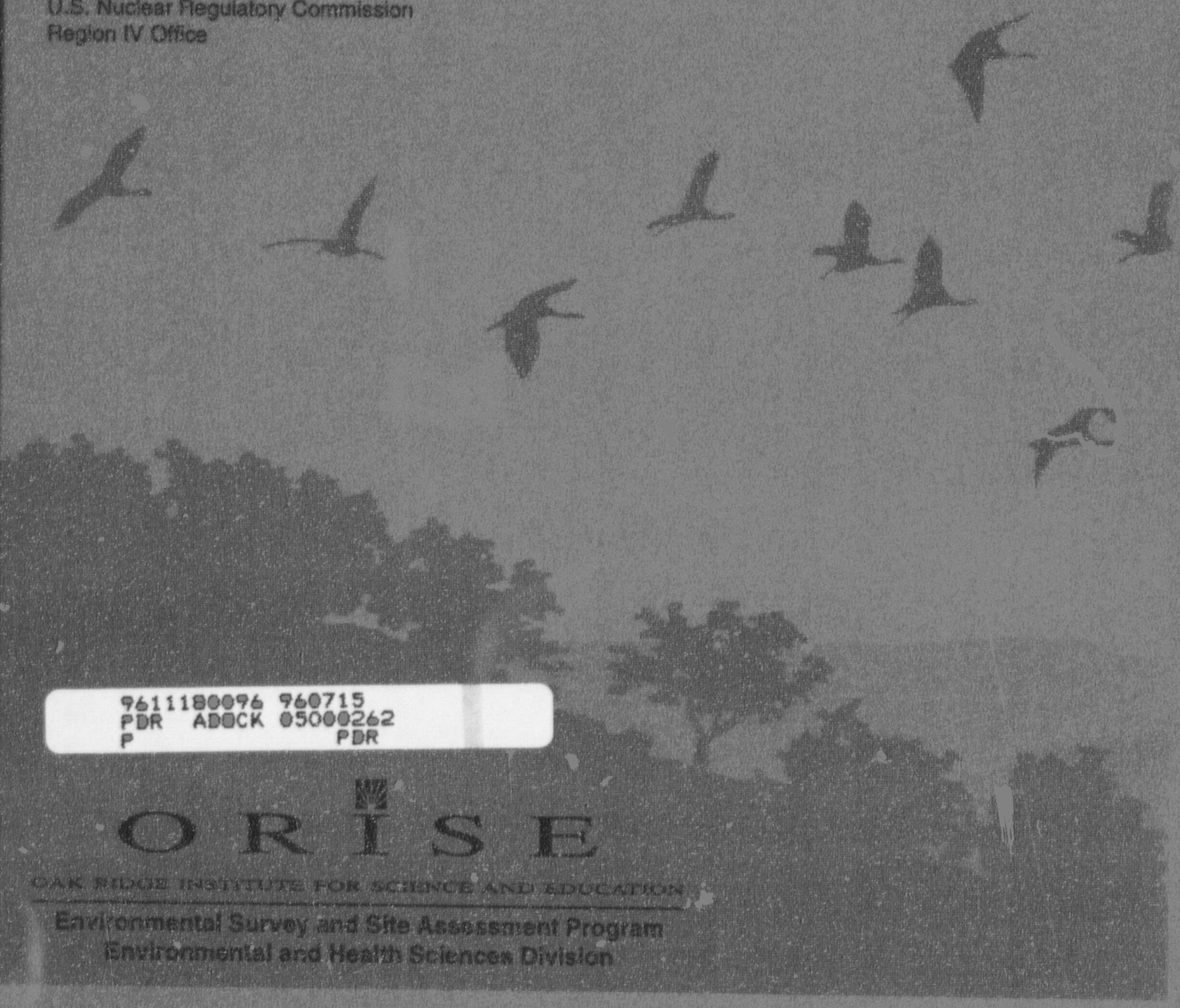
RADIOLOGICAL SURVEY FOR THE BRIGHAM YOUNG UNIVERSITY L-77 RESEARCH REACTOR PROVO, UTAH

[DOCKET 050-262]

W. C. ADAMS and J. R. MORTON

Prepared for the
U.S. Nuclear Regulatory Commission
Region IV Office



ORISE
OAK RIDGE INSTITUTE FOR SCIENCE AND EDUCATION
Environmental Survey and Site Assessment Program
Environmental and Health Sciences Division

The **Oak Ridge Institute for Science and Education** (ORISE) was established by the U.S. Department of Energy to undertake national and international programs in science and engineering education, training and management systems, energy and environment systems, and medical sciences. ORISE and its programs are operated by Oak Ridge Associated Universities (ORAU) through a management and operating contract with the U.S. Department of Energy. Established in 1946, ORAU is a consortium of 89 colleges and universities.

