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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	<p>Added data on complete resurvey of reactor room for ambient radiation and interpretation of dismantling order release criterion. (Page 33.1 through 33.8)</p> <p>Page 3 - added Figures:</p> <p>16. Results of Resurvey for Ambient Exposure Rate in Reactor Room (Instrument 596003)</p> <p>17. Results of Resurvey for Ambient Exposure Rate in Reactor Room (Instrument 596007)</p> <p>18. Ambient Exposure Rate Outside Reactor Room (Instrument 596003)</p> <p>19. Ambient Exposure Rate Outside Reactor Room (Instrument 596007)</p> <p>20. Net exposure rate (μ R/h) at Locations Exceeding Local Background in Reactor Room</p>	<p><i>F. E. Begley 2/21/86</i> F. E. Begley</p> <p><i>M. E. Remley 4/14/86</i> M. E. Remley</p> <p><i>R. J. Tuttle 2/21/86</i> R. J. Tuttle</p> <p><i>See Date 3-6-86</i></p>
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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 authorized¹ 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.

Table with multiple columns containing alphanumeric codes and descriptions of site features, including building names, room numbers, and facility types.

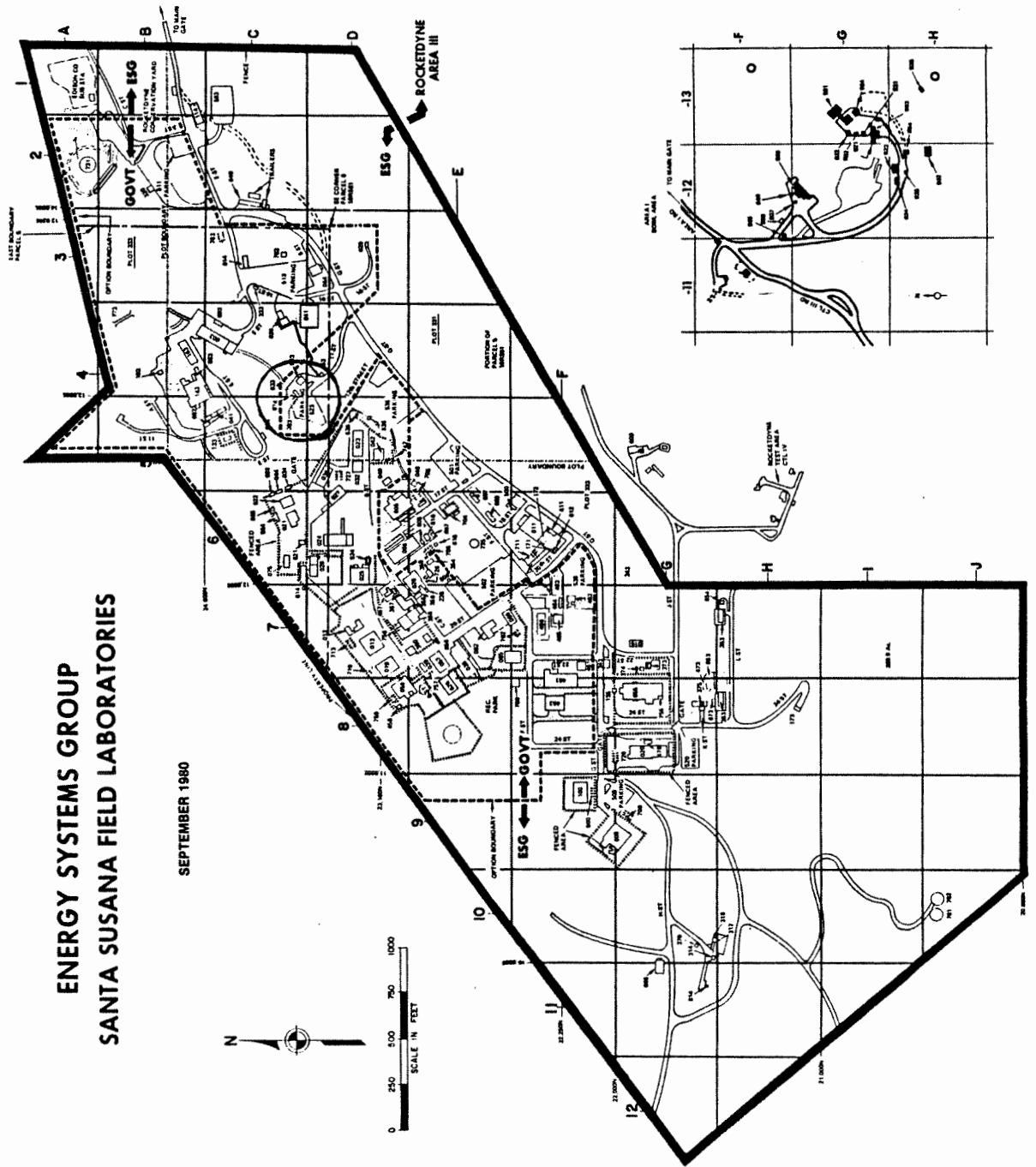


Figure 1. Santa Susana Site Map

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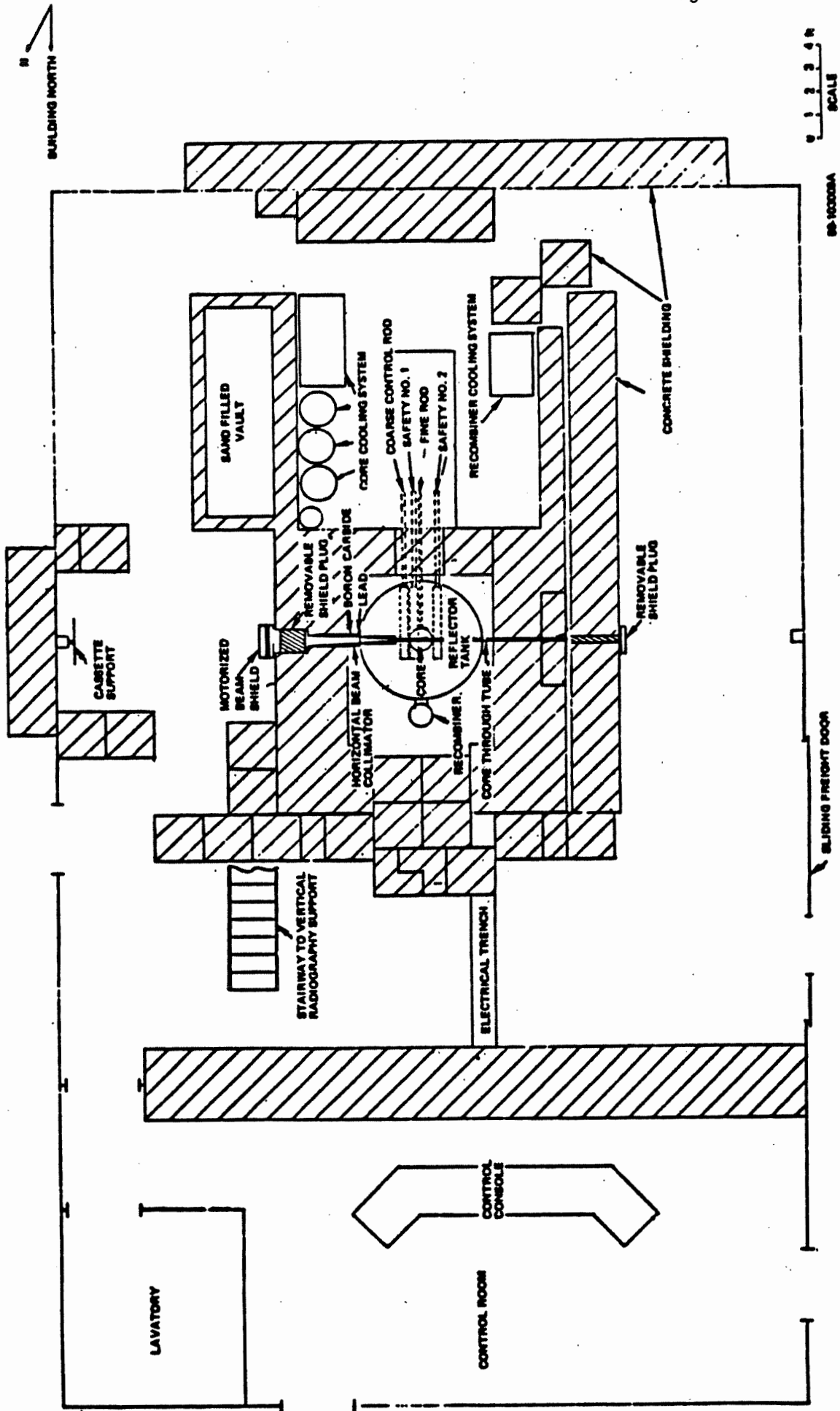


Figure 2. Floor Plan of the L-85 Reactor Building
(Before Decommissioning)

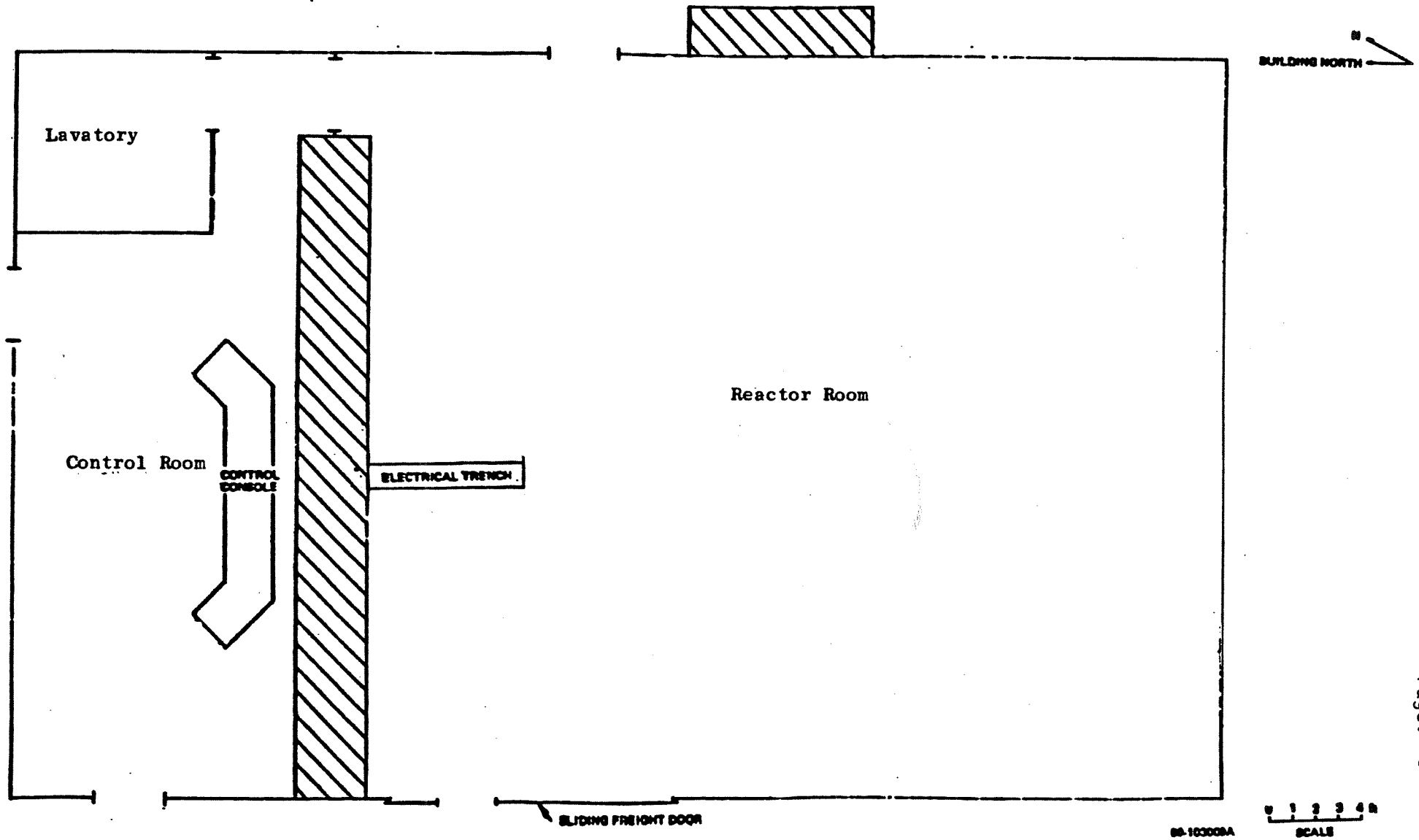


Figure 3. Floor Plan of the L-85 Reactor Building
(After Decommissioning)

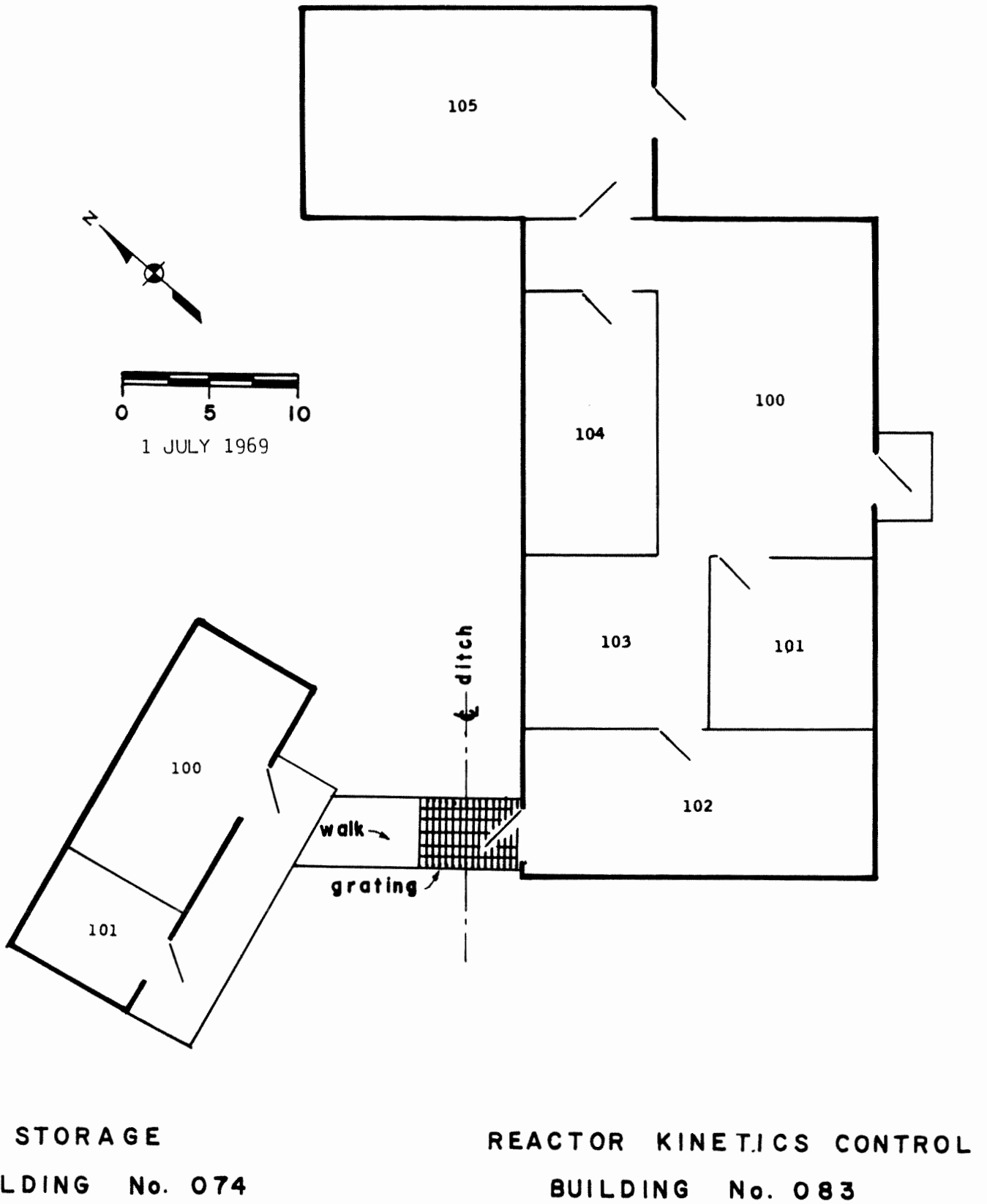
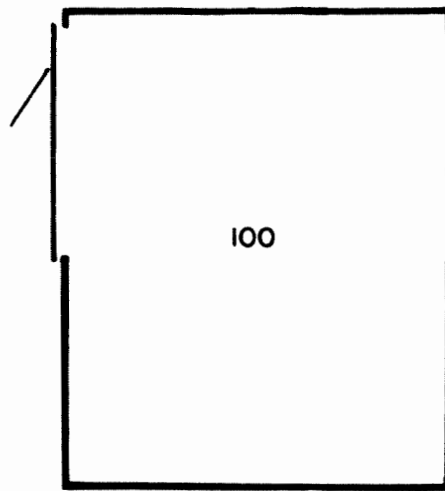


Figure 4.



1 JULY 1969

**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) version² of American National Standards Institute/Health Physics Society Standard ANSI N13.12, and in the DECON-1 document issued by the State of California³ 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.	5,000 α	15,000 α	1,000 α
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

^aWhere contamination by both alpha- and beta-gamma-emitting nuclides exist, the limits established for alpha- and beta-gamma-emitting nuclides should apply independently.

^bAs 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.

^cMeasurements 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.

^dThe maximum contamination level applies to an area of not more than 100 cm².

^eThe amount of removable radioactive material per 100 cm² 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-m² area was surveyed. Each area was outlined by felt marker or paint, with its coordinates marked within or beside the 1-m² area. The location of this 1-m² 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 sampling plan for selection of items to be tested, in this case, locations to be measured for radioactivity, and the method of analysis. The sampling plan 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:

