Prepared by
Oak Ridge Associated Universities

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Division of Radiation Safety and Safeguards, Emergency Preparedness and Radiological Protection Branch

CONFIRMATORY RADIOLOGICAL SURVEY OF THE L-85 REACTOR FACILITY ROCKETDYNE DIVISION ROCKWELL INTERNATIONAL CORPORATION SANTA SUSANA, CALIFORNIA

G. L. MURPHY

Radiological Site Assessment Program
Manpower Education, Research, and Training Division

FINAL REPORT
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SANTA SUSANA, CALIFORNIA

INTRODUCTION

Rockwell International Corporation operated a research reactor at the Santa Susana Field Laboratories from late 1956 until February 29, 1980. The facility operated under an Atomic Energy Commission (AEC) contract from 1956 through 1972, followed by Nuclear Regulatory Commission (NRC) license #R-118, from 1972 to present. The reactor was operated to provide a neutron source, for subcritical experiments, neutron radiography, and training functions.

In March 1980, Rockwell International applied for an NRC order authorizing dismantling of the facility, disposal of the component parts, and termination of facility license #R-118. The fuel solution was transferred to the Idaho Nuclear Engineering Laboratory in September 1982, and the NRC issued a decommissioning order on February 22, 1983. In March 1986, Rockwell submitted a radiation survey report (RI 86) of the decommissioned facility, indicating that the facility satisfied the NRC guidelines (RG 186) for release from licensing restrictions. At the request of the Nuclear Regulatory Commission's Region V Office, the Radiological Site Assessment Program (RSAP) of Oak Ridge Associated Universities (ORAU) conducted a confirmatory survey of the L-85 reactor facility (Building T-093). This report presents the procedures and results of that survey.

SITE DESCRIPTION

The L-85 reactor was operated by the Rocketdyne Division of Rockwell International Corporation, and was located at the Santa Susana Field Laboratories, slightly northwest of Canoga Park, California (Figure 1). The L-85 reactor was housed in Building T-093, which was constructed from concrete block and sheet metal on a steel girder frame. Building T-093 (Figure 2) consists of the reactor bay area and a small control room, which also contains a lavatory. The facility dimensions are 16.4 m x 9.7 m, and the reactor bay has an open ceiling approximately 10 meters high with an overhead crane (Figure 3).

DOCUMENT REVIEW

Prior to performing the site survey, ORAU performed a review of the licensee's documentation supporting the decommissioning project. ORAU reviewed the dismantling order (NRC 83), decommissioning plan (RI 85), the final survey (RI 86), licensee/NRC correspondence, and Rockwell internal documents supporting the decommissioning activity.

Rockwell developed and implemented a thorough, comprehensive decommissioning plan. Many items of concern addressed by the NRC, during the dismantling, were resolved by Rockwell. ORAU's review indicates the procedures, instrumentation, and data support the final survey report.

SURVEY PROCEDURES

On September 30 through October 2, 1986, ORAU conducted a confirmatory radiological survey of the L-85 reactor facility. The purpose of the survey was to verify the adequacy and accuracy of the licensee's final survey, and to confirm the radiological condition of the facility relative to the decommissioning guidelines.

A. Indoor Areas

Gridding

A 1 m x 1 m grid pattern was established on the floor, using the northwest corner of the building as the baseline coordinate (0,0). Grid block numbers were assigned sequentially from west to east, and each row was numbered starting from the west. A similar floor grid was established in the control room; the lavatory was not gridded due to its small size.

A 1 m x 1 m grid, two meters high, was established on lower wall surfaces. Grid block #1 was always established in the lower left hand corner of the wall, and the grid blocks were numbered sequentially from bottom to top.

Measurements on the upper walls and ceilings were referenced to building features, or the floor grid.

Surface Measurements

RSAP personnel scanned the floors and lower walls with an alpha and beta-gamma floor monitor and NaI(Tl) gamma scintillation detectors, respectively. Locations inaccessible to the floor monitors were scanned with hand held alpha scintillation and beta-gamma "pancake" detectors. Upper wall and overhead surface scanning (Figure 4) on ledges, beams, piping, fixtures, equipment, and ductwork was conducted using hand held alpha and beta-gamma detectors. Elevated areas were marked for additional measurements.

Measurements of total alpha and beta-gamma contamination levels (Figure 5) on random floor and lower wall grid blocks were performed at the center and four points, midway between the center and grid block corners. Smears (Figure 6) for removable alpha and beta contamination were performed at the location in each grid block where the highest total measurement was obtained. Total and removable contamination measurements were also performed on upper walls, ceilings, ledges, and other horizontal and vertical surfaces.

Exposure Rate Measurements

Gamma exposure rates at 1 meter above the floor were measured at random locations and areas of elevated gamma levels, identified by the surface scans, using a pressurized ionization chamber.

Gamma Spectroscopy Measurements

An in-situ gamma spectrum was collected at each location where gamma exposure rate measurements were made (Figure 7). The spectra was collectedusing a high purity germanium detector positioned one-half meter from the surface. The gamma spectra were used to identify the residual radionuclide contaminants.

Miscellaneous Sampling

Random samples of paint were collected in the reactor bay area and the control room. Samples of residue on horizontal surfaces, concrete dust and chips, standing water (located in electrical chase), and drain residues were collected and returned to ORAU for analyses.

B. Outside Areas

Surface Measurements

ORAU personnel performed walkover surface scans, using gamma scintillation detectors, to a distance of 10 meters from the building in all directions. Scans were extended to cover the access roads and parking lots to a distance of 80 meters from the building.

Soil Sampling

Surface soil samples were collected in east, south, and west areas adjacent to Building T-093. The area directly north of the building was covered by a small building and large rock out croppings.

C. Baseline Measurements

Indoor Areas

Building T-453 located approximately 50 meters south of Building T-093 was used to establish the baseline for gamma exposure rate measurements. Building T-453 has a similar construction history as T-093, and is located in a non-restricted area which has no history of non-sealed source radioactive materials use. Exposure rate measurements and gamma spectra were collected using the pressurized ionization chamber and the gamma spectroscopy system.

Outdoor Areas

Gamma exposure rate measurements and surface soil samples were collected at four off-site locations surrounding the Santa Susana Field Laboratory.

Sample Analysis and Interpretation of Data

Smears were counted to determine gross alpha and beta activity. Soil and miscellaneous residue samples were analyzed by gamma spectroscopy for cobalt-60, cesium-137, europium-152/154, uranium-238/235, and other identifiable photopeaks. Major analytical equipment used in support of this survey is listed in Appendix A, and Appendix B describes the measurement and analytical procedures.

Results were compared with guidelines established by the Nuclear Regulatory Commission for release of facilities for unrestricted use (Appendix C). An additional guideline, was established to limit the exposure rate to 5μR/h above ambient background at one meter from the surface, or the licensee may present a worst-case analysis which establishes the dose-rate to a theoretical individual occupying the area, to less than 10 mrem/year.

RESULTS

Indoor Areas

Baseline Exposure Rate

The baseline exposure rate of 12 μR/h, was established in Building T-453.

Surface Scans

Alpha and beta-gamma scans identified two areas of elevated alpha activity in grid blocks 16 and 81 on the floor of the reactor bay (Figure 8). These areas did not exceed the release guidelines, but Rockwell elected to perform further remedial action to ensure that residual contamination was as low as reasonably achievable.

Surface Contamination Levels

Tables 1 through 8 summarizes the results of surface contamination measurements performed on 56 random floor and lower wall grid blocks of Building T-093 (Figures 9 - 12). The total activity presented in the tables is a direct measurement which contains non-removable activity, as well as, removable activity, if present. The total alpha activity ranged from the minimum detectable activity of 11 dpm/100 cm^2 to 460 dpm/100 cm^2, in grid block 81. The highest average grid block results (160 dpm/100 cm^2) occurred in block 16. The removable alpha activity ranged from an MDA of 3 dpm/100 cm^2 to 14 dpm/100 cm^2. The total beta activity ranged from a MDA of 400 dpm/100 cm^2 to 4900 dpm/100 cm^2, in grid block 39. The highest average grid block results (1800 dpm/100 cm^2) occured in grid blocks 39 and 47. The removable beta activity ranged from an MDA of 7 dpm/100 cm^2 to 12 dpm/100 cm^2.

Tables 9 through 12 summarize the results of 30 single point surface contamination measurements performed on upper walls and ceilings of Building T-093. The total alpha activity ranged from an MDA of 18 dpm/100 cm^2 to 160 dpm/100 cm^2, and the removable activity ranged from an MDA of 3 dpm/100 cm^2 to 14 dpm/100 cm^2. The total beta activity ranged from an MDA of 400 dpm/100 cm^2 to 1400 dpm/100 cm^2, and the removable beta activity ranged from an MDA of 7 dpm/100 cm^2 to 20 dpm/100 cm^2.

Radionuclide Activity in Miscellaneous Media

Table 13 summarizes the radionuclide concentrations in 12 samples of miscellaneous media. Dust residues were collected at seven locations in the reactor bay area. Radionuclides in trace amounts were identified in several of these samples. One sample collected along the east wall ledge indicated the presence of Co-60, Cs-137, Eu-152, Eu-154, and U-238. Two paint samples were collected from the reactor bay and one from the control room, one reactor bay sample contained Cs-137 and U-238 in trace amounts. A smear from the sink trap of the control room lavatory had no detectable radionuclides. The minimum detectable activities reported are quite high due to the small sample size and relatively small area smeared. Concrete chips were collected from the scabbled area of the Reactor Bay. This sample contained 11 pCi/g of Eu-152, the highest europium content of any miscellaneous sample. Radiostrontium analysis on this sample indicated the Sr-89 was less than 0.3 pCi/g and Sr-90 was less than 0.4 pCi/g. A very small sample of standing water (less than 2 ml) was collected from the electrical chase in the reactor bay floor. No identifiable photopeaks were observed. The sample was analyzed for gross alpha and beta activity. The results were 1.4 ± 0.8 pCi/l alpha and 5.6 ± 1.2 pCi/l beta.

Exposure Rate Measurements

Table 14 summarizes the exposure rate measurements, taken at six random locations and five additional locations identified by surface scans (Figure 13). The exposure rate levels in Building T-093 ranged from 12 to 18 μR/h, compared to a baseline level of 12 μR/h measured in Building T-453. The gamma spectra collected in Building T-453 identified only the presence of natural radionuclides normally present in building materials. A review of the gamma spectra collected at each exposure rate measurement location indicated the presence of Co-60 and Eu-152 in the elevated exposure rate areas. Although Co-60 and Eu-152 photopeaks are present in the random locations, their relative peak areas are considerably smaller and are attributed to the close proximity of the hot spots from the scabbled floor.

Outside Areas

Surface Measurements

No locations of elevated direct radiation levels were identified by gamma scans of the areas within 10 meters of the buildings, parking lots and access roads, or drainage ditches. The exposure rates measured at one meter from the ground at the soil sampling locations, ranged from 16 to 18 μR/h, compared to 10 to 13 μR/h at offsite locations. Although the onsite locations are slightly elevated, this is probably related to variations in natural radionuclide concentrations in different types of soil.

