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ABSTRACT

Following cleanup of any previously detected radioactivity exceeding specified limits, a radiation survey was performed throughout the L-85 reactor building (T093) and associated buildings (T083, T074, and T453). The results of this survey show that this facility meets the criteria established by the U.S. Nuclear Regulatory Commission's Regulatory Guide 1.86 and NRC Dismantling Order, Docket No. 50-375, dated February 22, 1983, for release of facilities for unrestricted use.

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REV	SUMMARY OF CHANGE	APPROVALS AND DATE
A	Added data on complete resurvey of reactor room for ambient radiation and interpretation of dismantling order release criterion. (Page 33.1 through 33.8) Page 3 - added Figures: 16. Results of Resurvey for Ambient Exposure Rate in Reactor Room (Instrument 596003) 17. Results of Resurvey for Ambient Exposure Rate in Reactor Room (Instrument 596007) 18. Ambient Exposure Rate Outside Reactor Room (Instrument 596003) 19. Ambient Exposure Rate Outside Reactor Room (Instrument 596007) 20. Net exposure rate (μR/h) at Locations Exceeding Local Background in Reactor Room Release	F. E. Begley 3/2/86 F. E. Begley M E Remley 4 Mar 86 M. E. Remley RJ Tuttle 2/21/86 R. J. Tuttle Sw Date 3-6-86

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I. INTRODUCTION

The L-85 reactor in Building T093, at the Rockwell International Santa Susana Field Laboratories, was an NRC-licensed (R-118, Docket No. 50-375) operating facility since January 5, 1972. From 1952 until 1972, it was an AEC-owned facility. The L-85 reactor was initially located at Downey, California, under the designation WBNS (Water Boiler Neutron Source) from 1952 until 1956, where it was operated at a maximum power level of 0.5 W. It was moved to the Santa Susana Field Laboratories in the latter part of 1956, modified to increase the power level to 3 kWt, and redesignated as the AE-6 Reactor. After transfer of ownership from the U.S. Government to Rockwell International and licensing by the Nuclear Regulatory Commission, it was operated in support of commercial programs until February 29, 1980. An application for a dismantling order was made to the Nuclear Regulatory Commission on March 10, 1980.

The L-85 was a homogeneous aqueous solution research reactor. The fuel solution was highly enriched uranyl sulfate dissolved in water, and contained in a spherical graphite-reflected stainless steel core. Possession of the radioactive material produced by irradiation in the reactor was authorized[1] under the California Radioactive Material License No. 0015-70. The reactor was operated to provide a neutron source for subcritical experiments, neutron radiography, and training functions.

A complete description of the facility and reactor is presented in "Safety Analysis Report for the L-85 Nuclear Examination Reactor," AI-70-73, September 24, 1971, V. A. Swanson.

On July 29, 1982, the uranyl sulfate solution was removed from the reactor core, and on September 28, 1982, it was shipped to the Idaho National Engineering Laboratory for processing. The fuel draining operation was performed in accordance with the requirements of "Nuclear Safety Analysis and Procedure for Draining the L-85 Fuel Solution," N001NSA000001, V. A. Swanson, August 2, 1982.

The application for the dismantling order was amended on December 14, 1982, to include the changes in the facility and the impact on the detailed procedures required for implementation of the dismantling plan. The dismantling order was then issued on February 22, 1983, and is included here as Appendix C.

During the fuel draining operation, approximately 5 milliliters of U-235 contaminated rinse water spilled onto the floor. The area was decontaminated, but not completely at that time due to relatively high ambient radiation levels from equipment associated with the reactor. Further decontamination took place during decommissioning of the facility.

In summary, all detectable radioactive material was removed, with residual contamination well below the applicable limits specified in the Dismantling Order.

II. IDENTIFICATION OF PREMISES

The premises to be released consist of Buildings T093, T083, T074, and T453. The site is located at the Santa Susana Field Laboratories as shown in Figure 1.

Figure 2 shows the reactor building as it appeared before decommissioning; Figure 3 shows the building as it appears since decommissioning; Figure 4 shows Buildings T074 and T083; Figure 5 shows Building T453.

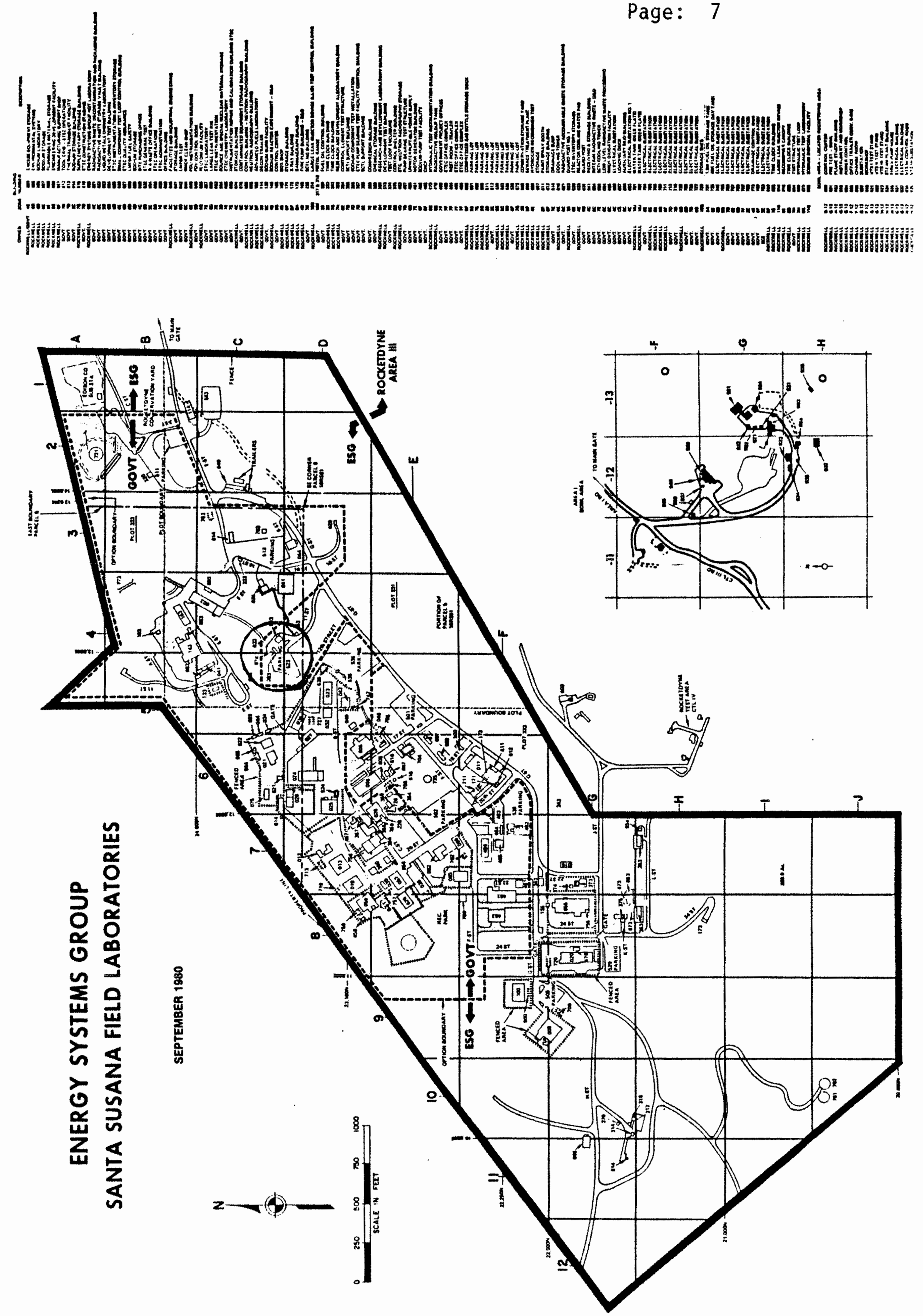


Figure 1. Santa Susana Site Map

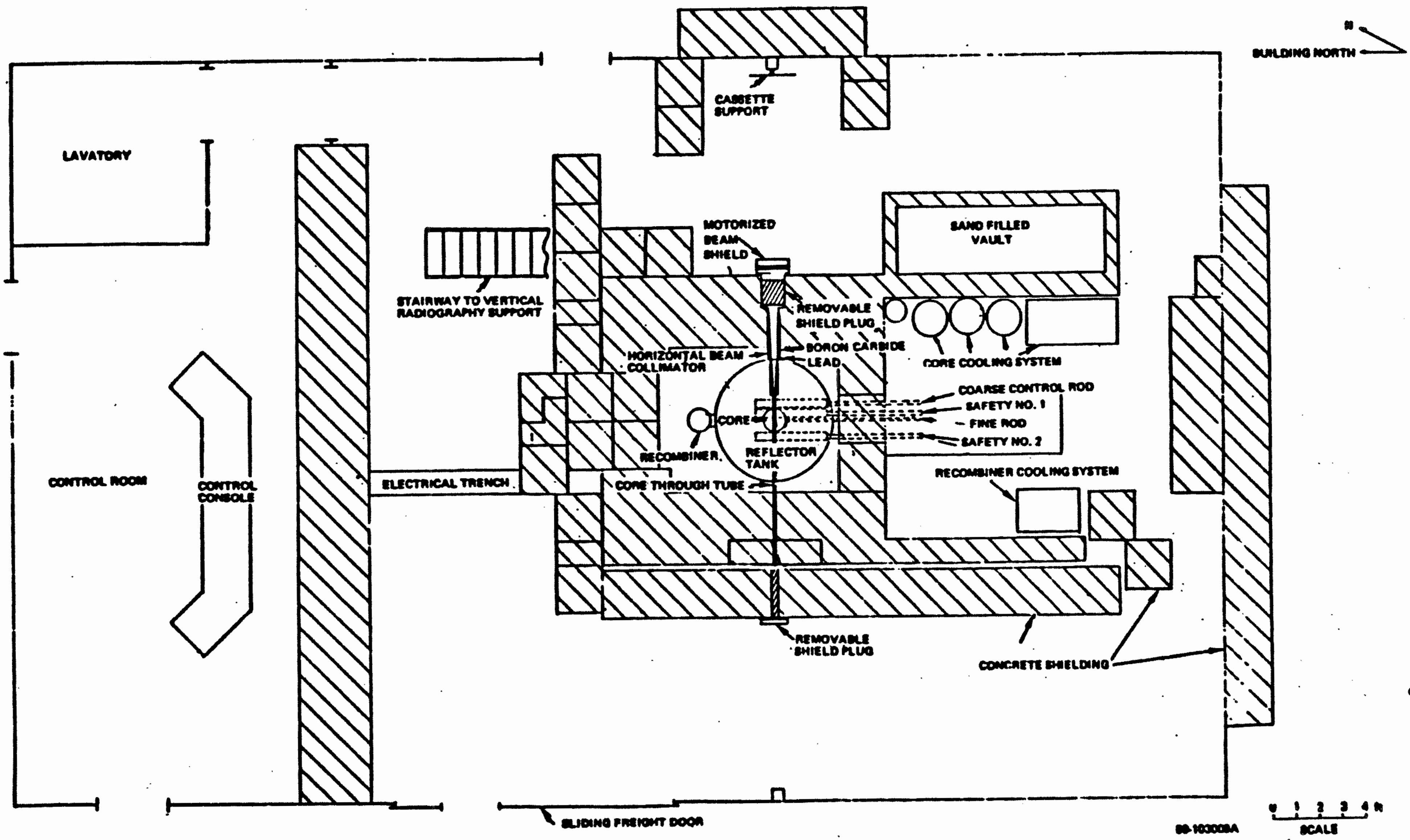


Figure 2. Floor Plan of the L-85 Reactor Building
(Before Decommissioning)

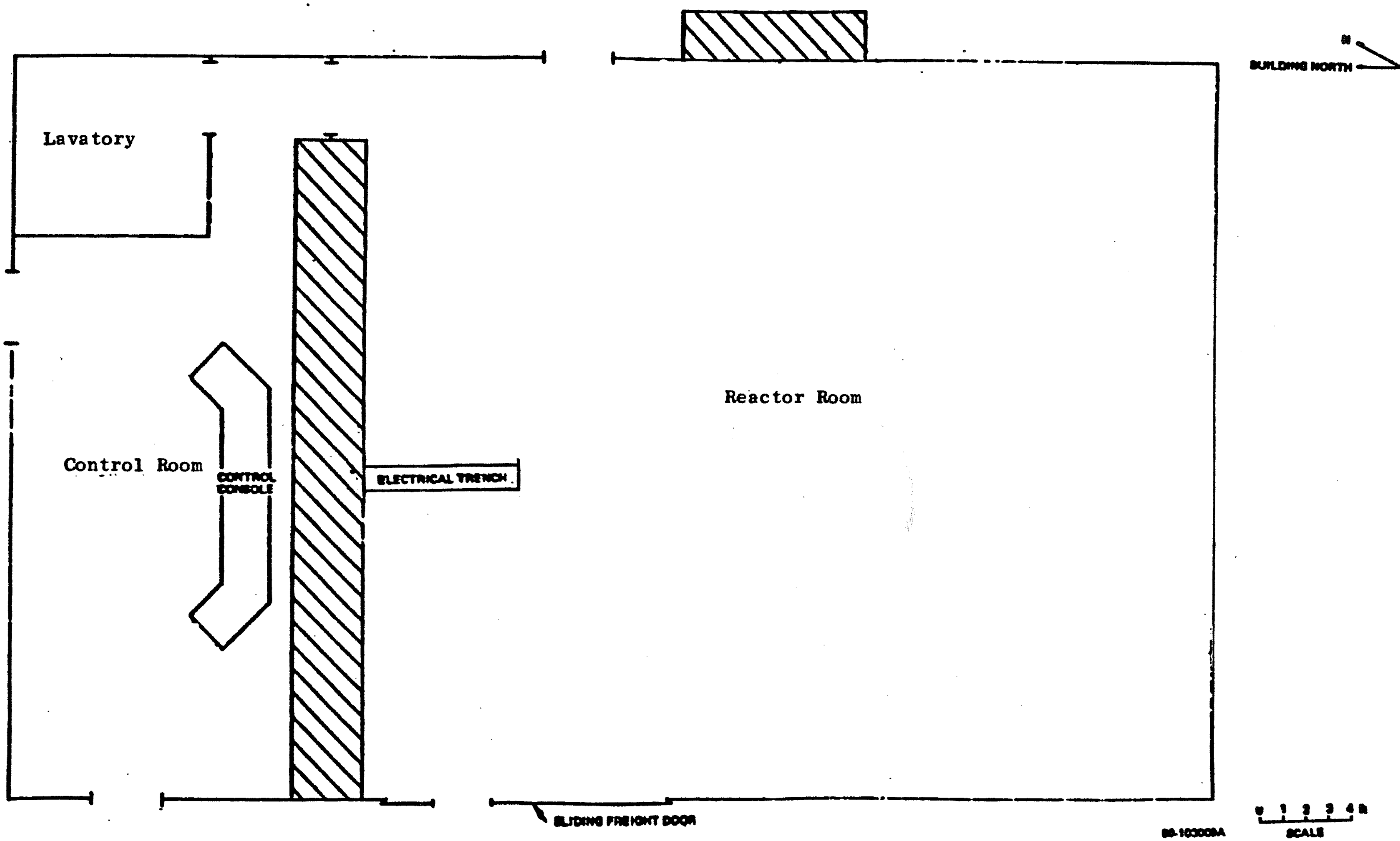
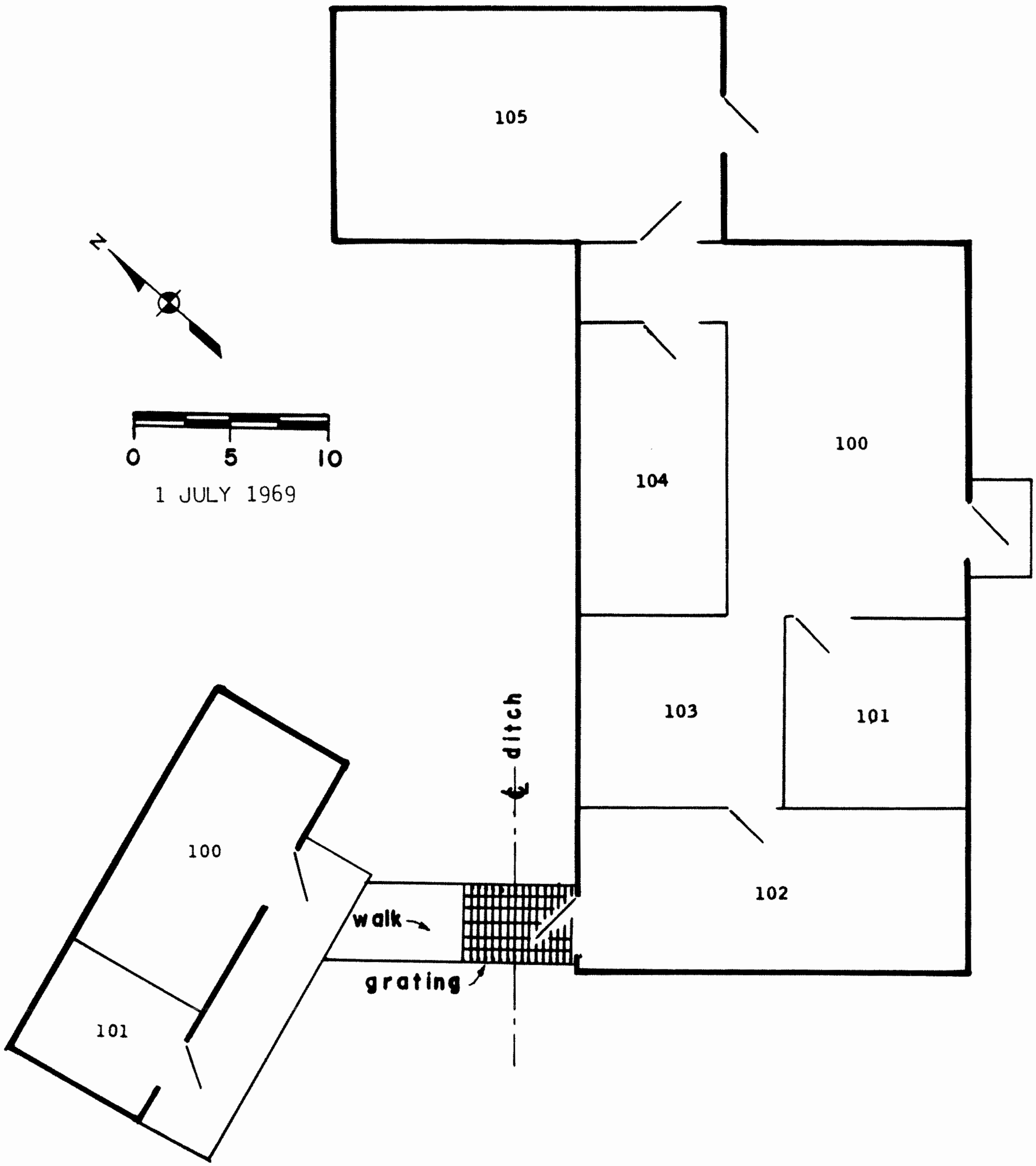


