RETURN TO 396-SS

70-824
PDR

Babcock & Wilcox
a McDermott company

Research & Development Division
Lynchburg Research Center

P. O. Box 11165
Lynchburg, Virginia 24506-1165
(804) 522-6000

May 27, 1986

Uranium Process Licensing Section
Uranium Fuel Licensing Branch
NMSS
U. S. Nuclear Regulatory Commission
Washington, DC 20555

'86 JUN -2 A11:02

SUBJECT: Response to Comments

REFERENCE: License SNM-778, Docket 70-824, Renewal Application

Gentlemen:

This is in response to your letter dated March 27, 1986, forwarding comments on the referenced application.

Responses to the sixty-one comments are given in Attachment 1, with the exception of Comment No. 48.a and b. The documents requested in that comment are attached to this letter.

Attachment No. 2 is an instruction sheet to assist in the proper insertion of the revised pages into the renewal application manual.

In Comment 49, you requested larger drawings. The drawings for Buildings A and B showed the two floors of each building on a single drawing. To enlarge the drawings, it was necessary to display a single floor on a drawing, and in the cases of Buildings A and B, I have designated the two drawings with an A and B, as being a two-part drawing, (e.g., Figure 10-1A and 10-1B). In these instances I have not changed the original reference in the text portion, (e.g., Figure 10-1).

The attached replacement pages are designated as Revision No. 2. Changes to the text have been marked with a vertical line in the right margin, opposite the line where a change has been made. Where no change is indicated on a page; preceding changes have caused the page content to be other than what appeared in the original issue, and the new page is given the Revision No. 2 designation for that reason. New figures that have been added in response to the comments are not designated by the vertical line in the margin.

If there are additional questions or clarifications needed, please don't hesitate to call me.

FEE NOT REQUIRED

Very truly yours,

BABCOCK & WILCOX
Lynchburg Research Center

Arne F. Olsen
Senior License Administrator

jhc
Attachments
pc: Region II

27032

DOCKET NO. 70-824

CONTROL NO. 27032

DATE OF DOC. 05/27/86

DATE RCVD. 06/02/86

FCUF ✓ PDR ✓

FCAF _____ LPDR _____

I & E REF. ✓

SAFEGUARDS ✓

FCTC _____ OTHER _____

DATE 06/02/86 INITIAL CEC

American Nuclear Insurers

MAELU

Mutual Atomic Energy Liability Underwriters

CERTIFICATE OF NUCLEAR ENERGY LIABILITY INSURANCE (FACILITY FORM)

This is to certify that there are in force as of the effective date of this Certificate two Nuclear Energy Liability Insurance Policies (Facility Form) one issued by the members of American Nuclear Insurers and the other issued by the members of Mutual Atomic Energy liability Underwriters respectively, to the insured named below with respect to the nuclear facility at the location described below. If either such policy is cancelled or terminated prior to the end of December 31 of the calendar year in which this Certificate becomes effective, 20 days advance written notice thereof will be mailed to the party designated below for whom this Certificate is issued and this Certificate shall thereupon terminate. Otherwise this Certificate shall terminate as of said December 31st. **

Name and Address of Insured:

Babcock & Wilcox Company
Tusc Building
20 S. Van Buren Avenue
Barberton,Ohio 44203

Policy No.(s)	Eff. Date of Policy	Limit of Liability
ANI / NF-111	October 1, 1961	$ 124,000,000.00 *
MAELU / MF- 85	March 7, 1975	$ 36,000,000.00 *

* Subject to all terms of the (respective) policy having reference thereto.

Location of the Nuclear Facility: The Babcock & Wilcox Company's 550 acre plant site, approximately 7 miles east of Lynchburg, Virginia.

Name and Address of Party for whom this Certificate is issued:

U. S. Nuclear Regulatory Commission
Washington, DC

Effective Date of this Certificate: March 12, 1986

AMERICAN NUCLEAR INSURERS — MUTUAL ATOMIC ENERGY LIABILITY UNDERWRITERS

BY: J. B. Quattrocchi
JOHN L. QUATTROCCHI, AUTHORIZED REPRESENTATIVE

DATE March 12, 1986
FARMINGTON, CT

** A Certificate will not be issued for the subsequent calendar year unless requested.

The Exchange, Suite 245/ 270 Farmington Avenue/ Farmington, Connecticut 06032/ (203) 677-7305
Eng. Dept. (203) 677-7715/ TLX. No. 643-029

Arkwright-Boston Insurance
Waltham, Massachusetts 02154

INSPECTION REPORT

MAERP RA

MAERP REINSURANCE ASSOCIATION

FOR MCDERMOTT INCORPORATED (THE BABCOCK & WILCOX COMPANY-RESEARCH CENTER)
Rte. 460
Lynchburg, Virginia 24505

INDEX 45336.30

BY C. D. Ratchford

ACCOUNT 2-05000

CONFERENCE WITH Mr. C. Bell, Facility Manager

ON March 26,1986

HOUSEKEEPING	is	Satisfactory	AREAS SPRINKLERED	Yes	% Additional Adequate	No 60	% Needed Lbs. Pressure
MAINTENANCE & REPAIR	is	Satisfactory	WATER SUPPLY	is			
SUPERVISION FIRE EQUIPMENT	is	Satisfactory	ALL VALVES FOUND OPEN	were			
P.E.O. & WATCHMEN	is	Satisfactory	CRITICALITY CONTROL			is	Satisfactory
RADIOISOTOPE HANDLING	is	Satisfactory	NUCLEAR REACTOR OPERATION			is	Satisfactory

SUMMARY

Management continues to express excellent interest in loss prevention at this nuclear research center.

GENERAL REMARKS

Mr. F. Dreamer, Maint., makes weekly recorded inspections of the fire protection equipment and these are reviewed by Mr. Dick Younger. Mr. Dick Younger is also in charge of proper valve supervision. Mr. J. P. Doran, Emergency Officer, is in charge of the well trained Plant Emergency Organization.

The "Feed" Building has been cleaned of all contaminants, but remains vacant. It is not known what this building will be used for.

A 1000 kVA transformer located along the north outside wall of the NFL Building contains PCB's and this transformer is to be replaced in the near future.

PRINTED IN USA

Prepared by Factory Mutual Engineering Association

ATTACHMENT 1
RESPONSE TO COMMENTS

SECTION I

1. All license conditions that were imposed at the time of the last renewal and subsequent amendments have been reviewed and are incorporated in this application with the exception of Conditions 9 and 11.

   Condition 9 incorporated the last renewal application and the revisions to that application. Condition 9 was not incorporated because the review process is still in progress and tne final revisions have not been determined.

   Condition 11 requires sealed sources to be leak tested in accordance with Commission guidance which was attached to the renewed license. Regulatory Guide 3.52, which was used for the development of the current renewal application does not make reference to the inclusion of Condition 11. It was assumed that a license condition and new guidance will also be included in the license at the time of renewal.

2. See Section 2.7.2.

SECTION II

1. FIGURES 1-1, 2, and 3 have been added to Section 1.

2. Items 1.4.1 and 1.4.2 have been changed to incorporate the requested confirmation.

3. Section 3.1.3 has been added. This condition requires the same control as is applied by the present license.

4. Section 1.6.10 has been changed in response.

5. Section 1.7.1 has been changed in response.

6. Section 1.7.2 and 1.7.4 have been deleted. The activities authorized by these two sections are specified in the regulations and are not necessary as license conditions.

7.a Section 1.8.1.1 has been revised to clarify that the maximum period between in vivo counting shall not exceed 12 months.

7.b Section 1.8.1.2 has been deleted.

8. Sections 2.2.4, 2.2.6, 2.2.9, and 2.3.1.1 have been revised in response to the comment.

9. Section 2.2.6 has been revised to respond to the comment.

10.a Assuming the "LRC" is an error and should have been "NRC," Section 2.3.1.5 has been revised in response to the comment.

10.b Section 2.3.1.5 has been revised in response to the comment.

11. This comment is similar to Comment I.2 The response is addressed in Section 2.7.2.

12.a This comment is responded to in Section 2.3.3.3 and Section 2.8.3.1.

12.b This comment is responded to in Section 2.3.3.3.

13. This comment is responded to in Section 2.5.12.

14. This comment is responded to in Section 2.5.3.

15. The response to this comment is found in Section 2.5.4.

16. The response to this comment is found in Section 2.5.11.

17. New Section 2.7.1 contains the response to this comment.

18.a Section 2.8.1.2 contains the response to this comment.

18.b In accordance to agreements reached during the March 12-14, 1986 meeting, this comment should have been deleted from the formal comments.

19.a My notes of our March 12-14 meeting reflect that it was understood that the audits referred to in this comment were the Safety Audit Subcommittee audits. Assuming that this understanding is still valid, the response is found in Section 2.8.3.1.

19.b The response to this comment is found in Section 2.8.3.1.

20. The response is found in Section 2.10.2.

21. The response is found in new Section 3.1.1.7.

22. Section 3.1.2.2 was deleted.

23.a Section 3.2.1.1 has been revised to respond to the comment and Section 3.2.1.3 was deleted.

23.b Subsection 3.2.1.2 has been deleted. The contents of 3.2.1.4 was renumbered 3.2.1.2.

24.a 3.2.1.2 (previously 3.2.1.4) has been revised to respond to the comment.

24.b 3.2.1.2 (previously 3.2.1.4) has been revised to respond to the comment.

25. Section 3.2.2.6 has been revised to respond to the comment.

26. Section 3.2.2.12 has been revised to respond to the comment.

27.a Section 3.2.3.1 and the table have been revised to respond to this comment.

27.b The table in section 3.2.3.1 has been revised to respond to the comment.

28. Section 3.2.3.4.1 has been revised to respond to the comment.

29. Section 3.2.4.3.3 is a new section which addresses this comment.

30.a Section 3.2.4.3.1 has been revised to respond to this comment.

30.b Section 3.2.3.4.1 has been revised to respond to this comment.

31. The values given in Section 3.2.4.3.2 have been changed in response to the comment. The new action level is 25% of the maximum permissible organ burden, which is very near the limit of detection for our in vivo counting contractor.

32.a In Section 12.8.5.1, the Table has been revised by adding "+ GAMMA" in response.

32.b The response to 32.a satisfies the comment.

33. Section 3.2.4.6.1 and 12.8.5.1 have been revised to indicate that the specified frequencies are minimum frequencies. When the work load in these areas increase, the frequency of the surveys also increases.

34. Section 3.2.4.8.1 has been revised to address this comment. Sections 3.2.4.8.2, 3, and 4 have been deleted. The response to Comment 45 also applies to Comment 34.

35. This comment was not among those discussed during the March meeting. Section 12.3.4.3 explains when supplemental dosimetry will be used.

36. Section 4.1.2 has been revised to name the position responsible for this function.

37.a A unit is defined in Section 1.6.12.

37.b Section 4.1.4 has been added to respond to the comment.

38.a Section 4.1.6 has been revised to respond to the comment.

38.b Section 4.1.6 has been revised to respond to the comment.

39. Section 4.2.3.6.1.1, Item 3, has been revised to respond to the comment.

40. In Section 5.1.4.4, Item 1, the manufacturer's name has been removed. The manufacturer's name did not appear to be pertinent to a demonstration of safety, therefore, the "type" of instrument is referred to in Section 13.2.

41. Section 6.0 has been revised to refer to Section 1.7.1 which addresses this comment.

42. Sections 7.1.2, 7.1.4, and 7.1.6 have been revised to respond to this comment.

43. Section 7.1.5 has been revised by deleting the reference to the 1/8-inch core sample. The reference to this sample was not intended to establish an action criterion but rather to indicate that core samples are to be used as a sampling method. Actions to be taken to core samples will be specified in a specific decommissioning plan which must be submitted and approved prior to beginning such an operation and the action to be taken will be based on the release criteria in effect at that time.

44. Section 7.2.1 has been revised to refer to the September 16, 1985 letter.

45. Section 8.0 has been revised to incorporate the February 28, 1984 amendment. The Lynchburg Research Center is identified.

46. The Figure 10.5 was mislabeled. This figure should have been labeled Figure 10.6. The drawing meant for Figure 10.5 was left out of the renewal application. New Figures 10.5 and 10.6 have been included in response.

47. Sections 10.4.2.3 and 10.4.2.5 have been revised in response.

48.a The Facility Form for the LRC is on file with Mr. Ira Dinitz, NRC. However, the requested form is attached to the cover letter of this submittal.

48.b The requested inspection report is attached to the cover letter of this submittal.

49. Larger figures have been provided.

50. Section 11.2.7 has been revised to include this recommended information.

51. Section 11.4.1 has been revised in response to the comment.

52. The reference has been corrected. See Section 12.3.1.

53. Section 12.8.1.1, 12.8.1.2, and 12.8.1.3 have been revised in response to the comment.

54. Section 12.8.1.1, 12.8.1.2, and 12.8.1.3 have been revised in response to the comment.

55. Section 12.8.1.4 has been revised in response to the comment.

56. Section 12.8.5.1 has been revised in response to the comment.

57. Section 12.8.5.2 has been revised in response to the comment.

58. Section 12.8.5.3 has been revised in response to the comment.

59. Section 12.10.1 has been revised in response to the comment.

60. Section 12.11 has been revised to provide examples of steps taken to implement ALARA.

61.a Figures 13-1 and 13-2 have been revised to provide the requested information.

61.b Section 13.1 has been revised to reference Tables 2-2 thru 2-6 in the LRC's Environment Report.

ATTACHMENT 2

INSTRUCTION SHEET

RENEWAL APPLICATION

DEMONSTRATION AND CONDITIONS FOR LICENSE SNM-778

REVISION 2 AMENDMENT 0

Remove Pages			Insert Pages		
Page	Rev.	Date	Page	Rev.	Date
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13-1	0	10/85	13-1	2	5/86
-	-	-	13-3	2	5/86
-	-	-	13-4	2	5/86

TABLE OF CONTENTS

Section		Page
1.0	STANDARD CONDITIONS AND SPECIAL AUTHORIZATIONS	1-1
1.1	NAME.	1-1
1.2	LOCATION	1-1
1.3	LICENSE NUMBER AND PERIOD	1-1
1.4	POSSESSION LIMITS	1-2
1.5	LOCATION OF POSSESSION AND USE	1-3
1.6	DEFINITIONS	1-3
1.7	AUTHORIZED ACTIVITIES	1-5
1.8	EXEMPTIONS AND SPECIAL AUTHORIZATIONS	1-5

List of Figures

Figure		Page
1-1	LRC LOCATION WITHIN VIRGINIA	1-6
1-2	LRC, FIVE MILE RADIUS	1-7
1-3	LRC, BUILDINGS	1-8

PART I

LICENSE CONDITIONS

1.0 STANDARD CONDITIONS AND SPECIAL AUTHORIZATIONS

1.1 NAME

Name - McDermott International, Inc.
Babcock & Wilcox
Research and Development Division
Lynchburg Research Center

McDermott International Inc. is incorporated under the laws of the Republic of Panama.

Principle Office - 1010 Common Street, New Orleans, Louisiana.

1.2 LOCATION

Address - Babcock & Wilcox
Lynchburg Research Center
P. O. Box 11165
Lynchburg, Virginia 24506-1165

The Lynchburg Research Center is located in Campbell County, Virginia, near the James River, approximately four miles East of the city of Lynchburg. Figure 1-1 shows the location of the LRC with respect to the Commonwealth of Virginia. Figure 1-2 shows the location of the LRC with respect to a five mile radius. Figure 1-3 shows the location of buildings and facility locations where licensed materials are handled and stored.

1.3 LICENSE NUMBER AND PERIOD

License Number - SNM-778

Docket Number - 70-824

Period of Time - It is requested that this license be renewed for a period of 10 years.

1.4 POSSESSION LIMITS

Material	Physical Form	Enrichment	Amount
1. Uranium enriched in U-235	Encapsulated or irradiated	$\geq$ 20 %	3.5 Kg contained U-235
2. Uranium enriched in U-235	Unencapsulated and unirradiated	$\geq$ 20 %	0.27 Kg contained U-235
3. Uranium enriched in U-235	Encapsulated or irradiated	5 % to <20%	1.2 Kg contained U-235
4. Uranium enriched in U-235	Unencapsulated and unirradiated	5 % to <20%	0.5 Kg contained U-235
5. Uranium enriched in U-235	Encapsulated or irradiated	.711 % to <5%	55 Kg contained U-235
6. Uranium enriched in U-235	Unencapsulated and unirradiated	.711 % to <5%	11 Kg contained U-235
7. Plutonium	Unencapsulated and unirradiated		0.31 Kg
8. Source Material	Any		6000 Kg
9. Fission Products & Transuranium Elements	Irradiated Fuel		Quantity contained in 4 irradiated fuel assemblies.
10. Fission products	Irradiated fuel		5,000,000 Ci.

11.	Any byproduct material	Irradiated structural materials & components	50,000 Ci.
12.	Byproduct material with at. nos. 3 thru 83	Any	3,000 Ci each total not to exceed 1,000,000 Ci.
13.	Transuranium elements	Any	20 milli-curries each
14.	Cf-252	Sealed Sources	4 milligrams
15.	Am-241	Sealed Sources	30 Ci
16.	H-3	Sealed Sources	100 Ci
17.	H-3	Oxide	3 Ci
18.	H-3	Ni plated Sc tritide foil	3 Ci

1.5 LOCATION OF POSSESSION AND USE

1.5.1 Licensed material shall be possessed and used at the Lynchburg Research Center.

1.5.2 Byproduct material in the form of sealed sources with activities of up to 500 millicuries may be possessed and used in locations other than the Lynchburg Research Center for performing instrument calibration, electronic noise analysis, shielding studies, or similar operations.

1.6 DEFINITIONS

1.6.1 LRC means Lynchburg Research Center.

1.6.2 SRC means Safety Review Committee.

1.6.3 SNM means Special Nuclear Material.

1.6.4 Licensed Material means source, byproduct, or SNM received, possessed, used or transferred under a general or specific license issued by the Nuclear Regulatory Commission.

1.6.5 Research and Development (R&D) means (1) theoretical analysis, exploration, or experimentation; or (2) the extension of investigative findings and theories of a scientific or technical nature into practical application for experimental and demonstration purposes, including the experimental production and testing of models, devices, equipment, materials and processes. The administration of licensed material, internally or externally, to human beings is not included in this definition.

1.6.6 Safety Audit Subcommittee (SAS) means the subcommittee established under the SRC to perform audit functions.

1.6.7 Manager, Lynchburg Technical Operations means the Manager of Technical Operations for the Lynchburg Research Center.

1.6.8 Qualified Person means a person who is assigned by his supervisor to work in an area where licensed material is handled and who is familiar with the hazards in the area. A qualified person may also be referred to as a Category A Person.

1.6.9 Authorized User means a person who may work with licensed material unsupervised and may supervise others, not so designated, in the handling of licensed material.

1.6.10 Calibration means a comparison of a measurement standard of known accuracy and traceable to NBS, with another standard or instrument to detect, correlate or adjust any variation in the accuracy of the item being compared. Calibration also includes standardization.

1.6.11 Standardization means, the act of using standards which are traceable to the NBS, a nationally accepted measurement system, or natural phenomena to set up an instrument. Standardization must be performed before and after use.

1.6.12 Unit means (1) a separate laboratory, room, or work area; (2) a transfer cart where SNM is separated from adjacent units by at least 8-inches edge-to-edge and 24-inches center-to-center. More than one unit may be on a cart provided the preceding edge-to-edge

and center-to-center values are maintained, and (3) a processing bench, glove box, furnace, fume hood, or other similar process equipment or container separated from adjacent units by at least 8-inches edge-to-edge and 24-inches center-to-center.

1.6.13 Standing RWP's are Radiation Work Permits issued for a term of more than 30-days, authorizing entry into High Radiation Areas and Airborne Radioactivity Areas to perform routine work.

1.7 AUTHORIZED ACTIVITIES

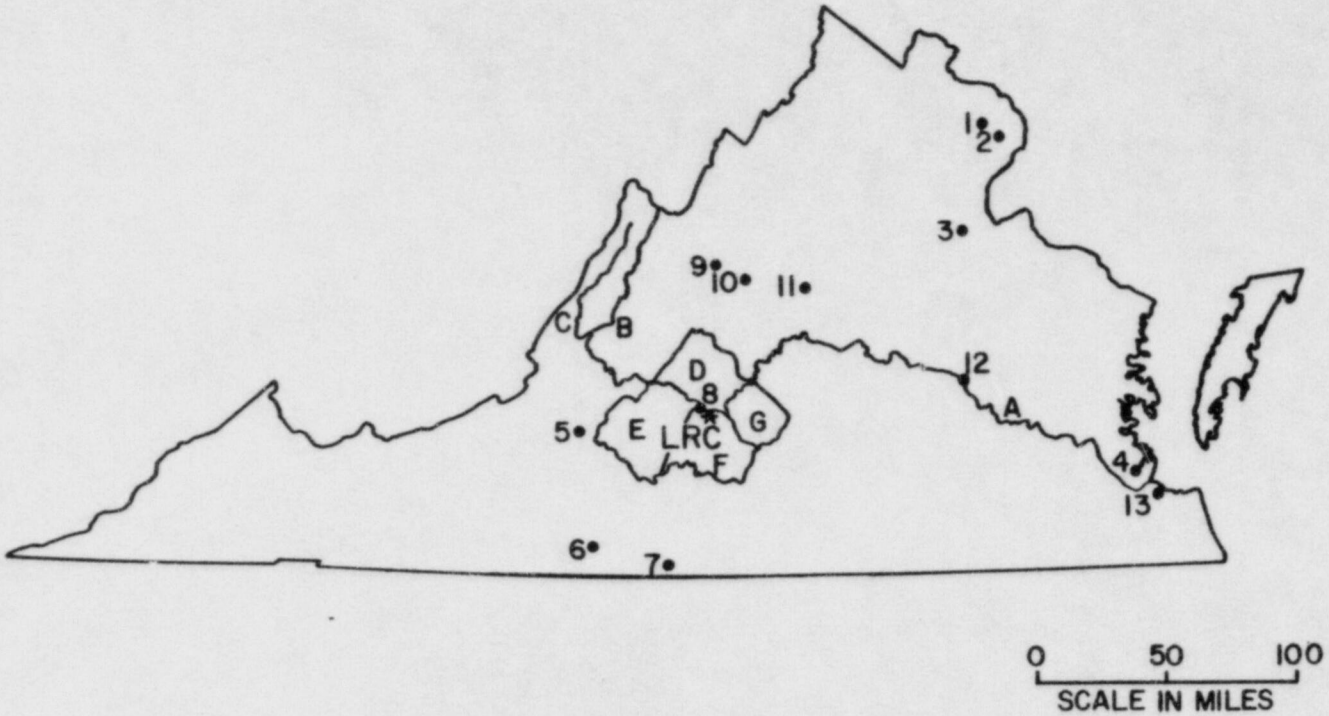
1.7.1 Licensed material shall be used in the performance of Research and Development (e.g., hot cell examination of irradiated and radioactive components including irradiated fuel; analytical activities for other companies or B&W divisions including laboratory analysis, preparation of and testing of materials and equipment; preparation and modification of radiation sources; and preparation and decontamination of reactor-related hardware for inspecting, evaluating, and measuring reactor components).

1.7.2 The LRC may transport and possess licensed material in private carriage between NRC licensed facilities within the United States pursuant to the regulations in 10 CFR 71 and 49 CFR.

1.8 EXEMPTIONS AND SPECIAL AUTHORIZATIONS

1.8.1 The uranium bioassay program sampling frequency shall comply with Tables 2 and 3 of Regulatory Guide 8.11, dated June, 1974, except as follows:

1.8.1.1 When an employee is absent from the LRC during a period when the bioassay counting service is on site, a special counting shall not be required for those employees for routine exposure control monitoring. The maximum amount of time between in vivo counts shall not exceed 12-months.



1. ARLINGTON (174, 284)
2. ALEXANDRIA (110, 938)
3. FREDERICKSBURG (14, 450)
4. NEWPORT NEWS (138, 177)
5. ROANOKE (92, 115)
6. MARTINSVILLE (19, 653)
7. DANVILLE (46, 391)
8. LYNCHBURG (54, 083)
9. STAUNTON (25, 504)
10. WAYNESBORO (16, 707)
11. CHARLOTTESVILLE (38, 880)
12. RICHMOND (249, 621)
13. NORFOLK (307, 951)

A. JAMES RIVER
B. COWPASTURE RIVER
C. JACKSON RIVER
D. AMHERST COUNTY (26, 072)
E. BEDFORD COUNTY (26, 728)
F. CAMPBELL COUNTY (43, 319)
G. APPOMATTOX COUNTY (9, 784)

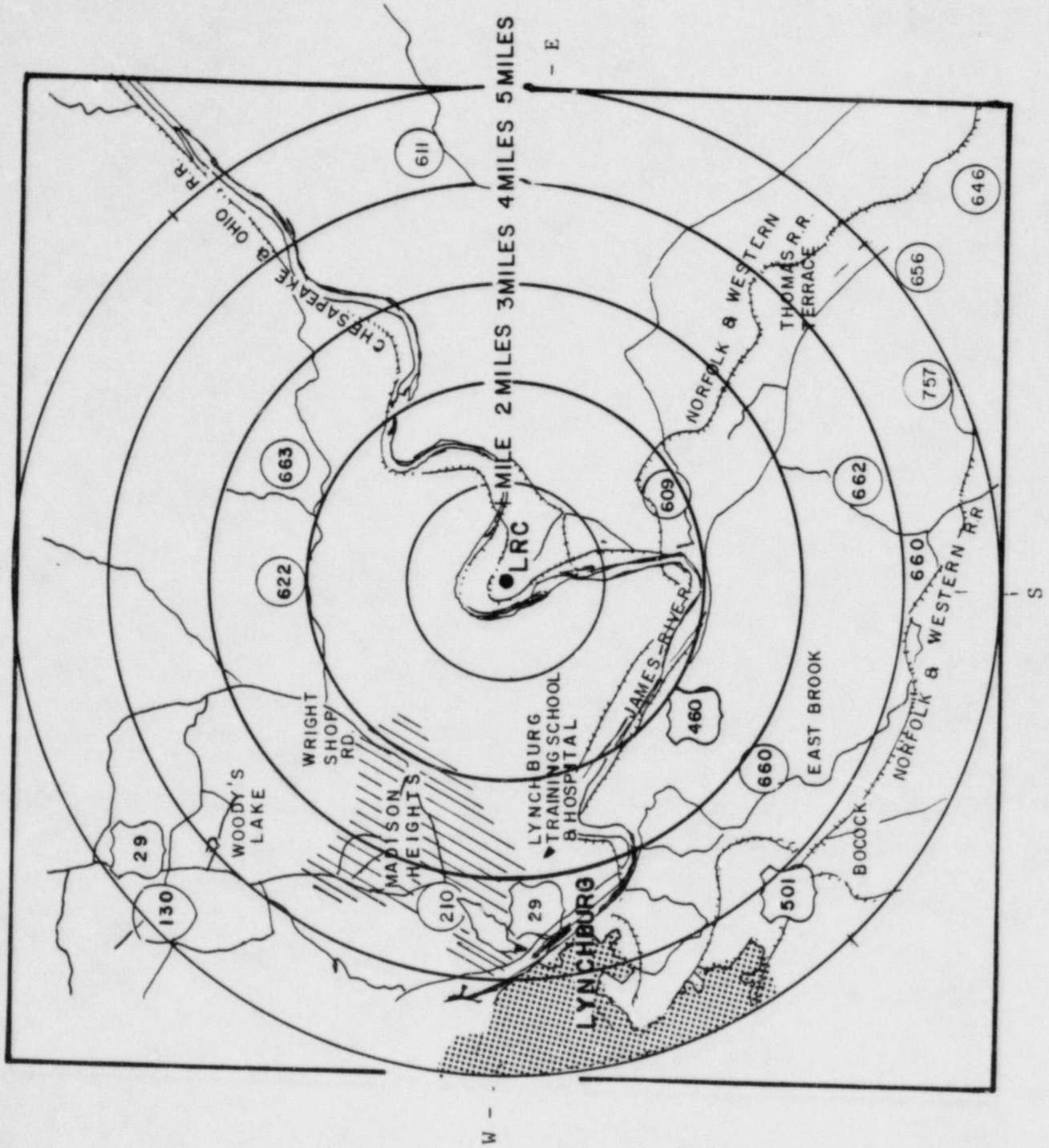
[NUMBERS IN PARENTHESES ( ) ARE 1970 CENSUS DATA]

FIGURE 1-1

License No	SNM-778	Docket No.	70-824	Date	May, 1986
Amendment No.	0	Revision No.	2	Page	1-6

Babcock & Wilcox
a McDermott company

FIGURE 1-2



License No	SNM-778	Docket No.	70-824	Date	May, 1986
Amendment No.	0	Revision No.	2	Page	1-7


N

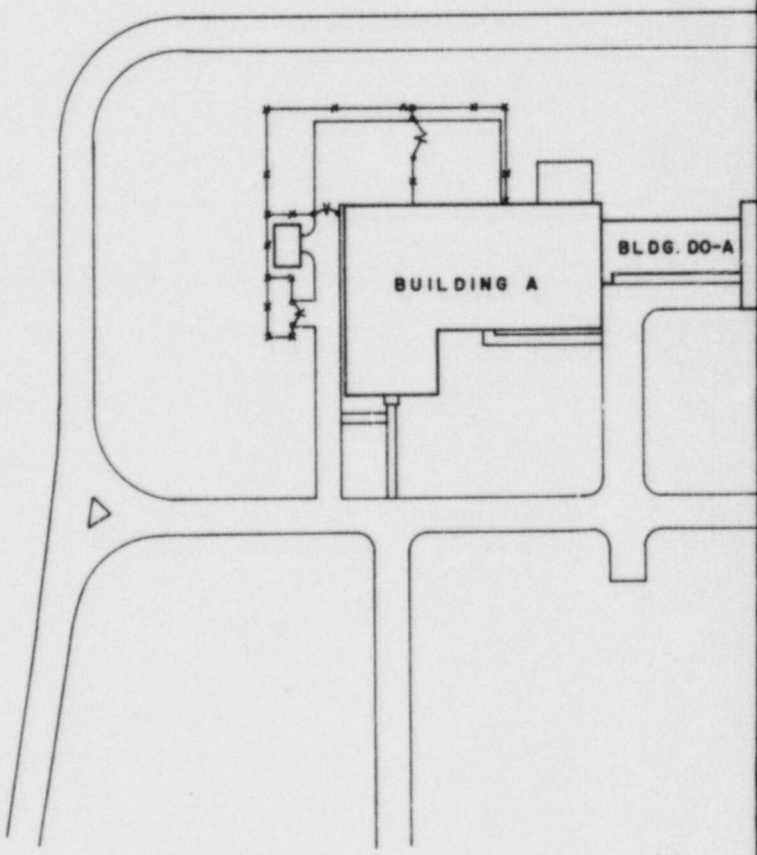
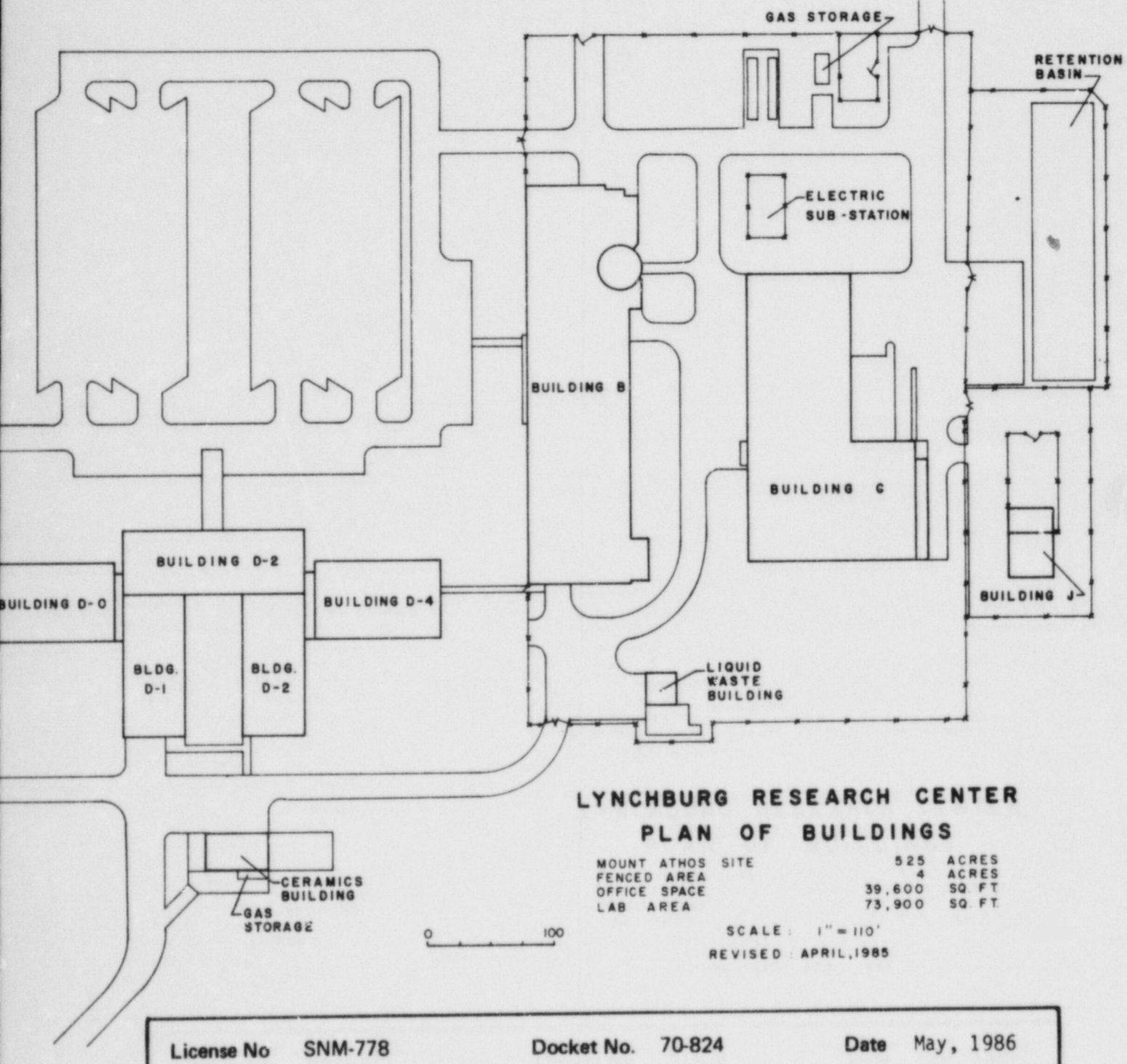

BUILDING A
BLDG. DO-A

FIGURE 1-3



LYNCHBURG RESEARCH CENTER
PLAN OF BUILDINGS

MOUNT ATHOS SITE	525	ACRES
FENCED AREA	4	ACRES
OFFICE SPACE	39,600	SQ. FT
LAB AREA	73,900	SQ. FT

SCALE: 1" = 110'

REVISED: APRIL, 1985

TABLE OF CONTENTS

Section		Page
2.0	GENERAL ORGANIZATIONAL AND ADMINISTRATIVE REQUIREMENTS . .	2-1
2.1	POLICY	2-1
2.2	ORGANIZATION RESPONSIBILTIES AND AUTHORITIES	2-1
2.3	SAFETY REVIEW COMMITTEE	2-3
2.4	APPROVAL AUTHORITY FOR PERSONNEL SELECTION	2-4
2.5	PERSONNEL EDUCATION AND EXPERIENCE REQUIREMENTS	2-5
2.6	TRAINING	2-6
2.7	OPERATING PROCEDURES	2-7
2.8	INTERNAL AUDITS AND INSPECTIONS	2-8
2.8.1	Nuclear Criticality Safety	2-8
2.8.2	Health Physics	2-8
2.8.3	General Safety and Compliance	2-8
2.9	INVESTIGATIONS AND REPORTING OF OFF-NORMAL OCCURRENCES . .	2-9
2.9.1	License Administrator	2-9
2.9.2	Supervisor, Health and Safety	2-10
2.9.3	Facility Supervisor	2-10
2.10	RECORDS	2-10
2.10.1	Health and Safety Group	2-10
2.10.2	Nuclear Safety Officer	2-11
2.10.3	License Administrator	2-11

2.0 GENERAL ORGANIZATIONAL AND ADMINISTRATIVE REQUIREMENTS

2.1 POLICY

It shall be the policy of the LRC to maintain radiation exposures to employees and the general public as low as is reasonably achievable. The facility procedures to ensure the safe handling of licensed material are the Area Operating Procedures.