Soil Samples

Six random soil samples were collected around Building T-093, and the results are summarized in Table 15 (Figure 14). The radionuclide concentrations are typical of concentrations present in four baseline samples collected offsite (Figure 15). Only photopeaks associated with naturally occurring radionuclides were detected in the surface soil samples.

COMPARISON OF RESULTS WITH GUIDELINES

The survey findings indicate the total residual contamination is less than the NRC guidelines of 5000 dpm/100 cm^2 average, 15000 dpm/100 cm^2 maximum, and 1000 dpm/100 cm^2 removable, for alpha residual contamination. The survey findings also indicate that the total beta residual contamination is less that the NRC guidelines for Sr-90 (most restrictive category), 1000 dpm/100 cm^2 average, 3000 dpm/100 cm^2 maximum, and 200 dpm/100 cm^2 removable except for grid blocks 5, 13, 32, 39, 41, 47, 59, and 67. However, strontium 89/90 results of randomly selected concrete samples, analyzed by ORAU and the licensee, indicate the strontium 89/90 contamination to be less than the minimum detectable activity of 16 dpm (ORAU data). No radionuclide contamination of surface soils was detected. The water sample results are less than the EPA drinking water guidelines of 15 pCi/l gross alpha and 50 pCi/l gross beta.

The only guideline which was not met, was the exposure rate criteria. The baseline exposure rate was established as 12 μR/h, and the highest exposure rate in the reactor bay at one meter above the intersection of grid blocks 41, 42, 50, 51 was 18 μR/h. This result is probably slightly elevated due to the "bowl" geometry of the scabbled floor. The licensee proposed a scenario (RAI 86) of an individual occupying a desk for 2000 hours/per year adjacent to the scabbled area prior to backfilling. At the highest exposure rate location, this individual would receive a dose of 4.4 mrem/year. Our data supports their scenario, although we differ slightly in predicted dose, 4 mrem per year, which would be within the 10 mrem/year guidelines. If this area were backfilled with concrete to render the facility usable, the exposure rate would likely be reduced to within the guideline level of 5 μR/h above background.

SUMMARY

At the request of the Nuclear Regulatory Commission, Region V Office, ORAU conduted a confirmatory radiological survey of the Rockwell International L-85 Reactor facility, located in Santa Susana, California. The survey was performed on September 30 through October 2, 1986. The purpose was to verify the radiological condition of the facility relative to NRC guidelines for release for unrestricted use. Radiological information collected included gamma exposure rates, surface contamination levels, gamma spectra, and radionuclide concentrations in soil and miscellaneous media.

Based on the final results of the ORAU survey, and a review of the decommissioning documentation, it is ORAU's opinion that the L-85 Reactor facility (Building T-093) has been remediated to the existing NRC guidelines, with the exception of the exposure rate criteria. ORAU suggests that the ALARA concept has been met, and that restoration of the remediated area will reduce the exposure rate to the levels established by the Dismantling Order.

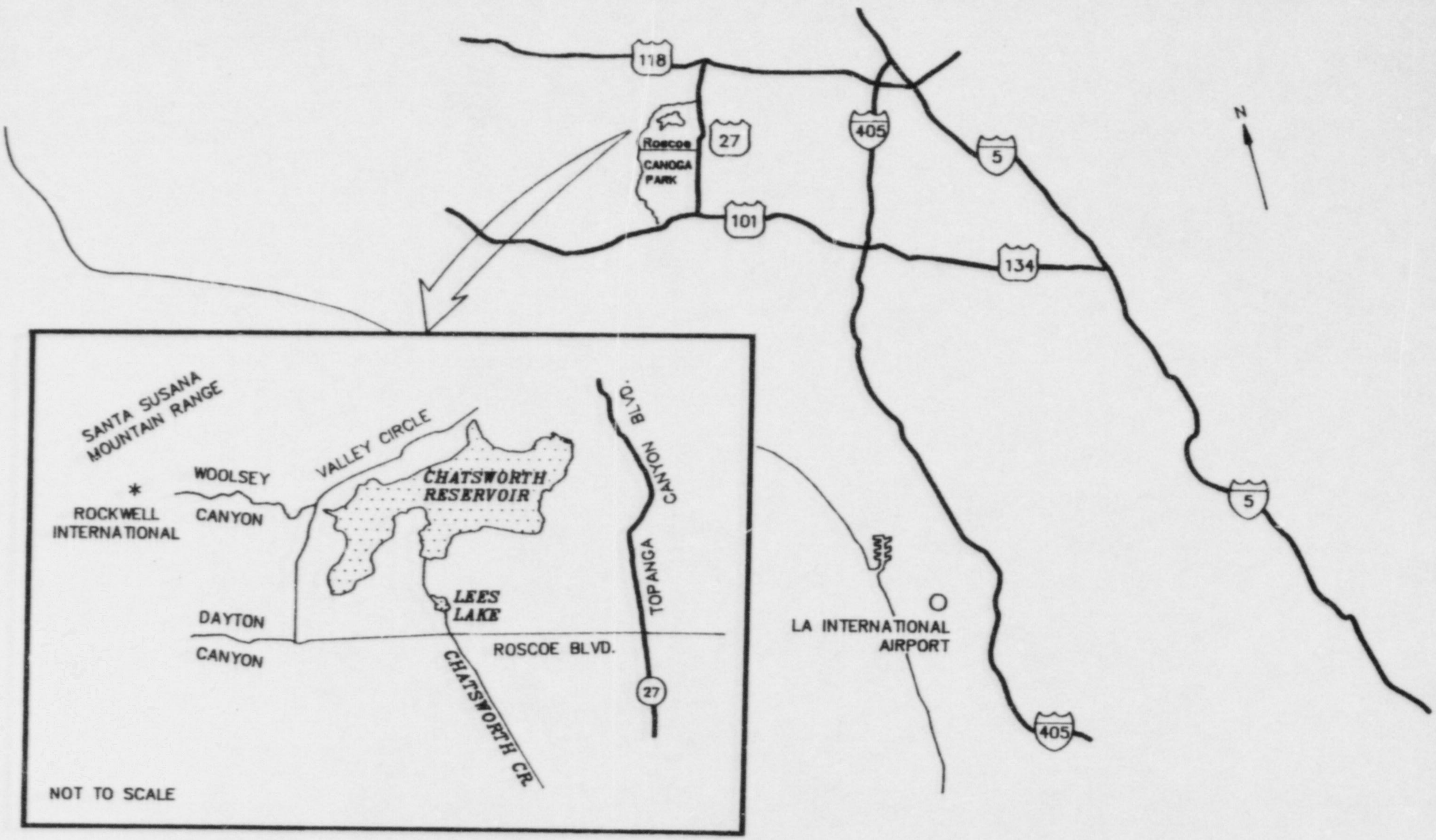

FIGURE 1: General Location of Rockwell International
Santa Susana, California

FIGURE 2: Building T-093. The high bay area of Building T-093 housed the L-85 reactor, and the low wing attached to the left served as the control room.

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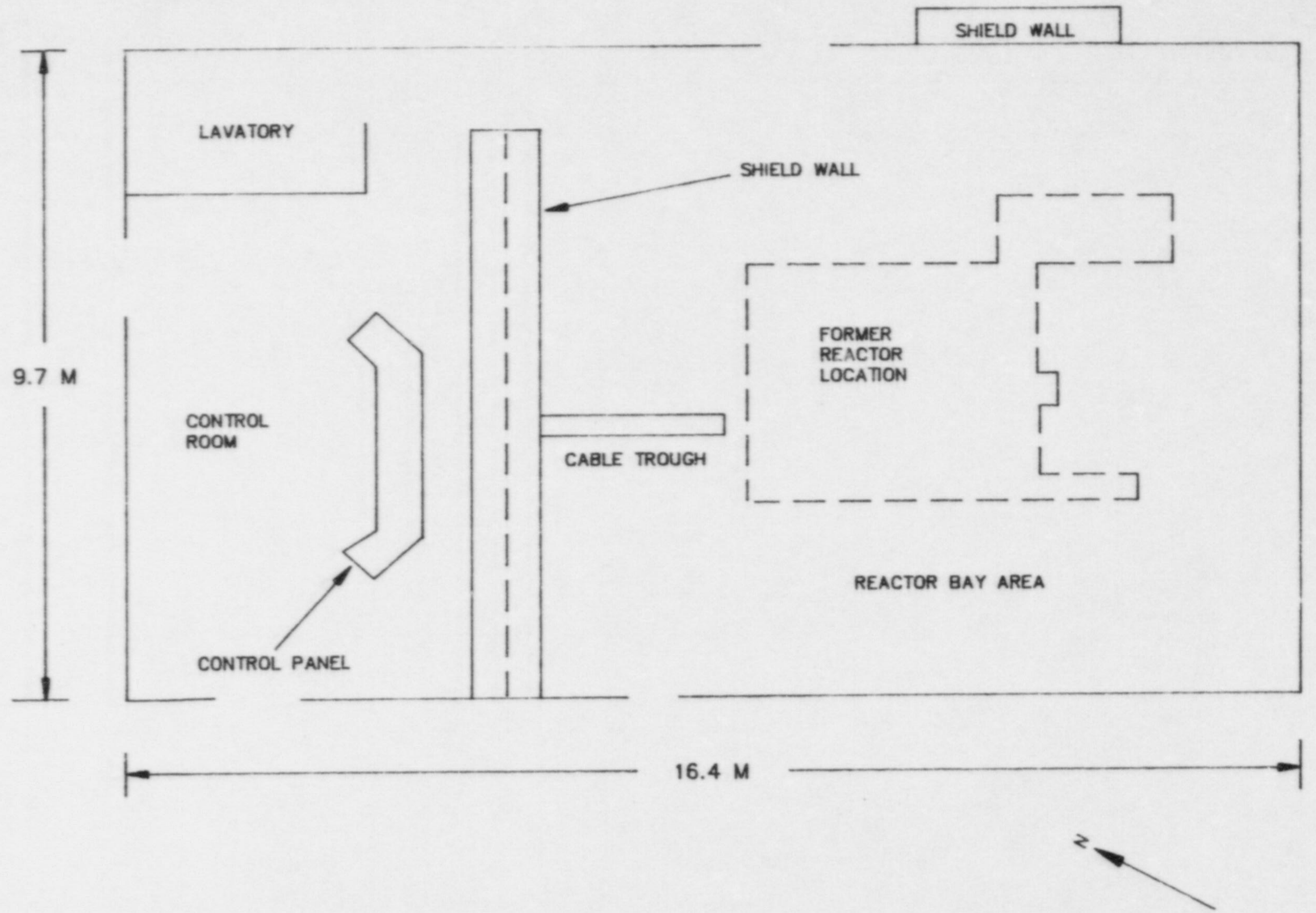

FIGURE 3: General Floor Plan of Building T-093

FIGURE 4: Surface Scans of Upper Walls and Ceiling. ORAU scanned the upper walls and ceiling using a hydraulic lift provided by Rockwell AI.

