50-267

CONFIRMATORY SURVEY FOR THE REPOWER AREA FORT ST. VRAIN PLATTEVILLE, COLORADO

[DOCKET NO. 50-267]

E.W. ABELQUIST

Prepared for the
Division of Waste Management
Headquarters Office
U.S. Nuclear Regulatory Commission

ORISE

Environmental Survey and Site Assessment Program
Energy/Environment Systems 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 88 colleges and universities.

NOTICES

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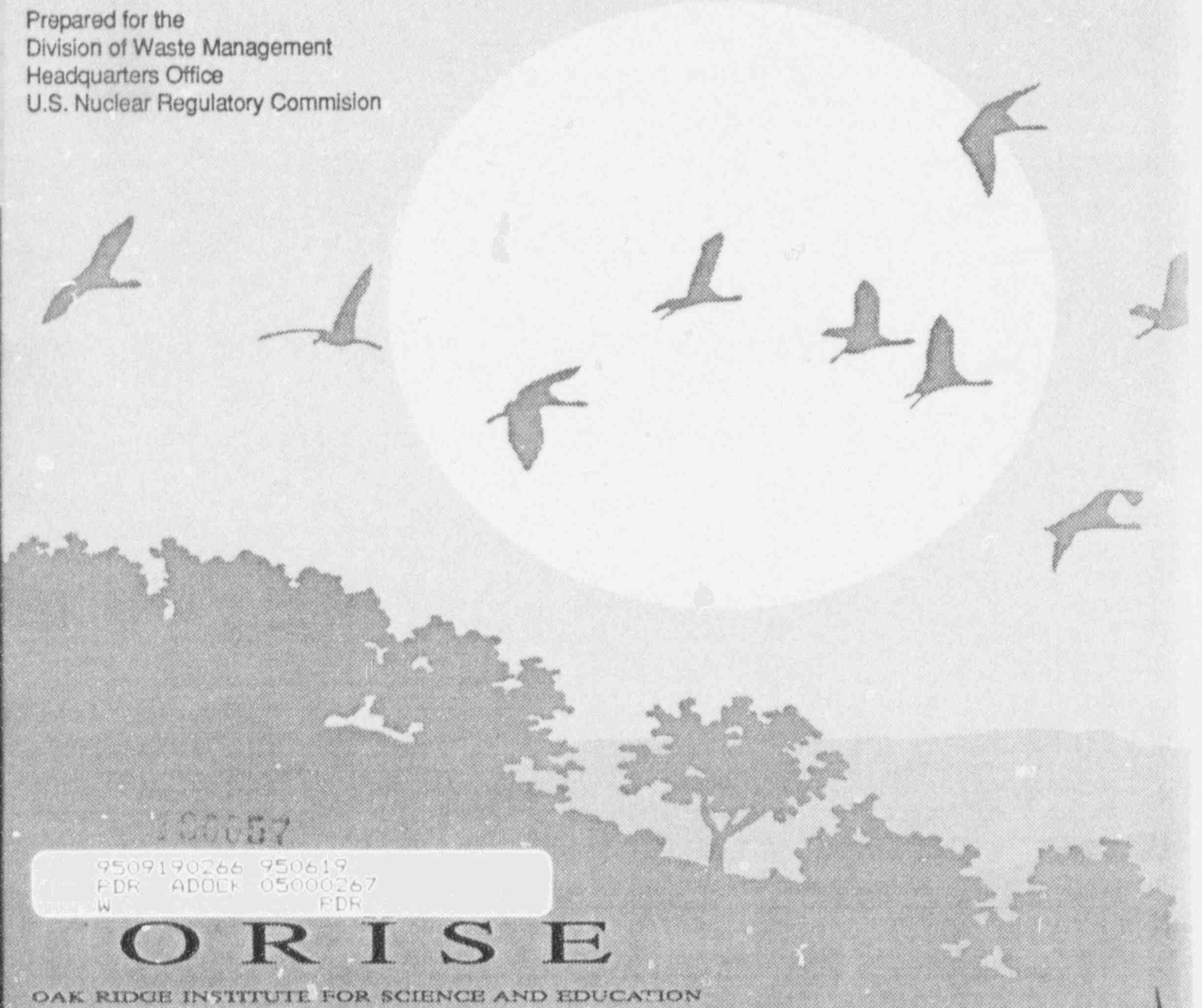

ORISE

OAK RIDGE INSTITUTE FOR SCIENCE AND EDUCATION

Environmental Survey and Site Assessment Program
Energy/Environment Systems Division

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ORISE 95/F-80

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Prepared by

E. W. Abelquist

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Division of Waste Management
Headquarters Office
U.S. Nuclear Regulatory Commission

FINAL REPORT

JUNE 1995

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CONFIRMATORY SURVEY
FOR THE
REPOWER AREA
FORT ST. VRAIN
PLATTEVILLE, COLORADO

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ABBREVIATIONS AND ACRONYMS

$\mu R/h$	microroentgens per hour
ASME	American Society of Mechanical Engineers
cm	centimeter
cm^2	square centimeter
cpm	counts per minute
DOE	Department of Energy
dpm/100 cm^2	disintegrations per minute/100 square centimeters
EML	Environmental Measurement Laboratory
EPA	Environmental Protection Agency
ESSAP	Environmental Survey and Site Assessment Program
FSV	Fort St. Vrain
HTGR	High Temperature Gas-Cooled Reactor
kg	kilogram
m	meter
m^2	square meter
mm	millimeter
MeV	million electron volts
mrem	millirem
MWe	Megawatts electric
NaI	sodium iodide
NIST	National Institute of Standards and Technology
NMSS	Office of Nuclear Material Safety and Safeguards
NRC	Nuclear Regulatory Commission
ORAU	Oak Ridge Associated Universities
ORISE	Oak Ridge Institute for Science and Education
pCi/g	picocuries per gram
PSC	Public Service Company of Colorado
TEDE	total effective dose equivalent

CONFIRMATORY SURVEY
FOR THE
REPOWER AREA
FORT ST. VRAIN
PLATTEVILLE, COLORADO

INTRODUCTION AND SITE HISTORY

Fort St. Vrain (FSV) was a 330 MWe High Temperature Gas-Cooled Reactor (HTGR) owned and operated by Public Service Company (PSC) of Colorado. The site consists of 6995 hectares (2798 acres) owned by PSC, of which approximately one square mile was designated as the exclusion area during plant operation. The licensee maintained complete control over this area. The basic installation included a reactor building, turbine building, cooling towers, and an electrical switchyard.

FSV was permanently shutdown in August 1989, with the decision to decommission the facility made during December 1989. On November 23, 1992, the Nuclear Regulatory Commission (NRC) issued the *Order to Authorize Decommissioning of Fort St. Vrain and Amendment No. 85 to Possession Only License No.* DPR-34.[1] During the period 1989 to 1991, a radiological characterization of the FSV site was performed. Currently, the FSV decommissioning is approximately 70 % complete, with completion expected early in 1996 (excluding the final site survey).

PSC has committed to the Colorado Public Utilities Commission to resume electrical generation at FSV through the installation of gas turbines. In order to perform this project, a small section of land in the southwest area of the site has been cleared in preparation for the repower effort. This area is referred to as the repower area, where PSC plans to install natural gas-fired combustion turbines and heat recovery boilers to repower the facility.

During plant operations, pre-fabricated steel buildings were located in the repower area. These buildings accommodated a construction workshop, a quality control facility that performed radiography, a small warehouse, and a flammable storage building.

