JUN 30 1987

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ORNL
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ORNL/RAP/LTR-86/35

DATE: July 1, 1986

SUBJECT: Inventory of ORNL Remedial Action Sites:
1. Solid Waste Storage Areas

TO: T. E. Myrick

FROM: T. Grizzard

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LETTER Report RAP86-35

Inventory of ORNL Remedial Action Sites: 1. Solid Waste Storage Areas*

T. Grizzard

Operations Division
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37831

July 1, 1986

Operated by

Martin Marietta Energy Systems, Inc.
for the
U.S. Department of Energy

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*Based on work performed at Oak Ridge National Laboratory, operated by Martin Marietta Energy Systems, Inc., under Contract No. DE-AC05-84OR21400 with the U.S. Department of Energy.

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ABSTRACT

Shallow land burial has been the method for disposal of solid low-level [both low activity (≤200 mR/h) and high activity (>200 mR/h)] radioactively contaminated waste at Oak Ridge National Laboratory (ORNL) since the beginning of operations in 1943. Most of the waste disposed at ORNL has come from various facilities within ORNL, but additional solid radioactively contaminated waste has been received from other sources (mainly DOE contractors).

Six solid waste storage areas (SWSAs) have been used since land disposal began at ORNL in 1943. The SWSAs are numbered consecutively in the order in which they were opened and used. This report discusses each of the six SWSAs used at ORNL as well as two other sites, the White Wing Road Storage Area and the Closed Contractor Landfill. Nuclide activity levels and volumes of waste disposed/stored on the Oak Ridge Reservations are presented, where available.

1. OBJECTIVES

The primary purpose of the Remedial Action Program is to provide surveillance and maintenance, corrective measures, and facility decommissioning necessary to place the Oak Ridge National Laboratory (ORNL) in compliance with applicable environmental and health regulations for radioactive and hazardous chemical wastes. Individual sites will be decontaminated in accordance with prioritites prescribed to control the potential for human exposure and relative magnitude of ecological impacts. Remedial action is a component of the Laboratory's Environmental Restoration and Facilities Upgrade Program (ERFU). In the program, remedial actions will be conducted at locations where past management practices or research activities have resulted in contamination of facilities or the environment that necessitate decommissioning or corrective actions. Remedial actions are divided into two activities: (1) Site Decommissioning Program (Surplus Contaminated Facilities Program-SCFP and Surplus Facilities Management Program-SFMP) and (2) the Site Corrective Measures Program (SCMP).

The Site Decommissioning Program is designed for management of surplus facilities (primarily structures), contaminated with radioactive materials and hazardous chemicals from the end of their operating life until final facility disposition. The Site Corrective Measures Program is designed for the management of inactive sites (process ponds, spill sites, environmental research areas, etc.),

contaminated with radioactive materials and hazardous wastes, leading to the eventual stabilization of these sites in accordance with environmental regulations.

Remedial Action Program Sites at ORNL comprise 13 distinct functional groups (of which the Solid Waste Storage Areas is one) among the several programmatic categories: (1) SCMP, (2) SCFP, and (3) SFMP. A series of reports is being prepared in which an inventory of each area is presented in a condensed summary. Each summary is the product of pertinent information abstracted from readily available literature and knowledgeable individuals on known contaminated sites requiring remedial action. The information includes site location and description; history; contaminants and their characteristics; extent of contamination, regulatory status, geological, hydrological, and ecological data; lists of major studies; identification of data deficiencies for environmental impact assessment and regulatory purposes; and knowledgeable individuals contacted.

The purpose of this report is to discuss the radioactively contaminated solid waste disposal/storage areas at ORNL and to summarize radionuclide concentrations and waste volumes disposed on the Oak Ridge Reservation. The data available for this report vary greatly, with the most current information being more nearly complete and reliable than information (if any) collected during the earlier years of ORNL operations.

2. BACKGROUND

The disposal/storage of radioactively contaminated solid waste at ORNL began in 1943. Since that time six SWSAs have been used to satisfy the requirement for space. Two additional areas, the White Wing Road Storage Area and the Contractor Landfill were also used and are also discussed in this report.

A computer program, The ORNL Radioactive Solid Waste Data File, for the SWSAs was initiated on January 1, 1962. This data inventory was identified as MOD 1 and was operational through December 31, 1974. From January 1, 1975 through March 31, 1977, the MOD 2 program was the official data program and from April 1, 1977 to the present, MOD 3 has been the data collection program.

Prior to January 1, 1962, scant records were kept, and not in a computer data base. Also, information was sparse as recordkeeping was lax and not standardized. A fire in 1957 destroyed the majority of records kept prior to that time (Binford and Gissel 1975, Bates 1983).

Starting on January 1, 1962, submitting an "Authorizations for Storage of Radioactive Contaminated Solid Waste" form (MOD 1 2822 form) was required of waste generators. Figure 1 is an example of a MOD 1 2822 entry. This form was required until January 1, 1975, when a modified 2822 form (MOD 2) was required. This modified form asked additional information of the generators. Figure 2 is an example of a completed MOD 2 2822 form. Then, on April 1, 1977, the MOD 3 2822 form was introduced. This form is currently used and a copy is presented in Figure 3. This form is vastly different from the two previous forms

and incorporates information that is requested by The Solid Waste Information Management System (SWIMS) which The United States Energy Research and Development Administration (ERDA) developed in 1977 (Dieckhoner 1979). Copies of The SWIMS form and the instruction sheet are presented in Figures 4 and 5.

Figure 1. MOD 1 burial authorization form for Oak Ridge National Laboratory.

BURIAL AUTHORIZATION FOR RADIOACTIVE-CONTAMINATED SOLID WASTE

REQUESTER: EXECUTES THIS SECTION BEFORE ARRANGING MATERIAL TRANSFER

THE FOLLOWING MATERIALS ARE TO BE BURIED
Contaminated Plywood, lead & debris

ORIGIN-BLDG	SECTION OF BLDG.	TYPE OF CONTAINER	PRINCIPLE RADIOACTIVE ISOTOPES PRESENT
3517	Cell 20	Doubly wrapped Plastic	^{90}Sr ^{137}Cs

PROTECTIVE CLOTHING REQUIRED FOR BOTH TRANSPORTATION AND BURIAL (BE SPECIFIC)
C clothing

PRECAUTIONS: HEALTH PHYSICIST MUST: ☐ ACCOMPANY SHIPMENT TRANSFER ☐ BE PRESENT AT BURIAL

☐ OTHER Do Not tear plastic. Use attached rope sling

BURIAL INSTRUCTIONS (See HPP-32, Responsibilities, 1.0 Area Supervision)
Bury as soon as possible

TELEPHONE BURIAL GROUND FOREMAN PRIOR TO MATERIAL TRANSFER (EXTENSION 6356 OR 6046)

REQUESTER'S AUTHORIZATION FOR DISPOSING MATERIALS	NAME	BADGE NUMBER	BUILDING
	Edward D. Matt	13488	3517
	DATE / DIVISION	DEPARTMENT	ACCOUNT CHARGE
	17 Feb / Isotopes	Process	3635-1

HEALTH PHYSICS: COMPLETED AT POINT OF ORIGIN OF SOLID WASTE AND BEFORE TRANSFER OF MATERIAL TO BURIAL GROUND

UNUSUAL HAZARDS

☑ CAUTION: DO NOT TEAR OR DAMAGE PACKAGING! ☐ CAUTION: FACE MASKS MUST BE USED DURING DUMPING OPERATIONS!

☐ OTHER

RADIATION LEVEL

BETA-GAMMA-SHIELDED, 10000 / 1500 mrem @ 12 / 36 inches; UNSHIELDED, mrem @ inches.

SURFACE CONTAMINATION- ~ 100 mR/hr minute Bo d/minute Alpha

PRECAUTIONARY MEASURES TO BE OBSERVED
plastic wrapped

HEALTH PHYSICS APPROVAL FOR MATERIAL TRANSFER TO BURIAL	NAME	DATE
	J H Spence	2-17-70

BURIAL GROUND: COMPLETED BEFORE RETURNING COPY TO ORIGINATOR

ALL INSTRUCTIONS NOTED WERE CARRIED OUT: ☐ YES ☐ NO ☐ SEE BELOW

UNUSUAL OCCURRENCES OBSERVED

COMMENTS REGARDING FIELD WORK

Trench 152 FR-120 FR-8 CF 30

BURIAL GROUND SUPERVISOR'S RECEIPT OF MATERIALS	NAME	BADGE NUMBER	DATE	TIME
	J Belinsky	15093	2-17-70	

DISTRIBUTION: Original - Burial Ground Foreman
1st Copy - Returns Completed to Originator
2nd Copy - Retained by Originator

UCN-2822
1123 3 641

AUTHORIZATION FOR STORAGE
OF RADIOACTIVE-CONTAMINATED SOLID WASTE

REQUESTER'S SECTION BEFORE AUTHORIZATION

ORIGIN (BLDG. IF ORNL): 3080
SECTION OF BLDG.:
DESCRIPTION OF CONTAINER(S): 2 Hot Cans
NO. OF CONTAINERS: 8
VOLUME OF WASTE: TOTAL 16 CU FT % BURNABLE

TYPE OF WASTE: [X] GENERAL RAD WASTE [ ] FISSILE (U-235) [ ] TRANSURANIUM OR U-233 [ ] OTHER (EXPLAIN IN REMARKS)

TOTAL RADIOACTIVITY 10 CURIES

REMARKS RE WASTE CONTENT AND/OR HANDLING PROCEDURES

Usual

TRANSURANIUM CONTAMINATED WASTE

SPECIFIC RADIOTOXICITY > 10μCi/Kg [ ] YES; [ ] NO

ISOTOPE(S) PRESENT AND ISOTOPE QUANTITIES:
[ ] Pu-239 _____ grams
[ ] U-233 _____ grams
[ ] OTHER (If Significant, Detail in Remarks)

FISSILE WASTE

[ ] U-235 _____ grams of Enrichment _____ %
[ ] OTHER (Detail in Remarks)

ACCOUNTABILITY NO.

NSR _____ ; SNM _____.

REQUESTER'S AUTHORIZATION FOR MATERIAL DISPOSAL

SIGNATURE: B P Phillips
PHONE NO.: 36314
BADGE NO.: 422264325
DATE: 3-3-76
DIVISION: Ops.
ACCOUNT NO.: 206921-01

MAIL FORM OR TELEPHONE STORAGE AREA FOREMAN FOR CONCURRENCE PRIOR TO MATERIAL TRANSFER (EXTENSION 3-6356)

RADIATION LEVEL

BETA GAMMA: SHIELDED _____ mrem/hr @ _____ inches; (OUTSIDE REUSABLE CARRIER)
UNSHIELDED 400 mrem/hr @ 1/2 inches. (OUTSIDE OF DISPOSABLE CONTAINER)

NEUTRON READING ——— mrem/hr

SURFACE CONTAMINATION — BETA GAMMA d/minute — ALPHA d/minute

[ ] H.P. TO ACCOMPANY SHIPMENT [ ] H.P. TO BE PRESENT DURING STORAGE [ ] RESPIRATORY PROTECTION REQUIRED

REMARKS REGARDING SAFETY IN HANDLING AND/OR STORAGE AND BURIAL

IN Clean CANS.

SPECIAL PROTECTIVE EQUIPMENT (For Storage or Burial)

[ ] COVERALLS, TAPED
[ ] COVERALLS, 2 PAIR
[ ] HEAD COVER
[ ] ASSAULT MASK
[ ] CHEMOX MASK
[ ] OTHER (DETAIL IN REMARKS)

HEALTH PHYSICS APPROVAL FOR MATERIAL TRANSFER

SIGNATURE: B. C. Burrell
BADGE NO.: 207-16-3801
PHONE NO.: 3-6585
DATE: 3/3/76

STORAGE AREA FOREMAN

ABOVE INFORMATION FOUND VALID? [X] YES [ ] NO (IF NO, EXPLAIN)

WASTE: [ ] STORED: [X] BURIED: IN SWSA NO. 6

REMARKS

TRENCH TYPE AND NO.: 192
WELL TYPE AND NO.:
FFO: 20
FFT: 8
BUILDING NO.:
COMPARTMENT:
LEVEL:
SWSA I.D.:
OTHER (EXPLAIN):

VOLUME (CU. FT.) TOTAL 16 : BURNABLE ✓ : NON-BURNABLE ✓ 16
CONTAINER TYPE: Hot cans

SUPERVISOR'S RECEIPT OF MATERIALS

SIGNATURE: H. R. Clowes
BADGE NO.: 408-90-8669
DATE STORED: 3-3-76

UCN-2822
(23 4-75)

DISTRIBUTION: WHITE - STORAGE AREA FOREMAN RETAINS
BLUE - COMPLETED AND RETURNED TO ORIGINATOR
CANARY - RETAINED BY ORIGINATOR

REQUEST FOR STORAGE OR DISPOSAL OF RADIOACTIVE SOLID WASTE OR SPECIAL MATERIALS

REQUESTER: EXECUTES THIS SECTION BEFORE ARRANGING MATERIAL TRANSFER

DATE	ORIGIN OF WASTE (BLDG. IF ORNL)	REQUESTER'S SIGNATURE	EMPLOYEE NO.	PHONE NO.	OFFICE AND BLDG. NO.	CHARGE NO.

TOTAL VOLUME (CU FT.)	COMBUSTIBLE VOL (CU. FT.)	WEIGHT (LBS)	ACCOUNTABILITY (MM NUMBER)	TOTAL CURIES IN WASTE (BEST ESTIMATE)

WASTE CLASSIFICATION (CHECK ONE)

0. ☐ TEMPORARY STORAGE
1. ☐ TRU OR ^{233}U (>100 μCi/Kg) - Retrievable Storage
2. ☐ URANIUM/THORIUM
3. ☐ FISSION PRODUCT
4. ☐ INDUCED ACTIVITY
5. ☐ TRITIUM
6. ☐ BETA-GAMMA TRU OR ^{233}U (>100 μCi/Kg) - Retrievable Storage
7. ☐ ALPHA TRU OR ^{233}U (<100 μ Ci/Kg) BETA-GAMMA TRU OR ^{233}U (<100 μ Ci/Kg)
8. ☐ OTHER _____
9. ☐ LANDFILL

TYPE OF WASTE (CHECK ONE)

1. ☐ BIOLOGICAL (BW)
2. ☐ CONTAMINATED EQUIPT. (CE)
3. ☐ DECONTAMINATION DEBRIS (DD)
4. ☐ DRY SOLIDS (DS)
5. ☐ SOLIDIFIED SLUDGE (SS)
6. ☐ NOT CLASSIFIED (NC)

BRIEF DESCRIPTION: _____

	CHARGE NO.
☐ CONSTRUCTION	
☐ RENOVATION	
☐ DECONTAMINATION & DECOMISSIONING (D&D)	

CONTAINER(S) IDENTIFICATION (INDICATE NUMBER OF EACH)

1. ☐ 55 GAL SS DRUM
2. ☐ 30 GAL SS DRUM
3. ☐ 4 1/2" WALL CONCRETE CASK
4. ☐ 6 IN. WALL CONCRETE CASK
5. ☐ 12 IN. WALL CONCRETE CASK
6. ☐ 55 GAL BI DRUM
7. ☐ 30 GAL BI DRUM
8. ☐ WOOD
9. ☐ OTHER
10. ☐ GI CAN
11. ☐ PLASTIC
12. ☐ DUMPSTER
13. ☐ NONE
14. ☐ CASK NO. _____ WALL THICKNESS _____ IN. SHIELDING MATERIAL _____

METAL BOX

15. ☐ 74.5" L x 50.5" W x 38.5" H
16. ☐ 68" L x 54" W x 38.5" H
17. ☐ 88" L x 54" W x 54" H
18. ☐ 112" L x 68" W x 77" H

PRINCIPAL ISOTOPE(S) (BEST ESTIMATE)

1. QUANTITY _____ ☐ GRAM(S) ☐ CURIE(S); IDENTITY _____
2. QUANTITY _____ ☐ GRAM(S) ☐ CURIE(S), IDENTITY _____
3. QUANTITY _____ ☐ GRAM(S) ☐ CURIE(S); IDENTITY _____
4. QUANTITY _____ ☐ GRAM(S) ☐ CURIE(S), IDENTITY _____

REQUESTER'S COMMENTS FOR THOSE HANDLING WASTE IN FIELD

QUANTITY _____ WATTS/M TOTAL

HEALTH PHYSICIST: EXECUTES THIS SECTION BEFORE MATERIAL TRANSFER

RADIATION DATA:

BETA-GAMMA: FOR PACKAGE _____ mrem/hr @ _____ IN. OR FOR SHIELDED CASK _____ mrem/hr. @ _____ IN.

SURFACE CONT. _____ dpm (β-γ); _____ dpm (α), NEUTRON READING _____ mrem/hr.

HP SURVEYOR'S COMMENTS FOR THOSE HANDLING WASTE IN FIELD

	HP'S SIGNATURE	PHONE NO

STORAGE AREA FOREMAN: COMPLETES AND SENDS COPY TO ORIGINATOR AFTER HANDLING WASTE

ACTION TAKEN (CHECK ONE)
WASTE WAS: ☐ STORED, ☐ BURIED, ☐ COMPACTED, ☐ OTHER _____

LOCATION SWSA NO. _____ DATE _____

FACILITY (Check One): ☐ BUILDING ☐ WELL ☐ RAVINE ☐ TRENCH ☐ ON GROUND ☐ OTHER

WASTE DESCRIPTION _____ FACILITY NUMBER _____ SWSA ATN _____

LOCATION WITHIN FACILITY
FFO FFT COMP. NO. LAYER FILE RANK

FISSILE WASTE DATA
WELL FULL: ☐ YES ☐ NO

CRITICALITY COMMITTEE
APPROVAL FOR _____ GRAMS

BALE NO _____

COMMENTS FROM SWSA FOREMAN REGARDING WASTE AND OR OPERATION

FOREMAN'S SIGNATURE _____

DISTRIBUTION
WHITE - STORAGE AREA FOREMAN RETAINS
BLUE - COMPLETED AND RETURNED TO ORIGINATOR
CANARY - RETAINED BY ORIGINATOR

UCN-2822
(3 9-83)

Form ERDA-735
(9-76)
ERDAM-0511

U.S. ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

SOLID RADIOACTIVE WASTE DATA-OPERATIONAL DATA

● *See Instructions on Reverse*

OPER OFFICE	CONTRACTOR CODE	SITE	OPER. TYPE	FISCAL YEAR	FISCAL QTR.	ACTUAL OR FORE	ACTION CODE	NUCLIDE
2	3 4	5	6	7 8	9	10	11	12

SECTION 0
To Be On All Cards

Prepared By

Date — Area Code — Phone

Approved By

Date — Area Code — Phone

SECTION 1

	C = Curie	F = Cu.Ft. M = Cu.M K = Gal. L = Liter	K = Kilogram P = Pound G = Gram	
	CURIE QUANTITY 1 3 – 2 1 2	GROSS VOLUME 2 3 – 3 1 3 2	GROSS WEIGHT 3 3 – 4 1 4 2	
2	+.E+	+.E+	+.E+	

SECTION 2

	NUCLIDE DESCRIPTION 13 – 26	*See Back of Form* Quantity or Avg. Conc. 27 – 35	36	COMMENTS 37 – 80
3		+.E		
3		+.E		
3		+.E		
3		+.E		
3		+.E		
3		+.E		
3		+.E		
3		+.E		
3		+.E		
3		+.E		
3		+.E		
3		+.E		
3		+.E		
3		+.E		
3		+.E		
3		+.E		
3		+.E		

SECTION 3

	WASTE CODE 13 14	F = Cu.Ft. M = Cu.M K = Gal. L = Liter VOLUME 15 – 23	24	Pct. Comb. 25 – 27	K = Kilograms P = Pound G = Gram WEIGHT 28 – 36	37	38	COMMENTS 38 – 80
4		+.E+			+.E+			
4		+.E+			+.E+			
4		+.E+			+.E+			
4		+.E+			+.E+			

SECTION 4

	TREAT. 13 14	F = Cu.Ft. M = Cu.M K = Gal. L = Liter VOLUME BEFORE TREATMENT 14 – 22	23	F = Cu.Ft. M = Cu.M K = Gal. L = Liter VOLUME AFTER TREATMENT 24 – 32	33	K = Kilograms P = Pound G = Gram WEIGHT BEFORE TREATMENT 34 – 42	43	K = Kilograms P = Pound G = Gram WEIGHT AFTER TREATMENT 44 – 52	53
5		+.E+		+.E+		+.E+		+.E+	
5		+.E+		+.E+		+.E+		+.E+	
5		+.E+		+.E+		+.E+		+.E+	

Figure 5. Solid Waste Information Management System (SWIMS) instruction sheet.

FORM ERDA-735
Solid Radioactive Waste Data Form — Operational Data

INSTRUCTIONS

GENERAL INSTRUCTIONS: All organizations generating, storing, or burying solid radioactive wastes are required to complete this operations data form. The information on existing conditions at the various facilities is required by ERDAM Chapter 0511, Radioactive Waste Management.

SPECIFIC INSTRUCTIONS:

- Data entered on the form must be clearly legible. Printing of required letters and numbers is preferred over writing or typing.
- Limit the number of characters in each block and do not use more than allotted spaces as indicated in various sections on this form.
- The alpha letter "O" shall be printed "Ø" to differentiate it from the number zero.
- The original and one copy of the form shall be submitted to the Division of Nuclear Fuel Cycle & Production, U.S. Energy Research and Development Administration, Washington, D.C. 20545, no later than one month following the end of the reporing period for which it is prepared.
- Scientific notation shall be used to record radioactivity, volume and weight in all sections of this form. This notation shall be expressed as follows:
  - + sign (preprinted on form)
  - Units (it is not necessary to code a decimal between units and tenths)
  - Tenths
  - Hundredths
  - Thousandths
  - E preprinted to denote exponential
  - Plus (+) or minus (-) sign of exponent (volumes and weights have preprinted plus sign)
  - Value of exponent

  Examples of this method of notation follow: .00515 is reported 5.105E-03; 5.15 is reported 5.150E+00; 5150. is reported 5.150E+03.

SECTION O

GENERAL INSTRUCTIONS: This section must always be completed. Enter the specific code in each space to identify organization, type of operation, year, and other required codes. The alphanumeric designations for the site identification required in Column 5 shall be established by contractor and/or operations office. Each organization shall ensure that there are no duplications within its jurisdiction when columns 2 through 12 are completed.

COLUMN:

2 Alphabetical designation for the Operations Office. Use office designations as shown below:

Albuquerque Operations Office	A
San Francisco Operations Office	B
Chicago Operations Office	C
Energy Research Centers	E
Headquarters - Germantown	G
Idaho Operations Office	I
Grand Junction Office	J
Nevada Operations Office	L
Oak Ridge Operations Office	Ø
Pittsburgh Naval Reactors Office	P
Richland Operations Office	R
Savannah River Operations Office	S
Schenectady Naval Reactors Office	Y

3-4 The alphabetical designation of the contractors, laboratories, research centers, and other ERDA organizations which handle solid radioactive wastes: this designation shall be compatible with the standard contractor name codes and titles provided by SWIMS program coordinators. Any necessary changes or additions must have prior approval of the SWIMS program coordinator.

5 The alphanumeric designation of the generating facility or storage site within the contractor organizations, such as fuel reprocessing plant or storage pad.

6 Identifying letter of the type of solid waste operation being reported:

B – Burial; G – Generation; R – Retrievable Storage

7-8 Fiscal Year - Enter identifying numbers for the fiscal year for which solid radioactive waste data is being reported. e.g. 78 for FY 1978.

9 Fiscal Quarter - Enter identifying number corresponding to the quarter of the fiscal year, e.g., a "4" would be entered for fourth quarter. A "5" will allow a full fiscal year of forecast data to be entered.

10 Type of Data - Alphabetical designation for type of data:

A – Actual; F – Forecast

11 Alphabetical designation of new or amended data:

Blank - New data; A – Amended Data

12 Identifying number of nuclide category as follows:

1 – TRU	4 – Induced activity	7 – Alpha
2 – Uranium/thorium	5 – Tritium	8 – Other
3 – Fission product	6 – Beta-Gamma-TRU	

SECTION 1

All organizations must complete and provide curie quantity, volume and weight for the appropriate reporting period.

COLUMN:

13-22 Indicate the curie quantity of radioactivity for nuclide category identified in Column 12 of Section 0.

23-32 Indicate the gross volume and related unit of measurement. If the volume has not been measured, the best estimate must be given.

33-42 Indicate the gross weight and related unit of measurement. If the weight has not been measured, the best estimate must be given.

43-80 Not used.

SECTION 2

The section is optional and is provided for those who maintain records by individual nuclides. A separate line entry must be made for each individual radionuclide of solid radioactive waste.

COLUMN:

13-26 Identify the individual nuclides contained in each category when this information is available from the organization generating the waste. Use normally accepted notations: Cobalt-60 is reported CØ-60. Uranium-235 is U-235. The designation should be compatible with the Standard Nuclide Identification list provided to Field Offices.

27-35 Indicate either the amount or average concentration (not both) of the nuclide reported in columns 13-26.

36 Indicate the units used to report the amount or concentration as follows:

C – Curies	G – Grams
F – Curies/cubic feet	H – Grams/cubic feet
K – Kilograms	M – Curies/cubic meter
P – Grams/cubic meter	

37-80 Any pertinent information on individual radionuclide.

SECTION 3

This section must be completed by all generators of solid radioactive wastes.

COLUMN:

13-14 Identify the type of waste contained in each nuclide category indicated in Column 12 of Section 0 as follows:

BW – Biological Waste	DS – Dry Solids
CE – Contaminated Equipment	SS – Solidified Sludge
DD – Decontaminations Debris	NC – Not classified

15-24 Indicate the volume for each individual type of solid radioactive waste. If the volume has not been measured, the best estimate must be given. The sum of volumes cannot exceed Section 1 gross volume.

25-27 Enter volume per cent (%) of waste which is combustible. Maximum entry is 100. For example, the waste is 80% combustible, would be entered as 080.

28-37 Indicate the gross weight and related unit of measurement. If the weight has not been measured the best estimate must be given. The sum of weights cannot exceed Section 1 gross weight.

38-80 Any pertinent information on type of solid radioactive waste.

SECTION 4

This section applies only to those organizations treating solid radioactive waste. A separate line entry must be made for each treatment process given to wastes. Note notations on the form.

COLUMN:

13 Identify the treatment process.

C – Compaction; N – Normal incineration; O – Other types

(Cost information must be supplied on ERDA Form 736 when treatment data is reported.)

14-23 Pretreatment volume and unit. The sum of volumes cannot exceed Section 1 gross volume.

24-33 Post-treatment volume and unit.

34-43 Pretreatment weight and unit. The sum of weights cannot exceed Section 1 gross weight.

44-53 Post-treatment weight and unit.

54-80 Not used.

3. GEOLOGY, HYDROLOGY, AND ECOLOGY

3.1 Geology

In the ORNL area, the succession of ridges and valleys from southeast to northwest is Copper Ridge, Melton Valley, Haw Ridge, Bethel Valley, and Chestnut Ridge. Webster (1976) summarizes results of geological studies of the ORNL area. The Chickamauga Limestone which underlies Bethel Valley (where ORNL, SWSAs 1, 2, and 3, and the Closed Contractor Landfill are located) is composed primarily of limestone, but shales, siltstones, and bedded chert are also present. Fractures and solution openings between beds of the Chickamauga Limestone result in uncertainty in regard to groundwater movement and this makes radionuclide monitoring difficult. As a result, this formation is regarded as undesirable for waste disposal. The lithologic character of the Conasauga Group (relative impervious) that underlies Melton Valley (where SWSAs 4, 5, and 6 are located) is variable, both along strike and in super position. The Conasauga Group contains shale, siltstone, and limestone.

There are no geologic data available for the White Wing Road Storage Area.

Soils in the Chickamauga Limestone area have been characterized as silty, with considerable amounts of clay present, and with a pH ranging from 4.5 to 5.7. The weathered zone in the areas underlain by the Chickamauga Limestone is thin, generally less than 3 meters (10 ft). In areas underlain by the Conasauga Group, the depth of weathering is related to topography: thin in low-laying areas and thicker on the

ridges. At SWSA 4, the weathered zone ranges from 1.2 to 4.9 meters (4 to 16 ft); at SWSA 5, up to 12 meters (from "a few feet" to 40 ft); at SWSA 6, generally, 1.5 meters or less to 12 meters (5 to 40 ft). The principal minerals in the weathered Conasauga Group are illite and vermiculite. The sorptive properties of each of the clay minerals vary for specific radionuclides (Webster 1976).

3.2 Hydrology

3.2.1 Surface Water

Debuchananne (Stockdale 1951) states that the section of Bethel Valley, in which the main part of ORNL is located, is drained by White Oak Creek and its tributaries. White Oak Creek flows out of Bethel Valley through Haw Gap at an elevation of 235 meters (770 ft). After passing through the gap, it is joined by Melton Branch and then flows south-southwest into the Clinch River, approximately 3.3 km (2.0 miles) away. Flow to the Clinch River is controlled by White Oak Dam, located approximately 1.0 km (0.6 mile) from the mouth of the creek which forms White Oak Lake.

A later study of the hydrology of the Oak Ridge area by McMaster (1967) provides data on runoff in area streams. The seasonal variation in runoff, averaged for the water years 1961-1964 and adjusted to the 1936-1960 water years, is as follows in terms of percentage of annual runoff: October-December, 17%; January-March, 49%; April-June, 23%; July-September, 11%. McMaster states that a large part of the runoff is derived from discharge of stored groundwater. The periods of maximum and minimum runoff correspond to the variations in rainfall.

These data represent averages for the Oak Ridge area and presumably apply to the burial ground sites as well.

Climatic conditions can have a significant effect on the near surface disposal of radioactive wastes. Richardson (1963) pointed out that climate is a primary factor affecting waste disposal at Oak Ridge. Richardson states:

> "The relatively large amount of rainfall at Oak Ridge has several effects. It causes the water table to occur at shallow depths and is responsible for seasonally large stream flow; it contributes to the development of a high drainage density, thereby reducing the distance between points of groundwater recharge and discharge and the length of time of groundwater residence; it lowers soil pH and influences the development of clay minerals that can control or modify the migration of radioactive ions; and in a less direct faction, it governs the composition of the natural biological community. Thus, the ultimate behavior and safety of the burial grounds are related to climate."

Webster has summarized geologic and hydrologic information that affects movement of radionuclides from waste disposal sites at ORNL. According to Webster (1976), White Oak Creek is both the natural drainage and an integral part of the Laboratory's waste distribution system. He also pointed out that the natural flow of water into the

creek is augmented by water piped into ORNL from outside the drainage basin and subsequently discharged to the creek as treated process waste water, laundry water, sanitary sewage, and reactor cooling water effluent. During prolonged periods of dry weather, the discharge from these sources often comprises a major fraction of creek flow.

3.2.2 Groundwater

Webster (1976) states that the depth to the water table varies seasonally and is generally greatest during September and October and least during the period January through March. For Melton Valley, water has been found at a depth of less than one foot near drainages and as deep as 67 feet beneath the line of low hills along the northwest side of the valley.

McMaster (1967) says that the residual material over the Conasauga is less compact and the water-bearing openings are larger than unweathered bedrock. As a consequence, the residuum bears most of the groundwater in the Conasauga Group outcrop belt. The thickness of the residuum in these belts is less than 10 meters (30 ft) in most places. Consequently, the volume of groundwater storage is small and is nearly depleted by September or October. He also states that annual water loss by evaporation and transpiration amounts to about 76 cm (30 inches), which is about 55% of the annual rainfall. During the July-September quarter at least 80% of the rainfall is returned to the atmosphere. Rainfall averages 29 cm (11.5 inches) and runoff, partly from groundwater storage, averages only 5.8 cm (2.3 inches).

Webster (1976) points out that water contour maps can be useful in predicting the direction of groundwater flow, indicating the areal hydraulic gradient from which the overall trend in movement can often be inferred. Webster (1976) also states that the water flow is toward White Oak Creek or its tributaries, except in a small area in and west of SWSA 3. In addition, Webster (1976) states that factors that control groundwater movement in the Conasauga Group change with depth.

Upward movement of contaminated groundwater can occur. Where the water table is shallow, evaporation of water into the atmosphere can be expected but, as Webster (1976) points out, the only radionuclide expected to migrate to any extent by this process is tritium. Studies near the waste pits (Auerbach 1958) showed that ^{106}Ru had been accumulated from groundwater by trees and deposited on the ground in fallen leaf litter. Other studies at ORNL (Struxness 1962) showed that ^{60}Co, ^{137}CS, and ^{90}Sr can be translocated from groundwater by this mechanism.

3.3 Ecology

3.3.1 Flora

The natural flora (oak-hickory overstory vegetation) typical of the Oak Ridge Reservation, that was probably originally present in the various areas used for waste disposal, was removed prior to their use for burial of solid wastes. In general, the growth of deep-rooted plants in these areas is discouraged both during use and after closure. The ground is seeded with a mixture of grass accompanied by application of lime and fertilizer in order to minimize erosion. One

exception to the above general statement may be noted in the case of SWSA 3 where trees were allowed to grow unhindered for a number of years after its closure in 1951. However, the trees were removed in 1979, along with items of contaminated equipment that were stored on the surface (Webster 1981). The area was seeded with grass after it was cleared. Another exception is in the case of the White Wing Road Storage Area where there has been no maintenance of the area and it is "grown up" in weeds and shrubs with cedars and pines inhabiting the area. There are no documented reports of rare or threatened plant species in any of the eight areas discussed in this report.

3.3.2 Fauna

The animals captured or observed at any of the eight sites discussed in this report include small mammals (rats, mice, and land shrews) and large mammals (oppossums, foxes, bobcats, groundhogs, skunks, and deer). Kitchings and Mann (1976) found that mammalian species inhabiting old field or disturbed areas are quite similar, whether the vegetative cover is grass or tree seedlings and shrubs. There are areas in SWSA 5 and SWSA 6 (as well as the grown-up White Wing Road Storage Area) that have overstory vegetation in locations unsuitable for trench disposal and animals characteristic of similar areas on the ORNL Reservation are present.

There are also bird populations in the eight areas discussed in this report not unlike other areas on the Reservation. There are, however, no documented reports of threatened or endangered mammal or bird species located here.

4. HAZARD ASSESSMENT AND REGULATORY STATUS

Martin Marietta Energy Systems (MMES) operates ORNL for the Department of Energy (DOE). MMES has the responsibility, by virtue of DOE Order 5480.14, to identify and to evaluate potential problems associated with inactive hazardous waste disposal sites at ORNL. The Order implements this policy by providing instructions for a DOE Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) program.

In order to estimate the relative hazard to human health, safety, and to the environment from potential migration of hazardous substances from the sites, as required by the DOE CERCLA Order, two hazard ranking systems were used. For nonradioactive sites, the Environmental Protection Agency's (EPA) Hazard Ranking System (HRS) was utilized. DOE's modified Hazard Ranking System (or m HRS) was used for evaluating radioactive sites. The potential hazard for both systems is estimated by calculating a score based on the migration potential of hazardous constituents via groundwater, surface water, or the atmosphere. Consideration is given to the waste characteristics, to specific pathway factors that may affect migration, and to the impact potential on humans or on the environment. The resulting computed migration score, Sm, then becomes an estimate of the relative hazard, with higher numbers indicating the greater risk.

Based on the m HRS, the highest Sm for a site addressed in this report was 7.2 (SWSAs 3, 4, 5, and 6) while the lowest Sm was 0.4 (Closed Contractor Landfill) (Nix et al. 1986). According to values

given and to the current EPA guidelines, none of these sites is an automatic candidate for the EPA's National Priority List (NPL) in which a score of 28.5 is required. The modified Hazard Ranking System scores (Sm) for the sites discussed in this report are as follows:

Site	Migration Score (Sm)
Solid Waste Storage Area 1	4.4
Solid Waste Storage Area 2	1.1
Solid Waste Storage Area 3	7.2
Solid Waste Storage Area 4	7.2
Solid Waste Storage Area 5	7.2
Solid Waste Storage Area 6	7.2
White Wing Road Storage Area	4.7
Closed Contractor Landfill	0.4

Since the completion of these hazard rankings, it has been determined that the regulatory status of the ORNL SWSAs is considerably more complex, and that most will also have to be dealt with under the terms of the Resource Conservation and Recovery Act, particularly Section 3004(u) dealing with corrective action for continuing release (Memorandum to Bobby Joe Davis, Chief, Environmental Protection Branch, DOE, Oak Ridge Operations, from P. S. Rohwer, ORNL, Environmental Coordinator, dated June 13, 1986). Table 1 identifies the specific units and the current assessment of the applicable regulations that govern each unit. These include RCRA regulations [permitted facilities

Table 1. Regulatory relationships for ORNL Solid Waste Storage Areas.

Solid Waste Storage Area (SWSA)	Regulatory Relationship			
	RCRA		DOE Orders	
	Permit	Potential 3004(u)	5480.14	5820.2
SWSA 1		X	X	
SWSA 2		X	X	
SWSA 3		X	X	
SWSA 4		X	X	
SWSA 5				
Inactive portion		X	X	
Active portion				X
TRU Waste Storage Area				
SWSA 6				
Inactive (closed) portion	X	X		X
Active portion		X		X
Closed contractor landfill		X	X	
White Wing Road Storage Yard		X	X	

and potential 3004(u) sites] as well as the applicable DOE Orders that govern remedial action (5480.14) and radioactive waste management (5820.2).

5. OPERATIONAL HISTORY AND INVENTORY OF SOLID WASTE STORAGE SITES

A summary of the eight sites discussed in this report includes the site history, physical and radiological conditions and past remedial actions of each. Table 1 lists each site, its operating dates, its status and its size.

5.1 Solid Waste Storage Area 1

The first storage area, SWSA 1, opened in 1943 and closed in 1944. It is located in Bethel Valley at the foot of Haw Ridge and the closest edge is approximately 7.6m (25 feet) south of White Oak Creek (Webster 1976, Myrick et al. 1984). The site is at the edge of the laboratory complex and it appears that the site was selected solely on the basis of its proximity to the laboratory. The site is triangle shaped and encompasses approximately 4,000 m^2 (one acre). The site is shown in Figure 6 (ORNL-DWG 80-16499).

The earliest record of burials in SWSA 1 is reflected in a memorandum of April 28, 1944. According to one account, the first waste was dumped in a long curving ravine. According to another account, the waste was first placed in auger holes at the east end of the area and later trenches were excavated to receive the waste. Only a small number of trenches were dug and it was reported that the site

TABLE 2. ORNL solid waste storage areas

Storage Area	Operating	Status	Acreage	ha
White Wing Road Storage Yard	1950-1964	closed	25	10.1
Closed Contractor Landfill	1974-1976	closed	3	1.2
SWSA 1	1943-1944	closed	1	0.4
SWSA 2	1944-1946	closed	4	1.6
SWSA 3	1946-1951	closed	7	2.8
SWSA 4	1951-1959	closed	23	9.3
SWSA 5	1959-	operating[1]	50	20.2
SWSA 6	1972-	operating[2]	68	27.5

[1]Retrievable storage facilities only. Area closed for Trench disposal in 1973.

[2]As of May 1, 1986, open only to Greater-Containment-Disposal (GCD) Techniques. Trench and auger hole disposal methods practiced prior to this time have been discontinued.

was abandoned when water was found in a trench excavated in the northern section of the site. A schematic diagram showing the orientation and approximate location of the trenches is given in figure 6.

The storage area is fenced and grassed and is maintained by Operations Division personnel. An access road to the newly constructed High Temperature Metals Laboratory and the south parking lot passes through the area. Some surface drainage from Haw Ridge flows across SWSA 1 to White Oak Creek. Groundwater movement in the area is also to White Oak Creek and is, therefore, monitored via The ORNL Stream Monitoring System.

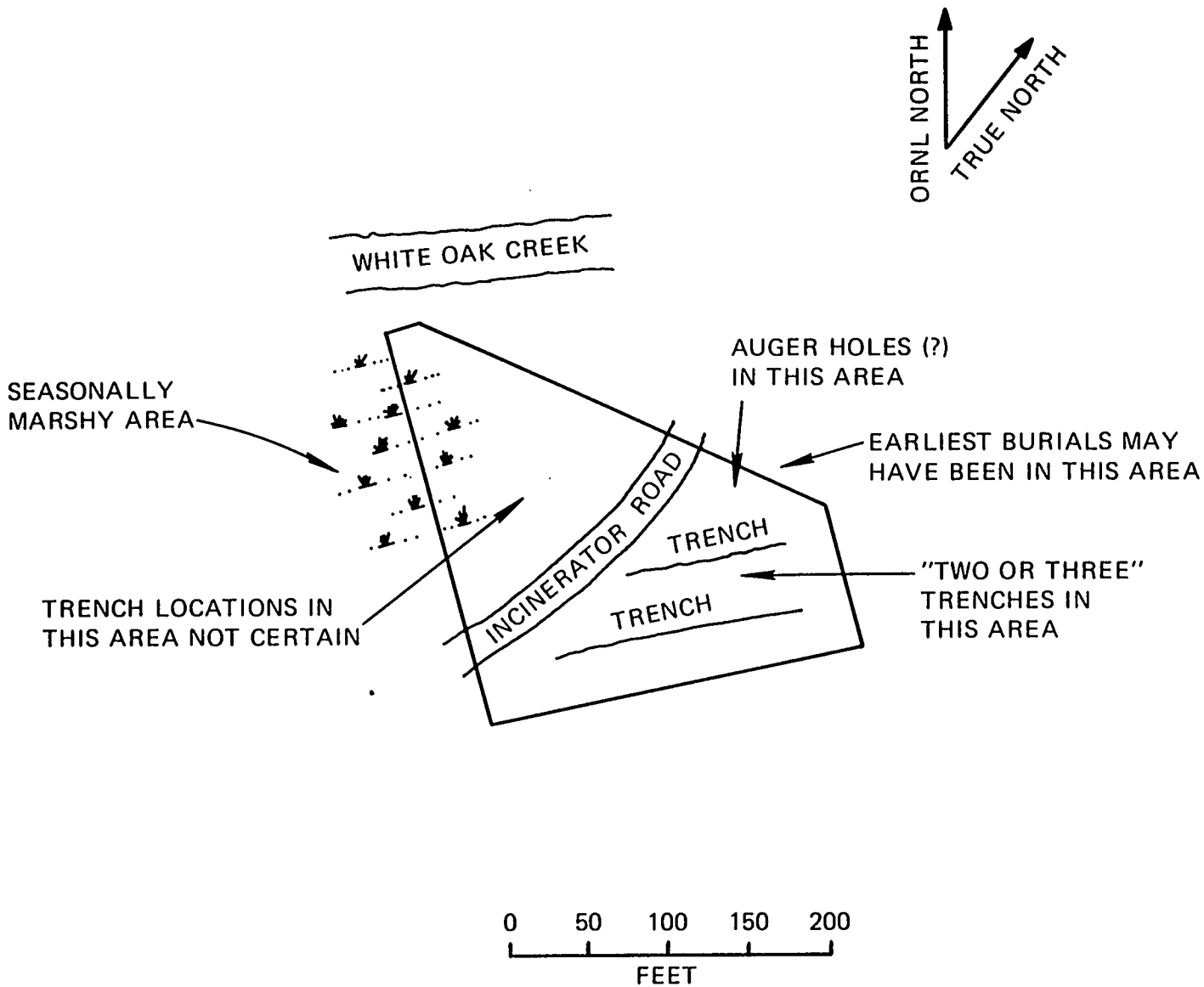
It is suspected that only a small amount (2,000-4,000 Ci from Bates 1983) of solid radioactive waste was buried in SWSA 1. There are no available records to show the quantity or type of solid waste disposed in this burial ground (Oakes and Shank 1979, Bates 1983).

The first record of burial ground monitoring was done in 1946 when the site was surveyed for general contamination. The survey showed only two spots that read above the background level of radioactivity and the survey did not report the analytical results (Myrick et al. 1984, Webster 1976).

The next recorded monitoring activity occurred in 1973 when water samples were taken from a seep and two monitoring wells(later removed) near, but outside of, the lower end of SWSA 1. Analyses indicated that water from one of the wells contained a minor concentration of ^{90}Sr.

Figure 6. Diagram showing the approximate location and orientation of trenches in SWSA 1. (Based on oral communication from J. E. Brandon and H. Dalton, 1974).

ORNL–DWG 80-16499



In 1975, water samples were taken from two wells, and a surface seep and analyzed for ^{90}Sr, ^{137}Cs, and Transuranic(TRU) elements. The results from one well indicated low concentrations of ^{90}Sr (9.4 dpm/ml) and no indication of ^{137}Cs or TRU elements (Evaluation Research Corporation 1982, Myrick et al. 1984).

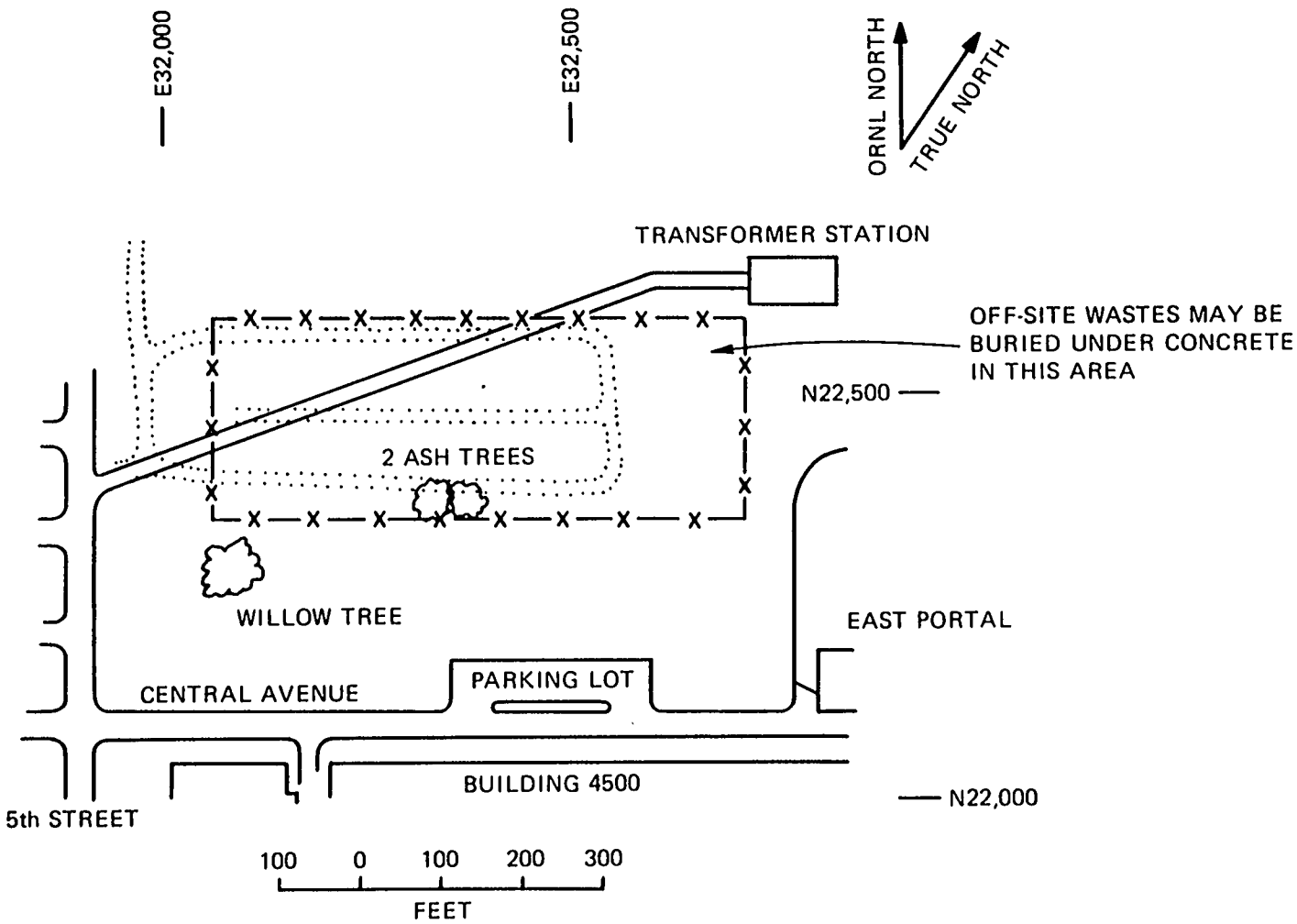
5.2 Solid Waste Storage Area 2

SWSA 2 (Active 1944-1946) is located on the south side of a hill near the east entrance and main parking area of the Laboratory, northwest of Building 4500 (Webster 1976, Oakes and Shank 1979). The area is currently unfenced and has a grass cover. An access road to a transformer station passes through the area. Figure 7 (ORNL-DWG 80-16504) is a diagram of the original site boundaries with respect to current landmarks. The disposal area encompassed approximately 1.4 ha (3.5 acres) and was opened shortly after SWSA 1 was closed. SWSA 2 was selected due to its accessibility and its proximity to the Graphite Reactor and various ORNL processing facilities.

Solid waste contaminated by beta or gamma activity was placed in black iron drums and then buried in trenches. Liquid waste contaminated with plutonium was placed in stainless steel drums and either buried in trenches or stored without burial in a "natural ravine" eroded in the site. The northeast corner of the site is believed to have been used for the burial of alpha-contaminated material from off-site. This material was covered with a slab of concrete after burial.

Figure 7. Diagram showing location of SWSA 2 relative to present features. Position of trees shown is approximate. Location of former fence and roads at the site is from Oak Ridge National Laboratory Engineering Drawing No. E-559.

ORNL–DWG 80-16504



EXPLANATION

X—X—X— FORMER FENCE AROUND BURIAL GROUND

:::::::::::::: ROADS IN AND NEAR BURIAL GROUND AT TIME OF USE.

Long-range land-use planning soon indicated that the area was unacceptable as a burial ground and the site was closed. After closure, all the waste was reported to have been exhumed and reburied at a new burial site (SWSA 3) (Oakes 1977, National Academy of Sciences 1976). The stainless steel drums containing liquid plutonium waste were removed intact, but the black iron drums containing the beta-gamma waste had deteriorated and required removal of the surrounding soil. The area was then bulldozed and seeded with grass. Surface water runoff from above the area has been diverted around the trenches to minimize erosion.

Conflicting reports have existed in regard to the removal of the buried waste from SWSA 2 with some accounts indicating that not all the waste was removed, in particular, larger equipment such as storage tanks and vehicles which had been disposed of at the site. Extensive core sampling has been completed in and around SWSA 2 since 1976 to determine the nature of the subsurface materials for a proposed Laboratory facility, the Energy Systems Research Laboratory. The proposed site includes the southwestern part of SWSA 2 and the area between SWSA 2 and Central Avenue. A total of 25 cores were taken, ranging in depth from 1.2 to 2.7 m (4 to 9 ft.), and the soil samples were analyzed for gamma-, beta-, and alpha-emitting isotopes (Webster 1976, Myrick et al. 1984). Water taken from 13 of the core holes was analyzed to determine tritium, gross beta, and gross alpha levels. None of the samples analyzed was found to be significantly higher than background samples collected throughout eastern and central Tennessee.

Results of these analyses indicate that there has been no significant migration of radionuclides from SWSA 2 and that probably most, if not all, of the contamination was removed from the area.

5.3 Solid Waste Storage Area 3

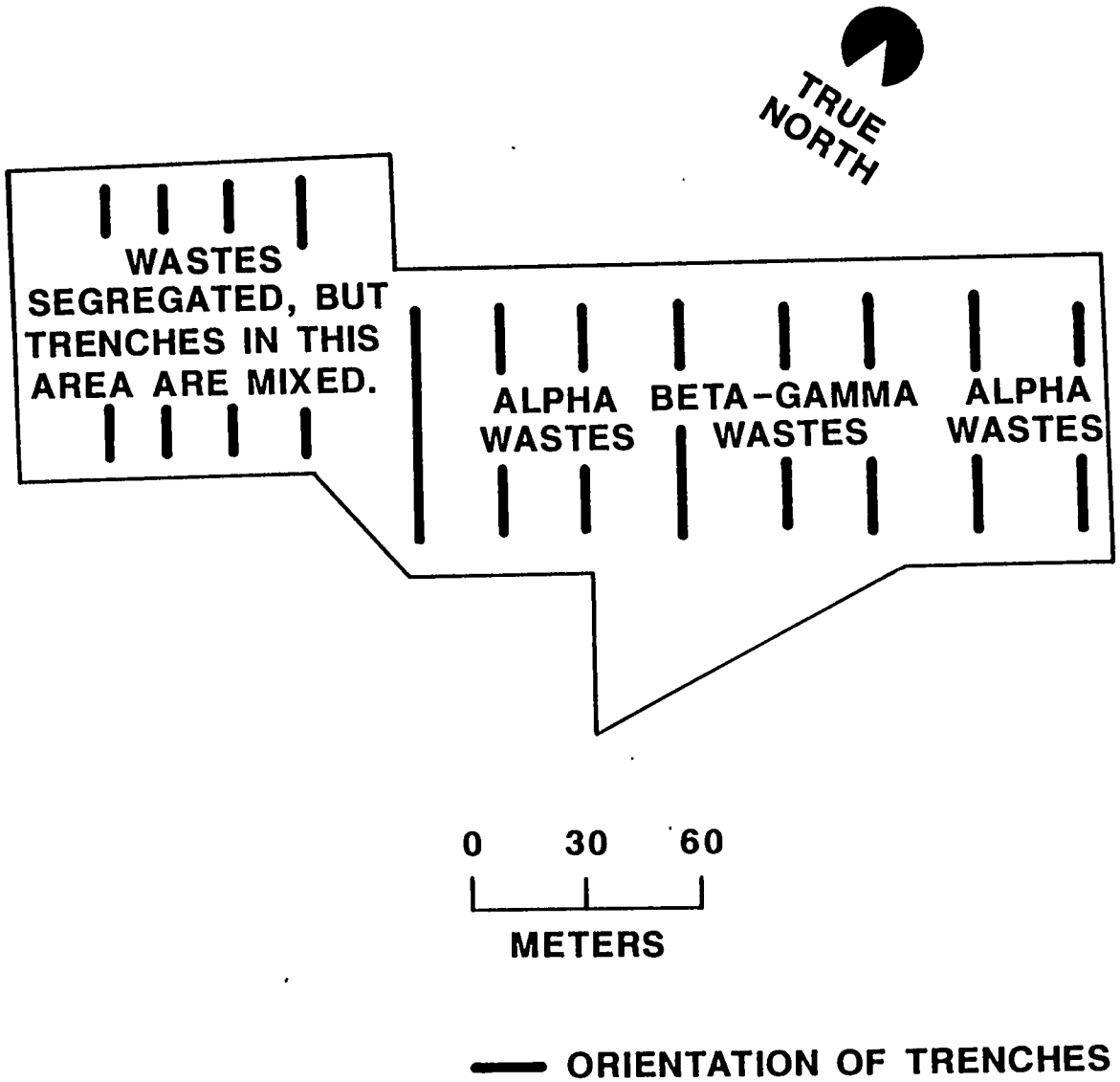
SWSA 3 (Active 1946-1951) is located in Bethel Valley in a fenced area at the foot of Haw Ridge approximately 0.6 miles west of the Laboratory complex (Oakes and Shank 1979). In recent years, after being closed for trench disposal, the site was used for aboveground storage of reuseable equipment. Bates (1983) showed that 44,000-56,000 Ci of radioactive waste were disposed of in SWSA 3.

Trenches were cut parallel to each other across the width of the site as shown in Figure 8. Alpha contaminated waste was placed in trenches and then covered with concrete, whereas the beta-gamma waste trenches were backfilled with native soil. Trenches in the area were generally 4.4 m (15 ft.) deep or less. The site was expanded westward until hard rock was encountered which made excavation difficult, and the site was closed in 1951 after utilizing some seven acres (Webster 1976, Myrick 1984)

During 1978 and 1979, a cleanup program was initiated to remove the aboveground stored material from SWSA 3 (Webster 1976). Stored items with detectable contamination levels were removed and buried in trenches in SWSA 6. Stored items with contamination levels undetectable by field survey instruments were buried in a landfill in SWSA 5. Radioactively contaminated items which were tagged for

Figure 8. Diagram showing location of SWSA 3.

ORNL-DWG 86-9757



continued storage were removed to a fenced aboveground storage area (approximately 0.8 ha) just west of SWSA 4 called "Storage Area 4 Extension". After removal of aboveground stored items, the east section of SWSA 3 was covered with about 8 in. of dirt and sown with grass. A ditch was dug along part of the south fence line to divert runoff from across the surface of the area. All trees which had grown in the area were removed.

In 1982, a small triangle-shaped section just south of a portion of the fenced area was found to contain buried contaminated metal. The material was not exhumed; instead, a fence was constructed to enclose the area. Upon completion of the fence extension, the enclosed older existing fence portion was removed (ORNL Eng. DWG. Location Plan SWSA No. 3 - Covered Scrap Metal Pile). This addition was leveled, fertilized and grassed and is now maintained along with the rest of SWSA 3.

SWSA 3, which was sited on limestones of the Chickamauga Group in Bethel Valley, has been shown to contribute ^{90}Sr via rapid subsurface flow to the Northwest Tributary (NWT) of White Oak Creek. In 1964, well water samples analyzed indicated small amounts of trivalent rare earths (TRE), ^{90}Sr, ^{89}Sr, and Tritium (Webster 1976). Well water samples collected in 1973 indicated ^{90}Sr levels up to 3.0 dpm/ml. Soil samples analysed in 1978 indicated higher than natural soil background levels which had been determined in 1976 (Oakes and Shank 1979, Oakes 1976, Eldridge 1979).

A connection from SWSA 3 to Raccoon Creek has also been demonstrated. The line of connection runs parallel to bedding strike and is thought to represent a solution channel that allows rapid transmission of water during storms. Dye tracing studies show rates of movement comparable to open channel flow velocities, although at present, quantities of radionuclides transported along these pathways are rather minor (Myrick et al. 1984).

Rothschild et al (1985) performed very-low-frequency-electromagnetic (VLF-EM) surveys in SWSA 3 which identified possible voids and/or fractures in the area. Results of this study may significantly influence the locations of where to drill future monitoring wells for best results.

5.4 Solid Waste Storage Area 4

Sufficient studies had been completed on radionuclide movement within burial areas by 1951 to indicate the potential of the Conasauga shale for adsorption of radionuclides. SWSA 4 was sited in Melton Valley in areas underlain by Conasauga shale to take advantage of this property (Oakes and Shank 1979). The site selected is located within the White Oak Creek drainage basin approximately 780 m (0.5 miles) southwest of the Laboratory complex. The SWSA is bounded on its northern border by Lagoon Road, an ORNL access road. The site is currently fenced and grassed (Webster 1979).

SWSA 4 was opened for routine burial in February 1951, and remained in operation for approximately 8.5 years encompassing a total area of 9.3 ha (23 acres). Development of SWSA 4 proceeded gradually from the low-lying northeast end to the higher southwest end. Trench orientation was variable and lacked any consistent relationship to the original site topography. A trench for burning noncontaminated material was also operated within the area (Webster 1976). Much of the site, primarily in the northeastern portion, has been covered with rock and dirt from building excavations at ORNL. In some places as much as 5.9 m (20 ft.) of this material overlie the original ground contour (Duguid 1975).

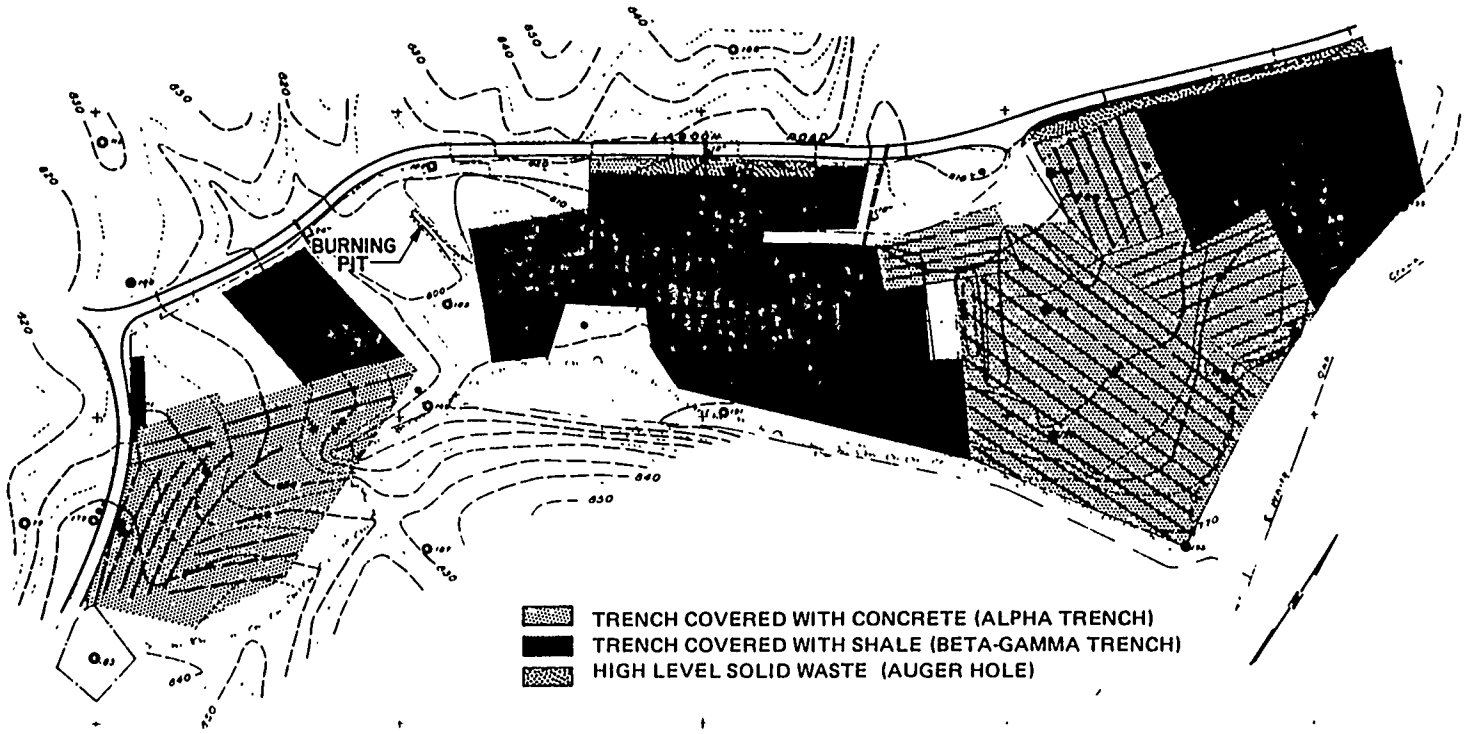
Trench burial was utilized in SWSA 4 for routine beta-gamma wastes. Trenches ranged from 14.8 to 118 m (50 to 400 ft.) in length, 2.4 to 8.9 m (8 to 30 ft.) in width, and 2.4 to 4.1 m (8 to 14 ft.) in depth. Burial was limited to high elevations during the wet periods, while the low topography was utilized during the drier summer months (Lomenick and Cowser 1961). Alpha-contaminated wastes were disposed of in trenches with the addition of a 0.44 m (1.5 ft.) concrete slab poured over each trench. Higher activity waste and some "Special" high activity waste were placed in concrete lined auger holes 0.3 to 0.6 m(1 to 2 ft.) in diameter and approximately 4.4 m (15 ft.) deep. Auger holes were also used for the "Temporary Storage" of material contaminated by fission products of short half life. About 50 of these auger holes are located in the northern part of SWSA 4 along Lagoon Road. To date, these "Temporary Storage" auger holes still have

material in them and there have been discussions as to whether to fill the holes with concrete or to retrieve the material and dispose of it in SWSA 6. The major concern, however, is identification of the material as no records exist. Figure 9 (ORNL-LR-DWG 58754) shows the SWSA 4 trench alignment, type of cover (shale or concrete), the high activity waste burial areas and the noncontaminated burning pit.

Little information exists to characterize the type, concentration or quantity of radionuclides placed in SWSA 4 for any of its years of operation. Bates' (1983) report estimates 90,000-120,000 Ci of waste were disposed of in SWSA 4 during 1951-1959. A wide variety of contaminated waste was buried in SWSA 4 in a wide variety of container types. The list includes materials such as paper, glassware, scrap metal, dirt, filters, oils, powders, depleted uranium, animal carcasses and large pieces of equipment. Records were maintained of the waste inventory of SWSA 4 but were destroyed by fire. Records are available for 1957 and 1958 which may be representative of the immediately previous years. During 1957 approximately 7,219 m^3 (255,000 ft^3) of waste were buried in SWSA 4 and during 1958 approximately 9,510 m^3 (336,000 ft^3) were buried (Lomenick and Cowser 1961). Approximately 50 percent of this waste was generated by ORNL with the remainder being shipped in from off-site (Lomenick and Cowser 1961). The curie content description of these off-site wastes was frequently incomplete and in many cases was specified in "less than" quantities which may differ greatly from the actual quantity.

Figure 9. Map of SWSA 4 showing location of auger holes, and orientation and general location of alpha and beta-gamma trenches (from Lomenick and Cowser 1961).

ORNL–DWG 86-12160



Although the existence of Conasauga shale was the determining factor in the site selection for SWSA 4, little consideration was given to other geologic and hydrologic parameters. Consequently, much of the waste in SWSA 4 is located in or very near the water table as illustrated by the shift to higher ground during the wet winter months. This factor has resulted in radionuclide transport within the burial area with the development of a number of "seeps" near the lower portion of the SWSA (Myrick et al. 1984). During 1959 and 1960, sampling of wells and streams in and near SWSA 4 showed that both ground water and surface water were contaminated. Water samples taken from eight of sixteen wells showed beta-gamma contamination and one contained alpha activity. Identifiable radioactive contaminants included ^{90}Sr, ^{106}Ru and trivalent rare earths (TRE). Alpha and beta activity were found in water from seeps and in the intermittent stream that borders the burial area to the south. Contaminants in two seeps included ^{90}Sr, ^{137}Cs, ^{60}Co, and TRE and in the intermittent stream, ^{106}Ru, ^{90}Sr, ^{210}Po, ^{239}Pu and TRE (Webster 1976).

In 1964, water samples were taken from six wells and one seep at SWSA 4 and each was found to contain ^{90}Sr, tritium, TRE and small amounts of ^{106}Ru (Webster 1976).

In 1973, eight wells sampled on the burial ground surface indicated contamination of ^{90}Sr while two of the wells showed measurable contamination of ^{125}Sb. In September of 1974 analysis was begun for tritium at SWSA 4 (Duguid and Reeves 1976). Although the

principal radionuclides present in the groundwater in and near SWSA 4 are tritium and ^{90}Sr, occasional water samples are found to contain small amounts of ^{60}Co, ^{125}Sb, and ^{137}Cs (Duguid 1976).

SWSA 4 has been shown to contribute approximately 35-50% of the ^{90}Sr that is discharged annually from the White Oak Creek basin at White Oak Dam. It has also been estimated that nearly 80% of the ^{90}Sr discharged from SWSA 4 is associated with stormflow. Recent studies also suggest that there is considerable radionuclide transport that involves the bathtub effect, where trenches collect water faster than it can drain away, and the result is overtopping of the trench at the lowest point along the rim (Duguid 1975). In extreme cases, positive pressure develops within the trench and the leachate is discharged as an artesian spring, primarily during storms.

In an effort to decrease the impact of water intrusion, a surface runoff collector and diversion system was constructed in 1975. This system consists partly of a shallow, asphalt-lined ditch along the north side of the burial ground. Surface water runoff is then diverted across the disposal area by means of three asphalt-lined conductor ditches and one unlined, natural drainage ditch at the northeast edge of the burial ground. The diversion system carries large amounts of water during heavy rains and small amounts of water for several days afterwards. In addition, in 1983, a second surface water diversion project was undertaken in an effort to channel the flow from the area north of the burial ground around the site rather than through it (Davis et al. 1985).

Evaluation of the effectiveness of this diversion is currently underway, and early indications are that a significant reduction is being achieved in ^{90}Sr releases to White Oak Creek (Myrick et al. 1984).

Spalding and Munro (1983) found the greatest concentration of ^{90}Sr groundwater contamination occurred mainly in the south central region of SWSA 4 and discharged into the headwaters of the primary tributary that drained the area. Also, SWSA 4 was found to be slightly supersaturated with respect to $CaCo_3$ (calcite).

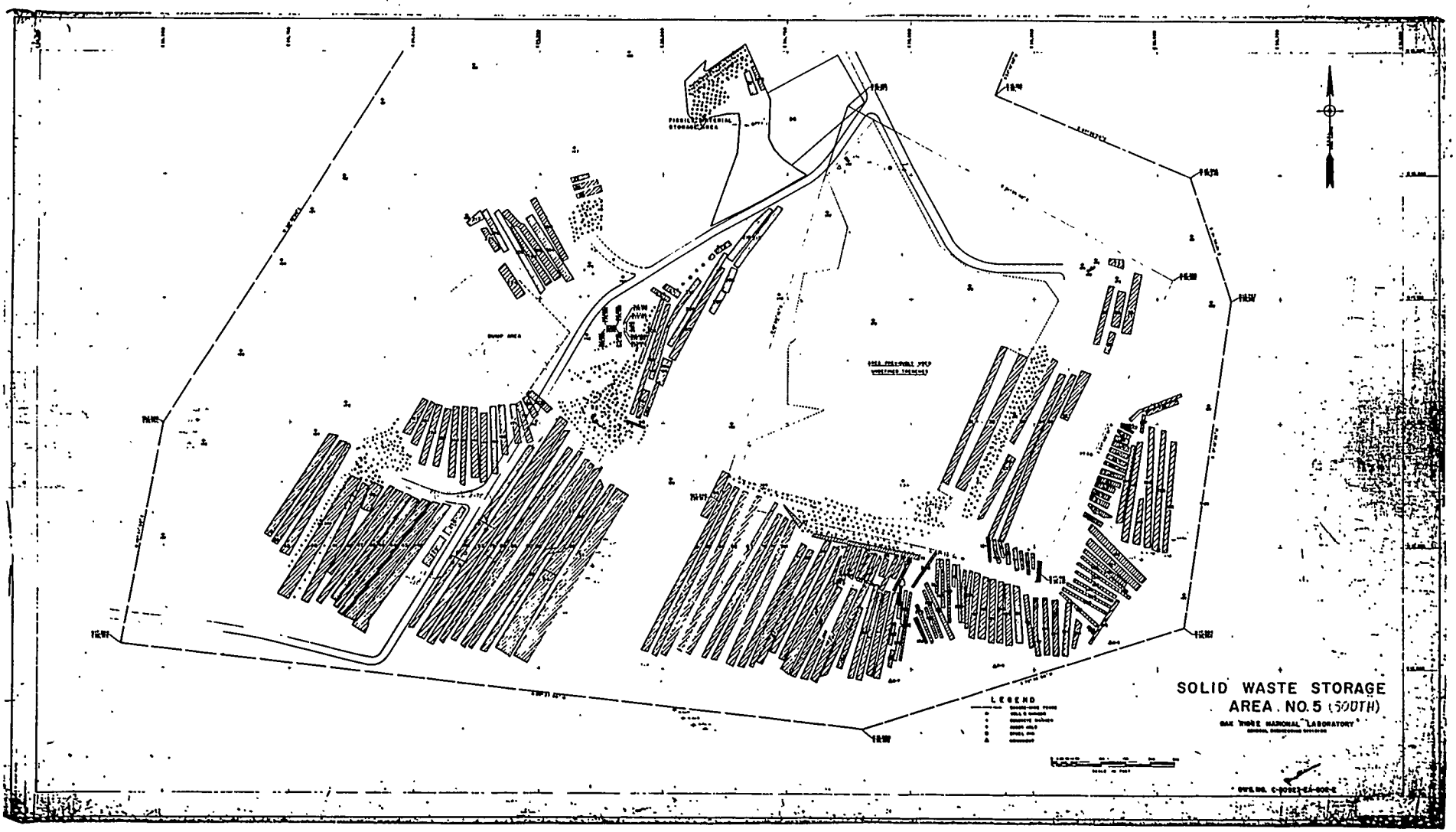
5.5 Solid Waste Storage Area 5

SWSA 5 was opened in 1959 as SWSA 4 neared full capacity. The selection of the site for SWSA 5 was based on experience gained as a result of previous burial operations as well as studies which were completed on radionuclide transport within soil. Criteria were established for selection of the SWSA 5 site including such considerations as depth to groundwater, ease of operation, flood potential, soil erosion by surface runoff, easy excavation, short hauling distance, restricted access roads, and underlying geology (Bates 1983). The site selected is located on a hillside east of White Oak Creek, between SWSA 4 and Melton Branch. SWSA 5 is composed of two distinct geographical areas providing different solid waste storage functions. These areas will be referred to as SWSA 5/north and SWSA 5/south. SWSA 5/south was used for disposal of routine buried waste; whereas, SWSA-5/north has been used mainly for retrievable storage of transuranic contaminated waste (Oakes and Shank 1979).

SWSA 5 (north and south) is a fenced area of approximately 32.3 ha. (80 acres) of which only approximately 20.2 ha (50 acres) have actually been used for waste disposal (Webster 1976, Myrick et al. 1984). The majority of this area is contained in SWSA 5/south. SWSA 5/south is a moderately sloping hillside with a grass cover (Figure 10) (ORNL-DWG No. C-20923-EA-002-E). The area is fenced and contains several unpaved access roads. Portions of the area are topographically or hydrologically unsuitable for trench disposal and thus have not been used for burial of waste. The original ORNL Hydrofracture Facility, utilized for many years for permanent disposal of ORNL generated liquid waste, but now deactivated, is located just outside the southwestern corner of SWSA 5/south.

Waste disposal in SWSA 5/south was accomplished in similar manner to SWSA 4. Beta-gamma contaminated waste was disposed of by normal trench burial (Bates 1983). Trenches in the area vary considerably in size from approximately 12 to 150 m (40 to 500 ft.) in length. They are approximately 4.4 m (15 ft.) deep and 3.5 (12 ft.) wide. The trenches were located at right angles to the strike of the shale to minimize collapse of the trench walls and thus, most trenches are located parallel to the slope. The "undefined area" shown was used primarily for disposal of segregated alpha contaminated waste which was trench buried and then covered with a slab of concrete. The records for the "undefined area" are missing. Auger holes were utilized for disposal of higher activity and fissile waste in the areas designated on Figure 10.

Figure 10. Location and alignment of trenches and location of auger holes in SWSA 5-south (from Oak Ridge National Laboratory Engineering Drawing No. C-20923-EA-002-E).


SOLID WASTE STORAGE
AREA . NO. 5 (SOUTH)
OAK RIDGE NATIONAL LABORATORY
LEGEND

SWSA 5/north has been used almost exclusively for the retrievable storage of transuranic waste (Webster 1976, Myrick et al. 1984, Bates 1983). Figure 11 (ORNL DWG No. C-20923-EA-001-E) shows the numerous retrievable facilities in SWSA 5/north. Figure 11 also shows these facilities along with the locations of trenches where waste has been retrievably buried.

TRU waste is stored in SWSA 5/north in three basic modes. Low-level activity waste (≤200 mR/h on contact) is placed in stainless steel drums which are held temporarily in a staging facility, Bldg. 7823, before being placed in partially underground storage facilities, Bldgs. 7826 and 7834. Hot cell waste requiring shielding (≤200 mR/h at the surface of the container) before storage is placed in concrete casks which were previously buried retrievably in the trenches shown in Figure 11. Currently, these concrete casks are being placed in the concrete storage facility, Building 7855. Transuranic Waste with a high-activity beta-gamma background is sealed in stainless steel containers and placed in stainless steel wells in Bldgs. 7827 and 7829.

Table 3 shows the contact handled (≤200 mR/h) alpha waste retrievably stored in SWSA 5-north. The table shows the number of drums stored by fiscal year along with the volumes and grams of TRU and ^{233}U. In FY 1984, 23 metal boxes of various dimensions were retrievably stored in Building 7834. The table also gives the total through September 1984 (fiscal 1984). Values given here are summarized from the MOD 3 ORNL Radioactive Solid Waste Data File. Bates (1983) has published the values taken from this file through FY 1981.

Figure 11. Location and alignment of trenches and location of auger holes in SWSA 5-north (from Oak Ridge National Laboratory Engineering Drawing No. C-20923-EA-001E).

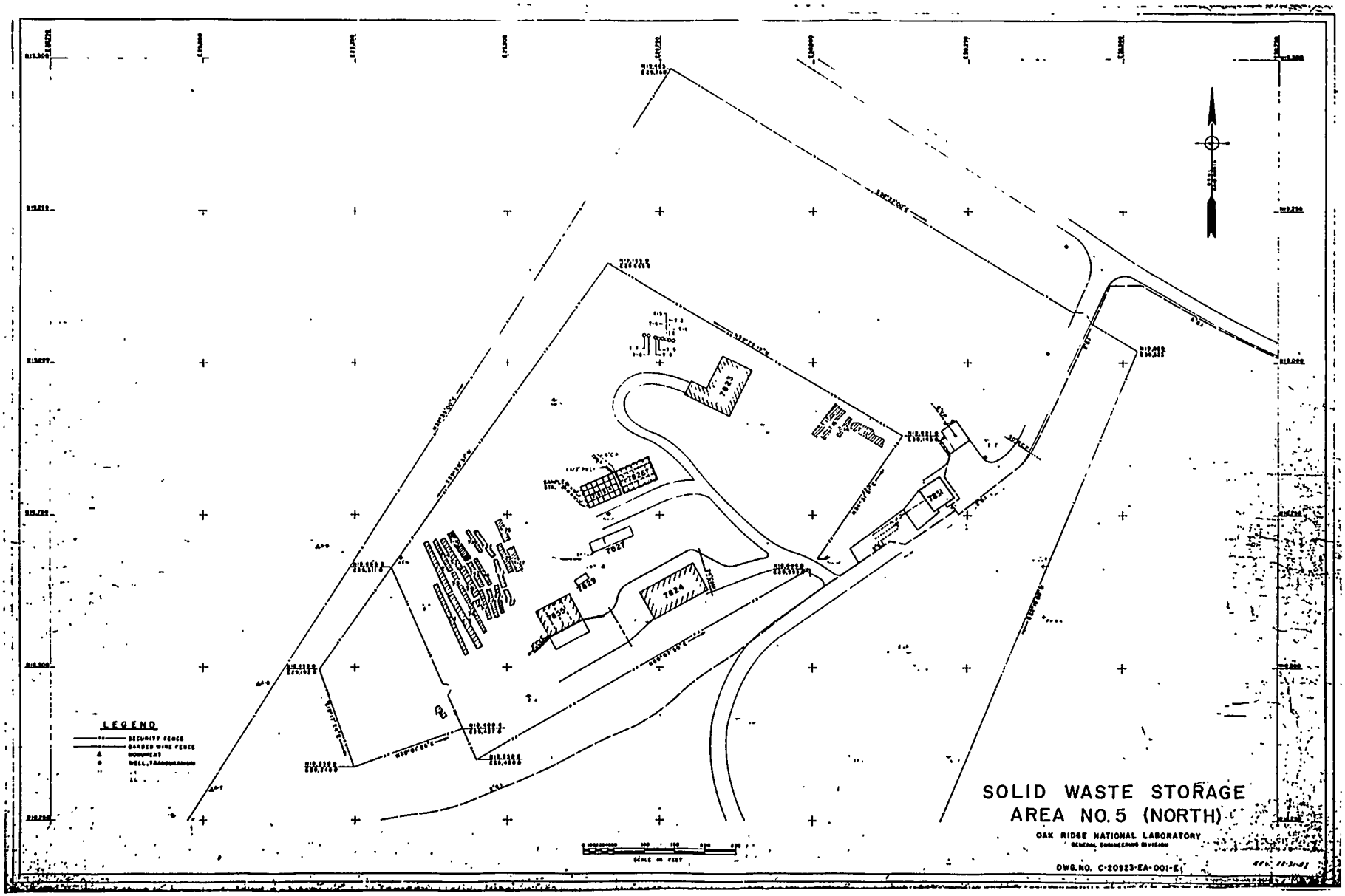

LEGEND
SECURITY FENCE
BARBED WIRE FENCE
MONUMENT
WELL, TRANSDUCAMOND
7823
7824
7827
7829
7831
SOLID WASTE STORAGE
AREA NO. 5 (NORTH)
OAK RIDGE NATIONAL LABORATORY
GENERAL ENGINEERING DIVISION
DWG. NO. C-20923-EA-001-E

Table 3. Low-level alpha wastes in retrievable storage in bldgs. 7826 and 7834 at Solid Waste Storage Area 5

Period	Volume (m^3)	Transuranium isotopes and ^{233}U (g)	Number of Stainless Steel Drums
Prior to FY 1974	124.7	2,138	282[2]
FY 1974	30.7[1]	1,176	137
FY 1975	42.1	1,761	190
FY 1976	46.3	2,352	205
Adj. Quarter FY 1976	12.9	1,053	57
FY 1977	15.5	1,294	69
FY 1978	34.1	585	151
FY 1979	32.7	818	145
FY 1980	30.0	335	133
FY 1981	35.4	814	129
FY 1982	20.3	103.4	86
FY 1983	25.0	480.6	106
FY 1984	53.3[3]	795.1	85
TOTAL	503.0	13,705.4	1,775

[1] Volumes corrected for blue boxes of low-level alpha waste buried in previous years.

[2] An additional 277 black iron drums were stored prior to FY-74.

[3] This includes 23 metal boxes stored retrievably.

The remotely handled transuranium/^{233}U waste in SWSA 5 is summarized in Table 4, including the number of concrete casks stored by fiscal year and the volume of TRU waste in the stainless-steel-lined wells. The amount (grams) of TRU/^{233}U and the total volume by fiscal year, as well as the grand total through FY 1984, are also illustrated. Values given here are summarized from the MOD 3 ORNL Radioactive Solid Waste Data File. Bates (1983) has published these values taken from this file through FY 1981.

Also located in SWSA 5/north is building 7824, a storage building for miscellaneous equipment and also currently housing "The Waste Examination Assay Facility" (WEAF). Some transuranic waste, which due to physical size or composition could not be stored in the manner described above, was disposed of by normal trench burial in the hillside just east of Building 7823 and across the access road from Buildings 7824 and 7826. However, regulations have changed and, since 1980, no TRU waste has been buried in SWSA 5.

SWSA 5/north is surrounded by a second security fence inside the primary SWSA 5 fence to further restrict access to the area. Outside the interior fence, near the main SWSA 5 entrance gate is located Bldg. 7831, the Solid Waste Compactor Facility. This facility is used for volume reduction and packaging of low-activity, nontransuranic and nonfissile, compactible waste (E. M. King 1979).

Beginning in April 1982, The WEAF was set up in a section of Building 7824 to examine the TRU contact handled waste contained in

Table 4. High-radiation level alpha wastes in retrievable below-grade storage in bldgs. 7827 and 7833, and trenches in Solid Waste Storage Area 5

Period	Total Volume (m^3)	Transuranium Isotopes and ^{233}U (g)	Volume in Stainless Steel Liners (m^3)	Number of Casks
FY 1971	2.8	12[1]	NA	-[2]
FY 1972	34.0	2,066[1]	NA	12[2]
FY 1973	124.6	889[1]	NA	44
FY 1974	82.2	140	NA	29
FY 1975	70.8	15	NA	25
FY 1976	45.3	168	0.9	15
Adj. Quarter FY 1976	22.9	34	0.2	8
FY 1977	69.0	565	1.1	24
FY 1978	40.9	151	0.8	14
FY 1979	42.5	9	0.01	15
FY 1980	25.5	9	0	7
FY 1981	25.5	2	0	9
FY 1982	31.1	1	0	11

Table 4. continued

Period	Total Volume (m^3)	Transuranium Isotopes and ^{233}U (g)	Volume in Stainless Steel Liners (m^3)	Number of Casks
FY 1983	25.5	13	0	13
FY 1984	37.1	27	0.33	9
Totals	448.6	4,101	3.34	235

[1]Quantities of isotopes present were most often given as less than the maximum quantity approved for the specific container used; best estimates are now required.

[2]Forty three casks for FY-71 and 44 casks for FY-72 are buried nonretrievably.

approximately 1800 55-gallon stainless steel and 275 mild steel drums already in storage at ORNL. This assay was to categorize waste equal to or above 100 Ci/kg TRU waste. All contact handled TRU waste drums (approximately 2400) since that time have been scanned and separated into those classes (100 μCi/kg and above or below 100 μCi/kg).

SWSA 5/south was closed in 1973 to routine trench burial, but SWSA 5/north continues to be used for storage of retrievable waste. The waste composition of buried waste in SWSA 5/south is believed to be characteristic of the heterogenous mixture described for the previous storage areas.

Normal problems caused by infiltration of precipitation were aggravated in SWSA 5 because of poor trench orientation (Webster 1976, Evaluation Research Corporation 1982). Trenches in this burial ground were excavated with their long axes in an upslope/downslope direction, paralleling the hydraulic gradient of the water table. The extent of water seepage might have been less if the long axis of the trenches paralleled the hillslope contours and crossed the hydraulic gradient; however, the trenches may have experienced considerably more sidewall collapse. Some of these trenches filled with water which seeped out the lower ends of the trenches.

In 1975, corrective actions were taken to reduce the seepage in an area found to have relatively high amounts of ^{90}Sr and measurable amounts of ^{244}Cm and ^{238}Pu (Evaluation Research Corporation 1982). Initially, about 0.6 m (2 ft.) of overburden was removed from the area overlying four of the burial trenches. Two underground dams, one of

concrete and one of bentonite-shale, were then installed across two parallel trenches. The stripped area was covered with a PVC membrane and the overburden was replaced. The surface was planted with grass to inhibit erosion. In addition to the placement of subsurface dams and a PVC membrane in SWSA 5/south, a near-surface seal consisting of a bentonite-shale mixture was placed over 14 trenches in the TRU waste area (SWSA 5/north) to prevent excessive infiltration of precipitation. Other corrective measures taken in SWSA 5 include filling of collapsed trench caps, installation of concrete drainage ditches, and surface contouring for better drainage (Tamaura et al. 1980).

Both groundwater and surface water drainage in SWSA 5 are predominantly southeast towards Melton Branch and southwest towards White Oak Creek. The water table contour map shows that the steepest gradient is in the direction of Melton Branch which implies that the prevailing movement is to the southeast. Therefore, most of the potential radionuclide transport by surface water is monitored at Station 4 on Melton Branch (Myrick 1984).

In 1964, radiochemical analyses were made on water samples collected from several wells and from the drainage that divides the site into two sections (5 south area and 5 north area). The principal contaminants found were ^{90}Sr, ^{106}Ru, ^{3}H, and trivalent rare earths. Only minor movement of ^{106}Ru, ^{137}Cs, and ^{60}Co from trenches was observed from analyzed core samples of new wells (Webster 1976, Oakes and Shank 1979).

Water samples collected from seeps from SWSA 5 in 1974 indicated that ^{90}Sr and ^{3}H were principal contaminants. The average concentration of ^{3}H in the samples was 3.9×10^{5} dpm/ml, or 0.2 mCi/ml. Water samples collected at a sampling station about 1.6 km (1 mile) downstream from the confluence of Melton Branch with White Oak Creek have indicated that several thousand curies of ^{3}H had passed that point annually since the mid-1960s. Most of the ^{3}H found at the station is believed to have been discharged to Melton Branch in groundwater from SWSA 5 (Webster 1976, Webster 1979).

Spalding and Munro (1983) found increased concentrations of ^{90}Sr in groundwater in two major areas along the perimeter of SWSA 5. One area was south of the old hydrofracture facility (Building 7860) just outside of and west of the SWSA 5 perimeter fence. Due to the directional flow path of the water in the area, it is assumed that the elevated ^{90}Sr concentrations were related to operations of the old hydrofracture facility. The second area of elevated ^{90}Sr concentrations is in a region downslope from specific trenches in the eastern portion of SWSA 5 that are known to contain a high inventory of ^{90}Sr.

5.6 Solid Waste Storage Area 6.

SWSA 6 is the site currently used for routine solid waste disposal. The site was opened in 1969; however, the first recorded disposal was in January, 1972 (Unpublished ORNL Radioactive Solid Waste Data File). The total fenced area is 27.5 ha (68 acres); but only approximately 5.9 ha (14.5 acres) of that area are usable due to the

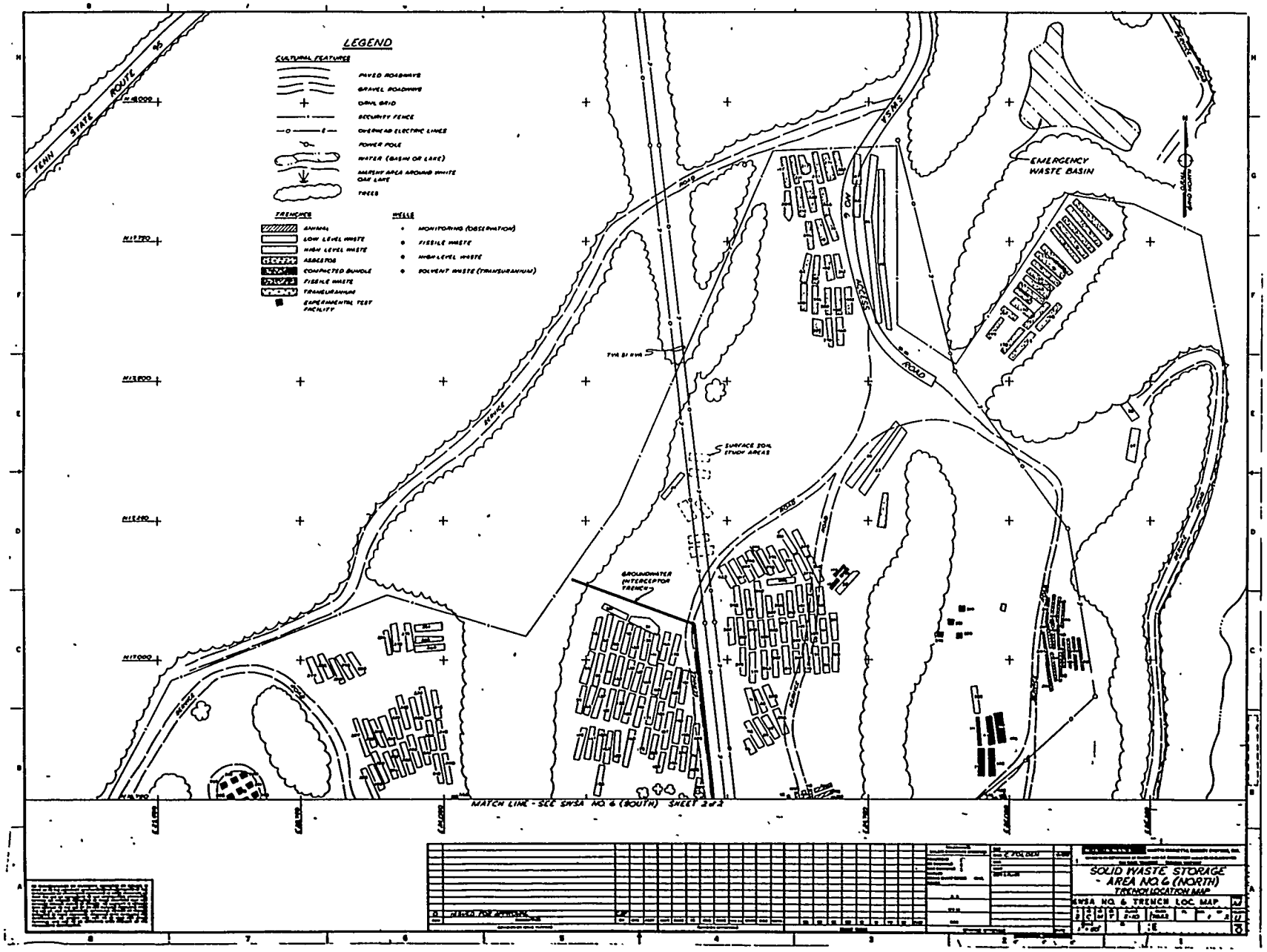
rough terrain (Webster 1976, Bates 1983, Myrick 1984). The site is located just northwest of White Oak Lake near White Oak Dam and Tennessee Highway 95. The area is generally sloping with wooded as well as cleared and grassed areas. Numerous unimproved access roads exist throughout the area. There is a heavy equipment storage building (building 7842) located in SWSA 6. Since SWSA 6 is currently in use, freshly filled trenches are found in the area. Figures 12 and 13 (ORNL DWGS-Solid Waste Storage No. 6-North and South) illustrate the topography and water table elevations of the site as well as location and alignment of existing trenches.

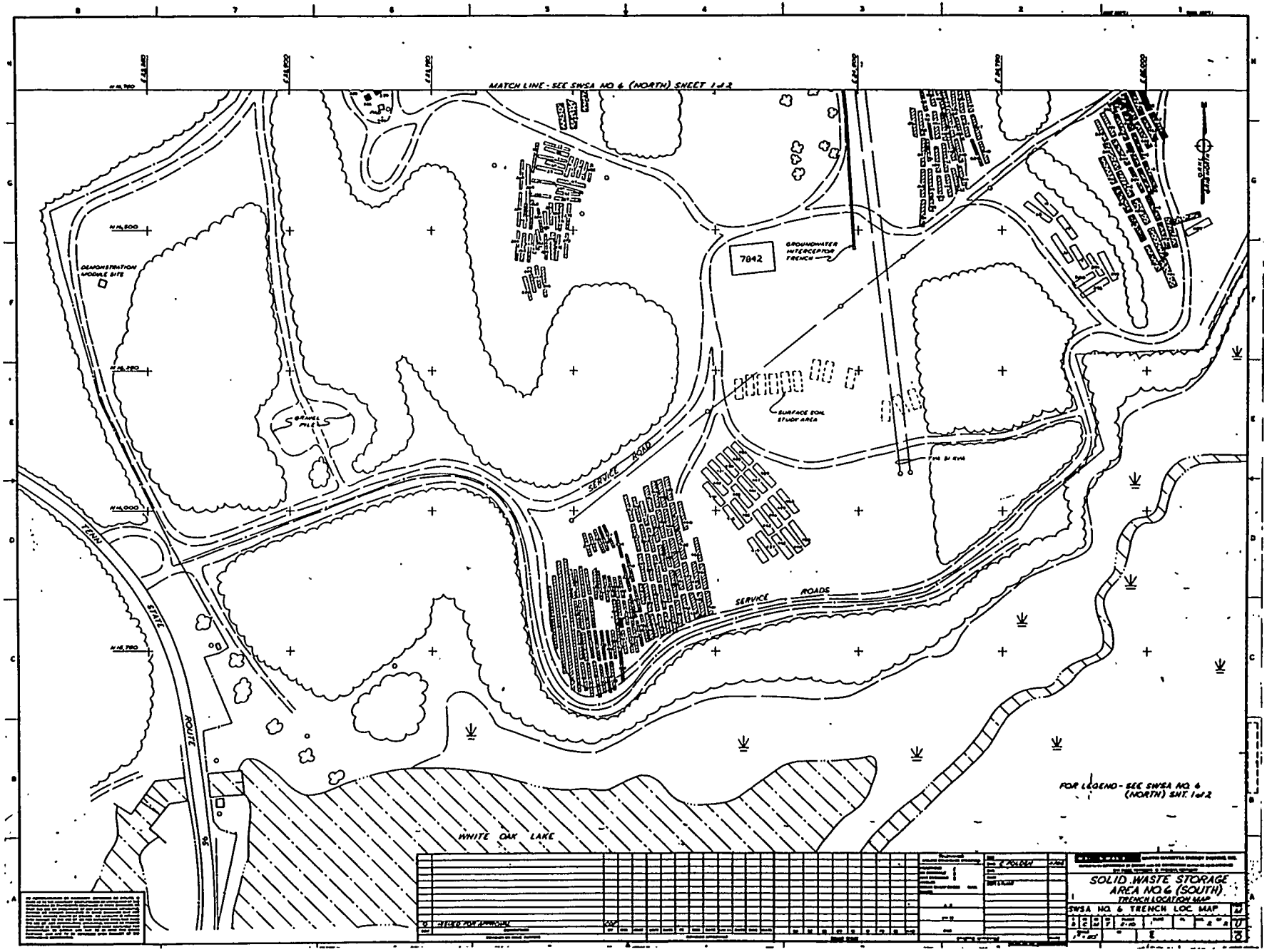
The ORNL Radioactive Solid Waste Data File (MOD 3 being more comprehensive than the two previous MODs) discussed previously was updated on April 1, 1977, and recordkeeping was made more informative. Data entries compiled by the generators are more informative and values submitted (volumes, curie content, weights, and nuclide identification) impart more knowledge of the waste being disposed.

The transuranics and ^{233}U material retrievably stored in SWSA 5 since October 1971 is also cataloged on the MOD 3 file. there is neither transuranics nor ^{233}U material stored or disposed in SWSA 6.

On May 1, 1986, in response to more stringent requirements for solid waste disposal in SWSA 6, a log-in data sheet was issued to all generators wishing to dispose of both low and high activity, non-TRU/^{233}U solid radioactive waste. This log-in sheet is required in addition to the standard 2822 request form. The log-in sheet and its instruction sheet are shown in Figures 14 and 15.

Figure 12. Location and alignment of trenches and auger holes in SWSA 6-north (from Oak Ridge National Laboratory Engineering SWSA No. 6 Trench Location Map).


LEGEND
CULTURAL FEATURES
PAVED ROADWAYS
GRAVEL ROADWAYS
CHAIN GRID
SECURITY FENCE
OVERHEAD ELECTRIC LINES
POWER POLE
WATER (BASIN OR LAKE)
MARSHY AREA AROUND WHITE OAK LAKE
TREES
TRENCHES
ANIMAL
LOW LEVEL WASTE
HIGH LEVEL WASTE
ASBESTOS
COMPACTED BUNDLE
FISSILE WASTE
TRANSURANIUM
EXPERIMENTAL TEST FACILITY
WELLS
MONITORING (OBSERVATION)
FISSILE WASTE
HIGH LEVEL WASTE
SOLVENT WASTE (TRANSURANIUM)
TENN STATE ROUTE 95
EMERGENCY WASTE BASIN
SURFACE SOIL STUDY AREAS
GROUNDWATER INTERCEPTOR TRENCH
MATCH LINE - SEE SWSA NO. 6 (SOUTH) SHEET 2 of 2
SOLID WASTE STORAGE - AREA NO. 6 (NORTH)
TRENCH LOCATION MAP
SWSA NO. 6 TRENCH LOC MAP


MATCH LINE - SEE SWSA NO. 6 (NORTH) SHEET 1 of 2
DEMONSTRATION MODULE SITE
GRAVEL PILES
7842
GROUNDWATER INTERCEPTOR TRENCH
SURFACE SOIL STUDY AREA
SERVICE ROAD
SERVICE ROADS
TENN STATE ROUTE 95
WHITE OAK LAKE
FOR LEGEND - SEE SWSA NO. 6 (NORTH) SHT. 1 of 2
SOLID WASTE STORAGE AREA NO. 6 (SOUTH)
TRENCH LOCATION MAP
SWSA NO. 6 TRENCH LOC. MAP

LOG-IN DATA SHEET FOR GENERATORS OF LLW

CERTIFICATION PROGRAM FOR LOW-LEVEL WASTES

[For each packet placed in this drum, an entry must be marked in every column (1 through 9) or the drum will not be removed for storage or disposal.]

LLW No.:

1. Packet Number	2. Date	3. Origin Bldg./Room No.	4. Radiation (mRem/h)		5. Curies Est.	6. Dominant Radionuclides	7. Physical Form	8. Chemical Form (see Note 3)	9. Signature	10. Log-In Procedures Verification	
			β,γ	Neut.						Date	Signature

NOTES:

1. Drum closure: Gasket has been put in place and closure ring bolt tightened to 65 ft-lb. Signature _____

2. Does this building handle TRU wastes? Yes_____ No _____ Signature _____

3. I have been trained in the approved procedures for exclusion of RCRA material in solid low-level waste and certify that this package was prepared in conformance with these procedures. Signature _____

This is to certify that the above named materials are properly described and packaged and are in proper condition for transportation according to applicable regulations of DOT for Type A waste.

_____ Authorized Signature _____ Title _____ Date

Figure 15. Generator instructions for filling out the "Log-in Data Sheet for Generators of LLW."

GENERATOR INSTRUCTIONS FOR FILLING OUT THE "LOG-IN DATA SHEET FOR GENERATORS OF LLW"

LLW No. Enter the bar-coded number affixed to the container (box, drum, other).

Column 1 PACKET NUMBER: Each article placed in the drum must be bagged or sealed. The first packet placed in each container should be labeled number one and each subsequent packet labeled and numbered sequentially thereafter.

Column 2 DATE: Enter the date that each packet is placed in the container.

Column 3 ORIGIN BLDG./ROOM NO: Enter the building, room number, or other information which identifies the location from which the waste originated.

Column 4 RADIATION (MREM/H); BETA, GAMMA; NEUTRON: Monitor each packet for beta and gamma radiation using a "Cutie Pie," Victoreen 440 or other comparable instrument and enter the result, if the appropriate monitor for neutrons using a BF_3 counter, fast neutron survey meter, or a comparable instrument and enter the result.

Column 5 CURIES EST.: An estimate of the curie content using the beta-gamma readings for each packet should be entered using the following table:

1 millirem/h	1.7 E-4 Ci	10 millirem/h	1.7 E-3 Ci	500 millirem/h	8.3 E-2 Ci
5 millirem/h	8.3 E-4 Ci	100 millirem/h	1.7 E-2 Ci	1 Rem/h	1.7 E-1 Ci

(Based on 1 MeV gamma)

Column 6 DOMINANT RADIONUCLIDES: List all radionuclides present starting with the dominant radionuclides first.

Column 7 PHYSICAL FORM: Acceptable entries for physical form include plastic, paper, glass, metal, soil, wood, rubber, cloth, etc.

Column 8 CHEMICAL FORM (radionuclide): Acceptable entries for chemical form include oxide, chloride, nitrate, metal oxide, elemental, inorganic salt, general chemical description, etc.

Column 9 SIGNATURE (see Note 3): The person placing each sealed or bagged article in a container will be required to put his/her signature (initials are not acceptable) in Column 9. This person must have taken and passed the training course on RCRA wastes or the waste package will not be approved for disposal or storage.

Column 10 LOG-IN PROCEDURES VERIFICATION; DATE; SIGNATURE: Verification that the log-in data sheet is being used properly must be made on a routine basis by either the QAR, RCO, GCO, HP assigned to that building. The date and legible signature (initials are not acceptable) of the person making the verification should be entered in the appropriate columns.

NOTE 1 If the waste container is a drum the head, gasket, and bolt ring must be placed on the drum and the ring bolt tightened to 65 ft-lb. The signature of the person closing the drum is required.

NOTE 2 Indicate if TRU wastes are handled in the building from which the low-level waste originates.

NOTE 3 The signature of the RCO or GDO that has taken and passed the training course on RCRA wastes is required.

SWSA PERSONNEL INSTRUCTIONS FOR CHECKING AND COMPLETING THE "LOG-IN DATA SHEET FOR GENERATORS OF LLW"

1. Check to see that a LLW No. is entered and that Columns 1 through 9 and Notes 1 through 3 are filled out completely and legibly. Check all signatures (no initials) against the master list to verify that all packers have been trained and passed the training course for RCRA wastes.
2. The SWSA foreman or SWSA supervisor must certify that the waste is in proper condition for transportation according to applicable DOT regulations for Type A packages.

Since 1982, a number of trenches (containing both high activity, low activity and biological waste) containing hazardous chemical wastes (lead, toluene and xylene) have recently been identified. Disposals containing those materials pose regulatory problems and those trench areas are currently being addressed.

Selection of the site for SWSA 6 was based on essentially the same criteria as were used in selecting the SWSA 5 site. The site is underlain by Conasauga shale and has similar hydrology to SWSA 5 (Webster 1976, Boegly 1984, Myrick et al. 1984).

Periodic water table measurements in SWSA 6 indicated the presence of groundwater in an isolated area known as the 49-Trench area. This is a small subsection of SWSA 6 containing 49 LLW disposal trenches which were filled with waste and closed in 1973-1974. In an attempt to prevent rainfall from infiltrating the cover material and collecting in the trenches, this area was sealed with a bentonite cover in 1976. The sealing procedure consisted of removing the top 2 ft. of cover material, applying bentonite at a rate of 4 lb/ft^2, disking the surface to yield a 5% bentonite-shale mixture, and compacting the seal in place. The seal was then covered with soil and seeded to prevent erosion.

Despite this cover, water was still observed to collect in the underlying trenches. Hence, a second engineered barrier was installed in 1983, consisting of a French drain designed to prevent the lateral movement of groundwater into the trenches. The drain was installed at a depth of approximately 30 ft, surrounding the group of trenches on

the north and east sides. Early monitoring activities have shown the dewatering of several trenches in close proximity to the drain, as well as a more general lowering of the groundwater table (Davis and Stanfield 1984, Davis et al. 1985).

All trenches are excavated with the bottoms being at least 0.6 m (2 ft.) above the known high water table (Radioactive Solid Waste Operational Manual 1985). Trenches in SWSA 6 are categorized into different types: low activity, high activity, biological, asbestos, compacted bale, and landfill. Unlined auger holes are also located in selected areas of SWSA 6.

5.6.1 Low Activity Trenches

Low activity trenches are used for disposing of waste with radiation readings on the container surface of less than 200 mR/h. Trench dimensions are approximately 14.8 m (50 ft.) long, 3.5 m (12 ft.) wide, and 5.7 m (18 ft.) deep. The operating time of this trench type is 6-8 weeks. As of December 31, 1984, 179 low activity trenches have been filled in SWSA 6. Material prior to June 1985 was dumped into this trench type but after that time most waste has been either packaged in plywood boxes and stacked in the trench or packaged in 55 gallon mild steel drums and stacked in a horizontal array in the trench. Contaminated dirt and concrete from excavation projects have also been dumped into selected low activity trenches.

Beginning in October 1985, in order to extend the burial life at SWSA 6, a decision to send selected drummed waste to an aboveground interim storage facility at K-25 was initiated. The drummed waste sent to K-25 had to satisfy at least three requirements: 1) radiation readings on container surface to be no greater than 10 mR/h, 2) no free liquids present [modified to accept drummed waste with less than one-tenth of one percent free liquid (100 ml)], and 3) no alpha waste. In order for this to take place, the drummed waste had to be surveyed by Environmental and Occupational Safety (E&OS) personnel, examined by the WEAF for alpha contamination, and also be scanned by a real-time radiography unit in Building 7824 to insure the absence of free liquid. Procedures are currently being developed to get the acceptable drummed waste to K-25 for storage. The unacceptable drummed waste is to be either sent back to the generator responsible for repackaging or to be set aside/stored until an appropriate disposal option has been selected; however, standard trench disposal, has been discontinued and the Greater-Confinement-Disposal (GCD) Techniques discussed in the following paragraphs are the current disposal methods. To date, neither has been done but a trial demonstration for shipment to K-25 has been made.

Currently, GCD Techniques are the disposal methods for both low activity and high activity low-level ORNL wastes at SWSA 6. Greater Confinement Disposal is to be used to assure that disposal method for low activity and high activity radioactive waste will provide long-term isolation and containment from

the surrounding environment. At present, there is no indication that past shallow-land burial techniques will be allowed, and most low activity waste other than low activity contaminated soil to be disposed will either be boxed in metal 1.2 x 1.2 x 2.1 m (4 x 4 x 7ft.) containers or in 55-gallon drums. The metal containers are to be stored on a concrete pad and 55 gallon drummed waste will be surveyed and stored until final disposition is resolved. Contaminated soil is to be disposed of by the GCD method discussed in the next paragraph.

Another method of low activity and high activity waste disposal being used at ORNL is the containment of radioactive waste within large precast, belowground, concrete cylinders. Each cylinder is 5.9 m (20 ft.) long by 2.4 m (8 ft.) inside diameter with a 0.15 m (6 inch) wall thickness. Each cylinder is buried in the vertical position, level with the ground surface. A concrete base 0.3 m (12 inch) thick with steel bar reinforcement is poured to seal the cylinder bottom. An application of tar is also poured around the inside of the cylinder base as a secondary seal. This type containment cylinder will give approximately 28.3 m^3 (1,000 ft^3) of usable burial space. Four cylinders placed side-by-side in SWSA 6 are currently in use. After each unit is filled with waste, a concrete cap, 0.6 m (2 ft.) thick, is poured over the top to assure a watertight seal and long-term waste confinement.

The use of precast concrete cylinders for waste burial containers has a secondary advantage in that it allows use of land area within SWSA 6 that could not be used for standard burial trenches due to

ground slope or depth of water table. To address the hydrological isolation of waste with the cylinders, two monitoring wells will be installed outside each cylinder to a depth several feet below the cylinder bottom. A single monitoring well will be placed inside each cylinder to rest on the concrete floor and will serve as a point to monitor for any leakage of groundwater into the cylinder of low-level waste.

5.6.2 High Activity Trenches

The high activity trench is used for disposal of waste that has a radiation reading on the surface of the container equal to or greater than 200 mR/h. This trench type is excavated in the higher elevations and, therefore, is farther away from the known high water table. Waste, after being dumped into this type of trench, is covered with enough soil, if needed, to lower the radiation reading at the trench edge to ≤100 mR/h level. The high activity trenches are excavated with dimensions being approximately 14.8 m (50 ft.) in length, 3.5 m (12 ft.) wide and 5.3 m (18 ft.) deep. The operating time of a trench of this type is approximately 6-8 months. By December 31, 1984, 42 high activity trenches had been used in SWSA 6.

Beginning on May 1, 1986, the normal procedures of trench disposal of high activity (and low activity) low-level waste were discontinued. Since that time a Greater-Confinement-Disposal (GCD) Technique has been instituted, similar to that described in the preceding section, under Low Activity Trenches.

5.6.3 Auger Holes

Unlined auger holes were used in SWSA 6 for the disposal of high activity low-level waste. These holes are excavated in the higher elevations (farther above the known high water table). These holes are 0.9 - 1.5m (3 - 5 ft.) in diameter, 5.9m (20 ft.) deep and spaced 0.9m (3 ft.) edge to edge. As of December 31, 1984, 516 auger holes have been excavated and filled.

Due to shielding and transportation requirements for shipment to SWSA 6 from the ORNL complex this high activity waste is loaded and transported via a lead lined shipping cask (carrier). Routinely 3 - 4 disposals may be made into a single auger hole. After a waste placement, if the radiation reading is >100 mR/h at the top of the hole, dirt is backfilled until a reading of ≤100 mR/h is observed. When the waste reaches approximately 1.2 m (4 ft.) from the surface, approximately 0.3 m (1 ft.) of dirt is spread on top of the waste, the hole is then capped with 0.3 m (1 ft.) of concrete and brought to ground level with additional soil.

Since May 1, 1986 there has been no unlined auger hole disposal. Greater-Confinement-Disposal (GCD) is currently being practiced for all auger hole disposals at SWSA 6. This disposal technique uses a 5.9 m (20 ft.) long, 2.4 m (8 ft.) inside diameter, precast concrete cylinder with a 0.2 (8 in.) wall thickness, buried vertically to ground level. A reinforced concrete base is poured for the bottom. As the bottom is poured, eleven 5.9 m (20 ft.) long, 0.5 m (20 in.) inside diameter cast iron pipes with 2 cm (3/4 in.) wall thickness are set in place (equally

spaced) inside the concrete cylinder and are anchored in place as the concrete dries. The space between each of the pipes inside the cylinder is later filled with concrete thereby filling all void spaces.

In order to empty a waste carrier, a steel plate (specific for each carrier type) is set in place above the hole to be used. Waste is then unloaded through the bottom discharge carrier directly into the hole. When the empty carrier is removed from the top of the hole, a 15 cm (0.5 ft.) thick concrete plug is then lowered on top of the waste. Additional disposals are made in the hole, with concrete spacers separating disposals, until the well is filled to within 1.8 m (6 ft.) from the top. A concrete plug is then poured in the remaining 1.8 m (6 ft.) to completely seal the hole. Approximately 10 disposals may be made in one hole. At present, the GCD disposal technique disposes of approximately 5 - 6 times the amount of waste as does the conventional method for the same land area used thereby greatly increasing the capacity of the present auger hole disposal area .

In order to evaluate the hydrological isolation of the GCD wells, two monitoring wells are installed outside the concrete cylinder to a depth approximately 5 meters below the cylinder bottom. Groundwater sampling will then be performed. Since these concrete cylinders are located in the same area as the unlined auger holes, and contamination from other sources may be present, a tracer (iodide, bromide or lithium) is to be used in this GCD disposal for positive radionuclide migration/leakage detection.

5.6.4 Biological Trenches

Biological trenches are used for disposal of carcasses, animal bedding, excreta, and any material associated with a biomedical laboratory, such as gloves, blotter paper, shoe covers, etc. Waste placed in the biological trench is predominately very low activity: low-level, non-alpha-contaminated waste (≤200 mR/h, most is approximately 5 mR/h).

Biological trenches are dug with dimensions approximately 2.4 m (8 ft.) wide, 2.4 m (8 ft.) deep and 14.8 m (50 ft.) long. Waste is dumped in this trench type one day a week. Five metal dividers are spaced equally (about 2.4 m apart) in the biological trench in order to conserve space due to the fact that one week's shipment is dumped between two dividers. Biological waste is covered the same day it is disposed; therefore, the dividers inhibit the cover dirt from spilling over in areas with no waste. With the use of the metal dividers in the biological trench, the trench operational life expectancy is at least six weeks. As of December 31, 1984, 185 biological trenches have been utilized in SWSA 6.

On May 1, 1986, trench disposal of biological waste was discontinued. Currently, biological waste is sent to the University of Tennessee to be incinerated. There will be no more shallow land burial of biological waste at SWSA 6.

5.6.5 Asbestos Trenches

Asbestos, both radioactively contaminated and non-contaminated, is disposed of in SWSA 6 in an asbestos trench that has the same dimensions as the biological waste trench. Metal dividers are used here also. Asbestos disposal takes place approximately once every 3 - 4 weeks, and the waste is covered with dirt that same day. As of December 31, 1984, 30 asbestos trenches have been filled in SWSA 6.

Non-contaminated asbestos disposal continues at SWSA 6 since asbestos must be disposed of in a secured, fenced area. However, since May 1, 1986, contaminated asbestos has been sent to the Y-12 Plant for disposal; contaminated asbestos disposal is no longer permitted in SWSA 6.

5.6.6 Compactible-Bale Trenches

In 1978, a baler was installed at SWSA 5 (Bldg. 7831) to compact non-alpha, radioactively contaminated software (gloves, shoe covers, blotter paper, etc.). This waste is segregated at the generator site and placed into plastic garbage bags and surveyed by E&OS personnel to insure the radiation reading at contact on the outside of the bag is ≤200 mR/h (most compactible waste has radiation readings of ≤10 mR/h).

Approximately 15% of the waste generated at ORNL is compacted (about 280 m^3/y). After compaction, the volume reduction ratio is nearly 8 to 1. The compactor is operated about twice a month. The waste is compacted into a rectangular 0.4 m^3 (15 $ft.^3$) bale weighing approximately 295 kg (650 lbs.). The compacted bales are

placed/stacked three deep and three wide in a trench with dimensions 3.5 m (12 ft.) wide, 14.8 m (50 ft.) long and 3 - 3.5 m (10 - 12 ft.) deep. As of December 31, 1984, there have been 6 bale trenches utilized at SWSA 6.

Since the first of May, 1986, no baled waste has been buried at SWSA 6. There is to be no more shallow land burial of baled waste. This baled waste is currently being stored above ground until an alternate disposal option has been selected for this waste type.

5.6.7 Landfill Area

At ORNL, there is a category of waste called "suspect" or "low-hazard contaminated waste". This waste type has no measurable contamination, but has been judged by E&OS and the generators to have come from a contamination area or to be such that all areas cannot be surveyed (such as inside a pipe). This material cannot have the ORNL "Health Physics Material Transfer Clearance Tag" (called green tag) affixed to it. The E&OS personnel assign a "Health Physics Restriction Tag" (brown tag) to this material. This brown tagged material is disposed of in an area in SWSA 6 that is unuseable for trench space and is called the Landfill Area (approximately 2 ha). This disposal method helps to conserve more valuable potential trench space.

The landfill waste is periodically covered with soil. In October 1985, a shredder was installed at the upslope end of the landfill area and low-hazard/suspect landfill material is reduced in volume by shredding and then dumping into the landfill area. The volume

reduction of the suspect material should prolong the lifetime of the landfill area. Currently, approximately 140m^3/y (5,000 ft^3/y) of suspect material is disposed of in this manner.

5.6.8 SWSA 6 Radionuclides

As stated previously, the ORNL Radioactive Solid Waste Data File MOD 3 (became active April 1, 1977) gives more information than do the two preceeding data files (MOD 1 and MOD 2). Boegly's (1984) Site Characterization Data Inventory sumarizes volumes (m^3) and radionuclide content (Ci). Table 5 gives the combustible, non-combustible, and total volumes of radioactive solid waste disposed in SWSA 6 by fiscal year and also for the total time period (1-1-72 through 9-30-84).

Table 6 (Boegly, 1985) gives the radionuclide inventory (Ci) derived from the MOD 3 file (from 4-1-77 through 9-30-84). The data file has an "unidentified" category which is also listed.

5.7 Other Estimates of Radionuclides Buried/Stored at ORNL in SWSAs

Beginning in 1977 at the request of the SWIMS program, annual estimates of activity (Ci), volume (m^3), weight (kg), associated with the SWSAs open at that time were given. These values were derived by Mr. John M. Miller, Waste Management Section of the Operations Division, ORNL. Bates (1983) published values through 1981, and values from 1982 through 1984 were derived from the ORNL Radioactive Solid

Table 5. Volume of low-level waste in Solid Waste Storage Area 6

Fiscal Year	Total volume (ft^3)	Total volume (m^3)	Combustible volume (ft^3)	Combustible volume (m^3)	Noncombustible volume (ft^3)	Noncombustible volume (m^3)
1972	29,600	840	(Data not available)			
1973	102,800	2,910	(Data not available)			
1974	113,500	3,200	(Data not available)			
1975	106,700	3,020	(Data not available)			
1976	91,900	2,600	(Data not available)			
1976A[a]	25,200	720	(Data not available)			
1977	102,400	2,890	(Data not available)			
1978	80,100	2,270	29,200	830	50,900	1,440
1979	67,800	1,920	15,100	430	52,700	1,490
1980	72,600	2,060	18,500	520	54,100	1,540
1981	57,700	1,630	17,900	510	39,800	1,120
1982	50,800	1,440	18,300	520	32,500	920
1983	64,100	1,810	15,800	450	48,300	1,360
1984	61,700	1,750	27,100	770	34,600	980
Total	1,026,000	29,060				

[a]From 1969 to 1976 the fiscal year covered the period July 1 to June 30; starting in 1976 the fiscal year (1977) covered the period October 1 to September 30. A transition quarter (termed 1976A in this table) is included to complete the records for SWSA 6. (Boegly 1984).

Table 6. Radionuclide inventory for period covering 1977-1984 in Solid Waste Storage Area 6 (Boegly 1985)

	Activity in Curies		
Radionuclide	Trenches	Wells or Auger Holes	Totals
Am-241	3.44	1.50E-04	3.44
Am-243	4.00E-05	1.06E-05	5.06E-05
Be-10	0.213	400.	400.
Bk-249	9.99E-07	0.000	9.99E-07
C-14	258.	9.99E-07	258.
Cd-113M	0.002	3.00	3.00
Cf-252	0.009	0.000	0.009
Cm-242	4.99E-06	0.000	4.99E-06
Cm-244	6.08	0.000	6.08
Co-60	9950.	22200.	32200.
Cs-134	4.42	0.000	4.42
CS-137	645.	4460.	5110.
Eu-152	7.12	50900.	50900.
Eu-154	3.34	72600.	72600.
Eu-155	0.0500	31300.	31300.
Fe-55	0.755	10.0	10.8
H-3	4320.	2790.	7110.
Na-22	0.541	0.000	0.541
Ni-63	0.0001	0.000	0.001

Table 6. Continued.

Radionuclide	Activity in Curies Trenches	Wells or Auger Holes	Totals
Np-237	0.000449	2.99E-07	0.00045
Pm-147	59.7	85.0	145.
Pu-238	0.027	0.00189	0.029
Pu-239	0.135	1.03E-05	0.141
Pu-242	0.005	0.000	0.005
Ru-106	10.0	0.000	10.0
Sm-151	3.25	200.	203.
Sn-121M	0.500	9.00	9.50
Sr-90	251.	2720.	2970.
Tc-99	36.2	0.612	36.8
Th-232	2.46	0.083	2.54
U-233	227.	0.000	227.
U-235	5.56	0.015	5.57
U-238	205.	0.445	205.
Zr-93	5.11	18.5	23.6
Unid	1650.	5690.	7340.
Totals	17,500	193,000	211,000

Note: Numbers do not add due to rounding and method of data entry.

Waste Data File (Table 7). This table shows approximately 602,258 Ci in 202,583 m^3 were stored or buried.

As results from Table 7 indicate there are overlaps in SWSA usage but ranges may be derived for each site. In addition, waste has been relocated among the SWSAs. Examples are 1) SWSA 2 material being exhumed and sent to SWSA 3 for disposal and 2) SWSA 3 aboveground material being relocated to SWSA 6 trenches and to SWSA 4 extension).

5.8 White Wing Road Storage Area

The White Wing Road Storage Area (WWRSA), sometimes called the White Wing Scrapyard, is located at the west end of the East Fork Ridge between highway 95 (White Wing Road) and the Oak Ridge turnpike. Figure 16 shows the WWRSA location. This area which covers approximately 10.1 ha (25 acres), was utilized mainly in the early 1950's for the storage of contaminated materials (equipment, tanks, and trucks) from the three Oak Ridge facilities (ORNL, Y-12 and K-25) (Personal communication with J. Gissel, Operations Division, X-10, now retired). The Oak Ridge National Laboratory (X-10) stored material in the portion nearest to McNew Hollow and farthest from the Oak Ridge turnpike. The Oak Ridge Gaseous Diffusion Plant (K-25) used the section farthest from McNew Hollow and closest to the turnpike, while Y-12 disposed of waste midway between the X-10 and K-25 disposal areas (Memorandum dated 2-7-66 to J. A. Elkins, Townsite, from H. E. Seagren, Plant and Equipment, ORNL).

Table 7. Estimates of total annual activity, volume, and weight of solid waste buried or stored at ORNL

Fiscal Year	Activity (Ci)	Volume (m^3)	Weight (kg)	SWSA
1943	2,000	700	1.4×10^5	1
1944	2,000	700	1.4×10^5	1,2
1945	2,000	700	1.4×10^5	2
1946	2,000	700	1.4×10^5	2,3
1947	10,000	3960	1.4×10^5	3
1948	10,000	3960	1.4×10^5	3
1949	10,000	3960	1.4×10^5	3
1950	10,000	3960	1.4×10^5	3
1951	10,000	3960	1.4×10^5	3,4
1952	10,000	5660	1.4×10^5	4
1953	10,000	5660	1.4×10^5	4
1954	10,000	5660	1.4×10^5	4
1955	10,000	5660	1.4×10^5	4
1956	10,000	5660	1.4×10^5	4
1957	20,000	9060	1.8×10^6	4
1958	20,000	9060	1.8×10^6	4
1959	20,000	9060	1.8×10^6	4,5

Table 7. continued

Fiscal Year	Activity (Ci)	Volume (m^3)	Weight (kg)	SWSA
1960	20,000	9060	1.8×10^6	5
1961	20,000	15000	2.7×10^6	5
1962	30,000	12000	2.3×10^6	5
1963	20,000	9430	1.8×10^6	5
1964	20,000	9090	1.8×10^6	5
1965	10,000	5350	9.1×10^5	5
1966	10,000	4500	9.1×10^5	5
1967	10,000	5640	9.1×10^5	5
1968	20,000	6850	1.4×10^6	5
1969	10,000	5440	9.1×10^5	5
1970	10,000	3630	4.5×10^5	5
1971	10,000	4730	1.0×10^6	5
1972	9.000	3650	8.6×10^5	5,6
1973	8,000	3030	7.1×10^5	5,6
1974	8,000	3400	7.0×10^5	5,6
1975	1,200	3170	6.4×10^5	5,6
1976*	7,000	3540	6.8×10^5	5,6
1977	4,046	2200	1.7×10^5	5,6

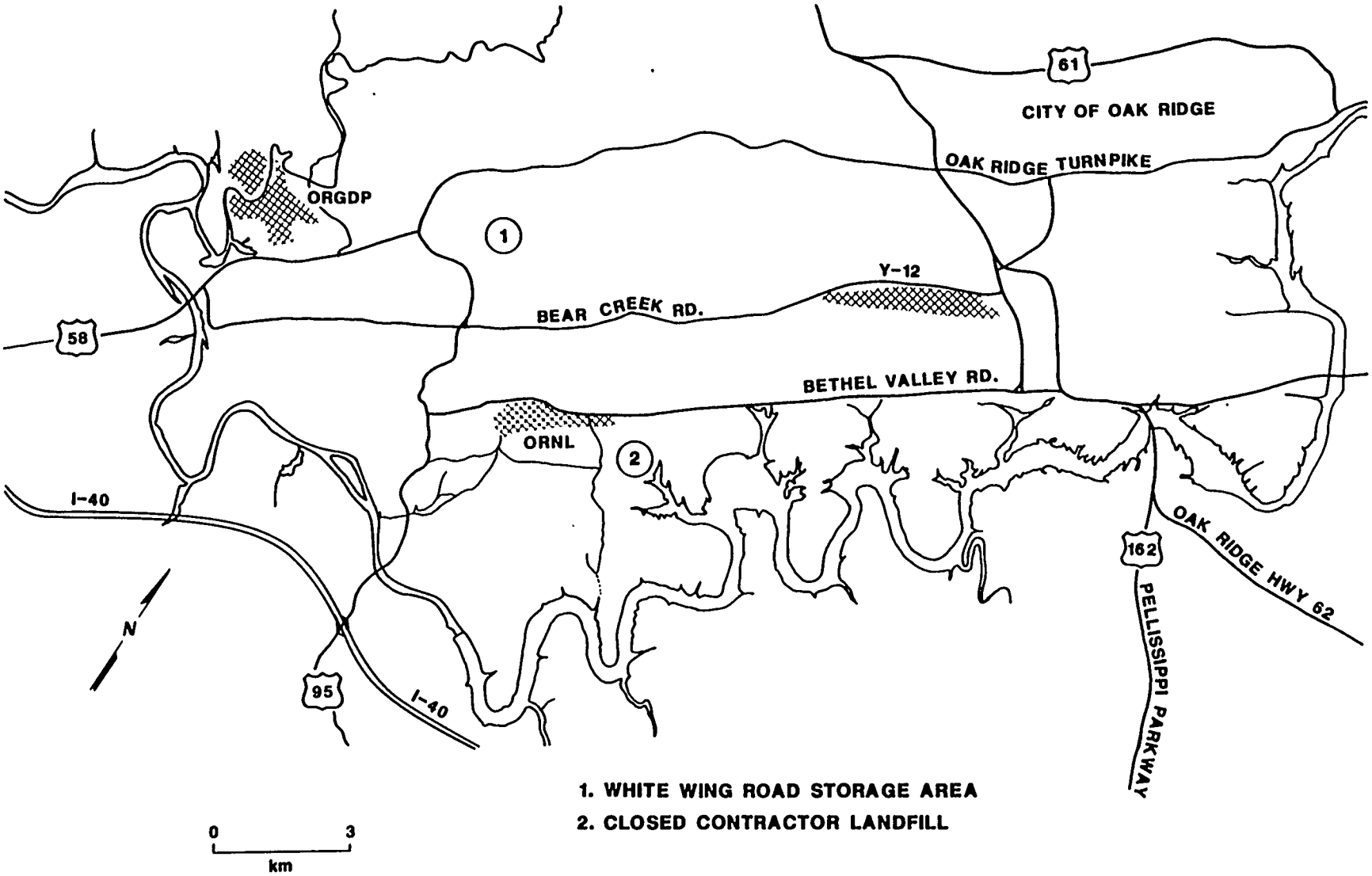
Table 7. continued

Fiscal Year	Activity (Ci)	Volume (m^3)	Weight (kg)	SWSA
1978	3,810	2360	4.2×10^5	5,6
1979	5,398	2110	6.1×10^5	5,6
1980	58,400	2350	6.0×10^5	5,6
1981	113,700	1800	7.6×10^5	5,6
1982	7,905	1491	3.7×10^5	5,6
1983	6,299	2238	7.6×10^5	5,6
1984	11,500	2444	7.0×10^5	5,6
TOTALS	602,258	202,583	3.223×10^7	

*July 1, 1975 through September 30, 1976 - Reflects change in fiscal year to begin in October.

Figure 16. Location of White Wing Road Storage Area and the Closed Contractor Landfill on the Oak Ridge Reservation.

ORNL-DWG 86-12130



1. WHITE WING ROAD STORAGE AREA
2. CLOSED CONTRACTOR LANDFILL

Some of the material stored at the WWRSA was suspected to be contaminated with plutonium. An estimate of the amount of ^{239}Pu that was in or on the material which came from ORNL was to be no more than 25 grams (Memorandum to D. M. Davis, ORNL, from Roy L. Clark, ORNL, dated 6-24-71).

The WWRSA was closed in 1964 and various clean-up activities took place between 1966 and 1971. A contract was let in 1966 with David Witherspoon, Inc. of Knoxville to haul off selected material to his place of business in Knoxville and to SWSA 6 for disposal. Also about 4590 m^3 (6,000 yds^3) of contaminated dirt were scraped from the site and sent to SWSA 5 for diposal. Clean, noncontaminated dirt was then brought in and the area was restored to original grade (Letter to T. E. Myrick, Operations Division, X-10, from J. G. Rogers, K-25, dated 11-8-85).

The site is abandoned and there has been no upkeep/maintenance of the area. The site is in the "Old Field" succession stage with cedars and pines being the predominant overstory vegetation. There is still some scrap metal, concrete and other trash at the WWRSA. In 1974 an aerial radiation survey, 1 m above ground, indicated the following; a) 0.8 to 6 mR/h gamma exposure rate and b) 0.5 to 4 mR/h for ^{137}Cs (Memorandum to D. M. Davis, ORNL, from Roy L. Clark, ORNL, dated 6-24-71).

There have been no additional environmental monitoring or sampling activities conducted in the WWRSA, but plans to do so are underway.

5.9 Closed Contractor Landfill

This area, located along Melton Valley Access Drive, was used to bury general, noncontaminated, construction debris generated by construction contractors performing work at ORNL. The landfill is only approximately 1.2 ha (3 acres) in size and its location is presented in Figure 16. The landfill was opened either in late 1974 or early 1975 and was closed around mid 1976, when the currently operated contractor landfill (located southeast of the intersection of Bethel Valley Road and highway 95) was opened (Clyde Thomas, P & E foreman, ORNL, personal communication).

The volume and nature of waste buried is unknown, other than the intent for which it was designed (see above). No waste-specific records were kept on the landfill operation. The site is not fenced but has been graded level and seeded with grass. The area is inactive and no radioactive contamination has been found [information from EPA Form T2070-3(10-79), 10 pages filled out by ORNL Department of Environmental Management].

Plans are underway to perform environmental monitoring or sampling at this site.

6. CONCLUSIONS

Since 1943, over 600,000 Ci and 200,000 m^3 of radioactive solid waste have been stored/buried in ORNL burial grounds. Very little is known about the precise radionuclide content or the total volume of the

waste from the time period of 1943 through 1961. Some information has been gathered for the time period of 1962 through March 1977. The greatest amount of information on storage/burial activities has been from April 1977 to date. One exception is that retrievably stored TRU waste data since October 1970 are more detailed.

It is difficult, if not impossible, to account for all of the radioactive waste by general inventory. Uncertainties noted here are compounded by changes over the years at the SWSA sites. Materials have been moved around, certain radionuclides have decayed naturally to much lower levels over the time span, and others may have leaked to the surrounding environment in significant quantities.

At present, generators are required to submit more accurate waste information than ever before and Waste Management personnel are closely screening every package of waste sent to the SWSAs for burial/storage. Waste Management personnel are also instituting greater confinement disposal techniques (GCD) to storage/burials. Current and future disposal regulations/techniques will result in more reliable estimates of waste inventories.

7. SUMMARY

Current regulations demand accurate information on radioactive solid waste disposed/stored at ORNL. From 1943 until 1962 no disposal records were kept. In 1962 a computer recordkeeping program began and its most advanced program (MOD 3) was initiated on April 1, 1977, and continues today. As of May 1, 1986, due to very stringent nuclide and

hazardous chemical data requirements imposed on generators wishing to dispose or store radioactive solid waste, a new data log-in sheet is also required which should give the data inventory a high degree of reliablilty.

Numerous monitoring wells have recently been and continue to be installed in order to monitor radionuclide concentrations and leaching/migration patterns. Boegly (1984) compiled site characterization data for SWSA 6 and similar data plans for the other sites discussed in this report may be appropriate in order to plan for remedial actions and eventual closure.

With the greater-confinement-disposal techniques currently being applied to all waste disposal at SWSA 6, leaching/migration problems should be minimized. By evaluating alternatives to shallow land burial and applying those acceptable, current disposal problems may, in the future, be eliminated.

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ORNL/RAP/LTR-86/14

DATE: July 15, 1986

SUBJECT: Inventory of ORNL Remedial Action Sites:
3. Process Ponds

TO: T. E. Myrick

FROM: F. G. Taylor

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LETTER REPORT RAP86-14

Inventory of ORNL Remedial Action Sites: 3. Process Ponds*

F. G. Taylor

Environmental Sciences Division
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37831

July 15, 1986

Operated By

Martin Marietta Energy Systems, Inc.
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SUMMARY

The Process Pond sites, identified as remedial action sites of the Environmental Restoration and Facilities Upgrade Program (ERFU), have been visited and the activities concerning each site reviewed by personal interview with persons knowledgeable about the site. Additional information was abstracted from reports, publications, and engineering drawings. Results of each review are presented to provide information on the pond location and description, history, contaminants, extent of contamination, major references (reports, drawings, etc.), and identification of knowledgeable persons concerning the site.