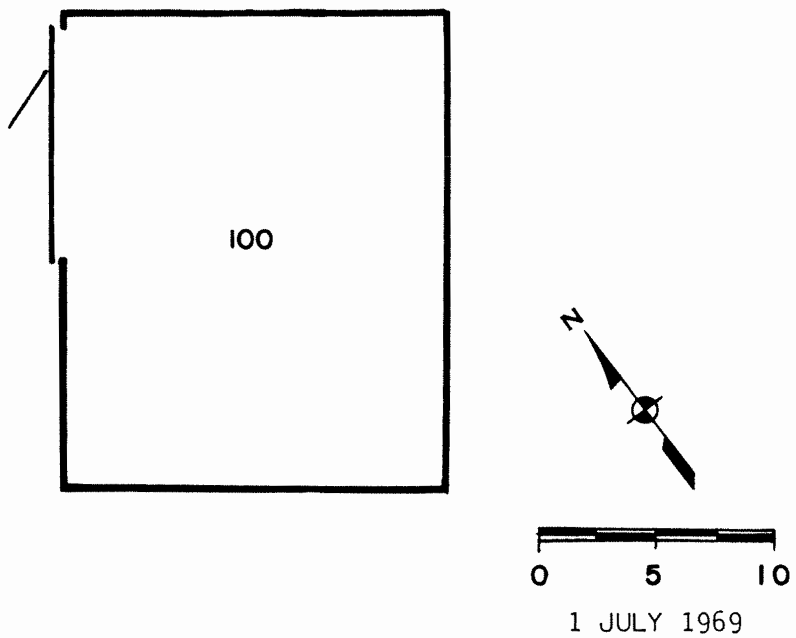
Figure 3. Floor Plan of the L-85 Reactor Building
(After Decommissioning)



STORAGE
BUILDING No. 074

REACTOR KINETICS CONTROL
BUILDING No. 083

Figure 4.



AE-6 FUEL HANDLING
BUILDING No. 453

Figure 5.

III. DECONTAMINATION EFFORTS

Prior to decommissioning, all peripheral items including benches, gas bottles, carts, cabinets, tools, etc., were surveyed and released from Building T093. Decommissioning was performed in accordance with the requirements set forth in "Procedure for Dismantling and Decontaminating the L-85 Reactor Facility," N001DWP000002, Rev. A, V. A. Swanson, July 9, 1985.

Decontamination efforts removed residual floor contamination and also activation-produced radioactivity that occurred during the years of operation. This was done by scabbling and concrete removal.

Except for the spill of rinse water in 1982, no areas of general contamination were historically allowed to exist. No radioactivity or radiation in excess of the Dismantling Order limits was found outside of the reactor room.

Standards for the release for unconditional use of this facility were taken from the U.S. Nuclear Regulatory Commission Regulatory Guide 1.86 (Appendix B), and the U.S. Nuclear Regulatory Commission Dismantling Order, Docket No. 50-375, "Order Authorizing Dismantling of Facility and Disposition of Component Parts" (Appendix C). It should be noted that these criteria are in agreement with the guidance found in the most recent (January 1985) version[2] of American National Standards Institute/Health Physics Society Standard ANSI N13.12, and in the DECON-1 document issued by the State of California[3] in 1977. U.S. Nuclear Regulatory Commission Regulatory Guide 1.86 is reproduced in Appendix B to this report, and pertinent sections are extracted to Table 1. In addition to the acceptance criteria shown in Table 1, the Dismantling Order imposed a limit on exposure rate of 5 microR/hr above ambient background.

TABLE 1

ACCEPTABLE SURFACE CONTAMINATION LEVELS

	Nuclides	dpm/100 cm^2		
		Average[b,c]	Maximum[b,d]	Removable[b,e]
1.	U-nat, U-235, U-238, & associated decay prod.	5,000α	15,000α	1,000α
2.	Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100	300	20
3.	Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1,000	3,000	200
4.	Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above	5,000	15,000	1,000

[a]Where contamination by both alpha- and beta-gamma-emitting nuclides exist, the limits established for alpha- and beta-gamma-emitting nuclides should apply independently.

[b]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 observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[c]Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

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

[e]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 assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

IV. SURVEY SCOPE AND PROCEDURES

A. SCOPE

A sampling inspection plan using variables has been used to demonstrate that the residual contamination in the area is below the limits shown in Table 2. These were taken from the U.S. Nuclear Regulatory Commission Dismantling Order as noted in Appendix C.

TABLE 2

RESIDUAL CONTAMINATION LIMITS FOR UNRESTRICTED RELEASE

Total average over 1 m^2	5,000 dpm/100 cm^2
Total maximum over 100 cm^2	15,000 dpm/100 cm^2
Removable contamination	1,000 dpm/100 cm^2
Ambient exposure rate (microR/hr)	Background + 5

The sampling inspection plan that was used is based on a uniform 3-meter square (10-ft square) grid superimposed on the area. A 3-meter square grid has been adopted to be consistent with both NRC and State of California guidance. The actual grid on the floor of each room was benchmarked in the northwest corner of the room. An identical grid pattern was reflected onto the ceiling. A similar grid structure was also applied to the walls, benchmarked in the upper left corner of each wall. Each survey area has been identified with codes indicating the surface (SB = steel beam; CB = center beam; P = pavement; D = drain; F = floor; C = ceiling; N, E, S, W = north, east, south, and west walls, respectively) and a two-figure Cartesian coordinate showing the distance in meters from a local benchmark in orthogonal directions.

Within each square defined by the grid lines, a single 1-m^2 area was surveyed. Each area was outlined by felt marker or paint, with its coordinates marked within or beside the 1-m^2 area. The location of this 1-m^2 area was left to the surveyor's judgment; it was to be the area that, in his judgment, was most likely to have retained the most residual contamination of any similar area within the grid square. The surveyor was instructed to do this conscientiously to assure that any significant residual contamination would be detected before a report of acceptability was made to a regulatory agency. The use of a predetermined grid with discretion for the exact location provides a biased-uniform survey; selection of one 1-m^2 area out of the nine within each grid square provides an 11% sampling of the surface.

As can be seen in Figure 1, there are three buildings around the perimeter of Building T093. Building T453 was a fuel handling building; T083 was an office and control room for the KEWB reactor (previously decommissioned under the jurisdiction of DOE); T074 was used for processing photographic oscillograph paper, and emergency supplies storage. Since radioactive materials were not used in Buildings T083 and T074, a reduction in sampling was applied to this area. In this area, a 1-m^2 sample was measured from every other 9-m^2 grid. This reduced inspection plan was also applied to Building T453 (fuel handling building) after a 100% floor survey, before and after the floor tile was removed, indicated no radioactive contamination.

Sampling inspection consists of a <u>sampling plan</u> for selection of items to be tested, in this case, locations to be measured for radioactivity, and the method of analysis. The <u>sampling plan</u> used for this phase was to inspect one 1-m^2 area out of every 3-meter square grid throughout the areas.

This 11% inspection (compared to 10% as recommended by the State of California) was used for this survey, except for a 5% survey in the non-reactor areas and the ceiling in the Reactor Room.

The 1-m^2 area chosen by the procedure described above is first measured for total alpha and beta activity and then for removable activity, as described in Section IV.B.

The values resulting from these measurements (converted to the proper units) are analyzed in the following manner:

The test statistic $\bar{x} + ks$ is compared to the acceptance limit U,

where

$\bar{x}$ = average (arithmetic mean of measured values)

s = observed sample standard deviation

k = tolerance factor calculated from the number of samples to achieve desired sensitivity for this test

u = acceptance limit.

The State of California[3] has stated that the consumer's risk of acceptance (beta) at 10% defective [Lot Tolerance Percent Defective (LTPD)] must be 0.1. For these choices of beta and LTPD, $K_\beta = K_2 = 1.282$.

The number of samples is n. Values of k for each sample size are calculated from (4):

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

For example, for n = 10, k = 2.41; n = 100, k = 1.47; n = 1000, k = 1.34.

The criteria for acceptance are presented as a <u>plan of action</u>. The <u>plan of action</u> is:

1) Acceptance: If the test statistic ($\bar{x} + ks$) is less than or equal to the limit (U) accept the region as clean. (If any single measured value exceeds the limit, decontaminate that location to below the limit, but do not change the value in the analysis.)

2) Collect additional measurements: If the test statistic ($\bar{x} + ks$) is greater than the limit (U), but $\bar{x}$ itself is less than U, independently resample and combine all measured values to determine if $\bar{x} + ks < U$ for the combined set; if so, accept the region as clean. If not, reject the region.

3) Rejection: If the test statistic ($\bar{x}$ + ks) is greater than the limit (U) and $\bar{x} > U$, reject the region.

Step 2 takes advantage of the improved discrimination of the acceptance test resulting from an increase in the number of samples to reduce the risk of rejecting a region that is acceptably clean. This false rejection should be avoided if possible to avoid the unnecessary expense of further decontamination. If the result of the additional inspection does not show acceptability, further decontamination is required. Step 3 assures that no truly contaminated area will be accepted. The contamination measurements made at the inspected location may be used to guide further decontamination, but these locations should be avoided in the subsequent inspection.

B. PROCEDURES

The following procedures were used in performing this survey.

1. Average Contamination Measurements

1) Identify 1-m^2 area to be measured.

2) With a portable scaler (Ludlum Model 2220 - ESG scaler, or equivalent) set for 5-min count time, use an alpha probe (Ludlum Model 43-1 or equivalent) or a beta probe (Ludlum Model 44-9, Technical Associates Model P-11, or equivalent) and uniformly scan the area. (Watch for and note any "hot spots" where the radioactivity may exceed the average limit. These are to be resurveyed later.)

3) Record the location and total count.

4) The total count is converted to dpm/100 cm^2 total surface activity by:

$$SA_T = \frac{(C - B)}{5}\frac{E(100)}{A}$$

where

SA_T = Total surface activity in dpm/100 cm^2

C = Total count in 5 min

5 = Count time, min

B = Background count in 5 min (generally 0-5 for alpha and about 200-220 for beta)

E = Efficiency factor, dpm/cpm (generally 4 for alpha and 7 for beta)

100 = 100 cm^2 standard area

A = Probe sensitive area (69 cm^2 for Ludlum Model 43-1 alpha scintillator; 20 cm^2 for Ludlum Model 44-9 and Technical Associates model P-11 pancake G-M).

2. Maximum Contamination Measurements

1) Return to any area identified as having a "hot spot."

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

3) Record the location and total count as a "hot spot" measurement.

4) The total count is converted to dpm/100 cm^2 as shown above.

3. Removable Contamination Measurements

1) Identify 1-m^2 area to be measured.

2) Using a Whatman 540 filter paper (2.4 cm diameter), wipe a "Z" or "S" pattern, with legs approximately 6 in. long, so as to sample removable contamination from an area of approximately 100 cm^2.

3) Place smear paper in file card "book" until ready for counting.

4) Count radioactivity using gas-flow proportional counter (NMC Model ACS-77 or equivalent) for 5 min.

5) Record the location and both the total alpha count and the total beta count.

6) The total counts are converted to dpm/100 cm^2 removable surface activity by:

$$SA_T = \frac{(C - B)E}{5}$$

where the appropriate alpha and beta backgrounds and efficiency factors are used. Backgrounds are typically 1-3 counts for alpha and 120-150 counts for beta. Efficiency factors are about 4 dpm/cpm for alpha and beta.

4. Soil

Samples of soil were taken from outside the facility to determine if any environmental release had occurred that could have resulted in soil contamination. Two samples were taken from the open dirt area directly across the driveway from the large door to the Reactor Room. Three others were taken from the drainage ditch leading away from the facility. All five samples showed similar amounts of naturally occurring radionuclides associated with natural uranium and natural thorium and K-40. In addition, three samples showed small amounts of Cs-137.

The average results for the five samples show the following activity concentrations (pCi/g).

Natural Uranium:	
U-235	0.06 ± .01
Ra-226	1.36 ± .18
Bi-214	0.89 ± .11
Pb-214	0.93 ± .24
Natural Thorium:	
Ac-228	1.50 ± .71
Pb-212	1.19 ± .06
Tl-208	0.42 ± .04
Cs-137	0.20 ± .13
K-40	23.37 ± 1.31

These results show an apparent loss of radon gas from the samples (Rn-222 from the natural uranium, Rn-220 from the natural thorium) and so the best estimate of the parent activities would be 1.36 pCi/g for U-238, and 1.50 pCi/g for Th-232. [The value for Tl-208 (0.42 pCi/g) indicates an activity of 1.16 pCi/g for its grandparent Pb-212, which is in good agreement with the value of 1.19 pCi/g measured.]

The Cs-137, found in three of the five samples, may be from global fallout from weapons testing. The maximum value found, 0.32 pCi/g, does not suggest contamination from the L-85. The two samples with no detectable Cs-137 were the two closest to the reactor facility. (For comparison, environmental monitoring at Hanford and Los Alamos in 1973 and 1976 showed Cs-137 concentrations in soil in the range of 0.4-2.8 pCi/g in noncontaminated areas.)

V. SURVEY RESULTS

The survey of this area was conducted using the survey plan previously described. A summary of the survey results appear below in Table 3 and 4. The results of the mathematical statistical analysis (Appendix A) are shown as "Inspection Test Statistic."