NOTICES

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ORISE 96/F-49

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Prepared by

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Prepared for the

U.S. Nuclear Regulatory Commission
Region IV Office

Sponsored by the
Office of Nuclear Materials Safety and Safeguards
U.S. Nuclear Regulatory Commission

JUNE 1996

FINAL REPORT

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RADIOLOGICAL SURVEY
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L-77 RESEARCH REACTOR
PROVO, UTAH

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ABBREVIATIONS AND ACRONYMS

α	alpha
β	beta
γ	gamma
μR/h	microroentgen per hour
μrem/h	microrem per hour
ASME	American Society of Mechanical Engineers
BKG	background
BYU	Brigham Young University
cm	centimeter
cm^2	square centimeter
cpm	counts per minute
Cs	cesium
DOE	Department of Energy
dpm/100 cm^2	disintegrations per minute per 100 square centimeters
EML	Environmental Measurements Laboratory
EPA	Environmental Protection Agency
ESSAP	Environmental Survey and Site Assessment Program
m	meter
m^2	square meter
MDC	minimum detectable concentration
NaI	sodium iodide
NIST	National Institute of Standards and Technology
NRC	Nuclear Regulatory Commission
ORISE	Oak Ridge Institute for Science and Education
Sr	strontium
Tc	technetium
Th	thorium
U	uranium
ZnS	zinc sulfide

RADIOLOGICAL SURVEY
OF THE
BRIGHAM YOUNG UNIVERSITY
L-77 RESEARCH REACTOR
PROVO, UTAH

INTRODUCTION AND SITE HISTORY

The Brigham Young University (BYU) L-77 Research Reactor was a small, solution-type nuclear reactor designed for laboratory use. It was manufactured by the Atomics International Division of North American Aviation, a predecessor of the Rocketdyne Division of Rockwell International and delivered to the university in August of 1967. Prior to that, this reactor had been used by Atomics International at several expositions in foreign countries where it had been loaded and operated for demonstration purposes. The reactor consisted of a core tank—which contained the liquid fuel solution—within an inner shield tank.

The reactor was fueled in September 1967 with an uranyl-sulfate solution with a final mass of 1447 grams of uranium-235 (U-235). During the next fifteen years, the reactor was used for 355 separate operations—for a total of 1799 watt-hours—in conjunction with physics classes taught at the university under U.S. Nuclear Regulatory Commission (NRC) License R-109, for research and training purposes. On May 12, 1982, all operations of the reactor terminated and the reactor was defueled on May 5, 1987.

There have been no recorded incidents, while loading or unloading fuel to the reactor, to indicate that there were any spills or other incidents that would have led to contamination of the reactor facility.

Operation of the reactor produced fission products, however only two—Sr-90 and Cs-137—are likely to remain following the ten years of inactivity. The liquid fuel is the only radioactive material associated with the reactor that presents a possibility of contaminating areas away from the reactor. Thus, any unreported spill involving the liquid fuel contaminated with fission products which may have occurred would have primarily included uranium contamination, along with Sr-90 and Cs-137 contamination. The licensee did not find any evidence of neutron activation of the stainless steel shell that divided the shielding water from the inner shielding components and the radiological analyses performed on samples of the shielding water indicated that there was no contamination in the sample. According to the licensee, the current radiological status of the facility, outside the reactor shield, is at background levels; therefore, because of the sealed and contained nature of the L-77 reactor, no contamination is expected outside of the core vessel.

BYU has decommissioned the facility and plans to release the site for unrestricted use. The final radiological survey of the reactor and its components and the reactor facility was performed by BYU and the results were provided to the NRC in April 1994 (BYU 1994). The reactor was disassembled and the contaminated materials identified during the decommissioning activities were shipped to Richland, Washington for disposal as low-level radioactive waste. The reactor fuel was shipped to EG&G in Idaho and the plutonium-beryllium neutron source has been transferred to the Physics department under a separate license. The remaining rooms in the facility—the Reactor Room, the Control Room, and the Accelerator Room—were surveyed and await release to unrestricted use pending the NRC's termination of License R-109.

The U.S. Nuclear Regulatory Commission, Region IV Office, requested that the Environmental Survey and Site Assessment Program (ESSAP) of the Oak Ridge Institute for Science and Education (ORISE) perform an independent radiological survey of the remaining reactor components and the reactor facility that are to be released for unrestricted use.

SITE DESCRIPTION

The research reactor facility is located on a hill just south of the Joseph Smith Building on the south edge of the Brigham Young University campus. Prior to the building being converted to a research reactor facility, the building had housed the main heating systems for the university. As the university expanded, the campus central heating systems were upgraded and moved to a new location—therefore, the Old Central Heating Facility was converted to the Nuclear Laboratory Building. The building is a steel, concrete, brick, and concrete block structure. Some of the interior walls consist of wood and sheetrock with tile or concrete flooring. There are three main rooms in the facility—the Reactor Room, the Control Room, and the Accelerator (Large) Room—which have a combined floor space of approximately 170 m^2 (Figure 1).

OBJECTIVES

The objectives of this radiological survey were to provide independent document reviews and radiological data for use by the NRC in evaluating the adequacy and accuracy of the licensee's final radiological survey report relative to established guidelines.

DOCUMENT/DATA REVIEW

ESSAP reviewed the licensee's radiological survey data (BYU 1994, 1995a, 1995b, and 1995c). Procedures and methods utilized by the licensee were reviewed for adequacy and appropriateness. The data were reviewed for accuracy, completeness, and compliance with guidelines. Comment letters, documenting these reviews, were submitted to the NRC on July 25, 1994; August 7, 1995; and, November 17, 1995(ORISE 1994, 1995a, and 1995b).

PROCEDURES

During the period of April 10 and 11, 1996 ESSAP performed a radiological survey of the Nuclear Laboratory and remaining L-77 Research Reactor components. The survey was performed in accordance with a site-specific survey plan which was submitted to and approved by the NRC Region IV Office (ORISE 1996). Survey activities included a visual inspection and independent measurements and sampling of the reactor facility and the reactor components to be released for unrestricted use. This report summarizes the procedures and results of the survey.

Survey Procedures

The following procedures apply to the reactor components and to the Reactor Room, Control Room, and Accelerator Room within the L-77 Research Reactor facility.

Reference Grid

The reference grid system established by the licensee was used where possible. Measurement locations on ungridded surfaces were referenced to prominent building features or the existing grid. Measurement and sampling locations on reactor components were referenced to reactor component nomenclature and identification numbers provided by the licensee.

