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GENERAL ELECTRIC

NUCLEAR ENERGY BUSINESS OPERATIONS
GENERAL ELECTRIC COMPANY • VALLECITOS NUCLEAR CENTER • PLEASANTON, CALIFORNIA 94566

April 5, 1984

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U. S. NUCLEAR REGULATORY COMMISSION
NMSS
Mail Section

L. C. Rouse, Chief
Advanced Fuel and Spent Fuel Licensing Branch
Office of Nuclear Material Safety and Safeguards
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

References: 1) License SNM-960, Docket 70-754.
2) Application for Renewal of License SNM-960, 8/20/71.
3) Additional Renewal Information, 5/13/77.

Dear Mr. Rouse:

Enclosed is an updated supportive information section for the renewal application for License SNM-960. This submittal totally supercedes our submittal of May 13, 1977 (Ref. 3).

Sincerely,

G. E. Cunningham

G. E. Cunningham
Senior Licensing Engineer
(415) 862-2211, Ext. 4330

/ca

Encl.

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CONTROL NO. 23524

DATE OF DOC. 04/05/84

DATE RCVD. 04/06/84

FCUF ____ PDR ✓

FCAF ✓ LPDR ✓

WM ____ I&E REF. ✓

WMUR ____ SAFEGUARDS ✓

FCTC ____ OTHER ____

DESCRIPTION:

enclosed is an update supportive information section for the renewal application

04/12/84 INITIAL OEL

GENERAL ELECTRIC

SPECIAL NUCLEAR MATERIAL
LICENSE RENEWAL APPLICATION
FOR THE
VALLECITOS NUCLEAR CENTER

MARCH, 1984

License SNM-960

Docket 70-754

General Electric Company
Vallecitos Nuclear Center
P.O. Box 460
Pleasanton, CA 94566

SPECIAL NUCLEAR MATERIAL

LICENSE RENEWAL APPLICATION

FOR THE

VALLECITOS NUCLEAR CENTER

INTRODUCTION

The General Electric Company has been engaged in nuclear energy work since 1955. Company products and services include complete nuclear power stations, research and test reactors, fuel element fabrication, reactor subsystems, special engineering studies, applied research and development services, and the production of medical and research radioisotopes.

One of the major installations at which the Company conducts this work is the Vallecitos Nuclear Center located near Pleasanton, California.

OBJECTIVES OF THIS APPLICATION

The basic objective of this application is to obtain a renewal of License SNM-960, Docket 70-754.

APPLICATION

General Electric hereby submits the enclosed information describing all activities at the Vallecitos Nuclear Center in which special nuclear materials are utilized, except those activities subject to licensing pursuant to 10 CFR Part 50, in support of our application for renewal of August 20, 1971. The information submitted herewith completely supersedes our previously submitted supportive information dated May 13, 1977. Several sections contained in the previous submittal have been eliminated or replaced with new sections describing different facilities. The facilities previously described in the eliminated and/or replaced sections have been discontinued. On the basis of this information, General Electric requests that License SNM-960 be renewed.

By G. E. C[signature]
G. E. Cunningham
Senior Licensing Engineer
Irradiation Processing Operation

CONTENTS

1.0 GENERAL INFORMATION

1.1 CORPORATE AND FINANCIAL

This application is filed by General Electric Company, a New York corporation with a principal place of business at One River Road, Schenectady, New York. A list of the names of the directors and officers appears in General Electric's latest Annual Report, a copy of which is forwarded to the Commission each year for inclusion in Docket 70-754. All may be addressed at 3135 Easton Turnpike, Fairfield, Connecticut 06431.

General Electric is not controlled by any alien, foreign corporation, or foreign government; it is controlled by its Board of Directors and the Officers elected by the Board.

General Electric is a publicly held corporation whose stock is traded on the principal security exchanges. The applicant has no knowledge or any information indicating any appreciable ownership of General Electric stock by an alien, foreign corporation, or foreign government. No person owns of record or is known by General Electric to own beneficially one percent or more of the outstanding shares of its capital stock.

Financial information required by 10CFR70.22 appears in the aforementioned Annual Report.

1.2 LOCATION AND GENERAL DESCRIPTION OF VALLECITOS NUCLEAR CENTER

The General Electric Vallecitos Nuclear Center (VNC) is operated in conjunction with the operation and manufacture of nuclear reactors, the development of advanced reactor concepts, and to provide nuclear products, processes and services. It is located near the center of the Pleasanton quadrangle of Alameda County, California. The laboratory is east of San Francisco Bay, approximately 35 air miles east-southeast of San Francisco and 20 air miles north of San Jose. The site is indicated on the Bay Area map, Figure 1.1. The nearest sizeable towns are Pleasanton with a population of approximately 30,000 located 4.1 air miles to the north-northwest, and Livermore with a population of approximately 50,000 located 6.2 air miles to the northeast. A United States Veterans Administration Hospital with a population of approximately 300 is located about four miles to the east.

The site is on the north side of Vallecitos Road (State Route 84), which is a two-lane paved highway. The Southern Pacific and Western Pacific railroads lie about two miles west of the site. The laboratory site consists of approximately 1,600 acres, about one-third of which is relatively flat. Approximately 1,400 acres of the site are leased for raising feed crops and cattle grazing.

There is very little industrial activity within a 10-mile radius of the plant. A small amount of light industry is located at Pleasanton, Fremont, and Livermore, but these towns are not industrial centers. The city of San Jose to the south, 20 miles distant, and Oakland and San Francisco, 30 and 35 miles respectively, to the northwest are the major industrial centers in the vicinity. In the southeast quadrant, there are no industries and very sparse population for 20 miles and beyond.

The property on which the laboratory buildings are located is drained by three branches of Vallecitos Creek and, on the west, by an unnamed creek. Both of these creeks discharge to Arroyo de la Laguna near the north end of Sunol Valley, two to three miles southwest of the property.

Water is supplied from the Hetch-Hetchy aqueduct by means of a 14-inch line capable of supplying over 3,000,000 gallons per day. A 500,000-gallon storage tank is provided on the laboratory site. One hundred thousand gallons are reserved for fire protection.

Electrical power is supplied by Pacific Gas & Electric to the main laboratory substation from whence it is distributed to each building on the site.

A sewage treatment system is provided in the southwest corner of the site. Effluent from this system is disposed to site land.

1.3 GENERAL PLANS AND USES OF SPECIAL NUCLEAR MATERIALS

This application requests authorization under Title 10, Code of Federal Regulations, Part 70, to receive and possess the special nuclear material designated in Section 1.4 herein; to receive and possess the special nuclear material and associated byproduct material produced by the irradiation thereof; to separate or process such special nuclear materials only for research and development activities, for analytical purposes, or research and development activities described in 10CFR50.2(a)(3)iii; and to use said special nuclear materials in research and development activities as defined in Section 70.4(j) and in chemical and physical analysis and examination and investigation of nuclear fuels, associated materials and devices at the Vallecitos Nuclear Center and at sites within the limits of the United States except in an Agreement State where General Electric is exempt from NRC licensing when such materials and devices are under the direct and personal possession of General Electric at all times.

1.4 SPECIAL NUCLEAR MATERIAL POSSESSION LIMITS

1.4.1 Vallecitos Nuclear Center

The special nuclear materials used in connection with activities at the Vallecitos Nuclear Center will not at any time exceed those listed in Section 2.1 of Appendix A to License SNM-960.

1.4.2 Locations Other Than the Vallecitos Nuclear Center

For purposes of demonstration, assembly or nonnuclear, nondestructive testing, 15 grams of U-235 or plutonium in any chemical or physical form or 170 grams of plutonium as sealed plutonium-neutron sources may be used at other sites within the limits of the United States except at locations in an

Agreement State as defined in Section 150.3 of 10CFR150 provided that any such special nuclear material be transferred in an approved container and that such devices are handled and used under the direct control of General Electric at all times. "Direct control" means that the source will be used in the presence and/or under the direction of General Electric personnel. General Electric personnel also shall ascertain that the material is placed in proper storage after use.

1.4.3 Form and Enrichment Specifications

The majority of the Vallecitos Nuclear Center activities are conducted in facilities and under procedural controls which accommodate any chemical or physical form and any U-235 isotopic content. If specific limitations are placed on these parameters in connection with an individual activity in order to assure the radiation or nuclear safety of that work, the limit is described in the appropriate section of this application entitled "General Plans and Uses of Materials".

1.5 PRINCIPAL VALLECITOS FACILITIES

Descriptions of the principal buildings and laboratories in which special nuclear materials are used at the Vallecitos Nuclear Center site are set forth in this section with the primary objective of general orientation. The locations of these facilities are shown in Figure 1.2. The specific activities conducted in each of these facilities and their safeguards equipment and procedures are discussed in later sections.

1.5.1 Radioactive Materials Laboratory

The Radioactive Materials Laboratory (RML) is located in Building 102. This laboratory is a shielded facility equipped with remote manipulators to conduct experiments and analyses with irradiated reactor fuels and other radioactive materials generated in Vallecitos reactors or received from customers and to separate specific radionuclides. The facility also includes a storage pool and dry pit storage.

1.5.2 Radiochemistry Laboratory

Adjacent to RML, on the main floor of Building 102, and providing analytical support to it, is a radiochemistry laboratory equipped with standard chemical and radiochemical apparatus. This laboratory primarily is used to analyze samples of materials prepared in the RML.

1.5.3 Core Materials Engineering Laboratory (CMEL)

Located in the basement of Building 102, the CMEL is equipped with glove boxes, fume hoods, and other apparatus appropriate to materials testing and properties studies and process development demonstrations.

1.5.4 Metallurgy, Chemistry, and Ceramics Building

A second major laboratory building in the 100 Area is the Metallurgy, Chemistry, and Ceramics Laboratory - Building 103. This two-story building consists of laboratories, variously equipped with laboratory apparatus designed to handle moderate quantities of radioactive materials, and offices. The functions served by this facility are research, development, and analytical chemistry services.

1.5.5 Building 105

Just north of Building 102 is Building 105. The principal facilities located in this building are an experimental reactor (the Nuclear Test Reactor) and laboratories.

The Nuclear Test Reactor serves as a source of neutrons for neutron radiography, exponential experiments, irradiations, and as a sensitive device for reactivity measurements. The laboratories in Building 105 use only minute quantities of special nuclear materials.

1.5.6 Engineering Shop

Building 106 contains machine shops, instrument calibration facilities, and the development shop. Special nuclear materials are brought to the development shop as encapsulated devices for equipment or mechanical modification and radiography.

1.5.7 Solid Radioactive Waste Storage Facility

Solid radioactive wastes generated at the various laboratory and facility locations are stored in the waste storage facility located approximately midway between the deactivated Vallecitos Boiling Water Reactor (VBWR) and General Electric Test Reactor (GETR) areas. This storage area includes vertical wells for storing drummed wastes and horizontal tubes for storing 5-inch and 7½-inch diameter waste liners.

1.5.8 Waste Treatment Plant

The Waste Treatment Plant is located adjacent to the deactivated VBWR site. This plant is used to concentrate and solidify liquid radioactive wastes generated at the Vallecitos Nuclear Center or other licensed facilities prior to transfer to authorized waste disposal firms or waste burial sites. Such wastes contain minimal quantities of special nuclear material.

1.5.9 Reactors and Auxiliary Facilities

The ESADA-Vallecitos Experimental Superheat Reactor (EVESR) and the VBWR are deactivated. The GETR with its auxiliary facilities, such as storage, maintenance and office buildings, is currently in a shut down condition.

1.5.10 400 Area

The 400 Area consists of two buildings, 400 and 401. Building 401 is devoted chiefly to offices, while Building 400 contains an experimental low-enrichment uranium scrap recovery system.

1.5.11 Building 104

This building includes warehousing and shipping and receiving facilities.

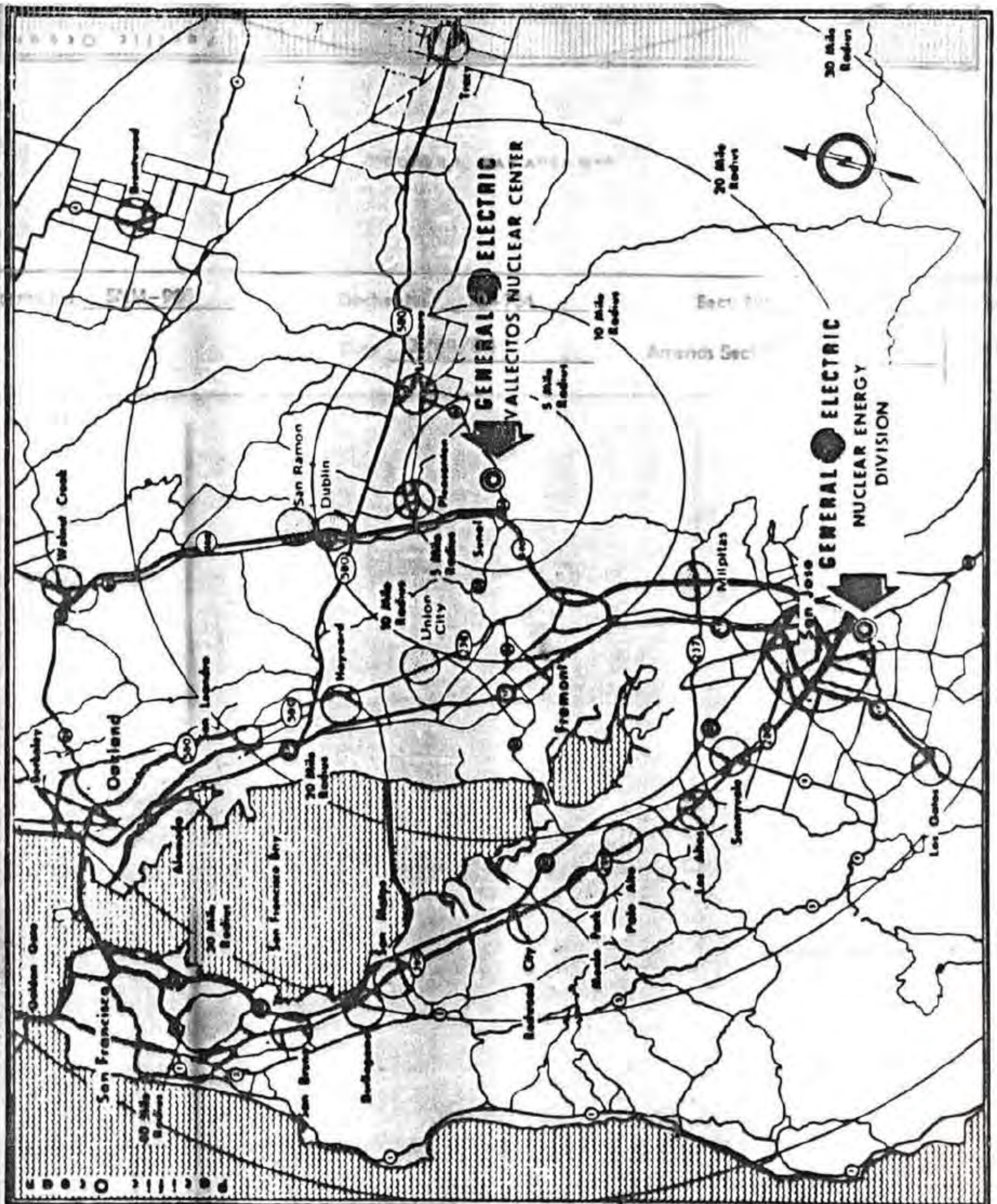

FIGURE 1.1. BAY AREA MAP

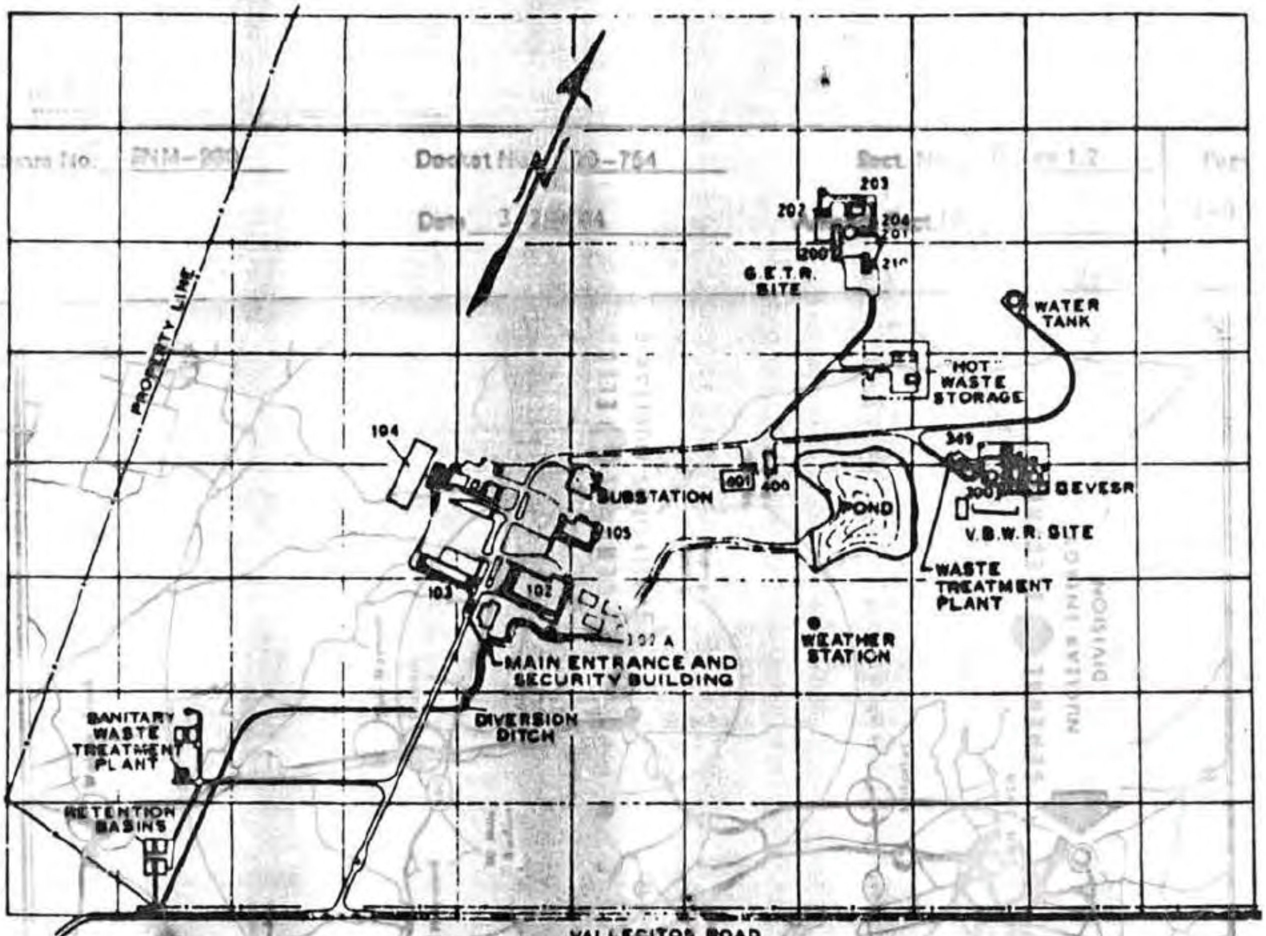

FIGURE 1.2. VALLECITOS NUCLEAR CENTER BUILDING LOCATIONS

2.0 ORGANIZATION AND ADMINISTRATION

All Vallecitos Nuclear Center (VNC) activities are conducted under the management of three organization components of the General Electric Company.

Nuclear Engineering Division (NED) - This Division provides the technical definition, quality requirements, and the design for all standard BWR product offerings except control and instrumentation and including the supporting research and development activities at the Vallecitos site.

Nuclear Fuel and Special Projects Division (NF&SPD) - This Division is represented at VNC by the Irradiation Processing Operation (IPO), which is responsible for the operation of the General Electric Nuclear Test Reactor, radioactive materials handling and inspection, radioisotope encapsulation, nuclear safety, and landlord activities at Vallecitos; by Core and Fuel Technology, which is responsible for development and characterization of reactor core materials; and by the Advanced Nuclear Technology Operation (ANTO). ANTO develops technology for advanced reactor fuels, performs materials testing and properties studies, and performs process development demonstrations. NF&SPD work is carried out for General Electric divisions, for government agencies, and for outside contractors.

Nuclear Services Operation (NSO) - This component provides consulting, retrofit, and upgrade services to operating nuclear power plant customers as required to operate and maintain BWR plants.

2.1 ORGANIZATION

An organization chart for VNC is included at the end of this section as Figure 2.1. An organization chart for the nuclear safety function is included as Figure 2.2.

2.2 DELEGATION OF RESPONSIBILITY

The General Managers of the components at VNC have established a policy of protection of employees, the public, and the environs from potential industrial, radiation, and nuclear hazards that could occur through activities conducted in each component's facilities. They have delegated the responsibility for implementing this basic policy through line managers to the manager and supervisor of each activity in which radioactive materials are handled, used, or stored. Additionally, IPO has experienced and competent staff personnel to provide expert advice and guidance to all components in matters of radiation and criticality safety. The Manager, IPO, has been delegated responsibility to act as the chief executive safety officer for all VNC operations involving radioactive materials. Industrial safety is provided for VNC by the Industrial Safety and Hygiene function.

With respect to the radiation and criticality safety programs, the basic line and staff relationships are illustrated in Figures 2.3 and 2.4, respectively.

2.3 SITE MANAGEMENT REVIEW FUNCTION

The Vallecitos Technological Safety Council (VTSC) is composed of senior site management and/or technical personnel and has been established as a site-wide evaluation function with responsibility for management evaluations and reviews of VNC activities. Assigned functions include review of VNC activities to assure that major risks are identified, considered, and resolved and that the application of policies and procedures are consistent as between the site components. This function will review changes in product design or services that are first of a kind or that require deviation from established parameters and reflect critically on product or operational safety.

2.4 RESPONSIBILITIES OF THE NUCLEAR SAFETY FUNCTION

a. Radiation Safety. Establish and administer a radiation safety program to ensure the protection of employees and the general public. Provide monitoring support, a dosimetry program, an environmental monitoring program, and employee training in the methods for minimizing exposure through proper use of survey instruments and protective clothing and devices. Maintain all radiation exposure records required by regulatory agencies.

b. Nuclear Safety Compliance and Review. Provide review of reportable incidents, new facilities or major changes to facilities and operations control standards, and professional advice and counsel on nuclear and radiation safety policy.

c. Licensing. Represent VNC components in regulatory activities concerning radiation protection and licensing. Procure, administer and interpret NRC licenses and regulations, state and local government licenses and regulations, and provide advice and counsel to VNC components and their customers on regulatory matters.

d. Radiological Engineering. Assist in improved operation of VNC facilities and the overall radiation protection program through the analyses of existing systems, equipment and operation thereof and the recommendation of improved systems, equipment or methods. Generally, these studies will be safety and plant life oriented.

e. Training. Establish and administer a site-wide training program at VNC to ensure adequate knowledge of radiation control procedures.

f. Criticality Safety. Perform criticality control analyses to establish safe batches, geometries, concentrations, and spacing of special nuclear materials and equipment. Audit the criticality control environment and conduct educational programs in criticality matters.

g. Emergency Preparedness and Response. Maintain the site emergency plans current; coordinate and audit training, tests, drills, and exercises.

2.5 OTHER COMPONENTS CONTRIBUTING TO OVERALL SAFETY

In addition to the principal radiation and nuclear safety functions of the staff components previously described, the following components and their functions make significant contributions to the overall safety program:

a. Nuclear Materials Safeguards. Establish and administer the basic system of special nuclear material control. Assure that SNM control policies and practices of all components are coordinated with the requirements of nuclear safety, licensing and shipping groups. Perform reviews of compliance with internal control procedures and regulatory requirements.

b. Physical Security. Develop and operate security programs to safeguard classified information, special nuclear materials, and corporate property.

c. Transportation and Materials Distribution. Within traffic regulations, plan the transportation and receiving of incoming materials and shipments of outgoing products. Assure all radioactive materials shipments are in compliance with applicable Federal and State regulations.

d. Industrial Safety and Hygiene Function. Develop programs to protect the employees from industrial hazards, including operation of medical and safety education programs.

2.6 TECHNICAL PERSONNEL CAPABILITIES

The ultimate responsibility for operational radiation safety for the operations conducted in the various Vallecitos facilities involving special nuclear material rests with the supervisor or manager of each facility. Equally important are the knowledge and experience of personnel in the nuclear safety function. Since the issuance of License SNM-960 in 1966, the Commission's Region V office has inspected VNC to assure that adequate levels of technical expertise are maintained in all positions. Resumes for key personnel are included as Addendum A to this section.

2.7 IMPLEMENTATION OF CRITICALITY CONTROL PROGRAM

The program for protection against accidental conditions of criticality is implemented by means of functional responsibility assignments. Managers whose operations require the use of quantities of special nuclear materials approaching a theoretical minimum critical mass, or greater, are responsible for integrating and measuring the efforts of line and staff participants in this program. The principal participants and their responsibilities are outlined below.

2.7.1 Responsibility of the Manager

It is the responsibility of any manager of an activity involving special nuclear materials to establish and maintain a records system to accurately account for the quantities of such materials used and transferred, thereby providing a basis for controlling against excessive accumulation or loss.

It is the responsibility of any manager of an activity involving more than 500 grams of U-235 or 300 grams of plutonium or U-233, through the designated supervisor or engineer with specific responsibility for a given detailed activity area, to:

a. Know and understand the specific limits of all nuclear parameters applicable to his area of responsibility.

b. Determine the need for a Criticality Analysis. This need arises whenever there is: (1) a change in the parameters of special nuclear material such as enrichment, chemical and physical form, density, etc.; (2) a change in the parameters such as quantity or dimensions of unit accumulations; (3) a change in the spacing or dimensions of processing, handling or storage equipment, or (4) a change in environment such as the addition of fire sprinkler systems of possible consequence to nuclear safety.

c. Prepare in writing a Request for Criticality Analysis stating the activity or area involved, including a description of the special nuclear material, process description, and equipment description with appropriate charts, diagrams, and drawings. All special nuclear material shall be included whether license or license-exempt and whether as usable forms or as scrap and waste.

d. Upon receipt of results of a Criticality Analysis in writing from the criticality safety function, take all necessary steps to implement the appropriate controls such as:

1. Posting of limitations at strategic locations at the criticality areas where they apply.

2. Issue appropriate procedures for establishing and maintaining equipment and materials in the arrangement and quantities specified by the Criticality Analysis.

3. Assure boundaries identifying each separate criticality area and position are conspicuously marked.

4. Assure all operating personnel are advised of any special limitations or controls other than as outlined above and are trained to follow the necessary precautionary procedures.

5. Establish procedures for handling off-standard processing conditions.

2.7.2 Responsibility of Criticality Safety Function

It is the responsibility of the criticality safety function to:

a. Review requests for Criticality Analysis to verify completeness of submitted information and, as appropriate, personally inspect the equipment or location involved to obtain a first-hand determination of the possibility of interactions with other special nuclear material in the environment.

b. Make criticality control calculations to establish safe batches, geometries, concentrations, and spacing of special nuclear materials and equipment in accordance with authorized computational methods.

c. Furnish a written Criticality Analysis to the requesting operation for each request received. The person in charge of the criticality safety function shall identify the analyses which involve systems in which the direct achievement of criticality may be the consequence of error in analysis and shall assure that such analyses are reviewed and verified.

d. Perform a periodic inspection of the degree to which actual operations conform to physical situations on which the criticality calculations were based.

e. Conduct a criticality control educational program to emphasize the need for following precise instructions.

2.7.3 Responsibility of Radiation Safety Function

It is the responsibility of the radiation safety function through its staff of qualified and experienced radiation monitors to:

a. In their normal rounds of duty be continuously alert to possible criticality safety practices inconsistent with the appropriate criticality analyses and/or procedures.

b. Be familiar with applicable parameters for criticality control and use these parameters as standards for inspection.

c. Verify that physical conditions and activities comply with the conditions specified in the Criticality Analysis.

d. Initiate immediate action to discontinue any SNM movement and shut down any operation which in its judgment does not fully comply with the Criticality Analysis.

2.8 IMPLEMENTATION OF RADIATION SAFETY PROGRAM

The program for protection against radiation hazards is implemented by means of functional responsibility assignments. Managers are responsible for integrating and measuring the efforts of line and staff participants in the program.

2.8.1 Responsibility of the Manager

It is the responsibility of the manager of an activity or area involving radioactive materials to:

a. Take all necessary steps to plan and organize the work of his component in accordance with approved radiation safety standards and operational procedures.

2.7.3 Responsibility of Radiation Safety Function

It is the responsibility of the radiation safety function through its staff of qualified and experienced radiation monitors to:

a. In their normal rounds of duty be continuously alert to possible criticality safety practices inconsistent with the appropriate criticality analyses and/or procedures.

b. Be familiar with applicable parameters for criticality control and use these parameters as standards for inspection.

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The program for protection against radiation hazards is implemented by means of functional responsibility assignments. Managers are responsible for integrating and measuring the efforts of line and staff participants in the program.

2.8.1 Responsibility of the Manager

It is the responsibility of the manager of an activity or area involving radioactive materials to:

a. Take all necessary steps to plan and organize the work of his component in accordance with approved radiation safety standards and operational procedures.

b. Identify needs for operational procedure revisions when there is a planned change in conditions such as types or quantities of radioactive materials or equipment modifications.

c. Integrate the results of reviews, inspections, engineering assessments and investigations made by the radiation safety function to correct or improve operational procedures, controls and performance.

2.8.2 Responsibility of Radiation Safety Function

It is the responsibility of the radiation safety function to:

a. Establish and administer through a system of instructions a radiation control program and maintain all radiation exposure records required by regulatory agencies.

b. Review and approve new operations in the design stage, identifying potential hazards and providing recommendations for their avoidance through education, training methods, and equipment application.

c. Review and approve requests for allocations of radioactive materials, within licensed quantities, to individual user groups after thorough investigation of the proposed detailed procedures, equipment and the qualifications of persons who will be responsible for using such materials. Maintain records of allocated quantities.

d. Inspect each radiation area to assure compliance with radiation safety procedures and limits.

e. Provide management with reports of inspection results showing trends in activities involving radioactive materials.

2.9 TRAINING PROGRAMS

2.9.1 Radiation Safety

Every employee, regularly or frequently exposed to radioactivity, is instructed in radiation protection such that he is able to protect himself and is made aware of the degree of hazard involved. A training program entitled, "Radiological Safety At Vallecitos Nuclear Center", is completed by each employee normally within one year of his starting date at VNC. This course is scheduled and conducted by the radiation safety function. Follow-up training commensurate with the work environment is determined by employee supervision.

The training course includes the following elements:

a. basic principles of radiation safety,
b. Company policies and operating procedures,
c. radioactive materials handling methods and shielding requirements,
d. emergency procedures,
e. requirements of NRC regulations, and
f. NRC license requirements.

2.9.2 Criticality

Every new employee who will work regularly in areas in which quantities of special nuclear material sufficient to form a critical mass are present is instructed in the principles of criticality safeguards and is made aware of the degree of hazard involved. This instruction normally is completed within one year after the employee's starting date at that facility. The instruction is conducted by the criticality safety function.

Area supervisors are responsible for informing all personnel at work or otherwise present in their area of specific procedures for criticality control, and for appropriate administrative action to assure compliance with these procedures.

2.10 INDEPENDENT REVIEW AND CHANGE AUTHORIZATION PROCEDURE

2.10.1 Definitions

a. Change Authorization. A Change Authorization is the VNC mechanism for independently reviewing and documenting changes. Any addition, alteration, deletion, modification, or substitution which results in a different position, course, or direction not previously analyzed, adds a new capability, performs a different function, modifies performance characteristics, or introduces a hazard not previously analyzed will constitute a change. Descriptions of existing facilities and standard methods of operation normally are contained in documents supporting Federal and State licenses or previously approved operating instructions and may be used to assist in determining if the proposed activity constitutes a change.

b. Facility. A facility consists of any permanent structure and that apparatus which, wholly or in part, is integrated with the structure, i.e., shielding, piping, ventilation, reactor core, etc.

c. Equipment. Equipment is defined as that apparatus which operates independently or in conjunction with the facilities to perform the various VNC activities.

d. Experiment or Test. An experiment or test consists of any device or combination of devices (not to include the facility or equipment wherein the experiment or test is housed) designed and operated for the purpose of obtaining a predetermined objective, or any nonroutine operation or manipulation of objects, systems or process streams for the purpose of gaining information.

e. Procedure. Procedures are the written guidelines established by facility management to describe and define the methods and instructions for operations.

2.10.2 Purpose

Any addition, alteration, deletion or substitution which adds a new capability, performs a different function, modifies performance characteristics, or introduces a hazard not previously analyzed requires an independent review by use of the Change Authorization procedure. A Change Authorization is prepared whenever the work involves changes to:

a. Facilities, equipment, or procedures so that safety or regulatory compliance considerations differ from those previously analyzed.

b. Preventive or corrective maintenance procedures for which no separate review procedures are established which could affect safety or violate a license condition or technical specification.

c. Radioactive material inventories and/or limits.

d. Hazardous or potentially hazardous industrial material inventories where such change is significant in terms of quantities or use.

The independent review of items is conducted so that the hazards (both direct and indirect) of the proposal are recognized and appropriate safeguards are provided to eliminate or reduce the probability and severity of potential accidents. While procurement, fabrication, selective installation or testing, etc., may proceed prior to the final CA review and approval, actual implementation of the proposed change may not proceed until this review and approval are received.

The Change Authorization is processed in accordance with a written procedure and reviewed by the Nuclear Safety function and by the Industrial Safety function as appropriate.

Early consultation with these review parties is important to the safety and economy of component plans. Should the operating component decide that the proposed activity does not require an independent review, that component is responsible to fully document its proposed activity, the results of its internal review, and the reasons an independent review was not required.

2.10.3 Procedures

a. Preparation of Change Authorization. Knowledgeable individuals prepare Change Authorizations which describe in sufficient detail the nature of the changes and the effect on safety, including applicable drawings and specifications, acceptance test procedures (ATPs), quality control requirements (if applicable), means for assuring personnel indoctrination for initiating method changes, and specifying the responsible supervision.

Certain minor changes and changes previously evaluated and documented need only internal review and documentation. The Nuclear Safety functions will assist the initiating component in making this determination.

b. Review of Change Authorization. All Change Authorizations (CA) are reviewed by the initiating component, by the Nuclear Safety function, and, as appropriate, by Industrial Safety and are approved by the appropriate area or facility manager or his designated alternate. Recommendations for the change needed for safety may be added by the reviewers following discussion with facility personnel as appropriate. The Nuclear Safety function has the responsibility for determining whether a proposed change constitutes an unreviewed safety question or other license or technical specification violation. No CA may be implemented until review is completed and the CA signed by the manager of the Nuclear Safety function or his designated alternate.

c. Distribution. A Change Authorization file will be maintained by each facility. A copy of the Change Authorization form with all supporting information must be filed with the Nuclear Safety function with other copies distributed as appropriate.

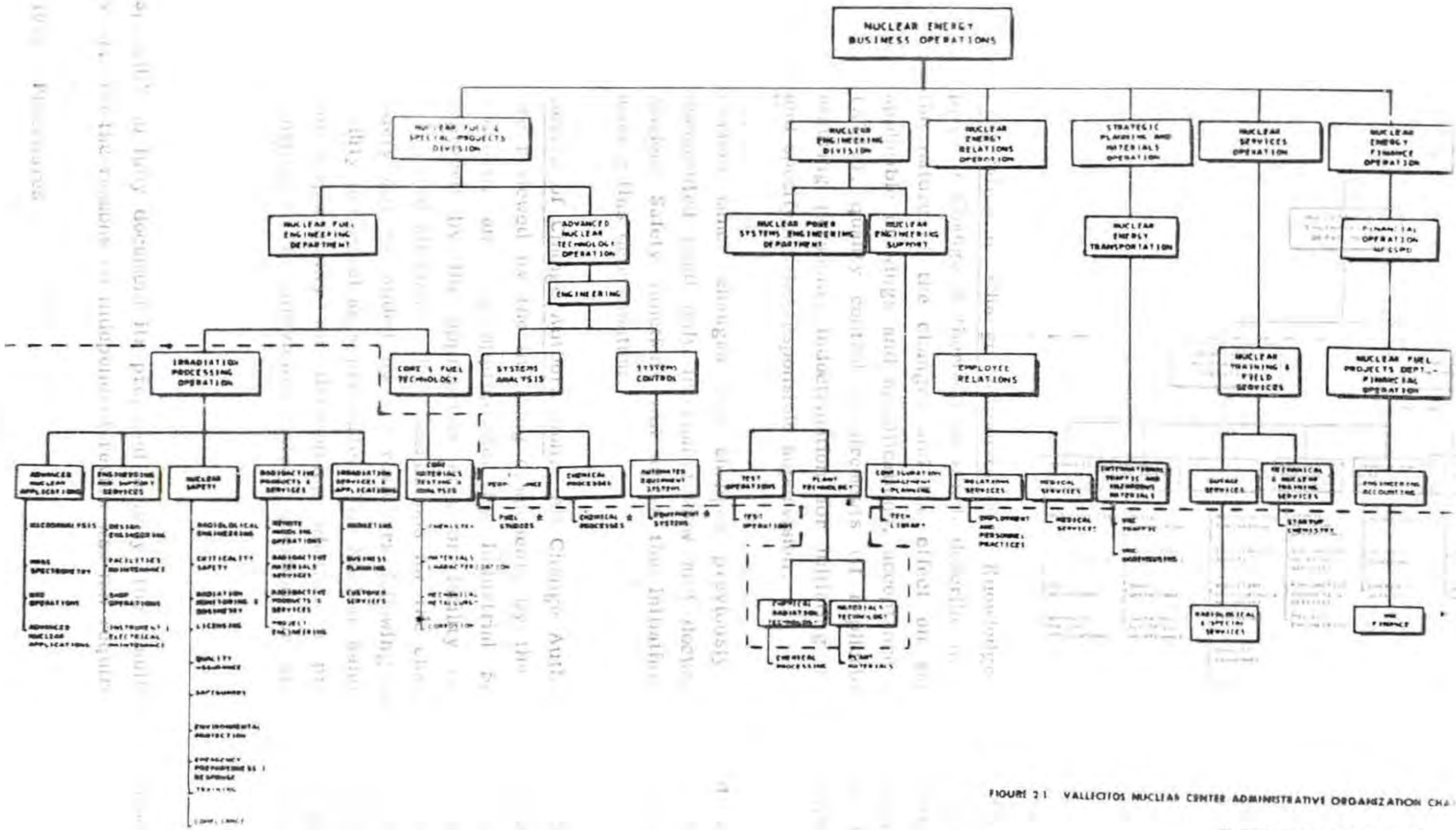

FIGURE 2.1 VALLECITOS NUCLEAR CENTER ADMINISTRATIVE ORGANIZATION CHA

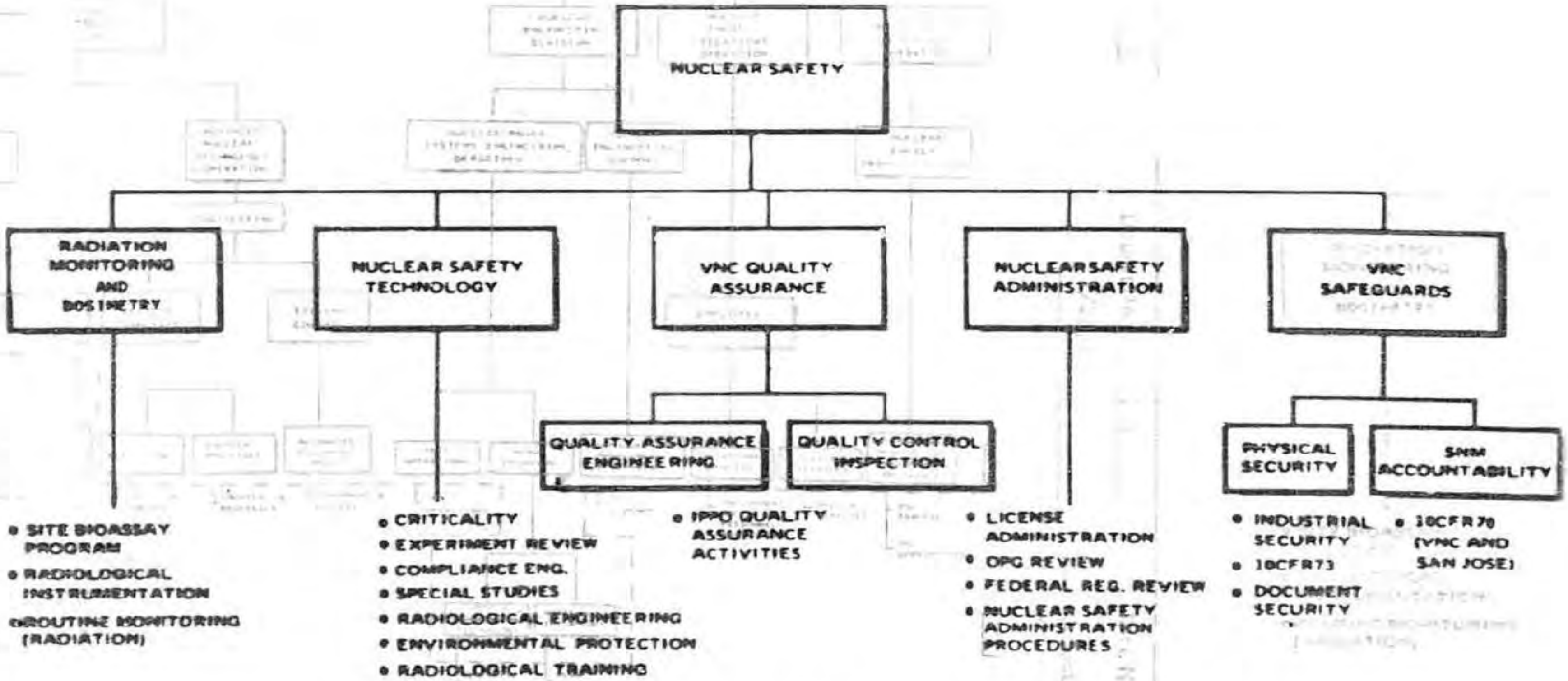

FIGURE 2.2. NUCLEAR SAFETY ORGANIZATION

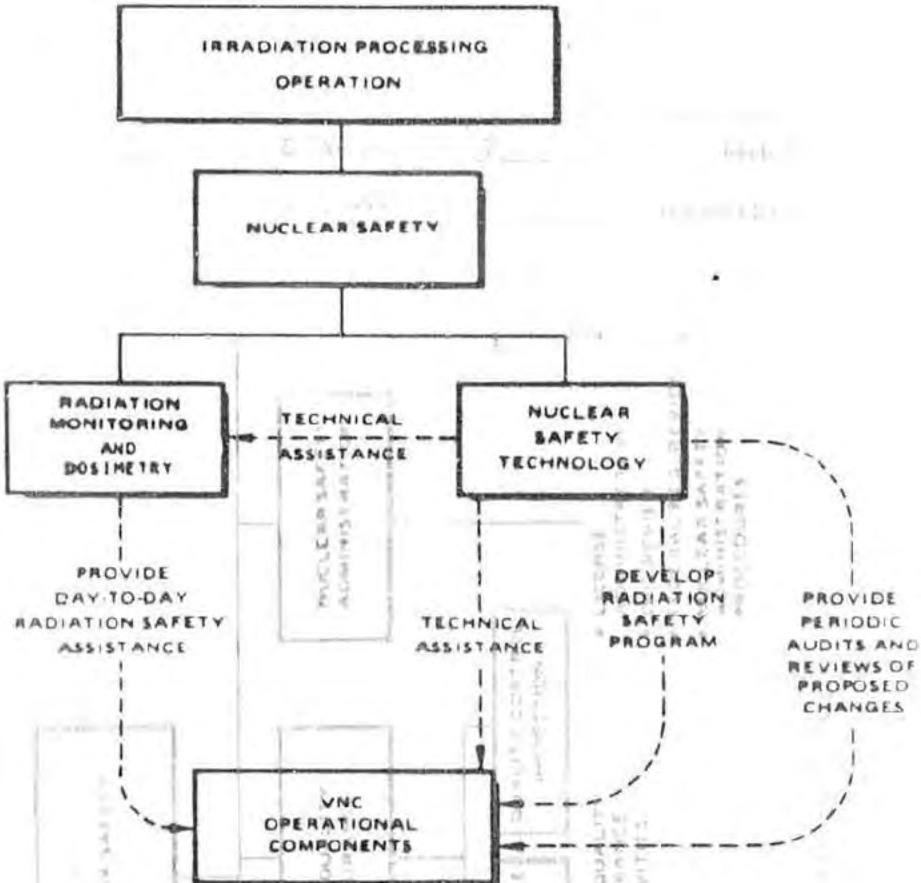

FIGURE 2.3. RADIATION SAFETY STRUCTURE

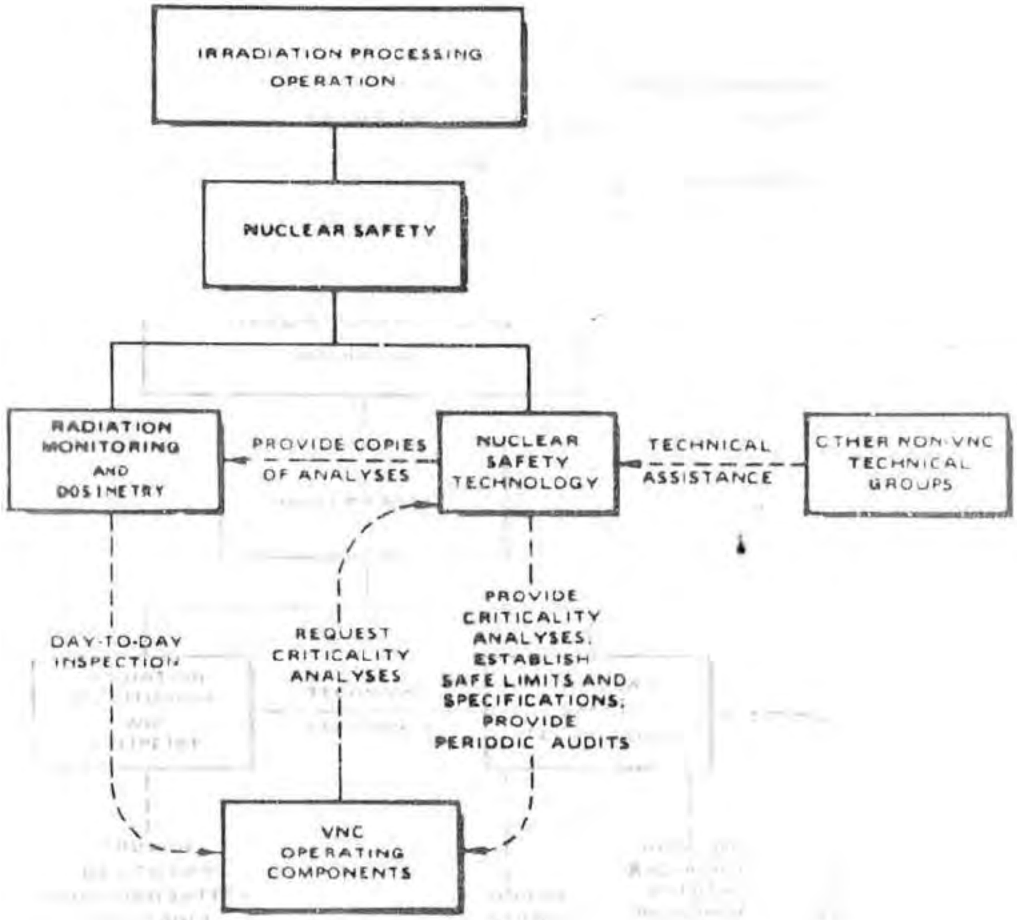

FIGURE 2.4. CRITICALITY SAFETY STRUCTURE

ADDENDUM A

TO

SECTION 2.0

PERSONNEL RESUMES

ADDENDUM A TO SECTION 2.0

RESUMES OF KEY MANAGERIAL AND SAFETY PERSONNEL

A.1 R. W. Darmitzel, Manager, Irradiation Processing Operation; B.S., Chemical Engineering, University of New Mexico, 1958.

Mr. Darmitzel joined the General Electric Company, Vallecitos Nuclear Center (VNC), in 1959 as an engineer responsible for the planning and performance of the post-irradiation examination of nuclear fuel materials. In 1966, Mr. Darmitzel became Manager, Isotope Production and Development, in which capacity he was responsible for the production of radioisotope products and the development of new radioisotope products, particularly for use in the field of nuclear medicine. This position required interaction with such regulatory agencies as the Food and Drug Administration, the Atomic Energy Commission, and the Department of Transportation. Mr. Darmitzel was active in the Atomic Industrial Forum and the American National Standards Institute.

From 1970 to 1975, Mr. Darmitzel became Manager, Radioactive Products and Services, and was responsible for both radioisotope production and the post-irradiation examination of reactor components in support of the Nuclear Energy Division's reactor development programs.

In 1975, Mr. Darmitzel assumed the position of Manager, IPO. He is responsible for the operation of the radioisotope production and post-irradiation examination facilities, and the operation of the Nuclear Test Reactor and its support functions, including Nuclear Safety. Mr. Darmitzel has been designated by the General Managers of the General Electric Divisions represented at VNC as the manager responsible for nuclear safety for the entire site.

A.2 W. H. King, Manager, Nuclear Safety Technology; B.S., Chemistry, University of Missouri, 1953.

Mr. King joined General Electric at its Vallecitos Nuclear Center in 1959 as a Radiological Engineer with responsibilities both at VNC and the San Jose facilities. In this capacity, he participated in the design and development of health physics and related methods and procedures including complete radiation, beryllium, and other health monitoring systems. He has been responsible for the conduct of specific technical investigations related to radiation and toxic chemical hazards including programs involving shielding, ventilation, waste processing and disposal, dosimetry, instrumentation and calibrations, environmental parameters, and studies relative to the toxicity of various chemicals on biological systems and the movement of radioactivity in humans.

During this period, Mr. King established numerous training programs and authored or co-authored numerous publications relative to dosimetry, instrumentation, calibrations, sources, and considerations of various isotopes in the biosphere.

In 1966, Mr. King transferred to Systems Engineering at the VNC site where he participated in the design, development and testing of engineered safeguards for reactor and laboratory systems as well as developing and placing into operation complete monitoring systems for criticality detection and site effluent control.

Following this assignment, Mr. King spent approximately one year in Technical Marketing at VNC and then became the Licensing Administrator for the VNC site.

In 1970, Mr. King was appointed Manager, Nuclear Safety Technology, in the site nuclear safety organization. In this capacity Mr. King has managed the technical personnel responsible for site radiation, criticality, and reactor safety.

Following various assignments, Mr. King was appointed Manager, Nuclear Safety and Quality Assurance, in 1981. Although retaining Nuclear Safety, Quality Assurance, and Safeguards functions, the title of the component was changed to site Nuclear Safety.

A.3 G. E. Cunningham, Senior Licensing Engineer, B.S., Physics, Louisiana State University, 1958.

In 1958, Mr. Cunningham joined the General Electric Company in Richland, Washington, under the Technical Graduate Rotational Training Program. From 1960 to 1966 while with General Electric in Richland, he was an engineer in the health physics group becoming responsible for all technical aspects of health physics for a chemical reprocessing facility. His responsibilities included exposure records, development of administrative and technical procedures, developing bioassay schedules, liaison work with other site components and with the Atomic Energy Commission, dosimetry studies, and environmental surveillance.

From 1966 to 1967, Mr. Cunningham was employed by Isochem, Incorporated, Richland, Washington. His work chiefly involved an investigation into the merits of Cobalt-60 and Cesium-137 for commercial irradiation application.

In 1967, Mr. Cunningham joined General Electric at VNC as a Criticality Specialist, providing criticality work and consultation for other General Electric sites and backup capabilities for site health physics.

In 1971, Mr. Cunningham became responsible for licensing and liaison activities between VNC and the various regulatory agencies.

From March 1975 to October 1975, Mr. Cunningham assumed the additional responsibility for the establishment of the VNC Physical Security and SNM Accountability Program. Since October 1975; he has reassumed his duties as the site licensing administrator.

Mr. Cunningham is a past member of the Board of Examiners of the American Board of Health Physics.

A.4 W. R. Lloyd, Senior Engineer; B.S., Mechanical Engineering, Stanford University, 1955; M.S., Mechanical Engineering, Stanford University, 1956.

In 1956, Mr. Lloyd was employed at Servomechanisms, Incorporated, as an engineer with prime responsibility for the design and development of electromechanical transducers for aircraft control systems.

In 1957, he joined the General Electric Company at the Flight Propulsion Laboratory Department in Cincinnati, Ohio, as a design engineer responsible for the preliminary design of a closed-cycle nuclear turboprop engine. In 1960, he was appointed Heat Transfer Research Engineer, alkali metal heat transfer research, where he contributed to the design and construction of one boiler and two condenser test sections for a high temperature, two-loop alkali-metal system.

Mr. Lloyd joined the Allison Division of General Motors Corporation in 1963 as a Senior Research Engineer with prime responsibility in the area of NaK-Argon two-phase flow investigation.

In 1966, Mr. Lloyd joined the General Electric Company Nuclear Thermionic Power Operation at VNC as an engineer. His prime duties in this position were the heat transfer, hydraulic, and stress analyses of core and system components, and the definition of certain fuel venting system design criteria for these plants.

Mr. Lloyd began his present assignment as engineer in Nuclear Safety in 1970. Prime duties involve the review of experiments prior to irradiation in the site test and research reactors, risk evaluation of certain business ventures and customer proposals, and technical analysis of special problems in the operation of the reactors.

In 1975, Mr. Lloyd was appointed criticality analyst for the VNC site. In this capacity, Mr. Lloyd is responsible for specifying certain features in nuclear fuel processing, storage and on-site transportation so that all operations are safe from a criticality viewpoint. Such operations include co-precipitation of mixed oxide powders from plutonium nitrate and uranyl nitrate, fabrication of ceramic pellets from these powders, and processing to obtain fission product molybdenum from dissolutions of uranium-aluminum alloy fuel.

A.5 B. M. Murray, Radiological Engineer; B.S., Mechanical Engineering, Chico State College, 1963.

Mr. Murray has had 21 years of experience in nuclear safety and operations activities at the Vallecitos Nuclear Center. He has expert capabilities in radiological engineering and risk evaluations, and his most recent assignment includes radiological engineering responsibilities for both the VNC and San Jose sites and Nuclear Safety Compliance Engineer for site activities licensed under State of California License 0017-59.

From 1963 to 1965, Mr. Murray's assignments on a training program included Nuclear Safety operations and technical analysis, BWR operations, and test reactor operations. He received a General Electric Test Reactor Operator's License in 1964.

From 1965 to 1974, Mr. Murray held the positions of Facility Engineer, Product Manager, and Isotope Production Supervisor in the Remote Handling Operation. His assignments included equipment design, project managing, sealed source fabrication, supervision of processed isotope production crew, and facility planning and modification.

Mr. Murray's assignments as Nuclear Safety Engineer during the period 1974 to the present time include: technical risk evaluations, computer-aided shielding calculations, internal and external dosimetry measurement and evaluation, radioactive effluent system evaluations and release limits, hypothetical accident evaluations, facility operating license application input preparation, stack monitoring system evaluations and recommendations, operating standards and procedures preparation and review, health physics instruments evaluation and specification, and review of proposed site activities. He became a California Registered Professional Mechanical Engineer in 1974, and he attended and completed course work for the Health Physics Society Summer School On Internal Radiation Dosimetry, Principles and Practices in June, 1983.

A.6 P. S. Webb, Specialist, Radiological Control Training; B.S., Texas Technical University, 1967; M.S., Biophysics (Health Physics), Texas A&M University, 1970.

Mr. Webb has had 12 years of experience in nuclear safety activities at the Vallecitos Nuclear Center. His most recent assignments have involved managing a radiological and environmental monitoring program, and structuring and conducting various courses for a radiological training program at GE-VNC. Currently, he is serving as Training Program Administrator and Respiratory Protection Program Administrator. Previous training assignments include training of radiation protection managers and health physicists for Hatch Nuclear Power Plant in Georgia, Duane Arnold Nuclear Power Plant in Iowa, and Taiwan Power Company in Taiwan, as well as nuclear power plant maintenance and refueling workers at the Oyster Creek Nuclear Power Plant near Tom's River, New Jersey. Additionally, Mr. Webb has expert capabilities in bioassay, internal and external radiation dosimetry, radiation monitoring instrumentation, and computer methods and applications.

From 1971 to 1978, Mr. Webb held the position of Radiological Engineer and was responsible for reactor health physics for on-site test reactor and off-site boiling water reactors, radiation protection in radioisotopes and plutonium laboratories, and training of customer utilities' reactor health physicists.

From 1978 to 1982, Mr. Webb held positions as Manager, Radiological and Environmental Protection; and Specialist, Radiological Training. He is a member of the Health Physics Society (HPS) and is a member of the HPS Subcommittee for Internal Dosimetry Standards--Plutonium Work Group. In addition, he is a member of the International Society for Respiratory Protection.

3.0 RADIATION PROTECTION FACILITIES AND EQUIPMENT

Permanent laboratory buildings and other nuclear facilities make up the Vallecitos Nuclear Center complex. The buildings and facilities are designed, equipped and maintained to perform work with radioactive materials in a manner providing a high degree of safety for persons employed at the laboratory and for residents of surrounding areas. Typical major equipment and facilities available for the protection of health, life, and property are set forth in this section.

3.1 PORTABLE MONITORING INSTRUMENTS

Monitoring instruments from the following list are available in adequate supply to provide for essential monitoring and for scheduled calibration and maintenance.

Portable Monitoring Instrumentation

	Instrument Type	Range
1.	GM Detector	0-500 k cpm, beta-gamma
2.	Ionization Chamber (low energy)	0-300 mrad/h, beta-gamma
3.	Ionization Chamber (CP)	1-250 k mR/h, gamma 4-1,000 k mrad/h, beta
4.	Ionization Chamber (gas multiplication)	1-1,000 k mR/h, gamma 20-20,000 k mrad/h, beta
5.	Geiger Tube (telescopic)	1-1,000 k mR/h, gamma
6.	Scintillation Counter - Sodium Iodide (Tl)	0-500 k cpm, gamma
7.	Neutron Rem Meter (BF_3)	0.5-5,000 mRem/h, neutron
8.	Alpha Survey Probes (gas proportional and ZnS)	200-1,000,000 dpm, alpha
9.	Portable Air Samplers	0-8 cfm

3.2 FIXED MONITORING EQUIPMENT

Listed below are types of equipment installed for monitoring quantities or concentrations of radioactivity:

Air samplers and monitors utilizing GM, proportional, scintillation, and semiconductor detection modes with moving and fixed filtering units which are capable of alarming at air concentrations equivalent to MPC's in less than four hours for most of the commonly encountered radioisotopes. Fixed filter units consist of 2-inch diameter millipore filters and constant flow control regulators. Stack sampling and monitoring units include isokinetic probes with GM, proportional, scintillation, semiconductor and/or flow-through ion chambers and appropriate filter media.

Fixed gamma monitors with ranges from 0.1 mR/h to 10^6 R/h are located in areas with potentially hazardous gamma fields.

Hand-and-shoe counters are provided at principal exit points for beta-gamma and alpha as required.

A whole body counter (shadow shield principle) utilizing a 5-in. x 5-in. NaI crystal is capable of detecting 0.01-0.1 of the maximum permissible body burden of several common gamma emitters and 1% for most other gamma emitters.

Environmental surveillance is provided by a number of TLD dosimeters located on the VNC site and at its perimeter. Four permanent environmental air sample stations also are located on site.

3.3 PROTECTIVE CLOTHING

Protective clothing is provided to assure the necessary protection of personnel. The amount and type of protective clothing required for a specific activity or area is determined on the basis of contamination potential. Available protective clothing includes caps, laboratory coats, coveralls, boots, overshoes, shoe covers, gloves, and respiratory protection, either filtered or independent air supplied types. Other special protective equipment is available for use from time to time.

3.4 FILM BADGES AND POCKET DOSIMETERS

Film badges are worn where there is potential for radiation exposure. In addition to beta-gamma sensitive film, each film holder also contains the neutron measuring materials indium and gold, both unshielded and cadmium-shielded, and a sulphur pellet. Neutron sensitive film or albedo neutron dosimeters are used where appropriate. Pocket ionization chamber dosimeters may be used. If pocket dosimeter results indicate an off-scale or unexpectedly high reading, the badge is processed; and if these results are confirmed, the circumstances are investigated and the individual is removed from radiation work, if appropriate. TLD extremity dosimeters are worn when radiation exposure to the hands is expected to exceed total body exposure to a significant degree. Film and TLD dosimeters are serviced by a commercial vendor.

3.5 RADIOACTIVE WASTE FACILITIES

The wastes containing the most significant quantities of special nuclear material are items such as irradiated fuel specimens which have been examined and analyzed at the Building 102 hot cell complex. Some of the wastes are delivered to a licensed waste contractor while others are stored at the Hillside Storage complex. Other radioactive wastes ordinarily contain only gram or milligram quantities of special nuclear materials. Facilities used for waste storage and handling at the site are specified in the sections which follow.

3.5.1 Dry Wastes

Dry contaminated wastes are placed in sealable drums, tubes, boxes, or casks available at each laboratory or facility where such wastes may be generated. Each laboratory or facility maintains a designated area for temporary waste storage. Dry wastes are transferred to Building 102 for final inspection and any necessary repackaging. Waste packages then are transferred to the site radioactive material storage facility, Hillside Storage.

The facility consists of two parts: (1) a vertical well facility for storage of 55-gallon drums of waste, and (2) a horizontal tube facility for storage of high level radioactive material contained in sealed encapsulations called "waste liners". The vertical well facility consists of 96 holes -- 4 groups of 24 -- made by vertically stacking two 3-foot-long sections of 36-inch outside diameter concrete pipe with 3-inch-thick walls and with the top lip about 3 inches above grade. Each group of pipes is spaced in two separate rows of twelve each with about 1 foot between pipes and rows. Compacted earth filled in around the pipes keeps them firmly in place. Wooden lids and the 3-inch berms minimize entrance of water. This facility is not used for wastes containing any significant amounts of SNM.

The horizontal tube facility is made of two rows of 40-foot-long concrete-lined steel pipes mounted horizontally and covered with earth. Eleven of the tubes have a 6-inch inside diameter, and seven have a 10-inch inside diameter. The tubes in either row are spaced on 3-foot centers, and the rows are spaced 3 feet apart with the tubes in the bottom row offset halfway between the tubes in the upper row. Shielding is provided on the top and sides of the facility by a minimum of 6 feet of compacted earth. Shielding at the front and back ends consists of 3 feet of concrete in which the pipes have been anchored, plus concrete-filled step plugs with a minimum of 3 feet of concrete shielding in the plug. Additional above-ground space for lower level waste or other materials is available within this fenced and posted facility.

3.5.2 Liquid Wastes

Liquid wastes are routed from laboratory sinks and gravity drains leading from sources known to be or potentially contaminated through regulated pipe lines to retention tanks located in each building where such wastes are generated. Such wastes are transferred periodically to a waste treatment plant for concentrating and solidifying the liquid wastes which are described in Section 13 of this application.

Other liquid waste (excluding sanitary waste) flows through a separate piping system into any three of four 60,000-gallon retention basins. After sampling and determining that radioactive contamination, if any, is within permissible discharge levels, the water in the basin is released.

Sanitary wastes are treated, and the waste waters are sprinklered on site.

3.6 EMERGENCY EQUIPMENT

A vehicle is available to Radiation Safety and can be quickly equipped with a supply of protective clothing, first aid equipment, respiratory protection equipment, and portable instrumentation and sampling equipment for use as an emergency vehicle. Emergency equipment also is stored in selected areas on site.

3.7 INDUSTRIAL SAFETY EQUIPMENT

In conjunction with the radiation safety program at VNC, industrial health and safety of VNC personnel also is emphasized. Some of the major protection facilities and equipment which are available include portable extinguishers, sprinkler systems, a fire truck with water tank and pumps, a dispensary attended by a registered nurse, and a wide range of typical industrial safety equipment.

3.8 CRITICALITY ALARM SYSTEMS

In any Vallecitos Nuclear Center area in which special nuclear material containing more than 500 grams of U-235 or 300 grams, computed by adding the weight of any U-233 and any plutonium to 0.6 times the weight of any U-235, is handled, used or stored and does not otherwise qualify as a "subcritical area" as defined in Section 3.14 of Appendix A, a monitoring system, including gamma or neutron sensing devices which will energize an audible alarm in the event of criticality, is installed and maintained. The system in use on site is described in the following paragraphs.

3.8.1 LiI Crystal System

This monitoring system consists of a lithium iodide crystal coupled to a photomultiplier tube. Failure of any detection circuit component which would prevent criticality detection activates a warning light on the unit. Failure of any signal-producing component is detected during the monthly test.

The alarm level of the system is pre-set at not less than 5 mRem/h nor more than 500 mRem/h (this setting is an exemption from 10CFR70.24 contained in License SNM-960). The system is capable of energizing the alarm when the radiation level at a distance of 1 foot from the special nuclear material is 300 R/h or 2.1×10^6 neutrons per square centimeter per second.

Sensing devices are positioned within 120 feet, air equivalent, of every required location where special nuclear material is handled, used or stored. The system is tested by exposing the detectors to appropriate sources and sounding the alarm monthly. The alarm system is designed so that the alarm continues to sound until reset. The alarm is clearly audible in all locations where radiation exposure may result from an accidental criticality incident. If a facility does not have emergency backup power, all movements of SNM are suspended during a power failure.

3.8.2 Excepted Areas

Criticality sensors are not provided below the surface of the water in the RML storage pool or within the RML cells, nor in the horizontal tube solid waste facility. Shielding surrounding the special nuclear material is as follows:

RML Storage Pool	16 feet of water
RML Hot Cells	1.5 to 3.0 feet of high density concrete
Horizontal Waste Storage	5.5 feet of compacted earth

Calculations indicate that in the unlikely event of accidental criticality in any of these facilities involving 10^{17} fissions, the maximum exposure dose would be less than the quarterly permissible whole body dose of 1.25 Rem. The detector above the RML dry storage pit, because of the shielding, would only detect a dose rate of about 1,500 R/h at 1 foot from the source when the shield plugs are in place. Therefore, for purposes of meeting the requirements of 10CFR70.24, the source of the criticality shall be taken as the building floor surface (the top of the shield plugs).

4.0 RADIATION PROTECTION PROCEDURES

A system of Vallecitos Safety Standards establishes the site radiation and criticality protection and regulatory compliance programs. The manager of the nuclear safety function issues the standards with review and comment of the managers of the major organizational components located on the site.

Currently, there are about 50 standards dealing with radiation protection matters. The principal features of these are summarized below.

4.1 PERSONNEL WORK RULES

Requirements are established to prevent or minimize the hazards of radioactivity and radioactive materials. Eating, storing, or preparing food is not permitted in radioactive materials areas. Smoking or storing tobacco is not permitted in areas where there is a potential for contamination with radioactive materials. Food containers may not be used for storing or handling radioactive materials.

General Electric furnishes protective clothing for service in areas where contamination is likely to contact personal clothing. Protective clothing standards are set by the site radiation safety function to assure effective quality, positive identification, and to avoid use for other than its intended purpose. The amount and type of clothing for any specific activity is assessed on the basis of potential for personnel contamination.

4.2 LIMITS OF RADIATION IN CONTROLLED WORK AREAS

All Vallecitos locations where there is a potential for radiation exposure are classified (radiation area, high radiation area, etc.) in accordance with the definitions of 10CFR20, Sections 20.202 and 20.203. General Electric's philosophy of protection is to keep radiation exposure at the lowest reasonably achievable level in all cases. Thus, design working limits substantially more conservative than the maximum permissible limits recommended by the National Committee on Radiation Protection are used as objectives of the radiation protection program. In no case may the calendar quarter limits of 10CFR20.101(a) be exceeded without written permission of the appropriate unit manager prior to the exposure and completion of data required by 10CFR20.101(b)(3). In no case may the whole body exposure during a calendar quarter exceed 3 Rems. Managers of facilities are responsible for compliance with the above exposure limits. Work requiring skills of multiple organizational components is implemented by means of the Radiation Work Permit (RWP) procedure. Work in controlled areas for which the necessary radiation controls remain essentially constant for extended periods of time may be governed by extended RWPs which set forth requirements and precautions to assure adequate radiation safeguards. The RWP is approved by the worker's direct supervisor and the supervisor of the area in which the work will be performed and by a person authorized to monitor and set time limits. The RWP is effective in preventing overexposure by assuring dose rates are known by monitoring measurements prior to the work. Employees who expect to receive a significant radiation dose maintain a current exposure estimate so that doses received by employees are known during film badge wearing periods. Exposures to dose rates in excess of 3 Rem per hour require appropriate management approvals as set forth in the site safety standards.

With respect to operations which could produce airborne radioactive contamination, managers of facilities are responsible for providing ventilation equipment to meet the concentration limits of 10CFR20.103 without the necessity, or credit, for personal respiratory devices during routine operations. In certain nonroutine situations where adequate ventilation equipment would be impractical or could not ensure control of airborne material, respiratory protection of demonstrated integrity is utilized.

For example, it occasionally is necessary to completely remove all equipment from a hot cell in the Radioactive Materials Laboratory prior to installation of new and different equipment. When this need arises, the inner cell surfaces are scrubbed by remote means until air samples show no further decrease in radioactive concentration. At that point, contact methods of further cell decontamination are instituted. Personnel entering the cell wear respiratory protection to accomplish this work.

The respiratory equipment currently in use at VNC is approved by the National Institute of Occupational Safety & Health (NIOSH) and, as such, achieves compliance with 10CFR20:

a. MSA Clearvue facepiece or Ultraview facepiece with a NIOSH-approved cartridge, or the MSA Ultra-Twin facepiece with a NIOSH-approved cartridge, or equivalent apparatus may be worn where concentrations do not exceed fifty times the concentrations of Appendix B, Table I. If radioactive iodine or other radioactive gases are present in concentrations in excess of concentrations of Appendix B, Table I, a similar NIOSH-approved respirator may be worn, or an air-supplied airline respirator may be substituted for the air-purifying type.

b. NIOSH-approved plastic hoods with positive pressure, continuous flow air supply may be worn where concentrations may exceed fifty times the concentrations of Appendix B, Table I.

c. U.S. Divers Company's Survivair self-contained breathing apparatus (SCBA) or other NIOSH-approved SCBA having a back pack air supply, hose, harness, pressure demand regulator, and a facepiece may be worn where concentrations exceed the limits of protection provided by air-purifying and airline respirators and for situations in which airborne concentrations have not been evaluated.

VNC has adopted only equipment that is approved by NIOSH.

No individual will be permitted to work more than twenty hours per week under conditions requiring masks. No individual will be permitted to work in a mask until he has received a medical clearance for respirator use and has been thoroughly instructed in methods of proper use, fitting, and field testing of respirators.

After use, masks are surveyed and thoroughly cleaned with detergents or soaps followed by a second survey when dry. No detectable contamination is permitted on the inside of the mask on the second survey, which must be conducted in a background of not more than 500 counts per minute (beta-gamma). Equipment parts and supplies are replaced as necessary by qualified technicians. The equipment is inspected to assure proper maintenance at all times. Masks which cannot be cleaned adequately or on which adequate surveys cannot be made are discarded to radioactive waste.

4.3 PERSONNEL MONITORING

Instructions for the use of film badges, finger TLD dosimeters and pocket dosimeters include the proper part of the body on which the device is to be worn and procedures to prevent spurious readings. In addition, personnel are instructed to use hand-and-shoe counters and portable monitoring instruments upon leaving a radioactive materials area.

In order to determine the extent to which individuals may have internally accumulated any radioactive materials, two bioassay procedures are established. Persons who work routinely with radioactive materials, particularly alpha and beta emitters, are tested by the urinalysis method approximately quarterly but at least annually by this means, or more frequently depending upon the potential for exposure to mechanisms which could cause ingestion, inhalation or absorption of such materials. Special analyses for other radioactive materials are performed whenever a need is indicated.

Persons who work routinely with radioactive materials which are readily detectable gamma emitters may be tested additionally, or alternatively, by the whole body counting technique. A whole body counter is operated by the nuclear safety function. These tests are performed at least annually. More frequent schedules are maintained for personnel who routinely handle radioactive materials. The whole body counter technique provides the advantage of producing results more accurately and more promptly than urinalysis and therefore is used wherever practicable.

The nuclear safety function is advised of all additions to the work force, changes in individual assignments, termination of employees, and any occurrence which may have resulted in internal deposition. This procedure assures that appropriate schedules are maintained for biological assay.

4.4 SURVEYS

Surveys to assure radiation safety are made routinely in order to detect any unfavorable trends or conditions. Special surveys also are conducted as warranted by the suspected or potential presence of radiation or radioactive material. Routine survey schedules are established whenever work programs are initiated and are changed when the work is discontinued or the scope of work is revised. Survey methods are formulated to meet the needs of the particular type of radioactive materials used and in the light of the equipment capability. Original survey reports are retained by the radiation safety function with copies to the supervisor of the surveyed area for information and action as appropriate.

Environmental surveys are made both within the VNC site and at off-site locations. These surveys involve samples of surface and ground water, soil, vegetation, and air and are used to verify compliance with the airborne concentration limits of 10CFR20.

The allowable stack discharge dilution factor, i.e., the ratio of the effluent concentration in a stack to the concentration at the site boundary, for any given discharge stack on site is a function of the actual annual meteorological conditions, the stack flow rate, and the distances from the stack to the site boundaries. Technically, there could be an allowable dilution factor for each radioactive effluent stack on site, but for simplicity and conservatism, the Building 102A stack which has the largest potential source term and the highest flow rate is used to determine a single allowable dilution factor for all SNM-960 licensed stacks on site.

The method of calculating the allowable dilution factor is to:

(1) Collect and process the hourly site meteorological conditions for one or more years.

(2) Using the annual meteorological data, calculate and tabulate a dilution-dispersion factor, χ/Q, and wind direction for each hour at 16 equal sectors of the site boundary.

(3) Tabulate and analyze these data to determine an annual average χ/Q value to the site boundary from the stack of interest. This factor represents the average air concentration (μCi/cc) during one year at a fixed point on the site boundary which would result from a continuous, steady release (μCi/sec) from the stack.

(4) This annual average χ/Q value then can be converted to a dilution factor by multiplying it by the stack flow rate (cc/sec) to form a ratio of boundary concentration to stack concentration. The reciprocal of this value gives the dilution factor as a number which is larger than one.

To calculate the worst case airborne effluent concentrations at the edge of the site in the event of a release, it is assumed that all releases are from ground level during moderately stable (Type F) atmospheric conditions with a one meter per second wind speed. No effective stack height is assumed because of limited stack heights and their proximity to buildings. The wind is assumed to blow toward the point of interest during the release. Using calculational methods and data outlined in Application Amendment No. 48 to Docket 50-18, HW-SA-2809, and Regulatory Guide 1.3, a concentration of 3.6×10^{-9} μc/cm³ will result from an average emission of 1 μc/sec from an 360 meter distance source.

Notwithstanding the above calculations, all effluent air from operations involving potential dust-producing activity is filtered; and, therefore, special nuclear material concentrations have been and are expected to continue to be routinely well below permissible levels for unrestricted locations at the points of discharge from the stacks.

Since the issuance of License SNM-960, data on stack releases have been collected. The actual doses at the site boundary have been calculated based on these data, and the results of the calculations are included in Section 4.0 of the Environmental Information Report for Special Nuclear Material, NEDO-21158, December, 1975.

Materials and equipment are surveyed prior to release from radioactive materials areas. Standards are established for each of two categories: Regulated Release and Unconditional Release. Regulated Release permits the removal of the item for reuse in another area at the site under conditions specified in writing on a tag attached to the item and permits the release of certified containers for off-site shipments. Unconditional Release permits reuse without restriction and requires survey using only the most sensitive detection instrumentation in background not exceeding 500 c/m beta-gamma or 200 d/m alpha. Unconditional Releases must be approved by radiation safety function personnel.

Routine calibrations of survey instruments are performed as follows:

a. Alpha calibrations by use of sources which are traceable to plutonium and uranium standards prepared by the National Bureau of Standards.

b. Beta calibrations by use of a slab of natural uranium in radioactive equilibrium with Th-234 and Pa-234.

c. Neutron instrument calibrations are performed by measuring a response to a pulse generator and to an Am-Be neutron source.

d. Gamma calibrations using Cobalt-60 sources standardized with a meter which in turn was calibrated with traceability to the National Bureau of Standards.

All radiation monitoring instruments are calibrated as frequently as deemed necessary to assure reliability during use. Portable radiation monitoring instruments are calibrated on an annual basis, before initial use, and after repair.

Stack particulate monitor systems are calibrated by placing a uniformly distributed radioactive source in the same geometry as the filter paper used for collecting particulates. Stack gas monitor systems are checked routinely by placing a reference gamma source on the side of the Kanne chamber and observing whether the response falls within prescribed limits. Calibration with a known radiogas standard has been performed to verify this procedure. Iodine monitors also are routinely source checked.

4.5 POSTING AND LABELING

Instructions are established implementing the posting and labeling requirements of 10 CFR Part 20. Additional precautionary signs are utilized to meet special requirements and detailed procedures; for example, "Regulated Lab Coats Required Beyond This Door", "Regulated Sink", etc. The area supervisor is responsible for maintaining the proper posting and labeling.

However, in view of the degree of control inherent in the security of the site and the required prerequisite training program for employees, there is an exemption from the provisions of 10CFR20.203 for high radiation area alarms which General Electric deems should be continued.

An exemption from the alarm provisions of 20.203(c)(2) is requested for any area or location which is:

1. a. Used for the temporary placement of strong, weatherproof containers such as shipping casks not exceeding the limits of 49CFR73.393, i.e., 200 mR/h at any point of readily accessible surface; or

b. An established area in which there may be radioactive materials in containers, the readily accessible surfaces of which do not exceed dose rates of 1,000 mR/h at any time and which do not exceed 100 mR/h at the barrier for any continuous period greater than 30 days, such as radioactive waste pickup areas from which wastes are collected on a weekly or monthly basis; and is

2. Located within the 94-acre inner fenced exclusion area of the site; and is

3. Bounded by a distinctively colored rope or chain providing a barrier at dose rate values not greater than 100 mR/h and posted as a High Radiation Area in accordance with 10CFR20.203(c).

4.6 WASTE DISPOSAL

Detailed procedures for the packaging, storage and removal of contaminated material which no longer is useful are established by operating components and reviewed by the nuclear safety function. The procedures define low, high, and intermediate levels of solid waste on the basis of a contact dose rate at the surface of outer packaging and provide specifications for container packaging to prevent loss of contents, repackaging of damaged units, labeling of contents and similar requirements. Procedures for handling or disposal of liquid wastes in the various waste treatment facilities described in Section 3 are detailed in similar instructions. Area supervision is responsible for adherence to proper handling procedures for obtaining approved containers and for arranging for transportation of wastes to the appropriate site storage or treatment facility.

4.7 REPORTS AND RECORDS

Records of surveys, personnel exposure records, and other records indicating the degree or nature of individuals' exposure are maintained by the radiation safety function. These records currently are retained for an indefinite period.

4.8 INDUSTRIAL SAFETY PROGRAM

The effectiveness of the radiation and criticality safety program is in large measure influenced by a sound industrial safety system, and vice versa. Good housekeeping practices, modern fire prevention equipment, and training in handling toxic or explosive chemicals are but a few benefits of industrial health and safety which carry into the nuclear areas. For that reason, compatible program philosophies and policies are maintained. General Electric places direct responsibility for safety on the individual employee and for directing safe practices on his area manager.

Other important elements include pre-employment medical examinations; medical surveillance of personnel; control of flammables, explosives, and toxic materials; and administration of fire prevention equipment requirements in accordance with applicable codes and regulations.

4.9 VALLECITOS EMERGENCY PLAN

Plans for the prompt and rapid response to emergency situations are set forth in the Vallecitos Nuclear Center Emergency Plan. The plan is revised from time to time to reflect changes in equipment and organization at the site. The plan addresses emergencies which could arise from accidental criticality or release of radiation, as well as from fire, explosion, or earthquake, whether localized at a single facility or of site-wide significance. A Radiological Contingency Plan dated November, 1982, has been submitted to the NRC Office of Nuclear Material Safety and Safeguards.

4.9.1 Nuclear Emergencies

The accidental criticality alarm is a distinct sound which is activated automatically when criticality detection instrumentation in any local area reaches a pre-set limit. Personnel in the affected area immediately evacuate.

At the same time, the origin of the signal is designated at the main security building. Upon determination of the cause of the alarm, the site-wide public address system is used to instruct all personnel at the site concerning further action. Upon the sounding of any alarm, Initial Response Emergency Operations Coordinator (EOC) makes an assessment of the indicated or reported situation and initiates whatever action is deemed necessary to minimize personnel injury and property damage. A dedicated telephone network can be used by the EOC for conference contact with designated VNC site managers representing most fields of technical competence at the site, with the person reporting the emergency, and with the security building.

If the nature and severity of the emergency requires evacuation, either complete or partial, instructions are broadcast.

Building emergency teams have been trained in the use of survey instruments and protective apparel. Guidelines to assist management in formulating dose limits for emergency exposures for the protection of human life, recovery of victims, and protection of health and property are available. A system which has undergone a criticality accident will be left undisturbed until competent review has produced a plan to cope with the situation. Calibrated instruments are available at strategic locations.

The Emergency Operations Coordinator will authorize reentry to an affected facility after safe conditions are restored. Entries for survey or decontamination purposes will be made at his direction in appropriate protective apparel and respiratory equipment.

Injured persons, if any, may be taken to the hospital in Livermore. Approximate travel time to the hospital is 15 minutes. Arrangements have been made with the hospital to receive and care for injured persons who are contaminated, with supervision by competent Nuclear Safety personnel. In the event of spurious alarms, the all-clear will be announced on the public address system. Periodic drills will be conducted to assure adequate personnel response to emergency situations.

The plan includes details for notification of regulatory and civil authorities, responsibilities of the emergency functional organization, responsibility for maintenance of communication facilities and standby equipment and instruments, location of assembly areas, responsibility for visitors, and other specific factors of importance to the effectiveness of the Emergency Plan.

Responsibility for determining the need for revisions of the Emergency Plan, for implementing such revisions, and for coordinating and evaluating periodic evacuation drills is delegated to the Site Emergency Planning Coordinator with the assistance of Nuclear Safety. The site-wide address system is tested weekly.

4.9.2 Nonnuclear Emergencies

Other alarms (automatically and manually operated) are used for fire, explosion, or other local or general nonnuclear emergency. The senior supervising person at a facility will decide when an emergency exists and will take appropriate action for the safety of people and equipment. Where appropriate, Area Managers maintain Building Emergency Procedures which are consonant with the Site Emergency Plan. Emergency crew activities are directed by the EOC.

4.10 RADIATION SAFETY REVIEW

The design of new VNC facilities, or major changes in existing facilities in which radioactive materials are to be handled, must be reviewed by the radiation safety function to insure that adequate health protection facilities are provided. Policy design and operating elements are considered in this review. Materials containment, exposure reduction (both operating and maintenance), design life, and assessments of potential hazards are used in the final design phase whereby such hazards are engineered out.

The Area Manager concerned, or his designated representative, is responsible for initiating requests for radiation safety review of proposed facilities and radiation safeguards.

4.11 HAZARDS ANALYSIS

The consequences of possible accident conditions at VNC have been analyzed, and the results are included in the Environmental Information Report for Special Nuclear Material, NEDO-21158, December, 1975.

5.0 CRITICALITY SAFETY PROGRAM TECHNOLOGY

5.1 INTRODUCTION

Administrative responsibilities for implementing the Criticality Safety Program at the Vallecitos Nuclear Center (VNC) are given in Section 2.0 of this application. The responsibility of qualified personnel to maintain activities at all times within specified limits is emphasized.

In making a criticality analysis, the two-contingency criterion is used: the occurrence of at least two unlikely, independent, and concurrent changes in one or more of the conditions originally specified as essential to nuclear safety is required before a nuclear accident is possible.

The manager of the criticality safety component shall assure that the analyses in which an error in an analysis could lead to conditions that do not fulfill the two-contingency criterion are identified and reviewed. This review shall be performed by qualified persons other than the person making the original analysis. The neutron physics advice function may provide technical review of the computation methods.

An organizational component competent in neutron physics and the technical methods for criticality computations shall verify the use of any new computational methods for criticality control.

In the discussion below, information about computational tools useful for performing criticality analyses at VNC is presented. Sources of nuclear parameter data useful in performing hand calculations are discussed, and a computer code useful in generating nuclear parameter data is described. Information is presented on the more powerful Monte Carlo criticality codes. Critical experiments useful in evaluating the computational tools for criticality analyses are identified, and results obtained in using the computational tools to calculate the critical experiments are presented. Any negative biases that must be applied are identified. Finally, the adjustment factors for normal operating conditions are presented.

5.2 MINIMUM CRITICAL QUANTITIES AT ISOLATED LOCATIONS

When only small quantities of special nuclear materials are required at any one time within an isolated facility, accidental criticality hazards are easily and definitely avoided by control of inventory to quantities substantially lower than the minimum quantities of such materials that have been experimentally found to be capable of criticality under optimum laboratory conditions. The literature* gives these minimum critical quantities as approximately:

820 grams U-235
570 grams U-233
531 grams Plutonium

Therefore, on this basis, activities are safe without consideration of factors other than administration of inventory controls to quantities not greater than:

500 grams U-235, or
300 grams U-233, or
300 grams Plutonium, or
any pro-rated combination thereof.

As used in this section, the term "isolated" means physically separated from other areas where special nuclear materials are used. The procedures for accounting for inventories of SNM normally will be under specifically designated custodians. The minimum neutronic isolation barrier is taken as the equivalent of eight inches of water or a distance of 12 feet, or the greatest distance across an orthographic projection of either accumulation or array on a plane perpendicular to a line joining their centers.

*ARH-600

5.3 DETERMINATION OF CRITICAL PARAMETERS OF ACCUMULATIONS AND ARRAYS OF FISSILE MATERIAL

5.3.1 Experimental Critical Data

Experimental critical data are available in the literature from which critical parameters for fissile materials such as mass, sphere volume, cylinder diameter, slab thickness, and concentration may be determined. Data are often presented so that values of critical parameters may be read directly from curves or tables. Some documents from which critical parameters may be obtained are:

a. Karsruhe Symposium papers, 1961;
b. A.H.S.B. Handbook 1, "Handbook of Criticality Data";
c. TID-7028, Critical Dimensions of Systems Containing U-235, Pu-239 and U-233;
d. TID-7016, Rev. 1 and Rev. 2, Nuclear Safety Guide;
e. LAMS-3067, Los Alamos Critical Mass Data;
f. LAMS-2537, Correlations of Experimental and Theoretical Critical Data;
g. Y-1272, Y-12 Plant Nuclear Safety Handbook;
h. ARH-600, Criticality Handbook;
i. Oak Ridge Criticality Data Center Report, ORNL-CDC-5;
j. DP-1014, "Critical and Safe Masses and Dimensions of Lattices of U and UO_2 Rods In Water".

When the geometry of the accumulation is other than a simple one, data obtained from this literature may be used to determine critical parameters by using simple buckling conversion methods or by using approximate, but conservative, values for the accumulation dimensions.

5.3.2 Computational Tools

Computational tools include hand calculations, the General Electric Boiling Lattice (TGEBLA) Code, and Monte Carlo criticality codes such as MERIT and

KENO-IV in SCALE. A brief description of each of these computational tools is given below.

a. Hand Calculations

Hand calculations are performed with the one-group approximation formula:

$$K_{eff} = K_{\infty}/(1 + M^2B^2)$$

where:

K_{eff} is the k-effective multiplication factor;

K_{∞} is the infinite multiplication factor for the fuel mixture under consideration;

M^2 (cm^2) is the migration area for the fuel mixture under consideration;

B^2 (cm^{-2}) is the geometrical buckling.

Values for the parameters K_{∞} and M^2 and the extrapolation lengths necessary to calculate the geometric buckling along with formulas for geometric buckling for simple geometric shapes are obtained from the ARHCØ Criticality Handbook, ARH600*.

The nuclear properties in ARH600* are primarily for homogeneous mixtures of uranium and plutonium with water. However, the effect of clumping of uranium fuel for enrichments below 5 w/o U-235 in uranium must be considered.

In DP-1014**, results from critical experiments and calculations for critical masses of homogeneous and heterogeneous UO_2-H_2O mixtures are given for U-235 enrichments ranging between 0.8 w/o and 5 w/o

*Carter, R. D., et al; "Criticality Handbook", ARH-600, Volumes 1-3, Atlantic Richfield Company; Richland, Washington; September 25, 1975.

**Clark, H. K., "Critical and Safe Masses and Dimensions of Lattices of U and UO_2 Rods In Water", DP-1014, Savannah River Laboratory; February, 1966.

U-235 in uranium. These results show the critical mass to be smaller for the heterogeneous mixtures than it is for homogeneous mixtures. For example, at a 5 w/o enrichment, the minimum heterogeneous critical mass is given as 1.56 kg U-235 for a 0.1-inch O.D. rod; the corresponding homogeneous minimum critical mass is given as 1.85 kg U-235. Similarly, for a 2 w/o enriched uranium, the heterogeneous minimum critical mass is given as 3.99 kg U-235 for a 0.4-inch O.D. rod; the homogeneous minimum critical mass is given as 6.98 kg U-235.

In DP-1014, values for migration area, material buckling, and extrapolation distance for full water reflection are given as a function of uranium concentration for both homogeneous and heterogeneous mixtures of $U-H_2O$ and UO_2-H_2O for uranium enrichments ranging from 0.8 w/o to 5 w/o U-235 in uranium. Values of k-effective for clumped, low-enriched $U-H_2O$ mixtures and UO_2-H_2O mixtures may be calculated using this information.

The General Electric Boiling Lattice Code, TGEBLA, also is used to generate the nuclear properties k_∞ and migration area for UO_2-H_2O heterogeneous mixtures (NEDE-32002*, "TGEBLA Lattice Physics Methods"; December, 1982). In the thermal energy range in TGEBLA, the rod-by-rod thermal spectra are calculated by a method similar to the THERMOS** formulisms. The major difference is that neutron leakage from rod to rod is taken into account in TGEBLA.

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*Submitted to the Office of Nuclear Reactor Regulation, NRC.
**Honeck, H. C., "THERMOS: A Thermalization Transport Code for Reactor Lattice Calculations", BNL-5826 (1961).

b. The MERIT Code

The MERIT* Criticality Code stems from the Battelle Monte Carlo Code (BMC). The BMC was converted by General Electric's Nuclear Energy Business Operation for use on the Honeywell 6000 and Control Data Corporation 7600 Computers. Several improvements were incorporated during this conversion. The MERIT Code is a neutron transport Monte Carlo Code and is designed primarily for use in calculating nuclear parameters associated with thermal reactors. However, it is not limited to this application. In general, MERIT can solve the eigenvalue (k-effective) fixed source and neutron shielding problems. The neutron flux, isotopic reaction rates, group average cross sections and leakages are calculated in three-spaced dimensions over the energy range from 0 to 10 MeV. The reaction types considered are fission, capture, inelastic, n-2n scatter, elastic scattering with isotopic and anisotopic angular distributions, and thermal elastic scattering. The energy distributions are continuous; however, the cross sections can be averaged over up to 2,000 microscopic energy groups. The isotopic material cross sections currently used with MERIT are processed from the ENDF/B-IV tapes and have a 190-group energy structure.

Improvements of the BMC code currently available in MERIT include the incorporation of a cross section library which interpolates cross sections as a function of temperature and the effective potential scattering cross section if they contain resonances; the incorporation of an improved representation of the ENDF/B Haywood scattering model for the thermal scattering by hydrogen in water with an interpolation routine which would interpolate the kernel to any temperature between 293°K and 1,000°K; the addition of infinite

*Kang, C. M., et al; "MERIT, A Monte Carlo Neutron Transport Program", NEDE-21377, 76 NED 34, Class II, General Electric Company; San Jose, California; August 1, 1976.

rectangular pipe boundary and box boundary options in the geometry routines; the modification of the code to allow neutron splitting and make the importance weighting and Russian Roulette parameters easier to input; the addition of a simple plotting routine to check the geometry input; and an improvement in the source routine to allow user specification of sources in a square array of rods.

c. The SCALE System

The SCALE* System that is running on a Control Data Corporation 7600 Computer at General Electric's Nuclear Energy Business Operation contains:

1. A number of stand-alone computer codes referred to here as functional modules for neutronic analysis that includes criticality analysis and shielding analysis;
2. A number of computer codes noted here as control modules which read simplified sets of input data and invoke one or more of the functional modules in a preestablished sequence to perform a specific type of criticality or shielding analysis;
3. A driver package which interfaces with the CDC System to provide the software and operating environment in which various control modules may be executed;
4. A number of data libraries containing nuclear cross section data and material property data.

The control modules read a simplified set of input describing a given problem, perform a number of auxiliary calculations formerly required by the program user, and call the necessary functional modules sometimes in an iterative fashion to achieve a desired solution.

*Bucholz, J. A., "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation", NUREG/CR-0200, ORNL/NUREG/CSD/2, Volume 1, Oak Ridge National Laboratory; July, 1980.

The Criticality Safety Analysis Sequence 2 (CSAS2) in the SCALE System is used to calculate the effective neutron multiplication factor (k-eff) for multidimensional systems which can be described in the KENO-IV geometry. An optional one-dimensional calculation in CSAS2 allows the user to 1) describe a unit cell in the fuel assembly, 2) perform a one-dimensional eigenvalue calculation of the unit cell to determine the spatially dependent flux spectrum, 3) cell-weight the microscopic cross section data with this spatially dependent spectrum, 4) homogenize the nuclide number densities in the unit cell (fuel assembly) and then, 5) use these homogenized cell-weighted cross sections in a subsequent multidimensional KENO-IV calculation. This analysis sequence includes two cross section processing codes, NITAWL and BONAMI; a one-dimensional transport code for cell-weighing cross section data called XSDRNPM; and a three-dimensional Monte Carlo code called KENO-IV for calculating the effective neutron multiplication factor (k-eff) for the entire system.

In the calculational sequence defined by CSAS2, the master cross section library selected first will be processed by BONAMI, which will perform a resonance self-shielding calculation for those nuclides which have Bondarenko data in lieu of resonance parameters. Dancoff factors are determined in DANCOF for lumped absorbers for the resonance self-shielding calculation. The self-shielding data for these nuclides, along with the original data for all the other nuclides then will be stored on-line in the same format as the original master library. Data for nonresonance nuclides and data for nuclides with resonance parameters will be copied onto this library essentially unchanged from the original library. The NITAWL code then will read the second master library, perform a resonance self-shielding calculation for those nuclides having resonance parameters, and collect results into a working library which may be used by the XSDRNPM and/or KENO-IV codes. This working library has the same number of energy groups and the same group structure as the original master library selected by the user. Unlike the original master library, it contains the self-shielded, group-averaged cross section data for the given physical situation. XSDRNPM is a one-dimensional discrete-ordinates transport code used by CSAS2 to obtain cell-weighted microscopic cross sections. To obtain the spacially dependent flux spectrum, XSDRNPM will perform a one-dimensional eigenvalue calculation for the unit cell. To ensure good resolution, the control module automatically will determine the amount of spatial mesh intervals to be used in each material zone. The cell-weighted working library produced by XSDRNPM then will have the same number of groups and the same group structure as the original master library selected by the user. Cross sections for the user-defined mixtures not found in the fuel assembly are effectively copied from the NITAWL working library to the XSDRNPM working library without change.

KENO-IV is a multigroup Monte Carlo code used by CSAS2 to determine the effective neutron multiplication factor of the multi-dimensional system specified by the user. KENO geometry is a three-dimensional geometry and allows for the simultaneous use of cuboids, spheres, hemispheres, cylinders, hemicylinders, and embedded arrays of such bodies.

At present, three cross section libraries have been assembled for use in the SCALE System on General Electric's CDC 7600 Computer. These include a 16-group cross section set based on earlier Hansen-Roach data, a 27-group cross section set collapsed from a 218-group cross section set based on ENDF/B-IV data, and a 123-group cross section set based on earlier GAM-THERMOS data.

5.4 CRITICAL EXPERIMENTS FOR COMPUTATIONAL TOOL EVALUATION

Uranium with enrichments varying from natural to fully enriched and plutonium are the fissile materials that need to be considered in the validation of VNC's computational tools. The forms include fully enriched homogeneous uranium-water mixtures, low-enriched heterogeneous uranium water mixtures, and homogeneous plutonium-water mixtures. Since the VNC license limit for Uranium-233 is 200 grams and a minimum of 570 grams is required for criticality without the use of supernormal moderators, validation of computational tools for use with U-233 is judged to be not necessary.

In Table 5.4 below, critical experiments suitable for validation of computational tools as based on the various combinations of fissile materials that may appear in each designated VNC facility are identified.

TABLE 5.4

Critical Experiments for
Computational Tool Evaluation

Experiment Name	References
A. TRX-1 & TRX-2 Low-Enriched Uranium Rods in Water	A.1. J. Hardy, Jr., D. Klein and J. J. Volpe; "A Study of Physics Parameters in Several Water-Moderated Lattices of Slightly Enriched and Natural Uranium", WAPD-TM-931; March, 1970.
	A.2. J. Hardy, Jr., D. Klein and J. J. Volpe; Nucl. Sci. Eng. 40, 101 (1970). J. J. Volpe, J. Hardy, Jr., and D. Klein, Nucl. Sci. Eng. 40, 116 (1970).
	A.3. J. Hardy, Jr., D. Klein and R. Dannels; Nucl. Sci. Eng. 26, 462 (1966).
	A.4. J. R. Brown et al., "Kinetics and Buckling Measurements in Lattices of Slightly Enriched U or UO_2 Rods in H_2O", WAPD-176 (January, 1958).
	A.5. R. Sher and S. Fiarman, "Studies of Thermal Reactor Benchmark Data Interpretation: Experimental Corrections", EPRI NP-209; October, 1976.

TABLE 5.4 (Continued)

Experiment Name	References
B. ORNL 1-4 & ORNL 10 Fully Enriched Uranium Spherical Solutions	B.1. R. Gwin and D. W. Magnuson, "Eta of U-233 and U-235 for Critical Experiments", Nuc. Sci. Eng. 12, 364 (1962).
	B.2. A. Staub et al., "Analysis of A Set of Critical Homogeneous $U-H_2O$ Spheres", Nuc. Sci. Eng. 34, 263 (1968).
C. PNL 1-5 Plutonium Spherical Solutions	C.1. R. C. Lloyd et al., "Criticality Studies With Plutonium Solutions", Nuc. Sci. Eng. 25, 165 (1966).
	C.2. L. E. Hansen and E. D. Clayton, "Theory-Experiment Tests Using ENDF/B Version II Cross-Section Data", Trans. Amer. Nuc. Soc. 15, 309 (June, 1972).
	C.3. F. E. Kruesi et al., "Critical Mass Studies of Plutonium-Nitrate Solution", HW-24514 (1952).
D. Babcock & Wilcox Small Lattice Facility Low-Enriched UO_2 Rods in Water	D.1. M. N. Baldwin et al., "Physics Verification Program - Part III", BAW-3647-6, Babcock & Wilcox, 1970.
MO_2 Rods in Water	D.2. G. T. Fairburn et al., "Pu Lattice Experiments In Uniform Test Lattice of UO_2-1.5% PuO_2 Fuel", BAW-1357, Babcock & Wilcox; August, 1970.

5.5 K-EFFECTIVE RESULTS FOR COMPUTATIONAL TOOL EVALUATION

K-effective results are presented for the ORNL, PNL, TRX, and B&W critical experiments in Table 5.5. Some comments on hand calculations follow.

The nuclear properties k-infinity, migration area, and extrapolation distance were taken from curves in ARH-600. If data were not available for the precise enrichment in the experiment, the data that were available for the next higher enrichment were used. For example, for ORNL-1 and ORNL-2, data for 93.5 w/o U-235 were used. For PNL-1 and PNL-2, data for 0 w/o Pu-240 were used. Also, ORNL-2 was poisoned with boron; this poisoning was not accounted for in the hand calculations.

For the hand calculations and the TRX-1 and TRX-2 experiments, the nuclear properties material buckling, migration area, and extrapolation distances were taken from Appendix C of DP-1014. An enrichment of 1.4 w/o U-235 in U was used compared to a 1.3 w/o U-235 in the TRX-1 and TRX-2 critical experiments. Further, linear interpolation of the values in Appendix C of DP-1014 was used to obtain the nuclear property values.

The conclusions are:

Use of the SCALE System using the 27-group cross section set requires no bias for fully enriched uranium solutions and for plutonium solutions; a negative bias of 2.5 w/o in the multiplication factor must be applied for low-enriched, clumped uranium rods in water.

The MERIT criticality code is adequate for determining the multiplication factor for all of the conditions in Tables 5.4 and 5.5 without application of a bias. The hand calculations, when conservatively applied, do not require application of a bias in the determination of the multiplication factor.

TABLE 5.5

K-Effective Results
Computational Tool Evaluation

Name of Critical Experiment	Feature	SCALE Results: K-eff ± 2 Sigma	SCALE Results: No. of Neutron Histories	MERIT Results: K-eff ± 2 Sigma	MERIT Results: No. of Neutron Histories	Hand Calc. Results K-eff
ORNL-1	Fully Enriched	1.0021 ± 0.0060	30,000	0.9911 ± 0.0056	50,000	1.000
ORNL-2	U (235) Nitrate	0.9977 ± 0.0068	30,000	0.9933 ± 0.0092	50,000	1.051
TRX-1	Low Enriched	0.9773 ± 0.0060	30,000	0.9998 ± 0.0026	150,000	1.014
TRX-2	U (235) Rods	0.9820 ± 0.0060	30,000	0.9983 ± 0.0030	150,000	1.019
PNL-1	Plutonium	1.0157 ± 0.0108	30,000	1.0194 ± 0.0110	50,000	1.034
PNL-2	Nitrate (5Pu240)	1.0105 ± 0.0114	30,000	1.0143 ± 0.0120	50,000	1.142
B&W	UO_2 Rod	0.9920*± 0.0046	30,000	0.9937*± 0.0292	50,000	--
B&W	MO_2 Rod	0.9972*± 0.0054	30,000	0.9936*± 0.0320	50,000	--

*k-infinity values

5.6 ADJUSTMENT FACTORS FOR NORMAL OPERATING CONDITIONS

Criticality analyses shall utilize calculative methods that have been evaluated by the calculation of models of critical experiments of similar geometrical configuration and material content; the maximum negative bias in the multiplication factor, defined as $(-\Delta k_{bias}) = k_{calculation} - k_{experiment}$, determined in the evaluation shall be applied to the analysis as given in the relationship below. The multiplication factor of the accumulation or array of accumulations shall satisfy the relationship:

$$k_{cal} + \Delta k_{uncertainty} + (-\Delta k_{bias}) \leq 0.95$$

where:

k_{cal} is the calculated multiplication factor by the calculative method, and $\Delta k_{uncertainty}$ is the statistical uncertainty (2σ) in the mean value of the calculated multiplication factor, and $(-\Delta k_{bias})$ is the maximum negative bias determined from the validation of the calculative method. If the bias is positive, the $(-\Delta k_{bias})$ term shall not be used, so that overstatements in the multiplication factor shall not be corrected.

Normally, subcritical values of nuclear parameters for individual accumulations under normal operating conditions shall be:

1. For accumulations limited by mass, the normally subcritical mass shall not exceed 90% of the critical mass.

2. For accumulations limited by volume, the normally subcritical volume shall not exceed 76% of the critical volume.

3. For accumulations limited by dimension, the normally subcritical cylinder diameter or slab thickness shall not exceed 90% or 88%, respectively, of the critical dimension where U-235 is the fissile constituent, nor 85% in all other cases.

4. For accumulations limited by concentration, the normally subcritical concentration shall be less than that concentration which is equivalent to the following atomic ratios of hydrogen to fissile material: H/U-235, 5200; H/U-233, 7600; and H/Pu (fissile), 7600.

5. Not more than 45% of the critical mass shall be contained in any accumulation in a criticality limit area in which double batching is credible.

Normally, subcritical values of nuclear parameters of interacting accumulations shall be:

1. Not more than 69% of the critical number of accumulations whenever the critical number has been determined by experiment; or

2. Not more than 77% of the critical number of accumulations in any array calculated by the density analog method (Ref: ARH-600); or

3. The number of accumulations determined by the solid angle method as described in TID-7016, Rev. 2, Paragraphs 4.29 through 4.34. For plutonium or uranium systems of more than five percent enrichment, the method may be applied only to accumulations that are well moderated (e.g., aqueous solutions) at the point of the maximum multiplication factor, and array reflection must be no more effective than a thick water reflector spaced at distances from the accumulations comparable to the spacing between accumulations.

6.0 LABORATORY BUILDING 102

6.1 LOCATION AND GENERAL DESCRIPTION

Building 102 is a single story with basement, concrete and steel structure located as shown in Figure 6.1. The predominant features of the building are the multikilocurie facility known as the Radioactive Materials Laboratory (RML) and the Core Materials Engineering Laboratory (CMEL) facilities. General support laboratories for these facilities also are located in the building. The current layout of the building is shown in Figures 6.2 and 6.3. The laboratory areas of the main floor are separated from general office areas by a fire wall. Inherent in the building design are general service facilities such as the main ventilation system, decontamination rooms and the like which are described later in this section.

6.2 VENTILATION SYSTEM

6.2.1 Air Supply

Inlet air to Laboratory Building 102 is provided by 30 air conditioning units furnishing modulated, filtered and tempered outside air to the operation and office areas. The capacity of the inlet air system is a nominal 65,000 cubic feet per minute.

6.2.2 Direction of Flow

Airflows in the laboratory are from areas of low radioactivity toward and through areas of higher radioactivity. The arrows in Figure 6.2 indicate general direction of flow. The system of directional airflow minimizes the possibility of accidental contamination of nonradioactive areas. Airflows are all single pass and continuous except a portion of the RML Operating Gallery air which is recirculated through absolute filters.

The operating areas are supplied at rates of 6 to 40 changes per hour. A minimum of 6 air changes per hour is being exhausted from the RML cell spaces. This amount of airflow to the cells is adequate for control of contamination. Air conditioners provide up to 11,000 cfm to the Core Materials Engineering Laboratory facilities.

6.2.3 Pressure and Flow Rates

Small static pressures (-0.01 to -0.03 inches of water) are used in the isolation of thin-walled general work areas, while static pressures from 0.02 to 0.20 inches of water are used to isolate the RML cells. The flow diagram as shown in Figure 6.2 indicates the airflow. Glove box contamination control is effected by maintaining from 0.5 to 1.0 inches of water negative static pressure. At this static pressure, air velocities of 125 linear feet per minute or greater are achieved through openings, an adequate rate to provide the necessary control.

6.2.4 Filtration and Exhaust

The main exhaust equipment is located in Building 102A (Figure 6.1). During regular working hours and any other times radioactive material is handled, four of six exhaust fans are used to provide a normal exhaust flow of 68,000 cfm with the other two fans in a standby status. At other times, two fans are employed. Exhaust air from Building 102 is routed to Building 102A via overhead ducts. At Building 102, this exhaust air is passed through 90 HEPA filters and then discharged to the atmosphere through a 66-inch diameter, 75-foot high stack.

Exhaust air from the RML Hot Cells is prefiltered at the outlet of each cell and is routed to a filter bank of 10 HEPA filters in the Building 102 basement before it joins the main exhaust stream at Building 102A. Two activated charcoal filters are available as needed. Two booster fans (one operating and one on automatic standby) are provided as required to aid in the transfer of the RML Hot Cell effluent from Building 102 to Building 102A.

Exhaust air from the glove boxes and hoods located in the Core Materials Engineering east Laboratory is filtered first through high efficiency filters located on each glove box or hood and then is passed through high efficiency filters in the laboratory exhaust duct before joining the main Building 102 exhaust stream. Exhaust air from glove boxes and hoods located in other CMEL areas is filtered at the glove box or hood.

The filter banks and the exhaust connections, at their point of origin, utilize high efficiency filters. The filter system is 99.95% efficient for 0.3 micrometers diameter homogeneous particles of dioctyl sebacate. The filters are constructed of fire-resistant materials and are housed in noncombustible duct work.

The exhaust connections have adjustable dampers where necessary to provide the capability for balancing flows throughout the system.

Figure 6.4 is a schematic diagram of the ventilation exhaust system.

Fire protection is provided to the exhaust system by: (1) a CO_2 suppression system and a water fog system for the Hot Cell basement exhaust filter bank; (2) a water spray system for the Hot Cell charcoal filters; and (3) a water fog spray for the main exhaust ducting from Building 102. None of the water suppression systems can discharge into facilities (e.g., Hot Cells, glove boxes) containing SNM.

6.2.5 Emergency Exhaust

In the event of the loss of utility power, emergency power is provided by a 335 kW diesel-driven electric generator. The generator is activated automatically by power loss and will reach full capacity within 1 minute. All critical equipment normally supplied with normal power through the main building switchboard will be supplied with emergency power. This equipment includes two exhaust fans, fans supplying air to areas where radioactive materials are handled, and the main stack monitoring equipment.

The generator is supplied with fuel from a 1,000-gallon storage tank. At an estimated fuel consumption rate of 20 to 25 gallons per hour, this supply should be sufficient for at least 40 hours of continuous, full load operation.

The building alarm panel and exhaust fans are supplied with normal power through the Motor Control Center (MCC). The solid-state logic portion of the MCC is a solid-state, battery-powered computer that provides for automatic switching of equipment during transitions between normal and emergency conditions. During normal and emergency power operations, it performs the following functions automatically:

1. If one of the operating fans should fail, it will provide an audible and visual alarm, identify and shut down the failed fan, and start a standby fan. The control panel will indicate which fan has failed.

2. In case a fire is identified by a detector in the exhaust ducting, the computer shuts down two of the four operating fans, and the exhaust system operates on reduced flow until the "fire" condition is cleared manually.

6.2.6 Stack Monitoring

The Building 102 stack monitoring system was designed for the monitoring of the primary isotopes released during normal operations in Building 102 and 102A. The system has been designed to detect beta particulate, gamma particulate, alpha particulate, halogens, and gaseous activities. Each detector output signal is transmitted to a recorder and to a central process computer which processes the data into a desired format and displays information on a teletype printout. The flow schematic of this system is shown in Figure 6.5.

6.2.6.1 Alpha Monitor

The alpha monitor continuously collects airborne particulates on a 47-mm Millipore-type filter, determines the amount of a pre-selected alpha-emitting isotope (Pu-239), and processes and presents the information for evaluation and recording. This continuous monitor will measure 1 MPC of Pu-239 in the presence of nominal amounts of radon-thoron daughters when sampling rate of 3 cfm is held constant for a period of 4 hours.

This monitor is an integrated system composed of an air particulate sample collecting mechanism, a solid-state alpha detector, and modular electronics. Its single-channel analyzer has an adjustable base line and an adjustable tracking-type window. Pulses passing through the window of the single-channel analyzer are fed simultaneously to a recorder and the central process computer.

An adjustable background compensator and an alarm delay have been added.

6.2.6.2 Noble Gas Monitor

The flow-through ion chamber is a detector to measure activity in gases. The test gas flows between the outer pair of electrodes called the "scrubber" to which a potential is applied. Here, ionization from previous decays is cleared from the gas. The uncharged test gas then passes into the 16.3-liter active volume. The ions produced by any radioactive decay occurring in the active region drift to the collecting electrode. The resulting current is measured by the electrometer. The sample flow rate is 3 cfm.

The response for Xe-133 is 1.45×10^{-7} amp/µCi/cc.

6.2.6.3 Iodine Monitor

The iodine monitor is utilized to evaluate the total amount and the rate of collection of radioactive iodine collected in the activated charcoal or silver zeolite cartridge filter in a sample air stream withdrawn from the effluent stack. The front of the filter cartridge is monitored by a gamma-sensitive NaI(Tl) crystal integrally mounted on the face of a photomultiplier tube. Pulses generated by this detector are preamplified and transmitted via cable to its respective single-channel analyzer.

The sample flow rate is 1.5 cfm. The monitor is calibrated for I-131 using a Ba-133 standard. The resulting efficiency is 6 to 8%.

A second (unmonitored) cartridge is connected in series with the monitored cartridge to assure maximum collection of iodine.

6.2.6.4 Beta-Gamma Monitor

The beta-gamma particulate monitor is utilized to evaluate the total amount of beta particulate activity and the total amount of a pre-selected gamma-emitting isotope in a sample air stream withdrawn from the stack. The sampled air (1.5 cfm) passes through a 47-mm (Millipore-type filter) which will remove approximately 99.9% of all particulates greater than 0.3 micrometers in diameter.

The beta-gamma activity collected on the particulate filter is monitored by a 2-inch diameter pancake-type GM detector having a window thickness of 7 to 8 mg/cm^2 which is located approximately 1/2-inch from the face of the filter.

The back of the filter is monitored by a gamma-sensitive NaI(Tl) crystal integrally mounted on the face of a photomultiplier tube. Pulses generated by this detector are preamplified and transmitted via cable to its respective single-channel analyzer.

The GM detector portion is calibrated with a Cl-36 beta source for which the efficiency is approximately 35% (4π), and the NaI crystal portion is set to selectively monitor Co-60 for which the average efficiency is 1 to 2%.

6.2.6.5 Permanent Records

The filters and cartridges described in Sections 6.2.6.1 through 6.2.6.4 are sent to the site counting room for analysis. These analyses provide the basis for the permanent stack effluent emission records.

6.3 DECONTAMINATION ROOM

A decontamination room is available for cleaning tools and equipment. Equipment in the decontamination room includes standard filtered airflow-type hoods connected to the building ventilation system, a large stainless steel sink draining to the building radioactive waste system, and miscellaneous other items of equipment typical of that used for decontamination work. The room is posted as a high radiation area and equipped with an alarm device in accordance with 20.203(c)(2).

6.4 AIR SAMPLERS

The air in various parts of laboratory rooms, offices and other occupied areas is sampled on a continuous basis. Approximately 98 sample stations (18 in the CMEL area and 80 in the RML and Radiochemistry areas) have been provided. Air is drawn through each sample station filter paper. Flow rates are routinely determined using standard equipment for such determinations. Filter papers are removed from the stations on a routine schedule and counted for alpha and/or beta-gamma activity. Continuously monitored, alarming air sample stations also are provided in the RML and CMEL areas.

In addition, spot check air sampling is done whenever a new operation is introduced to assure absence of significant contamination in the breathing air; and several continuous air monitors are in use for alpha, beta-gamma, and iodine determination. Alarms are sounded if concentrations increase significantly.

6.5 LIQUID WASTE HANDLING

Sinks and gravity drains in Building 102 radioactive materials areas are routed to a filter which removes particulates. The liquid then is pumped to a 3,000-gallon tank. The tank periodically is emptied, and the waste liquid is sent to the liquid waste evaporator facility to be concentrated, solidified, and transferred to a commercially licensed waste disposal contractor. This liquid waste material contains only minute quantities of SNM.

High-level liquid wastes and those originating from CMEL operations are solidified at the work stations for transfer to the waste contractor.

License No. SNM-960 Docket No. 70-754 Sect. No. Figure 6.1 Page 6-9

Date 3/28/84 Amends Sect.(s)

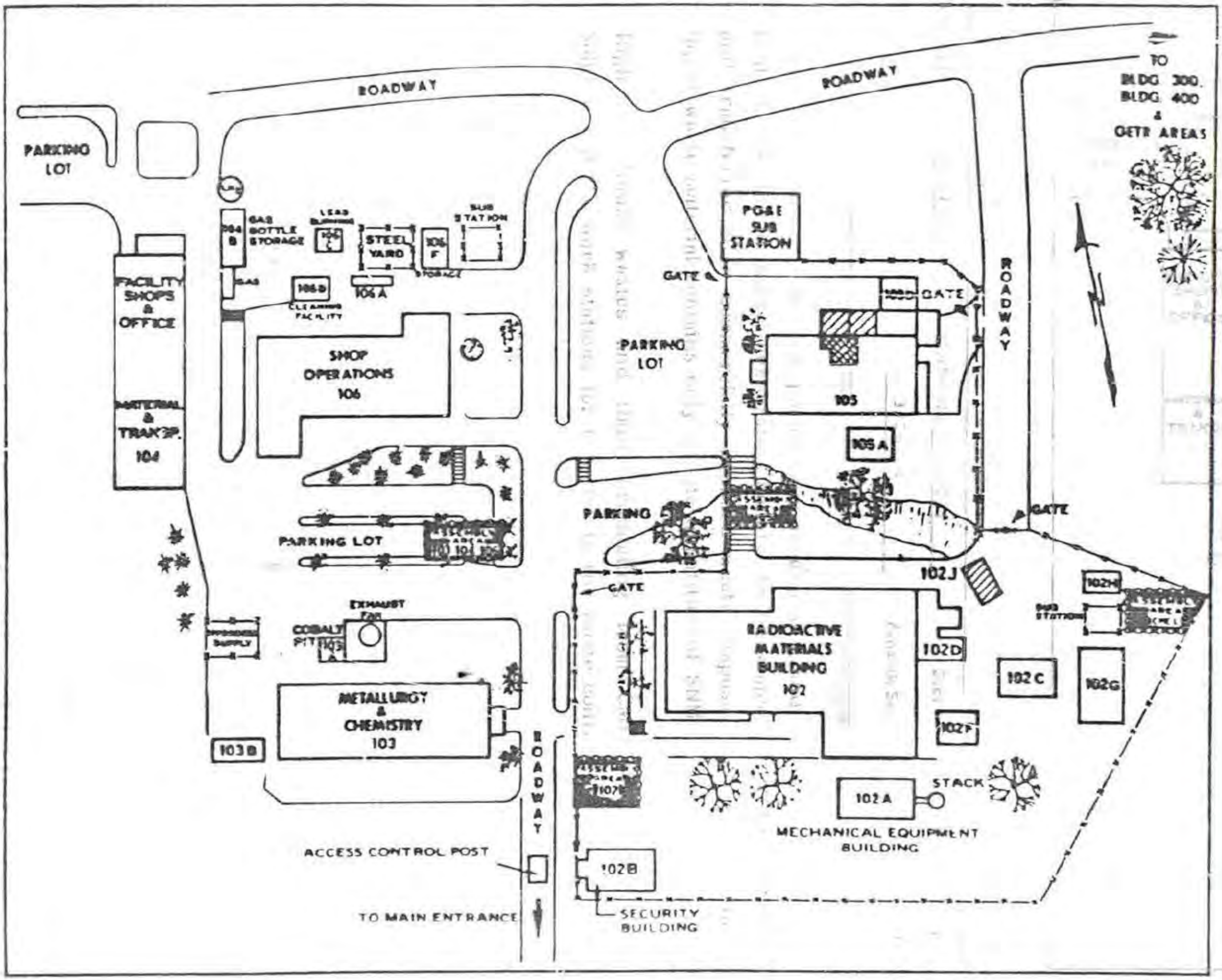


FIGURE 6.1. 100 AREA AND SURROUNDINGS

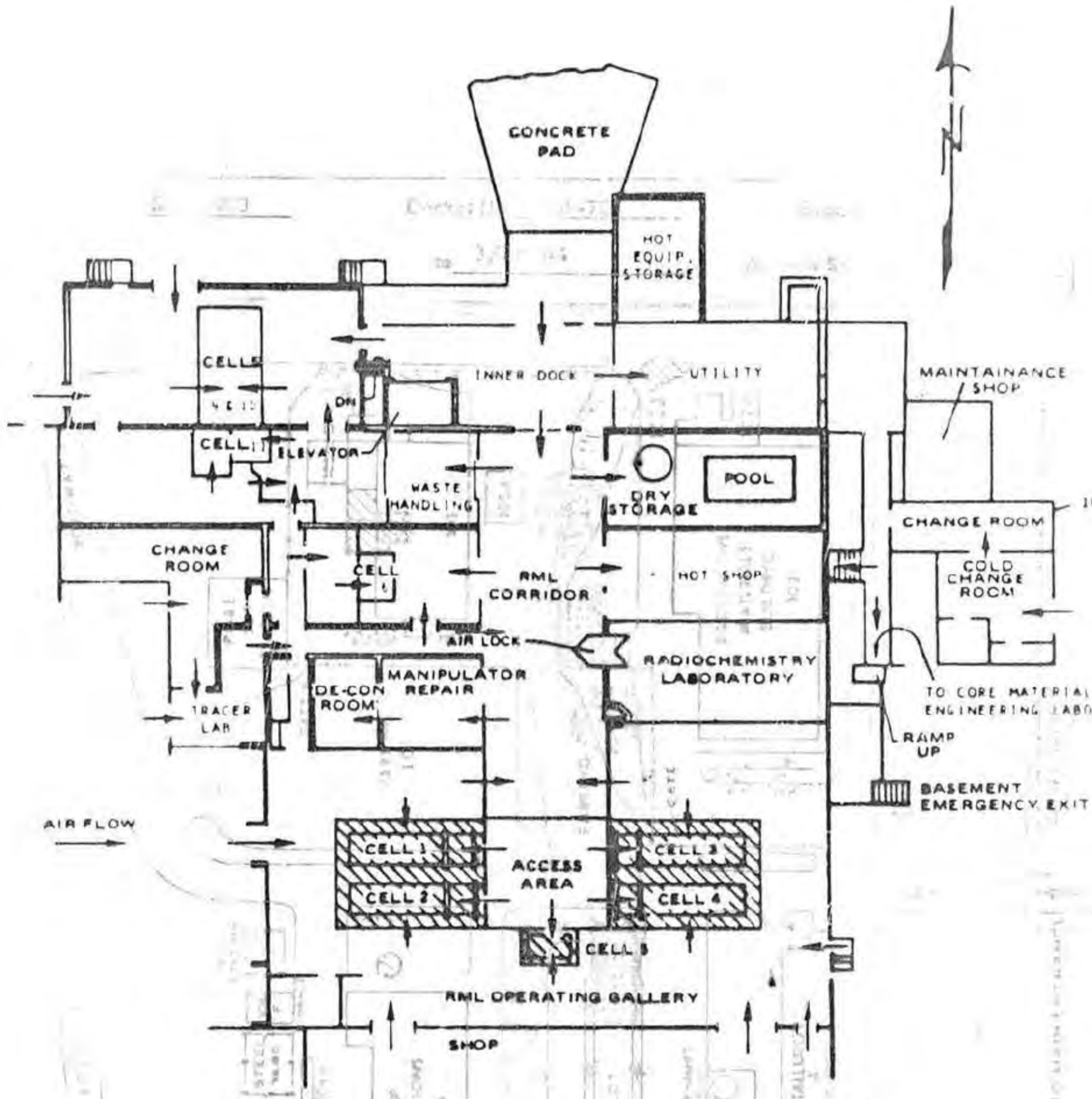


FIGURE 6.2. BUILDING 102 MAIN FLOOR – RADIOACTIVE MATERIALS AREAS

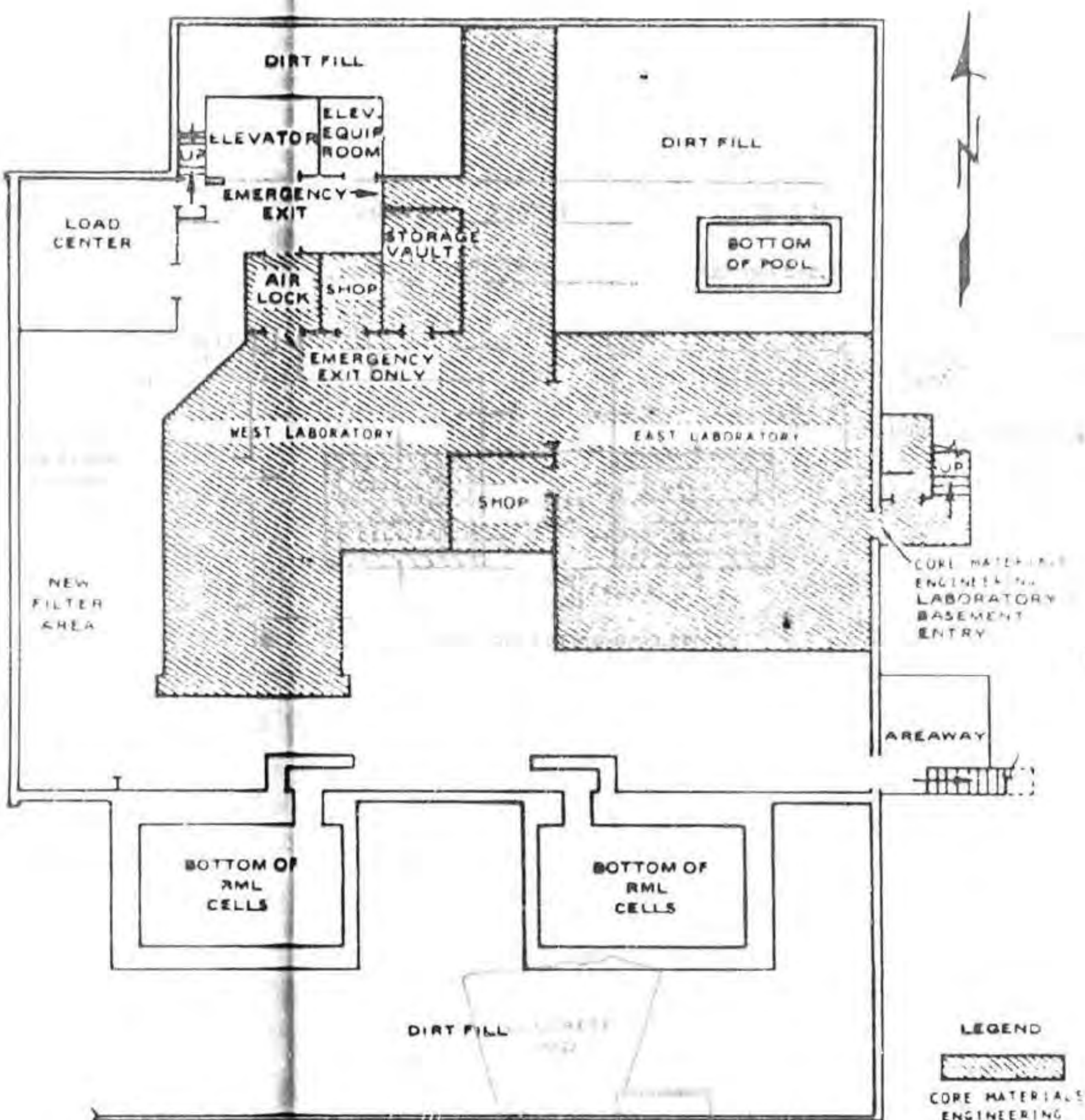


FIGURE 6.3. BUILDING 102 BASEMENT

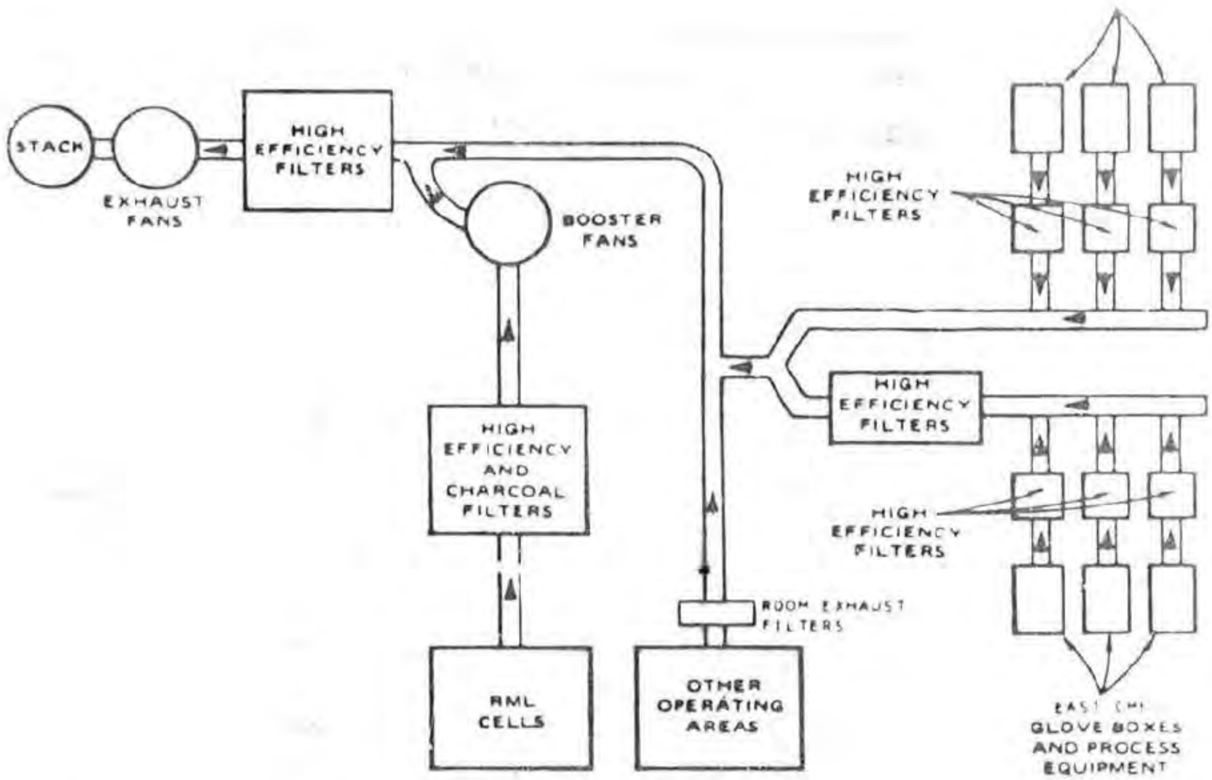


FIGURE 6.4. BUILDING 102 EXHAUST SYSTEM SCHEMATIC DIAGRAM

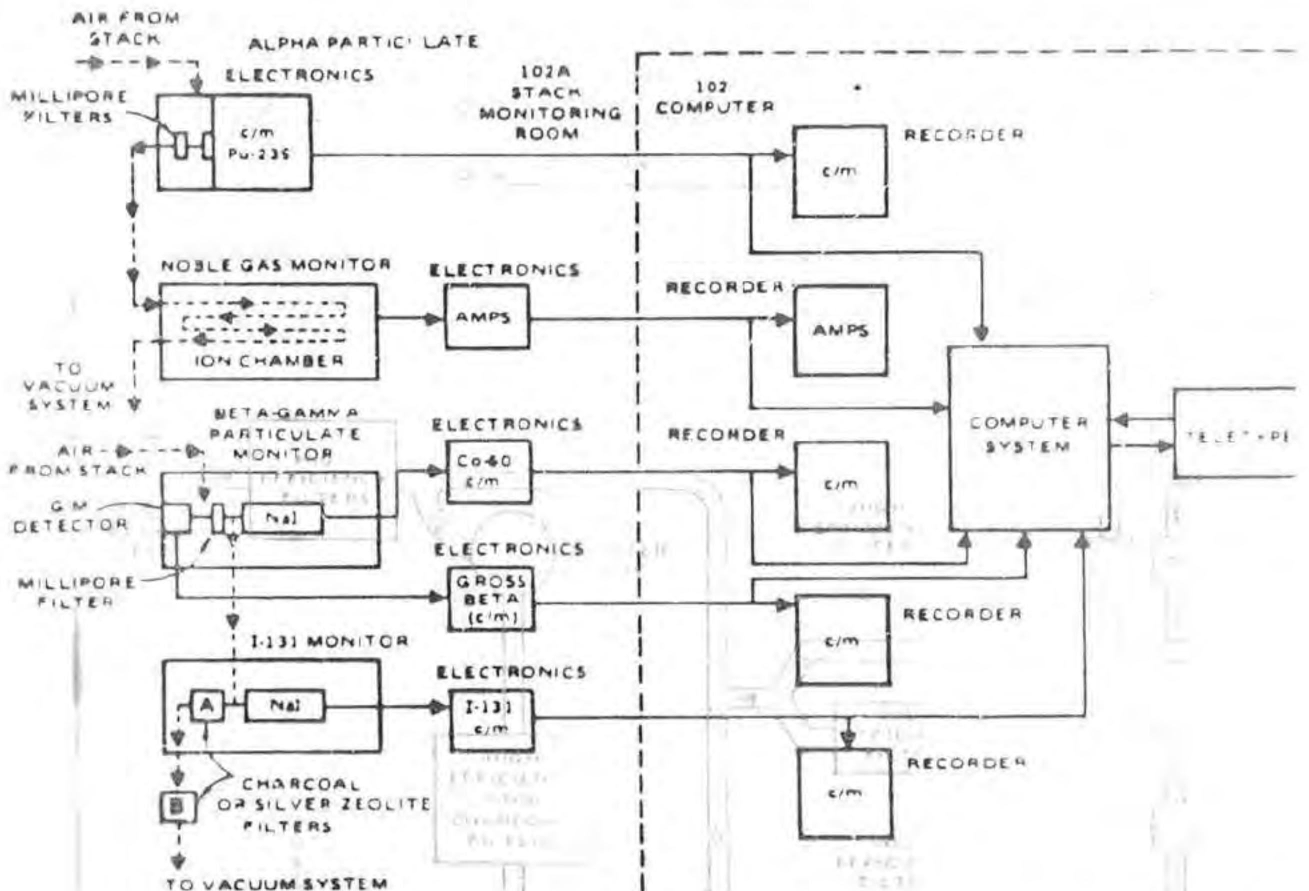


FIGURE 6.5. BASIC STACK SAMPLING SYSTEM FOR BUILDING 102

7.0 RADIOACTIVE MATERIALS LABORATORY

7.1 LOCATION AND GENERAL DESCRIPTION

The Radioactive Materials Laboratory (RML) is the term applied to a series of shielded facilities located in Building 102 used for the examination of irradiated reactor fuels and irradiated hardware and radioisotope production and including support and service areas. A plan of the RML area is shown in Figure 6.2. Although there are a number of hot cells in the RML complex, special nuclear material licensed under License SNM-960 is handled predominantly in Hot Cells 1, 2, 3, 4, and 5; and, accordingly, only those cells will be addressed in detail in this section. A sketch of these RML cells is included as Figure 7.1.

7.2 GENERAL PLANS AND USES OF MATERIALS

RML provides a high-level shielded facility in which safe and efficient nonnuclear testing and examination of irradiated reactor fuels, materials and components may be conducted. The facility also is equipped for the separation of specific byproduct radionuclides and the handling of large quantities of byproduct materials.

7.3 RML FACILITIES

7.3.1 General Description

Equipment and facilities of the Radioactive Materials Laboratory are classified into several major areas: the cell operating area; the general laboratory and cell service corridor; and the cell area. All areas are on the main floor of Building 102 as shown in Figure 6.2. These areas are described more fully in the sections immediately following.

7.3.2 High-Level Cells (Hot Cells 1, 2, 3, and 4)

The cell area is the heart of the Radioactive Materials Laboratory as it is here that work on irradiated material is performed. A double or twin cell block for high-level work is located on either side of the south end of the service corridor, constituting a total of four high-level cells. Each of these cells can contain safely in excess of one million curies of 1 MeV gammas. The high-level cell walls are 36 inches thick up to a height of 12 feet, where they are reduced to 24 inches to take advantage of reduced shielding requirements and to provide a set-back for overhead manipulator rails. The shielding material is a high-density concrete made with ferro-phosphorous aggregate with a poured weight of 300 pounds per cubic foot. Shielding in the vertical direction is provided by the 3-foot-thick concrete roof above the cells.

Each cell is equipped with an overhead bridge-mounted manipulator and a 3-ton bridge crane running on the same rails. The units are designed to traverse the entire length of the cell and the radiation lock and can be moved out of the cells on removable extension rails. This equipment provides handling capability for transfer of irradiated materials in heavy shielded casks.

The viewing window at each operating station is a 3-foot-thick, lead glass window with glass of sufficient density to give shielding equivalent to the cell wall.

Each of the high-level cells as shown by Figure 7.1 has a 6-foot-long radiation lock for entry from the access corridor. These radiation locks are formed by hydraulically operated bi-parting steel shielding doors. The outer door is 18 inches thick, and the inner door is 15 inches thick. A 3.5-foot-diameter by 9-foot-deep pit is provided in the radiation lock floor to accommodate very large casks and permit long irradiation assemblies to be withdrawn. The access corridor width and floor loading specifications were based on the use of a 15-ton capacity fork lift truck in this area. An intercell transfer system is provided in the common wall between each pair of cells.

Each high-level cell has an operating area approximately 17 feet long by 6.5 feet wide and is 14 feet high inside. There are four operating stations in each cell. Three of the stations are located along the side wall of the cell and the fourth at the end wall.

There is a total of 16 operating stations available. Typically, the elements of each cell operating station are a pair of through-wall master-slave manipulators, a viewing window, and the "in-cell" work or experiment apparatus required for carrying out one or more functional tests or operations. Six to 7 feet of width are available at each station internally and externally. This permits adequate space for one or two operators and the miscellaneous control and operating equipment which is required on the cold side.

Water is piped to the hot cells by an independent supply system. A 25-gallon vented storage tank located on the top of Cell 1 is filled by a water line not directly connected to the tank. This tank feeds an adjacent 42-gallon pressure tank through a connecting check valved pump line which, in turn, supplies pressurized water at a maximum of 60 psi to a process water header with branches to each hot cell. This independent system provides positive assurance against feedback of cell water to the potable water system as well as preventing large quantities of water (more than 42 gallons) to enter the cells in the event of piping failure. This system currently serves only Cells 1, 2, and 5.

7.3.3 Cell Door Interlock System

The high-level cells are equipped with inner and outer steel shielding doors which form a radiation lock. The cell doors are hydraulically operated and controlled by a panel at the cell operating face. The operating controls are interlocked such that the outer door normally cannot be opened when the inner door is open. A key-locked override switch is provided for unusual circumstances such as cell decontamination activities.

7.3.4 Cell 5

A smaller cell designed for conducting metallographic work occupies the south end of the main access corridor. This cell is used primarily for the preparation of samples for metallographic examination and micro-hardness testing. Remotely operated equipment for this cell includes sample mounting, polishing, cleaning, and etching equipment as well as a remotely operated micro-hardness tester. Metallographic samples after polishing and etching can be checked with the optics of the hardness tester before being transferred remotely out of the cell to a modified research metallograph contained in a shielded enclosure. The working area of the metallography cell is approximately 5.5 feet deep, 8 feet wide, and 8 feet high. The front wall contains two 18-inch by 13-inch lead glass windows for direct observation of the working area. The cell walls are 18-inch-thick magnetite concrete. The back of the cell, which opens onto the access corridor, contains a safe-type door for equipment and personnel access. Metallographic or micro-hardness test samples are introduced to this sample entry port by means of a special transfer cask.

7.3.5 Pool Facility

A 16-foot by 8-foot by 16-foot deep water pool is available for underwater transfer, examination, repair, assembly or disassembly, and storage of irradiated materials. Overhead crane facilities are available for cask handling in the pool or on the floor of the room. A filtered exhaust port is located adjacent to the pool. Dose rates f the surface of the pool are of the order of 20 mR/h at maximum storage capacity.

The pool is equipped with portable types of tools including tongs, hooks, tables, racks, and lights. The water is circulated at about 30 gpm through strainers and a tank containing 6 cubic feet of nuclear grade ion exchange resin, thereby maintaining excellent water clarity and decontamination. Resin is replenished as necessary. Spent resins are removed to waste drums for disposal

7.3.6 In-Floor Dry Storage Pit

Located immediately adjacent to the RML Storage Pool, a dry storage pit is available for temporary storage of irradiated fuel assemblies, rods or other rod-shaped materials. This storage device consists of 19 recessed pipes fabricated of 6-inch Schedule 40 steel pipe 46.5 inches long attached to two horizontal circular steel plates 91.25 inches in diameter and 6 inches thick. The device rests on a ledge cast in the concrete floor so that the top surface is flush with the floor with the tubes extending downward below the floor surface. The pipes are arranged to provide a minimum center-to-center spacing of 18 inches. Twelve-inch-thick stepped plugs provide shielding of sources up to 500 R/h at 1 MeV to dose rates of 2.5 mR/h.

Fuel materials are placed in sealed tubes called waste liners prior to insertion to the storage pipes. Movement of the material is accomplished by means of shielded transfer casks which have top and bottom entry ports. Handling tools which are attached remotely to the waste liner can permit insertion and withdrawal of stored objects without encountering direct radiation from the top of the open cask. Each tube has two 1-inch-diameter holes located 3 inches above the bottom and one 1/4-inch hole in the bottom plate to prevent accumulation of water.

7.3.7 Other Cells

Occasionally, other Cells, No.'s 6, 9, 10 and 11 (see Figure 6.2), are used in support of research and development activities authorized under License SNM-960. However, the quantities of SNM associated with these activities are limited to contamination adhering to such items as test specimens (e.g., irradiated fuel cladding samples). Brief descriptions of these cells are given below.

7.3.7.1 Cell 6

Cell 6 is a commercially produced cell with internal dimensions of 48-in. x 36-in. x 48-in. (height). The cell is provided with 6 inches of lead in front and four inches on other surfaces. Windows are of leaded glass. The cell surfaces are sheathed with stainless steel.

7.3.7.2 Cells 9 and 10

Cells 9 and 10 are twin cells with a common access lock and internal dimensions of 48-in. x 48-in. x 120-in. (height). They are constructed of 18-inch thick, precast, high-density concrete panels with leaded glass windows.

7.3.7.3 Cell 11

Cell 11 is constructed of laminated steel with a thickness totaling 10 inches. The inside dimensions are 96-in. x 42-in. x 108-in. (height), and the cell is equipped with leaded glass windows.

7.3.8 Radiochemistry Laboratory

The radiochemistry laboratory, used principally for the analysis of samples of irradiated fuel materials from the RML, is located immediately adjacent to the RML area. Hoods or glove boxes are connected to the previously described 102 ventilation system and are designed to provide a minimum face velocity of 125 linear feet per minute.

7.3.9 Servicing Areas and Equipment

The Radioactive Materials Laboratory section of Building 102 also contains several shielded cells used in non-SNM isotope work, a waste analyzing area, an equipment decontamination room, a machine shop, equipment storage areas, and a manipulator repair room.

7.3.10 Criticality Alarm Sensors

The Radioactive Materials Laboratory is monitored for criticality accidents by a detector located at the storage pool area entrance (Figure 6.2).

7.4 RML CRITICALITY CONTROL SYSTEM

Special nuclear materials used in connection with RML operations are principally in the form of oxides in irradiated fuel elements and experimental capsules. The spectrum of material and activity types may be quite broad. This required flexibility has been taken into account in the establishment of criticality controls and is reflected in considerably larger safety margins than might be appropriate to more routine or repetitive situations. For example, each CLA in each fuel examination cell is limited to 45 percent of a critical number of units (fuel rods, assemblies, etc.), and each fuel examination cell is limited further so that criticality is not possible if all of the fissile material in the cell comes together simultaneously under conditions of optimum water moderation and full water reflection; but normal activities preclude moderation to any degree, and sources of water to the cells are limited and virtually accident proof. Criticality controls governing RML and supporting activities are described in the following sections.

7.4.1 Receiving

RML work predominates in the post-irradiation fuel examination work but also involves work with various radioisotopes including encapsulations and the examination of experimental fuel capsules. Shielded casks used for SNM shipments or on-site transfers have received criticality analyses demonstrating their safety either to receive NRC certification or to meet the requirements of License SNM-960.

Quarantine zones in the radioactive materials storage area (Hillside Storage) and the north side of Building 102 are used for incoming shipments from off site awaiting evaluation of the contents. Shipments are spaced according to DoT and NRC requirements.

Casks of irradiated material may be received on the Building 102 dock.

7.4.2 Storage

Irradiated special nuclear materials may be stored in shielded shipping or transfer casks (temporary), in the RML pool, or in the dry pit storage facility. When material is stored in casks, the spacing between casks and the number accumulated in a given area will be consistent with the DoT requirements for safe shipments.

Pool storage is limited to locations which have been analyzed both for the critical parameters of the accumulation of material in each storage location and the interaction between storage of uranium or mixed oxide irradiated fuel rods or fueled experimental capsules. All storage locations are isolated from one another by a minimum of 12 inches of water, and the locations have been analyzed for all varying degrees of interspersed moderation. The fuel rods/capsules stored in each rack are limited to 45% of the smallest estimate of the initial number of pins with consideration given to the fissile composition, pellet diameter, and enrichment. When two or more types of rods/capsules are stored in the same location, the limit for the most restrictive type is used. The pool is further limited so that criticality is not possible even if all the SNM in the pool came together simultaneously.

The dry pit limit is currently 300 grams fissile for each of the 19 positions. Each position has a minimum edge-to-edge spacing of 12 inches. Fuel rods or assemblies are sealed in aluminum waste liner cans similar in design to DoT Specification 2R containers which exclude water in the "as-stored" situation.

There are no credible means for rearranging the tube spacing. If the pit were flooded accidentally, even with optimum moderation within the sealed liner units, the fully loaded facility would be safe since each unit would be isolated by 12 inches of water.

7.4.3 Operations

Criticality controls for the hot cells and for cold limit areas are based on the principal of limiting the quantity of fissile materials in terms of 45% of the minimum critical mass or number of units for each limit area. In determining the minimum critical mass or number of units (e.g., irradiated fuel rods), such factors as the form (e.g., fuel rod, powder, solution), enrichment, and composition of the special nuclear material and the available moderation and reflection are considered.

The limit area control is implemented by an electronic data processing system. A perpetual inventory log is kept for each limit area showing the quantity of fissile material in each unit transferred to and from the limit area.

Each cell is isolated by 36 inches of concrete and from its interlock by a 15-inch-thick steel door.

The analytical services provided to RML by the Radiochemistry Laboratory necessitate only small samples of special nuclear material in the lab area. Simple mass control therefore is established. Not more than 300 grams U-235, nor more than 150 grams U-233 or 200 grams plutonium, are permitted within the confines of the Radiochemistry Laboratory at any time. If mixtures of these fissile isotopes are present, the lowest limit applicable is the limit for the entire laboratory.

7.4.4 Waste Disposal

Upon completion of hot cell work, fuel-bearing materials and components are sealed in prenumbered waste liners and removed to one of the storage locations described above and in accordance with the criticality control limits of the storage location or transferred to the Hillside Storage location in a shielded cask.

Other forms of waste materials are handled in previously described Building 102 liquid waste and ventilation systems.

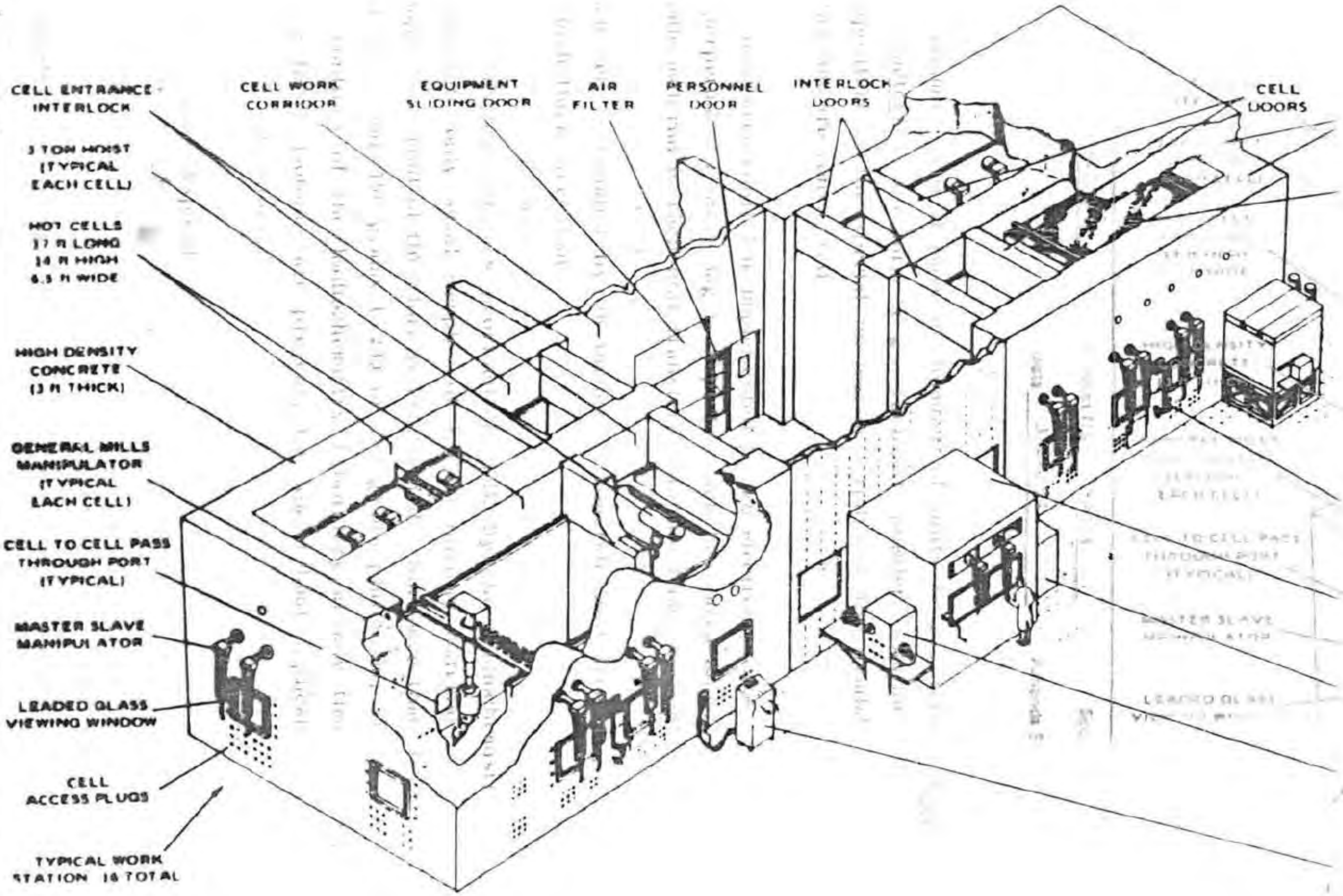


FIGURE 7.1. RADIOACTIVE MATERIALS LABORATORY HOT CELLS

8.0 PLANT CHEMICAL AND RADIATION TECHNOLOGY LABORATORY - BUILDING 400

8.1 LOCATION AND GENERAL DESCRIPTION

The Plant Chemical and Radiation Technology Laboratory (PC&RTL) is located in Building 400, which is a one-story building with an attached high bay area adjacent to Building 401 (Figure 1.2). The PC&RTL is the only portion of Building 400 in which special nuclear material is used and occupies approximately 2,400 square feet of space (Figure 8.1). The building is primarily of pre-cast concrete panel and metal panel construction, while the high bay is constructed of sheet metal over a steel frame.

8.2 GENERAL PLANS AND USES OF MATERIAL

Special nuclear material is used in research and development activities primarily in the area of low-enriched uranium oxide scrap recovery. No isotopic separation is performed.

8.3 LABORATORY FACILITIES AND EQUIPMENT

8.3.1 General

The purpose of the PC&RTL is to provide the development of chemical processes and prototype equipment to support fuel manufacturing components of General Electric. In order to provide these services, the PC&RTL is equipped with ventilated hoods for operations with laboratory quantities of SNM (Room 114), a machine shop area (Room 129), a prototype processing equipment area (high bay area), and a room for preparing SNM for prototype equipment testing (Room 115).

8.3.2 Scrap Recovery Pilot Process

The scrap recovery pilot process used for research and development activities is located along the south wall of the high bay area (Figure 8.1). This is an experimental facility used to investigate scrap recovery processes for low-enriched uranium oxide fuel scrap. Scrap material is brought into Room 115 in 5-gallon containers which are opened in a ventilated hood. The scrap then is sorted and ground, also in a ventilated hood.

Ground scrap is placed in the dissolver located in a fume hood in the high bay area. The fume hood is maintained at a -0.5-inch pressure with relation to the high bay area, and the dissolver fumes are exhausted through a NO_x catalytic destruct unit prior to entering the building exhaust system. The dissolver solution then is sent through a solvent extraction process. Typical chemicals used in the process are HNO_3 (0.15M to 8M), NH_3, $(NH_4)_2CO_3$, and organic solvent (30% tributyl phosphate and 70% dodecane). The solvent extraction equipment is located behind movable Plexiglas splash panels which also aid in maintaining proper ventilation control. As a safety measure, the organic solvent is maintained at 165°F or lower.

8.3.3 Hoods

Hoods are employed for handling powders and for dissolution operations. All hoods are maintained at a flow rate of at least 125 linear feet per minute across the openings.

8.3.4 Ventilation System

Inlet air supply for Building 400 is provided by air conditioning units furnishing filtered and tempered outside air to the building.

Air is withdrawn through the hoods in Rooms 114, 115, and 129. Each hood exhaust is equipped with a high-efficiency filter having a minimum efficiency of 99.97% for 0.3 micrometers diameter homogeneous particles of dioctyl

From the individual filters the effluent air is collected in a central duct and exhausted through a 45-foot high stack after passing through a bank of up to 24 HEPA filters (normally, 18 filters are used). This exhaust operates at approximately 21,000 cubic feet per minute. If the primary exhaust system fails, 4,000 to 6,000 cubic feet per minute is available through a backup emergency exhaust system. This system has its own 45-foot high stack with a bank of six high-efficiency filters.

8.3.5 Radioactive Waste Facilities

Dry contaminated waste materials generated in the PC&RTL are packaged as described in Section 3.5.1 and transferred to the site radioactive material storage facility (Hillside Storage).

Liquid wastes containing only trace quantities of SNM are collected in a 1,000-gallon waste tank (Figure 8.1). The tank is emptied periodically and the contents transported in the portable site transfer tank to the site liquid waste evaporator.

Liquid wastes containing accountable quantities of SNM and liquid organic wastes are solidified at the PC&RTL for transfer to the site radioactive material storage facility.

8.3.6 Criticality

Building 400 meets the requirements of a subcritical area as defined in Section 3.14 of Appendix A to License SNM-960.

8.3.7 Fire Protection

The PC&RTL is provided with an automatic sprinkler system. Fire extinguishers also are located strategically throughout the laboratory areas. The chief types of fuels available, except for the high bay area, are general paper and plastic combustibles and small quantities of flammable solvents. To reduce the potential for fire from the use of organic solvents in the high bay area, the solvent is kept at a temperature of 165°F or lower, and all electrical motors and switches associated with the solvent extraction columns are enclosed.

8.3.8 Laboratory and Effluent Air Monitoring

Approximately 20 air sample locations are operated within the PC&RTL facilities. The samples are collected on 50-mm Millipore-type filters. Each sample is changed on a weekly basis. In addition, alarming continuous alpha air monitors are operated in Room 115 and the high bay area.

The ventilation exhaust stack is sampled continuously using a 47-mm Millipore-type filter which is changed on a weekly basis.

8.3.9 Contamination Control

Monitoring and step-off procedures are observed at points of transition from controlled areas to uncontrolled areas. In addition, a hand-and-shoe monitoring station is provided in the corridor adjacent to the PC&RTL.

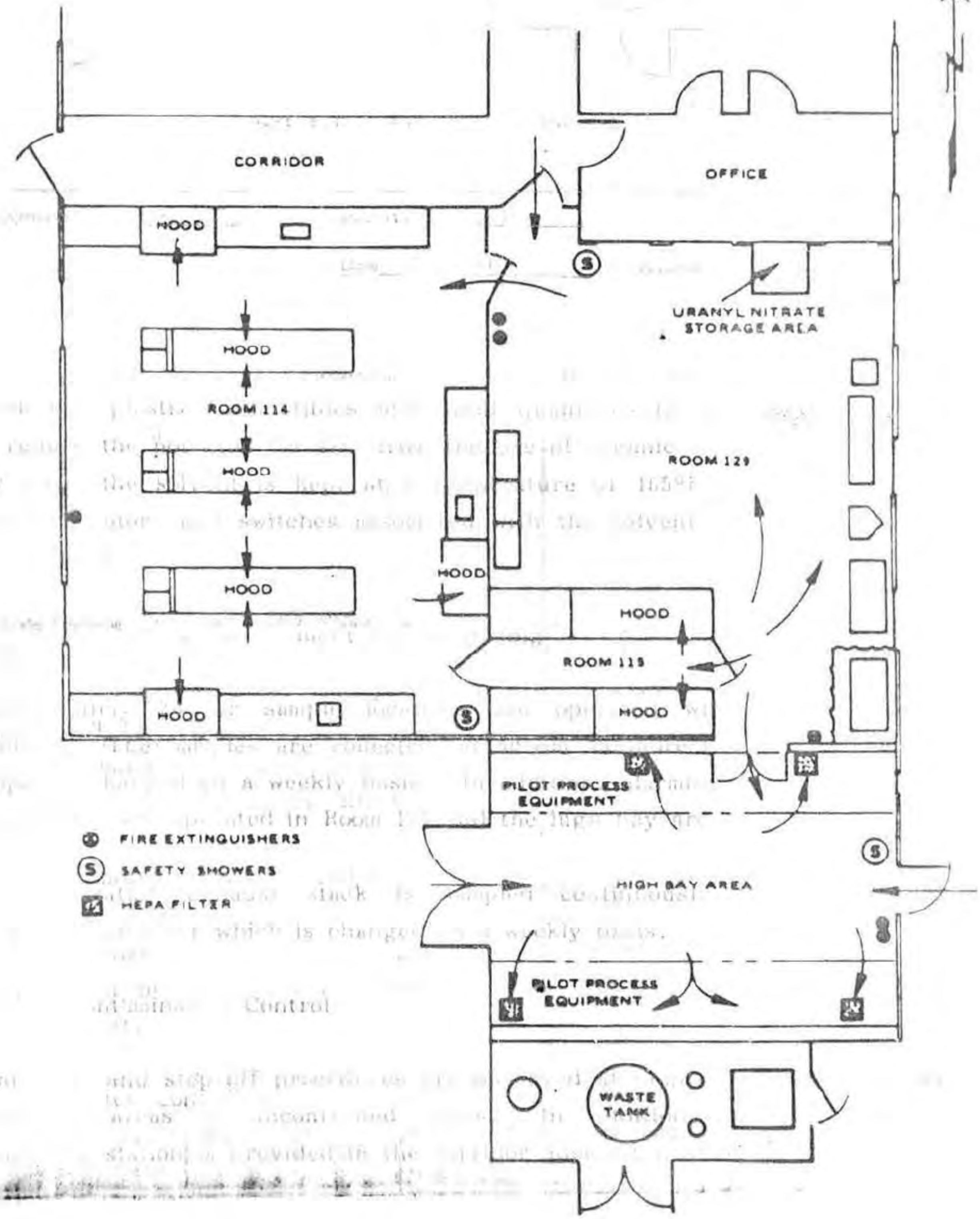


FIGURE 8.1 PLANT CHEMICAL AND RADIATION TECHNOLOGY

9.0 CORE MATERIALS ENGINEERING LABORATORY

9.1 LOCATION AND GENERAL DESCRIPTION

The main Core Materials Engineering Laboratory (CMEL) facilities are located in the basement of Building 102 as shown in Figure 9.1 and occupy approximately 4,000 square feet. The walls are composed of 8-inch reinforced and filled concrete block and 12-inch poured concrete. The CMEL facility is separated from activities conducted above on the main floor of Building 102 by a thickness of at least 12 inches of concrete. Additional CMEL facilities are located in Building 102J.

9.2 GENERAL PLANS AND USES OF MATERIALS

The material received, processed and used in the CMEL is employed in a variety of materials testing and properties studies and process development demonstrations. The materials are restricted to uranium as depleted, natural and/or $\leq$ 5% enriched and natural thorium. These materials generally are used in the form of pelletized solids (oxides), but other forms including nitrates and powdered oxides occasionally may be employed. With special permission from the nuclear safety function, small quantities of higher enrichment uranium (> 5% $\leq$ fully enriched) also may be brought into the CMEL for testing purposes.

The materials R&D activities occasionally involve heating sintered source material, such as UO_2 pellets, to high temperatures ($\leq$ 2,840°C). When such operations are conducted, the materials are doubly encapsulated. Primary encapsulation of materials is generally in tungsten, molybdenum, or stainless steel containments up to 12 inches in length and 1 inch in diameter. The U-235 content of U-based materials may vary up to fully enriched; however, it generally is restricted to $\leq$ 5.0%. Processes undertaken include drying, cutting, reduction, grinding/polishing, encapsulation, inspection, decontamination, and waste disposal.

Encapsulated specimens may be transferred to other VNC locations or to other facilities authorized to receive such materials.

9.3 CONTAINMENT OF RADIOACTIVE MATERIALS

Materials are confined, contained, and controlled by means of glove boxes, fume hoods, and similar equipment designed and constructed to meet the needs of the types and quantities of materials being used. Glove boxes are constructed of stainless steel or steel plate, the latter protected with corrosion-resistant coatings. Joints are sealed and rounded in order to eliminate surface interstices and to minimize crevice contamination. Viewing windows are removable to permit changing in case of discoloration or accidental cracking. These windows are of 0.25-inch or thicker Plexiglas.

Glove boxes are designed to be leak tight and normally are operated at a minimum positive pressure of 0.5 inches of water to preserve and maintain a controlled, purified, inert atmosphere. Special glove-box-like enclosures are used to contain developmental processing equipment, and these are equipped with absolute exhaust filters and airflow-controlling port covers. Fume hoods are of standard laboratory design, equipped with absolute exhaust filters and adjustable windows.

Procedures are established which provide for the transfer of materials to or from storage containers or between Process Development Operations (PDOs). Decontamination of equipment or containers may be carried out in fume hoods. Fume hoods and special glove-box-like enclosures will be provided with an airflow of at least 125 linear feet per minute face velocity.

Fire extinguishers are available where there is a potential for fire generation. A fire sprinkler system has been installed to suppress fires in the laboratory area; this system is interconnected to the site alarm system.

9.4 VENTILATION

The ventilation system for the laboratory is described in Section 6.0 of this application.

9.5 AIR MONITORING

Approximately 18 routine air sampling stations continuously sample the laboratory atmosphere and are changed and counted on a weekly or semi-weekly basis for alpha and beta-gamma activities. Three constant air monitors with appropriate alarm settings are used to give timely warning of significant increases in air activity concentrations.

9.6 PROCESS AND EQUIPMENT DESCRIPTION

9.6.1 Material Feed and Storage

Uranium and thorium are received in any of several forms; generally, as oxide, either as pellets or in powder form. Received materials are stored in their containment vessel in the laboratory vault (Figure 9.1) or in Building 102J in their shipping containers.

9.6.2 Encapsulation and Welding

Pellets or granules of material are loaded into capsules of varying dimensions up to 1.0 inch in diameter by 12 inches in length. The capsules are semi-closed containments with one end cap welded in place and the other end exposed for materials loading. Loading of the capsules is performed in a fume hood or controlled inert atmosphere glove box.

After loading, the capsules are transferred to the welding box for final welding closure. The sealed capsules are removed from the glove box, decontaminated as required, and transferred to the designated test facility.

9.6.3 Pellet Processing

Oxide pellets are received pre-fabricated and are distributed to locations within CMEL. Within the locations the pellets are ground, inspected, and classified or encapsulated.

9.6.4 Storage and Shipment

Oxide powders and pellets are stored in the Building 102 CMEL vault (Figure 9.1), in work locations, and in Building 102J. The CMEL vault has 25 cubicles which have a one-foot separation from one another in all directions, have restraining devices, and are secured to the vault walls.

PDOs are operating locations in which materials may be utilized for testing or demonstration purposes. PDOs are locations within the CMEL basement facility.

Materials may be stored in Building 102J (Figure 6.1) in approved shipping containers provided all conditions established by the Building 102J criticality analysis are met.

9.6.5 Waste Disposal

a. Liquid Waste Treatment and Solidification

Liquid wastes generated from cleaning of equipment and sectional cutting of materials or hardware either are evaporated or solidified in a mixture of Sorbitol and cement in metal containments. The solidified materials are labeled as to content and quantities and are transferred to the Radwaste Processing Facility for collection by a licensed waste disposal contractor. Other liquid wastes which normally do not contain or contain only trace amounts of radioactive materials are routed to the Building 102 waste tanks. No liquid effluents are released to the environment.

b. Solid Waste Disposal

All solid wastes containing or suspected of containing radioactive materials are segregated into compactible or noncompactible-type categories and deposited into Interim Waste Containers (IWC). The contents of these containers are recorded on the IWC Contents List. These containers then are transferred to the Radwaste Processing Facility for collection by a licensed waste disposal contractor.

9.6.6 Dry Analytical

Oxygen-to-metal (O:U) ratios are determined by oxidizing or reducing and weighing powdered pellets. Material is placed in a Vycor or quartz tube inside of an electric tube furnace in an atmosphere of argon-air, 6% hydrogen-94% inert gas, or argon-CO_2. Gravimetric analyses of this type also may be performed in a TGA system (see 9.6.7, Item d).

9.6.7 Materials and Properties Studies

Materials behavior studies currently are performed at Positions 20, 23, 24, 26 and 27 (Figure 9.1). The following are examples of current R&D work involving radioactive material:

a. Electrochemical Properties and Micro-Drilling

The potential (EMF) and resistivity of urania pellets are measured. These pellets are of varying density and oxygen-to-metal ratio and may contain inert diluents such as CeO_2. Certain experiments require that pellets be modified by micro-drilling and/or micro-machining.

b) High Temperature Property Studies

High temperature property experiments such as melting point, thermal conductivity, thermal creep, phase changes, diffusivity, etc., are conducted. These experiments are conducted under vacuum or in an inert or reducing atmosphere with the specimen doubly contained.

c) Metallography/Ceramography

Specimens to be examined are potted in a Bakelite mount with hysol resin. Specimens then are ground and polished and examined on a metallograph. Specimens then may be etched to reveal grain structure and re-examined. Grinding and polishing operations are performed inside a ventilated hood enclosure.

d) Micro Thermogravimetric Analysis

Milligram quantities of urania or thoria specimens are analyzed to determine the kinetics of reactions such as oxidation and reduction. This is accomplished using a micro-balance. The samples are heated in specified atmospheres and the weight change followed on a recorder.

e) Liquid Metal Test Facility

This facility is a small-scale, liquid alkali metal test loop in which liquid sodium or lithium can be circulated over small pelletized UO_2 or ThO_2-containing test specimens (≤ 10 gm) heated to temperatures as high as 750°C. The test loop is constructed from Type 316 stainless steel and is enclosed in an inert atmosphere glove box. The liquid metal capacity of the system is approximately 500 g, and it contains an automatic shut-down mechanism in the event of an alkali metal leak.

f. Controlled-Environment Mechanical Testing

Several test systems are used for measuring the mechanical properties of materials such as irradiated stainless steel at high temperatures (≤ 800°C) in controlled environments such as high purity argon. Compression and tensile testing is performed in an INSTRON 1123 Universal Test Machine, and miniature pressurized capsule burst tests are performed inside a dedicated high temperature furnace. The neutron-irradiated Type 316 stainless steel specimens employed in compression tests contain less than 0.5 mCi of activated radionuclides (^{60}Co, ^{54}Mn).

9.7 RADIATION AND CONTAMINATION CONTROL PROCEDURES

In addition to the procedures in effect in all laboratories at VNC, special procedures are imposed additionally in the laboratory.

Periodic surveys for alpha contamination are made of openings in the operating enclosures. Floors normally are surveyed daily. Surfaces external to the enclosures are maintained at less than 200 d/m alpha smearable and < 100 c/m beta-gamma smearable per 1 ft^2.

Minimum protective clothing requirements when working in the laboratories are shoe covering and laboratory coats. Hand-and-shoe surveys are required whenever personnel leave the laboratory.

9.8 CORROSION CONTROL

No highly corrosive materials are formed in any of the tests or processes. Normal laboratory reagents such as nitric acid and ammonium hydroxide are used in standard laboratory apparatus which safely entrains and confines any fumes that might be evolved. Air-reactive materials such as alkali metals are only handled inside inert atmosphere enclosures such as glove boxes.

9.9 CRITICALITY

The CMEL meets the requirements of a subcritical area as defined in Section 3.14 of Appendix A to the license.

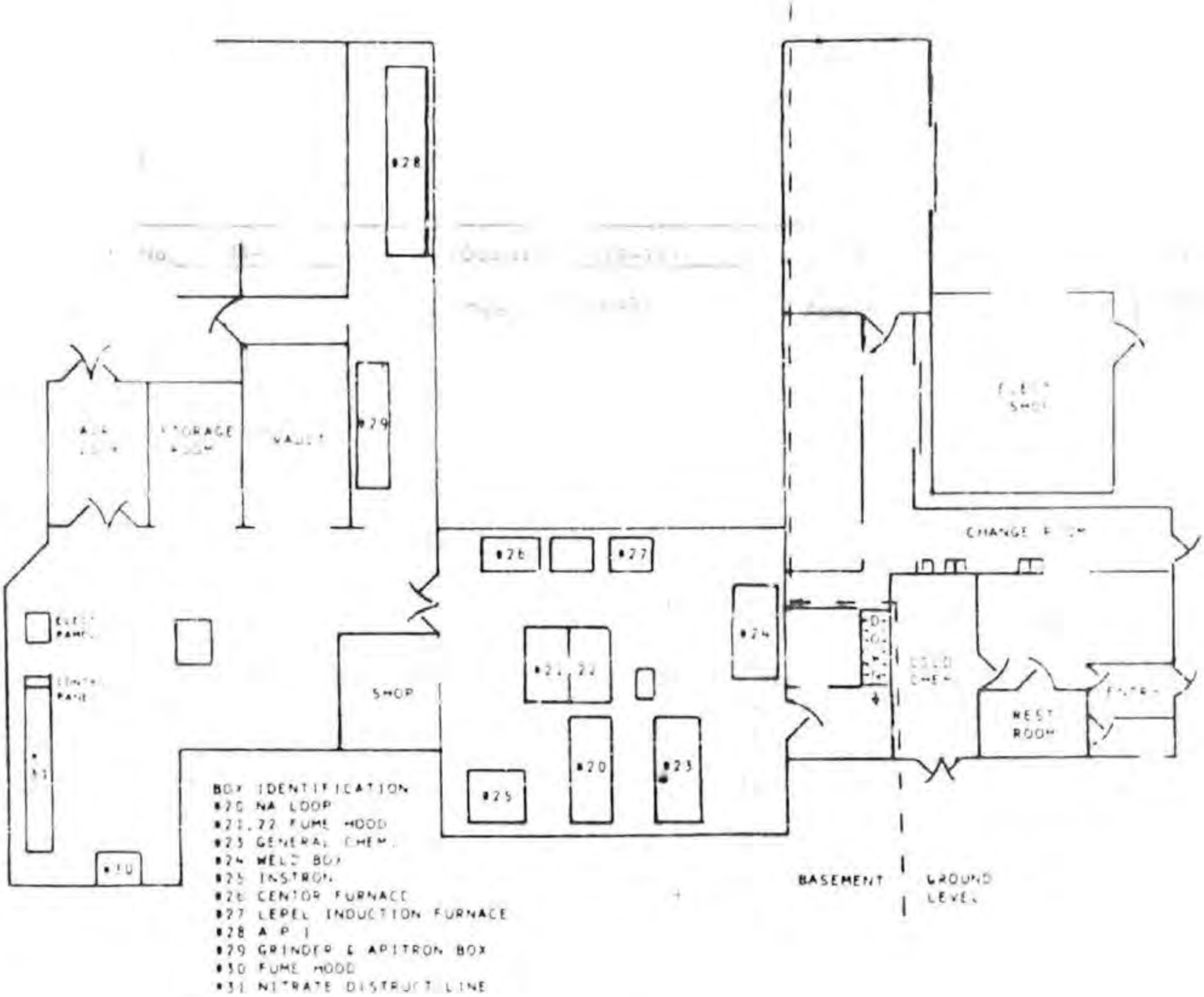


FIGURE 9.1 CORE MATERIALS ENGINEERING LABORATORY BUILDING 102

10.0 CHEMISTRY, METALLURGY AND CERAMICS LABORATORY - BUILDING 103

10.1 LOCATION AND GENERAL DESCRIPTION

The Chemistry, Metallurgy and Ceramics Laboratory, Building 103, is a two-story building located directly across the access road from Building 102 (Figure 6.1). The building has a total floor area of approximately 22,000 square feet including approximately 11,000 square feet of laboratory space. A corridor runs the length of the building on each floor. The laboratories are located on one side of the corridor and offices on the other as shown in Figures 10.1 and 10.2.

10.2 GENERAL PLANS AND USES OF MATERIAL

Special nuclear material is used in research and development activities including analytical, test specimen fabrication, maintenance, and calibration work and for sources of radiation necessary to support research and development programs. At the conclusion of experiments, radioactive materials usually are re-worked, re-used, stored, transferred off site to persons authorized to receive, or discarded as wastes. No isotopic separation or processing of special nuclear material is conducted except as necessary for experimental and analytical purposes.

10.3 LABORATORY FACILITIES AND EQUIPMENT

10.3.1 General

Metallurgy and ceramics areas of the building are located on the first floor and contain the following equipment: rolling mills, general purpose presses, swaging machines, hydrogen furnaces, vacuum furnaces, ceramic milling and mixing equipment, gas furnaces, atmosphere-controlled furnaces, arc furnaces, ceramic extrusion equipment, ovens, tensile machines, hardness testers, corrosion autoclaves, corrosion loops, metallographs, polishing wheels, electron microscope, x-ray equipment, microscopes, welders, induction heaters, machine shop equipment of various kinds, ultrasonic equipment, electrochemical equipment, mounting presses, gas purification trains, hoods, isolation boxes, sandblaster, and similar experimental equipment deemed necessary.

The chemistry areas on the second floor consist of typical chemical laboratories, a counting room, and an instrument room. Equipment in the chemistry laboratories includes the following: various types of spectrophotometers, fluorimeters, gas chromatographs and a plasma emission spectrometer; ovens, furnaces, centrifuges and other miscellaneous laboratory equipment; lead caves and glove boxes; vacuum systems, including necessary instrumentation; hoods designed for handling radioactive materials; counting instrumentation; and mass spectrometers of various types.

A concrete storage vault for special nuclear material is provided on the ground floor. The vault has walls and ceiling of 8-inch minimum thickness, without penetrations, and a single locked door.

A waste storage building of corrugated steel and aluminum on a concrete pad is provided for temporary storage of packaged wastes and scrap materials which result from the licensed chemistry, metallurgy, and ceramics activities. They consist of paper, glassware, plates, rods, wire, chips, samples, and other waste residue materials which result from such research, development, and analytical activities. These wastes normally will contain small amounts of byproduct, source, and special nuclear materials.

10.3.2 Hoods and Glove Boxes

Hoods and glove boxes are employed when plutonium, U-233, and other alpha-emitting materials with ALI's (ICRP-30 Annual Limits of Intake) < 1 mgm are handled which require particular precautions.

Glove boxes for plutonium and U-233 or similar operations are constructed of stainless steel or corrosion-resistant coated metal with Plexiglas or heavy safety plate glass used for viewing and lighting ports.

10.3.3 Ventilation System

Inlet air supply for the 103 Building is provided by air conditioning units furnishing filtered and tempered outside air to the building. Air passes from the office areas through 2-inch fiberglass roughing filters in the laboratory door grills, thereby minimizing the passage of lint and dust into the laboratories. These filters also minimize backflow of potentially contaminated material in the unlikely event of complete exhaust system failure. Laboratory doors are equipped with automatic closers.

Air is withdrawn through the hoods and glove boxes passing first through individual high-efficiency filters at each hood or glove box. Only filters having a minimum efficiency of 99.97% for 0.3 micrometers diameter homogeneous particles of dioctyl sebacate, DOS, are used in the laboratory effluent ventilation system. From the individual filters, the air is conducted through a second filtration in one of two parallel banks of high-efficiency filters. Thus filtered, it is discharged through a 48-foot high, 5-foot diameter stack. The high-efficiency filters are fabricated of fiberglass to provide high resistance to fire. Filter frames are metal or chemically impregnated for resistance to fire, and permanent duct work is metal or polyvinyl chloride. Each laboratory room used to conduct activities with radioactive materials is equipped with air sampling devices. However, airflow rates are adequate to permit routine operations with nuclear materials without the use of personnel respiratory protection. The main exhaust blower operates at approximately 36,000 cubic feet per minute. If complete ventilation failure occurs, an evacuation alarm is sounded automatically.

From 9 to 12 air changes per hour are provided for most laboratory rooms. However, in some rooms the airflow rate may be as high as 15 air changes per hour. Hood exhausts are dampered individually to maintain minimum face velocities on the order of 125 linear feet per minute across the openings. Glove boxes are operated at approximately -0.5 inches of water with respect to the room. Appropriate instrumentation indicating airflow and/or differential pressure is available.

The efficiency of the Building 103 ventilation filter system has been demonstrated by years of exhaust stack sampling data.

10.3.4 Radioactive Waste Facilities

Dry contaminated waste materials generated in Building 103 are packaged as indicated in Section 3.5.1. Waste containers are transferred directly to the site radioactive material storage facility or may be placed in a waste storage building approximately 11 feet by 12 feet constructed on a concrete pad adjacent to Building 103 (Figure 10.1). This building is of corrugated steel and aluminum construction. The waste storage building is conspicuously posted in accordance with 10CFR20.203. Waste packages stored in this area are removed periodically to the site waste storage facility for delivery to a licensed waste disposal contractor.

Liquid contaminated wastes originating in Building 103 are routed from laboratory sinks and gravity drains through regulated pipes to waste retention tanks. Two tanks, each of approximately 5,000-gallon capacity, are provided. The tanks are equipped for representative sampling and for draining to drums or tank trucks. All tank wastes are sent to the site waste evaporator for concentration and solidification.

10.3.5 Criticality Alarm

Building 103 currently is monitored for a criticality accident by three detectors as described in Section 3.8. Approximate locations of the sensors are shown in Figures 10.1 and 10.2.

10.3.6 Fire Protection

Building 103 is provided with an automatic sprinkler system. Fire extinguishers also are located strategically throughout the laboratory areas. Accidental fires in the hood enclosures are of low potential. Three types of fuel are available for such accidents: general paper and plastic combustibles, flammable solvents, and pyrophoric materials. VNC fire prevention procedures minimize the fire potential; however, extinguishing equipment and materials are provided at strategic locations in the building. Additionally, smothering agents such as Metl-l-x, asbestos cloth, etc., are available in each enclosure where sufficient fire probability exists. Special precautions will be taken when quantities of special nuclear materials are handled in hoods or glove boxes to minimize fire hazards in these enclosures. For example, metal containers will be used for pyrophoric materials.

10.4 BUILDING 103 PROCEDURES

In addition to the VNC general procedures described elsewhere herein, other procedures specific to the health and safety of experimental and supporting operations conducted in the chemistry, metallurgy and ceramics laboratories, Building 103, are employed. The essence of these procedures is as follows:

10.4.1 Survey and Contamination Control Procedures

Monitoring and step-off procedures are observed at points of transition from controlled areas to uncontrolled areas. In addition, hand-and-shoe monitoring stations are provided at the building exits. Surveys at the beginning of irradiated experiments assure the adequacy of local shielding.

10.4.2 Glove Box and Hood Station Procedures

Basic operations to be performed at glove box or hood stations normally volve gram or milligram quantities of uranium and TRU materials. TRU wders in excess of 50 microcuries are handled in glove boxes only.

Blending, sample and specimen preparation, and analytical operations are the most frequent activities utilizing such material. Standard procedures for these activities are used.

10.4.3 Ventilation System Maintenance

The filtered and monitored ventilation system for Building 103 is described in Section 10.3.3. The high degree of efficiency of this system is due in part to careful maintenance and operating procedures. The main filter bank in the system is dual so that the system can be run on one bank while the other bank is being changed. The duct velocities are low enough to allow the main flow control and balancing to be done at each of the primary filter box connections. The main system basically runs as a large manifold of relatively constant suction so that primary connections can be made to the system as needed allowing the overall blower capacity of the main system to be maintained. Damper locations are restricted to dampers at the exit of each primary absolute filter to control the flow of that primary branch, and dampers that normally are open at the main filter bank to permit running the system on one bank while the filters in the outer bank are changed.

All filter boxes are provided with differential pressure gages to measure pressure drop across the filters. The flows, correlated with the pressure changes, give a good indication of the state of the filters. Any filters found to be below standard are changed. Filter boxes are surveyed by Radiation Safety for contamination prior to changeout. This survey on the second bank of filters also is a measure of the effectiveness of the first filters.

Exhaust air from the 103 Building laboratories is monitored continuously. The filtered air that leaves the building is exhausted up a 48-foot stack which extends 20 feet above the building. The exhaust air is sampled continuously for iodine and particulate activity at a point 24 feet below the top. The particulates are collected on a high-efficiency, 47-mm Millipore filter; and iodine is collected on a charcoal cartridge. The filter and cartridge are evaluated on a weekly basis.

10.4.4 Waste Disposal

Normally, dry contaminated wastes occur in the form of paper towels, glassware, and similar nonsalvageable laboratory apparatus as well as small quantities of uranium as, for example, test specimens and analytical sample residues. These materials are packaged, sealed, recorded as to quantity and type of nuclear material, placed in the designated waste storage building (Figure 10.1), and transferred to the site radioactive materials storage facility until final disposal from the site. Storage of dry wastes in Building 103 is limited to packages having surface dose rates of 100 mR per hour or less.

The two 5,000-gallon capacity radioactive liquid waste tanks are monitored routinely and, when approximately filled, are sampled and their contents routed to the site waste evaporator.

10.5 BUILDING 103 CRITICALITY CONTROL

The activities and work programs conducted in Building 103 are experimental and analytical rather than industrial. Experimental operations rarely require large quantities of fissile special nuclear materials at a time but do require a wide variety of forms, compounds, and concentrations.

Criticality control of these materials is relatively simple for it is based on the issuance of a safe quantity by a materials custodian for use in any single laboratory room at any one time for the majority of the activities conducted. Stock supplies of special nuclear materials normally are stored in a vault when not in laboratory use. Directions for storage arrays in the vault limit the amount and spacing of the material to safe systems assuming optimum moderation and reflection. Full or partial flooding of the vault is unlikely; however, since there is no water service in the vault and, except for the steel vault door and electric lighting in the ceiling, there are no other penetrations.

10.5.1 Vault Description and Layout

The storage vault is isolated by concrete walls, ceiling and floor at least 8 inches thick. It is situated between a rest room and a lunch room. Areas immediately above the vault on the second floor do not contain special nuclear material. No special nuclear materials are permitted in any of the adjacent rooms. Within the storage vault, special nuclear materials are stored in specified locations. Figure 10.3 shows the current configuration.

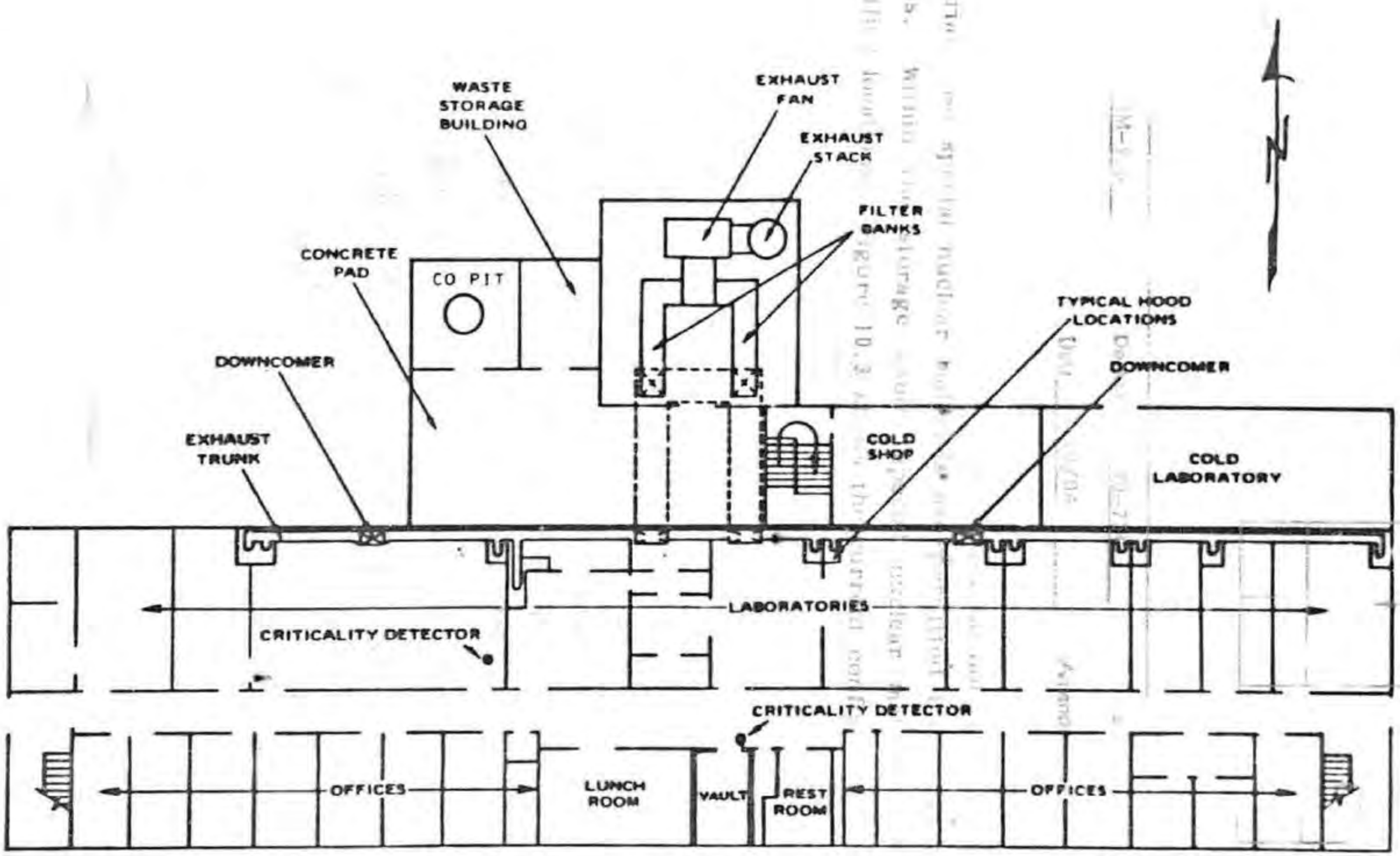


FIGURE 10.1. BUILDING 103 FLOOR PLAN (Ground Level)

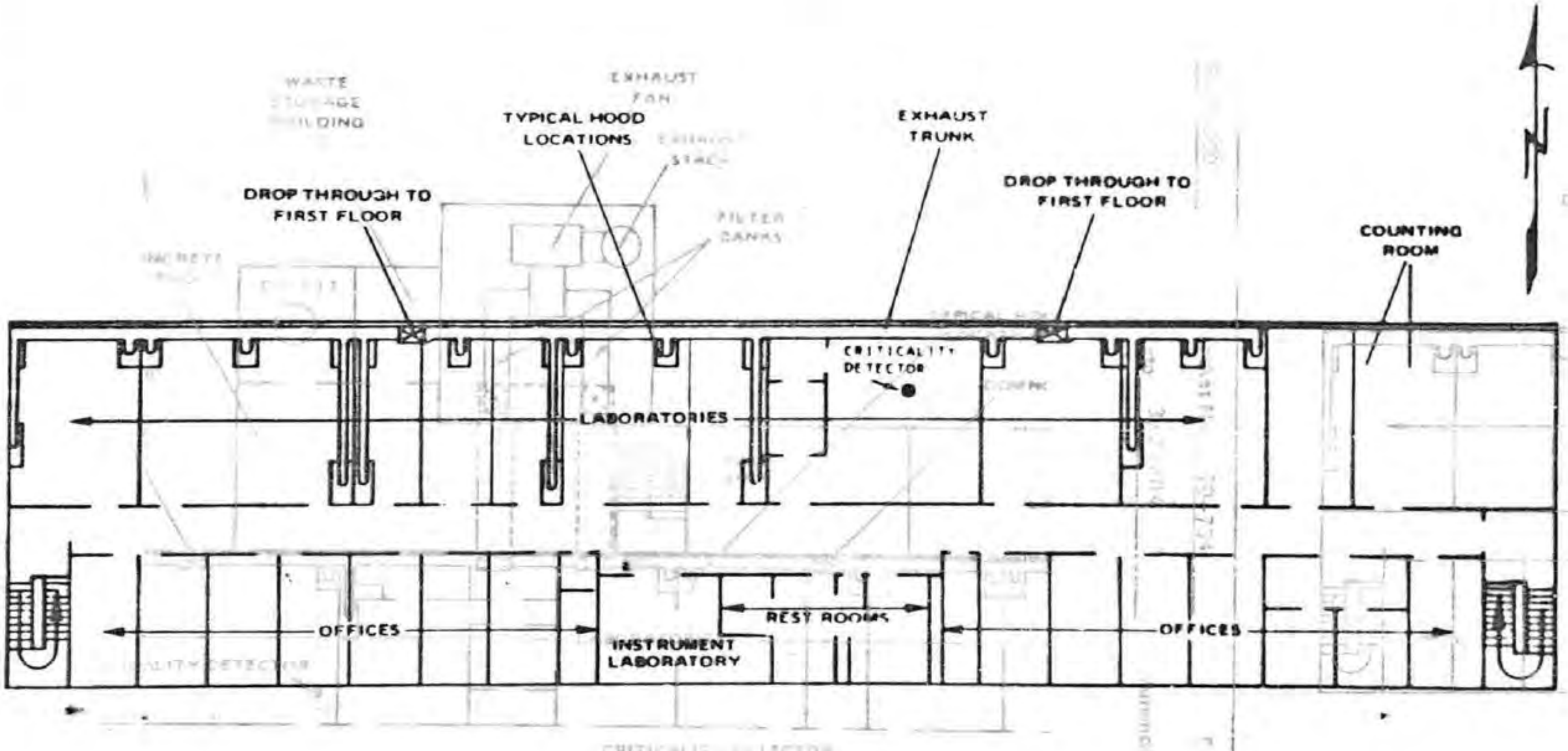
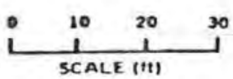


FIGURE 10.2. BUILDING 103 FLOOR PLAN (Second Floor)



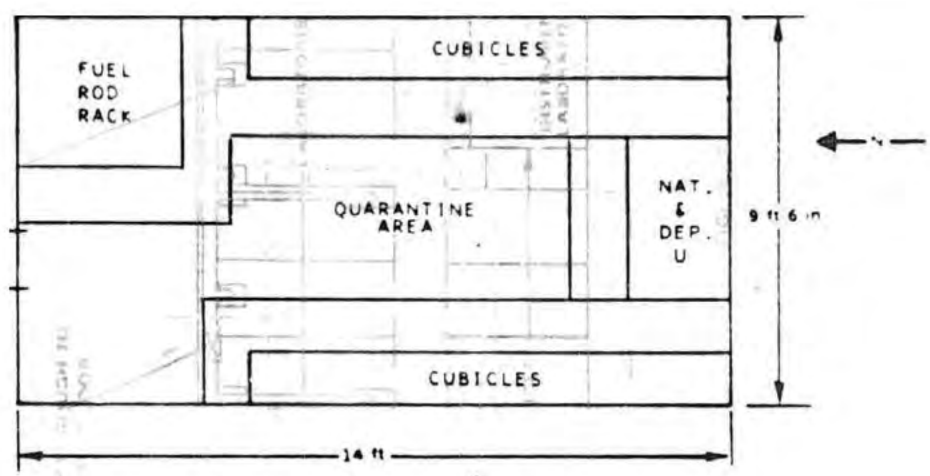


FIGURE 10.3. BUILDING 103 VAULT LAYOUT

11.0 DEVELOPMENT SHOP

11.1 LOCATION AND DESCRIPTION

The Development Shop currently is located in Building 106. This building is north of Building 103 and west of Building 105 (Figure 6.1). Building 106 is a largely single-story steel and concrete structure on a concrete pad with a floor area of approximately 15,500 square feet. The 2,540 square foot Development Shop is located in the central portion of the building and is enclosed totally by partitions (Figure 11.1). A portion of the building has a second story containing offices.

Activities in Building 106 include machining, welding, and sheet metal work. A small x-ray room is located in the weld and sheet metal shop area. The building contains a lunch room and several service functions.

11.2 GENERAL PLANS AND USES OF MATERIALS

Activities conducted in the Development Shop involving special nuclear materials consist of fabrication, modification, assembly and cleaning of small reactor components, capsules, capsule instrument leads, and minor repairs on fuel element hardware. These items contain only unirradiated solid forms of SNM. Generally, SNM is utilized in the encapsulated form; however, solid forms of SNM may be removed temporarily in a ventilated hood or glove box. Unencapsulated SNM is not subjected to actions that could change its chemical or physical form in any manner.

11.3 RADIATION SAFETY PROCEDURES

Activities conducted in the Development Shop as generally described above are of such nature that handling of unencapsulated special nuclear material rarely is necessary. Therefore, practically all possibilities for exposure of personnel to harmful quantities of such materials have been eliminated.

Equipment and procedures for maintaining a sound radiation safety program are set forth below.

11.3.1 Equipment

The Development Shop currently is equipped with machine tools, furnaces, welding equipment, leak detectors, controlled atmosphere chambers, hand tools, and conventional shop benches and cabinets. This equipment is used for work on hardware associated with totally encapsulated SNM.

Modification of SNM-bearing devices that could result in airborne activity will be accomplished in a ventilated hood. When a hood is used, the air is exhausted from this equipment through HEPA filters. This system is activated by two 3,500 cubic feet per minute exhaust blowers. Approximately six air changes per hour can be provided by this ventilation system. Hoods are operated at a hood face velocity of at least 125 linear feet per minute.

Although the potential for airborne contamination in this area is extremely low, each of the two stacks for the exhaust blowers is equipped to continuously sample the effluent using a 50-mm Millipore-type filter if activities with SNM warrant such sampling. A room air sampler also can be operated in the Development Shop area.

11.3.2 Procedures

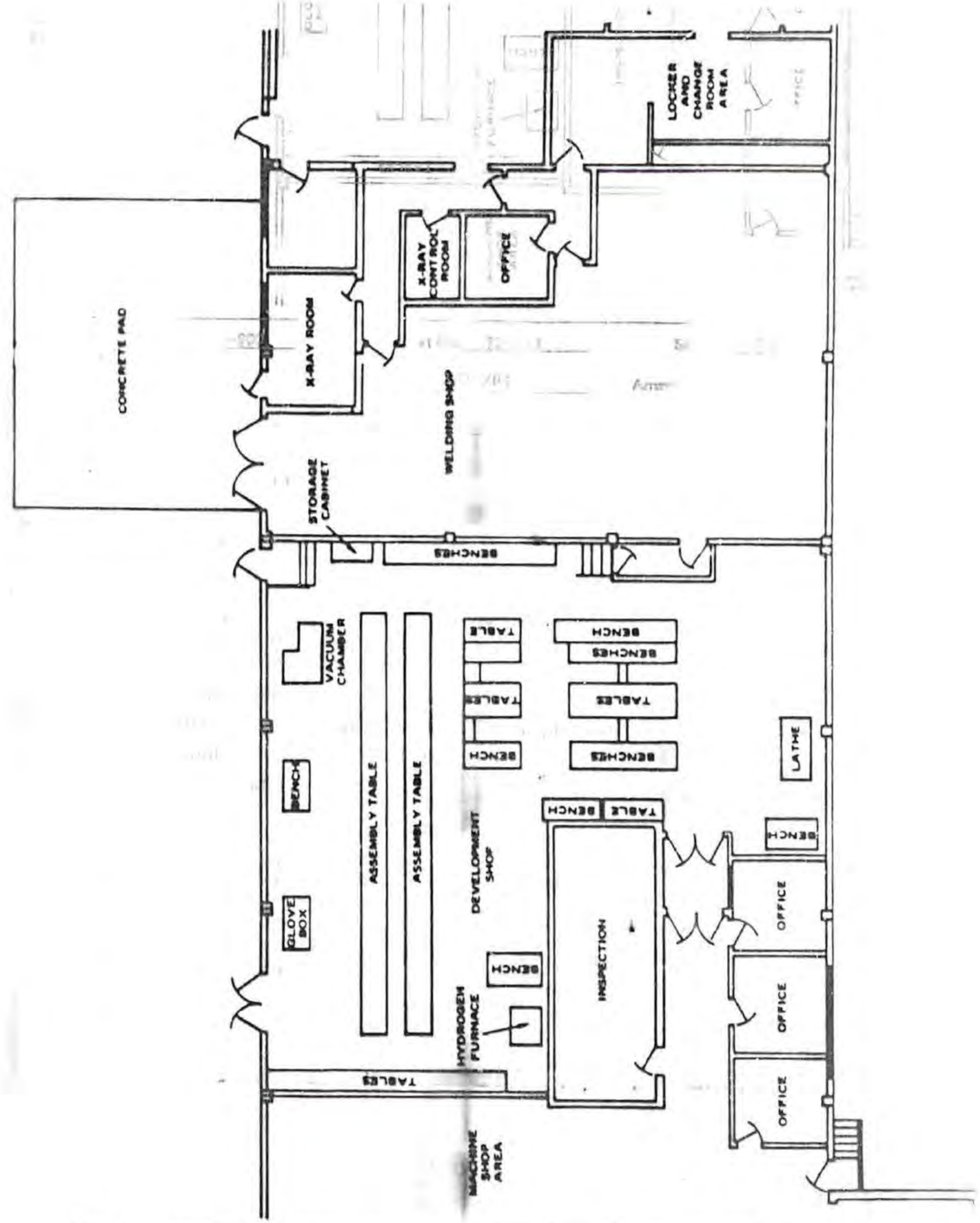
No irradiated special nuclear materials are permitted in the shop. No liquid forms of special nuclear materials are permitted. No work on unencapsulated SNM is permitted except in hoods. Equipment and floor areas are surveyed periodically by radiation safety function personnel. After each hood activity involving unencapsulated SNM, the hood and the immediate adjacent floor area are thoroughly surveyed, and all waste materials are removed in sealed cartons. Waste cartons are not allowed to accumulate but are removed to a designated waste collection area promptly after packaging.

11.4 CRITICALITY LIMITS

Building 106 meets the requirements of a subcriticality area as defined in Section 3.14 of Appendix A to License SNM-960.

11.5 SUPERVISION AND INSPECTION

The Area Manager for the Development Shop is responsible for establishing and maintaining detailed shop procedures to implement the conditions of the NRC license and the provisions of the general radiation safety program set forth in VNC Site Safety Standards.


CONCRETE PAD
X-RAY ROOM
X-RAY CONTROL ROOM
OFFICE
LOCKER AND CHANGE ROOM AREA
WELDING SHOP
STORAGE CABINET
BENCHES
VACUUM CHAMBER
ASSEMBLY TABLE
ASSEMBLY TABLE
TABLE
BENCH
BENCHES
TABLES
TABLES
BENCH
BENCHES
LATHE
BENCH
TABLE
BENCH
DEVELOPMENT SHOP
GLOVE BOX
BENCH
BENCH
INSPECTION
OFFICE
OFFICE
OFFICE
HYDROGEN FURNACE
TABLES
MACHINE SHOP AREA

12.0 BUILDING 105

12.1 LOCATION AND GENERAL DESCRIPTION

Building 105 is located immediately north of the 102 Building in the 100 Area (Figure 6.1). The building houses facilities licensed pursuant to 10 CFR Part 50 associated with the Nuclear Test Reactor (NTR) and the Advanced Nuclear Applications Laboratory (Figure 12.1).

12.2 NTR FACILITIES

The Nuclear Test Reactor (NTR) is licensed pursuant to 10CFR50 (License R-33). SNM licensed pursuant to License SNM-960 is taken to the NTR North Room or the NTR South Cell (Figure 12.1) primarily for purposes of neutrographic examination. Such material is in the form of sealed units, and only one safe batch (< 45% of the minimum critical accumulation) is permitted in the North Room or South Cell at any one time.

12.3 OTHER LABORATORY AREAS (ADVANCED NUCLEAR APPLICATIONS)

Other laboratory areas in Building 105 use SNM only as sealed sources, standards, foils, or as electronic components under general license and only in gram quantities.

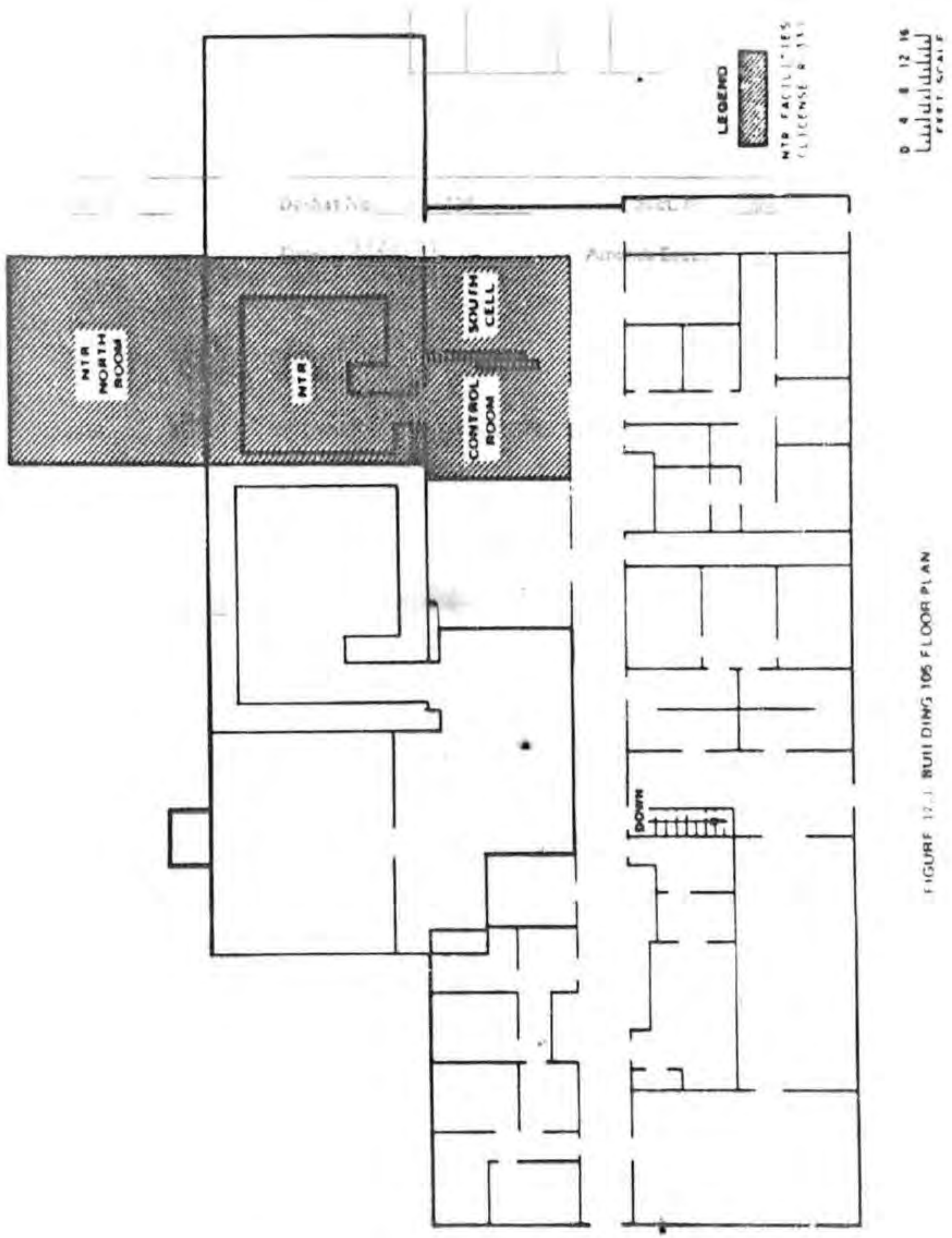


FIGURE 12.1 BUILDING 105 FLOOR PLAN

13.0 WASTE HANDLING

13.1 RADIOACTIVE LIQUID WASTE EVAPORATOR PLANT (BUILDING 349)

Activities described in this section involve the transfer of liquid radioactive wastes including those containing small quantities of special nuclear material generated at various facility and laboratory installations at the Vallecitos Nuclear Center site to the Radioactive Liquid Waste Evaporator Plant, Building 349; the concentration of these wastes in Building 349; the discharge of processed effluents; and the transfer of concentrated waste materials to a licensed waste disposal contractor. The estimated annual plant throughput is 120,000 gallons of liquid waste.

Each facility operated under License SNM-960 which generates liquid radioactive waste (i.e., Buildings 102, 103, and 400) are equipped with liquid radioactive waste retention tanks. All lines in each building which potentially could carry radioactive materials are connected to the retention tanks. The tanks are emptied as necessary.

Prior to transferring liquid wastes to the Waste Evaporator Plant, samples are analyzed for gross alpha and beta, total uranium and plutonium, and U-235. During the first half of 1983, uranium levels ranged from < 0.02 to 24.0 ppm with an average of 5.7 ppm. A log of the total quantity of U-235 in the plant at any one time is maintained. The entries to this log are based on the analyses of samples of the liquid waste, and the U-235 inventory has been typically between 50 to 70 grams. Analytical results for plutonium have shown concentrations to be less than 1×10^{-9} g Pu/ml of waste water.

The Waste Evaporator Plant is not a production or utilization facility within the meaning of 10 CFR Part 50, Section 50.2(a)(3) ("a facility designed or used for the processing of irradiated materials containing special nuclear materials...."). None of the components of the plant make possible the separation or purification of isotopes from each other.

13.1.1 Location and General Description

The Waste Evaporator Plant is housed in a metal building constructed on a poured concrete pad with integral concrete shielding walls around the processing vessels at high activity points. The shielding is designed to provide radiation levels in operating areas from normal wastes to less than 5 mR/h.

The building entrance opens into a change room and operations control room. General radiation levels in this area normally are less than 2 mR/h. Figure 13.1 shows the floor plan for the building.

The Waste Evaporator Plant is located in an area to the northwest of the deactivated VBWR facility (Figure 1.2).

13.1.2 Liquid Waste Processing

Liquid wastes are collected from the various site accumulation tanks and transported to the Waste Evaporator Plant by fork lift truck in a specially designed 1,500-gallon stainless steel tank. The waste transfer tank is equipped with internal vertical baffles to prevent cyclic shifting of the liquid material during transport. All external protuberances such as pipes, valves, gages, etc., are arranged or guarded in such a manner that they cannot come in contact with other vehicles or objects upon the roadway yet are readily accessible for manual operation for loading and unloading. Experience in transporting liquid wastes in this way shows dose rates to the fork lift truck operator generally have been less than 5 mR/h.

In the direct evaporation process the liquids are pumped continuously from the feed tank or chemical treatment tanks directly into the evaporator. Here the wastes are concentrated through a vertical tube natural convection evaporator. The vapor is treated in associated equipment including a high efficiency demister entrainment separator and a condenser. Effluent waters from the evaporator are collected in the monitoring tanks for analysis and disposal by evaporation. If further decontamination is necessary, the water can be re-routed to the feed storage tank for reprocessing or ion exchang treatment.

13.1.3 Concrete Mixing Station

Liquid waste concentrates from the evaporator are collected in a receiver and discharged into plastic-lined DOT Specification 17-H 55-gallon drums. The concentrates then are mixed with a cement-diatomaceous earth mixture or equivalent for solidification. After solidification, the drums are sealed and prepared for disposal as dry solid waste. The drums as loaded will provide compliance with appropriate DOT regulations. Solidified waste contained in drums is monitored, tagged with type, curie content and radiation level, and stored awaiting removal from the Vallecitos site by a licensed waste disposal contractor or other approved methods. Waste drums are not buried at the VNC site.

Small amounts of liquid wastes incompatible with evaporation may be solidified directly in 55-gallon drums.

Storage of dry wastes at the Evaporator Plant is limited to packages having surface dose rates of 100 mRem per hour or less. Waste drums having radiation readings in excess of 100 mRem per hour are transferred promptly to the high-level waste storage area. The building waste storage area is classified as a Radiation Area and so posted. No combination of drums is allowed to exceed 100 mRem/hr.

13.1.4 Ventilation

Effluent air from equipment and locations such as tank vents, hood vents, and sample points not requiring pre-filtering is exhausted directly to the main building filter exhaust system. Effluent air from the vented side of the condenser along with other points of suspected higher activity is manifolded into pre-filters before being exhausted into the main building exhaust ducts.

The main exhaust then is directed through a HEPA filter system (99.95% efficient for 0.3-micrometer particulate). Filters appropriate for high relative humidity service are used.

The filtered air then is discharged to the atmosphere through a stack at a point approximately 7 feet above the roof of the building. Approximately 18 air changes per hour maintain proper contamination control. Air discharged from the stack at the rate of 3,000 cfm is driven by an electrically powered blower mounted at the base of the stack.

13.1.5 Monitoring Procedures

Monitoring and/or step-off procedures are observed at points where each potentially contaminated regulated area exits into the clean or nonregulated areas. In addition, survey instruments are provided at convenient locations for final surveying. The doors to the process equipment area are alarmed or padlocked in accordance with 10CFR20.203(c)(2).

13.1.6 Criticality Control

The facility meets the requirements of a subcriticality area as defined in Section 3.14 of Appendix A to License SNM-960.

13.2 SOLID WASTE HANDLING

A single organization has been designated by VNC management as responsible to assure that all solid wastes leaving the site meet the appropriate regulatory requirements. For the purposes of this application, it will be referred to as the Waste Handling Function (WHF).

13.2.1 Solid Waste Accumulation

Solid wastes are accumulated at each location where radioactive materials are handled. The majority of wastes fits the Low Specific Activity (LSA) category as defined in the Department of Transportation regulations.

For each waste accumulation, the generating component is responsible for maintaining a listing of all material in the accumulation. Each accumulation with its listing is forwarded to the WHF for final inspection and/or repackaging.

13.2.2 Solid Waste Storage

Solid waste materials are stored in the site radioactive materials storage facility described in Section 3.5.1 and shown on Figure 13.2.

Waste materials that include or are associated with significant quantities of special nuclear materials are placed in containers called waste liners. Waste liners are stored in the horizontal tube facility. Fifty-five-gallon drums and boxes containing lower-level wastes or contaminated equipment are stored in the vertical well facility or above ground, depending on the activity levels.

Limits on the maximum quantity of special nuclear material that may be loaded into any containers for purposes of waste storage have been established by nuclear safety. For 55-gallon drum storage, calculations were made assuming optimum water moderation and spherical geometry for the individual masses within each drum. No credit was taken for neutron absorption by the materials between the individual units. On this basis, limits were established which provide criticality safety for an essentially infinite array of these drums. This limit was set in this fashion since it is possible to store drums above grade in almost any geometry. However, normal storage arrangement would be a planar array one drum high.

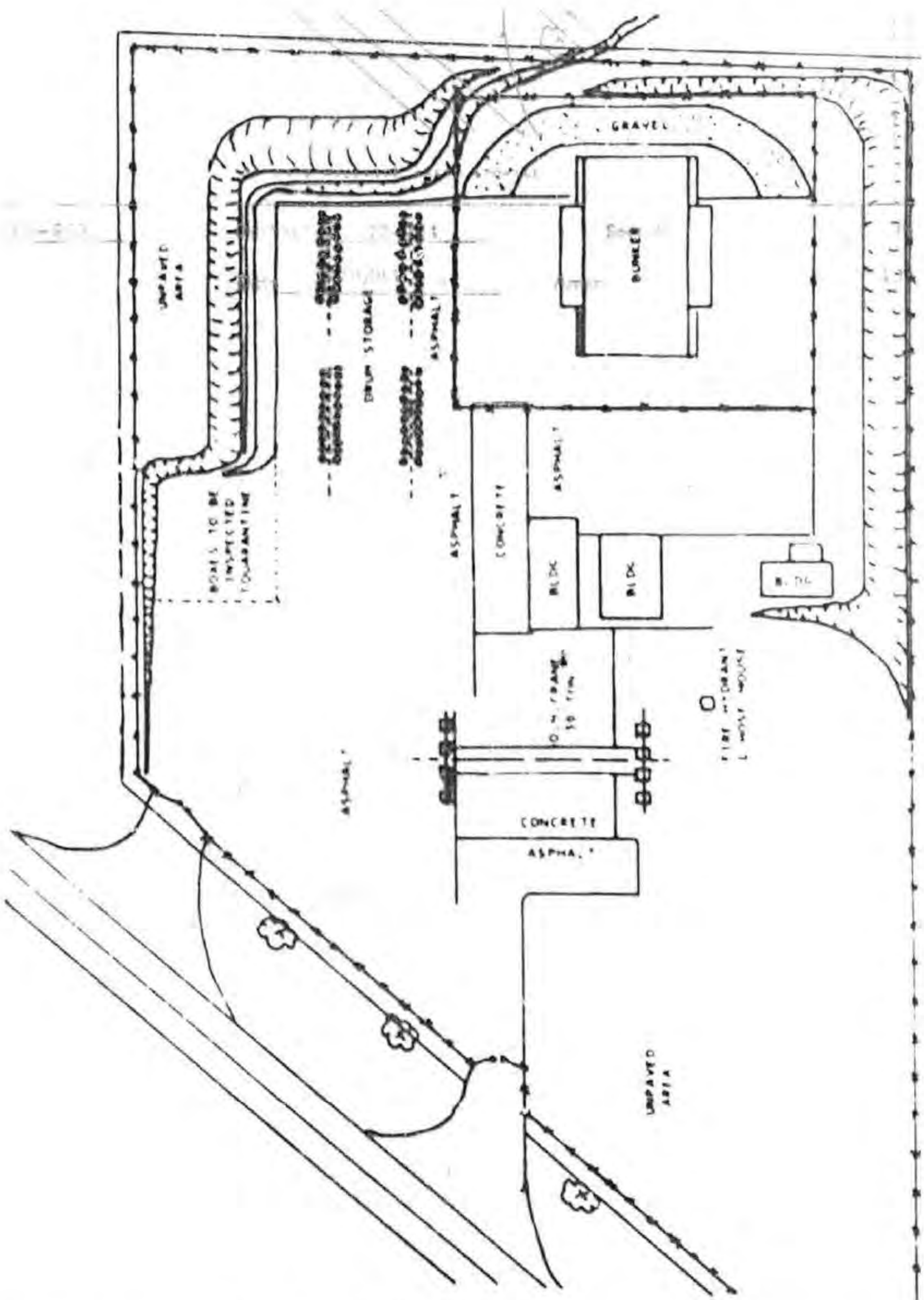


FIGURE 13.2 HILLSIDE WASTE STORAGE