TABLE 3
SUMMARY OF SURVEY RESULTS
(BUILDING T093)

Measurement	Number of Locations	Average Value (dpm/100 cm^2)	Maximum Value (dpm/100 cm^2)	Inspection Test Statistic	Limit
Average alpha	83	10.7	63.0	22.7	5,000
Maximum alpha	0	0	0	-	15,000
Removable alpha	83	0	0	1.6	1,000
Average beta	83	132.8	3102.0	1000.2	5,000
Maximum beta	0	0	0	-	15,000
Removable beta	83	9.4	98.0	33.4	1,000
Ambient exposure rate (microR/hr)	83	14.2	21.3	16.7	18.9

No hot spots were found; therefore, no maximum alpha or beta measurements were made.

One location in the Reactor Room that had not been included in the established inspection locations showed 4923 dpm ß/100 cm^2 during the final inspection. While this does not exceed the acceptance limit of 5000 dpm/100 cm^2, it was scabbled and remeasured. The result was 2542 dpm ß/100 cm^2. This value was not entered in the acceptance test data.

The mean ambient exposure rate (Figure 10) was found to be 13.9 microR/hr in the Reactor Room; thus, the acceptance criteria became 18.9 microR/hr.

During the final survey, the maximum ambient exposure rate found in the Reactor Room was 21.3 microR/hr. Remedial cleaning (concrete removal) at this location reduced the ambient exposure rate to 18.2 microR/hr, which is below the acceptance limit of 18.9 microR/hr. Two other locations were also resurveyed after the concrete removal, and the readings at the three locations are as follows:

Location	Before Remedial Cleanup (microR/hr)	After Remedial Cleanup (microR/hr)
F7, 7	19.2	17.3
F4, 5	20.2	17.8
F7, 4	21.3	18.2

The subsequent readings are indicated by the tips of the arrows extending down from the original values on Figure 10.

TABLE 4

SUMMARY OF SURVEY RESULTS
(BUILDINGS T083, T074, AND T453)

Measurement	Number of Locations	Average Value (dpm/100 cm^2)	Maximum Value (dpm/100 cm^2)	Inspection Test Statistic	Limit
Average alpha	98	4.1	17.2	17.1	5,000
Maximum alpha	0	0	0	-	15,000
Removable alpha	98	0.8	4.0	1.1	1,000
Average beta	98	4.5	1987.0	930.2	5,000
Maximum beta	0	0	0	-	15,000
Removable beta	98	2.3	93.0	53.6	1,000
Ambient exposure rate (microR/hr)	98	12.8	23.1	18.0	19.7

No hot spots were found; therefore, no maximum alpha or beta measurements were made.

In the non-reactor areas there was no activation. The observed variability and somewhat higher values are attributed to variation in the rock outcroppings pinup, radiation from the RMDF, and shielding by the rocks.

Soil samples, as previously discussed, showed no evidence of radioactivity due to facility operations.

With the exception of the slightly elevated ambient exposure rate values, the maximum radiation measurement values and the inspection test statistics are well below the appropriate limits. The results summarized in these tables confirm that all areas are acceptable for release for unrestricted use at the present time.

The survey data for each test characteristic are displayed as cumulative probability distributions in Figures 6 through 15. These figures show each value, arranged in order of magnitude from left to right, and a straight line representing the derived Gaussian distribution. The acceptance limit in each case is shown at or near the top edge of each graph. A vertical line at approximately 1.5 standard deviations above the mean represents the value of k used in the inspection test. The Gaussian distribution line must pass below the "x" marking the intersection of the "k" line and the acceptance limit line. In Figure 10 and in Figure 15, several measured values of the ambient exposure rate are shown above the acceptance limit. In the Reactor Building (Figure 10), these were reduced by removal of activated concrete near those locations. The over-limit values in the non-reactor areas resulted from radioactive material at the nearby Radioactive Materials Disposal Facility (RMDF).

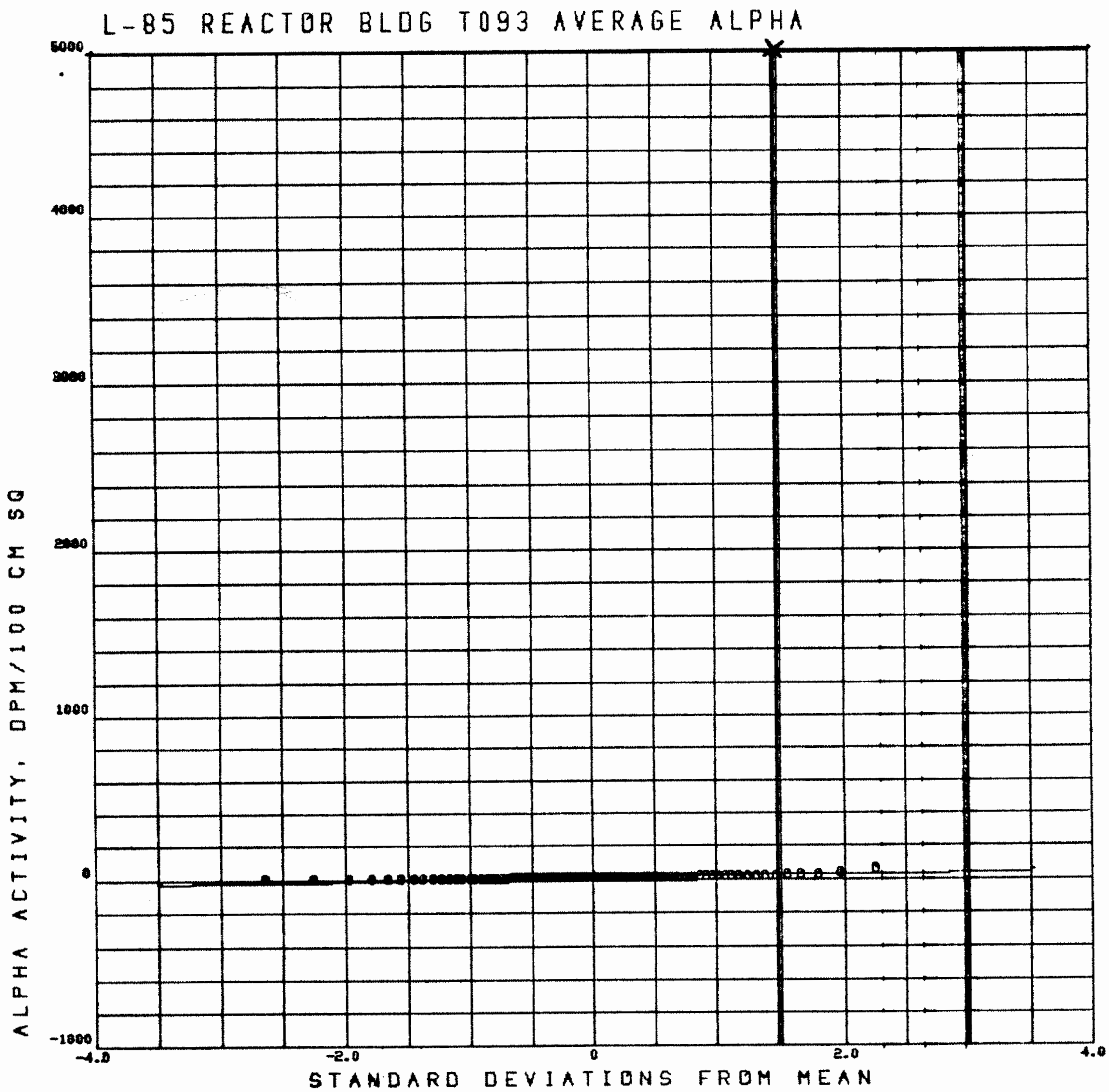


Figure 6. Average Alpha Activity (Reactor Building)

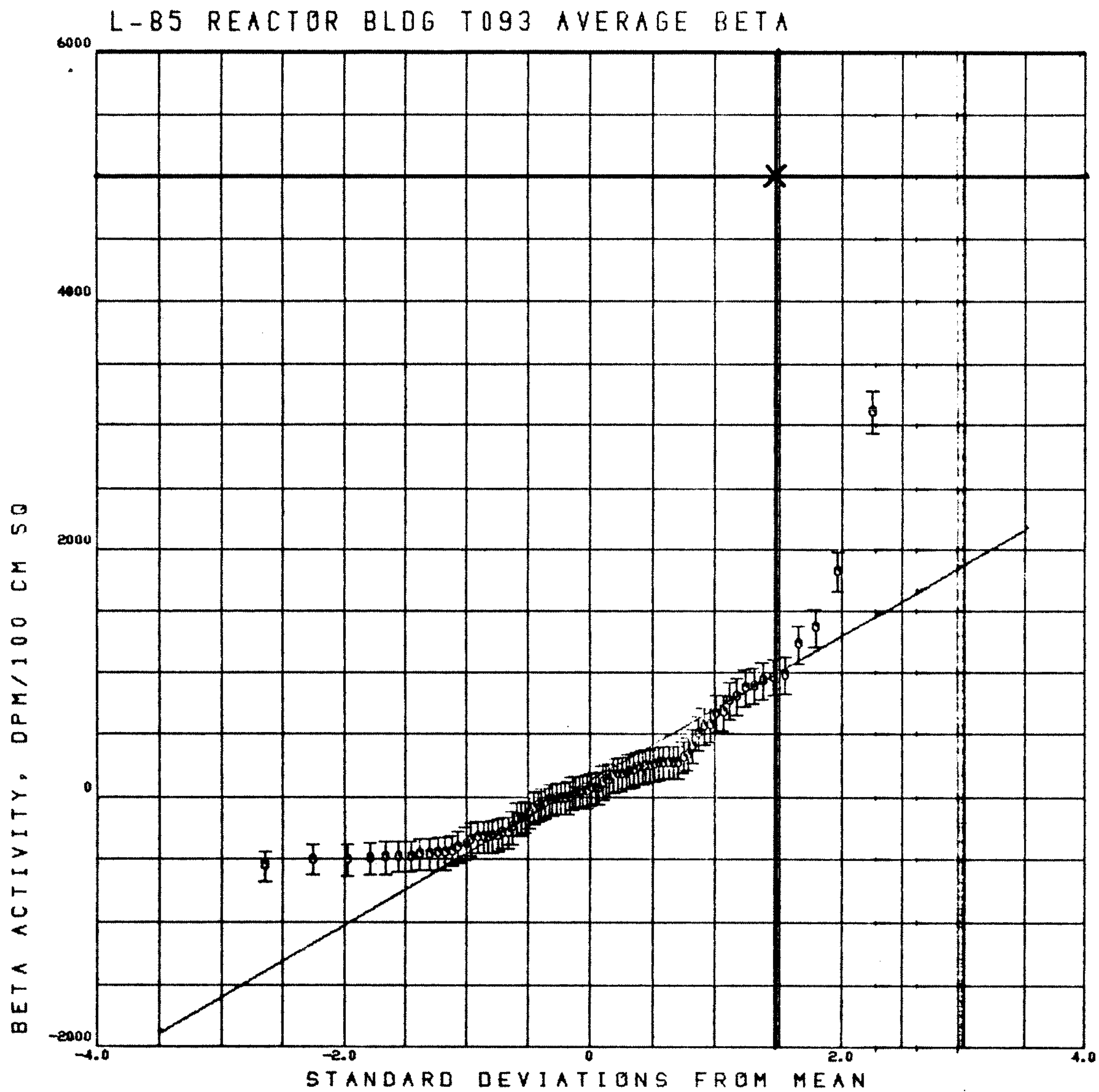


Figure 7. Average Beta Activity (Reactor Building)

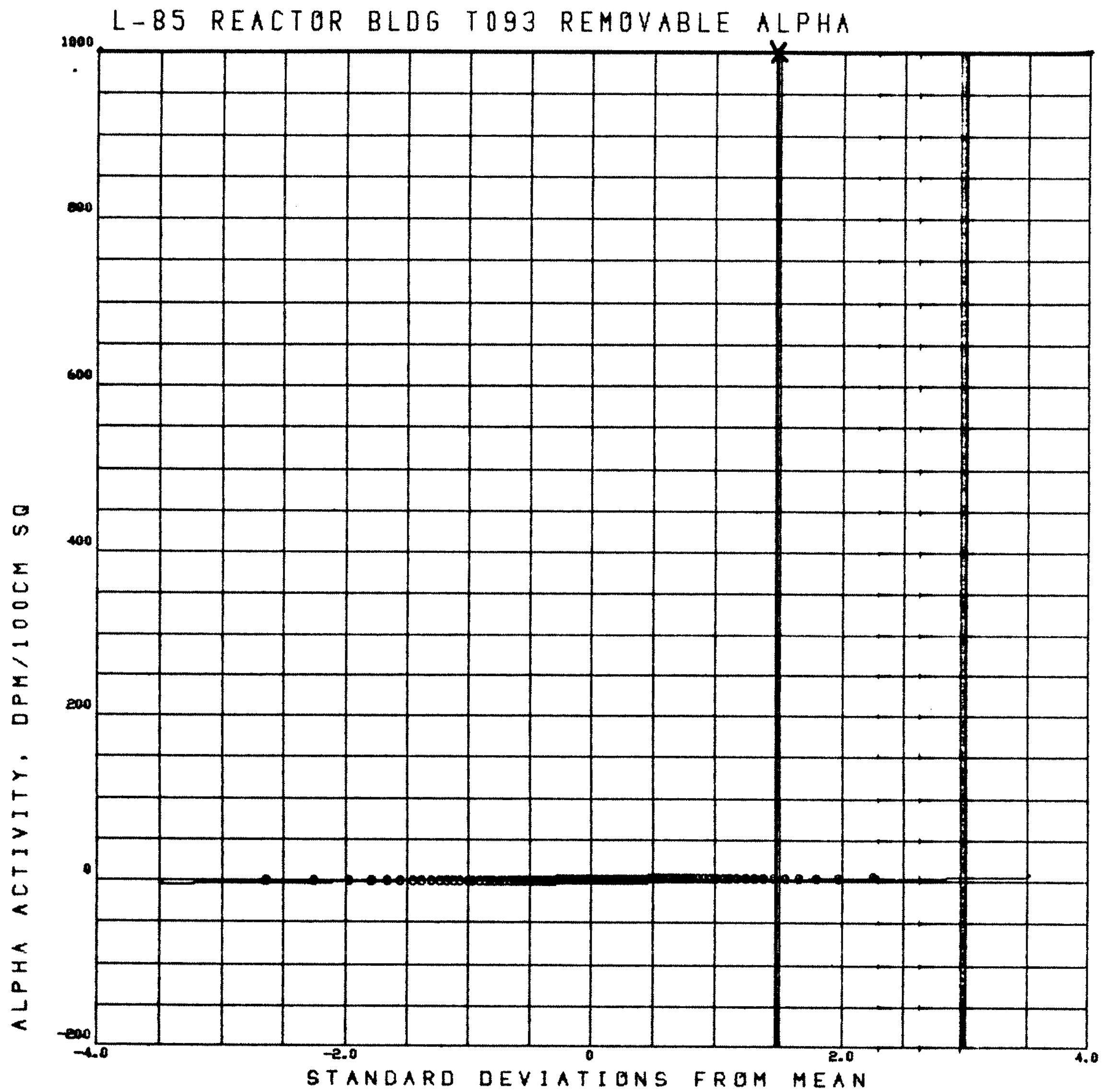


Figure 8. Removable Alpha Activity (Reactor Building)