1. OBJECTIVES

The primary purpose of the remedial action task is to provide corrective measures and facility decommissioning necessary to place the Oak Ridge National Laboratory (ORNL) in compliance with applicable environmental and health regulations for radioactive and hazardous chemical wastes. Individual sites will be managed in accordance with priorities prescribed to control the potential for human exposure and relative magnitude of ecological impacts. Remedial action is a component of the Laboratory's Environmental Restoration and Facilities Upgrade Program (ERFU). In the program, remedial actions will be conducted at locations where past management practices or research activities have resulted in contamination of facilities to the environment and consequently require decommissioning or corrective actions. Remedial actions are divided into two activities: (1) Site Decommissioning Program (Surplus Contaminated Facilities Program-SCFP and Surplus Facilities Management Program-SFMP) and (2) Site Corrective Measures Program (SCMP).

The Site Decommissioning Program is designed for the management of surplus facilities (primarily structures), contaminated with radioactive materials and hazardous chemicals, from the end of their operating life until final facility disposition. The Site Corrective Measures Program is designed for the management of inactive sites (process ponds, spill sites, environmental research areas), contaminated with radioactive materials and hazardous wastes, leading to the eventual stabilization of these sites in accordance with environmental regulations.

Remedial Action Sites at Oak Ridge National Laboratory (ORNL) comprise 13 distinct areas among the several programmatic categories: (1) Site Corrective Measures Program-SCMP, (2) Surplus Contaminated Facilities Program-SCFP, and (3) Surplus Facilities Management Program-SFMP. In a series of reports, an inventory of each area is presented. The summary is the product of pertinent information abstracted from readily available literature and personal interviews with individuals knowledgeable of the site. The information includes site location and description, history, contaminants and their characteristics, extent of contamination, regulatory status, lists of pertinent drawings, photographs, reports concerning the site, and individuals knowledgeable of the site and site activities. The subject of this report is the numerous process ponds that are currently being utilized for waste holding or processing, or have the potential for future waste handling, or have contained contaminated liquids in the past. The format (order of reports) follows the Compendium of ORNL Remedial Action Program Sites as presented by Grimsby (1986).

2. BACKGROUND

There are 25 ponds at ORNL, 14 of which are expected to require remedial actions to place ORNL in compliance with applicable environmental and health regulations for radioactive and hazardous chemical wastes (Cannon 1985). The remainder are not expected to require remedial action because of their known use. A summary, describing the current status of each pond and the programmatic

responsibility for conducting corrective measures, is presented in Table 2.1

Among the process ponds, ten are being considered for remedial action under the SCMP (Fig. 2.1), whereas the remainder are a part of the Surplus Facilities Management Program. Myrick et al. (1984) has provided a general overview of the ponds. The ten SCMP ponds are located in both the Bethel and Melton Valley areas at ORNL with the majority of the ponds in two central areas. Within Bethel Valley, the Equalization Basin (Site 3524), 190 Ponds (Sites 3539 and 3540), the 3512 Pond, and the East Sewage Plant Lagoon are situated on the south side of the main ORNL complex between White Oak Avenue and the White Oak Creek (Fig. 2.2). These ponds vary in design, purpose, size, and residual activity levels as noted in the summary (Table 2.2). The Equalization Basin and 190 Ponds are active parts of the ORNL radioactive waste management system, being utilized for holdup of process liquid waste prior to treatment through the Process Waste Treatment Plant (Building 3544) or discharge to the creek. Radiological characterization studies at the Equalization Basin have identified a significant layer of contaminated sediment on the bottom of the pond, resulting from the 40 years of active service. This open basin is unlined except for the natural clay bottom and rests in part on a rock base. The 190 Ponds are also open with a compacted clay liner and gravel riprap. These two ponds receive only the slightly or uncontaminated process liquid waste from the 4500 complex, and as such contain very little activity in either the holdup liquid or the thin layer of basin sediment. The inactive 3512 Pond was a catch basin for

Table 2.1 Summary of ORNL ponds expected to require remedial action.

Pond	Site	Status/design	Programmatic responsibility
Equalization	3524	Active/Waste Management	SCMP[a]
190 Process Waste Pond	3590	Active/Waste Management	SCMP
190 Process Waste Pond	3540	Active/Waste Management	SCMP
ewage Lagoon, East Basin	2521 area	Active/Waste Management	SCMP
HFIR Waste Basin	7905	Active/Waste Management	SCMP
HFIR Waste Basin	7906	Active/Waste Management	SCMP
TRU Waste Basin	7907	Active/Waste Management	SCMP
TRU Waste Basin	7908	Active/Waste Management	SCMP
Process Waste Sludge Basin, SWSA 5	7847	Inactive/Waste Management	SCMP
Abandoned Filled-In Waste Basin	3512	Inactive/Waste Management	SCMP
Waste Holding Basin	3513	Inactive/Waste Management	SFMP[b]
Old Hydrofracture Pond	7852 area	Inactive/Waste Management	SFMP
HRE Pond	7500 area	Inactive/Waste Management	SFMP
LITR Pond	3075	Inactive/Waste Management	SFMP

[a]SCMP refers to Site Corrective Measures Program.
[b]SFMP refers to Surplus Facilities Management Program.

Fig. 2.1 Location map for ORNL Site Corrective Measures Program (SCMP) sites. Source: Myrick et al. 1984.

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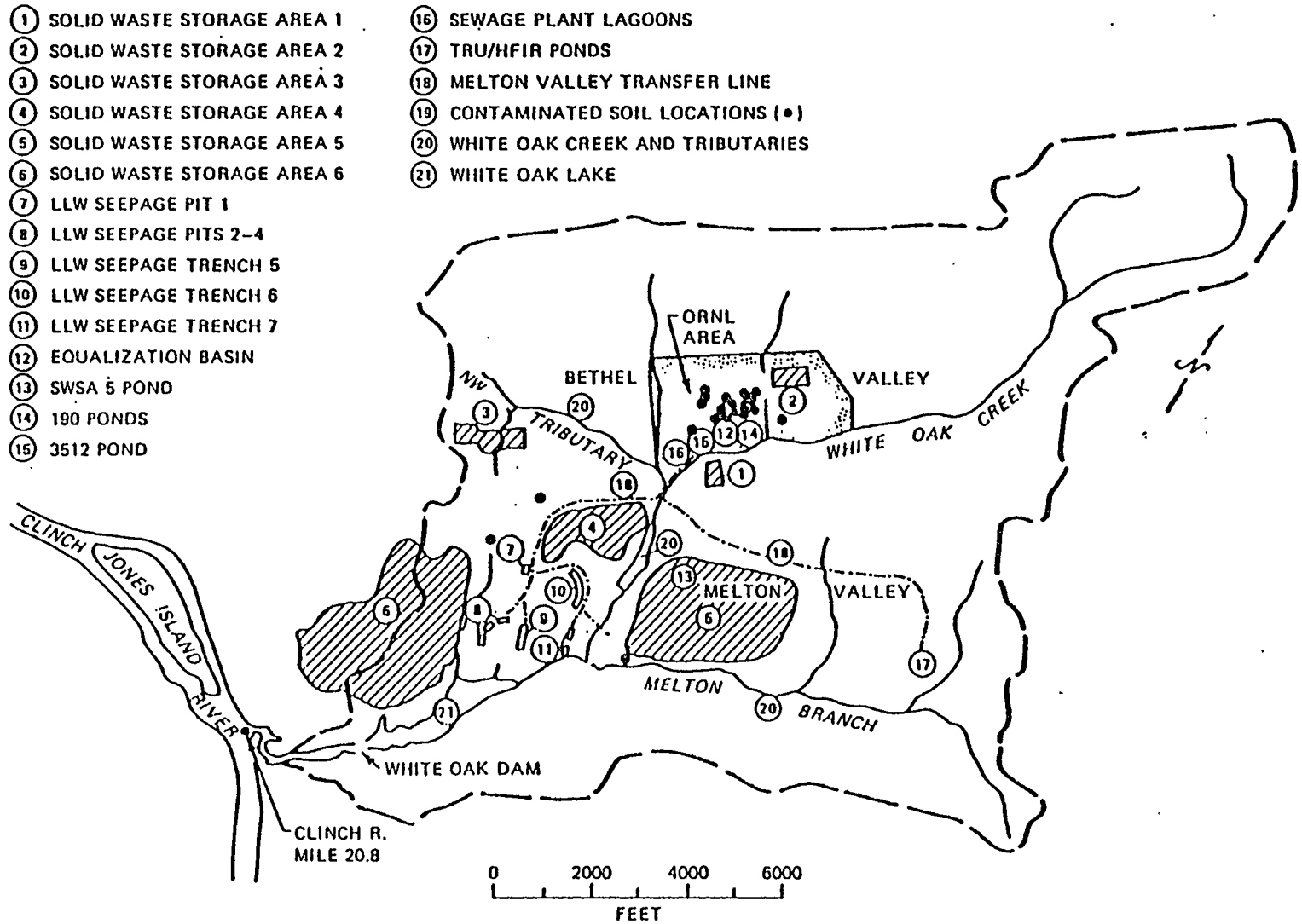

Fig. 2.2 Detailed map of SCMP process ponds in the central ORNL complex. Source: Myrick et al. 1984.

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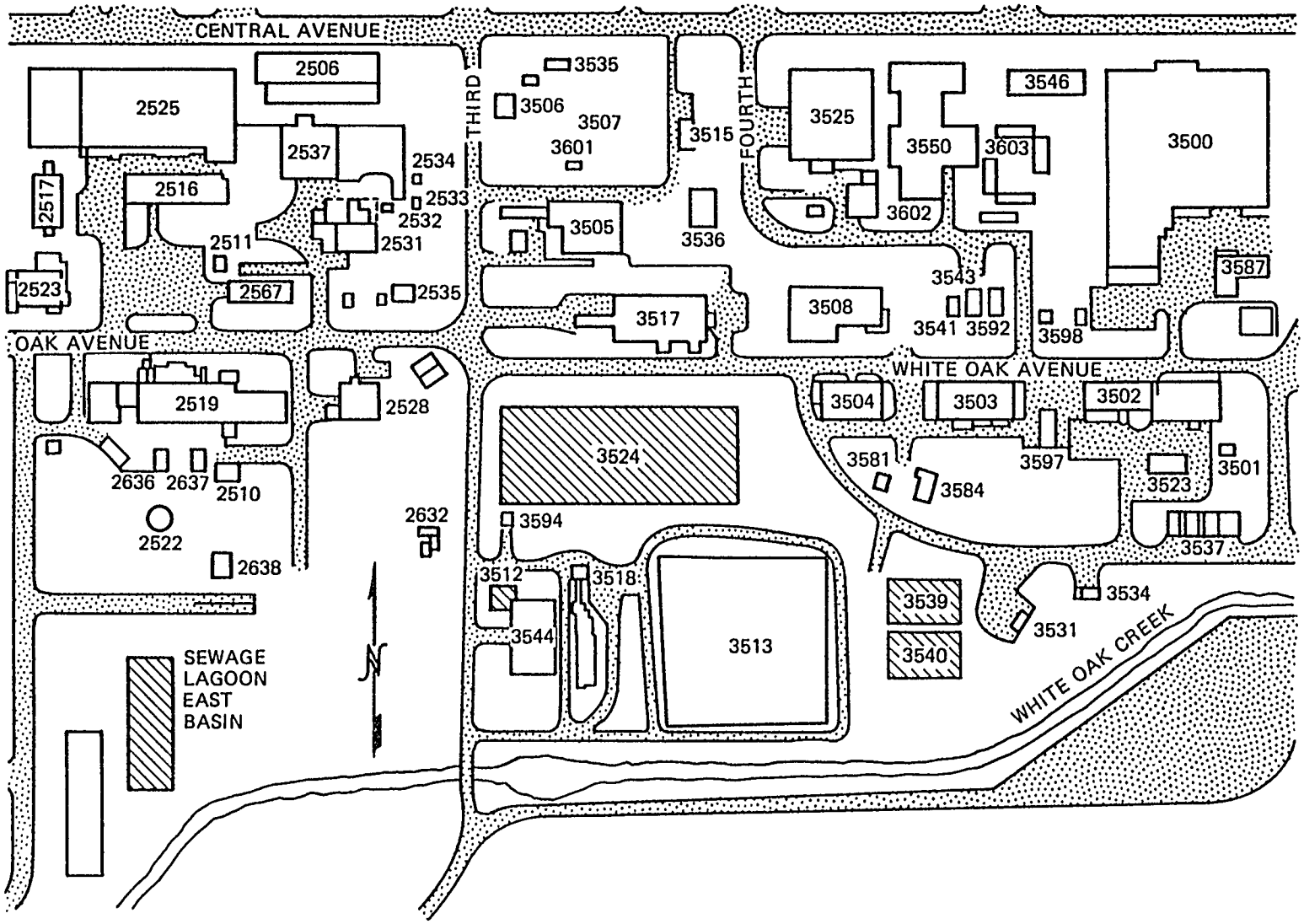

Table 2.2 Information summary of ORNL Remedial Action Pond Sites.

Site	Location	Status	Design	Service	Size and capacity	Estimated radioactivity	Reference
Equalization Basin (3524)	Bethel Valley	Active	Holdup of waste prior to feed to process waste treatment plant.	1945-present	$1x10^6$ gal. (~$3.8x10^6$ L)	150 Ci	Myrick et al. (1985)
190 Process Waste Ponds (3539, 3540)	Bethel Valley	Active	Collection of process waste from 4500 area.	1964-present	20x28 m $1.5x10^5$ gal. (~$5.7x10^5$ L)	<10 Ci	Myrick et al. (1985)
Sewage Lagoon East Basin	Bethel Valley	Inactive/ Standby	Sanitary waste.	1976-1985	15x63 m $1x10^6$ gal. (~$3.8x10^6$ L)	<10 Ci	Myrick et al. (1985)
HFIR Waste Basin (7905)	Melton Valley	Active	Basin for process liquid waste from HFIR and cooling tower blowdown.	1965-present	26x35 m $2.4x10^5$ gal. (~$9x10^5$ L)	<10 Ci[a]	Myrick et al. (1985)
HFIR Waste Basin (7906)	Melton Valley	Active	Basin for process liquid waste from HFIR and TRU.	1965-present	51x35 m $5x10^5$ gal. (~$1.89x10^6$ L)	<10 Ci[a]	Myrick et al. (1985)
TRU Waste Basin (7907)	Melton Valley	Active	Basin for process liquid waste from TRU facility.	1965-present	12x24 m $5x10^4$ gal. (~$1.89x10^5$ L)	<10 Ci[a]	Myrick et al. (1985)
TRU Waste Basin (7908)	Melton Valley	Active	Basin for process liquid waste from TRU facility.	1965-present	12x24 m $5x10^4$ gal. (~$1.89x10^5$ L)	<10 Ci[a]	Myrick et al. (1985)
Process Waste Basin, SWSA 5 (7835)	Melton Valley	Inactive	Basin for sludge from former Process Waste Treatment Plant.	1976-1981	24x25 m $2x10^5$ gal. (~$7.6x10^5$ L)	50 Ci	Myrick et al. (1985)

Table 2.2 (CONTINUED)

Site	Location	Status	Design	Service	Size and capacity	Estimated radioactivity	Reference
Abandoned Filled-In Waste Basin (3512)	Bethel Valley	Inactive	Retention pond for storage tank drainage.	1944-1950	15x15 m $3x10^4$ gal. (~$1.13x10^5$ L)	<10 Ci	Myrick et al. (1985)
Waste Holding Basin (3513)	Bethel Valley	Inactive	Radioactive chemical wastes from ORNL laboratory floor drains, supernatant from gunite tanks.	1944-1976	67x67 m $1.6x10^6$ gal. (~$6x10^6$ L)	<250 Ci	Stansfield and Francis (1985)
Old Hydrofracture Pond (7852 area)	Melton Valley	Inactive	Emergency retention pond for hydrofracture grout and equipment washdown waste.	1963-1980	6x30 m $1x10^5$ gal. (~$3.8x10^5$ L)	~550 Ci	Reed (1984a)
HRE Pond (7500 area)	Melton Valley	Inactive	Retention pond for shield water from the reactor shield tank.	1958-1961	23x24 m $3x10^5$ gal. (~$1.13x10^6$ L)	10 to 50 Ci	Reed (1984b)
LITR Pond (3075)	Bethel Valley	Inactive	Reactor process water and spill wash-off.	1951-1954	2.4x14 m $1.8x10^4$ gal. (~$6.8x10^4$ L)	<1 Ci	Rohwer (1985)

[a]Combined radioactivity of basins in TRU/HFIR area (7905, 7906, 7907, 7908).

liquid wastes early in the Laboratory operations. This pond (3512) was abandoned and filled in in the late 1940's or early 1950's when its use was no longer needed. During the construction of the current Process Waste Treatment Plant (3544), a portion of this pond was accidentally uncovered while the building footing was being dug. However, residual activity levels were not restrictive and the building was completed as designed. Subsequent sampling from the pond site has identified the presence of contamination, although not at levels requiring special monitoring or controls. The remaining pond in this Bethel Valley area is the East Sewage Treatment Plant Lagoon. This basin has been utilized as part of the ORNL Sewage System since its construction in 1954. Although not designed as part of the radioactive waste management system, minor amounts of contaminated groundwater have been collected through the underground piping network serving the Sewage Treatment Plant and have ended up in the sediment of this basin. The Lagoon is lined with a polyvinyl chloride liner, but the condition of the liner is not known.

Five additional SCMP ponds are located in Melton Valley, four in the HFIR/TRU complex (Fig. 2.3) and one in SWSA 5. The four ponds in the HFIR/TRU complex (Sites 7905, 7906, 7907, and 7908) all serve as process liquid waste collection and sampling basins. The 7905 pond also collects cooling tower blowdown. These basins are sampled on a regular basis and the contents discharged to the Process Waste Treatment Plant or White Oak Creek Watershed depending upon activity levels present. Due to the generally low levels of contamination entering these ponds, there are only minor amounts of residual activity

Fig. 2.3 Location map of SCMP process ponds in the HFIR/TRU complex in Melton Valley. Source: Myrick et al. 1984.

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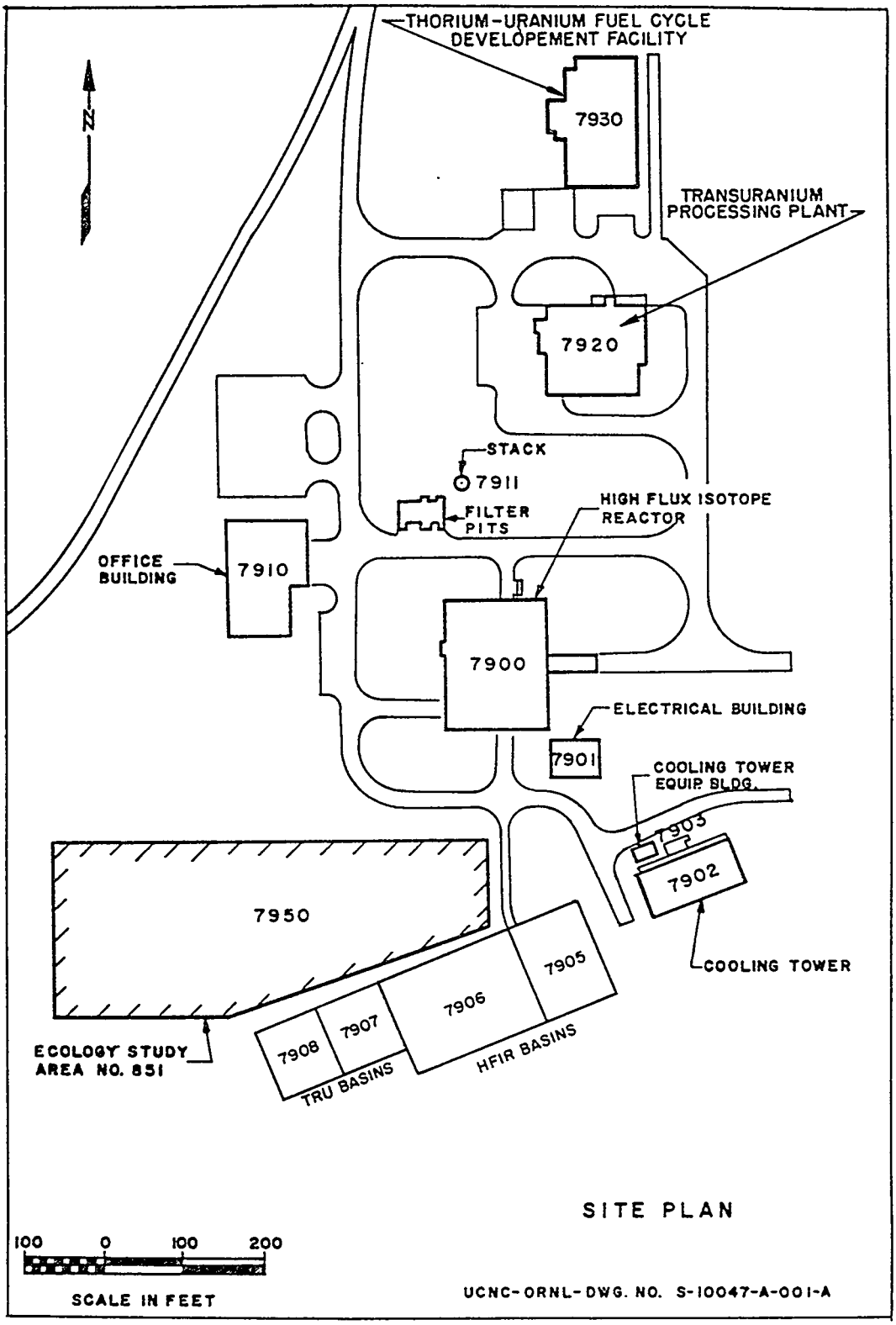

in the pond sediments. All four ponds are earth bermed, clay lined, and open with gravel riprap on the basin walls. At SWSA 5, there exists an inactive sludge pond (Site 7835), originally designed for storage of sludge generated by the Process Waste Treatment Plant. This pond is situated on a slope overlooking SWSA 5/north and was lined with thick plastic during construction. During operation, sludge was pumped from the treatment plant to this basin, the sludge was allowed to settle, and the supernatant pumped back to the Equalization Basin. This operation was discontinued in 1981 when the process was eliminated that generated the sludge. The SWSA 5 pond is open and essentially full of contaminated sludge.

3. GEOLOGY, HYDROLOGY, AND ECOLOGY

The process ponds described in this document range from ponds still in use to decommissioned ponds (filled-in and asphalt covered), all located on disturbed land areas. Remedial actions to minimize any potential hazard from these facilities are dependent on site-specific environmental factors such as geology, hydrology, and ecology. While no site-specific information is available, Nix et al. (1986) have provided generic descriptions for the Oak Ridge Reservation.

3.1 GEOLOGY

A geological map (McMasters 1963) of the Oak Ridge Reservation identifies four major stratigraphic units: Rome Formation, Conasauga Group, Knox Group, and Chickamauga Limestone. The Rome Formation underlies Haw Ridge, which separates Bethel and Melton Valleys, and consists of evenly bedded, fine-grained sandstone and shale. This

formation has a limited capacity for receiving, storing, and transporting water. In unweathered bedrock, water is largely limited to small openings along joints and bedding planes. Groundwater discharge during dry months is very low. The Conasauga Group, underlying Melton Valley, is composed of shales, limestones, and siltstones. Groundwater occurs principally in the weathered zone at openings along joints and bedding planes. These occur at shallow depths and consequently springs may be common during winter. During the summer, very little water is discharged. The Knox Group underlies much of Chestnut and Copper Ridges and is the most widely distributed geologic unit in East Tennessee. It is composed primarily of dolomitic limestone with prominent chert zones. Underground solution channels characterize this group and establish it as the principal water storage formation. The Chickamauga limestone, underlying Bethel Valley, is composed predominantly of limestone, shales, siltstones, and bedded chert. Because of the presence of extensive amounts of limestone, this formation is susceptible to underground solution by migrating groundwater and the development of a network of channels and voids. These often permit easy access of surface waters and free movement of waters underground. However, it is not an important aquifer because the channels are small with limited water storage capacity.

3.2 HYDROLOGY

The Oak Ridge Reservation consists of a series of drainage basins that feed into the Clinch River. The principal drainage system from the ORNL site is White Oak Creek. White Oak Creek originates from springs of the Knox Dolomite on Chestnut Ridge and drains an area of

16.4 km^2 (4052 acres). The creek flows from the main ORNL site in Bethel Valley (Chickamauga Limestone) through a gap in Haw Ridge (Rome Formation) to Melton Valley (Conasauga Shale). The stream varies from 0.6 m (2 ft) to 1.2 m (3.9 ft) in width and from 9.9 cm (3.9 in.) to 24.9 cm (9.8 in.) deep. Flow rates range from a maximum of 18.2 m^3/s (643 ft^3/s) to a minimum of zero, the average being 0.27 m^3/s (9.5 ft^3/s). White Oak Creek is joined by Melton Branch after it passes Haw Ridge and enters White Oak Lake 0.5 km (0.8 mile) downstream.

Melton Branch collects flow from both Haw Ridge and Copper Ridge and drains an area of 3.8 km^2 (939 acres). It is the principal drainage system of the ORNL Melton Valley complex.

Of the four stratigraphic groups, only the Knox Dolomite has significant water storage capacity. The storage usually occurs in solution cavities and may frequently result in springs. Water storage capacity of the Rome Formation, Conasauga Shale, and Chickamauga Limestone is small and occurs along joints and bedding planes. Most wells in these formations have flows less than 38 L/min (10 gal./min.).

Groundwater flow in the weathered residual soil on the ORNL site basically follows the water table conditions. In the Chickamauga Limestone underlying Bethel Valley, groundwater moves through small solution channels. Studies of groundwater movement in the Conasauga Shale of Melton Valley have suggested that the primary direction of groundwater movement parallels the strike. Groundwater discharge is through evapotranspiration, springs, and streams, and it contributes to the base flow of streams that ultimately reach the Clinch River water supply.

Depth to the water table varies both spatially and temporally. Depth to the water table is generally greatest during October to December and least between January and March. In Bethel Valley, depth to the water table ranges from 0.3 (~1 ft) to 11 m (36 ft), whereas in Melton Valley the range is from 0.3 (~1 ft) to 20 m (65.6 ft).

3.3 ECOLOGY

The land area of the reservation is ~15,000 ha (37,065 acres), with plant communities characteristic of those found in the ridge and valley region of Appalachia. The dominant oak-hickory association is typified by stands of shortleaf and Virginia pines, as well as the more extensive oaks and hickories. Approximately 75% of the reservation land area is forested. There are no threatened or endangered plant species on the reservation. Although several are proposed for status designation, none are present at any of the radwaste sites. Similarly, there are no endangered animal species on the radwaste sites, whereas the geographical ranges of several encompasses the reservation. While threatened or endangered fish species are known to exist in the headwaters of the Clinch River, none have been encountered in laboratory biological sampling programs. The reservation streams and ponds are not suitable habitat for sensitive species because they generally require a free-flowing habitat.

4. DESCRIPTION OF PROCESS PONDS

4.1 EQUALIZATION BASIN (3524)

The Equalization Basin (3524) is one of four holding basins in the southcentral portion of the Bethel Valley, ORNL complex (Figs. 2.1 and

2.2). This pond is an intermediate storage, collection, and mixing basin for the Process Waste Treatment System located in Building 3544. Basin 3524 measures 29 x 76 m (95 x 250 ft) with a maximum depth of 2 m (6.5 ft). Elevation at the top of the berm is 240 m (~787 ft). The present Basin 3524 was constructed at the site of two former basins. Two drawings provide details of the basin. They are Drawing D-36523, Site Plan and Drawing D-36524, Grading Plan. The 3524 Pond normally receives process wastes from the Bethel Valley, ORNL facility, but on occasion, some waste from the Melton Valley facilities is received. Process wastes consist of steam condensate from heating coils, process cooling water, condensate from the Low Level Waste Evaporator Facility, groundwater leakage to the pipes and tiles entering the pond, and discharges from numerous building sinks and floor drains. Several activities are responsible for the process wastes: reactor operations, radioisotope processing, hot cell operation, and general research and development. Waste streams are primarily effluents that contain traces of metals and solvents with little or no radioactivity. The basin is capable of storing one million gallons of liquids. Wastes from the 4500 Building complex are delivered to the 190 Ponds (3539 and 3540) for radionuclide monitoring, before transfer to the process waste treatment facility (3524 Pond), and discharge to White Oak Creek. A flow diagram of process waste lines is illustrated in Fig. 4.1.1.

Sludge samples were collected from the Equilization Basin (3524) during January and February, 1986, and analyzed for metal, pesticide, herbicide, and volatile organic constituents (Kitchings and Owenby 1986). A comparison of metal concentrations in EP TOX Extracts from sludge with EP Toxicity limits is presented in Table 4.1.1. Among the

Fig. 4.1.1 Flow diagram of process waste lines to the Basin 3524.
Source: MCI 1985.

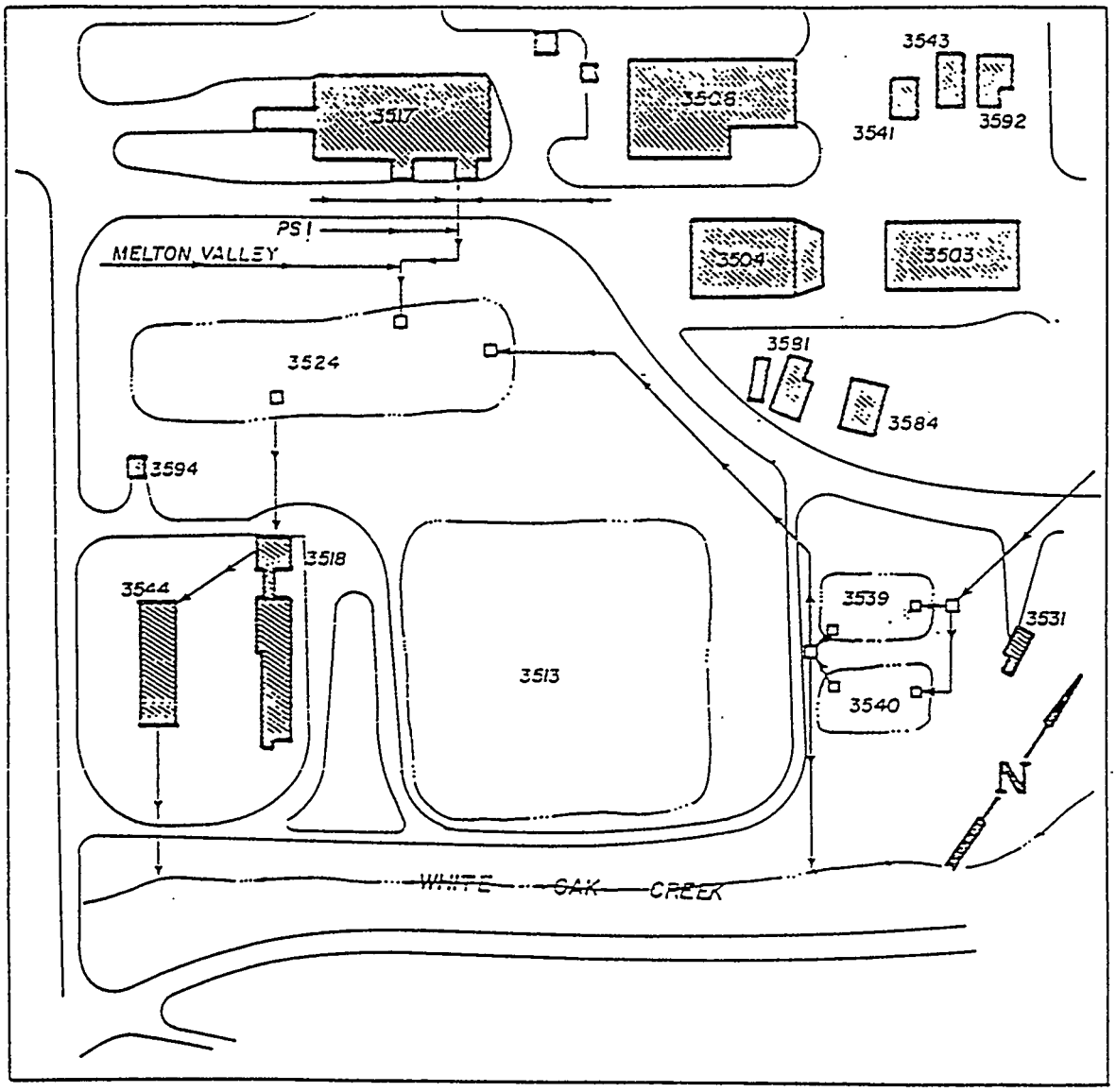
3543
3517
3508
3541
3592
PS1
MELTON VALLEY
3504
3503
3524
3591
3584
3594
3518
3544
3539
3531
3513
3540
N
WHITE OAK CREEK

Table 4.1.1 Comparison of metal concentrations in EP-TOX Extract from the 3524 Basin sludge with EP-TOX Limit.

Metal	Maximum µg/L	Minimum µg/L	Average µg/L	EP-TOX Limit µg/L
AG	<0.05	<0.05	<0.05	5
AS	<0.10	<0.10	<0.10	5
BA	1.20	1.20	1.20	100
CD	0.05	0.05	0.05	1
CR	<0.04	<0.04	<0.04	5
PB	0.62	0.37	0.50	5
SE	<0.20	<0.20	<0.20	1

metals which were present above the analytical detection limits, none exceeded the EP-TOX limit. Similarly, a comparison of pesticide/herbicide concentrations in EP-TOX Extracts from sludge with EP-TOX Limits is presented in Table 4.1.2. Among the constituents present, none exceeded the EP-TOX limit. Benzene, chloroform, and methylene chloride were three non EP-TOX Limited Volatile Organics present, with average concentrations of 6.8, 3.6, and 6.4 µg/L, respectively.

4.2 190 PROCESS WASTE POND (3539)

The 190 Process Pond (Basin 3539) is an open pond, located in the central ORNL complex (Bethel Valley), south of Building 3584 and east of the Holding Basin 3513. The pond has a capacity of 150,000 gallons (5.7×10^5 L) and measures 20 x 28 m at the top of the berm. Elevation at the top of the berm is 240 m MSL. Maximum depth is ~2.4 m (8 ft). A 15 cm thick compacted clay liner serves as the bottom of the basin.

4.3 190 PROCESS WASTE POND (3540)

This basin is located south of Basin 3539 and east of the Holding Basin 3513. The capacity and dimensions are the same as the Basin 3539. Location of both basins is depicted in Fig. 4.1.1.

The two 190 Process Waste Ponds are paired surge basins, designed to receive process waste streams from the Building 4500 complex. The basins have been in operation since 1964. A flow diagram (Fig. 4.3.1) provides details of building wastes to the two basins as well as to the Equalization Basin (3524). Waste streams from Building 4500 are split

Table 4.1.2 Comparison of pesticide/herbicide concentrations in EP-TOX Extracts from the 3524 Basin sludge with EP-TOX Limits.

Analysis	Maximum µg/L	Minimum µg/L	Average µg/L	Limit µg/L
Endrin	< 0.2	< 0.2	< 0.2	20
Lindane	< 2.0	< 2.0	< 2.0	400
Methoxychlor	< 8	< 8	< 8	10,000
2,4-D	<10	<10	<10	10,000
2,4,5-TP Silvex	<10	<10	<10	1,000
Toxophene	< 5	< 5	< 5	500

Fig. 4.3.1 Flow diagram of process waste to the 190 Process Waste Ponds. Source: MCI 1985.

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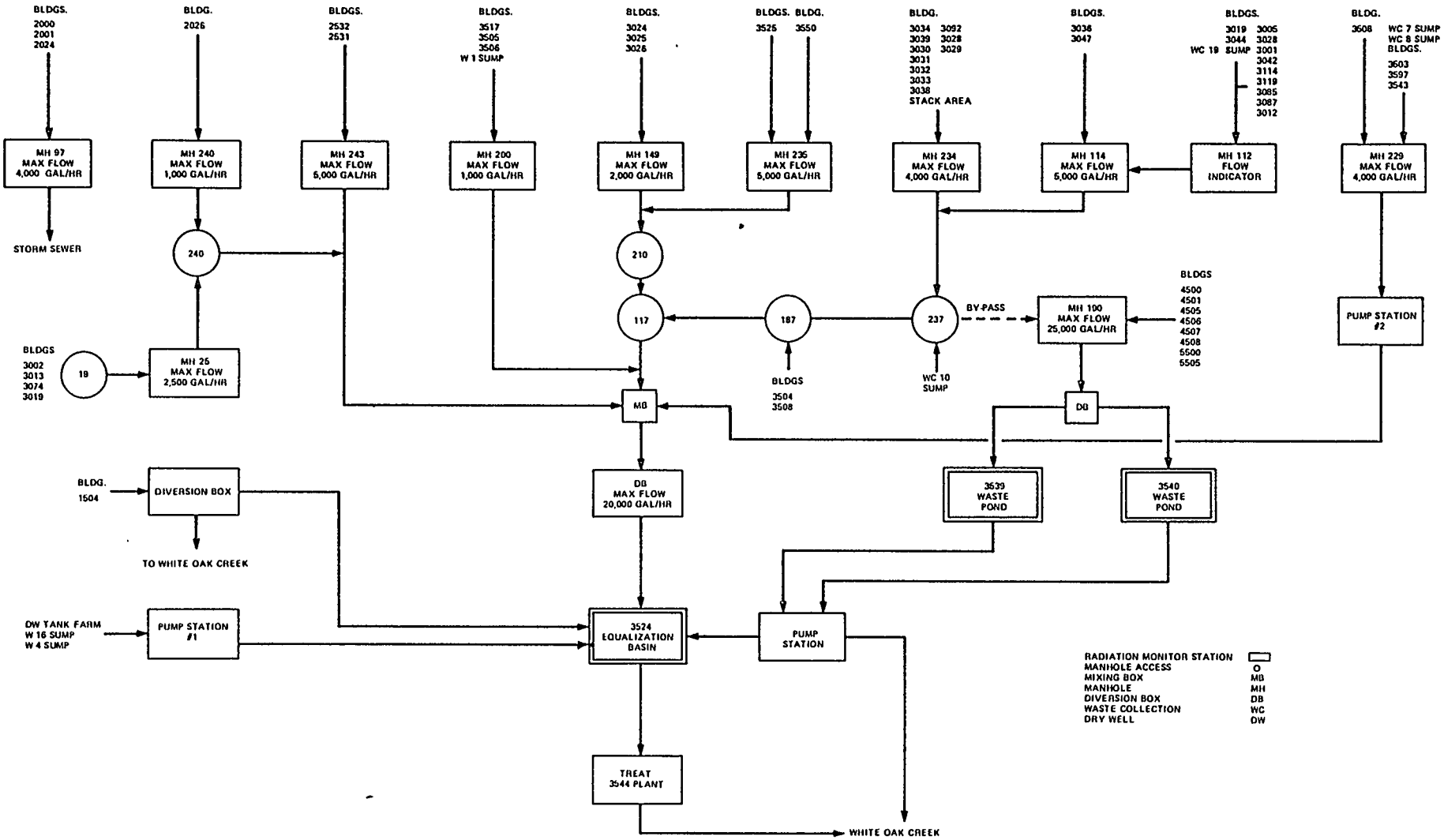


into the basins and monitored for radionuclide content before discharge to White Oak Creek or to the Equalization Basin (3524) and the process waste treatment system (Building 3544). Wastes are derived from floor drains, laboratory drains, steam condensates, and process vessel cooling waters. In the RCRA permit application, waste components are identified (MCI 1985). Some organic constituents were above detection limits. During January and February, 1986, sludge samples were collected from the 190 Process Waste Ponds (3539 and 3540) and analyzed for metal, pesticide, herbicide, and volatile organic constituents (Kitchings and Owenby 1986). A comparison of metal concentrations in EP-TOX Extracts from sludge with EP Toxicity limits is presented in Table 4.3.1. Among the metals which were present above the analytical detection limits, none exceeded the EP-TOX limit. A comparison of pesticide/herbicide concentrations in EP-TOX Extracts from sludge with EP-TOX limits is presented in Table 4.3.2. Among the constituents present, none exceeded the EP-TOX limits. Methylene chloride and 1,2-transdichloroethylene were two non EP-TOX Limited Volatile Organics present with average concentrations of 3.6 and 2.1 µg/L, respectively. The basins are operated in such a manner that when one pond is filling the other is empty or in the process of being emptied. Process wastes are ultimately discharged into White Oak Creek following verification of radionuclide content and pH adjustment. Total radioactivity present in the ponds is estimated to be less than 10 Ci (Myrick et al. 1984).

Table 4.3.1 Comparison of metal concentrations in EP-TOX Extract from sludge of the 3539 and 3540 Process Basins with EP-TOX Limit.

Basin	Analysis	Maximum µg/L	Minimum µg/L	Average µg/L	EP-TOX Limit µg/L
3539	AG	<0.0500	<0.0500	<0.05000	5.0
	AS	<0.1000	<0.1000	<0.10000	5.0
	BA	1.1000	0.9000	0.98333	100.0
	CD	0.0740	0.0630	0.06733	1.0
	CR	<0.0400	<0.0400	<0.04000	5.0
	HG	0.0012	0.0007	0.00093	0.2
	PB	<0.2000	<0.2000	<0.20000	5.0
	SE	<0.2000	<0.2000	<0.20000	1.0
3540	AG	<0.0500	<0.0500	<0.05000	5.0
	AS	<0.1000	<0.1000	<0.10000	5.0
	BA	0.9400	0.9300	0.93667	100.0
	CD	0.0510	0.0490	0.04967	1.0
	CR	<0.0400	<0.0400	<0.04000	5.0
	HG	0.0001	0.0001	0.00010	0.2
	PB	2.9000	2.7000	2.80000	5.0
	SE	0.2600	<0.2000	<0.22000	1.0

Table 4.3.2 Comparison of pesticide/herbicide concentrations in EP-TOX Extracts from sludge of the 3539 and 3540 Process Ponds with EP-TOX Limits.

Basin	Analysis	Maximum µg/L	Minimum µg/L	Average µg/L	Limit µg/L
3539	Endrin	< 0.2	< 0.2	< 0.2	20
	Lindane	< 2.0	< 2.0	< 2.0	400
	Methoxychlor	< 8	< 8	< 8	10,000
	2,4-D	<10	<10	<10	10,000
	2,4,5-TP Silvex	<10	<10	<10	1,000
	Toxophene	< 5	< 5	< 5	500
3540	Endrin	< 0.2	< 0.2	< 0.2	20
	Lindane	< 2.0	< 2.0	< 2.0	400
	Methoxychlor	< 8	< 8	< 8	10,000
	2,4-D	11	<10	<10	10,000
	2,4,5-TP Silvex	<10	<10	<10	1,000
	Toxaphene	< 5	< 5	< 5	500

Several drawings provide details of the paired ponds. These include:

S-10824-A002-D	Construction and Operation Area
S-10824-A003-D	
S-10824-A004-D	Grading Plan

Knowledgeable persons concerning the operation or construction of the basins include:

Breeding, Swan, Operations Division, ORNL, 615/574-7058
Odum, L. D., Plant & Equipment Division, ORNL, 615/574-4320

4.4 SEWAGE LAGOON, EAST BASIN (2521 area)

The East Basin Sewage Lagoon is located in the southwest ORNL complex (Fig. 2.2) and was utilized as a digestion pond for the laboratory's sewage from 1976 to 1985. The pond is 15 x 63 m, 2.4 m deep, with a capacity of 1×10^6 gallons (3.8×10^6 L). A compacted clay layer bottom is covered with a polyvinyl chloride (PVC) liner. The secondary sewage treatment (including flow through the east basin) was designed for 14 days retention before discharging the chlorinated water into White Oak Creek. It is estimated that the basin contains approximately 240 m^3 of sediments (Carmony 1986). Although the system is engineered to exclude nonsewage wastes, some radioactivity is present. Myrick et al. (1984) estimated the radionuclide content to be less than 10 Ci. The origin of that waste is unknown, but it is suspected that some laboratory drains may be connected to sewage lines. During the operation of the basin, gas developed under the PVC liner and inflated it. To remove the trapped gas, holes were cut in the liner to permit it to fall back into place.

A description of the basin is contained in the following file report:

Winsbro, W. R. 1973. Description of Aeration Lagoons. File Report.

Several engineering drawings are available which provide details of the basin. They include:

C 20219 EA 004 D
C 20219 EA 006 E
C 20219 EA 007 E
C 20219 EA 008 E
C 20219 EA 011 E

Few people presently at ORNL have detailed knowledge of the East Basin. The major source of information is:

Carmony, Vernon, Operations Division, ORNL, 615/574-7049

4.5 HFIR WASTE BASIN (7905)

Basin 7905 is one of four holding basins located south of the High Flux Isotope Reactor (HFIR) facility at the Melton Valley, ORNL facilities complex (Fig. 2.3). This basin has been in service since 1965. Basin 7905, frequently referred to as the HFIR "COLD" pond, is an intermediate storage and collection basin for the HFIR facility. The basin also serves as an emergency storage basin for radiologically contaminated blowdown water from the cooling tower in Building 7902. This provision is used several times a year and the influent flow from the HFIR facility to Basin 7905 is adjusted accordingly. The basin stores up to approximately 240,000 gallons of liquids (9×10^5 L) before discharge to Melton Branch or being pumped to Basin 3524 (Equalization Basin) in Bethel Valley. The 7905 Basin measures 26 x

35 m (86 x 116 ft) at the top of the berm and 12 x 21 m (40 x 70 ft) at the bottom of the pond. Maximum depth is 2.1 m (7 ft). Elevation at the top of the pond is 245 m (804 ft MSL). A major isotope in the 7905 Basin is cobalt-60. Sediment depth is 29 cm (12 in.). Myrick et al. (1984) estimated the combined total radionuclide content in all basins (7905, 7906, 7907, 7908) in the HFIR area to be less than 10 Ci. A schematic of the process flow to the 7905 Basin is illustrated in Fig. 4.5.1. The waste streams collected consist of flow drains, laboratory drains, steam condensates, process vessel cooling water, and precipitation falling directly on the basin. Wastes include nitric acid, sodium hydroxide, and sulfuric acid. Recent analyses of influent water and sludge indicate that none of the data for metals, pesticides, or herbicides exceeded the maximum concentration of contaminants for characteristics of EP Toxicity. Some organics exceeded detection limits (Kitchings and Owenby 1986). A comparison of metal concentrations in EP-TOX Extracts from the sludge of Basin 7905 with EP Toxicity limits is presented in Table 4.5.1. The presence of chromium is probably related to the early use of a chromate-corrosion inhibitor in the cooling tower recirculating water. A comparison of pesticide/herbicide concentrations in the EP-TOX Extracts with EP-TOX limits is illustrated in Table 4.5.2. Among the Non EP-TOX Limited Volatile Organics, chloroform and methylene chloride were identified as present above the analytical detection limits with mean concentrations of 6 and 5.7 μg/L, respectively. Major references concerning the Basin 7905 are presented by Berry et al. (1984) and MCI (1985). The site plan is illustrated in Drawing No. 1546-05-U-7306-3.

Fig. 4.5.1 Schematic of the process waste flow to the Basins 7905, 7906, 7907, and 7908 in the Melton Valley complex. Source: Berry et al. 1984.

ORNL-DWG 84-11653R2

SCHEMATIC OF THE MELTON VALLEY PROCESS WASTE SYSTEM

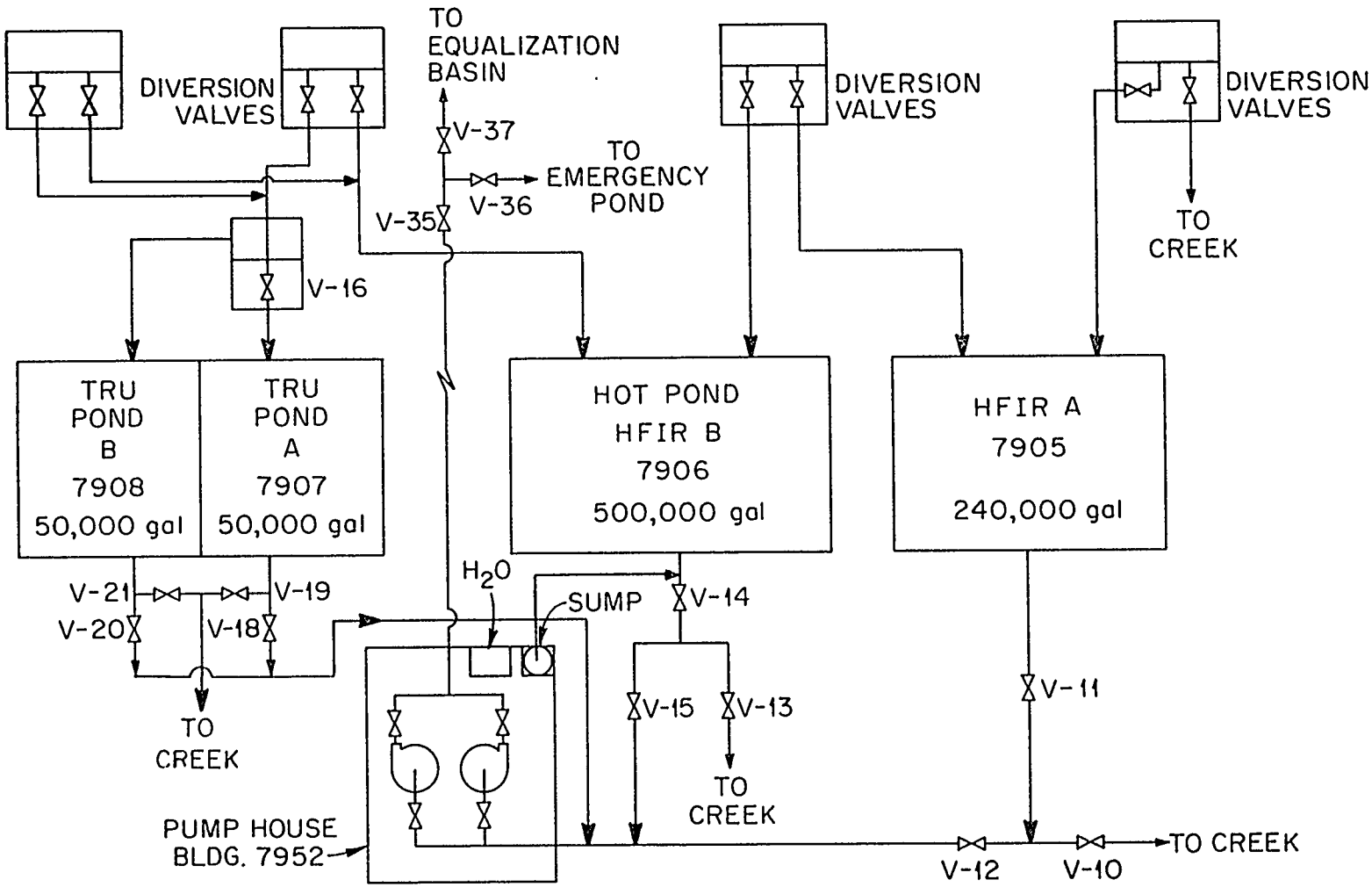


Table 4.5.1 Comparison of metal concentrations in EP-TOX Extracts from the sludge of Basin 7905 with EP-TOX Limits.

Metal	Maximum µg/L	Minimum µg/L	Average µg/L	EP-TOX Limit µg/L
AG	<0.0500	<0.0500	<0.05000	5.0
AS	<0.1000	<0.1000	<0.10000	5.0
BA	0.7300	0.6300	0.68667	100.0
CD	0.3600	0.3500	0.35667	1.0
CR	<0.0400	<0.0400	<0.04000	5.0
HG	<0.0001	<0.0001	<0.00010	0.2
PB	<0.2000	<0.2000	<0.20000	5.0
SE	<0.2000	<0.2000	<0.20000	1.0

Table 4.5.2 Comparison of pesticide/herbicide concentrations in EP-TOX Extracts from sludge of Basin 7905 with EP-TOX Limits.

Compound	Maximum µg/L	Minimum µg/L	Average µg/L	Limit µg/L
Endrin	< 0.2	< 0.2	< 0.2	20
Lindane	< 2.0	< 2.0	< 2.0	400
Methoxychlor	< 8.0	< 8.0	< 8.0	10,000
2,4-D	<10	<10	<10	10,000
2,4,5-TP Silvex	23	20	21	1,000
Toxaphene	< 5	< 5	< 5	500

A list of persons knowledgeable of the site and operations includes:

Breeding, Swan, Operations Division, ORNL, 615/574-7058
Lasher, Larry, Operations Division, ORNL, 615/576-5223
Odum, L. D., Plant & Equipment Division, ORNL, 615/574-4320

4.6 HFIR WASTE BASIN (7906)

Basin 7906 is located south of the High Flux Isotope Reactor (HFIR) in Melton Valley adjacent to the Basin 7905 (Fig. 2.3). The basin began operation in 1965. In addition to receiving wastes from HFIR, Basin 7906 can also receive diverted waste streams from the Thorium-Uranium Fuel Cycle Facility (TURF) and the Transuranium Processing Plant (TRU). A schematic of the process flow to Basin 7906 is illustrated in Fig. 4.5.1. This basin is referred to as the HFIR "HOT" Pond. The basin receives wastes which are suspected of containing radiation. Contaminants from the TRU and TURF facilities (if released) are primarily plutonium and daughter nuclides with alpha-emitting particles and atomic weights greater than 93. The major radionuclide from the HFIR is cobalt-60. Nonradioactive wastes include sodium and potassium hydroxide and acids. Recent analyses of influent water and sludge indicated that none of the data for metals or pesticides/herbicides exceeded the maximum concentration of contaminants for characteristics of EP Toxicity, and only a few of the organics exceeded detection limits (Kitchings and Owenby 1986). A comparison of metal concentrations in EP-TOX Extracts from the sludge of Basin 7906 with EP-TOX limits is presented in Table 4.6.1. A comparison of pesticide/herbicide concentrations with EP-TOX limits is

Table 4.6.1 Comparison of metal concentrations in EP-TOX Extracts from the sludge of Basin 7906 with EP-TOX Limits.

Metal	Maximum µg/L	Minimum µg/L	Average µg/L	EP-TOX Limit µg/L
AG	<0.0500	<0.0500	<0.05000	5.0
AS	<0.1000	<0.1000	<0.10000	5.0
BA	1.6000	1.3000	1.46667	100.0
CD	0.1000	0.0970	0.09833	1.0
CR	0.0970	0.0400	0.07600	5.0
HG	<0.0001	<0.0001	<0.00010	0.2
PB	<0.2000	<0.2000	<0.20000	5.0
SE	<0.2000	<0.2000	<0.20000	1.0

illustrated in Table 4.6.2. Among the Non EP-TOX Limited Volatile Organics, chloroform, methylene chloride, tetrachloroethylene, and trichloroethylene were identified as present above the analytical detection limits with average concentrations of 4.5, 7.0, 42.4, and 2.9 µg/L, respectively. If analysis indicates contamination, the waste is pumped to the Basin 3524 (Equalization Basin) in Bethel Valley. The storage capacity is 500,000 gallons (1.89×10^6 L). Elevation at the top of the basin berm is 245 m (804 ft MSL). The basin measures 51 x 35 m (167 x 116 ft) at the top and 37 x 21 m (121 x 70 ft) at the bottom. Average sediment depth is 20 cm (~7.8 in.). Myrick et al. (1984) estimated that Basin 7906 contained less than 10 curies of radioactivity (combined with Basins 7907 and 7908).

Persons knowledgeable of Basin 7906 include:

Breeding, Swan, Operations Division, ORNL, 615/574-7058
Odum, L. D., Plant & Equipment Division, ORNL, 615/574-4320
Belitz, K. S., Operations Division, ORNL, 615/576-5012

4.7 TRU WASTE BASIN (7907)

Basin 7907 is one of two ponds designed to receive process waste streams from the Transuranic Processing Facility (TRU). It is located south of Building 7900 (High Flux Isotope Reactor) in the Melton Valley area (Fig. 2.3). The basin is also known as the number 3 or TRU A Pond. The pond was put into service in 1965 and is currently receiving process waste streams from the TRUF/TRU facilities. During operation the Basin 7907 is filled and emptied alternately with Basin 7908. Waste streams are derived from floor drains, laboratory drains, steam

Table 4.6.2 Comparison of pesticide/herbicide concentrations in EP-TOX Extracts from sludge of Basin 7906 with EP-TOX Limits.

Compound	Maximum µg/L	Minimum µg/L	Average µg/L	Limit µg/L
Endrin	< 0.2	< 0.2	< 0.2	20
Lindane	< 2.0	< 2.0	< 2.0	400
Methoxychlor	< 8.0	< 8.0	< 8.0	10,000
2,4-D	<10	<10	<10	10,000
2,4,5-TP Silvex	<10	<10	<10	1,000
Toxaphene	< 5	< 5	< 5	500

condensates, and process vessel cooling waters. The major constituent in the process waste is curium-244. If radioactive contamination is detected, the wastes are pumped to Basin 3524 in the ORNL Bethel Valley complex. Noncontaminated water is discharged to Melton Branch. Figure 4.5.1 illustrates the process flow of the basin.

The pond has a capacity of 50,000 gallons (1.80×10^5 L) and measures 12 x 24 m (60 x 80 ft) at the top of the berm. Depth is approximately 3.4 m (4 ft). While there is no measure of radioactivity, Myrick et al. estimates that the combined radioactivity of all basins (7905, 7906, 7907, 7908) in the HFIR complex is less than 10 curies. Average depth of sediments is 6 cm (~2.4 in.). Some details of Basin 7907 are contained in the RCRA permit application (MCI 1985) and the operating manual (Berry et al. 1984). Recent analyses of influent water and sludge indicated that none of the data for metals or pesticides/herbicides exceeded the maximum concentration of contaminants for characteristics of EP Toxicity, and only a few of the organics exceeded detection limits (Kitchings and Owenby 1986). A comparison of metal concentrations in EP-TOX Extracts from sludge of Basin 7907 with EP-TOX limits is presented in Table 4.7.1. A comparison of pesticide/herbicide concentration with EP-TOX limits is illustrated in Table 4.7.2.

Two drawings provide details of the basin's physical characteristics. These are:

Drawing No. C-20998-Y-013-E-1 Site Plan
Drawing No. 2397-V-61-2 Underground Piping Plan

Table 4.7.1 Comparison of metal concentrations in EP-TOX Extracts from sludge of Basin 7907 with EP-TOX Limits.

Metal	Maximum µg/L	Minimum µg/L	Average µg/L	EP-TOX Limit µg/L
AG	<0.0500	<0.0500	<0.05000	5.0
AS	0.7400	0.7300	0.73667	5.0
BA	1.7000	1.7000	1.70000	100.0
CD	0.0240	0.0230	0.02333	1.0
CR	<0.0400	<0.0400	<0.04000	5.0
HG	<0.0001	<0.0001	<0.00010	0.2
PB	0.5000	0.4700	0.48667	5.0
SE	<0.2000	<0.2000	<0.20000	1.0

Table 4.7.2 Comparison of pesticide/herbicide concentrations in EP-TOX Extracts from sludge of Basin 7907 with EP-TOX Limits.

Compound	Maximum µg/L	Minimum µg/L	Average µg/L	Limit µg/L
Endrin	< 0.2	< 0.2	< 0.2	20
Lindane	< 2.0	< 2.0	< 2.0	400
Methoxychlor	< 8.0	< 8.0	< 8.0	10,000
2,4-D	<10	<10	<10	10,000
2,4,5-TP Silvex	11	<10	<10	1,000
Toxaphene	< 5	< 5	< 5	500

Several persons have knowledge of the Basin 7907 operation. They are:

Breeding, Swan, Operations Division, ORNL, 615/574-7058
Chattin, Fred, Chemical Technology Division, ORNL, 615/574-7071
Lasher, Larry, Operations Division, ORNL, 615/576-5223

4.8 TRU WASTE BASIN (7908)

Basin 7908 is located in the Melton Valley HFIR complex (Fig. 2.3) adjacent to the Basin 7907. The basin is identical in physical characteristics to the Basin 7907 and is known as the TRU B Pond. In operation, this pond is filled and emptied alternately with the Basin 7907. Particulars of the basin are contained in the description of Basin 7907 (Section 4.7). Recent analyses of influent water and sludge indicated that none of the data for metals or pesticides/herbicides exceeded the maximum concentration of contaminants for characteristics of EP Toxicity, and only a few of the organics exceeded detection (Kitchings and Owenby 1986). A comparison of metal concentrations in EP-TOX Extracts from the sludge with EP-TOX limits is presented in Table 4.8.1. The only Non EP-TOX Limited Volatile Organic identified as present above the analytical detection limit was methylene chloride with an average concentration of 15.2 μg/L. A comparison of pesticide/herbicide concentrations with EP-TOX limits is illustrated in Table 4.8.2.

4.9 PROCESS WASTE SLUDGE BASIN, SWSA 5 (7835)

The Basin 7835 is located north of SWSA 5 in Melton Valley (Fig. 2.1). A PVC liner is on top of a compacted clay layer. Sludge, generated by a water softening process at the Process Waste Treatment Plant, was transferred to the basin from 1976 to 1980 for storage.

Table 4.8.1 Comparison of metal concentrations in EP-TOX Extracts from sludge of Basin 7908 with EP-TOX Limits.

Metal	Maximum µg/L	Minimum µg/L	Average µg/L	EP-TOX Limit µg/L
AG	<0.0500	<0.0500	<0.05000	5.0
AS	<0.1000	<0.1000	<0.10000	5.0
BA	0.5000	0.4800	0.49333	100.0
CD	<0.0050	<0.0050	<0.00500	1.0
CR	<0.0400	<0.0400	<0.04000	5.0
HG	<0.0001	<0.0001	<0.00010	0.2
PB	<0.2000	<0.2000	<0.20000	5.0
SE	<0.2000	<0.2000	<0.20000	1.0

Table 4.8.2 Comparison of pesticide/herbicide concentrations in EP-TOX Extracts from sludge of Basin 7908 with EP-TOX Limits.

Compound	Maximum µg/L	Minimum µg/L	Average µg/L	Limit µg/L
Endrin	< 0.2	< 0.2	< 0.2	20
Lindane	< 2.0	< 2.0	< 2.0	400
Methoxychlor	< 8.0	< 8.0	< 8.0	10,000
2,4-D	<10	<10	<10	10,000
2,4,5-TP Silvex	18	<10	<13	10,000
Toxaphene	< 5	< 5	< 5	500

Sludge constituents are primarily ferrous sulfate and ferric hydroxide with mixed fission products present. Myrick et al. (1984) estimates that as much as 50 Ci may be contained in the 1.2 m (4 ft) thick sediment layer. The basin is 24 x 24 m (80 x 80 ft) with a maximum depth of 2.4 m (8 ft). Elevation at the top of the berm is 247 m MSL (809 ft).

Several drawings provide details of the basin. These include:

Drawing C-21246-EA-007D Grading Plan
Drawing C-21246-EA-008D Plot Plan

The most knowledgeable person contacted, concerning the operation of the basin, is L. Lasher, Operations Division, ORNL, 615/576-5223.

4.10 ABANDONED FILLED-IN WASTE BASIN (3512)

The Basin 3512 was located in Bethel Valley complex near the present Building 3544. More specifically, the site is the location of the present parking lot for Building 3544. Basin 3512 was a retention pond for storage tank drainage from 1944 to 1950. Capacity of the basin was 1.13×10^5 L (30,000 gallons). Myrick et al. (1984) estimates the total radioactive contamination to be less than 10 Ci. Few individuals were located who had any knowledge of the basin or its operation. One drawing (C-11116A, Process Waste Water Treatment Plant Plot Plan) gives details of size, but is not specific for exact location. No information was available relative to the kinds, extent, and quantity of contamination.

4.11 WASTE HOLDING BASIN (3513)

The 3513 Waste Holding Basin is located in Bethel Valley (Fig. 2.2) approximately 33 m (110 ft) north of White Oak Creek between Building 3544 and the two 190 Process Waste Ponds (3539 and 3540). The impoundment was constructed in 1944 to act as a settling basin for wastes prior to their discharge into White Oak Creek. The basin is 67 x 67 x 0.45 meters and has a storage capacity of 1.88×10^6 gallons (7.12×10^6 L) of water and sediment (Stansfield and Francis 1985). The basin was in operation until 1976 when a new process waste treatment plant began operation. Wastes to the basin included supernatant from the gunite tanks until 1949. Fly ash and soda lime were used to precipitate the radionuclides prior to releasing the basin water to White Oak Creek. Additional wastes were from laboratory floor and sink drains, chemical process cells, and cooling water from the graphite reactor. An operational flow diagram of the liquid wastes into Basin 3515 is depicted in Fig. 4.11.1.

A characterization study of Huang et al. (1984a) provides details of surface contamination in the perimeter areas and inventories of radionuclides in the pond. In a walk-over survey, elevated beta-gamma dose rates were detected around the pond. Average absorbed dose rates from areas outside and inside the fence were 0.3 mrad/hr (0.003 mGy/h) and 0.5 mrad/h (0.005 mGy/h), respectively.

To determine the degree of surface contamination in the perimeter areas, soil samples to 0.3 m deep were collected and analyzed. The primary radionuclides identified were Cs-137 and Sr-90 (Table 4.11.1). The highest Cs-137 concentration noted was 1.3 nCi/g (48 Bq/g), about

Fig. 4.11.1 Operational flow diagram of the liquid waste into WHB 3515.

ORNL–DWG 82-8416

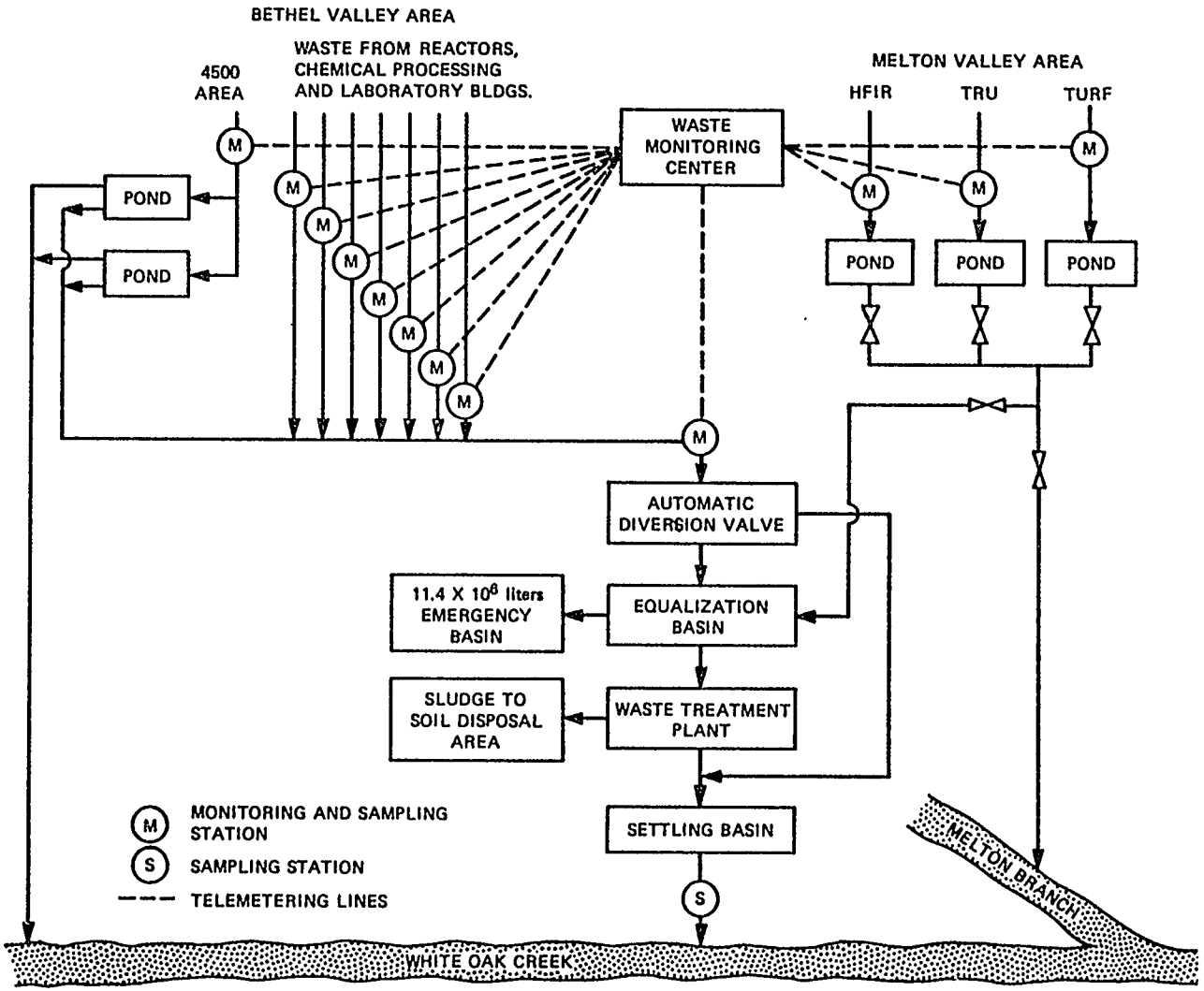


Table 4.11.1 Results of the preliminary gamma screening and the quantitative analysis of beta and gamma emitting radionuclides in soil and water samples near the Waste Holding Basin (3513). Source: Huang et al. 1984a.

Sample	Depth (m)	Gross gamma (cps/kg)[a]	^{137}Cs (Bq/g dry wt)	^{90}Sr (Bq/g dry wt)
		Hand cores		
1B-1	0-0.3	4600	37	3
3A-1	0-0.3	82	0.3	0.2
3B-1	0-0.3	150	0.8	0.2
3C-1	0-0.3	410	3.4	0.5
1K-1	0-0.3	150	0.53	0.53
1L-1	0-0.3	90	0.44	0.23
1M1-1[b]	0-0.1	1600	11	74
1M2-1	0.1-0.3	330	2.0	1.3
1N-1	0-0.3	900	7.4	NA[c]
1O-1	0-0.3	2100	19	0.43
15N-1[d]	0-0.3	77	0.25	0.15
6N-1	0-0.3	3800	44	0.8
6N-2	0.3-0.6	1100	7.6	0.4
6N-3	0.6-1.2	300	1.7	1.5
6N-4	1.2-1.8	230	1.4	3.0
6N-5	1.8-2.4	290	1.9	1.4
14H-1	0-0.3	4600	48	NA
14H-2	0.3-0.6	1200	8.5	0.44
14H-3	0.6-1.2	270	2.3	1.7
14H-4	1.2-1.8	170	1.1	0.05
		Deep Cores		
1-1	0-0.3	46	0.12	0.03
1-2	0.3-0.6	43	NA	NA
1-3	0.6-1.2	26	0.01	<0.05
1-4	1.2-1.8	25	NA	NA
1-5	1.8-2.4	38	NA	NA
1-6	2.4-3.1	25	NA	NA
1-7	3.1-3.7	12	NA	0.23
2-1	0-0.3	49	NA	NA
2-2	0.3-0.6	46	NA	NA
2-3	0.6-1.2	31	NA	NA
2-4	1.2-1.8	29	NA	NA
2-5	1.8-2.4	48	NA	NA
2-6	2.4-3.1	31	<0.01	0.82
2-7	3.1-3.7	32	NA	NA

Table 4.11.1 (CONTINUED)

Sample	Depth (m)	Gross gamma (cps/kg)[a]	^{137}Cs (Bq/g dry wt)	^{90}Sr (Bq/g dry wt)
		Water samples		
1-7	3.1-3.7	NA	ND	ND
2-7	3.1-3.7	NA	ND	0.033

[a]Total gamma activity between the 100- and 1500-keV energy range. Measured in counts per second (cps) per kilogram of moist soil. Background = 31 ± 28 cps/kg.

[b]Also contains trace quantities of ^{134}Cs. The sample from the 1M grid cell was divided into two portions, 1M1 and 1M2.

[c]NA indicates not analyzed; ND indicates nondetectable.

[d]A control sample from area with 0.3-mrad/h reading.

[e]Radionuclide concentrations in water are shown in units of Bq/mL.

2 m from the east pond boundary. Maximum Sr-90 concentration was 2 nCi/g (74 Bq/g) at 10 m from the southeast corner of the pond. Results for transuranic radionuclides are presented in Table 4.11.2. The most significant constituent among the transuranics was Pu-239, with a maximum concentration of 13.5 pCi/g (0.5 Bq/g) at 2 m from the southwest pond boundary.

The estimated direct-beta gamma absorbed dose rates measured 10 cm above the pond water were 0.16 mrad/h (0.0016 mGy/h) near the center of the pond and 0.32 mrad/h (0.0032 mGy/h) near the pond boundary. At the time of the survey (September 1983) it was estimated that the pond contained 4×10^6 L (~1.06×10^6 gal.) of water (Huang et al. 1984a). The pond water concentrations were 8 pCi/mL (0.3 Bq/mL) of Sr-90 and 11 pCi/mL (0.4 Bq/mL) of Cs-137. Consequently, inventories of Sr-90 and Cs-137 in the water are 50 mCi (2×10^9 Bq) and 30 mCi (1×10^9 Bq), respectively (Table 4.11.3).

The sediment layer ranged from 0.5 m to 1 m thick and contained approximately 2000 m^3 of residue (Huang et al. 1984a). In addition to Cs-137 and Sr-90, the sediment contained measurable quantities of Co-60, Eu-154, Pu-238, 239, Am-241, and Cm-244. Total inventory of radionuclides was estimated to be 155 Ci (5.9×10^{12} Bq), with cesium and strontium accounting for 84 and 13%, respectively. The remaining radionuclides comprise 3% of the total inventory. The inventory of radionuclides, estimated by Huang et al. (1984a), suggests from 20 to 35% less radioactivity (Table 4.11.4) than the earlier data from Tamura et al. (1977). The differences are probably related to sampling errors and illustrated the variability in estimating inventories.

Table 4.11.2 Results of quantitative analyses for transuranic radionuclides in soil and water samples near the Waste Holding Basin (3513). Source: Huang et al. 1984a.

Sample	^{239}Pu	^{238}Pu	^{241}Am	^{244}Cm
	(Bq/kg dry wt)[a]			
3A-1	4	<0.2	2	2
3B-1	8	<0.1	7	6
3C-1	20	0.3	5	2
1K-1	3	<0.1	1	3
1L-1	10	0.1	4	0.8
1M2-1	6	3	4	6
10	40	1	9	1
15N	3	<0.1	0.4	4
6N-1	500	7	100	3
6N-2	70	0.1	10	4
6N-3	20	<0.3	3	8
6N-4	10	0.4	9	6
6N-5	10	0.4	5	10
14H-2	200	10	20	10
14H-3	30	4	6	7
14H-4	20	2	3	1
1-1	1	1	2	0.3
1-7	ND[b]	ND	ND	ND
2-6	<0.1	1	1	0.4
2-7[c]	<0.001	0.003	0.003	0.02

[a]1 Bq = 27 pCi.
[b]Nondetectable.
[c]A water sample measured in Bq/L.

Table 4.11.3 Estimated inventories of water and sediment radionuclides in the Waste Holding Basin (3513). Source: Huang et al. 1984a.

Pond content	Average depth and volume	Major radionuclides	Average activity (Bq/mL or Bq/g)[a]	Inventory (GBq)	Inventory (Ci)
Water	1 m	^{137}Cs	0.4	2	0.05
	4×10^6 L	^{90}Sr	0.3	1	0.03
Sediment	0.5 m	^{137}Cs	2,400	5,000	130
	2×10^6 L	^{60}Co	21	50	1
		^{90}Sr	330	700	20
		^{238}Pu	1.7	4	0.1
		^{239}Pu	47	100	3
		^{241}Am	9.2	20	0.5
		^{244}Cm	2.3	5	0.1
		^{154}Eu	3.8	8	0.2

[a]Activity in sediment is on wet weight basis.

Table 4.11.4 Estimated inventory of radionuclides in sediment of 3513 impoundment.

Radionuclide	GBq		Ci	
Cesium-137	7300	(200a)	4800	(130)
Strontium-90	1100	(30)	730	(20)
Transuranics	200	(5)	140	(4)

Data in the second and third column are from Tamura et al. (1977) and the data in the last two columns are from Huang et al. (1984a).

In addition to the radionuclides, low levels of hazardous PCB's and heavy metals were detected in the sediment (Table 4.11.5). The sediment data of Huang et al. (1984a) are compared to the recent data of Stansfield and Francis (1986) in Table 4.11.5. In most instances the more recent estimates (1986) are higher. These differences are likely related to sampling errors. The concentration of PCB's in the water was 0.5 to 1.1 µg/L. In the characterization study by Stansfield and Francis (1986), chromium, lead, and silver concentrations in groundwater were identified as exceeding the National Interim Primary Drinking Water Standards (NIPDWS). Regulated concentrations for those three elements are equivalent (0.05 mg/L). Measured concentrations in the groundwater were 0.3388, 0.244, and 0.07 mg/L for chromium, lead, and silver, respectively. Those data are summarized in Table 4.11.6.

4.12 OLD HYDROFRACTURE FACILITY POND (7852 area)

The Old Hydrofracture Facility Pond (OHF) is located in the Bethel Valley complex (Fig. 4.12.1) near Building 7852 (Reed 1984). The site is in the southwest part of Solid Waste Storage Area No. 5 (SWSA 5) with drainage into White Oak Creek (120 m away), White Oak Lake, and ultimately into the Clinch River. Constructed in 1964, the pond measures 6 x 30 m x 1.8 m (20 x 100 x 6 ft deep) and has a storage capacity of 3.8×10^5 L (100,000 gallons). The sides of the pond are lined with limestone riprap. The primary purpose of the pond was to provide a catchment to receive any emergency backflow of grout from the hydrofracture well or any other source (equipment washdown). This would prevent accidental "grouting-in" of a well or equipment. A

Table 4.11.5 Concentrations and estimated inventories of nonradioactive contaminants in sediment of 3513 impoundment.* Data in the second and third column are from Stansfield and Francis (1986) and the data in the last two columns are from Huang et al. (1984a).

Contaminant	Concentration (mg/kg)	Inventory** (kg)	Concentration (mg/kg)	Inventory*** (kg)
Ag	48	22	NR	NR
As	<79	<37	2	0.9
Ba	378	177	NR	NR
Cd	13.1	6.1	7.2	3.2
Cr	1100	520	440	194
Cu	826	388	150	66
Hg	NR	NR	56	25
Pb	2800	1310	420	185
Se	156	73.2	2.8	1.2
Zn	632	297	130	57
PCB	6.9	3.2	16	7.0

*Mean concentration from five core analyses on dry weight basis. NR = not reported.

**Sediment volume estimated to be 2.35 x 10^6 L, bulk density of 1.2 kg/L and a moisture content of 83% (wet basis).

***Sediment volume estimated to be 2.0 x 10^6 L, bulk density of 1.1 kg/L and a moisture content of 80% (wet basis).

Table 4.11.6 Summary of selected groundwater concentrations from downgradient wells at 3513 Basin (from Stansfield and Francis 1986).

	Maximum level mg/L	Measured mg/L
National Interim Primary Drinking Water Standards (NIPDWS)		
Arsenic	0.05	<0.0018
Barium	1	0.2472
Cadmium	0.01	<0.0018
Chromium	0.05	0.3388
Fluoride	1.4-2.4	<1
Lead	0.05	0.244
Mercury	0.002	0.0002
Nitrate-N	10	3.5
Selenium	0.01	<0.005
Silver	0.05	<0.07
Endrin	0.0002	<0.001
Lindane	0.004	<0.0002
Methoxychlor	0.1	<0.0002
Toxaphene	0.005	<0.002
2,4-D	0.1	<0.0057
2,4,5-TP Silvex	0.01	<0.0058
	Bq/L	Bq/L
Radium-226	0.19	<0.0463
Gross alpha	0.556	1.47
	mR/yr	mR/yr
Gross beta	4	19.92
	counts/100 mL	counts/100 mL
Coliform bacteria	1	0.5
	mg/L	mg/L
Parameters establishing groundwater quality		
Chloride	ND	17.7
Iron	ND	19.8
Manganese	ND	3.73
Phenols	ND	<0.0025
Sodium	ND	30.2
Sulfate	ND	<12.8

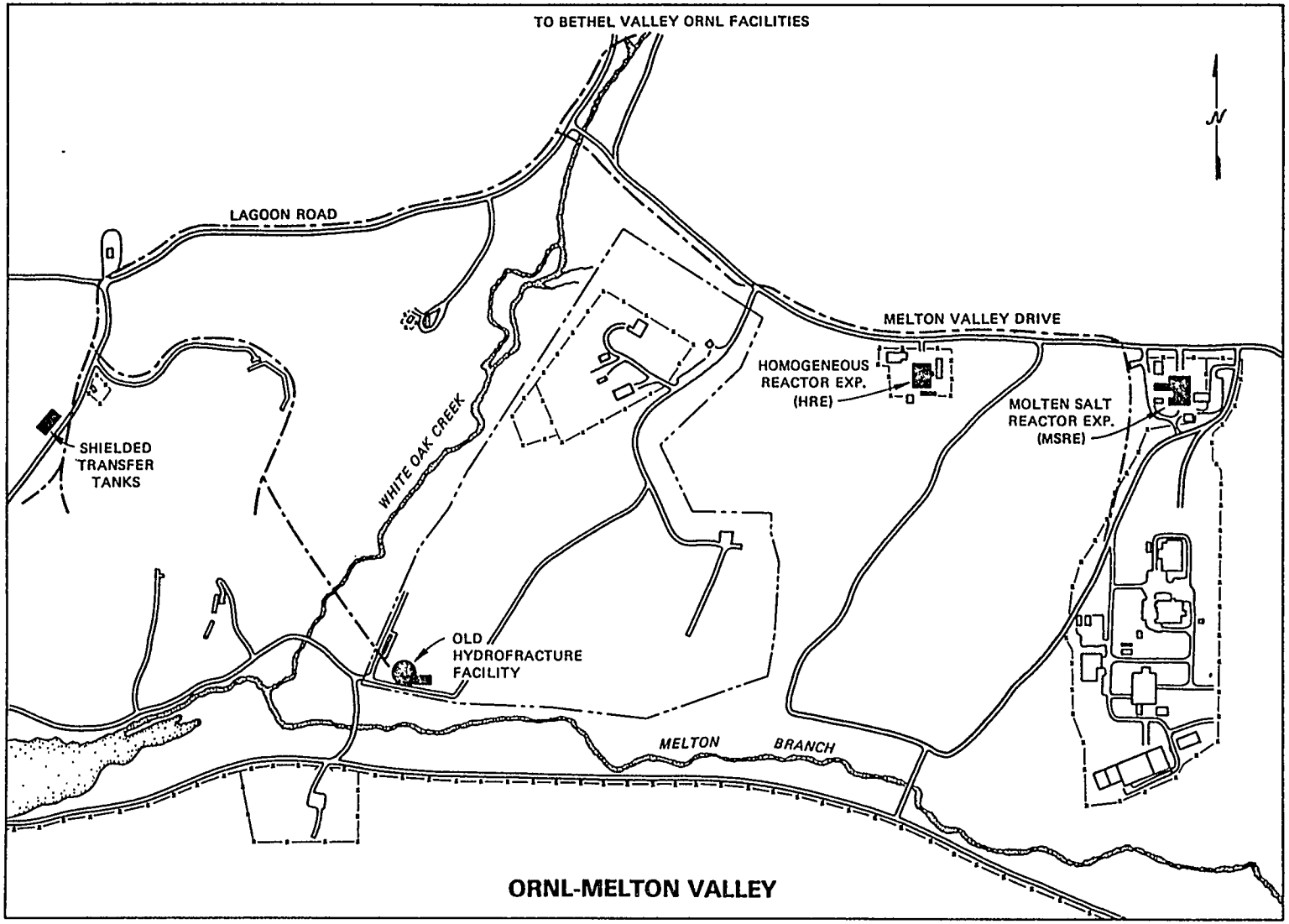
Table 4.11.6 (CONTINUED)

	mg/L	mg/L
Nonregulated parameters		
PCB's	ND	0.0001
	Bq/L	Bq/L
Tritium	ND	2700
Cesium-137	ND	0.39
Strontium-90	ND	10.6
Parameters used as indicators of groundwater contamination		
pH	ND	6.5
Specific conductance (uS/cm)	ND	592
Total organic carbon (mg/L)	ND	5.98
Total organic halogen (mg/L)	ND	0.0485

Measured concentrations are mean values of six determinations. 1 Bq = 27 pCi. ND = not defined.

Fig. 4.12.1 Area map of the ORNL facilities at Melton Valley (1 in. = 300 m).

ORNL–DWG 82-19188R2



ORNL-MELTON VALLEY

construction drawing (ORNL-DWG No. 540002-A-002-0) indicated that the pond bottom was to be sprayed with a liquid asphalt at 2 to 3 L/m^2 to control erosion. The presence of that liner has not been confirmed. The elevation of the bottom of the pond is ~233 m (764 ft) and the elevation of the dam is ~235 m (771 ft). During waste injection there existed a potential for wellhead rupture. Such a break would allow the grout to flow back up the well with no way of stopping the flow. In such an event, the waste grout mixture was to flow from the wellhead cell through an underground waste line [18 in. (46 cm) diameter] to the waste pond. During the operational life of the facility (1964-1969) there were no accidental breaks. Contamination in the pond resulted from wastewater rinsing (washdown) of equipment.

In a radiological characterization, Huang et al. (1984b) reported maximum dose rates of 100 and 300 mrad/hr at a height of 10 cm above the pond water. At that time (September 1983) there was a 40 cm layer of water and a 20 cm layer of sediment. It is estimated that the pond contains 37 m^3 (1306 ft^3) of sediment and 19 m^3 (670 ft^3) of soft clay. Samples of pond water, sediment, and underlying clay were all contaminated with cesium-137, strontium-90, cobalt-60, and cesium-134. The total activities in the water, sediment, and clay are 71 mCi, 404 Ci, and 11.7 Ci, respectively. In addition, significant quantities of transuranic alpha-emitting radionuclides were present in the sediment and clay-layer.

Pond water from the OHF impoundment was sampled in May of 1985 by Stansfield and Francis (1986). The sample was taken from the center of the impoundment when the depth of water was approximately 1 m (3 ft).

The sample, which was not filtered, was analyzed for heavy metals, herbicides/pesticides, and PCBs (Tables 4.12.1 and 4.12.2). The pH of the pond water was neutral (7.05) with specific conductance of 224 uS/cm. The analyses indicated that arsenic, barium, cadmium, chromium, fluoride, mercury, lead, and nitrate, as well as the pesticides and herbicides were below the maximum allowable NIPDWS (National Interim Primary Drinking Water Standards). The measured concentration of selenium (0.016 mg/L), slightly over the NIPDWS level (0.01 mg/L), and the detection limit of silver (0.07 mg/L) determined in the pond water were the only indications that chemical constituents exceeded the allowable NIPDWS levels. Detectable concentrations of PCBs were observed in the pond water (0.0001 mg/L) and counts of coliform bacteria (8 counts/100 mL) were in excess of the NIPDWS. The concentration of total organic halides (TOX) was 0.13 mg/L, and the total organic carbon content (TOC) was 16 mg/L. Other than the radionuclide concentrations, which are discussed below, the water quality of pond water from the impoundment (based on these chemical analyses) was surprisingly high.

The principal radionuclides measured in the pond water were cesium-137 and strontium-90 (Table 4.12.3). Stansfield and Francis (1986) observed similar concentrations of the two radionuclides [approximately 4000 Bq/L (equivalent to 109 pCi/mL)] while Huang et al. (1984b), who sampled the impoundment in September of 1983, observed approximately four times more cesium-137 than strontium-90 [29,000 and 7,100 Bq/L, respectively (equivalent to 800 and 193 pCi/mL)]. The concentrations of these two radionuclides measured by Huang et al.

Table 4.12.1 Concentration and inventory of nonradioactive contaminants in OHF pond water (from Stansfield and Francis 1986).

Constituent	Concentration pond mg/L		Inventory NIPDWS grams
Metals			
Antimony	<0.3	ND	<74
Arsenic	<0.001	0.05	<1
Barium	0.539	1	132
Beryllium	0.0021	ND	1
Boron	<0.1	ND	<25
Cadmium	0.0015	0.01	<1
Calcium	26.3	ND	6444
Chromium	0.0219	0.05	5
Cobalt	<0.02	ND	<5
Copper	<0.02	ND	<5
Iron	9.25	ND	2266
Lead	<0.001	0.05	<1
Lithium	<0.2	ND	<49
Magnesium	8.99	ND	2203
Manganese	0.2	ND	49
Mercury	0.0001	0.002	<1
Molybdenum	<0.02	ND	<5
Nickel	<0.06	ND	<15
Potassium	6.5	ND	1593
Selenium	0.016	0.01	4
Silver	<0.07	0.05	<17
Sodium	<0.5	ND	<123
Strontium	0.316	ND	77
Titanium	<0.02	ND	<5
Vanadium	<0.03	ND	<7
Zinc	0.134	ND	33
Anions			
Chloride	64	ND	15680
Fluoride	1	1.2-2.4	245
Nitrate-N	<1	10	<245
Phosphate	<0.93	ND	<228
Sulfate	19	ND	4655

Inventory based on pond water volume of 2.45×10^5 L. ND = level not defined by NIPDWS (National Interim Primary Drinking Water Standards).

Table 4.12.2 Concentration and inventory of organic chemicals in OHF pond water (from Stansfield and Francis 1986).

Constituent	Pond (mg/L)	NIPDWS (mg/L)	Inventory (grams)
Herbicides/pesticides			
Endrin	<0.0001	0.002	<0.1
Lindane	<0.0001	0.004	<0.1
Methoxychlor	<0.0002	0.1	<0.1
Toxaphene	<0.002	ND	<0.5
Organic compounds			
PCB's	0.0001	ND	<0.1
Phenols	<0.0001	ND	<0.1
TOC	16.5	ND	4043
TOX	0.132	ND	32

Inventory base on pond water volume of 2.45×10^5 L. TOC = total organic carbon. TOX = total organic halides. PCB = polychlorinated biphenyls. ND = limit not defined by NIPDWS (National Interim Primary Drinking Water Standards).

Table 4.12.3 Concentration and inventory of radionuclides in OHF pond water.* Data in the second and third column are from Stansfield and Francis (1986) and the data in the last two columns are from Huang et al. (1984b).

Radionuclide	Concentration Bq/L	Inventory MBq	Inventory mCi	Concentration Bq/L	Inventory MBq	Inventory mCi
Gross alpha	11	2.7	0.07	NR	NR	NR
Gross beta	9400	2303	63	NR	NR	NR
Americium-241	0.24	0.1	<0.01	NR	NR	NR
Cesium-137	3900	956	26	29000	2100	57
Cobalt-60	27	6.6	0.18	27	2	0.05
Curium-244	6.8	1.7	0.05	NR	NR	NR
Plutonium-238	0.17	<0.1	<0.01	NR	NR	NR
Plutonium-239	0.052	<0.1	<0.01	NR	NR	NR
Radium-226	0.015	<0.1	<0.01	NR	NR	NR
Strontium-90	4400	1080	29	7100	514	14
Uranium-234	1.5	0.4	0.01	NR	NR	NR
Uranium-238	0.375	0.1	<0.01	NR	NR	NR

*Value used for pond volume by Stansfield and Francis was 2.45×10^5 L. Value used for pond volume by Huang et al. (1984b) was 7.24×10^4 L. NR = not reported.

(1984b) were two to seven times that measured by Stansfield and Francis (1986). On the other hand, cobalt-60 concentrations in the pond water were identical. Stansfield and Francis (1986) also measured gross alpha and gross beta activity in the pond water. Gross beta activity was approximately twice the activity of strontium-90, implying that the gross beta activity resulted largely from the decay of strontium-90 and yttrium-90. Gross alpha activity was less than 1% of the gross beta and cesium-137 activities combined. Curium-244 appeared to be the dominant transuranic radionuclide measured (6.8 Bq/L or <0.2 pCi/mL). The volume of pond water retained in the impoundment was estimated by Stansfield and Francis (1986) to be approximately three times that estimated by Huang et al. (1984b). The difference between time of the year in sampling, May for Stansfield and Francis and September for Huang et al. (1984b), is likely the major cause for the differences in estimates of pond water volumes. As a consequence, there was not a major difference in estimates of the radionuclide inventory between the two studies. For example, the total inventory of strontium-90 and cesium-137 (approximately 1 GBq or 0.03 Ci for each radionuclide) was not substantially different than the estimate by Huang et al. (1984b) (2.1 and 0.5 GBq, for cesium-137 and strontium-90, respectively).

Huang et al. (1984b) and Stansfield and Francis (1986) have measured concentrations of potential contaminants in the sediment of the OHF impoundment. The principal purpose of the latter study was to determine if the sediment would be classified as a hazardous waste under CERCLA or RCRA regulations. Thus, the emphasis in this study was centered more along the analyses of sediment for nonradioactive

contaminants whereas the earlier study was principally concerned about the radiological characteristics of the impoundment in support of decontamination and decommissioning (D&D) activities. In the federal regulation 40 CFR 261 promulgated under RCRA (USEPA 1980), a solid waste is hazardous if it exhibits any of the defined characteristics of ignitability, corrosivity, reactivity, or extraction procedure (EP) toxicity. The EP toxicity is of primary concern as the inherent physical and chemical characteristics of the sediment rule out classification as a hazardous waste based on ignitability, or reactivity. Since the pH of the sediment was 5.2, the waste will not be classified as having the characteristic of corrosivity as defined under the regulations. The EP toxicity characteristic is based on measured concentrations of eight elements of the National Interim Primary Drinking Water Standard (NIPDWS) and six herbicides and pesticides in the filtrate of a 24-hr solid waste extraction test (USEPA 1982). If the levels of these constituents exceed established maximum permissible concentrations, then the waste is considered hazardous based on the characteristic of toxicity.

The sediment was thoroughly tested for its toxicity characteristics by replicate samplings and extractions of the sediment by the EP. For example, the sediment was tested for concentrations of metals in its EP extracts a total of six times (sediment collected from three sample locations at two dates). Analyses for the herbicides/pesticides in the extracts were conducted three times (sediment collected from three sample locations in the impoundment at a single sampling date). Sampling was conducted in November of 1984 and

February of 1985. A summary of these measurements are presented in Table 4.12.4.

Where analytical detection limits were low enough, the concentrations of the RCRA-regulated constituents measured in the EP extracts were well below the RCRA maximum allowable concentrations. Selenium concentrations in the EP extracts of the sediment samples collected in November of 1984 were determined by inductively coupled plasma (ICP) spectroscopy. For these samples, the detection limit was 2.4 mg/L, 1.4 mg/L in excess of the RCRA limit. The EP extracts of sediment samples taken in February 1985 were analyzed for selenium by atomic absorption spectroscopy (AA) in which the detection limit was 0.01, well below the maximum allowable RCRA concentration. In all six EP extracts, the selenium concentrations were below detection by the particular analytical technique used. The mean of the two detection limits (2.4 and 0.01) is 1.2 mg/L. Thus, the mean selenium concentration reported in Table 4.12.4 (1.2 mg/L) is an artifact in reporting the detection limits of the analytical procedure and should not imply that the concentrations of selenium in the EP extracts to be in excess of the maximum allowable RCRA concentration.

Total concentrations of nonradioactive contaminants in sediment samples of the OHF impoundment were determined by both Huang et al. (1984b) and Stansfield and Francis (1986). The method of expressing their concentration, however, differed. Stansfield and Francis expressed the concentrations on the dry weight basis while Huang et al. (1984b) expressed the concentrations on a wet weight basis (Table 4.12.5). All things considered equal, the concentrations

Table 4.12.4 Concentration of RCRA regulated constituents in EP Extracts of sediment from the OHF impoundment (Source: Stansfield and Francis 1986).

Constituent	Maximum allowable concentration	Measured concentration
	mg/L	
Arsenic	5	<0.60
Barium	100	0.307
Cadmium	1	0.0092
Chromium	5	0.0492
Lead	5	<0.0232
Mercury	0.2	0.065
Selenium	1	<1.2
Silver	5	<0.2
Endrin	0.02	<0.0001
Lindane	0.04	0.0001
Methoxychlor	10	<0.0002
Toxaphene	0.5	<0.002
2,4-D	10	<0.005
2,4,5-DTP	1	<0.005

Mean concentrations from six replicated EP Extracts for metals (sediment from three sample locations at two dates). For the herbicides/pesticides, the concentrations are mean concentrations from three replicated EP Extracts (sediment from three sample locations at a single date).

Table 4.12.5 Concentration and inventory of nonradioactive contaminants in sediment from the OHF impoundment. Data in the second and third column are from Stansfield and Francis (1986) and the data in the last two columns are from Huang et al. (1984b).

Contaminant	Concentration (mg/kg)	Inventory (kg)	Concentration (mg/kg)	Inventory (kg)
Antimony	<194	<3.69	<1.0	<0.08
Arsenic	<120	<2.28	1.1	0.09
Barium	320	6.07	NR	NR
Boron	132	25	NR	NR
Cadmium	<6.1	<0.12	<2.0	<1.60
Chromium	342	6.50	17.0	1.30
Cobalt	22.4	0.43	NR	NR
Copper	140	2.66	18.0	1.40
Iron	11200	403	NR	NR
Lead	<156	<2.97	6.0	0.48
Manganese	370	7.03	NR	NR
Mercury	NR	NR	<2.3	<0.18
Molybdenum	<15.5	<0.29	NR	NR
Nickel	149	2.84	NR	NR
PCB	2.9	0.05	3.5	2.80
Selenium	<240	<4.55	<0.8	<0.06
Silver	<40.9	<0.78	NR	NR
Vanadium	79.1	1.50	NR	NR
Zinc	162	3.07	16.0	1.20

Sediment concentrations by Stansfield and Francis (1986) are on a dry weight basis and represent a mean of three sediment samples taken from the north, center, and south sections of the impoundment. Inventory calculated by Stansfield and Francis (1986) was based on a total volume of sediment equal to 5.5×10^4 L, a wet bulk density of 1.2 kg/L, and a moisture content of 71%. Sediment concentrations by Huang et al. (1984b) are on a wet weight basis and represent a mean of five sediment samples taken from the center of the pond from a southwest to northeast direction. Inventory calculated by Huang et al. (1984b) was based on a total volume of sediment equal to 3.7×10^4 L and a wet bulk density of 2.16 kg/L. NR = not reported.

expressed on a dry weight basis should be higher than those on the same sediment on a wet weight basis. This is generally the case except for the case of PCBs in the sediment. Here, the concentrations determined by Huang et al. (1984b) are higher than those determined by Stansfield and Francis. Huang et al. (1984b) sampled the sediment in September of 1983 while Stansfield and Francis sampled in February of 1985, which should not be a major factor assuming the degradation rates of PCBs to be relatively slow. Different volumes of sediment and bulk densities of the sediment were also used by each study to determine the total inventory. The basis of calculation for each of the studies is footnoted at the bottom of Table 4.12.5. Generally speaking, there was not a profound difference in the contaminant inventory between the two studies. The exception was the major difference in the estimated inventory of PCBs. Huang et al. (1984b) estimated the PCB inventory to be close to 3 kg while that calculated by Stansfield and Francis (1986) was approximately 50 grams.

A similar trend in the concentrations of radionuclides in sediment taken from the OHF impoundment was noted between the two studies. For example, measurements by Huang et al. (1984b) were generally higher than those determined by Stansfield and Francis (1986), even though Huang et al. (1984b) expressed the concentration on the wet weight basis and Stansfield and Francis on a dry weight basis (see Table 4.12.6). The major exception was the measurements for strontium-90. In this case, concentrations determined by Huang et al. (1984b) were four times smaller than those determined by Stansfield and Francis (1986). Ironically, the inventory of strontium-90 in the

Table 4.12.6 Concentration and inventory of radionuclides in sediment from the OHF impoundment. Data in the second and third column are from Stansfield and Francis (1986) and the data in the last two columns are from Huang et al. (1984b).

Radionuclide	Concentration Bq/g	Inventory GBq	Inventory Ci	Concentration Bq/g	Inventory GBq	Inventory Ci
Gross alpha	102	1.93	0.05	NR	NR	NR
Gross beta	92700	1761	48	NR	NR	NR
Cesium-134	<20	<0.38	0.01	100	8.1	0.22
Cesium-137	125000	2370	64	180000	14000	388
Europium-154	<22.7	<0.43	<0.01	390	31	1
Uranium-234	<33.3	<0.63	<0.01	NR	NR	NR
Uranium-238	<612	<11.6	<1	0.23	0.0055	<0.01
Americium-241	<20.0	<0.38	<0.01	2.8	0.22	<0.01
Cobalt-60	608	11.6	0.31	740	60	1.6
Strontium-90	38800	737	20	9700	770	21

Sediment concentrations by Stansfield and Francis (1986) are on a dry weight basis and represent a mean of three sediment samples taken from the north, center, and south sections of the impoundment. Inventory calculated by Stansfield and Francis (1986) was based on a total volume of sediment equal to 5.5 x 10^4 L, a wet bulk density of 1.2 kg/L, and a moisture content of 71%. Sediment concentrations by Huang et al. (1984b) are on a wet weight basis and represent a mean of five sediment samples (except the Europium-154 analysis which is from a single sediment sample) taken from the center of the pond from a southwest to northeast direction. Inventory calculated by Huang et al. (1984b) was based on a total volume of sediment equal to 3.7 x 10^4 L and a wet bulk density of 2.16 kg/L. NR = not reported.

sediment by both studies were very close (737 and 770 GBq or 20 and 21 Ci). The basis for calculation of the inventories also differed with respect to volume of sediment, density, etc. (see footnotes at bottom of Table 4.12.6 for an explanation for the basis of calculation). As in the pond water, the bulk of the radioactivity is due to cesium-137 and strontium-90 and only a small but detectable fraction (<0.05%) is transuranic related.

The mean concentrations determined (Stansfield and Francis 1986) in groundwater sampled in the first two quarters of 1985 from three downgradient wells are presented in Table 4.12.7. The mean concentrations of all RCRA-regulated constituents were below maximum allowable levels except those for gross alpha, gross beta, and coliform bacteria. Radioactivity as well as PCBs were detected in the upgradient well, indicating potential contamination by either leakage from the waste pits or from the many underground pipes in the OHF area. Many of the underground pipes lead to the waste pits located uphill and slightly to the southeast of the impoundment. Activity in this well could also have heen caused by the leaching of radionuclides from wastes previously disposed of in the abandoned SWSA 5 burial ground. Counts of coliform bacteria in groundwater upgradient as well as downgradient were in excess of the NIPDWS. These counts may result from wildlife habitat such as waterfowl and terrestrial animals known to be in the area. Radioactivity was detected in all monitoring wells; however, well number 4 contained the highest concentration of cesium-137 and strontium-90.

Table 4.12.7 Groundwater concentrations in downgradient wells from the OHF Basin. Data taken from Stansfield and Francis (1986).

	Maximum level mg/L	Measured mg/L
National Interim Primary Drinking Water Standards (NIPDWS)		
Arsenic	0.05	<0.0017
Barium	1	0.3907
Cadmium	0.01	<0.0037
Chromium	0.05	<0.0456
Fluoride	1.4-2.4	<1
Lead	0.05	0.0267
Mercury	0.002	<0.0001
Nitrate-N	10	3.83
Selenium	0.01	<0.0023
Silver	0.05	<0.07
Endrin	0.0002	<0.0001
Lindane	0.004	<0.0001
Methoxychlor	0.1	<0.0002
Toxaphene	0.005	<0.002
2,4-D	0.1	0.005
2,4,5-TP Silvex	0.01	0.005
	Bq/L	Bq/L
Radium-226	0.19	0.104
Gross alpha	0.556	<3.08
	mR/yr	mR/yr
Gross beta	4	382
	counts/100 mL	counts/100 mL
Coliform bacteria	1	12.7
	mg/L	mg/L
Parameters establishing groundwater quality		
Chloride	ND	14.7
Iron	ND	12.1
Manganese	ND	2.37
Phenols	ND	<0.001
Sodium	ND	22.8
Sulfate	ND	17

Table 4.12.7 (CONTINUED)

	mg/L	mg/L
Nonregulated parameters		
PCB's	ND	0.0001
	Bq/L	Bq/L
Tritium	ND	94333
Cesium-137	ND	1.71
Strontium-90	ND	223
Parameters used as indicators of groundwater contamination		
pH	ND	6.49
Specific conductance (uS/cm)	ND	376
Total organic carbon (mg/L)	ND	6.37
Total organic halogen (mg/L)	ND	0.03

Mean concentrations [in mg/L, unless otherwise noted (gross beta is in Bq/L)] taken from three monitoring wells the first two quarters of 1985 (total of six analyses).

Soil samples were collected from the pond perimeter. A single sample 1.2 to 1.8 m below-grade, corresponding to the depth of the water-sediment layer in the pond, contained 78 pCi/g (2.9 Bq/g) suggesting some leakage from the pond.

Several additional reports contain information of the facility including the pond. The development and design of the OHF facility was described by deLaguna et al. (1968). The preliminary radiological characterization study (Huang et al. 1984b) preceeded the decommissioning study (Reed 1984).

The most knowledgeable persons of the OHF facility and the pond are:

C. W. Francis, Environmental Sciences Division, ORNL, 615/574-7257
T. Tamura, Environmental Sciences Division, ORNL, 615/576-8256
L. Lasher, Operations Division, ORNL, 615/576-5223

4.13 HOMOGENOUS REACTOR EXPERIMENT (HRE) POND (7500 area)

The HRE Pond is located in Melton Valley (Fig. 4.12.1), 0.4 km (0.6 mile) southeast of the ORNL complex. It is situated south of Building 7500, above Melton Branch. The pond, dedicated in 1955 to serve the Homogenous Reactor Test, a 5 Mw experimental reactor, operated between 1958 and 1961. With a capacity of 1.2×10^6 L (300,000 gallons), the pond was of rectangular design (23 x 24 m) with an average depth of 3.9 m (12.8 ft). Reed (1984b) describes the facility in a decommissioning study. Early in the program the pond received only slightly contaminated condensate from the evaporator. As the program progressed, the flow to the pond was shield water. This water was exposed to reactor circuits during maintenance. Pond water

was retained temporarily and released to the creek in allowable concentrations (~0.1 Ci/8 hr period).

In 1984 a characterization study was completed (Huang et al. 1984c). A schematic of the HRE facility (Fig. 4.13.1) identifies the location of the waste holding pond. Results of a walk-over survey are presented in Fig. 4.13.2. The maximum direct beta-gamma exposure detected near the pond location was 2.2 mrad/h (1 mrad = 10 μGy). Only eight areas exceeded 0.1 mrad/hr. At the locations of areas exceeding the 0.1 mrad/hr exposure, soil samples were collected from the top layer (30 cm). Radiological results of these eight samples are presented in Table 4.13.1. The contamination is primarily Cs-137 and Sr-90, with trace amounts of Pu-238,239, Am-241, and Cm-244. All samples contained Cs-137 activity between 0.06 and 6 pCi/g (1.6 and 160 Bq/g). All samples contained Sr-90 activity. The highest Cs-137 activity (6 pCi/g) was detected above the northeast corner of the pond. The highest Sr-90 activity, 1.8 pCi/g (48 Bq/g), was found approximately 15 m (50 ft) southeast of the pond area.

Subsurface soil cores were collected to 5.5 m deep (average depth 3.5 m). Cores 14-19 were collected in the vicinity of the filled-in pond (Fig. 4.13.2). The major gamma-emitting radionuclides in the cores were Cs-137 and Co-60. Samples from locations 15 and 17 identified Cs-137 concentrations above backgrounds typical of East Tennessee soils (Table 4.13.2). Samples from the same areas also contained trace amounts of Co-60 and generally higher Sr-90 concentrations. Concentrations of alpha-emitting radionuclides (Pu-238,239, Am-241, and Cm-244) were present but in insignificant

Fig. 4.13.1 HRE-2 facilities.

ORNL-DWG 83-18117

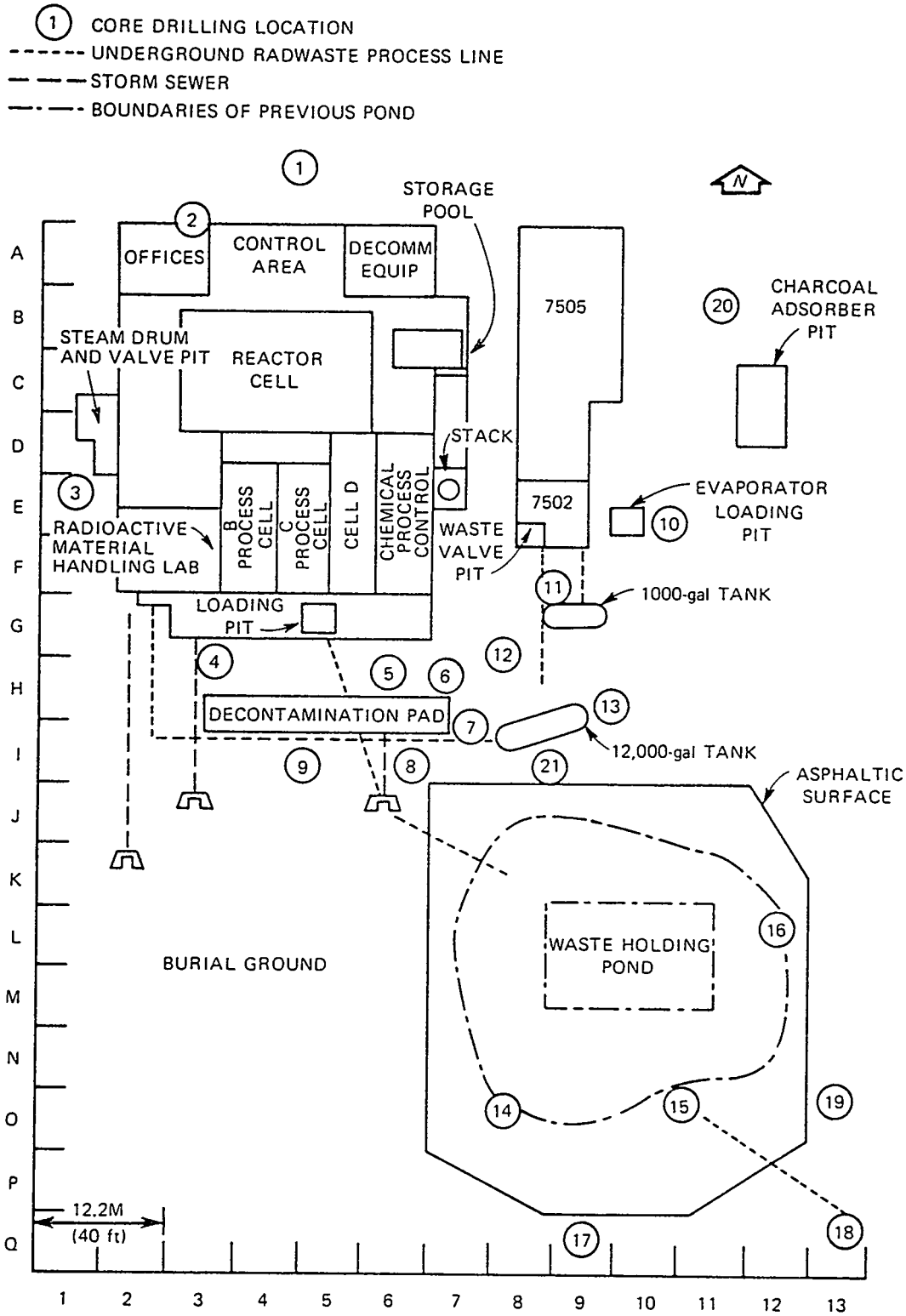


Fig. 4.13.2 Direct beta-gamma readings in mrad/h (1 mrad = 10 μGy) above 0.1 mrad/h at the ground surface near the HRE-2.

ORNL–DWG 83-18118

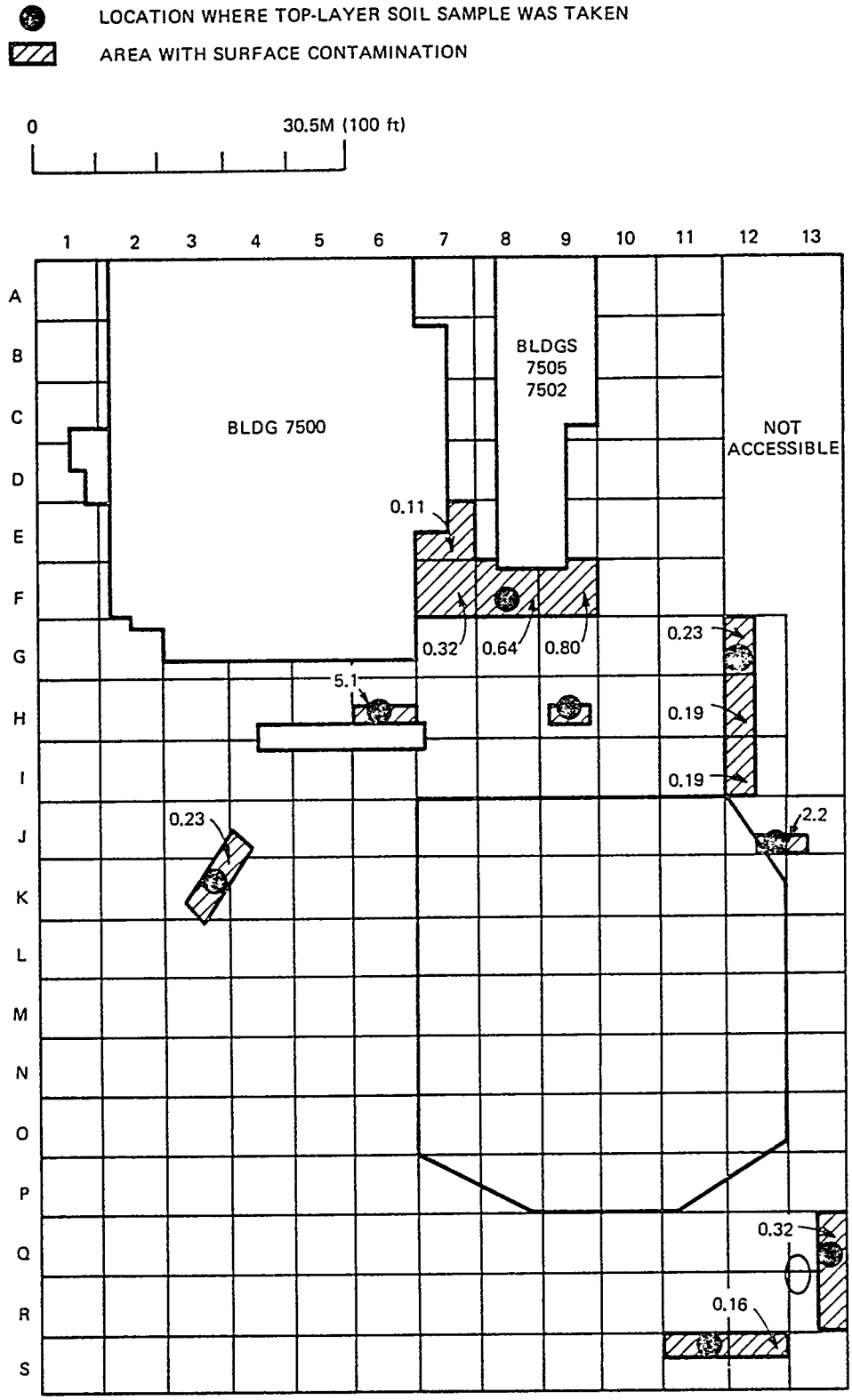


Table 4.13.1 Radionuclide concentrations of contaminated soil samples from the top layer (30 cm) of soil near the HRE-2 facility. Source: Huang 1984c.

Sample identification	Radioactivities (Bq/g)[a]						
	^{137}Cs	^{60}Co	^{90}Sr	^{239}Pu	^{238}Pu	^{241}Am	^{244}Cm
6H	43	ND[b]	10	0.25×10^{-3}	$<0.3\times10^{-3}$	0.31×10^{-2}	0.46×10^{-3}
9H	1.6	ND	14	0.15×10^{-2}	0.62×10^{-3}	0.22×10^{-2}	0.19×10^{-2}
8F	5.7	0.01	1.2	0.62×10^{-2}	0.4×10^{-3}	0.38×10^{-2}	0.38×10^{-3}
12G	4.8	ND	1.1	0.29×10^{-3}	$<0.4\times10^{-3}$	0.73×10^{-2}	0.13×10^{-3}
3K[c]	48	0.01	0.48	1.9	0.6	<0.1	0.4
12J	160	0.05	8.4	9.0	1.2	0.4	0.4
13Q	14	ND	8.2	$<0.3\times10^{-3}$	0.11×10^{-2}	0.13×10^{-2}	0.58×10^{-2}
11S	48	0.13	24	1.5	0.2	0.6	15

[a] 1 Bq/g = 27 pCi/g.
[b] ND indicates nondetectable.
[c] Contains ^{134}Cs (0.037 Bq/g).

Table 4.13.2 Radionuclide concentrations in deep coring soil samples near the HRE-2 facility. Source: Huang et al. 1984c.

Sample identification	^{137}Cs	^{90}Sr	Sample identification	^{137}Cs	^{90}Sr
	(Bq/g)[a]			(Bq/g)	
1-3	0.7×10^{-2}	0.25×10^{-1}	15-6[b]	0.11	45
1-4	ND[c]	0.16×10^{-1}	15-8	ND	0.27
1-5	0.4×10^{-2}	0.11×10^{-1}	16-3[b]	0.33	12
1-6	0.7×10^{-2}	0.11	16-4[b]	0.20	12
2-3[d]	ND	0.3×10^{-1}	16-5[b]	0.15×10^{-1}	1.1
2-5[d]	ND	0.7×10^{-1}	16-6	0.15×10^{-1}	0.18
2-6[d]	ND	0.74×10^{-1}	17-1[b]	5.9	2.4
3-3	0.7×10^{-2}	0.4×10^{-1}	17-2[b]	5.2	2.2
6-3	0.8×10^{-1}	0.13	17-3[b]	3.0	2.3
7-4[d]	ND	0.3×10^{-2}	17-4[b]	0.29	0.61
8-4[d]	ND	0.78×10^{-1}	17-5[b]	0.29	24
9-2	0.7×10^{-1}	0.7×10^{-2}	17-6[b]	ND	3.9
11-3	0.8×10^{-1}	0.1	18-1[b]	0.2	0.65
12-3[b]	0.15	0.2	18-4[d]	ND	0.36
13-2	0.4×10^{-1}	0.6×10^{-2}	18-5[d]	ND	0.17
13-3	0.54	0.17	19-4	0.6×10^{-1}	0.89×10^{-1}
13-5	0.9×10^{-1}	0.78×10^{-1}	19-7[d]	ND	0.14
13-6	ND	0.35×10^{-1}	20-5[d]	ND	0.3×10^{-1}
14-3[b]	0.48	0.16	20-6	0.1×10^{-1}	0.84×10^{-1}
14-6	ND	15	20-9[d]	0.3×10^{-2}	0.1
14-7	ND	0.21	21-3	ND	0.84×10^{-1}
15-3[b]	2.2	0.51	21-5[d]	ND	0.18
15-5[b]	0.03	0.15	21-7[d]	ND	0.17
DOE Guide	0.75	0.75		0.75	0.75

[a] 1 Bq = 27 pCi/g.
[b] Contained uranium activity.
[c] Nondetectable.
[d] The ^{90}Sr result is from Cerenkov counting.

quantities. In 1970, the pond was filled, graded and sprayed with weed killer before a top layer of crushed stone and asphalt was applied. Radiological content inside the perimeter of the filled-in pond was not determined, but it was estimated (Huang et al. 1984c) that from 50 to 100 Ci of radioactivity (Cs-137, Co-60, and Sr-90) are presently in the pond. A more recent study (Stansfield and Francis 1986a) estimated a total inventory of >840 Ci in the impoundment fill and sediment (Table 4.13.3). Approximately 89% of the total is attributed to gross-beta activity, 9% to strontium-90, and 2% to cesium-137.

Several persons have a good knowledge of the HRE facility and the retention pond. Most notably are:

Pugh, Luther P., Operations Division, ORNL, 615/576-5006
Yarboro, O. O., Fuel Recycle, ORNL, 615/574-7068

Yarboro was personally involved in the discharges into the pond and controlled releases from the pond to the creek.

4.14 LOW INTENSITY TEST REACTOR (LITR) POND (3075)

In 1951 the LITR served as a training reactor and later was converted to a testing reactor. Located in the northcentral area of the ORNL complex (Bethel Valley), the reactor used two 6.8×10^4 L (18,000 gallons) retention ponds for holding slightly contaminated waste water (Reed 1984c). These ponds (now stabilized by earth fill) are approximately 100 m east of Building 3075. Two engineering drawings (D-9236 and D-22210) provide details of locations and dimensions. Each pond was about 2.4 x 14 m in size and ranged from 1.8 to 2.4 m deep. A radiological characterization study (Rohwer 1985)

Table 4.13.3 Inventory of radionuclides in HRE impoundment fill/sediment (Source: Stansfield and Francis 1986a).

Constituent	Total inventory GBq	Total inventory Ci
Gross alpha	<0.001	<0.001
Gross beta	27600	750
^{137}Cs	604	16
^{234}U	0.123	0.003
^{235}U	0.017	<0.001
^{238}Pu	<0.001	<0.002
^{238}U	0.079	0.002
^{239}Pu	0.005	<0.002
^{241}Am	0.001	<0.001
^{60}Co	0.062	0.002
^{60}Sr	2750	75

confirmed that radioactivity materials exist at the site of the two ponds. In that study, soil and sediment samples were collected from each pond site. The maximum activities for cesium-137 and cobalt-60 were 32,000 Bq/kg (0.86 µCi/kg) and 6,200 Bq/kg (0.17 µCi/kg), respectively. Cesium-137 and cobalt-60 activities were detected in most samples (Table 4.14.1).

Results of Sr-90, Pu-238, and Pu-239 analyses of selected samples (samples in which elevated gamma activities were detected) are presented in Table 4.14.2. For comparison, background radioactivities in soil are provided. The average activities of Sr-90, Pu-238, and Pu-239 are significantly higher than background. Samples with significant amounts of Cs-137 and Co-60 are also contaminated with Sr-90, Pu-238, and Pu-239. It is estimated that the radioactive inventory consists of 20 mCi of Cs-137, 1 mCi of Sr-90, and 100 µCi of Pu-239.

Samples from the bottom of each pond, 2.1 to 2.4 m belowground, were analyzed for heavy metal content using spark source mass spectroscopy (SSMS). Results of the heavy-metal concentrations are shown in Table 4.14.3. For comparison with shallow soil, samples 2-2 and 2-3 are shown in Columns 7 and 8.

Due to the high detection limit for Hg (20 ppm) using SSMS, mercury concentrations were determined using the vapor atomic absorption method (AA). Results of mercury analyses using AA method are shown in Table 4.14.4. Concentrations in all samples are within the normal range for background concentrations of Hg in soil in the United States of 0.01 to 4.7 ppm (National Academy of Sciences 1978).

Table 4.14.1 Quantitative analysis of beta and gamma-emitting radionuclides in soil and sediment samples near the LITR ponds.

Sample number	Depth (cm)	Concentration[a] Cs-137	Co-60
1-1*	0.0-0.3	19	ND**
1-2	0.3-0.6	11	ND
1.3	0.6-0.9	7.4	ND
1-4	0.9-1.2	6.7	ND
1-5	1.2-1.5	13	ND
1-6	1.5-1.8	6.3	ND
1-7	1.8-2.1	3000	270
1-8	2.1-2.4	13	ND
2-1	0.0-0.3	85	11
2-2	0.3-0.6	59	5
2-3	0.6-0.9	81	9
2-4	0.9-1.2	590	120
2-5	1.2-1.5	11000	2400
2-6	1.5-1.8	32000	6200
2-7	1.8-2.1	4800	960
2-8	2.1-2.4	25000	5200
3-1	0.0-0.3	41	6
3-2	0.3-0.6	26	ND
3-3	0.6-0.9	30	ND
3-4	0.9-1.2	25	ND
3-5	1.2-1.5	33	ND
3-6	1.5-1.8	30	ND
3-7	1.8-2.1	36	2
3-8	2.1-2.4	41	13
4-1	0.0-0.3	37	11
4-2	0.3-0.6	12	ND
4-3	0.6-0.9	15	ND
4-4	0.9-1.2	17	ND
4-5	1.2-1.5	14	ND
4-6	1.5-1.8	7.8	ND
4-7	1.8-2.1	37	ND
4-8	2.1-2.4	18900	890

Table 4.14.1 (CONTINUED)

Sample number	Depth (cm)	Concentration[a] Cs-137	Co-60
5-1	0.0-0.3	15	ND
5-2	0.3-0.61	9	ND
5-3	0.6-0.9	23	ND
5-4	0.9-1.2	14	ND
5-5	1.2-1.5	18	ND
5-6	1.5-1.8	25	ND
5-7	1.8-2.1	2700	370
5-8	2.1-2.4	5900	560
6-1	0.0-0.3	15	370
6-2	0.3-0.6	18	4
6-3	0.6-0.9	20	3

[a]Bq/kg dry weight.

*The number following the dash represents the maximum depth in feet and the number preceding the dash represents the core site. 1 Bq = 27 pCi.

**ND indicates not detected.

Table 4.14.2 Strontium-90, Pu-238, and Pu-239 concentrations (Bq/kg dry weight) in LITR soil and sediment samples from 1.5 to 2.4 m below grade.

Identification	Depth (m)	Sr-90	Pu-238	Pu-239
1-7*	1.8-2.1	220	2.3	3.8
1-8	2.1-2.4	230	2.1	31
2-5	1.2-1.5	490	12	59
2-6	1.5-1.8	1500	24	230
2-7	1.8-2.1	300	2.9	20
2-8	2.1-2.4	1000	25	130
3-8	2.1-2.4	37	<0.2	0.46
4-7	1.8-2.1	<50	<2	1.2
4-8	2.1-2.4	700	20	42
5-7	1.8-2.1	180	1.3	33
5-8	2.1-2.4	240	3.5	40
Background**		7	0.2	2.7

*The number following the dash represents the same depth in feet and the number preceding the dash represents the core site.

**Annual average radioactivity in soil samples from uncontrolled areas in East Tennessee (Auxier, J. A. and T. W. Oakes, 1982).

Table 4.14.3 Heavy-metal concentrations in LITR sediment and soil samples concentration (mg/kg).*

Heavy metal	Sample identification						
	1-8**	2-8	3-8	4-8	5-8	2-2	2-3
As	50	<5	<5	30	<5	5	5
Cd	<1	<1	<1	<1	<1	<1	<1
Cr	2000	100	100	300	100	200	50
Cu	100	50	20	100	20	10	10
Ni	200	100	100	200	100	100	100
Pb	90	5	5	100	5	100	700
Se	40	30	30	30	30	30	30
Ti	10000	10000	10000	10000	10000	2000	1000
V	500	50	20	300	20	30	20
Y	200	3	10	70	10	10	10
Zn	300	100	30	200	5	100	100

*The results are within +100% and -50%.

**The number following the dash represents the maximum sample depth in feet and the number preceding the dash represents the core site.

Table 4.14.4 Mercury concentrations in samples from 2.1 to 2.4 m below ground at all LITR core sites.

Sample identification	mg/kg
1-8*	0.39
2-8	0.3
3-8	0.04
4-8	0.31
5-8	0.11

*The number following the dash represents the maximum sample depth in feet and the number preceding the dash represents the core site.

Results from gas chromatographic analyses show that PCB concentrations in samples from 2.1 to 2.4 m below ground are less than 0.15 ppm.

Sample inventory is considered incomplete as numerous stones were encountered during the survey and the inventory was not below the bottom (2.4 m) of the pond. This unsampled region contains an unknown portion of the precipitate in which additional contaminants may be concentrated.

5. REGULATORY STATUS

As promulgated in the Department of Energy (DOE) Order 5480.14, it is the policy of DOE to identify and evaluate potential problems associated with inactive hazardous waste disposal sites at DOE facilities. The order implements this policy by providing instructions for a Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) program.

Through the examination of DOE and ORNL information sources (records, reports, and personal interviews), approximately 141 sites have been identified as potentially requiring remedial actions. These sites included solid waste storage areas, waste ponds and seepage pits, radioactive waste processing and transfer facilities, research laboratories, environmental research areas, hazardous waste spill sites, reactors, and radioisotope development facilities. Of these sites, 81 were identified as potential CERCLA sites under the directive of DOE Order 5480.14. The remaining sites are controlled under the Resource Conservation and Recovery Act (RCRA), the Toxic Substances

Control Act (TSCA), the Underground Injection Control Program (UIC), or the DOE Surplus Facilities Management Order (DOE Order 5820.2).

The DOE CERCLA order requires that two hazard ranking systems be used to estimate the relative hazard to human health, safety, and the environment from the potential migration of hazardous substances from the site. For nonradioactive sites, the Environmental Protection Agency's (EPA) Hazard Ranking System (USEPA 1985) is to be used, whereas DOE's modified Hazard Ranking System (Hawley and Napier 1985) is to be used for radioactive sites. In both systems, the potential hazard is estimated by calculating a score (migration score S_M) based on the potential for migration of hazardous constituents through groundwater, surface water, or atmospheric pathways.

The scores are used by the EPA to distinguish between inactive waste sites that may pose a human health or environmental risk from those that do not, and to develop a national inventory of hazardous waste sites with the most hazardous site receiving top priority for remedial action. In order to be included on the NPL, a site must have a minimum migration score of 28.5. Nix et al. (1986) determined the modified Hazard Ranking Scores for several ORNL CERCLA sites included in this report. Included were six process ponds (SWSA Sludge Basin-7835, Abandoned Filled-In Waste Basin-3512, Waste Settling Basin-3513, Old Hydrofracture Pond-7852 area, Homogenous Reactor Experiment Pond-7500 area, and Low Intensity Test Reactor Pond-3075). The greater score between the combined pathway scores (air, groundwater, surface water) and the chemical-radioactive scores formulates the basis for ranking the site. Of the six ponds evaluated,

only three (Waste Settling Basin, Old Hydrofracture Facility Pond, and the Homogenous Reactor Experiment Pond) had been characterized for chemical and radiological constituents.

In the absence of adequate documentation or other evidence, such as the presence of 8.2×10^5 L of sludge in the 3512 sludge basin, calculations for waste characteristics were based on minimum values. Table 5.1 summarizes the modified Hazard Ranking System (S_M) for the six ponds. Based on the HRS scores and current EPA guidance, none of these sites is a candidate for inclusion on the National Priorities List.

6. DISCUSSION

It has been determined that all of the process ponds described in this report are potentially subject to the provisions of RCRA [Section 3004(u)], and a Preliminary Assessment/Site Investigation (PA/SI) will be conducted at each site to further document the site characteristics and determine the need for follow-on efforts (Rohwer 1986). Pending the outcome of the PA/SI, these sites may be governed by RCRA 3004(u) Regulations, CERCLA, or DOE Order 5820.2.

EP Toxicity analyses were recently completed for seven RCRA ponds (3539, 3540, 3524, 7905, 7906, 7907, and 7908). Included in the analyses were metals, pesticides, herbicides, and organic concentrations for influent water and sludge (sediment). None of the data for metals, pesticides, or herbicides exceeded the maximum concentration of contaminants for characteristics of EP Toxicity. Only a few of the organic constituents were present above detection limits

Table 5.1 Modified Hazard Ranking System migration scores (S_M) for some ORNL process ponds (from Nix et al. 1986).

Site	Process Ponds	Migration score (S_M)
SWSA 5 Sludge Basin (7847)		1.9
Abandoned Filled-In Waste Basin (3512)		0.6
Waste Holding Basin (3513)		5.3
Old Hydrofracture Pond (7852 area)		5.3
Homogenous Reactor Experiment Pond (7500 area)		5.3
Low Intensity Test Reactor (LITR) Pond (3075)		0

(Kitchings 1986). No quantitative data are available for radioactive characterization of these pond waters or sediments. There is a paucity of data for Basins 7835 (SWSA 5 Process Waste Sludge Basin) and the Abandoned Filled-In Basin (3512). No data is available for metal, organic, herbicide, or pesticide concentrations in the pond waters or sediments (sludge) of the Low Intensity Test Reactor (LITR) Basin (3075). Recently completed characterization studies of the Old Hydrofracture Pond (7852 area) and the Homogenous Reactor Experiment (HRE) Pond-7500 area provide data on EP-Toxic Organics and Metals.

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DATE: July 1986

SUBJECT: Inventory of ORNL Remedial Action Sites:
4. White Oak Creek Watershed

TO: T. E. Myrick

FROM: N. T. Edwards

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LETTER REPORT RAP86-23

(Internal Use Only)

Inventory of ORNL Remedial Action Sites: 4. White Oak Creek Watershed*

N. T. Edwards

July, 1986

Environmental Sciences Division
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37831

Operated by

Martin Marietta Energy Systems, Inc.
for the
U.S. Department of Energy

*Based on work performed at Oak Ridge National Laboratory, operated by Martin Marietta Energy Systems, Inc., under Contract No. DE-AC05-84OR21400 with the U.S. Department of Energy.

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ABSTRACT

This report contains an abbreviated account of the history of White Oak Creek Watershed after construction of The White Oak Dam and of laboratory activities and waste disposal operations which have contributed radionuclides and other potentially hazardous chemicals to both aquatic and terrestrial areas within the Watershed. It also summarizes radionuclide monitoring data in White Oak Lake, White Oak Creek, and Melton Branch Creek. Summarized data from much of radionuclide research conducted on the Watershed are presented, including concentration data in groundwater, lake sediments, gravels along stream beds, aquatic biota, and terrestrial biota and soil. These concentration data are used to identify key sources of contamination and for locating areas and biota some distance from these sources which contain relatively large amounts of contaminants.

The report also identifies some recent remedial action activities in the area of waste disposal aimed at reducing releases of pollutants to the Watershed environment. New research activities aimed at better understanding the fate and effects of pollutants of special concern in White Oak Creek Watershed, and recommendations for further research are discussed.

Key reports which address levels of non-radiochemical contaminants in the Watershed are referenced.

1. OBJECTIVES

The primary purpose of the Remedial Action Program is to provide surveillance and maintenance, corrective measures, and facility decommissioning necessary to place the Oak Ridge National Laboratory (ORNL) in compliance with applicable environmental and health regulations for radioactive and hazardous chemical wastes. Individual sites will be decontaminated in accordance with priorities prescribed to control the potential for human exposure and relative magnitude of ecological impacts. Remedial action is a component of Laboratory's Environmental Restoration and Facilities Upgrade Program (ERFU). In the program, remedial actions will be conducted at locations where past management practices or research activities have resulted in contamination of facilities or the environment that necessitate decommissioning or corrective actions. Remedial actions are divided into two activities: (1) Site Decommissioning Program (Surplus Contaminated Facilities Program-SCFP and Surplus Facilities Management Program-SFMP) and (2) the Site Corrective Measures Program (SCMP).

