MATERIALS LICENSE

Pursuant to the Atomic Energy Act of 1954, as amended, the Energy Reorganization Act of 1974 (Public Law 93–438), and Title 10, Code of Federal Regulations, Chapter I, Parts 30, 31, 32, 33, 34, 35, 40 and 70, and in reliance on statements and representations heretofore made by the licensee, a license is hereby issued authorizing the licensee to receive, acquire, possess, and transfer byproduct, source, and special nuclear material designated below; to use such material for the purpose(s) and at the place(s) designated below; to deliver or transfer such material to persons authorized to receive it in accordance with the regulations of the applicable Part(s). This license shall be deemed to contain the conditions specified in Section 183 of the Atomic Energy Act of 1954, as amended, and is subject to all applicable rules, regulations and orders of the Nuclear Regulatory Commission now or hereafter in effect and to any conditions specified below.

Licensee	
1. Union Carbide Corporation	3. License number: SNM-639 as renewed
2. Sterling Forest Research Center Tuxedo, New York 10987	4. Expiration date: October 31, 1989
	5. Docket or Reference No.: 70-687

6. Byproduct, source, and/or special nuclear material	7. Chemical and/or physical form	8. Maximum amount that licensee may possess at any one time under this license
A. Uranium-233	A. Encapsulated Sources	A. 10 grams
B. Uranium enriched in the U-235 isotope (> 20 percent enriched)	B. Any form (may be in the form of irradiated material with associated byproduct material)*	B. 23 kilograms of contained U-235
C. Plutonium	C. Encapsulated Sources	C. 2 milligrams of Pu-238 10 grams Pu-239 2 milligrams Pu-241 80 grams Pu-Be neutron source

* SNM mingled with radioactive material between Atomic Nos. 3 and 83 inclusive provided possession of the latter has been authorized under New York State License No. 729-0322.

COPY

MATERIALS LICENSE
SUPPLEMENTARY SHEET

License number
SNM-639 as renewed

Docket or Reference number
70-687

9. Authorized Use

For use in accordance with statements, representations and conditions contained in Part I of the revised application (Consolidated Application) submitted on June 6, 1984 and as amended on July 16, 1984 and September 6, 1984, except as modified by the conditions of this license. The effective pages of Part I of the application are identified in Appendix A of this license.

10. Insofar as applicable to radioactive materials subject to the authorities of this license and NRC regulations, the release of facilities, equipment and material from Building 2 to offsite, or from controlled to uncontrolled areas onsite, shall be in accordance with Appendix B (attached), "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material," dated July 1982. Records of the contamination survey and the final disposition of any equipment shall be kept for inspection by the NRC.

11. Upon completion of the decontamination of facilities in accordance with § 70.38 of Title 10 of the Code of Federal Regulations, the licensee shall submit with the report required by that regulation information that identifies all facilities where radioactive materials were used and stored, or disposed on the site. The report shall briefly describe operations conducted and radioactive materials used in the facilities and shall assess the results of the decontamination activities. The report shall provide the basis for unrestricted release of the facilities and the site, including a description of sampling and survey methods and instrumentation used, and shall include final contamination survey data for the facilities and grounds.

12. In addition to the reporting requirements specified on page I.8-1 of the application, the licensee shall provide a copy of any changes to the Radiological Contingency Plan to the Director, Division of Fuel Cycle and Material Safety, NRC.

13. The licensee shall provide to the NRC copies of its annual report summarizing the results of analyses of radiological environmental monitoring samples. This report shall be sent to the Director, Division of Fuel Cycle and Material Safety, U.S. NRC, Washington, D. C. 20555, and to the NRC Region I Office at the address specified in Appendix D of 10 CFR Part 20.

COPY

For the U.S. Nuclear Regulatory Commission

Date OCT 19 1984

By Leland C. Rouse
Division of Fuel Cycle and
Material Safety
Washington, D.C. 20555

Appendix A

List of Effective Pages
Union Carbide Corporation
SNM-639

Part I--License Conditions

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I.1-1	07/16/84
I.1-2	10/14/83
I.2-1	06/06/84
I.2-2	07/16/84
I.2-3	06/06/84
I.2-4	06/06/84
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I.2-7	06/06/84
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I.5-6	10/14/83
I.6-1	09/05/84
I.7-1	10/14/83
I.8-1	09/05/84

OCT 19 1984

APPENDIX B

GUIDELINES FOR DECONTAMINATION OF FACILITIES AND EQUIPMENT PRIOR TO RELEASE FOR UNRESTRICTED USE OR TERMINATION OF LICENSES FOR BYPRODUCT, SOURCE, OR SPECIAL NUCLEAR MATERIAL

U. S. Nuclear Regulatory Commission
Division of Fuel Cycle and Material Safety
Washington, D.C. 20555

July 1982

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The instructions in this guide, in conjunction with Table 1, specify the radionuclides and radiation exposure rate limits which should be used in decontamination and survey of surfaces or premises and equipment prior to abandonment or release for unrestricted use. The limits in Table 1 do not apply to premises, equipment, or scrap containing induced radioactivity for which the radiological considerations pertinent to their use may be different. The release of such facilities or items from regulatory control is considered on a case-by-case basis.

1. The licensee shall make a reasonable effort to eliminate residual contamination.

2. Radioactivity on equipment or surfaces shall not be covered by paint, plating, or other covering material unless contamination levels, as determined by a survey and documented, are below the limits specified in Table 1 prior to the application of the covering. A reasonable effort must be made to minimize the contamination prior to use of any covering.

3. The radioactivity on the interior surfaces of pipes, drain lines, or ductwork shall be determined by making measurements at all traps, and other appropriate access points, provided that contamination at these locations is likely to be representative of contamination on the interior of the pipes, drain lines, or ductwork. Surfaces of premises, equipment, or scrap which are likely to be contaminated but are of such size, construction, or location as to make the surface inaccessible for purposes of measurement shall be presumed to be contaminated in excess of the limits.

4. Upon request, the Commission may authorize a licensee to relinquish possession or control of premises, equipment, or scrap having surfaces contaminated with materials in excess of the limits specified. This may include, but would not be limited to, special circumstances such as razing of buildings, transfer of premises to another organization continuing work with radioactive materials, or conversion of facilities to a long-term storage or standby status. Such requests must:

    a. Provide detailed, specific information describing the premises, equipment or scrap, radioactive contaminants, and the nature, extent, and degree of residual surface contamination.

    b. Provide a detailed health and safety analysis which reflects that the residual amounts of materials on surface areas, together with other considerations such as prospective use of the premises, equipment or scrap, are unlikely to result in an unreasonable risk to the health and safety of the public.

5. Prior to release of premises for unrestricted use, the licensee shall make a comprehensive radiation survey which establishes that contamination is within the limits specified in Table 1. A copy of the survey report shall be filed with the Division of Fuel Cycle and Material Safety, USNRC, Washington, D.C. 20555, and also the Administrator of the NRC Regional Office having jurisdiction. The report should be filed at least 30 days prior to the planned date of abandonment. The survey report shall:

    a. Identify the premises.

    b. Show that reasonable effort has been made to eliminate residual contamination.

    c. Describe the scope of the survey and general procedures followed.

    d. State the findings of the survey in units specified in the instruction.

    Following review of the report, the NRC will consider visiting the facilities to confirm the survey.

TABLE 1

ACCEPTABLE SURFACE CONTAMINATION LEVELS

NUCLIDES[a]	AVERAGE[b c f]	MAXIMUM[b d f]	REMOVABLE[b e f]
U-nat, U-235, U-238, and associated decay products	5,000 dpm α/100 cm^2	15,000 dpm α/100 cm^2	1,000 dpm α/100 cm^2
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100 dpm/100 cm^2	300 dpm/100 cm^2	20 dpm/100 cm^2
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1000 dpm/100 cm^2	3000 dpm/100 cm^2	200 dpm/100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above.	5000 dpm $\beta\gamma$/100 cm^2	15,000 dpm $\beta\gamma$/100 cm^2	1000 dpm $\beta\gamma$/100 cm^2

[a] Where surface contamination by both alpha- and beta-gamma-emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides should apply independently.

[b] As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

[c] Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

[d] The maximum contamination level applies to an area of not more than 100 cm^2.

[e] The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

[f] The average and maximum radiation levels associated with surface contamination resulting from beta-gamma emitters should not exceed 0.2 mrad/hr at 1 cm and 1.0 mrad/hr at 1 cm, respectively, measured through not more than 7 milligrams per square centimeter of total absorber.

Nuclear Regulatory Commission Staff
Safety Evaluation Report
Related To The Renewal of
Special Nuclear Material License No. SNM-639
For The
Sterling Forest Research Center
Union Carbide Corporation
Tuxedo, New York
Docket No. 70-687

October 1984

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1. INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

By letter dated December 23, 1980, the Union Carbide Corporation (UCC or licensee) applied for renewal of Special Nuclear Material License No. SNM-639 pursuant to 10 CFR Part 70.* Upon request the licensee submitted a revised Consolidated Application, dated June 6, 1984, in support of the license renewal application. Supplements to this application were submitted July 16, September 6 and October 1, 1984. In accordance with 10 CFR Part 51.5(c)(3), the environmental aspects of the proposed license renewal have been assessed separately and are addressed in the Nuclear Regulatory Commission staff's (the staff) Environmental Impact Appraisal, issued in May 1984.

This report documents the staff's review and evaluation of the safety of the continued usage of special nuclear material (SNM) for generation of byproduct material to be used in medical applications at the Sterling Forest Research Center near Tuxedo, New York for a five-year term. Our technical review of radiological safety matters with respect to the renewal of License No. SNM-639 pursuant to 10 CFR Part 70 was based on the Consolidated Application, including amendments thereto. The Consolidated Application is available for public inspection at the NRC's Public Document Room at 1717 H Street, N.W., Washington, DC 20555.

In the course of the staff's safety review, a number of visits were made to the Tuxedo site to meet with the licensee to discuss and observe current operations with the licensee. During the staff's review, the licensee was requested to provide additional information needed for the evaluation. This information was provided by letters from the licensee. We have reviewed the design and operation of the Tuxedo operation to determine that the NRC's safety requirements have been met. Since the activities conducted under Special Nuclear Material License SNM-639 are performed in a facility contiguous with the Union Carbide reactor, the staff considered the relationship of these two licensed activities. In coordination with the Office of Nuclear Reactor Regulation we considered the influence any abnormal condition of either activity might have on the other, including severe accidents. We also considered the interfaces between licensed activities during normal operation; particularly the transfer of targets and fuel from the reactor pool to the transfer canal. All other reactor activities, not related to Building 2 operations, are considered in a separate licensing review under Docket 50-54 and are not addressed here. The licensee also has a radioactive materials license issued by New York State for possession and use of other radioactive material including byproduct material outside of the reactor. The relationship of the state licensed activities to those activities considered in this report is discussed in Section 1.4.1.

1.1 Background

UCC owns and operates the Sterling Forest Research Center, which contains a five megawatt swimming pool-type research reactor and is located on an industrial

*The renewal application was filed not less than thirty days prior to the expiration date of the license. Pursuant to 10 CFR § 70.33(b) the license shall not expire until the application for a renewal has been finally determined by the Commission.

site in the sparsely-settled Sterling Forest development region of Orange County, New York State. The UCC began commercial operation in September 1961 (first criticality) and has been operating on varying, but regular, schedules ever since. The reactor is used principally as a source of neutrons for irradiating target materials, used in fabricating radiopharmaceutical products. These products are used by the medical industry in diagnosis and treatment. A major product of the reactor is molybdenum-99 whose radioactive daughter, technicium-99, is used to meet medical diagnosis requirements in the United States, Japan, and Western Europe.

1.2 Authorized Activities

The renewed license will authorize UCC to possess and use special nuclear material (consisting primarily of uranium enriched to greater than 20% in the U-235 isotope) to prepare targets and process irradiated targets for recovery of selected radionuclides for subsequent medical use; to conduct tests and experiments related to these activities under the provisions of the license; and to conduct related receiving, packaging, waste treatment and shipping operations.

1.3 Possession Limits and Places of Use

The quantities of special nuclear materials that will be authorized by the license renewal are as follows:

Sterling Forest Research Center

A.	Uranium-233 (encapsulated sources)	10 grams
B.	Uranium enriched in U-235 isotope (>20% enriched)	23 kilograms of U-235
C.	Plutonium (encapsulated sources)	2 milligrams Pu-238 10 grams Pu-239 2 milligrams Pu-241 80 grams Pu-Be neutron source

1.4 Scope of Review

The safety review of UCC's license renewal application included review of applications related to the previous issuance of the license and subsequent amendments of the license to date as well as evaluation of the Consolidated Application filed for license renewal on June 6, 1984, as supplemented July 16, September 6, and October 1, 1984. Analyses and conclusions presented in the staff's Environmental Impact Appraisal issued in May 1984 were considered. The scope of review included review of the relationship of the licensed activities considered herein to those activities covered by the NRC reactor operating license and those activities authorized by the New York State radioactive materials license as well as NRC inspection history. The staff's safety evaluation, as summarized in this report, focused on the UCC organization, administrative controls, radiation protection program, nuclear criticality safety program, fire safety capability, accident analyses and emergency response capability as presented in the licensee's Consolidated Application, as supplemented, and the facility emergency plan.

An assessment of the capability of the UCC facilities to withstand natural phenomena events was performed. A portion of the seismic analysis is presented in Report No. SAI-1-148-08-781, "Evaluation Of Seismic Response Characteristics Of Hot Cells and Related Structures And Equipment At The UCC Sterling Forest Laboratory," dated November 30, 1983 and summarized in Section 8.1.1. The tornado and high winds analyses are summarized in Section 8.1.2.

Assessments were made against the requirements of 10 CFR 20 and 10 CFR 70, the guidelines of applicable regulatory guides, and industrial standards and practices. These are identified in the appropriate sections of the evaluation.

1.4.1 Relationship of SNM License to NYS Byproduct License

In connection with our review, we clarified the licensee-regulatory authority relationship between Union Carbide, New York State and the NRC with respect to possession and use of byproduct and special nuclear material by Union Carbide and its relationship to the NRC licensed research reactor.

Three licenses presently exist for the UCC facility: License No. R-81 issued by NRC's Office of Nuclear Reactor Regulation (NRR) for operation of a research reactor pursuant to 10 CFR Part 50; License No. SNM-639 by NRC's Office of Nuclear Material Safety and Safeguards (NMSS) for the possesssion and use of special nuclear material outside of the reactor pursuant to 10 CFR Part 70; and the Radioactive Materials License issued by New York State's Department of Labor for possession and use of other radioactive material including byproduct material outside of the reactor. License operations present a situation in which special nuclear material licensed by NRC is intimately intermingled with byproduct material licensed by New York State.

We concluded that the scope of our licensing review under 10 CFR Part 70 legitimately included the health and safety effects of byproduct materials intermingled with, or, otherwise affected by, special nuclear materials. In particular, 10 CFR 70.23(a)(3) requires a finding that the licensee's proposed equipment and facilities are adequate to protect health and minimize danger to life and property. Thus, our review included an evaluation of the radiological effects of the associated byproduct material, even though the latter is not licensed by the NRC. In performing the NRC review, the staff reviewed the byproduct material license (No. 729-0322) issued by the New York State Department of Labor and discussed its own review procedure with representatives of the State.