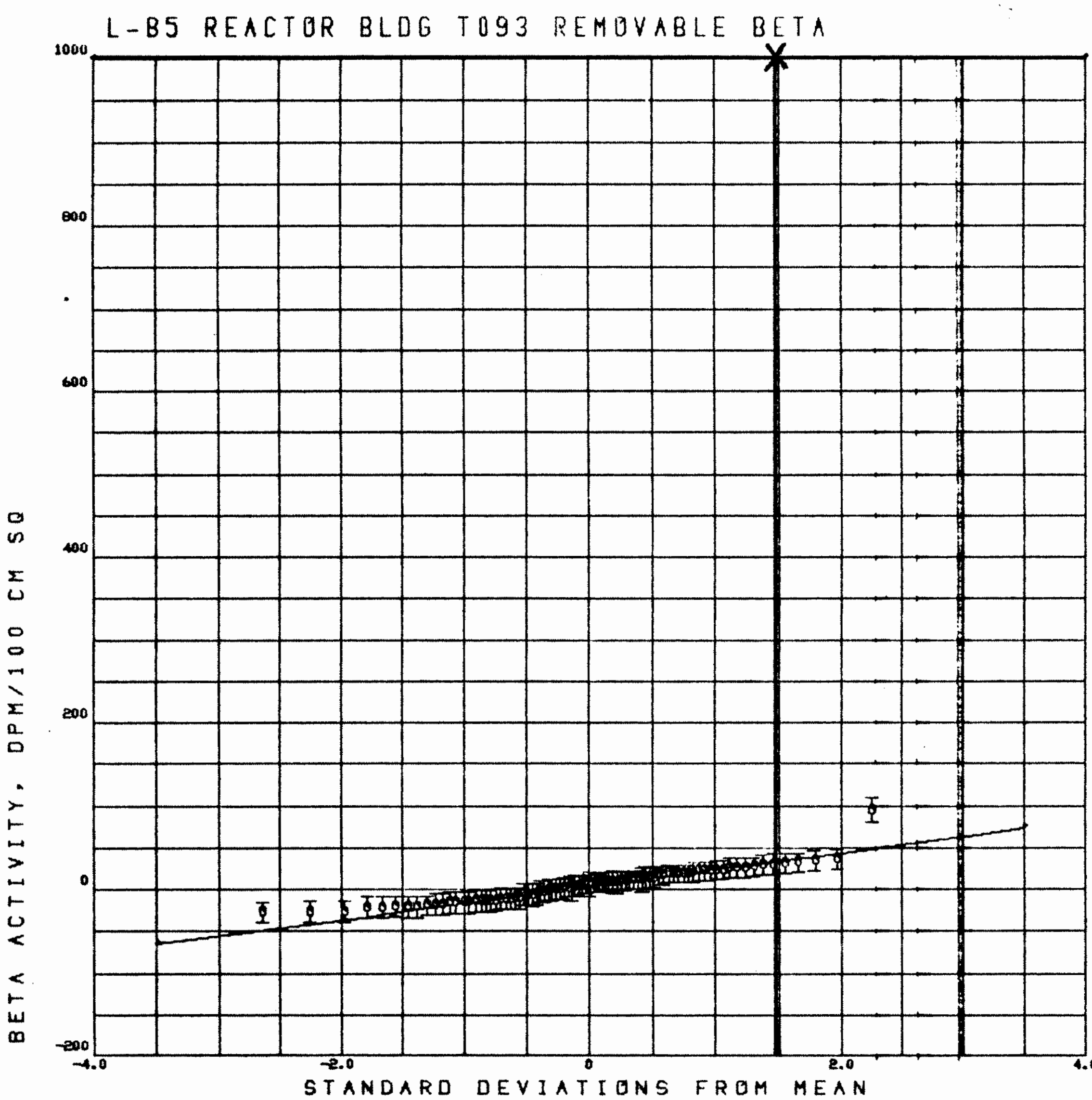


Figure 9. Removable Beta Activity (Reactor Building)

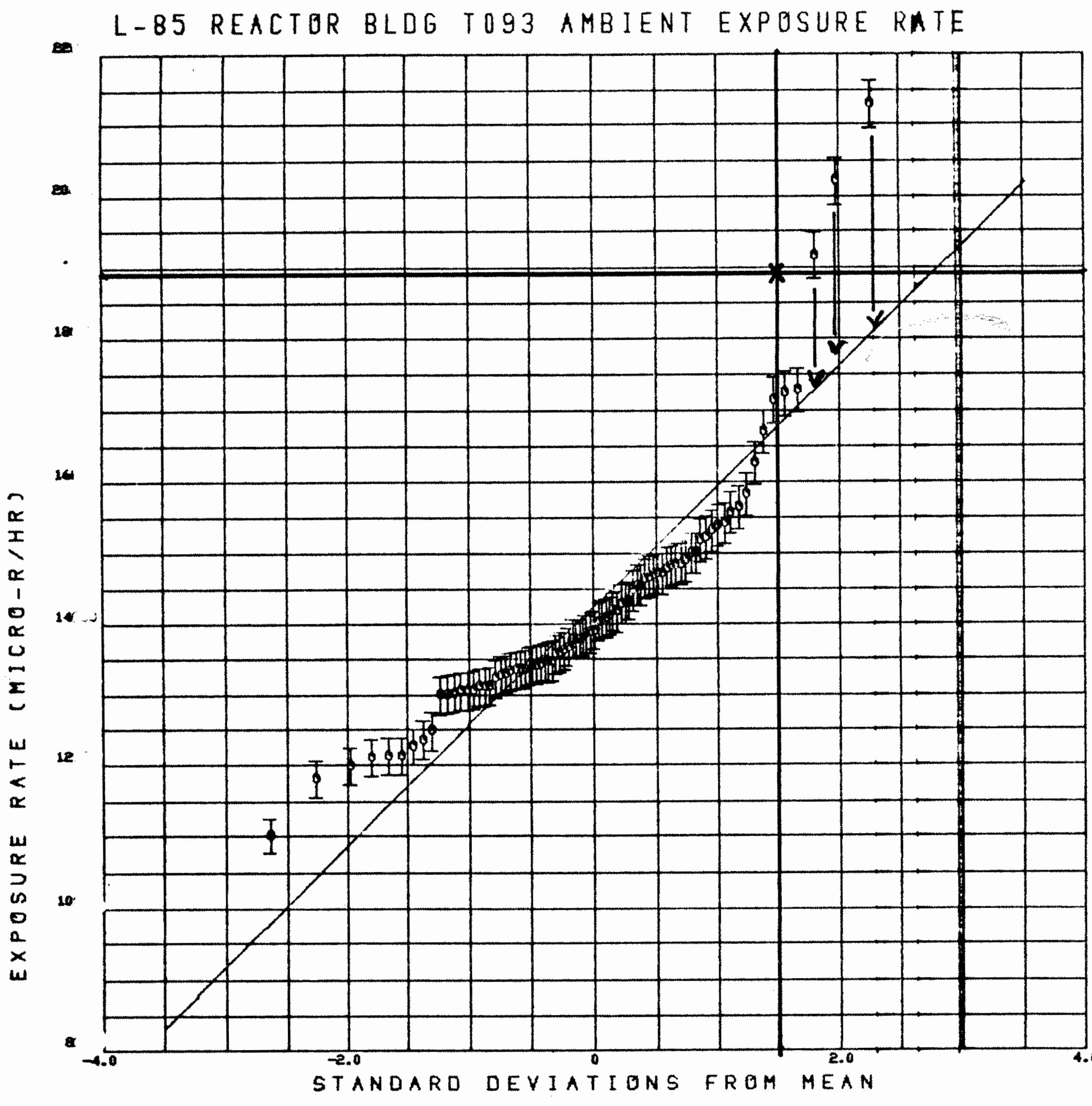


Figure 10. Ambient Exposure Rate (Reactor Building)

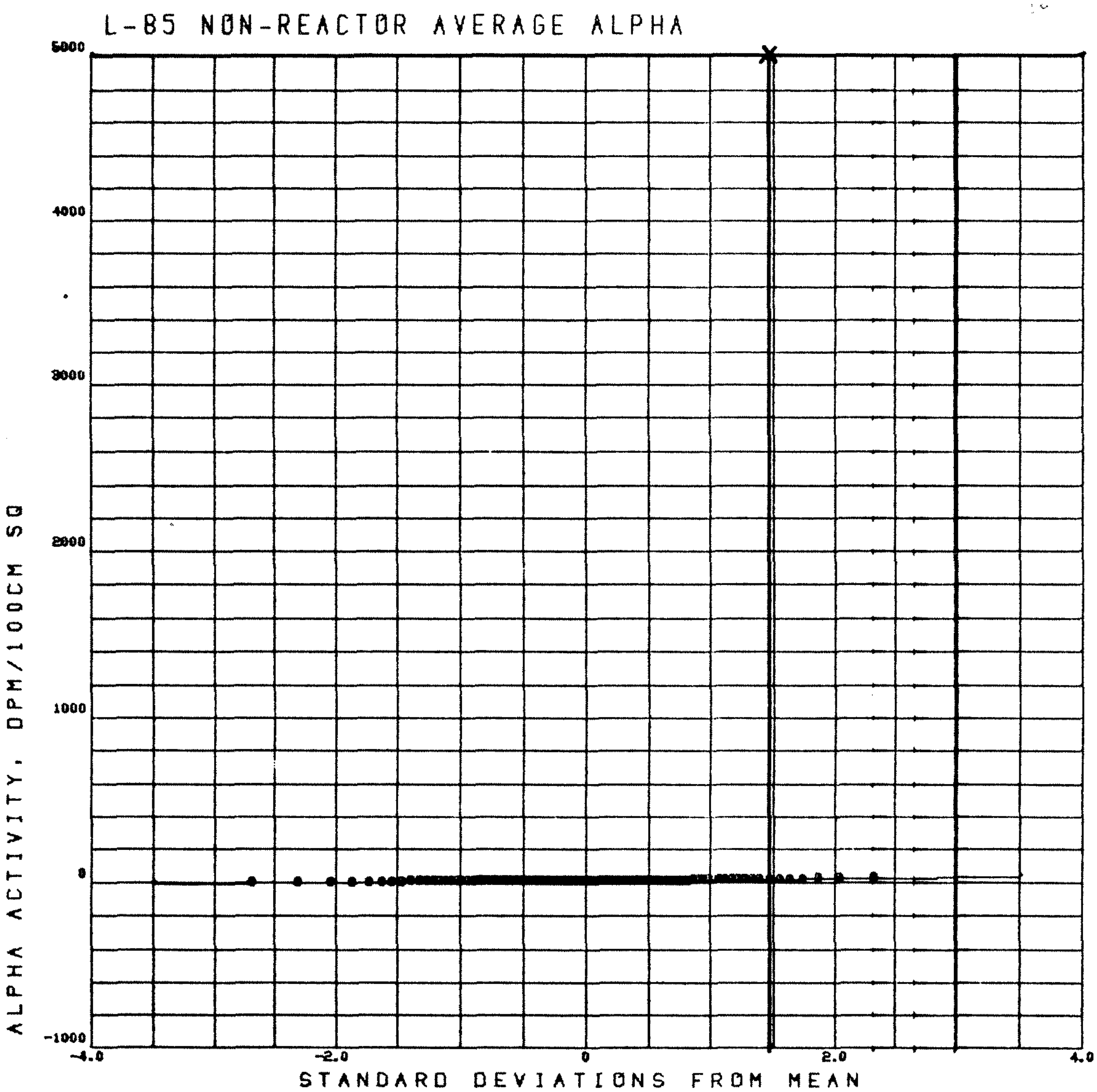


Figure 11. Average Alpha Activity (Non-Reactor Areas)

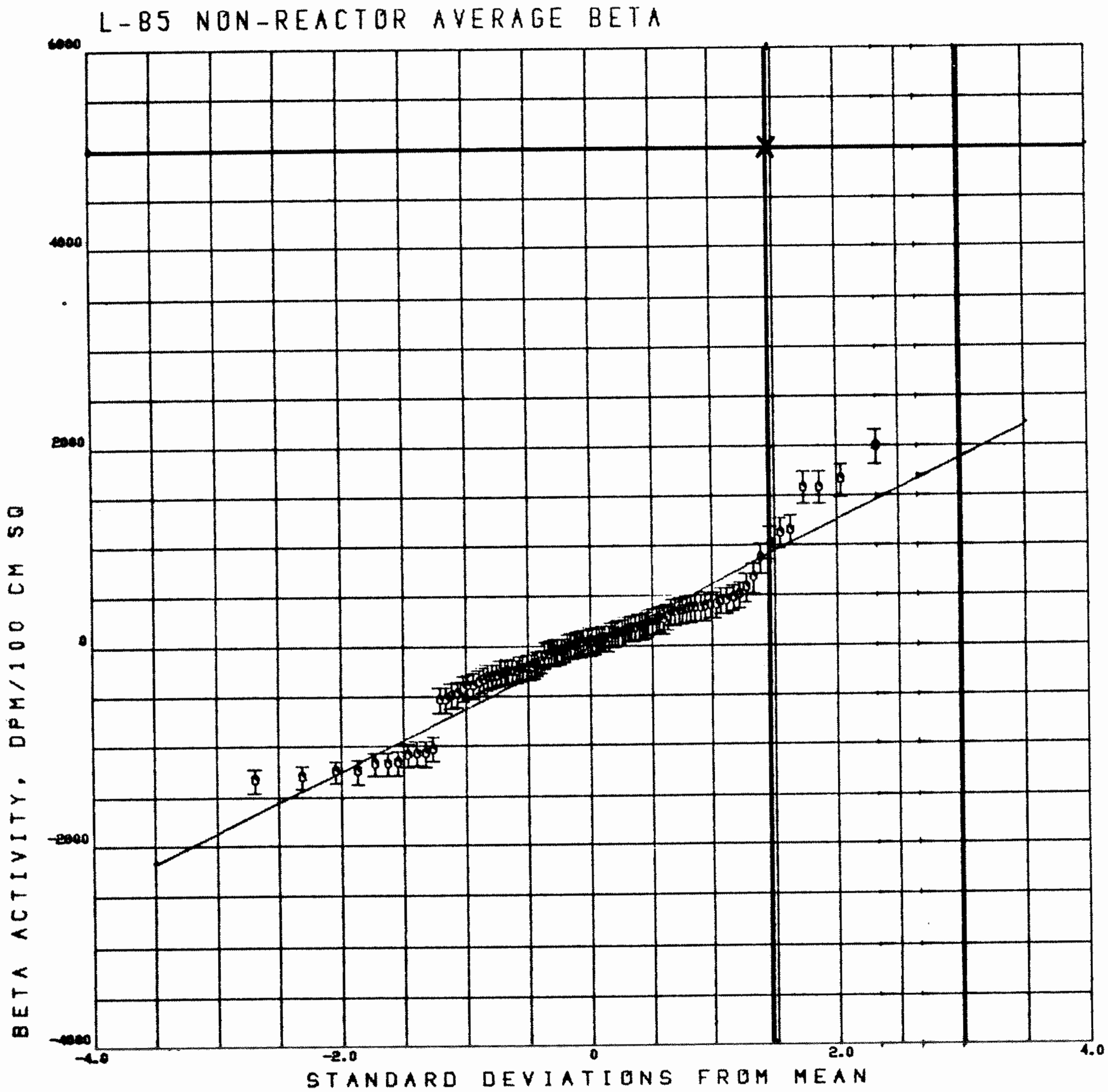


Figure 12. Average Beta Activity (Non-Reactor Areas)

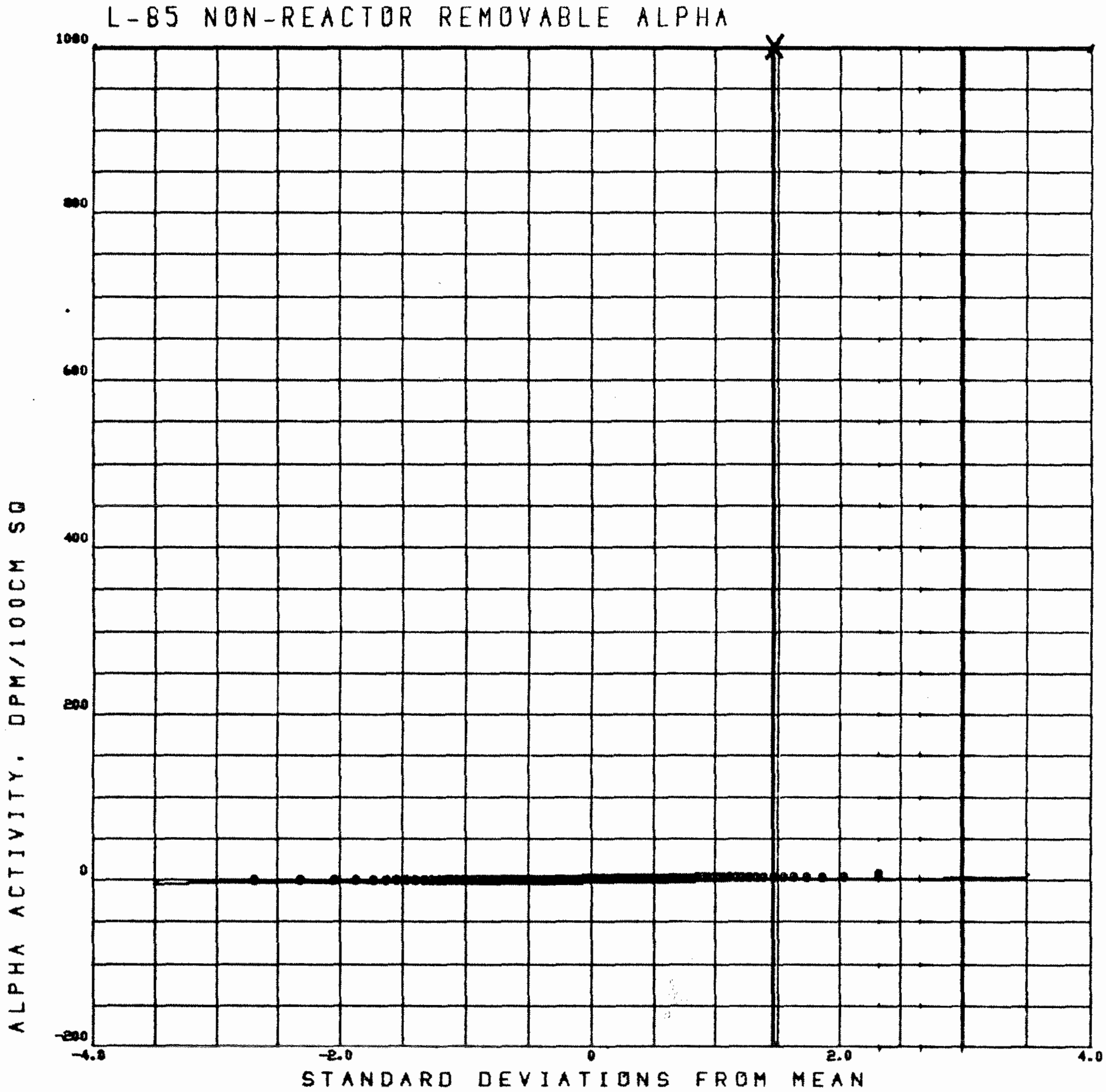


Figure 13. Removable Alpha Activity (Non-Reactor Areas)