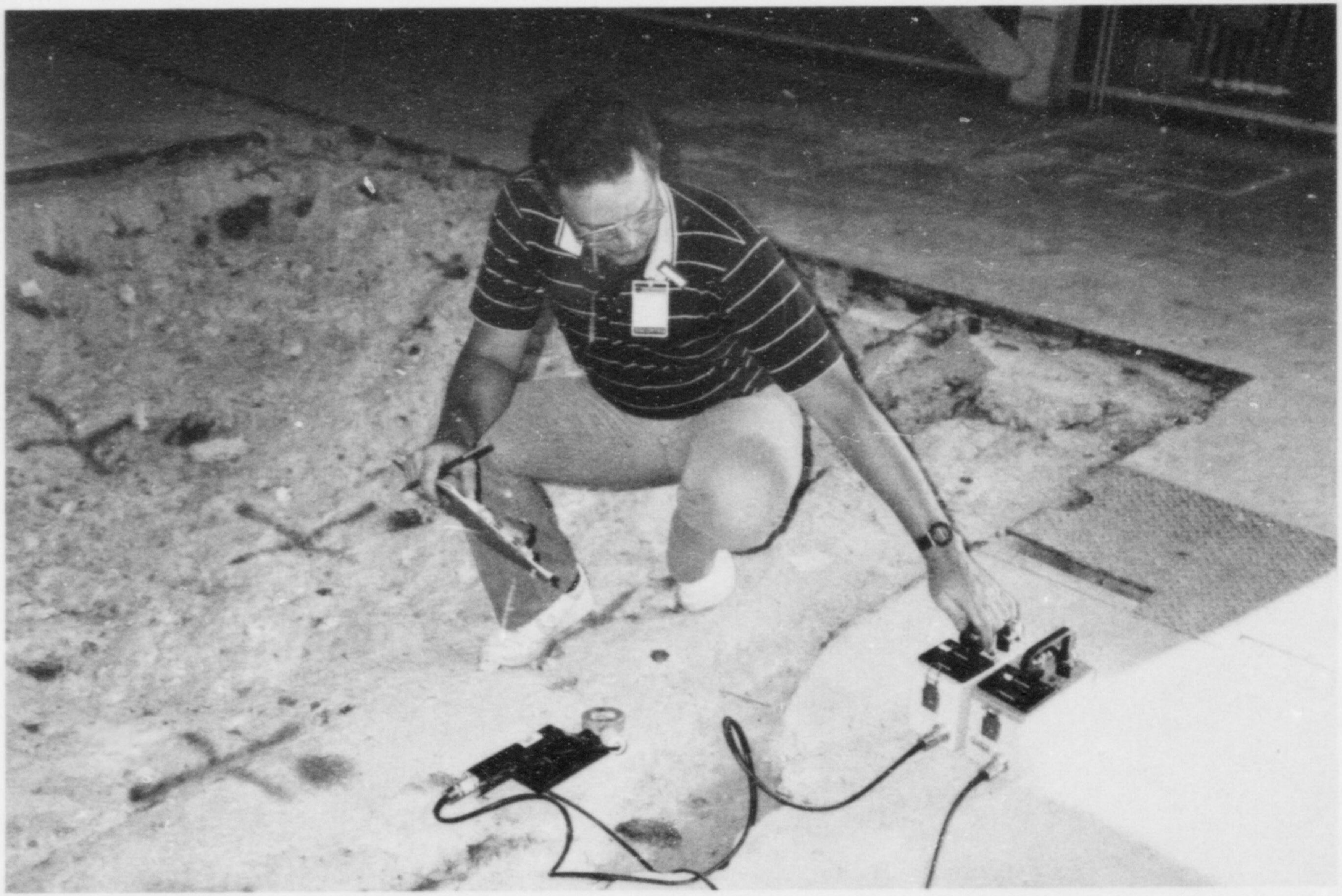

FIGURE 5: Direct Measurements of Floors and Lower Walls. Total alpha and beta-gamma contamination measurements were made using portable scalers and detectors. The grid system (X's on the floor) is visible in the remediated area of the concrete.

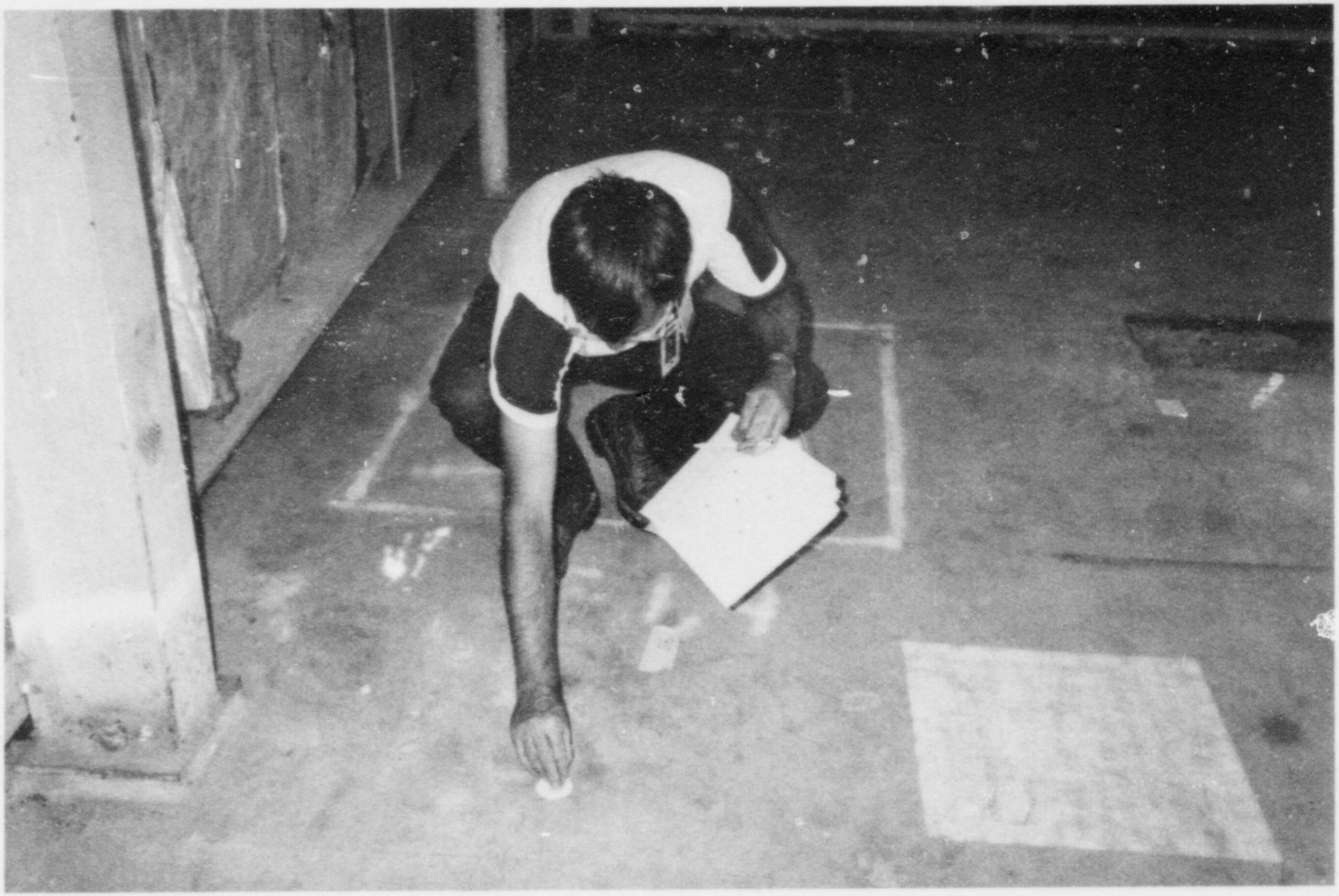

FIGURE 6: Transferable Contamination Surveys. Smears were taken at the location of the highest radiation measurement in each grid block.

FIGURE 7: In-situ Gamma Spectrometry Measurements. Gamma spectra were collected at random locations and locations of elevated radiation levels, using a high purity germanium detector and 4096 channel pulse height analyzer.