Our assessment of environmental effects is contained in our document, "Environmental Impact Appraisal, Union Carbide Corporation, Medical Products Division, Tuxedo, New York, Related to License Renewal of Nuclear Material License No. SNM-639, U.S. NRC, May 1984." This appraisal concluded that the environmental impacts related to the proposed license renewal were not significant and did not warrant preparation of an environmental impact statement. Accordingly, a

*However, NRC may not be able to condition its licenses to mitigate all possible environmental effects. For example, NRC cannot impose conditions that vary the terms of an NPDES permit for the same effluent stream. See Section 511(c)(2) of the Federal Water Pollution Control Act Amendments of 1972, 86 Stat. 893, (88 U.S.C. 1371(c)(2).

Negative Declaration was issued (49 FR 22169). The appraisal evaluated combined effluents resulting from the entire scope of operations conducted in Building 1 and Building 2 (reactor and hot cell laboratory operations).

1.5 Compliance History

The staff reviewed NRC inspection reports of UCC's activities under License SNM-639 and supplemented this analysis by discussion with Region I and UCC personnel. The following areas were examined:

- frequency of exposure incidents
- location of incidents
- cause of incidents
- Region I citations issued to UCC.

In the review, the staff did not identify a pattern of chronic or repetitious problems with respect to the NRC regulations and license conditions. This, in part, provides a basis for the renewal of Special Nuclear Material License, No. SNM-639.

2. CONDUCT OF OPERATIONS

2.1 Introduction

The staff reviewed the commitments made by the licensee to maintain technically qualified personnel in key safety positions and to implement adequate internal administrative control procedures. The staff, in its review, used Regulatory Guide 3.52, "Standard Format and Content for Uranium Fuel Fabrication Plant Applications," dated July 1982, as guidance. This document was prepared specifically for uranium fuel fabrication plants; however, the subjects of organization, audit and review, and policies and procedures, and many technical topics are generally applicable to other nuclear material facilities. The following description of the current management organization and administrative procedures is provided.

2.2 Management Organization

UCC conducts the Sterling Forest Research Center activities under the management of the Medical Products Division of its Union Carbide Subsidiary B. Subsidiary B was incorporated on December 12, 1980 in Delaware with its corporate office in Danbury, Connecticut. An organizational schematic of the Medical Products Division management is shown in Figure 2.1 where four levels of authority are provided. Level I consists of individuals responsible for the facility license and site administration; Level II of individuals responsible for the Reactor and Hot Laboratory facility operation and management; Level III of individuals responsible for the daily operations in the Reactor and Hot Laboratory; and Level IV of operating staff.

UCC has 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 at the Sterling Forest Research Center. The responsibility for implementing this basic policy is delegated through line managers to the manager and supervisor of each activity in which radioactive materials are handled, used, or stored. Additionally, UCC has a technical staff to review activities requiring adherence to regulations, license conditions and ALARA principles. The pertinent responsibilities of management within this organization structure are as follows:

- Site Manager - Also known as the General Manager, manages the radiochemical business including Reactor operations, Hot Laboratory operations, Radiochemical separations, facility maintenance, facility licensing and regulatory matters, and accounts for financial aspects of Reactor, Hot Laboratory and Radiochemical Production operations. This position reports to a Corporate Division President (currently, Medical Products Division).

- Nuclear Operations Manager - Manages Reactor and Hot Laboratory Operations and NRC Regulatory Affairs. Designs and implements changes to the facility. Assures compliance of operations with license conditions and regulatory requirements. Maintains current licenses and obtains revisions to licenses as they become necessary. Day-to-day operations and training are carried out through delegation of specific duties to the Reactor Supervisor, the Facilities and Services Engineer and the Project Engineer.

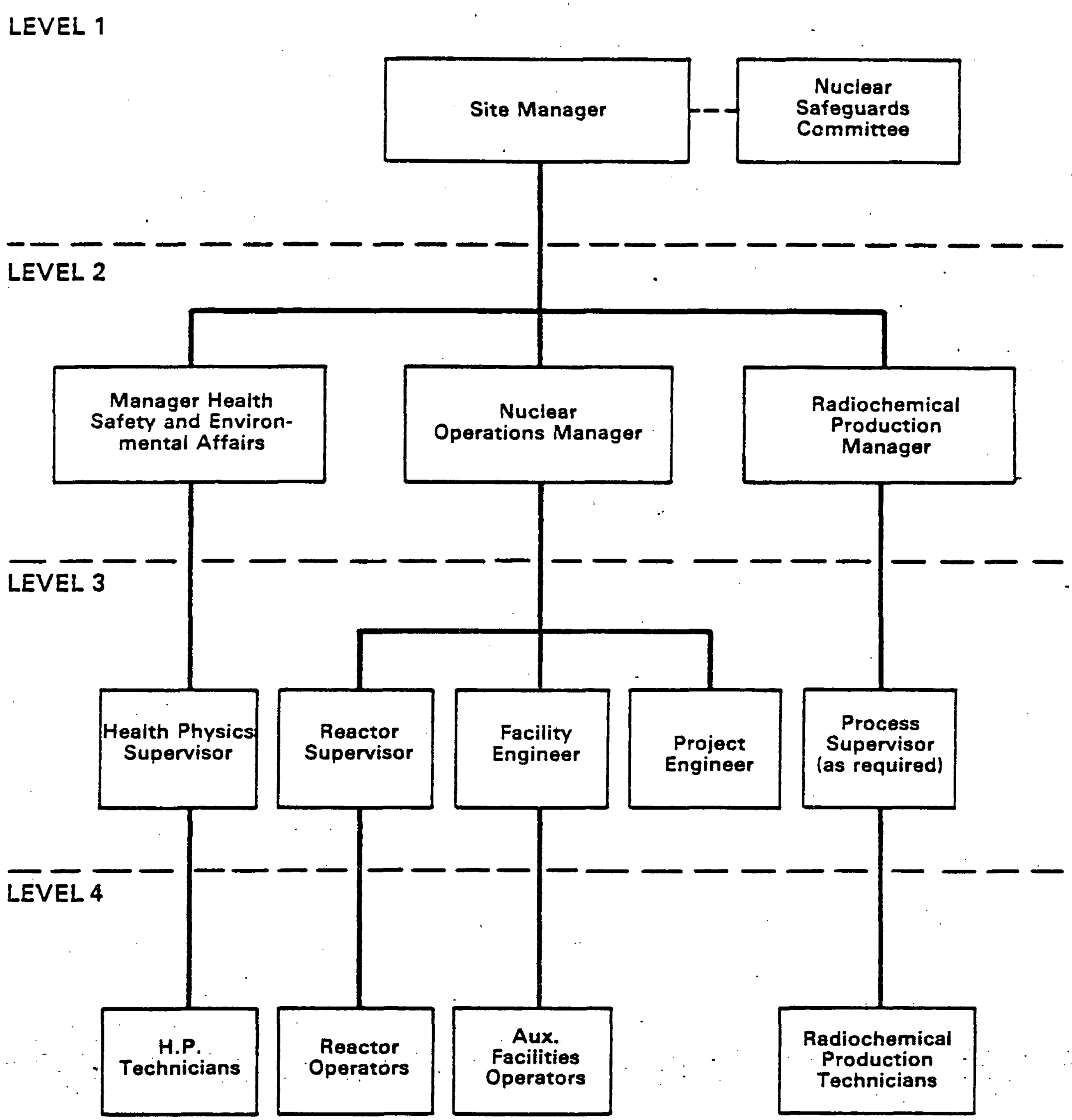


Figure 2.1 Union Carbide Medical Products Division Organization Chart

- Health, Safety and Environmental Affairs Manager - Manages the on-site and off-site environmental monitoring program per license and regulatory requirements, and ALARA principles. Carries out the duties of the Site Accountability Officer for special nuclear material under the Fundamental Nuclear Materials Control (FNMC) Plan. Coordinates activities of all operations groups involving health, safety and environmental protection against radiological and other, more common hazards. Day-to-day operations and training are carried out through delegation of specific duties to the Health Physics Supervisor, Radiation Safety Officer and General Safety Supervisor.

- Radiochemical Production Manager - Manages radiochemical production operations safely and efficiently. This scope of work includes target production, radiochemical separations, waste packaging and uranium accountability. Day-to-day operations and training are carried out through delegation of specific duties to the Hot Cell Production Supervisor, the Target Production Supervisor, the Inventory Control Supervisor and the Radiochemical Packaging Supervisor.

The responsibility for carrying out operations during routine absences of management personnel will normally be delegated to the next senior line manager. During emergencies the senior line manager present is the responsible individual to conduct operations until the emergency staff arrives.

2.2.1 Safety Review Committees

As part of its administrative control of site activities, UCC has established two safety review committees. One of the two committees is the Nuclear Safeguards Committee (NSC) which is composed of senior management and technical personnel. The NSC is responsible for reviewing and auditing operations with regard to nuclear hazards. The second is the General Safety Committee (GSC) which is composed of the managers of each major operating organization on site as well as the Health, Safety and Environmental Affairs Manager, or their designated alternates. The GSC is responsible for reviewing and auditing all operations and facilities on site concerning hazards such as fire, electrical shock, work practice restrictions, provision and maintenance of lifting equipment, etc.

2.3 Administrative Controls and Procedures

UCC, at its Tuxedo facility, has an internal review system to ensure that activities at the site are conducted in a safe and efficient manner. Procedures have been developed for initiation and review of all changes in the design or location of existing and new equipment, changes to storage areas, emergency procedures, operating limits, and/or operating procedures, and changes in nuclear criticality or radiation protection controls. Unless the proposed changes meet the requirements of the conditions of the license and applicable regulations, UCC will not administer these changes without specific review and approval of the NRC. As part of these procedures, a written safety review is prepared and approved by the Nuclear Safeguards Committee prior to implementation of the change. In case of emergency it is recognized that departure from established procedures may be necessary to minimize the consequences of the emergency conditions, and to regain control of radiation exposure conditions or prevent personnel injuries and dispersion of radioactive materials. Supervision authorizes these departures from procedure as discussed in Section 2.6.

UCC has also developed its Tuxedo site safety standards as part of its nuclear safety program. The following categories are included in the Tuxedo site safety standard framework: administration, dosimetry and bioassay, radiation surveying, air sampling, instrumentation, audits, reviews, training and emergency coordination. The principal features of several of these standards, respresentative of most of the categories, are summarized as follows:

- Administrative - These standards describe the UCC nuclear safety program and are intended to promote a more formal approach to keeping doses ALARA, to identify and promote continuance of good practices, and to promote further improvements where practicable.

- Dosimetry and Bioassay - The standards in this category include the administration of the bioassay program for monitoring possible internal contaminants by urine analysis and thyroid counting, and the issuance, collection, and data control of film badge and other dosimeters for monitoring external exposures.

- Radiation Surveys and Air Sampling - Standards in this category define radiation survey work routines, air sampling routines, action to be taken when performing personnel decontamination, and transfers, shipments and receipts of radioactive material.

- Instrumentation - Included in this category are standards describing various types of radiation protection instrumentation, and their operation and calibration.

- Audits - Included in this category is a standard describing the independent review and audit functions performed by the Nuclear Safeguards Committee.

- Reviews - There are several standards and procedures in this category, all dealing with informal and formal review activities of the Nuclear Safeguards Committee against operating procedures activities, technical specifications, and appropriate regulations. Included are compliance reviews, radiological safety reviews, criticality safety reviews, facility equipment/systems reviews and change authorization reviews.

- Emergency Plan - This procedure describes the detailed functions and responsibilities of nuclear safety personnel during emergency situations.

2.4 Audits and Reviews

A written review and audit program has been prepared and conducted under the direction of the Nuclear Safeguards Committee within the UCC. It defines the scope of the program, designates the personnel and their area of competence, the frequency of audits, required audit documentation, and defines the areas to be audited. The persons performing the audits do not have direct responsibility for the areas being audited. The specific areas audited at least annually include the quality assurance program, measurement procedures, SNM accountability, SNM criticality, packaging and transport of waste, radiation safety, and hot laboratory operation.

2.5 Personnel Training Programs

UCC has established training and requalification programs for those personnel responsible for handling radioactive material and those providing support to the various activities. The training and annual requalification programs are designed to assure an adequate understanding by the employees, who are authorized to handle SNM, of the hazards and complexities of handling radioactive materials from the standpoint of both radiological and nuclear criticality safety. The initial radiation safety training and continuing training program is designed to develop an understanding of rules and procedures and to promote safety consciousness and sound safety practices.

Responsibility for training in radiation safety lies with the Health Physics Department. The UCC safety standards require that Health Physics personnel determine the duration and depth of training for each employee. Such training is dependent on the job assignment and previous experience of each new employee. The Health Physics personnel also determine the need for additional formal training from follow-up observations and the results of personnel monitoring. Follow-up training commensurate with the work environment and the employee's work performance is determined by employee supervision. The employee also receives on-going training in the form of formal sessions and on-the-job reviews, etc.

UCC requires that each employee who may work with special nuclear material undergo an annual requalification program to demonstrate an understanding of criticality safety requirements as well as other license conditions. Completion of the requalification requirements includes a signoff by the employee and supervisor.

Training and retraining in fire protection and emergency procedures are also included in the UCC training program in addition to the maintenance of training records.

2.6 Emergency Plan

The staff has reviewed the Union Carbide Nuclear Facility Emergency Plan, which was submitted on September 3, 1982 and supplemented on August 8, 1983. The latter was UCC's response to additional emergency planning questions, requested by letter dated June 23, 1983.

The Emergency Plan is an integrated plan which encompasses both Reactor License R-81 and Materials License SNM-639. The plan is based on the requirements of 10 CFR 50, Appendix E, "Emergency Plans for Production and Utilization Facilities," and NUREG-0762, the "Standard Format and Content for Radiological Contingency Plans for Fuel Cycle Facilities." The guidelines of Regulatory

Guide 2.6 and ANS 15.16, "Emergency Planning for Research Reactors," were also used.

In our review, the staff determined that UCC's site-wide Emergency Plan is adequate to demonstrate that the licensee has accomplished the purpose and intent of radiological contingency planning, by assuring (1) that the facility is properly configured to limit releases of radioactive materials and radiation exposures in the event of an accident, (2) that a capability exists for measuring and assessing the significance of accidental releases of radioactive materials, (3) that appropriate emergency equipment and procedures are provided onsite to protect workers against radiation hazards that might be encountered following an accident, (4) that notifications are promptly made offsite to Federal, State, and local government agencies, and (5) that necessary recovery actions are taken in a timely fashion to return the plant to a safe condition following an accident.

The principal location where special nuclear material is handled is in the Hot Laboratory Building. The building, as described in the emergency plan, is properly configured to limit radioactive releases and radiation exposures from abnormal operations in that: all exhaust air from the Hot Lab passes through roughing filters, absolute (HEPA) filters, and when necessary through charcoal filters prior to passage to a monitoring system and 50-foot stack; the Hot Cell filter banks are protected by a CO_2 fire suppression system; ventilation systems are designed to assure positive, continuous flow of air from non-radioactive areas to radioactive areas; and eight radiation monitors with audio and visual alarms are located throughout the building.

