation systems. These two areas and their ventilation systems are isolated and completely independent of each other. The air flow remains towards these critical assembly areas, even if the doors leading from the low-bay areas open simultaneously.

Air intake to the SGR critical assembly room was filtered, to maintain the purity of the graphite, and entered through a duct which opens near the floor of the critical cell. The area is exhausted through roof mounted, motor-driven exhausters. There is one main exhauster with a HEPA filter unit and seven smaller exhausters without filters. The HEPA filter may be bypassed by operation of motor-driven dampers, and the small exhausters may be closed in a similar manner. During reactor shutdown periods the system was operated without exhaust filters (all dampers open), to maintain a high air change rate. During reactor operation air was exhausted through the filter unit only. Negative differential pressures, with respect to ambient and surrounding occupied areas, were maintained under all conditions. The controls for the dampers and blowers were interlocked with the reactor controls, to assure that the system was in proper condition prior to reactor startup.

2.2.2 Facility Characteristics of the OMR Side

Although the OMR project has ended and the reactor was removed, most of the facility remains essentially unchanged. This facility description is from Reference 25, which was written in 1962. The OMR side of T009 (see Figure 2.8) consists of a concrete-shielded high bay that was used as a critical assembly room, and an adjoining low-bay area which housed a control room, laboratory, offices, and miscellaneous support and utility areas. The entire facility is approximately 110 ft long by 63 ft wide.

The critical assembly room (high-bay area) is a concrete structure with shield walls extending to the roof eave (37 ft). Shield thicknesses vary from 4 ft to 1 ft, depending on the height and location of the wall. External fences surrounding the facility provided control over personnel and limited the approach to radiation areas. The critical assembly room is approximately 35 ft square and has a clear ceiling height of 33 ft. A 5-ton bridge crane services the area, and a service door is provided on the northeast corner for truck entry to the main high bay. The roof consists of built-up roofing over rigid insulation on metal decking (no shielding). A 19 by 12 by 10-ft deep pit which abuts the east wall was provided in the concrete floor for a moderator drain and storage tank.

A fuel storage room was provided west of the assembly room, adjacent to the west shield wall and truck entry area. The fuel storage room faces a corridor leading from the control room in the low-bay area to the assembly room. A change room, opening on the same corridor, was provided for use in access from the control room and low-bay area to the fuel storage room and assembly room.

The low bay is a single story structure constructed of insulated metal siding and metal lath and plaster interior. The roof is constructed of built-up roofing over insulation on metal decking (as over the high bay), except for the counting room, which has a 2-ft thick reinforced-concrete roof and walls. In addition to the counting room, the building contains the control room, machine shop, laboratory area, mechanical equipment area, office area, lavatory area, and miscellaneous utility areas. The control room is located next to the south shield wall of the assembly room and controls access to the high-bay area.

The ventilating system for the critical assembly room is separate from that used in the remaining portion of the building. The system for the critical assembly room maintains the high bay at a negative differential pressure relative to surrounding areas. During reactor operation, the exhaust was discharged through a stack terminating 45 ft above grade level (10 ft above roof). This effluent was continuously monitored and was automatically diverted through pre- and HEPA filters, if higher than permissible activity levels were detected. With the filters in use, approximately 3500 cfm of air are discharged to the stack. When the reactor limited the approach to radiation areas. The critical assembly room is approximately 35 ft square and has a clear ceiling height of 33 ft. A 5-ton bridge crane services the area, and a service door is provided on the northeast corner for truck entry to the main high bay. The roof consists of built-up roofing over rigid insulation on metal decking (no shielding). A 19 by 12 by 10-ft deep pit which abuts the east wall was provided in the concrete floor for a moderator drain and storage tank.

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2.3 Facility Utilization and Present Radiological Condition

Following construction in 1959, Building T009 was designated the Critical Experiment Building. Critical assemblies were made and tested for the Sodium Graphite Reactor (SGR) core for HNPF. Critical assemblies were also made and tested for the Organic Moderated Reactor (OMR) for the Piqua power plant.

The SGR began nuclear operation in January 1960, and a variety of experiments were performed on clean, graphite-moderated-reactor systems. Experiments were also performed in support of the Hallam Nuclear Power Facility program. Critical experiments were conducted with slightly enriched metal fuel, to study lattice parameters of SGR systems.

The SGR Critical Assembly was designed functionally to permit experiments on graphite-moderated reactors of various sizes and fuel-element configurations. The basic assembly consisted of a cylindrical array of hexagonal graphite logs in which fuel and poison elements were placed in a wide variety of lattice spacings and arrangements. The overall size of the assembly could be varied to 14 ft in diameter and 17 ft in height. To study temperature effects, provisions were made for air-heating of the assembly.

A general evaluation program for OMR systems using plate and circular-type, slightly enriched, metal fuel elements was underway in early 1959. The critical loading, control characteristics, and other reactor parameters were studied. Void coefficient measurements were performed on this system. Critical experiments in support of the Piqua Nuclear Power Facility were performed.

The OMR critical assembly consisted of a core vessel, thermal shield, fuel and control elements, source shield and drive mechanism, a moderator storage and drain tank with connecting lines, and an access stand for the assembly. The core was contained in a 6 ft diameter by 8-1/2-ft high mild-steel tank, which was supported 5 ft above the floor by a massive steel stand. The core mockup utilized slightly enriched fuel in a heterogenous, organic-moderated lattice. The uranium metal fuel elements which were used were of the flat plate, box-type, or were concentric cylinders. The moderator and reflector was a commercial mixture of terphenyl isomers maintained in a liquid state by means of electric heaters. A moderator dump system with a quick-opening valve was provided to drain the core vessel in the event of scram or shutdown. Experiments could be performed in the temperature range of 325 to 600F, in an unpressurized system. Boron carbide-filled shim and safety rods, activated by cables, were used for control.

The final critical assembly work ended in 1965 with the cancellation of the SRE Power Expansion Program. The fuel and critical assembly equipment were eventually removed from the building. In the early 1970s, the building was redesignated as the Engineering Development Facility. DOE-funded sodium fire experiments were conducted in the OMR high-bay, and In-Service Inspection (ISI) work was initiated in the SGR high-bay. At the present time the building is used for ISI work, and a variety of other work.

For conversion of the SGR high-bay to ISI work, the utilities were removed from the Change Room, and the interior walls of the Fuel Storage area were taken down. All fuel and equipment directly associated with the critical assembly and its operation were removed, and radiation surveys were performed. The ventilation system and exhaust filter banks for the SGR side (located on the high bay roof) are still in place and known to be uncontaminated. This system is not in use. All support labs were cleaned, are not in use, and known to be uncontaminated.

During operation of the SGR, there were no contamination problems. A later radiation survey showed a small amount of radioactivity in the SGR holdup tank (on the east side, and still in place but not used), thought to be thorium resulting from some chemistry work involving lantern mantles. Currently, some areas in the SGR high bay are subject to potential contamination from the ISI equipment, but there are no other known areas of contamination.

In the OMR high-bay, the reactor vessel and surrounding stand, the storage tank, piping, and electrical heating, and all other equipment associated with the critical assembly were removed. The fuel and the storage racks in the Fuel Storage room were removed. Radiation surveys were performed. After surveys were done, several areas were refurbished, repainted and floor tiles replaced. The ventilation system and exhaust filter plenum is still in place but thought to be uncontaminated. It has not been used since OMR operation. The radioactive material holdup tank for OMR has been removed. Drain lines leading to it probably are still in place.

On the OMR side, a minor, but continuing problem was experienced due to the peeling of the nickel plating on the uranium metal fuel plates. Release of radioactive contamination was limited to the organic moderator, and to work table surfaces and storage shelves. These areas were kept decontaminated.

3.0 SURVEY SCOPE

Building T009 was radiologically inspected for enriched uranium, depleted uranium, thorium, mixed fission products and activation products as specified by the "Radiological Survey Plan for SSFL," (Reference 4, Section 5.4.20):

* Survey perimeter pads and outside perimeter fence in area of old sanitary leach field.

* Survey the OMR side, including the high bay, fuel storage vault, counting room, hall closet, tool room, shop, and control room.

* Survey the ventilation system (of the OMR side).

* Survey the drain system (this applies to the R/A holdup tank on the east side of the building - SGR side).

* Survey the wall of west vault (OMR) for activated concrete.

Figure 3.1 shows these survey locations. Only the OMR side of T009 was surveyed. The SGR side is now used for ISI equipment which may be contaminated with radioactive material.

Interior building areas were radiologically inspected by measuring total and removable alpha/beta activity and ambient gamma exposure rate. Total alpha and beta surface activity was also measured "for indication" on selected grids and special building features. The surrounding northwest area (see Figure 2.7) was radiologically inspected by measuring ambient gamma exposure rates 1 meter above the surface, one in each 6-m square cell. If this gamma measurement indicated contamination, surface soil samples were to be acquired and analyzed by gamma spectrometry and for gross alpha/beta

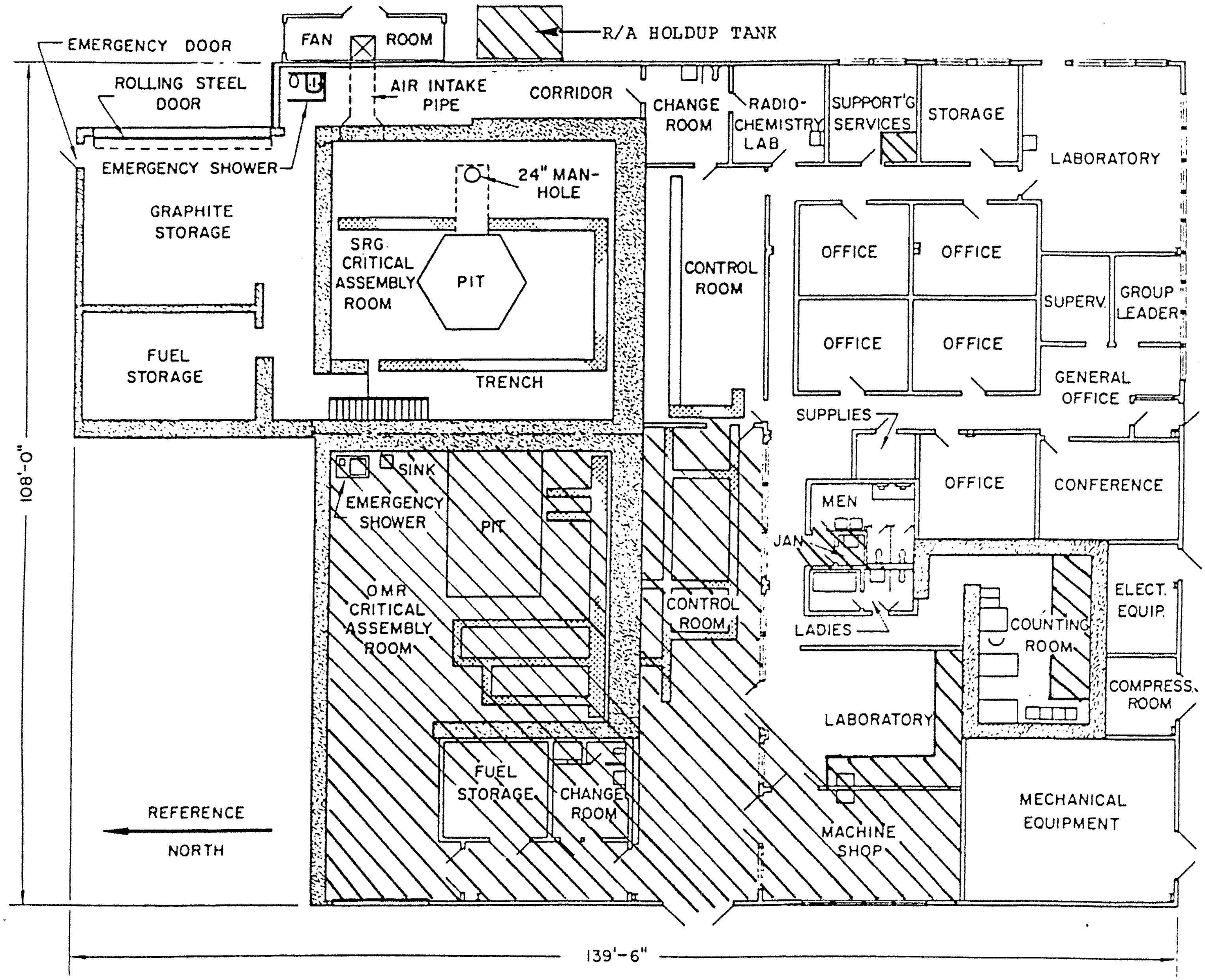

Figure 3.1 Sampling Lot Plan for Building T009

activity. Exhaust vents, filter plenums, sinks, drains, showers, and similar systems were surveyed for residual removable alpha/beta activity, and total alpha/beta contamination, and in some cases were scanned for gamma emitters by gamma spectrometry.

Total alpha/beta activity was measured in 57, 1 m^2 wall and floor locations inside the facility. Removable alpha/beta activity was assessed in those same locations plus several cabinets in rooms 114 and 116, and on miscellaneous building features. Measurements "for indication" of alpha and beta activity were also made on special building features. Gamma exposure rate measurements were made in 183 locations: 57 inside of T009; and 126 in the outside northwest area. Soil samples were not collected and analyzed because no indication of contamination was found. Ambient gamma exposure rates are reported in micro-roentgens per hour ($\mu R/h$). Removable alpha/beta activity is reported in disintegrations per minute per 100 cm^2 (dpm/100 cm^2). Alpha and beta activity measurements "for indication" are reported as No Detectable Activity (NDA), or less than 50 counts per minute (cpm), 60 cpm, ...etc. These data were analyzed statistically by sampling inspection by variables techniques against appropriate residual contamination acceptance limits.

3.1 Unrestricted-use Acceptable Contamination Limits

A sampling inspection plan using variables, discussed in Section 4.2, was used to compare radiological contamination quantities against unrestricted-use acceptable contamination limits prescribed in DOE guidelines (Reference 1), Regulatory Guide 1.86, NRC license SNM-21, and other references. The limits shown in Table 3.1 below have been adopted by Rocketdyne and are based on enriched uranium used for critical tests performed at T009. Measurements of average surface alpha/beta contamination are averaged over an area of no more than 1 m^2. The maximum allowable alpha/beta contamination level applies for a single area of not more than 100 cm^2 in that 1 m^2. Allowable removable alpha/beta contamination is based on a surface wipe with area equal to 100 cm^2.

Table 3.1 Maximum Acceptable Contamination Limits

Criteria	Alpha (dpm/100 cm^2)	Beta (dpm/100 cm^2)
Total Surface, averaged over 1 m^2	5000	5000
Maximum Surface, in 1 m^2	15000	15000
Removable Surface, over 100 cm^2	1000	1000
Ambient Gamma Exposure Rate*	5 μR/h above background	
Soil Activity Concentration**	46 pCi/g	100 pCi/g
Water Activity Concentration***	$1x10^{-4}$ 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 limit for enriched uranium is 30 pCi/g (Reference 26) plus that contribution from naturally occurring radioactivity, (about 16 pCi/g from Reference 17, p. 93). 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 for the survey. These are proactive action levels initiated when the surveyor detects radiation according to the following criteria:

1. <u>Characterization Level</u> - 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 this radiological survey, two sample lots were established for radiologic characterization and data interpretation. Interior and exterior radiological measurements were each treated as separate sample lots. Figure 3.1 shows the interior sample lot consisting of the following rooms:

* Room 126 (OMR Critical Assembly);
* Room 124 (OMR Fuel Storage);
* Room 122 (Change Room);
* Room 120 (Control Room);
* Room 118 (Machine Shop);
* Room 116 (Laboratory);
* Room 114 (Health Physics Counting Room); and
* Room 108 (Janitor's Closet).

These are areas of most probable residual contamination. Figure 2.7 shows the outside northwest area surveyed for residual radioactivity.

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

For the surrounding 1-acre northwestern area, 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 the fence-line).

3.3 Alpha and Beta Contamination Measurements

In order to determine alpha/beta contamination in each square meter surveyed per 9-m^2 area, four radiological characteristics were measured: total-average alpha surface activity, total-average beta surface activity, removable alpha surface activity, and removable beta surface activity. The location of the 1-m^2 area was left to the surveyor's judgement: it was to be the area that, in his judgement, was most likely to have retained the most residual contamination of any similar area within the 3-m square grid. The surveyor was instructed to do this conscientiously to assure that any significant residual contamination would be detected. The use of a predetermined grid with discretion for the exact location provides a uniform survey biased towards the high end of the distribution. An alpha probe and beta probe were each connected to a Ludlum Model 2220-ESG portable scaler.

Measurements of the average alpha surface activity were made by use of a large-diameter (9.5 cm) alpha scintillation detector, sensitive only to alpha particles with energy exceeding about 1.5 MeV. This detector was calibrated using a Th-230 alpha source. The energy of Th-230 alpha particles (4.6 MeV) is similar to that of the isotopes handled at T064, U-235, U-234, and U-238.

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

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

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

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

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

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

3.4 Ambient Gamma Exposure Rate Measurements

In each 9-m^2 cell (inside T009) and each 36-m^2 cell (in the northwestern outdoor area), a gamma exposure rate measurement was made 1 m from the surface. The particular location in each cell was chosen randomly, and identified on a map. A tripod was used to support a 1 in. x 1 in. NaI crystal coupled to a photomultiplier tube and fed to a Ludlum 2220-ESG scaler, at 1 m from the ground. In each cell, a 1-min. count was collected and converted to μR/h. The measurement location and exposure rate were recorded in tabular form. About 183 1-min. measurements were acquired.

3.5 Surface Alpha and Beta Radioactivity Measurements "For Indication"

Measurements of beta surface activity "for indication" were required by the Site Survey Plan (Reference 4) for better characterization of radiological condition only if gamma exposure rate measurements indicated possible radioactivity. No exposure rate measurements indicated possible radioactivity, so beta surface activity measurements were not made on a limited basis. However, a thorough survey for alpha and beta surface activity was performed in suspect-looking areas. Alpha/beta surveys were performed on sinks, drains, drain-trap sludge, showers, exhaust vents, filter banks, cracks, wall-to-floor joints, miscellaneous horizontal surfaces, and various residual test components. Alpha measurements were made using a Ludlum 43-1 alpha scintillator coupled to a Ludlum model 12 count-ratemeter. Beta measurements were made using a Ludlum 44-9 pancake Geiger-Mueller probe (active area = 20 cm^2) coupled to a Ludlum model 12 count rate meter. These measurements were made "for indication."

3.6 Sample Collection for Radioactivity Analysis

If an outdoor gamma exposure rate measurement indicated radioactive contamination, a 2-lb surface soil sample (no greater than 3" deep) was to be collected from that spot and analyzed for radioactivity. No soil samples were collected for this particular survey. Several other samples, however, were required by the Survey Plan to be collected and analyzed for radioactivity: 1) sink drain trap samples in rooms 121 and 108; 2) grease and crud from heavy machinery in room 118; 3) shower drain sediment from room 122; and 4) sludge from the R/A holdup tank on the east (SGR) side of T009. These samples were collected and analyzed by gamma spectrometry for gamma-emitting radionuclides.

3.7 Goals and Limitations of Survey Scope

The scope and detail of this radiological survey is based on the likelihood for residual radioactivity occurring in these areas because of the nuclear operations which were performed. These facilities are not suspect of containing residual radioactivity for several reasons:

1. Most nuclear materials handled at T009 were fully encapsulated;

2. The problem experienced at OMR relative to the peeling of nickel plating on uranium metal and consequential releases was controlled. The affected areas were kept decontaminated;

3. Activation of building materials was insignificant; and

4. When both the OMR and SGR sides were reassigned, a thorough radiation survey was performed to ensure no residual radioactivity remained undetected.

The scope of this survey was established in Reference 4 based on an unlikely occurrence of residual radioactivity being accidentally left behind from previous operations. The goal of this survey is to determine if contamination exists to such an extent that further surveying or remedial action is warranted.

Total and removable alpha/beta radiation measurements performed on an 11% sampling plan comply with State of California guidelines established for final radiological surveys to clear a facility for unrestricted use. Since those areas considered most likely to retain residual contamination were specifically surveyed, this sampling plan was biased to the high end of the measurement distribution.

Ambient gamma exposure rate measurements made outdoors are sensitive enough to detect contaminants accidentally left behind or unknowingly released. Most probable contaminants are mixed-fission products and activation products. The probability of existing residual enriched uranium is highly unlikely; no uranium powders or grinding fines were handled here. It is highly unlikely that any subsurface debris is currently in natural terrain areas; they were never used as dumping grounds or landfills. The leach field was for sanitary use only. Subsurface transport of contaminants is also considered negligible. If any contaminants do exist on-site, they are most likely still on the surface.

Because of the large area surveyed, outdoor exposure rates were measured every 36 m^2. This sampling plan is sufficient for two reasons:

1) Gamma measurements made on a 6-m square would detect Cs-137 at 100 pCi/g (the beta activity limit) if the surface layer was thicker than 1 cm. A 1 mCi Cs-137 source would be detectable at the greatest separation distance of 6 meters. These sensitivities meet the requirements of this survey; and

2) By applying Lot Tolerance Percent Defective techniques, we can determine with a statistical confidence of 0.90, that there is a probability of 90% that radioactive contamination does not exceed some predetermined acceptance limit. This determination varies inversely to the number of samples taken. This technique, along with the graphical representations of cumulative distribution functions will identify trends, anomalies, outliers, and perturbations in the radiation levels.

We are able to conclude whether:

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

2. Any relatively intense gamma-emitting debris is buried (see Section 5.6.4).

We cannot conclude whether:

1. Any slight subsurface migration has occurred; or if

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

The likelihood for occurrence of the above two conditions is small. First, migration periods of contaminants below the surface are typically very long. It is much easier for surface water flowing downslope to carry with it any contaminants. The settling out of these contaminants into the subsurface also takes a long time. Second, no known burial or dumping activities took place in the northwestern area of Building T009.

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

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

where C = the number of counts recorded in time, T, of the sample

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

Equal values of the time, T, must be used for the sample and background counts for equation 4-3 to apply. This Poisson distribution and standard deviation applies for single radiation measurements, of the discrete random variable, x, and is applicable only when the observation times are short compared with the half-life. This is the case for the site survey.

Because of the probabilistic nature of particles emitted by radioactive elements, repeated measurements of the average number of emissions per unit time shows a distribution approximated by the Gaussian (or normal) probability density function (pdf); this is known as the central limit theorem. This theorem holds for any random sample with finite standard deviation. If measurements are made at many similar locations, these measurements will show a greater variability, but the distribution will remain adequately represented by a Gaussian function. This Gaussian approximation is good when the number of samples collected is at least 30. Thus the number of occurrences of particular mean radiological contamination values, g(x), shows a Gaussian pdf relative to the contamination value, and the data can be plotted accordingly. Subsequently, based on the results of the data analysis, a conclusion can be made regarding the amount of radioactive material in an area, and any anomalous values can be identified.

The Gaussian probability density function, g(x), is given by:

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

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

m = average, or mean of the population distribution

σ = standard deviation of the population distribution.

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

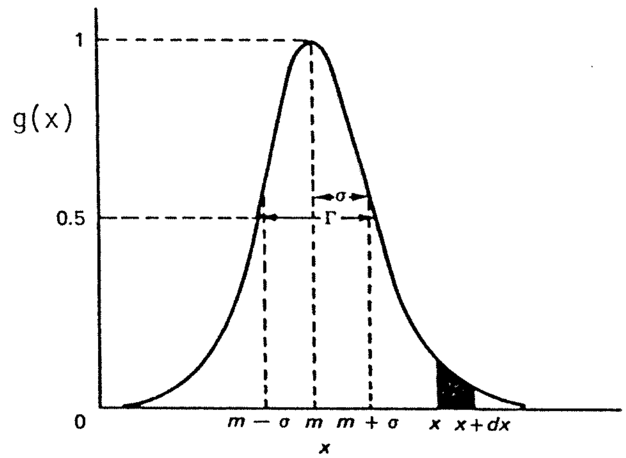

Figure 4.1 The Gaussian Probability Density Function

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

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

$$= P(x < X)$$

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

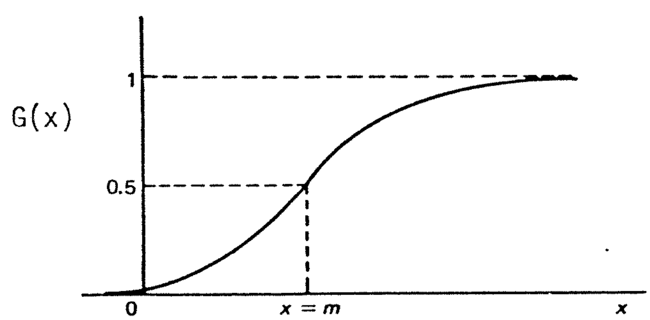

Figure 4.2 The Gaussian Cumulative Distribution Function

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

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

4.2 Sampling Inspection

4.2.1 By Variables

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

4.2.2 By Attributes

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

4.3 Sampling Inspection by Variables

4.3.1 Calculated Statistics of the Gaussian Distribution

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

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

The sample mean is given by:

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

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

The standard deviation, s is given by:

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

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

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

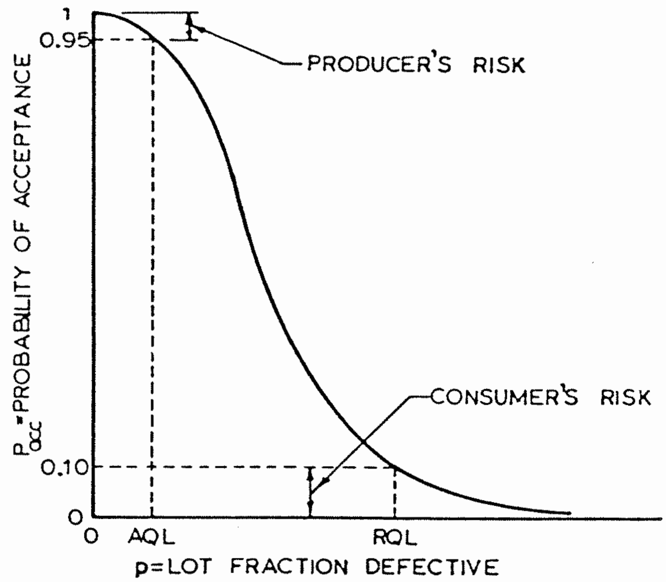

Figure 4.3 Operating Characteristics Curve

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

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

where:

k = tolerance factor

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

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

n = number of samples

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

The coefficients K_β 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 (Figure 4.2) on a linear-grade scale. For convenience of this survey and for readability, G(x) is plotted as the abscissa (x-axis) on a probability grade scale and the measurement value, x, is plotted as the ordinate (y-axis). 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.

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.

The coefficients K_β 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.

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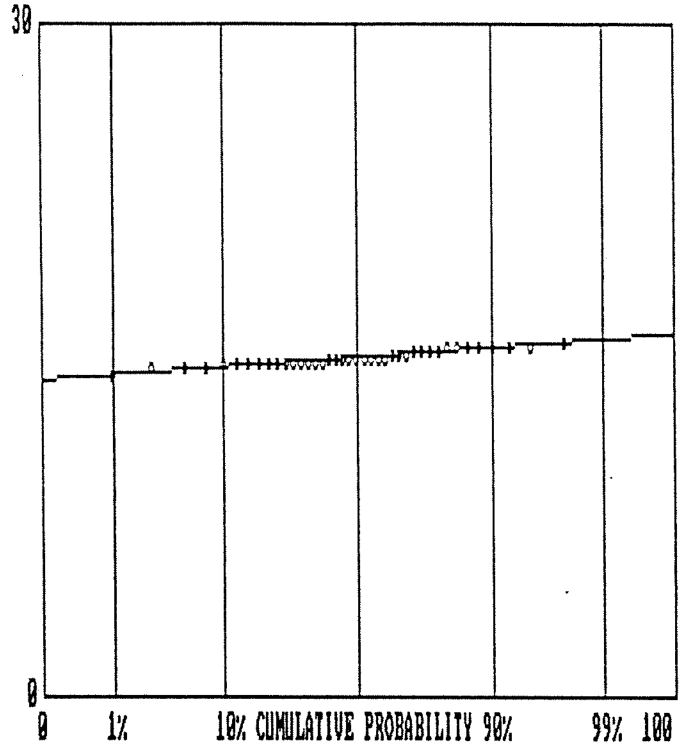
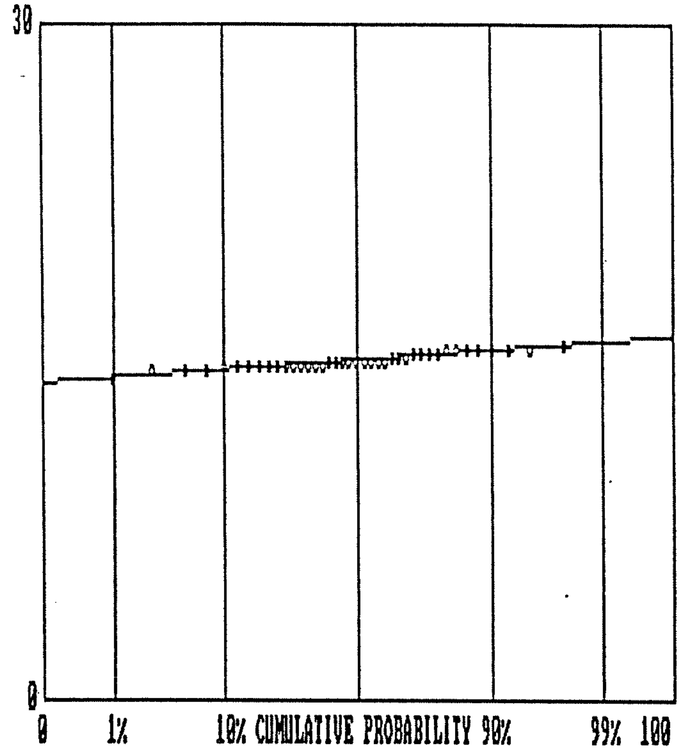


Figure 4.4 Gaussian cdf Plotted on Probability-Grade Paper

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.



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5.0 ANALYTICAL TECHNIQUES

Statistical methods presented in Section 4.0 were used to judge whether a sampling lot is not contaminated, slightly contaminated, contaminated above acceptance limits, or whether additional investigation is required. That decision is based on several radiological measurements. For interior surfaces and selected building features, these radiological measurements were:

1) Direct alpha and beta radiation;
2) Removable alpha and beta radioactivity; and
3) Ambient gamma exposure rate.

For exterior locations in the northwest area, ambient gamma exposure rate measurements were made. Locations showing elevated gamma exposure rates were to be selectively sampled and analyzed for gross alpha/beta radioactivity and for qualification and quantification of detectable gamma-emitting radionuclides.

Analytical techniques used to acquire, evaluate, and interpret these radiological measurements are presented in detail in this section. These techniques include instrument calibration, determination of "ambient background" radiation, evaluation of computer-generated gamma spectrometry output, and computerized data analysis through inspection by variables.

5.1 <u>Data Acquisition</u>

In each selected 1 m^2 area within a 3-m square grid inside T009, total and removable alpha/beta contamination were measured. In each 6-m square grid outside T009, ambient gamma exposure rate was measured. Each square grid was outlined and marked with its coordinates. The exact location within that square grid where the samples were collected was left to the surveyor's judgement: it was to be the area that, in his judgement, was most likely to have retained the greatest amount of contamination in

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5.2 Data Reduction Software Program

Each radiological measurement characteristic data value was input into SMART SPREADSHEET. This is an off-the-shelf computer software package which allows multiple computations to be performed on raw data values. Columns were established to calculate the alpha/beta total-average, maximum, and removable contamination per 1 m^2 in dpm/100 cm^2; and surface ambient gamma exposure rate in $\mu R/h$. 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 this data reduction was, for inside measurements:

1. Room number;
2. Grid location; ex. W(1,3), west wall, grid 1,3;
3. Alpha total activity, averaged over 1 m^2 (counts in 5 min.);
4. Alpha maximum activity for hot spot, if present (counts in 5 min.);

5. Alpha removable activity from 100 cm^2 smear (counts in 5 min.);
6. Beta total activity, averaged over 1 m^2 (counts in 5 min.);
7. Beta maximum activity for hot spot, if present (counts in 5 min.);
8. Beta removable activity from 100 cm^2 smear (counts in 5 min.);
9. Alpha survey instrument background (5 min.), efficiency factor (dpm/cpm), and area factor;
10. Alpha gas-proportional detector background (5 min.) and efficiency factor (dpm/cpm);
11. Beta survey instrument background (5 min.), efficiency factor (dpm/cpm), and area factor;
12. Beta gas-proportional detector background (5 min.) and efficiency factor (dpm/cpm).
13. Ambient gamma exposure rate (counts in 1 min.; cpm);
14. Gamma survey instrument background (1 min.);
15. Gamma survey instrument efficiency factor (μR/h/cpm).

Output for Gaussian Plots of inside measurements:

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

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

Input for data reduction of outside measurements was limited to gamma exposure rate data listed above.

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 using the acceptance limits in Table 3.1.

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

5.4 Direct Alpha/Beta Contamination Measurements

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

5.4.1 Instrument Calibration

Each detector was calibrated three times daily. The alpha detector was calibrated with Th-230; the beta detector with Tc-99. Background levels were determined in the counting room of Building T009.

5.4.2 Data Acquisition and Reduction

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

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

where:

SA	=	surface activity
C	=	total counts in 5 min.
5	=	count time, min.
B	=	background count in 5 min. (generally 0-5 for alpha and about 440-460 for beta)
EF	=	Efficiency factor, dpm/cpm (averages about 4.8 for alpha and about 3.7 for beta)
100	=	100 cm^2 standard area
A	=	probe sensitive area (71 cm^2 for Ludlum model 43-1 circular alpha scintillator; 20 cm^2 for Ludlum model 44-9 pancake G-M)

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

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

5.4.3 Data Analysis

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

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

5.5 Removable Alpha/Beta Contamination Measurements

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

5.5.1 Instrument Calibration

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

5.5.2 Data Acquisition and Reduction

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

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

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

The standard deviation of this measurement is:

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

5.5.3 Data Analysis

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

5.6 Ambient Gamma Exposure Rate

Measurements of ambient gamma exposure rate were made by use of a 1" x 1" NaI scintillation crystal coupled to a Ludlum Model 2220-ESG portable scaler, (Appendix A.3). This device was mounted on a tripod so that the sensitive crystal was 1 meter from the ground. The detector is nearly equally sensitive in all directions, i.e. 4 π geometry, and can detect variations in exposure rate down to one-half of a μR/h, using the digital scaler for a 1-min count time. Because of the natural variability of ambient radiation (particularly outdoors), however, a 3 to 5 μR/h exposure rate above "background" is considered the instrument sensitivity in practice. At this level, a surveyor would decide to collect additional measurements.

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 because of "instrument out of tolerance" was never necessary.

5.6.2 Data Acquisition and Reduction

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

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

where R = exposure rate (μR/h)
C = gross counts in 1 min. (cpm)
EF = efficiency factor (0.004632 μR/h/cpm) based on cross calibration with HPIC.

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

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

5.6.3 Data Analysis

Analysis and interpretation of gamma exposure rate data is a five step process:

1. Plot, in order of magnitude from left to right, total-gross exposure rates in μR/h against cumulative probability for three independent areas considered to be "natural background" at SSFL. These survey locations should be from areas where no radioactive material has ever been used, handled, stored, or disposed. These areas should be of similar geologic characteristics to those of the inspected areas. Calculate the average, standard deviation, and range for each distribution. These three distributions give the baseline for "natural" variability of exposure rate as a function of SSFL terrain.

2. Plot total-gross exposure rates in μR/h against cumulative probability for each subject sampling lot. Calculate the average, standard deviation, and range for each distribution. Compare these statistics and probability distributions against "natural background" distributions.

3. Determine if there are any trends indicated by the probability plots of each subject sampling lot which show a potentially contaminated area. If necessary, explain elevated measurements and/or trends in the distribution.

4. Determine whether the "natural background" distributions adequately represent "ambient background" for the tested areas. Determine if any nuclear-related operations in the local area are influencing "ambient background" in the test-areas. If so, make corrections.

5. Subtract "natural background" from each test-area measurement and compare the results against acceptance criteria in Table 3.1 and Section 4.3.3. Use inspection by variables techniques to test for acceptance. Calculate the average, standard deviation, and test statistic, $\overline{x}$ + ks, for each test-area distribution. If "ambient background" in the test-areas differs from "natural background," correct the data accordingly and retest. Often, "ambient background" is less than "natural background." When this is the case, a better estimate of "ambient background" is the median gross-total exposure rate value from the same uncorrected data set. The median is an unbiased estimator of "ambient background."

The most critical step in the analysis of gamma exposure rate measurements is assessing what true "ambient background" radiation is for a test area. "Ambient background" accounts for three effects which result in the production of an electronic pulse of the gamma instrument (a count), which under ideal measurement conditions would not occur:

1. "Natural background" radiation from the cosmos, and primordial radionuclides;

2. Secondary influence of gamma exposure rate due to nearby facilities which handle radioactive materials or radiation producing machines; and

3. Instrument noise.

These individual contributions to "ambient background" complicate data interpretation against acceptable limits because both the NRC and DOE criteria for acceptance for unrestricted use are given in μR/h above background, 5 and 20, respectively. In natural-terrain areas, significant deviations in "natural background" radiation occur as a function of landscape geometry. For example, when the detector is placed near a large sandstone outcropping, the exposure rate may increase by almost 4 μR/h. This increase is due to primordial radionuclides in the sandstone, and a change in source geometry, from a planar 2π-steradian surface to a rocky 3π-steradian surface. "Natural background" is also more variable when measurements are made over, at, or near large metal pieces, scrap components, and other objects -- such as those stored behind Building T009. "Natural background" is also different indoors and varies with construction materials. Fortunately no facilities are located nearby which increased ambient gamma conditions at T009. Finally, instrument noise fortunately, is fairly uniform.

The best solution for evaluating the potential or existence of residual contamination in an area where the ambient radiation field varies naturally by swings as large as the acceptance limit, is to first compare test-area total-gross exposure rates against "natural background" total-gross exposure rates. "Natural background" measurements were taken on flat and rugged terrain, with Chico Formation sandstone, similar to the characteristics outside of the Building T009 fence-line.

Best corrections for "ambient background" radiation for this survey, then, fall into two categories:

1. "Natural background" inside a combination concrete block/Butler-type building with concrete slab floor. Exposure rate is typically 6 to 12 μR/h. The best estimate of "background" for a typical facility is the median gross-total exposure rate measurement value for that facility; and

2. "Natural background" of a planar landscape composed of asphalt concrete. This is typical of the equipment staging area and storage yard outside of T009. Exposure rates are not highly variable, unless many large equipment items are stored there. Expect 13 to 15 μR/h.

Once all the best corrections for "ambient background" have been made, resulting distributions are compared against the 5 μR/h above "background" acceptance limit. The test statistic, $\overline{x}$ + ks, is calculated for each distribution. Statistical acceptance criteria presented in section 4.3.3 apply.

5.6.4 Sensitivity of Gamma Exposure Rate Measurements

The purpose of performing these exposure rate measurements is to detect any significant quantity of gamma-emitting radionuclides. Operational history and surveys performed years ago show that the most probable radiological contaminant in these areas is Cs-137, associated mixed-fission-products, and activation products. Since Cs-137 is a gamma emitter, it is detectable with the NaI detector.

The sensitivity of these measurements, or rather, the amount of contamination which could be there and which would not be detected, is based on two possibilities:

1) A uniformly contaminated region of soil or pavement; a layer on the surface, or a layer several feet below the surface; (this scenario is unlikely in this inspected area); or

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

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

2) A piece of contaminated debris located on the surface or buried several feet below, (this is the more credible scenario for residual radioactivity in this outside area).

Our acceptance criteria specify that no soil activity exceeding 100 pCi/g-beta is acceptable for unrestricted-use. In comparison, 10 μCi of Cs-137, total, is the limit for exempt quantity according to 10CFR30, Schedule B. If only Cs-137 were contained in the soil, 10 μCi of activity would be present in 100 kg of soil, or about 70,000 cm^2 of surface area, if the layer were 1 cm thick.

Natural ambient gamma "background" radiation is about 12-16 μR/h at 1 meter from the ground, so the radioactive material would have to produce an exposure rate of about 3 μR/h above background in order to detect it to such an extent that further investigation would commence. Table 5.1 shows theoretical exposure rates calculated for some uniformly contaminated soil and miscellaneous contaminated debris. The contaminant is assumed to be Cs-137. Condition (1) assumes a uniformly distributed layer of soil with 100 pCi/g Cs-137. Condition (2) assumes a point source of Cs-137 with total activity equal to 1 mCi.

For condition (1), 100 pCi/g Cs-137 layer of contaminated soil, these measurements would detect a surface layer greater than one cm thick, but would not detect a small thickness of soil (10 cm) buried much more than a half-foot from the surface. This is very good sensitivity, particularly since the likelihood of a thin layer of contaminated soil located more than 6 in. below the surface is small. For condition (2), contaminated debris, whose activity exceeded 1 mCi Cs-137 activity could be seen if it wasn't buried much deeper than about a foot. 10 mCi could probably be seen down to 2 feet. The likelihood of buried or scattered debris occurring in these areas is very small; however, this is probably the most likely scenario of the two for residual contamination. Concerning suspect activation products and their sensitivity levels, Co-60 is the most significant activation product. It is more easily detectable than Cs-137 because of higher energy gamma rays. Thus, this Cs-137 analysis gives the most conservative sensitivities for suspect contaminants.

5.7 Collection of Miscellaneous Samples for Radioactivity Analysis

Miscellaneous sludge samples from the SGR holdup tank, drain line clean-outs, and sink traps were analyzed for radioactivity by gamma spectrometry. Additionally, if radioactivity was indicated during performance of the filter bank survey, samples were to be collected and analyzed in the same manner. This section deals with the special techniques of performing gamma spectrometry.

5.7.1 Gamma Spectrometry

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

5.7.1.1 Instrument Calibration

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

It is particularly important when performing gamma spectrometry analysis, that the sample geometry be identical to the standard geometry. Efficiency is a function of geometry, and varies significantly in this case. Since these miscellaneous samples are not in the same counting geometry as the standard, these measurements are normally good "for indication" only.

5.7.1.2 Data Reduction and Analysis

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

Table 5.2 Probable Gamma Energies for Determining Radioactive Contaminants

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

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

Any major gamma emitting isotopes are readily distinguishable by the MCA. If by observation, it is determined that uranium daughter products are present above natural background concentrations, further interpretation is necessary. In which case the isotopes presented in Table 5.2 are used to estimate U-238, U-235, and Th-232 activity. Estimates of radionuclide content in each sample were derived based on corrections for:

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

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

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

U-238 activity is calculated by:

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

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

n = number of non-zero activity values

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

Th-232 activity is calculated by:

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

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

n = number of non-zero activity values

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

Cs-137, Co-60, and K-40 are easily identifiable radionuclides by gamma spectrometry; they each emit a single characteristic gamma-ray, (see Table 5.2).

6.0 PROCEDURES

The following radiological procedures were used in performing this survey.