The repower area does not have a known history of radioactive contamination. This was reaffirmed by the evaluation of the characterization results for the repower area, which did not identify radioactive contamination due to licensed activities. The repower area was therefore classified as an unaffected area. However, elevated levels of Cs-137 were identified in surface soil collected from a localized area outside, but adjacent to the repower area. This area had previously been used for temporary storage of spent fuel shipping casks.

At the request of the NRC's Division of Waste Management, Headquarters' Office, the Environmental Survey and Site Assessment Program (ESSAP) of the Oak Ridge Institute for Science and Education (ORISE) performed an independent confirmatory radiological survey of the repower area at the Fort St. Vrain site in Platteville, Colorado.

SITE DESCRIPTION

The FSV facility is located approximately 56 kilometers (35 miles) north of Denver and 5.6 kilometers northwest of the town of Platteville, in Weld County, Colorado (Figure 1). The site consists of 6995 hectares owned by PSC of which approximately one square mile was designated as the exclusion area during plant operation.

The repower survey area is located within the restricted area of the FSV facility on the east side of the turbine building, north of the electrical switchyard (Figure 2). This location is approximately 12,225 square meters in size. The area has been isolated from the balance of the restricted area by a chain link fence and locking gates controlled by FSV Security. The boundaries of the repower area include portions of the original restricted area fence on the south and east sides and newly erected fence on the west and north sides (Figure 3). The boundary also includes the east, and a portion of the south exterior walls of the evaporative cooler building.

OBJECTIVES

The objectives of the confirmatory 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 procedures and final status survey results.

DOCUMENT REVIEW

ESSAP has reviewed the licensee's final survey report and radiological survey data.[1] Procedures and methods utilized by the licensee were reviewed for adequacy and appropriateness. The data were reviewed for accuracy, completeness and compliance with guidelines.

PROCEDURES

During the period March 20 through 22, 1995, ESSAP performed a confirmatory survey at the Fort St. Vrain site in Platteville, Colorado. The survey was conducted in accordance with a survey plan dated March 17, 1995, submitted to and approved by the NRC's Division of Waste Management, Office of Nuclear Material Safety and Safeguards (NMSS).[2] This report summarizes the procedures and results of the survey. Additional information concerning major instrumentation, sampling equipment, and analytical procedures is provided in Appendices A and B.

SURVEY PROCEDURES

The licensee's final status survey of the repower area included two general categories of survey units: surfaces and structures, and open land areas. The area was further divided into survey units. The surface and structure survey units within the repower area included the Valve Pit, Miscellaneous Metal Surfaces, Concrete Slab at the Security Fence, Miscellaneous Concrete Surfaces, Evaporative Cooler Building (east and south walls below 2 m), and Evaporative Cooler Building (east and south foundation walls). The open land area survey units included the general soil area within the repower area, leach field soil area, septic system, and monitoring wells.

Reference System

ESSAP selected specific measurement and sampling locations from each of the survey units. Because survey maps illustrating the measurement locations were not provided by the licensee, ESSAP requested that the licensee identify some of their measurement and sampling locations for confirmatory measurements. Measurement and sampling locations were referenced to prominent site features and recorded on survey maps.

Surface Scans

Soil surfaces were scanned for gamma radiation using NaI scintillation detectors. Approximately 10% of the soil in the general and leach field soil areas was scanned. Structure surfaces were also scanned with gas proportional detectors over the 0.5 m^2 area surrounding each direct measurement location. Particular attention was given to cracks and joints in the surfaces and walls, ledges, drains, and other locations where material may have accumulated. All detectors were coupled to ratemeters or ratemeter-scalers with audible indicators. Locations of elevated direct radiation detected by scans were marked for further investigation.

Surface Activity Measurements

Direct measurements for total beta activity were performed at 38 locations, representing each of the survey units within the repower area. Measurements were performed using gas proportional detectors, coupled to portable ratemeter-scalers. Smear samples, for determining removable activity levels, were collected from each direct measurement location. Measurement and sampling locations are shown on Figures 4 and 5.

Exposure Rate Measurements

Exposure rate measurements were performed within the general soil area of the repower area. Exposure rates were measured at 1 m above surfaces at 10 locations using a pressurized ionization chamber (PIC). Background exposure rate measurements were performed using a PIC at five locations within a 0.5 to 10 km radius of the site. Measurement locations are shown on Figures 6 and 8.

Soil Sampling

Background soil samples were collected from each of the external background exposure rate measurement locations.

A total of ten soil samples was collected randomly from the general soil area within the repower area. Soil sampling locations are shown on Figures 7 and 8.

SAMPLE ANALYSIS AND DATA INTERPRETATION

Samples and data were returned to ESSAP's laboratory in Oak Ridge, Tennessee for analysis and interpretation. Soil samples were analyzed by solid state gamma spectrometry. Spectra were reviewed for Co-60, Cs-137, and any other identifiable photopeaks. Soil sample results were reported in units of picocuries per gram (pCi/g). Smear samples were analyzed for gross alpha and gross beta activity using a low background gas proportional counter, and the results converted to dpm/100 cm^2. Direct measurements for surface activity were converted to units of disintegrations per minute per 100 square centimeters (dpm/100 cm^2). Exposure rates were reported in units of microroentgens per hour ($\mu R/h$). Results were compared with the licensee's documentation and NRC guidelines established for release to unrestricted use, which are provided in Appendix C.

FINDINGS AND RESULTS

DOCUMENT REVIEW

ESSAP reviewed the licensee's final status survey report, including the final status survey data and provided comments to the NRC.[3] The survey instrumentation and procedures used, including the assessment of background contributions to surface activity measurements, were discussed at length. Guidelines for surface contamination and exposure rates were clearly stated, however, radionuclide concentrations in soil that correspond to the 10 millirem (mrem) per year site-specific soil guideline were not specified. The site operational history, decommissioning activities, and final survey results provided sufficient information on the radiological status of the repower area.

SURVEY RESULTS

Surface Scans

Surface scans for beta activity on structure surfaces did not identify any locations of elevated direct radiation. Surface scans for gamma activity within the general soil area also did not result in the identification of any locations of elevated direct radiation.

Surface Activity Measurements

Surface activity measurements for total beta activity are summarized in Table 1. Total beta activity levels for all measurement locations ranged from <340 to 710 dpm/100 cm^2. Removable activity levels were all less than the minimum detectable activity of the procedure which was 12 dpm/100 cm^2 for gross alpha and 16 dpm/100 cm^2 for gross beta.

Exposure Rates

Site exposure rates are summarized in Table 2. Gross exposure rates in the repower area ranged from 16.3 to 25.7 μR/h, and averaged 19 μR/h, at 1 m above the surface. The net exposure rates ranged from 0 to 9.4 μR/h, and averaged 2.7 μR/h. Background exposure rates ranged from 15.4 to 16.9 μR/h, and averaged 16.3 μR/h (Table 3).

Radionuclide Concentrations in Soil Samples

Radionuclide concentrations in background samples are summarized in Table 3 and were <0.2 pCi/g for Co-60, <0.1 to 0.2 pCi/g for Cs-137, 1.4 to 1.8 pCi/g for Th-228, 1.5 to 2.2 pCi/g for Th-232, <0.1 pCi/g for U-235, and <2.1 pCi/g for U-238.

Concentrations of radionuclides in surface soil samples collected randomly from the repower area summarized in Table 4. Radionuclide concentration ranges are as follows: <0.2 pCi/g for Co-60, <0.1 pCi/g for Cs-137, 1.1 to 2.0 pCi/g for Th-228, 1.1 to 1.8 pCi/g for Th-232, <0.1 pCi/g for U-235, and <2.3 pCi/g for U-238.