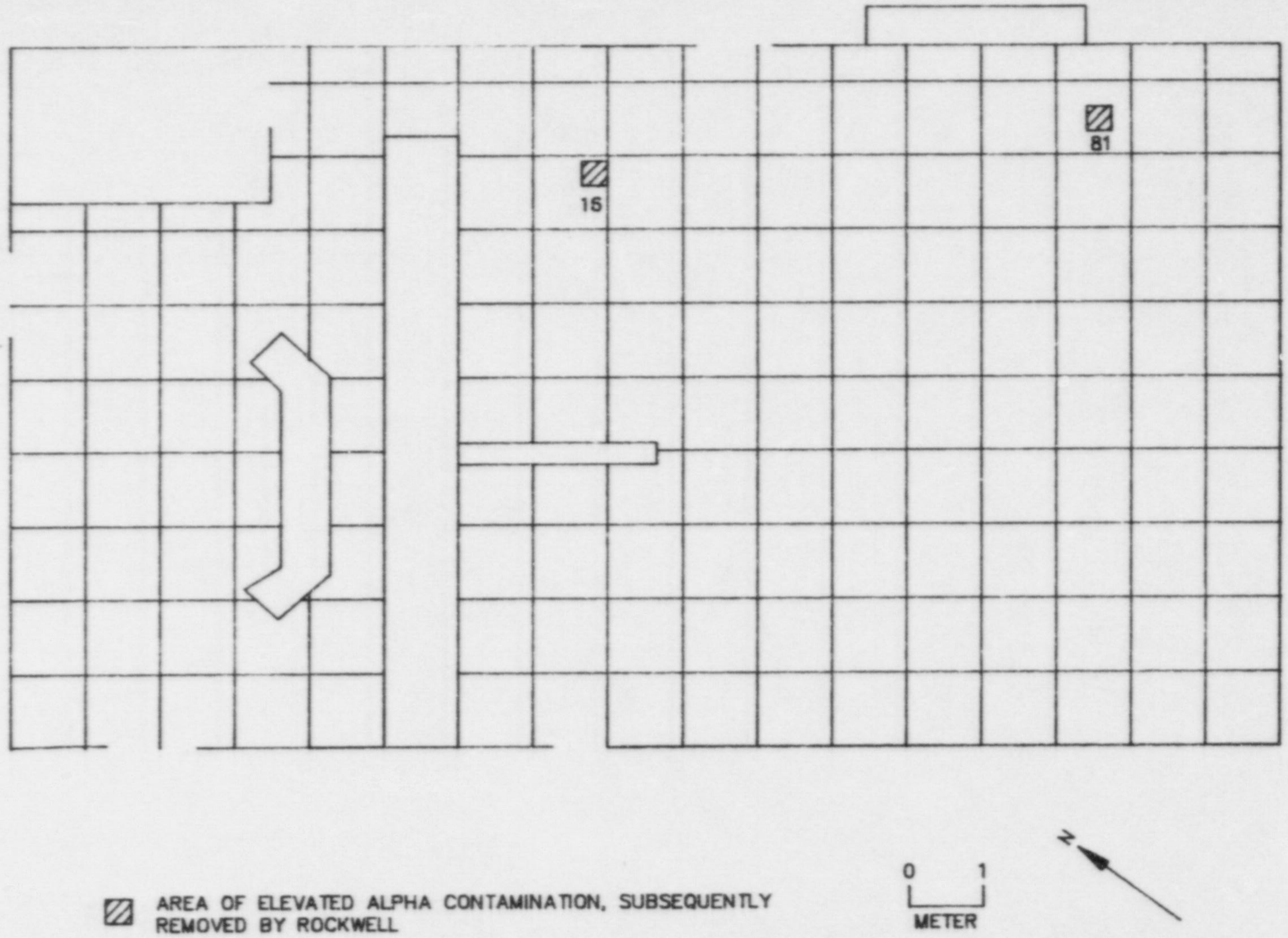

FIGURE 8: Elevated Alpha Levels in the Reactor Bay Area

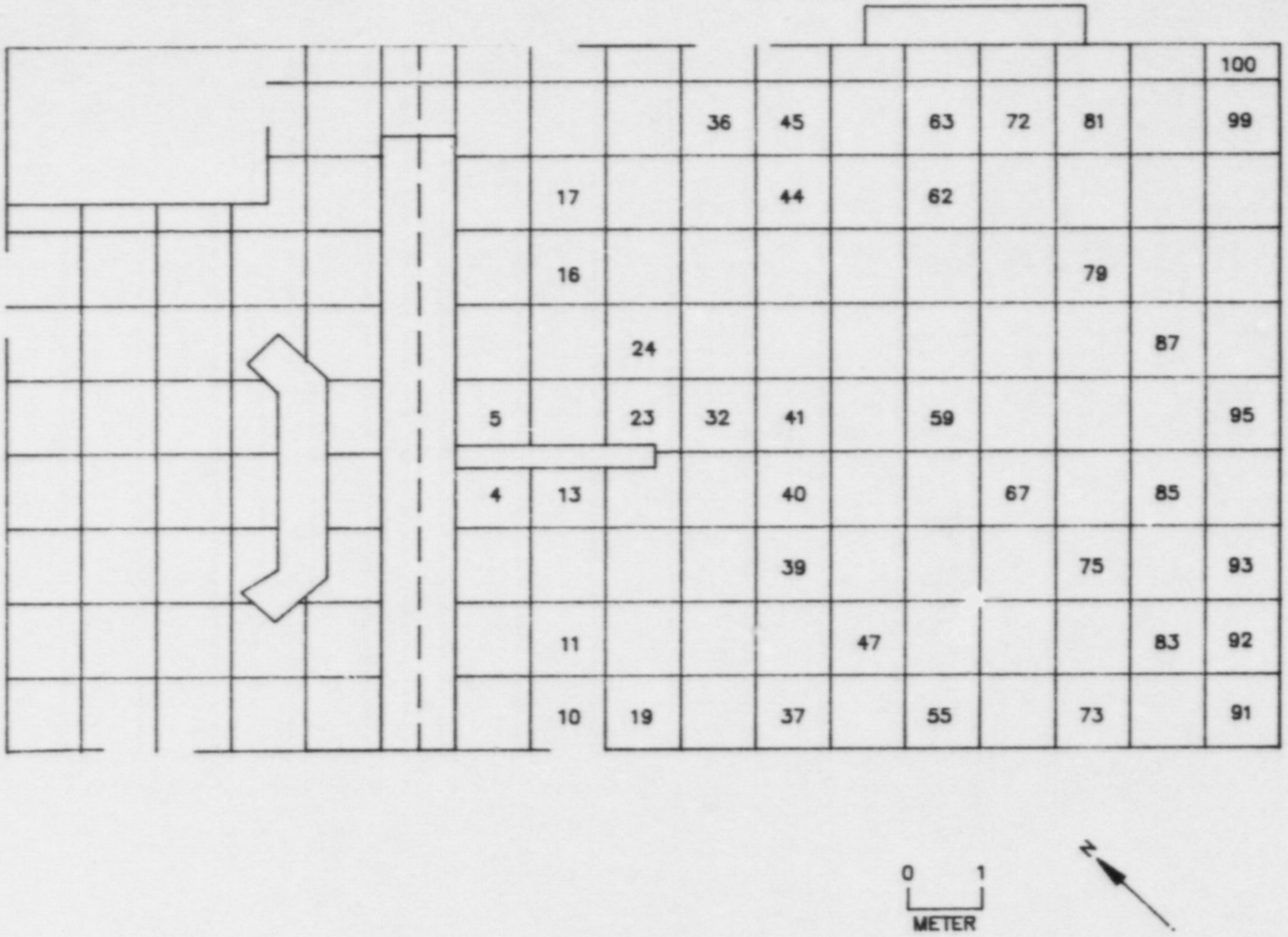

FIGURE 9: Location of 38 Random Grid Blocks on the Floor of the Reactor Bay Area

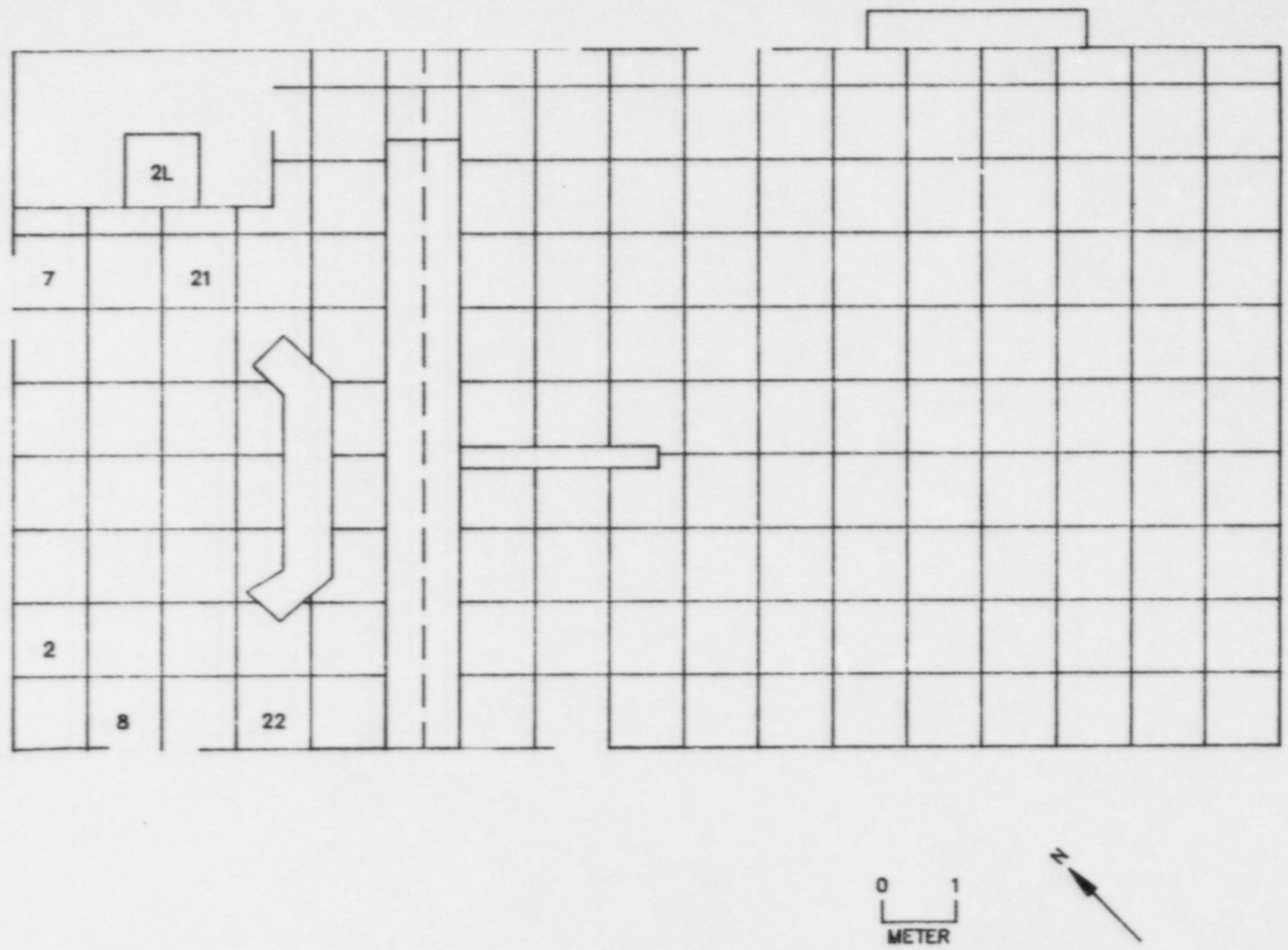

FIGURE 10: Location of 6 Random Grid Blocks on the Floor of the Control Room Area

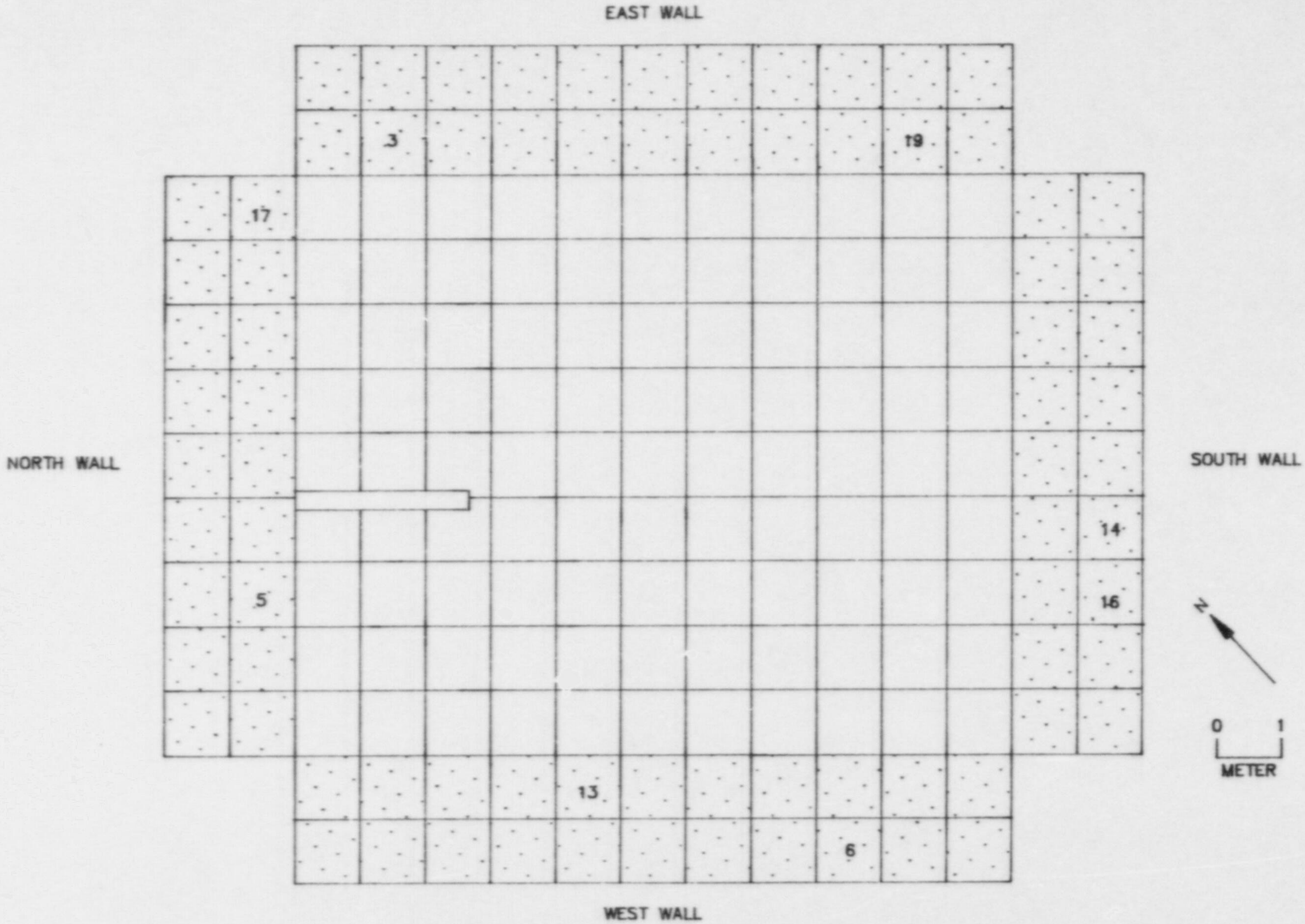

FIGURE 11: Location of 8 Random Grid Blocks on the Lower Walls of the Reactor Bay Area