6.1 Sample Selection Gridding

Superimpose 3-meter square grids on each indoor surface to be radiologically characterized. If a surface is less than 9 m^2 in area, then grid the area by square meters as appropriate. Designate each square meter in matrix notation with floor location (1,1) being the northwestern most square in a room. Wall location (1,1) is the lower left square meter as the surveyor views the wall.

6.1.1 Floor

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

6.1.2 Walls

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

6.1.3 Ceiling

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

6.1.4 Special Structural Features

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

6.1.5 Northwest Storage Yard and Surrounding Area

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

6.2 Calibration and Instrument Checks

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

Portable Ludlum 2220-ESG Survey Instruments:

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

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

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

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

5) Check battery (greater than 500).

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

5 min. for alpha and beta measurements. For ambient gamma exposure rate measurements, set time to 1 min.

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

8) Take and record a 5 min. count of known alpha and beta standards; an electroplated Th-230 and electroplated Tc-99 source, respectively. The efficiency factor (dpm/cpm) is calculated as the ratio of 2 times the 2π emission rate of the source (dpm) to the net count rate of the instrument. The radioactivity of the calibration sources is traceable to NBS. Similarly, use a Ra-226 check source located 1 ft from the NaI detector to check operability of the gamma instrument. The count rate should not vary by more than $\pm$ 5% from the initially established standard. The gamma calibration efficiency factor is determined by comparison against a Reuter Stokes High Pressure Ion Chamber (HPIC).

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

Gas-flow Proportional:

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

2) Using uncontaminated planchets, take four 5 min. background counts to determine the detector background for smear samples.

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

Average the daily results:

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

Gamma Spectrometer:

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

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

6.3 Radiological Measurements

6.3.1 Total-Average Alpha/Beta Contamination Measurements

1) Identify 1-m^2 area to be measured; 1 m^2 per 9 m^2 surface should be surveyed to be consistent with a minimum 11% sampling plan. Determine instrument background and efficiency as presented in Section 6.2.

2) With portable scaler instrumentation (Ludlum 2220-ESG) set for a 5-min. count time, use an alpha probe (Ludlum Model 43-1) on one instrument and a beta probe (Ludlum 44-9) on another, then uniformly scan the area. The probe transit velocity should be slow; less than one-third the numerical detector window diameter (in cm). This corresponds to a transit velocity not exceeding 3 cm/sec. The 5 min. count time per square meter was adopted based on this transit velocity limit for alpha contamination. (Watch and listen

for "hot spots" where radioactivity may exceed the average limit. These spots are to be resurveyed later).

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

4) Enter the data into SMART spreadsheet.

6.3.2 Maximum Alpha/Beta Contamination Measurements

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

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

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

4) Enter the data into SMART spreadsheet.

6.3.3 Removable Alpha/Beta Contamination Measurements

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

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

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

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

5) Enter the data into SMART spreadsheet.

6.3.4 Ambient Gamma Exposure Rate Measurements

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

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

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

4) Enter the data into SMART spreadsheet.

6.3.5 Surveys of Special Structural Features and Components

1) Using a Ludlum Model 12 count rate meter in connection with a Ludlum Model 43-5 rectangular alpha scintillation probe, survey various building features and components which are suspect of containing residual alpha contamination. Suspect areas include light fixtures, wall to floor joints, coving, beam supports, horizontal surfaces, cabinets in room 114 and 116, and miscellaneous equipment.

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

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

4) Do the same for beta contamination using a Ludlum model 44-9 pancake GM beta probe.

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

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

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7.0 SURVEY RESULTS

A radiological survey of Building T009 was performed using the survey plan previously described. Two sample lots for analyzing and interpreting radiological data were established: 1) Indoor area; and 2) Northwestern Outdoor Area. Uniform 3-m square grids were established inside to measure total and removable alpha/beta activity and gamma exposure rate. Within each 3-m square grid 1 m^2 was surveyed. Uniform 6-m square grids were established outside to measure ambient gamma exposure rate. Analytical interpretation of gamma exposure rate measurements, and total and removable alpha/beta contamination measurements show that a few locations are slightly contaminated at levels far less than acceptance limits. Further investigation is not required in these slightly contaminated locations.

Before the results of this survey are presented, a summary of historically-documented radiation surveys performed at the facility after OMR and SGR were removed is covered in Section 7.1. Results of these previous and ongoing routine surveys were the basis for the extent of this radiological survey. Next, the format used for presenting data, analyzing probability plots, and interpreting results of this survey is presented. Total and removable alpha/beta and gamma exposure rate measurement results are presented according to this statistical format.

7.1 Historical Radiological Survey Results

Although OMR and SGR were removed from Building T009 in the early 1970s, the facility has been used for various projects involving the use and handling of radioactive material since that time. Building T009 is designated a radiological orphan facility, meaning that on a periodic basis, random radiation surveys are performed. This section is provided for information, and is a summary of the results from documented radiation surveys performed from 1972 through 1988. This summary is taken from References 30, 31, and 32.

In 1972 a smear survey was performed in the OMR high bay and fuel storage vault. Some plastic wrapped equipment stored in the high bay was found contaminated with up to 362 α dpm and 1207 βdpm removable activity. No Detectable Activity (NDA) was determined for the high bay floor. The fuel vault floor was contaminated up to 185 α dpm and 336 βdpm removable activity. A request was made to decontaminate. Presumably decontamination was performed.

In 1978 a smear survey was performed in the OMR vault, change room, cold change room, restroom, and high bay. No removable alpha/beta activity was found.

In 1978 a smear survey of the front office, SGR control area, cable trenches, overhead crane and tramway, SGR fuel storage shelf crevices, SGR access hallway, SGR cold change room, control room, roof penthouse, filter plenum, and miscellaneous items was performed. 185 smears were collected. In all cases, no detectable removable activity was found.

In 1979, 20 perimeter soil samples were collected and analyzed for radioactivity. Results show an average beta activity concentration of 23.8 $\pm$ 13.6 βpCi/g; this is considered natural activity at SSFL.

Sludge samples from the OMR tank (now removed), the SGR tank, and from a manhole access to the old leach field were collected in 1979 and analyzed for radioactivity. Two samples were collected from each tank; one sample was wet, the other dry. The OMR tank results show 0.30 αpCi/g and 5.4 βpCi/g (wet); and 0.91 αpCi/g and 16.4 βpCi/g (dry). This amount of radioactivity is statistically insignificant. Drain lines which drain to the old OMR tank are currently thought to be in place. Based on measurements of this sludge, the drain lines are probably sufficiently clean. The SGR-tank sludge results show 25.2 αpCi/g and 138 βpCi/g (wet); and 131 αpCi/g and 716 βpCi/g (dry). This amount of radioactivity is statistically significant. Results of two samples collected from the leach field show a dry sample to be 12 αpCi/g and 49 βpCi/g; and a wet sample to be 1.9 αpCi/g and 7.3 βpCi/g.

In about 1982, a radiological survey was performed to verify the results of surveys performed up to that time and to investigate specific locations that may not have been checked by routine analyses, (Reference 30). Special emphasis was placed upon the critical assembly cells, overhead exhaust systems, liquid holdup tanks, and insides of laboratory cabinets, and drawers. Of all the smears taken, none exceeded 50 βdpm/100 cm^2 and 5 αdpm/100 cm^2, corresponding to the counting instrument lower limit of detection. Direct surface radiation surveys also showed No Detectable Activity. Water for the SGR holdup tank was found contaminated with about 700 pCi/g gross beta activity. Commercially available thorium was thought to be the contaminant. At the time of that survey, the counting room contained depleted uranium, but no problems were known. Except for the SGR holdup tank and counting room, the building was found "clean."

In 1986, water and sludge samples taken from the OMR holdup tank were analyzed by gamma spectrometry, (Reference 31). No gamma emitting radionuclides were found.

In May 1988, smears were collected throughout the SGR side and machine shop. Exposure rates were also measured in the same locations. No Detectable Activity was found.

7.2 Statistical Results Format

Radiological data collected for this survey are displayed as Gaussian cumulative distribution functions in Figures 7.1 through 7.9, and Figures 7.13 and 7.14. Figures 7.10 through 7.12 are distributions of gamma exposure rate measurements made at 3 independent SSFL locations to demonstrate the variability of "natural background" radiation. These distributions are a guide for comparing sample lot exposure rate results. Figures 7.1 through 7.7 are distributions of total and removable alpha/beta activity for the indoor sampling inspection lot. Figures 7.8, 7.9, 7.13, and 7.14 are distributions of gamma exposure rate results for both sample lots. Figures 7.9 and 7.14 have been corrected for "ambient background" based on the median value of gross-total measurements. The median value is an unbiased estimator for "ambient background" in a sampling area. Normally, an average of the results presented in Figures 7.10 through 7.12, (i.e. the "natural background" data) is used for background correction, but those "background" measurements were from natural terrain areas -- significantly different from the indoor inspected area, and slightly different from the outdoor area. Subtraction of "natural background" in this case would be an overestimate of "ambient background" in both test-areas. 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 of the fitted-Gaussian on the ordinate which corresponds to a 50% cumulative probability on the abscissa. The measurement value at 50% cumulative probability is the median. For a theoretical Gaussian, the median is equal to the mean. For a well-fitted Gaussian, the median is very close to the mean. One, two, and three standard deviations above the mean correspond to 84%, 97.7%, and 99.8% cumulative probability for a one-sided test, respectively. Inspection by variables is used to test only "background-corrected" data sets against acceptance limits. The value of k used in the inspection test is very nearly 1.5 for each case; thus, the Test Statistic (TS) line ($\bar{x}$ + ks) will run perpendicular to the abscissa corresponding to about a 93.3% cumulative probability. The Gaussian distribution line must pass below the intersection of the "TS" line (about 93%) and the horizontal line showing the acceptance limit at that point in order to accept the lot as being uncontaminated. "k" and thus the "TS" line increase as the number of samples in a lot decrease.

At the top left hand corner of each output is the data file name for the sample lot. For "uncorrected" gamma exposure rate data sets, 30 μR/h is normally used for convenience, as the maximum ordinate value. If gamma measurements exceed 30 μR/h, then the greatest measurement value is the upper bound of the ordinate axis. In cases where gamma measurements have been corrected for "natural 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; see Section 4.3.3.

For total and removable alpha/beta activity measurements, an acceptance limit is applicable in all cases. The test statistic is calculated and compared against the appropriate limit.

In cases where an acceptance limit is not appropriate, for instance, gamma exposure rate measurements not corrected for "natural 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, "ambient background," is subtracted and inspection by variables techniques are applied to prove or disprove the hypothesis that the area is not contaminated.

Surveys of total alpha and beta activity on selected building surfaces and materials were performed "for indication." Statistical interpretation is not applicable. Results are presented as No Detectable Activity (NDA), or less than 20 counts per minute (cpm), 30, 40, 50,...etc. above background. If these surveys were to determine a contaminated location, further investigation was to commence.

7.3 Building Interior

Total-average and removable alpha/beta measurements and gamma exposure rate measurements were made per square meter in each 3-m square grid. 57 measurements of this type were made, 5 minutes each. An additional 28 smears were collected for assessment of removable alpha/beta activity on various equipment, cabinets, drawers, and suspicious looking places. All of these measurements were evaluated by analytical interpretation using Gaussian statistics. Miscellaneous features were also surveyed "for indication" of alpha and beta activity.

7.3.1 Alpha/Beta Grid Measurements

Table 7.1 shows the results for 57 1 m^2 grid measurements taken indoors of total-average alpha, removable alpha, total-average beta, removable beta and background-corrected ambient gamma exposure rate. Also included in that table is the results of 126 background-corrected exposure rate measurements for the outdoor area. The table shows four important statistics for each distribution: average value, maximum value, standard deviation and the inspection test statistic ($TS = \bar{x} + ks$). Smears were not collected from all 12 wall measurements, (the likelihood of removable activity on walls is small) but were collected from 28 special features, particularly the storage cabinets in rooms 114 and 116. This sampling plan resulted in 73 smear surveys.

Descriptive statistics presented in Table 7.1 show that the average values of alpha/beta radioactivity recorded inside T009 agree with results taken previously of similar "clean" areas. However, the standard deviations calculated for each measurement type are abnormally high.

Table 7.1 Survey Results for Building T009

Measurement	Number of Locations	Average Value	Standard Deviation	Maximum Value	Inspection Test Statistic	Limit
Indoor						
Total-Average alpha (dpm/100 cm^2)	57	-3.7	19.2	92.0	25.7	5,000
Removable alpha (dpm/100 cm^2)	73	0.2	4.2	15.4	6.5	1,000
Total-Average beta (dpm/100 cm^2)	57	300	286	745	739	5,000
Removable beta (dpm/100 cm^2)	73	3.4	3.9	12.5	9.4	1,000
Ambient Gamma Exposure Rate Corrected for Background (μR/h)*	57	-0.4	1.9	2.6	2.6	5
Outdoor						
Ambient Gamma Exposure Rate Corrected for Background (μR/h)**	126	-0.03	0.92	2.8	1.3	5

Note: No maximum alpha/beta activity spots were found.
N/T: Not Tested - too few measurements for statistical significance.
* Indoor gamma exposure rate corrected for background of 12.0 μR/h.
** Outdoor gamma exposure rate corrected for background of 13.0 μR/h.

Although the inspection test statistics calculated are less than acceptance limits, and thus, we conclude that the indoor area is acceptably clean, further understanding of these abnormally high standard deviations is necessary. Before any judgments can be made about the existence of residual radioactivity, we must investigate the probability plots to determine outliers in each distribution and to formulate an understanding of these greater variances.

Figures 7.1 through 7.7 show the distributions for alpha/beta measurements made inside T009. Total-average alpha activity, Figure 7.1, is far below the acceptance limit of 5000 dpm/100 cm^2. Figure 7.2 is the same as Figure 7.1 except the ordinate scale has been expanded to show a variation in alpha activity observed on the south wall and floor of room 124 (the old fuel storage vault). Although far less than the acceptance limit, there is clearly two distinct Gaussian distributions; Figure 7.2 is a model plot showing a slightly contaminated area. The average alpha activity in room 124 is 47.6 $\pm$ 36.3 dpm/100 cm^2, with a maximum of 92 dpm/100 cm^2. A beta correlation in this manner was not observed. Figure 7.3 shows that the level of removable alpha activity inside T009 is far less than the acceptance limit of 1000 dpm/100 cm^2. However, when these data are plotted on an expanded scale as in Figure 7.4, it is clear that slight removable alpha contamination exists. The outliers in the distribution correspond to slight alpha contamination in the cabinets of rooms 114 and 116, and from the floor in the fuel vault (room 124). Figures 7.2 and 7.4 show the power of these cumulative probability plots, even at very low levels of contamination. This amount of detectable activity is not hazardous, and is well below acceptance limits for unrestricted use. All of the cabinets in rooms 114 and 116 were surveyed "for indication" of alpha and beta activity. In all places surveyed (corners, ledges, and shelves) No Detectable Activity was found. Slight removable contamination found in those cabinets is no cause for further inspection and investigation at these low levels.

Figure 7.5 shows the total-average beta activity distribution for the same indoor measurement locations. Measurements follow a Gaussian distribution. No single measurement exceeded 50% of the acceptance limit. Figure 7.6 shows that no statistically significant removable beta contamination is present. However, in order to observe the deviations in detail, the y-axis has been expanded for removable beta activity and shown in Figure 7.7. No identifiable trends or statistically significant beta activity is found.

Figure 7.1 Total-Average Alpha Activity Inside Building T009

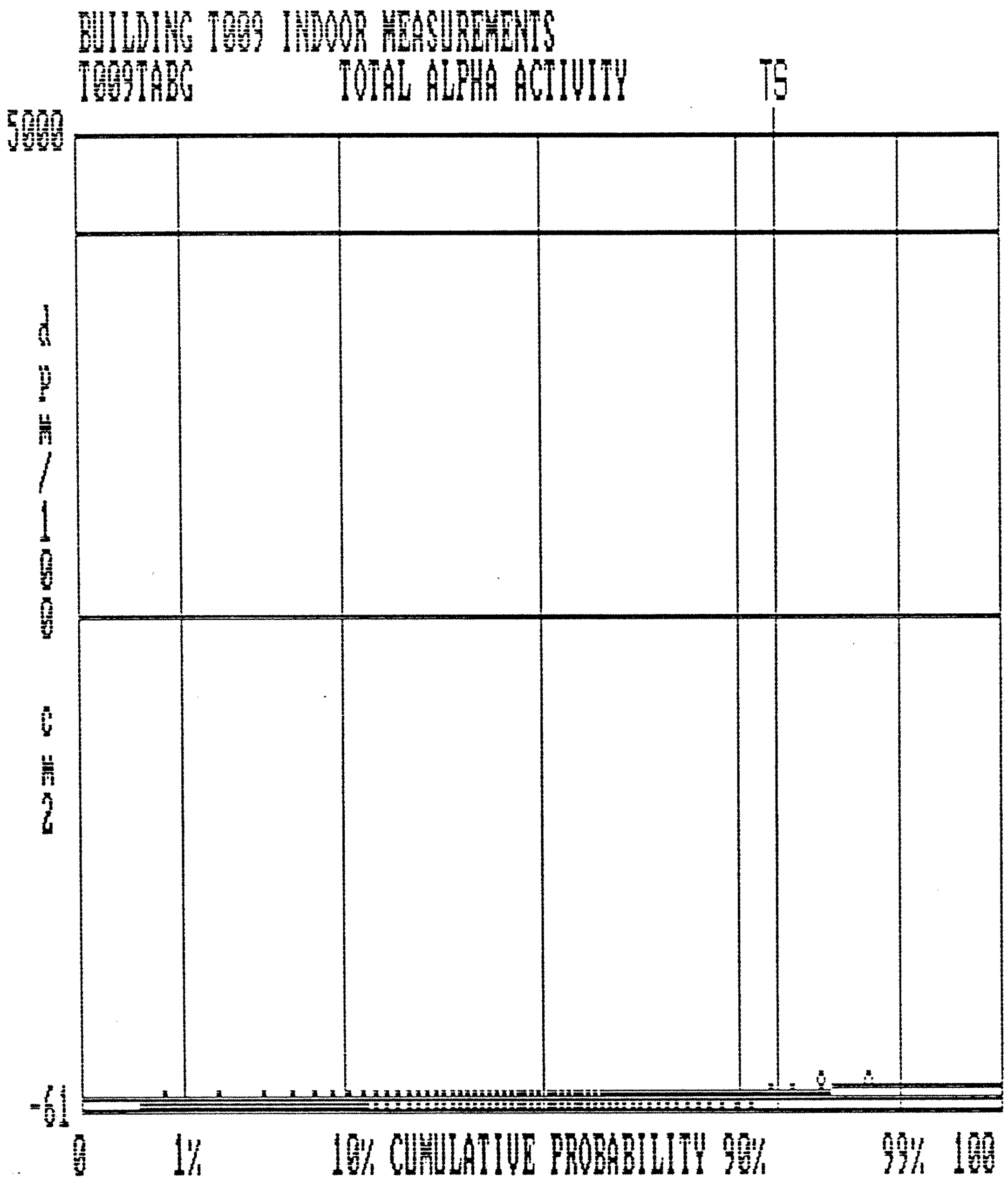


Figure 7.2 Total-Average Alpha Activity Inside Building T009
(Expanded Scale)

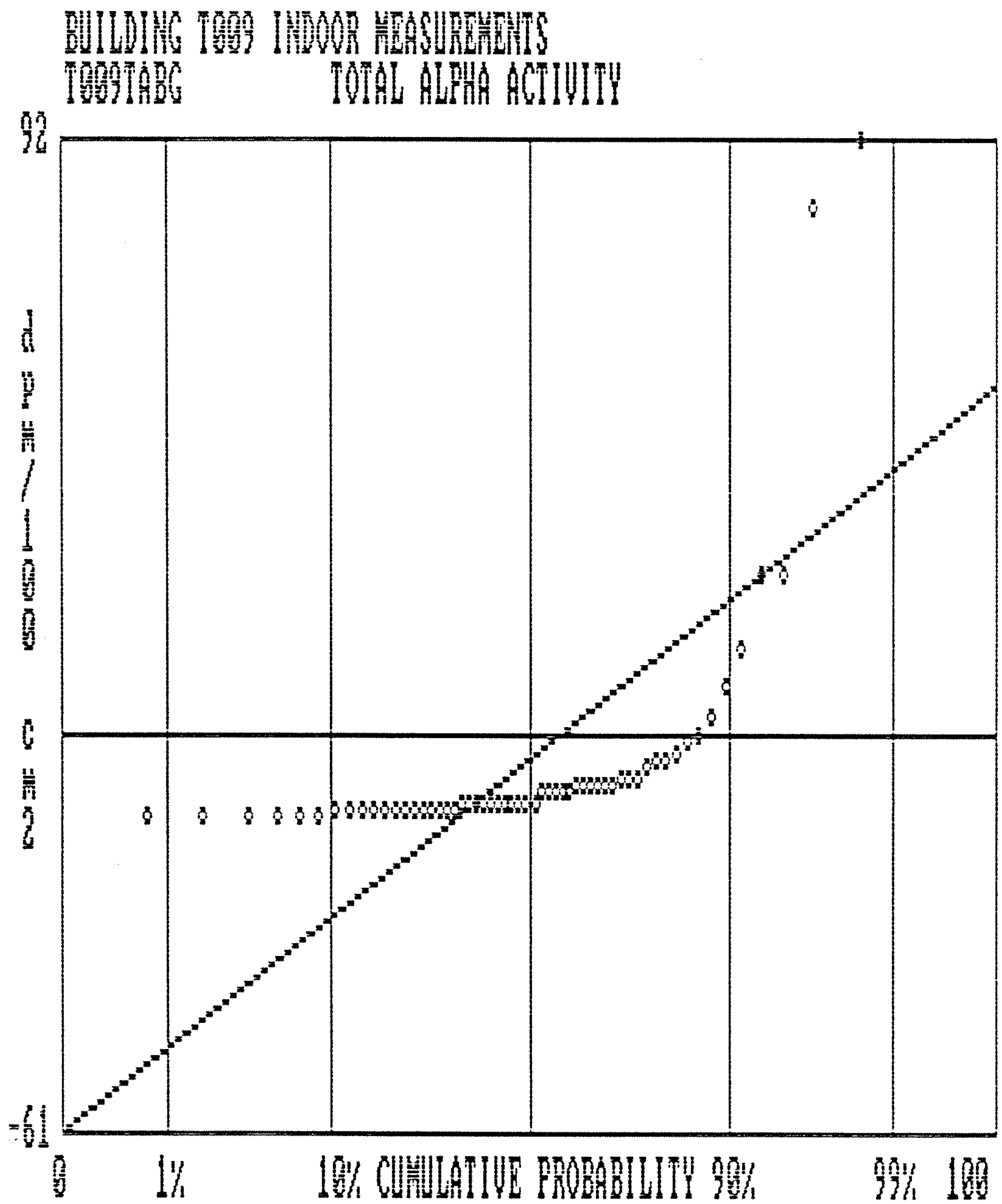


Figure 7.3 Removable Alpha Activity Inside Building T009

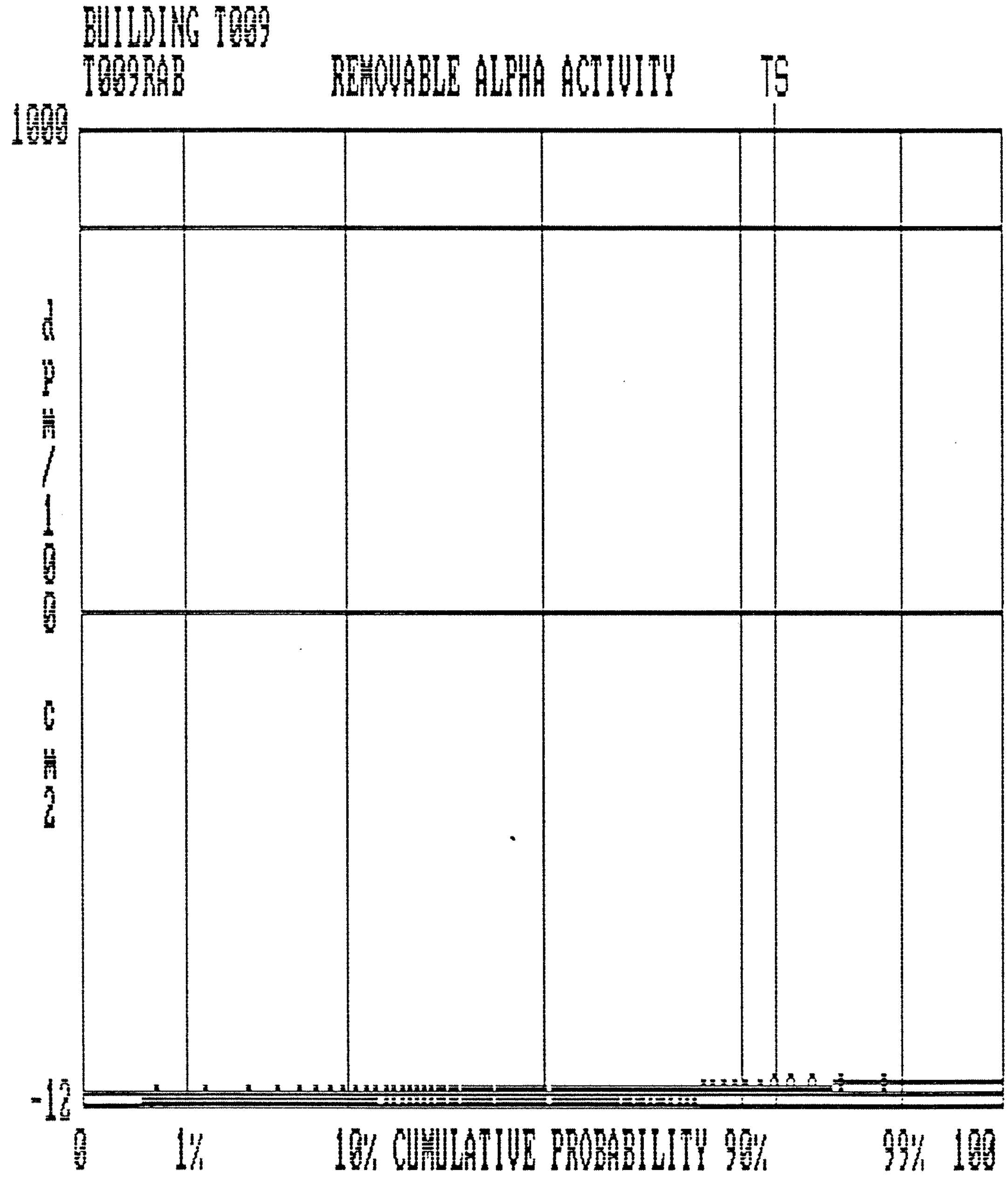


Figure 7.4 Removable Alpha Activity Inside Building T009
(Expanded Scale)

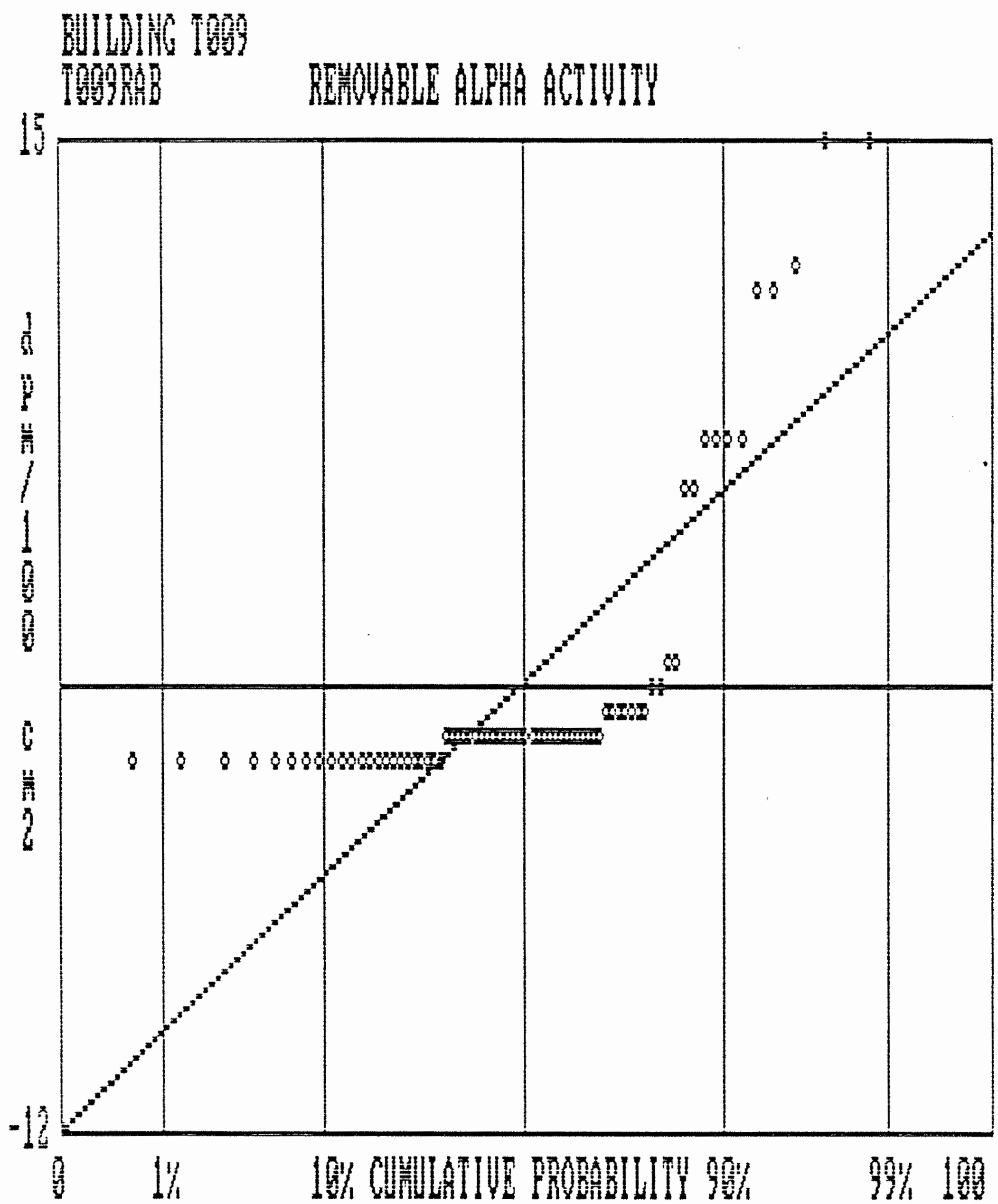


Figure 7.5 Total-Average Beta Activity Inside Building T009

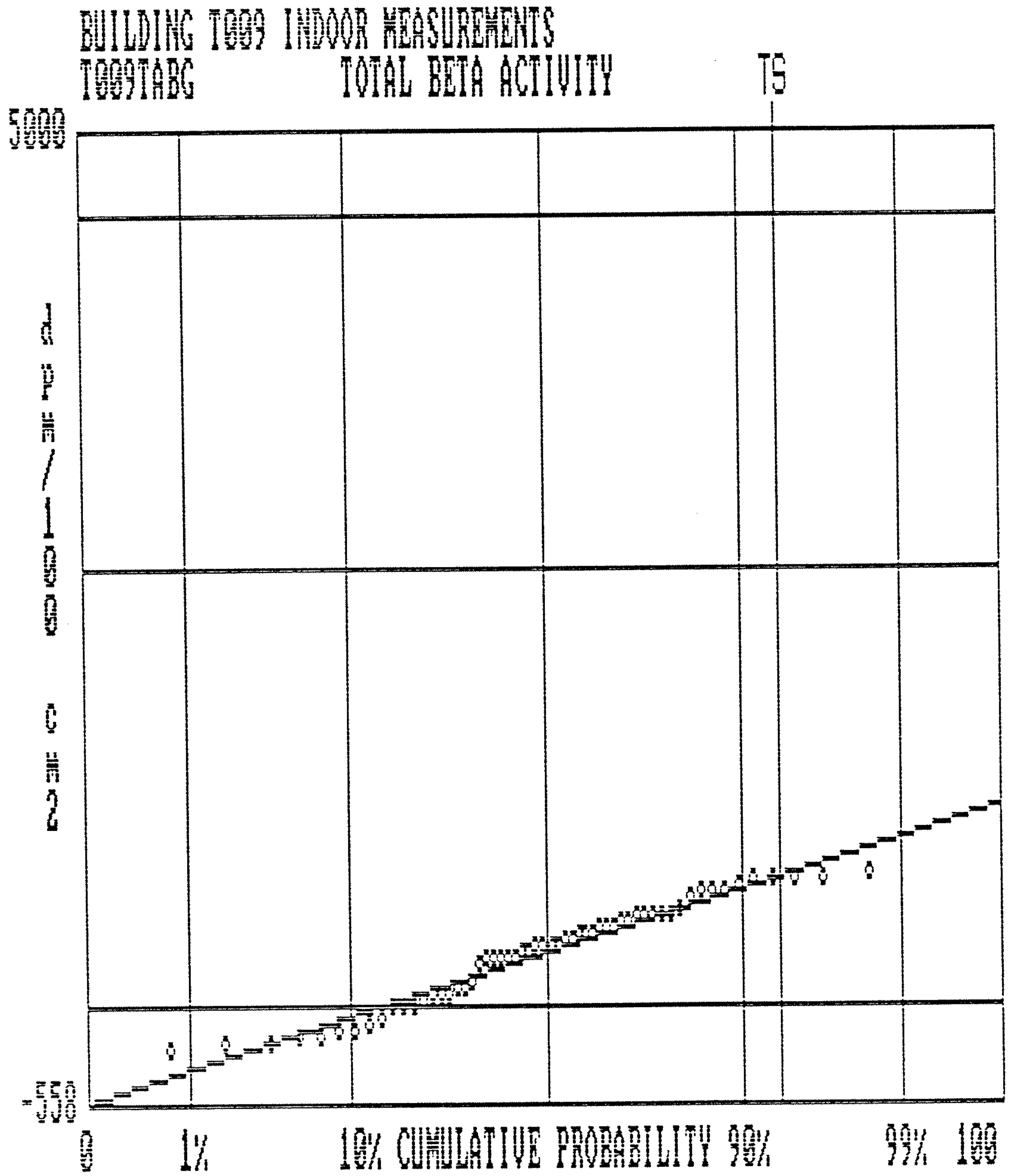


Figure 7.6 Removable Beta Activity Inside Building T009

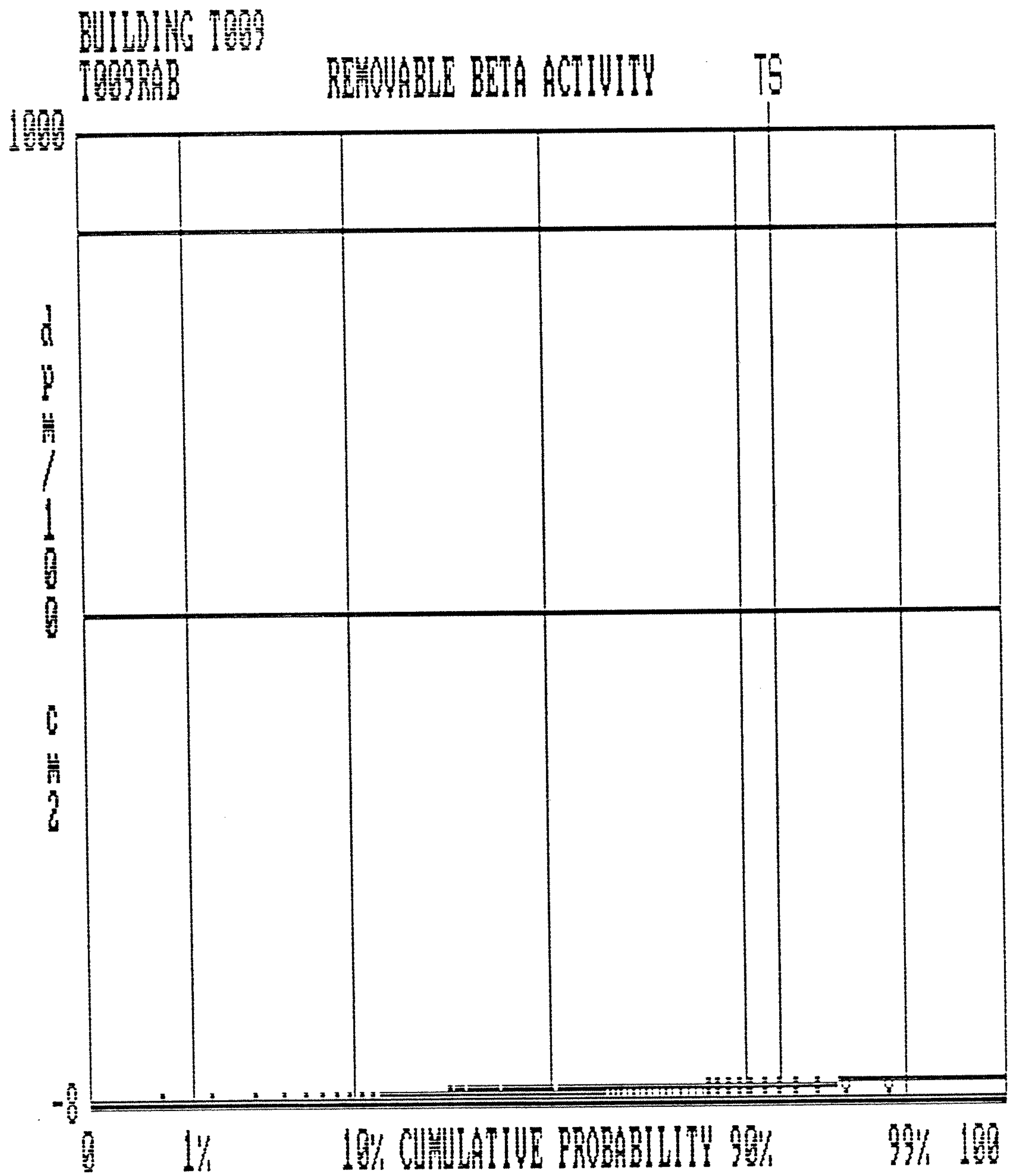
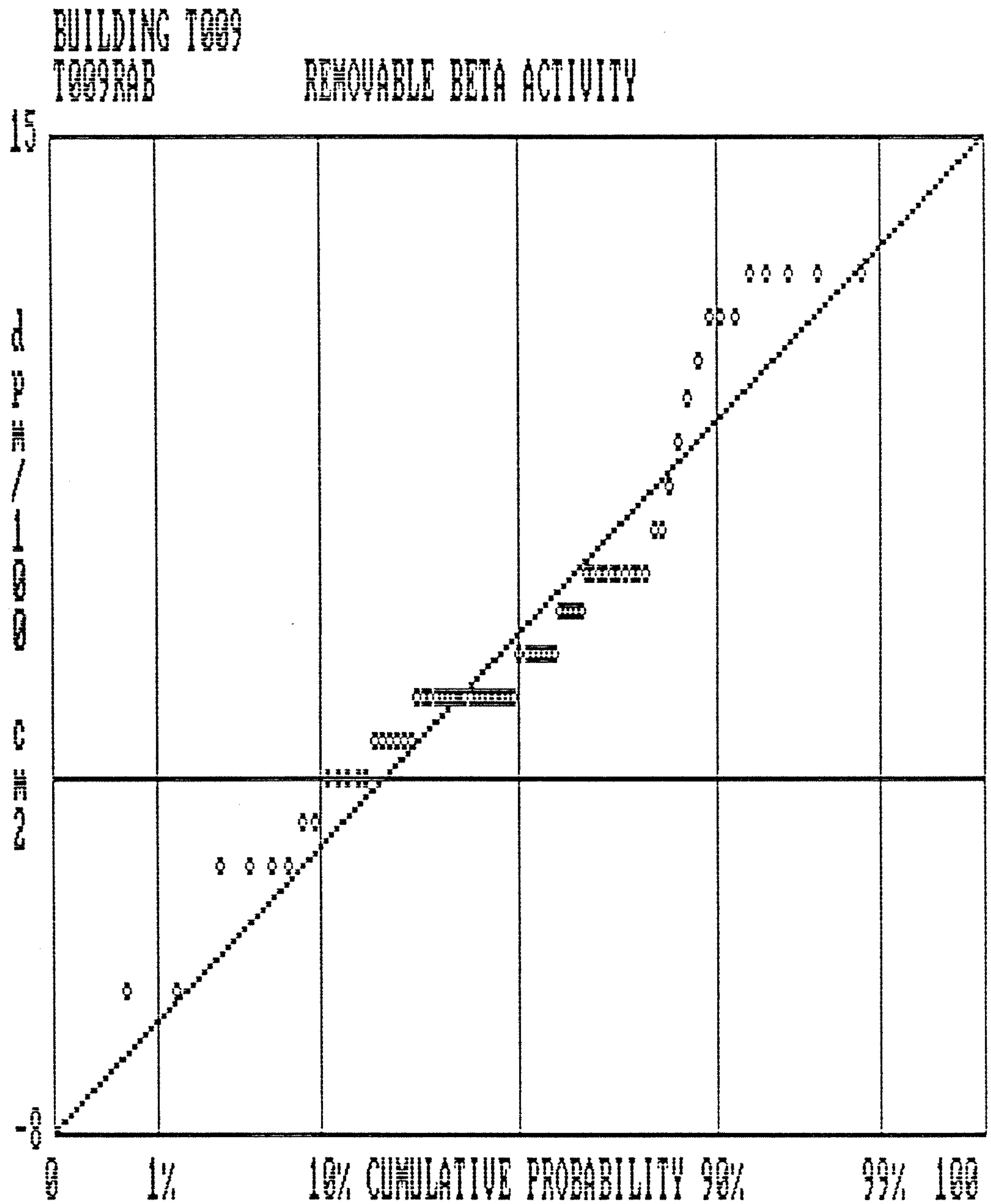


Figure 7.7 Removable Beta Activity Inside Building T009
(Expanded Scale)



Although some deviations were observed for total-average and removable alpha activity, they occurred far below the acceptance limit. Measurement values are presented in Appendix C. Survey locations are shown in Appendix D. Probability plots suggest that no unsurveyed areas would exceed acceptance limits had they been surveyed. This is a consumer's risk of acceptance of 0.10 at an LTPD of 10%. Further radiological investigation is not necessary.

7.3.2 Ambient Gamma Exposure Rate Measurements

Ambient gamma exposure rate measurements were acquired for 1 min. each in the same 57 locations as alpha/beta activity. Table 7.1 in the previous section shows the background-corrected results of these measurements to be less than acceptance criteria for unrestricted use.

Figure 7.8 shows the statistical distribution of gross-total gamma exposure rate measurements plotted against cumulative probability for indoor locations. A mean exposure rate of 11.4 $\pm$ 1.92 μR/h is typical of indoor locations such as Building T009. Most importantly, no outliers or trends indicating a contaminated area are observed. Table 7.3 shows a comparison of these results against results from "natural background" areas.