The radiological contingency planning organization, as described in the emergency plan, provides adequate preplanning for emergency response. The arrangements for offsite assistance as well as the responsibilities of various supporting organizations are established. Procedures for implementing emergency responses for the listed accident scenarios are described, and general plans for recovery and reentry are developed.

The emergency plan improves UCC's ability to protect against, respond to, and mitigate the consequences of an accident involving radioactive materials. The staff concludes that UCC's site-wide Emergency Plan satisfies the radiological contingency planning requirements of NUREG-0762 and provides a basis for an acceptable state of emergency preparedness.

2.7 Conceptual Decommissioning Plan

In its Consolidated Application, the UCC submitted a conceptual decommissioning plan for its Tuxedo facility. The plan was prepared for the Reactor License R-81 renewal application, Docket 50-51. The plan included two alternatives in decommissioning the facilities depending ultimately on the final utilization UCC would want for the decommissioned nuclear facilities. In one alternative, UCC will adhere to levels not exceeding those specified in Table I in the NRC prepared Annex B to the License, "Guidelines for Decontamination of Facilities and Equipment Prior to Termination of Licenses for Byproduct, Source, or Special Nuclear Material," so as to enable release of the property for unrestricted use.

The second alternative would enable the UCC to maintain the nuclear utilization capabilities of the facility along with restricted access under a byproduct materials license (subject to the regulatory jurisdiction of New York State).

These two alternatives in the decommissioning plan also include a discussion of the general considerations for plant decontamination, the procedures to be followed during decontamination and an estimate of the costs for decontaminating the Tuxedo facilities and site.

The UCC decommissioning plan was reviewed by the staff and is adequate in that it complies with NRC guidelines, the procedures proposed are reasonable, acceptable to the staff, and the estimated costs for decontamination are realistic.

2.8 Staff Evaluation - Conduct of Operations

The staff concludes on the basis of the review and comparison with the proposed guidelines in R.G. 3.52 standard that the licensee's management organization and administrative controls are adequate to protect health and minimize danger to life and property [10 CFR 70.23(a)(2) and 10 CFR 70.23(a)(4)]. The license application commitments and the demonstration portion of the license application present the level of detail necessary for the staff to make conclusions regarding safe operation.

3. SITE AND FACILITY CONSIDERATIONS

Licensed activities are conducted by Union Carbide at Sterling Forest Research Center near Tuxedo, New York. This chapter discusses the type of facilities, their uses and engineered safety-related features.

3.1 Site Description

The UCC plant is located within the town of Tuxedo in Orange County, New York. Orange County is in southeastern New York state, is bordered on the south by New Jersey, and is approximately 40 miles northwest of New York City. Tuxedo is in the extreme southeastern corner of Orange County approximately four miles north of the New Jersey state line. The plant site is located on 100 acres of land, owned by Union Carbide, in an area known as Sterling Forest and is about 3-1/4 miles northwest of the village of Tuxedo Park. Features within 10 miles of the site are shown on Figure 3.1. The plant itself has been constructed along Long Meadow Road on the eastern slope of Hogback Mountain at an average elevation 800 feet above mean sea level (msl).

There are five principal buildings at the plant site which are identified as follows:

Building 1	Reactor
Building 2	Hot Laboratory (buildings 1 and 2 are structurally joined)
Building 3	Maintenance
Building 4	Administration
Building 5	Heating Plant

There is an additional small concrete block structure at the north end of the plant site used for temporary storage of drummed, miscellaneous low-level radioactive wastes.

The plot plan of the plant site is shown in Figure 3.2.

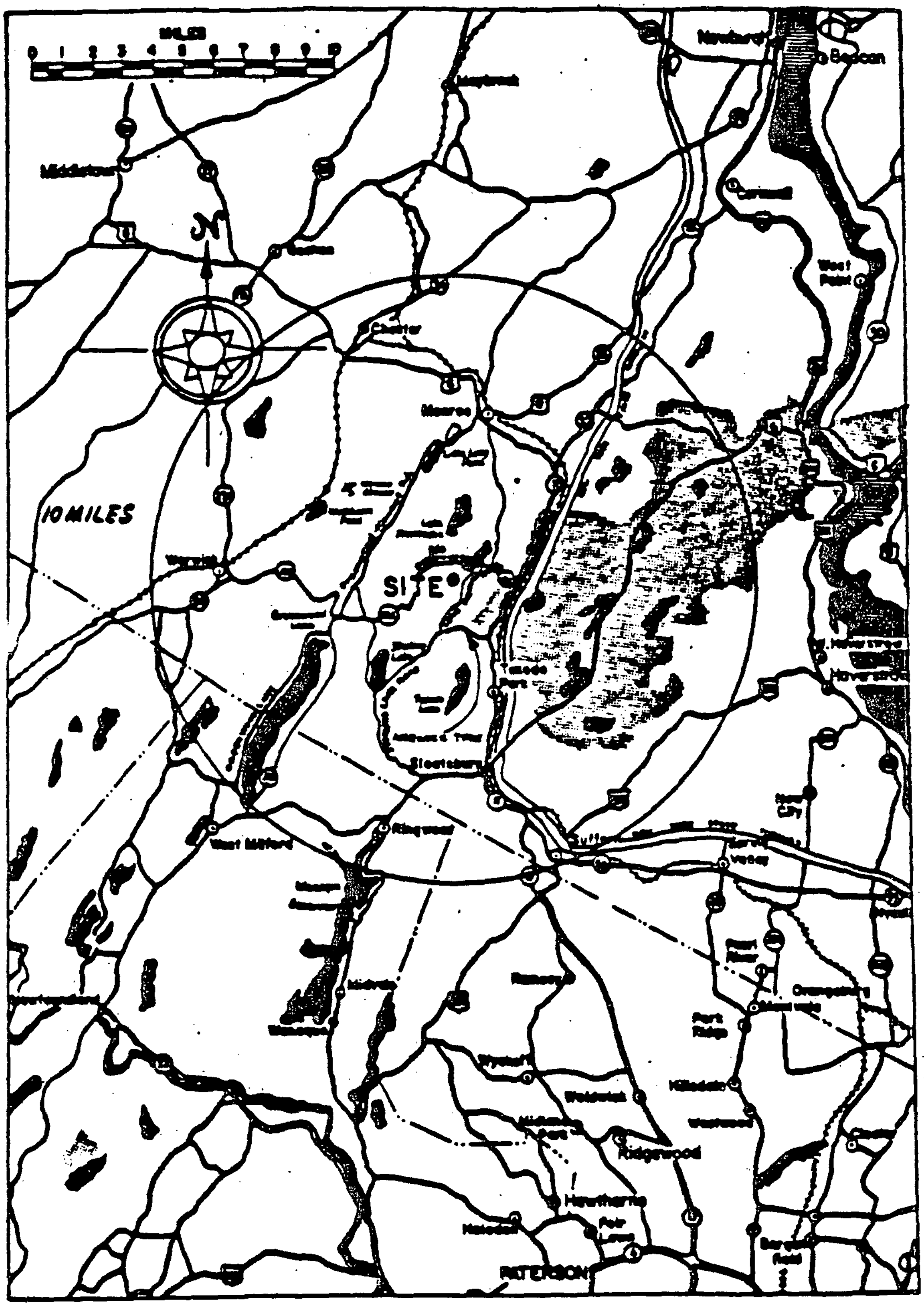
3.2 Facility Description

The Hot Laboratory (building 2) is a steel and concrete building 139 feet long, 57 feet wide and 37 feet high, which shares a common wall (its southern wall) with the Reactor Building (building 1). A 12-foot deep, concrete, water-filled canal connects the reactor pool to a location just below hot cell 1.

In the Hot Lab there are five hot cells, each having four-foot thick walls of high-density concrete, and the cells are separated from each other by four-foot thick high-density concrete walls. The floor plan for the Hot Lab is shown in Figure 3.3. The cells are general purpose units designed to accommodate a variety of operations, including chemical experiments, radiochemical separation of isotopes, physical testing for evaluation of irradiated material, solid state investigations and metallurgical work. A general description of the cells is presented below.

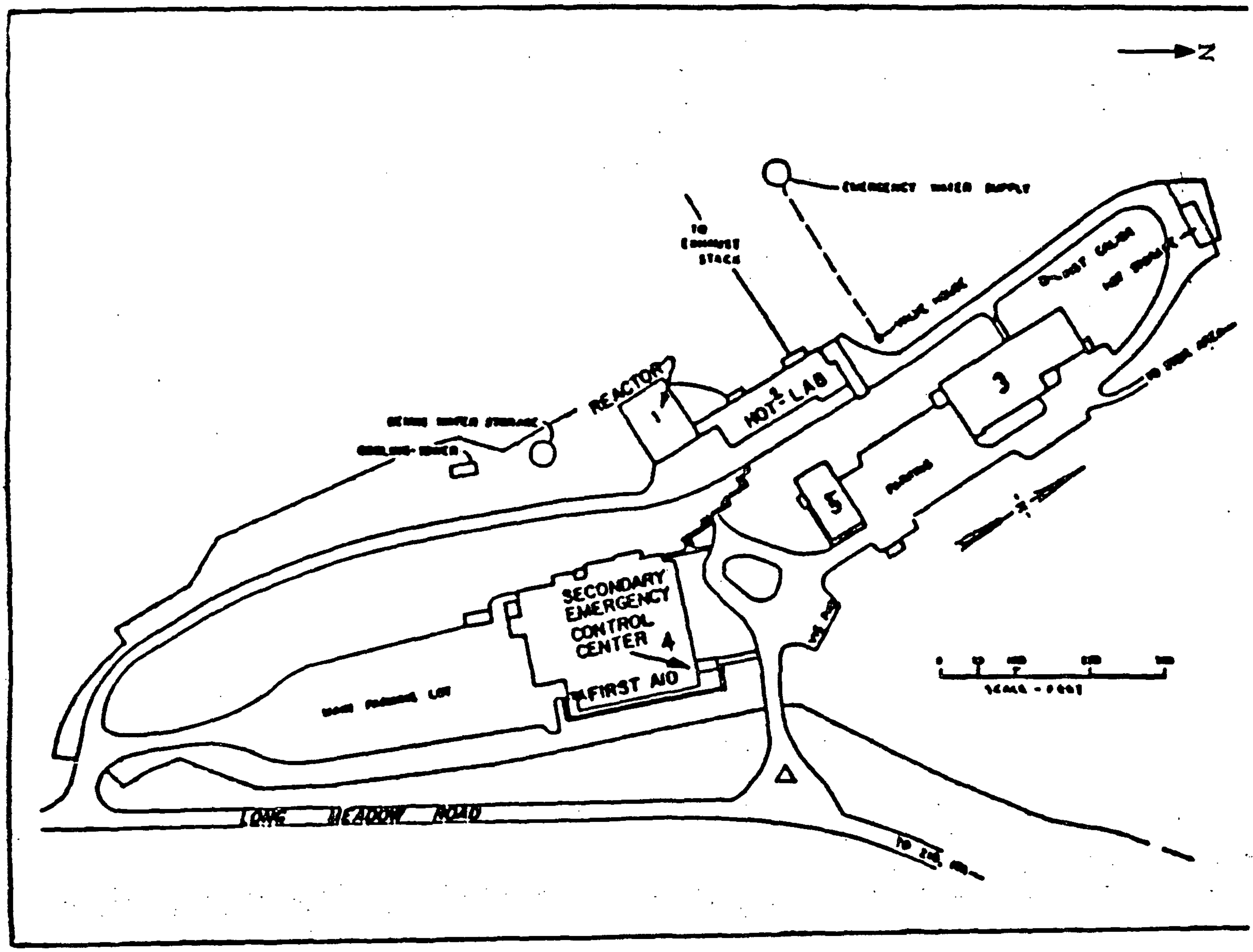
Features Within 10 Miles of the Tuxedo Site

Figure 3.1



UCC's Sterling Forest Research Center

Figure 3.2



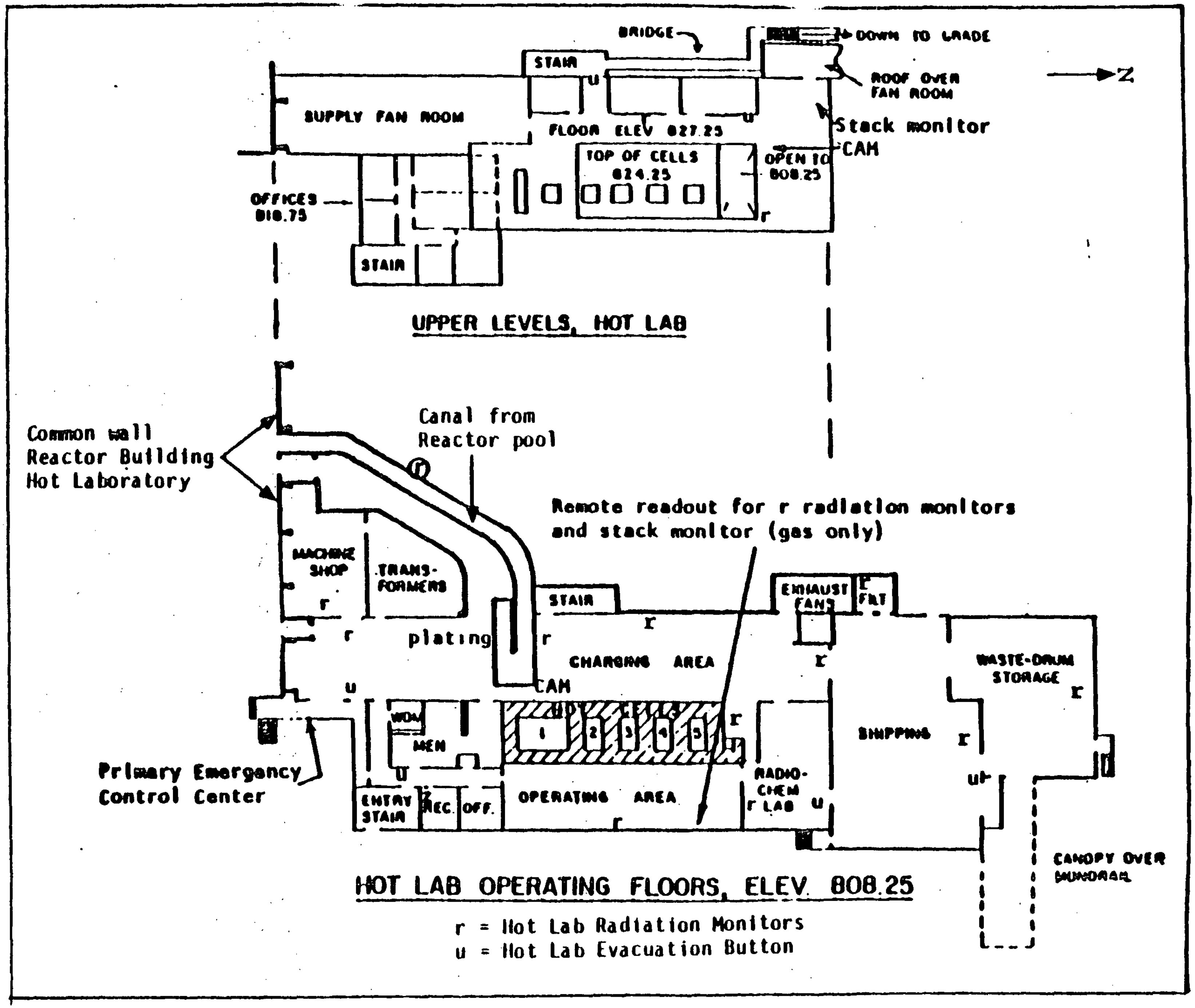


Figure 3.3

Hot Laboratory Building Floor Plan

Cell 1 is 16-feet wide by 10-feet long by 15-feet in height. This cell is equipped with a electromechanical remote handling arm (750 lb. capacity), one pair of Heavy Duty Model 8 manipulators and one pair of Standard Duty Model 8 manipulators. Two Corning 4-foot thick glass shielding windows are located in the front shielding wall of Cell 1. These viewing windows consist of Corning's "Radiation Shield Standard Assembly 1480," which is their standard unit for 4-foot shielding walls. The windows are constructed of five sections of 3.3 density lead glass each 9-1/2 inches thick.