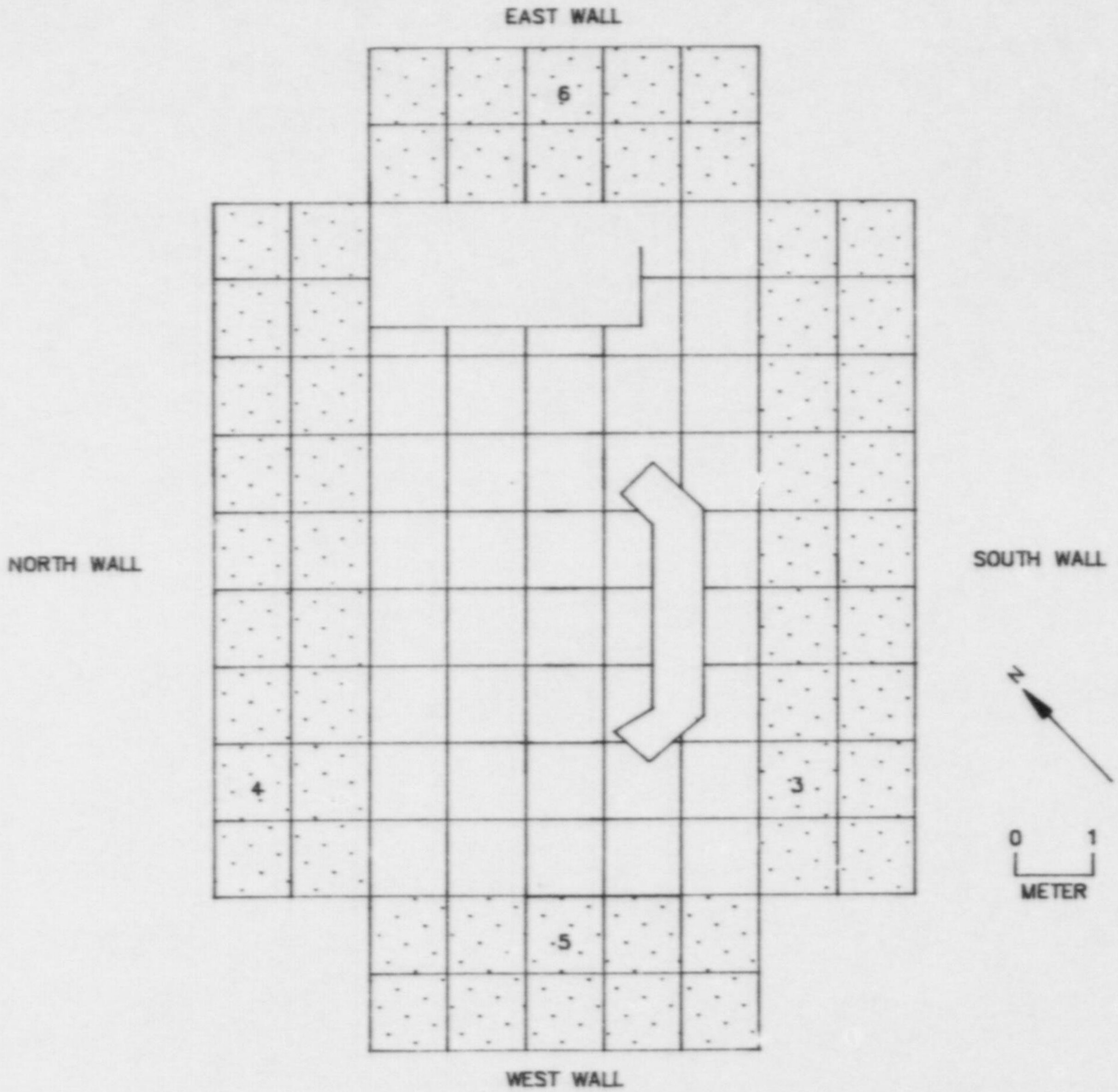

FIGURE 12: Location of 4 Random Grid Blocks on the Lower Walls of the Control Room