Figure 7.9 shows the same data set, in which case a correction for "ambient background" was made uniformly to each measurement value. 12.0 μR/h was used for "background" subtraction, corresponding to the median value of gross-total measurements shown in Figure 7.8. Deviations observed in the measurements because building structures are pronounced in this figure because the ordinate scale has been expanded. Table 7.1 shows that an average of -0.36 $\pm$ 1.92 μR/h is less than the 5 μR/h acceptance limit and all action levels. The inspection test statistic, 2.59 μR/h, is less than 5 μR/h. We accept this sample lot as uncontaminated by this inspection method.

Figure 7.8 Total-Gross Ambient Gamma Exposure Rates
Measured Inside Building T009

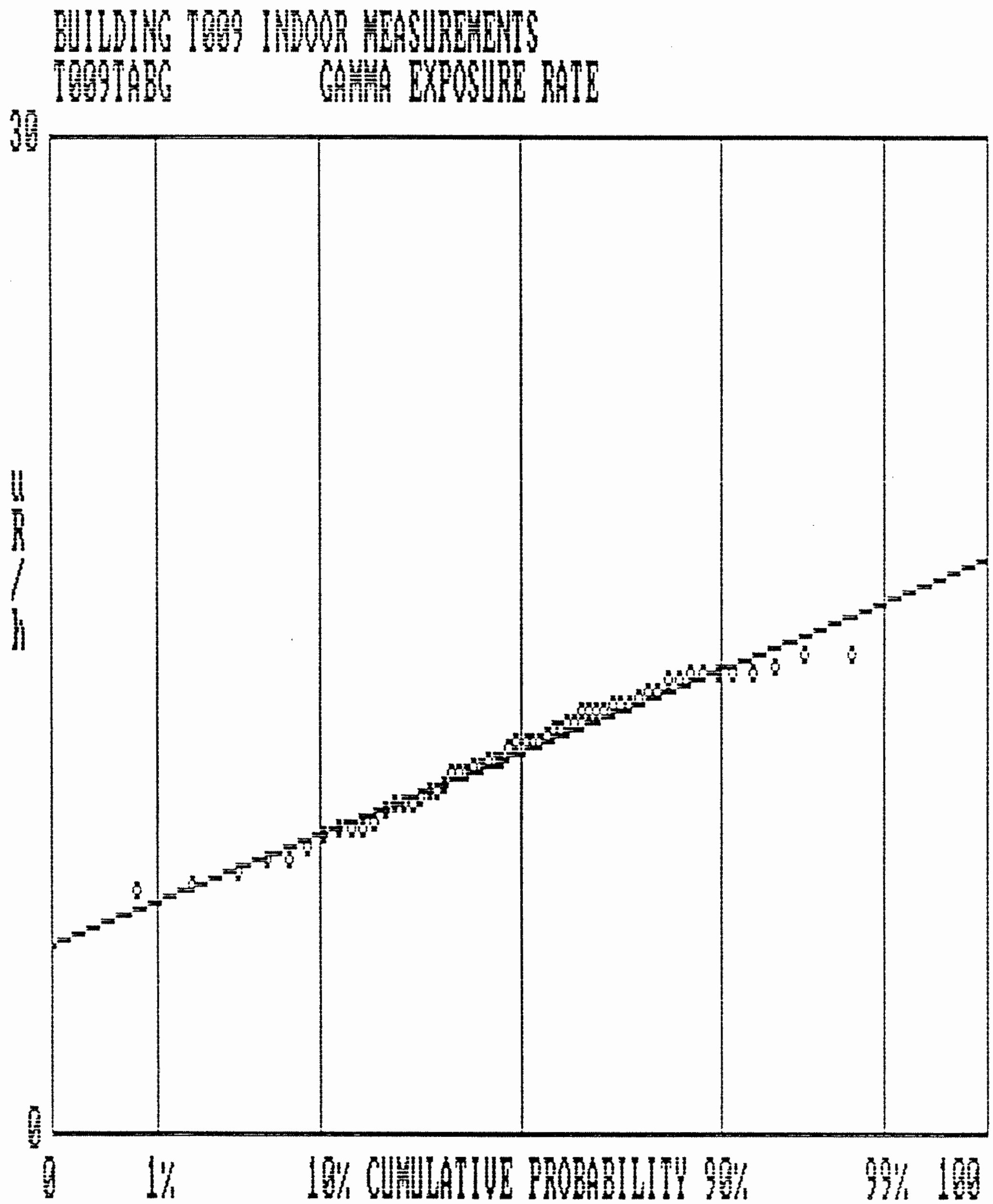
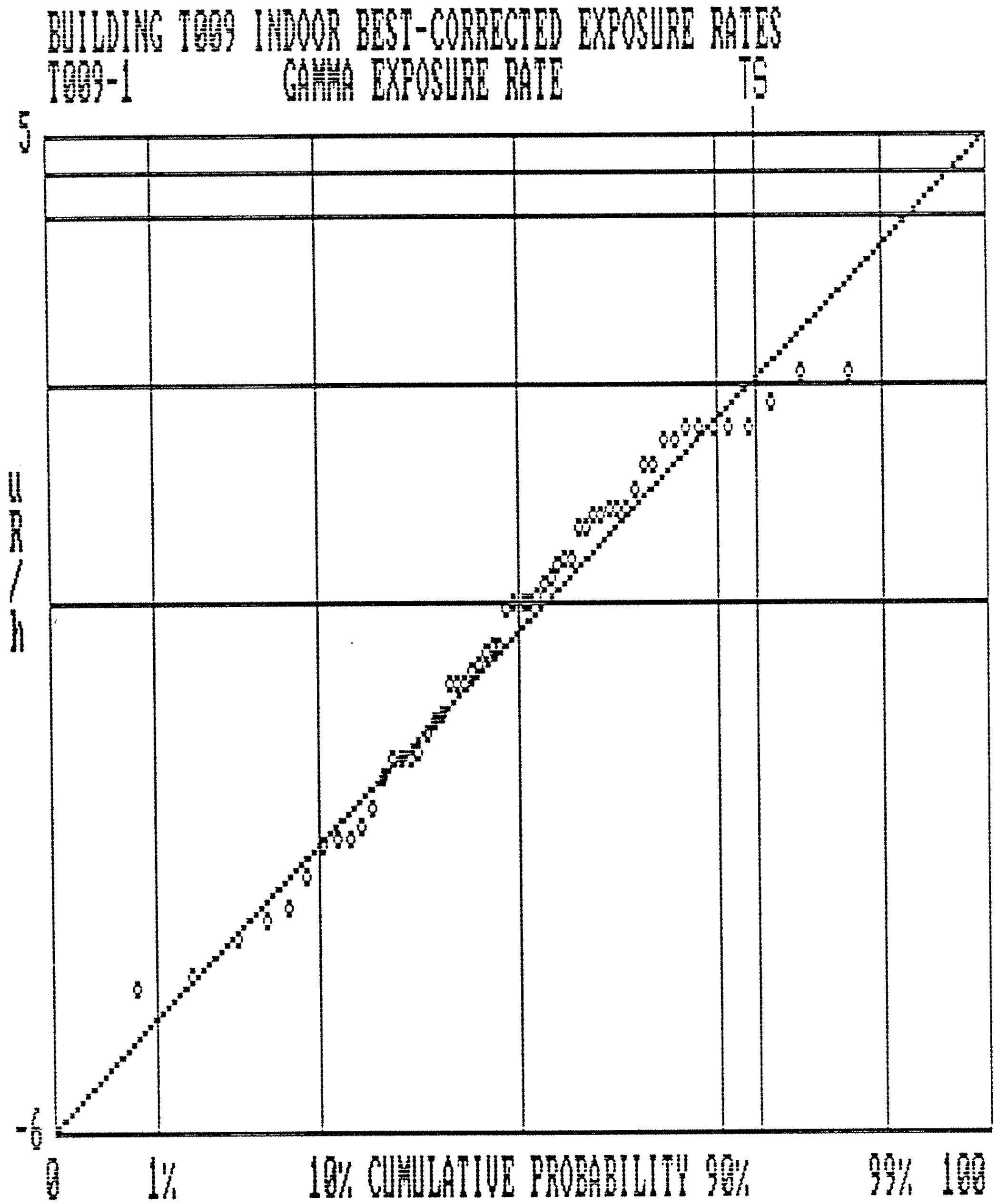


Figure 7.9 Background-Corrected Ambient Gamma Exposure Rate Measured Inside Building T009



7.4 Results of Special Surveys and Sample Analysis

Special building features and miscellaneous components were surveyed with hand held alpha and beta probes "for indication" of radioactivity. Additionally, samples of sludge, water, oxidized iron shavings, and residue from drain clean-outs, shower drains, catch pans in the machine shop, and the eastern R/A holdup tank were collected and analyzed for gamma emitters by gamma spectrometry. This section presents the results of these inspections.

Alpha/beta surveys were performed in the cabinets of rooms 114 and 116, along wall coving in rooms considered for this survey, sinks, shower drains, tile cracks, door jambs, and around machining tools (particularly in the grease residues). Results of this "search for radioactivity" show in all cases, No Detectable Activity.

As required by the Survey Plan (References 4, 29 and Appendix F), samples were collected and analyzed by gamma spectrometry. Table 7.2 shows the results of these analyses. No radioactivity was detected in any of the samples except for the SGR holdup tank. Section 7.4.2 and Table 7.3 show the specific results for sludge collected inside the tank.

7.4.1 OMR Exhaust Filter Plenum (West Side)

Also included as part of this radiological survey was an inspection for radioactivity in the OMR filter plenum. The OMR exhaust system has not been in use since the early 1970s -- it has been mothballed. Air flow out of the building was directed through a duct into two parallel banks of pre-filters and HEPA filters. Access doors are located on each side of the unit to allow changing each set of filters. The pair of filter banks (each having a pre and HEPA filter) was surveyed for total and removable alpha/beta radiation. The entire accessible surface of all filters was frisked with alpha and beta probes "for indication." Smears were collected in grooves and cracks. A maslin wipe of the northern HEPA filter was analyzed by gamma spectrometry. In all cases, No Detectable Activity was found.

Table 7.2 Results of Miscellaneous Samples Collected and Analyzed for Radioactivity

Sample	Result
(1) Crud and grease collected from catch pans located below machining tools in room 118 (the machine shop).	No gamma peaks identified.
(2) Crud collected from sink trap of room 108 (the janitor's closet).	No gamma peaks identified.
(3) Crud collected from shower drain in room 122 (the change room).	No gamma peaks identified.
(4) Crud collected from sink trap in room 121 (laboratory); this sink drains to the eastern R/A holdup tank.	No gamma peaks identified.
(5) SGR R/A Holdup Tank on east side of building (4 samples).	
1. About 1 liter of dry sludge from inside tank (put in poly-bottle).	* Beta probe placed against poly-bottle read 500 cpm-β (about 10,000 dpm/100 cm^2, or about 1 nCi). * Gamma spectrometry of the sample showed Ra-226, U-235, U-238, Th-232, Cs-137, Cs-134, and Mn-54. * Inside the tank is contaminated.
2. About 100 ml of damp soil and debris collected from east (high side) of pit.	* Beta probe showed NDA. * No gamma peaks identified.
3. About 500 ml of damp soil and water collected from west (low side) of pit.	* Beta probe showed NDA. * No gamma peaks identified.
4. About 500 ml of water/ sludge collected from sump.	* Beta probe showed NDA. * No gamma peaks identified.

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7.4.2 SGR Radioactive Liquid Holdup Tank (Inactive)

A 1350 g sample of dry scale, rust, sludge, and dirt was collected from the bottum inside circumference of the inactive SGR holdup tank. The sample was placed in a poly-bottle for analysis by gamma spectrometry. Results of this analysis are shown in Table 7.3.

Table 7.3 Results of Gamma Spectrometry for SGR Holdup Tank Sludge Sample

Decay Chain	Isotope Identified	Gamma Ray Energy (keV)	Activity Concentration (pCi/g)
U-238	Th-234	63.2	7.5
	Ra-226	186	18.0*
U-235	U-235	185.6	0.81*
Th-232	Pb-212	74.7	25.2**
	Ac-228	129.1	11.2**
	Ac-228	209.4	12.6
	Pb-212	238.5	20.1
	Ac-228	270.3	13.6
	Tl-208	277.3	19.2***
	Pb-212	300.0	17.7
	Ac-228	328.0	15.2
	Ac-228	338.0	18.1
	Ac-228	409.4	16.5
	Ac-228	463.0	18.4
	Tl-208	510.6	15.8***
	Tl-208	583.0	17.8***
	Bi-212	727.1	12.1
	Ac-228	794.8	16.4
	Tl-208	860.5	20.8***
	Ac-228	911.0	19.9
	Ac-228	964.5, 968.8	19.2
Cs-137	Cs-137	661.6	2.6

* U-235 and U-238 activities calculated assuming natural uranium. It is possible that only U-238 is present (depleted uranium).

** May be incorrectly greater because of low energy photon in Compton continuum region.

*** Corrected for 36% branching ratio from Bi-212.

It is evident from the analysis that Th-232 is present at an activity concentration of about 17.2 ± 3.5 pCi/g; 18 daughter photopeaks of Th-232 were observed. U-238 is also present, but its activity concentration is not well known. U-235 may also be present. Cs-137, a fission product is definitely present with an activity concentration of about 2.6 pCi/g, about 3 to 10 times greater than natural soil. No activation products were identified. When this tank is removed, it should be handled as radioactive material. Its current configuration is not a hazard, nor is it in a position of spreading contamination to surrounding areas.

7.5 Building Exterior (Northwestern Area)

Ambient gamma exposure rate measurements were made at 126 outdoor locations in the northwestern area. Appendix C shows the data sets. Appendix D shows measurement locations. Table 7.4 shows the computed statistics for this sampling lot compared against data from three independent outdoor areas where no radioactive material was ever handled, used, or stored. These outdoor areas are considered "natural background" at SSFL. This type of comparison is necessary for two reasons: 1) to demonstrate variability of "natural background" gamma radiation at SSFL; and 2) to estimate "natural background" at SSFL because the limits for unrestricted-use by which we use to demonstrate an "acceptable" area are based on above "background" criteria. So, unless we confidently know what "ambient 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. For the indoor survey, it would have been better to measure "ambient background" inside a facility of similar characteristics, but one was not available.

Descriptive statistics presented in Table 7.4 show that average exposure rates calculated for each test-area at Building T009 are slightly less than the three "natural background," control-group areas. Standard deviations of each test-area are greater than that observed for "natural background" in natural terrain. Greater variability observed in these test-areas is attributed to interference of exposure rate due to equipment items and nearby building materials. These properties make for a non-uniform

Table 7.4 Natural Background Gamma Radiation at SSFL Compared to Survey Data

Location	No. of Measurements	Mean Exposure Rate (μR/h)	Expected Standard Deviation at the Mean (μR/h)*	Standard Deviation of the Distribution (μR/h)**	Range μR/h
Interior	57	11.4	0.23	1.92	7.2
Exterior (Northwest Area)	126	13.0	0.24	0.92	5.8
Background					
Building 309 Area (1/19/88)	36	15.6	0.27	0.82	3.4
Well #13 Road (Dirt) (4/29/88)	43	16.2	0.27	0.49	2.2
Incinerator Road (Dirt) (4/29/88)	35	14.0	0.25	0.36	1.4

* The expected standard deviation at the mean is calculated based on counting statistics, equation 4.2.

** The standard deviation of the data points accounts for dispersion in the measurements, equation 4.7.

inspection lot. The range of measurements observed for these test cases is also greater than "natural background." Again, this observation is due to non-uniform deviations in exposure rate as a function of location, e.g. near a wall, indoors, partially indoors with an over-hang, outdoors, or near miscellaneous components. By observation of these descriptive statistics, Building T009 appears uncontaminated. However, before any judgments can be made about the existence of residual contamination, we must investigate the probability plots to determine outliers in each distribution and to formulate an understanding of the greater variations and ranges observed in the outdoor test-case.

Because the "natural background" gamma-radiation environment is quite variable at SSFL and because the limits for unrestricted use are based on limits above background, further demonstration of this natural variability is necessary. For comparison against test-area measurements, three independent areas were surveyed, all in locations where no radioactive material was ever handled, used, stored, or disposed. All three areas are located on the eastern side of SSFL: (1) Area surrounding building 309 on Area I Road; (2) well #13 Road; and (3) Incinerator Road. Table 7.4 shows the results of these measurements. These "natural background" areas are fairly similar in characteristics and topography to the outdoor inspected areas for this report, although the inspected lot at T009 is about 60% pavement. The purpose these "background" distributions serve is to show "natural" variability of gamma radiation on natural terrain at SSFL to be used as a guide for comparing sample lot results.

Figures 7.10 through 7.12 are probability plots of these three independent "background" areas. At least 30 measurements were made in each area on the same day. In the plots, a uniform background rate (unbiased by spatial effects), would appear as a straight line with a minimal slope. That slope would show that 1 standard deviation from the mean of values would be equivalent to the mean-value standard deviation (i.e. the square root of the counts of the mean multiplied by an appropriate efficiency factor). If this was the case, the values in columns 4 and 5 of Table 7.4 would be equivalent. Obviously, this ideal condition is impossible to achieve in this terrain at SSFL. All three plots show model Gaussian distributions, but with greater variability than would be expected from ideal laboratory-level measurements. Variability is greatest near Building 309.

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 each measurement is observable in this case. 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. "Natural background" variability approaches the NRC limit.

Figure 7.10 Ambient Gamma Radiation at Area Surrounding Building 309
(Background Distribution)

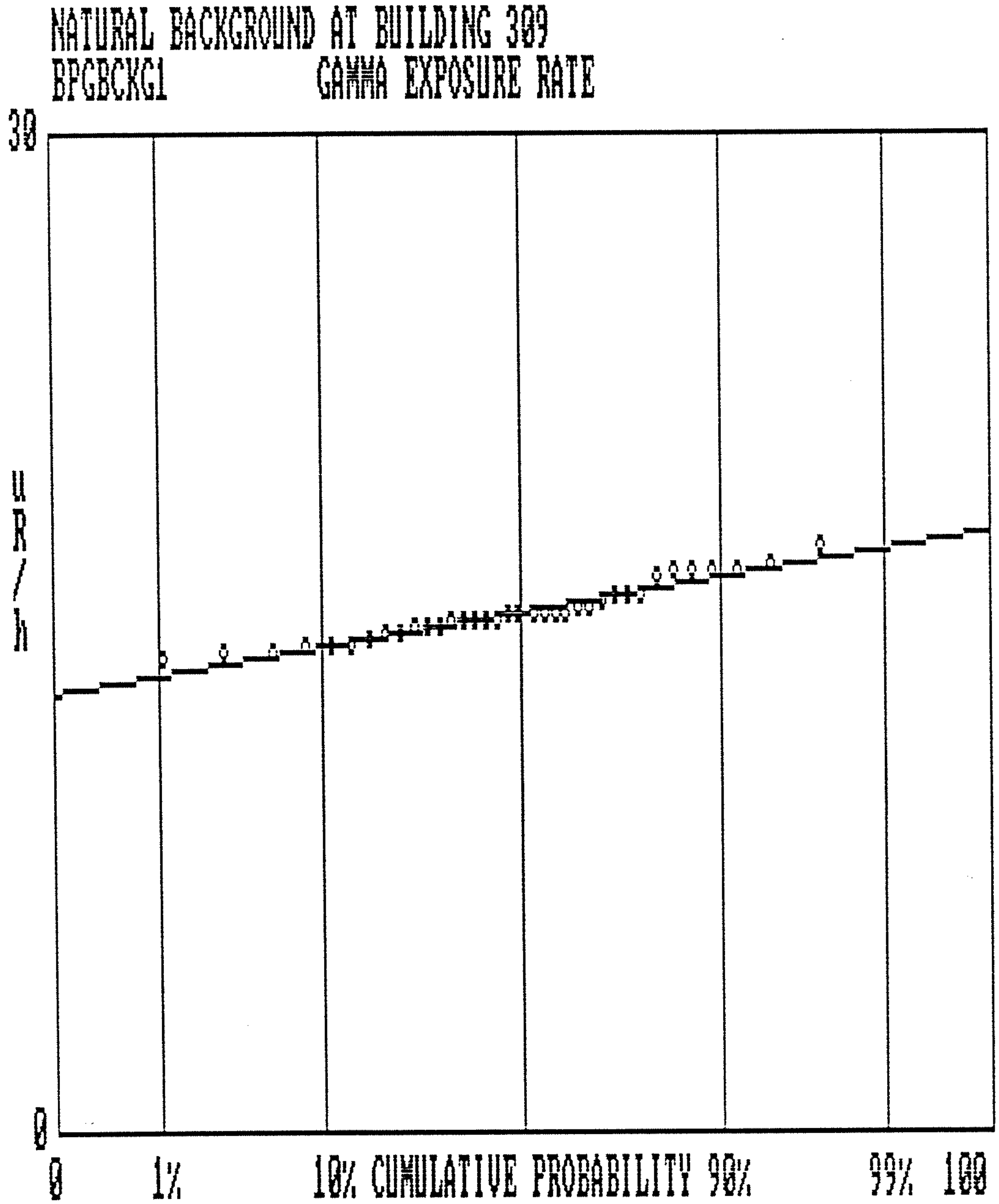


Figure 7.11 Ambient Gamma Radiation at Area Well #13 Road
(Background Distribution)