COMPARISON OF RESULTS WITH GUIDELINES

The primary contaminants of concern for this site are beta-gamma emitters resulting from the operation of the FSV facility. The applicable NRC guidelines for beta-gamma emitters in unaffected areas are provided in Regulatory Guide 1.86.[4] The guidelines are:

Total Activity

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

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

Removable Activity

1,000 dpm/100 cm^2

Surface activity measurements for total and removable activity were all within the surface contamination guidelines.

The guideline values for radionuclide concentrations in soil are the radionuclide-specific concentrations which could result in an average annual total effective dose equivalent (TEDE) of 10 mrem to an individual in a population group exposed to radioactive material following decommissioning. These values may be determined in accordance with the methodology contained in NUREG/CR-5512, Volume 1 and as presented in NUREG-1500.[5,6] Concentrations of radionuclides in soil samples are comparable to the concentrations measured in background samples (Tables 3 and 4). Therefore, compliance is demonstrated by the fact that soil samples collected from the repower area are indistinguishable from background levels.

The guideline for exposure rates, measured at 1 m above the surface, is 5 μR/h above background.[7] With the exception of one elevated exposure rates measured in the northwest corner of the repower area (Figure 6, #5), all exposure rate were within the guideline. This elevated exposure rate measurement was due to dismantlement activities to remove the core support floor from the reactor vessel and to place it in a segmenting area on the refueling floor of the reactor building. Because the segmenting area is only separated from the repower area by sheet metal walls, the core support floor has temporarily affected exposure rates in the repower area, most notably in areas closest to the reactor building.[8]

The licensee performed exposure rate measurements before and after the core support floor move. Licensee exposure rates prior to the core support floor move were consistent with their background measurements of exposure rate, and thus indicated compliance with the exposure rate guideline.[8] Their results indicate that the average exposure rate in the repower area increased by about 3.5 μR/h following the core support floor move. An instrument comparison between ESSAP and the licensee performed during the confirmatory survey indicated good agreement between exposure rate measurements in the repower area (Table 2). Specifically, based on a pair-wise comparison *t*-test, there are no statistically significant differences ($p > 0.2$) between ESSAP's and the licensee's corrected exposure rate data.

Furthermore, soil sampling in the areas affected by these increased exposure rates resulted in no indication of soil activity in excess of background levels.

SUMMARY

During the period March 20 through 22, 1995, at the request of the NRC's Division of Waste Management, NMSS, the Environmental Survey and Site Assessment Program of ORISE performed a confirmatory survey at the Fort St. Vrain site in Platteville, Colorado. Survey activities included document reviews, surface scans, surface activity measurements, exposure rate measurements, and soil sampling.

The confirmatory survey identified one location within the repower area that exhibited an elevated exposure rate measurement. This location within the repower area was influenced by elevated radiation from dismantling activities on the core support floor in the repower area. Soil sampling in this area confirmed that the elevated exposure rate measurement was not the result of elevated soil concentrations. The confirmatory survey results are consistent with those obtained by the licensee and support the licensee's conclusion that residual activity levels in the repower area satisfy the guidelines for release to unrestricted use.