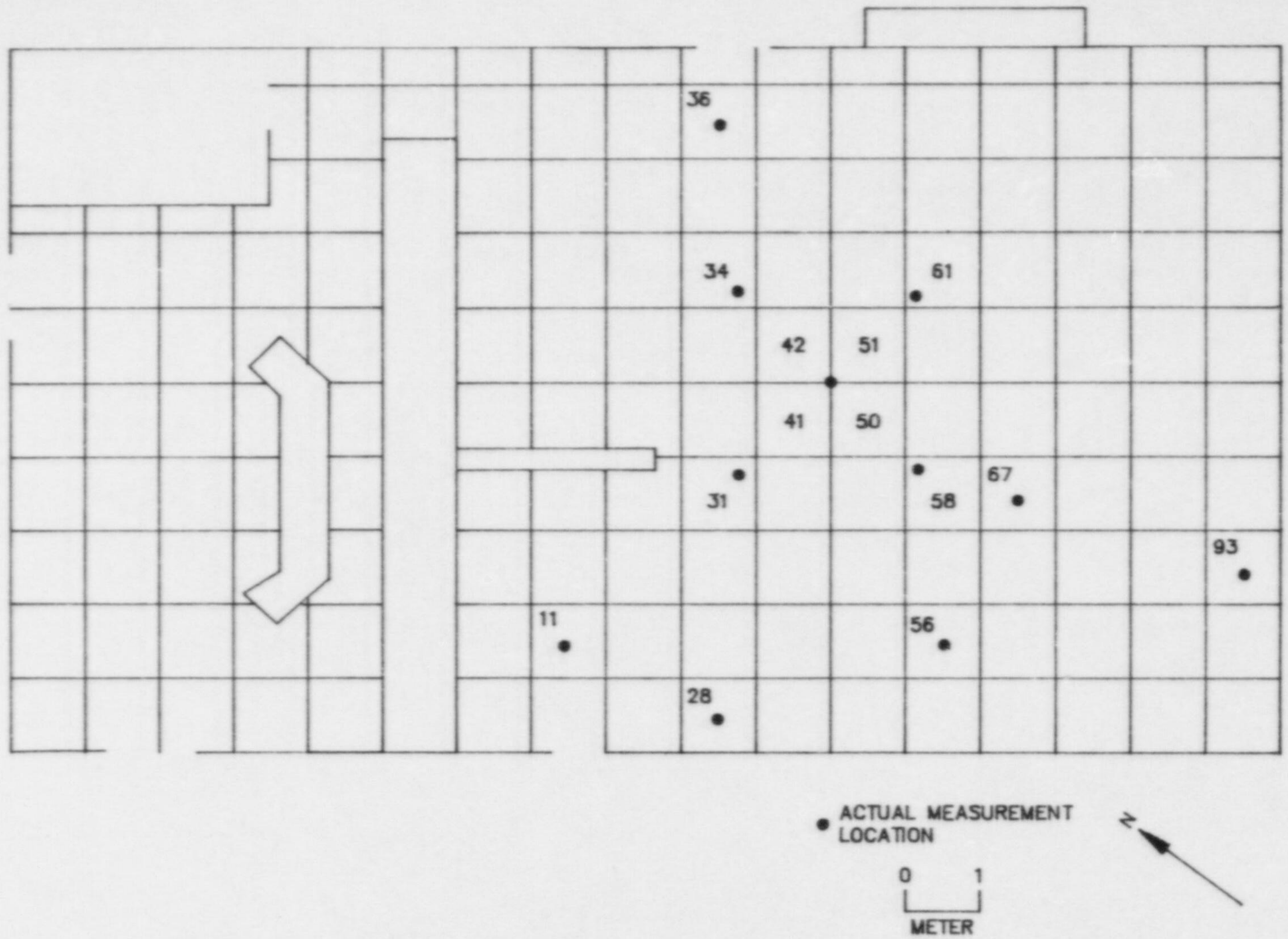

FIGURE 13: Location of Indoor Exposure Rate Measurements

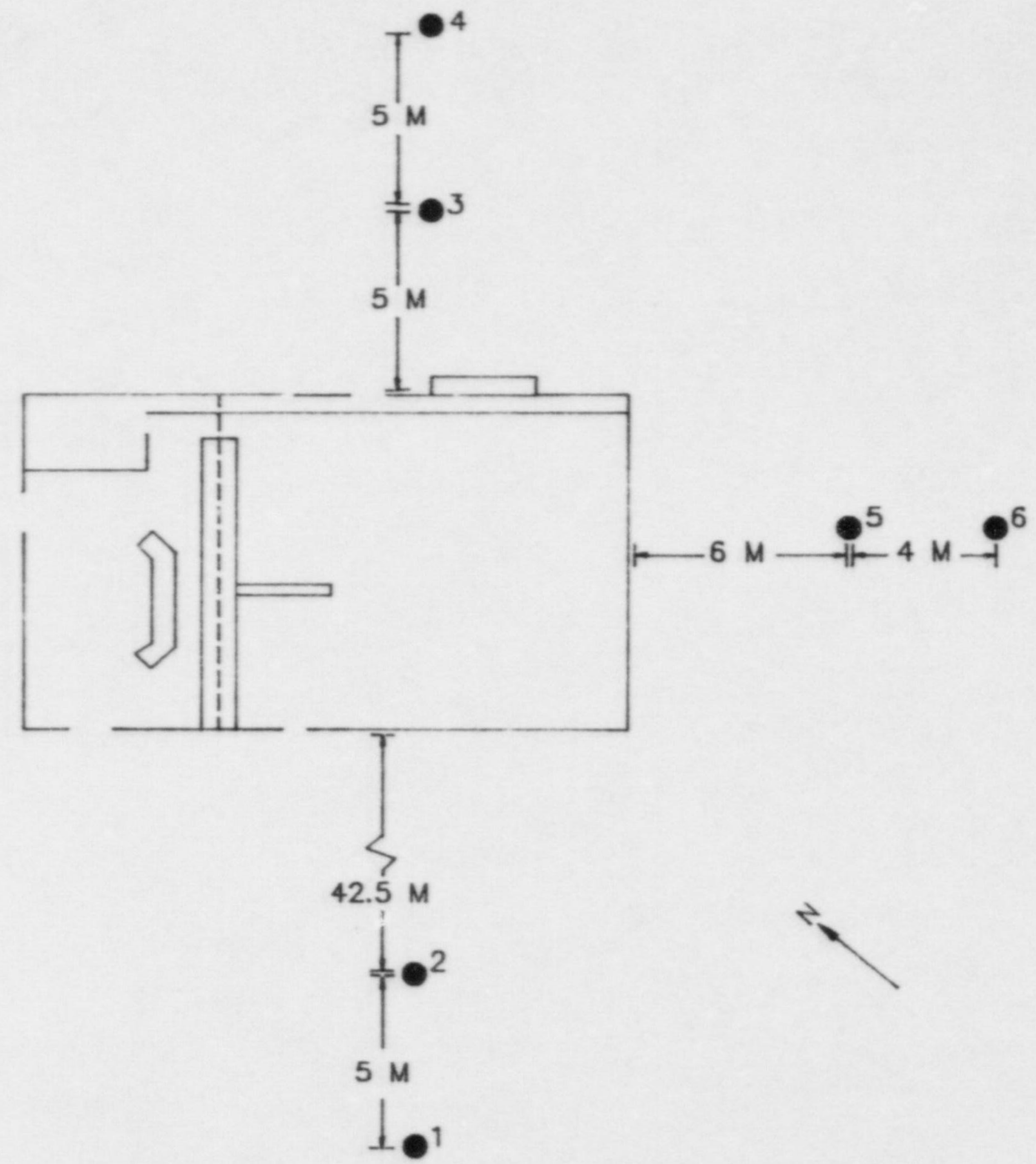

FIGURE 14: Location of 6 Random Surface Soil Samples Around Building T-093

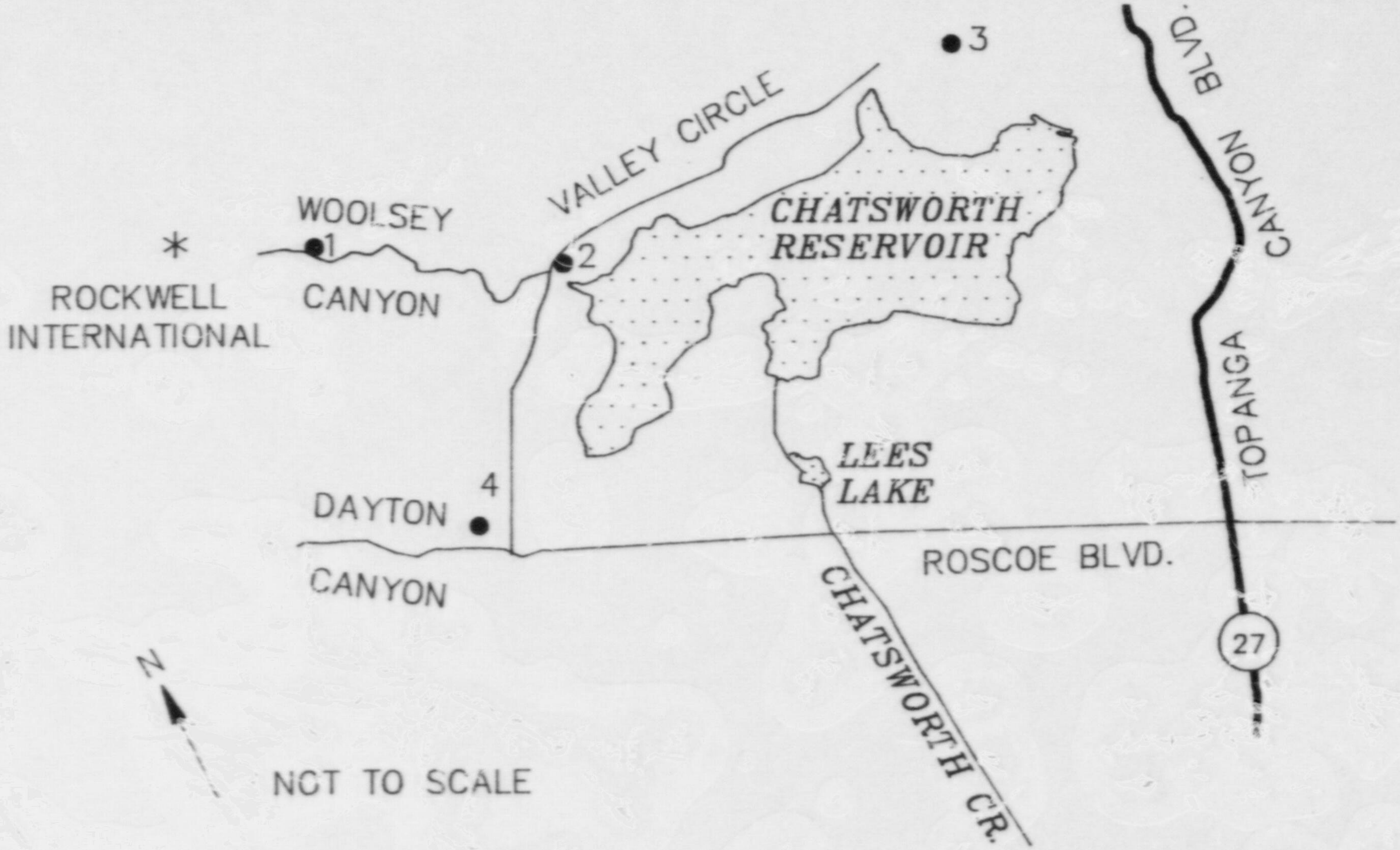

FIGURE 15: Location of 4 Baseline Samples in the Santa Susana Area

TABLE 1

GROSS ALPHA SURVEY RESULTS - REACTOR BAY FLOOR
ROCKWELL INTERNATIONAL
SANTA SUSANA, CALIFORNIA

Grid Block[a]	Total Activity (dpm/100 cm^2) Range	Average	Removable Activity (dpm/100 cm^2)
4	51 - 82	59	<3
5	<20 - 190	120	<3
10	51 - 82	61	<3
11	<20 - 100	61	<3
13	<20 - 120	59	<3
16	<14 - 1400[b]	330[b]	<3
17	19 - 100	45	<3
19	19 - 37	24	<3
23	<14 - 56	33	<3
24	<14 - 37	17	<3
32	<14 - 37	19	<3
36	<14 - 37	22	<3
37	<14 - 56	21	<3
39	41 - 210	80	<3
40	<14 - 37	26	<3
41	<14 - 28	<14	<3
44	<14 - 37	28	<3
45	19 - 46	30	<3
47	37 - 74	57	<3
55	<14 - 46	30	<3
59	<14 - 37	26	<3
62	<11 - 51	39	<3
63	<11 - 61	35	<3
67	20 - 41	35	<3
72	41 - 61	45	<3
73	28 - 93	45	<3
75	19 - 46	33	<3
79	41 - 61	51	<3
81	<20 - 5300[b]	930[b]	14 ± 10[c]
83	<14 - 56	28	<3
85	<14 - 56	35	<3
87	<14 - 46	22	<3
91	28 - 100	48	<3
92	<14 - 65	30	<3
93	19 - 56	34	<3
95	37 - 93	46	<3

TABLE 1 (Continued)

GROSS ALPHA SURVEY RESULTS - REACTOR BAY FLOOR
ROCKWELL INTERNATIONAL
SANTA SUSANA, CALIFORNIA

Grid Block	Total Activity (dpm/100 cm^2) Range	Average	Removable Activity (dpm/100 cm^2)
99	<14 - 110	39	<3
100	51 - 270	120	<3
16[b]	<14 - 390	160	<3
81[b]	<20 - 460	120	<3
Release Criteria	15000	5000	1000

[a]Refer to Figure 9.
[b]Area recleaned and surveyed. Additional data provided at bottom of table.
[c]Error is 2σ based only on counting statistics.

TABLE 2

GROSS BETA SURVEY RESULTS - REACTOR BAY FLOOR
ROCKWELL INTERNATIONAL
SANTA SUSANA, CALIFORNIA

Grid Block[a]	Total Activity (dpm/100 cm^2) Range	Average	Removable Activity (dpm/100 cm^2)
4	680 - 1200	950	<7
5	650 - 2500	1450[c]	<7
10	440 - 1000	820	<7
11	580 - 920	770	<7
13	850 - 1200	1000[c]	<7
16	750 - 1800	790	<7
17	<440 - 1400	840	<7
19	540 - 1100	960	<7
23	<490 - 1200	640	<7
24	<490 - 1200	670	<7
32	820 - 2700	1600[c]	<7
36	<490 - 880	790	<7
37	<490 - 780	590	<7
39	510 - 4900[c]	1800[c]	<7
40	<490 - 1600	840	<7
41	1200 - 1700	1500[c]	<7
44	540 - 990	760	<7
45	<490 - 1200	730	<7
47	990 - 3600[c]	1800[c]	<7
55	540 - 1000	730	<7
59	850 - 3300[c]	1600[c]	<7
62	680 - 1800	990	<7
63	<440 - 1100	780	<7
67	750 - 1300	1000[c]	<7
72	<440 - 1300	800	<7
73	<490 - 1300	710	<7
75	<490 - 1100	650	12 ± 8[b]
79	680 - 1300	980	<7
81	<440 - 1300	770	<7
83	<490 - 680	<490	<7
85	<490 - 990	550	8 ± 6
87	<490 - 1100	670	10 ± 7
91	<490 - 1500	730	<7
92	<490 - 950	650	<7

TABLE 2 (Continued)

GROSS ALPHA SURVEY RESULTS - REACTOR BAY FLOOR
ROCKWELL INTERNATIONAL
SANTA SUSANA, CALIFORNIA

Grid Block	Total Activity (dpm/100 cm^2) Range	Average	Removable Activity (dpm/100 cm^2)
93	<490 - 1100	<490	<7
95	<490 - 510	<490	<7
99	510 - 110C	710	<7
100	<440 - 1200	890	<7
Release Criteria	3000	1000	200

[a]Refer to Figure 9.
[b]Error is 2σ based only on counting statistics.
[c]Exceeds criteria for Sr-90. Sr-90 analysis of concrete samples indicates minimal presence of Sr-90.

TABLE 3

GROSS ALPHA SURVEY RESULTS - REACTOR BAY LOWER WALLS
ROCKWELL INTERNATIONAL
SANTA SUSANA, CALIFORNIA

Grid Block[a]	Wall Locations	Total Activity ($dpm/100\ cm^2$) Range	Average	Removable Activity ($dpm/100\ cm^2$)
5	North	31 - 100	57	<3
17	North	<20 - 100	65	<3
3	East	51 - 200	98	<3
19	East	51 - 170	110	<3
14	South	82 - 170	130	<3
16	South	61 - 210	130	<3
6	West	<20 - 200	94	<3
13	West	<20 - 37	22	<3
Release Criteria		15000	5000	1000

[a]Refer to Figure 11.

TABLE 4

GROSS BETA SURVEY RESULTS - REACTOR BAY LOWER WALLS
ROCKWELL INTERNATIONAL
SANTA SUSANA, CALIFORNIA

Grid Block[a]	Wall Location	Total Activity (dpm/100 cm^2) Range	Average	Removable Activity (dpm/100 cm^2)
5	North	<440 - 540	<440	<7
17	North	<440 - 540	<440	<7
3	East	<440 - 480	<440	<7
19	East	<440	<440	9 ± 7[b]
14	South	<440 - 610	<440	10 ± 7
16	South	<440 - 610	<440	<7
6	West	<440	<440	10 ± 7
13	West	<440	<440	<7
Release Criteria		3000	1000	200

[a]Refer to Figure 11.
[b]Error is 2σ based only on counting statistics.