The Site Decommissioning Program is designed for management of surplus facilities (primarily structures), contaminated with radioactive materials and hazardous chemicals from the end of their operating life until final facility disposition. The Site Corrective Measures Program is designed for the management of inactive sites (process ponds, spill sites, environmental research areas, etc.), contaminated with radioactive materials

and hazardous wastes, leading to the eventual stabilization of these sites in accordance with environmental regulations.

Remedial Action Program Sites at ORNL comprise 13 distinct functional groups (Table 1) among the several programmatic categories: (1) SCMP, (2) SCFP, and (3) SFMP. A series of reports is being prepared in which an inventory of each area is presented in a condensed summary. Each summary is the product of pertinent information abstracted from readily available literature and knowledgeable individuals on known contaminated sites requiring remedial action. The information includes site location and description, history, contaminants and their characteristics, extent of contamination, regulatory status, geological, hydrological, and ecological data, lists of major studies, identification of data deficiencies for environmental impact assessment and regulatory purposes, and knowledgeable individuals contacted.

The subject of this report is the White Oak Creek Watershed. Some of the contents overlap with that of other reports, because most of the other remedial action sites are located within the White Oak Creek Watershed. In this report the collective contribution of the other remedial action sites to the environment of the White Oak Creek Watershed are summarized. Emphasis is on identifying those areas of the Watershed that are impacted and possible contaminant source points.

Table 1. Inventory of ORNL Remedial Action Sites

1. Solid Waste Storage Areas
2. LLW Seepage Pits and Trenches
3. Process Ponds
4. White Oak Creek Watershed
5. LLW Line Leak Sites
6. Environmental Research Areas
7. Hazardous Waste Sites
8. Radioisotope Processing Facilities
9. Experimental Reactor Facilities
10. Radwaste Facilities
11. Research Laboratories
12. Abandoned Injection Wells
13. Other Contaminated Sites

2. BACKGROUND

White Oak Creek and its tributaries drain a 1550 ha area, receiving liquid wastes from various Oak Ridge National Laboratory operations and seepage from solid waste burial grounds, hydrofracture sites, and liquid waste burial sites. White Oak Dam was built in the fall of 1943 to form White Oak Lake for the purpose of providing a dilution and settling basin for these wastes.

2.1 WHITE OAK LAKE

By 1954 concentrations of radionuclides (the contaminants of primary concern at the time) were in equilibrium with White Oak Creek resulting in a loss of the lakes ability to dilute and hold up radioactive materials (Oakes et al., 1982a). Thus the lake was drained in 1955. Drainage was done slowly to prevent disturbance of the contaminated lake sediments. Fish in the lake were poisoned and removed before drainage. Drainage exposed a 14 ha area of lake bottom sediment. In 1961 a study revealed that about 4250 m^3 of silt had eroded from the lake bed since its drainage in 1955. In 1962 the lake was again allowed to fill for waste holdup. Presently it can be described as a shallow eutrophic reservoir with an area of about 10 ha. It serves as the final settling basin for waste released from ORNL operations and waste storage areas.

Detailed histories of White Oak Dam and lake can be found in reports by Struxness et al (1967) and Oakes et al (1982a). Lee and Auerbach (1960) provide an earlier account of the history of White Oak Lake.

2.2 SOURCES OF CONTAMINANTS RELEASED TO THE WATERSHED

Sources of contaminants released to the watershed have changed through the years concomitant with changes in laboratory activities, waste treatment techniques, and locations of waste disposal areas. Names and locations of facilities and waste disposal areas within the White Oak Creek Watershed are shown in fig. 1.

2.2.1 LIQUID WASTES

Information about liquid waste disposal facilities is summarized in Table 2. All of the pits and trenches used for liquid waste disposal have been capped with asphalt. Pits 1, 2, 3, and 4 were left open during use, then backfilled with either Conasauga shale or soil and capped with asphalt. However, pit #1 was not backfilled and capped until 30 years after its closure. Pit #2 was closed and backfilled in 1962 and capped in 1970. Pits 3 and 4 were capped 2 years and 4 years after their respective closures. Waste trenches 5, 6, and 7 were covered with a mound of earth as they were used. Trenches 5 and 7 were both closed in 1966 and capped in 1970, while trench #6, which was closed in 1961, was not capped with asphalt until 1981.

Table 2 . History of liquid waste treatment and storage within the White Oak Creek Watershed.

Site	Notes	Operating dates
Tank storage	Caustic treatment to precipitate particulates, decant to settling basin (pond 3513), released to White Oak Lake for further settling.	1943-1949
Tank storage	Evaporation of highly radioactive liquid waste, store in concrete tanks.	1949-1954
Waste pit #1	Size - 30 x 6 x 5 m depth. Short use because of leakage. Received discharges from decontamination building 7819 from 1962 to 1964.	7/51-10/51
Waste pit #2	Size - 61 x 30 x 5 m depth. Seepage of ^{106}Ru from 1959 - 1961.	1952-1962
Waste pit #3	Size - 61 x 30 x 5 m depth. Improved liquid monitoring devices and a sampling platform extending over it. Some Seepage.	1955-1961
First waste water treatment plant	A soda-lime and clay flocculation process. Reduced radioactivity in process waste water before discharging to White Oak Creek.	1957-1976
Waste pit #4	Size - 61 x 30 x 5 m depth. Severe seepage problems. Interceptor trenches were dug and seepage was pumped back to pit 4.	1956-1976
Fission Products Development Laboratory (Bldg. 3517)	Used to solidify and confine ^{90}Sr, ^{137}Cs, and other valuable isotopes thus preventing their entry into the environment.	1957-present

Table 2. (continued)

Site	Notes	Operating dates
Waste trench #5	Size - 93 x 1 x 5 m depth. Covered with earth and treated with copper sulfate and copper sulfide to reduce losses of ^{106}Ru.	1960-1966
Waste trench #6	Size - 154 x 1 x 5 m depth. Used only a short time because of severe seepage problems. Construction and treatment was similar to trench 5.	9/61-10/61
Waste trench #7	Two 30 m segments - width 1 m, depth 5 m. Seepage problems involving primarily ^{60}Co. ^{106}Ru leakage was minor.	1962-1966
First hydrofracture facility (OHF)	Low level liquid waste was concentrated in a newly constructed evaporator and mixed with cement grout and injected deep into the earth.	1964-1979
Shale fracture facility (NHF)	Used for disposal of remaining liquid waste concentrate and sludge that had been previously stored in tanks. Operations were suspended in 1985 to determine the probability of contaminating ground water.	1982-1984
Impoundment for first hydrofracture facility	30m x 6m x 1.5m depth. Served as an emergency containment for spills from grout injections. Received drilling fluid and cuttings from core borings into the radioactive grout in 1984-85.	1963-present

Table 2. (continued)

Site	Notes	Operating dates
Process waste settling basin (Pond 3513).	67m x 67m. Volume = 7,118,000 L. Received supernatant from gunite tanks 1944-49. Also received cooling water from graphite reactor, wastes from laboratory drains, and effluents from the waste treatment plant.	1944-1976
HFIR Ponds. Four waste ponds.	Volume (all Ponds)=3,179,400 L. Receive waste from the High Flux Isotope Reactor (HFIR), the Thorium-Uranium Fuel Cycle Facility and the Transuranium Processing Plant.	1965-present

In addition to the liquid waste sources included in Table 2, some waste was released directly into white Oak Creek from operating facilities such as research reactors, chemical processing plants, and research laboratories.

Liquid waste disposal at the two hydrofracture facilities, the first one operating from 1962 to 1979 and the second since 1982, has been suspended while studies for possible ground water contamination are being conducted.

Leaks found on low-level waste transfer lines are a continuing area of concern. For example, major leaks on transfer lines in the main laboratory area between building 3019 and storage tanks and also between buildings 3029 and 3028 were discovered during the winter of 1985. Remedial actions relative to these leaks are currently under study by outside contractors (D. Huff, ORNL, personal communication).

Details of the development of and problems associated with liquid waste disposal in pits and trenches were reviewed by Spalding and Boegly (1985), Gilbert/Commonwealth (1980)and Evaluation Research Corporation (1982).

2.2.2 SOLID WASTES

Information about solid waste storage areas (SWSAs) is summarized in Table 3. SWSAs are operated much like traditional municipal landfills except that extra care has been taken in areas used for disposal of high activity wastes (i.e.)200 mR on container contact), especially in SWSA #5 and #6. Experience

Table 3. History of solid waste burial in the White Oak Creek Watershed.

Site	Notes	Operating dates
SWSA 1	Size - 0.6 ha. No records are available on amounts of waste. Consisted of a few small trenches and auger holes. Probably only small amounts of radioactive waste were buried because the operations did not include isotope separation at this time.	1944
SWSA 2	Size - 1.2 ha. No records are available on amounts of waste buried. Iron drums containing radioactive solid waste and stainless steel drums containing plutonium were transferred to SWSA 3 after closure of SWSA 2.	1944-1946
SWSA 3	Size - 2.8 ha. Used primarily for disposal of contaminated trash and laboratory equipment. Little information is available on quantities buried. Some burial of radioactive material in concrete lined trenches and some in unlined trenches.	1946-1951
SWSA 4	Size - 9.3 ha. On Conasauga shale. Most records on amounts and types of waste were destroyed in a fire. Most of the area is water saturated because it receives runoff from upslope. Largest contributor of ^{90}Sr to White Oak Creek. It also contributes relatively large amounts of tritium.	1951-1959
SWSA 5	Size - 13.4 ha. TRU waste stored in northern section and low-level solid waste in the southern section. Seepage from this burial site contains high concentrations of tritium. ^{90}Sr is also a principal contaminant.	1959-1973, but still used for TRU waste.

Table 3.
(cont.)

Site	Notes	Operating dates
SWSA 6	27.5 ha. Only about 6 ha are useful for waste disposal, however. Coverage of trenches with bentonite-shale mixture has not been successful for hydrologic isolation of the wastes. Used for burial of radioactive wastes including ^{235}U. Special trenches are being used for biological waste and for baled compacted waste.	1969-present

gained in the operation of the earlier SWSAs dictated that the bottom of trenches be at least 0.6 m above the water table and this has been the practice in SWSA #5 and #6. However, some trenches in SWSA #5 still flooded. Measures have been taken to divert water around the burial trenches to reduce the amount of water infiltration into the trenches. Also, trenches containing alpha contamination have been covered with concrete prior to backfilling with soil, and some high activity waste is disposed of in auger holes and capped with concrete.

At present all of the solid waste disposal areas are closed except for SWSA 6 and a portion of SWSA 5. SWSA 5 is used for TRU wastes only and this is stored as retrievable waste above ground. SWSA #1 is presently fenced and grass covered. SWSA #2 is not fenced, but all waste material was removed from it and buried in SWSA #3 before SWSA #2 was returned to its original contour and sown to grass. Although SWSA #3 was closed in 1951 it continued to be used for surface storage of contaminated equipment until 1978-79. At that time the contaminated material was buried in SWSA #5 and #6, trees growing in the area were removed, 20 cm of soil was added to the surface and grass was sown. All of the SWSAs now have a grass cover, except for a few individual trenches that are covered with asphalt or concrete and some trenches in SWSA #6 that are still in use (Tom Grizzard, ORNL, personal communication). More information on solid waste disposal can be found in Binford and Gissel (1975),Eva. Res. Corp. (1982), Bates (1983), and Oakes et al. (1982b).

2.2.3 GASEOUS RELEASES TO THE ATMOSPHERE

The history of various sources of gaseous releases to the atmosphere from ORNL facilities is provided by Manneschmidt and Witkowski, 1962 and by Ohnesorge, et al., 1986. A brief summary of these sources is presented here. The two major sources of radionuclides to the air prior to 1950 were the RaLa (radioactive lanthanum) facility (operating from 1945 to 1956) and the Graphite Reactor (1943 - 1963). A central off-gas and ventilation system with an improved scrubber system (stack 3039) was built in 1950 that served both of these facilities. Filters were added to stack 3039 in 1961 for scrubbing radioiodine and an improved stack gas sampling system was installed in 1962 and again in 1970. There have also been a few accidental and unmonitored discharges of radionuclides to the atmosphere. Details are presented by Ohnesorge, et al., 1986.

All current sources of airborne releases that might impact the White Oak Creek watershed including steam plant emissions, process discharges, and cooling tower drift are presented by Oakes, et al., 1984.

2.3 SITE CHARACTERISTICS

2.3.1 GEOLOGY AND HYDROLOGY

Bethel Valley, the area in which the first three burial grounds (SWSAs 1, 2, and 3) are located is underlain with limestone of the Chickamauga group. Because of solution channels

in this formation, later burial grounds were located in Melton Valley since it is underlain by Conasauga Shale, a relatively impermeable formation (Duguid, 1975). Fractures in Conasauga Shale, however, may allow groundwater to flow through the formation. This was thought to be the cause of losses of ^{90}Sr and ^{137}Cs from trench 6 before its closure in 1961. SWSA 4, the burial ground with a history of radionuclide leakage problems, is underlain by Conasauga Shale and is interbedded with thin limestone units. The shale is weathered to a depth of 4.6 m in the upper portion of the burial ground and to about 1.5 m near the streams (Duguid, 1975).

The water table in the White Oak Creek Watershed is a subdued replica of the land surface (Stockdale, 1951). In SWSA 4 the water table tends to be close to the surface in low elevation areas increasing in depth to about 4.3 m at the highest ground surface elevations. The disposal of uncontaminated fill on the surface of SWSA 4 prior to 1975 caused an increase in water infiltration and a rise in the water table at this site. One of the major problems causing radionuclide losses from trenches in SWSAs 4 and 5 is that water drains through the trenches from upslope and pools in the lower end of the trench until it overflows to surface streams (the "bathtub effect"). Past corrective measures have been directed mainly at elimination of this problem.

Groundwater flow and surface flow is determined primarily by the surface contour, and is generally toward the streams which

drain the area. The two major streams in the Watershed are White Oak Creek and Melton Branch. White Oak Creek receives drainage from the main plant areas and from SWSAs 1,2,3,4,and from the west side of SWSA 5 (fig. 1). Melton Branch receives drainage from most of SWSA 5 and from the HFIR facilities. White Oak Creek, after its confluence with Melton Branch receives drainage from the Pit and Trench areas before forming White Oak Lake. The Lake is bounded on its north side by SWSA 6.

From 1943 to 1948 White Oak Lake was maintained at an elevation of about 228.6 m and covered an area of 17.7 ha. From 1948 to 1955 the Lake covered an area ranging from 12.9 to 16 ha. It was drained in 1955. In 1962 the lake was again filled for waste holdup. It covered an average area of about 10 ha until 1979 when it was gradually lowered to reduce hydrostatic pressure on the dam structure. After lowering was completed the lake covered a land area of about 8 ha.

Between 1980 and 1984 modifications to the dam have added to its structural integrity, reduced likelihood of internal erosion of the dam, and decreased the likelihood of solids being discharged into the Clinch River.

Details of the geology and hydrology of the White Oak Creek Watershed can be found in Stockdale (1951).

2.3.2 TERRESTRIAL FLORA

About 480 ha of vegetation within the White Oak Creek Watershed is in close enough proximity to ORNL to be potentially

impacted by the laboratory's operations and associated waste disposal operations. Vegetation cover type and land usage areas in a 661 ha area to the south and west of the fenced X-10 site are provided in Table 4. This area is bounded by Bethel Valley Road to the north, Melton Valley and the Health Physics Research Reactor access Roads to the east, the North Reactor Evacuation Road to the South, and Highway 95 to the west. Vegetation most likely to be impacted occupies low elevation areas on floodplains along streams. Vegetation types found in these areas are primarily Sweet Gum-Sycamore-Southern Yellow Pine and Sweet Gum-Southern Yellow Pine-Ash. However, the most recent vegetation map of the area (Forestry Compartment Map Alt. No. 2, obtainable from D.M. Bradburn, ORNL) does not identify vegetation types along the banks of White Oak Lake and lower portions of White Oak Creek. During the period between 1956 and 1960 when White Oak Lake bed was exposed due to drainage, the lake bed re-vegetated to a woody thicket of willow, sycamore, box elder, red maple, dogwood, ash, poplar, and other tree shrub species (Deselm and Shanks, 1967; Auerbach et al., 1961). When the lake was allowed to refill in 1960 much of this woody scrub area was covered by water. Similar species still occupy the banks of the lake. Sedges and cattails are the principal species that occupy marshy areas. The floodplain south of SWSA 4 (contaminated in 1944 when it was used as a temporary impoundment for liquid radioactive effluents from ORNL) had been vegetated by the early 1970's by herbaceous species and trees; primarily sycamore, white ash,

Table 4. Vegetation cover and land use in the vicinity of White Oak Creek, Melton Branch, and the waste disposal facilities. The fenced X-10 main plant area is not included. Data were synthesized from Forestry Compartment Map Alt. No. 2 (obtainable from D.M. Bradburn, ORNL).

Vegetation Cover Type and Land Usage	Area (ha)
Short leaf pine plantation	16.2
Southern yellow pine	95.5
Southern yellow pine-hardwood	57.5
Southern yellow pine-cedar-hardwood	4.0
Southern yellow pine-red maple regeneration	4.0
Cedar	9.3
Cedar-red oak-southern yellow pine	0.8
Cedar-sweet gum	7.3
White pine-virginia pine	0.8
White pine-white Oak	1.2
White pine-white oak-southern yellow pine	3.2
White Oak-white pine	17.0
White oak-southern yellow pine-hickory	4.9
White oak-mixed hardwoods	132.0
Chestnut oak-hickory-red oak	26.0
Red oak-white oak-hickory	59.0
Red oak-sycamore-elm	0.8
Tulip poplar-mixed hardwood	2.8
Sweet gum-southern yellow pine-ash	15.0
Sweet gum-sycamore-southern yellow pine	8.9
Beetle killed pine	3.2
Regenerating after tree harvest	8.5
Non-forested (gravel pits, etc.)	14.2
Lakes and streams	13.4
Roads and rights of way	79.3
Burial grounds and surrounding scrub	58.2
Buildings, facilities, construction	17.8
TOTAL	661.0

and mixed hardwoods (Van Voris and Dahlman, 1976). Low ridges to the south of SWSA 4 are vegetated by white oak-mixed hardwood and southern yellow pine forest types. To the south and west of SWSA 5 the predominant vegetation is sweet gum, sycamore, white ash, southern yellow pine, and early successional hardwoods that are re-vegetating beetle killed pine stands.

Thirteen official Tennessee listed endangered and threatened plant species are found on the Oak Ridge Reservation (Parr, 1984a), but none of these are found in the White Oak Creek Watershed (Parr, 1984b).

2.3.3 TERRESTRIAL FAUNA

Plant communities in the Oak Ridge Reservation provide a variety of habitats for a large number of animal species. More than 120 species of terrestrial birds, 32 species of waterfowl, wading birds, and shorebirds, 60 species of reptiles and amphibians and about 40 species of mammals have been recorded (Kitchings and Mann 1976). The habitat provided in White Oak Creek Watershed is typical of that found on much of the rest of the Reservation and is probably almost as rich in species diversity. Rather than list the species here, those species for which contamination data is available are included in the Contaminant Inventory section (Section 4).

Seven federally listed endangered or threatened animal species (the gray bat, the Indiana bat, the eastern cougar, the bald eagle, the peregrine falcon, the Bachman's warbler, and the

red-cockaded woodpecker) could find suitable habitat on and adjacent to the Reservation, but none are established (Boyle et al., 1982). The eastern cougar has been cited in the vicinity of the Watershed on numerous occasions, but the U.S. Fish and Wildlife Service has failed to find an established cougar population (Boyle et. al., 1982). Two Tennessee listed endangered species (the osprey and Bachman's sparrow) have been increasing in number on and adjacent to the Reservation. However their presence in the White Oak Creek Watershed is unlikely (R.L. Kroodsma, ORNL, personal communication).

2.3.4 AQUATIC BIOTA

A relatively complete inventory of aquatic biota found in White Oak Creek, Melton Branch, and White Oak Lake is provided by Loar, Solomon, and Cada (1982). A very brief summary of their findings are presented in Tables 5 and 6. An earlier (1952-53) description of the benthic macroinvertebrates in White Oak Creek was published by Krumholz (1954b). Previous studies of the biota in White Oak Lake include those by Krumholz (1954b and 1954c); Lackey (1957); Neal, Patten, and DePoe (1967); Kolehmainen and Nelson (1969); and Auerbach et al., (1974).

3. CONTAMINANT INVENTORY OF THE WATERSHED

3.1 RADIONUCLIDE CONTAMINATION

Because of the concern for release of radionuclides into the

Table 5. The aquatic biota in White Oak Creek and Melton Branch in 1979 - 80 (summarized from Loar et al., 1982).

Dominant Taxonomic Groups	Notes
Periphyton	
Diatoms	91 to 94% of all periphyton
Green Algae	6 to 8.7% " "
Benthic Macroinvertebrates	
Chironomids	94 to 98% of all species
Dipterans	1.5 to 1.9% " " (mostly Empipdae). Maximum diversity of benthic organisms was found in a control area upstream from ORNL.
Ichthyoplankton (fish eggs and larvae)	none found
Fishes	
Compostoma anomalum (Stone roller)	Bluegill and Mosquitofish were found only in White Oak Creek near White Oak Dam. The other 3 species were found only at a control station upstream from ORNL. Very low densities were found at both stations.
Cottus carolinae (Banded sculpin)	
Rhinichthys atratulus (Blacknose dace)	
Gambusia affinis (Mosquitofish)	
Lepomis macrochirus (Bluegill)	

Table 6. The biota of White Oak Lake in 1979 - 80 (summarized from Loar et al., 1982)

Dominant Taxonomic Groups	Notes
Phytoplankton	
Green algae	93% of all phytoplankton found. 38 of 68 phytoplankton genera found.
Diatoms	14 genera - very low abundance.
Zooplankton	
Rotifers	>80% of all zooplankton found. 44 of 70 species.
Ichthyoplankton	
Clupeidae	Probably gizzard shad.
Lepomis	Bluegill, redear sunfish.
Periphyton	
Diatoms	63% of all periphyton found.
Green algae	32% " " " "
Bluegreen algae	3% " " " "
Benthic Macroinvertebrates	
Arthropoda (Insecta)	79% of all species found.
Molluska	14% " " " "
Annelida	7% " " " "
Fishes	
Bluegill	78% of all fish collected. Other less abundant species included redear-bluegill hybrids, largemouth bass, and mosquitofish.

environment adjacent to the Oak Ridge Reservation, a systematic monitoring of all radionuclides released to the environment from ORNL facilities has been conducted only at White Oak Dam, the point where water from the White Oak Creek Watershed enters the Clinch River. However, numerous other monitoring programs and environmental research projects have been conducted within the Watershed. Earlier research on the Watershed stressed the fate and effects of specific radionuclides in ecological systems. Information on radionuclide concentrations in the biota is available primarily from these kinds of studies. Such studies have been conducted intermittently over a relatively long time period and in areas known to be highly contaminated with the radionuclide(s) of interest. Historically, monitoring of groundwater, stream, soil, and sediment contamination within the Watershed has focused on ^{90}Sr, ^{137}Cs, and ^{60}Co with emphasis on a broader spectrum of contaminants only recently initiated.

Collectively, these projects and monitoring programs identify areas within the Watershed where the radionuclides of major concern are found, what their concentrations are, and clues as to their origin (i.e., waste burial areas, settling ponds, etc.). An abbreviated synthesis of the data from these studies follows.

3.1.1. GROUNDWATER

A number of monitoring wells have been installed in White Oak Creek Watershed at various times since 1950. Most of the

wells were installed in or adjacent to the waste burial areas. Beginning in 1985 several new wells have been installed near settling basins associated with process waste, hydrofracture facilities, and the HFIR ponds. The older wells have been used primarily for studies of groundwater levels and movement, with occasional analysis of samples for radionuclides. The newer wells, such as those around the process waste settling basin are being used for determining concentrations of non-radioactive contaminants as well as radionuclides. A summary of well locations and depths is provided in Table 7.

The average concentrations of radionuclides of major concern in groundwater near sources equipped with wells are summarized in Table 8. Tritium (^{3}H) concentrations exceeded those of all other radionuclides in each of the areas sampled, ranging from an average of 13 pCi/mL near SWSA 6 to 47000 pCi/mL near SWSA 5. SWSA 5 also contributed the highest concentrations of ^{90}Sr (55 pCi/mL) to groundwater. Groundwater downslope from Trench 7, a burial pit for intermediate level liquid waste in the Pits and Trenches waste disposal area, contributed relatively high concentrations of ^{60}Co, ^{3}H, and ^{99}Tc (Olsen et al.,1986).

3.1.2 WATER IN WHITE OAK LAKE AND ITS TRIBUTARIES

Releases of radionuclides to the Clinch River from White Oak Lake have been monitored at White Oak Dam since 1949. Prior to 1949 daily discharges of gross beta activity were estimated from direct radiation readings. New weirs installed at White Oak Dam

Table 7. History of well construction in the various waste disposal areas and settling basins.

Location	Year Installed	Number of Wells	Depth Range (m)	References
SWSA 3	1950	19	2 to 77	Webster et. al., 1981a
SWSA 4	1974	31	1 to 3	" " 1981b
SWSA 5	1958-59	28	1 to 47	" " 1982
	1964	5	4 to 6	" " "
	1975-76	62	1 to 12	" " "
	1977	2	4	" " "
SWSA 6	1954	30	1 to 3	" " 1980
	1956	4	37 to 38	" " "
	1964	44	2 to 8	" " "
	1976	4	6 to 14	" " "
	1977	19	6 to 24	" " "
	1978	3	3 to 4	" " "
Pond 3513	1985	5	4 to 8	Stansfield & Francis, 1986
Pond 3524				
3539				
3540	1985	15	5 to 18	R.G. Stansfield, p.c.
Pits area	1955	55		Spalding and Boegly, 1985
Trench 5	1955-63	18		ORNL Map #S-10063-A-001-D
Trench 6	1955-63	8		" " " " " " "
Trench 7	1955-63	16		" " " " " " "
" 7	1963-75	14		Duguid, 1975
" 7	1975-85	8		ORNL Map #C3E20965

Table 8. Average concentrations of radionuclides in groundwater collected from wells near contaminated sites in White Oak Creek Watershed.

Radio-nuclide	Trench[a] 7	Pond[b] 3513	OHF[c]	SWSA[d] 4	SWSA[d] 5	SWSA[d] 6	Pits & Trenches[d]
	pCi/mL						
^{60}Co	23.0		0.08	0.02	<0.01	<0.01	4.1
^{137}Cs		0.01	0.05	0.32	<0.06	<0.10	0.16
^{3}H	202.0	73.0		280.0	47000.0	13.0	100.0
^{90}Sr	5.5	0.3	6.03	11.0	55.0	1.4	0.29
^{226}Ra		0.001					
^{233}U	0.17						
^{99}Tc	21.0						8.05[e]
^{238}Pu	0.0003						
^{244}Cm	0.004						
^{241}Am	0.002						

[a] Samples collected in 1980-82 (Olsen, et al., 1986).

[b] Samples collected in 1985 (Stansfield and Francis, 1985b).

[c] Samples collected in 1985 (Stansfield and Francis, 1985b).

[d] Samples collected in 1984 (Env. Monitoring Report, 1984).

[e] Samples collected from wells near pit # 5 in 1984. (Garten et al., 1986).

in 1984 provide accurate measurements of flow rates thus reducing uncertainties in the data that have existed from 1949 to 1984 (especially during periods of high flow rates).

Annual amounts of 10 radionuclides, the rare earths, and the transuranium nuclides in White Oak Lake water that were discharged into the Clinch River during 1949 through 1984 are given in Table 9. This data reflects an historic and current estimate of radionuclide contamination levels in the Lake water. In 1984 the radionuclides discharged in greatest amounts were ^{3}H and ^{90}Sr. Amounts of ^{3}H declined between 1975 and 1981 but have since shown a slight increase. ^{137}Cs and ^{90}Sr showed a drastic decline in the early 1960's and are currently <1 and ~3 curies per year, respectively.

In addition to the sampling station at White Oak Dam, 4 other monitoring stations, located at strategic points along White Oak Creek and Melton Branch, have been in use for the past several years. The purpose of these stations is to help identify point sources of contaminants. In order to more accurately identify contaminant sources, 5 additional stations were constructed (Bates, 1983). Figure 1 shows the locations of the stations relative to contaminant sources. Records from these stations have been consistently kept for only ^{90}Sr because ^{90}Sr is probably the most mobile of the radionuclide wastes and has historically presented the most problems in terms of containment. Samples from these stations as well as grab samples from other

Table 9. Annual discharges and total discharges of radionuclides to the Clinch River, 1949 to 1984 (data from Oakes et al. 1982 and Env. Monitoring Report, 1985).

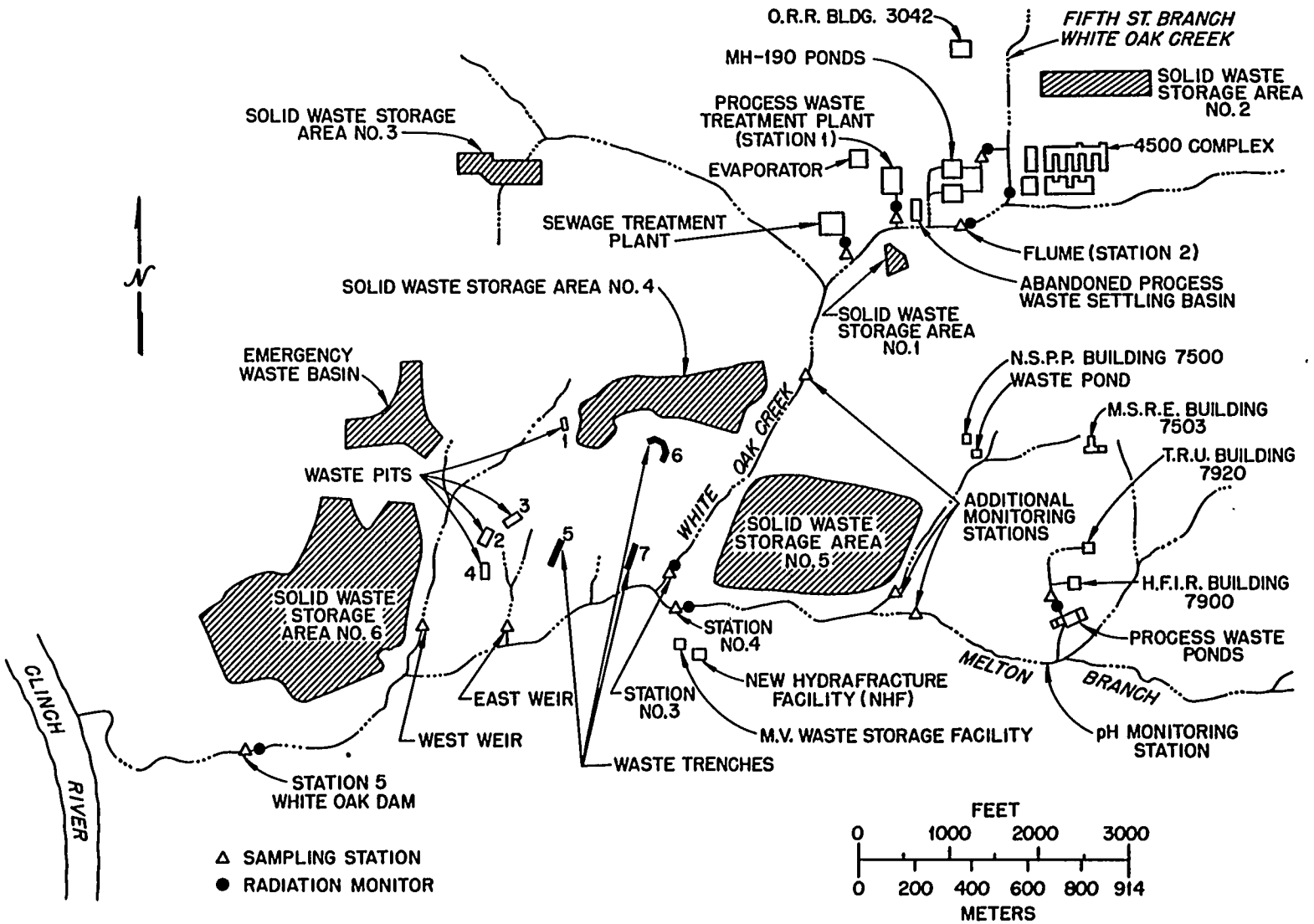
Table 9. Annual discharges and total discharges of radionuclides to the Clinch River, 1949 to 1984 (data from Oakes et al., 1982; Environmental Monitoring Report, 1985; and Environment, Safety, and Health Report, 1986).

Year	^{137}Cs	^{106}Ru	^{89}Sr	^{90}Sr	TRE[a] (-Ce)	^{144}Ce	^{95}Zr	^{95}Nb	^{131}I	^{60}Co	^{3}H	TRU[b]
							Curies					
1949	77	110		150	77	18	180	22	77		NA[c]	0.009 (from 8/12/49)
1950	19	23		38	30	NA	15	42	19			0.04
1951	20	18		29	11	NA	5	2	18			0.08
1952	10	15		72	26	23	19	18	20			0.03
1953	6	26		130	110	7	8	4	2			0.08
1954	22	11		140	160	24	14	9	4	NA		0.07
1955	63	31		93	150	85	5	6	7	7		0.25
1956	170	29		100	140	59	12	15	4	46		0.28
1957	89	60		83	110	13	23	7	1	5		0.15
1958	55	42	NA	150	240	30	6	6	8	9		0.08
1959	76	520	0.3	60	94	48	27	30	1	77		0.68
1960	31	1900	1.9	28	48	27	38	45	5	72		0.19
1961	15	2000	2.0	22	24	4	20	70	4	31		0.07
1962	6	1400	1.7	9	11	1	2	8	0.4	14		0.06
1963	4	430	1.0	8	9	2	0.3	0.7	0.4	14		0.17
1964	6	191	0.8	7	13	0.3	0.2	0.1	0.3	15	1,930	0.08
1965	2	69	0.6	3	6	0.1	0.3	0.3	0.2	12	1,160	0.50
1966	2	29	0.9	3	5	0.1	0.7	0.7	0.2	7	3,090	0.16
1967	3	17	0.7	5	9	0.2	0.5	0.5	0.9	3	13,300	1.03
1968	1	5	0.6	3	4	0.03	0.3	0.3	0.3	1	9,690	0.04
1969	1	2	0.3	3.1	5	0.02	0.2	0.2	0.5	1	12,200	0.20
1970	2	1	0.3	3.9	5	0.06	0.02	0.02	0.3	1	9,470	0.40
1971	1	0.5	0.2	3.4	3	0.05	0.01	0.01	0.2	1	8,950	0.05
1972	2	0.5	NA	6.5	5	0.03	0.01	0.01	0.3	1	10,600	0.07
1973	2	0.7		6.7	NA	0.02	0.05	0.05	0.5	1	15,000	0.08
1974	1	0.2		6.0		0.02	0.02	0.02	0.2	0.6	8,630	0.02
1975	0.6	0.3		7.2		NA	NA	NA	0.3	0.5	11,100	0.02
1976	0.2	0.2		4.5					0.03	0.9	7,420	0.01
1977	0.2	0.2		2.7					0.03	0.4	6,250	0.03
1978	0.3	0.2		2.0					0.04	0.4	6,290	0.03
1979	0.2	0.1		2.4					0.04	0.4	7,700	0.03
1980	0.6	0.0		1.5					0.04	0.4	4,554	0.04
1981	0.2	0.1		1.5					0.04	0.4	2,876	0.04
1982	1.5			2.7					0.06	1.0	5,400	0.03
1983	1.2	0.2		2.1					<0.01	0.3	5,600	0.05
1984	0.6	0.3		2.6					0.06	0.2	6,400	0.03
1985	0.42	<0.010		3.0					Not Detected	0.62	3,700	0.008
TOTAL	692	6933	11.3	1195	1295	342	375	287	175	325	161,310	5.18

[a]Total rare earths minus cerium.
[b]Transuranium nuclides.
[c]No analysis performed.

Figure 1. Location plan for radioactive liquid and solid waste facilities, monitoring stations, and other ORNL facilities.

ORNL-DWG 81-21824



LOCATION PLAN FOR RADIOACTIVE LIQUID AND SOLID WASTE FACILITIES

key locations have also been periodically analyzed for non-radioactive chemical pollutants. The first reliable measurements did not begin at these stations, however, until 1962 (Duguid, 1975). Even after 1962, deficiencies existed at the water monitoring stations during periods of unusually high rainfall (N. Cutshall, ORNL, personal communication). Rainfall was intense enough 2 or 3 times a year to cause the weirs at stations 3 and 4 and station 5 at White Oak Dam to overflow, thus rendering them ineffective for collecting flow proportional samples (Bates, 1983). In 1984 new weirs were constructed at White Oak Dam, at station 3 in White Creek, and at station 4 on Melton Branch. These new weirs were designed to provide more accurate flow measurements over a wide range of flow rates.

Stream water routinely collected at permanent monitoring stations along White Oak Creek and Melton Branch and some of their tributaries has traditionally been analyzed for ^{90}Sr, one of the more mobile radionuclides, as an indicator of point sources of contamination (Table 10). Using this method an estimated 94% of all the ^{90}Sr entering the Lake between January and June 1985 could be traced to SWSA's 1, 3, 4, and 5, and the sewage treatment plant, with 40% attributed to SWSA's 1 and 3. In 1978-79 SWSA 4 contributed about 50% to the total ^{90}Sr input to White Oak Lake (Steuber et al., 1981) as compared to only 10% in 1985 (Table 10). However, a more recent study (Cerling, in prep.) indicates that currently the most significant source of

Table 10. Monthly average ^{90}Sr (mCi) discharges into White Oak Lake and percent contribution from major sources.

Monitoring Periods			Major Sources[b]	Source Contribution[b]
1/64-12/78[a]	9/78-1/79[a]	1/85-6/85[b]	1985	1985
		White Oak Creek (Station 3)		
425	137	105	SWSA 1 & 3	40%
			Sewage Treatment Plant	24%
			SWSA 4	10%
		Melton Branch (Station 4)		
82	37	32	SWSA 5	20%
507	174	139[c]		94%

[a] Synthesized from Steuber et al. 1981.

[b] Lasher and Swanks, 1985; Scott and Swanks, 1985. The values given are the percentages of the total input to White Oak Lake from all sources.

[c] An average of 2 mCi/month were discharged through the east and west weirs downslope from the liquid LLW disposal pits.

^{90}Sr is the main plant area, most likely the Process Waste Treatment Plant.

The average monthly discharge <u>into</u> the lake during the 1985 period was 139 mCi as compared to 187 mCi/month discharged <u>from</u> the Lake into the Clinch River. However, the amount discharged from the Lake exceeded measured inputs during only one of the 6 months. In January, 410 mCi were monitored at the dam while only 185 mCi were monitored entering the Lake. This was apparently not due to losses from the Lake in excess of inputs but the inability to obtain accurate measurements at all stream monitoring stations during seasonal flood water fluctuations. According to Lasher and Swanks (1985), most of the ^{90}Sr released in January came from the sanitary sewer system which became contaminated by a groundwater source near Building 3019. This contamination was apparently from a leak on a transfer line between Building 3019 and storage tanks (See Section 2.2.1).

The corrected average monthly amount of ^{90}Sr entering the Lake in the first half of 1985 was only about 190 mCi (corrected for the monitoring error in January) as compared to 507 mCi/month during 1964 through 1978. This may be due to decreases in amounts of ^{90}Sr available for release from sources and/or to improved methods of containment.

3.1.3 STREAMBED GRAVELS AND LAKE SEDIMENTS

Analysis of streambed gravels has been used successfully for identification of source contamination points for ^{60}Co, ^{90}Sr, and

^{137}Cs (Cerling and Spalding, 1981 and 1982; Cerling and Turner, 1982), and more recently for a wider range of contaminants including heavy metals and organic pollutants (Cerling, in prep.). The advantages of gravel analysis is due primarily to the stability of the gravels to downstream movement and the higher radionuclide concentrations in gravel (due to adsorption) than the associated stream water. From sampling points located at 35 m intervals along all contaminated tributaries in White Oak Creek Watershed, Cerling and Spalding (1981) found concentrations of ^{90}Sr ranging from 1 to 485 pCi/g of gravel; ^{60}Co ranging from <0.5 to 9910 pCi/g; and ^{137}Cs ranging from 0.5 to 1973 pCi/g. The strategy and intensity of their sampling scheme permitted a higher degree of resolution on source identification than had previously been possible. They attributed 50% of the ^{90}Sr releases to ORNL plant effluents (in close agreement with Stueber et al., 1978), 30% to SWSA 4, 10% to SWSA 5 and 10% to combined effluents from SWSA 3, the Molten Salt Reactor Facility, and Pit 1 in the intermediate-level liquid waste storage area. Most of the ^{60}Co was found in gravels of Melton Branch in an area that pinpointed the major source to be from the High Flux Isotope Reactor (HFIR) cooling water effluent. Almost all of the ^{137}Cs was found above monitoring station 2 which pointed to ORNL plant effluents as the major contributor.

The Lake bed sediments have been intermittently sampled for 6 radionuclides and the transuranics. Summarized results from these studies are presented in Table 11. Total activity due to

Table 11. Concentrations and amounts of radionuclides in White Oak Lake bed sediments in 1962, 1972 and 1979.

Radio-nuclide	Concentration Range (pCi/g wet) 1972	1979[a]	Total Activity (Ci) 1962[b]	1979[c]
^{106}Ru	4 - 547		1038	
^{137}Cs	67 - 4425	58 - 131	704	591
^{60}Co	4 - 1534	133 - 725	152	33
^{90}Sr		29 - 85	10	20
^{65}Zn	0 - 138			
^{125}Sb	3 - 280			
TRU[d]				<1

Sources: 1962 data from Lomenick and Gardiner, 1965; 1972 data from Blaylock et al., 1973; 1979 data from Oakes et al., 1982a;

[a] This is an average range for the 0 - 15 cm depth except for ^{90}Sr which is the average for the 0 - 2.5 cm and the 12.5 - 15 cm depths.

[b] Samples were collected to a depth of 61 cm. There was little activity below 61 cm (Lomenick and Gardiner, 1965)

[c] These data from Oakes et al. (1982a) were calculated from their own measurements for the top 15 cm and from Blaylock et al., (1973) for the 15 - 98 cm depth.

[d] The TRU elements measured were ^{238}Pu - 0.096 Ci; ^{239}Pu - 0.250; ^{244}Cm - 0.498; and ^{241}Am - 0.024; measurements for the top 15cm only.

^{137}Cs and ^{60}Co was 1.2 and 4.6 times greater in 1962 than in 1979, while ^{90}Sr amounts in the Lake sediments were twice as high in 1979 as in 1962. Oakes et al., (1982b) estimated the total activity in the Lake sediments to be 644 Ci made up of the following nuclides: ^{137}Cs, 591 Ci; ^{60}Co, 33 Ci; ^{90}Sr, 20 Ci. Less than 1 Ci of transuranic nuclides was found in the upper 15 cm of sediment: ^{238}Pu (0.096 ci); ^{239}Pu (0.25 Ci); ^{241}Am (0.024); and ^{244}Cm (0.498 Ci).

3.1.4 AQUATIC BIOTA

Most of the radionuclide research in the aquatic environment of the Watershed has addressed transfer of specific radionuclides through food chains, bioconcentration factors and elimination rates in key organisms of the system, and effects of radiation exposure on the organisms. Many of the studies, specifically designed to determine concentrations of radionuclides in aquatic organisms, were done in the Clinch River below White Oak Dam and are therefore not included here. However, concentration data for a few of the radionuclides in algae and fish were gleaned from published reports of research performed in White Oak Lake. These data are summarized in Table 12. Krumholz (1954d), using the ^{32}P concentration data for Spirogyra, calculated that a mat of filamentous algae (weighing more than a ton) in White Oak Lake contained 100 mCi of radioactivity and that the concentration factor for ^{32}P was about 850,000. However, only small amounts of ^{32}P were found in White Oak Lake fish. Conversely, ^{90}Sr was

Table 12. Radionuclides concentrations in algae and fish of White Oak Lake.

Radio-nuclide	Year	Biota	Radionuclide Concentration	References
^{32}P	1950-53	Algae[a]	32160 pCi/g wet wt.	Krumholz, 1954d
Rare earths	"	"	29090 "	"
^{90}Sr	"	"	15310 "	"
^{60}Co	1964	Algae[b]	>5 to 16000 pCi/gdw[g]	Neal et al. 1967
^{106}Ru	"	"	>5 to 20000 "	"
^{137}Cs	"	"	>5 to 9000 "	"
^{60}Co	1979	Algae[c]	21 to 32 pCi/mL H_2O	Oakes et al. 1982a
^{137}Cs	"	"	1.7 to 54 "	"
^{90}Sr	"	"	110 "	"
^{60}Co	1965	Fish[d]	6 to 83 pCi/g wet wt.	Kevern and Griffith, 1966
^{137}Cs	"	"	20 to 38 "	
^{90}Sr	"	"	46 to 240 "	"
^{60}Co	1969	Fish[e]	3 to 5 "	Nelson et al. 1970
^{137}Cs	"	"	24 to 49 "	"
^{137}Cs	1970	Fish[f]	27 to 63 "	Oakes et al. 1982a

[a] Based on analysis of a filamentous algae (Spirogyra spp.).

[b] Based on analysis of Periphyton. The lower number in the range represents algae near the surface and the higher number represents algae near the bottom.

[c] Algae floating on the surface.

[d] Five species. Sample size = 5 per species.

[e] Three species. Sample sizes = 73, 84, and 115 per species.

[f] Seven species. Sample size ranged from 6 to 186.

[g] gdw=grams dry weight.

found in relatively high concentrations in fish bones even though its concentration in algae was lower than ^{32}P concentrations. The vertical distribution of radionuclide concentrations in periphyton (lowest concentrations at the surface and the highest near the bottom) reported by Neal et al. (1967) supported earlier findings by Krumholz (1954d).

Tritium concentrations in the aquatic biota have not been routinely measured. However, Blaylock et al. (1973) placed bluegill fish from an uncontaminated lake into White Oak Lake for a period of 36 days and assayed them for ^{3}H. They reported that the fish did not concentrate 3H in either their body water or in the tissue-bound component.

Radionuclide concentration data for the Lake waterfowl is based on a single study of migratory birds in the early 1950's. At that time, 649 migratory waterfowl were captured from White Oak Lake area, banded and released (Krumholz, 1954d).

Based on analysis of tissues of birds returned by hunters, Oakes et al. (1982a) estimated a total body burden for all birds of about 5,000,000 pCi. Almost all of the activity was due to ^{32}P concentrated in the muscle, skin, and giblets. Data for other radionuclides in waterfowl are available from a study conducted at a radioactive leaching pond at the Idaho National Engineering Laboratory (Halford et al., 1978).

3.1.5 SOIL AND TERRESTRIAL BIOTA

Radionuclide contamination studies of soil and terrestrial vegetation have been conducted in areas of the Watershed known to be highly contaminated with radionuclides such as in the dry lakebed during the late 1950's after drainage of the lake and in floodplain areas that were exposed after drainage of diked settling basins along White Oak Creek (Table 13). Thus, the levels of contamination presented are not expected to be typical of those in the rest of the Watershed. For example, the high concentrations of ^{90}Sr, ^{137}Cs, and ^{60}Co found in soil and vegetation in 1975 is from samples collected in a small area (about 3m by 8m) near leakage from a transfer line to Trench 7 (Duguid, 1976). In this particular case concentrations of ^{90}Sr were much greater in vegetation than in the soil indicating that ^{90}Sr bioaccumulates in vegetation. ^{99}Tc was also found to bioaccumulate in vegetation (Table 13).

A number of studies on the biogeochemistry of uranium, thorium, plutonium and other transuranic elements have been conducted in a contaminated floodplain below SWSA 4. The concentration of $^{239,240}Pu$, in floodplain soil, which ranges from about 10 to 150 pCi/g dw over the 3-ha area of the floodplain, is highest in the 0-10 cm zone of the soil profile (Dahlman et al., 1980). Dahlman et al. (1980) estimated the total $^{239,240}Pu$ inventory in the 3- ha floodplain to be 0.5 Ci in the top 20 cm

Table 13. Concentrations of radionuclides in terrestrial vegetation and soil in the White Oak Creek Watershed.

Radio-nuclide	Year	Soil	Herbaceous veg. Stems	Herbaceous veg. Leaves	Trees Stems	Trees Leaves
		pCi/g dry wt				
^{137}Cs	1956-58	7300	157	613	-	32
	1962	23,100	-	500	-	-
	1964	14	-	0.3	-	-
	1974	6200	-	33	35	28
	1975	38,000	-	-	-	260
	1975	90,000	-	-	-	-
	1978	6800	-	86	-	-
	1983	15,300	-	-	-	-
^{106}Ru	1958	-	-	-	-	3747
	1962	104,900	-	9500	-	-
	1964	28	-	2	-	-
^{90}Sr	1956	1370	560	902	-	-
	1957	360	-	1200	-	-
	1959	400	-	-	-	-
	1962	300	-	120	-	-
	1975	200,000	-	-	-	600,000
	1975	134,000	-	-	-	-
	1982	-	-	-	-	45
	1983	175	-	-	-	-
^{60}Co	1958	-	-	-	-	121
	1962	7000	-	359	-	-
	1964	4570	-	265	-	-
	1975	660	-	-	-	310
	1982	-	-	359	-	1
^{99}Tc	1982	-	-	-	-	777
	1984	899	-	18,532	1,027	432
^{239}Pu	1975	67	0.16		0.003	0.003
	1978	61				0.032

Sources: Dunaway, et al., 1963; Auerbach and Crossley, 1958; Crossley and Howden, 1961; Auerbach et al., 1958, 1959 and 1960; Crossley, 1969; Duguid, 1975;Van Voris and Dahlman, 1976; Oakes et al., 1982a; Garten and Dahlman, 1978; Dahlman et al., 1980; Olsen et al., in press: Garten et al., in press. In some instances the data shown represent averages or weighted averages. Samples were collected from several locally contaminated areas on the watershed.

of soil. Concentrations of uranium and thorium in the floodplain soil averaged 3 x 10^{-6} and 4 x 10^{-6} pCi/g, respectively (Bondietti and Sweeton, 1977).

Soils collected from the floodplain in 1978 contained an average of 28, 4.3, 2.6 and 0.5 pCi of ^{241}Am, ^{234}U, ^{238}U, and ^{244}Cm, respectively, per gram (Garten 1980). Concentrations (pCi/g dry wt) of these radionuclides in leaves collected from trees that were growing in this soil ranged from 0.001 to 0.013 pCi ^{241}Am, 0.005 to 0.05 pCi ^{234}U, 0.002 to 0.03 pCi ^{238}U and 0.001 to 0.009 pCi ^{244}Cm.

Radionuclide concentrations in terrestrial animals are presented in Table 14. Except for deer and birds, these animals were collected from areas where their primary food sources were expected to contain high concentrations of radionuclides. The deer samples were from near White Oak Lake but not immediately adjacent to it. They did have access to the area. In addition to the ^{131}I found in deer thyroid (Table 14), ^{125}I and ^{129}I have also been found in varying amounts in thyroid glands of deer on the reservation, with highest concentrations found in deer close to ORNL (J.S. Eldridge, ORNL, personal communication). It has been speculated that the reactor stacks in the ORNL plant area are the sources of the ^{125}I and ^{129}I.

The historical ^{137}Cs and ^{90}Sr data for birds is for muscle and bone tissue, respectively, from small birds feeding in terrestrial areas of the Watershed.

Table 14. Radionuclide concentrations in terrestrial animals of the White Oak Creek Watershed.

Radio-nuclides	Year	Arthropods Herbaceous	Arthropods Predacious	Birds	Small Mammals	Deer
		pCi/g dry wt				
^{137}Cs	1957-58	87	97	-	-	-
	1958-59	92		3137	-	-
	1960	-	-	-	193	-
	1961	78	73	-	514	-
	1962	-	-	-	58	-
	1974	-	-	-	27	-
	1978	-	-	-	343	-
	1984	0.03 - 69	-	-	-	0.2-muscle
^{106}Ru	1960	-	-	-	715	-
	1961	-	-	-	2075	-
	1962	-	-	-	63	-
	1964	721	2059	-	-	-
^{90}Sr	1957-58	91	81	-	-	-
	1958-59	110		234	-	-
	1960	-	-	-	2146	-
	1961	117	55	-	264	-
	1962	-	-	-	624	-
	1984	-	-	-	-	0.3-muscle
	1985	-	-	-	-	36-306-bone
^{60}Co	1960	-	-	-	21	-
	1961	-	-	-	123	-
	1962	-	-	-	29	-
	1964	114	122	-	-	-
	1984	0.3 - 23	-	-	-	1.3-liver
^{131}I	1984	-	-	-	-	67-thyroid
^{40}K	1984	-	-	-	-	11-muscle
^{239}Pu	1976	181 -3798		-	4-27	-

Notes: In a few cases data were originally reported per gram fresh weight and were converted to dry wt. assuming muscle and internal organs to be 70% water; bone - 15% (marrow was not included in the analysis); arthropods - 65%.

Deer samples were from near, but not on, the Watershed.

Sources: Willard, 1960; Crossley and Howden, 1961; Auerbach, et al., 1959 & 1962; Kaye and Dunaway, 1962; Crossley, 1969; Van Voris and Dahlman, 1976; Garten and Dahlman, 1978; Garten, 1980; Oakes et al., 1982a; Env. Monitoring Report, 1985; J. Eldridge, ORNL, personal Comm.

3.2 NON-RADIOACTIVE CHEMICAL CONTAMINATION

The water and lake sediments in White Oak Creek Watershed contain metals and organic chemicals as a result of discharges from ORNL (Boyle et al. 1982). Historically, the concern at ORNL has been with radionuclides, and therefore data for non-radioactive chemical contamination of the Watershed are very sparse. Thus, an inventory of these chemicals for all components of the Watershed is not possible at this time. The Industrial Hygiene Department did begin a water surveillance program within the Watershed in 1968. They reported average concentrations of 16 non-radioactive pollutants, including one organic (phenols), in the settling basin, White Oak Creek and Lake, and Melton Branch for the years 1970 and 1971 (Bolton and Bonner, 1973). Similar data for 1979 and 1980 and including PCB data were published in the Environmental Monitoring Report (1984). The 1984 Environmental Monitoring Report also contains concentration values for about 50 chemicals, including 7 organics in groundwater near SWSAs 4, 5, and 6. This report and other Environmental Monitoring reports dating back to 1976 also present current and past concentration data for Cr, Zn, NO^3 (N), and Hg in water collected at White Oak Dam. The most recent study of heavy metals and organics in the Watershed water and sediments is that of Cerling (in prep.). Non-radioactive chemical data for the biota of the Watershed is very limited. Concentrations of Hg and other trace metals in White Oak Lake fish were published by Loar et al. (1981) and by TVA (1985).

4. REGULATORY STATUS AND HAZARD RANKING

The White Oak Creek and its tributaries and the White Oak Lake may be regulated under the Comprehensive Environment Response, Compensation, and Liability Act (CERCLA). The CERCLA order requires that DOE's modified Hazard Ranking System be used for radioactive sites. This system is a modification of the Environmental Protection Agency's (EPA) Hazard Ranking System. Both systems utilize a ranking scheme in which the highest numbers (on a scale of 0 to 100) indicate the greatest risks. The current "cut-off" point for EPA's National Priorities List for sites requiring cleanup is 28.5 (C.E. Nix, ORNL, personal communication).

The hazard ranking scores for White Oak Lake and for White Oak Creek and its tributaries are both 5.2 (Nix, et al., 1986). These scores were based on the presence of significant quantities of radionuclides, heavy metals, and PCBs in certain areas of the Watershed. Minimum values for quantity of heavy metals and PCBs were used. Air pathway scores and hazards from fire and explosions were both assumed to be zero. Maximum scores were used for release to groundwater unless there was evidence of specific containment features at specific sites. However, scores for potential targets from contaminated groundwater were low because there is no well water usage in the area. Surface waters were assumed to be contaminated but scores were based entirely on recreational usage because the nearest drinking water supply is

beyond 3 miles and there are no federally listed endangered species within the Watershed.

5. RECOMMENDATIONS

A number of remedial action activities designed to reduce the input of pollutants into White Oak Creek Watershed have been conducted over the past several years (e.g., diversion of water around problem waste storage areas such as SWSA 4, White Oak Creek floodplain, and covering pits with asphalt to reduce water infiltration, etc). The next steps needed before further major remedial activities are undertaken are the completion and careful analysis of current research and monitoring activities. Some of the recently initiated work is discussed below.

Water and sediment samples recently collected at strategic locations throughout the Watershed are currently being analyzed for total organic carbon, EPA priority pollutants including heavy metals and toxic organics, and the more mobile and/or relatively long-lived radionuclides (^{90}Sr, ^{137}Cs, ^{60}Co, and ^{99}Tc). Sediments were collected in 2 cm depth increments down to clay in White Oak Lake. Water samples were collected from seepage areas below SWSA 4, from wells in and adjacent to SWSAs 3, 4, 5, and 6, and from wells in the pits and trenches area. A report of the results from this study, defined as the Contaminant Scoping Survey, will be issued in June, 1986 (F.G. Taylor and T. Doyle, ORNL).

The wells sampled in the above study do not meet RCRA standards but this will present a problem only with a few of the

non-radioactive contaminants. New RCRA quality wells will be in place throughout the Watershed by about 1990 (personal communication, S.H. Stow, ORNL). New wells are currently being installed throughout the main X-10 plant area and plans call for strategic placement of some additional wells in the liquid and solid waste disposal areas. These new wells will be used primarily for determining the depth to groundwater and the direction and rate of flow. Water from the wells will be analyzed periodically for radionuclides. Results from this study and the Contaminant Scoping Survey will be evaluated to aid in the strategic placement of the more expensive RCRA quality wells.

Plans also call for the installation of RCRA quality wells in the vicinity of auger hole waste disposal locations known to contain organic solvents (E.C. Davis, ORNL, personal communication). Water from these wells will be analyzed for organic and heavy metal pollutants, and radionuclides. A comprehensive monitoring plan for identifying and quantifying point sources of pollutants in SWSA 4 is also currently being developed.

Fish will be collected annually from strategic locations within the White Oak Creek drainage basin and analyzed for a suite of metals, PCBs, PAHs, and screened for other organics (George Southworth, ORNL, Personal communication). The role of turtles in the dispersal of radionuclides from sediments of White Oak Creek to the surrounding terrestrial environment will be investigated (B.T. Walton, ORNL, personal communication). The

interactions of ^{3}H between soil, vegetation, and the atmosphere will be investigated in a high ^{3}H activity area below SWSA #5 (C.T. Garten, ORNL, personal communication).

Data from these studies will be inadequate to analyze the existing inventory of radionuclides and non-radioactive chemicals in soil, vegetation, and animals occupying the watershed. Data from the terrestrial system is especially insufficient for such purposes.

Data on contaminant uptake by waterfowl are also deficient. While studies have indicated that relatively small amounts are removed by migratory birds, doses to hunters eating such birds have not been well documented. Studies of the large population of non-migratory Canada Geese, some of which nest on White Oak Lake is recommended.

The relatively large data bases from the aquatic system and terrestrial flood plains are sufficient for estimating only concentrations (not total amounts because of the lack of biomass standing crop data) of some (not all) of the radionuclides released and relatively few of the non-radioactive chemicals. The feasibility of doing a complete inventory is questionable, even for only the contaminants of major concern. However, the bioaccumulation factors for most problematic radionuclides have been well documented. Therefore, a systematic soil sampling program for the entire watershed is recommended. This data alone would not permit an accurate inventory, but areas of high contamination that may have been overlooked in the past, would be

identified. This information is also necessary for the development of a comprehensive land-use plan for the watershed to identify areas in need of cleanup and contaminated areas to be preserved as Ecological Study Sites by virtue of their unique nature and opportunity for long term environmental research. An advisory board should be appointed to laboratory management for guidance in the selection of such Ecological Study Sites.

The White Oak Creek Watershed is a unique "field laboratory", which should continue to be used for gathering information that will increase our knowledge of safe waste disposal and handling techniques, the fate of pollutants in the environment, and their effects on organisms. However, a more systematic approach (i.e. "before" and "after" studies) is needed in order to adequately evaluate effectiveness of remedial action activities and to identify areas needing corrective actions. For example, a workable model of the Watershed, designed to integrate biota, hydrology, and erosion, is needed to help facilitate research planning and data analysis. Such a model, coupled with more research projects directed toward evaluation of specific planned remedial actions, would expedite both new knowledge gained and effectiveness of remedial actions.

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ORNL/RAP/LTR-86/18

DATE: June 16, 1986

SUBJECT: Inventory of ORNL Remedial Action Sites:
6. Environmental Research Areas

TO: T. E. Myrick

FROM: F. G. Taylor

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LETTER REPORT RAP86-18

Inventory of ORNL Remedial Action Sites: 6. Environmental Research Areas*

F. G. Taylor

June 16, 1986

Environmental Sciences Division
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37831

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SUMMARY

Environmental Research Areas identified as potential remedial action program sites have been visited and the activities concerning the area reviewed by personal interview with persons knowledgeable about the sites. Additional information was abstracted from reports and publications. Results of the site review are summarized. Most areas involved radioisotopes with relatively short half-lives, with numerous half-lives lapsed since radionuclide application. The exception is Areas 1 and 2, both of which contained cesium-137 and involved up to hectare size areas. One deficiency concerning site characterization is the absence of information or persons knowledgeable of Area 7 (cesium-137 tagged field). Another deficiency is knowledge of how the radioactive materials were disposed of at the termination of the study.

1. OBJECTIVES

The primary purpose of the remedial action task is to provide corrective measures and facility decommissioning necessary to place the Oak Ridge National Laboratory (ORNL) in compliance with applicable environmental and health regulations for radioactive and hazardous chemical wastes. Individual sites will be decontaminated in accordance with priorities prescribed to control the potential for human exposure and relative magnitude of ecological impacts. Remedial action is a component of Laboratory's Environmental Restoration and Facilities Upgrade Program (ERFU). In the program, remedial actions will be conducted at locations where past management practices or research activities have resulted in contamination of facilities or the environment that necessitate decommissioning or corrective actions. Remedial actions are divided into two activities: (1) Site Decommissioning Program (Surplus Contaminated Facility Program-SCFP and Surplus Facility Management Program-SFMP) and (2) Site Corrective Measures Program (SCMP).

The Site Decommissioning Program is designed for the management of surplus facilities (primarily structures), contaminated with radioactive materials and hazardous chemicals, from the end of their operating life until final facility disposition. The Site Corrective Measures Program is designed for the management of inactive sites (process ponds, spill sites, environmental research areas), contaminated with radioactive materials and hazardous wastes, leading to the eventual stabilization of these sites in accordance with environmental regulations.

Remedial Action Program Sites at ORNL comprise 13 distinct areas (Table 1.1) among the several programmatic categories: (1) Site

Table 1.1 Inventory of ORNL Remedial Action Sites

1. Solid waste storage areas
2. LLW seepage pits and trenches
3. Process ponds
4. White Oak Creek Watershed
5. LLW line leak sites
6. Environmental research areas
7. Hazardous waste sites
8. Radioisotope processing facilities
9. Experimental reactor facilities
10. Radwaste facilities
11. Research laboratories
12. Other contaminated sites
13. Abandoned injection wells

Corrective Measures Program-SCMP, (2) Surplus Contaminated Facility Program-SCFP, and (3) Surplus Facility Management Program-SFMP. In a series of reports, an inventory of each area is presented in a condensed summary. The summary is the product of pertinent information abstracted from readily available literature and knowledgeable individuals on known contaminated sites requiring remedial action. The information includes site location and description, history, contaminants and their characteristics, extent of contamination, regulatory status, geological, hydrological, and ecological data, list of major studies, and knowledgeable individuals contacted. The subject of this report is the several research areas utilized for environmental and ecological research.

2. BACKGROUND

The establishment of Environmental Research Areas began with activities in the Health Physics Division (HPD). Historically, that division was concerned with the protection of man from ionizing radiation. That concern was manifested in an interdisciplinary program in the environmental sciences in relation to the behavior of radionuclides released to the environment. In 1953 it was recognized that many aspects of the radiation issue were ecological in nature. Consequently, in 1954 a program was started in radiation ecology. In 1955 White Oak Lake (a radioactive waste impoundment) was drained. This resulted in the first relatively large terrestrial area (Environmental Research Area) contaminated by radioactive waste and provided the means and impetus for development of a field program directly related to nuclear technology. Initial research activities emphasized the processes

of movement and behavior of radionuclides and other chemical elements in natural ecosystems. Laboratory and field studies utilized radioactive tracers and waste radionuclides to determine (1) uptake, accumulation, and movement of isotopes in terrestrial and aquatic food chains; (2) rates of translocation in plants; (3) consumption of food and turnover of isotopes by terrestrial and aquatic insects and other invertebrates, fish, and mammals; (4) reentry of nutrients and isotopes into the soil through the breakdown of organic litter by fungi, bacteria, and soil animals.

As research interests expanded, the organization of ecological research evolved from the Radiation Ecology Section (HPD) to the Ecological Sciences Division in 1971 and the present Environmental Sciences Division in 1973. With the expansion of diverse interests and programs came the addition of numerous environmental research areas. Some of the areas may contain residual radioactivity following completion of specific studies, while others have decayed beyond radiological detection. Some area locations and histories are known only by employees nearing retirement. The purpose of this task is to obtain pertinent information from available literature and knowledgeable individuals on each site requiring remedial action. The compilation will provide information to determine what actions are necessary to bring each site into compliance with appropriate regulations while identifying data deficiencies.

2.1 SITE CHARACTERISTICS

There are several documents which describe in a generic sense the land and water resources on the Oak Ridge Reservation. The most complete document (Dahlman et al. 1977) provides some basic insight to the diverse geology, hydrology, and ecology of the area. Excerpts from that document are summarized, with appropriate references for more detailed descriptions.

2.2.1 Geology

The Oak Ridge Reservation is in the Tennessee section of the Ridge and Valley province and is underlain by consolidated marine sediments of Paleozoic age. The topography is characterized by subparallel northeast-trending ridges and valleys and reflects the geologic structure of the area which consists of generally southeast-dipping strata. The different lithologies of the formations result in different rates of weathering and erosion. The more resistant ridge-producing formations are the Rome formation, the Knox group, the Rockwood formation, and the Fort Payne chert. Most of the valleys are underlain by the Conasauga group and the Chickamauga limestone (McMaster 1963).

The western part of the Tennessee section of the Ridge and Valley province is structurally characterized by major subparallel thrust faults that trend northeast and dip southeast. Along most of these faults, the Rome formation has been thrust over younger formations, which causes repetition of the geologic sequence within the area. The two major faults on the reservation are the Copper Creek fault and the

White Oak Mountain fault. The latter exhibits several subsidiary features including branch faults, a syncline, and two slices of dolomite of the Knox group. There is no evidence of recent displacement along either of the faults.

The reservation is underlain by nine geologic formations ranging in age from Cambrian to Mississippian. The formations are of sedimentary origin: limestone, dolomite, sandstone, and shale. From oldest to youngest, the formations are the Rome formation, the Conasauga group, the Knox group, the Chickamauga limestone, the Reedsville shale and Maury formation, and the Fort Payne chert.

The Rome formation is a fine-grained shaly sandstone that is generally well cemented. Its ability to hold and transmit groundwater is due almost entirely to fractures. However, because of the composition of the formation, the fractures have not been enlarged by solution.

The Conasauga group is primarily silty, slightly calcareous shale interbedded with thin limestone units. The permeability seems to be associated with changes due to weathering of the rock above and somewhat below the water table. Where the shale is saturated, movement of groundwater is concentrated in a zone near the water table and perhaps locally deeper along some fractures. The fractures in the Conasauga group, and therefore the permeability, are evenly distributed; this results in a more uniform flow of groundwater than occurs in other formations.

The Knox group is composed largely of thick beds of dolomite and limestone, and, where it is unweathered and undeformed, it has low

porosity and low permeability. However, where it has been fractured by folding or faulting, its permeability has been greatly increased. In many places these fractures have been enlarged by solution, and large openings have been formed. Sinkholes and caverns are common. These openings give rise locally to large springs and wells of high capacity, although there are wells that are almost dry. The permeability and porosity of the formation are unevenly distributed and have been localized by fracturing and later solution by groundwater movement.

The Chickamauga limestone is composed largely of thin beds of shaly limestone and shale. The porosity of the formation is low; fractures have been enlarged by solution but not, however, to the extent of the caverns Knox group. These fractures and solution channels form a network of open channels and voids which permit the free movement of groundwater.

2.2.2 Hydrology

The Oak Ridge Reservation is in the lower portion of the Clinch River drainage basin which originates in southwestern Virginia and drains an area of approximately 11,430 km. The Clinch River bounds the reservation on the south and west for a length of approximately 39 miles (63 km), extending from Clinch River Mile (CRM) 49 on Melton Hill Reservoir to CRM 10.0 just downstream from the mouth of Poplar Creek. Surface waters on and contiguous with the reservation, including the Clinch River, are important resources for research, recreation, municipal and industrial activities in the Oak Ridge area. The primary surface waters on the reservation are small streams which generally

originate from contributing springs in the limestone on the ridge slopes. Average annual discharge rate of the small streams is less than 0.57 m^3 sec^{-1}.

The groundwater table of the reservation can be described as a subdued replica of the surface topography. Thus, the groundwater flows from areas of high elevation to areas of low elevation and ultimately discharges into surface streams. In general, this description may be extended to the entire reservation where groundwater occurs under water table conditions in a humid region.

Storm runoff in the vicinity of Oak Ridge has been described for a variety of catchment basins from 0.38 to 82 square miles (1.0 to 212 km^2) in area (Sheppard 1974). Streamflow (Q) of Walker Branch was best described by the relationship $Q = 1.9\ p^2$ where Q has units of cubic feet per second and P is precipitation in inches (1 to 9 inches per event). A slightly modified relationship ($Q = 4.7\ A^{0.8}p^2$, where A is square miles) holds for other catchments of the Oak Ridge areas. It was estimated that a storm involving 9 inches of precipitation in 48 hours (100-year recurrence interval) would generate 150 to 600 cfs (4200 to 1700 liters sec^{-1}) of discharge per square mile of catchment basin.

Water quality of surface streams on the reservation is, in general, of the calcium-magnesium-bicarbonate type, reflecting the limestone geology of the area. Water quality and ecology of several of the streams on the reservation, however, have been altered as a result of industrial and municipal effluents. The major drainage basins on and flowing through the Oak Ridge Reservation are White Oak Creek with

a drainage area of 16.9 km^2 and Poplar Creek with a drainage area of 352 km^2. Several smaller catchments on the reservation with small, perennial streams which are aquatic research sites and control areas for environmental monitoring programs include Walker Branch (drainage area = 1 km^2), McCoy Branch, Scarboro-Kerr Hollow (3 km^2), and Grassy Creek (3 km^2). In addition, there are two small quarries on the reservation (Roger's Quarry and Lambert Quarry) that serve as experimental sites for aquatic research.

2.2.3 Flora

The general ecology of the Oak Ridge Reservation was described in detail by Kitchings and Mann (1976), Dahlman et al. (1977), and Mann et al. (1985). Except for forest management practices, the nearly 15,000 ha has been relatively undisturbed since the 1940's. The site includes a wide variety of habitats ranging from open water to mesic forests. An inventory of vascular plant habitats (Mann and Bierner 1975) and plant species checklist identifies a total of 842 species, subspecies, and varieties as present on the reservation (Mann et al. 1985). Approximately one-third of the total area either has been planted in pine, primarily loblolly pine, or is in natural shortleaf and Virginia pine stands. In many areas, the natural pine stands are being replaced by upland hardwoods, dominated by oaks, hickories, blackgum, sourwood, and occasionally white pines. Some areas, such as pipelines and powerline corridors and a few field research sites, are maintained as open fields.

The more sheltered coves and bluffs support vegetation communities more typical of mesic forests of the nearby Cumberland Plateau (e.g., beech, hemlock, basswood, cucumber tree, tripetal magnolia, and sugar maple). Although most narrow, steep-sided, perennial stream drainages on the reservation have this mesophytic character, it is well developed and extensive on only a few steep north-facing slopes and bluffs. Mesic bluff sites occur only along the Clinch River and its major embayments and Poplar Creek. Many of the more uncommon species are restricted to shallow, dry soils on Chickamauga limestone.

Among the species known to exist on the reservation, several are noted as endangered, threatened, or of special concern in Tennessee (Parr 1984). The occurrence of any of these species at any environmental research site requiring remedial action is noted in the appropriate summary of the research site.

2.2.4 Fauna

The variety of wooded and open areas, as well as extensive edge communities, create habitat for numerous mammalian and avian species residing on the Oak Ridge Reservation. Both bird and mammal fauna population densities are generally correlated with vegetation cover and type. An important factor in determining whether a given species will occur in a given area is the nature of the habitat in the area. Small mammals, such as rodents, may be confined to a single habitat type, whereas the larger sized species may range over several habitats in order to fulfill their existence requirements. The same can be said for avian populations. Most of the birds and mammals found on the Oak

Ridge Reservation have the capability of tolerating and adapting to a variety of habitats and therefore may be found in habitats other than those which are typical for the respective species. Habitat preference for average conditions and some representative animal species of the habitats is discussed in detail by Dahlman et al. (1977) and Kitchings and Mann (1976).

3. DESCRIPTION OF ORNL REMEDIAL ACTION PROGRAM SITES: ENVIRONMENTAL RESEARCH AREAS (1954-1971)

Environmental Research Areas are scattered over the DOE reservation. A total of 14 candidate sites (Table 3.1) for possible remedial actions were initially identified (Cannon 1986) and located on a map (Figure 3.1). The extent of contamination at these sites is largely unknown, but it is believed not to be significant. A tabulation of Environmental Research Areas, with prominent landmarks, is presented in Table 3.1. The listing is derived from the compendium of research areas as presented by Grimsby (1986). The source of information leading to that list is unknown, but only sites between 1954 and 1971 were included. Table 3.2 identifies the areas by number and key identifiers (medium, isotope). Additional research areas were active from 1972 to the present and are discussed in a later section (Section 4).

3.1 AREA 1: CS-137 CONTAMINATED FIELD (0800 area)

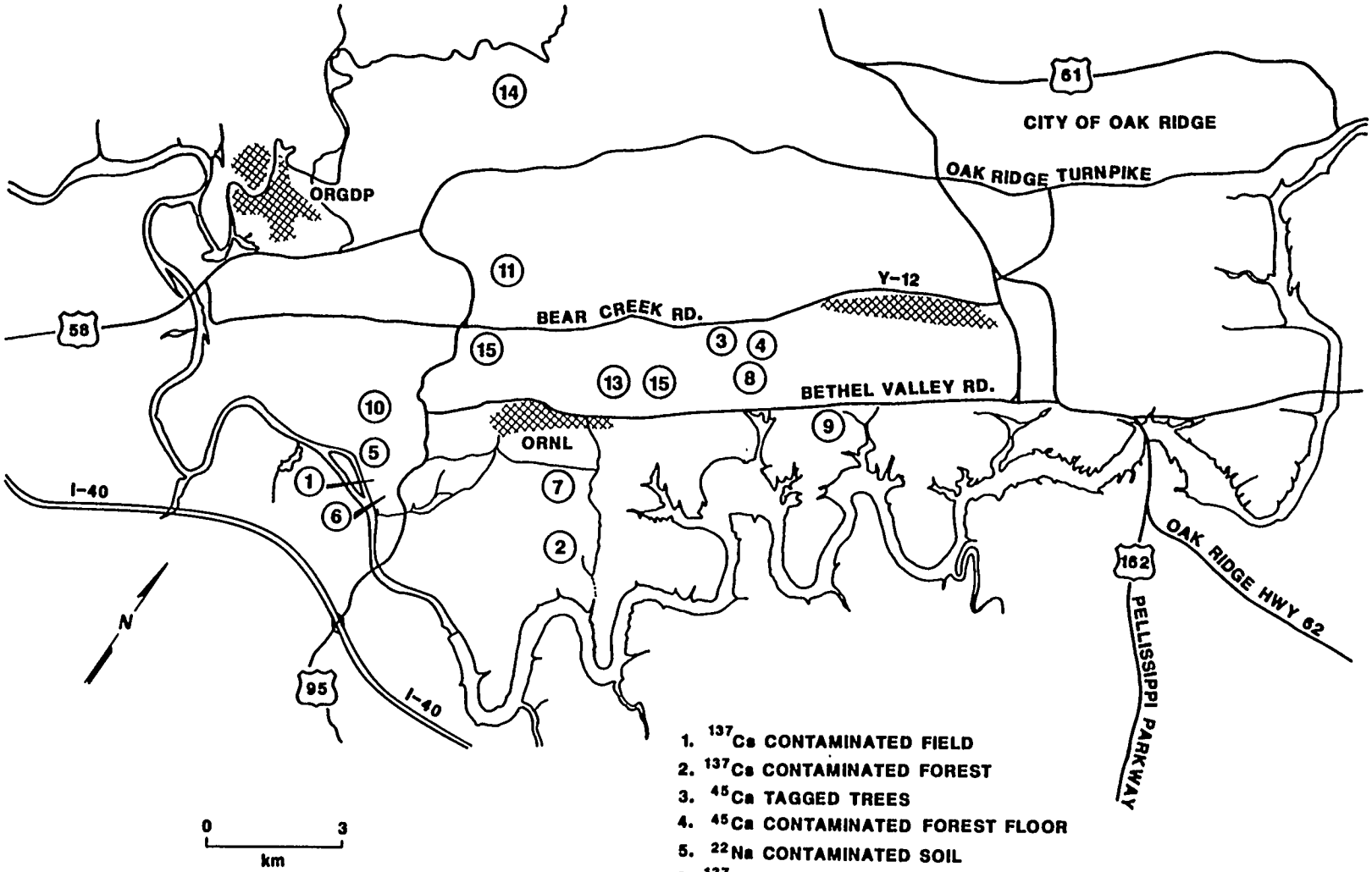
The 0800 area is a 50 acre (~20 ha) fescue-dominated field. The area is best known for simulated fallout studies utilizing cesium-137. In addition, a number of studies have been conducted outside the fenced

Table 3.1 ORNL Remedial Action Sites: Environmental research areas

Area 1:	Cs-137 contaminated field (0800 area)
Area 2:	Cs-137 contaminated forest, soil, and vegetation (near Health Physics Research Reactor)
Area 3:	Ca-45 tagged trees and dogwood (Walker Branch Watershed)
Area 4:	Ca-45 tagged soil and vegetation (Walker Branch Watershed)
Area 5:	Na-22 contaminated field (0800 area)
Area 6:	Cs-137 tagged area for radionuclide runoff studies (0800 area)
Area 7:	Cs-137 tagged field (Melton Valley near HFIR & TRU facilities)
Area 8:	$^{197}HgNO_3$ contamination (Walker Branch Watershed)
Area 9:	Cs-134 tagged field (Melton Valley near Mt. Vernon Road)
Area 10:	Ca-45 tagged forest (New Zion Patrol Road)
Area 11:	McNew Hollow contaminated area (north of Pine Ridge)
Area 12:	White Wing storage area removed as a research area and included with solid waste storage areas (Grizzard 1986)
Area 13:	$CH_3\ ^{203}HgCl$ contaminated area (headwater of White Oak Creek)
Area 14:	Tritium-tagged trees and soil (0804 area, Blackoak Ridge)
Area 15:	Cesium contaminated area (contamination removed, Chestnut Ridge)

Fig. 3.1 Location of major environmental research areas in the Remedial Action Program.

ORNL-DWG 86-12052



1. ^{137}Cs CONTAMINATED FIELD
2. ^{137}Cs CONTAMINATED FOREST
3. ^{45}Ca TAGGED TREES
4. ^{45}Ca CONTAMINATED FOREST FLOOR
5. ^{22}Na CONTAMINATED SOIL
6. ^{137}Cs CONTAMINATED SOIL
7. ^{137}Cs CONTAMINATED LITTER BAGS
8. $^{197}HgNO_3$ CONTAMINATED STREAM
9. ^{134}Cs CONTAMINATED TREE
10. ^{45}Ca TAGGED TREES
11. McNEW HOLLOW ^{137}Cs AND ^{59}Fe CONTAMINATED
13. $^{203}CH_3HgCl$ CONTAMINATED STREAM
14. TRITIUM TAGGED TREES
15. ^{137}Cs CONTAMINATED AREA

Table 3.2 Detail summary concerning radioisotope usage at ORNL Remedial Action Sites: Environmental research areas

Site number	Date of contamination	Principal radionuclide	Half-life	Form of contaminant	Quantity of radioactivity	Environmental matrix	Extent of contamination	Status
1	August, 1968	Cesium-137	30 years	Silica particles	8.8 Ci	Vegetation, soil	2 ha	Inactive
2	May 20-23, 1962	Cesium-137	30 years	Liquid	467 mCi	Vegetation, wood, soil	500 m^2	Inactive
3	June 7, 1969	Calcium-45	165 days	Liquid	1.25 Ci	Wood, foliage	Probably <0.5 ha	Inactive, radioactivity removed
4	Dec. 20, 1969	Calcium-45	165 days	Leaves	136 mCi	Leaves, soil, soil-water solution	15 m^2	Inactive, partially removed
5	1968 and 1969	Sodium-22	2.62 years	Grass leaves	Unknown	Insects, spiders	~500 m^2	Inactive
6	Oct. 20, 1964	Cesium-137	30 years	Liquid	15 mCi	Soil	~20 m^2	Inactive
7	Before 1962	Cesium-137 Cobalt-60	30 years 5.26 years	Leaves	~2 mCi	Leaves	Unknown	Inactive, contaminated leaves removed
8	Oct. 5, 1971	$^{197}Hg(NO_3)_2$	65 hours	Liquid	4.48 mCi	Water, fish, plants, periphyton, sediments	100 m section of stream	Inactive
9	Aug. 15, 1964	Cesium-134	2.05 years	Liquid	5.69 mCi	Foliage, insects	100 m^2	Inactive
10	May 4, 1966	Calcium-45	165 days	Liquid	~30 mCi	Foliage, wood, soil	Unknown, probably <1 ha	Inactive
11	Jan. 28, 1969	Cesium-137 Iron-59	30 years 45.6 days	Animals, contaminated by injection	32 µCi 12.8 µCi	Animal tissue Animal tissue	400 m^2	Inactive, radioactivity removed
13	Sept. 1, 1971	$CH_3\ ^{203}HgCl$	46.9 days	Liquid	1.65 mCi	Water, fish, plants, periphyton, sediment	100 m section of stream	Inactive
14	May 16, 1971	Tritium (H-3)	12.26 years	Liquid	180 mCi	leaves, wood, soil	<0.25 ha	Inactive, contaminated trees removed
15	July 31, 1969	Cesium-137 Cobalt-60	30 years 5.26 years	Liquid Liquid	11.6 mCi 99.4 mCi	Seeds, animals	4 ha 4 ha	Inactive Inactive

enclosure dedicated for cesium-137 application. Among the isotopes applied were cesium-134, chromium-51, rubidium-86, cadmium-109, mercury-203, mercury-197, and technetium-95m. This environmental research area is probably of major significance among potential remedial action of research sites since it occupies a large land area, involves a moderately long-lived isotope, and significant radioactivity.

Cesium-137 is an important long-lived (30 years) component of weapons fall-out. This radionuclide emits both beta (0.52 and 1.18 MeV) and gamma (0.66 MeV) radiation. Because of civil defense interests in postattack survival, a large field study was commissioned beginning in 1966. The study area (Fig. 3.1) is located north of the Clinch River (~100 m) at mile 20.5, on an azimuth south, 2.1 km from the intersection between Bethel Valley Road and State Highway 95 (see Map S-16A). The site consists of a 2 hectare-fenced area in a fescue grassland community. Within the fenced area are eight 100 m^2 treatment enclosures. Sides (metal sheeting) of each enclosure are extended 18 inches (45 cm) below the surface and 24 inches (61 cm) above ground. In August of 1968, four of the enclosures were contaminated and four remained as controls.

The contaminant consisted of cesium-137 fused at high temperatures to silica particles (100 μCi/g). The particles ranged from 88 to 177 μm in diameter. Each enclosure received approximately 2.2 Ci of radioactivity or a total of 8.8 Ci to the site. The particle size distribution was selected to simulate particle diameters characteristic of weapons fallout. Foliage and soil were contaminated directly by

sedimentation, with the fraction intercepted by foliage subject to weathering losses to the litter and soil.

Sampling for radiometric analysis was by vegetation clipping and soil coring. After analyses, all samples were disposed of according to ORNL procedures. The remaining radioactivity associated with each enclosure has been subject to weathering and radiological decay. Nearly 18 years (~0.6 half-life) have passed since the pens were contaminated. Correcting for radiological decay, ~5.7 Ci of activity would remain today, assuming no particle losses due to weathering, runoff, wind transport, etc. The site is presently inactive. Pertinent photos and drawings describing the site are numerous and contained in the various publications and reports resulting from the activity.

A number of short-lived radioisotopes were applied in field studies (Cr-51, Rb-96) outside the fenced enclosure. However, as numerous half-lives have passed, the isotopes are below detection limits. They are identified by bibliographic inclusion in the references listed below and discussed in Sections 4.3 through 4.7.

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Dahlman, R. C., C. W. Francis, and T. Tamura. 1975. Radiocesium cycling in vegetation and soil. pp. 462-481. IN F. G. Howell, J. B. Gentry, and M. H. Smith (eds.), Mineral Cycling in Southeastern Ecosystems. CONF-740513. 898 pp.

Matti, C. S., J. P. Witherspoon, and B. G. Blaylock. 1975. Cycling of mercury and cadmium in an old-field ecosystem during one growing season. ORNL/NSF/EATC-10. Oak Ridge National Laboratory, Oak Ridge, TN. 75 pp.

Huckabee, J. W., R. A. Goldstein, S. A. Janzen, and S. E. Woock. 1977. Methylmercury in a freshwater food chain. pp. 199-216. IN Proceedings, International Conference on Heavy Metals in the Environment, Toronto, Ontario, Canada, October 27-31, 1975.

Taylor, F. G., P. D. Parr, and F. Ball. 1978. Interception and retention of simulated cooling tower drift by vegetation. pp. I-39 - I-48. IN R. Nietubicz and L. Green (eds.), Cooling Tower Environment - 1978. Water Resources Center Special Report No. 9, University of Maryland, College Park, MD; and 1980. Atmos. Environ. 14:19-25.

Styron, C. E., G. J. Dodson, J. J. Beauchamp, and F. L. Miller, Jr. 1976. Responses of a grassland arthropod community to a chronic beta and gamma radiation. pp. 381-385. IN C. E. Cushing (ed.), Radioecology and Energy Resources. Dowden, Hutchinson and Ross, Inc., Stroudsburg, PA.

Huckabee, J. W., D. M. Lucas, and J. M. Baird. 1981. Occurrence of methylated mercury in a terrestrial food chain. Environ. Res. 26:174-181.

Hoffman, F. Owen, Charles T. Garten, Jr., John W. Huckabee, and Deborah Lucas. 1982. Interception and retention of technetium by vegetation and soil. J. Environ. Qual. 11:134-141.

Garten, Charles T., Jr., F. Owen Hoffman, and Ernest A. Bondietti. 1984. Field and greenhouse experiments on the fate of technetium in plants and soil. Health Phys. 46:647-656.

Because publications and reports concerning the Cs-137 site and the 0800 area are extensive, a number of individuals are knowledgeable concerning the site. Among those are several major sources, notably:

Abner, C. H., Plant and Equipment Division, ORNL, 615/574-4385
Auerbach, S. I., Environmental Sciences Division, ORNL, 615/574-7302
Dahlman, R. C., DOE, Washington, DC, FTS 233-3281
Dunaway, P. B., DOE, Oak Ridge, TN, 615/576-0744
Garten, C. T., Jr., Environmental Sciences Division, ORNL, 615/574-7355

Kaye, S. V., Health and Safety Research Division, ORNL, 615/574-5845
Taylor, F. G., Environmental Sciences Division, ORNL, 615/574-7354
Witherspoon, J. P., Health and Safety Research Division, ORNL, 615/576-2105

3.2 AREA 2: CS-137 CONTAMINATED FOREST, SOIL, AND VEGETATION

This site is known as the "cesium forest." It represents a major accomplishment in radioecological research, as a small forest ecosystem was contaminated with cesium-137. Thirty trees, ranging up to 30 m (~100 feet) tall, were inoculated with cesium to determine the movement of this nuclide and as an analog to the essential element, potassium. A total of 467 mCi cesium-137 was introduced into the transpiration stream of a number of yellow-poplar trees. The site containing the contaminated trees is a 20x25 m plot (500 m^2). It is located on an azimuth bearing of E66°S, a distance of 2.92 km (1.8 miles) from the intersection of Bethel Valley Road and Melton Valley Access Road (Fig. 3.1). The site is a lower slope forest of Copper Ridge. The isotope was introduced into the tree stems (trunks) over a three-day period (May 20-23, 1962). Since the isotope was injected into the trees, nearly 24 years have passed (~0.8 half-life). Correcting only for radiological decay, approximately 270 mCi would remain today. This amount has been further decreased by wind distribution of leaves, movement through soil, and runoff. Numerous publications were derived from research conducted on the site over the years. The most significant, relative to providing detailed information concerning the introduction of the isotope, are:

Auerbach, S. I., J. S. Olson, and H. D. Waller. 1964. Landscape investigations using cesium-137. Nature 201:761-764.

Peters, L. N., J. S. Olson, and R. M. Anderson. 1969. Eight-year summary of plant-soil data on ^{137}Cs in a tagged Appalachian forest dominated by Liriodendron tulipifera. ORNL/TM-2737. Oak Ridge National Laboratory, Oak Ridge, TN. 235 pp.

Waller, H. D., and J. S. Olson. 1967. Prompt transfer of cesium-137 to the soils of a tagged Liriodendron forest. Ecology 48:15-25.

Several persons knowledgeable of the site are currently employees at ORNL. Primary contacts are:

Auerbach, S. I., Environmental Sciences Division, ORNL, 615/574-7302
Witherspoon, J. P., Health and Safety Research Division, ORNL, 615/576-2105

Another important source of information is an employee retired from ORNL:

Olson, J. S., ORNL Retiree, 615/376-2250

3.3 AREA 3: CA-45 TAGGED TREES

The radioactivity at this site is associated with the contamination discussed in Site 4. Notably, several red maple trees were inoculated with 1.25 Ci of calcium-45 and allowed to move in the transpiration stream throughout the tree. The site is on Walker Branch Watershed (Chestnut Ridge), a distance of 3.2 km (2 miles) on an azimuth N24°E from the intersection of Bethel Valley Road and Melton Valley Access Road (Fig. 3.1). The isotope was introduced in the trees on June 7, 1969. Following several days posttagging, all foliage from the trees was removed, processed, and prepared for introduction to the soil site described as Site 4. Few persons at the laboratory have first-hand information about the contamination. Individuals who are knowledgeable include:

Grizzard, T., Operations Division, ORNL, 615/576-5224
Grigal, D. F., University of Minnesota, St. Paul, FTS 612/373-1443
Kelly, J. M., TVA-ORNL Project, 615/574-7815

Some details concerning isotope introduction to the tree are contained in the following publication:

Grigal, D. F. Calcium cycling: Diffusion into a forest soil. pp. 218-224. IN Third National Symposium on Radioecology. CONF-710501.

3.4 AREA 4: CA-45 TAGGED SOIL AND VEGETATION

A small plot in an oak-hickory forest on the Walker Branch Watershed was contaminated with calcium-45 (165-day half-life) labelled foliage to determine the movement of the isotope into soil and soil-water solution. The duration of the study was 72 weeks. The site is located N31°E on an azimuth, 3.6 km (2.2 miles) from the intersection of Bethel Valley Road and Melton Valley Access Road (Fig. 3.1). The plot was contaminated by introducing maple leaves to the soil, which had previously been inoculated with Ca-45 at another location (Site 3). In addition, a quantity of contaminated leaves were placed in mesh bags and dispersed on the site. Total radioactivity introduced was 136 mCi on December 20, 1969. At the termination of the experiment, the leaves contained in the mesh bags were removed and disposed of. A single publication, as noted below, provides details of the study.

Grigal, D. F. Calcium cycling: Diffusion into a forest soil. pp. 218-224. IN Third National Symposium on Radioecology. CONF-710501.

Knowledgeable persons concerning the site are:

Grizzard, T., Operations Division, ORNL, 615/576-5224
Grigal, D. F., University of Minnesota, St. Paul, FTS 612/373-1443

3.5 AREA 5: NA-22 CONTAMINATED SOIL

This site is in the vicinity of the 0800 area (Site 1). It is located off Jones Island Patrol Road near the Clinch River on a bearing S10°W, a distance of 1.6 km (1.1 miles) from the intersection of Bethel Valley Road and State Highway 95 (Fig. 3.1). The area originally was a grassland, but it is now a dense thicket of shrubs. Remnants of a two-strand wire fence indicate that the site was 0.5 ha (~1.2 acres) in area. Activities at the site span several years (1968-1969) and involved ^{47}calcium and ^{42}potassium in addition to the ^{22}sodium. Sodium was the most significant isotope because of its longer half-life (2.62 years). The exact date of isotope application and quantity used is not known. It is believed that the isotope was not actually applied to the soil, but instead to vegetation which was contaminated in the laboratory, fed to grasshoppers and crickets, which were housed in cages in the field. The primary information concerning the site is in the following reports.

Van Hook, R. I., D. E. Reichle, and S. I. Auerbach. 1970. Energy and nutrient dynamics of predator and prey arthropod populations in a grassland ecosystem. ORNL-4509. Oak Ridge National Laboratory, Oak Ridge, TN.

Van Hook, Robert I., Jr. 1971. Energy and nutrient dynamics of spiders and orthopteran populations in a grassland ecosystem. Ecological Monographs 4:1-26.

Persons knowledgeable of the site and the isotope application are:

Van Hook, R. I., Environmental Sciences Division, ORNL, 615/574-7374
Reichle, D. E., Environmental Sciences Division, ORNL, 615/574-7301

3.6 AREA 6: CS-137 TAGGED AREA FOR RADIONUCLIDE RUNOFF STUDIES

The objective of this field contamination was to investigate runoff, erosion, and infiltration of cesium-137 on a silt-loam soil. On a bearing of E83°S, the site is located approximately 2 km (1.2 miles) from the junction of Bethel Valley Road and State Highway 95 (Fig. 3.1). The site is due north of the confluence of White Oak Creek and the Clinch River, a distance of 0.3 km (0.2 mile). The isotope (15 mCi total) was introduced to the soil in a liquid spray on October 20, 1964. Since the date of contamination, 21.6 years have passed (~0.72 half-life). Correcting for radiological decay, 9.15 mCi of activity would remain, providing no losses from the system. Total land actually contaminated was <20 m^2. Presently, the inactive site is a grass-shrub dominated old field. Pertinent drawings and illustrations are contained in the following publications:

Rogowski, A. S., and Tsuneo Tamura. 1965. Movement of ^{137}Cs by runoff, erosion, and infiltration on the alluvial captina silt loam. Health Phys. 11:1333-1340.

Tamura, Tsuneo. 1967. Movement of cesium-137 by runoff, erosion, and infiltration from a soil under different cover conditions. pp. 149-167. IN Postattack Recovery from Nuclear War. NAS/NAE/NRC.

Rogowski, A. S., and Tsuneo Tamura. 1970. Erosional behavior of cesium-137. Health Phys. 18:467-477.

Rogowski, A. S., and T. Tamura. 1970. Environmental mobility of cesium-137. Radiation Botany 10:35-45.

Few persons are knowledgeable of the activities concerning the site. They are:

Tamura, T. S., Environmental Sciences Division, ORNL, 615/576-8256
Rogowski, A. S., Northeast Watershed Research Center, University Park, PA 16802

3.7 AREA 7: CS-137 TAGGED LITTER BAGS

Information concerning this environmental research area is the most elusive of all the sites. The site was utilized as a litter bag study area to determine the differences in isotope solubility from leaves under natural conditions. The site is located south of Melton Valley Drive on an azimuth E83°S, a distance of 1.3 km (0.8 mile) from the intersection between Bethel Valley Road and Melton Valley Access Road (Fig. 3.1). Leaves contaminated with cesium-137 and cobalt-60 were contained in the litter bags, placed in the pine-oak forest for a year, and periodically removed for radiological analysis. The amount of radioactivity is not known, but assumed to be approximately 2 mCi, equally divided between the cesium and cobalt. Date of contamination is not known, but probably early 1961. More than 25 years have lapsed (0.8 half-life for the cesium-137 and 4.8 half-lives for the cobalt-60) since the litter bags were first placed in the field. If all bags and contents remained at the site, approximately 0.55 mCi and 36 μCi would remain for cesium and cobalt, respectively. At the completion of the study all experimental materials were removed. Additional litter bags containing leaves tagged with ruthenium-106 and strontium-85 were studied at the site. The quantity of those radioisotopes used is

unknown. A very brief mention of the activity is contained in the following reports:

Witkamp, Martin, and Marilyn L. Frank. 1965. Losses of ^{137}Cs and ^{60}Co from leaf litter. pp. 72. IN Health Physics Division Annual Progress Report for the period ending July 31, 1965. ORNL-3849. Oak Ridge National Laboratory, Oak Ridge, TN.

Olson, J. S. 1963. Tracer studies of the breakdown of forest litter. pp. 411-416. IN First National Symposium on Radioecology. Reinhold Publishing Corporation, NY.

Only one person had a knowledge of the site, but only peripherally as the study was nearly complete at the time of her employment.

Frank, M. L., Environmental Sciences Division, ORNL, 615/574-7390

3.8 AREA 8: $^{197}Hg(NO_3)_2$ CONTAMINATION STREAM

A 100 m section of Walker Branch was tagged with radioactive mercuric nitrate [$^{197}Hg(NO_3)_2$] to determine the fate of that mercury compound in a natural stream ecosystem. Mercury-197 has a 65-hour half-life. The location of radionuclide introduction was downstream from the Walker Branch Watershed project weirs. The site is on a bearing N41°E, 2.2 miles (3.6 km) from the intersection of Bethel Valley Road and Melton Valley Access Road (Fig. 3.1). A total of 4.48 mCi of radioactivity was introduced into the stream with a flow rate of 347.8 l/min on October 5, 1971. The sample matrix consisted of water, fish, watercress, periphyton, and sediments. Seventy-five percent of the isotope was retained in the first 100 m following tagging. Literature concerning the study is contained in two progress

reports identified below. These contain drawings and illustrations pertinent to the site.

Huckabee, J. W., B. G. Blaylock, and N. A. Griffith. 1971. Behavior of methylmercury and mercuric nitrate in natural stream ecosystems. pp. 135-137. IN Ecological Sciences Division Annual Progress Report. ORNL-4759. Oak Ridge National Laboratory, Oak Ridge, TN.

Blaylock, B. G., R. A. Goldstein, J. W. Huckabee, Sherry Janzen, Carol Matti, R. G. Olmstead, Mara Slawsky, R. A. Stella, and J. P. Witherspoon. 1973. Ecology of toxic metals. pp. 121-150. IN Ecology and Analysis of Trace Contaminants. ORNL-NSF-EATC-1. Oak Ridge National Laboratory, Oak Ridge, TN.

Several individuals are knowledgeable concerning the site and the isotope application. They are:

Blaylock, B. G., Environmental Sciences Division, ORNL, 615/574-7397
Elwood, J. W., Environmental Sciences Division, ORNL, 615/574-7370
Huckabee, J. W., Electric Power Research Institute, Menlo Park, CA, 414/855-2589

3.9 AREA 9: CS-134 TAGGED TREE (inoculated red cedar tree)

Actual contamination at this site involved a single red cedar tree in a field inoculated on August 15, 1964, with 5.69 mCi of cesium-134. Approximately 10.7 half-lives have lapsed since the tree was contaminated, potentially leaving 3.4 µCi of radioactivity. This quantity has been further diluted by natural processes, weathering, leaf transport, and soil leaching. The study was designed to determine the uptake and transfer of the radionuclide through metamorphosis of the bagworm moth. Approximately 100 m^2 of area was involved. Insects were contained in cages suspended by wires near the canopy of the host tree. The site is located 4.5 km (2.2 miles) on an azimuth of N55°E from the intersection of Bethel Valley Road and Melton Valley Access Road

(Fig. 3.1). Details of the isotope application are contained in the following:

Reichle, D. E., and G. J. Dodson. 1965. Radiocesium movement in a foliage-feeding canopy insect. pp. 68-69. IN Health Physics Division Annual Progress Report for the period ending July 31, 1965. ORNL-3849. Oak Ridge National Laboratory, Oak Ridge, TN.

The most knowledgable person concerning the site and activities is:

Reichle, D. E., Environmental Sciences Division, ORNL, 615/574-7301

3.10 AREA 10: CA-45 TAGGED FOREST

Twelve dogwood trees were inoculated on May 4, 1966 with calcium-45. Inoculum ranged from 0.67 to 2.51 mCi among the trees. If one assumes the maximum radioactivity, a conservative estimate of ~30 mCi calcium-45 was introduced into the biota of the site. The purposes of the contamination study were to document the accumulation of calcium by various plant organs and to determine the rate of calcium cycling in the tree-soil systems. The site is in Roane County, 0.7 km (0.45 mile) on an azimuth of W25°S from the intersection of Bethel Valley Road and State Highway 95 (Fig. 3.1). Sample matrix consisted of wood, foliage, and plant organs (flowers, etc.). The extent of contamination is unknown, but assumed to be less than 1 ha (2.47 acres). Few people at the laboratory have personal knowledge concerning the site or the application of the radioisotope. Advisors to the study were:

Auerbach, S. I., Environmental Sciences Division, ORNL, 615/574-7302
Olson, J. S., retired ORNL employee, 615/376-2250

Major reports containing details of the isotope introduction in the trees are:

Thomas, William A. 1969. Cycling of ^{45}Ca by dogwood trees. pp. 661-664. IN Second National Symposium on Radioecology. CONF-670503.

Thomas, William A. 1969. Accumulation and cycling of calcium by dogwood trees. Ecological Monographs 39:101-120.

Thomas, William A. 1970. Retention of calcium-45 by dogwood trees. Plant Phys. 45:510-511.

3.11 AREA 11: MCNEW HOLLOW, CESIUM-137, IRON-59 CONTAMINATED AREA

This study was conducted to determine the elimination of cesium-137 and iron-59 by wild small rodents in the field. Cotton rats, contaminated in a dual isotope technique, were released into 10x10 m plots in McNew Hollow. A total of four animals were placed in each of four pens. Total radioactivity (introduced in January 1969) consisted of 12.8 µCi $^{59}FeCl$ and 32 µCi $^{137}CsCl$. To date, iron-59 has decayed through 139 half-lives, whereas the cesium-137 has lapsed a fraction (0.58) of a half-life. Correcting for radiological decay, ~21 µCi of cesium would remain. Iron-59 would no longer be detectable. McNew Hollow is located on an azimuth N18°W, a distance of 3.4 km (2.1 miles) from the intersection of Bethel Valley Road and State Highway 95 (Fig. 3.1). Animals were periodically live-trapped and taken into the laboratory for radiological analysis. At the termination of the study, all animals were removed, sacrificed in the laboratory for the determination of radionuclide distribution among body tissues. Two major publications, containing pertinent photographs and illustrations, are as follows:

Baker, C. E., P. B. Dunaway, and S. I. Auerbach. 1970. Relationship of cesium-137 and iron-59 elimination rates to metabolic rates for small rodents. ORNL-4568. Oak Ridge National Laboratory, Oak Ridge, TN.

Baker, Charles E., and Paul B. Dunaway. 1975. Elimination of ^{137}Cs and ^{59}Fe and its relationship to metabolic rates of wild small rodents. J. Exp. Zoology 192:223-236.

Individuals knowledgeable about the site and the isotope application are:

Story, J. D., Environmental and Occupational Safety Division, ORNL, 615/576-7712.
Dunaway, P. D., DOE, Oak Ridge, TN, 615/576-0744

3.12 AREA 13: $CH_3\ ^{203}HgCl$ CONTAMINATED STREAM

A 100 m section of White Oak Creek was tagged with radioactive methylmercury ($CH_3\ ^{203}HgCl$) to clarify the fate of that mercury compound in a natural stream ecosystem. Mercury-203 has a 46.9 day half-life. The point of stream inoculation is located one-half mile (~0.8 km) on a bearing N12°E from the junction of Bethel Valley Road and Melton Valley Access Road (Fig. 3.1). The isotope (1.65 m Ci) was introduced on September 1, 1971. The sample matrix consisted of water, fish, watercress, periphyton, and sediments. At the time of the tagging the streamflow was 325.6 l/min. Eighty-one percent of the radioisotope was retained in the first 100 m after release. Detailed information concerning the research site is contained in the two progress reports noted below:

Huckabee, J. W., B. G. Blaylock, and N. A. Griffith. 1971. Behavior of methylmercury and mercuric nitrate in natural stream ecosystems. pp. 135-137. IN Ecological Sciences Division Annual Progress Report. ORNL-4759. Oak Ridge National Laboratory, Oak Ridge, TN.

Blaylock, B. G., R. A. Goldstein, J. W. Huckabee, Sherry Janzen, Carol Matti, R. G. Olmstead, Mara Slawsky, R. A. Stella, and J. P. Witherspoon. 1973. Ecology of toxic metals. pp. 121-150. IN Ecology and Analysis of Trace Contaminants. ORNL-NSF-EATC-1. Oak Ridge National Laboratory, Oak Ridge, TN.

Those individuals with knowledge concerning the site and the isotope application are:

Blaylock, B. G., Environmental Sciences Division, ORNL, 615/574-7397
Elwood, J. W., Environmental Sciences Division, ORNL, 615/574-7370
Huckabee, J. W., Electric Power Research Institute, Menlo Park, CA, 414/855-2589

Drawing and illustrations providing detailed data are contained in the reports noted.

3.13 AREA 14: TRITIUM-TAGGED TREES AND SOIL (0800 area)

This site is located on Blackoak Ridge adjacent to the East Ridge Patrol Road, on an azimuth W8°S, 2.6 km from the confluence of East Ridge Patrol Road and the Oak Ridge Turnpike (Fig. 3.1). This ridgetop site was utilized in a pilot study to investigate the feasibility of using tritiated water to measure rates of transpiration of deciduous tree species under field conditions. Tagging consisted of introducing tritium (12.26 yr half-life) into two yellow-poplar trees, one hickory tree, and a 4 m^2 soil plot. The quantity of tritium used is subject to question. A file report (Wing 1973) suggests from 80 to 180 mCi of tritiated water was used. As the latter estimate was obtained from the principal investigator, it is assumed to be correct. Tritium was introduced to the trees on May 16, 1971. Correcting for radiological decay, ~77 mCi would be present today. Area involved in the

experiment was about 0.6 acre (~0.25 ha). Following the study all materials were removed. The site is currently an inactive research area. As this was a pilot study, the data were only summarized in annual report form. Two sources of detailed information concerning the tritium-tagged trees and soil are noted below. These reports contain drawings and illustrations pertinent to the site.

Auerbach, S. I., et al. 1971. Ecological Sciences Division Annual Progress Report for period ending September 30, 1971. ORNL-4759. Oak Ridge National Laboratory, Oak Ridge, TN. 188 pp.

Wing, J. F. 1973. Tritium-tagged ecology plot. Safety and Environmental Control Division Files, U.S. Atomic Energy Commission.

Several individuals have knowledge concerning the site. They are:

Auerbach, S. I., Environmental Sciences Division, ORNL, 615/574-7302
Henderson, G., School of Forestry, University of Missouri, Columbia, MO, 314/755-8373
Kaye, S. V., Health and Safety Research Division, ORNL, 615/574-5845
Watson, A. P., Health and Safety Research Division, ORNL, 615/576-2125

3.14 AREA 15: CESIUM-137 AND CO-60 CONTAMINATED AREA

More specifically, this study involved cesium-137 and cobalt-60. Three plots served as the study design, a control and two radioisotope treatment plots. The two treatment plots were 1.96 hectares each (4.85 acres) and 2 km distance apart. Each plot was subdivided into a 10 m grid system, with feeder stations containing radiolabelled seeds dispersed in feeders on a 20 m grid. Each plot, therefore, had 64 feeders. Each plot received radioactive seeds at weekly intervals for 58 weeks. Treatment plots were located on Chestnut Ridge (Fig. 3.1), one on a bearing of S84°W, a distance of 1.35 km (0.8 mile)

from Building 2001 at ORNL, the other on a bearing N49°E, a distance of 2.51 km (1.55 miles). Fifty-nine kg of seeds were soaked in 71 mCi of Cs-137 and 554 mCi of Co-60. The percents retained after water and acid leaching were 19 and 21, respectively. The concentrations in the seeds were 0.23 mCi/kg for cesium-137 and 1.97 mCi/kg for cobalt-60. During the course of the study, 0.435 kg of radioactive seeds were dispersed weekly for 58 weeks, amounting to ~50.5 kg for the two plots. This resulted in each plot receiving a total of ~5.8 mCi of cesium-137 and 49.7 mCi of cobalt-60 between July 31, 1969, and September 3, 1970. The objective of the study was to utilize the radioisotope to determine annual and seasonal consumption rates of white pine seeds by small forest mammals. Animals in the study plots were live-trapped, isotopic body burden determined in the laboratory, and animals returned to the forest at the point of trapping. As a result of the long-term study, the radioactive materials were constantly being removed from the site by feeding, but a fraction was being returned through body-elimination processes. Correcting for radiological decay, ~4 mCi of cesium-137 could possibly be present today, in comparison to 5.5 mCi of cobalt-60. These quantities were further diluted by feed (food) placement by the resident animals, in excrement of animals meandering from the site, and scavenging by animals passing through the area. The primary publication describing this environmental research site is noted below:

Mathies, John B., P. B. Dunaway, Gary Schneider, and S. I. Auerbach. 1972. Annual consumption of cesium-137 and cobalt-60 labelled pine seeds by small mammals in an oak-hickory forest. ORNL/TM-3912. Oak Ridge National Laboratory, Oak Ridge, TN.

Several persons are knowledgeable concerning the research site, notably:

Auerbach, S. I., Environmental Sciences Division, ORNL, 615/574-7302
Dunaway, P. B., DOE, Oak Ridge, TN, 615/576-0744
Story, J. D., Environmental and Occupational Safety Division, ORNL, 615/576-7712

Pertinent drawings and photographs are contained in the report. The site is presently in an inactive status.

4. DESCRIPTION OF ADDITIONAL ENVIRONMENTAL RESEARCH SITES (1972-PRESENT)

During the almost three decades of ecological research at ORNL, there were several research areas which were not included in the tabulations that led to the Compendium of Remedial Action Sites (Grimsby 1986). Those areas included in the compendium (Tables 3.1 and 3.2) represent major activities up to 1971. Even though there were less emphases on radionuclide studies since 1971, several significant projects were initiated and completed (Table 4.1). Most of these later studies involved radioisotopes of minimum energies and radiation doses (e.g., C-14) or short half-lives (e.g., Rb-86, Zn-65, Cr-51). In this section 23 additional environmental research areas between 1972 and 1986 are discussed. Two areas, active sites in the early 1960's, are also reviewed. Table 4.1 provides a historical summary of the additional areas. The area numbers in Table 4.1 represent a means to catalogue the additional areas and should not be confused with ERFU program designated numbers.

Table 4.1 Detail summary concerning radioisotope usage at other environmental research areas

Area number	Date of contamination	Principal radionuclide	Half-life	Form of contaminant	Quantity of radioactivity	Environmental matrix	Extent of contamination	Status
16	April 1960	Cesium-134 Potassium-42	2.05 years 12.4 hours	Liquid Liquid	24 mCi 4 mCi	Foliage, wood, soil	~400 m^2	Inactive, trees removed
17	June 1965	Zinc-65	245 days	Liquid	1.3 mCi	Soil, stem, foliage	~100 m^2	Inactive, all materials (plants, pots, etc.) removed
18	June 1968	Cesium-134	2.05 years	Particles	261 µCi	Soil, stem, foliage	100 m^2	Inactive, all materials removed
19	June 1969	Rubidium-86	18.66 days	Particles	32.28 mCi	Soil, stem, foliage	200 m^2	Inactive
20	July 1970	Cesium-134	2.05 years	Particles	682 µCi	Plants, soil	200 m^2	Inactive
21	June 1970	Cesium-134	2.05 years	Particles	1.23 mCi	Grass, soil	100 m^2	Inactive
22	June 1971	Cesium-134	2.05 years	Particles	147 µCi	Lichens, moss, soil	100 m^2	Inactive
23	September 1978	Technetium-95m Technetium-95m Technetium-95m	61 days 61 days 61 days	Liquid Liquid Liquid	3 mCi 40 µCi 30 µCi	Grass, soil Grass, soil Grass, soil	100 m^2 100 m^2 100 m^2	Inactive Inactive Inactive
24	February 1981	Technetium-95m	61 days	Liquid	336 µCi	Grass, soil	<10 m^2	Inactive
25	September 1983	Iodine-131 Technetium-95m	8.05 days 61 days	Liquid Liquid	60 mCi 10 mCi	Grass, soil, goats	4 ha	Inactive
26	July 1976	Chromium-51	27.8 days	Liquid	125 µCi	Grass, soil	100 m^2	Inactive
27	June 1984	Neptunium-237 Technetium-99	2.14E6 years 2.12E5 years	Liquid Contaminated Soil	64 µCi 32 µCi	Soil, plants Soil, plants	<3 m^2 <1 m^2	Inactive Inactive
28	April 1964	Cesium-137	30 years	Liquid	~1 mCi	Leaf litter, soil	25 m^2	Inactive

Table 4.1 (CONT'D)

Area number	Date of contamination	Principal radionuclide	Half-life	Form of contaminant	Quantity of radioactivity	Environmental matrix	Extent of contamination	Status
29	June 1966	Cesium-137	30 years	Liquid	350 µCi	Plants, litter, soil	10 m^2	Inactive
30	June 1964	Cesium-137	30 years	Liquid	5 mCi	Soil, grass	<6 m^2	Inactive
31	June 1970	Cesium-134	2.05 years	Liquid	2 mCi	Wood, foliage, soil	~35 m^2	Inactive
32	July 1970	Cobalt-60 Manganese-54	5.2 years 0.82 years	Liquid Liquid	17 µCi 17 µCi	Animal tissue, excretae	~1 ha	Inactive
33	June 1983 June 1984	Carbon-14	5479 years	Gas	1 mCi 2 mCi	Foliage, wood, roots	~200 m^2	Inactive
34	October 1972	Carbon-14	5479 years	Liquid	5 mCi 4.95 mCi	Foliage, wood, roots	~100 m^2	Inactive
35	Between 1972 and 1977	Carbon-14	5479 years	Gas and liquid	100 µCi 10 µCi	Foliage, wood, roots	~200 m^2	Inactive
36	June and November 1979	Carbon-14	5479 years	Gas	360 µCi	Foliage, wood, roots	~0.5 ha	Inactive
37	July 1976	Carbon-14	5479 years	Liquid	3 mCi	Foliage, wood, roots, and soil	~200 m^2	Inactive
38	June 1984	Carbon-14	5479 years	Gas	~1 mCi	Foliage, wood, roots	~0.5 ha	Inactive

4.1 AREA 16: CESIUM-134 CONTAMINATED WHITE OAK TREES

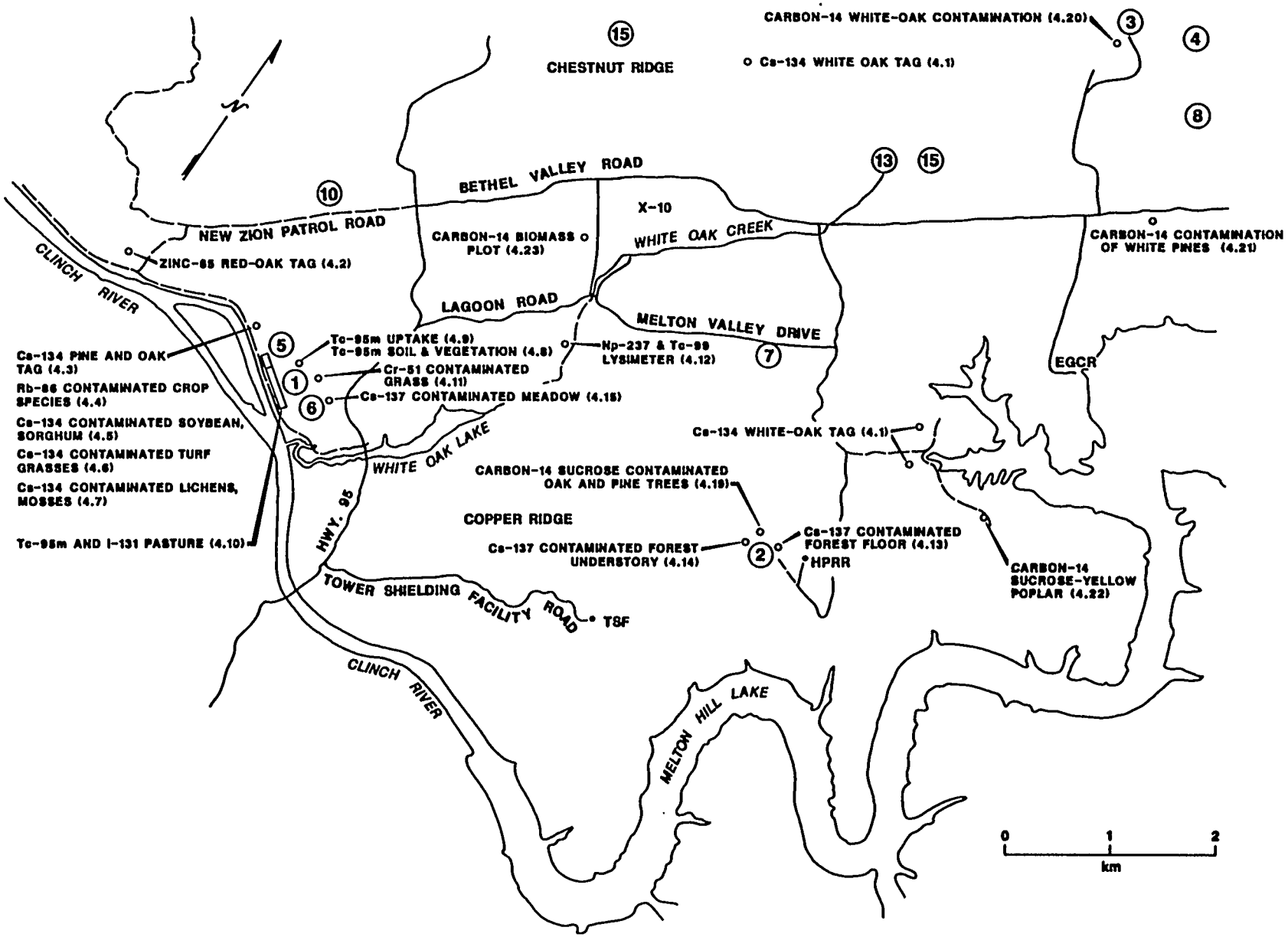
Among the earliest deliberate isotope applications was the inoculation of 12 white oak trees on sites of contrasting soil type and moisture conditions (Witherspoon 1964). Three trees on each of four soils were injected with 2 mCi each of Cs-134 in April of 1960. Two of the trees were double-tagged with 2 mCi of K-42 to ascertain distribution similarities between potassium and cesium. Each site covered an area of <50 m^2. Throughout the growing season leaves were collected and analyzed for radionuclide distribution, and at the end of the study each tree was cut and the roots excavated for analyses. The four contrasting soil types were as follows:

Landisburg soil on Conasauga shale
Fullerton soil on Conasauga shale
Monongahela soil on Knox dolomite
Sequoria soil on Knox dolomite.

The Landisburg and Fullerton sites are on the north slope of Chestnut Ridge along azimuths of N29°E and W32°N and distances of 2.7 miles (4.4 km) and 1 mile (1.6 km), respectively, from the confluence of Bethel Valley Road and Melton Valley Access Road. The Monongahela and Sequoria sites are in Melton Valley along azimuths of E30°S and E36°S from the intersection of Bethel Valley Road and Melton Valley Access Road and distances of 1.3 miles (2.1 km) and 1.4 miles (2.2 km), respectively. A detailed map of the plots is depicted in Fig. 4.1. Since isotope introduction to the trees, thousands of half-lives have lapsed for the potassium (K-42) isotope while more than 12 half-lives have lapsed for cesium-134. If all of the original cesium inoculum

Fig. 4.1 Locations of environmental research areas not included in the Compendium of Remedial Action Sites (Grimsby 1986). Circled numbers are keyed to areas identified in Tables 3.1 and 3.2. Numbers in parenthesis are keyed to sections in the text.

ORNL-DWG 88-12055



remained in any single tree, ≤0.3 μCi of activity would remain. Because the trees were harvested at the end of the study, it is estimated that only 25% of the inoculum would remain. This would result in <0.1 μCi remaining at any tree site (Witherspoon 1986). Details of the study are presented in the monograph by Witherspoon (1964). The primary contact concerning information of this site is J. P. Witherspoon, Health and Safety Research Division, ORNL, 615/576-2105.

4.2 AREA 17: ZINC-65 CONTAMINATED RED OAK SEEDLINGS

In June of 1965, gamma-irradiated red oak seedlings were contaminated with zinc-65 to investigate the effects of ionizing radiation on zinc uptake (Brown and Taylor 1966). A total of 1.3 mCi was introduced into 26 (50 μCi/pot) containers. Foliage was subsampled periodically during a 12-week period to ascertain the uptake and distribution of the nuclide. At the end of the study all experimental materials were removed. Considering that 31 half-lives have lapsed, <1 pCi (total) of activity would remain if all the experimental materials remained. The site is located about 200 m north of Clinch River Mile 19.25 and 1.6 miles (2.5 km) on an azimuth of S46°W from the intersection of State Highway 95 and New Zion Patrol Road (Fig. 4.1). The primary contact concerning information of this site is Fred G. Taylor, Environmental Sciences Division, ORNL, 615/574-7354.

4.3 AREA 18: CESIUM-134 CONTAMINATION OF PINE AND OAK TREES

A number of short-term experiments, conducted in the 0800 area (see Section 3.1) were designed to determine initial interception and retention of fallout particles by various plant taxa. In the initial study, white pine and red oak seedlings were contaminated (June 1968) with cesium-134 particles (Witherspoon and Taylor 1969). A total of 261 µCi was applied to 30 seedlings (8.7 µCi/pot) and sampled periodically over 33 days. The manner of sampling removed 10% of the experimental units (pot-soil-plant) with each collection. By the end of the study all experimental materials had been removed with possible contamination of the soil and grass by dislodged particles. Considering that nine half-lives have lapsed, <1 µCi (total) of activity would be present if all the experimental materials were left in the field. The site is located on an azimuth of S18°W, a distance of 1.1 miles (1.7 km) from the intersection of State Highway 95 and the New Zion Patrol Road (Fig. 4.1). Contacts for information concerning the site are:

Witherspoon, J. P., Health and Safety Division, ORNL, 615/576-2105
Taylor, F. G., Environmental Sciences Division, ORNL, 615/574-7354

4.4 AREA 19: RUBIDIUM-86 CONTAMINATION OF AGRICULTURAL PLANTS

In another fallout-related study (Witherspoon and Taylor 1970) agricultural species were contaminated with rubidium-86 to determine fractional interception and retention times of particles in two size classes. That study consisted of two 10x10 m plots located at the site (Fig. 4.1) of the pine-oak contamination (see Section 4.3). The

particles containing the rubidium-86 isotope were applied to the areas in June 1969. As the half-life of the radionuclide is 18.7 days, approximately 230 half-lives have lapsed. A total of 12.48 mCi were applied to one plot, whereas the other received 19.8 mCi. After seven weeks the study was completed. Two individuals who are knowledgeable concerning the site and the study are:

Taylor, F. G., Environmental Sciences Division, ORNL, 615/574-7354
Witherspoon, J. P., Health and Safety Research Division, ORNL, 615/576-2105

4.5 AREA 20: CESIUM-134 CONTAMINATED SOYBEAN AND SORGHUM PLANTS

This study was a continuation of research interests in plants and fallout (Witherspoon and Taylor 1971). Earlier studies utilized particles up to 175 μ in diameter, whereas this study was designed to investigate the behavior of smaller <10 μ diameter particles. Twenty-five plants each of soybean and sorghum received a total of 682 μCi (total) of cesium-134 in July of 1970. Three plants of each species were collected every week for seven weeks for radiological determination. The removal of plants and radiological decay reduced the radioactivity with each sampling period. Considering that 7.8 half-lives have lapsed, ~3 μCi (total) of activity would remain if all the experimental materials remained. The location (Fig. 4.1) of the site is the same as used in previous fallout studies (see Sections 4.3 and 4.4). Knowledgeable persons concerning the site and the studies include:

Taylor, F. G., Environmental Sciences Division, ORNL, 615/574-7354
Witherspoon, J. P., Health and Safety Research Division, ORNL, 615/576-2105

4.6 AREA 21: CESIUM-134 CONTAMINATION OF TURF AND FORAGE GRASSES

In another simulated fallout study, the interception and retention of cesium-134 contaminated particles on forage and turf grasses were investigated (Peters and Witherspoon 1970). Each of four plots (2.5x10 m) contained a different grass species, and each received 307 µCi of cesium-134 in June of 1970. The total activity to the site was 1.23 mCi. Location of the site (Fig. 4.1) is the same as described for the previous fallout studies (see Sections 4.3, 4.4, and 4.5). The sampling design was comparable to the other studies with samples collected weekly during the study (seven weeks). The removal of plant materials combined with radiological decay of the isotope reduced the radionuclide with each sampling period. Considering that nearly eight half-lives have lapsed since isotope introduction to the site, ~5 µCi (total) of activity would remain if all the experimental materials remained. The most knowledgeable person of the site and study is J. P. Witherspoon, Health and Safety Research Division, ORNL, 615/576-2105.

4.7 AREA 22: CESIUM-134 CONTAMINATION OF LICHENS AND MOSSES

In a final particulate fallout study, lower plants (lichens and mosses) were contaminated with particles containing cesium-134 (Taylor and Witherspoon 1972). The study site (Fig. 4.1) was the same 10x10 m plot used in the previous fallout studies (see Sections 4.3, 4.4, 4.5, and 4.6). The actual contamination was to individual moss and lichen tussocks (21 for each species). Each contaminated point (~1000 cm^2) received 3.5 µCi in June 1971 for a total of 147 µCi. At weekly

intervals three tussocks of each species were removed from the site for radiological analyses. If all of the original radioisotopes remained at the site, less than 1 μCi (total) would remain after nearly 7.3 half-lives of radiological decay. Knowledgeable persons concerning the site include:

Taylor, F. G., Environmental Sciences Division, ORNL, 615/574-7354
Witherspoon, J. P., Health and Safety Research Division, ORNL, 615/576-2105

4.8 AREA 23: TECHNETIUM-95m CONTAMINATION OF VEGETATION AND SOIL

Interest in the interception and retention of fission products continued with the contamination of fescue grass plots with technetium-95m. The manner of radioisotope application differed from previous retention studies in that the nuclide was applied by liquid spray (Hoffman et al. 1982). The location of the site was in the 0800 area (see Section 3.1). Specifically, the site is located (Fig. 4.1) about 100 m north of the western end of the enclosure referenced in Section 3.1. In the study, the isotope was applied to the vegetation at three different times. The first consisted of fifteen 1 m^2 plots which received 200 μCi each in September of 1978; the next consisted of 10 μCi to each of four 1 m^2 plots in April 1979; and, the final application was 10 μCi to each of three 1 m^2 plots in July of 1979. Because of the lapsed time since contamination (>7 yr) and the short half-life (61 d) of the isotope, the radionuclide is not detectable. The most knowledgeable persons of the site and the isotope applications include:

Garten, C. T., Environmental Sciences Division, ORNL, 615/574-7355
Hoffman, F. O., Environmental Sciences Division, ORNL, 615/576-2118

4.9 AREA 24: TECHNETIUM-95m UPTAKE STUDIES

In the same general location (Fig. 4.1) as described in Section 4.8, three 1 m^2 plots were contaminated with 336 µCi (112 µCi each) of technetium-95m (Garten et al. 1984). The isotope was applied directly to the bare, undisturbed soil. The uptake of the technetium by emerging plants in the field was compared to uptake in a companion, greenhouse study in which soil columns were contaminated and emerging plants analyzed. The field contamination was in late February 1981 with approximately 31 half-lives lapsed. Considering the original microcurie quantities and the radiological decay, the presence of the isotope is below detection. The major contact concerning information of the site and the application of the radionuclide is C. T. Garten, Environmental Sciences Division, ORNL, 615/574-7355.

4.10 AREA 25: TECHNETIUM-95m AND IODINE-131 CONTAMINATION OF PASTURE

In May, July, and September of 1983, iodine-131 was sprayed on a pasture area of 4000 m^2. The total activity was 60 mCi. In September, 10 mCi of technetium-95m was also applied. The purpose of the study was to assess the transfer of these radioisotopes from forage grass to milk in goats (Bondietti and Garten 1984). The study site (Fig. 4.1) was the same enclosure and described in Section 3.1, except that the areas contaminated with technetium and iodine were isolated from the original cesium-137 contaminated pens located within the 2 ha enclosure. Radioactivity of both isotopes is not detectable because

of radiological decay, the technetium-95m having gone through approximately 16 half-lives and the iodine-131, 126 half-lives. Contacts concerning the site and the study are:

Bondietti, E. A., Environmental Sciences Division, ORNL, 615-574-7286
Garten, C. T., Environmental Sciences Division, ORNL, 615/574-7355

4.11 AREA 26: CHROMIUM-51 CONTAMINATION OF GRASS PLOTS

In July of 1976, chromium-51 was applied to grass plots in an aerosol spray (Taylor et al. 1980). A total of twenty-five 1 m^2 plots received 125 μCi of the radionuclide. The purpose of the contamination was to investigate the interception and retention of simulated cooling tower drift on vegetation (pine, yellow-poplar, and fescue grass foliage). The study duration was nine weeks with experimental units collected at one-week intervals. Because the isotope has a short half-life (27.8 d) and over 3600 days have passed since contamination, no radioactivity remains. The location (Fig. 4.1) of the research site was in the 0800 area (see Section 3.1) approximately 0.2 m (3 km) on an azimuth N6°E of the northeast corner of the 0800 area enclosure. The most knowledgeable person concerning the site and the isotope application is F. G. Taylor, Environmental Sciences Division, ORNL, 615/574-7354.

4.12 AREA 27: NEPTUNIUM-237 AND TECHNETIUM-99 CONTAMINATION OF SOIL LYSIMETERS

In June of 1984 eight lysimeter cylinders (28 cm diameter) were contaminated with a total of 64 μCi of neptunium-237 and four

additional cylinders were contaminated with a total of 32 μCi of technetium-99. These tubes were covered at the bottom with a mesh such that a possibility existed for downward radionuclide migration to the surrounding soil. After contamination the cylinders were lowered into a hole so that the contaminated soil and surrounding soil levels were the same. Agricultural species were cultured in the lysimeters, harvested, and analyzed for concentration ratio calculations. At the termination of the study, all cylinders were removed from the soil, encased in plastic, and presently await proper disposal. The site is located (Fig. 4.1) on an azimuth of E75°S, a distance of 0.5 km (0.3 mile) from the intersection of Melton Valley Drive and Lagoon Road. The site is commonly referred to as the plutonium floodplain, southeast of Burial Ground No. 4. Specifics concerning the site and the study can be obtained from C. T. Garten, Environmental Sciences Division, ORNL, 615/574-7355.

4.13 AREA 28: CESIUM-137 CONTAMINATED FOREST FLOOR

A 5x5 m plot near (50 m east) the cesium-tagged forest (see Section 3.2 and Fig. 4.1) was contaminated in April 1964 with an undisclosed quantity of cesium-137 (Waller and Olson 1965). The contamination consisted of a spray and was probably in the order of 1 mCi or less. The purpose of the study was to determine the transfer of the cesium from the litter to successive soil depths following rain-leaching and decay of the litter. Since the initial contamination, 22.2 years (0.74 half-life) have lapsed. If the total radioactivity applied was in the order of 1 mCi, approximately

~0.6 mCi would remain, subject to dilution by leaching and decay of the litter. Few individuals remain at ORNL who have knowledge of this site or the contamination. The major contact is F. G. Taylor, Environmental Sciences Division, ORNL, 615/574-7354.

4.14 AREA 29: CESIUM-137 CONTAMINATION OF FOREST UNDERSTORY SPECIES

On June 27, 1966, 360 µCi of cesium-137 was sprayed on a 10 m^2 plot (Whitford and Olson 1967). The plot is located (Fig. 4.1) approximately 10 m west of the cesium forest study enclosure (see Section 3.2). The study was designed to determine the movement of radiocesium to litter and soil from the understory canopy. Nearly 20 years have passed (0.67 half-life) since the radionuclide was applied. The maximum residual radioactivity today would be 220 µCi. Because of annual leaffall and wind dispersal, decomposition of litter, and runoff, that quantity is greatly reduced. No individual is presently at the laboratory who has first hand knowledge of the contamination.

4.15 AREA 30: CESIUM-137 CONTAMINATED MEADOW

Information in the 1965 Health Physics Division Annual Progress Report (Waller 1965) identifies an 8x8 ft (2.4 x 2.4 m) plot contaminated as a pilot study for what eventually became the runoff soil-erosion study (see Section 3.6). The location of the plot is thought to be an area of detectable radioactive contamination, adjacent (20 m north) to the runoff plot (Fig. 4.1). In the study 5 mCi was applied in the form of a spray to clipped (short meadow) and unclipped (long meadow) grass cover in the plot. The date of isotope application

is not known but thought to be in June of 1964. If that date is correct, nearly 22 years (~0.73 half-life) have lapsed since the field contamination. If the only loss of radioactivity was through radiological decay, ~3 mCi would remain today. Much less would actually be present due to losses from rainout, plant fragmentation, and wind dispersal.