2.2 ORGANIZATION RESPONSIBILITIES AND AUTHORITIES.

2.2.1 Manager, Lynchburg Technical Operations - The Manager, Lynchburg Technical Operations is ultimately responsible for all safety at the LRC.

2.2.2 Laboratory Managers - The Laboratory Managers are responsible for the safety of personnel in their laboratories. The Laboratory Managers report to the Manager, Lynchburg Technical Operations.

2.2.3 Section Managers - Section Managers are responsible for the safe performance of projects under their purview. To this end, they are responsible for ensuring that personnel in their sections follow all applicable Area Operating Procedures. The Section Managers report to the Laboratory Managers.

2.2.4 Facility Supervisor - The Facility Supervisor is responsible to the Manager, Lynchburg Technical Operations for the safe conduct of all operations at the LRC and for ensuring that all applicable operations are conducted in compliance with the license and applicable regulations. To fulfill these responsibilities the Facility Supervisor shall have the authority to stop any operation that he feels is unsafe or in violation of license. The Facility Supervisor shall review all new Area Operating Procedures and RWP's, and changes there to, for license and regulatory compliance and for facility safety; and he shall have approval authority for them. He shall submit items for review to the SRC.

2.2.5 Manager, Safety and Licensing - The Manager, Safety and Licensing reports to the Manager, Lynchburg Technical Operations. The Supervisor, Health and Safety, the Accountability Specialist, and the License Administrator report to this manager.

2.2.6 Supervisor, Health and Safety - The Supervisor, Health and Safety is responsible for providing adequate facilities, procedures, and properly trained personnel to implement the Health Physics and Industrial Safety Programs. He is responsible for health physics and industrial safety activities. The Supervisor, Health and Safety reports to the Manager, Safety and Licensing, and has direct access to the Manager, Lynchburg Technical Operations in matters pertaining to Health and Safety. The Supervisor, Health and Safety has the authority to stop any operation that he believes is contrary to accepted safety practices or license requirements. He shall review all new Area Operatiang Procedures, Radiation Work Permits and changes there to, for the radiation safety aspects of the procedure RWP, or change, and he shall have approval authority for them. He shall conduct training programs for new employees and Authorized Users of Radioactive Material. He shall be responsible for the shipment of licensed material. The Supervisor, Health and Safety shall be a member of the Safety Review Committee but shall not be a member of the Safety Audit Subcommittee.

2.2.7 Health Physics Engineer - A Health Physics Engineer shall administer activities of the Health Physics Staff. He shall report to the Supervisor, Health and Safety.

2.2.8 Industrial Safety Officer - The Industrial Safety Officer shall administer the industrial safety program. He shall report to the Supervisor, Health and Safety.

2.2.9 Nuclear Safety Officer - The Nuclear Safety Officer shall be responsible for ensuring that no operation at the LRC results in the inadvertent assembly of a critical mass. He shall review all new Area Operating Procedures and changes there to, for nuclear criticality safety and shall have approval authority for them. He shall conduct training programs in criticality safety and perform criticality safety calculations. He shall report to the Manager, Lynchburg Technical Operations.

2.2.10 License Administrator - The License Administrator shall be responsible for administering the license. He is the primary liaison with the NRC and other federal, state, and local agencies in matters that pertain to nuclear activities. He shall be the coordinator of the SRC and the Safety Audit Subcommittee and shall represent management on both. He shall maintain the permanent records of the SRC and shall be responsible for assuring that appropriate action is taken to correct SAS audit findings that are

approved by the Manager, Lynchburg Technical Operations. He shall report to the Manager, Safety and Licensing.

2.2.11 Accountability Specialist - The Accountability Specialist shall be responsible for the maintenance and retention of SNM accountability records. The Accountability Specialist shall report to the Manager, Safety and Licensing.

2.3 SAFETY REVIEW COMMITTEE

2.3.1 Function

2.3.1.1 The SRC shall review and approve all new Area Operating Procedures, and shall concur with all changes made to them in the time interval since their last regular meeting.

2.3.1.2 The SRC shall review and approve new projects and major changes to existing projects that utilize licensed materials.

2.3.1.3 The SRC shall review the annual report prepared by the Supervisor, Health and Safety.

2.3.1.4 The SRC shall provide the LRC with general consulting services in the field of radiation protection and the safe handling of licensed material.

2.3.1.5 The SRC shall review all SAS audit findings, all overexposures and unusual occurrences which must be reported to the NRC. These reviews shall be conducted during the next regularly scheduled meeting following the event and the results of the review shall be documetned in the minutes.

2.3.2 Frequency of Meetings

2.3.2.1 The SRC shall meet at least four times annually for the purposes of conducting its business as specified in Section 2.3.1.

2.3.3 Safety Audit Subcommittee

2.3.3.1 The SAS shall perform audits of the LRC for the Safety Review Committee.

2.3.3.2 The SAS shall audit facilities, procedures, records, and

operations at the LRC for compliance with written requirements and the exercise of acceptable safety practices.

2.3.3.3 The SAS shall perform at least three audits annually, distributed over a 12-month period. Audits shall be made in accordance with written guidance to assure all aspect of 2.3.3.2 are audited.

2.3.3.4 SAS membership shall be appointed by the Manager, Lynchburg Technical Operations.

2.3.4 Reporting

2.3.4.1 The SRC shall report to the Manager, Lynchburg Technical Operations.

2.3.4.2 The SAS shall report to the Chairman, SRC.

2.3.5 Recordkeeping

2.3.5.1 Minutes of the SRC proceedings shall be prepared by the Chairman, SRC.

2.3.5.2 SRC Minutes shall be forwarded to the Manager, Lynchburg Technical Operations by the Chairman, SRC.

2.3.5.3 The permanent records of the SRC shall be kept by the SRC Coordinator.

2.3.5.4 SAS audit reports shall be prepared by the Chairman, SAS.

2.3.5.5 SAS audit reports shall be forwarded to the Chairman, SRC by the Chairman, SAS.

2.3.5.6 SAS audit reports shall be forwarded to the Manager, Lynchburg Technical Operations by the Chairman, SRC with comments, as he deems appropriate.

2.4 APPROVAL AUTHORITY FOR PERSONNEL SELECTION

2.4.1 The Manager, Lynchburg Technical Operations shall approve the personnel selected for safety-related positions specified in Section 2.2 of this Part. The Manager, Lynchburg Technical Operations is appointed by the R & D Division Vice President.

2.5 PERSONNEL EDUCATION AND EXPERIENCE REQUIREMENTS

2.5.1 Manager, Lynchburg Technical Operations - The Manager, Lynchburg Technical Operations shall be appointed in accordance with Company policy.

2.5.2 Laboratory Managers - The Laboratory Managers are appointed by the Manager, Lynchburg Technical Operations in accordance with Company policy.

2.5.3 Section Managers - The Section Managers shall have a BS degree and three years post graduate work or equivalent experience in the pertinent technical field. Managers of sections handling licensed materials shall have demonstrated knowledge of the application of radiation and nuclear criticality safety requirements relative to their projects.

2.5.4 Facility Supervisor - The Facility Supervisor shall have a degree in his related work and three years experience in the use and handling of licensed material, or five years experience in the use and handling of licensed material. He must demonstrate to management proficiency in the application of good principles of radiation protection, industrial safety, and nuclear safety as related to the activities at the LRC.

2.5.5 Manager, Safety and Licensing - The Manager, Safety and Licensing shall have a BS degree in a technical field and five years experience in the nuclear field.

2.5.6 Supervisor, Health and Safety - The Supervisor, Health and Safety shall have a BS degree in a technical field and professional experience in assignments involving radiation protection at the supervisory level. He must have four years experience and demonstrate proficiency in the application of radiation safety principles and be knowledgeable in fields related to radiation protection.

2.5.7 Health Physics Engineer - A Health Physics Engineer shall have a BS degree which shall include at least 20 quarter hours health physics related course work or the equivalent in work experience.

2.5.8 Industrial Safety Officer - The Industrial Safety Officer shall have at least one year's experience in radiation and industrial safety. He shall be familiar with the codes and requirements of

the Occupational Health and Safety Act of 1970 and the National Fire Protection Association.

2.5.9 Nuclear Safety Officer - The Nuclear Safety Officer shall have a BS degree in science or engineering. He must have either two years experience with nuclear criticality safety calculations similar to those associated with LRC activities or he must have one year's experience with nuclear criticality safety calculations similar to those associated with LRC activities if he has at least an additional two years' experience in nuclear reactor physics calculations.

2.5.10 Accountability Specialist - The Accountability Specialist shall have at least a high school education and three years' experience in the use of licensed material. He must demonstrate to Company management his knowledge of the principles necessary for the accountability and safeguarding of special nuclear materials.

2.5.11 License Administrator - The License Administrator shall have a BS degree in science or engineering and three years experience in nuclear technology or an AS degree in science or nuclear technology with five years experience in nuclear technology.

2.5.12 Safety Review Committee - The SCR membership, as a body, shall have expertise in chemistry, nuclear physics, health physics, and the safe handling of radioactive material. The SRC membership shall have a general understanding of nuclear criticality safety as it pertains to LRC operation.

2.6 TRAINING

2.6.1 Program I - Each new employee shall receive training within thirty days of reporting to work. This training, denoted as Program I, provides an introduction to radioactivity and a thorough coverage of safety rules and procedures including emergency procedures.

2.6.2 Program II - New laboratory employees who will be working with licensed material shall be required to complete Program II training. Completion of this program requires passing a written examination. Parts of Program II may be waived by the Supervisor, Health and Safety for technical and scientific personnel already knowledgeable and experienced in working in radiation areas and with licensed material. However, such personnel must pass the

written examination required for Program II. Persons who complete this course may be designated as an Authorized User.

2.6.3 Retraining - Persons who are designated as Authorized Users shall be retrained annually. Satisfactory completion of the retraining shall be determined by passing a written examination.

2.6.4 Respiratory Protection Training - Training in respiratory protection techniques and equipment shall be required of all employees before the use of such equipment will be permitted. Satisfactory completion of this training shall be determined by passing a written examination.

2.7 OPERATING PROCEDURES

2.7.1 All operations with licensed material shall be conducted in accordance with Area Operating Procedures or Radiation Work Permits (see 3.1.1).

2.7.2 Area Operating Procedures (AOP) - Area Operating Procedures shall be established for all operations in which SNM, source and byproduct materials are stored or handled. AOP's shall include those nuclear criticality and radiation safety controls and limits that apply to the operation. Each AOP shall be approved by the Nuclear Safety Officer or his designated alternate, the Supervisor, Health and Safety or his designated alternate, the Facility Supervisor or his designated alternate, and the Safety Review Committee.

2.7.3 AOP's may be revised with the approval of the Nuclear Safety Officer or his designated alternate, the Supervisor, Health and Safety or his designated alternate, and the Facility Supervisor or his designated alternate. The revised procedure may be used with these approvals until the next scheduled regular meeting of the Safety Review Committee when the revision must be approved by the SRC.

2.7.4 AOP's shall be available in each operations area where they apply and shall be followed by operations personnel.

2.7.5 Distribution of new and revised procedures shall be made in accordance with a document control system which assures that the procedure manuals contain only the most current revision of the procedures.

2.7.6 AOP manuals shall be reviewed annually by the Facility Supervisor to assure that the manuals contain the most current revision of the procedures.

2.8 INTERNAL AUDITS AND INSPECTIONS

2.8.1 Nuclear Criticality Safety

2.8.1.1 The Nuclear Safety Officer or his designated alternate shall conduct internal audits of the LRC for the purpose of evaluating the nuclear criticality safety aspects of operations. This audit shall be conducted in accordance with written audit guidance. This audit shall be conducted once each calendar quarter. A report of his findings shall be made to the Manager, Lynchburg Technical Operations within two weeks of completing the audit. The audit reports shall be forwarded to the Facility Supervisor and the License Administrator. The License Administrator shall be responsible for assuring that the appropriate corrective actions are taken to address the audit findings.

2.8.1.2 The Facility Supervisor shall perform an inspection weekly for compliance with the nuclear criticality safety aspects of the operations. Findings resulting from these inspections shall be reported to the Nuclear Safety Officer.

2.8.2 Health Physics

2.8.2.1 The Supervisor, Health and Safety or his designated alternate shall conduct internal audits of the LRC for the purpose of evaluating the health physics aspects of operations. This audit shall be conducted in accordance with written audit guidance. This audit shall be conducted once each month. A report of his findings shall be made to the Manager, Lynchburg Technical Operations within two weeks of completing the audit. The audit reports shall be forwarded to the Manager, Lynchburg Technical Operations and the License Administrator. The License Administrator shall be responsible for assuring the appropriate corrective actions are taken to address the audit findings.

2.8.3 General Safety and Compliance

2.8.3.1 The SAS performs audits of general safety and compliance at the LRC. These audits shall be conducted three times annually. The

audits shall be distributed over a 12-month period. The SAS shall include an audit of the Health and Safety Group at least once annually. This annual audit shall be performed by a qualified individual who is independent of the Health and Safety Group. Other areas of LRC operations shall be audited for compliance with written requirements and the exercise of acceptable safety practices. Audits shall be made in accordance with written guidance to assure all aspects of Section 2.3.3.2 are audited. The Chairman, SAS shall file a report of the audit findings with the Chairman, SRC, with a copy to the License Administrator and the Facility Supervisor and members of the SRC. The Chairman, SRC shall forward the report to the Manager, Lynchburg Technical Operations with comments, as he deems appropriate. The License Administrator shall be responsible for assuring that the appropriate corrective actions are taken to address the audit findings.

2.9 INVESTIGATIONS AND REPORTING OF OFF-NORMAL OCCURRENCES

2.9.1 License Administrator

The License Administrator shall investigate and report, when required, the following types of off-normal occurrences:

2.9.1.1 Excessive levels of radiation from or contamination on packages upon receipt.

2.9.1.2 Thefts, attempted thefts, or losses of licensed material, other than normal operating losses.

2.9.1.3 Incidents as specified in 10 CFR 20.403

2.9.1.4 Overexposure of individuals and excessive levels and concentrations of radioactivity.

2.9.1.5 Failures to comply and defects pursuant to 10 CFR 21.

2.9.1.6 Changes to security, safeguards, or emergency plans made without prior NRC approval, when prior approval is required.

2.9.1.7 Failures to comply with license requirements.

2.9.1.8 Unapproved storage or use of licensed material.

2.9.2 Supervisor, Health and Safety

The Supervisor, Health and Safety shall perform investigations and issue reports of the following:

2.9.2.1 Higher than expected personnel exposures.

2.9.2.2 Higher than expected concentration of airborne activity in the facility.

2.9.2.3 Unauthorized entry into a High Radiation or Airborne Radioactive Material area.

2.9.2.4 Failure of equipment or instrumentation to meet Health and Safety requirements.

2.9.3 Facility Supervisor

The Facility Supervisor shall perform investigations of the following:

2.9.3.1 Any violation of nuclear criticality safety criteria.

2.9.3.2 Any violation of Area Operating Procedures or RWP's.

2.10 RECORDS

The following positions or organizations shall be responsible for maintaining the indicated records, for the period specified. Records may be kept in original form, microfilm or in computer storage. The symbol (*) indicates that the record will be retained until the NRC authorizes its disposition.

2.10.1 Health and Safety Group

Health and Safety Supervisor audits	2 years
Shipping and receiving RM forms	5 years
Waste disposal records	(*)
Personnel dosimetry records	(*)
Results of Bioassays and Whole Body Counting	(*)
Releases to the environment	(*)

Radiation survey data	2 years
Contamination survey data	2 years
Radiation Work Permits (completed)	5 years
Radiation detection instrument calibration	2 years
Leak tests of sealed sources	2 years
Employee training	(*)
Employee retraining	(*)
Airborne radioactivity sampling data	(*)
NRC-4 forms	(*)
NRC-5 forms	(*)

2.10.2 Nuclear Safety Officer

Nuclear safety evaluations and calculations	6 months after termination of the approved process.

2.10.3 License Administrator

Safety Review Committee Minutes	(*)
Safety Audit Subcommittee Audit Reports	2 years
Investigation reports of off-normal occurrences	2 years

3.0 RADIATION PROTECTION

3.1 SPECIAL ADMINISTRATIVE REQUIREMENTS

3.1.1 Radiation Work Permits (RWP)

3.1.1.1 RWP's shall be issued whenever the activity is not covered by an Area Operating Procedure and personnel are likely to be exposed to levels of radiation or concentrations of radioactive material in excess of those specified in 10 CFR 20.

3.1.1.2 RWP's shall be approved by the Work Area Supervisor, Employee's Supervisor, Health Physics Supervisor, and the Facility Supervisor.

3.1.1.3 The RWP form shall specify levels of personnel exposure above which a documented ALARA evaluation shall be required. RWP's that require a documented ALARA evaluation must, in addition to 3.1.1.2, be approved by the Manager, Lynchburg Technical Operations.

3.1.1.4 RWP's shall be approved at a meeting of all the signators of the form.

3.1.1.5 The RWP form shall provide space for entering the estimated exposures to the whole body, extremities, and for the job. These are used to identify the areas of exposure concern and do not constitute an exposure goal or limit.

3.1.1.6 The RWP form shall provide space for the workers' supervisor to sign or initial, attesting that the workers have been instructed in the requirements of the RWP.

3.1.1.7 The term of an RWP shall not exceed 30-days except for Standing RWP's.

3.1.2 ALARA Policy

The management of the LRC is committed to a policy of maintaining exposures as low as is reasonably achievable.

3.1.2.1 Employees shall be introduced to this policy during their initial training and shall be reinforced during the annual retraining of Authorized Users.

3.1.2.2 The ALARA policy shall be implemented through the Area Operating Procedures and Radiation Work Permits.

3.1.2.3 The ALARA policy shall be enforced by the Facility Supervisor and the Supervisor, Health and Safety in the exercise of their review and approval authority, their authority to terminate operations, and audits.

3.1.2.4 The SRC shall evaluate ALARA performance in exercising their review authority over procedures and proposed new projects and their review of the annual report from the Supervisor, Health and Safety.

3.1.3 Offsite Possession

Offsite possession and use of licensed material shall be the responsibility of and under the control of LRC employees specifically approved by the Safety Review Committee.

3.2 TECHNICAL REQUIREMENTS

3.2.1 Access Control

3.2.1.1 High Radiation Areas - Entry into a High Radiation Area or an Airborne Radioactivity Area shall be controlled by an RWP.

3.2.1.2 Contamination Areas - Areas which are determined by the Health and Safety Group to present a risk of spreading radioactive contamination into non-contaminated areas shall be clearly marked at each entrance. Step-off pads shall be provided. Personnel survey instrumentation shall be provided at the step-off pad. The minimum protective clothing required in Contamination Areas shall be shoe covers and lab coats. Exiting such areas shall require personnel to remove their protective clothing and survey themselves with the instrumentation provided. Persons found to be contaminated above background levels must receive the approval of a member of the Health Physics Staff before leaving the Contaminated Area.

3.2.2 Ventilation Requirements

3.2.2.1 Potentially contaminated exhaust air from hood, hot cells, and glove boxes shall be discharged through the fifty meter high stack, except as noted in 3.2.2.6.

3.2.2.2 The exhaust stack shall be sampled isokinetically.

3.2.2.3 The minimum air flow rate in the stack sampling system shall be 2 cfm.

3.2.2.4 The stack sampling and monitoring system shall operate continuously except for periods when repair or calibration is required.

3.2.2.5 The following table presents the release limits and action levels associated with the exhaust stack. The Health and Safety Group shall be responsible for responding to releases in excess of these action levels. An operation that results in action levels being exceeded for 4-consecutive time periods, shall be shutdown until the cause is corrected.

STACK RELEASE LIMITS AND ACTION LEVELS

Release Product	Release Limit	Action Level
Beta Particulate	2 mCi/yr	200 uCi/week
Alpha Particulate (long lived)	20 uCi/yr	1 uCi/2 weeks
Kr-85	2500 Ci/yr	70 Ci/week
H-3	130 Ci/yr	3 Ci/week
I-131	6 mCi/yr or 300 uCi/week	200 uCi/week

3.2.2.6 Exhaust systems that cannot be practicably discharged through the 50-meter stack, and where there exists a reasonable probability that the discharges to the atmosphere could exceed 10% of the applicable MPC for an unrestricted area, shall be monitored for gaseous and particulate activity in the effluent.

3.2.2.7 Exhaust air from areas in which there is no airborne radioactive material may be exhausted directly to the roof either with or without continuous sampling, if approved by the Safety Review Committee.

3.2.2.8 Areas equipped with an air monitor may be exhausted to the roof through HEPA filters if the concentration of airborne radioactive material is below the appropriate MPC for an unrestricted area, if approved by the Safety Review Committee.

3.2.2.9 All hoods used for the handling of licensed material shall exhaust through at least one prefilter and one HEPA filter, except for hoods that are specifically designed and installed for use with perchloric acid.

3.2.2.10 Fume hoods utilized for the handling of unirradiated Pu shall be provided with two HEPA filters in series.

3.2.2.11 Hot cells shall be provided with two stages of HEPA filters.

3.2.2.12 Final HEPA filters which service facilities where licensed material is handled shall be tested, using the cold DOP test, annually or after a final HEPA filter is changed, whichever comes sooner.

3.2.2.13 The acceptance criteria for the testing of final HEPA filters (3.2.2.12) shall be 99.95% of all particles having a light-scattering mean diameter of approximately 0.7 micrometers.

3.2.3 Instrumentation

3.2.3.1 Portable instruments - The LRC maintains a relatively large and diverse inventory of portable survey instruments. These instruments vary in range and sensitivity. The below listing is a representative sampling of the instruments on hand:

Instrument	Sensitivity	Characteristics	Type Radiation
Ionization Chamber	0 - 20K R/hr	6.5Kev - 1.2Mev	Beta & Gamma
Geiger Counter	0 - 1K R/hr	23Kev - 1.2Mev	Beta & Gamma

Proportional Counter (gas flow)	25 - 500K cpm		Alpha & Beta
Scintillation Detector	0 - 50K cpm		Alpha
Geiger Counter	0 - 50K cpm	>40Kev	Beta
Scintillation Detector	0 - 5 mR/hr		Gamma
Neutron Dose	0 - 5K mR/hr	25Kev - 3Mev	Neutron

3.2.3.2 Air Monitors

3.2.3.2.1 Nuclear Measurements Corp. (NMC) Model AM-2A - This instrument utilizes a gas flow proportional detector with a 1.0 mg/cm^2 thick end window. These instruments are operated as alpha or beta-gamma monitors. They utilize a fixed filter with a nominal air flow of 2.5 to 3 ft^3/min. The alarm setting is set at less than 40 MPC hours above normal background including Radon and Thoron daughters.

3.2.3.2.2 Eberline Model AIM-3S - These monitors are used for alpha monitoring only. They are typically located in areas where Pu or U is being processed. They use a ZnS(Ag) scintillation detector with a fixed filter. The monitor air flow is nominally 20 ft^3/hr. The alarm is set at less than 40 MPC hours above the normal background for Radon and Thoron daughters.

3.2.3.3 Air Samplers

3.2.3.3.1 Mine Safety Appliance (MSA) Model G - These personal air samplers utilize a Millipore field sample cassette. The nominal air flow rate is 2 liters/min. Samples are collected for counting on a low background counter with sufficient sensitivity to detect 25% of the applicable MPC for 8 hour sampling intervals.

3.2.3.3.2 Fixed samplers are located at work stations where the concentration of airborne radioactive material potentially exceeds 25% of the applicable MPC. These samplers which may be used to determine concentrations in the workers breathing zone shall be

evaluated for representativeness at least once every 12 months or when a licensed process or equipment change is made that could disturb the air flow pattern.

3.2.3.4 Criticality Monitors

3.2.3.4.1 Nuclear Measurements Corp. (NMC) Model GA-2TO and GA-2A - These monitors are designed as criticality alarm systems. Detection is by a NaI (Tl) detector operated in the constant current mode. Response is logrithmic and non-saturating. Emergency power is provided. The nominal alarm setpoint is 20 mR/hr. Failure alarm function is provided. Criticality monitors shall be calibrated semi-annually.

3.2.3.5 Counting Equipment

3.2.3.5.1 Sharp Low Beta - Air samples and effluent samples may be counted on this instrument. This instrument utilizes a 4.5-inch and a 2.5-inch very thin end window proportional detector. Backgrounds and counter response are tested weekly and the instrument is calibrated annually.

3.2.3.5.2 Beckman Wide Beta - Air samples and effluent samples may be counted on this instrument. It utilizes two 2.5-inch very thin end window proportional detectors. Backgrounds and counter response are tested weekly and the system is calibrated annually. The manual detector is used infrequently and it is tested when used.

3.2.4 Internal and External Exposure

3.2.4.1 Ventilation

3.2.4.1.1 The minimum air velocity across the opening of fume hoods that are used to handle licensed material shall be at least 100 fpm. Hood face velocities shall be measured monthly. Those hoods that do not meet the minimum requirement shall be placed out of service.

3.2.4.1.2 The maximum differential pressure across HEPA filters is limited to 4-inches of water. HEPA filters shall be changed to prevent exceeding this limit.

3.2.4.1.3 The minimum differential pressure across the hot cell face shall

be 0.25-inches of water. An additional hot cell fan will be automatically or manually started when the differential pressure reaches 0.25-inches of water.

3.2.4.1.4 The minimum air flow rate through any opened door to a hot cell shall be 100 fpm. An additional hot cell fan will be automatically or manually started when the differential pressure reaches 0.25-inches of water.

3.2.4.2 Air Sampling and Analysis

3.2.4.2.1 Air Samples shall be taken in all areas where operations could cause personnel to be exposed to airborne radioactive materials.

3.2.4.2.2 Any area in which the concentration of airborne radioactive material potentially exceeds 25% of the applicable MPC shall be continuously monitored for as long as the process that caused the airborne activity is in progress.

3.2.4.2.3 Permanently mounted air sampling equipment used to determine concentrations in the worker's breathing zone shall be evaluated for representativeness at least once every 12 months. This evaluation shall also be performed when changes are made in a process or equipment that could effect the sample's representativeness.

3.2.4.2.4 An evaluation of the representativeness of air sampling equipment shall be performed at the start up of a process that has been shutdown for more that 6 months.

3.2.4.2.5 Permanently mounted air samplers used to determine concentrations in a worker's breathing zone shall be changed as follows:

1. Process areas during normal operations - once/shift.

2. All areas during periods when normal operations are shut down - 48 hours.

3. Samples that are less than 10% of the MPC - weekly.

4. Samples that exceed 25% of the MPC during one sample change period - once/shift until the situation is corrected and at least one month's samples are less than 10% of the MPC.

3.2.4.2.6 Permanently mounted air samplers shall have a minimum sample flow rate of 8 LPM except when a personnel air sampler is used as a temporary replacement, in which case the minimum flow rate shall be 1.8 LPM.

3.2.4.3 Bioassay

3.2.4.3.1 Uranium Bioassay Program

1. The uranium bioassay program sampling frequency shall comply with Regulatory Guide 8.11, June, 1974, except as specified in section 1.8 of this application.

2. The action levels for the results of urinalysis sampling shall be <25 pCi/L for Class W and Y compounds and <21 pCi/L for Class D compounds. If these levels are exceeded, diagnostic evaluation must be performed. If the diagnostic evaluation yields positive results, then the worker shall be removed from further uranium work and the Supervisor, Health and Safety shall estimate the worker's exposure utilizing additional urine samples, fecal samples, in vivo counting or other means at his disposal. Any worker whose estimated body burden is greater than 50% shall be in vivo counted as soon as practicable. Any worker whose estimated body burden is between 10 and 50% will be in vivo counted during the next time that the body counting service is at the B&W site.

3.2.4.3.2 Plutonium Bioassay Program

1. All personnel who routinely work in plutonium handling areas shall be subject to the plutonium bioassay program. The minimum frequency for urine sampling shall be six months. The minimum frequency for in vivo counting shall be annual.

2. The following are the action criteria for W and Y compounds exposure for the plutonium bioassay program:

Analysis	Action Level	Action to be Taken
Urinalysis	< 0.2 dpm/L	None
Urinalysis	> 0.2 dpm/L	1. Resample the individual within 5 working days.

		2. Determine if area surveys support the analysis results. 3. If #2 is positive, investigate the cause and correct. 4. If the exposure is confirmed by #1 investigate to determine how exposure was incurred and correct it. If the exposure exceeds 50% of the maximum permissible annual dose, the worker shall be restricted from further exposure until the Supervisor, Health and Safety authorizes the lifting of this restriction.
Urinalysis	> 4 dpm/L	1. Restrict the individual from further Pu work. 2. Resample the individual with 24 hours. 3. Initiate an investigation. 4. The Supervisor, Health and Safety only, may lift the work restriction.
In vivo	< 4E-9 Ci	None
In vivo	≥ 4E-9 Ci	1. Restrict the worker from further exposure. 2. Resample the individual within 10 working days. 3. Determine if area surveys support the analysis results. 4. If #3 is positive, investigate the cause and correct as needed.

5. If exposure is confirmed by #2, the Supervisor, Health and Safety shall determine the organ dose. If the confirmed exposure exceeds 50% of the maximum permissible annual dose, the worker shall be restricted from further exposures until the Supervisor, Health and Safety authorizes the lifting of this restriction.

6. The restriction in #1 may be lifted by the Supervisor, Health and Safety if the results of the analysis performed under #2 fails to confirm the analysis.

3.2.4.3.3 Fission Product Bioassay Program

1. The fission product bioassay program sampling frequency shall comply with Regulatory Guide 8.26, September, 1980.

2. Additional bioassays shall be performed when in the opinion of the Supervisor, Health and Safety, conditions during the job were such that significant internal exposure may have occurred. The following are action criteria for additional bioassays.

Analysis	Action Level	Action to be Taken
In vivo	>10% MPOB	Remeasure subject to determine effective half life of the contaminant and plot decay curves. Follow-up program will continue until the contamination present is <5% MPOB or the effective half life has been determined.

Estimation from nasal smears or air sample	>10% MPOB	Submit in vitro sample for analysis within 5 working days.
In vitro	>5% MPOB	Resample excreta to confirm presence of contamination and to establish rate of elimination. Perform isotopic analysis if >10% of MPOB is a possibility.
In vitro	>10% MPOB	In vivo measurement to be made as soon as practicable.

3. The Supervisor, Health and Safety, shall be responsible for evaluations to determine the location and amount of deposition; to provide data necessary for estimating internal dose rates, retention functions, and dose commitments; and to determine if work restrictions or referrals for therapeutic treatment are required for any case where a result indicating a greater than 10%/MPOB deposition of a radionuclide is verified.

3.2.4.4 Protective Clothing

3.2.4.4.1 The use of protective clothing shall be specified in Area Operating Procedures and Radiation Work Permits.

3.2.4.4.2 Protective clothing may also be specified by the Health and Safety Group. In the event of conflicts between the Area Operating Procedure, Radiation Work Permit, and the Health and Safety Group, the decision of the latter shall prevail.

3.2.4.5 Respiratory Protection

3.2.4.5.1 The Respiratory Protection Program shall be a responsibility of the Health and Safety Group.

3.2.4.5.2 The Respiratory Protection Program shall be implemented through written and approved procedures.

3.2.4.6 Surface Contamination Monitoring

3.2.4.6.1 The Health and Safety Group shall perform smear surveys in the

below listed areas at the indicated minimum frequencies:

Area	Frequency	Action Level ($dpm/100cm^2$)
<----------ALPHA---------->		
Unirradiated, unencapsulated fuel handling areas	Weekly	5000
Building B Counting Lab.	Monthly	200
Building A Labs.	Monthly	200
Hot Cell Oper. Area	Monthly	200
Scanning Electron Microscopy Lab.	Monthly	200
Exit portals from controlled areas	Biweekly	200
<----------BETA + GAMMA---------->		
Building A Labs.	Monthly	2000
Building B Counting Lab.	Monthly	2000
Scanning Electron Microscopy Lab.	Monthly	2000
Hot Cell Operations Area	Bimonthly	2000
Cask Handling Area	Bimonthly	22000
Radiochemistry Lab.	Bimonthly	22000
Exit portals from controlled areas	Biweekly	2000

3.2.4.6.2 Large area smears are used to survey many square meters of surface area. To determine if these smears indicate that an action level has been exceeded, the assumed area covered shall not exceed 1-square meter.

3.2.4.6.3 Daily surveys shall be performed in the cafeteria, snack bars, and vending machine areas. If contamination is detected in any of these areas, corrective action shall be taken at once.

3.2.4.7 Decontamination

3.2.4.7.1 The Health and Safety Group shall determine and direct the action to be taken to protect personnel and reduce the levels of contamination below those specified in Section 3.2.4.6.

3.2.4.7.2 Decontamination to reduce levels of contamination shall begin within within 24-hours of discovery. If discovery is made just prior to the beginning of a holiday or weekend, the contamination shall be marked and labeled, and decontamination shall commence during the first regular workday after discovery.

3.2.4.7.3 Fixed contamination that, in the opinion of the Supervisor, Health and Safety, does not substantially contribute to a worker's exposure, shall be posted and its location and radiation level recorded and its removal shall be scheduled as soon as practicable.

3.2.4.7.4 Fixed contamination that, in the opinion of the Supervisor, Health and Safety, may substantially contribute to workers' exposure shall be posted and removed as soon as practicable.

3.2.4.8 Emergency Evacuation

3.2.4.8.1 Refer to Radiological Contingency Plan, required by Order dated February 11, 1981, as amended.

3.2.4.9 Personnel Monitoring

3.2.4.9.1 All employees of the LRC shall be issued a TLD monitor. This monitor has a range of from 10 mRem to 10,000 Rem. This dosimetry shall be attached to the employee's identification badge.

3.2.4.9.2 Employees whose annual exposure, as projected by the Supervisor, Health and Safety, will exceed 100 millirem (Radiation Workers) shall be issued a film badge and two indirect reading pocket dosimeters or one self-reading dosimeter and one TLD. This dosimetry shall be worn when the individual is in a radiation, high radiation or airborne activity area.

3.2.4.9.3 Visitors shall wear one TLD which shall be changed weekly. This monitor is inserted in the visitor identification badge and shall be worn at all times while on site.

3.2.4.9.4 Visitors in large tour groups shall be issued one TLD dosimeter for each 10 persons. At least one monitored visitor shall be in each subgroup.

3.2.4.9.5 B & W employees from facilities other than the LRC will not be issued LRC dosimetry if they wear the dosimetry from their own facility. If they do not have their own dosimetry, they shall be monitored the same as other visitors.

3.2.4.9.6 Visitors who perform special work at the LRC may be badged the same as Radiation Workers after appropriate training.

3.2.4.9.7 Delivery truck drivers shall be issued one self-reading dosimeter and a film badge.

4.0 NUCLEAR CRITICALITY SAFETY

4.1 SPECIAL ADMINISTRATIVE REQUIREMENTS

4.1.1 Double Contingency Policy - The Double Contingency Policy as defined in the American National Standard ANSI/ANS-8.1-1983 shall be followed in establishing the basis for nuclear criticality safety of all operations.

4.1.2 Structural Integrity - Where structural integrity is necessary to provide assurance for nuclear criticality safety, the design and construction of those structures will be evaluated with due regard to load capacity and foreseeable abnormal loads, accidents, and deterioration. The Manager, Facilities shall be responsible for determining the structural integrity of equipment when it is necessary to provide assurance of nuclear criticality safety.

4.1.3 Nuclear Criticality Safety Evaluation - All modifications or additions or both to any operation, system or equipment must be approved by the Facility Supervisor. It is the responsibility of the Facility Supervisor, in consultation with the Nuclear Safety Officer, to determine whether or not a nuclear criticality safety evaluation is required for the proposed modification or addition. The Nuclear Safety Officer or a person designated by him shall provide any required evaluations, including calculational support. Nuclear safety evaluations shall be reviewed by a second individual, either the Nuclear Safety Officer or by a person with the same minimum qualifications required for the Nuclear Safety Officer.

4.1.4 Posting - Each unit shall be posted with the limits of SNM permitted in the unit. The Facility Supervisor shall be responsible for approving the posting of nuclear criticality safety limits.

4.1.5 Labeling - Each container containing greater than 0.5 grams of SNM shall be labeled to show the amount of element, the percent enrichment, when applicable, and the amount of fissile isotope. This condition does not apply to irradiated SNM.

4.1.6 Compliance - Compliance with the nuclear criticality safety requirements shall be in accordance with written area operating procedures, reviewed and approved by the Facility Supervisor, the Supervisor, Health and Safety, the Nuclear Safety Officer, and the

SRC. Area operating procedures shall include all the controls and limits significant to the nuclear criticality safety of the operation. In addition, the Nuclear Safety Officer shall perform a quarterly audit for compliance with nuclear criticality safety requirements and verifies that process conditions have not been altered that may affect nuclear criticality safety. The results of the audit shall be documented and submitted to the Manager, Lynchburg Technical Operations.

4.2 TECHNICAL REQUIREMENTS

4.2.1 Nuclear Isolation - When nuclear isolation is required (the potential neutronic interaction between units is negligible) the unit or units isolated shall be separated from all other SNM by one of the following or equivalent conditions:

1. Twelve inches of water.

2. Twelve inches of concrete with density of at least 140 lb/ft3 provided that the isolated unit or units cannot be representable as a slab which interacts with the other SNM primarily through its major face.

3. The edge-to-edge separation of 12-feet, or the greatest distance across an orthographic projection of either accumulation on a plane perpendicular to a line joining their centers, whichever is larger.

4.2.2 Building A

4.2.2.1 General - Building A shall be limited to 40 units as defined in section 1.6 of this Part. Each unit shall be separated from adjacent units by at least 8-inches edge-to-edge and 24-inches center-to-center.

4.2.2.2 Unit Limits - Each unit shall be limited to one of the following:

4.2.2.2.1 Mass limits for mixtures of plutonium and U-235

Pu (wt%)	Limit (total grams fissile)
0	350

1 to 20	313
20 to 40	283
40 to 60	258
60 to 80	237
80 to 100	220

4.2.2.2.2 Mass Limits for Low Enriched Uranium - 850 grams of U-235 as contained in uranium enriched in the isotope U-235 to and including 4 wt%.

1. Whenever uranium enriched to 4 wt% U-235 is being processed under the 850 gram limit, no unit shall be permitted to have fissile material at an enrichment greater than 4 wt% within that laboratory, room, or work area.

2. Whenever an 850 gram, enriched controlled unit is in use in the building, the Facility Supervisor must approve all transfers involving materials with enrichments greater than 4 wt% within the building.

4.2.3 Building B

4.2.3.1 General - Building B shall be limited to 40 units, excluding the hot cells, underwater storage, and the examination of power reactor fuel assemblies. Each unit shall be separated from adjacent units by at least 8-inches edge-to-edge and 24-inches center-to-center.

4.2.3.2 Unit Limits - Each unit shall be limited to the values specified in Section 4.2.2.2 of this Part.

4.2.3.3 Hot Cell - The hot cells, except for examination of power reactor fuel assemblies, shall be limited to the following units:

1. Three units in hot cell no. 1, separated by at least 12-inches edge-to-edge.

2. One unit in each of the other hot cells.

4.2.3.4 Underwater Storage (Transfer Canal & Storage Pool) - SNM in storage under water in the Transfer Canal & Storage Pool shall be in racks or containers limited to the values specified in 4.2.2.2, excluding power reactor fuel assemblies, and separated by 12-inches edge-to-edge.

4.2.3.5 Storage Tubes - SNM in storage tubes shall be limited to the values specified in 4.2.2.2 for each tube. Storage tubes shall be spaced a minimum of 17-inches center-to-center, are approximately 5-inches in diameter, and totally immersed in concrete.

4.2.3.6 Power Reactor Fuel Assemblies - Examination of unirradiated and irradiated power reactor fuel assemblies, including both non-destructive and destructive testing is carried out in Building B subject to existing nuclear criticality safety limits and controls except as modified by the following conditions.

4.2.3.6.1 Fuel assemblies to be studied are identified as:

1. Each assembly shall be of the enriched uranium oxide PWR type with a 15 X 15, or 17 X 17 square pin lattice not greater than 8.6-inches on a side (further identified as a Babcock & Wilcox Mark B or Mark C canless assembly).

2. The maximum initial enrichment in an unirradiated assembly shall not exceed 4.05 wt%.

3. Damaged fuel assemblies may be examined in air. Fuel assemblies which have been damaged can be examined in water if they maintain their 8.6-inches on a side dimensions.

4.2.3.6.1.1 Other PWR or BWR fuel assemblies which do not meet the above may be studied, provided:

1. The unirradiated, fully reflected fuel assembly (fueled with UO_2 only) with all control rods removed is shown by an appropriate nuclear safety evaluation to be subcritical by at least 5 % (K-eff <0.95).

2. The fuel assembly is shown by an appropriate nuclear safety evaluation to be subcritical by at least 5 % (K-eff <0.95) under specific conditions of disassembly.

3. In 4.2.3.6.1.1 Items 1 and 2 above, the primary source for validation data shall be:

   - DP-1014, "Critical and Safe Masses and Dimensions of Lattices of U and UO_2 Rods in Water," by H. K. Clark of Savannah River Laboratories, or;

- Actual critical experiments having approximately the same enrichment and metal-to-water ratio as is present in the fuel assembly to be studied, or;

- Criticality data supplied by the reactor designer for the actual fuel assemblies.

An appropriate cross-section set will be selected and used to calculate the validation data from one of the above described sources. Any bias between calculational results with that cross-section set and validation data will be included in the results of the safety evaluation called for in #1 and #2 above. A description of and the results of the validation and the nuclear safety analysis will be assembled into a report which shall contain the items specified in Section 4.3.6 of ANSI/ ANS-8.1-1983, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."

4.2.3.6.1.2 BWR fuel assemblies may be received and studied provided:

1. They are evaluated pursuant to Section 4.2.3.6.1.1 of this Part, or

2. The BWR fuel assemblies have a maximum initial unirradiated enrichment of 4.05 wt% U-235 and have a cross sectional area not exceeding that of a 22.5 cm (8.85 in.) diameter cylinder.

4.2.3.6.2 Receipt and Storage

4.2.3.6.2.1 Unirradiated Fuel Assemblies - Unirradiated fuel assemblies will be received at a maximum of two at a time in a shipping container licensed for two assemblies, or one assembly in a shipping container licensed for one assembly. Unirradiated fuel assemblies may be stored in air in the Crane & Cask Handling Area, the Assembly & Machine Shop Area, or the Development Test Area subject to the following conditions:

1. Assemblies may be stored in their shipping container as received.

2. Assemblies may be stored no less than 21-inches apart center-to-center.

3. Assemblies may be stored under water in the hot cell pool, mockup pool, or development test area pool in racks constructed to maintain a 1-foot minimum surface-to-surface separation between assemblies and any other SNM.

4. No more than four unirradiated assemblies may be kept at the LRC at one time.

4.2.3.6.2.2 Irradiated Fuel Assemblies - Irradiated fuel assemblies shall be received one at a time in a licensed single assembly shipping container or two at a time in a shipping container licensed for two assemblies. Fuel assemblies that have been irradiated will be stored in the hot cell pool which is limited to the following conditions:

1. A maximum of four assemblies or portions thereof may be in the pool at a time.

2. The assemblies shall be stored in a storage rack which is so constructed as to maintain a 1-foot minimum surface-to-surface separation between the stored assemblies and any other fissile material which might be in the pool.

3. Only one assembly may be in a designated work area of the pool at any one time. There shall be at least 1-foot minimum surface-to-surface separation between the assembly in the work area and any other fissile material.

4. Fuel rods which have been removed from an assembly shall be stored in a storage rack providing space in each position for a maximum of 75 rods. All positions shall be spaced from any other fissile material by a minimum surface-to-surface separation of 1-foot.

5. Partially dismantled assemblies will be stored in the assembly storage rack.

6. Each position in the assembly storage rack and in the fuel rod storage rack must limit contained fuel to a square not to exceed the dimensions of a fresh fuel assembly or to a cross sectional area not exceeding that of a 22.5 cm (8.85 in.) diameter cylinder.

4.2.3.6.3 Work Area of Pool Under Hot Cell No. 1 - The work area position

under Cell No. 1 is used to load and unload irradiated fuel assemblies into and out of shipping casks and to dismantle both irradiated and unirradiated fuel assemblies. The following conditions govern operations in this work area:

1. Only two assemblies at a time shall be permitted outside of their shipping container provided they are separated by a minimum surface-to-surface separation of 1-foot.

2. An associated storage position shall be permitted for fuel rods or components which have been removed from the assemblies.

3. The assemblies and associated rod storage positions shall be separated from each other and any other fissile material by a minimum surface-to-surface separation of 1-foot.

4. Fissile material and fuel rods or components in the associated storage positions shall each be restricted to a square not exceeding the dimensions of a fresh fuel assembly or to a cross sectional area not exceeding that of a 22.5 cm (8.85 in.) diameter cylinder.

5. Only one fuel rod at a time shall be removed from or inserted into the assembly or the rod storage position. A Maximum of 75 rods shall be permitted in the rod storage position.

6. A fuel assembly may be completely dismantled by withdrawing one fuel rod at a time from the assembly; during all stages of dismantlement, the partially dismantled assembly shall be maintained within the confines of a square not exceeding the dimensions of a fresh fuel assembly or to a cross sectional area not exceeding that of a 22.5 cm (8.85 in.) diameter cylinder.

7. An assembly and its associated rod storage position may be withdrawn from the pool into the cell. Free water drainage from both the assembly and rod storage position as well as 1-foot separation from other fissile materials and each other shall be assured.

4.2.3.6.4 Assembly and Machine Shop and Development Test Area - The work areas on the first floor of Building B may be used to

disassemble unirradiated fuel assemblies for testing. The following conditions govern operations in the work area:

1. Only one assembly at a time shall be permitted to be dismantled.

2. An associated storage position will be permitted for fuel rods which have been removed from the assembly, and shall be spaced and stored as stated in items 3 and 4 (4.2.3.6.4) below.

3. The assembly and associated rod storage position shall be separated from each other and any other fissile material by a minimum surface-to-surface separation of 21-inches.

4. The associated rod storage position shall be no larger in any dimension than the fuel assembly. There shall be one such storage position for each assembly to be dismantled. Rods may be stored or handled in a slab up to 4-inches thick provided the slab is separated from other fissile material by a minimum of 12-feet.

5. Only one fuel rod at a time may be removed from or inserted into the assembly or any rod storage position. Only one rod may be in transit to any one location at a time.

6. The fuel assembly may be completely disassembled by withdrawing one fuel rod at a time from the assembly; during all stages of disassembly, the partially disassembled assembly shall be maintained within the confines of the assembly whether damaged or undamaged.

7. Fuel rods may be removed one at a time from this area as required. These rods shall be subject to all fuel handling requirements pertinent to the area they are in.

8. Assemblies may be handled and dismantled under water in these areas (mock-up pool and development test area pool) subject to the same requirements of the hot cell pool.

4.2.3.6.5 Hot Cell Operations - Fuel rods removed from irradiated assemblies may be examined including destructive examination in the hot cells. Operations in the hot cells shall be governed according to the following conditions:

1. An assembly and its associated rod storage position may be withdrawn from the pool into Hot Cell No. 1 pursuant to Item No. 7 of Section 4.2.3.6.3 of this Part.

2. Two units in Hot Cell No. 1 may have a total of 64 fuel rods each, stored, provided that rods shall be confined within a cross sectional area not exceeding that of a 22.5 cm (8.85 in.) diameter cylinder, drainage of any free water within the unit shall be assured and the units must be maintained 1-foot from each other and any other SNM in the cell.

3. In addition to the two units of stored rods, another unit limited to the values in 4.2.2.2.1 may be present in Hot Cell No. 1. In this unit under mass control, rods may be destructively examined.

4.2.3.6.6 Fuel Rod Dismantlement - Fuel rods from unirradiated assemblies can be dismantled in any area where the license permits handling of unirradiated fuel. The following conditions must also be met in areas to dismantle fuel rods:

1. The area shall be mass limited to 350 grams of U-235. This area must be separated from the assembly and slab storage area by minimum of 12-feet.

2. Dismantlement must be completed under approved procedures.

4.2.3.6.7 Shipment and Disposal - After examination, assemblies, partially dismantled assemblies, fuel rods, and scrap generated during destructive examination shall be disposed of according to the following conditions.

1. Fuel rods, including fuel rod segments may be placed in any available hole in a fuel assembly, including the instrument and control rod guide tube positions, i.e., 225 and 285 fuel rods in Mark B and Mark C assemblies, respectively. Fuel rod segments shall have their ends sealed, and shall be encapsulated in steel tubing with ends sealed, prior to insertion into an available hole in a fuel assembly.

2. Unirradiated assemblies may be reassembled (one rod at a time) for later use.

3. Assemblies, including partially dismantled assemblies, shall be loaded into shipping casks approved for such assemblies for shipment.

4. Scrap, including rod segments, shall be disposed of according to present LRC procedures and limits.

4.2.4 Building C

4.2.4.1 General - Building C is limited to 90 units. Each unit shall be separated from adjacent units by at least 8-inches edge-to edge and 36-inches center-to-center.

4.2.4.2 Unit Limits - Each unit shall be limited to the values specified in section 4.2.2.2 of this Part.

4.2.5 Outside Storage

4.2.5.1 General - Outside storage consists of underground storage, shipments, and the fenced storage area located adjacent to Building J.

4.2.5.2 Underground Storage - Radioactive materials stored in underground storage tubes shall be limited to one SNM unit per tube, with values as specified in section 4.2.2.2 of this Part. Tubes shall be spaced 20-inches center-to-center and are 5-inches in diameter, and totally immersed in concrete.

4.2.5.3 Shipments - Each shipment of fissile material being stored outside must conform with all license requirements for the type of shipping container. Additionally, each shipment must be nuclearly isolated from all other SNM.

4.2.6 Dry Waste - Dry waste is accumulated in 55-gallon drums, or other suitable containers, with a maximum of 45 grams of SNM per container. These containers may be located throughout the laboratories as required to collect contaminated laboratory waste. Filled containers are transferred, to the radioactive waste storage building after scanning. Dry waste containing 0.5 grams of SNM or less per container may be stored in a fenced, locked and paved outside storage area adjacent to Building J.

5.1.3.6 Waste tanks that indicate concentrations of activity greater than those specified in section 5.1.3.2 shall be appropriately diluted prior to release.

5.1.3.7 The NNFD must approve the release of liquid waste to their waste treatment facility prior to the release.

5.1.3.8 The 10,000 sq. ft. Storm Drain Collection Pond shall be grab sampled quarterly. The sample shall be analyzed for gross alpha and gross beta.

5.1.4 Gaseous Effluent - Discharge air from process areas is released to the general environment through the 50-meter high stack. The discharge rate of the stack is approximately 20,000 cubic feet per minute. The annual discharge volume is 1.1E10 cubic feet.

5.1.4.1 Action levels - The action levels for releases from the stack are specified in section 3.2.2.5 of this Part I.

5.1.4.2 Analyses - The fixed filter of the stack particulate monitor shall be counted on the Low Beta or Wide Beta counting system, after an appropriate decay period. The results shall be recorded and maintained on file.

5.1.4.3 Sampling - The stack shall be sampled isokinetically on a continuous basis.

5.1.4.4 Monitoring - The stack sample shall be passed through a monitoring system that consists of the following:

1. Particulate Monitor - The stack particulate monitor consists of an alpha and beta channel, with a dual channel ratio detector. This monitor uses a fixed filter and a nominal sampling flow rate of 2 - 3 cubic feet per minute. The detector is a thin window (1.0 mg/cm2) gas flow proportional detector. Alpha and Beta-gamma radiations are monitored through two single channel analyzers and log rate meters. The ratio between these two channels is also displayed as a logrithmic ratio. This system effectively compensates for the presence of Radon and Thoron daughters and increases the sensitivity of the system. Alpha and Beta-gamma

6.0 SPECIAL PROCESS COMMITMENTS

The Lynchburg Research Center is engaged in research and development and for this reason there are a large number of small special processes that are special only because they are outside of the few repetitive operations. Examples of operations performd at the LRC are given in section 1.7.1. The LRC relies on established administrative controls to determine what proposals fall outside of the bounds of work that is authorized by the license, in which case amendments are applied for. Those proposals that are authorized by the license but are significantly different from previously reviewed proposals shall be brought before the Safety Review Committee for review and approval.

7.0 DECOMMISSIONING PLAN

The Lynchburg Research Center is committed to decommissioning the facilities which have been used for the use and storage of licensed material at the end of their useful life. At the time of this application for renewal of License No. SNM-778, two programs are underway to decommission Buildings A and C. It is presently estimated that these two facilities will be decontaminated and ready for release for unrestricted use by March, 1986.

7.1 PLANNING CONSIDERATIONS

7.1.1 The history of the facility shall be determined to facilitate the identification of services, equipment, and areas that should be included in the survey plan.

7.1.2 The decontamination of facilities and equipment must meet the levels of contamination specified in Table 1, Annex C to License SNM-778, Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material, dated November, 1976. In addition, a reasonable effort will be made to further reduce contamination levels to those which are as low as reasonably achievable.

7.1.3 No covering will be applied to remaining surfaces until it has been determined that contamination levels are below those of Table 1, Annex C to License SNM-778, Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material, dated November, 1976, and until it has been determined that a reasonable effort has been made to further reduce contamination below those specified above.

7.1.4 The radioactive contamination of interior surfaces of pipes, ductwork, and other conduits will be determined by taking measurements at all traps and other appropriate access points, provided contamination at these locations is likely to be representative of interior conditions. If such access locations are not likely to be representative, or if interior surfaces are inaccessible, the interior surfaces will be assumed to be contaminated in excess of levels specified in Table 1, Annex C to License SNM-778, Guidelines for Decontamination of Facilities and

Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material, dated November, 1976.

7.1.5 A radiological survey plan shall be designed and implemented to assess the extent of decontamination necessary. The survey plan will include the taking of smear samples of roofs, ceilings, walls, floors, and equipment. It shall also include the taking of core samples of concrete floors, walls, ceilings and pools. Core samples shall also be taken of soil beneath floors and pools and in the vicinity of under ground drainlines. Records of the survey and sample results shall be prepared and maintained.

7.1.6 Equipment will be disposed of by burial or sale. The decontamination of equipment scheduled for burial will be sufficient to meet the requirements of transportation and those of the receiving facility. The decontamination of equipment scheduled for sale to a licensed facility will be sufficient to meet the requirements of transportation and those of the receiving facility. Equipment scheduled for receipt by a nonlicensee shall be decontaminated to levels specified in Table 1, Annex C to License SNM-778, Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material, dated November, 1976.

7.1.7 At the completion of decommissioning activities, a report of the final survey shall be prepared and submitted to the NRC with a request that a confiratory inspection be made and that the facility be released for unrestricted use.

7.2 COST AND FINANCIAL ARRANGEMENTS

7.2.1 Financial Arrangements - Financial arrangements to cover the cost of decommissioning are set forth in the letter dated September 16, 1985, from L. V. Jordan, Assistant Controller.

7.2.2 Cost Estimate Guidelines - The following are the general guidelines upon which the estimates are based.

7.2.2.1 No facility will be razed.

7.2.2.2 No "mothballing" of any facility or perpetual surveillance is contemplated. All facilitis will be released for unrestricted use upon approval by NRC.

7.2.2.3 "Reasonable" efforts to decontaminate below levels specified in Table 1, Regulatory Guide 1.86, dated June 1974, shall be required only to the extent that the benefits derived clearly justiry the additional cost expenditures.

7.2.2.4 There will be no requirement to prepare or furnish prior to, concurrent with, or subsequent to decommissioning, any environmental data or impact reports.

7.2.2.5 Physical security pursuant to 10 CFR 73 will not be required at the time the Decommissioning Plan is implemented.

7.2.2.6 All estimates are given in 1985 dollars. No factor for escalation has been included.

7.2.3 Technical Guidelines - The following are the technical guidelines upon which the estimated costs are based.

7.2.3.1 Surface removal will be required for decontamination of the interior surface of hot cells, transfer canal and storage pool, liquid waste retention basins 5K-1, 5K-2, 10K-1, 10K-2 and the Building A basin, primary equipment cell pump pit, and the Building B Containment Pool.

7.2.3.2 All plutonium contaminated glove boxes will be disposed of at burial facilities in Richland, Washington.

7.2.3.3 No decontamination of cold areas will be required.

7.2.3.4 Removal of ventilation ducting beyond the first stage of HEPA filtration will not be required.

7.2.3.5 Removal of piping in the liquid waste system will be required.

7.2.3.6 Disassembly and off site burial of liquid waste tanks 2K-1, 2K-2, 2K-3, 13K, 4K-1, 4K-2, 300-1 and 300-2 will be required.

7.2.3.7 Disassembly and off site burial of the waste storage tubes will be required.

7.2.3.8 Health and Safety labor will be approximately 20 percent of the total labor cost.

7.2.3.9 Rental of special shipping containers will be required.

7.2.4 Cost Estimate

7.2.4.1	Building A	$500,000
	Building B	$8,500,000
	Building C	$4,500,000
	Building J	$500,000
	Liquid Waste Disposal Facility	$1,000,000
	TOTAL	$15,000,000

8.0 RADIOLOGICAL CONTINGENCY PLAN

The Lynchburg Research Center shall maintain and execute the response measures of the Radiological Contingency Plan submitted to the NRC on November 15, 1983, in accordance with provisions of the February 11, 1981 order. The LRC shall maintain implementing procedures for the Radiological Contingency Plan as necessary to implement the Plan. The LRC shall make no change in the Radiological Contingency Plan that would decrease the response effectiveness of the Plan without NRC approval. The LRC may make changes to the Radiological Contingency Plan without prior NRC approval if the changes do not decrease the response effectiveness of the Plan. The LRC shall maintain records of changes that were made to the Plan without prior approval for a period of two years from the date of change and shall furnish the Chief, Uranium Fuel Licensing Branch, Division of Fuel Cycle and Material Safety, NMSS, USNRC, Washington, DC 20555 and Region II, a report containing a description of each change within six months after the change is made.

TABLE OF CONTENTS

Section		Page
10.0	FACILITY DESCRIPTION	10-1
10.1	PLANT LAYOUT	10-1
10.2	UTILITIES INCLUDING EMERGENCY POWER	10-1
10.2.1	Potable Water	10-1
10.2.2	Process Water	10-1
10.2.3	Gas	10-1
10.2.4	Fuel Oil	10-1
10.2.5	Electricity	10-1
10.3	HEATING, VENTILATION, AND AIR CONDITIONING	10-2
10.3.1	Heating	10-2
10.3.2	Ventilation	10-2
10.4	WASTE HANDLING	10-4
10.4.1	Liquid Wastes	10-4
10.4.2	Solid Wastes	10-5
10.5	FIRE PROTECTION	10-6
10.5.1	Codes and Standards	10-6
10.5.2	Insurance Inspection Reports	10-8
10.5.3	Fire Protection Equipment	10-9
10.5.4	Combustible Waste Storage	10-9

List of Figures

Figure		Page
10-1A	BUILDING A (SECOND FLOOR)	10-10
10-1B	BUILDING A (FIRST FLOOR)	10-11
10-2A	BUILDING B (FIRST FLOOR)	10-12
10-2B	BUILDING B (SECOND FLOOR)	10-13
10-3	BUILDING C	10-14
10-4	BUILDING B VENTILATION SYSTEM	10-15
10-5	BUILDING B RADIOACTIVE LIQUID WASTE PIPING DIAGRAM . .	10-16
10-6	FLOW DIAGRAM FOR LIQUID WASTE BUILDING	10-17

10.4.2 Solid Wastes

10.4.2.1 Solid wastes are generated in Building A and C as a result of the decommissioning efforts in progress in those buildings and in Building B as a result of normal operations associated with the Hot Cells, Radiochemistry Laboratory, Failure Analysis Laboratory, Fatigue and Fracture Laboratory, and support facilities. Solid waste that is generated in the Cask Handling Area and the Radiochemistry Laboratory is compacted. These wastes are generally low level byproduct materials. High level wastes generated in the Hot Cells are not compacted.

10.4.2.2 Solid radioactive waste is stored, awaiting shipment for offsite disposal, in Building J, the Building J Annex, the high level waste storage tubes, and the outside storage area. Waste is stored in closed containers suitable for offsite shipment. In Building J these containers may be any approved by DOT or NRC. Containers that are stored outdoors must be constructed of metal. Any container may be stored outdoors for short periods of time incidental to transportation. Waste stored in the high level storage tubes must be packaged in containers used in the removal of waste from the hot cells.

10.4.2.3 Building J provides 1400 square feet of storage space. The building is equipped with a smoke detector, air sampler, and a criticality monitor. The interior of the building is divided into three areas that are partitioned by concrete block walls. This permits storage of high, intermediate, and low level waste in the same facility in a manner that results in minimum exposure to personnel maneuvering waste within the building. The maximum quantity of SNM permitted in Building J is 45 grams per container. Containment of stored waste is assured by monthly smear sampling of the Building J floor. The continuous air monitor in Building J is provided with an alarm to alert personnel of the presence of airborne radioactivity within the Building.

10.4.2.4 The Building J Annex is constructed of unmortared concrete block. Three walls consist of four courses of block which provide about a four-foot thick shield wall. The wall that is adjacent to Building J consists of three courses of block. The Annex is provided with exhaust ventilation through ducting which connects Building J with the Annex thereby permitting the smoke detector and air sampler in Building J to serve both. The Annex is provided with a metal roof which is hinged to Building J, capable

of being locked and provided with side panels which permit the roof to fit flush with the top of the block walls. Containers are loaded into the Annex from the top. A curbing will be placed on the approach side of the addition to prevent a loading vehicle from accidently contacting the wall. Two individuals are involved in loading containers into this facility to prevent a container from striking the walls. This facility provides storage of waste that is contaminated with irradiated fuel and is being stored on site until it is accepted by the DOE under the Nuclear Waste Policy Act of 1982. The maximum quantity of SNM per container shall be limited to 45 grams.

10.4.2.5 The Outside Waste Storage Area is located adjacent to Building J. This area is fenced, locked and paved. Waste stored in this area is limited to that contained in closed metal containers. Each container is limited to not more than a Type A quantity (10 CFR 71.4) or 0.5 grams of fissile material or both. Pu shall not be stored in this area. Containment of stored waste is assured by a quarterly visual inspection by the Supervisor, Health and Safety.

10.4.2.6 The High Level Waste Storage Tubes are located adjacent to the south side of the Liquid Waste Disposal Facility. These tubes are constructed of two sections of iron pipe, immersed in concrete, and below ground level. The upper section of pipe (approximately 42-inches long) is 6-inches in diameter. The lower section (approximately 80-inches long) is welded to the upper section and is 5-inches in diameter. Each tube is fitted with a concrete-filled iron plug. These tubes are locked and under the direct control of the Health and Safety Group. Waste stored in these tubes is limited to that which is produced in the Hot Cells and must be in closed metal containers. The quantity of fissile material permitted in each tube is limited to one unit.

10.5 FIRE PROTECTION

10.5.1 Codes and Standards - The development and building construction program of the Lynchburg Research Center complex has taken place over the period 1955 to the present. For the three main buildings under consideration in this renewal request, the design and construction efforts took place from 1955 to 1969. There have been a number of alterations and use changes over the past ten years, but generally these changes have not significantly altered the

structural characteristics of the buildings.

All three buildings were built as staged or "added-on" phased construction. Building A was built in four distinct phases, Building B was built in two stages, and Building C was constructed in five phases. Building A was designed in-house by B&W engineering personnel. Building B was designed by Wiley & Wilson, a Lynchburg consulting engineering firm. Building C was also primarily designed by Wiley & Wilson, with some design by B&W.

The physical layout of all three buildings is highly functional, i.e., based on the specialized requirements of research work related to the nuclear industry. For the most part, the building structure envelopes are quite conventional in nature, both from a design and construction materials standpoint. With the exception of highly specialized portions of these buildings, such as the hot cells, engineering design of the buildings would be considered as state-of-the-art for light industrial/heavy commercial class buildings (for each of the design and construction time periods involved).

The overall quality of the building construction is well above average. Aside from some roof leakage problems and minor settlement cracking in some of the masonry construction, the performance of the building structures and envelopes has been good. There have been no repairs related to significant structural defects in any of the three buildings. As would be anticipated for a complex of this type and importance, maintenance of the buildings has been excellent and contributes to the overall good condition of such a facility.

During the period of design and construction for Buildings A, B, and C, it should be noted that there was very little in the way of code requirements or guidance for construction of such a facility. Virginia did not adopt a state-wide building code until September 1, 1973. Up until that time, various localities in the state had adopted their own local building codes; the Southern Building Code being the one generally used. Many counties, however, had no code at all; Campbell County, in which the LRC complex is located, had no building code during this time period. The only state-wide code directly applicable to building construction prior to 1973, was the Virginia Fire Safety Regulations, enacted in 1949.

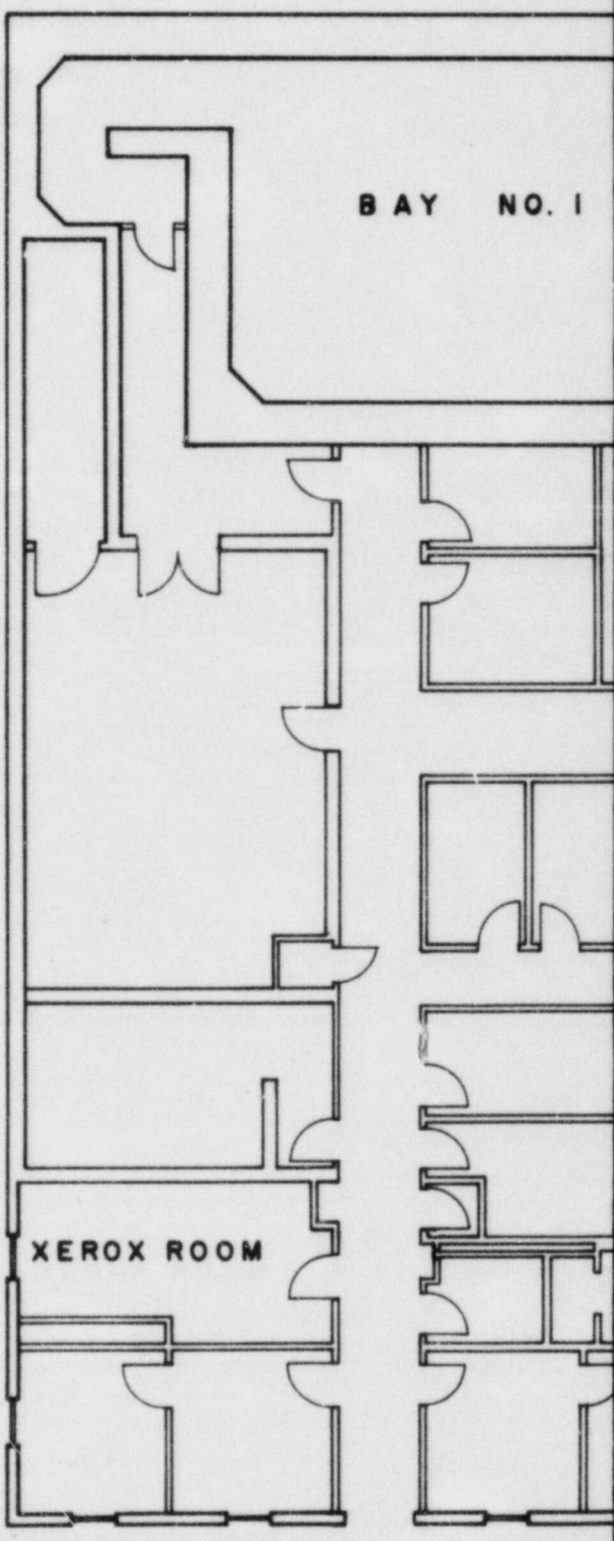
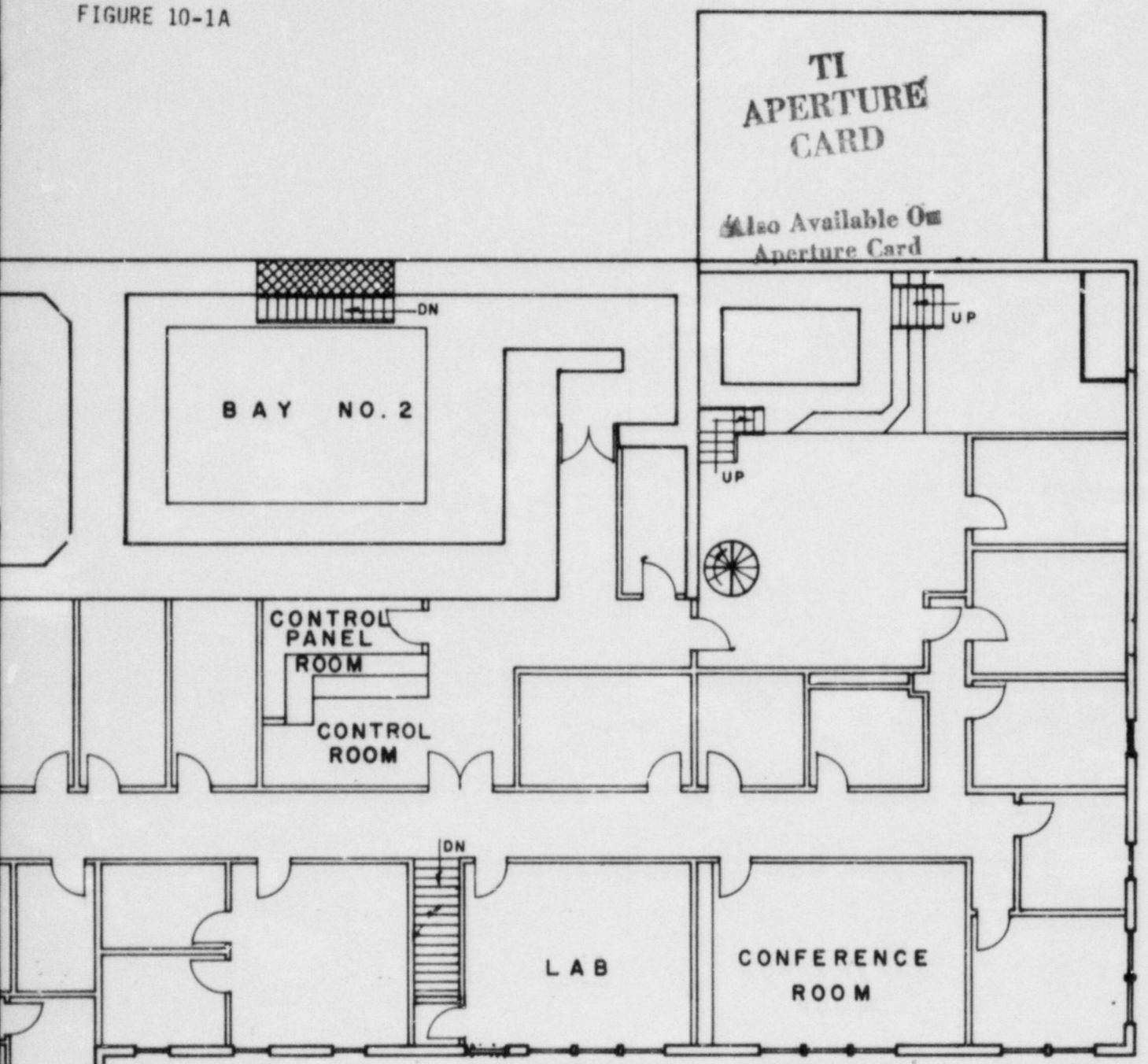

BAY NO. 1
XEROX ROOM

FIGURE 10-1A



BUILDING "A"

SECOND FLOOR

License No	SNM-778	Docket No.	70-824	Date	May, 1986
Amendment No.	0	Revision No.	2	Page	10-10

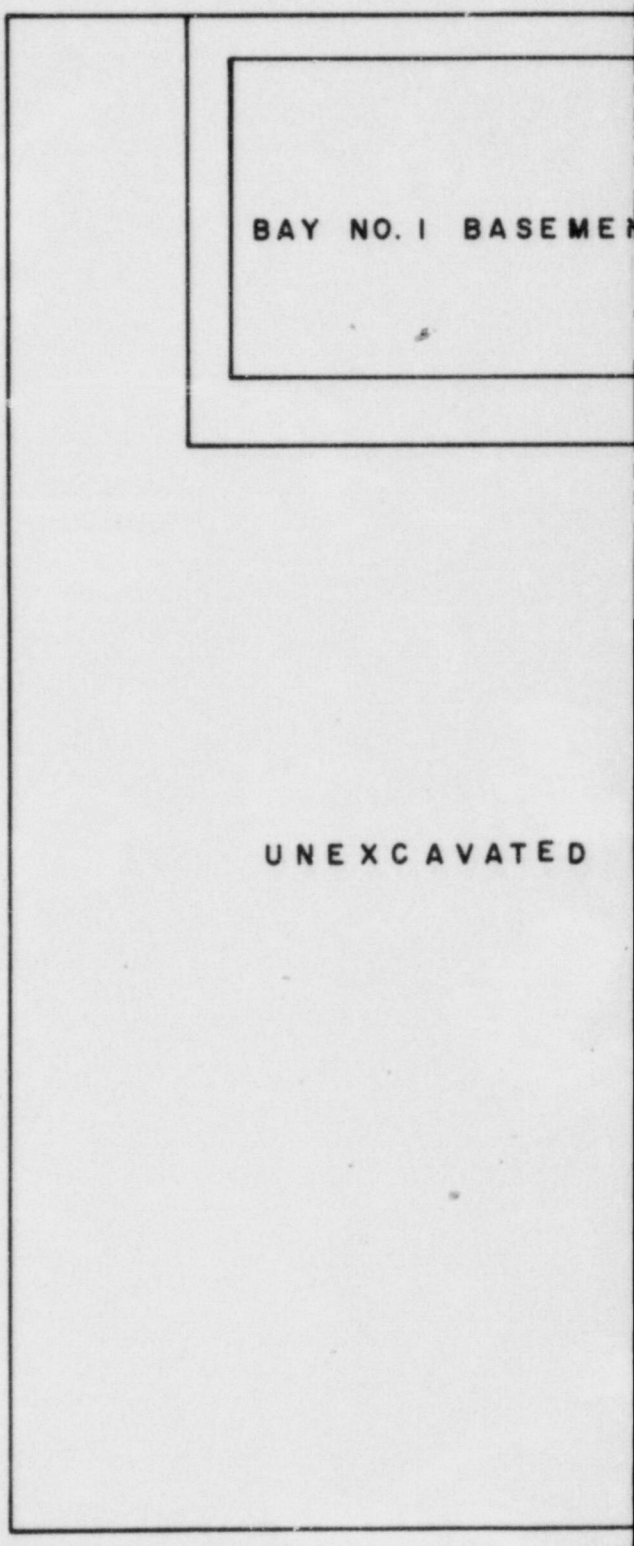
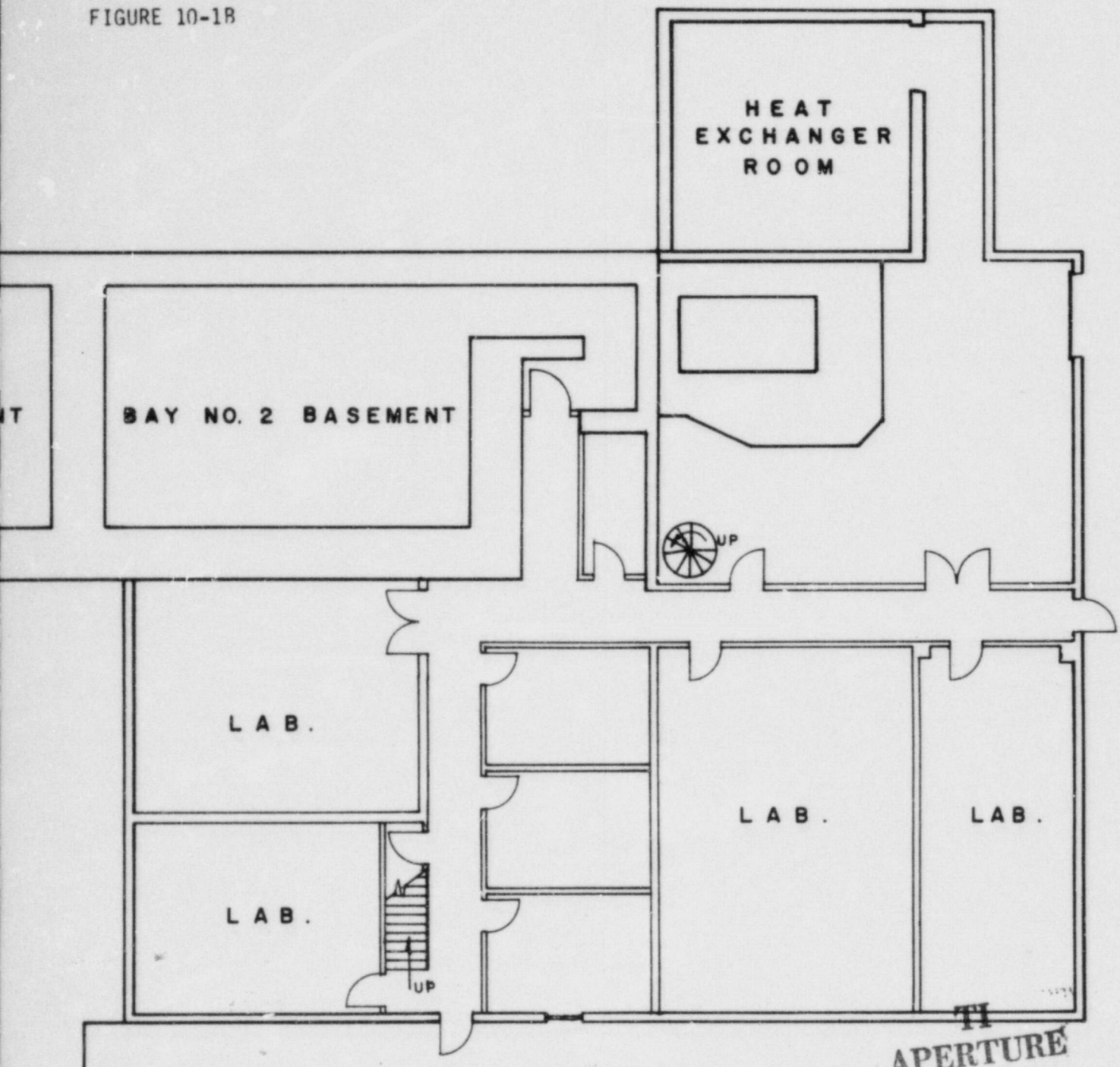

BAY NO. 1 BASEME
UNEXCAVATED

FIGURE 10-1B



BUILDING "A"

FIRST FLOOR

License No	SNM-778	Docket No.	70-824	Date	May, 1986
Amendment No.	0	Revision No.	2	Page	10-11

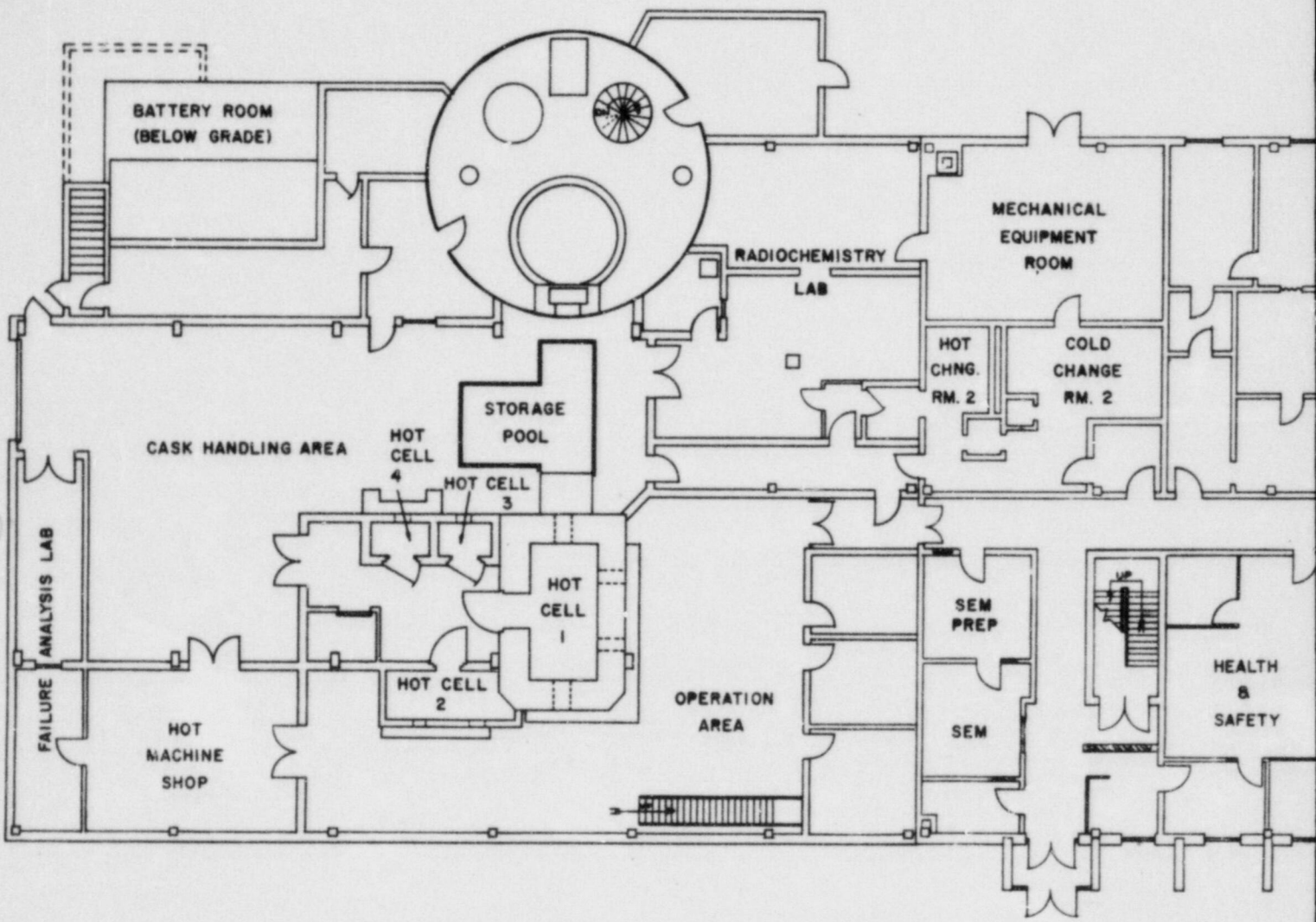
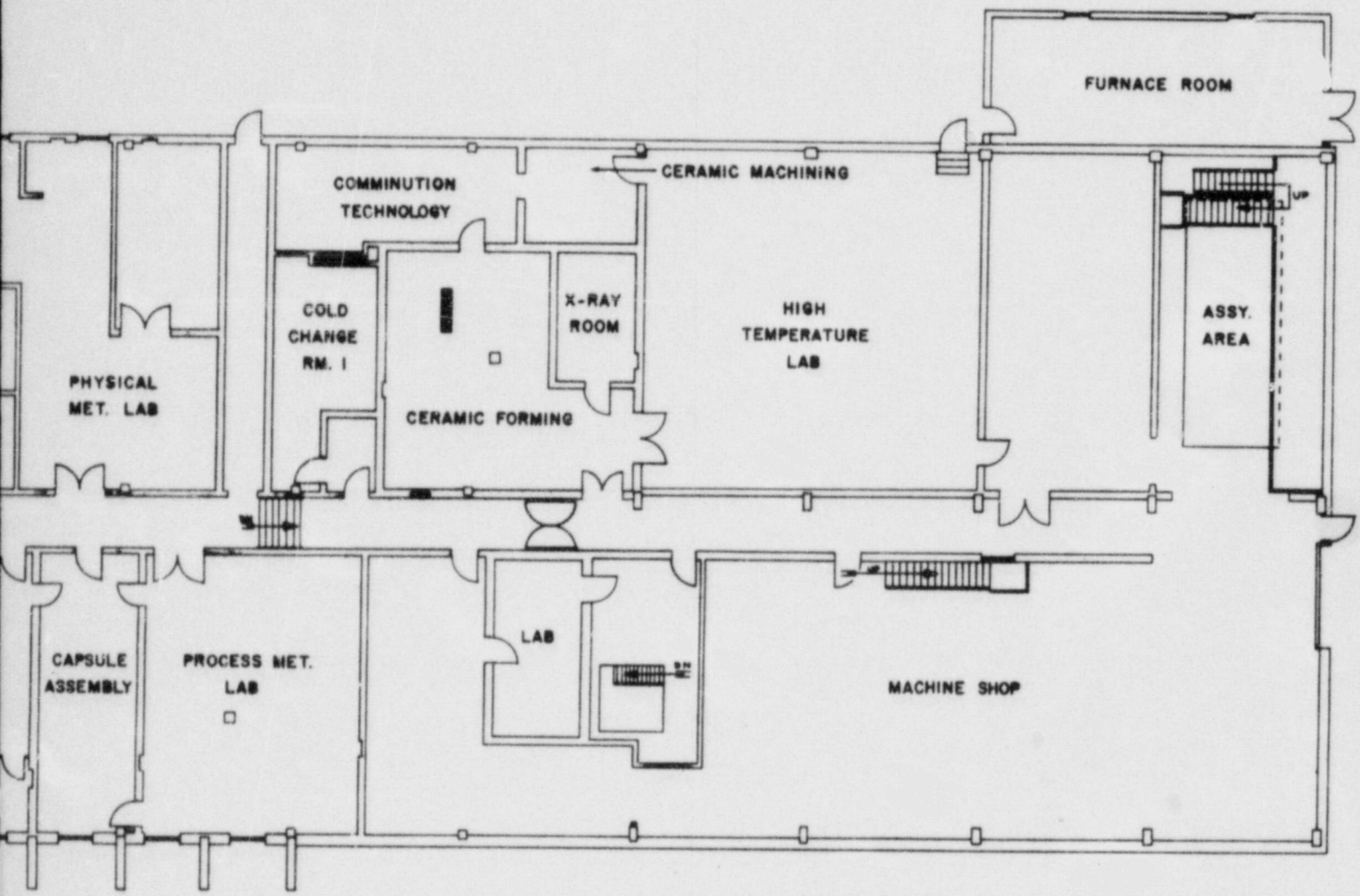

BATTERY ROOM
(BELOW GRADE)
RADIOCHEMISTRY
LAB
MECHANICAL
EQUIPMENT
ROOM
HOT
CHNG.
RM. 2
COLD
CHANGE
RM. 2
STORAGE
POOL
CASK HANDLING AREA
HOT
CELL
4
HOT CELL
3
HOT
CELL
1
HOT CELL
2
FAILURE ANALYSIS LAB
HOT
MACHINE
SHOP
OPERATION
AREA
SEM
PREP
SEM
UP
HEALTH
&
SAFETY

FIGURE 10-2A



BUILDING "B"
1ST FLOOR
SCALE: 1" = 20'
REVISED APRIL 1984

8606200356-04

License No	SNM-778	Docket No.	70-824	Date	May, 1986
Amendment No.	0	Revision No.	2	Page	10-12

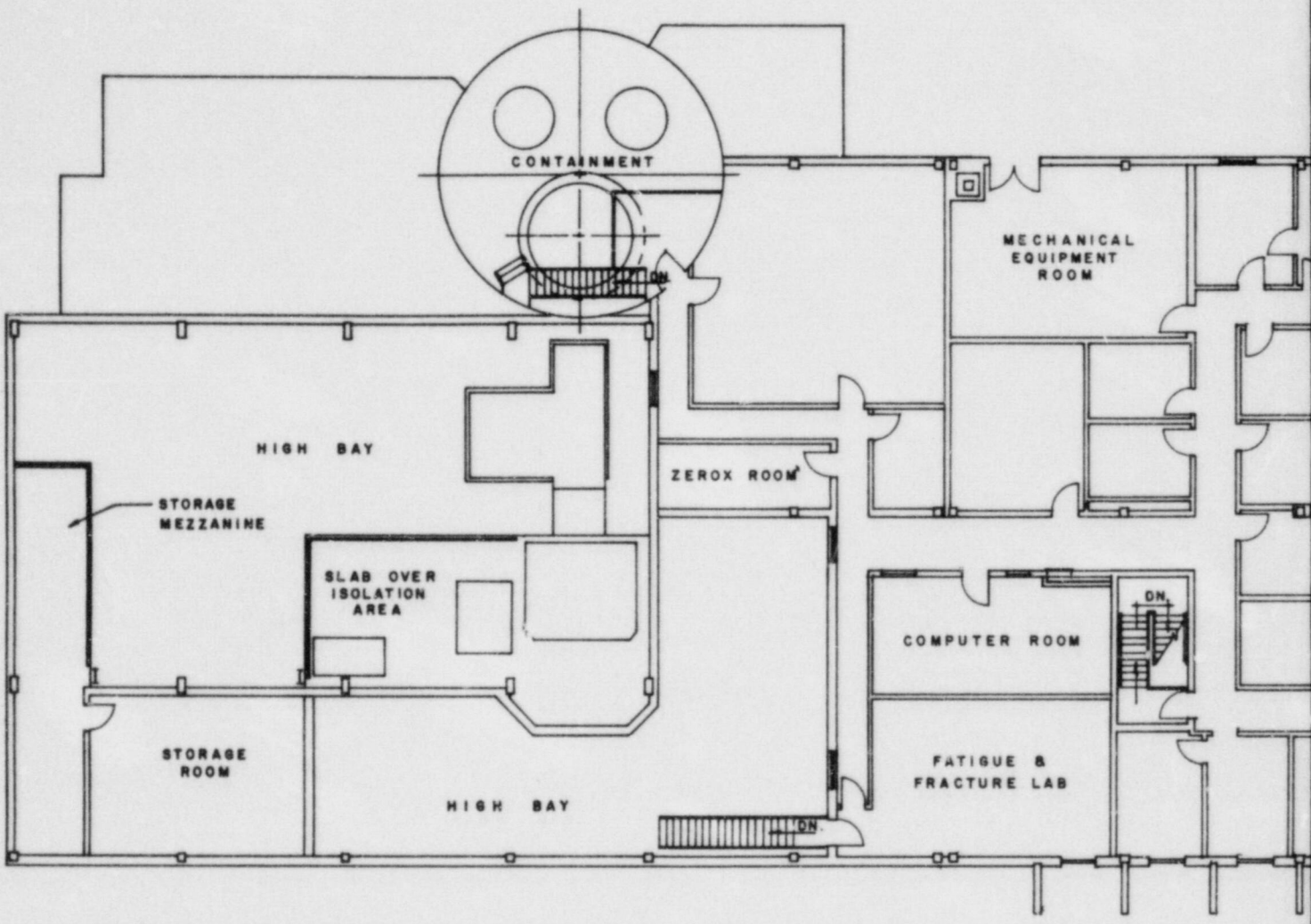
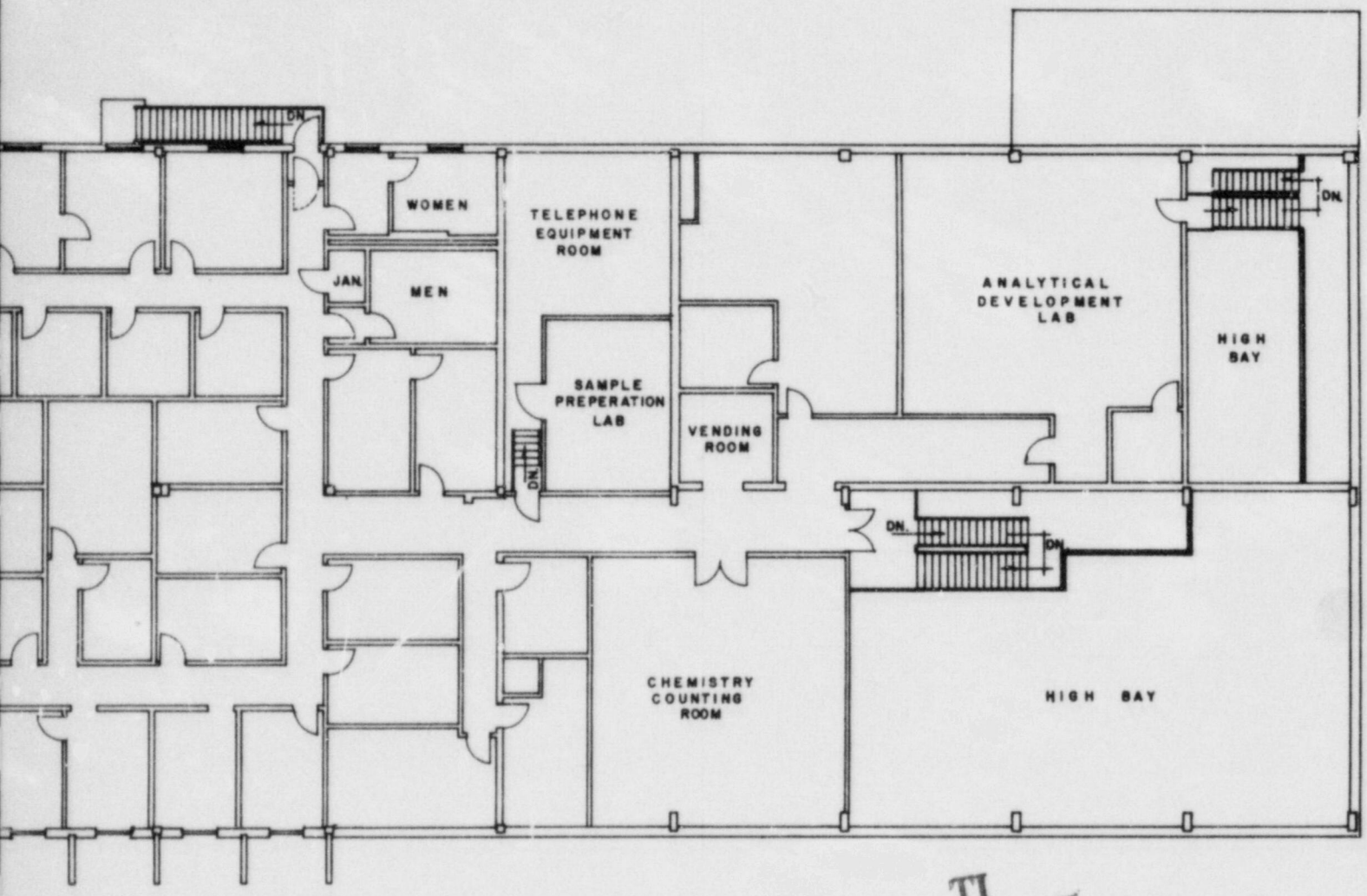

CONTAINMENT
DN.
MECHANICAL EQUIPMENT ROOM
HIGH BAY
ZEROX ROOM
STORAGE MEZZANINE
SLAB OVER ISOLATION AREA
DN.
COMPUTER ROOM
STORAGE ROOM
HIGH BAY
DN.
FATIGUE & FRACTURE LAB

FIGURE 10-2B



BUILDING "B"

SECOND FLOOR PLAN

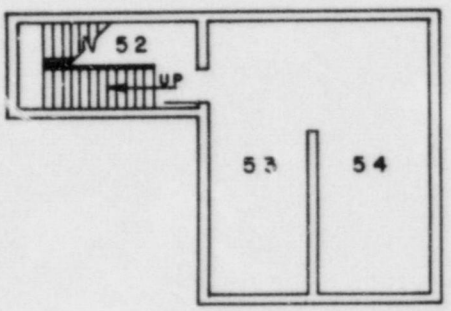
SCALE : 1" = 20'

REVISED : APRIL 1985

0' 80'

8606200356-05

License No	SNM-778	Docket No.	70-824	Date	May, 1986
Amendment No.	0	Revision No.	2	Page	10-13



BASEMENT

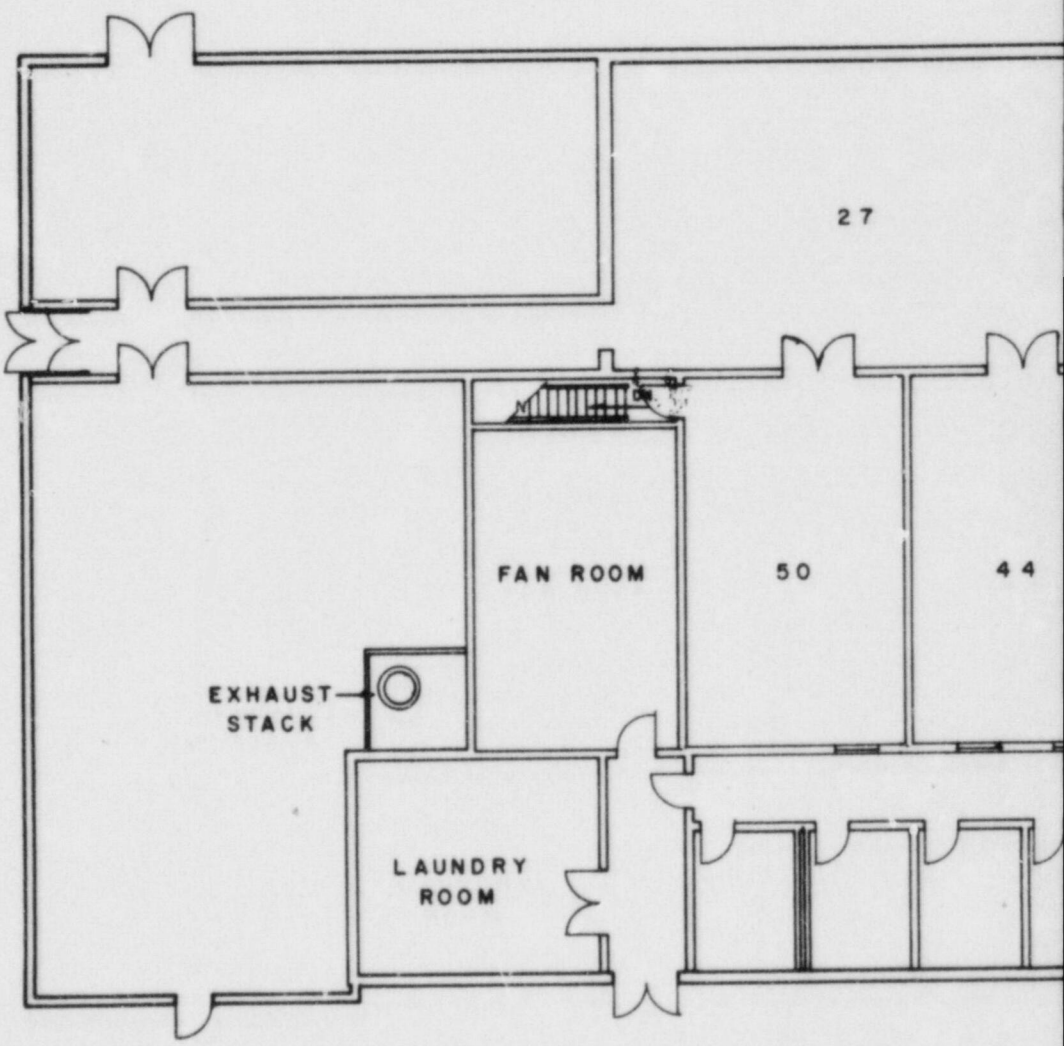
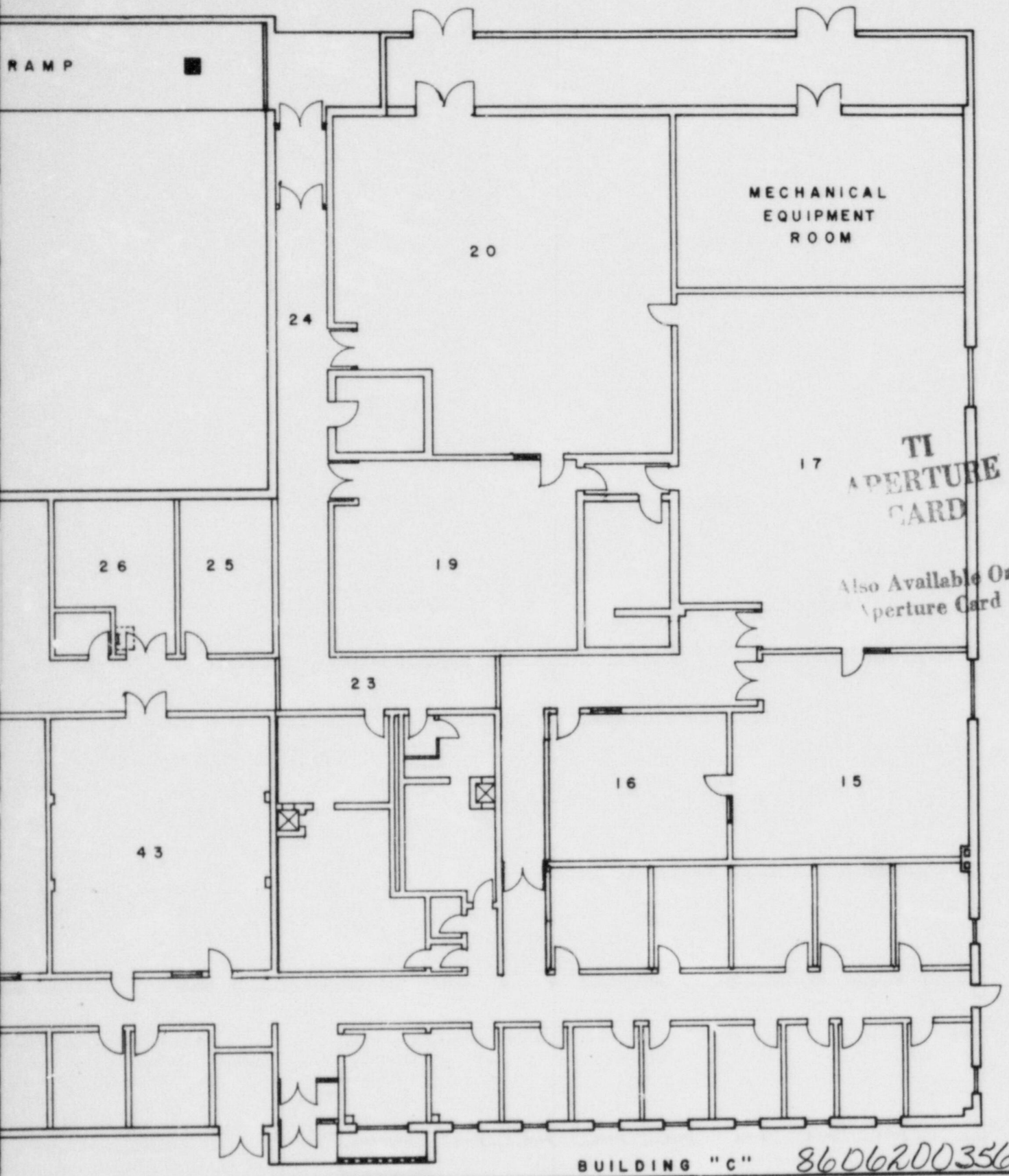


FIGURE 10-3



BUILDING "C"

8606200356-06

License No	SNM-778	Docket No.	70-824	Date	May, 1986
Amendment No.	0	Revision No.	2	Page	10-14

Babcock & Wilcox
a McDermott company

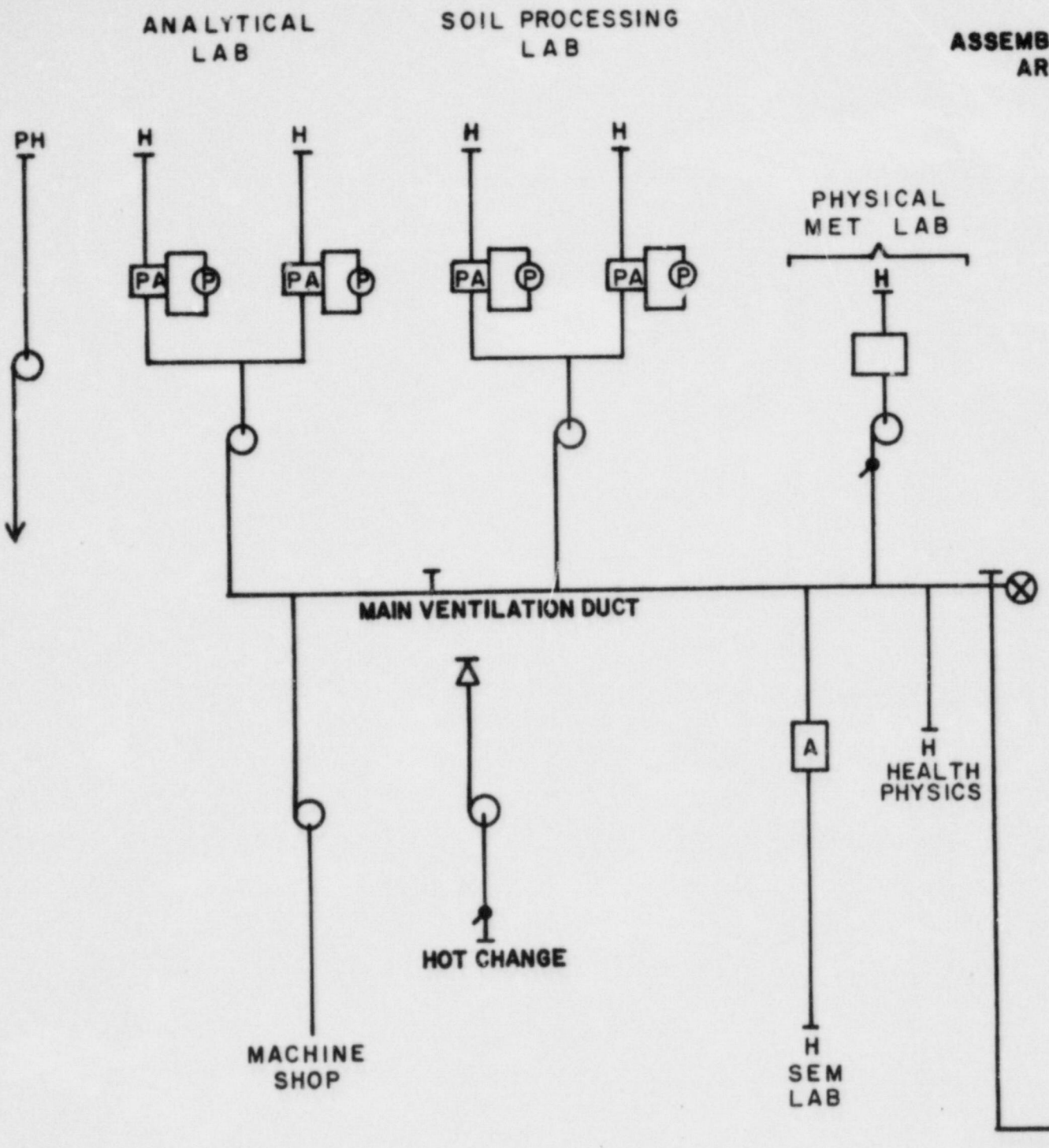
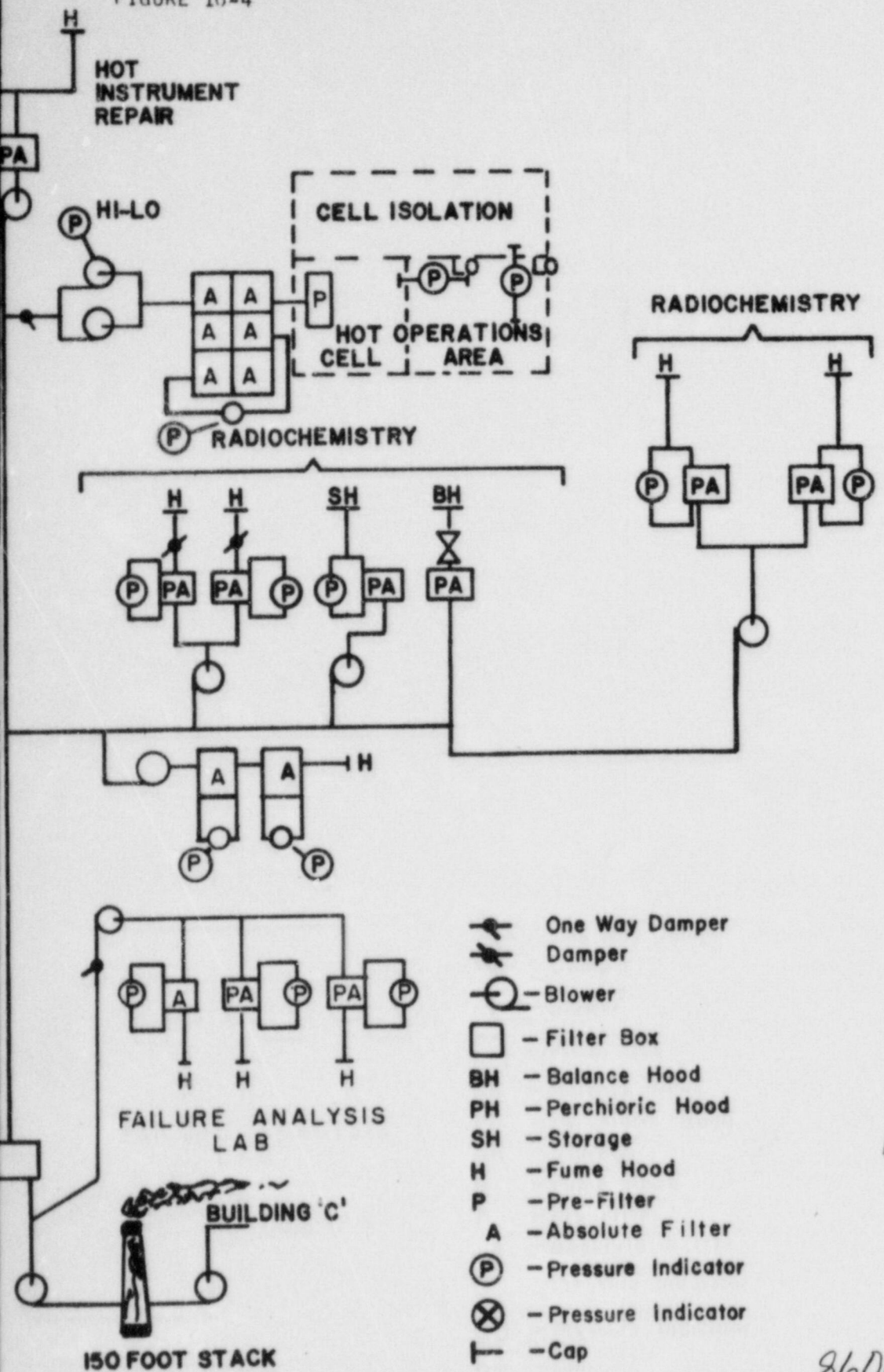

ANALYTICAL LAB
SOIL PROCESSING LAB
ASSEMBLY AREA
PH
H
H
H
H
PA
P
PA
P
PA
P
PA
P
PHYSICAL MET LAB
H
MAIN VENTILATION DUCT
A
H
HEALTH PHYSICS
HOT CHANGE
MACHINE SHOP
H
SEM LAB

FIGURE 10-4



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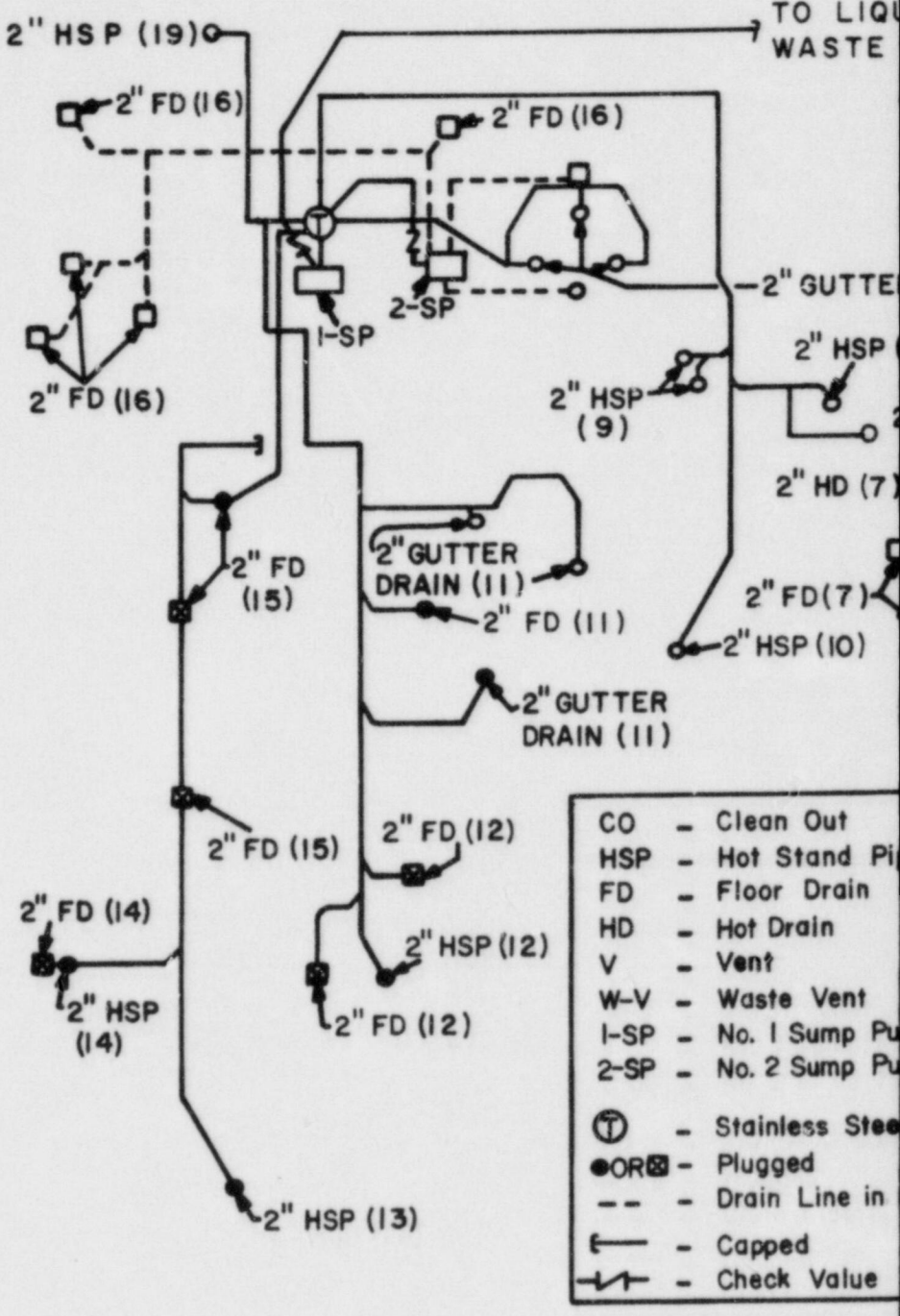
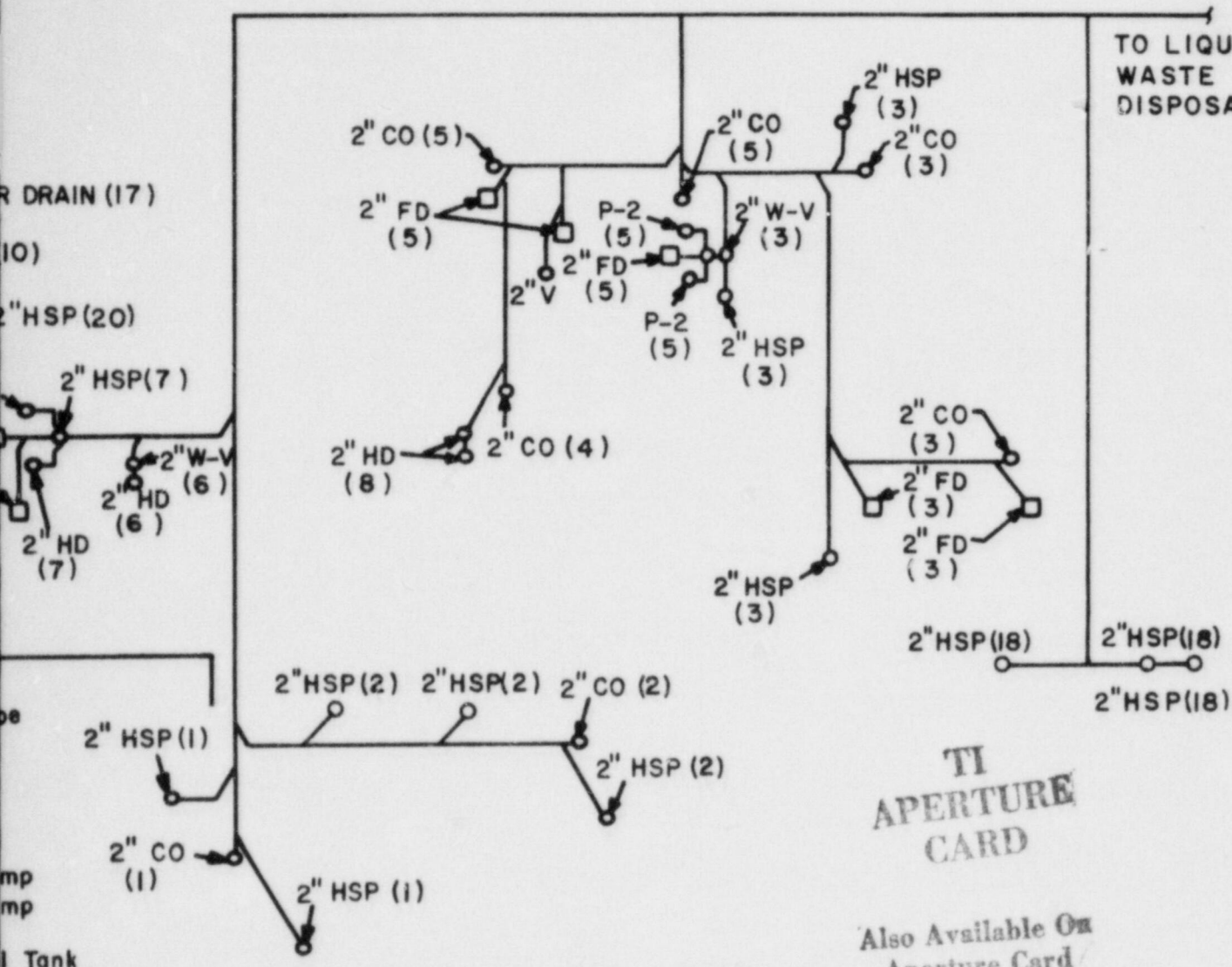
License No	SNM-778	Docket No.	70-824	Date	May, 1986
Amendment No.	0	Revision No.	2	Page	10-15



REVISED:
APRIL, 1985 RED

FIGURE 10-5



ROOM NUMBERS

1 - HEALTH PHYSICS
2 - PROCESS MET. LAB
3 - HIGH TEMPERATURE LAB
4 - CERAMIC FORMING
5 - COMMINUTION TECHNOLOGY
6 - HEALTH PHYSICS
7 - HOT CHANGE ROOM
8 - HEALTH PHYSICS
9 - LOOP SAMPLING ROOM
10 - RADIOCHEMISTRY LAB
11 - STORAGE POOL
12 - ISOLATION AREA
13 - HOT INSTRUMENT REPAIR
14 - EXPERIMENTAL ASSEMBLY AREA
15 - CASK HANDLING AREA
16 - PRIMARY CELL (BASEMENT)
17 - EXPERIMENTAL POOL
18 - SAMPLE PREP. LAB
19 - FAILURE ANALYSIS LAB
20 - CHEMICAL DEVELOPMENT LAB

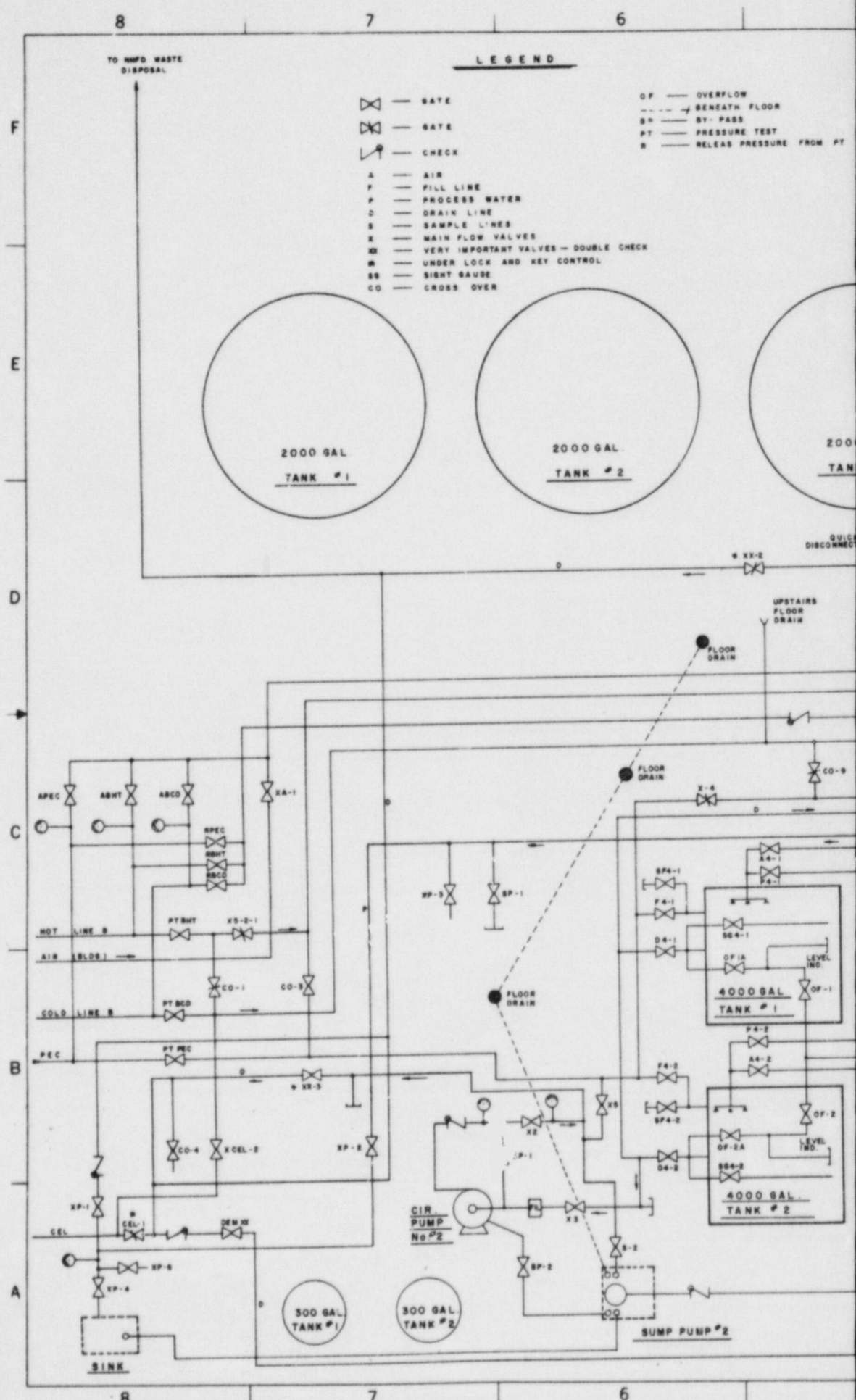
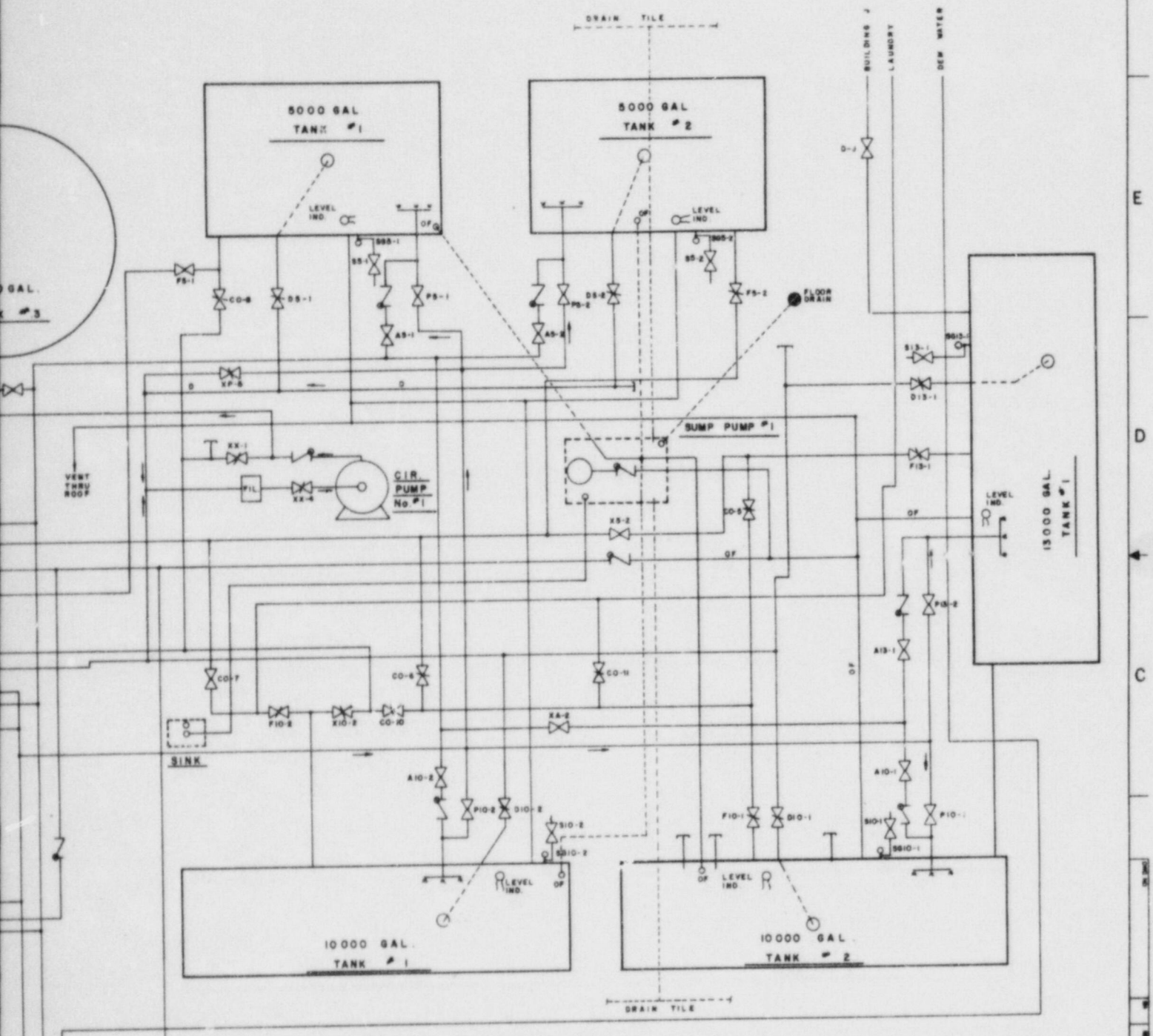

8
7
6
F
E
D
C
B
A
TO NMFD WASTE DISPOSAL
LEGEND
GATE
GATE
CHECK
A — AIR
F — FILL LINE
P — PROCESS WATER
D — DRAIN LINE
S — SAMPLE LINES
X — MAIN FLOW VALVES
XX — VERY IMPORTANT VALVES — DOUBLE CHECK
— UNDER LOCK AND KEY CONTROL
SG — SIGHT GAUGE
CO — CROSS OVER
OF — OVERFLOW
— BENEATH FLOOR
BP — BY- PASS
PT — PRESSURE TEST
R — RELEAS PRESSURE FROM PT
2000 GAL TANK #1
2000 GAL TANK #2
XX-2
UPSTAIRS FLOOR DRAIN
FLOOR DRAIN
FLOOR DRAIN
FLOOR DRAIN
APEC
ABHT
ABCD
XA-1
HOT LINE B
AIR (BLDG)
COLD LINE B
PEC
PT BHT
PT BCD
PT PEC
XS-2-1
CO-1
CO-3
XX-3
CO-4
XCEL-2
XP-5
XP-1
CEL
XP-6
XP-4
SINK
300 GAL TANK #1
300 GAL TANK #2
CIR. PUMP No #2
XP-3
SP-1
SP-2
X2
X3
X5
S-2
P-1
SUMP PUMP #2
CO-9
X-4
SP4-1
F4-1
D4-1
SG4-1
OF-1A
LEVEL IND.
4000 GAL TANK #1
OF-1
P4-2
A4-2
F4-2
SP4-2
D4-2
SG4-2
OF-2A
OF-2
4000 GAL TANK #2

FIGURE 10-6



License No	SNM-778	Docket No.	70-824	Date	May, 1986
Amendment No.	0	Revision No.	2	Page	10-17

liaison between the LRC and the NRC and other federal, state, and local agencies regarding nuclear matters. He is the coordinator of the Safety Review Committee and Chairman of the Safety Audit Subcommittee and represents LRC management on both. The License Administrator is responsible for ensuring that corrective action is taken in response to audit findings as they pertain to licensed activities.

11.2.7 Nuclear Safety Officer - The Nuclear Safety Officer is appointed by and reports to the Manager, Lynchburg Technical Operations. The Nuclear Safety Officer is responsible for ensuring that no operation at the LRC can lead to the inadvertent assembly of a critical mass. To help assure this, he has signature authority for all new Area Operating Procedures and changes to these procedures, he observes operations, institutes educational programs if and when he deems them necessary, and carries out confirming nuclear criticality safety calculations.

The Nuclear Safety Officer will inspect all LRC operations where special nuclear material is being processed, quarterly. Other areas may be inspected less frequently, but all licensed facilities will be inspected at least twice a year. He will consider area operations when scheduling these inspections and will, if necessary, schedule his inspection at more frequent intervals. His consideration should include inspection of new operations, an audit of nuclear safety records, a check for area posting, a review of current practices and a review of corrective actions recommended during previous audits and the status of the recommended actions. He shall submit a report of his finding to the Manager, Lynchburg Technical Operations, with a copy to the License Administrator. Prior to the submission of the report, he will discuss its contents with the Facility Supervisor. The following information is to be included:

1. Areas visited
2. Operations observed
3. Unsafe practices and situations noted
4. Nuclear safety activity of the quarter
5. Recommendations.

11.2.8 Facility Supervisor - The Facility Supervisor is appointed by and reports to the Manager, Lynchburg Technical Operations. He shall be responsible to the Manager, Lynchburg Technical Operations for the safe conduct of all operations at the LRC and for ensuring that these operations are conducted in accordance with all license conditions. The Facility Supervisor shall review and have approval authority for Area Operating Procedures. He shall have authority to terminate any operation that he deems contrary to license conditions, Area Operating Procedures, or general safety conditions. The Facility Supervisor shall become familiar with all license conditions and procedures concerned with radiation safety, nuclear safety, industrial safety, and nuclear materials safeguards. He may consult with the following personnel to ensure compliance with all safety regulations and principles:

Supervisor, Health and Safety

Nuclear Safety Officer

Industrial Safety Officer

Accountability Specialist

11.2.9 Safety Review Committee - The Safety Review Committee (SRC) shall be comprised of at least five technically trained and experienced members appointed by the Manager, Lynchburg Technical Operations. One member shall be selected by the Manager, Lynchburg Technical Operations to be the SRC Chairman. The Chairman shall preside at the meetings and keep the minutes. The Manager, Lynchburg Technical Operations shall appoint an Alternate Chairman who shall act for the Chairman during absences. One member shall be appointed by the Manager, Lynchburg Technical Operations to be the SRC Coordinator. The Coordinator shall represent LRC management on the SRC, set the meeting agenda, and maintains the permanent files of the Committee.

The SRC membership shall have expertise in chemistry, nuclear physics, health physics, and the safe handling of radioactive material. The SRC membership shall have a general understanding of nuclear criticality safety as it pertains to LRC operations. Consultants with special expertise are available to the Committee when needed.

The SRC shall meet at least four times a year. A quorum shall consist of a simple majority of the membership including the Chairman. The SRC shall review and approve all Area Operating Procedures. It shall review and approve new projects that utilize licensed material that are significantly different from previously reviewed and approved projects. The SRC shall review the annual report issued by the Supervisor, Health and Safety which summarizes LRC personnel exposures, environmental releases, and a summary of the ALARA program accomplishments. The SRC Chairman shall forward the Committee minutes to the Manager, Lynchburg Technical Operations, with a copy to the SRC Coordinator.

The Manager, Lynchburg Technical Operations shall appoint the members of the Safety Audit Sub-committee (SAS). The SAS shall be comprised of at least two individuals, one of whom shall be designated as Chairman and he shall report to the Chairman, SRC. The SAS shall audit operations at the LRC at least three times annually, with successive audits separated by at least two months. Additional audits may be performed at any time. The SAS Chairman shall develop the audit report and submit it to the SRC Chairman. The SRC Chairman shall submit the audit report to the Manager, Lynchburg Technical Operations with appropriate comments, with a copy to the License Administrator.

11.3 EDUCATION AND EXPERIENCE OF KEY PERSONNEL

11.3.1 Safety and Licensing Manager - Richard L. Bennett

Education:

B.Ch.E. - Chemical Engineering, University of Delaware, 1958

Experience:

(1985-Present) Babcock & Wilcox, Manager, Safety and Licensing, Lynchburg Research Center, Lynchburg, Virginia.

See Section 11.2.1

(1982-1985) Babcock & Wilcox, Manager, Building C Decommissioning, Lynchburg Research Center, Lynchburg, Virginia

He was responsible for decontaminating facilities that were used for preparation of experimental quantities of nuclear fuels containing plutonium.

(1973-1982) Babcock & Wilcox, Supervisor, Process Technology Group, Lynchburg Research Center, Lynchburg, Virginia

This group was responsible for long-range studies, design assistance, start-up assistance, and preparation of environmental reports and safety analyses related to nuclear fuel conversion. Some of the specific projects performed by the group were preparation of the designs for a low-enriched nuclear fuel conversion plant, preparation of a conceptual design for a spiked nuclear fuel fabrication plant, process engineering assistance to nuclear fuel conversion plants, development of a halide volatility scrap recovery process, development of alternative effluent treatment systems for various nuclear fuel conversion processes, and evaluation of fabrication methods for advanced fuels.

(1971-1973) Babcock & Wilcox, Senior Research Engineer, Lynchburg Research Center, Lynchburg, Virginia

He was responsible for the conceptual design of a facility to treat the effluent from a nuclear fuel plant and developing and evaluating processes for recovering byproducts from B&W wastes.

(1959-1971) American Cyanamid Company, Process Engineer, Piney River, Virginia

He has had broad experience in chemical engineering. This includes research and development, designing equipment and processes, testing and operating new equipment, pilot plant operation, process engineering, and economic evaluation. He has specific knowledge in pigment manufacture, effluent treatment, and byproduct recovery.

Professional Affiliations:

American Institute of Chemical Engineers (Member)
American Nuclear Society (Member)

11.3.2 Health and Safety Supervisor - John W. Cure, III

Education:

B. S. - Electrical Engineering, Virginia Military Institute, 1952
M. S. - Physics, Vanderbilt University, 1956
- AEC-Sponsored Radiological Physics Fellowship, 1952-54 (in conjunction with Vanderbilt University and Oak Ridge National Laboratory)
- Nuclear Safety Training Course, Oak Ridge, 1957
- Certified Health Physicist, American Board of Health Physics, 1961
- In-Place filter Testing, Harvard, 1976

Experience:

(1956-Present) Babcock & Wilcox, Supervisor, Health and Safety, Lynchburg Research Center, Lynchburg, Virginia

Mr. Cure established the Health Physics Program at B&W's Critical Experiment Laboratory, which is now Building A, at the Lynchburg Research Center. This program was expanded as the nuclear activities at the Center grew. In 1972 this program provided health physics coverage for four critical experiment reactors, a one-megawatt pool-type research reactor, a six-megawatt test reactor, hot cells, a radiochemistry laboratory, and uranium, thorium, and plutonium fuel laboratories. The Health and Safety Group, in addition to providing operational surveillance, is responsible for the solid waste disposal program, shipping and receipt of radioactive materials, liquid waste disposal program, and environmental surveillance. This group is responsible for administering the bioassay program, implementing the respiratory protection program and maintaining the personnel exposure records system. At the present time the Health and Safety Group consists of three health physics engineers, three technicians, and the survey monitors associated with two decommissioning projects. Mr. Cure also provides calculational support for the Health and Safety Group.

Mr. Cure is also responsible for Industrial Safety which includes compliance with the regulations of the Occupational Health and Safety Administration, administering the fire prevention and fire protection programs, and enforcing the safety standards of the Company's insurance underwriters.

Mr. Cure serves on the LRC's Safety Review Committee and the Safety Committee at the Alliance Research Center. He has served on the Radiation Safety Committee at the Virginia Polytechnic

Institute and State University. In 1973, he was a member of the Oak Ridge National Laboratories Applied Health Physics Advisory Committee. He has taught formal courses in health physics and radiation safety in B&W educational programs such as that provided for the N.S. Savannah deck officers (first and second crews), CAMEN research reactor operators, and operator training for Arkansas Power and Light Company, Sacramento Municipal Utility District, Florida Power Company, Metropolitan Edison Power Company and Toledo Edison Power Company.

(1954-1956) U. S. Air Force, Kirtland Air Force Base, New Mexico

Mr. Cure was a nuclear research officer. He participated as a health physicist during operation "Tea Pot."

(1953-1954) Junior Health Physicist, Oak Ridge National Laboratory, Junior Health Physicist, Oak Ridge, Tennessee

Mr. Cure was employed as a junior health physicist at the Oak Ridge National Laboratory and received experience in the design and testing of new instruments and also in field work.

Professional Affiliations:

American Nuclear Society (Member)
Health Physics Society (Member)
Subcommittee on Internal Dosimetry Working Committee - 1972-75
Virginia Health Physics Society - President, 1975-76, 1981-82
- Councilman, 1976-77
American Industrial Hygiene Association
Virginia Manufacturers Association, Water and Air Control Comm.
Virginia Safety Association - Board of Directors

11.3.3 Health Physics Engineer - W. Scott Pennington

Education:

B.S. - Biology with an option in Health Physics, Virginia Polytechnic Institute and State University, 1978

Experience:

(1979-Present) Babcock & Wilcox, Senior Health Physicist, Lynchburg Research Center, Lynchburg, Virginia

Mr. Pennington administers the LRC's health physics program. The program includes the measurement and control of the external exposure, internal exposure, environmental sampling, tne respiratory protection program, the solid waste disposal program, the liquid waste disposal program, and providing radiation and contamination surveillance for the Center's decommissioning projects. He provides expertise in radiation safety to project engineers. He implements the bioassay (in vivo and in vitro) program. He has approval authority for radiation work permits, facility work orders, and area operating procedures, in the absence of the Supervisor, Health and Safety.

(1978-1979) Southwest Research Institute, Environmental Science Technician, Environmental Science Department, Houston, Texas

Mr. Pennington was involved in a study for the Bureau of Land Management of off-shore oil platforms in the Gulf of Mexico, and studies for the Houston Lighting and Power Company on thermal pollution and commercial fish survivability.

Professional Affiliation:

Health Physics Society (Member)
Virginia Health Physics Society (Member)
American Nuclear Society (Member)

11.3.4 Industrial Safety Officer - Reginald R. Spradlin

Education:
- Graduate, Appomattox County High School
- Certified Instructor Trainer, Basic Cardiac Life Support, American Heart Association
- Certified Instructor, First Aid & Advanced First Aid, American Red Cross
- Training in the following areas:
  Industrial Safety
  Fire Fighting
  Rescue
  Extrication
  Fire Protection
  Fire Extinguishing Equipment and Materials
  Arson Investigation.

Experience:

(1972-Present) Babcock & Wilcox, Industrial Safety Officer, Lynchburg Research Center, Lynchburg, Virginia

Mr. Spradlin is the LRC's Industrial Safety Officer. As such he is responsible for compliance with the regulations of the Occupational Health and Safety Administration. He advises the LRC on the standards and requirements of the National Fire Protection Association and performs reviews of equipment and systems for compliance with NFPA standards. He performs inspections of facilities and equipment for fire protection purposes. He reviews facility changes and modifications to ensure fire safety. Mr. Spradlin performs tests, maintenance, and inspection of fire protection, control and extinguishing equipment. He is responsible for investigating all accidents, and keeping his management informed of safety activities. He performs fire and rescue training for the members of the LRC's Fire and Rescue Team, and serves as the Captain of the team. He is a certified Shock Trauma Technician, an Emergency Medical Technician, and certified instructor in CPR and Standard and Advanced First Aid.

(1971-1972) Babcock & Wilcox, Accountability Technician, Lynchburg Research Center, Lynchburg, Virginia

Mr. Spradlin served as the Accountability Technician. In this capacity he was responsible for the recordkeeping system for SNM accountability in the Plutonium Development Laboratory. He recorded all transfers of SNM, performed inventories, and updated the unit log records.

(1969-1971) Babcock & Wilcox, Health Physics Technician, Lynchburg Research Center, Lynchburg, Virginia

Mr. Spradlin was a health physics technician in the Plutonium Development Laboratory. He was responsible for performing contamination surveys of the facility, assisting in the monitoring of bagging operations, and supervising decontamination. He implemented the surveillanace program for airborne radioactive material. He performed maintenance, testing, and calibration of alpha particle survey instrumentation and counting equipment. He implemented the respiratory protection program in that laboratory.

(1967-1969) Babcock & Wilcox, Plant Engineering Technician,

Lynchburg Research Center, Lynchburg, Virginia

As a plant engineering technician, Mr. Spradlin performed installation, modification, and repair of facilities, equipment, and experimental apparatus at the LRC. He performed these duties on electrical, mechanical and plumbing systems.

(1952-1967) Mead Corporation, Maintenance Superintendent, Mead Paper Company, Lynchburg, Virginia

Mr. Spradlin served in several capacities during this period, including: finishing operation, paper machine operation, Millwright, Maintenance Foreman, Maintenance Superintendent, Safety Inspector and Accident Investigator.

Professional Affiliations:

Concord Rescue Squad - Founding President
American Heart Association - Cardiac Care Committee

11.3.5 Accountability Specialist - Kenneth D. Long

Education:

Graduate - White Sulphur Springs High School, 1958
Certificate - Bookkeeping, Central Virginia Community College, 1983

Experience:

(1974-Present) Babcock & Wilcox, Accountability Specialist
Lynchburg Research Center, Lynchburg, Virginia

Mr. Long, as the Accountability Specialist, is responsible to the Manager of Safety and Licensing for the accurate accounting of all Special Nuclear, Source, and Byproduct material at the LRC. He is responsible for recording all transfers of SNM that are made within the LRC and for preparing the reports and records of off site transfers. He prepares all NRC/DOE 741 Transaction Forms. He is responsible for the timely completion of inventories of licensed material. He initiates the paper work required for all shipments of licensed material.

In addition to his normal duties he is a Document Custodian. In this capacity, he is responsible for the safe storage of all

classified DOE and DOD documents at the LRC. He is also an authorized classifier and an authorized courier of classified material.

(1970-1974) Babcock & Wilcox, Shipping & Receiving Clerk
Lynchburg Research Center, Lynchburg, Virginia

Mr. Long was responsible for the shipment and receipt of all materials at the LRC. This assignment included the processing of all the necessary forms and documents used for shipping and receiving licensed materials as well as the many items that are required for operation of a research and development laboratory.

(1967-1970) Babcock & Wilcox, Technician
Lynchburg Research Center, Lynchburg, Virginia

Mr. Long was a technician in the Plutonium Development Laboratory during this period. He performed chemical operations utilizing uranium and plutonium materials and was responsible for the accountability of SNM materials into and out of his area.

Professional Affiliations:

Institute of Nuclear Materials Management (Senior Member)
Nuclear Materials Control Committee, B&W (Secretary)
American Nuclear Society, Virginia Chapter (Member)

11.3.6 License Administrator - Arne F. Olsen
Facility Supervisor - Arne F. Olsen

Education:

AAS - Nuclear Technology, Central Virginia Community College, 1978

Experience:

(1972-Present) Babcock & Wilcox, Senior License Administrator and Facility Supervisor, Lynchburg Research Center, Lynchburg, Virginia

Mr. Olsen is responsible for preparing, amending, and administering the licenses that the LRC possesses with the NRC and the Commonwealth of Virginia. He acts as the primary liaison between the LRC and the NRC and other federal, state, and local agencies regarding

nuclear matters. He coordinates the visits made by the NRC's Office of Inspection and Enforcement, and coordinates the LRC's compliance with NRC and state regulations and the licenses. He is the coordinator of the Safety Review Committee and is Chairman of the Safety Audit Subcommittee, and represents LRC management on both. Mr. Olsen is the Facility Supervisor and as such is responsible to the Manager, Lynchburg Technical Operations for the safety of all operations at the LRC.

Mr. Olsen is the Alternate LRC Security Officer, Alternate Emergency Officer and an internal auditor.

(1968-1972) Babcock & Wilcox, Health Physics Technologist, Lynchburg Research Center, Lynchburg, Virginia

In this capacity, Mr. Olsen was responsible to the site Health Physicist (Supervisor, Health and Safety) for the implementation of the Health Physics Program in the Plutonium Development Laboratory. This responsibility included the implementation of the smearing, survey, air sampling, environmental sampling, and waste disposal programs.

(1964-1968) Babcock & Wilcox, Technician and Shift Leader, Babcock & Wilcox Test Reactor, Lynchburg Research Center, Lynchburg, Virginia

Mr. Olsen possessed a Senior Reactor Operator's License for the BAWTR. He was in charge on one of four shifts of reactor operators charged with the proper operation and maintenance of the BAWTR. He supervised the loading and unloading of fuel and experiments in the reactor and kept all required records of operations and maintenance performed on his shift.

(1960-1964) U. S. Navy, Reactor Plant Electrical Supervisor, USS Enterprise CVA(N)-65

Mr. Olsen was an Electrician, First Class and was responsible for the proper operation and maintenance of all electrical equipment serving one of the reactor plants aboard the Enterprise.

Professional Affiliation:

Health Physics Society (Member)
Site Environmental Committee, B&W (Member)

11.3.7 Nuclear Safety Officer - Francis M. Alcorn

Education:

B.S. - Nuclear Engineering, North Carolina State College, 1957
M.B.A - Business Administration, Lynchburg College, 1974
- Graduate study in Nuclear Engineering, University of Virginia

Experience:

(1971-Present) Babcock & Wilcox, Supervisor, Nuclear Criticality Safety Group, Lynchburg Research Center, Lynchburg, Virginia

This group is the Company's central organization which provides guidance, develops and validates the analytical methods needed for criticality evaluations, does criticality calculations, performs nuclear safety audits, and gives assistance to the various divisions of the Company and the Company's customers in matters related to nuclear criticality safety. In addition to his responsibility as supervisor of this group, he is the Nuclear Safety Officer for the Lynchburg Research Center.

(1969-1971) Babcock & Wilcox, Criticality Specialist, Nuclear Safety Engineer, Lynchburg Research Center, Lynchburg, Virginia

Transferred to the LRC as Nuclear Criticality Safety Specialist for Babcock & Wilcox's Naval Nuclear Fuel Plant, Commercial Nuclear Fuel Plant, and the LRC. He was appointed Nuclear Safety Officer for the LRC.

(1964-1969) Babcock & Wilcox, Power Generation Division, Lynchburg, Virginia

Mr. Alcorn was a physicist in the PWR Development Section and was responsible for determining the most economical method for utilizing plutonium as a recycle fuel in B&W's pressurized water reactor concepts. In addition, he was Nuclear Criticality Safety Advisor to the Company's Naval Nuclear Fuel Division.

(1961-1964) Babcock & Wilcox, Nuclear Power Generation Division Lynchburg, Virginia

He has been concerned with core neutron physics analysis and design of the Consolidated Edison Reactor, the Liquid Metal Fuel Reactor, the Babcock & Wilcox Test Reactor, the Advanced Test Reactor, the Heavy Water-Organic Cooled Reactor Concept, and Babcock & Wilcox Pressurized Water Reactor Concepts. He developed methods for and performed calculations for criticality, fuel depletion, nuclear safety coefficients, power profiles, nuclear fuel costs and critical experiment analysis. He has also worked in the areas of kinetic safety analysis.

(1957-1960) Babcock & Wilcox, Atomic Energy Division
Lynchburg, Virginia

He functioned as a nuclear engineer doing both core neutron physics and shielding calculations.

(1960-1961) General Nuclear Engineering Corporation, Staff Physicist

Mr. Alcorn engaged in core neutron physics design and analysis of the Boiling Nuclear Superheat Reactor. He also wrote physics articles for Power Reactor Technology which were published by GNEC for the AEC.

Professional Affiliations:

Sigma Pi Sigma (Member)
Tau Beta Pi (Member)
American Nuclear Society - Past Chairman of ANS Nuclear Criticality Safety Division
- Member Standards Subcommittee ANS-8.

11.4 OPERATING PROCEDURES

11.4.1 Area Operating Procedures (AOP) - All operations with licensed material shall be conducted in accordance with Area Operating Procedures or a Radiation Work Permit. Area Operating Procedures are prepared by any technically competent person. The proposed procedure is delivered to the Facility Supervisor who ensures that the procedure is in the proper format. The Facility Supervisor routes the procedure to the Nuclear Safety Officer who reviews it to assure that any nuclear criticality safety issues are properly

addressed. If the Nuclear Safety Officer (NSO) has additions or corrections, he notes them on the procedure and forwards it to the Supervisor, Health and Safety (S.H&S). If the NSO approves it, he signs the procedure in the space provided and forwards it to the S.H&S. The S.H&S reviews it for proper radiological and industrial safety content. If he has additions or corrections, he notes them on the procedure and forwards it to the Facility Supervisor. If the S.H&S approves the procedure, he signs the procedure in the space provided and forwards it to the Facility Supervisor. The Facility Supervisor reviews it for general safety and determines its impact on other work and facilities. The Facility Supervisor is responsible for resolving all additions or changes recommended by the previous reviewers. When the procedure is approved by the three reviewers, the Facility Supervisor forwards it to the Safety Review Committee. The Safety Review Committee (SRC) may approve the procedure as written, approve the procedure conditionally with specific changes to be made prior to issuance or the SRC can disapprove it. The SRC coordinator signs for the SRC when approval is voted. The procedure may be implemented subsequent to SRC approval.

Revisions to AOP's will follow this same approval route, except that the revised procedure may be implemented after receiving the approval signatures of the NSO, S.H&S and the Facility Supervisor. The revised procedure will be placed on the agenda for the next regularly scheduled meeting of the SRC.

11.4.2 Availability - AOP's are entered in 3-ring binder manuals. Manuals are issued to individual workers and placed in areas where the procedures apply.

11.5 TRAINING

11.5.1 General Radiation Protection Training

The LRC provides two training programs covering the nature, use and control of radiation, and radioactivity. These courses are presented to ensure that all LRC personnel receive training appropriate to their activities and to fulfill obligations under the NRC license to provide such training.

The courses consist of a series of lectures intended to present the proper background and technical base to allow workers to understand

the principles of radiation safety. The Health and Safety Group administers the course and, in general, teaches each course. Where practical, basic general procedures and federal regulations are included and discussed. Training aids, such as motion pictures and self-study materials, are used as appropriate.

Program I is intended for new employees who will be scheduled for such training within 30 days of reporting to work at the LRC. Program II is intended for personnel working with radioactive materials. Personnel selected by their section manager to be an Authorized User (Section 1.6) of radioactive materials (i.e., employees who may handle licensed material unsupervised, health physics technicians, etc.) will be scheduled for these courses. Personnel will not be permitted to work unsupervised with licensed material until they are trained in radiation protection and criticality safety and designated an Authorized User. Retraining of Authorized Users of radioactive materials is performed annually.

Workers who are exposed to ionizing radiation are classified as radiation workers and will receive training commensurate with their exposure as required by Title 10, Code of Federal Regulations, Part 19 (10 CFR 19). This training will include Program II as necessary.

Training in area operating procedures and special area procedures is the responsibility of the line supervisor. This training should be accompanied with appropriate formal and on-the-job training as the job requirements dictate.

11.5.2 Program I

This course is available to new office employees and is presented to employees within 30 days of reporting to work at the LRC. It provides an introduction to radiation and radioactivity (understandable to the employee with no technical education or experience) and a thorough coverage of safety rules and procedures, including the site emergency procedures. Subjects include types of radiation, radiation effects on humans, permissible levels, basic health physics rules, a history of radiation protection, and personal hygiene.

11.5.3 Program II

New laboratory employees who work with radioactive materials are

required to complete this course and pass a written test. Subjects include the following:

1. Radioactivity

    a. Types of radiation

    b. Radioactive decay

    c. Radiation dose and dose rates

    d. Exposure control methods - time, distance, shielding

    e. External and internal exposure hazards

    f. Respiratory protection

    g. The importance of maintaining exposures as low as is reasonably achievable (ALARA)

    h. Risks from radiation exposure including exposure of females and the embryo/fetus

    i. Radiation exposure compared to other hazards in the work place.

2. Health Physics Instruments

    a. Personnel monitoring devices

    b. Cutie pie and Geiger-Mueller counter

    c. Alpha survey meter

    d. Air monitors

    e. Criticality alarm system

    f. Emergency equipment

    g. Instructions in field use of instruments.

3. Regulations and Procedures

a. Code of Federal Regulations (including 10 CFR 19)

b. License requirements

c. Shipment of radioactive materials

d. Waste disposal

e. Internal procedures.

Parts of Program II may be waived as appropriate for technical and scientific personnel already knowledgeable and experienced in working in radiation areas and with licensed material. However, such personnel must pass the written examination required for Program II.

11.5.4 Respiratory Protection Training

Training in respiratory protection techniques will be required of all employees before the use of such equipment will be allowed. This training will be carried out by a qualified individual, as defined in NUREG-0041 (Section 12.1), who will document that such training as been completed. Those persons who direct the work of employees using respiratory protection will be included in the training courses. Periodic retraining will be scheduled, at the discretion of the qualified individual, to ensure that a high degree of employee proficiency in the use of respiratory protective devices is maintained.

Training in respiratory protection shall include the following subjects:

a. Discussion of the airborne contaminants present in the work environment including their physical properties, physiological actions, toxicity, means of detection, and maximum permissible concentrations (MPC's).

b. Discussion of the importance of selecting the proper respirator based on the hazard and the dangers of using respirators for a purpose other than that intended.

c. Discussion of the construction, operating principles, and limitations of the available respirators.

d. Discussion of the use of engineering controls as a substitute for respiratory protection and the need to make every reasonable effort to reduce or eliminate the need for respiratory protection.

e. Instruction in methods to be used to determine that the respirator is in proper working order.

f. Instruction in fitting the respirator properly, field testing for proper fit, and factors that may influence a proper fit.

g. Instructions in the proper use and maintenance of the respirator.

h. Discussion of the uses of various cartridges and canisters available for air-purifying respirators.

i. Review of radiation and contamination hazards, including a review of other protective equipment that may be used with respirators.

j. Instruction in emergency actions to be taken in the event of respirator malfunction.

k. Classroom instruction to recognize and cope with emergency situations while working with a respirator.

l. Any additional training as needed for special use.

m. The wearer must pass a written examination on the material presented on respiratory protection.

11.6 FACILITY CHANGE

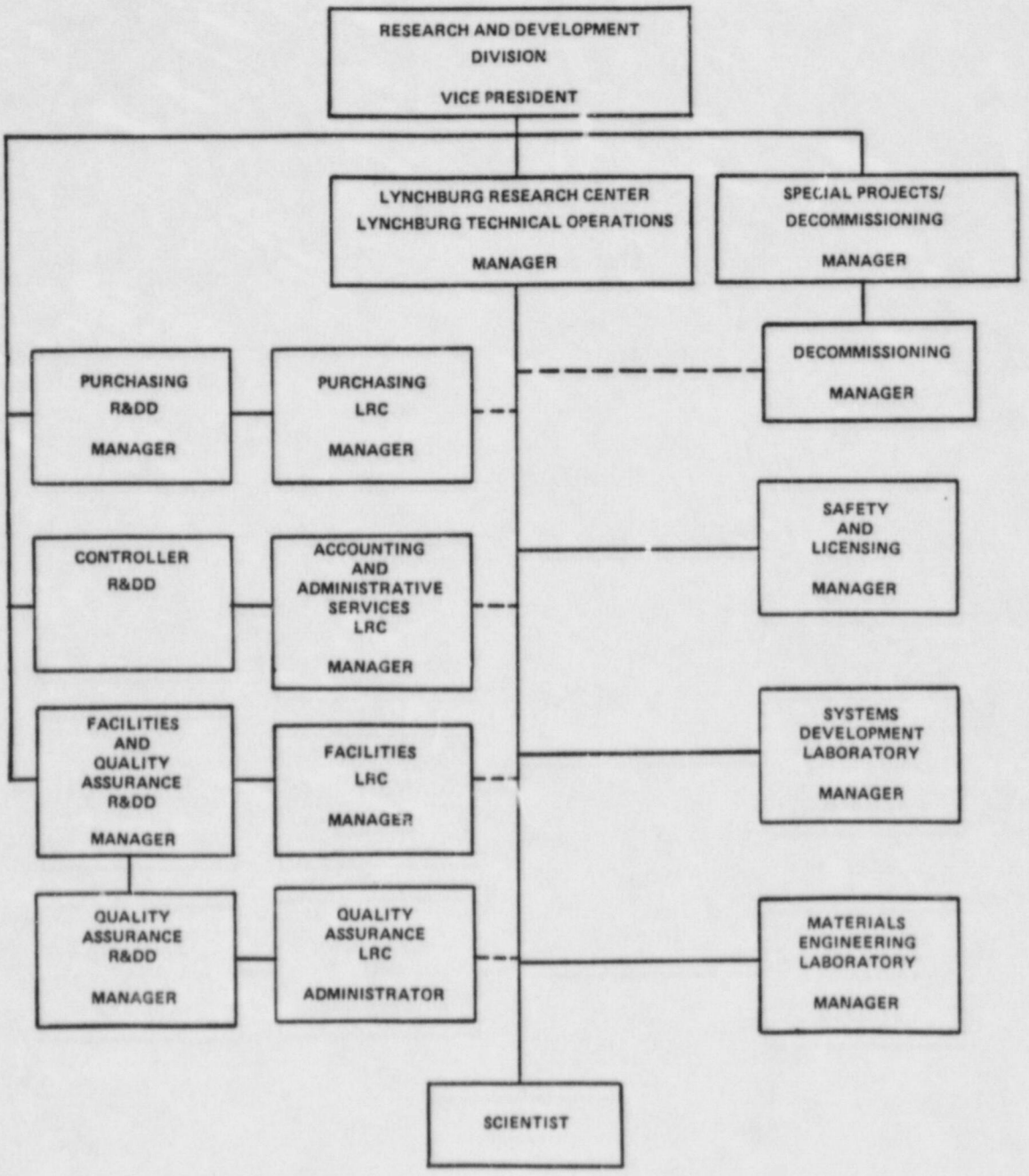
Changes and modifications to buildings, exhaust ventilation systems, gas supply systems, emergency electrical systems, etc. are requested on Form LRC-229, "Facilities Work Order Form" (Figure 9-4). All work orders are forwarded to the maintenance supervisor. The Plant Engineering Supervisor determines if the request involves a facility change. If a facility change is involved, the work order is forwarded to the Facility Supervisor. It is the Facility Supervisor's responsibility to determine that all safety and licensing considerations have been addressed and if the request must be

approved by the Safety Review Committee. Space is provided on the form for the approval signatures of the Supervisor, Health and Safety, the Industrial Safety Officer, and the Facility Supervisor.

Completed forms are kept on file by the maintenance supervisor and are audited once a month by the Health Physics Group.

FIGURE 11-1

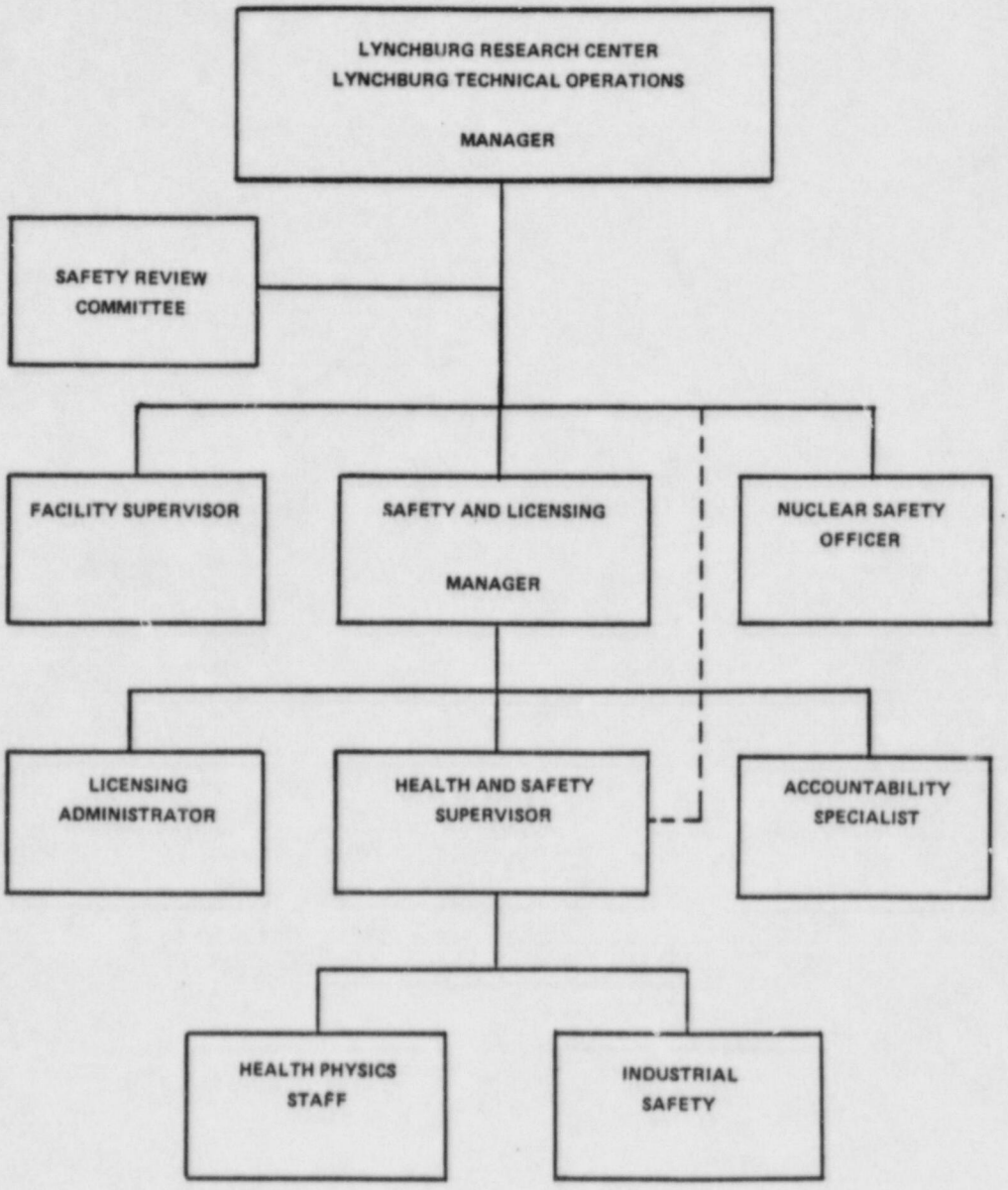
LRC LINE ORGANIZATION



– – –INDICATES FUNCTIONAL REPORTING

FIGURE 11-2

LRC SAFETY ORGANIZATION



– – – INDICATES FUNCTIONAL REPORTING

FIGURE 11-3

LRC 229

FACILITIES WORK ORDER FORM

TO: Plant Engineering Date _____

From: _____ Section: _____ Signed: _____

Section Mgr.: _____ Date: _____

Date Required: _____ Charge No.: _____ (Labor) _____ (Material)

DESCRIPTION OF WORK TO BE DONE

_____

_____

_____

_____

_____

SIGNATURE REQUIRED: Industrial Safety Officer: _____

Health Physics: _____ Facility Supervisor: _____

Space Below This Line For Plant Engineering Use Only

Order Received Date: _____ Signed: _____

Planned Starting Date: _____ Planned Completion Date: _____

Order Completed: | Work Order Number

Date: _____ Signature: _____

TABLE OF CONTENTS

Section		Page
12.0	RADIATION PROTECTION	12-1
12.1	PROGRAM	12-1
12.2	POSTING AND LABELING	12-1
12.2.1	Radioactive Materials Area	12-1
12.2.2	Contamination Area	12-2
12.2.3	Radiation Area	12-2
12.2.4	High Radiation Area	12-2
12.2.5	Airborne Radioactivity Area	12-2
12.3	EXTERNAL RADIATION - PERSONNEL MONITORING	12-3
12.3.1	Administrative Exposure Control	12-3
12.3.2	Personnel Monitoring for LRC Employees	12-3
12.3.3	Personnel Monitoring and Escort Requirements for Non-LRC Employees	12-5
12.3.4	Monitoring Devices	12-7
12.4	DIRECT RADIATION SURVEYS	12-8
12.5	REPORTS AND RECORDS	12-9
12.6	INSTRUMENTS	12-9
12.6.1	Types	12-10
12.6.2	Calibration	12-11
12.7	PROTECTIVE CLOTHING	12-12

TABLE OF CONTENTS (Cont'd)

Section		Page
12.7.1	Clothing	12-12
12.7.2	Emergency Clothing	12-12
12.8	ADMINISTRATIVE CONTROL LEVELS	12-12
12.8.1	Internal Occupational Exposure	12-12
12.8.2	External Occupational Exposure	12-19
12.8.3	Airborne Activity	12-19
12.8.4	Liquid Activity	12-20
12.8.5	Surface Contamination	12-21
12.8.6	Compliance with 40 CFR 190	12-24
12.9	RESPIRATORY PROTECTION	12-25
12.10	OCCUPATIONAL EXPOSURE ANALYSIS	12-26
12.10.1	External Exposure Analysis	12-26
12.10.2	Internal Exposure	12-31
12.11	MEASURES TAKEN TO IMPLEMENT ALARA	12-39
12.12	BIOASSAY PROGRAM	12-40
12.13	AIR SAMPLING AND MONITORING	12-41
12.13.1	Air Sampling Program	12-41
12.13.2	Air Monitoring Program	12-42

TABLE OF CONTENTS (Cont'd)

Section		Page
12.14	SURFACE CONTAMINATION	12-43
12.14.1	Smear Surveying	12-43
12.14.2	Direct Radiation Surveys	12-46
12.14.3	Personnel Contamination Surveys	12-48

List of Tables

Table		Page
12-1	PORTABLE RADIATION PROTECTION INSTRUMENTATION	12-10
12-2	STATIONARY RADIATION PROTECTION INSTRUMENTATION	12-11
12-3	PLUTONIUM BIOASSAY ACTION CRITERIA	12-13
12-4	PLUTONIUM BIOASSAY ACTION CRITERIA	12-14
12-5	URANIUM BIOASSAY ACTION CRITERIA	12-15
12-6	URANIUM BIOASSAY ACTION CRITERIA	12-17
12-7	BETA-GAMMA ACTION CRITERIA	12-18
12-8	STACK RELEASE ACTION LEVELS	12-20
12-9	SMEAR SURVEYS IN WORK AREAS	12-21
12-10	ACTION LEVELS FOR LARGE AREA SMEARS	12-22
12-11	MAXIMUM PERMISSIBLE CONTAMINATION FOR SKIN SURFACES	12-23

List of Tables (Cont'd)

Table		Page
12-12	MAXIMUM PERMISSIBLE CONTAMINATION OF CLOTHING	12-24
12-13	1984 LRC EXPOSURES BY RANGE	12-27
12-14	1985 LRC EXPOSURES BY RANGE	12-28
12-15	LRC RADIATION EXPOSURE	12-29
12-16	EXPOSURE BY GROUP (PERSON REMS)	12-30
12-17	NUMBER OF URINE BIOASSAY SAMPLES	12-31
12-18	1983 AIR ACTIVITY	12-32
12-19	1984 AIR ACTIVITY	12-34
12-20	WHOLE BODY COUNTS 1983	12-36
12-21	Am - Pu LUNG COUNTING 1983	12-37
12-22	URANIUM LUNG COUNTING 1983	12-38
12-23	WHOLE BODY COUNTS 1984	12-38
12-24	ACTION LEVELS FOR LARGE AREA SMEARS	12-44
12-25	SMEAR SURVEY FREQUENCIES AND ACTION LEVELS	12-45
12-26	CONTAMINATION ACTION LEVELS	12-47

and the words - CAUTION - AIRBORNE RADIOACTIVITY AREA. Entry is limited to those qualified persons classified as radiation workers, working under an approved radiation work permit. No entry is permitted until an appropriate area survey has been made and a member of the Health and Safety Group is present. Protective clothing, protective equipment, and personnel monitoring devices to be worn in the area will be specified by the Health and Safety Group and must be worn. When exiting these areas, each person must remove the protective clothing and monitor himself in accordance with established procedures.

12.3 EXTERNAL RADIATION - PERSONNEL MONITORING

12.3.1 Administrative Exposure Control - Limits for external radiation exposure are set forth in 10 CFR 20.101 and these general limits are used at the LRC. The applicable exposure limits to be used for operations at the LRC are:

1. Whole body - 300 mRem/week (with long-term exposure controlled within the 1.25 Rem/quarter limit by the worker's immediate supervisor)

2. Skin of the whole body - 1.5 Rem/week

3. Hands and forearms, feet and ankles - 3.0 Rem/week.

The Manager, Lynchburg Technical Operations, has the authority to approve whole body exposures up to, but not exceeding, 3.0 Rem/calendar quarter. In emergencies, the Emergency Officer is authorized to allow personnel exposures to the whole body of up to 3.0 Rem/calendar quarter. Higher exposures may be authorized by the Emergency Officer in accordance with the Radiological Contingency Plan.

12.3.2 Personnel Monitoring for LRC Employees - All LRC employees will be monitored for radiation exposure while on site. This monitoring will be accomplished in two ways:

1. All employees will be issued a thermoluminescent dosimeter (called an Annual TLD).

2. Employees will be classified as radiation or non-radiation workers. In addition to the Annual TLD, those employees

2. Resample the individual within 24 hours.

3. Investigate the exposure and take corrective action as needed.

4. Evaluate for possible referral to a competent physician.

5. Remove work restriction only with the approval of the Supervisor, Health and Safety.

TABLE 12-4

PLUTONIUM BIOASSAY ACTION CRITERIA

(CLASS W AND Y COMPOUNDS ONLY)

Bioassay Technique	Action Level	Action To Be Taken
In-vivo	< 4E-9 Ci	None
	≥ 4E-9 Ci (16 nanocuries)	1. Restrict worker from further exposure. 2. Resample the individual within 10 working days. 3. Determine if area surveys support the analysis results. 4. If area surveys confirm result, investigate the cause and take corrective actions.

5. If the resample results do not confirm the exposure, the Supervisor, Health and Safety may lift the work restrictions.

6. If resample results confirm the exposure, the Supervisor, Health and Safety shall determine the organ dose.

7. If the exposure has exceeded 50% of the maximum permissible annual dose, the worker shall remain on a work restriction until the Supervisor, Health and Safety authorizes the removal of the restriction.

12.8.1.2 Uranium bioassay action criteria.

TABLE 12-5

URANIUM BIOASSAY ACTION CRITERIA

(CLASS D COMPOUNDS)

Bioassay Technique	Action Level	Action To Be Taken
Urinalysis	<21 picocuries/L	No action
	≥21 picocuries/L	1. Resample within 5 working days. 2. Perform an area survey.

TABLE 12-6

URANIUM BIOASSAY ACTION CRITERIA

(CLASS W AND Y COMPOUNDS)

Bioassay Technique	Action Level	Action To Be Taken
Urinalysis	<25 picocuries/L	No action
	≥25 picocuries/L	1. Resample within 5 working days. 2. Survey the work area. 3. If #2 confirms an exposure, investigate and correct the cause. 4. If the resample results (#1) confirm a deposition >50% of a body burden, the worker will be restricted from further exposure and will be in-vivo counted as soon as practicable. 5. If the resample results (#1) indicate >10% but <50% of a body burden, the worker will be in-vivo counted during the next scheduled period when the service is on site. 6. If the in-vivo counting results (#4) indicated >50% of a body burden, the worker will be restricted from further exposure until the Supervisor, Health and Safety removes the restriction.

12.8.1.4 Beta-gamma activity - Workers who work in areas where beta-gamma internal exposure is likely (Hot Cells, Radiochemistry, Health Physics) shall be in-vivo counted at approximately annual intervals.

TABLE 12-7

FISSION PRODUCT ACTION CRITERIA

Analysis	Action Level	Action to be Taken
In-vivo	>10% MPOB	Remeasure subject to determine effective half life of the contaminant and plot decay curves. Followup program will continue until the contamination present is <5% MPOB or the effective half life has been determined.
Estimation from nasal smears or air sample	>10% MPOB	Submit in vitro sample for analysis within 5 working days.
In-vitro	>5% MPOB	Resample excreta to confirm presence of contamination and to establish rate of elimination. Perform isotopic analysis if >10% MPOB is a possibility.
In-vitro	>10% MPOB	In vivo measurement to be made as soon as practicable.

The Supervisor, Health and Safety shall be responsible for evaluations to determine the location and amount of deposition; to provide data necessary for estimating internal dose rates, retention functions, and dose commitments; and to determine whether work restrictions or referrals for therapeutic treatment are required for any case where a result indicating a greater than 10% MPOB deposition of a radionuclide is verified.

exceeds 25% of the MPC values listed in Table I, Col. 2, of 10 CFR 20, Appendix B, the waste must be diluted to levels that meet this specification. Liquid waste is discharged to the liquid waste processing system at the NNFD. The NNFD must be notified and approve of each discharge from the LRC prior to discharge. No alarms are associated with this system because its operation is under the positive control of the Health and Safety Group.

12.8.5 Surface Contamination

12.8.5.1 Work Areas - The Health and Safety Group performs smear surveys in the work areas listed in Table 12-9. The frequencies specified in Table 12-9 are minimum frequencies. More frequent surveys are performed based on the level of work performed in the specified areas. Action is taken to protect personnel and reduce the levels of contamination below those specified. The Health and Safety Group will supervise and direct the protection and decontamination activities. Decontamination to reduce levels of contamination will commence within 24 hours of discovery. The Supervisor, Health and Safety shall evaluate and approve any delays on decontamination work that are longer than 24 hours.

TABLE 12-9

SMEAR SURVEYS IN WORK AREAS

Area	Frequency *	Action Level (dpm/100 cm^2)
<----------ALPHA---------->		
Unirradiated, unencapsulated fuel handling areas	Weekly	5000
Building B Counting Lab.	Monthly	200
Building A Labs.	Monthly	200
Hot Cell Oper. Area	Monthly	200
Scanning Electron Microscopy Lab.	Monthly	200

Exit portals from controlled areas	Biweekly	200
<----------BETA + GAMMA---------->		
Building A Labs.	Monthly	2000
Building B Counting Lab.	Monthly	2000
Scanning Electron Microscopy Lab.	Monthly	2000
Hot Cell Operations Area	Bimonthly	2000
Cask Handling Area	Bimonthly	22000
Radiochemistry Lab.	Bimonthly	22000
Exit portals from controlled areas	Biweekly	2000

*Minimum frequency specified. More frequent surveys are performed, based on work loads.

Large area smears are usd to survey many square meters of surface area. Action levels for large area smears are given below.

TABLE 12-10

ACTION LEVELS FOR LARGE AREA SMEARS

1. Routine Large Area Smears (1000-5000 dpm) - Repeat the large area smear. If results show levels of contamination above 1000 dpm, take smears in smaller areas to locate the source. Decontaminate all areas in which the smear results indicate contamination above 1000 dpm/100 square feet.

2. Routine Large Area Smears (5000-10,000 dpm) - Repeat the large area smear. If results show levels of contamination above 5000 dpm, isolate the contaminated area. Take smears in smaller areas to locate the source. Decontaminate all

areas in which the smear results show contamination in excess of 1000 dpm/100 square feet.

3. Routine Large Area Smears (>10,000 dpm) - Isolate the contaminated area. Survey all personnel in the contaminated area. Take smaller smears in the area to locate the source. Decontaminate all areas in which the smear results show contamination in excess of 1000 dpm/100 square feet. Survey all persons leaving the building.

12.8.5.2 Personnel Contamination Surveys - Personnel are required to monitor themselves for activity present on their hands, shoes, clothing and person before exiting a contamination area. Contamination monitors (friskers) are located at all normal exits from contamination areas for this purpose. The detector should be held as close to the surface of the item being monitored as possible, without touching the item, and the probe should be moved at a slow speed over the surface. Allowable levels of contamination on skin surfaces and on items of clothing are given in Tables 12-11 & 12-12. Any contamination in excess of these limits should be reported immediately to the Health and Safety Group. The Health and Safety Group will supervise the decontamination and determine if clothing must be discarded. The approval of the Health and Safety Group shall be required to allow any individual to leave a contaminated area who is contaminated above background radiation levels.

TABLE 12-11

MAXIMUM PERMISSIBLE CONTAMINATION FOR SKIN SURFACES

Surface	Fixed Alpha dpm/100 sq. cm.	Fixed Beta-Gamma dpm/100 sq. cm.*	Smearable (Alpha, Beta-gamma)
Body	220	2200	None Detectable
Hands	220	2200	None Detectable

TABLE 12-12

MAXIMUM PERMISSIBLE CONTAMINATION OF CLOTHING

(dpm/100 sq. cm)

Item	Fixed Alpha	Fixed Beta-Gamma*	Smearable Alpha	Smearable Beta-Gamma
Shoes:				
Contaminated Zone				
Inside	2,200	22,000	220	2,200
Outside	22,000	220,000	2,200	22,000
Personal				
Inside	2,200	2,200	220	2,200
Outside	22,000	22,000	220	2,200
Clothing:				
Contaminated Zone	2,200	2,200	Not Detectable	
Personal	2,200	2,200	Not Detectable	

12.8.5.3 Release of Equipment or Packages - Packages and equipment are surveyed by the Health and Safety Group. The Health and Safety Group has the authority to prohibit the release of items that are found to exceed the limits specified in Annex C to License SNM-778 "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use of Termination of Licenses for Byproduct, Source, or Special Nuclear Material, dated November, 1976."

12.8.6 Compliance with 40 CFR 190 - Compliance with 40 CFR 190 has been demonstrated by calculation. Two release pathways were considered; release through the 50-meter stack and release of liquids. The stack release assumed all alpha activity was plutonium. The lung dose to a member of the general public at the maximum point was calculated to be 8.6E-6 millirems/yr. The exposure calculation for release of liquid to the James River was based on the assumption that exposure was achieved by ingestion of fish caught and eaten at

a rate of 10 kilograms each year. The exposure was calculated to be 5.3E-7 millirem/year.

12.9 RESPIRATORY PROTECTION

The primary objective of a respiratory protection program is to limit the inhalation of airborne radioactive materials and other hazardous materials. This objective is normally accomplished through the use of engineering controls, including process, containment, and ventilation equipment. When engineering controls are not feasible or cannot be applied, respiratory protection must be used. The Health and Safety Group is responsible for the implementation of the respiratory protection program at the LRC. The program is based on the guidance contained in 10 CFR 20, Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection," and NUREG-0041, "Manual of Respiratory Protection Against Airborne Radioactive Materials."

The respiratory protection program will include the following:

1. Air sampling and other surveys sufficient to identify the hazard, to evaluate individual exposures, and to permit proper selection of respiratory protection equipment.

2. Written procedures to ensure proper selection, supervision, and training of personnel using such protective equipment.

3. Written procedures to ensure the adequate individual fitting of respirators, as well as procedures to ensure the testing of respiratory protective equipment for operability immediately prior to each use.

4. Written procedures for maintenance to ensure full effectiveness of respiratory protective equipment, including procedures for cleaning and disinfecting, decontaminating, inspecting, repairing, and storing.

5. Written operational and administrative procedures for the control, issuance, proper use, and return of respiratory protective equipment, including provisions for planned limitations on duration of respirator use for any individual as necessitated by operational conditions.

6. Bioassays and other surveys, as appropriate, to evaluate individual exposures and to assess the protection actually provided.

7. Records sufficient to permit periodic evaluation of the adequacy of the respiratory protection program.

8. Determination prior to assignment of any individual to tasks requiring the use of respirators that such an individual is physically able to perform the work and use the respiratory protective equipment. A physician is to determine what health and physical conditions are pertinent. The medical status of each respirator user is to be reviewed at least annually.

Other details of an effective respiratory protection program can be found in the above mentioned documents and the LRC health physics procedures.

12.10 OCCUPATIONAL EXPOSURE ANALYSIS

12.10.1 External Exposure - The external radiation exposure received by LRC employees is presented in Tables 12-13 through 12-16. Tables 12-13 and 12-14 show the exposures by ranges and the number of employees in each range for calendar years 1984 and 1985 respectively.

TABLE 12-13

1984 LRC EXPOSURES BY RANGE

Annual Whole Body Dose Ranges (Rems)	Number of Individuals In Each Range
No Measurable Exposure	87
Measurable Exposure <0.100	77
0.100 to 0.250	33
0.250 to 0.500	12
0.500 to 0.750	6
0.750 to 1.000	1
1.000 to 2.000	3
2.000 to 3.000	1
3.000 to 4.000	0
4.000 to 5.000	0
5.000 to 6.000	0
6.000 to 7.000	0
7.000 to 8.000	0
8.000 to 9.000	0
9.000 to 10.000	0
10.000 to 11.000	0
11.000 to 12.000	0
>12.000	0
	220

TABLE 12-14

1985 LRC EXPOSURES BY RANGES

Annual Whole Body Dose Ranges (Rems)	Number of Individuals In Each Range
No Measurable Exposure	185
Measurable Exposure <0.100	83
0.100 to 0.250	21
0.250 to 0.500	17
0.500 to 0.750	5
0.750 to 1.000	3
1.000 to 2.000	2
2.000 to 3.000	2
3.000 to 4.000	0
4.000 to 5.000	0
5.000 to 6.000	0
6.000 to 7.000	0
7.000 to 8.000	0
8.000 to 9.000	0
9.000 to 10.000	0
10.000 to 11.000	0
11.000 to 12.000	0
>12.000	0
	318

Table 12-15 presents the exposures received by LRC employees for calendar years 1981 through 1984. The row entitled "Off Site" gives the exposures received by LRC employees at other licensed facilities.

TABLE 12-15

LRC RADIATION EXPOSURE

	1984	1983	1982	1981
Total Person Rems	23.5	18.4	19.4	26.3
Off Site	3.5	2.0	2.5	3.0
LRC	20.0	16.4	16.9	23.3
Average Exposure	0.09	0.088	.105	.137
Number of Workers	220	208	184	192
Highest Exposure	2.25	2.04	1.9	1.7

The exposure received by LRC employees is categorized by group in Table 12-16 for exposures received for calendar years 1983 and 1984.

TABLE 12-16

EXPOSURE BY GROUP (PERSON REMS)

Group	1984	1983
Plant Engineering	5.10	2.25
Project Services	0.07	0.05
Health & Safety	1.85	2.16
Nuclear Materials	11.40	9.60
Chemical & Nuclear Engineering	1.30	1.53
Nondestructive Methods	0.58	0.17
Process Control	0.00	0.49
Systems Design & Engineering	2.12	2.09

Calendar year 1984 brought increased activity in our hot cell facility. This typically results in increased exposures to personnel in the Nuclear Materials, Plant Engineering, and Health and Safety Groups. Table 12-15 reflects this in all categories. Table 12-16 also reflects this increase in two of the three affected groups. Only Health and Safety saw a reduction in the group's exposure. The amount of exposure received from off-site work reversed a three year period of decreases. Table 12-16 reflects this in the increase in the Systems Design & Engineering Group's exposure.

The increases noted in Tables 12-15 and 12-16 do not indicate a decrease in the vigilance given by LRC management to personnel exposures nor do they suggest a decreased ALARA emphasis. Exposure history at the LRC shows wide variances because of the variety of work that is performed here. Clear trends have not been evident. If the amount of hot cell work is considered and the fact that objects received for examination exhibit higher levels of radioactivity, the effectiveness of the ALARA program can be appreciated. The preliminary exposure information required on the Radiation Work Permit form was increased in early 1985. This has resulted in many improvements in the manner that cell entries are made.

12.10.2 Internal Exposure - The bioassay sampling, lung counting, and air sampling programs show that the worker is exposed to extremely low levels of respirable activity.

12.10.2.1 Bioassay Results - Urine bioassay samples are taken primarily of workers who perform work with unclad uranium and those involved in any work with plutonium. Table 12-17 below presents the number of urine bioassay samples taken during 1983 and 1984.

TABLE 12-17

NUMBER OF URINE BIOASSAY SAMPLES

Month	1983 U	1983 Pu	1984 U	1984 Pu
January	5	-	20	4
February	-	-	13	6
March	-	-	15	12
April	19	19	18	9
May	-	-	13	5
June	16	16	17	8
July	11	8	15	6
August	10	8	15	7
September	11	14	14	7
October	11	9	16	5
November	3	1	-	-
December	5	5	14	6

In 1983, all samples for uranium were less than 5 grams/liter (lower limit of detection), except on four occasions when the analysis indicated the presence of uranium but none met the resample limit of 20 grams/liter. All plutonium analyses were below the minimum sensitivity which varied from 0.00 $\pm$ 0.1 to 0.3 $\pm$ 0.4 dpm per sample.

In 1984, all samples for uranium were less than 5 grams/liter (lower limit of detection), except on one occasion 27 g/liter was reported. A resample showed that the level had returned below the lower limit of detection. All plutonium samples indicated 0.0 $\pm$ (0.01 to 0.6).

12.10.2.2 Air Sampling Results - The air sampling program is the first line of defense for all operations of this type, but the bioassay program, along with lung counts, is the final step in the estimation of exposure that may occur.

12.10.2.2.1 Table 12-18 presents a summary of the air sampling program at the LRC for calendar year 1983, for fixed air samplers.

TABLE 12-18

1983 AIR ACTIVITY

(VALUES IN μCi/ml)

Labs	Approximate Average	Maximum Concentration	MPC
15	3×10^{-15}	1.2×10^{-14}	1×10^{-10}
16	3×10^{-15}	1.5×10^{-14}	1×10^{-10}
17	8.7×10^{-15}	1.8×10^{-13}	4×10^{-11}
19	7×10^{-15}	8.7×10^{-14}	1×10^{-10}
27	2.4×10^{-15}	5.7×10^{-15}	1×10^{-10}
44*	2×10^{-15}	6.5×10^{-15}	1×10^{-10}
Cask Handling Area	1.9×10^{-12}	1.27×10^{-10}	9×10^{-9}
	6.7×10^{-15}	4.5×10^{-13}	4×10^{-11}
Hot Cell	1×10^{-14}	1.25×10^{-13}	9×10^{-9}
	5×10^{-16}	1.2×10^{-15}	4×10^{-11}
Recirculated Air "C"	1.5×10^{-14}	3.5×10^{-13}	9×10^{-9}
	4×10^{-15}	1.93×10^{-14}	4×10^{-11}
Waste Storage Area	1.5×10^{-14}	2.6×10^{-14}	9×10^{-9}
	7×10^{-16}	1.7×10^{-15}	4×10^{-11}
Laundry	3×10^{-14}	1.5×10^{-13}	9×10^{-9}
	3×10^{-15}	2.5×10^{-14}	4×10^{-11}
Radio Chem Lab	7×10^{-14}	2.3×10^{-12}	9×10^{-9}
	1.5×10^{-15}	1.5×10^{-14}	4×10^{-11}

*Discontinued in Sept.

12.10.2.2.2 On 338 occasions in 1983, breathing zone air samples were taken to measure the airborne activity to which workers were exposed. In no case was anyone exposed to greater than 2 MPC of airborne activity in any one week. In most cases, respiratory protection was used and exposure levels were at least a factor of 1,000 below the limits.

There are three major operations which require respiratory protection, and several minor ones.

1. Entries into the isolation area behind the hot cell. A supplied air respiratory system was installed in January, 1980, in the hot cell area which has a protection factor of at least 1,000. This system incorporates a double bibb hood which has reduced airborne activity to which a worker is exposed to below detectable levels.

2. Operations outside of the isolation area in the cask handling area using the 3M hood and the supplied air respiratory system. This system incorporates the 3M hard hat which is NIOSH approved with a protection factor of 1,000. Breathing zone samples are taken outside of the hood each time this system is used.

3. Operations in Building C may involve bagging operations with plutonium glove boxes. All operations of this type require respiratory protection. When it is used, a breathing zone sample is taken. Normally, the powered respirator with 1,000 protection factor is used; however, the full face masks with a protection factor of 50 may be used.

4. Other minor operations requiring respiratory protection are: changing HEPA filters, repair work on NPD site support equipment, and any other operations where Health and Safety believes that there is a potential of airborne activity.

5. It should to be noted that a major operation is occurring in the decommissioning of Building C that is requiring the use of respiratory protection for industrial safety reasons, not for protection from radioactive materials. A number of operations are very dusty (paint chipping,

concrete destruction, etc.). A NIOSH approved full flow hard hat system is used. With no protection factor, no one in Building C has been exposed in excess of 2 MPC hr in one week. In most cases, radioactivity above background is undetectable.

12.10.2.2.3 Table 12-19 presents a summary of the air sampling program at the LRC for calendar year 1984, for fixed air samplers.

TABLE 12-19

1984 AIR ACTIVITY

(VALUES IN μCi/ml)

Labs	Approximate Average	Maximum Concentration	MPC
15*	3E-15	1.6E-14	1E-10
16*	2E-15	5E-15	1E-10
17*	5E-15	3.9E-13	4E-11
19	7E-15	1E-13	1E-10
27**	2.4E-14	7.5E-15	1E-10
Soil Processing***	1E-15	7.4E-15	4E-11
Cask Handling Area	5E-13 5E-15	1.2E-11 5E-13	9E-9 4E-11
Hot Cell	8E-15 5E-16	6.7E-13 1. E-14	9E-9 4E-11
Recirculated Air Building C	1.5E-14 1.5E-15	1.1E-13 3.3E-13	9E-9 4E-11
Waste Storage	1.5E-14 7E-16	2.9E-14 3.3E-15	9E-9 4E-11

Laundry	3E-14	6.3E-14	9E-9
	2E-15	3.3E-15	4E-11
Radio Chem	3E-14	1.0E-12	9E-9
	1.5E-15	2.4E-15	4E-11

*Discontinued November 1984
**Discontinued June 1984
***Begun May 1984

12.10.2.2.4 On 278 occasions in 1984, breathing zone air samples were taken to measure the airborne activity to which workers were exposed. In no case was anyone exposed to greater than 3 MPC hour of airborne activity in any one week. In most cases, respiratory protection was used and exposure levels were at least a factor of 1000 below the limits.

There are three major operations which require respiratory protection, and several minor ones.

1. Entries into the isolation area behind the hot cell. A supplied air respiratory system was installed in January, 1980 in the hot cell area which has a protection factor of at least 1000. This system incorporates a double bibb hood which has reduced airborne activity to which a worker is exposed to below measurable levels.

2. Operations outside of the isolation area in the cask handling area use the 3M hood and the supplied air respiratory system. This system incorporated the 3M hard hat which is NIOSH approved with a protection factor of 1000. Breathing zone samples are taken outside of the hood each time this system is used.

3. Operations in Building C may involve bagging operations with plutonium glove boxes. All operations of this type require respiratory protection. When it is used, a breathing zone sample is taken. Normally, a 20T air line respirator with a 1000 protection factor is used; however, the full face mask with a protection factor of 50 may be used.

4. Other minor operations requiring respiratory protection are: changing of HEPA filters, repair work on NPD site support equipment, and any other operation where Health and Safety believes that there is a potential of airborne activity.

5. It should to be noted that a major operation is occurring in the decommissioning of Building C that is requiring the use of respiratory protection for industrial safety reasons, not for protection from radioactive materials. A number of operations are very dusty (paint chipping, concrete destruction, etc.). A NIOSH approved full flow hard hat system is used. With no protection factor, no one in Building C has been exposed in excess of 2 MPC hours in one week. In most cases, radioactivity above background is undetectable.

12.10.2.3 In-vivo Results (1983) - Whole body counting was performed by Helgeson Scientific Services, Inc. on 32 employees during 1983. Three had detectable activities, no other workers indicated detectable activity. The results of the three employees with detectable activity is presented in Table 12-20.

TABLE 12-20

WHOLE BODY COUNTS
1983

(ALL VALUES IN NANOCURIES)

		Employee		
Isotope	MPBB	1	2	3
Cs-137	3E4	8±2	4±2	
Mn-54	3.6E3	5±2		4±1
Co-60	1.1E3		3±1	7±1

In-vivo counting was performed on seven employees during 1983, for plutonium and Americium-241. These results are summarized in Table 12-21.

TABLE 12-21

Am - Pu LUNG COUNTING
1983

(ALL VALUES IN NANOCURIES)

Employee	Pu	Am
1	0	0.00±0.10
2	0	0.00±0.11
3	0	0.13±0.13
4	0	0.00±0.14
5	0	0.00±0.15
6	0	0.00±0.19
7	0	0.00±0.16

In-vivo lung counting was performed on nine employees in 1983, for uranium. The results are listed in Table 12-22. Four of the nine indicated positive results. However, these results were not confirmed in followup urinalyses.

TABLE 12-22

URANIUM LUNG COUNTING
1983

(ALL VALUES IN MICROGRAMS)

Employee	U-235
1	0±30
2	0±43
3	0±39
4	42±37
5	0±41
6	38±33
7	76±45
8	0±39
9	49±44

12.10.2.4 In-vivo results (1984) - Whole body counting was performed by Helgeson Scientific Services, Inc. on 99 employees during 1984. Twelve had positive results but these were very low levels. A summary is presented in Table 12-23.

TABLE 12-23

WHOLE BODY COUNTS
1984

(EXPOSURE VALUES IN NANOCURIES)

Isotope	Number of Employees	Maximum Observed	MPBB
Cs-134	1	3.0	2E4
Cs-137	7	9.0	3E4
Co-60	4	4.0	1.1E3

In-vivo lung counting was performed on 14 employees during 1984 for Plutonium-239 and Americium-241. No plutonium was reported. The presence of Americium-241 was indicated for 5 employees with the highest quantity being 0.26 Nanocuries (±0.14) for one person.

In-vivo lung counting was performed on 20 employees during 1984 for Uranium-235. In 5 instances, the results were positive with the highest result being 48 micrograms (±37) for one person.

12.11 MEASURES TAKEN TO IMPLEMENT ALARA

12.11.1 Irradiated metal specimens had been stored on the roof of the hot cells in an open top cave. This configuration caused this roof to be designated as a high radiation area. Several small totally enclosed caves have been constructed for the storage of these specimens which has eliminated the high radiation area on the cell roof, thus reducing exposures received by personnel who periodically enter the area for maintenance on the HEPA filters and to calibrate an area monitor. It also eliminated the radiation area on the roof of Building B which no longer contributes to the exposure of workers who maintain the building ventillation system.

12.11.2 Cleaning of the hot cells contributed significantly to exposure doses of workers. This cleaning operation, which is performed at three or four year intervals, requires the set-up table in the cell to be dismantled. In 1985, this operation was performed remotely with a modified saw so that personnel did not enter the cell for this high exposure work.

12.11.3 Trash removal from the hot cell during cell cleaning operations was significant in the past. During the cleaning operation in 1985, trash was remotely loaded into special metal drum liners that were designated to fit into 30-gallon drums and to be handled with long poles. This process modification reduced personnel exposures for this part of the operation considerably.

12.11.4 The LRC has purchased a TLD reader which provides immediate information on worker exposure. This system is not intended to replace the normal contract service for dose measurement but rather to provide prompt indication of unexpected exposures for non-routine operations. The system makes possible the estimation of exposures to hard to measure areas of the body such as the soles of feet, hands and fingers.

12.11.5 A supplied air respiratory system has been installed to support hot cell work, principally during hot cell entries. This system provides a greater protection factor for workers in addition to providing greater worker comfort while performing the strenuous work.

12.11.6 The Radiation Work Permit (RWP) approval process has been revised. Previously, the worker or his supervisor completed the RWP form and carried it to those personnel who were required to sign it. This method has been changed such that the workers, supervisors and signators of the RWP gather at a meeting where the proposed work scope and methods are discussed in detail. All facets of work are agreed to before any authorization signatures are placed on the RWP. This new approval process requires more time being spent for the planning stage of a task but considerable exposure savings have results.

12.12 BIOASSAY PROGRAM

Those employees routinely working in contamination or airborne radioactivity areas will be scheduled for participation in the bioassay program. The Health and Safety Group will select those employees to be sampled in the program. This selection will be based on the probability of exposure, the employee's work habits, the type of work in the area, air sample data, previous bioassay data, etc. Routine bioassay may consist of check or whole-body counting (in-vivo bioassay) or excretion analysis (in-vitro bioassay). In-vivo bioassay is performed routinely by a bioassay service which comes on-site for the evaluations. In-vitro bioassay is performed by a commercial laboratory located off-site.

Bioassay action criteria for plutonium are outlined in Table 12-3 & 12-4. In general, no action is required if the excretion result (i.e., urinalysis) is less than 0.2 dpm/liter or the in-vivo measurement of material in the lung is less than 16 nanoCuries. All compounds of plutonium are considered to be either class W or Y. This classification refers to the most recent evaluation of the ICRP for internal dose calculations. Class W compounds are moderately soluble and clear from the pulmonary region of the lung with half-times in the range 10 to 100 days. Class Y compounds are essentially insoluble and are considered to clear from the pulmonary region with halftimes of >100 days. No compounds of plutonium are

considered by the ICRP to be readily soluble (i.e., class D compounds which clear from the lungs in <10 days).

The bioassay program for uranium generally follows that outlined in Regulatory Guide 8.11, "Application of Bioassay For Uranium," June 1974. There are two exceptions to this general guidance:

1. Employees off-site during the regular visit of the bioassay service will not be scheduled for a special, make-up count, if the count was scheduled only for routine exposure control monitoring.

2. Bioassays of employees working in areas in which both plutonium and uranium may be airborne shall be evaluated for both plutonium and uranium. The Supervisor, Health and Safety may decide to analyze for only one of these elements, if it can be demonstrated that the analysis for a single element is a more sensitive indicator of an uptake.

Bioassay action criteria for uranium are outlined in Table 12-5 & 12-6.

Employees working primarily with beta and gamma emitting radionuclides will also be included in the in-vivo bioassay analysis program. Any employee suspected of an exposure greater than 40 MPC-hours will be scheduled for a bioassay evaluation as soon as practicable after the exposure. Bioassay action criteria for beta-gamma are outlined in table 12-7.

12.13 AIR SAMPLING AND MONITORING

The presence of airborne radioactive materials in the working areas of the LRC is determined through the combined use of air samplers and monitors. These programs are discussed below:

12.13.1 Air Sampling Program

The air sampling program can be divided into two categories; fixed and portable. Selection of the sampling category and the frequency of sampling is left to the discretion of the Supervisor, Health and Safety.

12.13.1.1 Fixed Air Samplers - Air samples are obtained at designated points through the use of a central vacuum system. Sampling points are located as close as possible to a permanent operator station to permit continuous sampling of the air near the worker's breathing zone. These samples are usually collected weekly. However, the frequency may vary as the situation dictates.

Normally, these are evaluated within two weeks, after allowing the appropriate decay period for the radon daughter products. However, based on the particular operation, etc., a Health Physics Engineer may determine that it is necessary to evaluate the samples without allowing for the decay period. In these cases, an applicable radon decay correction factor must be applied to the results.

12.13.1.2 Portable Samplers - Air samples in the approximate breathing zone of a worker may be obtained through the use of a lapel sampler. The lapel sampler consists of a small sampling head attached to the worker's lapel (or collar) connected through a small flexible tube to a small air-pump worn at the waist. The flow rates through these samplers are quite low when compared to the fixed system. However, since the sampler is located near the nose and mouth and moves with the worker as he moves about the area, it provides a reasonable estimate of the concentration of airborne radioactivity in the breathing zone of the worker.

Air samples obtained with these samplers are evaluated on a low background, proportional counting system. Factors are applied to the counting results to account for background activity and detector efficiency. All results are reported in units of activity/unit volume of air sampled.

12.13.2 Air Monitoring Program

Air monitoring in operating areas of the LRC is accomplished with continuous monitors in predetermined, fixed locations. Normally, a monitor is placed in each radioactive materials handling area in which there is a potential for the release of airborne radioactivity. Locations are selected based upon the ability of the monitor to provide a reasonable evaluation of the airborne activity in a particular area and to provide adequate warnings to those in the area of changing conditions. These determinations are made by the Health and Safety Group based upon the operations in the area, the potential for release, and the quantity and chemical form of the material.

Alarms are set in accordance with the particular operation, the material being handled, and the potential for release. Actual alarm points are set as low as possible commensurate with the ambient radiation levels in the area.

12.14 SURFACE CONTAMINATION

12.14.1 Smear Surveying

Smear surveys are performed in all areas specified in the license and which, in the judgment of the Supervisor, Health and Safety, have a potential for surface contamination. The frequency of these surveys will be based upon the potential for contamination in the area, previous experience with contamination in the area, and the need to keep the area free from contamination. Typical areas and survey schedules are listed in Table 12-9, however, both the areas included and the frequencies of surveys are subject to change based upon the current research activities at the LRC. The frequency of smear surveys in areas not included in the table are generally specified in the procedure covering the particular area.

12.14.1.1 Smear Samples - Smear samples are obtained with small, absorbent filter papers. The smear paper is moved across an area of approximately 100 sq. cm. using about 5 pounds of pressure. The smear may be counted with a portable gas-flow proportional counter capable of detecting alpha or beta radiation. Normally, smear samples are evaluated in a stationary counter located in the Health Physics Laboratory. Appropriate conversion factors are applied to the net counts to express the smear results in units of disintegrations per minute.

12.14.1.2 Large Area Smears - Large area smears are obtained using the dust mop technique in areas around the site, the hot cell operations area, the change room and main hallways in Building B. These smears are intended to indicate the general contamination environment in an area and may lead to a more extensive survey, if unexpected contamination is indicated. Normally, large area smears are evaluated with a hand-held, portable survey instrument (e.g., a gas-flow proportional counter such as the PAC 4G). Actions to be taken in response to the results of large area smears are outlined in Table 12-24.

12.14.1.3 Action Levels - Included in Table 12-25 are the appropriate action levels to be used in designated areas of the LRC. Decontamination shall be initiated in areas in which the removable surface contamination levels exceed these action levels. The Health and Safety Group shall determine and direct the actions to be taken to protect LRC personnel working in these areas and to reduce contamination levels as far below those listed in Table 12-1 as is possible. Normally, decontamination of an identified area shall begin within 24 hours of the discovery.

In some cases, for example, if the contamination is discovered just prior to a weekend or a regularly scheduled holiday, the contaminated area may be marked and posted appropriately. Such a determination shall be made by the Health and Safety Group based upon the severity and extent of the contamination and the potential for further contamination of equipment and/or personnel during the interval. Decontamination of the area shall begin on the first regular work-day after discovery.

TABLE 12-24

ACTION LEVELS FOR LARGE AREA SMEARS

1. Routine Large Area Smears (1000 - 5000 dpm)

   Repeat the large area smear. If results show levels of contamination above 1000 dpm, take smears in smaller areas to locate the source.

   Decontaminate all areas in which the smear results indicate contamination above 1000 dpm per 100 sq. ft.

2. Routine Large Area Smears (5000 - 10,000 dpm)

   Repeat the large area smear. If results show levels of contamination above 5000 dpm, isolate the contaminated area. Take smears in smaller areas to locate the source.

   Decontaminate all areas in which the smear results show contamination in excess of 1000 dpm per 100 sq. ft.

3. Routine Large Area Smears (>10,000 dpm)

Isolate the contaminated area.

Survey all personnel in the contaminated area.

Take smaller smears in the area to locate the source.

Decontaminate all areas in which the smear results show contamination in excess of 1000 dpm per 100 sq. ft.

Survey all persons leaving the building.

NOTE:

Routine large area smears are normally taken in the early afternoon to facilitate clean-up of areas found to be contaminated before the end of the normal work-day.

TABLE 12-25

SMEAR SURVEY FREQUENCIES AND ACTION LEVELS

Alpha Radiation Smear Survey

Area	Frequency	Action Level (dpm/100 sq. cm.)
Unirradiated, unencapsulated fuel handling areas	weekly	5,000
Building B counting laboratory	monthly	200
Building A laboratories	monthly	200
Hot cell operations area	monthly	200
Scanning electron microscopy laboratory	monthly	200

Exit portals from controlled	twice weekly	200

Beta Radiation Smear Survey

Area	Frequency	Action Level (dpm/100 sq. cm.)
Building A Laboratories	monthly	2,000
Building B Counting Laboratory	monthly	2,000
Scanning Electron Microscopy Laboratory	monthly	2,000
Hot Cell Operations Area	twice monthly	2,000
Cask Handling Area	twice monthly	22,000
Radiochemistry Laboratory	twice monthly	22,000
Exit Portals From Controlled Areas	twice monthly	2,000

12.14.2 Direct Radiation Surveys

Surveys of the direct radiation exposure in areas of the LRC are to be performed on a frequency established by a Health Physics engineer. In general, these surveys require the selection of the appropriate portable survey instruments based upon the anticipated radiation levels, the types of radiation expected, and the nature or type of survey to be performed. General maps of the areas to be surveyed may be used to record the measured ambient radiation levels and/or, in some cases, to designate specific areas in which the exposure rates should be measured. The survey should also include a visual examination of the area for any unusual conditions or work habits which could affect the exposures received by personnel working in these areas. Items of this nature should be reported immediately to the Supervisor, Health and Safety, or corrected immediately, if practical.

Results of these surveys should be reviewed by a Health Physics Engineer to ensure that the proper posting requirements are in effect for the area and to ensure that appropriate actions are taken to keep all exposures ALARA.

Action levels for direct radiation surveys are presented in Table 12-26.

TABLE 12-26

CONTAMINATION ACTION LEVELS

Area	Type of Radiation	Fixed Surface Reading	Transferable Surface Contamination (dpm/100 sq. cm.)
Uncontrolled	Alpha	300 dpm/100 sq. cm.	30
	Beta-Gamma	0.1 mrad/h	220
Contamination*	Alpha	3000 dpm/100 sq. cm.	2,200
	Beta-Gamma	1.0 mrad/h**	22,000

* The Supervisor, Health and Safety may raise these action levels. Justification for this action must be documented and forwarded to the Safety Review Committee for their review and approval.

** This action limit applies to contamination areas which are normally radiation areas. This level of contamination will not cause a significant increase in radiation exposure.

NOTE:

This table provides limits above which decontamination must be initiated. These action levels pertain to areas normally accessible to personnel performing normal work functions. The levels do not apply to areas requiring extraordinary precautions for entry, e.g., the Isolation Area, waste water tanks, etc. In these cases, direct health physics coverage is the primary control mechanism.

12.14.3 Personnel Contamination Surveys

Personnel are required to monitor themselves for activity present on their hands, shoes, clothing, and person before exiting a contamination area. Contamination monitors (friskers) are located at all exits from contamination areas for this purpose. The detector (probe) should be held as close to the surface of the item being monitored as possible (without touching the item) and the probe should be moved at a speed of about 0.5 inch/second. Allowable levels of contamination on skin surfaces and on items of clothing are given in Tables 12-11 and 12-12. Any contamination in excess of these limits should be reported immediately to the Health and Safety Group.

13.0 ENVIRONMENTAL SAFETY

13.1 ENVIRONMENTAL MONITORING

Environmental sampling of the area surrounding the LRC is performed on a regular basis to evaluate changes in the levels of radioactivity in air, water, and vegetation. The minimum environmental program consists of the following.

- o one continuous on-site background air sample (Figure 13-1)
- o monthly water samples from the James River collected above and below the liquid discharge point (Figure 13-2)
- o continuous sampling of rain water on-site (Figure 13-1)
- o quarterly samples of river silt and near-river vegetation (Figure 13-2).

Normally, LRC personnel are responsible for collecting the environmental samples. Analysis of these samples may be performed on-site or the samples may be analyzed by a commercial laboratory.

Environmental sampling data for the calendar years 1982, 1983, and 1984 is given in "Lynchburg Research Center, Environmental Report, October, 1985," Tables 2-2 through 2-6.

13.2 EFFLUENT AIR MONITORING

Potentially contaminated air from chemical hoods, hot cells, and glove boxes is discharged ultimately through the 50-meter stack. Generally, exhaust air containing beta-gamma activity is passed through a single-stage HEPA filter which is sufficient to remove airborne particulates. Air from more hazardous operations, e.g., from glove boxes, is routed through a two-stage HEPA filter.

Discharge through the stack is accomplished with a large blower, powered normally by a large electric motor operated on off-site power. Emergency power is supplied by an internal combustion engine coupled to the blower shaft through a centrifugal clutch. On loss of off-site power, the engine starts automatically and takes over the load upon reaching the proper speed.

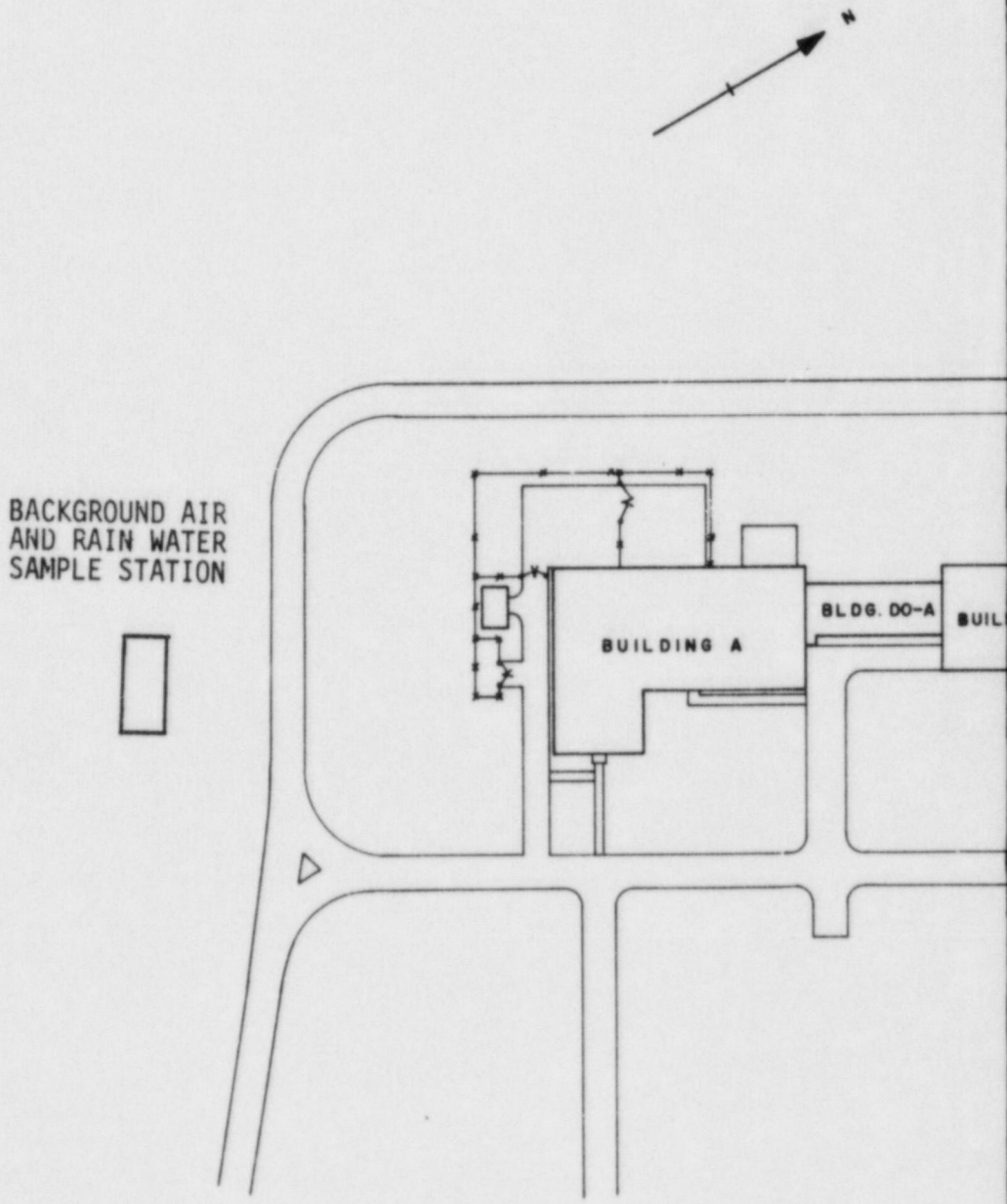
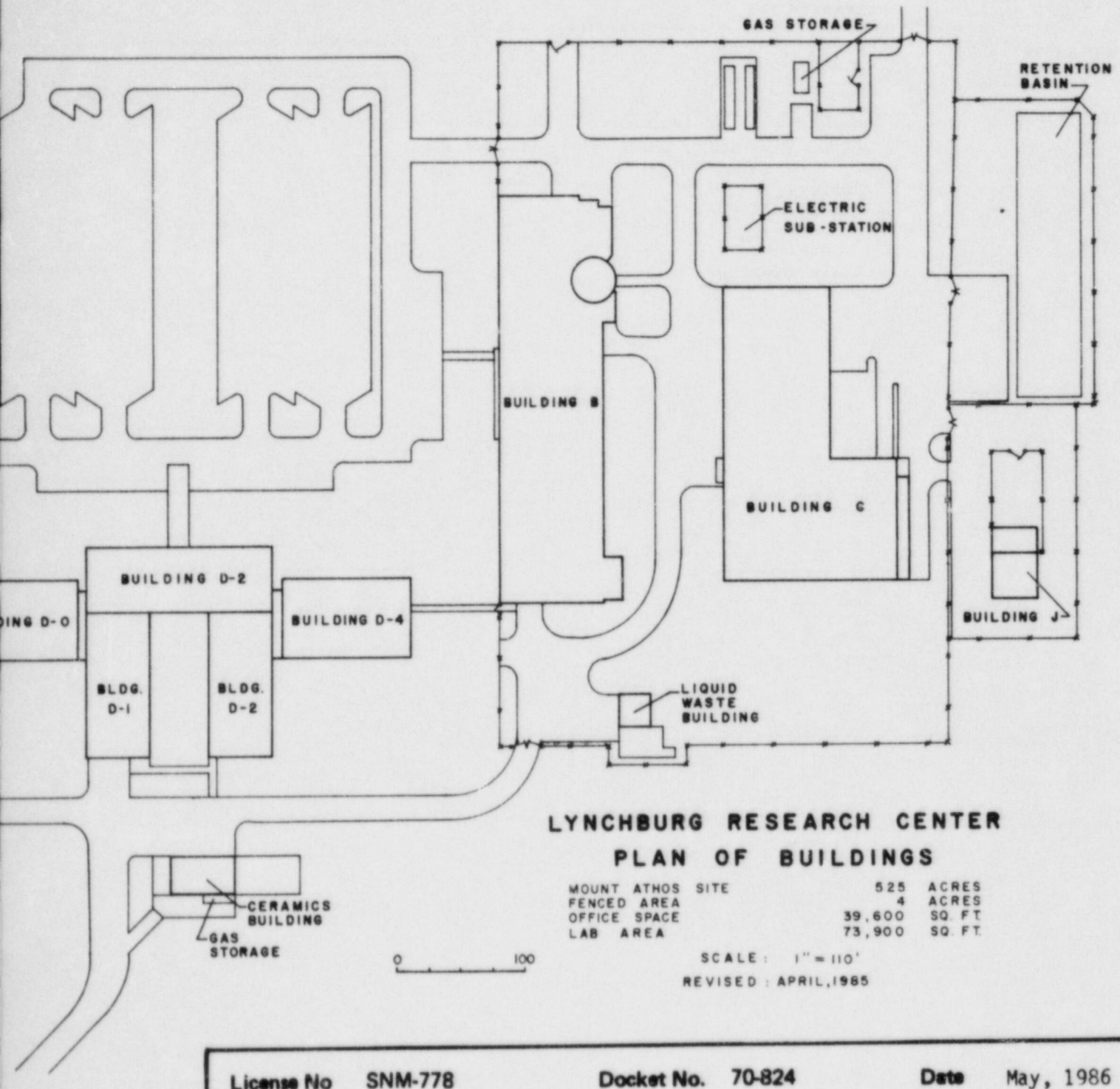

N
BACKGROUND AIR
AND RAIN WATER
SAMPLE STATION
BUILDING A
BLDG. DO-A

FIGURE 13-1

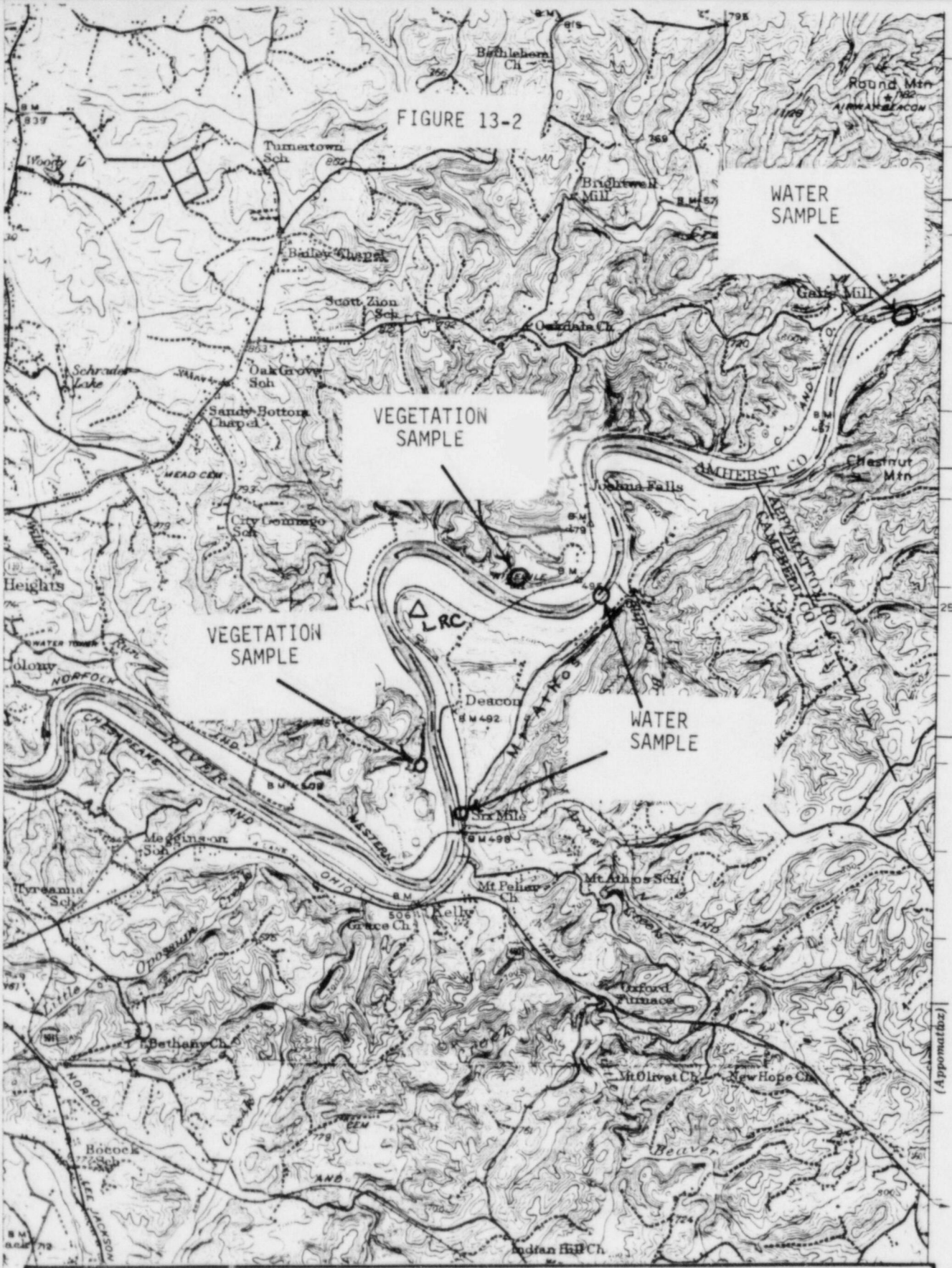


LYNCHBURG RESEARCH CENTER
PLAN OF BUILDINGS

MOUNT ATHOS SITE	525	ACRES
FENCED AREA	4	ACRES
OFFICE SPACE	39,600	SQ. FT.
LAB AREA	73,900	SQ. FT.

SCALE: 1" = 110'

REVISED: APRIL, 1985



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Amendment No.	0	Revision No.	2	Page	13-4