WESTINGHOUSE ELECTRIC CORPORATION
NUCLEAR ENERGY SYSTEMS

APPLICATION FOR RENEWAL
OF A
SPECIAL NUCLEAR MATERIALS LICENSE
FOR THE
NUCLEAR FUEL DIVISION
AT THE
COLUMBIA, S.C. SITE

LICENSE NUMBER
SNM-1107

APRIL 30, 1983

U.S. NUCLEAR REGULATORY COMMISSION
DOCKET 70-1151

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REVISION RECORD

Revision Number	Date of Revision	Pages Revised	Revision Reason

SECTION 1

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1.1 Introduction and Corporate Information

For manufacturing operations involving the use of a wide variety of radioactive materials in complex facilities, it becomes impractical to delineate in a license either the detailed operating procedures or the detailed facility and equipment designs which may be employed over an extended period of time to insure nuclear criticality safety and adequate radiation protection. Frequent changes are necessary to insure continuing nuclear criticality safety and radiation protection while permitting the performance of the many activities that are within the capability of the facilities. Detailed descriptions in an application for such a Materials License seem useful primarily in evaluating whether effective protective measures are being provided.

This license application, based on a comprehensive outline, has been prepared so that the first part (Section 1) provides sufficient descriptions of procedures, equipment and facilities, as they are currently planned, to permit proper evaluation. The first part of the application will become the "safety demonstration" portion of the license, which may be changed by the licensee in accordance with established internal procedures defined in the "license conditions" (Sections 2 and 3). It is expected that such changes will occur during the course of normal operations in the facility.

The "license conditions" (Sections 2 and 3) presents the principal criteria which will limit the changes in facilities or procedures, the type of activities that will be performed and the quantities of materials that will be used under the license. Sections 2 and 3 will not be subject to change without prior authorization from the NRC.

Certain alternative provisions of the pertinent regulations and certain exemptions from requirements of the regulations have been specified in a third part (Section 4). These provisions will not be changed without prior authorization.

1.1 (Continued)

This application is for the renewal of License SNM-1107, Docket 70-1151 which authorizes the receipt, possession, storage, use, and transfer of special nuclear material at the Columbia Fuel Fabrication Facility. This facility is operated by the Nuclear Fuel Division of the Westinghouse Electric Corporation. In accordance with the requirements of 10 CFR 70.22(a), the following information is submitted:

(a) Applicant:

Westinghouse Electric Corporation
Westinghouse Building
Gateway Center
Pittsburgh, Pennsylvania 15222

(b) Mailing address for all correspondence pertaining to this license:

Mr. A. Joseph Nardi, Manager
NES License Administration
P.O. Box 355,
Pittsburgh, Pennsylvania 15230

(c) Mailing address for location where these materials will be used:

Westinghouse Electric Corporation
Nuclear Fuel Division
P.O. Drawer R,
Columbia, South Carolina 29250

Additional corporate information as required by 10 CFR 70.22(a) is provided in a separate transmittal. That transmittal contains current financial and technical information on the Westinghouse Electric Corporation and is updated approximately annually. The latest submittal made was dated December 9, 1982, from Mr. A. J. Nardi to Mr. John G. Davis and Mr. Harold R. Denton, titled "Corporate Information for Licenses."

1.2 General Description of Operations

The Manufacturing Department of the Nuclear Fuel Division (NFD) is primarily engaged in the manufacture of fuel assemblies for commercial nuclear reactors. The manufacturing operations to be authorized by this license consist of receiving low enriched (less or equal to than 5.0 w/o U-235) uranium hexafluoride; processing the hexafluoride to produce uranium dioxide powder; and processing the uraniuam dioxide through pellet pressing and sintering, fuel rod loading and sealing, and fuel assembly fabrication. These operations are accompanied by appropriate radiation protection, nuclear criticality safety, SNM safeguards, quality assurance and environmental pollution controls.

Two distinct processes are used to convert uranium hexafluoride to uranium dioxide powder: Integrated Dry Route (IDR) and Ammonium Diuranate (ADU). The IDR conversion lines are designed to receive uranium hexafluoride in enrichments up to a maximum of 5.0 w/o U-235 and process this material through rod loading. The ADU conversion lines are presently designed for 4.15 w/o U-235 for conversion. For enrichments greater than 4.15 w/o U-235 but less than or equal to 5.0 w/o U-235, material for such fabrication is procured as uranium dioxide powder, only, that can then be processed through pellet pressing and sintering, rod loading and sealing, and fuel assembly fabrication and shipping as required.

The ADU lines may be used to process enrichments up to 5.0 w/o U-235 provided that properly approved and documented evaluations demonstrate the nuclear criticality safety of the wet processing systems.

The Columbia Site Analytical Laboratory is part of the Quality Control Department within the Product Assurance Department of NFD. This laboratory performs chemical analyses on products produced in manufacturing operations. It also supplies similar services for other Westinghouse facilities or outside

1.2 (Continued)

customers. The laboratory possesses and uses calibration standards which consist of gram quantity samples of uranium containing U-235 at a variety of enrichments up to and including fully enriched.

The Radiation Protection Component described in paragraph 3.1.2 is the Regulatory Compliance Department. It provides comprehensive radiological and environmental pollution control engineering, nuclear criticality safety administration, radiation protection and SNM safeguards services for all Corporate activities at the Columbia Site.

This group maintains facilities which are typical for such activities and is responsible for the safe handling and use of sealed sources, instrument checking and calibration, operation of the liquid and gaseous effluent monitoring systems and for providing nuclear criticality, radiation protection and SNM safeguards engineering services.

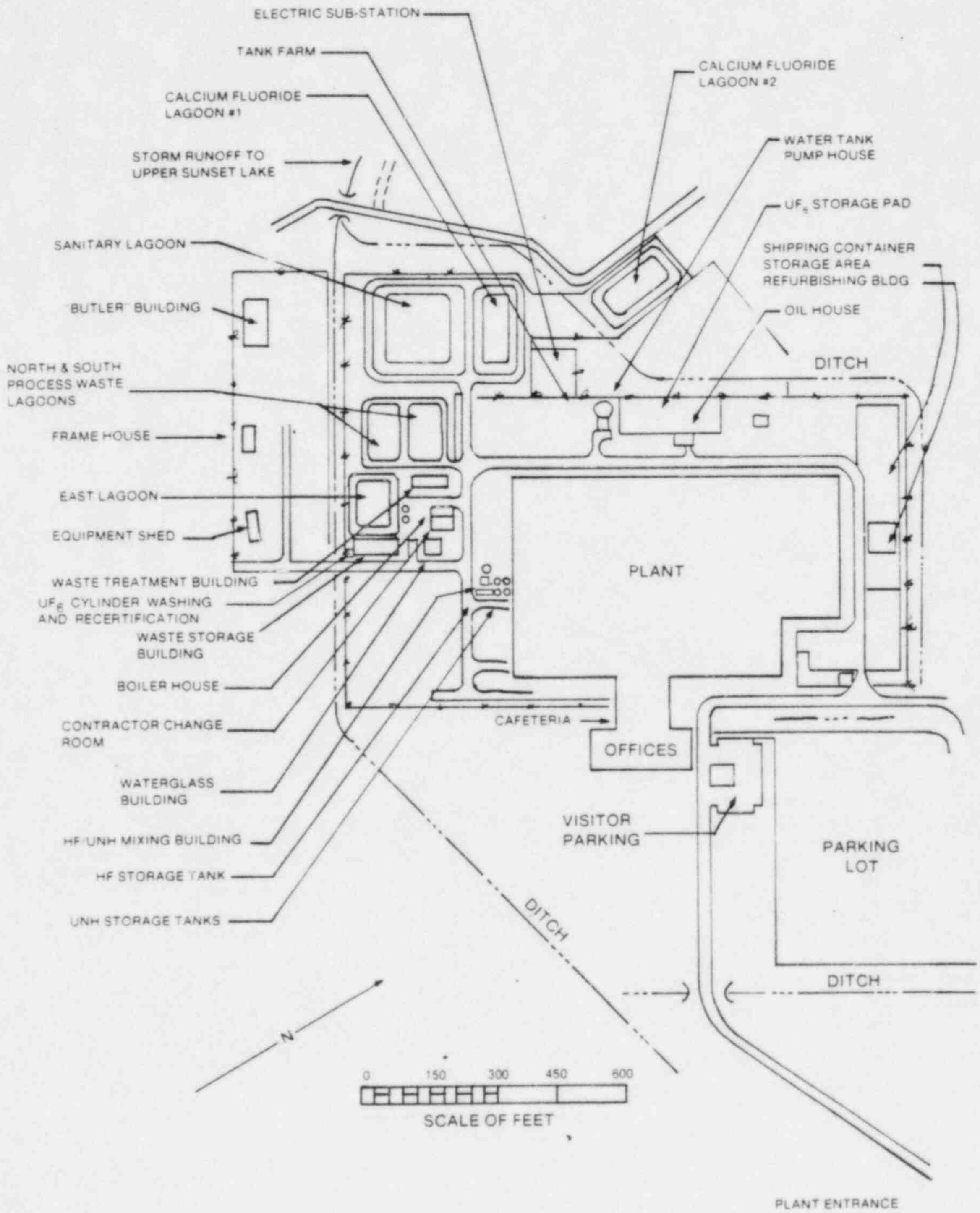
1.3 Site Description

1.3.1 Geographical Location and Topography

The Westinghouse Electric Corporation, Nuclear Fuel Division, Columbia Site, is located on South Carolina Highway No. 48 (Bluff Road) in Richland County approximately 8 miles southeast of Columbia, South Carolina. The plant is situated approximately 1,800 feet from Highway No. 48 on a semirural plot of approximately 1,156 acres. The licensed facility is on a knoll approximately 40 feet above the water level of the nearby Congaree River. Details of the plant location, including proximity to nearby towns, industries, public facilities, the Congaree River, transportation links and topography are shown in the Site Emergency Plan, Figures 1.1 through 1.4.

1.3.2 Location of Buildings within the Site

Figure 1.3.2.1 shows the relative locations of the buildings on the site. All existing buildings are occupied by the



DETAILED COLUMBIA SITE PLAN
Figure 1.3.2.1

1.3.2 (Continued)

Nuclear Fuel Division or contractors.

1.3.3 Description of NFD Buildings

The building in which the Nuclear Fuel Division operations are conducted consists mainly of high-bay manufacturing areas. It is constructed with a structural steel framework and cast concrete curtain walls. Adjoining the east side of the building is a two-story office building. Adjoining the south side of the building is a facility for the fabrication of poison fuel rods. The operations conducted within this building are similar to those described in this license except for the fact that nuclear poisons are incorporated in the fuel. An isolated facility is required to prevent cross-contamination of other fuel and the nuclear poison. All significant quantities of material authorized under this license are received, processed and stored within the Manufacturing Building.

Low-level contaminated wastes are stored in the Waste Storage Building located to the south of the Manufacturing Building as shown in Figure 1.3.2.1. Prior to transfer to this area, contaminated items are visually inspected to ensure that no accumulation of radioactive material is present, and then surveyed and released by the Radiation Protection Component in accordance with the appropriate contamination limits as defined in this license. Contaminated items which exceed these limits, or contain inaccessible surfaces or areas where radioactive material accumulations could exist, are bagged and taped or otherwise containerized as appropriate to prevent the spread of contamination.

The specific buildings and areas in which licensed materials may be handled and/or stored are shown on Figure 1.3.2.1. In addition to these locations, miscellaneous licensed materials may be stored in various areas within the perimeter chain link

1.3.3 (Continued)

fence. These include (1) UF6 cylinders awaiting transfer into the Manufacturing Building, (2) containerized SNM within covered trailers, (3) shipping containers or equivalent awaiting transport or transfer to other areas, (4) miscellaneous drum storage (empty UNH contaminated drums, contaminated waste oil, etc.). Existing license conditions are used to control outside storage of SNM with regard to nuclear criticality safety and contamination control.

Contiguous to the Waste Storage Building are facilities for UF6 cylinder washing and recertification.

The Site Waste Treatment Facilities and associated lagoons are located in an area southwest of the Manufacturing Building. An equipment shed, a small vacant house, and a Butler building used for equipment and materials storage are also located in the same direction, but farther away.

A chemical tank farm, water storage tank and pump house, UF6 cylinder storage pad and oil house are located to the west side of the Manufacturing Building. The UF6 cylinder storage pad is surrounded by a security fence as described in the Security Plan and equipped with criticality evacuation alarms.

1.3.4 Site Security

Physical security for the Columbia Plant is described in the Westinghouse Electric Corporation Site Physical Security Plan and Security Procedures Manual. The Plan describes the measures employed by Westinghouse to mitigate the opportunity for theft and to detect the loss of special nuclear material of low strategic significance in accordance with 10 CFR 73.

1.3.5 Utilities and Services

The Columbia Site is served by a single electrical supply line. Four diesel-powered standby generators are installed to meet the emergency electrical power requirements of the site in the event of a temporary outage of the normal supply source. Emergency power is automatically provided to critical process equipment, emergency lighting systems, cooling system

1.3.5 (Continued)

pumps, all safety alarm systems (fire alarm, hazard alarms, etc.), Conversion Control Room alarms, health physics sampling systems and emergency ventilation systems, including scrubbers. The emergency systems are tested once each month to ensure proper operations.

A ten-inch main from the Columbia Municipal Water Authority supplies water to the site.

Liquid process wastes are treated in the Waste Treatment Facilities and then pumped to the Congaree River via a 4-inch pipeline. Waste treatment consists of filtration, flocculation, lime addition, distillation and precipitation in a series of holding lagoons for the removal of uranium, ammonia and fluorides.

Site sanitary sewage is treated in an extended aeration package plant and discharged into a biological oxidation/settling, polishing lagoon. The lagoon effluent is then chlorinated and mixed with treated liquid process waste at the facility lift station.

The combined waste is passed through a final aerater, followed by pH adjustment as required, and subsequently pumped to the Congaree River.

Storm water from the site enters a system of drainage ditches and ultimately flows to the Congaree River.

1.3.6 Effluent Considerations

Gaseous exhausts from the Controlled Area are routed through HEPA filtration as a minimum to remove entrained uranium particulates. Exhausts containing uranium in soluble form are normally passed through aqueous scrubbers, passed through HEPA filter and then discharged to the environment.

A continuous on-line monitor system is provided for automatically sampling, monitoring and recording the radioactivity concentrations in liquid process wastes prior to

1.3.6 (Continued)

discharge to the Waste Treatment Facility. Discharge to the Waste Treatment Facility is made in accordance with the provisions of 10 CFR 20. The on-line monitor system has sufficient holdup capacity, if additional treatment is required to meet the limits specified in 10 CFR 20.

An advanced wastewater treatment system provides additional uranium removal from the waste stream. Other small waste streams are batch collected in quarantine tanks, sampled, analyzed and discharged to the Waste Treatment Facility in accordance with the provisions of 10 CFR 20. Additional treatment is provided as required to meet the limits specified in 10 CFR 20.

Contaminated liquid wastes from sources such as laboratory drains and controlled area sinks are discharged directly to the Contaminated Waste Disposal System where they are collected, filtered, sampled, analyzed and released to the Waste Treatment Facility lift station in accordance with 10 CFR 20 limits.

The Radiation Protection Component maintains records of the estimated total quantities of SNM that are disposed of in this fashion.

Wastes processed through the Waste Treatment Facility as described in Paragraph 1.3.5 are sampled and analyzed at the point of discharge. A continuous proportional sample is collected. Each day's collection is then sampled for analysis.

1.3.7 Solid Waste Storage and Disposal

Solid wastes are accumulated in specially designated containers which are placed throughout the work area. Wastes consist of paper, wood, plastics, metals, floor sweepings and similar material which is contaminated by, or contain, radioactive materials.

Due to the small quantity of SNM adhering to the base material,

1.3.7 (Continued)

the low enrichment in the isotope U-235 which is typical of most products and administrative controls, it is not credible that any quantity of SNM approaching 350 grams of U-235 will accumulate in any one container.

Contents of filled containers are transferred to the waste processing area currently located within the Contamination Controlled Area. Noncombustible solid wastes are loaded into containers, compacted when appropriate, gamma scanned to verify the SNM content and placed in storage to await shipment for burial or recovery. Filled boxes are normally stored in a designated area inside the plant, but they may be stored outdoors if adequately protected from the environment.

Disposal of most combustible wastes is accomplished by on-site incineration in accordance with the provisions of Paragraph 1.9.6 of our application. Combustible wastes with low uranium content and all noncombustible wastes are shipped to a licensed burial facility. Shipments are made in compliance with all applicable NRC, DOT, and State regulations and license requirements.

1.3.8 Fire Protection

Onsite Emergency Brigade members have been trained in fire fighting techniques. An agreement has been made with the Columbia Fire Department to respond in fire fighting missions if required. They have been brought to the site for indoctrination and special training. Visits for reindoctrination are arranged as deemed necessary.

The ten-inch water main which supplies process and drinking water to the site also supplies a water tank which has a capacity of 200,000 gallons. The pump, which is connected to the emergency power supply, is rated at 1,000 gpm at 100 psi.

1.3.8 (Continued)

Several 6-inch fire hydrants are located on the Columbia Site with 2.5-inch hose connectors. Inch and a half standpipes are strategically located throughout the facilities; although in fuel processing areas, water may be used to combat a fire only with the specific consent of designated individuals.

Automatic fire sprinklers are located in areas of the building where no uranium is processed, handled or stored under "moderation" criticality control criteria.

Fire extinguishers of appropriate types are strategically located throughout the facilities. These include CO2, Met-L-X powder, dry chemical and halon types.

1.4 Organization

1.4.1 Organization Chart

The organization chart shown in Figure 1.4.1.1 indicates the relationships of the functions of interest in this application.

1.4.2 Functions of Organizational Operating Units

The NFD Columbia Plant operates fuel fabrication facilities at Columbia, South Carolina for the Westinghouse Nuclear Energy Systems group (NES), a major Westinghouse operation. NES has designed the nuclear steam supply systems for a number of nuclear power plants, carried on intensive research and experimentation with various types of reactor technologies, and has manufactured many cores and auxiliary components for nuclear power plants.

Within NES, the WRD Licensing, Safeguards and Safety Group encompasses a variety of activities which primarily provide services to other divisions. NES Industrial Safety and the Nuclear Materials Management and Safeguards Representative are both members of this group.

The Nuclear Fuel Division, which also is a division within NES, is responsible for the manufacture of fuel-bearing components for commercial reactors. NFD includes a competent engineering group which is responsible for assigned fuel

1.4.2 (Continued)

design and development projects. However, the Division's primary function, to fabricate and ship commercial reactor fuel assemblies, is the responsibility of the NFD Columbia Plant.

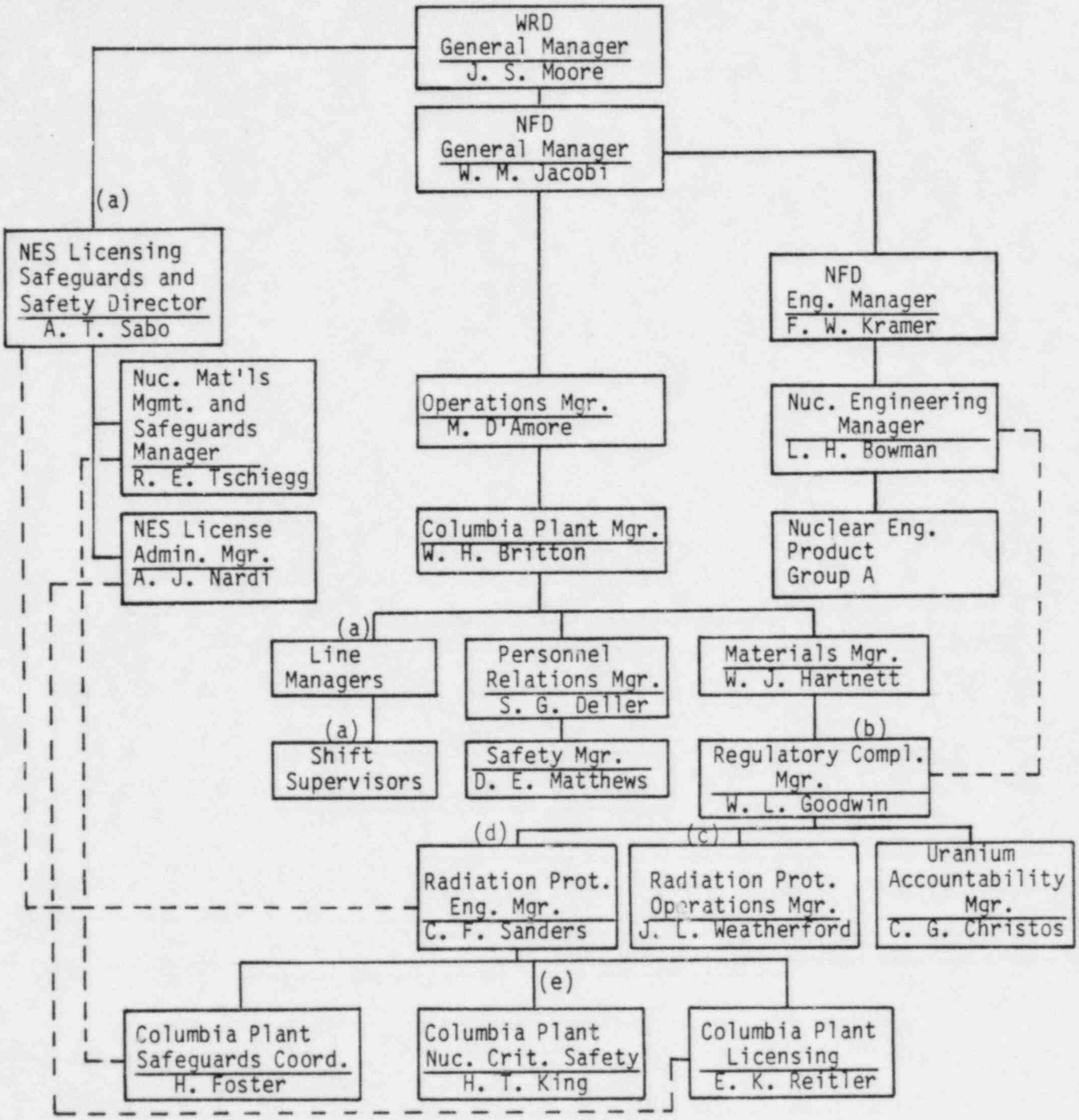
The NFD Columbia Plant operates the Columbia, South Carolina, facility as a significant portion of its activities toward carrying out this assigned responsibility to fabricate SNM from various raw material forms into finished fuel assemblies. In addition, the NFD Columbia Plant acts as the "landlord activity" at the Columbia site, and is responsible to supply all required site supporting functions such as maintenance, industrial relations, public relations, security, health physics, industrial hygiene and safety, nuclear criticality safety administration, SNM safeguards and environmental pollution control.

1.4.3 <u>Functions of Management Positions within Organizational Operating Units</u>

Each Westinghouse management position is covered by a written description, spelling out in detail its scope, purpose, duties, responsibilities, difficulties and requirements. The description identifies the incumbent's authority for decisions which he may make unilaterally and those requiring higher management approval. It delineates relationships with other functions. It specifies his responsibility for personnel and for the control and maintenance of facilities and equipment. These descriptions are reviewed and approved by two higher levels of line management. The NES Management Position Committee, which consists of key members of the NES staff, review and evaluate these positions. These reviews determine that all key functions are covered, inter-relationships are clear and conflicts eliminated. Persons are selected to fill these management positions by evaluating their capability to perform the various activities specified in the position

FIGURE 1.4.1.1

WESTINGHOUSE ORGANIZATION



(a) Certain Intermediate Levels Omitted for Simplification
(b) Radiation Protection Component
(c) Radiation Protection Operations Component
(d) Radiation Protection Engineering Component
(e) Criticality Engineer

1.4.3 (Continued)

description. Two higher levels of management, as a minimum, must approve each selection or change of an incumbent. Adequate continuing performance of these individuals is assured through an organized program of annual review.

The operation of the Columbia Plant of the Nuclear Fuel Division is in accordance with the general operating philosophy and procedure that is employed in Westinghouse plants and facilities. Briefly, this philosophy provides that responsibility for all phases of operations, including safety and health protection, follows the usual lines of organizational authority. Advisory and service groups are provided to assist line management in the analysis of operations within their control, and to provide measurements, determinations and information which aid in the analysis of specific operations and situations. Such a service and staff function in no way relieves the individual line manager from responsibility for the safe operation of his function and facility and for ascertaining and assuring, through appropriate management channels, that adequate service is provided. Basic policies and procedures are established by line management with the approval of cognizant staff groups; and within the framework of these policies, the responsibility for making decisions at the operating level rests with the first level manager (shift supervisor). He has a basic responsibility to operate his facility in a safe and orderly manner.

The shift supervisor is responsible for providing operating instructions for the guidance and direction of nonmanagment personnel. Written manuals or procedures are prepared which become the bases for performing specific operations. The shift supervisor cannot make unilateral changes in such written procedures or in posted limits.

The shift supervisor is also responsible for assuring that the personnel under his jurisdiction receive adequate training.

1.4.3 (Continued)

The Radiation Protection Component presents an orientation to new employes. The topics of limiting exposure to external radiation, prevention of internal exposures, use of protective clothing and personnel monitoring devices, fundamental radiation safety rules and policies, the concept of nuclear criticality safety, and plant emergency procedures are discussed. To acquaint the new employe with basic NRC regulations, several parts of Title 10, Code of Federal Regulations, are covered. Emphasis is placed upon 10 CFR Parts 19 and 20.

The cognizant shift supervisor assigns an experienced employe the responsibility of indoctrinating and training the new employe in the proper procedures and precautions in performing his specific job. The shift supervisor evaluates the progress of the new employe and gradually increases his job assignments until he fulfills the requirements of his job description. Failure to achieve minimum performance requirements is cause for a change in assignment or release.

Periodic instruction is conducted on the job by the employe's supervisor and/or personnel from the Radiation Protection Component. As the need arises, changes in regulations, operating conditions and procedures, and administrative policies are covered. To ascertain that all employes are familiar with the site emergency evacuation procedures, drills are conducted at least annually to simulate emergencies. After each drill is evaluated, appropriate management is informed of any shortcomings, and they instruct personnel under their supervision regarding any clarifications required.

NFD Columbia Plant Manager

Overall responsibility for NFD activities at the Columbia Site is vested with this individual. He directs all the Columbia Manufacturing operations and staff functions either personally or through other management personnel. He effects and

1.4.3 (Continued)

coordinates necessary support activities obtained from higher Westinghouse management and performs management functions in accordance with Westinghouse policies and higher management directives.

Shift Supervisors

A Shift Supervisor has the responsibility for the activities in his areas. He schedules and assigns work tasks according to equipment availability, personnel availability and overall processing requirements. He ensures that operating procedures are available and are clearly understood by all of his personnel, as necessary. He ensures that all prescribed radiation protection and nuclear criticality safety controls and procedures are followed by all personnel under his jurisdiction. He requires compliance with Federal, State and licensing requirements. He ensures that his equipment is operating properly and that his personnel have the necessary tools and supplies. He is directly responsible for the health and safety of personnel under his supervision.

Materials Manager

The Materials Manager is responsible for the direction of the Radiation Protection Component, Transportation, Purchasing, Inventory Control, and Shipping and Receiving. He is responsible for the management of fissile materials from the time they leave the enrichment facility until such time as completed fuel assemblies are delivered to the customer at the reactor site. Included in the scope of the function is the processing of uranium in subcontractors' facilities.

Regulatory Compliance Manager

The Manager, Regulatory Compliance, is responsible for the establishment and guidance of programs in nuclear criticality safety, radiation protection, SNM safeguards and environmental pollution control for the Columbia Site. The nuclear criticality safety, SNM safeguards and radiation protection

1.4.3 (Continued)

procedures have been written into the Regulatory Compliance Procedures Manual, for the Columbia Site. This Manual serves as the basic guide to supervision in matters related to nuclear criticality safety, SNM safeguards and radiation protection. Any revision of this Manual is first prepared by the Radiation Protection Component in conjunction with recommendations of cognizant first level managers. The revisions require approval by the Regulatory Compliance Manager and the NFD Columbia Plant Manager.

Nuclear Engineering Manager

THe Manager, Nuclear Engineering, is responsible for the nuclear design and analysis of pressurized water reactor initial and replacement cores. In addition, he provides consultation for Nuclear Fuel Division personnel on nuclear criticality safety issues.

Safety Manager

The Manager, Safety is responsible for the establishment and guidance of programs in industrial safety, industrial hygiene and fire protection for the Columbia Site.

The safety procedures have been written into the Safety Manual for the Columbia Site. This Manual serves as the basic safety guide to supervision in matters related to industrial safety, industrial hygiene and fire protection. Any revision of this Manual is first prepared by Safety in conjunction with recommendations of cognizant first level managers. The revisions are reviewed by the Site Safety Policy Committee and require approval by the Safety Manager (Safety Policy Committee member) and the NFD Columbia Plant Manager (Safety Policy Committee chairman).

1.4.4 Functions of Radiological Health and Safety Personnel Assigned within Organizational Operating Units

A staff activity with responsibility for radiological safety maintains a general surveillance program and reports system

1.4.4 (Continued)

and instigates action at appropriate management levels, as required. For purposes of efficiency and functional homogeneity, such services as Health Physics, Nuclear Criticality Safety, SNM Safeguards and Environmental Pollution Control may be provided by one group.

Customary industrial hygiene and safety responsibilities are exercised by a Safety Manager who functions as part of the personnel relations staff in accordance with conventional practices.

The Regulatory Compliance group at Columbia functions on a site-wide basis to assure consistent, coherent and comprehensive planning and application of overall radiological safety programs. Consultation, general surveillance and coordination of radiological safety and industrial hygiene policies among the various NES sites is provided by the Director, Licensing, Safeguards and Safety, who functions at the NES staff level. He has a broad background of experience in safety, industrial hygiene and radiation protection.

In the case of the Nuclear Fuel Division operations, any and all personnel concerned with the operation of the facility have the right to review or request review of the safety of any operating step or procedure. Specifically, the Regulatory Compliance staff member on duty has the authority to forbid, through the cognizant supervisor, any operation, which in his opinion involves undue immediate hazard, until the situation is reviewed with cognizant supervision and there is a satisfactory resolution of the methods and procedures to be used. Continuous daily contact between the personnel of the Regulatory Compliance group and operational staff members assures appropriate consideration of new problems as they arise.

1.5 Personnel

The technical qualifications of the personnel currently holding positions of interest are given below:

1.5 Personnel

W. H. Britton - NFD Columbia Plant Manager - B.S., Carnegie-Mellon University - Metallurgy

Mr. Britton has had twenty-eight years of experience in the nuclear industry and two additional years as an engineer in the steel industry. His nuclear experience includes assignments in nuclear fuels engineering, manufacturing development, and manufacturing operations.

He was employed by the Westinghouse Electric Corporation Bettis Lab from 1955 to 1978 in various engineering, supervisory and management capacities associated with nuclear fuel development and manufacturing projects in the naval and breeder reactor programs.

Mr. Britton joined the Westinghouse NFD Columbia Plant in 1978 as Manager, Process Development. In this capacity, he was responsible for the development of new chemical and mechanical processes related to the manufacture of nuclear fuel assemblies.

In 1979, he was appointed as Manager, Chemical Operations at the Columbia Plant, with overall responsibility for the manufacturing of uranium powder and pellets and fuel rod loading and welding. In 1980, his responsibilities were expanded to all nuclear fuel manufacturing.

He was appointed as NFD Columbia Plant Manager in April 1983.

W. J. Hartnett - Materials Manager - B.S. Industrial Engineering, University of Pittsburgh (1964).

Mr. Hartnett has twenty-six years experience with Westinghouse in the Commercial nuclear field. He joined the Commercial Atomic Power Activities of Westinghouse in the Purchasing Department in 1956. His main responsibilities involved the procurement of fuel assembly components for BR-3, Yankee (Rowe) and other reactor projects.

He later was appointed to management positions in purchasing where he directed and personally negotiated contracts for reactor internals, fuel cladding, uranium conversion services,

1.5 (Continued)

uranium concentrate purchases and various control rod components.

In 1969 he was assigned to the startup of the Columbia Plant of the Nuclear Fuel Division where he was designatea the Materials Manager. He currently directs the efforts of the following departments: Regulatory Compliance, Transportation and Purchasing.

W. L. Goodwin - Regulatory Compliance Manager - B.S., University of Tennessee - Nuclear Engineering.

Mr. Goodwin received his B.S. degree in Nuclear Engineering from the University of Tennessee in 1965. From September 1965 to June 1968 he was employed by E. I. DuPont de Nemours and Company at the Savannah River Plant (DOE installation) as a Nuclear Engineer. His responsibilities included design, modification and evaluation of both health physics instrumentation and in-process radiation monitoring instruments and systems.

Mr. Goodwin joined Westinghouse in July 1968 as a Criticality and Health Physics Engineer in the Manufacturing Denartment of the Nuclear Fuel Division. He had responsibilities, primarily, for the nuclear safety evaluation of all new manufacturing equipment, processes and facilities. He also conducted nuclear criticality safety audits and performed safety analyses to permit the processing of uranium at low enrichments.

In 1973 Mr. Goodwin was appointed Manager, Health Physics and Criticality, for the the NFD Manufacturing Department. In this position, he had responsibility for the maintenance and coordination of all aspects of the radiation protection and nuclear criticality safety functions for the entire Columbia Site.

In April 1974, Mr. Goodwin was reassigned as Manager, Radiological and Environmental Services, for the NFD

1.5 (Continued)

Manufacturing Department. In this position he has responsibility for the establishment, implementation and maintenance of all environmental pollution control and surveillance programs as well as that for radiation protection and nuclear criticality safety for the entire Columbia Site. He serves as a member of the Columbia Plant Safety Policy Committee. In 1978 the title of this department was changed to Regulatory Compliance, and in 1982, SNM Accountability was included as an additional responsibility.

Mr. Goodwin is a member of ANS 8.13 Subcommittee for Nuclear Criticality Safety and Health Physics Society Subcommittee for Internal Dosimetry Standards.

C. F. Sanders - Radiation Protection Engineering Manager - B.S., University of Tennessee - Nuclear Engineering.

Mr. Sanders has had seventeen years of experience in the nuclear industry. His nuclear experience includes assignments in process development, fuels performance, nuclear criticality safety and radiation protection.

He was employed by Union Carbide Corporation at the Oak Ridge National Laboratory from 1965 to 1973. He held the position of Engineering Technician until he received his degree in 1969. From 1969 to 1973 he held the positions of Development Engineer and Research Associate. His responsibilities included research and development assignments associated with fuel fabrication processes and inspection techniques, and fuels performance testing.

Mr. Sanders joined the Manufacturing Department of the Westinghouse Nuclear Fuel Division as a Senior Health Physicist in 1973. In this position he was responsible for performing nuclear criticality safety analyses to permit the processing of low enriched uranium, conducting audits of plant processes and operations, and providing personnel training.

1.5 (Continued)

In 1974 Mr. Sanders was appointed Manager, Health Physics. In this position he was responsible for the maintenance and operation of the Plant Health Physics Counting Room, for the formulation, monitoring and enforcement of all radiation protection programs, and for health physics training.

In 1978 he was appointed as Regulatory Compliance Engineering Manager, responsible for health physics and nuclear criticality safety engineering functions within the Regulatory Compliance group. In 1981, SNM safeguards functions were included as an additional responsibility.

He is a member of INMM Safeguards Committee for low-enriched uranium, Health Physics Society and AIF subcommittees concerned with rewriting 10 CFR 20 and the ANS Subcommittee on Measurement of Radioactive Material in Waste.

H. T. King - Criticality Engineer - B.S., Wake Forest University - Math and Physics - M.S., University of South Carolina - Nuclear Engineering.

Mr. King has approximately thirty-three years of experience in the nuclear industry. His nuclear experience includes assignments in reactor technology, separations chemistry, health physics and nuclear criticality safety.

In 1949 Mr. King accepted an engineering position in the Radiation Chemistry Group of the NEPA Division of Fairchild Engine and Aircraft Corporation at the USAEC's Oak Ridge facility. His primary responsibility was the determination of high temperature radiation effects in materials.

He was employed by the E. I. DuPont de Nemours and Company from 1951 to 1959. While employed by DuPont, he held positions in the Separations Training Group at the Oak Ridge National Laboratory, Operations Group for the Reactor Prototype at the New York Shipbuilding Company in Camden, New Jersey, and Health Physics and Reactor Technology Sections at the US-ERDA's Savannah River Plant.

1.5 (Continued)

Mr. King accepted a position as Environmental Physicist with Carolinas-Virginia Nuclear Power Associates in 1959. There he had responsibilities associated with effluent and environmental monitoring programs, bioassay program and the maintenance and operation of the Health Physics Counting Room. He also had specific responsibility for the health physics program associated with a Vapor Container Leakage and Maximum Credible Accident Study performed for the USAEC by Phillips Petroleum Company.

From 1969 to 1970 he worked as a Shift Health Physicist in the Radiation Monitoring Division of Norfolk Naval Shipyard. He was responsible for the maintenance of radiation and contamination control programs during the repair and overhaul of nuclear-powered submarines.

In 1970 Mr. King accepted a position as Senior Health Physicist in the Manufacturing Department of the Westinghouse Nuclear Fuel Division. From 1970 to 1974 he was responsible for the engineering aspects of the Columbia Site health physics and environmental pollution control programs. Presently he has prime responsibility for nuclear criticality safety administration.

B. F. Cooney - Senior Engineer, Nuclear Engineering - B.S., Carnegie-Mellon University - Physics - M.S., Carnegie-Mellon University - Nuclear Engineering

Mr. Cooney accepted an engineering position with the Nuclear Engineering section of the Nuclear Fuel Division in 1977. From 1977 to 1980, he was responsible for the nuclear design of several domestic PWR cores. In 1980, he was assigned responsibility for the criticality analysis efforts in Nuclear Engineering, which included shipping containers and fuel storage racks. As part of these responsibilities, he has directed a large number of criticality analyses, including the analyses to support the January 1981 renewal of SNM-5450 (RCC

1.5 (Continued)

shipping container license). Concurrent with his criticality responsibilities, he was heavily involved in nuclear fuel product development.

A. T. Sabo, Director, Safety and Industrial Hygiene, Nuclear Energy Systems

Mr. Sabo is responsible for directing a Division staff function for the purpose of providing overall coordination and general surveillance of the effectiveness of the safety and health protection activities throughout the NES Divisions.

Mr. Sabo received his B.S. in Biochemistry from the University of Pittsburgh in 1951 and has taken additional course work in industrial hygiene, health physics, and ventilation at the University of Pittsburgh's Graduate School of Public Health and at Michigan State University.

From 1951 to 1956 he was employed as an Industrial Hygienist in the Pennsylvania Department of Health, Pittsburgh District, conducting scientific, industrial, and environmental surveys throughout Western Pennsylvania.

Joining the AEC's Bettis Atomic Power Laboratory (operated by Westinghouse) in 1956, Mr. Sabo worked as an area coordinator for the industrial hygiene, helath physics, and safety programs dealing with fuel fabrication and associated material processing. At this time he was also a member of the PNROO team (Bettis Radiological Monitoring Team) which served the NYOO district.

He was employed by the Nuclear Materials and Equipment Corporation from early 1962 to January 1963, where he was in charge of the health and safety programs at the Advanced Materials Center near Leechburg. His duties involved directing all health physics, industrial hygiene, and safety policies for plants primarily geared to the development and production of plutonium-bearing materials, powdered metals and radioisotopes.

1.5 (Continued)

Mr. Sabo joined the Westinghouse Atomic Power Divisions as Manager, Industrial Hygiene and Safety in 1963, and was appointed to his present position in 1969.

Mr. Sabo is a member of the Health Physics Society, the American Nuclear Society, the American Industrial Hygiene Association, and the American Society of Safety Engineers.

1.6 Emergency Procedures

The emergency procedures and equipment are described in the Columbia Plant Site Emergency Plan, as amended.

1.7 Radiological Safety Procedures

1.7.1 Health Physics Operations Procedures Manual

The radiation protection procedures, practices and equipment employed at the Columbia Site are described in the Health Physics Operations Procedures Manual. This manual deals with such general topics as bioassays, personnel exposure monitoring and control, contamination control, effluent and environmental monitoring, and health physics technician training. It contains the radiation protection procedures and limits which have been instituted to ensure compliance with 10 CFR Part 20 and NRC license requirements.

The Health Physics Operations Procedures Manual is maintained by the Radiation Protection Component and is used primarily as a training manual and procedures document for the technical and professional personnel within the component. Copies of the manual, however, are distributed to key management personnel for informational purposes, and are made readily available to any Columbia Site employe upon request.

The Radiation Protection Component periodically reviews and revises this manual as required. Revision of the manual requires the written approval of the Radiation Protection Component Manager. No revision is made which would compromise compliance with applicable NRC regulations and license conditions.

1.7.2 Regulatory Compliance Procedures Manual

General criteria, rules and requirements are contained in the Regulatory Compliance Procedures Manual. This manual deals with such general topics as radiation protection, nuclear criticality safety, environmental control and nuclear material safeguards. The procedures contained therein establish the respective policies and define the responsibilities for implementation and enforcement thereof. These procedures also establish the respective criteria and limits that have been instituted to ensure compliance with applicable NRC regulations and license conditions.

1.7.2 (Continued)

The Radiation Protection Component maintains the Regulatory Compliance Procedures Manual and distributes copies of all new and revised procedures to designated manual holders. Manual distribution includes line management in each functional area of the plant where SNM is processed or stored, as well as other key management personnel. The Radiation Protection Component periodically reviews the manual and revises it as required. When new or revised procedures are issued, line management is required to promptly communicate these procedures to affected personnel. Revision of the manual requires the written approval of the Manager of the Radiation Protection Component and the Plant Manager. No revision is made which would compromise compliance with applicable NRC regulations and license conditions.

1.7.3 Safety Manual

The procedures documented in the Safety Manual constitute the general industrial safety procedures followed in the day-to-day manufacturing operations conducted at the Columbia Site. This manual deals with such general topics as industrial safety, industrial hygiene and fire protection. The procedures contained therein establish the respective policies and define the responsibilities for implementation and enforcement thereof.

The Safety Component maintains the Safety Manual and distributes copies of all new and revised procedures to designated manual holders. Manual distribution includes line management in each functional area of the plant as well as other key management personnel. The Safety Component periodically reviews the manual and revises it as required. When new or revised procedures are issued, line management is required to promptly communicate these procedures to affected personnel. Revision of the manual requires the written approval of the Manager of the Safety Component and the Plant

1.7.3 (Continued)

Manager. No revision is made which would compromise the health, safety or welfare of employes and other personnel at the Columbia Site.

1.8 Nuclear Criticality Safety

The nuclear criticality safety of most processing operations may be established using approved maximum permissible values (MPV) for simple geometries. The MPV is based on published values and is chosen to be nuclearly safe assuming optimum light water moderation and reflection for isolated units. Such units are called subcrits. The spacing of these subcrits may be established using approved empirical rules which assure nuclearly safe spacing. The nuclear criticality safety of some subcrits, fuel assemblies and fuel assembly arrays is based on computer calculations. Specific evaluations of situations which are too complex to be subject to approved controls and evaluations of new types of processes and equipment are submitted for review and approval by the NRC.

1.8.1 Nuclear Criticality Safety Values

1.8.1.1 General Maximum Permissible Values

Maximum Permissible Values (MPV) for isolated units have been established using published data. The data were converted to minimum critical values in standard units, as required, and plotted. These minimum critical values assume optimum water moderation and full reflection. MPV's are established using the safety factors listed below,

where $MPV = \frac{\text{Minimum Critical Value}}{\text{Safety Factor}}$

Mass (Administrative Controls)	2.3
Mass (Engineered Controls)	1.3
Volume	1.3
Cylinder Diameter	1.1
Slab Thickness	1.2

1.8.1.1 (Continued)

MPV's have been established for the three principal forms of SNM which will be present in the facility: aqueous solutions, homogeneous UO2-water mixtures, and heterogeneous UO2-water mixtures. The values established for aqueous solutions are applicable to UO2F2 systems, ammonium diuranate (ADU) slurries and materials which have a theoretical uranium density less than 3.2 g U/cm^3. Measurements of ADU slurries in the range of 20-50 w/o ADU, which is the density range obtained by centrifuging, indicates that the ADU slurries fulfill the uranium density-moderator relationships for UO2F2 solutions. The H/U of these slurries is greater than 30 at 50 w/o ADU. Maximum densities obtained by settling are 25 w/o ADU. When the ADU slurry is converted to oxide, the limits for UO2 are applicable to the operation. The UO2 values apply to all uranium oxides.

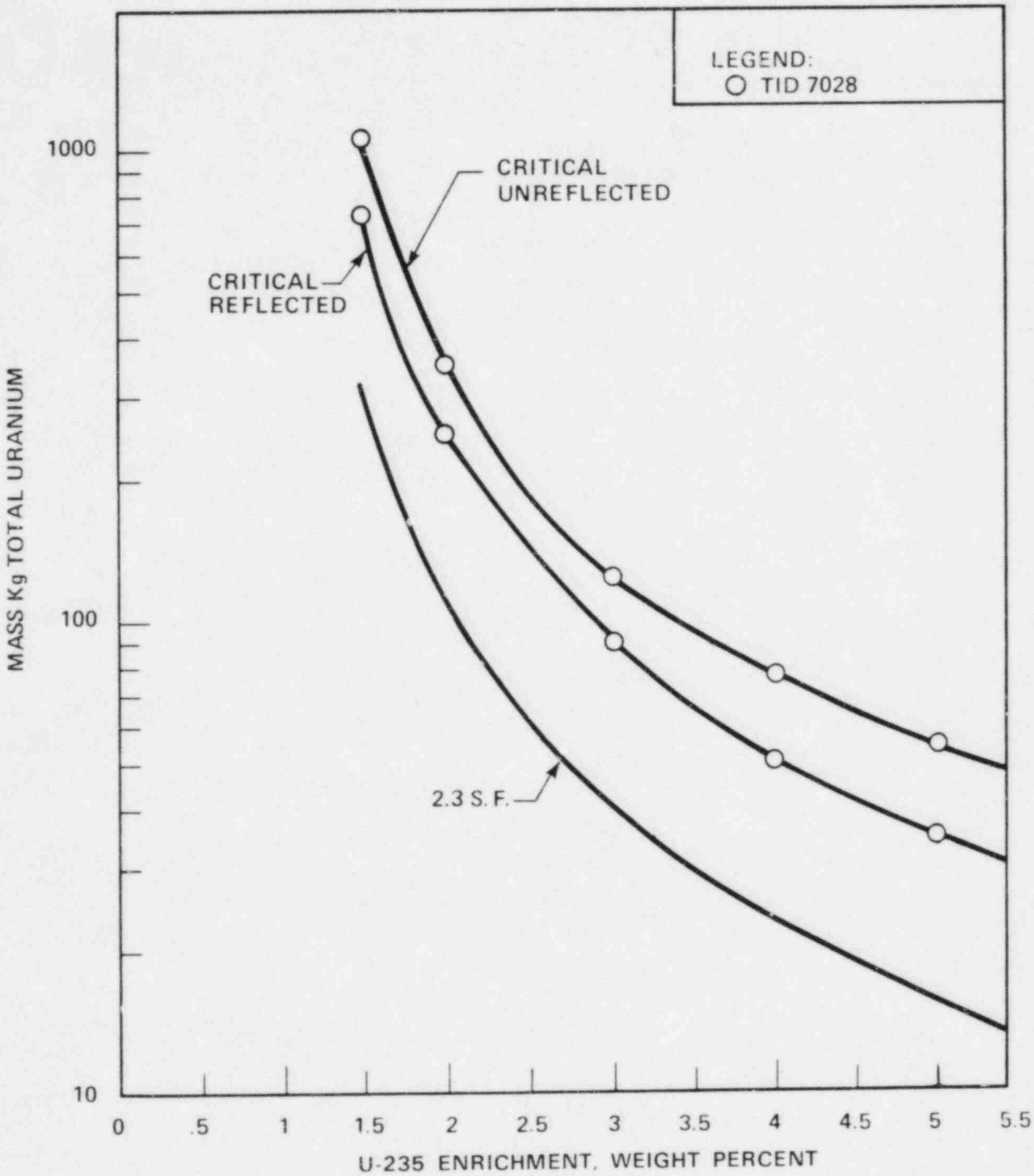
Heterogeneous values are conservatively applied to clad oxide rods. Cylinder data are applied on an equal cross sectional area to other shapes.

Minimum critical values and MPV's are summarized in Figures 1.8.1.1 through 1.8.1.12. Uranium enriched to more than 5 w/o is limited to a nuclearly safe value of 350 grams of contained U-235 by license condition and is isolated from other SNM.

1.8.1.2 Moderation-Controlled Subcrits

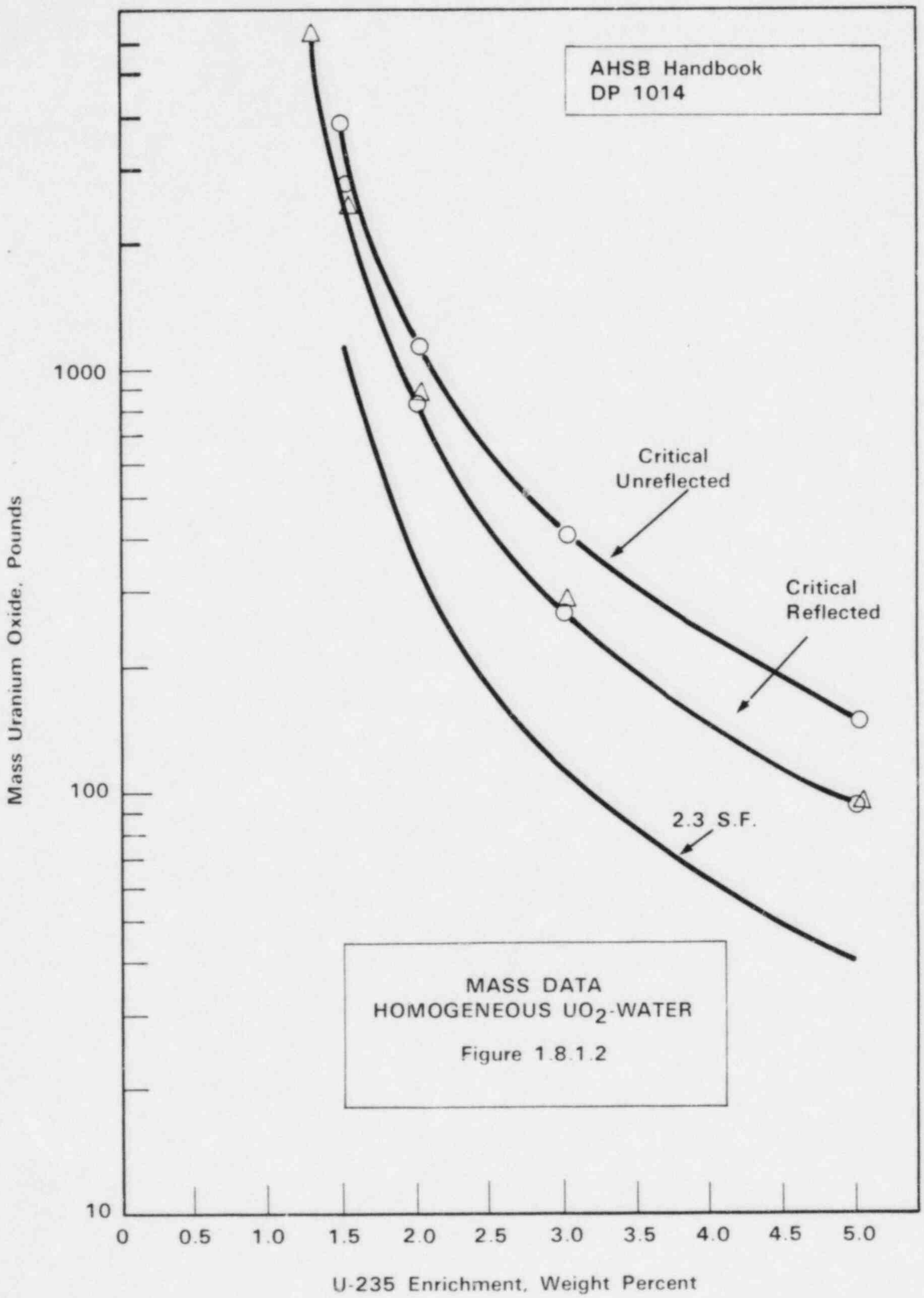
UF6 cylinders -- The nuclear criticality safety aspect of UF6 in 30-inch diameter, UF6 cylinders based on moderation-control is documented in K-1920 and K-1663. These cylinders are stored with a minimum of one foot edge-to-edge separation with no interaction considerations.

Dry UO2 in closed equipment and subcrit container -- Calculations for UF6 reported in K-1663 indicate the K infinite for U(5.0) F6 with an H/U-235 $\leq$10 is less than 1.0. Based on the nuclear similarity of oxygen and fluorine, and on the shape of the curve in Figure 3 of K-1663, this value is



Mass Data for Solutions

Figure 1.8.1.1



MASS DATA
HOMOGENEOUS UO_2-WATER

Figure 1.8.1.2

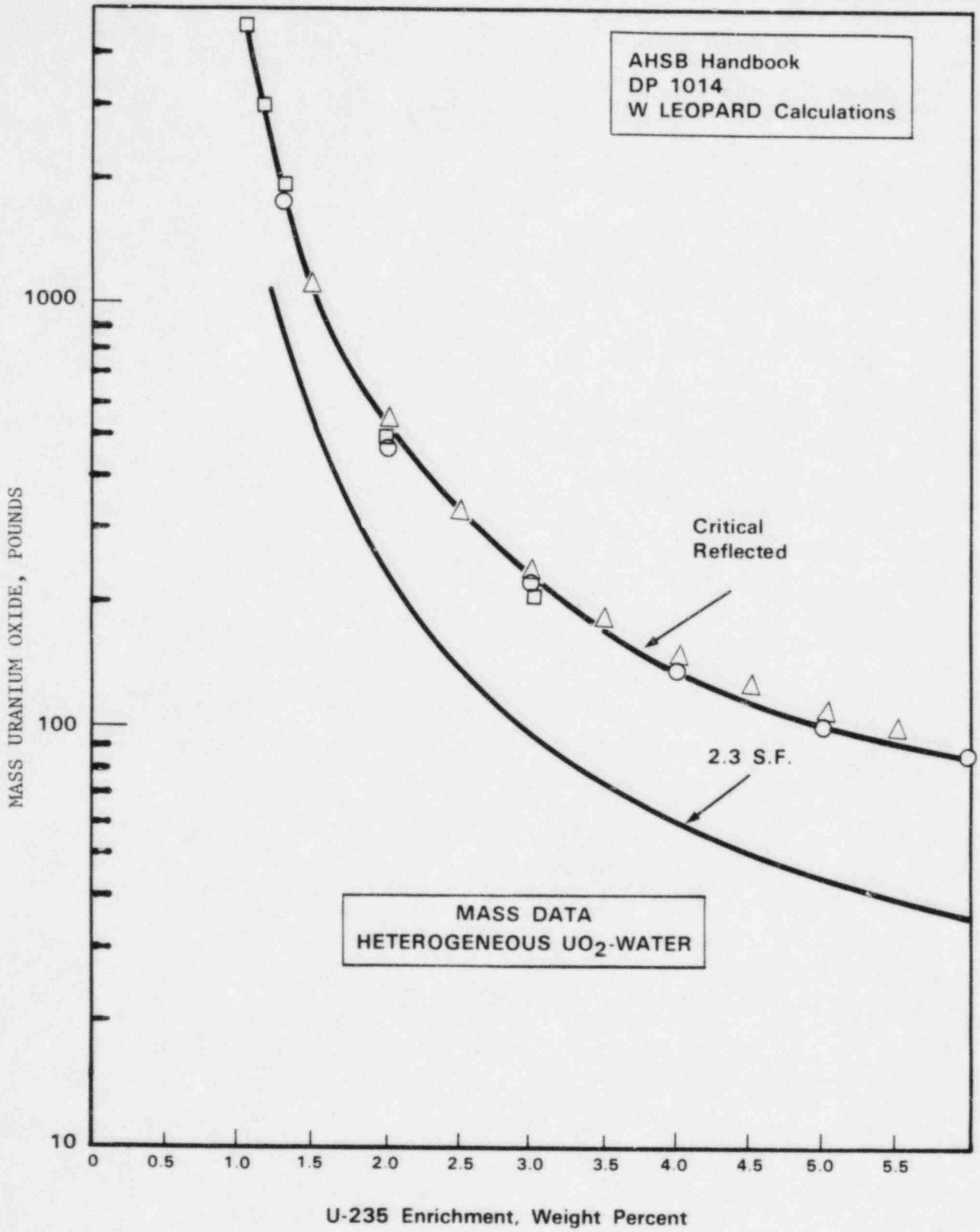
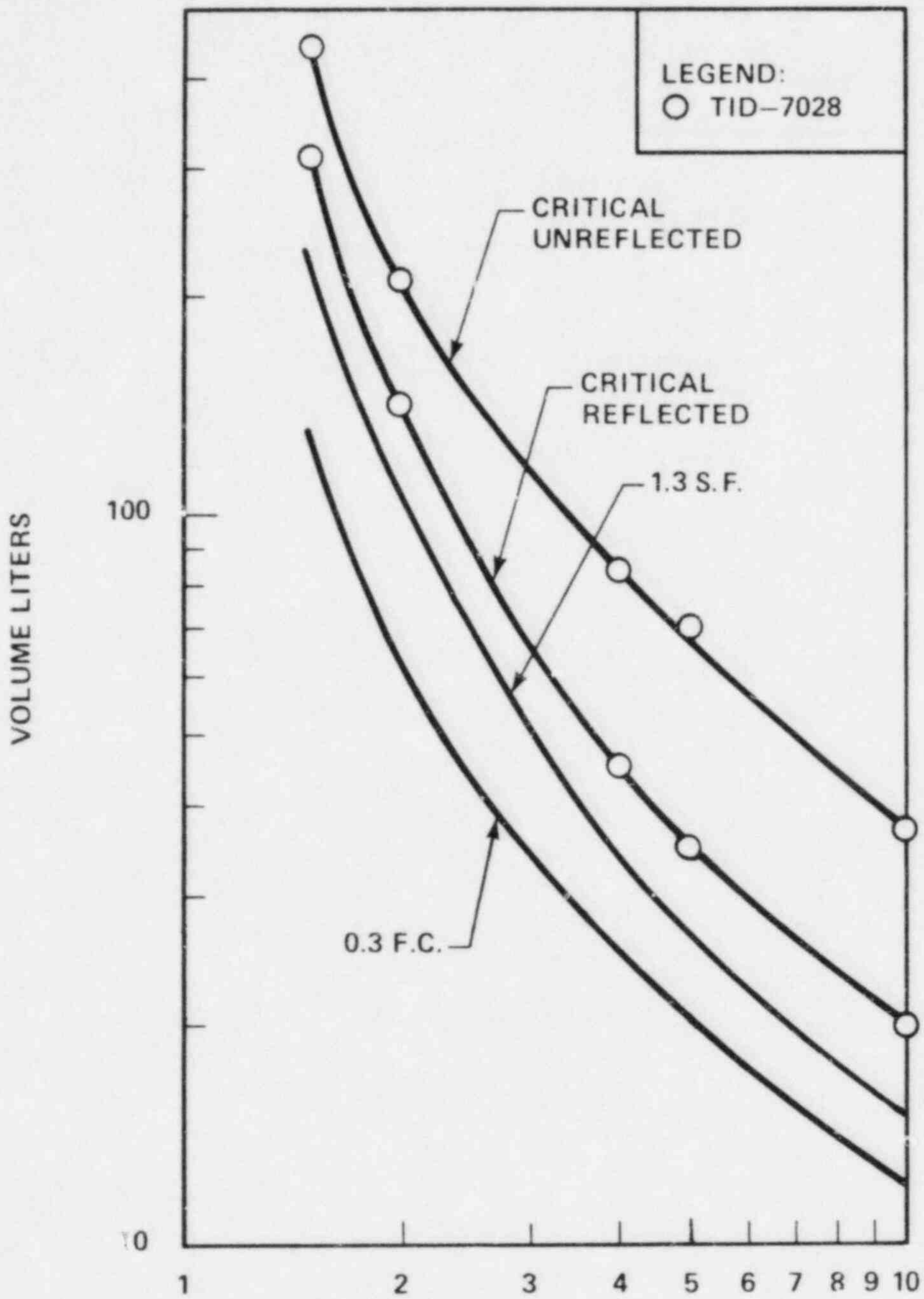
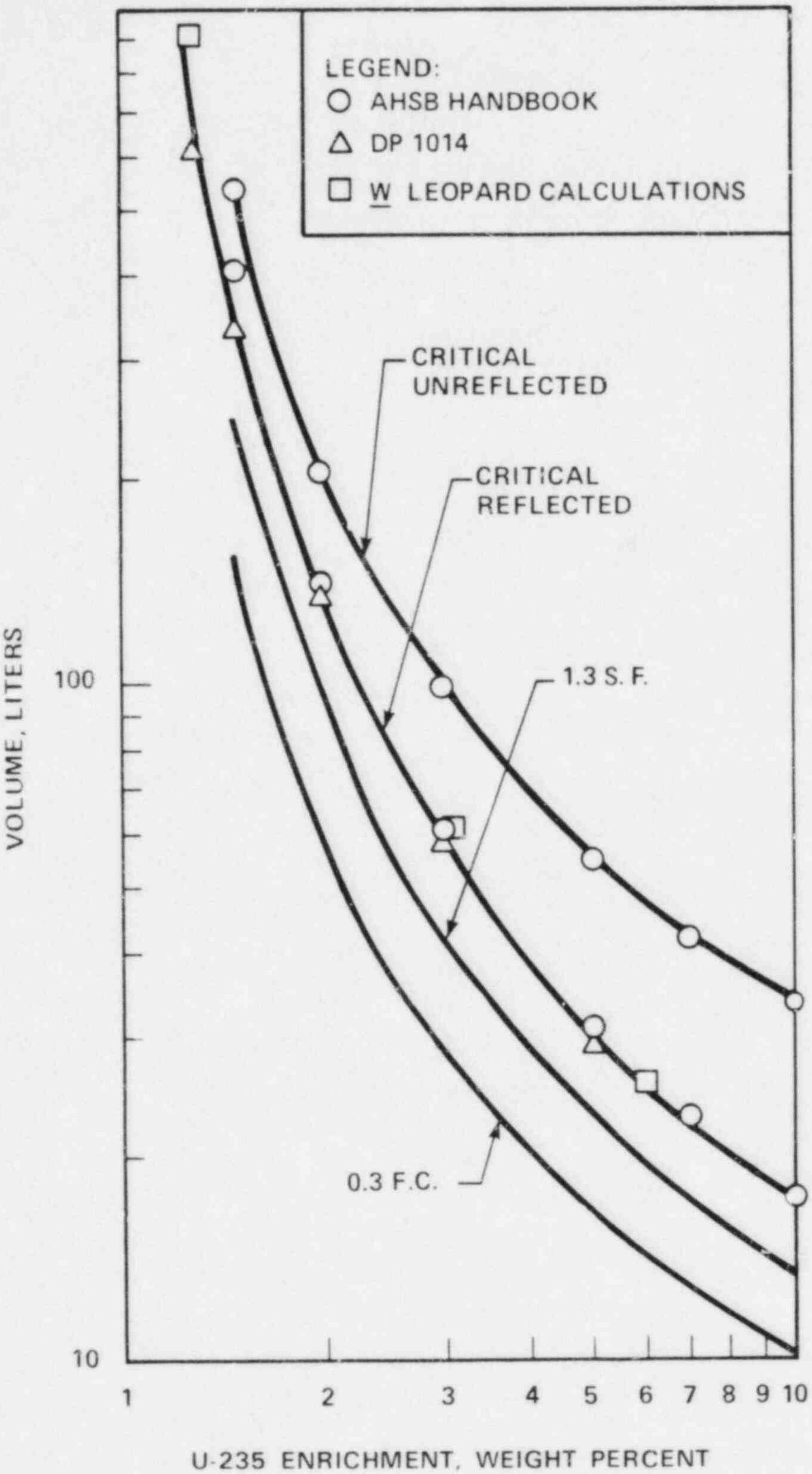

AHSB Handbook
DP 1014
W LEOPARD Calculations
MASS URANIUM OXIDE, POUNDS
1000
100
10
Critical
Reflected
2.3 S.F.
MASS DATA
HETEROGENEOUS UO_2-WATER
0
0.5
1.0
1.5
2.0
2.5
3.0
3.5
4.0
4.5
5.0
5.5
U-235 Enrichment, Weight Percent

Figure 1.8.1.3

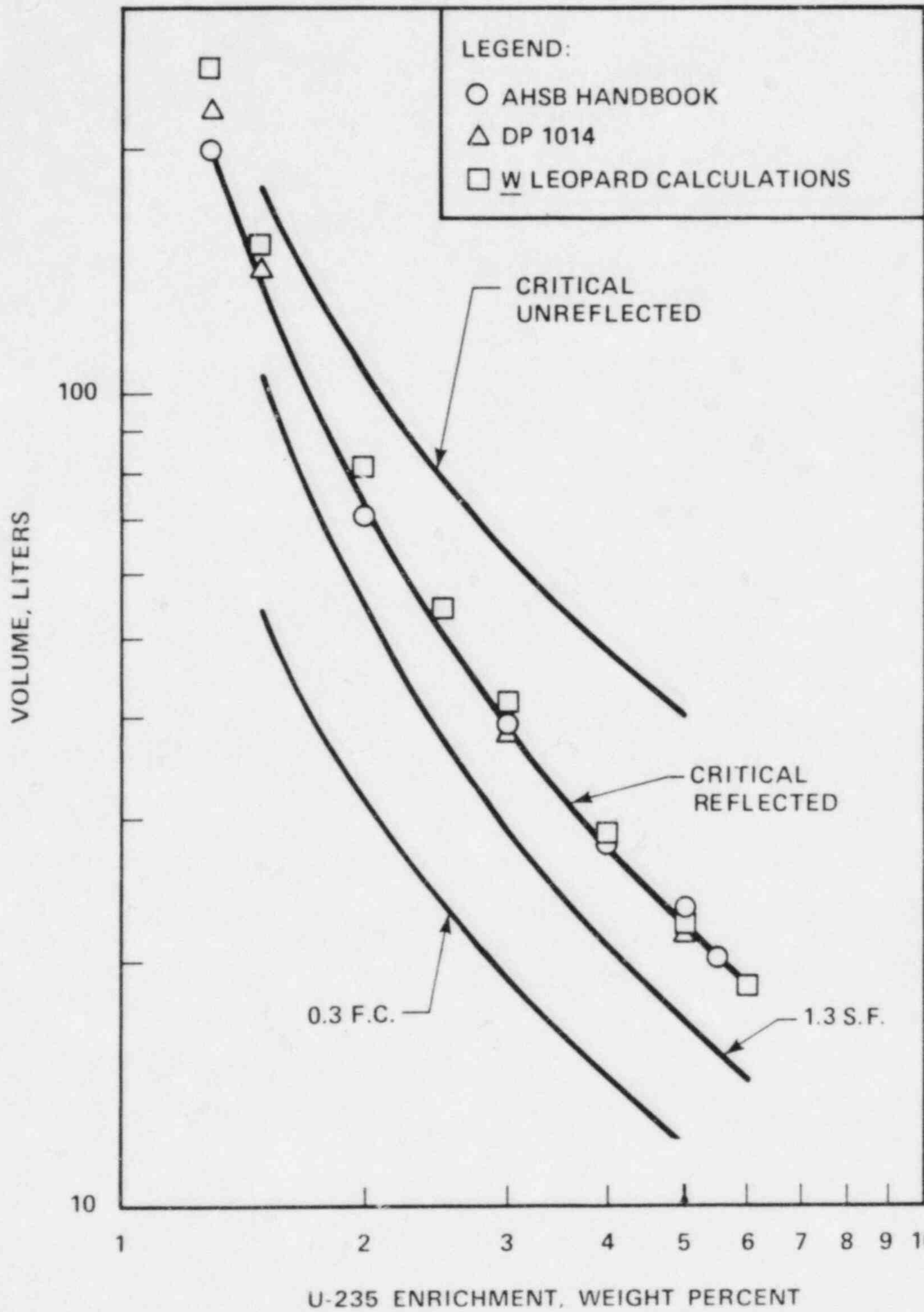


U-235 ENRICHMENT, WEIGHT PERCENT
Volume Data for Solution
Figure 1.8.1.4



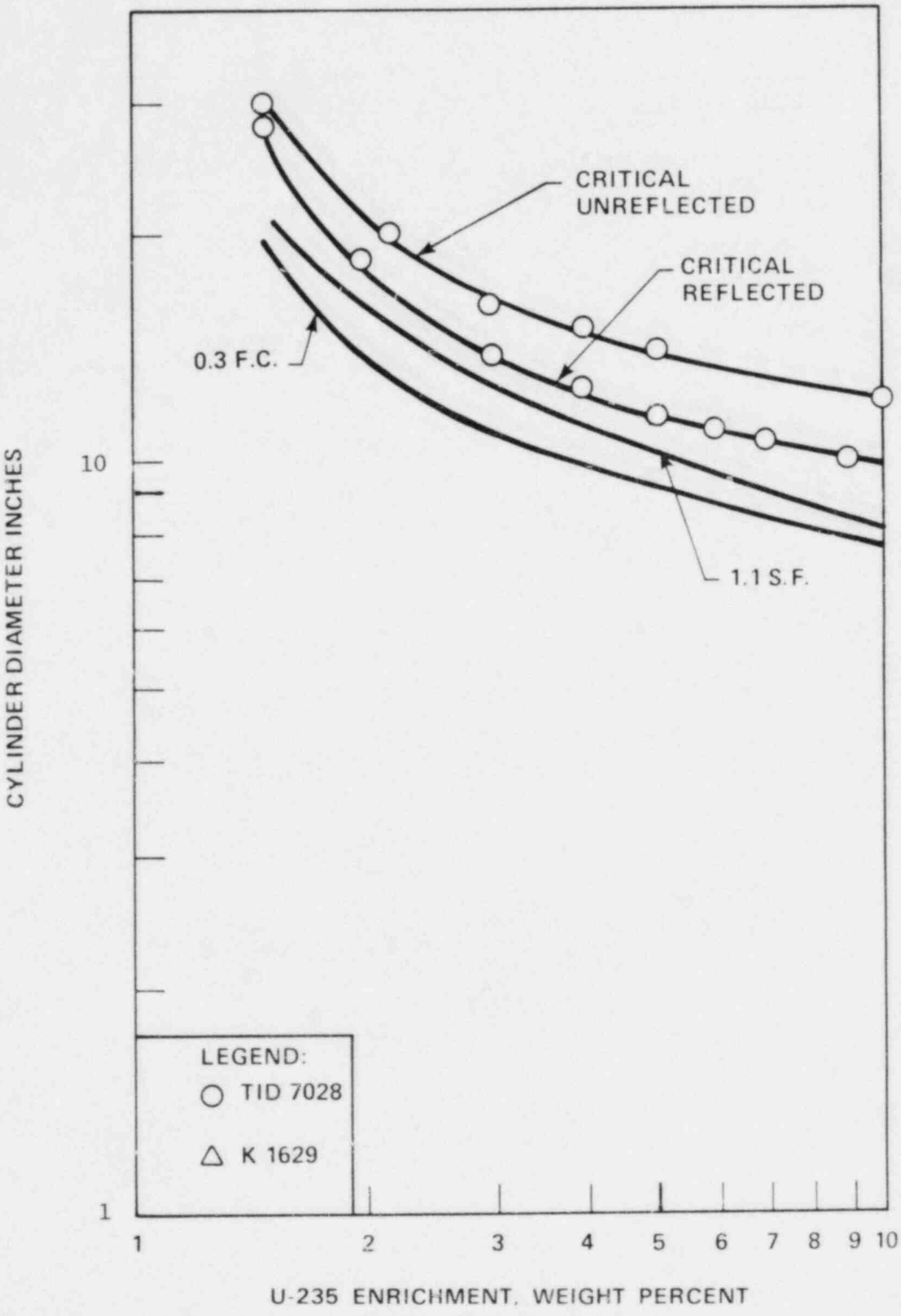
Volume Data Homogeneous UO_2-Water

Figure 1.8.1.5



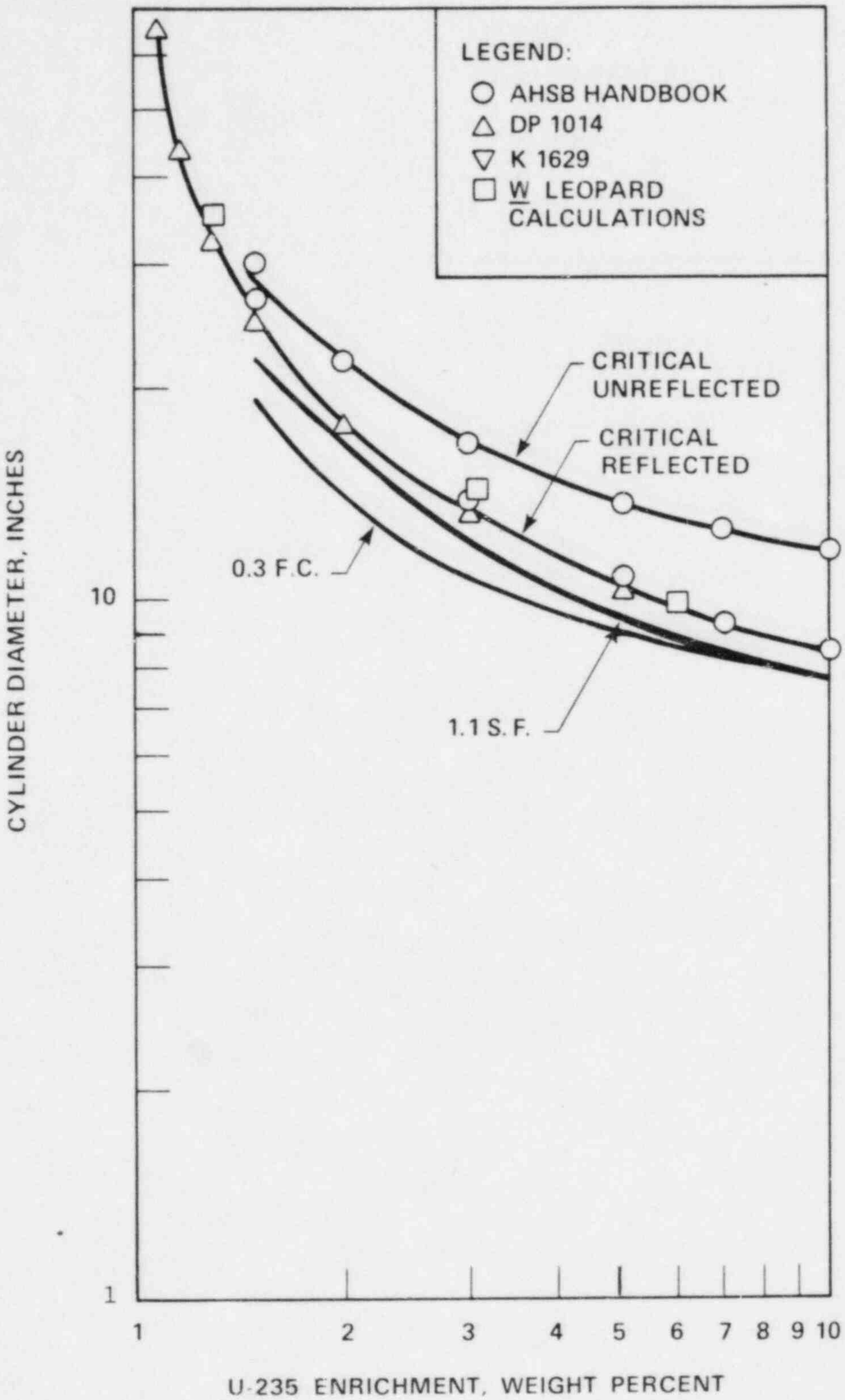
Volume Data Heterogeneous UO_2-Water

Figure 1.8.1.6



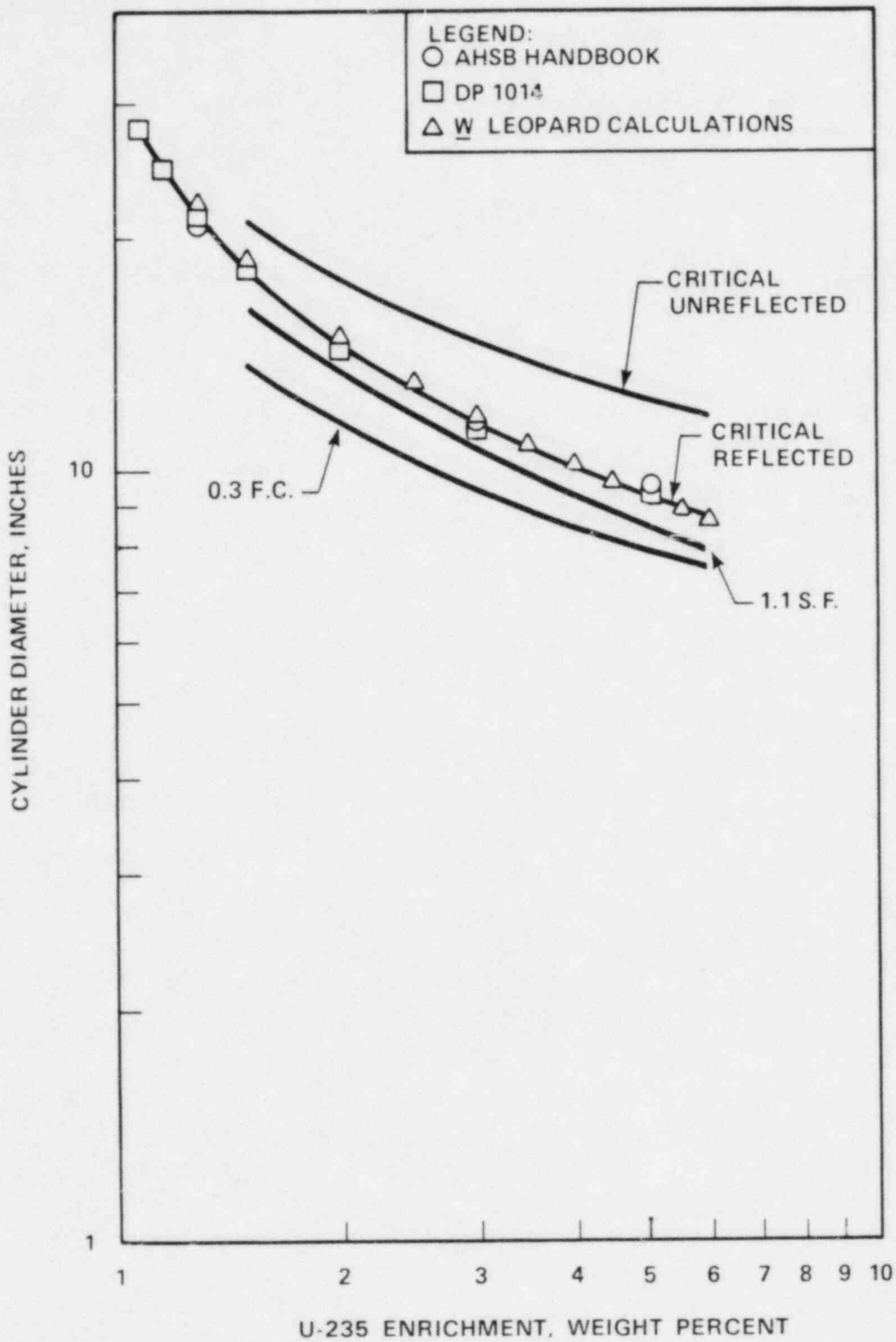
Cylinder Data for Solutions

Figure 1.8.1.7



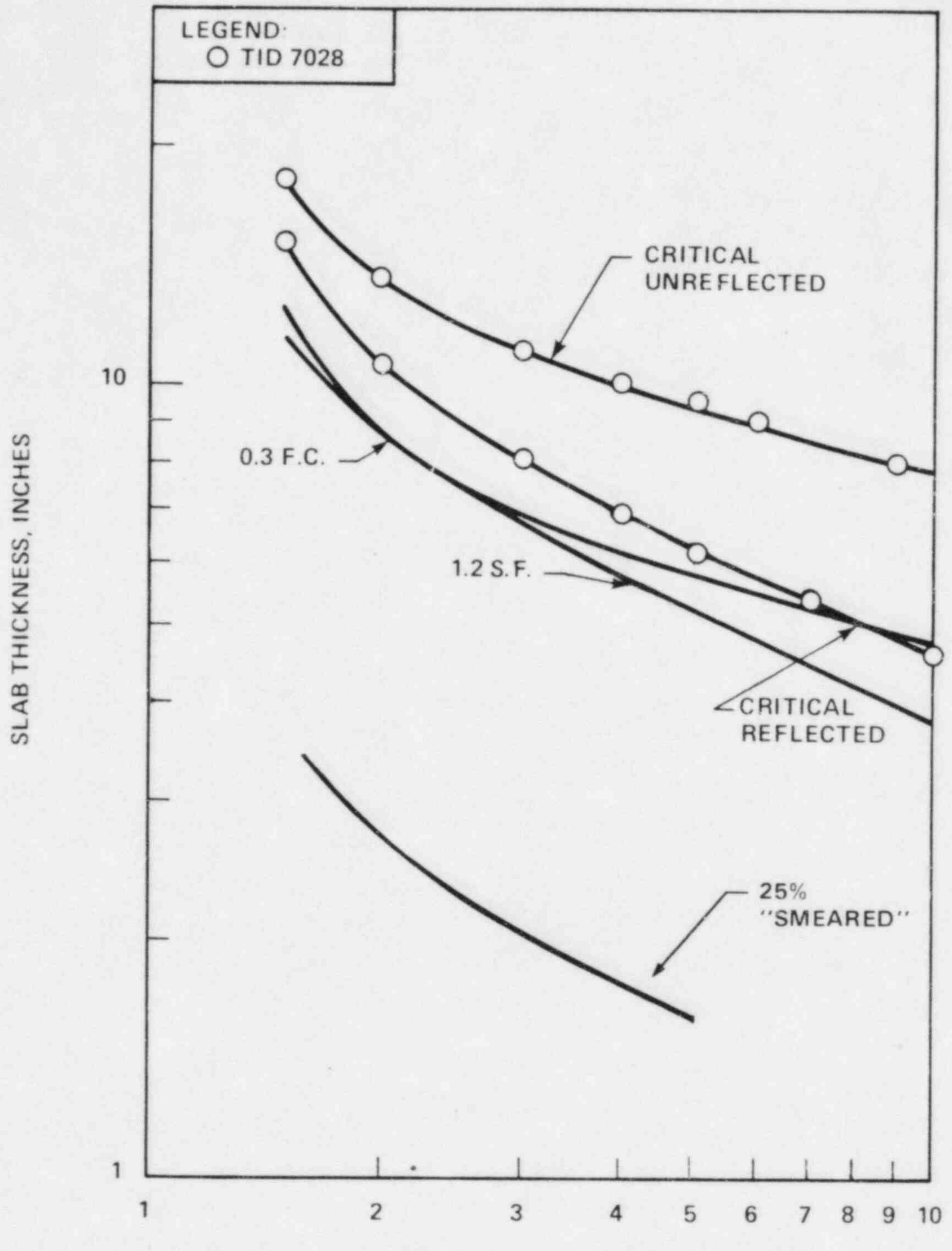
Cylinder Data Homogeneous UO_2-Water

Figure 1.8.1.8



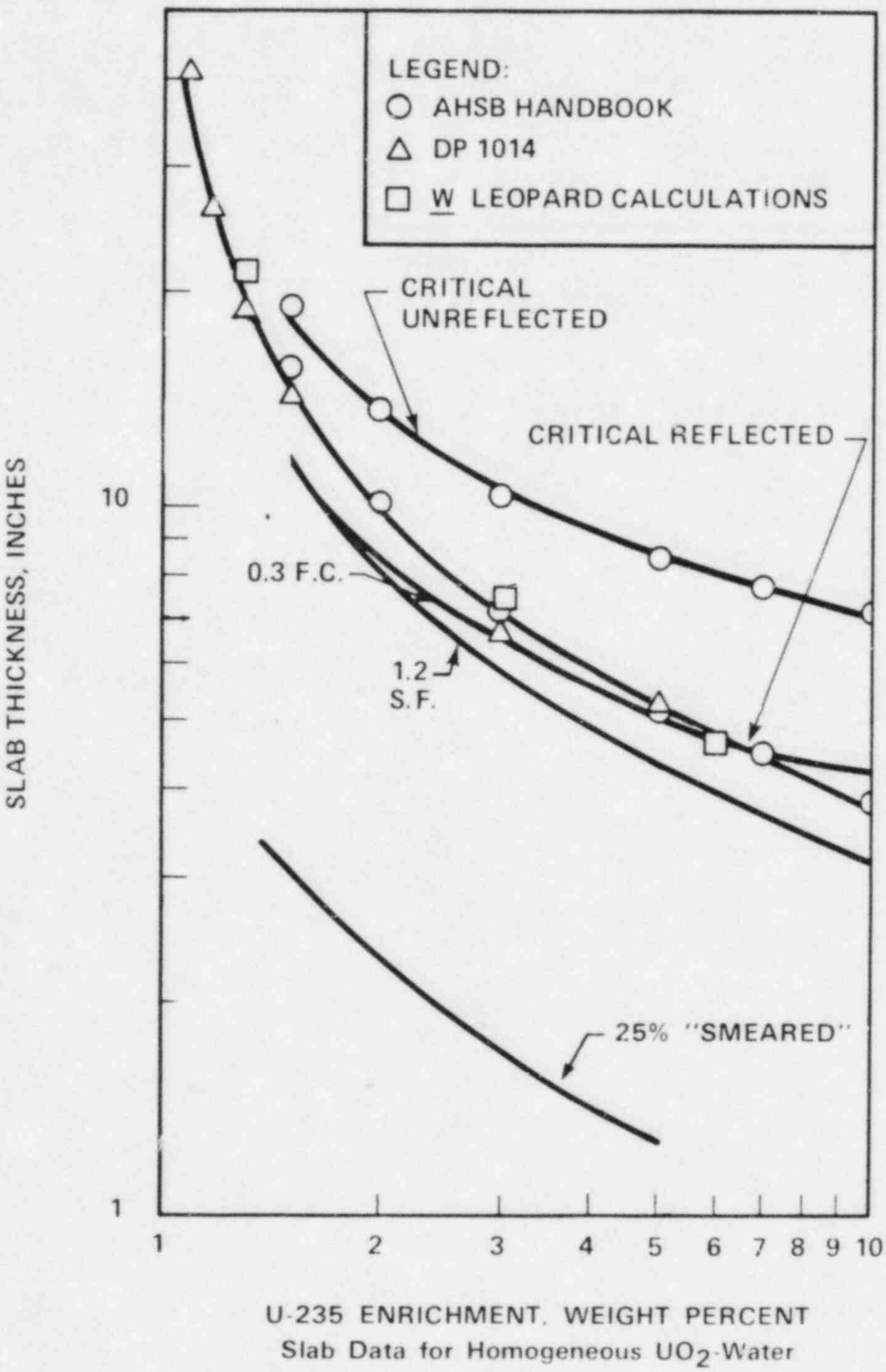
Cylinder Data Heterogeneous UO_2-Water

Figure 1.8.1.9



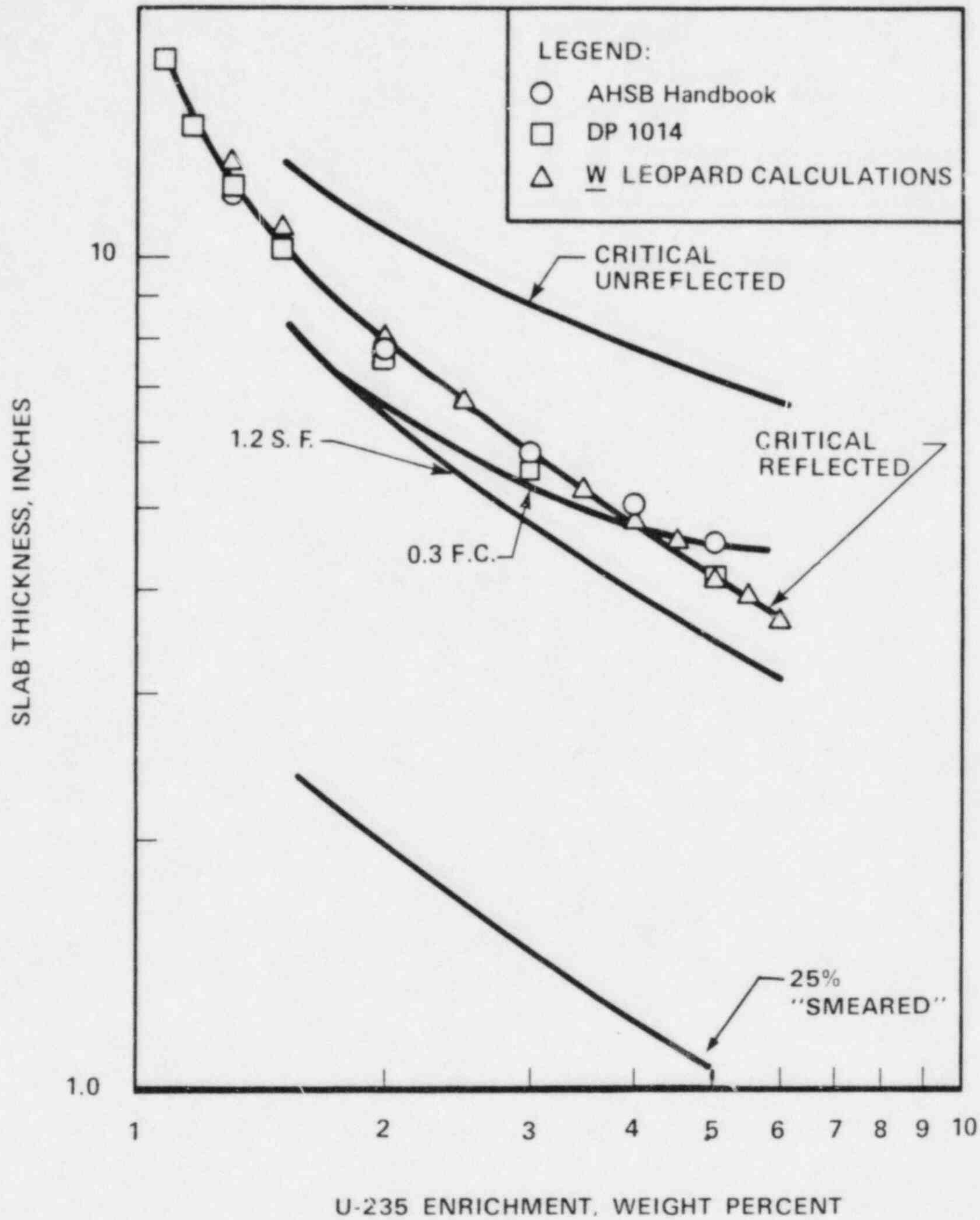
Slab Data for Solutions

Figure 1.8.1.10



Slab Data for Homogeneous UO_2-Water

Figure 1.8.1.11



Slab Data for Heterogeneous UO_2-Water

Figure 1.8.1.12

1.8.1.2 (Continued)

considered to apply to oxides of uranium enriched $\leq$4.15 w/o in the isotope U-235, provided that H/U $\leq$0.5 and the moderator is uniformly distributed in the fissile material. Therefore, such subcrits require no interaction considerations. However, subcrits must be closed or sealed and must meet one of the maximum permissible values established in Subparagraph 1.8.1.1. An array must provide a minimum of one foot edge-to-edge spacing between subcrits. These requirements assure nuclear criticality safety if a subcrit or the array were to be accidentally moderated. The same requirements apply to the oxides of uranium enriched between 4.15 w/o and 5 w/o in the isotope U-235, except that the H/U ratio is limited to $\leq$0.3.

Dry Uranium compounds in large process equipment -- The nuclear criticality safety of limited moderation (moderation controlled) processing operations takes advantage of the fact that, for low-enriched uranium fuels (enriched to less than or equal to 5 w/o U-235) a moderator must be present to attain criticality.

Water Barrier Moderation Controls -- For portions of the Manufacturing Building operated on a basis of nuclear criticality safety control by moderation limitations ("the Moderation Controlled Area"), it is essential to prevent entry of water or other hydrogenous liquids (and certain solids) into process equipment, vessels and containers. The following design principles are therefore followed:

(1) The Moderation Controlled Area is protected against the ingress of external water (or other hydrogenous liquids), under normal and accident conditions, from areas not subject to moderation control.

1.8.1.2 (Continued)

(2) The building construction (roof and walls) is of high integrity, and surround a secondary high integrity "water barrier" enclosing the Moderation Control Area.

(3) Equipment is designed to minimize the use of hydrogenous materials as services; only process feed lines essential to the enclosed operaton are permitted to pass through the water barrier, and only analyzed hydrogenous materials and evaluated material containers specifically authorized by the Radiation Protection Component may be introduced to the Moderation Control Area.

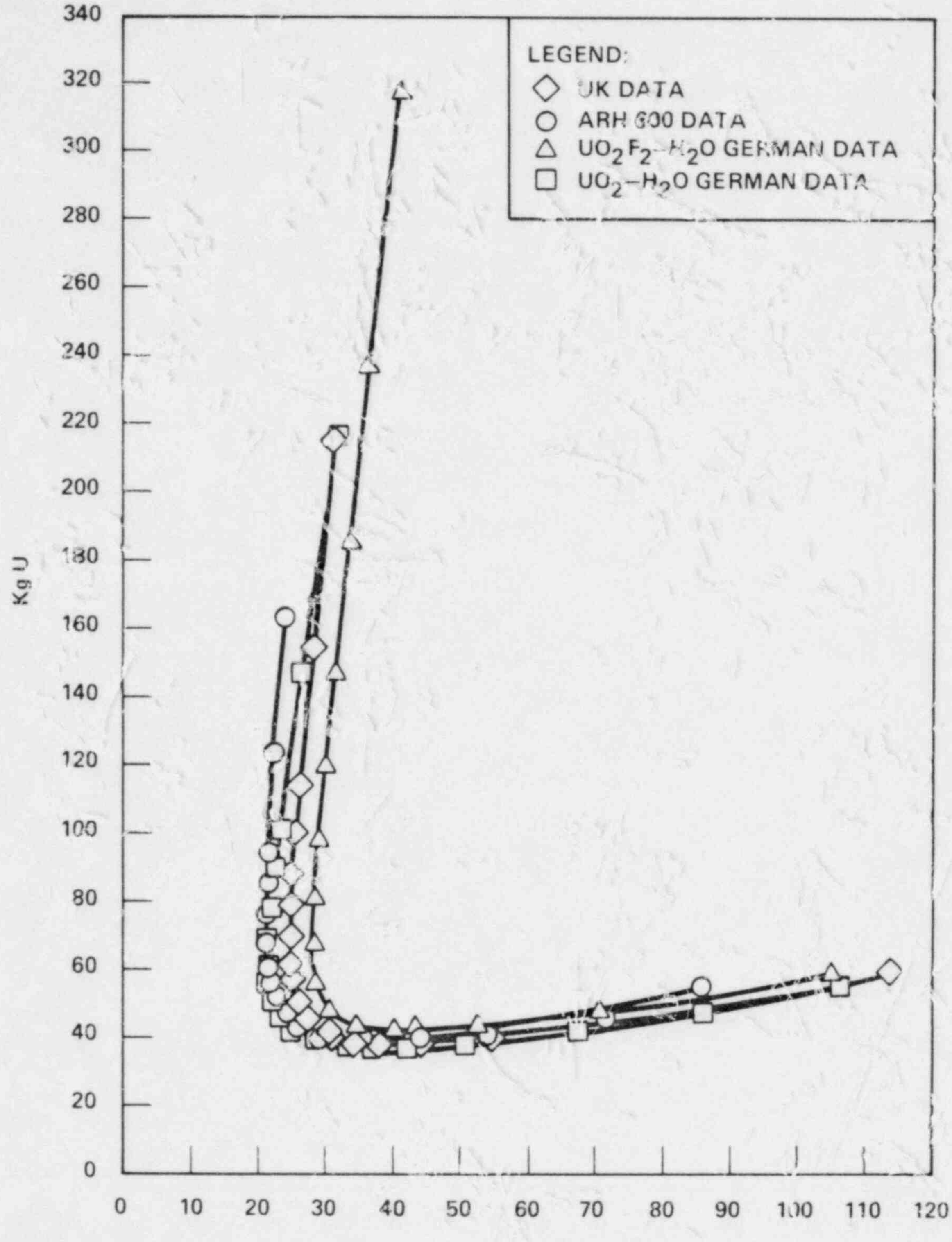
Processing Operation Moderation Controls -- The nuclear criticality safety of large units of low-enriched uranium, and arrays of such units, based on moderation control where the moderator is assumed to be uniformly distributed throughout the fissile material, has been well documented in the literature (and has been in general use by the nuclear industry for many years). Moderation control can be logically extended to additional applications by establishing suitable administrative and/or engineered safeguards, which necessarily involve consideration of postulated normal and accident conditions in which the moderator might not be so uniformly distributed. A nearly infinite number of combinations of uranium mass and density, moderator mass and density, and uranium moderator system geometry can be postulated for such applications. (Some configurations would require more moderator than a uniform distribution; for example, as a result of tunneling, layering or concentration on vessel walls. Likewise, configurations in which the moderator is concentrated in a reactive geometry could require much less moderator than a uniform distribution, with the limiting case being the optimum fuel moderator ratio in a fully reflected spherical geometry.) Therefore, to develop boundary

1.8.1.2 (Continued)

conditions for moderation control safety analyses, the most restrictive conditions possible are first determined, using UO2 water data from ARH-600 (Reference 11 in Subparagraph 1.8.1.6 of this License Application), and plotted in Figure 1.8.1.13. The ARH-600 data are found to be in substantial agreement with the data in the UKAEA Handbook (Reference 1 in Subparagraph 1.8.1.6) at concentrations up through the minimum critical mass (but are shown to result in somewhat smaller critical volumes at higher uranium concentration); the ARH-600 data are also found to be in substantial agreement with UO2-H2O and UO2F2-H2O data in the Handbook der Kritikalitat' (Reference 12 in Subparagraph 1.8.1.6). These data are also plotted in Figure 1.8.1.13 (which graphically illustrates the wide range of critical conditions which can exist).

Figure 1.8.1.14 is an envelope plot of the published critical data which shows the relationship of uranium mass, water volume, and uranium-water system volume (as a function of system uranium density) for 5 w/o enriched UO2 in a spherical geometry with a full-water reflector. The minimum critical mass of uranium is shown to occur in the range attained by product powders in some units of facility processing equipment, and the minimum critical water and uranium-water system volume are within the range attained by product powders in some facility handling processes. (The broad minimum water volume is relatively constant for a wide range of UO2 masses and densities.)

The previously described plotted data supports the following maximum subcritical moderation levels in homogenous UO2 at an enrichment of 5 w/o U-235:



UO_2 (5%) Water Reflected Sphere Critical Mass Vs. Water Volume

Figure 1.8.1.13

1.8.1.2 (Continued)

Maximum permitted weight percent water equivalent in an infinite dry UO2 system - weight percent:	1.0
Maximum permitted hydrogen to uranium atomic ratio in an infinite dry UO2 system - H/U:	
- Enrichment ≤4.15 weight percent	0.5
- Enrichment >4.15 weight percent but ≤5.0 weight percent	0.3
Maximum permitted water volume to a dry (moderation controlled), aqueous reflected UO2 system - liters:	19

The minimum quantity of water required for criticality increases as uranium enrichment falls below 5 w/o U-235, and as less than full reflection is provided. This is shown in Figure 1.8.1.15 which was extracted from ARH-600 data. Therefore, controls on water equivalent volumes based on 5 w/o enrichment and full reflection are shown to be conservative at lower enrichments and/or less than full reflection.

Hygroscopicity

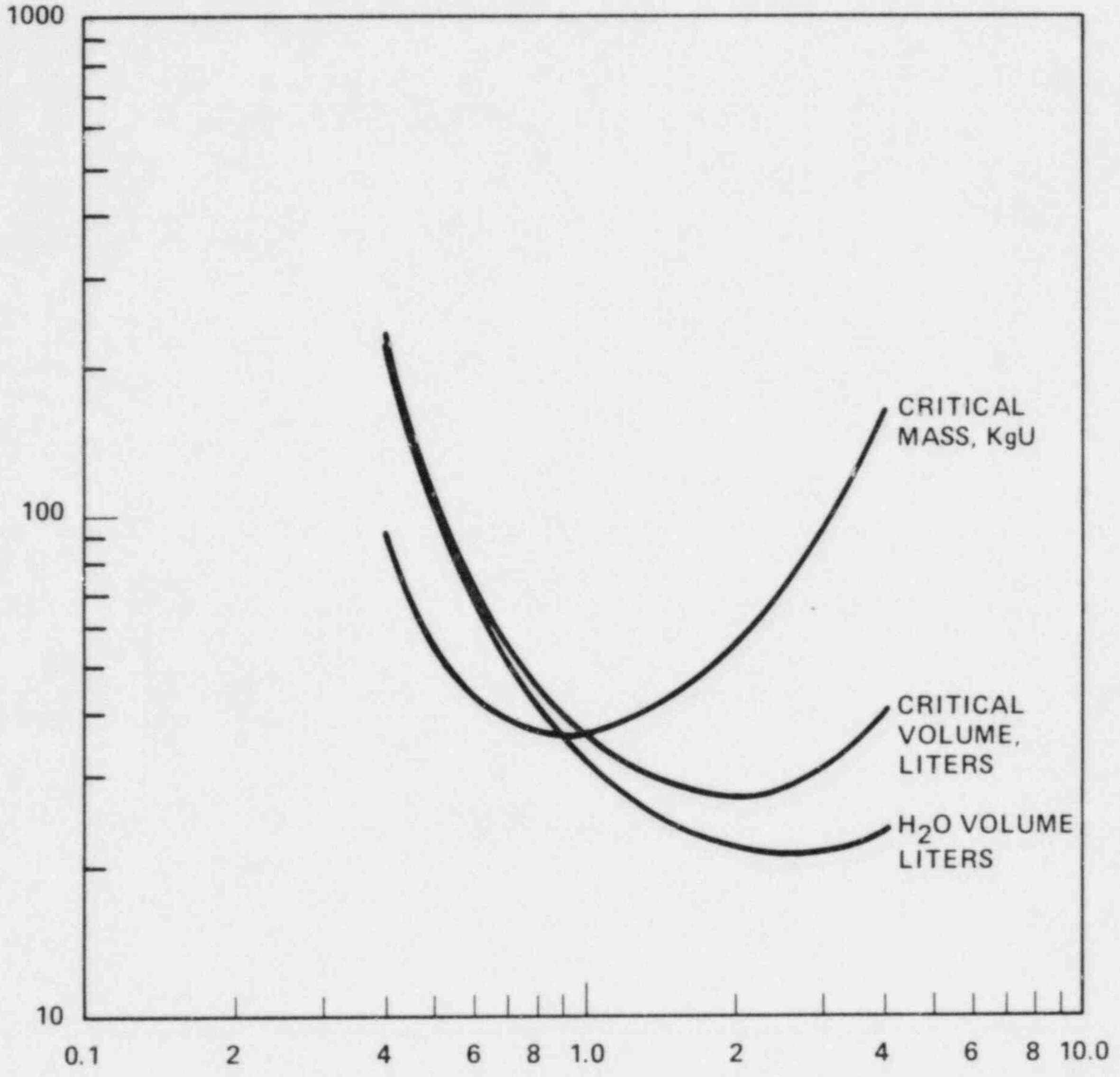
Oxides of uranium are transported and stored in closed and sealed containers. Such containers are equipped with a gasket seal to minimize hygroscopic effects. Studies of hygroscopic effects indicate that oxides of uranium to be used at the Columbia Plant will attain an equilibrium H/U of less than 0.26 when subjected to a 100 per cent relative humidity atmosphere.

Operator Training

All operations/maintenance personnel authorized to perform moderation controlled operations have received training from the Radiation Protection Component with emphasis on the following:

(1) types of moderators and their effects on special nuclear materials,

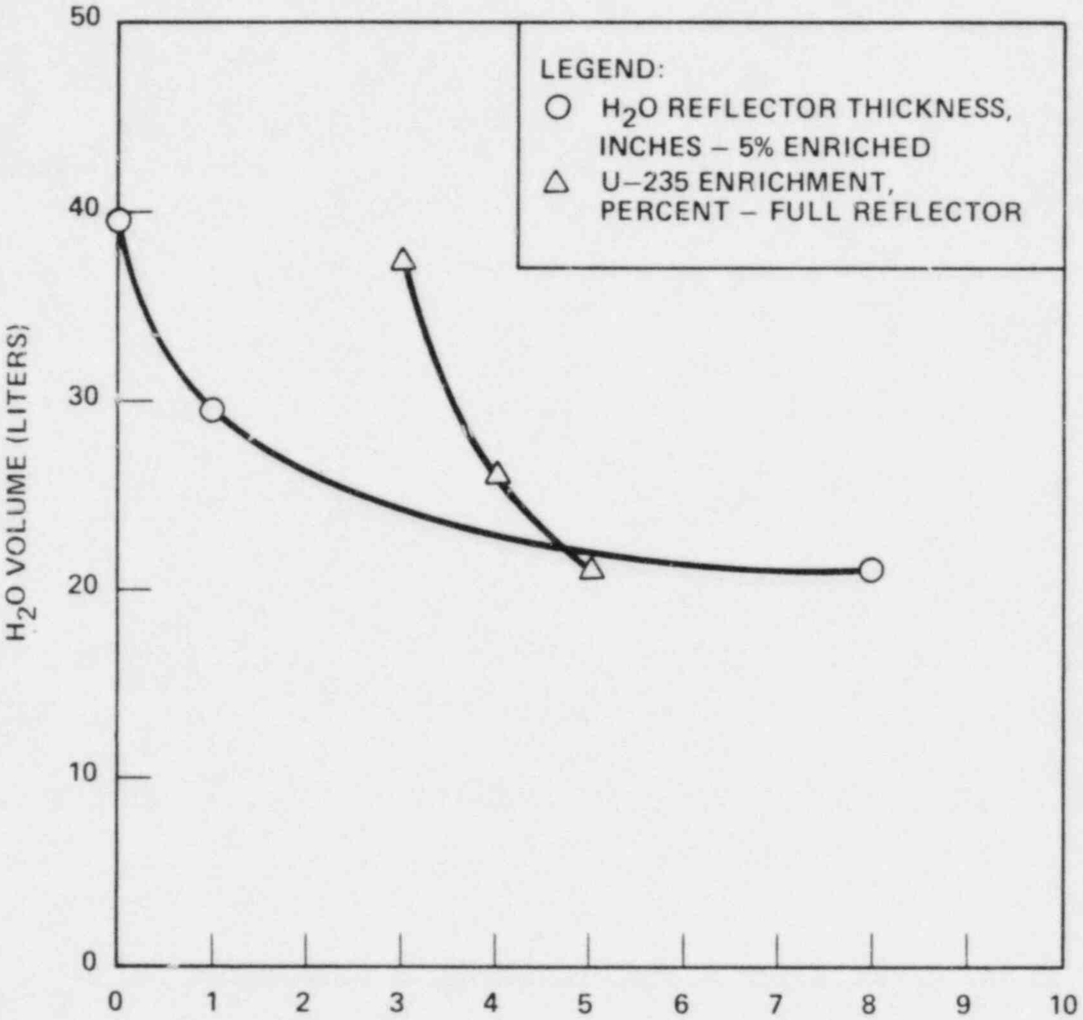
(2) effects of moderator additions to bulk powders,



URANIUM CONCENTRATION, KgU/LITER

Comparison of Critical Data UO_2-H_2O Reflected Sphere

Figure 1.8.1.14



MINIMUM WATER VOLUME VERSUS REFLECTOR THICKNESS AND U-235 ENRICHMENT
Figure 1.8.1.15

1.8.1.2 (Continued)

(3) importance of strict adherence to procedures and nuclear criticality safety parameters,

(4) double contingency aspects of nuclear criticality safety,

(5) spacing requirements,

(6) process controls and their effects on moderation, and

(7) batching controls

1.8.1.3 Nuclear Poisons

Borosilicate glass Raschig rings, when used as the primary control for nuclear criticality safety, are used only in accordance with Regulatory Guide 3.1, "Use of Borosilicate Glass Raschig Rings as a Neutron Absorber in Solutions of Fissile Material." Raschig rings may also be used as a secondary control to provide additional nuclear criticality safety assurance in vessels of uncontrolled geometry where uranium concentrations are normally maintained less than the maximum permissible solution concentrations (5.0 g U-235/liter for aqueous solutions) as the primary nuclear criticality safety control. Typical of such uses would be the addition of Raschig rings into the process quarantine tanks, which are sampled prior to release to the Waste Treatment Facility.

The solutions used in these tanks have a pH <11.0 and a fluoride concentration of approximately 4 per cent as NH_4F.

Twelve years of operating experience at the Columbia Plant have shown that corrosion of the rings is negligible as confirmed by the data given in Table 1.8.1.1.

The following tests are performed annually to determine the various parameters of the Raschig rings. Test results are reported to the Radiation Protection Component for review and evaluation:

TABLE 1.8.1.1

RASCHIG RING ANALYSES FOR QUARANTINE TANKS

WEIGHT PERCENT B_2O_3

	3/1/71	2/15/72	11/15/72	1/23/75	6/21/77	6/5/78	11/13/79	5/20/80	11/17/81	12/10/82
V-116A	12.9		11.45	12.34	13.02	12.0	12.0	12.7	13.0	12.5
V-116B	11.3			12.04	13.06	12.3	12.2	12.8	13.4	12.1
V-116C	10.2		12.90	12.90	12.11	12.1	11.9	12.7	13.0	12.4
V-216A	12.4	12.5	12.49	12.99	12.73	12.1	12.1	12.6	12.8	12.4
V-216B	13.2	12.7		12.80	12.59	12.1	11.9	12.7	13.3	12.1
V-216C	11.4	12.6	11.53	12.52	13.26	12.1	12.2	12.5	13.2	12.7
Unused Ring	10.8	13.4		12.99		12.4				
Unused Ring	10.6									

1.8.1.3 (Continued)

(1) Settling

The levels of rings in the tanks are determined by visual and physical measurements in accordance with written procedure. Rings are added when the required cumulative addition will become equal to 10 per cent of the original loading. Since 1971 settling of the rings has been negligible.

(2) Physical Properties

The volume occupied by the rings is determined by calculating the arithmetic difference between the actual tank volume and the liquid capacity of the tank with the rings. The glass content volume of the tanks is approximately 39 per cent.

(3) Boron Content

Boron content is determined in accordance with Appendix A of ANSI/ANI-8.5-1979. Rings have been sampled and analyzed since 1971, with no significant boron leaching detected. The average boron content of the rings is approximately 12.5 weight per cent B203.

A publication by N. Ketzlach entitled "Proposed Extension of Raschig Ring Standard to Low Enriched Uranium Fuels," concludes that a 24-volume per cent borosilicate glass Raschig ring poisoned mixture (<5 per cent U-235 enriched) cannot be made critical independent of uranium concentration when B203 in the glass is 12.8 per cent. Reducing the B203 content in the glass to 11.8 per cent with 5.4 g U/ml in the matrix solution resulted in an increase of approximately 3.3 per cent in the calculated K infinite compared to the K infinite for 12.8 per cent B203. The average B203 content of rings in the tanks is approximately 12.5 per cent. Consequently, the mixture is safe (K infinite <0.8). Actually, the margin of safety is much greater since the concentration of 5.4 g U/ml is greater than the solubility limit for uranium in

1.8.1.3 (Continued)

solutions. In addition, the average volume occupied by the rings in the existing tanks is approximately 39 volume percent.

The use of any other nuclear poisons or any use of poisons for primary criticality control requires a specific license amendment.

1.8.1.4 Nuclear Interaction Evaluations

All subcrits must meet the MPV's and must have a separation of at least one foot edge-to-edge between any two adjacent subcrits.

Interaction between subcrits which are not isolated (Subparagraph 2.1.1) is evaluated using one or more of three methods: (1) surface density, (2) solid angle and (3) KENO-IV Monte Carlo computer code. The surface density method requires that each subcrit be limited to the nuclear equivalent of a sphere having ≤ 0.3 times the volume of an unreflected critical sphere of the same material (≤ 0.3 "fraction critical"). Values are established for each control using equal buckling conversions as follows:

$$\left(\frac{\pi}{R_s+\lambda_s}\right)^2 = \left(\frac{\pi}{T+2\lambda_t}\right)^2 + \left(\frac{\pi}{L+2\lambda_t}\right)^2 + \left(\frac{\pi}{W+2\lambda_t}\right)^2 \quad \text{for a}$$

slab with finite dimensions, or

$$= \left(\frac{2.405}{R_c+\lambda_c}\right)^2 + \left(\frac{\pi}{H+2\lambda_c}\right)^2 \quad \text{for a cylinder of finite length}$$

Where R_s = the radius of a sphere

T = finite slab thickness

L = finite slab length

W = finite slab width

R_c = finite cylinder radius

H = finite cylinder height

λ_s = 2.1 cm for unreflected spheres

λ_t = 2.8 cm for unreflected slabs

λ_c = 2.5 cm for unreflected cylinders

The MPV for mass is sufficiently low (.435 of reflected

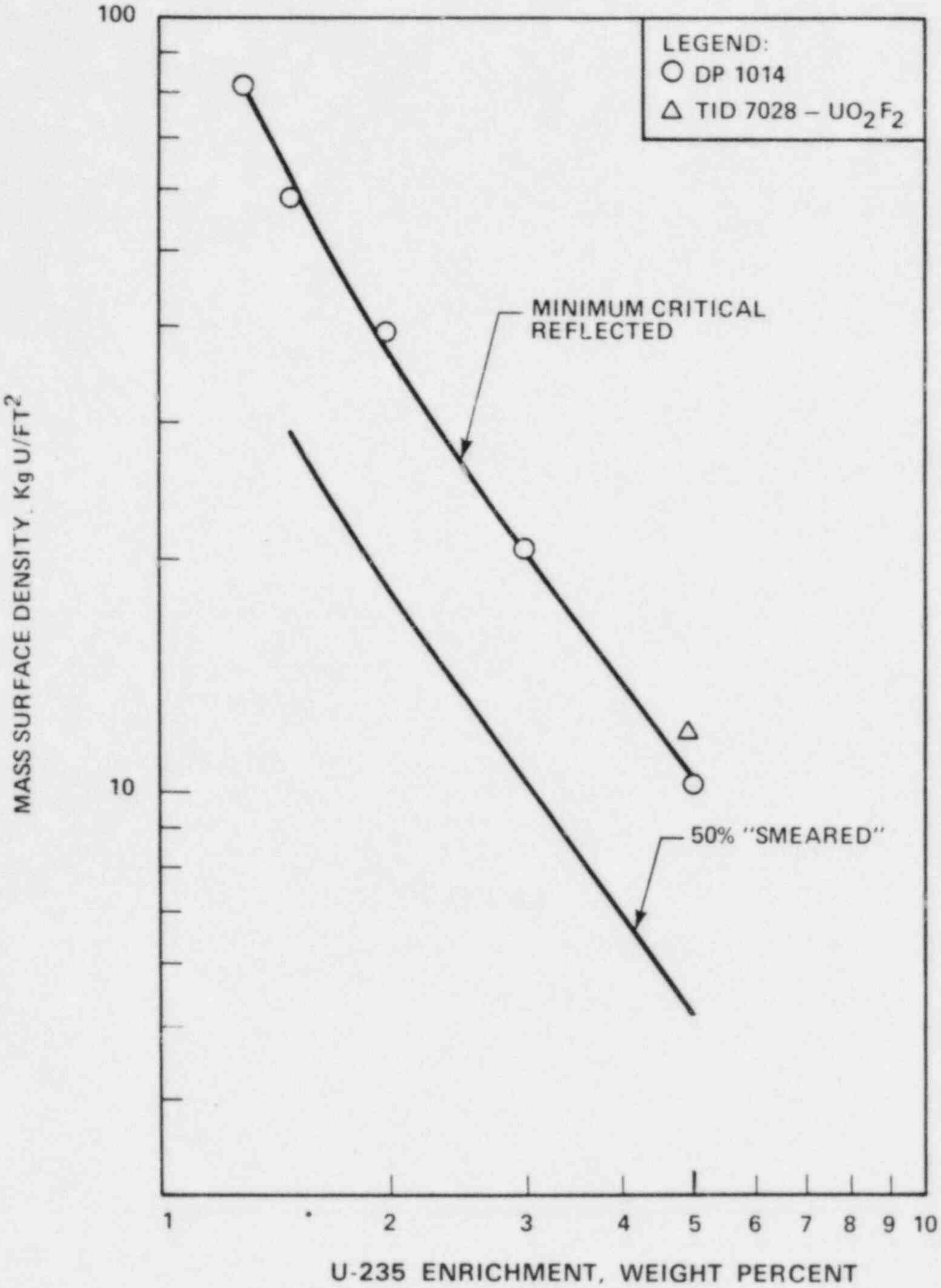
1.8.1.4 (Continued)

critical mass) that it is much less than 0.3 of the unreflected critical mass assuming a 3.9 cm reflector savings for volumes with concentrations above safe concentration limits. The calculated 0.3 fraction critical values for geometry controlled subcrits are shown in Figures 1.8.1.4 through 1.8.1.12.

The subcrits are centered in a space determined by the ratio t_s/t_c where t_s is the "smeared" thickness (the thickness of the slab which would be formed if all the SNM permitted in the subcrit were to flow out over its assigned space) and t_c is the minimum critical reflected infinite slab thickness for the SNM being considered. For a given subcrit the fraction critical value will be limited to $\leq$0.3 and the ratio t_s/t_c will be limited to $\leq$0.25.

For geometry controlled subcrits, the respective fraction critical values are determined on an equal volume basis. That is, the assigned surface density area times the appropriate t_s/t_c ($\leq$0.25) times the minimum critical slab thickness from Figure 2.3.2.8 is $\geq$ the volume of the subcrit. Mass controlled subcrits are spaced such that the average surface density in the assigned area is $\leq$50 per cent of the minimum critical surface density for UO2 (homogeneous or heterogeneous) or UO2 reflected slabs. This is plotted on Figure 1.8.1.16.

If array interaction is evaluated using the solid angle method, the point-to-plane spacing criteria specified in TID-7016, Revision 2, will apply as supplemented by the reflector conditions given in Subparagraph 2.3.2. Alternatively, average solid angles determined from the curves in Section F of Y-1272 may be used for similar and parallel subcrits. Appropriate subcrit Keff's will be determined from the curves given in Figure 2.3.2.10 or Figure 2.3.2.11 dependent upon the form of SNM that is present in the array under evaluation. In any case, the maximum allowable subcrit



Data for Mass Surface Density

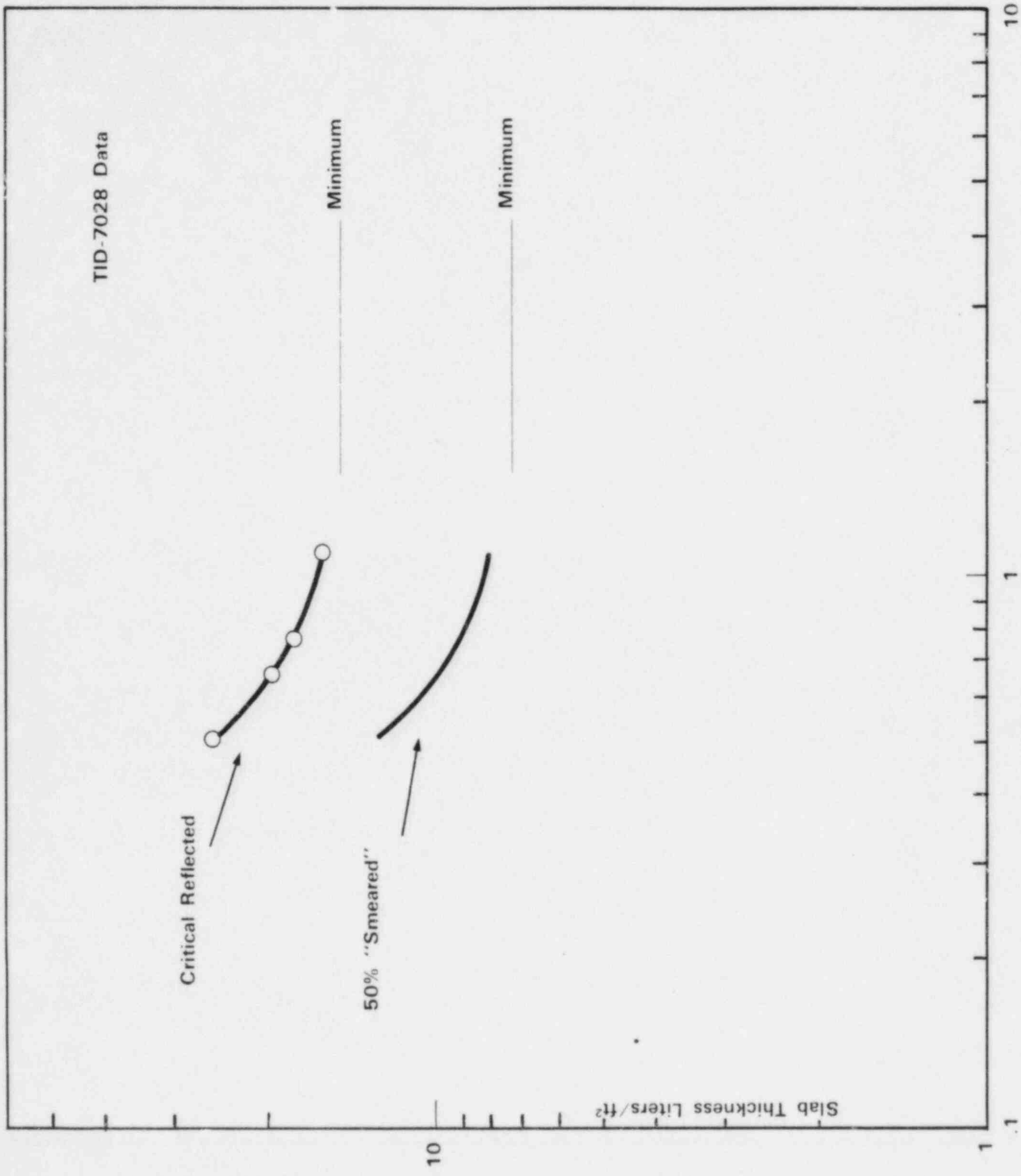
Figure 1.8.1.16

1.8.1.4 (Continued)

multiplication factor (Keff) will be limited to 0.8. When array interaction is evaluated using KENO-IV Monte Carlo computer codes, only validated codes will be used. In each case, the maximum allowable value of Keff + 2 sigma will be limited to $\leq$0.95. In addition, the test cases prepared by the licensee analyst, the calculational results provided by the vendor analyst and the application of those results will be reviewed and approved in accordance with the conditions specified in Subparagraph 2.3.2.

In certain subcrits, such as the UF6 hydrolysis tanks and the tanks which receive the liquid discharged from the centrifuges, the concentration of SNM in the solutions cannot exceed 500 grams U/liter. When it has been established that the concentration of the solution in a subcrit cannot credibly exceed 500 g U/liter, the minimum critical slab thickness used to determine surface density spacing requirements for that subcrit is that of a UO2F2 slab at a concentration of 500 g U/liter and a uranium enrichment of 4.9 w/o. Data for UO2F2 at this enrichment is plotted in Figure 1.8.1.17. The specified minimum critical slab thickness is conservative when applied to solutions having lower uranium concentrations or for solutions containing uranium at lower enrichments. This choice of minimum critical slab thickness for spacing purposes has no relationship to the establishment of the subcrit MPV as described in Subparagraph 1.8.1.1.

Moderation controlled units (such as UF6 cylinders, closed UO2 containers and equipment, and fuel assemblies) and poisoned systems are excluded from interaction considerations provided that they are not located in an area designated by the surface density method for another subcrit and are located at least one foot from other SNM. In addition, it is demonstrated that moderation control is maintained under both normal and accident conditions.



$U^{(4.9)}O_2F_2$ CONCENTRATION SLAB DATA
Figure 1.8.1.17

1.8.1.4 (Continued)

Transfer piping $\leq$2 inches in diameter and safe diameter vent lines are neglected in the interaction calculations.

1.8.1.5 Fuel Assemblies and Rods

A safe slab thickness for close-packed clad UO2 fuel rod storage has been calculated by the NFD Nuclear Engineering Department. These calculations were conservatively based on a square lattice. K infinite for a triangular lattice was less than 0.95 for zircaloy clad rods containing uranium enriched $\leq$5.0 w/o. The maximum permissible slab thickness (0.84 of the minimum water moderated and reflected critical slab thickness) is 32 inches for UO2 rods with an enrichment $\leq$3.0 w/o, 20 inches for UO2 rods with an enrichment $\leq$4.0 w/o, and 15 inches for UO2 rods with an enrichment $\leq$5.0 w/o. The unreflected critical slab thickness is (Max. Permissible Thickness/.84) + (2 X 3.9/2.54). This equals 41.2 inches for enrichments $\leq$3.0 w/o, 26.9 inches for enrichments $\leq$4.0 w/o, and 20.9 inches for enrichments $\leq$5.0 w/o. The "0.3 fraction critical" limit for use with the surface density spacing of moderated subcrits is conservatively determined using 0.3 times this critical unreflected slab thickness of 12.4 inches, 8.1 inches, and 6.3 inches respectively, for $\leq$3.0 w/o, $\leq$4.0 w/o and $\leq$5.0 w/o U-235 enrichment. Similarly, the 25 per cent of reflected critical thickness for surface density spacing is 9.5 inches, 6 inches, and 4.5 inches, respectively, for $\leq$3.0 w/o, $\leq$4.0 w/o and $\leq$5.0 w/o U-235 enrichment. The maximum permissible slabs are separated by at least one foot edge-to-edge. The arrays, which are dry under normal conditions and would not retain water, are not considered in interaction evaluations.

The maximum fuel assembly keff is 0.95. Fuel assembly keff's are calculated assuming optimum moderation and full reflection by light water, since assemblies have consistently been designed to be undermoderated when flooded. The calculations

1.8.1.5 (Continued)

would assume optimum moderation when such moderation became physically possible. Fuel assemblies are separated by at least one foot edge-to-edge and will not retain water. Interaction with other assemblies is, therefore, not considered.

1.8.1.6 References

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3. Clark, H. K., AEC Research and Development Report, Critical and Safe Masses and Dimensions of Lattices of U and UO2 Rods in Water, DP-1014, Savannah River Laboratory, February 1966.
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5. Newlon, C. E., AEC Research and Development Report, Minimum Critical Cylinder Diameters of Hydrogen Moderated U (4.9) System, K-1629, Union Carbide Corporation, ORGDP, March 15, 1965.
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8. Standard Shipping Container for 30-inch Diameter UF6 Cylinders, K-1920, Union Carbide Corp., ORGDP, July 20, 1966.

1.8.1.6 (Continued)

9. Paxton, H. C., Criticality Control in Operations with Fissile Material, LA-3366, Los Alamos Scientific Laboratory, 1966

10. Wachter, J. W., Editor, AEC Research and Development Report, Y-12 Plant Nuclear Safety Handbook, Y-1272, Union Carbide Corp., July 19, 1963.

11. Carter, R. D., Criticality Handbook, ARH-600, Revision 1, May 23, 1969.

12. Thomas, W., et al, 6th Supplement to Handbook der Kritikalitat, January 1975.

1.8.2 Nuclear Criticality Safety Administration

The principle staff component contributing to the administration of radiation protection and nuclear criticality safety in the NFD Manufacturing Department is the Materials Department. This group inherently is involved in the early stages of the planning for all changes which may occur on the production floor. It has been assigned responsibility for negotiating all required licensing authorizations, and thus is thoroughly cognizant of NRC regulatory requirements. The Radiation Protection Component (Figure 1.4.1.1) is assigned to this department.

In practice, any proposed modification of, or addition to, existing SNM processing, handling or storage equipment or operations, is referred directly to the Radiation Protection Component for review and approval prior to its adoption on the floor. Included in this review are those changes which could affect radiation protection , nuclear criticality safety or SNM safeguards.

Changes which could affect radiation protection are reviewed during the conceptual or initial design stage by one of the site Health Physicists. This individual establishes the radiation and contamination control criteria that are

1.8.2 (Continued)

applicable to the proposed change in accordance with NRC regulations, license requirements and good health physics practices. In addition, he specifies any engineered or administrative controls that are required to ensure full regulatory compliance. Operations conducted subsequent to changes and modifications are subject to review and surveillance by Radiation Protection Component personnel to confirm successful operation.

Changes which could affect nuclear criticality safety are reviewed during the conceptual or initial design stage by the site Criticality Engineer. This individual ascertains that the proposed change conforms with NRC regulations, license requirements and good nuclear criticality control practices. He selects appropriate MPV's from Subparagraph 2.3.2 and determines the individual unit and/or array spacing requirements. He specifies any engineered or administrative controls that are required to ensure full regulatory compliance. If required, the Criticality Engineer obtains additional or specific data on fuel assembly reactivity values from an individual with qualifications as defined in Subparagraph 3.3.7. Operations conducted subsequent to changes and modifications are subject to review and surveillance by Radiation Protection Component personnel to confirm successful operation.

NFD Nuclear Engineering has established a routine procedure which requires that all transmittals to other operations which contain nuclear criticality safety data must be in writing and must be reviewed and approved by at least one level of supervision before it is transmitted.

All radiation protection and nuclear criticality safety requirements, including MPV's and spacing data, are transmitted to a Radiation Protection Component representative

1.8.2 (Continued)

who reviews and approves their application to the proposed change. Records of approved changes, plus any supporting data and information, are maintained for at least one year. Operations conducted subsequent to changes and modifications are subject to review and surveillance by Radiation Protection Component personnel to confirm successful operation.

The cognizant shift supervisor is responsible to assure that radiation protection requirements, MPV's and spacing criteria are supplied to him as pa[illegible] of the information supplied by the Radiation Protection Component. The MPV's and spacing criteria are posted and the shift supervisor uses any technique he considers appropriate to assure that employes under his supervision are made aware of each change. Observed violations are reported to the cognizant shift supervisor, who is responsible to take immediate steps to correct them. Repeated or continuing violations are reported to the Manager of the Radiation Protection Component, who in turn reports them to higher levels of management for appropriate action.

1.8.3. <u>Nuclear Criticality Safety Equipment</u>

1.8.3.1 <u>Mechanical Design Criteria</u>

In recognition of the importance of the sound mechanical design of isolation structures, racks, lifting fixtures, etc., to general plant safety as well as to nuclear criticality safety, an established procedure exists which requires that all such devices be designed with a minimum safety factor of 3 to 1, unless otherwise specfically approved by the Manager, Safety. Lifting devices are reviewed and approved at the design stage by the Manager, Safety. The devices are inspected on receipt.

The general condition of structures, lifting devices, isolation media, etc., is routinely audited during the plant safety tours made by Safety Department personnel.

1.8.3.2 Immediate Evacuation Signal System

The nuclear criticality alarm system which initiates immediate evacuation of the facility is described in the Site Emergency Plan. Employes are trained in recognizing the evacuation signal as a continuous sounding siren.

Radiation monitoring units are located to assure compliance with the requirements of 10 CFR 70.24. Insofar as possible, the location and spacing of the detectors is chosen to avoid the effect of shielding by massive equipment or materials. Low density materials of construction such as 2 x 4 stud construction walls, plaster or metal corrugated panels, asbestos panels, doors, panel walls, and steel office partitions are disregarded in determining the spacing. The spacing is reduced where high density building materials such as brick, concrete, or concrete or cinder blocks shield a potential accident area from the detector. The distance, X, equivalent to 120 feet for unshielded detectors is determined from the relation $X = 120\ T^{1/2}$ where T is the transmission through the shielding. Where more than one wall shields the source, the value of T is the product of the transmission factors for each of the intervening walls.

The transmission is determined from curves for cobalt-60 in NBS Handbook 73, "Protection Against Radiation from sealed Gamma Sources," or meaured using a cobalt-60 source. A constant transmission factor, based on normal incidence, is assumed for each wall and is used for all areas shielded by that wall.

1.9 Processing Operations

The processing operations to be performed on radioactive materials under this license may be divided into a number of distinct categories. The first category is the ADU and IDR conversion operations which employ chemical means to convert UF6 to uranium oxide powder. The second category is the

1.9 (Continued)

fabrication operations which use essentially mechanical processes to produce fuel assemblies containing encapsulated UO_2 pellets.

Another category is the analytical operations which use a variety of spectrographic and wet chemical operations on small samples of material to assure that material specifications are met.

Still another category is the treatment of scrap to permit it to be recycled into production operations or more closely controlled prior to discarding. This treatment includes chemical dissolution and precipitation, solvent extraction and dry processes.

A fifth category is the incineration of combustible contaminated wastes. This operation greatly reduces the volume of waste materials to be disposed of by burial, and it permits increased accuracy in the measurements made to determine the nature and quantity of SNM being discarded.

Figure 1.9.0.1 shows the general arrangement of the various areas in the Columbia Facility.

1.9.1 Conversion Operations (ADU Lines)

Conversion Lines 1 through 5 are of the ADU process type for converting UF_6 or uranyl nitrate to uranium oxide. This involves the continuous processing of solutions and slurries of SNM. These lines are established on a geometry controlled basis for uranium with a maximum enrichment of 4.15 w/o. Line flow rate is up to 200 pounds of uranium per hour. Future changes to the process will be spaced using the surface density criteria given in Subparagraph 1.8.1.4.

The spacings employed for Lines 1 and 2 are the "tightest," or closest together. The components of subsequent lines using the ADU process were spaced further apart, primarily to simplify access for routine operation and maintenance. The

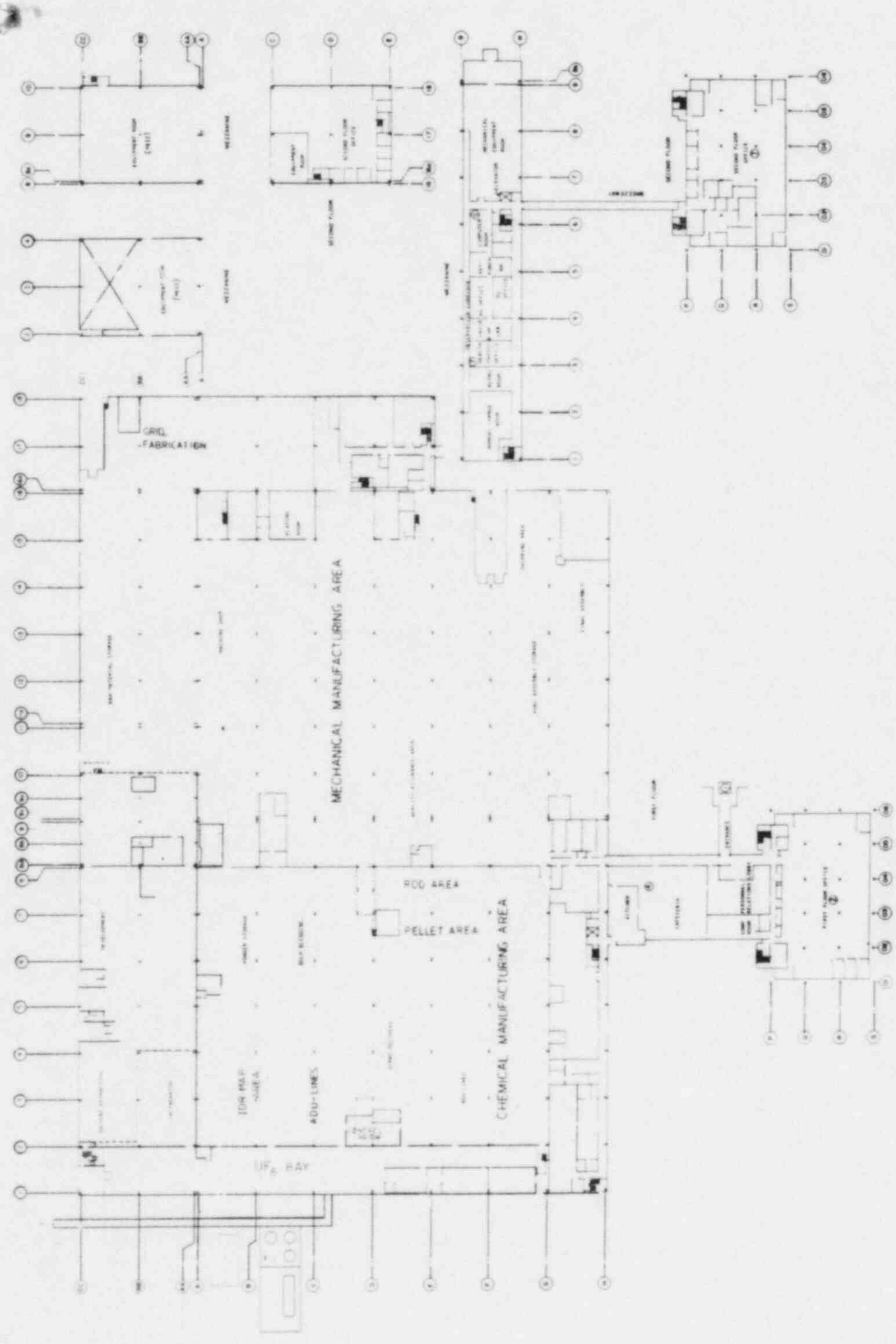


Figure 1.9.0.1 PLANT LAYOUT

1.9.1 (Continued)

spacings for all lines were determined by permissible surface density criteria for 4.15 w/o uranium-235. As a check on the use of those parameters, the spacings have been verified by the solid angle method.

The mechanical design of the system has been considered under normal and abnormal operating conditions, including consideration of drainage from safe to unsafe geometries. The system was tested with water and depleted uranium, and was found to be safe.

The U-235 enrichment is verified at withdrawal at the enrichment plant and in the Westinghouse Analytical Laboratory prior to processing the material. The maximum U-235 enrichment shipped in Model 30A and 30B cylinders is 5 w/o. If 5 w/o enriched UF6 were accidentally mislabelled, shipped, received and processed as 4.15 w/o material the conversion line would remain subcritical under normal operating conditions. This is based on a comparison of the MPV's for 4.15 w/o solutions and homogeneous oxides with the respective critical values for 5 w/o solutions and homogeneous oxides. The line is primarily geometrically controlled, with solution and homogeneous oxide MPV's for cylinder diameter of 11.0 and 10.4 inches, respectively.

1.9.1.1 UF6 Receipt and Storage (ADU and IDR Lines)

UF6 is received in Model 30A or 30B cylinders in NRC and DOT authorized packagings. Upon arrival, the packages are unloaded after survey by the Radiation Protection Component. These moderation controlled cylinders of UF6 are then removed from the outer packages, weighed and transferred to the outside UF6 Storage Pad for storage. Here the cylinders are stored horizontally on wooden chocks or in protective overpaks with a minimum of 1 foot edge-to-edge spacing. Cylinders of UF6 are transferred from the UF6 Storage Pad to the

1.9.1.1 (Continued)

Manufacturing Building UF6 Bay, upended and stored vertically in fixed steel racks prior to processing. These racks also provide a minimum of 1 foot edge-to-edge spacing.

Combustible materials are kept at a minimum in these areas. Inside transfers are made by overhead crane and outside transfers are made using an appropriate transport vehicle.

Low enrichment uranium in sealed cylinders requires only routine radiation protection precautions. The cylinders are resurveyed (contamination and direct radiation) after removal of the UF6.

1.9.1.2 <u>Vaporization and Hydrolysis</u>

After release by Quality Control, each cylinder containing UF6 is transferred vertically by crane from the storage rack and loaded into a vaporizer (steel steam chest) designed to enclose the cylinder and all connections to the conversion system. Lids on all vaporizers are equipped with pneumatically operated, manually controlled closure systems. Lids are equipped with gaskets and 6-12 bolts to insure leak tightness after closure. Each vaporizer is also equipped with a pressure relief valve. A flexible pigtail is connected to the cylinder valve to permit the flow of UF6 from the cylinder to the conversion processing system. Operations personnel are required to wear protective clothing and equipment such as coveralls, shoe covers, rubber gloves and full-face shields while the connection is being made. In addition, a ventilated tent, which is an integral part of the conversion area air effluent treatment system, is used. All connections are pressure tested to assure leak tightness. Externally controlled cylinder valve "operators" are provided for opening and closing the cylinder valves during conversion line operations. A quick-quench carbon dioxide system is available to reduce the temperature in the event of abnormal conditions.

1.9.1.2 (Continued)

The system consists of a movable cart loaded with two liquid carbon dioxide bottles connected in parallel and equipped with a quick-disconnect line for connection to the steam chest lid. The lid is designed so that the carbon dioxide will be directed toward the UF6 cylinder valve and pigtail connections. Spare carbon dioxide cylinders are provided.

An accumulation of condensate in the bottom of the vaporizer would be detected by a liquid level detector, which would actuate an alarm in the Control Room.

UF6 leakage into the vaporizer would be detected by an electrical conductivity cell in the steam condensate line, which would actuate an alarm in the Control Room, and automatically shut off the steam supply and the condensate flow lines. A manually operated valve in the bottom of the varporizer permits the drainage of the condensate to a nuclearly safe sump. This U-shaped sump in the east sump is enclosed in concrete which provides nuclear isolation. The vertical open ends are 10 inches in diameter and the horizontal bottom connecting section is 8 inches in diameter. The sump in the west trench is a 10.5-inch I.D. pipe which is a safe diameter.

The increased reflector savings provided by the close-fitting, thick concrete is estimated from Figure 46 of TID-7028. These data indicate an increase in reflector savings of 1 cm. Decreasing the critical water reflected cylinder radius values by this 1 cm results in an estimated critical concrete reflected diameter of 11.2 inches (12.0 inches - 0.8 inch). Applying a 1.1 safety factor results in an MPV of 10.2 inches. Then applying the 0.92 reduction factor, the permissible reduced diameter for ells is 9.4 inches. The equivalent diameter of the 10 inch - 8 inch pipe intersection

1.9.1.2 (Continued)

is 9.1 inches, which satisfies the reduced diameter criteria for concrete reflected ells.

Vaporized UF6 flows through installed corrosion-resistant piping to a vertical 10.02-inch diameter x 16-foot, polypropylene or teflon lined, steel hydrolysis tower. The hydrolysis tower consists of a water reservoir section (maintained on level control) at the bottom, a void section with a UF6 entrance nozzle located approximately 18 inches above the water level, a polypropylene tellerette-packed contact region located above the UF6 entrance nozzle, and a water spray section at the top of the tower. The hydrolysis tank is vented and thus provides a siphon break to the UF6 cylinder. In addition, manual block valves (two valves in series) are provided to isolate the UF6 cylinder from the main gas header. A positive flow of N2 is maintained, by procedure, through the main header when the UF6 flow is shut off. Thus, the siphoning of water from the hydrolysis tower to the UF6 cylinder is precluded.

The hydrolysis tower is vented to the Conversion Area air effluent treatment system which consists of two treatment stages, water scrubbing and HEPA filtration. In the hydrolysis tower, UF6 is contacted with water to produce a uranyl fluoride (UO2F2) solution containing hydrofluoric acid (HF). The relative quantities of UF6 and water are accurately metered to produce a UO2F2 solution which is closely controlled about a specified uranium concentration (approximately 160 gm U/liter). Control of the concentration is maintained by sampling and analysis for specific gravity. Attaining a concentration of 500 grams U/liter is not credible since UF6 crystallization and subsequent plugging would occur. The 10.02-inch cylinder diameter is nuclearly safe.

1.9.1.2 (Continued)

From the hydrolysis column, the UO2F2 is partially neutralized with ammonium hydroxide.

"Heels" are removed from the UF6 cylinders by educting to the hydrolysis tower with N2 or UF6 from a full cylinder. Alternatively, cylinders may be evacuated using a "cold trap" vacuum system.

Leaks in the system outside the vaporizer are controlled by valving off the line. Most leaks within the vaporizer would be contained and would be stopped with the CO2 system. A gross leak within the chest could cause a release through the pressure relief valve, to the UF6 Bay. The release would stop when the pressure dropped below the relief pressure due to vaporization cooling. In this event, the entire Controlled Area would be evacuated and the HEPA filtered emergency ventilation system for the UF6 Bay would be activated. The release would normally be confined within the Controlled Area which is designed as containment.

Leaks within the vaporizer, which are assumed to clog the steam trap and result in the accumulation of solution containing SNM in the bottom of the chest, would activate the conductivity cell and liquid level detector. Activation of the conductivity cell would cause an audible alarm and shut off the steam supply and condensate flows. The conductivity cell and liquid level detector settings provide ample margin to allow for any level increase resulting from the condensation of that steam present in the chest when the supply line valve is closed. A manually operated valve in the bottom of the vaporizer permits the drainage of the condensate to the nuclearly safe sump.

Local ventilated tents, capable of handling releases normally encountered, are provided for use during connecting and

1.9.1.2 (Continued)

disconnecting operations on the UF6 systems. These systems are an integral part of the Conversion Area air effluent treatment system.

UF6 cylinders containing any significant quantities of UF6 are normally transferred with the valve cover in place. In the event that it is not practical to attach a valve cover to the cylinder, they may be transported without the valve covers in accordance with Subparagraph 2.4.4.

Low enrichment uranium in a closed system requires only routine radiation protection precautions. The precautions specified above will effectively minimize the possibility of a significant material release.

1.9.1.3 Precipitation and Centrifugation

The uranyl fluoride solution (UO2F2) is converted to ammonium diuranate (ADU) by reacting the UO2F2 with ammonium hydroxide in the precipitation tank where the ADU precipitates. The vertical precipitator tanks are 10 inches in diameter by approximately 6 feet tall and the liquid is maintained at a a constant depth by a level controller. The 10-inch diameter is nuclearly safe.

A 10-inch diameter by approximately 20 feet high tank is used as a nitric acid feed tank for acid washing of vessels and as a uranyl nitrate feed tank when uranyl nitrate is used as feed material to the ADU lines. The 10-inch diameter is nuclearly safe.

From the precipitation tanks, the slurry is pumped to the first stage clarifier. This unit is a centrifuge with a rapidly revolving horizontal bowl in which the solids are separated by centrifugal force and are removed from the unit by a conveyor screw that turns at a speed slightly faster than the bowl. The solids discharge to the ADU receiver tank and the centrate flows by gravity into the second stage clarifier

1.9.1.3 (Continued)

feed tank. The centrifuge is located on a platform above the floor.

The centrifuge is essentially a horizontal rotating cylinder and consists of an inlet along the central axis, a discharge screw surrounding the inlet, and an outer "bowl" concentric with the inlet. The outer bowl has a section 19.6 inches long which has a 14-inch inside diameter. The prolongation of this section is 12 inches long and tapers from 14 inches I.D. to about 9 3/4 inches I.D. The discharge screw extends the full length of the "bowl," tapering to suit. The slurry enters along the axis and is flung outward, becoming concentrated along the outer wall. The discharge screw forces the thickened slurry to the tapered section where it is forced through outlet holes into the exit chute. The concentrate overflow slots at the full size end of the bowl extend to within 1 inch of the wall, limiting any slurry buildup to an annular region with a 14-inch O.D., a 12-inch I. D., and extending the length of the bowl. The bowl housing and material exits are designed to prevent buildup and provide a free gravity flow of material.

The total volume of the central section is 4.4 liters and the maximum operating volume is conservatively estimated as the total volume of the central section plus a 1-inch annular region with a 14-inch outside diameter by the full length of the centrifuge bowl or 4.4 plus 21.2 liters. Thus the maximum operating volume is approximately 26 liters although the total volume in the bowl is 54 liters. The dewatered ADU is sufficiently fluid to flow out of the various outlets if the centrifuge stops, so that rotation is necessary to retain a volume approaching the size of the maximum operating volume. The centrifuge feed control valve will automatically close if

1.9.1.3 (Continued)

the drive motor power is interrupted, and the bowl will be flushed with water for one to two minutes while the bowl coasts to a stop.

The ADU is discharged through the centrifuge discharge chute to the ADU receiver tank which is a 4.75-inch thick slab tank. The centrate flows by gravity through the exit chute and a 4-inch diameter pipe to the second stage clarifier feed tank. The transition sections at the bottom of the exit chutes are ventilated and if these collection tanks overflowed, the material would back up in the vent drain tank which has a liquid alarm. However, the collection tanks are operated on level control.

The 26 liter maximum operating volume of the centrifuge is a safe volume and significant material holdup in the centrifuge bowl, housing, or exit chutes is not credible because of the elevation and gravity flow design.

The top of the ADU receiver tank is about 2 1/2 feet below the centrifuge bowl. The 4.75-inch slab thickness is nuclearly safe.

The vertical vent drain tank is a nuclearly safe 10.02-inch diameter by 7.0-foot cylinder. The tank has a liquid level alarm and is drained on a regular basis. The cleanout port flange is equipped with a plug to maintain the 10.02-inch diameter at this point.

The centrate from the first stage clarifier flows by gravity into the second stage clarifier feed tank, a vertical 10.02-inch diameter by 5.0-foot tank, which operates on level control. The 10.02-inch cylinder diameter is nuclearly safe.

The centrate is then pumped to the second stage centrifuge. The total volume of the centrifuge bowl, sludge holding space and sludge catcher is 30 liters, which is a nuclearly safe volume. The underflow, containing the solids and centrifuge

1.9.1.3 (Continued)

overflow are recycled to the first stage clarifier inlet. The centrate waste solution flows to the liquid waste treatment and disposal system via the on-line monitor.

1.9.1.4 Drying

Wet ADU slurry is pumped from the receiver tank to a 10.25-inch diameter, surface heated dryer. The ADU is conveyed through the dryer by rotating paddles where the water content is reduced from approximately 50 per cent to 5 per cent. The ADU is then discharged from the dryer into a 4-inch thick, slab type bucket elevator transfer system.

Off-gases from the dryer pass through 9.9-inch equivalent diameter (enclosure) bag filters prior to entering a 10-inch diameter vapor condenser where condensibles and entrained particles are removed from the filtered gases. Both the condensate and the noncondensible vapors from the condenser are discharged into the first stage clarifier feed tank which is vented to the Conversion Area air effluent treatment system. The 9.9-inch, 10.25-inch and 10-inch diameters for the bag filter enclosure, the dryer, and the condenser respectively are nuclearly safe diameters for solutions. The bag filter enclosure is connected to one end of, and is perpendicular to the dryer. This intersection satisfies the reduced effective diameter requirement of 10.21 inches for ells associated with solution systems. The 4-inch slab thickness of the bucket elevator transfer system is also a nuclearly safe thickness for solutions.

Low enriched uranium in a closed system requires only routine radiation protection precautions.

1.9.1.5 Calcining

A motor driven auger conveys dried ADU, into a 10-inch diameter, rotating, calcining furnace. The ADU decomposes to U308 and is subsequently reduced to UO2. The furnace empties

1.9.1.5 (Continued

into a 7.5-inch equivalent diameter discharge screw conveyor. An inspection sight port on the product end of the furnace is used to check the condition of the product feed stream.

The horizontal 10-inch diameter by 26-foot, 4-inch calciner furnace is a nuclearly safe diameter for homogeneous oxides.

The furnace uses a reducing atmosphere. Off-gases pass through a water scrubber and are vented to the Conversion Area air effluent treatment system. The scrubber reservoir is a slab tank with a nuclearly safe thickness of 4.75 inches. It is separated from other conversion process vessels by 12 inches of concrete on either side. The increased reflector savings provided by the thick concrete is estimated from Figure 46 of TID-7028. These data indicate an increase in reflector savings of 2.2 cm. Decreasing the critical water reflected slab thickness value for 4.15 w/o solutions by this 2.2 cm. results in an estimated critical concrete-reflected slab thickness of 5.9 inches (6.8 inches - 0.9 inch).

Applying a 1.2 safety factor results in an MPV of 4.9 inches for 4.15 w/o solutions, which is greater than the actual value of 4.75 inches. The scrubber water is filtered to remove any UO2, monitored and then pumped to the Waste Treatment Facility. Low enrichment uranium in a closed system requires only routine radiation protection precautions.

1.9.1.6 Comminution

Dry uranium oxide powder is conveyed from the calciner exit chute, through a milling operation to achieve the specified particle properties, and into polypak drums. These containers are sampled, closed, identified, decontaminated and placed onto movable storage carts. Product samples are composited and sent to the Analytical Laboratory for analysis to verify dryness and other specification requirements.

1.9.1.6 (Continued)

The discharge screw conveyor, which has an external water cooling jacket, consists of two 5-foot, 9-inch long conveyors, one below and perpendicular to the other, satisfying the 12-inch edge-to-edge and 3 diameters axis-to-axis spacing requirement and connected at one point with a 7.1-inch equivalent diameter gravity flow section. The complete system satisfies the reduced diameter requirements for ells and tees (9.4 inches and 8.6 inches) for homogeneous oxide systems. A 6-inch diameter section connects one end of the bottom screw conveyor loop to a 6-inch vane feeder. The connecting hoppers, feed chutes, and hammer mill form a slab type gravity flow arrangement with a maximum cross section of 5 1/2 x 15 1/2 inches which is equal in cross sectional area to the MPV for homogeneous oxides. In addition, the material in this section is dry and would not contribute to nuclear interaction. The mill and associated conveyors form a closed system which is maintained at a negative pressure relative to its surroundings. The system and ventilation exhausts are passed through HEPA filters before being discharged.

1.9.1.7 <u>Blending</u>

Dry uranium oxide powder is normally blended to achieve uniform physical and chemical characteristics as required by product or process specifications. The blending system consists of a powder feed hopper contained inside a ventilation hood equipped with a HEPA filter, transfer system, dispensing hopper with a discharge screw, and an electrically powered, oblong shaped carousel for automatically conveying the collection containers under the discharge screw. A specially designed containment and ventilation system equipped with HEPA filters is provided for the carousel.

Although the operation only functions adequately when using dry powder, the feed and dispensing hoppers are each operated

1.9.1.7 (Continued)

on a nuclearly safe mass basis and are also geometrically controlled by their 10-inch diameter which is nuclearly safe for homogeneous oxides.

Collection containers, 10 inches long X 10 inches wide X 7 inches deep, travel end-to-end on the carousel, a conveyor, forming a horizontal oval approximately 6 feet X 45 feet. When loaded on the carousel the containers form the equivalent of two horizontal, parallel cylinders connected together at each end by two "ells" in series. The cross-sectional area of each "equivalent cylinder" is equal to that of a cylinder with its diameter reduced in accordance with the pipe intersection criteria for "ells" described in Subparagraph 1.8.1.1.

In addition, dry uranium oxide powder may also be blended to adjust the enrichment of the uranium to permit greater utilization of all licensed SNM. Normally, only small amounts of excess material are processed in this manner. These blending operations are performed in electrically powered, rotating blenders also provided with containment and ventilation systems equipped with HEPA filters. The operations are administratively controlled on a nuclearly safe mass basis as specified in Figure 2.3.2.1.

1.9.1.8 Storage of Uranium Oxide Powder

The dry uranium oxide powder is stored in closed polypak containers on movable powder storage carts of all-steel construction. The storage carts are designed with a minimum safety factor of 4 and without any water retention capability. Each 4-foot X 6-foot rack provides conveyor type storage for 8 continuous parallel, nearly horizontal, cylinders of polypaks (72 inches long) in a 2 wide X 4 high array. The arrangement provides at least 12 inches of spacing between cylinders. Carts are stored side by side to form continuous arrays.

1.9.1.8 (Continued)

The 9 1/2-inch inside diameter of the polypaks satisfies the MPV for homogeneous oxide cylinders and since the uranium oxide is dry and in closed containers, nuclear interaction is not considered.

Low enrichment uranium in closed packages requires only routine radiation protection precautions.

1.9.1.9 Process Waste Treatment

The aqueous waste stream from the second stage clarifier is collected in a tank. This vertical tank is 10 inches in diameter by 23 feet-5 inches tall. The 10-inch diameter is nuclearly safe for solutions. Solutions entering this tank have been through at least two stages of clarification and one stage of filtration, so it is not credible for the concentration to exceed 500 gm U/liter.

From this system, the waste solution is pumped through a $\leq$10-inch diameter cartridge filter, and to the final filtration tanks in a semicontinuous operation.

After passing through the second stage clarifiers, waste solutions from all ADU conversion lines are individually monitored through the programmed radioactivity monitor and then are combined in a common header and pumped to the quarantine tank system. The quarantine tank system consists of two (primary and final) on-line radioactivity monitors, two final filtration tanks, two pump-out tanks and two diversion tanks. Each on-line radioactivity monitor consists of a one-liter, flow through sample container, well scintillation detector, lead shield and single channel analyzer. The final filtration tanks, diversion tanks and pump out tanks are equipped with cartridge filters for removing trace amounts of insoluble uranium from the waste streams.

The programmed on-line radioactivity monitor sequentially measures the uranium concentration in the waste stream from

1.9.1.9 (Continued)

each ADU conversion line prior to combination in the header. This is accomplished by means of a timer and stepping switch and an air operated valve in each waste stream line for routing the waste stream through the monitor.

The combined waste streams are first pumped to the final filtration tanks where they are circulated through final filters. An amount equal to that entering the tanks is continuously pumped out, passing through the primary on-line radioactivity monitor. If the uranium concentration in this waste stream is below a specified level (normally MPC) as measured by the primary monitor, the waste stream is then pumped to one of the two pump-out tanks which are operated on level control. If the uranium concentration exceeds this specified level, the waste stream is diverted to the diversion tanks where it is circulated and filtered again.

Contents of the pump-out tanks are automatically discharged to the Waste Treatment Facility described in Paragraph 1.3.5 for chemical treatment. As the waste solution is discharged, it passes through cartridge filters, the final radioactivity monitor and an on-line volume meter and automatic proportional sampler located in the Conversion Section of the Controlled Area. Radiation Protection Component personnel records the volume discharged and collect and analyze the proportional sample for radioactivity at least daily.

After circulation through cartridge filters, contents of the diversion tanks are manually sampled and analyzed by Radiation Protection Component personnel. If the uranium concentration is acceptable, the contents are pumped back to the final filtration tanks. If not, they are tranferred to Scrap Recovery for recovery of the uranium and ultimate disposal.

Each of the three on-line monitors is calibrated at least semiannually. In addition, the daily proportional sample

1.9.1.9 (Continued)

analysis is used to confirm the proper functioning of the monitors on a daily basis.

Although the uranium concentration in the final filtration tanks is normally less than the maximum permissible concentration (MPC) values, these tanks as well as the diversion tanks are filled with borosilicate glass Raschig rings to provide a secondary means of criticality control.

The pH of the solution is normally <11 (approximately 0.25 normal ammonium hydroxide) and the fluoride concentration (present as NH_4F) is approximately 4 percent. At the Waste Treatment Facility, uranium is removed (see Subparagraph 1.9.1.9(1)), fluoride is precipated by the addition of calcium hydroxide and free ammonia is removed via distillation. The still bottoms are pumped to a settling lagoon where the CaF_2 settles out and the treated effluent water is then discharged to waste-holding lagoons. The wastewater is further diluted as it mixes with other non-nuclear wastewater from the sewage system at the lift station. From the lift station, the combined liquid effluent is pumped through a final aerator and then pumped via pipeline to the Congaree River.

The precipitated material is treated as low-level solid waste until the final disposition is determined based on the analytical result. The final disposition may be shipment to a licensed burial location, burial on site, or disposal as nonradioactive industrial wastes, depending on the level of activity indicated by the analysis.

(1) Advanced Waste Treatment Description and Operation

(a) Purpose

Aqueous wastes from the ADU conversion process are treated to further reduce uranium concentrations. The recovered uranium is returned to the Manufacturing Building for purification and

1.9.1.9 (Continued)

recycle; the wastewater stream is processed further to remove fluorides and ammonia.

(b) Waste Characterization

The aqueous wastewater stream from ADU conversion process operations is filtered to lower uranium concentrations to approximately 15 mg U/liter. The wastewater treatment and monitoring system to control aqueous wastewater concentrations from the ADU Conversion lines are described in this subparagraph. The Advanced Waste Treatment process reduces uranium concentrations to less than 0.3 mg U/liter.

(c) Location

The Advanced Waste Treatment Process Building is located adjacent to the Waste Storage Building as shown on Figure 1.3.2.1.

(d) Process Description and Operation

Aqueous wastes are transferred from the Manufacturing Building to one of several 30,000-gallon storage tanks and then to the Advanced Waste Treatment Building. The process liquid is mixed with a stream containing a flocculating and bonding compound. The compound precipitates, bonding soluble uranium in a gel, and flocculating the insoluble uranium. Leaving the tank, the liquid-solid stream enters a rotary pressure filter where the uranium-bearing slurry is removed. The "uranium depleted" filtrate aqueous stream is then transferred to one of several 30,000-gallon surge tanks for further processing to remove ammonia and fluoride.

The flocculating and bonding stock solution is stored in carbon steel storage tanks. This

1.9.1.9 (Continued)

(d) (Continued)

solution is diluted with water and metered into the aqueous waste as it enters the storage tanks.

The uranium-bearing slurry leaving the pressurized rotary filter is stored in acid leach tanks which are mechanically stirred to prevent solids settling. Nitric acid is slowly added to the slurry to lower the pH, break up the gel and form uranyl nitrate. This mixture is stirred and then pumped through a second rotary pressure filter to separate and wash the solids until the uranium level is reduced to an acceptable level. This "uranium-depleted" slurry is solidified and disposed as waste. The uranium-bearing solution is sent to the solvent extraction units in the Manufacturing Building for uranium recovery. Refer to Figure 1.9.1.1 which presents a schematic diagram of the treatment process.

(e) Radiological Safety

The aqueous wastes are processed within the Advanced Waste Treatment Building as shown on Figure 1.3.2.1. Aqueous wastes enter the process with a concentration of approximately 15 mg U/liter; uranyl nitrate at approximately 5,000 mg U/liter is leached from the slurry as the final product.

Uranium-bearing materials are either in a liquid or solid/slurry form throughout the system and are contained within tanks, piping, pumps, filters, etc. Consequently, no significant airborne radioactivity concentrations have been measured.

The Advanced Waste Treatment Building is maintained as a limited area, with surveillance and control

1.9.1.9 (Continued)

(e) Continued)

level guides as defined in Subparagraph 3.2 of the existing license. Air sampling was performed to verify the expected low airborne radioactivity concentrations. Building ventilation consists of natural draft ventilation via louvers in the sides and roof of the structure.

All personnel which have access to the building are subject to the bioassay program as defined in Subparagraph 3.2.4 of the existing license.

The "uranium-depleted" slurry is solidified and disposed as waste. Since the uranium concentrations in the slurry are low (e.g., <0.001 mg U/liter), no airborne radioactivity or contamination problems have been encountered.

The uranyl nitrate product is either pumped or containerized and transferred to the Manufacturing Building for purification and recycle. Containers used for the aqueous uranyl nitrate transfers are equivalent to Type A packaging as defined in Title 49 of the Code of Federal Regulations such as DOT Specification 6D, 55-gallon capacity steel drums with DOT Specification 2 SL polyethylene inner liners.

(f) Nuclear Criticality Safety

During the aqueous waste processing, uranium is concentrated in a slurry by a bonding of the soluble uranium in a gel and flocculation of the insoluble uranium. Based upon experiments performed at Westinghouse Research Laboratories, the maximum uranium concentration occurs at the slurry discharge of the first rotary pressure

1.9.1.9 (Continued)

(f) (Continued)

filter. This slurry contains approximately 16 w/o solids and approximately 5,000 mg U/liter or approxiately 0.21 gm U-235/liter, so that neutron interaction is of limited concern (see Subparagraph 2.3.2.2 of the existing license).

Slurry from the rotary pressure filter is transferred to 400-gallon leach tanks and combined with nitric acid to form uranyl nitrate. Consequently, uranium concentrations will be further reduced in these tanks. If the tanks were full of slurry and in the unlikely event that total uranium precipitation were to occur, the quantity of uranium in the tanks can be calculated as follows:

$$\frac{5.0 \text{ gm U}}{\text{liter}} \times \underline{400 \text{ gallons}} \times \frac{3.785 \text{ liters}}{\text{gallon}} \times 0.05 = 379 \text{ gm U-235}$$

where the maximum enrichment is 5.0 w/o U-235. This quantity represents a safe mass (see Figure 2.3.2.1).

All vessels within the building are bottom drained to minimize buildup of uranium. Where uranium accumulations are possible, periodic vessel inspections are conducted and actions taken to remove the material.

(g) Environmental Safety

The Advanced Waste Treatment Process removes over 90 per cent of the uranium from the aqueous waste stream discharged from the Manufacturing Building. The technical goal is to lower the uranium level in the aqueous waste from the present average of approximately 15 mg U/liter to less than 0.3 mg U/liter. Based upon the present average aqueous

1.9.1.9 (Continued)

(g) (Continued)

waste discharge rate of 12 gallons per minute per line, over 300 Kg of uranium per year will be recovered by this process for recycle as nuclear fuel.

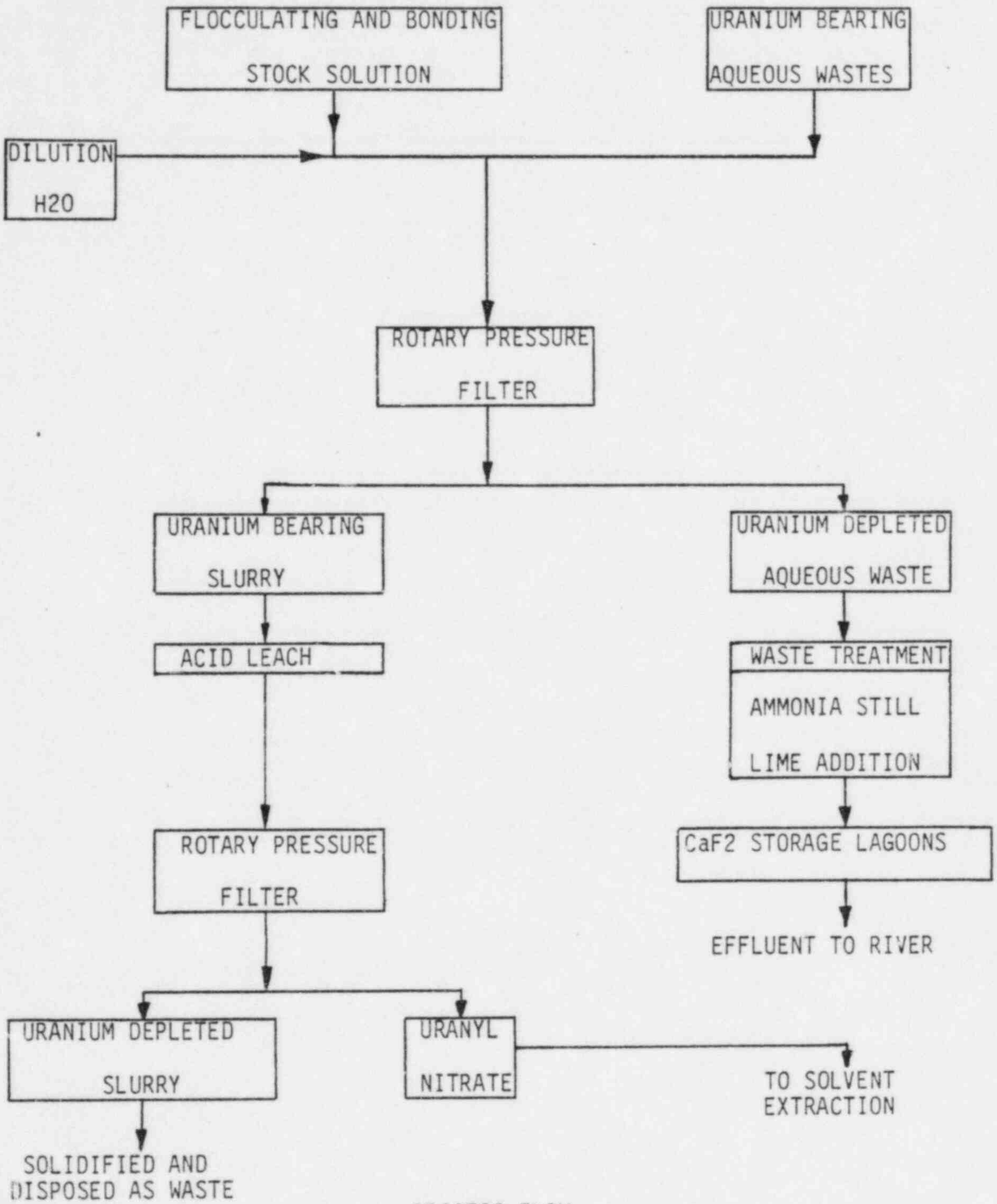
This process is an addition to the existing Waste Treatment System which consists of filtration, ammonia recovery, fluoride precipitation and pH adjustment. The feed materials to the process are aqueous wastes from the Manufacturing Building; discharges from the process are uranium depleted aqueous wastes which will be further treated to remove ammonia and fluorides, uranyl nitrate which is transferred back to the manufacturing plant, and "spent" flocculating and bonding agents which are solidified and disposed as waste. Consequently, there are no additional aqueous discharges to the environment.

Airborne radioactivity discharges from the facility have been minimal since uranium-bearing materials are handled in aqueous forms.

1.9.1.10 Solution Storage

Solutions are stored in tanks or containers under concentration ($\leq$5 grams U-235/l) control or in safe ($\leq$35 l) volume containers in a square lattice with at least a 1-foot edge-to-edge spacing. Alternately, they may be stored under geometric control as provided by safe ($\leq$11-inch) diameter containers. Edge-to-edge spacings between these containers will be provided in accordance with the surface density criteria described in Subparagraph 2.3.2.

ADVANCED WASTE WATER TREATMENT



PROCESS FLOW
Figure 1.9.1.1

1.9.2 Fabrication Operations

The fabrication operations which are described in detail in the following subparagraphs represent the fabrication operations which have been carried out safely and economically for the past 14 years at the NFD Manufacturing Department's Columbia Plant under License SNM-1107, Docket 70-1151. The success of the Columbia Operation, while not a substitute for the detailed analyses contained in the following subparagraphs, is strong supporting evidence of the adequacy of the radiation protection and nuclear criticality safety controls which are designed into these operations. The descriptions given below describe the processing of 4.15 w/o enriched uranium, unless specifically stated otherwise.

1.9.2.1 Receipt of Material (from Outside Vendors)

Dry (as defined in Subparagraph 2.1.1) uranium oxide is received from suppliers in licensed packages. These are opened inside the plant where individual containers, sized and loaded to Westinghouse specifications, are inventoried, weighed and stored. Samples of incoming material are taken and sent to the Analytical Laboratory for analyses to verify the U-235 enrichment, dryness, and other specification requirements. Each separate enrichment is assigned a distinctive identification tape. This tape is used on in-process containers as required to maintain the enrichment identity during processing.

The powder containers are closed and sealed when received. Most of them remain closed during the unloading and weighing steps, and therefore require only routine radiation protection precautions. Those which are opened for sampling purposes are opened in a ventilated enclosure equipped with a HEPA filter.

The individual shipping packages are sized to assure nuclear criticality safety under credible accident conditions.

1.9.2.1 (Continued)

Uranium oxide powder, enriched ≤5.0 w/o in U-235, is received in ≤9.4-inch diameter inner containers which may be stacked in coaxial cylinders. As specified in Subparagraph 1.8.1.2, interaction between cylinders need not be considered due to the "Dryness" of the powder. Analyses are used to verify that the powder is "Dry." Incoming material is statistically sampled and analyzed to verify the shipper's values.

1.9.2.2 UO2 Powder Storage

Closed powder containers, whether received from an outside vendor or from the Conversion Facility are placed onto one of several portable steel storage carts. These carts are designed to assure that the containers on each cart form a tier of long cylinders with their axes parallel and near horizontal. Alternatively, the containers may be placed on fixed steel storage shelves. These shelves are designed to assure that the containers on the shelves form vertical columns of long cylinders with their axes parallel. In both cases, the cylinders are spaced 12 inches edge-to-edge from each other.

The containers remain closed during storage, and therefore require only routine radiation protection precautions.

The inside diameter of each of the cylinders formed by the stored containers of uranium oxide powder is ≤9.4 inches, assuring nuclear criticality safety under any conditons for uranium enriched ≤5.0 w/o U-235. All storage carts and shelves are constructed and spaced to assure at least 12 inches edge-to-edge separation between any two cylinders. All material is "Dry" and the containers are tightly closed. No sprinkler systems are installed in the area. Furthermore, the use of water or other hydrogenous fire fighting agents is specifically prohibited in moderation controlled areas, and such areas are posted accordingly. For these reasons,

1.9.2.2 (Continued)

interaction between parallel carts and shelves need not be considered.

This concept of spaced, infinite cylinders is extended throughout the plant to additional storage and conveyance systems where "Dry" uranium is maintained in closed containers.

Moderated material, including grinder sludge and broken pellets, generally is dried before being stored in polyethylene drums (polypaks) or fiberboard drums (fiberpaks) which are placed on steel shelves in a solid concrete block structure. Each container of SNM is limited to approximately 40 lbs. UO2, and has a diameter of $\leq$9.5 inches which is less than the MPV of 10.2 inches for $\leq$4.15 w/o homogenous oxides. Stacked containers, therefore, constitute vertical, safe diameter columns, the height of which does not exceed 72 inches. Each column within the structure is separated from any adjacent column in the subarray by 8 inches of high density ($\geq$140 pounds/ft^3) concrete or the nuclear equivalent. At least a two-foot aisle separation is maintained between subarrays. The degree of subcriticality of the entire array has been evaluated using the KENO-II Monte Carlo computer code. The array Keff was conservatively calculated to be 0.908 + 0.008.

Alternatively, moderated material may be stored on nuclear safe steel racks under mass, volume or geometric criticality control criteria. These racks are designed to assure the required spacings as determined from the surface density criteria described in Subparagraph 2.3.2.

1.9.2.3 <u>Powder Preparation and Pelleting</u>

Uranium oxide powder is manually transferred from inprocess storage to a ventilated hood equipped with a HEPA filter. Here, the powder is opened and the contents transferred to an elevator tray which is put into a fully enclosed elevator

1.9.2.3 (Continued)

system. This system transports the powder to a hopper which gravity feeds into a press or roll type powder compaction unit. Compacted powder is automatically transferred to a mechanical granulator.

Granulated uranium oxide powder is collected in nuclearly safe containers in a HEPA filtered ventilation hood where it is blended with a small percentage of binder-lubricant (not sufficient to increase the H/U beyond 0.5 for $\leq$4.15 w/o U-235 or H/U of 0.3 for >4.15 w/o U-235 but less than or equal to 5 w/o U-235 unless cleared by computer calculation. Blending is accomplished by rolling the material in a container. This feed material is then transferred to the $\leq$10.1-inch diameter feed hopper of the pellet press. Powder flows by gravity from the hopper to the pellet press die, where product pellets are formed. These are transferred either into nuclearly safe, intermediate storage containers or directly into molybdenum sintering boats for further processing.

Negative pressure is provided at the opening of the press feed hoppers to preclude dusting when uranium oxide powder is added. Similar ventilation and containment are also provided at other work areas of each of the presses where exposed uranium oxide is processed.

The other operations described (press feed addition, granulation, binder-lubricant addition, and blending) are carried out in ventilated hoods. The effectiveness of the protective features is continuously evaluated using air sampling techniques. Air filtration is provided by HEPA filtering systems. The filtered air is then recirculated.

Each step described herein is controlled using limits which assure nuclear criticality safety under any conditions of moderation and reflection. SNM accumulations in the filtering systems are maintained below the MPV for mass, cylinder diameter, or slab thickness by engineered or

1.9.2.3 (Continued)

administrative controls. Systems under administrative control are visually inspected at least weekly during operatons, and any observed SNM accumulations are removed for subsequent processing.

Uranium oxide powder is transferred to the press feed hoppers. These are either mass limited to one 60-pound batch, or geometry limited to a 10.1-inch diameter cylinder.

Granulated powder is collected in containers having a diameter $\leq$10.1 inches or a volume $\leq$26 liters. Containers having this same diameter or volume are used for the subsequent binder-lubricant addition and blending operations. The press feed hopper spacing area of >8 ft^2 provides a "smeared" slab thickness less than the allowable value of 2.0 inches.

Other spacing requirements are generally satisfied by the fact that any one hood or press is normally limited to one safe subcrit. More than one subcrit is permitted only when each subcrit is provided with, and maintained within, its own nuclearly safe area. Each hood or press occupies a floor area of from 12.5 to more than 25 ft^2.

1.9.2.4 Sintering

Sintering boats are conveyed to the sintering furnace area and charged one at a time in a single line into the appropriate furnace. As the boats pass through the controlled-atmosphere furnace, the pellets sinter to about 95 per cent of theoretical U02 density. Upon discharge from the furnace, the boats are transferred to the grinder area.

Exhaust and supply air fans are provided to remove heat generated by the furnaces. In addition, building room air is discharged through the exhaust fans and subsequently HEPA filtered prior to release to the environment.

1.9.2.4 (Continued)

Although protective clothing is worn in the furnace area, the pelletized form of the material minimizes the generation of dusts. However, continuous air sampling is conducted in this area.

The movement and storage of loaded sintering boats in the furnace area and its associated conveyors are controlled on a slab basis by the $\leq$4.0-inch depth of the boats or on an equivalent 8-inch diameter cylinder basis. Administrative and/or physical controls are employed to prevent the stacking of loaded boats beyond the allowable slab thickness.

1.9.2.5 Pellet Processing

To obtain precise pellet dimensions, all pellets are processed through a grinding operation. The grinding is conducted dry or under a stream of coolant which also carries off abraded particles. This stream is passed through a centrifuge to remove solids and is recirculated. The pellets to be ground are contained in sintering boats, or temporary storage containers having a depth $\leq$4.0 inches on the input conveyor adjacent to the grinder. As pellets are ground to the specified dimension, they are automatically loaded onto trays on the output conveyor and inspected for defects. Loaded trays may be transferred to an electrically heated dryer oven to evaporate moisture from the pellets. After drying, the trays are transferred to pellet tray storage carts.

During the wet grinding operation, a sludge containing a mixture of uranium oxide and abrasive is formed. Also, some pellet production may fail to meet quality standards. Sludge and/or off-standard pellets are accumulated in maximum permissible mass batches or $\leq$4-inch slabs and stored pending disposition.

Frequent surveys and the use of protective clothing provide adequate contamination control. The grinders are provided

1.9.2.5 (Continued)

with ventilated containments to minimize airborne radioactivity. The drying ovens are vented through a HEPA filter to the working environment.

The pellet grinding operating is a slab-controlled process. Pellets are fed and unloaded on a slab basis.

Pellets on special pellet trays which do not retain water are stacked in slabs (each ≤ 4 inches thick pellets) on the conveyors or in the drying ovens.

Pellets are stored on special pellet trays in portable storage carts or fixed shelves. Each cart or set of shelves provides for slabs separated vertically by a twelve-inch spacing. Each slab meets the MPV for pellet thickness. Neither design will permit the retention of water. Adjacent carts or shelves form continuous slab arrays.

Alternately, pellet trays can be stored on each of the 52 shelves, provided that every other shelf is at least 3/16 inch boral. The boral shelves are fixed to preclude inadvertent removal of the shelves; installation of the shelves is confirmed by the Radiation Protection Component in accordance with Subparagraph 2.3.2.1(6). The Keff for an infinite array of carts with continuous pellet shelves assuming that every other shelf is at least 3/16 inch boral is 0.714 + 0.007 as determined by computer calculations.

The nuclear criticality safety of the uranium contained in the sludge in the closed coolant system is based on a mass MPV. In addition, the system geometry is limited to a ≤ 7.0 gallon centrifuge and a ≤ 4.8 inch slab shaped reservoir.

1.9.2.6 <u>Rod Loading</u>

Pellet trays are transferred to rod loading fixtures. The fuel pellets are placed on fuel rod mockup channels, and then loaded into empty fuel capsules. Loaded capsules are plugged, placed on trays ("cookie sheets") and transferred to

1.9.2.6 (Continued)

weld stations where they are inserted into a fixture and welded. Finished rods are then transferred to the final fuel assembly area.

Standard protective clothing and contamination control procedures apply.

These operations are all controlled on a slab basis (one rod or pellet diameter is the present practical maximum thickness).

1.9.2.7 Quality Assurance Operations

Finished fuel rods are subjected to a variety of inspection operations to assure the quality of the final product. These operations may include visual inspection, weighing, dimensional inspection, radiography, gamma scanning and leak testing.

Fuel rods to be leak tested are placed in a chamber, which is designed to permit the evacuation of the atmosphere in the chamber. Defective seal welds are detected by checking for the presence of rod filler gas in the chamber, following evacuation.

These operations are performed in an area free from surface contamination and present no contamination problems.

Quality assurance operations are controlled on a slab ($\leq$4 inches) basis. Close packed fuel rods are stored throughout the plant as slabs with a thickness less than or equal to the MPV of 20 inches.

1.9.2.8 Fuel Assembly Processing

Fuel rods are fabricated into completed fuel assemblies in an area free from surface contamination. The fuel assemblies are inspected and cleaned using an appropriate material. They may then be stored, or packaged immediately in NRC and DOT approved packagings as dictated by production needs.

These operations are performed in an area free from surface contamination, and present no contamination problems.

1.9.2.8 (Continued)

Except for the cleaning operation, all fuel assembly processing operations are performed dry. Completed fuel assemblies are handled and stored in a manner which provides a minimum 12-inch edge-to-edge spacing.

Cleaning operations are performed in a pit containing 3 wash tanks in a line below the monorail, each with assembly spacing features. The center-to-center spacing between fuel assemblies is 30 inches and there is at least 12 inches of water between moderated assemblies. The same solution provides isolation and moderation so that moderation without isolation is not credible for these open type assemblies. The monorail cranes assure at least a 12-inch edge-to-edge separation between assemblies.

The MPV applied is the assembly reactivity, which does not exceed 0.95, as determined by the NFD Nuclear Enginering group. The tanks contain no strongly corrosive agents which could dissolve the cladding and permit a more reactive arrangement of the fuel.

1.9.2.9 <u>Processing UO2 (4.15 < U w/o < 5)</u>

Controls similar to those described in Subparagraphs 1.9.2.1 through 1.9.2.8 are imposed during the processing of uranium oxide enriched between 4.15 w/o and 5 w/o, except that MPV's are appropriately reduced as specified in Figures 2.3.2.1 through 2.3.2.5.

Since operations involving any given enrichment are carried out only in a single designated processing line as a routine precaution to avoid commingling enrichments, such control will be readily administered. No SNM enriched to >4.15 w/o will be permitted in the scrap preparation area unless such operations are reviewed and approved by the Radiation Protection Component.

1.9.3 Laboratory Analysis

Samples of fuel material which have been selected at various stages during fuel conversion and fuel assembly fabrication are analyzed in the Analytical Laboratory. Appropriate techniques for wet chemical, spectrographic and other forms of analysis are used. Most of the samples are handled in solution form. When preliminary operations, such as weighing out powder samples or crushing sample pellets prior to dissolution, must be performed, they are carried out in a suitable ventilated hood with a HEPA filtered exhaust.

Samples from other Westinghouse Divisions or outside customers also may be received and analyzed.

Uranium-233 may be used in the preparation of precisely controlled, standard solutions. Samples of the solutions may be used in the determination of total uranium in Zircaloy (3.5 ppm maximum) by the isotopic dilution method.

A maximum permissible mass limit of 700 grams of SNM as contained U-235, with up to 5 grams of U-233, is imposed on each laboratory.

The normal industrial hygiene requirements associated with the operation of a chemical laboratory, together with the technical competence and training of the personnel provide adequate radiation protection measures. Special care is exercised when dry oxides are being handled to avoid the possibility of ingesting fuel material.

1.9.4 Scrap Recovery Operations

1.9.4.1 General

Scrap recovery process operations are characterized as batch operations involving a variety of input forms. The preliminary operations concentrate and purify the materials if needed and convert it to forms readily processed into U_3O_8 powder. Not all materials require processing through the entire sequence of operations.

1.9.4.1 (Continued)

The basic processing sequence includes dissolution of solid forms, conversion to slurry form by precipitating ADU from the solution, dewatering the slurry form by wet mechanical separation, calcining the resulting sludge in furnaces, and packaging and storing the resulting product. The product is sampled and analyzed before release to manufacturing.

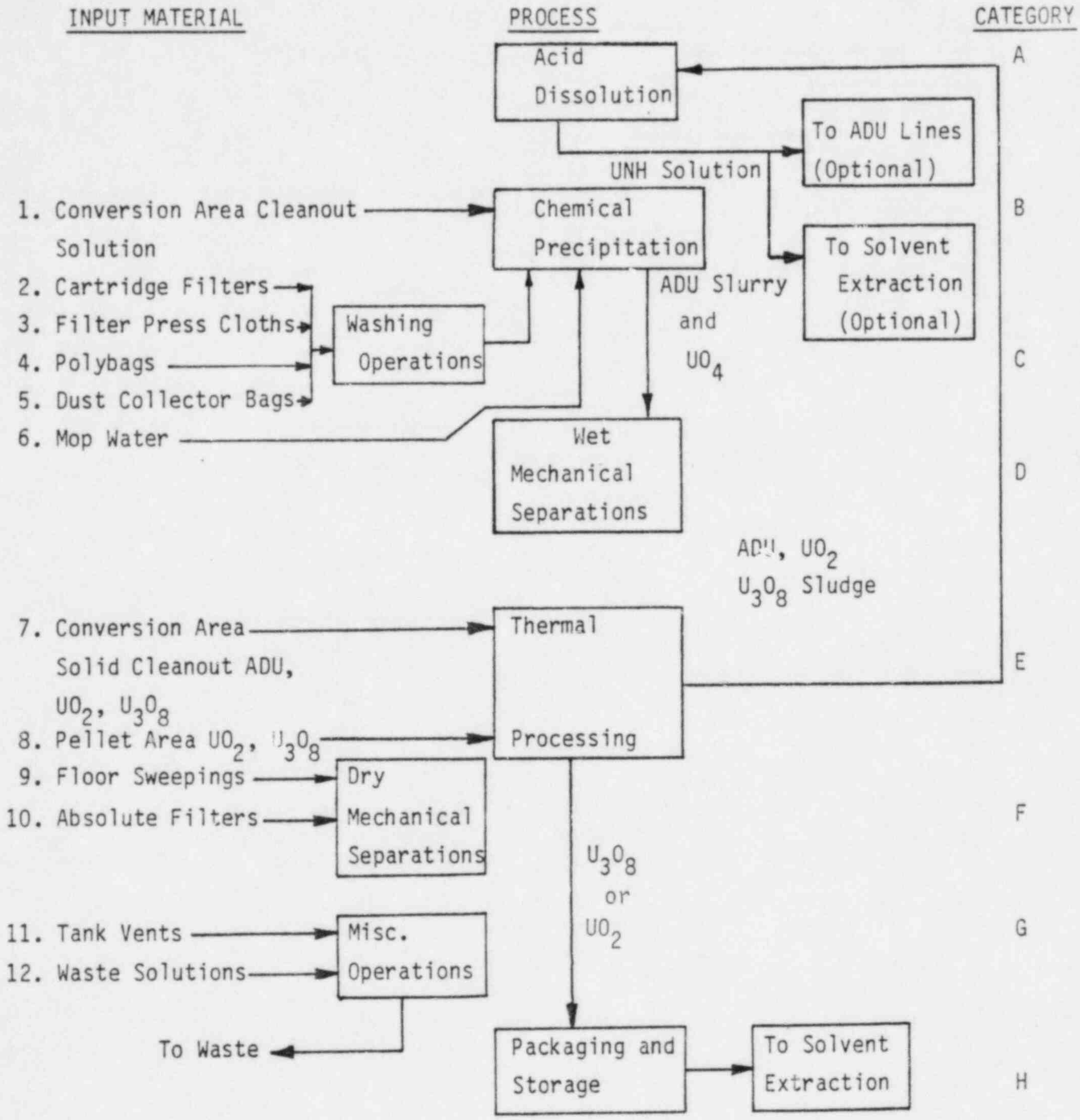
Various other inputs are fed into the basic sequence at appropriate points. Liquids containing dissolved uranium are introduced into the solution hold tanks used to supply the precipitator columns. Clean aqueous suspensions from mopping or off-gas scrubbing operations and from laundering cloth filters, poly bags, etc. are introduced prior to the wet mechanical separation operation. Clean powder scrap, scrapped pellets and other quality solids may be subjected to a dry mechanical separation step and then are introduced, along with the wet sludge, to the calcining operation. Thermal processing capabilities also exist to convert this clean scrap material directly into usable UO_2 or U_3O_8 powder.

Figure 1.9.4.1 is a flow diagram demonstrating the types of material inputs into the system and the principle operations applied to each type. The operations are designated by categories to simplify subsequent discussions.

1.9.4.2 <u>Dissolution Operation (Category A)</u>

Acid dissolution of oxidized scrap materials is carried out in stainless steel tanks. Prior to its introduction into the tank, the material is granulated and screened to promote the speed of dissolution.

The off-gases from the dissolution tank are routed through a reflux condenser and a scrubber to remove entrained particles and condensible vapors prior to releasing the gases through the HEPA filtered exhaust system to the atmosphere. The



SCRAP RECOVERY PROCESS OPERATIONS FLOW SHEET

Figure 1.9.4.1

1.9.4.2 (Continued)

reflux condenser is mounted vertically and directly above the dissolution tank so that any condensate formed gravity drains back into the dissolution tank.

The solid special nuclear materials introduced into the dissolution tanks are restricted to powdered forms. Initially, a slurry, or oxide water mixture is formed. Following dissolution and filtration, the contents form true solution. Acid dissolution is performed using administrative controls to assure less than the maximum permissible mass (Figure 2.3.2.1 of our application) or uranium in the system at any time.

The entire recovery process area has been evaluated using both the surface density criteria specified in Subparagraph 2.3.2 and the solid angle method of TID-7016 to assure the nuclear criticality safety of the array of processing vessels and equipment.

1.9.4.3 Chemical Precipitation (Category B)

Precipitation operations and subsequent slurry storage also are carried out in 10-inch diameter tanks. The precipitate is in the form of discrete, homogeneous particles, and the 10-inch diameter is less than the maximum permissible diameter for homogeneous oxide water mixtures.

1.9.4.4 Wet Mechanical Separation (Category D)

Wet mechanical separation can be performed using either a centrifuge or a filter press. Both of these devices are sized to be less than the 28 liters listed on Figure 2.3.2.2 as the maximum permissible volume for oxide-water mixtures containing uranium enriched to $\leq$4.15 w/o.

Plate-and frame filter presses, each with a volume of $\leq$28 liters, are used for filtering various uranium solutions. Ventilation hoods connected to the scrubber intake are installed directly above the filter presses for removing fumes

1.9.4.4 (Continued)

and vapors from the filter press areas. The scrubber removes any condensible vapors prior to releasing the off-gases to the atmosphere through the HEPA filtered exhaust system.

The filter press cloths, dust collection bags, cartridge filters and similar materials impregnated with uranium (Category C) are laundered in a front loading, washing machine. Conservative tare weights for each type of material to be laundered are established by randomly selecting dry samples of each type, weighing the samples, and assigning each type a tare weight equal to the maximum sample weight. For administrative purposes, the uranium-bearing materials are divided into batches, each with a net weight $\leq$56 pounds, which is the maximum permissible mass specified in Figure 2.3.2.1 for $\leq$4.15 w/o uranium oxide-water mixtures. Since each type of material to be laundered except dust collection bags, is wet and the total weight of entrained liquid is assumed to be uranium, the calculated net weights are also conservative.

The washing machine has a wet capacity of about 130 pounds so each batch may be laundered as one or more washer loads. The inner bowl and the drain of the washing machine are checked and cleaned (if required) after the completion of each batch. The design of the outer bowl and gravity flow drain effectively prevents any significant accumulation of SNM in the outer bowl.

The liquids usually discharged to a drain are instead collected in tanks having a 10-inch diameter which is less than the maximum permissible diameter (specified in Figure 2.3.2.3) for $\leq$4.15 w/o uranium oxide-water mixtures. These liquids provide feed slurries to the wet mechanical separation operation.

1.9.4.5 Thermal Processing (Category E)

Filter press "cake" and/or wet cleanout materials are dried and converted to U3O8 powder in furnaces in pans which assure a safe slab geometry control ($\leq$4.75 inches given in Figure 2.3.2.4).

Off-gases from the furnaces are vented through hoods connected to the scrubber intake. The hoods are installed directly above the furnace doors near the points of release for the off-gases. Entrained particles and condensible vapors are removed in the scrubber prior to release to the atmosphere through the HEPA filtered exhaust system.

The resulting U3O8 powder is packaged and stored while awaiting further processing.

Various solid materials which are uncontaminated may be introduced into a thermal processing step directly to convert them into the desired uranium oxide form.

1.9.4.6 Dry Mechanical Separation (Category F)

Nonhomogeneous uranium oxides are crushed to a powder in a mechanical, granulator prior to the dissolution operation to promote the speed of the reaction. Contaminated UO2 and U3O8 such as that found in floor sweepings is sorted to remove trash and gross impurities. In addition, accumulations of chemically uncontaminated UO2 or U3O8 such as those occurring on absolute filters in the Controlled Area are removed for further processing by manual techniques such as shaking, scraping, etc.

Dry materials charging, dry mechanical separation, pellet and powder granulation, and final packaging are performed in hoods, hopper dryboxes, or similar enclosures equipped with ventilation and HEPA filters to minimize airborne contamination levels. Continuous air samplers are strategically located throughout the scrap recovery area to obtain representative air samples. Face velocity measurements

1.9.4.6 (Continued

for each ventilation enclosure are performed at least weekly. Any reduced face velocity which approaches the minimum effective capture velocity is corrected immediately.

Dry mechanical separation operations are performed using administrative controls to assure less than the maximum permissible values such as mass (Figure 2.3.2.1) and slab (Figure 2.3.2.4) in each ventilated hood or enclosure.

1.9.4.7 Miscellaneous Operations (Category G)

The off-gases vented from the process tanks and furnaces in the scrap recovery area are routed through a scrubber to remove entrained particles and condensible vapors prior to release to the atmosphere through the HEPA filtered exhaust system. The scrubber water is periodically transferred to one of the Conversion Area quarantine tanks. While being used in the scrubber system, the water is maintained in a reservoir having a 10.1-inch maximum permissible cylinder diameter as specified for $\leq$4.15 w/o uranium oxide-water mixtures in Figure 2.3.2.3.

Liquid effluents from the scrubber, precipitation and wet separation operations are pumped to Conversion Area diversion tanks for subsequent monitoring or sampling and analysis to provide assurance that the material can be disposed of via the Waste Treatment Facility without exceeding the allowable release level specified in 10 CFR 20. Although the uranium concentration in the diversion tanks is normally less than the maximum permissible concentration (MPC) values, the diversion tanks are filled with borosilicate glass Raschig rings.

1.9.5 Moderation Controlled Processing Operations

1.9.5.1 Bulk UO2 Powder Handling

1.9.5.1.1 General Process Description

Oxides of uranium are discharged from the calciner, through a hammermill and into a safe geometry container. Following

1.9.5.1.1 (Continued)

representative sampling of this material and moisture analyses, the containers are placed in interim storage and transported to a moderation controlled Blending/Storage Area for further processing..

In the Blending/Storage Area, the material is transferred into larger containers (approximately 60 cubic feet), blended and stored while awaiting further processing. Additional operations to be performed in this area include sampling, consolidation, remilling, addition of recycle materials and installation and removal of the cover plates and feeders from the bulk containers. All such operations are conducted with appropriate containment and ventilation. As needed, bulk containers are transferred to the head end of the Pellet Area and attached to the Feed Prep Station via a feeder device. Powder is transferred from the bulk container to the Feed Prep Station as required. Further processing of this materal is described in other portions of this license and include roll compaction, granulation, lubricant addition, pressing, sintering and grinding.

1.9.5.1.2 <u>Conversion Area Operations</u>

<u>Comminution</u>

Dry uranium oxide powder is conveyed from the calciner exit chute, through a milling operation into a receiving container (approximately 150 Kg U capacity). The receiving containers are 304 stainless steel cylinders, approximately 64 inches in length and 10.4 inches inside diameter. The 10.4-inch inside diameter of the receiving containers satisfies the MPV for homogeneous oxides for enrichments ≤ 4 w/o U-235. Powder dryness is verified by representatively sampling the powder as it flows from the milling exit chute to the receiving container and analyzing the sample for moisture content. Administrative controls (e.g., visual inspection of empty

1.9.5.1.2 (Continued)

containers) assure that there are no moderator materials present prior to filling. Calciner contents empty into a 6-inch equivalent diameter discharge hopper and then to a 4-inch equivalent diameter, 6 feet long screw conveyor which empties to a 6-inch equivalent diameter gravity flow section and then to the mill. Powder exiting the mill gravity flows to the 10.4-inch equivalent diameter receiving container described above.

Alternatively, powder can flow from the calciner discharge hopper through a 6-inch equivalent diameter gravity flow section to an approximately 9.5-inch diameter polypak. Addback materials are introduced to the mill via a 6-inch equivalent diameter gravity flow section into the discharge point of the screw conveyor. Polypaks are automatically discharged into a hopper which feeds the 6-inch equivalent diameter section. The total volume of the hopper is 20.4 liters which represents a safe volume. The screw conveyor and hoppers satisfy the reduced diameter requirements for ells and tees (9.4 inches and 8.6 inches respectively) for homogeneous oxide systems for enrichments $\leq$4.15 w/o U-235. The connecting hoppers, feed chutes and mill form a slab type gravity flow arrangement with a maximum cross section of 5.5 x 15.5 inches which is equal in cross sectional area to the MPV for homogeneous oxides for enrichments $\leq$4.15 w/o U-235.

The above powder transport and milling equipment form a closed system which is maintained at a negative pressure relative to its surroundings. Ventilation exhausts are HEPA filtered before being discharged.

Powder Sampling

Powder exiting the mill is representatively batch sampled by automatically withdrawing duplicate samples as the powder

1.9.5.1.2 (Continued)

flows into the 150 Kg receiving containers. Sample withdrawal frequencies are established based upon the maximum rate of powder flow from the mill. The volume of powder collected in the sample container is used to determine the representativeness of the collected powder. Samples of powder are analyzed for moisture content using standard laboratory techniques.

When the 150 Kg container is filled with powder, the lid is closed, gasket sealed and affixed with a tamper-safe seal to indicate that the container integrity has been maintained. Administrative controls, e.g., operator documentation and overchecks, shall be used to verify that the 150 Kg receiving containers have been appropriately sampled, analyzed and sealed.

Powder Storage

Sealed 150 Kg receiving containers are removed from the mill enclosure and stored in the process area, with spacing as specified by license criteria, while awaiting moisture analysis results. Following receipt of moisture analyses, the 150 Kg receiving containers are transported to transfer carts which assure a minimum spacing of 12 inches between containers. Off-spec materials in 150 Kg receiving containers are stored in an area approved by the Radiation Protection Component. These off-spec materials may be transferred back to polypaks for recycle through the conversion lines or through scrap recovery in accordance with this license.

Radiological Safety

The powder transport and milling equipment form a closed system which is maintained at a negative pressure relative to its surroundings. Furthermore, the mill and powder discharge station are equipped with secondary containment which is also maintained at a negative pressure to minimize airborne radioactivity during container transfer and filling operations.

1.9.5.1.2 (Continued)

Transfers of powder between containers are conducted using ventilated enclosures designed to minimize airborne radioactivity.

Air exhausted from ventilated enclosures is HEPA filtered prior to discharge. HEPA filters are leak tested upon installation and after each subsequent HEPA filter change. Continuous, representative air sampling is performed to monitor airborne radioactivity concentrations.

Nuclear Criticality Safety

Nuclear criticality safety for Conversion Area operations utilizes geometry control from the calciner discharge to the filling of the 150 Kg receiving container. Storage controls for 150 Kg receiving containers include spacing of containers in accordance with existing surface density or solid angle criteria prior to verification of dryness and moderation control for containers following verification of dryness.

Powder dryness is asured by exercising the following controls: (1) Containers are closed and sealed while empty and not in use. Inspection procedures require visual checking of each container prior to hookup for filling. (2) Uranium powder discharged from the calciner is normally "dry" as defined in this license (paragraph 2.1.1). Higher moisture content would only result from a gross malfunction of the calciner, such as a large reduction in temperature, which would be immediately obvious to the process operator. Changes in the physical characteristics of the powder would also be an indicator. (3) There are no sources of moderator in the powder transfer or milling equipment. (4) Powder dryness is verified by representative sampling and moisture analyses. (5) The 150 Kg receiving containers are closed using a hinged, gasketed and bolted cover. (6) A seal is applied to the cover to verify that proper sampling, moisture analyses and

1.9.5.1.2 (Continued)

closure have been completed. (7) Administrative controls, e.g., operator documentation and overcheck, are in effect to assure that the above steps are completed.

1.9.5.1.3 <u>Powder Blending and Storage</u>

Uranium powder is transferred to the Blending/Storage Area, which is maintained as a Moderation-Controlled Area, for further processing.

<u>Blending/Storage Area Description</u>

The Blending/Storage Area is an enclosed room located within the Controlled Area, designed to exclude moderator materials under both routine and nonroutine conditions. The walls consist of formed metal panels which are steel reinforced to provide an impact barrier for credible accidents, e.g., a lift truck impacting the walls. The ceiling consists of formed metal panels designed to exclude water in the event of a water leak above this area such as a roof leak or water pipe break. No hydrogenous-containing pipes are permitted to penetrate the walls or ceiling. All other penetrations (ventilation ducts, service piping, electrical, etc.) are sealed. Ventilation ducts are equipped with drainage capabilities to prevent water accumulations in the ducts.

<u>Operations</u>

Uranium oxides transferred to the Blending/Storage Area from existing facilities include powder from the milling operations and recycle materials. All process materials are determined to be "dry" before being transferred into the Blending/Storage Area. The following operations are conducted in this area:

<u>Loading of Bulk Containers</u> - Uranium oxides are transferred from geometry control containers into bulk containers by gravity feed from overhead containers via feeders. The feeders are physically attached to the bulk containers and to the feed containers. Where the feed container is a polypak,

1.9.5.1.3 (Continued)

the feeder is physically connected to a ventilated enclosure which provides containment during the powder transfers. Scales are used to measure the weight of powder transferred into the bulk containers. Total mass of uranium oxides in the bulk containers is approximately 1700 Kg.

Installation and Removal of Cover Plate from Bulk Container - Bulk containers are loaded with a feeder affixed (bolted and gasketed) to the container. This feeder is replaced with a cover plate (bolted and gasketed) which is attached to the bulk container during blending and storage. Secondary containment and ventilation are provided to minimize airborne radioactivity during installation and removal of the cover plate and feeder.

Blending - Bulk containers are transferred via lift truck to the blending station consisting of a hydraulic clamping device to hold the bulk container and a hydraulic container rotating device. The clamping device contains approximately three gallons of low-hydrogenous hydraulic fluid (approximately 5.5 ppm hydrogen). The rotating mechanism contains approximately fifty gallons of hydraulic fluid which is physically shielded from the container being blended and from other nongeometry controlled containers in the area. Since the bulk containers are sealed during blending, no special radiological safety controls are required.

Container Sampling - Samples of powder are extracted from the containers as required by product specifications. A sampler station with appropriate containment and ventilation is provided to minimize airborne radioactivity.

Powder Transfer and Rework -- Powder is transferred between containers for purposes such as remilling, consolidation, and off-loading powder into preproduction containers. Remilling

1.9.5.1.3 (Continued)

and consolidation involve powder transfers between bulk containers and from 150 Kg containers to bulk containers as required, using gravity feed from overhead containers via a feeder. Although the connectors between the containers are sealed, secondary containment and ventilation are provided to minimize airborne radioactivity during connecting and disconnecting steps.

Powder can be off-loaded into moderation controlled containers (approximately 60 Kg capacity) or geometry controlled containers (polypaks) depending upon the required sample size for preproduction purposes.

Powder is transferred to the 60 Kg container by connecting the 60 Kg container to the bulk container, using gravity powder feed from the bulk container. Secondary containment and ventilation are provided to minimize airborne radioactivity during connecting and disconnecting steps. When filled, the 60 Kg containers are sampled within ventilated enclosures, transported to a pellet line and off-loaded within the enclosure described in Section 1.9.5.1.5. The 60 Kg containers are subject to the same philosophy as the bulk container, when in transit from the Blending/Storage Area to the pellet lines, to assure that hydrogenous materials are excluded from the containers. The containers are transported within a cart designed to protect the container during transport.

Powder can be transferred from the bulk container to polypaks using gravity flow through a feeder to a polypak which is mated to the feeder via a pressure fit between the top of the polypak and a gasket. The polypak and connecting surfaces are enclosed within ventilated confinement.

1.9.5.1.3 (Continued)

The following controls are in effect for polypaks used within the moderation controlled area:

(1) Prior to bringing polypaks containing SNM into the moderation controlled area, appropriate moisture determinations shall be completed to confirm "dryness" and the containers appropriately labelled.

(2) Polypaks shall be stored in a designated area to assure that there are no commingling of polypaks and moderation controlled powder (e.g., preclude polypak storage between bulk containers).

(3) Procedural controls are in effect which provide specific handling criteria for polypaks, particularly during powder transfers, polypak transfers and storage. Procedures are reviewed and approved by the Radiation Protection Component.

Storage - Uranium-bearing polypaks and 150 Kg containers on geometry-controlled transfer carts are permitted to be stored in this area. Storage is provided for geometry controlled and bulk containers to assure a minimum 12-inch edge-to-edge spacing. Designated storage areas are provided for polypaks to assure that there are no commingling of polypaks and moderation controlled powder.

Bulk Container Description

The bulk containers are approximately 60 cubic feet capacity, with a 48 x 42 inch rectangular cross section, approximately 80 inches in length with a conical end which tapers to approximately 16 inches. The tapered end is equipped with a flange to which a flange cover or vibratory feeder can be attached. The units are fabricated from minimum 0.140 inch thick 304 stainless steel. Flange covers and vibratory feeder flanges are constructed of minimum 0.250 inch thick 304 stainless steel with approximately one inch wide by

1.9.5.1.3 (Continued)

approximately 1/8 inch thick gasket and bolted to the cone with 1/2 inch bolts at approximately 5-inch spacings. The container is equipped with an angle iron frame which provides structural support and protects the container from physical damage during transport and storage.

A vent located near the bottom/top of the rectangular area is equipped with two metal filters and a quick disconnect fitting. Each end of the frame is equipped with a steel plate approximately 3/16 inch thick to protect the container during transport. (Physical protection is also provided for the other four sides of the container during transport.)

Bulk containers are designed, constructed and tested in accordance with applicable portions of 49 CFR 178.251 and 178.252 for specification 56 portable tanks.

<u>Radiological Safety</u>

Powder transfer operations utilize sealed connections for powder containment. Furthermore, secondary ventilated enclosures are provided to minimize airborne radioactivity during connecting and disconnecting operations. Exhaust air is discharged through HEPA filtration.

Radiation protection programs are conducted in accordance with this license.

<u>Nuclear Criticality Safety</u>

The following administrative controls are in effect for operations conducted in the Blending/Storage Area:

- The types and amounts of hydrogenous materials permitted in the Blending/Storage Area are specifically authorized by the Radiation Protection Component.
- Vessels are limited to safe geometry or safe volume except for process vessels and LSA waste containers.

1.9.5.1.3 (Continued)

- Before introduction of SNM into unsafe geometry containers, the containers are inspected to verify that they are free of hydrogenous materials and that the materials to be introduced are acceptable. This inspection is documen+ed and is verified by a second person.
- Only authorized individuals are permitted access to the Blending/Storage Area.
- When maintenance is performed on the Blending/Storage Area water barrier, procedural controls are instituted to limit operations until the barrier integrity is reestablished.

1.9.5.1.4 Powder Transport

Method of Transport

Bulk containers are transported using a hydraulic lift truck which is equipped with physical shields to protect the containers during transport to and from the Blending/Storage Area. Hydraulic fluid associated with the lift truck is physically shielded from the bulk container being transported and from other bulk containers in the Blending/Storage Area except where the volume of hydraulic fluid is less than 19 liters.

Bulk containers are transported, via lift truck, from the Blending/Storage Area to an appropriate pellet line for further processing. The following controls are in effect during bulk container transport: (1) the bulk containers are physically protected from damage during transport, (2) transport speed is limited to approximately 3 feet per second by a governor device, (3) the bulk containers are always transported with the feeder mechanism in the down position, (4) the feeder valve is closed and equipped with a water-tight gasketed cap in place, (5) during bulk container transports,

1.9.5.1.4 (Continued

the activities are subject to continuous operator surveillance, (6) except for maintenance operations on empty containers, bulk containers are transported directly from the Blending/Storage Area to the Pellet Area and returned, (7) all bolts used to seal the bulk containers are provided with lock wires or equivalent to assure that a water-tight closure is maintained.

Radiological Safety

During transport of the bulk containers, all uranium is contained. Consequently, there are no radiological safety concerns.

Nuclear Criticality Safety

Nuclear criticality safety is assured by maintaining the powder "dry" during both routine and credible accident conditions. Following powder loading, the bulk containers are gasket sealed prior to transport or storage. The bulk containers are then physically protected during transport.

1.9.5.1.5 Pellet Line Operations

Powder Transfers

Bulk containers in the Pellet Area are lifted by hoist to a support stand and placed within a secondary ventilated enclosure (the bulk containers are maintained in an inverted position, i.e., with the feeder in a down position, to minimize the possibility of water ingress). The bulk container feeder is then connected to a second feeder which transports powder, on demand, to a ventilated station for further processing.

Radiological Safety

All powder transfers are conducted within ventilated, HEPA filtered enclosures in accordance with license conditions.

1.9.5.1.4 (Continued)

Nuclear Criticality Safety

The bulk container and powder transport feeder form a closed system which provides physical protection against moderator ingress. The powder transport feeder consists of a 6-inch equivalent diameter conveyor which feeds a 10-inch equivalent diameter dump hopper and 10-inch equivalent diameter powder chute, which are interfaced with a ventilated hood, prior to discharge into a container within the hood. The bulk container feeder is interconnected with the transport feeder to prevent powder buildup in the event that the transport feeder stops. The dump hopper is equipped with a weight control system to automatically shut off the transport feeder and bulk container feeder when a given weight is reached. The hood is maintained under geometric controls.

1.9.5.2 Manufacturing Automation Project (MAP)

1.9.5.2.1 General Process Description

The MAP Line is designed to manufacture fuel--beginning with uranium hexafluoride--through inspected, loaded fuel rods. It uses the Integrated Dry Route (IDR) UF6 to UO2 conversion process. The conversion line consists of four (4) vaporizers, two (2) kiln systems, and three (3) blender systems.

The radiological safety, occupational safety and health, and much of the nuclear criticality safety, of processing operations in the MAP Line take advantage of the some 12 years of safe operating experience in the existing fuel fabrication facility. Except as addressed in this safety analysis, processing operations are as described in other portions of this license and include powder compaction, granulation, pressing, pellet sintering, grinding, fuel rod fabrication, and scrap reprocessing.

1.9.5.2 (Continued)

The described IDR (dry) conversion process relies on moderation control, as the primary nuclear criticality safety control, to a somewhat greater extent than previously used in the existing fuel fabrication facility; however, this process and its support systems have been thoroughly proven in commercial application with some years of safe operation at British Nuclear Fuels Limited (BNFL - Great Britain) and some 3 years of safe operation at Franco-Belge de Fabrication de Combustibles (FBFC - France). In these processes and systems, equipment geometries and uranium masses can exceed the Maximum Permissible Values (MPV) specified in this license; thus, the nuclear criticality safety of individual units, and arrays of such units, is primarily based upon rigorous control of moderating materials (including water, plastics, porosity control and binder/die lubricant materials for ceramic fuel, and lubricating oils). A systematic analytical approach is used to demonstrate application of the "double contingency principle" and to identify administrative and engineered controls required for safe operation with process-regulated, limited moderation systems.

Figure 1.9.5.1 is used to show IDR Conversion Process Equipment instrumentation, which has been incorporated into the design, to detect low temperature upset conditions, and to apply automatic actions well before potentially unsafe conditions could develop. The applicable interlocks are:

Interlock: Low superheated steam temperature shuts off steam feeds to kilns (Disentrainment Chamber 1 and Kiln barrel 6).

Control Instruments: Steam superheater temperature control and thermocouple at superheater outlet; steam flow control and safety shutoff valves in series; UF6 flow control and safety shutoff valves in series; H2 flow control and safety shutoff valves in series.

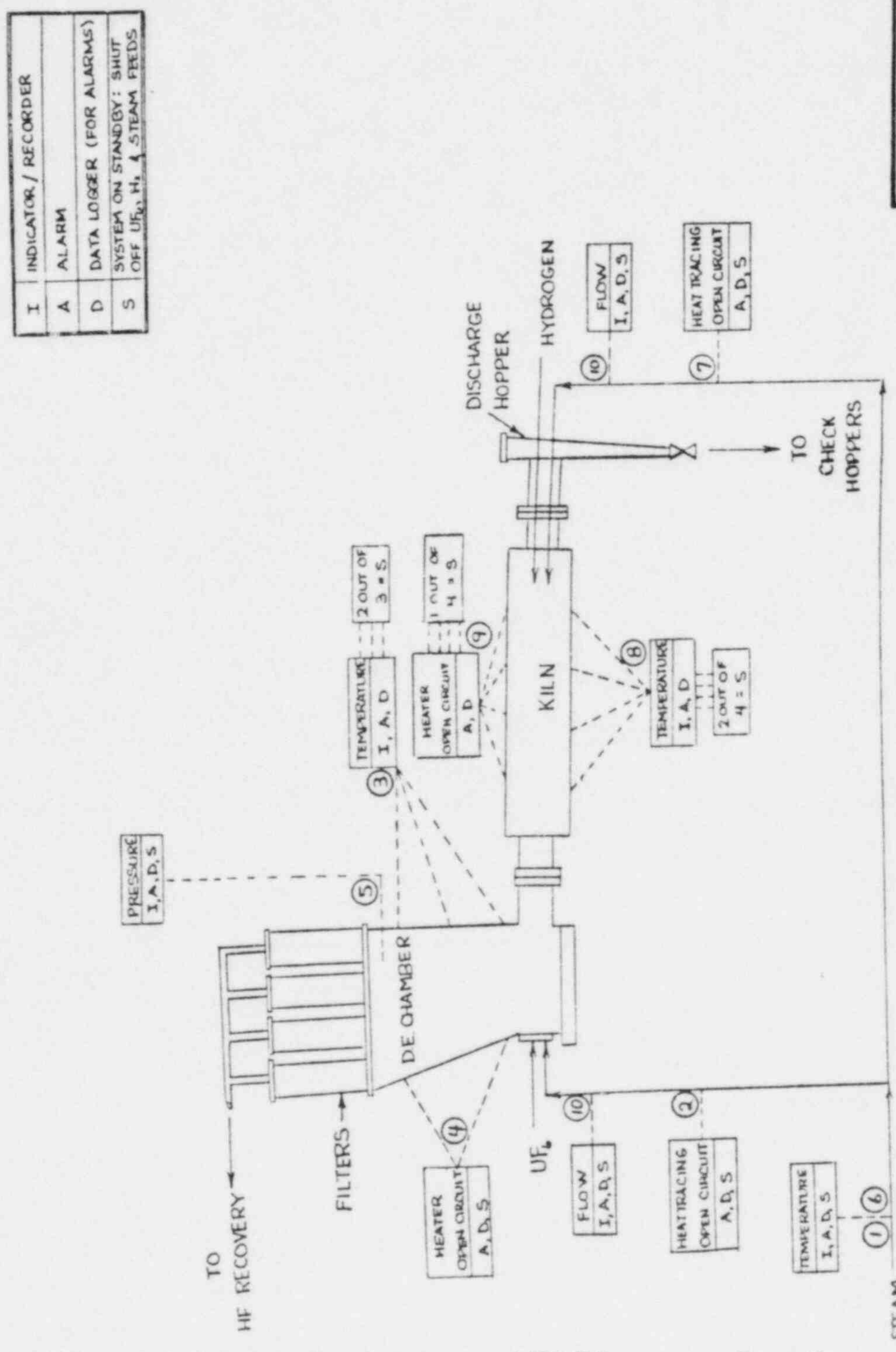


FIGURE 1.9.5.1

Reference:
2.2.14 IDR Conversion Process Equipment

1.9.5.2 (Continued)

Priority Alarm: Yes - signal of superheater temperature control activates audible and visual alarms; signal of outlet thermocouple activates alarms and initiates automatic actions.

Automatic Actions: Low steam temperature at superheater outlet initiates automatic actions to shut off steam, H2, and UF6 flows to both kilns and to introduce nitrogen purge flows to the kilns.

Required for Startup: Yes - operational steam temperature required before initiating gas flows to kiln.

Required for Operation: Yes - operational steam temperature required.

Reportable Event: Yes - failure of interlocks to shut off steam to an affected system, when called upon by system conditions to do so, must be promptly reported to NRC Region II.

Interlock: Failure of heat tracing on steam line to Disentrainment Chamber jet shuts off steam feeds to Kiln 2.

Control Instruments: Heat tracing control loop thermocouple and two heat tracing burnout protection thermocouples; steam flow control and safety shutoff valves in series; UF6 flow control and safety shutoff valves in series; H2 flow control and safety shutoff valves in series.

Priority Alarm: Yes - signal of control loop thermocouple activites audible and visual alarms; signal of either burnout protection thermocouple activates alarms and initiates automatic actions.

Automatic Actions: Signal of either burnout protection thermocouple initiates automatic shutoff of steam, UF6, and H2 flows to kiln and introduces nitrogen purge flow to kiln.

Required for Startup: Yes - operational steam line heat tracing required.

1.9.5.2 (Continued)

Required for Operations: Yes - operational steam line heat tracing required.

Reportable Event: No.

Interlock: Low Disentrainment Chamber temperature shuts off steam feeds to Kiln 3 .

Control Instruments: Three thermocouples: two measuring and controlling chamber heat tracing, and one measuring internal temperatures; steam flow control and safety shutoff valves in series; UF6 flow control and safety shutoff valves in series; H2 flow control and safety shutoff valves in series.

Priority Alarm: Yes - low temperature signal by any one of the three thermocouples activates audible and visual alarms.

Automatic Actions: Low chamber temperature registered by any two of the three thermocouples initiates automatic shutoff of steam, UF6, and H2 flows to kiln and replacement of these flows with nitrogen purges.

Required for Startup: Yes - operational chamber temperature required to initiate steam, UF6, and hydrogen flows.

Required for Operation: Yes - operational chamber temperature required.

Reportable Event: Yes - failure of interlocks to shut off steam to an affected system, when called upon by system conditions to do so, must be promptly reported to NRC Region II.

Interlock: Failure of Disentrainment Chamber heater shuts off steam feeds to Kiln 4 .

Control Instruments: Heater control loop thermocouples and heater burnout protection thermocouples (1 per heater); steam flow control and safety shutoff valves in series; UF6 flow control and safety shutoff valves in series; H2 flow control and safety shutoff valves in series.

Priority Alarm: Yes - signal of either of the two heater control loop thermocouples activates audible and visual

1.9.5.2 (Continued)

alarms; signal of either burnout protection thermocouple activates alarms and initiates automatic actions.

Automatic Actions: Signal of either heater burnout protection thermocouple initiates automatic shutoff of steam, UF6, and H2 flows to kiln and replacement of these flows with nitrogen purges.

Required for Startup: Yes - operational chamber heaters required to initiate gas flows.

Required for Operation: Yes - operational chamber heaters required.

Reportable Event: No

Interlock: High Disentrainment Chamber pressure shuts off steam feeds to Kiln 5.

Control Instruments: Pressure transmitter; steam flow control and safety shutoff valves in series; UF6 flow control and safety shutoff valves in series; H_2 flow control and safety shutoff valves in series.

Priority Alarm: Yes - high chamber pressure signal activates audible and visual alarms.

Automatic Actions: Detection of high chamber pressure initiates automatic shutoff of steam and UF6 flows to kiln and replacement of these flows with nitrogen purges.

Required for Startup: Yes - normal chamber operating pressure required.

Required for Operation: Yes - normal operating pressure required.

Reportable Event: Yes - Failure of interlocks to shutoff steam to an affected system, when called upon by system conditions to do so, must be promptly reported to NRC Region II.

Interlock: Failure of heat tracing on steam line to kiln barrel probe (steam poker) shuts off steam feeds to Kiln 7 .

1.9.5.2 (Continued)

Control Instruments: Heat tracing control loop thermocouple and two heat tracing burnout protection thermocouples; steam flow control and safety shutoff valves in series; UF6 flow control and safety shutoff vales in series; H2 flow control and safety shutoff valves in series.

Priority Alarm: Yes - Signal of control loop thermocouple activates audible and visual alarms; signal of either burnout protection thermocouple activates alarms and initiates automatic action.

Automatic Actions: Signal of either burnout protection thermocouple activates automatic shutoff of steam, UF6, and H2 flows to kiln and introduces nitrogen purge flow to kiln.

Required for Startup: Yes - operational steam line heat tracing required.

Required for Operation: Yes - operational steam line heat tracing required.

Reportable Event: No

Interlock: Low kiln barrel temperature shuts off steam feeds to Kiln (8).

Control Instruments: Four furnace heater control loop thermocouples (1 per zone) measuring and controlling kiln barrel temperature; steam flow control and safety shutoff valves in series; UF6 flow control and safety shutoff valves in series; H2 flow control and safety shutoff valves in series.

Priority Alarm: Yes - low temperature signal by any 1 of the 4 thermocouples activates audible and visual alarms.

Automatic Actions: Low temperature registered by any two of the four barrel zones initiates automatic shutoff of steam, UF6, and H2 flows to kiln and replacement of those flows with nitrogen purges.

Required for Startup: Yes - operational kiln furnace temperatures required to initiate gas flows.

1.9.5.2 (Continued)

Required for Operation: Yes - operational kiln furnace temperatures required.

Reportable Event: Yes - Failure of interlocks to shut off steam to an affected system, when called upon by system conditions to do so, must be promptly reported to NRC Region II.

Interlock: Failure of kiln barrel heater shuts off steam feeds to Kiln (9).

Control Instruments: Furnace heater control loop thermocouples (1 per zone), and 4 burnout protection thermocouples (1 per heater); steam flow control and safety shutoff valves in series; UF6 flow control and safety shutoff valves in series; H2 flow control and safety shutoff valves in series.

Priority Alarm: Yes - failure of any 1 of the 4 control loop thermocouples activates audible and visual alarms; failure of any 1 of the 4 burnout protection thermocouples activates alarms and initiates automatic actions.

Automatic Actions: Failure of any 1 burnout protection thermocouple initiates automatic shutoff of steam, UF6 and H2 flows to kiln and replacement of these flows with nitrogen purges.

Required for Startup: Yes - operational kiln barrel heaters required to initiate gas flows.

Required for Operation: Yes - operational kiln barrel heaters required.

Reportable Event: No

Interlock: Loss of steam flow shuts off U6 and H2 flows to Kiln (10) .

Control Instruments: Mass flow meter on each steam line to kiln; steam flow control and safety shutoff valves in series; UF6 flow control and safety shutoff valves in series; H2 flow control and safety shutoff valves in series.

1.9.5.2 (Continued)

Priority Alarm: Yes - loss of either steam flow activates audible and visual alarms and initiates automatic actions.

Automatic Actions: Low steam flow to either disentrainment chamber jet or kiln barrel probe (steam poker) initiates automatic shutoff of any remaining steam flow and of UF6 and H2 flows, and replacement of these flows with nitrogen purges.

Required for Startup: Yes - operational steam flow rates required before H2 flow begins; steam and H2 flows required before UF6 flow begins.

Required for Operation: Yes - operational steam flow rates required.

Reportable Event: No

Aspects of the MAP Line addressed in this safety analysis include: (1) the "water spray heating and emergency cooling" portion of vaporization; (2) conversion (i.e., all operations inside the "water barrier," plus HF recovery); and (3) the "die lubricant mixer" portion of fabrication.

1.9.5.2.2 Conversion Area Operations (Outside the Water Barrier)

UF6 Vaporization

UF6 cylinders, containing uranium enriched up to 5 weight percent (w/o) in U-235, are heated to processing temperature in vaporizer chests, by use of circulating hot water sprays. A vaporizer chest is connected to its liquid reservoir by downcomers which provide emergency hydrolysis of released UF6 and pressure relief for the chest interior. A cold water spray system is provided to enable emergency cylinder cooling.

Recovery of HF for Resale

The conversion kiln offgas is cooled to recover byproduct hydrofluoric acid by condensation.

Offgas from each conversion kiln flows to its own individual cylindrical condenser in which the gases are cooled and HF is recovered as aqueous solution condensate. The condensate is

1.9.5.2.2 (Continued)

drained continuously from the condenser (alternatively) into a pair of rectangular HF quarantine ("Q") receiver-sampling tanks in which the condensate is held for analysis of uranium compounds that could result from incomplete reaction in the kiln or from carryover of solids through failed process filters.

Acceptable low ($\leq$10 ppm) uranium content HF solution is pump transferred to a storage/loadout tank; unacceptable high (>10 ppm) uranium conent HF solution is transferred to a rectangular neutralization tank for adjustment with ammonium hydroxide prior to pump transfer to the existing facility liquid effluent treatment system for recovery of uranium.

Condenser offgas containing hydrogen, nitrogen, and traces of HF and water vapor is transferred to a condenser offgas scrubber system. From within the HF recovery area, HF vessel vent gases containing traces of HF are transferred to an HF vent scrubber system.

Radiological Safety

Low-enriched uranium in the closed vaporization system requires only those routine radiological safety precautions that have become a part of existing facility practice.

Emergency UF6 cylinder cooling is provided by a manually-controlled spray system connected to an external plant cooling water supply.

A vaporizer chest is connected to its integral liquid reservoir through downcomers which extend below the surface of liquid in the reservoir, thus enabling any gases or vapors from the vaporizer chest to be bubbled through the liquid reservoir to the gas space of the vaporizer safe sump, for hydrolysis of inadvertently released UF6, before venting through an HF vent scrubber system and HEPA filtration prior to release from the facility.

1.9.5.2.2 (Continued)

HF recovery operations are designed such that significant quantities of uranium are excluded from the recovery system. The uranium hexafluoride and steam supplies on each individual kiln are alarmed and interlocked by a high integrity control sytem. In the event of steam failure, the gaseous UF6 feed to the system is automatically shut off. Each kiln system is fitted with high integrity blow-back process filters, two filtration units in series, to remove entrained uranium compounds from the system offgases; filter integrity is continuously monitored for high and low differential pressure, and is alarmed and interlocked to automatically shut off UF6 and steam feed to the system in the event of process filter failure.

As a further safeguard, a continuous uranium-in-liquid monitor is included in each condenser system, which is also alarmed and interlocked to shut off UF6 and steam feed to the kiln system if uranium enters the HF recovery system. Recovered HF is quarantined for uranium analysis before release to an outside HF storage/loadout tank if uranium content is less than or equal to 10 parts per million, or, to a rectangular neutralization tank in the HF recovery area if uranium content is greater than 10 parts per million.

Process gases from the condenser offgas and HF vent scrubber systems are HEPA filtered prior to release from the facility.

Nuclear Criticality Safety

All process vessels and equipment (except for the UF6 cylinders already evaluated in this license) in the vaporization and HF recovery areas, normally containing uranium or having the potential to contain uranium as a result of accident conditions, are designed safe by geometry for 5 w/o U-235 materials under full reflection and optimum moderation, and are design spaced in accordance with criticality safety criteria approved elsewhere in this license.

1.9.5.2.2 (Continued)

During vaporization heating or emergency cooling operations, UF6 cylinders are subject to water sprays only. Engineered normal operation overflows, and engineered emergency cooling water volume controls, keep the cylinders from being immersed in water.

The lower portion of the vaporizer chest is designed to provide a liquid reservoir of slab geometry less than 5.2 inches depth. This liquid reservoir is normally pumped to the existing facility liquid effluent treatment system; however, a conductivity cell inside each vaporizer will provide early warning of any UF6 leakage, from resultant chest reservoir liquid contamination, and will divert the vaporizer condensate to a remote vaporizer safe sump. · This diversion system is designed to provide an unimpeded condensate flow path to either the liquid effluent treatment system (normal operation) or the vaporizer safe sump (emergency operation). The vaporizer safe sump is a closed floor unit designed to contain uranium in liquids in a safe slab geometry of less than 5.2 inches depth. Further, the vaporizer safe sump is arranged to also act as an emergency vent duct to carry uranium in gases (as a result of any leakage from UF6 vessels, vaporizers, and/or connective piping) to an HF vent scrubber; and to provide the necessary holdup for anlaysis of any liquids for uranium content, prior to potential transfer to unfavorable geometries.

Since UF6 cylinders are subject to regular test and inspection, it is not considered likely that a major failure would occur under operating pressure; however, on the conservative hypothesis that a minor failure might occur, the low holdup of water in a vaporizer chest's safe slab geometry liquid reservoir (with the described emergency interlock transfer to the vaporizer safe sump) provides contingent

1.9.5.2.2 (Continued)

nuclear criticality safety control. A high conductivity indication automatically shuts off hot water spray to the system, to enable manual introduction of the cold water spray. During subsequent cylinder cooling, a negative (condensation) pressure could develop in the vessel, but the maximum differential pressure would be low. This might lead to the entry of some water into the cylinder in the event of a failure, but a significant influx is unlikely for the following reasons: (1) even on a most conservative assumption of failure it is likely that this would occur only at a weld joint, or in connective apparatus, and hence would be small in total area; (2) manual emergency spray cooling is used to reduce the amount of water likely to be drawn in; (3) a positive pressure would likely develop due to production of HF gas from the reaction between UF6 and water; and (4) provision is made for introducton of nitrogen to relieve condensation vacuum.

Nuclear criticality safety within the HF recovery area is enhanced by radiological safety design features, which are designed such that only very small quantities of uranium can enter the sensitive items of recovery equipment, and which provide backup uranium-in-liquid monitors to give an early warning of any such uranium entry into the recovery system.

Additional nuclear criticality safety controls arise from the design, dimensions, and operation of the sensitive recovery equipment items. The cylindrical (less than 9.0-inch-diameter) condenser is not designed to hold HF; the condensate gravity-drains to the HF "Q" tanks as it forms. The spaced slab (less than 4.3-inch-thick) HF "Q" tanks are each designed for sufficient holdup time to permit uranium analysis before pump transfer of collected acid, either to storage loadout or to neutralization and uranium recovery;

1.9.5.2.2 (Continued)

each tank is equipped with vent lines which are arranged to act as overflows to the alternate "Q" tank. The neutralization tank is a safe slab (less than 4.3 inches thick).

1.9.5.2.3 Conversion Area Operations (Inside the Water Barrier)

Dry Conversion of UF6 to UO2

Reactants, consisting of gaseous UF6 from the vaporization process and superheated steam, are continuously introduced to the slab-shaped disentrainment chamber of a conversion kiln. Here, the UF6 vapor and superheated steam undergo dry hydrolysis to form uranyl fluoride (UO2F2) powder and hydrogen fluoride (HF) gas. The UO2F2 powder falls to, and is continuously removed from, the bottom of the chamber and slowly tumbles down a slightly slanted, rotating cylindrical kiln barrel. Hydrogen gas and superheated steam are continuously introduced at the product discharge end of a kiln. In the kiln barrel the powder is further contacted with a countercurrent flow of hydrogen and steam, to strip the remaining fluoride and reduce the uranium to uranium dioxide (UO2) powder.

Off-gases (hydrogen, hydrogen fluoride, nitrogen, and steam) are continuously removed from the kiln system through eight cylindrical process filters at the top of the disentrainment chamber, for temporary retention and subsequent nitrogen blowback of uranium powder. Hydrogen concentration in nitrogen is below the lower explosive limit prior to mixing with air in the ventilation system; thus, no additional treatment is required for industrial safety.

A conversion kiln is maintained under a controlled positive pressure, up to about 12 inches of water, by means of the ventilation off-gas system. Nitrogen-purged lip seals at each end of the kiln barrel (seal pressure > kiln pressure) enable rotation without loss of confinement. Hydrogen detection

1.9.5.2.3 (Continued)

monitors are located above the kiln barrel seals to sample for potential leaks of kiln gases; these monitors will initiate an alarm and activate a safe-shutdown interlock, if hydrogen is detected. The interlock which shuts off uranium and steam input to a kiln system also automatically shuts off hydrogen input, and introduces nitrogen purge, to the system. The conversion kiln system is operated with zone-controlled temperature conditions (the hydrolysis reaction is highly exothermic and takes place in the disentrainment chamber at temperatures from 450 to 500^{o}F; the reduction reaction takes place in the remainder of the kiln barrel at temperatures in excess of 900^{o}F). Excess pressure alarms and interlocks (for nuclear criticality safety), and excess temperature alarms and trips, also enhance industrial safety.

Uranium dioxide powder is removed continuously from the product discharge end of a kiln, into one of a pair of cylindrical U02 check hoppers. Provision is made for continuously purging the U02 check hoppers with dry nitrogen during filling and analytical hold time. Product entering a U02 check hopper is continuously sampled by a proportional sampler. The samples are analyzed for acceptable quality with respect to moisture content. Acceptable product is discharged to a pneumatic line for direct (dry nitrogen) transfer to powder preparation processes; powder in a check hopper found to be unacceptable is discharged to special geocon containers for transfer to existing facilities for recovery of uranium. Bulk powder (modcon) containers are also provided as a contingency for transfer of acceptable product to powder preparation processes.

Uranium Powder Transport, Blending, and Storage

The process uses bulk powder processing techniques for homogenization of U02 and U308 powders with porosity control additive powder to produce powder blends for pellet

1.9.5.2.3 (Continued)

fabrication; the process can also utilize bulk powder processing techniques for enrichment blending of UO2 and U3O8 powders. The basic process system consists of: a pneumatic (dry nitrogen) transport system which conveys acceptable UO2 powder from the kiln check hoppers to a blender inlet, and a pneumatic (dry air) transport system which conveys homogenized powder from a blender outlet to a pellet line feed hopper; conical 5000 Kg orbital screw blenders for powder homogenization; and moderation controlled (modcon) 500 Kg bulk containers for transport and storage of U3O8 recycle material, and also for contingency backup to the UO2 pneumatic transport system.

The system is designed such that UO2 powder is pneumatically transported from the kiln check hoppers, which are mounted on load cells for weight monitoring, to a filtered receiver vessel and into a blender. As previously stated, a contingency backup system is provided for this powder to be collected in modcon containers which can be filled, weighed, identified and transported to the blender area. UO2 powder from other sources outside the the MAP Line, such as the existing facility scrap recovery areas, could also be introduced by this system. Transfer and handling of bulk containers are in accord with applicable lifting device safety practices. An area is provided for bulk container storage. UO2 powder collected in modcon containers can also be transported to the blender, via the filtered receiver vessel, using a UO2 unloading station and the pneumatic transport system. Pneumatically transferred UO2 powder is disengaged from the dry nitrogen transport gas in a filtered receiver vessel. The powder is then rate-fed, by a vibratory feeder, into the blender for homogenization. When the backup container system is utilized, the filled and weighed UO2 container can be mounted on a blender for discharge. The

1.9.5.2.3 (Continued)

container port is connected to the blender system using a flexible connector seal. The powder is then rate-fed, by a vibratory feeder, into the blender. When a predetermined amount (minimum working capacity) of UO2 powder has been collected in a blender, operation is initiated. The homogenization operation is conducted in an atmosphere of dry nitrogen. An "off-off" cycle of the blender is used; each period of rotation is followed by a rest period. The dry nitrogen prevents oxidation of the uranium powder at temperatures induced by the mechanical energy input; the "on-off" cycling of the blender prevents excessive powder temperatures. Alarms and interlocks in the blender rotation system will automatically prohibit blender operation upon detection of low blender purge gas flow, high blender temperature, or high blender pressure. Modcon containers, filled with U3O8 powder from the geometry controlled clean scrap reprocessing system (adjacent to the water barrier) are also weighed, identified and transported to the blender area. These containers are kept in the bulk container storage area. U3O8 powder is fed into the blender by the same procedure previously described for the UO2 powder backup container system.

Porosity control blending additive is kept in a carefully controlled storage area, outside the water barrier, awaiting transaction for preparation and use in subsequent powder blending operations. The additive is stored in sealed containers when outside of ventilated containment or process vessels; the additive preparation station, and the additive rate feeder to the blender, are enclosed within ventilated containment. At the additive preparation station, the porosity control agent (which has previously been sampled, analyzed, milled, and sieved; and, which has had silica base flow promoting agent mixed in) is batch-weighed as preparation

1.9.5.2.3 (Continued)

for blender loading. This additive is transferred into a specially designed feed hopper above the blender and the additive is then charged intermittently to the blender during the subsequent homogenization operation.

When powder homogenization is complete, representative samples of the resultant mix are taken to ancillary service areas in the existing facility and are quality control evaluated for physical properties, chemical assay, and verification of isotopic enrichment. Acceptable product is released for subsequent fabrication operations, based on results of the blend analytical parameters. The homogenized powder can be fed to an intermediate storage hopper; then rate-fed, by vibratory feeder, to an aerated transition pad for (dry air) pneumatic transport to the pellet line feed hopper. Alternately, the powder can be discharged from the blender system directly to bulk containers for storage and/or subsequent processing, or into appropriate containers for off-site shipment. A powder probe is mechanically inserted into the blender to assure the blender has emptied.

The blender system can also be used to mix and transfer batches of uranium powders (UO2 and U3O8) of different enrichments up to 5 w/o U-235, using applicable procedures previously described for powder homogenization. Upon completion of enrichment blending, the mix is quality control sampled and analyzed. This blended powder can also then be prepared for release to the pelleting area, or can be discharged to bulk containers for storage and subsequent processing, or placed into appropriate containers for off-site shipment.

Radiological Safety

Low-enriched uranium in the closed dry conversion and powder blending systems requires only those routine radiological safety precautions that have become a part of existing

1.9.5.2.3 (Continued)

facility practice. Uranium powder storage containers are fitted with efficient sealing lids which are kept tightly closed unless the container is in a fill or discharge position. Spill prevention alarms and interlocks for nuclear criticality safety, and excess temperature and pressure alarms and interlocks previously described for occupational safety and health, also enhance radiological safety. At each end of the kiln, nitrogen-purged lip seals are used to allow rotation of the barrel without loss of process gases or solids. A low-pressure monitor is fitted on the primary section of each seal unit and seal purge flows are continuously monitored; and the kiln barrel has a low pressure monitor. Given a fault condition, these monitors will initiate a safety-related process alarm for corrective actions to prevent release of kiln gases to room atmosphere.

The following processing sections of the dry conversion and blender systems are enclosed within ventilated containment: uranium container loading and unloading stations, uranium vibratory feeders, and inspection and sample ports.

UF6 vapor-bearing lines to the disentrainment chamber are enclosed within containment, maintained at a negative pressure with respect to the operating area, and are passed through the HF vent scrubber system and HEPA filtration prior to release from the facility. Kiln system process offgas passes through, and is treated by, the HF recovery system previously discussed. Blender process off-gas (dry nitrogen purge) passes through a roughing filter followed by a HEPA filter, prior to discharge from the facility. Transport off-gases (dry nitrogen) from the blender receiver vessel, and transport off-gases (dry air) from the pellet line feed hopper, are passed through sintered metal filters contained in the respective vessels. The filtered gas then flows through a HEPA filter prior to release from the facility.

1.9.5.2.3 (Continued)

Nuclear Criticality Safety

Moderation Controls - The described conversion and blending operations rely principally on moderation control for nuclear criticality safety. Operations are designed such that multiple contingencies--such as process upsets, accidents, and loss of administrative control--must occur before conditions for a criticality accident are possible. The systematic analytical approach used to evaluate the conversion and blending operations postulated unsafe conditions which could result in sufficient moderation getting into sufficient enriched uranium in an unsafe geometrical configuration, and identified the controls to be applied to prevent these events from occurring. At least double-contingency criteria were satisfied for each potential unsafe condition.

A substantial part of the evaluation identified controls to exclude uncontained moderators from the moderation controlled area; another part identified controls to prevent uranium spills to this area. It is notable that these "moderator in area" and "uranium spill" scenarios would also have to happen within a short time of each other in order for the potential criticality condition to occur (otherwise, administrative procedures requiring prompt cleanup corrective actions would also have to fail).

Because superheated dry steam is introduced into the IDR conversion process, and gaseous HF is produced in the subsequent reactions, system temperatures are carefully maintained at elevated levels to keep these hydrogenous compounds in the gaseous phase--so they do not act as significant moderators. Nuclear criticality safety control in the process equipment is largely based on instrumentation incorporated into the design to detect low temperature upset conditions, and to automatically apply prompt corrective actions well before potentially unsafe conditions could

1.9.5.2.3 (Continued)

develop. Such temperature safety alarm and interlock circuits have redundant thermocouples. If a single thermocouple fails, a priority audible and visual alarm is actuated; if two or more thermocouples fail, the controlled system automatically goes to safe shutdown. Further, such safety interlock valves are designed to provide a redundant shutoff function independent of respective process control valves; that is, in a steam line, separate block valves-in series--are used for process control and for safety shutoff.

Because hydrogenous porosity control additive is introduced into the blending process, administrative and engineered controls are applied to procure and verify the correct additive, to limit the maximum quantity of additive introduced to a single batch, and to rate-feed the additive during blending to minimize local concentrations. The supplier is requested to clearly identify additive provided; then when the additive is received a chemical analysis is performed to confirm the identification. The additive is stored in a controlled area outside the water barrier; an administratively controlled transaction takes place prior to moving a batch into the moderation controlled area. The additive feed hopper is specifially designed to hold no more than single batch. Blender operation logic is interlocked such that, once charged with a single additive batch and closed, the feed hopper cannot be reopened until the hopper has emptied and subsequent total blender discharge is complete. (The blender was modeled in KENO as an optimized configuration which minimizes neutron leakage from the system while maximizing the effect of reflectors (both water and concrete) external to the system. Additional model input included 5 w/o U-235 enriched uranium with a maximum expected moisture content, and a double batch of porosity control additive. Further, the uranium and moderator were modeled to mix in a sphere sized to result in

1.9.5.2.3 (Continued)

optimum moderation of the UO2, with the sphere located in a position to maximize the reflection by surrounding dry UO2. The calculated keff was 0.948 ± 0.008.)

Because the blender gear box is internal to the system, administrative controls are applied to procure and verify a nonhydrogenous lubricant. The supplier is requested to clearly identify the nonhydrogenous lubricant provided; then when the lubricant is received a chemical analysis is performed to confirm the identification. The Radiation Protection Component specifically authorizes initial lubricant loading, and subsequent lubricant loadings, internal to the system.

Ancillary Controls - Specific attention has been paid to processing operations at interfaces with the water barriers, and at the interface of the high-temperature kiln operation and the room-temperature check hoppers. Also, discharge to and from containers, and container storage, has been specifically addressed.

Design and layout of equipment in the HF recovery area enhances nuclear criticality safety of the disentrainment chambers; because, controlled vessel overflows and level alarms prevent the HF system from becoming solid with aqueous solution and transfer lines have siphon breaks to prevent aqueous solution backflow through the system. Because of the previously evaluated multiplicity of "dryness" controls on powder formed in the disentrainment chamber, it is very unlikely that significant quantities of moderated material could be transferred to the heated process blowback filters at the top of the chamber.

Further, it is even more unlikely that moderation of powder already within these cylindrical filters could occur, since this would require a failure of the dry nitrogen process blowback system in addition to moderation by faulty

1.9.5.2.3 (Continued)

disentrainment chamber products. (The filter array was modeled on KENO as filled with 5 w/o U-235 enriched uranium having optimum moderation and nominal (1-inch graphite) reflection. The calculated keff was 0.695 $\pm$ 0.012.)

Because of the substantial impairment of IDR powder flow properties at even very low moisture content, it is unlikely that significant quantities of moderated material could be transferred to the check hoppers. Moderation of powder already within the (unheated) hoppers could only arise from the back-diffusion of the kiln barrel probe steam supply via the discharge end of the kiln. This is a most unlikely event since the steam flow would have to overcome the resistance of the main (kiln) hydrogen gas flow as well as the nitrogen purge which is fed through the base of the hopper. However, as an additional contingency, each cylindrical check hopper is fitted with a stainless steel annular insert containing boron carbide, which runs coaxially within the hopper and serves as a nuclear poison. (This check hopper arrangement was modeled in KENO as filled with 5 w/o U-235 enriched uranium having optimum moderation and nominal (1-inch water) reflection. The calculated keff was 0.927 $\pm$ 0.014.) Administrative procurement controls, and inspections by the Radiation Protection Component, are used to confirm the poison rods are initially placed in check hoppers. The rods are inspected for damage during periodic check hopper maintenance. Appropriate engineered and administrative controls (including postings, procedures, and Radiation Protection Component inspection) are used to confirm the poison rods are replaced following check hopper maintenance.

Only uranium powder that has been sampled, analyzed, and confirmed dry (H/U $\leq$0.3) is released for transfer from geometry-controlled containers into moderation controlled containers. As a contingency to this analysis, the following

1.9.5.2.3 (Continued)

alarms and interlocks in the check hoppers and the scrap reprocessing product receiver vessel will automatically prevent discharge of the vessels upon detection of the indicated fault:

1. Moisture in the vessel purge gas (including four alarms and interlocks in the dew point analyzer protection system: (1.1) low gas flow to the analyzer, (1.2) low gas flow to the analyzer filter blowback system, (1.3) failure of the analyzer filter blowback system timer, and (1.4) loss of electrical continuity in the analyzer circuit); or
2. Low vessel purge gas flow.

Engineered controls have been designed to prevent discharge of accidently moderated check hopper powder to anything but geometry-controlled containers, and to prevent container discharge to moderation controlled processing equipment from anything but moderation controlled containers. The interface between check-hopper discharge and container receiving is designed to "recognize" only specially made geometry-controlled containers, before allowing manual override of the interlock preventing moderated material discharge. The interfaces between container discharge and moderation controlled process equipment receiving are designed to "recognize" only specially made moderation controlled containers, before allowing container discharge.

Filled UO_2/U_3O_8 moderation controlled containers are stored in a one-high planar array with a minimum of one foot edge-to-edge container separation, thus they require no interaction considerations; in powder storage areas, engineered controls maintain the required spacing. The cylindrical ($\leq$9-inch diameter) geometry-controlled containers potentially containing moderated material, are removed from

1.9.5.2.3 (Continued)

the moderation controlled area as they are filled; they are not stored inside the water barrier.

1.9.5.2.4 Fabrication Area Operations

Except for the following processing step, MAP Line fabrication area operations are as described in other portions of this license for the existing facility.

Uranium and Die Lubricant Powder Mixing

Uranium powder granules are proportionally mixed with powdered die lubricant (such as stearic acid), in a urethane-lined vibratory machine, to produce pellet press feed material.

Radiological Safety

Low-enriched uranium in the closed torus bowl vibrator requires only those routine radiological safety precautions that have become a part of existing facility practice.

Nuclear Criticality Safety

The mixer was modeled in KENO as an optimized configuration which minimizes neutron leakage from the system while maximizing the effect of reflectors (both water and urethane) external to the system. Additional model input included fully moderated 5 w/o U-235 enriched uranium (to account for the maximum credible hydrogen content from a mixture of intrinsic moisture, porosity control additive, and stearic acid). Further, the uranium and moderator were conservatively modeled to mix in a rectangular volume sized 8.6 percent greater than the maximum physical dimensions of the roller's toroid volume. The calculated keff was 0.935 ± 0.014.

1.9.6 Incinerator

1.9.6.1 Purpose of Incineration

Combustible waste may be treated by incineration to reduce the volume of waste disposed to licensed burial grounds and to permit the recovery of SNM when practical.

1.9.6.2 Typical Materials to be Incinerated

Typical materials are paper, plastic shoe covers, gloves, mops, plastic bags, tape, fiberboard containers, burnable liquids, etc. from licensee's various laboratories and fuel fabrication processes.

1.9.6.3 Incinerator System Location

The incineration system is located in the Contamination Controlled Area of the Westinghouse Columbia Plant. The specific location is shown in Figure 1.9.0.1.

1.9.6.4 Incinerator System Description and Operation

The incinerator is a controlled air, dual chambered, gas fired unit with two chambers. The ignition chamber (lower) operates at approximately 1300^{O}F and the (upper) combustion chamber operates at approximately 1600^{O}F. In addition, the ignition chamber is equipped with a combustible liquid burner.

There is also a continuous Ash Removal System opening at the rear of the ignition chamber. Exhaust gases leaving the upper combustion chamber are transferred into a Quench Tower.

The exhaust gases are sprayed with conditioned water and are condensed and cooled. This condensed liquid is adjusted for pH and reused in the gas scrubbing system.

Material movement through the incineration system is described as follows:

(1) Boxes of contaminated waste, with the amount of contamination recorded, are delivered to the incinerator.

(2) The boxes are fed into the incincerator feed system.

(3) The incinerator feed door raises automatically as a hydraulic RAM pushes the box into the ignition chamber of the incinerator.

(4) The feed system can be adjusted from four (4) boxes to twelve (12) boxes per hour.

(5) Live steam may be injected into the chamber to assist control of the combustion process.

1.9.6.4 (Continued)

(6) The gaseous products of the combustion at approximately 1600°F enter the heat recovery boiler where the temperature is lowered to approximately 600°F and then to the Quench Tower where the temperature is further reduced to approximately 160°F.

(7) The gases pass through a Venturi Scrubber section and into a HCl Acid Stripper (packed column).

(8) The gas flow then travels through a condenser and any liquid removed is returned to the scrubber pump.

(9) The dewatered exhaust gases are reheated by the duct heater before entering the HEPA filter house. All duct work between the condenser and the exhaust blower is heated. The top, bottom and sides of the HEPA filter house are also heated.

(10) The exhaust blower is mounted in the second floor equipment room with a stack up through the roof. An isokinetic air sampling probe measuring particulate uranium is installed above the blower. A small backup blower is installed in parallel with the primary blower. The backup blower will only operate when the primary blower fails. The backup blower is only to permit an orderly shutdown.

1.9.6.5 Radiological Safety Control

The incinerator system is installed within the Contamination Controlled Area of the plant. Only authorized personnel are allowed into this area. Operating personnel are required to submit to the bioassay program for routine urinalyses. Lung burden determinations (Subparagraph 3.2.4) and the use of external radiation exposure monitoring devices (Subparagraph 2.2.3) may also be required.

An isokinetic air sampler, described in Subparagraph 1.9.6.4 is installed downstream of the HEPA filters in the

1.9.6.5 (Continued)

incinerator filter house. This air sampler continuously collects samples representative of the exhaust effluents discharged to the atmosphere. The samples are analyzed daily during operations. If the exhaust effluent at the point of release reaches a level of 1×10^{-12} microcuries per milliliter, an investigation will be made and the results evaluated. If the investigation reveals that 4×10^{-12} microcuries per milliliter may be exceeded as an annual average concentration, the cause will be determined and corrective action taken. These release and action limits are consistent with those applied to all other exhaust effluents.

Effluent exhaust concentrations from the existing incinerator have averaged less than 20 per cent of MPC.

Air sampling requirements for the incincerator are evaluated in accordance with Subparagraph 2.2.6 and 3.2.3 of this license. Permanently mounted continuous air sampling stations are established where the greatest concentration of airborne activity is expected under adverse circumstances and consistent with operator work locations during the initial loading operations, and during ash removal operations.

The exhaust blower maintains the combustion chambers at a negative pressure with respect to atmospheric. Under normal conditions, the combustion chamber operates at a negative pressure of 0.1" H2O or more. The exhaust system is maintained (filter change, scrubber maintenance, etc.) to assure that this minimum negative pressure drop of 0.1" H2O is maintained. This ensures adequate containment since the combustion chambers are relatively air tight to assure that a proper combustion atmosphere is retained. Minimum instrumentation and controls are described in Subparagraph 2.2.12 of this license. Incinerator ashes are continuously transferred from the combustion chamber directly to a ventilated enclosure. This is performed within containment

1.9.6.5 (Continued)

under negative pressure. After transfer to an approved container, appropriate precautions are exercised during removal of the containers to minimize airborne radioactivity. Ash is then transferred to a ventilated enclosure designed to meet the requirements of Subparagraph 2.2.5 of this license.

At any time, liquid from the scrubber can be manually or automatically transferred to the waste treatment process where uranium content is determined and if within acceptable limits is pumped to the plant waste treatment facility. Here they are refiltered and discharged to the environment in accordance with the limits specified in 10 CFR 20.

1.9.6.6 Nuclear Safety Control

The area is monitored by a criticality alarm station which is part of the plant system and alarms both throughout manufacturing areas and at the plant Health Physics Laboratory. The incinerator is operated on a nuclearly safe batch basis in accordance with the maximum permissible values of mass listed in Figure 2.3.2.1. The enrichment used to determine the maximum permissible mass is the highest value assigned to any of the wastes accumulated for incineration. All associated equipment, such as the scrubber pump tank, heat exchanger, filter housing in the scrubber recirculating system, and the vacuum cleaner used for ash removal is designed for processing wastes having a maximum enrichment of 4.15 w/o U-235.

However, if it becomes necessary to process wastes having enrichments between 4.15 w/o U-235 and 5.0 w/o U-235, this equipment (if required) will be resized and respaced in accordance with the applicable criteria given in Figures 2.3.2.1 through 2.3.2.5. In addition, the batch size will be appropriately reduced.

1.9.6.6 (Continued)

Combustible wastes delivered to the incinerator area are contained in 30-66 gallon, metal or fiberboard drums, or fiberboard boxes. These containers of waste are scanned in a gamma counting facility. After scanning, the containers are stored under established criteria for the Contamination Controlled Area.

Wastes accumulated without regard to origins are assumed to contain uranium having a maximum enrichment, normally 4.15 per cent U-235. Proper storage of this material and storage area posting requirements are monitored routinely by the Plant Criticality Engineer.

Waste accumulated with known origins or enrichments are assigned the known enrichment. They are stored correspondingly segregated in accordance with established operating procedures. Storage areas for these wastes are reviewed and approved by the Manager of the Radiation Protection Component.

All containers are marked with grams of U-235 contained therein as determined by licensee's gamma counting facilities. Only precounted containers of combustible waste may be charged to the incinerator.

The gamma counting facilities are calibrated at least semiannually, and checked twice weekly during counting operations, using phantoms loaded with known quantities of U-235. Size and geometry of the phantoms are equivalent to those of the waste containers.

The incinerator is operated on a safe batch mass control basis. The operator maintains an awareness of the U-235 content of waste charged to the system and ash removed from the system. The system is charged to the allowable safe batch limit reduced by the inherent error of the measurement system. Waste may be charged until this limit is reached. When the incinerator system is cleaned and the ash removed,

1.9.6.6 (Continued)

the nuclearly safe containers of ash are gamma counted and the SNM content determined. Based upon these data, additional charging may continue until a safe batch is reached. Incinerator operations are suspended, ash removed and gamma counted, and the chambers thoroughly cleaned and visually inspected for uranium accumulations. Alternatively, the MUF may be adjusted to zero after each burn. In this case, the ash would be removed and the entire incinerator system would be thoroughly cleaned and visually inspected, prior to release by the supervisor.

After operation of the incinerator, ash is removed in trays or with a vacuum cleaner or other means and loaded into containers such as polypaks, fiberpaks or metal pails. Both the vacuum cleaner and the ash containers are either limited to a nuclearly safe volume or a nuclearly safe diameter. The applicable maximum permissible values specified in Figures 2.3.2.2 and 2.3.2.3 respectively for 4.15 w/o homogeneous oxides are used. Each container of ash is limited to a maximum of 21.5 pounds of total uranium which is less than a maximum permissible value for 4.15 per cent enriched material as specified in Figure 2.3.2.1. Containers of ash are stored in a designated section of the Contamination Controlled Area in accordance with the applicable nuclear criticality control criteria established in Subparagraph 2.3.2. Ash may be processed for recovery of the SNM or disposed of to a licensed burial facility.

The wet scrubber system sump tank (water reservoir), filters and heat exchanger are each in the form of cylinders with an effective safe inside diameter of 10.4 inches or less. The heat exchanger is considered a flow through device and the sump, tank and filters are spaced in accordance with surface density criteria.

1.9.6.7 Safety Mechanisms

Safety controls exist in several areas of the system to insure safe operation of the system as well as control operational upsets and/or malfunctions that could occur. These are listed as follows:

(1) High temperature (approximate 250°F) detected in the Quench Tower Sump. Alarm Indication. Appropriate actions are taken to correct the situation.

(2) Low indications in the scrubber system weir liquid flow. Alarm Indication. Appropriate actions are taken to correct the situation.

(3) Low indication on system air flow. Alarm Indication. Appropriate actions are taken to correct the situation

(4) ΔP HEPA filter high. Alarm Indication. Appropriate actions are taken to correct the situation.

As enumerated above, sufficient means are provioed for immediate detection and correction of problems that could occur in the incinerator system. For this reason, adverse inplant and offsite effects due to system failures are not considered likely.

However, the provisions of Subparagraph 2.2.12 apply to the incinerator, including the automatic shutdown requirements described therein.

1.9.6.8 Improvements in the operation, logic and functional control of this system may be made as indicated by new regulations and/or operating experience.

1.9.7 Chemical-Manufacturing Development Laboratory

1.9.7.1 Purpose of Chemical-Manufacturing Development Laboratory

The purpose of the Chemical-Manufacturing Development Laboratory is to provide a separate area where development of processes and equipment can be accomplished with minimum impact on normal production operations. This area is used for development, prototype development and equipment checkout prior to installation in the production operations.

1.9.7.2 Typical Operations That Occur in the Development Laboratory

The three typical areas of operation that may occur are listed below:

1.9.7.2.1 Chemical Development such as Waste Treatment Studies, Uranium Chemical Processing, Uranium Recovery, etc.

1.9.7.2.2 Ceramic Development such as Powder Preparation and Characterization, Pelletization Studies, Sintering Studies, etc.

1.9.7.2.3 Mechanical Development such as Rod Loading Devices, Rod Plugging and Welding Development, etc.

1.9.7.3 Chemical-Manufacturing Development Laboratory Location

The Chemical-Manufacturing Development Laboratory is located in the Contamination Controlled Area of the Westinghouse Columbia Plant. The specific location is shown in Figure 1.9.0.1.

1.9.7.4 Chemical-Manufacturing Development Laboratory Description

The laboratory occupies approximately 4,400 square feet total area of floor space. It is located adjacent to the Production Manufacturing Area and is isolated by a wall or fence.

Services such as ventilation, water chemicals, etc. are provided as needed. Equipment and operations used for development are evaluated by the Radiation Protection Component for compliance with existing license and regulatory requirements prior to operation.

The laboratory is monitored by the area criticality alarm system. The personnel bioassay program described in Section 3.2.4 applies to persons working in this area.

1.9.7.5 Chemical-Manufacturing Development Laboratory Controls

The scope of projects carried out in this area is necessarily very broad. This requires the following administrative controls to assure that radiological and nuclear safety concerns are addressed and appropriate controls are implemented.

1.9.7.5 (Continued)

In this regard, lines of organizational authority for the laboratory follow the philosophy outlined in Subparagraph 3.1 of the license; i.e., line management is responsible for all aspects of the operations, including safety.

The Radiation Protection Component's responsibility is to review all development work from a radiological and nuclear criticality safety standpoint. All work involving radioactive materials requires either an effective detailed procedure or a job safety analysis. Procedures are reviewed by the Radiation Protection Component for radiological and nuclear criticality safety. Whether the work is performed under approved procedures or job safety analyses, the responsible line manager is required to submit sufficient information to permit the proper review. The radiation protection function then issues the nuclear safety posting criteria and inspects installed equipment as appropriate.

Air sampling requirements for the laboratory are evaluated in accordance with Subparagraphs 2.2.6 and 3.2.3 of this license. Permanently mounted continuous air sampling stations are established where the greatest concentration of airborne activity is expected under adverse circumstances and consistent with operator work locations. However, engineering controls are used where possible to control airborne radioactivity at its source.

Ventilated enclosures are used as appropriate to control airborne concentrations. Examples include a walkin ventilated hood for testing large pieces of equipment, small fume hoods, dust collectors, etc.

All enclosures are designed and operated to meet the ventilation specifications in Subparagraph 2.2.5 of this license; ventilation control procedures conform to Subparagraph 3.2.3 of this license.

1.9.8 Purification of Contaminated Scrap through Solvent Extraction

1.9.8.1 Purpose of Solvent Extraction

Contaminated uranium scrap may be purified through inplant solvent extraction and thus virtually eliminate the shipment of uranium-bearing scrap material over the public highway system.

1.9.8.2 Typical Materials to be Purified through Solvent Extraction

Typical materials are uranyl nitrate solutions prepared utilizing the following scrap components requiring purification. These include scrap filter press cake, contaminated UO2, and incinerator ash.

These scrap materials result from licensee's fuel fabrication process operations.

1.9.8.3 Solvent Extraction System Location

The solvent extraction system is located in the Contaminated Controlled Area of the Westinghouse Columbia Plant. The specific location is shown in Figure 1.9.0.1.

1.9.8.4 Process Description and Equipment

1.9.8.4.1 Solvent Extraction System

The solvent extraction system consists of scrap preparatory equipment, dissolver, mechanically pulsed extraction and stripping columns, product evaporator, a nitric acid recovery system, associated liquid storage vessels and related instruments and controls. Scrap uranium is charged to the dissolver in preweighed critically-safe batches where nitric acid and water dissolve the uranium and forms a uranyl nitrate solution. The dissolver product is filtered, pumped to the feed storage tanks and adjusted to a maximum of 6 grams U-235 per liter. The feed adjustment tank, feed storage tanks and all other uranium-bearing tanks meet the MPV for diameter criteria. All liquid transfer lines are less than 2 inches in diameter and do not require criticality consideration. The process flow diagram is shown in Figure 1.9.8.4.1.

1.9.8.4.2 Preparatory Equipment

Prior to dissolution, uranium-bearing scrap is treated to remove impurities such as volatile materials. These operations include thermal processing, solids removal and concentration of the scrap materials. Thermal processing and concentration are conducted in furnaces on a batch basis. Solids removal is performed in a series of nuclearly safe diameter tanks designed to sequentially dissolve the scrap in a nitric acid solution and settle out the nondissolved solids.

1.9.8.4.3 Dissolution

Dissolution is performed in closed vessels in accordance with the techniques described in Subparagraph 1.9.4.2.

1.9.8.4.4 Extraction

Uranium is extracted from the acidified feed solution in a four inch safe diameter column containing spaced sieve plates and pulsed by a piston pump. The extracting solvent is a tributyl phosphate (TBP) mixture. The solvent mixture is metered into the top of the column. The uranium-bearing aqueous feed is charged in slightly below the column midpoint and flows toward the top of the column countercurrent to the solvent mixture flow.

A nitric acid scrub solution flows into the bottom of the column and toward the top of the column combining with the aqueous feed stream in the upper region of the column. Superimposed on these stream flows is an oscillation provided by the pulsing pump which forces the liquids back and forth through the sieve plate holes creating a dispersion of solvent droplets in the aqueous continuum. The uranium transfers from the aqueous phase into the solvent phase aided by the "salting out" action of the excess nitric acid present. The extracted aqueous phase collects in a "quiet zone" provided at the top of the column, overflows into a ten-inch

1.9.8.4.4 (Continued)

nuclearly safe diameter holding tank and is directed through a solvent trap to a nitric acid recovery unit. A radiation detector monitors the column overflow and automatically diverts the stream into a ten-inch nuclearly safe diameter auxiliary tank for recycle to the feed tanks if a high uranium content is indicated (>1 gram U-235 per liter). The uranium laden solvent phase collects in a quiet transition zone provided at the bottom of the column and then flows to the top of the stripping column.

1.9.8.4.5 <u>Uranium Stripping</u>

The uranium laden solvent from the extraction column flows into the top of a six-inch safe diameter column containing spaced sieve plates and pulsed by a piston pump. The solvent phase flows toward the bottom of the stripping column countercurent to a stream of weakly acidic nitric acid/stripping water that is metered into the bottom of the column. Superimposed on the flows of these two streams is an oscillation provided by the pulsing pump which forces the liquids back and forth through the sieve plate holes creating a dispersion of solvent droplets in the aqueous phase. The uranium transfers from the solvent phase into the stripping water. The uranium laden stripping water collects in the transition zone provided at the top of the column and overflows through a solvent trap into a ten-inch nuclearly safe diameter holding tank. The stripped solvent collects in a transition zone provided at the bottom of the stripping column and flows into a ten-inch nuclearly safe diameter cylindrical holding tank. From there it recycles to the extraction column again to extract uranium from a new increment of feed.

1.9.8.4.6 Uranyl Nitrate Concentration

Dilute aqueous phase product from the stripping column is transferred into a safe diameter, cylindrical, flash evaporater in order to remove excess water from the uranium solution. The evaporated water is condensed and recycled to a storage tank for reuse in the extraction process. The uranyl nitrate product, concentrated to a maximum of 5 gm U-235/liter, is pumped from the evaporator to product holding tanks (10-inch nuclearly safe diameter cylinders) where it is sampled for isotopic content, acidity (as free HNO_3) and uranium content.

Adjustments to acid content and isotopic content (when needed) of the product are continuously made as the product moves through a ten-inch safe diameter cylindrical mixing tank. The material is then sent to product storage tanks for use in the uranium conversion lines.

1.9.8.4.7 Nitric Acid Recovery

The aqueous effluent from the extraction column is transferred into a nuclearly safe diameter cylindrical evaporator which vaporizes almost all of the liquid. A small amount of liquid is necessary to keep the metal salts carried by the effluent in a slurry solution. The vapor stream is piped into a safe diameter distillation column which separates the nitric acid as approximately 30 per cent acid and this is returned to a storage tank for reuse in the uranium scrap dissolver. The distillation overhead, which is essentially water, is recycled to a storage tank for reuse in the extraction process.

1.9.8.4.8 Process Waste Treatment

Tank vents are connected to the plant vessel ventilation system which controls atmospheric pollutants to the lowest practical level. Noncondensable offgas from the uranyl nitrate evaporator and nitric acid still are removed through

1.9.8.4.8 (Continued)

the plant vessel ventilation system. The vessel ventilation system contains a direct contact, recirculating, venturi scrubber with a liquid disengagement section followed by a demister and finally, HEPA filters prior to discharge of gases to the atmosphere. Filtered gases discharged to the atmosphere are continuously sampled and are analyzed for airborne particulate radioactivity on a daily basis. Scrubber water is transferred to the final filtration tanks where it is combined with conversion operations process wastes. These materials are then pumped through an online monitor system where uranium content is determined and, if within acceptable limits, is pumped to the plant waste treatment facility. Liquid discharges from the solvent extraction system are filtered through carbon filters before being processed through the plant waste treatment facility.

A periodic solvent renewal of approximately 30 gallons for each 300 hours of column operation is expected to be needed. The spent solvent mixture is drawn from the holding tank into a drum and replenished with fresh solvent mixture.

The preferred method of disposal of the degraded solvent will be to wash the spent solvent solution with a sodium carbonate solution to remove the degradation products and then reuse the solvent in the process. Two additional possibilities for disposal of the spent solvent exist: (1) further salvage uranium from the spent solvent such that the material may be disposed of by incineration, (2) solidify and ship to a licensed burial facility.

The metal salt slurry residue from the nitric acid recovery vaporizer also contains traces of uranium. This slurry is recycled through the solvent extraction system to further remove uranium from this material. The remaining residue is precipitated, filtered, solidified and disposed of through

1.9.8.4.8 (Continued)

burial. All other streams from the solvent extraction process are internally recycled as described in previous paragraphs.

1.9.8.5 Radiological Safety Control

The solvent extraction system is installed within the Contamination Controlled Area of the plant. Only authorized personnel are allowed into this area. Operating personnel are required to submit to the bioassay program for routine urinalyses. Lung burden determinations and the use of the external radiation exposure monitoring devices may also be required.

Permanently mounted, continuous air samplers are located at points within the system where greatest concentration of airborne activity is expected under adverse circumstances and the points where operators work during routine operation of the system.

All tanks, columns, enclosures and furnaces are ventilated into the existing plant exhaust system.

All tanks and columns are vented into the present scrap recovery scrubber system. Treatment of these off-gases include scrubbing and HEPA filtration.

Ventilated enclosures used for containment of dry scrap are directed through the existing plant exhaust system which includes HEPA filtration. An average of 100 lfpm is maintained across all openings. Ventilation control procedures and administrative controls are described in Subparagraphs 2.2.5 and 3.2.3 of the license.

With the effluent treatment described above, no significant exhaust concentrations have been experienced.

1.9.8.6 Nuclear Safety Control

The nuclear safety of the system is assured on the basis of safe mass, geometry, volume and by solid angle criteria. The

1.9.8.6 (Continued)

double contingency criteria is met by solid angle criteria and one or more of the MPV's for mass, geometry or volume.

The dissolver is identical with the ones now used in Dissolution Operations Section 1.9.4.2 of the license and are operated on a safe mass basis. All solutions down stream of the feed adjustment tanks are $\leq$6 gm U-235/liter so that neutron interaction is of limited concern (see next paragraph for solid angle analyses). All process vessels and equipment meet the MPV's for mass, geometry or volume.

Solid angle analyses were performed using solid angle criteria as given in TID-7016, Revision 2, and the supplemental reflector conditions as specified in Subparagraph 2.3.2 of the license.

The area is monitored by two criticality alarm stations of the plant gamma alarm system with dual scintillation detectors and three siren type audible alarm horns.

Tanks which exceed the MPV for cylinder diameter criteria such as the HNO3 head tanks are vented to preclude an inadvertent vacuum transfer of uranium-bearing solution from the process vessels to the tanks. All process vessels are vented into a safe geometry scrubber.

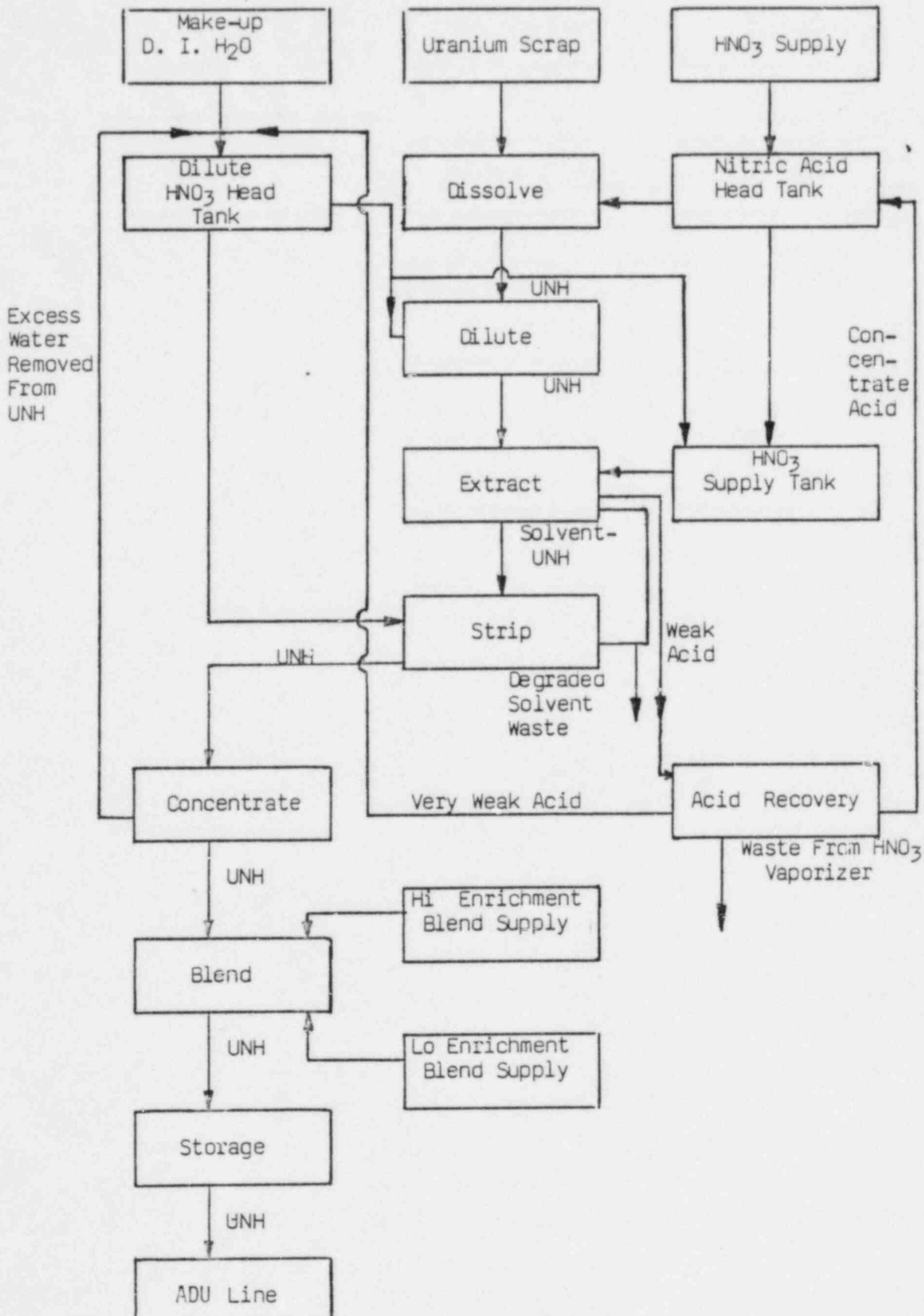
1.9.9 <u>UF6 Cylinder Washing and Recertification</u>

The UF6 cylinder washing and recertification facility is located adjacent to the Waste Storage Building as shown on Figure 1.3.2.1.

Cylinders are washed by introducing five-gallon batches of aqueous solutions into the cylinders, rotating the cylinders to assure that all surfaces are thoroughly washed and discharging the solutions into geometry controlled containers. These steps are repeated until the uranium concentrations are below an acceptable level. Contents of the geometry controlled containers are sampled and anlayzed

SOLVENT EXTRACTION
SYSTEM FLOW DIAGRAM

FIGURE 1.9.8.4.1



1.9.9 (Continued)

for uranium; if the uranium content is acceptable, the solutions are transfered to a larger holding tank for storage and for radiological decay of the short half-life uranium daughter products.

When the solutions in the holding tanks have decayed to an acceptable level, the liquid is transferred to the Manufacturing Building for recovery of the uranium or to the liquid waste treatment facility for disposal.

Cylinder recertification procedures include hydrostatic testing, visual checks for cracks, etc., wall thickness and valve inspections. Water used in the hydrostatic tests is stored in hold tanks, sampled, analyzed and disposed based upon results.

Nuclear Safety

Rinse solutions are limited to five-gallon batches which represent a safe volume. These rinse solutions are collected in a cylindrical tank with diameter less than 9.4 inches which represents an MPV for homogenous oxides.

Radiological Safety

Uranium in solution form represents a minimal potential for airborne.

Appropriate shielding is utilized for aqueous storage tanks to minimize personnel exposures associated with the uranium daughter products.

1.10 Offsite Release Evaluations

Westinghouse has postulated a variety of accidents in the handling of large quantities of source and low enriched special nuclear materials and has found the consequences of any of them to be well within established guidelines. The most significant type of accident would be a release of an appreciable quantity of radioactive material off site. A quantity of uranium may be established, which, if released,

1.10 (Continued)

would produce a downwind concentration at ground level equal to the maximum allowable concentration specified in 10 CFR 20.106(a).

For calculational purposes the following assumptions were made:

(1) The specific activity of uranium enriched to 4.15 w/o in the isotope U-235 is taken to be 2.04×10^{-3} Ci/kg, or 9.27×10^{-4} Ci/lb.

(2) The duration of the release is 24 hours.

(3) During this period, the allowable concentration at the restricted area boundary is 1.46×10^{-9} Ci/m3 [10 CFR 20.106(a)].

(4) An individual remains at a point in the unrestricted area downwind of the release point and at ground level, for the duration of the release.

(5) The effect of particulate deposition is conservatively neglected.

Based on the above assumptions, the release rate is calculated using the equation

$$X = \frac{Q}{(\pi\sigma_y\sigma_z + CA)\bar{u}} e^{-(Y^2/2\sigma_y^2)} \quad /1$$

where: X = concentration at ground level = 1.46×10^{-9} Ci/m^3

Q = release rate, Ci/sec

Y = crosswind distance in meters = 0 downwind

C = building shape factor = 2 for a bluff building in level terrain

σ_y = crosswind plume standard deviation for maximum concentration downwind

σ_z = vertical plume standard deviation for maximum concentration downwind

A = cross-sectional area of building = 506 meter^2

$\bar{u}$ = wind velocity = 3 meters/sec

1.10 (Continued)

For these calculations, it is conservative to neglect the plume standard deviation. Then:

$X = Q/CA\bar{u}$

$Q = (2)(5.06 \times 10^{2})(3)(1.46 \times 10^{-9})$

$Q = (3.03 \times 10^{3})(1.46 \times 10^{-9})$

$Q = 4.43 \times 10^{-6}$ Ci/sec

Q = 0.383 Ci released over a 24-hour period

The total quantity of 4.15 w/o enriched uranium that would produce this amount of activity is equal to

$M = 0.383$ Ci/9.27×0^{-4} Ci/lb = 413 pounds of uranium

This quantity decreases to approximately 355 pounds for uranium enriched to 5 w/o and is increased for enrichments below 4.15 w/o and for source material. The quantities computed above are conservative, since the calculation does not consider material deposited between the release point and the unrestricted area handled in any one operation and the containment provided by the equipment and buildings, a material release from the facility of this magnitude is not considered credible.

/1 "Recommended Guide for the Prediction [illegible] Dispersion of Airborne Effluents," by Maynard Smith, Brookha[illegible] [illegible]al Laboratory, American Society of Engineer's Committee on Air Pollution Controls. May 1968.

SECTION 2

TABLE OF CONTENTS

Minimum Specifications and Capabilities

2.1 Scope of Licensed Activities

2.1.1 Definitions

2.1.1.1 Clean Area - an area where radioactive material, if present, is completely contained and there is negligible contamination on the floors or accessible surfaces (e.g., Machining Area, Grid Assembly Area, Final Assembly Area, Office Areas, Cafeteria, etc.).

2.1.1.2 Component - when used administratively, an organization unit, distinguishable by its assigned responsibilities; e.g., the Line Management Component, the Radiation Protection Component, etc.

2.1.1.3 Controlled Access Area - is represented on three sides (north, west and south) by a seven-foot high fence fabric barrier of Number 11 American Wire Gauge topped by three strands of barbed wire and on the fourth side by the Administration and Manufacturing Buildings.

2.1.1.4 Controlled Area - an area where uncontained radioactive materials are processed and probability of contamination on floors and accessible surfaces is high. Protective clothing is required (e.g., Conversion Area, Pelleting Area, Rod Loading Area, UF6 Bay, etc.).

2.1.1.5 Dry - when used to describe special nuclear material (SNM), having a moderation ratio (H/U = hydrogen to uranium atomic ratio) less than, or equal to, 0.5 for uranium enriched up to 4.15 weight percent in U-235, and less than or equal to 0.3 for uranium enriched between 4.15 and 5.0 weight percent.

2.1.1.6 Equivalent Diameter - when evaluating the adequacy, for purposes of nuclear criticality safety, of the geometry control of a subcrit unit having a noncircular cross section, the diameter of that circle that has the same area as the area of the cross section of the subcrit unit. Dequiv. = 2 $(\text{Area}/\pi)^{1/2}$ where Area is the cross sectional area of the subcrit unit under reivew.

2.1.1 (Continued)

2.1.1.7 Equivalent Experience - when used to equate experience with education, ten years of applicable experience is equivalent to a bachelor's degree.

2.1.1.8 Fraction Critical - the ratio of the equivalent spherical volume of a subcrit, determined from geometric buckling conversions (for example, see Subparagraph 1.8.1.4), to the unreflected critical volume of a sphere of the same fissile material.

2.1.1.9 Function - when used administratively, an individual (usually the component manager), acting in coordination with the other personnel of a component, having the capability and authority to make and implement decisions required to carry out the component's assigned responsibilities.

2.1.1.10 Licensed Activity - that combination of personnel, plant and equipment established by the Corporation to carry out the processing of radioactive material authorized by this license.

2.1.1.11 Limited Area - An area where small quantities of uncontained radioactive materials are processed such that the probability of contamination on floors and accessible surfaces is low. (e.g., Analytical Chemistry Lab, Health Physics Lab, Manufacturing Development Lab, Metallurgical Lab, Step-Off Pads, etc.)

2.1.1.12 Maximum Permissible Value (MPV) - that value or combination of values of a selected nuclear criticality safety control parameter(s) which:

(1) has been established to assure nuclear criticality safety for the subcrit under all foreseeable conditions, including interaction and the maximum credible accident,

(2) is the value communicated to the operator(s) as the value that must be observed.

2.1.1 (Continued)

2.1.1.13 Nuclearly Isolated - a subcrit or array of subcrits located with respect to any other subcrit such that an edge-to-edge separation exists between them which is not less than one of the following or its nuclear equivalent:

(1) Eight inches of water.

(2) A distance through air which is the larger of: 12 feet, or the greatest distance across an orthographic projection of either subcrit or array of subcrits on a plane perpendicular to a line joining their centers.

2.1.1.14 Restricted Area - An area to which access is controlled by physical or administrative means and routinely monitored by site security personnel.

2.1.1.15 Subcrit - An accumulation of fissionable material which is treated as a whole and within which one or more physical limitation(s) is incorporated to assure nuclear criticality safety.

2.1.2 Summary Description

2.1.2.1 The objective of the licensed activity is the ADU or IDR process conversion of uranium hexafluoride to uranium dioxide and the manufacture of fuel-bearing components for nuclear reactor cores. The licensed material is composed of unirradiated special nuclear materials received principally as uraium hexafluoride containing uranium enriched up to 5.0 w/o in the isotope U-235.

2.1.2 (Continued)

2.1.2.2 Specifically, activities authorized at the Columbia Site include:

(1) Receipt of SNM (UF6 cylinders, samples, uranium oxides, uranyl nitrates, uranium pellets, fuel rods and fuel assemblies)
(2) Chemical conversion processing,
(3) Pelletizing
(4) Pellet sintering
(5) Fuel rod loading and sealing,
(6) Fuel assembly fabrication
(7) Chemical and mechanical development,
(8) Chemical laboratory analyses and services
(9) Uranium scrap recovery
(10) Incineration,
(11) Radioactive waste packaging for disposal (solids, liquids and gases),
(12) Maintenance activities,
(13) Health physics activities (lab analyses, respirator testing, bioassay, etc.),
(14) Respirator cleaning and HEPA filter testing,
(15) QA/QC inspection activities,
(16) Decontamination activities, including clothing and equipment
(17) Mechanical (nonradioactive component fabrication and assembly),
(18) Shipment and receipt of radioactive materials,
(19) Liquid waste treatment,
(20) Lagoon storage,
(21) Solid waste storage,
(22) UF6 cylinder washing and recertification,
(23) Shipping container refurbishment, and
(24) Poison rod fabrication

2.1.2 (Continued)

2.1.2.3 Scrap or waste licensed material resulting from site operations are possessed for concentration, recovery and/or reuse. These operations may involve chemical separation (e.g., acid treatment and dissolution or acid leaching, followed by chemical precipitation), mechanical separation and thermal decomposition.

2.1.2.4 The licensed activity performs work for other Westinghouse Divisions or outside customers which is adapted to the capabilities of the facility. The work may consist of uranium oxide fuel fabrication and quality assurance testing operations and laboratory analyses of uranium or byproduct material.

2.1.3 Material

2.1.3.1 Special Nuclear Material

Listed below are the maximum quantities of special nuclear materials which will be possessed by the licensed activity at any one time.

Material	Form	Quantity
U-235	Unirradiated, any chemical or physical form at any enrichment	350 grams
U-235	Unirradiated, any chemical or physical form at any enrichment $\leq$ 5.00 w/o	75,000 kilograms
U-233	Any chemical or physical form, for laboratory uses only	5 grams *
PU 238, 239	Sealed sources	1.5 grams

*Activities involving U-233 shall be limited to the handling of individual one gram quantities in ventilated hoods.

2.1.3.2 Administration

The procurement of these materials are according to program needs. No production, consumption or significant loss of special nuclear material is anticipated. Transfers of material are arranged as required with facilities authorized to possess such materials.

2.2 Facility and Equipment Design Features

2.2.1 General Requirements

2.2.1.1 The licensed activity maintains personnel, equipment and facilities adequate for processing material in a safe and efficient manner. The equipment and facilities include: storage facilities for licensed material at various stages of processing: UF6 to UO2 conversion equipment, such as tanks, piping, pumps, and filters; uranium oxide processing equipment such as ventilated hoods, blenders, presses, furnaces, grinders, and fixtures and tooling; receiving and shipping facilities including approved shipping packagings; and a solid waste storage and packaging facility.

2.2.1.2 A Radiation Protection Component is maintained which provides support services contributing to radiation protection, nuclear criticality safety, SNM safeguards and environmental pollution control.

2.2.1.3 Conventional firefighting equipment is maintained at strategic locations on the Columbia Site.

2.2.2 Emergency Equipment

Four emergency standby generators with rated capacities of 250 kw, 300 kw, and two at 500 kw are maintained. The standby generators are activated automatically in the event of a facility power outage.

2.2.3 Personnel Monitoring Devices

2.2.3.1 Film badges or thermoluminescent dosimeters (TLD's) provided by a commercial supplier, capable of detecting and measuring beta-gamma and x-radiation shall be used. Badges for

2.2.3 (Continued)

beta-gamma and x-radiation are evaluated at least quarterly. In addition, neutron detection capability shall be available as specified by the Radiation Protection Component.

2.2.3.2 All personnel who have access to areas with criticality monitors are provided with criticality monitoring devices to provide the means for identifying quickly which individuals have received doses of 10 rads or more.

2.2.4 Radiation Protection Measuring Instruments

The instruments used for radiation protection measurements shall have the capabilities listed below. (More than one instrument may be utilized to cover the entire range specified.)

2.2.4.1 Portable Survey Instruments

Alpha	$50\text{-}10^6$ dpm
Beta-gamma	0.1 mR/hr - 300 R/hr

2.2.4.2 Laboratory Assay Instruments

Alpha	0.1 MPC

2.2.5 Ventilation Equipment

2.2.5.1 All ventilation systems shall be designed and normally operated to move air from uncontaminated areas to suspect contaminated areas to contaminated areas. Exceptions to such air flows shall be limited to unusual conditions resulting from equipment failure or be of sufficiently limited duration as not to constitute a significant exposure condition. When adverse air flows are detected, corrective actions shall be taken as soon as practicable.

2.2.5.2 Ventilation systems servicing laboratory type hoods, production hoods and/or primary containments shall provide an average face velocity of at least 100 linear fpm for all openings during normal operations. Exceptions to the velocity requirements are those containments which are installed to primarily confine chemical fumes, mists and dusts. Velocity determinations shall be made and documented at least monthly when the equipment is in operation.

2.2.5.3 When containment of uranium dust by conventional ventilation hoods is not possible or impractical, gloveboxes may be used. Ventilation systems for gloveboxes and other completely enclosed devices shall be designed and operated with a nominal negative internal pressure (approximately 0.1 inch water minimum with respect to ambient air) sufficient to maintain the concentration of airborne radioactivity from the serviced equipment below 10CFR20 limits. Gloveboxes shall be equipped with instrumentation for measuring differential pressure, which shall be checked at least monthly when the equipment is in operation.

2.2.5.4 Ventilation hoods and gloveboxes shall be constructed primarily of metal, glass and/or fire resistant plastic. Plastics shall conform to a Class I fire rating (e.g., Lexan).

2.2.5.5 Ducts shall be designed to minimize accumulations of SNM and shall be inspected for SNM on a routine basis commensurate with the potential for SNM accumulations.

2.2.5.6 Exhausts from gloveboxes, hoods, local exhaust and similar devices, when employed for radiation protection purposes, shall be passed through HEPA filtration. HEPA filters shall be replaced either on a routine schedule or when airborne concentrations, hood velocities, differential pressure drop or DOP (or acceptable substitute) penetration measurements indicate that replacement is necessary. The maximum differential pressure permitted across a HEPA filter shall be 8 inches water for negative pressure systems and 4 inches water for recirculating positive pressurized systems.

2.2.5.7 Exhausts from inplant recirculating air cleanup systems, including gloveboxes, hoods, local exhaust and similar devices shall either be penetration tested or sampled for airborne radioactivity concentrations at least monthly.

2.2.6 Air Sampling

2.2.6.1 Portable and permanently mounted air sampling equipment selected to be suitable for detecting and evaluating the

2.2.6 (Continued)

concentration of airborne radioactive particulate matter shall be maintained. Airborne radioactivity areas are established in accordance with 10CFR20. The air in these areas is continuously sampled by permanently mounted samplers located where potential airborne concentrations are deemed to exist as determined by the Radiation Protection Component. The filters from these samples shall be normally changed each work shift, allowed to decay and processed through calibrated counting equipment. Using appropriate correction factors such as instrument efficiency, collection efficiency and self-absorption, airborne concentrations are calculated.

2.2.6.2 When existing equipment is substantially modified, or when new operations are introduced, those which have a potential for generating significant airborne radioactivity shall be air sampled to determine the need for permanent sampling. The Radiation Protection Component shall be responsible for evaluating the need for such sampling.

2.2.6.3 Air sampling representativeness of existing samplers shall be reviewed every two years, or following substantive equipment or ventilation changes. Such studies shall be performed in accordance with procedures specified by the Radiation Protection Component.

2.2.6.4 Counting equipment and techniques shall be used which will detect concentrations of airborne radioactive particulates at levels which are 10 per cent of the limits specified in 10CFR20.

2.2.7 <u>Liquid Waste Disposal</u>

2.2.7.1 A waste treatment faclity is provided and maintained, with sufficient capacity to permit the holdup of liquid wastes for treatment, sampling and analysis, and to permit their disposal in accordance with the provisions of 10CFR20.

2.2.7.2 Any violation of the NPDES Permit or changes in the Permit shall be submitted within ten days of the determination of the event to the NRC Region II Office of Inspection and Enforcement.

2.2.8 Solid Waste Disposal

Facilities designed to be adequate for the preparation, packaging and transferring of solid and high level liquid radioactive wastes to a licensed disposal facility shall be provided and maintained. Materials containing trace quantities of radioactive constituents which fall under the guidelines outlined in NRC Branch Technical Position SECY 81-576, Option 1 shall be disposed of in a chemical disposal site or sanitary landfill without continuing NRC licensing controls and shall be considered an nonradioactive industrial waste products.

2.2.9 Criticality Radiation Detection and Alarm System

2.2.9.1 An immediate evacuation signal system activated by radiation detection devices, which fulfills the requirements of 10 CFR 70.24(a)(2) shall be maintained. In the event of a normal power outage, emergency power is connected to the system automatically.

2.2.9.2 The immediate evacuation signal devices shall be located so that the signal will be heard by all affected personnel. The ability of the employes to hear the signal shall be confirmed during training drills and functional tests.

2.2.9.3 The control units for the immediate evacuation signal system shall be equipped with devices which will activate a visible or audible signal such that personnel will be made aware of a failure in the system.

2.2.10 Processing Equipment Incorporating Nuclear Criticality Safety Controls

All isolation structures, vessels, racks, furnaces and other processing equipment which incorporate nuclear criticality safety limiting parameters shall be designed with an adequate strength factor to assure against failure under foreseeable loads. They shall also be designed or mounted to assure stability in the event of any foreseeable shocks or collisions. Design safety analyses shall be reviewed and

2.2.10 (Continued)

approved by the Safety Component as required. The materials of construction shall be selected or coated to resist leakage, fire, corrosion or other deterioration which might tend to reduce their effectiveness.

2.2.11 Chemical Equipment

The equipment specified below shall be provided as part of the UF6 vaporization system:

2.2.11.1 A UF6 detection method in the steam condensate system, with an alarm to alert operating personnel to a leak in steam-type vaporizers and provision to shut off the steam in the event of a UF6 leak; and with an interlock and an alarm to alert operating personnel to a leak in hot-water-type vaporizers.

2.2.11.2 A pressure relief valve and a liquid level detector and alarm in the steam-type UF6 vaporizers; or, with favorable geometry/overflow-type sumps in the hot-water-type UF6 vaporizers. The liquid level detector probe shall be sealed to prevent problems associated with plugging.

2.2.11.3 Provisions to permit the leak testing of the UF6-cylinder-to-conversion-system connections each time a cylinder is connected.

2.2.11.4 A means for cooling the UF6 cylinder.

2.2.11.5 A means to prevent the backflow of water from the hdyrolysis tank to the UF6 cylinder, in ADU process systems.

2.2.11.6 Devices to permit remote closing of the UF6 cylinder valves without the need to remove the vaporizer heads.

2.2.12 Incinerator Equipment

The equipment, controls and safety interlocks specified below shall be provided as part of the incinerator system:

2.2.12.1 Temperature sensor/controllers in the primary combustion chamber, breech and scrubber exhaust.

2.2.12.2 Safety interlocks to inhibit feeding additional wastes if any overtemperature occurs in the primary combustion chamber.

2.2.12.3 A means for automatic shutdown of the incinerator in case of overtemperature in the scrubber exhaust.

2.2.12.4 An auxiliary means for cooling the incinerator exhaust gases.

2.2.12.5 Cooling water flow monitoring devices.

2.2.12.6 Pressure monitoring devices in the breech and scrubber exhaust.

2.2.12.7 A means for automatic shutdown of the incinerator in case of insufficient negative pressure in the breech and scrubber exhaust.

2.2.12.8 Continuous, representative gaseous effluent sampler

2.2.12.9 HEPA filtered exhaust and ash removal systems.

2.2.12.10 A means of monitoring and adjustment of the pH of scrubber solutions.

2.2.12.11 Maintenance ˜ a record indicating the maximum permissible quantity of 235 in the incinerator at any one time, for each container of waste charged and each container of ash removed and counted.

2.2.13 <u>Moderation Control Areas</u>

2.2.13.1 Fire control in areas of the Manufacturing Building where uranium is processed, handled, or stored under moderation control criteria contained in Subparagraph 2.2.13 shall receive particular attention, as follows:

(1) Special consideration shall be given to use of fire-resistive or noncombustible building components, equipment, and materials. Enclosures shall be constructed using noncombustible components where practicable; where visibility or other operating requirements dictate their use, plastics or other enclosure materials shall conform to Class 1 fire rating.

(2) Personnel shall receive training in the importance of minimizing the accumulation of combustible materials. Monthly audits by the Radiation Protection Component shall include this item.

2.2.13 (Continued)

(3) Special consideration shall be given to the prompt disposal of combustible waste. Waste generated during work activities shall be stored in metal containers having fire protection covers with containers emptied promptly when filled.

(4) A readily available supply of portable fire extinguishers suitable for use on the specific hazards encountered shall be provided.

(5) Such areas shall be subject to administrative controls, including specific personnel training and appropriate posting of instructions, to assure that only permissible firefighting means and materials are used.

2.2.13.2 All sources of moderator to the Moderation-Controlled Area (through service flows, etc.) shall be minimized to the extent practicable; then, those remaining shall be evaluated by the Radiation Protection Component using the double contingency principle. All sources of moderator to the moderation-controlled nuclear material (process equipment, bulk containers, etc.) shall be minimized to the extent practicable; then those remaining shall be investigated by the Radiation Protection Component, and controls shall be installed, which conform with the double contingency principle. The Radiation Protection Component shall review the initial designs, subsequent modifications to equipment and introductions of moderators into moderation controlled areas and materials, prior to implementation and following any changes to assure that double contingency criteria are maintained. Moderator additions may be controlled on a generic or case-by-case basis.

2.2.13.3 During nonprocess operations, such as maintenance involving temporary openings in the area hydrogenous barrier, moderation-controlled equipment and bulk containers within the affected area shall not be opened without prior unloading of the fissile material.

2.2.13.4 Moderation-Controlled Areas shall be conspicuously posted with instructions that only evaluated material containers, and other authorized moderators specifically approved by the Radiation Protection Component may be introduced to the area. Individual containers of moderator (e.g., water) shall be limited to 5-gallon capacity.

2.2.13.5 Moderator additions to moderation-controlled processes (e.g., IDR operations) shall be made batchwise so that accidental addition of large quantities of moderator has a low probability.

2.2.13.6 Controls associated with transfers of enriched uranium from geometry-controlled containers to moderation-controlled containers (e.g., completion of moisture determinations, container sealing, verifying that moderation-controlled containers are free of hydrogenous materials) shall be documented. This documentation shall be overchecked for completeness and accuracy by an individual other than the one performing the original determination prior to further processing of the material. Moisture determinations shall be completed before the material is transferred into moderation-controlled areas.

2.2.13.7 Monthly audits shall be conducted by the Radiation Protection Component to verify continued proper functioning of moderation-control systems and controls.

2.2.13.8 Nuclear criticality safety analysis of moderation-controlled operations shall include consideration of the effect of accidental moderator double batching (e.g., IDR operations).

2.2.13.9 The following controls shall apply when bulk containers containing fissile material, are removed from the Moderation-Controlled Area:

(1) During transport, bulk containers shall be physically protected (barrier shield, etc.) against conditions incident to transport.

2.2.13.9 (Continued)

(2) During transport, storage and powder off-loading, the bulk containers shall be oriented with the discharge feeder in the "down" position.

(3) During powder off-loading and container breaches at the unloading station, controls shall be established to assure that a closed system is maintained, providing positive containment of powder and precluding accidental introduction of moderating materials. This containment shall consist of a ventilated confinement surrounding the bulk powder container and discharge equipment and designed to exclude moderator material.

(4) Bulk containers shall be designed, constructed, and tested in accordance with applicable specifications in Title 49, Code of Federal Regulations, Part 178.251 and 178.252, for Specification 56 portable tanks.

(5) Container closure devices (gaskets, flanges, etc.) shall be inspected for integrity prior to each closure.

(6) Bulk container hoists shall be designed and operated to provide a margin of safety for both routine and accident conditions, including excess speed braking provision, brake-motor interlocks, semiannual load testing of hoists and chains or cables.

2.2.14 <u>IDR Conversion Process Equipment</u>

2.2.14.1 The following equipment will be provided as part of the IDR conversion system:

(1)* Disentrainment chamber steam temperature detectors

(2) Disentrainment chamber steam trace heater circuit detectors

(3)* Disentrainment chamber temperature detectors

(4) Disentrainment chamber heater circuit detectors

(5)* Disentrainment chamber pressure detectors

(6)* Kiln barrel steam temperature detectors

2.2.14.1 (Continued)

(7) Kiln barrel steam trace heater circuit detectors

(8)* Kiln barrel temperature detectors

(9) Kiln Barrel heater circuit detectors

(10) Shutoff of steam feed shuts off UF6 feed controllers

(11) Associated transmitters, alarms, alarm documentation capability and interlocks.

2.2.14.2 This equipment must be installed and operable as a condition of system startup and operation. This equipment will be calibrated at least semiannually during line operations; the respective calibration records will be maintained for one year.

2.2.14.3 Operating procedures for the IDR Conversion System will include functional parameter values for the respective equipment which will be conservative with respect to the limiting safe parameter values.

2.2.14.4 Failure of the interlocks, on the indicated (*) controllers, to shut off steam to an affected system when called upon by system operating conditions to do so, will be promptly reported to the Region II Office of Inspection and Enforcement.

2.3 <u>Nuclear Criticality Safety Limits</u>

2.3.1 <u>General Nuclear Criticality Control Procedures and Criteria</u>

2.3.1.1 Written procedures describing general nuclear criticality control requirements shall be maintained in the Regulatory Compliance Procedures Manual by the Radiation Protection Component. Copies of this manual shall be maintained by the line supervision in each functional area and the procedures contained therein shall be communicated to affected personnel. Operations shall be conducted in accordance with these procedures to ensure compliance with NRC regulations and license conditions.

2.3.1.2 A process or operation may be considered nuclearly safe if it requires at least two unlikely and unrelated accidents or errors to occur concurrently to cause a criticality accident.

2.3.1.3 Materials of unknown enrichment shall be handled as if it were at the maximum enrichment credible under the circumstances.

2.3.1.4 Where the mechanical design of equipment is important to radiation protection and nuclear criticality safety, the mechanical integrity under normal and credible accident conditions shall be considered.

2.3.1.5 All transfers of enriched uranium oxides from geometry-controlled equipment or containers to moderation-controlled equipment or containers shall require moisture analyses prior to the transfers.

2.3.1.6 Prior to addition of moderating materials (such as porosity control additive) to enriched uranium under moderation control, an analysis shall be performed to confirm the correct additive.

2.3.1.7 Prior to addition of lubricating materials to gearboxes, etc., internal to enriched uranium processing equipment (such as MAP Line blenders) under moderation-control, an analysis shall be performed to confirm the correct lubricant.

2.3.1.8 The records of nuclear criticality safety analyses shall include a floor plan of the SNM process and fixed storage areas showing the spacing requirements imposed to control neutron interaction except for portable storage devices and shipping containers which have been analyzed for such storage, and have established spacing requirements to control interaction.

2.3.2 <u>Nuclear Criticality Safety Values</u>

2.3.2.1 <u>Maximum Permissible Values</u>

(1) Maximum Permissible Values for subcrits with a maximum U-235 enrichment up to 5.0 w/o are established in tabular form as follows:

Figure 2.3.2.1	Batch or Mass Controlled Subcrits
Figure 2.3.2.2	Volume Controlled Subcrits
Figure 2.3.2.3	Cylinder Diameter or Equal Cross Sectional Area Controlled Subcrits

2.3.2 (Continued)

Figure 2.3.2.4 Slab Thickness Controlled Subcrits

Figure 2.3.2.5 Batch or Mass Controlled Subcrits (Engineering Controls)

(2) Subcrits containing uranium enriched to greater than 5.0 w/o will be limited to 350 grams of contained U-235.

(3) Moderation-Controlled Subcrits - Systems shall be considered under moderation-control for nuclear criticality safety when the following conditions are met:

The contained special nuclear material is "dry" under normal operating conditions.

The containment precludes introduction of moderator; or, system controls, procedures, and interlocks preclude introduction of sufficient moderator to compromise the nuclear criticality safety of the system.

Moderation-controlled UF6 in approved shipping cylinders will constitute a specific MPV.

When validated KENO computer codes are used to evaluate the Keff of individual isolated subcrits, the following criteria shall apply:

The computed Keff shall not exceed 0.95 $\pm$ 2 sigma; and moderation control and neutron reflection assumptions shall include conditions expected to be encountered during both routine operations and credible accident situations, as determined by the Radiation Protection Component, taking into consideration the following: (a) potential heterogeneous effects due to particle sizes, neutron absorbers, material densities, or the separation of fissile material and absorber, (b) the possibility of accumulations of fissile material in inaccessible or unplanned locations and (c) neutron absorption capability of the materials of construction.

(4) Uranium concentration controlled subcrits will be limited to a maximum allowable concentration of 5 grams U-235 per liter (liquids). This MPV will not be applied unless it

Maximum U-235 Enrichment	Solutions Kg U	Homogeneous Oxides Pounds Oxide	Heterogeneous Oxides Pounds Oxide
1.50	315	1150	460
1.75	165	560	290
2.00	110	350	220
2.25	78	240	170
2.50	60	180	130
2.75	48	140	110
3.00	40	110	93
3.25	33	93	81
3.50	27	79	72
3.75	25	68	65
4.00	22	60	60
4.25	20	55	55
4.50	18	49	51
4.75	17	44	48
5.00	16	40	45

MAXIMUM PERMISSIBLE VALUES (MPV)
MASS

(Administrative Controls)

Figure 2.3.2.1

Maximum U-235 Enrichment	Solutions	Homogeneous Oxides	Heterogeneous Oxides
1.50	238	252	111
1.75	150	151	73
2.00	106	100	55
2.25	83	74	44
2.50	68	59	37
2.75	58	49	32
3.00	50	42	29
3.25	45	37	26
3.50	40	34	24
3.75	37	31	23
4.00	35	29	21
4.25	32	27	19
4.50	30	26	18
4.75	28	24	17
5.00	27	23	16

MAXIMUM PERMISSIBLE VALUE (MPV)
VOLUMES IN LITERS

Figure 2.3.2.2

Maximum U-235 Enrichment	Solutions	Homogeneous Oxides	Heterogeneous Oxides
1.50	24.6	22.5	16.2
1.75	19.0	18.7	14.3
2.00	16.5	16.4	13.0
2.25	15.0	14.8	12.1
2.50	13.8	13.6	11.5
2.75	13.1	12.7	11.0
3.00	12.5	12.1	10.5
3.25	12.0	11.5	10.2
3.50	11.7	11.0	9.8
3.75	11.4	10.6	9.5
4.00	11.1	10.4	9.3
4.25	10.9	10.0	9.0
4.50	10.7	9.8	8.8
4.75	10.5	9.6	8.6
5.00	10.4	9.4	8.4

MAXIMUM PERMISSIBLE VALUES (MPV)

CYLINDER DIAMETER IN INCHES OR EQUAL CROSS-SECTIONAL AREA

Figure 2.3.2.3

Maximum U-235 Enrichment	Solutions	Homogeneous Oxides	Heterogeneous Oxides	Close Packed Clad Rods
1.50	12.5	11.8	8.4	32
1.75	10.0	9.6	7.2	32
2.00	8.8	8.2	6.4	32
2.25	8.0	7.3	5.9	32
2.50	7.4	6.7	5.5	32
2.75	7.0	6.2	5.1	32
3.00	6.7	5.8	4.8	32
3.25	6.4	5.5	4.6	20
3.50	6.2	5.2	4.4	20
3.75	6.0	5.0	4.2	20
4.00	5.8	4.8	4.0	20
4.25	5.6	4.7	3.8	15
4.50	5.5	4.6	3.7	15
4.75	5.3	4.4	3.6	15
5.00	5.2	4.3	3.5	15

MAXIMUM PERMISSIBLE VALUES (MPV)
SLAB THICKNESS IN INCHES

Figure 2.3.2.4

Maximum U-235 Enrichment	Solutions (Kg U)	Homogeneous Oxides (Pounds Oxide)	Heterogeneous Oxides (Pounds Oxide)
1.50	525	1918	766
1.75	275	935	484
2.00	183	583	367
2.25	130	400	282
2.50	100	301	216
2.75	80	235	183
3.00	67	183	154
3.25	55	154	136
3.50	45	132	121
3.75	42	114	110
4.00	37	99	99
4.25	33	92	92
4.50	30	81	84
4.75	28	73	81
5.00	27	66	73

MAXIMUM PERMISSIBLE VALUES (MPV)
MASS

(Engineering Controls)

Figure 2.3.2.5

2.3.2 (Continued)

can be demonstrated that the precipitation of the SNM at higher concentrations due to process failures are not credible.

(5) The process quarantine tanks which are safe by concentration control and which are part of a continuous processing line shall be filled with a glass volume fraction of at least 30 per cent borosilicate glass Raschig rings. The Raschig rings shall be purchased in accordance with the specifications listed in ANSI/ANS-8.5-1979. During operation the glass Raschig rings will be exposed to a basic solution (NH4OH) which will have a pH less than 11.0, temperature of 60 degrees C or less, and a fluoride concentration of about 4 per cent. The Raschig rings shall be acid washed on a routine basis with a weak acid solution containing about 15-30 per cent nitric acid. The Raschig rings shall be inspected at least once per year for (1) settling, (2) physical properties, and (3) boron content in accordance with ANSI/ANS-8.5-1979, "Use of Borosilicate-Glass Raschig Rings as A Neutron Absorber in Solutions of Fissile Material."

(6) Fixed poisons may be used in nonliquid special nuclear material systems when the following restrictions are both met: The poison shall be physically protected from abrasive action by the special nuclear material; and, nuclear criticality safety of such poisoned systems shall be verified by computer calculations. The Radiation Protection Component shall confirm that these fixed poisons are initially installed in such systems, and are replaced after maintenance of such systems.

(7) Subcrits composed of fuel assemblies shall be limited by reactivity. The computed Keff, including allowances for computational error, shall not exceed 0.95. These

2.3.2 (Continued)

computations shall be performed by individuals with qualifications contained in Subparagraph 3.3.7 or 3.3.8. The results of these computations shall be independently reviewed within the department performing the calculations and approved by the department manager before being transmitted to the NFD Manufacturing Department.

2.3.2.2 Nuclear Interaction Criteria

Inasmuch as subcrits limited in accordance with Maximum Permissible Values (MPV) or validated KENO Computer codes may contribute to array interaction, one of the following subcrit spacing requirements shall be observed for subcrits which are not isolated from other SNM.

(1) Geometry controlled subcrits shall be evaluated to determine their corresponding "fraction critical" values. Assigned "fraction critical" values shall be limited to a maximum value of 0.3. Physical dimensions corresponding to "0.3 fraction critical" values are given in Figures 2.3.2.6 and 2.3.2.7.

These subcrits shall be spaced to provide a maximum "smeared" slab thickness (ts) not to exceed 25 per cent of the respective minimum critical reflected slab thickness (tc), i.e., ts/tc $\leq$0.25. The appropriate minimum critical reflected slab thicknesses shall be determined from Figure 2.3.2.8.

Alternatively, if the subcrit "fraction critical" value and "smeared" slab thickness limitation specified above cannot be met for subcrits $\leq$MPV, the spacing of the array shall be (a) verified using solid angle criteria of TID-7016 Revision 2, or (b) verified by validated KENO calculations using the ANSI N16.9-1975 validation standard with the following two exceptions relating respectively to paragraphs 4.3 and 4.8 of the standard:

2.3.2.2 (Continued)

(a) Whenever available, the uncertainties in experimental data shall be considered in establishing bias; otherwise, the quoted experimental data shall be used directly in establishing the bias.

(b) When appropriate experimental data are available, they shall be used in validation; otherwise, calculations shall be validated using appropriate calculations performed independently by a recognized authority with demonstrated competence in the application of nuclear criticality safety computer calculations.

(2) The solid angle criteria of TID-7016, Revision 2, shall apply as supplemented by the following reflector conditions: (a) no more than full water reflection on six sides of the array or (b) no more than the equivalent of full concrete reflection on three sides of the array. The solid angles shall be determined by the point-to-plane method described in this document, except that the average solid angles determined from the curves in Section F of Y-1272 may be used for similar and parallel subcrits. The multiplication factor (Keff) of a subcrit shall be determined from Figure 2.3.2.10 or 2.3.2.11, but shall be limited to a maximum value of 0.8.

(3) When array interaction is evaluated using validated KENO computer codes, the maximum allowable calculated Keff + 2 sigma, corresponding to conservatively defined conditions, shall be limited to ≤ 0.95. In addition, the following three conditions shall also apply: (a) test cases developed by the original licensee analyst shall be reviewed and approved by a manager, or a senior staff member of the Radiation Protection Component other than the original analyst prior to presentation to the vendor, (b) results of computer calculations performed by the

VOLUME IN LITERS

Maximum U-235 Enrichment	SOLUTIONS .3 Fraction Critical Restriction	HOMOGENEOUS OXIDES .3 Fraction Critical Restriction	HETEROGENEOUS OXIDE .3 Fraction Critical Restriction
1.50	132	156	54
1.75	83	90	40.5
2.00	63	60	31.8
2.25	51	46.2	26.7
2.50	43	38.1	23.4
2.75	38	33	21.0
3.00	34	29.1	19.2
3.25	31	26.1	17.7
3.50	29	23.7	16.2
3.75	27	21.9	15.3
4.00	25	20.7	14.4
4.25	23.4	19.5	13.5
4.50	22	18.3	12.9
4.75	21	17.4	12.5
5.00	20.4	16.5	12.0

VOLUME SUBCRIT CRITERIA FOR NUCLEAR INTERACTION EVALUATIONS

Figure 2.3.2.6

CYLINDER DIAMETER IN INCHES
OR EQUAL CROSS SECTIONAL AREA

Maximum U-235 Enrichment	SOLUTIONS .3 Fraction Critical Restriction	HOMOGENEOUS OXIDES .3 Fraction Critical Restriction	HETEROGENEOUS OXIDE .3 Fraction Critical Restriction
1.50	19.9	19.3	13.9
1.75	16.5	16.5	12.5
2.00	14.2	14.4	11.5
2.25	13.0	13.1	11.0
2.50	12.0	12.0	10.4
2.75	11.5	11.5	10.0
3.00	10.8	10.9	9.6
3.25	10.5	10.5	9.3
3.50	10.2	10.1	9.0
3.75	10.0	9.9	8.8
4.00	9.9	9.7	8.7
4.25	9.6	9.4	8.5
4.50	9.4	9.2	8.4
4.75	9.3	9.1	8.2
5.00	9.2	9.0	8.0

CYLINDRICAL SUBCRIT CRITERIA FOR
NUCLEAR INTERACTION EVALUATIONS

Figure 2.3.2.7

Maximum U-235 Enrichment	Solutions	Homogeneous Oxides	Heterogeneous Oxides	Close Packed Fuel Rods
1.50	15.0	14.2	10.0	38.1
1.75	12.0	11.5	8.6	38.1
2.00	10.5	10.0	7.7	38.1
2.25	9.6	8.9	7.1	38.1
2.50	8.9	8.0	6.6	38.1
2.75	8.4	7.4	6.2	38.1
3.00	8.0	7.0	5.8	38.1
3.25	7.7	6.6	5.5	23.8
3.50	7.4	6.3	5.2	23.8
3.75	7.1	6.0	5.0	23.8
4.00	6.9	5.8	4.8	23.8
4.25	6.7	5.6	4.6	17.8
4.50	6.5	5.4	4.4	17.8
4.75	6.4	5.3	4.3	17.8
5.00	6.2	5.2	4.1	17.8

MINIMUM CRITICAL REFLECTED SLAB THICKNESSES
FOR GEOMETRY CONTROLLED SUBCRITS IN INCHES

Figure 2.3.2.8

2.3.2.2 (Continued)

licensee's vendor shall be reviewed by the manager, or a senior staff member of the vendor's Nuclear Criticality Safety Component, other than the original analyst, prior to transmittal to the licensee and (c) upon receipt of the vendor's analytical results, the Radiation Protection Engineering Manager or a qualified designee shall review them prior to their application at the facility.

Personnel qualification requirements shall meet those specified in Subparagraphs 3.3.7 or 3.3.8.

(4) Mass controlled subcrits shall be spaced to provide a maximum "smeared" surface density as given in Figure 2.3.2.9.

(5) Surface density area requirements shall be specified as follows:

(a) Mass Controlled Subcrits: Squares, circles or hexagons.

(b) Volume Controlled Subcrits: Squares, circles or hexagons.

(c) Vertical cylinders or similar geometry controlled subcrits: Squares, circles or hexagons.

(d) Horizontal and inclined cylinders or similar geometry controlled subcrits: Rectangular areas with equal distance spacing between the subcrit's normal projection on the area and the 4 sides of the surface area boundary.

(e) Slab Controlled Subcrits: Rectangular areas with equal distance spacing between the subcrit and surface area boundary.

However, two or more adjacent coplanar subcrits can be considered to be parts of a single continuous slab, and interaction between them need not be considered. The required surface area spacing around the redefined slab shall then be based on the cumulative quantity of material permitted.

Maximum U-235 Enrichment	Kg U/Ft.^{2}	or	Pounds Uranium Oxide/Ft.^{2}
1.50	29.5		73.8
1.75	22.5		56.3
2.00	18.3		45.8
2.25	15.4		38.5
2.50	13.3		33.2
2.75	11.8		29.5
3.00	10.3		25.7
3.25	9.2		23.0
3.50	8.3		20.7
3.75	7.6		19.0
4.00	7.0		17.5
4.25	6.4		16.0
4.50	5.95		14.9
4.75	5.50		13.7
5.00	5.15		12.9

MASS CONTROLLED SUBCRITS
MAXIMUM "SMEARED" OR AVERAGE SURFACE DENSITY
VALUES APPLY TO SOLUTIONS,
HOMOGENEOUS OXIDES AND HETEROGENEOUS OXIDES

Figure 2.3.2.9

2.3.2.2 (Continued)

(f) Stacked subcrits: Surface areas are established on both a horizontal and vertical plane by projecting the subcrits onto the respective planes. Surface areas are symmetrical to the respective normal projections of the subcrits, and have equidistant spacing between their boundaries and the normal projections of the subcrits on the areas. Horizontal (floor) surface areas are established for the sum of the subcrits in each stack. Vertical surface areas are established for individual subcrits. Minimum vertical spacings between subcrits within a stack, and minimum horizontal spacings between stacked subcrits, are determined by arranging the respective surface areas, as defined above, such that no overlaps exist.

(6) Moderation-controlled subcrits ("Dry") of 5 w/o or less enriched uranium, when limited as follows shall not be considered to contribute to interacting arrays: (a) Maximum Permissible Values, (b) in closed containers or configurations which would not retain water, (c) located outside of areas assigned to interacting subcrits, (d) no sprinkler system in the area, (e) no use of water or other hydrogenous agents for fire fighting purposes and (f) appropriate nuclear criticality safety signs posted in the controlled area.

(7) Concentration controlled subcrits (with or without borosilicate glass Raschig rings) are not considered to contribute to nuclear interaction provided that they are outside of areas assigned to interacting subcrits.

(8) Notwithstanding other spacing requirements, any subcrit shall be separated by at least 12 inches from any other subcrit.

(9) Surface density areas shall be assigned to interacting subcrits and identified on related criticality layout

2.3.2.2 (Continued)

drawings or sketches. When they provide for more than the 12-inch minimum edge-to-edge separation between adjacent subcrits, the surface density areas shall be identified individually or collectively by taping or painting the manufacturing floor. Alternatively, these areas may be indentified by posting the respective criticality layouts in visible location(s) within the manufacturing lines or functional areas of the plant.

2.3.2.3 <u>Pipe Intersections</u>

The diameters of intersecting pipes, within 3 diameters of the intersection, shall be reduced to maximum values determined by multiplying the applicable MPV or .3 fraction critical restriction value by a factor as follows:

Ells	.92
Tees	.84
Crosses or Wyes	.76

No two intersections may occur within 3 diameters (axis to axis) of one another. Other intersections of individually safe pipes are also permissible if the sum of the cross sectional areas of all pipes is equal to or less than the corresponding area of the permissible intersection pipe diameters derived using the above reduction factors. Thus the effective diameter, d_e, of an intersection is

$$d_e = [(\sum_i^n d_i^2)/n]^{1/2}$$

where d_i = diameter of the "i^{th}" branch of the intersection

n = number of branches:

2 for ells

3 for tees and wyes

4 for crosses

2.3.3 <u>Radiation Protection, Nuclear Criticality Safety and SNM Safeguards Administration</u>

2.3.3.1 Plant policies and procedures related to radiation protection, nuclear criticality safety and SNM safeguards are established

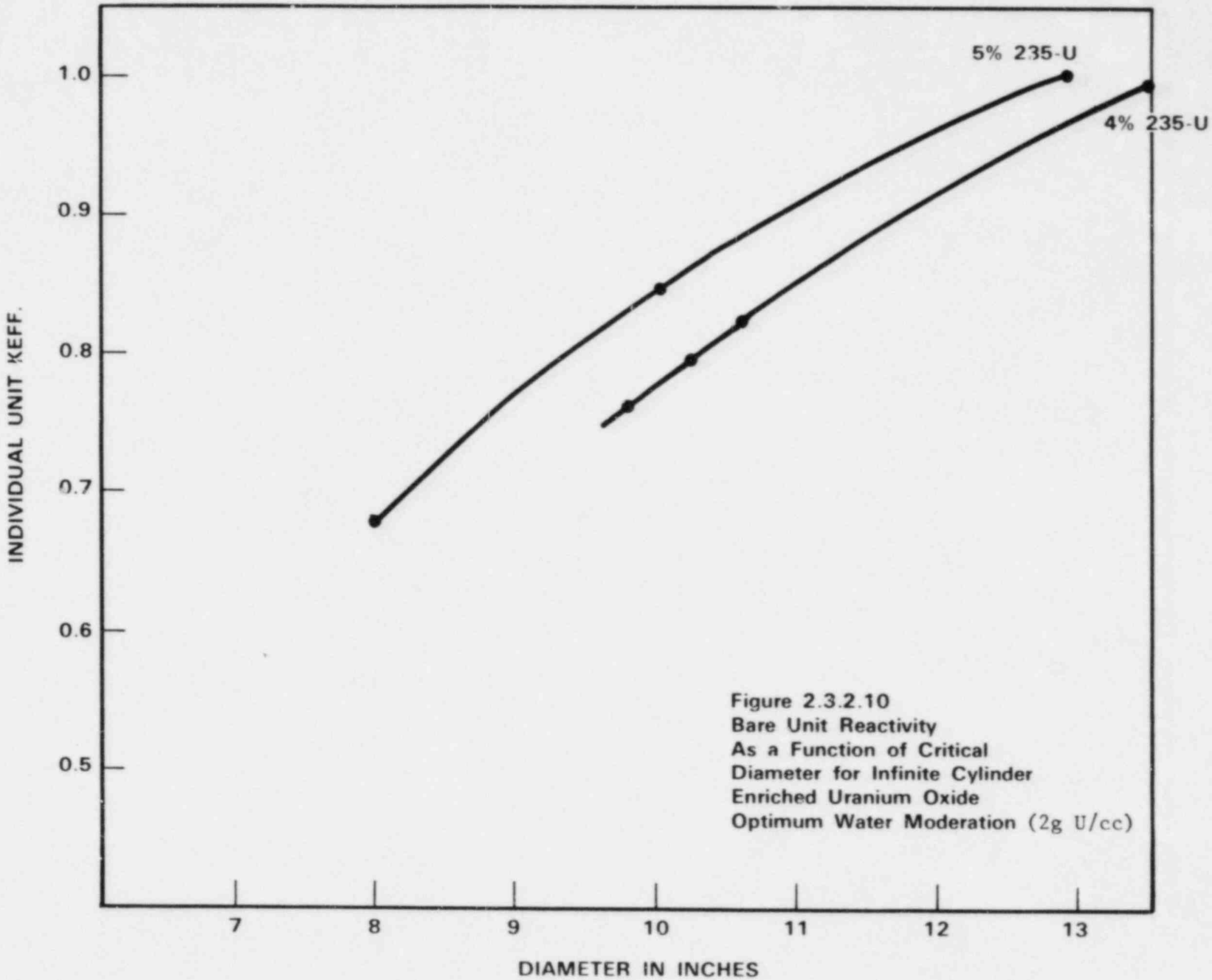


Figure 2.3.2.10
Bare Unit Reactivity
As a Function of Critical
Diameter for Infinite Cylinder
Enriched Uranium Oxide
Optimum Water Moderation (2g U/cc)

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License No. SNM-1107 Revision Submittal Date: ______ Revision No. ______

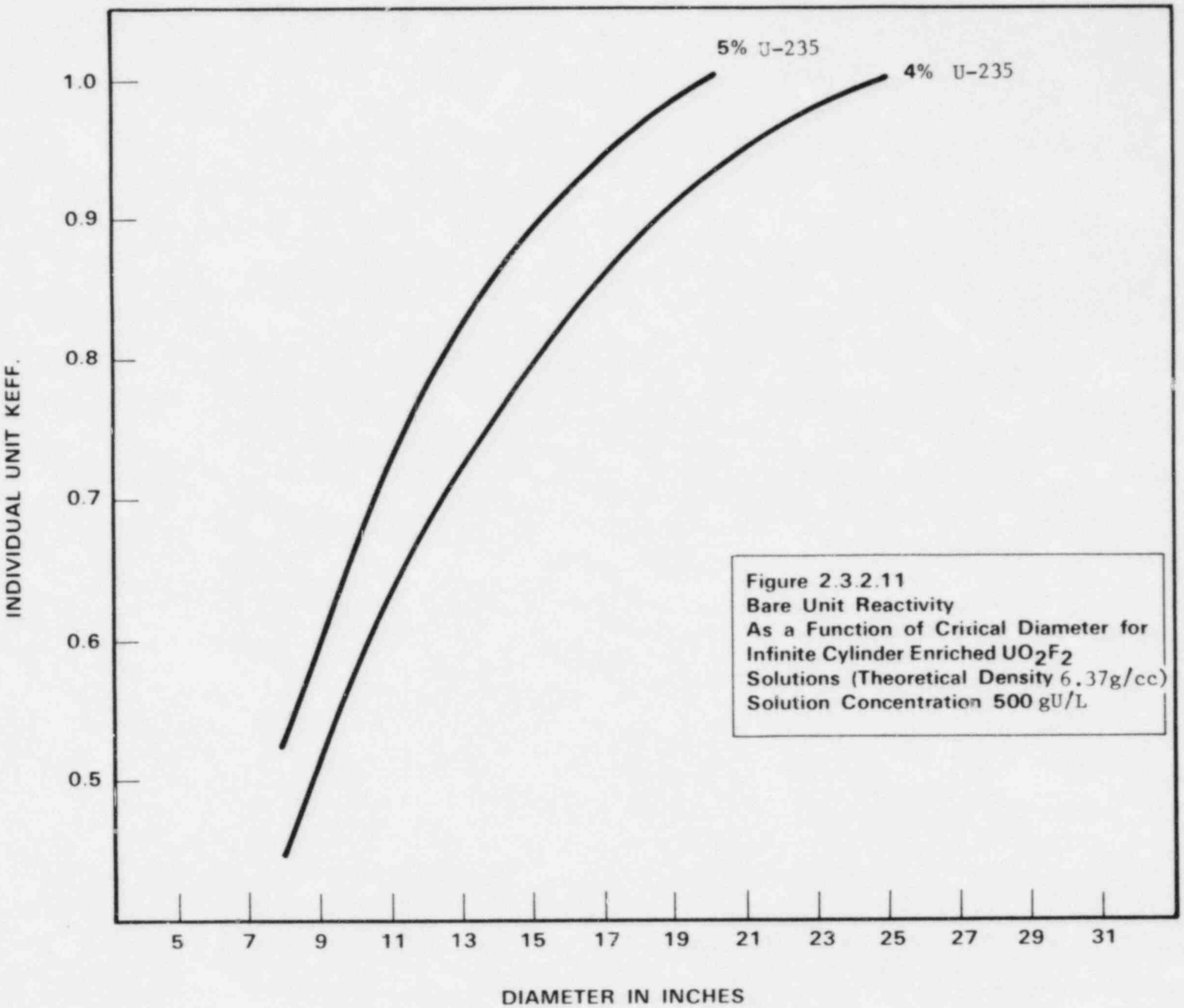


Figure 2.3.2.11
Bare Unit Reactivity
As a Function of Critical Diameter for
Infinite Cylinder Enriched UO_2F_2
Solutions (Theoretical Density 6.37g/cc)
Solution Concentration 500 gU/L

2.3.2 (Continued)

in the Regulatory Compliance Procedures Manual. Accordingly, all modifications of, planned additions to, or relocations of existing SNM processing, storage or transfer equipment shall be reviewed and approved by the Radiation Protection Component prior to effecting the change in the manufacturing operations. Such reviews shall specifically evaluate abnormal as well as normal conditions. Based on these considerations, control parameters complying with subcrit MPV and spacing requirements of this section shall be selected. These control parameters shall be reviewed and approved by the Manager of the Radiation Protection Component or a qualified designee as per Subparagraph 3.3.7, and then transmitted to cognizant line management pursuant to Subparagraph 3.1.3. Successful operations shall be confirmed following initiation of the changes.

2.3.3.2 Analysis of all systems in which an error in the analysis could lead to conditions that would not fulfill the double contingency principle, as determined independently by the Radiation Protection Engineering Manager or his designee, shall be reviewed by a senior member (not the original analyst of the relevant system) of the component, prior to implementation of the system. These individuals shall meet the minimum qualifications in Subparagraphs 3.3.7 or 3.3.8.

2.4 Minimum Conditions of Operation

2.4.1 Processing operations involving SNM shall be performed routinely only when appropriate equipment having the capabilities specified in this license has been provided and is operative and when qualified line management personnel are present.

2.4.2 Nonroutine and emergency operations involving SNM shall be performed only after the particular operation has been approved by the appropriate line management function and the Radiation Protection Component, consistent with the urgency of the situation in the event of an emergency. All equipment

2.4.2 (Continued)

specified as necessary to the operation by the line management function and radiation protection function shall be provided and operative.

2.4.3 A continuing program of surveillance, air sampling and smear sampling which is adequate to detect ventilation deficiencies and assure compliance with 10 CFR 20 limits, shall be conducted.

2.4.4 UF6 cylinders shall have valve covers in place when being moved except for those which have been processed and evacuated. In the event that it is not practical to attach a valve cover to the cylinder, they may be transported without the valve covers provided that a supervisor or designee escort is provided (in addition to the individual operating the transport equipment), and the personnel involved with the transfers shall be knowledgeable in emegency actions to be taken in event of damage to the cylinder valve during the transfers.

2.4.5 Unless specifically exempted by license conditions, Westinghouse shall comply with the provisions of Title 10, Chapter 1, Code of Federal Regulations, Parts 19 and 20.

2.4.6 A Site Physical Security Plan and implementing procedures shall be maintained in accordance with 10 CFR 73.67.

2.5 Emergency Plan

A Site Emergency Plan and implementing procedures section shall be maintained in accordance with 10 CFR 70 and NUREG 0762, "Standard Format and Content for Radiological Contingency Plan for Fuel Cycle and Materials Facilities" dated July 1981.

2.6 Operating Procedures

Special nuclear material processing shall be conducted in accordance with approved written procedures or instructions.

2.6.1 Adopting and Issuing Procedures

2.6.1.1 Safety-related procedures and procedural changes shall be issued and distributed by department managers in accordance with written policies, including preparation, review and applicable approvals. Line managers shall be responsible for assuring that safety-related procedures are made available to appropriate personnel through posting of limits, training programs and other appropriate written or verbal notifications.

2.6.1.2 Radiation protection and nuclear criticality safety parameters shall be incorporated into policies and procedures which are issued by the Radiation Protection Component, approved by the Plant Manager and distributed by line management to appropriate personnel and incorporated into applicable operating procedures.

Radiation Protection Component approvals are required for all safety-related procedures and changes involving the processing, handling, storage, inspection or movement of special nuclear materials.

2.6.2 Procedure Reviews

Document	Review Frequency
Operating Procedures	As Required
Regulatory Compliance Procedures	Annual
Health Physics Operating Procedures	Annual

2.7 Environmental Monitoring

2.7.1 The Columbia Site radiological environmental monitoring program is summarized in Figure 2.7.1. Analytical sensitivities are shown on Figure 2.7.2. The location of wells and surface water monitoring stations are shown on Figure 2.7.3 and 2.7.4 respectively. The locations of air monitoring stations are shown on Figure 2.7.5.

COLUMBIA SITE RADIOLOGICAL MONITORING PROGRAM

Types of Sample	No. of Locations *	Minimum Analysis	Monitoring Frequency
Ambient Air Particulate	4 (1)	Gross Alpha	Continuous sampling
Vegetation	4	Gross Alpha and Beta	Annually
Ground Water	4	Gross Alpha and Beta	Quarterly
Surface Water (Onsite)	3	Gross Alpha and Beta	Quarterly
Surface Water (Congaree River)	3	Gross Alpha and Beta	Quarterly
Fish	1	Gross Alpha and Beta	Annually
Sediment	1	Total Uranium	Annually
Soil	4	Total Uranium	Annually

* Sampling locations shown on attached figures.

1 All samplers are located on W property.

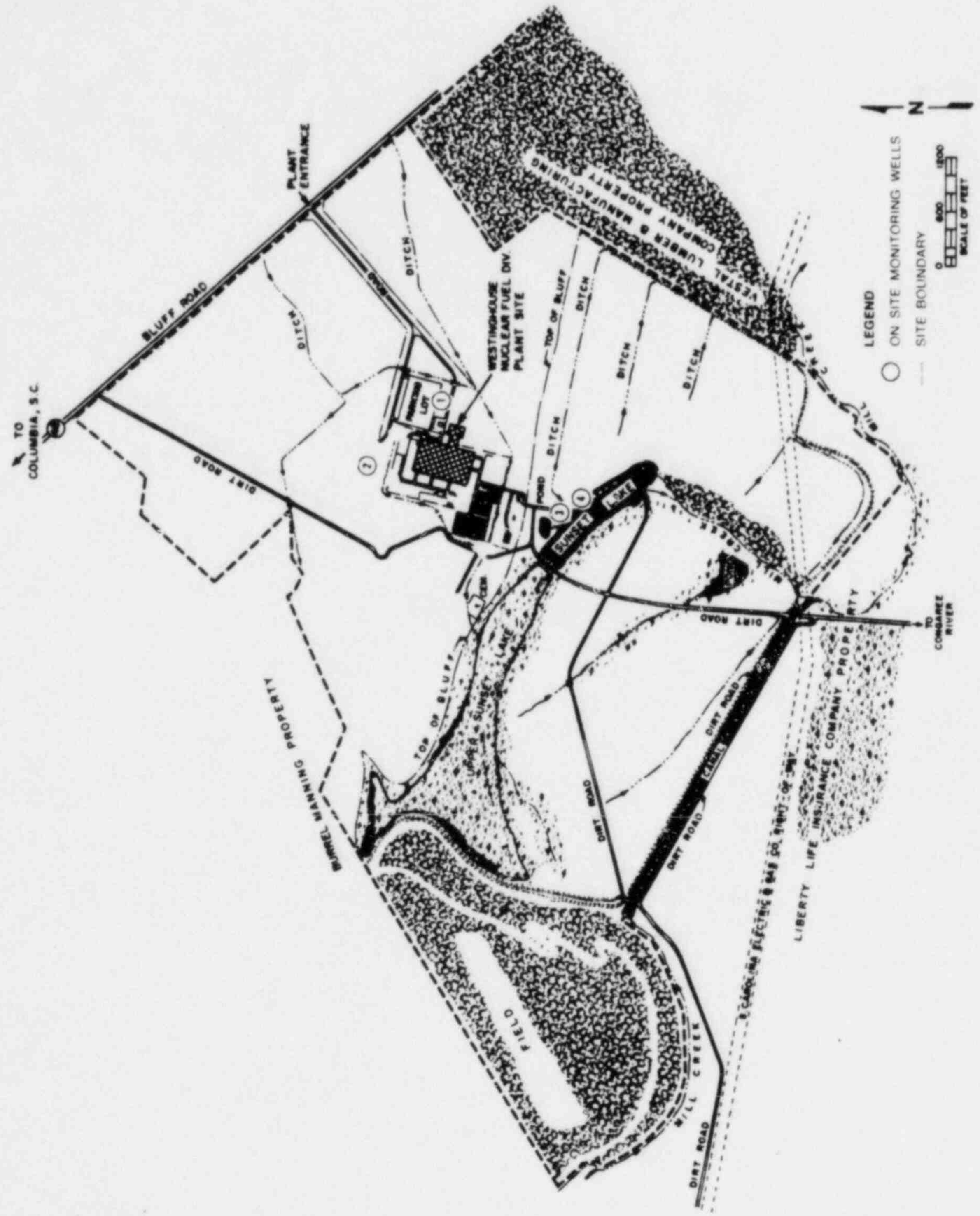
Figure 2.7.1

TYPICAL ANALYTICAL SENSITIVITIES *

Sample	Analysis	Typical Sample Size	Minimum Detectable Level (MDL)
Air Particulate	Gross Alpha	571 m^3	2.0×10^{-15} uCi/ml
Vegetation	Gross Alpha	100 g	3.0 pCi/g
	Gross Beta	100 g	0.5 pCi/g
Water Samples	Gross Alpha	1 liter	2.2×10^{-9} uCi/ml
	Gross Beta	1 liter	2.5×10^{-8} uCi/ml
	Uranium	1 liter	5.0×10^{-7} uCi/ml
Fish	Gross Alpha	30 g	1.0 pCi/g
	Gross Beta	30 g	3.0 pCi/g
	Uranium	1 kg	0.02 pCi/g
Sediment and Soil	Gross Alpha	100 g	1.0 pCi/g
	Gross Beta	100 g	3.0 pCi/g
	Uranium	100 g	0.02 pCi/g

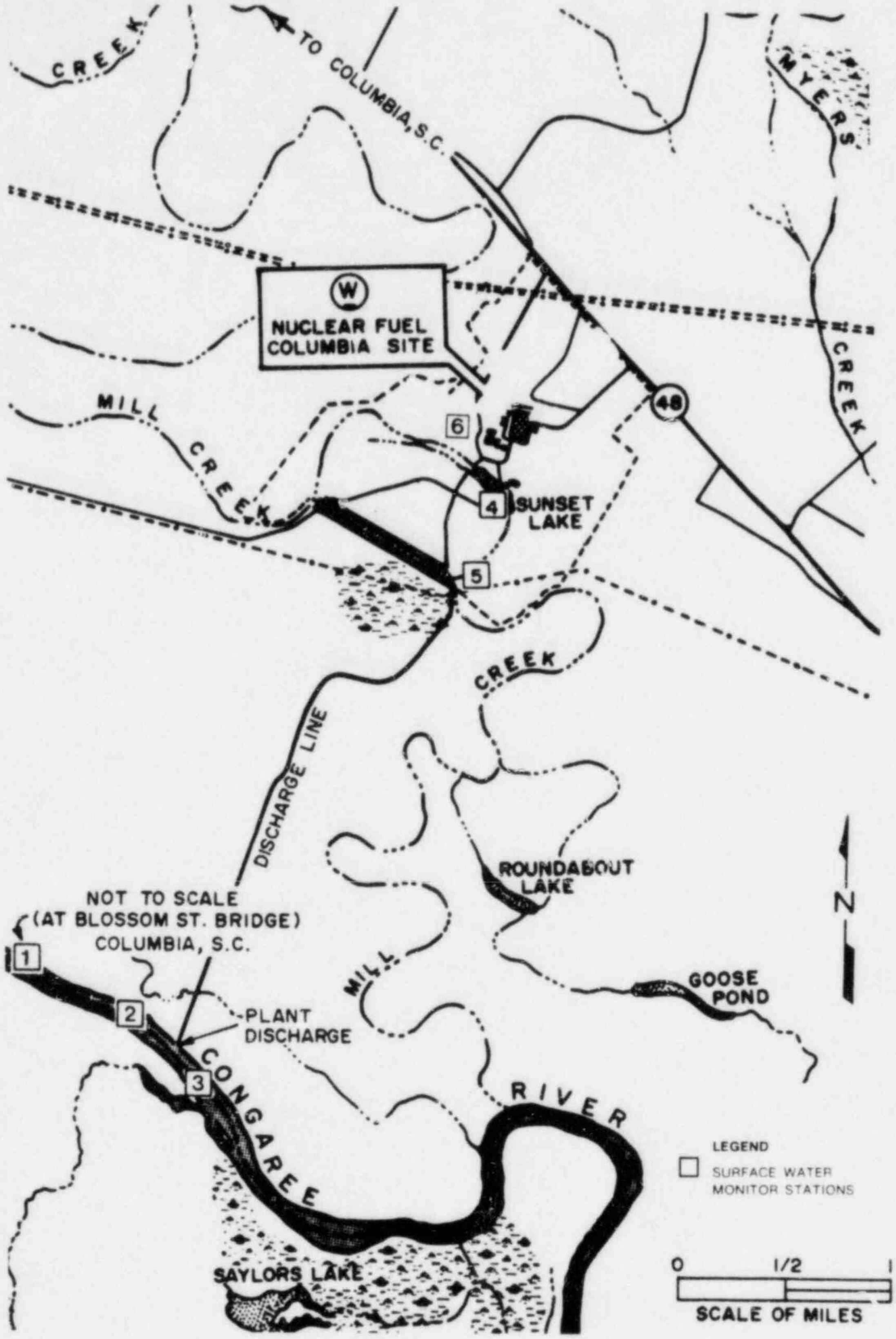
* Sensitivities are statistically representative of vendors used by Westinghouse to perform these analyses with the exception of the air particulate analyses.

Figure 2.7.2



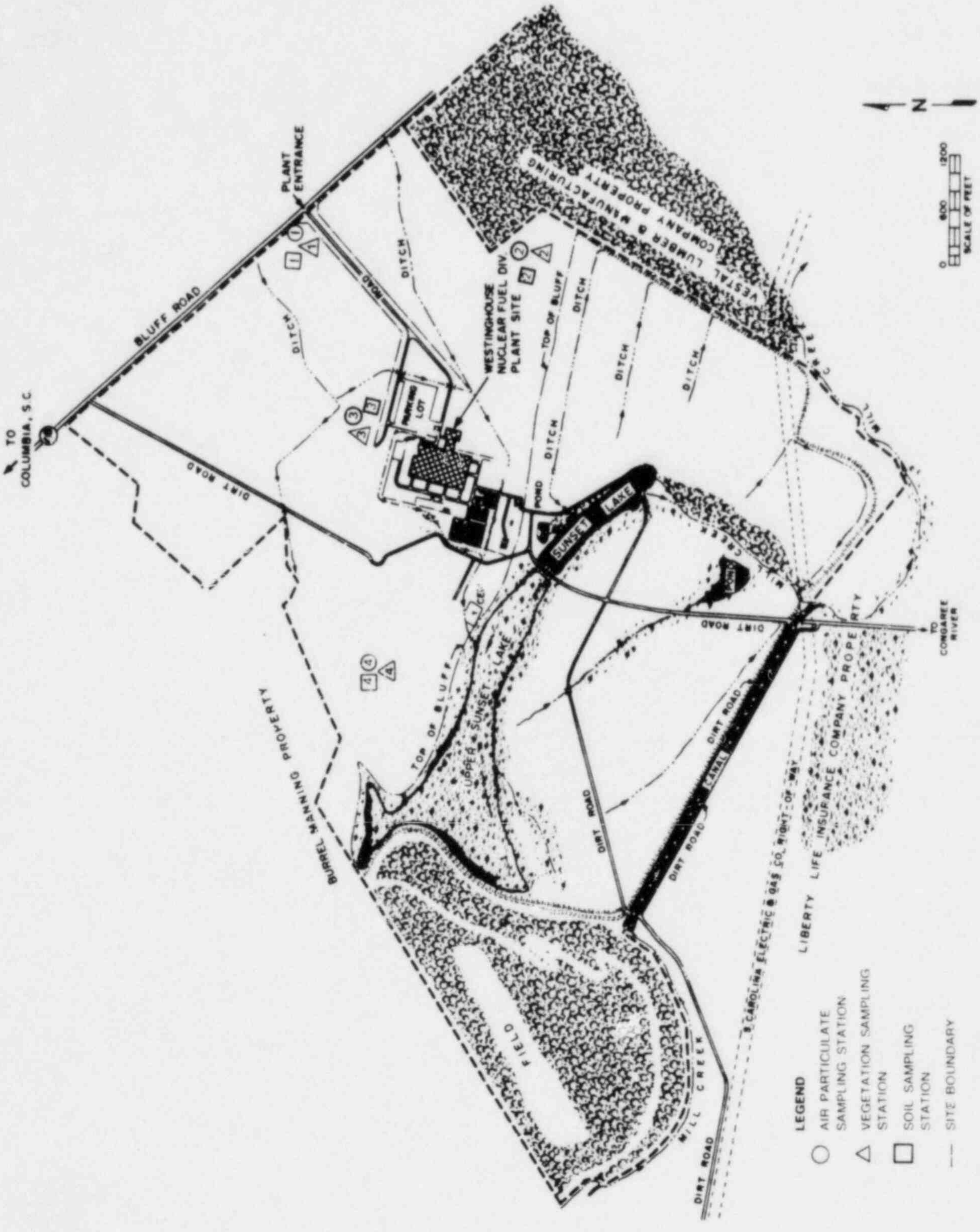
LOCATIONS OF 4" MONITORING WELLS

FIGURE 2.7.3



LOCATIONS OF SURFACE WATER MONITORING STATIONS

FIGURE 2.7.4



LOCATIONS OF AIR, VEGETATION, AND SOIL MONITORING STATIONS

FIGURE 2.7.5

SECTION 3

TABLE OF CONTENTS

3.1 Administrative Controls

3.1.1 Administrative Philosophy

The licensed activity shall be operated in accordance with the general operating philosophy that is employed in Westinghouse plants and facilities. Briefly, this philosophy provides that responsibility for all phases of operations, including safety and health protection, follows the usual lines of organizational authority. Advisory and service components are provided to assist the line management component in the analysis of operations within their control and to provide measurements, determinations and information which aid in the analysis of specific operations and situations. Such a service and staff function in no way relieves the individual line manager of responsibility for the operation of his area and for ascertaining and assuring, through appropriate management channels, that adequate service is provided. Basic policies and procedures are established by the line management component with the approval of cognizant staff components; and within the framework of these policies, the responsibility for making decisions at the operating level rests with the individual line manager. He has a basic responsibility to operate his area in a safe and orderly manner. Two higher levels of management shall approve each selection or change of an incumbent for management and professional positions.

3.1.2 Position Responsibilities

3.1.2.1 NFD Columbia Plant Manager

Overall responsibility for NFD Manufacturing activities at the Columbia Site are vested with this individual. He directs all activities of the Columbia operations and Columbia staff functions, either personally or through other management personnel. He effects and coordinates necessary

3.1.2.1 (Continued)

support activities obtained from higher Westinghouse management and performs management functions in accordance with Westinghouse policies and higher management directives.

3.1.2.2 Line Managers

A line manager shall supervise operations and activities involving licensed materials. He shall plan, direct and control activities either personally or through other management personnel. He shall perform management duties in accordance with Westinghouse policy and higher management directives. A line manager may be responsible for more than one work area. The line manager shall be directly responsible for the safe operation and control of activities in his work area(s) and for the safety of the environs as influenced by the activities conducted therein. He shall be responsible for nuclear criticality safety, radiation protection, SNM safeguards, environmental pollution control, and fire protection in all areas over which he has authority, with appropriate support from cognizant service groups.

A shift supervisor is a first-level line manager who supervises operating personnel. He shall fulfill his responsibilities as a line manager by assuring that all operations under his control are carried out in accordance with the schedules, processing procedures, nuclear criticality safety, SNM safeguards and radiation protection limits, and other instructions supplied by higher management.

3.1.2.3 Radiation Protection Component

The Radiation Protection Component shall be that organizational component of the licensed activity with the responsibility of providing authoritative professional advice and counsel to line managers on matters of radiation protection, nuclear criticality safety, SNM safeguards, environmental pollution control and emergency

3.1.2.3 (Continued)

planning, and of evaluating the effectiveness of the radiation protection program.

The Radiation Protection Component shall, to the extent practicable, be administratively independent of process supervision. The Radiation Protection Component shall be responsible for the establishment, conduct and continuing evaluation of radiation protection programs to ensure the protection of the employes and environs at the licensed facility. In particular, for any processing change which could result in a maximum credible accident (MCA) not previously evaluated or in excess of those previously evaluated, the Radiation Protection Component shall perform an MCA analysis to assure that no off-site exposure in excess of those specified in Title 10, Code of Federal Regulations, would occur. Any process change for which the analysis indicates that the MCA could produce effects in excess of those previously evaluated, however, shall be submitted for review and approval of the NRC prior to implementation.

The radiation protection program shall include as a minimum: the evaluation of releases of radioactive effluents and materials from the site; the establishment of procedures to control contamination, exposure of individuals to radiation, and integrity and reliability of radiation detection instruments; and the maintenance of the appropriate records and reports associated therewith.

Nuclear criticality safety services shall be provided to supply computed effective neutron multiplication factors (keff) for fuel assemblies, to determine subcrit MPV's and spacing requirements within the scope of Subparagraph 2.3 of this license, and to provide inspection and surveillance services against accidental criticality.

3.1.2.3 (Continued)

Specific responsibilities of the Radiation Protection Component shall include, but may not be limited to, the following:

(1) Conduct and review of nuclear criticality safety analyses.

(2) Review of nuclear criticality safety, radiation protection and SNM safeguards aspects of changes to equipment and operations associated with the processing, handling or storage of SNM.

(3) SNM safeguards and accountability of SNM.

(4) Training and monitoring of training effectiveness in

(a) radiation protection,

(b) nuclear criticality safety,

(c) emergency planning, and

(d) SNM safeguards

(5) Review and approval of design drawings of equipment and layouts associated with the processing, handling or storage of SNM.

(6) Inspection of installed equipment for conformance with radiation protection, nuclear criticality safety, and SNM safeguards requirements and documentation thereof.

(7) Audits of licensed activities for compliance with applicable State and Federal regulations, licenses and permits.

(8) Maintenance of site Health Physics Procedures Manual.

(9) Maintenance of site Regulatory Compliance Procedures Manual.

(10) Maintenance of site Fundamental Nuclear Materials Control Manual

(11) Review and approval of all site procedures specifically related to radiation protection, nuclear criticality safety, emergency planning and SNM safeguards.

(12) License administration.

(13) Routine surveillance of operations.

3.1.2.4 Materials Component

The Materials Component shall be that organizational component of the licensed activity with the designated responsibility for radiation protection, nuclear criticality safety, nuclear materials safeguards, inventory control, purchasing, and transportation of SNM. The Materials Component shall be specifically responsible for assuring that the applicable license, nuclear criticality safety, radiation protection and SNM safeguards requirements have been evaluated and communicated to line management to be incorporated into manufacturing equipment and procedures prior to their use for processing licensed material.

3.1.2.5 Safety Component

The Safety Component shall be that organizational component of the licensed activity with the designated responsibility for providing authoritative professional advice and counsel to line management on matters related to industrial safety, industrial hygiene and fire protection. It shall, to the extent practicable, be administratively independent of process supervision.

The Safety Component shall be responsible for ensuring that all applicable industrial-safety-related requirements have been evaluated, communicated to line management and incorporated into manufacturing equipment and procedures prior to their use in processing licensed material. It shall also be responsible for the establishment, conduct and continuing evaluation of all industrial safety related programs to ensure the protection of the employes and the environs at the licensed facility. The safety program shall include as a minimum: the evaluation of potential physical, chemical and fire hazards; the development and implementation of safety programs and procedures designed to minimize accidents and injuries to employes; the purchase and maintenance of industrial safety protection and monitoring

3.1.2.5 (Continued)

equipment; and the maintenance of the appropriate records and reports associated therewith.

3.1.2.6 <u>Regulatory Compliance Committee and ALARA Responsibilities</u>

The Regulatory Compliance Committee (RCC) shall be that organizational component of the licensed activity responsible for overall coordination of licensing, compliance, regulatory and radiation protection matters, and for developing policies and procedures relating to the use and storage of special nuclear materials. Special responsibilities shall include:

(1) Review and assessment of the ALARA programs.

(2) Review and assessment of the SNM control, accountability and safeguards programs.

(3) Review and assessment of occupational internal and external radiation exposures, radioactive material releases to unrestricted areas and unusual occurrences.

(4) Review of noncompliance items and assurance of implementation of corrective actions.

(5) Serving as the 10 CFR 21 Safety Review Committee for the Columbia Plant.

The Regulatory Compliance Committee shall be chaired by a qualified individual selected by the Plant Manager, and its membership shall consist of the Manager of the Radiation Protection Component and at least three other management personnel who are qualified to evaluate plant operations from the ALARA standpoint. The committee shall convene at least quarterly on a routine basis, following any major radiation incident, and when otherwise warranted by circumstances. The committee's findings, conclusions and recommendations shall be reported directly to the Plant Manager following each meeting. Appropriate action shall be taken, as required, to maintain and demonstrate compliance with ALARA regulatory requirements.

3.1.2.6.1 <u>ALARA Report</u>

A formal report shall be made by the Regulatory Compliance Committee to the Plant Manager annually reviewing personnel exposures and effluent release data to determine:

(1) if there are any upward trends developing in personnel exposures for identifiable categories of workers or types of operations, effluents or concentrations of effluents in environmental samples,

(2) if exposures and effluents might be lowered in accordance with the ALARA concept, and

(3) if equipment for effluent and exposure control is being properly used, maintained and inspected.

The report shall include review of other required audits and inspections performed during the reporting period and review of the data from the following areas: personnel exposures, bioassay results, unusual occurrences, effluent releases, inplant airborne radioactivity and environmental monitoring.

3.1.2.6.2 <u>ALARA Responsibility and Authority</u>

ALARA responsibility and authority are vested with line management, with assistance from upper management and cognizant service groups. ALARA policies, goals and expectations are identified and provided to line management by the Regulatory Compliance Committee. Specific ALARA recommendations and requirements are generated by the Radiation Protection Component in the form of radiation protection and nuclear criticality safety policies and procedures; selected ALARA criteria are also incorporated into operating procedures as applicable.

ALARA considerations are incorporated into the designs of new or modified facilities and equipment and are verified and approved by the Radiation Protection Component.

Individuals responsible for facility and equipment designs are encouraged to interface with the Radiation Protection

3.1.2.6.2 (Continued)

Component at an early stage in the design to assure that ALARA philosophies are appropriately incorporated into the project.

ALARA principles and requirements are also provided to employes during routine training sessions.

3.1.2.7 Safety Policy Committee

The Columbia Plant Safety Policy Committee functions as a mangement advisory group to assure that operations are conducted in a manner to provide maximum possible protection from injury to employes, and additionally, to ensure that health hazard concerns are adequately addressed. Each appropriate staff function shall be represented on the Committee.

3.1.3 Notification

Changes made in accordance with the provisions of the license shall be appropriately communicated by the use of circular letters, circulation of revised procedures or documents, work place meetings, postings or other suitable means. In particular, changes affecting nuclear criticality safety or radiation protection requirements shall be communicated to the affected operating personnel by suitable posting, issuance of revised, written operating procedures or instructions and word-of-mouth instructions from the shift supervisor.

3.1.4 Compliance

3.1.4.1 The shift supervisor shall enforce compliance with internal operating procedures which assure that the conditions of the license and NRC regulations will be satisfied. This shall be accomplished by providing copies of the necessary procedures and manuals to appropriate personnel and by evaluating employe performance on a continuing basis. An employe shall receive additional instruction or reindoctrination as determined by these evaluations. If prescribed procedures and policies are repeatedly or deliberately violated,

3.1.4.1 (Continued)

disciplinary action shall be taken in accordance with established corporate and Columbia Site procedures. The Radiation Protection Component staff member on duty shall have the prerogative of forbidding, through the cognizant shift supervisor, any operations, which in his opinion, involves undue immediate hazard, until the situation is reviewed with the cognizant manager or his designated alternate and there is a satisfactory resolution of the methods and procedures to be used.

3.1.4.2 Radiation Protection Component personnel on duty shall conduct continuing inspections of nuclear criticality safety and radiation protection requirements in the course of their normal activities and in accordance with written guidance. Observed items shall be immediately reported to the cognizant shift supervisor.

Repeated or continuing items shall be reported to the Manager of the Radiation Protection Component, who in turn shall report them to higher levels of line management for appropriate action.

3.1.4.3 Radiation protection, nuclear criticality safety and SNM safeguards audits shall be performed in accordance with a written plan at least monthly during operations. These audits shall be performed by a Radiation Protection Component representative.

(1) He (they) shall be accompanied by at least one cognizant line manager or his designee from each area under inspection.

(2) Any observed violations and the required corrective actions, assigned responsibilities and completion dates shall be documented in an inspection report. These inspection reports shall be transmitted to the Plant Manager, Manufacturing management and responsible managers of other areas inspected. The Radiation

3.1.4.3 (Continued)

Protection Component shall provide followup to ensure that corrective action is taken as required.

(3) AuJit findings and recommendations are reviewed to determine whether there are any undesirable trends, with results of these trend analyses documented in the ALARA Reports.

3.1.4.4 The Safety Component shall conduct bimonthly audits of fire prevention and protection in accordance with a written plan. Auditors shall be accompanied by at least one cognizant line manager or his designee from each area under inspection. Audit reports shall be transmitted to the Plant Manager, Manufacturing management and responsible managers of other aras inspected and shall contain observed violations, required corrective actions, assigned responsibilities, completion dates and shall identify what was inspected. The Safety Component shall provide followup as required to ensure that appropriate corrective actions are taken.

3.1.4.5 The Radiation Protection Component shall perform quarterly data audits of environmental quality to assure that chemical and radiological effluents and environmental concentrations are within license and permit conditions. Audit reports shall be transmitted to the Plant Manager and other cognizant managers. The Radiation Protection Component shall provide followup as required to ensure that appropriate corrective actions are taken. Environmental data are reviewed at least semiannually to determine whether there are any undesirable trends, with results of these trend analyses documented in the ALARA Reports.

3.1.5 Training

3.1.5.1 All new employes assigned to licensed activities shall receive training relative to the behavior of radiation and radioactive materials, risks involved in receiving low level radiaton exposure and basic necessities for radiation protection, ALARA practices and nuclear criticality safety.

3.1.5.1 (Continued)

Visitors shall receive indoctrination commensurate with their assignments, or shall be escorted by trained personnel.

3.1.5.2 Personnel for whom respiratory protection credit is given shall be trained in the proper use of respiratory protection equipment.

3.1.5.3 Employes designated to take part in emergency responses shall receive training commensurate with their responsibilities during such activities.

3.1.5.4 Employees who handle unencapsulated uranium shall receive instructions on handling precautions.

3.1.5.5 Employes who work with SNM shall receive general refresher training at least once every two years. This training shall be provided as part of a continuing program conducted by the Radiation Protection Component or line management to maintain a constant awareness by the employe of the necessity for radiation protection and nuclear criticality safety requirements. The effectiveness of this program shall be evaluated through written exams which are signed by the individual being tested.

Personnel who were absent from their general refresher training shall receive such retraining within one month of their return to work.

3.1.5.6 Records of formal training sessions offered by the Radiation Protection Component shall be maintained for a minimum of two years.

3.1.5.7 Line management shall be responsible for instructing personnel in the safety aspects of plant operations, including methods of dealing with process upsets or malfunctions.

3.1.5.8 Each employe shall be promptly informed of all changes in processes involving SNM, radiation protection, nuclear criticality safety controls, emergency procedures or fire protection provisions affecting his work assignment by the cognizant line manager.

3.1.6 Records

3.1.6.1 Written procedures shall be maintained, delineating the records management program for records such as plant alterations or additions, abnormal and atypical occurrences and events associated with radioactivity releases, nuclear criticality analyses, audit and inspection reports, instrument calibration, ALARA Reports, employe training and retraining, personnel exposures, routine radiation and contamination surveys, environmental surveys and Regulatory Compliance Committee meeting minutes. The procedures shall identify the type, methods of collection and retention periods of applicable documents. Records of tests, measurements and surveys required to demonstrate compliance with this license shall be retained for at least two years unless otherwise specified in 10 CFR 20.

3.1.6.2 Records of nuclear criticality safety analyses shall be maintained during the conduct of the activity and for at least six months following termination of such activities.

3.1.7 Shipping Procedures

Special nuclear material shall be delivered to a carrier for transport in packagings with a valid NRC Certificate of Compliance or DOT Specification Package.

Waste materials and quantities of SNM less than those exempted or generally licensed in 10 CFR 71 shall be shipped in accordance with the applicable DOT and IAEA regulations.

3.1.8 Radiation Work Permit Procedures

3.1.8.1 A Radiation Work Permit (RWP) shall be required for work, where radiation protection requirements are not covered by effective operating procedures, which involves the potential for internal exposures exceeding 40 MPC-hours in seven consecutive days or external doses exceeding 25 percent of the limits contained in 10 CFR 20.101(a). The RWP's are initiated by the responsible department and submitted to the Radiation Protection Component for review and approval. The

3.1.8.1 (Continued)

Radiation Protection Component shall specify the applicable requirements for the work to be performed. Approvals are also obtained from the cognizant area supervisor/manager prior to starting an activity.

3.1.8.2 The initiating department shall be responsible for assuring that only personnel who have completed the appropriate radiation safety or nuclear criticality safety training are assigned to perform work under an RWP.

3.1.8.3 A copy of the RWP shall be made available to personnel working under the RWP, and work shall be monitored by the Radiation Protection Component as required.

3.1.8.4 Specific requirements of the RWP program shall be contained in approved procedures.

3.2 Surveillance Requirements

3.2.1 Radiation Protection Procedures

3.2.1.1 General Radiation Protection Procedures

Written procedures describing general radiation protection requirements shall be maintained.

3.2.1.2 Measurement Control Procedures

The necessity for and application of instrumentation shall be established by the Radiation Protection Component.

(1) Instruments shall be routinely calibrated on a schedule established by the Radiation Protection Component. The schedule shall require calibration following acquisition, and thereafter at least following major repairs or semiannually, whichever occurs first. Alpha counting equipment used in the laboratory also shall be checked each working day when in use to determine background activity and a calibrated source shall be counted to assure proper instrument functioning. A voltage plateau and the proper counting voltage of each

3.2.1.2 (Continued)

laboratory alpha counting instrument shall be determined quarterly. Instrument calibration records shall be maintained for a period of one year.

(2) Criticality detection and alarm systems shall be checked for proper functioning on a schedule established by the Radiation Protection Component. The schedule shall require a check following installation, and thereafter at least following major repairs or semiannually, whichever occurs first.

(3) Radiation protection personnel shall check the criticality detection devices for indications of instrument failure on a random, continuing basis as their normal shop duties take them in the vicinity of a detection device or a monitoring station.

(4) Operability of portable survey instruments shall be determined prior to use.

(5) Air flow measurement devices on the fixed inplant air sample system, the air effluent sampling system and the environmental air samplers shall be calibrated annually.

(6) The fixed inplant and effluent air sample systems shall be subject to surveillance by the Radiation Protection Component (e.g., flow meters working and properly adjusted, integrity of the vacuum system and proper insertion of the filter media).

(7) The Radiation Protection Component shall perform an evaluation every two years (or more frequently if a problem is identified) of vendors used to analyze environmental or bioassay samples.

(8) The invivo counter shall be routinely calibrated at least annually and functionally tested daily when the counter is in operation.

3.2.2 Access Control

Adequate change rooms and step-off pads shall be provided in the Manufacturing Building to prevent the spread of contamination.

3.2.2.1 The Controlled Areas, which require the use of protective clothing, include the Chemical Manufacturing Area and other areas involved in the processing, use and storage of unencapsulated radioactive materials. Encapsulated radioactive materials are processed, used and stored in the Mechanical Manufacturing Areas.

3.2.2.2 Access points to the Controlled Areas are established through change rooms and step-off pads. Each access point includes a "hot" side and a "cold" side, with a step-off area provided between the hot and cold sides.

3.2.2.3 Clean protective clothing is provided for personnel entering the Controlled Area in the form of coveralls, underwear, socks, labcoats, shoecovers, safety shoes, etc. consistent with the individual's work assignment. Personal clothing, for those dressing out completely in protective clothing, is stored on the cold side. Used protective clothing is stored on the hot side and collected there for laundering.

3.2.2.4 Each entrance or access point to the Controlled Access Area shall be posted in acordance with 10 CFR 20.203 except for 10 CFR 20.303(f). In lieu thereof, a sign bearing the legend, "Every container or vessel in this area may contain radioactive material," shall be posted at entrances to each area in which radioactive materials are processed, used, or stored.

3.2.2.5 Instructions shall be posted at exit points from the Controlled Areas describing survey techniques and procedures for decontamination and instances of instrument inoperability.

3.2.2.6 Personnel survey instruments shall be provided in the change rooms and step-off pads for use by personnel leaving the Controlled Areas.

3.2.2.7 Personnel are authorized to enter Controlled Areas by virtue of management approval in accordance with the Site Security Plan and after completing the required radiation protection training.

3.2.3 Ventilation Control Procedures

3.2.3.1 Adequacy of ventilation and containment controls within the licensed activity shall be determined by continuous air sampling.

3.2.3.2 In ventilation and containment equipment and systems, the effectiveness of the final HEPA filters placed just before release of effluents to the atmosphere, shall be determined by "in-place" DOP or other suitable testing means following filter changes. For other HEPA filters in such systems, the effectiveness shall be determined using means selected by the Radiation Protection Component, unless omission of testing is authorized by the Radiation Protection Component.

3.2.3.3 The use of gloveboxes, ventilation hoods and similar devices shall be instituted when they are determined to be necessary as a result of radiation protection evaluation or measurements.

3.2.3.4 All areas where exposed radioactive materials are handled shall be sampled for airborne radioactive particulate matter by using appropriate samplers to obtain reasonably representative samples during operations, on a schedule established by the Radiation Protection Component, and when indicated by a particular operation. The frequency of surveys shall be such that any deterioration in ventilation and containment controls and operating procedures, resulting in increased concentrations, will be detected so that corrective action may be instituted.

3.2.3.4 (Continued)

(1) The location of permanently mounted continuous air sampling equipment and the frequency with which the samples will be removed and counted shall be established by the Radiation Protection Component.

(2) Permanently mounted continuous air sampling heads shall be located where potential airborne contamination hazards exist as determined by the Radiation Protection Component to obtain reasonably representative samples of air in operator locations for specific operations or "general air levels" in operational areas.

(3) Initially, all new operations shall be surveyed using portable air sampling equipment to determine the need for permanently mounted air sampling stations. Air samples shall normally be changed and counted on an eight (8) hour shift basis during production operations, unless otherwise established by the Radiation Protection Component.

3.2.3.5 All monitoring programs shall provide for investigation and/or increased sampling if the airborne radioactive concentration, not directly the result of a specific known cause, exceeds the action levels as specified in Table 3.2.3.1.

Records of the investigation and/or instructions regarding increased sampling shall be maintained for a period of one year.

3.2.3.6 For those operations which may result in the exhausting of radioactive materials to unrestricted areas, the adequacy of air effluent control shall be determined by representative stack sampling to demonstrate compliance with 40 CFR 190 by limiting the total stack discharge to 1,500 microcuries per calendar quarter. Sampling shall be performed continuously

TABLE 3.2.3.1

RESTRICTED AREA AIRBORNE RADIOACTIVITY CONCENTRATION ACTION LEVELS

Action Level *	Action Required
1.0 One sample >1.0 MPC	1.1 Investigation
2.0 Weekly average >1.0 MPC	2.1 Investigation
	2.2 Special air sampling
3.0 Quarterly average >0.5 MPC	3.1 Investigation
	3.2 Special air sampling
	3.3 Engineering evaluation of equipment, operations and/or procedures.
	3.4 Transmittal of evaluation results to the Regulatory Compliance Committee
4.0 One sample >10 MPC	4.1 Shutdown of equipment and/or operation
	4.2 Initiate corrective action(s)

* MPC - Maximum Permissible Concentration specified in 10 CFR 20.103.

3.2.3.6 (Continued)

during production operations. Samples shall be collected and analyzed on a schedule determined by the Radiation Protection Component to assure the required sensitivity of the measurement.

3.2.3.7 Velocity determinations shall be made and documented at least monthly when the equipment is in operation.

3.2.3.8 For those operations which may result in the exhausting of ammonia and fluorides to unrestricted areas, the adequacy of air effluent control shall be determined by stack sampling. Sampling shall be performed quarterly during production operations. An investigation shall be conducted of any stack discharging ammonia or fluoride concentrations that would result in site boundary concentrations in excess of 10 per cent of their ACGIH Threshold Limit Values. Appropriate corrective action, as determined from the results of the investigation, shall be taken to reduce the concentrations to levels below these values.

3.2.4 <u>Bioassay</u>

3.2.4.1 <u>Urinalysis Program</u>

Routine urine sampling frequencies are established for operators and maintenance personnel assigned to work in areas where transportable uranium compounds are processed. These personnel are requested to submit urine samples based upon the activities performed.

(1) These frequencies for submitting urine samples shall be consistent with or exceed those listed in Regulatory Guide 8.11, Table 3, for appropriate personnel.

(2) The actions specified in Table 3.2.4.1 shall be taken when routine urine samples are submitted in accordance with established plant procedures and the results exceed these limits.

3.2.4.1 (Continued)

(3) Samples so submitted shall be staggered to better represent the ongoing potential uptake of the groups being sampled.

3.2.4.2 Invivo Counts

Routine invivo counting frequencies are established for operators and maintenance personnel who normally work in areas where nontransportable uranium compounds are processed.

(1) The invivo frequencies are based upon the activities performed by the individuals and shall be consistent with or exceed Table 3 of Regulatory Guide 8.11, with the exception that fecal analyses may be employed in lieu of invivo counts for those personnel who cannot be invivo counted (e.g., claustrophobics).

(2) The actions specified in Table 3.2.4.2 shall apply when those results are exceeded.

3.2.4.3 Nonroutine Bioassay

In the event that unusual occurrences cause exposures in excess of 40 MPC hours in a seven consecutive day period, diagnostic bioassay shall be used to evaluate the extent of the exposures. Analysis of results shall be based upon models published by the ICRP, NCRP or the NRC, and may make use of site specific data or individual (personal) characteristics.

3.2.5 Contamination Surveys

3.2.5.1 Contamination surveys shall be performed on a continuing basis to evaluate the potential spread of radioactive contamination. Surveys shall be made with sufficient frequency to assure that acceptable contamination levels are maintained. The contamination levels and minimum survey frequencies given in Table 3.2.5.1 shall define the action levels described below (not applicable to process equipment). Portions of a Contamination Controlled Area may be assigned higher limits if an evaluation has been made by

URINALYSIS ACTION LEVELS
NONTRANSPORTABLE AREA
(URANIUM OXIDE AREAS)

Urinalysis Result	Action(s)
1. Sample >100 dpm/l	1.1 Confirm results
	1.2 Perform invivo count

ROUTINE SAMPLING PROGRAM: TRANSPORTABLE AREAS
CHEMICAL CONVERSION AREAS

Urinalysis Result	Action(s)
1. Sample >35 - μgU/l	1.1 Confirm result
	1.2 If results are confirmed:
	1.2.1 Impose work restriction
	1.2.2 Collect diagnostic urine samples
	1.2.3 Conduct job investigation to determine probable cause
	1.2.4 Initiate corrective action(s)
2. Sample >100 ug - U/l	2.1 Confirm result
	2.2 Impose work restriction
	2.3 Have additional urine specimen tested for albuminuria under the direction of a physician
	2.4 Conduct job investigation to determine probable cause
	2.5 Initiate corrective action(s)

Table 3.2.4.1

INVIVO MEASUREMENT ACTION LEVELS

Invivo Result	Action(s)
1. Lung Burden >150 ug U-235	1.1 Confirm results 1.2 If results were confirmed and unexpected: 1.2.1 Identify probable cause and correct or initiate additional control measures. 1.2..2 Review data to determine if others were exposed and perform bioassay measurements for them.
2. Lung Burden >200 ug U-235	2.1 All of the above 2.2 Establish work restrictions 2.3 Perform additional bioassay measurements. The specific program will be determined on a case by case basis, but shall include additional invivo counting and measurements using at least one other bioassay technique.

Table 3.2.4.2.

3.2.5.1 (Continued)

the Radiation Protection Component which demonstrates that the protective measures for that area will assure fulfillment of 10 CFR 20 provisions.

3.2.5.2 The frequency with which areas will be surveyed shall depend on the potential for contamination. Initially, new operations shall be subjected to increased contamination surveillance until experience is gained on the need for this increased frequency.

3.2.5.3 Decontamination shall be required within 3 working shifts if the average contamination level in an area exceeds the appropriate action level. Immediate decontamination or area isolation shall be required if the average contamination level in an area is greater than five times the appropriate action level. If a given smear is greater than five times the action level for that area, additional smears shall be taken to define the area requiring immediate decontamination. Follow-up surveys shall be performed prior to releasing any area requiring immediate decontamination.

3.2.5.4 A counting technique shall be used which is capable of detecting 10 dpm/smear sample at a 90 per cent confidence level (or the equivalent) when surveying unrestricted type areas.

3.2.6 External Personnel Exposures

Investigations of external personnel exposures shall be conducted if valid results exceed 75 per cent of the limits specified in 10 CFR 20.201. Corrective actions, if practicable, shall be taken when the investigation level is exceeded.

3.2.7 Nuclear Materials Management Procedures

The fundamental material controls approved by the USNRC Division of Safeguards shall be maintained.

TABLE 3.2.5.1

Guides to Alpha Activity Control Levels

Area Type	Total Activity (Direct Reading cpm)	Removable on Smear ($dpm/100\ cm^2$)	Minimum Survey Frequency
Clean Area	250	50	Monthly
Limited Area	1,100	220	Biweekly
Controlled Area	15,000	3,000	Weekly

NOTE: Values for alpha are direct readings in "counts per minute" (cpm) as read on a PAC-4G survey instrument with alpha probe (face area approximately 60 cm^2, window approximately .85 mg/cm^2, approximately 2 pi geometry) or equivalent.

3.3 Technical Qualifications

3.3.1 NFD Columbia Plant Manager

The minimum qualifications for this individual shall be a bachelor's degree, or equivalent experience, in one of the sciences or engineering, with five years' experience in nuclear plants or laboratories. In addition, he shall be generally knowledgeable in the policies and procedures used at the NFD Columbia Plant.

3.3.2 Line Manager

The minimum qualifications of a line manager, above the lead supervisor level, shall be a bachelor's degree, or equivalent experience, in one of the sciences or engineering, with two years' experience in nuclear plants or laboratories. A shift supervisor shall have demonstrated management capability by his work accomplishments.

All line managers shall have demonstrated proficiency in the application of the licensed activity's radiation protection program as it relates to limitations and radiological controls on work activities in his assigned radiation or radioactive materials area. Additionally, each line manager of an area where SNM is handled shall have demonstrated proficiency in the application of nuclear criticality safety control procedures and will be knowledgeable in the procedures applicable to the work area(s) under his management.

3.3.3 Manager, Radiation Protection Component

The minimum qualifications of the Manager of the Radiation Protection Component shall be a bachelor's degree in one of the physical or biological sciences or engineering, with two years' experience in assignments involving radiation protection and nuclear criticality safety. He shall have demonstrated proficiency in health physics and nuclear criticality safety and administration of the Radiation

3.3.3 (Continued)

Protection Component functions. His experience shall include at least one year of directly relevant nuclear criticality safety analysis.

3.3.4 Manager, Radiation Protection - Engineering Component

The minimum qualifications of the Manager of the Radiation Protection - Engineering Component shall be a bachelor's degree in one of the physical or biological sciences or engineering, with two years' experience in assignments involving radiation protection and nuclear criticality safety. He shall have demonstrated proficiency in radiation protection and nuclear criticality safety, and his experience shall include at least one year of directly relevant nuclear criticality safety analysis.

3.3.5 Manager, Radiation Protection - Operations Component

The minimum qualifications of the Manager of the Radiation Protection - Operations Component shall be five years of experience in the nuclear industry. His experience shall include at least two years of directly relevant experience in radiation protection. He shall have demonstrated proficiency in health physics and administration of the health physics component functions.

3.3.6 Manager, Safety Component

The minimum qualifications of the Manager of the Safety Component shall be a bachelor's degree, or equivalent experience, in engineering, industrial management or one of the physical or biological sciences, with two years' experience in assignments involving industrial safety and hygiene. He shall have demonstrated proficiency in safety engineering methods, techniques, procedures and standards, as well as the administration of the component functions.

3.3.7 Nuclear Criticality Safety Engineer

The minimum qualifications of the Nuclear Criticality Safety Engineer shall be a bachelor's degree in one of the physical

3.3.7 (Continued)

sciences or engineering, with two years' experience in related engineering. His experience shall include at least one year of directly relevant nuclear criticality safety analysis. He shall have demonstrated proficiency in nuclear criticality safety and the performance of the position function.

3.3.8 Nuclear Criticality Safety Services Personnel

Nuclear criticality safety services shall be provided to supply computed effective neutron multiplication factors (keff) for fuel assemblies and to determine subcrit MPV's and spacing requrements within the scope of Subparagraph 2.3.2 of this license.

Individuals who compute keff values shall have at least a bachelor's degree in science or engineering. They shall have at least five years' experience in a responsible position in a nuclear field such as engineering, physics or chemistry, including at least one year of directly relevant experience in reactivity calculations.

3.3.9 Manager, Materials Component

The minimum qualifications of the Manager of the Materials Component shall be a bachelor's degree with five years' experience in nuclear plants or laboratories. In addition, he shall be generally knowledgeable in the procedures used in the NFD Manufacturing Department.

SECTION 4

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Authorizations and Exemptions

4.1 Release for Unrestricted Use

4.1.1 Materials and equipment may be removed from a contamination controlled area for transfer to individuals for the purpose of unrestricted release when each of the following conditions is satisfied:

4.1.1.1 A reasonable effort to eliminate residual contamination has been made.

4.1.1.2 All surfaces are accessible for survey, or it is reasonable to assume from the design and usage that no radioactive materials could have contaminated inaccessible surfaces without having contaminated the outer surfaces as well.

4.1.1.3 Survey of accessible surfaces by the radiation protection function has verified that contamination release limits have been met.

4.1.1.4 Fixed contamination is less than 15,000 dpm/100 cm^2 maximum and 5,000 dpm/100 cm^2 average. The maximum contamination level applies to an area of not more than 100 cm^2. Average contamination levels are averaged over an area of one square meter or less.

4.1.1.5 Removable contamination is less than 1,000 dpm/100 cm^2 as determined by wiping the 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.

4.1.1.6 Beta-gamma radiation levels are less than 1 millirad per hour maximum with an average less than 0.2 millirad per hour as measured one centimeter from the surface through not more than 7 milligrams per square centimeter of total absorber.

4.1.2 Prior to release of premises (buildings) for unrestricted use, a comprehensive survey shall be made which establishes that contamination is within the limits of Subparagraph 4.1.1. A copy of the survey report shall be filed with the

4.1.2 (Continued)

Division of Fuel Cycle and Material Safety, USNRC, Washington, D. C. 20555, and also the Director of the Regional Office of the Office of Inspection and Enforcement, USNRC, Region II. The report should be filed at least 30 days prior to the planned date of abandonment.

The survey report shall:

4.1.2.1 Identify the premises.

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

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

4.1.2.4 State the findings of the survey including radiation and contamination units.

4.2 <u>Use of Material at Offsite Locations</u>

4.2.1 Up to 15 grams of unirradiated U-235 may be used for demonstration and testing purposes at offsite locations within the United States as defined in Section 70.4(n), Title 10, Part 70, Code of Federal Regulations, Chapter 1, except in Agreement States as defined in Section 70.4(o).

4.2.2 The material shall be subject to the following controls: (1) The material, while not being transported, shall be subject to control by a Westinghouse employe approved by the Manager of the Radiation Protection Component, and (2) The individual specified in (1) shall be cognizant of, and shall ensure compliance with, the radiation protection requirements of 10 CFR 20 and applicable Westinghouse policies and procedures. In addition, he shall ensure that there will be no comingling with other special nuclear material which would cause the total mass to exceed 350 grams U-235.

4.3 Record Storage

4.3.1 Small quantities of source, special nuclear material and byproduct materials may be abandoned or disposed of when present as trace contamination on certain papers, notebooks, films and other items which are being retained for record retention purposes. These records may be transferred to locations which will be used primarily for their storage. No special controls will be imposed on them, and they may be randomly mingled with other noncontaminated records.

4.3.2 The records from operations, where radioactive contamination may occur, shall be surveyed to determine the levels of radioactivity. They will not be released if the following levels, as determined by survey instrument measurements, are exceeded:

	Activity (dpm/100 cm^2)	
	Alpha	Beta-gamma
Average	220	660
Maximum	2,200	6,600

4.4 Exemptions from the Requirements of 10 CFR 70.24

The following areas are exempted from the requirements of 10 CFR 70.24 when the specified conditions are met:

4.4.1 Isolated Areas

The following administratively controlled areas are limited to 700 grams U-235 per area and are remote from other SNM operations:

4.4.1.1 Machine Shop Area

4.4.1.2 Office and Conference Room Areas

4.4.1.3 Analytical Chemistry Laboratories (includes U-233 as authorized in this license)

4.4.1.4 Health Physics Laboratory

4.4.1.5 Metallurgical Laboratories

4.4.1.6 Development Laboratories

4.4.2 Low Concentration Storage Areas

Certain areas are administratively controlled to assure that all SNM present is in quantities (1) less than 350 grams U-235 per package, provided that there is not more than 5 grams U-235 in any cubic foot within the package, or (2) less than or equal to 50 grams U-235 per package, provided that the average concentration within the package does not exceed 5 grams U-235 per cubic foot. Each area shall be nuclearly isolated from SNM which is more highly concentrated. Concentrations shall be determined and the packages shall be labelled prior to their transfer to these areas.

4.4.3 Shipping Package Storage Areas

Certain areas contain only SNM which is contained in USNRC-approved shipping packages. The maximum number of packages permitted in such an area shall be as follows:

Fissile Class I	250 packages
Fissile Class II	Packages totaling 50 transport units
Fissile Class III	One approved shipping quantity

A maximum of four such arrays shall be permitted, each of which shall be nuclearly isolated from other SNM.

4.5 Nonradioactive Industrial Wastes

Under the auspices of 10 CFR 20.302, industrial waste treatment products (primarily calcium fluoride) in which the mean concentration of uranium constituents does not exceed 30 picocuries per gram shall be released for disposition in a chemical disposal site or industrial landfill without continuing USNRC licensing controls. A sampling plan shall be implemented which will characterize the industrial wastes in accordance with NUREG/CR-2082 as follows:

4.5.1 The estimation of the population mean for the uranium concentration shall be representative of the industrial wastes located onsite and any subsequent shipment of this material.

4.5.2 The sample size, used to calculate the mean value, shall be determined such that the 95 per cent confidence limit of the mean value shall be less than 25 per cent of this value (e.g., $\pm ts/\sqrt{n} < .25\ \bar{x}$) where "t" is from students t tables, "s" is the standard deviation, "$\bar{x}$" is the mean, and "n" is the population size.

In addition to these requirements, the sampling plan shall provide a minimum confidence level of 95 per cent that the true mean value determined for these industrial waste products is less than the maximum allowable limit of 30 picocuries per gram of dry material. Records pertaining to the release of these materials shall be maintained along with the names of the individuals receiving these materials. Reasonable efforts shall be made to reduce the level of contamination in line with the ALARA concept.

4.6 Possession of Finished Fuel Assemblies at Reactor Sites

4.6.1 Finished fuel assemblies may be possessed by Westinghouse at reactor facilities anywhere within the United States for the purpose of loading them into shipping packages which will then be delivered to a carrier for transport in accordance with DOT regulations and, when appropriate, the provisions of the general licenses contained in 10 CFR 71.11 and 10 CFR 71.12.

4.6.2 Operations incident to the loading shall be subject to the control of a Westinghouse employe to assure that the loaded package complies with the requirements of the USNRC and USDOT regulations and the appropriate USNRC Certificate of Compliance or USDOT specification.

4.6.3 Westinghouse requests exemption, at the loading location, from the requirements of 10 CFR 70.24, provided that when the exemption is exercised, the following additional controls shall be assured by the cognizant Westinghouse employe.

4.6.3.1 The total number of finished fuel assemblies in process at any one time shall not exceed the maximum authorized contents of the packaging being loaded.

4.6.3.2 As finished fuel assemblies are removed from the approval storage facilities, they shall be constrained in an arrangement no more reactive than that which they will assume in the shipping package.

4.6.3.3 If two (2) fuel assemblies are in movement at the same time, a 12-inch minimum edge-to-edge separation shall be maintained between them, and only one (1) fuel assembly at a time shall be loaded into the shipping package.

4.6.3.4 Once started, a loading operation shall be completed, or all the fissile materials involved shall be returned to the approved storage facility.

4.6.3.5 Loaded packages shall be stored in the approved shipping array, pending delivery to a carrier.

4.6.3.6 No more than the maximum number of packages authorized for a single shipment shall be loaded and possessed at one location by Westinghouse under this authorization at any one time.

4.6.4 The regulations of the USNRC require that the nuclear criticality safety of a single package and an array of loaded packages be conclusively demonstrated with a wide margin of safety as a prerequisite to issuing the Certificate of Compliance authorization to deliver the packages to a carrier. By establishing that the same requirements are applicable during loading operations, the same degree of nuclear criticality safety is assured and the requirement for monitoring may be relaxed.

4.7 Authorization to Act As A Private Carrier

In addition to the authorizations provided to Westinghouse under License No. SNM-1506 Docket 70-1810, special nuclear material may be possessed for the purpose of private carriage in accordance with the following provisions: (1) The materials to be transported are those described in Subparagraphs 4.2 and 4.6 of this license and for UF6 transported from DOE facilities, and (2) The private carriage is authorized from locations at which Westinghouse is authorized to possess the material to persons or locations authorized by law to receive such materials within the United States.

4.8 Transfer of Special Nuclear Material

The licensed activity shall be authorized to sell, or otherwise transfer, hydrofluoric acid containing trace quanitities of uranium (enriched up to five weight percent in U-235) to nonlicensed persons, provided that:

4.8.1 The concentration of uranium in the acid does not exceed 10 parts per million.

4.8.2 Each such sale or transfer shall be accompanied by a written instruction that the acid is not to be used for any purpose involving human consumption.

SNM 1107

UPDATE FOR ENVIRONMENTAL IMPACT APPRAISAL

WESTINGHOUSE ELECTRIC CORPORATION, NFD PLANT

COLUMBIA, SOUTH CAROLINA

DOCKET NO. 70-1151

APRIL 1983

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SECTION 1

INTRODUCTION

1. INTRODUCTION

In conjunction with the renewal application for SNM-1107, the following update is submitted to supplement the original "Environmental Evaluation Report", March 1975 and the "Environmental Impact Appraisal Report, NR-FM-013", April 1977. The aforementioned reports have concluded and documented in detail that the environmental impact of operating the Westinghouse Nuclear Fuel Division, Columbia Plant is either minimal or insignificant. Recommendations were made to file a negative declaration as provided in 10 CFR 51.7 relative to the NEPA Act of 1969 (P.L.91-190) and the President's Council on Environmental Quality Guidelines. In the interim no events or significant changes in the mode of operation have occurred that would reverse those conclusions.

1.1 DESCRIPTION OF THE PROPOSED ACTION

This report has been prepared to provide an environmental impact appraisal for continuing operations at the 700 MTU level and eventually expanding to the 1600 MTU/year level. The facility began operation in September 1969 and was constructed to fabricate low-enriched uranium fuel (5% U-235) assemblies for use in light water commercial power reactors. Data in this report are consistent with time and production capacity referenced. Where necessary, recent update information has been provided to illustrate continued performance and operation according to ALARA concepts and prudent judgement. No operations have been conducted or are planned which utilize mixed oxide plutonium fuel.

1.2 SUMMARY OF CURRENT STATUS

The original plant was constructed in 1969 and designed with a production capacity equivalent to 400 metric tons of uranium per year. Subsequent estimated load projections were made that determined the plant should eventually be expanded to 1600 MTU. In 1979 the facility was expanded by 100,000 square feet to allow space for the solvent extraction process, a chemical development laboratory, a new incinerator, and additional office space. No additional environmental impacts are expected as a result of these modifications. Approximately 5% of the total site area has been developed. Typical gaseous discharges average less than 25 uCi/wk (700 MTU/yr). Discharge at this rate conforms to the EPA regulation 40 CFR 190 limiting the dose at the site boundary to 25 millirem/year. The radioactivity in the liquid effluent discharged typically averages less than 5% of the Maximum Permissible Concentration at the discharge point. All NPDES/EPA chemical liquid discharge parameter limits are routinely complied with. A long term NPDES/EPA discharge permit has been granted by the SC-DHEC expiring in 1986. Procedures have been implemented to meet the requirements of the Resource Conservation and Recovery Act and Hazardous Waste Regulations according to rules implemented by the SCDHEC.

SECTION 2

DESCRIPTION OF ACTIVITY

2.1 INTRODUCTION

The Westinghouse Electric Corporation, Nuclear Fuel Division, Columbia Site, is located on South Carolina Highway #48 (Bluff Road) in Richland County approximately 8 miles southeast of Columbia, South Carolina. The Westinghouse plant fabricates nuclear fuel assemblies containing low-enriched ($\leq$5% U-235) uranium oxide fuel for use in commercial light-water nuclear-powered reactors.

The fabrication process involves the chemical conversion of uranium hexafluoride (UF_6) to uranium dioxide (UO_2) using the ammonium diuranate process. This is followed by milling, pressing, sintering, and machining of the uranium to form fuel pellets approximately one-quarter inch in diameter and one-half inch in length. These pellets are loaded and encapsulated into fuel rods approximately 12 feet long, and the rods are stacked into the fuel assembly hardware fixture frame for eventual use in nuclear reactors. Various recovery operations including thermal oxidation, dissolution of scrap powders with nitric acid, chemical precipitation, wet mechanical separation, washing operations, and solvent extraction are carried out to support the conversion process in the recycle of material. Incineration is conducted to decrease the volume of low-level waste and to economically recover uranium contained in combustible wastes.

In the Mechanical Manufacturing Area of the plant, additional machining, welding, electroplating, quality control testing, and other miscellaneous operations involved in the production of the assembly's hardware are performed. Sanitary and industrial waste treatment processes are conducted external to the main facility.

Radioactive material authorization is granted under USNRC License No. SNM-1107, Docket No. 1151, as listed in Figure 2.0.

The ammonium diuranate process generates both liquid and solid wastes. Prior to chemical waste treatment, the aqueous waste stream contains 1-3% ammonium fluoride and 3-10% free ammonia. All final liquid effluent discharged must meet EPA NPDES permit requirements. The ammonia is recovered using steam distillation and recycled. Fluoride is removed by precipitation with calcium hydroxide and stored in holding lagoons. If the calcium fluoride is contaminated greater than 30 pCi/g with uranium, the material will be fixed with a cement-like binder and buried at a licensed SNM low-level waste burial site. Combustible wastes are incinerated and the uranium in the ash is recovered and recycled. Contaminated noncombustible waste is packaged properly and buried in a licensed low-level waste burial site.

RADIOACTIVE MATERIALS
QUANTITY CERTIFICATION

This is to certify that the Westinghouse Electric Corporation is authorized under USNRC License No. SNM-1107 to possess the following radioactive material at its site at Columbia, South Carolina. The current expiration date of the above license is May 31, 1983.

MATERIAL	FORM	ENRICHMENT	QUANTITY
Uranium-233	Any	---	5 Grams
Uranium	Any	Any	350 Grams ^{235}U
Uranium (low-enriched)	Any	$\leq$ 4.15 w/o	50,000 Kilograms ^{235}U
Uranium (low-enriched)	Oxides (dry)	$>$4.15$<$5.0 w/o	2,500 Kilograms U
Plutonium-238	Sealed Source	---	1.5 Grams ^{238}Pu

Figure 2.0 SNM-1107, Radioactive Materials Certification

2.2 SITE AND FACILITY DESCRIPTION

The Westinghouse Nuclear Fuel Plant is located in the central part of South Carolina in Richland County, approximately 8 miles southeast of Columbia on South Carolina Highway #48. The site coordinates are latitude 33° 50' 60" and longitude 80° 56' 45". The plant is set back approximately 1800 feet from the roadway on a semi-rural plot of approximately 1156 acres. Nearby towns, industries, public facilities, the Congaree River, and transportation links are shown in Figure 2.1.

The site is bounded by Route 48 to the north; the Vestal Lumber Manufacturing property to the east; the Liberty Life Insurance property to the south, and Burrell Manning property to the west. The fuel fabrication facility, waste treatment area, parking lot, and other miscellaneous building occupy approximately 5 percent of the site area. The plant floor is located at 142 feet above mean sea level. Plant site drainage flow follows original drainage patterns to Sunset Lake, Mill Creek, and Congaree River. Figure 2.2 shows the site plan with relationship to external boundaries.

The plant work force consists of approximately 850 personnel working over three shifts. A typical percentile distribution by shift would place 50% on first, 25% on second; and 25% on third. Figure 2.3 shows the estimated resident population within a five mile radius of the plant. The maximum population densities occur in the NW and ENE sectors. The projected population for 1980 for the 5 mile area is 5655. Transient population in schools and industrial workers would increase the population approximately 40% within the 5 mile area.

Figure 2.4 depicts the "Detailed Columbia Site Plan". The major site facilities include the manufacturing plant building, UF_6 storage area, waste treatment area, shipping container refurbishing building and storage area, waste storage building, waterglass advanced wastewater treatment building, tank farm raw material storage area. The main plant building utilizes approximately 345,000 square feet of floor space for the administration and manufacture of nuclear fuel assemblies.

The main plant building, containing the chemical and mechanical manufacturing process, consists of a large high bay manufacturing building with an expansion addition constructed of structual steel framework and concrete curtain walls.

Elevated roof expansions have been made in the solvent extraction area and the proposed IDR production area. Adjoining the east and north sides of the manufacturing building are two office areas. The auxillary support structures are typically prefabricated steel buildings.

The floor plan and operational layout within the manufacturing plant are displayed in Figure 2.5. The original building was designed by Giffels and Rossetti and constructed by McCrory-Sumwalt Construction to meet compliance with the Southern Building Code, 1965 Edition with minor exceptions.

2.3 PROCESS DESCRIPTION

Only "fresh" unirradiated low enriched uranium is processed at the Columbia Plant. Therefore, typical external dose levels associated with the material are small, and operating personnel can work in close proximity with the uranium without undue concern over external dose rates.

Various Chemical and Mechanical Operations are performed in the fabrication of nuclear fuel assemblies. Plant equipment is designed to keep chemical and radioactivity discharges both in the plant and environment as low as is reasonably achievable (ALARA).

2.3.1 AMMONIUM DIURANATE PROCESS

Uranium oxide product material is prepared using the ammonium diuranate process. This process is well established and is accepted as representing the current state of applied technology. The process consists of hydrolyzing UF_6 in water to form uranyl fluoride. The uranyl fluoride reacts with ammonium hydroxide in a precipitation column to form ammonium diuranate (ADU). The ADU is dried and then heated in a rotary tube calciner in the presence of hydrogen and steam to form uranium oxide (UO_2).

Five ADU chemical conversion lines are available to process the UF_6 into UO_2. The multiple line concept allows the processing of different U_{235} enrichments concurrently to meet fuel delivery requirements.

2.3.2 UF_6 VAPORIZATION AND HYDROLYSIS

Uranium hexafluoride (UF_6) is received from DOE gaseous diffusion plants in 2 1/2 ton cylinders as a solid. UF_6 is vaporized from a solid to gas by placing the cylinder into a steam chest and heating to approximately 250 degrees F. The gas is transported under pressure to a water filled hydrolysis column, forming an aqueous solution of uranyl fluoride.

$$\underset{\text{Uranium Hexafluoride}}{UF_6} + \underset{\text{Water}}{2\,H_2O} \longrightarrow \underset{\text{Uranyl Fluoride}}{UO_2F_2} + \underset{\text{Hydrofluoric Acid}}{4HF}$$

2.3.3 pH ADJUSTMENT AND PRECIPITATION

Ammonium hydroxide is added to the hydrolyzed product to partially neutralize the hydrofluoric acid. Ammonium hydroxide then is added to the uranyl fluoride solution in large quantities to precipitate the uranium as ADU. The ADU slurry that is formed in the ammonium hydroxide is routed to a centrifuge for dewatering.

$$\underset{\text{Uranyl Fluoride}}{2UO_2F_2} + \underset{\text{Ammonium Hydroxide}}{6NH_4OH} \longrightarrow \underset{\text{Ammonium Diuranate}}{(NH_4)_2\,U_2O_7} + \underset{\text{Ammonium Fluoride}}{4NH_4F + H_2O}$$

2.3.4 CALCINATION

The centrifuged material is further dried and/or directly fed to a heated rotary tube calciner where the ADU is converted by heat in a steam and hydrogen atmosphere to uranium dioxide (UO_2). The calciner is fired with natural gas. The effluent reaction gases are scrubbed to remove both particulate and chemical contaminants and passed through HEPA filters.

$$\underset{\text{ADU}}{(NH_4)_2U_2O_7} + \underset{\text{Hydrogen}}{2H_2} \longrightarrow \underset{\text{Uranium Dioxide}}{2UO_2} + \underset{\text{Ammonia}}{NH_3} + \underset{\text{Water}}{H_2O}$$

Liquid waste is decanted and filtered to remove excess uranium contamination. An online gamma ray spectroscopy monitoring system is used to continuously analyze the liquid effluent and insure the uranium concentration is less than 3×10^{-5} uCi/ml, a maximum permissible concentration (MPC) limit allowed by the Nuclear Regulatory Commission. Quarantine tanks are available to increase settling times and allow filtration if the activity level is above release limits. A unit operation advanced wastewater treatment process has been installed external to the main plant to insure the uranium level in the liquid effluent stream is limited to extremely low values.

The uranium dioxide from the calcination step is milled to smaller uniform particles and blended to insure uniform product characteristics. A bulk handling process has been installed to improve product handling procedure. After completing quality control testing this material is transferred to the Pellet and Rod Area.

2.3.5 DIRECT CONVERSION FLUIDIZED BED PROCESS (DCFB)

This process has been discontinued in favor of the IDR process as described in 2.3.6. The equipment has been dismantled and removed.

2.3.6 INTEGRATED DRY ROUTE (IDR) CONVERSION PROCESS

A major expansion of the facility has been undertaken to allow for installation of equipment to convert UF_6 to UO_2 utilizing a dry conversion process.

This process, licensed from British Nuclear Fuel Limited is well established, and it will significantly reduce the volume of liquid waste generation. Vaporized UF_6 is directly reacted with superheated stream in a conversion kiln forming uranyl fluoride and hydrogen fluoride gas. The uranyl fluoride stream further reacts in the kiln with a countercurrent stream of hydrogen, nitrogen, and superheated steam forming a uranium oxide (UO_2) powder.

Process gases are continuously vented and filtered through the top end of the kiln. UO_2 product is discharged from the product end of the kiln. Hydrofluoric acid is condensed and recovered as a by-product. Off gas streams are thoroughly scrubbed with caustic to insure complete removal of hydrofluoric acid products. Finally all effluent gases discharged will be vented through HEPA-filtered exhaust systems. Converted powder will be handled using processes similiar to ADU processed material. Extensive use of automated equipment is planned.

2.3.7 SCRAP RECOVERY

Scrap recovery process operations are characterized as batch operations involving a variety of input forms. The preliminary operations concentrate the material and convert it to forms readily processed as U_3O_8 powder and uranyl nitrate. Not all materials require processing through the entire sequence of operatons. The basic processing sequence includes dissolution of solid forms, conversion to slurry form by precipitating ADU from the solution, dewatering the slurry form by wet mechanical separation, calcining the resulting sludge in regular or controlled atmosphere furnaces, and packaging and storing the resulting product.

Before being released through the HEPA-filtered exhaust system to the atmosphere, off-gases from the uranyl nitrate dissolvers are routed through a reflux condenser and a scrubber to remove entrained particles and condensible vapors. The reflux condenser is mounted vertically and directly above the dissolution tank so that any condensation formed can drain back into the tank. An incineration process is conducted to minimize burial of low-level combustible contaminated waste and to economically recycle product grade material. A solvent extraction process is operated to recover and purify various contaminated uranium materials.

2.3.8 PELLET AND ROD AREA

The product UO_2 powder from the Conversion Area is brought to a feed preparation hood in the Pellet Area where it is mixed with U_3O_8 and UO_2 add back material. The material is transferred via a bucket elevator system to a roll compactor and pre-compacted. The material is then granulated and mixed with zinc stearate binder-lubricant. The granules of uranium are next fed into high speed pellet presses where the fuel is compacted into a green pellet. The green pellets are then loaded into molybdenum boats and sintered in an electrically heated furnace at about 1750 degrees C in a hydrogen atmosphere. This process produces a denser more compact pellet. To obtain precise dimensions all pellets are processed through a grinding operation and dimensionally checked.

Pending quality control release, the pellets are then loaded onto trays for interim storage prior to being loaded into the fuel rods. Then the pellets are loaded into the empty fuel rods using a vibratory feed mechanism. A spring is inserted into the plenum section, and end plugs are inserted and girth welded to the rod. Next the rod is pressurized with helium and seal welded. Finished fuel rods are transferred to Quality Assurance Operations.

2.3.9 MECHANICAL MANUFACTURING AREA

The production area of the plant is divided into two general Controlled Areas: the Chemical Manufacturing Area (an airborne radioactivity controlled area) and the Mechanical Manufacturing Area. All uranium material that is transferred to the mechanical manufacturing area has been encapsulated and sealed. A small additional room is proposed for the south end of the facility to manufacture poison rods for nuclear fuel assemblies.

Various quality control and quality assurance operations are performed in the manufacturing area on sealed rods including: X-ray testing, helium leak testing, gamma scanning, visual checks, and dimensional checks. Machining operations are performed to fabricate various internal parts of the nuclear fuel assembly "skeleton" structure including: grid straps, bottom nozzle, top nozzle, and guide tubes. Individual rods are loaded into the skeleton assembly. Final weighing and testing operations are performed on completed assemblies. Other machining operations are performed in fabrication of the boron carbide burnable poison assemblies and silver cadmium control rod "spyder" assemblies. A nickel plating shop is maintained to assist with the brazing of the inconel grid straps. Zirconium grid strap fabrication using laser welding techniques has been introduced for certain fuel assemblies.

As a final step, the assembly is given a complete wash in soap and water, and a deionized water rinse. The assemblies can either be stored for shipment or shipped immediately in approved RCC containers. A substantial quantity of assemblies are stored prior to shipment to the utility in the fuel assembly storage area.

2.3.10 SHIPPING

All shipments of nuclear materials and wastes from the Columbia plant are carried out in conformance with NRC, DOT, and the State of South Carolina requirements. Completed fuel assemblies are shipped to utility customers in approved containers license by the Nuclear Regulatory Commission. Low-level waste shipments are appropriately packaged and analyzed for uranium content prior to shipment to the low-level waste burial grounds.

Figure 2.1 Westinghouse NFD Site Area Location

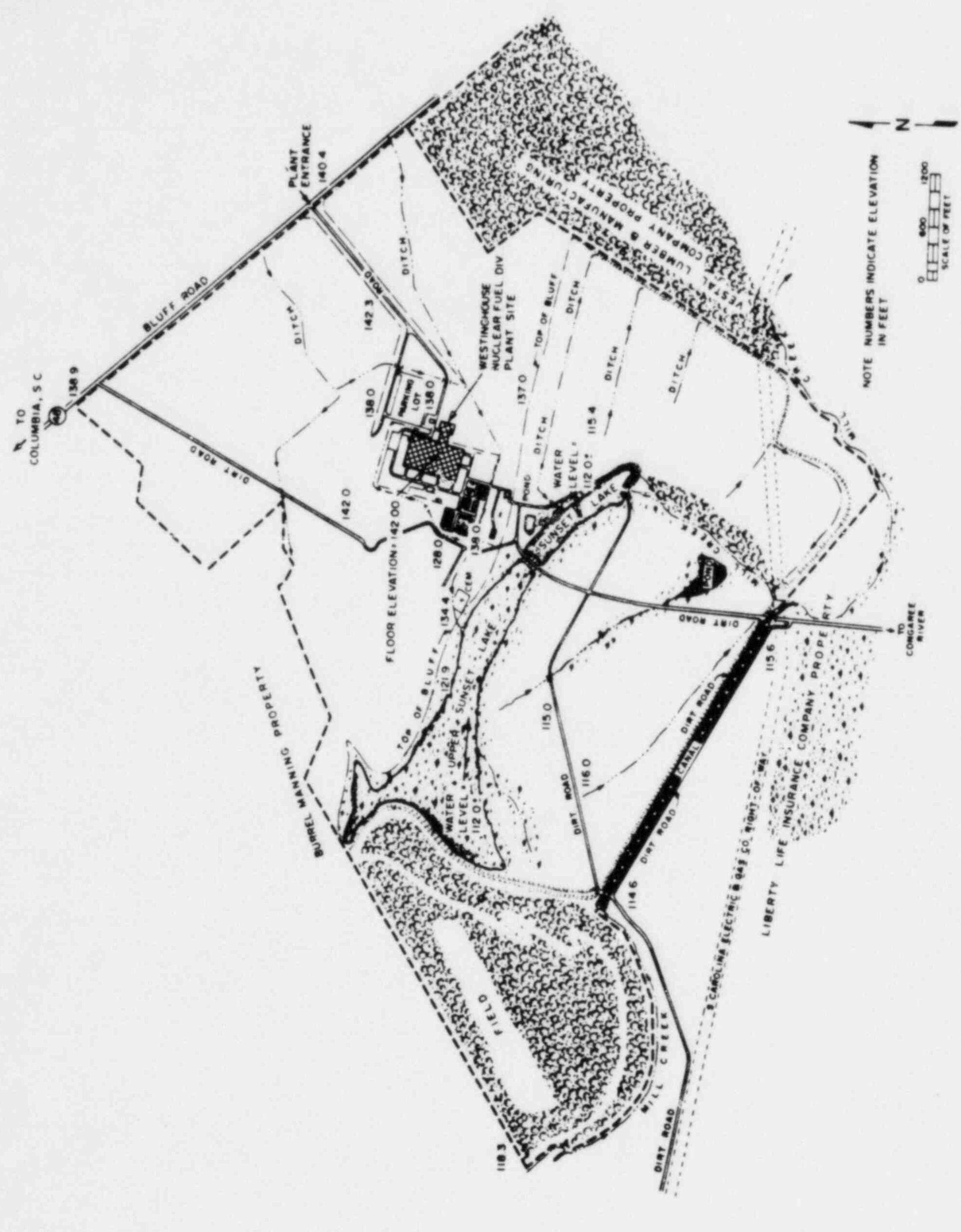


Figure 2.2. Westinghouse NFD Site Property Map.

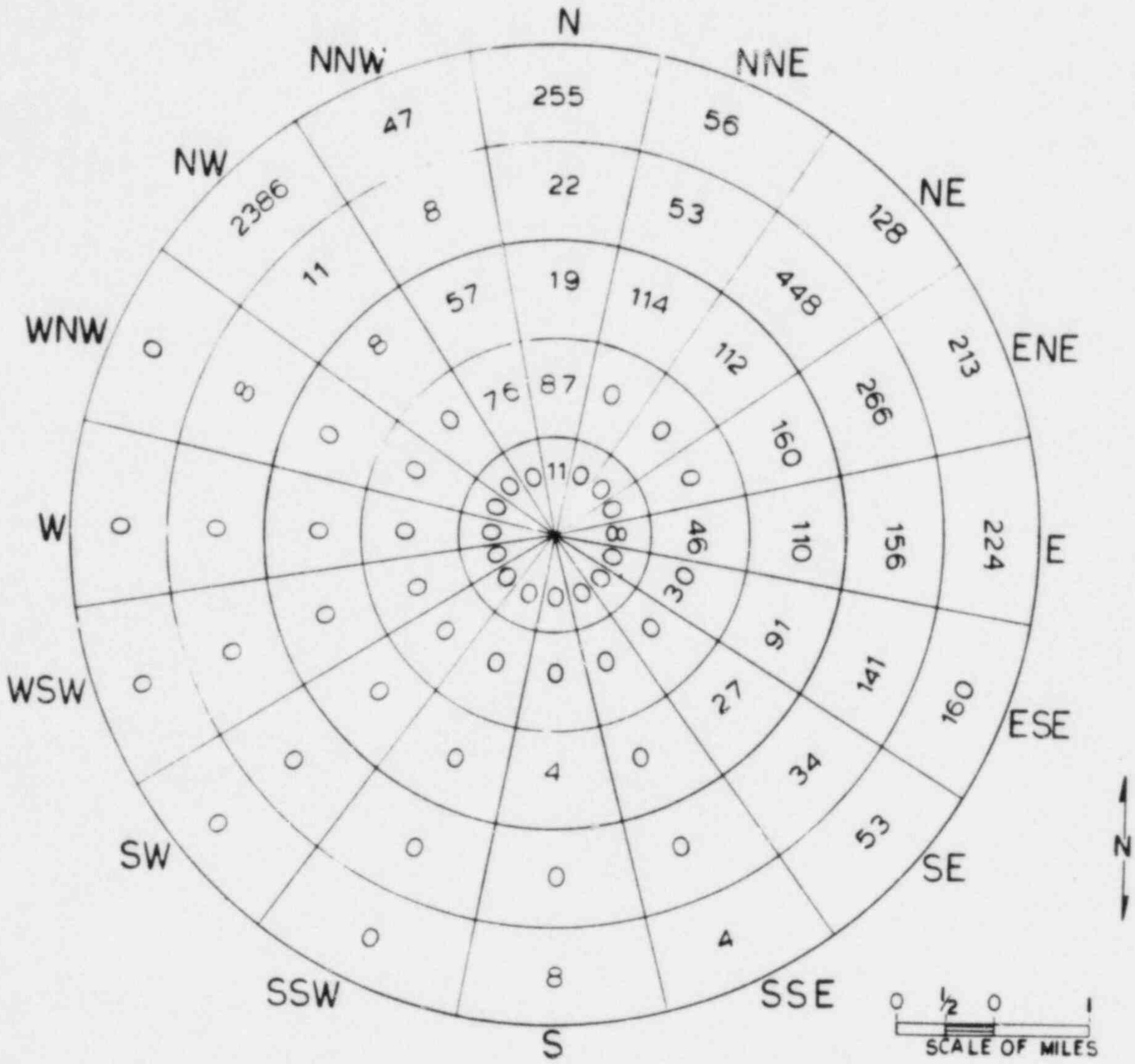


Figure 2.3 Estimated Population Distribution Within 5 Mile Radius of the Columbia Plant

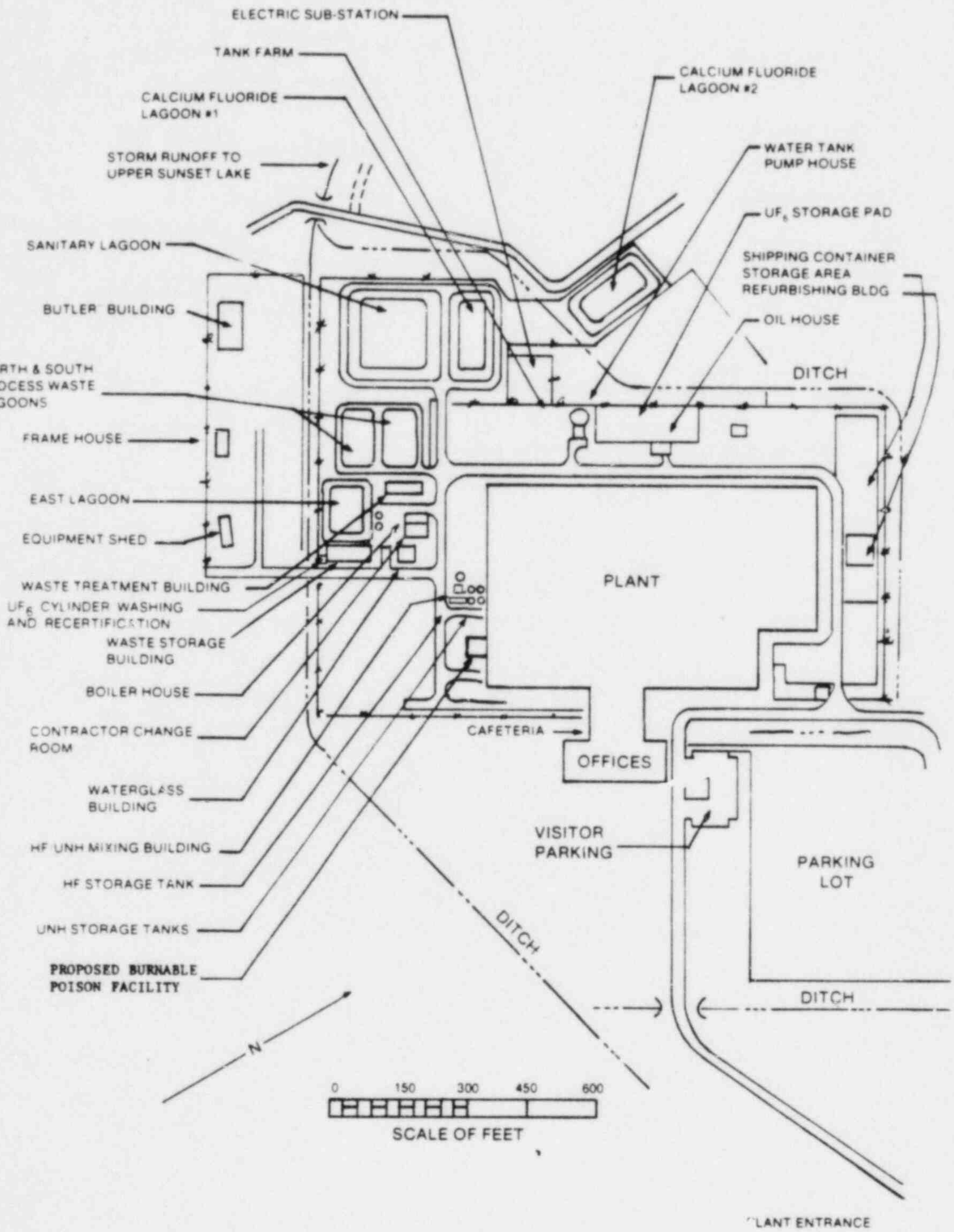


Figure 2.4 Detailed Columbia Site Plan

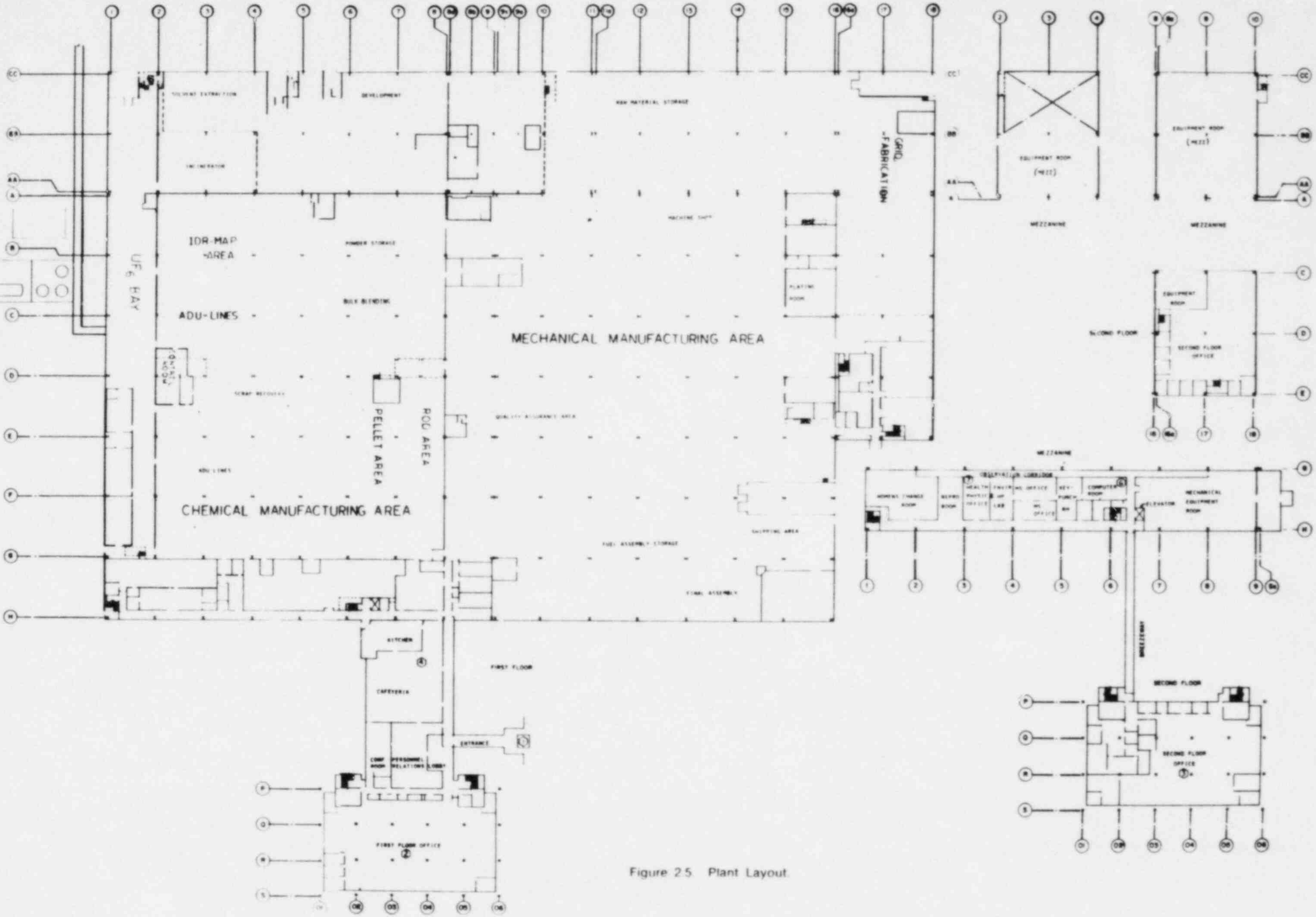


Figure 2.5 Plant Layout

SECTION 3

DESCRIPTION OF SITE ENVIRONMENT

3.1 INTRODUCTION

This section presents basic information concerning the physical, biological, and human characteristics of the regional environment that might be affected by the operation of the Westinghouse Nuclear Fuel Division manufacturing plant site (NFCS) near Columbia, South Carolina.

3.2 DEMOGRAPHY

3.2.1 RESIDENT POPULATION WITHIN 5 MILES

The local population can be affected by plant operations an/or expansion through its exposure to the atmospheric effluents from the plant and to the negative effects of possible accidents at the plant.

A detailed analysis of the resident population distribution within a 5-mile radius of the plant was performed utilizing Richland County tax maps and 1980 census data.

The total population withing a 5-mile radius of the plant is 5655. The maximum population densities occur in areas northwest and east northeast the plant site.

The population density in each sector is 6.1 per square mile, 25.4 per square mile, 45.3 per square mile, 52.4 per square mile, and 125 per square mile within the 0- to 1-mile, 1- to 2-mile, 2- to 3-mile, 3- to 4-mile, and 4- to 5-mile sectors, respectively, from the plant site.

The transient population is similarly sparse and favorably distributed in a 180^{o} compass sector. All schools are over 3 miles away, and only one of 13 businesses with more than 5 employees is less than 3 miles from NFCS. The total student body and school staff from the three public schools in the study area increases the overall population by 16% (900 school personnel in addition to 5655 area residents) each day school is in session. Two out of three school persons are permanent residents of the study area.

There are no hospitals in the study area. Industrial and business facilities with more than five employees add 1369 (24% of resident population) persons to the total population of the study area during work days, assuming that all employes reside outside the area.

Attendance at the two parks in the area (Bluff Road County Park, 3.8 miles northwest of the NFCS; Hopkins County Park, 1.75 miles east of the NFCS) is estimated to be 300 persons per day or 5% the area resident. In the unlikely event that none of the park attendees were local residents, the total daily transient population would add about 45% to the population of the area. Overall, the low resident and transient population density and its favorable distribution away from the Columbia Plant are positive aspects of the site situation.

3.2.2 INDUSTRIES AND BUSINESSES

The recreational parks and sports facilities within the study area are listed in Table 3.3. There are no state parks or commercial sports stadiums within a 5-mile radius of the plant.

Estimates prepared from Central Midlands Regional Planning Council data using the 1981 Richland County Tax Maps and 1980 Census data and interpretations from U. S. Geological Survey Maps.

TABLE 3.1

1980 POPULATION ESTIMATES
BY SECTORS WITHIN A 5 MILE RADIUS
OF THE WESTINGHOUSE NUCLEAR FUEL PLANT, COLUMBIA, S.C.

SECTORS	1 MILE	2 MILES	3 MILES	4 MILES	5 MILES	TOTAL
N	11	87	19	22	255	394
NNE	0	0	114	53	56	223
NE	0	0	122	448	128	698
ENE	0	0	160	266	213	639
E	8	46	110	156	224	544
ESE	0	30	91	141	160	422
SE	0	0	27	34	53	114
SSE	0	0	0	0	4	4
S	0	0	4	0	8	12
SSW	0	0	0	0	0	0
SW	0	0	0	0	0	0
WSW	0	0	0	0	0	0
W	0	0	0	0	0	0
WNW	0	0	0	8	4	12
NW	0	0	8	11	2,386	2,405
NNW	0	76	57	8	47	188
TOTALS	19	239	712	1,147	3,538	5,655

Estimates prepared from Central Midlands Regional Planning Council data using the 1981 Richland County Tax Maps and 1980 Census data and interpretations from U.S. Geological Survey Maps.

TABLE 3.2

INDUSTRIES AND BUSINESSES WITHIN 5 MILES OF THE

WESTINGHOUSE NUCLEAR FUEL COLUMBIA SITE

Name	Type of Business	Employees	Distance and Direction From Plant
Columbia Container Co.	Garbage containers	20	4-3/4 miles NW
Dan Dee Specialties	Picture frames	30	4-1/5 miles NW
Southco	Supplies for gift shops	10	4-1/5 miles NW
Seefab	Metal Fabrication	51	4-1/2 miles NW
Metal Equip. Fabricators	Metal Fabrication	30	4-1/2 miles NW
Piedmont Products	Fiberglass Products	150	4-7/8 miles NW
Roadway Express	General freight handling	24	4-1/5 miles NW
Columbia Eggs Division of Farmers Cooperative Exchange	Egg grading and packaging	35	3-miles ENE
Square D Company	Manufacture heavy industrial motor controls	930	4-2/3 miles NNE
Wallace Concrete Products, Manhole Div.	Manholes	12	4-4/5 miles NNE
McGregor's Dairy	Milk	7	4-4/5 miles NNE

TABLE 3.3

RECREATIONAL PARKS AND SPORTS FACILITIES

WITHIN A 5-MILE RADIUS OF THE PLANT

NAME	DISTANCE AND DIRECTION DIRECTION FROM PLANT	1981-82 ESTIMATED ATTENDANCE
Bluff Road County Park	3-3/4 miles NW	58,000
Hopkins County Park	1-3/4 miles E	39,561

3.3 LAND USE

This portion of the Appraisal describes characteristics of local (within a 5-mile radius of the NFCS) land use that are important in the environmental assessment of the NFCS operation and/or expansion. The distribution and nature of agriculture, the important historic and prehistoric landmarks, and the distribution of undeveloped nonagricultural land are evaluated in this section..

3.3.1 LAND-USE DISTRIBUTION

The nature, extent, and distribution of local land uses are important in the environmental assessment of NFCS operations and/or expansion. The primary interaction to evaluate is that of NFCS radiological and chemical atmospheric effluents with local human and biological populations and with farming and manufacturing activities. Interactions involving water supplies to these populations and activities are of equal importance but are treated elsewhere in this assessment.

Approximately 5% of the land within 5 miles of the NFCS is residential; less than 1% (exclusive of the NFCS) is industrial and about 20% is agricultural. Seventy percent of the land is uninhabited forest or swamp forest.

3.3.1.1 MANUFACTURING

All firms with five or more employees are within the 180° sector (west-east-northeast). Those plants with potentially significant atmosphere or aquatic effluent loads with which the NFCS effluents could interact include the Carolina Eastman plant (man-made production fibers), Wallace Concrete Products (man-hole production), and Square D Company (industrial motor control production).

However, information obtained from the Santee-Cooper River Basin water quality management plan lists only Carolina Eastman as a water user, and there, only cooling water (3.2×10^6 gal/day) is indicated. Other point sources of effluent water are described for the nearby drainage systems, but only volume and BOD_5 estimates are given for the facilities producing effluents.

3.3.1.2 AGRICULTURE

Agricultural land occupies about 20% of the land area, primarily in the northern and eastern portions of the study area. Crops include soybeans, corn, hay, cotton, wheat, and oats. Pecan orchards are present to the east. In general, the cultivation is neither extensive nor does it appear to be economically important.

McGregor's Dairy, 4.8 miles north-northeast of the NFCS, has 150 milk cows. No other important crop or livestock production appears to occur in the study area. Light agricultural production and its possible decreasing frequency with greater proximity to the NFCS plant are positive aspects of the Columbia site.

3.3.1.3 UNDEVELOPED NONAGRICULTURAL LAND AND LANDMARKS

Seventy percent of the land in the study area is covered by forest or swamp forest. Forest and swamp land is more frequent to the north and less frequent toward the south. The distribution of the three forest communities seems to be controlled by drainage. Water tupelo sweet gum dominated forests occur in swamps and on wet alluvial substrates along the Congaree River. A more mesic oak forest dominates the better-drained sites, whereas the driest sites in the area may be dominated by loblolly pine and hardwoods (oak species, red maple, yellow poplar, etc.). The distribution of vegetation and the biotic relationships existing within them are discussed in Section 3.8.1.

Presently there are no important logging activities in the forests on the Columbia Plant property; however, the swamp-forest communities southeast of the NCFS may be significant in another context. This southeast area of some 21,000 acres has been named as a national park. Its forests have been largely undisturbed for 200 years. Not only is this area a rare remnant of previously extensive southern-river floodplain forests, but it also contains several of the largest trees of their species. There are numerous known and probably many more undiscovered archaeological sites in the Congaree Swamp and along S. C. Highway 48.

Other historical, cultural, and natural landmarks occur in the study area. According to correspondence from the South Carolina Department of Archives and History, the following historical sites, listed in the Central Midlands Survey of 1974, are potentially eligible for National Register, but none are presently of high priority for nomination. All sites are within a 5-mile radius of the plant.

1. <u>Raiford's Mill Creek (Mill Creek)</u> - 18th century

   The first settlements in the county were made along Mill Creek in the 1740's. Hopewell Ferry, across the Congaree River below the creek's mouth, was used in 1756 and thoughout the Revolution. The creek was named for Philip Raiford, who settled on the creek below Adams' Mill Pond. The creek was later called Hays' Creek for William Hays, who built a mill there in 1748-1750. Known by 1800 simply as Mill Creek.

2. <u>Cabin Branch (John Hopkins, Jr., Plantation House)</u> - 1796

   Off County Road 1159, one-fourth mile south of intersection with Road 223, near Congaree Community. An 18th century house with two large front rooms, a center hall, and an open loggia. About 1835, two large rooms were added at the rear, and the loggia was extended into a hall. Still owned by the Hopkins.

3. <u>Claytor House - 1887</u>

   On Highway 37 at Hopkins, wooden cottage built by Dr. Hubert Claytor, with porch and fish-scale gable; architecturally distinctive.

4. Chappel Cabin Branch (Hicks Plantation House and Garden) - 1781

On a dirt road off County Road 37, one-half mile south of Hopkins; a two-story rectangular frame house with a single-story front porch; recent alterations; a garden with original plantings remains; still occupied by Chappel family.

5. Hopkins Overseers Dwelling - 19th century

In the Hopkins Community on County Road 37, one-fourth mile south of intersection of County Roads 37 and 55; center section is pedimented frame cottage; the Hopkins family cemetery is nearby.

3.3.2 SUMMARY - LOCAL POPULATION AND LAND-USE CHARACTERISTICS

The present facility is well located with respect to land use. Resident and transient population density is low, particularly near the site. Gathering places (schools, recreational areas, business and manufacturing sites) are sparse, and agricultural activities are quite limited. Some commercial logging activities are conducted south of the facility.

3.4 GEOLOGY

The features of underlying rocks pertinent to the NFCS environmental assessment include stability with respect to possible land subsidence, structure and seismicity with respect to potential for earthquake damage, and stratigraphy with respect to surface drainage and subsurface movement of water through aquifers. The surface expression of geological parameters (topography, soil thickness, etc.) controls other parameters such as vegetation composition, land use, and microclimate. Each of these characteristics will be disussed below.

3.4.1 PHYSIOGRAPHY - LOCAL AND REGIONAL SITING

The following discussion of physiography provides the conceptual framework for understanding specific subdisciplinary topics.

Like Washington, D.C., Raleigh, N. C., and Augusta, Ga., Columbia, S. C., is located along the zone of small waterfalls and river rapids known as the fall line. Here the Paleozoic (270 to 600 million years old) metamorphic rocks of the Piedmont physiographic province on the west dip beneath Cretaceous (135 million years old) sedimentary rocks of the Coastal Plain physiographic province on the east. The vertical gradient that produces the fall zone is due to the difference in erosion resistance of the adjacent metamorphic (hard) and sedimentary (soft) surface rocks.

Northwest of Columbia, the Piedmont materials include those composing the Carolina slate belt, a zone of rocks that are somewhat less erosion-resistant than the neighboring Piedmont metamorphics. Consequently, they form wider valleys and are favored for reservoirs on the Saluda River (Lake Murray, Lake Greenwood) above Columbia. A granitic pluton originating in the Piedmont slate belt occurs within Coastal Plain sediments and crops out of the north edge of Columbia.

It is radiometrically dated as mississipian (350 million years old) and includes metargillites, tuffs, and volcanic breccias. The southeastern contact of the pluton is covered by Coastal Plain sediments, so its location is unknown.

The Coastal Plain on which the NFCS is located extends from Cape Cod, Massachusetts, south to Mexico, and from the fall line eastward beneath the Atlantic Ocean to the edge of the continental shelf. It is characterized by gently rolling hills and generally well-drained, mature valleys. River systems are presently incising their beds and migrating headward in the vicinity of Columbia. "It is thought that this rejuvenation is more likely related to the impact of man and the clearing of the land in downstream directions, than to changes in base level initiated by tectonic activity." Drainage headward of the incisement is dendritic and mature, whereas below this point the major creeks draining into the Congaree have developed in an adverted fashion as a response to former topography.

These soils are developed on the Okefenokee Formation, which is the principal deep aquifer in this portion of Richland County. Groundwater movements within the aquifier should parallel the impermeable surface of the Tuscaloosa alluvial beds about 30 ft below the ground surface at the NFCS and then follow the expected southwest dip toward either Mill Creek or onto the Congaree River floodplain. Migration updip toward the northeast or north is unlikely.

The Coastal Plain sediments pinch out at the fall-line area some 10 miles northwest of the NFCS. Beneath the NFCS, the sediments are some 165 ft thick (25 ft below MSL), whereas at the Atlantic coast, they are between 1100 ft (North Carolina coast) and 3700 ft (Georgia coast). Beneath these unconsolidated or semiconsolidated sediments, the Palezoic metamorphic rocks of the Piedmont are found.

The bottommost Coastal Plain sediments of late Cretaceous age (approximately 100 million years old) are part of the Tuscaloosa Formation. The lower units are alluvial (deltaic) in nature and consist of quartz gravel, with lenses of sandy clay (Kaolin), coarse sands, and pure clays that are nonporous and of low permeability. At the point of outcrop, these lower units are marked by standing water or small springs that reflect perched water tables. The upper Tuscaloosa marine phase occurs only to the east of the NFCS, being eroded locally. Where the phase does occur, its fine sands lack clay lenses, thus it is highly permeable and quite porous.

The lower Tuscaloosa sediments are some 35 ft thick below the NFCS and dip toward the southeast at 20 to 30 ft/mile. Thus the aquifers they contain similarly dip away from the NFCS and toward the Congaree River.

Unconsolidated sediments of more recent alluvial origin cover the nonporous Tuscaloosa materials at the NFCS. The sediments consist of medium-to-coarse clayey sand and fine gravel beds that form the Okefenokee terrace upon which the NFCS rests) at about 140 ft above MLA. The Wicomico terrace below the NFCS on the southeast contains the Congaree River floodplain at about 120 ft above MSL, whereas above the

NFCS on the northwest, the Sunderland terrace occurs at about 150 ft above MSL. These terraces are overlain by sandy soils.

3.4.2 SEISMOLOGY

Most of the seismology information in this report is based on the Colquhoun report, which was in turn based on recent work by Bollinger. Bollinger is responsible for much of the current literature on seismology of the southeast generally, and of South Carolina in particular. He suggests that "broadly viewed, the region is a minor seismic zone, characterized by a low level of seismic energy release." He also suggests that "earthquate, frequency per unit time per unit area in this region is about one-tenth that of the west coast, but the seismograph station density that exists even today is inadequate." However, he notes from other research that areas affected by shocks east of the Rocky Mountains may be felt over a wider area than those from equal magnitude events in the western United States.

The earthquake activity in the NFCS area is probably associated with tensional faulting and apparently does not affect the Paleozoic and Cretaceous rocks below the NFCS. Instead, the Oligocene, early Miocene, and perhaps middle Miocene sediments (40, 25, and 15 million years old respectively) that are so affected occur some 30 miles southeast of the site.

The most intense shock in South Carolina cited by Bollinger was the August 1886 event at Charleston. The quake was felt as far west as Missouri and as far north as Vermont. The quake intensity in the Charleston area corresponded to a Modified Mercalli Index (MMI) scale of X, while most of the remainder of South Carolina, including the NFCS area underwent a quake intensity of MMI-VII. The expected damage at an MMI value of VII includes cracked masonry, broken chimneys, falling plaster, loose bricks and cornices, damage to concrete irrigation ditches, and caving in along sand and gravel banks.

Table 1 of the Bollinger Report indicated that at least 14 quakes of MMI-VII or greater intensity have occurred in the southeast since 1754.

The distribution of quakes from 1754 to 1974 is shown in Fig. 3.1. Recent seismic activity within 50 miles of the NFCS include two MMI-III and IV intensity shocks in 1971; an MMI-V shock near Bowman (43 miles south of the NFCS) in 1972; a quake (magnitude not noted) March 28, 1973, in the Pelion-Gaston area (18 miles west of the NFCS); and a MMI-V shock August 2, 1974, near Edgefield County (50 miles west of the NFCS).

Bollinger suggests that "up to 1950, the seismic activity within the state is seen to be concentrated in the Charleston-Summerville area, but subsequent to that time has been primarily outside that locale...unexplained is the apparent shift, during the past two decades, of seismic activity away from the coastal Charleston-Summerville area to the interior portions of the state. This apparent shift now includes three shocks in the central part that has been historically free of earthquake epicenters." Apparently, this suggested trend of Coastal Plain seismicity is quite localized, for Bollinger notes that

"appreciable earthquake activity in the Coastal Plain province appears only in South Carolina." The uncertainty of the suggested trend is magnified by the sparse and often unreliable data upon which it is based. "The southeastern region has seismic monitoring inadequate to specify completely its seismicity. This in turn implies the possibility of missing any buildup or decline in that activity. Figure 3.1 illustrates the epicenter location of the various quakes and the respective dates of occurrence.

3.4.3 SOIL

The nature of soils in the area is important in the assessment of NFCS operations or in its expansion. Problems occur if soils will not support structures or lagoons, if soil permeability allows effluents to escape into aquifers, or if the engineering limitations of soils (swelling, shrinking, corrosibility to concrete and soil, and flooding potential) cannot be overcome.

Soils groups for the NFCS region are mapped in Fig. 3.2. From this figure it is apparent that the plant site occurs on the Craven-Leaf-Johns association. Craven series soils are moderately well-drained level or sloping Coastal Plain soils. The surface layer is loam, with a clay subsoil that is very firm and slowly permeable. Clayey sediments interfinger with sand lenses below. The Leaf association is poorly drained, with a silt-loam surface and silty-clay subsoil.

Both soil series in the association have certain limitations. They are highly corrosive to both concrete and steel, and they have severe shrink-swell potential and severly wetness and flooding potential because of seasonal-high water tables. The latter characteristic also decreases their suitability for septic tanks. The wetness of the soils also limits sewage lagoons and sanitary landfills.

3.4.4 SUMMARY OF GEOLOGICAL CHARACTERISTICS

The geological characteristics of the site are favorable both for current operations and for the proposed expansion. The bed-rock geology is relatively stable and structurally sound. Although tectonic activity exists, no earthquakes have been recorded within 6 miles of the site.

3.5 HYDROLOGY

3.5.1 GROUNDWATER

3.5.1.1 SURFICIAL TERRACE AQUIFERS

The principal shallow aquifers in the region of the NFCS plant lie within three general terrace formations, including the Okefenokee, the Sunderland, and the Coharie. Site specific hydrogeological data indicates that the Westinghoue site is underlain by four units. The Terrace Unit (Unit I); the Upper Black Mingo (Unite II); the lower Black Mingo, and the Tuscaloosa Formation (Unit IV). Production from shallow holes is possible within any of these terrace formations. Most of the wells in the vicinity of the NFCS plant produce water within 25 to 40 ft of the surface.

The Black Mingo Formation underlies the Terrace Unit, and two hydrogeologic units have been defined within this formation: the upper unit (Unit II) is a confining bed, and the lower unit (Unit III) is an artesian sand aquifer. Hydrogeologic Unit II is composed predominantly of clay and shale. The lithology, thickness, and estimated extremely low vertical hydraulic conductivity (K') of this unit in the area south of the main plant indicates that this unit should be an effective confining bed. It should greatly reduce the probability of vertical migration of contaminants through this unit. Even if vertical hydraulic conductivities were several orders of magnitude greater than estimates provided in this report, much greater vertical hydraulic gradients than now exist would be needed to cause downward vertical leakage of contaminants into underlying aquifers.

A survey conducted in April 1974 by the South Carolina Health Department reported that more than 700 wells used for domestic and agricultural purposes are located within 5 miles of the NFCS plant. All wells are located updip of the NFCS plant. Since there are no wells that draw groundwater from the surface aquifer downdip of the plant, substances entering the aquifer from the plant would probably discharge toward the river into Mill Creek without entering any wells that are in use.

3.5.1.2 UPPER TUSCALOOSA FORMATION AQUIFERS

The principal deep aquifers in this region of Richland County are limited to the upper, marine phase of the Tuscaloosa Formation. The nearest subcrop of these aquifers (Kti and Ktm aquifers) is approximately 14 miles to the east of the NFCS plant (Fig. 3.3). The piezometric surface of these aquifers indicates flow to the southeast, away from the NFCS. Since the aquifer is eroded, the NFCS plant is not connected hydrologically with land areas toward the northeast and east in Richland County.

3.5.1.3 LOWER TUSCALOOSA FORMATION AQUIFERS

In the NFCS vicinity, a deep aquifer of unknown regional nature is present in the lower beds of the Tuscaloosa Formation (alluvial phase) about 70 ft below the surface. The number of wells penetrating this deep aquifer is limited and confined to that portion of Richland County lying west of the Ktm aquifer subcrop. Production from these lower Tuscaloosa beds is unpredictable. Of five wells drilled into the lower Tuscaloosa Formation in the vicinity of the NFCS plant, four produced water. In all cases, production from the alluvial phase occurred very low in the stratigraphic sequence, and there was no production from the higher beds because of the occurrences of impermeable to low permeability clays. In the vicinity of the NFCS plant, the upper beds of the alluvial phase of the Tuscaloosa Formation may be regarded as an aquiclude with regard to deeper aquifers and the overlying Okefenokee terrace aquifer. Therefore, accidental discharge of industrial wastes into the surficial aquifer underlying the NFCS holding ponds would be confined to the surfical aquifer and would not migrate to the lower aquifer as long as present casings in the deep holes remain effective.

3.5.2 SURFACE WATER

Figure 3.4 shows the surface-water features in the vicinity of the NFCS facility. The major stream near the plant is the Congaree River. Principal tributaries to the Congaree include Gills Creek at Columbia, Mill Creek near the NFCS plant, and Beaver Creek near Hopkins. Sunset Lake is a shallow, artificial impoundment on Mill Creek on the NFCS property (Fig. 3.4).

3.5.2.1 CONGAREE RIVER

The Congaree River is a typical South Atlantic Piedmont stream that is characterized by high levels of suspended solids, sandy bottoms, and sand beaches. The Congaree is formed by the confluence of the Broad and Saluda Rivers, 10 miles northwest of the Westinghouse Plant at Columbia, and flows southeast for approximately 60 miles to its confluence with the Wateree River and ultimate discharge into Lake Marion near Fort Motte, South Carolina. The flow of the Congaree River is regulated by Lake Murray and Lake Greenwood on the Saluda River, and to some extent, by power plants on the Broad River. For approximately 1 mile below the confluence at Columbia, the river flows over Piedmont granites and schists before jossing sediments of the Coastal Plain province. The river valley within the Coastal Plain province is exceptionally wide, extending over 8 miles from the terraced northeastern to the scarped southwestern valley walls. The Congaree River generally occupies the southwestern part of the floodplain, flowing in a meandering pattern. At the plant discharge point, the Congaree is about 500 ft wide and approximately 9 ft deep.

The average flow of the Congaree River was calculated from computer printouts available from the South Carolina Water Resources Commission, and equals approximately 9326 cfs for the period 1939 to 1981. The lowest seven-day average flow that might be expected in a ten-year period is 1740 cfs. The minimum daily flow for the period 1941-1976 is 964 cfs. The lowest flows occur generally during the summer months. Since 1892, the highest stage of the Congaree River attained at Columbia, South Carolina, was the flood of August 27, 1908, when the stage reached 35.8 ft and discharge was 364,000 cfs. Since the flow of the Congaree has been regulated by lakes on the Saluda River, the maximum flow was 155,000 cfs on October 11, 1964, when the gage height attained was 29.74 ft. The NFCS plant is about 17 ft above the level reached by the 1908 flood water of the Congaree. On the 100-year flood-line maps of the Congaree River valley constructed by the U.S. Army Corps of Engineers, the line separating flood-prone and proximate areas occasionally flooded from higher land areas is drawn at 130 ft above MSL in the vicinity of the NFCS which is 140 ft above MSL.

3.5.2.2 MILL CREEK AND SUNSET LAKE

Mill Creek is a tributary of the Congaree River, entering the river approximately 3 miles south of the NFCS, downstream of the plant's discharge point. Sunset Lake is a shallow (6 ft maximum depth) artificial impoundment on Mill Creek one-fourth mile south the NFCS plant (Fig. 3.5). The lake originally consisted of two open-water areas, an Upper Lake with a surface area of 44 acres and a Lower Lake of 8 acres.

Mill Creek enter the Upper Lake from the north and exits the Lower Lake by passing over a a small dam located at the south end of the lake. Mill Creek then meanders through swamplands until discharging into the Congaree River 2.5 miles downstream from the NFCS waste discharge. The flow from Upper Sunset Lake to Lower Sunset Lake is by way of a narrow channel passing under a causeway. Water from the meander of Mill Creek that fed Upper Sunset Lake has been partially diverted from the lake by means of a canal (Fig. 3.5). The area that was originally the Upper Lake is now a swamp. Lower Sunset Lake still maintains an open-water area. A small dam separates Upper and Lower Sunset Lakes, and flow to the lower lake may be controlled by this dam.

Both Sunset Lake and Mill Creek could possibly receive accidental spillage from the holding ponds at the NFCS facility. These spills could enter the swampy region of the former Upper Lake through a storm drain and subsequently enter the Lower Lake and Mill Creek. Impact would probably be minimal, since all lagoons contain treated waste.

3.5.3 WATER USE

3.5.3.1 GROUNDWATER

Of the 700 wells in use within a 5-mile radius of the NFCS facility, none are located downdip of the plant. Because of the erosion of the upper Tuscaloosa aquifers, the NFCS plant is not connected hydrologically with wells that draw from these aquifers to the east and northeast of the plant. Groundwater quality will be discussed in Sect. 3.7.

3.5.3.2 SURFACE WATER

The city of Columbia uses 50 cfs (32 Mgd) of the Broad River waters for municipal use. There are no industrial or municpal water users of the Congaree River along its course from its confluence with the Saluda and Broad Rivers to its joining with the Wateree River to form the Santee River. The NFCS plant receives its water from the Columbia Municipal Water System. Projected water consumption by the NFCS plant at the expanded capacity of 1600 MTU/year is 10.6 million gallons per month (0.5 cfs).

The Congaree River receives discharges directly from the cities of Columbia, Cayce, and West Columbia, and from the NFCS plant. The city of Columbia has installed a metropolitan area waste-treatment plant, which consists of a trickling-filter and activated-sludge process. Columbia discharges currently average 42 cfs (27 Mgd) and peak at 93 cfs (60 Mgd). Cayce and West Columbia together discharge 3 cfs (1.9 Mgd) of raw sewage into the Congaree. NFCS currently discharges 0.2 cfs of process and sanitary wastes into the river. At 1600-MTU capacity, discharges will potentially increase to 0.3 cfs. The only other major industrial discharge to the Congaree subbasin is from Carolina Eastman, which discharges 49 cfs (32 Mgd) of cooling-tower water into the Congaree via Hale's Branch, upstream from the NFCS. Water quality of the river will be discussed in Section 3.7

3.6 METEOROLOGY AND CLIMATOLOGY

This section describes the climatological features and the severe weather potential for the NFCS. The atmospheric dispersion characteristics relevant to the transport of gaseous effluents from the facility are also evaluated.

A summary of local climatological features for the U.S. Weather Bureau Station at Columbia Metropolitan Airport located about 16 miles northeast of the site, is given in Table 3.4, which includes temperature, relative humidity, wind, precipitation, and mean annual number of days of climatological events. The weather in the region of the Columbia site reflects a temperate climate, with high relative humidity, moderate rainfall throughout the year, moderate winds, and normal diurnal temperature changes. Winters are mild, with rare cold waves accompanied by temperatures of zero or below. Freezing temperatures (32^{o}F) occur on an average of 77 days per year, generally during the months of November through March.

3.6.1 WINDS, TORNADOES, AND HURRICANES

In South Carolina, severe tornadoes occur almost every year, most often in the spring. During the interval 1956 through 1973, 172 tornadoes were reported in this State. Data from Richland County, where the site is located, shows that over the past 24 years (1950-1973), nine tornadoes have been reported. Thom developed an empirical formula to compute the mean recurrence interval for a tornado striking any location by approximating the location with a geometrical point. Based on the mean path area of a tornado, the number of tornadoes per year, and the area over which tornadoes may occur (Richland County), the probability of a tornado striking any location within Richland County, which includes the site, is once in more than 700 years.

During a 30-year period (1941-1970), four to five North Atlantic hurricanes out of a total of 31 penetrated into the central part of South Carolina. There was no severe damage due to winds; however, flash floods caused damage to farmlands and public utilities in the Columbia regions. The fastest wind recorded in the Columbia region is 60 mph; the calculated fastest mile of wind expected to occur in a 100-year period is 100 mph.

3.6.2 ATMOSPHERIC DISPERSION

High air pollution is caused by low mixing heights and light winds. Holzworth's data on the frequency of high air pollution potential (designated HAPP) indicated that from 1960 to 1965 the Columbia region experienced no HAPP cases of low mixing heights and light winds. Mixing heights of less than 1000m coupled with winds of less than 4 m/sec lasting two days or more occurred only once in autumn during the five-year period. Similar conditions lasting five days or more did not occur. These data indicate that central South Carolina is in a region of extremely favorable dispersion.

3.6.3 DIFFUSION CLIMATOLOGY

The annual and seasonal summary of the joint wind stability frequency was obtained from onsite meteorological data collected on site from August 1, 1972, through July 31, 1973. The data indicate that stable conditions exist 47% of the time, neutral conditions occur about 43% of the time, and unstable atmospheric conditions prevail about 10% of the time.

Seasonal stability distribution for the varous stability classes indicates that spring experiences the greatest number of hours (310) of unstable conditions as well as slightly stable conditions (412 hours); winter, the greatest number of hours (1047) of neutral conditions; and summer, the greatest number of hours (964) of stable conditions.

The annual wind rose from NFCS (Aug. 1, 1972 to July 31, 1973) is shown in Fig. 3.6. An updated map (Figure 3.7) has been included to depict the average annual wind rose at the Columbia Airport from 1948 through 1961 using NOAA data.

3.6.4 SHORT-TERM (ACCIDENT) DIFFUSION ESTIMATES

Estimates of atmospheric dilution factors (x/Q) representative of post-accident time periods up to 30 days for the 50% confidence level are given in Table 3.5 for downwind distances as far as 20 miles from the Westinghouse site, assuming a ground-level release.

TABLE 3.4
CLIMATOLOGICAL DATA FROM
COLUMBIA METROPOLITAN AIRPORT FOR 1981[a]

Parameter	Data
Temperature, °F	
Annual average	61.7
Maximum	74.1
Minimum	49.2
Record high	100
Record low	10
Degree days, heating	2999
Relative Humidity, %	
Annual average	69
Wind annual average speed mph	6.5
Prevailing direction	SW
Fastest mile	
Speed, mph	30
Direction	SW
Precipitation, in.	
Annual average	40.7
Monthly maximum	8.5
Monthly minimum	0.23
24-hr maximum	2.81
Snowfall, in.	
Annual average	T
Monthly maximum	T
24-hr maximum	T
Mean Annual (No. of Days)	
Precipitation 0.1 in.	105
Snow, sleet, hail 1.0 in.	0
Thunderstorms	46
Heavy fog	12
Maximum temperature > 90°F	62
Minimum temperature < 32°F	77

[a]NOAA Local Climatological Summary Data 1981

Table 3.5. Atmospheric dilution factors (x/Q) for accident conditions
(Data in sec/M^3)

Distance (miles)	Hours 0-8	Hours 8 to 24	Days 1 to 4	Days 4 to 30
0.07	4.539E-2	4.55E-2	1.940E-3	2.320E-4
0.1	1.255E-2	3.429E-3	1.479E-3	1.802E-4
0.2	3.760E-3	1.099E-3	4.190E-4	5.538E-5
0.3	1.916E-3	5.287E-4	2.020E-4	2.662E-5
0.5	8.400E-4	2.163E-4	8.290E-5	1.089E-5
0.568	7.71E-4	1.816E-4	6.910E-5	8.880E-6
0.7	5.74E-4	1.251E-4	4.798E-5	6.273E-6
1.0	4.23E-4	6.926E-5	2.655E-5	3.467E-6
2.0	2.07E-4	2.329E-5	8.790E-6	1.149E-6
5.0	7.48E-5	6.146E-6	2.310E-6	3.171E-7
10.0	3.28E-5	2.314E-6	8.670E-7	1.101E-7
20.0	1.509E-5	9.351E-7	3.488E-7	4.378E-8

Data taken at NFCS from August 1, 1972 through July 31, 1973
Includes building wake factors.
Exclusion distance, 114 m (374 ft).
Read as 4.539×10^{-2}

3.6.5 LONG-TERM (ROUTINE) DIFFUSION ESTIMATES

Estimates of atmospheric dilution factors (x/Q) on an annual basis at downwind distances up to 50 miles in 16 compass directions at the 50-ft level as provided in Table 3.6, assuming ground-level release. The highest x/Q values occur in the northeast sector, whereas the lowest values occur in the southern sector.

3.7 BACKGROUND CHARACTERISTICS

3.7.1 RADIOLOGICAL CHARACTERISTICS

To evaluate the significance of future releases of radioactive and nonradioactive materials to environment from expanded operations at the NFCS, present background characteristics of the environs must be determined. The background radiological characteristics presented in the section were developed from selected data from numerous published reports and from the plant's environment monitoring program. Where available, these data include uranium and gross alpha and beta levels as measured in air, water, fallout deposition, soil, and vegetation.

TABLE 3.6

ESTIMATES OF ATMOSPHERIC DILUTION FACTORS FOR NFCS ANNUAL AVERAGE x/Q VALUES

(Data in sec/m^3)

Distance Downwind (miles)	Direction from plant location															
	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
0.1	1.11E-4	1.21E-4	2.62E-4	2.18E-4	1.90E-4	1.46E-4	1.11E-4	9.83E-5	8.99E-5	1.15E-4	1.57E-4	1.72E-4	1.86E-4	1.49E-4	1.46E-4	1.17E-4
0.2	3.18E-5	3.44E-5	7.51E-5	6.27E-5	5.47E-5	4.21E-5	3.19E-5	2.84E-5	2.60E-5	3.33E-5	4.54E-5	5.00E-5	5.39E-5	4.32E-5	4.21E-5	3.36E-5
0.3	1.54E-5	1.66E-5	3.62E-5	3.03E-5	2.65E-5	2.04E-5	1.55E-5	1.38E-5	1.26E-5	1.62E-5	2.20E-5	2.43E-5	2.62E-5	2.10E-5	2.05E-5	1.63E-5
0.5	6.22E-6	6.71E-6	1.47E-5	1.23E-5	1.08E-5	8.32E-6	6.32E-6	5.63E-6	5.15E-6	6.59E-6	8.97E-6	9.90E-6	1.07E-5	8.58E-6	8.35E-6	6.64E-6
0.8	2.76E-6	2.97E-6	6.52E-6	5.48E-6	4.80E-6	3.70E-6	2.81E-6	2.51E-6	2.29E-6	2.93E-6	3.99E-6	4.41E-6	4.77E-6	3.83E-6	3.72E-6	2.95E-6
1.0	1.89E-6	2.04E-6	4.49E-7	3.78E-6	3.31E-6	2.55E-6	1.94E-6	1.73E-6	1.58E-6	2.02E-6	2.75E-6	3.04E-6	3.29E-6	2.64E-6	2.56E-6	2.03E-6
1.5	9.80E-7	1.05E-6	2.34E-6	1.97E-6	1.72E-6	1.33E-7	1.01E-6	9.00E-7	8.24E-7	1.05E-6	1.43E-6	1.59E-6	1.72E-6	1.38E-6	1.33E-6	1.06E-6
2.5	4.42E-7	4.76E-7	1.07E-6	9.03E-7	7.87E-7	6.0[illegible]-7	4.58E-7	4.10E-7	3.76E-7	4.79E-7	6.53E-7	7.28E-7	7.87E-7	6.30E-7	6.07E-7	4.80E-7
3.5	2.65E-7	2.85E-7	6.49E-7	5.47E-7	4.76E-7	3.64E-7	2.76E-7	2.47E-7	2.27E-7	2.89E-7	3.94E-7	4.41E-7	4.77E-7	3.82E-7	3.67E-7	2.90E-7
4.5	1.82E-7	1.96E-7	4.49E-7	3.79E-7	3.29E-7	2.51E-7	1.90E-7	1.71E-7	1.57E-7	1.99E-7	2.72E-7	3.06E-7	3.31E-7	2.64E-7	2.53E-7	1.99E-7
7.5	8.69E-8	9.38E-8	2.19E-7	1.84E-7	1.59E-7	1.21E-7	9.12E-7	8.21E-8	7.55E-8	9.59E-8	1.31E-7	1.48E-7	1.61E-7	1.28E-7	1.22E-7	9.55E-8
10.0	5.78E-8	6.25E-8	1.47E-7	1.24E-7	1.07E-7	8.07E-8	6.08E-8	5.48E-8	5.05E-8	6.41E-8	8.77E-8	9.97E-8	1.08E-7	8.60E-8	8.14E-8	6.36E-8
15.0	3.16E-8	3.41E-8	8.07E-8	6.80E-8	5.85E-8	4.44E-8	3.34E-8	3.01E-8	2.78E-8	3.52E-8	4.82E-8	5.49E-8	5.95E-8	4.74E-8	4.49E-8	3.50E-8
20.0	2.20E-8	2.39E-8	5.79E-8	4.86E-8	4.14E-8	3.10E-8	2.32E-8	2.11E-8	1.95E-8	2.47E-8	3.38E-8	3.98E-8	4.21E-8	3.34E-8	3.13E-8	2.43E-8
25.0	1.66E-8	1.82E-8	4.48E-8	3.75E-8	3.17E-8	2.36E-8	1.76E-8	1.60E-8	1.49E-8	1.88E-8	2.58E-8	2.99E-8	3.23E-8	2.55E-8	2.37E-8	1.84E-8
30.0	1.32E-8	1.46E-8	3.64E-8	3.04E-8	2.56E-8	1.89E-8	1.40E-8	1.28E-8	1.19E-8	1.50E-8	2.07E-8	2.41E-8	2.60E-8	2.05E-8	1.90E-8	1.47E-8
35.0	1.09E-8	1.21E-8	3.06E-8	2.54E-8	2.13E-8	1.56E-8	1.16E-8	1.06E-8	9.91E-9	1.25E-8	1.72E-8	2.01E-8	2.17E-8	1.71E-8	1.57E-8	1.21E-8
40.0	9.28E-9	1.03E-8	2.63E-8	2.18E-8	1.82E-8	1.33E-8	9.82E-9	9.02E-9	8.44E-9	1.06E-8	1.47E-8	1.72E-8	1.85E-8	1.46E-8	1.33E-8	1.03E-8
45.0	8.03E-9	8.93E-9	2.30E-8	1.91E-8	1.59E-8	1.15E-8	8.50E-9	7.82E-9	7.34E-9	9.22E-9	1.27E-8	1.50E-8	1.61E-8	1.27E-8	1.15E-8	8.89E-9
50.0	7.06E-9	7.88E-9	2.04E-8	1.69E-8	1.41E-8	1.02E-8	7.47E-9	6.88E-9	6.47E-9	8.13E-9	1.12E-8	1.32E-8	1.43E-8	1.12E-8	1.01E-8	7.82E-9

TABLE 3.7

AVERAGE DIRECT RADIATION MEASUREMENTS

(MILLIREMS/DAY) WITH TLDS[a]

LOCATION OF STATIONS RELEVANT TO THE W PLANT

Sample Period	8 miles northwest	3000 ft. northeast	2250 ft. east	2250 ft. northwest	7miles northeast
12/17/80 - 03/25/81	0.21	0.22	0.23	0.21	0.20
03/25/81 - 06/24/81	0.18	0.14	0.17	0.18	0.15
06/24/81 - 09/23/81	0.22	0.23	0.22	0.21	0.23
09/23/81 - 12/10/81	0.25	0.25	0.26	0.27	0.23
12/10/81 - 03/31/82	0.27	0.23	0.25	0.25	0.26

[a]SC-DHEC FILE DATA

3.7.1.1 TOTAL-BODY DOSE RATES

Based on Estimates of Ionizing Radiation Doses in the U.S. 1960-2000, the total-body dose rate from natural background radiation in the vicinity of Columbia is expected to be similar to that for South Carolina in general, that is, about 135 millirems/year (70 millirems/year from external terrestrial radiation, 40 millirems/year from cosmic rays, and 25 millirems/year from internal terrestrial radiation). This value compares favorably with an average of 0.32 millirem/day (117 millirems/year) previously reported by the State for areas in South Carolina where there are no nuclear facilities.

The above dose rates can be compared with those recently measured at six locations in the vicinity of the plant. These data, listed in Table 3.7, indicate an average dose rate of 0.14 to 0.27 millirem/day, or 51 to 99 millirems/year.

3.7.1.2 SUMMARY

A summary of present background radiological characteristics typical of the NFCS environs is presented in Table 3.8. Because of the variations in reported data, typical ranges rather than specific values are given for some of the parameters.

3.7.2 NONRADIOLOGICAL CHARACTERISTICS

3.7.2.1 ATMOSPHERIC EFFLUENTS

Fluorides and ammonia are the main atmospheric chemicals discharged from the NFCS plant. Sporadic NO_x emissions from the nickel-plating-room stack were measured at 2 ppm. Sporadic NO_x emissions equal to 15 ppm have been detected in the scrap recovery exhaust.

No data were available to describe ambient atmospheric concentration values for either ammonia or fluorides or to describe deposition values for ammonia. Average annual fluorides deposition was less than 0.3 ug cm^{-2} $year^{-1}$. The average ambient atmospheric concentration for SO_2 was 32 ug/m^3. A maximum 1-hr value of 100 ug/m^3 was recorded.

3.7.2.2 BACKGROUND NONRADIOLOGICAL CHARACTERISTICS OF WATER

Groundwater quality

Table 3.9 lists groundwater quality for the surface aquifer northeast of the NFCS plant.

Table 3.10 documents the water quality (Well #7) of the surface aquifer south-southwest of the plant in the direction of predominate flow of groundwater from the plant and waste treatment lagoons. See section 5.2.3 for a further discussion.

Congaree River

Tables 3.11 and 3.12 address water quality data in 1981 for the Congaree River as calculated from SC-DHEC records. No seasonal trends or yearly maximums for the values in the tables are obtainable. One station is located above the Columbia plant and one is below.

Effects of municipal discharges of sewage from the city of Columbia and other communities upstream are reflected in high bacteria counts in the downstream Congaree River. The elevated ammonia levels in the downstream sample are probably a result of the presence of agricultural runoff and sewage in the river.

3.8 ECOLOGY

3.8.1 TERRESTRIAL BIOTA

3.8.1.1 GENERAL FEATURES

Characteristics of the terrestrial biota that are important in the assessment of the current operations or proposed expansion of the NFCS relate primarily to atmospheric effluent loads derived from the facility. The kinds of biotic problems that occur in association with long linear structures or with disturbance of large land areas need not be considered here because the plant is compact and covers only 60 acres. Instead, biotic relationships must be evaluated with respect to atmospheric contaminant concentrations, their downwind diminution, and their gradual accumulation in plant and animal tissues and in soils. Therefore the nature and composition of the plant and animal communities at increasing distances from the NFCS will be discussed in the following paragraphs.

The potential vegetation of the area is classified by Kuchler as southern floodplain forest along the Congaree River, with oak-hickory-pine forest on the uplands and southern mixed forest immediately west of Columbia, Table 3.14 lists dominant and associated species in each of these plant associations. Species common to all three associations and those found in any two of the three communities are footnoted.

That so few important species are common to the three associations indicates that physical environmental limiting factors exert strong control on the composition and character of the plant communities. This is particularly apparent in the case of the swamp forest, which shares only 11% of its major species with the southern mixed forest and 18% with the oak-hickory-pine forest. The second two are more similar to each other, but they still share only 43% of their important species.

TABLE 3.8
SUMMARY OF BACKGROUND RADIOLOGICAL
CHARACTERISTICS TYPICAL OF THE COLUMBIA PLANT ENVIRONS (1981-1982)

Measured parameter	Gross Alpha (Range)	Average
Ambient Air	2.0×10^{-15} to $6.4\ 10^{-14}$ uCi/ml	3.9×10^{-15} uCi/ml
Surface water		
Congaree River	2.2×10^{-9} uCi/ml	2.2×10^{-9} uCi/ml
Sunset Lake	1 to 56	4×10^{-9} uCi/ml
Road (storm drain)	1 to 87 $\times 10^{-9}$ uCi/ml	18×10^{-9} uCi/ml
Well water		
Onsite	1 to 110 $\times 10^{-9}$ uCi/ml	5×10^{-9} uCi/ml
Offsite	1×10^{-9} uCi/ml	
Drinking water	<0.5 to 1.4×10^{-9} uCi/ml	1×10^{-9} uCi/ml
Vegetation	1 to 5 pCi/g	2.2 pCi/g
Soil	1 to 5 pCi/g	3.6 pCi/g
TLD's, Whole body dose	51 to 160 mR/year	80 mR/year

TABLE 3.9
ANALYSIS OF GROUNDWATER
SURFACE AQUIFER NORTHEAST OF PLANT (WELL #24)

Parameter	mg/l, or as indicated
Arsenic	< 0.005
Barium	< 0.1
Cadmium	< 0.005
Chromium	< 0.01
Fluoride	< 1.0
Lead	< 0.05
Mercury, ug/l	< 0.2
Nitrate Nitrogen	0.7
Ammonia Nitrogen	< 1
Selenium	< 0.01
Silver	< 0.01
Turbidity	1.7
Chloride	2.0
Hydrogen Sulfide	< 0.01
Copper	< 0.01
Iron	0.04
Manganese	0.018
Sulfate	18
Dissolved Solids	50
Zinc	0.042
Color	5
pH, units	6.0
Surfactants(MBAS)	0.47
Nickel	< 0.01
Conductance, uMHOS	180

TABLE 3.10
ANALYSIS OF GROUNDWATER
SURFACE AQUIFER SOUTHWEST OF PLANT (WELL #7)

Parameter	mg/l, or as indicated
Arsenic	< 0.005
Barium	0.1
Cadmium	< 0.005
Chromium	< 0.01
Fluoride	49
Lead	< 0.05
Mercury, ug/l	< 0.2
Nitrate Nitrogen	310
Ammonia Nitrogen	602
Selenium	< 0.01
Silver	< 0.01
Turbidity	1.2
Chloride	20
Hydrogen Sulfide	< 0.01
Copper	< 0.01
Iron	0.04
Manganese	< 0.005
Sulfate	125
Dissolved Solids	642
Zinc	.021
Color	25
pH, units	9.4
Surfactants(MBAS)	0.54
Nickel	0.02
Conductance, uMHOS	2950

TABLE 3.11
CONGAREE RIVER ANNUAL (1981) WATER QUALITY AVERAGES
AT BLOSSOM STREET BRIDGE UPSTREAM OF PLANT[a]

Constituent	Value
Water, temp, °C	16
Turbidity, Hach	14
Conductivity uMHOS	68
D.O., mg/l	10.1
BOD_5, mg/l	3.4
pH, units	7.1
Total Alkalinity, mg/l	16
$NH_3 + NH_4$, mg/l	0.53
$NO_2 + NO_3$, mg/l	0.34
Phosphates, mg/l	.13
Total Organic Carbon, mg/l	4.3
Cadmium, ug/l	< 10
Chromium, ug/l	50
Copper, ug/l	< 50
Iron, ug/l	787
Nickel, ug/l	< 50
Lead, ug/l	55
Mercury, ug/l	0.2
Fecal Coliform, MFN/100ml	249

[a]SCDHEC Data

TABLE 3.12
CONGAREE RIVER ANNUAL DOWNSTREAM OF PLANT
WATER QUALITY AVERAGES AT US 601 BRIDGE (1981)[a]

Constituent	Value
Water, temp, ^{o}C	16.8
Turbidity, Hach	14.1
Conductivity, uMHOS	70
D. O. mg/l	8.1
BOD_5, mg/l	3.9
pH, units	6.8
Total Alkalinity mg/l	17
$NH_3 + NH_4$, mg/l	.095
$NO_2 + NO_3$, mg/l	.042
Phosphates, total, mg/l	0.32
Total Organic Carbon, mg/l	5.1
Cadmium, ug/l	< 10
Chromium, ug/l	< 50
Copper, ug/l	< 50
Iron ug/l	1300
Nickel ug/l	< 50
Lead ug/l	< 50
Mercury ug/l	0.3
Fecal Coliform, MFN/100 ml	1490

[a]SCDHEC Data

TABLE 3.13
RECOMMENDED SURFACE WATER CRITERIA FOR PUBLIC WATER SUPPLIES

Constituent or characteristic	Permissible criteria (mg/liter)	Desirable criteria mg/liter
Ammonia	0.5 (as nitrogen)	0.01
Arsenic	0.10	Absent
Barium	1.0	Absent
Boron	1.0	Absent
Cadmium	0.01	Absent
Chloride	250	25
Chromium (hexavalent)	0.05	Absent
Copper	1.0	Virtually absent
Dissolved oxygen		
Monthly mean	> 4	Near saturation
Individual sample	> 3	
Iron (filterable)	0.3	Virtually absent
Lead	0.05	Absent
Manganese (filterable)	0.05	Absent
Nitrates plus nitrites	10 (as N)	Virtually absent
pH (range)	6.0-8.5	
Selenium	0.01	Absent
Silver	0.05	Absent
Sulfate	250	50
Total dissolved solids (filterable residue)	500	200
Zinc	5	Virtually absent
Cyanide	0.2	Absent
Oil and grease	Virtually absent	Absent
Total carbon	0.15	0.04
DDT	0.042	Absent
Dieldrin	0.017	Absent
Heptaepoxide	0.018	Absent
2-4-D	0.1	Absent
Total coliform	20,000 per 100 ml	
Fecal coliform	2,000 per 100 ml	
Mercury	0.002	Absent

Sources: Committee on Water Quality Criteria, Water Quality Criteria, 1972, National Academy of Sciences - National Academy of Engineering, Washington, D. C. 1972.

TABLE 3.14
POTENTIAL NATURAL VEGETATION OF THE COLUMBIA, SOUTH CAROLINA AREA

SOUTHERN FLOODPLAIN FOREST (Quercus-Nyssa-Taxodium)

Physiognomy:	Dense, medium tall to tall forest of broadleaf deciduous and evergreen trees and shrubs and needleleaf deciduous trees
Dominants:	Tupelo (Nyssa aquatica) Oak (Quercus spp.) Bald cypress (Taxodium distichum)
Other components:	Acer rubrum var. drummondii, Ampelopsis arborea, Berchemia scandens, Campsis radicans, Carya aquatica, C. illinoensis, Celtis laevigata, Forestiera acuminata, Franxinus caroliniana, f. profunday, Gleditsia aquatica, Ilex decidua, Liquidambar styraciflua,[a] Nyssa silvatica,[b] N. silvatica var. biflora, Persea borbonia,[a] Planera aquatica, Platanus occidentalis, populus deltoides, P. heterophylla, Quercus falcata[a] var. pagadoefolia, Q. lyrata, Q. michauxii, Q. nigra, Q. shumardii,[b] Salix nigra, Ulmus americana, Vitis spp.

OAK-HICKORY-PINE FOREST (Quercus-Carya-Pinus)

Physiognomy:	Medium tall to tall forest of broadleaf deciduous and needleleaf evergreen trees
Dominants:	Hickory (Carya spp.) Shortleaf pine (Pinus echinata)[b] Loblolly pine (P. taeda)[b] White oak (Quercus alba)[b] Post oak (Q. stellata)[b]
Other components:	Carya cordiformis, C. glabra,[b] C. ovata, C. tomentosa,[b] Cornus florida,[b] Diospyros virginiana, Liquidambar styraciflua,[a] Liriodendron tulipifera,[b] Nyssa sylvatica,[b] Oxydendrum arboreum, Persea borbonia (lower elevations),[a] Pinus virginiana, Quercus coccinea, Q. falcata,[a] Q. marilandica,[b] Q. prinus, Q. rubra, Q. shumardii,[b] Q. velutina

SOUTHERN MIXED FOREST (Fagus-Liquidambar-Magnolia-Pinus-Quercus)

Physiognomy	Tall forest of broadleaf deciduous and evergreen and needleleaf evergreen trees

Dominants:	Beech (Fagus grandifolia) Sweet gum (Liquidambar styraciflua)[a] Souther magnolia (Magnolia grandiflora) Slash pine (Pinus elliottii Loblolly pine (P. taeda)[b] White oak (Quercus alba)[b] Laurel oak (Q. laurifolia)
Other components:	Acer barbatum, Carpinus caroliniana, Carya glabra (northern part),[b] C. tomentosa (northern part),[b] Cornus florida,[b] Ilex glabra, I. opaca, Liriodendron tulipifera,[b] Myrica cerifera, Ostrya virginiana, Persea borbonia,[a] Pinus echinata (northern part,[b] P. palustris, Quercus falcata,[a] Q. incana, Q. laevis, Q. marilandica,[b] Q. stellata,[b] var. margaretta, Q. virginiana (Florida and coastal regions), Sabal palmetto (eastern part, Serenoa repens

[a]Species common to all three plant associations.
[b]Species found in any two of the plant associations.

The physical parameters that control plant community composition near the Columbia plant may be a function of soils and the associated water drainage. For example, specialized morphological adaptations are required for survival of trees that grow in the permanently standing water of swampland. The well-drained sandy soils of the Orangeburg soil series are distinctly zeric (dry) in comparison to the wet, poorly drained clays and silty clays of the Leaf and Craven soil series. Unfortunately, data were not available from the NFCS to demonstrate that these specific soils and soil moisture conditions produce the distinct community boundaries as described in the data mentioned above.

The nature of the parameters that control plant community composition, structure, and density are important when the impact of proposed land uses is considered. If the vegetation is primarily under biological control (i.e., competition, predation, parasitism, etc.), the impact of a minor change in physical or chemical parameters may induce significant community-wide changes in species composition, structure, and density. Thus the environmental change is translated via the intricate dependent relationships among the organisms into responses by a large number of species.

On the other hand, if a larger portion of the control upon vegetation is exerted by physical parameters (i.e., depth to water table, soil structure, drainage characteristics, etc.), the impact of a minor change in physical or chemical parameters may induce little significant change in species composition, structure, and density. In this case, the environmental change is translated more independently into responses by individual species rather than into responses by groups of interdependent species.

Because the animal populations depend either directly or indirectly upon characteristics of the plant communities, they change similarly to the plant communities. Hypothetically, greater impact from changes in physical and chemical parameters may occur among animal populations that are associated with plant communities controlled by biological paramters rather than those controlled by physical parameters.

It seems likely that the latter situation occurs in the biotic communities surrounding the NFCS, that is, physical parameters control biotic composition and density.

3.8.1.2 FOOD WEB RELATIONSHIPS

A food web was constructed to show the relationships between some of the common species on the site. The web illustrates possible pathways for uptake and concentration of pollutants in the food chain. This food web includes man as a consumer of several species that could be harvested primarily by hunting (Fig. 3.8).

3.8.1.3 THREATENED AND ENDANGERED SPECIES

Although several threatened or endangered vertebrate species are known to exist in South Carolina, few are likely to be found in the immediate site area. Previous reports suggest that visits to the site by the endagered Souther bald eagle (Haliaeltus leucocephalus) and American peregrine

falcon (Falco peregrinus anatum) are rare. The endangered red-cocaded woodpecker (Denerocopus borealis) and threatened Eastern brown pelican (Pelicanus occidentalis), both present in South Carolina, are birds of restricted habitats, and their presence onsite is unlikely. The endangered Bachman's warbler (Vermivora bachanii) may occur in the swampy river bottom habitat on the site, but its presence was not detected. No Eastern cougars (Felis concolor), either resident or transient, are expected onsite. American alligators (Alligator mississippiensis) may have occurred onsite.

There are nine endangered and 35 threatened plant species that occur in South Carolina. None of the nine endangered species are know to occur in R hland County or the six immediately adjacent counties. The nine threatened species occurring in this area are listed in Appendix A.12. Of these, three occur in Richland County (Isoetes melanospora, Nestronia umbellula, and Sarracenia ruba). Two of the species are restricted to habitats or physiographic areas that do not exist at the NFCS (Appendix A.12). The lone exception is Isoetes melanospora, which prefers stream edges. Isoetes melanospora therefore may occur along Mill Creek. Even if other threatened species listed in Appendix A. 12 occurred at the plant in the past, disturbance for farming, then for plant construction, would have eliminated them.

3.8.2 AQUATIC BIOTA

3.8.2.1 INTRODUCTION

Aquatic habitats occuring in the vicinity of the plant include the Congaree River, Mill Creek, and Sunset Lake. The Congaree River is a typical South Atlantic Piedmont stream characterized by high levels of suspended solids, a sandy bottom, and sand beaches. River flow averages 9326 cfs. At the NFCS discharge point, the river is approximately 500 ft. wide and 9 ft. deep. Water quality has been described in the preceding section.

Sunset Lake is a shallow, artificial impoundment on Mill Creek. Upper Sunset Lake is now a swamp supporting a mixed stand of swamp tupelo (Nyssa aquatica) and Carolina ash (Fraxinus caroliniana). The water surface is covered by a dense duckweed mat (Spirodela polyrihiza and Lemna minor). Lower Sunset Lake still maintains an open-water area, although the encroachment of swamp tupelo and macrophyte growth is evident. This small lake exhibits a brown coloration, probably due to the presence of humic substances.

3.8.2.2 AVAILABLE ENVIRONMENTAL DATA

A review of information available on the Congaree River indicates that very little biological survey work has been done in the area. Contacts with State and Federal agencies and the University of South Carolina provided minimal information. According to the South Carolina Fish and Wildlife Department, there have been no creel census of the Congaree River. A listing of typical fish species found in the Congaree is provided in Table 3.15. Various expeditions conducted with the South Carolina Department of Health and Environmental Control have verified the presence of these species. Westinghouse Environmental Systems Group

performed a detailed survey in 1974 with the original Environmental report. No substantial deviations from that data would be expected.

3.8.2.3 CONGAREE RIVER

The Congaree River, near the Columbia plant discharge, is close enough to the sea that its flow is relatively slow, the waters are warm (+20°C), and the bottom is generally composed of muds and sands. Recreational use of the river is somewhat limited due to the remoteness of the location and limited launch sites. Fauna of such warm, slow rivers are either adapted to warm water or are tolerant to a wide range of temperatures; there is often a rich plankton fauna. Detritus and plant debris washed into the river from the surrounding lands makes up a large part of the nutrients at the base of the food web. The food web of rivers is a complex structure, with organisms often feeding at two or more trophic levels. For example, benthic invertebrates may feed on submerged vegetation as primary consumers or on zooplankton as secondary consumers. Fish may function as primary consumers on plants, as secondary consumers on invertebrates, and as tertiary consumers on other fish.

In the past, the Congaree River has received a large amount of raw sewage. At the time WESD conducted the biological survey, the City of Columbia was still discharging some raw sewage into the river. Many of the organisms collected from the Congaree River, such as tubificid worms, midge larvae (Chironomidae), and phytoplankton of the genera Nitzchia, Melospira, and Pteromonas, are representative of waters rich in organic matter matter with little dissolved oxygen.

Phytoplankton collected by two samples from the river in the vicinity of the NFCS discharge in October 1974 were predominately the colonial green algae, Eudorina elegans. Twenty-two species of phtoplankton were identified. Of the total number of individuals collected, 73% were members of Chorophyta, 14% of Chrysophta, 12% Cyanophyta, and 1% of Euglenophya and Pyrrophyta. Average cell numbers in the river were 500 cells per milliliter. Some of the phytoplankton present may have been washed down from the lakes on the Saluda River.

Zooplanton collected from tow samples were very sparse. Thirty-three species were identified. The larval stage of bivalve mollusks (glochidia), copepods, oligochaete worms, nemotods, and the protozoan Epistilis plicatilis were most frequently encountered. Glochidia comprised 21% of the individuals collected. Rotifers, cladeocerans, and insects were also collected.

Samples from a variety of substrates (logs, leaves, and rocks) in the river yielded 112 species of periphyton. Ninety-seven percent of the species were diatoms. Some of the more abundant diatoms observed were Achnanthes deflexa, Navicula minima, Navicula mutica, and Navivula crytocephala. Green algae, ostly Ulothrix sp., and blue-green algae, Microcoleius vaginatus and Oscillatoria sp., were observed infrequently.

Benthic invertebrates were collected by ponar dredge samples from the river above and below the NFCS discharge. Of the four phyla collected, 43% were mollusks, 29% were annelids, 27% were arthropods (insects), and 1% were nematodes. Fingernail clams, Sphaerium sp., were the most numerous organisms. Corbiculid clams occurred only below the NFCS discharge. Midge larvae (Chironomidae) and tubificid worms were numerous both above and below the discharge.

Recent electroshocking expedition in July, 1982 with the SC-DHEC, Bureau of Radiological Health have yielded the following species: striped bass, catfish, long nosed gar, and shad.

TABLE 3.15
MAJOR FISH SPECIES THAT PRESENTLY OCCUR IN THE CONGAREE RIVER

Scientific Name	Common Names
Lepisoteidae	
lepisosteus osseus	Long-nose gar
Amiidae	
Amia calva	Bowfin
Clupeidae	
Dorosoma cepedianum	Gizzard shad
Cyprinidae	
Cyprinus carpio	Carp
Ictaluridae	
Ictalurus natalis	Yellow bullhead
I. nebulosus	Brown bullhead
I. punctatus	Channel catfish
Serranidae	
Morone saxatilis	Striped bass
M. chrysops	White bass
Centrarchidae	
Lepomis macrochirus	Bluegill
Micropterus dolomieui	Smallmouth bass
M. salmoides	Largemouth bass
Pomoxis annularis	White crappie
P. nigromaculatus	Black crappie

Source: (South Carolina Fish and Wildlife Department).

Fish are generally at the top levels of the food web in aquatic ecosystems. However, because many species will eat a variety of foods, including plants, detritus, and invertebrates, it is impossible to assign a definite position in the food web to a large proportion of the species. The most widespread and important food for river fishes is invertebrates, which form the major part of the diet of a wide range of types of fishes. The majority of the fish likely to be found in the Congaree River, such as bass, bluegill, and crappies, feed on invertebrates. The range of invertebrates eaten is wide, and fish generally eat whatever is available. Largemouth bass and large catfish may eat other fishes as well as other vertebrates such as frogs, salamanders, and young ducklings. Carp, gar, and catfish are bottom feeders and often function as scavengers. Detritus and plant debris washed into the river may make up a large part of the diet of some fishes, especially gizzard shad.

3.8.2.4 SUNSET LAKE AND MILL CREEK

Sunset Lake is a shallow, humic lake with abundant decomposing organic matter and stands of emergent vegetation such as yellow water lily (Nuphar advena), lizard tails (Saururus cernus), and St. John's wort (Hypericum spathulatum). Dissolved oxygen levels in the lake were low (less than 4 ppm) due to high temperatures, low flow, and decomposing organic matter. The only benthic invertebrate collected was the abundant phantom-midge (Chaobor s punctipennis), which is tolerant of low dissolved oxygen levels.

The plankton fauna of Sunset Lake were abundant. Phytoplankton densities averaged 60,000 plankters per milliliter. Predominant phytoplankters in the lake were the colonial green algae Eudorina elegans. In general, green algae constituted the majority of the phytoplankton community, although diatoms, euglenoids, bluegreens, and dinoflagellates were also represented. Zooplankton species were predominately protozoans (Difflugia lobostoma and Difflugia oblonga) and the rotifer Asplanchna priodonta. Both zooplankton and phytoplankton were more abundant at the inflow end of the lake, probably as a result of the inflow of swamp water from Upper Sunset Lake.

Table 3.16 lists fish species collected from Sunset Lake and Mill Creek in a 1974 expedition. Bluegill and golden shiners (Notemigonus crysoleucas) were the most abundant species. Recent samplings in 1981 and 1982 have yielded the following species: dogfish, carp, catfish, crappie, and bluegill. Employe fishing is allowed on Lower Sunset Lake and on Mill Creek on the plant property.

3.8.2.5 THREATENED AND ENDANGERED AQUATIC SPECIES

The only threatened and endangered aquatic species that might occur in the region of South Carolina near the Columbia Plant is the American alligator, Alligator mississippiensis.

TABLE 3.16
ENUMERATION OF FISH SPECIES COLLECTED FROM THE CONGAREE RIVER
AND SUNSET LAKE FROM OCT. 1-3, 1974

Scientific Name	Common Name	No. Collected by gill nets	No. Collected by electroshocking	Total No. Collected
Lepisosteidae				
Lepisosteus osseus	Long-nose gar	3	0	3
Amiidae				
Amia calva	Bowfin	1	0	1
Cyprinidae				
Notemigonus crysoleucas	Golden shiner	18	0	18
Catostomidae				
Minytrema melanops	Spotted sucker	2	0	2
Ictaluridae				
Ictalurus natalis	Yellow bullhead catfish	1	0	1
Ictalurus mebulosus	Brown bullhead catfish	1	0	1
Ictalurus punctatus[a]	Channel catfish	0	1	1
	Unidentified catfish	4	0	4
Centrarchidae				
Centrarchus macropterus	Flier	1	1	2
Chaenobryttus gulosus	Warmouth	6	0	6
Lepomis machrochirus	Bluegill	68	1	69
Pomoxis nigromaculatus	Black crappie	8	0	8
Unidentified fish		2	0	2
Totals		115	3	118

[a]This was the only fish of its kind collected from the Congaree River.

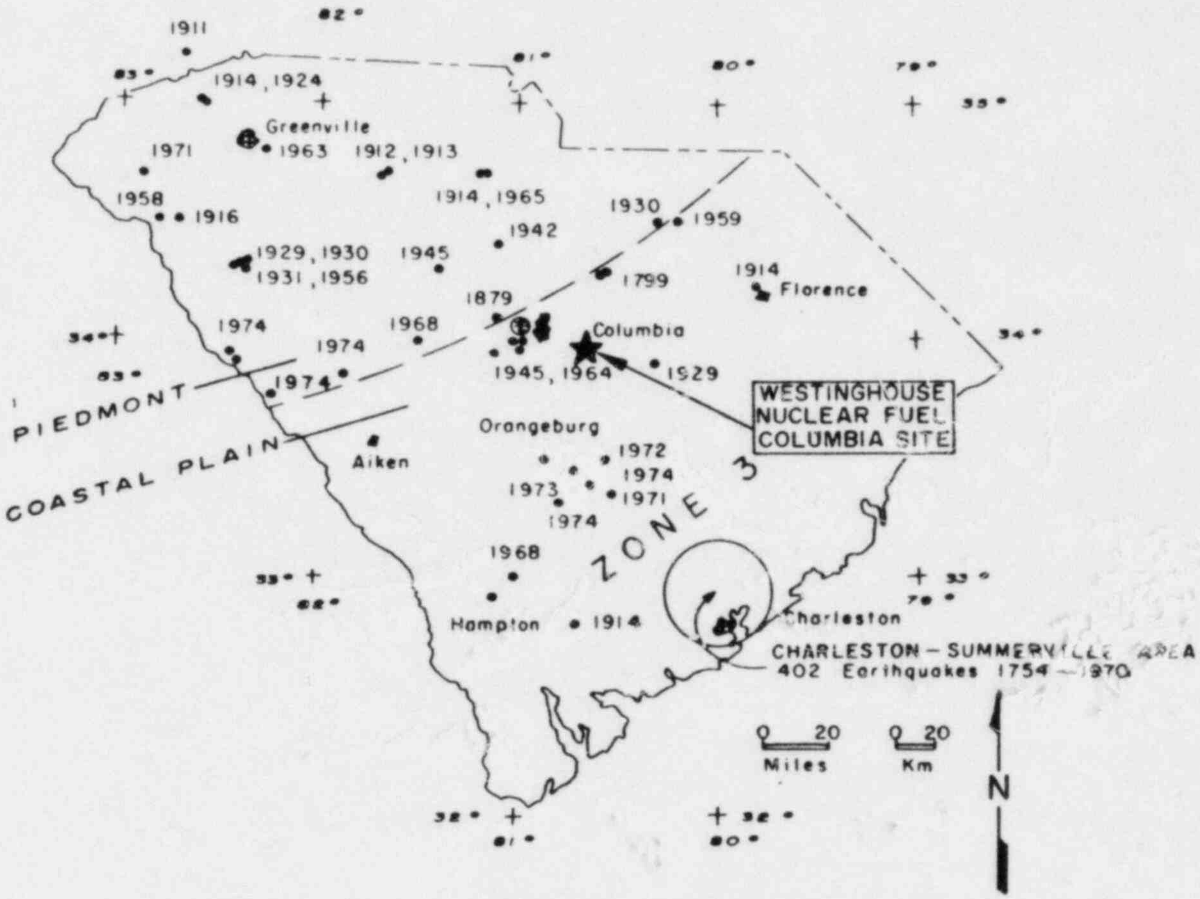


Figure 3.1. Seismicity Map of South Carolina With Year of Occurrence Noted by Epicenter. Seismic Risk Zone 3 Circled.

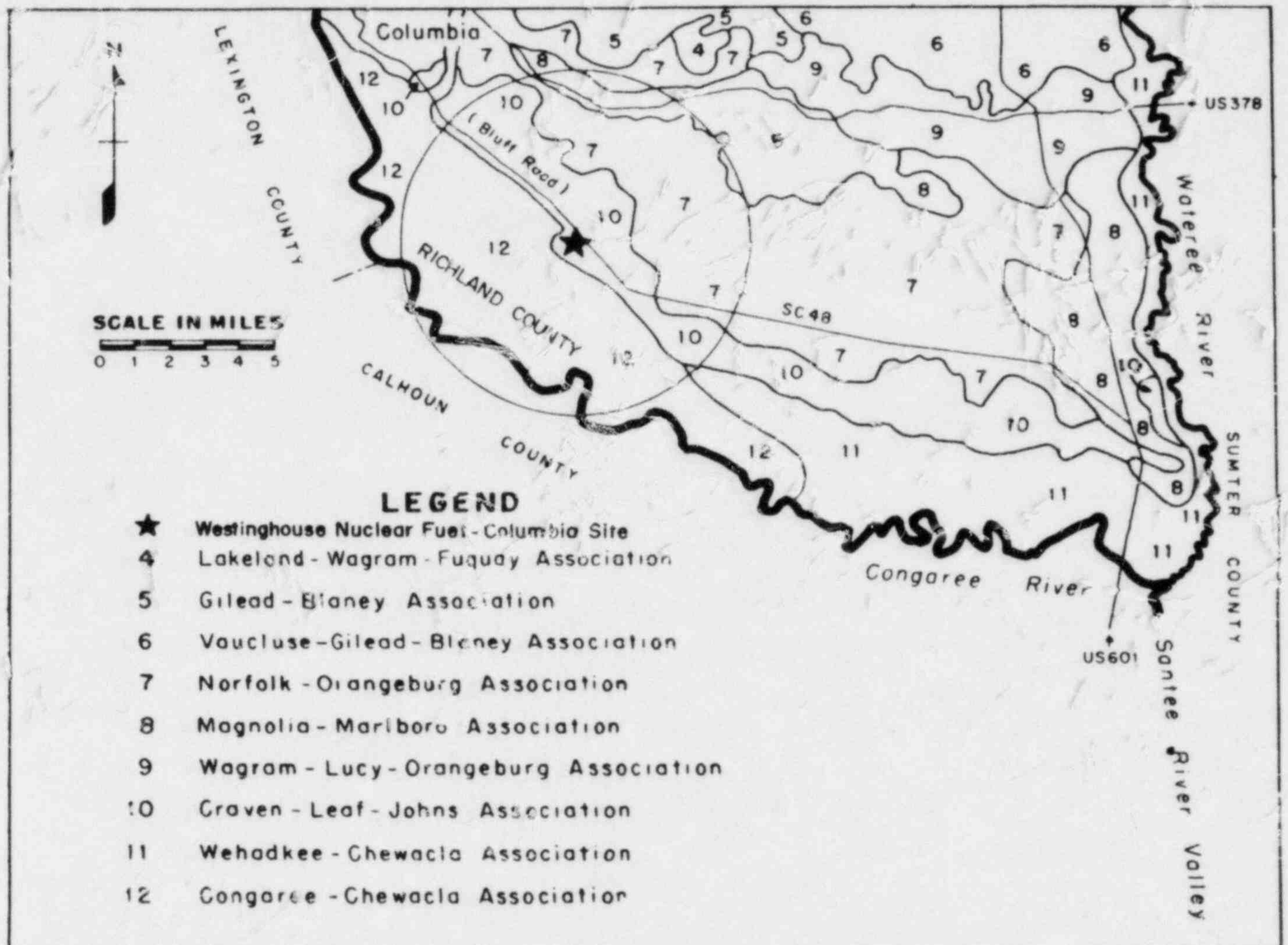


Figure 3.2. Generalized Soil Associations of Southern Richland County, South Carolina.

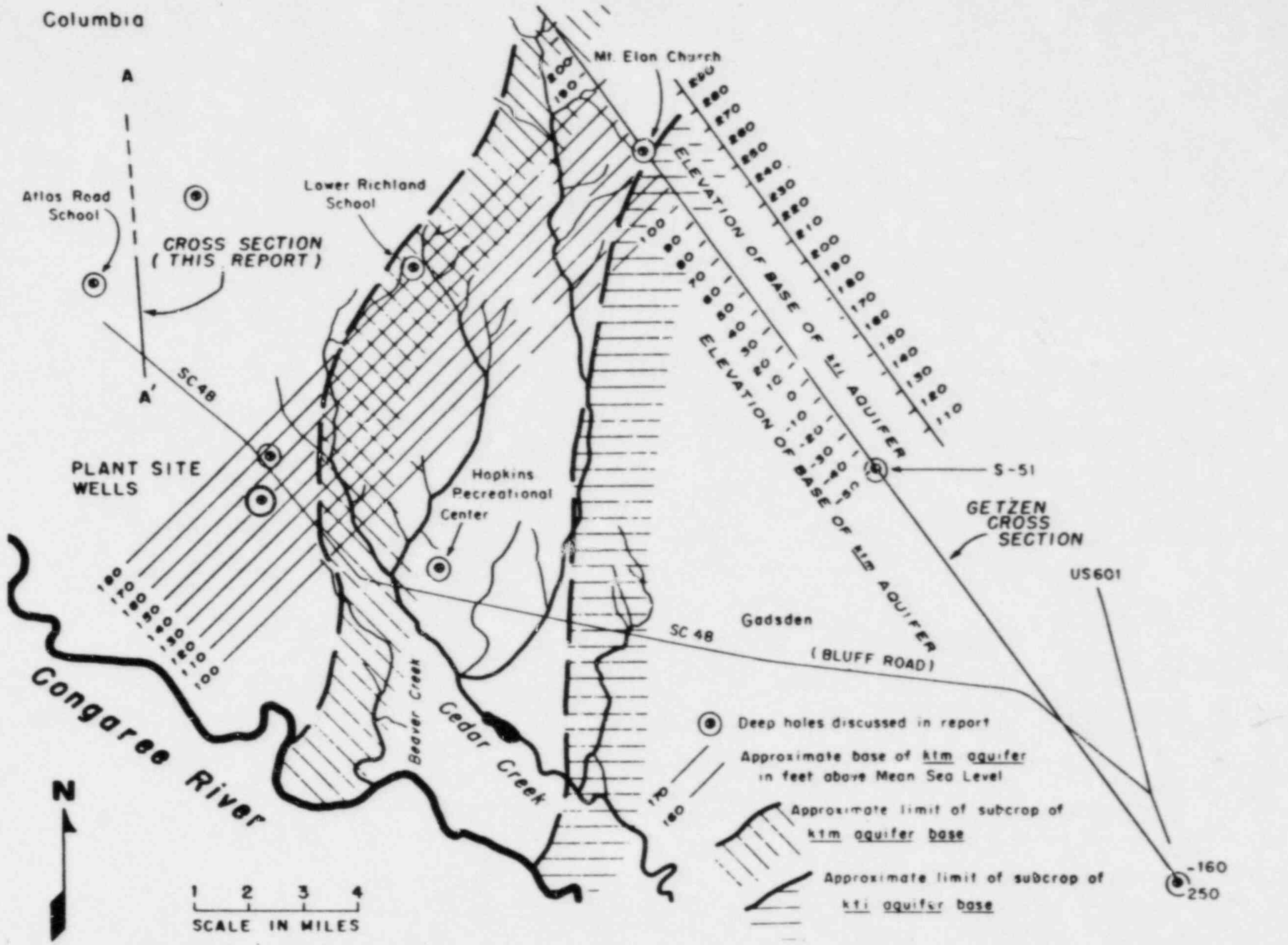


Figure 3.3. Principle Deep Aquifers in the Vicinity of Westinghouse Plant.

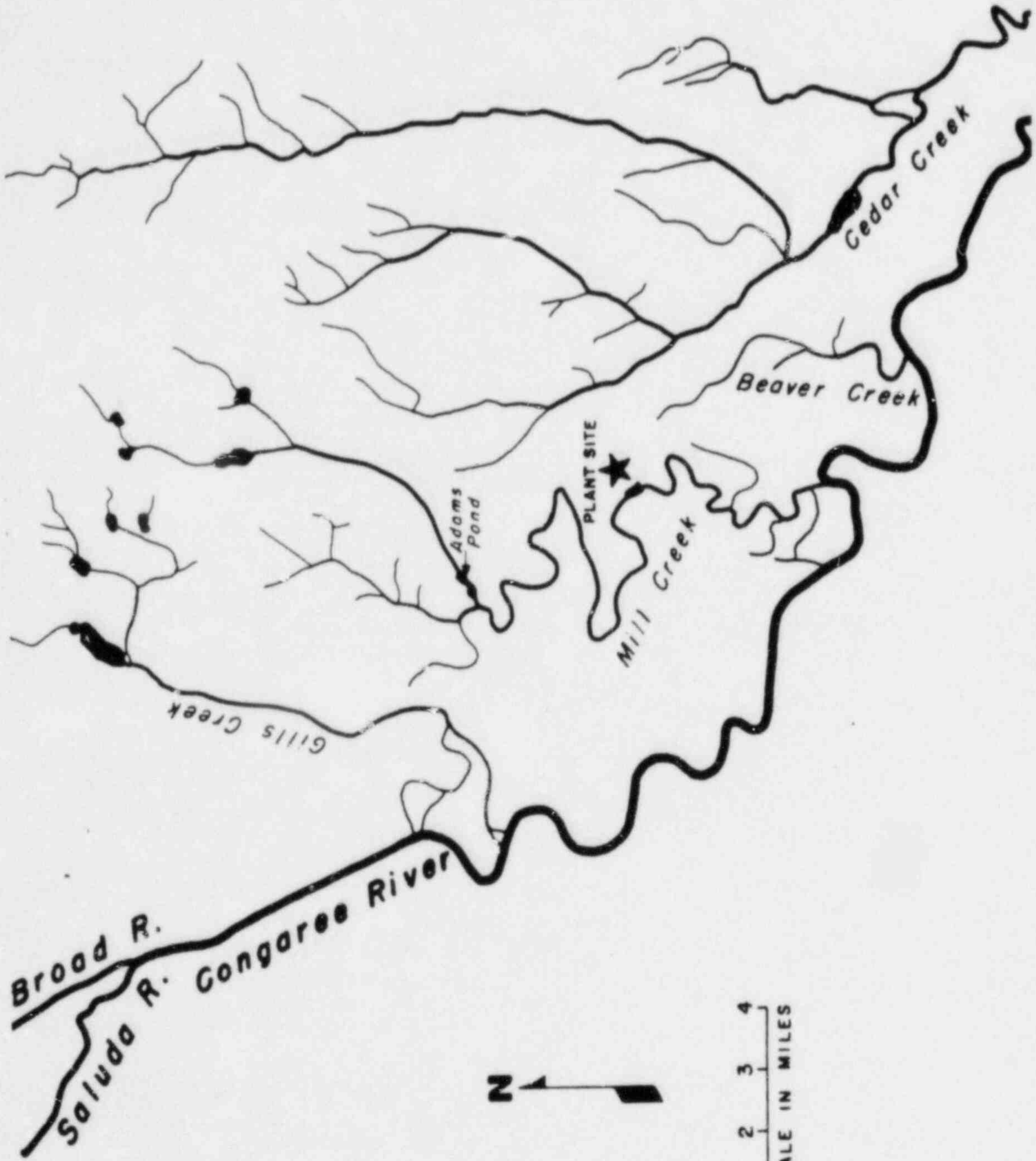


Figure 3.4. Surface Water Locations Near Columbia Plant.

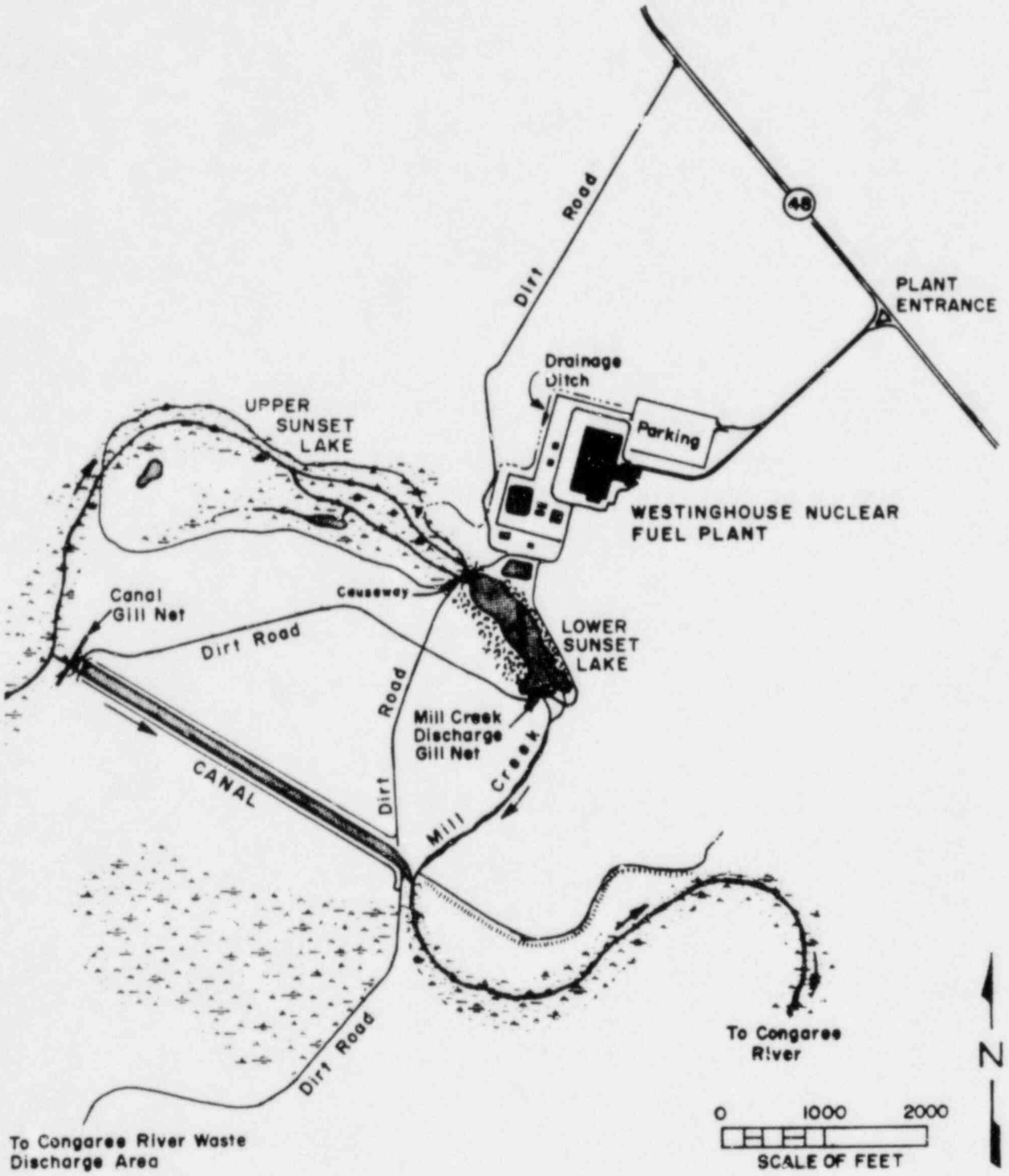


Figure 3.5 Sunset Lake in the Area of the Plant.

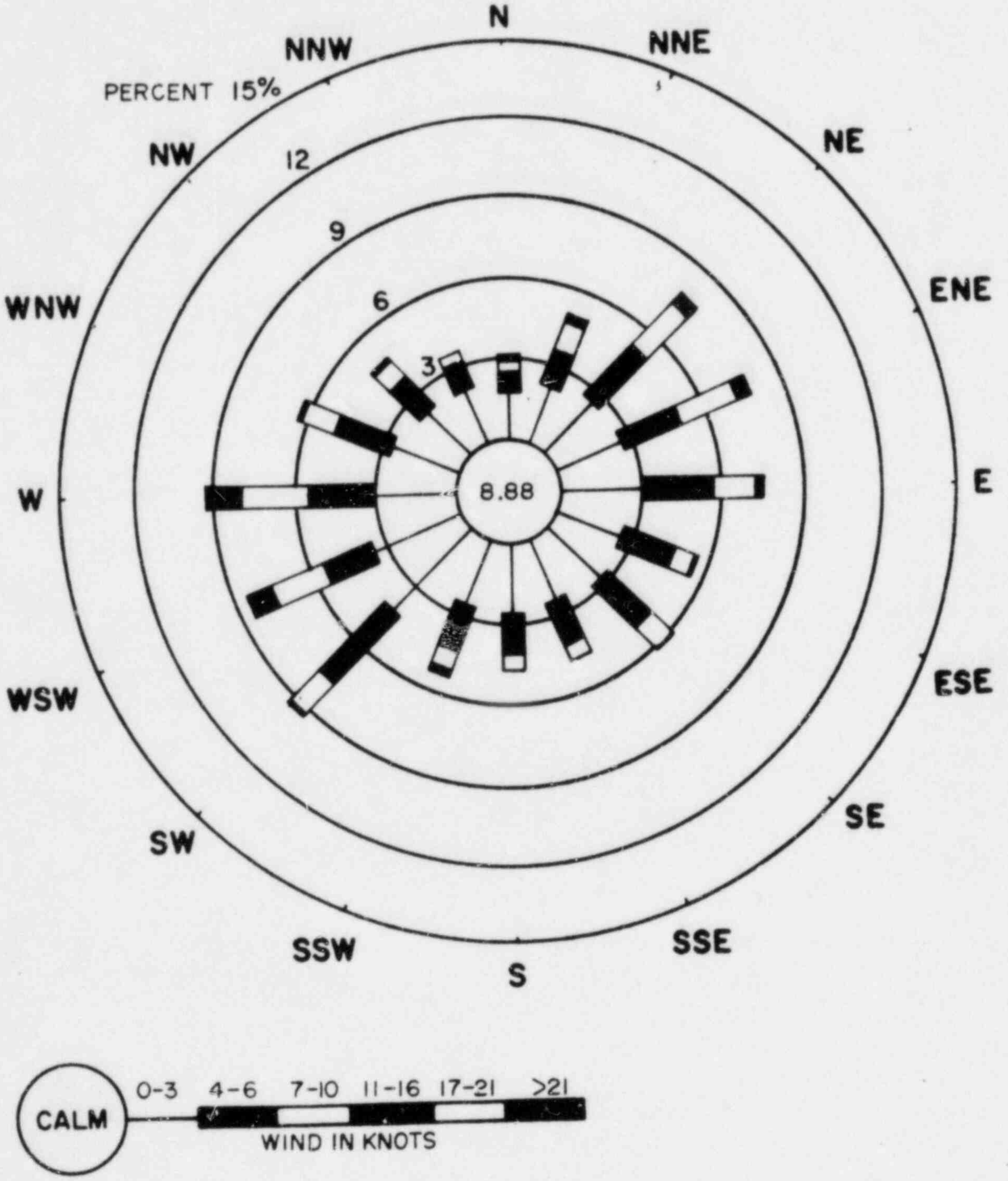


Figure 3.6. Annual Wind Rose Drawn From Site Specific Data (8-72 through 7-73).

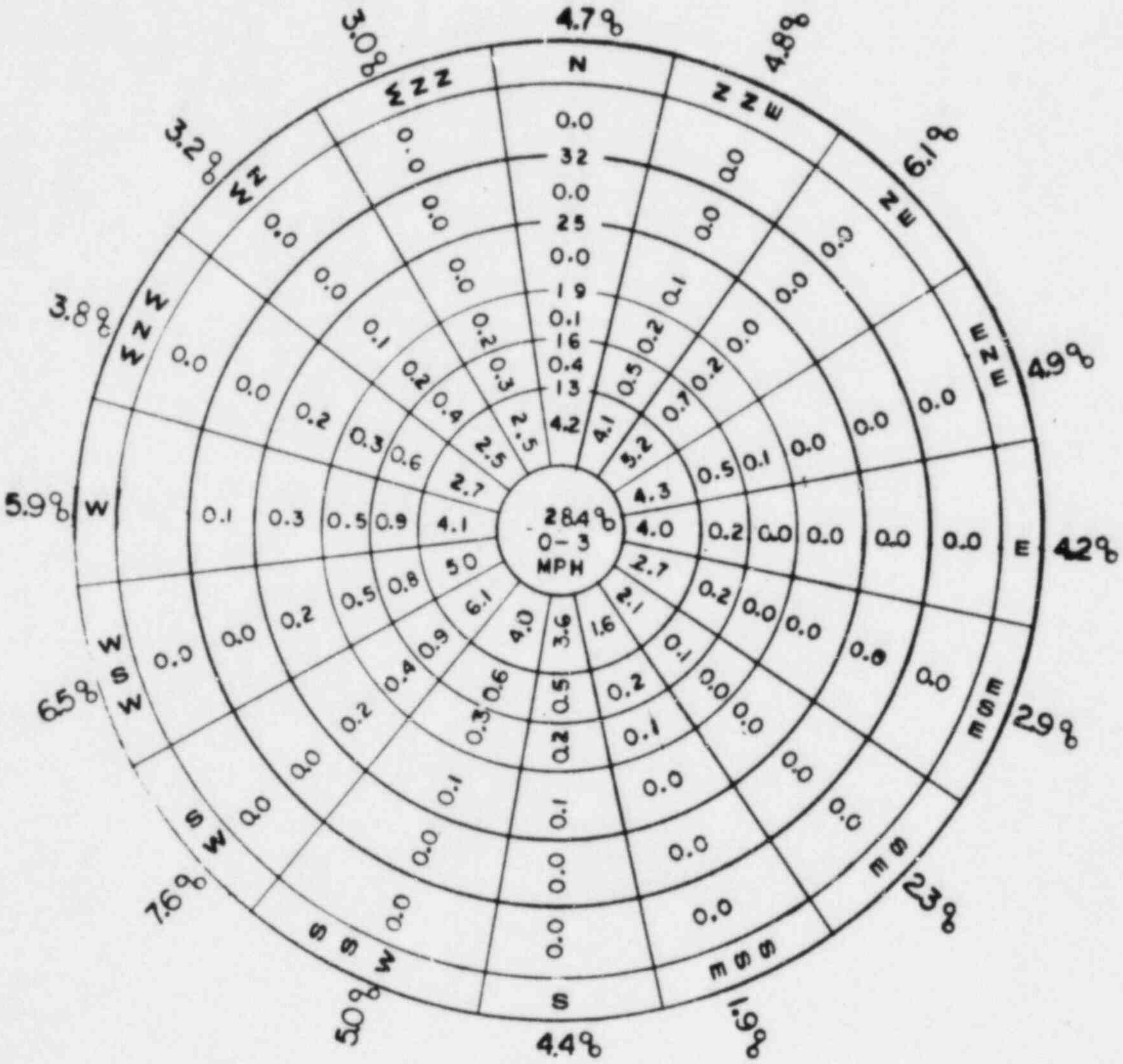
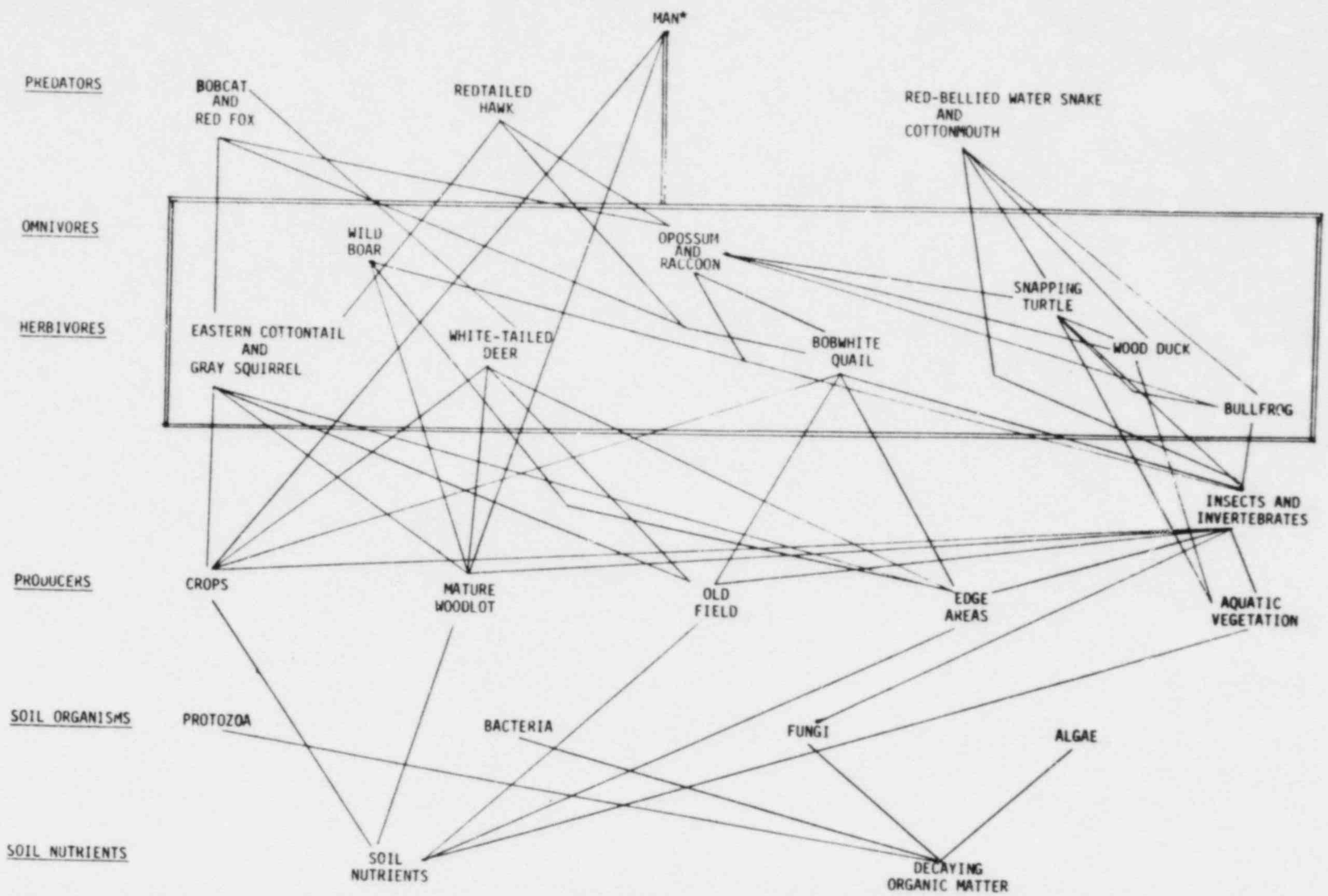


Figure 3.7 Average Annual Wind Rose at Columbia Airport 1948-1981 (NOAADATA)



*All animals within double line may be consumed by man.

Figure 3.8. Generalized Food Web of W Columbia Site.

SECTION 4
DESCRIPTION OF WASTE STREAMS AND EFFLUENT CONTROL

4.1 LIQUID WASTE TREAEMENT

4.1.1 RADIOLOGICAL CONTROL

Compliance with 10 CFR 20 discharge activity limits regarding discharge of liquid waste to the unrestricted area is assured by a continuous on-line gamma ray spectroscopy system within the main plant's controlled access area. Quarantine tanks and diversion tanks are available to increase settling times and allow sufficient filtration if the liquid activity is above release limits. When the liquid has been successfully scanned and approved for discharge, the liquid will be pumped from the final pump out tank to the advanced wastewater treatment facility for uranium removal external to the main plant. This polishing operation will assure that the last remnant of uranium is removed from the liquid stream and recycled through scrap recovery. The liquid stream is then discharged to the chemical waste treatment removal system.

4.1.2 CHEMICAL WASTE TREATMENT CONTROL

An NPDES/EPA liquid waste discharge permit was originally effective in April 1974, implementing compliance with the Federal Water Pollution Control Act. Compliance was effective in phases striving to meet the goal of zero discharge. During the final phase negotiations were initiated to modify the permit. A consultant was employed to recommend processes and suggest achievable limits. A modified permit was issued in 1978. Authority for program administration was transferred from the EPA to the South Carolina Department of Health and Environmental Control. A renewal modification of the permit was made effective February 1, 1983, implementing the condition of best available technology.

Since the 1978 modification, Westinghouse has routinely complied with all NPDES discharge limitation parameters. The major liquid waste generated in converting uranium hexafluoride into uranium dioxide is ammonium fluoride. The basic chemical reaction involved in the treatment of process waste is:

$$2NH_4F + Ca(OH)_2 \longrightarrow 2NH_4OH + CaF_2$$

Slaked lime simultaneously supplies the necessary pH adjustment required for ammonia distillation and fluoride removal capability. The slurry is then pumped to a 15 sieve tray distillation column to strip the free ammonia from the waste stream for recycle in the ADU process. Sodium hydroxide is used for supplemental pH adjustment to assure all the bound ammonia is converted to free ammonia in accordance with the following equation:

$$NH_4 + OH^- \longrightarrow NH_3 + H_2O$$

The recycled ammonium hydroxide (30%) is stored for reuse in the ADU conversion process. On line specific ion electrodes and pH monitors allow the operator to control the process and meet discharge requirements.

After the ammonia is removed the still bottoms containing CaF_2 waste are further treated by pumping them to a lagoon for solids removal and fluoride removal using quiescent settling. Following settling, the clarified liquid is transferred to either the north or south lagoon for discharge via the plant outfall. Chemical analyses of the lagoons dictate the allowable daily discharge rates. The process waste is mixed with the sanitary waste and other miscellaneous streams prior to discharge via a single 4" line to Congaree River. A monitoring system has been installed to continuously sample this stream and assure permit limitations are complied with. Daily samples are analyzed for chemical constituents and radiological components.

4.1.3 SANITARY WASTE SYSTEM

To meet the intentions of PL-92-500 secondary treatment criteria, including an 85% reduction in BOD_5 and a maximum 30 mg/l total suspended solids, the sanitary treatment system was modified in December 1975 with the installation of a 75,000 gallon Clow extended aeration package plant. The 1,500,00 gallon sanitary lagoon is used as a polishing pond. Piping has been installed to accomodate discharge directly from the package plant should a problem develop in the sanitary lagoon. Initially dry chlorination was used to insure adequate disinfection. Recently a gaseous chlorine injection system was installed to facilitate addition and insure compliance with the fecal coliform parameter.

All domestic type wastes, shower water, cafeteria water and several miscellaneous streams are routed to the sanitary system. Contaminated laundry cleaning is performed outside the Columbia plant by an approved vendor, and no routine laundry wastes are encountered.

4.2 INDUSTRIAL WASTES

At design capacity, water will be obtained from the Columbia Municipal Water System at a rate of approximately 10,600,000 gal/month. This water will be used for potable and process cooling requirements. About 45% of the incoming water will be returned to the atmosphere in the form of water vapor from the lagoon ponds and the cooling towers. The balance of water, approximately 5,700,000 gal/month, will be discharged to the Congaree River in the form of process and sanitary-treated water, 2,850,000 gal for each stream. A 4-in. -diam pipeline transfers these wastes to the Congaree River about 3.5 miles south of the facility. The pipe submerges into the river and discharges directly into the river current, near the bottom and about 20 ft from the shoreline.

4.3 WASTE CONFINEMENT AND EFFLUENT CONTROL

Effluents from the various processes occur in three forms: gaseous, liquid, and solid. The effluents may contain small quantities of the radioisotopes U-234, U-235, U-236, and U-238. The composition of the mixture will vary depending upon the enrichment of the material being processed; however, in all cases, the bulk of the material will be U-238 (95% by weight or more), whereas the predominant activity will be U-234 (up to 86% of the total activity). For this study, the reference mixture given in Table 4.2 will be used.

TABLE 4.1
AVERAGE AIRBORNE URANIUM RELEASES

Period Ending	uCi Discharged
12-31-79	1008
06-30-80	537
12-31-80	659
06-30-81	462
12-31-81	485
06-30-82	502

Average Discharge/week = 23.4 uCi

TABLE 4.2
REFERENCE ISOTOPIC MIXTURE

Isotope	Mass fraction (%)	Activity fraction (%)	Activity (uCi/g)
U-235	0.04	85.38	2.48
U-235	4.15	3.07	0.09
U-236	0.025	0.55	0.016
U-238	95.78	11.00	0.32

The process effluent consist primarily of ammonia (NH_3) and fluorides such as NH_4F, CaF_2, and a very small amount of gaseous HF.

4.3.1 GASEOUS EFFLUENTS

4.3.1.1 VENTILATION SYSTEMS - RADIOACTIVE MATERIAL AREAS

Operations involving the use of radioactive materials in unsealed physical forms are limited to low-enriched (5% U-235) uranium in the fuel manufacturing facilities or in the associated analytical laboratory. The high efficiency ventilation systems installed in these facilities are designed so that all air from zones used to handle or process uranium is treated to remove essentially all uranium prior to its release to the atmosphere. At the design capacity of 1600 MTU/year, the maximum air flow from the plant is estimated not to exceed 160,000 cfm.

The normal gaseous radioactive effluent from the plant operating at 1600 MTU has been estimated to be about three times that of the present 700-MTU operation. These effluents are released from a number of short stacks and roof vents, as shown in Fig. 4.1. Process gas scrubbers are listed in Table 4.4.

4.3.1.2 NONRADIOACTIVE PROCESS GASES

Process gases are also vented through several short stacks and vents on the roof of the manufacturing building, as shown in Fig. 4.1. Prior to discharge to the atmosphere, process gases that may affect the environment are scrubbed for chemical removal.

Based upon an extrapolation of data during 700-MTU/year operation, the average and maximum release rates for ammonia and fluorides during normal operation at 1600 MTU have been estimated. They are listed in Table 4.3.

TABLE 4.3
AVERAGE AND MAXIMUM EMISSION RATES (g/sec) OF PROCESS GASES

	Average at		Maximum at	
Chemical	700 MTU	Est. 1600 MTU	700 MTU	Est. 1600 MTU
Ammonia (NH_3)	1.8	4.94	2.3	6.2
Fluorides (F^-)	0.00015	0.0004	0.0003	0.0008

The IDR process will not require the use of ammonia. Maximum ammonia levels may be somewhat lower than those estimated at 1600 MTU.

4.3.1.3 MONITORING PROCEDURES

Each release stack monitored is equipped with an isokinetic probe device that continuously draws a sample through a fiberglass filter paper. The filter paper is then removed periodically and analyzed for uranium. The past analysis of air concentrations and flow rates have been utilized to give the total release rate at the present capacity. These calculations were then extrapolated to estimate the releases at the projected capacity of 1600 MTU. Waste gases from chemical processing are also periodically analyzed for ammonia and fluorides. The average and maximum ammonia and fluoride gaseous effluent releases for the present operating load of 700 MTU have been utilized to estimate the values to be expected at the projected capacity of 1600 MTU as indicated in Table 4.3.

4.3.2 LIQUID EFFLUENTS

Liquid wastes consist of two components: sanitary wastewater generated by plant employes and industrial wastewater generated by the manufacturing process. Process waste which may contain uranium, are processed through quarantine holding tanks and circulated through cartridge filters. The fluoride-containing wastes are treated with lime to form a slurry of CaF_2, which is then distilled to remove the ammonia for reuse. The slurry is then discharged to the west lagoon for settling of the solids.

In addition to the isotopes of uranium, the liquid-waste streams contain small amounts of the daughter products Th-231, Th-234, and Pa-234m. These radionuclides account for the presence of beta-gamma activity in the liquid-waste stream.

The average discharge concentrations and the total annual release of radioactivity to the river for the present 700-MTU operation, in addition to estimated values for the projected 1600-MTU operation, are given in Table 4.6.

The total annual flow rates are 47 million gallons for the 700-MTU capacity and are estimated to be 69 million gallons for the 1600-MTU-capacity plant. A schematic diagram of the waste system is shown in Fig. 4.2. The components are identified in Table 4.5.

4.3.2.1 RADIOACTIVE LIQUID EFFLUENTS

The raw waste streams are monitored for radioactivity before leaving the plant conversion area. If the uranium concentration exceeds a specified level, the stream is diverted for additional processing. Liquid wastes from the line process scrubbers, the scrap-recovery, and other process streams are stored in quarantine tanks on a batch basis and then sampled before release. Only uranium concentrations in liquid-waste streams below the specified level of 30 pCi/ml (the MPC for U-234 given in 10 CFR Part 20) are permitted to leave the plant area. This waste is then processed through the Advanced Wastewater Treatment Process to remove uranium to levels less than 1 ppm.

4.3.2.2 NONRADIOACTIVE LIQUID EFFLUENTS

After precipitation of the fluorides with lime and the removal of the ammonia in the stripping still, the treated effluent is discharged to the west lagoon to permit settling of the solids. The liquid is decanted from the top on a batch basis to the north and south lagoon where additional settling takes place. After a one-to three-day settling time, the supernate is pumped to the Congaree River, usually together with overflow from the sanitary stabilization pond. The average annual total quantities discharged at the 700-MTU/year level are given in Table 4.7.

TABLE 4.4
PROCESS GAS SCRUBBERS

Scrubber	Quantity	Location	Type	Efficiency Chemical %	Efficiency Particulate (wt %)
S-2A S-2B	2	Plant air effluent	High-energy venturi cyclone	70-85 (NH_3, HF)	90
S-3	1	Vessel vent header	Packed tower	90 (NH_3, HF)	
S-1	2	Scrap Recovery	Venturi	90 (NH_4F)	
Calciner reaction gas effluent	10	Calciners	Venturi	75-80 (NH_3) 75-90 (HF)	90
Incinerator effluent	1	Incinerator	Packed tower	90 (Acids)	95
Scrap Recovery (S-1056)	1	Scrap Recovery	Spray Tower	95%	90

TABLE 4.5
IDENTIFICATION OF LIQUID WASTE TRANSFER LINES FOR THE BUILDING AND WASTE TREATMENT SYSTEM

Line No	
1	Process waste Water UO_2 NH_4 NH_4OH
2	Groundwater
3	Roof drain to storm sewer
4	Sanitary drain
5	Closed-cooling-tower loop overflow and blowdown of system to storm sewer (Note: Line No. 5 stream deleted because of EPA requirements and now flows to line No. 4) Water Nalco 39 Nalpac 8241 Nalco 918 Nalpac 8240 Nalco 7313
6	Contaminated waste Water UO_2 powder
7	Roof drain to storm sewer
8	Sanitary drain Has boiler blowdown Nalco 19 Nalco 711 Nalco 356 Nalco 752 Has deionizer flush and backwash Sulfuric acid (H_2SO_4) Caustic NaOH Serves as waste drain for x-ray film development chemicals Gavaert fixer and developer is used in automatic developing equipment Contains wastes from cafeteria and kitchen
9	Process waste (treated) Water NH_4F, NH_4NO_3 NH_4OH, CaF_2,CaOH
10	Storm sewer groundwater
11	Treated unclassified waste

TABLE 4.5 (continued)
IDENTIFICAION OF TRANSFER LINES FOR THE BUILDING
AND WASTE TREATMENT SYSTEM

Line No	
12	Deonized water regeneration H_2SO_4, NaOH Lab wastes
13	Test water, cylinder recertification
14	Waste storage pad drain

TABLE 4.6
DISCHARGE CONCENTRATIONS AND TOTAL ANNUAL RELEASE
OF RADIOACTIVITY IN NFCS LIQUID WASTES

	At 700 MTU/year			Estimated At 1600 MTU/year		
Radiation Component	Concen-tration (pCi/ml)	Percent-age of MPC	Total Release rate (mCi/yr)	Concen-tration (pCi/ml)	Percent-age of MPC	Total Release rate (mCi/yr)
Alpha	0.64	2.2	97	1.77	5.9	262
Beta	0.305	0.9	45.2	0.823	2.4	122

[a]Based on 10 CFR 20 limits for U-234
[b]Based on calculated 10 CFR 20 limits for combined daughter products Th-234, Pu234, Th-231 = 3.3×10^{-5}

4.3.2.3 MONITORING PROCEDURES

A continuous recorder measures the flow and pH of waste discharged to the river. Twenty-four-hour composite samples are obtained and measured on a daily basis for ammonia and fluoride. Also, 24-hr composite samples are obtained and measured on a weekly basis for BOD_5, and for total suspended solids. Grab samples are taken from the final aeration tank once a week and once a month, respectively, and analyzed for grease and oil and fecal coliform. A monthly composite sample is also analyzed for gross alpha, beta, and gamma activity.

4.3.3 SOLID WASTES

4.3.3.1 MANUFACTURING WASTE

Materials such as used packaging, worn-out clothing, paper, wood-floor sweepings, discarded tools, etc., are collected and stored prior to disposal, which is made according to two primary classifications: uranium contaminated or contamination free. The contaminated material is further segregated into combustible and noncombustible classifications. Noncombustible waste is examined to determine the feasibility of recovery, and is then either processed chemically or collected in boxes for ultimate disposal at a Government-licensed waste disposal site. Combustible items are reduced to ash in a specially designed incinerator, and the ash is dissolved in a mixer-settler dissolver system. Solvent extraction will recover and purify the uranium for recycle back to the product material stream.

The incinerator off-gas is scrubbed and filtered through high efficiency filters. The activity in the stack averages 4×10^{-13} uCi/ml of uranium. The ash is sampled, analyzed for uranium, and either reprocessed or disposed of by burial.

Approximately 40 to 60 bales of combustible contaminated waste and 4 bales of noncombustible waste will be generated per day at the 1600-MTU/year level.

4.3.3.2 CALCIUM FLUORIDE

Following fixation with a cement like binder the calcium fluoride, contaminated with uranium, was buried at the low level radioactive waste burial site in Barnwell, South Carolina. All of the calcium fluoride generated prior to 1981 or approximately 575,000 ft^3 of material was handled in this manner. An advanced wastewater treatment system was installed in 1980 to remove the last remnant of uranium and avoid contaminating the calcium fluoride. Future calcium fluoride should contain less than 30 pCi/gram of uranium activity and constitute de minimus material for radioactivity classification purposes, which will permit disposal of this material in a chemical or sanitary landfill.

TABLE 4.7

AVERAGE WATER QUALITY FOR LIQUID EFFLUENT DISCHARGE
AT 700 MTU/YEAR OPERATION

Parameter	Concentration (mg/l)	Quantity (lbs/day)
pH (units)	8.6	
BOD_5	18.9	16.2
Fecal Coliform (MPN/100 ml)	50	
TSS	23.3	20.0
COD	89	76.4
Oil & Grease	3.5	3.0
Phenol (ug/l)	< 1	< 0.001
Surfactants	0.17	0.15
Nitrate	160	137
Sulfate	140	120
Sulfide	< 0.05	0.04
Ammonia (N)	17.5	15.0
Phosphorus	2.3	2.0
Cyanide	< 0.02	< 0.02
Fluoride	17.4	14.9
Barium	0.10	0.09
Iron	5.0	4.3
Manganese	0.04	0.03
Magnesium	6.1	5.2
Zinc	2.2	1.9
Aluminum	0.41	.35
Cobalt	< 0.01	< 0.01
Molybdenum	0.04	0.03
Sodium	194	166
Boron	0.267	0.23
Bromide	< 0.1	< 0.09

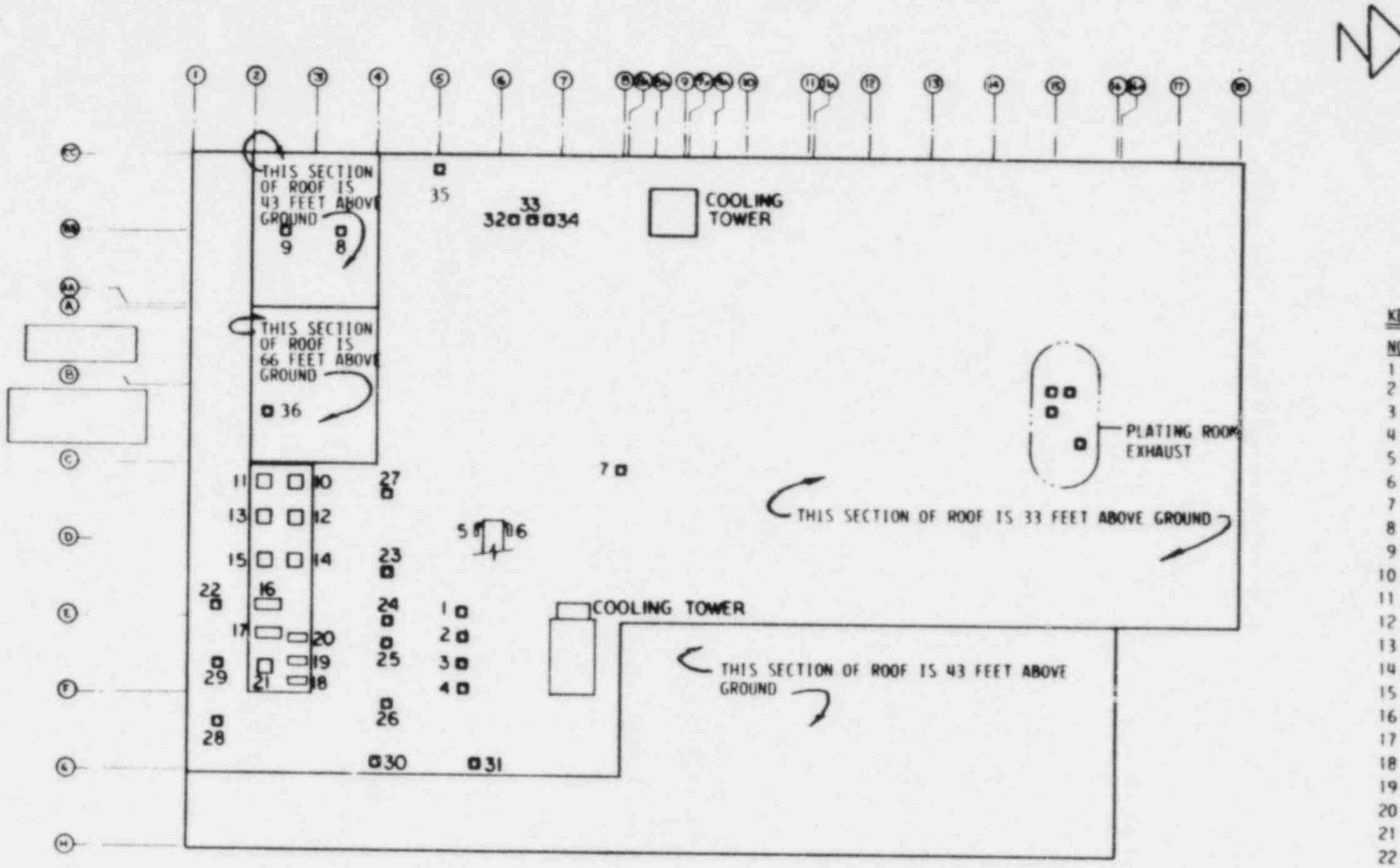


Figure 4.1. Process Stack Discharge Locations From Top View of Roof

KEY:

NO.		STATION DESCRIPTION
1	-	FURNACE EX. LINE 1
2	-	FURNACE EX. LINE 2
3	-	FURNACE EX. LINE 3
4	-	FURNACE EX. LINE 4
5	-	FURNACE EX. LINE 5-S
6	-	FURNACE EX. LINE 5-N
7	-	MET. LAB EXHAUST
8	-	INCINERATOR EX.
9	-	SUPPLEMENTAL INC. EX.
10	-	CONVERSION 1-A SYSTEM EX.
11	-	CONVERSION 1-B SYSTEM EX.
12	-	SCRAP RECOVERY 2-A SYSTEM EX.
13	-	SCRAP RECOVERY 2-B SYSTEM EX.
14	-	ADU LINE SCRUBBER 3A
15	-	ADU LINE SCRUBBER 3B
16	-	MAINT VENT SYSTEM 4-A
17	-	MAINT VENT SYSTEM 4-B
18	-	CONVERSION ENCLOSURE EX. #4
19	-	CONVERSION ENCLOSURE EX. #3
20	-	CONVERSION ENCLOSURE EX. #2
21	-	CHEM. LAB FILTERED EX.
22	-	DECON ROOM EX.
23	-	CALCINER COMBUST GAS LINE 1
24	-	CALCINER COMBUST GAS LINE 2
25	-	CALCINER COMBUST GAS LINE 3
26	-	CALCINER COMBUST GAS LINE 4
27	-	CALCINER COMBUST GAS LINE 5
28	-	AIR COMPRESSOR ROOM EX.
29	-	BOILER ROOM EX.
30	-	CHEM. LAB EX. #2
31	-	H.P. LAB EX.
32	-	DEVELOPMENT LAB EX. 1
33	-	DEVELOPMENT LAB EX. 2
34	-	DEVELOPMENT LAB EX. 3
35	-	SOLVENT EXTRACTION AREA EX.
36	-	PROPOSED IDR EX.

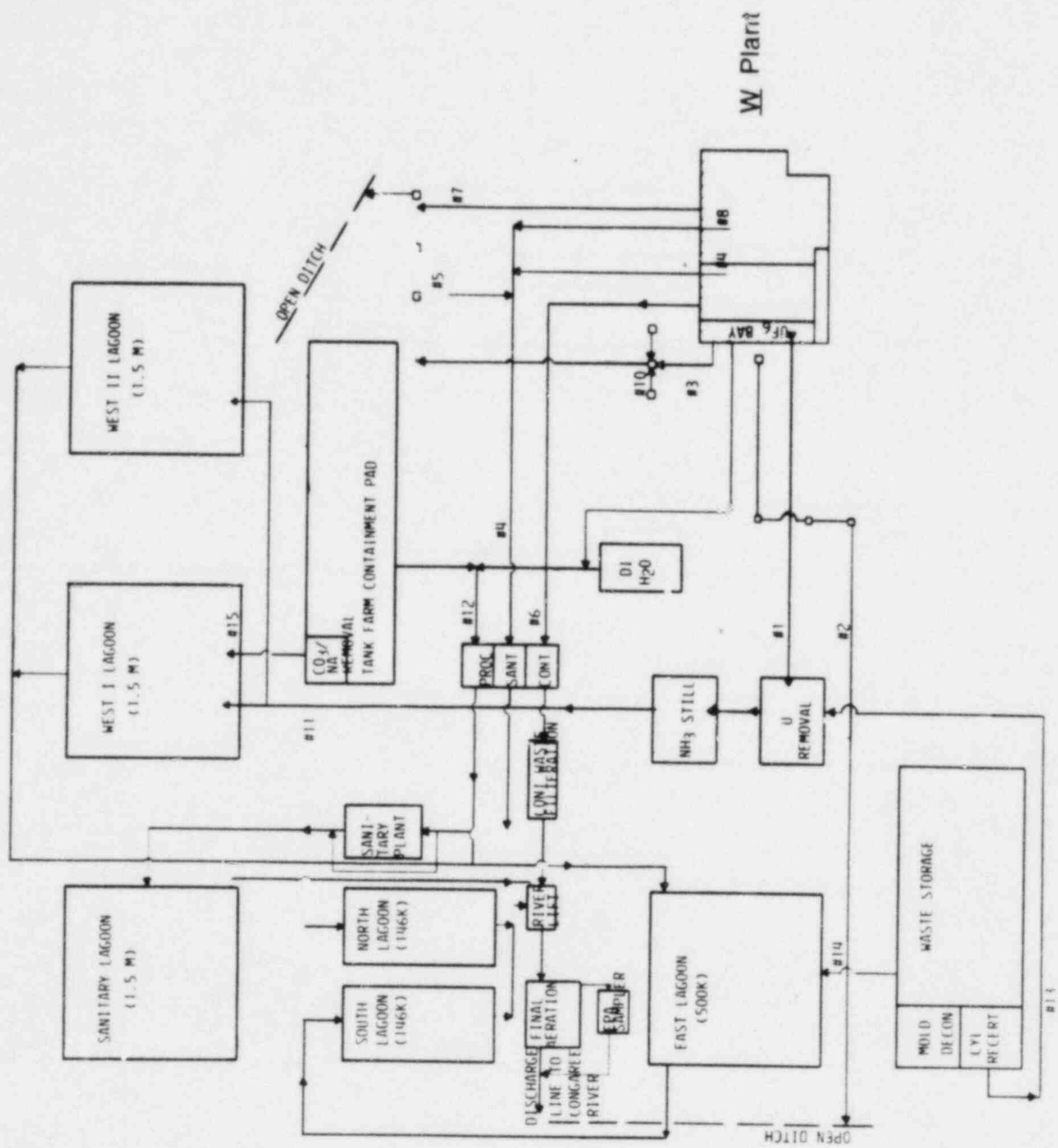


Figure 4.2 Building and Liquid Waste Treatment Flow Sheet

SECTION 5

ENVIRONMENTAL IMPACT OF FACILITY OPERATIONS

5.1 RADIOLOGICAL IMPACTS

The radiological impacts of the NFCS plant were assessed by calculating the maximum dose to the individual living at the site boundary closest to the highest concentration of airborne radionuclides released from a 1600-MTU/year plant. In addition, the dose to an individual from drinking water and eating fish from the nearest water supply was also assessed. The doses are actually 50-year dose commitments, that is, the total dose to a reference organ, resulting from one year of intake, that will accrue during the remaining lifetime (50 years) of the individual. Site-specific data were used where available.

5.1.1 TERRESTRIAL

5.1.1.1 INDIVIDUAL DOSE

The doses from airborne radioactive effluents released from the plant stack are calculated for an individual living at the property line, 1800 ft north-northwest of the plant, where the maximum offsite ground-level concentration of radionuclides occurs. The average annual x/Q value at this location was estimated to be 1.63×10^{-5} sec/m^3. It is assumed that the individual spends all of his time at this location and that all of the food consumed is produced at the point of reference.

The highest dose received, 1.1 millirems/year, was to the bone. The total-body dose (0.07 millirem/year) and doses to the kidney, GI tract, and lungs were all less than 1 millirem/year. These doses are well below the allowable limit of 25 millirems/year to the total body, 75 millirems to the thyroid, and 25 millirems to any other organ. The annual average total-body dose from natural radiation in the State of South Carolina is about 135 millirems/year and may be compared with the maximum total-body dose of 0.07 millirem/year resulting from the plant airborne releases.

5.1.1.2 POPULATION DOSE

The annual total-body dose to the population within 50 miles of the plant (see Sect. 2.2) is 2.4×10^{-2} man-rem and is due primarily to U-234 (46%) and U-235 (39%). The population total-body dose may be compared with the similar population dose resulting from the area natural background of 8.75×10^4 man-rems. The average total-body dose resulting from the plant airborne effluents to the 6.48×10^5 persons within a 50-mile radius is only 32.7×10^{-5} millirem/year.

The highest annual population organ doses are to the bone, 1.8×10^{-1}, man-rem, due primarily to U-234 (77%); lungs, 1.1×10^{-1} man-rem, due primarily to U-234 (79%); and kidney, 4.7×10^{-2} man-rem, due primarily to U-234 (72%) and U-235 (16%). The dose to all other organs is less than that of the total body.

5.1.2 AQUATIC

The doses resulting from the release of liquid effluents into the Congaree River are based on the estimated measurements in the river for a projected 1600-MTU/year plant capacity. The bone received the highest combined dose from eating fish and drinking water (1.0 millirem/year), based on an intake of 50 g/day and 1.2 liters/day respectively. The dose to the total body from aquatic sources was only 0.06 millirem/year and, similarly, doses to the kidney and GI tract were less than 1 millirem/year. These annual doses are well below the allowable limits of 25 millirems/year to the total body, 75 millirems to the thyroid, and 25 millirems to any other organ.

5.2 NONRADIOLOGICAL IMPACTS

5.2.1 TERRESTRIAL IMPACTS

The primary potential impact of the Columbia plant operation on terrestrial biotic communities will result from the atmospheric effluents emitted from the plant. The only measurable effluents are ammonia and chemical. Under certain conditions, the concentrations and duration of ammonia and fluoride exposure can be toxic to terrestrial biota.

5.2.1.1 IMPACT ON PLANTS

All plants require a source of nitrogen, especially for protein synthesis. Ammonia is a form of already-fixed nitrogen that can be used by plants. However, very high concentrations of ammonia in the atmosphere can cause collapse of foliar tissue without loss of chlorophyll. Bioassay tests (5-min exposures) of 10 plant species indicated that a concentration of 1000 ug/m^3 of atmospheric ammonia decreased photosynthesis by about 10%. Offsite vegetation would not undergo ammonia concentrations great enough (1000 ug/m^3) to reduce photosynthesis even under extended Pasquill atype F or G stability conditions.

At the current 700 MTU/year production, no vegetation damage attributable to atmospheric ammonia has been observed. Because ammonia and other nitrogen compounds are common products of microbial decomposition of dead plant debris, ammonia accumulation in soils from the NFCS is not likely to be measurable.

The processes and problems involved with atmospheric fluoride concentrations are more complex than those concerning ammonia. Fluorides are highly soluble in biological systems, where they may accumulate to toxic levels. The effects of atmospheric fluorides on plants are generally induced through fluoride absorption by leaves rather than through absorption by roots. Although root uptake of fluorides can be important in acid soils, even that is a little consequence in toxicity.

The effect of atmospheric fluoride absorbed by leaves varies with the continuity of the exposure. Plants appear to respond least to intermittent fluoride exposure, especially when nonexposure periods cover single or multiple days. Intermittent exposure is more characteristic

of areas that surround a point source of atmospheric fluoride such as the NFCS. Shifts in wind direction may reduce otherwise toxic concentrations to harmless levels at a given locale several times a day.

Low fluoride levels in leaf tissue may merely reduce photosynthesis, thereby reducing growth. They may also cause death of plants, which symptomatically appears first as chlorosis (loss of chlorophyll) then as necrosis (death) of leaves in broad-leafed species or tip-burn of conifer needles.

McCune has synthesized the literature on plant responses to atmospheric fluoride. In all plants, the cumulative nature of the fluoride damage is apparent from the gradual reduction in fluoride concentrations necessary to induce toxicity with increased time of exposure. None of the experiments on which the work is based exceeded 100 days. Therefore, actual long-term toxic values may be lower, especially for perennials. Among perennials, conifers in particular should be affected because they carry their needles for several years. On the other hand, most of McCune's data are based on continual exposure to fluorides rather than to the intermittent exposure expected to occur at the NFCS. Boundary fluoride concentration values from the NFCS (Table 5.1) and suggested air quality criteria for fluorides are given in Table 5.2.

TABLE 5.1
AMBIENT GROUND LEVEL CONCENTRATIONS OF AMMONIA AND FLUROIDE
AT 700 MTU/YR PRODUCTION AND ESTIMATED FOR 1600 MTU/YR PRODUCTION
AT THE NEAREST SITE BOUNDARY (1800 FT NNW, 0.3 MILE)

	700 MTU/YR		1600 MTU/YR	
	Annual Average ug/m^3	Annual Maximum ug/m^3	Annual Average ug/m^3	Annual Maximum ug/m^3
Ammonia	64	80	145	180
Fluoride	0.005	0.010	0.014	0.027

TABLE 5.2
AMBIENT ATMOSPHERIC FLUORIDE LEVELS (AT THE PLANT BOUNDARY EXPECTED FROM WESTINGHOUSE NFD THE COLUMBIA PLANT (NFCS) OPERATION AT 1600 MTU/YEAR), AIR QUALITY STANDARDS, AND LEVELS AT WHICH SYMPTOMS APPEAR AND TOXIC CONDITIONS PREVAIL AMONG SELECTED PLANTS[a]

	Exposure Period	Average concentration of (ug/m^3)	Maximum concentration of (ug/m^3)
NFCS fluoride concentrations at plant boundary	Yearly	0.014	0.027
Washington state HF ambient air standard[b]	Yearly	0.500	
	Monthly	0.800	
	Exposure Period (days)	**Appearance of symptoms (ug/m^3)**	**Toxicity (ug/m^3)**
Fluoride concentrations			
Tree fruits	70	1.5	4.0
Corn	10	1.5	2.0
Alfalfa	12	1.0	100.0
Larch	5		1.0
Western conifers	8		1.0
one- to two-month-old pine needles	10		0.6

[a]Toxic levels are those at which reduced productivity or yield have been measured.

[b]Washington maintains the most restrictive fluoride air quality standards; South Carolina has none.

Source: D. C. McCune, "On the Establishment of Air Quality Criteria, with Reference to the Effects of Atmospheric Fluorine on Vegetation", American Petroleum Institute, Air Qual. Monogr. 69(3) (1969).

From the calculated levels of fluoride concentrations (0.014 to 0.27 ug/m^3) expected at the NFCS boundary at 1600-MTU/year operation and from the toxic levels of atmospheric fluorine shown for various plant species in Table 5.2, little, if any, plant biological damage can be expected near the NFCS.

The most important natural biotic community is the Congaree Swamp National Forest. It is unlikely that fluoride emissions from the NFCS would reach the swamp 4-1/2 miles distant.

The loblolly pines (Pinus taeda) and other pine species near the NFCS would also be unlikely to undergo fluoride damage as would those on the plant site. However, if damage were to occur and were great enough to actually kill trees, some changes in forest composition would follow.

Effects of fluoride concentrations on agricultural activities could also occur. The pecan orchards near the eastern border of the study area are probably too far from the NFCS to be affected. Examination of cultivated and native species within 3 miles of the NFCS, by WESD field ecologists, in 1974 indicated no fluoride damage after five years plant operation.

5.2.1.2 IMPACT ON ANIMALS (INCLUDING MAN)

There is generally no direct effect of ammonia on animals or humans at the concentrations calculated for the NFCS boundary conditions (145 to 160 ug/m^3). The lowest ammonia concentrations found to affect experimental animals was 2000 ug/M^3; ammonia concentrations two orders of magnitude greater are generally required to produce measurable damage.[11] The indirect effects of ammonia on humans and animals via food ingestions is also not expected because this compound is a plant nutrient and is used in fertilization.

Atmospheric fluorides cannot be evaluated as simply as ammonia. Direct inhalation of atmospheric fluoride contributes a negligible amount to total fluoride intake by man or animals. Instead, effects are felt through food chains that contain fluorides accumulated by plants being ingested by cattle that are, in turn, ingested by man.

Although soil fluoride concentrations are irrelevant (Sect. 5.2.1.1), the relationship of atmospheric fluoride concentrations (usually in micrograms per cubic meter) to levels of fluoride in plant tissue (usually in parts per million per gram of dry dry weight) is complex. Natural fluoride accumulation rates differ among plant species, and within a species, rates differ among different climatic and edaphic conditions. However, certain important conclusions can be drawn.

First, a direct (if not linear) relationship between atmospheric exposure to fluorides and plant accumulation of fluorides does exist. Second, the relationship can be used to translate one variable into the other. For example, Benedict et al. and Guderian et al. used a 1.1-ug/m^3 contration of fluorine with a 47-day exposure period to test fluoride accumulation in forage plants. Fluoride levels in orchard grass and alfalfa were raised 125 ppm/g in both species. Under exposure periods of two months, plant fluoride concentrations of 40 ppm in these two species would be supplied by a concentration of 0.5 ug of fluoride per cubic meter. Other forage crop fluoride accumulation values are similar. Because maximum concentration values of fluoride at NFCS (Table 4.2) are little different, similar plant tissue fluoride values may be expected at the NFCS boundary.

The concentration of fluoride in plant tissue that can affect cattle are 50 ppm; sheep and pigs, 100 ppm; and chickens, 300 ppm. At plant tissue fluoride concentrations greater than 50 ppm, for example, chronic fluorosis may develop in dairy and beef cattled, producing lameness, loss of appetite with a resulting weight loss, and decreased milk production.

Dietary intake of fluorides by humans is not a problem because contaminated plant and/or animal foods that humans ingest do not constitute a large enough portion of the total human diet.

Review is performed on the level of fluoride in the forage crops since livestock consume the hay harvested from plant site pastureland. The monitoring program will insure no livestock are affected by ingestion of chronic levels of fluoride.

5.2.1.3 SUMMARY OF IMPACTS ON TERRESTRIAL BIOTA

No deleterious effects of ammonia generated at the NFCS have been identified. At the present 700-MTU/year production level or at the projected 1600-MTU year level, it is unlikely that fluoride emissions will damage native plants and livestock off the NFCS property.

Potential damage to onsite needle-leafed trees, with acompanying potential changes in forest composition, and subsequent changes in animal species composition and population densities will not occur at current discharge rates. Forage and hay crops grown on the NFCS property may be effected by fluoride accumulation. A monitoring program is conducted to insure against excessive accumulation of fluoride.

No other significant impact of the NFCS plant operation or expansion upon agricultural or natural biota is foreseen.

5.2.2 AQUATIC IMPACTS

5.2.2.1 IMPACTS OF THE NFCS FACILITY ON WATER USE

The NFCS plant receives its water from the Columbia Municipal Water system. The projected maximum water consumption at the expanded capacity of 1600 MTU/year is 0.5 cfs (10.6 million gallons per month), which represents 1% of the city's water use. The effect of the NFCS on the city's water availability will be negligible.

At the expanded 1600-MTU capacity, the NFCS will return 0.3 cfs of combined process and sanitary wastes to the Congaree River, resulting in a consumptive water use of 0.2 cfs. Since the lowest seven-day average flow of the Congaree River to be expected in a ten-year period is 1590 cfs, water use by the NFCS plant will have a negligible effect on the quantity of water available from the river for downstream use.

5.2.2.2 IMPACTS OF NFCS LIQUID EFFLUENTS

Liquid effluents and the waste treatment system are described in detail in Sect. 5.3 of this appraisal. Liquid wastes consist of two waste streams: sanitary wastewater generated by plant employes and industrial wastewater generated by the chemical process. At the expanded 1600-MTU capacity it is estimated that each waste stream will consist of 0.15 cfs, totaling 0.3 cfs of liquid effluent.

The major wastes generated by the manufacturing processes are fluoride and ammonia. No nickel waste is currently generated, since the addition of an ion exchange removal system and an evaporator unit for nickel

recovery. The treatment of process wastes consists of the precipitation of fluorides with lime and removal of ammonia by distillation. Treated process effluent is then discharged to a settling lagoon for solids removal. The clarified liquid is then pumped to either the north or south lagoon for further settling before being discharged along with sanitary-waste effluents.

The treatment of sanitary wastes consists of an extended aeration package plant and polishing lagoon. The NFCS presently chlorinates the sanitary-waste stream using gaseous chlorine. Residual chlorine concentrations in the sanitary effluent average approximately 2.0 ppm.

5.2.2.3 COMPLIANCE WITH FEDERAL AND STATE WATER QUALITY REGULATIONS

Beginning in April 1974 the Westinghouse plant's liquid waste discharge were made in compliance with the National Pollutant Discharge Elimination System (NPDES) Permit. A four phased permit was issued for eight effluent parameters. Limits of the final phase were adjusted following Westinghouse petition of the EPA requesting revised discharge limits in accordance with best practicable technology. An expert wastewater engineering consultant, Metcalf and Eddy, was retained to determine what actually constituted best available technology and appropriate discharge limits. A new NPDES permit was negotiated under the auspices of the EPA and the South Carolina Department of Health and Control effective April 1978 and at that point regulatory authority was transferred to SC-DHEC. The permit was extended and reissued for a period of five years ending in December 1986. The recent permit implemented the EPA's program of best available technology.

Recent review of regulatory compliance history is given in Table 5.4. From this table it is concluded that Westinghouse meets the intent of the regulations maintaining discharges in abeyance with the daily maximum discharge rates. Over the recent period of two years only eight permit noncompliances have been recorded. All other discharges during this period have met limits listed in Table 5.3. Actual NPDES discharge averages are listed in Table 5.5. Other recommended state water quality standards for Class B waters are listed in Table 5.6.

5.2.2.4 IMPACTS OF LIQUID EFFLUENTS ON THE CONGAREE RIVER

Due to the tremendous dilution of the Westinghouse effluent stream with the Congaree River, it is difficult to postulate any condition beyond the mixing zone where the constituents in the Westinghouse stream would have any significant impact on the water quality of the Congaree River. Typical dilution factors are listed in Table 5.7 for mixing under both low river flow conditions and daily average flow conditions at current production rates and 1600 MTU. Recent Westinghouse radiological and chemical monitoring results of the Congaree River are documented in Table 5.8.

5.2.2.5 IMPACTS OF DISCHARGES TO SUNSET LAKE AND MILL CREEK

No industrial or sanitary waste is discharged to surface water such as Sunset Lake or Mill Creek. Possible contamination of Sunset Lake could result as a result of lagoon failure or leakage. All industrial waste

storage lagoons have been relined in the period 1981-1982 with 36 mil hypalon liners to eliminate such occurrences. All process lagoons now contain treated process waste rather than raw material waste. Raw material storage areas have been appropriately diked to prevent leakage to storm drains which may eventually impact Sunset Lake. If an event did occur, the upper portion of Sunset Lake could be isolated from the lower portion using a flow gate as a dam.

Sunset Lake would offer tremendous dilution volume with its estimated 43,000,000 gallon capacity. Typical flow rates should range between 0.1-0.6 ft^3/sec. Sunset Lake could possibly be influenced by groundwater contamination. Recent review by consultant geologists have determined that current level of chemical contamination present no problems. Table 5.9 discusses results of recent monitoring of Sunset Lake and other surface water.

The road sample, where various storm drain samples combine into one outfall is the only point where detectable concentrations of ammonia and fluoride are noted. This is not an aquatic habitat, and therefore concentrations of the noted magnitude are not of concern.

TABLE 5.3
NPDES PERMIT LIMITS FOR
WESTINGHOUSE ELECTRIC CORPORATION'S DISCHARGE
TO CONGAREE RIVER, LBS/DAY (UNLESS LISTED)[a]

Parameter	Daily Average	Daily Maximum
BOD_5	18.7	40
TSS	32.2	64.4
Fluoride	30	60
Ammonia	35	70
Oil and Grease	10 mg/l	15 mg/l
Fecal Coliform	200 MPN/100 ml	400 MPN/100 ml
pH	6.9 (min)	10.5 (maximum)

[a]Nickel was deleted as a regulated parameter following elimination of routine discharge from the plating room.

TABLE 5.4
WESTINGHOUSE NPDES REGULATORY NONCOMPLIANCE HISTORY
1981 - 1982

Date	Parameter	Amount, lbs/day
1-01-81	Total suspended solids	77.5
1-02-81	Total suspended solids	119.4
1-06-81	Total suspended solids	90.3
4-16-81	Total suspended solids	84.0
5-06-81	BOD_5	71.0
5-16-81	Ammonia	360.0
2-18-82	BOD_5	77.4
11-25-82	BOD_5	50.2

TABLE 5.5
ACTUAL NPDES WESTINGHOUSE DAILY AVERAGE
DISCHARGE RESULTS (1982)

Parameter	Pounds/Day (unless listed)
Ammonia	14.8
Fluoride	14.9
Nickel	0.05
Oil and Grease	3.5 mg/l
BOD_5	14.6
Total Suspended Solids	17.2
Fecal Coliform	51 colonies/100 ml
pH	8.7 units
Flow	87,800 gal/day

TABLE 5.6
QUALITY STANDARDS FOR CLASS B WATERS[a]

Items	Specifications
Fecal Coliform	Not to exceed a log mean of 1000/100 ml based on five consecutive smaples during any 30-day period; nor to exceed 2000/100 ml in more than 20% of the samples examined during such period (not applicable during or following periods of rainfall)
pH	Range between 6.0 and 8.5, except that swamp waters may range from 5.0 to 8.5
Dissolved oxygen	Daily average not less than 5 mg/liter with a low of 4 mg/liter, except that swamp waters may have an average of 4 mg/liter
Phenolic compounds	Not greater than 1 ug/liter unless caused by natural conditions

[a]Waters suitable for domestic supply after complete treatment in accordance with requirements of the South Carolina Department of Health and Environmental Control. Suitable also for propagation of fish, industrial and agricultural uses, and other uses requiring water of lesser quality.

TABLE 5.7
DILUTION FACTORS OF WESTINGHOUSE STREAM
ASSUMING COMPLETE MIXING IN THE CONGAREE RIVER

Production Rate	River flow of 1740 cfs[a]	River flow of of 9326 cfs
At 700 MTU	1.1×10^{-4}	2.1×10^{-5}
At 1600 MTU	1.7×10^{-4}	3.2×10^{-5}

[a]Lowest seven day average for a 10-year period.

Table 5.8 provides a listing of actual Westinghouse data for selected parameters in Congaree River during October 1981 through September 1982. Both the average ammonia and fluoride concentrations are below recommended safe levels at various points upstream and downstream of the Columbia Plant, verifying statements made in first section of 5.2.2.4.

TABLE 5.8
RESULTS OF 12 MONTHS (OCTOBER 1981 TO SEPTEMBER 1982)
OF WNFD MONITORING OF THE CONGAREE RIVER AT FIVE POINTS

	Ammonia NH_3(N) (mg/liter) Av.	Max.	Fluoride (mg/liter) Av.	Max.	pH Av.	Max.	Activity pCi/l Avg.
1. Ten miles above WNFD discharge at Blossom Street Bridge, Columbia, S.C.	0.18	0.35	0.34	0.43	6.8	7.1	≤ 2.3
2. 500 yds. above discharge	0.18	0.35	0.33	0.39	6.8	7.1	≤ 2.3
3. At discharge	0.19	0.37	0.33	0.40	6.8	7.2	≤ 2.3
4. Mouth of Mill Creek, less than one mile below discharge	0.22	0.61	0.33	0.42	6.7	7.2	≤ 2.3
5. 601 Bridge, 25 miles below WNFD discharge point	0.17	0.42	0.31	0.42	6.8	7.2	≤ 2.3
Recommended maximum safe concentration for aquatic biota	0.9[a]		1.5[b]		6.5-8.5[c]		

[a]The recommended safe level is 0.02 mg/liter of undissociated ammonia. At at temperature of 25°C, which would be expected in the Congaree River in summer, and at a pH of 7.6, the maximum recorded pH above, a total ammonia concentration of 0.9 mg/liter would yield a concentration of 0.02 mg/liter of undissociated ammonia. (From Environmental Protectionn Agency, Offices of Water and Hazardous Material, Quality Criteria for Water, Washington, D.C., 1976.)

[b]From Resource Agency of California, State Water Quality Control Board, WAter Quality Critera, 2nd ed., J. E. McKee and H. W. Wolfe (Eds.), 1963.

[c]From Environmental Protection Agency, Offices of Water and Hazardous Material, Quality Criteria for Water, Washington, D.C. 1976.

TABLE 5.9
RESULTS OF MONITORING SITE SURFACE WATER
INCLUDING SUNSET LAKE AND MILL CREEK IN 1982

Station	$NH_3(N)$ Avg.	$NH_3(N)$ Max.	Fluoride Avg.	Fluoride Max.	pH Avg.	pH Max.
Entrance of Mill Creek to Sunset Lake	≤ 1	≤ 1	≤ 1	1.5	6.2	6.7
Road storm drain	≤ 4	22	≤ 4	32	6.8	7.8
Dam causeway between upper and lower lake	< 1	1.3	< 1	3.0	6.4	7.1
Spillway of lower lake into Mill Creek	< 1	1.7	< 1	3.0	6.6	7.1
Exit point where Mill Creek joins diversion canal	< 1	1	< 1	1.4	6.3	6.4

5.2.3 IMPACTS ON GROUNDWATER

Following notation of altered chemical and biological conditions in a small man made pond between the Westinghouse NFD Plant and Sunset Lake in April 1980, a special groundwater investigation was initiated to determine the cause. Prior to this time, four on-site wells were routinely sampled and evaluated; groundwater contamination was not perceived to be a problem from that data analysis.

Previous reports had indicated that substances entering the surficial aquifer, or the Terrance unit, would not penetrate into the deeper aquifers, the Black Mingo or the Tuscaloosa units. At the time of the incident the pond was sampled and found to contain elevated concentrations of fluoride (60 mg/l) and ammonia (69 mg/l). A small spring discharging into the northern end of the pond was analyzed to contain fluoride and ammonia, at concentrations of 180 mg/l and 295 mg/l respectively.

Consultants were employed to assist the investigation and determine the magnitude of the plume contamination. Twenty-eight additional two inch wells were drilled and sampled to determine the source of the contamination and boundaries of the leachate plume. The investigation determined that the shallow aquifer was contaminated with ammonia and fluoride to a maximum level of 900 mg/l NH_3 and 150 mg/l F^-. No abnormal radioactivity concentrations were noted in the wells, other than the well immediately adjacent to the waste treatment dike pad area. The activity in this well was determined to be a factor of 200 below NRC approved discharge limits as stipulated in 10 CFR 20. Sources of contamination were determined to be the concentrated process waste treatment tanks, ammonia tank storage area, and possible lagoon leaks.

Westinghouse responded by constructing improved concrete dikes and internment pads for storage of process waste and raw materials. All lagoon liners were inspected and replaced with 36 mil hypalon liners. A complete review of piping networks was performed to insure a current leak did not exist. Extensive geological surveys were performed by Davis & Floyd and Law Engineering. The improvements have been effective in eliminating groundwater contamination from known sources.

The vertical extent of the contamination appears to be limited to the surface aquifer or the upper 20-25 foot of groundwater. The area wherein altered groundwater conditions were detected involves approximately 20 acres located in the center of the 1,175 acre site. Hydrogelogical work demonstrates that the predominate groundwater movement is south-southwest from the wastewater treatment area in the direction of the pond and Sunset Lake. These bodies of water would act to intercept the plume's movement. The geological units beneath the Terrace Unit were analyzed to determine the vertical hydraulic conductivity. This lower unit is composed of clay and shale. The lithology thickness and extremely low vertical conductivity (K ') of this unit in the area south of the main plant indicates that this unit should be an effective confining bed. No intrusion problems should be encountered in the Middendorf, Black Mingo, or Tuscaloosa aquifer systems.

There are no potable wells downdip of the plant between Sunset Lake and Congaree River. Therefore, no potable aquifer systems should be impacted by the incident. Using estimated permeabilities the conditions should take approximately ten years to return to normal. Well sampling procedures have been developed and monitoring will continue until background concentrations are attained.

5.2.4 IMPACTS OF SOLID WASTE

Noncontaminated solid waste is generated in the cafeteria, offices, and the contamination free manufacturing areas. These materials are disposed of by a local waste hauler and buried at county landfills.

Solids produced by chemical precipitation at waste treatment lagoons are basically CaF_2. CaF_2 solids contaminated with uranium are buried at low level waste burial sites. Future declassification of the material is expected, if the activity is less than 30 pCi/gram which would allow disposal at alternate sites. Calcium fluoride is an inert chemical material with extremely low solubility. Therefore, minimal impact should be involved with the burial of the material at current rates.

5.2.5 NOISE

No significant audible noise problems are discernible at the respective site boundaries.

5.2.6 APPEARANCE - VISUAL IMPACT

The Columbia plant was designed as a modern concrete structure to complement the flat rural surroundings. The grounds are landscaped with shrubs and wide expanses of lawn which is in sharp contrast to neighboring land use, where a chemical dump site conducted storage operations.

5.2.7 LOCAL EMPLOYMENT AND ECONOMICS

The present work force of approximately 850 is expected to grow gradually to approximately 1700 during the proposed five-year expansion period, which significantly reduces the influx during any given year. Most of the demand for labor can be met within the local labor area.

The plant currently generates approximately $18.5 million of income annually. Based on present salary levels, this amount should increase to over $36 million per year at the end of the expansion period.

Total gross tax revenues are expected to increase throughout the license renewal. The local community currently receives about $356,000 in property taxes, with $169,000 going to the State government each year as sales and use taxes.

A community relations file maintained by the Westinghouse Personnel Relations Department reveals that there has been minimal adverse public reaction to the plant since its establishment. Westinghouse actively participates in the annual United Way campaign soliciting nearly $90,000

in 1982. Approximately two oral presentations per month are given by company representatives to local civic groups, schools, and organizations concerning energy and nuclear power. The plant has been well accepted and is considered a favorable addition to the community.

SECTION 6

ENVIRONMENTAL MONITORING PROGRAM

6.1 DESCRIPTION OF PROGRAM

A comprehensive environmental monitoring program is conducted at the W-NFD Columbia Plant to assure that operations are conducted in compliance with all state and federal regulations. Various samples are typically analyzed at the point of discharge and at the site boundary or mixing zone for both chemical and radioactivity content. The environmental monitoring program documents site boundary and mixing zone results. The data is utilized to establish a statistical data base for routine and long-term environmental impact assessment.

The program is conducted as outlined in Table 6.1. Figures 6.1 and 6.2 illustrate sample locations.

6.2 CONCLUSIONS

The radiological monitoring program adequately measures the impact of plant effluents on the environmental during normal operations or following an accident situation.

TABLE 6.1

COLUMBIA SITE RADIOLOGICAL MONITORING PROGRAM

Type of Sample	Number of Locations[a]	Minimum Analysis	Monitoring Frequency[b]
Ambient Air Particulate	4	Gross Alpha	Continuous Sampling
Vegetation	4	Gross Alpha & Beta	Annually
Ground Water (Routine)	4	Gross Alpha & Beta	Quarterly
Surface Water (On-Site)	3	Gross Alpha & Beta	Quarterly
Surface Water (Congaree River)	3	Gross Alpha & Beta	Quarterly
Soil	4	Total Uranium	Annually
Sediment	1	Total Uranium	Annually
Fish	1	Gross Alpha & Beta	Annually

[a]Refer to Figures 6.1 and 6.2 for exact locations.
[b]Monitoring frequency may be increased following accident conditions

TABLE 6.2

TYPICAL ANALYTICAL SENSITIVITIES*1

Sample	Analysis	Sample Size	Minimum Detectable Level (MDL)
Air Particulate	Gross Alpha	571 m^3	2.0×10^{-15} uCi/ml
Vegetation	Gross Alpha	100 g	3.0 pCi/g
	Gross Beta	100 g	0.5 pCi/g
Water Samples	Gross Alpha	1 liter	2.2×10^{-9} uCi/ml
	Gross Beta	1 liter	2.5×10^{-8} uCi/ml
	Uranium	1 liter	5.0×10^{-7} uCi/ml
Fish	Gross Alpha	30 g	1.0 pCi/g
	Gross Beta	30 g	3.0 pCi/g
	Uranium	1 kg	0.02 pCi/g
Sediment and Soil	Gross Alpha	100 g	1.0 pCi/g
	Gross Beta	100 g	3.0 pCi/g
	Uranium	100 g	0.02 pCi/g

*1 - Sensitivities are statistically representative of vendors used by Westinghouse to perform these analyses with the exception of the air particulate analyses.

6.3 Nonradiological Monitoring Program

Sample	Locations	Analysis	Frequency
Stack gas	4	Particulate fluoride	Monthly
Stack gas	4	NH_3	Quarterly
Congaree River	3	NH_3, F^-, pH	Quarterly
Surface Water	3	NH_3, F^-, pH	Quarterly
Soil	4	Fluoride	Annually
Vegetation	4	Fluoride	Annually
Ground Water (Routine)	4	NH_3, F^-, pH	Quarterly
Ground Water (Non-Routine)	28	NH_3, F^-, pH	Annually
Fish	1	Fluoride	Annually

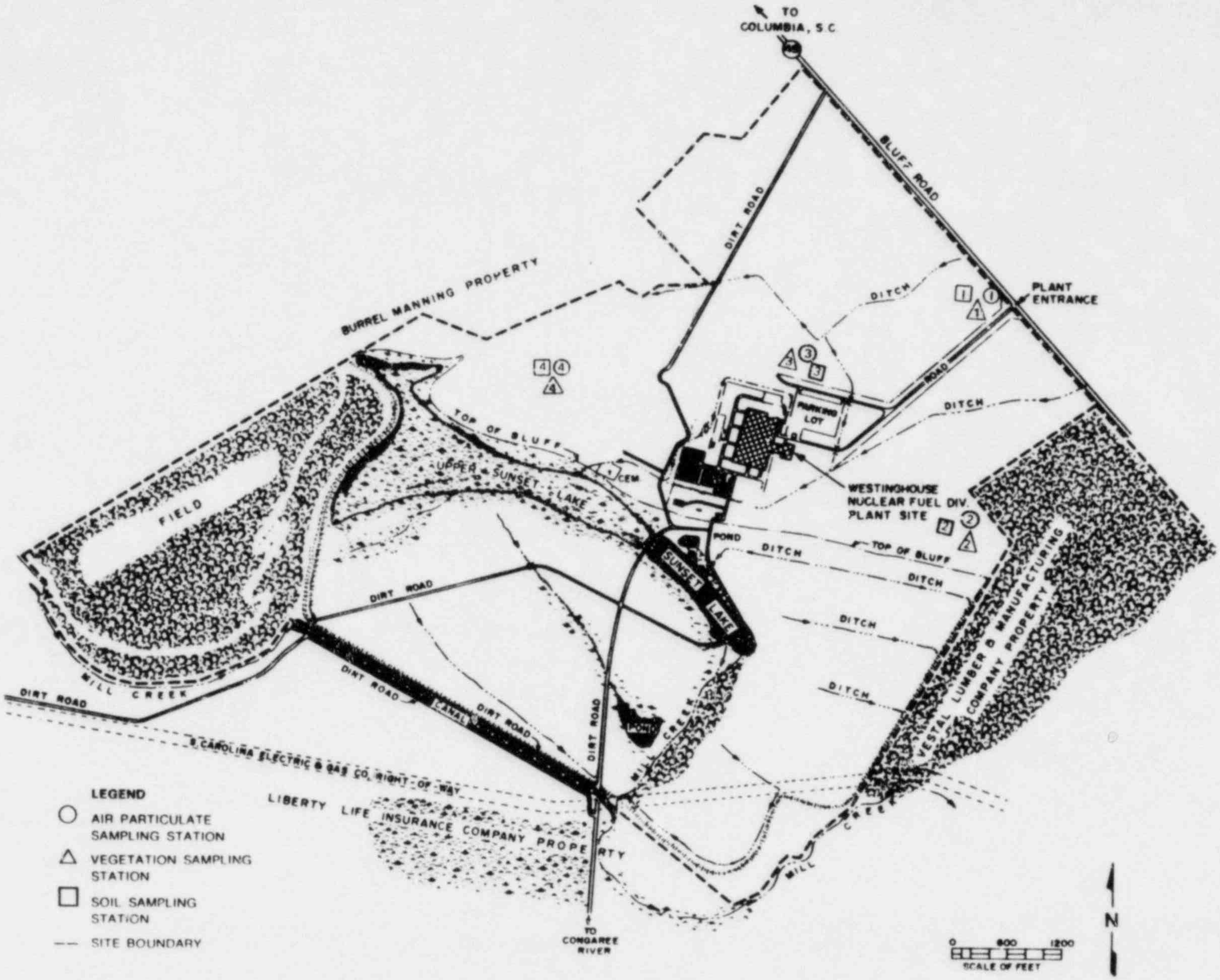


Figure 6.1. Locations of Air, Vegetation, and Soil Monitoring Stations.

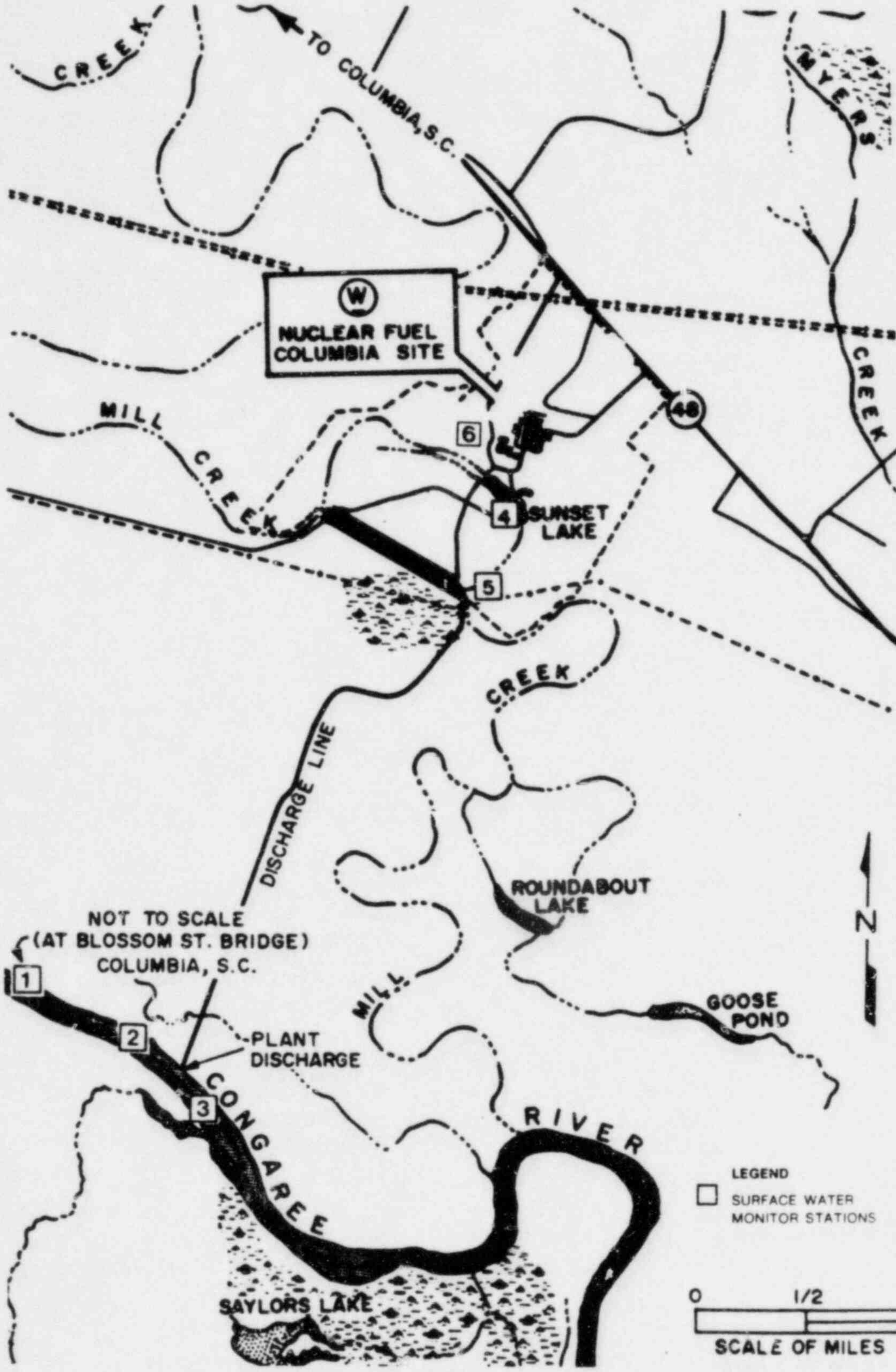


Figure 6.2. Locations of Surface Water Monitoring Stations.

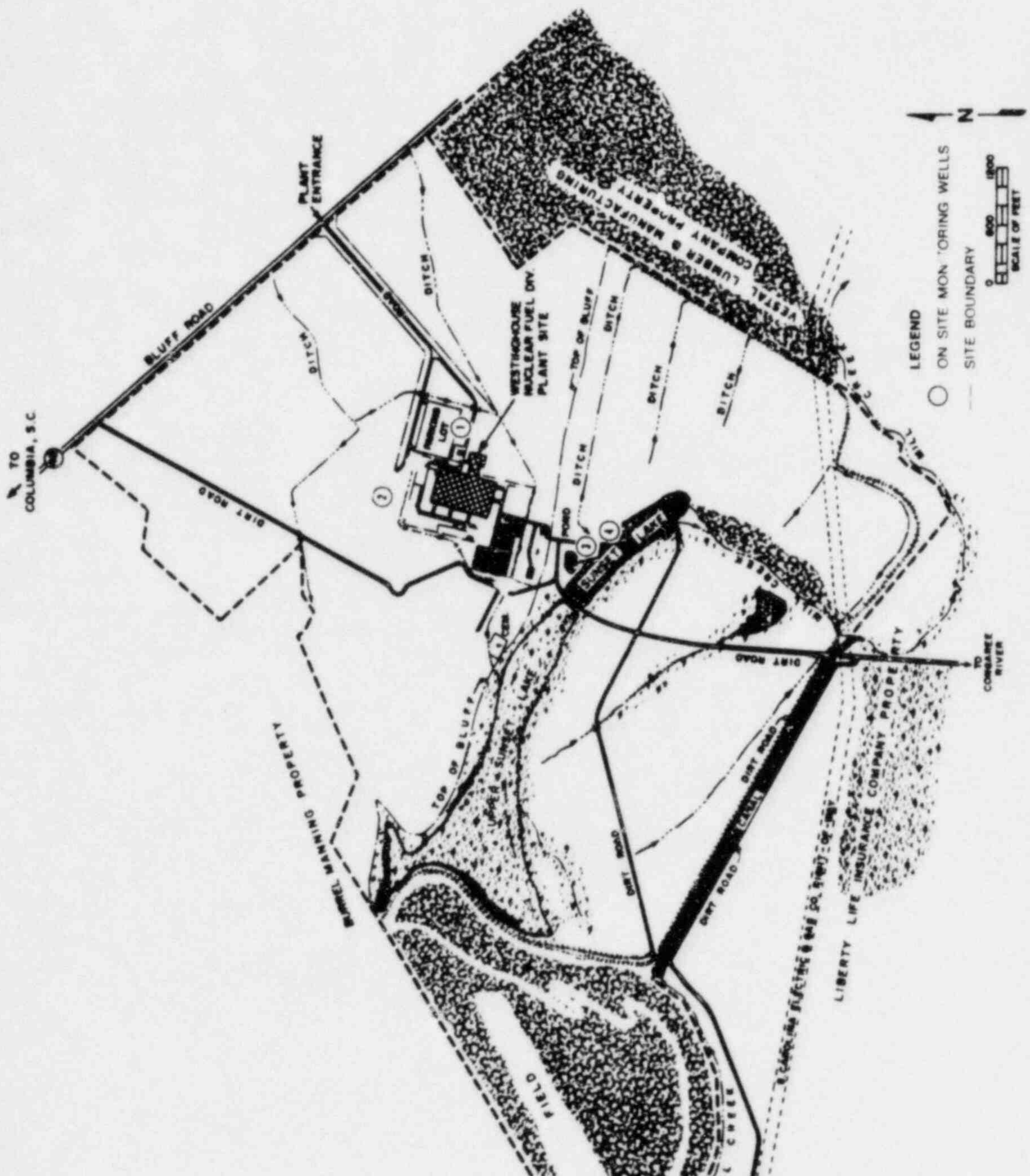


Figure 6.3. Locations of 4" Monitoring Wells.

SECTION 7

IMPACT OF ACCIDENTS

Accidents have been divided into three categories of severity according to the following classifications:

<u>Category 1 events</u>

Category 1 events would include plant accidents most likely to occur in the course of plant operations. These accidents would have the least severe consequences of the three categories considered. Although some adverse onsite effects could occur as a result of these accidents, the effects on the offsite environment would be minimal and not likely to exceed those from normal effluent releases.

<u>Category 2 events</u>

Category 2 events are accidents that may occur infrequently during the operating life of the plant. These accidents could release to the environment amounts of uranium or harmful chemicals that would be several times greater than those from normal effluent releases.

<u>Category 3 events</u>

Category 3 events include postulated accidents that are not expected to occur during the life of the plant; however, these accidents are considered because their consequences would include the potential for release to the environment of significant amounts of radioactive or otherwise harmful chemical materials. Thus, these accidents represent limiting-design conditions and are known as design-basis accidents.

In cases where atmospheric transport is of interest calculations were performed using the 50% confidence short-term values of X/Q.

7.1 RADIOLOGICAL ACCIDENT EVALUATION

The radiological and, where applicable, the chemical consequences of radiological accidents in all three categories both within and outside the manufacturing plant are summarized in Table 7.1.

For evaluation purposes, the radiological consequences have been presented in terms of the equivalent time of exposure to occupational MPC values given in 10 CFR Part 20, Appendix B. These are based on an annual dose commitment of 5 rems to the whole body, bone, or gonads, 30 rems to the skin and thyroid, and 15 rems to other organs.

A uranium chemical toxicity guide is shown in Table 7.2, based on recommendations of the Federal Water Pollution Control Administration; the percentage of the concentrations given in this guide are shown in Table 7.1.

In accident category 1, the event evaluated as having the greatest effect on the environment to humans results from a postulated spill of solid radioactive wastes in combination with a fire. This accident results in an airborne concentration of uranium at the site boundary that results in an exposure to the lung equivalent to about 2.6 hour at occupational MPC, assuming that all of the uranium is converted to the insoluble oxide form. The accident with the next highest effect is a minor leak in a holding lagoon equivalent to 12.5 gal/min remaining undetected for a 24 hour period. This release to Upper Sunset Lake results in uranium concentrations equivalent to only a 30 second exposure at nonoccupational MPC and 1.6×10^{-2}% of the chemical toxicity guide.

TABLE 7.1

SUMMARY OF RADIOLOGICAL PLANT ACCIDENT EVALUATION (700 MTU/YEAR CAPACITY)

Accident Category	Accident Description	Radiological Dose (rems/occurrence)	Uranium Chemical Toxicity (mg/day)	Equivalent time at Occupational MPC	Percentage of chemical toxicity guide
INSIDE					
1	Break in transfer line	normal releases		normal releases	
1	Solution spill	normal releases		normal releases	
1	UO_2 powder spill	normal releases		normal releases	
1	UF_6 release within vaporizer area	4.4×10^{-8} (bone)	1×10^{-6} (kidney)	1 sec (bone)	1.7×10^{-3} (kidney)
1	Loss of electric or water supply	normal releases		normal releases	
1	Spill of radioactive wastes	1.9×10^{-2} (lung)		2.6 hr (lung)	
1	Localized fire and/or explosion	1.9×10^{-2} (lung)		2.6 hr (lung)	
1	Failure of HVAC system	normal releases		normal releases	
OUTSIDE					
1	Leak of UF_6 container	normal releases		normal releases	
1	Leak in a lagoon	6.3×10^{-5} (GI tract)	8.0×10^{-4} (kidney)	30 sec (GI tract	1.6×10^{-2} (kidney)
INSIDE					
2	Cracked calciner tube	4.1×10^{-1} (lung)		2.4 days (lung)	
2	Failure of a single HEPA filter	3.0×10^{-2} (lung) 1.1×10^{-2} (bone)	1.4×10^{-1} (kidney)	4.2 hr (lung) 4.7 hr (bone)	5.8×10^{0} (kidney)
2	Rupture of a UO_2 fuel rod	normal releases		normal releases	
2	Rupture of a mixed oxide fuel rod	1.2×10^{-2} (lung)		1.7 hr (lung)	
2	Massive failure of a UF_6 shipping container	2.6×10^{-3} (lung) 1.0×10^{-3} (bone	2.4×10^{-2} (kidney)	22 min (lung) 17 min (bone)	1.0×10^{0} (kidney)
OUTSIDE					
2	Complete failure of UO_2 shipping container	9.0×10^{-1} (lung)		5.2 days (lung)	
2	Breach of a uranyl nitrate drum	1.6×10^{-3} (GI tract)	4.1×10^{-2} (kidney)	13 min (GI tract)	4.1×10^{-1} (kidney)
2	Massive failure of lagoon	7.3×10^{-3} (GI tract)	9.0×10^{-2}	1 hr (GI tract)	9.0×10^{-1} (kidney)
INSIDE					
3	Criticality	3.4×10^{0} thyroid) 1.4×10^{-1} (whole body)		9.8 days (thyroid) 2.4 days (whole body)	
3	Sintering furnace explosion	2.4×10^{-3} (lung)		20 min (lung)	
3	Calciner explosion	1.1×10^{-3} (lung) 6.1×10^{-5} (bone)	2.0×10^{-3} (kidney)	9 min (lung) 2 min (bone)	8.0×10^{-2} (kidney)
3	Fires in bank of filters	8.2×10^{0} (lung)		47 days (lung	

TABLE 7.2

URANIUM CHEMICAL TOXICITY RELEASE GUIDE ON AIR AND WATER INTAKE LIMITS (in mg/day) FOR HUMANS

Accident Category	Description	Air	Water
1	Minor faults and operational transients (accidents of moderate (frequency)	0.06	5
2	Infrequent faults	2.5	10
3	Limiting faults (design basis accidents)	2.5	10

An accident in category 1 with the third largest effect is a UF_6 release in the vaporizer area. The UF_6 leak is calculated to result in a uranium concentration at the site boundary that is equivalent to less than a 1-sec exposure at MPC and 1.7×10^{-3}% of the chemical-toxicity-guide value. All other accidents in category 1 are expected to be equal to or less than the allowable releases for normal operation. Thus, the consequences of the most frequent category 1-type accidents are calculated to be so far below the safe guide limits that they are expected to result in negligible effects to offsite residents or transients and to have virtually no other environmental effects.

Category 2 accidents within the manufacturing building that are evaluated as having the greatest effect on an individual located at the nearest site boundary include a cracked calciner tube, failure of a single HEPA filter, or a massive failure of a UF_6 container. These accidents result in calculated radiological doses that are equivalent to 2.4 days, 4.2 hr. and 22 min, respectively, at nonoccupational MPC. Additionally, a single HEPA filter failure and a massive failure of a UF_6 container are also calculated to result in inhaled concentrations of soluble uranium that are 5.8% and 1.0%, respectively, of the applicable chemical-toxicity-guide values.

Other category 2 accidents outside the manufacturing building that lead to similar radiological and/or chemical-toxicity effects include a complete failure of a UO_2 shipping container, a breach of a uranyl nitrate drum, and a massive failure of a lagoon. For the latter two events, the uranium is assumed to have contaminated all drains into Upper Sunset Lake and is diluted there by 43 million gallons of water. Since none of the category 2 accident consequences exceed 10% of the guide-value limits, these accidents were evaluated using conservative and/or realistic parameters; because such accidents would be expected to take place infrequently, no measurable effect on individuals living in the vicinity of the plant would be expected.

In category 3, the events leading to the largest effects on an individual located at the site boundary were postulated as a criticality event or a

fire in the bank of HEPA filters controlling the furnace exhausts. These two events, should they occur, would be expected to result in concentrations of airborne radioactive materials to the environment such that an individual located at the nearest site boundary would receive a thyroid dose commitment equivalent to 9.8 days exposure at nonoccupational MPC in the former case and a lung exposure equivalent to about 47 days at MPC in the latter case. No effects from chemical toxicity would be expected to result from these accidents. Some decontamination in the downwind direction would probably be required.

A calciner explosion, however, could result in an individual offsite receiving the equivalent of a 9-min exposure to the lung and 0.08% of the chemical-toxicity-guide value for the kidney.

Thus, these events, should they happen, would not be expected to result in measurable effects on residents or transients located nearby. Furthermore, as mentioned previously, category 3 events would not be expected to happen during the life of the plant.

7.2 NONRADIOLOGICAL ACCIDENT EVALUATION

The most significant environmental problems that may occur at a low-level-enrichment nuclear fuel fabrication plant would most likely result from possible accidents associated with potentially harmful chemicals rather than from radioactive materials. Thus, the NFCS can be considered in the same class as any other manufacturing plant in which significant quantities of nonradioactive chemicals are processed.

The chemicals presently stored onsite are listed in Table 7.3; estimated quantities for 1600-MTU capacity are also included.

The important chemical accidents would involve chemicals associated with uranium, anhydrous ammonia, aqueous ammonia, nitric acid, and hydrogen.

7.2.1 CATEGORY 1 ACCIDENTS

An accident of this type within the manufacturing building in the chemical processing area would be typified by minor liquid spills (i.e., 10 gal or less) including acid, ammonium diuranate, uranyl nitrate, and oil spills. A leak of this nature would be quickly detected by operators, and corrective action (such as isolation of the leaking-line section) would be taken. The spilled liquids would be quickly cleaned up and transferred to appropriate waste containers or, if appropriate, returned to the process for recovery. No floor drains are installed in the processing area of the main plant building, and therefore, there would be no release to the environment through either airborne or liquid pathways.

Category 1 accidents (external to the manufacturing building) that are likely to happen during the life of the plant include minor process-equipment leaks (50 gal or less). A leak of this type would be located rapidly by operators, and corrective action would be implemented.

Another possible accident of this type could result from the release of chemicals due to a leak in the liner of a waste-holding pond. In previous spills that occurred before improvements in the waste treatment process (as required by the NPDES permit) were implemented, a chemical analysis of surface water indicated that fluoride concentrations increased from less than 1 mg/liter for all sampling stations to 120 mg/liter at the causeway station, to 108 mg/liter at the spillway station, and to 96 mg/liter at the exit station. Ammonia concentrations at the causeway station increased from less than 1 mg/liter to 40.1 mg/liter. Typical incident conditions should not produce concentrations of this magnitude due to the previously mentioned decrease in concentration of lagoon contents. These concentrations can be hazardous to aquatic life.

7.2.2 CATEGORY 2 ACCIDENTS

In general, category 2 accidents within the manufacturing building (leakages of approximately 50 gal) are of greater magnitude than category 1; however, category 2 accidents still could not result in releases that would be of concern to the external environment because of the reasons given for accidents of category 1.

Greater environmental consideration must be given to the case when the water spray to the plant gaseous effluent scrubber fails. In this case, ammonia and ammonium fluoride would escape to the atmosphere unscrubbed. This condition could go undected for a maximum of 8 hour since the scrubber is checked at last once each shift. This scrubber is 70% to 85% effective; therefore, when discounting the scrubber efficiency, the maximum concentration of ammonia and ammonium fluoride that would escape to the environmental stmosphere during a maximum of 8 hr is estimated to be 36.28 mg/m^3 and 107 ug/m^3 respectively.

Category 2 accidents occurring in the chemical storage areas outside the manufacturing building could result in complete or partial emptying of a storage tank. The releases would flow through the storm drainage ditch to Upper Sunset Lake, where it would mix and flow into Lower Sunset Lake via a causeway. Lower Sunset Lake drains into Mills Creek, which eventually enters the Congaree River via a meandering route of about seven miles. Both Sunset Lakes are on Westinghouse property.

In the event of a major spill, the upper lake can be closed off at the causeway and then diluted by increasing the diverted flow of incoming Mill Creek water. The continuous chemical monitoring and prompt dilution of these waters can prevent significant liquid releases to the offsite environment. As part of the 1975 plan improvement program, protective dikes that could contain approximately 36,000 gal of liquid wastes in the event of complete tank failure were placed around the chemical tank farm. The largest bulk storage tank contains anhydrous ammonia, which has a capacity of 18,000 gal. The dikes were further upgraded in 1982 to assure leaks do not reach the groundwater.

7.2.3 CATEGORY 3 ACCIDENTS

Category 3 accidents are catastrophic in magnitude and are not expected in the plant's lifetime. All are extremely unlikely; they would involve either container rupture, failure, explosion, fire, natural disaster, or an extremely improbable criticality-type accident. Storage vessels are designed using good engineering practices and are filled according to safe operating procedures. To experience a rupture or failure, some unforeseen catastrophic disaster would have to occur, or all current safety systems would have to deteriorate simultaneously.

A category 3 accident in the chemical storage area would release no more potentially harmful chemicals than the contents of any industrial chemical storage tank or the overflow resulting from a massive lagoon break.

Another major accident could result from a hydrogen explosion, and as a result, tanks containing chemicals could be ruptured. Environmental hazards associated with such ruptures have been discussed in previous sections. To prevent such an accident, the hydrogen gas tanks are stored and handled according to OSHA regulation No. 1910.10 and maintained according to procedures recommended by the vendor.

7.3 EVALUATION OF POTENTIAL ENVIRONMENTAL IMPACT OF ACCIDENTS

It is concluded that even the most severe accidents will have little or no radiological environmental effect outside the site boundary.

TABLE 7.3

BULK CHEMICAL AND GAS STORAGE[a]

Product	Location	At 700 MTU/Year Total Gallons	At 700 MTU/Year Cubic Feet	At 700 MTU/Year Pounds	At 1600 MTU/Year (estimated) Total Gallons	At 1600 MTU/Year (estimated) Cubic feet	At 1600 MTU/Year (estimated) Pounds
Liquid NH_4OH	Tank farm	15,700			20,700		
Anhydrous ammonia	Tank farm	60,000			60,000		
Sodium Hydroxide	Tank farm	20,000			20,000		
Nitric acid (68%)	Tank farm	5,000			10,000		
Hydrogen (liquid)	Tank farm	18,000	2,044,000		36,000	2,044,000	
Nitrogen (liquid)	Tank farm	6,000	541,000		12,000	541,000	
Argon (gas)	Tank farm	1,200	116,000		1,200	116,000	
Helium (liquid)	Tank farm		138,000			276,000	
Uranium Hexafluoride	Outside Pad			550,000			1,100,000
Uranyl Nitrate (liquid)	Outside plant	30,000			30,000		
Lime (CaO)	Waste Treatment Hopper			140,000			200,000
Zinc stearate	Inside plant			2,500			5,000
Miscellaneous							
Acetone	55-gal drums (oil house)	825			825		
Sulfuric acid (66 Baume + 45°)	South pad	1500			700		
Nitric acid (68%)	55-gal drums (North pad)	275			275		
Muriatic acid (22% HCl)	55-gal drums (North pad)			800			800
Sodium Carbonate	Plating Room			800			800
Caustic soda (50% NaOH solution)	(North pad) (100-lb drum)			500			500
Nickel sulfate	Plating Room			500			500

[a]Other small amounts of miscellaneous chemicals (500 lb or 55 gal) are stored at the plant site.

APPENDIX A

FLORA AND FAUNA OBSERVED ON AND NEAR THE W NFD PLANT

(During 1974 Survey)

Appendix A

TABLE A.1

PLANTS OBSERVED ONSITE

Loblolly pine	Pinus taeda
Overcup oak	Quercus lyrata
Swamp chestnut oak	Q. michauxii
Southern red oak	Q. falcata
Cherrybark oak	Q. falcata var. pagodaefolia
Scarlet oak	Q. coccinea
Water oak	Q. nigra
White oak	Q. alba
Willow oak	Q. phellos
White ash	Fraxinus americana
Caolina ash	F. caroniniana
Shagbark hickory	Carya ovata
Mockernut hickory	C. tomentosa
Shellbark hickory	C. laciniosa
Bitternut hickory	C. cordiformes
Sweet gum	Liquidambar styraciflua
American elm	Ulmus americana
Winged elm	U. alata
Red maple	Acer rubrum
Yellow poplar	Liriodendron tulipifera
American beech	Fagus grandifolia
Black cherry	Prunus serotina
Water tupelo	Nyssa aquatica
Black locust	Robinia pseudoacacia
Redbud	Cercis canadensis
Poison ivy	Rhus radicans
Smooth sumac	R. glabra
Japanese honeysuckle	Lonicera japonica
Greenbrier	Smilax bona-nox
Trumpet vine	Campsis radicans
Virginia creeper	Parthenocissus quinquefolia
Common privet	Ligustrum vulgare
Cross vine	Bignonia capreolata
Blackberry	Rubus sp.
Lead bush	Amporpha fruticosa
Wild onion	Allium sp.
Smartweed	Polygonum sp.
Broomsedge	Andropogon virginicus
Great mullein	Verbascum thapsus
Sheep sorrel	Rumex hastatulus
Queen Anne's lace	Daucus carota
Water meal	Spilodella sp.
Duckweed	Lemna sp.

TABLE A.2

AVIFAUNA OBSERVED IN OPEN-FIELD AREAS

Turkey vulture	Cathartes aura
Red-tailed hawk	Buteo jamaicensis
Red-shouldered hawk	B. lineatus
Bobwhite	Colinus virginianus
Killdeer	Charadrius vociferous
Mourning Dove	Zenaidura macroura
Chimney swift	Chaetura pelagica
Tree sparrow	Iridoprocne bicolor
English sparrow	Passer domesticus
Bobolink	Dolichonyx oryzivorous
Eastern meadowlark	Sturnella magna
Common grackle	Quiscalus quiscula
Brown-headed cowbird	Molothrus ater
Savannah sparrow	Passerculus sandwichensis
Vesper sparrow	Pooecetes gramineus
Field sparrow	Spizella pusilla
Chipping sparrow	S. passerina
Song sparrow	Melospiza melodia

TABLE A.3
AVIFAUNA OBSERVED IN SWAMP-EDGE AREAS

Black duck	Anas rubripes
Wood duck	Aix sponsa
Marsh hawk	Circus cyaneus
Cattle egret	Bubulcus ibis
Great blue heron	Ardea herodias
Solitary sandpiper	Tringa solitaria
Spotted sandpiper	Actitis macularia
Barred owl	Strix varia
Fish crow	Corvus ossifragus
Acadian flycatcher	Empidonax virescens
Carolina wren	Thryothorus ludovicianus
Gray catbird	Dumetella carolinensis
White-eyed vireo	Vireo griseus
Red-eyed vireo	V. olivaceus
Prothonotary warbler	Protonotaria citrea
Northern parula warbler	Parula americana
American redstart	Setophaga ruticilla
Red-winged blackbird	Agelaius phoeniceus
American goldfinch	Spinus tristis
Song sparrow	Melospiza melodia

TABLE A.4

AVIFAUNA OBSERVED ALONG BORDERS OF OLD FIELDS AND WOODLOTS

Turkey vulture	Cathartes aura
Red-tailed hawk	Buteo jamaicensis
Red-shouldered hawk	B. lineatus
Common flicker	Colaptes auratus
Red-bellied woodpecker	Centurus carolinus
Hairy woodpecker	Dendrocopus villosus
Ruby-throated hummingbird	Archilochus colubris
Eastern kingbird	Tyrannus tyrannus
Great crested flycatcher	Myiarchus crinitus
Acadian flycatcher	Empidonax virescens
Blue jay	Cyanocitta cristata
Fish crow	Corvus ossifragus
Tufted titmouse	Parus bicolor
White-breasted nuthatch	Sitta carolinensis
Carolina wren	Thryothorus ludovicianus
Mockingbird	Mimus polyglottos
Gray catbird	Dumetella carolinensis
Robin	Turdus migratorius
Wood thrush	Hylocichla mustelina
Blue-gray gnatcatcher	Polioptila caerulea
Ruby-crowned kinglet	Regulus calendula
Cedar waxwing	Bombycilla cedrorum
Loggerhead shrike	Lanius ludovicianus
Starling	Sturnus vulgaris
Solitary vireo	Vireo solitarius
White-eyed vireo	V. griseus
Worm-eating warbler	Helmitheros vermivorous
Northern parula warbler	Parula americana
Yellow-rumped warbler	Dendroica coronata
Black-throated green warbler	D. virens
Black-throated blue warbler	D. caerulescens
Preairie warbler	D. discolor
Yellowthroat	Geothlypis trichas
Yellow-breasted chat	Icteria virens
Summer tanager	Piranga rubra
Cardinal	Richmondena cardinalsis
Rose-breasted grosbeak	Pheucticus ludovicianus
Indigo bunting	Passerina cyanea
American goldfinch	Spinus tristis
Rufous-sided towhee	Pipilo erythrophtalmus
Savannah sparrow	Passserculus sandwichensis
Vesper sparrow	Pooecetes gramineus
Field sparrow	Spizella pusilla
White-throated sparrow	Zonotrichia albicollis
Song sparrow	Melospiza melodia

TABLE A.5

AVIFAUNA OBSERVED IN ALL AREAS ON AND NEAR THE SITE

Black duck	Anas rubripes
Wood duck	Aix sponsa
Turkey vulture	Cathartes aura
Marsh hawk	Circus cyaneus
Red-tailed hawk	Buteo jamaicensis
Red-shouldered hawk	B. lineatus
Bobwhite	Colinus virginianus
Cattle egret	Bubulcus ibis
Great blue heron	Ardea herodias
Killdeer	Charadrius vociferous
Solitary sandpiper	Tringa solitaria
Spotted sandpiper	Actitis macularia
Mourning dove	Zenaidura macroura
Barred owl	Strix varia
Chimney swift	Chaetura pelagica
Ruby-throated hummingbird	Archilochus colubris
Common flicker	Calaptes auratus
Red-bellied woodpecker	Centurus carolinus
Hairy woodpecker	Dendrocopus villosus
Eastern kingbird	Tyrannus tyrannus
Great crested flycatcher	Myiarcus crinitus
Acadian flycatcher	Empidonax virescens
Tree swallow	Iridoprocne bicolor
Blue jay	Cyanocitta cristata
Fish crow	Corvus ossifragus
Tufted titmouse	Parus bicolor
White-breasted nuthatch	Sitta carolinensis
Carolina wren	Thryothorus ludovicianus
Mockingbird	Minus polyglottos
Gray catbird	Dumetella carolinensis
Robin	Turdus migratorius
Wood thrush	Hylocichla mustelina
Blue-gray gnatcatcher	Polioptila caerulea
Ruby-crowned kinglet	Regulus calendula
Cedar waxwing	Bombycilla cedrorum
Loggerhead shrike	Lanius ludovicianus
Starling	Sturnus vulgaris
Solitary vireo	Vireo solitarius
White-eyed vireo	V. griseus
Pronthonotary warbler	Protonotaria citrea
Worm-eating warbler	Helmitheros vermivorous
Parula warbler	Parula americana
Northern yellow-rumped warbler	Dendroica coronata
Black-throated green warbler	D. virens
Black-throated blue warbler	D. caerulescens
Prairie warbler	D. discolor
Yellowthroat	Geothlypis trichas
Yellow-breasted chat	Icteria virens
American redstart	Setophaga ruticilla
English sparrow	Passer domesticus

TABLE A.5 (CONTINUED)

AVIFAUNA OBSERVED IN ALL AREAS ON AND NEAR THE SITE

Bobolink	Dolichonyx oryzivorous
Eastern meadowlark	Sturnella magna
Red-winged blackbird	Agelaius phoeniceus
Common grackle	Quiscalus quiscula
Brown-headed cowbird	Molothrus ater
Summer tanager	Piranga rubra
Cardinal	Richmondena cardinalis
Rose-breasted grosbeak	Pheucticus ludovicianus
Indigo bunting	Passerina cyanea
American goldfinch	Spinus tristis
Rufous-sided towhee	Pipilo erythrophthalmus
Savannah sparrow	Passerculus sandwichensis
Vesper sparrow	Pooecetes gramineus
Chipping sparrow	Spizella passerina
Field sparrow	S. pusilla
White-throated sparrow	Zonotrichia albicolis
Song sparrow	Melospiza melodia

TABLE A.6

MAMMALS OF THE PLANT SITE AND ADJACENT AREAS

Oppossum	Didelphis marsupialis
Shorttail shrew	Blarina brevicauda
Least shrew	Cryptotis parva
Eastern mole	Scalopus aquaticus
Little brown myotis	Myotis lucifugus
Racoon	Procyon lotor
River otter	Lutra canadensis
Striped skunk	Mephitis mephitis
Red fox	Vulpes fulva
Gray fox	Urocyon cinereoargenteus
Bobcat	Lynx rufus
Woodchuck	Marmota monax
Eastern chipmunk	Tamias striatus
Eastern gray squirrel	Sciurus carolinensis
Red squirrel	Tamiasciurus hudsonicus
White-footed mouse	Peromyscus leucopus
Golden mouse	P. nuttalli
Meadow vole	Microtus pennsylvanicus
Muskrat	Ondatra zibethica
Meadow jumping mouse	Zapus hudsonius
Eastern cottontail	Sylvilagus floridanus
Wild boar	Sus scrofa
Eastern whitetail deer	Odocoileus virginianus

TABLE A.7

MAMMALS COMMON TO SHORE-LINE AREAS

Oppossum	Didelphis marsupialis
Shorttail shrew	Blarina brevicauda
Least shrew	Cryptotis parva
Little brown myotis	Myotis lucifugus
Racoon	Procyon lotor
River otter	Lutra canadensis
Striped skunk	Mephites mephites
Red fox	Vulpes fulva
Gray fox	Urocyon cinereoargenteus
Bobcat	Lynx rufus
Eastern chipmunk	Tamias striatus
White-footed mouse	Peromyscus leucopus
Muskrat	Ondatra zibethica
Eastern whitetail deer	Odocoileus virginianus

TABLE A.8

MAMMALS COMMON TO EDGE AREAS OF OLD FIELDS AND WOODLOTS

Oppossum	Didelphis marsupialis
Shorttail shrew	Blarina brevicauda
Raccoon	Procyon lotor
Red fox	Vulpes fulva
Gray fox	Urocyon cinereoargenteus
Eastern chipmunk	Tamias striatus
Eastern gray squirrel	Sciurus carolinensis
Red squirrel	Tamiasciurus hudsonicus
White-footed mouse	Peromyscus leucopus
Eastern cottontail	Sylvilagus fluoridanus
Eastern whitetail deer	Odocoileus virginianus

TABLE A.9

MAMMALS COMMON TO GRASSY AREAS AND OPEN FIELDS

Oppossum	Didelphis marsupialis
Least shrew	Cryptotis parva
Eastern mole	Scalopus aquaticus
Striped skunk	Mephites mephites
Woodchuck	Marmota monax
Meadow vole	Microtus pennsylvanicus
Meadow jumping mouse	Zapus hudsonius

TABLE A.10

HERPETOFAUNA OBSERVED ONSITE AND IN ADJOINING AREAS

Amphibians

Green frog	Rana clamitans melanota
Bullfrog	R. catesbeiana
Southern cricket frog	Acris gryllus gryllus
Southern leopard frog	Rana pipiens sphenocephala
American toad	Bufo americanus

Reptiles

Snapping turtle	Chelydra serpentina
Yellow-bellied turtle	Pseudemys scripta scripta
Eastern mud turtle	Kinosternon subrubrum subrubrum
Eastern painted turtle	Chrysemys picta picta
Eastern box turtle	Terrapene carolina carolina
Five-lined skink	Eumeces fasciatus
Banded water snake	Natrix sipedon fasciata
Red-bellied water snake	N. erythrogaster erythrogaster
Black rat snake	Elaphe obsoleta obsoleta
Eastern king snake	Lampropeltis getulus
Canebrake rattlesnake	Crotalus horridus atricaudatus

A. 11

THREATENED OR ENDANGERED SPECIES IN SOUTH CAROLINA
POSSIBLY OCCURING ON THE SITE

Eastern brown pelican	Pelicanus occidentalis
Southern bald eagle	Haliaeetus l. leucocephalus
American peregrine falcon	Falco peregrinus aratum
Red-cockaded woodpecker	Dendrocopus borealis
Bachman's warbler	Vermivora bachmanii
Eastern cougar	Felis concolor
American alligator	Alligator mississippiensis

TABLE A.12

THREATENED PLANT SPECIES IN RICHLAND AND ADJACENT COUNTIES
SOUTH CAROLINA

Family	Genus, species, variety	Habitat Preference	County
Asteraceae	Helianthus schweinitzii	Upland wood, thickets, pastures on Piedmont	Lexington
Crassulaceae	Sedum pussillum	Vernal pools in granite on Piedmont	Fairfield Kershaw
Fagaceae	Quercus georgiana	Granitic hills	Kershaw
Haloragaceae	Myriophyllum laxum	Sinks and pools	Kershaw
Isoetaceae	Isoetes melanospora	Ponds on granite, low wet fields edges of sluggish streams on Piedmont and Coastal Plain	Richland
Liliaceae	Trillium pusillum var.	(var. not mentioned) alluvial woods, pocosins	Calhourn
Poaceae	Sporobolus teretifolius	Savannas on Coastal Plain	Kershaw
Santalaceae	Nestronia umbellula	Parasitic on pine roots, in woods primarily on Piedmont	Richland Calhoun
Sarraceniaceae	Sarracenia rubra	Shrubby bogs, savannas on Coastal Plain	Richland Calhoun Lexington Sumter

Source: Smithstonian Institution, Report on Endangered and Threatened Plant Species of the United States, Ser. No. 94-A, Washington, D. C., 1975; A. E. Radford, H. E. Ahles, and C. R. Bell Manual of the Vascular Flora of the Carolinas, University of North Carolina Press, Chapel Hill, 1968.

Appendix B

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CONCEPTUAL PLAN AND COST ESTIMATE

FOR DECOMMISSIONING THE WESTINGHOUSE

NUCLEAR FUEL DIVISION - COLUMBIA, SOUTH CAROLINA PLANT

APRIL 1983

TABLE OF CONTENTS

SECTION I

STATEMENT OF PURPOSE

STATEMENT OF PURPOSE

This report was prepared for the Westinghouse Nuclear Fuel Division, Columbia, South Carolina, for the purpose of estimating the cost of decommissioning the Columbia facility. In addition to the estimate of cost, a discussion of decommissioning methods and a conceptual plan of decommissioning activities are included. This report is not intended to be a decommissioning plan, but rather a preliminary planning tool.

The concepts, costs, and regulatory requirements presented in this report are based on conditions as they exist in January 1983. The contents of this report represent a judgment and best estimate for the impact of a future event and should be viewed accordingly.

SECTION II

SUMMARY

SUMMARY

Westinghouse contracted with Chem-Nuclear Systems, Inc. for the preparation of a conceptual plan in sufficient detail to devise the broad scope of decommissioning activities and an estimate of the cost involved. This document was submitted to the USNRC in 1978 in support of the SNM-1107 license renewal application. The costs have been updated annually since 1978. This plan represents a complete revision which incorporates all modifications to the facility since 1978. The plan and resulting cost estimate were prepared using the following assumptions:

1. Costs were expressed in 1983 dollars.
2. Decommissioning activities were to be performed by an independent contractor.
3. Packaging, transportation and disposal charges were calculated using the Chem-Nuclear Systems, Inc., Barnwell, South Carolina burial facility.
4. Current radiological limits and decontamination technology were assumed.
5. All process equipment and ancillary equipment in the contaminated areas would be decontaminated as much as possible, packaged and buried at a licensed burial facility.
6. All buildings would be decontaminated to levels acceptable for release for unrestricted use.
7. All contaminated underground piping would be removed, decontaminated as much as possible, packaged and buried at a licensed burial facility. The ground surrounding the piping would be surveyed and also removed for burial if contaminated.
8. Most of the material in the waste lagoons was assumed to contain less than 30 picocuries per gram uranium; the material would be removed from the lagoons, processed and buried in a chemical or sanitary landfill.
9. The USNRC has promulgated regulations establishing de minimus quantities of SNM that could be excluded from regulations.

Using these parameters, the contractor inventoried the equipment and material in the facility and established the portion likely to remain uncontaminated or capable of being satisfactorily decontaminated. The remainder would be prepared and transported for disposal to a licensed burial facility. Similarly, those portions of the facility, such as the walls, floors, etc., requiring decontamination were estimated.

The estimate provided an allowance for professional health physics staffing to perform a comprehensive initial survey and the preparation of a specific overall plan prior to the initiation of decommissioning activities. It also provided for the required health physics surveillance during the entire operation, including the final contamination surveys of the "clean" facility and an acceptance inspection by the appropriate licensing agency as a condition of terminating the license.

By costing out these activities, an estimate of costs was then determined by the contractor for the decommissioning of the Columbia facility. Using the above assumptions and the contractor's estimate, the total costs for decommissioning the Columbia facility are estimated by Westinghouse to be approximately 17 million dollars. On the same basis, Westinghouse estimates that it will take approximately one year to complete the decommissioning task.

SECTION III

SITE DESCRIPTION

SITE DESCRIPTION

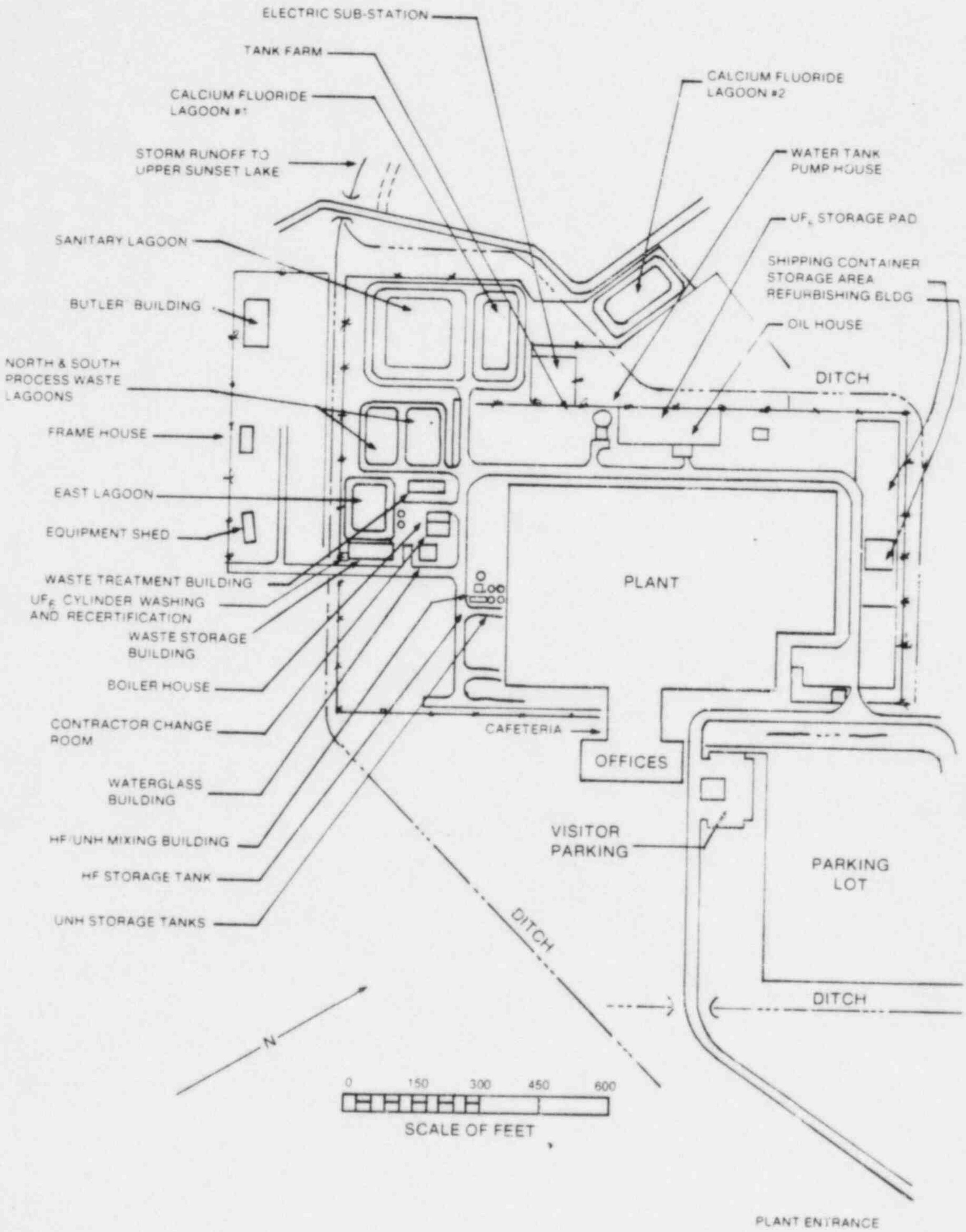
The Westinghouse Electric Corporation, Nuclear Fuel Division, Columbia Site is located on South Carolina Highway No. 48 (Bluff Road) in Richland County approximately 8 miles southeast of Columbia, South Carolina. The plant is situated approximately 1,800 feet from Highway No. 48 on a semirural plot of approximately 1,156 acres. The licensed facility is on a knoll approximately 40 feet above the water level of the nearby Congaree River. The Site Emergency Plan shows the geographical location of the site.

Relative locations of the buildings on the site are shown in Figure III-1. All existing buildings are occupied by the Manufacturing Department of the Nuclear Fuel Division (NFD).

The building in which NFD Manufacturing operations are conducted consists mainly of high-bay manufacturing areas. It is constructed with a structural steel framework and case concrete curtain walls. Adjoining the east side of the building is a two-story office building. All significant quantities of material authorized under License SNM-1107 are received, processed and stored (excluding outside UF6 storage) within these NFD manufacturing areas.

The site Waste Treatment Facilities and associated lagoons are located in an area southwest of the Manufacturing Building. Located adjacent to this facility is the Waste Storage Building which is used as a contaminated solid waste storage area and for other miscellaneous activities. An equipment shed, a small vacant house, and a Butler building used for equipment and materials storage are also located in the same direction, but farther away.

A chemical tank farm, water storage tank and pump house, UF6 cylinder storage pad, oil house and shipping container refurbishing building are located to the west side of the Manufacturing Building. The UF6 cylinder storage pad is surrounded by a security fence and equipped with criticality evacuation alarms.



COLUMBIA SITE BUILDING LOCATIONS

FIGURE III-1

Processing Operations

The NFD Columbia Site is primarily engaged in the manufacture of fuel assemblies for commercial nuclear reactors. The processing operations authorized under License SNM-1107 consist of receiving low-enriched (less than 5.0 w/o U-235) uranium hexafluoride; processing the hexafluoride to produce uranium dioxide powder; and processing the uranium dioxide through pellet pressing and sintering, fuel rod loading and sealing, and fuel assembly fabrication. These operations are accompanied by appropriate radiation protection, nuclear criticality safety, quality assurance, SNM safeguards and environmental pollution controls.

Scrap or waste licensed material resulting from site operations are processed to permit them to be recycled into production operations, or more closely controlled prior to discarding. These operations may involve chemical separation (e.g., acid treatment and dissolution or acid leaching, followed by chemical precipitation), solvent extraction, mechanical separation and thermal decomposition.

In addition, combustible contaminated wastes are incinerated on site. The operation greatly reduces the volume of waste materials to be disposed of by burial, and it permits increased accuracy in the measurements made to determine the nature and quantity of SNM being discarded.

SECTION IV

DISPOSAL AND RELEASE CRITERIA

DISPOSAL AND RELEASE CRITERIA

The cost estimates for transportation and disposal listed in Section IX are based on 1983 dollars and existing license conditions for the Chem-Nuclear Systems, Inc. Barnwell, South Carolina disposal site. Under existing license conditions, SNM received at the Barnwell Site must be packaged in accordance with DOT requirements.

The packaging costs listed in Section IX, are based on Low Specific Activity (LSA) packaging requirements. This position is predicated upon the exemption permitted under 49 CFR 173.396 (a) (5), which states in part, "The following materials are not classified as fissile radioactive materials, are exempted from this section, and must be packaged in accordance with other provisions of this subpart as appropriate:

(5) "A package containing less than 350 grams of fissile material, if there is not more than 5 grams of fissile mateiial in any cubic foot within the package."

"Low Specific Activity" is defined in 49 CFR 173.389 which states in part:

(5) "Objects of nonradioactive material externally contaminated with radioactive material, provided that the radioactive material is not readily dispersible and the surface contamination when averaged over an area of one (1) square meter, does not exceed 0.0001 millicurie (220,000 disintegrations per minute) per square meter of Group I radionuclides or 0.001 millicurie (2,200,000 disintegrations per minute) per square meter of other radionuclides."

The packaging requirements for LSA are provided in 49 CFR 173.392.

To summarize, the cost est mates presented in this report are based on LSA packaging and transportation requirements and Chem-Nuclear Systems' license conditions. All material packaged for disposal will be assayed

under an approved procedure prior to release from the NFD-Columbia controlled area. Items which do not meet the release criteria given in Table IV-1, will require decontamination to the appropriate levels.

TABLE [illegible]1: UNRESTRICTED AREA RADIOLOGICAL CONTAMINATION LIMITS

	OPTION 1		OPTION 2	
Isotope	Total (2)	Removable (1)(2)	Total (2)(3)	Removable (1)(2)
U nat, U-235, U-238 Th nat, Th-232 and associated decay products	5,000 dpm/100 cm^2	1,000 dpm/100 cm^2	Avg: 5,000 dpm/100 cm^2 Max: 15,000 dpm/100 cm^2	1,000 dpm/100 cm^2
Other isotopes which decay by alpha emission or by spontaneous fission	1,000 dpm/100 cm^2	100 dpm/100 cm^2	Avg: 500 dpm/100 cm^2 Max: 2,500 dpm/100 cm^2	100 dpm/100 cm^2
Beta-Gamma emitters (isotopes with decay modes other than alpha emission or spontaneous fission)	0.4 mrad/hr at 1 cm*	1,000 dpm/100 cm^2	Avg: 0.2 mrad/hr at 1 cm* Max: 1.0 mrad/hr at 1 cm*	1,000 dpm/100 cm^2

* Where surface contamination by both alpha and beta-gamma emitting isotopes exists, the limits established for alpha and beta-gamma emitting isotopes shall apply independently.

(1) The amount of removable radioactive material per 100 cm^2 of surface area shall be determined by wiping that area with dry filter or soft absorbent paper and with the application of moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. In determining removable contamination on objects of lesser surface area, the pertinent levels shall be reduced proportionally, and the entire surface shall be wiped.

(2) 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 and count rate meter, for background, efficiency, and geometric factors associated with the instrumentation.

(3) Measurements of total contaminant shall not be averaged over more than 10 square meters. For objects of lesser surface area, the average shall be derived for each such object.

* Measured through not more than 7 mg/cm^2 of total absorber.

NOTE: Either Option 1 or Option 2 may be used. For example, if all beta-gamma readings were less than 0.4 mrad/hr at 1 cm, Option 1 could be used; but if the maximum reading were 0.8 mrad/hr, material could be released under Option 2, providing the average was less than 0.2 mrad/hr.

SECTION V

CONCEPTUAL DECOMMISSIONING PLAN

CONCEPTUAL DECOMMISSIONING PLAN

Planning and Preparation

Prior to the commencement of decommissioning, an extensive study and evaluation program would be initiated. The following considerations would be included in this study:

a. Compliance with regulations
b. Dose rates and activities of components
c. Physical size and weights of the various components
d. Excavation requirements
e. Equipment available to transport the waste products
f. Containers available in the industry for packaging of the equipment and components and other waste materials
g. Limits in which to function such as: (1) acceptable dose rates, (2) work to be done in controlled areas, (3) work that can be done in restricted areas, (4) types of contamination control measures to be used.
h. Assignment of a category to each item or group of items which would involve placing items in categories for packaging and handling where consideration can be given to dose rate, volume and weight.
i. Disposal of equipment, materials and piece parts
j. Disposal of radioactive sources
k. Disposal of contaminated earth
l. Razing of structures and facilities as required
m. Cleanup to meet limits for levels of radioactivity

Radiation and Contamination Control

Control of radiation exposure and the spread of contamination is of primary importance during the conduct of decommissioning activities. Appropriate measures would be taken to minimize or completely eliminate radiation exposure to personnel and spread of contamination to the environment.

Upon completion of each phase, such as dismantling and decontamination of a portion of a process area, radiation and contamination surveys would be performed. After the surveys are analyzed for that work area, it would be released or further decontaminated in order to achieve acceptable limits. During all dismantling operations, continuous air sampling would be conducted. Also, special samples would be taken at the point of worker involvement. Controls would be implemented to limit personnel exposures (both external and internal) to As Low As Reasonably Achievable levels.

The contamination control area would encompass all process areas where radiation or contamination exists in amounts above the limits established for clean areas. Entry to and exit from contamination control areas for individuals would be through the designated access control points only. All equipment entry and exit from the process work areas would have a designated control point. This would serve as a survey point for initial release and proper documentation for transport loading.

All transport vehicles arriving at the Columbia facility would be cleared, logged in, and issued passes. The vehicles would then be weighed, surveyed, and moved to the designated loading area. To avoid overweight during loading, portable scales would be used to determine the individual axle weight of each truck and trailer. When the loading operation was completed, the truck would be moved to a designated area and surveyed for release.

Soil and water samples would be collected and analyzed to confirm that the surrounding environment had not been contaminated. If any contamination levels in excess of allowable limits specified in USNRC regulations were found, soil from the affected areas would be removed, sampled for radioactivity and transferred to an appropriate disposal facility.

Conceptual Plan

Should it become necessary for Westinghouse to decommission the Columbia facility, a contractor would be selected and all affected parties would be included in the decision making process to ensure an orderly progression of decommissioning activities.

The incinerator and support systems would be left in service until all other process equipment has been removed, packaged, and possession transferred to the receiver. This would permit SNM recovery throughout the entire effort and would provide a means of recovering SNM removed during the decontamination process. All piping would be flushed thoroughly to ensure internal contamination has been removed before any dismantling occurs.

Utility service would be left intact to the facility, but isolation and disconnect would be made prior to the removal of the equipment in each service area. A schedule of service disruption and disconnect would be developed during the initial survey period.

The initial radiological survey and decontamination tests would be in-depth and complete. After the results of these surveys had been analyzed and a final decommissioning plan developed and approved, decommissioning would begin (See Figure V-1).

The removal of non-process related equipment, materials, and hardware would be made along with the removal of equipment from non-SNM areas. This will allow for flexibility in planning due to the amount of usable floor space that will become available.

All SNM and process system equipment and hardware such as heavy equipment, fuel handling devices, and pellet furnaces would be removed. A radiolngical assessment of process structures and areas would be performed. These structures would have to meet contamination criteria

for packaging or further decontamination would be performed. After these criteria have been met, the structures would be dismantled and packaged in accordance with the disposal criteria. The overhead piping and connecting piping would be the next order of removal.

Following the removal of all equipment from the process areas, a decision concerning the buildings and improvements would be made. Some portions of these buildings and improvements might be salvaged through conventional means while other portions would require special disposal techniques.

Contaminated areas and equipment located in open, uncovered areas would be addressed during the latter phase of decommissioning. Equipment and small structures that are contaminated, however, would be included in shipments of material from the process area. In areas where contaminated undergound piping had been removed, the ground surrounding the piping would be surveyed and removed for disposal if contaminated.

The final survey of the Columbia Site would begin only after the complete removal of all contaminated material from the site, and the final survey and release of all heavy equipment, vehicles, and support equipment. Prior to beginning the final survey, Westinghose personnel and the contractor would meet with the appropriate NRC staff to determine current views on acceptable survey criteria and to identify those NRC personnel which may be authenticating the results of the final survey.

SECTION VI

DECONTAMINATION METHODS

DECONTAMINATION METHODS

It is believed that decommissioning methods and techniques should be tested and evaluated at the site of the task prior to the actual commencement of decommissioning activities. This affords the advantage of a "dry-run" under the same conditions as the actual decontamination project. For instance, testing and evaluation would be performed with field-installed radiation detection instrumentation and "split" samples would be returned to selected laboratories for comparison evaluations to establish levels of detectability.

Enclosed within this section are Table VI-1, which lists decontamination aids, and Table VI-2, which lists decontamination methods and the order in which testing would be done in accordance with the various conditions at the Columbia facility.

TABLE [illegible]1

DECONTAMINATION AIDS

Method	Surface	Advantage	Precautions
Vacuum Cleaning (absolute filter at exhaust side)	Dry or wet	Rapidly removes loose dirt, helps prevent airborne problems	Minimize airborne during the vacuuming
Steam	Paint or oily surfaces	90% reduction of contamination on painted surfaces	Need drainage for water or way to collect
Detergents	Nonporous	Dissolves film	Requires contact with surface. Good only for loose contamination
Complexing Agents Oxalates Carbonates Citrates	Nonporous, nonrust, or oxidized surface	Holds contamination in solution	Has no penetration powers
Organic Solvents	Nonporous	Quick dissolving action very little solution to dispose of as waste	Requires good ventilation and fire precaution. Toxic.
Inorganic Acids	Metal surfaces	Corrosive action	Good ventilation. Toxic, corrosive
Caustics Tri-sodium Phosphate	Painted surfaces (vertical)	Quick reduction in contamination	Destructive, not to be used on aluminum
Abrasion	Nonporous	Reduce level as low as desired	Control must be tight. Prevent contaminton spread

TABLE VI-2

DECONTAMINATION METHODS

(Listed in Order of Trial)

Material	Method of Decontamination
Concrete	1. Vacuum 2. Scrub with detergent and water 3. Wetting agent mixture 4. Steam cleaner 5. 10 per cent ammonia citrate solution 6. HCl solution (30-60 percent) 7. Chip away area
Steel	1. Vacuum 2. Soap and water 3. Decon 4306-D* (if oxidized) 4. TiO_2 paste (keep wet) 5. Remove section
Transite	1. Vacuum 2. Detergent and water 3. Wetting agent mixture (versene) 4. Remove section 5. Use abrasize powder (keep wet)
Aluminum	1. Vacuum 2. Detergent and water 3. Wetting agent 4. Decon 4306-D* (if oxidized) 4. TiO_2 paste (keep wet) 5. Remove section
Glass	1. Window cleaner 2. Detergent or $Na_3\ Po_4$ solution 3. 10 per cent ammonia citrate
Tile	1. Vacuum 2. Detergent and water (wet vacuum) 3. Trisodium phosphate 4. 10 per cent ammonia citrate solution 5. Acetone or other solvent 6. Remove section

Material	Method of Decontamination
Linoleum	1. Vacuum 2. Detergent and water 3. Solvent 4. Dilute HCl 5. Ammonia citrate solution 6. Remove section
Painted Surfaces	1. Vacuum 2. Detergent and water 3. Wetting agents 4. Decon 345 or decon 4182-A* 5. Steam clean 6. Dilute HCl 7. Paint remover
Wood	1. Vacuum 2. Remove section
Soil	1. Process (if water U removed) 2. Dispose

*Turco Products

SECTION VII

EQUIPMENT VOLUME LIST AND AREA DATA

EQUIPMENT VOLUME LIST AND AREA DATA

Equipment Volume List

Area	Cubic Feet
Inplant Office	942
Blender Area	1,231
Incinerator	3,140
Process Filter Housing	97,767
Pellet Loading and Welding Area	4,870
Pellet Area Equipment	62,708
Conversion Area	142,405
UF_6 Bay	2,836
Total	315,899

Area Survey

Area	Square Feet
Total Wall Area	36,980
Total Floor and Walkway Area	84,000
Ceiling Area	84,000
Total	204,980

SECTION VIII

OUTSIDE AREAS

OUTSIDE AREAS

Waste Lagoons

1 - Settling Lagoon West - 1	190' long x 56' wide x 4' deep 1.75×10^6 gallons
1 - Settling Lagoon West - 2	190' long x 56' wide x 4' deep 1.75×10^6 gallons
1 - Process Lagoon North	125' long x 65' wide x 4' deep 9×10^4 gallons
1 - Process Lagoon South	125' long x 65' wide x 4' deep 9×10^4 gallons
1 - Process lagoon East	120' long x 90' wide x 4' deep 5×10^5 gallons

Storage Tanks

3 - 2320-Gallon Tanks	Low-level contaminated liquid waste
4 - 7500-Gallon Tanks	Uranyl nitrate solution - $UO_2(NO_3)_2$
6 - 30,000-Gallon Tanks	Low-level contaminated liquid waste

Waste Storage Building

1 - Waste Storage Building is approximately 150' long by 44' wide and is used for waste storage, waste packaging, decontamination, and UF_6 cylinder washing and recertification.

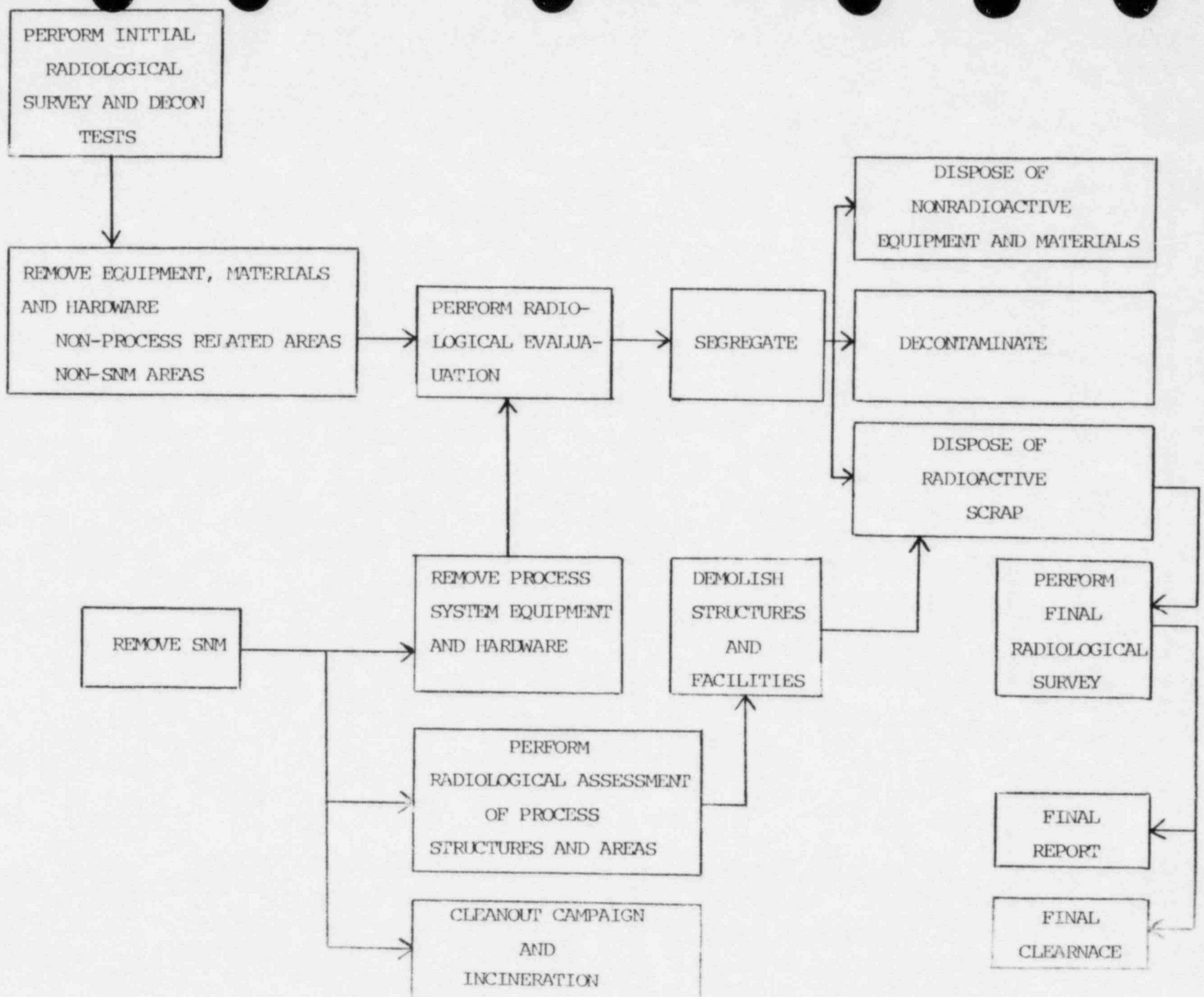
Calcium Fluoride Disposal

The following information concerns the disposal of the material in the waste lagoons.

It is estimated that the settling lagoons will contain approximately 1.75 $\times 10^6$ gallons of liquid and solid waste material. Approximately 50 per cent of the total is calcium fluoride (CaF_2). The specific activity of the waste is less than 30 picocuries per gram. This material would be appropriately sampled, solidified and transported to a chemical or sanitary disposal site in accordance with SM-1107. Assuming that 50 per cent of the total lagoon volume is CaF_2 and that the CaF_2 has a density of 13.5 pounds per gallon, the total quantity of CaF_2 is estimated at 11.8 $\times 10^6$ pounds. Absorbent material would be added at a rate of 35 pounds of absorbent for each 100 pounds of CaF_2. Consequently, the total quantity of solidified sludge would be 15.9 $\times$ 10^6 pounds. The material would be shipped in bulk form via trucks.

One process lagoon is reserved for calcium fluoride which does not meet the requirements for disposal via a chemical landfill. It is estimated that this lagoon contains approximately 10^5 cubic feet of CaF_2. This material will be solidified, containerized and shipped for burial as radioactive waste.

FIGURE V-1: FLOW OF DECOMMISSIONING ACTIVITIES



SECTION IX

COST ESTIMATE

COST ESTIMATE

Manufacturing Plant		
Labor		
Craft Personnel	\$3,165,562	
Health Physics Personnel	\$1,392,912	
Transportation and Disposal		
Transportation - 342 loads at \$1,194/load	\$ 408,348	
Packaging - 486 boxes at \$720/box	\$ 349,920	
Burial - 434,349 ft^3 at \$17.04/ft^3	\$7,401,307	
Equipment Cost	\$ 172,088	
Subtotal		\$12,890,137
Calcium Fluoride		
Transportation and Disposal		
Chemical Landfill Disposal (15.9×10^6 pounds at \$.03/pound)	\$ 477,000	
Chem Nuclear Barnwell Disposal 10^5 cubic feet at \$17.04/ft^3	\$1,704,000	
Subtotal		\$ 2,181,000
Waste Storage Area		
Combined Disposal Cost	\$ 24,403	
Subtotal		\$ 24,403
Surveys		
Initial Site Survey	\$ 98,109	
Final Release Survey	\$ 195,295	
Subtotal		\$ 293,404
10 Per Cent Contingency		\$ 1,538,894
TOTAL COST		\$16,927,838

WESTINGHOUSE ELECTRIC CORPORATION

NUCLEAR FUEL DIVISION

COLUMBIA PLANT ALARA REPORT

JANUARY 1 TO JUNE 30, 1982

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CONTENTS

I. SUMMARY

This report summarizes ALARA programs and data for the first six months of 1982 and evaluates short- and long-term trends in the following topical areas: personnel exposures (external and internal), airborne radioactivity concentrations, effluent and environmental monitoring, unusual occurrences, audit findings, uses of equipment, and ALARA programs. The most significant trends are summarized below:

GENERAL

The ALARA program for the Columbia Plant is directed by the Regulatory Compliance Committee (RCC), an organization of technical staff managers and other cognizant personnel. This Committee meets routinely to consider pertinent subjects related to the ALARA program (airborne radioactivity concentrations, regulatory requirements, significant ALARA trends). Specific ALARA efforts associated with airborne reduction programs have been delegated to the Airborne Radiation Task Force, a subcommittee of the RCC. This group meets routinely to review specific airborne data and trends, identify corrective actions where applicable and schedule implementation programs. ALARA philosophies are transmitted to appropriate personnel via policies, procedures and training. A detailed description of specific ALARA programs for this reporting period is found in Section IX.

AIRBORNE RADIOACTIVITY CONCENTRATIONS

Controlled Area airborne radioactivity averages were approximately 8% MPC for the first six months of 1982 versus 8% MPC for calendar year 1981. The Conversion Area was unchanged at 11% MPC; the Pellet Area increased from 7% MPC to 9% MPC; the Conversion Services Area decreased from 9% MPC to 8% MPC.

The most significant trend occurred in the Pellet Area where airborne radioactivity averages increased, primarily because of problems experienced with the pellet presses. The principle causes have been identified and corrective actions have been planned and/or implemented to improve equipment and administrative controls.

The number of air sample stations exceeding an average of 25% MPC has increased from four during the last six months of 1981 to five during the current reporting period because of the addition of the Blue M Furnace air sample station. Corrective actions have been identified for all five air sample stations.

The number of airborne spikes exceeding 100% MPC increased by approximately 4% during the first six months of 1982 compared with the last six months of 1981. This was attributed to an increase in the number of airborne spikes on the pellet presses, Conversion Line 3 and Blue M Furnace.

PERSONNEL EXPOSURES

External personnel exposures for both whole body and skin continue to average less than 25% of the NRC maximum permissible quarterly dose of 1,250 mrem and 7,500 mrem respectively. Decreasing trends since 1978 are apparent for all departments as a result of equipment and material handling improvement in the Controlled and Mechanical Areas.

Internal personnel exposures, as monitored by airborne exposures in MPC-hours and bioassay measurements (urine and in-vivo) are consistent with reductions in airborne radioactivity concentrations in the Controlled Area.

EFFLUENTS AND ENVIRONMENTAL

Effluents from the Columbia Plant are sampled and analyzed to determine releases to the environment. Based upon NRC meteorological and dispersion models and calculations, airborne effluents result in an annual dose to the nearest resident to the Columbia Plant Site which is less than 7% of Federal limits, based upon a permitted annual dose of 25 mrem.

CONCLUSIONS

(1) Inplant airborne radioactivity concentrations have stabilized during the past three ALARA reporting periods. Programs have been identified to address those areas with the highest airborne potentials.

(2) External personnel exposures continue to average well below permissible levels, with continued decreasing trends.

(3) Environmental discharges remain well within license limits and permitted values.

II. EXTERNAL PERSONNEL EXPOSURES

External personnel dosimeters (TLD's) are supplied to those individuals who may be exposed to greater than 25% of the applicable quarterly radiation dose limits specified in paragraph (a) of 10 CFR 20.101. Personnel dosimeters are also distributed, upon request or at the discretion of Radiological and Environmental Engineering, to those individuals who would not normally be monitored. Badges are provided for approximately 450 individuals and are changed and evaluated quarterly for men and monthly for women (the latter frequency was implemented to provide additional administrative controls for fertile women in minimizing fetal exposures).

Whole body and skin doses by job function are summarized in Figures 1-8 for 1977 through the first quarter of 1982 for each major job category. (Because quarterly exposure reports are provided by a vendor, data for the second quarter of 1982 were not available for this report; for this reason, external exposure data reporting will always be one quarter behind.)

All departmental whole body dose averages for each quarter to date were below 25% of the NRC Maximum Permissible Dose Limit of 1,250 mrem. A comparison of quarterly average whole body dose results for the last four quarters versus the average for the four prior quarters shows the following: Conversion (-42%), Pellet (-32%), Rod (-35%), QC Chemical (-35%), Maintenance (-41%), Final Inspection (-17%), Rod Inspection (-22%), Final Assembly (-18%). Reductions were observed in all areas. Trend analyses have shown significant decreases in the past three years, particularly for Controlled Areas. The following modifications have contributed to these favorable trends: (1) improvements in housekeeping which have significantly lowered contamination levels on floors, process equipment, personnel dosimeters, etc.,

(2) relocations of uranium powder to central storage areas to reduce personnel exposures at the process lines, (3) storage of pellets in shielded storage carts, (4) removals of contaminated scrap and surplus equpment from the facility, (5) reduction in the inventory of UF_6 cylinder "heels" in the UF_6 Bay, (6) personnel training which emphasized dose reduction techniques and increased personnel awareness, (7) enclosure improvements (e.g., stratification blender and QC inspection hoods) to provide for more remote handling of uranium powder and pellets, and (8) increased mechanization in the Rod Area.

All departmental skin dose averages were also less than 25% of the NRC Maximum Permissible Dose of 7,500 mrem for each quarter to date. A comparison of quarterly average skin dose results for the last four quarters versus the average for the four prior quarters shows the following: Conversion (-41%), Pellet (-34%), Rod (-56%), QC Chemical (-30%), Maintenance (-23%), Rod Inspection (-18%), Final Inspection (-25%), Final Assembly (-17%). Similar trends were observed for skin doses: a decline in the average departmental doses since 1978. This is consistent with the improvements cited above.

The highest individual whole body doses for the fourth quarter of 1981 and the first quarter of 1982 were 640 mrem (Uranium Inventory) and 620 mrem (Pellet Area). These doses are consistent with the work conducted and material handled in these areas. The highest individual skin doses for this period were 1,250 mrem (Maintenance) and 1,250 (Pellet Area). No unusual problems were identified.

Approximately 47 badges are placed throughout the Manufacturing areas to monitor general area exposure rates. Exposure rates are consistent with powder, pellet, rod and assembly inventories in these areas. Average quarterly badge exposures range from approximately 2 mrem in the Maintenance Area to approximately 212 mrem in the Final Assembly Area.

The integrated whole body doses for the fourth quarter of 1981 and the first quarter of 1982 were approximately 28 and 30 person-rem respectively, or an average of approximately 64 mrem per quarter per person badged. Integrated dose reductions have paralleled the departmental dose reductions because the total badged population has remained relatively stable.

The ALARA Program to minimize external personnel exposures consists of the following: (1) routine surveillance of radioactive material areas for external radiation, (2) personnel and area dosimetry, (3) Radiological and Environmental Engineering reviews of surveillance and dosimetry results, (4) investigations of unusual occurrences, and (5) evaluations of area and personnel trends. From data obtained through these means, recommendations are made for process and/or equipment modifications which will further reduce radiation exposures.

The following equipment modifications and administrative controls have also contributed to lower external personnel exposures: (1) a large quantity of uranium-contaminated scrap and surplus equipment has been removed from the Controlled Area, thereby reducing radiation exposure rates. Since the beginning of 1982, over 24,000 cubic feet of radioactive waste have been removed from the facility, (2) UF_6 cylinders containing uranium "heels" are significant contributors to external personnel exposures in the UF_6 Bay because of elevated beta-gamma exposure rates from the cylinders. The residence time for these cylinders in the UF_6 Bay has been reduced significantly, thereby reducing personnel exposures, (3) the powder cart storage area was rearranged to provide a greater distance between the fuel and adjacent aisleways, thus reducing radiation exposure rates. Provisions have also been made for the storage of powder carts within the newly installed bulk powder storage area which will further reduce external radiation exposures.

III. INTERNAL PERSONNEL EXPOSURES

Personnel internal exposures are based upon Controlled Area air sampling results at representative work stations and an accounting of time spent by each individual at these locations. Daily internal exposures (in MPC-hours) are calculated for each individual who spends a significant portion of his time working with unencapsulated forms of uranium. The average internal exposures in MPC-hours for each job category since 1978 are presented graphically in Figures 9-13.

Bioassay (urine, fecal and in-vivo) results are used to verify the internal exposures. Urinalyses and in-vivo counts are performed on a routine frequency based upon the exposure potential. Fecal sampling is performed to verify uranium uptakes in the event of an unusual occurrence. The average bioassay results (urine and in-vivo) per quarter per job category since 1977 are presented graphically in Figures 9-13.

MPC-Hours

All departments averaged less than 10% of the maximum permissible quarterly limit of 520 MPC-hours except for the Pellet Area (57 MPC-hours or approximately 11% of this limit). The Pellet Area continues to experience elevated airborne radioactivity, primarily from the pellet presses (see Section IV and IX for descriptions of programs designed to reduce these concentrations). The highest individual exposure occurred in the first quarter (100 MPC-hours) to an individual in the Pellet Area.

As reported in the last ALARA Report, Daniel Construction averaged 62 MPC-hours during the third quarter of 1981 which represented a significant increase over previous results. This was attributed to increased maintenance and construction activities on Conversion Lines 3 and 5. For the first half of 1982, the average exposure for Daniel Construction was reduced to 31 MPC-hours, reflecting a reduced workload on process equipment.

Urinalyses

Urine samples are collected routinely from potentially exposed personnel to assist in the evaluation of exposures to transportable and nontransportable uranium. Urinalysis frequencies are determined routinely by Radiological and Environmental Engineering based upon criteria such as exposure levels in MPC-hours, scope of work in the Controlled Area, etc.

Urinalysis results are plotted in Figures 9-13 for each major department. There were no significant changes in departmental averages during the first two quarters of 1982. Individuals were encouraged to submit urine samples immediately upon reporting to work following the December 1981 Christmas plant shutdown. Of the 51 samples submitted, all were below the minimum detectable level of 2 micrograms uranium per liter, confirming that, for these individuals, there is no significant long term systemic uranium deposition.

In-vivo

In-vivo departmental averages for the first and second quarters of 1982 continue to average less than the minimum detectable quantity of U-235 for the in-vivo counter as shown in Figure 9-13. The Pellet Area experienced an increase in average in-vivo counts for the first and second quarters of 1982 versus 1981 (+35%). This is attributed to a recalibration of the in-vivo counter. The Conversion Area experienced a corresponding increase in in-vivo counts during the first quarter of 1982; however, in-vivo count averages decreased during the second quarter because of improvements in scheduling of counts to minimize the possibility of skin contamination, a surface phenomenon which can significantly affect counting results.

Personnel Work Restrictions

Personnel are restricted from working with radioactive materials for a variety of reasons, including routine urinalyses and in-vivo counts exceeding

certain internal limits, elevated nasal smears, unusual occurrences, etc. During this reporting period, thirteen individuals were restricted, three because of elevated urine sample results, five due to elevated in-vivo analyses and five because of high airborne. This represents a decrease compared with restrictions in the last half of 1981 (19). Of the thirteen restrictions, six occurred in the Conversion Area, four in the Pellet Area, two in Maintenance and one in the Waste Recovery and Disposal Area. All individuals were subsequently released from restriction and returned to their routine work assignment.

NOTE: Some work restrictions result from failure to submit timely bioassay samples. These restrictions were not included in the above data analyses.

IV. AIRBORNE RADIOACTIVITY CONCENTRATIONS

Airborne radioactivity concentrations in the Controlled Area are sampled continuously at approximately 177 individual, fixed work stations, with samples changed and counted each shift. In addition, special air samples are taken to evaluate airborne radioactivity during nonroutine activities and engineering evaluations. The following parameters are used to evaluate airborne radioactivity trends: (1) Controlled Area average airborne radioactivity concentrations, (2) individual area average airborne radioactivity concentrations, (3) individual air sample stations exceeding 25% MPC, and (4) airborne radioactivity spikes. These are discussed below:

The Controlled Area airborne radioactivity concentration average for the first six months of 1982 was 8% MPC versus 8% MPC for the calender year 1981. Individual area averages are presented in Figure 14 from 1976 to the present, showing trends for this period.

The locations and average results of individual Controlled Area air sample stations averaging greater than 25% MPC for the first six months of 1982 are listed in Table 1, together with the trends for this period. During the first six months of 1982, approximately 3% (5) of the air sampler stations averaged over 25% MPC; this represents an increase over the last six months of 1981 (2% or 4 sampler stations). In the Conversion Area, only one air sample station, Line 3 Hot Filter Hood, exceeded 25% MPC; compared with one for the last six months of 1981 (Line 5 Feed End Bottom Vent Star). The Line 3 Hot Filter Hood sample average of 27% MPC represented a nominal increase from the last six months of 1981 (25% MPC), and resulted primarily from UO_2 powder spills during vent line rod-outs of the vertical calciner vent. Vent line rod-outs were previously performed by inserting a rod from beneath the vent; when the powder plug became dislodged, UO_2 powder would fall into the hood

and then onto the floor. To correct this problem, a vent access port was installed on the upper portion of the hot filter to provide access from above for rod-outs. Valve isolation between the vent line and the hood assures that powder releases into the polypak in the hood are controlled. Since this modification was completed, there have been no significant releases from this operation.

In the Conversion Services area, there was one individual air sample station, Blue-M Furnace, which exceeded 25% MPC for the first six months of 1982. During this period, the granulator was replaced with a jaw crusher and relocated. In conjunction with this activity, the two Blue-M Oxidation Furnaces were placed adjacent to each other with the air sampler relocated between the furnaces. The proximity of the two furnaces increased the airborne radioactivity concentrations at this station. To correct this problem, new ventilation hoods are being designed for the furnaces.

The stratification blender input hood sample increased from 12% MPC during the last reporting period to 23% MPC during the first six months of 1982. Efforts are underway to reduce these concentrations (routine changing of hood prefilters and installation of permanent containment between the carousel and the input hood).

In the Pellet Area, Pellet Presses 1, 2, and 4 exceeded 25% MPC for the first six months of 1982 (for the previous reporting period, Pellet Presses 1, 2 and 3 exceeded 25% MPC). The four operating Pellet Presses averaged 28% MPC for the first six months of 1982 vs. 25% MPC for the last six months of 1981. Elevated airborne concentrations resulted primarily from containment leaks, maintenance activities and transfers of uranium powder outside of containment by personnel during routine operations such as pellet density checks and press adjustments. Airborne Milestones have been developed to address both

equipment modifications and improvements in administrative controls. The following containment improvements have been implemented: (1) redesign of a portion of the containment on Pellet Lines 2 and 3 to minimize powder leaks around the drive belts, (2) removal of the pellet weigh hoods from Pellet Lines 2 and 3 to increase ventilation flows, and (3) modifications to hood access doors. For pellet operators, airborne awareness sessions have been conducted to emphasize airborne reduction techniques and methods.

The total number of air samples exceeding specific percent MPC ranges is presented in Table 2 and shown graphically for samples greater than 100% MPC in Figure 15. For the first half of 1982, there was a 4% increase in the number of spikes exceeding 100% MPC when compared with the last half of 1981. This is attributed to increased numbers of spikes on the pellet presses, Conversion Line 3, and the Blue M furnaces.

A list of air sample stations with the highest frequency of spikes greater than 100% MPC is shown in Table 3, including the trends for the first half of 1982 versus 1981. Each of these is discussed below:

Pellet Presses--Airborne spikes continue to occur at the pellet presses during routine operations and result from containment leaks, maintenance activities and operator transfers of material outside the enclosures. Airborne Milestones were developed during the second quarter of 1982 to modify containment and improve administrative controls to reduce the potential for powder transfers outside the enclosures (see previous discussion).

Conversion Line 3 Feed End Top--The principle causes of airborne spikes at this location were (1) powder spills resulting from vent system rod-outs (see above discussion concerning Conversion Area airborne radioactivity concentrations), (2) calciner and duplex valve maintenance activities, and (3) system leaks. Most of the airborne radioactivity and the majority of

the spikes occurred during the first quarter with a subsequent reduction in the second quarter. For example, there were eleven spikes during the first quarter and only five during the second. Airborne radioactivity concentrations were reduced from 23% MPC during the first quarter to 17% MPC during the second quarter. The actions described above to improve vent system rod-outs should further improve the airborne radioactivity concentrations at this station.

Conversion Line 4-Feed End Bottom-Vent Star--The following contributed to airborne spikes: (1) vent system rod-outs,, (2) maintenance activities on the feed end (calciner, duplex valves, ADU pump, dryer and feed screw), and (3) clean-outs. The Airborne Radiation Task Force has initiated actions to evaluate alternatives to the existing methods for vent system clean-outs to reduce airborne releases during these operations.

Blue M Furnaces--During this reporting period, the two Blue-M furnaces were moved adjacent to each other, and the air sample station was relocated to a position between them. Hoods are currently being designed to contain this operation.

Incinerator Output--The number of airborne radioactivity spikes decreased during 1982, primarily because of additional containment and ventilation installed at the ash removal location.

V. EFFLUENT AND ENVIRONMENTAL MONITORING

Effluent releases from the Columbia Plant consist of two pathways, gaseous and liquid. Gaseous and particulate effluents are HEPA filtered and scrubbed as appropriate before being discharged to the environment. The total activity discharged during the first six months of 1982 was 502 microcuries of uranium, compared with 485 microcuries during the last six months of 1981. This represents a small increase over the previous reporting period, but within expected statistical variations. Minor problems were noted with the incinerator exhaust during the week ending May 16, 1982 due to a HEPA filter failure. There has been no recurrence of this problem.

Gaseous effluents represent an average discharge concentration of less than 10% of the Maximum Permissible Concentration for unrestricted areas. Figure 16 shows the trends since 1977.

The total annual discharge of 987 microcuries uranium during the last twelve months corresponds to an annual lung dose of approximately 3.5 mrem to an infant living at the nearest plant boundary or approximately 1.7 mrem to an infant at the nearest residence. This represents less than 14% (nearest plant boundary) and 7% (nearest resident) respectively of the environmental standards contained in 40CFR190, as derived from NRC models and calculations.

Liquid wastes are treated to remove chemical and radioactive constituents and discharged to the Congaree River. Sampling and analyses are performed to confirm that effluent concentrations are well below applicable regulatory limits. Typically, liquid discharges average less than 10% MPC. An advanced waste treatment facility was installed in late 1980 to recover uranium from the conversion process liquid waste effluent stream. Uranium concentrations are lowered from approximately 15-20 ppm to less than 0.1 ppm by this

treatment process. Since installation of this system, over 11 million gallons of liquid waste have been treated, with a total recovery of over 660 Kgs uranium, which would otherwise have been discharged to the environment or buried as low-level radioactive waste.

During the first half of 1982, approximately 70,000 microcuries of uranium were discharged to the Congaree River, compared with approximately 75,000 microcuries during the last six months of 1981. This reduction is within statistical variations expected in this area.

Nonradiological effluents of ammonia and fluorides are shown in Figure 17. During the first half of 1982, ammonia discharges averaged approximately 18 pounds per day which is well within the limits of the National Pollutant Discharge Elimination System (NPDES) Permit of 35 pounds per day. Flouride effluents averaged approximately 13 pounds per day which is also well within the NPDES Permit limit of 30 pounds per day.

A new NPDES Permit became effective on January 1, 1982 and will be in effect for five years. There were no changes in the ammonia or fluoride discharge limits.

Environmental airborne radioactivity concentrations are sampled continuously to verify that plant effluents are ALARA. Air concentrations at the site boundaries continue to average approximately 0.1% of the 10CFR20 Maximum Permissible Concentration for unrestricted areas.

VI. UNUSUAL OCCURRENCES

During the first six months of 1982, there were no unusual occurrences requiring complete evacuation of the Controlled Area. Furthermore, there were no significant releases of UF_6 requiring evacuation of the UF_6 Bay. This reflects the equipment improvements and administrative controls which have been implemented.

As shown in Table 7, there were no personnel exposures during the last six months of 1981 which exceeded 40 MPC-hours in any seven-day period. This is attributed to the reductions in airborne radioactive concentrations and airborne spikes.

VII. AUDITS

Routine Regulatory Compliance audits were conducted during the first six months of 1982 in the areas of radiation protection, nuclear criticality safety and SNM safeguards. Inspections are conducted jointly by Radiological and Environmental Engineering personnel and line management in accordance with written checklists. Results of inspections are discussed with cognizant supervisors and managers, scheduled for correction, and followed up to assure that timely corrective actions have been taken.

Radiation protection items identified during this period were associated with uranium contamination, containment capture velocities, air sampling, containment integrity, ventilation indicators, wearing of peronnel dosimetry and housekeeping. Nuclear criticality safety items included SNM storage, moderation controls, spacing of SNM, container labelling, and criticality postings. Safeguards findings included item control, SNM identification, and storage. In most instances, effective corrective actions were taken to prevent recurrence. No undesirable trends were noted.

A total of twelve NRC inspections were conducted during the first six months of 1982 involving approximately 79 inspection person-days versus ten inspections and approximately 71 person-days for the last half of 1981. The inspections were conducted in the areas of health physics, nuclear criticality safety, safeguards (SNM measurements, accountability and SNM physical inventory), physical security and fire protection. There were four items of noncompliance versus eight during the previous reporting period. Table 8 summarizes the results of NRC inspections.

Three of the items of noncompliance (all Severity Level V) were received in the Safeguards area: (1) failure to develop item control procedures (this

represented a repeat item from a November 1981 NRC inspection), (2) failure to distribute timely Form 741's, and (3) failure to have procedures reviewed and approved by the Measurement Control Coordinator. The fourth item was received in the area of fire protection for failure to follow an internal procedure. Appropriate corrective actions have been taken to address the items and to prevent recurrence.

The four items of noncompliance were received during three inspections. There were no items of noncompliance during nine inspections.

VIII. USES OF EQUIPMENT

During the last reporting period, major modifications were made to Conversion Line 2 to improve electronic data monitoring, provide feedback and control of processing parameters, install more efficient ventilated containment for the feed end and upgrade effluent control equipment. Air sampling performed during operations of this line with enriched uranium (see Table 4) show considerable improvement in both airborne radioactivity concentrations and number of airborne spikes.

Emergency ventilation for the UF_6 Bay was formerly supplied by the DCFB fan and filter house which consisted of a positive pressure HEPA filter system. The filter house was removed to accommodate facility roof construction. Emergency ventilation for the UF_6 Bay is now supplied by the Conversion Area scrubber system which incorporates state-of-the-art ventilation techniques (negative pressure system and bag-in/bag out HEPA filter capabilities).

The new nondestructive analyzer became operational during this reporting period. This equiment is being used for uranium assay of baled waste to minimize errors associated with these operations and maximize equipment efficiency such as incinerator loading. Significant improvements have been realized as follows: (1) The incinerator can operate for longer uninterrupted periods because of an improved measurement accuracy for baled waste and ash, (2) Fewer shutdowns result in lower airborne radioactivity concentrations in the area because of fewer breaches of equipment, (3) Personnel exposures are reduced, and (4) Effluent control systems operate more efficiently with fewer shutdowns.

All flowmeters were calibrated during this reporting period, including fixed air sample locations, recirculating air sample stations and those associated with effluent sampling.

Routine personnel contamination checks by Health Physics Operations were initiated on all shifts to verify the effectiveness of change room and step-off pad practices.

An improved breathing zone air sampler was designed for use in the Controlled Area to determine representativeness of air sample stations. The new sampler minimizes the variables associated with fixed versus lapel sampling to provide a greater degree of confidence in the sampling technique. Initial results have been encouraging.

IX. ALARA PROGRAM SUMMARY

A. General

Management commitment to the As-Low-As-Reasonably-Achievable (ALARA) principle continued in order to minimize airborne radioactivity concentrations, personnel exposures and effluents to the environment. The overall ALARA program is directed by the Regulatory Compliance Committee which establishes plant policies and procedures and monitors progress of implementing groups in achieving ALARA. One of these groups, the Airborne Radiation Task Force, met on two occasions during the first six months of 1982 to identify and report progress on "Airborne Milestones," review airborne radioactivity concentrations and evaluate facility trends in several important areas. Results of Task Force activities are reported to the Regulatory Compliance Committee for information and concurrence. In addition to these efforts, the General Manager of the Nuclear Fuel Division monitors progress towards achieving ALARA goals.

B. Airborne and Effluent Reduction Programs

The following programs have been effective in reducing airborne radioactive concentrations in the Controlled Areas during the current reporting period of January to June 1982.

1. Process Line Improvements

   a. Conversion Line 3 Vent System

      The Conversion Line 3 vent system was modified to provide access to the vent line from above the hot filter to prevent powder spills during vent system rod-outs (previously, rod-outs were performed from beneath the vent, resulting in occasional powder spills into the hood and onto the floor).

b. Pellet Press Enclosures

A program has been initiated to upgrade the pellet press enclosures to improve the hood-to-press interfaces to control powder releases, to incorporate improved design features for press operators during routine operations, and to upgrade ventilation air flows.

2. Facilities Improvements

a. Conversion Line 2 Modifications

Conversion Line 2 was modified to provide state-of-the-art instrumentation for improved monitoring and control of processes to decrease downtime associated with equipment plugging and leaks. Concurrently, containment systems were redesigned to take advantage of experience gained on the other conversion lines. Operations were initiated in November 1981 with depleted uranium; enriched uranium operations began in early 1982. Table 4 summarizes airborne data for this line during a period of full production and demonstrates the effectiveness of the modified containment.

b. New Pellet Area Oxidation Furnace and Hood

A new oxidation furnace and hood were installed on Pellet Line 3 to improve the containment surrounding the furnace and increase the efficiency of the oxidation process. When the equipment is in full service, airborne radioactivity concentrations are expected to be lowered at this location.

c. New Rod Scanner

A new device for nondestructively analyzing fuel rods was installed in the rod inspection area in June 1982. The equipment incorporates a more direct method for transferring sources from the shipping containers to the rod scanner, thereby decreasing personnel exposures during these transfers.

d. Incinerator Ash Enclosure

A secondary enclosure was installed at the discharge end of the incinerator to upgrade containment and improve ventilation for ash handling. This improvement has been effective in reducing the number of airborne spikes at this location.

e. Stratification Blender

The containment on the stratification blender was modified to provide ventilation of the entire volume beneath the input hood. An additional enclosure was fabricated for the input hood feed screw to provide containment for removals during enrichment cleanouts and maintenance. These changes have been effective in reducing airborne radioactivity concentrations.

f. Jaw Crusher

A new jaw crusher and associated containment were installed to replace the granulator in the Scrap Recovery Area. This resulted in a significant reduction in airborne radioactivity concentrations (24% MPC during the last half of 1981 versus 9% MPC during the first six months of 1982).

g. Scrubber Relocation

The main Conversion Area scrubbers were transferred to a new location within the Controlled Area. Concurrently, the gas fired heaters were replaced with electric heaters to provide greater reliability.

Table 1

LOCATIONS AND AVERAGE READINGS OF ALL AIR SAMPLERS AVERAGING > 25% MPC

1/1/82 to 6/30/82

Conversion Area	% MPC	Trend (1/1/82 - 6/30/82) (1)	
Line 3 Hot Filter Hood	27%	↑	(2)
Conversion Services			
Blue-M Furnace	34%	↑	(2)
Pellet Area			
Line 1 Pellet Press	33	↑	
Line 2 Pellet Press	30	↓	
Line 4 Pellet Press	30	↑	(2)
Rod Area			
All sample stations averaged less than 25% MPC	N/A		N/A

(1) Only 5 out of 177 samples averaged above 25% MPC

(2) These samples averaged less than 25% MPC during the previous reporting period.

Table 2

NUMBER OF AIRBORNE RADIOACTIVITY SPIKES *

TIME PERIOD	100-200% MPC	201-500% MPC	>500% MPC	TOTAL >100% MPC
1/1/77 - 6/30/77	1,297	249	40	1,586
7/1/77 - 12/31/77	1,216	273	55	1,544
1/1/78 - 6/30/78	656	175	49	880
7/1/78 - 12/31/78	654	160	44	858
1/1/79 - 6/30/79	965	293	79	1,337
7/1/79 - 12/31/79	1,172	389	89	1,650
1/1/80 - 6/30/80	540	177	29	746
7/1/80 - 12/31/80	372	140	42	554
1/1/81 - 6/30/81	270	54	13	337
7/1/81 - 12/31/81	267	83	18	368
1/1/82 - 6/30/82	285	80	17	382

*Based upon approximately 101,000 samples each semi-annual period.

Table 3

HIGHEST FREQUENCY SPIKE STATIONS (>100% MPC)

1/1/82 to 6/30/82

STATION	NUMBER OF SHIFT SAMPLES >100% MPC	% OF TOTAL	TREND 1ST HALF 1982 VS. 1981
Line 1 Pellet Press	24	6	↑
Line 2 Pellet Press	23	6	↑
Line 4 Pellet Press	19	5	↑
Conv. Line 3 Feed End Top	16	4	↑
Conv. Line 4 Feed End Bottom Vent Star	15	4	↑
Blue M Furnaces	15	4	↑
Incinerator Output	15	4	↓
Line 3 Pellet Press	14	4	↓

TABLE 4

CONVERSION LINE 2 AIRBORNE SUMMARY

MODIFIED FEED END CONTAINMENT

WEEK [1] ENDING	% MPC FEED END TOP	FEED END BOTTOM	FEED END BOTTOM VENT. STAR	# SPIKES >100% MPC
5/1/82	9	9	10	0
5/8/82	8	9	5	0
5/15/82	21	11	8	0
5/22/82	9	8	8	0
5/29/82	13	13	10	0
6/5/82	13	12	14	0
6/12/82	9	9	10	0
6/19/82	19	15	12	0
6/26/82	17	13	48 [2]	3
Average	13	11	14	Total 3

(1) Enriched uranium processed during this period following line modifications in late 1981 and early 1982.

(2) Enrichment cleanout and maintenance activities.

Table 5

CONVERSION LINE SUMMARY (FIRST HALF OF 1982 VS. 1981)

OPERATING LINES ONLY

	Line 2			Line 3			Line 4			Line 5		
	1982	1981	Trend	1982	1981	Trend	1982	1981	Trend	1982	1981	Trend
Feed End Top Average	17%	3%	↑	20%	22%	↓	10%	18%	↓	13%	12%	↑
Feed End Bottom Average	11%	13%	↓	13%	22%	↓	12%	(2)	—	9%	8%	↑
Vent Star Average	12%	3%	↑	27%	16%	↑	25%	20%	↑	17%	22%	↓
Calciner Recycle Average	20%	(2)	—	23%	(1)	—	5%	13%	↓	23%	16%	↑
Feed End Average	15%	6%	↑	19%	20%	↓	13%	16%	↓	16%	15%	
Product End Average	14%	7%	↑	14%	10%	↑	13%	14%	↓	12%	13%	↓
Number of Airborne Spikes >100% MPC (Feed End)	22	4(4)	↑	53	54(4)	↓	21	16(4)	↑	28	39(4)	↓
Number of Airborne Spikes >100% MPC (Product End)	13	10(4)	↑	18	7(4)	↑	12	19(4)	↓	11	10(4)	↑
Production Throughput	72	0(3)	↑	80	84(3)	↓	105	102(3)	↑	26	105(3)	↓

(1) Hot Filter Air Sample

(2) No Data for This Sample

(3) Semiannual Average

(4) Last Half of 1981

Table 6

EFFLUENT DISCHARGES, uCi TOTAL URANIUM

TIME PERIOD	GASEOUS EFFLUENT		LIQUID EFFLUENT	
	uCi	Kg U*	uCi	Kg U*
7/1/77 - 12/31/77	2,447	1.22	80,568	40.3
1/1/78 - 6/30/78	1,519	0.76	62,822	31.4
7/1/78 - 12/31/78	1,142	0.57	97,033	48.5
1/1/79 - 6/30/79	1,040	0.52	107,745	53.9
7/1/79 - 12/31/79	1,008	0.50	198,090	99.0
1/1/80 - 6/30/80	537	0.27	97,699	48.8
7/1/80 - 12/31/80	659	0.33	46,204	23.1
1/1/81 - 6/30/81	462	0.22	55,000	27.5
7/1/81 - 12/31/81	485	0.24	75,751	37.9
1/1/82 - 6/30/82	502	0.25	70,003	35.0

* Assumes a specific activity of 2 uCi/gram.

Table 7

PERSONNEL EXPOSURES EXCEEDING 40 MPC-HOURS IN ANY

SEVEN CONSECUTIVE DAY PERIOD

TIME PERIOD	NUMBER OF EXPOSURES
1/1/77 - 6/30/77	13
7/1/77 - 12/31/77	9
1/1/78 - 6/30/78	3
7/1/78 - 12/31/78	3
1/1/79 - 6/30/79	3
7/1/79 - 12/31/79	2
1/1/80 - 6/30/80	2
7/1/80 - 12/31/80	0
1/1/81 - 6/30/81	0
7/1/81 - 12/31/81	0
1/1/82 - 6/30/82	0

Table 8

NRC INSPECTIONS

TIME PERIOD	DEFICIENCIES HP & CRIT.	DEFICIENCIES SAFEGUARDS	INFRACTIONS HP & CRIT.	INFRACTIONS SAFEGUARDS	NUMBER OF INSPECTIONS*	TOTAL POINTS
1/1/76 - 6/30/76	2	1	4	0	3	46
7/1/76 - 12/31/76	1	1	2	0	3	24
1/1/77 - 6/30/77	1	1	7	1	8	84
7/1/77 - 12/31/77	5	6	6	0	13	82
1/1/78 - 6/30/78	0	1	5	0	8	52
7/1/78 - 12/31/78	0	6	2	0	10	32
1/1/79 - 6/30/79	3	8	0	2	10	42
7/1/79 - 12/31/79	1	2 (1)	3 (2)	0	8	38
1/1/80 - 6/30/80	2	9	0	0	10	22
7/1/80 - 12/31/80	0	3	0	2	8	26

ENFORCEMENT CATEGORY (3)

TIME PERIOD	IV HP&CR.	IV SGDS.	V HP&CR.	V SGDS.	VI HP&CR.	VI SGDS.	OTHERS HP&CR.	OTHERS SGDS.	NUMBER OF INSPECTIONS*	TOTAL POINTS
1/1/81 - 6/30/81	0	0	0	4	0	0	0	0	10	NA
7/1/81 - 12/31/81	0	0	3	2	2	1	0	0	10	NA
1/1/82 - 6/30/82	0	0	1 (4)	3 (5)	0	0	0	0	12	NA

*Includes Health Physics, Emergency Planning, Criticality, Security, Environmental Monitoring, Safeguards and Management Review.

(1) One deficiency was a repeat.

(2) Includes a transportation infraction which occurred in 1979 but was not imposed until 1980.

(3) NRC IE Enforcement Criteria

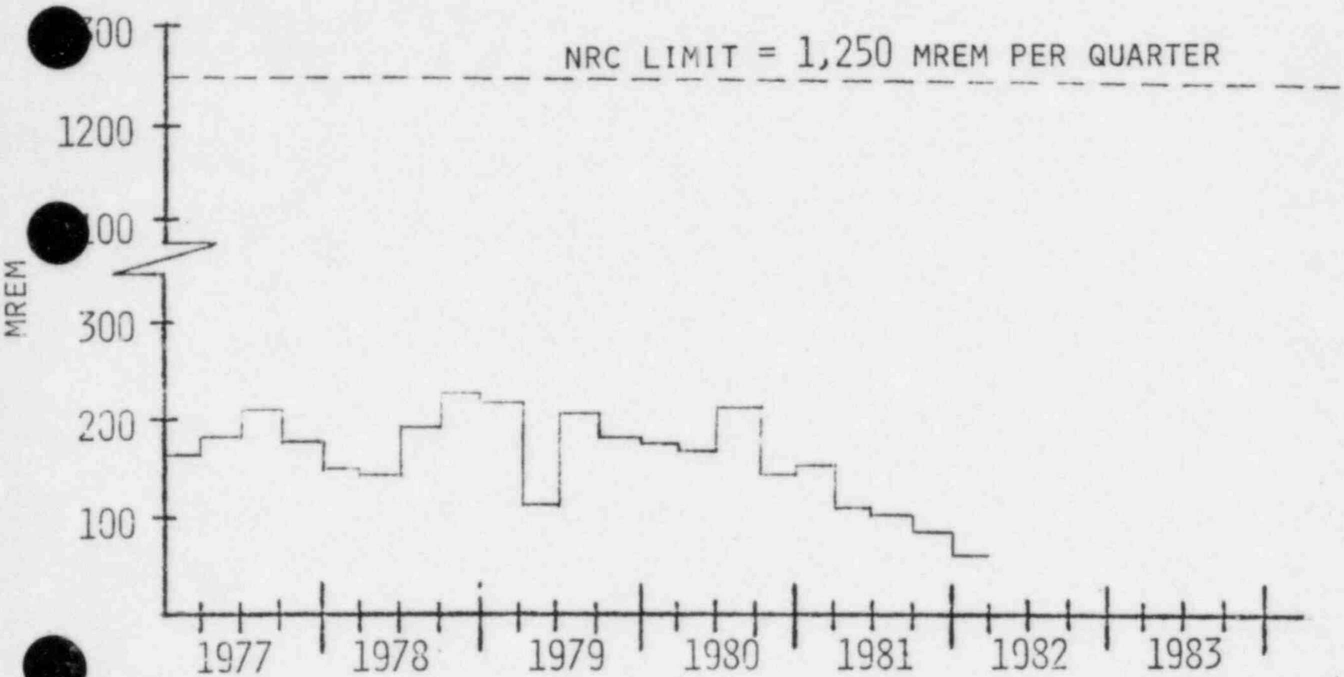
(4) Fire Protection Violation

(5) One Violation was a Repeat

AVERAGE QUARTERLY EXTERNAL DOSE PER EMPLOYEE

AREA: CONVERSION

WHOLE BODY DOSE



SKIN DOSE

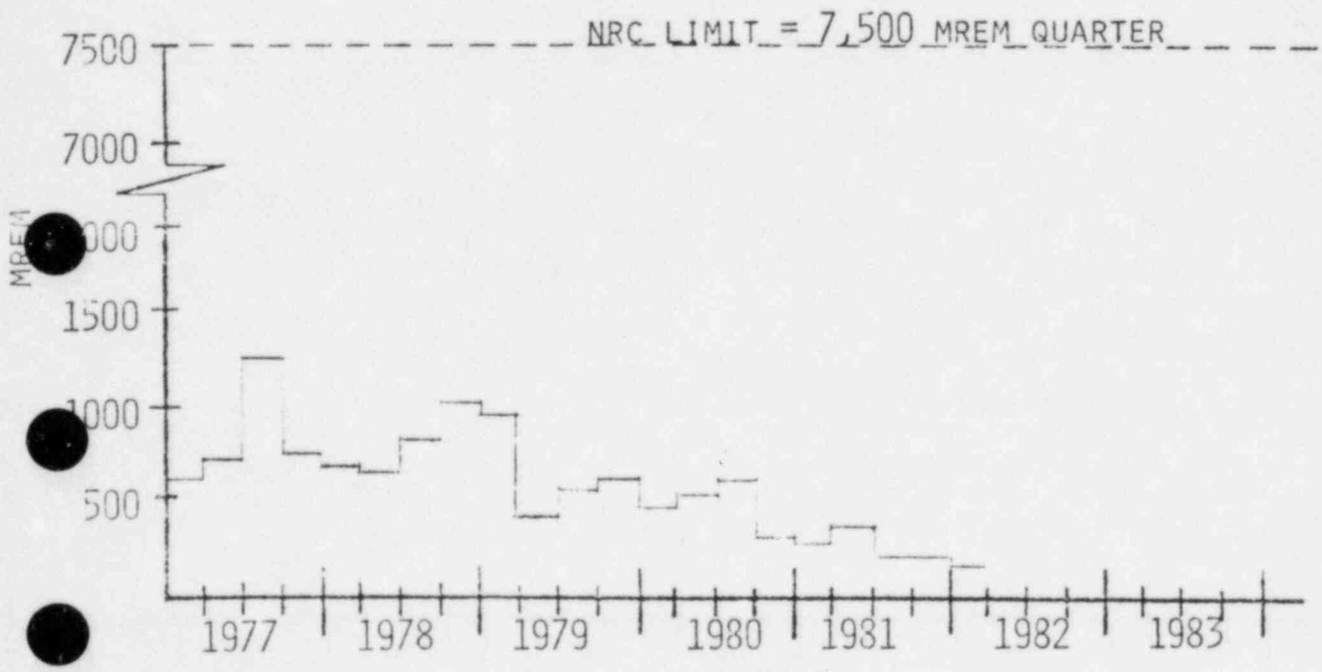


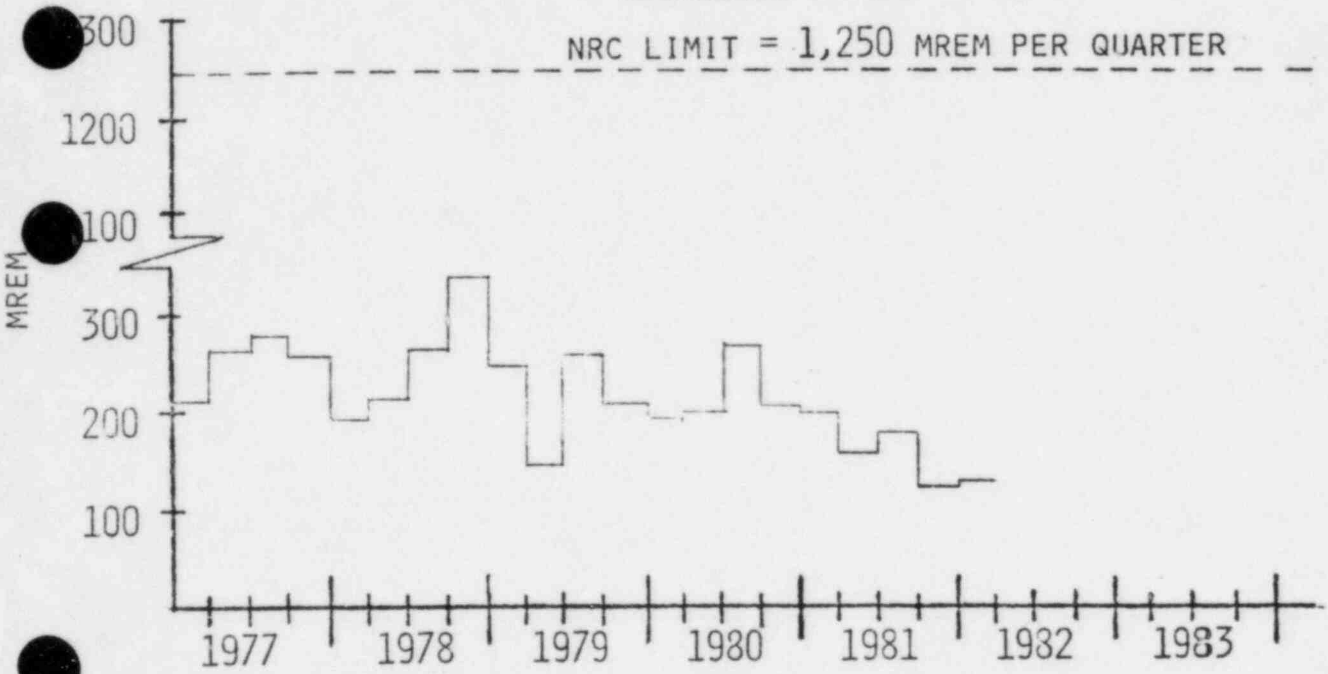
FIGURE 1

AVERAGE QUARTERLY EXTERNAL DOSE

PER EMPLOYEE

AREA: PELLET

WHOLE BODY DOSE



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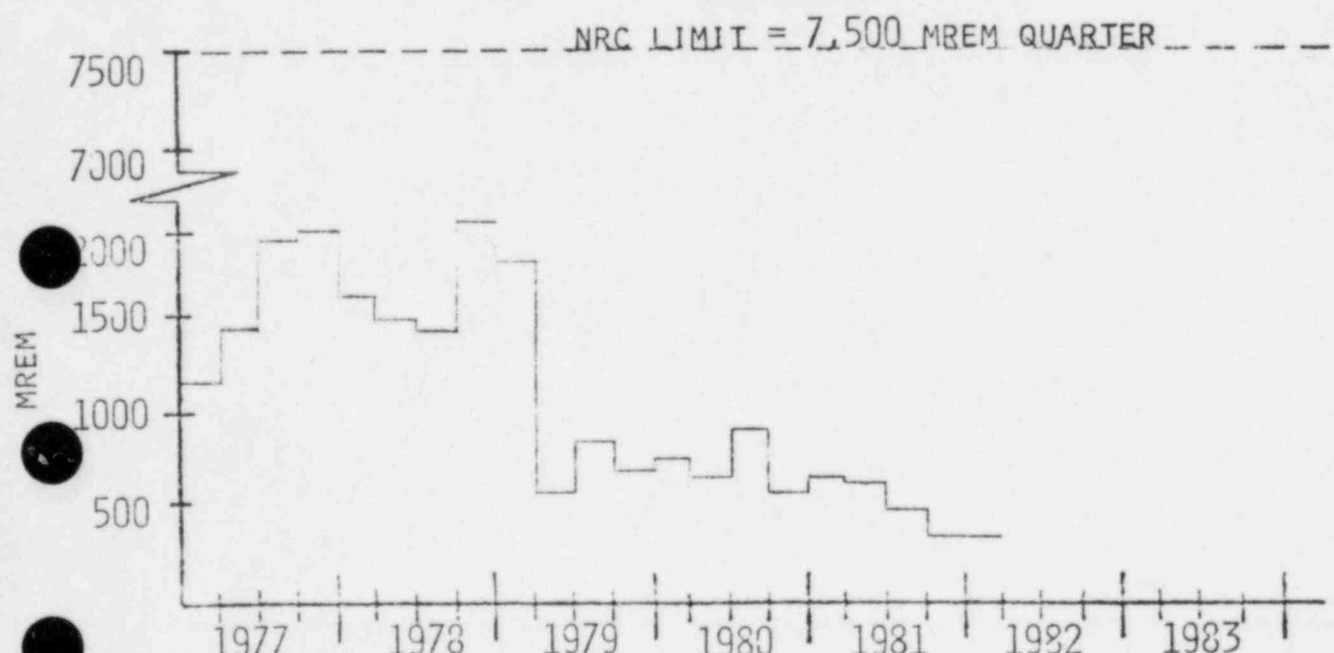


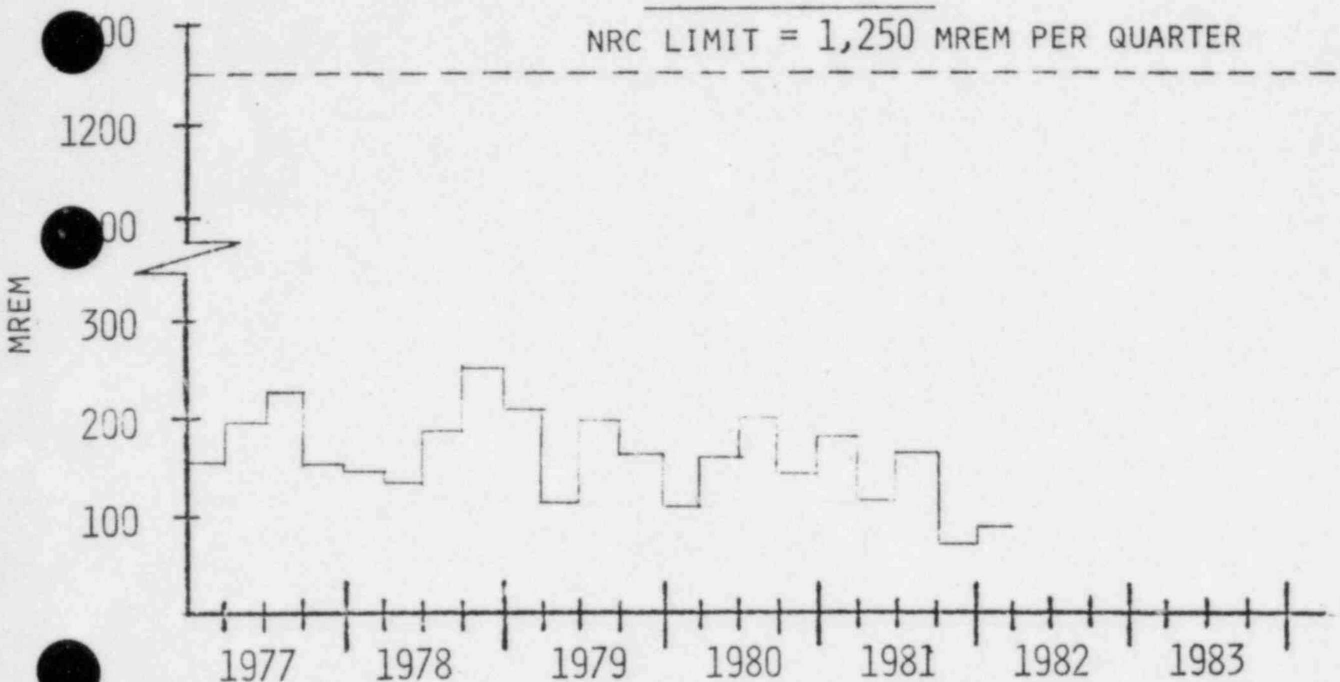
FIGURE 2

AVERAGE QUARTERLY EXTERNAL DOSE

PER EMPLOYEE

AREA: ROD

WHOLE BODY DOSE



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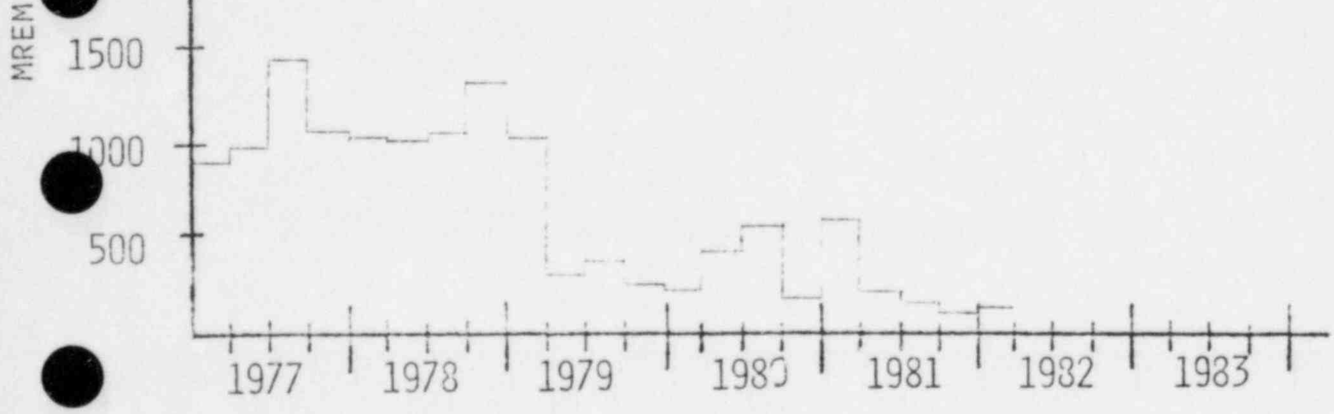


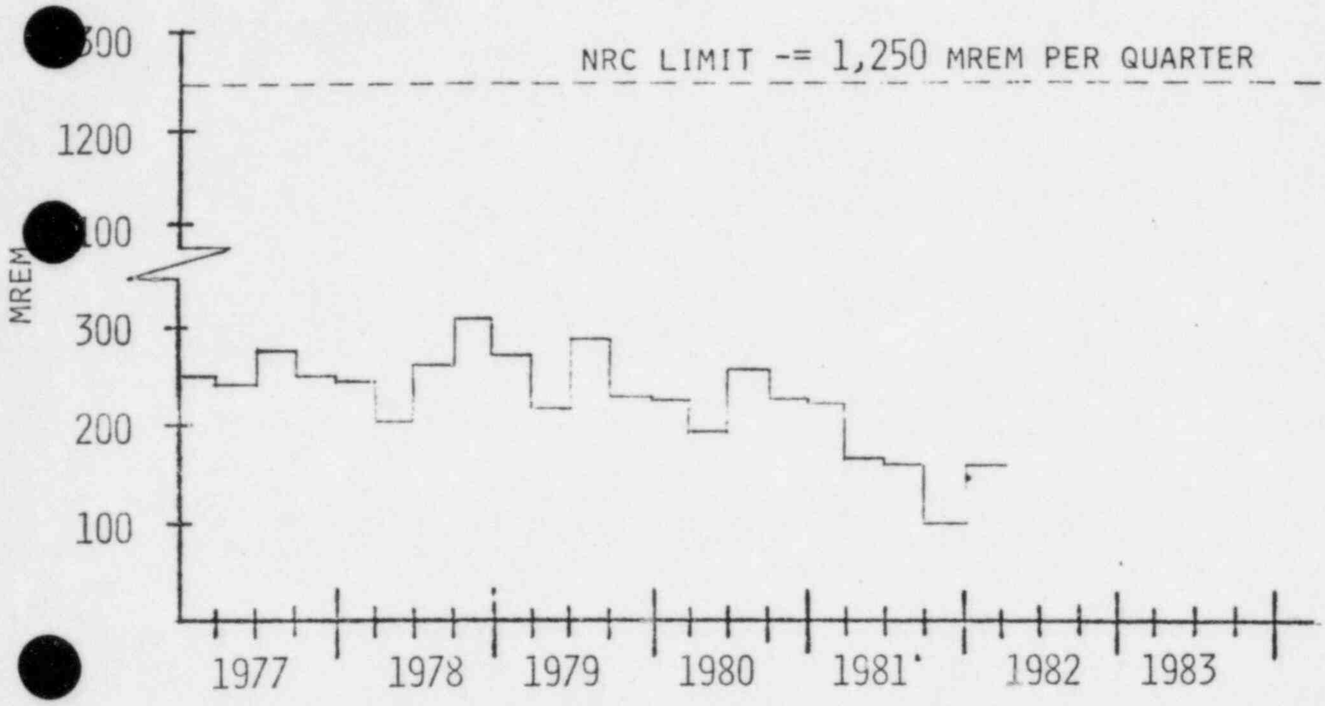
FIGURE 3

AVERAGE QUARTERLY EXTERNAL DOSE

PER EMPLOYEE

AREA: Q.C. CHEMICAL

WHOLE BODY DOSE



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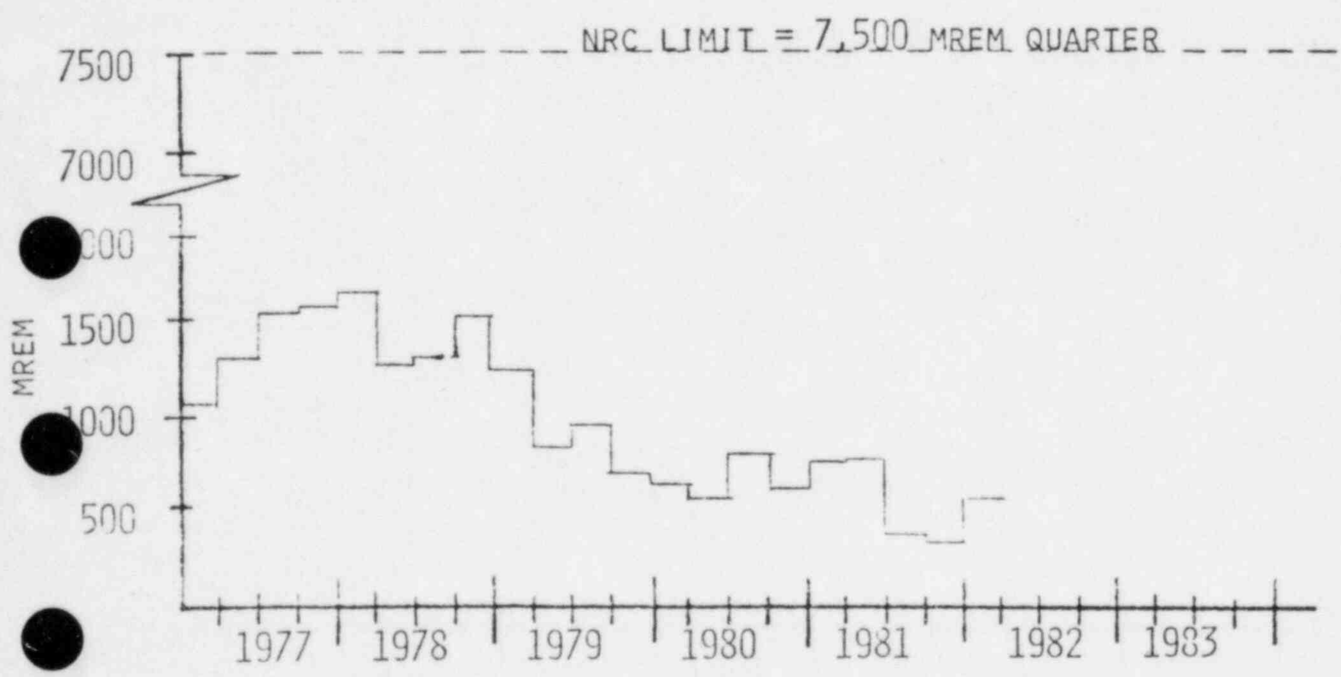


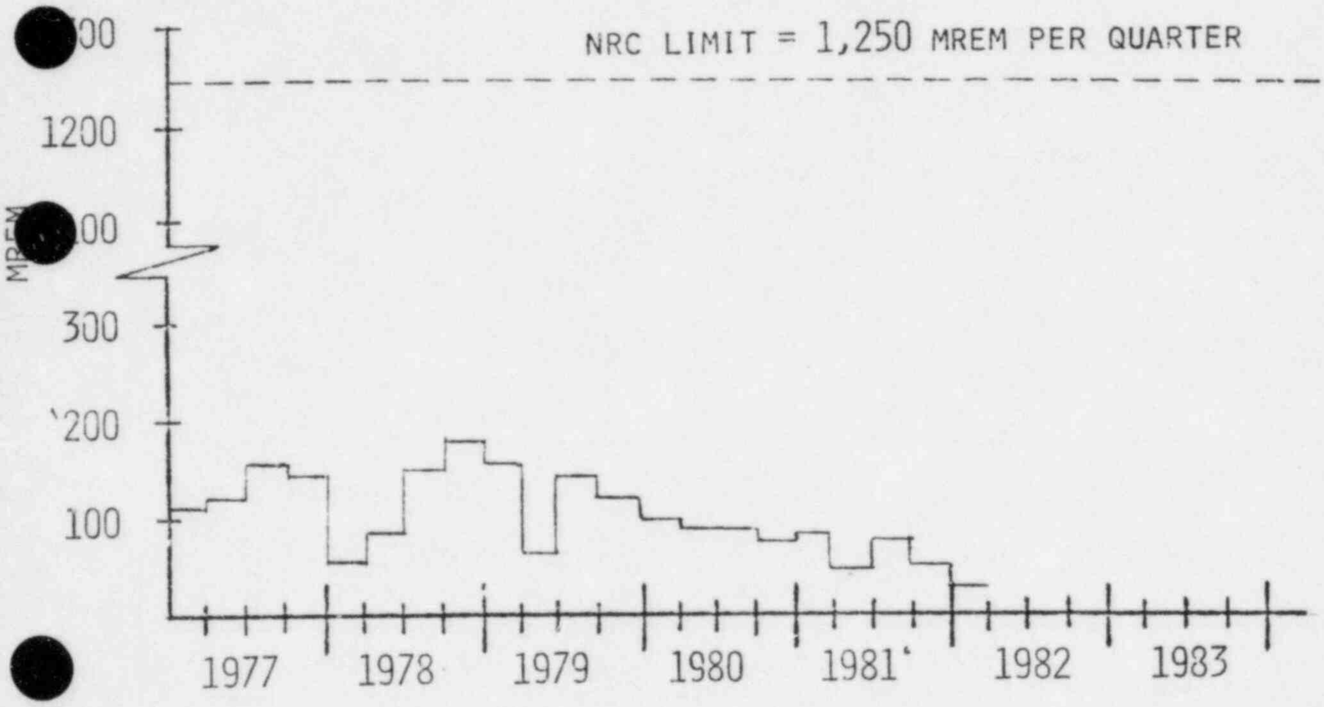
FIGURE 4

AVERAGE QUARTERLY EXTERNAL DOSE

PER EMPLOYEE

AREA: MAINTENANCE

WHOLE BODY DOSE



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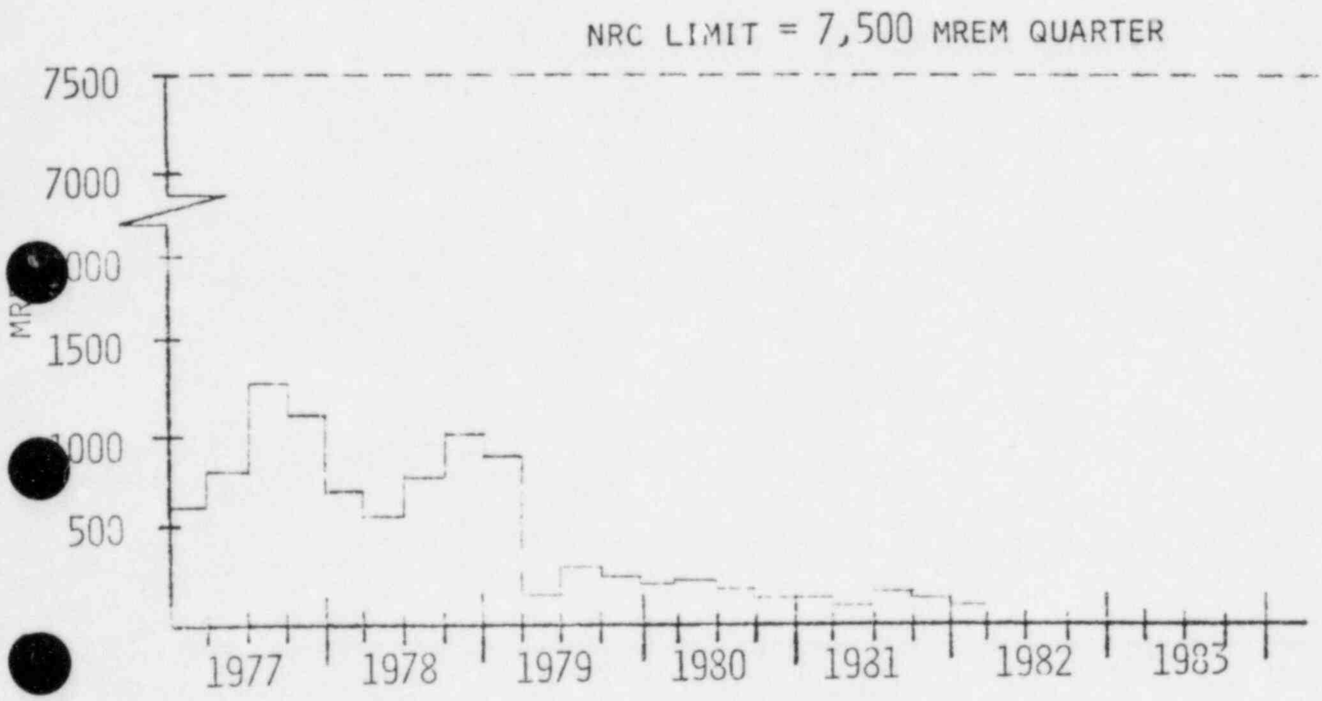


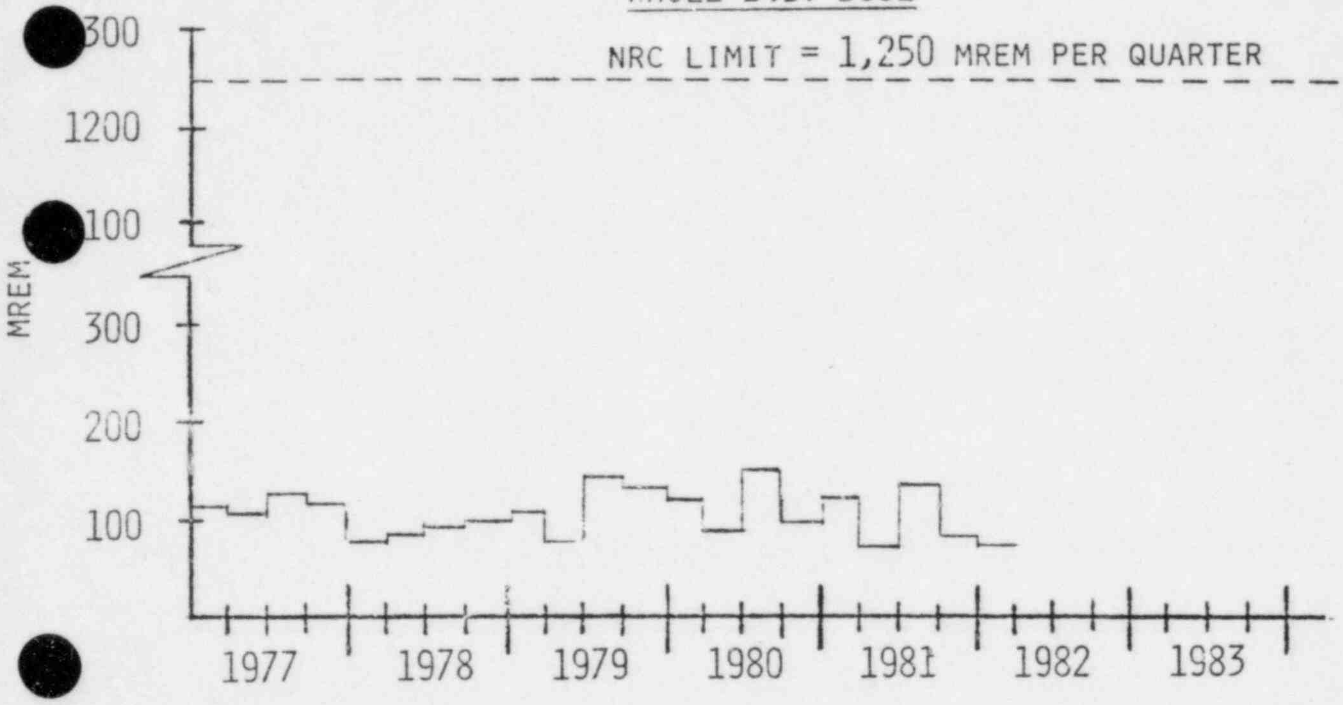
FIGURE 5

AVERAGE QUARTERLY EXTERNAL DOSE

PER EMPLOYEE

AREA: ROD INSPECTION

WHOLE BODY DOSE



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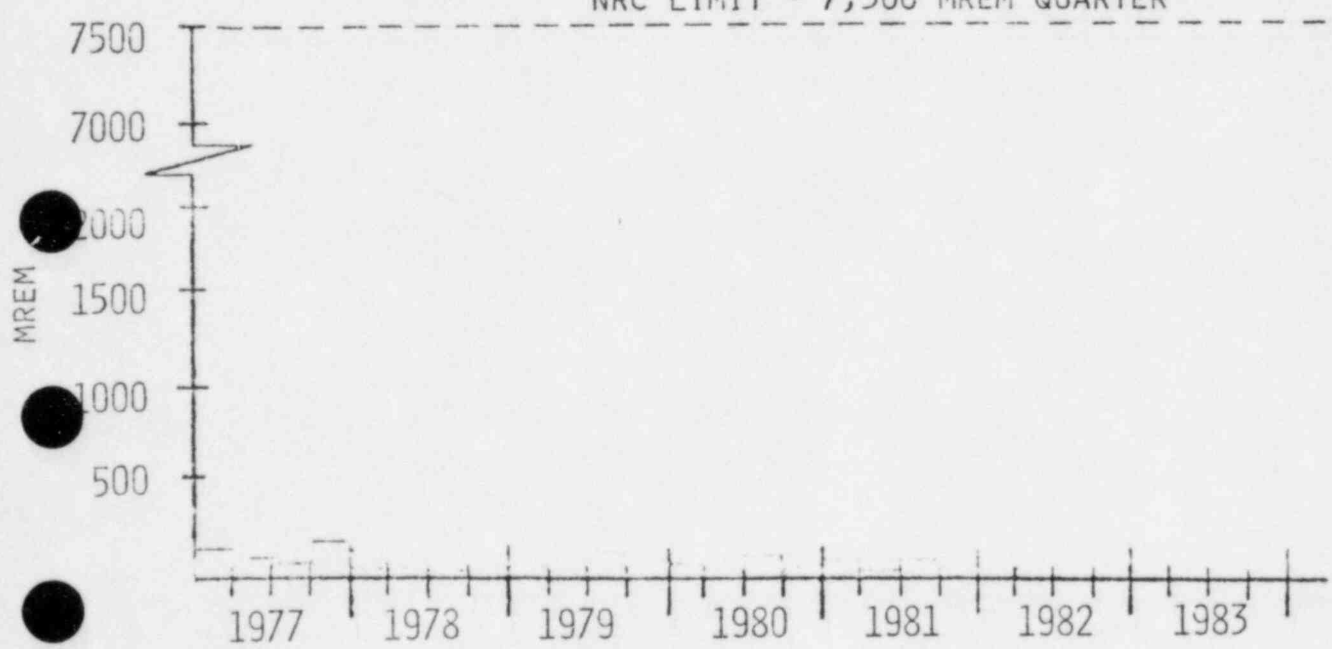