Detailed information of the study and exact plot location is vague. Persons with some knowledge of the site are S. I. Auerbach and F. G. Taylor, Environmental Sciences Division, ORNL, 615/574-7302 and 615/574-7354, respectively.

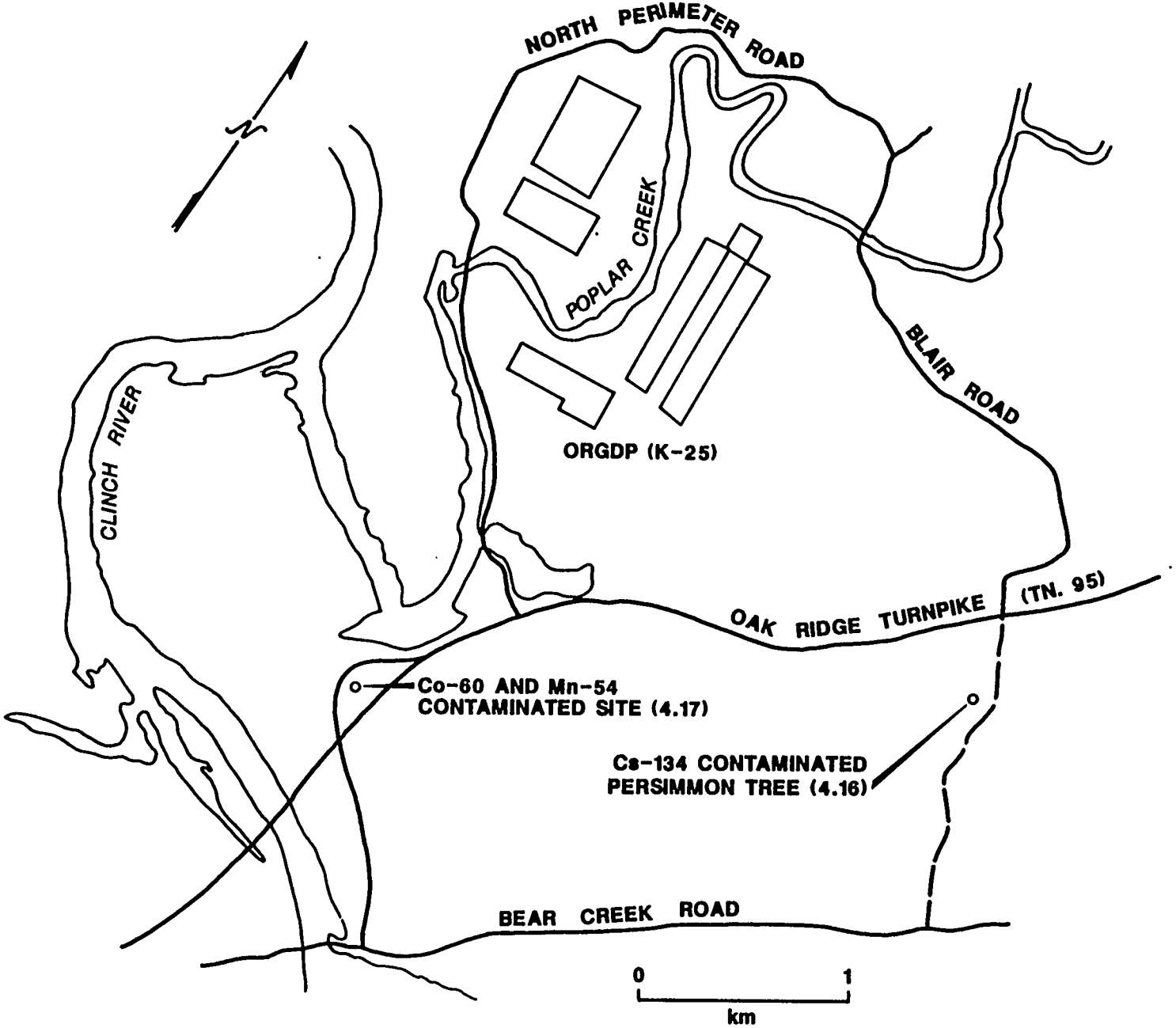
4.16 AREA 31: CESIUM-134 CONTAMINATION OF A PERSIMMON TREE

In June 1970, a single persimmon tree was inoculated with ~2 mCi of cesium-134. The study was designed to determine the transfer of the radionuclide from a contaminated canopy to the understory by rainout. The location (Fig. 4.2) of the tree was in an old field on an azimuth of E73°S from the intersection of the Oak Ridge Turnpike and Blair Road, a distance of 0.3 mile (0.4 km). Duration of the study was approximately 10 weeks. Nearly 16 years or 8 half-lives have passed since the radionuclide was introduced at the site. The maximum possible contaminant remaining would be ~8 μCi, providing there were no losses from site by rain (runoff or leaching), wind, or animal consumption. Persons knowledgeable of the site and the isotope contamination are:

Witherspoon, J. P., Health and Safety Research Division, ORNL, 615/576-2105
Taylor, F. G., Environmental Sciences Division, ORNL, 615/574-7354

Fig. 4.2 Locations of cobalt-60 and manganese-54 injected small mammal study and cesium-134 inoculated persimmon tree. Numbers in parenthesis are keyed to sections in the text.

ORNL-DWG 86-12054



4.17 AREA 32: COBALT-60 AND MANGANESE-54 CONTAMINATED SITE

In July of 1970, 34 pine voles were gamma-irradiated; 17 were injected with 1 μCi of cobalt-60, while the other 17 were injected with manganese-54 (Dunaway et al. 1971). The purpose of the experiment was to determine the effects of near-lethal irradiation and natural environmental factors on the retention and excretion of the two nuclides in members of a field population. Following isotope injection, the pine voles were returned to the area from which they were trapped. The area (Fig. 4.2) was a 1 ha field, 0.52 mile (0.8 km) distance from the intersection of the Oak Ridge Turnpike and the North Perimeter Patrol Road (K-25) on an azimuth of S40°W. Considering that three half-lives have lapsed in the decay of the cobalt-60 contaminant, ~2.6 μCi (total) of activity would remain today providing that all 17 Co-60 contaminated animals remained within the 1 ha field. In the case of manganese-54, 19 half-lives have lapsed in the radiological decay scheme with <16 pCi remaining today. Primary contacts concerning the study and the site include:

Dunaway, P. B., U.S. Department of Energy, ORO, 615/576-0744
Kitchings, J. T., Environmental and Occupational Safety Division, ORNL, 615/574-7363
Story, J. D., Environmental and Occupational Safety Division, ORNL, 615/576-7712

4.18 AREA 33: CARBON-14 MAINTENANCE-RESPIRATION STUDIES

In June 1983, 20 trees were contaminated with ~1 mCi (total) of carbon-14. The isotope was introduced as carbon dioxide. In June 1984, 16 additional trees were contaminated with 2 mCi in the same

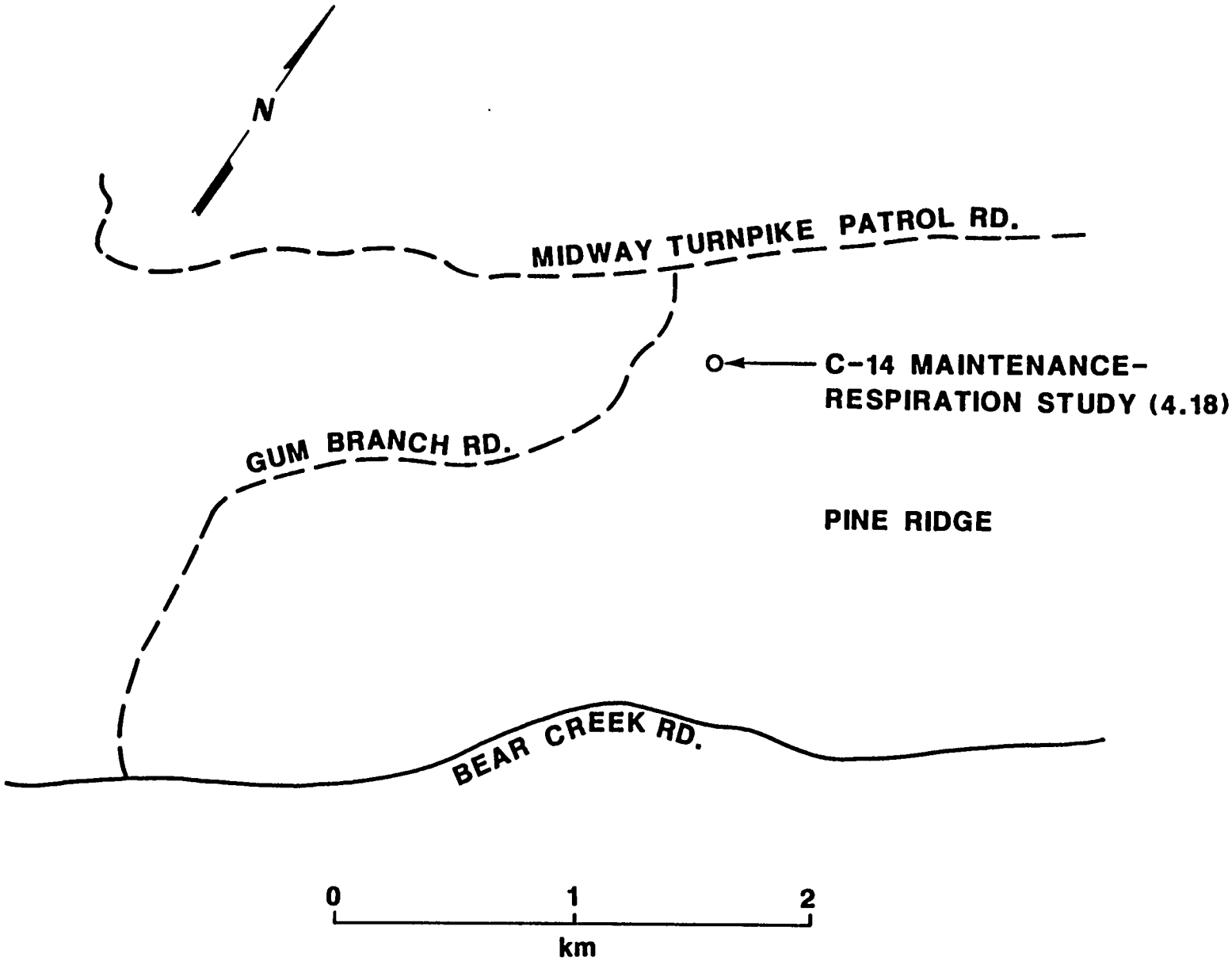
manner. The site is located on the north side of Pine Ridge on an azimuth N22°E from the intersection of Bear Creek Road and Gum Branch Road (Fig. 4.3), a distance of 1.7 miles (2.75 km). Radiocarbon has a long half-life (5479 yr), but because of the weak beta energy (0.156 MeV) it does not constitute a hazard. As the plant materials decomposed, a portion of the radiocarbon was dispersed into the atmosphere as carbon dioxide. Persons with knowledge of the site and the isotope use are S. B. McLaughlin and N. T. Edwards, Environmental Sciences Division, ORNL, 615/574-7358 and 615/576-4868, respectively.

4.19 AREA 34: CARBON-14 SUCROSE INOCULATION OF OAK AND PINE TREES

Two studies were conducted at this site, both designed to determine the spatial and temporal pattern of carbohydrate movement and respiratory use by the root systems of trees (McLaughlin et al. 1977 and Edwards et al. 1977). In October of 1976, a single white oak was inoculated with 5 mCi of carbon-14 sucrose dissolved in water. In October of 1972, three trees received 1.65 mCi each of the carbon-14 sucrose. The location is near the cesium-134 contaminated forest (Section 3.2). It is about 1.8 miles (2.9 km) on an azimuth E66°S of the intersection of Bethel Valley Road and Melton Valley Access Road (Fig. 4.1). Major contacts for details of the contamination are S. B. McLaughlin and N. T. Edwards, both of the Environmental Sciences Division, ORNL. Their respective telephone numbers are 615/574-7358 and 615/576-4868.

Fig. 4.3 Location of carbon-14 maintenance-respiration study. Number in parenthesis is keyed to the section in the text.

ORNL-DWG 86-12053



4.20 AREA 35: CARBON-14 ALLOCATION IN WHITE OAK TREES

Two studies were conducted at this area between 1972 and 1977 (McLaughlin and McConathy 1979, McLaughlin et al. 1979, and McLaughlin et al. 1980). In one, two trees were contaminated by foliar tagging using carbon-14. Each tree received 50 µCi. In the other study, two trees received 5 µCi each of carbon-14 sucrose. The purpose of the first was to determine seasonal changes in photosynthate translocation and allocation by following the rates of movement of the carbon-14 from labelled foliage. The second was to follow the rate and efficiency of the utilization of food reserves. The location of the area (Fig. 4.1) is on an azimuth of N25°E from the intersection of Bethel Valley Road and Melton Valley Access Road, a distance of 1.9 miles (3.1 km). The site occupies approximately 200 m^2. Information of the studies are available from S. B. McLaughlin and R. K. McConathy, Environmental Sciences Division, ORNL, whose telephone numbers are 615/574-7358 and 615/574-7307, respectively.

4.21 AREA 36: CARBON-14 ALLOCATION AND GROWTH OF WHITE PINE TREES

Nine white pine trees were labelled with carbon-14 between June and November of 1979. A total of 360 µCi was introduced to foliage at four times during the growing season (McLaughlin et al. 1980). The location of the pine stand is south of Bethel Valley Road near the Walker Branch embayment of Melton Hill Lake. The distance is approximately 1.8 miles (2.9 km) on an azimuth N55°E of the intersection of Bethel Valley Road and Melton Valley Access Road (Fig. 4.1). The purpose of the study was to determine the rate and

cause of declining vigor of oxidant-stressed trees by following the fate of photosynthetically fixed carbon-14. Details of the contamination can be obtained from S. B. McLaughlin and R. K. McConathy, Environmental Sciences Division, ORNL. Their respective telephone numbers are 615/574-7358 and 615/574-7307.

4.22 AREA 37: CARBON-14 EFFLUX IN YELLOW-POPLAR STAND

In July of 1976, three yellow-poplar trees were contaminated with carbon-14 sucrose. A total of 3 mCi was introduced into the trees. Stems of the trees were girdled to inhibit upward translocation. The design of the study was to measure carbon dioxide efflux of the roots through the soil. The site is located on the north slope of Copper Ridge, a distance of 1.8 miles (2.9 km) from the intersection of Bethel Valley Road and Melton Valley Access Road (Fig. 4.1). The directional azimuth is E76°S. Details of the study can be obtained from N. T. Edwards, Environmental Sciences Division, ORNL, 615/576-4868. Results of the study were not published.

4.23 AREA 38: CARBON-14 ALLOCATION IN WOODY BIOMASS PLANTATION SPECIES

Several species of trees in the Woody Biomass Plantation were contaminated with a total of 1 mCi of carbon-14 dioxide. The radiocarbon was introduced (June 1984) as a gas to foliage contained in exposure cuvettes. The purpose of the study was to determine photosynthate allocation. The plantation is located south of Building 1503 (Figure 4.1). Following the study, all plant materials were removed and disposed. Results have not been published to date.

Information concerning specifics of the study can be obtained from S. B. McLaughlin, Environmental Sciences Division, ORNL, 615/574-7358.

5. PRELIMINARY WALK-OVER RADIOLOGICAL SURVEY

In May of 1986, a preliminary walk-over radiological survey (radiation exposure) was conducted of the areas identified in the Compendium of Remedial Action Sites (Grimsby 1986). Personnel of the Health and Safety Research Division conducted the survey to confirm the location of the research areas, determine the approximate area of contamination, and to prioritize areas for a detailed radiological survey (soil, water, and vegetation analyses). The survey was made with Geiger-Mueller type beta-gamma survey meters and NaI gamma scintillation detectors. The results of that survey are summarized in Table 5.1. Among the 14 areas visited, measurements were made at ten. No radioactivity was detectable at six areas, indicating that the isotope had either decayed or that it was so widely distributed that it was below detection. As suspected, the cesium-137 contaminated pens (Section 3.1) had the highest exposure (8-10 mR/h). The exposure rates in the cesium-137 contaminated forest (Section 3.2) ranged from 1 to 5 mR/h, with the highest rates near the tower. Radiation was also detectable (1 mR/h) at the three pits of the cesium-137 runoff study area (Section 3.6). The cesium-137 and cobalt-60 small mammal forest site (Section 3.15) is of considerable area (~4 ha combined) and revealed hot spots (1 to 2 mR/h) at the feeding stations. No survey was made of the additional areas identified in Section 4.

Table 5.1 Preliminary walk-over survey of ORNL Remedial Action Sites: Environmental research areas

Area	Description	Exposure
1	Cs-137 field	8-10 mR/h
2	Cs-137 forest	1-5 mR/h
3	Ca-45 tagged trees	nd
4	Ca-45 tagged soil	nd
5	Na-22 tagged soil	nm
6	Cs-137 runoff	1 mR/h
7	Cs-137 tagged litter bags	nd
8	Hg-197 contaminated stream	nm
9	Cs-134 tagged tree	nd
10	Ca-45 tagged trees	nd
11	Cs-137 and Fe-59 contaminated field, McNew Hollow	nd
13	Hg-203 tagged stream	nm
14	Tritium-tagged trees	nm
15	Cs-137 and Co-60 contaminated forest	1-2 mR/h

nd = not detected, nm = not measured.

6. DISCUSSION

Areas 1, 2, and 6 utilized a long-life isotope (Cs-137, 30 years), presently contain a significant fraction of the initial radioactivity, and involved significant land areas. Consequently, these sites constitute the most significant areas of contamination among the environmental research areas and were therefore subjected to evaluation by the modified Hazard Ranking System (mHRS)(Hawley and Napier 1985). The mHRS is an alternative ranking system, designed to remove the bias against radioactive waste sites inherent in the Hazard Ranking System as described in 40 CFR-300 Appendix A. Scores are assigned to waste sites after considering: (1) if releases of hazardous substances are known to have occurred or are likely to occur; (2) the toxicity, persistence, and amount of hazardous waste present; and (3) the number of people and the existence of sensitive environments potentially at risk because of the site. The composite score of the evaluation (S_m) reflects the potential for harm to humans or the environment from migration of a hazardous substance from a site along ground water, surface water, or atmospheric pathways. Sites with the high scores are subject to review by the Environmental Protection Agency according to the National Priorities List.

The cesium-137 contaminated field (Area 1) is singly the most significant environmental research area. The migration scores, S_m, as determined by the mHRS were 0.2 rad score and 2.3 chemical score (Nix 1986). The chemical score is dominated by the persistence of cesium in the environment. Using the conservative chemical score the site is not a candidate for inclusion on the National Priorities List.

Because Sites 2 and 6 contain the same isotope, with similar pathways, the migration scores (S_m) are the same as Site 1. Collectively, the three sites have a relatively low potential for harmful effects to humans or the environment.

An attempt has been made to identify additional research areas which were omitted from the Compendium of Remedial Action Program Sites (Grimsby 1986). A total of 23 additional areas are discussed. The areas are identified by sequential numbers and should not be confused with numerical designations provided by the ERFU program. Six of the areas utilized carbon-14, a low energy (0.156 MeV) beta source, to study physiological processes in trees. Most of the contaminated materials were removed at the completion of the studies. Trees that were left in the forests have been subjected to biological decay, thereby releasing a portion of isotope to the atmosphere as $^{14}CO_2$. One area utilized radionuclides (Tc-99 and Np-237) of extremely long half-lives in closed systems which have since been contained and await disposal. The relatively short half-life isotopes (Cs-134, Zn-75, Rb-86, Cr-51, Co-60, and Mn-54) utilized would not constitute a problem as radiological decay has reduced the residual radioactivity to microcurie quantities or less. The three additional cesium-137 areas are adjacent to areas already discussed (Sections 3.2 and 3.6) and should be included with the larger area for remedial action consideration.

7. RECOMMENDATIONS

The design of this report is not to make a final decision as to which environmental research areas qualify as Remedial Action Program Sites. Rather, it is intended to provide a basis for recommending those sites which require remedial action in order to place the Oak Ridge National Laboratory in compliance with applicable environmental regulations.

The radioactive contamination at several environmental research areas was either moved to other sites for execution of the designed study or the contaminated materials were removed at the completion of the study. Therefore, it is recommended that Areas 3, 4, 5, 7, 11, and 14 be removed from consideration as Remedial Action Program Sites. Areas 1, 2, and 6 are subject to remedial action consideration because they utilized a relatively long-life isotope, Cs-137 (30 years), contain a significant fraction of the initial radioactivity after several years of decay, and involved significant land areas. The remainder of the areas (8, 9, 10, and 13) should be removed from potential remedial action consideration because the quantity of radioactivity is now below radiological detection after many lapsed half-lives. Although the modified Hazard Ranking System resulted in a low score for potential human health and environmental effects, it is recommended that risk analyses and radiological surveys be performed for Areas 1, 2, and 6 because of the residual radioactivity. Such an assessment may provide rationale for a field survey to confirm or deny the ranking analysis. Area 15 should be surveyed in detail by a health physics field unit to determine the total number and area of "hot spots."

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ORNL/RAP/LTR-86/21

DATE: February 28, 1986

SUBJECT: Inventory of ORNL Remedial Action Sites:
7. Hazardous Waste Sites

TO: T. E. Myrick

FROM: R. E. Saylor

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Inventory of ORNL Remedial Action Sites: 7. Hazardous Waste Sites*

R. E. Saylor

February 28, 1986

Environmental Sciences Division
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37831

Operated by

Martin Marietta Energy Systems, Inc.
for the
U.S. Department of Energy

*Based on work performed at Oak Ridge National Laboratory, operated by Martin Marietta Energy Systems, Inc., under Contract No. DE-AC05-84OR21400 with the U.S. Department of Enrgy.

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1.0 OBJECTIVES

The primary purpose of the remedial action task is to provide corrective measures and facility decommissioning necessary to place the Oak Ridge National Laboratory (ORNL) in compliance with applicable environmental and health regulations for radioactive and hazardous chemical wastes. Individual sites will be decontaminated in accordance with priorities prescribed to control the potential for human exposure and relative magnitude of ecological impacts. Remedial action is a component of Laboratory's Environmental Restoration and Facilities Upgrade Program (ERFU). In the program, remedial actions will be conducted at locations where past management practices or research activities have resulted in contamination of facilities or the environment that necessitate decommissioning or corrective actions. Remedial actions are divided into two activities: (1) Site Decommissioning Program (Surplus Contaminated Facility Program-SCFP, and Surplus Facility Management Program-SFMP) and (2) Site Corrective Measures Program (SCMP).

The Site Decommissioning Program is designed for the management of surplus facilities (primarily structures), contaminated with radioactive materials and hazardous chemicals, from the end of their operating life until final facility disposition. The Site Corrective Measures Program is designed for the management of inactive sites (process ponds, spill sites, environmental research areas) contaminated with radioactive materials and hazardous chemical wastes, leading to the eventual stabilization of these sites in accordance with environmental regulations.

Remedial Action Program sites at ORNL comprise 13 distinct areas among the several programmatic categories: (1) Site Corrective Measures

Program-SCMP, (2) Surplus Contaminated Facility Program-SCFP, and (3) Surplus Facility Management Program-SFMP. In a series of reports, an inventory of each area is presented in a condensed summary. The summary is the product of pertinent information abstracted from readily available literature and knowledgeable individuals on known contaminated sites requiring remedial action. The information includes site location and description, history, contaminants and their characteristics, extent of contamination, regulatory status, geological, hydrological, and ecological data, list of major studies, and knowledgeable individuals contacted. The subject of this report is several hazardous waste sites (Table 1-1). These sites include four areas contaminated with mercury, PCB containing and contaminated equipment, ORNL storage areas located at the Y-12 Plant and the contaminated oil storage tank at the new Hydrofracture Facility (Building 7860).

Table 1-1. Remedial Action Program Hazardous Waste Sites (Category 7)

Hazardous Waste Sites[a]	Location	Program Category	Regulatory Status
Mercury Contaminated Soil	4501	SCMP	CERCLA
Mercury Contaminated Soil	4508	"	"
Mercury Contaminated Soil	3503	"	"
Mercury Contaminated Soil	3592	"	"
Oil Storage Tank	New Hydro Fac.	SCFP	RCRA(?)
ORNL at Y-12			
PCB Transformers	9204-3	SCFP	TSCA

[a]"Compendium of ORNL Remedial Action Program Sites," a January 14, 1986 memo from J. H. Grimsby to L. E. Stratton, contains complete Remedial Action Program site listing.

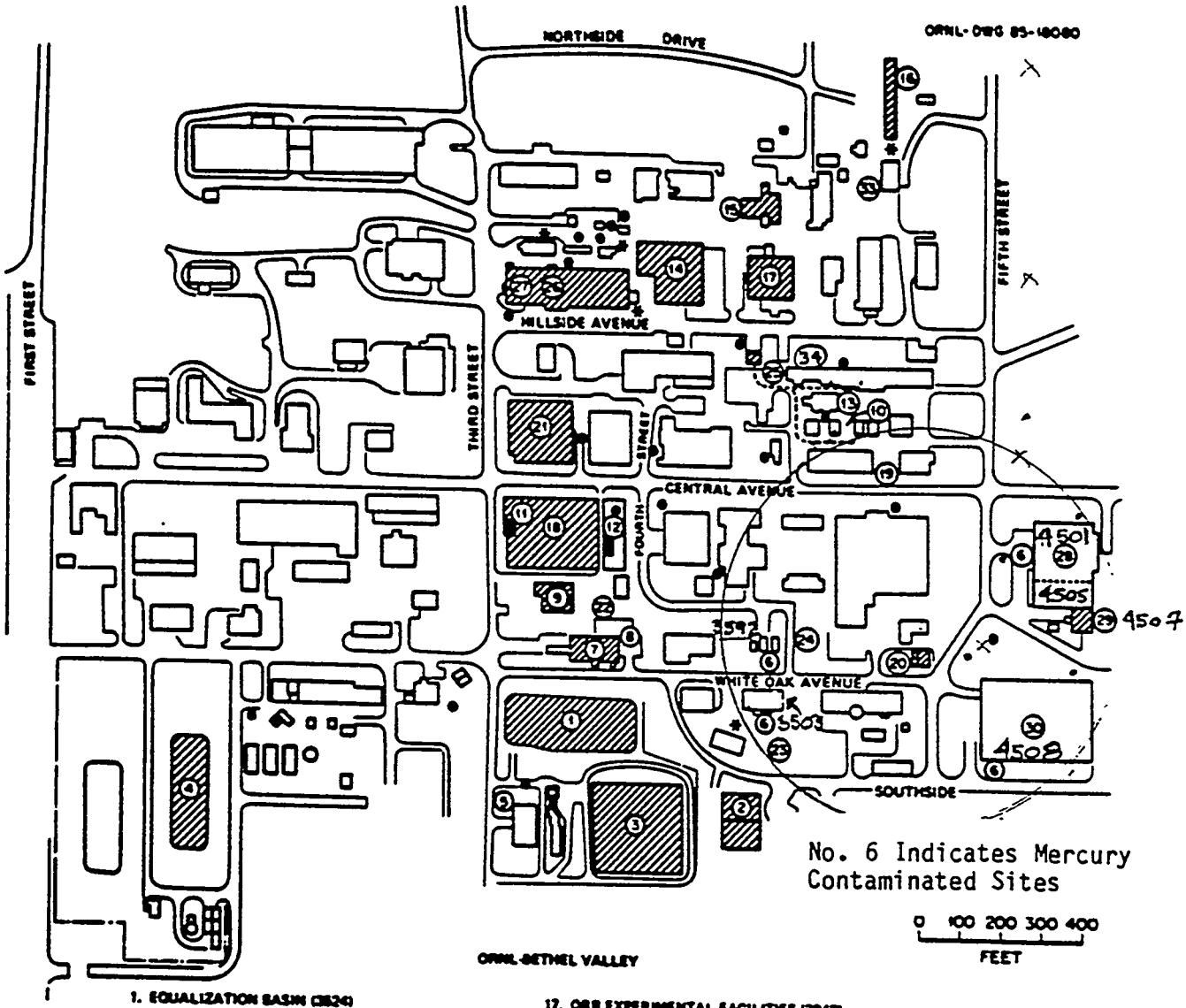
2.0 MERCURY CONTAMINATED SOILS

During the '50s and '60s, ORNL performed experimental work which required the use of large quantities of mercury. At the time of operations, unknown amounts of mercury spills took place. The location of known and suspected mercury spill sites is given in Figure 2-1. These sites include Buildings 4501, 4508, 3592, and 3503. Since these sites were contaminated with mercury prior to November 1980, they are regulated by the Comprehensive Environmental Response, Compensation and Liability Act of 1980 (CERCLA). The implementation of CERCLA for Department of Energy (DOE) facilities is given in DOE Order 5480.14. DOE Order 5480.14 Phase I, Installation Assessment, has been completed for these sites and will be submitted prior to the April 26, 1986 due date.

2.1 BACKGROUND

In 1983, ORNL's Department of Environmental Management (DEM), analyzed soil samples for mercury for three of the four listed mercury-contaminated soil sites. This mercury soil sampling is partially documented in a June 9, 1983 letter from T. W. Oakes (DEM) to S. F. Wing (Environmental Protection Branch of DOE Oak Ridge Operations). Detailed sampling procedure documentation appears to be unavailable. The data from this study and other DEM file documents are presented in Sections 2.2 - 2.5 of this report.

Area background mercury concentrations are reported in "Environmental Monitoring Report 1983 LY/UB-19." Soil testing was performed by Oak Ridge Associated Universities (ORAU) for sites at Black Oak Ridge, Freels Bend,



1. EQUALIZATION BASIN (3524)
2. 180 PROCESS WASTE PONDS (3538)(3540)
3. WASTE HOLDING BASIN (3513)
4. SEWAGE LAGOON, EAST BASIN
5. ABANDONED WASTE BASIN (FILLED-IN) (3512)
6. MERCURY CONTAMINATED SOIL
7. FISSION PRODUCT DEVELOPMENT LAB (3517)
8. FPDL FILTER PIT/LLW TRANSFER LINE (3517)
9. METAL RECOVERY FACILITY (3505)
10. STORAGE GARDEN 3033
11. WASTE EVAPORATOR FACILITY (3508)
12. FISSION PRODUCT PILOT PLANT (3515)
13. COBALT-60 STORAGE GARDEN (3029)
14. ORNL GRAPHITE REACTOR (3001)
15. LOW INTENSITY TEST REACTOR (3005)
16. ORR WATER-TO-AIR HEAT EXCHANGER (3087)
17. ORR EXPERIMENTAL FACILITIES (3042)
18. GUNITE WASTE TANKS W5-W10 (3507)
19. WASTE TANK WC-1
20. WASTE TANKS WC-15, WC-17
21. WASTE TANKS W1-W4, W13-W15, W1-A (3023)
22. WASTE TANK W-11
23. WASTE TANKS TH1-TH3
24. WASTE TANK TH4
25. ISOTOPES DUCTWORK/3110 FILTER HOUSE
26. RADIOCHEMICAL PROCESSING LAB (3019)
27. HIGH LEVEL RADIOCHEMICAL ANALYTICAL LAB (3019B)
28. HIGH LEVEL RADIOCHEMICAL LAB (4501-CELL D)
29. HIGH LEVEL CHEMICAL DEVELOPMENT LAB (4507)
30. REMOTE COATING FURNACE LOOP/CERAMIC PROCESSING LAB (4508)
31. LLW LINE LEAK SITES ●
32. OTHER CONTAMINATED SITES ✱
33. LITR POND (3075)
34. STRONTIUM 90 POWER GENERATORS

Fig. 2-1. ORNL Remedial Action Sites, Including Mercury Contaminated Soil Sites (No. 6).

Raccoon Creek and Gallaher Quarry. Nine soil samples ranged from .004 to .2 ppm mercury.

The Tennessee Division of Solid Waste Management, "Hazardous Substance Guidelines" (May 1985) defines the maximum limit of mercury in water to be 0.002 ppm (mg/l) and the maximum concentration of mercury in soil to be 0.2 ppm (mg/Kg). These values are currently used by the Division's Super Fund Section and by Permits and Quality Assurance as guidelines for clean closure considerations. These values along with background concentrations will be used in this report as a guideline for estimating the extent of mercury contaminations.

Extensive research into mercury spills that have occurred at the Y-12 Plant has shown that elemental mercury is not transported in solution within groundwater. If a mercury contaminated water sample is filtered, the mercury concentration almost goes to zero, thus indicating that the mercury is attached to soil particles. Mercury tends to adhere to soil particles because of its high surface tension and molecular charge.

There currently is no site-specific geological or hydrogeological data available for the mercury contaminated sites. However, current characterization plans have led to the installation of several piezometers and a series of boreholes in this and surrounding areas. These monitoring sites will allow for the development of site-specific geology, depth to groundwater table, water table gradient and groundwater flow direction. A report addressing site-specific geology and hydrogeology of this area (R. H. Ketelle and D. D. Huff) is expected by the end of FY 86. The mercury contaminated areas discussed below are located (Fig. 2-1) within the main laboratory complex, which has a standard industrial setting. It is unlikely

that the ecology of the area would be disturbed significantly if remedial actions were necessary for these areas. Therefore, no additional ecological data are needed for this area.

2.2 BUILDING 4501

A small pilot plant supporting Y-12 programs, located in Building 4501, for the OREX lithium separation process involved ton quantities of mercury. This operation took place from April until November, 1954 in support of Y-12's thermonuclear weapons program. At the time of operations, unknown amounts of mercury spills took place. Although these spills were cleaned up, it is evident that considerable quantities escaped through cracks in the concrete floor and reached the environment. However, there is no accurate measure of mercury loss at this site.

Soil samples were collected by ORNL's Department of Environmental Management around the 4501 building complex in 1983 and analyzed for mercury concentration. Table 2-1 gives a summary of the soil mercury concentrations and Figure 2-2 shows the location of the sampling sites. Soil samples mercury concentrations ranged from 0.05 ppm to 4.4 ppm with the exception of a site located at Fifth Creek which had a concentration of 465 ppm. This high value may be due to mercury-contaminated water discharged from a sump in Building 4501 into Fifth Creek. These values are generally higher than background mercury soil level concentrations. The mercury concentrations listed in Table 2-1 are also generally higher than the Tennessee Division of Solid Waste Management's guideline maximum limit of 0.2 ppm.

Table 2-1. Summary of Mercury Analyses in Soil Samples - Buildings 4501, 4505, and 4507

Sample Area	Mercury Concentration (ppm dry wt.)
4501-1[a,b]	0.29
4501-2	0.25
4501-3	0.32
4501-4	0.23
4501-5	0.70
4501-8	0.44
4501-10	0.29
4501-13	4.4
4501-15	2.3
4501-16	1.1
4501-17	0.12
4501-18	0.38
4501-19	1.4
4501-20	0.05
4501-22	0.33
4501-24	0.70
4501-26	13
4501-27	3.8
4501-29	1.4
Fifth Creek-31	465
Fifth Creek-32	3.5

[a]Figure 2-2 shows location of sampling area.
[b]Sampling was performed by ORNL's Department of Environmental Management in 1983, sampling method undocumented.

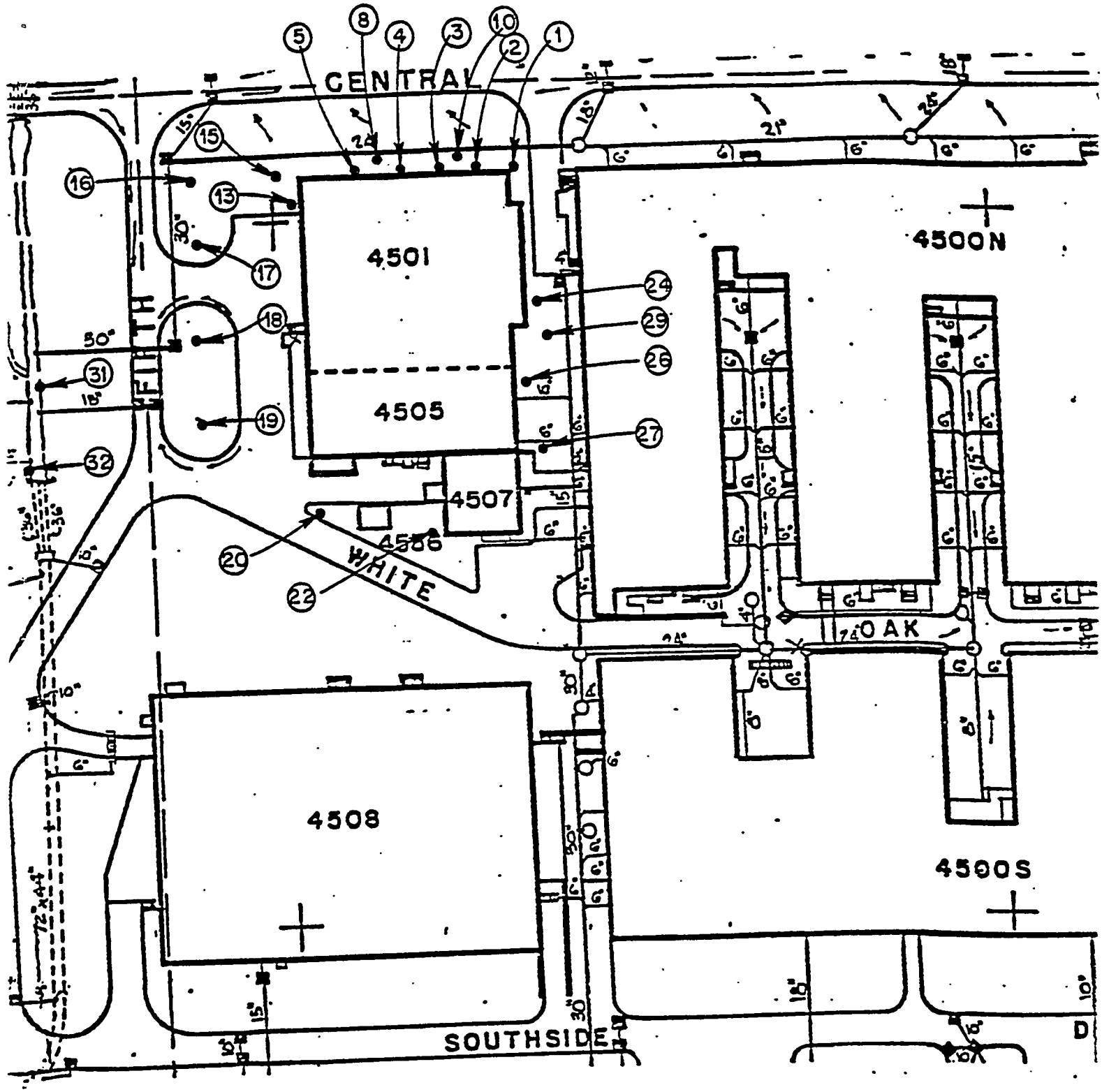


Fig. 2-2. Location of Soil Sampling Areas for Buildings 4501, 4505, and 4507.

2.3 BUILDING 3503

Building 3503 was used as an R&D pilot plant in support of the PUREX spent fuel reprocessing program. Some mercury (unknown quantity) used in connection with tests on a uranium-recovery process was spilled in the early 1960s. Mercury soil concentration around Building 3503 is presented in Table 2-2. The location of Building 3503 is shown in Figure 2.3; however, sampling sites are not specified on map. The soil samples studied near this building ranged from 0.8 to 25 parts per million. Mercury soil concentrations are above both the area mercury background level and the State of Tennessee's maximum mercury soil level concentrations guideline.

2.4 BUILDING 3592

In 1954, component development work was done in Building 3592 in conjunction with the OREX process. During this time, large quantities of mercury were used at this facility with some spillage. One large spill is known to have seeped through the building into the ground. There is no accurate measure of mercury loss at Building 3592, but operating personnel have estimated a total of approximately 2,000 to 3,000 pounds lost due to spills and leakage.

Table 2-2 presents mercury soil concentrations for three locations around Building 3592. Figure 2-3 shows the location of 3592; however, sampling site locations and methodology are unclear. The soil samples ranged from 4.1 to 320 ppm mercury, these values are significantly higher than both

Table 2-2. Summary of Mercury Analyses in Soil Samples - Buildings 3503 and 3593

Sample Location[a,b]	Mercury Concentration (ppm dry wt.)
South of Bldg. 3503	
Storage Area 1	25
Storage Area 2	3.2
Storage Area 3	6.5
Storage Area 4	0.8
Building 3592	
Sample 1	4.1
Sample 2	8.8
Sample 3	320

[a]There is no record of exact location of sampling areas, see Figure 2-3 for building locations.
[b]Sampling was performed by ORNL's Department of Environmental Management in 1983, sampling method undocumented.

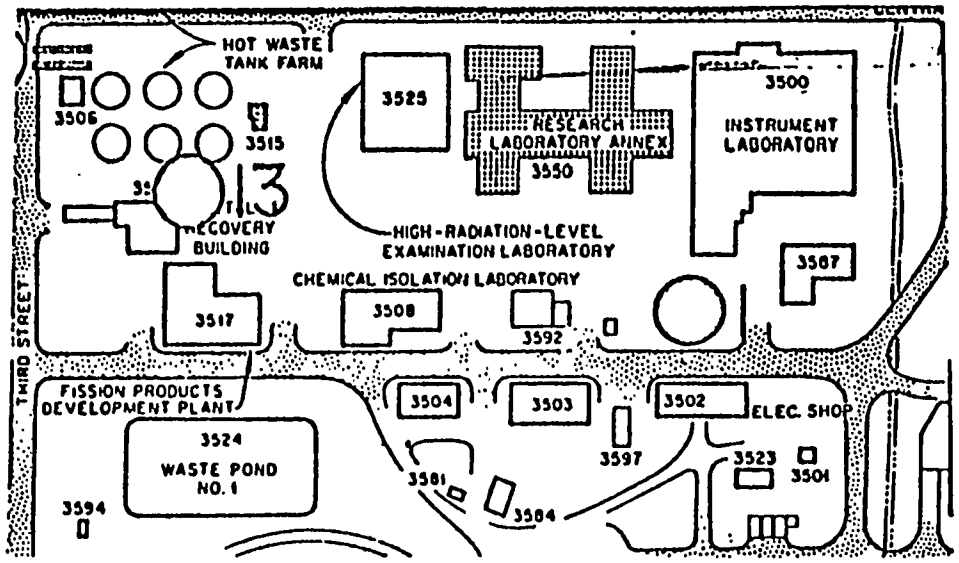


Fig. 2-3. Map Showing Location of Buildings 3592 and 3503.

mercury background concentrations and the State of Tennessee guideline for maximum mercury soil level concentration in CERCLA site cleanups.

2.5 BUILDING 4508

Discussions with individuals involved with research activities that took place in Building 4508 failed to reveal any activities that used large quantities of mercury. Current and past inventories of mercury, used within this building, appear to be in quantities of approximately 100 pounds. Discussions with personnel failed to establish any known mercury spills in or around the building. It is possible that this site has been incorrectly listed as a mercury spill site.

ORNL's Department of Environmental Management 1983 soil sampling did not include sampling around Building 4508. A sampling program needs to be instituted around this building to determine if the soil is contaminated with mercury.

2.6 SUMMARY AND SUGGESTIONS

Past mercury contaminated soil cleanup activities in the Oak Ridge community can provide insight into levels of effort that may be needed to cleanup ORNL mercury contaminated soils. If a site had less than 100 ppm of mercury in the soil the site was left alone. A site that had between 100 to 500 ppm of mercury was cleaned-up if there was an injestion or inhalation pathway. A site that contained greater than 500 ppm of mercury was automatically cleaned-up. Mercury contaminated soils in the Oak Ridge

community were considered clean at 10 ppm. It appears that the Y-12 Plant is developing a similar strategy to address their mercury contaminated sites. It should be noted that site cleanup standards would be much more stringent for a mercury contaminated site that could end up contaminating an aquatic system.

These cleanup activities indicate that site cleanup standards may be negotiable with the State of Tennessee and the Environmental Protection Agency. It would therefore be advisable to contact state and federal agencies at the earliest possible time.

An environmental assessment will have to be performed at these sites after the completion of initial site characterization. This assessment will provide useful in site cleanup negotiations with regulators and also provide input into future site cleanup cost benefit analyses.

As has been stated, documentation of mercury spills at ORNL is non existent. Interviewing of personnel has substantiated that large (undefined) quantities of mercury were spilled. However, it is believed that further interviewing of personnel would not substantially aid in quantifying the magnitude of mercury contamination.

Information dealing with the operations of research activities carried out at Buildings 4501, 3592, and 3503, along with contaminated soil samples located around these building, definitely indicate that these specific sites must be considered mercury contaminated. Building 4508 may not be a spill site; however this site must be characterized before the soil can be proven not to be mercury contaminated. A near-term project might be the institution of soil sampling around Building 4508. It is possible that a

spill has not occurred at this site, and that in the near future this site might be removed from the official remedial actions list.

As has been mentioned in Section 2.1, geological and hydrological characterization of this site is underway and will be completed by the end of FY86. This information will generate site-specific geological information, depth to groundwater, groundwater gradient and groundwater flow data. A deep borehole, along with other boleholes in this are, will provide information necessary to perform geologic characterization of this site.

Further site characterization data will be necessary for this area to complete an environmental impact appraisal. This characterization work should include mercury vapor detection in soil within the spill site area. This work should lead into an intensive three dimensional soil sampling project. To address the extent and magnitude of ground water contamination, a series of groundwater monitoring wells should be installed. At this time site-specific groundwater monitoring plans have not been developed for this area.

Contact the following persons for information on groundwater monitoring wells:

Steve Stowe, Operations Division, 615/574-7428
D. D. Huff, Environmental Sciences Division, 615/574-7859
R. H. Ketelle, Energy Division, 615/574-5762
J. W. Boegly, Environmental Sciences Division, 615/574-7858

Environmental assessment activities personnel may find the storm water characterization report, which is due at the end of FY 86 of interest. This

report is being developed by the Environmental Upgrade Section of ORNL's Operations Division. The report will characterize water in the stormwater sewers lines, White Oak Creek, and Fifth Creek. A review of this report, along with other planned site characterization should allow for better understanding of the source of mercury contamination and the migration of mercury throughout the White Oak Creek Watershed. Contact the following person for more information on this project:

J. B. Berry, Operations Division, 615/574-6907.

3.0 SURPLUS PCB EQUIPMENT WAITING DISPOSAL: ORNL FACILITIES AT THE Y-12 PLANT

Six locations (Table 3-1) at the Y-12 Plant have been identified which house a variety of surplus ORNL equipment (transformers, capacitors, and PCB containers) that are either contaminated with PCBs or contain PCB contaminated oils. The handling of PCBs and PCB items in use, storage for reuse, and storage for disposal are regulated by the Toxic Substances Control Act (TSCA). Three important requirements of this law are: (1) a provision for an annual inventory of PCBs and PCB items in use; (2) the storage for reuse and storage for disposal of PCBs and PCB contaminated materials in approved facilities and; (3) that PCBs must be destroyed in an appropriate manner within one year of being placed in storage for disposal (see 40 CFR 761.180 for specifics).

The report "Internal Environmental Audit Report of the Use, Storage, and Disposal of PCB's at the Y-12 Plant," issued September 5, 1985, sited noncompliances of the use, storage and disposal of PCBs by ORNL facilities at the Y-12 Plant. This audit was conducted the week of June 5-10, 1985. The audit team included a representative from Y-12's Electrical Maintenance Department of Plant TSCA Coordinator, Plant Spill Coordinator, Plant Internal Environmental Audit Coordinator, and a representative from the Waste Transportation, Storage, and Disposal Department (WTSD). Eleven recommendations were made by the audit team as actions that must be taken to correct the noncompliances cited in the audit.

In response to the internal audit report, ORNL's Department of Environmental Management (DEM) has proposed a plan of action to comply with

Table 3-1. Surplus PCB Equipment - ORNL at Y-12

Location	Equipment	Estimated Concentration
SW of 9204-3	37 PCB contaminated transformer carcasses	50-500 ppm
Basement 9204-3	22 PCB contaminated transformers	50-500 ppm
East of 9204-1	13 PCB contaminated transformers	50-500 ppm
SY200 Yard	5-2200 gal. transformers & 60 cooling racks	<2-29 ppm
Cyclotron cubicles 9201-2	143 PCB transformers 360 capacitors (∿ 3000 gal. oil)	>500 ppm
9201-2	Cyclotron Z oil	∿ 27 ppm

[a]It has been decided that Y-12 will dispose of PCB items stored in 9723-3

stipulated recommendations. The DEM report, "Internal PCB Audit Response Report ORNL Facilities at Y-12," January 1986 (no report number), outlines the plan of action that DEM will be taking for each recommendation that was made by the audit team. Another pertinent documentation of PCB sites is "Immediate Repairs and Improvements of ORNL Surplus Contaminated Facilities," RAP86-4.

Appendix A of the internal audit response report presents a complete inventory of the PCB containing and contaminated equipment. This inventory includes: ID number, location, oil holding capacity, concentration of PCBs in oil, and division responsible for the equipment. A summary of this information and a map of the Y-12 PCB sites are given in Table 3-1 and Figure 3-1 respectively.

Immediate improvement of the PCB contaminated equipment at Y-12 is scheduled for FY 86. The sites have been prioritized, scoping estimates made and initial contacts made with disposal agencies. In general this program involves the incineration of PCB contaminated oils in approved TSCA incinerators, and the triple rinsing and burial of the containers (equipment) in an approved burial ground. The scope of the work and funding available indicate that the majority of the sites can be cleared in fiscal year 86.

The PCB storage sites are located over the Maynardville limestone, a very cavernous formation, which is characterized by solution channel flow. Soils tend to be silty clays of variable thickness. There is significant interaction between groundwater and surface water in this area. Groundwater flow has been characterized for many of these areas, but may not be comprehensive enough for all sites. The PCB storage areas are located

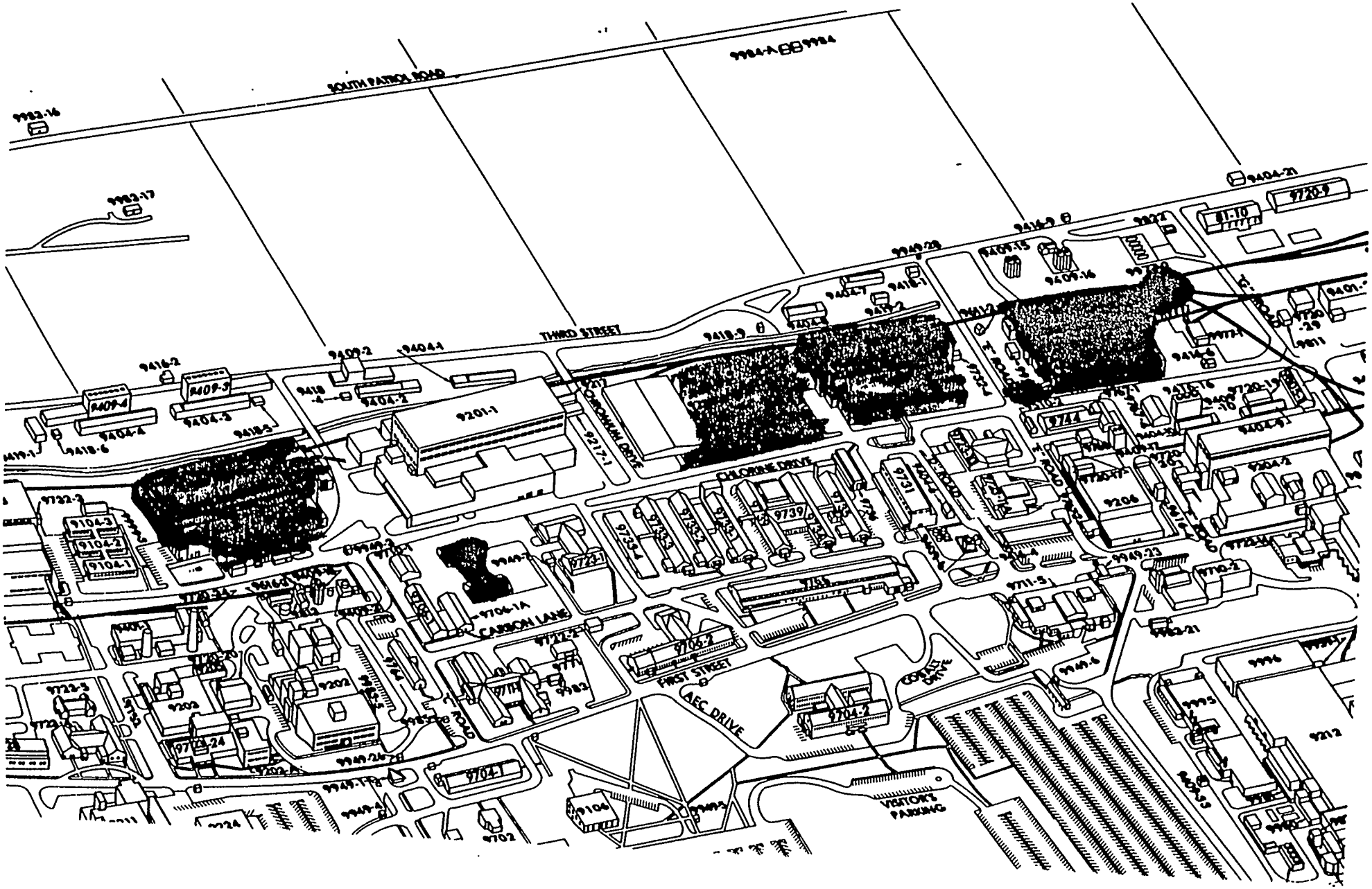


Fig. 3-1. Location of ORNL PCB Storage Sites at Y-12.

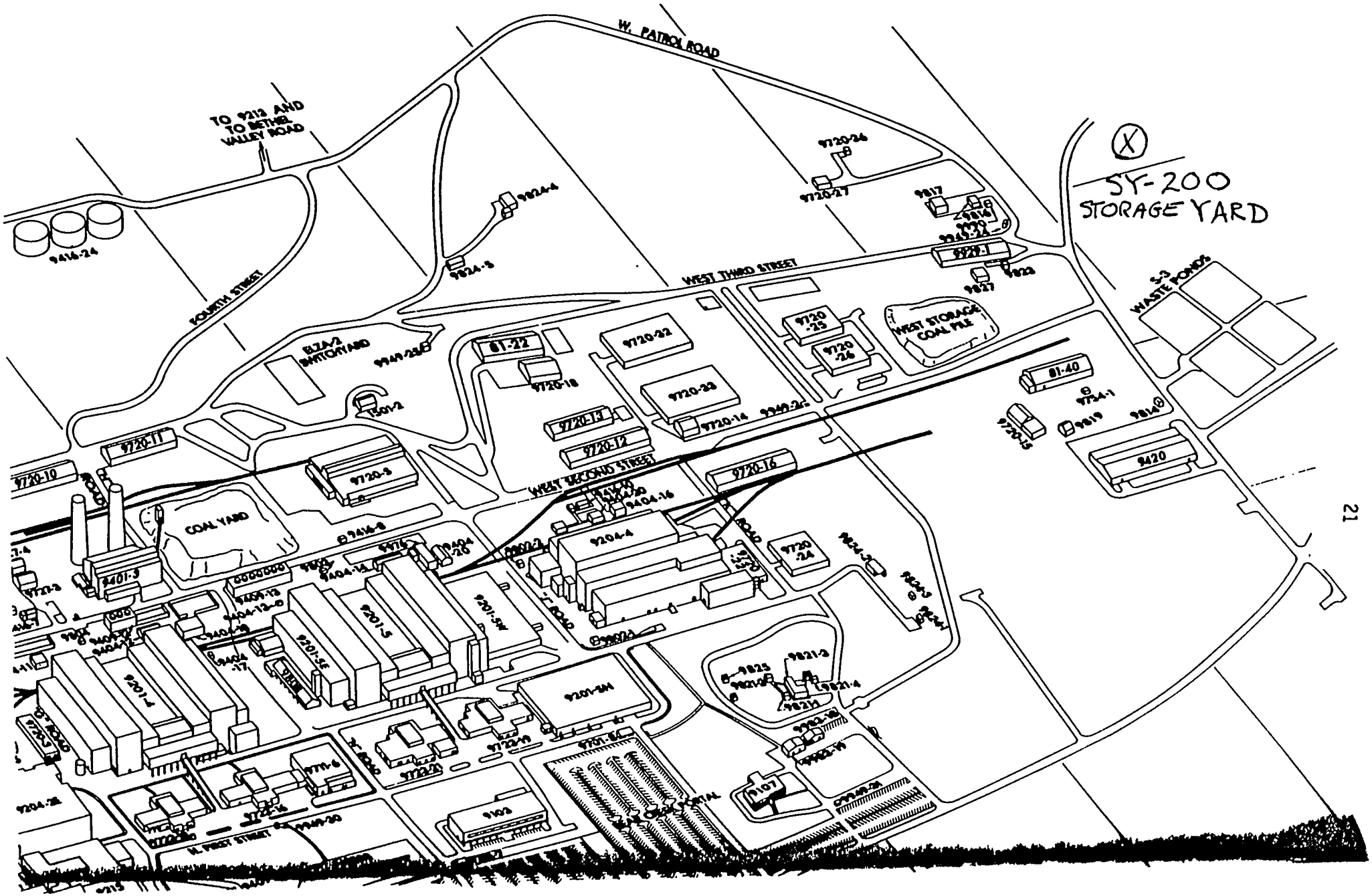


Fig. 3-1. (Continued) Location of ORNL PCB Storage Sites at Y-12.

within the Y-12 complex, which has a standard industrial setting. It is unlikely that the ecology of the area would be disturbed significantly if remedial actions proved to be necessary. Therefore, no additional ecological data are needed for this area.

There is work currently being done and proposed at Y-12 that deals with PCB contamination of the environment. It is possible that that the PCB site contamination scoping studies could be integrated with this work. Contact the following person for more information on this program:

R. H. Kingrea, Y-12 Health Safety and Environmental Rept., 615/576-7897.

There will be very little need for environmental analyses for these areas if the PCB storage sites are shown to be free of contamination. However, if an area is shown to be contaminated with PCB, the site would have to be characterized, cleaned-up (in accordance with both RCRA and TSCA) and an environmental appraisal performed. Records indicate that PCB contaminated oils have leaked from a transformer at SY-200. Therefore, this site, as well as other sites specific, must be studied to ensure that PCBs have not contaminated the storage area and the environment. This task has yet to be addressed.

Knowledgeable personnel include:

L. L. Attwood, Department of Environmental Management, 615/574-0369
B. D. Barkenbus, Department of Environmental Management, 615/574-6665
T. W. Burwinkle, Operations Division, 615/576-0423.

4.0 CONTAMINATED OIL STORAGE TANK AT HYDROFRACTURE SITE (7860)

A 4500-gallon mild steel storage tank was installed underground at the new Hydrofracture Facility (Building 7860) in 1981 or 1982. Figure 4.1 shows the location of this storage tank. The intent of the tank is to hold radioactively contaminated waste oils and water that were being stored in drums. Estimates are that the tank is about half full of radioactively contaminated waste oils and water. The level of radioactivity is estimated to be 5 Ci/gal. The exact nature of the radioactivity is unknown at this time.

The Operations Division is currently collecting detailed information on this tank. This is to include characterization of the contaminated waste by the end of FY86 or early FY 87. The 7860 Storage Tank Data Sheet, with known information, is given in Figure 4.2. The remainder of this information is expected to be completed by March 1986. For further information on Storage Tank 7860 contact:

Pete Helms, Operations Division, 615/576-2624
L. C. Lasher, Operations Division, 615/574-7057.

This tank is currently being handled as a RCRA Subtitle I Tank (underground storage tank). The DEM will be producing appropriate storage tank documentation in April 1986. For information on the 7860 storage tank regulatory status contact:

J. T. Kitchings, Department of Environmental Management, 615/574-7363.

Remedial action will probably involve the removal and disposal of the contaminated waste oil and water from this tank. An environmental assessment dealing with the disposal of the waste materials and storage tank may be necessary before remedial action is instituted. Storage tank

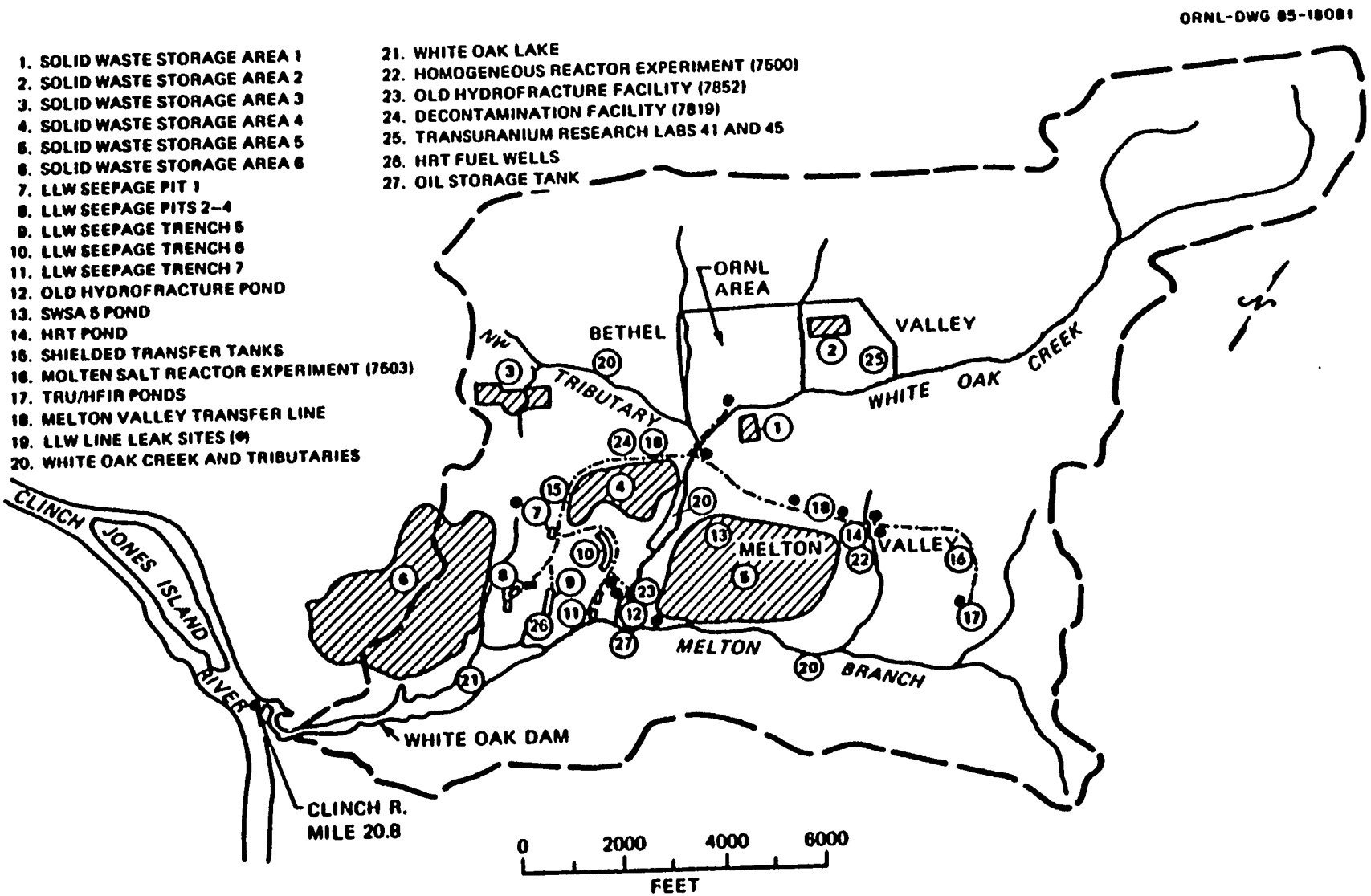


Fig. 4-1. Location (no. 27) of the New Hydrofracture Contaminated Waste Oil Storage Tank.

Fig. 4-2. New Hydrofracture Contaminated Oil Storage Tank (7860) Data Sheet.

1. Tank No./Name 7860 NEW IN HYDROFRACTURING FACILITY
2. Description of location _____
3. Plant ID or serial no. _____
4. Building Engineer S.R LARKINS
5. Division responsible OPERATIONS Contact L.C. LASHER
6. Tank Manufacturer _____ Model no. _____
7. Age of tank _____ Date installed in present location _____
8. Total tank capacity 4500 gallons
9. Function of tank: _____
10. Status of tank:
    - ____ currently in use
    - ____ temporarily out of service
    - ____ permanently out of use
    - ____ to be brought into service in the future; (date: _____)
11. If tank is permanently out of service, estimate:
    - date last used (mo/yr) _____
    - quantity of contents remaining _____ gallons
    - Was tank filled with inert material? _____
12. Tank shape:
    - ____ cylinder (upright)
    - ____ cylinder (horizontal)
    - ____ sphere
    - ____ other (specify) _____
13. Dimensions:
    - diameter _____ ft
    - height _____ ft
    - shell thickness _____ inches
    - other _____ ft
14. Is tank (mark all that apply):
    - ____ fixed roof
    - ____ floating roof
    - ____ open top
    - ____ totally below grade
    - ____ totally above grade
    - ____ partially below grade (% below)
15. Is attached piping:
    - ____ totally above grade
    - ____ totally below grade
    - ____ partially below grade (%) _____.
16. Is 10% or more of the total volume (including the volume of attached underground pipes) of the tank underground? (Do not include volumes situated in underground areas, such as basements, where the tank is situated upon or above the floor.) ____ yes ____ no
17. What material is currently or has been most recently stored in tank (mark all that apply):
    - ____ empty
    - ____ petroleum (specify): ____ diesel ____ kerosene
      - ____ gasoline (including alcohol blends)
      - ____ used oil
      - ____ other (specify) _____
    - ____ hazardous substance(s) (specify, giving relative amounts if several are present): RADIOACTIVE CONTAMINATED OIL
    - ____ other (specify) _____

Fig. 4-2. (Continued) New Hydrofracture Contaminated Oil Storage Tank (7860) Data Sheet.

18. Material of construction of tank (mark one only):
    ____steel (type)_____ ____concrete
    ____fiberglass reinforced plastic ____other (specify)_____
19. Internal protection (mark all that apply):
    ____cathodic protection ____interior lining (e.g. epoxy resins)
    ____none ____other (specify)
20. External protection (mark all that apply):
    ____painted (e.g. asphaltic) ____fiberglass reinforced plastic coated
    ____none ____other (specify)_____
    ____secondary containment (specify)_____
21. Piping (mark all that apply):
    ____bare steel ____fiberglass reinforced plastic
    ____galvanized steel ____cathodically protected
    ____other (specify)_____
22. Is groundwater monitoring provided for:____tank ____piping
23. General condition of tank: ____known leaks ____suspected leaks
    ____no leaks ____unknown
24. General condition of attached piping:
    ____known leaks ____suspected leaks
    ____no leaks ____unknown
25. If leaks are known to have occurred, what material(s) leaked_____
    _____
    Please estimate quantities of leaked materials_____
    Dates of leakage_____ Describe any corrective action taken_____
26. Describe any leak detection system associated with the tank:_____
    _____
    Are monitoring records available?_____ If so, where?_____
    Are inventory control records available?____ If so, where?_____
27. Describe the history of maintenance for the tank/piping_____
    _____
28. Has the tank system been tested?_____ Where are the records?_____
    _____
29. What is the quantity of material added to the tank?
    _____gallons/(day)(week)(month) (circle one)
30. What is the average distance from the top of the tank to the liquid surface?_____
31. What percent of the annual flow through the system is in:
    Dec. - Feb. _____% March - May _____%
    June - Aug. _____% Sept. - Nov._____%
32. In this space, discuss any other information that you feel would be valuable to the survey or complete previous questions where space provided was inadequate._____
    _____
    _____
    _____
    _____
    _____
    _____
    _____
    _____
    _____
    _____

509Z

material characterization must be completed before detailed remedial actions can be designated and implemented. Since this tank has been installed recently, it is unlikely that a leak has developed. Therefore, geologic, hydrologic and ecologic site characterization should not be instituted at this time.

5.0 RECOMMENDATIONS

This section will summarize recommedations presented in this report. Refer to individual report sections for a complete analysis of Hazardous Waste Remedial Action site recommendations.

Mercury contaminated soils sites require further site characterization. This could include the initiation of mercury vapor detection in soils, three dimensional soil sampling for the presents of mercury, and the installation of groundwater monitoring wells. Near-term cursory characterization around Building 4508 is necessary to determine if the soil has been contaminated. An environmental assessment must be performed for mercury contaminated sites. This will include a review of mercury contaminated soil cleanup activities within the City of Oak Ridge and at the Y-12 Plant.

Surplus PCB equipment storage sites at Y-12 require that a cursory study be instituted to determine if PCB contamination has occurred. Further site characterization activities may have to be initiated if this study indicates PCB contamination. An immediate review of possible contamination of storage area SY200 should begin. This activity, as well as other activities, should be integrated with characterization and cleanup activities at the Y-12 Plant.

The contaminated waste oil storage tank located at the new hydrofracture facility may require an environmental review dealing with the disposal of the waste oil and storage tank. This will be determined after waste characterization has been completed. Due to the fact that the waste storage tank is relatively new, it is unlikely that site characterization will be necessary.

DATE ISSUED JUN 20 1987

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POST OFFICE BOX X, OAK RIDGE, TENNESSEE 37831

ORNL/RAP/LTR-86/37

DATE: July 1986

SUBJECT: Inventory of ORNL Remedial Action Sites:
8. Radioisotope Processing Facilities

TO: T. E. Myrick

FROM: N. T. Edwards
F. G. Taylor

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LETTER REPORT RAP86-37

(Internal Use Only)

Inventory of ORNL Remedial Action Sites: 8. Radioisotope Processing Facilities

N.T. Edwards
and
F.G. Taylor, Jr.

July, 1986

Environmental Sciences Division
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37831

Operated by

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for the
U.S. Department of Energy

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ABSTRACT

This report summarizes the history and contamination levels of facilities, equipment, and storage areas utilized for radioisotope processing at ORNL. Thirteen contaminated areas are identified and discussed. Alpha and beta-gamma dose rates range from near background in some areas to as high as 1000 rad/h in isolated hot spots near some equipment and facilities. Radionuclide contaminants include isotopes of cesium, strontium, cerium, promethium, uranium, plutonium, americium, ruthenium, cobalt, carbon, gallium, germanium, silver, cadmium, and yttrium. Amounts of radionuclides found in these facilities are generally very small with the exception or radioisotope storage facilities (^{90}Sr Power Generators and the ^{60}Co Storage Garden). These facilities contain 10^4 to 10^5 Ci of their respective radionuclides. Surface contamination ranges from near background to $>5 \times 10^5$ dpm/100 cm^2 on the inside walls of some used storage tanks. Transferable contaminant levels range from <200 dpm to about 10,000 dpm/100 cm^2.

1. OBJECTIVES

The primary purpose of the Remedial Action Program is to provide surveillance and maintenance, corrective measures, and facility decommissioning necessary to place the Oak Ridge National Laboratory (ORNL) in compliance with applicable environmental and health regulations for radioactive and hazardous chemical wastes. Individual sites will be decontaminated in accordance with priorities prescribed to control the potential for human exposure and relative magnitude of ecological impacts. Remedial action is a component of the Laboratory's Environmental Restoration and Facilities Upgrade Program (ERFU). In the program, remedial actions will be conducted at locations where past management practices or research activities have resulted in contamination of facilities or the environment that necessitate decommissioning or corrective actions. Remedial actions are divided into two activities: (1) site decommissioning (Surplus Contaminated Facilities Program-SCFP and Surplus Facilities Management Program-SFMP) and (2)the Site Corrective Measures Program (SCMP).

The Site Decommissioning Program is designed for management of surplus facilities (primarily structures), contaminated with radioactive materials and hazardous chemicals from the end of their operating life until final facility disposition. The Site Corrective Measures Program (SCMP) is designed for the management of inactive sites (process ponds, spill sites, environmental

research areas, etc.), contaminated with radioactive materials and hazardous wastes, leading to the eventual stabilization of these sites in accordance with environmental regulations.

Remedial Action Program sites at ORNL comprise 13 distinct functional groups (Table 1) among the several programmatic categories: SCMP, (2) SCFP, and (3) SFMP. In a series of reports, an inventory of each area is presented in a condensed summary. The summary is the product of pertinent information abstracted from readily available literature and knowledgeable individuals on known contaminated sites requiring remedial action. The information includes site location and description, history, contaminants and their characteristics, extent of contamination, regulatory status, geological, hydrological, and ecological data, hazard ranking evaluation, and data deficiencies.

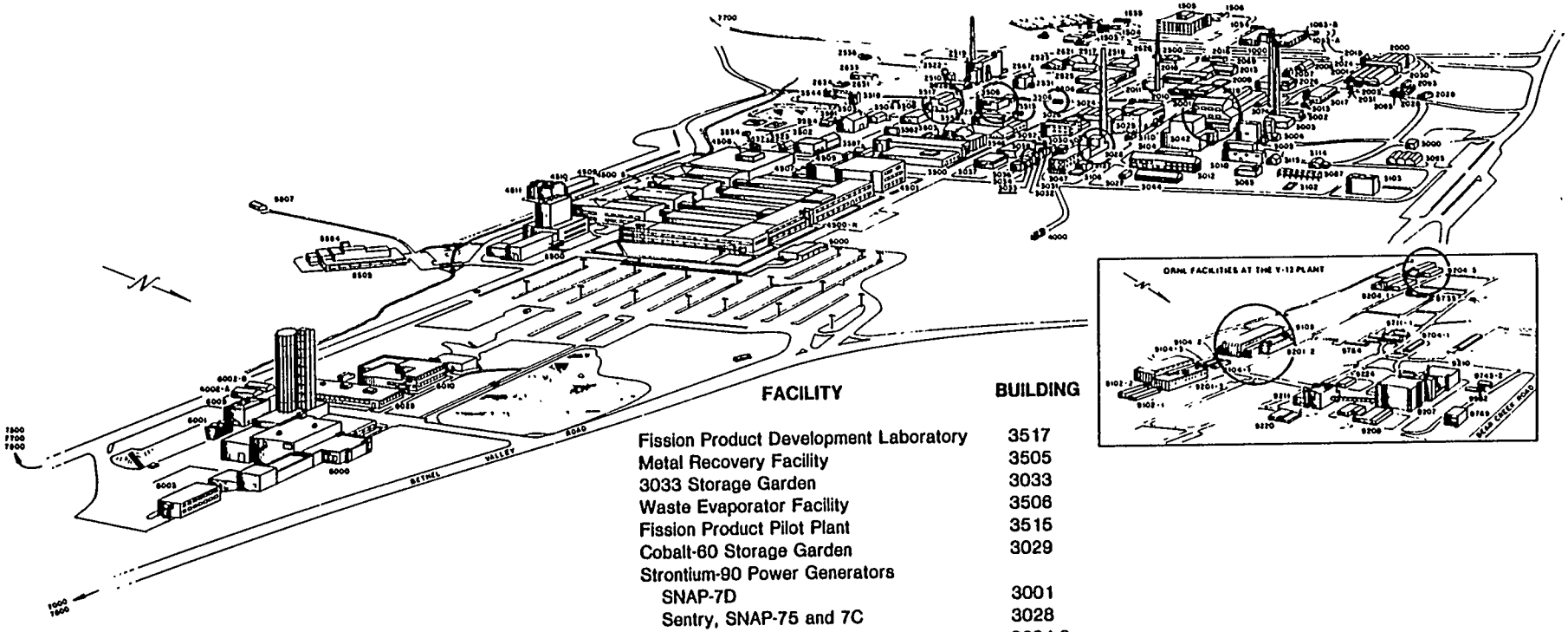
The subject of this report is the Radioisotope Processing Facilities. An abbreviated history of the facilities and levels of radioactivity in various parts of the facilities are included. Locations of most of the facilities are provided in figure 1. Most of the information in sections 2.1 through 2.6 were obtained from a report by Myrick (1984). Other sources are cited in the text.

Table 1. Inventory of ORNL Remedial Action Sites

1.	Solid Waste Storage Areas
2.	LLW Seepage Pits and Trenches
3.	Process Ponds
4.	White Oak Creek Watershed
5.	LLW Line Leak Sites
6.	Environmental Research Areas
7.	Hazardous Waste Sites
8.	Radioisotope Processing Facilities
9.	Experimental Reactor Facilities
10.	Radwaste Facilities
11.	Research Laboratories
12.	Abandoned Injection Wells
13.	Other Contaminated Sites

Figure 1. Locations of the Radioisotope Processing Facilities (not including the Shielded Transfer Tanks stored near SWSA 4 and the Plutonium Process Condensate tank stored near Building 9720-8 at Y-12).

ORNL-DWG 86-12162



FACILITY	BUILDING
Fission Product Development Laboratory	3517
Metal Recovery Facility	3505
3033 Storage Garden	3033
Waste Evaporator Facility	3506
Fission Product Pilot Plant	3515
Cobalt-60 Storage Garden	3029
Strontium-90 Power Generators	
SNAP-7D	3001
Sentry, SNAP-75 and 7C	3028
Beta Cubicle	9204-3
Plutonium Process Facility	9204-3
Curium Handling Glovebox	9204-3
ORNL 86-inch Cyclotron Facility	9201-2

2. DESCRIPTIONS OF FACILITIES AND CONTAMINANT INVENTORIES

2.1 FISSION PRODUCT DEVELOPMENT LABORATORY (FPDL)

The FPDL (Figs. 2 and 3) was constructed in 1958 to separate kilocurie quantities of ^{137}Cs, ^{90}Sr, ^{144}Ce, and ^{147}Pm. It was modified in 1963 to allow production of megacurie amounts of ^{137}Cs, ^{90}Sr, and ^{144}Ce. The FPDL (building 3517) consists of 23 large-volume , stainless steel lined, concrete-shielded hot cells with associated manipulator galleries and operating areas. The facilities are enclosed in a reinforced concrete, steel, and brick structure with adjacent below-grade tank farm cells. It contains a cell ventilation and off-gas system, a process chilled water system, radiation and contamination monitoring systems, general building services, and a process waste and low level waste collection system. In 1975 the facility was placed in standby condition and initial decontamination efforts undertaken. Since that time, portions of the facility have been reactivated for additional fission product processing and for use as ORNL's decontamination facility. The decontamination facilities consists of a vibratory finisher and electropolishing unit.

Currently (June, 1986), ~1.5 megacuries of ^{90}Sr and ~0.25 megacuries of ^{137}Cs re stored at this facility (Joe Devore, ORNL, personal communication). The ^{90}Sr is in the form of strontium fluoride and is doubly contained in Hastelloy-C Tubes. The ^{137}Cs is in the form of cesium chloride and is stored in stainless steel capsules. The amounts of these isotopes stored at the FPDL

Figure 2. Schematic of the Fission Product Development Laboratory.

ORNL-LR-DWG 10845

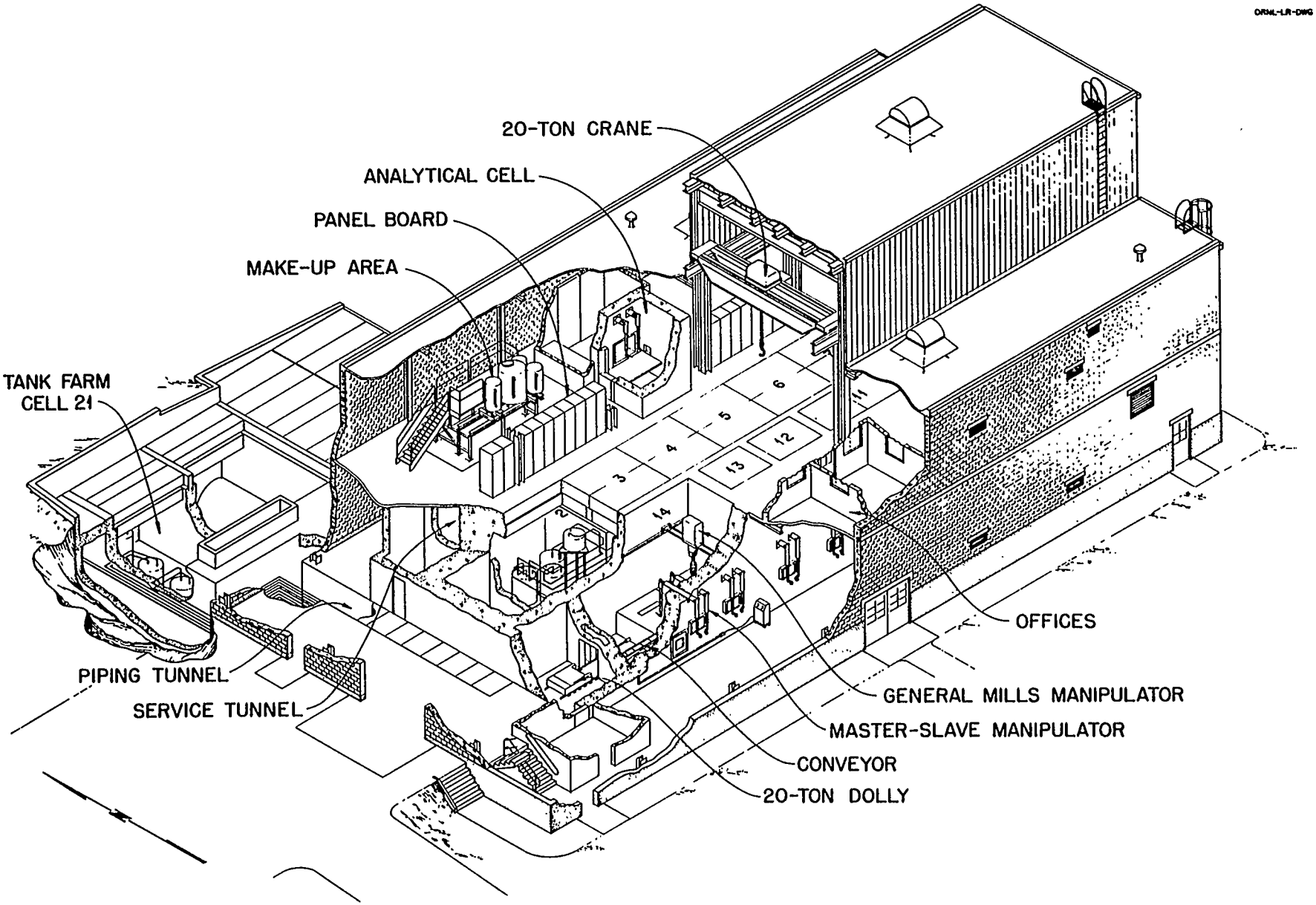
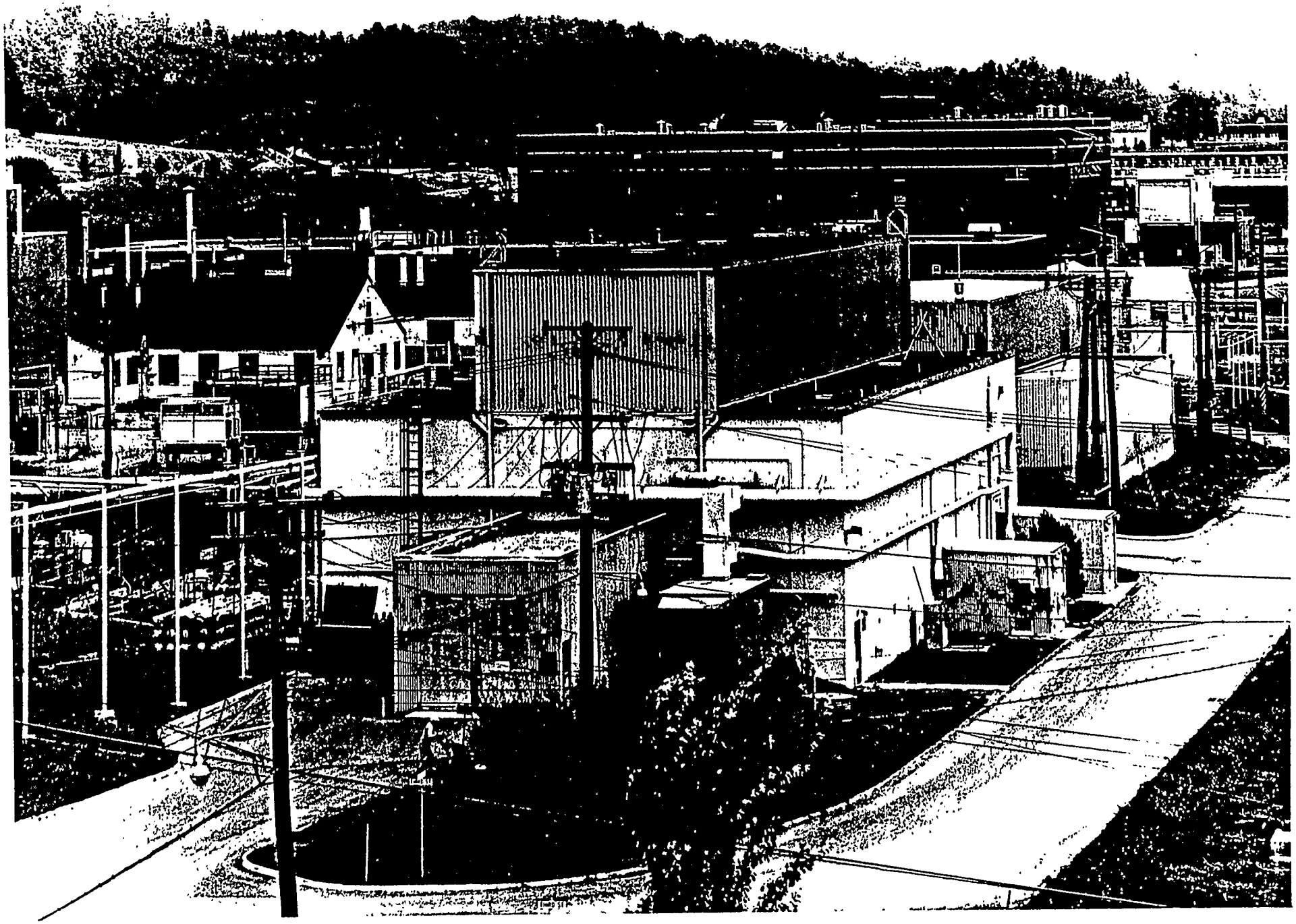


Figure 3. Photograph of building 3517 which houses the Fission Product Development Laboratory.



vary substantially over time. All of the ^{90}Sr is scheduled to be loaded into thermoelectric generators (Joe Devore, ORNL, personal communication to N.T. Edwards, June, 1986).

The contamination problems at this facility are not due to the large amounts of ^{90}Sr and ^{137}Cs periodically stored there, because they are well contained, are associated with active source fabrication programs, and are not the concern of the RAP. However, the inactive process cells contain an array of contaminated tanks, piping, samplers, services, and instrumentation with background radiation levels ranging from 1 - 100 rad/h, with isolated hot spots of 100 - 1000 rad/h. During decontamination, which has been underway for the last several years and is continuing, about 227 m^3 of solid LLW waste and 265 m^3 of liquid LLW waste is expected to be generated. About half of this waste has already been disposed (T.W. Burwinkle, ORNL, personal communication to N.T. Edwards, June 23, 1986).

The proposed disposition of the FPDL cells is decontamination for reuse.

2.2 METAL RECOVERY FACILITY (MRF)

The MRF (Figs. 4 and 5) was a small-scale production reprocessing plant, originally constructed in 1951 for the recovery of uranium from fuel and waste solutions. The facility was later found to be useful for recovering uranium, plutonium, americium, and other miscellaneous materials from a variety of low-burnup reactor fuels and other feed materials. The facility

Figure 4. Schematic of the Metal Recovery Facility.

ORNL-DWG 83-18106

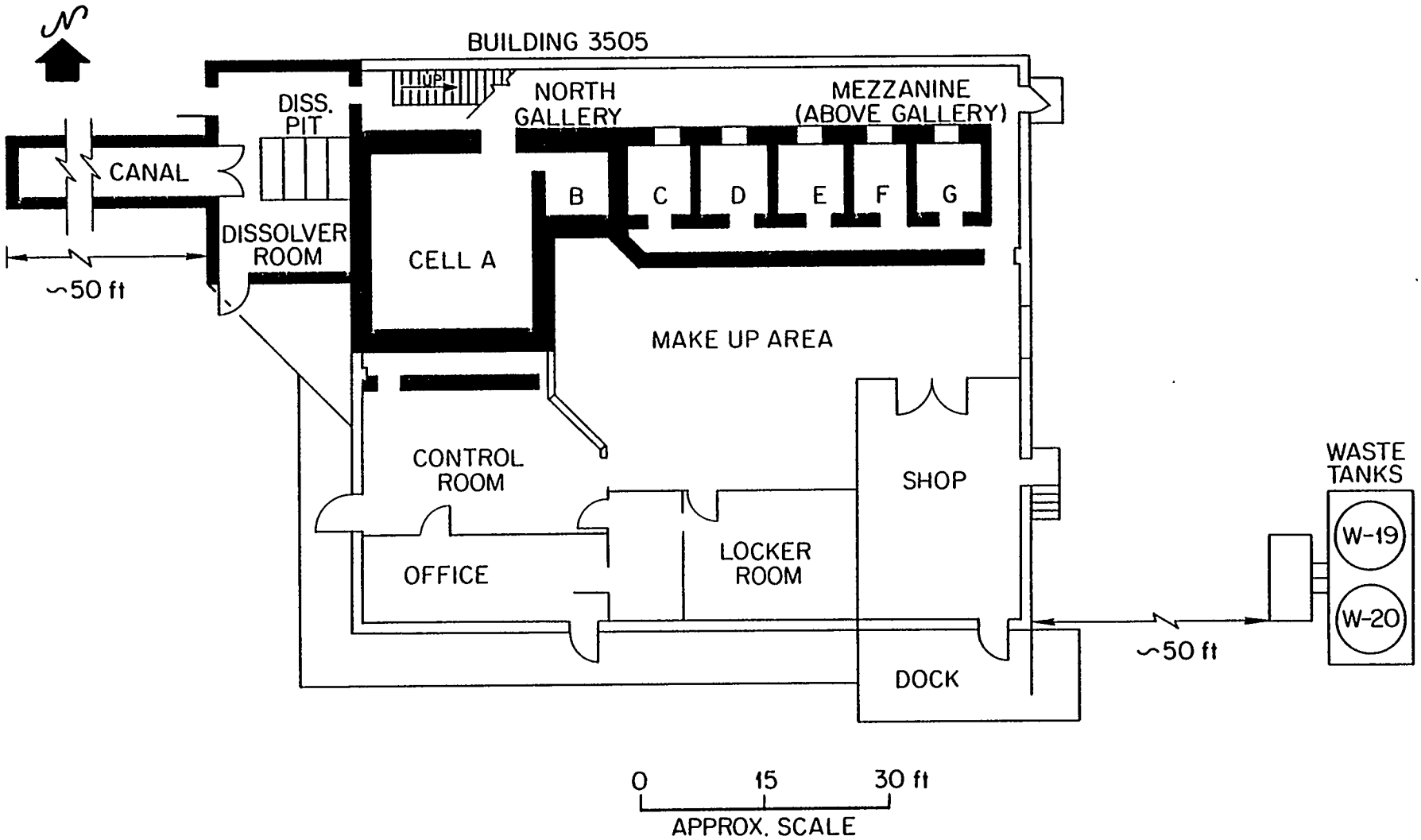
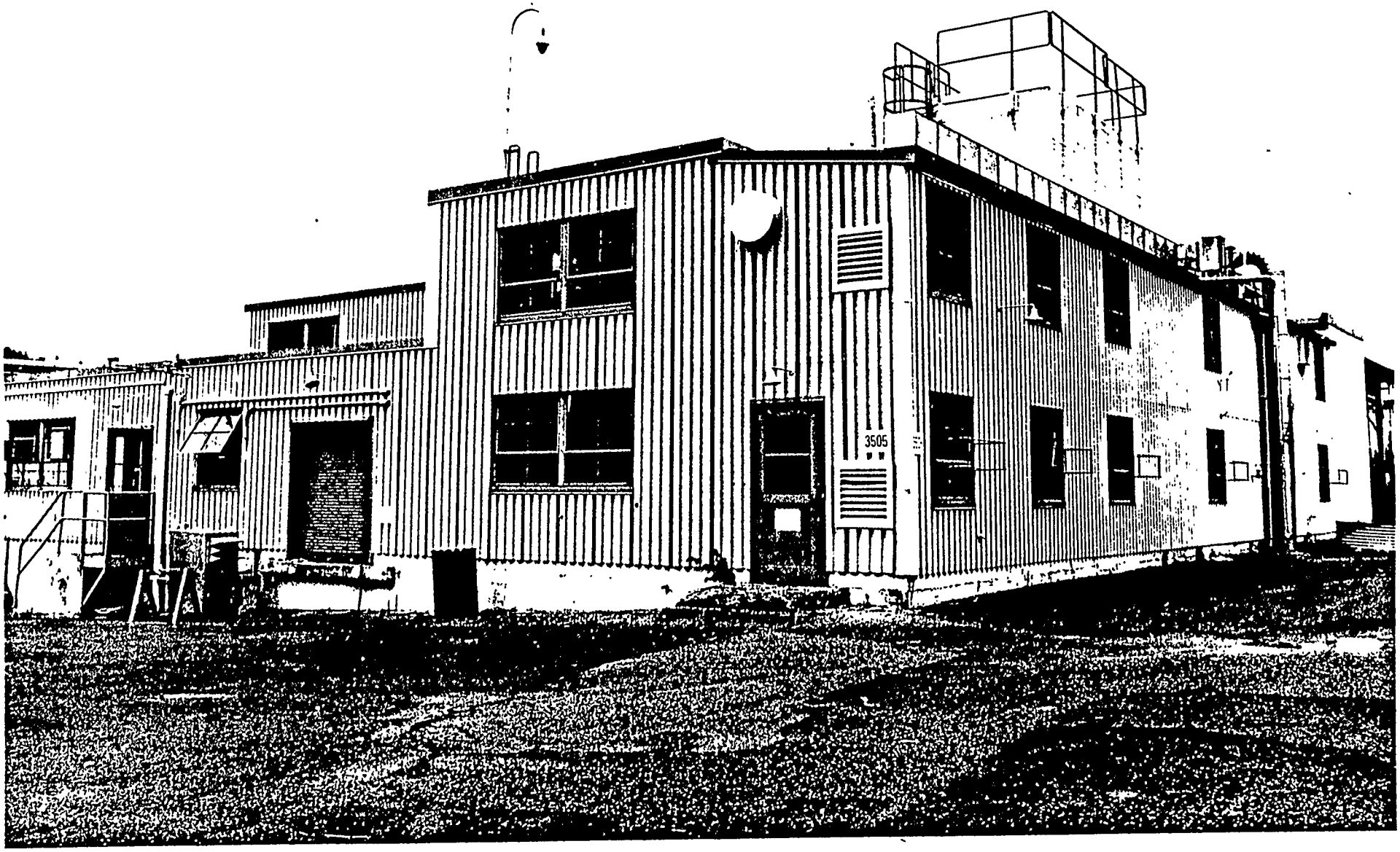


Figure 5. Photograph of the Metal Recovery Facility (Bldg. 3505).


3505

operated from 1952 to 1960 producing about 320 Mg of uranium, 140 kg of plutonium, 1.3 Mg of neptunium, and 55 kg of americium. The MRF consists of building 3505, an adjacent below-grade canal, and two nearby buried waste tanks. The building is basically a one-story, steel siding structure constructed around seven above-grade concrete process cells and a below-grade dissolver tank pit. The canal is a 4.3 meter-deep water-filled concrete basin that was utilized for storage and handling of fuel slugs. The two waste tanks (8516 L capacities) are located below grade approximately 15 m east of the building.

Abandoned contaminated process equipment, located in the process cells, the dissolver pit, and the canal, exhibit significant levels of alpha and beta-gamma contamination, with isolated spots in almost every cell exceeding 500,000 dpm/100 cm^2 and direct alpha and beta-gamma dose rates in one cell of 400 mrad/h. Samples from the cell walls and floors indicate the presence of significant transuranic contamination. Dose rates in the canal range from 1 to 100 rad/h, and the dissolver pit walls are also heavily contaminated with alpha and beta-gamma emitters. The buried waste tanks are known to be internally contaminated. Radionuclide inventories of the MRF are estimated to be 10 Ci of ^{90}Sr and ^{137}Cs and 1 Ci of ^{238,239} and ^{240}Pu. Decontamination of this facility has been underway for the last several years and is continuing. About 380 m^3 of liquid radioactive waste and 330 m^3 of solid radioactive waste is expected to be generated during decontamination of this facility. About half of this waste has

already been disposed (T.W. Burwinkle, ORNL, personal communication to N.T. Edwards, June 23, 1986).

The proposed disposition of the MRF is decontamination for reuse.

2.3 THE 3033 STORAGE GARDEN

This storage area, located just north of building 3033, consists of seven buried stainless steel cylinders (or wells) about 0.3 m in diameter and 1.5 m long, set in concrete with about 8 cm extending above ground level (Fig. 6). It was used from 1956 to 1975 to store sealed radioactive sources, miscellaneous contaminated items, and irradiated targets prior to processing.

No stored radioactive materials or radiation sources remain in any of the seven wells. Beta-gamma dose rates on the interior of the steel wells range from <0.1 to 40 mrad/h, with measured transferable contamination levels from <200 to 9,400 dpm/100cm^2. About 2 m^3 of solid waste will be generated during decontamination of the facility.

The proposed disposition of this facility is initial decontamination and then entombment.

2.4 WASTE EVAPORATOR FACILITY (BUILDING 3506)

The Waste Evaporator Facility (Fig. 7) received low-level liquid waste from ORNL during the period 1949 - 1954. It was used for concentration and volume reduction of the liquid waste

Figure 6. Photograph of the 3033 Storage Garden north of building 3033.

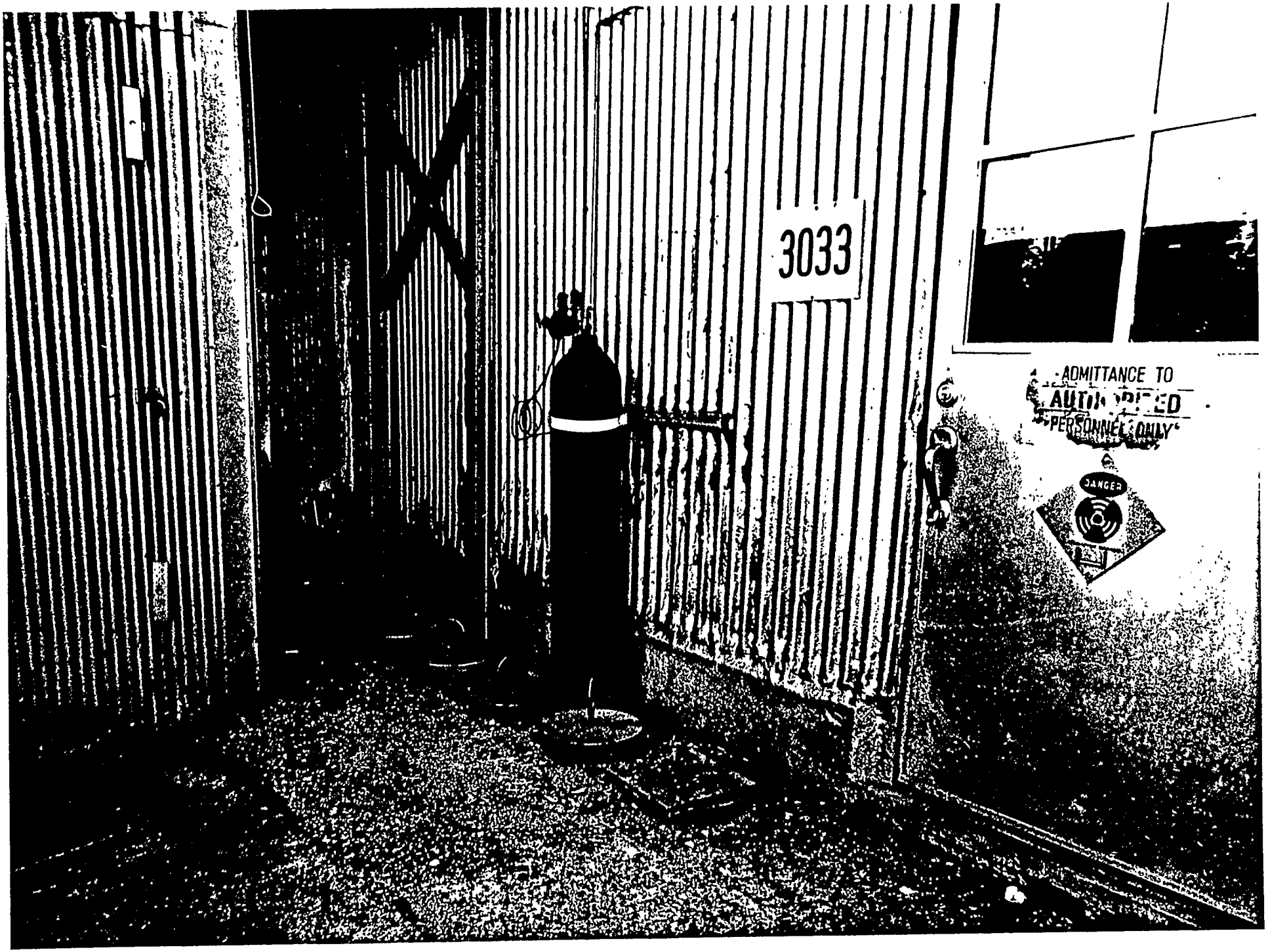
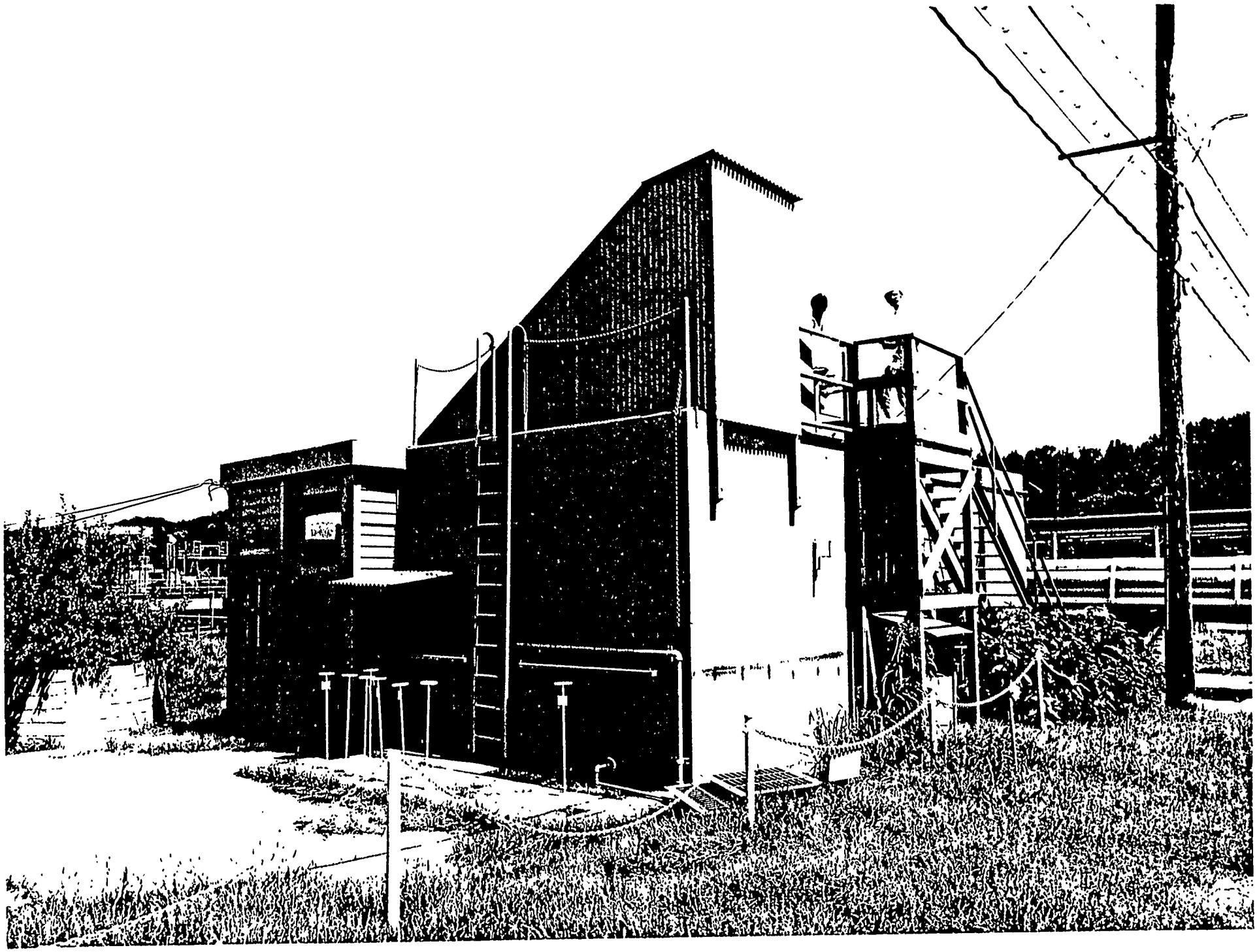

3033
ADMITTANCE TO
AUTHORIZED
PERSONNEL ONLY
DANGER

Figure 7. Photograph of building 3506 which housed the old Waste Evaporator Facility from 1949 to 1954.



prior to its final disposal in excavated pits and trenches in adjacent Melton Valley (see report by Edwards, 1986). The facility was partially decontaminated when the new Waste Evaporator Facility (building 2531) was brought on-line, to provide room for installation of experimental equipment for fission product purification studies and for demonstration of radioactive waste incineration. A new concrete floor was poured over the existing floor in the cell area during decontamination in order to "fix" any remaining radioactivity. The facility was used as a temporary storage area and change house for about 12 years. The structure developed leaks around cell blocks on the roof and all miscellaneous debris and materials were removed and buried (Harry McNabb, ORNL, personal communication to N.T. Edwards, June 13,1986). Water from the leaks was pumped from the floor to process drains. These leaks are currently being repaired and the building is not being used.

The facility consists of a reinforced concrete cell with underground piping, valve pit and an attached wood-framed operating area. Contaminated pipe chases and surplus support equipment associated with the previous evaporator operations exhibit dose rates up to 10 mrad/h, with transferable surface contamination of several thousand dpm/100 cm^2. Curie levels of activity remain in the abandoned valve pit on the north side of the cell and the fixed contamination in the cell floor. Estimated exposure rates in these areas are 1 - 100R/h. Soil in the vicinity of the site is also contaminated. About 645 m^3 of solid

radioactive waste and 30 m^3 of liquid radioactive waste will be generated during the decommissioning of this facility.

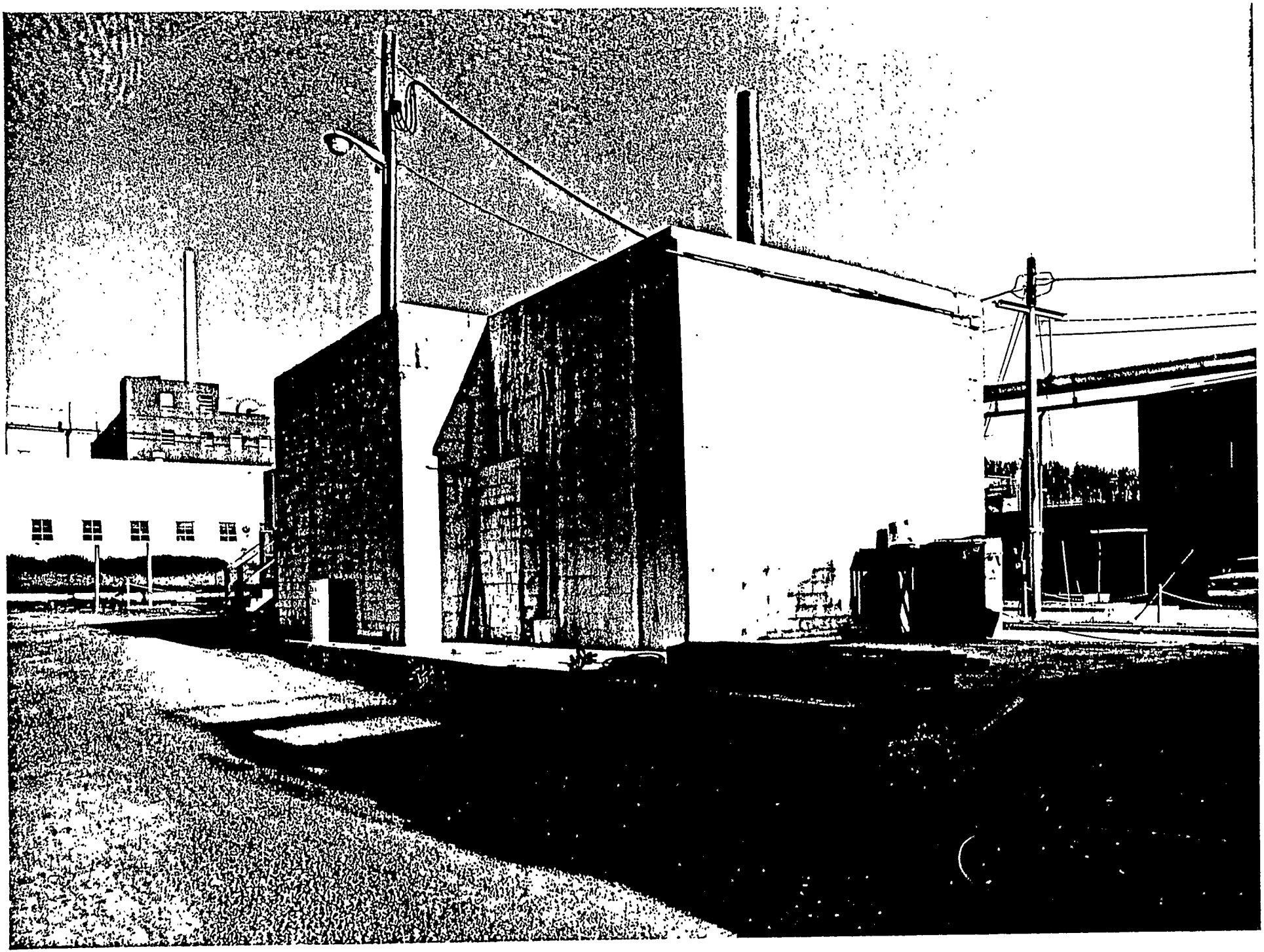
The proposed disposition of the Waste Evaporator Facility is dismantlement with further decontamination being coordinated with the final disposition of the adjacent facilities in the south tank farm area and the surrounding contaminated soil.

2.5 FISSION PRODUCT PILOT PLANT (FPPP)

The FPPP (formerly called the ^{106}Ru tank arrangement) was operated between 1948 and 1958. The ^{106}Ru facility consisted of a concrete pad with tanks, surrounded by stacks of concrete blocks. This was modified in 1950-51 to operate as a hot cell facility. It was then used to separate curie quantities of various radionuclides from low-level liquid wastes. The present facility consists primarily of an unlined concrete-shielded cell, approximately 6 m by 3m by 2.4m high, with an adjacent shielded operating area. After shutdown, the hot cell portion of the building was sealed with concrete block to reduce the radiation levels in the surrounding area. All entrances to the building are sealed. The FPPP (Building 3515) is located on the east side of the south tank farm (Fig. 8).

The current residual radionuclide inventory is believed to be in the range of 10 to 100 Ci, although no recent survey information is available due to the lack of direct access. About 1010 m^3 of solid radioactive waste and 200 m^3 of liquid

Figure 8. Photograph of building 3515 which houses the Fission Product Pilot Plant.



radioactive waste will be generated during decommissioning of this facility.

The proposed disposition of this facility is dismantlement with decommissioning coordinated with the final disposition of the adjacent tank farm area and the Waste Evaporation Facility.

2.6 SHIELDED TRANSFER TANKS

Five heavily shielded transfer tanks (Figs. 9 and 10) are stored outdoors in a materials storage yard near Solid Waste Storage Area 4. These tanks were used repeatedly during the 1960s for shipment of ^{137}Cs-loaded ion exchange resins from Hanford Atomic Products Operations to ORNL. They were used earlier (late 1950,s) to transport lower concentrations of aqueous cesium and strontium waste from Arco, Idaho to ORNL. All were removed from service in 1967 except for one which was used in 1970-71 at ORNL for transferring a caustic solution containing a high-level fission product waste.

The principal contaminant remaining in the tanks is residual ^{137}Cs, with a range of activity between 50 and 700 Ci each. There is no evidence of loss of containment in any of the tanks.

The proposed disposition of these tanks is shallow land burial.

2.7 COBALT-60 STORAGE GARDEN

This 1.82 m^2 subterranean (2.6 m deep) facility (Fig. 11), located in Building 3029, is capable of gamma irradiation doses

Figure 9. Schematic of Shielded Transfer Tank.

ORNL-LR-DWG 63529R

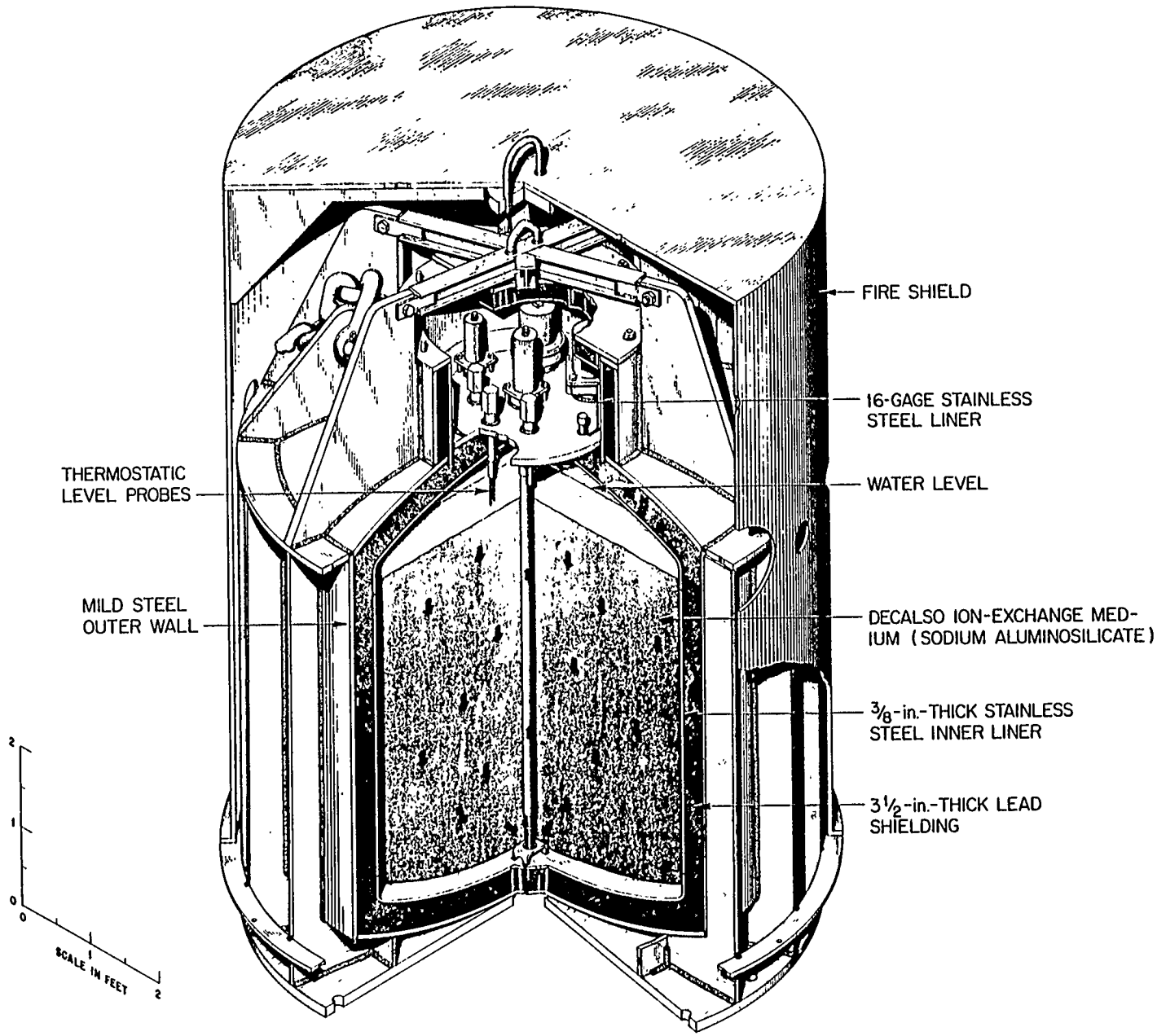
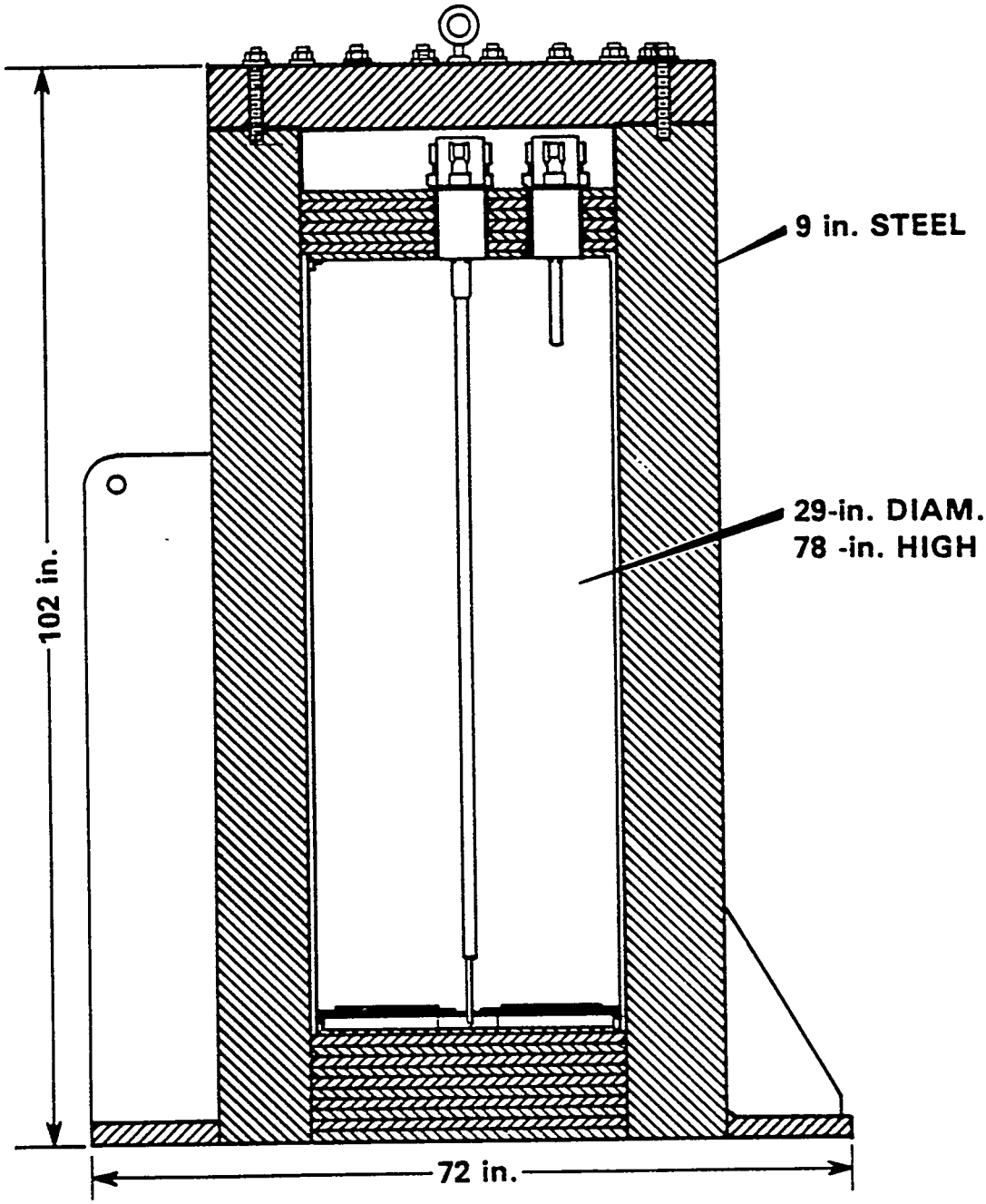


Figure 10. Schematic of Model III Shielded Transfer Tank.

ORNL-DWG 79-20020

DOT SP 5174

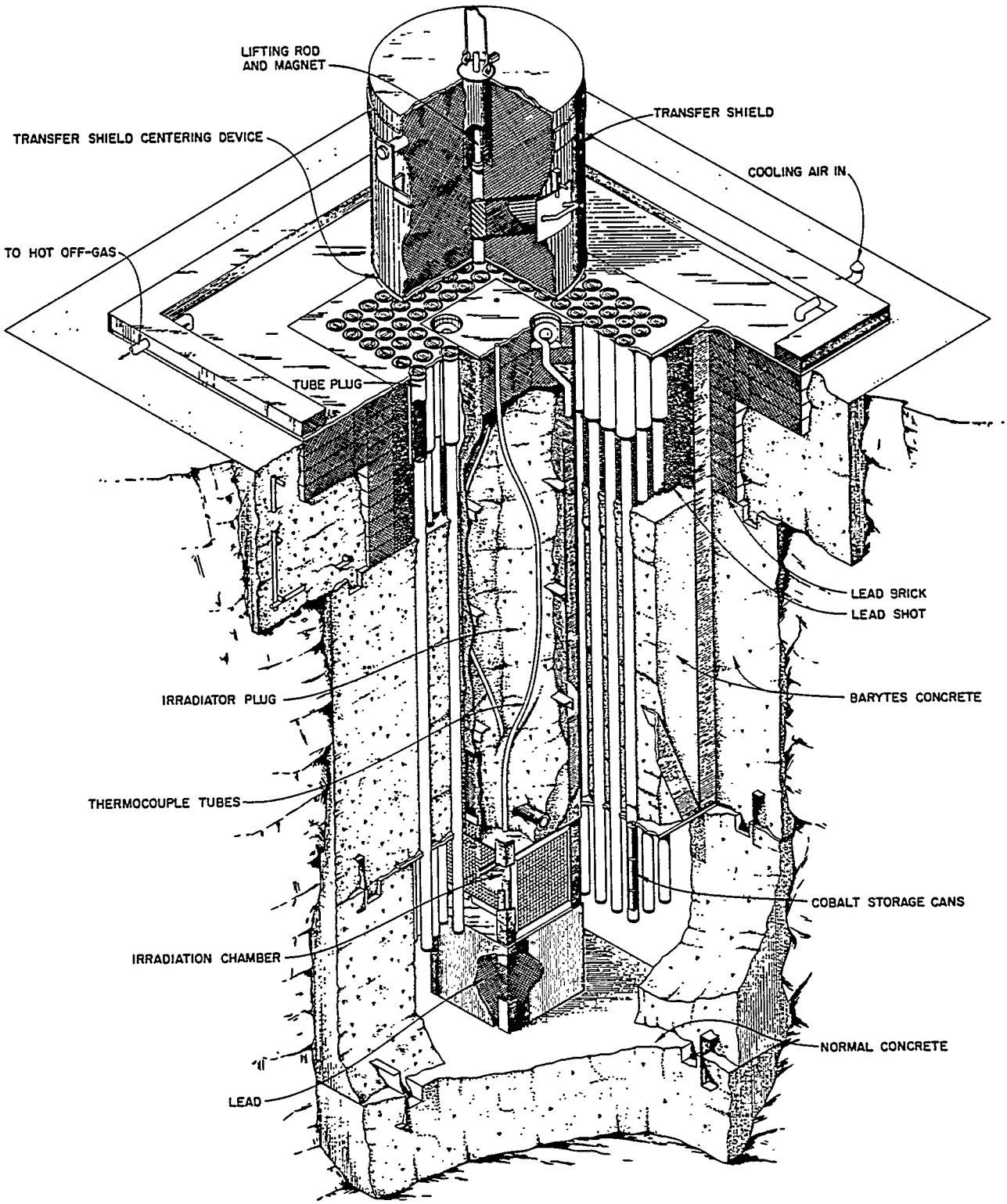
SHIELDED TRANSFER TANK-MODEL III



WEIGHT—42,000 lb.
AUTHORIZED CONTENTS—90,000 Ci ^{137}Cs.

Figure 11. Schematic of the Cobalt-60 Storage Garden.

ORNL-LR-DWG 14492



COBALT STORAGE GARDEN

of up to 1.5 X 10^6 rad/h (C.L. Ottinger, personal communication to N.T. Edwards, April 3, 1986). It currently has an inventory of about ~50,000 Ci of metallic form ^{60}Co contained in 51 aluminum targets or steel cans (T.W. Burwinkle, ORNL, personal communication, June 23, 1986). The facility is comprised of 92 stationary stainless steel tubes surrounding a 0.3 m^2 x 2.1 m long irradiator plug. Each tube is plugged at the top with a lead filled stainless steel tube. The ^{60}Co storage cans located at the bottom of the tubes, which terminate about 0.3 m above the floor of the facility, surround an irradiation chamber. This irradiation chamber is shielded at the bottom with 0.03 m^3 of lead to protect operating personnel during loading and unloading of specimens. The entire facility is shielded with 56 cm of lead bricks plus 56 cm of barytes concrete. This facility is currently operable but not being used.

2.8 STRONTIUM-90 POWER GENERATORS

Four ^{90}Sr thermoelectric generators identified as Sentry, SNAP-7B, SNAP-7C, and SNAP-7D are stored at ORNL. SNAP-7D is stored in Building 3001 and the other three are stored in Building 3028. Each unit consists of one or more welded ^{90}Sr (strontium titanate) sources inside of a shielded thermoelectric device. The ^{90}Sr pellets are enclosed in welded "superalloy" containers. The current ^{90}Sr inventories of these generators are as follows: Sentry - 8,895 Ci; SNAP-7B - 119,480 Ci;

SNAP-7C - 20,260; and SNAP-7D - 117,470 Ci (C.L. Ottinger, personal communication to N.T. Edwards, April 3, 1986).

2.9 BETA CUBICLE (Building 9204-3, Y-12)

The Beta Cubicle was a power supply and control system located in Building 9204-3 at Y-12. Operation of the Cubicle began in the mid 1940's. Most of the contents have been removed. The major contaminant of the Cubicle was polychlorinated biphenyls (PCB), and not radioactive contamination. Information on the date of decommissioning and equipment removal is not known. Sometime after an attempt to salvage the cubicle, the PCB residue was removed. A radiation survey indicated no transferable contamination greater than 20 dpm/100 cm^2 alpha or greater than 200 dpm/100 cm^2 beta-gamma (Houser, 1986a).

This facility is being considered for removal from the Remedial Action Program Listing (T.W. Burwinkle, ORNL, personal communication to N.T. Edwards, June 12, 1986).

2.10 PLUTONIUM PROCESS CONDENSATE TANK (Building 9720-8, Y-12)

During the 1950's and 1960's, a 500 gal (1892 L) steel tank was used to transport plutonium process condensate from Building 9204-3 to ORNL for disposal. The tank is attached to a permanent stand, designed for truck mounting and transport. It is presently located south of Building 9720-8 at Y-12. It has been determined that the tank interior is contaminated with radiation levels in excess of 5×10^5 dpm/100 cm^2 alpha. Exterior portions

of the tank have been protected with an application of a paint sealant. The transferable contamination on the exterior tank is less than 30 dpm/100 cm^2 alpha and less than 200 dpm/100 cm^2 beta-gamma (Houser 1986b). Radiation exposure at the exterior surface is less than 1 mR/h.

2.11 PLUTONIUM PROCESS FACILITY (Building 9204-3, Y-12)

The plutonium process facility in Building 9204-3 consists of an L-shaped glove box in Room 116. The box resulted from an attempt to isolate aspects of the plutonium processing operations, by enclosing a portion of the room. This utilized the existing room walls and ceilings. Operation of the facility began in the early 1950's until decommissioning in early 1960. Isotopes generated were primarily ^{240}Pu and ^{242}Pu (J.G. Tracy, ORNL, personal communication to N.T. Edwards, June 27, 1986). Presently the facility contains equipment (tanks, pipes, valves, etc.).

Both the equipment and box components are highly contaminated with detectable radiation levels greater than 5 X 10^5 dpm/100 cm^2 alpha (Houser, 1986c). No transferable contamination was detected on external surfaces, except at the bottom of the enclosure where the radiation level is less than 2 mR/h.

2.12 CURIUM HANDLING GLOVEBOX (Building 9204-3, Y-12)

The curium handling facility in Building 9204-3 includes a small glovebox used in support of the curium operations from the late 1960's to the mid 1970's. Isotopes generated were primarily ^{244}Cm and ^{246}Cm (J.G. Tracy, ORNL, personal communication to N.T. Edwards June 27, 1986). The operating area and the glovebox are presently in stand-by status.

In a radiological characterization, Houser (1986d) reported interior contamination in the magnitude of 5 X 10^5 dpm/100 cm^2 alpha. External radiation was determined to be less than 2 mR/h, with a transferable contamination of less than 5000 dpm/100 cm^2 alpha.

2.13 ORNL 86-INCH CYCLOTRON FACILITY (Building 9201-2)

In 1950, the Cyclotron Facility was constructed in Building 9201-2 at the Y-12 plant. The cyclotron was dedicated to the production of radionuclides used in medical diagnosis, as trace elements in coal liquefaction and gasification studies, and for treatment of coronary diseases. The facility operated until 1983. It is a large facility, occupying much of Building 9201-2.

In a radiological characterization, it was determined that the external surfaces of the cyclotron are relatively free of radioactivity (Houser, 1986e). Maximum radiation levels of 70

mR/h were identified, with transferable contaminations of <300 dpm/100 cm^2 beta-gamma and <20 dpm/100 cm^2 alpha.

Internally the cyclotron contains varying amounts of beta-gamma emitting radionuclides with short half-lives (^{57}Co, ^{67}Ga, ^{65}Zn, ^{109}Cd, ^{110}Ag, and ^{88}Y). It is estimated that 10 Ci of activity are in the vacuum pumps and oil. Another 100 Ci may be present from residual quantities of radionuclides from reactions of the cyclotron beam with the targets. Lead shielded stainless steel shipping casks and storage bins in the cyclotron area contain abandoned targets with a radiation level <40 mR/h. A small amount (10 to 30 mCi) of at least one longer lived radionuclide (^{207}Bi, half-life= 38y) may be stored in the cyclotron pit in the form of a lead target (J.W. Terry, ORNL, personal communication to N.T. Edwards, July 2, 1986).

Another potential contaminant is the magnet cooling coils contents. These coils and piping contain approximately 15,000 gallons (5.68×10^4 L) of transformer oil which contains less than 50 ppm polychlorinated biphenyls (PCB). A leak in the pit area or outdoor above-ground piping could result in the contamination of the adjacent creek. Currently water in the sump underneath the cyclotron must be pumped out periodically (every week or two during periods of high rainfall) and the pipes in the sump area must be checked for leaks before each pumping (R.D. Parten, ORNL, personal communication to N.T. Edwards, July 2, 1986).

3. GEOLOGY, HYDROLOGY, AND ECOLOGY

The radioisotope processing facilities described in this document range from stainless steel lined, concrete shielded hot cells to gloveboxes and contaminated electric generators. All are located in buildings or on disturbed land areas. Remedial actions to minimize any potential hazard from these facilities are dependent on site-specific environmental factors such as geology, hydrology, and ecology. While no site-specific information is available, Nix et al. (1986) have provided generic descriptions for the Oak Ridge Reservation.

3.1 GEOLOGY

A geological map (McMasters 1963) of the Oak Ridge Reservation identifies four major stratigraphic units: Rome Formation, Conasauga Group, Knox Group, and Chickamauga Limestone. The Rome Formation underlies Haw Ridge, which separates Bethel and Melton Valleys, and consists of evenly bedded, fine-grained sandstone and shale. This formation has a limited capacity for receiving, storing, and transporting water. In unweathered bedrock, water is largely limited to small openings along joints and bedding planes. Groundwater discharge during dry months is very low. The Conasauga Group, underlying Melton Valley, is composed of shales, limestones, and siltstones. Groundwater occurs principally in the weathered zone at openings

along joints and bedding planes. These occur at shallow depths and consequently springs may be common during winter. During the summer, very little water is discharged. The Knox Group underlies much of Chestnut and Copper Ridges and is the most widely distributed geologic unit in East Tennessee. It is composed primarily of dolomitic limestone with prominent chert zones. Underground solution channels characterize this group and establish it as the principal water storage formation. Because of the presence of extensive amounts of limestone, this formation is susceptible to underground solution by migrating groundwater and the development of a network of channels and voids. These often permit easy access of surface waters and free movement of waters underground. However, it is not an important aquifer because the channels are small with limited water storage capacity.

3.2 HYDROLOGY

The Oak Ridge Reservation consists of a series of drainage basins that feed into the Clinch River. The principal drainage system from the ORNL site is White Oak Creek. White Oak Creek originates from springs of the Knox Dolomite on Chestnut Ridge and drains an area of 16.4 km^2 (4052 acres). The creek flows from the main ORNL site in Bethel Valley (Chickamauga LImestone) through a gap in Haw Ridge (Rome Formation) to Melton Valley (Conasauga Shale). The stream varies from 0.6 m (2 ft) to 1.2 m (3.9 ft) in width and from 9.9 cm (3.9 in.) to 24.9 cm (9.8 in.)

deep. Flow rates range from a maximum on 18.2 m^3/s (643 ft^3/s) to a minimum of zero, the average being o.27 m^3/s (9.5 ft^3/s). White Oak Creek is joined by Melton Branch after it passes Haw Ridge and enters White Oak Lake 0.5 km (0.8 mile) downstream.

Melton Branch collects flow from both Haw Ridge and Copper Ridge and drains an area of 3.8 km^2 (939 acres). It is the principal drainage system of the ORNL Melton Valley complex.

Of the four stratigraphic groups, only the Knox Dolomite has significant water storage capacity. The storage usually occurs in solution cavities and may frequently result in springs. Water storage capacity of the Rome Formation, Conasauga Shale, and Chickamauga Limestone is small and occurs along joints and bedding planes. Most wells in these formations have flows less than 38 L/min (10 gal./min.).

Groundwater flow in the weathered residual soil on the ORNL site basically follows the water table conditions. In the Chickamauga Limestone underlying Bethel Valley, groundwater moves through small solution channels. Studies of groundwater movement in the Conasauga Shale of Melton Valley have suggested that the primary direction of groundwater movement parallels the strike. Groundwater discharge is through evapotranspiration, springs, and streams, and it contributes to the base flow to streams that ultimately reach the Clinch River water supply.

Depth to the water table varies both spatially and temporally. Depth to the water table is generally greatest during October to December and least between January and March.

In Bethel Valley, depth to the water table ranges from 0.3 (~1 ft) to 11 m (36 ft), whereas in Melton Valley the range is from 0.3 (~1 ft) to 20 m (65.6 ft).

3.3 ECOLOGY

The land area of the reservation is ~15,000 ha (37,065 acres), with plant communities characteristic of those found in the ridge and valley region of Appalachia. The dominant Oak-hickory association is typified by stands of shortleaf and Virginia pines, as well as the more extensive oaks and hickories. Approximately 75% of the reservation land area is forested. There are no threatened or endangered plant species in areas occupied by the radioisotope processing facilities. There are no endangered animal species on the sites, but the geographical ranges of several encompass the reservation. While threatened or endangered fish species are known to exist in the headwaters of the Clinch River, none have been encountered in laboratory biological sampling programs. The reservation streams and ponds are not suitable habitat for sensitive species because they generally require a free-flowing habitat.

4. REGULATORY STATUS

The radioisotope processing facilities are regulated under the Department of Energy Order 5820.2 which has no provisions for hazard ranking. Therefore Hazard Ranking System analyses for these facilities have not been performed.

5. DISCUSSION

The several radioisotope processing facilities at ORNL (X-10 and Y-12) have been reviewed with the objective of providing descriptive information and identification of data needs for assessment tasks. The types of facilities range from buildings to glove boxes, with remedial action control and supervision in the Surplus Facilities Management Program (SFMP) and the Surplus Contaminated Facilities Program (SCFP).

Among the facilities identified for potential remedial action (Grimsby, 1986), radiological characterization data was available for eight sites. Radiation dose rates ranged from <0.1 to 40 mrad/h for the interior steel wells of the 3033 Storage Garden, to exposure rates between 1 and 100 R/h for the Waste Evaporator Facility. Transferable contamination was not detectable for some facilities, but was significant for others (e.g. the Waste Evaporator Facility).

Major information deficiencies among the various isotope processing facilities are measurements of radiation dose, exposure, and transferable contamination (table 2). Another major data deficiency for many of these facilities (e.g. FPDL, FPPP, and MRF) is characterization data concerning surrounding soil contamination from past operations, or continuing release. It is possible that some radiation information has been overlooked in the assembly of this report, and is actually known for those facilities noted as deficient. Nonetheless, efforts

should be initiated to insure that all facilities are characterized radiologically.

A non radiological problem that needs to be addressed is the PCB contaminated oil in the pipes at the 86-inch cyclotron facility. The current practice of checking the pipes beneath the cyclotron for leaks before pumping water from the sump to the creek would not be necessary if the oil were properly stored or incinerated.

Table 2. Summary of radiological data for the radioisotope processing facilities

	Total Activity	Radiation	Rate	Probe, Surface Reading (dpm/100 cm^2)	Transferrable Contamination, Smear (dpm/100 cm^2)	Isotope
Fission Product Development Lab.	~1000 Ci	β,γ	1-100 rad/h 100-1000 rad/h[a]	—	—	Sr-90, Cs-137, Ce-144, Pm-147
Metal Recovery Facility	~11 Ci	α,β,γ	400 mrad/h 1-100 rad/h[b]	$>5x10^5$	—	Sr-90, Cs-137, Pu-238, Pu-239
Storage Garden 3033	—	β,γ	<0.1-40 mrad/h	—	<200 to 9400	Mixed fission products
Waste Evaporator Facility	Ci levels	β,γ	10 mrad/h 1-100 R/h[c]	—	Thousands	Mixed fission products
Fission Product Pilot Plant	10 to 100 Ci	β,γ	—	—	—	Ru-106
Shielded Transfer Tanks	50 to 700 Ci	β,γ	—	—	—	Cs-137
Cobalt-60 Storage Garden	6000 Ci	β,γ	See footnote*	—	—	Co-60
Strontium-90 Power Generators (4)	266, 105 Ci total	β	—	—	—	Sr-90
Beta Cubicle	—	α,β,γ	—	—	20 α 200 β,γ	—

Table 2. Summary of radiological data for the radioisotope processing facilities (CONT'D)

	Total Activity	Radiation	Rate	Probe, Surface Reading (dpm/100 cm^2)	Transferrable Contamination, Smear (dpm/100 cm^2)	Isotope
Pu Process Condensate Tank	—	α,β	<1 mR/h	>5x10^5 α	<30 α <200 β,γ	Pu-239
Pu Process Facility	—	α,β	<2 mR/h	>5x10^5	—	Pu-240, Pu-242
Curium Handling Glovebox	—	α	<2 mR/h	>5x10^5	5,000	Cm-244, Cm-246
86-inch Cyclotron	10 Ci[d] 100 Ci[e]	α,β,γ	70 mR/h	—	<20 α <300 β,γ	Co-57, Ga-67, Zn-65, Cd-109, Ag-110, Y-88

a = isolated spots; b = canal for water storage of fuel slugs; c = valve pit; d = vacuum pumps and oil; e = target components.
*The Co-60 storage garden (gamma irradiator) can deliver exposures up to 1.5x10^6 rem/h.

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ORNL/RAP/LTR-86/51

DATE: June 30, 1986

SUBJECT: Inventory of ORNL Remedial Action Sites:
9. Experimental Reactor Facilities

TO: T. E. Myrick

FROM: T. J. Blasing

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LETTER REPORT RAP86-51

Inventory of ORNL Remedial Action Sites: 9. Experimental Reactor Facilities*

T. J. Blasing

June 30, 1986

Environmental Sciences Division
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37831

Operated By

Martin Marietta Energy Systems, Inc.
for the
U.S. Department of Energy

*Based on work performed at Oak Ridge National Laboratory, operated by Martin Marietta Energy Systems, Inc., under Contract No. DE-AC05-84OR21400 with the U.S. Department of Energy.

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SUMMARY

This report provides a description of sites within the Remedial Action Program that are classified as experimental reactor facilities. That category includes six sites within the ORNL complex that are part of the Surplus Facilities Management Program (SFMP) and one site about 4 km south of ORNL (Tower Shielding Facility) that is part of the Surplus Contaminated Facilities Program (SCFP). The first six sites have been the subjects of radiological characterizations and decommissioning reports that have provided much of the information contained in the present report. Decommissioning priorities have been considered and suggested for these facilities, and that material is summarized in Chapter 5 of the present report.

The Tower Shielding Facility Equipment has not been investigated nearly as thoroughly as the other six experimental reactor sites in the Remedial Action Program. The facility itself is not surplus, but contains surplus equipment and material that has accumulated over the last 30 years as a result of the activities at the site. While there does not appear to be any reason to assign this site a high decommissioning priority, a decommissioning study should probably be carried out to provide estimates of cost and of waste volume, and to assign an appropriate priority in the decommissioning sequence.

1. OBJECTIVES AND BACKGROUND

The primary purpose of the remedial action task is to provide corrective measures and facility decommissioning necessary to place Oak Ridge National Laboratory (ORNL) in compliance with applicable environmental and health regulations for radioactive and hazardous chemical wastes. Individual sites will be decontaminated in accordance with priorities prescribed to control the potential for human exposure and magnitude of ecological impacts. Remedial actions will be conducted at sites where past management practices or research activities have resulted in contamination of facilities or of the environment that require corrective actions or decommissioning. Remedial actions are divided into two types of activity: (1) the Site Corrective Measures Program (SCMP) and (2) the Site Decommissioning Program. The Site Decommissioning Program is further divided into the Surplus Contaminated Facilities Program (SCFP) and the Surplus Facilities Management Program (SFMP).

The Site Corrective Measures Program is designed for the management of inactive sites (e.g., process ponds, spill sites, and environmental research areas) that have been contaminated with radioactive materials and hazardous wastes, leading to the eventual stabilization of these sites in accord with environmental regulations. The Site Decommissioning Program is designed for the management of contaminated surplus facilities (primarily structures) from the end of their operating life until final facility disposition.

Each of the approximately 140 Remedial Action Program sites at ORNL is divided into one of the three programmatic areas described above (SCMP, SCFP, SFMP). Each site is also classified into one of 13 categories (Solid Waste Storage Areas, Process Ponds, etc.). A list of these categories is provided in Table 1. A series of reports provides a current inventory of the remedial action sites within each category. Each report covers one category and contains a summary of pertinent information abstracted from readily available literature and knowledgeable individuals on known contaminated sites requiring remedial action. Information presented includes that which is known about site location and description, geology, hydrology, climatology, ecology, history, contaminants and their characteristics, extent of contamination, a reference list of major studies concerning the site, regulatory status, names of persons knowledgeable about the site, and other relevant material. This report provides a current inventory of the experimental reactor facilities (category 9, Table 1).

There are seven sites in the category of experimental reactor facilities. These are listed in Table 2. All seven sites are regulated under DOE Order 5820.2. All of these facilities are within the Surplus Facilities Management Program except for the Tower Shielding Facility Equipment (Site 9.7) which is within the Surplus Contaminated Facilities Program. All sites except for the Tower Shielding Facility have been the subject of preliminary decommissioning study reports and preliminary radiological surveys. Much of the material in section 4.1 through 4.6 is a summary of these decommissioning reports and their accompanying radiological characterizations. The decommissioning reports are excellent sources of detailed information about each site. The

Table 1. Functional Categories of Remedial Action Program Sites.

1. Solid Waste Storage Areas
2. Low-Level Waste Seepage Pits and Trenches
3. Process Ponds
4. White Oak Creek Watershed
5. Low-Level Waste Line Leak Sites
6. Environmental Research Sites
7. Hazardous Waste Sites
8. Radioisotope Processing Facilities
9. Experimental Reactor Facilities
10. Radwaste Facilities
11. Research Laboratories
12. Abandoned Injection Wells
13. Other Contaminated Sites

Table 2. Experimental Reactor Facilities Designated as Remedial Action Program Sites at Oak Ridge National Laboratory. All of these sites are regulated under DOE Order 5820.2

Name of Site	ORNL Building	Program Category
9.1 ORNL Graphite Reactor	3001	SFMP
9.2 Molten Salt Reactor Experiment	7503	"
9.3 Low Intensity Test Reactor	3005	"
9.4 Homogeneous Reactor Experiment	7500	"
9.5 ORR Heat Exchanger	3087	"
9.6 ORR Experimental Facilities	3042	"
9.7 Tower Shielding Facility Equipment	7702	SCFP

radiological characterizations itemize radiological hazards and, in some cases, known or suspected nonradiological hazards. The decommissioning reports, radiological characterizations, and other relevant documents are referenced in the discussion of each respective site. Details of the geology, hydrology, climatology, flora, and fauna of the ORNL site are discussed extensively in Boyle et al. (ORNL-5870, 1982, Chapter 3) and are similar from one site to the next within the ORNL complex. Some site-specific comments on those topics will be included in the Section 4.7 dealing with the Tower Shielding Facility Equipment which is about 4 Km south of the main ORNL complex. That site differs in many respects from the other six sites, and it will therefore be treated in a somewhat different format.

The SFMP sites, which include sites 9.1 through 9.6 in Table 2, are currently managed in a structured program of routine maintenance and surveillance (Myrick 1985). The Tower Shielding Facility Equipment (site 9.7) is within the SCFP and will be similarly managed beginning in FY 1987 (Burwinkle 1986).

1.1 References

Burwinkle, T. W. 1986. The ORNL Surplus Contaminated Facilities Program Maintenance and Surveillance Plan, FY 1987-1996. ORNL/CF-86/226. Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Myrick, T. E., 1985. The ORNL Surplus Facilities Management Program Maintenance and Surveillance Plan, FY 1985-1994. ORNL/CF-85/4.

2. ENVIRONMENTAL CHARACTERISTICS

Several environmental characterizations of the areas around Oak Ridge National Laboratory (ORNL) have been published. Probably the most recent and comprehensive of these is the report of Boyle et al. (1982). Much of the material that follows is excerpted or updated from Chapter 3 of that report, or from an earlier report of Dahlman, et al. (1977).

2.1 Geology:

The ORNL site is located in the Appalachian Physiographic Division of the eastern United States, within the Valley and Ridge Physiographic Province. The local topography is characterized by subparallel northeast-trending ridges and valleys associated with southeast-dipping strata. Nine geologic formations or groups ranging in age from Early Cambrian to early Mississippian have been mapped within the Oak Ridge Reservation. All are of sedimentary origin. From oldest to youngest they include the Rome formation, the Conasauga group, the Knox group, the Chickamauga Limestone, the Sequatchie formation, the Rockwood formation, the Chattanooga shale, the Maury formation, and the Fort Payne chert. Much of the younger material has eroded so that the first four strata listed above form the major underlying strata of the region. A northwest-southeast cross section of these major strata is shown in Figure 1. The Rome Formation, of Cambrian age, has been thrust over younger formations of later Cambrian and Ordovician ages. This has occurred in several places, causing the repetition of the geologic sequence within the area as seen in Figure 1.

The differing lithologies of the formations result in different erosion rates. More resistant, ridge-producing, formations include in the

Figure 1. Geologic cross section of the ORNL area

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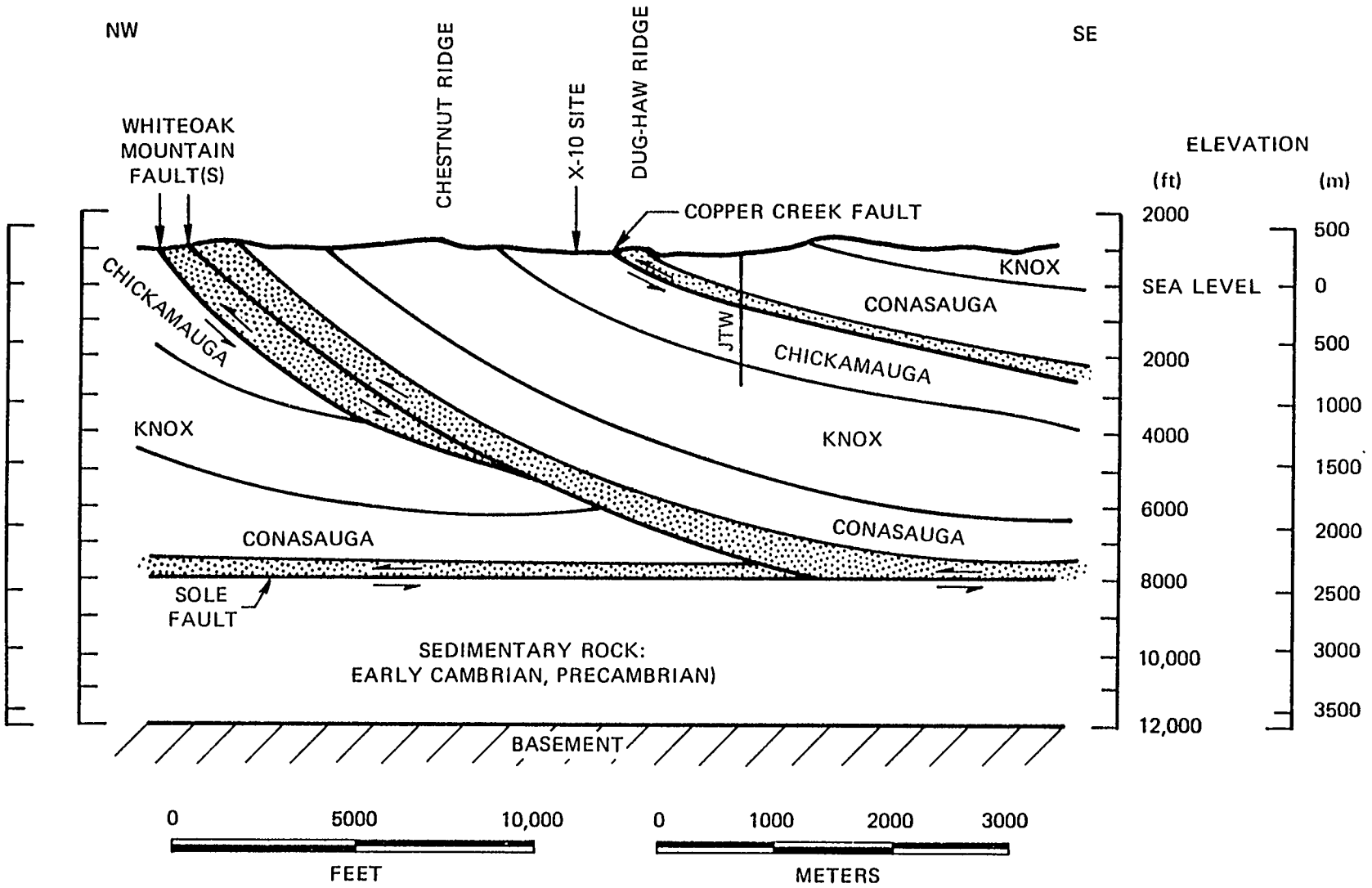

Rome formation and the Knox group. Most of the valleys are underlain by the Conasauga group or Chickamauga limestone. Surface elevations range from about 226 m at the Clinch River on the southern border of the site to about 413 m at the crest of Melton Hill.

Although numerous faults exist within the area, they apparently originated prior to the end of the Paleozoic era. Major tectonic activity seems to have ceased thereafter. No physiographic evidence indicating such activity has been observed along the thrust fault areas in the region. Consequently, there is no reason to expect tectonic activity will occur there in the future.

The Oak Ridge area experiences a moderate amount of seismic activity (5 earthquakes in the last 165 years with a Modified Mercalli intensity of V to VI), though no incidence of surface deformation has been documented. The type of seismic activity that occurs in this region is common throughout the world. No earthquakes of Modified Mercalli intensity VII were reported. Intensity VII earthquakes occur approximately one tenth as often as intensity V to VI earthquakes This suggests a recurrence interval of 300 to 1000 years for intensity VII earthquakes, which are not severe. Examples of damage to be expected from such earthquakes are: weak chimneys broken off at roofline, damage to weak masonry of low standards of workmanship, fall of plaster, and damage to concrete irrigation ditches.

Although a moderate level of seismic activity is experienced in the region, it is highly improbable that a major shock will occur in the Oak Ridge area for several thousand years. Forces from more seismically active areas such as Charleston, South Carolina or Memphis, Tennessee would be dissipated by distances of 600 km or greater.

General characteristics of the major strata in the region are given below.

The oldest formation in the area is the Rome formation. It consists of fine-grained shaley sandstone that is generally well cemented. Its ability to hold and transmit groundwater is due almost entirely to fractures.

The Conasauga group is primarily silty, slightly calcareous, shale interbedded with thin limestone units. The permeability seems to be associated with changes due to weathering of the rock above and somewhat below the water table. Where the shale is saturated, movement of groundwater is concentrated in a zone near the water table and perhaps locally deeper along some fractures. The fractures in the conasauga group, and therefore the permeability, are evenly distributed. This results in a more uniform flow of groundwater than in other formations.

The Knox group is composed largely of thick beds of dolomite and limestone, and where it is not weathered or deformed it has low porosity and low permeability. However, where it has been fractured by folding or faulting its permeability has been greatly increased. In many places these fractures have been enlarged by solution, and large openings have been formed. Sinkholes and caverns are common. These openings give rise locally to springs and wells. The porosity and permeability of the formation are unevenly distributed and have been localized by fracturing and later solution by groundwater movement.

The Chickamauga Limestone is composed of thin beds of shaly limestone and shale. The porosity is low. Fractures have been enlarged by solution but not to the extent of the Knox group. Fractures and solution channels form a network of open channels and voids that permit relatively free movement of groundwater.

Four of the sites discussed in this report are within the Oak Ridge National Laboratory (ORNL) main complex in Bethel Valley, which is underlain by Chickamauga Limestone. These four sites are the Oak Ridge Graphite Reactor, the Low Intensity Test Reactor, the water-to-air heat exchangers, and the experimental facilities associated with the Oak Ridge Research Reactor. The two sites in Melton Valley (the Molten Salt Reactor Experiment and the Homogeneous Reactor Experiment) are underlain by rocks of the Conasauga group. The farthest-south site, the Tower Shielding Facility, is underlain by dolomite and limestone of the Knox group.

2.2 Soils

Soils in the ORNL area are generally of the group known as ultisols, formerly called red-yellow podzolic and reddish-brown lateritic soils. Other soil types are found locally. These are mostly entisols, found in steeply sloping areas, or inceptisols found mainly in alluvial areas adjacent to streams. The ultisols develop in humid areas of temperate to tropical zones on old or highly weathered parent material under forest or savannah vegetation. In general, the soils in the ORNL area are moist, strongly leached, acidic, and low in organic matter. Soil depth in the vicinity of ORNL varies from about 0.15 to 5 m, and entrance of water into the soil can range from less than 0.5 cm/h to 25 cm/h. Soils derived from the Knox group contain kaolinite as their principal clay mineral, soils from the Conasauga group contain illite and vermiculite as principal clay minerals, and soils derived from the Chickamauga Limestone contain a mixture of kaolinitic and illitic minerals with some units having appreciable amounts of montmorillonitic clay minerals. Generally, illitic and vermiculitic clay minerals are more efficient in fixation of potassium

and comparable ions into less available positions than are the kaolinitic materials.

2.3 Hydrology

Oak Ridge National Laboratory is located in the White Oak Creek watershed which drains an area of about 16.5 km^2. Melton Branch is the primary tributary, as seen in Figure 2. Surface runoff from the vicinity of experimental reactor facility sites 9.1 to 9.6 (Table 2) flows into White Oak Creek or one of its tributaries. Surface flow from the Tower Shielding Facility (site 9.7) flows along a natural drain just south of the site and subsequently flows into the Clinch River just below Melton Hill Dam and about 3 to 4 km above the mouth of White Oak creek. The waters of White Oak Creek are impounded by White Oak Dam which is located along the creek just to the northeast of Tennessee State Route 95. The resulting White Oak Lake is a small shallow impoundment. The water level is controlled by a vertical sluice gate that remains in a fixed position during normal operations. The normal lake level since 1960 has been 227 m above mean sea level. The normal surface area of White Oak Lake is about 10 ha, and the retention time is about two days.

Water that drains from White Oak Lake and from the area around the Tower Shielding Facility enters the Clinch River and subsequently the Tennessee River. The Clinch River originates in southwestern Virginia and drains approximately 11,340 km^2. Flow in this system is controlled by the Tennessee Valley Authority with three dams. The nearest upstream dam is the Melton Hill Dam, approximately 3.5 km upstream from White Oak creek. Water level from White Oak Dam to the mouth of White Oak Creek is strongly influenced by the level of the Clinch River which is, in turn,

Figure 2. Map showing the White Oak Creek drainage basin and other creeks in the ORNL area in relation to the Clinch River

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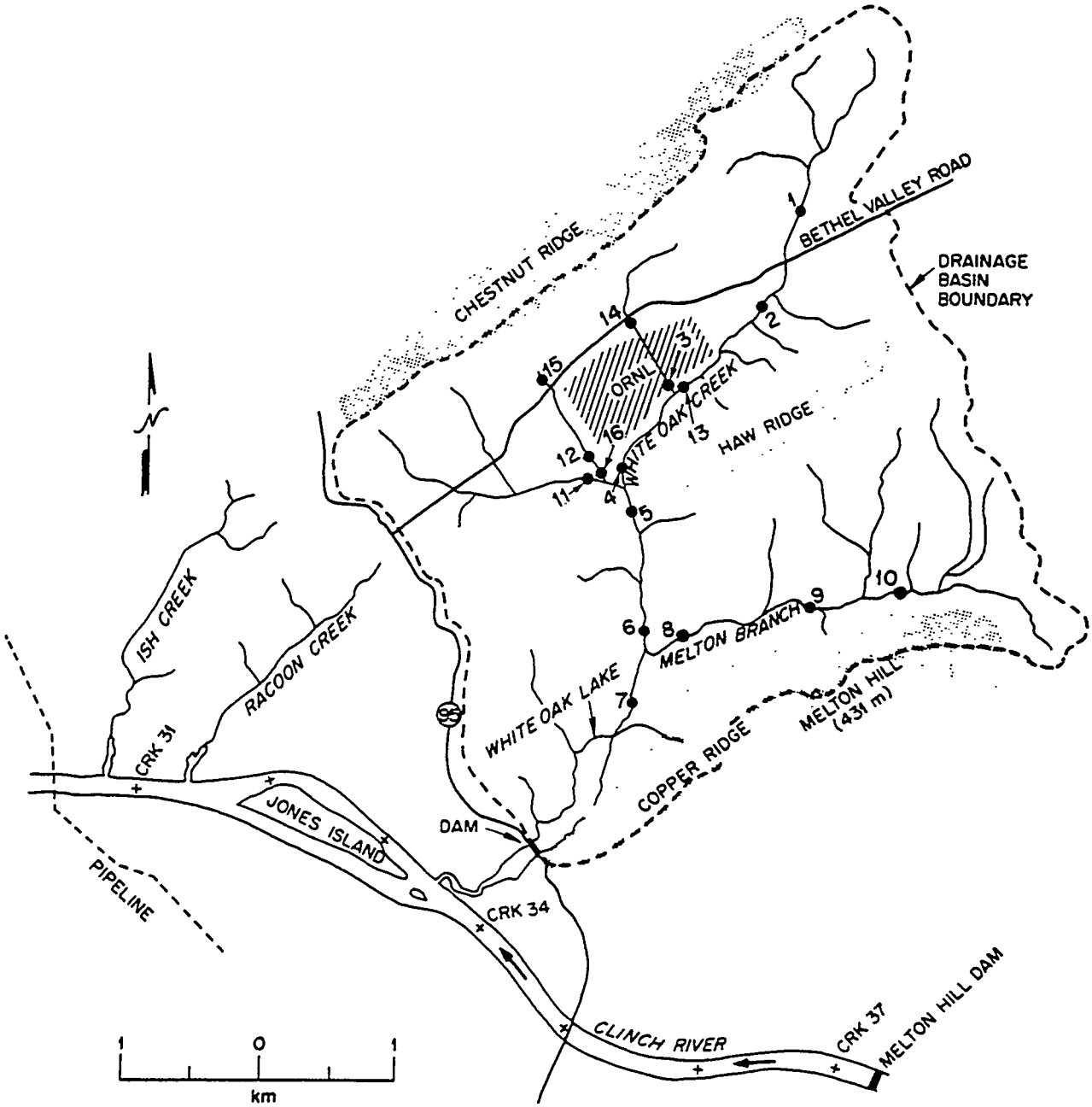

influenced by the pulsating pattern of water witholding and discharge at Melton Hill Dam. Watts Bar Dam on the Tennessee River creates backwaters that can affect White Oak Creek by raising the water level between the mouth of the creek and White Oak Dam. This normally occurs during the summer months (April–October) when the Watts Bar Reservoir pool surface is maintained at a higher elevation. During the winter the area between the mouth of White Oak Creek and White Oak Dam often resembles a mudflat.

Water quality of surface streams on the Oak Ridge Reservation is, generally of the calcium–magnesium–bicarbonate type, reflecting the limestone geology of the area. Water quality and ecology of streams near Oak Ridge National Laboratory have, however, been altered as a result of industrial effluents.

There are nine public water supply systems that withdraw surface water from within an approximately 32 km (20 mi) radius of Oak Ridge National Laboratory. Only one of these is downstream of White Oak Dam. The intake for the city of Kingston is located along the Tennessee River, about 0.6 river km above the confluence of the Clinch and Tennessee rivers and about 34 river km below White Oak Dam. Surface waters in the vicinity of Oak Ridge National Laboratory, including the Clinch River, are important resources for research, recreation (boating, fishing, swimming), and municipal and industrial activities. Most of the surface waters near the experimental reactor facilities originate from contributing springs in the limestone on the ridge slopes.

Over 100 private wells and springs and 8 industrial and 16 public groundwater supplies exist within about 32 km of Oak Ridge National Laboratory. However, due to the stratigraphic and structural control of groundwater flow in the region, groundwater beneath the sites is expected

to migrate along the strike and to discharge into surface water bodies. There is a low probability of groundwater migration from Laboratory facilities to offsite wells. Further details of surface- and groundwater withdrawls within about a 32 km (20 mi) radius of ORNL can be found in Boyle et al (1982).

2.4 Climatology

The climate of the Oak Ridge area may be broadly classified as humid continental. The Cumberland Mountains to the northwest and the Great Smoky Mountains to the southeast both influence the climate of the area around ORNL. The Cumberland Mountains serve to retard the flow of cold winter air that often penetrates far south over the plains and prairies in the Central United States. The prevailing surface winds usually blow up the valley from the southwest or down the valley from the northeast. Local topography reduces wind speed substantially. Severe wind storms and tornadoes are rare in the Oak Ridge area which has one of the lowest tornado probabilities in the state. Air stagnation, however, is relatively common. Periods of air stagnation that have the potential for being multi-day air-pollution episodes occur frequently in eastern Tennessee, relative to most other areas of the eastern United States. About two such episodes occur per year, on the average, to cover an average of about 8 days per year (Korshover 1976).

The approximately 140 cm of precipitation that falls in an average year is distributed quite equally over the months, relative to other regions in the eastern United States. The driest month is October with an average of about 7 cm of precipitation. July and the winter months are

the wettest, averaging about 15 cm. Monthly total precipitation amounts exceeding 25 cm can occur in any season of the year, though such extremes are least likely in the autumn. The largest amount ever recorded in October was about 16 cm. At least a trace of snow has been reported in each winter of record. Amounts in individual winters have ranged from only a trace to over one meter. The average amount is about 26 cm, or about 26 mm water equivalent. Snow from any single storm will usually melt in a few days. Episodes of freezing rain are not uncommon during winter, especially when cold air is trapped in the valley and relatively lighter (warm and moist) air from the Gulf of Mexico is advected at higher elevations. Thus, it may be raining in the Cumberland or Great Smoky Mountains while freezing rain is occurring in the valley between.

Temperatures seldom rise much above 35°C or drop below 0°C. However on January 21, 1985, the temperature dropped to −27.7°C to provide the record minimum. The record maximum is 40.6°C.

2.5 Ecology

The ecological systems around Oak Ridge National Laboratory are typical for the Appalachian region. The area was primarily agricultural land before federal acquisition in 1942, and much of the area has since reverted back to natural plant cover. The dominant plant community is oak–hickory forest, though elements of mixed mesophytic (buckeye, beech, maple, etc.) are also often present in coves and on sheltered slopes. Although most narrow, steep-sided, perennial stream drainages near ORNL have mixed mesophytic characteristics, the type is well developed and extensive only on a few steep north–facing slopes and bluffs. The laboratory area is a developed site surrounded by grasslands.

2.5.1 Terrestrial Flora:

Over 800 species, subspecies, and varieties have been identified as present in the area. Approximately one-third of the area either has been planted in pine, or is in natural pine stands. In many areas the natural pine stands are being replaced by oaks, hickories backgum, sourwood, and occasional white pines. Some areas, such as pipelines, power-line corridors, and a few field research sites are maintained as open fields. No plants on the federal list of endangered species have been observed near the Laboratory. The nearest such species is the green pitcherplant (Sarracenia oreophila) which has been recorded about 65 km to the north. The state of Tennessee has separate lists of plant species that are endangered, threatened, or of special concern within the state. Three plant species that occur near ORNL are listed by the state as endangered, threatened, or of special concern within the state. Three plant species that occur near ORNL are listed by the state as endangered and four are under review for inclusion on the federal list of endangered species (Parr 1984). These plants occur in the general vicinity of the Laboratory, but are in areas designated as natural areas within the Oak Ridge National Environmental Research Park and are not affected by activities in the developed areas around the experimental reactor sites.

2.5.2 Fauna:

There are seven federally listed endangered species that have ranges encompassing the Laboratory or for which suitable habitat may exist near the Laboratory. The bald eagle is the only endangered species that has been conclusively recorded in the area. Most of these bald eagles are probably of the northern races that are listed as threatened rather than

endangered, but eagles seen in summer may originate from the endangered breeding population in Florida, where nesting occurs in winter. No eagles are known to nest near the Laboratory. In addition to the federal list of endangered animal species, the Tennessee Wildlife Resources Agency lists five species as being endangered within the state. None of these species appear to have been significantly affected by activities of the Laboratory. Detailed information on endangered species in and around the Laboratory is given by Boyle et al. (1982).

2.5.3 Aquatic Ecology:

A survey of the aquatic ecology around ORNL was carried out during parts of 1979 and 1980 by Loar et al. (1981). Much of what follows is based on their report and supplemented by subsequent observations.

The upper part of White Oak Creek, above the ORNL site, is believed to resemble conditions before construction of ORNL. There were 24 taxa of benthic invertebrates reported, which represented the greatest diversity of species in anywhere in the creek. Fish identified were mostly stoneroller and blacknose dace, which are typical of small southeastern streams.

Between ORNL and White Oak Lake the benthic invertebrates were dominated by midge larvae. Bluegill and mosquitofish were found, but there was little or no evidence of spawning and no fish were found at one of the sampling locations. Also, no fish were found in the Melton Branch.

In White Oak Lake, the benthic invertebrates were mostly midge larvae. Fish included bluegill, sunfish, mosquitofish, and largemouth bass. Other species are also believed present.

Between White Oak Lake and the Clinch River the benthic invertebrates are still dominated by midge larvae, but the asiatic clam is also present as it is in the Clinch River. The abundance of fish eggs strongly suggests that this is a spawning area. The fish population somewhat resembles that of the Clinch River and includes gizzard shad, and channel catfish, with lesser abundances of sauger and bluegill.

The benthic invertebrates of the Clinch River near ORNL are usually dominated by midge larvae, though aquatic earthworms and asiatic clams are occasionally dominant. Gizzard shad were the most abundant of the 21 species of fish identified. Several popular sport fishes were also identified. The Clinch River near White Oak Creek appears diverse, productive, and free of excessive loadings of nutrients or organic material.

2.5.4 Establishment of Wildlife Management Area

The Oak Ridge Wildlife Management Area has been established on the reservation as the result of the Cooperative Agreement Between the Department of Energy and the Tennessee Wildlife Resources Agency for the Establishment of a Wildlife Management Area, November 30, 1984. This Agreement enables DOE and the Tennessee Wildlife Resources Agency to work together in areas of wildlife management and research. Results of this program include the establishment of a series of organized deer hunts and the release of several wild turkeys as part of an effort to reestablish turkey populations within the state.

2.6 References

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3. REGULATORY STATUS AND HAZARD RANKING

Hazard Ranking System analyses for the experimental reactor facilities have not been performed, because these facilities are regulated under Department of Energy Order 5820.2 which has no provisions for hazard ranking.

4. DESCRIPTION OF THE EXPERIMENTAL REACTOR FACILITIES

This section summarizes pertinent information about each of the seven sites listed as Experimental Reactor Facilities (Table 2). Sites 1-6 fall under the Surplus Facilities Management Program. The Tower Shielding equipment (Site 7) that is surplus consists of several cubic meters of "junked" equipment that has simply been left to lie around the site. This equipment falls under the Surplus Contaminated Facilities Program.

4.1 Site 9.1: Oak Ridge National Laboratory (ORNL) Graphite Reactor

The following material is mostly abstracted from the Preliminary Decommissioning Study Report (Reed 1984) pertaining to this site, and from associated references and appendices as indicated.

4.1.1 Purpose of Facility

The ORNL Graphite Reactor was the world's first continuously operated nuclear reactor, and is today a registered historic landmark. It was built to produce the first gram size quantities of plutonium to provide information for the construction of the large plutonium production reactors located at Hanford, Washington. Although originally designed for a 1 MW power level, in 1944 improvements in the cooling system and fuel cladding allowed the power level to be increased to an average rate of 3.6 MW. Following its contribution to the Hanford work, the facility was used for training reactor operators and for several experiments. After exactly 20 years of operation, the reactor was shut down on November 4, 1963. In 1966 the fuel was removed from the reactor.

4.1.2 Location and Description of Site

The ORNL Graphite Reactor (OGR) is located in a five-story corrugated metal structure (Figure 3) designated as Building 3001 and located as site 4 on the map in Figure 4. Associated structures include the filter building (Building 3002), the fan house (Building 3003), and an approximately 60 m (200 feet) high stack (numbered 3018). These associated structures are all located near to the north side of the reactor building (Figure 5). The floor plans for the reactor building (3001) are shown in Figures 6, 7, and 8.

Besides these associated structures, there is a subsurface, water-filled, transfer canal that was originally constructed for the purpose of transferring fuel slugs from Building 3001 to Building 3019-A (Radiochemical Processing Pilot Plant). Details and drawings of this canal are given by Reed (1984). The canal runs southward from the west

Figure 3. View of Building 3001, looking north

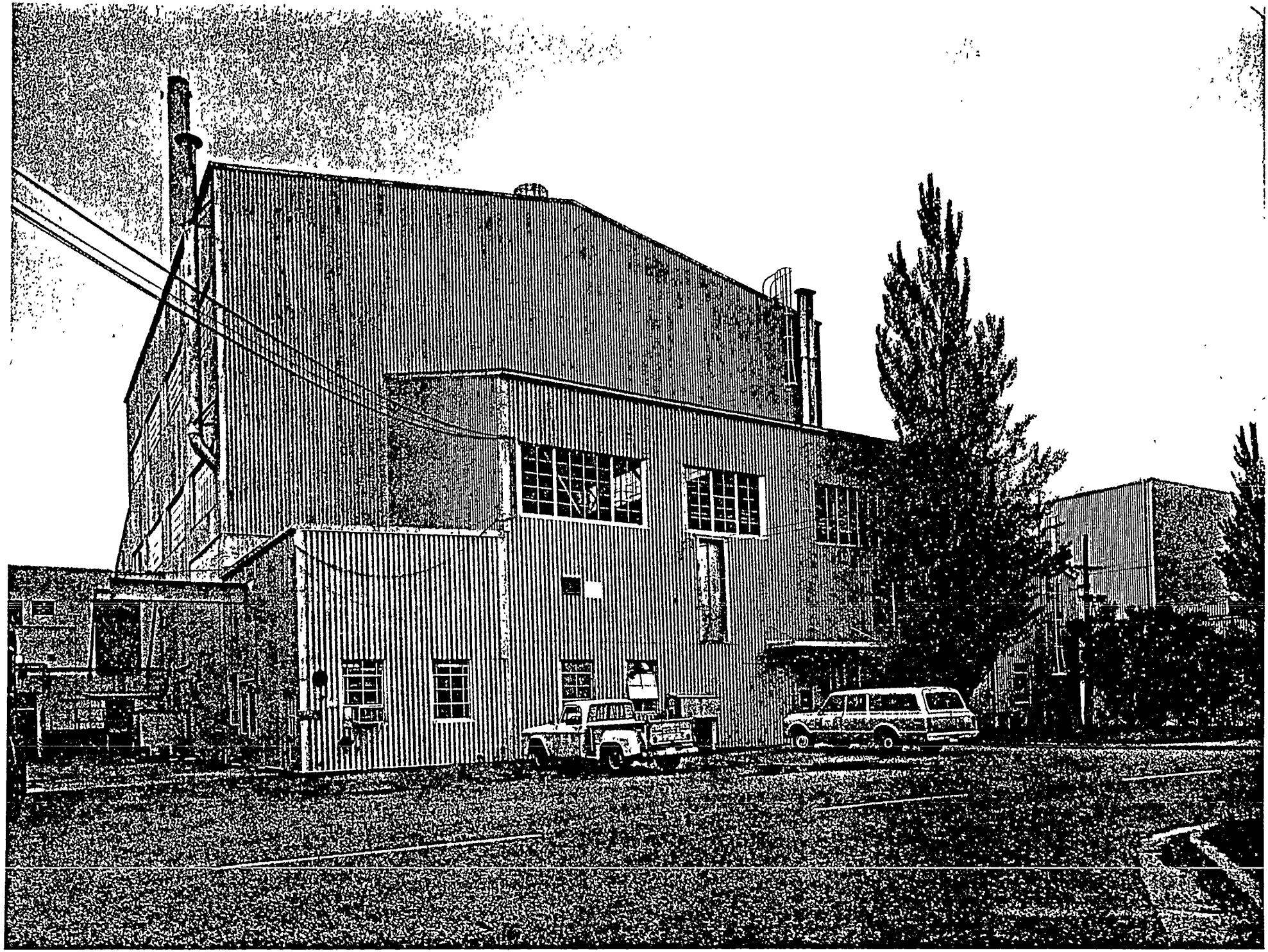

Figure 4. Locations of some of the experimental reactor facilities that have been designated as remedial action sites at ORNL.

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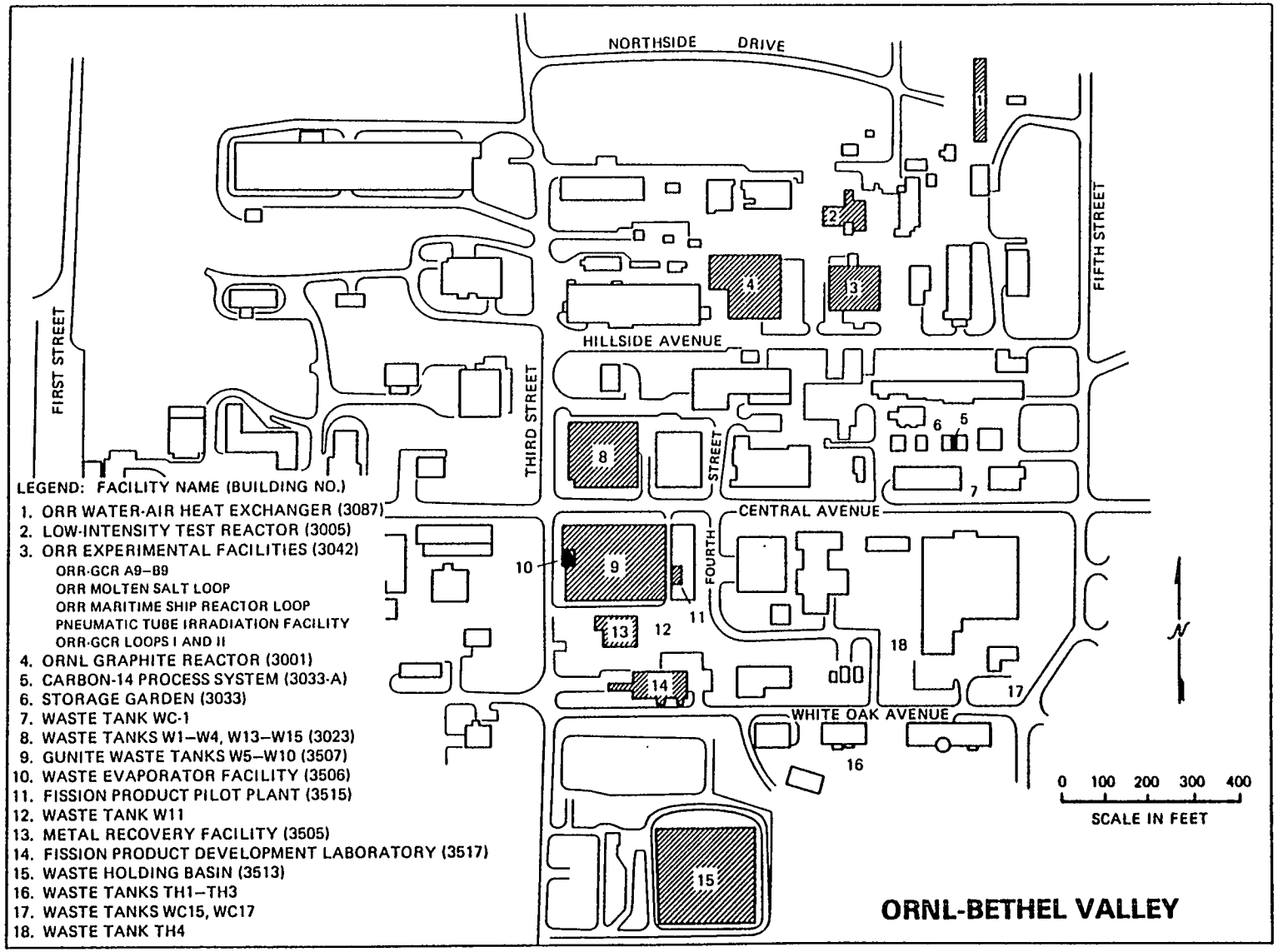

Figure 5. Aerial view of Building 3001 and nearby buildings, looking southwest. Building 3001 is the large building with the dark roof (near center). Just to the north (right, in photo) of 3001 is the 3018 stack (closest stack to viewer) which has its base in the southwest corner of the fan house (3003). Just west of the the fan house is the filter house (Building 3002) with a van parked on its north side. Other buildings shown in this photo are 3019 (just west of the southern half of Building 3001); Building 3042 (large building just east of 3001) that houses the Oak Ridge Research Reactor and associated experimental facilities; and the Low Intensity Test Reactor (Buildings 3004 and 3005) just to the north of 3042 and east of 3003.

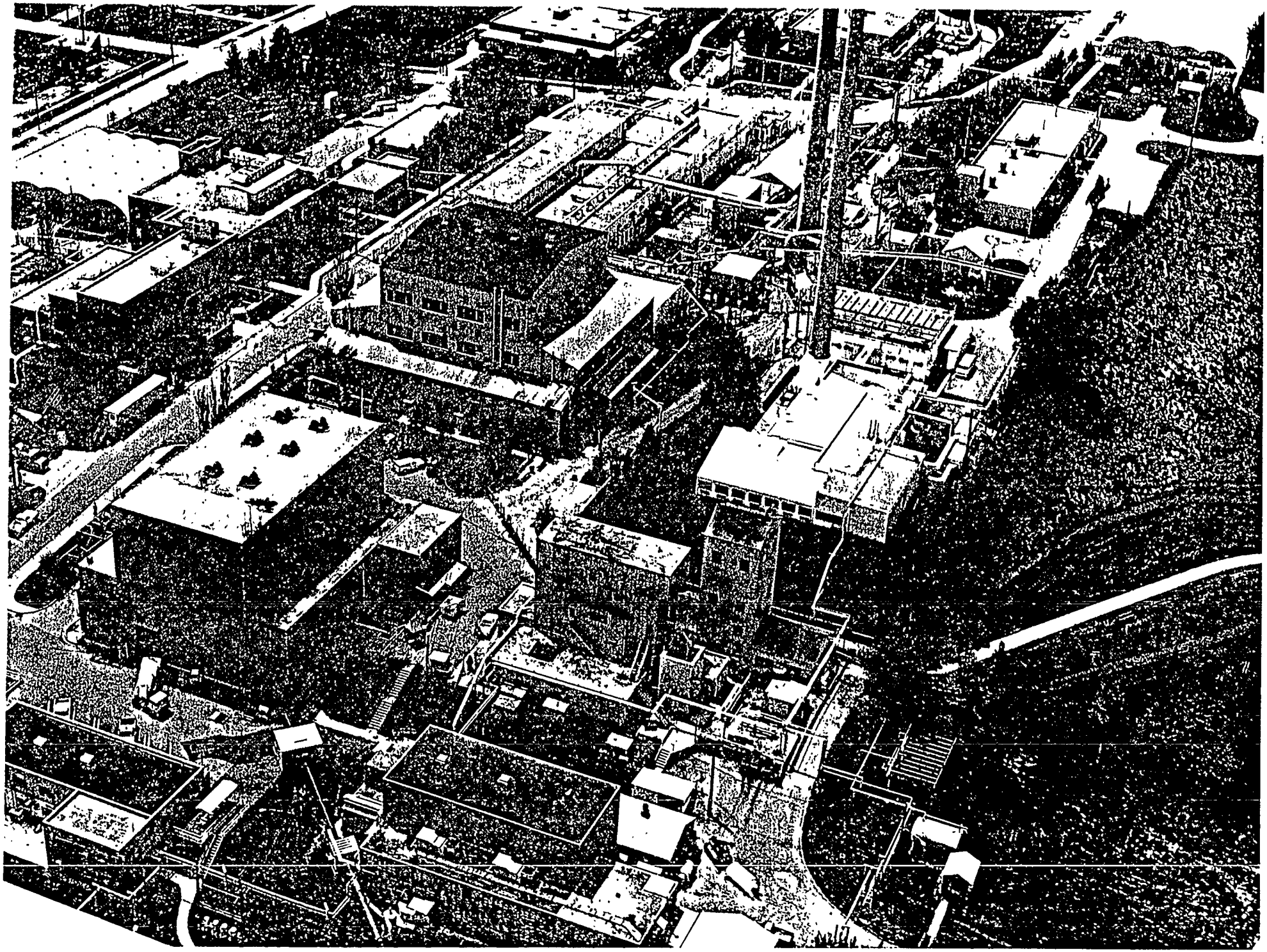

Figure 6. First-floor plan of the ORNL Graphite Reactor (Building 3001)

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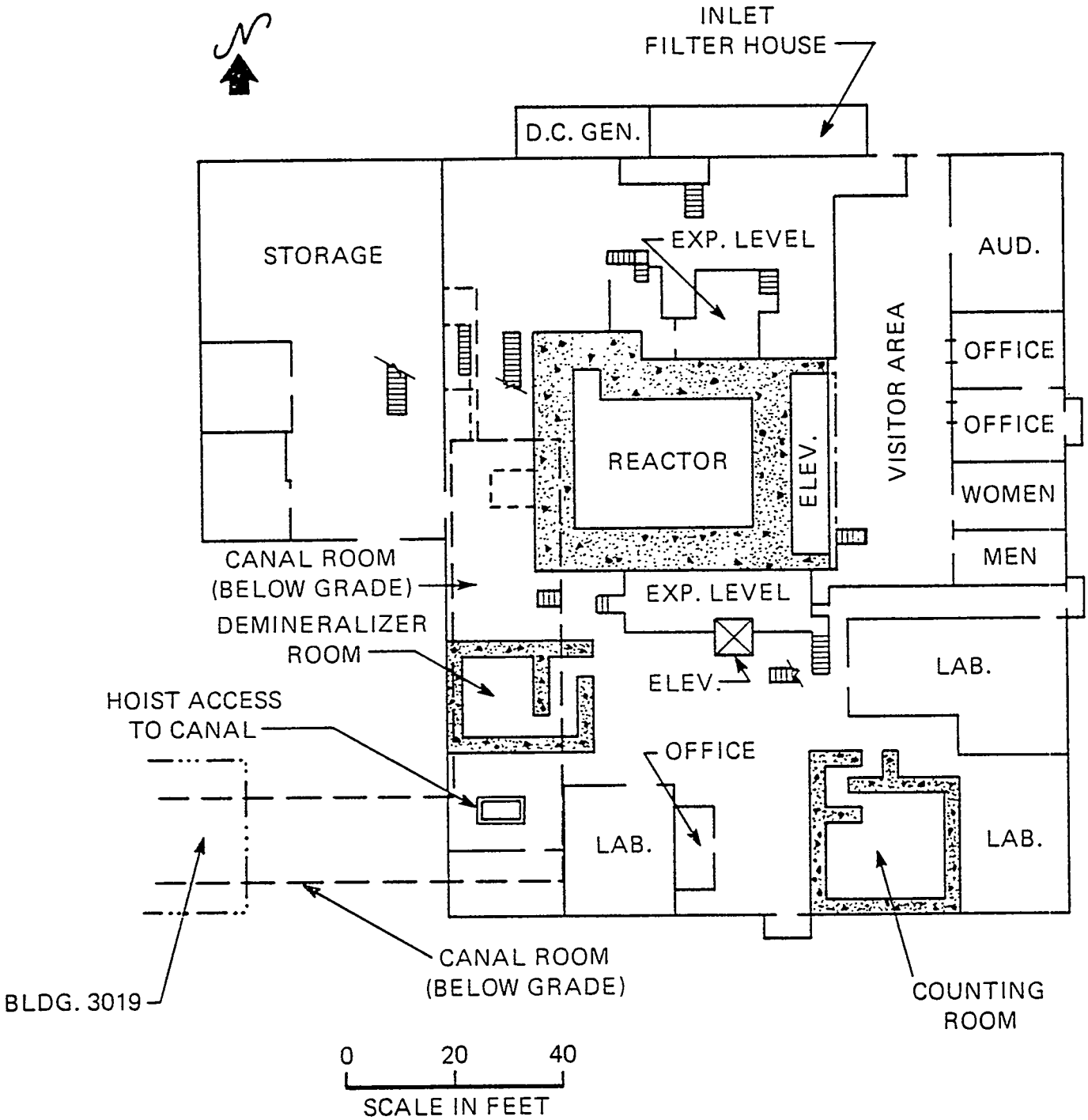
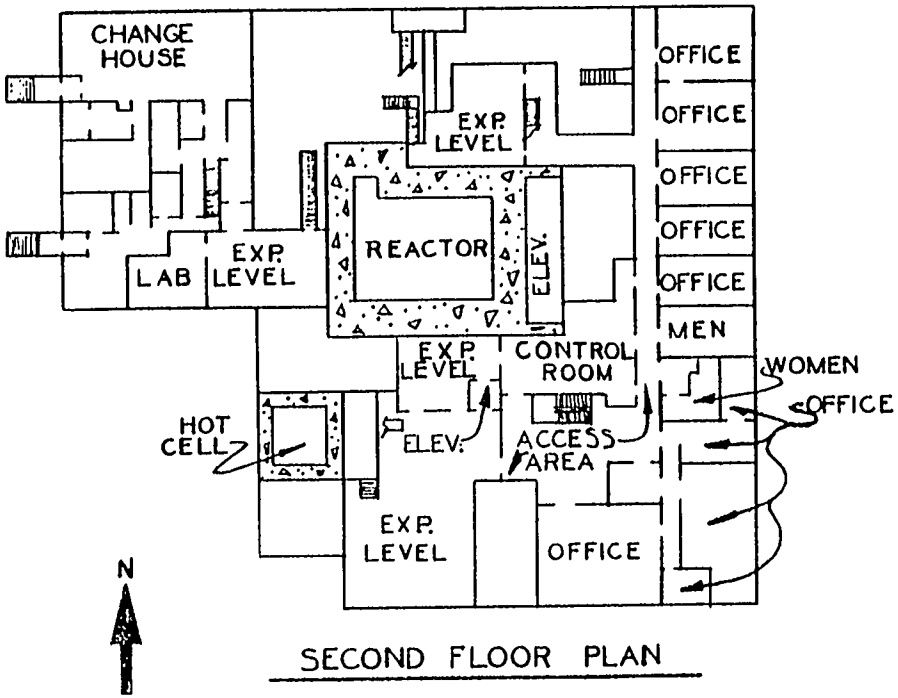
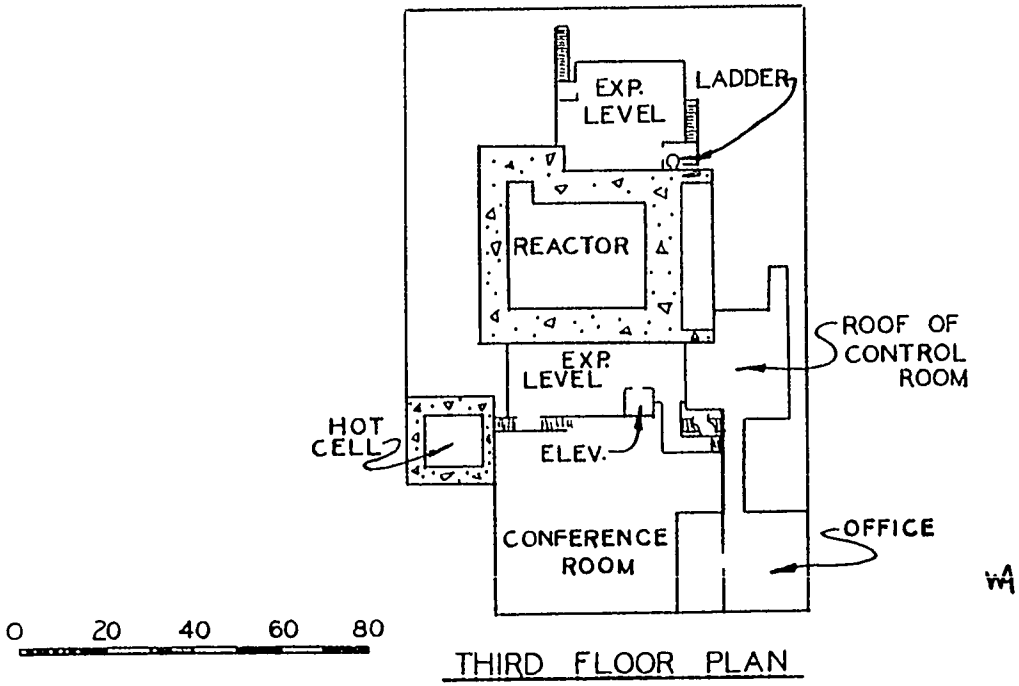


Figure 7. Second-and third floor plans of the ORNL Graphite Reactor (Building 3001)

ORNL DWG 76-12390



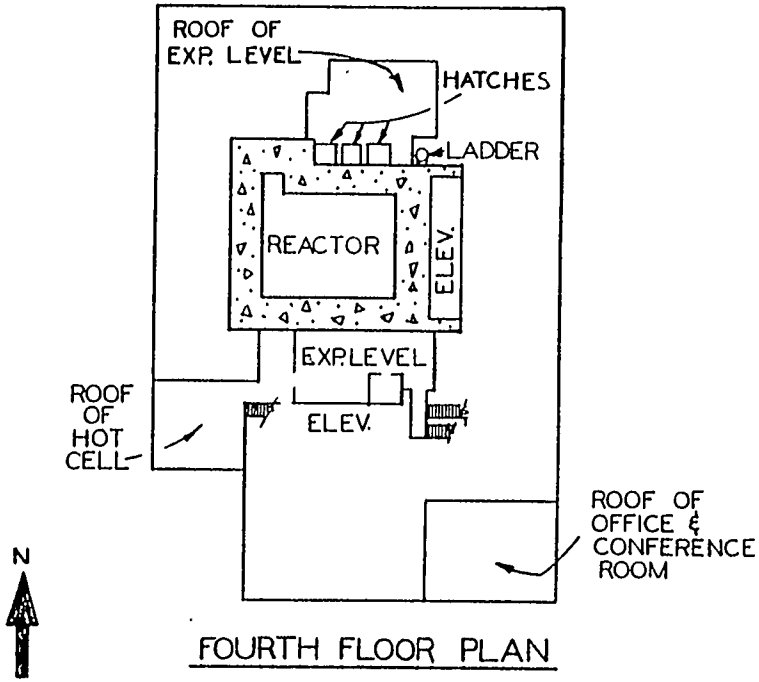
SECOND FLOOR PLAN



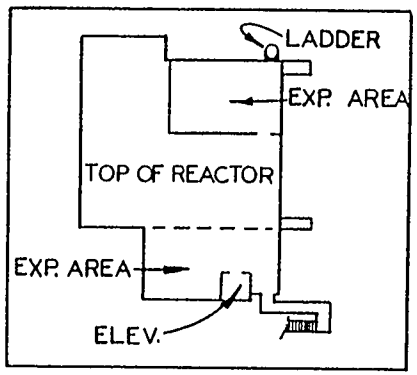
THIRD FLOOR PLAN

Figure 8. Fourth- and fifth-floor plans of the ORNL Graphite Reactor (Building 3001)

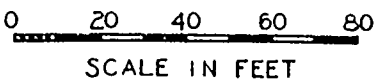
ORNL DWG 76-12391



FOURTH FLOOR PLAN



FIFTH FLOOR PLAN



side of the reactor for about 20 m (66 feet) and then westward for about 13.5 m (44 feet) to building 3019. The canal is about 2 m wide and 4 m deep, and is flanked by two walkways. In recent years radioactive material has been stored in this canal.

4.1.3 Description of Contamination

Detailed descriptions of the contaminants are found in Reed (1984) and Simpson et al. (1984).

Building 3001 (Reactor Building): The reactor is reported to contain 16 curies of carbon-14, 5000 microcuries (80 mg) of plutonium-239, 200 microcuries (<0.5 Kg) of uranium oxide, and lesser radiation sources as described in Simpson et al. (1984). A gamma spectroscopy scan indicated that cesium-137 and cobalt-60 are present. An order-of-magnitude estimate of less than 10 Ci of each is provided (Reed 1984). Thus, the radionuclide inventory in the reactor probably totals less than 50 curies. The demineralization room (or "hot cell") was checked and cesium-137 and cobalt-60 were detected in the gamma spectrum. Five percent of the alpha readings (only one reading) were above the ORNL guidelines for establishing a "contamination zone" (Simpson et al. 1984). The beta-gamma contamination levels ranged from < 500 dpm/100 cm^2 to 1,670 dpm/100 cm^2 with an average of approximately 500 dpm/100 cm^2. The only alpha contamination was found at a level of 32 dpm/100 cm^2. The transfer canal in the basement of the reactor building is reported to contain approximately 112,000 Ci of strontium-90 stored in cans (9 cans, approximately 0.35 m^3 total volume, both values are estimates) and approximately 50,000 Ci of cobalt-60 pellets stored in containers (Reed 1984). Also stored in the canal are various empty buckets, drums, scrap pipe, etc. A more complete list of these items is given in the decommissioning study of Reed (1984).

There are three subsurface air ducts connecting Buildings 3001, 3002, and 3003 that have been contaminated. Cesium-137, strontium-90 and plutonium-239 are all apparently present. Contamination levels, as estimated from wall scrapings (in 1976), were around 0.8 to 0.9 microcuries/gm (of cesium-137 and strontium-90). Plutonium-239 contamination was similarly estimated at 1 to 12 nanocuries/Kg throughout the duct system (Simpson 1984). Dose rates in the interior of the duct work between the fans and the stack were found to be up to 500 mrad/h with transferable beta-gamma contamination levels up to 300,000 dpm/100cm^2 (Cagle and Pugh, 1976, Simpson et al. 1984).

Building 3002: The survey of accessible areas of Building 3002 showed transferable contamination of 1760 dpm/100cm^2 beta-gamma, and 56 dpm/100cm^2 alpha.

Building 3003: No significant contamination was detected.

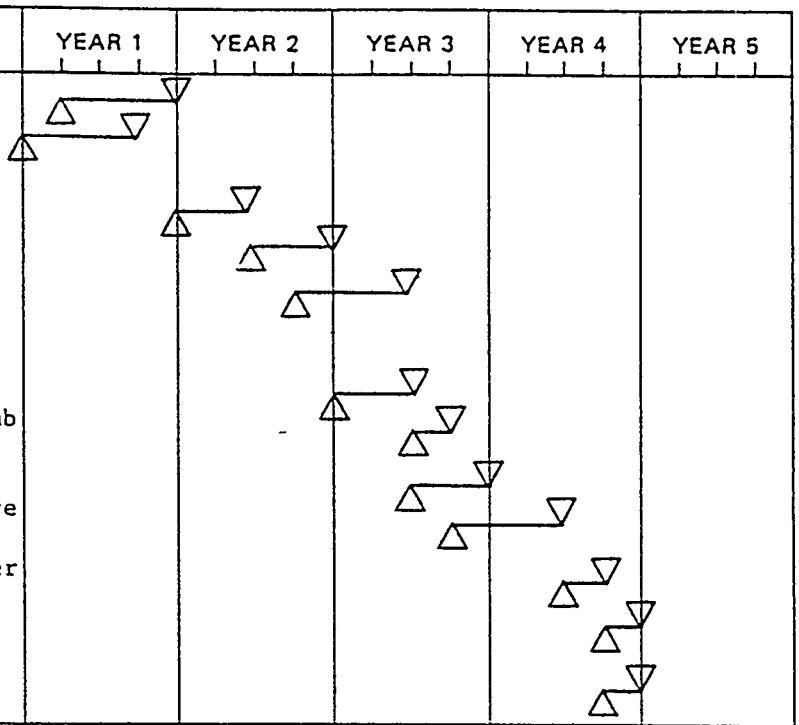
Stack: Rainwater collected in the stack is cleaned out from time to time. Reed (1984) refers to a cleanout about 1980 in which the water was sufficiently low in radioactivity that it could be sent down the process drain, and that there is "good likelihood that it could be sent to the storm sewer during the next cleanout".

In 1959, there was some plutonium contamination from Building 3019 that affected the surrounding area, including buildings in the area of the reactor building. Some of the contamination was fixed with paint or tar (on the roof). Some of this contamination could conceivably become exposed during a demolition.

Further details of the contamination at this site are given in Simpson et al. (1984) and Reed (1984). An itemization of the wastes expected to be generated during decommissioning is given by Reed (1984) and summarized in Table 3.

Table 3. Timing, cost, and waste-volume projections for decommissioning the ORNL Graphite Reactor. The cost estimates are in first-quarter FY 1985 dollars. (Source: Myrick 1984)

ORNL-DWG 83-14840R3



	YEAR 1	YEAR 2	YEAR 3	YEAR 4	YEAR 5
PROJECT COSTS (SK)					
Annual Expense	390	1275	3700	2560	
Capital Equipment	–	–	50	25	
Total	390	1275	3750	2585	
WASTE VOLUME PROJECTIONS					
Solid (m^3)					
Soil (LLW)	–	–	–	–	
(TRU)	–	–	–	–	
Rubble (LLW)	–	–	10	1055	
(TRU)	–	–	–	–	
Metal (LLW)	–	–	110	5	
(TRU)	–	–	–	–	
Misc. (LLW)	–	–	15	60	
(TRU)	–	–	–	–	
Total (LLW)	–	–	135	1120	
(TRU)	–	–	–	–	
Liquid (m^3)					
Process Waste	–	–	–	–	
LLW	–	–	475	70	

Simpson et al. (1984) also stated that "a review of the OGR and a walk-around survey of the exterior of Building 3001 indicated no environmental concern associated with the present status of the OGR facility. Absorbed dose rates around the facility were in the range of 0.02-0.05 mrad/h, typical background values for this area."

4.1.4 Data Deficiencies

There is asbestos packing between the graphite and concrete in the reactor core. This asbestos is safely confined and will not present a hazard until the reactor is dismantled. However, other sources of asbestos may exist at the OGR that were not evident in this survey.

There also exists a potential of encountering entrapped radionuclides in the paint and concrete. Literature describing the plutonium release incident at Building 3019 indicate that Building 3001 was heavily contaminated in this event. The interior of Building 3001 was completely cleaned; however some exterior areas of the facility, particularly the roof, were coated with tar or paint to prevent the spread of contamination (King and McCanley, 1961). This contamination would not be detectable by routine-portable survey instrumentation due to the short range of the alpha radiation, but would require a large number of samples drilled or cored from the exterior walls and roof or the use of special low-energy gamma spectroscopy equipment. This survey would be required when the exterior of the building is dismantled and the contamination possibly exposed.

Decommissioning of the canal will require that buckets of highly radioactive materials currently stored there be shielded and removed. A more detailed survey can then be conducted to locate isolated areas of radioactivity on walls, floors, or loose debris on the bottom of the canal.

4.1.5 Decommissioning Options

The decommissioning study of Reed (1984) reported an estimate of 4 years of time and $8,000,000 (FY 1985 first-quarter dollars) to decommission the site (Table 3). Because the reactor is a Registered National Historic Landmark, decommissioning should preserve the principal structure and enough of its contents so that visitors can associate the site with the event that gave it historical significance. According to the decommissioning study of Reed (1984), the major decommissioning activities should be primarily in areas that will not be visible to visitors once the work is accomplished.

4.1.6 References:

Cagle, C. D., and L. P. Pugh. 1976. Decommissioning Study for the ORNL Graphite Reactor. ORNL/CF-76/196. Oak Ridge National Laboratory, Oak Ridge, Tennessee.

King, L. J., and W. T. McCarley, 1961, Plutonium Release Incident of November 20, 1959, ORNL-2989, Oak Ridge National Laboratory, Oak Ridge, Tennessee 37831.

Myrick, T. E. 1984. The ORNL Surplus Facilities Management Program Long-Range Plan. ORNL/TM-8957, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Reed, W. R. 1984. Preliminary Decommissioning Study Reports: Volume 7: Oak Ridge Graphite Reactor. X-OE-231, Vol. 7. Oak Ridge National Laboratory, Oak Ridge, Tennessee 37831.

Simpson, D. R., J. H. Pemberton, and R. C. Cooper. 1984. Preliminary Radiological Characterization of the Oak Ridge Graphite Reactor (OGR) Facility (Building 3001, 3002, and 3003). ORNL/CF-84/30. Oak Ridge National Laboratory, Oak Ridge, Tennessee 37831.

4.1.7 Persons Knowledgeable about the Site:

T. P. (Tom) Hamrick (ORNL), 615/574-7078

4.2 Site 9.2: Molten Salt Reactor Experiment

Much of the following material is abstracted from the Preliminary Decommissioning Study Report for this facility (Peretz 1984) and associated references and appendices as indicated.

4.2.1 Purpose of Facility:

The Molten Salt Reactor Experiment (MSRE) was an 8 MW (thermal) fluid-fueled reactor constructed to demonstrate the molten-salt breeder concept for commercial power generation. It operated from 1965 to 1969.

The experiment made use of existing facilities that had been constructed during the 1950s for use in the Aircraft Nuclear Propulsion program. The reactor building (Building 7503) was constructed in 1951 to house the Aircraft Reactor Experiment. That experiment was cancelled in 1957. In 1961 construction began on the Molten Salt Reactor Experiment, using Building 7503 and support equipment left over from the Aircraft Reactor experiment. Reactor operation was completed in December 1969.

4.2.2 Location and Description of Site

The site is located on Melton Valley Drive, about 1 km (0.6 mile) south of the main ORNL complex. It is the most northeastern site in Figure 9. The main reactor building (7503) houses the reactor and support cells. An aerial view of the facility, looking south, is shown in Figure 10. Figure 11 is a schematic guide to the structures comprising this facility. Figure 12 shows a schematic vertical cross section of the main plant (Building 7503), and Figures 13 and 14 show, respectively, the basement and first-floor plans of Building 7503. Preliminary radiological surveys, including subsurface soil corings, indicated that the only appreciable contamination outside building 7503 involves the off-gas filter and charcoal pits, blowers and stack. Other structures in the facility will therefore not be discussed here. A LLW pumping station is just west of the 7500 area, and LLW collection tank WC-20 is across Melton Valley Drive. Underlying bedrock in Melton Valley is Conasauga Shale.

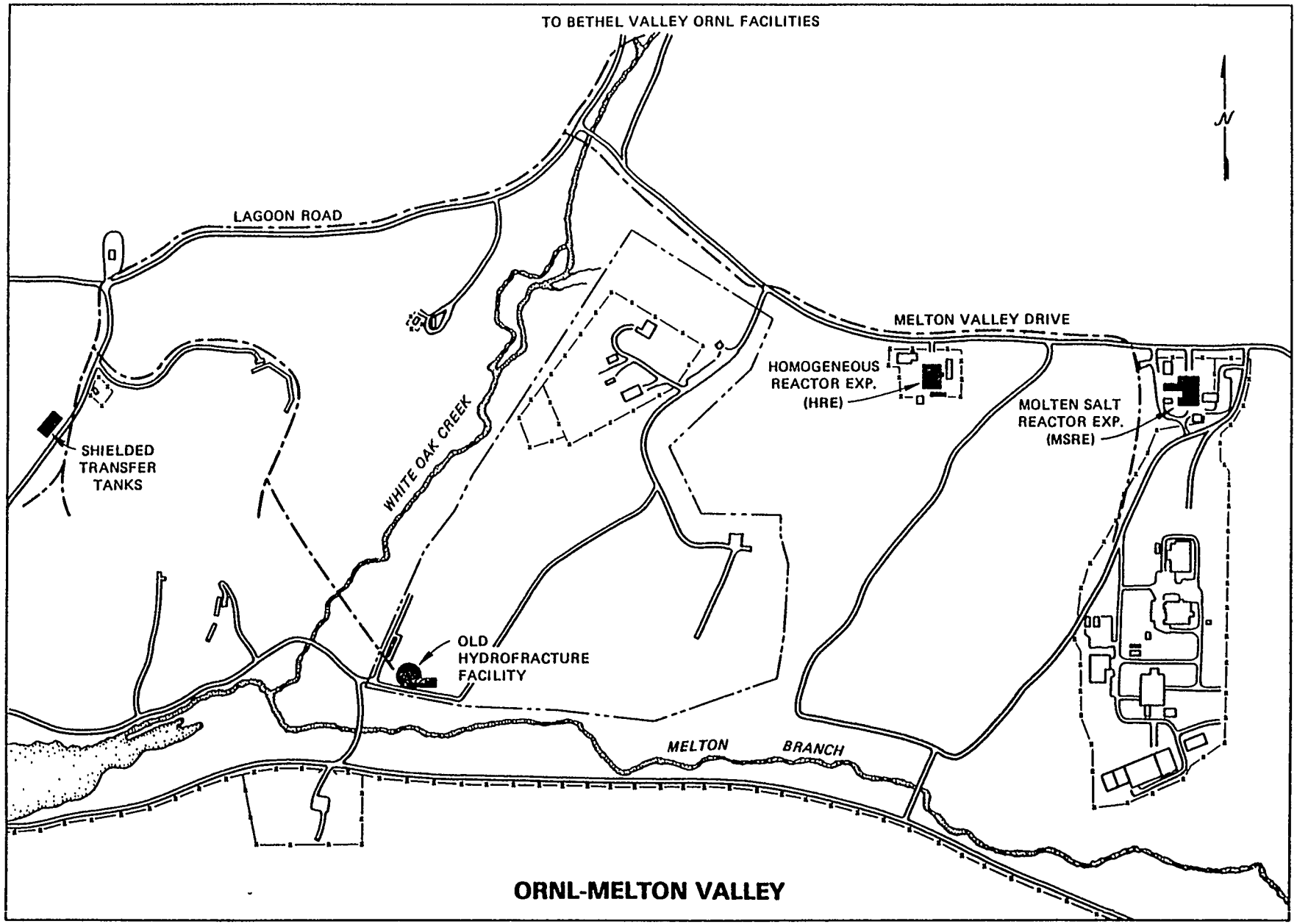
Building 7503 is a large building of irregular dimensions, resulting from additions and modifications for its various uses over the last 35 years. The filter house (Building 7511) is a small (approximately 8 m by 6 m) pit with concrete walls approximately 0.3 m thick.

4.2.3 Description of Contamination

Following reactor shutdown, the fuel salt was drained into two tanks and a flush salt was run through the reactor and drained into a separate tank. A study of options for disposal of the salts was conducted (by Haubenreich and Lindauer (1972). Three salt batches remain at MSRE. The fuel salt is contained in two fuel-drain tanks: FD-1 (2480 Kg of salt) and

Figure 9. ORNL Facilities in Melton Valley. The two experimental reactor facilities are in the northeast section, along Melton Valley Drive

ORNL-DWG 82-19186R2



ORNL-MELTON VALLEY

Figure 10. Aerial view of the Molten Salt Reactor Experiment facilities, looking south. Building 7503 is the large building near the center. Figure 11 provides a schematic that can be used to identify specific facilities in this photo.

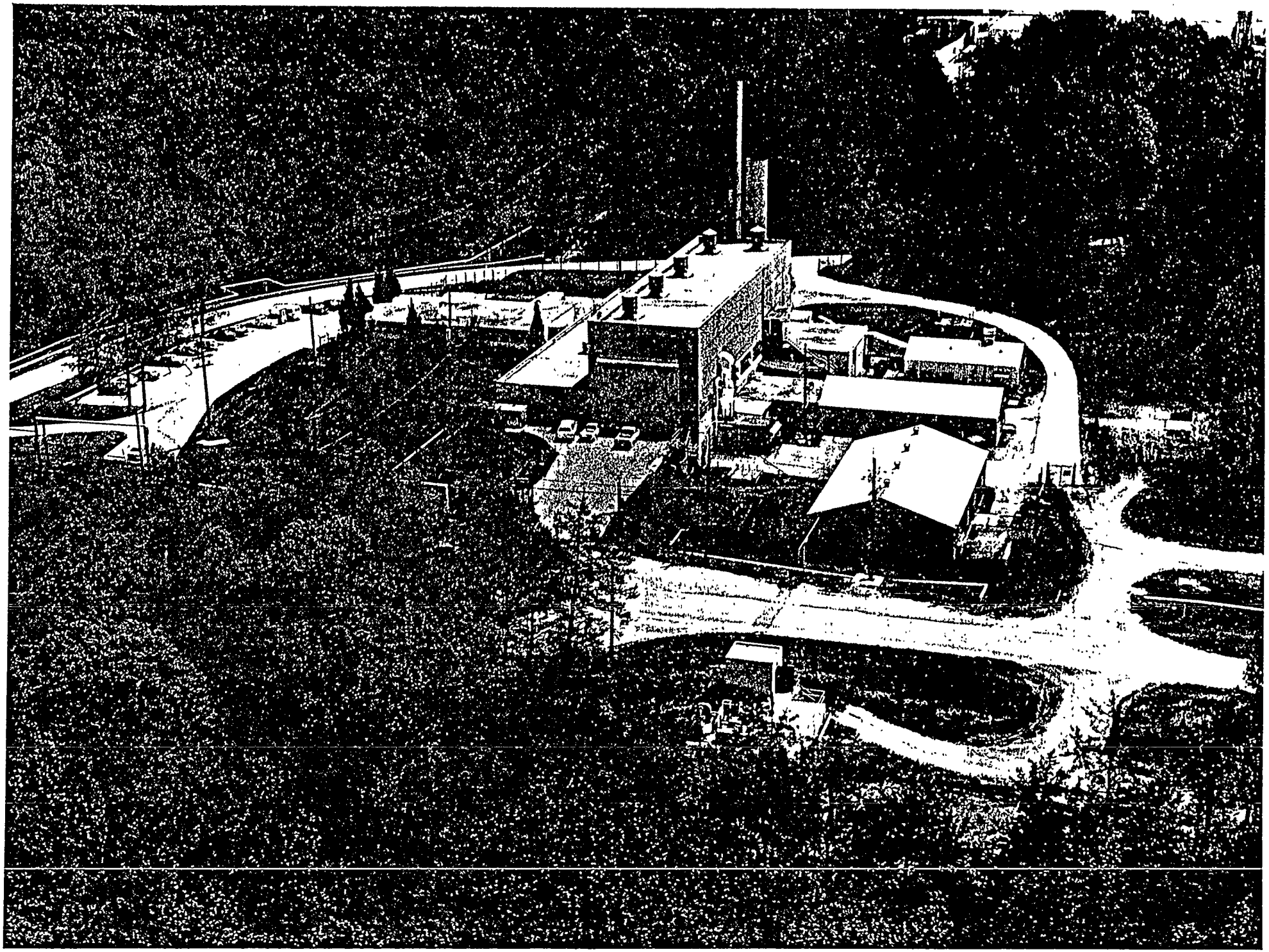


Figure 11. Schematic layout of the Molten Salt Reactor Experiment facilities

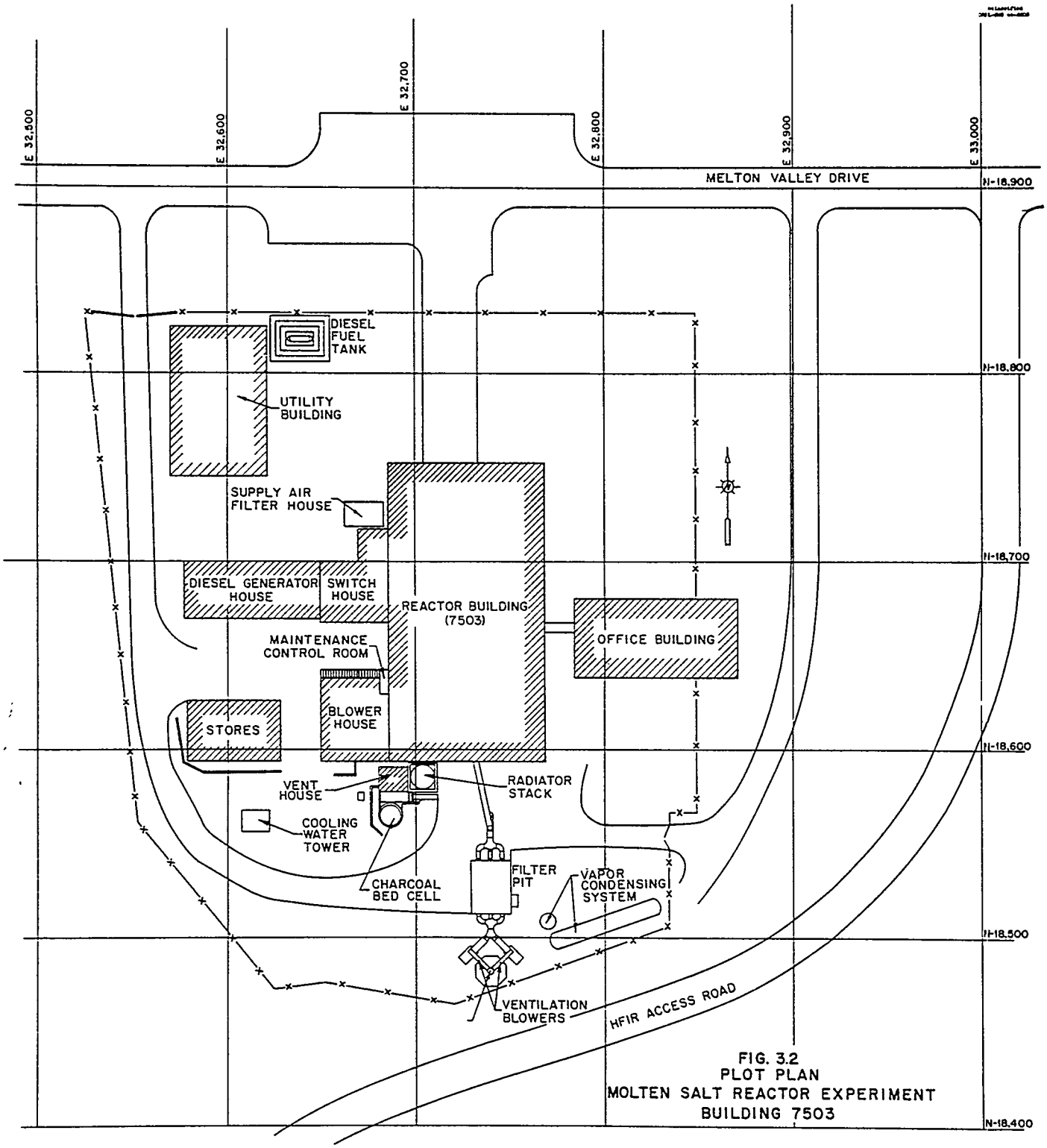
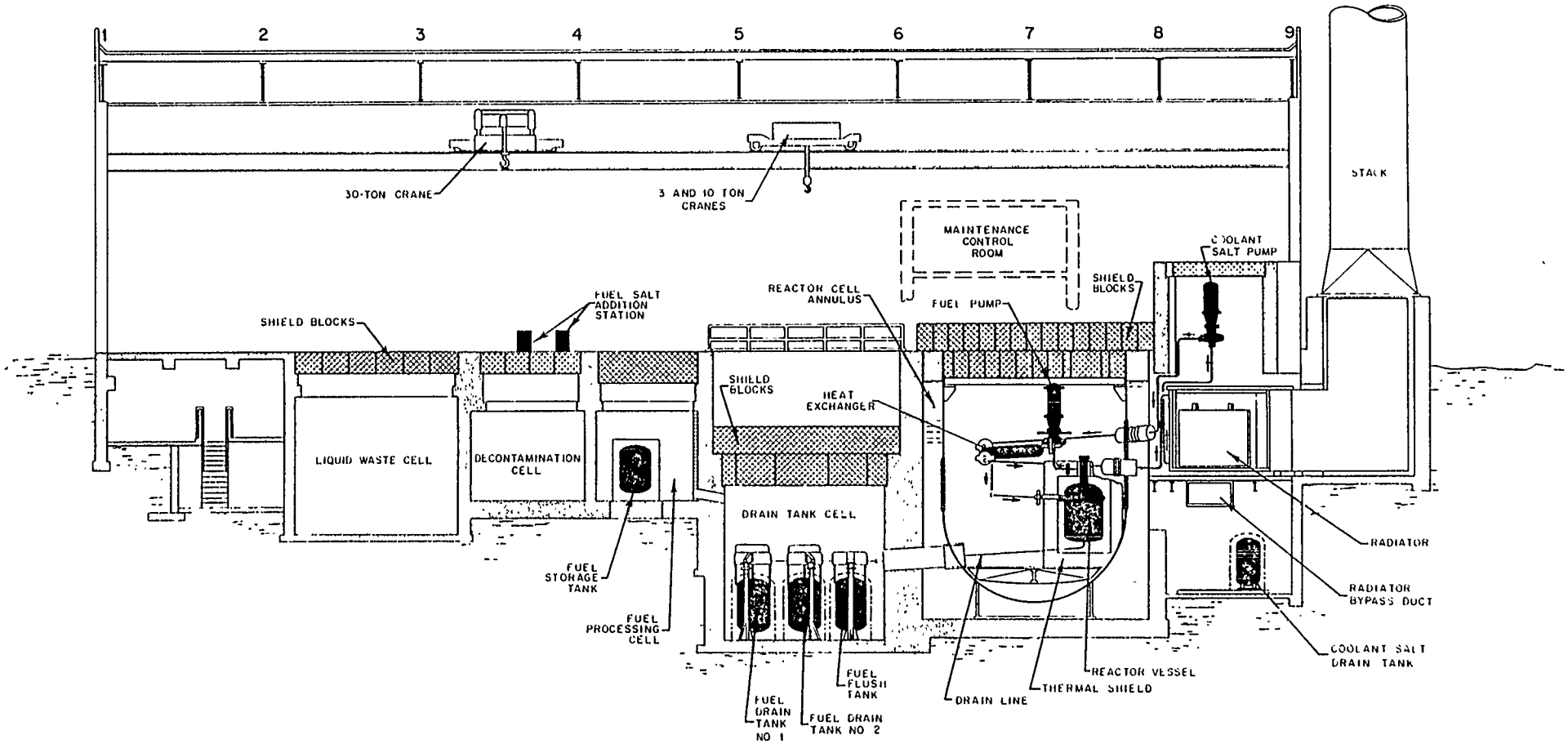


FIG. 3.2
PLOT PLAN
MOLTEN SALT REACTOR EXPERIMENT
BUILDING 7503

Figure 12. Schematic vertical cross section of the Molten Salt Reactor Experiment.

ORNL DWG. 64-597



MSRE PLANT LAYOUT, ELEVATION

Figure 13. Basement floor plan for the Molten Salt Reactor Experiment.

ORNL DWG. 63-4347

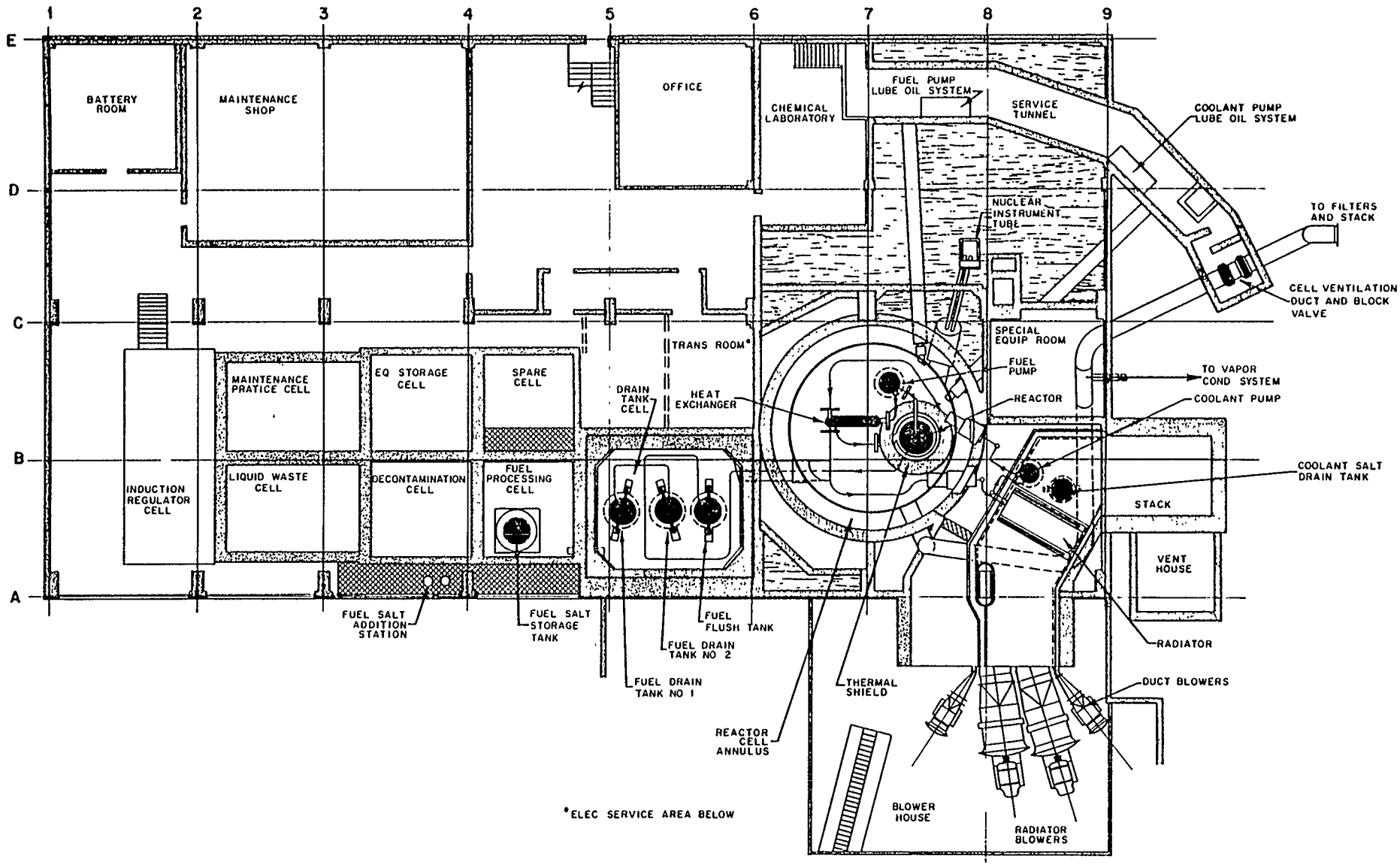
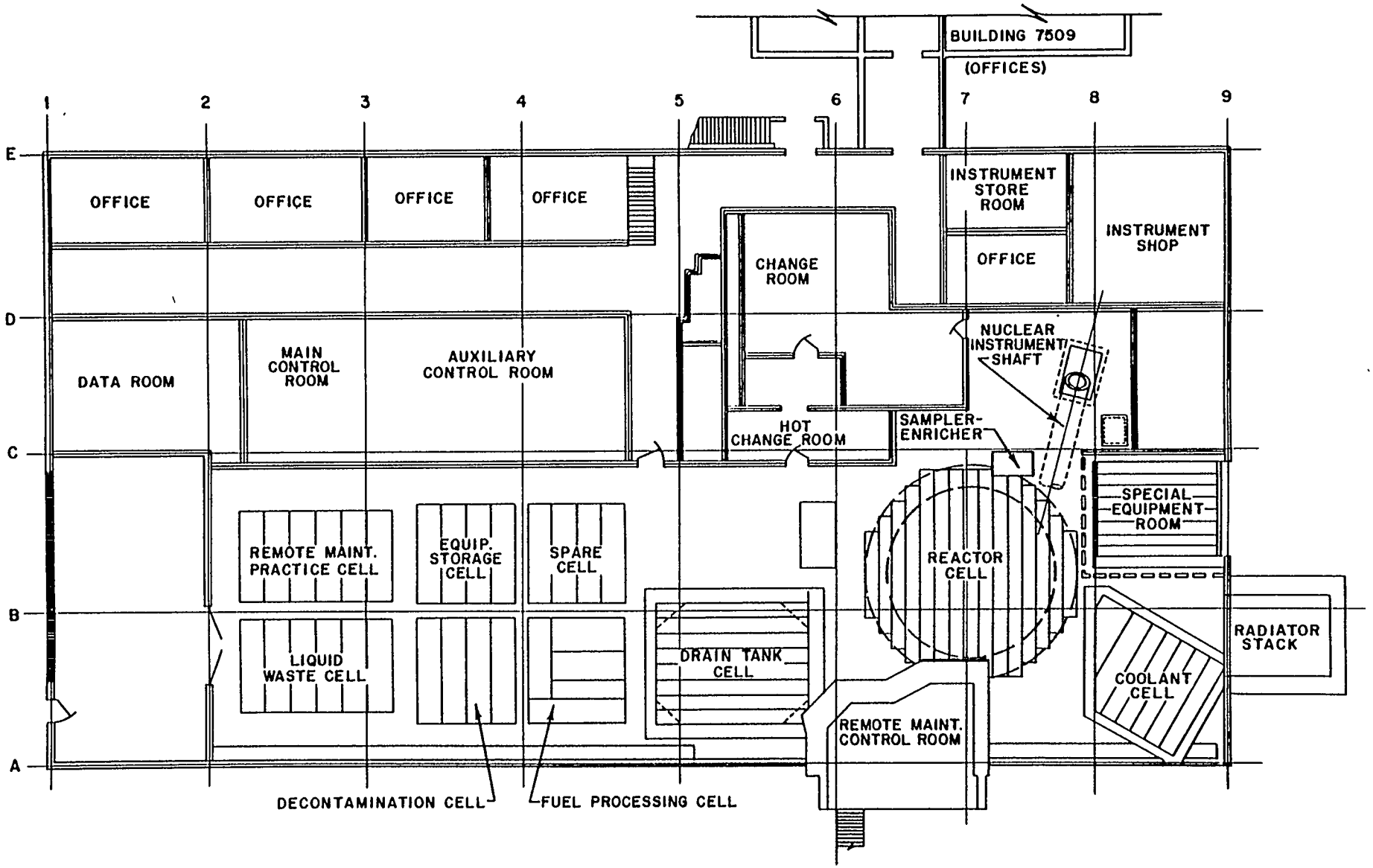


Figure 14. First-floor plan for the Molten Salt Reactor Experiment

ORNL-DWG 64-8726



MSRE PLANT LAYOUT, PLAN
(ABOVE 852'0" LEVEL)

FD-2 (2170 Kg of salt), the flush salt tank contains about 4290 Kg of flush salt, and the coolant drain tank contains 1620 Kg of coolant salt. The coolant salt contains only trace amounts of radioactivity. Heavy nuclides in the salts account for an estimated 1834 Ci of which over 99% was in the fuel salt (Table 4). Most of the isotopes involved have long half lives (much greater than 100 years).

As of 1984, most of the remaining fission products were strontium-90 and cesium-137 (half lives of 28.1 and 30.2 years, respectively) or their very-short-lived daughters (yttrium-90 and barium-137, respectively) which are often included with them in radioactivity assessments. The total radioactivity from these sources is estimated at 35,800 Ci (Table 5). The distribution of these fission products is given as approximately 98% in the fuel salt and 2% in the flush salt. More complete summaries of alpha, beta, gamma, and neutron radiation are given in Peretz (1984) and Simpson (1984).

In addition to the above major sources of radioactivity, an inventory of several thousand Ci of Cobalt-60 in the reactor cell is possible. In general, radiological hazards within Building 7503 are believed to be confined to the reactor and process cells. The only appreciable contamination found outside Building 7503 is associated with the off-gas filter and charcoal pits to the south of Building 7503 (Figure 11) and the associated blowers and stack. Most of the interior surfaces of the off-gas system are presumed to be contaminated to some extent.

Potential chemical toxicity of the fuel and flush salts is summarized in the decommissioning report of Peretz (1984). Although considered minor compared to radiotoxicity for the fuel and flush salts, it is recommended that chemical toxicity be considered in handling the coolant salt. This

Table 4. Calculated radioactivity of fuel-associated nuclides in MSRE Salts.

Table 4. Calculated Radioactivity of Fuel-Associated Nuclides in MSRE Salts (From Simpson 1984)

Nuclide	Half-life (Years)	Inventories in 1984 (curies)	
		Fuel Salt	Flush Salt
^{208}Tl	a	54	0.3
^{212}Po	a	96	0.6
^{212}Bi	a	150	0.9
^{212}Pb	a	150	0.9
^{216}Po	a	150	0.9
^{220}Rn	a	150	0.9
^{224}Ra	a	150	0.9
^{228}Th	a	150	0.9
^{232}U	72	146	0.9
^{233}U	1.62×10^5	370	2.3
^{234}U	2.47×10^5	19	0.1
^{238}Pu	87.8	5	0.1
^{239}Pu	24,130	45	0.9
^{240}Pu	6,570	18	0.5
^{241}Pu	14.4	162	3.2
^{241}Am	432	3	0.1
		1820	14

[a]Activities are in secular equilibrium, decreasing with the 72-y half-life of ^{232}U.

Table 5. Calculated Fission product activities in MSRE salts.

Table 5. Calculated Fission Product Activities in MSRE Salts[a] (From Simpson 1984)

Nuclide	Half-life (y)	Activities (curies) 1984
^{89}Sr	0.14	0
^{90}Sr	28.1	9,520
^{90}Y	0.0	9,520
^{91}Y	0.16	0
^{95}Zr	0.18	0
^{95}Nb	0.10	0
^{106}Ru	1.0	0.4
^{106}Rh	0.0	0.4
^{125}Sb	2.7	18.2
^{125}Te	0.16	8.6
^{127}Te	0.30	0
^{137}Cs	30	8,110
^{137}Ba	0.0	7,570
^{144}Ce	0.78	0.48
^{144}Pr	0.0	0.48
^{147}Pm	2.6	918
^{151}Sm	90	132
^{154}Eu	16	19
^{155}Eu	1.8	1.60
		35,800

[a]Total in fuel and flush salts. Long-lived fission products are distributed 98.1% in the fuel and 1.9% in the flush salt.

salt contains only trace amounts of radioactivity and they may not therefore be handled with sufficient care if chemical toxicity is overlooked. A fuel salt storage cell upgrade project is being planned for FY 1988 to enhance the safety of continued storage-in-place of the fuel and flush salt.

Waste lines leading away from the MSRE are not known to have leaked and are assumed to be in good condition. Some LLW lines may have been disconnected and, if so, they may have to be reconnected or alternative provisions made for LLW disposal prior to decontamination and decommissioning. No site contamination concerns outside of Building 7503 other than the filter pit area have been identified. The facility appears to be structurally sound and capable of continuing to retain its current radionuclide inventory (Simpson 1984).

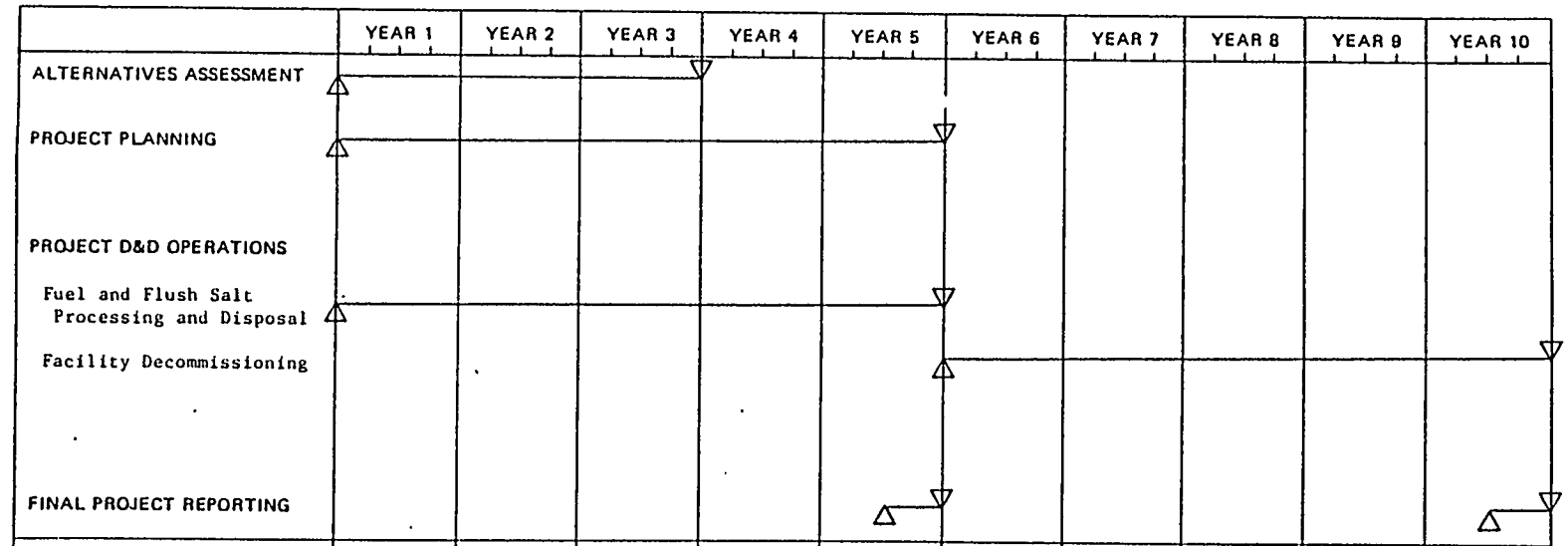
The amount of waste expected to be generated from various decommissioning activities is itemized in the decommissioning report of Peretz (1984) and summarized in Table 6. Further details about the site are given by Peretz (1984) and in the preliminary radiological characterization by Simpson (1984).

4.2.4 Data Deficiencies

Relatively little information was found on the radiation or contamination levels of the cells other than the drain tank and reactor cells; however, the analysis by Cagle and Pugh (1977) indicates that no significant problems are expected. Likewise, the ventilation system is known to be slightly contaminated and a storage pipe outside appears to be slightly contaminated, but both represent only very minor problems in comparison to others previously discussed. Asbestos was also used

Table 6. Timing, cost, and waste-volume projections for decommissioning the Molten Salt Reactor Experiment. The cost estimates are in first-quarter FY 1985 dollars. (Source: Myrick 1984).

ORNL-DWG 83-14840R2



	YEAR 1	YEAR 2	YEAR 3	YEAR 4	YEAR 5	YEAR 6	YEAR 7	YEAR 8	YEAR 9	YEAR 10
PROJECT COSTS ($K)										
Annual Expense	330	1040	2000	2760	4510	5500	5450	4000	2300	510
Capital Equipment	-	100	130	600	200	110	100	260	100	-
Total	330	1140	2130	3360	4710	5610	5550	4260	2400	510
WASTE VOLUME PROJECTIONS										
Solid (m^3)										
Soil (LLW)	-	-	-	-	-	-	-	-	-	-
(TRU)	-	-	-	-	-	-	-	-	-	-
Rubble (LLW)	-	-	-	-	-	-	-	-	-	-
(TRU)	-	-	-	-	-	-	-	-	-	-
Metal (LLW)	-	-	-	-	-	50	50	50	50	50
(TRU)	-	-	-	-	-	25	25	25	25	25
Misc. (LLW)	-	-	-	-	-	75	75	75	75	75
(TRU)	-	-	-	-	-	-	-	-	-	-
Total (LLW)	-	-	-	-	-	125	125	125	125	125
(TRU)	-	-	-	-	-	25	25	25	25	25
Liquid (m^3)										
Process Waste	-	-	-	-	-	5	5	5	5	5
LLW	-	-	-	-	-	15	15	15	15	15

extensively within the facility and will need to be located, removed and buried.

The MSRE decommissioning project will be, by far, the most complex and costly single effort undertaken by the ORNL SFMP, and will require a more extensive radiological and engineering review than possible in the preliminary radiological characterization prepared by Simpson (1984). This more detailed review should emphasize the safe handling and removal of the fuel salts and the dismantlement or entombing of the reactor vessel and associated components.

It was noted by Braunstein et al. (1984) that disposal of the fuel salts will require expertise that is "available almost exclusively at ORNL among personnel, many of whom will be eligible for retirement in the next several years." As a result of this situation and because disposal of the fuel salts is probably the greatest single issue yet to be resolved in the decontamination and decommissioning of the Molten Salt Reactor Experiment, extensive analyses are currently underway to explore pertinent technical issues and salt-disposition options. A description of some results of these studies, as well as other background material including a summary of monitoring activities and explanations of some of the technical problems encountered, has been presented by Notz (1985).

4.2.5 Decommissioning Options

Decommissioning options are discussed in the Preliminary Decommissioning Study Report of Peretz (1984). It is noted in that study (page 62) that evaluations and decision making for decommissioning the MSRE facilities are much easier than decision making for disposal of the fuel and flush salts.

The total cost estimated for disposal of fuel and flush salts and decommissioning the facility is given as over $30,000,000 (First-quarter FY 1985 dollars), spent over a 10-year period.

4.2.6 References

Braunstein H. M., et al. 1984. Preliminary Assessment for Decommissioning Surplus Facilities at ORNL: Environmental and Health Risk. ORNL/CF-84/310, Oak Ridge National laboratory, Oak Ridge, Tennessee.

Cagle, C. D., and L. P. Pugh. 1977. Decommissioning Study for the ORNL Molten Salt Reactor Experiment (MSRE). ORNL/CF-77/391. Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Haubenreich, P. N. and R. B. Lindauer, Consideration of Possible Methods of Disposal of MSRE Salts, ORNL/CF-72-1-1, January 28, 1972.

Myrick, T. E. 1984. The ORNL Surplus Facilities Management Program Long-Range Plan. ORNL/TM-8957, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Notz, K. J. 1985. Extended Storage-in-Place of MSRE Fuel Salt and Flush Salt. ORNL/TM-9756. Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Peretz, F. J. 1984. Preliminary Decommissioning Study Reports Volume 5: Molten Salt Reactor Experiment. X-OE-231, Vol. 5. Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Simpson, D. R. 1984. Preliminary Radiological Characterization of the Molten Salt Reactor Experiment (MSRE). ORNL/CF-84/92. Oak Ridge National Laboratory, Oak Ridge, Tennessee.

4.2.7 Persons Knowledgeable about the Site

(Technical) L. P. (Luther Pugh), ORNL, 615/576-5006

(Administrative) B. L. (Bernie) Corbett, ORNL, 615/574-7001

4.3 Site 9.3: Low Intensity Test Reactor (LITR)

Much of the material that follows is abstracted from the Preliminary Decommissioning Study Report on this facility, prepared by W. R. Reed (1984), and from associated references and appendices as indicated.

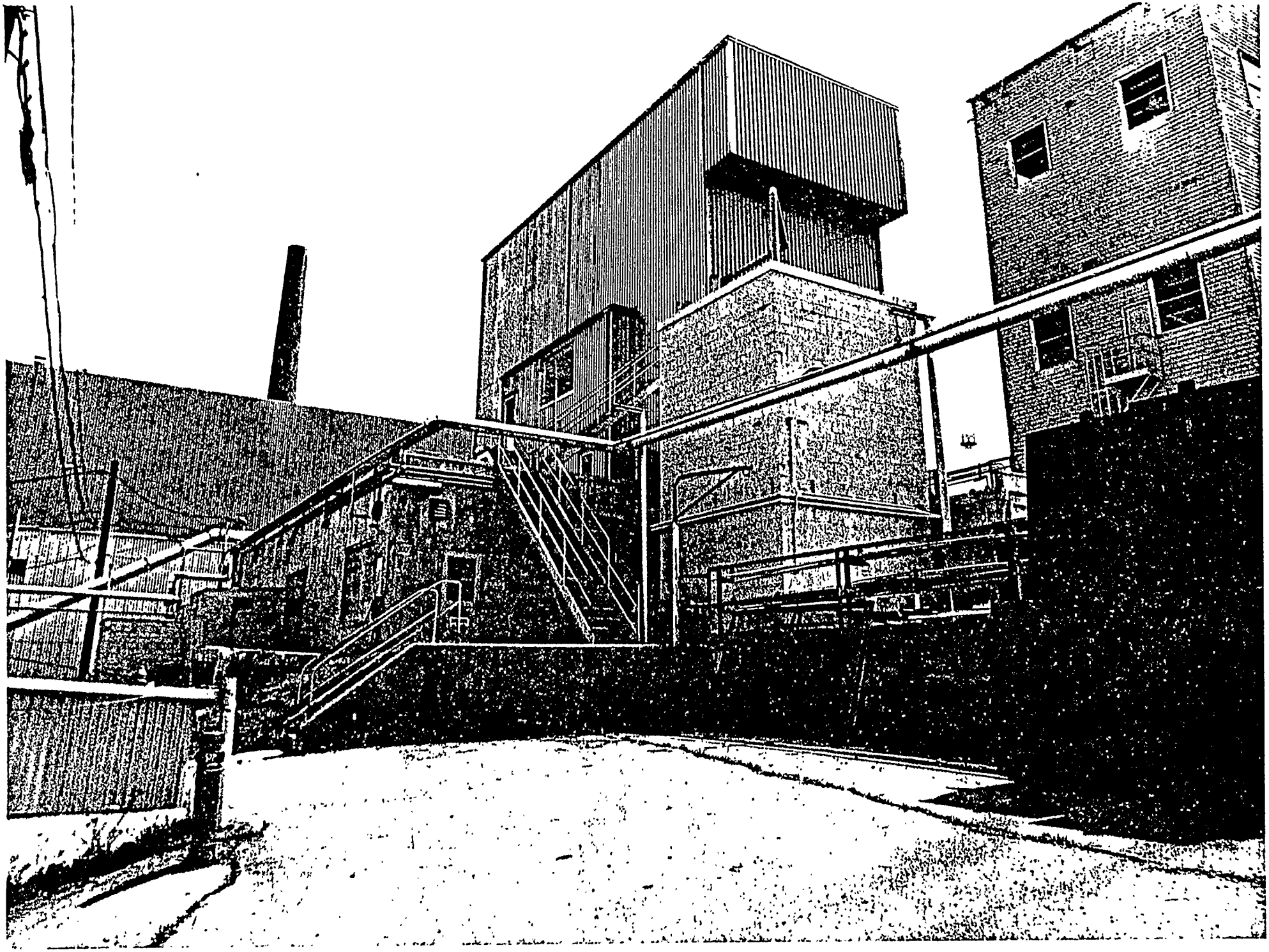
4.3.1 Purpose of Facility

The LITR began operation as a 500-Kw training reactor in March 1951. It was later converted to a testing reactor with a final power level of 3000 Kw, reached in 1953. It was shut down in October 1968.

4.3.2 Location and Description of Site

The reactor is located in Building 3005, which is shown in Figure 15. Its location with respect to other buildings can be seen from the schematic layout in Figure 4 and the aerial photo in Figure 5. These sites are all underlain by rocks of the Chicamauga group. The LITR facility includes "a modest amount" of cooling-water-associated equipment in the yard north of Building 3005. Two retention ponds, approximately 100 yards to the east of Building 3005, have since been earth filled. These ponds are discussed in a separate report on process ponds that have been designated as Remedial Action Program Sites (Category 3, Table 1).

Figure 15. Building 3005, looking southwest, with Building 3004 visible in the upper right.



More information on the dimensions of the LITR facilities is given by Reed (1984). A first-floor plan of Building 3005 is given in Figure 16.

4.3.3. Description of Contamination

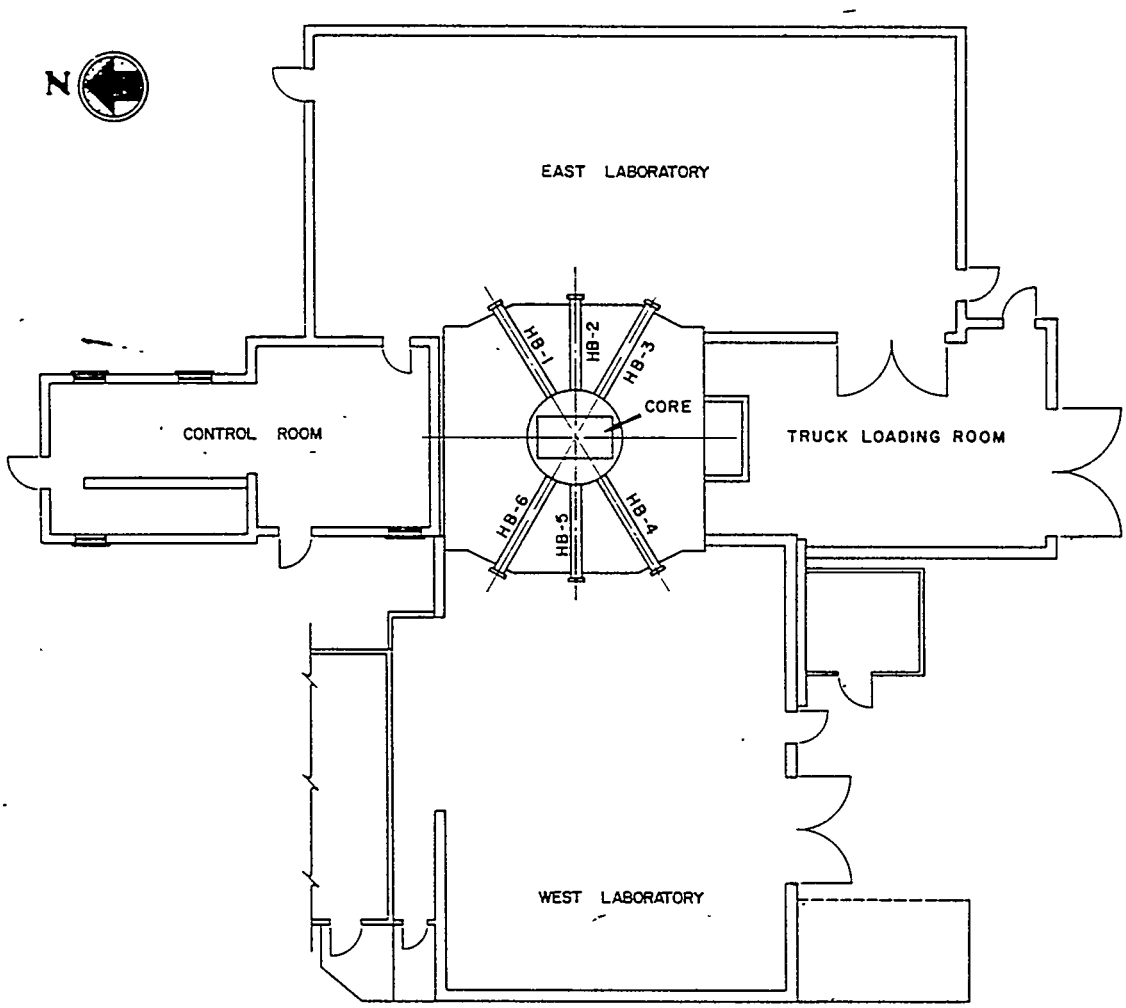
After shutdown in 1968, all fuel and shim-safety rods were removed from the reactor tank, water was drained, and connections were made to exhaust it continuously to the normal off-gas system. Components still inside the reactor are listed in the Preliminary Decommissioning Study Report of Reed (1984). Interior surfaces of the reactor tank and primary water piping system are contaminated with radioactive corrosion products and fission products. Potential sources of radioactive dust that represent additional hazards for decommissioning workers are also described by Reed (1984).

A preliminary radiological survey was prepared by Simpson et al. (1984). That study estimated that the reactor core might contain between 6 and 45 Ci of cobalt-60, less than 50 Ci of nickel-63, and less than 10 Ci of iron-55, which are all activation products. It was also estimated that the core might give a reading of about 200-300 rad/hr. The core still contains beryllium reflector elements and precautions may be required when this potentially hazardous material is removed (Simpson et al. 1984).

The radiological survey found that "only very slight amounts of contamination are indicated on the interiors of the LITR heat exchanger and on pipes in the upper levels of Building 3005." No evidence of soil contamination was found in accessible areas around the facility. An itemized account of expected wastes from decommissioning work is given in the decommissioning report (Reed 1984) and summarized in Table 7.

Figure 16. First-floor plan of Building 3005.

ORNL DWG. 65-1906 R



FIRST LEVEL

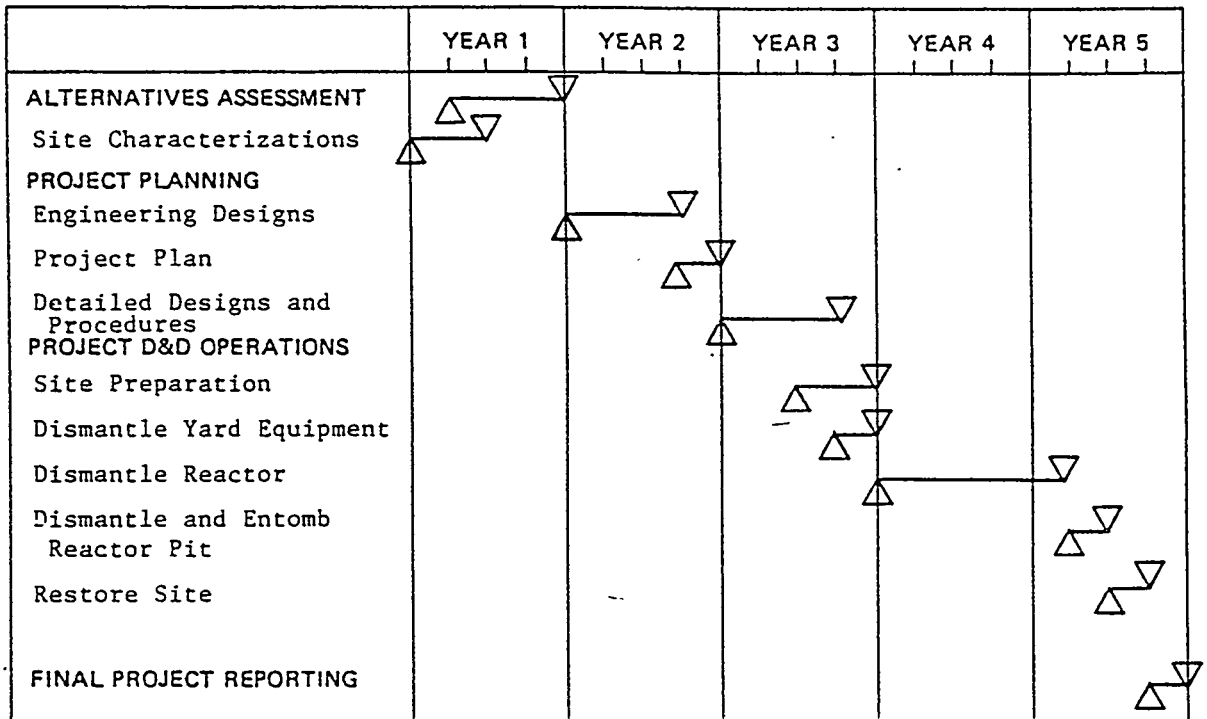
3.2.1

MICROFILMED

I-20300-EF-002-C-1

Table 7. Timing, cost, and waste-volume projections for decommissioning the Low Intensity Test Reactor. The cost estimates are in first-quarter FY 1985 dollars. (Source: Myrick 1984).

ORNL-DWG 83-14840R3



	YEAR 1	YEAR 2	YEAR 3	YEAR 4	YEAR 5
PROJECT COSTS ($K)					
Annual Expense	120	210	1720	1370	770
Capital Equipment	–	–	110	50	150
Total	120	210	1830	1420	920
WASTE VOLUME PROJECTIONS					
Solid (m^3)					
Soil (LLW)	–	–	25	–	–
(TRU)	–	–	–	–	–
Rubble (LLW)	–	–	–	200	5
(TRU)	–	–	–	–	–
Metal (LLW)	–	–	140	10	25
(TRU)	–	–	–	30	5
Misc. (LLW)	–	–	–	5	–
(TRU)	–	–	–	–	–
Total (LLW)	–	–	165	215	30
(TRU)	–	–	–	30	5
Liquid (m^3)					
Process Waste	–	–	–	–	–
LLW	–	–	–	125	125

4.3.4 Data Deficiencies

It was recommended by Reed (1984) that decommissioning work should include checking of the buried water piping for accumulated sludge from early years of operation when no filter was used. That report also recommended checking the soil around an early water leak in the buried waterline to LITR.

Also, the sand and blocks surrounding the reactor tank may be potential sources of radioactive contamination as the borated plastic may have deteriorated or shifted, allowing neutrons to reach the sand and inner concrete blocks. This could cause a potential contamination problem if the shield blocks are disassembled (Simpson, et al. 1984, Cagle and Pugh, 1975).

4.3.5 Decommissioning Options

Decommissioning alternatives are discussed in Chapter 3 of the report by Reed (1984). Much of the discussion is based on an earlier decommissioning study by Cagle and Pugh (1975). According to the estimates in these studies, the decommissioning of this facility will take about 5 years and will cost about $4,500,000 first-quarter FY 1985 dollars (Table 6).

4.3.6 References

Cagle, C. D., and L. P. Pugh. 1975. Decommissioning Study for the Low Intensity Testing Reactor. ORNL/CF-75-6-67. Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Myrick, T. E. 1984. The ORNL Surplus Facilities Management Program Long-Range Plan. ORNL/TM-8957, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Reed, W. R., 1984. Preliminary Decommissioning Study Report, Volume 8: Low Intensity Test Reactor. X-OE-231, Vol. 8. Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Simpson, D. R., J. H. Pemberton, and R. C. Cooper. 1984. Preliminary Radiological Characterization of the Low Intensity Testing Reactor (LITR). ORNL CF/84-/37. Oak Ridge National Laboratory, Oak Ridge, Tennessee.

4.3.7. Persons Knowledgable about the Site

T. P. (Tom) Hamrick, ORNL, 615/574-7078

4.4 Site 9.4: Homogeneous Reactor Experiment

This facility is also known as the Homogeneous Reactor Test. Much of the material that follows in this section was abstracted from the Preliminary Decommission Study Report pertinent to the Homogeneous Reactor Experiment (Reed 1984) and associated references and appendices as indicated.

4.4.1 Background Information

This facility was constructed around 1951, and was originally intended for three phases of experimentation. However, early in the first experiment, accelerated corrosion caused a hole in the core vessel. The hole was patched successfully, but another hole formed later which could not be closed. Operation continued, with leakage between the fuel system and the blanket system, from 1958 until 1961 when the facility was closed. The second and third experiments were never initiated.

4.4.2 Location and Description of Site

The reactor is located in Building 7500, about 1 km southeast of the main ORNL complex, and is the westernmost of the two experimental sites along the Melton Valley Road (Figure 9). It is adjacent to Solid Waste Storage Area No. 5 (SWSA 5). Underlying bedrock material is of the Conasauga group. Building 7502, a ground-level reinforced concrete cell to the east of Building 7500 and just to the south of Building 7505, is also of interest because it contains "the most serious potential hazard outside of Building 7500," namely, the waste evaporator (Reed 1984). Other structures associated with this facility are discussed in detail by Reed (1984) and in supporting references. It was noted by Reed (1984) that the cooling tower southwest of Building 7500 has deteriorated badly, and is no longer useful.

Except for a five year period from 1970 to 1975, parts of Building 7500 and its associated waste system have been used for the Nuclear Safety Pilot Plant. Figure 17 is an aerial photo of the Homogeneous Reactor Experiment site. Figure 18 is a schematic layout of the facilities, and Figure 19 is a first-floor plan of Building 7500.

4.4.3 Description of Contamination

A Preliminary Radiological Characterization of the site has been prepared by Simpson and DeLaGarza (1982). That report was updated by Huang et al. (1984).

Fuel: Some alpha contamination is expected in the reactor pit due to residual fuel. No alpha contamination was found in any accessible area surveyed. Accessible areas remain relatively free of contamination despite high contamination levels known to exist in the reactor pit. The major components enclosed in the reactor pit were not included in the

Figure 17. Aerial photo of the Homogeneous Reactor Experiment Facilities, looking northwest.

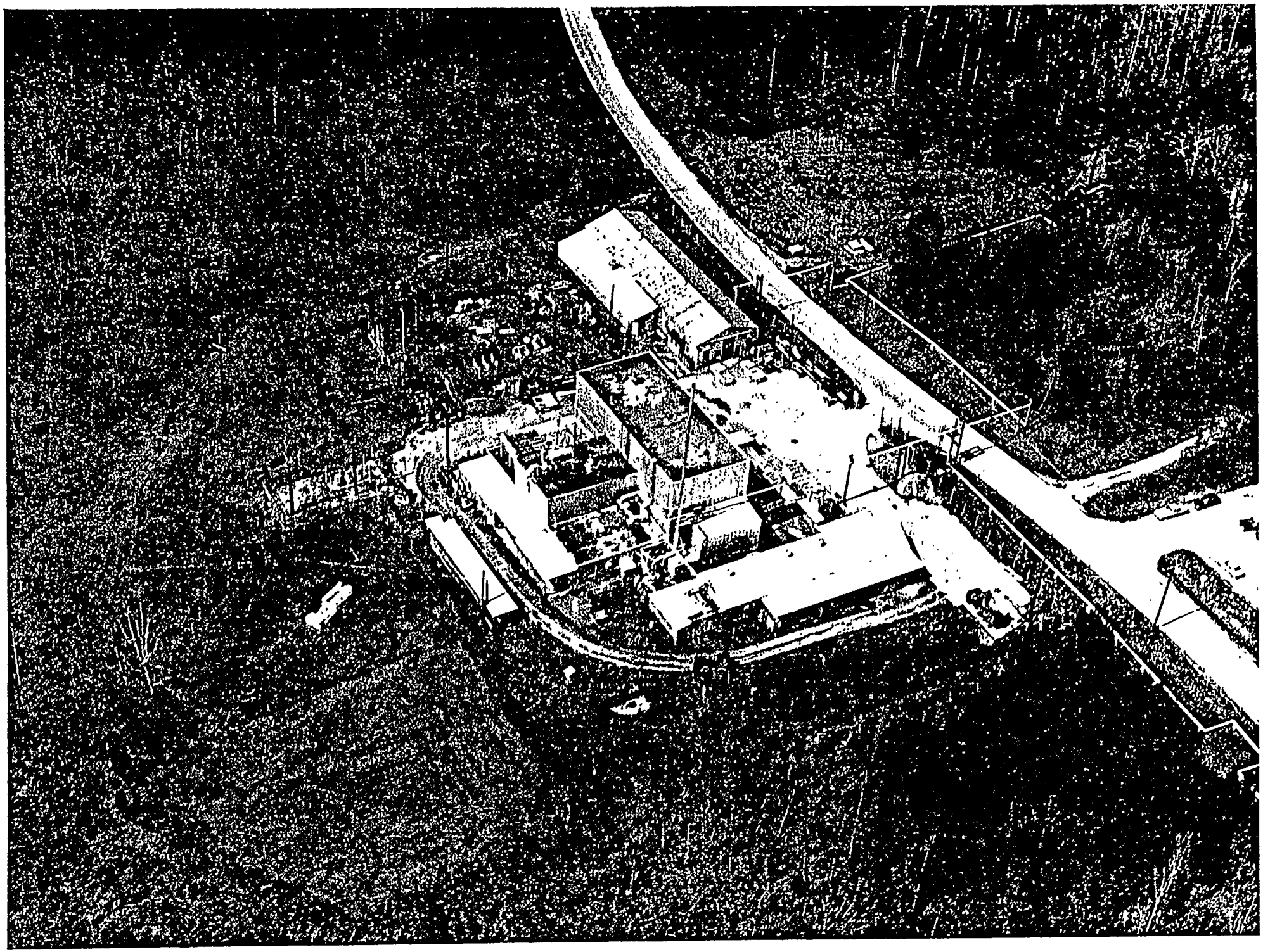


Figure 18. Schematic diagram of the Homogeneous Reactor Experiment facilities. The waste evaporator just south of Building 7505 is also known as Building 7502.

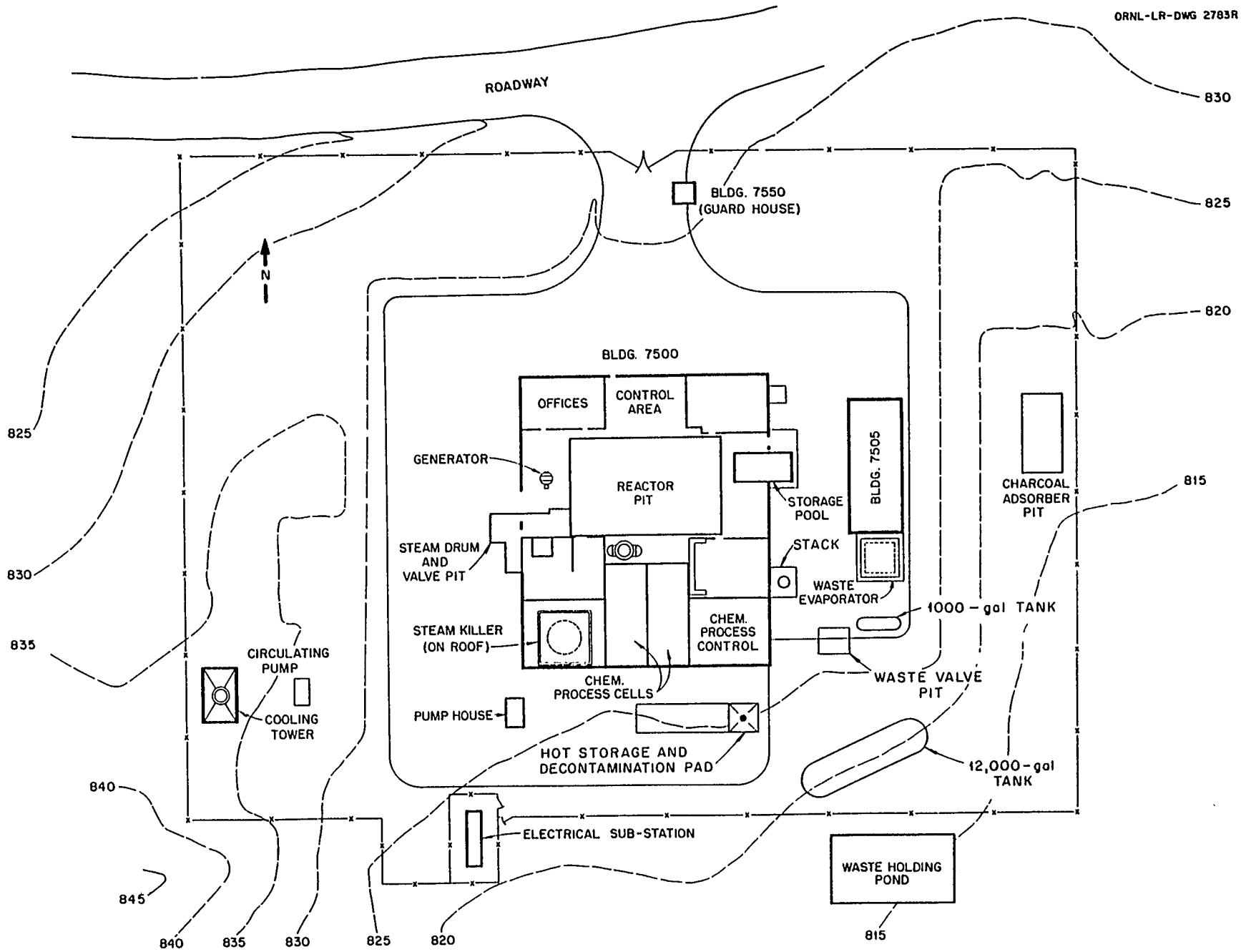
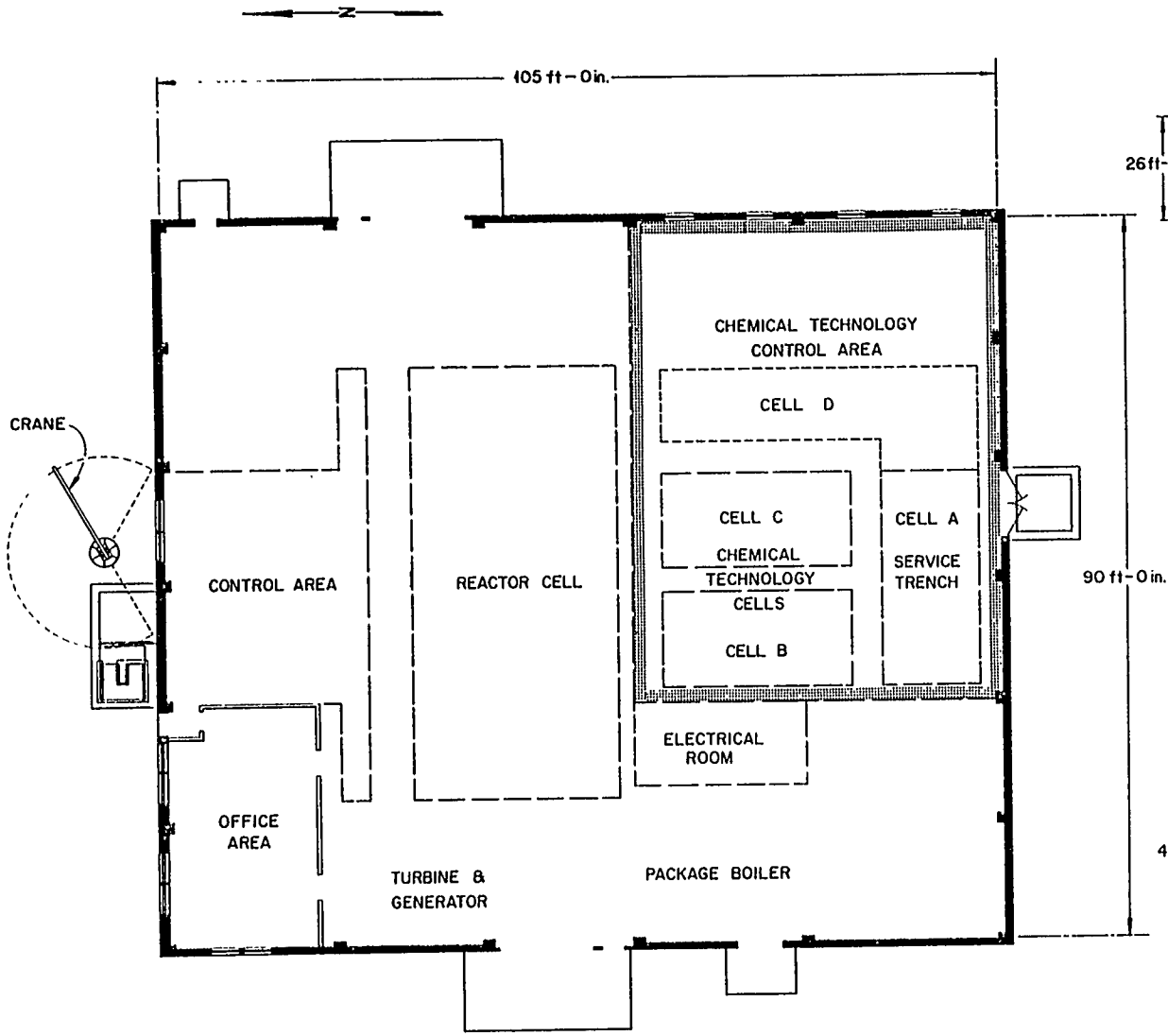

ORNL-LR-DWG 2783R
ROADWAY
BLDG. 7550
(GUARD HOUSE)
N
BLDG. 7500
OFFICES
CONTROL
AREA
GENERATOR
REACTOR
PIT
STORAGE
POOL
BLDG. 7505
CHARCOAL
ADSORBER
PIT
STEAM DRUM
AND
VALVE PIT
STACK
WASTE
EVAPORATOR
1000 - gal TANK
STEAM KILLER
(ON ROOF)
CHEM.
PROCESS
CONTROL
CIRCULATING
PUMP
COOLING
TOWER
CHEM.
PROCESS CELLS
WASTE VALVE
PIT
PUMP HOUSE
HOT STORAGE AND
DECONTAMINATION PAD
12,000-gal
TANK
ELECTRICAL SUB-STATION
WASTE HOLDING
POND
830
825
820
815
825
830
835
840
845
840
835
830
825
820
815

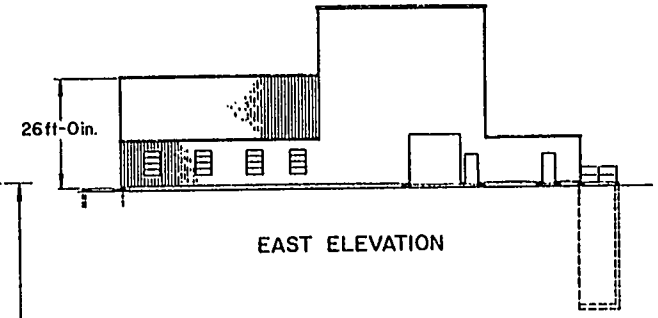
Figure 19. First-floor plan of Building 7500.

ORNL-LH-DWG 2784
REVISION I 4-6-55

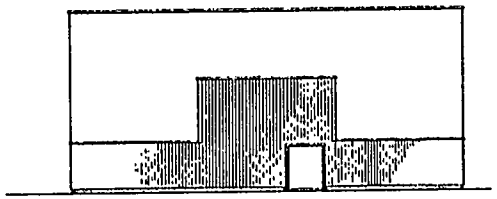


GROUND FLOOR PLAN

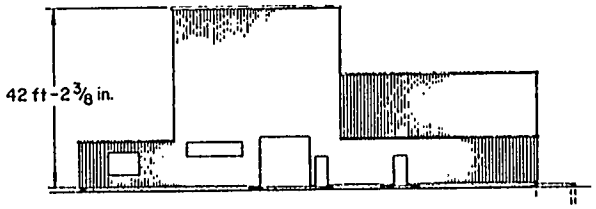
10 0 10 20
SCALE IN FEET



EAST ELEVATION



SOUTH ELEVATION



WEST ELEVATION

10 0 10 20
SCALE IN FEET

radiological surveys mentioned above (Reed 1984). The direct beta-gamma readings in the reactor cell could be as high as 600 rad/h (Reed 1984, Huang et al. 1984). Radioactive gases could exist in the cell. The evaporator cell, chemical processing cell C, and the storage pool also provide a small probability for significant exposure (Braunstein et al. 1984, Huang et al. 1984).

Fission and Corrosion Products: Isotopic analysis indicated that these consisted almost entirely of cesium-137 and strontium-90, which both have half lives of about 30 years. Cagle and Pugh (1976), Huang et al (1984) and Reed (1984) all noted that an estimated 30-40 Kg of highly radioactive isoluble corrosion and fission products remain in the process piping. This was also noted by Myrick (1984) who parenthetically provided a radioactivity estimate of less than 500 Ci for that material.

Environmental or Peripheral Contamination: The waste evaporator (in Building 7502) represents the most serious potential hazard outside of Building 7500. Accessible areas of Building 7502 are extensively contaminated and much of the area is blocked with no easy way of determining contamination levels inside. The building had been deteriorating (Reed 1984), but a containment upgrade project has since been initiated to address that situation.

This facility does not seriously interfere with the operation of the Nuclear Safety Pilot Plant with which it shares Building 7500. Further, this facility is deemed not to pose any suspected current or immediate danger to ORNL personnel, the public, or the environment (Reed 1984). However, it is natural for personnel needing workspace to gravitate to the area. The presence of personnel who may be ignorant of the radiological

history and conditions of the site would presumably increase the incentive to investigate and eliminate potential hazards left over from the facility.

The waste-retention pond associated with the Homogeneous Reactor Experiment is discussed in a separate report on process ponds that have been designated as Remedial Action Program Sites (Category 3, Table 1). However, a cross-reference paragraph seems appropriate here. Most of this material is summarized from Braunstein et al. (1984).

The waste-retention pond was filled in during 1970. Estimates of the amount of buried radioactivity range from 100 to 500 Ci. This activity is thought to be comprised mostly of cesium-137, strontium-90, and cobalt-60. The pond is located about 50 meters upgradient from a small creek which flows into Melton Branch of White Oak Creek and, eventually, into the Clinch River. Radioactive materials, notably strontium-90, can be transported by groundwater to the creek. Groundwater upgradient and downgradient from the pond site is currently monitored for contamination. The potential for contaminant transport increases the incentive to more fully characterize the site on a priority basis.

4.4.4 Data Deficiencies

Several items of interest were mentioned in the preliminary decommissioning report by Reed (1984).

The major reactor components enclosed in the reactor pit were not included in the radiological surveys mentioned above (Reed 1984).

The waste evaporator (in Building 7502) represents the most serious potential hazard outside of Building 7500. Accessible areas of Building 7502 are extensively contaminated and much of the area is blocked with no easy way of determining contamination levels inside.

According to Huang et al (1984), between 1951 and 1953 "A large quantity of fuel solution containing enriched uranium-235 was accidentally released into the small creek near Building 7500. By now, after 30 years, because of the high water solubility most (if not all) of the fuel solution should be washed away. However, it is possible that some residual radionuclides, especially uranium-235, are left on the creek bank and in the sediment. To verify this, soil and sediment samples from these areas would be needed."

Huang et al (1984) noted strontium-90 contamination near the sediment layer of the waste-retention pond (which has since been filled in). They recommended drilling to "take inventory of the radiological content of this earth layer of probable high contamination."

Radiation readings on the outside of the curb around the decontamination pad indicated that some leaching of contamination into the ground may have occurred in the area of pad (Building 7561). It was recommended that prior to any extensive work in the north sublevel of Building 7500, a safety inspection be made to verify that no hazards have been created due to deteriorating conditions in the walls and floor.

The exact condition of buried waste tanks and the buried carbon absorbers (pipe coils) is not known.

Several lines and pipes penetrate the walls of the north sublevels into the reactor pit. These lines may contain small amounts of contamination and some consideration should be given as to how they will be removed.

4.4.5 Decommissioning Options

Decommissioning of this facility has been estimated to take 6 years and to cost between $13,605,000 (Reed 1984) and $14,475,000 (Myrick

1984). Both estimates are in FY 1985 first-quarter dollars. Estimated waste material resulting from decommissioning is itemized in Table 8.

4.4.6 References

Braunstein, H. M., et al. 1984. Preliminary Assessment for Decommissioning Surplus Facilities at ORNL: Environmental and Health Risk. ORNL/CF-84/310, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Cagle, C. D., and L. P. Pugh. 1976. Decommissioning Study for the Homogeneous Reactor Experiment No. 2 (HRE-2). ORNL CF-76-66. Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Campbell, D. O. 1956. Decontamination of the Homogeneous Reactor Experiment. ORNL-1839. Oak Ridge National Laboratory, Oak Ridge, Tennessee.

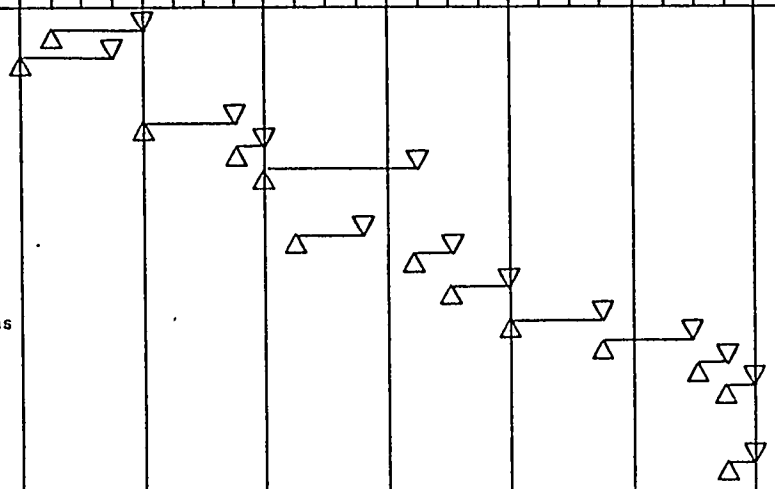
Huang, S. F., D. R. Simpson, L. K. Podolak, and T. W. Oakes. 1984. Preliminary Radiological Characterization of the Homogeneous Reactor Experiment No. 2 (HRE-2). ORNL CF 84/201. Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Myrick, T. E. 1984. The ORNL Surplus Facilities Management Program Long-Range Plan. ORNL/TM-8957, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Reed, W. R. 1984. Preliminary Decommissioning Study Reports Volume 6: Homogeneous Reactor Test. X-OE-231, Vol. 6. Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Table 8. Timing, cost, and waste-volume projections for decommissioning the Homogeneous Reactor Experiment. The cost estimates are in first-quarter FY 1985 dollars. (Source: Myrick 1984).

ORNL-DWG 83-14840R2



	YEAR 1	YEAR 2	YEAR 3	YEAR 4	YEAR 5	YEAR 6	YEAR 7	YEAR 8	YEAR 9	YEAR 10
ALTERNATIVES ASSESSMENT										
Site Characterization										
PROJECT PLANNING										
Engineering Designs										
Project Plan										
Detailed Designs										
PROJECT D&D OPERATIONS										
Entomb Pond										
Site Preparation										
Clean and Consolidate Reactor Cell										
Dismantle Bldg. 7500 Areas										
Dismantle Outlying Areas										
Entomb Reactor Cell										
Site Restoration										
FINAL PROJECT REPORTING										
PROJECT COSTS ($K)										
Annual Expense	350	1040	2185	4130	3335	3290				
Capital Equipment	-	-	20	85	30	10				
Total	350	1040	2205	4215	3365	3300				
WASTE VOLUME PROJECTIONS										
Solid (m^3)										
Soil (LLW)	-	-	30	-	50	170				
(TRU)	-	-	-	-	-	-				
Rubble (LLW)	-	-	-	-	15	60				
(TRU)	-	-	-	-	-	-				
Metal (LLW)	-	-	5	-	5	20				
(TRU)	-	-	-	-	-	-				
Misc. (LLW)	-	-	-	-	-	-				
(TRU)	-	-	-	-	-	-				
Total (LLW)	-	-	35	-	70	250				
(TRU)	-	-	-	-	-	-				
Liquid (m^3)										
Process Waste	-	-	-	-	-	-				
LLW	-	-	-	40	60	15				

Simpson, D. R., and S. DeLaGarza. 1982. Preliminary Radiological Characterization of the Homogeneous Reactor Experiment No. 2 (HRE-2). ORNL CF 82-/288.

Several other reports pertinent to decommissioning (Design manuals, drawings, etc.) are also listed by Reed (1984).

4.4.7 Persons Knowledgeable about the Site

(Technical) L. P. (Luther) Pugh, ORNL, 615/576-5006

(Administrative) B. L. (Bernie) Corbett, ORNL, 615/574-7001

4.5 Site 9.5: ORR Water-to-Air Heat Exchanger (Building 3087)

This facility has been described in a Preliminary Decommissioning Study Report (Horton 1984). Much of the material presented in this section is abstracted from that report and from associated references and appendices as indicated.

4.5.1 Purpose of Facility

This facility, known locally as the Oak-Ridge-Research Reactor (ORR) Trane Coolers, provided the primary cooling for the ORR Reactor during the first several months of reactor operation. However, it did not provide adequate cooling when reactor capacity was increased, and in 1961 it was replaced by a cooling tower. The structures have not been used or maintained operational since 1961. Because of its past relationship to the ORR, this facility is often grouped together with the ORR Experimental Facilities, discussed in section 4.6.

4.5.2 Location and Description of Site

The facility is identified as Building 3087. It is located approximately 300 feet northeast of the ORR (Building 3042) and is designed as site 1 in Figure 4. A ground-level photo is shown in Figure 20. Figure 21 is an aerial view, looking northwest, showing the Trane coolers near the center. For orientation with respect to previously discussed facilities, the fan house for the Graphite Reactor (Building 3003) and the Low Intensity Test Reactor (Buildings 3004 and 3005) are visible in the upper-left part of the picture. In 1970, one of the eight original coolers was removed, decontaminated, and salvaged. The remaining seven coolers are deteriorating and show considerable surface rust (much of which is covered by snow in the winter photograph of the site shown as Figure 20).

4.5.3 Description of Contamination

A preliminary radiological characterization of the facility has been completed by Simpson and Huang (1983). Results of that survey indicate only limited residual contamination of the interiors of pipes that carried cooling water. Exterior surfaces and environmental areas around the site are indicated to be free of radioactive contamination. The cooler that was removed in 1970 was examined before salvage. It contained no alpha activity but was contaminated with beta-gamma activity (Pemberton, 1970). Most contamination was confined to the inlet and exit primary water lines. The contaminants were then identified as cesium-137 (half life = 30.2 years), cobalt-60 (half life = 5.3 years), and cerium-144 and chromium-51 (both of which have half lives of less than one year).

Figure 20. Ground-level photo of the water-to-air heat exchangers, known locally as Trane coolers.

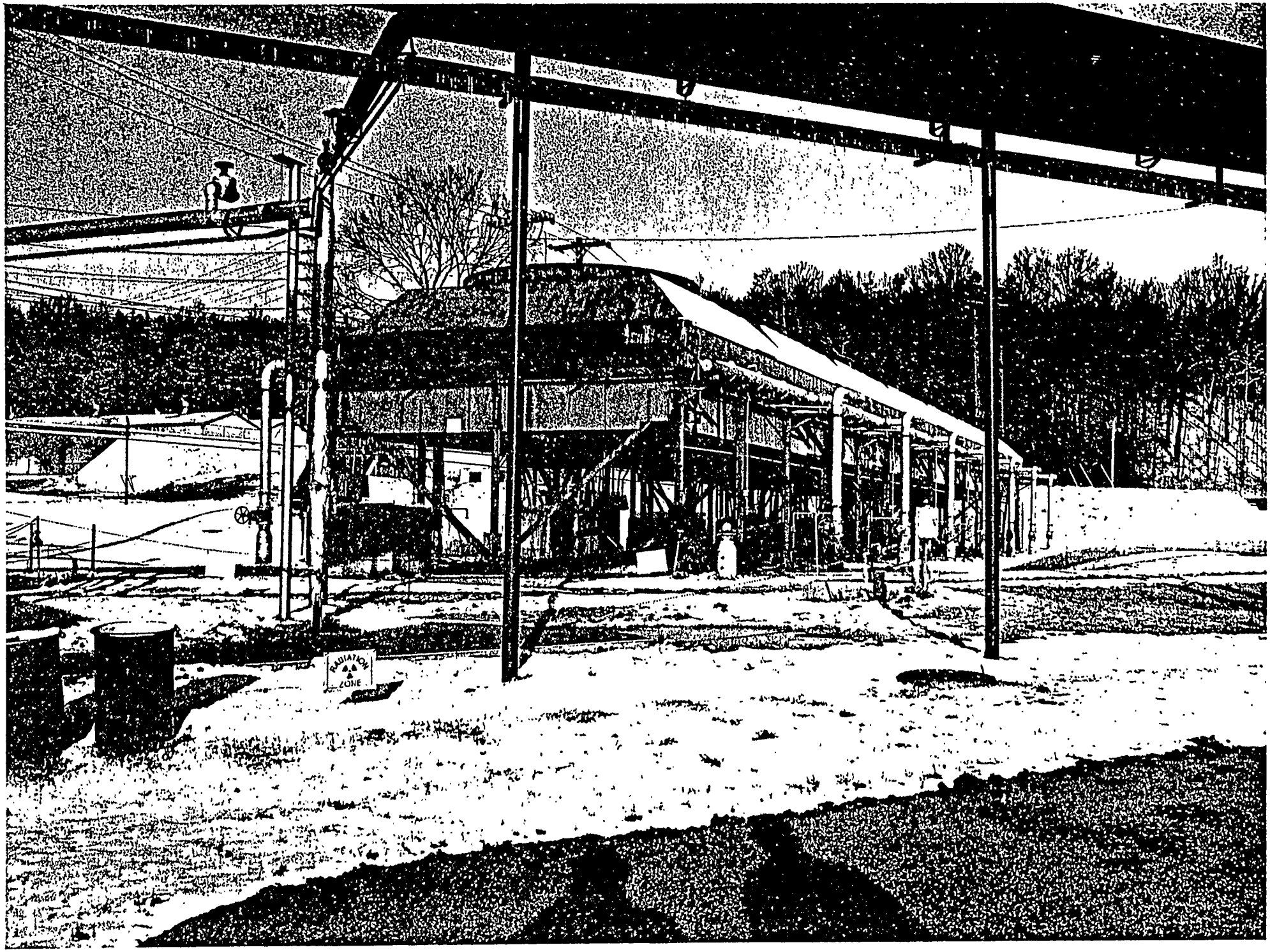
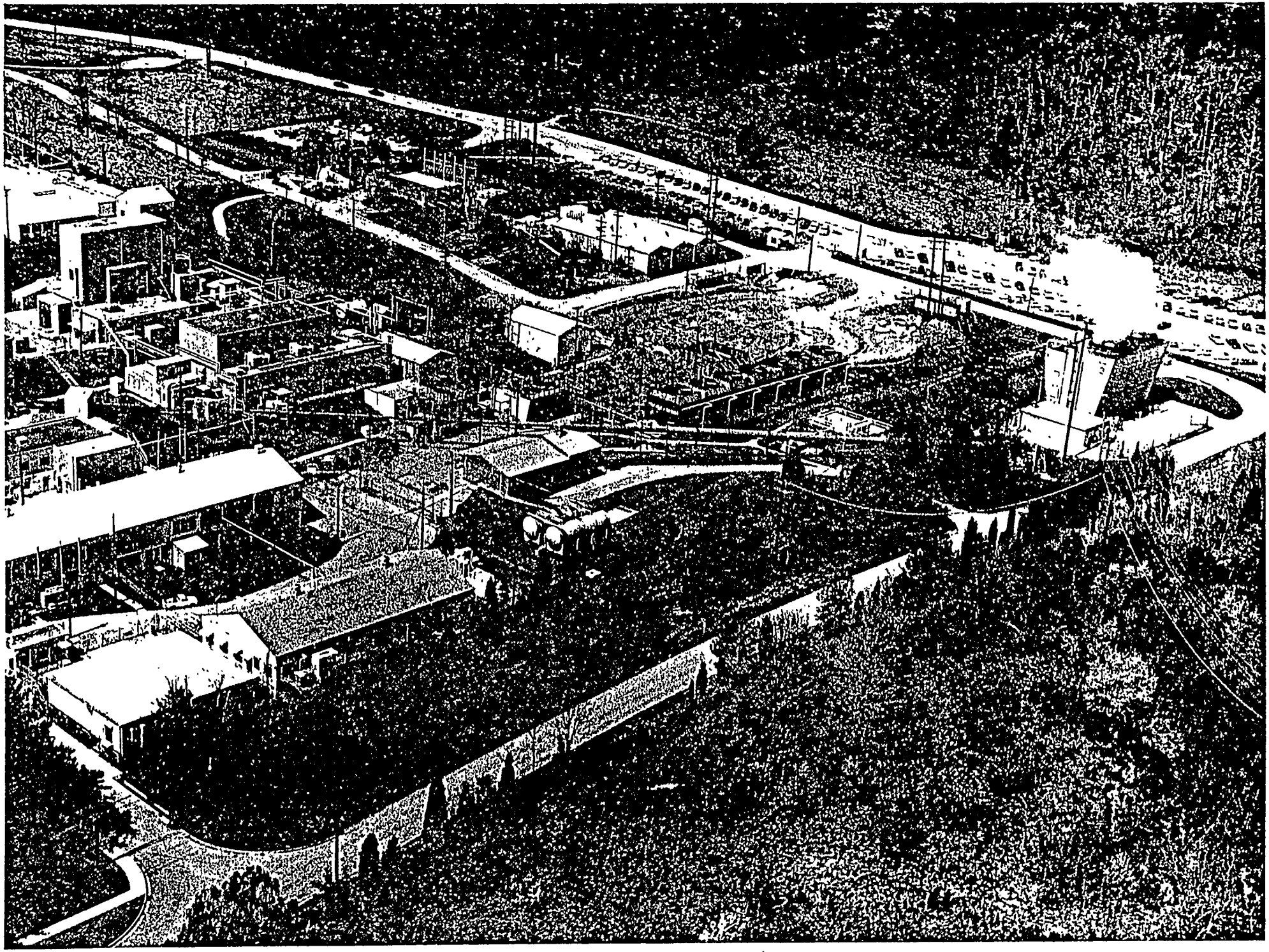

RADIATION
ZONE

Figure 21. Aerial view, looking northwest, showing the heat exchangers near the center. For orientation with respect to facilities described previously in this report, the Low Intensity Test Reactor (Buildings 3005 and 3004) and the fan house for the Graphite Reactor (Building 3003) are visible in the upper left.



4.5.4 Decommissioning Options

Continued deterioration of this facility is inevitable. There is potential for loss of containment of the low levels of radioactive material present if major repairs or decommissioning activities are not conducted. This equipment is surplus, having little or no potential for further use except possibly for scrap. Removal of the facility would make the site available for reuse. The cost of dismantlement and disposal has been estimated as $212,000 FY 1985 first-quarter dollars. The work should take about 8 to 14 weeks and would generate an estimated 82 m^3 of contaminated metal, 82 m^3 of rubble, and 20 m^3 of contaminated soil (Horton 1984).

4.5.5 Data Deficiencies

No data deficiencies were noted.

4.5.6 References

Horton, J. R. 1984. Preliminary Decommissioning Study Reports Volume 10: Oak Ridge Research Reactor Water-to-Air Heat Exchangers. X-OE-231, Vol. 10. Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Pemberton, J. H. 1970. "Radiation-Contamination Survey of One Section of the ORR Trane Coolers," Interoffice Correspondence, ORNL Industrial Safety and Health Physics Division, September 2, 1970).

Simpson, D. R., and S. F. Huang. 1983. Preliminary Radiological Characterization of the ORR Water-to-Air Heat Exchangers (Building 3087). ORNL/CF-83/204. Oak Ridge National Laboratory, Oak Ridge, Tennessee.

4.5.7 Persons Knowledgable about the Site

(General Operation) T. P. Hamrick, ORNL, 615/574-7078

(Health Physics) J. H. Pemberton, 615/574-6705

4.6 Site 9.6: Oak Ridge Research Reactor Experimental Facilities

Most of the material in this section is abstracted from the Preliminary Decommissioning Study report pertinent to this site (Horton 1984) and associated references and appendices as indicated.

4.6.1 Purpose of Facility

Since 1959 several experimental facilities have been installed in the Oak Ridge Research Reactor (ORR) for testing of various materials, analysis of liquid and gaseous coolant systems, and transfer of irradiated samples. Six of these facilities have been designated as surplus and are included in the ORNL Surplus Facilities Management Program. These six will each be discussed separately in subsection 4.6.3.

4.6.2 Location and Description of Site

The ORR is a 30 KW pool-type research reactor located in Building 3042 (see map, Figure 4). An aerial view of this building in relation to other facilities is shown in Figure 5. All of the experimental facilities are housed in the reactor building except for some equipment associated with the Maritime Ship Reactor Loop that is buried underground outside Building 3042 (discussed in section 4.6.3.3) and a portion of the pneumatic tube facility (discussed in section 4.6.3.4) that runs between Building 3042 and Building 3001. The six surplus

experimental facilities are listed and discussed in detail by Horton (1984). Each facility includes, or once included, an in-reactor section with associated piping, instrumentation and controls leading to processing or experimental areas located either immediately adjacent to the reactor or at more remote locations, primarily in the basement of the ORR. The out-of-reactor equipment is typically contained in shielded cells of lead, concrete block, or concrete and steel, with separate instrument and control panels. These systems encompass a wide range of complexity (Horton 1984). The original and present location of this equipment with respect to the ORR is described by Horton (1984).

Following completion of the respective experiments, the in-reactor portions of the facilities were usually removed and the remaining systems placed in standby. Most of the facilities remain as left, with only limited removal of equipment to provide room for active experiments or for reactor maintenance. Horton (1984) characterizes the abandoned experiments as being "in various states of disrepair and deterioration."

The six experimental facilities designated as surplus are:

(1) ORR Gas-Cooled Reactor Loops A9, B9, and C1

(2) ORR Molten Salt Loop

(3) ORR Maritime Ship Reactor Loop

(4) Pneumatic Tube Irradiation Facility

(5) ORR Gas-Cooled Reactor Loop I

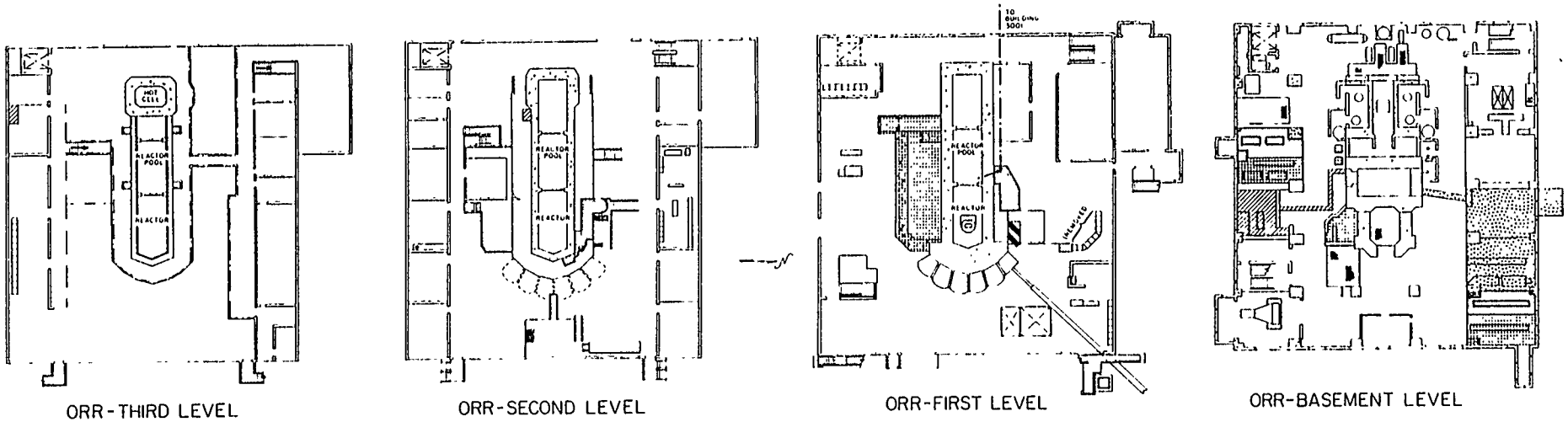
(6) ORR Gas-Cooled Reactor Loop II

The locations of these facilities with respect to the Oak Ridge Research Reactor, are shown in the floor plans presented as Figure 22.

Figure 22. Floor plans for Building 3042, showing the locations of the six surplus facilities.

ORNL-DWG 83-16505R

OAK RIDGE RESEARCH REACTOR (ORR) BUILDING PLAN



GCR A9-B9 EXPERIMENT
MOLTEN SALT LOOP
MARITIME SHIP REACTOR LOOP
PNEUMATIC TUBE IRRADIATION FACILITY
GCR LOOP I
GCR LOOP II

The material that follows was largely excerpted from the ORR Experimental Facilities preliminary decommissioning report by Horton (1984) which contains more detailed accounts of facility designs, operating histories, and current conditions for each of the facilities listed above. Preliminary radiological characterizations have been done for each facility (Simpson and Pemberton 1984), and these are cited extensively and often quoted verbatim in Horton's decommissioning study and in the present report. Results of the radiological survey indicated that the ORR Maritime Ship Reactor Loop and the ORR-GCR Loop II, will require extensive health physics coverage during decommissioning. The Maritime Ship Reactor Loop equipment room contains several pieces of highly contaminated equipment, and the GCR Loop II has sections still extending into the reactor core. The remaining four facilities do not appear to present serious health physics problems, primarily because in-core sections were previously removed and the remaining sections are not extensively contaminated (Simpson and Pemberton 1984).

Some of the experimental equipment could not be surveyed unless the reactor were shut down and the pool water level lowered. Therefore, such items have not been thoroughly surveyed for contamination. These cases are noted by Horton (1984) and by Simpson and Pemberton (1984), and contamination estimates are given, where possible, based on available information. For five of the six facilities, it was asserted in the preliminary radiological survey that no environmental considerations are involved because each of the facilities is located completely within Building 3042. The exception appears to be the Maritime Ship Reactor Loop, for which a drain and tank are buried underground outside Building 3042 and it apparently is not known whether

those items were ever used, were ever leaking contamination, etc. This subject is discussed in section 4.6.3.3 of this report, and in more detail by Horton (1984) and Simpson and Pemberton (1984). The following section of this report summarizes material found in those other two, more detailed, reports. The six experimental facilities will each be discussed separately, in order of the above list.

4.6.3 History and Extent of Contamination for each Facility

4.6.3.1 ORR Gas-Cooled Reactor Loops A9, B9, and C1

These facilities consisted of in-pile radiation loops connected to the ORR. The loops were designated A9, B9, and C1. Loop C1 was a prototype, installed before A9 and B9. These loops were installed in the early 1960s to provide facilities in which feed materials could be irradiated to determine their ability to retain gaseous fission products during irradiation. The loops continued in use until the late 1960s and have since been abandoned. The equipment that remains consists mainly of piping that is stored in the reactor pool or nearby areas. The control room has been stripped of most of the equipment, but has not been decommissioned.

The following text is excerpted from the preliminary radiological characterization of the site by Simpson and Pemberton (1984).

> "Essentially, the only equipment that remains from the experiment is the piping. Piping for A9-B9 remains in the reactor pool, the north stand pipe, under the reactor, and in a trench in the basement floor leading to the A9-B9 control room. This piping is sealed off below water level at the reactor end and at the end of the trench in the basement control room. The C1 horizontal

piping, also associated with this experiment, is located on the inside south wall of the pool and is sealed off under water on both ends. The control room in the basement and the C1 cubicle on the outside of the pool have been cleared of essentially all equipment. In-core equipment for this experiment was removed previously.

"The survey plan for this facility was to primarily assess the radiation hazard involved with the removal of the A9-B9 piping leading from the reactor pool to the basement control room. Most of this pipe is inaccessible as it is in a shielded trench, under the reactor, or in a pipe stand; however, one end is open and accessible in the basement control room. This section was smeared for contamination and surveyed for dose rate. The remainder of the control room was given a brief survey to ensure that it remained free of contamination. The sections of the A9-B9 piping remaining in the reactor pool and the C1 piping which is entirely in the pool were not surveyed as they remain underwater.

"The piping associated with the A9-B9 experiment remains highly contaminated. A smear reading from the open end of the piping in the basement control room indicates transferable contamination of 100,000 dpm/100 cm^2 beta-gamma, but less than 20 dpm/100 cm^2 alpha. An isotopic analysis indicated the contamination to be almost exclusively ^{60}Co. A direct reading of absorbed dose rate at the open end of the piping indicated 10 mrad/h.

"The remainder of the control room was free of contamination and at the background radiation levels for this area (< 0.1 mrem/h). The cubicle formerly housing equipment from the C1 piping

was also surveyed and found free of contamination. The general radiation level in the C1 cubicle was 1 mrad/h; however a maximum reading of 20 mrad/h at 2.5 cm (1 in.) was noted on a line labeled NOG (OG-13) in the northeast section of the cubicle near the ceiling. Another line in the cubicle from an old GB-10 experiment read 3 mrad/h at 2.5 cm (1 in.). It is not known if these lines were ever associated with the C1 experiment.

"No environmental considerations are involved in this facility as it is completely housed within Bldg. 3042.

"No significant radiological or nonradiological hazards were observed. Routine ORNL radiation and contamination control procedures would be adequate during decommissioning.

"Piping from this facility located in the reactor pool was not surveyed. Measurement of transferable contamination is impractical underwater; and radiation from the reactor core and spent fuel would interfere with a direct reading of the dose rate associated with the piping. A survey of this piping would require very little time but would require that the reactor be shut down and the water lowered below the piping (if possible). Neutron activation of the piping would be minimal due to its distance from the reactor core.

"Also, the A9-B9 piping could only be readily surveyed at the control room, though this should be a reasonable estimate of the contamination throughout. If it is proposed to remove all of the A9-B9 piping, areas such as the trench under the reactor and the north pipe stand would need to be assessed. However, due to their location, these areas would be difficult or impossible to fully assess prior to decommissioning of the ORR.

"The accessible piping associated with the A9-B9 experiment remains highly contaminated internally, primarily with ^{60}Co. No alpha contamination was detected. The control room is free of contamination other than that associated with the piping.

"The C1 piping was not surveyed as it remains in the reactor pool and the A9-B9 piping in the trench under the reactor and in the north stand pipe also was not accessible.

"Based upon the contamination levels detected, some care will be required to remove the A9-B9 piping. Contamination clothing will be required and respirators needed if there is any risk of suspending this contamination in the air. Pieces of piping will need to be wrapped in plastic if they are to be removed and buried as contaminated waste. Because of the high levels of contamination and small diameter of the pipes, consideration should be given to currently available devices that will pinch the end of the pipe together, sealing in the contamination as they cut the piping."

4.6.3.2 ORR Molten Salt Loop

Installation of this facility began around 1959. Its purpose was to investigate the behavior of aqueous homogeneous fuel solutions and potential construction materials under thermal neutron radiation. This activity was in support of the Homogeneous Reactor Experiment (section 4.4). The facility has not been used since 1967.

The following text is excerpted from the preliminary radiological characterization of this facility by Simpson and Pemberton (1984).

"The major area of the Molten Salt Loop (MSL) still remaining is enclosed in lead shielding with some sections as large as 0.9 m x 2.4 m x 25 cm (3 ft x 8 ft x 10 in.) thick. Removal of this shielding would involve removal of grating from the floors above to position the hoist and disruption of experiments on either side of the HN-1 beam hole; therefore, it was considered beyond the scope of this survey. The survey plan was therefore modified to include a survey of the outside of the lead shielding for transferable contamination and dose rate hazards and a review of the 1982 work in the small equipment chamber to indicate levels of contamination behind the lead shielding.

"The survey of the external areas of the MSL indicated no radiation hazards and no transferable contamination. A scan of the external surfaces indicates no radiation levels above 2 mrad/h. Local absorbed dose rates below 2 mrad/h are not readily measurable in this area due to the background from the reactor. Contamination smears were taken from various locations on the shielding. All smears indicated levels well below limits for establishment of contamination zones (less than 500 dpm/100 cm^2 beta-gamma contamination and less than 20 dpm 100 cm^2 alpha contamination).

"In 1982, the small equipment chamber shielding was opened and extensive work done in that area to replace one of the plugs. Survey and health physics requirements during this work indicate that this chamber is not extensively contaminated. Smears in the area indicated generally less than 3000 dpm/100 cm^2 beta-gamma and less than 20 dpm/100

cm^2 alpha. Clothing requirements were one set of protective clothing. Absorbed dose rates in the area ranged from 10 to 30 mrad/h. These readings were taken while the reactor was shut down, as required for the plug replacement. The large equipment chamber would be expected to be contaminated to levels no higher than the small equipment chamber.

"No environmental considerations are involved in this facility as it is completely housed within Bldg. 3042.

"No significant radiological or nonradiological hazards were observed. Routine ORNL radiation and contamination control procedures would be adequate during decommissioning.

"Prior to D&D activities, a direct survey of the interior of the small and large equipment chambers should be made. This survey would monitor transferable contamination and dose rate levels and should only require a few hours of survey time due to the small area involved. The major factor would be coordinating the movement of the lead shields with the Operations Division and experimenters at ORR to avoid major disruptions of work. However, based on measurements from work done in the area in 1982, engineering evaluations should be possible prior to this survey.

"The major components of the MSL that remain to be decommissioned are the large and small equipment chambers and any material and equipment in them. The exterior surfaces of these chambers have been surveyed and found free of contamination. The interiors were not surveyed, but work done in the small equipment chamber in 1982 indicates that the

radiation hazards are not great. At that time, contamination levels were generally less than 3000 dpm/100 cm^2 and the absorbed dose rate was approximately 10–30 mrad/h with the reactor shut down. One set of contamination clothing was required. Requirement for final facility decontamination and decommissioning would be expected to be the same barring the discovery of large amounts of contamination in the remaining equipment."

4.6.3.3. ORR Maritime Ship Reactor Loop

While early reactors were being developed, this test loop served as a facility in which fuel and other materials could be subjected to radiation and other environmental conditions simulating a pressurized-water reactor system. Early work was in support of the Maritime Reactor Program. The facility was installed in December 1959 and operated until 1962. The system has since been abandoned and is believed to be empty except for possible remnants of the aqueous coolant and its impurities still in the piping system. This loop (also known as the Pressurized Water Loop) is adjacent to the reactor. Process equipment is enclosed in a concrete cubicle next to the control room in the north side of Building 3042, approximately 6 m from the reactor. Process lines in a pipe trench previously connected the system to an in-pile section of the reactor. Ancillary equipment is located on top of the control room.

Earlier work indicated that longer-lived water-borne activity during normal loop operation was principally Cobalt-60 (half life =

5.3 years) and Manganese-54, Iron-59, and Cobalt-58 (all having half lives of less than one year), present as dissolved ionic material.

The following text was excerpted from the preliminary radiological characterization of the site by Simpson and Pemberton (1984).

"The survey plan for the MSRL was primarily aimed at assessing the contamination and dose rate hazards in the shielded equipment room in the basement. This area was known to be somewhat contaminated and to have areas with elevated dose rates. Therefore, the area was smeared and dose rates taken in a number of locations with a "cutie pie" exposure rate meter and an extendable GM probe. The adjacent control room was also surveyed for transferable contamination and dose rate hazards. No attempt was made to survey the process lines in the trench or the underground waste tank as these areas were inaccessible.

"Absorbed dose rates in the shielded equipment room varied from 2 mrad/h at the entrance to 200 mrad/h in the southeast corner. General background over most of the room was 4-6 mrad/h, however higher readings were found around bottoms of the ion exchange columns (up to 30 mrad/h), the cooling water pumps (60 mrad/h) and a pump and surge tank (up to 200 mrad/h), in the southeast corner. The absorbed dose rate in the control room was always less than 1 mrad/h, except in the sample station which read 3 mrad/h.

"A total of 58 smear samples were taken in the shielded equipment room and the control room for the MSRL. Results from the equipment room indicated scattered low level beta-gamma contamination throughout the room. Levels of contamination varied from less than 500 to 1300 dpm/100 cm^2 beta-gamma with an average of 640 dpm/100 cm^2. There was no indication of alpha contamination.

"The sample station in the control room was more highly contaminated, with transferable beta-gamma contamination levels varying from 720 to 72,000 dpm/100 cm^2 with an average of 18,000 dpm/100 cm^2 beta-gamma, indicating the likelihood of spills or leakage in the sample station during operation. Again, no alpha contamination was found. The remainder of the control room was free of contamination. Isotopic analysis of a smear from the sample station indicates the radiation to be primarily from ^{60}Co. A trace of ^{137}Cs was also detected.

"The flash condenser tank and associated hot drain line of this facility are underground on the north side of the building and are inaccessible for surveying. Prior to decommissioning the tank, an environmental assessment including deep soil coring and sampling will be required to determine if there has been any leakage and determine the health physics requirements.

"In addition to the radiation hazards identified, there are also indications that thallium, a highly poisonous

material, is present in the water purification system, and that asbestos may have been used as insulation for process piping. Appropriate hazardous waste handling and disposal practices would need to be specified prior to conduct of decommissioning activities.

"The areas of this facility that are inaccessible for assessment in this project are the dump tank buried outside the building and process lines that extend under the basement floor, up a pipe stand, and into the reactor pool. In the case of the dump tank, no information was found indicating whether it was ever used and contaminated. In the case of the process lines, the only accessible area is in the shielded equipment room where the direct dose rate from the process lines is relatively low compared to other dose rates in the area. Further survey efforts of these areas would require more detailed planning and survey support.

"Decontamination and decommissioning of the MSRL will be a major project requiring extensive health physics coverage. While contamination in the shielded equipment room is currently relatively low-level, the dose rates in the area indicate significant amounts of contamination remaining in the system.

"Contaminated clothing and respirators will be required, and frequent health physics monitoring will be needed. The average dose rate in the room is approximately 4-6 mrad/h and would limit working time to 3-5 hours based on the ORNL daily limit of 20 mrad/day; but much shorter working times would be imposed in higher dose rate areas.

"The sample station in the control room area will also need to be removed under health physics supervision. The absorbed dose rate is relatively low at sample station (3 mrad/h), but transferable contamination levels were high (up to 72,000 dpm/100 cm^2 beta-gamma). Care will be required to prevent the spread of this contamination into the clean area of the control room as the sample station is dismantled. Note that the contamination level is indicative of what might be expected in the removal of components in the equipment room, if dismantlement is necessary.

"The process lines to the reactor were only surveyed in the equipment room but the absorbed dose rates there were not high at their entry point (4 mrad/h) and the problems of removal should be no greater than those discussed in the section on the A9-B9 process lines. The dump tank was not accessible for survey, though there is some uncertainty as to whether it was ever used."

4.6.3.4 Pneumatic Tube Irradiation Facility

This is an experimental facility connecting the ORR to a sampling station in Building 3001. The system was used for rapid transfer of reactor-irradiated liquid samples between the ORR and the sampling station. It was installed in 1965 and operated for about one year. It provided an opportunity to study the direct fission yield of short half-life elements. The tubes are believed to be empty, but little is known about their internal condition. It is not known for certain if a transfer container ever leaked and internally contaminated the tubes.

The following text was excerpted from the preliminary radiological characterization of the site by Simpson and Pemberton (1984).

"The survey plan for this very simple facility was to survey the external areas of the pneumatic tube with a GM survey meter where it was accessible and smear the internal area of the tube at the plugged ends on the north side of the reactor and in the hood in Room 101 of Bldg. 3001.

"Radiation results from this facility indicate no radiation or contamination in the accessible areas.

"Smears taken from inside the tubes at the plugged sections in Bldg. 3042 and from the end of the tube in Bldg. 3001 indicate transferable contamination less than 20 dpm/100 cm^2 alpha and less than 200 dpm/100 cm^2 beta-gamma. Surveys of the external areas of the tube indicate no radiation above background. Background varied from 2 mrad/h near the reactor to less than 0.1 mrad/h at the hood in Bldg. 3001.

"No environmental considerations are involved in this facility despite the section of the tube that is underground beneath Bldgs. 3042 and 3001. Both ends of the tube are free of radioactive contamination and even if some contamination had settled in the low part of the tube underground and somehow leaked through the tube, it would still be contained in the concrete shield surrounding the tube.

"No significant radiological or nonradiological hazards were observed.

"Two areas of the pneumatic tube facility cannot be directly assessed at this time. The first is the tip of the tube extending into the reactor pool. This tip is not expected to be radioactive but may be contaminated due to the reactor water. It cannot be smeared since it is underwater and cannot be directly probed with an underwater probe since the radiation from the reactor core would overwhelm any reading from the tip. A survey of this tip would require very little time but would require that the reactor be shut down and the water lowered below the tip.

"The second area not surveyed is the section of pneumatic tube underground. This section is encased in concrete so there is little point in taking core samples of soil, since any leak from the tube would have been contained in the concrete. The only method for direct assessment would be to unearth and break the concrete exposing the tube. This would appear to be extreme, however, since the exposed sections of the tube are free of contamination.

"The accessible areas of the pneumatic tube appear to be completely free of contamination. These sections should be able to be dismantled with only normal precautions that would apply to Bldg. 3042. That is, the local health physicist should be informed prior to starting the work; and the sections of the tube that are removed should be monitored to ensure that all parts remain free of contamination.

"If the capped section of the tube in the reactor pool is removed, it will require that the reactor be shut down and the

water level lowered. The health physics requirements for this work will depend more upon the reactor than the pneumatic tube. That is, the dose rate in the area would primarily be determined by the amount and location of radioactivity (spent fuel, etc.) in the fuel pool at that time; and the contamination would be determined by the purity of the reactor pool water.

"The section of the pneumatic tube underground has also not been surveyed as it is inaccessible. Since both ends of the tube are clean, this would strongly indicate that the section underground is also clean. Furthermore, the experimental use of the line in 1965 was primarily to measure very short-lived fission products, so only small volumes (a few milliliters) could be handled at any one time due to the large dose rate from these products. Therefore, the amounts that could have leaked would be very small and most of the activity would have decayed long ago. The primary isotope that would remain would most likely be uranium in very small quantities. Due to the relatively low radiotoxicity of uranium, this should present little problem."

4.6.3.5 ORR Gas-Cooled Reactor Loop I

Gas-Cooled-Reactor (GCR) Loop I was installed in 1960 to test gas-cooled, metal-clad fuel elements in a reactor environment. The system operated until 1967. Radiological characteristics are given by Simpson and Pemberton (1984). The following text is excerpted from their report.

"The survey plan for this area primarily consisted of assessing the condition of the cubicles on the second-level balcony, including the ductwork leading to the containment tank. In particular, the ductwork was of concern since the containment tank and ductwork represent the bulk of the material that would have to be removed. Therefore, the plan was to enter the cubicles and crawl down the tunnel as far as possible, checking for contamination and measuring the dose rates in the areas. The control room was also checked for contamination to ensure that it was still clean.

"The tunnel ductwork leading to the containment tank was probed and smeared to the drop-off leading to the tank on the main floor. The absorbed dose rate was in the range of 1-2 mrad/h and never more than 2 mrad/h. This is typical of background dose rates near the reactor. Fifteen smear samples were taken in the ductwork and all were well below limits for establishing a contamination zone (less than 20 dpm/100 cm^2 alpha and less than 500 dpm/100 cm^2 beta-gamma). The cubicles at the northeast corner of the second-level balcony were also surveyed. These areas are small (3.4 m^3, 120 ft^3) and very congested with equipment. Twenty smears were taken of floors and equipment, and all were well below contamination zone limits. The absorbed dose rate in the area was approximately 5 mrad/h. This would appear to be slightly above background, but the increase in dose due to a reduction in shielding at the thimble opening into the pool is uncertain. No particular area or component was identified as

significantly more radioactive than others. Nineteen smear samples were taken in the Loops I and II control room in the northeast corner of the basement. Again, all results were well below the contamination zone limits.

"No environmental considerations are involved in this facility as it is completely housed with Bldg. 3042.

"Besides the risk from low-level amounts of radiation the only hazard observed was what appeared to be asbestos insulation on the inside of the duct leading to the containment tank. Appropriate hazardous waste handling and disposal practices would need to be specified prior to decommissioning activities.

"Two areas were not directly assessed in this study. The first is the section of the thimble extending into the fuel pool which is underwater. An assessment of this component will require that the reactor be shut down and the water level lowered. Note that the radiation and contamination levels at that time will be primarily a matter of reactor conditions rather than radiation associated with the thimble. Also, before equipment is dismantled in the cubicles, a review of all lines possibly connected to the thimble should be done to ensure that there is no possibility of cutting a line that is still open to the reactor pool. This would appear to be important since the design of the experiment implies that the possibility existed of reactor water escaping into the cubicle and containment tank.

"The second area not assessed was the containment tank, as it is completely enclosed in shielding. However, the duct leading to the interior of the tank was surveyed and found free of contamination, so it would appear likely that the tank is clean.

"The major structures associated with the GCR-Loop I appear to be free of contamination. In particular, the ductwork leading to to the containment tank is clean, implying that the internals of the tank and structures enclosed in shielding on the main floor are free of contamination. This is in agreement with operators who did not believe that the tank designed for emergency containment of reactor water had ever been used.

"The equipment in the cubicles on the second-level balcony are free of external contamination and appear to have only low levels of contamination (if any) internally, based on external dose rates. No ready access to pipe internals was found to measure internal contamination and plug samples were not considered advisable without confirmation that no access to the pool existed. However, assuming the external dose rates found (5 mrad/h) were completely due to internal contamination, only moderate levels of contamination would be expected, requiring only one set of protective clothing. It is expected that respirators would only be required during cutting or grinding operations."

"The control room area in the basement was also surveyed and found free of contamination."

4.6.3.6 ORR Gas-Cooled Reactor Loop II

This facility was built in 1962 and operated briefly through 1963. Its purpose was to study problems of fuel element stability, rate of fission-gas release from fuels, and maintenance of a system contaminated with radioactive fission products.

The following text is excerpted from the preliminary radiological characterization prepared by Simpson and Pemberton (1984).

> "The survey plan for this area consisted of a complete survey of the hot cell area to detect transferable contamination and dose rate surveys during reactor shutdown and operation to determine dose rates. The absorbed dose rate (mrad/h) during shutdown was determined by an ORNL "cutie pie" ionization chamber. The dose equivalent rate (mrem/h) during operation was determined by summing the measurement from the ionization chamber with measurements made with ORNL fast and thermal neutron meters.
>
> "During shutdown, the Loop II hot cell had radiation readings less than 1 mrad/h except in the east end. Even in the east end, radiation levels were relatively low (2-5 mrad/h) except for a valve labeled HV7000 which read 10 mrad/h. During operation, the dose equivalent rate was considerably higher, due to radioactivity from reactor coolant water flowing in pipes in and near the cell and from low levels of neutron leakage from the reactor core. The east end of the room was not surveyed during reactor operations due to elevated readings (> 100 mrad/h) and the possibility of narrow, intense beams of radiation.

"Seventy-three smears were taken throughout the Loop II facility. All smear samples were well below contamination zone limits (less than 20 dpm/100 cm^2 alpha and 500 dpm/100 cm^2 beta-gamma) indicating the area was free of transferable contamination. The Loop II control room was surveyed as part of the Loop I survey and was also found to be free of contamination.

"No environmental considerations are involved in this facility as it is completely housed within Bldg. 3042.

"Other than radiation, the only risk consideration observed was asbestos insulation used in the Loop II hot cell area. Appropriate hazardous waste handling and disposal practices would need to be specified prior to the conduct of decommissioning activities.

"Areas of GCR-Loop II which will require more detailed analysis are those associated with the in-core sections. In particular, any lines which are open to reactor pool water should be identified and isolated if possible. Also, an up-to-date detailed plan must be developed for removal of the in-core plugs.

"The GCR-Loop II facility is a large, intricate experimental area which is going to require careful planning and preparation to dismantle. The major task appears to be isolating the in-core components and removing them from the system.

"Other than the in-core section, this survey indicates that the remainder of the facility has little, if any, contamination. Also, there appears to be some question as to

how much of the facility was ever used. If, in fact, little of the facility was operated, much of the equipment in the hot cell can be removed with only limited health physics coverage if done while the reactor is shut down. Note that work during reaction operations will be limited due to the increased dose rate. One important step prior to the removal of equipment will be the verification that all components are isolated from the reactor pool so that reactor water is not accidentally released. If possible, all uncontaminated equipment should be removed prior to the removal of the in-core plugs to prevent the contamination of potentially reusable items."

4.6.4 Data Deficiencies

Several items could not be radiologically surveyed due to inaccessibility. In some cases a survey would require that the reactor be shut down and the reactor pool water level be lowered. Specific data deficiencies for each facility are summarized below. Much of the material in the following sections is taken from Simpson and Pemberton (1984).

4.6.4.1 Gas-Cooled Reactor Loops A9, B9, and C1

Piping located in the reactor pool has not been surveyed. A survey would require very little time but would require that the reactor be shut down and the water lowered below the piping (if possible).

Also, Simpson and Pemberton (1984) state that "the A9-B9 piping can only be readily surveyed at the control room, though

this should be a reasonable estimate of the contamination throughout. If it is proposed to remove all of the A9–B9 piping, areas such as the trench under the reactor and the north pipe stand would need to be assessed. However, due to their location, these areas would be difficult or impossible to fully assess prior to decommissioning of the ORR."

4.6.4.2 Molten Salt Loop

Simpson and Pemberton (1984) recommend that prior to decontamination and decommissioning activities, a direct survey of the interior of the small and large equipment chambers be made to monitor transferable contamination and dose rate levels. This survey should only take a few hours due to the small area involved.

4.6.4.3 Maritime Ship Reactor Loop

Areas of this facility that were inaccessible for assessment are the dump tank buried outside the building and process lines that extend under the basement floor, up a pipe stand, and into the reactor pool. No information was found indicating whether the dump tank was ever used and contaminated. In the case of the process lines, the only accessible area is in the shielded equipment room where the direct dose rate from the process lines is relatively low compared to other dose rates in the area.

4.6.4.4 Pneumatic Tube Irradiation Facility

Two areas of this facility have not been directly assessed. The first is the tip of the tube extending into the reactor pool.

A survey of this tip would require very little time but would require that the reactor be shut down and the water lowered below the tip.

The second area not surveyed is the section of pneumatic tube underground. The only method or direct assessment would be to unearth and break the concrete exposing the tube.

4.4.6.5 Gas-Cooled Reactor Loop I

Two areas of this facility have not been directly assessed. The first is the section of the thimble extending into the fuel pool which is underwater. An assessment would require that the reactor be shut down and the water level lowered. Also, before equipment is dismantled in the cubicles, a review of all lines possibly connected to the thimble should be done to ensure that there is no possibility of cutting a line that is still open to the reactor pool. This would appear to be important since the design of the experiment implies that the possibility existed of reactor water escaping into the cubicle and containment tank.

The second area not assessed was the containment tank, as it is completely enclosed in shielding. However, the duct leading to the interior of the tank was surveyed and found free of contamination, so it would appear likely that the tank is clean.

4.4.6.6 Gas-Cooled Reactor Loop II

Areas requiring more detailed analysis are those associated with in-core sections. In particular, any lines which are open to reactor pool water should be identified and isolated if possible.

4.6.5 Decommissioning Options

The preliminary decommissioning report of Horton (1984) summarizes the recommendation for each facility. Complete dismantlement was recommended for ORR Gas-Cooled Reactor Loops A9, B9, and C1 (Facility 1), The Molten Salt Loop (Facility 2), and the ORR Gas-Cooled Reactor Loop I (Facility 5); partial dismantlement was recommended for the other facilities (Table 9). It is estimated that the decommissioning of these six facilities would take five years and cost $5,225,000 FY 1985 first-quarter dollars. Further details, including cost-estimate work sheets, are provided by Horton (1984, including Appendix I). Timing, costs, and waste volume projections for each year of decommissioning activity are summarized in Table 10. The water-to-air heat exchanger, discussed in Section 4.5, is also included in this table.

Table 9. Recommended Decommissioning Alternatives for the ORR Experimental Facilities (from Table 1 of Horton, 1984).

Table 9. Recommended Decommissioning Alternatives and Cost Estimates for the ORR Experimental Facilities. (Source: Horton 1984)

Facility	Recommended Alternative	Comments	Cost 1st-Quarter 1985 Dollars
1. ORR Gas-Cooled Reactor A9-B9, C1	Complete Dismantlement	Including piping inside pool and standpipes	430,000
2. Molten Salt Loop	Complete Dismantlement	Will require fabrication of new shield plug and relocation of some lines	679,000
3. Maritime Ship Reactor	Partial Dismantlement	Leave the buried tank outdoors; dismantle shield walls as required; relocate active instrumentation	1,440,000
4. Pneumatic Tube	Partial Dismantlement	Leave underground piping in place	248,000
5. ORR Gas-Cooled Reactor Loop I	Complete Dismantlement		998,000
6. ORR Gas-Cooled Reactor Loop II	Partial Dismantlement	Leave the structural walls of the hot cell, the beam hole plug, and crane; re-pipe cooling water lines as required	1,440,000
7. Heat Exchangers	Complete Dismantlement		212,000

Table 10. Timing, cost, and waste volume projections for decommissiong the ORR Experimental Facilities, including the heat exchangers discussed in Section 4.5. The cost estimates are in first-quarter FY 1985 dollars. (Source: Myrick 1984).

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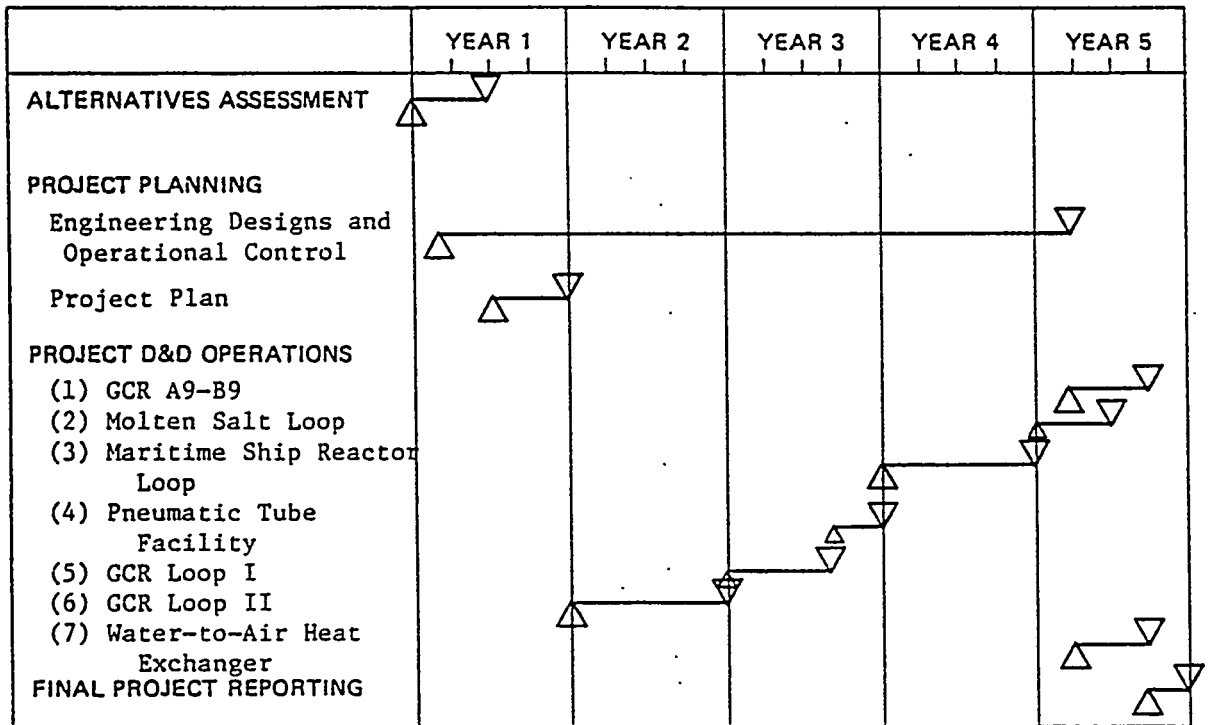


	YEAR 1	YEAR 2	YEAR 3	YEAR 4	YEAR 5
PROJECT COSTS (SK)					
Annual Expense	260	1350	1300	1340	1150
Capital Equipment	–	10	10	40	–
Total	260	1360	1310	1380	1150
WASTE VOLUME PROJECTIONS					
Solid (m^3)					
Soil (LLW)	–	–	–	–	20
(TRU)	–	–	–	–	–
Rubble (LLW)	–	–	–	–	80
(TRU)	–	–	–	–	–
Metal (LLW)	–	65	75	95	80*
(TRU)	–	–	–	–	–
Misc. (LLW)	–	–	–	–	–
(TRU)	–	–	–	–	–
Total (LLW)	–	65	75	95	180
(TRU)	–	–	–	–	–
Liquid (m^3)					
Process Waste	–	–	–	–	–
LLW	–	–	–	–	–

*Estimates of LLW metal waste during the fifth year are $82m^3$ for the heat exchangers (Horton 1984, referenced in section 4.5 of the present report), and $75m^3$ for the ORR experimental facilities (Horton 1984, referenced in present section). The latter figure ($75m^3$) was probably inadvertently neglected in arriving at the $80m^3$ figure given above.

4.6.5 References

Several references to detailed studies, design reports, etc., for individual facilities are given by Horton (1984). More general references are given below.

Hamrick, T. P., and J. H. Swanks. The Oak Ridge Research Reactor - A Functional Description. ORNL-4169 (Vol. 1). Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Horton, J. R. 1984. Preliminary Decommissioning Study Reports Volume 9: ORR Experimental Facilities. X-OE-231, Vol. 9. Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Myrick, T. E. 1984. The ORNL Surplus Facilities Management Program Long-Range Plan. ORNL/TM-8957, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Simpson, D. R., and J. H. Pemberton. 1984. Preliminary Radiological Characterization of the ORR Experimental Facilities. ORNL/CF-83/250. Oak Ridge National Laboratory, Oak Ridge, Tennessee.

4.6.6 Persons Knowledgeable about the Site.

(General Operations) T. P. Hamrick, ORNL, 615/574-7078

(Health Physics) J. H. Pemberton, ORNL, 615/574-6705

4.7 Site 7: Tower Shielding Facility Equipment

4.7.1 Purpose of Facility

This facility was constructed to provide an economical means for obtaining shielding performance data, free from ground scatter or structure scatter, for use in the development of reactors. This is accomplished by placing a small, low-power reactor in a cylindrical tank and measuring the radiation from the reactor at various angles while the reactor and tank are suspended in the air about 60 meters above the ground.

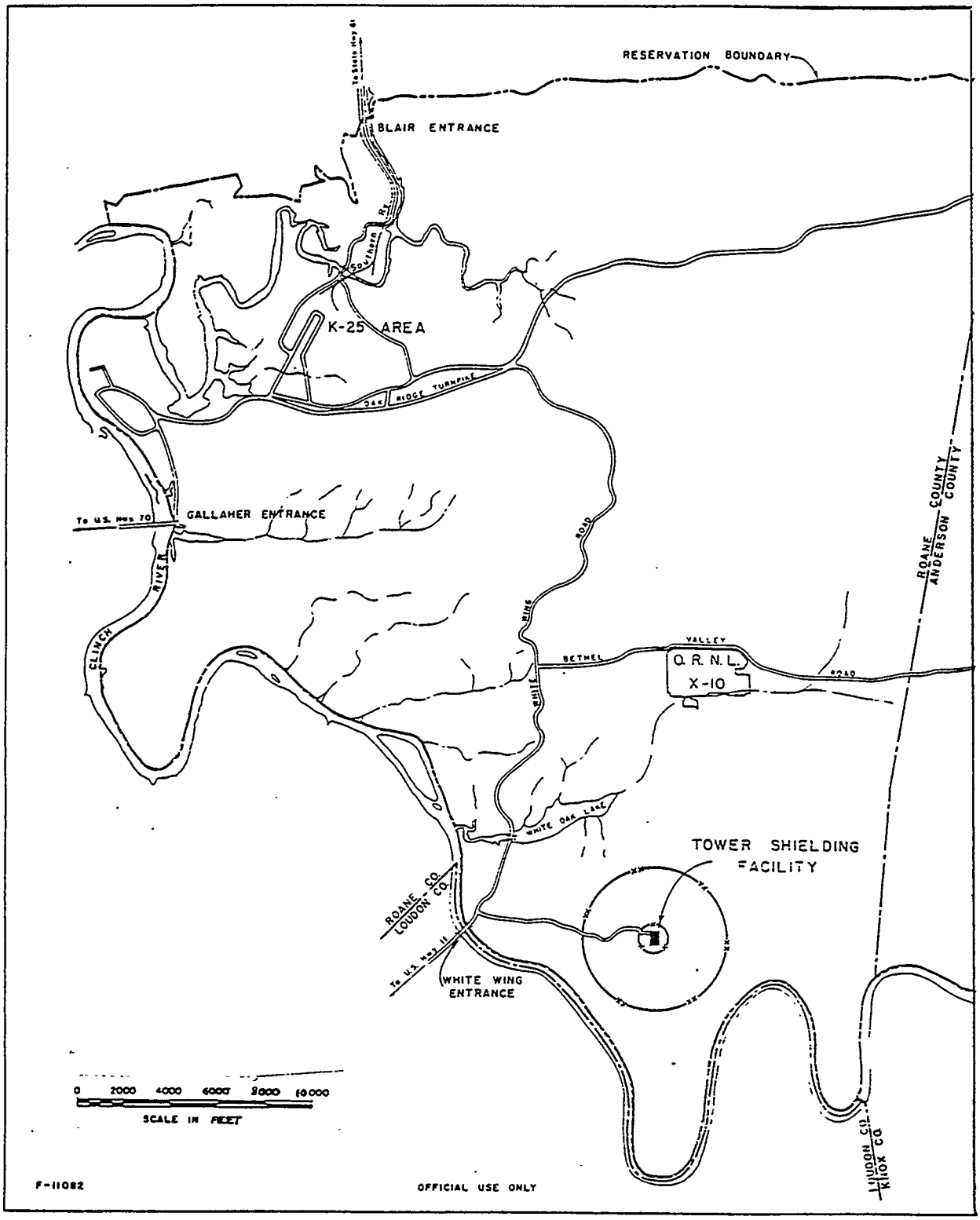
The tall towers have also proven useful for numerous drop tests.

4.7.2 Location and Description of Site

The facility is located about 3.5 to 4 Kilometers south of the ORNL main complex (Figure 23). It was completed in 1954, is still in use, and will continue to operate. The surplus equipment considered here consists of a large volume of leftover materials many of which may be radioactively contaminated to some extent.

Structures on the site include the four large towers that are used to elevate the equipment being studied. These towers are arranged in rectangular fashion and extend about 100 meters above the ground. An earth-covered blockhouse beneath the towers provides a control house. However, the surplus items considered here are the large volumes of miscellaneous equipment and other material that has accumulated over the years as a result of work conducted at the site. Much of this material has simply been junked near the site so that from several approaches the site has

Figure 23. Location of the Tower Shielding Facility.


RESERVATION BOUNDARY
BLAIR ENTRANCE
K-25 AREA
OAK RIDGE TURNPIKE
To U.S. Hwy 70
GALLAHER ENTRANCE
CLINCH RIVER
WHITE WING ROAD
BETHEL VALLEY ROAD
O. R. N. L.
X-10
WHITE OAK LAKE
TOWER SHIELDING FACILITY
ROANE CO.
LOUDON CO.
To U.S. Hwy 11
WHITE WING ENTRANCE
ROANE COUNTY
ANDERSON COUNTY
LOUDON CO.
KNOX CO.
0 2000 4000 6000 8000 10000
SCALE IN FEET
F-11082
OFFICIAL USE ONLY

the appearance of a junkyard. Some typical photos from around the site are presentedas Figures 24, 25, and 26. Figure 24 shows the Towers in the background.

Underlying the site is gray, bedded, dolomitic limestone of the Knox formation interbedded with conspicuous chert zones. The formation is about 800 meters thick and without prominent faults in the vicinity of the site. The overlying soil is mainly silty clay, usually red but in some places yellow, interspersed with hard cherts of various sizes. More details on the geology and hydrology of the site are found in Chapter 2 and indicated references. The climate and ecological characteristics of the site are the same as for the other six sites in this report, and are discussed in Chapter 2.

Deer are often seen in large numbers at the site. The vegetation surrounding the site appears normal, with apparently healthy trees often touching items of the surplus equipment in question (Figures 24, 25, and 26).

4.7.3 Description of Contamination

According to a recent (March 1986), scoping survey conducted by Houser and Simpson (1986), no significant radiation hazards within the Tower Shielding Facility present an exposure potential to personnel. Although much of the surplus equipment has been exposed to neutron fluxes, only a small portion of that equipment (< 15%) contains measurable levels of radioactivity. The radioactivity is of an induced nature, primarily gamma, and is adequately shielded such that chances of exposure to personnel are minimal. Radioisotope containers, shields, and other such

Figure 24. Surplus material that has been abandoned near the Tower Shielding Facility, with the towers visible in the background.

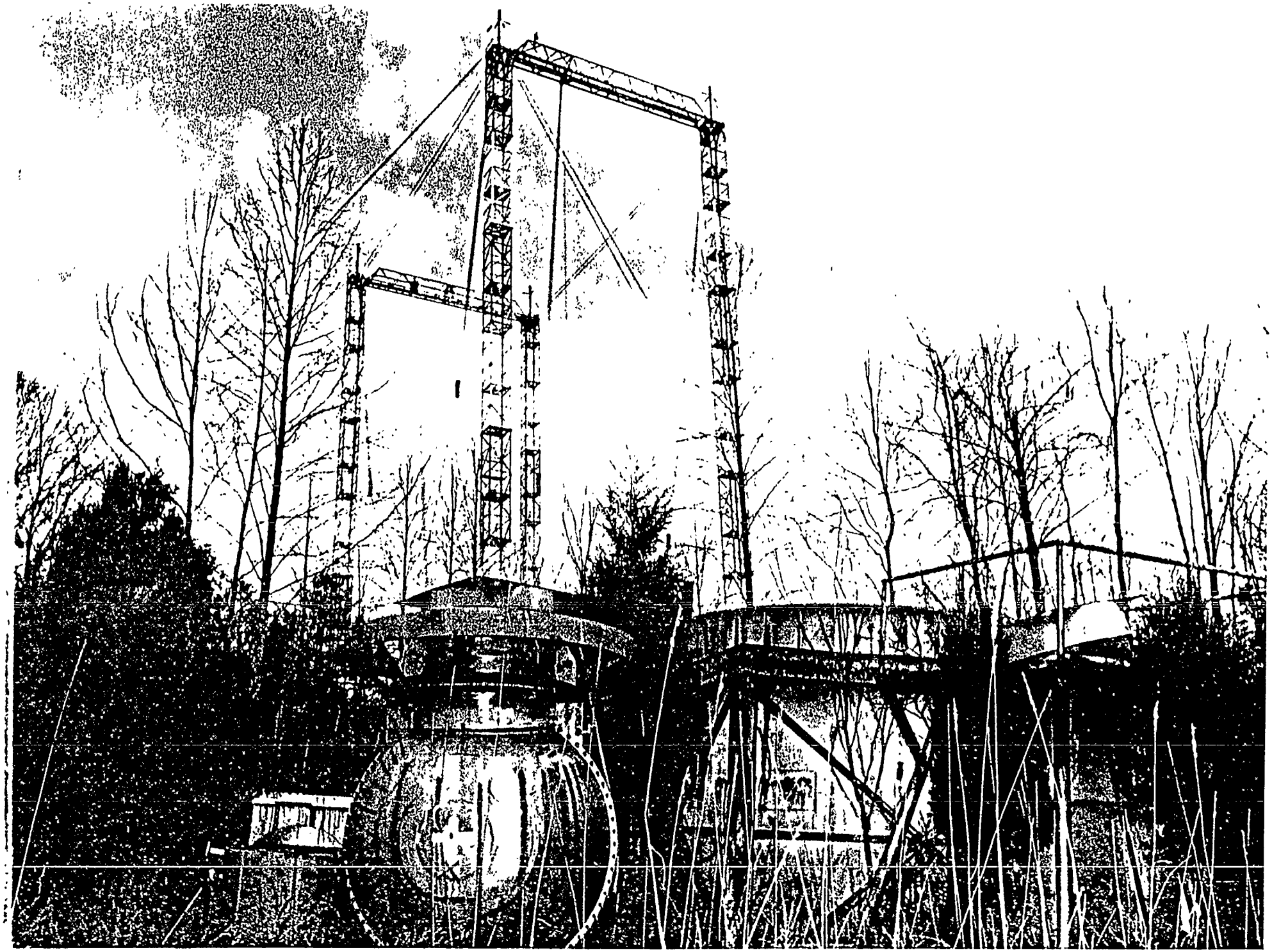


Figure 25. Surplus material that has been abandoned near the Tower Shielding Facility.

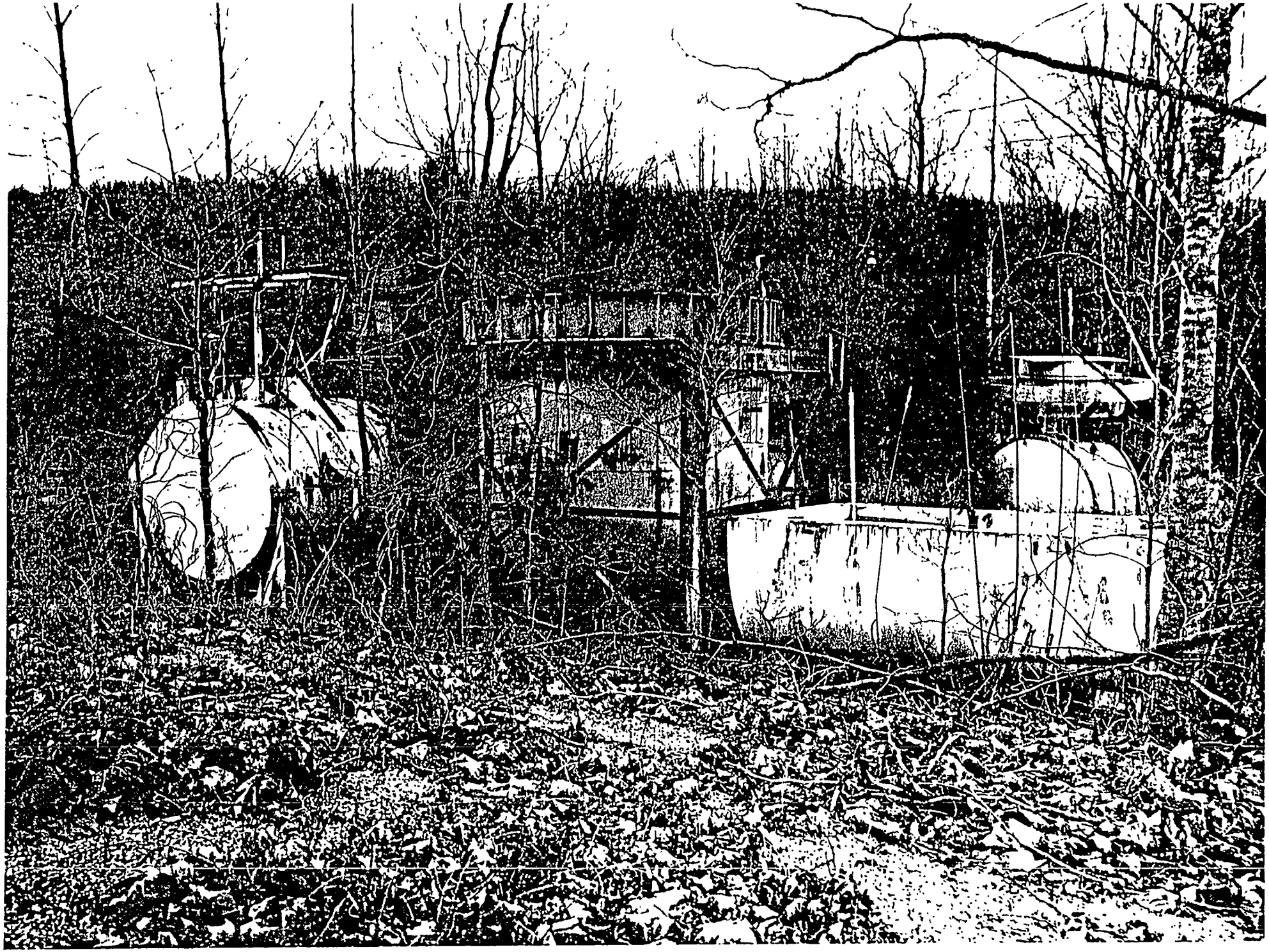


Figure 26. Another typical view of surplus material that has been abandoned near the Tower Shielding Facility.

equipment contains internal radiation levels of up to 10 R/h. However, shielding reduces radiation levels to < 5 mR/h at the surface. According to the scoping survey of Houser and Simpson (1986), "there is no evidence of transferable contamination within the facility."

A major source of shielding at the TSF has utilized sodium. One source of sodium contains 300 55-gallon drums. Also, several large shields contain sodium. The sodium is adequately contained, except for one shield that is slightly ruptured and has evidence of sodium leakage. The leak may have sealed itself, but this phenomenon should probably be given some attention.

Another area of concern noted by Houser and Simpson (1986) involves a group of 10 depleted uranium sheets (1/2" x 1' x 1') and 3 larger sheets (1" x 5' x 5') that are excessively corroded and are contained in an aluminum box. These sheets are a radiological and environmental hazard. Though presently adequately contained, continued corrosion could cause a serious problem.

The volume of surplus material is nontrivial. Many of the surplus items have volumes on the order of several cubic meters, and several of these items can be seen along any pathway on the hill to the south and southwest of the facility (Figures 24, 25, and 26).

Much of the surplus material on top of the hill ("on the pad" above the control house) has been transported to the K-25 site for use there. Apparently some internal funding was made available to dispose of much of the material that was on the pad. J. W. Paul (listed below) is likely to be the best source of current information about these matters.

4.7.4 Data Deficiencies

As indicated earlier, there are many pieces of assembled equipment scattered within the facility boundaries which will require detailed radiological evaluations in the event decontamination or scrapping operations are attempted. Waste-volume estimates have not yet been made, though it is obvious that the volume is large. Except as noted below, cost estimates for decommissioning have not yet been made.

4.7.5 Decommissioning Options

A decommissioning report has not been prepared for this site. Some personnel who are familiar with the site (J. W. Paul, Leo Holland) have estimated that it would cost $65,000 just to clean up the surplus material near the site at the top of the hill. However, to clean up the items strewn over the south and west sides of the hill would likely cost much more.

4.7.6 References

ORNL Engineering and Mechanical Division. 1955. Completion Report on the Tower Shielding Facility ORNL-CF 55-1-165 Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Holland, L. B. and J. O. Kolb. 1970. Tower Shielding Reactor II Design and Operation Report: Vol. 2 - Safety Analysis. ORNL/TM-2893, Vol. 2. Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Houser, B. S., and D. R. Simpson. 1986. Scoping surveys for Surplus Contaminated Facilities Program (SCFP); Tower Shielding Facility.

4.7.6 Persons Knowledgeable about the Site

L. B. Holland, Tower Shielding Facility, ORNL, 615/574-7210

J. L. Hull, Tower Shielding Facility, ORNL, 615/574-7210

F. J. Muckenthaler, Tower Shielding Facility, ORNL, 615/574-7211

J. W. Paul, Plant and Equipment Division, ORNL, 615/574-4257

5. PRIORITY AND RISK CONSIDERATIONS

The first six sites described in this report are within the Surplus Facilities Management Program (SFMP). Decommissioning studies have already been carried out for these facilities. Some of the resulting information has been summarized by Myrick (1984; Reference given in Section 4.6.5), as presented in Table 11. Project priority considerations and rankings for decommissioning these facilities, relative to other SFMP sites, are given in Table 12 (also modified from Myrick 1984). A similar table, including all the Remedial Action Program sites (in the 13 categories in Table 1), is still a long way from preparation because much of the necessary information has yet to be obtained and summarized.

Table 11. ORNL SFMP project description summary for the Experimental Reactor Facilities (Modified from Myrick 1984)

Table 11. ORNL SFMP project description summary for the Experimental Reactor Facilities (modified by Myrick, 1984)

Facility	Service Dates	Proposed Disposition Mode	Estimated[a] Cost ($ x 10^3)	Estimated Project Duration	Special Considerations
ORNL Graphite Reactor	1943-1963	Passive Protective Storage	$8,000	4 years	Registered National Historical Landmark
Molten Salt Reactor Experiment	1965-1969	Fuel Disposal, Facility Entombment	$30,000	10 years	Uranium fuel remains on-site and will require disposal prior to facility decommissioning
Low Intensity Test Reactor	1951-1968	Dismantlement	$4,500	5 years	Location and structural condition limits decommissioning options.
Homogeneous Reactor Experiment	1957-1961	Entombment	$14,000	6 years	Associated waste retention pond will require entombment.
ORR Experimental Facilities[b]	1958-1973	Dismantlement	$5,500	5 years	Experimental areas have potential for reuse.

a. In First-Quarter FY 1985 dollars.
b. Note that the water-to-air heat exchangers (Trane coolers) are included.

Table 12. ORNL SFMP project priorities for the Experimental Reactor Facilities (Modified from Myrick 1984)

Table 12. ORNL SFMP project priorities for the Experimental Reactor Facilities (Modified from Myrick 1984)

Facility	Priority assessment categories						
	Programmatic Concerns		Economic Considerations Ranking	Health Risks Ranking	Cost Effectiveness Ranking	Weighted Rank Totals[a]	Project priority[b]
	Ranking	Comment					
Molten Salt Reactor	High	Complex, long-term project requiring available key personnel	3	2	2	25	4
Homogeneous Reactor Experiment	Medium	Environmental concern over pond and other outlying areas	3	3	2	28	7
ORNL Graphite Reactor	Low	National Historic Landmark; building occupied	1	5	1	22	11
Low Intensity Test Reactor	Low	Low-levels of contamination in a contained structure	2	4	2	26	12
ORR Experimental Facilities including water-to-air heat exchangers.	Low	Low potential for near-term facility reuse	3	6	3	39	14

a. The weighted rank totals are calculated as 5X (Economic Conside ratings Ranking) + 3X (Health Risks Ranking) + 2X (Cost Effectiveness Ranking). A low rank total implies a high priority - i.e., weightings are inversely proportional to assigned importance.

b. Project priorities are given in terms of ranking, from 1 to 16, of all 16 SFMP sites.

The Molten Salt Reactor Experiment and the Homogeneous Reactor Experiment were given high and medium priorities, respectively, for reasons summarized in Table 12. The other three experimental reactor sites in the SCMP were given "low" priority.

The Tower Shielding Facility Equipment (Site 7 in this report) is within the Surplus Contaminated Facility Program (SCFP). As such, it has not been investigated as thoroughly for decommissioning as have the SFMP sites. Decommissioning studies of SCFP sites are planned for FY 1987.

Houser and Simpson (1986) noted that the Tower Shielding Facility surplus equipment is, for the most part, in sound condition and that the facility presents no significant problems except for the storage and disposal of some sodium and the presence of some oxidized uranium sheets (that are presently adequately contained). These problems appear correctable with appropriate actions. There does not appear to be any great urgency in initiating decontamination and decommissioning operations at this site, though it should be noted that the site has not yet been throughly examined and that the large volumes of surplus material will need to be cleaned up eventually. A more thorough examination would include checking for hazardous chemical storage/contamination by various materials such as lead, cadmium, beryllium, etc.

Radiological hazards are, of course, of primary interest in the decommissioning of experimental reactor facilities. However, nonradiological hazards could become a problem if they are neglected. For example, asbestos is an item of concern in at least three facilities. Chemical hazards should be considered in

decommissioning the Molten Salt Reactor Experiment. Thallium may be present in the water-purification system of the Maritime Ship Reactor Loop of the surplus ORR Experimental Facilities. These and other potential nonradiological hazards have been identified in the respective decommissioning reports or, for the Tower Shielding Facility, in the scoping survey. these potential hazards can normally be handled routinely, and should present no appreciable threats so long as they are not overlooked amidst the generally greater concern with radiological hazards.

As noted at the end of Chapter 1, these sites are to be managed in a structured program of routine maintenance and surveillance until appropriate decontamination and decommissioning activities are completed.

LETTER REPORT RAP86-22/R1

#1827 (add)

Inventory of ORNL Remedial Action Sites: 10. Radwaste Facilities*

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Oak Ridge, Tennessee 37831

July 31, 1986

Operated By

Martin Marietta Energy Systems, Inc.
for the
U.S. Department of Energy

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#1827 (add)

*Based on work performed at Oak Ridge National Laboratory, operated by Martin Marietta Energy Systems, Inc., under Contract No. DE-AC05-84OR21400 with the U. S. Department of Energy.

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SUMMARY

The Radwaste Facilities, identified as remedial action sites of the Environmental Restoration and Facilities Upgrade (ERFU) program, have been reviewed by personal interview with persons knowledgeable about the site. Additional information was abstracted from reports, publications, and engineering drawings. Results of each review are designed to provide information on the facility location and description, history, contaminants, extent of contamination, geological, hydrological and ecological data, regulatory status, major references (reports, drawings, etc.), and identification of knowledgeable persons concerning the site.

1. OBJECTIVES

The primary purpose of the remedial action task is to provide corrective measures and facility decommissioning necessary to place the Oak Ridge National Laboratory (ORNL) in compliance with applicable environmental and health regulations for radioactive and hazardous chemical wastes. Individual sites will be decontaminated in accordance with priorities prescribed to control the potential for human exposure and relative magnitude of ecological impacts. Remedial action is a component of the Laboratory's Environmental Restoration and Facilities Upgrade program (ERFU). In the program, remedial actions will be conducted at locations where past management practices or research activities have resulted in contamination of facilities or the environment that necessitate decommissioning or corrective actions. Remedial actions are divided into two activities: (1) Site Decommissioning Program (Surplus Contaminated Facilities Program-SCFP and Surplus Facilities Management Program-SFMP) and (2) Site Corrective Measures Program (SCMP).

The Site Decommissioning Program is designed for the management of surplus facilities (primarily structures), contaminated with radioactive materials and hazardous chemicals, from the end of their operating life until final facility disposition. The Site Corrective Measures Program is designed for the management of inactive sites (process ponds, spill sites, environmental research areas), contaminated with radioactive materials and hazardous wastes, leading to the eventual stabilization of these sites in accordance with environmental regulations.

Remedial Action Program sites at Oak Ridge National Laboratory (ORNL) comprise 13 distinct areas among the several programmatic categories: (1) SCMP, (2) SCFP, and (3) SFMP. In a series of reports, an inventory of each area is presented. The summary is the product of pertinent information abstracted from readily available literature and knowledgeable individuals on known contaminated sites requiring remedial action. The information includes site location and description, history, contaminants and their characteristics, extent of contamination, regulatory status, lists of pertinent drawings, photographs, reports concerning the site, and individuals knowledgeable of the site and site activities. The subject of this report is the numerous radwaste facilities that are currently being utilized for waste holding or processing, or have the potential for future waste handling, or have contained contaminated liquids in the past and are now abandoned.

2. BACKGROUND

Since 1943, several storage tanks have been installed at ORNL for storage of liquid radioactive waste. Twenty-one of these tanks, designated as surplus, are involved in the ORNL Surplus Facilities Management Program. Those tanks are identified in Table 2.1.1, and each is discussed in an informative summary. Locations of the waste tanks and other facilities are noted in Fig. 2.1.1. The summaries are designed to provide information of site location, history, contaminants and their characteristics, and extent of contamination. Also included are major references on each site and identification of knowledgeable

Table 2.1.1 Summary of histories and radiological inventories of waste tanks in the Surplus Facilities Management Program (SFMP)

Tank	Location	Service dates	Reason for abandonment	Construction	Capacity (ℓ)	Present content	Present volume (ℓ)	Major isotopes	Quantity of radioactivity	Sources of information
W-1	Site 3023	1943-1960's	Leaks	Gunite	1.9E4	Liquid	3.9E3	Cs-137, Sr-90, Eu-152, Eu-154, Cm-244, U-238, and TRU	0.142 Ci	Horton 1984b Huang 1984b
W-2	Site 3023	1943-1960's	Leaks	Gunite	1.9E4	Liquid/ sludge	5.0E3	Cs-137, Co-60, Sr-90, Eu-152, Eu-155, Eu-154, Cm-244, U-233, U-238, TRU	230 Ci	Horton 1984b Huang 1984b
W-3	Site 3023	1943-1960's	Collects water	Gunite	1.6E5	Liquid/ sludge	1.0E5	Cs-137, Sr-90, Cm-244, U-233, U-238, TRU	1.066 Ci	Myrick 1984 Horton 1984b Huang 1984b
W-4	Site 3023	1943-1960's	Collects water	Gunite	1.6E5	Liquid/ sludge	6.6E4	Cs-137, Co-60, Sr-90, Cm-244, U-233, U-235, U-238, TRU	207 Ci	Horton 1984b Huang 1984b
W-5	Site 3507	1943-1978	Wall deterioration	Gunite	6.6E5	Liquid/ sludge	1.4E4	Sr-90, Cs-137	330 Ci	Huang 1984a Horton 1984a Myrick 1984
W-6	Site 3507	1943-1978	Wall deterioration	Gunite	6.6E5	Liquid/ sludge	2.6E5	Sr-90, Cs-137	2190 Ci	Huang 1984a Horton 1984a Myrick 1984
W-7	Site 3507	1943-1978	Wall deterioration	Gunite	6.6E5	Liquid/ sludge	7.2E4	Sr-90, Cs-137	1850 Ci	Huang 1984a Horton 1984a Myrick 1984

Table 2.1.1 Summary of histories and radiological inventories of waste tanks in the Surplus Facilities Management Program (SFMP) (CONT'D)

Tank	Location	Service dates	Reason for abandonment	Construction	Capacity (ℓ)	Present content	Present volume (ℓ)	Major isotopes	Quantity of radioactivity	Sources of information
W-8	Site 3507	1943-1978	Wall deterioration	Gunite	6.6E5	Liquid/ sludge	1.1E5	Sr-90, Cs-137	8900 Ci	Huang 1984a Horton 1984a Myrick 1984
W-9	Site 3507	1943-1978	Wall deterioration	Gunite	6.6E5	Liquid	8.5E4	Sr-90, Cs-137	730 Ci	Huang 1984a Horton 1984a Myrick 1984
W-10	Site 3507	1943-1978	Wall deterioration	Gunite	6.4E5	Liquid	3.5E5	Sr-90, Cs-137	8290 Ci	Horton 1984a Myrick 1984
W-11	Site 3023	1943-1948	Leaks	Gunite	5.8E3	Liquid/ sludge	1.2E3	Cs-137, Sr-90, Cm-244, TRU	<0.0003 Ci	Myrick 1984 Horton 1984b Huang 1984b
W-13	Site 3023	1940's-1958	No use	Stainless-steel	7.8E3	Liquid	1.7E3	Cs-137, Co-60, Sr-90, Eu-154, Cm-244, U-233, U-235, U-238, TRU	330 Ci	Myrick 1984 Horton 1984b Huang 1984b
W-14	Site 3023	1940's-1958	No use	Stainless-steel	7.8E3	Liquid	~0.5E3	Cs-137, Sr-90, Co-60, Cm-244, U-233, U-238, TRU	14 Ci	Huang 1984b
W-15	Site 3023	1940's-1958	No use	Stainless-steel	7.8E3	Liquid/ sludge	N.S.	Cs-137, Sr-90	N.S.	Myrick 1984 Horton 1984b

Table 2.1.1 Summary of histories and radiological inventories of waste tanks in the Surplus Facilities Management Program (SFMP) (CONT'D)

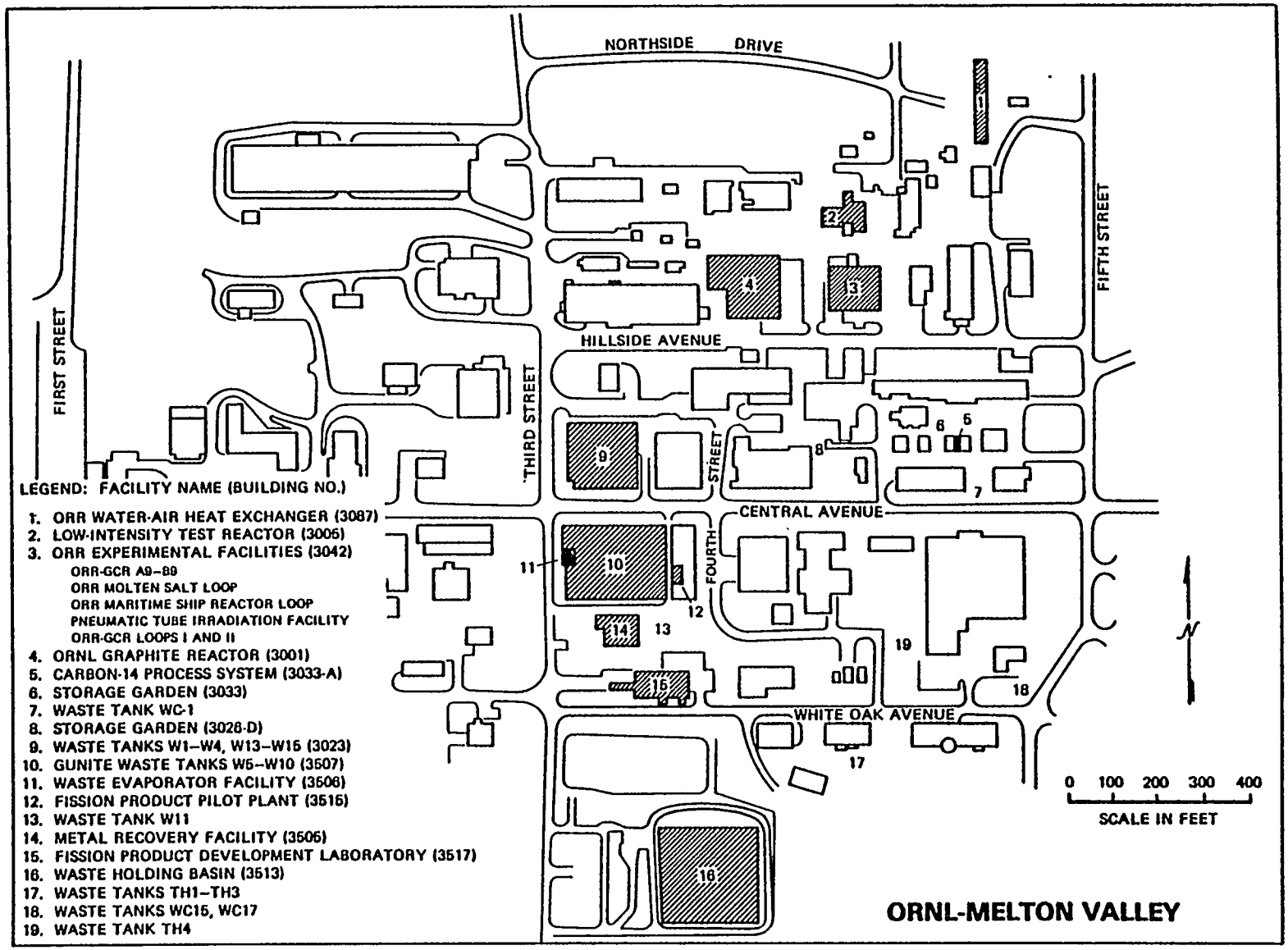
Tank	Location	Service dates	Reason for abandonment	Construction	Capacity (ℓ)	Present content	Present volume (ℓ)	Major isotopes	Quantity of radioactivity	Sources of information
WC-1	SW of 3037	1950-1968	Leaks	Stainless-steel	7.8E3	Empty	N.S.	Cs-137, Sr-90, Co-60	N.S.	Myrick 1984 Horton 1984b Huang 1984b
WC-15	SE of 3587	1953-1960's	Leaks	Stainless-steel	3.9E3	Liquid/ sludge	N.S.	Cs-137, Sr-90, Co-60, TRU	N.S.	Myrick 1984 Horton 1984b Huang 1984b
WC-17	SE of 3587	1953-1960's	Leaks	Stainless-steel	3.9E3	Liquid/ sludge	3.9E3	Cs-137, Sr-90, Co-60, Eu-154, Cm-244, TRU	5 mCi	Huang 1984b
TH-1	S of 3503	1952-1970	No use	Stainless-steel	9.7E3	Liquid	1.8E3	Cs-137, Sr-90, Co-60, Cm-244, Th-232, TRU	1.1 Ci	Myrick 1984 Horton 1984b Huang 1984b
TH-2	S of 3503	1952-1970	No use	Stainless-steel	9.7E3	Empty	N.S.	N.S.	N.S.	Myrick 1984
TH-3	S of 3503	1952-1970	No use	Stainless-steel	1.3E4	Liquid	2.6E3	Cs-137, Co-60, Sr-90, Cm-244, Th-232, TRU	1.2 Ci	Myrick 1984 Horton 1984b Huang 1984b
TH-4	S of 3500	1952-1970's	Filled/sludge	Gunite	5.4E4	Liquid/ sludge	5.4E4	Cs-137, Sr-90, Cm-244, Th-232, TRU	0.71 Ci	Huang 1984b

N.S. = not sampled.

TRU = transuranics.

Fig. 2.1.1 Location of the LLW collection and storage tanks at the Oak Ridge National Laboratory (from Horton 1984b).

ORNL-DWG 82-19187



persons concerning the site. The order of presentation follows the format as presented in the compendium of Remedial Action Program sites (Grimsby 1986).

3. GEOLOGY, HYDROLOGY, AND ECOLOGY

The radwaste facilities described in this document range from a belowground transfer line (Section 5.9) to buildings (Section 5.8), all located on disturbed land areas. Remedial actions to minimize any potential hazard from these facilities are dependent on site-specific environmental factors such as geology, hydrology, and ecology. While no site-specific information is available, Nix et al. (1986) have provided generic descriptions for the Oak Ridge Reservation.

3.1 GEOLOGY

A geological map (McMasters 1963) of the Oak Ridge Reservation identifies four major stratigraphic units: Rome Formation, Conasauga Group, Knox Group, and Chickamauga Limestone. The Rome Formation underlies Haw Ridge, which separates Bethel and Melton Valleys, and consists of evenly bedded, fine-grained sandstone and shale. This formation has a limited capacity for receiving, storing, and transporting water. In unweathered bedrock, water is largely limited to small openings along joints and bedding planes. Groundwater discharge during dry months is very low. The Conasauga Group, underlying Melton Valley, is composed of shales, limestones, and siltstones. Groundwater occurs principally in the weathered zone at openings along joints and bedding planes. These occur at shallow depths and consequently springs may be common during winter. During

the summer, very little water is discharged. The Knox Group underlies much of Chestnut and Copper Ridges and is the most widely distributed geologic unit in East Tennessee. It is composed primarily of dolomitic limestone with prominent chert zones. Underground solution channels characterize this group and establish it as the principal water storage formation. The Chickamauga limestone, underlying Bethel Valley, is composed predominantly of limestone, shales, siltstones, and bedded chert. Because of the presence of extensive amounts of limestone, this formation is susceptible to underground solution by migrating groundwater and the development of a network of channels and voids. These often permit easy access of surface waters and free movement of waters underground. However, it is not an important aquifer because the channels are small with limited water storage capacity.

3.2 HYDROLOGY

The Oak Ridge Reservation consists of a series of drainage basins that feed into the Clinch River. The principal drainage system from the ORNL site is White Oak Creek. White Oak Creek originates from springs of the Knox Dolomite on Chestnut Ridge and drains an area of 16.4 km^2 (4052 acres). The creek flows from the main ORNL site in Bethel Valley (Chickamauga Limestone) through a gap in Haw Ridge (Rome Formation) to Melton Valley (Conasauga Shale). The stream varies from 0.6 m (2 ft) to 1.2 m (3.9 ft) in width and from 9.9 cm (3.9 in.) to 24.9 cm (9.8 in.) deep. Flow rates range from a maximum of 18.2 m^3/s (643 ft^3/s) to a minimum of zero, the average being 0.27 m^3/s (9.5 ft^3/s). White Oak Creek is joined by Melton Branch after it passes Haw Ridge and enters White Oak Lake 0.5 km (0.8 mile) downstream.

Melton Branch collects flow from both Haw Ridge and Copper Ridge and drains an area of 3.8 km^2 (939 acres). It is the principal drainage system of the ORNL Melton Valley complex.

Of the four stratigraphic groups, only the Knox Dolomite has significant water storage capacity. The storage usually occurs in solution cavities and may frequently result in springs. Water storage capacity of the Rome Formation, Conasauga Shale, and Chickamauga Limestone is small and occurs along joints and bedding planes. Most wells in these formations have flows less than 38 L/min (10 gal./min.).

Groundwater flow in the weathered residual soil on the ORNL site basically follows the water table conditions. In the Chickamauga Limestone underlying Bethel Valley, groundwater moves through small solution channels. Studies of groundwater movement in the Conasauga Shale of Melton Valley have suggested that the primary direction of groundwater movement parallels the strike. Groundwater discharge is through evapotranspiration, springs, and streams, and it contributes to the base flow of streams that ultimately reach the Clinch River water supply.

Depth to the water table varies both spatially and temporally. Depth to the water table is generally greatest during October to December and least between January and March. In Bethel Valley, depth to the water table ranges from 0.3 (~1 ft) to 11 m (36 ft), whereas in Melton Valley the range is from 0.3 (~1 ft) to 20 m (65.6 ft).

3.3 ECOLOGY

The land area of the reservation is ~15,000 ha (37,065 acres), with plant communities characteristic of those found in the ridge and

valley region of Appalachia. The dominant oak-hickory association is typified by stands of shortleaf and Virginia pines, as well as the more extensive oaks and hickories. Approximately 75% of the reservation land area is forested. There are no threatened or endangered plant species on the reservation. Although several are proposed for status designation, none are present at any of the radwaste sites. Similarly, there are no endangered animal species on the radwaste sites, whereas the geographical ranges of several encompasses the reservation. While threatened or endangered fish species are known to exist in the headwaters of the Clinch River, none have been encountered in laboratory biological sampling programs. The reservation streams and ponds are not suitable habitat for sensitive species because they generally require a free-flowing habitat.

4. REGULATORY STATUS

As promulgated in the Department of Energy (DOE) Order 5480.14, it is the policy of DOE to identify and evaluate potential problems associated with inactive hazardous waste disposal sites at DOE facilities. The order implements this policy by providing instructions for a Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) program.

Through the examination of DOE and ORNL information sources (records, reports, and personal interviews), approximately 141 sites have been identified as potentially requiring remedial actions. These sites included solid waste storage areas, waste ponds and seepage pits, radioactive waste processing and transfer facilities, research

laboratories, environmental research areas, hazardous waste spill sites, reactors, and radioisotope development facilities. Of these sites, 81 were identified as CERCLA sites under the directive of DOE Order 5480.14. The remaining sites are controlled under the Resource Conservation and Recovery Act (RCRA), the Toxic Substances Control Act (TSCA), or the DOE Surplus Facilities Management Order (DOE Order 5820.2).

Hazard Ranking System analyses, e.g., as required by DOE Order 5480.14, for the radwaste facilities have not been performed, because they are regulated under provisions of the Resource Conservation and Recovery Act (RCRA) and the Department of Energy Order 5820.2 (Rohwer 1986), neither of which have provisions for hazard ranking.

5. RADWASTE FACILITIES

5.1 GUNITE WASTE STORAGE TANKS W-5 THROUGH W-10

Gunite Tanks W-5 through W-10 are located in the center of the ORNL complex, near the intersection of Central Avenue and Third Street (Fig. 2.1.1), in an area known as the South Tank Farm (3507 area). The tanks were constructed in 1943 of reinforced gunite. Gunite is a trade name for a method of spraying a mixture of sand, cement, and water against a form. The six tanks are underground in two adjacent rows of three tanks each on a 60 ft (18.29 m) center-to-center matrix. The inside diameter of the tanks is 50 ft (15.2 m), with a height of 12 ft (3.7 m). A schematic of the gunite tanks and associated valves and wells is illustrated in Fig. 5.1.1. Each tank is covered with a 6 ft (1.8 m) layer of compacted soil overburden.

Fig. 5.1.1 Gunite liquid waste storage tanks (1 ft = 0.31 m).

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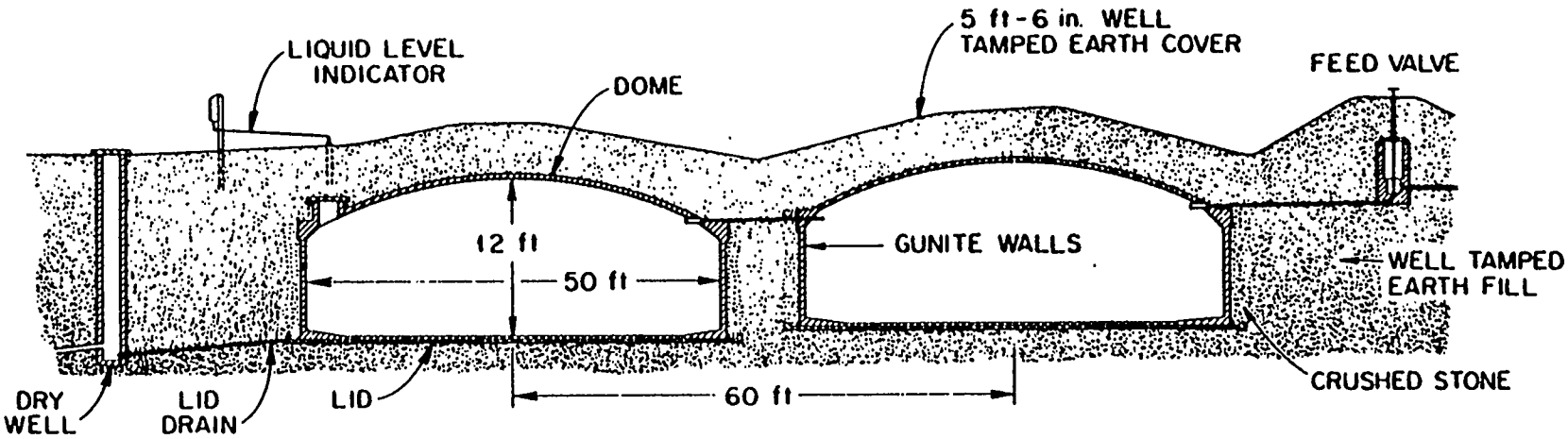

The tanks began to receive low-level liquid waste in 1943 and continued until 1978. Waste was derived from numerous ORNL programs (Graphite Reactor and other pilot plants). Typically, the waste acidity was neutralized with NaOH or $CaCO_3$ until the waste was basic before transfer to one of the gunite tanks. From the gunite tank the waste was transferred to an evaporator and concentrated before returning to another gunite tank. Several chemicals precipitated during storage in the tanks and resulted in a sludge layer. Precipitates included strontium hydroxide, barium hydroxide, and other hydroxides, carbonates, and sulfates.

The tanks contained highly radioactive materials prior to being emptied in the early 1980's. Details of radiochemical and stable chemical analyses of the sludge and a program for the transfer of the residue from the tanks was described by Weeren and Mackey (1980). After the sludge and water was removed, a radiological characterization was made (Huang et al. 1984a). Some residual materials were left following the sludge removal program. The six tanks presently contain various quantities of waste. Table 5.1.1 provides an early summary of estimated residual waste volume and radioactivity. Since the characterization study was completed, there has been infiltration of rainwater into the gunite tanks (Scott 1986). The current volumes for Tanks W-5 through W-10 (Table 2.1.1) include that leakage. A radiological survey was made after tank contents were removed (Huang et al. 1984a). The direct radiation dose rates in each tank were measured at three distances (0.6, 2.4, and 4 m) above the bottom of the tanks. The results are summarized in Table 5.1.2. The direct beta-gamma dose

Table 5.1.1 Estimated inventory of residual waste in the gunite tanks (from Huang 1984a)

Tank[a]	Type of waste[b] Slurry (ℓ)	Solids (ℓ)[d]	Total volume (ℓ)	Major radionuclides	Total inventory (TBq)	(kCi)
W-5	14,000	9,700	24,000	^{90}Sr	11	0.3
				^{137}Cs	0.60	0.02
				others	0.43	0.01
W-6	39,000	19,000	58,000	^{90}Sr	76	2.0
				^{137}Cs	5.5	0.15
				others	1.6	0.04
W-7	39,000	7,800	47,000	^{90}Sr	62	1.7
				^{137}Cs	4.5	0.12
				others	1.3	0.03
W-8	39,000	3,900	43,000	^{90}Sr	290	7.8
				^{137}Cs	35	0.96
				others	5.1	0.14
W-9	54,000	0	54,000	^{90}Sr	15	0.41
				^{137}Cs	9.1	0.25
				others	2.6	0.07
W-10	120,000	0	120,000	^{90}Sr	260	7.0
				^{137}Cs	36	0.96
				others	12	0.33
Total	305,000	40,000	345,000		820	22

[a]The capacity of each tank is 6.6×10^5 ℓ.
[b]Estimated volume based on photographic information.
[c]Based on radioactivity/volume ratios multiplied by the volume shown in column 4 of this table.
[d]High density sludge.

Table 5.1.2 Radiological survey results in the waste tanks in the South Tank Farm (from Huang et al. 1984a)

Tank	TLD absorbed dose rate (rad/h)			Direct beta-gamma absorbed dose rate[a] (rad/h)
	B[b]	M[b]	S[b]	
W-5	0.74	0.76	0.94	0.015
W-6	2.5	1.2	0.03	0.020
W-7	1.8	1.6	0.83	0.300
W-8	17.0	2.5	0.62	0.150
W-9	6.5	5.6	0.22	0.100
W-10	20.0	4.4	0.43	0.250

[a]Radiation measured at the opening of the tank access with lid open.

[b]B, M, and S indicate different depth where TLD chips were placed in the tank. B indicates 0.6 m above the bottom of Tank W-5 and 0.9 m above the bottom of other tanks. M and S indicate 1.8 m and 3.4 m above B, respectively.

rates at the tank access openings ranged from 0.015 rad/hr (Tank W-5) to 0.3 rad/hr (Tank W-7). The tanks with the highest radiation absorbed dose rates near the surface of the waste were W-10, W-8, and W-9, with rates of 20, 17, and 6.5 rad/hr, respectively. These high rates are believed to be associated with the residual waste left in each tank and embedded in the tank walls. The direct beta-gamma radiation levels in the South Tank Farm are generally low. Background was on the order of 0.5 mrad/hr. Elevated rates (3.2-150 mrad/hr) were detected with lids open (Huang et al. 1984a). Most of the radioactivity is Cs-137 and Sr-90. Tanks W-8 and W-10 contain most of the radioactive waste with 8.9 and 8.3 kCi, respectively. Tank W-5 contains the least with 0.33 kCi.

Two references contain significant information concerning the gunite tanks. They are:

Horton, J. R. 1984a. Preliminary Decommissioning Study Reports, Volume 4: Gunite Storage Tanks. X-OE-231, Vol. 4.

Huang, S. F., W. A. Alexander, J. B. Watson, and T. W. Oakes. 1984a. Preliminary Radiological Survey of the Gunite Waste Tanks in the South Tank Farm at Oak Ridge National Laboratory. ORNL/CF-84/206.

Persons knowledgeable of the gunite tanks include:

Breeding, Swan, Operations Division, ORNL, 615/574-7058
Lasher, Larry, Operations Division, ORNL, 615/576-5223

5.2 WASTE TANK WC-1

Tank WC-1 is a 2,000 gal. (7600 L) capacity stainless-steel tank. The tank is subsurface east of Building 3038 and west of Building 3037 near the center of the ORNL complex (Fig. 2.1.1). The Fifth Avenue

Creek is approximately 46 m east of the tank. The tank is 8.66 ft (2.64 m) in diameter and 5.75 ft (1.75 m) tall. A typical stainless-steel tank is illustrated in Fig. 5.2.1. Over a period of years (1950-1968), the tank was used for the collection and monitoring of process liquid waste from isotope production and development laboratories (Buildings 3038, 3028, 3029, 3030, 3031, 3032, 3033, 3047). Additional waste was from the Building 3110 filter, 3039 stack, and the 3092 scrubber. In 1968 the tank was abandoned because of leaking discharge lines. It presently contains curie amounts of Cs-137, Sr-90, and Co-60. Information of direct or transferrable radioactivity is not available.

A walk-over survey for direct beta-gamma absorbed dose indicated a maximum rate of 0.23 mrad/hr at 1-3 cm above the surface, and a distance of 3 m from the tank (Huang et al. 1984b). Soil samples were collected from 0 to 0.9 m deep and analyzed for radioactivity. The highest Cs-137 and Co-60 activities were 44 Bq/g (1188 pCi/g) and 21 Bq/g (567 pCi/g), respectively. All samples were not analyzed for Sr-90, but the highest contamination identified 200 Bq/g (5.4 nCi/g). Two groundwater samples indicated Sr-90 concentrations of 7.7 Bq/mL (207 pCi/mL) and 3.6 Bq/mL (97 pCi/mL). A summary of the soil and water analyses is presented in Table 5.2.1. No radiological characterization or estimate of residual materials have been made. Results of uranium analyses on samples from Tank WC-1 and analyses for transuranics on soil and water samples near the tank are presented in Appendices A and B, respectively.

Fig. 5.2.1 A typical stainless-steel tank.

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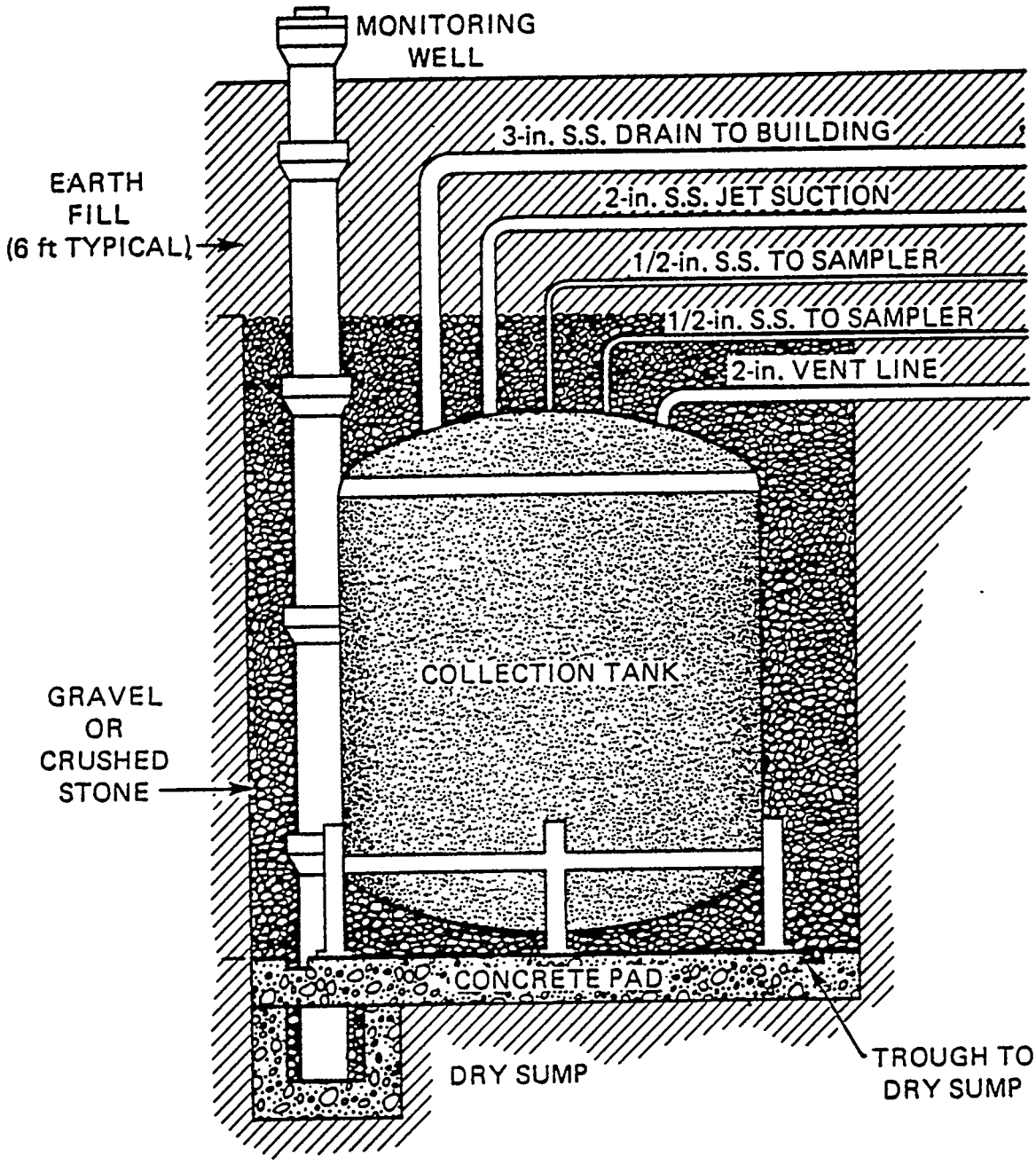

Table 5.2.1 Results of the preliminary gamma screening[a] and the quantitative beta and gamma analyses on soil and water samples from Tank Site WC-1 (Source: Huang et al. 1984b)

Sample identification	Gamma activity (cps/kg)	^{137}Cs	^{60}Co	^{90}Sr
		(Bq/g)[b]		
3B-1	NA[c]	1.2	0.03	NA
3B-2	NA	12.0	0.35	NA
3B-3	NA	7.0	0.37	NA
1D-1	NA	3.2	0.16	NA
1D-3	NA	2.9	0.11	NA
1D-3	NA	2.0	0.48	NA
1-1	230	1.2	0.02	ND[c]
1-2	220	1.6	0.04	0.14
1-3	NS[c]	NS	NS	NS
1-4	34	0.04	0.02	NA
1-5	1,100	1.1	0.63	200
1-6	1,300	0.27	1.9	140
1-7	2,000	0.48	4.8	NA
1-8	2,200	0.37	8.5	NA
1-9	2,800	0.41	10.0	110
1-10	3,600	0.32	16.0	NA
1-10[d]	NA	ND	<0.01	7.7
2-1	130	0.63	0.04	2.2
2-2	62	0.13	0.02	NA
2-3	NS	NS	NS	NS
2-4	27	0.04	ND	NA
2-5	130	0.63	0.08	NA
2-6	42	0.08	ND	NA
2-7	730	0.20	0.67	150
3-1	1,000	6.4	0.24	0.83
3-2	230	1.6	0.02	0.3
3-3	NS	NS	NS	NS
3-4	2,800	0.44	9.6	79
3-5[e]	6,100	44	0.93	1.4
3-6	670	2.3	0.85	NA
3-7	3,200	19	5.2	40
3-8	6,900	4.4	21	190
3-9	5,000	7.0	15	NA
3-10	2,800	2.2	14	37
3-11	2,400	1.0	9.3	12
3-11[d]	NA	ND	<0.01	3.6

Table 5.2.1 (CONT'D)

Sample identification	Gamma activity (cps/kg)	^{137}Cs	^{60}Co	^{90}Sr
		(Bq/g)[b]		
4-1	100	0.41	0.03	0.37
4-2	46	0.1	<0.01	NA
4-3	NS	NS	NS	NS
4-4	46	0.19	0.16	NA
4-5	380	0.05	1.7	13
4-6	340	0.02	1.9	NA
4-7	290	0.03	1.4	6.6

[a]Total gamma activity was between 100 and 1,500 keV measured in counts per second (cps) per kilogram of moist soil.
[b]1 Bq = 27 pCi.
[c]NA indicates not analyzed or no available data, NS indicates no sample, and ND indicates not detectable.
[d]Indicates a water sample. The radionuclide concentrations are measured in Bq/mL.
[e]This sample also contains 2.9 Bq/g of ^{134}Cs, 0.19 Bq/g of ^{152}Eu, and 0.26 Bq of ^{154}Eu.

Tank WC-1 is described in a decommissioning study report by Horton (1984b). Knowledgeable persons concerning this tank are:

Breeding, Swan, Operations Division, ORNL, 615/574-7058
Lasher, Larry, Operations Division, ORNL, 615/576-5223

5.3 WASTE TANKS WC-15 AND WC-17

Waste Tanks WC-15 and WC-17 are located south of Building 3587 and west of Building 4508 near the intersection of Fifth Street and White Oak Avenue (Fig. 2.1.1). Fifth Creek is approximately 20 m east of the tank site. Both tanks are underground with approximately 10 ft (3 m) of soil and gravel cover. The tanks are of stainless-steel construction and measure 6.5 ft (2 m) tall by 5 ft (1.5 m) in diameter. Each tank has a capacity of ~1,000 gal. (3900 L). These tanks were put into service in 1951 to receive waste from research laboratories in Building 4500. Because of leaks, the tanks were taken out of service in the 1960's. In a radiological characterization study (Huang et al., 1984b) it was noted that Tank WC-17 contained 3900 L of oil and water. Radionuclides consist of Cs-137, Sr-90, and a variety of transuranics. It is estimated that the total radioactivity is approximately 5 mCi. The inventory of the contents in Tank WC-17 is summarized in Table 5.3.1. Tank WC-15 is not accessible from ground level and was not sampled extensively for radionuclide, sludge, water content, or direct absorbed dose.

Low levels of direct beta-gamma absorbed dose rates were measured at various depths inside Tank WC-17. The maximum was 0.15 mrad/hr at the surface of the tank contents. No direct or transferrable alpha activity above background was detected at the tank access port. Direct

Table 5.3.1 Radiological inventory of the contents of Waste Tank WC-17 (Source: Huang et al. 1984b)

Tank (capacity)	Tank contents			Major radionuclides	Concentration[a] (Bq/mL or Bq/g)	Inventory	
	Type	Depth (m)	Volume (L)			(GBq)[b]	(mCi)
WC-17 (3.9×10^3 L)	Oil	0.2	300	^{137}Cs	2.8	0.0008	0.022
				^{90}Sr	24	0.008	0.22
				^{60}Co	2.8	0.0008	0.022
				^{154}Eu	1.5	0.0004	0.011
				^{238}Pu	40	0.012	0.32
				^{239}Pu	20	0.006	0.16
				^{241}Am	37	0.012	0.32
				^{244}Cm	53	0.016	0.44
	Water	2.1	3,600	^{137}Cs	4.6	0.02	0.5
				^{90}Sr	18	0.07	2
				^{238}Pu	1.7	0.006	0.2
				^{239}Pu	1.0	0.004	0.1
				^{241}Am	<1.0	0.004	0.1
				^{244}Cm	<1.0	<0.004	0.1

[a]Results for sludge samples are expressed in Becquerels per gram of wet sample. Water sample results are in Becquerels per milliliter.
[b]1 Bq = 27 pCi.

beta-gamma dose at 1-3 cm above the surface was 90 mrad/hr inside a containment tent over the pump pits. Measurements over the tank site were generally <1 mrad/hr. Results of soil analyses indicated maximum subsurface soil concentrations of 12 Bq/g (324 pCi/g) and 11 Bq/g (297 pCi/g) for Cs-137 and Sr-90, respectively (Table 5.3.2). No groundwater was encountered at this tank site, but there is a potential for subsurface contamination to migrate into nearby (20 m) Fifth Creek. Results of uranium analysis on samples from Tanks WC-15 and WC-17 and transuranics in soils near the tanks are summarized in Appendices A and B, respectively.

Two reports provide details of Tanks WC-15 and WC-17 (Horton 1984b and Huang et al. 1984b). Personnel of the Operations Division should be consulted for specific information. Knowledgeable persons concerning Tanks WC-15 and WC-17 are:

Breeding, Swan, Operations Division, ORNL, 615/574-7058
Lasher, Larry, Operations Division, ORNL, 615/576-5223

5.4 WASTE TANKS W-1 THROUGH W-4 AND W-13 THROUGH W-15

Tanks W-1 and W-2 are located in the North Tank Farm (3023 area) near the intersection of Third Street and Central Avenue (Fig. 2.1.1). These tanks were constructed in 1943 to receive low-level waste primarily from Building 3019 and operated until 1960 when they were removed from service because of leaks. The tanks are 12 ft (3.66 m) in diameter and 8 ft (2.44 m) tall with a capacity of 4800 gal. (~18,000 L). Tanks are dome-shaped with a compacted soil cover of 6 ft (1.82 m). The construction material was gunite, and the details

Table 5.3.2 Results of the preliminary gamma screening[a] and the quantitative gamma and beta analyses[b] on soil samples from Tank Site WC-15, 17 (Source: Huang et al. 1984b)

Sample identification	Gamma activity (cps/kg)	^{137}Cs (Bq/g)[c]	^{90}Sr (Bq/g)	Others (trace amount)
1-1	NS[d]	NS	NS	
1-2	NS	NS	NS	
1-3	43	0.07	NA[d]	
1-4	39	0.06	ND[d]	
1-5	45	0.15	NA	
1-6	34	0.09	NA	
1-7	49	0.11	NA	
1-8	36	0.07	1.4	
2-1	NS	NS	NS	
2-2	NS	NS	NS	
2-3	57	0.12	0.07	
2-4	54	NA	NA	
2-5	34	NA	NA	
2-6	25	0.06	NA	
3-1	48	NA	NA	
3-2	37	0.03	NA	
4-1	NS	NS	NS	
4-2	NS	NS	NS	
4-3	810	0.02	1.8	^{60}Co, ^{154}Eu, ^{241}Am, ^{244}Cm
4-4	34	0.03	3.8	^{154}Eu
4-5	320	0.03	NA	^{154}Eu
4-6	44	0.02	NA	^{154}Eu
4-7	1,200	0.04	4.0	^{60}Co, ^{154}Eu, ^{241}Am, ^{244}Cm
4-8	1,500	12	0.94	^{60}Co, ^{154}Eu
4-9	ND	9.3	1.0	^{60}Co, ^{154}Eu
5-1	27	0.10	NA	
5-2	23	NA	NA	
5-3	30	0.02	0.04	
5-4	32	NA	NA	
5-5	27	NA	NA	
5-6	29	0.07	NA	
5-7	670	0.03	0.06	
6-1	40	0.02	NA	
6-2	42	NA	NA	
6-3	690	7.4	0.05	^{134}Cs
6-4	210	1.6	ND	^{154}Eu, ^{134}Cs
6-5	48	0.20	NA	
6-6	31	0.10	NA	
6-7	35	0.14	11	^{239}Pu, ^{244}Cm

[a]The gamma activity was between 10 and 1500 keV, measured in counts per second per kilogram of moist soil.
[b]^{137}Cs and ^{90}Sr are reported on a dry weight basis.
[c]1 Bq = 27 pCi.
[d]NS indicates no sample, NA indicates not analyzed, and ND indicates not detectable.

of tank installation are depicted in Fig. 5.4.1. Horton (1984b) provided an excellent report with details of construction, tank history, and contamination.

Tank waste was primarily from Building 3019, a radiochemical processing facility. Tank W-1 presently contains ~3.9×10^3 L of liquid, primarily rainwater and some solid deposits. Tank W-1 received both metal waste with highly radioactive source and fissionable materials. Tank W-2 was designed to receive overflow from Tank W-1 and therefore received the same kinds of materials. It is estimated that Tank W-2 contains 5,000 L of liquid/sludge residue. Major contaminants in Tank W-1 include Cs-137 and Sr-90, whereas the principal radionuclide in Tank W-2 is thought to be Pu-238. Huang et al. (1984b) indicated that additional samples would be collected and analyzed. Total inventories of individual radionuclides identified indicate that the total radioactivity in Tank W-1 is about 142 mCi and 230 Ci in Tank W-2 (Horton 1984b). A summary of the tank contents is presented in Table 5.4.1. Major radionuclides found in the soil at the North Tank Farm (3023 area) are Cs-137, Sr-90, Co-60, and Th-232. Results of the radiological analyses are contained in Appendix C.

Tanks W-3 and W-4 are also located in the North Tank Farm. These tanks are somewhat larger than the W-1 and W-2 tanks, with a capacity of 42,500 gal. (1.6×10^5 L). The tanks have a diameter of 25 ft (7.6 m) and a sidewall height of 12 ft (3.66 m). Both tanks are of gunite construction and are covered with a 6 ft (1.8 m) layer of compacted soil. Like Tanks W-1 and W-2, these tanks were operated for 35 years. During the period of service, the tanks received metal waste

Fig. 5.4.1 Typical concrete waste tank installation (from Horton 1984b).

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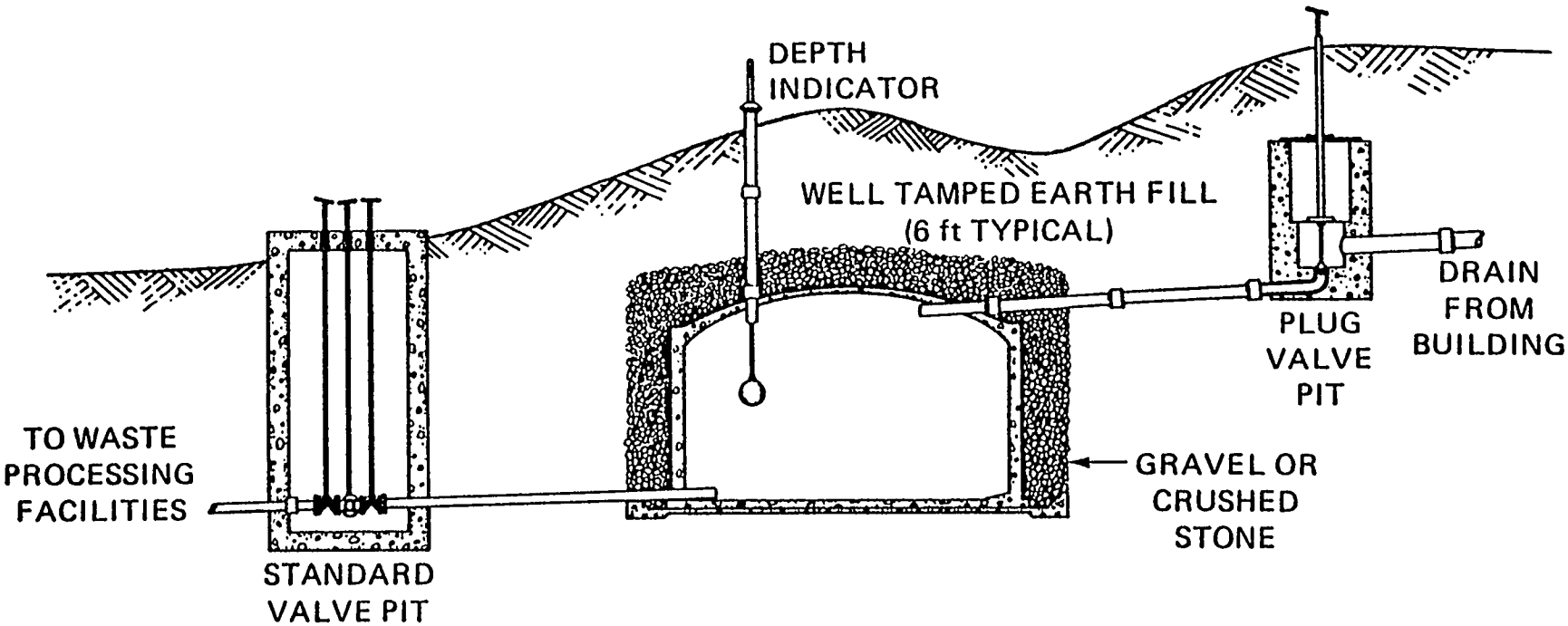


Table 5.4.1 Radiological inventories of the contents of Tanks W-1 through W-4 (Source: Huang et al. 1984b)

Tank (capacity)	Tank contents: Type	Depth (m)	Volume (L)	Major radionuclides	Concentration[a] (Bq/mL or Bq/g)	Inventory (GBq)[b]	Inventory (mCi)
W-1 (1.9×10^4 L)	Water	0.6	3,900	^{137}Cs	1.7×10^2	0.7	20
				^{90}Sr	8.4×10^2	3.3	90
				^{152}Eu	2.6×10^2	1	30
				^{154}Eu	5.2	0.02	0.5
				^{238}Pu	1.8	0.007	0.2
				^{239}Pu	2.9	0.01	0.3
				^{241}Am	1.7	0.007	0.2
				^{244}Cm	13	0.05	1
				^{238}U	<0.1	0.0004	0.1
W-2 (1.9×10^4 L)	Water	0.45	3,000	^{137}Cs	13	0.04	1
				^{90}Sr	5.1×10^2	2	4
				^{238}Pu	1.9	0.006	0.2
				^{239}Pu	5.1	0.02	0.4
				^{241}Am	2.7	0.008	0.2
				^{244}Cm	4.1	0.01	0.3
				^{233}U	0.9	0.003	0.07
				^{238}U	0.2	0.0006	0.02
	Sludge	0.3	2,000	^{137}Cs	1.7×10^5	400	10,000
				^{90}Sr	2.1×10^5	500	10,000
				^{60}Co	5.7×10^2	1	40
				^{238}Pu	2.8×10^{6}[c]	7,000	200,000
				^{239}Pu	3.3×10^4	80	2,000
				^{241}Am	5.0×10^4	100	3,000
				^{244}Cm	1.3×10^3	3	80
				^{152}Eu	7.3×10^4	200	5,000
				^{154}Eu	1.2×10^4	30	800
				^{155}Eu	4.8×10^2	1	30
				^{233}U	1.2×10^2	0.3	8
				^{238}U	4.6	0.01	0.3

Table 5.4.1 (CONT'D)

Tank (capacity)	Tank contents Type	Depth (m)	Volume (L)	Major radionuclides	Concentration[a] (Bq/mL or Bq/g)	Inventory (GBq)[b]	(mCi)
W-3 (1.6×10^5 L)	Water	2.3	84,000	^{137}Cs	4.3×10^2	40	1,000
				^{90}Sr	14	1	30
				^{238}Pu	2.9	0.2	7
				^{239}Pu	1.4	0.1	3
				^{241}Am	1.7	0.1	4
				^{244}Cm	4.7	0.4	10
				^{233}U	4.2	0.4	10
				^{238}U	1.0	0.08	2
	Sludge	0.45	16,000	NS[d]	NS	NS	NS
W-4 (1.6×10^5 L)	Water	1.2	44,000	^{137}Cs	7.5×10^2	33	900
				^{90}Sr	3.0×10^2	13	400
				^{238}Pu	1.8	0.08	2.1
				^{239}Pu	0.8	0.04	1
				^{241}Am	2.7	0.1	3
				^{244}Cm	2.5	0.1	3
				^{233}U	12	0.5	14
				^{235}U	0.8	0.04	1
				^{238}U	13	0.6	20
	Sludge	0.6	22,000	^{137}Cs	1.4×10^5	4,000	100,000
				^{60}Co	28	0.7	20
				^{90}Sr	2.0×10^5	5,000	100,000
				^{238}Pu	80	2	60
				^{239}Pu	2.4×10^3	60	2,000
				^{241}Am	1.3×10^2	3	90
				^{244}Cm	2.4×10^2	6	200
				^{233}U	2.2×10^3	60	2,000
				^{235}U	1.1×10^2	3	80
				^{238}U	2.2×10^3	60	2,000

[a]Results for sludge samples are expressed in Becquerels per gram of wet sample. Water sample results are in Becquerels per milliliter.
[b]1 Bq = 27 pCi.
[c]Data questionable. Another sample will be taken and analyzed.
[d]No sample can be collected.

from Building 3019. Tank W-3 collected plutonium waste, whereas Tank W-4 collected uranium waste. Tank W-3 presently contains $1x10^5$ L of liquid and sludge, while Tank W-4 contains $6.6x10^4$ L of similar materials. A summary of the tank inventory is presented in Table 5.4.1.

The radionuclide inventory on Tank W-3 is estimated at 1 Ci, while the total inventory in Tank W-4 is approximately 207 Ci (Table 2.1.1). Major radionuclides are Cs-137 and Sr-90. A summary of the tank contents is presented in Table 5.4.1. In addition to the tanks, the soil around the tanks contain some Cs-137 and Sr-90. Results of soil analyses are given in Appendix C.

Tanks W-13, W-14, and W-15 are stainless-steel tanks located in the North Tank Farm (area 3023). These tanks are buried in concrete to a depth of 7.5 ft (2.3 m). Tanks W-13 and W-14 are horizontal tanks, measuring 11 ft (3.35 m) long and 6 ft (1.83 m) in diameter. Each has a capacity of 2,000 gal. (7,570 L). Tank W-15 is a vertical cylinder tank 6 ft (1.83 m) tall with a diameter of 8 ft (2.44 m). It also has a capacity of 2,000 gal. Tank W-13 collected waste from the Chemistry Division Hot Laboratory Group, whereas Tanks W-14 and W-15 served the Operations Division. All were removed from service in 1958. The total radionuclide inventory in Tank W-13 was estimated to be about 330 Ci, with Tank W-14 containing approximately 14 Ci (Table 2.1.1). No estimate of radioactive content was available for Tank W-15. Major nuclides in Tanks W-13 and W-14 are Cs-137 and Sr-90. A radiological inventory of the tank contents is presented in Table 5.4.2. The detailed inventory of these tanks is presented in a decommissioning study report (Horton 1984b).

Table 5.4.2 Radiological inventories of the contents of Tanks W-13 and W-14 (Source: Huang et al. 1984b)

Tank (capacity)	Tank contents: Type	Tank contents: Depth (m)	Tank contents: Volume (L)	Major radionuclides	Concentration[a] (Bq/mL or Bq/g)	Inventory (GBq)[b]	Inventory (mCi)
W-13 (7.8×10^3 L)	Water	0.4	1,700	^{137}Cs	7.1×10^5	1,000	30,000
				^{60}Co	2.6×10^2	0.4	10
				^{90}Sr	7.4×10^6	10,000	300,000
				^{154}Eu	2×10^3	4	100
				^{238}Pu	5.6×10^2	1	30
				^{239}Pu	58	0.1	3
				^{241}Am	3.0×10^2	0.5	10
				^{244}Cm	70	0.1	3
				^{233}U	4.6	0.008	0.2
				^{235}U	0.4	0.0007	0.02
				^{238}U	2.1	0.004	0.1
W-14 (7.8×10^3 L)	Water	0.2	470	^{137}Cs	4.9×10^5	200	6,000
				^{90}Sr	5.9×10^5	300	8,000
				^{60}Co	2.2×10^3	1	30
				^{238}Pu	12	0.006	0.2
				^{239}Pu	20	0.009	0.3
				^{241}Am	10	0.005	0.1
				^{244}Cm	62	0.03	0.8
				^{233}U	17	0.008	0.2
				^{238}U	2.1	0.001	0.03

[a]Results for sludge samples are expressed in Becquerels per gram of wet sample. Water sample results are in Becquerels per milliliter.
[b]1 Bq = 27 pCi.

A radiological characterization of the North Tank Farm (area 3023) found that all tanks were heavily contaminated (Huang 1984b). The highest direct dose rates inside the tanks ranged from 250 to 6500 mrad/hr. During a walk-over survey the direct beta-gamma dose rates were determined at 1-3 cm above the surface. The maximum dose rate was 50 mrad/hr at an aboveground pipe, approximately 18 m west of Tanks W-13, W-14, and W-15, but generally, the dose rate was less than 1 mrad/hr throughout the tank farm area. Soil samples were collected from 0 to 0.9 m deep and analyzed for radioactivity. The maximum Cs-137 and Sr-90 activities were 73 and 67.5 nCi/g, respectively. Additional radionuclides (Co-60, Th-232, Eu-154, Cs-134) were present in trace amounts.

Knowledgeable persons of these tanks and their operations include:

Breeding, Swan, Operations Division, ORNL, 615/574-7058
Lasher, Larry, Operations Division, ORNL, 615/574-7058

5.5 WASTE TANK W-11

Tank W-11 is located on the southeast corner of the South Tank Farm (3507 area) near Buildings 3517 and 3515 (Fig. 2.1.1). This tank is a gunite tank covered with 6 ft (1.8 m) of soil overburden and a capacity of 1500 gal. (5677 L). It measures 8 ft (2.4 m) in diameter and 5.5 ft (1.7 m) tall. The W-11 tank was placed in service in 1943 to serve as a waste collection and monitoring tank for research laboratories in the Building 3550. It operated for about 40 years, before being removed from service because of leaks. Horton (1984b) estimates that the tank presently contains ~1200 L of liquid/sludge residue. In a radiological characterization study, Cs-137, Sr-90, and

Pu-238 were identified as the major radionuclides present (Table 5.5.1). The total estimated radioactivity present is less than 1 mCi (Table 2.1.1).

Due to the contributions from other radiation sources at the tank site, the absorbed dose rate measured at the tank opening (1 mrad/hr) was slightly higher than the rates measured inside the tank (Huang 1984b). The interior surface at the tank access was contaminated with transferrable beta-gamma and alpha activities of 700 and 300 dpm/100 cm^2, respectively. In a walk-over survey, elevated direct beta-gamma absorbed dose rates were generally below 1 mrad/hr, except in the pump pit area where a maximum rate of 28 mrad/hr was detected. Surface soil samples (0 to 0.6 m deep) revealed that the entire tank site is contaminated. The maximum Cs-137 and Sr-90 activities were 120 Bq/g (3.24 nCi/g) and 54 Bq/g (1.46 nCi/g), respectively. Results of the soil analyses are presented in Table 5.5.2.

The following individuals are knowledgeable concerning the tank and its operation:

Breeding, Swan, Operations Division, ORNL, 615/574-7058
Lasher, Larry, Operations Division, ORNL, 615/576-5223

5.6 WASTE TANKS TH-1, TH-2, AND TH-3

Several tanks are located near Building 3503 (Tanks TH-1, TH-2, and TH-3) south of White Oak Avenue (Fig. 2.1.1). These tanks received waste from the thorium pilot plant. All the tanks are of stainless-steel construction and covered with a 5 ft (1.5 m) layer of soil. Volumes of Tanks TH-1, TH-2, and TH-3 are 2500 gal. (9462 L),

Table 5.5.1 Radiological inventory of the contents of Waste Tank WC-11 (Source: Huang et al. 1984b)

Tank (capacity)	Tank contents			Major radionuclides	Concentration[a] (Bq/mL or Bq/g)	Inventory	
	Type	Depth (m)	Volume (L)			(GBq)[b]	(mCi)
W-11 (5.8×10^3 L)	Water	0.3	1,000	^{137}Cs	1.1	0.001	0.03
				^{90}Sr	2.8	0.003	0.08
				^{238}Pu	2.3	0.002	0.05
				^{239}Pu	1.1	0.001	0.03
				^{241}Am	<1.0	<0.001	<0.03
				^{244}Cm	<1.0	<0.001	<0.03
	Sludge	0.05	170	NS[c]	NS	NS	NS

[a]Results for sludge samples are expressed in Becquerels per gram of wet sample. Water sample results are in Becquerels per milliliter.
[b]1 Bq = 27 pCi.
[c]No sample can be collected.

Table 5.5.2 Results of the preliminary gamma screening[a] and the quantitative beta and gamma analyses[b] on soil and water samples from Tank Site W-11 (Source: Huang et al. 1984b)

Sample identification	Gamma activity (cps/kg)	^{137}Cs (Bq/g)[c]	^{90}Sr (Bq/g)	Others (trace amount)
1-1	700	7.0	7.5	
1-2	300	2.6	12	
1-3	NS[c]	NS	ND[c]	
1-4	210	1.5	2.2	
1-5	100	1.3	NA	
1-6	62	0.71	NA	
1-7	250	2.4	3.2	
1-8	63	NA[d]	NA	
2-1	1,400	11	4.1	
2-2	460	4.4	2.6	
2-3	150	0.78	NA	^{60}Co
2-4	78	0.34	NA	^{60}Co
2-5	58	0.31	28	
2-6	350	4.8	NA	
2-7	130	NA	NA	
2-8	8,100	120	42	^{60}Co, ^{154}Eu
3-1	1,400	9.6	29	
3-2	550	3.0	34	
3-3	1,000	5.9	32	^{60}Co, ^{154}Eu
3-4	310	26	14	^{60}Co, ^{154}Eu
3-5	37	0.07	5.6	
3-6	63	0.02	NA	
3-7	49	0.03	0.2	
3-8	1,800	3.3	14	^{60}Co, ^{154}Eu
4-1	2,500	16	38	
4-2	410	1.5	54	
4-3	120	0.52	24	
5-1	180	1.1	5.8	
5-2	170	1.5	1.7	
5-3	57	0.1	NA	
5-4	31	0.03	NA	
5-5	38	0.03	NA	
5-6	37	0.01	NA	
5-7	65	0.07	NA	
5-8	83	0.09	0.9	^{60}Co

[a]Gamma activity measured in counts per second (cps) per kilogram of moist soil in the energy range 100-1,500 keV.
[b]^{137}Cs and ^{90}Sr are reported on a dry weight basis.
[c]1 Bq = 27 pCi.
[d]NS indicates no sample was taken, NA indicates not analyzed, and ND indicates not detectable.

2400 gal. (9084 L), and 3300 gal. (12,490 L), respectively. Tank TH-1 was installed in 1948, whereas Tanks TH-2 and TH-3 were installed in 1952. Tank TH-1 is 9.5 ft (2.9 m) long and 7 ft (2.1 m) in diameter. Tank TH-2 is 7 ft (2.1 m) in diameter and stands 10 ft (3 m) tall. Similarly, Tank TH-3 is 10 ft (3 m) tall, but has a diameter of 9.17 ft (2.8 m). All of the thorium waste tanks have a common pump pit.

Horton (1984b) discusses the thorium waste tanks in detail. Because of high radiation levels, Tank TH-2 contents were not sampled for radioactivity. In a radiological survey (Huang et al. 1984b) the direct reading at the top of Tank TH-2 was 150 mrad/h, and the transferrable counts were 50 dpm alpha and 3000 dpm beta-gamma per 100 cm^2. Dose rate increased when the shielding was removed; the direct dose rate was 6 rad/hr, and transferrable contamination was 100 dpm alpha and 22,000 dpm beta-gamma per 100 cm^2, respectively. Tank TH-1 is estimated to contain 1800 L of liquid. Radionuclides identified in TH-1 include Cs-137, Co-60, Sr-90, and small traces of transuranics (Table 5.6.1). Total radioactivity is estimated to be approximately 1 Ci. Tank TH-3 contains about 2600 L of liquid residue. The residue is composed of Cs-137, Co-60, Sr-90, and some transuranic waste. The estimated total radioactivity is about 1.2 Ci (Table 5.6.1). Results of uranium analysis on samples from the thorium tanks and analysis of transuranics in soils and water near the tanks are presented in Appendices A and B, respectively.

In a radiological characterization study (Huang 1984b) Tanks TH-1 and TH-3 were opened. The highest direct absorbed dose rates measured inside TH-1 and TH-3 were 53 and 9 mrad/hr, respectively.

Table 5.6.1 Radiological inventories of the contents of Waste Tanks TH-1 and TH-3 (Source: Huang et al. 1984b)

Tank (capacity)	Tank contents: Type	Depth (m)	Volume (L)	Major radionuclides	Concentration[a] (Bq/mL or Bq/g)	Inventory (GBq)[b]	Inventory (mCi)
TH-1 (9.7×10^3 L)	Water	0.45	1,800	^{137}Cs	9.5×10^3	20	500
				^{90}Sr	1.2×10^4	22	600
				^{60}Co	2.7×10^2	0.5	10
				^{238}Pu	1.9	0.003	0.09
				^{239}Pu	1.2	0.002	0.06
				^{241}Am	<1.0	0.002	0.05
				^{232}Th	0.07	0.0001	0.003
TH-3 (1.3×10^4 L)	Water	0.6	2,600	^{137}Cs	8.3×10^3	22	600
				^{60}Co	6.5	0.02	0.5
				^{90}Sr	7.9×10^3	20	600
				^{238}Pu	1.1	0.003	0.08
				^{239}Pu	1.3	0.003	0.09
				^{241}Am	<1.0	<0.003	<0.07
				^{244}Cm	<1.0	<0.003	<0.07
				^{232}Th	0.02	0.00005	0.001

[a]Results for sludge samples are expressed in Becquerels per gram of wet sample. Water sample results are in Becquerels per milliliter.
[b]1 Bq = 27 pCi.

Transferrable activities at the tank access were below background. Tank TH-2 was not studied because of difficult access and heavy contamination. The direct beta-gamma readings outside and inside the lead shielding was 3.2 and 150 mrad/hr, respectively. The absorbed dose rate above the ground surface between the tank and the lead shielding was 6 rad/hr. Surface soil was heavily contaminated between the shielding and the tank; for example, 5 g of moist soil gave direct beta-gamma and gamma dose rates of 30 and 10 mrad/hr, respectively.

From a walk-over survey, absorbed dose rates up to 3.2 mrad/hr were measured in the tank site (TH-1, TH-3, and TH-3). Subsurface soil samples indicated that the major radionuclides were Cs-137 and Sr-90. The highest concentrations of Cs-137 and Sr-90 were 7.6 nCi/g (280 Bq/g) and 4.6 nCi/g (170 Bq/g), respectively.

Groundwater was encountered at a core site (4.9 m below-grade) and determined to be contaminated with Sr-90 at a concentration of 97 pCi/mL (3.6 Bq/mL). Analytical results of the soil and water determinations are contained in Appendix D.

The following persons are considered as knowledgeable about the tanks and their operation:

Breeding, Swan, Operations Division, ORNL, 615/574-7058
Lasher, Larry, Operations Division, ORNL, 615/576-5223

5.7 WASTE TANK TH-4

Waste Tank TH-4 is located in the central ORNL complex between Central Avenue and White Oak Avenue (Fig. 2.1.1). The nearest Building is 3500. The tank is of gunite construction and was placed in service in 1943 to receive waste from the irradiated thorium and uranium pilot

plant development projects. This tank has a capacity of 14,265 gal. (5.4×10^4 L). Diameter of the tank is 20 ft (6.1 m) with a height of 9 ft (2.7 m). The tank is covered with approximately 6 ft (1.8 m) of soil.

Presently, the tank is filled with alkaline thorium and uranium sludge (5.4×10^4 L). Most of the radioactivity is due to the presence of Cs-137, except for small quantities of transuranics (Huang 1984b). Total radioactivity is estimated to be less than 1 Ci. The radiological inventory is summarized in Table 5.7.1. Results of uranium analysis on Tank TH-4 and transuranics in soil and water samples from the tank site are depicted in Appendices A and B, respectively. Radiation absorbed dose rates were determined inside Tank TH-4 (Huang 1984b). The maximum rate observed above the tank contents was 8.1 mrad/hr. The direct reading at the tank opening was 0.5 mrad/hr. No direct dose rate greater than 0.1 mrad/hr was detected at the tank site, except for readings of 0.26 and 0.32 mrad/hr at pipe locations. Soil samples revealed that there was no significant contamination at this site with no activity exceeding 1 Bq/g (27 pCi/g) (Table 5.7.2).

The following individuals are knowledgeable concerning Tank TH-4 and its operation:

Breeding, Swan, Operations Division, ORNL, 615/574-7058
Lasher, Larry, Operations Division, ORNL, 615/576-5223

5.8 OLD HYDROFRACTURE FACILITY

The Old Hydrofracture Facility (OHF) is located 1.1 miles (1.8 km) southwest of the main ORNL complex (Fig. 5.8.1) west of the Solid Waste

Table 5.7.1 Radiological inventory of the contents of Waste Tank TH-4 (Source: Huang et al. 1984b)

Tank (capacity)	Tank contents			Major radionuclides	Concentration[a] (Bq/mL or Bq/g)	Inventory	
	Type	Depth (m)	Volume (L)			(GBq)[b]	(mCi)
TH-4 (5.4×10^4 L)	Water	2.0[d]	37,000[c]	^{137}Cs	7.6	0.3	8
				^{90}Sr	0.26	0.01	0.3
				^{238}Pu	1.7	0.06	2
				^{239}Pu	1.7	0.06	2
				^{241}Am	<1.0	<0.04	<1
				^{244}Cm	<1.0	<0.04	<1
				^{232}Th	0.01	0.0004	0.01
	Sludge	1.1[d]	21,000[c]	^{137}Cs	6.6×10^2	20	500
				^{90}Sr	92	2	60
				^{238}Pu	14	0.4	10
				^{239}Pu	27	0.7	20
				^{241}Am	80	2	50
				^{244}Cm	60	2	40

[a]Results for sludge samples are expressed in Becquerels per gram of wet sample. Water sample results are in Becquerels per milliliter.
[b]1 Bq = 27 pCi.
[c]Tank TH-4 is filled with sludge and water. The volume of water and sludge may be overestimated.

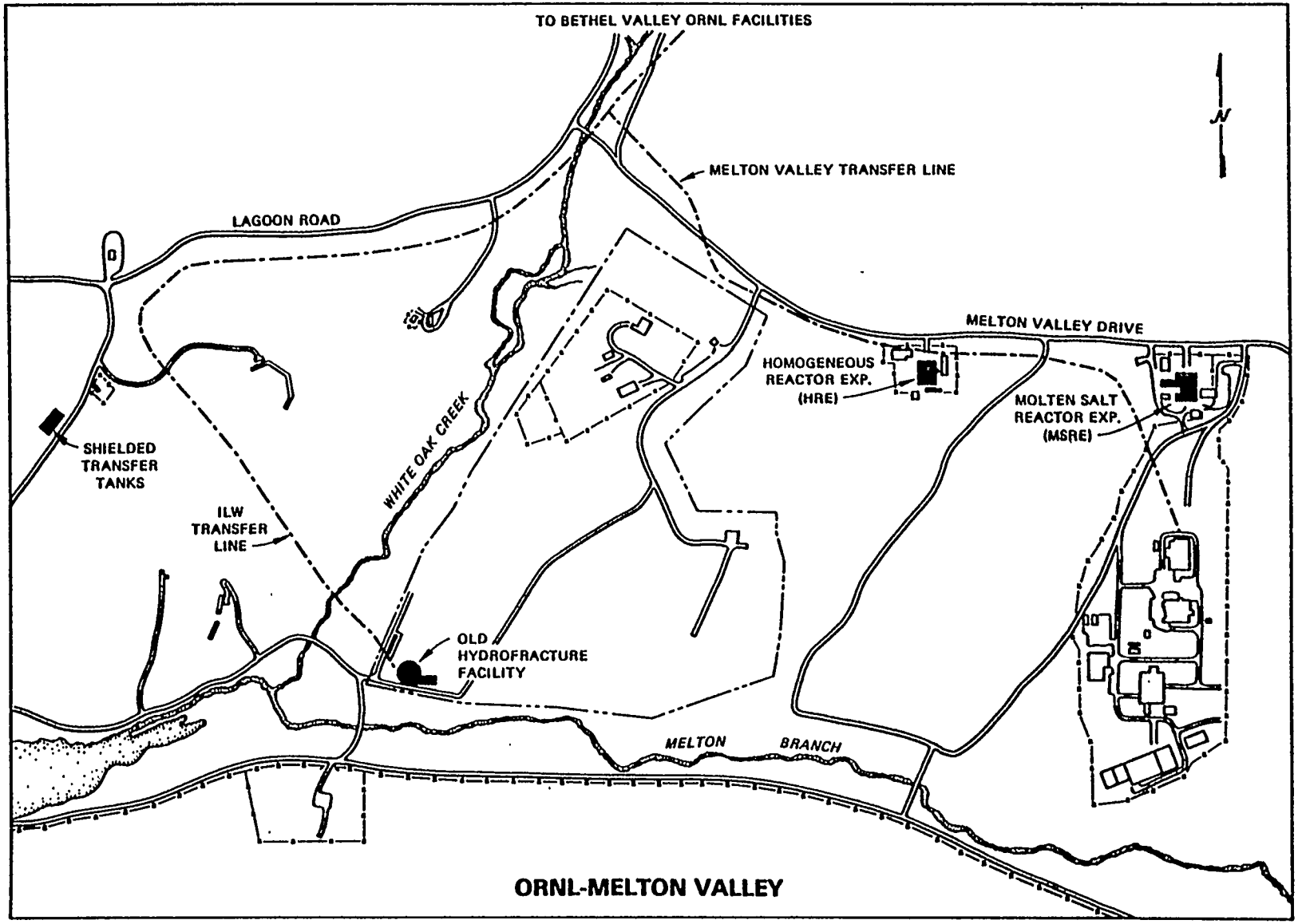
Table 5.7.2 Results of the preliminary gamma screening[a] and the quantitative gamma and beta analyses[b] on soil samples from Tank Site TH-4

Sample identification	Gamma activity (cps/kg)	^{137}Cs (Bq/g)[c]	^{90}Sr (Bq/g)	Others (trace amount)
1-1	41	NA[d]	NA	
1-2	48	0.10	^{60}Co	
1-3	30	NA	NA	
1-4	27	NA	NA	
1-5	25	0.01	NA	
1-6	29	NA	NA	
1-7	35	NA	NA	
1-8	33	0.14	0.11	
2-1	45	NA	NA	
2-2	35	0.27	0.007	^{60}Co
2-3	160	1.2	0.004	^{60}Co
2-4	29	0.13	0.007	
2-5	33	NA	NA	
2-6	33	NA	NA	
2-7	25	NA	NA	
2-8	43	0.08	NA	
3-1	46	NA	NA	
3-2	45	0.07	0.005	
3-3	30	NA	NA	
3-4	31	NA	NA	
3-5	29	NA	NA	
3-6	50	0.04	NA	
3-7	35	NA	NA	
3-8	33	0.04	0.001	

[a]Gamma activity was measured in counts per second (cps) per kilogram of moist soil in 100-1500 keV energy range.
[b]^{137}Cs and ^{90}Sr are reported on a dry weight basis.
[c]1 Bq = 27 pCi.
[d]NA indicates not analyzed.

Fig. 5.8.1 Location of the Old Hydrofracture Facility at the Oak Ridge National Laboratory (from Reed 1984).

ORNL-DWG 82-19186



ORNL-MELTON VALLEY

Storage Area #5. The facility was constructed in 1963 and operated until 1980 as a pilot plant to demonstrate the feasibility of permanent disposal of liquid radioactive waste in impermeable shale formations by hydrofracture methods. The facility consists of an uncontaminated building (7853) and its equipment. Contaminated structures include Building 7852, well and well head, pumphouse and valve pit, waste pits, waste tanks, and underground piping. An emergency retention pond is discussed in a separate report (Taylor 1986). Information on major structures of the facility has been abstracted from the study of Reed (1984) and Huang et al. (1984c), and each is presented in this report in a condensed summary. A site plan of the facility is depicted in Fig. 5.8.2.

5.8.1 Building 7852

Building 7852 includes a mixing cell, a pump cell for the head end of the injection pump, a well cell, a transite-roof-covered engine pad, and a control room. The mixing cell contains a mixer and a grout mix tub with its agitators. The pump cell contains minor equipment such as hoses, ladders, etc. A small drum-sized tank, piping, valves, and top of the injection well are contained in the well cell. The control room consists of an observation platform and several instruments.

A radiological characterization was made for the Old Hydrofracture Facility (Huang et al. 1984c). The levels of contamination for Building 7852 are summarized in Table 5.8.1 along with data for the pumphouse. In the control room, absorbed dose rates ranged from 75 to 600 mrad/hr, while the maximum transferrable beta-gamma activity and alpha activities were 49,000 dpm/100 cm^2 and 40 dpm/100 cm^2,

Fig. 5.8.2 Site plan of the Old Hydrofracture Facility (from Reed 1984).

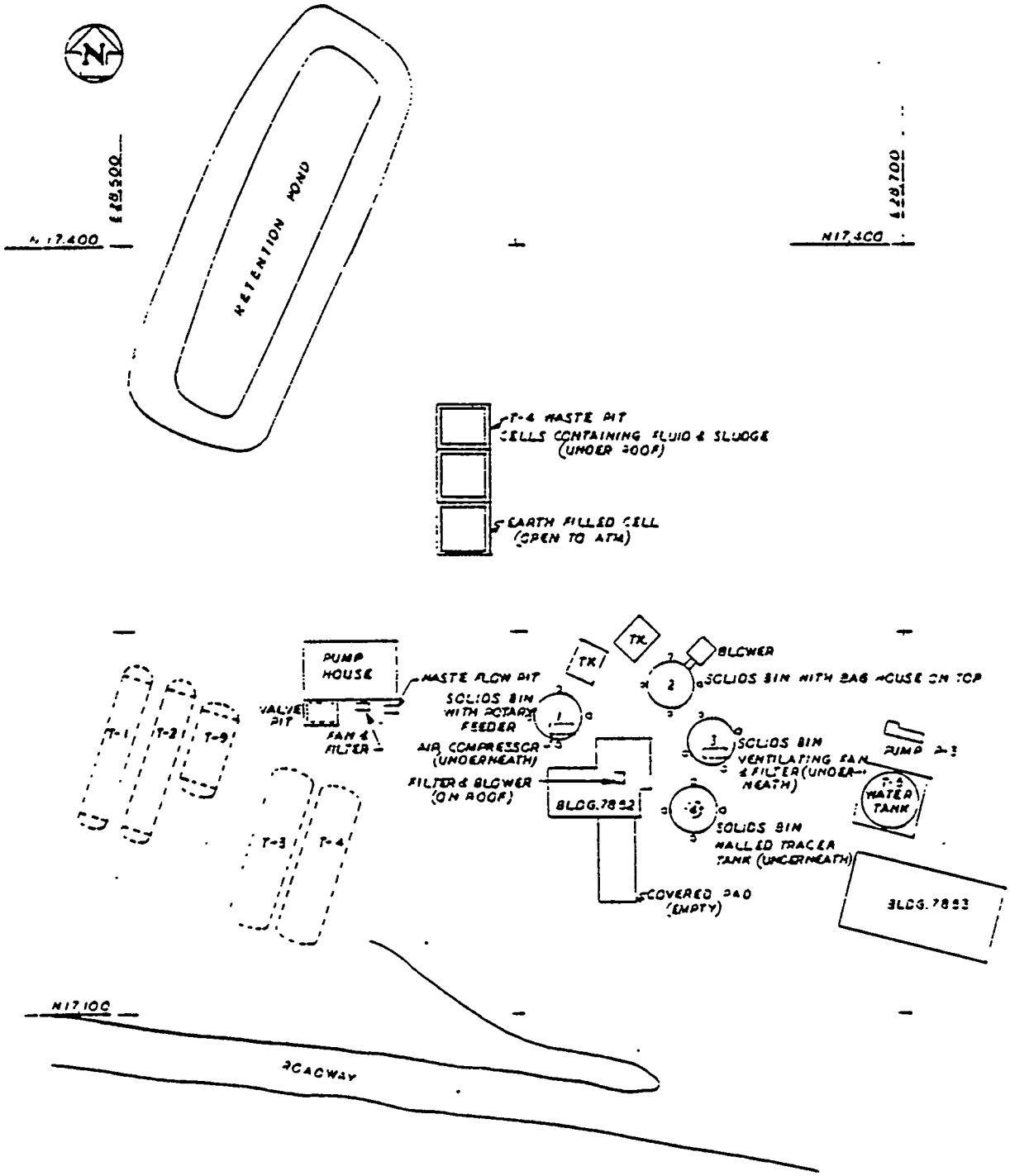

N
RETENTION POND
E 28,500
E 28,700
N 17,400
N17,300
T-4 WASTE PIT
CELLS CONTAINING FLUID & SLUDGE
(UNDER ROOF)
EARTH FILLED CELL
(OPEN TO ATM)
PUMP HOUSE
WASTE FLOW PIT
VALVE PIT
FAN & FILTER
T-1
T-2
T-3
T-4
SOLIDS BIN WITH ROTARY FEEDER
AIR COMPRESSOR (UNDERNEATH)
FILTER & BLOWER (ON ROOF)
TK
TK
BLOWER
SOLIDS BIN WITH BAG HOUSE ON TOP
SOLIDS BIN
VENTILATING FAN & FILTER (UNDERNEATH)
PUMP
WATER TANK
BLDG. 7852
SOLIDS BIN
HALLED TRACER TANK (UNDERNEATH)
COVERED PAD (EMPTY)
BLDG. 7853
N17100
ROADWAY

Table 5.8.1 Radiation levels in 7852 and in Pump House (from Reed 1984)

Location	Radiation level		
		Smearable (per 100 cm^2)	
	Direct $\beta - \gamma$ (mrad/hr)	α (dpm)	$\beta - \gamma$ (mrad/hr) except as noted
7852			
Mixing cell	Avg. 2500 Max. 4000	Generally <20 Max. 102 (1 location)	Avg. 16 Max. 30
Pump cell	Avg. 625 Max. 2000	<20	Avg. 2.4 Max. 5
Well cell	Avg. 1320 Max. 3000	<20	Avg. 23 Max. 36
Engine pad	Avg. 130 Max. 300	Generally <20 Max. 27 (1 location)	Avg. 0.85 Max. 1
Control room	Avg. 75-80 except max. 600 (at stored items)	21 - 36	21,000-49,000 dpm
Pump House			
Pump room	Avg. 1340 except for max. 8,000 (under lead shield)	<20	0.5-16 mrad/hr
Valve pit	5	Not given	Not given

respectively. Higher dose rates were observed in the mixing, pump, and well cells. For example, dose rates ranged from 400 to 4,000 mrad/hr in the mixing cell, 150 to 2,000 mrad/hr in the pump cell, and 200 to 3,000 mrad/hr in the well cell. Data for the remaining structures (pump room and engine pad), including a summary of transferrable contamination, are presented in Table 5.8.2.

5.8.2 Waste Pit Cells

There are three waste pit cells of concrete construction associated with this facility. One cell is filled with grout which is known to be radioactive. The other two cells contain contaminated water and sediments over an undetermined layer of set-up grout. The direct dose rates under the roof were 10-40 mrad/hr. The middle cell had a layer of 18 in. (0.45 m) of water over a layer of 4 in. (10 cm) of sediment. The north cell had a layer of 36 in. (0.9 m) of sediment. Presumably, the two cells contain solidified grout in the very bottom (beneath the sediment). Perhaps as much as 3 ft (0.9 m) of grout could be in each cell. As a total, there are ~20,000 L (5300 gal.) of water containing 5 GBq (0.1 curies and ~3000 L (800 gal.) of sediment containing 10 GBq (0.3 Ci) in the two cells. An inventory of the radionuclides in the waste pits is presented in Table 5.8.3. The middle cell sediments contained more Pu-238 with 0.8 nCi/g in comparison to the north cell with 0.5 nCi/g. Also, Pu-239 was more abundant in the middle cell sediments (2.6 nCi/g) than in the north cell (1.1 nCi/g). Sediments in the middle cell were lower in Cm-244 (1.5 nCi/g) than the sediments in the north cell (16 nCi/g). Sediments in both cells contained similar amounts of Am-241 (0.35 and

Table 5.8.2 Summary of the interior survey results at the OHF (Source: Huang et al. 1984c)

Location	Direct radiation levels (mrad/h)	Smearable activity[a] per 100 cm^2		
		Beta-gamma		Alpha
		µGy/h[b]	mrad/h[b]	dpm
Control room	75-600	21,000-49,000 dpm		21-36
Mixing cell	400-4000	50-300	5-30	<20-100
Pump cell	150-2000	20-50	2-5	<20[c]
Well cell	200-3000	100-350	10-35	<20
Pump room	20-8000	5-150	0.5-15	<20
Engine pad	20-300	5-10	0.5-1	<20-27

[a]Wet paper towel techniques were used because of the high levels of transferrable contamination and the rough surfaces.
[b]These smears were surveyed with portable instrumentation rather than smear counters to prevent contamination of the smear counters.
[c]Alpha smearable activity $\geq$20 dpm is the current ORNL guide for defining a contamination zone or contaminated equipment.

Table 5.8.3 Waste pit inventories of radionuclides (Source: Huang et al. 1984c)

	Average depth and volume	Radionuclide	Concentration (Bq/mL or Bq/g)[a]	Inventory[b] (GBq)	Inventory[b] (mCi)
South Cell Water	45 cm 6.3×10^3 L	^{137}Cs	1.2×10^2	0.8	20
		^{239}Pu	1.1	0.007	0.2
		^{238}Pu	2.2	0.01	0.4
		^{241}Am	<1	0.006	0.2
		^{244}Cm	1.0	0.006	0.2
		^{90}Sr	15	0.09	3
South Cell Sediment	10 cm 1.4×10^3 L	^{137}Cs	4×10^3	6	200
		^{60}Co	31	0.04	1
		^{239}Pu	97	0.1	4
		^{238}Pu	30	0.04	1
		^{241}Am	17	0.02	0.06
		^{244}Cm	54	0.08	2
		^{90}Sr	7.8×10^2	1	30
North Cell Water	90 cm 1.3×10^4 L	^{137}Cs	2.1×10^2	3	80
		^{239}Pu	1	0.01	0.4
		^{238}Pu	1.5	0.02	0.5
		^{241}Am	<1	<0.01	<0.4
		^{244}Cm	<1	<0.01	<0.4
		^{90}Sr	16	0.2	6
North Cell Sediment	10 cm 1.4×10^3 L	^{137}Cs	3.1×10^3	4	100
		^{60}Co	14	0.02	0.6
		^{239}Pu	42	0.06	2
		^{238}Pu	17	0.02	0.6
		^{241}Am	13	0.02	0.6
		^{244}Cm	6×10^2	0.8	20
		^{90}Sr	4×10^2	0.6	20

[a] 1 Bq = 27 pCi.
[b] Inventory in the sediment was estimated by assuming that the total volume was equal to the total dry weight.

0.45 nCi/g). Average concentrations for transuranics in sediments were 3.9 nCi/g for the middle cell and 2 nCi/g for the north cell (Huang et al. 1984c).

5.8.3 Waste Storage Tanks

There are five carbon steel waste tanks at the Old Hydrofracture Facility. These tanks were for the storage of waste prior to blending in the preparation of the grout. All tanks were installed on concrete pads in open pits with concrete-dividing walls to separate the tanks and permit the monitoring of each tank for leakage. There is a 4 ft (1.2 m) layer of soil over the tanks.

No measure of total residual radioactivity was made for the tanks. Major radionuclides include Cs-137, Sr-90, Co-60, and TRU-type materials. In a radiological characterization of the OHF, Huang et al. (1984c) made an approximation of radioactivity present by assuming that each tank contained a 12 in. (0.3 m) layer of sludge. It was further assumed that the radioactive concentration was similar to the original grout (0.26 mCi/ml). On those bases each tank might contain from 30 to 40 TBq (600 to 1000 Ci) or a total of 190 TBq (4600 Ci) in $2x10^4$ L of sludge. A summary of tank characteristics is presented in Table 5.8.4. In a walk-over survey, the direct beta-gamma absorbed dose rate in the vicinity of the waste storage tanks was less than 1 mrad/hr at 1 to 3 cm above the surface.

A gamma screening and quantitative gamma and beta survey was made on soil samples of the OHF. Those data are contained in Appendix E. The most detailed information of the Old Hydrofracture Facility

Table 5.8.4 Estimate[a] of residual radioactivity in waste tanks at the OHF (Source: Huang et al. 1984c)

Tank[b]	Tank capacity (L)	Tank diameter (m)	Tank length (m)	Sludge volume (L)	Sludge activities (TBq)	Sludge activities (kCi)
T-1	5.9×10^4	2.4	13	4.2×10^3	40	1
T-2	5.9×10^4	2.4	13	4.2×10^3	40	1
T-9	4.4×10^4	3.1	5.9	2.2×10^3	30	0.6
T-3	1.1×10^5	3.2	13	4.9×10^3	40	1
T-4	1.1×10^5	3.2	13	4.9×10^3	40	1

[a]The estimate was based on the assumption of 30 cm thickness of residual sludge (a rough estimate from operational experience) at a concentration of 0.26 Ci/L.
[b]Tanks are positioned horizontally.

(including drawings, maps, and photographs) is presented in an excellent report by deLaguna et al. (1968).

The following individuals are knowledgeable concerning the operation of the facility:

Lasher, Larry, Operations Division, ORNL, 615/576-5223
Tamura, Tsuneo, Environmental Sciences Division, ORNL, 615/576-8256

5.9 FISSION PRODUCTS DEVELOPMENT LABORATORY (FPDL) LLW TRANSFER LINE (BUILDING 3517)

Between 1958 and 1978, low-level waste was transferred from the Fission Products Development Laboratory (Building 3517) to the South Tank Farm (area 3507) through a divided line with waste streams to gunite storage Tanks W-5 and W-6. Construction design consisted of a stainless-steel pipe approximately 30 ft (9.1 in.) long, 1.5 in. (3.8 cm) diameter, and covered with a 3 ft (0.9 m) layer of soil. This portion of the transfer line was abandoned when the two gunite tanks (W-5 and W-6) were removed from service. While there has not been a radiological inventory of the line, it is expected that it is contaminated with similar radionuclides (Cs-137 and Sr-90) as noted for Tanks W-5 and W-6 (DeVore 1986). No estimate of total radioactivity for this line is available.

The low-level wastes from the Fission Products Laboratory (Building 3517) is currently transferred to a collection header on the west side of the South Tank Farm. The line is 3.8 cm (1.5 in.) in diameter and 96 m (317 ft) long, and has been in operation since 1958 without any history of leakage. While there has not been a radiological inventory of the line, it is expected that it is

contaminated with similar radionuclides (Cs-137 and Sr-90) as noted for Tanks W-5 and W-6. A double-contained line is currently under construction and is expected to replace the present line in fiscal year 1987.

The following individuals are knowledgeable concerning the transfer line and its operation:

DeVore, Joe, Operations Division, ORNL, 615/574-7092
Scott, Chris, Operations Division, ORNL, 615/574-7057

5.10 FISSION PRODUCTS DEVELOPMENT LABORATORY (FPDL) FILTER PIT (BUILDING 3517)

A filter pit east of Building 3517 was put in service in 1958 to filter building air exhaust from the Fission Products Development Laboratory (Fig. 2.1.1). Two banks of fiber roughing filters were arranged in series. The roughing filters were replaced in the 1960's with stainless-steel back-washable filters. These were removed in the mid-1970's and replaced with High Efficiency Particulate Absolute (HEPA) finishing filters.

The practice of acid-backwashing of the stainless-steel filters contributed to the present level of contamination of the filter pit. The back-wash solution fell to the floor, where the present radiation exposure from concrete surface is estimated to be ~200 R/hr (DeVore 1986). The contaminant radionuclides are principally Cs-137 and Sr-90.

Adjacent to the filter pit is the filter house, an integral part of the exhaust air filtration system of the Fission Products Development Laboratory (Building 3517). It is located east of Building 3517 in the central ORNL complex. Filtered air from the filter pit

moves to the HEPA filters of the filter house. The filter house is of steel-plate and concrete block construction and consists of three sets of three parallel rows of HEPA filters. It has been in service since the mid-1960's. No systematic inventory of radioactive contamination has been made of the filter house. A Health Physics survey indicated an exposure of 5 to 10 R/hr on the concrete and steel-plate surfaces (DeVore 1986). Excavation at the site removed large quantities of contaminated soil, indicating that more contamination will likely be found just outside the excavation area. The pit is known to accumulate ground water in its sump which contributes to the facility's LLW disposal burden (Burwinkle 1986). These findings indicate that some upgrade work will be necessary once the system becomes surplus. It is suspected that the principal radionuclide contaminants are Cs-137 and Sr-90.

Detailed information of the filter house and its operation is available from Joe DeVore, Operations Division, ORNL, 615/574-7092.

5.11 ISOTOPES DUCTWORK/3110 FILTER HOUSE

This remedial action site presently consists of a filter house to handle the cell ventilation air exhaust in the isotopes area. The filter house receives exhaust from the various buildings through underground ducts. Those buildings are 3028 (Radioisotope Production Laboratory-A), 3029 (Radioisotope Production Laboratory-B), 3030 (Radioisotope Production Laboratory-C), 3031 (Radioisotope Production Laboratory-D), 3032 (Radioisotope Production Laboratory-E), and 3033, 3033A (Radioisotope Laboratories-F). The facility also served Building

3038 (Radioisotope Laboratory), and Building 3047 (the Isotope Technology Building). From the filter house (3110), the air flow was to the 3029 stack. There are 42 HEPA filters in the house. The filter house began operation in the early 1960's. Prior to that time (early 1940's to 1984), a section of duct connected the filter house with a discharge stack. This duct was removed in 1984.

A floor drain in Building 3110 collected ground water and transported the water to sump by gravity flow where it was pumped to the LLW system (Tank WC-10). The interior of the ductwork and filter house is contaminated with a variety of radionuclides from past operations, as are the filter house and the filters still contained within (Burwinkle 1986). The duct is known to accumulate ground water through leakage which collects in a sump for transfer to the Process Waste System.

A number of engineering drawings provide details of the filter house. They are:

D39120	D43475
D39121	D45152
D39122	D45155
D39123	
D39124	
D39125	

Specific information can be obtained from Sam Gheesling, Plant and Equipment Division, ORNL, 615/574-4243. There is a need for more precise details of the kinds of radionuclides, extent of contamination, and radiation dose rates.

5.12 LOW-LEVEL WASTE TANK W1-A

Tank W1-A is located in the North Tank Farm (area 3023), northeast of the intersection of Third Street and Central Avenue (Fig. 2.1.1). The tank is of stainless-steel construction, has a capacity of 5,000 gal. (1.9×10^4 L), and is covered with a layer of soil in excess of 6 ft (1.8 m). Installed in the mid-1950's, Tank W1-A received wastes from the high radiation level analytical facilities of Buildings 2026, 3019-B, and the Radiochemical Processing Pilot Plant (3019) until early 1986 when it was removed from service. No waste has been collected from 3019-B in about ten years. Waste from Building 3019 contained U-233 while waste from Building 2026 was a mixture of fission product radionuclides. In a scoping survey it was estimated that radiation exposure rates in excess of 100 R/hr may be present in the tank and inlet process waste lines (Houser 1986a). Because of the relatively low background measured at ground level (3 mR/hr), it is assumed that there is no external contamination from tank leakage. The transfer line to Tank W1-A from Buildings 2026 and 3019 is strongly suspected of leaking and contaminating soil near the tank. This is the reason Tank W1-A was removed from service.

Knowledgeable persons concerning this tank include:

Breeding, Swan, Operations Division, ORNL, 615/574-7058
Lasher, Larry, Operations Division, ORNL, 615/576-5223

5.13 DECONTAMINATION FACILITY (BUILDING 7819), ORNL

The ORNL Decontamination Facility (Building 7819) is located north of Lagoon Road near the entrance to SWSA 4. It began operation in the early 1960's and was closed in the late 1970's. The facility was an

unshielded decontamination operation, consisting of a "C" zone for acid baths and a "clean" area. Sand blasting was also used for more rigorous cleaning. Isotope carriers from laboratories and hot cells were among the items decontaminated. Some contaminated equipment remains at the facility. The contamination consists of the building, equipment, and the land area within a 50 ft (15 m) radius of the building. In a radiological characterization survey, the radiation exposure rate (β and γ) ranged from a maximum of 3.5 R/hr (equipment and shelves) in the "C" zone to <10 mR/hr at floor level in the clean area (Houser 1986b). Wet towel smears indicated the transferrable radioactivities (β and γ) at the same locations were much less (e.g., 2-27 mR/hr, "C" zone equipment and shelves; and 1-5 mR/hr, clean area floor level). The uncovered sand utilized in blast cleaning is located about 40 ft (12 m) from the building and has a maximum exposure rate of 2 mR/hr.

Most of the information concerning the facility was abstracted from the scoping survey (Houser 1986b). Knowledgeable persons concerning the site include:

Lasher, Larry, Operations Division, ORNL, 615/576-5223
McNabb, Harry, Operations Division, ORNL, 615/574-7085

5.14 DECONTAMINATION FACILITY (ORNL AT Y-12), BUILDING 9419-1

This facility was in service between the mid-1960's and the early 1970's. It was dedicated to the Molten Salt Reactor Experiment (MSRE). It is located in the southeast portion of Y-12 in Bear Creek Valley. The facility was primarily a steam-cleaning facility, housed in a 20x30 ft (6x9 m) building. Thorium is the principal contaminant,

and the contamination is largely surface in nature. This facility also includes an outdoor concrete, walled pit which is uncovered and is connected with Building 9419-1 and East Fork Poplar Creek by drain lines. The pit is about 4.2 x 4.3 m (14 x 14 ft). The drain lines are between 20 and 25 m (65 and 82 ft) from Building 9419-1 and East Fork Poplar Creek, respectively. A cursory survey of the pit indicated that it was free of contamination, but an in-depth characterization of the pit was not possible in the initial survey (Burwinkle 1986). To assess the potential for environmental impact from this facility, the pit will need to be more fully explored including water and sludge analyses. There is no measure of radiation exposure rate or total radioactivity. T. G. Ramsey, Engineering Technology, Y-12, 615/574-0403, should be contacted for details of the facility and its operation.

6. DISCUSSION

The several radwaste facilities at ORNL have been reviewed with the objective of providing descriptive information and identification of data needs for assessment tasks. A number of the facilities are waste collection and storage tanks, with remedial action control and supervision in the Surplus Facilities Management Program (SFMP). Additional facilities (e.g., pipelines, ductwork, filter pits, etc.) are contained in the Surplus Contaminated Facilities Program (SCFP). Both of these programs are in the Site Decommissioning Program, designed for the management of surplus facilities (primarily structures), contaminated with radioactive and hazardous chemical wastes.

Information on radiation dose rate, total radioactivity (Ci), and volume of residue (sludge/water) is presented for many tanks in the Surplus Facilities Management Program. There are data deficiencies for these parameters among Tanks W-15, WC-1, and WC-15. The suspected Pu-238 contamination in Tank W-2 needs to be confirmed by additional sampling and analyses. Estimates of total radioactivity and the volume of water/sludge residue is available for Tank WC-17, but measurements of radiation dose rate are not readily available. The Thorium Pilot Plant (Building 3503) waste tank (TH-2) is of major concern as the radionuclide inventories may exceed those in any of the tanks sampled. No radiological inventory was made for the tank contents.

There is very little information concerning some of the radwaste-related facilities in the Surplus Contaminated Facilities Program. Included are Fission Products Development Laboratory (3517 Building) Transfer Line and Filter Pit, the Isotopes Ductwork/3110 Filter House, Tank W1-A, Tank TH-2, and the ORNL Decontamination Facilities at Y-12 (9419-1) and X-10 (7819). Work should be initiated soon to provide necessary information. An important consideration should be measurements of radiation dose rate. Without this information risk analyses cannot be made.

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Appendix A. Results of uranium analyses on samples from all tank sites.

Tank site	Sample	^{235}U (ng/g)	^{235}U (mBq/g)[a]
WC-1	1-6	54	4
	1-9	11	1
	1-10	220	20
	3-1	27	2
	3-4	29	2
	3-7	35	2
	3-8	46	4
	3-10	14	1
W-11	2-1	61	0.1
	2-8		
	3-1	1,000	100
	3-2	510	40
	3-3	410	40
	3-4	280	20
	3-5	130	10
	3-7	420	40
	3-8	54	40
	4-3	27	0.2
	5-4	30	2
	5-7	63	50
	5-8	2,700	200
WC-15,17	1-3	23	2
	1-5	21	2
	1-7	23	2
	4-4	23	2
	4-8	19	2
	5-3	21	2
	5-7	21	2
	6-3	20	2
	6-5	23	2
	6-7	16	1
TH-1,2,3	1-7	31	2
	1-8	40	3
	1-9	75	6
	2-9	57	5
	6-7	24	20
	6-9	23	2
	6-9[b]	0.1	1
	7-8	22	2

Appendix A. (CONT'D)

Tank site	Sample	^{235}U (ng/g)	^{235}U (mBq/g)[a]
TH-4	1-2	590	50
	1-8	200	20
	2-2	29	2
	2-3	22	2
	2-4	26	2
	3-2	66	5
	3-8	74	6
NTF	1-1	19	2
	1-2	19	2
	1-8	17	2
	2-9	110	90
	2-10	57	5
	7-3	210	20
	7-6	91	7
	9-3	21	2
	9-4	23	2
	16-1	40	3
	16-5	22	2
	16-8	22	2
	17-1	24	2
	19-9	6,200	5
	23-7	22	2
	24-1	20	2
	24-9	41	3
	24-10	25	2

[a]1 Bq = 27 pCi.
[b]A water sample, measured in Bq/mL.

Appendix B. Results of quantitative analyses for transuranics on selected soil and water samples from all tank sites.

Tank site	Sample	^{239}Pu	^{238}Pu	^{241}Am	^{244}Cm
		(mBq/g)[a]			
WC-1	1-1	ND[c]	ND	ND	ND
	1-2	<0.3	1	3	0.5
	1-5	1	<0.1	3	0.9
	1-10[b]	<0.1	<0.1	1	0.5
	2-1	0.05	2	3	3
	2-7	0.7	2	3	4
	3-1	2	6	6	0.9
	3-11[b]	<0.1	<0.1	0.2	<0.1
	4-5	0.3	0.6	2	1
	4-7	2	0.3	8	5
W-11	1-1	100	0.7	8	10
	1-2	30	0.5	0.7	6
	1-4	10	<0.4	0.8	1
	1-7	40	<0.1	1	1
	2-1	200	1	10	3
	2-2	20	<0.1	<0.2	<0.2
	2-5	2	0.2	0.3	0.9
	3-1	90	4	30	80
	3-2	40	2	60	10
	4-1	5	0.7	0.9	9
	4-2	20	3	10	60
	5-1	10	0.9	2	20
	5-2	40	0.7	3	3
WC-15,17	1-8	0.4	3	7	8
	4-3	5	30	1,000	800
	4-4	0.9	0.2	10	6
	4-7	5	30	4,000	3,000
	5-7	<0.3	<0.3	0.8	0.9
	6-7	200	400	70	1,000
TH-1,2,3	1-3	1	0.4	9	6
	1-4	2	<0.4	3	1
	1-5	100	2	90	50
	1-6	6	0.9	5	6
	1-10	20	0.3	6	8
	2-4	20	0.2	2	8
	2-6	30	0.9	20	40
	3-6	1	0.4	9	3
	6-7	<0.2	<0.2	2	2
	6-9	1	1	0.09	0.4
	7-8	70	30	30	60

Appendix B. (CONT'D)

Tank site	Sample	^{239}Pu	^{238}Pu	^{241}Am	^{244}Cm
		(mBq/g)[a]			
TH-4	1-2	2	0.1	3	<0.2
	1-8	0.2	2	2	4
NTF	2-1	10	0.3	6	3
	3-1	200	10	30	300
	3-3	8	0.2	4	0.6
	6-8	10	4	0.3	0.3
	7-2	2	<0.3	3	<0.2
	7-9	10	0.5	2	0.1
	9-1	8	<0.1	2	2
	10-1	10	0.5	5	1
	12-2	30	0.6	3	2
	13-1	40	<0.1	10	0.9
	13-2	20	<0.2	4	<0.1
	13-5	20	3	2	2
	13-10	0.1	2	4	10
	16-1	20	0.8	2	4
	16-2	9	<0.4	2	2
	17-2	0.6	<0.3	0.5	0.5
	18-1	2	1	2	7
	19-2	60	0.1	2	5
	21-2	10	0.2	9	5
	25-1	80	1	80	1

[a]1 Bq = 27 pCi.
[b]A water sample, measured in Bq/mL.
[c]ND indicates not detectable.

Appendix C. Results of the preliminary gamma screening[a] and the quantitative gamma and beta analyses[b] on soil samples from the site of North Tank Farm, Tanks W-1 through W-4, and W-13 through W-15.

Sample identification	Gamma activity (cps/kg)	^{137}Cs (Bq/g)[c]	^{90}Sr (Bq/g)	Others (trace amount)
2D-1[d]	NA[e]	8.1	NA	^{60}Co
2D-2[d]	NA	6.7	NA	^{60}Co
2D-3[d]	NA	4.1	NA	^{60}Co
1-1	150	0.85	ND[e]	
1-2	120	0.63	ND	
1-3	25	NA[d]	NA	
1-4	25	0.01	NA	
1-5	26	NA	NA	
1-6	39	0.01	NA	
1-7	43	0.004	NA	
1-8	170	0.31	NA	^{232}Th
2-1	110	0.44	0.26	^{60}Co
2-2	49	NA	NA	
2-3	37	0.09	NA	
2-4	26	NA	NA	
2-5	25	NA	NA	
2-6	23	NA	NA	
2-7	33	0.03	ND	
2-8	140	0.05	NA	^{60}Co, ^{232}Th
2-9	7,500	ND	1,800	
2-10	5,200	0.15	NA	^{60}Co
3-1	650	4.0	0.38	^{60}Co
3-2	520	3.5	ND	
3-3	42	0.29	0.53	
3-4	23	NA	NA	
3-5	25	0.004	NA	
3-6	31	NA	NA	
3-7	41	0.01	NA	
3-8	37	NA	NA	
3-9	27	NA	NA	
4-1	76	0.19	ND	
4-2	72	NA	NA	
4-3	53	0.19	NA	
4-4	39	NA	NA	
4-5	45	NA	NA	
4-6	41	NA	NA	
4-7	49	NA	NA	
4-8	47	NA	NA	
4-9	54	0.25	ND	

Appendix C. (CONT'D)

Sample identification	Gamma activity (cps/kg)	^{137}Cs (Bq/g)[c]	^{90}Sr (Bq/g)	Others (trace amount)
6-1	NS[e]	NA	NA	
6-2	NS	NA	NA	
6-3	35	0.09	NA	^{60}Co
6-4	27	NA	NA	
6-5	29	0.01	NA	
6-6	38	NA	NA	
6-7	24	NA	NA	
6-8	570	0.93	43	^{60}Co
7-1	510	4.1	NA	
7-2	1,000	7.4	1.8	
7-3	80,000	1,300	NA	^{57}Co, ^{134}Cs, ^{232}Th, ^{154}Eu
7-4	19,000	130	48	^{134}Cs, ^{232}Th, $^{152,154}Eu$
7-5	78,000	560	NA	^{134}Cs, ^{232}Th, ^{154}Eu
7-6	39,000	520	NA	^{60}Co, ^{232}Th, ^{154}Eu
7-7	20,000	2,700	19	$^{57,60}Co$, ^{134}Cs, ^{232}Th
7-8	5,600	63	NA	$^{57,60}Co$, ^{232}Th
7-9	140	1.6	0.35	^{154}Eu
7-10	49	0.33	NA	^{154}Eu
8-1	170	0.85	ND	
8-2	62	NA	NA	
8-3	28	NA	NA	
8-4	41	NA	NA	
8-5	130	0.1	NA	
8-6	44	NA	NA	
8-7	31	NA	NA	
8-8	41	NA	NA	
8-9	38	NA	NA	
8-10	31	0.01	NA	
9-1	160	0.94	0.64	
9-2	54	NA	NA	
9-3	38	NA	NA	
9-4	44	NA	NA	
9-5	33	NA	NA	
9-6	37	NA	NA	
9-7	26	NA	NA	
9-8	31	NA	NA	
9-9	24	0.01	NA	
10-1	150	0.67	0.62	^{60}Co
10-2	54	NA	NA	
10-3	30	NA	NA	
10-4	39	NA	NA	
10-5	30	NA	NA	
10-6	40	0.01	NA	

Appendix C. (CONT'D)

Sample identification	Gamma activity (cps/kg)	^{137}Cs (Bq/g)[c]	^{90}Sr (Bq/g)	Others (trace amount)
10-7	36	NA	NA	
10-8	33	NA	NA	
10-9	29	NA	NA	
10-10	42	0.01	NA	
12-1	110	NA	ND	
12-2	310	NA	0.24	^{60}Co
12-3	40	NA	NA	
12-4	45	NA	NA	
12-5	31	NA	NA	
13-1	290	2.1	0.28	
13-2	220	1.4	0.57	
13-3	110	0.85	NA	
13-4	130	0.67	NA	
13-5	380	3.1	0.51	
13-6	330	2.7	NA	
13-7	60	0.19	NA	
13-8	46	2.2	0.16	
13-9	110	0.81	NA	
13-10	28	0.05	0.09	
15-1	170	1.0	ND	
15-2	58	NA	NA	
15-3	54	0.33	NA	
15-4	NS	NA	NA	
15-5	28	NA	NA	
15-6	27	NA	NA	
15-7	31	0.06	ND	
16-1	400	2.9	0.35	
16-2	120	0.78	0.28	
16-3	160	1.1	NA	
16-4	34	0.15	NA	
16-5	39	0.10	NA	
16-6	76	0.41	NA	
16-7	48	0.10	NA	
16-8	28	0.09	NA	
16-9	50	0.27	ND	
16-10	78	0.67	NA	
17-1	2,200	19	NA	^{134}Cs
17-2	130	0.85	0.95	
17-3	180	NA	NA	
17-4	45	NA	NA	
17-5	54	0.11	NA	
17-6	60	NA	NA	
17-7	32	NA	NA	
17-8	42	0.06	NA	

Appendix C. (CONT'D)

Sample identification	Gamma activity (cps/kg)	^{137}Cs (Bq/g)[c]	^{90}Sr (Bq/g)	Others (trace amount)
18-1	70	0.22	0.29	
18-2	50	NA	NA	
18-3	81	0.41	NA	
18-4	49	NA	NA	
18-5	63	NA	NA	
18-6	33	NA	NA	
18-7	63	0.31	NA	
19-1	100	0.52	ND	^{60}Co
19-2	140	0.70	0.38	^{60}Co
19-3	31	0.04	NA	^{60}Co
19-4	30	NA	NA	
19-5	90	0.74	NA	
19-6	40	0.26	NA	
19-7	27	NA	NA	
19-8	37	0.06	ND	
19-9	NA	17,000	2,500	^{60}Co
21-1	110	0.56	NA	
21-2	340	2.5	0.18	
21-3	41	NA	NA	
21-4	40	NA	NA	
21-5	47	NA	NA	
21-6	57	0.01	NA	
21-7	44	NA	NA	
21-8	25	NA	NA	
21-9	31	NA	NA	
21-10	28	0.01	NA	
23-1	49	NA	NA	
23-2	100	0.37	ND	
23-3	45	0.35	NA	
23-4	56	0.26	NA	
23-5	64	0.34	NA	
23-6	20,000	180	NA	
23-7	10,000	100	5.8	
24-1	76	0.37	ND	^{60}Co
24-2	81	0.31	ND	
24-3	64	0.70	NA	
24-4	38	0.22	NA	
24-5	46	0.10	NA	
24-6	28	NA	NA	
24-7	38	0.05	NA	
24-8	29	NA	NA	
24-9	130	0.07	16	
24-10	50	0.14	21	

Appendix C. (CONT'D)

Sample identification	Gamma activity (cps/kg)	^{137}Cs (Bq/g)[c]	^{90}Sr (Bq/g)	Others (trace amount)
25-1	420	2.8	0.33	^{60}Co
25-2	150	0.89	ND	
25-3	28	NA	NA	
25-4	27	NA	NA	
25-5	34	0.1	NA	
25-6	28	NA	NA	
25-7	36	NA	NA	
25-8	41	0.06	NA	
25-9	5,900	63.0	91	^{60}Co, ^{154}Eu

[a]Gamma activity was measured in counts per second (cps) per kilogram of moist soil in the energy range 100-1500 keV.
[b]^{137}Cs and ^{90}Sr are reported on a dry weight basis.
[c]1 Bq = 27 pCi.
[d]Hand coring samples, 0.3 m in length.
[e]ND indicates not detectable, NS indicates no sample, and NA indicates not analyzed.

Appendix D. Results of the preliminary gamma screening[a] and the quantitative gamma and beta analyses[b] on soil and water samples from Tank Sites TH-1, 2, and 3.

Sample identification	Gamma activity (cps/kg)	^{137}Cs (Bq/g)[c]	^{90}Sr (Bq/g)	Others (trace amount)
1-1	NS[d]	NS	NS	
1-2	NS	NS	NS	
1-3	34	0.03	0.59	
1-4	75	0.52	0.32	
1-5	1,200	13	14	^{60}Co
1-6	100	0.59	5.3	
1-7	1,900	22	6.9	^{60}Co, ^{232}Th
1-8	790	15	NA[c]	^{60}Co, ^{232}Th
1-9	580	7.0	1.4	^{60}Co
1-10	230	1.9	3.1	
2-1	NS	NS	NS	
2-2	NS	NS	NS	
2-3	250	2.2	ND[d]	
2-4	87	0.48	170	^{60}Co
2-5	1,200	11	NA	
2-6	900	2.3	33	
2-7	2,100	NA	NA	
2-8	430	3.7	NA	^{60}Co
2-9	890	8.5	5.9	^{60}Co, ^{154}Eu
3-1	NS	NS	NA	
3-2	NS	NS	NA	
3-3	34	0.18	NA	
3-4	34	0.26	NA	
3-5	34	0.20	NA	^{60}Co
3-6	29	0.05	0.18	
6-1	NS	NS	NS	
6-2	NS	NS	NS	
6-3	29	NA	NA	
6-4	45	NA	NA	
6-5	120	NA	NA	
6-6	52	0.22	NA	
6-7	440	2.2	0.02	^{60}Co
6-8	NS	NS	NS	
6-9	17,000	280	7.9	
6-9[e]	NA	<0.01	1.3	^{60}Co

Appendix D. (CONT'D)

Sample identification	Gamma activity (cps/kg)	^{137}Cs (Bq/g)[c]	^{90}Sr (Bq/g)	Others (trace amount)
7-1	NS	NS	NS	
7-2	NS	NS	NS	
7-3	NA	NA	NA	
7-4	NA	NA	NA	
7-5	27	NA	NA	
7-6	48	NA	NA	
7-7	45	NA	NA	
7-8	230	2.0	0.56	^{60}Co

[a]Gamma activity was measured in counts per second per kilogram of moist soil in the energy range 100-1500 keV.
[b]^{137}Cs and ^{90}Sr are reported on a dry weight basis.
[c]1 Bq = 27 pCi.
[d]NS indicates no sample, NA indicates not analyzed, and ND indicates not detectable.
[e]A water sample.

Appendix E. Preliminary gamma screening results and quantitative gamma and beta analyses on soil samples of the Old Hydrofracture Facility.

Sample	Gamma activity (cps/kg)[a]	^{137}Cs (Bq/g)[b]	Sample	Gamma activity (cps/kg)	^{137}Cs (Bq/g)
1-1	<80	NA[c]	8-10	35	ND
1-2	33	NA	9-1[d]	<30	0.05
1-3	44	NA	9-2	28	NA
1-4	44	NA	9-3	60	0.56
1-5	34	NA	9-4	31	NA
1-6	42	NA	9-5	37	NA
3-1	66	0.37	9-6	35	NA
3-2	29	0.04	9-7	37	NA
3-3[d]	55	0.12	9-8	35	NA
3-4[d]	420	2.9	9-9	28	NA
3-5	65	0.2	10-1[d]	470	3.5
3-6	45	NA	10-2	120	1.0
3-7	32	NA	10-3	240	NA
3-8	34	NA	10-4[d]	34	0.03
5-1	35	0.09	10-5	38	NA
5-2	41	NA	10-6	25	NA
5-3	30	NA	10-7	32	NA
5-4	28	NA	10-8	38	NA
5-5	32	NA	10-9[d]	66	NA
5-6	33	NA	10-10	46	NA
5-7	31	NA	11-3	35	NA
6-1	41	0.2	11-4	36	NA
6-2	32	NA	11-5	38	NA
6-3	34	NA	11-6	38	NA
6-4	37	NA	11-7	34	NA
6-5	29	NA	11-8	30	NA
6-6	45	NA	11-9	30	NA
6-7	46	NA	11-10	29	NA
7-1[d]	96	1.1	11-11	32	NA
7-2	38	NA	11-12	32	NA
7-3	38	NA	11-13[d]	38	ND
7-4	31	NA	12-3	36	NA
7-5	32	NA	12-4	39	NA
7-6	34	NA	12-5	34	NA
7-7	34	NA	12-6	34	NA
8-3	44	NA	12-7	81	ND
8-4	24	NA	12-8[d]	240	ND
8-5	34	NA	13-3[d]	130	0.74
8-6	28	NA	13-4	68	0.35
8-7	53	NA	13-5[d]	70	0.26
8-8	26	NA	13-6	46	0.07
8-9	37	NA	13-7	69	0.52

Appendix E. (CONT'D)

Sample	Gamma activity (cps/kg)[a]	^{137}Cs (Bq/g)[b]	Sample	Gamma activity (cps/kg)	^{137}Cs (Bq/g)
13-8[d]	260	3.2	16-6	54	NA
13-9	40	0.02	16-8	33	NA
13-10	50	0.32	16-9	32	NA
13-11	58	0.27	16-10	67	NA
13-12	37	0.02	17-1	29	0.01
14-3	81	0.67	17-2	29	NA
14-4	84	0.31	17-3	30	NA
14-5	43	NA	17-4	32	NA
14-6	45	NA	17-5	40	NA
14-7	37	NA	18-1	43	0.17
14-8	41	NA	18-2	32	NA
14-9	34	NA	18-3	41	NA
14-10	53	<0.01	18-4	33	NA
15-1[d]	51	0.19	18-5	30	NA
15-2	43	NA	18-6	34	NA
15-3	40	NA	18-7	31	NA
15-4	33	NA	18-8	31	NA
15-5	40	NA	18-9	34	ND
15-6	31	NA	20-1	<30	0.05
15-7	64	NA	20-2	20	NA
15-8	41	NA	20-3	27	NA
15-9	42	NA	20-4	36	NA
15-10[d]	35	ND	20-5	31	NA
16-1[d]	88	0.48	20-6	31	NA
16-2	43	NA	20-7	37	NA
16-3	36	NA	20-8	39	NA
16-4	42	NA	20-9[d]	41	NA
16-5[d]	53	0.33			

[a]cps/kg indicates counts per second per kilogram of moist soil. Natural background is approximately 20-40 cps/kg on the detector used for this study [15 x 15 cm NaI (TI) detector; energy span 100-1500 keV].
[b]1 Bq = 27 pCi.
[c]NA indicates not analyzed and ND indicates not detectable.
[d]Sample also analyzed for ^{90}Sr, uranium, and transuranic elements.

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ORNL/RAP/LTR-86/52

DATE: June 30, 1986

SUBJECT: Inventory of ORNL Remedial Action Sites:
11. Research Laboratories

TO: T. E. Myrick

FROM: T. J. Blasing
B. S. Houser

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LETTER REPORT RAP86-52

Inventory of ORNL Remedial Action Sites: 11. Research Laboratories*

T. J. Blasing and B. S. Houser

June 30, 1986

Environmental Sciences Division
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37831

Operated by

Martin Marietta Energy Systems, Inc.
for the
U.S. Department of Energy

*Based on work performed at Oak Ridge National Laboratory, operated by Martin Marietta Energy Systems, Inc., under Contract No. DE-AC05-84OR21400 with the U.S. Department of Energy.

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SUMMARY

This report provides a current description of sites within the Remedial Action Program that are classified as research laboratories. This category presently includes five sites within the ORNL main complex plus two sites in ORNL areas at Y-12. These laboratories, and surrounding areas, are often still used for various activities not involving the surplus equipment that is to be decommissioned. Therefore, continuous radiological monitoring is often necessary. Equipment stored in these laboratories that is part of the Remedial Action Program is usually marked, roped off, or contained in some way to prevent unnecessary health hazards. This surplus equipment is mostly in good condition. However, the facilities in Building 3019-B (High-Level Radiation Analytical Laboratory) are deteriorating and should probably receive prompt attention and a high priority in the sequence of decommissioning projects.

Relatively little information is available regarding these sites, as compared to many other sites within the Remedial Action Program. Scoping surveys of the research laboratories have recently been completed but more detailed decommissioning studies including estimates of waste volume, identification of the preferred remedial action alternative for each site, and assignment of priorities have not yet been carried out.

1. OBJECTIVES

The primary purpose of the remedial action task is to provide corrective measures and facility decommissioning necessary to place Oak Ridge National Laboratory (ORNL) in compliance with applicable environmental and health regulations for radioactive and hazardous chemical wastes. Individual sites will be decontaminated in accordance with priorities prescribed to control the potential for human exposure and magnitude of ecological impacts. Remedial actions will be conducted at sites where past management practices or research activities have resulted in contamination of facilities or the environment that require corrective actions or decommissioning. Remedial actions are divided into two types of activity: (1) the Site Corrective Measures Program (SCMP) and (2) the Site Decommissioning Program. The Site Decommissioning Program is further divided into the Surplus Contaminated Facilities Program (SCFP) and the Surplus Facilities Management Program (SFMP).

The Site Corrective Measures Program is designed for the management of inactive wastes (e.g., process ponds, spill sites, and environmental research areas) that have been contaminated with radioactive materials and hazardous wastes, leading to the eventual stabilization of these sites in accord with environmental regulations. The Site Decommissioning Program is designed for the management of surplus facilities (primarily structures) from the end of their operating life until final facility disposition.

Each of the approximately 140 remedial action sites at ORNL is within one of three programmatic areas described above (SCMP,

SCFP, SFMP). Each site is also classified into one of 13 functional categories (e.g., Solid Waste Storage Areas, Process Ponds, etc.). A list of these categories is provided in Table 1. A series of reports provides a current inventory of the remedial action sites within each category. Each report covers one category and contains a summary of pertinent information abstracted from readily available literature and knowledgeable individuals on known or contaminated sites requiring remedial action. Information summarized includes site location and description, geology, hydrology, climatology, ecology, history, contaminants and their characteristics, extent of contamination, a reference list of major studies concerning the site, names of persons knowledgable about the site, and other relevant material. This report provides a current inventory of the seven research laboratories presently identified as remedial action sites (Table 2).

2. ENVIRONMENTAL CHARACTERISTICS

Several environmental characterizations of the areas around Oak Ridge National Laboratory (ORNL) have been published. Probably the most recent and comprehensive of these is the report of Boyle et al. (1982). Much of the material that follows is excerpted or updated from Chapter 3 of that report, or from an earlier report of Dahlman, et al. (1977).

2.1 Geology

The ORNL site is located in the Appalachian Physiographic Division of the eastern United States, within the Valley and Ridge Physiographic Province. The local topography is characterized by subparallel

Table 1. Functional Categories of Remedial Action Program Sites.

1. Solid Waste Storage Areas
2. Low-Level Waste Seepage Pits and Trenches
3. Process Ponds
4. White Oak Creek Watershed
5. Low-Level Waste Line Leak Sites
6. Environmental Research Sites
7. Hazardous Waste Sites
8. Radioisotope Processing Facilities
9. Experimental Reactor Facilities
10. Radwaste Facilities
11. Research Laboratories
12. Abandoned Injection Wells
13. Other Contaminated Sites

Table 2. Research Laboratories designated as Remedial Action Program Sites at Oak Ridge National Laboratory. All of these sites are regulated under DOE Order 5820.2 and all are in the Surplus Contaminated Facilities Program.

Name of Site	Plant	Building Number
1. High Level Chemical Development Laboratory	ORNL	4507
2. High Level Radiation Analytical Laboratory	"	3019-B
3. Remote Coating Furnace Loop	"	4508
4. Ceramic Processing Laboratory	"	4508
5. Transuranium Research Laboratories 41 and 45	"	5505
6. Molten Salt Reactor Experiment Fuel Handling Facility	Y-12	9201-3
7. Coolant Salt Technology Facility	Y-12	9201-3

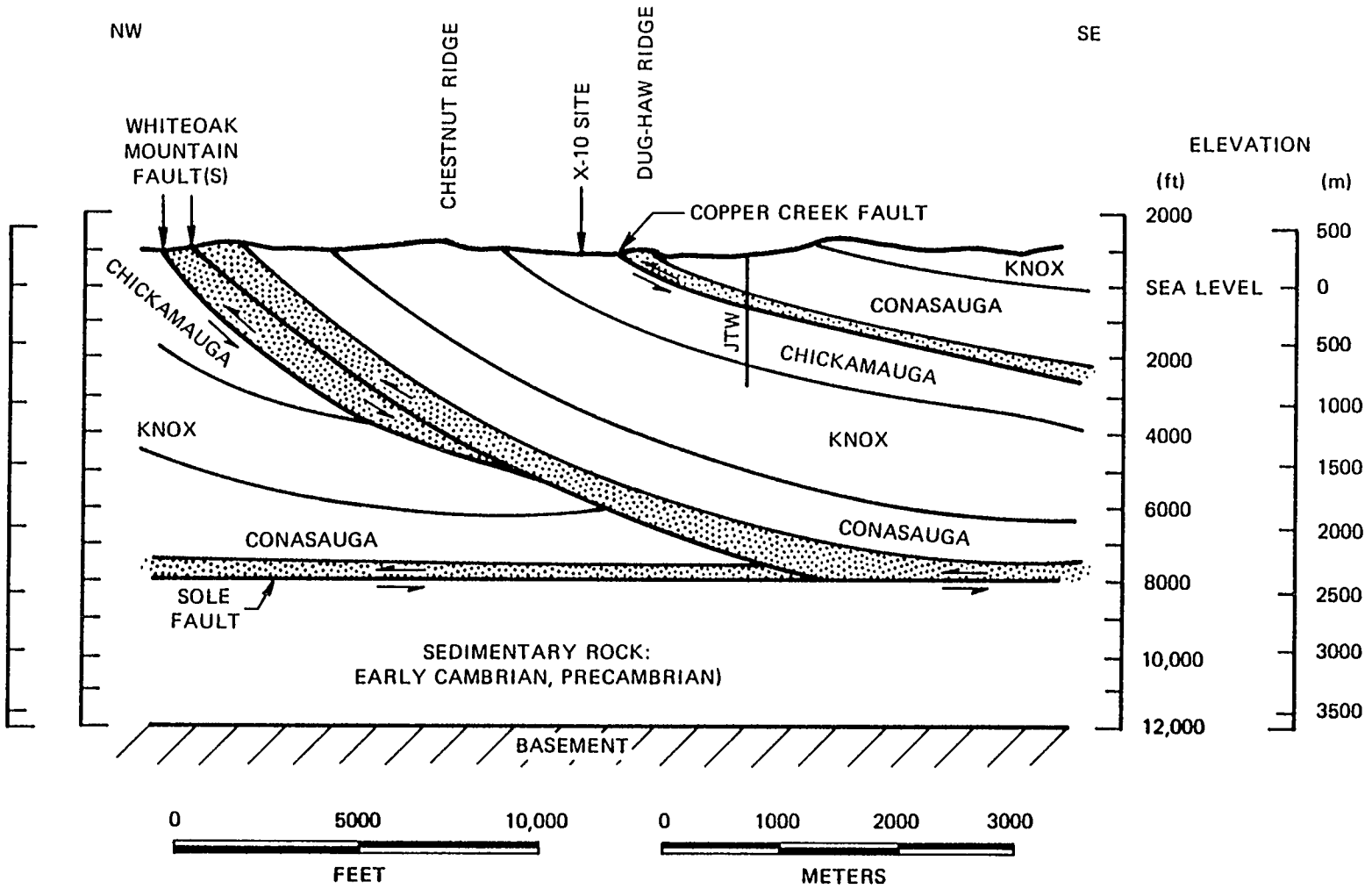
northeast-trending ridges and valleys associated with southeast-dipping strata. Nine geologic formations or groups ranging in age from Early Cambrian to early Mississippian have been mapped within the Oak Ridge Reservation. All are of sedimentary origin. From oldest to youngest they include the Rome formation, the Conasauga group, the Knox group, the Chickamauga Limestone, the Sequatchie formation, the Rockwood formation, the Chattanooga shale, the Maury formation, and the Fort Payne chert. Much of the younger material has eroded so that the first four strata listed above form the major underlying strata of the region. A northwest-southeast cross section of these major strata is shown in Figure 1. The Rome Formation, of Cambrian age, has been thrust over younger formations of later Cambrian and Ordovician ages. This has occurred in several places, causing the repetition of the geologic sequence within the area as seen in Figure 1.

The differing lithologies of the formations result in different erosion rates. More resistant, ridge-producing, formations include in the Rome formation and the Knox group. Most of the valleys are underlain by the Conasauga group or Chickamauga limestone. Surface elevations range from about 226 m at the Clinch River on the southern border of the site to about 413 m at the crest of Melton Hill.

Although numerous faults exist within the area, they apparently originated prior to the end of the Paleozoic era. Major tectonic activity seems to have ceased thereafter. No physiographic evidence indicating such activity has been observed along the thrust fault areas in the region. Consequently, there is no reason to expect tectonic activity will occur there in the future.

Figure 1. Geologic cross section of the ORNL area.

ORNL–DWG 82-13318



The Oak Ridge area experiences a moderate amount of seismic activity (5 earthquakes in the last 165 years with a Modified Mercalli intensity of V to VI), though no incidence of surface deformation has been documented. The type of seismic activity that occurs in this region is common throughout the world. No earthquakes of Modified Mercalli intensity VII were reported. Intensity VII earthquakes occur approximately one tenth as often as intensity V to VI earthquakes This suggests a recurrence interval of 300 to 1000 years for intensity VII earthquakes, which are not severe. Examples of damage to be expected from such earthquakes are: weak chimneys broken off at roofline, damage to weak masonry of low standards of workmanship, fall of plaster, and damage to concrete irrigation ditches.

Although a moderate level of seismic activity is experienced in the region, it is highly improbable that a major shock will occur in the Oak Ridge area for several thousand years. Forces from more seismically active areas such as those near Charleston, South Carolina or Memphis, Tennessee would be dissipated by distances of 600 km or greater.

General characteristics of the major strata in the region are given below.

The oldest formation in the area is the Rome formation. It consists of fine-grained shaley sandstone that is generally well cemented. Its ability to hold and transmit groundwater is due almost entirely to fractures.

The Conasauga group is primarily silty, slightly calcareous, shale interbedded with thin limestone units. The permeability seems to be associated with changes due to weathering of the rock above and somewhat below the water table. Where the shale is saturated, movement of groundwater is concentrated in a zone near the water table and perhaps

locally deeper along some fractures. The fractures in the conasauga group, and therefore the permeability, are evenly distributed. This results in a more uniform flow of groundwater than in other formations. The Knox group is composed largely of thick beds of dolomite and limestone, and where it is not weathered or deformed it has low porosity and low permeability. However, where it has been fractured by folding or faulting its permeability has been greatly increased. In many places these fractures have been enlarged by solution, and large openings have been formed. Sinkholes and caverns are common. These openings give rise locally to springs and wells. The porosity and permeability of the formation are unevenly distributed and have been localized by fracturing and later solution by groundwater movement.

The Chickamauga Limestone is composed of thin beds of shaly limestone and shale. The porosity is low. Fractures have been enlarged by solution but not to the extent of the Knox group. Fractures and solution channels form a network of open channels and voids that permit relatively free movement of groundwater.

The first five sites discussed in this report are within the Oak Ridge National Laboratory (ORNL) main complex, which is underlain by Chickamauga Limestone. However, the sixth and seventh sites are about 10 km northeast of the main ORNL complex at the Y-12 site which is underlain by Conasauga Shale.

2.2 Soils

Soils in the ORNL area are generally of the group known as ultisols, formerly called red-yellow podzolic and reddish-brown lateritic soils. Other soil types are found locally. These are mostly entisols, found in

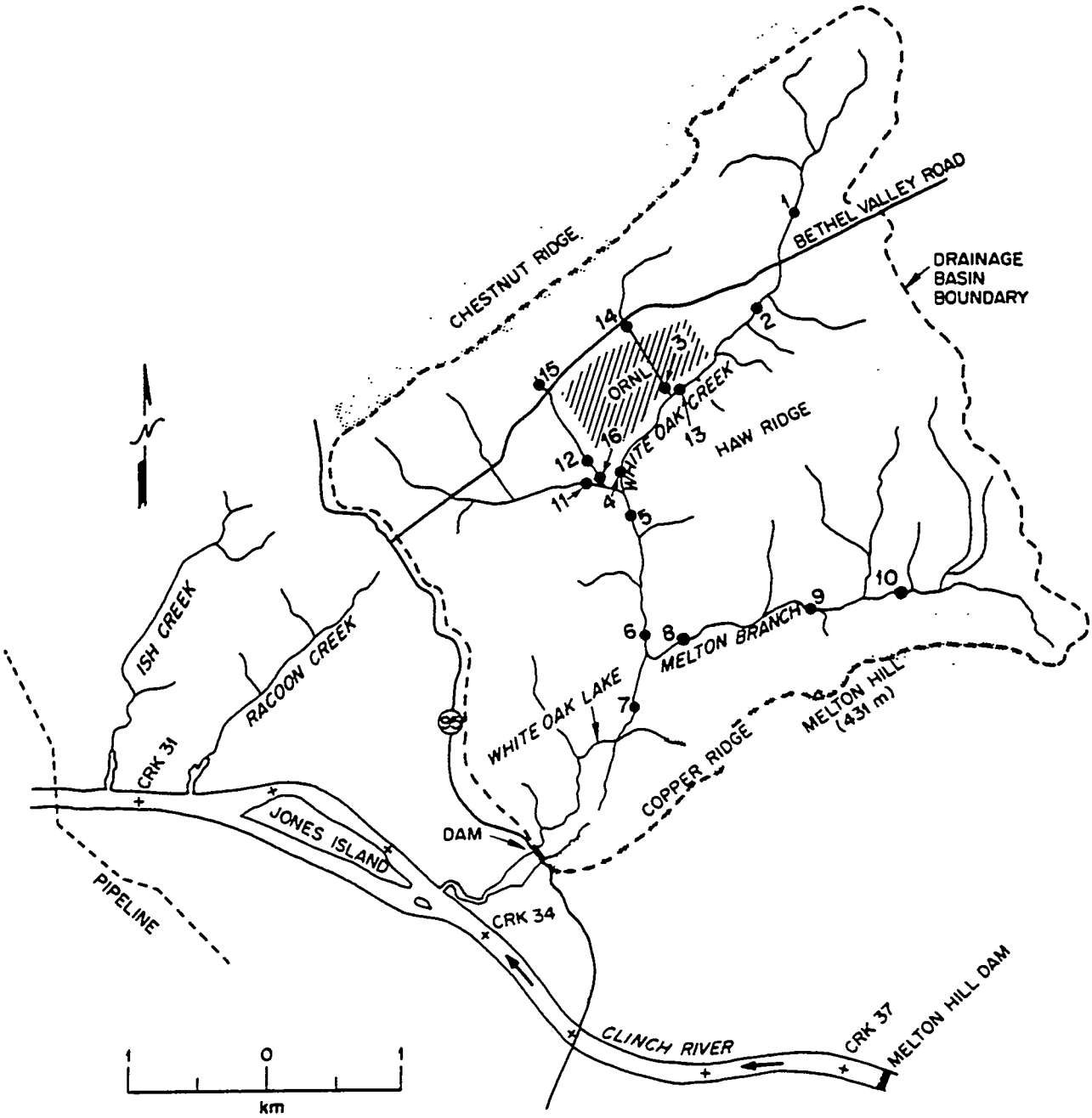
steeply sloping areas, or inceptisols found mainly in alluvial areas adjacent to streams. The ultisols develop in humid areas of temperate to tropical zones on old or highly weathered parent material under forest or savannah vegetation. In general, the soils in the ORNL area are moist, strongly leached, acidic, and low in organic matter. Soil depth in the vicinity of ORNL varies from about 0.15 to 5 m, and entrance of water into the soil can range from less than 0.5 cm/h to 25 cm/h. Soils derived from the Knox group contain kaolinite as their principal clay mineral, soils from the Conasauga group contain illite and vermiculite as principal clay minerals, and soils derived from the Chickamauga Limestone contain a mixture of kaolinitic and illitic minerals with some units having appreciable amounts of montmorillonitic clay minerals. Generally, illitic and vermiculitic clay minerals are more efficient in fixation of potassium and comparable ions into less available positions than are the kaolinitic materials.

2.3 Hydrology

Oak Ridge National Laboratory is located in the White Oak Creek watershed which drains an area of about 16.5 km^2. Melton Branch is the primary tributary, as seen in Figure 2. Surface runoff from the vicinity of research laboratory sites 1 – 5 (Table 2) flows into White Oak Creek or one of its tributaries. The waters of White Oak Creek are impounded by White Oak Dam which is located along the creek just to the northeast of Tennessee State Route 95. The resulting White Oak Lake is a small shallow impoundment. The water level is controlled by a vertical sluice gate that remains in a fixed position during normal operations. The normal lake level since 1960 has been 227 m above mean sea level. The normal surface

Figure 2. Map showing the White Oak Creek drainage basin and other creeks in the ORNL area in relation to the Clinch River

ORNL-DWG 80-15923R



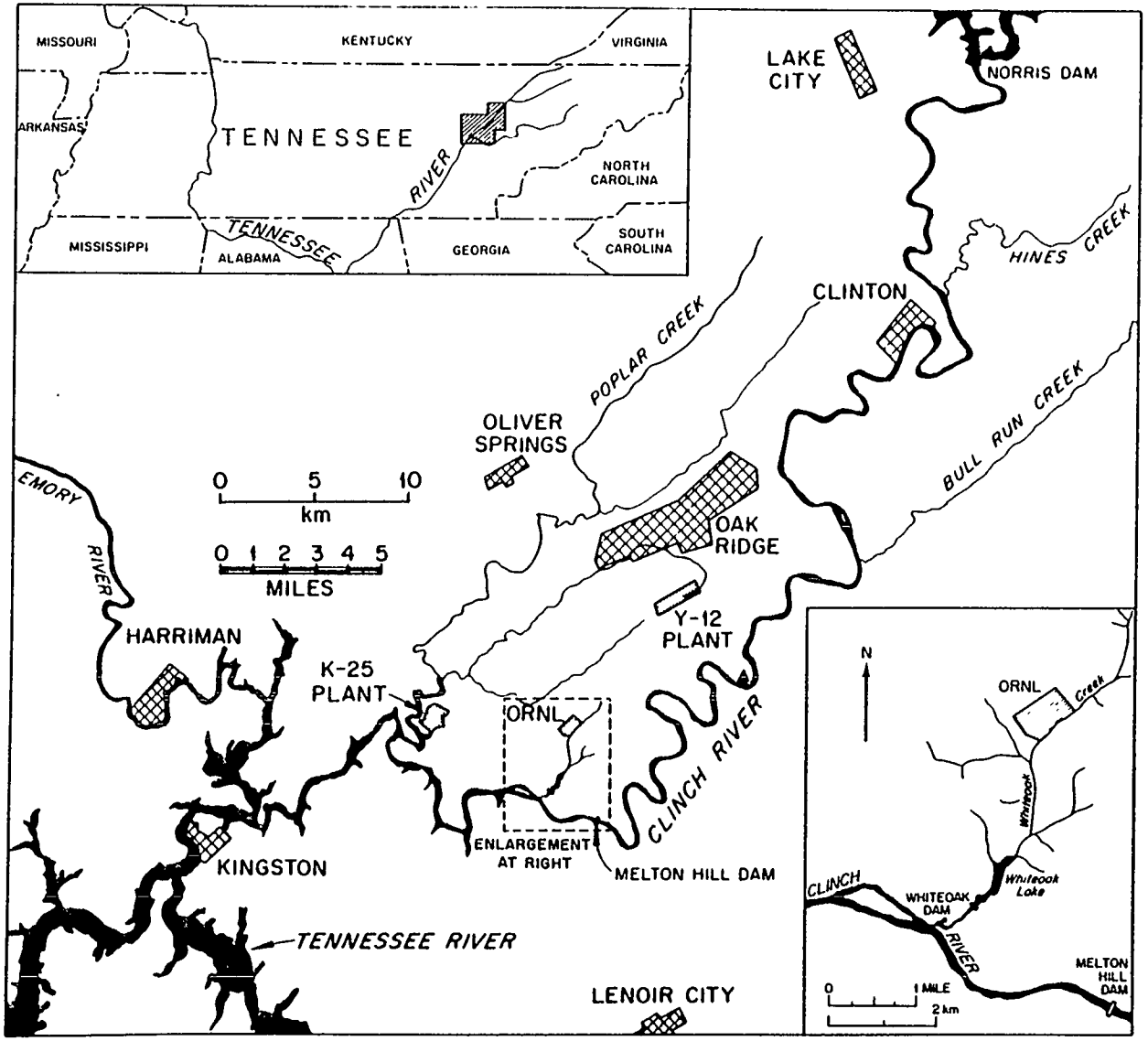
area of White Oak Lake is about 10 ha, and the retention time is about two days. Water that drains from White Oak Lake enters the Clinch River and subsequently the Tennessee River. The Clinch River originates in southwestern Virginia and drains approximately 11,340 km^2. Flow in this system is controlled by the Tennessee Valley Authority with three dams. The nearest upstream dam from the mouth of White Oak Creek is the Melton Hill Dam, approximately 3.5 km upstream. Water level from White Oak Dam to the mouth of White Oak Creek is strongly influenced by the level of the Clinch River which is, in turn, influenced by the pulsating pattern of water witholding and discharge at Melton Hill Dam. Watts Bar Dam on the Tennessee River creates backwaters that can affect White Oak Creek by raising the water level between the mouth of the creek and White Oak Dam. This normally occurs during the summer months (April-October) when the Watts Bar Reservoir pool surface is maintained at a higher elevation. During the winter the area between the mouth of White Oak Creek and White Oak Dam often resembles a mudflat.

Surface flow from the Y-12 facilities enters East Fork Poplar Creek which flows northward through the city of Oak Ridge and then southwestward to join the main branch of Poplar Creek about 12 km southwest (straight-line distance) of the Y-12 site. Poplar Creek then meanders through the K-25 site and into the Clinch River about 18 km downstream from the Melton Hill Dam (Figure 3).

Water quality of surface streams on the Oak Ridge Reservation is, generally of the calcium-magnesium-bicarbonate type, reflecting the limestone geology of the area. Water quality and ecology of many streams in the area have, however, been altered as a result of industrial effluents.

Figure 3. Map showing rivers and streams in the Oak Ridge area

ORNL-DWG 82-14318



There are nine public water supply systems that withdraw surface water from within an approximately 32 km (20 mi) radius of Oak Ridge National Laboratory. Only one of these is downstream of White Oak Dam. The intake for the city of Kingston is located along the Tennessee River about 0.6 river km above the confluence of the Clinch and Tennessee rivers and about 34 river km below White Oak Dam. Surface waters in the vicinity of Oak Ridge National Laboratory, including the Clinch River, are important resources for research, recreation (boating, fishing, swimming), and municipal and industrial activities. Most of the surface waters near the research laboratories originate from contributing springs in the limestone on the ridge slopes.

Over 100 private wells and springs and 8 industrial and 16 public groundwater supplies exist within about 32 km of Oak Ridge National Laboratory. However, due to the stratigraphic and structural control of groundwater flow in the region, groundwater beneath the sites is expected to migrate along the strike and to discharge into surface water bodies. There is a low probability of groundwater migration from Laboratory facilities to offsite wells. Further details of surface- and groundwater withdrawls within about a 32 km (20 mi) radius of ORNL can be found in Boyle et. al. 1982.

2.4 Climatology

The climate of the Oak Ridge area may be broadly classified as humid continental. The Cumberland Mountains to the northwest and the Great Smoky Mountains to the southeast both influence the climate of the area around ORNL. The Cumberland Mountains serve to retard the flow of cold winter air that often penetrates far south over the plains and prairies in

the Central United States. The prevailing surface winds usually blow up the valley from the southwest or down the valley from the northeast. Local topography reduces wind speed substantially. Severe wind storms and tornadoes are rare in the Oak Ridge area which has one of the lowest tornado probabilities in the state. Air stagnation, however, is relatively common. Periods of air stagnation that have the potential for being multi-day air-pollution episodes occur frequently in eastern Tennessee, relative to most other areas of the eastern United States. About two such episodes occur per year, on the average, to cover an average of about 8 days per year (Korshover 1976).

The approximately 140 cm of precipitation that falls in an average year is distributed quite equally over the months, relative to other regions in the eastern United States. The driest month is October with an average of about 7 cm of precipitation. July and the winter months are the wettest, averaging about 15 cm. Monthly total precipitation amounts exceeding 25 cm can occur in any season of the year, though such extremes are least likely in the autumn. The largest amount ever recorded in October was about 16 cm. At least a trace of snow has been reported in each winter of record. Amounts in individual winters have ranged from only a trace to over one meter. The average amount is about 26 cm, or about 26 mm water equivalent. Snow from any single storm will usually melt in a few days. Episodes of freezing rain are not uncommon during winter, especially when cold air is trapped in the valley and relatively lighter (warm and moist) air from the Gulf of Mexico is advected at higher elevations. Thus, it may be raining in the Cumberland or Great Smoky Mountains while freezing rain is occurring in the valley between.

Temperatures seldom rise much above 35°C or drop below 0°C. However on January 21, 1985, the temperature dropped to −27.7°C to provide the record minimum. The record maximum is 40.6°C.

2.5 Ecology

The ecological systems around Oak Ridge National laboratory are typical for the Appalachian region. The area was primarily agricultural land before federal acquisition in 1942, and much of the area has since reverted back to natural plant cover. The dominant plant community is oak-hickory forest, though elements of mixed mesophytic (buckeye, beech, maple, etc.) are also often present in coves and on sheltered slopes. Although most narrow, steep-sided, perennial stream drainages near ORNL have mixed mesophytic characteristics, the type is well developed and extensive only on a few steep north-facing slopes and bluffs. The laboratory area is a developed site surrounded by grasslands.

2.5.1 Terrestrial Flora

Over 800 species, subspecies, and varieties have been identified as present in the area. Approximately one-third of the area either has been planted in pine, or is in natural pine stands. In many areas the natural pine stands are being replaced by oaks, hickories backgum, sourwood, and occasional white pines. Some areas, such as pipelines, power-line corridors, and a few field research sites are maintained as open fields. No plants on the federal list of endangered species have been observed near the Laboratory. The nearest such species is the green pitcherplant (Sarracenia oreophila) which has been recorded about 65 km to the north. The state of Tennessee has separate lists of plant species that are

endangered, threatened, or of special concern within the state. Three plant species that occur near ORNL are listed by the state as endangered, threatened, or of special concern within the state. Three plant species that occur near ORNL are listed by the state as endangered and four are under review for inclusion on the federal list of endangered species (Parr 1984). These plants occur in the general vicinity of the Laboratory, but are in areas designated as natural areas within the Oak Ridge National Environmental Research Park and are not affected by activities in the developed areas around the experimental reactor sites.

2.5.2 Fauna

There are seven federally listed endangered species that have ranges encompassing the Laboratory or for which suitable habitat may exist near the laboratory. The bald eagle is the only endangered species that has been conclusively recorded in the area. Most of these bald eagles are probably of the northern races that are listed as threatened rather than endangered, but eagles seen in summer may originate from the endangered breeding population in Florida, where nesting occurs in winter. No eagles are known to nest near the Laboratory. In addition to the federal list of endangered animal species, the Tennessee Wildlife Resources Agency lists five species as being endangered within the state. None of these species appear to have been significantly affected by activities of the Laboratory. Detailed information on endangered species in and around the Laboratory is given by Boyle et al. (1982).

2.5.3 Aquatic Ecology

A survey of the aquatic ecology around ORNL was carried out during parts of 1979 and 1980 by Loar et al. (1981). Much of what follows is

based on their report and supplemented by subsequent observations.

The upper part of White Oak Creek, above the ORNL site, is believed to resemble conditions before construction of ORNL. There were 24 taxa of benthic invertebrates reported, which represented the greatest diversity of species anywhere in the creek. Fish identified were mostly stoneroller and blacknose dace, which are typical of small southeastern streams.

Between ORNL and White Oak Lake the benthic invertebrates were dominated by midge larvae. Bluegill and mosquitofish were found, but there was little or no evidence of spawning and no fish were found at one of the sampling locations. Also, no fish were found in the Melton Branch.

In White Oak Lake, the benthic invertebrates were mostly midge larvae. Fish included bluegill, sunfish, mosquitofish, and largemouth bass. Other species are also believed present.

Between White Oak Lake and the Clinch River the benthic invertebrates are still dominated by midge larvae, but the asiatic clam is also present as it is in the Clinch River. The abundance of fish eggs strongly suggests that this is a spawning area. The fish population somewhat resembles that of the Clinch River and includes gizzard shad, and channel catfish, with lesser abundances of sauger and bluegill.

The benthic invertebrates of the Clinch River near ORNL are usually dominated by midge larvae, though aquatic earthworms and asiatic clams are occasionally dominant. Gizzard shad were the most abundant of the 21 species of fish identified. Several popular sport fishes were also identified. The Clinch River near White Oak Creek appears diverse, productive, and free of excessive loadings of nutrients or organic material.

East Fork Poplar Creek has received loadings from a variety of pollutants and other material resulting from municipal and industrial activities in Oak Ridge and the nearby Y-12 plant. This may at least partially account for the low number of benthic invertebrate species observed in 1975 (ERDA 1975). However, recent water conservation procedures (e.g., the implementation of zero discharge by the Oak Ridge water treatment plant in 1978) may lead to some observed increases in benthic invertebrate species when a recently proposed biological monitoring plan is implemented (Loar et al. 1985).

The Tennessee Valley Authority (TVA) has recently carried out some fish sampling and analyses in East Fork Poplar Creek and nearby Bear Creek (TVA 1985). A total of 241 fish, including 18 species and one hybrid sunfish were collected. Only two species (redbreast sunfish and bluegill) were collected from all four stations. Relative abundance estimates of fish (number/sample run) in 100 m sections of East Fork Poplar Creek suggested greater densities in the upper reaches of the stream. The stations at miles 8.8 and 13.8 had significantly ($P > 0.01$) greater numbers of fish than at mile 4.0. Normally, fish densities are much greater near the mouth of a stream and decrease in an upstream direction to a low in the headwater areas. The West End Sewage Treatment Plant could be affecting the distribution and abundance of fish in East Fork Poplar Creek. Fish densities at a sampling station along a nearby creek (Bear Creek) were considerably higher than at any of the East Fork Poplar Creek stations. A total of 24 species of fish were collected from East Fork Poplar Creek sites. No darter species and only a limited number of minnow species were found.

2.5.4 Establishment of Wildlife Management Area

The Oak Ridge Wildlife Management Area has been established on the reservation as the result of the Cooperative Agreement Between the Department of Energy and the Tennessee Wildlife Resources Agency for the establishment of a Wildlife Management Area, November 30, 1984. This agreement enables DOE and the Tennessee Wildlife Resources Agency to work together in areas of wildlife management and research. Results of this program include the establishment of a series of organized deer hunts and the release of several wild turkeys as part of an effort to establish turkey populations within the state.

3. REGULATORY STATUS AND HAZARD RANKING

Hazard Ranking System analyses for the research laboratory facilities, covered in this report, have not been performed because these facilities are regulated under the Department of Energy Order 5820.2 which has no provisions for hazard ranking.

4. INVENTORY OF THE RESEARCH LABORATORY FACILITIES

Of the Remedial Action Program sites, relatively little is known about the research laboratories (Category 11, Table 1). However, many of these labs are housed in buildings that are currently used for various other activities, so routine radiological monitoring is often necessary. Scoping surveys have been prepared for each site as a first step in acquiring more information for the Remedial Action Program. Results of these surveys have been incorporated into this report. The following sections provide a summary of the history and use of each facility, radiological characteristics, presence and nature of any environmental

concerns, any other risk considerations noted, areas requiring further study, and conclusions relevant to decommissioning drawn from the survey of each site. The names of persons knowledgeable about each site are also listed. These people can usually be quite helpful to anyone interested in visiting a site and they can usually provide very detailed and up-to-date information about their respective sites.

Up-to-date information on administrative matters for these and other SCFP sites can be provided by T. W. (Tom) Burwinkle (615/576-0423). Other information presented was obtained from phone conversations with persons listed below as being knowledgeable about the respective sites. Two sites originally listed as Remedial Action Program sites (the Radiochemical Processing Facility and Cell D of the High-Level Radiochemical Laboratory) are still in use and have been deleted from the current list. Site-by-site summaries for the remaining seven facilities are given below.

4.1 High Level Chemical Development Laboratory

4.1.1 Location

This facility is located in Building 4507 of the main ORNL complex. Its location within that complex is shown on the map in Figure 4. A photo of the Building is shown as Figure 5.

4.1.2. History and Use of the Facility

The entire Building (4507) is considered in this survey. It was constructed in 1957 to be used as a small-scale pilot plant laboratory for research on reactor fuel reprocessing. This facility has four hot cells and associated equipment for handling highly irradiated alpha-beta-gamma sources on the ground level and a chemical make-up area on the second

Figure 4. Location of Building 4507 and emergency evacuation route

ORNL DWG 73-8623

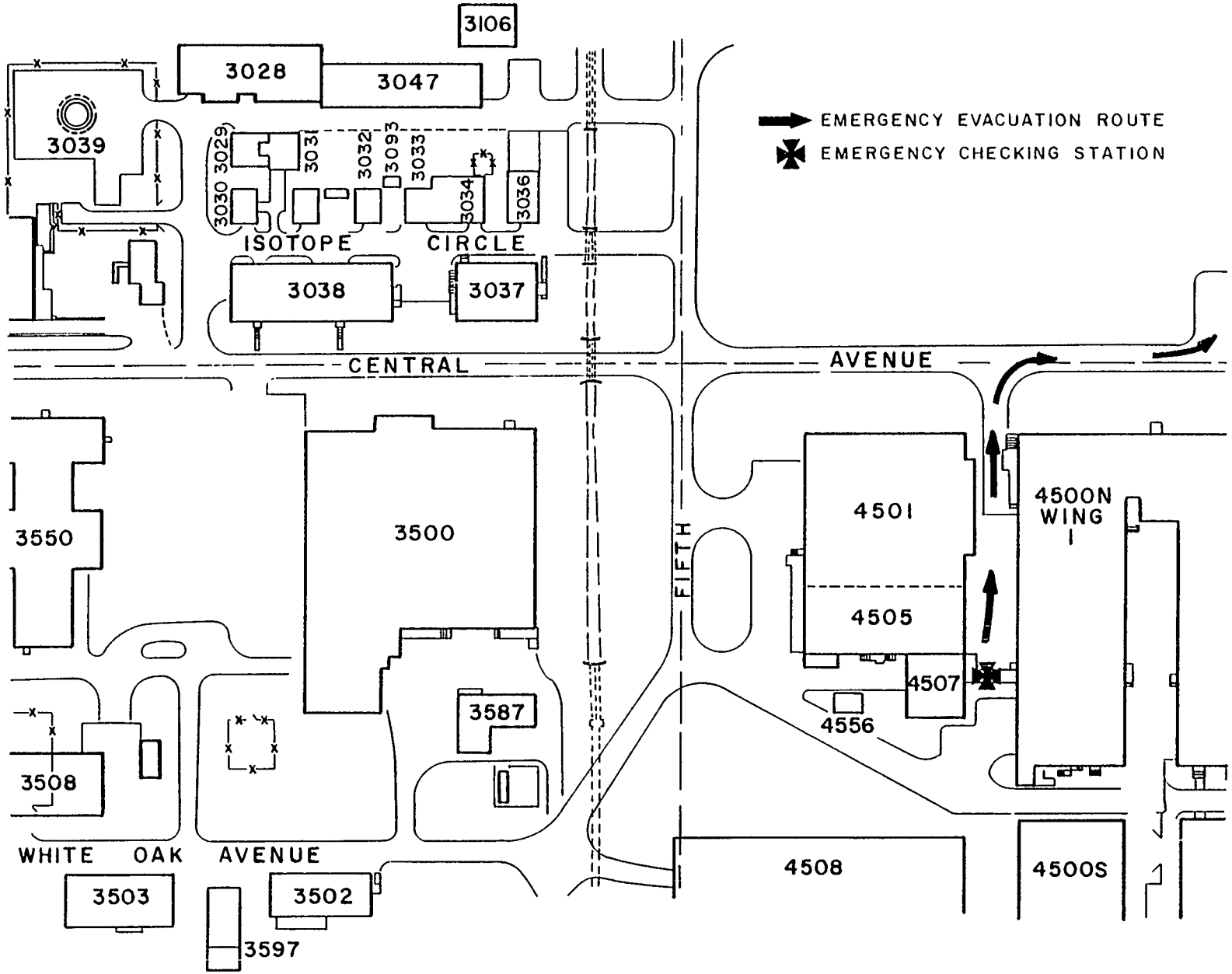
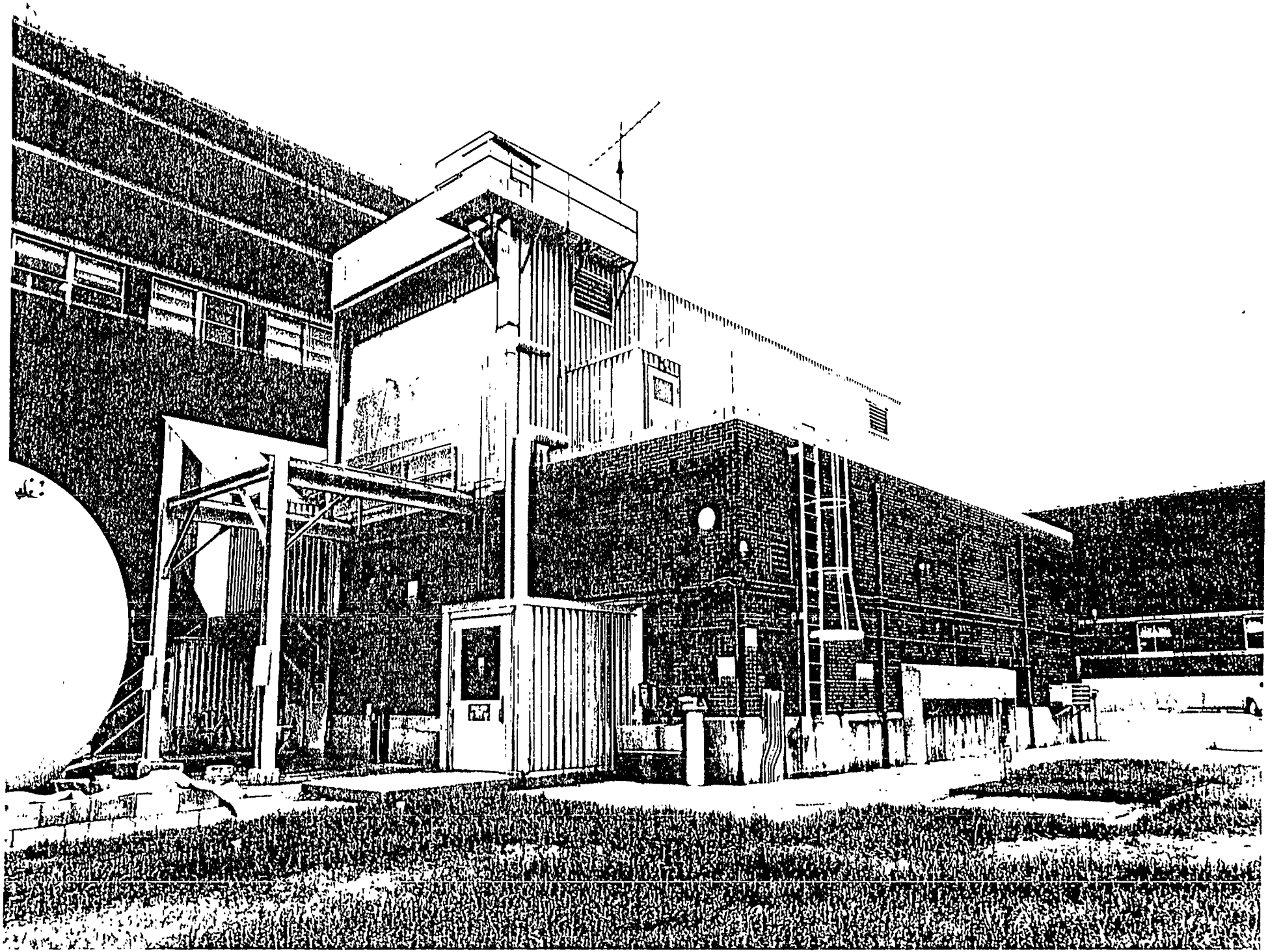


Figure 5. Building 4507 with Building 4505
in the left background and Building
4500 in the right background



level. The building is also equipped with a penthouse superstructure above the hot cells which contains glove-boxes, a shielded manipulator cave for small scale work, and a gantry crane for handling shielded casks. Schematic plans for Building 4507 are shown in Figures 6 and 7.

The facilities in Bldg. 4507 have been used in development studies on fuels, separation and recovery of transuranic isotopes, and separation of fission products during the 1960s and 1970s. Research activities were terminated in 1980 and the building subsequently placed in it's present "mothballed" condition.

4.1.3 Radiological Characterization

Although the building is presently inactive, some air-monitoring is routinely conducted and periodic sampling for transferable contamination is performed. External surfaces of the cells and operating equipment are relatively free of transferable contamination with levels being less than 20 d/m/100 cm^2 alpha or 200 d/m/100 cm^2 beta-gamma. Radiation probe readings at the external surfaces of the cells were less than 0.5 mR/h. There is one area on the ground floor 20 cm x 20 cm located near the east wall which is contaminated to a level of 10^4d/m/100 cm^2 alpha. This area is appropriately painted and a notification sign placed on top of the area. Several of the valves and service piping are contaminated internally to levels of 5,000 d/m/100 cm^2 alpha and 1500 d/m/100 cm^2 beta-gamma. This equipment is appropriately tagged. Although the interior of the hot cells could not be surveyed, contamination levels greater than 2.5 x 10^8 d/m/100 cm^2 alpha can be reasonably

Figure 6. Schematic ventilation plan for Building 4507 ventilation system

ORNL DWG 73-8627

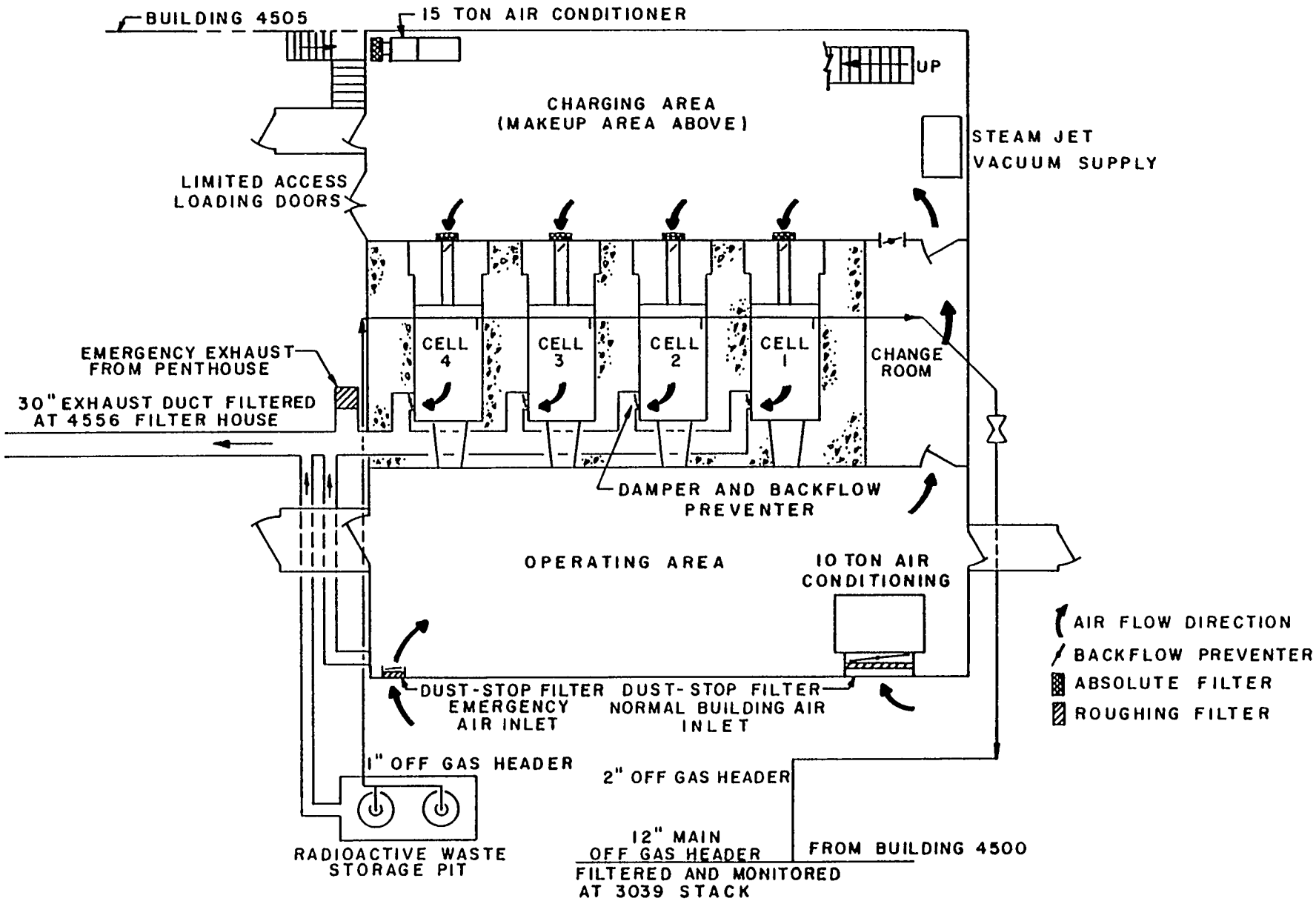
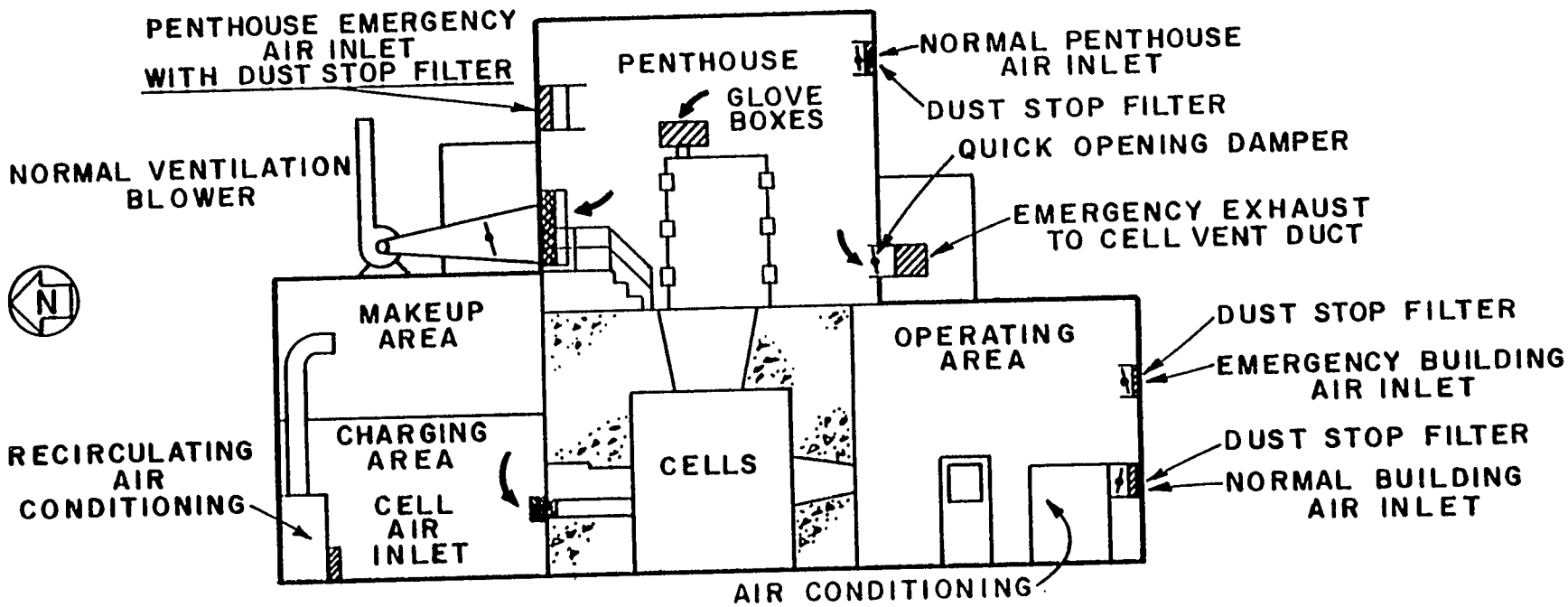


Figure 7. Schematic elevation of Building 4507 ventilation system

ORNL DWG 73-8622



anticipated. The penthouse is classified as a contaminated zone due to an accident several years ago. However, the alpha contamination is covered with paint and surveys have not revealed any exposed contamination in recent years.

The majority of the work in this facility has involved transuranic elements. Therefore appropriate decontamination procedures will need to be used.

4.1.4 Environmental Concerns

Building 4507 is maintained at a slight negative pressure relative to the atmosphere along with appropriate air locks. Therefore any airborne contaminants should remain contained. The building and associated equipment appear to be in good condition. Assuming appropriate decontamination procedures are used, no environmental concerns were identified.

4.1.5 Other Risk Considerations

Other than the risks involved in the decontamination of the cells, no significant risks were identified.

4.1.6 Areas Requiring Further Study

Once decontamination/decommissioning activities begin in Building 4507 it will be necessary to do an in-depth radiation survey within the cells to accurately characterize the radioactivity contained within.

4.1.7 Summary and Conclusions

The decontamination/decommissioning of Building 4507 should not

present any unusual problems. Radiation levels within the cells are relatively high, but similar situations have been addressed on previous occasions at ORNL. The building structure, cells and associated equipment all appear to be in good condition and the radiation is contained thereby reducing the urgency for decontamination.

4.1.8: Persons Knowledgeable about the Site:

R. G. (Bob) Stacy, ORNL, 615/574-7071

D. G. Noe, ORNL, 615/574-6710

4.2: High Level Radiation Analytical Laboratory

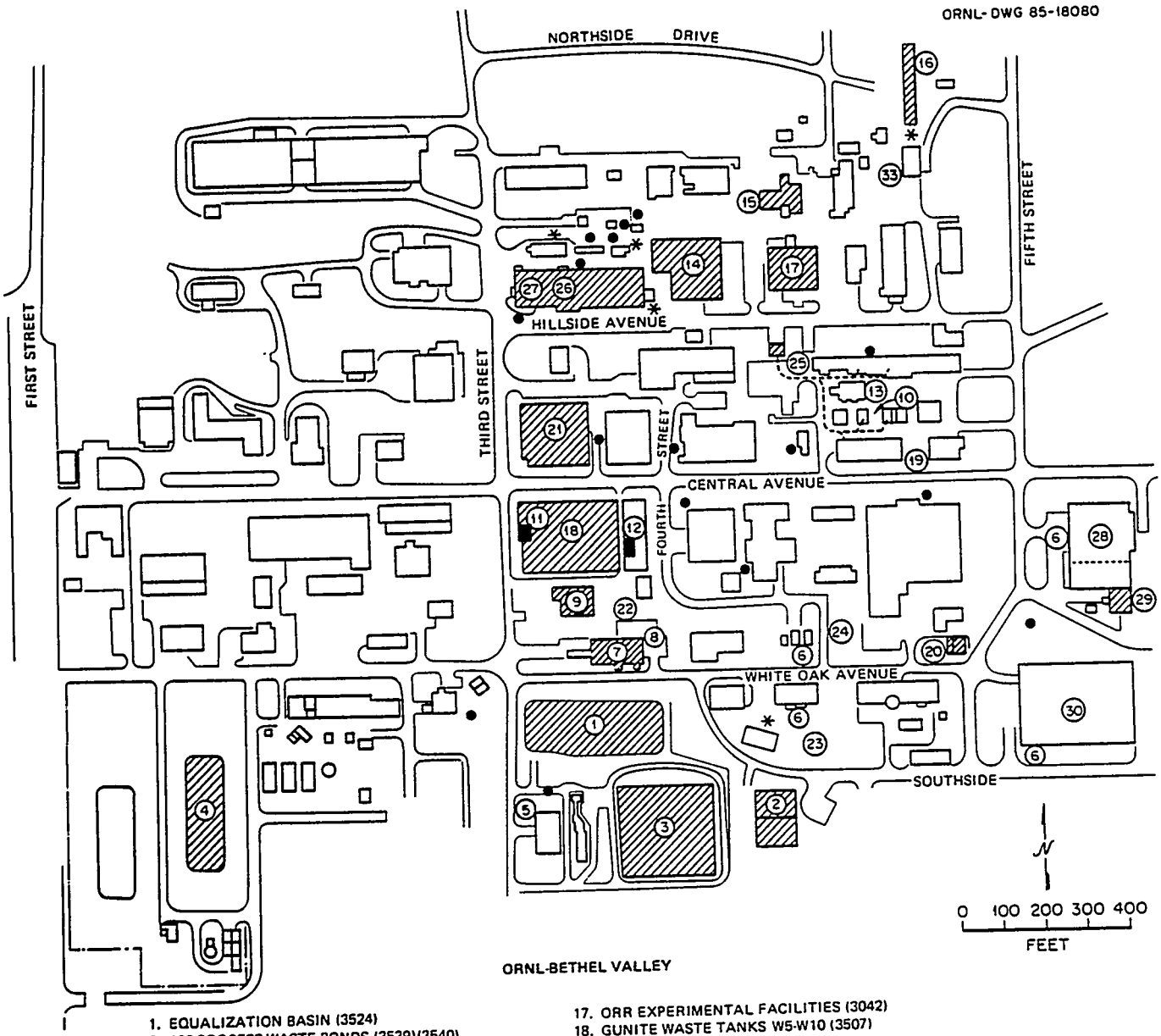
4.2.1 Location:

This facility is located in Building 3019-B of the main ORNL complex, and is site No. 27 near the top (north) of the map in Figure 8.

4.2.2 History and Use of the Facility

This facility was constructed in 1955 to analyze highly radioactive materials in support of existing and projected ORNL programs. The facility contains a hot-cell bank which consists of a storage cell and seven analytical cells. Figure 9 shows a floor plan of this area. Figure 10 is a photograph of the area, looking south. Casks with highly radioactive samples were brought into the storage cell on an electrically powered cart where samples were unloaded. Solid radioactive wastes were removed from the storage cell via this cart also. Samples were distributed via an inter-cell conveyor to the seven analytical cells for processing and analysis. Hoods in the cell operating area were used largely for storage and limited analysis of low level samples and preparation of uncontaminated agents. An electrically driven fan discharging to an adjacent stack (designated as Building 3020)

Figure 8. Locations of some remedial action sites at ORNL.



ORNL-BETHEL VALLEY

1. EQUALIZATION BASIN (3524)
2. 190 PROCESS WASTE PONDS (3539)(3540)
3. WASTE HOLDING BASIN (3513)
4. SEWAGE LAGOON, EAST BASIN
5. ABANDONED WASTE BASIN (FILLED-IN) (3512)
6. MERCURY CONTAMINATED SOIL
7. FISSION PRODUCT DEVELOPMENT LAB (3517)
8. FPDL FILTER PIT/LLW TRANSFER LINE (3517)
9. METAL RECOVERY FACILITY (3505)
10. STORAGE GARDEN 3033
11. WASTE EVAPORATOR FACILITY (3506)
12. FISSION PRODUCT PILOT PLANT (3515)
13. COBALT-60 STORAGE GARDEN (3029)
14. ORNL GRAPHITE REACTOR (3001)
15. LOW INTENSITY TEST REACTOR (3005)
16. ORR WATER-TO-AIR HEAT EXCHANGER (3087)
17. ORR EXPERIMENTAL FACILITIES (3042)
18. GUNITE WASTE TANKS W5-W10 (3507)
19. WASTE TANK WC-1
20. WASTE TANKS WC-15, WC-17
21. WASTE TANKS W1-W4, W13-W15, W1-A (3023)
22. WASTE TANK W-11
23. WASTE TANKS TH1-TH3
24. WASTE TANK TH4
25. ISOTOPES DUCTWORK/3110 FILTER HOUSE
26. RADIOCHEMICAL PROCESSING LAB (3019)
27. HIGH LEVEL RADIOCHEMICAL ANALYTICAL LAB (3019B)
28. HIGH LEVEL RADIOCHEMICAL LAB (4501-CELL D)
29. HIGH LEVEL CHEMICAL DEVELOPMENT LAB (4507)
30. REMOTE COATING FURNACE LOOP/CERAMIC PROCESSING LAB (4508)
31. LLW LINE LEAK SITES ●
32. OTHER CONTAMINATED SITES *
33. LITR POND (3075)

Figure 9. Floor plan showing the hot cell area in Building 3019-B

ORNL-DWG. 74-5163

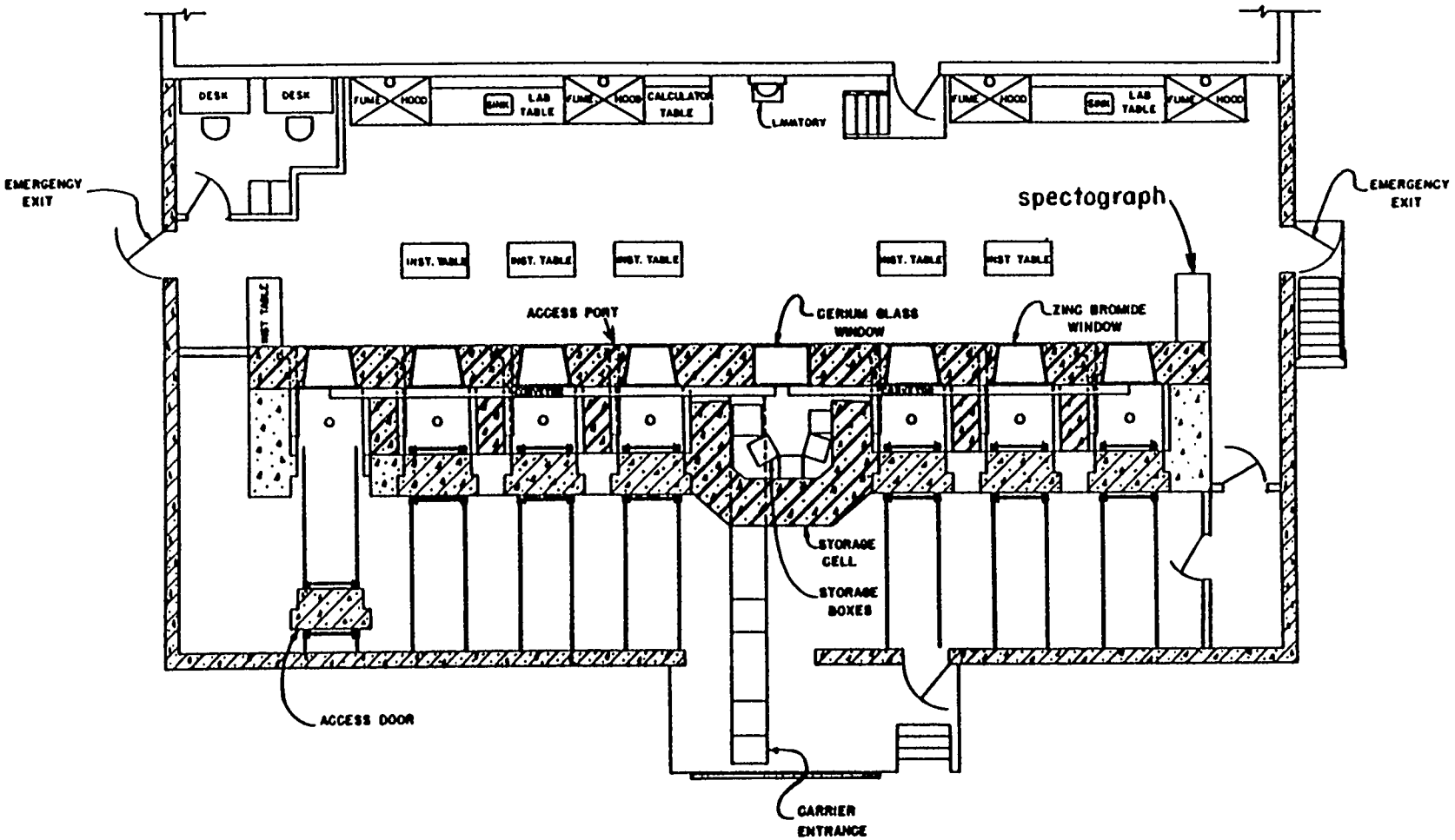
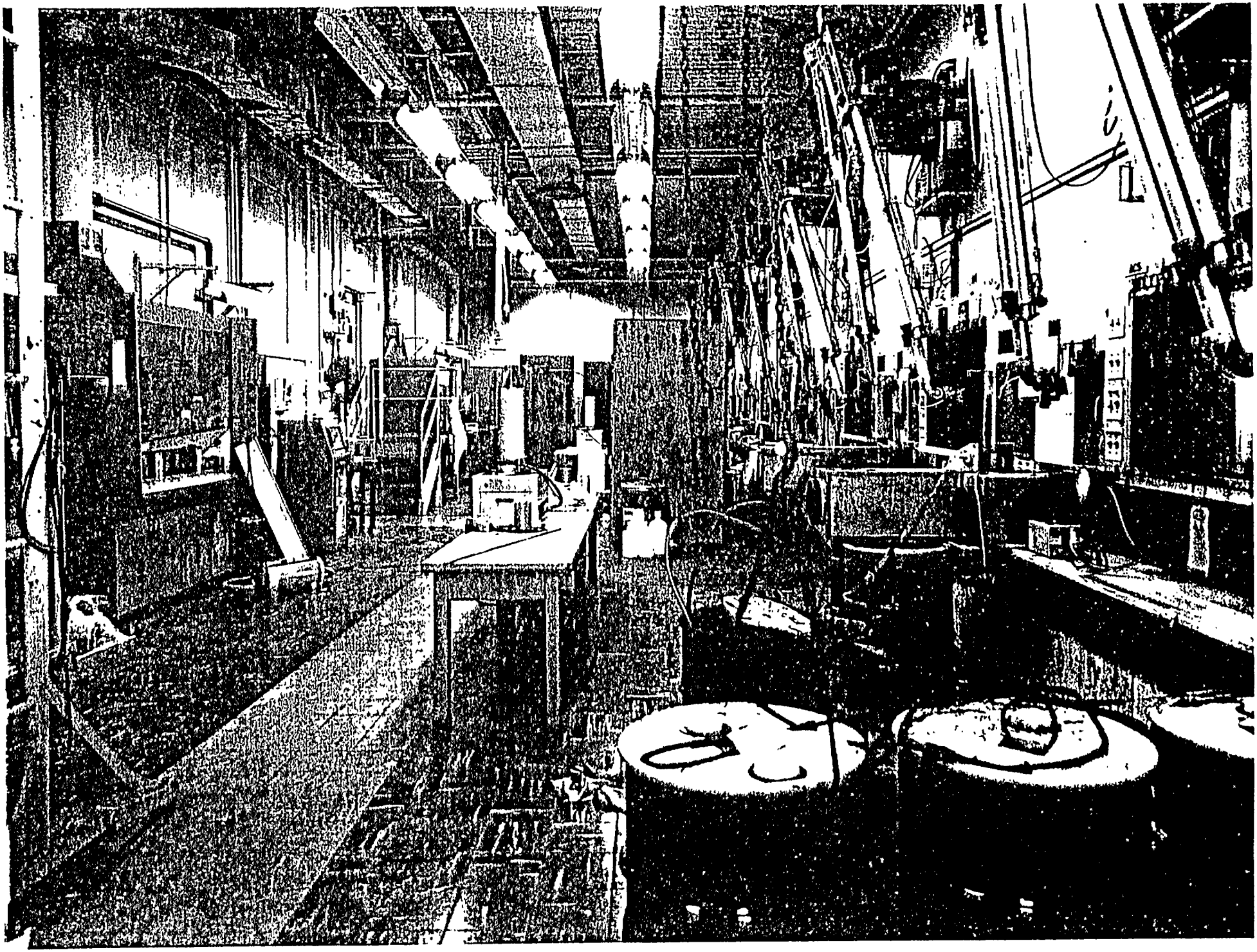


Figure 10. Photograph of the hot-cell area in Building 3019-B, looking south



exhausts air from the cells and hoods via ductwork routed through the Building 3108 filter pit also adjacent to the stack. Until 1965, when an alternative facility was constructed in Building 2026, 3019-B was used extensively for processing highly radioactive samples containing uranium, plutonium, trans-plutonium elements, activation products and fission products. After startup of the other facility in Building 2026, use of 3019-B was sporadic and it has been inactive since 1980.

4.2.3 Radiological Characterization

The cells, access tunnel to the storage cell, cell access drawers and drawer containment boxes, cell exhaust ducts, and miscellaneous equipment and materials within the cells are grossly contaminated with long-lived fission products and alpha emitters. Also, there are present about a dozen five-gallon cans filled with highly gamma-active solid waste (Cesium-137, spent fuel, etc.). Maximum levels with the cells and on materials and equipment in the cells are probably in excess of 100 rad/h and 10^6 d/m/100cm^2. Most of the levels associated with these areas are estimates based on the history of the facility and prior surveys, as these areas are not readily accessible for monitoring.

Visibility through all cell windows is impaired and is essentially opaque for cells 1, 2, and 3. The inner pane of the window cell 2 is broken and zinc bromide is leaking out, resulting in a radiation field of 50 mR/h at the outer pane. Approximately half of the zinc bromide was transferred from the cell window resulting in a radiation field of 35 mR/h at the outer pane.

Over the lifetime of the facility there have been several leaks and spills resulting in some contamination in the cell access area (room 14)

and the cell operating area (room 15). Paint bonding has been applied to some surfaces. Current migration from some cells, due to past operations, continues to result in transferable contamination or localized contamination in these areas. Accessible surfaces of hoods in room 15 generally are less than 20 d/m/100cm^2 alpha or 200 d/m/100cm^2 beta-gamma. Somewhat higher levels are anticipated in the hood exhaust plenums and ducts and under paint or tape bonding.

Table 3 provides a summary of radiological status of this facility.

4.2.4 Environmental Concerns

The exhaust ventilation system maintains the cells and hoods at a greater negative pressure than the cell operating and access areas. Also the cell access and operating areas are slightly negative with respect to atmosphere. Therefore environmental concerns relative to airborne contaminants are minimal at present. However, with a few exceptions, air exhausted from hoods and cells is not filtered prior to discharge to the duct leading to Building 3108 filter pit. The interior of this duct is grossly contaminated. If the duct integrity is not maintained, radioactive materials could leak from the duct onto the roof or ground with further dispersal by wind and rain. Failure of a cell window could result in hundreds of gallons of zinc bromide being discharged into the process or LLW waste drain systems.

4.2.5 Other Risk Considerations

Asbestos insulation has probably been used on steam and water lines associated with the cells. Also, zinc bromide is slowly leaking from some of the cell windows and presents a chemical hazard to workers and corrosive impacts to anything it touches. On the underside of one

Table 3. Radiological Status of the High Level

Radiation Analytical Laboratory,

November 1985

Table 3. Radiological Status of the High Level Radiation Analytical Laboratory, November 1985

	SURFACE CONTAMINATION (MAXIMUM)				RADIATION FIELD
	TRANSFERABLE d/m/100cm²		INSTRUMENT READINGS		mR/h
	ALPHA	BETA-GAMMA	ALPHA/dm/100cm²	mrad/hr	
CELLS (INTERIOR)	> 10^5*	> 10^5*	> 10^6*	> 100 *	10^2 to > 10^5
CELL ACCESS TUNNEL	> 10^4*	> 10^5*	> 10^5	> 10^4	10^2 to > 10^3
Room 14 (Floor Level)	10^3	9.6×10^4	10^4**	140	0.2 to 5
Room 14 (top of Cells & at Exhaust Duct System)	6×10^2	4.2×10^3	-	-	5 to 2.2×10^3
Room 15 (at cells)	90	9.6×10^4	1.2×10^4**	150	0.2 to 35
Room 15 (at hoods)	2×10^3 (SPOT)	< 2×10^2	10^6 (SPOT)	< .5	0.1 to 0.5

*ESTIMATED

**POSSIBLY HIGHER UNDER PAINT/TAPE ON SOME SURFACES

of the cells, a bulge has developed and it appears about to rupture. Vinyl sheet covering is peeling off noticeably in many cell areas.

4.2.6 Areas Requiring Further Study

Radiation and contamination levels in the cells are estimates. Access to the cells for monitoring purposes would require removal of materials, extensive remote decontamination and then construction of containment "tents" before the rear cell shielding doors could be opened. Some areas also exist where contamination is covered by paint or tape bonding in the cell access and operating areas.

4.2.7 Summary and Conclusions

Decommissioning of the cells and equipment in this laboratory will require careful planning for contamination and personnel exposure control. Extensive decontamination of cells and cell equipment will be required after all viewing window visibility is improved. Maintenance of LLW drain service (now unavailable) or provision of an alternative LLW removal capability will be essential to carry out in-cell decontamination. Shielded containers will be required to transfer solid wastes and equipment from the cells to the solid waste storage areas. Some waste, particularly the cell 6 spectograph spark source exhaust system and filter will likely qualify as transuranic wastes and should be packaged accordingly. Cell walls and floors are unlined porous concrete or concrete block and will be difficult to decontaminate.

Decommissioning of the hoods and hood ducts in the cell operating area should be accomplished without difficulty using appropriate contamination control and personnel protective procedures.

As a result of the physical condition of the cells and cell operating area, it is recommended this facility be given considerable priority when setting priorities for decommissioning.

4.2.8: Persons Knowledgeable about the Site

E. D. (Emory) Collins, ORNL, 615/574-6928

D. A. (Dan) Costanzo, ORNL, 615/574-4885

C. H. (Charlie) Miller, ORNL, 615/574-6703

Either of the first two persons listed above should be contacted before visiting this facility.

4.3: Remote Coating Furnace Loop

4.3.1: Location

This facility and associated support equipment is located in room 265A of Building 4508 of the main ORNL complex, identified as site 30 on the (east) of the map in Figure 8. The location of Room 265A is shown on the floor plan in Figure 11.

4.3.2 History and Use of the Facility

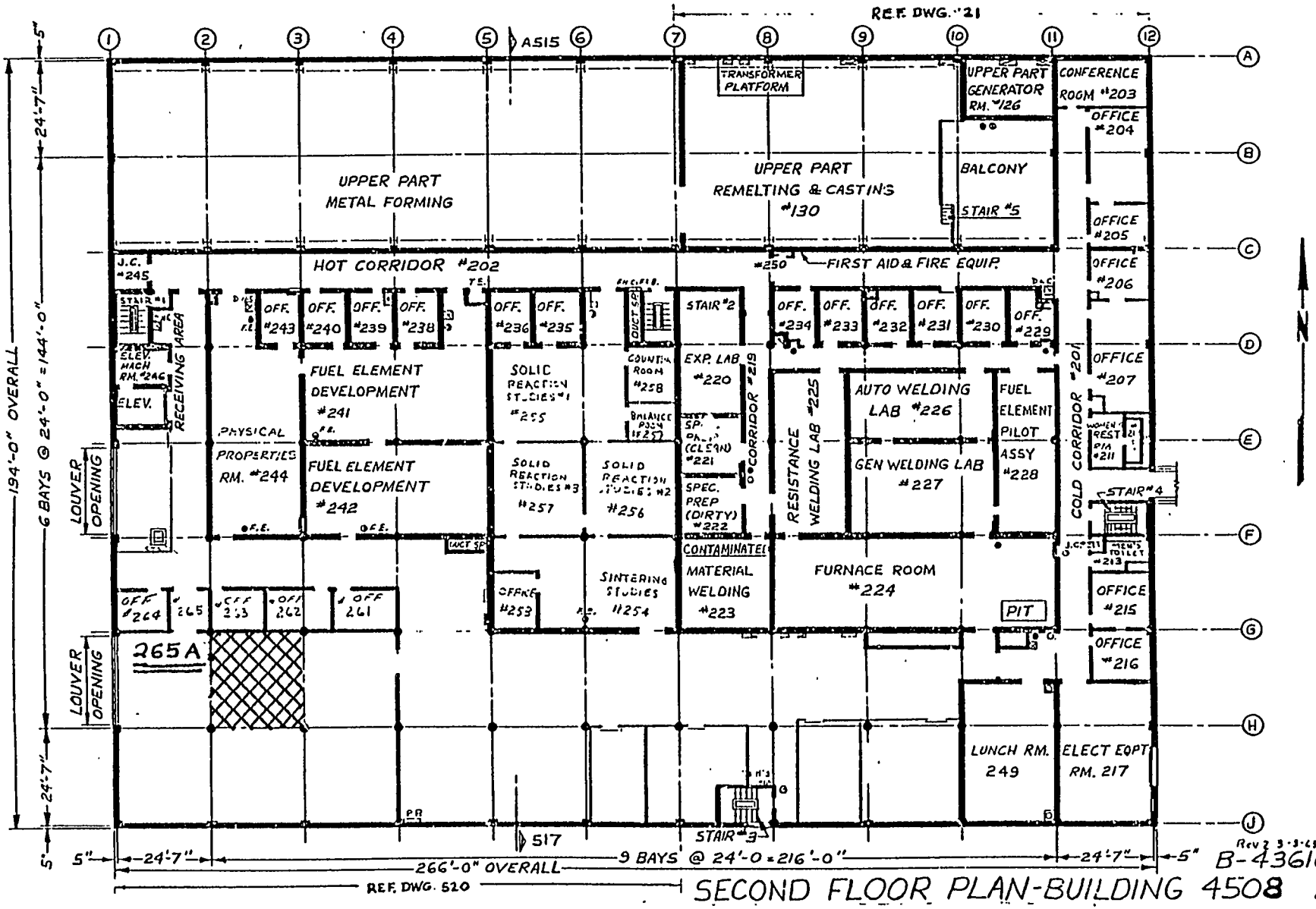
The Remote Coating Furnace Loop was initially operated in the late 1960s in support of the High Temperature Gas-Cooled Reactor (HTGR) Program to prepare coated-particle fuels for the HTGR fuel refabrication cycle. The use of this facility concluded in 1980.

Equipment associated with this facility, with the exception of the controls and instruments, is contained in a transparent plexiglas enclosure covering a floor area of approximately 58 m^2. The equipment is in good physical condition.

Figure 11. Second-floor plan of Building 4508

Crosshatching in the lower left portion

marks the area of interest



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SECOND FLOOR PLAN-BUILDING 4508

4.3.3 Radiological Characterization

Room 265A of the building is routinely surveyed for radioactivity and most of the surfaces are easily accessible therefore, an accurate projection of the present radioactive contamination can be made. The Remote Coating Furnace Loop was used to prepare HTGR fuel materials containing ^{238}U or ^{232}Th. Maximum transferable contamination levels within the enclosure are less than 5,000 d/m/100cm^2 alpha and the maximum background levels are less than 10 mR/h. Radiation measurements within the internals of the furnace loop were less than 6000 d/m 100cm^2 alpha. Standard decontamination procedures should be sufficient for appropriately handling the coating equipment.

4.3.4 Environmental Concerns

No environmental risks were identified in the decontamination and decommissioning activities of this facility.

4.3.5 Other Risk Considerations

No other risk was identified.

4.3.6 Areas Requiring Further Study

There were no areas identified as requiring further study.

4.3.7 Summary and Conclusions

Decommissioning of room 265A, Building 4508, should not present any unusual problems. Radiation levels are relatively low, and the equipment is accessible and in sound physical condition. There does not appear to be any urgency in decommissioning this facility due to the good condition of the equipment and adequacy of the shielding offered by the enclosure.

4.3.8 Persons Knowledgeable about the Site

D. P. (Dave) Stinton, ORNL, 615/574-4556.

R. E. (Ray) Keny, ORNL 615/574-4349

4.4 Ceramic Processing Laboratory

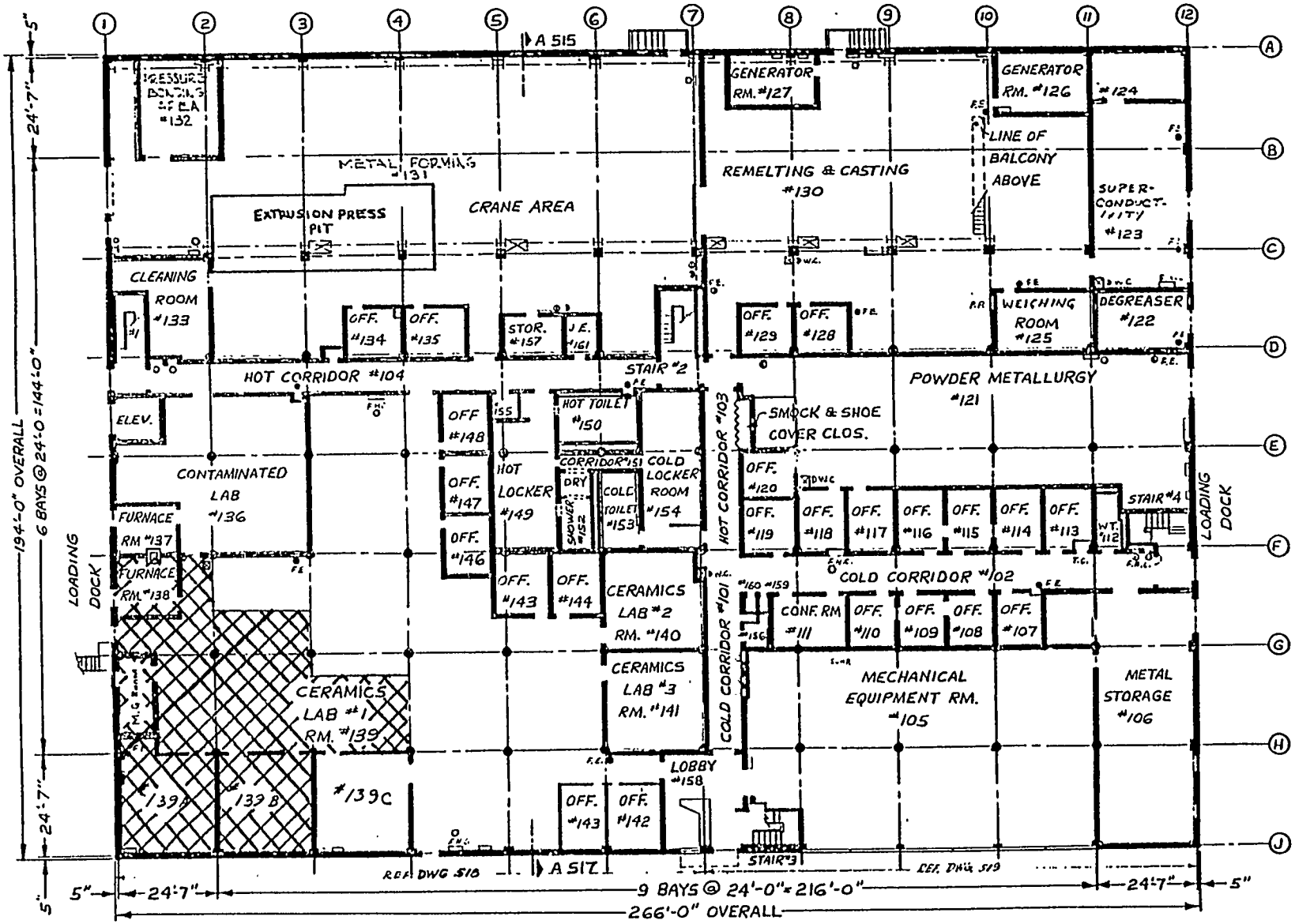
4.4.1 Location:

This facility is located in the Room 139 complex of Building 4508 of the main ORNL complex, and is jointly identified with the Remote Coating Furnace Loop as site No. 30 on the left (east) side of the map in Figure 8. This laboratory is a vaguely defined area, the boundaries of which are sometimes defined by storage cabinets, lab benches etc. Figure 12 is a first-floor plan of Building 4508, showing the location of the Ceramic Processing Laboratory.

4.4.2 History and Use of the Facility

This facility has been used for the Chemical Vapor Deposition (CVD) coating process in support of the High-Temperature Gas-Cooled Reactor (HTGR) Program. It was put into service in 1960 and is still used

Figure 12. First-floor plan of Building 4508. crosshatching in the lower left marks the area of interest.



FIRST FLOOR PLAN - BUILDING 4508

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periodically. It is expected to become surplus around January 1987. Uranium-233 and lesser quantities of Uranium-232 have been used for the coating process in the past. However, current activities do not involve the use of radioactive materials as was the case during the HTGR Program. The major activity performed in this facility was the coating of various types of fuel elements with uranium impregnated graphite. This process emphasized remote handling processes, using various types of laboratory equipment (e.g., blenders, coating furnaces, and balances) for handling graphite, both in powder and solid form.

4.4.3 Radiological Characterization

Room 139 is routinely surveyed for radioactivity, and most of the surfaces are easily accessible. Therefore, accurate radiation levels can be obtained. Although much of the equipment has been decontaminated on occasions, recent surveys indicate that measurable levels of radioactivity presently exist. The isotope ^{233}U was used during the coating process along with lesser quantities of ^{232}U. At ORNL, waste contaminated with ^{233}U is handled as if it were transuranic. However, the measured levels of radioactivity appear to be well below the minimum for transuranic waste [3.7×10^6 Bq/kg (100 nCi/g)]; therefore, low-level waste procedures will most likely apply.

Radiation surveys indicate maximum transferable contamination levels to be less than 1,000 d/m/100cm^2 alpha and maximum radiation levels to be less than 5,000 d/m/100cm^2 beta-gamma.

There are also several small lead "pigs" (approximately 7.5 cm x 15 cm) stored in a hood in room 139 which contain small thorium/uranium impregnated graphite sources. The maximum unshielded radiation levels of these sources is less than 2 R/h.

There is also one piece of equipment in this area (a press) which was contaminated with nontransferable alpha radiation. The press was subsequently painted and appropriately marked to identify the shielded contamination. No measurable radiation can be detected on the press now; however, the components will need to be treated as contaminated during disposal activities.

4.4.4 Environmental Concerns

Three areas of concern were identified during a review of this facility. (1) A power supply (capacitor) is located in Room 139 which contains polychlorinated biphenyls (PCBs) and will require special disposal procedures. This power supply is appropriately labeled. (2) The possibility of existing duct-work servicing Room 139 containing graphite dust should be considered during dismantling activities. (3) Polynuclear aromatic hydrocarbons (PNAs, sometimes called PAHs), formed at elevated temperatures during the coating process, have been partially trapped in local scrubbers or vented into the exhaust duct work.

4.4.5 Other Risk Considerations

No risks other than those previously mentioned were identified.

4.4.6 Areas Requiring Further Study

Other than assuring that appropriate disposal procedures are used,

no other areas were identified as requiring further study.

4.4.7 Summary and Conclusions

Decontamination/decommissioning of Room 139, Building 4508, should not present any unusual problems. Radiation levels are relatively low, and the equipment is in sound physical condition. The majority of the equipment involved is also accessible, therefore, no risks are involved due to a lack of needed information. Caution is recommended regarding the PCBs in the transformers, PNAs in the local scrubbers and exhaust duct work, and the possibility of graphite dust in the duct work. Standard disposal procedures should be sufficient for all disposal activities involving this facility.

Personnel occupying the area expect to be moved to the High Temperature Materials Laboratory in 1987, so the area will become vacant at that time. It could be used for a variety of purposes if it were radioactively "cleaned up". J. O. Stiegler (listed below) indicated a desire to assist in any way he could to help get the area decontaminated so that it could be made more useful.

4.4.8 Persons Knowledgeable about the Site

A. J. (Art) Moorhead, ORNL, 615/574-5153

J. O. (Jim) Stiegler, Metals and Ceramics Division, ORNL, 615/574-4065.

R. E. (Ray) Keny, ORNL, 615/574-4349

W. H. (Bill) Miller, Jr., ORNL, 574-5171

4.5 Transuranium Research Laboratories 41 and 45

4.5.1 Location:

These facilities are located in Rooms 41 and 45 of Building 5505 of

Figure 13. Schematic of the ORNL complex showing the location of Building 5505

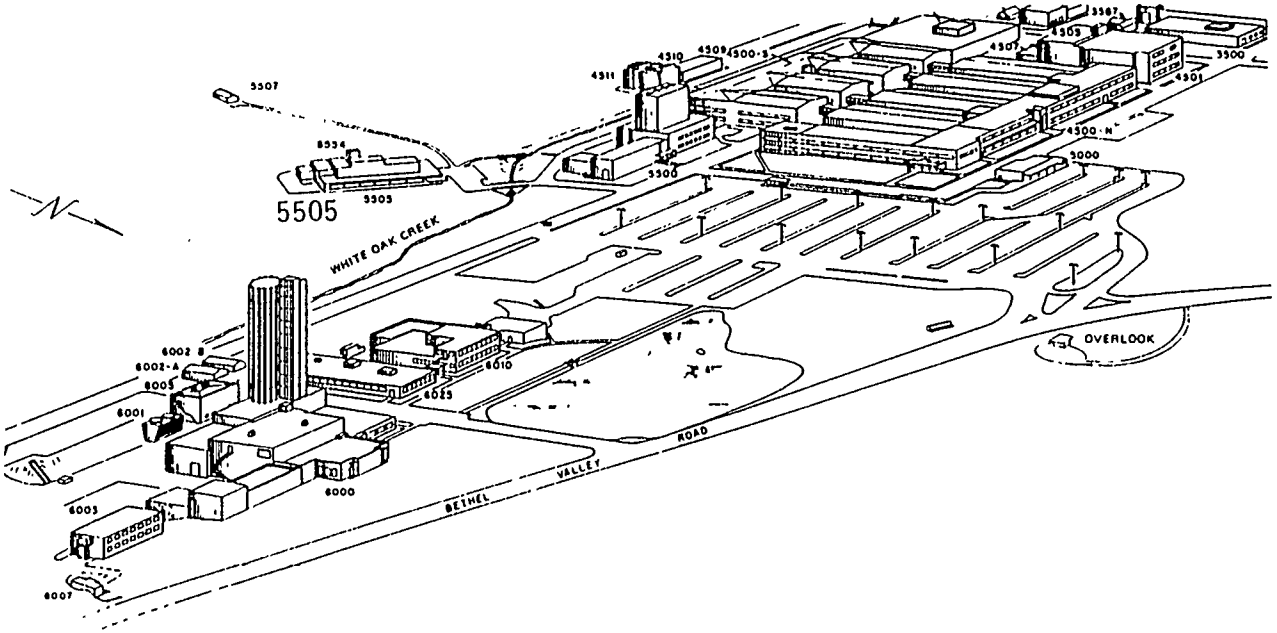
N
5507
5505
5504
5503
WHITE OAK CREEK
4311
4310
4309
4500-S
3507
4503
4507
3500
4501
4500-N
3000
3500
6002-B
6002-A
6005
6001
6003
6007
6000
6025
6010
BETHEL VALLEY ROAD
OVERLOOK

Figure 14. First-floor plan of Building 5505

Rooms 41 and 45 are located near the lower right

hand corner

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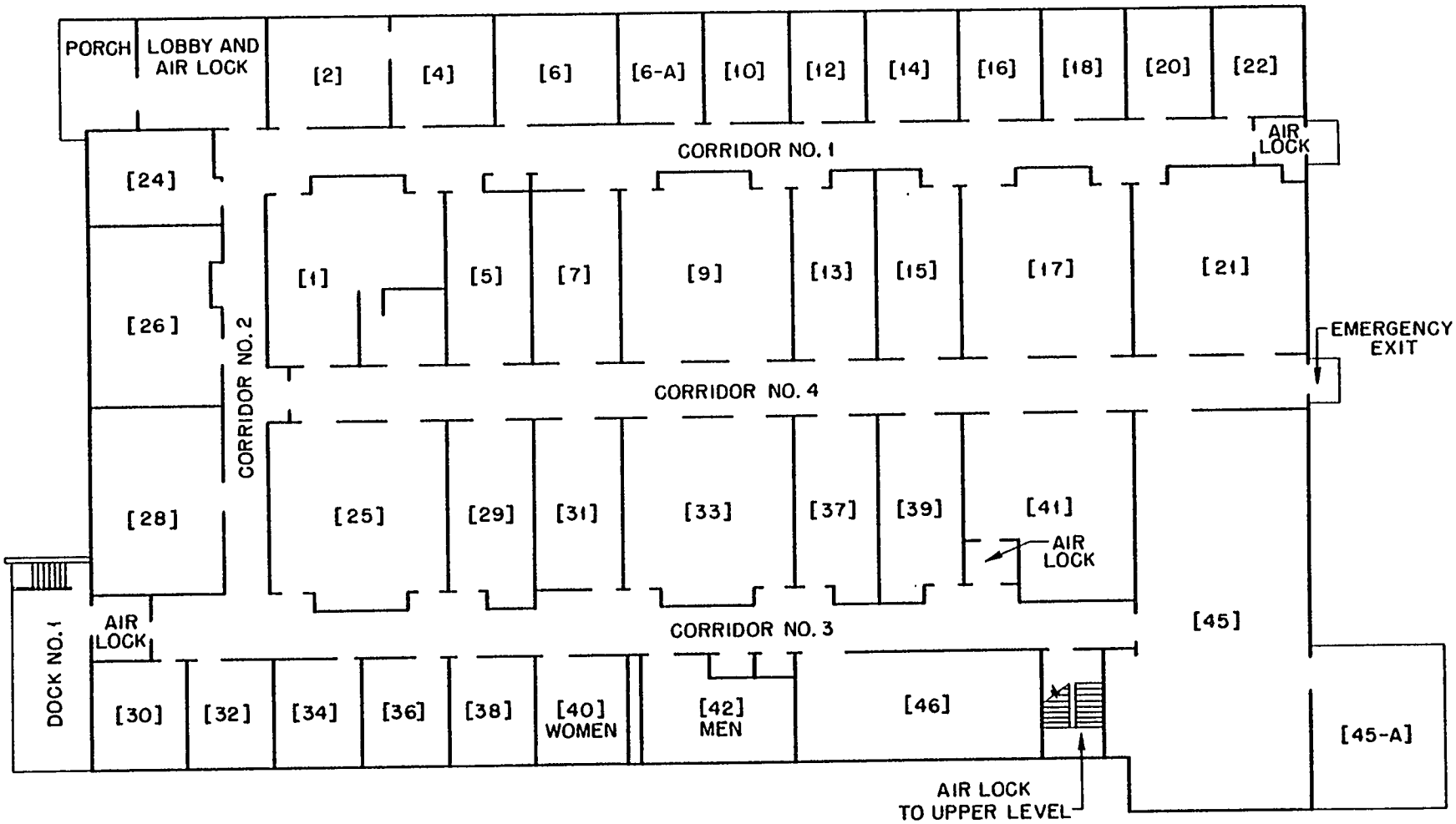


FIRST FLOOR PLAN
TRL-BLDG-5505

the main ORNL complex. The location of Building 5505 is shown in Figure 13. The locations of Rooms 41 and 45 are shown in Figure 14.

4.5.2 History and Use of the Facility

Building 5505 is an office and laboratory building for use by technical personnel conducting experiments with isotopes of the transuranic elements. This facility, comprised of 16 laboratories and a similar number of offices along with other support facilities, began operation in 1967 and continues today with an occupancy of some 20 personnel. The two areas of interest in this building are Rooms 41 and 45, both of which were used in the past but are presently inactive except for the storage of research equipment. Room 45 was initially used to provide a central holding area for radionuclides and to provide additional security and accountability for special materials. This room has been used for equipment storage since 1973. Room 41 was actively used for research through 1981 but has been relatively inactive since that time.

4.5.3 Radiological Characterization

Although the two rooms of interest are considered inactive, radiation surveys are routinely conducted; therefore, an accurate assessment of the radioactive contamination is possible. Equipment contained within these two rooms is relatively free of external transferable contamination; however, most of the surplus equipment does contain significant levels of radioactivity internally. Due to the significant levels of transuranic waste in both areas, special waste disposal procedures will be necessary.

The research-scale isotope separator in Room 41 has no external transferable contamination greater than 20 d/m/100cm^2 alpha or 200 d/m/100cm^2 beta-gamma. Surveys indicate a radioactivity level of 4 mR/h at the glove box end of the separator and a fast neutron activity of ~ 10 mrem/h. Due to the configuration of the separator, it is not possible to measure the internal contamination; however, based on prior use of the separator it is projected to have significant contamination from transuranics internally. It should also be noted that in the early 1970s, an accident occurred in this room resulting in extensive radioactive contamination to the room and its contents. Although the room was subsequently decontaminated to safe working levels, the possibility exists that radioactivity might be encountered during the removal of the separator.

The 13 portable glove boxes in Room 45 have no external transferable contamination greater than 20 d/m/100cm^2 alpha or 200 d/m/100cm^2 beta-gamma. Maximum radioactivity levels of ~ 4 mR/h were measured at the glove boxes. Alpha contamination within the glove boxes varies considerably with levels ranging from 1,000 d/m/100cm^2 to an approximate maximum of 1.0×10^6 d/m/100cm^2. In addition to the portable glove boxes, Room 45 also contains one built-in hood which has been converted to a glove box. Exterior contamination levels do not exceed those listed for the portable glove boxes, however, due to the inaccessibility of all the surfaces of this glove box, the possibility exists that additional sources of radioactivity might be exposed during dismantling operations.

Research equipment contained within the glove boxes consists of furnaces, ring stands, racks, lights, hot plates, pumps and other similar equipment which will be surveyed and disposed of using

appropriate standard or transuranic disposal procedures.

4.5.4 Environmental Concerns

As a result of the inherent ventilation/filtration system in Building 5505, environmental risks are minimal. Glove boxes are operated at a greater negative pressure than the surrounding laboratory room, and the room is at a greater negative pressure than the hallways and offices.

4.5.5 Other Risk Considerations

Risks, other than those previously mentioned, associated with decommissioning of the subject facilities appear to be minimal. Although some of the glove boxes presently contain chemicals along with the transuranic waste, the chemicals are in small quantities and should present no abnormal problems during their removal.

4.5.6 Areas Requiring Further Study

Other than a more detailed review of the potential contamination within the interior of the isotope separator in Room 41, no areas were identified as needing further study.

4.5.7 Summary and Conclusions

Decommissioning of the identified equipment in Rooms 41 and 45 of Building 5505 should present no unusual problems. Although the transferable contamination from the transuranics present a significant health hazard, past experience and established procedures should assure no undue risks. The isotope separator in Room 41 is difficult to

characterize internally; however, a thorough analysis of past operating parameters should result in an accurate projection of internal levels.

4.5.8 Persons Knowledgeable about the Site

O. L. (Lew) Keller. Chemistry Division, ORNL, 615/574-4987

W. D. (Bill) Carden, ORNL, 615/574-4997

R. L. (Dick) Hahn, ORNL, 615/574-4995

4.6 Molten Salt Reactor Experiment Fuel Handling Facility

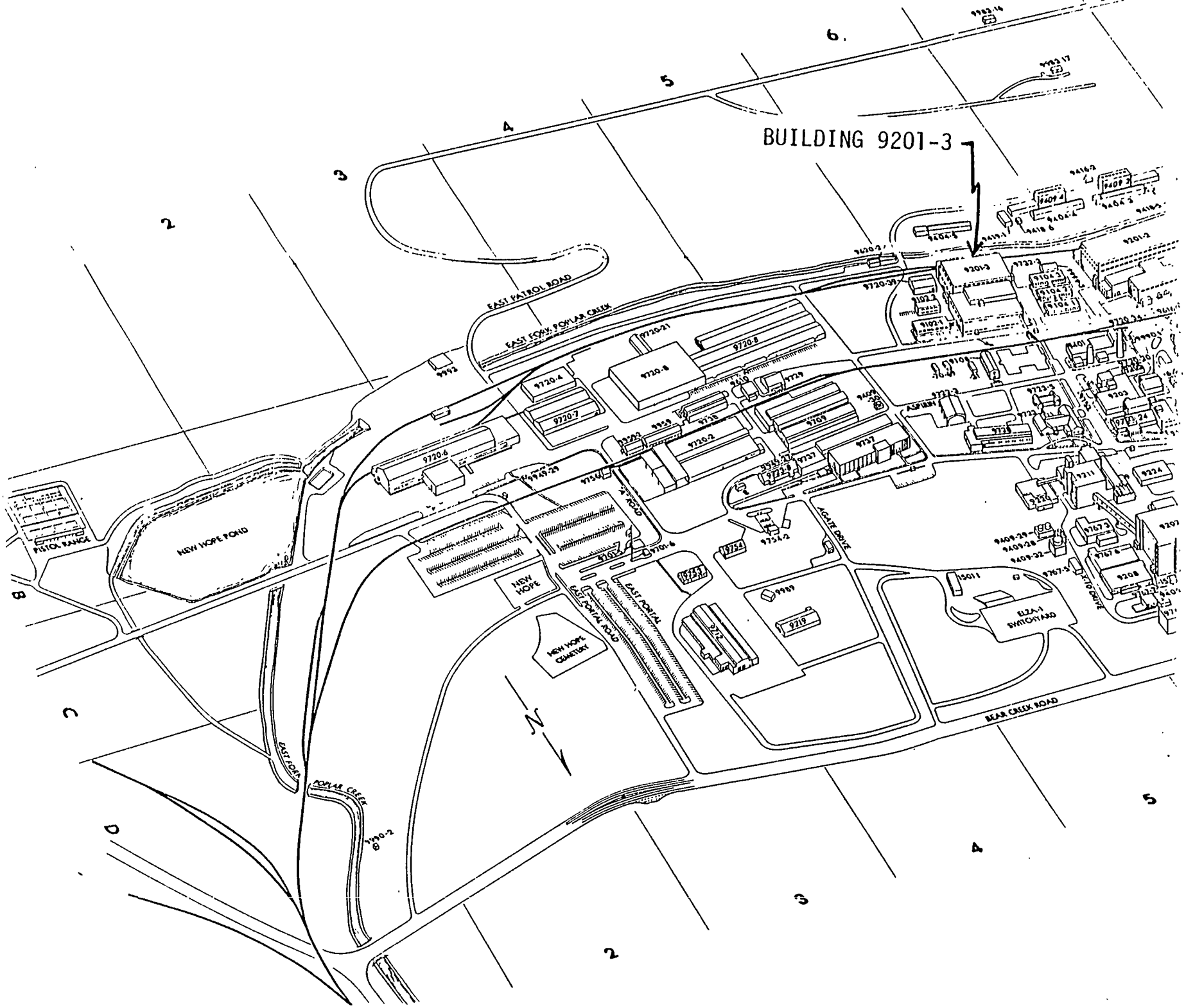
4.6.1 Location and Use

This facility is located in room 235 of Building 9201-3 of the Y-12 complex. The location of Building 9201-3 within the Y-12 complex is shown in Figure 15. The item of interest is a large cell (about 5 m X 5 m X 3 levels high). It was used in fuel preparation for the Molten Salt Reactor Experiment, which terminated in 1969.

4.6.2 Radiological Characterization

The cell is surveyed by health physicists routinely and entrances are appropriately tagged. The cell is generally contaminated throughout including the floors, sink, piping, and ductwork. Radiation surveys completed on August 7, 1985 indicated probe readings ranging from 5,000 - 50,000 d/m/100cm^2 alpha and transferable levels of 1,236 d/m/100 cm^2 alpha and 3,500 d/m/100cm^2 beta-gamma. The radiation levels cited are the result of the residual contamination from research, primarily with ^{233}U and ^{228}Th.

Figure 15. Location of Building 9201-3 within the Y-12 Complex


BUILDING 9201-3
EAST PATROL ROAD
EAST FORK POPLAR CREEK
NEW HOPE POND
PISTOL RANGE
NEW HOPE
NEW HOPE CEMETERY
EAST PORTAL ROAD
EAST PORTAL
A ROAD
AGATE DRIVE
ASPHALT
BEAR CREEK ROAD
POPLAR CREEK
EAST FORK
E.D.A.-1 SWITCHYARD
N

4.6.3 Environmental Concerns

No specific environmental concerns were identified.

4.6.4 Other Risk Considerations

No risks were identified during this survey.

4.6.5 Areas Requiring Further Study

No areas requiring further study were identified.

4.6.6 Summary and Conclusions

This facility should not present any problems in the event decontamination/decommissioning is initiated. It is in good condition and is well maintained. The persons listed below can probably provide answers to any specific questions about this facility and its past applications.

4.6.7 Persons Knowledgeable about the Site

T. G. Ramsey, Engineering Technology Division, Y-12, 615/574-0403 is a good first contact. For historical information, he may refer you to any of the following "old hands":

N. E. (Norm) Morgan, Y-12, 615/574-0233

E. L. (Ed) Biddle, Y-12, 615/574-0252

F. E. Lynch, Y-12, 615-574-0413

4.7 Coolant-Salt Technology Facility (CSTF)

4.7.1 Location:

This facility is located near the Molten-Salt Reactor Experiment Fuel Handling Facility (section 4.6 above) on the second floor of

Building 9201-3 of the Y-12 complex. The location of Building 9201-3 within the Y-12 complex is shown in Figure 3.

4.7.2 History and Use of the Facility

This facility consists of a flow loop system that was used to inject tritium into a boron trifluoride flow in support of the Molten Salt Reactor Experiment. It has been inactive for several years.

4.7.3 Radiological Characterization

This facility is also known as the CSTF Tritium Test Loop. The only residual radioactivity is located in the piping and valves located on the external surfaces of the facility. Radiation levels ranging up to 40,000 d/m/100cm^2 alpha are present with transferable levels of approximately 1,000 d/m/100cm^2 alpha and 3,500 d/m/100cm^2 beta-gamma. The piping is appropriately tagged with radiation warning tags.

4.7.4 Environmental Concerns

None identified.

4.7.5 Other Risk Considerations

No risks were identified during this survey.

4.7.6 Areas Requiring Further Study

No areas requiring further study were identified.

4.7.7 Summary and Conclusions

This facility should not present any problems in the event decontamination/decommissioning is initiated. It is in good condition and is well maintained. The persons listed below can probably provide answers to any specific questions about this facility and its past applications.

4.7.8 Persons Knowledgeable about the Site

T. G. Ramsey, Engineering Technology Division, Y-12, 615/574-0403 is a good first contact. For historical information, he may refer you to any of the following "old hands":

N. E. (Norm) Morgan, Y-12, 615/574-0233

E. L. (Ed) Biddle, Y-12, 615/574-0252

F. E. Lynch, Y-12, 615/574-0413

5. DISCUSSION AND CONCLUSIONS

Most of the facilities considered in this report are used for work or storage areas and are therefore routinely monitored for radioactivity and are maintained in good condition. The notable exception to this is the High Level Radiation Analytical Laboratory in Building 3019-B. The hot cells in that area have deteriorated appreciably (for example, zinc bromide is slowly leaking from some of the cell windows), and it is recommended that the High-Level Radiation Analytical Laboratory in Building 3019-B be given considerable priority in the decontamination/decommissioning sequence. In order to provide for early decommissioning of that particular site, high priority consideration must be given to restoration of or provision of alternative LLW service to that facility. Other than that concern, there does not appear to be any great urgency called for in decontamination and decommissioning these research laboratory facilities.

The information presented in this report is primarily from scoping surveys. Areas requiring further study have been identified for some of the facilities, as indicated in their respective sections of Chapter 4. An important unknown for all these facilities is the volume of waste to be generated during decommissioning operations. Accurate estimates of the volume of each category of waste (e.g., contaminated metal, rubble, liquid) need to be established for each site. Cost-of-decommissioning estimates and a menu of decommissioning options (e.g., dismantlement, passive protective storage) also need to be prepared for each site, probably along lines similar to those of the decommissioning reports already prepared for the surplus reactor facilities (see ORNL Document Series X-OE-231, Volumes 5-10).

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DATE ISSUED JUN 30 1987

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ORNL/RAP/LTR-86/54

DATE: August 15, 1986

SUBJECT: Inventory of ORNL Remedial Action Sites:
12. Other Contaminated Sites

TO: T. E. Myrick

FROM: H. J. Grimsby

#1827 (add)

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RAP86-54

Inventory of ORNL Remedial Action Sites: 12. Other Contaminated Sites*

H. J. Grimsby

August 15, 1986

Energy Division
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37831

Operated by

Martin Marietta Energy Systems, Inc.
for the
U.S. Department of Energy

*Based on work performed at Oak Ridge National Laboratory, operated by Martin Marietta Energy Systems, Inc., under Contract No. DE-AC05-84OR21400 with the U.S. Department of Energy.

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1.0 OBJECTIVES

The primary purpose of the remedial action task is to provide corrective measures and facility decommissioning necessary to place the Oak Ridge National Laboratory (ORNL) in compliance with applicable environmental and health regulations for radioactive and hazardous chemical wastes. Remedial Investigation and Feasibility Studies of the sites will be addressed within Waste Area Groupings (WAGs) such that decontamination prescribed to control the potential for human exposure and environmental impact from each site is integrated into the program as a whole. Remedial actions will be conducted at locations where past activities have resulted in contamination of facilities or the environment that necessitate corrective actions or decommissioning. Remedial actons are divided into two activities: (1) Site Decommissioning Program: Surplus Contaminated Facility Program (SCFP) and Surplus Facility Management Program (SFMP), and (2) Site Corrective Measures Program (SCMP).

Remedial Action Program sites at ORNL comprise 12 distinct categories, listed in Table 1, within the SCFP, SFMP, and SCMP and one category which is comprised of the miscellaneous sites and facilities which do not fall within the definition of one of the 12 distinct areas.

This summary is the product of information abstracted from historcal records and knowledgeable personnel. The information includes, where available, location, description, history, contaminants, extent of contamination, regulatory status, results of past studies, and environmental considerations.

Table 1. Inventory of ORNL Remedial Action Program Sites

1. Solid Waste Storage Areas
2. LLW Seepage Pits and Trenches
3. Process Ponds
4. White Oak Creek Watershed
5. LLW Line Leak Sites
6. Environmental Research Areas
7. Hazardous Waste Sites
8. Radioisotope Processing Facilities
9. Experimental Reactor Facilities
10. Radwaste Facilities
11. Research Facilities
12. Other Contaminated Sites
13. Inactive Injection Wells

2.0 BACKGROUND

The areas included in this category are located within the Oak Ridge National Laboratory Main Complex WAG and ORNL Operations at Y-12 WAG. The sites described herin involve varied miscellaneous types of contamination and do not fall within any of the other twelve specific categories such as the ponds, radwaste facilites, or hazardous waste sites.

The sites, located on Figures 1 and 2, include the following:

ORNL at Y-12	
East End Basement	9204-1
Attic	9204-1
Storage Tank	9201-3
ORNL Main Complex	
Building 3503 Storage Pad	3503 Area
Overflow of Graphite Reactor Storage Canal	3001/3019 Area
1959 Explosion of Building 3019 Cell	3019 Area
Oak Ridge Reactor (ORR) Decay Tank Rupture	3087 Area
Base of 3019 Stack	3019/3091 Area

The Building 3503 Storage Pad and the ORNL sites at Y-12 are part of the Surplus Contaminated Facilities Program (SCFP) and are regulated under the Department of Energy (DOE) Order 5820.2. Preliminary radiological scoping surveys have been performed for these sites as part of the SCFP and are summarized within this report. The four (4) remaining sites were originally regulated under the Comprehensive Environmental Restoration, Compensation, and Liability Act (CERCLA) as part of the SCMP. Recent

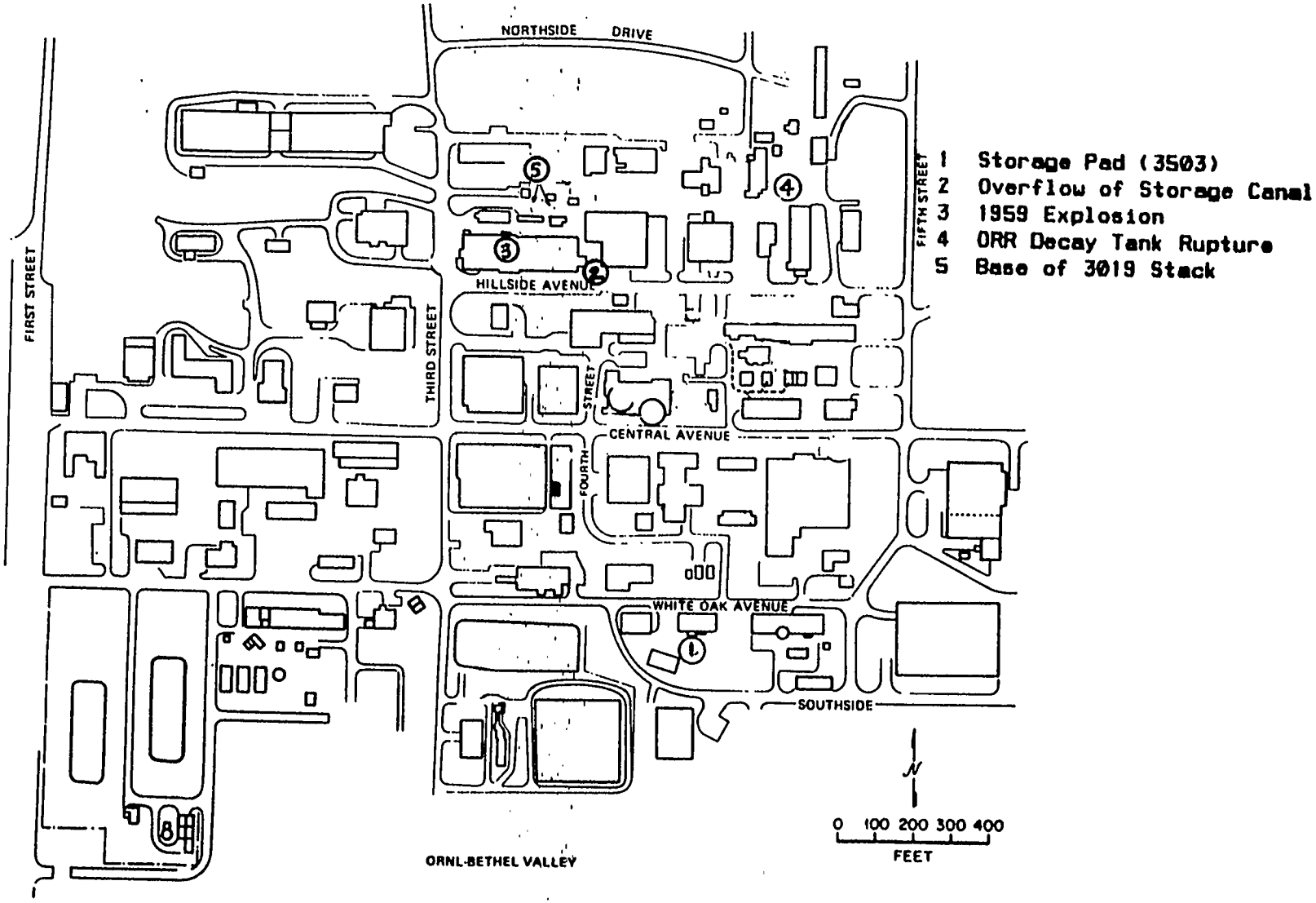
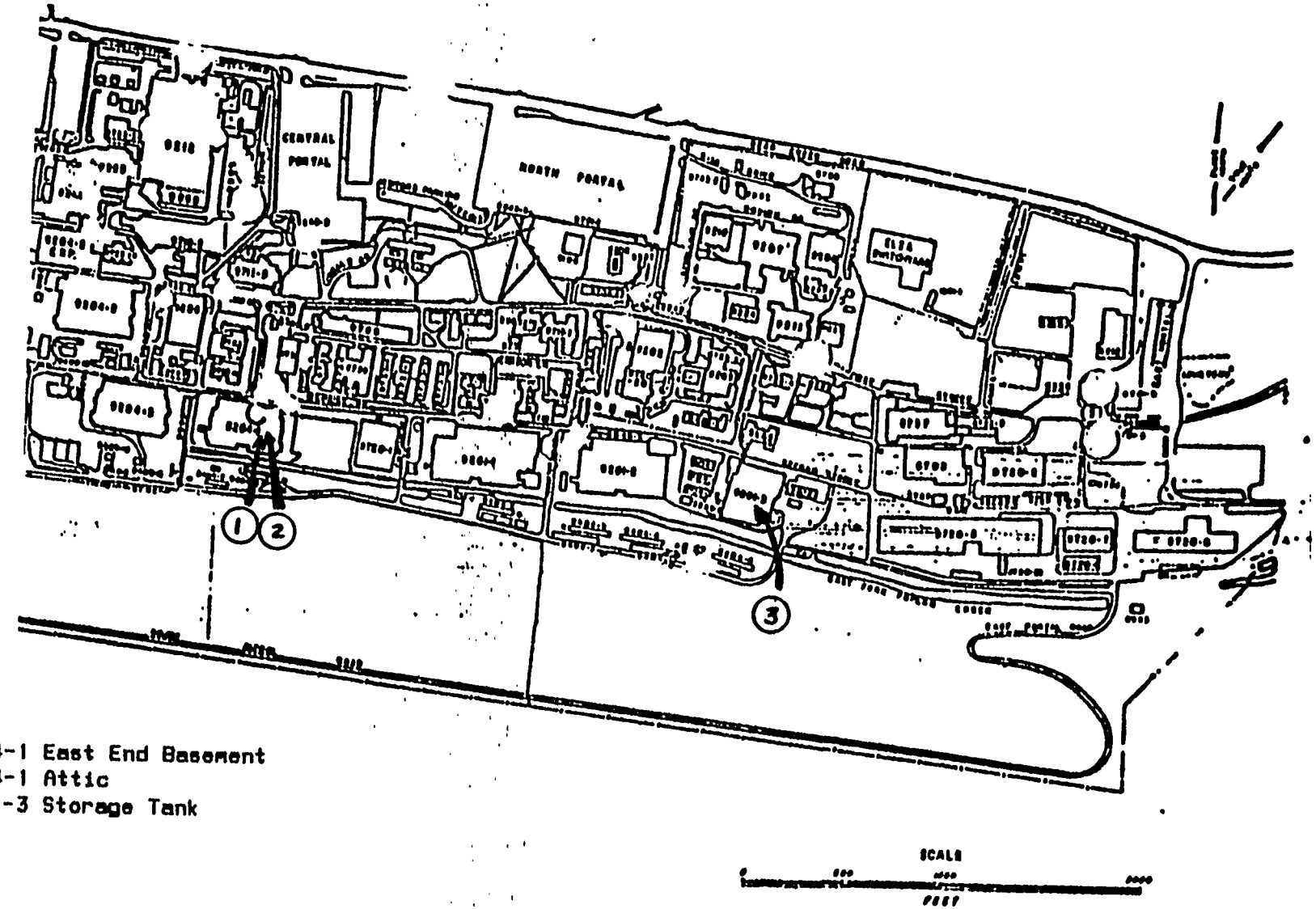


Figure 1. ORNL Main Complex: Other Contaminated Sites



1. Bldg. 9204-1 East End Basement
2. Bldg. 9204-1 Attic
3. Bldg. 9201-3 Storage Tank

Figure 2. ORNL at Y-12: Other Contaminated Sites.

revisions to the program emphasized compliance under 3004(u) provisions under the Remedial Conservation and Recovery Act (RCRA) as well.

The modified Hazard Ranking System (mHRS) migration scores (S_m) were determined for the SCMP sites (Overflow of Graphite Reactor Storage Canal, 1959 Explosion of Building 3019 Cell, ORR Decay Tank Rupture, and Base of 3019 Stack) and reported in ORNL/TM-9989 "CERCLA Phase I Report: Identification and Preliminary Assessment of Inactive Hazardous Waste Disposal Sites and Other Contaminated Areas at ORNL" (Nix et al., March 1986) to be as follows:

	mHRS(S_m)
3019 Stack, 1959 Accident	1.4
Overflow of ORG Canal	0
Rupture of ORRR Tank	0

The parameter S_m reflects the potential for harm to humans or the environment from migration of a hazardous substance away from the facility by groundwater, surface water, or air pathways. The mHRS ranking for these sites were low either because of lack of information (3019 stack, Overflow of ORG Canal, and Rupture of ORRR Tank) or the small quantity of contaminants present (1959 Accident).

3.0 SITE DESCRIPTIONS AND ENVIRONMENTAL CONSIDERATIONS

3.1 ORNL at Y-12: East End Basement of 9204-1

Site Description. This site has two adjacent areas (separated by a 1-m concrete walkway) which are located in the basement crawl space of the east end of Building 9204-1. Both areas are dirt floors which are approximately 7.5-m by 7.5-m. The area has been roped off to restrict access and is not used frequently by employees (nor is it easily assessible to employee traffic).

The source of contamination at this site is unknown. Two possible explanations for the contamination are noted: (1) a pump leaking a uranium solution through the floor above the dirt; or (2) contaminated dirt being brought into the area. Recent scoping surveys indicate contamination with low-level alpha emitters (alpha radiation levels of up to 2000 d/m/100 sq-cm). No beta-gamma radiation was detected. Although the depth of soil contamination has not been verified, the present estimate is that removal of 15 to 20 cm of soil would be necessary to remove the contamination (Houser and Simpson, April 3, 1986). Houser and Simpson also report background beta-gamma levels of 300 counts per minute (c/m) (< 1 mR/h) from process pipes located approximately 1.5 m above the dirt floor areas. The maximum contact levels are reported to be 2000 c/m (< 1 mR/h). The process pipes in the area (approximately 5 feet above the dirt floor) are not included in the scope of this remedial activity.

Environmental Considerations. Surface and groundwater contamination are the primary environmental concerns for this area. During periods of

heavy rainfall, flooding can occur (and has occurred) with the potential for transport of contaminants to adjacent land areas, surface waters, and groundwater.

Removal of the contaminated dirt would require transport and disposal of approximately 1000 cubic feet of dirt. Before removal and disposal, the extent and specific type(s) of contamination would have to be determined by soil analysis.

Due to the close and relatively closed nature of the area, health and safety precautions required would be somewhat cumbersome during excavation (or other remedial action alternatives), but no real difficulties are foreseen in providing proper health and safety protection during remediation.

3.2 ORNL at Y-12: Building 9204-1 Attic

Site Description. The contaminated area is located on the second level of Building 9204-1. The contamination is confined to paint which was used to "fix" earlier contamination of the ceiling area. The paint, with adherring contamination, is now flaking off and landing on top of a false ceiling which was installed prior to the painting of the original ceiling. The original ceiling apparently was contaminated as a result of washing operations which occurred during the 1940's period. Splatter contaminated the beams and ceiling area above the washing area requiring the application of paint for fixation purposes.

The total attic area marked as a contamination zone is approximately 18 m by 24 m; however, according to the most recent radiological survey

(April 18, 1985), there is no measurable radioactivity beneath the false ceiling. The contamination is confined to the original ceiling and beams, which are still partially covered with the paint, and the top of the false ceiling, which contains flakes of the paint. Transferable alpha levels in the area ranged between 10 and 275 d/m/100 sq. cm. with an average level of 20 d/m/100 sq cm. The maximum beta-gamma level was 50 c/m (< 1 mR/h) (Houser and Simpson, April 23, 1986). The specific isotope(s) contaminating this area is (are) unknown.

Environmental Considerations. Since the source of radioactivity in this area is unknown, the nature of contamination should be determined prior to any cleanup or disposal activities. Direct contact exposure appears to be the pathway of concern since the contaminants present are "fixed" and are not likely to be tranported by air or water pathways.

3.3 ORNL at Y-12: Storage Tank in Building 9201-3

Site Description. The storage tank is located in a mezzanine room in Building 9201-3 shown in Figure 3. Most of the surplus contaminated facilities and equipment in this building were operated in support of the Molten Salt Reactor Experiment (MSRE) during the period 1958 through 1969. The glass-lined tank was used to hold cutting oil for a machine shop but is now empty. The scoping survey revealed no alpha or beta-gamma radioactivity external or internal to the tank (Houser and Simpson, March 31, 1986). The tank is located in a room designated as a C-zone area because of existing low-level alpha contamination at the room's entrance. The contamination in this area of the room was sealed by an application of paint.

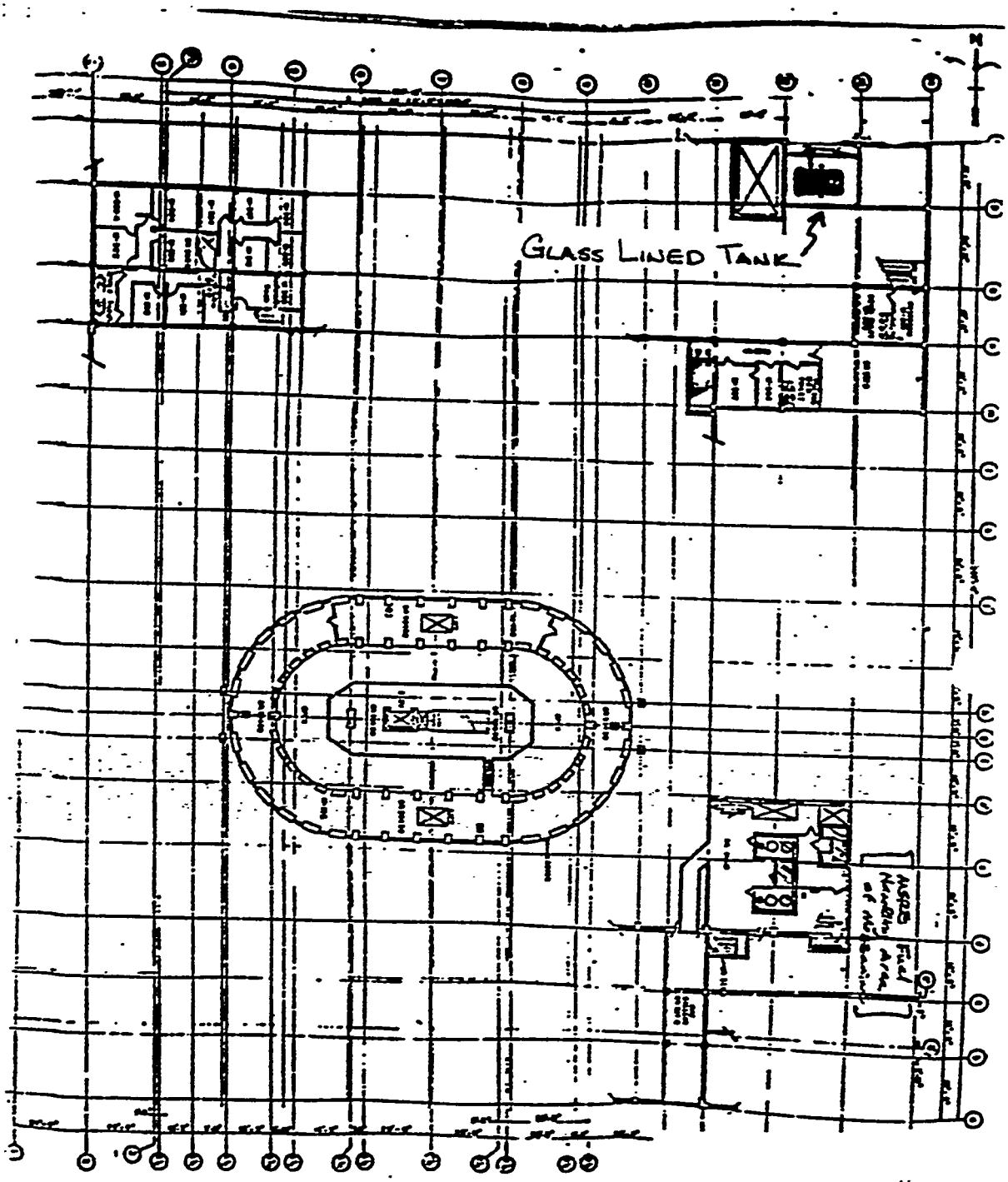


Figure 3. Building 9201-3 Storage Tank Location.

Environmental Considerations. Since the scoping survey revealed no alpha or beta-gamma contamination external or internal to the tank and there are no tank contents, no environmental problems are expected with respect to demolition/disposal of the tank or leaving the tank in place. Residuals from the cutting oil, if any, should present no unique disposal problems.

3.4 ORNL Main Complex: Storage Pad Southwest of Building 3503

Site Description. The concrete pad located southwest of Building 3503, as shown in Figure 1, is approximately 12 m by 15 m. The pad, built in the late 1950's, has been used as storage for containers of radionuclide contaminated materials, scrap material, and metal recovery operations equipment. A portion of the pad has covering over two areas which are used for storage of barrels and "surplus" miscellaneous equipment and crates. The two areas have metal tray floorings. Personnel in the area indicated that the items currently stored on the pad are to be disposed in the near future.

The major radioactivity present is believed to be associated with the storage of U-233 and Pu-239. Prior to the addition of a 10-cm layer of concrete to the pad in the 1970's, the surface contamination was estimated at 100,000 d/m/100 sq cm (Houser and Simpson, April 1, 1986). The most recent survey performed by Houser and Simpson shows detectable radioactivity in the areas under the shed and within approximately 1 m of the south and west side of the pad at ground level. No alpha radioactivity was detected during the latest survey. Although the major radioactivity beneath the top layer of concrete is believed to be U-233 and Pu-239, the source of radioactivity in the areas surrounding the pad and the metal trays has not

been determined. In addition, although the extent of lateral surface contamination has been determined, the amount of subsurface contamination is not known.

Environmental Considerations. The pad is located just north of White Oak Creek (WOC) and migration of contaminants by groundwater or by surface water run-off is possible. Before implementing cleanup activities, the nature and extent of contamination should be determined in the subsurface concrete layer, the metal tray area, and the ground area located southwest of the pad and to this end, a more detailed mapping and identification of contaminants has been scheduled. In addition, disposal of equipment and barrels currently on the site, after proper characterization, is in progress. The presence of Pu-239 should be considered during any remedial action involving disturbance of the concrete. In addition, since this pad has been used to store a variety of materials associated with metal recovery operations, the characterization should include analysis for chemicals other than radionuclides, such as heavy metals and organics.

3.5 ORNL Main Complex: Overflow of Graphite Reactor Storage Canal

Site Description. Information concerning contamination in the area of the storage canal has not been found. The overflow is mentioned in notes accompanying ORNL Drawing A-90015-0-063F, Revision 5; however, no mention was made in the "Decommissioning Study for the ORNL Graphite Reactor (OGR)" published in 1976 nor could information be found in the various Operations Division Monthly reports which usually contain references to such 'unusual' occurrences.

Since the incident cannot be described, the following background data on the canal and typical depleted fuel components are provided such that future characterization of the 3019/3001 area (located within the ORNL Main Complex WAG) may be planned (and/or more easily understood if encountered). As part of any future characterization, a survey in the vicinity of the canal would seem appropriate.

The reactor fuel was natural-uranium metal clad with high-purity aluminum. Fuel was removed from the reactor and guided into an 28-cm square chute which opened into the bottom of a 6-m deep by 2-m wide by 2.5-m long water filled pit located below the ground-floor level on the west side of the reactor (Building 3001). The general layout of the canal in relation to the reactor is shown in Figures 4, 5, and 6. The 2-m wide by 3.5-m deep canal connected to the fuel discharge pit and extended south 20 m, then west 13.5 m, and terminated beneath Building 3019 (the Radiochemical-Processing Pilot Plant). The canal was used for storage and handling of irradiated fuel and radioisotope targets.

Since the fuel was aluminum-clad natural-uranium metal, the contaminants would most likely be fission products, such as Cs-137 and Sr-90, and fissile species of U-235 and Pu-239.

Environmental Considerations. Although no data has been found to collaborate that an overflow did occur, the potential for this being a major source of groundwater contamination in the 3019/3001 area exists. If excavation occurs in this general area, transport of contaminants by surface water to WOC could occur unless surface waters were properly diverted. Characterization of the 3019 area includes dye tracing activities, but no known characterization activities are known or planned for this individual site.

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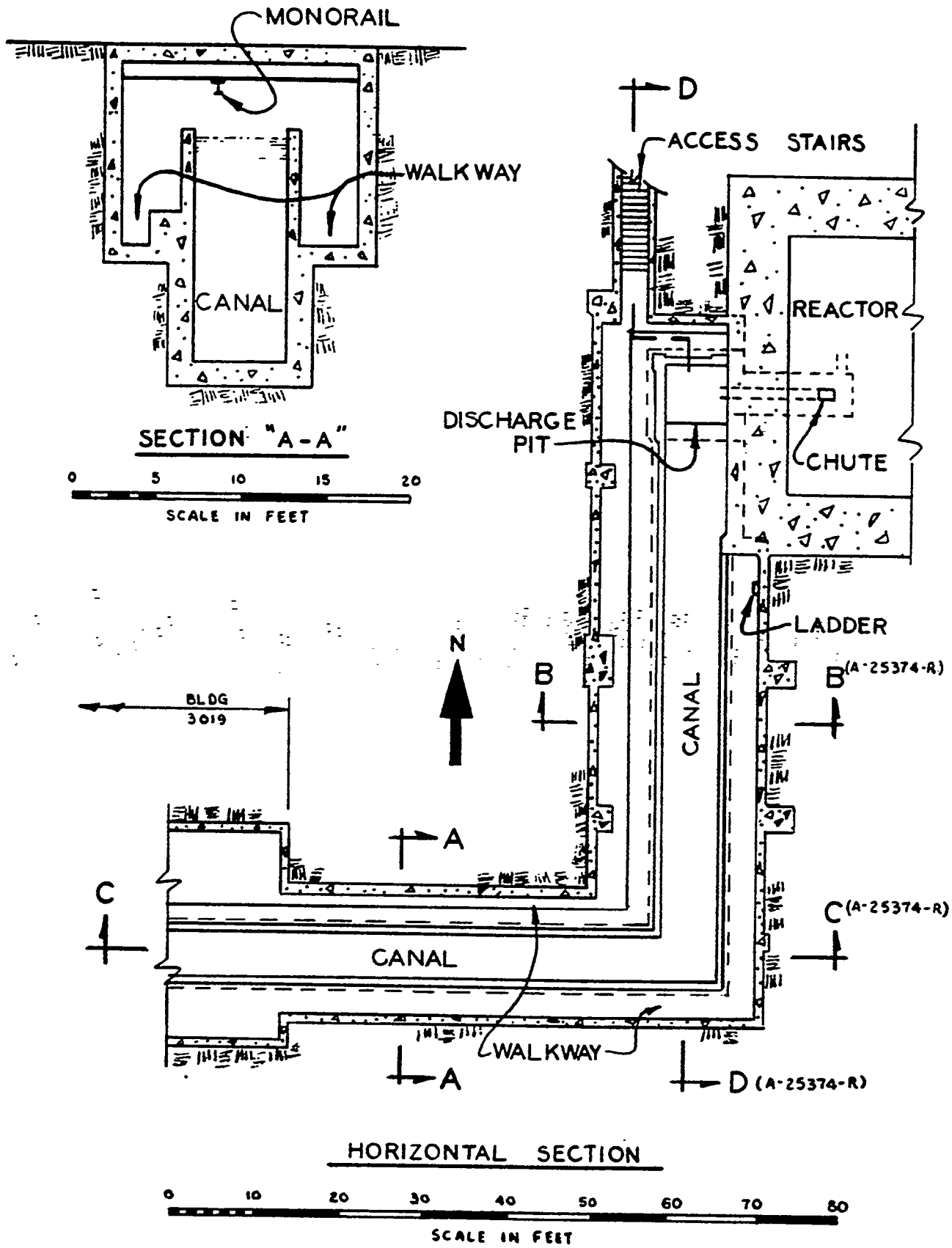

Figure 4. Layout and Section of the Fuel Discharge and Storage Canal.

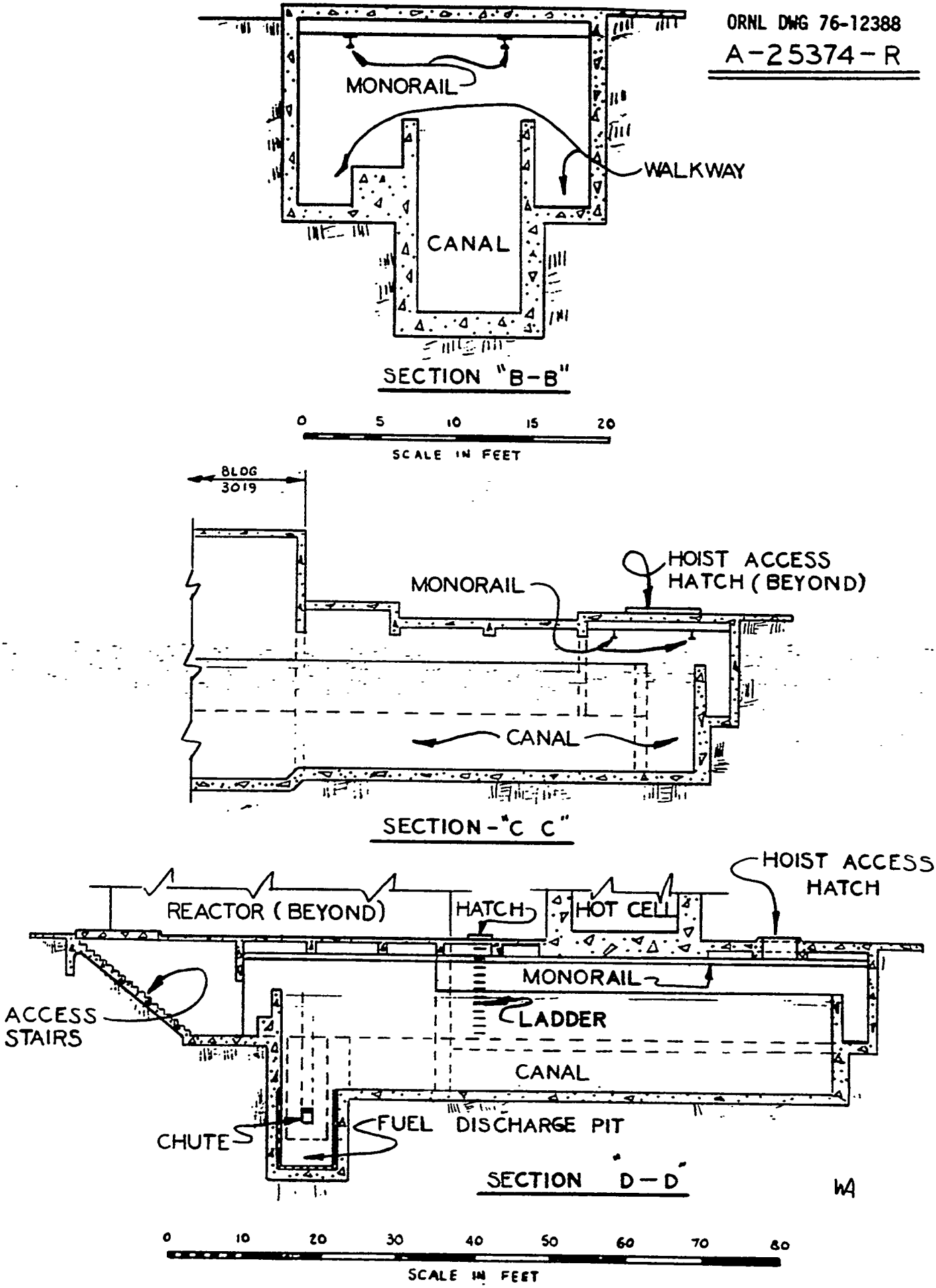

Figure 5. Sections of the Fuel Discharge and Storage Canal

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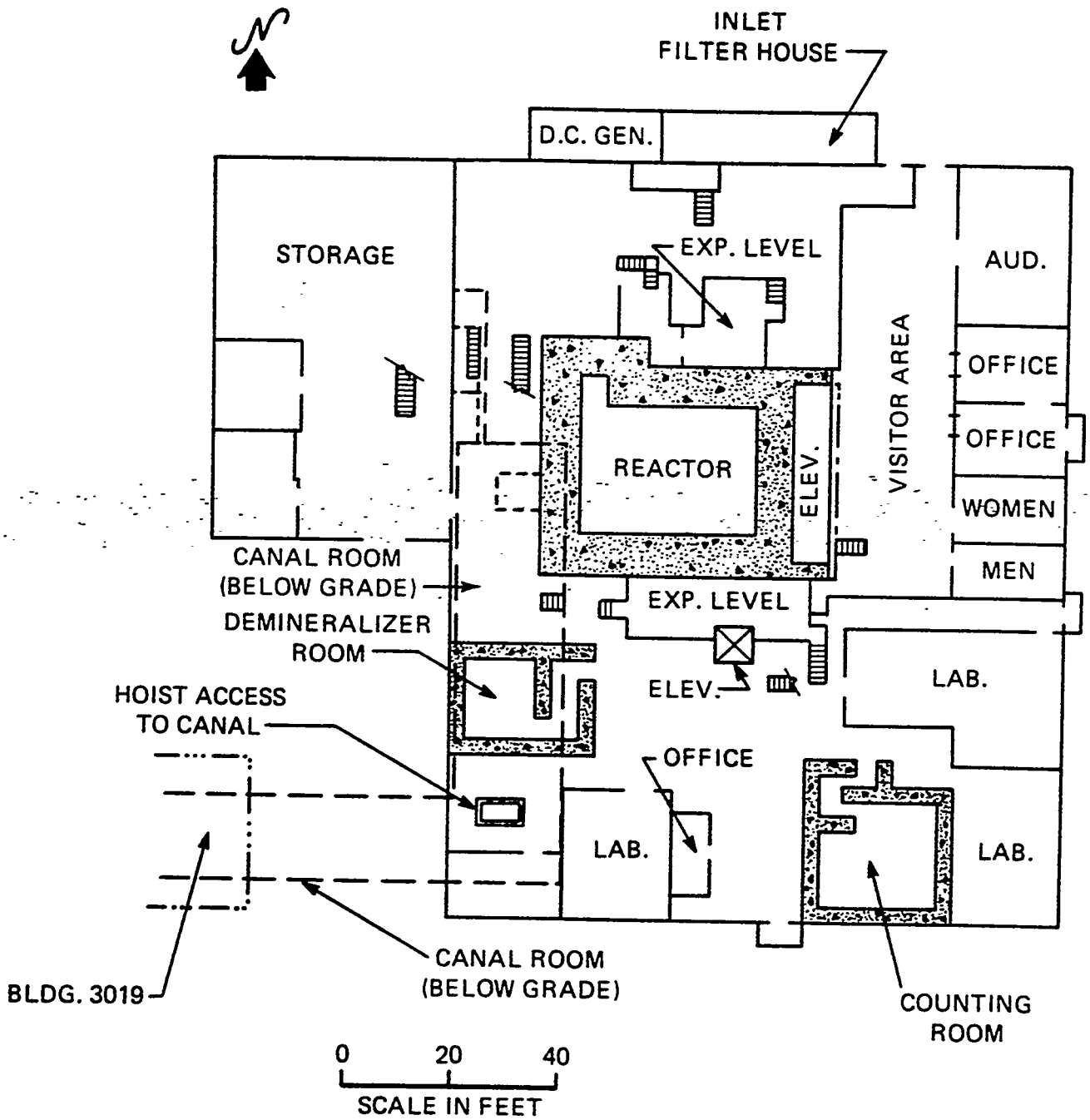


Figure 6. Ground Floor Plan of Building 3001

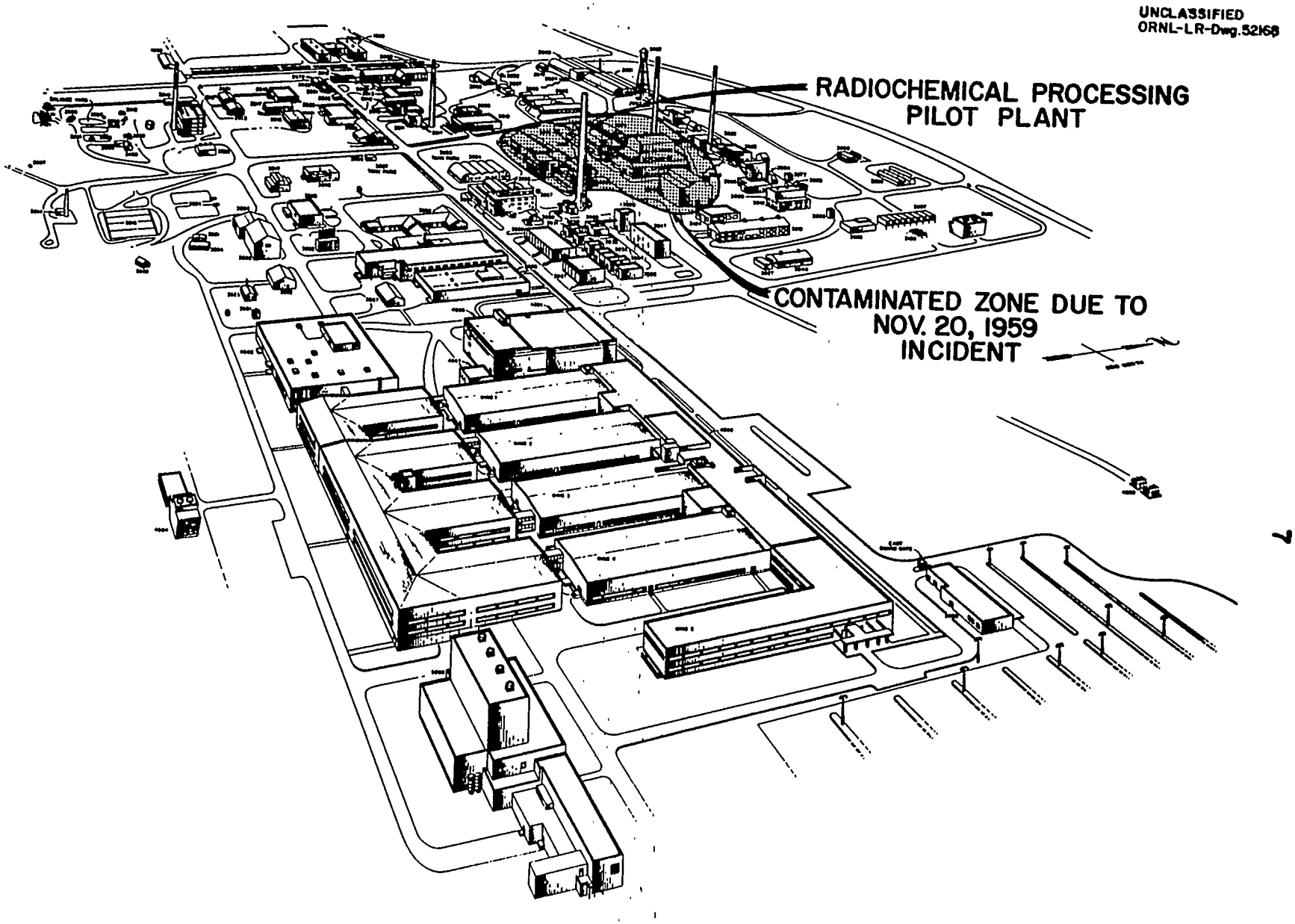
3.6 ORNL Main Complex: 1959 Explosion of Building 3019 Cell

Site Description. On November 20, 1959, a nonnuclear explosion involving an evaporator occurred in a shielded cell in the Radiochemical Processing Pilot Plant, Building 3019. Plutonium released from the processing cell contaminated areas in the Pilot Plant building and nearby streets and building surfaces. The explosion was considered the result of rapid reaction of nitrated organic compounds formed by the nitration of 14 liters of Turco Decon 4501, a decontaminating reagent.

Figure 7 shows the location of Building 3019 and the ORNL Main Complex circa 1960. The evaporator which ruptured was shielded in cell 6 of the building as shown in Figure 8.

Radioactivity was released to the processing cell, the Chemical Technology Division Pilot Plant operating and office areas, and buildings and grounds in the vicinity of the Radiochemical Processing Pilot Plant. Fallout of the radioactivity was reported as rapid, and the contaminated area was only a small fraction of the Laboratory area as shown in Figure 9.

Alpha-emitting plutonium was the major contaminant and beta-gamma emitting zirconium-95 and niobium-95 were considered second in importance. The potential hazard presented by the plutonium and the cleanup effort required to contain or remove the plutonium alpha activity apparently overshadowed the consequences of the zirconium-niobium contamination. Figure 8 is a cross-section of the facility showing contamination levels. The highest concentration was inside the cell, and the second highest level was in the area directly above the cell.



Oak Ridge National Laboratory,
X-10 Site

Figure 7. Location of Contaminated Zone due to November 20, 1959 Incident

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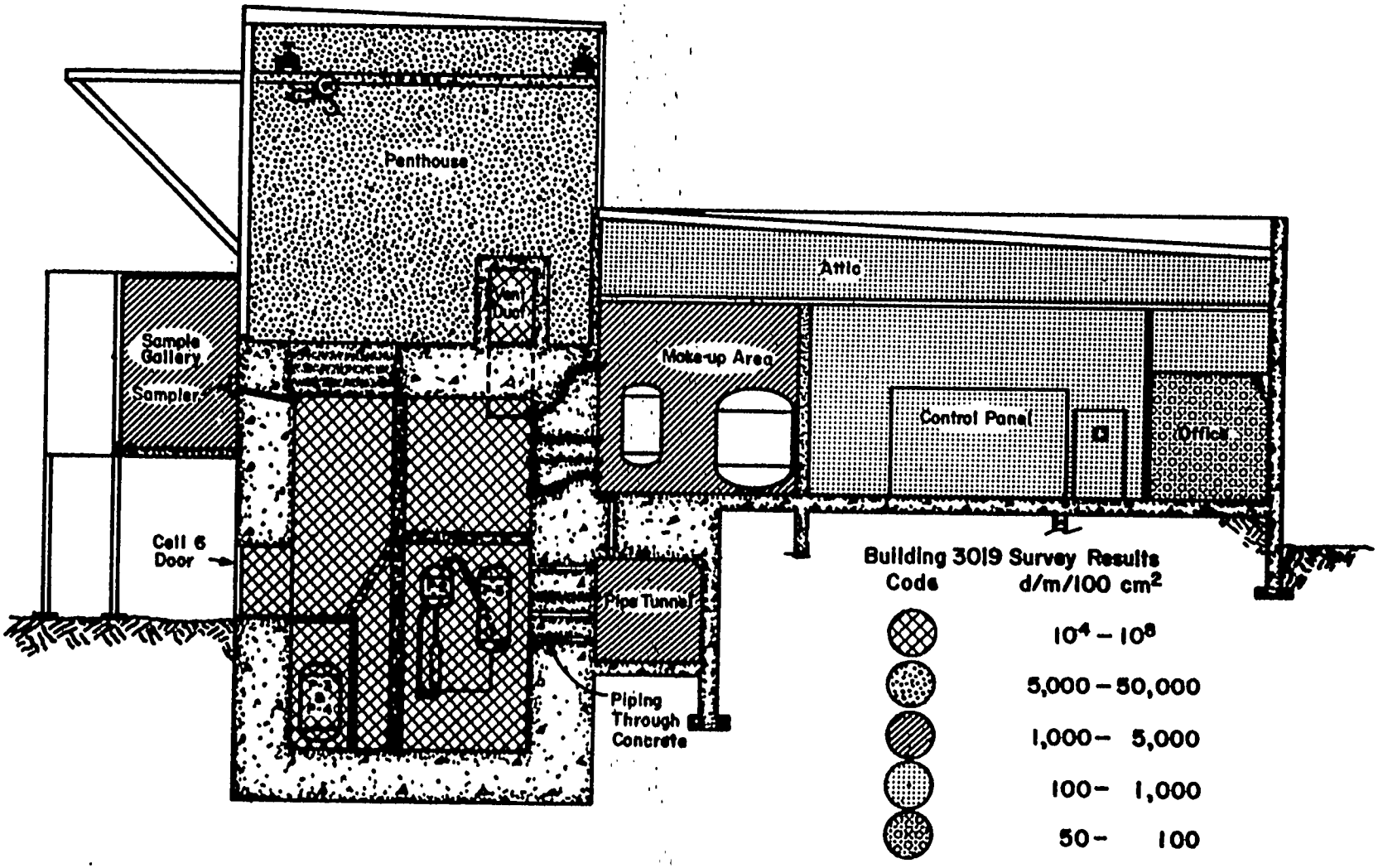

Figure 8. Radiochemical Processing Pilot Plant

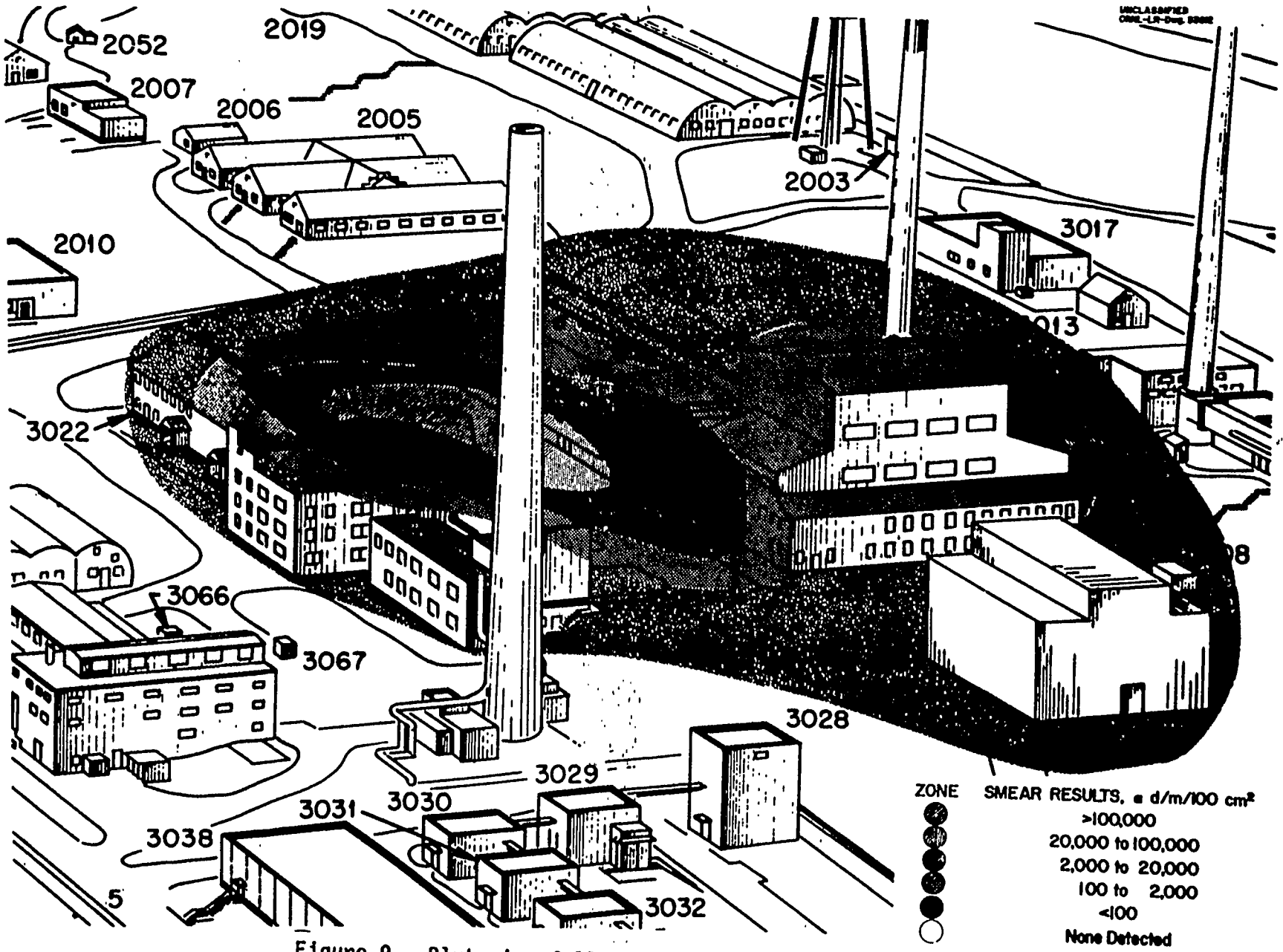

Figure 9. Plutonium fallout zones after explosion.

Standard Health Physics techniques of the day were used to define contaminated areas and to indicate the degree to which these areas were contaminated. This apparently consisted of using gas-purged proportional counters for measuring direct alpha activity, cutie pies and Geiger counters for beta-gamma measurements, and filter paper smear techniques for determining the alpha and beta-gamma transferable radioactivity. Calculations of the amount of plutonium dispersed by the explosion were considered to be order-of-magnitude estimates only, due to questionable data (King and McCarley, February 1, 1961).

The required limits for residual decontamination at the time were:

	Direct Reading	Transferable
Max	300 d/m/100 sq cm	30 d/m/100 sq cm
Avg*	<= 30 d/m/100 sq cm	<= 3 d/m/100 sq cm

*The number of samples considered in deriving an average included at least 10 samples and at least one sample from each square meter of the projected surface area.

If cleaning to these specifications was not feasible, the areas were to be decontaminated to within 10 times these specifications and the surfaces were to be painted. Three coats of paint were applied: the first coat fixed the contamination, the second coat was an orange enamel, and the third coat matched the decor. These areas were called "controlled zones" and specifications called for surfaces to be repainted if the orange paint showed. Reportedly, most of the areas designated "controlled zones" were "contamination zones" before the plutonium release and as such were designated as restricted access.

The only areas which met the specifications following decontamination were the office areas. All other areas, except the cells were decontaminated to a factor of 10 above the limits and repainted with three coats of paint.

The following is a summary of measures taken during the two years following the explosion. Detailed accounts of the decontamination activities can be found in ORNL-3100, "Decontamination of Cells 6 and 7, Building 3019 Following Plutonium-Release Incident," and ORNL-2989, "Plutonium Release Incident of November 20, 1959".

Decontamination of Outdoor Areas. The exterior surfaces of the buildings shown in Figure 9 and the surfaces of the roads and grounds in the area were treated as described in the following. No pretreatment was given to any of surfaces, and no exterior surfaces were washed before painting.

Bldg 3019. All exposed surfaces south of 3019 were fixed with paint, and everything except three stainless steel tanks in a permanent storage pit were removed to the burial ground. The roof of the facility (including the penthouse and the sampling gallery) was covered with tar and chips to a depth of up to 5 cm. The aluminum siding on the sampling gallery and penthouse was painted as was the duct work between the building and the 3019 stack. The south side of 3019 below the sampling gallery was painted white, the paint was then removed and the wall was repainted.

Bldg 3022. The roof and north wall were sprayed with green enamel paint to fix the alpha activity. Two coats of aluminum paints were subsequently added to the roof, and the north and west sides of 3022 were painted to match the original color. This building was later razed.

Bldg 3025. The roof was sprayed with tar and chips, painted, and the cooling tower and part of the duct work on top of the building were painted.

The north wall, constructed of red brick, was wiped with detergent and water and sealed with two coats of masonry sealing compound.

Bldg 3001. The roof was coated with aluminum fiber roofing compound to a maximum depth of 0.6 cm. All four external walls of the building were repainted the original alumninum color.

Bldg 3005. The roof on the east and west wings was coated with alumninum fiber roofing compound to a maximum of 0.6 cm and the building was repainted its original color.

Buildings 3003, 3004, and 3008 were completely repainted.

Roads and Grounds. The surface of Hillside Avenue between Third Street and Bldg 3042 and the paved area between Bldgs 3001 and 3042 were covered with tar and chips and then excavated to a depth of 30 cm and replaced with gravel. These sections were then repaved. The road north of Bldg 3019 and south of 3074 was covered with emulsified asphalt and gravel. Ground was removed within the area of Bldgs 3019, 3022, 3025, 3001, and 3042.

Decontamination of the Graphite Reactor Building (3001). Every laboratory and work area in the building was contaminated with plutonium drawn into and through the building ventilation system. The building was vacuumed, washed, and then surveyed for contamination.

Decontamination of the Radiochemical Processing Pilot Plant (3019). All Chemical Technology Division areas of the pilot plant were contaminated by plutonium as a result of the explosion. During the cleanup, contamination was uncovered that had been deposited and then covered during previous years of operation.

The amount of plutonium dispersed to the area south and east of building 3019 was estimated to be approximately 600 mg in the form of Pu-239 and Pu-240 and equivalent to approximately 0.047 curies (Parrott 1979). Most contamination was removed with decontamination activities following the incident.

Environmental Consideration. Cleanup actions following the incident were extensive and immobilization of residual contamination appeared adequate. Continued effectiveness of the immobilization activities taken should be confirmed. Demolition of structures, which were within this area at the time of the explosion, would have a small potential for causing environmental releases. Precautionary steps might be needed to limit air and direct contact exposures. In addition, disposal of the demolition waste would require consideration of the presence of radionuclides, particularly Pu.

3.7 ORNL Main Complex: Oak Ridge Reactor Decay Tank Rupture

Site Description. The following note in the "Health Physics Division Applied Health Physics and Safety Annual Report for 1974" describes the decay rupture.

"Surveillance and assistance were provided during the excavation of the 11,000 gallon Decay Tank for repair of a leak. The leak had been releasing primary coolant water at a rate of 1.5 gallons/minute. Radiation levels up to 2 R/hr were encountered along with levels of transferable contamination up to 35 mR/hr at 1 inch. The contamination was effectively confined to the immediate work area. Internal exposures were avoided by the use of plastic

suits with a positive air supply. The external exposure was kept well below the permissible limits."

In addition to the reference cited above, the incident is described briefly in the following note which accompanied ORNL Drawing A-90015-0-063 F, rev 5.

"The ORR Decay Tank, Dwg. 10000-T-0460-D, Sheet 30, coordinate N-22738 E31660, was ruptured and in March 1974 its contents were found to be leaking into the surrounding earth. The tank was removed, repaired, and reinstalled during April 1974."

Environmental Considerations. Transport of contaminants by groundwater is the major environmental concern of this incident. Excavation performed in this area could cause surface water transport if contaminants were exposed; therefore, provisions for diverting runoff or surface water should be included in any plans for excavation in the area.

3.8 ORNL Main Complex: Base of 3019 Stack

Site Description. Uncertainties exist as to the area intended for investigation under the heading: Base of 3019 Stack. Discussions with Building 3019 Operations and Remedial Action Program personnel have not resolved the question totally. Although no definitive resolution was reached, personnel generally reflect the belief that the area of concern is actually now referred to as the "3019 Hot Bank" located south of Building 3020 and west of Building 3091 (the building that houses the 3019 stack).

Building 3019 stack emissions may have contributed to the overall contamination of the bank; however, other contributors to contamination in

this area include low-level waste (LLW) line leak sites located in this area. Specifically, LLW sites 1, 2 and 3 as reported in RAP-8 (Grimsby, January 1986).

Soil analyses taken in August 1985 of the 3019 bank are summarized in Table 2. The approximate locations of the soil samples and surface readings recorded at that time are indicated on Figure 10.

Environmental Considerations. The 3019 bank has been a source of surface and surface water contamination. Groundwater contamination is also probable. Since the bank is a composite of several contaminating sources, including LLW line leaks and probably stack emissions, remedial actions will need to address the area as a whole in addition to the known individual sites. As in the case with all potential subsurface contamination, subsurface disturbance activities will necessitate active surface water diversion/control.

Table 2. Soil Analyses for the 3019 Bank Area, June 12, 1985

ACTIVITY (bq/kg)	LOCATION[a] A	B	C	D
Gross Alpha	1.7 E5 (.1 E5)[b]	2.7 E4 (.4 E4)	1.7 E3 (.9 E3)	1.0 E3 (.9 E3)
Gross Beta	4.1 E5 (.1 E5)	1.5 E5 (.1 E5)	1.6 E3 (.5 E3)	1.8 E3 (.8 E3)
Pu-239	2.0 E4 (.1 E4)	42 (6)	13 (1)	50 (6)
Pu-238	1.2 E3 (.1 E3)	74 (8)	1.2 (.3)	1.3 (.5)
Am-241	700 (90)	3.6 E3 (.2 E3)	23 (2)	25 (2)
Cm-244	3.2 E4 (.2 E4)	3.2 E4 (.2 E4)	130 (10)	27 (2)
Cs-137	6.3 E5 (.4 E5)	1.4 E5 (.1 E5)	640 (40)	870 (60)
Co-60	1.4 E3 (.2 E3)	690 (100)	14 (3)	3.1 (2.6)

[a] Approximate soil sample locations are shown on Figure 10.

[b] Degree of uncertainty (+/-) of analysis.

SOURCE: Data sheet furnished by C. H. Miller, November 1985.

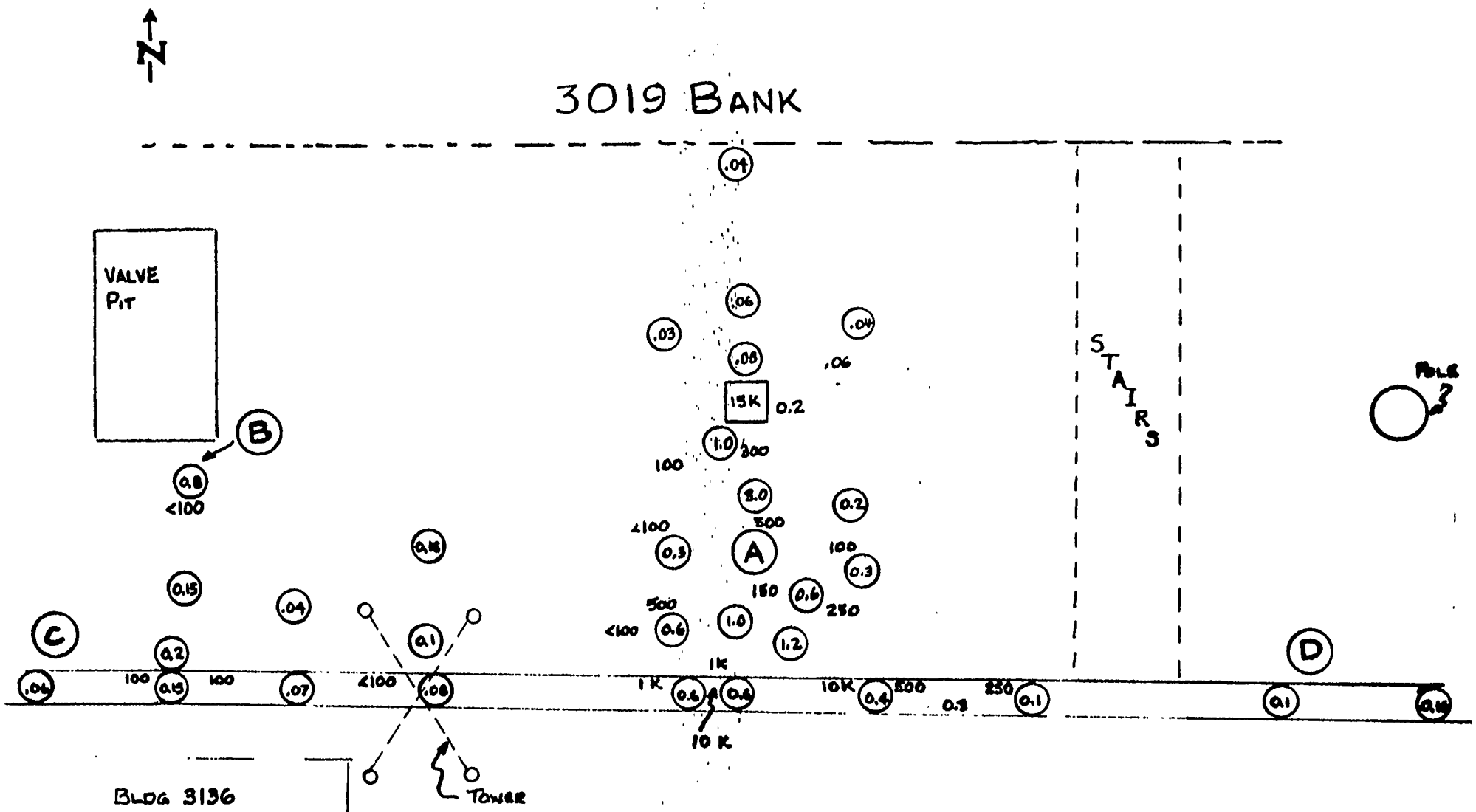

Figure 10. Location of Soil Samples and Soil Surface Readings for 3019 Bank

4.0 DISCUSSION AND RECOMMENDATIONS

With the exception of the storage tank at 9201-3 (Y-12), the nature and extent of contamination should be determined for the SCFP sites [the 9204-1 East End Basement and Attic (Y-12) and the 3503 Storage Pad (ORNL)]. The East End Basement and the 3503 Storage Pad are potential sources of surface water and groundwater contamination. The storage tank at 9201-3 stored cutting oil of an unknown nature; contamination has not been confirmed from scoping surveys and to-date no problems relating to disposal of the tank has been determined.

The Overflow of the Graphite Reactor Storage Canal was unconfirmed from review of old laboratory records; therefore, the location, nature, and extent of contamination of the event, if it occurred, is unknown and would need to be determined. If the incident did occur, groundwater transport is the major concern of the undisturbed site. Similarly, the nature and extent of contamination should be determined for the ORR Decay Tank rupture. Groundwater transport is the environmental concern of this undisturbed site as well. Surface water contamination could occur from excavation or drilling which occurred in either area; therefore, if drilling or other ground disturbing activities in these areas are planned, equipment and procedures for diverting runoff from the site should be readily available.

The 1959 explosion was followed by cleanup activities in the general 3019/3001 area. Most of the contamination was removed by the decontamination activities; the contamination which remained on the building surfaces was "fixed" using various paints. Demolition or closure activities of the buildings in this area (existing at the time of the explosion),

would require further surveys before disposal. Precautions would be necessary to limit direct contact or air born emissions. The 3019 bank (Base of 3019 stack) is part of the general 3019 area contamination which has major contamination components from LLW line leaks. Releases to surface water and groundwater pathways from the area should be addressed as part of the 3019 area characterization. Site characterization planning for the ORNL Main Complex WAG should consider the presence of the ORNL sites described herein, such that proper procedures can be employed to restrict runoff from subsurface ground disturbance (i.e. well drilling, excavation) performed near these sites. In addition, the presence of these sites in the ORNL Main Complex WAG should be considered when evaluating potential site mitigative technologies.

5.0 REFERENCES

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ORNL/RAP/LTR-86/55

DATE: August 29, 1986

SUBJECT: Inventory of ORNL Remedial Action Sites:
13. Abandoned Injection Wells

TO: T. E. Myrick

FROM: H. J. Grimsby

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Inventory of ORNL Remedial Action Sites: 13. Abandoned Injection Wells*

H. J. Grimsby

August 29, 1986

Energy Division
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37831

Operated by

Martin Marietta Energy Systems, Inc.
for the
U.S. Department of Energy

*Based on work performed at Oak Ridge National Laboratory, operated by Martin Marietta Energy Systems, Inc., under Contract No. DE-AC05-84OR21400 with the U.S. Department of Energy.

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1.0 OBJECTIVES

The primary purpose of the remedial action task is to provide corrective measures and facility decommissioning necessary to place the Oak Ridge National Laboratory (ORNL) in compliance with applicable environmental and health regulations for radioactive and hazardous chemical wastes. Remedial Investigation and Feasibility Studies of the sites will be addressed within Waste Area Groupings (WAGs) such that decontamination prescribed to control the potential for human exposure and environmental impact from each site is integrated into the program as a whole. Remedial actions will be conducted at locations where past activities have resulted in contamination of facilities or the environment that necessitate corrective actions or decommissioning. Remedial actions are divided into two activities: (1) Site Decommissioning Program: Surplus Contaminated Facilities Program (SCFP) and Surplus Facilities Management Program (SFMP) and (2) Site Corrective Measures Program (SCMP).

Remedial Action Program (RAP) sites at ORNL comprise 13 categories, listed in Table 1, within the SCFP, SFMP, and SCMP programs. Category 13, Abandoned Injection Wells, is comprised of the wells located on four different sites at ORNL.

As part of the RAP site identification task, a report summarizing information about each site is to be prepared; however, hydrofracture information of this type has been reported in "Closure Plan for ORNL Hydrofracture Operations" (Myrick and Stow, Unpublished) and "Background

Document for Application for an Underground Injection Permit - DRAFT* (Baldwin, unpublished). To avoid duplicating much of this information, summaries of the background information (process, locations, well and waste inventories, operational problems of consideration, and regulatory status) are provided along with a bibliography (sources of more detailed information). Environmental considerations and data deficiencies related to final closure of the injection wells are summarized with emphasis placed on known and potential operational deviations.

Table 1. Inventory of ORNL Remedial Action Sites

1. Solid Waste Storage Areas
2. LLW Seepage Pits and Trenches
3. Process Ponds
4. White Oak Creek Watershed
5. LLW Line Leak Sites
6. Environmental Research Areas
7. Hazardous Waste Sites
8. Radioisotope Processing Facilities
9. Experimental Reactor Facilities
10. Radwaste Facilities
11. Research Facilities
12. Other Contaminated Sites
13. Inactive Injection Wells

2.0 BACKGROUND

2.1 Process

Hydrofracture is a process developed at the ORNL for disposing locally-generated liquid low-level waste (LLW). Hydraulic pressure is used to initiate the formation of a fracture between layers of shale; LLW solution is mixed with solids composed of cement and other additives; and the mixture is injected under pressure into the fractured shale.

2.2 Site Locations

Four sites located in Melton Valley of the ORNL (as shown in Fig. 1) were used for hydrofracture. The hydrofracture process was developed through a series of experiments between 1959 and 1965 at three experimental sites: (1) Experimental Site No. 1 (4-acre site), (2) Experimental Site No. 2, and (3) the Old (or Original) Hydrofracture Facility (OHF). [See de Laguna et al. for experimental injection details.] The OHF was modified in 1966 and used for routine disposal of liquid LLW until 1980. The New Hydrofracture Facility (NHF) operated from 1982 to 1984 for routine disposals.

2.3 Waste Injection Summary

The waste injection summaries for individual injections conducted at each of the four sites are contained in Tables 2 and 3. The primary focus for waste characterization was radionuclide concentration. Grouts used for

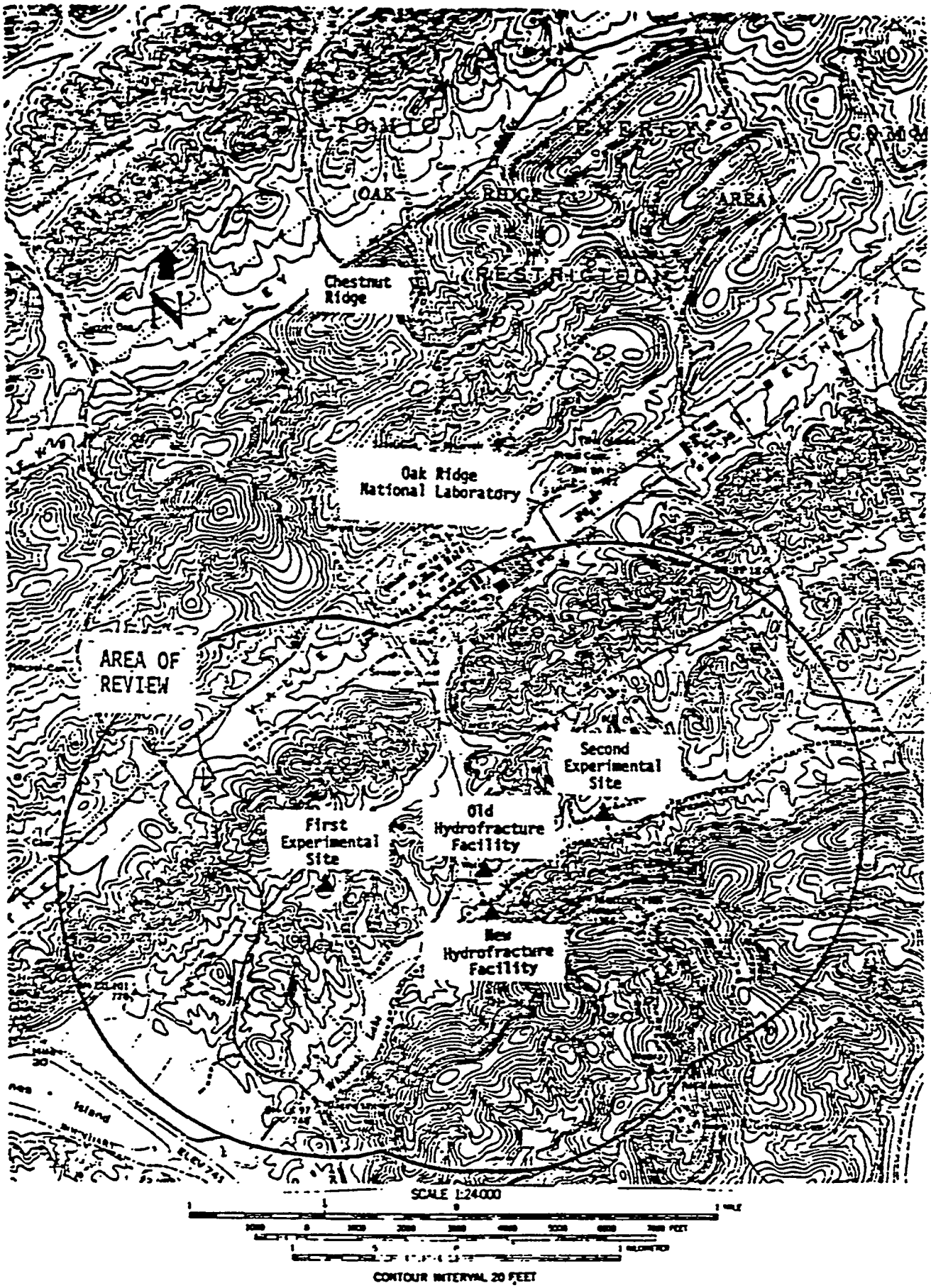

Fig. 1. Location of hydrofracture sites at ORNL. Circles around each site

Table 2. Experimental and OHF Operational Injection Parameters

INJECTION	DATE	DEPTH (ft)	WASTE VOLUME (gal)	WASTE PLUS WATER VOLUME (gal)	GROUT VOLUME (gal)	MIX RATIO[a]	^{90}Sr[b] (Ci)	^{137}Cs[c] (Ci)	^{244}Cm (Ci)	^{239}Pu (Ci)	Other Activity (Ci)
Experimental Site No. 1											
1	10/59	290			27,000			35			8.7 ^{141}Ce
Source: de Laguna et al. 1968.											
Experimental Site No. 2											
1	Sept. 1960	934			91,500			25			
2	1960	694			132,500			25			
Source: de Laguna et al. 1968.											
OHF Experimental Injections											
1	2/13/64	945		40,300	43,383	0.63					
2	2/20/64	924		33,400	42,891	5.87					30 ^{198}Au
3	4/8/64	912		43,090	67,944	12.8	1	74			
4	4/17/64	900		39,070	60,137	9.5	1	50			
5	5/28/64	890		153,683	217,468	6.6	608	193			
6A	5/19/65	880		23,200	24,470	5.7	330	1,562			
6B	5/22/65	872		68,000	96,800						
7	8/16/65	872		86,550	124,961	6.8	492	3,358			
	Totals of test injections			487,293	678,054						
Source: de Laguna et al. 1971.											
OHF Operational Injections											
ILW-1A	12/12/66	872	36,000	69,931	95,197	6.2	3	19,950	NA[d]	NA	
ILW-1B	12/13/66	872	26,000								
ILW-2A	4/20/67	862	86,000	164,800	230,405	6.1	1,050	58,500	NA	NA	
ILW-2B	4/24/67	862	62,000								

Table 2. Experimental and OHF Operational Injection Parameters (Cont'd.)

INJECTION	DATE	DEPTH (ft)	WASTE VOLUME (gal)	WASTE PLUS WATER VOLUME (gal)	GROUT VOLUME (gal)	MIX RATIO[a]	^{90}Sr[b] (Ci)	^{137}Cs[c] (Ci)	^{244}Cm (Ci)	^{239}Pu (Ci)	Other Activity (Ci)
ILW-3A	11/28/67	862	31,000	99,050	146,751	5.5	9,000	17,000	NA	NA	
ILW-3B	11/29/67	862	52,000								
Water Test	12/13/67	852									
ILW-4A	4/3/68	852	24,010	97,090	130,675	5.1	4,300	51,900	NA	1.10	
ILW-4B	4/4/68	852	62,180								
ILW-5	10/30/68	842	81,800	87,110	115,174	5.6	500	69,400	NA	1.15	
ILW-6	6/11/69	842	79,350	91,750	126,331	5.4	8,900	89,000	NA	0.24	
ILW-7	9/23/70	842	83,000	107,650	145,670	5.5	2,747	44,833	19.2	1.77	
Source: de Laguna et al. 1971.											
ILW-8	9/29/72	832	72,700	81,400	108,605	7.3	45	28,000	0.20	0.13	
ILW-9	10/17/72	832	68,300	75,600	114,000	7.8	231	23,400	6.51	None	
ILW-10	11/8/72	832	84,760	93,570	132,960	7.1	1,330	18,800	26.67	0.37	
ILW-11	12/5/72	832	75,760	82,110	125,490	7.2	1,100	23,500	155.74	None	
ILW-12	1/24/75	822	25,710	30,100	42,100	6.6	1,324	12,752	1.02	None	
ILW-13	4/29/75	822	81,000	85,900	126,100	6.3	3,368	35,750	17.83	0.03	
ILW-14	6/20/75	822	82,970	92,470	138,700	6.7	2,874	30,592	3.58	None	
ILW-15	6/30/77	822	91,000	104,000	145,037	5.5	138	26,390	None	0.66	
ILW-16	11/17/77	812	55,200	59,200	79,500	7.2	1,618	14,964	None	None	
ILW-17	9/1/78	802	82,300	89,500	137,500	6.7	90	22,270	2.27	0.07	
ILW-18	5/18/79	792	83,014	97,434	139,000	6.4	28	16,880	0.19	0.29	
TOTAL ILW			1,426,054	1,653,374	2,323,907		38,640	603,881			

[a]lb SOLID/gal LIQUID.
[b]^{90}Sr IS IN EQUILIBRIUM WITH ^{90}Y. FOR EVERY Ci OF ^{90}Sr, THERE IS A Ci OF ^{90}Y.
[c]^{137}Ca IS IN EQUILIBRIUM WITH ^{137m}Ba. FOR EVERY Ci OF ^{137}Cs, THERE IS A Ci OF ^{137m}Ba.
[d]NA - NOT ANALYZED.

Source: Baldwin, unpublished.

Table 3. NHF INJECTION PARAMETERS

INJECTION	DATE	WASTE VOLUME GAL	GROUT VOLUME GAL	ACTIVITY INJECTED (Ci)				
				TRU	^{244}Cm	$^{90}Sr^{a}$	$^{137}Cs^{b}$	OTHER
ILW-19	JUN 16-17, 1982	160,000	228,000	2	5	156	17,330	347
SI-1	AUG 10-15, 1982	192,000	315,000	72	710	28,500	5,500	2,000
SI-2	SEP 23-24, 1982	116,000	154,000	73		57,200	4,800	1,400
SI-3	OCT 26-29, 1982	248,000	309,000	290	510	61,000	4,100	1,800
SI-4	APR 8-10, 1983	194,000	244,000	130	96	11,000	450	230
SI-5	MAY 17-18, 1983	158,000	164,000	65	76	7,200	410	160
ILW-20	JUN 14-15, 1983	111,000	155,000	14	53	3,266	7,140	627
SI-6	JUL 12-14, 1983	204,000	224,000	240	1,060	67,553	2,750	930
SI-7	AUG 9-10, 1983	162,900	189,800	84	220	21,630	2,585	160
SI-8	OCT 25-26, 1983	196,000	242,000	357	2,980	217,400	14,800	3,400
SI-9	DEC 1-2, 1983	190,600	238,600	404	920	125,000	16,200	990
SI-10	JAN 25-27, 1984	185,000	250,000	375	763	41,100	5,600	760
ILW-21	JAN 27-28, 1984	122,000	160,000	19	71	3,500	2,100	510
TOTALS		2,239,500	2,873,400	2,125	7,464	644,505	83,765	13,314

[a] ^{90}Sr IS IN EQUILIBRIUM WITH ^{90}Y. FOR EVERY Ci OF ^{90}Sr, THERE IS A Ci OF ^{90}Y.

[b] ^{137}Cs IS IN EQUILIBRIUM WITH ^{137m}Ba. FOR EVERY Ci OF ^{137}Cs, THERE IS A Ci OF ^{137m}Ba.

Source: Baldwin, unpublished.

the injections were formulated on the basis of (1) compatablility with waste solution to be injected, (2) ability to be pumped over extended periods of time, (3) high retention of associated water upon setting, and (4) providing a relatively low leach rate such that radionuclide migration is retarded. In theory, the proportions of the different grout ingredients could be adjusted to allow for variations in the composition of the waste solution being injected; however, essentially the same mix composition was used for all injections (except for incidents where proper mix operations may not have been maintained). A typical range of compositions used during operational injections is shown in Table 4. The range resulted primarily from differences in component weights as delivered by suppliers.

2.4 Wells

Three basic types of wells are present at the OHF and NHF sites: an injection well for injecting waste grout, observation wells for determining grout sheet orientation, and rock cover monitoring wells for verifying shale impermeability. The observation wells are cased and cemented; the wells extend into (and usually through) the disposal zone. The rock cover monitoring wells are cased and cemented to a depth of approximately 500-600 ft with about 100 ft of open hole below the case. The three basic well configurations are shown in Fig. 2, and a summary of the well inventories at each site is contained in Table 5. Selected data for the wells at each site are presented in Tables 6, 7, 8, and 9.

Table 4. Composition Range of Dry Solids Mix

Ingredient	Weight Percent
Cement (Type I)	35.2 - 41.4 (38.5 desired)
Fly Ash	33.2 - 44.1 (38.5 desired)
Drilling Clay	13.2 - 16.9 (17.4 desired)
Pottery Clay	7.1 - 8.5 (7.7 desired)
Retarder	0.04 - 0.045 (0.05 desired)

Source: Weeren et al. May 14, 1982.

2.5 Operational Problems of Consideration

The operational problems of interest to well closure activities involve:

(1) Waste-Solids Mix Ratio. Unpublished records show unacceptably lean mix ratios in several injected waste-grout mixes. Variability in grout characteristics, as a result of lean or inadequately blended liquid and solid constituents, may have led to less than ideal waste-form stability in the disposal formation. (Myrick and Stow, unpublished)

(2) Injection Well Plugging. The NHF injection well became plugged in 1982 when the tubing string inside the well became cemented to the well casing as a result of grout leaking through cracks in the stringer piping. The failed well was recovered through drilling operations; however, the original injection well casing was breached and a new configuration established. (See Weeren et al. April 1984 for details.)

(3) Loss of Observation Well Integrity. Injected grout ruptured and, in some cases, plugged well casings at both the OHF and NHF. Contaminated water is reported to have leaked up one well at the OHF resulting in surface contamination (Myrick and Stow, unpublished).

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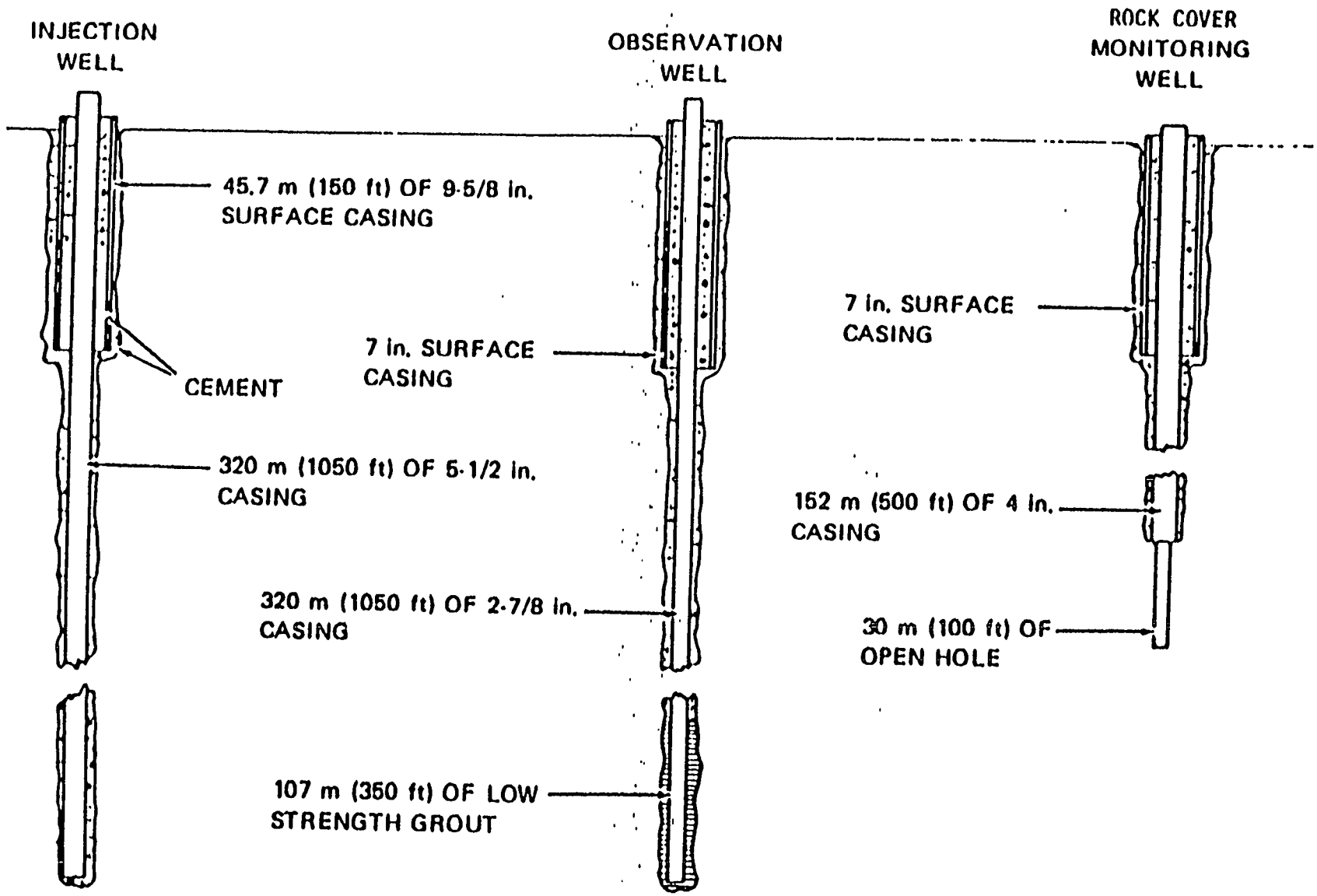

Fig. 2. Schematic of typical well construction for hydrofracture operations.

Table 5. Well Inventory Summary

Site	Type	Number	Depth (ft)	Status
4 Acre	Injection	1	300	Capped
	Core holes	24	150-300	Open
Test 2	Injection	1	now 660	Plugged
	Monitoring	1	1050	Plugged
	Core holes	23	800-1000	Open
OHF	Injection	1	now 782	Capped
	Observation	5	1000	Capped
	Observation	3	1000	Plugged
	Test	1	3200	Plugged
	Rock cover	9	550-650	Capped
	Pilot	4	500	Plugged
	Core	1	1040	Capped
NHF	Injection	1	now 890	Capped
	Observation	6	1000	Capped
	Rock cover	8	600	Capped
	Deep monitoring	7	750-1505	Capped

Source: Myrick and Stow, unpublished.

Table 6. Data on Wells at Experimental Site No. 1

Well	ORNL Coordinates N	E	Elevation ft	Depth ft	Diameter in.	Casing	Cemented	Deviation	Other
Injection	18920	25890	816	300	6	3-1/2 J55	yes	unknown	Slotted at 290 ft
N-199	19119	25890	820	200	3	surface	no	unknown	^{137}Cs contaminated
N-231	19151	25890	820	200	3	surface	no	unknown	Grout sheet
N-343	19263	25890	825	165	3	surface	no	unknown	Grout sheet
N-399	19319	25890	825	165	3	surface	no	unknown	Grout sheet
N-449	19369	25890	827	147	3	surface	no	unknown	---
NE-359	19173	26143	820	200	3	surface	no	unknown	Grout sheet
NE-410	19210	26180	825	200	3	surface	no	unknown	---
E-37	18920	25927	817	307	3	surface	no	unknown	Grout sheet
E-199	18920	26089	800	200	3	surface	no	unknown	Plugged with sand
E-273	18920	26163	797	287	3	surface	no	unknown	Grout seam
E-362	18920	26252	810	330	3	surface	no	unknown	---
S-88	18832	25890	813	360	3	surface	no	unknown	Grout seam
S-134	18786	25890	810	300	3	surface	no	unknown	---
S-202	18718	25890	805	300	3	surface	no	unknown	Plugged with sand
W-79	18920	25811	805	300	3	surface	no	unknown	Grout seam
W-127	18920	25763	800	340	3	surface	no	unknown	---
W-184	18920	25706	800	340	3	surface	no	unknown	---
W-200	18920	25690	800	200	3	surface	no	unknown	Plugged with sand
W-237	18920	25653	800	325	3	surface	no	unknown	---
W-309	18920	25581	808	333	3	surface	no	unknown	---
NW-259	19156	25790	813	154	3	surface	no	unknown	Grout sheet
NW-309	19171	25710	822	150	3	surface	no	unknown	Grout sheet
Terry	19132	25884	820	220	3	surface	no	unknown	Grout sheet. Drilled 10/82
Joy	19146	25905	820	220	3	50 ft 4 in.	no	unknown	Grout sheet. Drilled 2/84

Source: Baldwin, unpublished.

Table 7. Data on Wells at Experimental Site No. 2

Well	ORNL Coordinates N	E	Elevation ft	Depth ft	Diameter in.	Casing	Cemented	Deviation	Other
Injection	16817	31260	790	940	8	4-1/2	yes	none	
	Slotted at 934 ft and injection made. Slotted at 694 and injection made. Shaped charge tests at 686, 676, and 661. Water injected.								
N-200	17013	31287	800	815	2	surface	no	180 ft N 17 W [ORNL]	Grout sheets
N-400	17217	31290	824	780	2	surface	no	110 ft N 35 W at bottom	Lower grout sheet
N-600	17410	31290	834	700	2	surface	no	145 ft N 12 W at bottom	
NE-200	16918	31441	793	840	2	surface	no	N 16 W 90 ft at 580	Upper grout sheet
NE-300	16962	31515	799	860	2	surface	no	N 14 W 70 ft at 694	Upper grout sheet
NE-400	17026	31600	799	820	2	surface	no	N 44 E 50 ft at 560	Upper grout sheet
NE-500	17073	31690	796	810	2	surface	no	N 19 W 40 ft at 545	Upper grout sheet
E-100	16793	31355	788	910	2	surface	no	N 15 W 100 ft at 600	Both grout sheets water injection
E-200	16772	31450	789	900	2	surface	no	N 3 W 95 ft at 620	Upper grout sheet
E-400	16718	31645	794	900	2	surface	no	N 14 W 140 ft at 630	Upper grout sheet
E-500	16686	31750	798	900	2	surface	no	N 9 W 135 ft at 660	Upper grout sheet
E-600	16666	31849	794	900	2	surface	no	N 3 E 145 ft at 662	
S-100	16725	31237	784	894	2	surface	no	N 12 W 140 ft at 890 Silted to 300 ft	Both grout sheets Water injection
S-200	16615	31237	775	935	2	surface	no	N 17 W 160 ft (bottom)	Lower grout sheet Water injection, silted
S-400	16433	31150	780	960	2	surface	no	North 320 ft (bottom)	Water injection
SW-200	16719	31082	773	900	2	surface	no	N 14 W 100 ft at 630	Water injection

Table 7. Data on Wells at Experimental Site No. 2 (Cont'd.)

Well	ORNL Coordinates N	E	Elevation ft	Depth ft	Diameter in.	Casing	Cemented	Deviation	Other
W-15	16825	31244	788	860	2	surface	no	N 17 W 120 ft at 589	
W-30	16832	31227	788	1050	3	see below	yes	unknown	Lower grout sheet
	from top - 892 ft 3/4 in. pipe cemented in place, 150 ft perforated 1-1/2 in. pipe below								
W-100	16840	31166	784	840	2	surface	no	N 5 E 130 ft at 588	Lower grout sheet
W-200	16868	31062	783	840	2	surface	no	N 5 E 125 ft at 600	Water injection
W-400	16915	30875	785	840	2	surface	no	N 16 E 45 ft at 625	
W-600	16967	30677	773	840	2	surface	no	N 18 W 65 ft at 615	
NW-200	17012	31160	800	820	2	surface	no	unknown	Lower grout sheet
NW-400	17166	31054	817	780	2	surface	no	unknown	

Source: Baldwin, unpublished.

Table 8. Data on Wells at Old Hydrofracture Facility

Well	ORNL Coordinates N	E	Elevation ft	Depth ft	Diameter in.	Casing	Cemented	Deviation	Other
Injection	17155	28617	799	1080		9-5/8 to 150 5-1/2 to 1080	yes	none	Slotted at intervals 988 to 782 ft
N-100	17255	28617	786	1000		1-1/4 in.	yes	to 13°	Drilled 8/64
N-150	17310	28611	786	1050		2-7/8	yes	none	Latex cement used Bottom 300 ft
NE-125	17245	28708	801	1030		1-1/4 in.	yes	<2°	Casing parted (ILW-13) Attempted plugging 5/82
E-320	17013	28921	782	840		2-7/8	yes	none	Drilled summer, 1972 Shaped charge test at 810 ft (12/83)
S-100	17055	28618	786	1032		1-1/4	yes	<2°	Tubing ruptured. Well plugged (700 ft)
S-220(68)	16934	28619	761	1050		1-1/4	yes	N 11° W 20°	Tubing ruptured (ILW-5)
S-220(72)	16938	28596	761	820		2-7/8	yes	none	Drilled summer, 1972
W-300	17115	28317	761	3263		1-1/4 in. to 2900 ft	yes	none	Cored, plugged during '75 series. Plugged May 1982
NW-100	17224	28548	778	1000		1-1/4 in.	yes	to 14°	
N-200RC	17360	28646	792	650		4 in. to 500	yes	N 5° W 27 ft at 600	Drilled 1963
N-275RC	17430	28616	777	600		4 in. to 550	yes	N 18° E to 13.5°	
NE-200RC	17300	28762	806	600		4 in. to 500	yes	N 44° E to 4.5°	50 ft open hole
NE-125RC	17227	28723	800	600		4 in. to 500	yes	N 26° W to 9°	
E-300RC	17003	28893	783	650		4 in. to 600	yes	N 4° W	50 ft open hole
S-200RC	16954	28619	762	600		4 in. to 550	yes	N 31° E to 11°	
W-300RC	17109	28327	761	650		4 in. to 600	yes	N 31° E to 34°	50 ft open hole
NW-175RC	17278	28494	772	570		4 in. to 500	yes	N 7° W to 15°	
NW-250RC	17330	28441	760	600		4 in. to 540	yes	E 12° S to 10°	60 ft open hole
N-375P	17522	28608	772	500		4 in.	yes	unknown	

Table 8. Data on Wells at Old Hydrofracture Facility (Cont'd.)

Well	ORNL Coordinates N	E	Elevation ft	Depth ft	Diameter in.	Casing	Cemented	Deviation	Other
NE-300P	17378	28839	816	500		4 in.	yes	unknown	
E-320P	16999	28924	782	500		4 in.	yes	unknown	
NW-290P	17358	28412	759	500		4 in.	yes	unknown	
NE-100	17225	28690	795	1040	7	4 in. to 500 ft	yes		Cored - 500 to 1040 ft

Source: Baldwin, unpublished.

Table 9. Data on Wells at New Hydrofracture Facility

Well	ORNL Coordinates N	E	Elevation ft	Depth ft	Diameter in.	Casing	Cemented	Deviation	Other
Injection	16502	28178	789	990		9-5/8 to 250 5-1/2 to 750 2-7/8 to bottom	yes	N 37 W 150 ft	Details in ORNL-TM-4713, ORNL/TM-8823
N-200 (OW-4)	16702	28178	771	1072		2-7/8	yes	N 7 E 215 ft	Intercepted grout Sheets from old facility
E-200 (OW-3)	16503	28378	788	1130		2-7/8	yes	N 35 E 125 ft	1-1/4 in. pipe set and cemented to 1130
SE-150 (OW-5)	16445	28308	794	1145		4-1/2	yes	N 29 E 130 ft	
S-200 (OW-2)	16304	28179	799	1168		2-7/8	yes	N 34 E 225 ft	Cored
W-200 (OW-1)	16502	27978	782	1127		2-7/8	yes	N 38 E 205 ft	
NW-400 (OW-6)	16702	27923	770	910		2-7/8	yes	Unknown	Casing ruptured at 900 ft
N-200RC (17)	16702	28158	770	580		4 in. to 480	yes	5 ft	Open hole filled with sand and seismic sensors
NE-280RC (13)	16673	28367	765	580		4 in. to 480	yes	N 15 W 40 ft	
E-200RC (16)	16483	28377	785	650		4 in. to 550	yes	30 ft NW	
SE-280RC (12)	16313	27879	810	700		4 in. to 600	yes	N 22 W 130 ft	
S-200RC (15)	16304	27231	800	700		4 in. to 600	yes	S 27 E 140 ft	
SW-280RC (11)	16387	72929	785	700		4 in. to 600	yes	N 37 W 90 ft	
W-190RC (14)	16489	27991	785	650		4 in. to 550	yes	N 43 W 118 ft	Open hole filled with sand and seismic sensors
NW-340RC (10)	16702	27896	770	580		4 in. to 480	yes	N 55 W 33 ft	

Source: Baldwin, unpublished.

2.6 Regulatory Status

The NHF injection well does not meet the standards established by the Tennessee Department of Health and Environment (TDHE) and the Environmental Protection Agency (EPA) for injection of hazardous materials. Therefore, the NHF will be closed under the requirements of the Resource Conservation and Recovery Act (RCRA) and the Underground Injection Control (UIC) regulations. A closure plan (Myrick and Stow, unpublished) has been written which addresses the four hydrofracture sites.

3.0 ENVIRONMENTAL CONSIDERATIONS

The hydrofracture grouts were designed to mix the LLW with cement and other solids, such that after injection, the mix would harden fixing the solids within the host rock. For those injections where acceptable mix ratios were utilized, the solids should be incapable of moving and surface or groundwater contamination as a result of migration of the solids is unlikely.

No gaseous fission products would be expected (trace amounts present in the LLW solution, if released, could be retained by the surrounding shale).

Water would be the major vehicle for transport of the hazardous and radioactive materials injected. There are two major potential sources of water: (1) water contained in the waste, and (2) groundwater contact from water contained by the shale or either upper or deep aquifers. The primary source of groundwater contact is believed to be from the water contained by the surrounding shale (Stow, personal communication).

A credible long-term hazard is associated with the possibility that proper mix ratios (and/or proper operating procedures) were not maintained for all injections. Not only could an excess volume of free water be produced with a corresponding possiblity of liquid entrapment, but unset (unhardened) grout could also be present. In both cases, the radionuclides would be mobile (ideally the grout would immobilize the waste and incorporate the liquids) and the only barrier to transport would be the ion exchange capacity of the surrounding shale. Any groundwater contact could potentially leach the grout sheets.

Radionuclide contamination, primarily ^{90}Sr, has been detected in wells DM-2 and DM-1. The nature of contamination and the depth at which contamination is occurring, 950 to 1000 ft, indicate the source is related to NHF operations. Preliminary hydrogeologic characterization at the NHF suggests that radionuclides are present in the groundwater of the injection interval and have migrated an unknown distance from the grout sheet. The composition of the groundwater indicates mixing of natural groundwater and water from injection wells. Water chemistries also suggest the possibility of deep and shallow groundwater mixing (Myrick and Stow, unpublished).

Contamination of wells in contact with the grout sheets will necessitate proper worker protection and decontamination capabilities during remedial investigation activities. Most wells at the OHF and NHF are either capped or plugged at present whereas wells located at Experimental Sites 1 and 2 are not.

Data are needed to confirm the deep and shallow groundwater mixing (including its extent and rate). Data to determine the horizontal and vertical extent of contamination associated with the injections are required. A Remedial Investigation Plan is currently being developed to these ends by RAP personnel. The area of interest, as proposed, will extend horizontally for one mile from each injection well, as shown on Figure 1, and vertically from Rutledge - Rogersville through the upper Rome Foundation which lie immediately over and under the injection zone. Data review will be necessary to determine whether the area of interest, as currently defined, will require expansion.

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