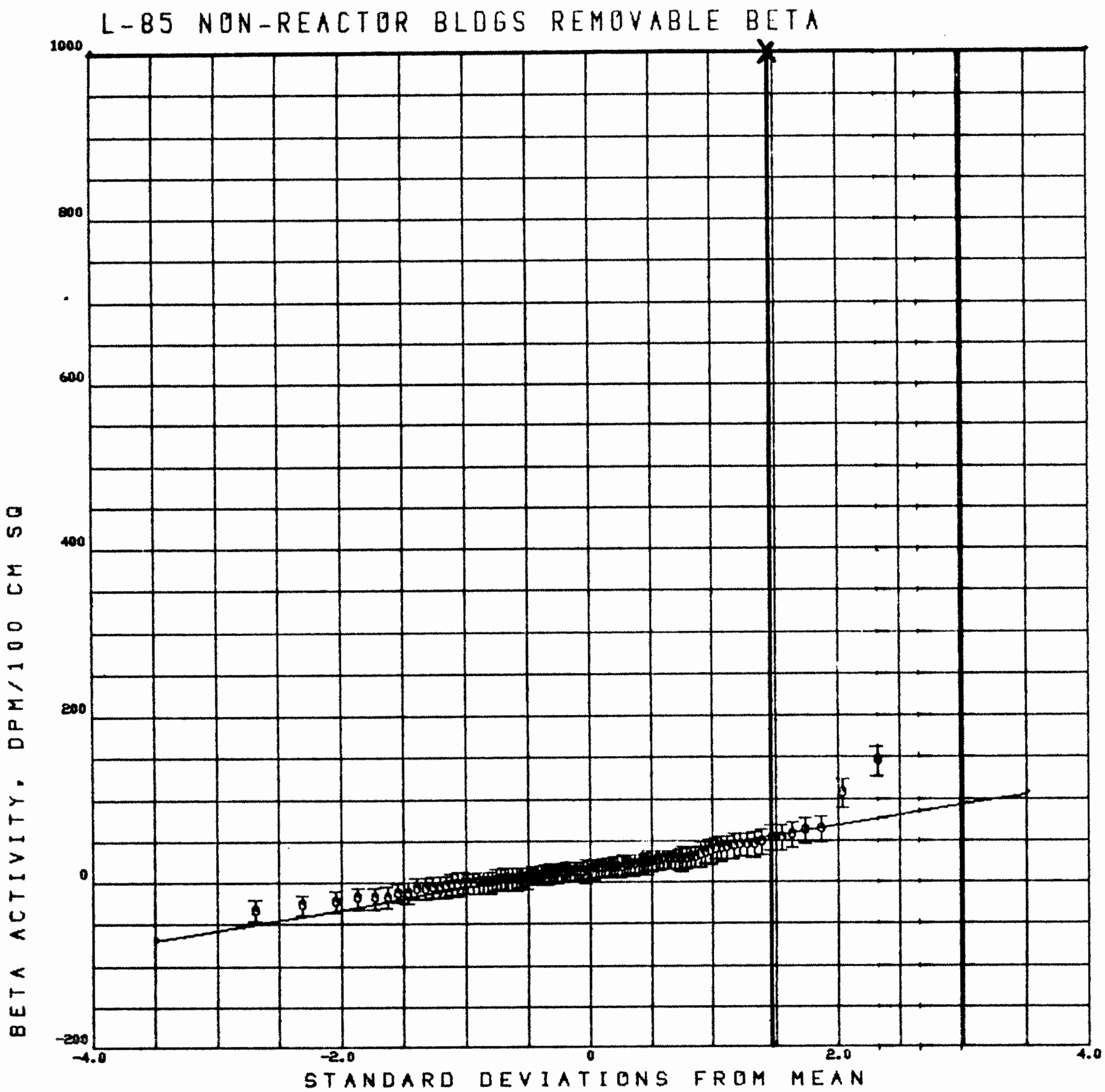


Figure 14. Removable Beta Activity (Non-Reactor Areas)

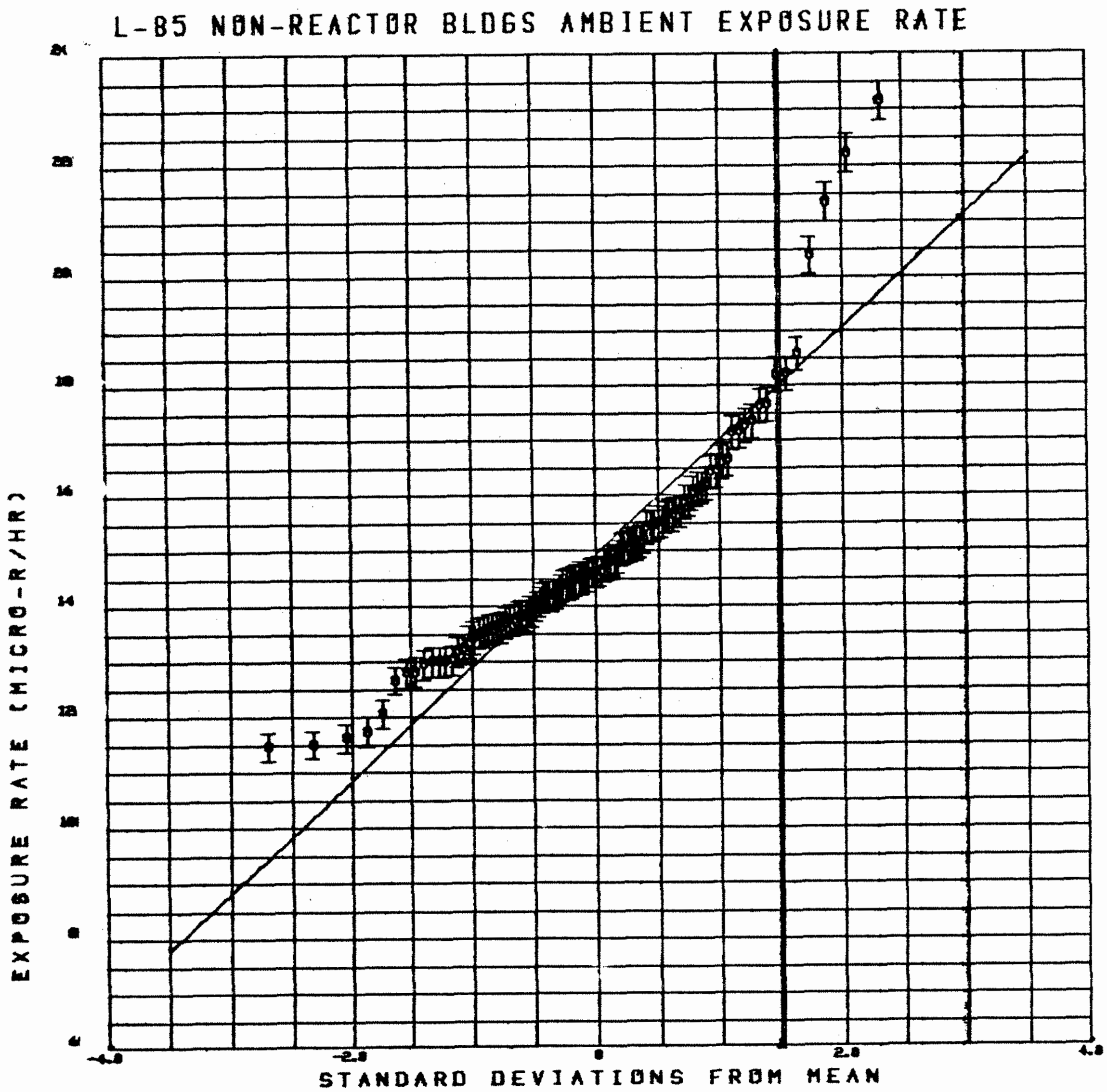


Figure 15. Ambient Exposure Rate (Non-Reactor Areas)

Resurvey of Reactor Room

On January 23, 1986, the reactor room floor was resurveyed for ambient radiation. This was done because of concerns expressed by NRC Region V regarding the appropriateness of using NaI (Tl) scintillation detectors for measuring exposure rate. These scintillation detectors record interactions of gamma-ray photons with the detector material and, strictly, cannot be calibrated by use of a calibration source to read exposure rate (μR/h) in general. To correct this shortcoming, and to provide high-quality data, two scintillation detectors with scalers were calibrated to the ambient gamma-ray spectrum by direct comparison to two Reuter-Stokes high pressure ion chambers (HPIC). The HPIC monitors provide true exposure rate in principle, and had been calibrated by use of a Co-60 calibration source.

The intercomparison was done by taking 5 sets of readings from each of the two HPICs, by two independent observers. Each set covered a time span of approximately 1 min. The HPIC near the Industrial Security Control Center averaged 12.4 μR/h. The HPIC near Building T363 averaged 12.0 μR/h. Concurrent readings with the scintillation detectors gave conversion factors of 4.738 x 10^{-3} (μR/h)/cpm for instrument number 596003 and 4.669 x 10^{-3} (μR/h)/cpm for instrument number 596007. 211 214

Measurements were made in the reactor room with the two scintillation detectors suspended 1 meter above the local surface, over the center of the 1-meter-square grids defined during the earlier survey. The results of these measurements are shown separately in Figures 16 and 17. These figures show readings corresponding to unactivated concrete, in the range of 10 to 15 μR/h, and several values clearly indicating activated concrete, above 16 μR/h.

Measurements were also made just southeast of the reactor room, outside, over natural soil. These results are shown in Figures 18 and 19 and demonstrate the inherent variability of the measurements. Roughly half the

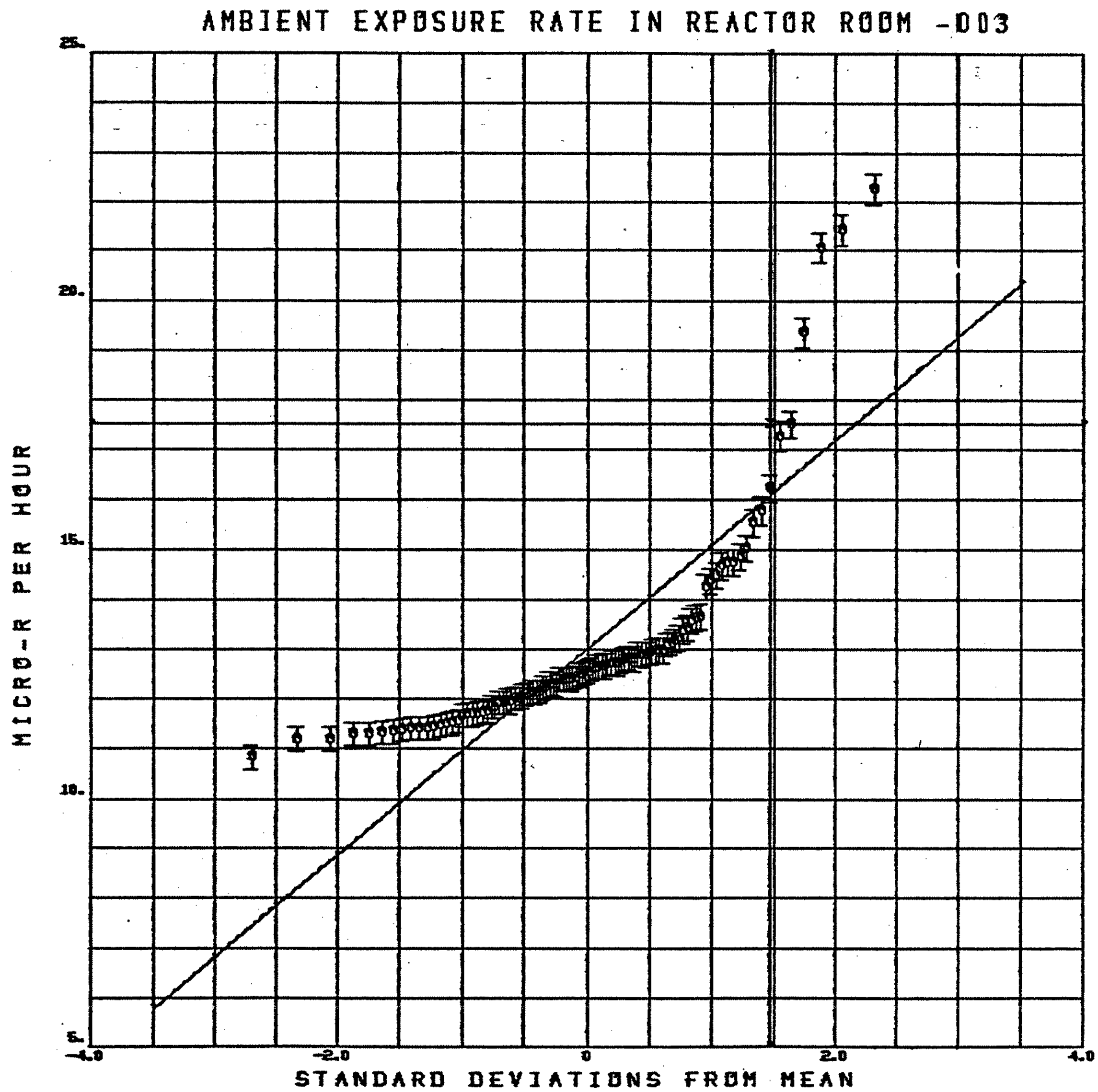


Figure 16. Results of Resurvey for Ambient Exposure Rate in Reactor Room (Instrument 596003)