Surface Scans

Surface scans for alpha, beta, and gamma activity were performed on the floor and lower wall surfaces in the Reactor Room, the Accelerator Room, and the Control Room, using large-area and hand-held gas proportional, ZnS scintillation, and NaI scintillation detectors coupled to ratemeters or ratemeter-scalers with audible indicators. A 75% scan of the floor and 50% scan of the lower wall surfaces was performed in each surveyed room. Scans were also performed over 50 to 100% of the reactor component surfaces.

Surface Activity Measurements

Background measurements of surface activity on poured concrete, concrete blocks, and sheet metal were performed at building locations not having a history of radioactive materials use.

Direct measurements for total alpha and total beta activity were performed at 47 locations on the floor, walls, and overhead surfaces. An additional 14 direct measurements were performed on reactor components. One smear for the detection of removable activity was collected at each direct measurement location, with the exception of lead shot used for shielding, where no smear was taken.

These measurements were performed using gas proportional and ZnS scintillation detectors coupled to ratemeter-scalers. Measurement locations are shown on Figures 2 through 4.

BYU stored the lead shot used for shielding in 55 gallon drums labeled as LDP 1&2, LDP 3&4, etc. A lead shot sample from three of the drums was collected from the surface at the top of the drum and the total beta activity determined. From drum LDP 1&2, a second sample, at 8 inches from the top surface, was also collected and the total beta activity was determined.

Exposure Rate Measurements

Background exposure rates were determined for the building interior at 3 locations of similar construction but without a history of radioactive materials use. Facility exposure rates were measured at 5 locations (Figure 1). Measurements were performed at 1 m above the surface using a microrem meter.

Sample Analysis And Data Interpretation

Samples and data were returned to ORISE's ESSAP laboratory in Oak Ridge, Tennessee for analysis and interpretation. Smears were analyzed for gross alpha and gross beta activity using a low-background gas proportional counter. Smear results and direct measurements for surface activity were converted to units of disintegrations per minute per 100 cm^2 (dpm/100 cm^2). Exposure rates were reported in units of microroentgens per hour (μR/h). The data generated were compared to NRC guidelines established for unrestricted use (NRC 1974).

FINDINGS AND RESULTS

SURFACE SCANS

Surface scans did not identify any areas of elevated direct radiation throughout the L-77 Research Reactor facility or on any of the reactor component surfaces.

SURFACE ACTIVITY LEVELS

Survey activity measurements and smear results are presented in Table 1. All total alpha activity measurements were below the minimum detectable concentration (MDC) of 59 dpm/100 cm^2. Total beta activity ranged from less than 290 to 510 dpm/100 cm^2. Removable activity at all locations was less than the minimum detectable concentrations of 10 and 14 dpm/100 cm^2 for gross alpha and gross beta, respectively.

EXPOSURE RATES

Exposure rates performed at one meter above the surface at the three background and five site locations are presented in Table 2. Background exposure rates ranged from 8 to 9 μR/h with an average of 9 μR/h. Exposure rates inside the reactor facility ranged from 8 to 10 μR/h.

COMPARISON OF RESULTS WITH GUIDELINES

Due to historical records which indicate the presence of beta-gamma emitters, strontium-90, and uranium, several NRC guidelines are applicable. The most restrictive surface activity guidelines are those for Sr-90. The applicable NRC guidelines for Sr-90 surface activity levels are (NRC 1974):

<u>Total Activity</u>

1,000 β-γ dpm/100 cm^2, averaged over a 1 m^2 area

3,000 β-γ dpm/100 cm^2, maximum in a 100 cm^2 area

Removable Activity

200 β-γ dpm/100 cm^2

The applicable NRC guidelines for uranium surface activity levels are (NRC 1974):

Total Activity

5,000 α dpm/100 cm^2, averaged over a 1 m^2 area

15,000 α dpm/100 cm^2, maximum in a 100 cm^2 area

Removable Activity

1,000 α dpm/100 cm^2

Surface activity levels at all direct measurement locations were below the guidelines for both average and maximum alpha and beta activity. All removable activity levels were below the applicable guidelines, as well.

The exposure rate guideline is 5 μR/hr above background (NRC 1991). All site exposure rate measurements were within the guideline limit.

SUMMARY

On April 10 and 11, 1996, ESSAP performed a radiological survey of the L-77 Research Reactor components and the Nuclear Laboratory Building at Brigham Young University in Provo, Utah. Survey activities included surface scans, direct measurements of total and removable activity, and exposure rate measurements.

Survey results indicated that fixed and removable activity levels were less than the applicable guidelines. All exposure rates were below guidelines and were consistent with background levels. ESSAP's survey findings are consistent with the licensee's measurements and support the licensee's conclusion that the radiological conditions of the surveyed areas and reactor components satisfy the NRC guidelines for release to unrestricted use.

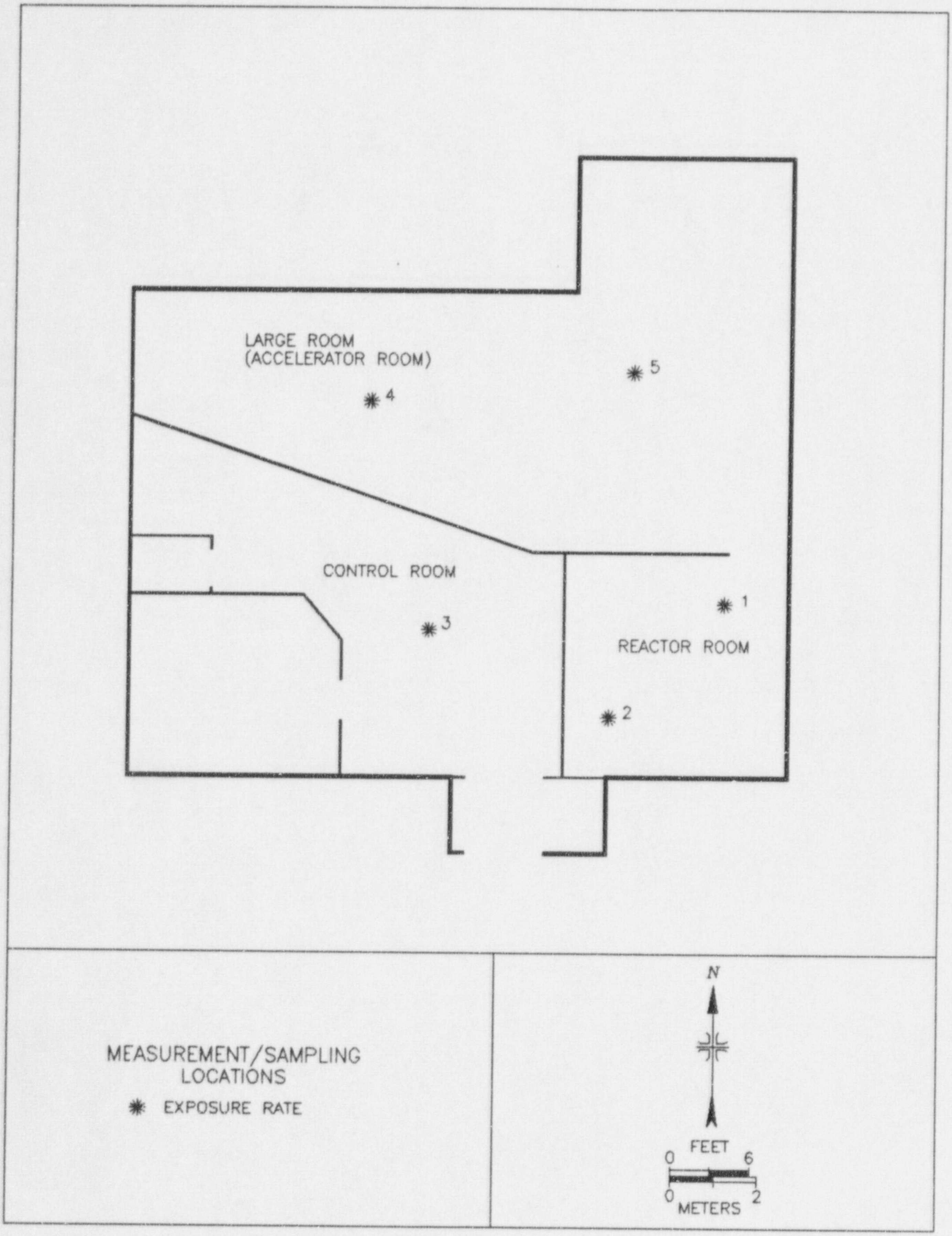

FIGURE 1: Research Reactor Facility – Exposure Rate Measurement Locations

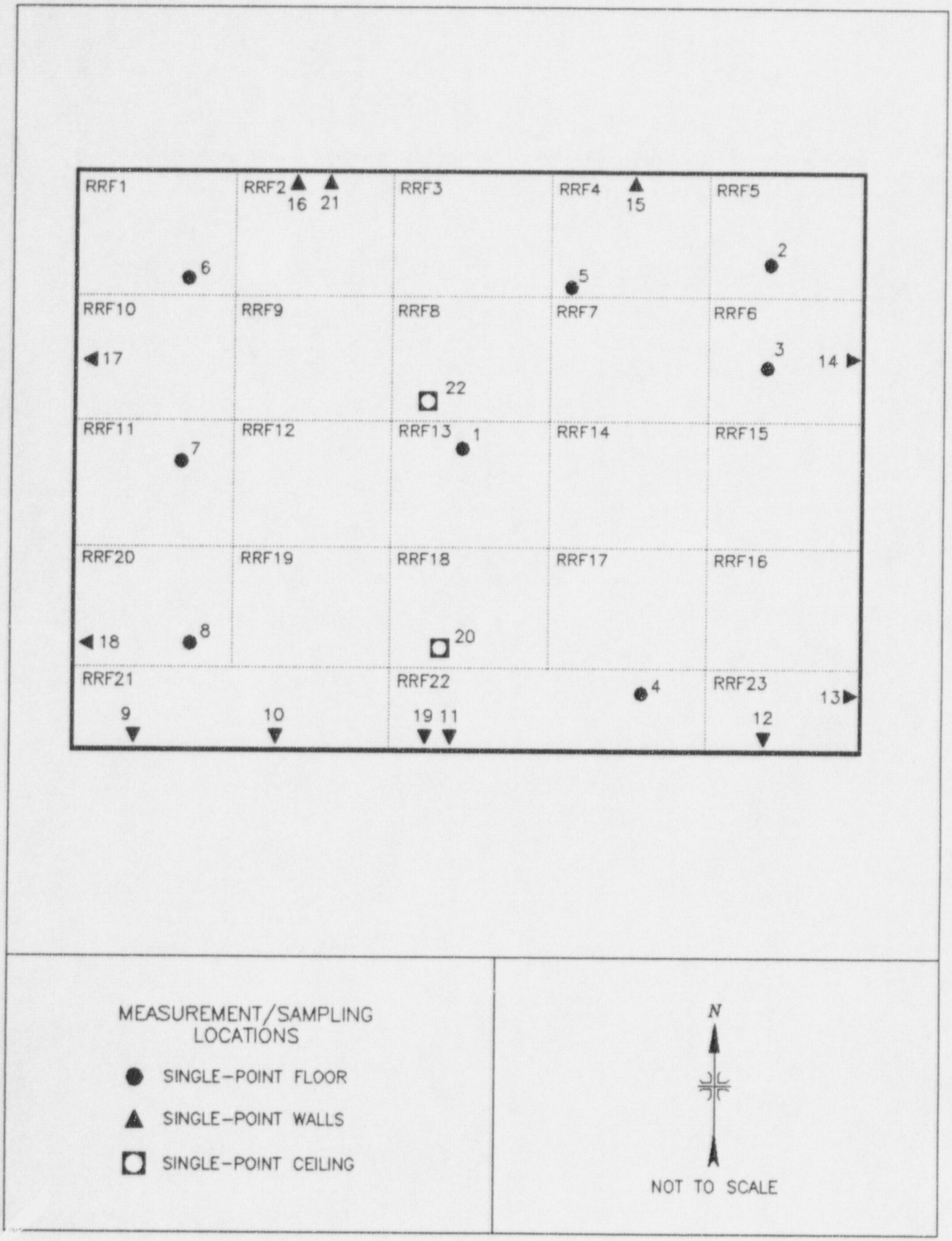

FIGURE 2: Reactor Room – Measurement and Sampling Locations

262-003(1)

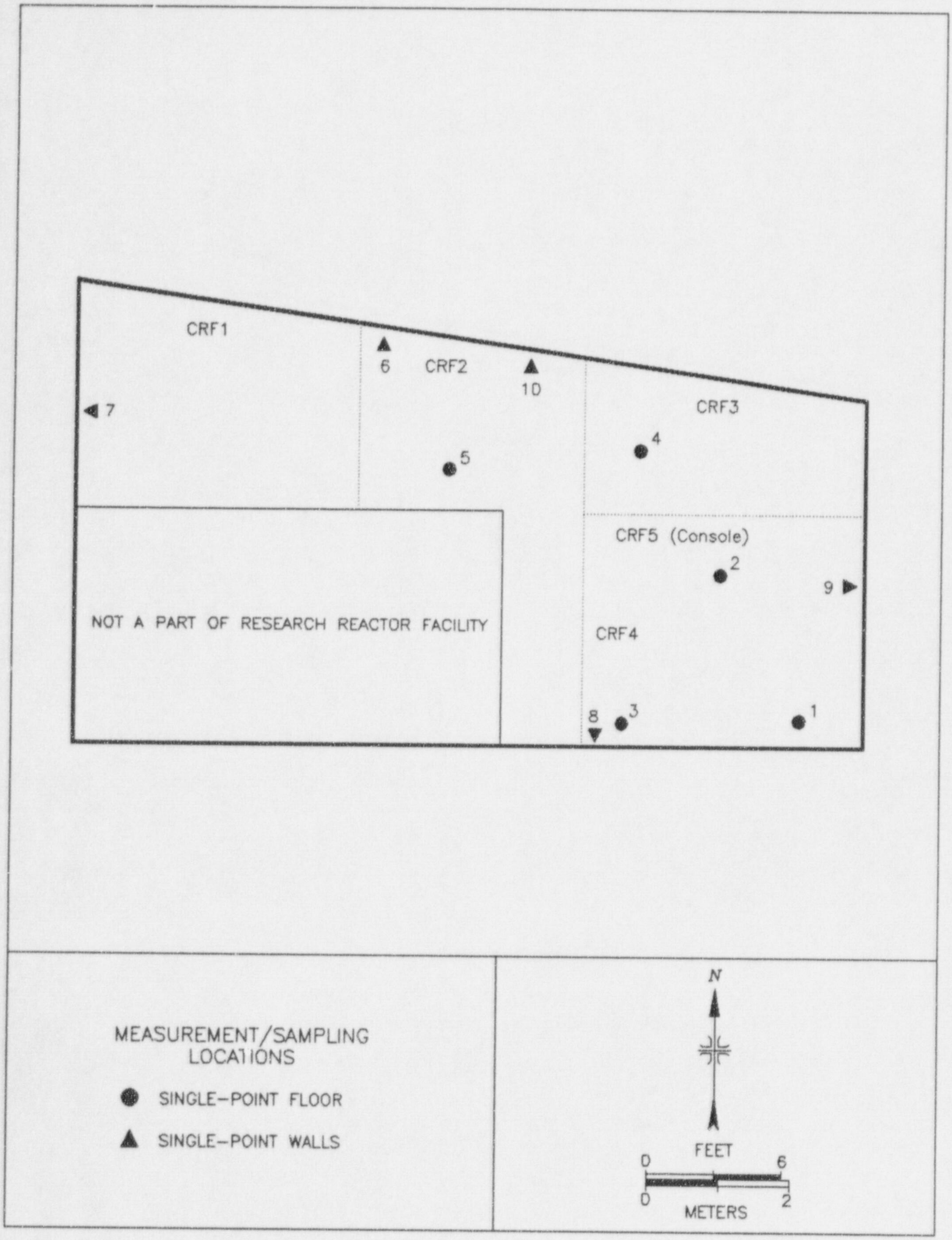

FIGURE 3: Control Room – Measurement and Sampling Locations

262-005 (1)

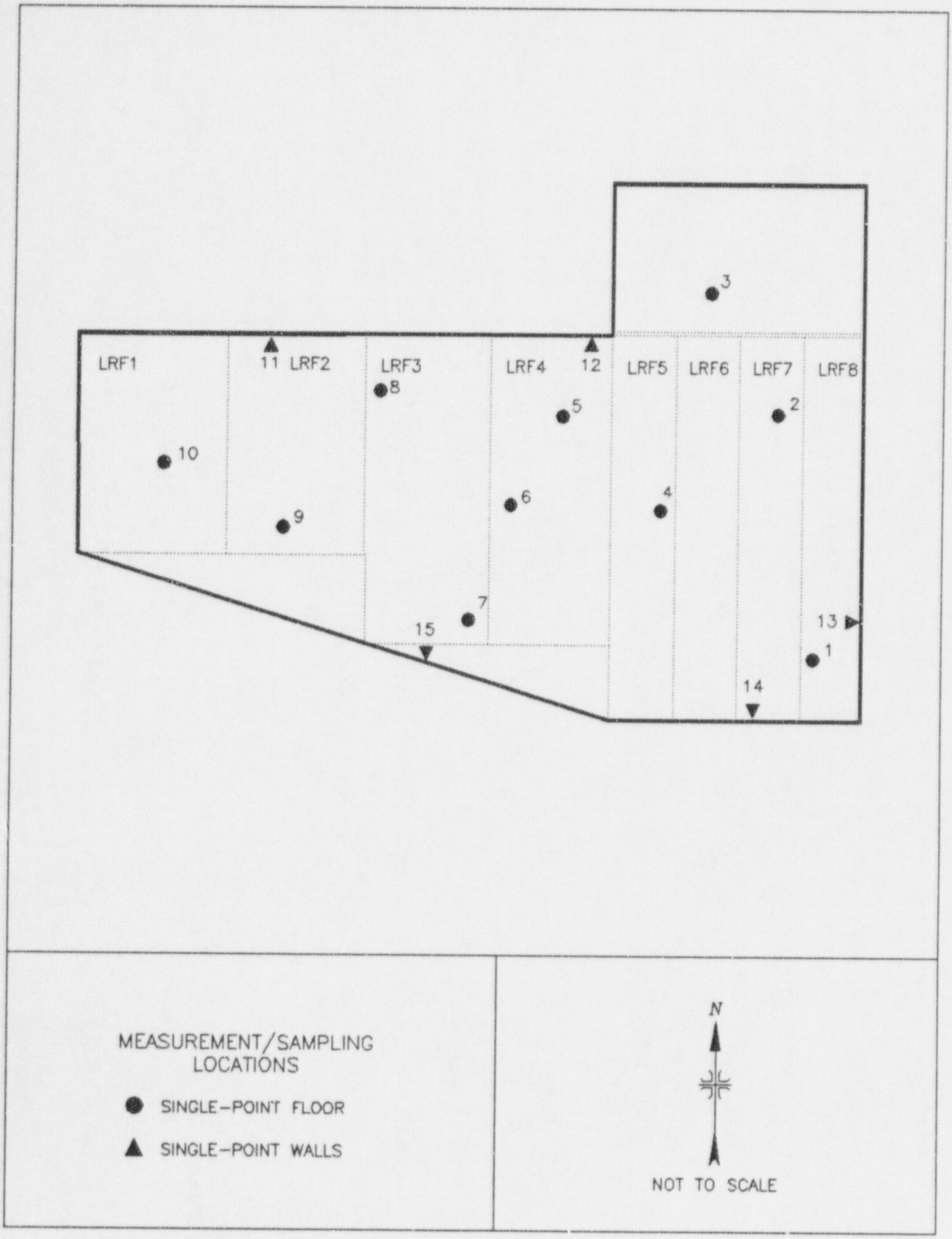

FIGURE 4: Large Room (Accelerator Room) – Measurement and Sampling Locations

TABLE 1

SUMMARY OF SURFACE ACTIVITY LEVELS FOR THE L-77 RESEARCH REACTOR BRIGHAM YOUNG UNIVERSITY PROVO, UTAH

Location/ Grid Block[a]	Total Activity (dpm/100 cm^2)		Removable Activity (dpm/100 cm^2)	
	Alpha	Beta	Alpha	Beta
Reactor Room[b]				
1/RRF 13	<59	350	<10	<14
2/RRF 5	<59	<290	<10	<14
3/RRF 6	<59	<290	<10	<14
4/RRF 22	<59	<290	<10	<14
5/RRF 4	<59	<290	<10	<14
6/RRF 1	<59	<290	<10	<14
7/RRF 11	<59	<290	<10	<14
8/RRF 20	<59	<290	<10	<14
9/RRSW 3	<59	<230	<10	<14
10/RRSW 5	<59	<290	<10	<14
11/RRSW 8	<59	<290	<10	<14
10/RRSW 14	<59	<290	<10	<14
13/RREW 10	<59	<290	<10	<14
14/RREW 2	<59	<290	<10	<14
15/RRNW 9	<59	<290	<10	<14
15/RRNW 2	<59	<290	<10	<14
17/RRWW 1	<59	<230	<10	<14
18/RRWW 5	<59	<230	<10	<14
19/RRSW 7	<59	240	<10	<14

TABLE 1 (Continued)

SUMMARY OF SURFACE ACTIVITY LEVELS
FOR THE
L-77 RESEARCH REACTOR
BRIGHAM YOUNG UNIVERSITY
PROVO, UTAH

Location/ Grid Block	Total Activity (dpm/100 cm^2)		Removable Activity (dpm/100 cm^2)	
	Alpha	Beta	Alpha	Beta
20/RRC 18	<59	<270	<10	<14
21/RRNW 2	<59	450	<10	<14
22/RRC 8	<59	<270	<10	<14
Control Room[c]				
1/CRF 4	<59	<290	<10	<14
2/CRF 5	<59	<290	<10	<14
3/CRF 4	<59	<290	<10	<14
4/CRF 3	<59	<290	<10	<14
5/CRF 2	<59	<290	<10	<14
6/CRNW 3	<59	<290	<10	<14
7/IWB 7	<59	<290	<10	<14
8/CRSW 10	<59	<290	<10	<14
9/CREW 2	<59	420	<10	<14
10/CRNW 2	<59	510	<10	<14
Accelerator (Large) Room[d]				
1/LRF 8	<28	<290	<10	<14
2/LRF 7	<28	<290	<10	<14
3/LRF	<28	<290	<10	<14
4/LRF 5	<28	<290	<10	<14
5/LRF 4	<28	<290	<10	<14
6/LRF 4	<28	<290	<10	<14

TABLE 1 (Continued)

SUMMARY OF SURFACE ACTIVITY LEVELS FOR THE L-77 RESEARCH REACTOR BRIGHAM YOUNG UNIVERSITY PROVO, UTAH

Location/ Grid Block	Total Activity (dpm/100 cm^2)		Removable Activity (dpm/100 cm^2)	
	Alpha	Beta	Alpha	Beta
7/LRF 3	<28	<290	<10	<14
8/LRF 3	<28	<290	<10	<14
9/LRF 2	<28	<290	<10	<14
10/LRF 1	<28	<290	<10	<14
11/LRW 4	<28	<290	<10	<14
10/LRW 5	<28	<290	<10	<14
13/LRW 9	<28	<290	<10	<14
14/LRT 10	<28	<290	<10	<14
15/LRW 11	<28	<290	<10	<14
Reactor Components[e]				
MB 41	<28	<270	<10	<14
SS 17	<28	<270	<10	<14
A 2	<28	<270	<10	<14
MB 18	<28	<270	<10	<14
M 26	<28	<270	<10	<14
MB 46	<28	<270	<10	<14
A 10	<28	<270	<10	<14
MB 50	<28	<270	<10	<14
SS 5	<28	<270	<10	<14
HEPA Filter	NA	<270	NA	NA

TABLE 1 (Continued)

SUMMARY OF SURFACE ACTIVITY LEVELS FOR THE L-77 RESEARCH REACTOR BRIGHAM YOUNG UNIVERSITY PROVO, UTAH

Location/ Grid Block	Total Activity (dpm/100 cm^2)		Removable Activity (dpm/100 cm^2)	
	Alpha	Beta	Alpha	Beta
Lead Diphenyl Shot[f]				
LDP 1&2	NA[g]	<270	NA	NA
LDP 1&2, 8"	NA	<270	NA	NA
LDP 3&4	NA	<270	NA	NA
LDP 5&6	NA	<270	NA	NA

[a]The grid identification numbers, as identified by BYU are as follows: RR = Reactor Room; CR = Control Room; LR = Large Room; F = floor; SW = south wall; EW = east wall; NW = north wall; WW = west wall; C = ceiling; MP = metal peices from reactor that were scanned only; SS = stainless steel pieces from reactor; A = aluminum pieces from reactor; MB = mild steel beta measurement; IWB = internal wall bathroom; and LRT = Large Room Top.

[b]Refer to Figure 2.

[c]Refer to Figure 3.

[d]Refer to Figure 4.

[e]The L-77 research reactor was dismantled by BYU; the identification numbers are those placed on each individual piece by BYU.

[f]LDP is the lead diphenyl shot used as shielding in the reactor. BYU had the lead shot stored in 55 gallon drums labeled as LDP 1&2, LDP 3&4, etc. A lead shot sample from the drum was collected from the surface of the drum and the total beta activity determined. From drum LDP 1&2, a second sample, at 8 inches was also collected and the total beta activity was determined.

[g]NA = Not Applicable.

TABLE 2

EXPOSURE RATES
FOR THE
L-77 RESEARCH REACTOR
BRIGHAM YOUNG UNIVERSITY
PROVO, UTAH

Location	Exposure Rate at 1 m (μR/h)
Reactor Facility[a]	
1 - Reactor Room	10
2 - Reactor Room	10
3 - Control Room	9
4 - Accelerator Room	9
5 - Accelerator Room	8
Background[b]	
Cluff Building, Southwest Foyer	9
Cluff Building, Southeast Foyer	8
Cluff Building, Northeast Foyer	9

[a]Refer to Figure 1.
[b]Figure not provided.

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APPENDIX A

MAJOR INSTRUMENTATION

APPENDIX A

MAJOR INSTRUMENTATION

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

DIRECT RADIATION MEASUREMENT

Instruments

Bicron Micro-Rem Meter
(Bicron Corporation, Newburg, OH)

Eberline Pulse Ratemeter
Model PRM-6
(Eberline, Santa Fe, NM)

Ludlum Floor Monitor
Model 239-1
(Ludlum Measurements, Inc.,
Sweetwater, TX)

Ludlum Ratemeter-Scaler
Model 2221
(Ludlum Measurements, Inc.,
Sweetwater, TX)

Detectors

Eberline ZnS Scintillation Detector
Model AC-3-7
Effective Area, 74 cm^2
(Eberline, Santa Fe, NM)

Ludlum Gas Proportional Detector
Model 43-37
Effective Area, 550 cm^2
(Ludlum Measurements, Inc.,
Sweetwater, TX)

Ludlum Gas Proportional Detector
Model 43-68
Effective Area, 126 cm^2
(Ludlum Measurements, Inc.,
Sweetwater, TX)

Victoreen NaI Scintillation Detector
Model 489-55
3.2 cm x 3.8 cm Crystal
(Victoreen, Cleveland, OH)

LABORATORY ANALYTICAL INSTRUMENTATION

Low Background Gas Proportional Counter
Model LB-5100-W
(Oxford, Oak Ridge, TN)

APPENDIX B

SURVEY AND ANALYTICAL PROCEDURES

APPENDIX B
SURVEY AND ANALYTICAL PROCEDURES

SURVEY PROCEDURES

Surface Scans

Surface scans 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. A large surface area, gas proportional floor monitor was used to scan the floors of the surveyed areas. Other surfaces were scanned using small area (74 cm^2 or 126 cm^2) hand-held detectors. Identification of elevated levels was based on increases in the audible signal from the recording and/or indicating instrument. Combinations of detectors and instruments used for the scans were:

Alpha	—	gas proportional detector with ratemeter-scaler
	—	ZnS scintillation detector with ratemeter-scaler
Beta	—	gas proportional detector with ratemeter-scaler
Gamma	—	NaI scintillation detector with ratemeter

Surface Activity Measurements

Measurements of total alpha and total beta activity levels were performed using ZnS scintillation and gas proportional detectors with ratemeters-scalers.

Count rates (cpm), which were integrated over 1 minute in a static position, were converted to activity levels (dpm/100 cm^2) by dividing the net rate by the 4π efficiency and correcting for the active area of the detector. Because different materials (poured concrete and concrete block, brick, metal, and wood) may have very different background levels, average background counts were determined for each material encountered during the survey activities--on a similar material having no known radiological history at a location separate from the surveyed area. The alpha activity background countrate for the ZnS scintillation detector and the gas proportional detector averaged 1 cpm for each detector. The alpha efficiency factor was 0.17 for the ZnS scintillation detector and 0.21 for the gas proportional detector. Both detector types were calibrated with a Th-230 calibration source. The beta activity background count rate for the gas proportional detector averaged 337 cpm for concrete and brick surfaces, 287 cpm for metal surfaces, and 206 cpm for wood surfaces. The beta efficiency factor was 0.24 for the gas proportional detector which was calibrated with a Tc-99 calibration source. The alpha minimum detectable concentration (MDC) was 59 dpm/100 cm^2 for the ZnS scintillation detector and 28 dpm/100 cm^2 for the gas proportional detector. The beta activity MDCs for the gas proportional detector were 290 dpm/100 cm^2 for concrete and brick surfaces, 270 dpm/100 cm^2 for metal surfaces, and 230 dpm/100 cm^2 for wood surfaces. The effective windows for the ZnS scintillation and gas proportional detectors were 74 cm^2 and 126 cm^2, respectively.

Removable Activity Measurements

Removable activity levels were determined using numbered filter paper disks, 47 mm in diameter. Moderate pressure was applied to the smear and approximately 100 cm^2 of the surface was wiped. Smears were placed in labeled envelopes with the location and other pertinent information recorded.

Exposure Rate Measurements

Measurements of dose equivalent rates (μrem/h) were performed at 1 m above the surface using a Bicron microrem meter. Although the instrument displays data in μrem/h, the μrem/h to μR/h conversion is essentially unity.

ANALYTICAL PROCEDURES

Removable Activity

Smears were counted on a low background gas proportional system for gross alpha and gross beta activity.

UNCERTAINTIES AND DETECTION LIMITS

Detection limits, referred to as minimum detectable concentrations (MDC), were based on 2.71 plus 4.65 times the standard deviation of the background count [$2.71 + (4.65\sqrt{BKG})$]. When the activity was determined to be less than the MDC of the measurement procedure, the result was reported as less than MDC. Because of variations in background levels and measurement efficiencies, the detection limits differ from instrument to instrument.

CALIBRATION AND QUALITY ASSURANCE

Calibration of all field and laboratory instrumentation was based on standards/sources, traceable to NIST, when such standards/sources were available. In cases where they were not available, standards of an industry recognized organization was used. Calibration of the Bicron microrem meter was performed by an in-house instrument specialist.

Analytical and field survey activities were conducted in accordance with procedures from the following ESSAP documents:

- Survey Procedures Manual, Revision 9 (April 1995)
- Laboratory Procedures Manual, Revision 9 (January 1995)
- Quality Assurance Manual, Revision 7 (January 1995)

The procedures contained in these manuals were developed to meet the requirements of DOE Order 5700.6C and ASME NQA-1 for Quality Assurance and contain measures to assess processes during their performance.

Quality control procedures include:

- Daily instrument background and check-source measurements—at the beginning, during the middle, and at the end of each day—are performed to confirm that equipment operation is within acceptable statistical fluctuations.
- Participation in EPA and EML laboratory Quality Assurance Programs.
- Training and certification of all individuals performing procedures.
- Periodic internal and external audits.

APPENDIX C

REGULATORY GUIDE 1.86, TERMINATION OF OPERATING LICENSES FOR NUCLEAR REACTORS

U.S. ATOMIC ENERGY COMMISSION

June 1974

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 from the reactor and placing it in storage (either onsite or 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 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 off-site 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 1).

The hazard associated with the returned 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 1 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 1) 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 must 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 of 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 1.

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 permissible 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 1. 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 1, the Commission may terminate the license. If 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 1

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	1,000 dpm/100 cm^2	3,000 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.	5,000 dpm $\beta\gamma$/100 cm^2	15,000 dpm $\beta\gamma$/100 cm^2	1,000 dpm $\beta\gamma$/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.