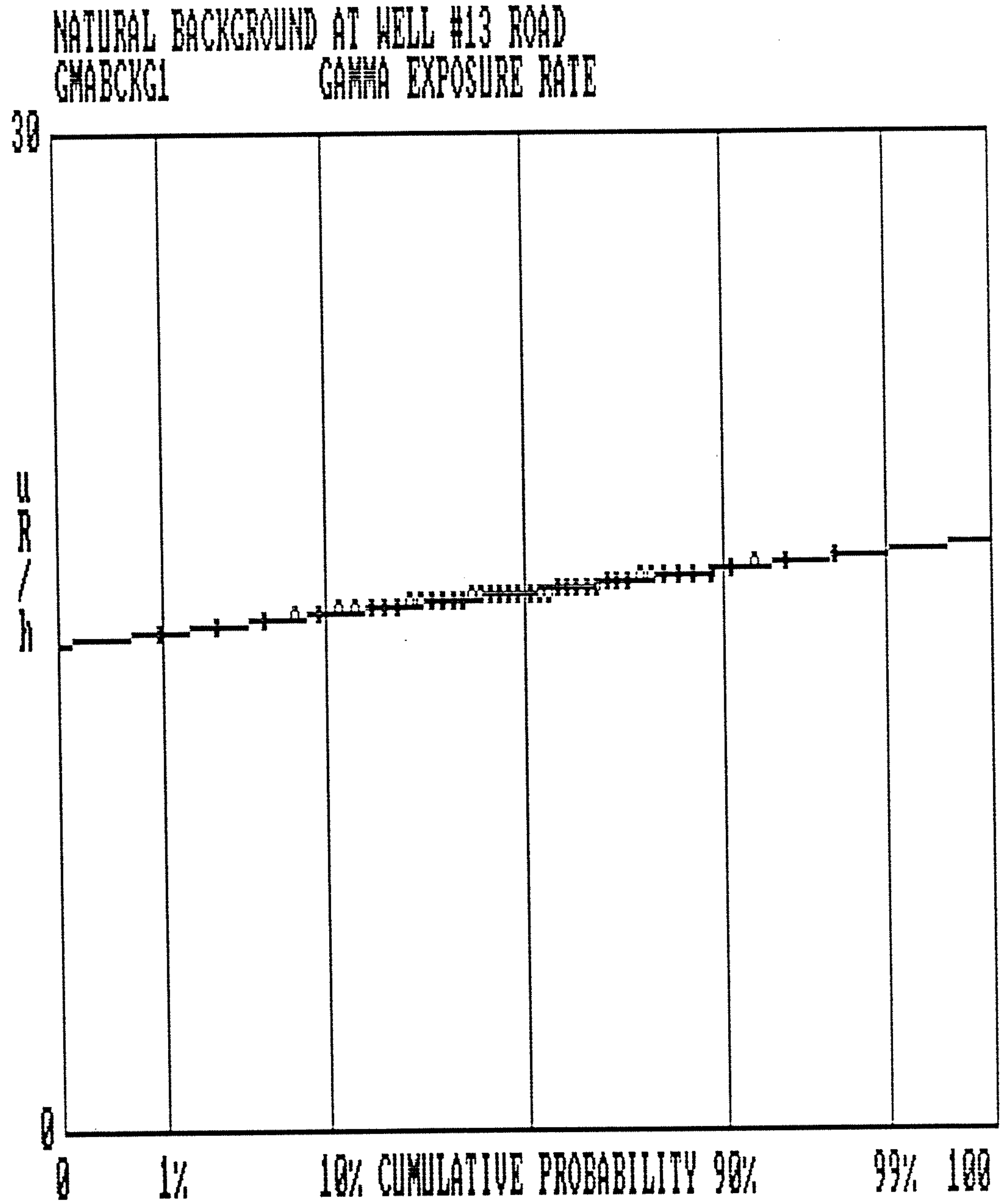
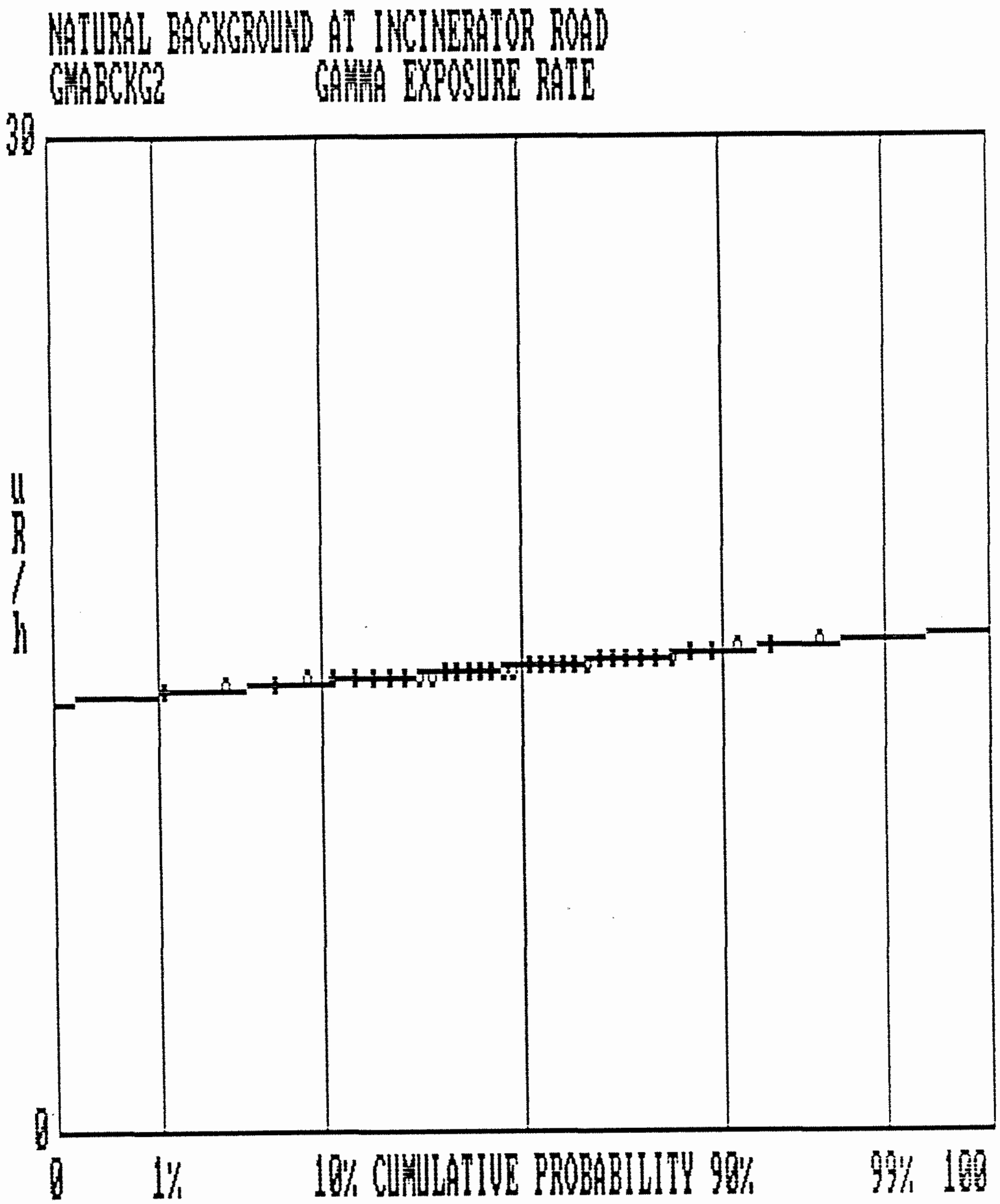


Figure 7.12 Ambient Gamma Radiation at Incinerator Road (Background Distribution)



This "natural background" analysis shows the great difficulty in assessing whether an area is contaminated based on the NRC acceptance limit of 5 μR/h above background. The DOE limit of 20 μR/h is more reasonable. Natural gamma radiation is significantly variable at SSFL. We'll now compare this "natural" variability against the test-area measurements presented in this report. We'll see that "natural" variability is even more significant inside and near facilities.

Figure 7.13 shows the statistical distribution of gross-total gamma exposure rate measurements plotted against cumulative probability for the outside area. An average of 13.0 $\pm$ 0.24 μR/h is in good agreement with results from the "natural background" areas, Table 7.4. No trends indicating a contaminated area are observed.

Figure 7.14 shows the same data set, in which case a correction for "ambient background" was made uniformly to each measurement value. 13.0 μR/h was used for "background" subtraction, corresponding to the median measurement value of gross-total measurements shown in Figure 7.13. Slight deviations observed in the measurements because of equipment stored in the paved yard are pronounced in this figure because the ordinate scale has been expanded. The thee outliers observed at the high end (one is above the 50% Reinspection level) were all acquired next to the building on the west side near the exit doors. This 1 μR/h increase can be expected when a measurement is made adjacent to a facility. Table 7.1 shows that an average of -0.03 $\pm$ 0.92 is less than the 5 μR/h acceptance limit and all action levels. The inspection test statistic, 1.3 μR/h, is less than 5 μR/h. We accept this sample lot as uncontaminated by this inspection method.

Figure 7.13 Total-Gross Ambient Gamma Exposure Rates Measured Outside Building T009 in Northwestern Area

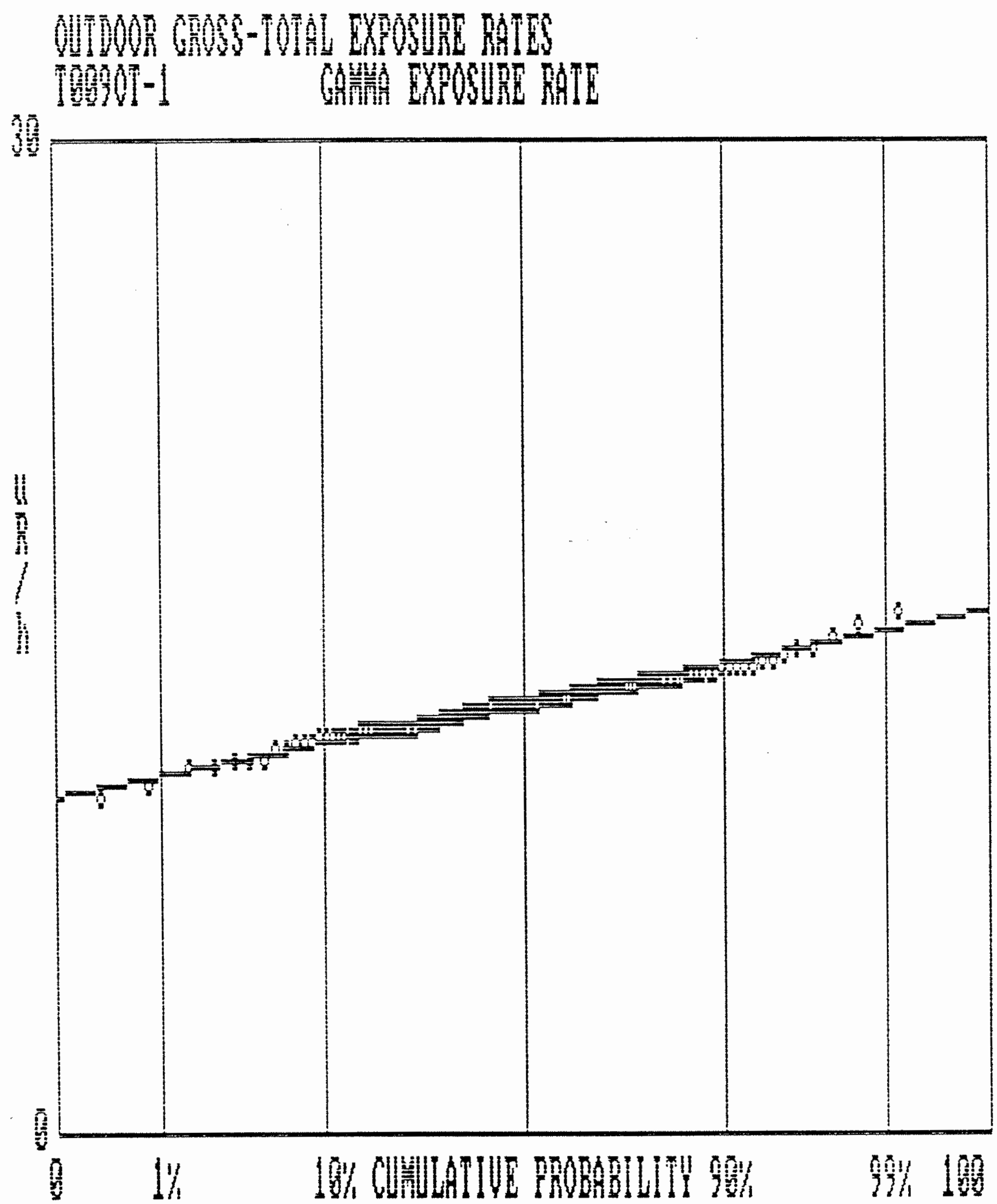
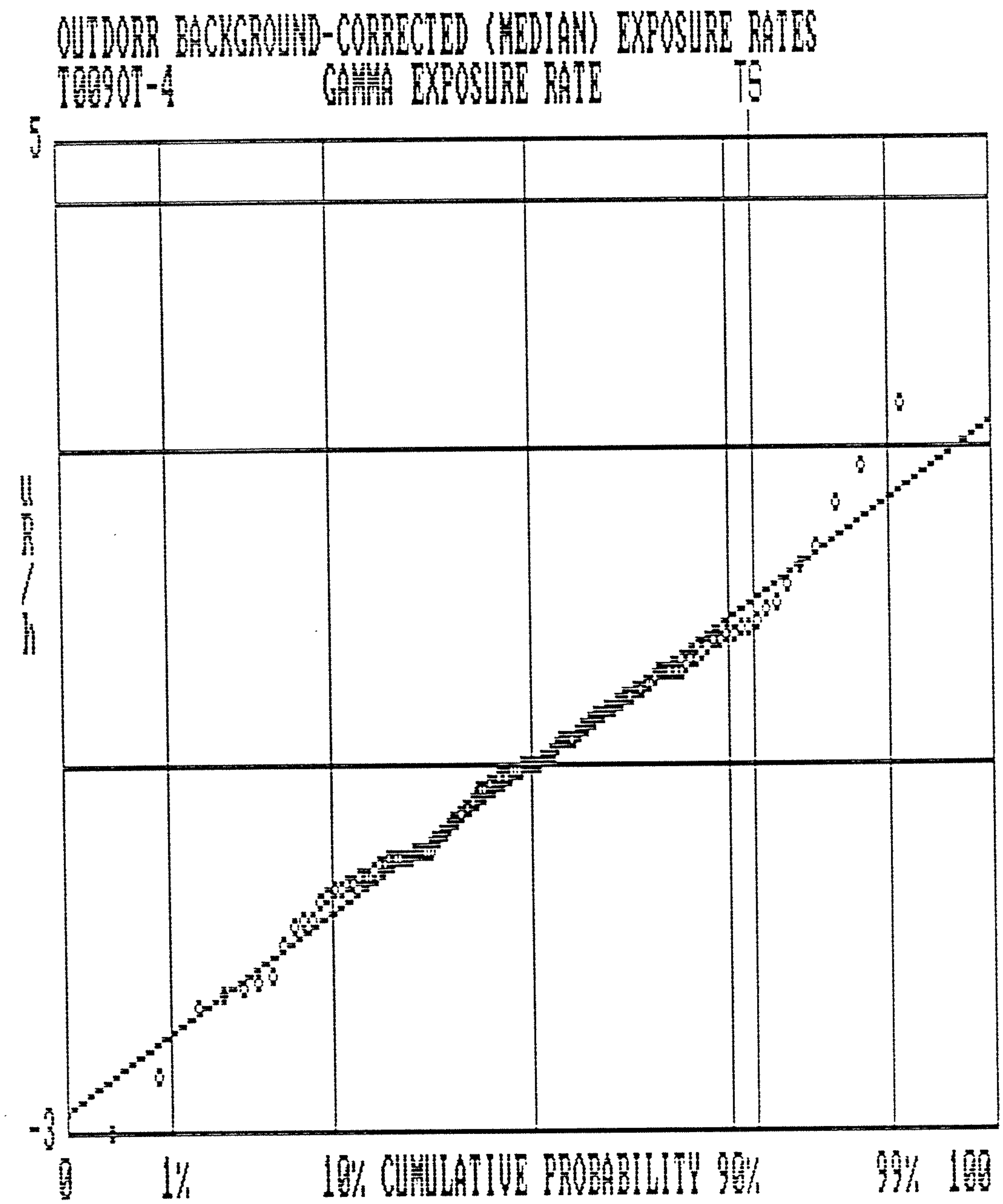


Figure 7.14 Background-Corrected Ambient Gamma Exposure Rate Measured Outside Building T009 in Northwestern Area



8.0 CONCLUSIONS

The OMR-side and outside northwestern area of Building T009 were inspected for radioactive contaminants. Total and removable alpha/beta radioactivity measurements and gamma exposure rate measurements were made on a 3-m square sampling grid inside the facility. 1 m^2 was surveyed for 5 min. in each 9-m^2 area for total alpha/beta activity. A 100 cm^2 smear from that 1 m^2 was collected to measure removable activity. An exposure rate measurement was acquired 1 m from the floor. Outdoor exposure rate measurements were made on a 6-m square sampling grid. Total alpha/beta surface activity was checked "for indication" on interior special features. Smears and alpha/beta surveys were performed on cabinets, sinks, drains, the OMR exhaust and ventilation system, and the OMR filter plenum -- these systems are more likely to have retained residual contamination. Total and removable alpha/beta activity measurements plotted against cumulative probability are far less than acceptance limits; however, some perturbations in the Gaussian distribution were observed. Slight alpha contamination was detected in the old fuel vault and in the cabinets in rooms 114 and 116. Maximum total-average alpha measurements showed (only in the fuel vault) 92 dpm/100 cm^2 -- about 50 times below the acceptance limit. Removable alpha activity in the cabinets showed a maximum 15 dpm/100 cm^2 -- about 70 times below the acceptance limit. Additionally, analysis of a sludge sample removed from the SGR radioactive material holdup tank showed the presence of fission products, U-238, Th-232, and possibly U-235. In all cases, the contamination found was slight and far below hazard levels and unrestricted-use acceptance limits. All alpha/beta surface activity measurements made "for indication" show No Detectable Activity above ambient background.

Based on these statistical distributions of exposure rate measurements corrected for what we found to be "ambient background" in each sample lot, and on total and removable alpha/beta activity, we conclude through inspection by variables, that all locations surveyed pass criteria for unrestricted use of the facility. This statistical test assumes a consumer's risk of acceptance of 0.1 at an LTPD of 10%. No further inspection is required in these locations based on this statistical approach. The SGR holdup tank does contain slight quantities of radioactive material. It should be removed and handled as radioactive waste, under the supervision of a health physicist.

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APPENDIX A. DESCRIPTION OF NUCLEAR INSTRUMENTATION

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

A.1 Gamma Spectrometry Analyzer

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

A.2 Gross Alpha/Beta Automatic Proportional Counter

Smear test wipes were analyzed for gross alpha and gross beta radioactivity with a Canberra Industries Model 2201 Ultra Low Level Counting System. Model 2201 consists of a highly efficient gas-flow sample detector operating in the proportional gas amplification region. The system detects radiation in a 2π geometry using P-10 gas (90% methane, 10% argon). A cosmic-ray detector provides coincidence event cancellation to reduce instrument background. The two detectors operate in an anticoincidence mode to reduce the count rate due to cosmic-ray events. When cosmic-ray or background events occur, the input circuit to the count integrator is gated off and the simultaneous event is discarded. Thus, only true alpha and/or beta radiation events are recorded. The detectors are coupled through dual Model 2006A preamplifiers to a Model 2015A system amplifier then through a Model 2209A coincidence analyzer to the alpha or beta event scaling unit. The Series 2201 has a sample capacity of 99 samples contained in a magazine designed to accept sample planchets having a 2-inch diameter. Calibration of the sample detector for alpha and for beta radiation on smear-wipes is done with NBS traceable certified thorium-230 (alpha) and technicium-99 (beta) radiation sources having a configuration essentially equivalent to that of the smear wipes.

A.3 Portable Instruments

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

A Ludlum model 43-1 alpha scintillation detector was coupled to one 2220 for alpha contamination measurements. The scintillator is ZnS(Ag). The window (0.8 mg/cm^2) is aluminized mylar with an active area of about 72 cm^2. Background for this probe is less than 2 counts per 5 minutes. Efficiency for Pu-239 or Th-230 alpha particles is between 25% and 30%.

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

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

A Ludlum model 12 count-ratemeter was coupled to a Ludlum model 43-1 alpha scintillator to detect alpha radiation. This instrument is best suited "for indication" determinations.

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

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APPENDIX B. COPY OF DOE REPORT,

"GUIDELINES FOR RESIDUAL RADIOACTIVITY AT

FUSRAP AND REMOTE SFMP SITES," March, 1985

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

MAR 05 1985

CORRESPONDENCE

Addressees

GUIDELINES FOR RESIDUAL RADIOACTIVITY AT FUSRAP AND REMOTE SFMP SITES

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

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

As a result of a recent court decision, the Environmental Protection Agency (EPA) has issued 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, Jr.

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

SFMPO:PFXD

Attachment:
As stated

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

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

(February 1985)

A. INTRODUCTION

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

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

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

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

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

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

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

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

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

B. BASIC DOSE LIMITS

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

C. GUIDELINES FOR RESIDUAL RADIOACTIVITY

C.1 Residual Radionuclides in Soil Material

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

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

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

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

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

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

C.2 Airborne Radon Decay Products

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

C.3. External Gamma Radiation

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

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

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C.4 Surface Contamination

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

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

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

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

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

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

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

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

D. AUTHORIZED LIMITS FOR RESIDUAL RADIOACTIVITY

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

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

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

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

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

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

c. 5480.2, Hazardous and Radioactive Mixed Waste Management

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

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

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

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

h. 5484.2, Unusual Occurrence Reporting System

i. 5820.2, Radioactive Waste Management

E.1 Interim Storage

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

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

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

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

E.2 Long-Term Management

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

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

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

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

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

F. EXCEPTIONS

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

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

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

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

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

e. Where there is no feasible remedial action.