A Kollmorgan periscope, currently in use in Cell 1, can be relocated to any of the other cells. With auxiliary attachments on the periscope it is possible to do in-cell microscopy and to take photographs of specimens in the cell.

Cells 2, 3, and 4 are 6-feet wide by 10-feet long by 12.5-feet in height, while Cell 5 is 6-feet wide by 10-feet long by 25-feet in height. Cells 2, 3, 4, and 5 are each equipped with a Corning 4-feet thick glass shielding window and all cells are equipped with one pair of Model 8 Master Slave Manipulators.

Major access to all the cells is possible through the rear doors (7-feet wide by 6-feet high by 4-feet thick) which can be withdrawn utilizing electrical drives. The electrical connection and power supply to drive these doors are kept locked to prevent unauthorized entrance. An alarm sounds when any of these rear access doors is opened. The access doors for the cells are motor driven through a 1200:1 reduction worm gear and move on steel rails located in the floor of the charging area.

Access to all cells is also possible via top roof openings containing removable plugs. The roof and roof plugs of all cells are 3-1/4-feet thick magnetite concrete. The roof plug is made up of three 14-inch thick concrete slabs which must be removed individually with a 10-ton capacity overhead crane. A 6-inch diameter charging sleeve located in the center of the roof plug is fitted with an 8-inch long lead-filled steel plug. Two 4-inch diameter charging sleeves also are provided through the roof. They have magnetite plugs 6 inches in diameter at the exterior surface and are stepped to 4 inches in diameter 18-inches from the interior surface. There are laboratories and a solution make-up room above the charging area, but areas directly above the cells are not occupied.

Radioactive samples, specimens, isotopes, etc., are transferred through the previously mentioned canal and brought into Cell 1 via an automatic elevator mechanism.

The area on the front side of the cells is the operating zone and is maintained as a radioactively clean area. The viewing windows, manipulator controls, intercell conveyor controls, in-cell service controls (air, water, vacuum, gas) and periscope are located in this area. The operating control panels for the ventilation system and the Radioactive Waste Water Treatment System are located at the north end of this area.

Radiation levels in the Hot Lab are monitored by a combined area radiation monitoring/criticality monitoring system. The system is linked to a master panel located in the operating area. Area radiation monitors are normally set to alarm at 5 mR/hr. The setpoint at which each monitor activates the building criticality evacuation alarm is determined by the detector location and consideration of intervening shielding from the location of a potential criticality.

In the front shielding wall of each cell there are twelve removable 2-inch diameter stepped pipe sleeves, one 8-1/2-inch diameter sleeve (to accommodate the Model 8 manipulators). When the sleeves are not in use magnetite shielding plugs are placed in the sleeves. Special services not available within the cell (such as inert gas, high pressure air, natural gas) can be fed into the cells through special plugs which can be inserted in place of the standard 2-inch diameter stepped pipe sleeves. Locking bars are used to prevent accidental removal of any of these plugs.

In the rear shielding wall of each cell there are five 2-inch diameter stepped pipe sleeves. Each rear cell door also contains one 8-inch diameter stepped sleeve. These sleeves provide additional access ports from the charging area in the cells. They contain magnetite shielding plugs when not in use.

An intercell conveyor permits transfer of samples or equipment between cells from an external loading station. The conveyor loading/operating station is provided with a shielded viewing window, a remote manipulator, and a radiation controlled interlock on the access door. The drive unit for the conveyor is located outside the shielded area so that maintenance can be performed with minimum radiation exposure.

The charging area is located to the rear of the cells. Controls for the rear access doors to the cells are located here. Access to the decontamination room, exhaust fan room, waste treatment facility and conveyor loading station are from the charging area.

Just north of the charging area is the shipping area with access to a loading dock along its eastern wall. The north end of the shipping area is, in turn, separated from the Waste Drum Storage Facility by swinging doors.

The Waste Drum Storage Facility is a recent addition. The top of this structure (adjacent to the north end of the Hot Lab) is level with the ground floor of the Lab. The bottom of the structure is approximately eight feet below ground level. It is essentially a concrete block 35 feet by 44 feet and eight feet thick in which 121 cylindrical pits, seven feet deep, and 28 inches in diameter are provided for waste storage. Each storage pit has a removable shield plug four feet deep with a diameter of 32 inches in its top 18 inches, and a 28-inch diameter over its remaining length. The reinforced concrete at the bottom of the storage pits is one foot thick. No storage pit is located in the concrete closer than four feet from the outside wall of the block, i.e., four feet of concrete shielding is provided between any waste drum and all exposed surfaces. A special mobile crane has been manufactured for shield plug removal, waste drum placement and shield plug replacement. The facility is roofed and windowless. It has three personnel access doors and swinging doors at the exterior truck loading platform.

This facility will provide for lag storage of high specific activity, low-level radiological waste for six to eight months during which time radioactive decay will take place so that the specific activity of the waste will be lower at the time of retrieval and shipment for ultimate disposal.

Low-level radioactive specimens or samples will be handled in the Radiochemical Laboratory. Equipment available in the laboratory includes standard laboratory benches with stainless steel tops, glove boxes and hoods.

Operations in this laboratory involving higher level radioactive gases will be conducted within special hoods. There are three hoods; two regular and one walk-in unit. These hoods, with all interior surfaces of stainless steel, are 6 feet wide and are designed for work with radioactive materials. All flows from these hoods pass through roughing filters, and absolute filters (these are standard units) prior to passage to an exhaust fan and monitoring system. Supporting non-radioactive analytical work also is done in this laboratory.

Three target make-up laboratories are located in the second floor area. These labs are currently used for the uranium target preparation. They can also be used for work similar to that described for the Radiochemical Lab. All operations involving uncontained radioactive materials will be carried out in hoods or glove boxes of the type used in the Radiochemical Lab.

3.3 Service and Support Systems

The licensee has designed and operates a ventilation system to continuously confine and control radioactivity within its Tuxedo facility. In consonance with "As Low As Reasonably Achievable" (ALARA) criteria, the ventilation system has been improved to remove most of the radioiodine released to cell air during processing. Three carbon filters in series are used for this purpose, each with a design efficiency exceeding 99 percent.

A negative pressure of about 0.3 inches of water referenced to atmospheric pressure is maintained in the hot cells to assure that the direction of air flow is from operating areas into the cells. This is also true of hoods and glove boxes in which radioactive materials are handled.

A simplified diagram of the ventilation system is shown in Figure 3.4. A reserve fan is maintained on standby to assure proper air flow and negative pressure if the main exhaust fan should fail. Procedures require that process operations cease at that time. Two emergency generators are also available to assure continued operation of the ventilation system.

The principal service and support systems of ventilation, confinement, and utilities as they exist at the Tuxedo facility are described in the following sections.

3.3.1 Ventilation

In order to assure confinement of radioactivity during all operations, the licensee has provided a ventilation system shown in simplified form in Figure 3.4. The principle that air flow from clean areas to areas of successively greater potential contamination has been followed. Air is supplied to offices and other occupied areas by two separate fans. It is drawn into laboratory hoods, hot cells, and exhaust ducting by a single exhaust fan. Normal flow rates are shown on Figure 3.4. Exit air joins with the reactor exhaust air and thence to a common stack. The stack is located near the crest of Hogback Mountain on the west side of the facility. The fifty-foot stack, and exhaust duct leading to it, rise 187 feet above grade level.

The hot cells are not sealed and about 2,500 cubic feet of air per minute flows around doors, cell covers, and equipment ports into the cells. If a door at

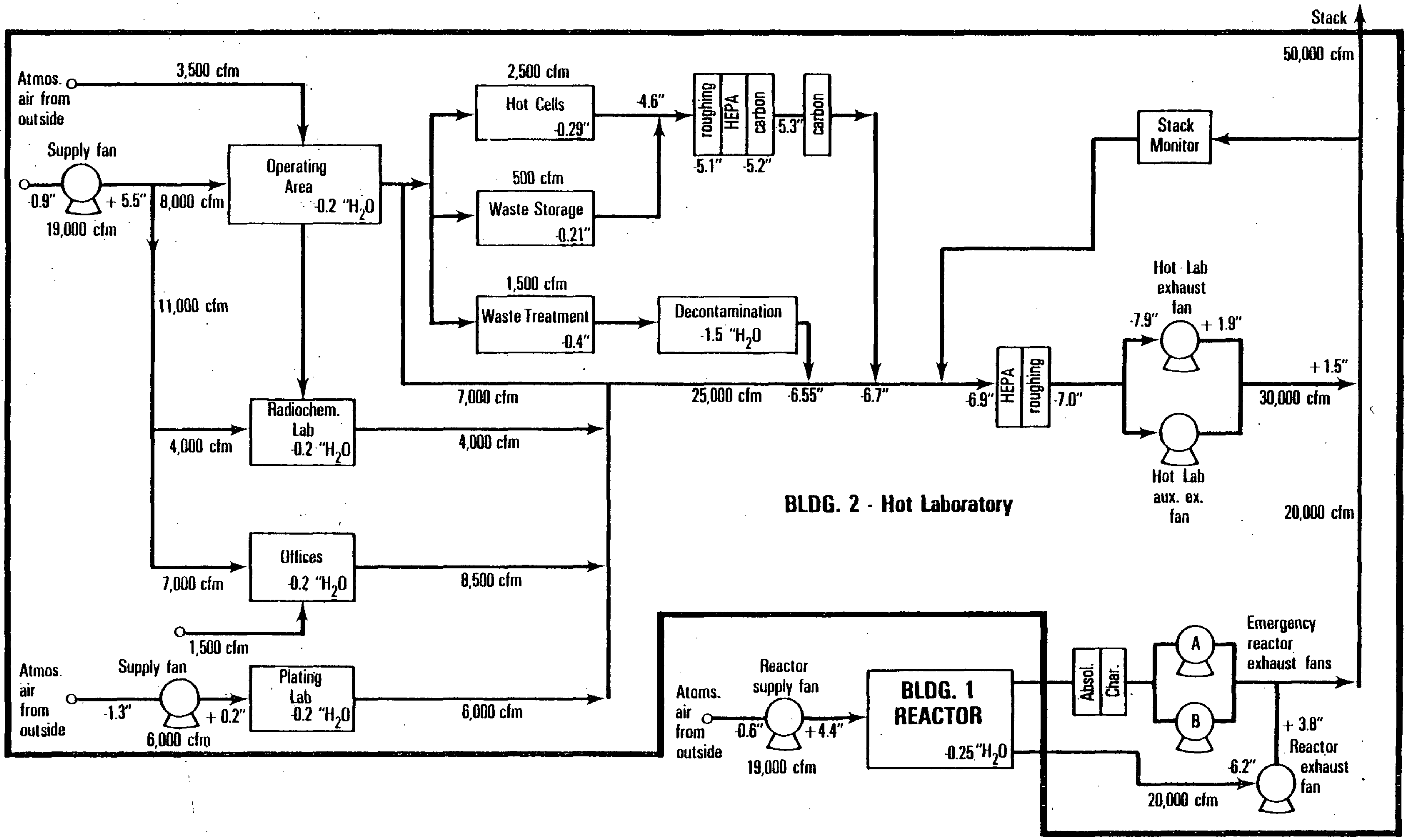


Figure 3.4 Simplified Ventilation System of Union Carbide Facility Tuxedo, N.Y.

the rear of the hot cells is opened, the total flow inward might increase to about 6,500 cubic feet per minute. A door is opened about once per week.

All air effluent from the hot cells passes through a series of filters prior to joining additional exhaust air from other facility areas. The hot cell exhaust splits into two parallel trains consisting of a roughing filter, a high efficiency particulate air (HEPA) filter, and two carbon filters (shown as one in Figure 3.4). Downstream of the parallel trains is a single train consisting of an additional roughing filter and an additional carbon filter. During maintenance an auxiliary fan and exhaust train is used consisting of a roughing, HEPA and carbon filter.

The HEPA filters are used to remove particulate radioactivity at efficiencies of 99.95% or greater. They are individually tested for efficiency prior to installation and then tested in place after installation. The carbon filters are used specifically to remove radioiodine from the exhaust air from the hot cells. These filters usually exhibit efficiencies for halogen removal of greater than 99%. The series of three carbon filters, therefore, can be expected to remove all but 10^{-6} of incident iodine. The minimum overall efficiency stipulated by the technical requirements for radioiodine is 99.5%. The combined stream of exhaust air is then further passed through a roughing filter and final HEPA filter.

Normally the fans are powered by a three phase, 440 volt electrical supply. If normal power fails, either exhaust fan (main and auxiliary) will be powered by either of two emergency generators, as described in Section 3.3.3. The switch-over is automatic; the main exhaust fan operates in this mode at half speed.

Based on the staff's review and evaluation of the licensee's ventilation system as described above, we conclude that the system will provide adequate confinement capability during normal and foreseeable abnormal circumstances.

3.3.2 Confinement

Primary barriers for SNM confinement at the facility are provided by the process equipment, hoods, and, glove boxes. The glove box system works with the ventilation system described in Section 3.3.1, so that leakage of air is normally into the glove box. The glove boxes are enclosures constructed of welded stainless steel or aluminum framework onto which are clamped gasketed panels or windows. Manual operations within the glove boxes are performed through gloves which are sealed to open ports in the glove box panels.

The secondary confinement barrier consists of rooms, building walls, and the building ventilation systems. The Hot Laboratory and Reactor buildings are of concrete construction.

3.3.3 Utilities

Normal operating power is provided by a commercial utility company and is standard 3 phase high voltage supply which is stepped down to 440 V by a dedicated transformer which is located in the motor control center in the Hot Laboratory building.

Emergency power for the facility is provided by two auxiliary generators that are driven by internal combustion engines. One is rated at 50 Kw and is fueled with gasoline. The other is 45 Kw and is fueled with natural gas. When normal power is lost these auxiliary generators automatically start in sequence and when they attain their rated capacity an automatic transfer of the emergency load is accomplished. The following safety-related equipment receives electrical power from these generators:

1. Partial lighting of all areas.
2. Reactor control console.
3. Reactor and Hot Laboratory supply fans (at half speed).
4. Reactor and Hot Laboratory exhaust fans (at half speed).
5. Reactor Building doors.
6. Reactor Emergency exhaust fan and evacuation and containment system equipment.
7. Reactor beam tube ventilation and flushing system.
8. Fuel pump for emergency generator storage fuel tank.
9. Hot Laboratory auxiliary exhaust fan.
10. Certain electrical power circuits for standing and mobile emergency equipment (i.e., radiation monitors) criticality monitoring and alarm system, etc.

When normal power is restored to the motor control center the load is automatically retransferred from the emergency generators after the normal supply of power has been sustained for 1 minute.

The Hot Laboratory exhaust-fan load can be switched to either emergency generator by a manual transfer switch. There is a 6-day gasoline supply for the 50 Kw generator and the 45 Kw generator can be continuously fueled with natural gas from the utility company distribution system. Both generators are tested weekly for operability.

Battery powered emergency lighting is provided initially to aid the safe movement throughout the buildings before the emergency load is supplied from the auxiliary generators.

Compressed air for operation of safety-related ventilation dampers and various other non-safety related functions is supplied from the site utilities building (Heating Plant, Building 5). There are two compressors, one service unit and one on stand-by, that supply the site requirements. One compressor is powered by an emergency generator located in the heating plant. In the event of loss of compressed air the ventilation dampers are monitored and programmed to automatically go into emergency sequence. When normal pressure is resumed the ventilation system must be reset manually.

Water to the site is supplied by the local water company. It is drawn from the Indian Kill reservoir, filtered and supplied through a network of mains that supplies all local residents and industrial facilities. The water main system includes a head tank to ensure adequate pressure. The water main system includes a nearby head tank to assure an adequate supply of water for the fire main and for emergency cooling of the reactor in the event of interruption of the normal supply.

3.3.4 Staff Evaluation - Service and Support Systems

Based on the above discussion, we conclude that the ventilation, confinement and utilities systems provided by UCC for Sterling Forest Research Center's special nuclear material handling facilities are satisfactory for normal and potential abnormal operating conditions. The integrity of secondary confinement barriers and systems in all process building is adequate. Well-engineered equipment and responsible operation and control of the systems have contributed to safe confinement practices.

3.4 Seismic Analysis

For our seismic review the staff has relied on seismologic information submitted by UCC in its Consolidated Application, the scientific literature, and information obtained in connection with the Indian Point Station, a nearby power reactor located in New York. This information was developed as a result of general interest in seismic safety of power reactors and has resulted in intense study of the seismic activity in the vicinity of the Ramapo Fault in the New Jersey-New York Highlands and estimates of the probability of occurrence of strong earthquakes at those reactor sites.

Geologically, the UCC site is in the Precambrian Hudson Highlands, an area that has historically had seismic activity, but at relatively low magnitude levels. Local seismic networks in the last decade have helped define the features of the seismicity in the New York City metropolitan region of which the Hudson Highland is a part. The pattern of earthquakes as located by instruments in this region has the same general features as the longer term (several hundred years) historical record. The earthquake locations, which have been determined from the seismic network data, are accurate to within a few kilometers, while the historical data are only accurate to within a few tens of kilometers.

UCC in its discussion of the seismicity for the site relies almost exclusively on one reference (Aggarwal and Sykes, 1978) and the position that much of the seismicity in the region is attributable to the Ramapo fault which is located about 12 kilometers at its closest point to the site. Aggarwal and Sykes (1978) found two lines of evidence which they interpreted to support their hypothesis that the Ramapo fault is an active techonic structure, viz.,

(1) proximity of hypocenters of several small earthquakes to the fault, and,

(2) focal mechanism and inferred stress orientation consistent with movement on the fault.

More recent work (Kafka, 1983) indicates that there is no evidence from the microearthquakes recorded by the local seismic network from 1970 to 1982 to suggest that northeast trending faults that lie to the northwest of the Newark basin (such as the Ramapo fault) are any more active than other structures which lie around the Newark basin. Woodward Clyde Consultants (1982) report that earthquakes in the site region as located by instruments do not lie preferentially along either the Ramapo fault or along other northeast trending structures. No spatial correlation is observed between the distribution of epicenters and geologic structures or terrains that are mapped at the surface. The two largest events since the local seismic network has been operating in the New York metropolitan area were not associated with the Ramapo fault, but rather, were located in the Coastal Plain east of the Newark basin (Cheesequake, N.J.) and in the Valley and Ridge province north of the Newark basin (Wappingers Falls, N.Y.). Epicenter locations of Magnitude 2 and larger earthquakes appear to be in the region surrounding the Newark basin. Although the Ramapo fault shows a spatial correlation with some of the earthquakes, fault plane solutions for many of these events indicate primarily thrust-type faulting on north to west striking planes which is inconsistent with movement on the Ramapo fault (Woodward-Clyde Consultants, 1982). There is, therefore, a difference of opinion as to whether seismic activity is solely associated with the Ramapo fault.

The largest historical earthquakes within 100 miles of Tuxedo, N.Y., had maximum Modified Mercalli intensities of VII. (Stover et al., 1980, 1981). For facilities being reviewed under the criteria of 10 CFR Part 100, Appendix A, the safe shutdown earthquake (SSE) for this site would probably be based on an event of about maximum Modified Mercalli intensity VII occurring in the site vicinty. A definitive conclusion on this matter in a nuclear power plant licensing application would require additional review and evaluation. An appropriate peak acceleration derived from intensity data might be used to anchor a standard spectral shape. In recent reviews of nuclear power plant sites, it has been the staff's position that magnitude is a more suitable characterization of earthquake size than is intensity. An earthquake's magnitude is used in conjunction with site geology to select time histories for use in obtaining site specific response spectra. For the eastern United States, the staff equates earthquakes of maximum intensity VII with magnitude (m_b) of approximately 5.3.

Since there is no seismic design criteria established for the Hot Laboratory at the UCC site at Tuxedo, N.Y., the staff must use a greater degree of judgment to evaluate the adequacy of the facility with respect to seismic hazard. If criteria were to be established, a more general consideration of the regional geology and seismology should be made rather than simply assigning all earthquake to the Ramapo fault and assuming that the ground motion from the resulting earthquake would not affect the site. A discussion of the effects of an earthquake on structures, systems, and components important to safety is presented in Section 8.1.1 of this report.

3.5 Fire Protection

The licensee has considered the potential for dispersal of radioactivity as a result of fire. Each hot cell is limited to no more than two liters of any flammable liquid. Since air inleakage to the cells is limited, the rate of combustion would be limited to about one pound per minute. Such a fire would last only a few minutes unless other combustibles were ignited. The licensee has limited the total of "loose" combustibles to two hundred pounds, but operator action would be expected to douse the fire before a sizeable fraction of other combustibles could burn.

Since the heated air from a cell fire would mix with cooler air from the other cells, the maximum temperature downstream at the filter bank would be reduced to less than 200°C. The carbon filters would be protected by a carbon dioxide fire extinguishing-cooling system which would be activated if the heated air entering the filter bank exceeded 300°C. Because of the limited air inleakage and consequent low burn rate, all filters are expected to function during any cell fire, and the consequent release would be minimal.

A fire could occur in areas of the Hot Laboratory Building external to the hot cells. Smaller quantities of radioactivity would be associated with such fires, primarily in the radiochemical laboratory. Hood or glove box exhausts are filtered, but if by-passed, the main building ventilation system would filter the contaminated air before release.

Because of the reliance on the ventilation system to maintain the safe confinement of radioactivity, particularly in the hot cells, the staff evaluated the potential for a fire damaging that system. As described in Section 3.3.1, the main exhaust fan for the Hot Laboratory Building is located in a separate enclosure (fan room) at the northwest end of the charging area (see Figure 3.4.1). The final polishing filters are located in a separate enclosure adjacent to the fan room. These enclosures are constructed of reinforced concrete with steel doors. No combustibles are stored within these enclosures. Although unlikely, a fire external to these enclosures could affect fan or filter operation. Loss of the main exhaust fan might be overcome by the auxiliary fan located on the second floor.

The licensee has taken precautions to extinguish any fire occurring in the Hot Laboratory Building. There are 22 portable fire extinguishers located throughout the building. Operators are trained and retrained in fire protection, including the use of fire extinguishers (annually). The local volunteer fire department substation is located on Long Meadow Road within 2,000 feet of the site. The fire department is instructed in the rudiments of radiation protection and the compatibility of hose connections and their locations has been assured. An automatic sprinkler system is installed in the first floor area north of the radiochemical laboratory in which combustible material is stored.

3.5.1 Staff Evaluation - Fire Safety

Based on the staff's review and evaluation of the licensee's precautions to prevent a fire and ameliorate its effects, should one occur as discussed above, we find that the Hot Laboratory is adequately protected. However, a fire involving radioactive materials is a credible event for the facility. Our analysis indicates that the radiological impact of such an event would be no more severe than the design basis accidents described in Section 16 of the Consolidated Application.

4. PROCESS SAFETY CONSIDERATIONS

Approval of a license application, including a license renewal, requires that the equipment and facilities be determined "adequate to protect health and minimize danger to life or property" [10 CFR 70.22]. The process safety considerations evaluated by the staff included the adequacy of the design, operation of the individual unit operations to minimize occupational exposure, the design and administrative controls to prevent a nuclear criticality accident, and control of the various radioactive effluents from the plant.

4.1 Process Operations

The staff reviewed the three processes involving SNM for safety consideration. These processes are currently being conducted in the Hot Laboratory facility. They are (1) target preparation, (2) isotope processing, and (3) uranium waste form processing. The staff evaluation was based on the information provided by UCC and knowledge supplemented by site visits and UCC responses to questions requesting additional information. Target preparation is conducted in hoods. Other processes described in the following sections are housed in hot cells used as secondary confinement enclosures. Primary enclosures are process vessels, themselves.

4.1.1 Target Preparation

Target preparation involves a unique process for preparing a primary target containing fissionable materials (e.g., uranium). This process was developed and patented by the UCC. The target is designed so that a thin, uniform layer of fissile material can be bonded to its inner wall and can then be used as both an irradiation chamber and container for chemical processing. The principal radiological safety consideration in this area is criticality prevention.

4.1.2 Isotopes Processing

Isotope processing involves the separation of fission product isotopes from irradiated, uranium-fueled targets. In the principal process, Mo-99 is separated by precipitation and packaged in accordance with 10 CFR Part 71 for delivery to medical industries. There has been ample demonstration during the more than twenty years of operation that the thick shielding is adequate protection against direct radiation. Moreover, the containment of process radiochemicals in specially designed glassware has proved to be a positive control over the spread of radioactivity.

4.1.3 Uranium Waste Form Processing

Uranium waste form processing involves the conversion of the waste solution, resulting from the isotope process, to a form acceptable for transfer offsite to reprocessing to reclaim the unfissioned uranium. These operations are similar to isotopes processing with limited potential, because of small batch sizes, for energetic reactions. Prior to the incorporation of these steps as a routine operation, UCC conducted several tests to demonstrate the limited impact of this process. These tests are described in the UCC report, "Startup Report for Uranium Waste Form Process," dated October 27, 1980.

4.1.4 Staff Evaluation - Process Operations

Information, as above, provided by the licensee, the history of the safety of UCC's operations contained in NRC inspection reports, and staff site visits were used in assessing the safety of the process operations. In addition to the staff's review of the safety of normal operation, the staff reviewed the capability for response to abnormal conditions. This aspect is discussed in Section 2.6.

4.2 Nuclear Criticality Safety

4.2.1 Organizational and Administrative Requirements

The organizational and administrative requirements for the nuclear criticality safety control of the operations at the Uranium Carbide Corporation (UCC) facility are described in Chapters 2, 4, and 15 of the Consolidated Application together with the requirements for radiological safety controls. This includes minimum qualifications of key personnel, training, operating procedure reviews, audits and inspections, and recordkeeping. Although the staff has found no unsafe situations in the facility, it became apparent from the review of the application that it is necessary for UCC to strengthen its capability in this area to provide for audits by competent personnel of both current operating practices and of all calculations prior to instituting any changes in operations. In this regard, UCC has secured the services of an outside nuclear criticality safety consultant,* who will perform an annual audit of the nuclear criticality safety functions and related process controls of the operations at the facility and will perform an independent review of all nuclear criticality safety calculations prior to instituting any changes in operations.

In addition to the annual audit made by the consultant, a bi-monthly (every two months) inspection of all SNM criticality control areas is performed by an appointee of the Nuclear Safeguards Committee to assure that compliance with criticality limits is maintained in each control area.

4.2.2 Technical Criteria

The UCC nuclear criticality safety criteria provide for reviews by a member of the Nuclear Safeguards Committee, who is knowledgeable in the area of nuclear criticality safety and an independent review by the nuclear criticality safety consultant. These reviewers use approved technical criteria provided in detail in Chapter 4 of the licence application. The important nuclear criticality safety criteria are as follows:

1. The basic policy is the double contingency policy. Process designs incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible. This is the basic policy endorsed by Regulatory

*See the licensee's letter, dated June 8, 1984, James J. McGovern to Leland C. Rouse, concerning this aspect.

Guide 3.4, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."

2. A safe mass limit (350 g contained U-235), when there is no control of container geometry or presence of a container seal against inleakage of water, is no more than 45% of the minimum critical mass independent of container geometry or the degree of water moderation and reflection. Dry (unmoderated) SNM, packaged in DOT or NRC approved containers (for dry shipment), may contain no more than 700 g of contained U-235. The safety factor of the latter limiting mass of 700 g is even greater than the 350 gram limit regardless of the degree of water reflection. Combinations of the two types of masses (dry and "wet") in a given work area are governed by the "unity rule" (sum of reciprocals).

3. Although several of the containers used in the processing operations are safe-by-geometry, the nuclear criticality safety of a batch is not dependent on container geometry. The U-235 mass is the controlling parameter.

4. The total quantity of contained U-235 in the Waste Storage and Waste Form Process Hot Cells are limited by specified mass limits. Each container in the cell, having a maximum inside diameter of 5.0 and 3.0 inches, respectively, for the Waste Storage and Waste Form Process Hot Cells, is separated from the others so that the surface density of the arrays meets the safety criteria specified in NUREG/CR-0095, "Nuclear Safety Guide," TID-7016, Revision 2.

5. Criteria for the Isotope Process Cell are based on safe, individual batch sizes (independent of container geometry or degree of water flooding) in safe geometry containers and filled with borosilicate-glass Raschig rings in accordance with modifications to ANSI/ANS-8.5-1979, "Use of Raschig Rings as a Neutron Absorber in Solutions of Fissile Material." The licensee plans to use very small rings compared to the 1.50-inch diameter rings generally used in large fuel fabrication facilities (for which the standard was primarily developed). The Demonstration section of the UCC consolidated application states, "The mechanical shock-resistance test and the maintenance inspections specified in the standard are not required because of conditions and duration of storage."

   The exemption from these two tests is acceptable because of the following commitment. The licensee shall replace all rings no later than 60 days from the start of use, in lieu of the mechanical shock-resistance test and the maintenance inspections specified in ANSI/ANS-8.5-1979.

4.2.3 Staff Evaluation - Nuclear Criticality Safety

The nuclear criticality safety review and our conclusions that the controls are acceptable are based on:

1. The history of safe plant operations with respect to nuclear criticality safety since the original license was issued.

2. The nuclear criticality safety program has been strengthened by the additional of an outside consultant experienced in outside-of-reactor nuclear criticality safety.

3. The application is clear and adequately covers all aspects of the nuclear criticality safety program in the form of a document that should be equally well understood by the licensee, the NRC, and the public.

The basic policy underlying the conditions sections of the license application is in accordance with Regulatory Guide 3.4, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors."

4.3 Radioactive Waste Effluents

The staff reviewed the controls UCC has provided for reducing radioactive contaminants in airborne and liquid effluents to the environment and in solid waste shipments to offsite burial facilities.

4.3.1 Airborne Radioactive Effluents

The reactor building is maintained at a negative pressure (-0.25 in WG) whenever the reactor is operating so as to direct any air leakage through the containment boundary into the building. The ventilation system is shown in Figure 3.4. The potential gaseous effluent of primary concern is radioactivity leaving the building via the main exhaust duct which leads to an elevated stack shared with the Hot Laboratory building. In the event of a release of radioactivity from the reactor core, the ventilation system will automatically isolate the Reactor Building by tripping the main supply and exhaust fans and dampers and initiating the emergency exhaust fans which exhaust through absolute and charcoal filters. Such an isolation will occur when initiated manually, or when the facility experiences a loss of commercial power and upon high radiation from the bridge monitors sensing a gaseous release from the core. Double doors at each access point to the building provide an airlock. The ventilation system for the Hot Laboratory is shown in Figure 3.4 and described in Section 3.3.1.

The discharge from the Hot Laboratory exhaust fan joins the reactor building exhaust and passes to the common elevated stack. The stack monitor for the two buildings is located in Hot Laboratory Building 2, with local readouts and alarms for the iodine, noble gas, and particulate monitors. Remote recorders and alarms for each of these three parameters are located in the reactor control room. No other gaseous effluents of significance originate in Hot Laboratory and Reactor buildings.

Hot Laboratory and Reactor buildings share a commercial power feed transformer and emergency generators. Upon loss of commercial power, the emergency generators start and pick-up load automatically. The main exhaust fans automatically transfer to half speed on loss of commercial power. If either fan fails to transfer, the backup fan (emergency reactor exhaust fan or the auxiliary hot lab exhaust fan) starts automatically.

4.3.2 Liquid Radioactive Waste

Figure 4.1 gives an overview of the liquid waste disposal system at the Tuxedo site. It has three sub-systems: radioactive waste, non-radioactive process waste, and sanitary waste. The liquid radioactive waste system is discussed in this section.

All radioactive liquid waste from the Tuxedo site is directed to a collector tank in Building 2, which feeds an evaporator for separating the liquid waste into radioactive sludge and decontaminated water. This system is shown in Figure 4.2. The potential sources for liquid radioactive waste are in buildings 1, 2 and 4. Upon purifying the contaminated water, the hold tank is sampled to ensure that the water has been adequately decontaminated. If the water is pure, it is either returned to the canal, which is part of the primary water system for the reactor, or discharged to one of the 5,000 gallon mall tanks. If the water has not been decontaminated, it is returned to the collector tank and reprocessed. When a mall tank is full, it is sampled and released as a batch. Nonradioactive process waste from Buildings 1 and 2 are also fed into the mall tanks for sampling and batch releasing.

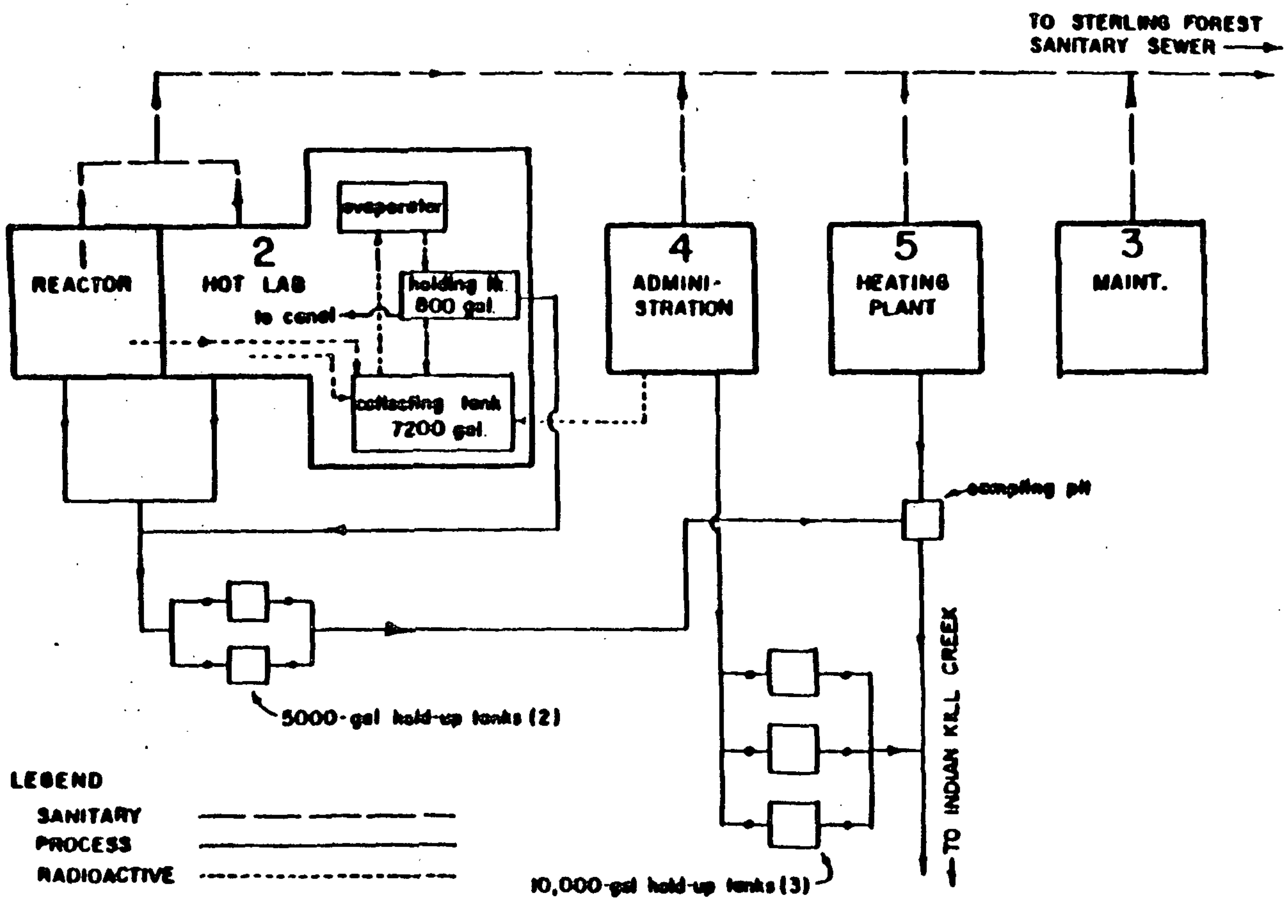
The system for process liquid waste is made up of drains directly from Buildings 4 and 5 and indirectly by way of the mall tanks from Buildings 1 and 2. Liquid process wastes from Building 5 and the mall tanks flow through a sample pit where periodic grab samples are taken to analyze the continuous waste stream. Process waste from Building 4 can be batch-released from hold tanks after sampling.

Small volumes of liquid wastes identified as hazardous substances are collected and disposed of by proper techniques. These are not handled routinely and are therefore treated on a case-by-case basis by qualified technicians on a small scale in a laboratory environment.

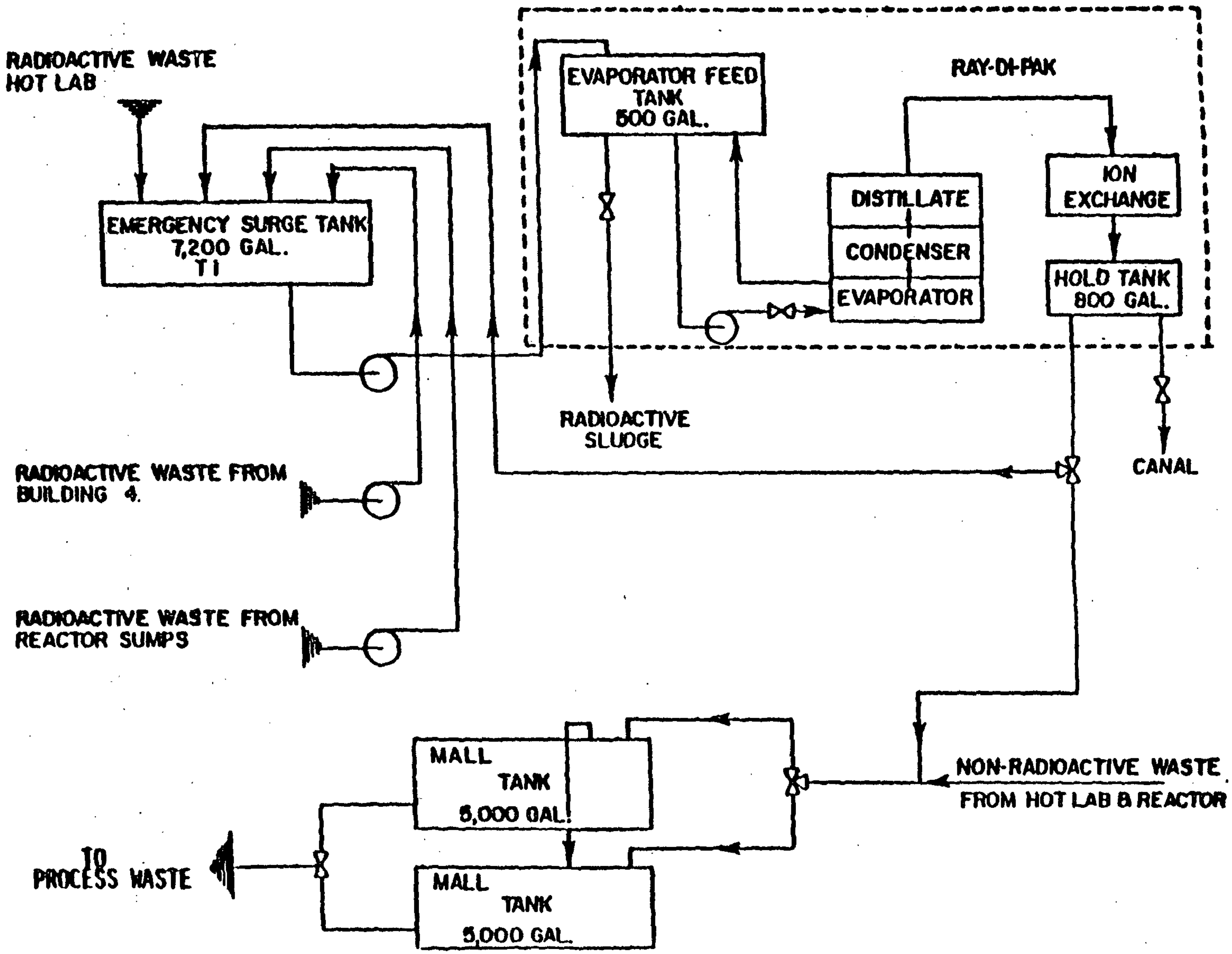
Storm water flows from roofs and paved areas by way of the natural terrain of the land, ultimately flowing past a sampling point enroute to the Indian Kill reservoir as is shown in Figure 4.3.

4.3.3 Solid Radioactive Waste

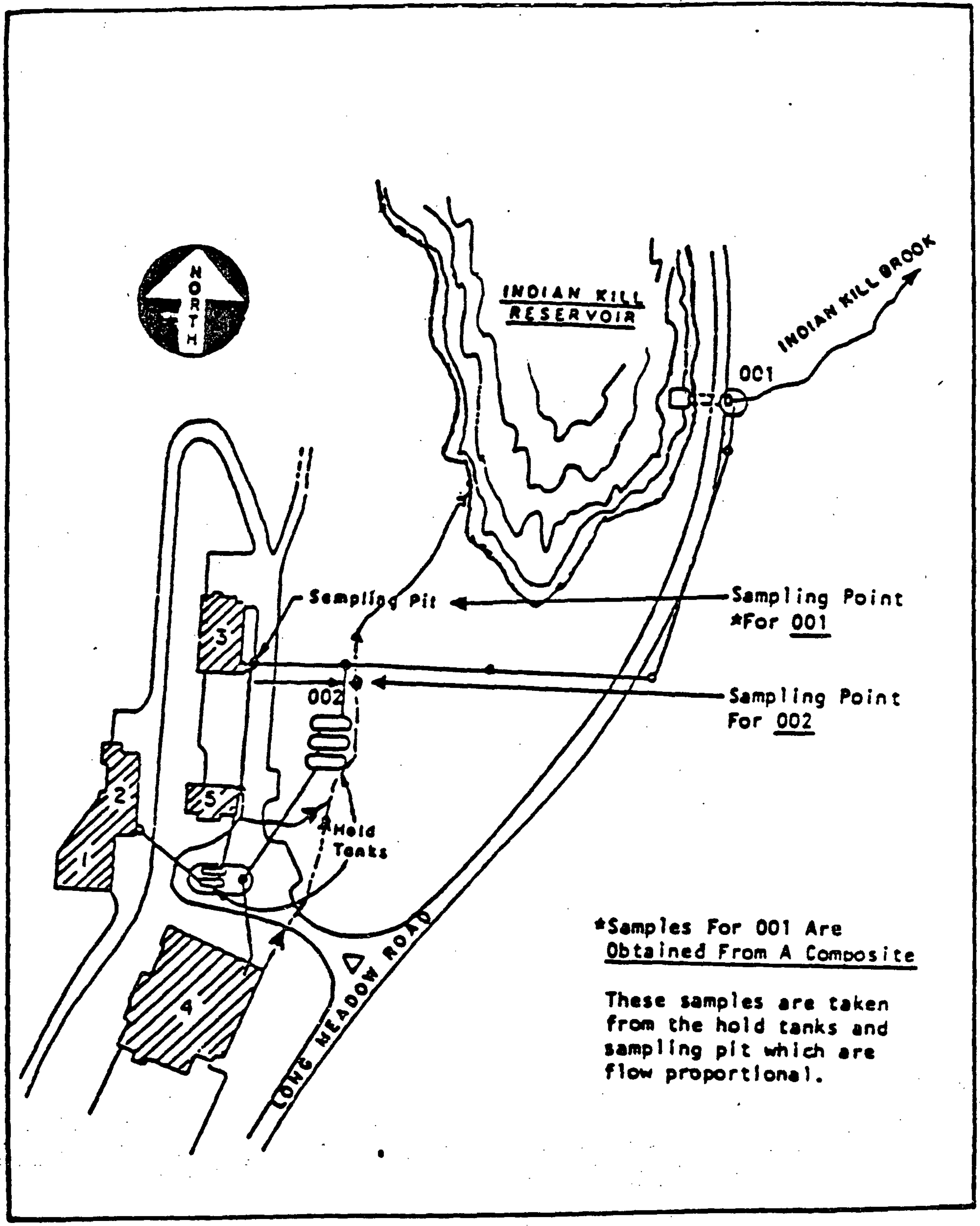
All solid radioactive waste originating at the Tuxedo site is classified as low-level waste with the exception of spent reactor fuel. Approximately 24 reactor fuel elements are shipped to U.S. Department of Energy reprocessing facilities each year. The low-level radioactive waste is further divided into low specific activity and high specific activity material. The low specific activity is made up of resins from the reactor water cleanup system which have been solidified, liquid wastes from the target preparation process which have been solidified, and compacted laboratory trash. While the volume of low specific activity waste is sizeable, the amount of radioactivity is very small. The high specific activity wastes, in contrast, are low in volume but high in radioactivity. These wastes are packaged in 55-gallon drums inside of the hot cells and require substantial shielding when removed from the cells. After removal from the hot cells where storage space is limited these waste drums are put into the previously described, specially designed storage pits where they are allowed to decay for an additional 6 to 8 months before disposal. All low-level radioactive waste is shipped to licensed commercial burial facilities.



Site Liquid Waste Disposal System
Figure 4.1



Radioactive Liquid Waste Disposal System
Figure 4.2



Storm Water Drainage System

Figure 4.3

In recent years the following steps were taken by the licensee to reduce the volume of solid radioactive wastes:

- A commercial compactor was installed for low specific activity waste drums.

- Packaging procedures were reviewed to minimize the unnecessary volume of material placed in waste drums.

- Short half-life wastes were segregated so that after decaying to insignificant levels they could be disposed of by conventional methods.

- Special equipment was procured and built and procedures were developed such that the high specific activity waste packaged in hot cells could be better packed and compacted.

By reducing the volume of wastes, onsite storage time is increased, thus allowing additional decay onsite and consequently less activity at the time of offsite shipment and disposal.

Although receipt of these solid wastes at offsite disposal areas may not be within the jurisdiction of the U.S. Nuclear Regulatory Commission's rules in 10 CFR Part 61, the licensee has committed to packaging and classifying the wastes in accordance with Subsections 61.55 and 61.56 of 10 CFR Part 61. It is likely that any Agreement State accepting the wastes will have requirements similar to those promulgated in 10 CFR Part 61. We note here also that the additional requirements of 10 CFR 20.311 for assuring that the wastes are properly identified have been implemented in the shipping procedures of the licensee.

4.3.4 Staff Conclusions on Radioactive Waste Effluents

Based on the discussions above considering the treatment of airborne and liquid effluents and solid wastes, the licensee has demonstrated an adherence to the principle of ALARA in reducing contaminant levels in radioactive wastes and effluents. The staff has concluded, therefore, that not only radiologically safe levels have been maintained, but the licensee has continued to improve treatment techniques to reduce contaminants to even lower levels.

5. RADIATION SAFETY

The review covered the description of the UCC management policy, design of facilities, organizational structure regarding radiation protection, and the use of operating experience in reducing occupational exposure. Also examined were the method and techniques used for developing plans and procedures for assuring that occupational radiation exposures will be as low as reasonably achievable (ALARA). The review included an analysis of UCC's policy, plans, and organization as compared with Regulatory Guide 8.10, "Operating Philosphy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable," September 1975. Also, where appropriate, it was compared with applicable portions of Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable," June 1978.

5.1 Radiation Safety Administration

5.1.1 Organization and Authority

Radiation protection supervision for this facility consists of the Manager, Health, Safety and Environmental Affairs, assisted by the Health Physics Supervisor and members of the Health, Safety and Environmental Affairs group. The staff has reviewed the qualifications and experience of the individuals filling these positions as compared with UCC's stated requirements, those found in industry for similar responsibilities, and the guidance in Regulatory Guide 8.10. The individuals occupying these positions possess satisfactory education, training, and experience to provide and administer an acceptable health and safety program.

The Manager, Health, Safety and Environmental Affairs, is responsible for administering the overall safety program. This includes providing technical bases, criteria, and methods; providing documents that incorporate procedures and detailed instructions for administering them; and providing technical input for license applications. This individual is also charged with providing justification and approval for exceptions to personnel exposure limits, establishing frequency for health physics routines, providing safety analyses of proposed operational changes and/or modifications, and implementing ALARA policies.

The Health Physics Supervisor is responsible for establishing and maintaining the plant radiological safety program, preparing and reviewing Radiation Contamination Work Permits (RCWPs), assisting in health physics and radiation safety training, and providing health physics supervision at all times including emergencies.

The daily activities of the health physics program are carried out by Health Physics Technicians. Direct radiation protection assistance is provided on day shifts; on other shifts the HP technicians are on call for support. All nonroutine events are documented by the Health Physics Technicians in Unusual Occurrence Reports. Situations involving potential overexposures and/or body intakes are documented by the Health Physics supervision in Incident Reports. These documents are used to identify recurring or potential problem situations.

5.1.2 Radiation/Contamination Work Permit

For any operation or maintenance work involving work on or entry into a system containing special nuclear material not already covered by an effective operating procedure, a Radiation/Contamination Work Permit (RCWP) normally is prepared if there is a potential for release of contamination. The RCWP documents the information used to delineate all potential trouble points and to identify proper safety procedures. The RCWP is approved by a Health Physics Supervisor. All work performed under an RCWP requires health and safety coverage. This health and safety coverage shows the ability of management to control contamination and limit personnel exposures.

5.1.3 ALARA Commitment

UCC management subscribes to the philosophy of maintaining occupational radiation exposures as low as reasonably achievable (ALARA). Shielding has been installed at appropriate work stations to minimize exposures to external radiation. To prevent internal exposures, a total confinement approach has been used for all production-scale operations. Detailed written procedures prepared for the operations are reviewed for methods of reducing potential exposures to the operating staff.

The General Safety Committe (GSC) consisting of representation from Health Physics and site management, reviews problem areas and/or operations for ways to reduce occupational radiation exposures. Planning and process procedures are under the surveillance of the Nuclear Safeguards Committee (NSC). ALARA considerations are formulated by NSC and GSC. ALARA recommendations are directed to the General Safety Committee concerning operational radiation control. ALARA for new projects is handled by NSC. A file is maintained at the facility that documents ALARA recommendations.

5.2 Radiation Safety Controls

5.2.1 External Radiation Exposure Control

Whole body dose from external radiation is minimized by limiting the exposure rate at each work station and by administratively limiting the quarterly accumulative dose of each individual. Pocket chamber dosimeters and film badges are used to monitor personnel and criticality whole-body and extremity exposures.

The Health Physics staff has measured the radiation levels at all work stations and other routinely occupied areas under normal working conditions and anticipated glove box inventories. Trend and unusual conditions are identified by changes in the measured radiation levels. If adverse trends occur, affected persons are requested to give a written account as to the cause of the exposure. In this way, exposure can be maintained ALARA by lowering any exposure trends as they develop. In general, dose rates of less than 2 mR/hr in occupied work areas is used as a criterion.

5.2.2 Internal Radiation Exposure Control

UCC uses a confinement approach, when possible, to minimize the possibility of bodily intake. The confinement system consists of primary containers, glove boxes, hot cells, and the ventilation system. This provides barriers between the workers and potential hazardous materials. To monitor the effectiveness of the confinement techniques, alpha and beta-gamma continous air monitors (CAMS) are located in areas where there is a possibility for airborne radioactive material. The CAMS are fitted with audible alarms.

Continuous air sampling, contamination surveys and a bioassay program are also used as part of the radiation surveillance program. UCC is extensively covered by fixed air sample locations. The samples are taken continuously and analyzed weekly. Routine contamination surveys and a bioassay program provide back-up measurements for internal exposure monitoring. These are discussed in subsequent sections.

5.2.3 Contamination Control

Contamination surveys are an integral part of the ALARA program for potentially contaminated areas such as the solution make-up lab, the plating lab, the target preparation area, and other working locations in the Hot Laboratory building. Approximately 55 smear samples are collected and evaluated daily. Survey results are then transmitted to the supervisors of the areas surveyed so that clean-up action may be taken if required. The action levels for surface contamination through daily sampling are prescribed in the licensee's Consolidated Application. Radiation level surveys are conducted at least monthly in work areas, and more frequently in areas where significant quantities of radioactive materials are used.

Area monitors and criticality monitors in radiation areas of the facility are displayed at a central location and help ensure that exposure to occupational radiation is kept ALARA. Special surveys performed for the purpose of pre-planning operational work in radiation areas include an evaluation of radiation dose rate, contamination conditions, and airborne radioactivity.

5.2.4 Biossay Program

The UCC bioassay program serves as a backup for the radioactive material contamination surveys as well as serving to establish the quantities of internal contaminants from known exposures. Bioassay at UCC includes both urine and thyroid. Urine is sampled on an annual basis for all individuals working with open sources of radioactive material and immediately when an overexposure is suspected. Thyroid uptake sampling frequency is at least quarterly for all employees processing and dispersing iodine. Acceptable action points have been established for resample, whole body counting, and technical evaluation of potential exposures.

Internal exposure is also evaluated at UCC through measurement of activity at the fixed air sampling sites in the reactor and Hot Laboratory buildings. In addition, internal exposure of employees is controlled through hand monitoring to prevent radioactive material from entering the body through ingestion. Beta-gamma hand and foot monitors are provided at all the normal exits from radiation control areas.

5.3 Staff Evaluation - Radiation Protection

Based on its review and evaluation of the radiation safety information in the licensee's Consolidated Application, other supportive information and inspection history, the staff has concluded that UCC has the necessary technical staff, administrative and technical procedures, and equipment to provide effective and safe radiation programs. Conformance by UCC to their conditions should ensure a safe operation and that unfavorable trends or effects can be detected quickly by UCC or by NRC's Regional personnel and corrective action initiated.

6. EFFLUENT AND ENVIRONMENTAL MONITORING

The staff review covered the organization of the program, the description of the equipment available in the health physics and environmental laboratories, and the procedures for monitoring effluents and releases. It also included a study of the concentrations of airborne radioactive materials in the effluent stack and at the various sampling point locations; the concentrations of radioactive/nonradioactive materials in the liquid hold-up tank prior to either reprocessing or release directly to the environment (Indian Kill Reservoir, Indian Kill Brook and indirectly by way of Indian Kill to Warwick Brook and the Ramapo River); and the concentrations of low-level solid radioactive wastes before reprocessing or disposal.

6.1 Organization and Authority

The Manager of Health, Safety and Environment Affairs, is responsible for ensuring that an effective effluent control and monitoring program and a representative environmental surveillance program is established and maintained. The effluent monitoring portion of this program is performed by the Health Physics staff. This involves the monitoring of stack sample filter results, the collection of representative batch samples from the liquid hold-up tanks, the periodic grab samples or evaluation of the concentrations of radioactive materials that these stream or volumes contain, and the shipment of solid radioactive waste to either U.S. Department of Energy reprocessing facilities or licensed commercial burial facilities.

The licensee and the New York State Department of Environmental Conservation (NYS-DEC) have cooperated in an environmental monitoring program. The programs are directed toward measuring airborne activities and measuring direct radiation from the byproduct material used within the Tuxedo facility. The NYS-DEC has conducted an environmental monitoring program for the facility for many years. In addition, several other agencies have performed special monitoring and their results have been compared to UCC monitoring results. The staff also analyzed the radiological impacts of UCC effluent releases. This is discussed in detail in the Section 5.0 of the NRC's Environmental Impact Appraisal, dated May 1984.

6.2 Administrative Program (Methods and Procedures)

6.2.1 Effluent Monitoring

The effluent stack is equipped with a continuous air sampling system. The sample line inlet is located in the second floor of the Hot Laboratory building after the final carbon filter; thus the air being sampled is representative of that being discharged to the environment. The collecting filter is used once through and moved continually at 1 inch per hour. The amount of material on the moving sample filter is counted continuously by the particulate detector using an anthracence beta scintillation crystal. The particulate detector is one of the three monitoring systems employed by the stack continuous air sampling system. The other two are the gas and iodine detection systems. All three use different scintillation crystals to measure different radioactive species. Action points have been established on the basis of sample activity that may indicate the possibility of some operational event that must be immediately investigated to determine the cause of elevated activity.

All potentially contaminated liquid wastes from the UCC operations are collected in hold tanks. Prior to release from the hold tanks, the contaminated water is sampled, and the activity level analyzed. Liquid waste will not be released from the site unless its activity concentration, including dilution with non-radioactive waste water, is below that specified in 10 CFR Part 20. This activity concentration will be determined at least once per month by an analysis of a composite sample of all tanks released during that period.

6.2.2 Environmental Monitoring

Routine environmental monitoring includes a total of five fixed sampling stations and twelve water sampling stations, which are located in different directions and various distances from the facility. Sampling point locations were chosen to provide measurements of the maximum environmental effects from operation and consideration was given to residential areas and the prevailing wind direction.

6.3 Equipment, Instrumentation and Counting Facilities

Within UCC, the Health Physics facilities are equipped with instrumentation for alpha, beta, and gamma analysis of liquid and airborne radioactive materials. The health physics instrumentation is located in a laboratory suitably segregated from the operating areas of the facility, but close enough to these areas to permit rapid identification and quantification of radioactive samples. Showers and standard decontamination agents are available for use in personnel decontamination. Other contamination control equipment available includes shoe covers, lab coats, coveralls, and rubber gloves. Alpha and beta-gamma personnel friskers are used to detect contamination on personnel. A supplied air respiratory protection system is available for use in potential airborne radioactivity areas. The facility is designed and maintained so that respiratory protection is normally not required. However, in the event of an emergency, self-contained breathing apparatus (Scott Packs) are available. This emergency protection equipment is located at each entrance to the facility. Other protection gear for contamination control is located at the entrances to the areas where it is used.

Radiation detectors and monitors presently available include the following:

Type	Number Available	Calibration Method*	Range	
Ion Chamber	5	Cesium-137	1 mR/h	1000 r/h
Ion Chamber	11	Cesium-137	1 mR/h	50 r/h
Portable Geiger Counter	7	Cesium-137	0.1 mR/h	50 mR/h
Alpha Scintillation Detector	6	Alpha Source	1 dpm	$6x10^6$ dpm
Gas Flow Proportional Counter	1	Alpha Source	1 dpm	$1x10^6$ dpm
Criticality Monitors	13	Cesium-137	0.1 mR/h	10,000 mR/h
Gamma Multichannel Analyzer	2	Point Source Isotope	Up to 10^5 cpm	

Calibration frequency for these instruments is at least quarterly.

*All calibration sources are traceable to National Bureau of Standards.

Personnel are monitored for gamma, beta, and fast neutrons using film dosimetry. In addition, personnel wear gamma pocket chambers that are evaluated daily by Health Physics. A criticality accident, should one occur, would also be evaluated with this dosimetry. Reports of exposure are listed for each individual in an exposure history record, which clearly shows his total lifetime and permitted occupational exposure. Bioassay is performed at least once per year on radiation workers. Special bioassay samples are obtained whenever it appears that an individual may have been exposed to excessive airborne radioactivity.

Airborne radioactivity concentrations are evaluated and controlled by approximately 25 fixed air monitors. The air monitors' collection filters are analyzed each day for radioactivity. Appropriate warnings are placed at boundaries to areas in which an exposure to more than 25 percent of MPC might occur. Special breathing zone air samplers are used by personnel in areas in which breathing zone air activity might be higher than that recorded by the fixed air sample near that location.

All portable and laboratory technical equipment and instrumentation are maintained by an electronics department having responsibility for repair. Calibration of equipment is performed on a periodic basis by the Health Physics Department. Under abnormal conditions, an extensive criticality monitoring system senses the higher levels of radiation produced and activates the emergency evacuation system.

6.4 Staff Evaluation - Effluent and Environmental Monitoring

Upon review of the UCC effluent and environmental program, its implementation and past performance, and our analysis of radiological impact of UCC environmental monitoring results, the staff concludes that UCC has an adequate program for normal operation. UCC also has suitable corrective procedures established should abnormalities be identified. The basis for acceptance is conformance to current industrial practice and the recommendations of applicable regulatory guides.

7. MATERIALS AND PLANT PROTECTION

7.1 Material Control and Accounting and Physical Protection

The current regulations in Part 70 provide for material accounting and control requirements with respect to facility organization, material control arrangements, accountability measurements, statistical controls, inventory methods, shipping and receiving procedures, material storage practices, records and reports, and management control.

The current regulations in 10 CFR Part 73 provide requirements for the physical security and protection of fixed sites and transportation involving strategic quantities of nuclear materials. Physical security requirements for protecting fixed sites include the establishment and training of security organization (including armed guards), provision for physical barriers, and establishing of response plans.

The licensee has an approved material control and accounting plan and an approved physical security plan which meet the current requirements of 10 CFR Parts 70 and 73 (Materials and Plant Protection Amendment MPP-3, issued May 26, 1983).

8. ACCIDENT ANALYSIS

As part of its safety review for the continued usage of special nuclear material at the Tuxedo facility, the staff reviewed those accidents the licensee evaluated in its Consolidated Application and Emergency Plan. We performed independent evaluations which considered the postulated accident scenarios and natural phenomena that could release radioactive materials to the environment. These are: (1) earthquake effects; (2) tornado and hurricane; and (3) breach of a target tube.

Section 3.4 describes earthquake potential for the Tuxedo area. The frequency of earthquake which could have any noticeable effect on the facility is low. Based on the information discussed in Section 3.4, an earthquake of magnitude 5.3 is approximately the upper bound to any credible occurrence over the life of the facility. The staff has evaluated reasonable effects attributable to an earthquake of this size. Of principal interest was the confinement of the radioactive materials normally being processed in the hot cells. For process reasons, not particularly related to earthquake, process materials containing radioactivity are confined within vessels designed to provide substantial resistance against impact. The sturdiness of the process equipment prevents the escape of materials during an earthquake.

The results of our analysis and the radiological consequences for each of the accidents are presented below.

8.1 Accident Discussion

8.1.1 Consideration of Earthquake Effects

In Section 3.4 the staff discusses seismicity in the region of the site. It was concluded that an earthquake of magnitude (mb) 5.3 would probably be an appropriate maximum for design purposes for a power reactor at that location. A rough estimate of the acceleration corresponding to a magnitude 5.3 earthquake is about 3.9 ft/sec^2 or 12% of gravity (see Figure 8.1). This value was used for calculational purposes as a reference point in Figure 1 of Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants." As such, the value would normally be used by design engineers to obtain the correct forces to be withstood by all safety-related structures, systems, and components for protection against earthquakes. In this instance, the plant has been designed, constructed, and in operation for over twenty years. Our analysis, therefore, considered the impact an earthquake of that size might have on the existing facility. In this report we do not consider the pool reactor; it is considered in a separate licensing action.

These effects are considered in a Science Applications, Inc. report, "Evaluation of Seismic Response Characteristics of Hot Cells and Related Structures and Equipment at the UCC Sterling Forest Research Center," Report No. SAI-1-148-08-781. The principal conclusion of that report is "that no potential failure has been identified that would be caused by an earthquake ground acceleration level of less than 0.2 g" (20% of gravity). It should be understood that the analysis of the report considered only effects on significant structures, systems, and components, whose failure could result in the release of radioactive material. As stated in the report, these effects were considered up to an acceleration of

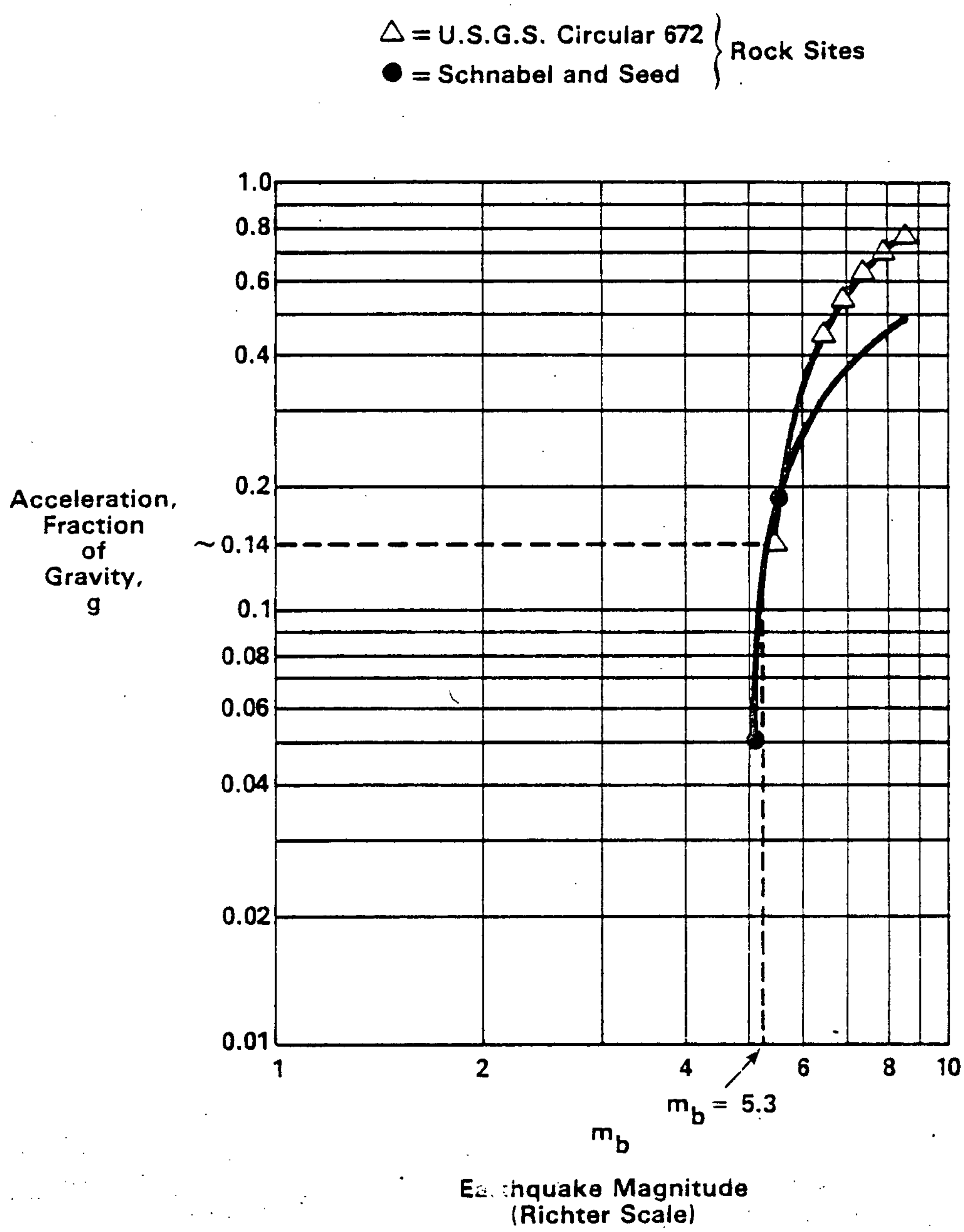


Figure 8.1 Estimation of Design Acceleration

0.2 g, sufficiently greater than the more likely maximum of 0.12 g. This was done to provide added assurance that uncertainty in the relationship of acceleration to earthquake magnitude and magnitude itself were encompassed. It also may be inferred from the report that much of the structure essential to confinement, such as the five hot cells, are likely to afford protection well beyond 0.2 g.

In its analysis of earthquake effects the staff also considered the form and confinement of radioactive materials as a potential source for releases greater than for normal operation. External to the hot cells, radioactive material is confined either in target tubes under water or as waste material in confinement containers, as discussed elsewhere in this report. These materials are considered to some extent in the SAI report described above; we also note that other circumstances, such as dropping containers, would result in stresses for release greater than from an earthquake.

The radioactive material in the hot cells can be classified as "in process" or as cell contamination. The licensee has considered the release of radioactive material from process operations for circumstances similar to earthquake stress. Special vessels of glass encased in terephthalate with septum seals assure that only minor quantities of radioactive materials would be released as a result of shaking or vessel dropping. The amount of radioactive material resuspended into the cell air from all vibration would be small. The filtration system described in Section 3.3.1 will adequately confine the material.

8.1.2 Tornado and Hurricane

In our considerations of possible effects of high winds external to the facility we have relied upon the extensive analyses of the Indian Point site. Figure 8.2 is taken from that analysis. Hurricanes, in general, offer less threat since their large size permits plant ventilation system equilibration, i.e., although the absolute pressure is reduced, pressure differences within the plant remain constant and proper flow direction is maintained. Conversely, tornado pressure changes, related to their high rotational winds, are rapid, often occurring within a few seconds. In our analysis we, therefore, principally studied the effects of tornado, although their occurrence in the region is rare.

From Figure 8.2 we observe that a tornado strike of any size has a frequency in the range of one in ten thousand years. At winds speeds of 250 miles per hour the frequency is lowered to one in a million years. The staff considers that 200 miles per hour winds are sufficient for purpose of analysis. At this high wind speed a tornado may develop a pressure change of about one pound per square inch occurring in six seconds.

Unlike the reactor containment, the interior of the hot laboratory building (Building No. 2) would sense the pressure change virtually at once.

Analysis of the off-site effect indicates that doses would be insignificant. A dose of 0.003 millirem was estimated to a person near the site who might be outside in such severe weather. The principal reason for incurring such low doses is the extreme turbulence of the tornado which very rapidly disperses any radioactivity withdrawn from the cells throughout a very large volume of air.

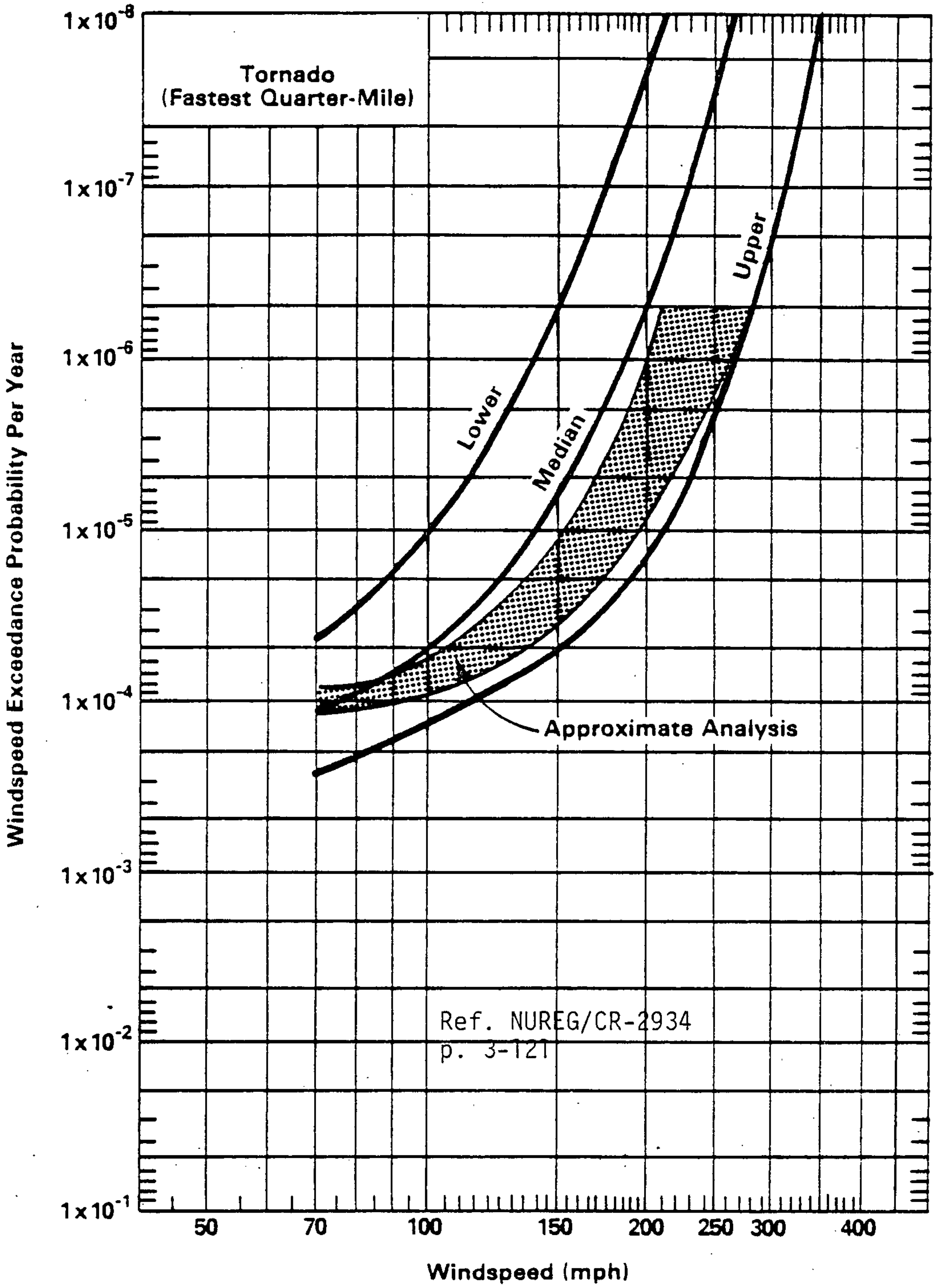


Figure 8.2 Tornado Windspeed Exceedance Probabilities at 33 ft. Elevation

8.1.3 Breach of Target Tube

The license has considered the effects of breaching a target tube six hours following its removal from the reactor. We have analyzed the radiological impact off-site of a possible target breach upon immediate removal from reactor. This is a more severe condition than analyzed by the licensee and is considered to be highly unlikely. We also very conservatively have considered the effect of by-passing any filtering capability.

Even under these extreme conditions, the off-site doses to a nearby individual would be on the order of 20% of the accident guidance[1] used by the staff to judge the acceptability of such events. We note that it is difficult to release any radioactivity in the Hot Laboratory building which could by-pass the ventilation filtering system. This is discussed in detail in Section 3.3.1.

8.1.4 Interactive Effects with Reactor

As discussed in Section 1.0 we have considered credible impacts of reactor accidents upon hot cell operation. The nature of these accidents is considered in a separate staff report.[2] Although the reactor and hot cell buildings are contiguous, they are separated for confinement control. The release from a melt-down accident, which is considered to be a maximum credible event, is confined within the reactor building except for a portion released from the common stack. There is substantial shielding between the reactor operating area and the hot cells operating area. Although it might be decided to curtail hot cell operations, even this most severe accident would not result in any further release from a hot cell accident caused by a reactor accident. It is to be noted that the obverse also holds that the target tube breach accident described above will not result in unsafe operation of the reactor.

[1]In evaluating upper limit accidents the staff uses fifty times the values in Column 1, Table II of Appendix B to 10 CFR Part 20 as concentration limits consequent to dose. The same "sum of the ratios" formula is used as indicated in the footnote to Appendix B.

[2]Technical Evaluation report to be published by the NRC in October 1984.