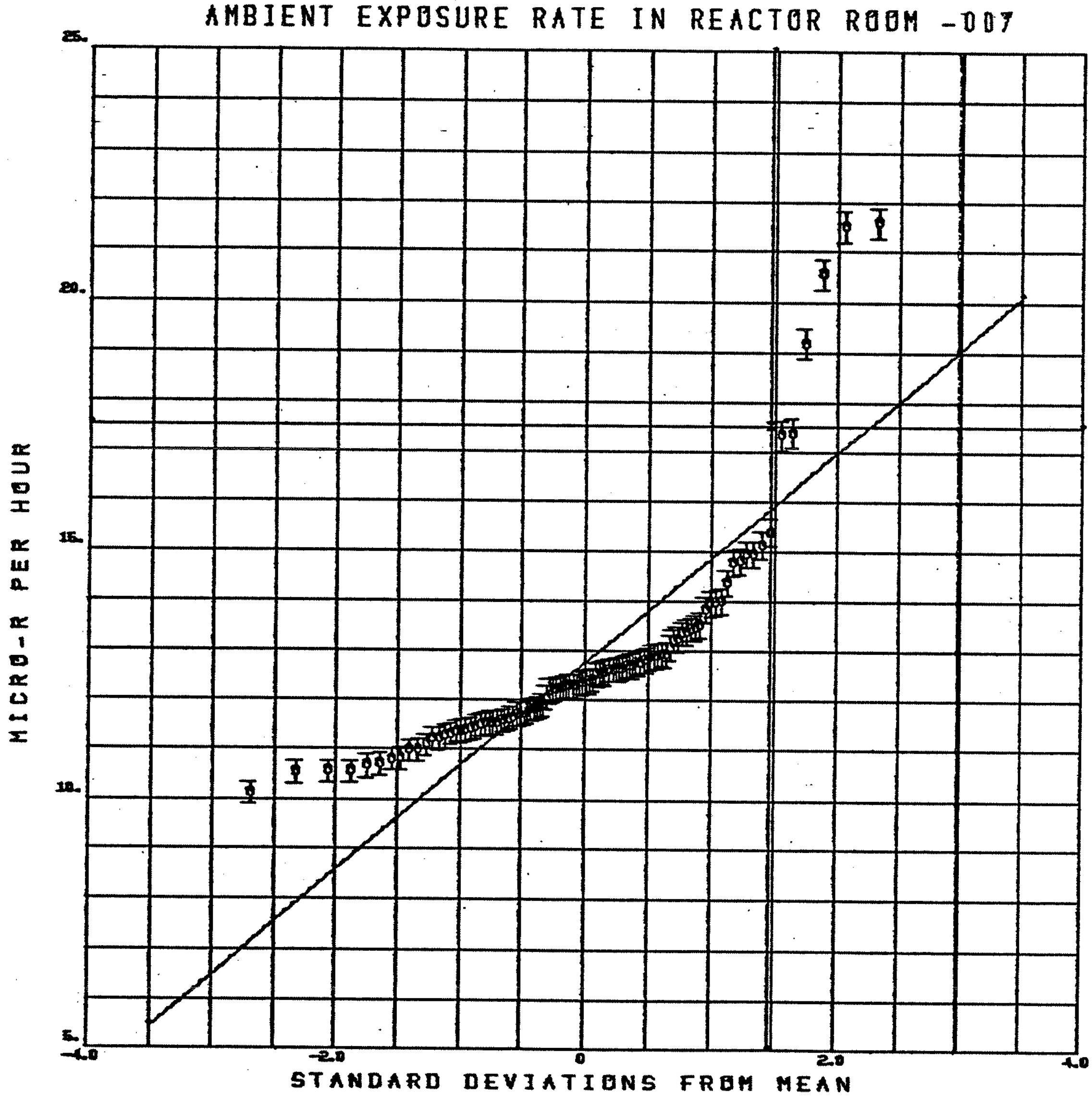


Figure 17. Results of Resurvey for Ambient Exposure Rate in Reactor Room (Instrument 596007)

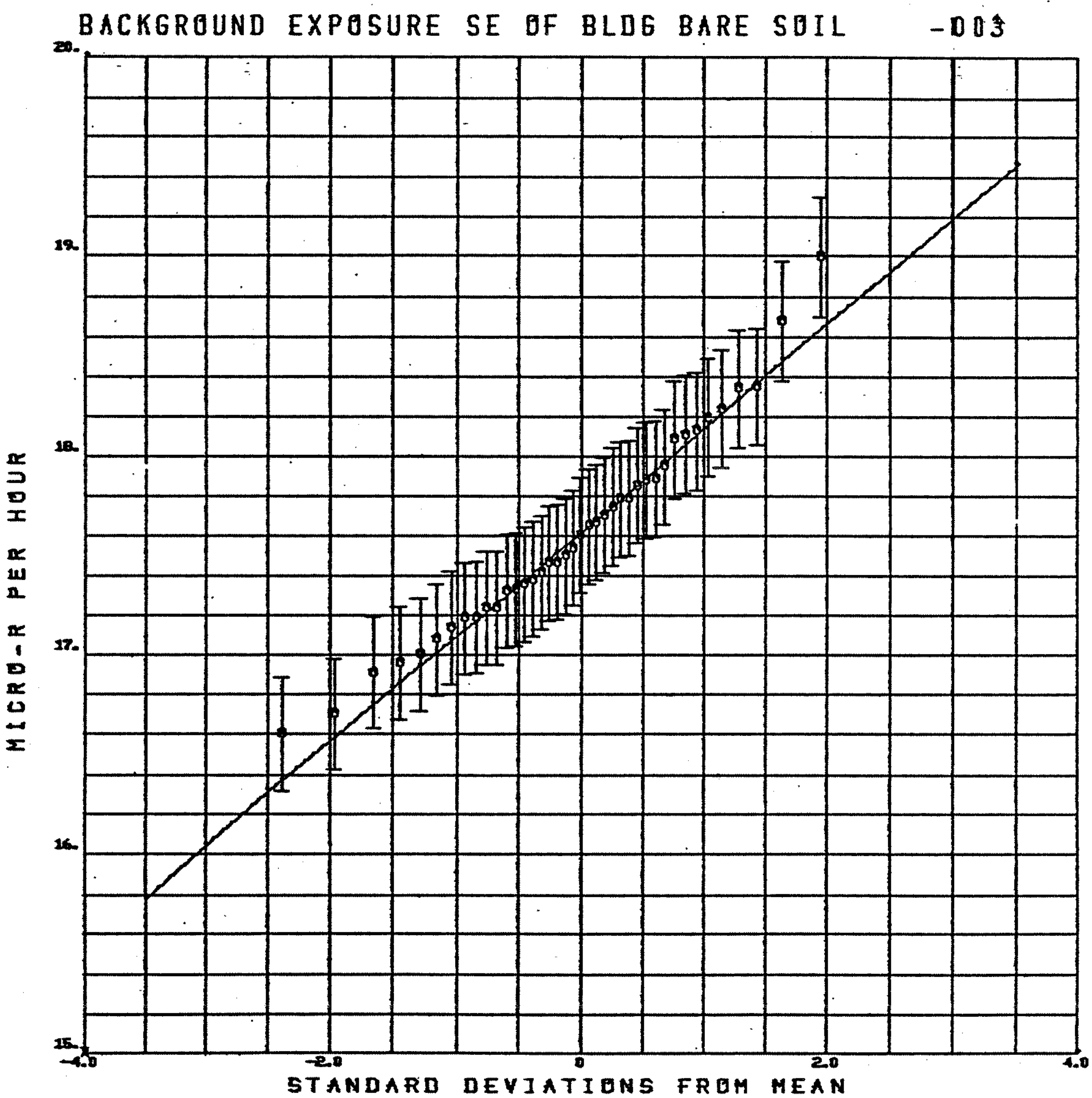


Figure 18. Ambient Exposure Rate Outside Reactor Room
(Instrument 596003)

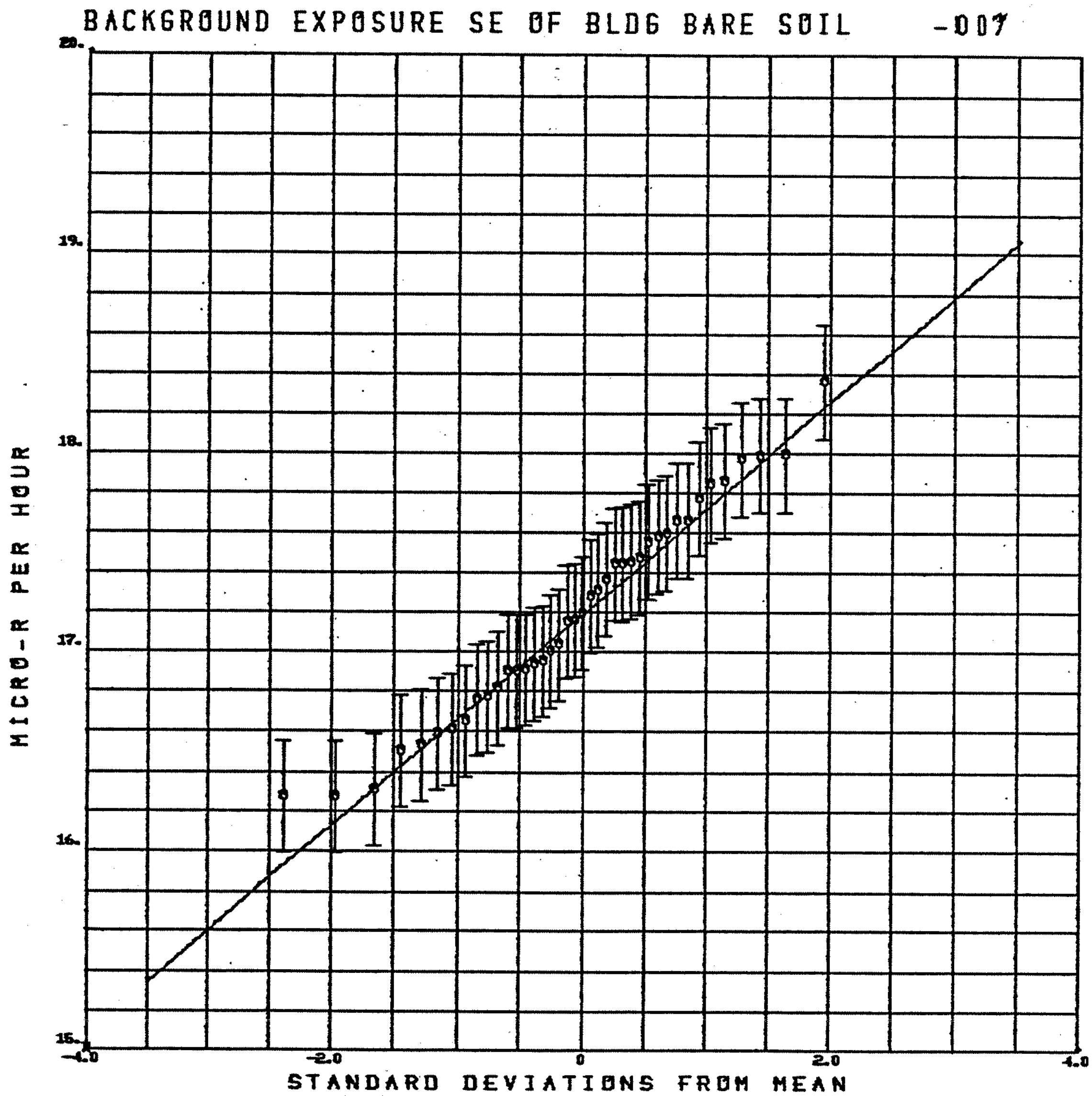


Figure 19. Ambient Exposure Rate Outside Reactor Room (Instrument 596007)

variability is due to "counting statistics" with the balance due to variations in the local exposure rate. This area is adjacent to a sandstone outcropping and the somewhat higher average rate measured here (17.4 μR/h) compared to the measurements at the HPIC locations (12.0 and 12.4 μR/h) may reflect both a geometric effect and an increased level of natural radioactive minerals.

Discussions with NRC Region V had indicated that the intention of the ambient radiation exposure rate criterion was to eliminate, to the extent reasonable, radioactivity induced during operation of the facility. Therefore, the appropriate "natural background" is that determined for unactivated concrete.

The average value of this "natural background" can be taken as the observed median of the measurements in the reactor room, the value of the measurements at 0 standard deviations from the mean. (The median and the mean of a Gaussian distribution are identical. Departure from a true Gaussian distribution, as in this case, by elevated readings, will not significantly affect the median, which may be taken as the mean of the unperturbed distribution.) For instrument number 596003 (Figure 16) the median is 12.55 μR/h. For instrument number 596007 (Figure 17), it is 12.36 μR/h. An average value of 12.5 μR/h is taken as representing unactivated concrete.

Figures 16 and 17 show the acceptance criterion line of 5 μR/h above natural background at 17.5 μR/h. This shows that four locations exceed the limit and two others are marginal. To reduce these to below the limit would require the removal and disposal of approximately 10-20 ft^3 of extremely low level activated concrete. This is not desirable either from consideration of cost or unnecessary use of disposal site space.

The alternative acceptance criterion established by the dismantling order, is that "no person will receive more than 10 mrem/year." This condition can be shown to be assured for several different uses of the building. The locations that show exposure rates distinguishably above background are shown in Figure 20. The dashed rectangle in the middle of the reactor room represents the support block for the reactor. This is probably the only area with

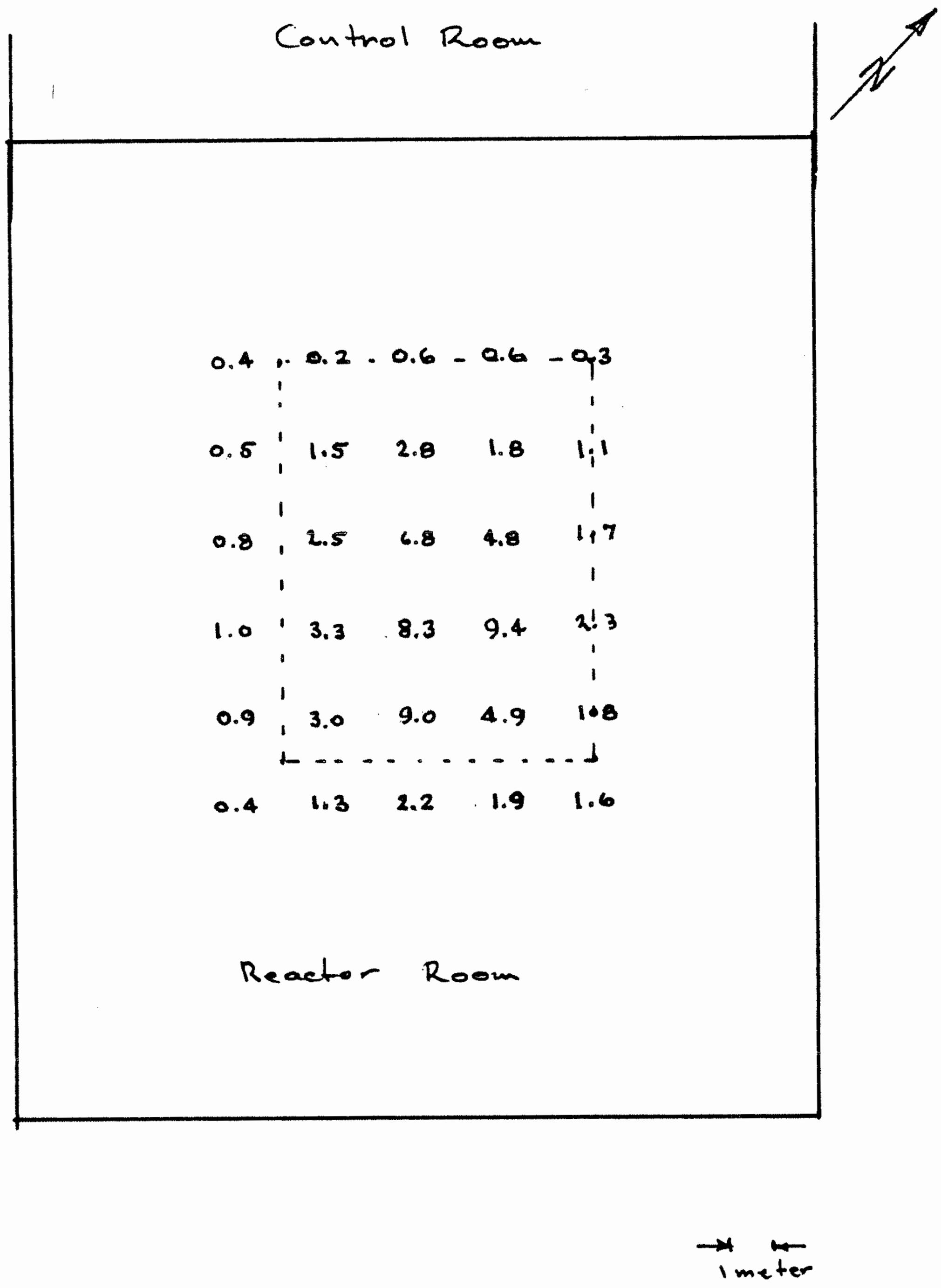


Figure 20. Net Exposure Rate (μR/h) at Location Exceeding Local Background in Reactor Room

true activation, and the adjacent locations probably show elevated readings due to activation within the support block, rather than in the floor outside it.