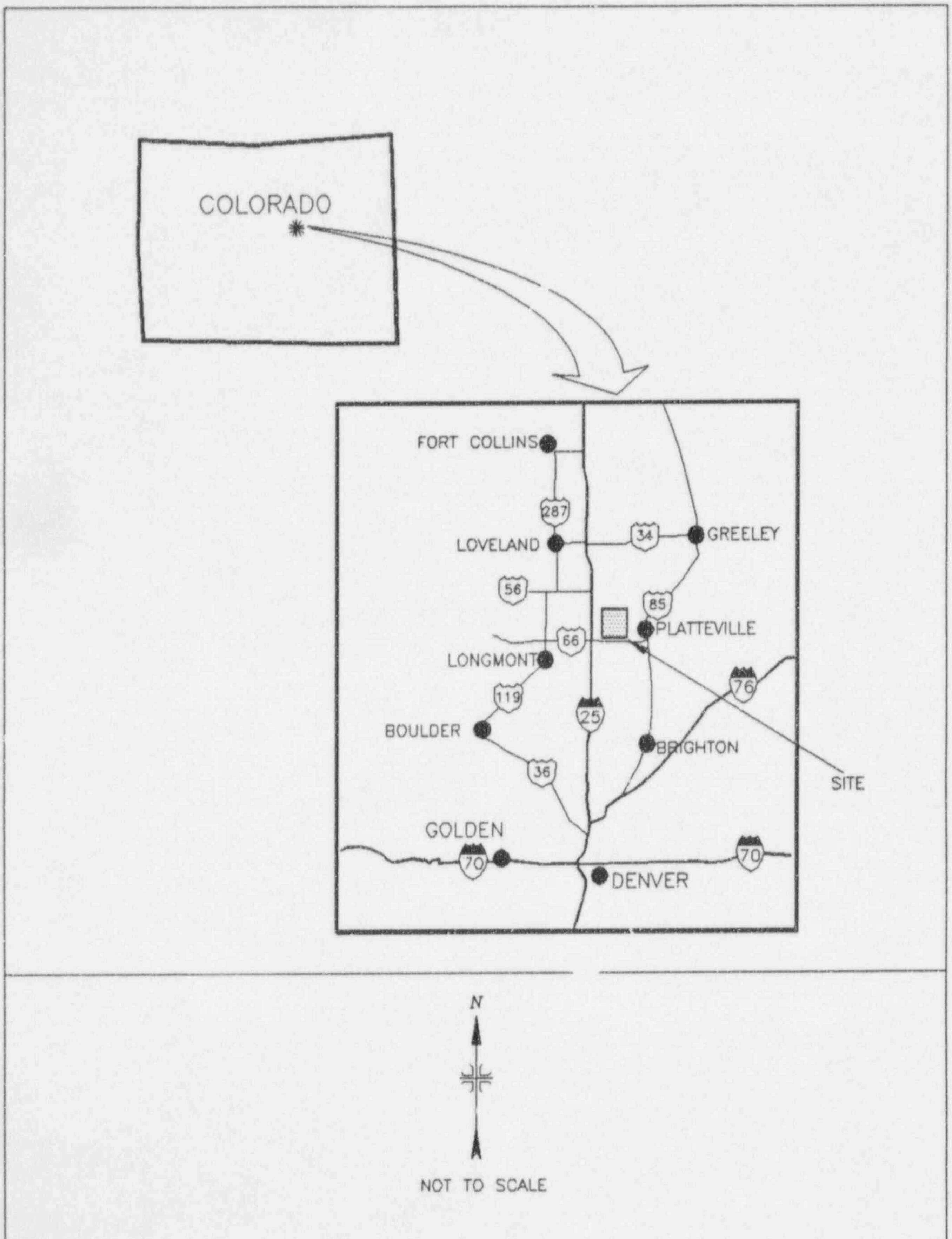

FIGURE 1: Location of the Fort St. Vrain Site – Platteville, Colorado

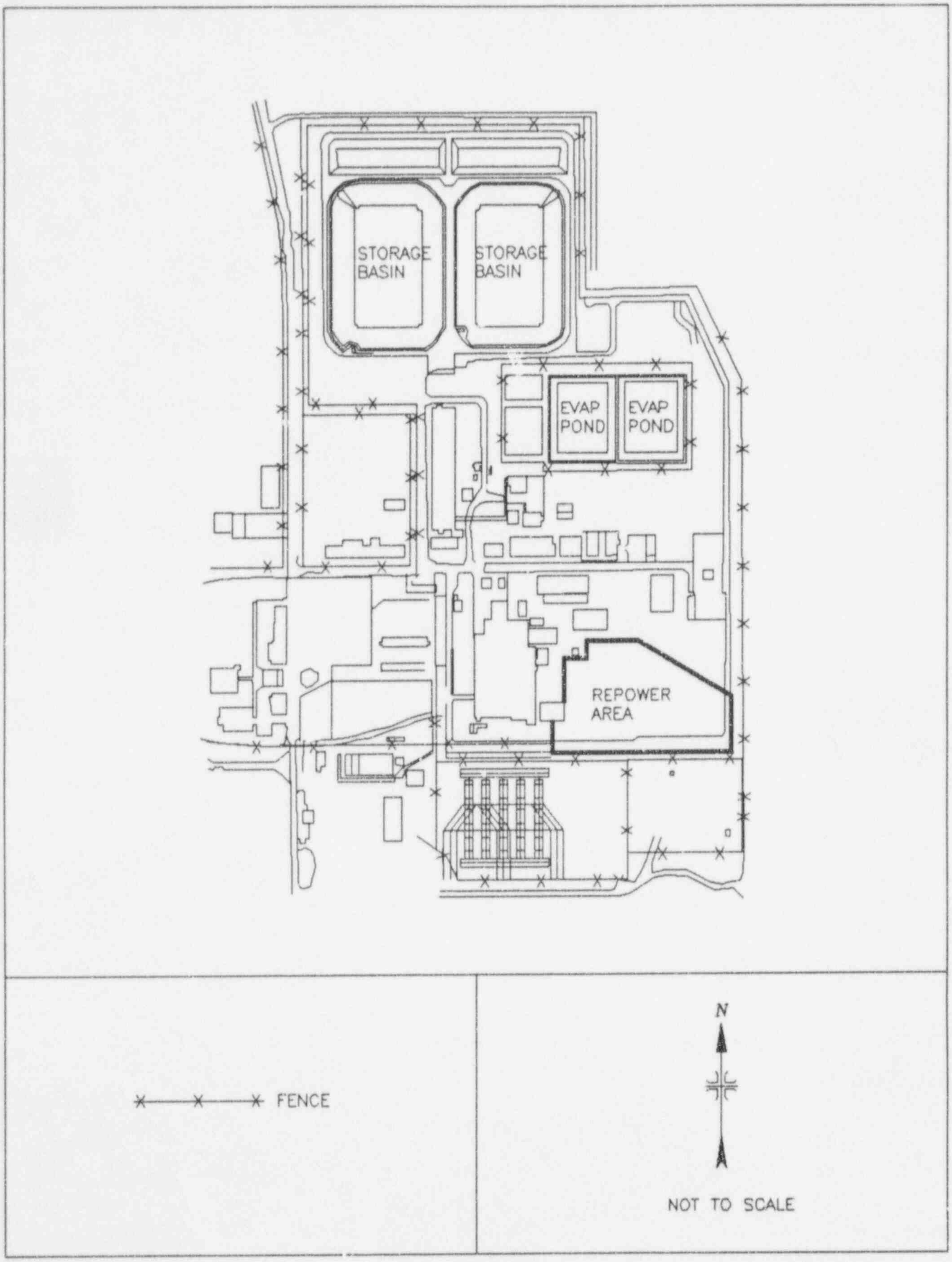

FIGURE 2: Plot Plan of the Fort St. Vrain Nuclear Station

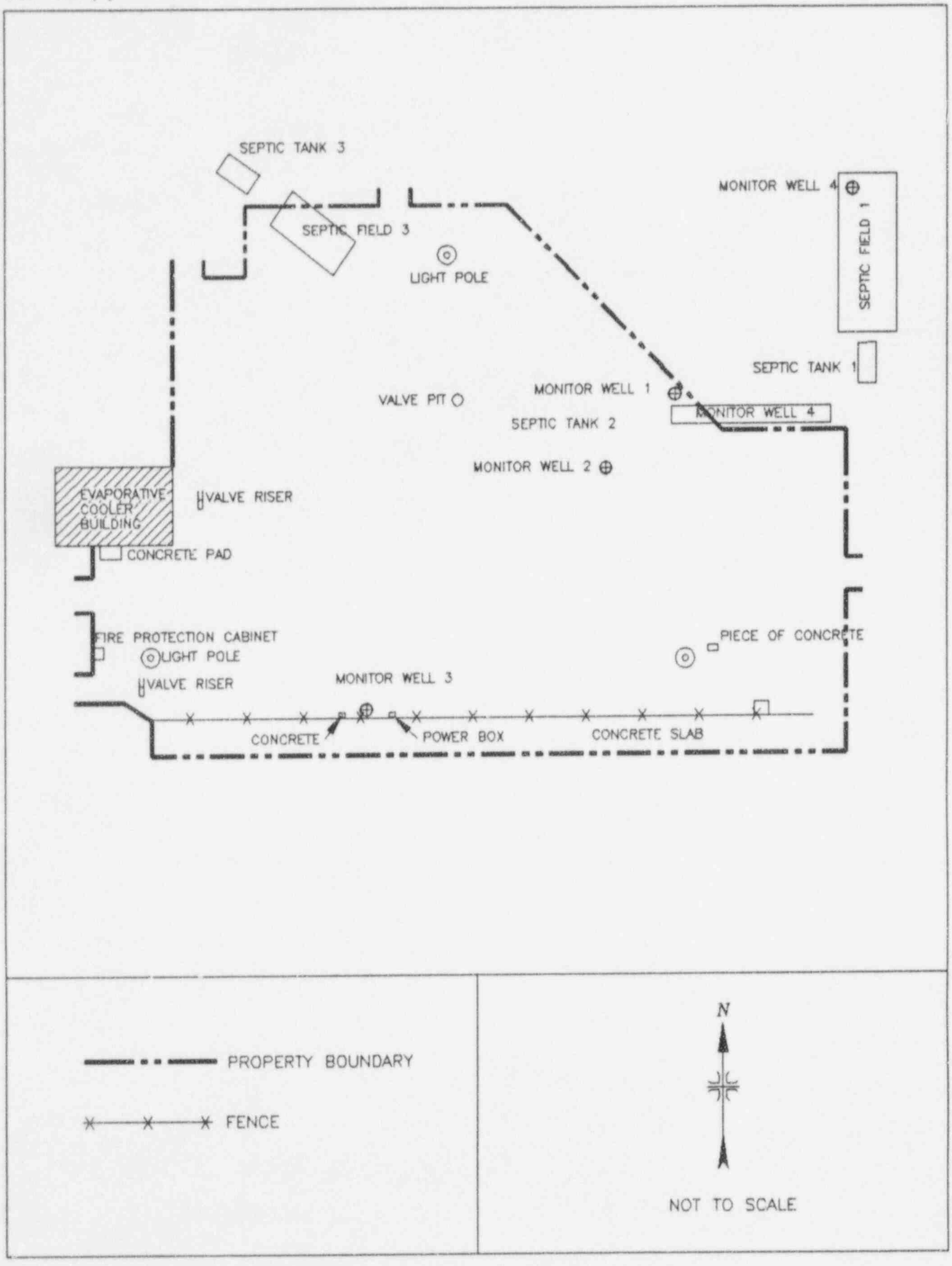

FIGURE 3: Fort St. Vrain – Repower Area