TABLE 5

GROSS ALPHA SURVEY RESULTS - CONTROL ROOM FLOOR
ROCKWELL INTERNATIONAL
SANTA SUSANA, CALIFORNIA

Grid Block[a]	Total Activity (dpm/100 cm^2) Range	Average	Removable Activity (dpm/100 cm^2)
2	<18 - 73	25	<3
7	<18 - 82	58	<3
8	<18 - 46	<18	<3
21	<18 - 64	35	<3
22	<18 - 82	36	<3
2L[b]	<18 - 270	70	<3
Release Criteria	15000	5000	1000

[a]Refer to Figure 10.
[b]L refers to area in lavatory.

TABLE 6

GROSS BETA SURVEY RESULTS - CONTROL ROOM FLOOR
ROCKWELL INTERNATIONAL
SANTA SUSANA, CALIFORNIA

Grid Block[a]	Total Activity (dpm/100 cm^2) Range	Average	Removable Activity (dpm/100 cm^2)
2	<400 - 530	<400	<7
7	<400	<400	<7
8	<400	<400	<7
21	<400 - 560	<400	<7
22	<400 - 680	<400	<7
2L[b]	<400 - 650	<400	<7
Release Criteria	3000	1000	200

[a]Refer to Figure 10.
[b]L refers to area in lavatory.

TABLE 7

GROSS ALPHA SURVEY RESULTS - CONTROL ROOM LOWER WALLS
ROCKWELL INTERNATIONAL
SANTA SUSANA, CALIFORNIA

Grid Block[a]	Wall Location	Total Activity ($dpm/100\ cm^2$) Range	Average	Removable Activity ($dpm/100\ cm^2$)
4	North	<18 - 42	25	<3
6	East	<18 - 55	37	<3
3	South	<18	<18	<3
5	West	<18 - 27	<18	<3
Release Criteria		15000	5000	1000

[a]Refer to Figure 12.

TABLE 8

GROSS BETA SURVEY RESULTS - CONTROL ROOM LOWER WALLS
ROCKWELL INTERNATIONAL
SANTA SUSANA, CALIFORNIA

Grid Block[a]	Wall Location	Total Activity (dpm/100 cm^2) Range	Total Activity (dpm/100 cm^2) Average	Removable Activity (dpm/100 cm^2)
4	North	<400	<400	<7
6	East	<400	<400	<7
3	South	<400 - 650	<400	<7
5	West	<400 - 470	<400	<7
Release Criteria		3000	1000	200

[a]Refer to Figure 12.

TABLE 9

GROSS ALPHA SURVEY RESULTS - REACTOR BAY UPPER WALLS/CEILINGS
ROCKWELL INTERNATIONAL
SANTA SUSANA, CALIFORNIA

Location	Total Activity[a] (dpm/100 cm^2)	Removable Activity (dpm/100 cm^2)
West Wall-4m high @ 9m N	37	<3
West Wall-6m high @ 6m N	37	<3
Ceiling(Grid Block 38 Floor)	27	<3
West Wall-7m high @ 11m N	<18	<3
Traveling Crane Support NW Corner	<18	<3
North Wall-5m high @ 2m E	37	<3
North Wall-6m high @ 6m E	55	<3
North Wall-7m high @ 5m E	27	<3
East Wall-8m high @ 9.5m N	64	<3
East Wall-4m high @ 10m N	46	<3
East Wall-4m high @ 8m N	37	<3
Ceiling Beam-7m high @ 6.5m N x 4.5m E	160	10 ± 8[b]
Ceiling (Grid Block 34 Floor)	<18	<3
East Wall-5m high @ 5m N	64	<3
Ceiling Beam-8m high @ 6m N x 4m E	27	<3
Ceiling (Grid Block 81 Floor)	110	<3
East Wall-4m high @ 3m N	37	<3
South Wall-3m high @ 9m E	<18	<3
South Wall-5m high @ 8m E	110	<3
Ceiling (Grid Block 95 Floor)	120	<3
South Wall-5m high @ 3m E	120	<3
Ceiling (Grid Block 91 Floor)	<18	<3
West Wall-6m high @ 3m N	82	<3
West Wall-7m high @ 6m N	130	<3
West Wall-5m high @ 5m N	120	14 ± 10
Release Criteria	15000	1000

[a]Single point measurements performed at random locations.
[b]Error is 2σ based only on counting statistics.

TABLE 10

GROSS BETA SURVEY RESULTS - REACTOR BAY UPPER WALLS/CEILINGS
ROCKWELL INTERNATIONAL
SANTA SUSANA, CALIFORNIA

Location	Total Activity[a] (dpm/100 cm^2)	Removable Activity (dpm/100 cm^2)
West Wall-4m high @ 9m N	450	14 ± 7[b]
West Wall-6m high @ 6m N	<440	<7
Ceiling(Grid Block 38 Floor)	<440	<7
West Wall-7m high @ 11m N	<440	<7
Traveling Crane Support NW Corner	<440	<7
North Wall-5m high @ 2m E	<440	<7
North Wall-6m high @ 6m E	1400	<7
North Wall-7m high @ 5m E	<440	<7
East Wall-8m high @ 9.5m N	<440	<7
East Wall-4m high @ 10m N	820	20 ± 9
East Wall-4m high @ 8m N	1100	<7
Ceiling Beam-7m high @ 6.5m N x 4.5m E	1400	13 ± 7
Ceiling (Grid Block 34 Floor)	<440	<7
East Wall-5m high @ 5m N	<440	<7
Ceiling Beam-8m high @ 6m N x 4m E	<440	<7
Ceiling (Grid Block 81 Floor)	450	<7
East Wall-4m high @ 3m N	990	<7
South Wall-3m high @ 9m E	<440	9 ± 6
South Wall-5m high @ 8m E	1300	12 ± 7
Ceiling (Grid Block 95 Floor)	<440	<7
South Wall-5m high @ 3m E	1200	<7
Ceiling (Grid Block 91 Floor)	650	<7
West Wall-6m high @ 3m N	650	13 ± 7
West Wall-7m high @ 6m N	710	<7
West Wall-5m high @ 5m N	<440	12 ± 7
Release Criteria	3000	200

[a] Single point measurements performed at random locations.
[b] Error is 2σ based only on counting statistics.

TABLE 11

GROSS ALPHA SURVEY RESULTS - CONTROL ROOM UPPER WALLS/CEILING
ROCKWELL INTERNATIONAL
SANTA SUSANA, CALIFORNIA

Location	Total Activity[a] (dpm/100 cm^2)	Removable Activity (dpm/100 cm^2)
Ceiling (Grid Block 20 Floor)	27	<3
South Wall-3m high @ 1.5m E	<18	<3
Top of Ventilation Duct @ 5m E of West Wall	27	<3
Ceiling (Grid Block 2 Floor)	<18	<3
Ceiling (Grid Block 3L[b] Floor	<18	<3
Release Criteria	15000	1000

[a]Single point measurements performed at random locations.
[b]L refers to area in lavatory.

TABLE 12

GROSS BETA SURVEY RESULTS - CONTROL ROOM UPPER WALLS/CEILING
ROCKWELL INTERNATIONAL
SANTA SUSANA, CALIFORNIA

Location	Total Activity[a] (dpm/100 cm^2)	Removable Activity (dpm/100 cm^2)
Ceiling (Grid Block 20 Floor)	<400	<7
South Wall-3m high @ 1.5m E	<400	<7
Top of Ventilation Duct @ 5m E of West Wall	<400	<7
Ceiling (Grid Block 2 Floor)	<400	<7
Ceiling (Grid Block 3L[b] Floor)	<400	<7
Release Criteria	3000	200

[a]Single point measurements performed at random locations.
[b]L refers to area in lavatory.

TABLE 13

RADIONUCLIDE ACTIVITIES IN MISCELLANEOUS MEDIA
ROCKWELL INTERNATIONAL
SANTA SUSANA, CALIFORNIA

Sample No.	Location	Radionuclide Activity (dpm/100 cm^2)[a] Co-60	Cs-137	Eu-152	Eu-154	U-238	U-235
1- Residue	Reactor Bay North Wall Beam	<1.5	17 ± 6[b]	<3.6	<1.8	<0.9	<7.1
2- Residue	Reactor Bay East Wall Beam	<0.7	<1.0	<1.7	<0.8	15 ± 7	<2.0
3- Residue	Reactor Bay East Wall Beam	<0.7	2.1 ± 1.5	<1.5	<0.7		<3.1
4- Residue	Reactor Bay East Wall Ledge	1.0 ± 0.7	6.3 ± 1.6	1.8 ± 1.6	0.9 ± 0.8	22 ± 8	<1.7
5- Residue	Reactor Bay South Wall Ledge	<2.9	2.6 ± 0.9	<0.9	<0.5	1.7 ± 0.7	<1.9
6- Residue	Reactor Bay West Wall Ledge	<0.8	<2.1	<1.6	<0.9	23 ± 10	<5.0
7- Residue	Reactor Bay Floor	<0.2	0.7 ± 0.2	0.3 ± 0.2	0.2 ± 0.1	1.6 ± 0.7	<0.3
8- Paint	Reactor Bay North Wall	<12	<7.9	<13	<7.0	<160	<33
9- Paint	Reactor Bay South Wall	<10	35 ± 18	<15	<7.3	303 ± 173	<44
10-Paint	Control Room North Wall	<6.6	<6.6	<11	<6.6	<56	<35
11-Smear	Lavatory Sink Trap	<11	<11	<17	<8.2	<70	<41
12-Conrete Chips[c]	Reactor Bay Floor	0.9 ± 0.7	<0.4	11 ± 2	<0.9	<4.1	<1.4

[a]Sample was collected from an area exceeding 100 cm^2, but the data has been normalized to 100 cm^2.
[b]Error is 2σ based only on counting statistics.
[c]Concrete chips and dust were collected from scabbled area of the Reactor Bay Floor. Results are reported in pCi/g.

TABLE 14

INDOOR EXPOSURE RATE MEASUREMENTS
ROCKWELL INTERNATIONAL
SANTA SUSANA, CALIFORNIA

Location	Grid Block[a]	Exposure Rate (μR/h) at 1 m Above Surface
Reactor Bay	11	13
Reactor Bay	28	13
Reactor Bay	36	14
Reactor Bay	56	13
Reactor Bay	67	14
Reactor Bay	93	12
Reactor Bay	34	14
Reactor Bay	61	17
Reactor Bay	58	15
Reactor Bay	31	14
Reactor Bay	Corner 41/42 and 50/51	18
Bldg. T-453 (Background Location)	Center of Bldg.	12

[a]Refer to Figure 13.