The results of Figures 16 and 17 show that the overall average exposure rate in the reactor room is 12.9 μR/h compared to the average "natural background" of 12.5 μR/h, for a net increase of 0.4 μR/h. Assuming occupancy of the room by a person managing a store-room, for example, for 2000 h per year, the exposure of that person would amount to approximately 0.8 mrem per year.

The surface of the support block was excavated to depths (relative to the adjacent floor) ranging from about 2 in. at the edges to 15-22 in. in the center. Therefore, this surface is not directly useable as a floor.

If a person worked for 2000 h a year at a desk located adjacent to the support block area, at the highest exposure rate location (2.2 μR/h above background), his dose would amount to 4.4 mrem per year.

In order to make the facility fully useable the depression must be filled with new concrete. This would shield the most active areas with a minimum of 15 in. of concrete. Assuming that all of the excess radiation is from Co-60 at the surface, the resulting exposure rate at 1 meter above the new level floor surface would be reduced by at least a factor of 55. This would reduce the maximum observed value (9.4 μR/h above background) to less than 0.2 μR/h above background. Assuming that an individual worked at that location 2000 h per year, his dose would be less than 0.4 mrem per year.

Thus, reasonable uses of the facility will not result in a person receiving more than 10 mrem per year.

5034Y/slw

VI. CONCLUSIONS

An appropriate survey has been conducted thoughout the area to be released. All measured values of residual radioactivity are below the acceptance limit. The results of this survey show statistically that no residual contamination remains in this area and demonstrate a negligible risk of there being any undetected contamination exceeding the acceptance limits. With the concurrence of the U.S. Nuclear Regulatory Commission, the facility license will be voluntarily terminated and the area will be released for unrestricted use.

VII. REFERENCES

1. California Radioactive Material License No. 0015-70

2. "Surface Radioactivity Guides for Materials, Equipment and Facilities to Be Released for Uncontrolled Use," draft American National Standards Institute N13.12, January 1985

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

APPENDIX A

INTERPRETATION OF GRAPHIC PRESENTATION

APPENDIX A

The purpose of statistical analysis is to convert a large amount of data into a manageable amount of understandable information. This process can involve a variety of mathematical techniques, the simplest being the determination of an average (or mean) value for a given set of data. This simple determination is improved upon by also calculating the standard deviation of the data about the mean, which gives an estimate of the variability of the data. In many cases, this variability represents variations both in the characteristics being measured (average alpha activity in one square meter, for example) and in measurement (due to random fluctuations in the detector efficiency and background radiation levels).

The significance of these quantities (mean and standard deviation) depends upon the distribution assumed for the data. Sometimes there is a theoretically known distribution for a particular measurement process, such as the binominal, or the Poisson distribution for counting radioactivity. These distributions are relatively well approximated by the Gaussian, or normal, distribution. In fact, the Gaussian distribution approximates the distribution of many different kinds of measurements and for simplicity is generally assumed to be the proper distribution. The Gaussian distribution is frequently seen in the form of a bell-shaped curve, with most values occurring near the mean value and fewer and fewer values existing at increasing distance from the mean, both greater than and less than the mean.

However, it is difficult to derive this bell-shaped curve from experimental data unless the data are specifically selected to demonstrate the curve, and deviations from the distribution are difficult to see. A better version is the so-called "cumulative probability function," which forms an S-shaped curve when plotted in the usual manner. This can be further improved by adjusting the abscissa (the X-values on an X-Y graph) so that the S-curve becomes a straight line. This is a standard statistical technique and is the basis for special graph paper used for probability analysis of data. The parameters of the Gaussian distribution (the mean and the standard deviation) are determined by the usual calculational methods:

$$\text{Mean} = \overline{X} = \frac{\Sigma X_i}{N}$$

$$\text{Standard deviation} = s = \left[\frac{\Sigma (X_i - \overline{X})^2}{N-1}\right]^{1/2}$$

where X_i represents the individual data values, and N is the number of points.

This method is the basis for the figures presented earlier in this report, where the measured values are plotted against the distance from the mean value, using the standard deviation of the assumed Gaussian distribution as the unit.

Where the data is not well represented by a Gaussian distribution (and this is true of most cases) the departure is readily apparent: the data points do not lie along a straight line representing the Gaussian distribution. In most cases, this departure takes a single typical form. Much of the data lies along the theoretical straight line with a few points at either extreme lying somewhat above it.

This form can usually be interpreted as showing a large number of uncontaminated locations where the variability is due to random fluctuations in the measurements themselves, with the balance being locations that harbor more or less residual contamination.

If the contaminated area is large, there will be many points departing from the curve. In these cases, the points will not fit the theoretical straight line. If most of the region in question is contaminated, the distribution will be dominated by the contaminated data points, in a line of points generally sloping from the lower left to the upper right, fitting more or less closely, a theoretical straight line.

To promote the quantitative use of sampling inspection in radiological surveys, several governmental agencies (the U.S. Nuclear Regulatory Commission and the State of California) have established a policy for the interpretation of survey data. All survey results must be below the appropriate limits and, in addition, the set of data, when interpreted statistically, must indicate that there is less than a 10% risk of accepting a facility in which 10% of the area is contaminated in excess of the limits. The mathematical methods used for this interpretation are explained in the next section.

In this report, this analysis has been extended to provide a sampling inspection test. This analysis uses a standard quality control technique called inspection by variables, in which the distribution of the measured values is used to predict the probability that other unmeasured values would exceed a specified limit. The standard test method requires calculating the mean ($\bar{X}$) and the standard deviation(s). Then, depending on the values chosen for certain parameters that reflect the performance of the test accepting bad lots, or rejecting good lots, the necessary number of samples is determined and a multiplier, k, is computed so that the inequality

$$\bar{X} + ks < U$$

where U is the acceptance limit, representing an acceptable lot. In the present application, "lot" is used to refer to a major segment of the survey effort.

The parameters used in this test are those recommended by the State of California, Radiologic Health Section, for the release of a facility for unrestricted use. These are the so-called "consumers risk" (or beta) and the "lot-tolerance percent defective" (LTPD). The values recommended for these are beta = 0.1 and LTPD = 10%. This means that, if a lot passes the acceptance test, there is one chance in ten (0.1) that 10% of the total number of locations in the facility would have residual contamination exceeding the limit.

The usual manner of applying this inspection test is to use tables giving the values of the sample size (N) and multiplier (k) for the selected values of beta and LTPD. In the present application, the number of measured values (N) in each lot was used to compute k, and this value was used to calculate $\bar{X} + ks$. The computation of k is somewhat complicated, but once programmed for the computer as part of a data analysis program, the complication is no obstacle to its use.

$$k = \frac{K_2 + \sqrt{K_2^2 - ab}}{a}$$

$$\text{with } a = 1 - \frac{K_\beta^2}{2(n - 1)}$$

$$\text{and } b = K_2^2 - \frac{K_\beta^2}{n}$$

The value of K_2 is that for the variable of a Gaussian distribution corresponding to the LTPD value, and the value of K_β is that for the Gaussian variable corresponding to beta.

An area that shows detectable contamination may still be acceptable for release according to the regulations if the levels of contamination are low enough. Acceptable limits have been established by the U.S. Nuclear Regulatory Commission, as shown in Appendix B, and several Agreement States. Clearly, all measured values must be less than the specified limits for an area to be acceptable. In the figures, (6-15) these limits are shown as horizontal lines marked in the graph by an "x." Review of the figures shows that, in most cases, all data points lie below the limit. The inspection test results in a vertical line on each graph, marked by an x where it crosses the horizontal limit line. A theoretical straight line is calculated for each distribution of data points; this shows as a line sloping more or less from the lower left to the upper right. The cleaner an area is, the closer to the horizontal this line will be. If this line passes below the x, the survey area is acceptable according to this set of well established statistical criteria. (Any locations within the area that were measured to be contaminated in excess of the limit, would still need to be decontaminated to a level less than the limit.)

APPENDIX B

REGULATORY GUIDE 1.86
TERMINATION OF OPERATING LICENSES FOR NUCLEAR REACTORS

APPENDIX B

U.S. ATOMIC ENERGY COMMISSION

REGULATORY GUIDE

DIRECTORATE OF REGULATORY STANDARDS

REGULATORY GUIDE 1.86

TERMINATION OF OPERATING LICENSES FOR NUCLEAR REACTORS

A. INTRODUCTION

Section 50.51, "Duration of license, renewal," of 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires that each license to operate a production and utilization facility be issued for a specified duration. Upon expiration of the specified period, the license may be either renewed or terminated by the Commission. Section 50.82, "Applications for termination of licenses," specifies the requirements that must be satisfied to terminate an operating license, including the requirement that the dismantlement of the facility and disposal of the component parts not be inimical to the common defense and security or to the health and safety of the public. This guide describes methods and procedures considered acceptable by the Regulatory staff for the termination of operating licenses for nuclear reactors. The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

B. DISCUSSION

When a licensee decides to terminate his nuclear reactor operating license, he may, as a first step in the process, request that his operating license be amended to restrict him to possess but not operate the facility. The advantage to the licensee of converting to such a possession-only license is reduced surveillance requirements in that periodic surveillance of equipment important to the safety of reactor operation is no longer required. Once this possession-only license is issued, reactor operation is not permitted. Other activities related to cessation of operations such as unloading fuel from the reactor and placing it in storage (either onsite of offsite) may be continued.

A licensee having a possession-only license must retain, with the Part 50 license, authorization for special nuclear material (10 CFR Part 70, "Special Nuclear Material"), byproduct material (10 CFR Part 30, "Rules of General Applicability to Licensing of Byproduct Material"), and source material (10 CFR Part 40, "Licensing of Source Material"), until the fuel, radioactive components, and sources are removed from the facility. Appropriate administrative controls and facility requirements are imposed by the Part 50 license and the technical specifications to assure that proper surveillance is performed and that the reactor facility is maintained in a safe condition and not operated.

A possession-only license permits various options and procedures for decommissioning, such as mothballing, entombment, or dismantling. The requirements imposed depend on the option selected.

Section 50.82 provides that the licensee may dismantle and dispose of the component parts of a nuclear reactor in accordance with existing regulations. For research reactors and critical facilities, this has usually meant the disassembly of a reactor and its shipment offsite, sometimes to another appropriately licensed organization for further use. The site from which a reactor has been removed must be decontaminated, as necessary, and inspected by the Commission to determine whether unrestricted access can be approved. In the case of nuclear power reactors, dismantling has usually been accomplished by shipping fuel offsite, making the reactor inoperable, and disposing of some of the radioactive components.

Radioactive components may be either shipped offsite for burial at an authorized burial ground or secured on the site. Those radioactive materials remaining on the site must be isolated from the public by physical barriers or other means to prevent public access to hazardous levels of radiation. Surveillance is necessary to assure the long term integrity of the barriers. The amount of surveillance required depends upon (1) the potential hazard to the health and safety of the public from radioactive material remaining on the site and (2) the integrity of the physical barriers. Before areas may be released for unrestricted use, they must have been decontaminated or the radioactivity must have decayed to less than prescribed limits (Table I).

The hazard associated with the retired facility is evaluated by considering the amount and type of remaining contamination, the degree of confinement of the remaining radioactive materials, the physical security provided by the confinement, the susceptibility to release of radiation as a result of natural phenomena, and the duration of required surveillance.

C. REGULATORY POSITION

1. APPLICATION FOR A LICENSE TO POSSESS BUT NOT OPERATE (POSSESSION-ONLY LICENSE)

A request to amend an operating license to a possession-only license should be made to the Director of Licensing, U.S. Atomic Energy Commission, Washington, D.C. 20545. The request should include the following information:

a. A description of the current status of the facility.

b. A description of measures that will be taken to prevent criticality or reactivity changes and to minimize releases of radioactivity from the facility.

c. Any proposed changes to the technical specifications that reflect the possession-only facility status and the necessary disassembly/retirement activities to be performed.

d. A safety analysis of both the activities to be accomplished and the proposed changes to the technical specifications.

e. An inventory of activated materials and their location in the facility.

2. ALTERNATIVES FOR REACTOR RETIREMENT

Four alternatives for retirement of nuclear reactor facilities are considered acceptable by the Regulatory staff. These are:

a. **Mothballing.** Mothballing of a nuclear reactor facility consists of putting the facility in a state of protective storage. In general, the facility may be left intact except that all fuel assemblies and the radioactive fluids and waste should be removed from the site. Adequate radiation monitoring, environmental surveillance, and appropriate security procedures should be established under a possession-only license to ensure that the health and safety of the public is not endangered.

b. **In-Place Entombment.** In-place entombment consists of sealing all the remaining highly radioactive or contaminated components (e.g., the pressure vessel and reactor internals) within a structure integral with the biological shield after having all fuel assemblies, radioactive fluids and wastes, and certain selected components shipped offsite. The structure should provide integrity over the period of time in which significant quantities (greater than Table I levels) of radioactivity remain with the material in the entombment. An appropriate and continuing surveillance program should be established under a possession-only license.

c. **Removal of Radioactive Components and Dismantling.** All fuel assemblies, radioactive fluids and waste, and other materials having activities above accepted unrestricted activity levels (Table I) should be removed from the site. The facility owner may then have unrestricted use of the site with no requirement for a license. If the facility owner so desires, the remainder of the reactor facility may be dismantled and all vestiges removed and disposed of.

d. **Conversion to a New Nuclear System or a Fossil Fuel System.** This alternative, which applies only to nuclear power plants, utilizes the existing turbine system with a new steam supply system. The original nuclear steam supply system should be separated from the electric generating system and disposed of in accordance with one of the previous three retirement alternatives.

3. SURVEILLANCE AND SECURITY FOR THE RETIREMENT ALTERNATIVES WHOSE FINAL STATUS REQUIRES A POSSESSION-ONLY LICENSE

A facility which has been licensed under a possession-only license may contain a significant amount of radioactivity in the form of activated and contaminated hardware and structural materials. Surveillance and commensurate security should be provided to assure that the public health and safety are not endangered.

a. Physical security to prevent inadvertent exposure of personnel should be provided by multiple locked barriers. The presence of these barriers should make it extremely difficult for an unauthorized person to gain access to areas where radiation or contamination levels exceed those specified in Regulatory Position C.4. To prevent inadvertent exposure, radiation areas above 5 mR/hr, such as near the activated primary system of a power plant, should be appropriately marked and should not be accessible except by cutting of welded closures or the disassembly and removal of substantial structures and/or shielding material. Means such as a remote-readout intrusion alarm system should be provided to indicate to designated personnel when a physical barrier is penetrated. Security personnel that provide access control to the facility may be used instead of the physical barriers and the intrusion alarm systems.

b. The physical barriers to unauthorized entrance into the facility, e.g., fences, buildings, welded doors, and access openings, should be inspected at least quarterly to assure that these barriers have not deteriorated and that locks and locking apparatus are intact.

c. A facility radiation survey should be performed at least quarterly to verify that no radioactive material is escaping or being transported through the containment barriers in the facility. Sampling should be done along the most probable path by which radioactive material such as that stored in the inner containment regions could be transported to the outer regions of the facility and ultimately to the environs.

d. An environmental radiation survey should be performed at least semiannually to verify that no signficant amounts of radiation have been released to the environment from the facility. Samples such as soil, vegetation, and water should be taken at locations for which statistical data has been established during reactor operations.

e. A site representative should be designated to be responsible for controlling authorized access into and movement within the facility.

f. Administrative procedures should be established for the notification and reporting of abnormal occurrences such as (1) the entrance of an unauthorized person or persons into the facility and (2) a significant change in the radiation or contamination levels in the facility or the offsite environment.

g. The following reports should be made:

(1) An annual report to the Director of Licensing, U.S. Atomic Energy Commission, Washington, D.C. 20545, describing the results of the environmental and facility radiation surveys, the status of the facility, and an evaluation of the performance of security and surveillance measures.

(2) An abnormal occurrence report to the Regulatory Operations Regional Office by telephone within 24 hours of discovery of an abnormal occurrence. The abnormal occurrence will also be reported in the annual report described in the preceding item.

h. Records or logs relative to the following items should be kept and retained until the license is terminated, after which they may be stored with other plant records:

(1) Environmental surveys,

(2) Facility radiation surveys,

(3) Inspections of the physical barriers, and

(4) Abnormal occurrences.

4. DECONTAMINATION FOR RELEASE FOR UNRESTRICTED USE

If it is desired to terminate a license and to eliminate any further surveillance requirements, the facility should be sufficiently decontaminated to prevent risk to the public health and safety. After the decontamination is satisfactorily accomplished and the site inspected by the Commission, the Commission may authorize the license to be terminated and the facility abandoned or released for unrestricted use. The licensee should perform the decontamination using the following guidelines:

a. The licensee should make a reasonable effort to eliminate residual contamination.

b. No covering should be applied to radioactive surfaces of equipment or structures by paint, plating, or other covering material until it is known that contamination levels (determined by a survey and documented) are below the limits specified in Table I. In addition, a reasonable effort should be made (and documented) to further minimize contamination prior to any such covering.

c. The radioactivity of the interior surfaces of pipes, drain lines, or ductwork should be determined by making measurements at all traps and other appropriate access points, provided contamination at these locations is likely to be representative of contamination on the interior of the pipes, drain lines, or ductwork. Surfaces of premises, equipment, or scrap which are likely to be contaminated but are of such size, construction, or location as to make the surface inaccessible for purposes of measurement should be assumed to be contaminated in excess of the permissable radiation limits.

d. Upon request, the Commission may authorize a licensee to relinquish possession or control of premises, equipment, or scrap having surfaces contaminated in excess of the limits specified. This may include, but is not limited to, special circumstances such as the transfer of premises to another licensed organization that will continue to work with radioactive materials. Requests for such authorization should provide:

(1) Detailed, specific information describing the premises, equipment, scrap, and radioactive contaminants and the nature, extent, and degree of residual surface contamination.

(2) A detailed health and safety analysis indicating that the residual amounts of materials on surface areas, together with other considerations such as the prospective use of the premises, equipment, or scrap, are unlikely to result in an unreasonable risk to the health and safety of the public.

e. Prior to release of the premises for unrestricted use, the licensee should make a comprehensive radiation survey establishing that contamination is within the limits specified in Table I. A survey report should be filed with the Director of Licensing, U.S. Atomic Energy Commission, Washington, D.C. 20545, with a copy to the Director of the Regulatory Operations Regional Office having jurisdiction. The report should be filed at least 30 days prior to the planned date of abandonment. The survey report should:

(1) Identify the premises;

(2) Show that reasonable effort has been made to reduce residual contamination to as low as practicable levels;

(3) Describe the scope of the survey and the general procedures followed; and

(4) State the finding of the survey in units specified in Table 1.

After review of the report, the Commission may inspect the facilities to confirm the survey prior to granting approval for abandonment.

5. REACTOR RETIREMENT PROCEDURES

As indicated in Regulatory Position C.2, several alternatives are acceptable for reactor facility retirement. If minor disassembly or "mothballing" is planned, this could be done by the existing operating and maintenance procedures under the license in effect. Any planned actions involving an unreviewed safety question or a change in the technical specifications should be reviewed and approved in accordance with the requirements of 10 CFR §50.59.

If major structural changes to radioactive components of the facility are planned, such as removal of the pressure vessel or major components of the primary system, a dismantlement plan including the information required by §50.82 should be submitted to the Commission. A dismantlement plan should be submitted for all the alternatives of Regulatory Position C.2 except mothballing. However, minor disassembly activities may still be performed in the absence of such a plan, provided they are permitted by existing operating and maintenance procedures. A dismantlement plan should include the following:

a. A description of the ultimate status of the facility

b. A description of the dismantling activities and the precautions to be taken.

c. A safety analysis of the dismantling activities including any effluents which may be released.

d. A safety analysis of the facility in its ultimate status.

Upon satisfactory review and approval of the dismantling plan, a dismantling order is issued by the Commission in accordance with §50.82. When dismantling is completed and the Commission has been notified by letter, the appropriate Regulatory Operations Regional Office inspects the facility and verifies completion in accordance with the dismantlement plan. If residual radiation levels do not exceed the values in Table I, the Commission may terminate the license. If these levels are exceeded, the licensee retains the possession-only license under which the dismantling activities have been conducted or, as an alternative, may make application to the State (if an Agreement State) for a byproduct materials license.

TABLE I

ACCEPTABLE SURFACE CONTAMINATION LEVELS

NUCLIDE[a]	AVERAGE[b c]	MAXIMUM[b d]	REMOVABLE[b e]
U-nat, U-235, U-238, and associated decay products	5,000 dpm α/100 cm^2	15,000 dpm α/100 cm^2	1,000 dpm α/100 cm^2
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100 dpm/100 cm^2	300 dpm/100 cm^2	20 dpm/100 cm^2
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1000 dpm/100 cm^2	3000 dpm/100 cm^2	200 dpm/100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above.	5000 dpm β-γ/100 cm^2	15,000 dpm β-γ/100 cm^2	1000 dpm β-γ/100 cm^2

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

[b]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 observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[c]Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

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

[e]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 assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

APPENDIX C

U.S. NRC DISMANTLING ORDER

APPENDIX C

UNITED STATES
NUCLEAR REGULATORY COMMIS
WASHINGTON, D. C. 20555

No.: N001SRR140087
Page: 49

FEB 22 1983

MAR 1 1983
M.E. REMLEY

ket No. 50-375

RECEIVED
CORRESPONDENCE
FEB 28 1983
ENERGY SYSTEMS GROUP

Dr. M. E. Remley, Director
Health, Safety and Radiation Services
Rockwell International Corporation
8900 DeSoto Avenue
Canoga Park, California 91304

Dear Dr. Remley:

The Commission has issued the enclosed Order that authorizes you to dismantle the Rockwell International L-85 Nuclear Examination Reactor in accordance with your application dated March 10, 1980, as amended by letter dated December 14, 1982. The dismantling plan replaces the Technical Specifications in their entirety. Since the fuel and radioactive sources have been shipped offsite to authorized receivers, we will consider termination of License No. R-118 after the reactor has been dismantled and residual radioactivity has been reduced to levels specified in the enclosed order authorizing dismantling.

The related Safety Evaluation, Environmental Impact Appraisal, and Negative Declaration are also enclosed.

A copy of the Order and Negative Declaration are being filed with the Office of the Federal Register for publication.

Sincerely,

Darrell G. Eisenhut, Director
Division of Licensing

Enclosures:
1. Order Authorizing Dismantling
2. Safety Evaluation
3. Environmental Impact Appraisal
4. Negative Declaration

cc w/enclosures:
See next page

1623ES6

Energy Systems Group
D/786 Correspondence

ACTION

REMARKS

NDERSON, R. V.	LA24		
SHWORTH	LB22		
ABBE	NB14		
AUMEISTER	LB11		
EELEY	LB01		
ELL, J.	LA26		
RUNINGS	LB02		
ARDENAS	NB01		
ONNERS	NB02		
ETTERMAN	LA17		
POL	LA21		
AST	LA27		
LLIS, L.	LA34		
MPEY, D.	KB44		
EILER	LA30	X	X
OWLER	KB04		
ARDNER, R. R.	LA06		
YLFE	LA39		
ALLINAN	LA24		
ARTZLER	KB05		
AYS, E.	MA03		
OLBROOK	LA07		
OVER	LA34		
TTER	LB30		
OHNSON, R. A.	LA34		
ULIAN	KB03		
ATZ	LA19		
GER	NB02		
	LA35		
	MA03		
AM	MA03		
ANCET	LB02		
UCAS	LA34		
ASON, D.	NB02		
COURT	LA02		
DONALD	LB11		
EYERS, G. W.	LA10		
OORE, K. A.	MA03		
OSS	NB02		
AGAMATSU	KB08		
LDENKAMP	LB17		
ARKER, T.	NB02		
EICHE, L.	055-AA89		
EINECKER	LA30		
EMLEY	NB13	X	X
OBERTS, W. J.	LA21	X	X
ANDERS	NB01		
CHIRM	LA23		
CHMIDT, D.	LA20		
CHMITT, A.	LA17		
LVERMAN	LB16		
MITH, J. V.	LA38		
PENCER, C.	MA03		
PRINGER	LA18		
WANSON	LB09	X	X
RILLING	LB22		
ALTER, J. H.	LB07		
HEELER	LA33		
IESENECK	T038		
ILLIAMS, R. O.	LA04		
2 CORRES	001		
UTTLE		X	X
NB13			

FORM N 77-H-1 REV. 11-82

SUPPLIED

cc w/enclosure(s):

Sacramento County Board of
Supervisors
827 7th Street, Room 424
Sacramento, California 95814

Office of Intergovernmental
Management - State of California
1400 10th Street, Room 108
Sacramento, California 95814

California Department of Health
ATTN: Chief, Environmental
Radiation Control Unit
Radiological Health Section
714 P Street, Room 498
Sacramento, California 95814

ROCKWELL INTERNATIONAL

DOCKET NO. 50-375

ORDER AUTHORIZING DISMANTLING OF FACILITY AND DISPOSITION OF COMPONENT PARTS

By application dated March 10, 1980, as amended by letter dated December 14, 1982, Rockwell International (the licensee) requested authorization to dismantle its L-85 Nuclear Examination Reactor (the facility), located at the licensee's site at Santa Susana Field Laboratory, Ventura County, California, and to dispose of the component parts, in accordance with the plan submitted as part of the application. A "Notice of Proposed Issuance of Orders Authorizing Dismantling of Facility, Disposition of Component Parts, and Termination of Facility License" was published in the Federal Register on April 30, 1980 (45 FR 30759). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Nuclear Regulatory Commission (the Commission) has reviewed the application in accordance with the provisions of the Commission's rules and regulations and has found that the dismantling and disposal of component parts under the licensee's dismantling plan will be in accordance with the regulations in 10 CFR Chapter I, and will not be inimical to the common defense and security or to the health and safety of the public. The basis for the findings is set forth in the concurrently issued Safety Evaluation by the Office of Nuclear Reactor Regulation.

The Commission has prepared an environmental impact appraisal for this action. Based on that appraisal, the Commission has determined that this action will not result in any significant environmental impact and that an environmental impact statement need not be prepared.

Accordingly, Rockwell International is hereby authorized to dismantle the facility covered by Facility License No. R-118, and dispose of the component parts in accordance with its corrected dismantling plan dated December 14, 1982 and the Commission's rules and regulations.

After completion of the dismantling and decontamination of the reactor, the submission of a report on the radiation survey to confirm that radiation levels in the facility area meet the values defined in the dismantling plan, and inspection by representatives of the Commission, consideration will be given to whether a further order should be issued terminating Facility License No. R-118.

For further details with respect to this action see (1) the application for authorization to dismantle facility and dispose of component parts dated March 10, 1980, as revised by letter dated December 14, 1982, (2) the Commission's related Safety Evaluation, (3) the Commission's Environmental Impact Appraisal, and (4) the Commission's Negative Declaration dated FEB 22 1983 (which is also being published in the Federal Register). All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 22nd day of February 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

Darrell G. Eisenhut, Director
Division of Licensing

SAFETY EVALUATION BY THE
OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING ORDER AUTHORIZING DISMANTLING
OF FACILITY AND DISPOSITION OF COMPONENT PARTS
ROCKWELL INTERNATIONAL CORPORATION
L-85 REACTOR
DOCKET NO. 50-375

Introduction

By application dated March 10, 1980, as amended by letter dated December 14, 1982, Rockwell International Corporation (the licensee) requested authorization to dismantle its L-85 Nuclear Examination Reactor and dispose of its component parts in accordance with its dismantling plan.

The L-85 reactor is a homogeneous, solution-type research reactor licensed to operate at a maximum power level of 3 kilowatts, thermal. The fuel is a solution of water and fully enriched uranyl sulfate. The fuel solution is contained in a 1-foot diameter, spherical, stainless steel vessel that is surrounded by a graphite reflector. Two safety rods and two control rods are inserted.

On July 29, 1982, the uranyl sulfate solution was removed from the reactor core and on September 28, 1982, it was shipped to the Idaho Nuclear Engineering Laboratory for processing. Coolant water with radioactivity concentration of about .01 MPC has been drained, so the only radioactivity remaining is that produced by activation during the years of operation. Subsequent to the removal and offsite shipment of the fuel, Rockwell International (RI) submitted an amended application for dismantling by letter dated December 14, 1982.

Evaluation

The RI application indicates that only about 1.3 Ci of total residual activity remains, mostly in core vessel, steel reflector tank and the control rods.

The dismantling plan indicates dismantling and removal of all components and activated structural materials will be conducted in a manner such that radioactivity readings will be consistent with Table 1 of Regulatory Guide 1.86.

Table 1 in the RI application indicates that decontamination activities will reduce contamination to a level of 5 μR/hr above the background... "or the occupancy of the facility must be limited so that no person will receive more than 10 mRem/yr."

ENVIRONMENTAL IMPACT APPRAISAL

SUPPORTING ORDER AUTHORIZING DISMANTLING OF

FACILITY AND DISPOSITION OF COMPONENT PARTS

ROCKWELL INTERNATIONAL CORPORATION

DOCKET NO. 50-375

Introduction

By application dated March 10, 1980, as revised December 14, 1982, Rockwell Internation Corporation (RI) applied for authorization to dismantle its L-85 Nuclear Examination Reactor, dispose of its component parts, and terminate the facility license. This evaluation deals with those features and characteristics of reactor dismantling and disposition of component parts which may affect the environment.

Discussion

The L-85 Nuclear Examination Reactor is a small research reactor that operated at a maximum of 3 kW thermal. The concurrently issued Safety Evaluation discusses the construction of the reactor and the safety aspects of dismantling. The reactor was originally located at corporate facilities in Downey, California until 1956, where it operated at 0.3 watts. It was moved to its present location in the RI Santa Susana Field Laboratory where it was modified to increase its power level to the current level of 3 kW.

Environmental Considerations

Radioactive waste material produced during dismantling, such as paper towels, gloves and wipes, will be disposed of at an authorized radioactive waste burial site. The reactor components, other than the reactor core, will be decontaminated, stored, or disposed as scrap. Those reactor components that remain radioactively contaminated or activated will be shipped to an authorized burial site. The fuel has already been drained and shipped to a Department of Energy (DOE) facility for processing. RI proposes to remove all byproduct materials radioactive wastes, and radioactive components from the reactor facility. Radioactivity will be reduced to 5 μR/hr above background. RI has indicated that they do not now have plans to use the space occupied by the reactor for other activities following dismantling, at least not in the near future (personal communications with M. A. Remley by H. Bernard).

Therefore, dismantling will reduce radioactivity to virtually indistinguishable background and will cause no significant environmental impact.

Alternatives to Dismantling of Reactor and Disposal of Components

The reactor has not been operated for about 2 years and there are no plans or need for future operation. The other reasonable alternative to dismantling is to leave the reactor where it is, secure the facility and continue monitoring. However, as all of the short-lived radioactivity has already decayed, the measures necessary for dismantling in the future would be similar to those considered for dismantling and disposal at this time.

Long Term Effects of Dismantling and Disposal of Components

As the reactor fuel has already been shipped to a DOE facility for processing and any radioactive reactor components or structures will be disposed of at an authorized burial site, and decontamination will be accomplished to 5 μR/hr above background, there will no long term effects due to dismantling and disposal of this facility.

Conclusion

We conclude that there will be no significant environmental impact associated with the dismantling of the L-85 Nuclear Examination reactor facility and disposal of its component parts, and that no environmental impact statement is required to be written for dismantling the facility and disposal of its component parts.

Dated: February 22, 1983

NEGATIVE DECLARATION

FOR THE

ROCKWELL INTERNATIONAL CORPORATION

L-85 NUCLEAR EXAMINATION REACTOR

DOCKET NO. 50-375

The U. S. Nuclear Regulatory Commission (the Commission) has considered the Order authorizing dismantling of facility and disposition of component parts for the Rockwell International Corporation (the licensee) L-85 Nuclear Examination Reactor operated under Facility License No. R-118. The Order authorizes the licensee to disassemble the reactor which had operated at power levels up to 3 kW (thermal), and to dispose of the component parts.

The Commission's Office of Nuclear Reactor Regulation has prepared an environmental impact appraisal for this training reactor. On the basis of this appraisal, the Commission has concluded that an environmental impact statement for this particular action is not warranted because there will be no significant environmental impact attributable to the proposed action. The environmental impact appraisal is available for public inspection at the Commission's Public Document Room at 1717 H Street, N.W., Washington, D.C.

Dated at Bethesda, Maryland, this 22nd day of February 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

Darrell G. Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing