70-27

Babcock & Wilcox

a McDermott company

Naval Nuclear Fuel Division

P. O. Box 785
Lynchburg, Virginia 24505-0785
(804) 522-6000

August 10, 1994
94-069

U. S. Nuclear Regulatory Commission
ATTN: Mr. R. C. Pierson, Chief
Licensing Branch
Division of Fuel Cycle Safety and
Safeguards, NMSS
Washington, DC 20555

Gentlemen:

Our application for consolidation of license SNM-778 under license SNM-42 was first submitted on May 5, 1994. Comments on the application were addressed in the re-submittal July 15, 1994. B&W also agreed to submit a demonstration section for inclusion in the license SNM-42 application.

Enclosed with this letter is the demonstration section for the Lynchburg Technology Center. This is an addition to Section V of the NNFD license. This is a new chapter to be placed in the unclassified portion of Section V. It provides essentially the same information as that which exists in the demonstration section of License SNM-778. Position titles have been changed to reflect the revised and combined organization and new site drawings are provided.

If additional information is needed, please contact me.

Sincerely,

Arne F. Olsen

Arne F. Olsen
Licensing Officer

cc: U.S.N.R.C.
Region II

NRC Resident Inspector

NF12
1/1

SNM-42

SECTION V

CHAPTER 18

LTC DEMONSTRATION

SECTION V, CHAPTER 18.1

TABLE OF APPLICABLE PAGES

Page	Date
18.1-i	08/05/94
18.1-1	08/05/94
18.1-2	08/05/94
18.1-3	08/05/94
18.1-4	08/05/94
18.1-5	08/05/94
18.1-6	08/05/94
18.1-7	08/05/94
18.1-8	08/05/94

CHAPTER 18.1

OVERVIEW OF OPERATION

TABLE OF CONTENTS

List of Figures

18.1 OVERVIEW OF OPERATION

18.1.1 SUMMARY OF OPERATING OBJECTIVE AND PROCESS

Activities, utilizing licensed material, are conducted at the Lynchburg Technology Center (LTC) in support of the operating divisions of Babcock & Wilcox and for other companies and government organizations. The broad range of projects that have been conducted pursuant to the license cannot be described in terms of through-put or any single process. Radioactive materials are handled and stored, principally in Building B. That building houses the Hot Cells, Radiochemistry Laboratory, Scanning Electron Microscopy Laboratory, Metallurgy Laboratories, Analytical Chemistry Laboratory, Fatigue and Fracture Laboratory, Failure Analysis Laboratory, Crane and Cask Handling Area, a Hot Machine Shop, the Counting Room, and a Health Physics Laboratory.

Licensed material in the form of liquid waste is collected in tanks that are located in the Liquid Waste Disposal Facility. A laundry is also located in the Liquid Waste Disposal Facility. Solid radioactive waste is stored in Building J, the Annex to Building J, the storage area adjacent to Building J, and the Temporary Storage Facility (In-Ground Storage).

18.1.2 SITE DESCRIPTION

The LTC is located on the James River about four miles east of Lynchburg, Virginia. The site, which comprises 525 acres, lies within Campbell County and borders on Amherst County. The site occupies about 13.6 acres of the site.

The irregularly shaped property is bounded on three sides by a large loop of the James River and on the fourth side by State Route 726, which closely follows the base of Mount Athos. This mountain rises rapidly from about 500 feet MSL to 900 feet MSL, making it the dominant feature of the surrounding landscape. The Babcock & Wilcox property consists of large sections of relatively flat floodplain along the James River lying at about 470 feet MSL. The interior of the property is largely composed of rolling hills, one of which rises to almost 700 feet MSL.

The land in the immediate vicinity of the plant is sparsely inhabited. The severe topography makes it unsuitable for commercial farming. The Lynchburg Foundry, a producer of light metal castings, occupies a parcel of land which abuts the south boundary of the Babcock & Wilcox property. The Foundry is approximately 0.5 miles from the LTC.

The site is serviced by a spur of the CSX Railroad which runs through the Babcock & Wilcox property. The property is also conveniently located for truck and automobile access. About three miles from the site, State Route 726 connects with U.S. Highway 460, a major link between Roanoke and Richmond. The LTC is located about 100 feet above the James River and for that reason no dams on the river would threaten the LTC should they fail.

18.1.3 LOCATION OF SITE BUILDINGS

Figure 18.1-1 shows the layout of buildings at the site. All buildings are of masonry construction.

18.1.4 LICENSE HISTORY

The LTC was originally operated under NRC License No. SNM-778, which was issued on September 16, 1966. Since that time, the following renewals and amendments were approved to that license prior to its incorporation into NRC License No. SNM-42:

February 15, 1974	First renewal
July 21, 1980	Second renewal

August 28, 1981	Amendment No. 1, approved a change in the organization.
February 25, 1982	Amendment No. 2, approved the Radiological Contingency Plan
May 11, 1982	Amendment No. 3, approved the reduction in the possession limit for unirradiated Pu to 0.9 kilograms
May 14, 1982	Amendment No. 4, added four license conditions requiring surveys of B & W and railroad property and established a restricted zone.
August 11, 1983	Amendment No. 5, approved the reduction of the possession limits of SNM to below one effective kilogram.
February 28, 1984	Amendment No. 6, approved a revision to the Radiological Contingency Plan.
November 6, 1984	Amendment No. 7, approved a revision to the definition of the Restricted Area.
July 25, 1985	Amendment No. 8, approved the extension of the expiration date of the license to January 1, 1986.
October 15, 1985	Amendment No. 9, approved a change in the organization.
July 30, 1987	Renewal
August 18, 1987	Amendment No. 1, to remove Building A as an authorized place of use.
September 22, 1987	Amendment No. 2, to incorporate a revision to the Radiological Contingency Plan.

October 2, 1987	Amendment No. 3, to incorporate organizational changes.
August 4, 1988	Amendment No. 4, to remove Building C as an authorized place of use and to authorize the storage of dry radioactive waste in the Temporary Storage Facility.
August 22, 1988	Amendment No. 5, authorized laundry operations; changes to the surface survey frequency in snack bars and vending machine areas in the unrestricted areas; and the performance of repairs and modifications to radioactive and contaminated components.
November 8, 1988	Amendment No. 6, was issued at the convenience of the Commission to supersede Amendment No. 4. and clarify the status of Building C.
December 15, 1988	Amendment No.7, authorized the release of the soil which was removed from beneath Building C to an unrestricted area.
May 17, 1989	Amendment No. 8, incorporated the ground water monitoring program.
August 8, 1989	Amendment No. 9, to incorporate organizational changes.
August 6, 1990	Amendment No. 10, to incorporate a change in the corporate ownership.
October 6, 1992	Amendment No. 11, to incorporate organizational changes.
May 10, 1993	Amendment No. 12, to incorporate changes to our Radiological Contingency Plan.

June 9, 1993 — Amendment No. 13, to incorporate minor changes to our radiation safety program and administrative procedures.

18.1.5 CHANGES IN PROCEDURES, FACILITIES, AND EQUIPMENT

18.1.5.1 Changes in Procedures - Changes, revisions, or additions to the LTC's Area Operating Procedures are submitted to the Group/Section Manager. The Group/Section Manager is responsible for assuring that changes are reviewed as required on the procedure form. Area Operating Procedures (AOP) must be reviewed and approved by Nuclear Criticality Safety, Health Physics, and Industrial Health and Safety.

18.1.5.2 Changes in Facilities - Changes and modifications to buildings, exhaust ventilation systems, emergency electrical systems, and other facilities and services that may affect the safe handling of licensed material are requested on the Facilities Work Order Form (Figure 18.1-2). These forms are first submitted to the Facilities Engineering Supervisor. He shall determine if the requested change involves a "Facility Change." If a facility change is involved, the work order is forwarded to Health Physics and Industrial Health and Safety for their review and approval to assure that all safety and licensing considerations have been addressed.

Completed forms are kept on file by the Facilities Engineering Supervisor and are audited monthly by the Health Physics Group.

18.1.5.3 Changes in Equipment - Changes or modifications in equipment which is used to handle licensed material are requested on the Facility Work Order Form. This form is submitted to the Facilities Engineering Supervisor who determines that the change or modification involves a piece of equipment involved in licensed work. If he determines that the change does involve such equipment, he forwards the form to Health Physics and Industrial Health and Safety. The Health Physics and Industrial Health and Safety Officer exercises the same responsibilities as those described in Section 18.1.5.2, above.

SECTION V, CHAPTER 18.2

TABLE OF APPLICABLE PAGES

Page	Date
18.2-i	08/05/94
18.2-1	08/05/94
18.2-2	08/05/94
18.2-3	08/05/94
18.2-4	08/05/94
18.2-5	08/05/94
18.2-6	08/05/94
18.2-7	08/05/94
18.2-8	08/05/94
18.2-9	08/05/94
18.2-10	08/05/94
18.2-11	08/05/94
18.2-12	08/05/94
18.2-13	08/05/94
18.2-14	08/05/94

CHAPTER 18.2

FACILITY DESCRIPTION

TABLE OF CONTENTS

List of Figures

18.2 FACILITY DESCRIPTION

18.2.1 PLANT LAYOUT

Figures 18.2-1 through 18.2-3 show the layout of the Lynchburg Technology Center (LTC) buildings and Building B, which is the main location handling licensed materials at the LTC.

18.2.2 UTILITIES INCLUDING EMERGENCY POWER

18.2.2.1 Potable Water

Potable water is provided to the LTC by the NNFD. It is pumped from wells. It is stored and treated at the NNFD and is gravity fed to the site.

18.2.2.2 Process Water

Process water is provided to the LTC by the NNFD. The source of process water is the James River. It is pumped from the river, filtered by the NNFD and is gravity fed to the LTC. There is a storage capacity of 6,000,000 gallons on the entire Mt. Athos site. Process water is also used for fire fighting.

18.2.2.3 Gas

Natural gas is supplied at the site via pipeline which enters the property on the western side of the LTC. Natural gas is used for space heating, fuel for emergency engines, and laboratory uses.

18.2.2.4 Fuel Oil

Fuel oil is available for space heating to provide a backup source in the event of curtailed availability of natural gas. Fuel oil is purchased locally and stored at the LTC. There is a storage capacity of approximately 8,800 gallons.

18.2.2.5 Electricity

Electricity is furnished to the LTC from a substation located on the west side of the Mt Athos site. This source provides the normal source of power for the stack fans, hot cell fans, criticality monitors, emergency evacuation alarm, and lighting. When normal site power is lost, emergency sources are provided for these loads in the following manner.

18.2.2.5.1 The Onan emergency generator will start automatically on loss of LTC power and provides emergency power to one Hot Cell exhaust fan, the stack monitoring system, and emergency lighting. This engine/ generator is fueled by natural gas. The Onan system is tested weekly.

18.2.2.5.2 An Invertastat provides an uninterruptable source of electrical power for the criticality monitors, emergency evacuation alarm, and the alarm panel in Building B. The invertastat operates, upon loss of normal LTC power, from a battery, with sufficient capacity to provide its loads with power for up to 12 hours. The invertastat and battery are tested weekly.

18.2.2.5.3 The stack fan is provided with a backup engine which will start automatically upon loss of normal LTC power. The fan supplies off gas for Building B. The engine is fueled with natural gas and is tested weekly.

18.2.3 <u>HEATING, VENTILATION, AND AIR CONDITIONING</u>

18.2.3.1 Heating - Space heating in Building B is provided by gas fired boilers. The room air in Building B is partially recirculated.

18.2.3.2 Ventilation

18.2.3.2.1 Ventilation, Building B - The overall ventilation system for Building B has been designed so that the flow of air is toward the areas of highest potential airborne contamination; the lowest pressure being within the hot cell. All processing of licensed material in a form that could result in harmful airborne contamination or could generally contaminate personnel, equipment, or buildings, is carried out in the hot cells, glove boxes, and fume hoods. The minimum air flow rate through the hood opening is maintained at 100 linear feet per minute or greater. Fume hoods and glove boxes for use with licensed material are provided with a HEPA filter, except for those specified for use with perchloric acid, in which case they exhaust directly to the atmosphere. Fume hoods and glove boxes used in the handling of unirradiated Pu must exhaust through two HEPA filters connected in series. Final HEPA filters are provided with differential pressure indication and filters are changed when this differential pressure reaches 4-inches of water. When handling Post Accident Samples (PAS), the hoods may also be equipped with charcoal filters.

18.2.3.2.2 Hot Cells - The hot cells are provided with off-gas that is passed through two stages of HEPA filters in series. The ducting between the hot cells and the HEPA filters is constructed of steel. There is a fire damper positioned upstream of the filters that is operated by a fusible link that will close the damper in the event of fire and protect the filter. The cells are provided with two off-gas fans that operate in "and/or" modes. This provision permits increased off-gas when the cells are opened or maintenance of the off-gas system when one fan requires maintenance or fails. The off-gas exhausts to the stack fan. The off-gas system provides sufficient capacity to maintain an air flow of 100 linear feet per minute through an opening. The minimum differential pressure across the hot cell face is 0.25 inches of water. The system was designed so that either the roof slab or personnel door can be opened and a differential pressure of 0.6 inch of water would be maintained across the cell face. The opening of either of these major openings is controlled so that both are not opened at the same time. The volume of air removed is sufficient to remove the heat generated by lighting plus 7.5 kW from other sources and to maintain ambient temperatures below 120 degrees F during normal operations.

18.2.4 WASTE HANDLING

18.2.4.1 Liquid Wastes

18.2.4.1.1 Liquid wastes that are potentially contaminated are piped to the Liquid Waste Disposal Facility (LWDF). Wastes generated in the North-east end of Building B, which includes the Hot Cell Area, Radiochemistry Laboratory, Failure Analysis Laboratory, the Primary Equipment Cell (PEC), and the Pool Test Facility, collect in a large tank located in the PEC and are pumped to the LWDF. The remainder of the building is drained by gravity to the LWDF.

The LWDF is a below grade tank farm. Potentially contaminated waste water is piped from material handling areas to specific tanks so that the type of activity in any given tank can be anticipated. Each tank is provided with piping for thorough mixing by air spurge. The drain lines from each tank are located in the bottom of the tank to permit complete emptying. Exceptions to this are the two 4000 gallon waste tanks which have floating drains. The tanks are constructed of concrete with a water-proofing material, steel, stainless steel, carbon steel treated with an epoxy lining.

18.2.4.1.2 Liquids with high concentrations of radioactive materials are solidified and disposed of as solid wastes.

18.2.4.2 Solid Wastes

18.2.4.2.1 Solid wastes are generated 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 that are generated in the Hot Cells are not compacted.

18.2.4.2.2 Solid radioactive waste is stored, awaiting shipment for off-site disposal, in Building J, the Building J Annex, the High Level Waste Storage Tubes, the Temporary Storage Facility, and the Outside Storage Area. Waste is stored in closed containers suitable for off-site 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.

18.2.4.2.3 Building J provides 1400 square feet of storage space. The building is equipped with a smoke detector, 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. Containment of stored waste is assured by monthly smear sampling of the Building J floor.

18.2.4.2.4 The Outside 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. Containment of stored waste is assured by a quarterly visual inspection by the Health Physics Group.

18.2.4.2.5 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 Physics Group. Waste stored in these tubes is limited to that which is produced in the Hot Cells and must be in closed metal containers.

18.2.4.2.6 The Temporary Storage Facility is located adjacent to the Retention Basin, within the fenced and locked restricted area. This facility consists of an in-ground array of eight vertical, concrete "silos" arranged in two rows of four. The silos rest on a common concrete pad and are surrounded at the top by a concrete apron that promotes rain water runoff. Each silo is equipped with a 24-inch thick concrete lid. The lid thickness is based on a radiation level of 5 R/hr from the column of drums within a silo. The foundation drain system, at the base, is piped to a collection tank to permit periodic sampling. Each silo is equipped with a stainless steel drain pipe which leads to a common sampling pit where individual silo samples can be obtained. The waste that is stored in this facility is in long-lived containers (e.g., stainless steel or galvanized drums), that have been tested to withstand the maximum expected load of a column of drums.

18.2.5 FIRE PROTECTION

18.2.5.1 Codes and Standards - The development and building construction program of the site has taken place over the period 1955 to the present. For the main building handling licensed materials, Building B, the design and construction efforts took place from 1955 to 1969. There have been a number of alterations and use changes over the years, but generally these changes have not significantly altered the structural characteristics of the buildings.

The LTC buildings were built as staged or "added-on" phased construction. Building B was built in two stages and designed by Wiley & Wilson, a Lynchburg consulting engineering firm.

The physical layout of the 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 Building B 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 site 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.

The lack of a state-wide building code should not be taken as implication that the design and construction of the site facilities were accomplished on an inferior basis. On the contrary, where good accepted engineering and construction practices, coupled with stringent requirements from insuring companies such as Factory Mutual, are used as the main criteria for such facilities, the resulting structures usually far exceed the minimum requirements of various building codes. Such is typically the case for the building under consideration, when examined from the load capacity standpoint as specified in the Virginia building code, BOCA (Building Officials and Code Administrators).

Conventional construction materials are used throughout the building. Structural steel yield strength varies from 33,000 PSI (ASTM A7 steel) to 36,000 PSI (ASTM A36 steel). Concrete strength is 5000 PSI for the precast prestressed concrete elements found in Building B. Concrete reinforcing steel is typically ASTM A615, Grade 40 (40,000 PSI yield strength). Working stress design was used as the basis for concrete and steel design for all structures on site. Applicable design criteria used for the facility includes the standards of the American Concrete Institute (ACI), the Pre-stressed Concrete Institute (PCI), and the American Institute of Steel Construction (AISC). These various standards serve as both design and code basis for the respective types of construction, both at the time of original design as well as the present.

18.2.5.2 Insurance Inspection Reports - The site is inspected annually by the American Nuclear Insurers (ANI) Insurance Company. These reports have consistently found that the site meets the requirements in each of their inspection categories for a "satisfactory" rating. On a few occasions there have been recommendations that the site add fire protection equipment when the use of an area has been changed. Each such recommendation has been addressed at the site in a manner that has been found acceptable to the inspectors upon their reinspection.

18.2.5.3 Fire protection equipment is installed in accordance with recommendations made by the Industrial Safety Officer, the Corporate Fire Protection Engineer, or the insurance underwriters. Installed systems are approved and inspected by Factory Mutual Engineering Association. Routine inspection and maintenance is described below:

EQUIPMENT	MAINTENANCE	RESPONSIBILITY
Portable fire extinguishers	Inspection/ test	Industrial Health and Safety
Fire hoses	Inspection/ test	Industrial Health and Safety
Sprinklers	Test	Facilities Engineering
Fire suppres. systems (Halon)	Inspection	Facilities Engineering
Housekeeping	Inspection	Industrial Health and Safety
Emergency equipment	Inspection	Industrial Health and Safety

18.2.5.4 Combustible Waste Storage - Combustibles are not routinely stored at the site except; when work requiring such materials is in progress, in containers for shipping and receiving, or in sprinkled areas. Combustible wastes are discarded in metal containers and disposed of by an off site disposal firm. Contaminated combustible waste is discarded in metal containers and shipped to an off-site licensed disposal facility.

FIGURE 18.1-2

Lynchburg Technology Center

RL-229
8/94

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 Health & Safety ______________________

Health Physics Supervisor: ______________________

Space Below This Line for Plant Engineering Use Only

Order Received Date: ______________________ Signed: ______________________

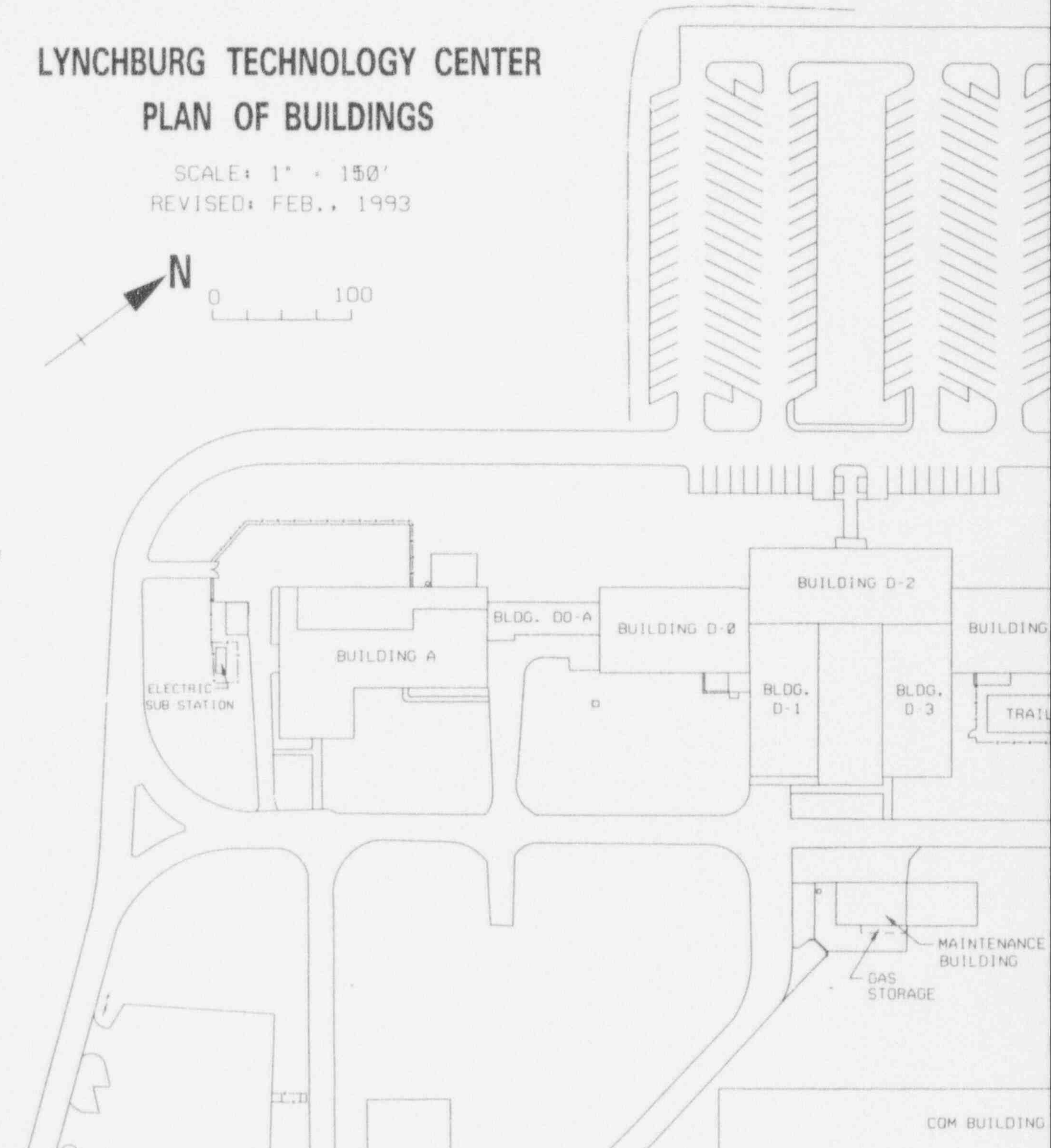
Planned Starting Date: ______________________ Planned Completion Date: ______________________

Order Completed: ______________________ Work Order Number

Date: ____________ Signature: ______________________

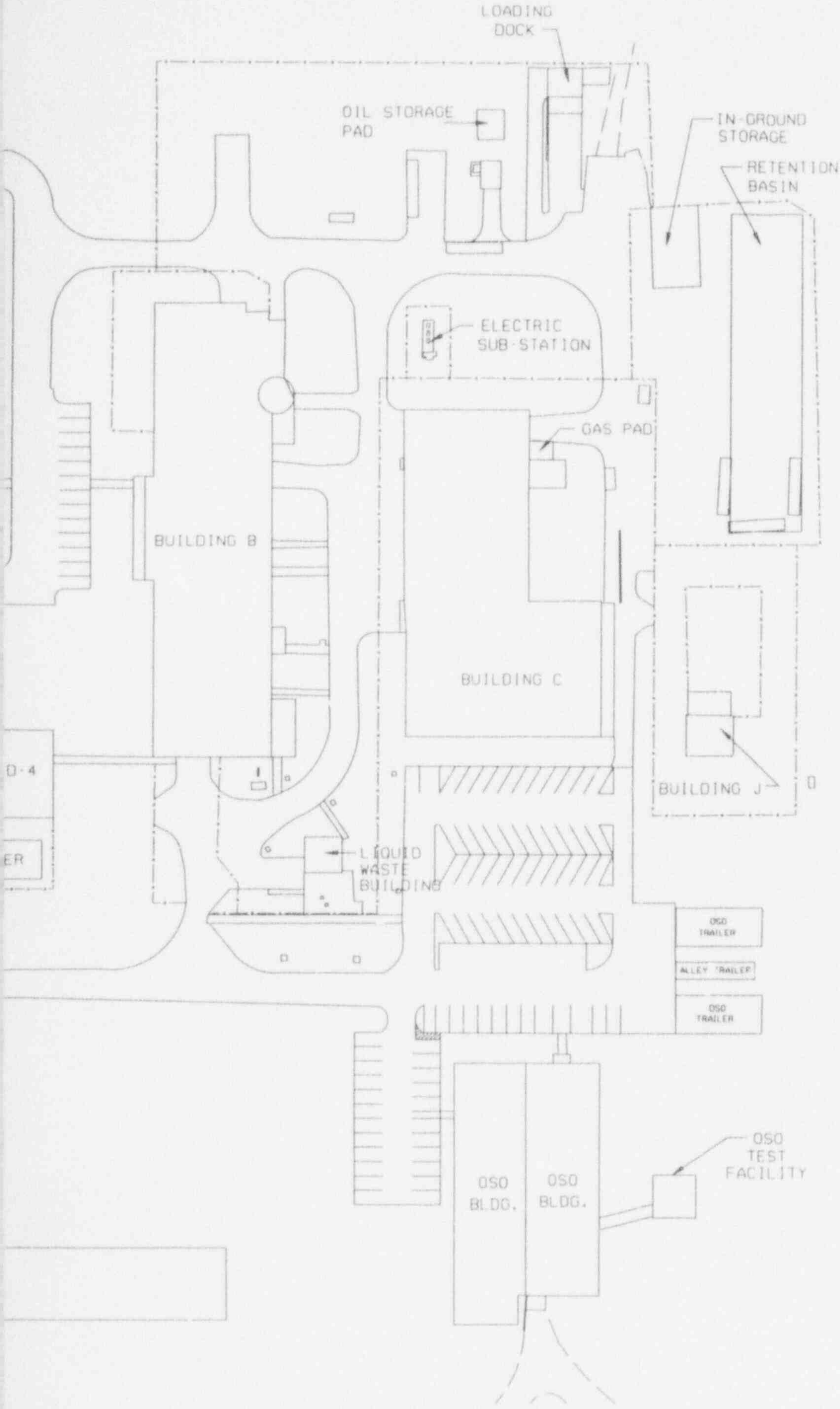
LYNCHBURG TECHNOLOGY CENTER
PLAN OF BUILDINGS

SCALE: 1" = 150'
REVISED: FEB., 1993



Date: 08/05/94

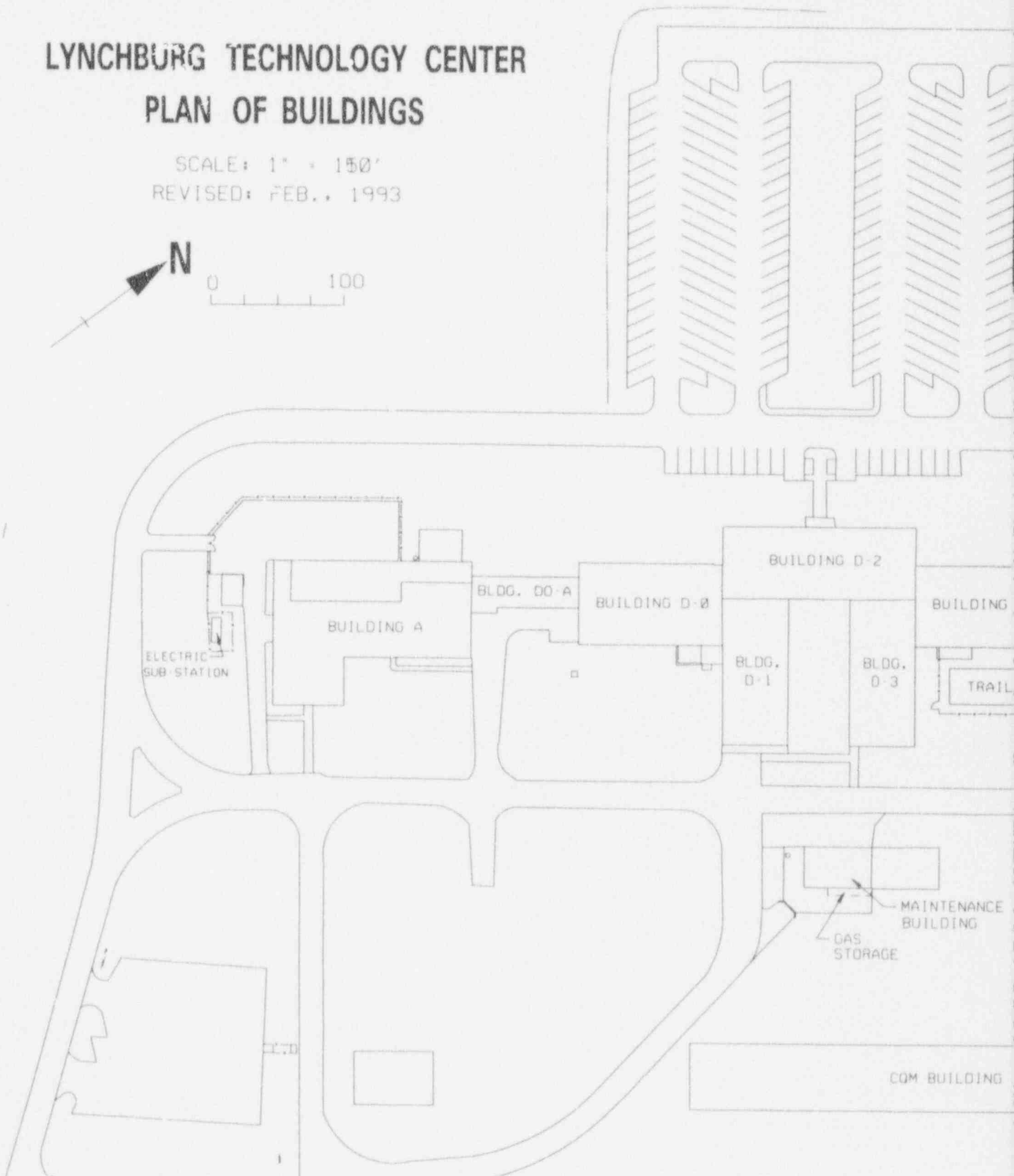
FIGURE 18.1-1



9408180208-01

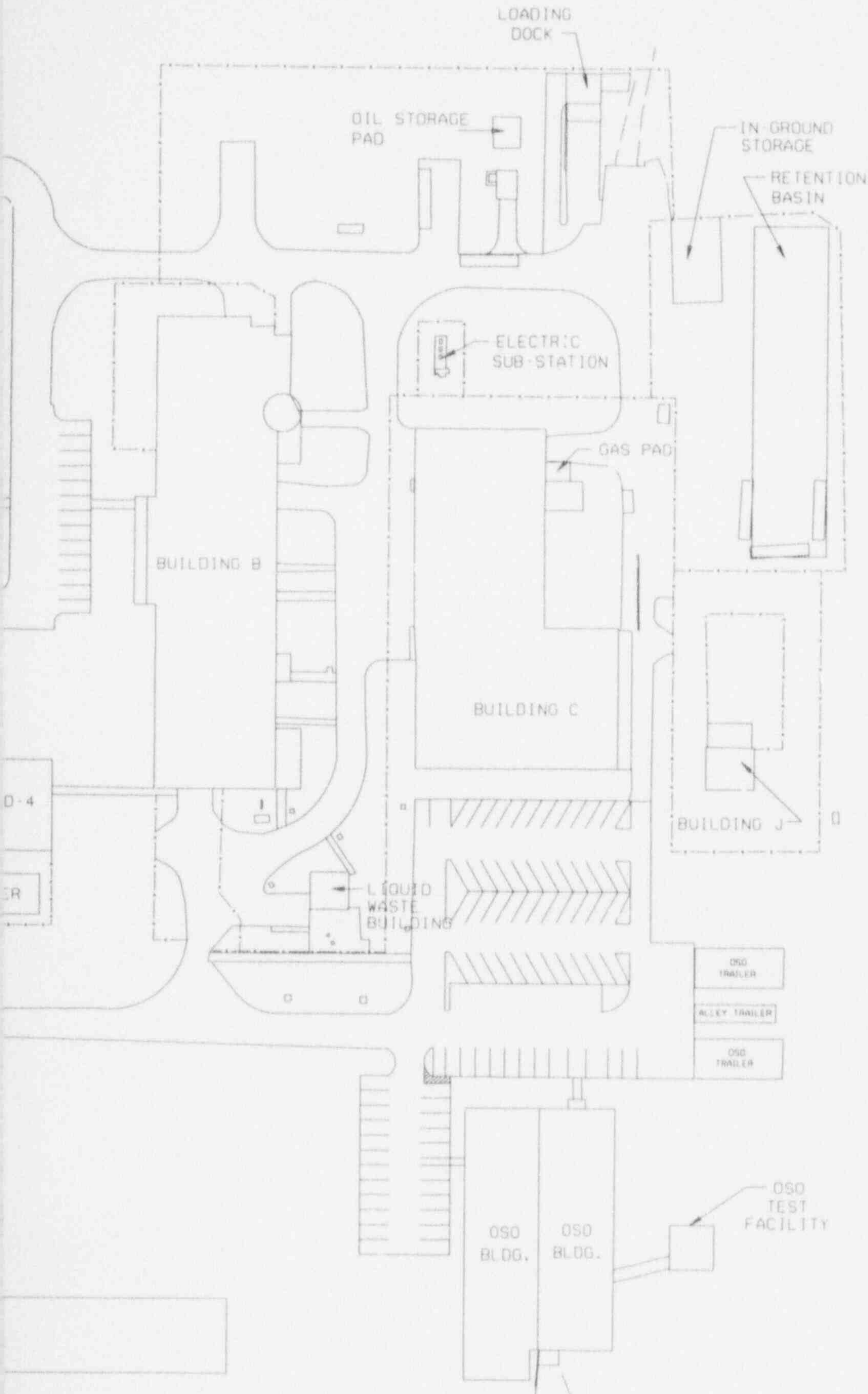
LYNCHBURG TECHNOLOGY CENTER
PLAN OF BUILDINGS

SCALE: 1" = 150'
REVISED: FEB., 1993

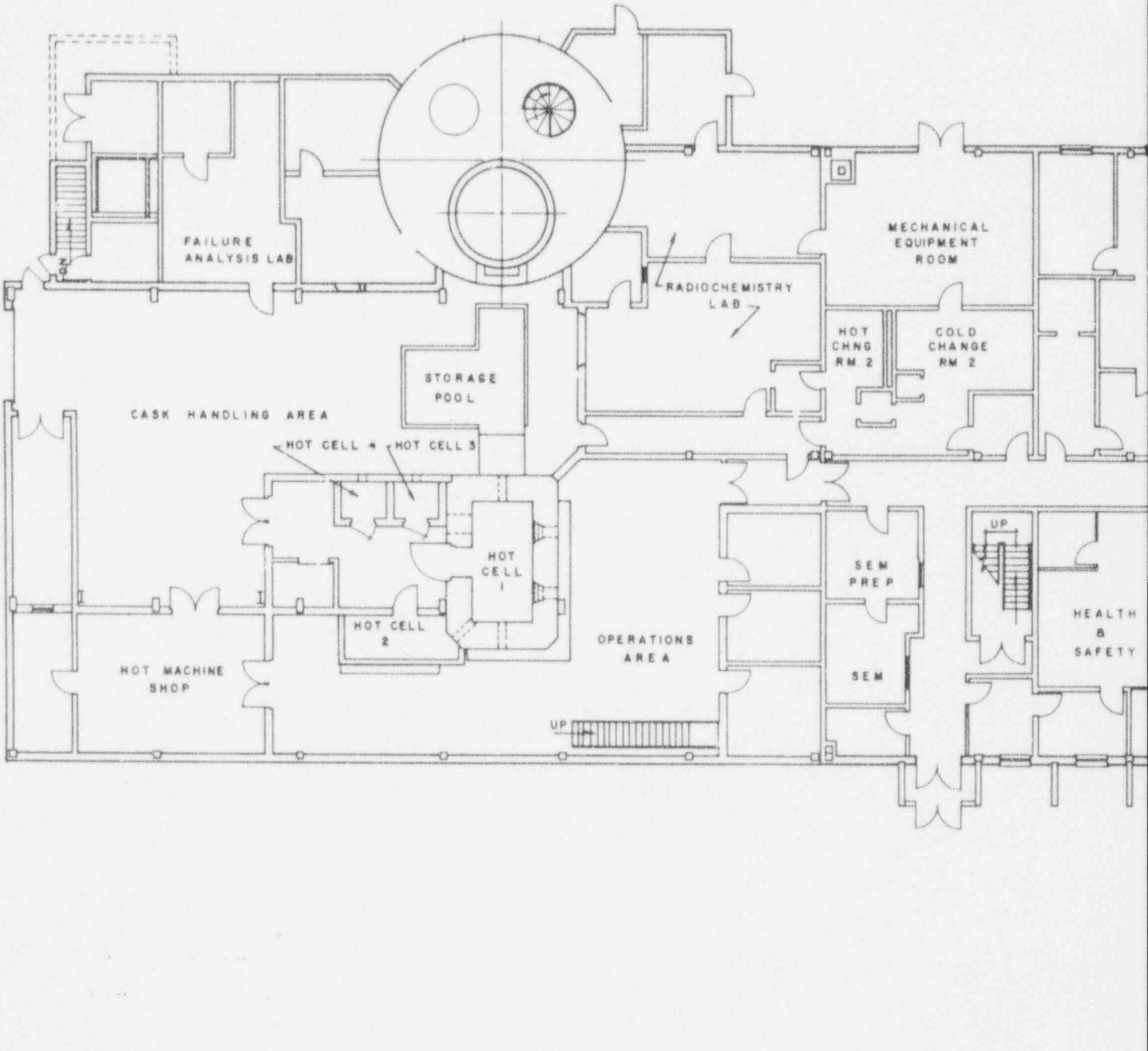


Date: 08/05/94

FIGURE 18.2-1

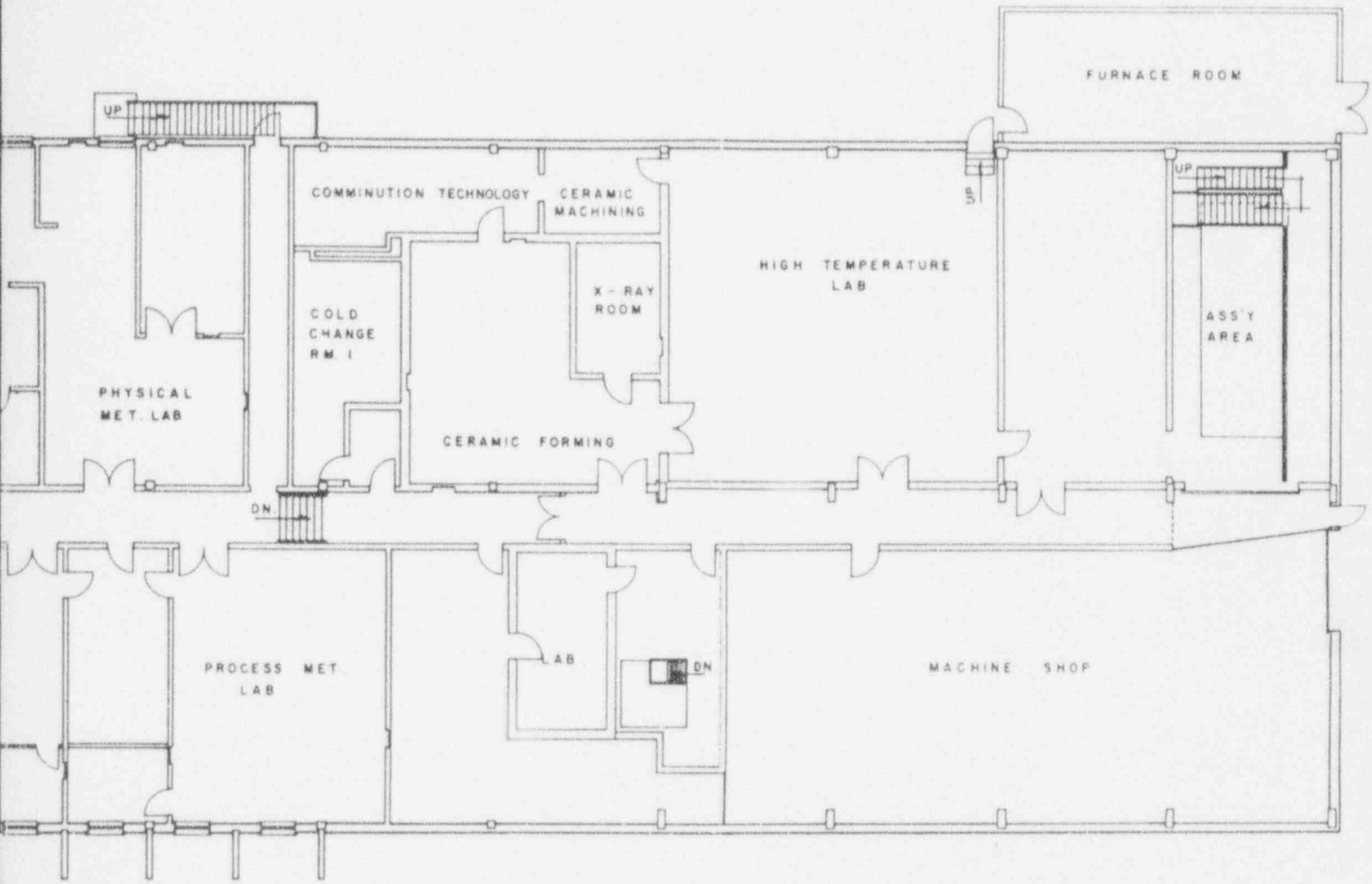


9408180208-02


FAILURE ANALYSIS LAB
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
OPERATIONS AREA
HOT MACHINE SHOP
SEM PREP
SEM
HEALTH & SAFETY
UP
UP

Date: 08/05/94

FIGURE 18.2-2

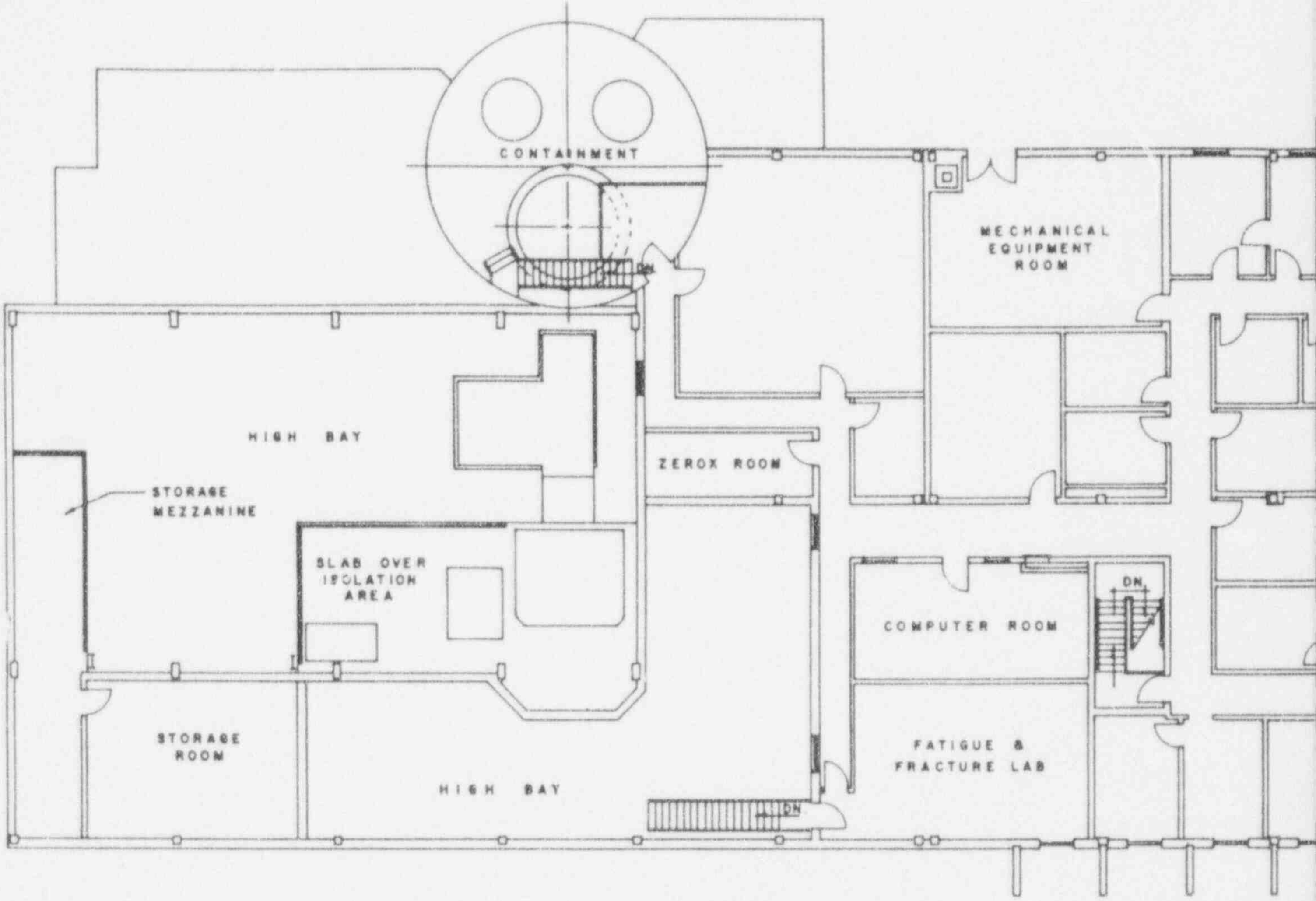


BUILDING "B"

FIRST FLOOR PLAN

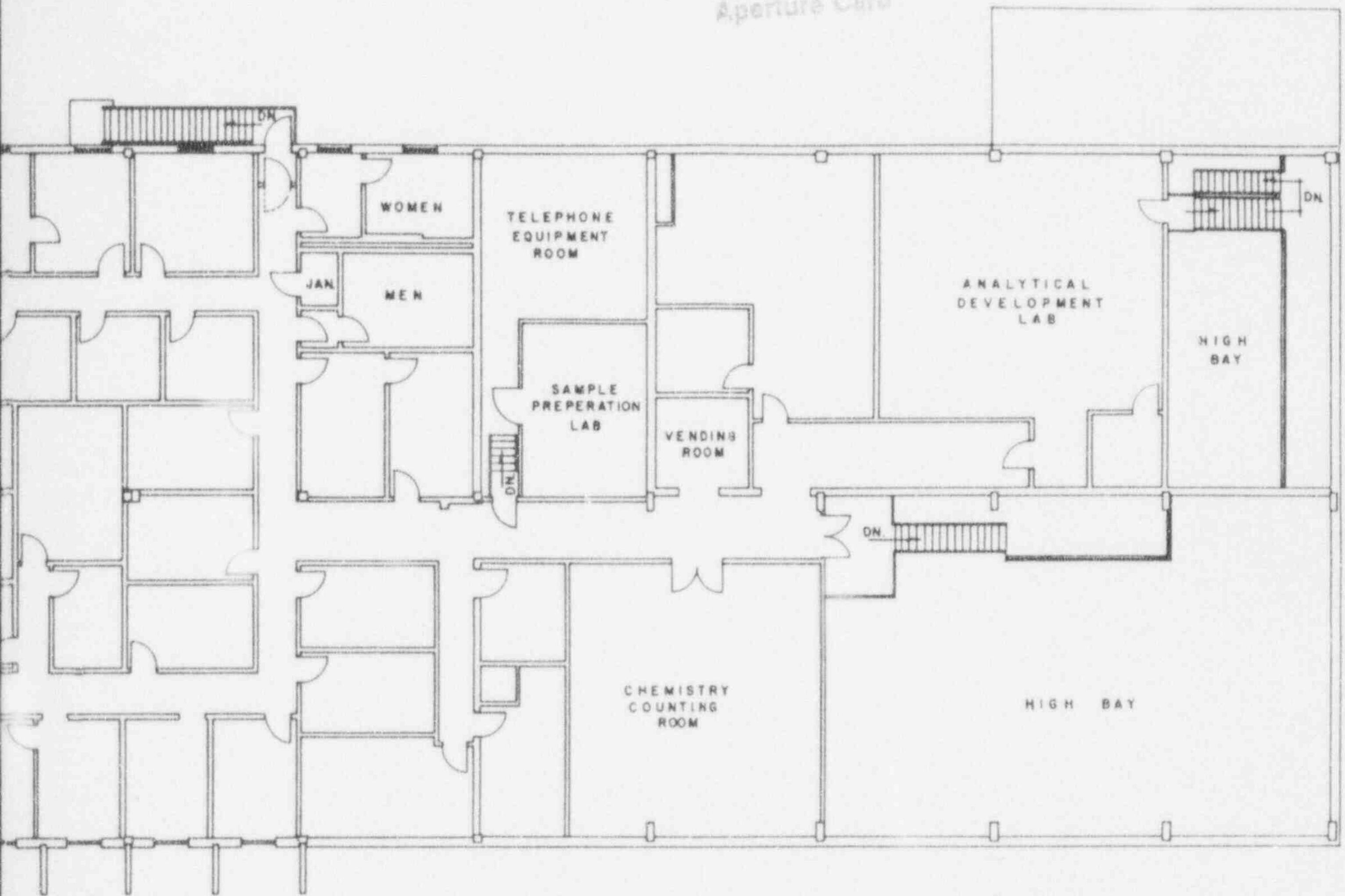
SCALE: 1" = 20'

9408180208-03


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

Date: 08/05/94

FIGURE 18.2-3



BUILDING "B"

SECOND FLOOR PLAN

SCALE : 1" = 20''

9408180208-04

SECTION V, CHAPTER 18.3

TABLE OF CONTENTS

CHAPTER 18.3

NUCLEAR CRITICALITY SAFETY

TABLE OF CONTENTS

List of Figures

List of Tables

18.3 NUCLEAR CRITICALITY SAFETY

18.3.1 ADMINISTRATIVE AND TECHNICAL PROCEDURES

The ultimate responsibility for nuclear criticality safety rests with the NNFD Management. This responsibility has been delegated to the Manager, Nuclear Criticality Safety. However, these responsibilities do not relieve supervisors of their responsibility for insuring that all operations under their control are conducted in a safe manner according to approved procedures.

The Manager, Nuclear Criticality Safety is responsible for establishing the nuclear criticality safety limits for the Lynchburg Technology Center (LTC), assessing potential changes to these limits, ensuring the validity of assumptions, and the accuracy of results. Educational programs may be developed and implemented if and when the Manager, Nuclear Criticality Safety deems them necessary.

Once a quarter the Manager, Nuclear Criticality Safety or qualified person designated by him will inspect selected LTC activities where special nuclear materials are being processed. Other SNM handling areas may be inspected less frequently; however, all areas shall be inspected at least once every two quarters. He shall consider area operations when scheduling these inspections and shall, if necessary, schedule his inspection at more frequent intervals. His consideration should include inspection of new facilities, inspection of hazardous non-routine operations, an audit of nuclear criticality safety records, a check for area posting and a review of current practices.

A written report will be filed with the Manager, Radiation and Criticality Safety, appropriate area management, and the NNFD General Manager. The report shall be brief, concerning itself with any inspections made and nuclear criticality safety activity during the specified time period. The following information is to be included:

- Areas visited
- Operations observed
- Unsafe practices or situations noted
- Nuclear safety activity of the quarter (brief summary)
- Recommendations
- Resolution of previous recommendations.

18.3.2 PREFERRED APPROACH TO DESIGN

Research and Development activities are performed at the LTC. While the use of favorable geometry is the preferred approach in a production facility, it is not appropriate nor practical at a research laboratory. Since most projects require only small amounts of SNM on laboratory benches and hoods, the preferred approach is through safe masses in simple arrays; the lattice density model or arrays found in TID-7016, Rev. 1 is the adopted model. The one exception to use of safe masses is when examining and testing reactor fuel assemblies. The approach to such uses is to accept only a limited number of fuel assemblies and then to maintain the fuel of an assembly within the dimensional envelope of the original assembly's dimensions. Where this is not possible, the fuel of an assembly is handled within the dimensions of favorable geometry or as a safe mass.

18.3.3 BASIC ASSUMPTIONS

This section describes basic assumptions and evaluations that have been made to demonstrate nuclear criticality safety for the specifications of Section III, Chapter 2 (Nuclear Criticality Safety Criteria).

18.3.3.1 Nuclear Isolation - Special nuclear material is isolated from all other special nuclear material for nuclear criticality safety purposes if any of

$$\frac{grams\ Pu\ fissile}{220} + \frac{grams\ U\text{-}235}{350} < 1$$

2. The values for U-233 - Pu and U-233 - U-235 mixtures were found by taking the lowest limit of any isotope in the mixture.

All computer calculations were made using either the NULIF code for fully reflected spheres or with the Monte Carlo code KENO. Four series of computer calculations were made. Tables 18.3-1 and 18.3-2 summarize the results.

18.3-1 Table for determination of K_{eff} for the mass limits listed in TID-7016 for lattice density model at the upper H/X limit (made with NULIF).

18.3-2 Table for determination of K_{eff} Vs H/X for 349 grams of U-235 contained in fully enriched uranium metal (made with NULIF).

The K_{eff} values for the lattice density limits ranged from 0.800 to 0.854 and are tabulated in the following table.

2. The values for U-233 - Pu and U-233 - U-235 mixtures were found by taking the lowest limit of any isotope in the mixture.

All computer calculations were made using either the NULIF code for fully reflected spheres or with the Monte Carlo code KENO. Four series of computer calculations were made. Tables 18.3-1 and 18.3-2 summarize the results.

18.3-1 Table for determination of K_{eff} for the mass limits listed in TID-7016 for lattice density model at the upper H/X limit (made with NULIF).

18.3-2 Table for determination of K_{eff} Vs H/X for 349 grams of U-235 contained in fully enriched uranium metal (made with NULIF).

The K_{eff} values for the lattice density limits ranged from 0.800 to 0.854 and are tabulated in the following table.

TABLE 18.3-1

K_{eff} TID-7016

Mass, Kg U-235	Sphere Radius, cm	H/U-235	H/Total U		
10.0	6.828	2.0	1.87	1.86	0.800
9.0	7.182	3.0	2.81	1.84	0.804
7.3	7.562	5.0	4.68	1.82	0.803
5.2	8.178	10.0	9.36	1.81	0.815
3.6	8.922	20.0	18.71	1.84	0.854

The K_{eff} values for the present limit (349 was used instead of 350) Vs H/X are given in the following table.

TABLE 18.3-2

K_{eff} FOR PRESENT LIMIT

Mass, g U-235	Sphere Radius, cm	H/U-235	H/Total U		
349	13.32	736.8	689	1.49	0.780
349	9.82	293.8	275	1.76	0.753
349	7.79	146.2	137	1.86	0.671

By analogy, values for Pu would be similar.

3. The unit and its limit (laboratory, furnace, transfer cart, etc.) are established by the Manager, Nuclear Criticality Safety, who authorizes posting the limit showing the maximum quantity of plutonium, U-233, and U-235 allowed. The fissile material content of the material transferred to or from a unit is established from process records, analyses, or previous analytical data. Only authorized users of SNM may transfer SNM between units and must do so only according to approved procedures. A board, sign, or other acceptable device is used to record the new balance and compares to balance with the unit limit.

18.3.3.2.3 Hot Cell - The demonstration for the units and array is identical to that of 18.3.3.2.1 and 18.3.3.2.2. The individual hot cells are isolated from all other arrays by a minimum of 2 feet of high density concrete.

18.3.3.2.4 Underwater Storage - Transfer Canal - Underwater aluminum or stainless steel storage racks are constructed to ensure 12-inch edge-to-edge spacing of each unit. Units are limited to those in Section III, Chapter 2, §2.8, excluding PWR fuel assemblies and, since they are separated by 12 inches of water, units are considered isolated. Therefore, any number of these units may be used.

Racks and fixtures are constructed with sufficient integrity and strength to withstand reasonable structural deformity, thereby providing the spacing previously outlined. Supervisor approval is required for removing or inserting any subcritical unit out of or into its storage rack.

There is no credible way in which water can be lost from the storage pool and transfer canal. However, assuming loss of water, stored units would drain and be unmoderated and sub-critical.

18.3.3.2.5 Underground Storage Tubes - Underground storage tubes are 5 inches in diameter, approximately 10 feet long, and on 17 inch centers (minimum) in a straight line. Material stored is first placed in a storage can with an inside diameter of 4-1/2 inches. Maximum units demonstrated safe in Section 18.3.3.2.2 are stored one per tube. These are nuclearly isolated from each other by 12 inches of concrete (minimum). The average edge-to-edge separation approximates 13 inches of concrete.

18.3.3.2.6 Power Reactor Fuel Assemblies

18.3.3.2.6.1 General - The LTC will receive and examine PWR fuel assemblies for both nondestructive and destructive examination. Irradiated assemblies will have been subjected to a reactor environment. From a nuclear criticality safety viewpoint, these assemblies are in their most reactive state when fresh or unirradiated. Therefore, nuclear safety is demonstrated by appropriate evaluation of the unirradiated assembly. The current plans call for examination of B&W-manufactured fuel assemblies from B&W power reactors. The current models of interest are designated as the Mark B and Mark C can-less assembly. The Mark B assembly was described in an earlier version of the SNM license for B&W's Commercial Nuclear Fuel Plant (SNM License No. 1168, Docket 70-1201). The K_{eff} of the unrodded and fully moderated and reflected assembly was shown to be 0.92 at maximum enrichment. Maximum enrichment is defined as 4.0 percent nominal which could go to 4.05 percent in manufacturing. Table 18.3-5 shows a comparison of the Mark C and Mark B assemblies. The K_{eff} of the Mark C assembly under the same conditions listed above has a value of 0.92. The reactivity as well as the spectral and physics kinetics of these assemblies are essentially the same. All of the nuclear safety calculations shown in this section were made with the Mark B assembly model (except Tables 18.3-5 and 18.3-6). Results were obtained for a fully reflected infinite array 12 inch edge-to-edge of maximumly enriched assemblies that were fully moderated, i.e., under water. The Mark B and C assemblies are to be disassembled in air only in an unirradiated state. The similarity in nuclear characteristics and the large decrease in reactivity in air-moderated assemblies ensure nuclear safety during the disassembly operations. Conditions given in Section III, Chapter 2, § 2.6.8, are sufficient to ensure that these two assembly types are indeed those to be examined. A damaged assembly which is restrained to 8.6 inches on a side will be no more reactive in air or water even if part of the fuel is missing; this will be demonstrated in Section 18.3.3.2.6.3.

TABLE 18.3-3

COMPARISON OF THE MARK B AND MARK C FUEL ASSEMBLIES

	Mark B	Mark C
Fuel assembly array	15 x 15	17 x 17
Fuel assembly dimensions, in.	8.45 x 8.45	8.536 x 8.536
Control rod tubes per assembly	16	24
Instrument tube per assembly	1	1
Fuel rods per assembly	208	264
Fuel rod pitch, in.	0.568	0.501
Fuel active height, in.	144	143
Pellet OD, in.	0.370	0.324
Theoretical density, %	92.5	94.0
Enrichment, %	4.0	4.0
Fuel rod clad ID, in.	0.377	0.332
Fuel rod clad OD, in.	0.430	0.379
Fuel rod clad material	Zr-4	Zr-4
V_{water}/V_{fuel} in fuel rod cell	1.65	1.68
V_{water}/V_{fuel} in assembly with water completely filling control rod and instrument cells	1.90	1.98
K_{eff} of one assembly in H_2O	0.92	0.92

Since industry is continuing to improve its assemblies, the LTC may destructively examine other types of assemblies. The conditions given in Section III, Chapter 2, § 2.6.9 and 2.6.10, for additional evaluation are adequate to ensure nuclear safety for different assemblies.

Acceptance of BWR fuel assemblies for study is acceptable if the assemblies have a maximum enrichment of 4.05 wt% U-235 and have a cross sectional area not exceeding that of a 22.5 cm diameter cylinder. By reference to DP-1014 this is 90% of the minimum critical cylinder diameter for an infinitely long, water reflected, optimally moderated cylinder with four wt% enriched heterogeneous UO_2. This value is further supported by Figure 2 in ANSI/ANS 8.1-1983 and Figure 2.15 in TID-7016, Rev. 2.

18.3.3.2.6.2 Receipt and Storage

A. Unirradiated Assemblies - Unirradiated fuel assemblies may be stored in their shipping containers since their nuclear safety has been proven prior to their licensing. Assemblies that are unirradiated may also be stored in air if the distance between assemblies is no less than 21 by 38 inches. This distance assures criticality safety for less than 100 assemblies of either the Mark B and/or Mark C assembly types. This ensures the safety of the maximum of four assemblies stored at the LTC. Unirradiated assemblies may also be stored under water (hot cell pool, mock-up pool, or development test area pool).

Assemblies stored in air will be stored either:

1. Horizontally - on the floor or on tables constructed with sufficient integrity and strength to withstand reasonable structural deformity thus assuring the above mentioned spacing.

2. Vertically - in racks and fixtures constructed with sufficient integrity and strength to withstand reasonable structural deformity and assuring the above mentioned spacing.

Approval by the Manager, Nuclear Criticality Safety is required to move any other fissile material into the area where the assemblies are stored. No more than four unirradiated assemblies may be stored at once. The limit of four assemblies is an arbitrary limit which the LTC imposes upon itself and does not affect nuclear safety.

Partially disassembled unirradiated Mark B or Mark C assemblies may also be stored in air. This is safe due to the lower moderation characteristics of air compared to water. Air moderated values of K_{eff} will be less than those shown in Table 18.3-4.

Fuel rods from unirradiated, disassembled Mark B or Mark C assemblies will be stored in air in slabs not to exceed 4 inches in height (see Section 18.3.3.2.6.4, statement 2).

B. Irradiated Fuel Assemblies - Assemblies which have been irradiated may also be stored in their shipping containers or in the hot cell pool. Storage in the hot cell pool is limited to four irradiated assemblies. The limit of four assemblies in the pool is an arbitrary limit which the LTC imposes upon itself and does not affect nuclear safety since each fuel assembly or rod storage position is neutronically isolated from any other fissile material by a minimum of 1 foot of water.

Racks and fixtures in the pool are constructed with sufficient integrity and strength to withstand reasonable structural deformity, thereby providing the spacing previously outlined. The racks are also constructed to preclude inadvertently placing other fissile material closer than the 1-foot minimum spacing. Supervisor approval is required for removing or inserting fissile material into or out of any of the racks or fixtures. Storage of Mark B and Mark C fuel rods and partially dismantled assemblies into storage racks which restrain the size of each position to a square not exceeding the dimensions of a fresh fuel assembly, i.e., 8.6 inches, is safe based upon the analysis demonstrating safety of as assembly during dismantlement. Fuel rods may also be stored in an ever safe cross sectional area fixture, i.e., a cross sectional area not exceeding that of a 22.5 cm diameter cylinder.

18.3.3.2.6.3 Work Area Of Pool Under Hot Cell No. 1 - This area will be used to dismantle irradiated and unirradiated assemblies. Nuclear criticality safety for Mark B and

Mark C assemblies under varying stages of dismantlement has been demonstrated via use of NULIF and PDQ-07 physics computer codes.

Reactivity was calculated by PDQ (coefficients having been generated by NULIF) for a fully reflected and flooded, unrodded fresh assembly and for the same assembly under five conditions of dismantlement. The cases run with the number of rods removed in each case and the resulting K_{eff} is given in Table 18.3-4.

TABLE 18.3-4

REACTIVITY FOR MARK B AND MARK C FUEL ASSEMBLIES UNDER DISMANTLEMENT

Case No.	No. of Removed Rods	Calculated K_{eff} Mark B	Mark C
1	0	0.891	0.921
2	4	0.894	0.920
3	12	0.897	0.919
4	24	0.898	0.922
5	36	0.896	0.917
6	8	0.888	0.918

Reactivity was also calculated by KENO-IV using the 123-group XSDRN cross section set for a fresh assembly fully submerged in water under conditions of 24 rods removed and with all instrument and control rod guide tube positions loaded with fuel rods. The cases run with the number of rods removed or added and the resulting K-effectives are given in Table 18.3-5.

TABLE 8.3-5

K-EFFECTIVE OF INDIVIDUAL MARK B AND MARK C FUEL ASSEMBLIES

Change in No. Of Fuel Rods From Normal	K-effective ± 2 Mark B	Mark C
0	.895 ± .010	.900 ± .013
-24*	.900 ± .015	.906 ± .014
+17	.890 ± .015	
+25		.876 ± .016

*Same configurations as case 4 in Table 18.3-4.

Cases 2 through 5 represent removal of rods "uniformly" through the assembly while Case 6 represents the removal of eight rods clustered about the center. Rods removed are shown schematically in Figures 18.3-1 and 18.3-2 for Mark B and C assemblies, respectively. The calculations reported above demonstrate the nuclear safety of an assembly under various conditions of assembly and reloading. If any of the fuel rods inserted into the fuel assembly are further encased in metal tubing, the assembly would still be safe due to the tubing displacing moderator with absorber. A grouping of 75 fuel rods confined within a 8.6-inch square merely describes a dismantled assembly and is also safe. Fuel rods inserted into instrument and control rod guide tubes shall be held in place with a flat metal plate which shall be bolted to the top of the assembly.

The safety of withdrawing an assembly and its associated rod storage position partially into the cell is demonstrated safe by comparison to a series of KENO runs made for pool storage at a reactor site. To demonstrate the safety of flooding a reactor site storage pool filled with fresh Mark B fuel assemblies, an array of fuel assemblies 14 units wide, infinitely long, and reflected on the sides and bottom by concrete was calculated by KENO. Each assembly was spaced 1 foot from the other on the concrete reflector, as appropriate. Four cases at different degrees of pool flooding were evaluated and are described in Table 18.3-6.

TABLE 18.3-6

REACTIVITY FOR AN INFINITE BY 14-UNIT ARRAY OF FUEL ASSEMBLIES

Water Height	
Fully Flooded	0.951 ± 0.006
3/4	0.946 ± 0.007
1/2	0.928 ± 0.007
0 (dry)	0.506 ± 0.004

The confidence levels quoted above are one standard deviation. K_{eff} for the fully flooded condition is higher than that calculated by PDQ because of simplifications made in running the cases. The series of runs were to demonstrate safety of a partially flooded pool, a much more restrictive condition than partial withdrawal into one cell. The similarity in the Mark B and Mark C nuclear characteristics and the simplifying assumptions assure these calculations are also valid for the Mark C assembly type.

18.3.3.2.6.4 Assembly and Machine Shop and Development Test Areas - Assemblies of either Mark B or C disassembled in air are far less reactive than the cases listed in Table 18.3-4. Either assembly type may be disassembled in air. A safe reactivity level (<<0.95) is assured provided the handling in Section III, Chapter 2, § 2.6, is followed. The conditions stated in Section III, Chapter 2, § 2.6, are based on KENO calculations that show:

1. Two assemblies in air 21 inches or more apart are nuclearly safe.

2. Fuel pins at a maximum enrichment when optimumly moderated are fully reflected in an infinite slab have a K_{eff} = 0.95 if the slab is no more than 4 inches thick.

3. Fuel rods in any configuration or number, up to the number in the assembly, when limited to the confines of the assembly size are no more reactive than the intact assembly (Ref. Table 18.3-4).

18.3.3.2.6.5 Hot Cell Operations - Work within the hot cell will, by and large, follow existing controls. Three units in addition to an assembly and its associated rod storage position are permitted within Cell No. 1. Two of the three units are restricted to rods confined within an ever safe cross sectional area, i.e., a cross sectional area not exceeding that of a 22.5 cm diameter cylinder; in addition these two units must be free draining of any water. The third unit of Cell No. 1 under mass control is permitted. All other Hot Cells are limited to one unit each.

18.3.3.2.6.6 Fuel Rod Dismantlement - Fuel rods of either assembly type may be dismantled. Dismantlement can be performed in any area which present licensing conditions permit fuel handling. In addition, mass control must be limited to 350 grams of U-235, proper spacing must be maintained, and approved procedures must be followed.

18.3.3.2.6.7 Shipment and Disposal - The conditions of 18.3.3.2.6 are consistent with the above demonstration and/or current limits.

18.3.3.3 Outside Storage

18.3.3.3.1 General - The underground storage and shipments are nuclearly isolated by distance or matter.

18.3.3.3.2 Underground Storage - The underground storage tubes are 5 inches in diameter, approximately 20 feet long and 20-inch centers. Maximum units demonstrated safe in Section 18.3.3.2.2 are stored, one per tube. These are neutronically isolated from each other by 15 inches of concrete.

18.3.3.4 Dry Waste

Dry waste is waste that is free of liquids but not necessarily free of hydrogenous material. Liquid waste which cannot be put through the hot drain system is solidified before disposal. Nuclear Criticality Safety of dry waste is ensured by maintaining the concentration of SNM to a value much less than an ever safe concentration. Forty-five grams of SNM in a 30-gallon drum yields a concentration of less than 0.40 grams per liter. By comparison, TID-7016 (Rev. 2) reports the subcritical limit for fully enriched, optimumly moderated U-235 to be 11.5 grams/liter and for Pu-239 the value is 7.0 grams/liter. The waste, as borne out by more than 25 years of experience, will maintain an approximate uniform distribution within the containers. Containers with unirradiated fuel are gamma scanned before transfer to verify that the maximum amount of SNM permitted per container (45 grams) is not exceeded. Assurance that containers with irradiated fuel do not exceed the maximum permitted amount of SNM is based on a mass balance difference. The amount of SNM contained within irradiated fuel is specified by the customer when received. Irradiated material is transferred into and out of units and hot cells based on the received value. The difference between the amount of material transferred into a hot cell or unit and that transferred out of a hot cell or unit establishes the amount of SNM remaining in the hot cell or unit, in the form of contamination. Experience indicates that the contamination, which includes the SNM, is uniformly distributed in the waste. There is therefore no requirement in the number or arrangement of containers within the radioactive waste building or in the temporary storage facility. Even if the 45 grams of SNM were concentrated in the corners of each container, the maximum amount of SNM at the intersection of eight containers would be 360 grams of SNM and each 45 gram segment would be separated from the other by two thicknesses of steel. Therefore, this is of no concern.

18.3.4 ANALYTICAL METHODS AND VALIDATION REFERENCES

Nuclear criticality safety computer calculations presented in this chapter have used the computer codes NULIF, PDQ-07 and/or KENO. The physics codes NULIF and PDQ-07 are not only routinely used in nuclear criticality safety to evaluate highly moderated low-enriched systems but were also the standard codes used by the reactor design group of the Babcock & Wilcox Company (both codes were certified by the Company's Quality Assurance Program for reactor calculations). The Monte Carlo code KENO is state-of-the-art in industry for nuclear criticality safety evaluations. Future calculations for nuclear criticality safety will make use of these codes and the Nuclear Criticality Safety Codes in SCALE 3 (NITAWL-S, XSDRNPM-S, KENO-IVS and KENO-Va). SCALE 3 is described in NUREG/CR-0200. Before use of the SCALE 3 package, the proper wording of the various codes will be assured and appropriate benchmarking activity will be carried out.

18.3.5 DATA SOURCES

Data and Guidance for Nuclear Criticality Safety is taken from one or more of the sources specified below.

1. Calculations using methods described in Section 18.3.4.

2. "Nuclear Safety Guide, TID-7016, Revision 2," NUREG/CR-0095 (ORNL/NUREG/CSO-6), (June, 1978.)

3. "Nuclear Safety Guide, TID-7016, Revision 1," (1961). TID-7016, Revision 1 is used only for application of the lattices density method which is Table IV and Figure 22 on page 26. Table IV has been modified according to information published in the Federal Register, March 5, 1963 on page 2130.

4. American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Oxide Reactors, ANSI/ANS-8.1-1983).

5. H. K. Clark, "Critical and Safe Masses and Dimensions of Lattices of U and UO_2 Rods in Water" DP-1014, Savannah

River Laboratory (1966).

6. H. C. Paxton, "Criticality Control in Operations with Fissile Material," LA-3366(Rev), Los Alamos Scientific Laboratory, (1972).

7. R. D. Carter, et al, "Criticality Handbook," ARH-600 Revised last November 6, 1973, Atlantic Richfield Hanford Company.

18.3.6 FIXED POISONS

The LTC does not now use Fixed Poisons to maintain nuclear criticality safety.

18.3.7 STRUCTURAL INTEGRITY

Where structural integrity is necessary to provide assurance for nuclear criticality safety in any operation, the design and construction of those structures will be evaluated with due regard to load capacity and fore-seeable abnormal loads, accidents and deterioration. This engineering activity is the responsibility of the Manager, Industrial Engineering with review and approval by a qualified person.

18.3.8 SPECIAL CONTROLS

There are no special controls for nuclear criticality safety at the LTC.

FIGURE 18.3-1

FUEL ROD REMOVAL SCHEMATIC - MARK B

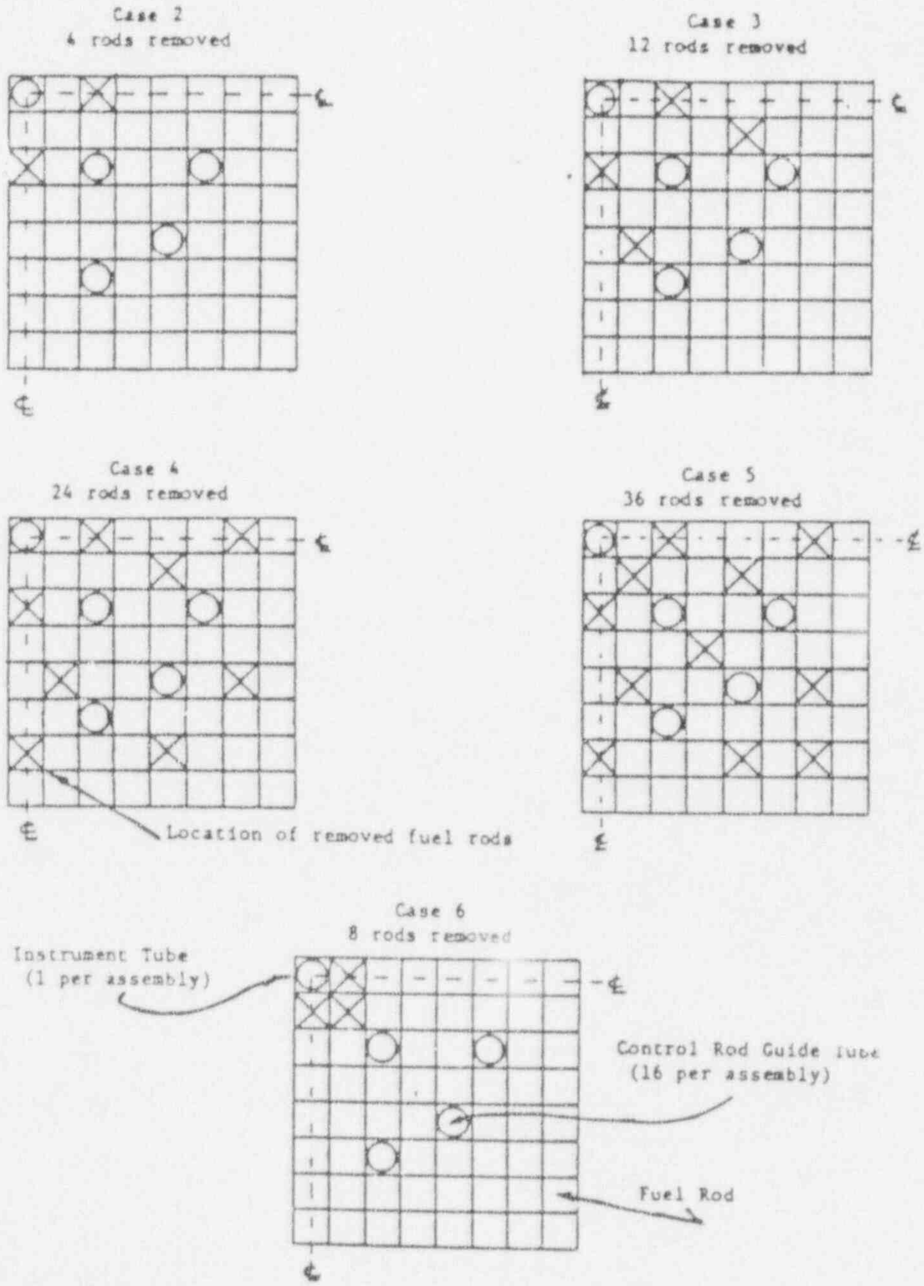
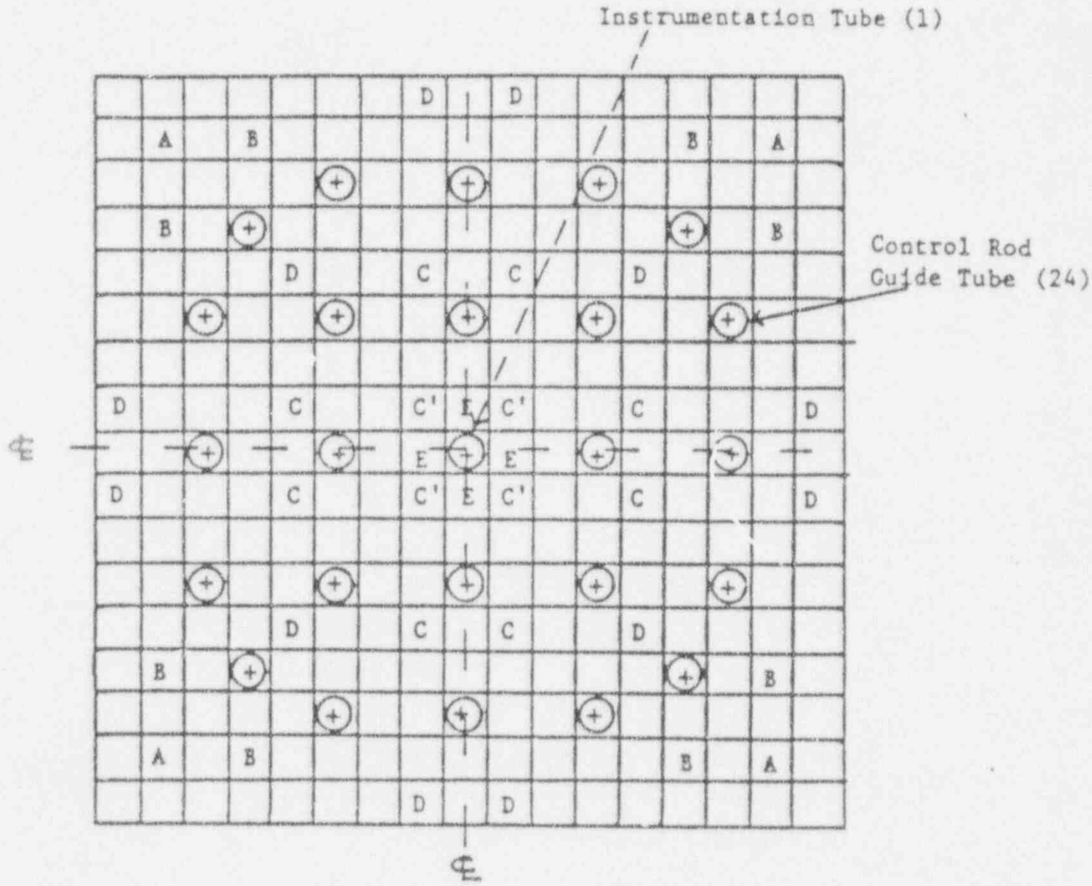


FIGURE 18.3-2

FUEL ROD REMOVAL SCHEMATIC - MARK C



MARK C FUEL ASSEMBLY (17 × 17)

Fuel Rods Removed for Reactivity Under Dismantlement
(Section 4.4.7.3)

No. of Rods Removed	Pins Shown By Letter
4	A
12	A, B
24	A, B, C, C'
36	A, B, C, C', D
8	C', E

SECTION V, CHAPTER 18.4

TABLE OF CONTENTS

CHAPTER 18.4

TABLE OF CONTENTS

18.4.1 PROCEDURES

18.4.1.1 Area Operating Procedures (AOP) - All operations with licensed material at the Lynchburg Technology Center (LTC) shall be conducted in accordance with Area Operating Procedures (AOP's) or a Radiation Work Permit. Area Operating Procedures are prepared by any technically competent person. The procedures are reviewed and approved by the Manager of the operation or affected area, Nuclear Criticality Safety, Industrial Health and Safety, and Health Physics. When the procedure is approved by all the required reviewers, the Supervisor, Health Physics will issue the AOP for implementation.

18.4.3.2 Technical Procedures - Technical procedures provide detailed technical standards and instructions for performing specific tasks. They are generally not intended for use by operations personnel for the safe handling of licensed materials and are not distributed in the same manner as AOP's.

Technical procedures for Health Physics and Nuclear Criticality Safety are reviewed and approved by a Health Physicist and the Nuclear Criticality Safety Specialist, respectively, or by their designated alternates. The distribution list for each procedure is specified in the procedure.

18.4.2 TRAINING

18.4.2.1 General Radiation Protection Training

The LTC provides three LTC specific training programs covering the nature, use and control of radiation, and radioactivity. These courses are presented to ensure that all LTC personnel receive training appropriate to their activities.

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 Supervisor, Health Physics or his designated alternate 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 1 is intended for LTC workers and non-LTC workers who will be authorized access to the restricted area. Program 2 is intended for LTC and non-LTC workers who may enter the restricted and controlled areas but who will not be permitted to work with licensed material at the LTC without supervision. Program 3 is intended for Authorized Users (those who will be authorized to work with licensed material and to supervise such work).

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

18.4.2.2 Program 1

This course is presented to LTC workers and non-LTC workers who will be granted access to the restricted area but who will not be granted unescorted access to the controlled areas. The course provides an introduction to radiation and radioactivity (understandable to a non-technical person) and a thorough coverage of safety rules and procedures, including the site emergency procedures. Subjects include the types of radiation, ALARA, radiation effects on humans, decontamination procedures, radiation exposure to females, warning signs, basic health physics rules, a history of radiation protection, worker's rights and responsibilities, and health physics terms.

18.4.2.3 Program 2

This course is presented to LTC workers and non-LTC workers who will be granted unescorted access to the restricted area and controlled areas but who will not be permitted to work with radioactive materials without supervision. This course is intended to provide the workers with a knowledge of the hazards of working in radiation and controlled areas and ways to minimize their dose. Subjects include types of radiation, radiation exposure limits, ALARA, personnel dosimetry and its use, dose calculation, biological effects, radiation exposure to females, radiation protection measures, warning signs and labels, radiation work permits, emergency procedures, rights and responsibilities of workers, and health physics terms.

18.4.2.4 Program 3

This course is presented to LTC workers and non-LTC workers who will be granted unescorted access to the restricted area and controlled areas and will be permitted to work with radioactive materials and supervise such work. This course is intended for meeting the requirements for designation of a worker as an authorized user. Subjects include fundamentals of radiation, external and internal radiation protection, biological effects, radiation detection, instrumentation, contamination control, license requirements, site organization, rights and responsibilities under 10 CFR 19, ALARA, dose calculation, personnel dosimetry requirements and use, posting and labeling, and health physics terms.

18.4.2.5 Respiratory Protection Training

Training in respiratory protection techniques will be required of LTC workers 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 workers using respiratory protection will be included in the training courses. Biennial retraining will be scheduled, at the discretion of the qualified individual, to ensure that a high degree of 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 allowable limits.

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.

SECTION V, CHAPTER 18.5

TABLE OF APPLICABLE PAGES

Page	Date
18.5-i	08/05/94
18.5-1	08/05/94
18.5-2	08/05/94
18.5-3	08/05/94
18.5-4	08/05/94
18.5-5	08/05/94
18.5-6	08/05/94
18.5-7	08/05/94
18.5-8	08/05/94
18.5-9	08/05/94
18.5-10	08/05/94

CHAPTER 18.5

TABLE OF CONTENTS

List of Figures

18.5 ENVIRONMENTAL SAFETY

18.5.1 ENVIRONMENTAL MONITORING

Environmental sampling of the area surrounding the Lynchburg Technology Center (LTC) 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.

- one continuous on-site boundary air sample (Figure 18.5-1)
- quarterly water samples from the James River collected above and below the liquid discharge point (Figure 18.5-2)
- continuous sampling of rain water on-site (Figure 18.5-1)
- quarterly samples of river silt and near-river vegetation (Figure 18.5-2).

Normally, site 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 is presented in the Annual ALARA Report.

18.5.2 EFFLUENT AIR MONITORING

Planned discharges of air to the environment shall be in compliance with the limits specified in 40 CFR 61. Compliance with 40 CFR 61 demonstrated by calculation that the total annual release limits permitted by the license will not exceed the specified 40 CFR 61 dose limits for persons located at the point of maximum ground level concentration.

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.

Discharges through the stack are monitored with a sampling head located in the stack about 25 feet above the base. Air removed by the sampler passes through a fixed filter, into the chamber of the gas monitor, and is returned to the stack. The fixed filter is monitored continuously for alpha and beta activity. The second monitor is a sealed proportional counter that monitors continuously for noble gases. The stack monitor has an adjustable flow rate and is set to sample the stack isokinetically. Both monitors are equipped with adjustable alarms. Set points for these alarms are determined by the Health Physics Group. These alarms are connected to an alarm panel located at the LTC in the Health Physics Laboratory in Building B.

Air from areas equipped with continuous air samplers (and which is below the applicable airborne concentration levels for an unrestricted area) may be exhausted, through HEPA filters, directly to the roof of the building. Air from areas which have a low potential for airborne activity may be exhausted directly to the roof of the building.

18.5.3 LIQUID EFFLUENT MONITORING

All potentially radioactive liquids at the LTC are collected in tanks located in the Liquid Waste Disposal Facility. The contents of each tank are mixed, samples are obtained, and are analyzed for radioactivity before the liquids are released to the NNFD waste treatment plant.

Water samples are also obtained on a quarterly basis from the retention basin located behind Building C, the sample pit and collection tank located adjacent to the Temporary Storage Facility, and the holding pond located near Building J.

18.5.4 GROUNDWATER MONITORING

18.5.4.1 Monitoring System

The groundwater monitoring system is installed in compliance with Condition No. 13, License SNM-778, as renewed on July 30, 1987. The purpose of the monitoring system is to monitor the groundwater under the Liquid Waste Disposal Facility (LWDF) for radioactivity that may be released as a result of tank leakage. There are seven wells roughly evenly positioned around the LWDF. The wells are drilled to depths that permit the sampling of the groundwater at the interface of the soil and bedrock. The depths of the wells are shown in Table 18.5-3.

TABLE 18.5-1

MONITOR WELL CONSTRUCTION DETAILS

Well No.[2]	Bottom[1] of Screen	Top[1] of Screen	Total[1] Depth
1	63	43	65
2	46	31	52
3	57	37	60
4	48	33	50
5	60	40	104
6	81.2	61.2	85
7	72	57	72

[1]Depth below grade in feet.

[2]Well locations are specified in Fig. 18.5-3.

18.5.4.2 Constructions Details

The wells are constructed of four-inch Schedule-40 PVC pipe and PVC screen. They are drilled using the air rotary technique to avoid the introduction of any contaminants or substances from the drilling mud. The screens are set inside a temporary casing and a sorted sand filter pack placed around the well screens. Above the well screens, bentonite pellets were installed to seal off the sand and then the wells were grouted to the surface. A protective well pad and lock system has been placed around each well.

The groundwater monitoring system is designed to provide an early warning of a leak, should one occur, from any of the tanks or piping located at the LWDF. To ensure this, the wells are roughly equally spaced around the perimeter of the LWDF and associated underground storage tanks. Groundwater flow is to the west. Wells MW-RL06 and MW-RL07 are located to the west and hydrologically down-gradient of the tanks.

The depth of the wells (close to the bedrock interface) were based on the following:

- Flow in the aquifer is primarily through fractures in the bedrock.
- The water table is near but not limited to the bedrock/alluvial sediment interface.
- No evidence was found of impermeable layers capable of forming perched water tables.

The monitoring wells were installed in general accord with EPA's RCRA Ground-Water Technical Enforcement Guidance Document (September 1986).

The initial analysis results from the November, 1987 samples (only five wells) are presented in Table 18.5-2:

TABLE 18.5-2

WELL SAMPLE ANALYSIS FOR NOVEMBER, 1987

Well #1	2.12 E-9 μCi/ml
Well #2	2.03 E-9 μCi/ml
Well #3	2.72 E-9 μCi/ml
Well #4	1.34 E-9 μCi/ml
Well #5	No Water

18.5.4.3 Sampling

Each of the wells will be sampled on a quarterly basis for two years, beginning in November, 1987. After this period, if no action level is exceeded, the sampling interval will be once each year, in the winter or spring.

18.5.4.4 Analysis

Each well sample will be analyzed for gross beta-gamma activity to detect the presence of Cesium-137 and Cobalt-60, the principle isotopes present in the liquid waste system. Currently there is no reason to suspect that alpha activity would be present in the groundwater since appreciable levels are not being detected during the sampling of the storage tanks. If significant alpha activity is detected during the sampling of the storage tanks, then a decision will be made to sample the wells for alpha activity as well as for beta-gamma.

18.5.4.5 Action Levels and Actions to be Taken

Gross beta-gamma activity exceeding 30 pCi/L will require the following actions:

1. Prepare and implement a plan to determine the source of the activity. This may include the taking of soil core samples, installation of additional wells, checking for alpha activity, and isotopic analysis of the samples.

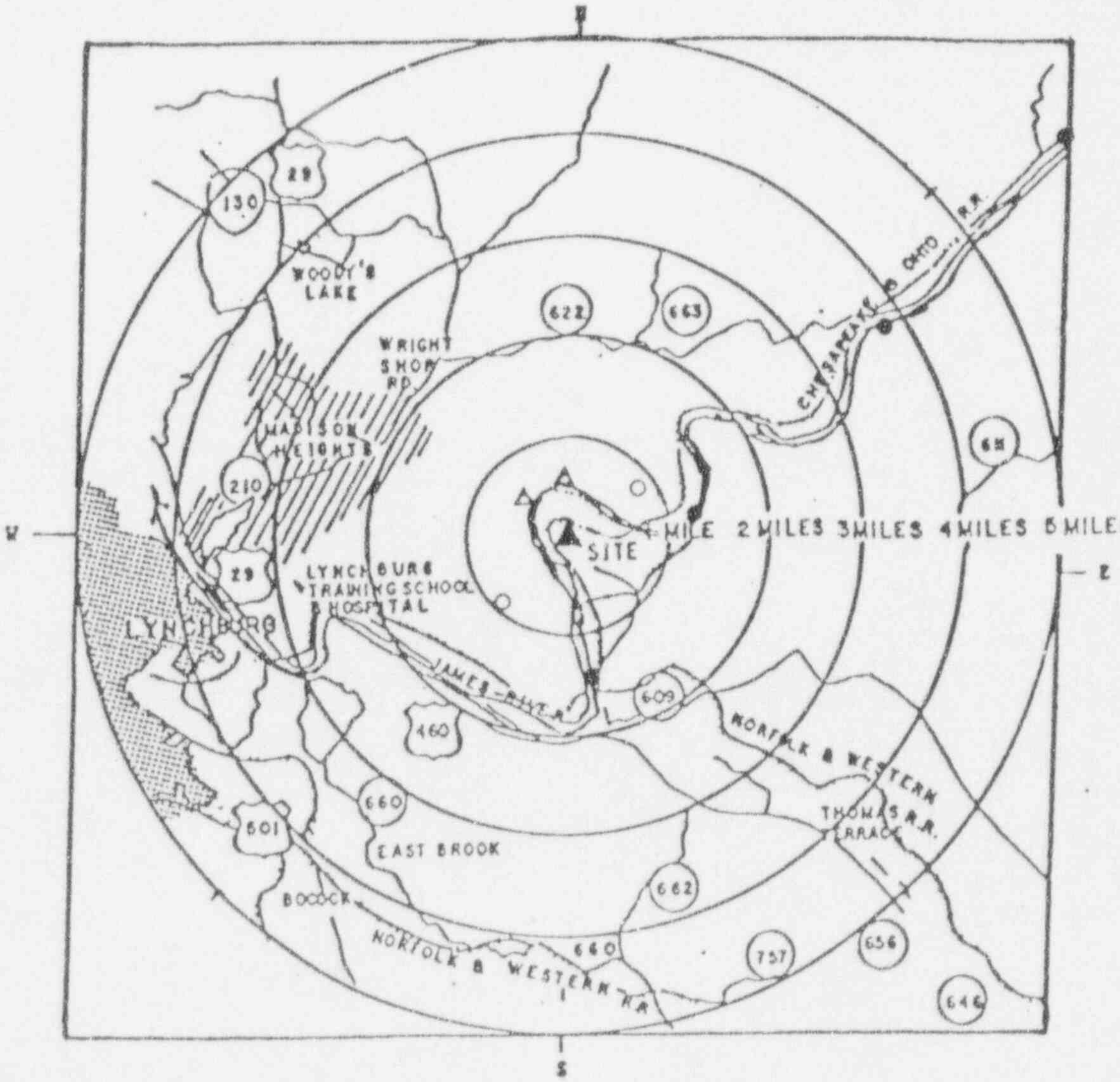
2. Submission of the report to NRC Region II in the event that the investigation of the elevated levels shows leakage in the tanks.

18.5.4.6 Records

Records of well sampling and analyses shall be retained until the NRC authorizes their disposition. These records shall include the dates when the samples were taken, sample analysis results, each instance that the action level was exceeded, the actions taken in response to the exceeded level, any changes made to the system, and all reports made to the NRC in connection with this program.

FIGURE 18.5-2

SEDIMENT, WATER, AND VEGETATION SAMPLING LOCATIONS



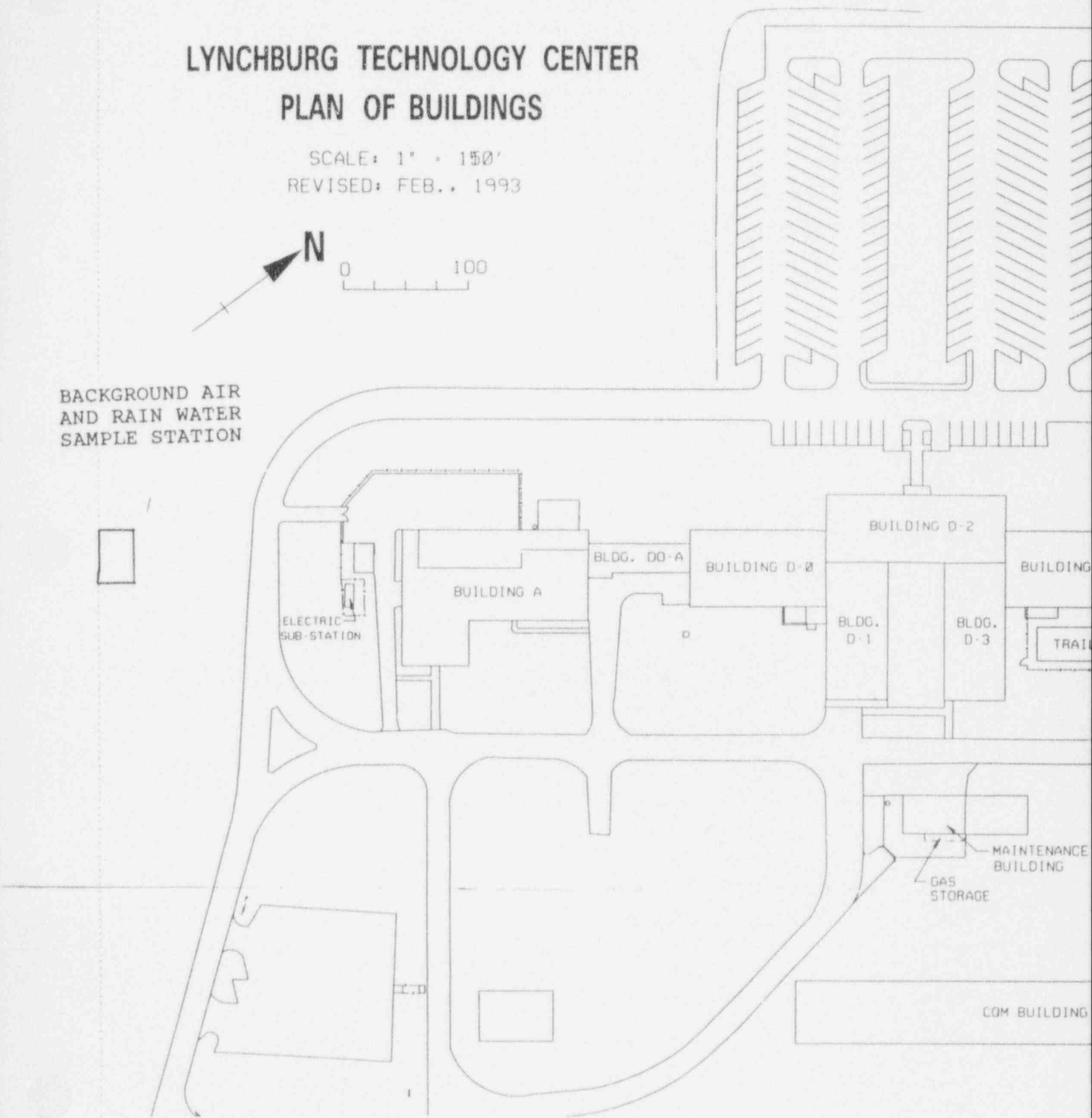
● Sediment

○ Vegetation

△ Water

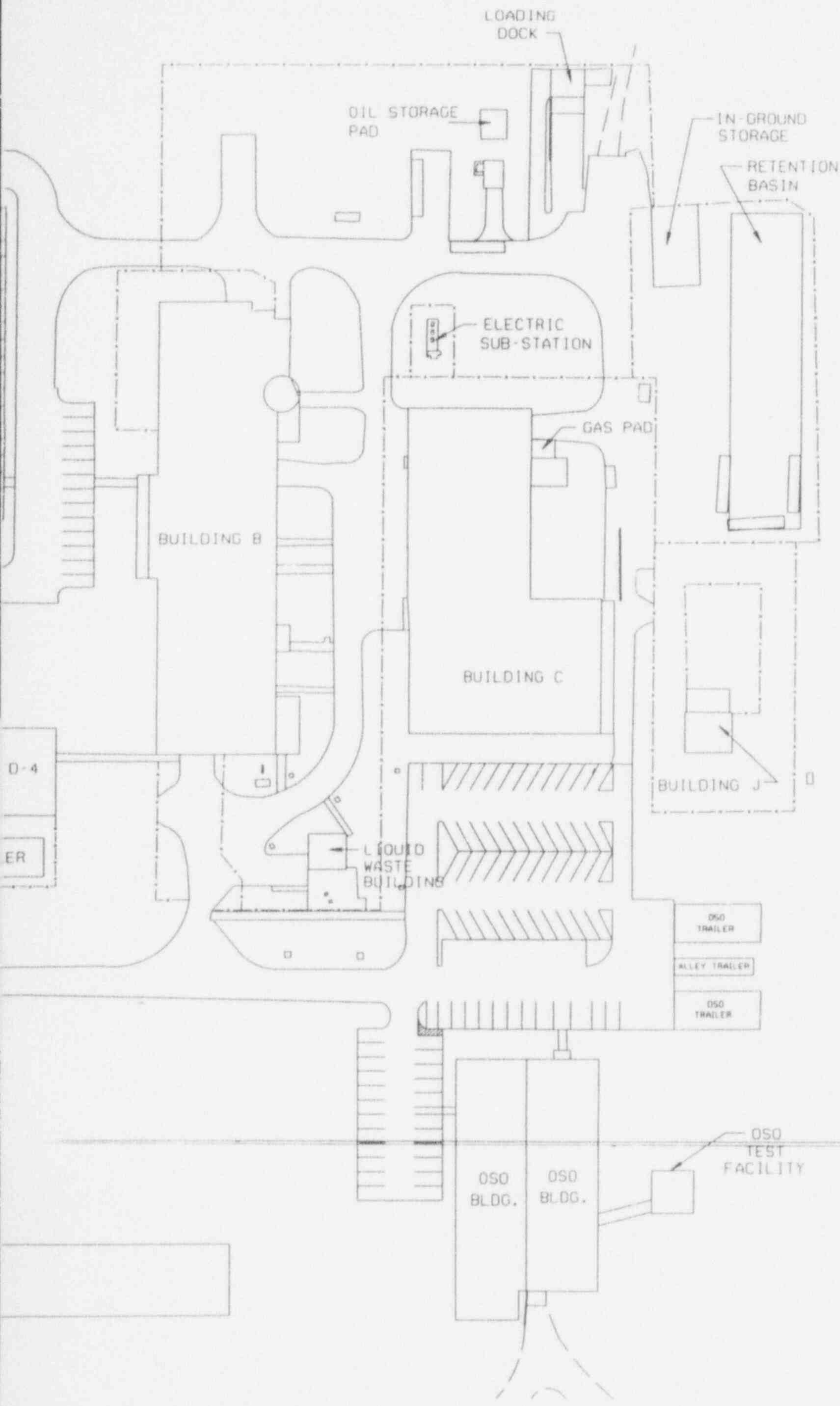
LYNCHBURG TECHNOLOGY CENTER
PLAN OF BUILDINGS

SCALE: 1" = 150'
REVISED: FEB., 1993

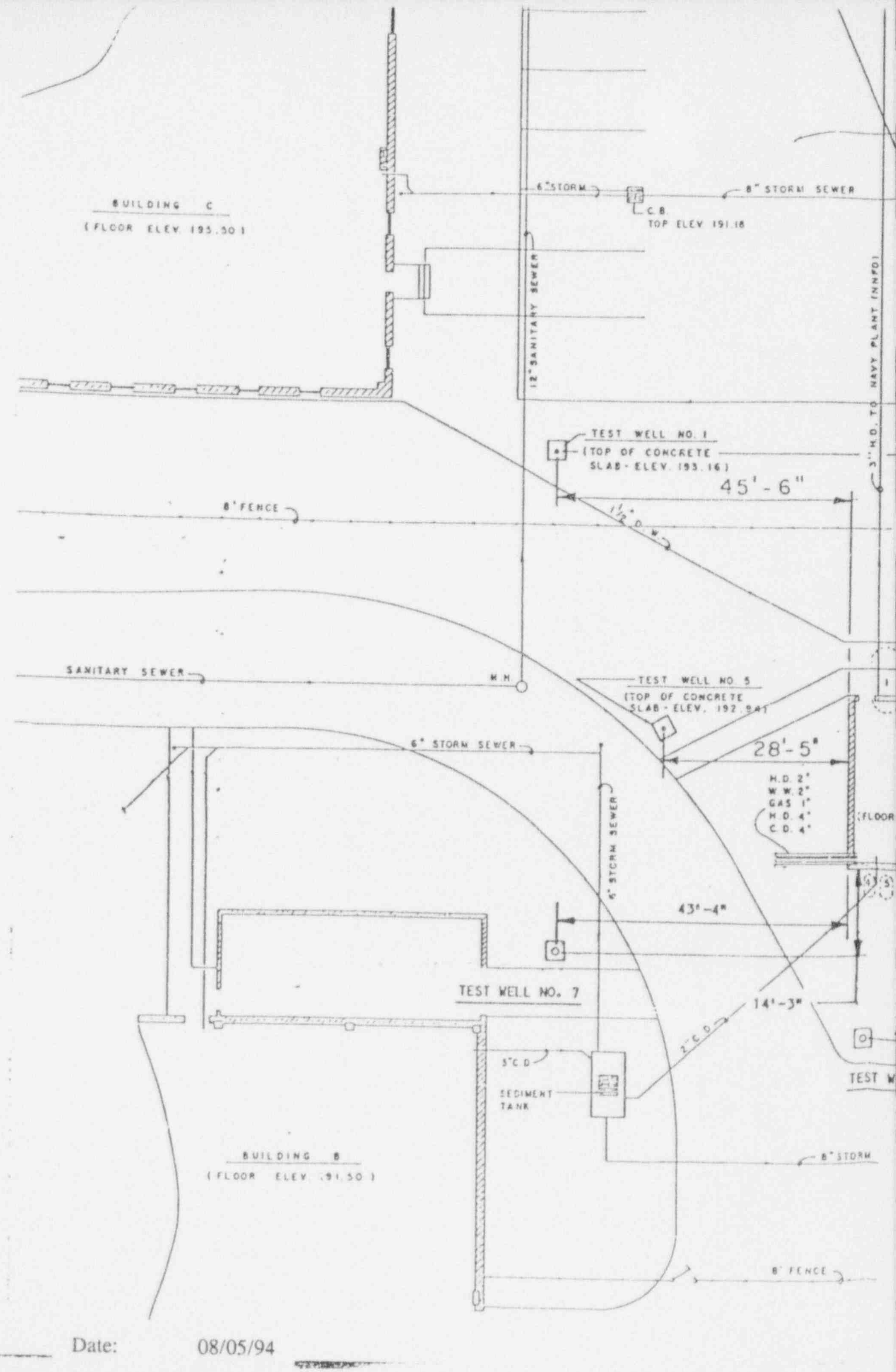


Date: 08/05/94

FIGURE 18.5-1



9408180208-05


BUILDING C
(FLOOR ELEV. 195.50)
6" STORM
C.B.
TOP ELEV. 191.18
8" STORM SEWER
12" SANITARY SEWER
3" H.D. TO NAVY PLANT (NNPD)
TEST WELL NO. 1
(TOP OF CONCRETE SLAB - ELEV. 195.16)
45'-6"
8' FENCE
1 1/2" D.W.
SANITARY SEWER
M.H.
TEST WELL NO. 5
(TOP OF CONCRETE SLAB - ELEV. 192.94)
6" STORM SEWER
28'-5"
H.D. 2"
W.W. 2"
GAS 1"
H.D. 4"
C.D. 4"
(FLOOR
6" STORM SEWER
43'-4"
TEST WELL NO. 7
14'-3"
2" C.D.
3" C.D.
SEDIMENT TANK
TEST W
8" STORM
BUILDING B
(FLOOR ELEV. 191.50)
8' FENCE

Date: 08/05/94

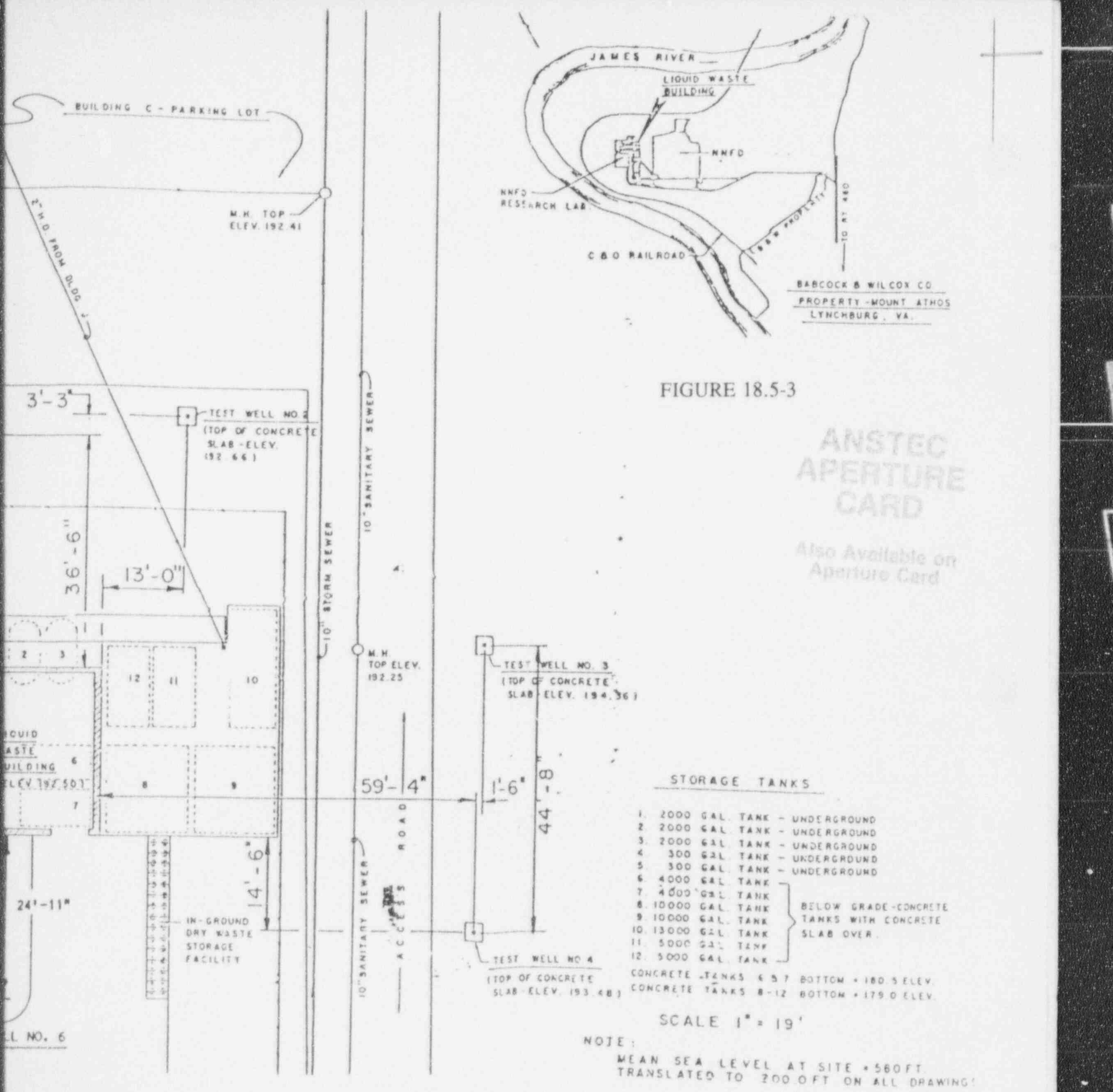


FIGURE 18.5-3

9408180208-06