G. SOURCES

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

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

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

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

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

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APPENDIX C. RADIOLOGICAL SURVEY DATA

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C.1 Indoor Measurements

T009.WS		<----- ALPHA				----- BETA				----->	<----- GAMMA ----->	
ROOM	GRID	DPM/100CM2				DPM/100CM2					uR/h	
NUMBER	NAME	TOTAL	STD DEV	REM	STD DEV	TOTAL	STD DEV	MAX	REM	STD DEV	TOTAL	STD DEV
126	W1,2	-8.8	4.9	N/D	N/D	260.2	103.0	0	N/D	N/D	12.9	0.2
126	W2,5	-11.3	4.5	N/D	N/D	330.8	104.1		N/D	N/D	13.1	0.2
126	W1,8	-11.3	4.5	N/D	N/D	308.4	103.8		N/D	N/D	13.8	0.3
124	N3,1	-1.3	5.8	N/D	N/D	45.0	99.6		N/D	N/D	11.8	0.2
124	N2,4	-2.5	5.6	N/D	N/D	-9.6	98.7		N/D	N/D	11.8	0.2
124	E2,2	-5.0	5.3	N/D	N/D	70.7	100.0		N/D	N/D	12.0	0.2
124	E3,3	7.6	6.7	N/D	N/D	-70.7	97.7		N/D	N/D	10.5	0.2
124	S3,2	0.0	5.9	N/D	N/D	-144.5	96.5		N/D	N/D	10.9	0.2
124	S2,4	25.2	8.2	N/D	N/D	67.5	100.0		N/D	N/D	10.9	0.2
124	W2,1	-6.3	5.2	N/D	N/D	-167.0	96.1		N/D	N/D	11.9	0.2
124	W2,2	-6.3	5.2	N/D	N/D	-224.8	95.2		N/D	N/D	11.1	0.2
124	W3,3	2.5	6.2	N/D	N/D	-167.0	96.1		N/D	N/D	11.7	0.2
126	F2,2	-12.6	4.7	-1.4	1.4	664.9	108.5		0.0	3.3	12.7	0.2
126	F2,5	-11.3	4.9	-2.1	1.2	742.0	109.6		0.8	3.4	12.8	0.2
126	F2,8	-12.6	4.7	-1.4	1.4	745.2	109.7		1.6	3.5	13.8	0.3
126	F5,2	-12.6	4.7	-2.1	1.2	677.7	108.7		6.2	4.0	11.3	0.2
126	F5,5	-3.8	5.8	-2.1	1.2	539.6	106.6		5.5	3.9	12.9	0.2
126	F5,8	-8.8	5.2	-0.7	1.6	642.4	108.2		3.1	3.7	13.7	0.3
126	F2,11	-10.1	5.0	-2.1	1.2	501.1	106.0		1.6	3.5	14.5	0.3
126	F2,14	-11.3	4.9	-2.1	1.2	478.6	105.7		1.6	3.5	13.8	0.3
126	F2,17	-8.8	5.2	-2.1	1.2	424.0	104.9		-1.6	3.1	13.7	0.3
126	F5,11	-6.3	5.5	-2.1	1.2	456.1	105.4		3.1	3.7	14.5	0.3
126	F5,14	-10.1	5.0	-1.4	1.4	459.3	105.4		3.9	3.7	14.1	0.3
126	F5,17	-10.1	5.0	-1.4	1.4	523.6	106.4		0.0	3.3	13.8	0.3
126	F8,11	-11.3	4.9	-0.7	1.6	658.5	108.4		5.5	3.9	13.8	0.3
126	F8,14	-10.1	5.0	-1.4	1.4	-199.1	94.8		2.3	3.6	12.2	0.2
126	F8,17	-11.3	4.9	-2.1	1.2	-234.5	94.2		1.6	3.5	12.9	0.2
126	F11,11	-11.3	4.9	-2.1	1.2	334.0	103.5		-1.6	3.1	12.8	0.2
126	F12,14	-10.1	5.0	-2.1	1.2	388.7	104.3		0.8	3.4	13.4	0.2
126	F11,17	-8.8	5.2	-2.1	1.2	725.9	109.4		1.6	3.5	12.7	0.2
124COR	F2,2	-7.6	5.3	-1.4	1.4	359.7	103.9		-5.5	2.6	10.4	0.2
124COR	F5,2	-11.3	4.9	-1.4	1.4	289.1	102.8		5.5	3.9	13.4	0.2
124COR	F7,2	-8.8	5.2	-2.1	1.2	276.2	102.6		-1.6	3.1	12.3	0.2
124	F1,1	92.0	12.4	11.9	3.4	269.8	102.5		2.3	3.6	11.8	0.2
124	F2,2	25.2	8.4	-0.7	1.6	276.2	102.6		2.3	3.6	12.1	0.2
124	F3,3	81.9	11.9	-0.7	1.6	147.8	100.6		4.7	3.8	12.3	0.2
124	F4,1	13.9	7.5	-2.1	1.2	407.9	104.6		1.6	3.5	11.1	0.2
120	F2,2	-10.1	5.0	-1.4	1.4	725.9	109.4		3.1	3.7	11.2	0.2
120	F2,5	-10.1	5.0	-1.4	1.4	610.3	107.7		4.7	3.8	10.9	0.2
120	F2,8	-10.1	5.0	0.7	1.9	86.7	99.6		3.1	3.7	10.0	0.2
120	F2,11	-12.6	4.7	-2.1	1.2	346.9	103.7		3.9	3.7	9.4	0.2
120	F2,14	-10.1	5.0	-0.7	1.6	247.3	102.1		0.8	3.4	10.0	0.2
120	F2,17	-12.6	4.7	-0.7	1.6	298.7	102.9		3.9	3.7	11.3	0.2
120	F2,19	-7.6	5.3	-2.1	1.2	382.2	104.2		2.3	3.6	10.3	0.2
120	F5,2	-10.1	5.0	-1.4	1.4	510.7	106.2		-4.7	2.7	10.0	0.2
120	F5,5	-11.3	4.9	-1.4	1.4	507.5	106.1		-0.8	3.2	9.1	0.2
120	F5,8	-12.6	4.7	-1.4	1.4	716.3	109.3		-2.3	3.0	9.8	0.2
120	F5,11	-11.3	4.9	-2.1	1.2	530.0	106.5		2.3	3.6	10.1	0.2
120	F5,14	-7.6	5.3	-0.7	1.6	414.3	104.7		0.8	3.4	9.0	0.2
120	F5,17	-11.3	4.9	-1.4	1.4	526.8	106.4		4.7	3.8	9.1	0.2
120	F5,19	-10.1	5.0	-2.1	1.2	436.8	105.1		3.9	3.7	9.2	0.2
118	F2,2	-3.8	5.8	-2.1	1.2	0.0	98.2		3.1	3.5	8.2	0.2
118	F2,5	-11.3	4.9	-2.1	1.2	-19.3	97.8		0.0	3.1	8.6	0.2
118	F5,2	-7.6	5.3	-2.1	1.2	-125.3	96.1		0.8	3.2	8.3	0.2

T009.WS		<---- ALPHA				---- BETA				---->	<----- GAMMA	---->
ROOM	GRID	DPM/100CM2				DPM/100CM2					uR/h	
NUMBER	NAME	TOTAL	STD DEV	REM	STD DEV	TOTAL	STD DEV	MAX	REM	STD DEV	TOTAL	STD DEV
118	F5,5	-7.6	5.3	-1.4	1.4	83.5	99.5		5.5	3.7	7.9	0.2
118	F7,2	-11.3	4.9	-2.1	1.2	99.6	99.8		0.8	3.2	7.3	0.2
118	F7,5	-7.6	5.3	-1.4	1.4	-96.4	96.6		3.1	3.5	7.5	0.2
114	CAB1			11.2	3.3				10.9	4.4		
114	CAB2			5.6	2.6				7.8	4.1		
114	CAB3			7.0	2.8				10.9	4.4		
114	CAB4			15.4	3.7				11.7	4.5		
114	CAB5			7.0	2.8				10.1	4.3		
114	FW1			0.7	1.9				3.9	3.7		
114	FW2			-2.1	1.2				7.0	4.1		
114	FW3			-1.4	1.4				2.3	3.6		
114	FW4			-1.4	1.4				2.3	3.6		
114	FW5			-1.4	1.4				2.3	3.6		
114	FW6			-2.1	1.2				-0.8	3.2		
114	FW7			-1.4	1.4				0.0	3.3		
116	E1			-1.4	1.4				5.5	3.7		
116	E2			-1.4	1.4				2.3	3.4		
116	S1			-1.4	1.4				4.7	3.7		
116	S2			-1.4	1.4				0.0	3.1		
116	S3			-1.4	1.4				1.6	3.3		
116	S4			0.0	1.7				3.1	3.5		
116	S5			-1.4	1.4				2.3	3.4		
116	W1			-1.4	1.4				2.3	3.4		
116	W2			0.0	1.7				5.5	3.7		
116	W3			-1.4	1.4				11.7	4.3		
116	W4			-1.4	1.4				6.2	3.8		
116	CAB1			11.2	3.3				11.7	4.3		
116	CAB2			5.6	2.6				8.6	4.1		
116	CAB3			7.0	2.8				11.7	4.3		
116	CAB4			15.4	3.7				12.5	4.4		
116	CAB5			7.0	2.8				10.9	4.3		
NUMBER OF MEAS.:		57		73		57			73		57	
AVERAGE:		-3.69		0.18		300.63			3.47		11.45	
STD. DEV.:		19.17		4.22		256.33			3.91		1.92	
MAXIMUM:		91.98		15.40		745.18			12.48		14.49	
MINIMUM:		-12.60		-2.10		-234.48			-5.46		7.33	
RANGE:		104.58		17.50		979.66			17.94		7.16	

C.2 Outdoor Exposure Rates (Stored by Location)

T009OUT.WS SORTED BY LOCATION

ROOM NUMBER	GRID NAME	GAMMA TOTAL	uR/h TOTAL	STD DEV
OUTSIDE	1-9	3014	13.96	0.25
OUTSIDE	1-10	3002	13.91	0.25
OUTSIDE	1-11	2968	13.75	0.25
OUTSIDE	1-12	2986	13.83	0.25
OUTSIDE	1-13	2847	13.19	0.25
OUTSIDE	2-7	2887	13.37	0.25
OUTSIDE	2-8	2904	13.45	0.25
OUTSIDE	2-9	2878	13.33	0.25
OUTSIDE	2-10	2807	13.00	0.25
OUTSIDE	2-11	2728	12.64	0.24
OUTSIDE	2-12	2804	12.99	0.25
OUTSIDE	2-13	2772	12.84	0.24
OUTSIDE	3-7	2966	13.74	0.25
OUTSIDE	3-8	2765	12.81	0.24
OUTSIDE	3-9	2840	13.15	0.25
OUTSIDE	3-10	2603	12.06	0.24
OUTSIDE	3-11	2794	12.94	0.24
OUTSIDE	3-12	2928	13.56	0.25
OUTSIDE	3-13	2894	13.41	0.25
OUTSIDE	4-5	2960	13.71	0.25
OUTSIDE	4-6	3034	14.05	0.26
OUTSIDE	4-7	2808	13.01	0.25
OUTSIDE	4-8	2640	12.23	0.24
OUTSIDE	4-9	2714	12.57	0.24
OUTSIDE	4-10	2782	12.89	0.24
OUTSIDE	4-11	2780	12.88	0.24
OUTSIDE	4-12	2968	13.75	0.25
OUTSIDE	4-13	3024	14.01	0.25
OUTSIDE	5-5	2987	13.84	0.25
OUTSIDE	5-6	2824	13.08	0.25
OUTSIDE	5-7	2692	12.47	0.24
OUTSIDE	5-8	2716	12.58	0.24
OUTSIDE	5-9	2797	12.96	0.24
OUTSIDE	5-10	2639	12.22	0.24
OUTSIDE	5-11	2637	12.21	0.24
OUTSIDE	5-12	2930	13.57	0.25
OUTSIDE	5-13	3143	14.56	0.26
OUTSIDE	6-4	2999	13.89	0.25
OUTSIDE	6-5	2972	13.77	0.25
OUTSIDE	6-6	2704	12.52	0.24
OUTSIDE	6-7	2628	12.17	0.24
OUTSIDE	6-8	2917	13.51	0.25
OUTSIDE	6-9	2909	13.47	0.25
OUTSIDE	6-10	2862	13.26	0.25
OUTSIDE	6-11	2405	11.14	0.23
OUTSIDE	6-12	3044	14.10	0.26
OUTSIDE	6-13	3037	14.07	0.26
OUTSIDE	7-3	3114	14.42	0.26
OUTSIDE	7-4	2852	13.21	0.25
OUTSIDE	7-5	2943	13.63	0.25

T009OUT.WS SORTED BY LOCATION

ROOM NUMBER	GRID NAME	GAMMA TOTAL	uR/h TOTAL	STD DEV
OUTSIDE	7-6	2747	12.72	0.24
OUTSIDE	7-7	2781	12.88	0.24
OUTSIDE	7-8	2896	13.41	0.25
OUTSIDE	7-9	2787	12.91	0.24
OUTSIDE	7-10	3075	14.24	0.26
OUTSIDE	7-11	2615	12.11	0.24
OUTSIDE	8-2	3013	13.96	0.25
OUTSIDE	8-3	2803	12.98	0.25
OUTSIDE	8-4	2638	12.22	0.24
OUTSIDE	8-5	2704	12.52	0.24
OUTSIDE	8-6	2684	12.43	0.24
OUTSIDE	8-7	2652	12.28	0.24
OUTSIDE	8-8	2496	11.56	0.23
OUTSIDE	8-9	2891	13.39	0.25
OUTSIDE	8-10	2632	12.19	0.24
OUTSIDE	8-11	3069	14.22	0.26
OUTSIDE	9-1	2965	13.73	0.25
OUTSIDE	9-2	2930	13.57	0.25
OUTSIDE	9-3	2766	12.81	0.24
OUTSIDE	9-4	2812	13.03	0.25
OUTSIDE	9-5	3024	14.01	0.25
OUTSIDE	9-6	2765	12.81	0.24
OUTSIDE	9-7	2535	11.74	0.23
OUTSIDE	9-8	2443	11.32	0.23
OUTSIDE	9-9	2389	11.07	0.23
OUTSIDE	9-10	2909	13.47	0.25
OUTSIDE	9-11	2857	13.23	0.25
OUTSIDE	10-1	2950	13.66	0.25
OUTSIDE	10-2	2848	13.19	0.25
OUTSIDE	10-3	2848	13.19	0.25
OUTSIDE	10-4	2673	12.38	0.24
OUTSIDE	10-5	2609	12.08	0.24
OUTSIDE	10-6	2959	13.71	0.25
OUTSIDE	10-7	2820	13.06	0.25
OUTSIDE	11-1	2916	13.51	0.25
OUTSIDE	11-2	2816	13.04	0.25
OUTSIDE	11-3	2864	13.27	0.25
OUTSIDE	11-4	2839	13.15	0.25
OUTSIDE	11-5	2893	13.40	0.25
OUTSIDE	11-6	3251	15.06	0.26
OUTSIDE	11-7	3175	14.71	0.26
OUTSIDE	12-1	2874	13.31	0.25
OUTSIDE	12-2	2690	12.46	0.24
OUTSIDE	12-3	2644	12.25	0.24
OUTSIDE	12-4	2743	12.71	0.24
OUTSIDE	12-5	2917	13.51	0.25
OUTSIDE	12-6	3319	15.37	0.27
OUTSIDE	12-7	3419	15.84	0.27
OUTSIDE	13-1	2782	12.89	0.24
OUTSIDE	13-2	2652	12.28	0.24

T009OUT.WS SORTED BY LOCATION

ROOM NUMBER	GRID NAME	GAMMA TOTAL	uR/h TOTAL	STD DEV
OUTSIDE	13-3	2172	10.06	0.22
OUTSIDE	13-4	2610	12.09	0.24
OUTSIDE	13-5	2804	12.99	0.25
OUTSIDE	13-6	2767	12.82	0.24
OUTSIDE	13-7	2528	11.71	0.23
OUTSIDE	14-1	2796	12.95	0.24
OUTSIDE	14-2	2536	11.75	0.23
OUTSIDE	14-3	2269	10.51	0.22
OUTSIDE	14-4	2582	11.96	0.24
OUTSIDE	14-5	2656	12.30	0.24
OUTSIDE	14-6	2648	12.27	0.24
OUTSIDE	14-7	2431	11.26	0.23
OUTSIDE	15-1	2840	13.15	0.25
OUTSIDE	15-2	2731	12.65	0.24
OUTSIDE	15-3	2591	12.00	0.24
OUTSIDE	15-4	2643	12.24	0.24
OUTSIDE	15-5	2654	12.29	0.24
OUTSIDE	15-6	2601	12.05	0.24
OUTSIDE	15-7	2563	11.87	0.23
OUTSIDE	16-1	2940	13.62	0.25
OUTSIDE	16-2	2586	11.98	0.24
OUTSIDE	16-3	2415	11.19	0.23
OUTSIDE	16-4	2792	12.93	0.24
OUTSIDE	16-5	2767	12.82	0.24
OUTSIDE	16-6	2671	12.37	0.24
OUTSIDE	16-7	2624	12.15	0.24

NUMBER OF MEAS.	126	
AVERAGE/SQRT(SUMSQ)	12.96	2.75
STD. DEV.	0.92	
MAXIMUM	15.84	
MIMIMUM	10.06	
RANGE	5.78	

Outdoor Exposure Rates (Stored by Exposure Rate)

T009OUT.WS SORTED BY EXPOSURE RATE

ROOM NUMBER	GRID NAME	GAMMA TOTAL	uR/h TOTAL	uR/h STD DEV
OUTSIDE	12-7	3419	15.84	0.27
OUTSIDE	12-6	3319	15.37	0.27
OUTSIDE	11-6	3251	15.06	0.26
OUTSIDE	11-7	3175	14.71	0.26
OUTSIDE	5-13	3143	14.56	0.26
OUTSIDE	7-3	3114	14.42	0.26
OUTSIDE	7-10	3075	14.24	0.26
OUTSIDE	8-11	3069	14.22	0.26
OUTSIDE	6-12	3044	14.10	0.26
OUTSIDE	6-13	3037	14.07	0.26
OUTSIDE	4-6	3034	14.05	0.26
OUTSIDE	4-13	3024	14.01	0.25
OUTSIDE	9-5	3024	14.01	0.25
OUTSIDE	1-9	3014	13.96	0.25
OUTSIDE	8-2	3013	13.96	0.25
OUTSIDE	1-10	3002	13.91	0.25
OUTSIDE	6-4	2999	13.89	0.25
OUTSIDE	5-5	2987	13.84	0.25
OUTSIDE	1-12	2986	13.83	0.25
OUTSIDE	6-5	2972	13.77	0.25
OUTSIDE	4-12	2968	13.75	0.25
OUTSIDE	1-11	2968	13.75	0.25
OUTSIDE	3-7	2966	13.74	0.25
OUTSIDE	9-1	2965	13.73	0.25
OUTSIDE	4-5	2960	13.71	0.25
OUTSIDE	10-6	2959	13.71	0.25
OUTSIDE	10-1	2950	13.66	0.25
OUTSIDE	7-5	2943	13.63	0.25
OUTSIDE	16-1	2940	13.62	0.25
OUTSIDE	5-12	2930	13.57	0.25
OUTSIDE	9-2	2930	13.57	0.25
OUTSIDE	3-12	2928	13.56	0.25
OUTSIDE	6-8	2917	13.51	0.25
OUTSIDE	12-5	2917	13.51	0.25
OUTSIDE	11-1	2916	13.51	0.25
OUTSIDE	6-9	2909	13.47	0.25
OUTSIDE	9-10	2909	13.47	0.25
OUTSIDE	2-8	2904	13.45	0.25
OUTSIDE	7-8	2896	13.41	0.25
OUTSIDE	3-13	2894	13.41	0.25
OUTSIDE	11-5	2893	13.40	0.25
OUTSIDE	8-9	2891	13.39	0.25
OUTSIDE	2-7	2887	13.37	0.25
OUTSIDE	2-9	2878	13.33	0.25
OUTSIDE	12-1	2874	13.31	0.25
OUTSIDE	11-3	2864	13.27	0.25
OUTSIDE	6-10	2862	13.26	0.25
OUTSIDE	9-11	2857	13.23	0.25
OUTSIDE	7-4	2852	13.21	0.25
OUTSIDE	10-2	2848	13.19	0.25

T009OUT.WS SORTED BY EXPOSURE RATE

ROOM NUMBER	GRID NAME	GAMMA TOTAL	uR/h TOTAL	uR/h STD DEV
OUTSIDE	10-3	2848	13.19	0.25
OUTSIDE	1-13	2847	13.19	0.25
OUTSIDE	3-9	2840	13.15	0.25
OUTSIDE	15-1	2840	13.15	0.25
OUTSIDE	11-4	2839	13.15	0.25
OUTSIDE	5-6	2824	13.08	0.25
OUTSIDE	10-7	2820	13.06	0.25
OUTSIDE	11-2	2816	13.04	0.25
OUTSIDE	9-4	2812	13.03	0.25
OUTSIDE	4-7	2808	13.01	0.25
OUTSIDE	2-10	2807	13.00	0.25
OUTSIDE	2-12	2804	12.99	0.25
OUTSIDE	13-5	2804	12.99	0.25
OUTSIDE	8-3	2803	12.98	0.25
OUTSIDE	5-9	2797	12.96	0.24
OUTSIDE	14-1	2796	12.95	0.24
OUTSIDE	3-11	2794	12.94	0.24
OUTSIDE	16-4	2792	12.93	0.24
OUTSIDE	7-9	2787	12.91	0.24
OUTSIDE	4-10	2782	12.89	0.24
OUTSIDE	13-1	2782	12.89	0.24
OUTSIDE	7-7	2781	12.88	0.24
OUTSIDE	4-11	2780	12.88	0.24
OUTSIDE	2-13	2772	12.84	0.24
OUTSIDE	13-6	2767	12.82	0.24
OUTSIDE	16-5	2767	12.82	0.24
OUTSIDE	9-3	2766	12.81	0.24
OUTSIDE	3-8	2765	12.81	0.24
OUTSIDE	9-6	2765	12.81	0.24
OUTSIDE	7-6	2747	12.72	0.24
OUTSIDE	12-4	2743	12.71	0.24
OUTSIDE	15-2	2731	12.65	0.24
OUTSIDE	2-11	2728	12.64	0.24
OUTSIDE	5-8	2716	12.58	0.24
OUTSIDE	4-9	2714	12.57	0.24
OUTSIDE	8-5	2704	12.52	0.24
OUTSIDE	6-6	2704	12.52	0.24
OUTSIDE	5-7	2692	12.47	0.24
OUTSIDE	12-2	2690	12.46	0.24
OUTSIDE	8-6	2684	12.43	0.24
OUTSIDE	10-4	2673	12.38	0.24
OUTSIDE	16-6	2671	12.37	0.24
OUTSIDE	14-5	2656	12.30	0.24
OUTSIDE	15-5	2654	12.29	0.24
OUTSIDE	8-7	2652	12.28	0.24
OUTSIDE	13-2	2652	12.28	0.24
OUTSIDE	14-6	2648	12.27	0.24
OUTSIDE	12-3	2644	12.25	0.24
OUTSIDE	15-4	2643	12.24	0.24
OUTSIDE	4-8	2640	12.23	0.24

T009OUT.WS SORTED BY EXPOSURE RATE

ROOM NUMBER	GRID NAME	GAMMA TOTAL	uR/h TOTAL	STD DEV
OUTSIDE	5-10	2639	12.22	0.24
OUTSIDE	8-4	2638	12.22	0.24
OUTSIDE	5-11	2637	12.21	0.24
OUTSIDE	8-10	2632	12.19	0.24
OUTSIDE	6-7	2628	12.17	0.24
OUTSIDE	16-7	2624	12.15	0.24
OUTSIDE	7-11	2615	12.11	0.24
OUTSIDE	13-4	2610	12.09	0.24
OUTSIDE	10-5	2609	12.08	0.24
OUTSIDE	3-10	2603	12.06	0.24
OUTSIDE	15-6	2601	12.05	0.24
OUTSIDE	15-3	2591	12.00	0.24
OUTSIDE	16-2	2586	11.98	0.24
OUTSIDE	14-4	2582	11.96	0.24
OUTSIDE	15-7	2563	11.87	0.23
OUTSIDE	14-2	2536	11.75	0.23
OUTSIDE	9-7	2535	11.74	0.23
OUTSIDE	13-7	2528	11.71	0.23
OUTSIDE	8-8	2496	11.56	0.23
OUTSIDE	9-8	2443	11.32	0.23
OUTSIDE	14-7	2431	11.26	0.23
OUTSIDE	16-3	2415	11.19	0.23
OUTSIDE	6-11	2405	11.14	0.23
OUTSIDE	9-9	2389	11.07	0.23
OUTSIDE	14-3	2269	10.51	0.22
OUTSIDE	13-3	2172	10.06	0.22

NUMBER OF MEAS.	126	
AVERAGE/SQRT(SUMSQ)	12.96	2.75
STD. DEV.	0.92	
MAXIMUM	15.84	
MIMIMUM	10.06	
RANGE	5.78	

APPENDIX D. SURVEYOR MAPS USED DURING RADIOLOGICAL SURVEY

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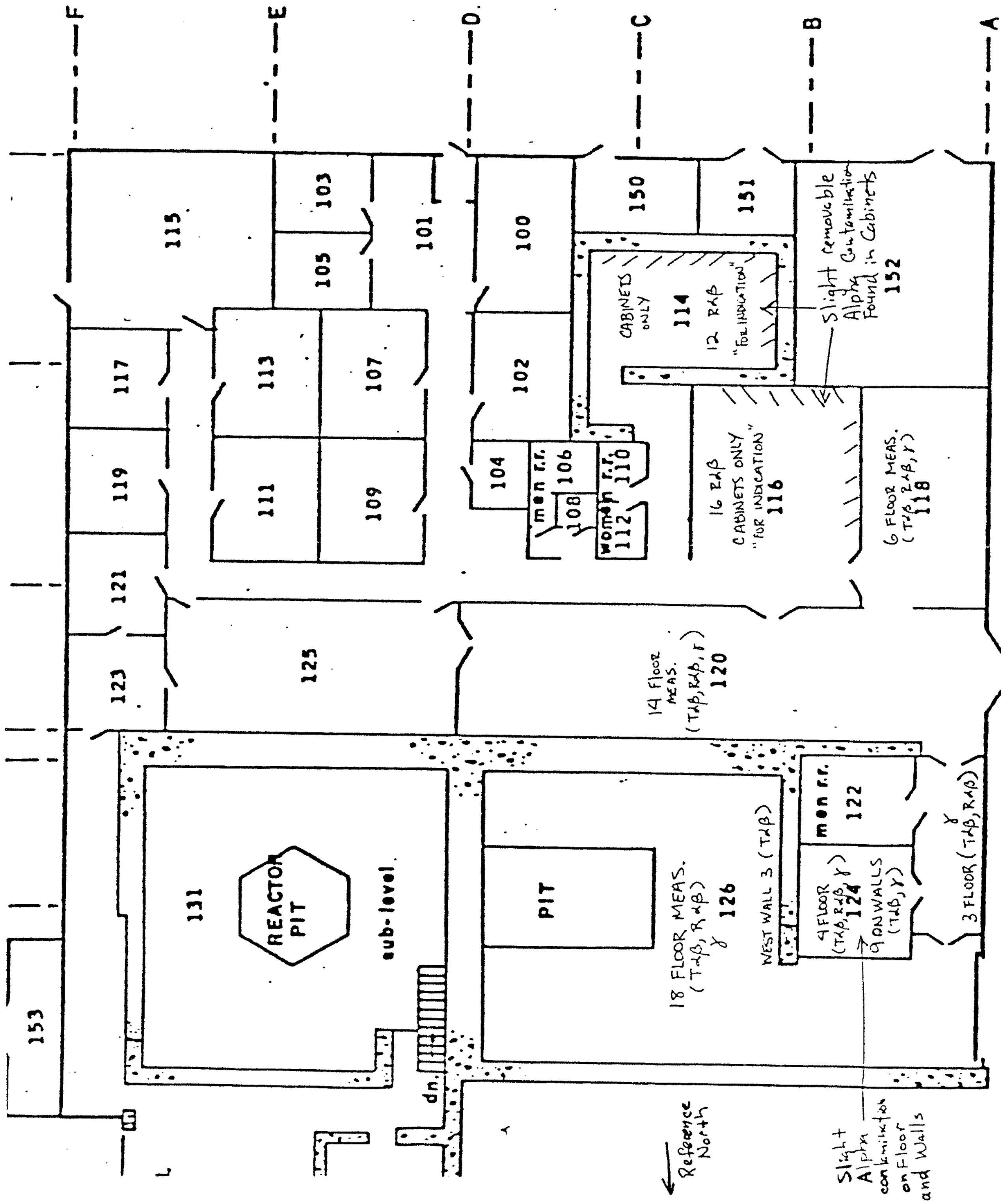

D.1 Indoor Survey Locations

D.2 Outdoor Survey Locations

NORTH & WEST SIDES

Reference North

	1	2	3	4	5	6	7	8	9	10	11	12	13
1									3014	3002	2968	2986	284
2							2887	2909	2878	2807	2728	2804	277
3							2966	2765	2840	2603	2794	2928	2894
4					2960	3034	2808	2640	2714	2782	2780	2968	3029
5					2987	2824	2692	2716	2797	2639	2637	2930	3143
6				2999	2972	2704	2628	2917	2909	2862	2405	3094	3037
7			3114	2852	2943	2797 (C)	2781	2896	2787	3075	2615		
8		3013	2803	2638	2704	2684	2652	2496	2891	2632	3069		
9	2965	2930	2766	2812	3024	2765 (C)	2535 (C)	2443 (T)	2389 (T)	2909	2857		
10	2950	2848	2848	2673	2609	2959 3051	2820						
11	2916	2816	2864	2839	2893	3251	3175 3008						
12	2874	2690	2644	2743	2917	3319	3419						
13	2782	2652	2172	2610	2804	2767	2528						
14	2796	2596	2269	2582	2656	2648	2431						
15	2840	2731	2591	2643	2654	2601	2563						
16	2940	2586	2415	2792	2767	2671	2629						

DRAIN

BLDG. 009

APPENDIX E. GAMMA SPECTROMETRY RADIONUCLIDE GAMMA-RAY ENERGY AND YIELD LIBRARY

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	Isotope Energy (keV)	Half-Life % Yield								
1.	Zr-95	64.40 D								
	724.0	44%	756.6	55%						
2.	Nb-95	35.15 D								
	765.7	99%								
3.	Ru-103	39.35 D								
	497.0	86%	610.0	5%						
4.	Sb-125	0.1011E04 D								
	176.2	6%	428.0	29%	463.5	10%	606.7	5%	636.1	11%
5.	I-131	8.04 D								
	284.2	6%	364.5	81%	636.9	7%				
6.	Cs-134	752.63 D								
	563.2	8%	569.2	15%	604.6	98%	795.7	85%	801.7	9%
7.	Cs-136	12.98 D								
	66.8	12%	86.2	6%	153.1	7%	176.5	14%	273.5	13%
	340.5	47%	818.5	100%	1048.0	80%	1235.2	20%		
8.	Cs-137	0.1095E05 D								
	661.6	85%								
9.	Ba-140	12.80 D								
	162.5	5%	537.3	20%						
10.	La-140	1.68 D								
	328.7	18%	487.0	43%	815.7	22%	867.8	5%	925.0	6%
	1596.0	95%								
11.	Ce-141	32.50 D								
	36.0	8%	145.1	48%						
12.	Ce-144	284.19 D								
	133.5	11%								
13.	Cr-51	27.70 D								
	320.0	9%								
14.	Mn-54	312.19 D								
	834.7	100%								
15.	Fe-59	45.10 D								
	1099.1	56%	1291.5	43%						

	Isotope / Energy (keV)	Half-Life / % Yield								
16.	Co-58	70.78 D								
	511.0	30%	810.7	99%						
17.	Co-60	0.1924E04 D								
	1173.1	100%	1332.5	100%						
18.	Zn-65	243.80 D								
	511.0	3%	1115.5	51%						
19.	Rh-102	0.1054E04 D								
	418.2	10%	475.0	93%	628.0	6%	631.0	56%	697.0	45%
	766.7	33%	1046.5	33%	1112.6	17%				
20.	Rh-102M	206.00D								
	475.0	44%	511.0	23%						
21.	Sb-124	60.20 D								
	602.6	98%	645.7	7%	722.7	12%	1691.0	50%	2091.1	6%
22.	Be-07	53.40 D								
	477.5	10%								
23.	Na-22	949.00 D								
	511.0	180%	1274.5	100%						
24.	K-040	0.46E12 D								
	1460.7	11%								
25.	Ra-226	0.584E06 D								
	186.0	3%								
26.	Pb-214	0.02 D								
	74.7 6%	77.0 11%	241.8	7%	295.1	19%	352.0	37%		
27.	Bi-214	0.01 D								
	609.2	46%	1120.2	15%	1238.0	6%	1764.5	15%		
28.	Ra-224	3.66 D								
	241.0	4%								
29.	Pb-212	0.44 D								
	74.7	9%	77.0	18%	87.1	6%	238.5	43%		
30.	Bi-212	0.04 D								
	727.1	12%	1620.5	3%						

	Isotope Energy (keV)	Half-Life % Yield								
31.	Tl-208	0.00 D								
	277.3	6%	510.6	22%	583.0	86%	860.5	12%		
32.	Ac-228	0.25 D								
	338.3	12%	911.0	29%	964.5	5%	968.8	17%		
33.	Th-234	24.10 D								
	63.2	4%	92.3	2%	92.7	3%				
34.	U-232	0.263E05 D								
	269.0	4%								
35.	U-235	0.26E12 D								
	93.3	2%	143.7	11%	163.3	5%	185.6	54%	205.2	5%
36.	Am-241	0.158E06 D								
	59.5	36%								
37.	Np-237	0.7817E09 D								
	29.0	9%	86.1	13%						
38.	Pu-242	0.1409E09 D								
	44.5	3%								
39.	Am-243	0.2699E07 D								
	74.6	66%								
40.	Np-239	2.35 D								
	99.5	15%	103.6	24%	106.0	23%	117.6	8%	228.1	11%
	277.5	14%								
41.	Al-26	0.2612E10 D								
	511.0	164%	1808.6	100%						
42.	Nb-94	0.7409E07 D								
	702.5	100%	871.0	100%						
43.	Ag-108M	0.4635E05 D								
	79.5	7%	433.6	90%	614.3	90%	722.9	90%		
44.	Cd-109	453.00 D								
	88.0	3%								
45.	Ba-133	0.3906E04 D								
	81.0	33%	276.2	7%	302.6	19%	355.8	62%	383.6	9%

	Isotope Energy (keV)	Half-Life % Yield								
46.	Eu-148	54.00 D								
	413.8	11%	414.0	7%	550.1	99%	553.1	17%	571.8	9%
	611.2	19%	629.8	71%	725.6	12%	1034.0	8%		
47.	Eu-152	0.4636E04 D								
	121.7	29%	244.6	8%	344.2	27%	778.8	13%	964.0	14%
	1085.7	10%	1112.0	13%	1408.0	21%				
48.	Eu-154	0.3102E04 D								
	123.0	40%	248.0	7%	723.2	20%	873.1	11%	996.2	11%
	1004.7	18%	1274.7	35%						
49.	Eu-155	0.181E04 D								
	86.3	33%	105.2	22%						
50.	Tb-158	0.5475E05 D								
	79.5	11%	181.8	9%	780.1	9%	944.1	43%	962.1	20%
51.	Pt-193	0.1825E05 D								
	63.2	24%	64.8	44%	73.5	15%				
52.	Co-57	270.00 D								
	122.0	86%	136.3	11%						
53.	Sr-85	64.73 D								
	513.9	99%								
54.	Y-88	106.60 D								
	898.0	94%	1836.0	99%						
55.	Sn-113	115.10 D								
	391.6	64%								
56.	Ce-139	137.50 D								
	165.7	80%								
57.	Hg-203	46.59 D								
	72.8	6%	279.1	81%						
58.	Ta-182	115.00 D								
	67.7	41%	100.1	14%	152.4	7%	222.0	7%	1121.2	35%
	1189.0	16%	1221.4	27%	1230.9	11%				

APPENDIX F. COPY OF INTERNAL LETTER, "RADIOLOGICAL SURVEY PLAN FOR BUILDING T009," MARCH 30, 1988

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3/30/88

rdsrvy17.ws

Those Listed

J. A. Chapman
641, 055-T100

5766

Radiological Survey Plan for Building T009

Building T009 was a test facility for the OMR and SGR. Low enriched uranium, thorium, activation and mixed fission products were handled or produced there as a result of this activity. The following survey plan specifies the radiological characterization required at T009 as part of the DOE SSFL survey.

Survey Task Nomenclature

Direct Measurements: a=alpha, b=beta, g=gamma exposure rate (Made in one square meter, on a 3mx3m grid- inside only)
Removable Measurements: ra=alpha, rb=beta
Sampling Analysis: gs=gamma spectrometry, abp=alpha/beta proportional

Survey Plan

Room No.	Survey Task
R/A Holdup Tank (East side)	Grab sample from tank (gs)
Northwest perimeter (see map)	(g, on 6mx6m grid)
121	sink drain trap sample (gs)
108-Janitor's closet	sink drain trap sample (gs)
114-counting room	bottom shelf of cabinets (a/b/ra/rb) floor to wall joint (a/b/ra/rb) do not survey floor
116-tool room	20% of drawers (a/b/ra/rb) cabinet to floor joint (a/b/ra/rb)
118-machine shop	floor survey (a/b/g/ra/rb) lathe, drill press, mill (a/b) grease, crud in catch pans (gs)
120-control room (LMR, SGR)	floor (a/b/ra/rb/g)
126-west vault and corridor	floor (a/b/ra/rb/g) southwest vault wall up to 2 meters (a/b/g)

page 2

Room No.	Survey Task
124-fuel vault	floor (a/b/ra/rb/g) walls up to 2m, and doors (a/b/g)
122-change room	shower drain sediment (gs)
Ventilation System	TBD-survey of both filter plenums (a/b/g/ra/rb)
Drain System	See rooms 121, 122, 108, and holdup tank

Floor tile removal is not necessary. This survey should take about 5 man days.

J. A. Chapman

J. A. Chapman
Radiation & Nuclear Safety

attachments: 1) T009 Plot Plan
2) SSFl Radiological Survey Plan-T009

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

ATTACHMENT 1 : T009 PLOT-SURVEY PLAN
3/30/88

GEN-ZR-0014
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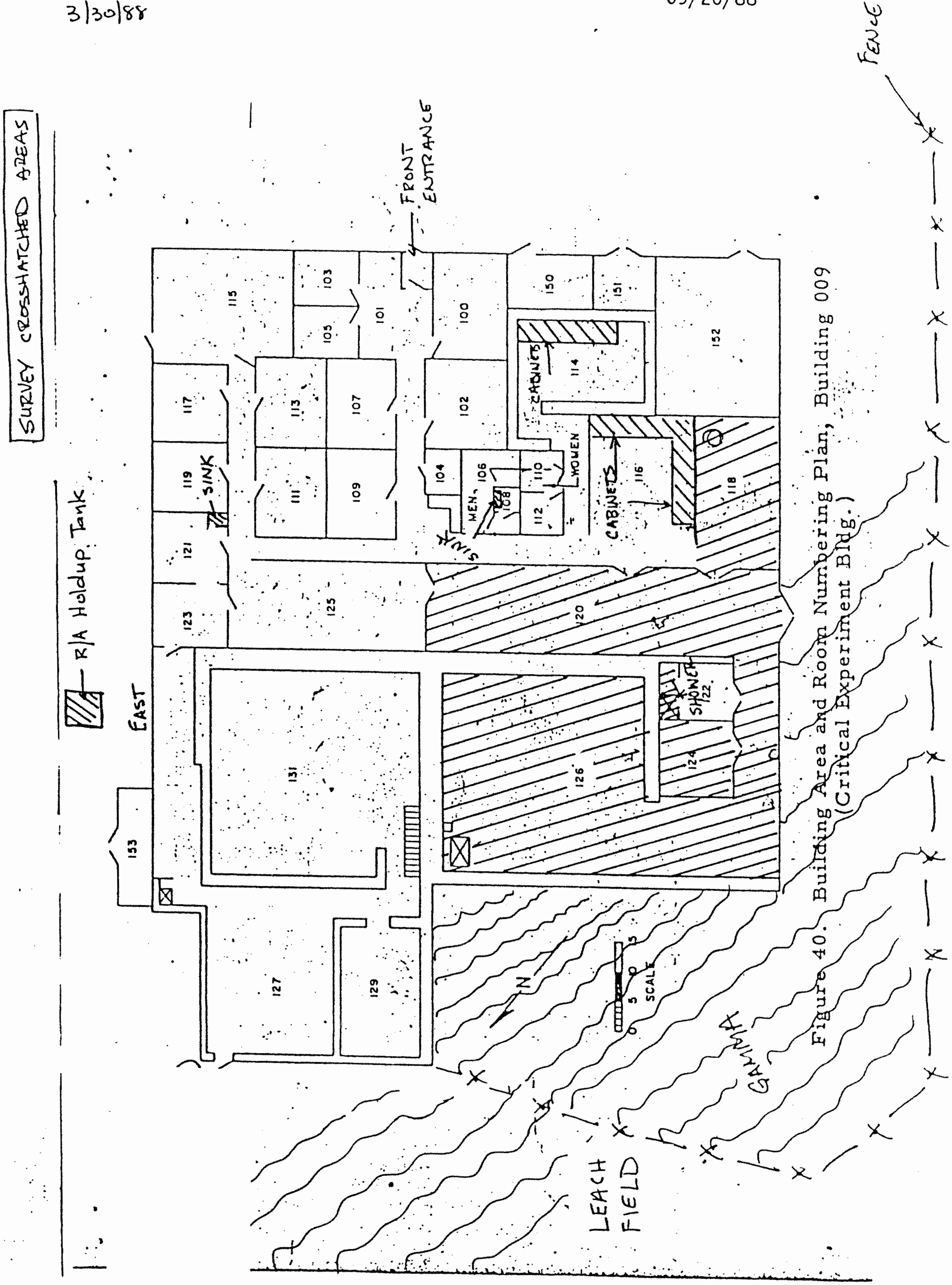

Figure 40. Building Area and Room Numbering Plan, Building 009 (Critical Experiment Bldg.)

ATTACHMENT 2:

5.4.20 T-009 - Building

Project Name: OMR and SGR Test Facility

Isotopes: Enriched uranium, depleted uranium, thorium, mixed fission and activation products

Physical and Chemical Form of Radioisotopes: Various

Process Conducted: Critical experiments, some chemistry

Known Problems: Depleted uranium from T-028 in janitor's closet. Some found in counting room. Thorium in holdup tank, minor amounts. ISI work ongoing in NE high bay with fresh fission products. Contamination in machine shop.

1. Perimeter pads and outside perimeter fence (leach field?)

2. Surface survey - high bay - west; vault - counting room hall closet, tool room, shop, and control room

3. Survey ventilation system

4. Drain system.

5. Wall of west valut for neutron-activated concrete.