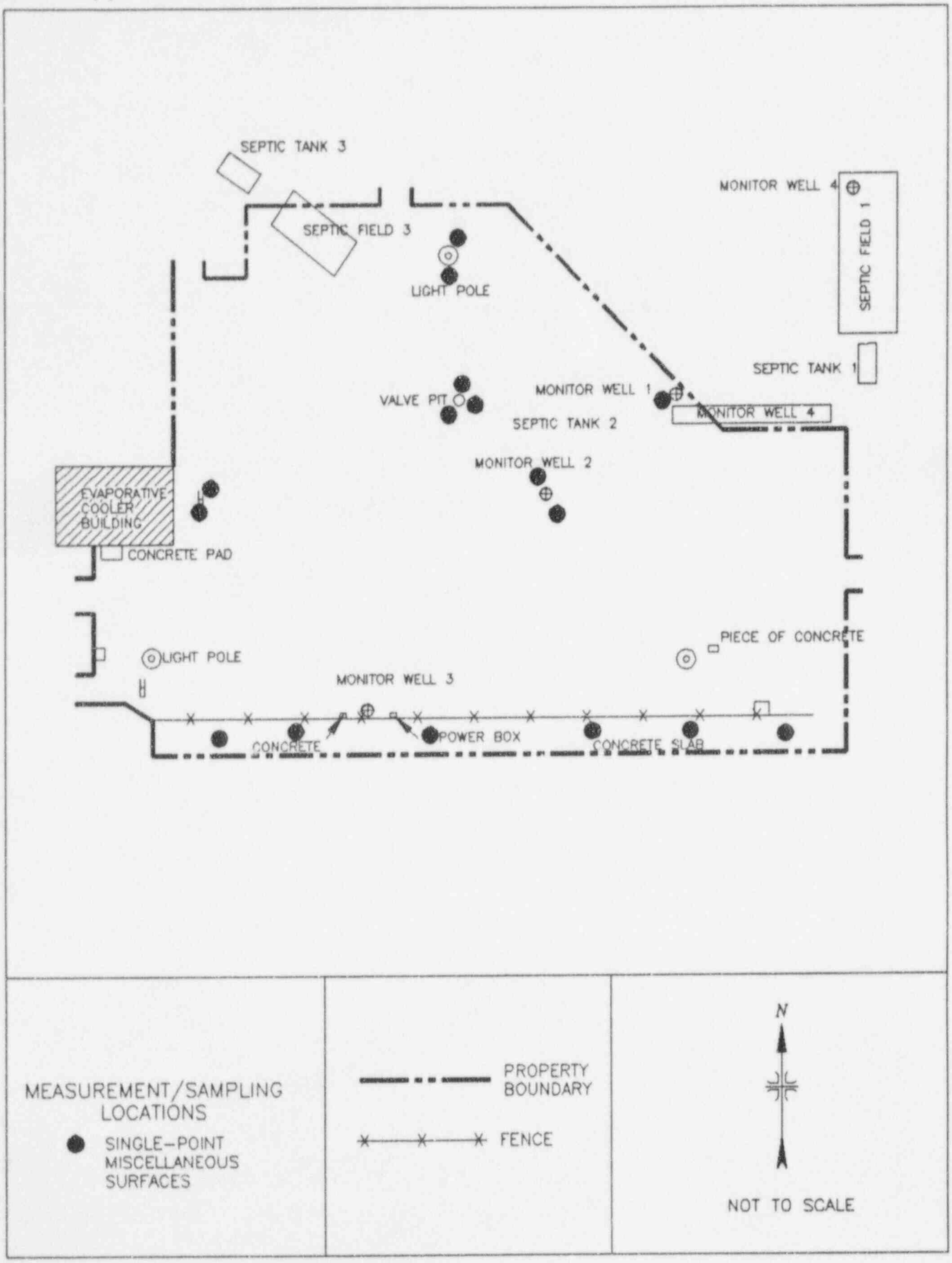

FIGURE 4: Repower Area, Miscellaneous Concrete and Metal Surfaces — Measurement and Sampling Locations

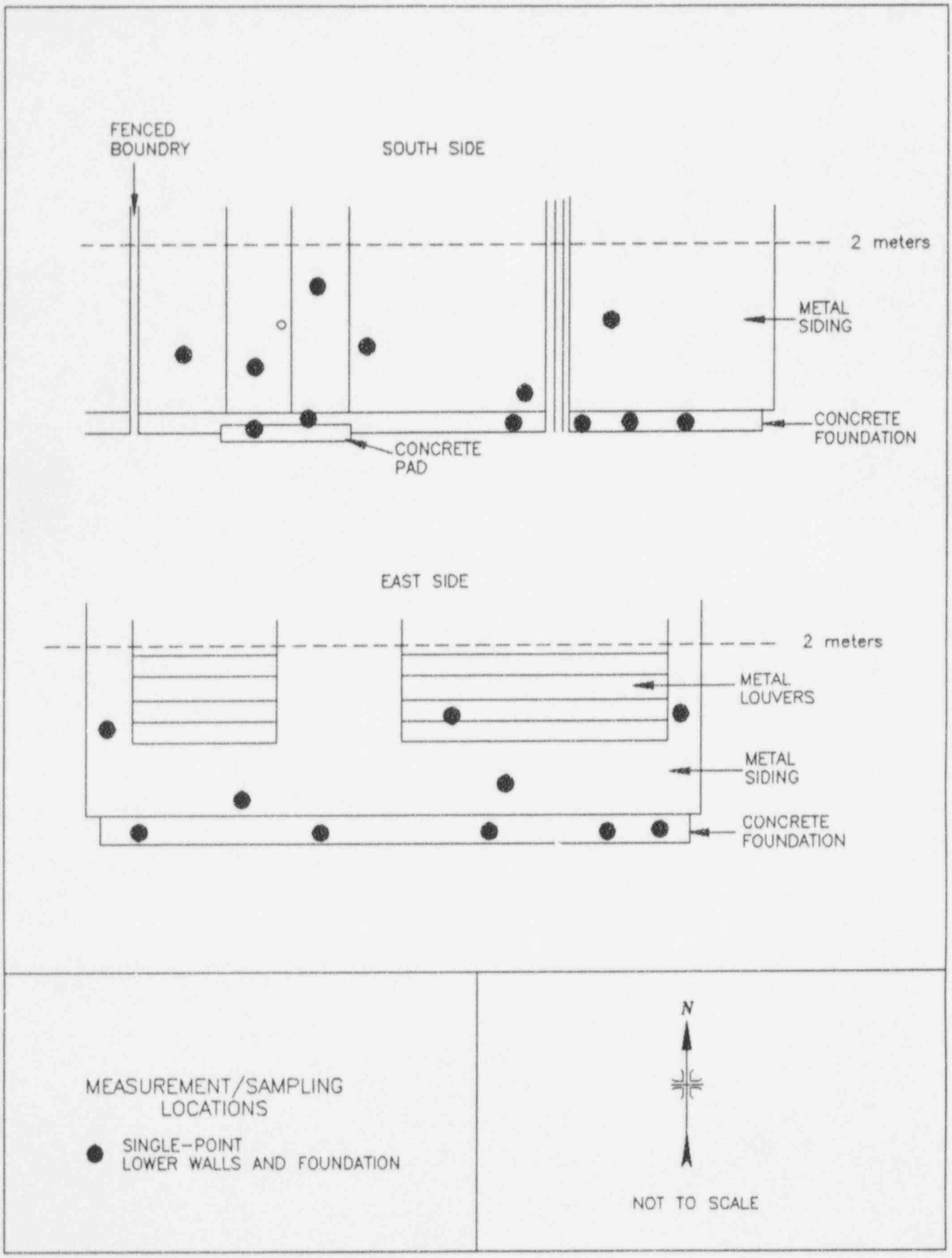

FIGURE 5: Repower Area, Evaporative Cooler Building – Measurement and Sampling Locations

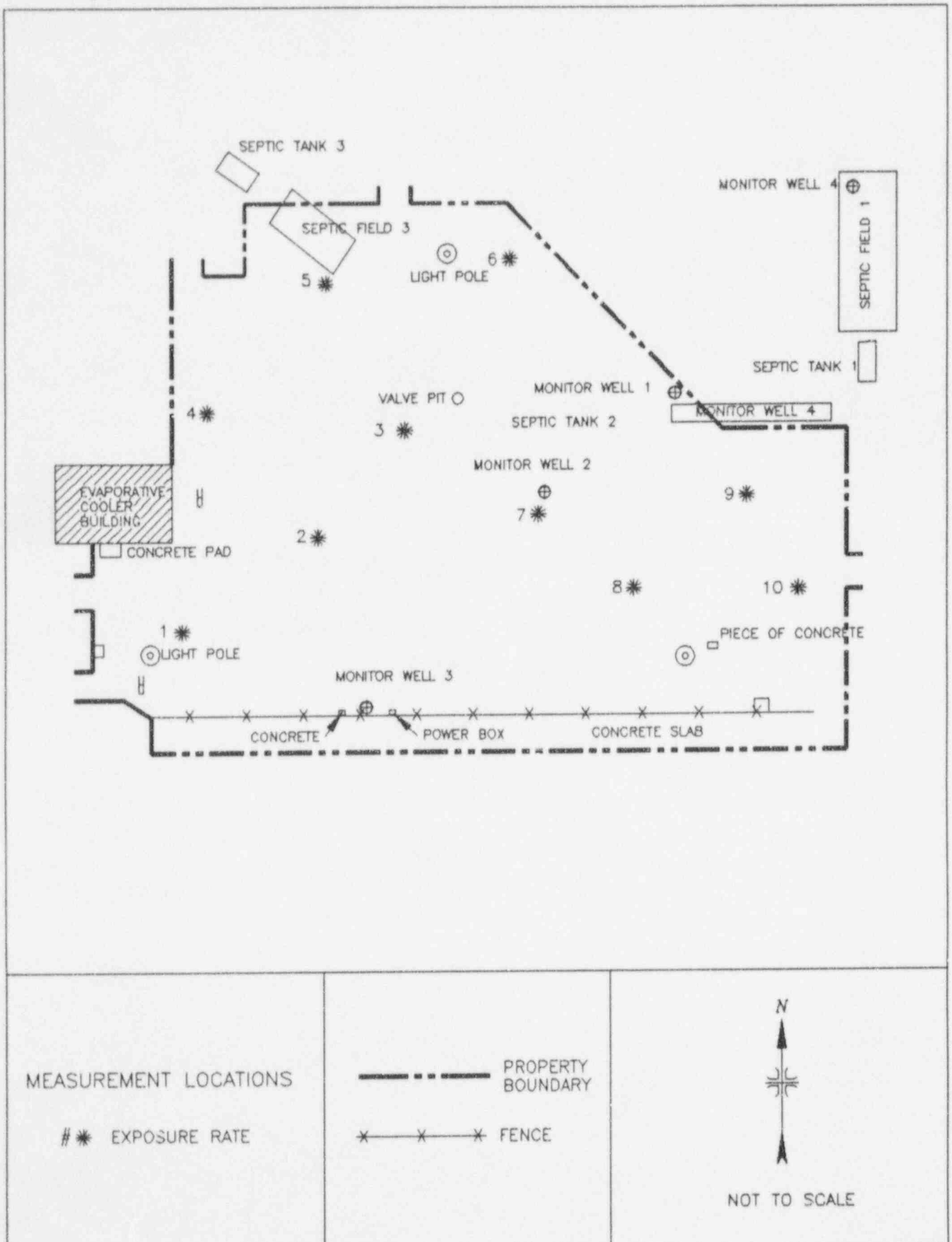

FIGURE 6: Repower Area – Exposure Rate Measurement Locations

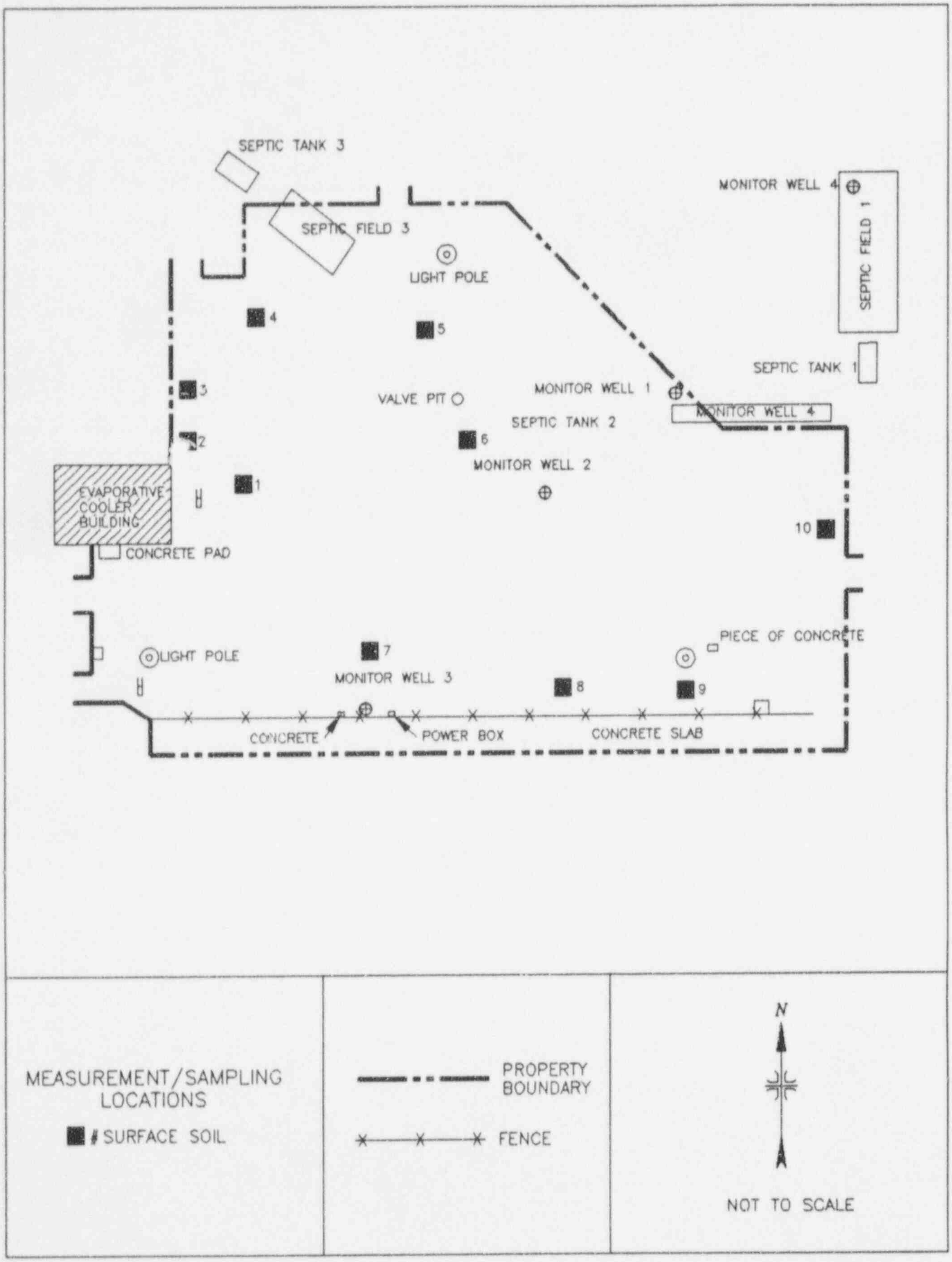

FIGURE 7: Repower Area – Soil Sampling Locations

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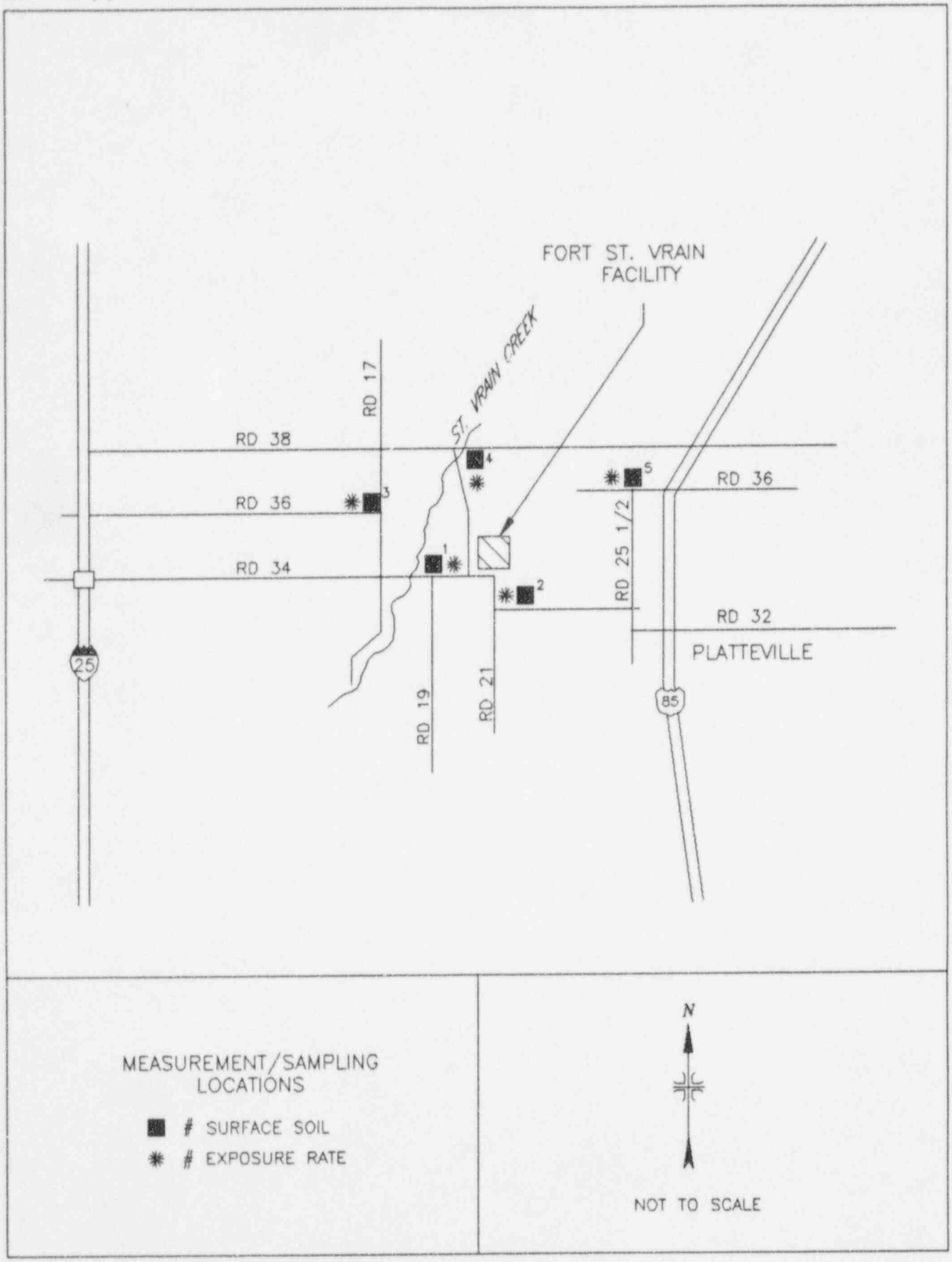

FIGURE 8: Fort St. Vrain – Background Soil Sampling and Exposure Rate Measurement Locations

TABLE 1

SUMMARY OF SURFACE ACTIVITY MEASUREMENTS
REPOWER AREA
FORT ST. VRAIN
PLATTEVILLE, COLORADO

Location[a]	Number of Measurement Locations	Range of Total Beta Activity (dpm/100 cm^2)	Range of Removable Activity (dpm/100 cm^2)	
			Alpha	Beta
Concrete Slab (at security fence)	6	<420	<12	<16
Miscellaneous Concrete	3	<420	<12	<16
Valve Pit	3	<420 - 710	<12	<16
Miscellaneous Metal	4	<340	<12	<16
Evaporative Cooler Bldg. (East and South Walls)	11	<340	<12	<16
Evaporative Cooler Bldg. (East and South Foundation)	11	<420	<12	<16

[a]Refer to Figures 4 and 5.

TABLE 2

EXPOSURE RATES
REPOWER AREA
FORT ST. VRAIN
PLATTEVILLE, COLORADO

Location[a]	Gross Exposure Rates (μR/h) at 1 m Above Surface		ESSAP Net Exposure Rates (μR/h) at 1 m Above Surface[c]
	ESSAP	Licensee[b]	
1	17.7	17.00	1.4
2	18.0	18.27	1.7
3	20.5	22.66	4.2
4	18.3	21.88	2.0
5	25.7	31.65	9.4
6	21.0	21.09	4.7
7	19.0	18.65	2.7
8	17.1	16.54	0.8
9	16.4	15.47	0.1
10	16.3	14.55	0

[a]Refer to Figure 6.
[b]Licensee gross exposure rates in repower area were measured with a NaI detector. PIC correction factor of 0.64 was applied to these measurements.
[c]Net exposure rates were determined by subtracting the background exposure rate (16.3 μR/h) from each gross exposure rate.

TABLE 3

BACKGROUND EXPOSURE RATES AND RADIONUCLIDE CONCENTRATIONS IN SOIL REPOWER AREA FORT ST. VRAIN PLATTEVILLE, COLORADO

Location[a]	Radionuclide Concentration (pCi/g)						Exposure Rate (μR/h) at 1 m Above Surface
	Co-60	Cs-137	Th-228	Th-232	U-235	U-238	
1	<0.1	<0.1	1.8 ± 0.2[b]	1.7 ± 0.4	<0.1	0.9 ± 1.4	16.9
2	<0.2	<0.1	1.5 ± 0.2	1.6 ± 0.4	<0.1	<2.1	16.3
3	<0.1	<0.1	1.4 ± 0.1	1.8 ± 0.4	<0.1	<1.5	15.4
4	<0.1	<0.1	1.6 ± 0.1	1.5 ± 0.5	<0.1	<1.9	16.7
5	<0.1	0.2 ± 0.1	1.7 ± 0.2	2.2 ± 0.6	<0.1	<1.8	16.3

[a]Refer to Figure 8.

[b]Uncertainties represent the 95% confidence level, based only on counting statistics.

TABLE 4

RADIONUCLIDE CONCENTRATIONS IN SOIL SAMPLES
REPOWER AREA
FORT ST. VRAIN
PLATTEVILLE, COLORADO

Location[a]	Radionuclide Concentration (pCi/g)					
	Co-60	Cs-137	Th-228	Th-232	U-235	U-238
1	<0.1	<0.1	1.6 ± 0.2[b]	1.5 ± 0.5	<0.1	<1.7
2	<0.1	<0.1	1.7 ± 0.2	1.6 ± 0.5	<0.1	<2.1
3	<0.1	<0.1	1.5 ± 0.1	1.6 ± 0.4	<0.1	0.9 ± 1.3
4	<0.2	<0.1	2.0 ± 0.2	1.8 ± 0.4	<0.1	<2.3
5	<0.1	<0.1	1.5 ± 0.1	1.6 ± 0.4	<0.1	1.0 ± 0.9
6	<0.1	<0.1	1.7 ± 0.1	1.6 ± 0.4	<0.1	<1.4
7	<0.2	<0.1	2.0 ± 0.2	1.8 ± 0.4	<0.1	<2.2
8	<0.1	<0.1	1.9 ± 0.2	1.8 ± 0.6	<0.1	<1.7
9	<0.2	<0.1	1.6 ± 0.2	1.4 ± 0.4	<0.1	<2.1
10	<0.1	<0.1	1.1 ± 0.1	1.1 ± 0.4	<0.1	<1.4

[a]Refer to Figure 7.
[b]Uncertainties represent the 95% confidence level, based only on counting statistics.

REFERENCES

1. Public Service Company of Colorado,"Final Status Survey Plan and Report," Cintichem, Inc., December 5, 1994.

2. Oak Ridge Institute for Science and Education "Confirmatory Survey Plan for the Repower Area, Fort St. Vrain, Platteville, Colorado (Docket No. 50-267)," March 17, 1995.

3. Oak Ridge Institute for Science and Education, letter from E. W. Abelquist to D. Fauver, NRC/NMSS, "Document Review - Final Survey Report for Release of the Repower Area, Fort St. Vrain, Platteville, Colorado (Docket No. 50-267)," March 13, 1995.

4. U.S. Nuclear Regulatory Commission, "Termination of Operating Licenses for Nuclear Reactors," Regulatory Guide 1.86, Washington, D.C., June 1974.

5. NUREG/CR-5512, "Residual Radioactive Contamination from Decommissioning," Volume 1, October 1992.

6. NUREG-1500, "Working Draft Regulatory Guide on Release Criteria for Decommissioning: NRC Staff's Draft for Comment," August 1994.

7. Public Service Company of Colorado, letter from D. Waremburg to J. Austin (NRC), "Final Survey Plan for Site Release," February 17, 1994.

8. Public Service Company of Colorado, letter from M. Fisher to M. Weber (NRC), "Transitory Radiation Levels in Fort St. Vrain Repower Area", March 15, 1995.

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

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

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

Detectors

Ludlum Gas Proportional Detector
Model 43-68
Effective Area, 100 cm^2
(Ludlum Measurements, Inc.,
Sweetwater, TX)

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

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

LABORATORY ANALYTICAL INSTRUMENTATION

High Purity Extended Range Intrinsic Detectors
Model No: ERVDS30-25195
(Tennelec, Oak Ridge, TN)
Used in conjunction with:
Lead Shield Model G-11
(Nuclear Lead, Oak Ridge, TN) and

Multichannel Analyzer
3100 Vax Workstation
(Canberra, Meriden, CT)

High-Purity Germanium Detector
Model GMX-23195-S, 23% Eff.
(EG&G ORTEC, Oak Ridge, TN)
Used in conjunction with:
Lead Shield Model G-16
(Gamma Products, Palos Hills, IL) and
Multichannel Analyzer
3100 Vax Workstation
(Canberra, Meriden, CT)

Low Background Gas Proportional Counter
Model LB-5110-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. Hand-held gas proportional detectors were used to scan the structural surfaces. 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:

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

Surface Activity Measurements

Measurements of total beta activity levels were performed using gas proportional detectors with ratemeter-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. The beta activity background count rate for the gas proportional detectors was 507 and 761 cpm on metal and concrete surfaces, respectively. The beta efficiency factors ranged from 0.24 to 0.25 for the gas proportional detectors calibrated to Tc-99. The probe area for the gas proportional detectors is 126 cm^2.

Soil Sampling

Approximately 1 kg of soil was collected at each sample location. Collected samples were placed in a plastic bag, sealed, and labeled in accordance with ESSAP survey procedures.

ANALYTICAL PROCEDURES

Gamma Spectrometry

Samples of soil materials were dried, mixed, crushed, and/or homogenized as necessary, and a portion sealed in 0.5-liter Marinelli beaker or other appropriate container. The quantity placed in the beaker was chosen to reproduce the calibrated counting geometry. Net material weights were determined and the samples counted using intrinsic germanium detectors coupled to a 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.

All photopeaks associated with the radionuclides of concern were reviewed for consistency of activity. Energy peaks used for determining the activities of radionuclides of concern were:

Co-60	1.173 MeV
Cs-137	0.662 MeV
Th-228	0.239 MeV from Pb-212*
Th-232	0.911 MeV from Ac-228*
U-235	0.186 MeV
U-238	0.063 MeV from Th-234*

*Secular equilibrium assumed.

Spectra were also reviewed for other identifiable photopeaks.

UNCERTAINTIES AND DETECTION LIMITS

The uncertainties associated with the analytical data presented in the tables of this report represent the 95% confidence level for that data. These uncertainties were calculated based on both the gross sample count levels and the associated background count levels. Additional uncertainties, associated with sampling and measurement procedures, have not been propagated into the data presented in this report.

Detection limits, referred to as minimum detectable activity (MDA), 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 MDA of the measurement procedure, the result was reported as less than MDA. Because of variations in background levels, measurement efficiencies, and contributions from other radionuclide in samples, the detection limits differ from sample to sample and 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 were used. Calibration of pressurized ionization chambers was performed by the manufacturer.

Analytical and field survey activities were conducted in accordance with procedures from the following documents of the Environmental Survey and Site Assessment Program:

- Survey Procedures Manual, Revision 8 (December 1993)
- 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 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

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USAEC REGULATORY GUIDES

Regulatory Guides are issued to describe and make available to the public methods acceptable to the AEC regulatory staff of implementing specific parts of the Commission's regulations, to originate techniques used by the staff in evaluating specific problems or postulated accidents, or to provide guidance to applicants. Regulatory Guides are not substitutes for regulations and compliance with them is not required. Methods and solutions different from those set out in the guides will be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission.

Published guides will be revised periodically, as appropriate, to accommodate comments and to reflect new information or experience.

Copies of published guides may be obtained by request indicating the division desired to the U.S. Atomic Energy Commission, Washington, D.C. 20545. Attention: Director of Regulatory Standards. Comments and suggestions for improvements in these guides are encouraged and should be sent to the Secretary of the Commission, U.S. Atomic Energy commission, Washington, D.C. 20545. Attention: Chief, Public Proceedings Staff.

The guides are issued in the following ten broad divisions.

1. Power Reactors
2. Research and test Reactors
3. Fuels and Materials Facilities
4. Environmental and Siting
5. Materials and Plan Protection
6. Products
7. Transportation
8. Occupational Health
9. Antitrust Review
10. General

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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 alpna- 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.