TABLE 15

EXPOSURE RATES AND RADIONUCLIDE CONCENTRATIONS IN SURFACE SOIL SAMPLES
ROCKWELL INTERNATIONAL
SANTA SUSANA, CALIFORNIA

Location[a]	Exposure Rate At 1 m From Surface (μR/h)	Radionuclide Concentrations (pCi/g)					
		Co-60	Cs-137	Eu-152	Eu-154	U-238	U-235
47.5m West of Reactor Bldg.	18	<0.1	0.4 ± 0.1[b]	<0.1	<0.1	1.4 ± 0.9	<0.2
42.5m West of Reactor Bldg.	16	<0.1	<0.1	<0.1	<0.1	2.5 ± 0.7	<0.2
5m East of Reactor Bldg.	16	<0.1	<0.1	<0.1	<0.1	1.0 ± 0.5	<0.2
10m East of Reactor Bldg.	16	<0.1	0.4 ± 0.1	<0.1	<0.1	1.3 ± 0.8	<0.2
6m South of Reactor Bldg.	16	<0.1	0.1 ± 0.1	<0.1	<0.1	2.1 ± 0.6	<0.2
10m South of Reactor Bldg.	16	<0.1	<0.1	<0.1	<0.1	2.1 ± 0.6	<0.2

[a]Refer to Figure 14.
[b]Error is 2σ based only on counting statistics.

TABLE 16

EXPOSURE RATES AND RADIONUCLIDE CONCENTRATIONS IN SURFACE SOIL BASELINE LOCATIONS
ROCKWELL INTERNATIONAL
SANTA SUSANA, CALIFORNIA

Location[a]	Exposure Rate At 1 m From Surface	Radionuclide Concentrations (pCi/g)					
		Co-60	Cs-137	Eu-152	Eu-154	U-238	U-235
Woolsey Canyon Rd. 1.6 km from gate	13	<0.1	<0.1	<0.1	<0.1	<0.7	<0.2
Woolsey Canyon Rd. @ Valley Circle	13	<0.1	<0.1	<0.1	<0.1	0.7 ± 0.4[b]	<0.2
Valley Circle @ Schumann Drive	13	<0.1	<0.1	<0.1	<0.1	<1.5	<0.3
Valley Circle @ Roscoe Dr.	10	<0.1	<0.1	<0.1	<0.1	<1.5	<0.3
Bldg. T-453 (Indoor)	12	--[c]	--	--	--	--	--

[a]Refer to Figure 15.
[b]Error is 2σ based only on counting statistics.
[c]Exposure rate data only.

REFERENCES

NRC 83 Rockwell International, Docket No. 50-375, Order Authorizing Dismantling of Facility and Disposition of Component Parts, USNRC, Division of Licensing, February 22, 1983.

RAI 86 Letter from M.E. Remley, Rockwell A.I. to H. Denton, USNRC, dated 03/06/86. Attachment: Revised Radiation Survey Report for L-85 Nuclear Examination Reactor (7).

RG 186 Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors", USNRC.

RI 85 Procedure for Dismantling and Decontaminating the L-85 Reactor Facility, Rocketdyne Division, Rockwell International Corporation, July 9, 1985.

RI 86 Radiation Survey Report of the L-85 Reactor Facility Following Dismantlement and Decontaminaton of the Facility, Docket No. 50-375, Rocketdyne Division, Rockwell International Corporation, March 6, 1986.

APPENDIX A

MAJOR ANALYTICAL EQUIPMENT

APPENDIX A

Major Analytical Equipment

The display or description of a specific product is not to be construed as an endorsement of that product or its manufacturer by the authors or their employer.

A. Direct Radiation Measurements

Eberline "RASCAL"
Portable Ratemeter-Scaler
Model PRS-1
(Eberline, Sante Fe, NM)

Eberline PRM-6
Portable Ratemeter
(Eberline, Sante Fe, NM)

Ludlum Alpha Beta Floor Monitor
Model 239-1
(Ludlum, Sweetwater, TX)

Ludlum Model 2220
Portable Scaler-Ratemeter
(Ludlum, Sweetwater, TX)

Eberline Alpha Scintillation Probe
Models AC-3-7
(Eberline, Sante Fe, NM)

Eberline Beta-Gamma "Pancake" Probe
Model HP-260
(Eberline, Sante Fe, NM)

Victoreen Beta-Gamma "Pancake" Probe
Model 489-110
(Victoreen, Inc., Cleveland, OH)

Reuter-Stokes Pressurized Ionization Chamber
Model RSS-111
(Reuter-Stokes, Cleveland, OH)

Victoreen NaI Gamma Scintilltion Probe
Model 489-55
(Victoreen, Inc., Cleveland, OH)

High Purity Germanium Detector
Model GEM-13180-S, 13% efficiency
(EG&G ORTEC, Oak Ridge, TN)

Multichannel Analyzer
Canberra Series 30
(Canberra Instruments, Meridian, CT)

B. Laboratory Analyses

Low Background Alpha-Beta Counter
Model LB5110-2080
(Tennelec, Inc., Oak Ridge, TN)

Source	Serial #	Calibration Efficiency
Am-241	1777-3-84	.276
Cs-137	2069-2-15	.411

Ge(Li) Detectors
Model LGCC2220SD, 23% efficiency
(Princeton Gamma-Tech, Princeton, NJ)

Used in conjunction with:
Lead Shield, SPG-16
(Applied Physical Technology, Smyrna, GA)

High-Purity Germanium Detector
Model GMX-23195-S, 23% efficiency
(EG&G ORTEC, Oak Ridge, TN)

Used in conjunction with:
Lead Shield, G-16
(Gamma Products Inc., Palos Hills, IL)

Multichannel Analyzer
ND-66/ND 680 System
(Nuclear Data, Inc., Schaumburg, IL)

High Purity Germanium Coaxial Well Detecter
Model GWL 110210-PWS-S, 23% efficiency
(EG&G ORTEC, Oak Ridge, TN)

C. Site Specific Equipment List

Scaler/Ratemeter		Detector		Calibration Date
PRM-6	#5	NaI	#11	Daily Onsite
PRM-6	#6	NaI	# 8	Daily Onsite
PRM-6	#8	NaI	#14	Daily Onsite
PRM-6	#9	NaI	#13	Daily Onsite
RASCAL	#11	AC-3	# 3	09/24/86
RASCAL	#13	AC-3	#12	09/23/86
RASCAL	#14	AC-3	#14	09/23/86
RASCAL	#13	HP-260	#13	09/24/86
RASCAL	#14	HP-260	# 7	09/24/86
RASCAL	#14	HP-260	#11	09/24/86

APPENDIX B

MEASUREMENT AND ANALYTICAL PROCEDURES

APPENDIX B

Analytical Procedures

Alpha and Beta-gamma Measurements

Measurements of total alpha radiation levels were performed using Eberline Model PRS-1 portable scaler/ratemeters with Model AC-3-7 alpha scintillation probes. Measurements of total beta-gamma radiation levels were performed using Eberline Model PRS-1 portable scaler/ratemeters with Model HP-260 thin-window "pancake" G-M probes. Count rates (cpm) were converted to disintegration rates (dpm/100 cm^2) by dividing the net rate by the 4π efficiency and correcting for the active area of the detector. Although other factors (i.e. backscatter) can affect the calibration, they are considered insignificant for the measurements performed. Effective window areas were 59 cm^2 for the ZnS detectors and 15 cm^2 for the G-M detectors. Background count rates for ZnS alpha probes averaged approximately 1 cpm; the average background count rate was approximately 40 cpm for the G-M detectors.

Surface Scan

Surface scans of grid blocks in the facility were performed by passing the probes slowly over the surface. The distance between the probe and the surface was maintained at a minimum - nominally about 1 cm. Identification of elevated levels was based on increases in the audible signal from the recording or indicating instrument. Alpha and beta-gamma scans of large surface areas on the floor of the facility were accomplished by use of a gas proportional floor monitor, with a 600 cm^2 sensitive area. The instrument is slowly moved in a systematic pattern to cover 100% of the accessible area. Combinations of detectors and instrument for the scans were:

Beta-Gamma	- G-M probe with PRM-6 ratemeter.
Beta-Gamma	- G-M probe with "RASCAL" scaler/ratemeter.
Gamma	- NaI scintillation detector (3.2 cm x 3.8 cm crystal) with PRM-6 ratemeter.
Alpha	- ZnS probe with "RASCAL" scaler/ratemeter.
Alpha/Beta	- Gas proportional floor monitor with PRM-6 ratemeter or Ludlum Model 2220 scaler/ratemeter.

Gamma Exposure Rate Measurements

Measurements of gamma exposure rates were performed using a Reuter-Stokes pressurized ionization chamber. The average of a minimum of five readings was determined at a distance of 1 meter from the surface to the center of the chamber. Gamma spectra was collected at each location where exposure rate measurements were made, using a portable high purity germanium detector and MCA system.

Removable Contamination Measurements

Smears for determination of removable contamination levels were collected on numbered filter paper disks 47 mm in diameter, then placed in individually labeled envelopes with the location and other pertinent information recorded. The smears were counted on a low background alpha-beta counter.

Soil Sample Analysis

Soil samples were dried, mixed, and a portion sealed in 0.5-liter Marinelli beaker. The quantity placed in each beaker was chosen to reproduce the calibrated counting geometry and ranged from 400 to 900 g of soil. Net soils weights were determined and the samples counted using Ge(Li) and intrinsic germanium detectors coupled to a Nuclear Data Model ND-680 pulse height analyzer system. Background and Compton stripping, peak search, peak identification, and concentration calculations were performed using the computer capabilities inherent in the analyzer system. The individual spectra were reviewed for identifiable photopeaks related to nuclear fuels, activation, or fission products.

Miscellaneous Media

Miscellaneous media of very small sample size (dust, residue, standing water, concrete chips, etc.) were analyzed in a high-purity Germanium Coaxial well detector coupled to a Nuclear Data Model ND-680 pulse height analyzer. The individual spectra were reviewed for identifiable photopeaks related to fuel, activation, or fission products.

Errors and Detection Limits

The errors associated with the analytical data presented in the tables of this report, represent the 95% (2σ) confidence levels for that data. These errors were calculated, based on both the gross sample count levels and the associated background count levels. When the net sample count was less than the 2σ statistical deviation of the background count, the sample concentration was reported as less than the minimum detectable activity (<MDA) or concentration (<MDC). Because of variation in Compton contribution from other radionuclides in the samples, the MDA's/MDC's for specific radionuclides differ from sample to sample.

Calibration and Quality Assurance

Laboratory and field survey procedures are documented in manuals developed specifically for the Oak Ridge Associated Universities' Radiological Site Assessment Program.

With the exception of the measurements conducted with portable gamma scintillation survey meters, instruments were calibrated with NBS-traceable standards. The calibration procedures for the portable gamma instruments are performed by comparison with an NBS calibrated pressurized ionization chamber. Gamma spectrometry systems are calibrated as needed using NBS traceable standards; quarterly NBS traceable standards are counted to verify efficiency factors, and daily backgrounds and standards are counted to check calibration drift or loss of resolution. Quality assurance results are maintained on file.

Quality control procedures on all instruments included daily background and check-source measurements to confirm equipment operation within acceptable statistical fluctuations. The ORAU laboratory participates in the EPA and EML Quality Assurance Programs.

APPENDIX C

REGULATORY GUIDE 1.86
TERMINATION OF OPERATING LICENSES
FOR NUCLEAR REACTORS

June 1974

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 significant 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.