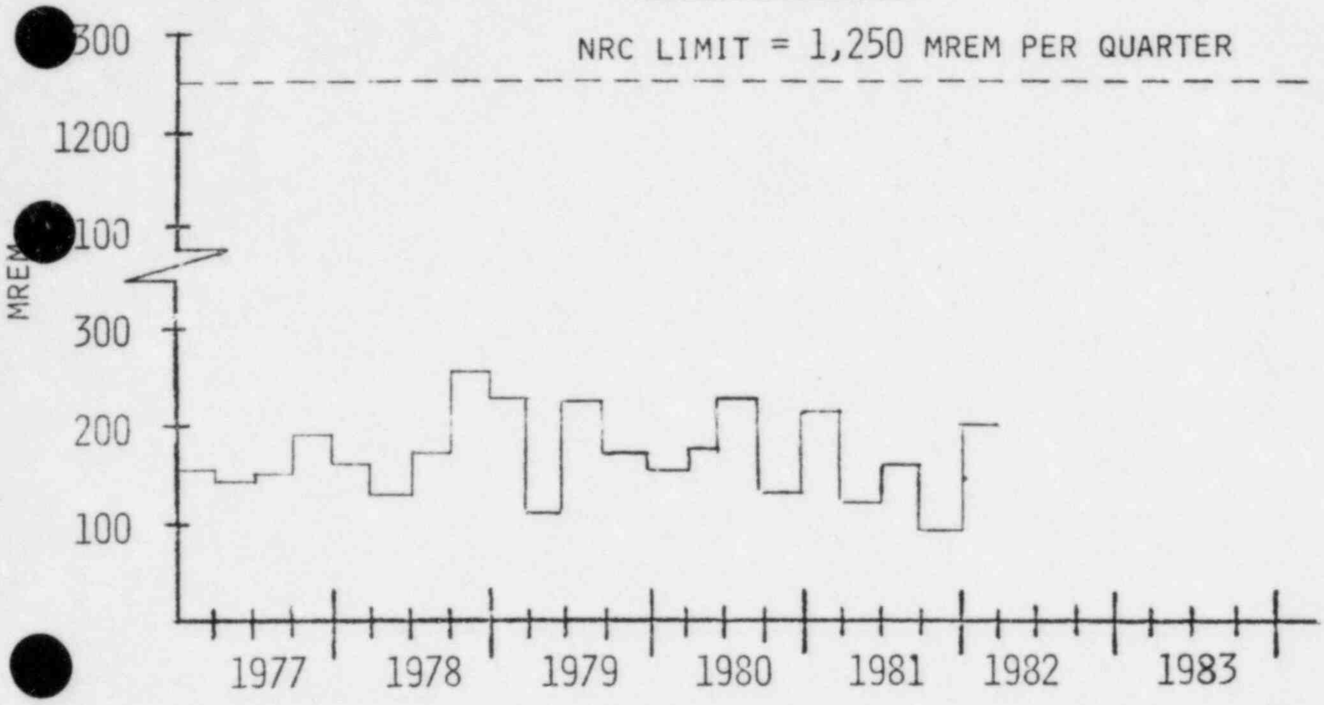
FIGURE 6

AVERAGE QUARTERLY EXTERNAL DOSE

PER EMPLOYEE

AREA: FINAL INSPECTION

WHOLE BODY DOSE



SKIN DOSE

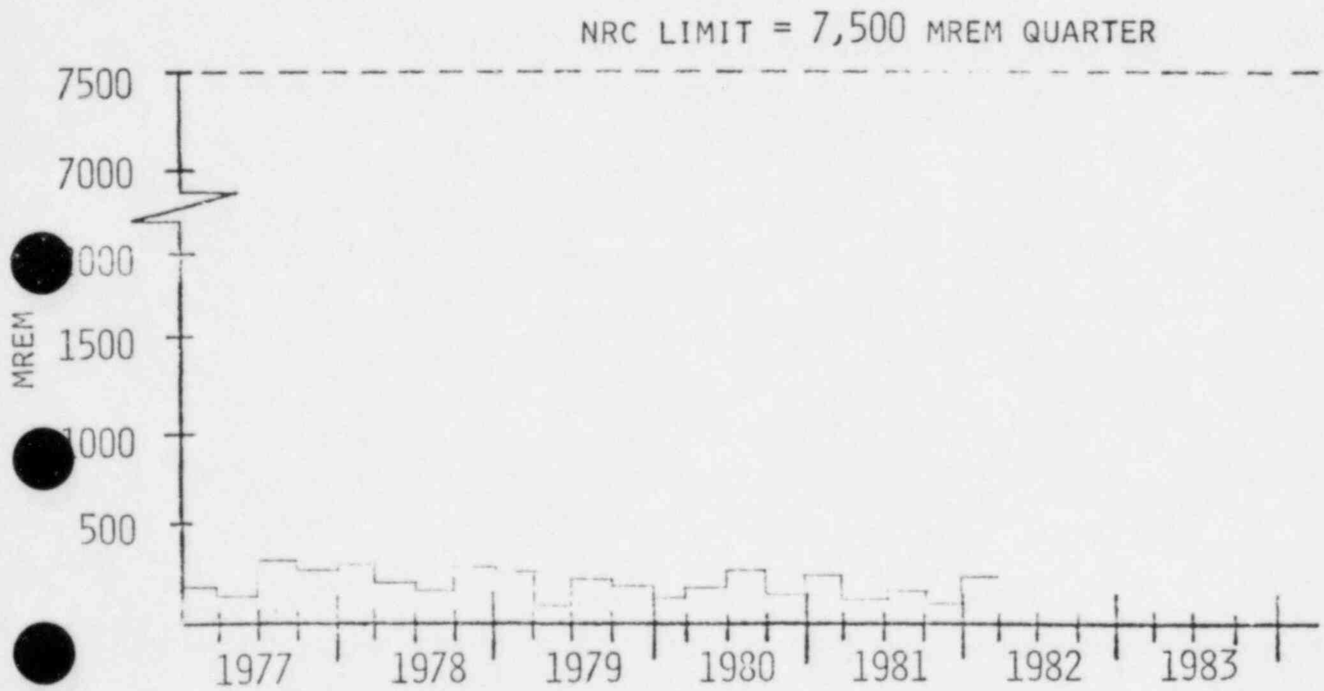


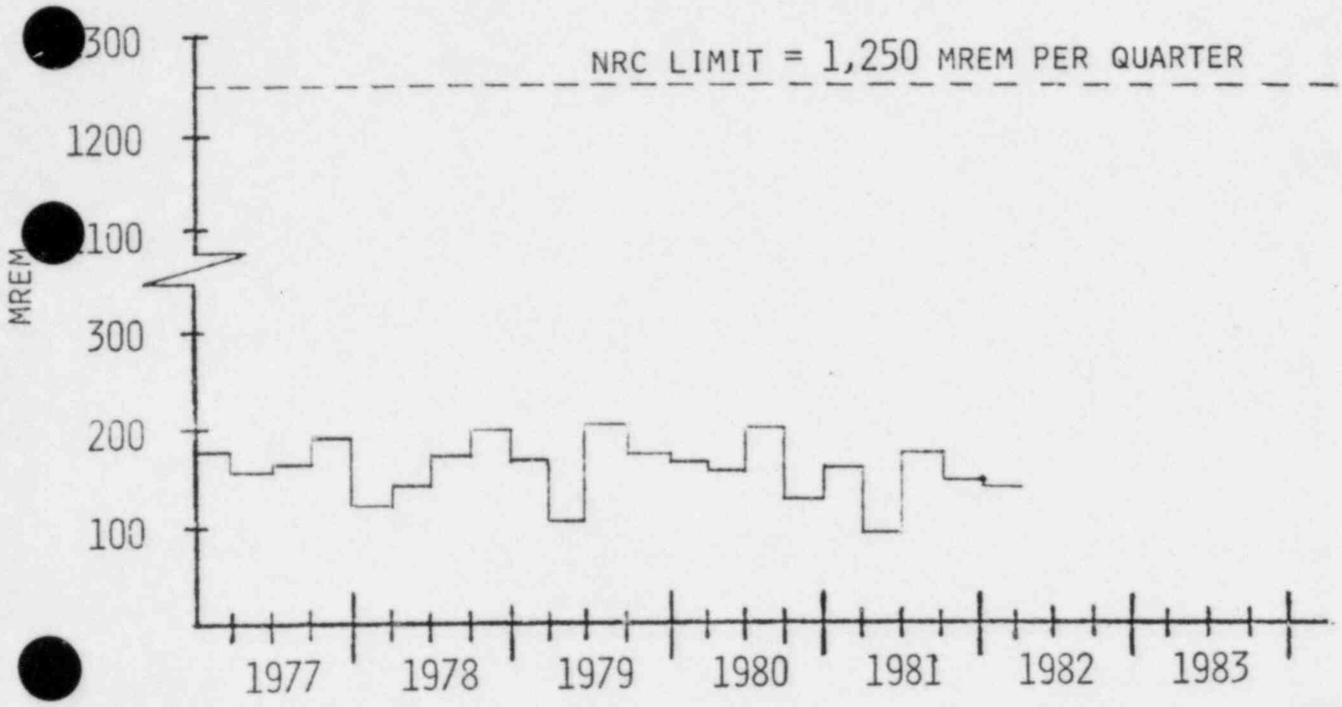
FIGURE 7

AVERAGE QUARTERLY EXTERNAL DOSE

PER EMPLOYEE

AREA: FINAL ASSEMBLY

WHOLE BODY DOSE



SKIN DOSE

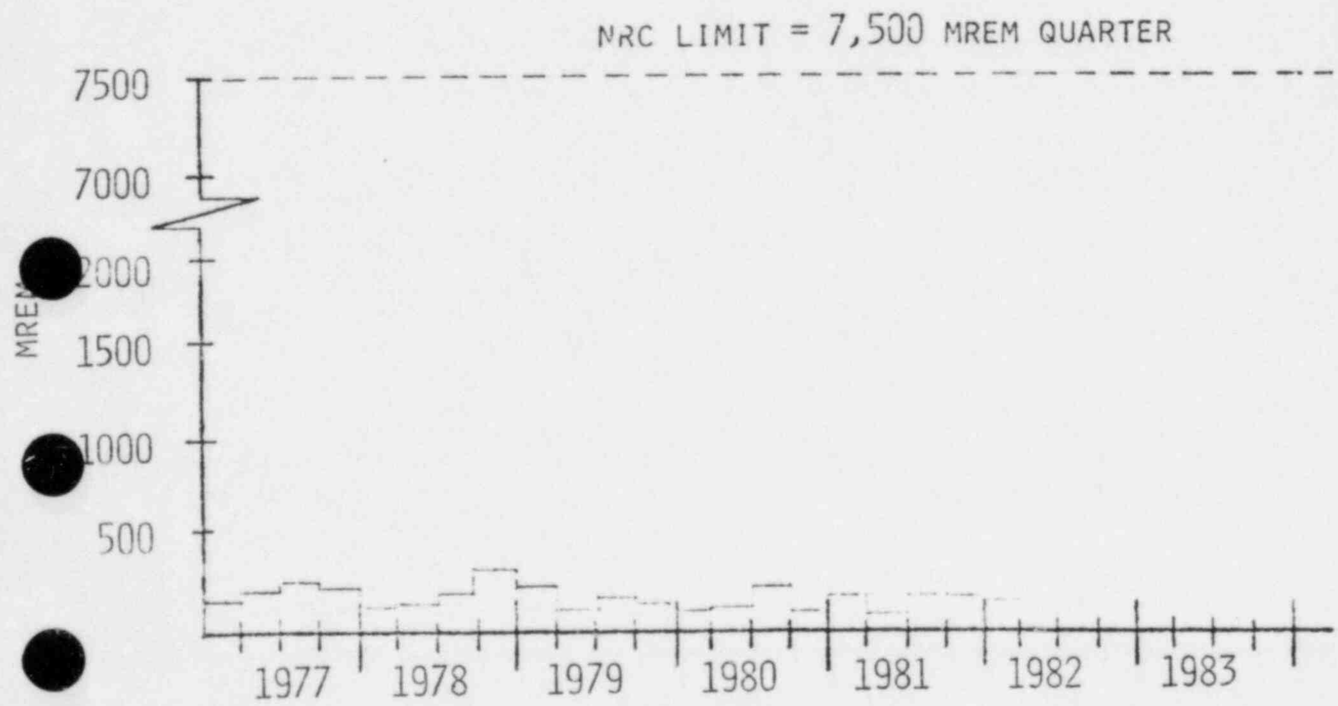


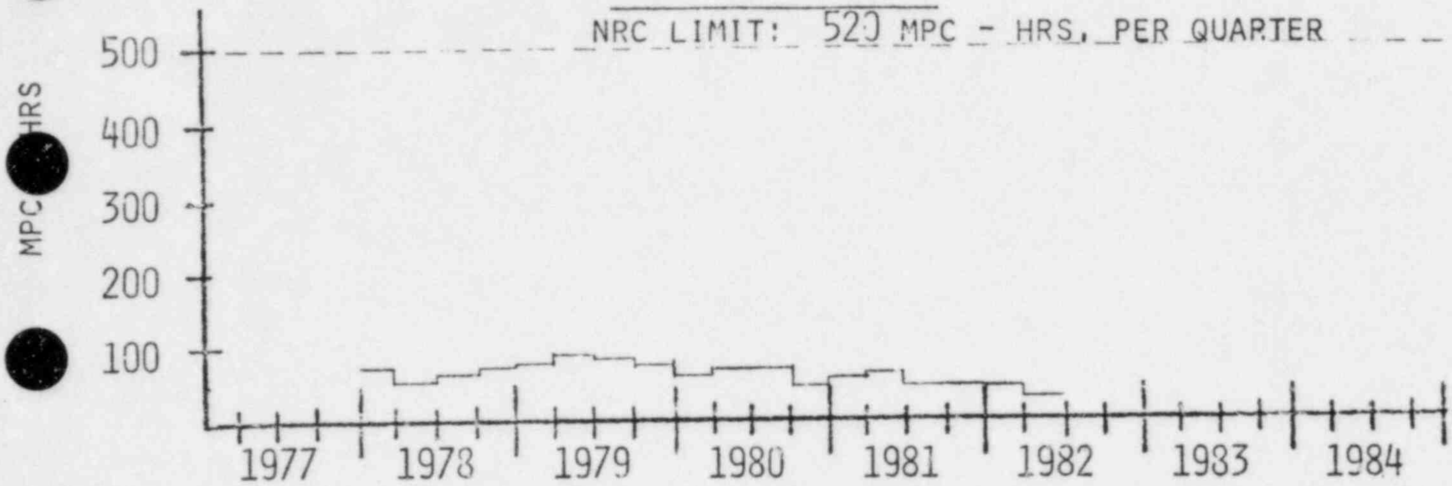
FIGURE 8

AVERAGE QUARTERLY EXTERNAL EXPOSURE

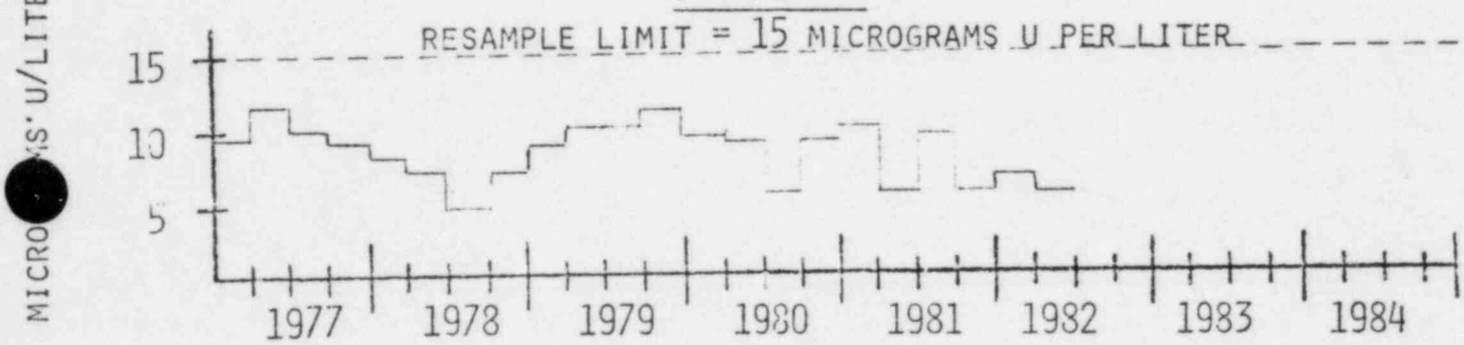
AND BIOASSAY RESULTS PER EMPLOYEE

AREA: CONVERSION

INTERNAL EXPOSURE



URINALYSIS



IN-VIVO

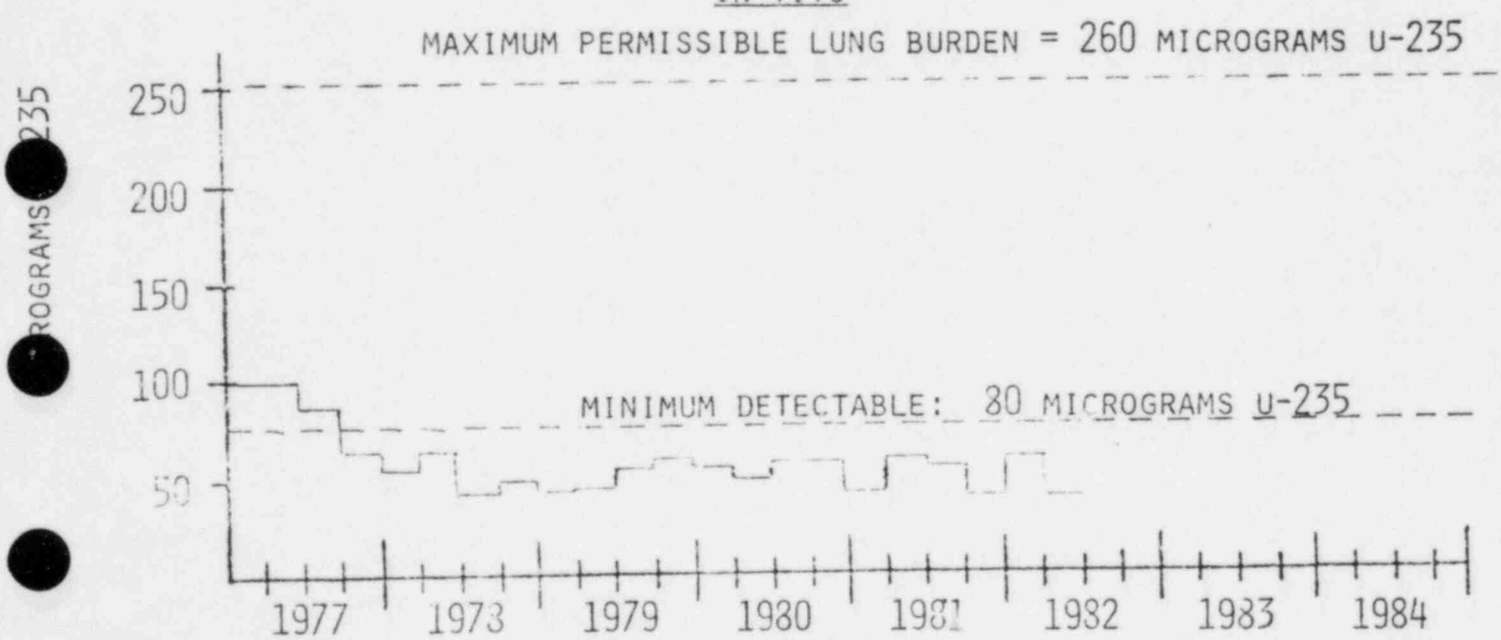


FIGURE 9

AVERAGE QUARTERLY INTERNAL EXPOSURE

AND BIOASSAY RESULTS PER EMPLOYEE

AREA: PELLET

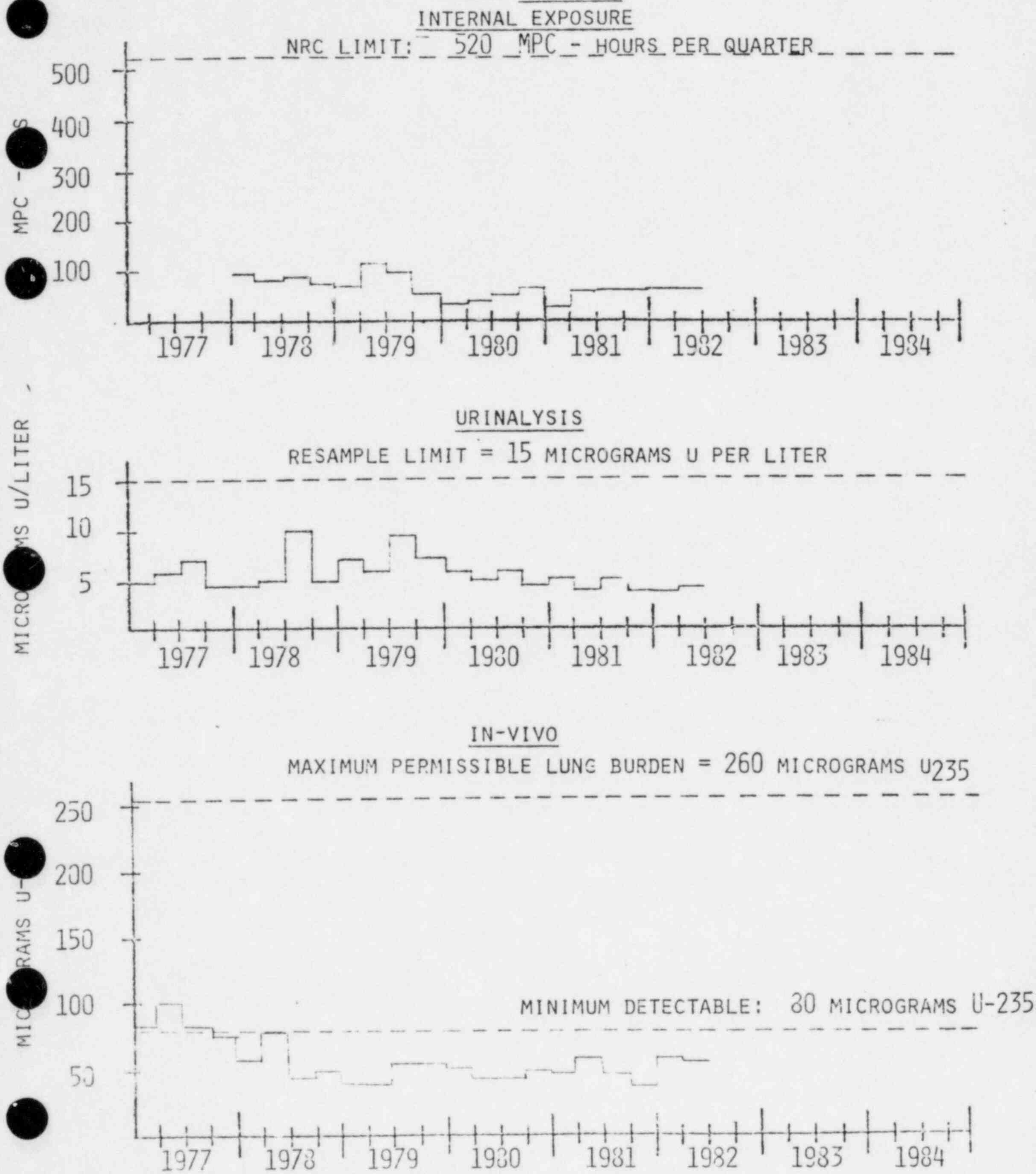


FIGURE 10

AVERAGE QUARTERLY INTERNAL EXPOSURE AND BIOASSAY RESULTS PER EMPLOYEE

AREA: ROD

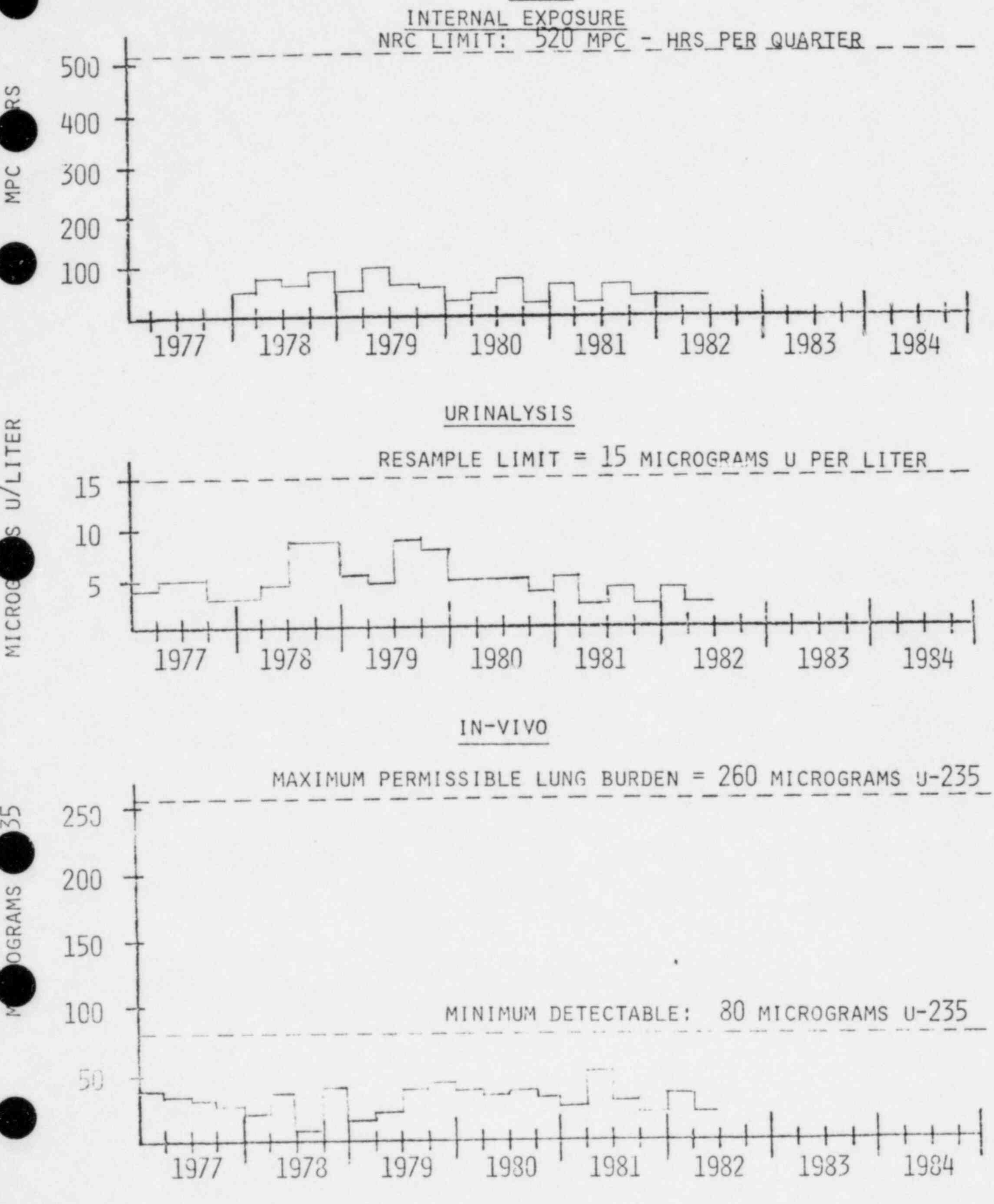


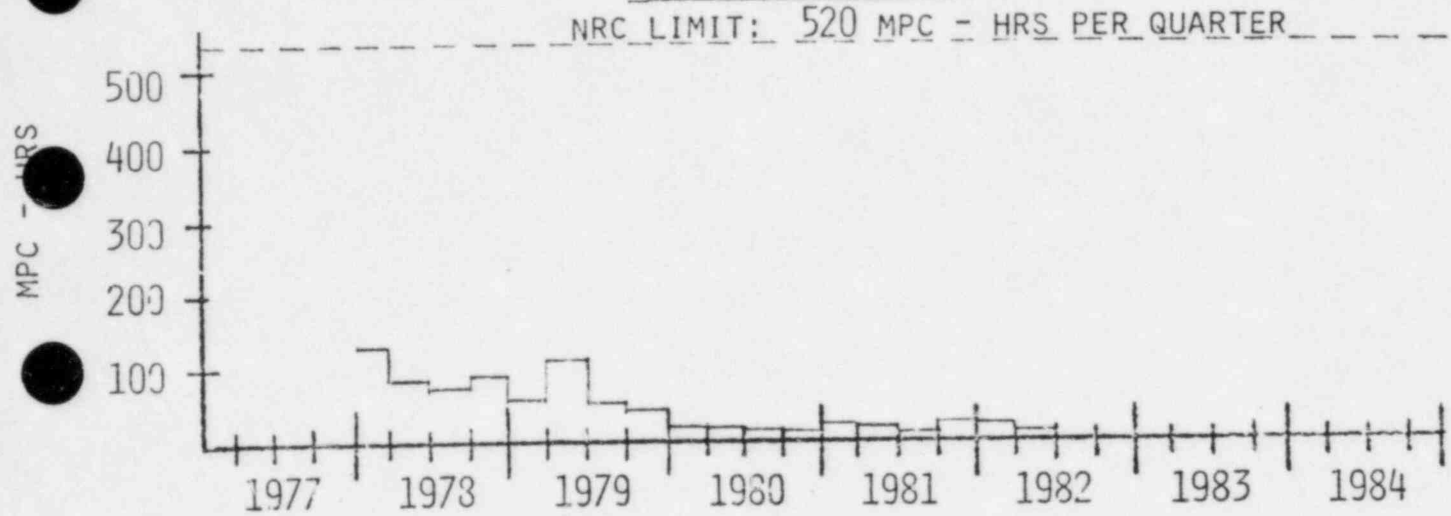
FIGURE 11

AVERAGE QUARTERLY INTERNAL EXPOSURE

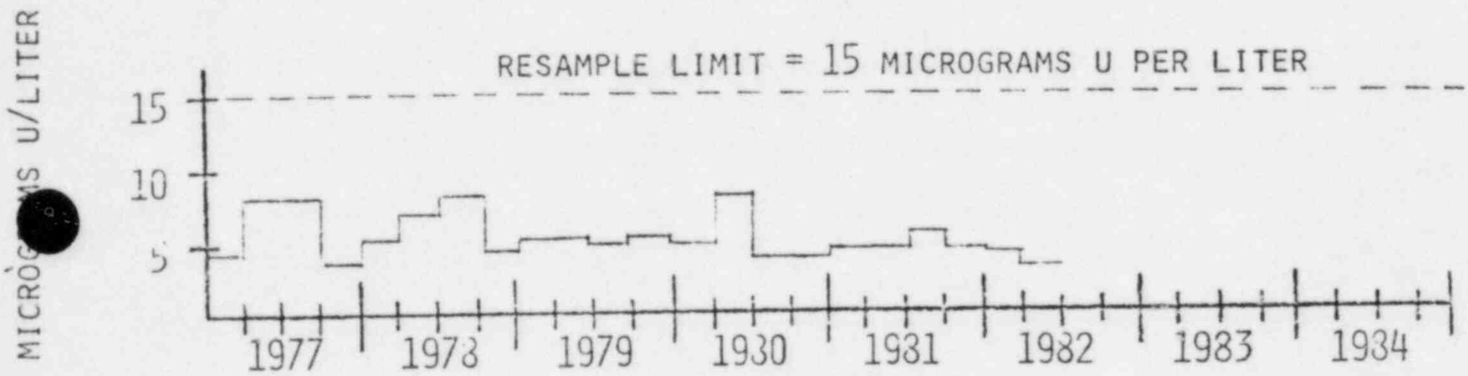
AND BIOASSAY RESULTS PER EMPLOYEE

AREA: Q.C. CHEMICAL

INTERNAL EXPOSURE



URINALYSIS



IN-VIVO

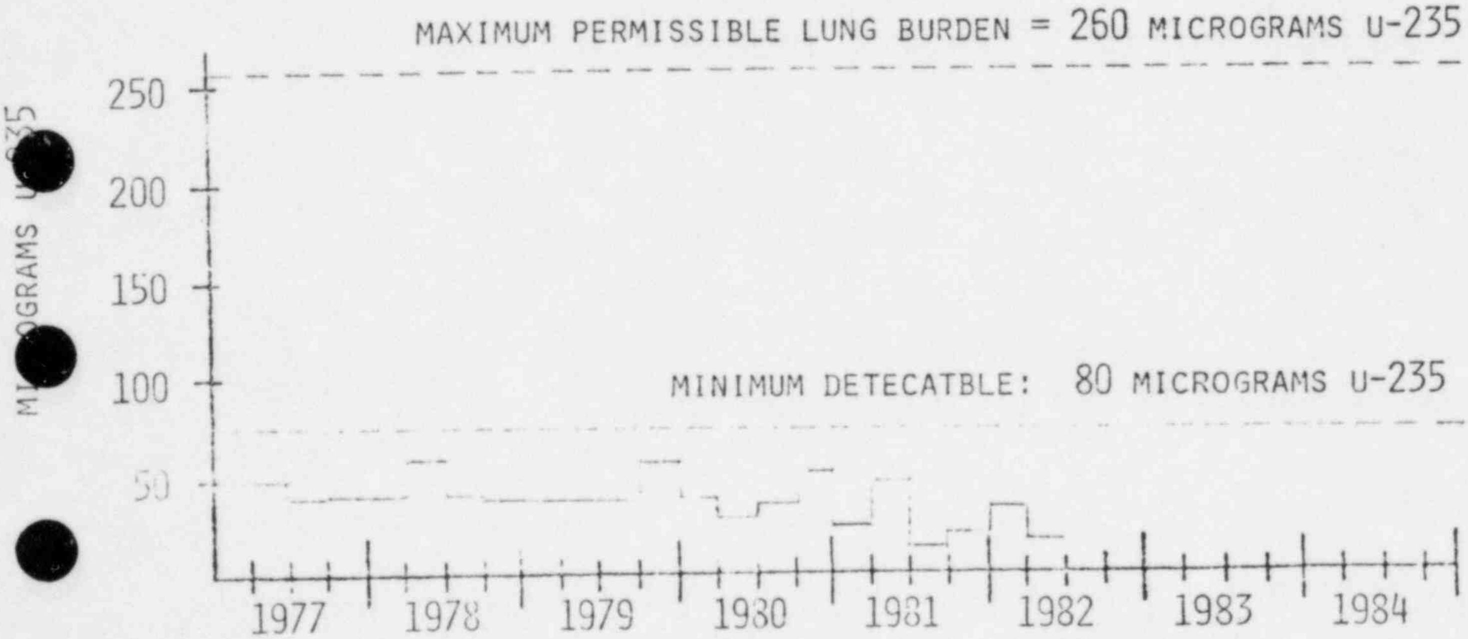


FIGURE 12

AVERAGE QUARTERLY INTERNAL EXPOSURE

AND BIOASSAY RESULTS PER EMPLOYEE

AREA: MAINTENANCE

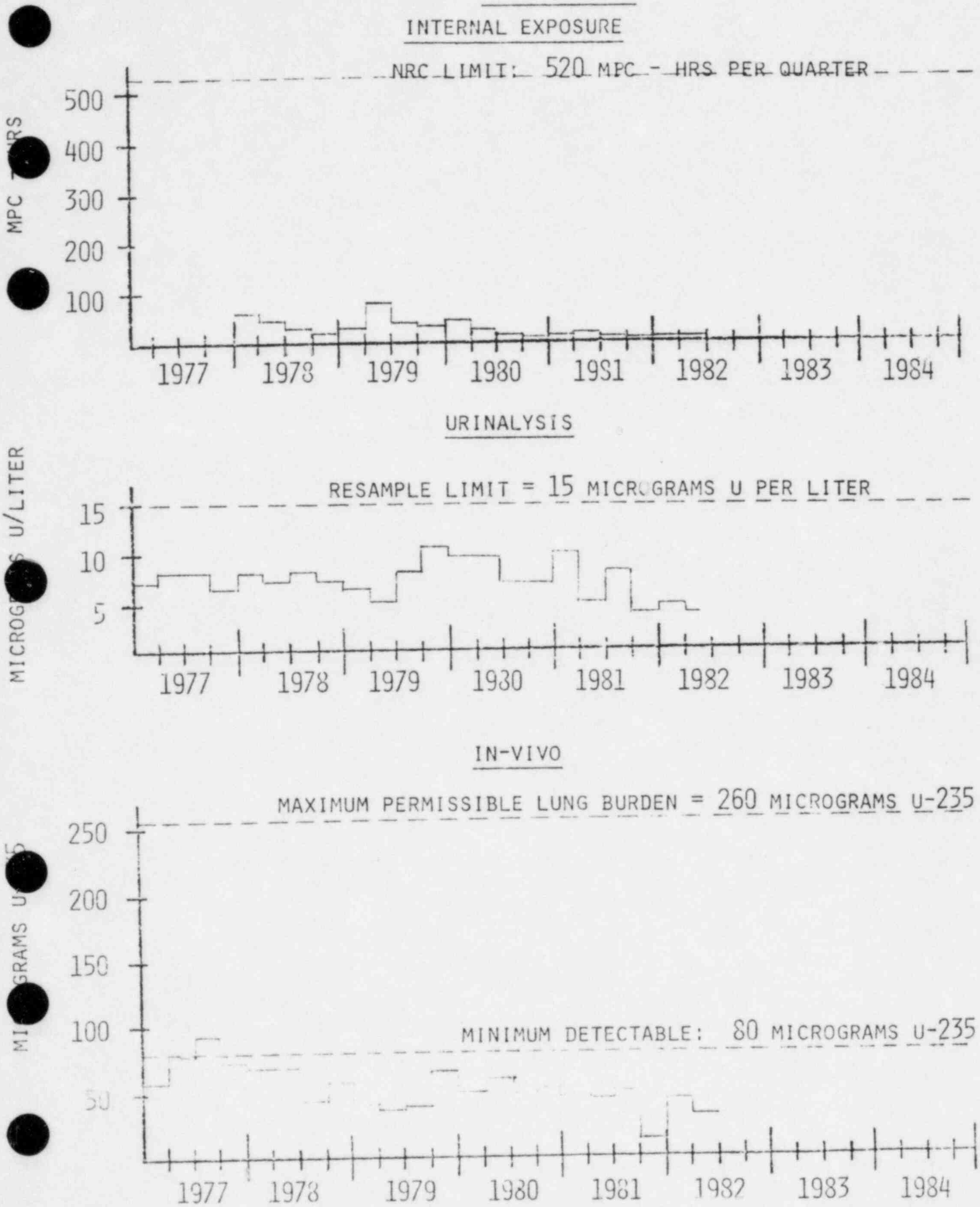


FIGURE 13

AVERAGE AREA AIR CONCENTRATIONS

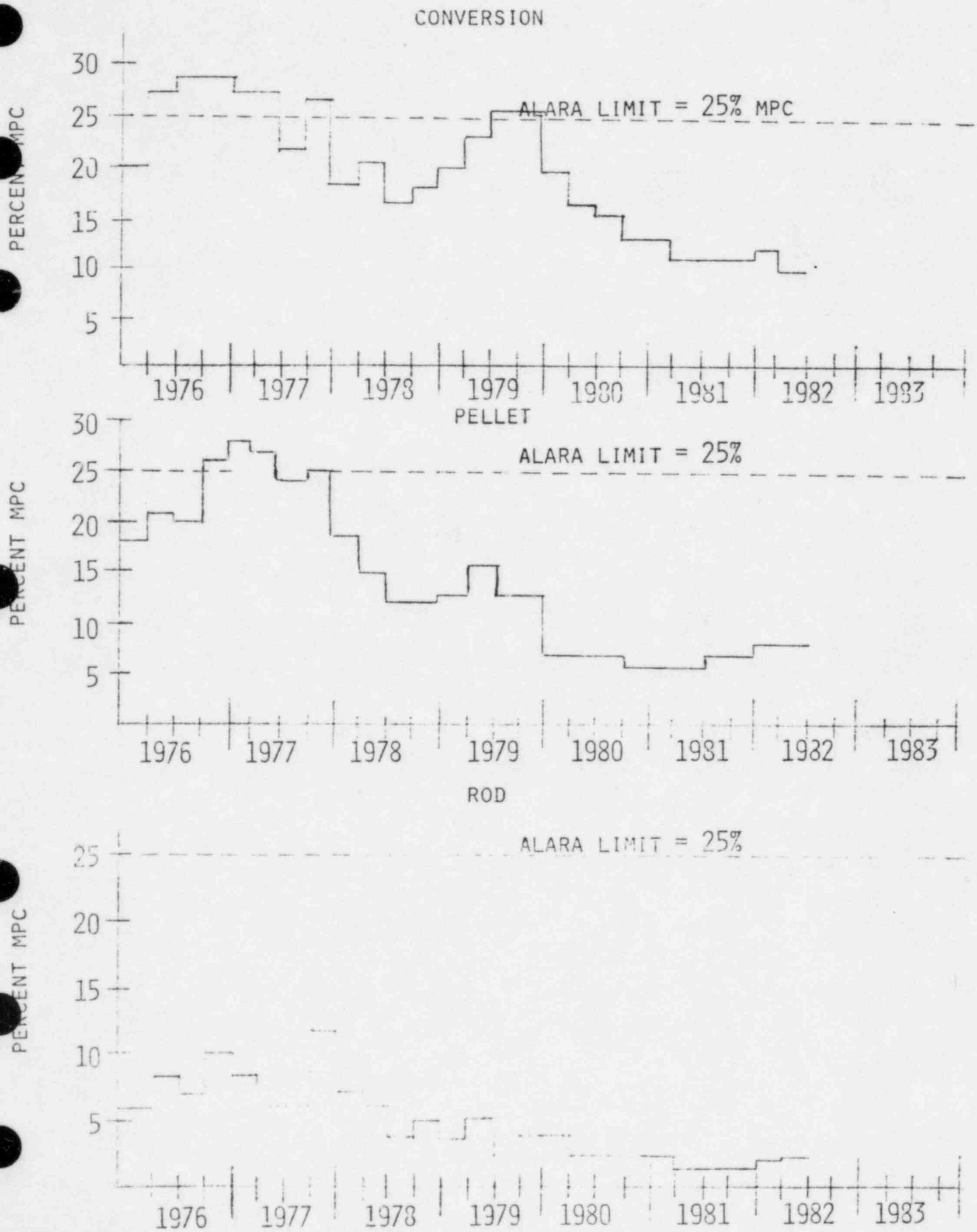


FIGURE 14

NUMBER OF AIR SAMPLES

> 100 % MPC

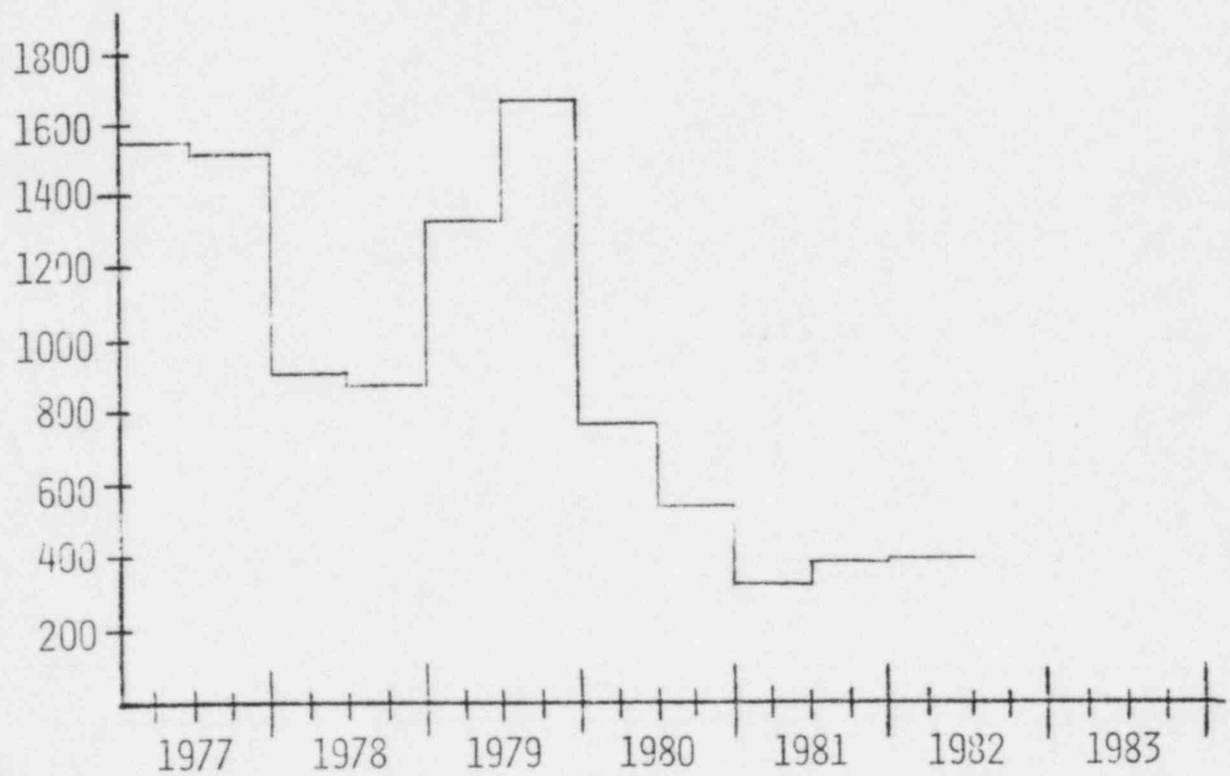
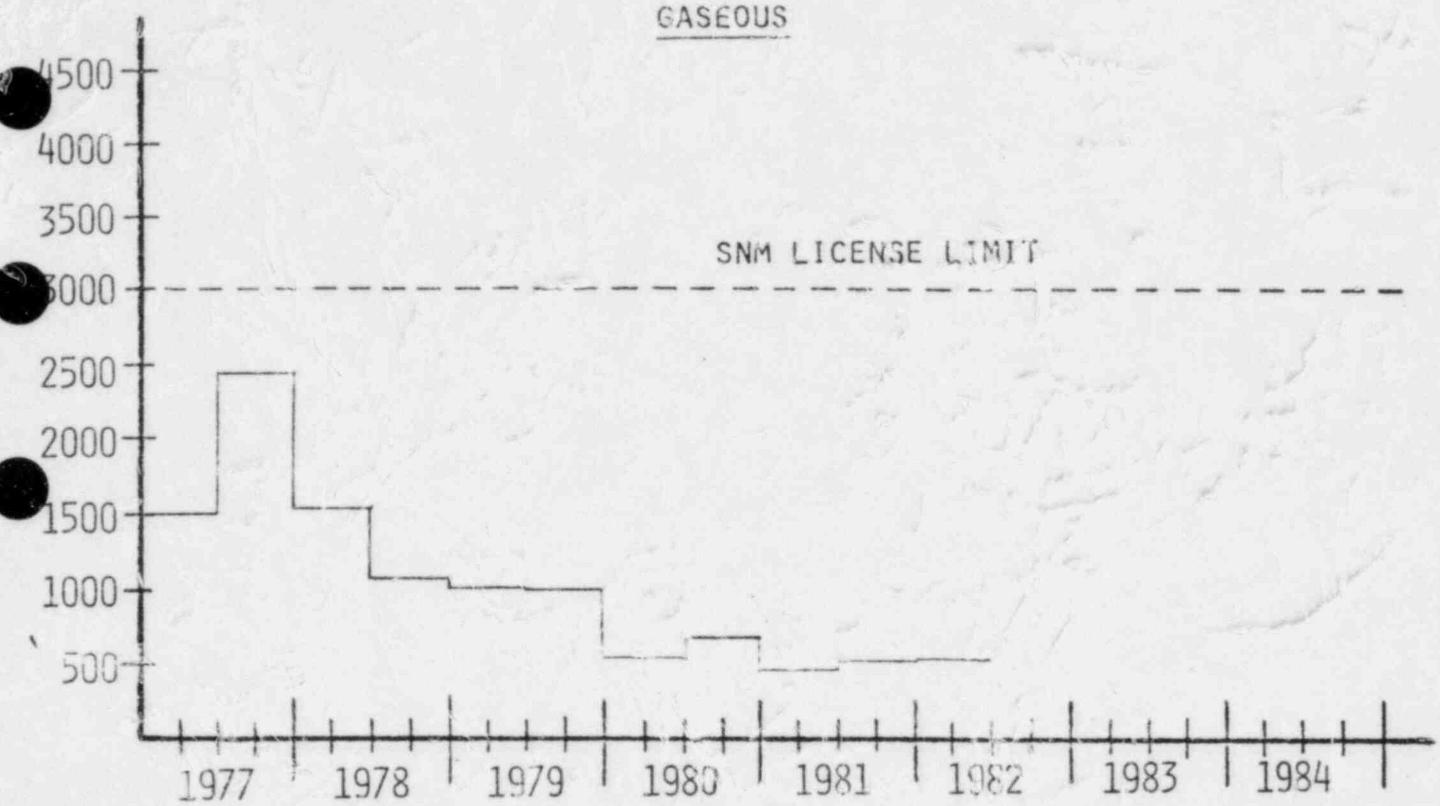


FIGURE 15

LIQUID AND GASEOUS EFFLUENTS



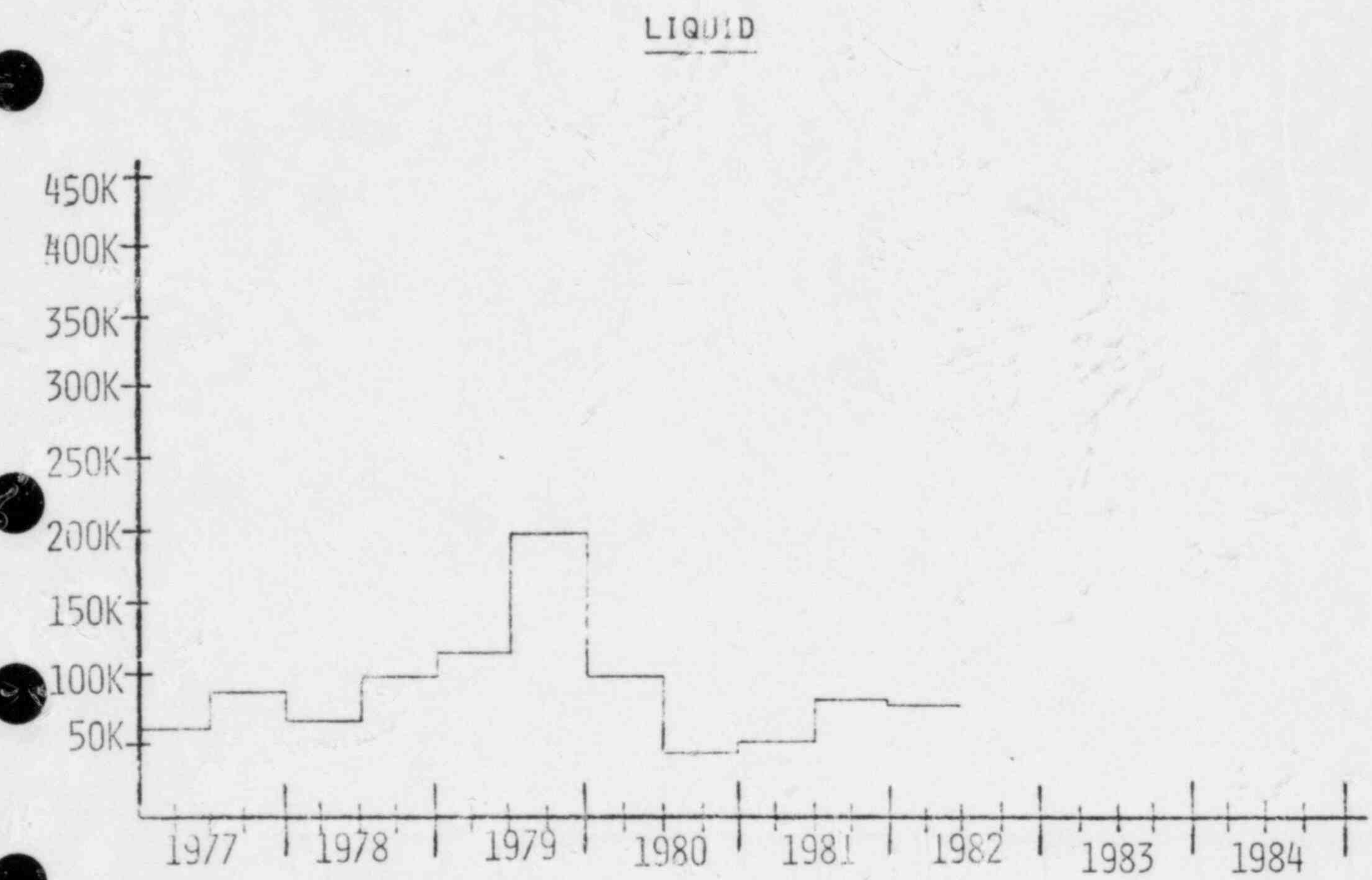
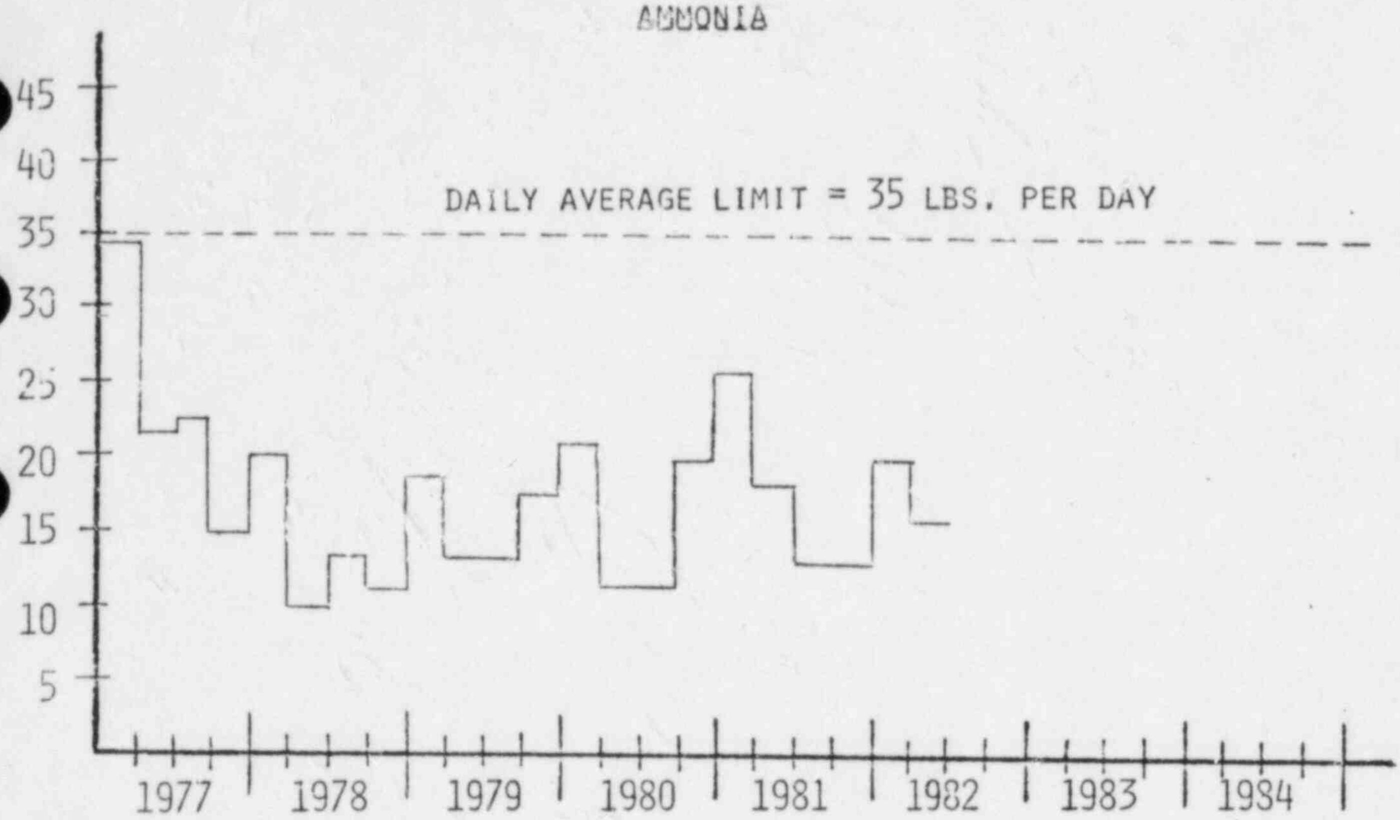


FIGURE 16

AMMONIA AND FLUORIDE EFFLUENTS



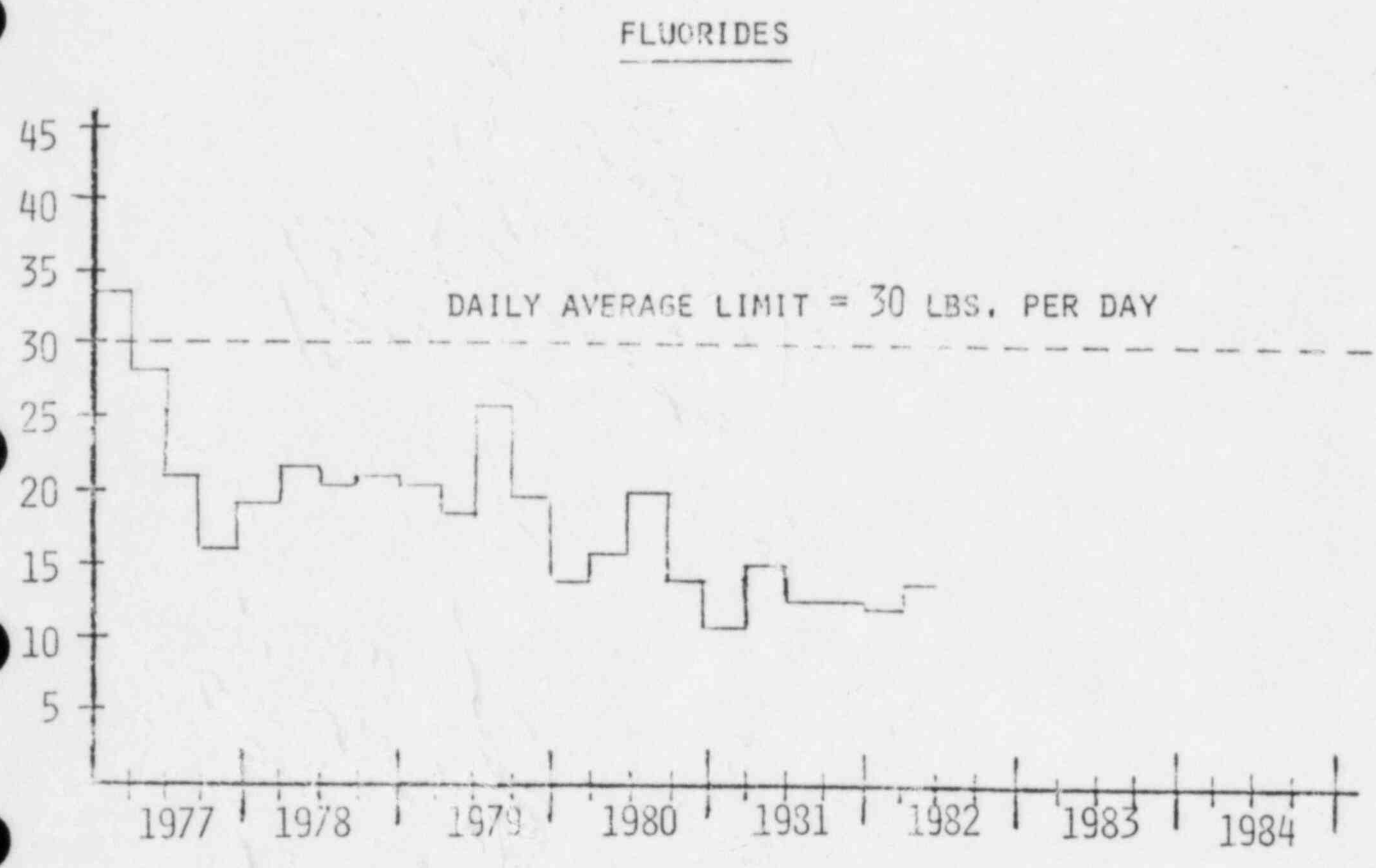


FIGURE 17

WESTINGHOUSE ELECTRIC CORPORATION

NUCLEAR FUEL DIVISION

COLUMBIA PLANT ALARA REPORT

JULY 1 TO DECEMBER 31, 1982

Prepared by: Edward Reitler Date: 2/18/83
E. K. Reitler, Fellow Engineer

Approved by: C. F. Sanders Date: 2/18/83
C. F. Sanders, Manager
Radiological & Environmental Engineering

W. L. Goodwin Date: 2/18/83
W. L. Goodwin, Manager
Regulatory Compliance

Mead D'Amore Date: 2/18/83
Mead D'Amore, Manager
Columbia Plant

CONTENTS

I. SUMMARY

This report summarizes ALARA programs and data for the last six months of 1982 and evaluates short- and long-term trends in the following topical areas: personnel exposures (external and internal), airborne radioactivity concentrations, effluent and environmental monitoring, unusual occurrences, audit findings, uses of equipment, and ALARA programs. The most significant trends are summarized below:

GENERAL

The ALARA program for the Columbia Plant is directed by the Regulatory Compliance Committee (RCC), an organization of technical staff managers and other cognizant personnel. This Committee meets routinely to consider pertinent subjects related to the ALARA program such as airborne radioactivity concentrations, regulatory requirements, significant ALARA trends and compliance inspections. Specific ALARA efforts associated with airborne reduction programs have been delegated to the Airborne Radiation Task Force, a subcommittee of the RCC. This group meets routinely to review specific airborne data and trends, identifies corrective actions where applicable and schedules implementation programs. ALARA philosophies are transmitted to appropriate personnel via policies, procedures and training. A detailed description of specific ALARA programs for this reporting period is found in Section IX.

AIRBORNE RADIOACTIVITY CONCENTRATIONS

Controlled Area airborne radioactivity averages were approximately 10% MPC for the last six months of 1982 versus 8% MPC for the first six months of 1982. Increases were observed throughout the Controlled Area. This is

attributed to specific problems encountered at certain air sample stations which resulted in elevated concentrations and to the unusually high activites associated with facility modifications (i.e. MAP, bulk powder handling and ventilation changes).

The number of air sample stations exceeding an average of 25% MPC has increased from five during the first six months of 1982 to eleven during the current reporting period because of problems at the Stratification Blender, Conversion Line 3 and the Pellet Area. Corrective actions have been identified or evaluations are underway for all such air sample stations.

The number of airborne spikes exceeding 100% MPC increased by approximately 17% during the last six months of 1982 compared with the first six months of 1982. This was attributed to an increase in the number of airborne spikes on Conversion Line 3, "Blue M" Furnaces and the incinerator.

PERSONNEL EXPOSURES

External personnel exposures for both whole body and skin continue to average less than 25% of the NRC maximum permissible quarterly dose of 1,250 mrem and 7,500 mrem respectively. Decreasing trends since 1978 are apparent for all departments as a result of equipment and material handling improvements in the Controlled and Mechanical Areas.

Internal personnel exposures, as monitored by airborne exposures in MPC-hours and bioassay measurements (urine and in-vivo) are consistent with airborne radioactivity concentrations in the Controlled Area.

EFFLUENTS AND ENVIRONMENTAL

Effluents from the Columbia Plant are sampled and analyzed to determine releases to the environment. Based upon NRC meteorological and dispersion models and calculations, airborne effluents result in an annual dose to the

nearest resident to the Columbia Plant Site which is less than 8% of Federal limits, based upon a permitted annual dose of 25 mrem.

CONCLUSIONS

(1) Inplant airborne radioactivity concentrations have increased during the last ALARA reporting period. Programs have been identified to address those areas with the highest airborne potentials.

(2) External personnel exposures continue to average well below permissible levels, with continued decreasing trends.

(3) Environmental discharges remain well within license limits and permitted values.

II. EXTERNAL PERSONNEL EXPOSURES

External personnel dosimeters (TLD's) are supplied to those individuals who may be exposed to greater than 25% of the applicable quarterly radiation dose limits specified in paragraph (a) of 10 CFR 20.101. Personnel dosimeters are also distributed, upon request or at the discretion of Radiological and Environmental Engineering, to those individuals who would not normally be monitored. Badges are provided for approximately 500 individuals and are changed and evaluated quarterly for men and monthly for women (the latter frequency was implemented to provide additional administrative controls for fertile women in minimizing fetal exposures).

Whole body and skin doses by job function are summarized in Figures 1-8 for 1977 through the third quarter of 1982 for each major job category. (Because quarterly exposure reports are provided by a vendor, data for the fourth quarter of 1982 were not available for this report; for this reason, external exposure data reporting will always be one quarter behind.)

All departmental whole body dose averages for each quarter to date were below 25% of the NRC Maximum Permissible Dose Limit of 1,250 mrem. A comparison of quarterly average whole body dose results for the last four quarters versus the average for the four prior quarters shows the following: Conversion (-30%), Pellet (-19%), Rod (-44%), QC Chemical (-26%), Maintenance (-31%), Final Inspection (-4%), Rod Inspection (-20%), Final Assembly (-2%). Reductions were observed in all areas. Trend analyses have shown significant decreases in the past three years, particularly for Controlled Areas. The following modifications have contributed to these favorable trends: (1) improvements in housekeeping which have significantly lowered contamination levels on floors, process equipment, personnel dosimeters, etc.,

(2) relocations of uranium powder to central storage areas to reduce personnel exposures at the process lines, (3) storage of pellets in shielded storage carts, (4) removals of contaminated scrap and surplus equipment from the facility, (5) reduction in the inventory of UF_6 cylinder "heels" in the UF_6 Bay, (6) personnel training which emphasized dose reduction techniques and increased personnel awareness, (7) enclosure improvements (e.g., stratification blender and QC inspection hoods) to provide for more remote handling of uranium powder and pellets, and (8) increased mechanization in the Rod Area.

All departmental skin dose averages were also less than 25% of the NRC Maximum Permissible Dose of 7,500 mrem for each quarter to date. A comparison of quarterly average skin dose results for the last four quarters versus the average for the four prior quarters shows the following: Conversion (-22%), Pellet (-24%), Rod (-50%), QC Chemical (-27%), Maintenance (-20%), Rod Inspection (-16%), Final Inspection (+10%), Final Assembly (+7%). For the Controlled Areas, the trends for skin doses follow the pattern for whole body doses (i.e., a decreasing trend since 1980). For the Mechanical Area there are no positive or negative trends.

No unusual individual whole body or skin doses were noted. The highest individual whole body dose for the second and third quarters was 560 mrem (Conversion Area); the highest individual skin dose was 1,600 mrem (Conversion Area).

Approximately 47 badges are placed throughout the Manufacturing areas to monitor general area exposure rates. Exposure rates are consistent with powder, pellet, rod and assembly inventories in these areas.

The integrated whole body doses for the second and third quarters of 1982 were approximately 46 and 37 person-rem respectively, or an average of approximately 82 mrem per quarter per person badged. The integrated dose has increased somewhat during the last two quarters; a portion of this increase is attributed to an increase in the number of badged individuals.

The ALARA Program to minimize external personnel exposures consists of the following: (1) routine surveillance of radioactive material areas for external radiation, (2) personnel and area dosimetry, (3) Radiological and Environmental Engineering reviews of surveillance and dosimetry results, (4) investigations of unusual occurrences, and (5) evaluations of area and personnel trends. From data obtained through these means, recommendations are made for process and/or equipment modifications which will further reduce radiation exposures.

During August and September 1982, a study was performed to quantify external extremity doses to a representative sampling of personnel in the Conversion, Pellet, Rod, QC Chemical and Mechanical Manufacturing Areas. Finger rings were provided for approximately 35 individuals for five or six weeks. The maximum dose for the badged period was 1,210 mrem for six weeks which corresponds to a maximum quarterly dose of 2,621 mrem. The average dose for the six-week period was approximately 62 mrem per person per week. Based upon this study, the maximum extremity dose to fuel handlers should be well below 25% of the Maximum Permissible Dose defined in 10 CFR 20.101.

III. INTERNAL PERSONNEL EXPOSURES

Personnel internal exposures are based upon Controlled Area air sampling results at representative work stations and an accounting of time spent by each individual at these locations. Daily internal exposures (in MPC-hours) are calculated for each individual who spends a significant portion of his time working with unencapsulated forms of uranium. The average internal exposures in MPC-hours for each job category since 1978 are presented graphically in Figures 9-13.

Bioassay (urine, fecal and in-vivo) results are used to verify the internal exposures. Urinalyses and in-vivo counts are performed on a routine frequency based upon the exposure potential. Fecal sampling is performed to verify uranium uptakes in the event of an unusual occurrence. The average bioassay results (urine and in-vivo) per quarter per job category since 1977 are presented graphically in Figures 9-13.

MPC-Hours

Personnel exposures in MPC-hours increased somewhat during the fourth quarter of 1982, reflecting increases in airborne radioactivity concentrations for this same period (Conversion: 56 MPC-hours, Pellet: 74 MPC-hours, Rod: 31 MPC-hours, QC Chemical: 33 MPC-hours, Maintenance: 30 MPC-hours). The highest quarterly value (Pellet Area: 74 MPC-hours) represents approximately 14% of the maximum permissible quarterly limit of 520 MPC-hours. The highest individual exposure occurred in the fourth quarter (124 MPC-hours) to a Pellet Area employe. There were no exposures exceeding 40 MPC-hours in any seven consecutive day period.

Urinalyses

Urine samples are collected routinely from potentially exposed personnel to assist in the evaluation of exposures to transportable and nontransportable uranium. Urinalysis frequencies are determined routinely by Radiological and Environmental Engineering based upon criteria such as exposure levels in MPC-hours, scope of work in the Controlled Area, etc.

Urinalysis results are plotted in Figures 9-13 for each major department. There were no significant changes in departmental averages during the last two quarters of 1982.

In-vivo

In-vivo departmental averages for the third and fourth quarters of 1982 continue to average less than the minimum detectable quantity of U-235 for the in-vivo counter as shown in Figures 9-13. There were no significant trends in the departmental averages for this period. Only one individual exceeded 200 micrograms U-235 (the restriction limit); this individual worked in the Conversion Area as an operator. Subsequent counts decreased initially but have recently reached a plateau of approximately 100 micrograms U-235 as of January 1983. This individual will remain on restriction until his invivo count falls below 100 micrograms U-235.

Personnel Work Restrictions

Personnel are restricted from working with radioactive materials for a variety of reasons, including routine urinalyses and in-vivo counts exceeding certain internal limits, elevated nasal smears, unusual occurrences, etc. During this reporting period, seven individuals were restricted, two because of elevated urine sample results, one due to elevated in-vivo analyses and four because of high airborne radioactivity. This represents a decrease

compared with restrictions in the first half of 1982 (13). Of the seven restrictions, four occurred in the Conversion Area, two in Waste Recovery and Disposal, and one in Maintenance. All individuals were subsequently released from restriction and returned to their routine work assignment except for one Conversion Area opeator who was restricted because of an elevated invivo measurement. Although this individual's invivo count has fallen below the level normally used to remove a person from restriction, he has been retained on restriction because of a longer than expected biological halflife.

NOTE: Some work restrictions result from failure to submit timely bioassay samples. These restrictions were not included in the above data analyses.

IV. AIRBORNE RADIOACTIVITY CONCENTRATIONS

Airborne radioactivity concentrations in the Controlled Area are sampled continuously at approximately 179 individual work stations, with samples changed and counted each shift. In addition, special air samples are taken to evaluate airborne radioactivity during nonroutine activities and to support engineering evaluations. The following parameters are used to evaluate airborne radioactivity trends: (1) Controlled Area average airborne radioactivity concentrations, (2) individual area average airborne radioactivity concentrations, (3) individual air sample stations exceeding 25% MPC, and (4) airborne radioactivity spikes. These are discussed below:

The Controlled Area airborne radioactivity concentration average for the last six months of 1982 was 10% MPC versus 8% MPC for the first six months of 1982. Individual area averages are presented in Figure 14 from 1976 to the present, showing trends for this period.

The locations and average results of individual Controlled Area air sample stations averaging greater than 25% MPC for the last six months of 1982 are listed in Table 1, together with the trends for this period. During the last six months of 1982, approximately 6% (11) of the air sampler stations averaged over 25% MPC; this represents an increase over the first six months of 1982 (3% or 5 sampler stations). In the Conversion Area, three air sample stations, Line 3 Feed End Top, Elevator Bottom and Product End Bottom Mill, exceeded 25% MPC; compared with one for the first six months of 1982 (Line 3 Hot Filter Hood).

Line 3 Feed End Top averaged 28% MPC for the six-month period; however, during the week ending December 18, 1982, this sample averaged 171% MPC because of a leaking duplex valve and associated containment problems on the

feed end of Line 3. If this week were deleted from the data, the sample would have averaged 22% MPC for the six-month period. Most of the airborne problems on Line 3 were associated with process upsets such as plugging and leaking of equipment and inadequate containment.

Line 3 Elevator Bottom also averaged 28% MPC for the six-month period, and resulted from the same causes described above for the Line 3 Feed End Top sample. A new hood was installed at the vent star valve to reduce airborne associated with this operation. Furthermore, Line 3 was shut down in January 1983 to upgrade the feed end enclosure by reducing the size to facilitate access to the elevator and duplex valves and to expedite decontamination of the enclosure. Improvements are also planned for the ADU dryer system (improved design for the ADU dryer filter bags to reduce airborne during bag changes) and modifications to the calciner off-gas system to minimize releases from this area.

The Line 3 Product End Mill sample averaged 34% MPC for the six-month period versus 19% MPC and 15% MPC for similar samples located on Lines 2 and 4 respectively (the other lines which operated during this period). The higher airborne concentrations on Line 3 have been attributed to an unusually high frequency of maintenance activities resulting from equipment failures and from enrichment cleanouts. Several of these activities required the removal of the secondary containment surrounding the Fitzmill. Increased emphasis has been placed on the need for temporary ventilated enclosures such as tents during these activities. The air sampler for this station is located in a position which makes it vulnerable to powder spills when the containment is opened. To correct this situation, an evaluation will be made to move the sampler to another representative position where powder spills are less likely.

In the Conversion Services Area, three air sample stations, Stratification Blender Input Hood, "Blue M" Furnaces 1 and 2 and Incinerator output, exceeded 25% MPC; compared with one for the first six months of 1982 (Blue M Furnaces 1 and 2).

The Stratification Blender Input Hood averaged 30% MPC for the last six months of 1982 versus 23% for the first six months of 1982. The primary contributors to this airborne were hood ventilation and containment problems, material handling and operator-related problems. An evaluation is underway to determine the corrective actions required to minimize the airborne concentrations at this station.

The "Blue M" Furnaces 1 and 2 sample averaged 42% MPC for the last six months of 1982 versus 34% MPC for the first six months of 1982. The two furnaces had been previously relocated adjacent to each other with the air sampler positioned between them, thus resulting in a higher airborne radioactivity concentrations. To correct this problem, new ventilated hoods are being designed for the furnaces to contain fumes generated during material processing. Furthermore, additional ventilation capacity is being provided for the furnaces.

The incinerator output sample averaged 26% MPC for the last six months of 1982, versus 21% for the first six months of 1982. Most of the airborne resulted from ash transfers and blowouts during the third quarter, resulting in an average of 34% MPC. During the fourth quarter, the average at this station was reduced to 18% MPC. Equipment is being designed to automate the ash handling between the incinerators and the ash hood.

In the Pellet Area, Pellet Presses 1, 2, 3, and 4 and Pellet Line 2 Furnace exceeded 25% MPC for the last six months of 1982.

Pellet Presses--Four pellet presses exceeded 25% MPC for this period versus three presses for the first six months of 1982. Line 1 Press decreased from 33% MPC to 30% MPC, Line 2 Press increased from 30% MPC to 32% MPC, Line 4 Press increased from 30% MPC to 33% MPC and Line 3 Press increased from 20% MPC to 30% MPC. The number of spikes greater than 100% MPC was reduced from 85 during the first six months of 1982 to 57 during the last six months of 1982. The primary causes of airborne at the presses are containment leaks, powder spills within containment, maintenance activities and operator transfers of material outside the enclosures. Several improvements were implemented for the presses including modifications to the enclosures to reduce their size and improve the enclosure-to-press sealing surfaces, refurbishment of the presses to minimize powder spills within the enclosures, increasing operator awareness of airborne related problems and increased emphasis on decontamination. These efforts are continuing into 1983.

Pellet Line 2 Furnace increased from 16% MPC to 27% MPC. There were several possible contributors to this increase including airborne carryover from both the feed and product ends of Pellet Line 2 and from adjacent lines and operator-generated airborne. Several actions were taken to correct this problem including an increased emphasis on decontamination and additional containment. The line was shut down during the fourth quarter for modifications. When the line is restarted, the airborne trends will be reevaluated to determine the effectiveness of these actions.

For the Conversion, Conversion Services and Pellet Areas, approximately 74% of the air sample stations experienced a net increase for the last six months of 1982 versus the first six months of 1982; 15% experienced a decrease; and 11% were unchanged. This has been attributed to an unusually high frequency of modifications within the Controlled Area including installation of bulk powder handling, MAP construction and Conversion Line 3 construction. Furthermore, ventilation was disrupted on two occasions during the construction of the MAP area and the relocation of the main Conversion Area scrubbers.

A list of air sample stations with the highest frequency of spikes greater than 100% MPC is shown in Table 3, including the trends for the last half of 1982 versus the first half of 1982. Each of these is discussed below:

Incinerator Output--The number of spikes increased from 15 during the first half of 1982 to 27 during the last half of 1982. The majority of them (22) occurred during the third quarter as a result of operational problems resulting from the burning of oils. Following the completion of corrective actions, the number of spikes was reduced to 5 during the fourth quarter.

A new system is being designed for the automation of ash handling at this station to further reduce the number of spikes and airborne radioactivity concentrations. This system should be implemented by late 1983.

Incinerator Ash Hood--A large volume of ash generated by the incinerator has resulted in a higher than normal throughput at this location. The instantaneous hood inventory of ash has also increased, which leads to

problems such as housekeeping and contamination. Recently, a greater emphasis has been placed upon operating parameters to improve this situation. As a result, airborne radioactivity concentrations were reduced substantially during the last 8 weeks of 1982 (approximately 10% MPC) as well as the number of spikes (one spike in the last 8 weeks of 1982).

Pellet Presses--The number of spikes greater than 100% MPC was reduced from 85 during the first six months of 1982 to 57 during the last six months of 1982. This can be attributed to a number of factors including: improved containment for the presses, improved housekeeping for ventilated enclosures and in the general area, and upgrading operator awareness of techniques to minimize airborne radioactivity.

Conversion Line 3 Elevator Bottom and Feed End Bottom--Refer to the previous discussion of these two air sample stations in this section.

"Blue M" Furnaces 1 and 2--Refer to the previous discussion of this air sample station in this section.

Conversion Line 3 Product End Mill--Refer to the previous discussion of this air sample station in this section.

V. EFFLUENT AND ENVIRONMENTAL MONITORING

Effluent releases from the Columbia Plant consist of two pathways, gaseous and liquid. Gaseous and particulate effluents are HEPA filtered and scrubbed as appropriate before being discharged to the environment. The total activity discharged during the last six months of 1982 was 574 microcuries of uranium, compared with 502 microcuries during the first six months of 1982. This represents a small increase over the previous reporting period, but within expected statistical variations.

Gaseous effluents represent an average discharge concentration of less than 10% of the Maximum Permissible Concentration for unrestricted areas. Figure 16 shows the trends since 1977.

The total annual discharge of 1,076 microcuries uranium during the last twelve months corresponds to an annual lung dose of approximately 3.8 mrem to an infant living at the nearest plant boundary or approximately 1.8 mrem to an infant at the nearest residence. This represents less than 15% (nearest plant boundary) and 8% (nearest resident) respectively of the environmental standards contained in 40CFR190, as derived from NRC models and calculations.

Liquid wastes are treated to remove chemical and radioactive constituents and discharged to the Congaree River. Sampling and analyses are performed to confirm that effluent concentrations are well below applicable regulatory limits. Typically, liquid discharges average less than 10% MPC. An advanced waste treatment facility was installed in late 1980 to recover uranium from the conversion process liquid waste effluent stream. Uranium concentrations are lowered from approximately 15-20 ppm to less than 0.1 ppm by this treatment process.

During the last half of 1982, approximately 34,700 microcuries of uranium were discharged to the Congaree River, compared with approximately 70,000 microcuries during the first six months of 1982. This reduction is attributed to the low concentration of uranium in the calcium fluoride stored in the process lagoons resulting from the advanced waste water treatment process described above.

Nonradiological effluents of ammonia and fluorides are shown in Figure 17. During the last half of 1982, ammonia discharges averaged approximately 12 pounds per day which is well within the limits of the National Pollutant Discharge Elimination System (NPDES) Permit of 35 pounds per day. Flouride effluents averaged approximately 16 pounds per day which is also well within the NPDES Permit limit of 30 pounds per day.

There was one violation of the NPDES Permit during this period for exceeding the daily maximum discharge limit for BODS on November 25, 1982. Appropriate corrective actions have been taken.

A revised NPDES Permit has been approved and will become effective on February 1, 1983. The major changes were: (a) deletion of the nickel monitoring requirement, (b) relaxation of the pH compliance requirement and (c) a requirement to develop a "Best Management Practices" plan within six months.

Environmental airborne radioactivity concentrations are sampled continuously to verify that plant effluents are ALARA. Air concentrations at the site boundaries continue to average approximately 0.1% of the 10CFR20 Maximum Permissible Concentration for unrestricted areas.

VI. UNUSUAL OCCURRENCES

During the last six months of 1982, there were no unusual occurrences requiring complete evacuation of the Controlled Area. There was a brief power outage in July 1982 which resulted in a partial evacuation for less than one hour.

As shown in Table 7, there were no personnel exposures during the last six months of 1982 which exceeded 40 MPC-hours in any seven-day period. This is attributed to the controls on airborne radioactivity and personnel to minimize intakes of uranium.

VII. AUDITS

Routine Regulatory Compliance audits were conducted during the last six months of 1982 in the areas of radiation protection, nuclear criticality safety and SNM safeguards. Inspections are conducted jointly by Radiological and Environmental Engineering personnel and line management in accordance with written checklists. Results of inspections are discussed with cognizant supervisors and managers, scheduled for correction, and followed up to assure that timely corrective actions have been taken.

Radiation protection items identified during this period were associated with uranium contamination, containment capture velocities, air sampling, containment integrity, ventilation indicators, wearing of personnel dosimetry, completion of ventilation logs, and housekeeping. Nuclear criticality safety items included SNM storage, moderation controls, spacing of SNM, container labelling, and criticality postings. Safeguards findings included item control, SNM identification, residual SNM in polypaks and tamper-seal records storage. In most instances, effective corrective actions were taken to prevent recurrence. No undesirable trends were noted.

A total of eleven NRC inspections were conducted during the last six months of 1982 involving approximately 69 inspection person-days versus twelve inspections and approximately 79 person-days for the first half of 1982. The inspections were conducted in the areas of health physics, nuclear criticality safety, safeguards (SNM measurements, accountability and SNM physical inventory), environmental monitoring, liquid and gaseous effluents and emergency planning. There were two items of noncompliance versus four during the previous reporting period. Table 8 summarizes the results of NRC inspections.

Both of the items of noncompliance were received in the Safeguards area: (1) failure to follow a procedure regarding the storage of tampersafing seals in secured areas and recording tampersafing information on inappropriate forms that were devoid of information containing the time of seal application and the identification of individuals applying the seals (Severity Level V) and (2) failure to maintain current knowledge of the identity, quantity and location of loose fuel rods as evidenced by the detection of up to 50 fuel rods unaccounted for on the October 30-31, 1982 SNM physical inventory (Severity Level IV). Regarding item (2) above, the NRC requested that Westinghouse address other contributing factors which may exist that led to the failure of the item control program to appropriately account for all fuel rods. Appropriate corrective actions have been taken to address the violations and to prevent recurrence.

The two items of noncompliance were received during two inspections; there were no items of noncompliance during nine inspections.

During July 1982 the internal audit program was expanded to enhance the effectiveness of the Columbia Plant management control systems and to minimize regulatory and license violations. The following specific programs have been implemented: (1) The formal scope of the Regulatory Compliance Committee (RCC) was expanded to include SNM control, accountability and safeguards, (2) Items of noncompliance are formally reviewed by the RCC, including responses to the NRC and assurance of completion of corrective actions, and (3) Formal quarterly audits have been established by the RCC in the areas of SNM control, accountability and safeguards.

On September 22, 1983, a violation was received from South Carolina Department of Health and Environmental Control (DHEC) as a result of holes in

four drums of low-level radioactive waste received at the Chem Nuclear Barnwell Disposal Site. A $1,000 penalty was assessed by DHEC. Appropriate corrective actions were taken including procedure review and updating and the use of 17-H drums rather than "strong tight containers."

VIII. USES OF EQUIPMENT

The Health Physics Operations Department has maintained an on-the-floor laboratory to evaluate surveys and to provide a base of operations for the Controlled Area. Counting and survey instruments are provided, including an automatic planchet sample changer. During the last six months of 1982, the laboratory was relocated to the Conversion Area Control Room. This has resulted in a communications improvement between area supervision and Health Physics Technicians assigned to the Controlled Area.

An evaluation was performed for the stack and inplant air sampling systems to determine the accuracy of flow measuring devices (e.g., rotometers). As a result of this evaluation, correction factors have been applied to both systems to compensate for the errors introduced when rotometers are operated in a negative pressure mode. Furthermore, all rotometers were calibrated during this period, including fixed air samplers, recirculating air sample stations and those associated with effluent sampling.

The Health Physics nonroutine activities surveillance coverage has been modified to provide for improved feedback of results to cognizant personnel to facilitate timely corrective actions and for protection of air samplers from contamination from powder spills during these activities.

Modifications were made in the Waste Treatment Area to improve the mixing of treated process wastewater with lime to assure proper homogenization of the two streams prior to ammonia distillation.

Improvements were incorporated into the low-level waste handling and shipping area to assure adherence to all waste packaging, labeling and transportation requirements including: (a) engineering overchecks of waste

packages and paperwork, (b) additional operator training and (c) upgrading the low-level waste containers from "strong tight packages" to Type A containers.

Environmental sampling stations are protected from the weather by an enclosure which surrounds the samplers. New enclosures have been designed and manufactured to replace the existing units which are deteriorating with age.

Source checks were performed for the invivo counter using calibrated standards from a commercial vendor. Based upon the results of these tests, the current calibration of the counter was confirmed.

IX. ALARA PROGRAM SUMMARY

A. General

Management commitment to the As-Low-As-Reasonably-Achievable (ALARA) principle continued in order to minimize airborne radioactivity concentrations, personnel exposures and effluents to the environment. The overall ALARA program is directed by the Regulatory Compliance Committee which establishes plant policies and procedures and monitors progress of implementing groups in achieving ALARA. One of these groups, the Airborne Radiation Task Force, met on two occasions during the last six months of 1982 to identify and report progress on "Airborne Milestones," review airborne radioactivity concentrations and evaluate facility trends in several important areas. Results of Task Force activities are reported to the Regulatory Compliance Committee for information and concurrence. In addition to these efforts, the General Manager of the Nuclear Fuel Division monitors progress towards achieving ALARA goals.

B. Airborne and Effluent Reduction Programs

The following programs have been effective in reducing airborne radioactive concentrations in the Controlled Areas during the current reporting period of July to December 1982.

1. Process Line Improvements

   a. Calciner 3 Vent System

   Calciner offgases are routed to the scrubber for treatment via the calciner vent system. Occasionally, this system will become plugged with UO_2F_2 accumulations, thus requiring cleanouts. To reduce airborne radioactivity releases during these

cleanouts, ventilated enclosures have been installed at the cleanout ports on two operating conversion lines. Furthermore, preventive maintenance type routine cleanouts and a more efficient cleanout tool have been instituted to reduce airborne.

b. ADU Dryer Bags

The ADU dryer bags which filter dryer offgases were modified on Conversion Line 4 to shorten the bags and frame. This improved the procedure for changing the bags when they become plugged, thus reducing the airborne potential.

c. Pellet Press Enclosures

Efforts continued to improve the pellet press enclosures to minimize airborne leaks at the enclosure to press interface; procurement of a new scale to enable operators to weigh and measure pellets within the enclosure (Line 1); reduction in the enclosure size to increase ventilation flows across openings; enclosure improvements to facilitate maintenance, including removal and refitting the enclosures to the presses.

d. Conversion Line 3 Polypak Hood

A new hood was installed on the feed end of Conversion Line 3 to contain the polypak dump station.

e. Four pellet presses were refurbished to reduce the quantity of powder spilled within the enclosures. Improvements included refinishing of the press table and upgrading the die feeders.

2. Facilities Improvements

a. Conversion Line 2 Modifications

The discharge end of Conversion Line 2 was modified to accommodate the bulk powder handling system. Improvements include upgraded powder transport equipment and new enclosures. The line is being operated with polypaks until the remainder of the bulk powder handling equipment is installed. When this system is fully operational, a reduction in the number of polypaks is expected to result in airborne reductions as well.

b. Conversion Scrubber Relocation

The main Conversion Area scrubber relocation was completed during this period. Concurrently, the gas-fired heaters were replaced with electric heaters to provide greater reliability of heating scrubber off-gases.

c. Backup Precipitator Vessels

Backup precipitator vessels were installed on one conversion line. By providing a second precipitator, the process stream can be diverted to the clean vessel when the first vessel requires cleaning, thus reducing line downtime and possible airborne radioactivity.

d. Solvent Extraction Containment and Ventilation

Additional containment and ventilation were installed on the process vessels (extraction and stripping columns), filter presses and recovery units in the solvent extraction area to minimize airborne problems.

e. Pellet Storage

Additional pellet tray storage carts were obtained for the purpose of minimizing airborne and radiation levels associated with pellet storage.

Table 1

LOCATIONS AND AVERAGE READINGS OF ALL AIR SAMPLERS AVERAGING 25% MPC

7/1/82 to 12/31/82

Conversion Area	% MPC	Trend (7/1/82 - 12/31/82) (1)	
Line 3 Feed End Top	28%	↑	(2)
Line 3 Elevator Bottom	28%	↑	(2)
Line 3 Product End Bottom Mill	34%	↑	(2)
Conversion Services			
"Blue-M" Furnace 1 and 2	42%	↑	
Stratification Blender Input Hood	30%	↑	(2)
Incinerator Output	26%	↑	(2)
Pellet Area			
Line 1 Pellet Press	30	↓	
Line 2 Pellet Press	32	↑	
Line 3 Pellet Press	30	↑	(2)
Line 4 Pellet Press	33	↑	
Pellet Line 2 Furnace	27	↑	(2)
Rod Area			
All sample stations averaged less than 25% MPC	N/A		N/A

(1) Eleven out of 179 samples averaged above 25% MPC

(2) These samples averaged less than 25% MPC during the previous reporting period.

Table 2

NUMBER OF AIRBORNE RADIOACTIVITY SPIKES *

TIME PERIOD	100-200% MPC	201-500% MPC	500% MPC	TOTAL 100% MPC
1/1/77 - 6/30/77	1,297	249	40	1,586
7/1/77 - 12/31/77	1,216	273	55	1,544
1/1/78 - 6/30/78	656	175	49	880
7/1/78 - 12/31/78	654	160	44	858
1/1/79 - 6/30/79	965	293	79	1,337
7/1/79 - 12/31/79	1,172	389	89	1,650
1/1/80 - 6/30/80	540	177	29	746
7/1/80 - 12/31/80	372	140	42	554
1/1/81 - 6/30/81	270	54	13	337
7/1/81 - 12/31/81	267	83	18	368
1/1/82 - 6/30/82	285	80	17	382
7/1/82 - 12/31/82	339	81	17	447

*Based upon approximately 101,000 samples each semi-annual period.

Table 3

HIGHEST FREQUENCY SPIKE STATIONS (>100% MPC)

7/1/82 to 12/31/82

STATION	NUMBER OF SHIFT SAMPLES 100% MPC	% OF TOTAL	TREND LAST HALF 1982 VS. FIRST HALF 1982.
Incinerator Output	27	6	↑
Incinerator Ash Hood	24	5	↑
Line 3 Pellet Press	24	5	↑
Conv. Line 3 Elevator Bottom	20	4	↑
"Blue-M" Furnaces 1 and 2	18	4	↑
Conv. Line 3 Feed End Top	17	4	↑
Conv. Line 3 Product End Mill	17	4	↑
Line 4 Pellet Press	16	4	↓
Line 1 Pellet Press	16	4	↓

Table 4

CONVERSION LINE SUMMARY (LAST HALF OF 1982 VS. FIRST HALF OF 1982)

OPERATING LINES ONLY

	Line 2			Line 3			Line 4		
	First Half 1982	Last Half 1982	Trend	First Half 1982	Last Half 1982	Trend	First Half 1982	Last Half 1982	Trend
Feed End Top Average	17%	18%	↑	20%	28%	↑	10%	16%	↑
Feed End Bottom Average	11%	15%	↑	13%	17%	↑	12%	12%	→
Vent Star Average	12%	20%	↑	27%	21%	↓	25%	18%	↓
Calciner Recycle Average	20%	19%	↓	23%	23%	→	5%	10%	↑
Feed End Average	15%	18%	↑	19%	22%	↑	13%	14%	↑
Product End Average	14%	16%	↑	14%	18%	↑	13%	16%	↑
Number of Airborne Spikes 100% MPC (Feed End)	22	31	↑	53	50	↓	21	24	↑
Number of Airborne Spikes 100% MPC (Product End)	13	12	↓	18	27	↑	12	5	↓
Production Throughput	72	90	↑	80	75	↓	105	106	↑

Table 5

EFFLUENT DISCHARGES, uCi TOTAL URANIUM

TIME PERIOD	GASEOUS EFFLUENT		LIQUID EFFLUENT	
	uCi	Kg U*	uCi	Kg U*
7/1/77 - 12/31/77	2,447	1.22	80,568	40.3
1/1/78 - 6/30/78	1,519	0.76	62,822	31.4
7/1/78 - 12/31/78	1,142	0.57	97,033	48.5
1/1/79 - 6/30/79	1,040	0.52	107,745	53.9
7/1/79 - 12/31/79	1,008	0.50	198,090	99.0
1/1/80 - 6/30/80	537	0.27	97,699	48.8
7/1/80 - 12/31/80	659	0.33	46,204	23.1
1/1/81 - 6/30/81	462	0.22	55,000	27.5
7/1/81 - 12/31/81	485	0.24	75,751	37.9
1/1/82 - 6/30/82	502	0.25	70,003	35.0
7/1/82 - 12/31/82	574	0.29	34,765	17.4

* Assumes a specific activity of 2 uCi/gram.

Table 6

PERSONNEL EXPOSURES EXCEEDING 40 MPC-HOURS IN ANY

SEVEN CONSECUTIVE DAY PERIOD

TIME PERIOD	NUMBER OF EXPOSURES
1/1/77 - 6/30/77	13
7/1/77 - 12/31/77	9
1/1/78 - 6/30/78	3
7/1/78 - 12/31/78	3
1/1/79 - 6/30/79	3
7/1/79 - 12/31/79	2
1/1/80 - 6/30/80	2
7/1/80 - 12/31/80	0
1/1/81 - 6/30/81	0
7/1/81 - 12/31/81	0
1/1/82 - 6/30/82	0
7/1/82 - 12/31/82	0

Table 7

NRC INSPECTIONS

TIME PERIOD	DEFICIENCIES HP & CRIT.	DEFICIENCIES SAFEGUARDS	INFRACTIONS HP & CRIT.	INFRACTIONS SAFEGUARDS	NUMBER OF INSPECTIONS*	TOTAL PO[illegible]
1/1/76 - 6/30/76	2	1	4	0	3	46
7/1/76 - 12/31/76	1	1	2	0	3	24
1/1/77 - 6/30/77	1	1	7	1	8	[illegible]
7/1/77 - 12/31/77	5	6	6	0	13	82
1/1/78 - 6/30/78	0	1	5	0	8	52
7/1/78 - 12/31/78	0	6	2	0	10	32
1/1/79 - 6/30/79	3	8	0	2	10	42
7/1/79 - 12/31/79	1	$2^{(1)}$	$3^{(2)}$	0	8	38
1/1/80 - 6/30/80	2	9	0	0	10	22
7/1/80 - 12/31/80	0	3	0	2	8	[illegible]

ENFORCEMENT CATEGORY (3)

	IV HP&CR.	IV SGDS.	V HP&CR.	V SGDS.	VI HP&CR.	VI SGDS.	OTHERS HP&CR.	OTHERS SGDS.		
1/1/81 - 6/30/81	0	0	0	4	0	0	0	0	10	NA
7/1/81 - 12/31/81	0	0	3	2	2	1	0	0	10	NA
1/1/82 - 6/30/82	0	0	$1^{(4)}$	$3^{(5)}$	0	0	0	0	12	[illegible]
7/1/82 - 12/31/82	0	$1^{(6)}$	0	1	0	0	0	0	11	NA

(1) One deficiency was a repeat.

(2) Includes a transportation infraction which occurred in 1979 but was not imposed un[illegible] 1980.

(3) NRC IE Enforcement Criteria implemented during 1981

(4) Fire Protection Violation

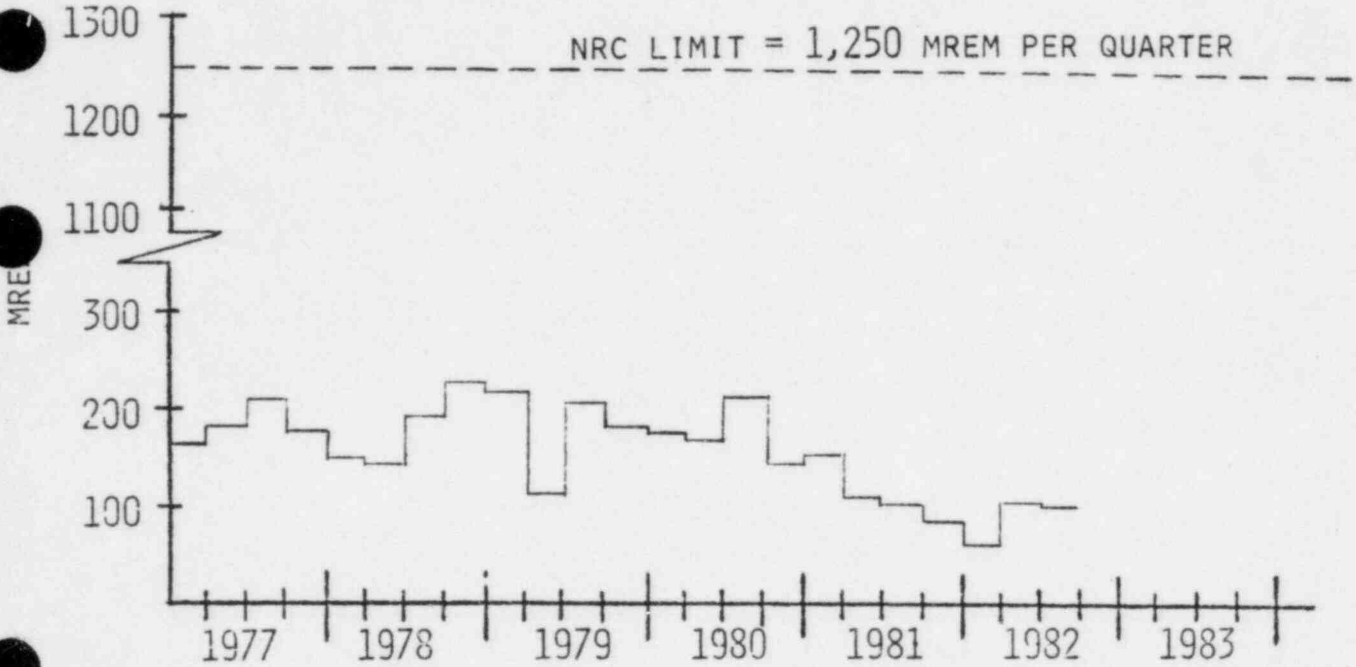
(5) One Violation was a Repeat

(6) Failure to Follow Procedures Regarding Item Control of Fuel Rods

AVERAGE QUARTERLY EXTERNAL DOSE PER EMPLOYEE

AREA: CONVERSION

WHOLE BODY DOSE



SKIN DOSE

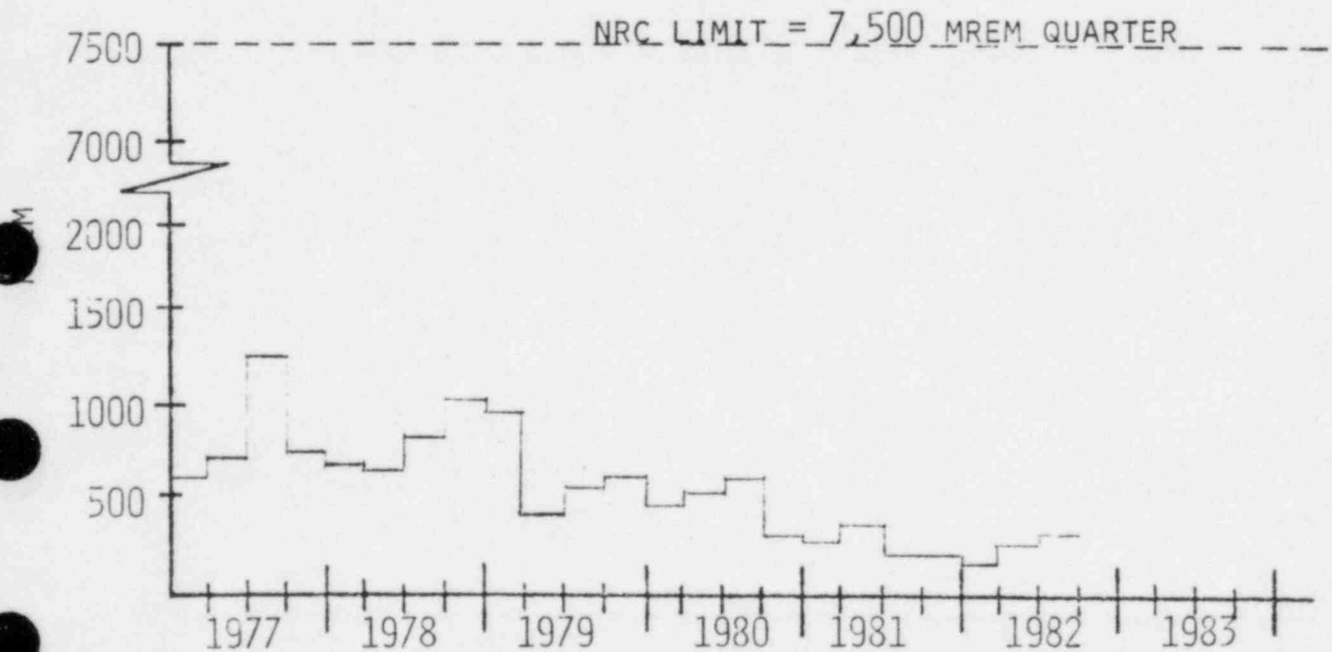


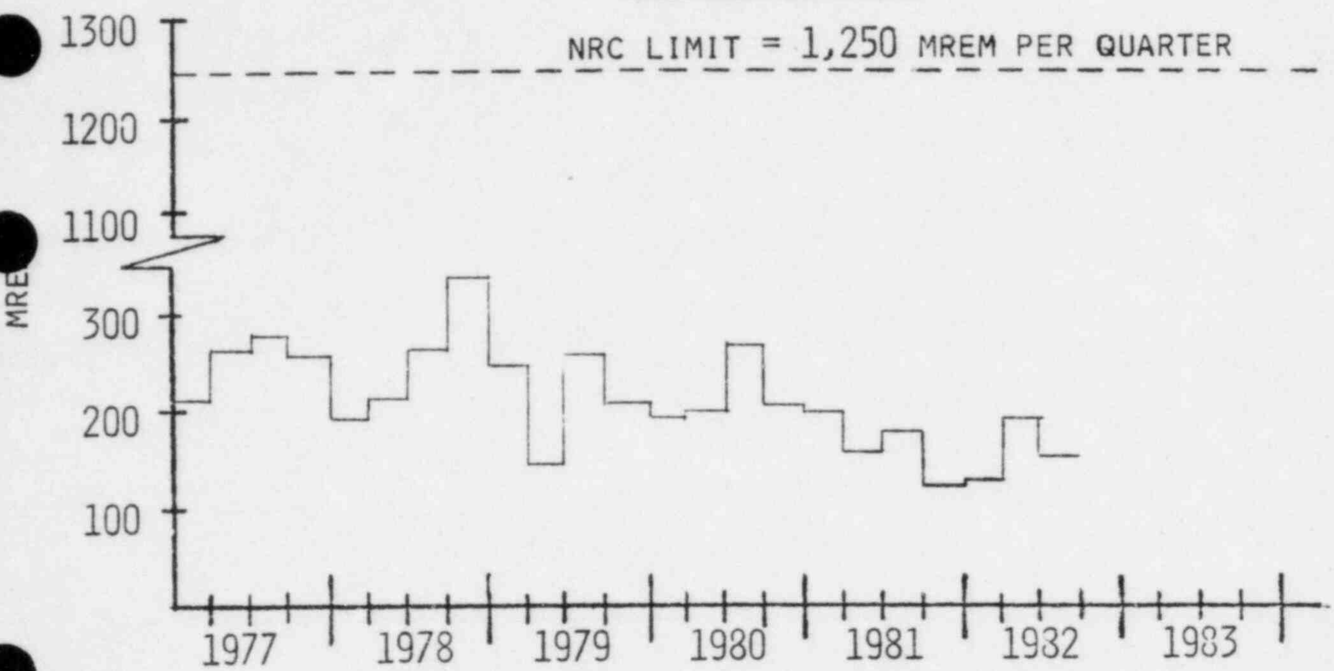
FIGURE 1

AVERAGE QUARTERLY EXTERNAL DOSE

PER EMPLOYEE

AREA: PELLET

WHOLE BODY DOSE



SKIN DOSE

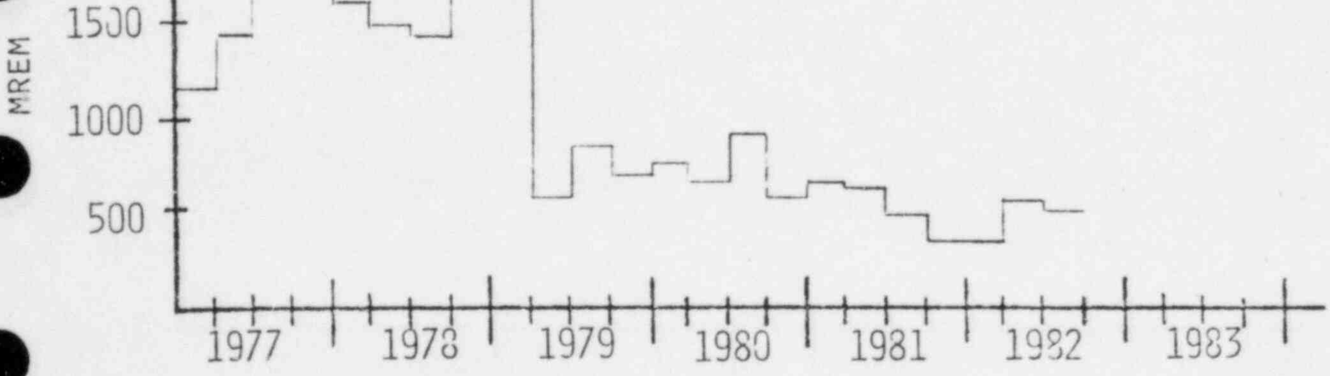


FIGURE 2

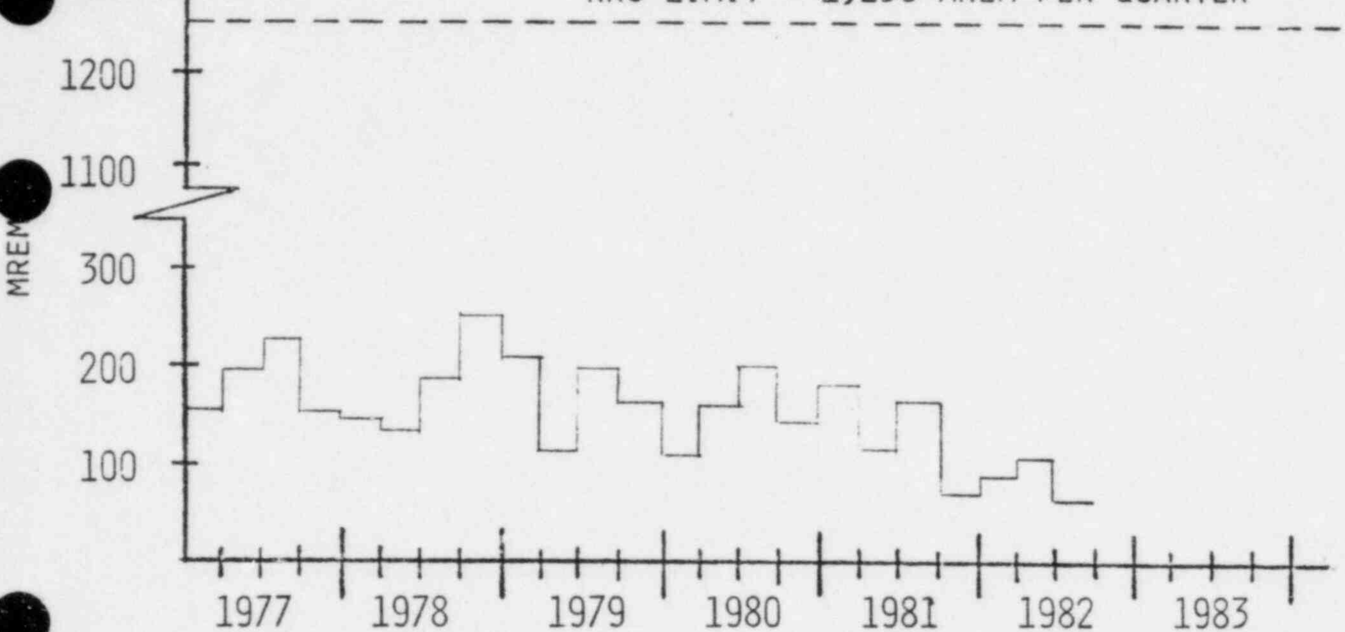
AVERAGE QUARTERLY EXTERNAL DOSE

PER EMPLOYEE

AREA: ROD

WHOLE BODY DOSE

NRC LIMIT = 1,250 MREM PER QUARTER



SKIN DOSE

NRC LIMIT = 7,500 MREM QUARTER

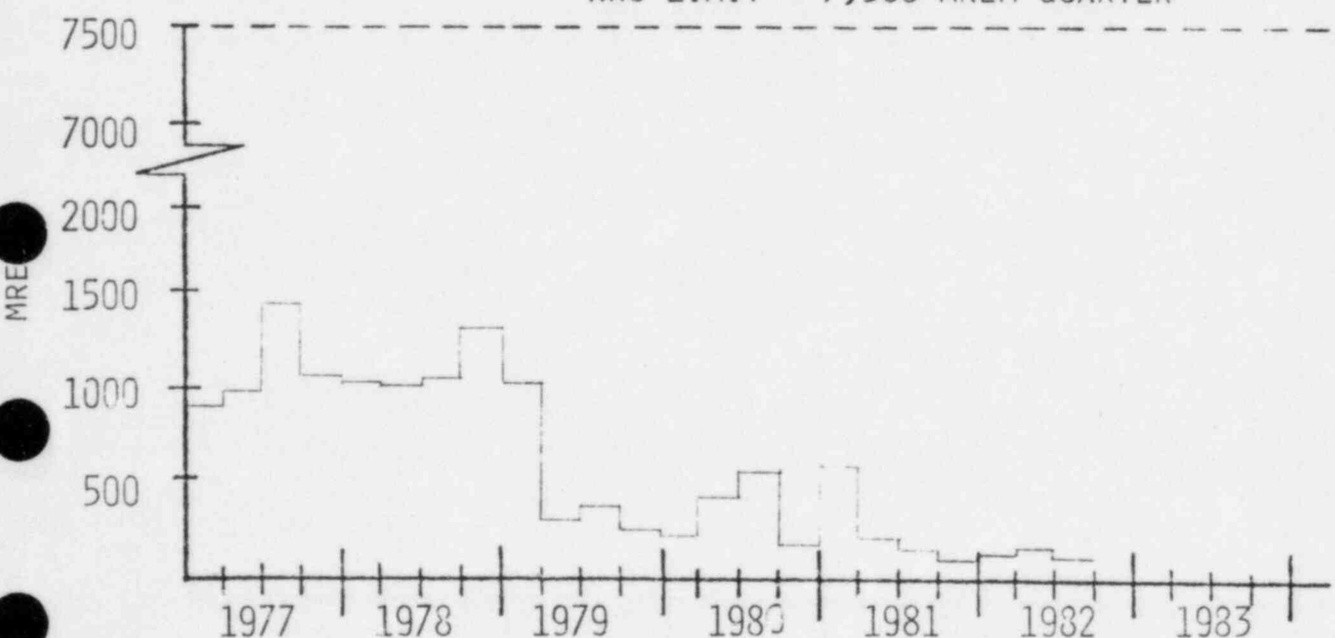


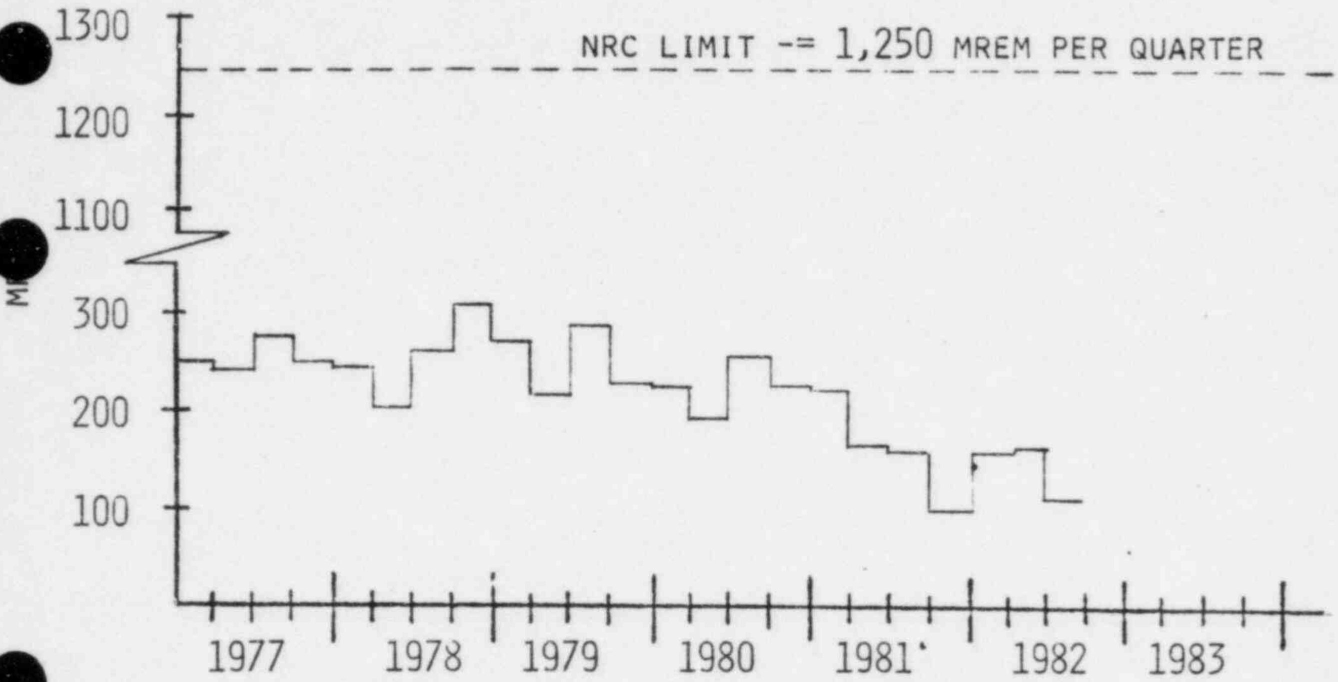
FIGURE 3

AVERAGE QUARTERLY EXTERNAL DOSE

PER EMPLOYEE

AREA: Q.C. CHEMICAL

WHOLE BODY DOSE



SKIN DOSE

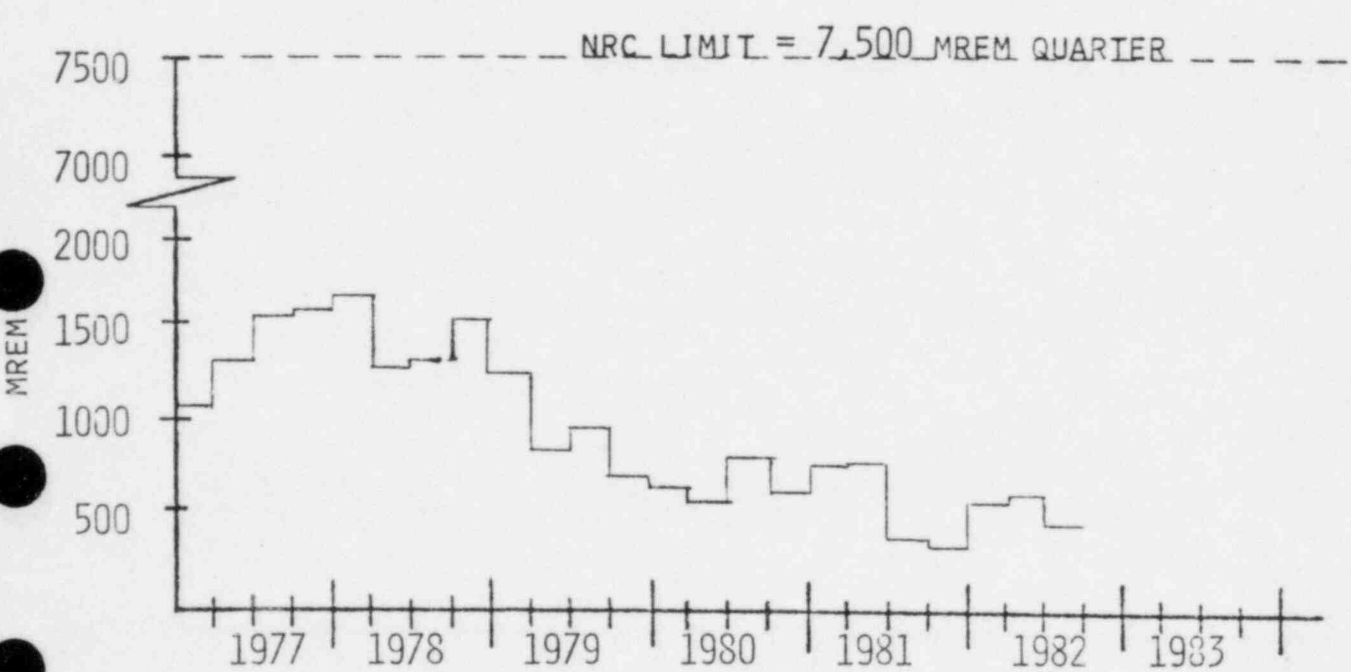


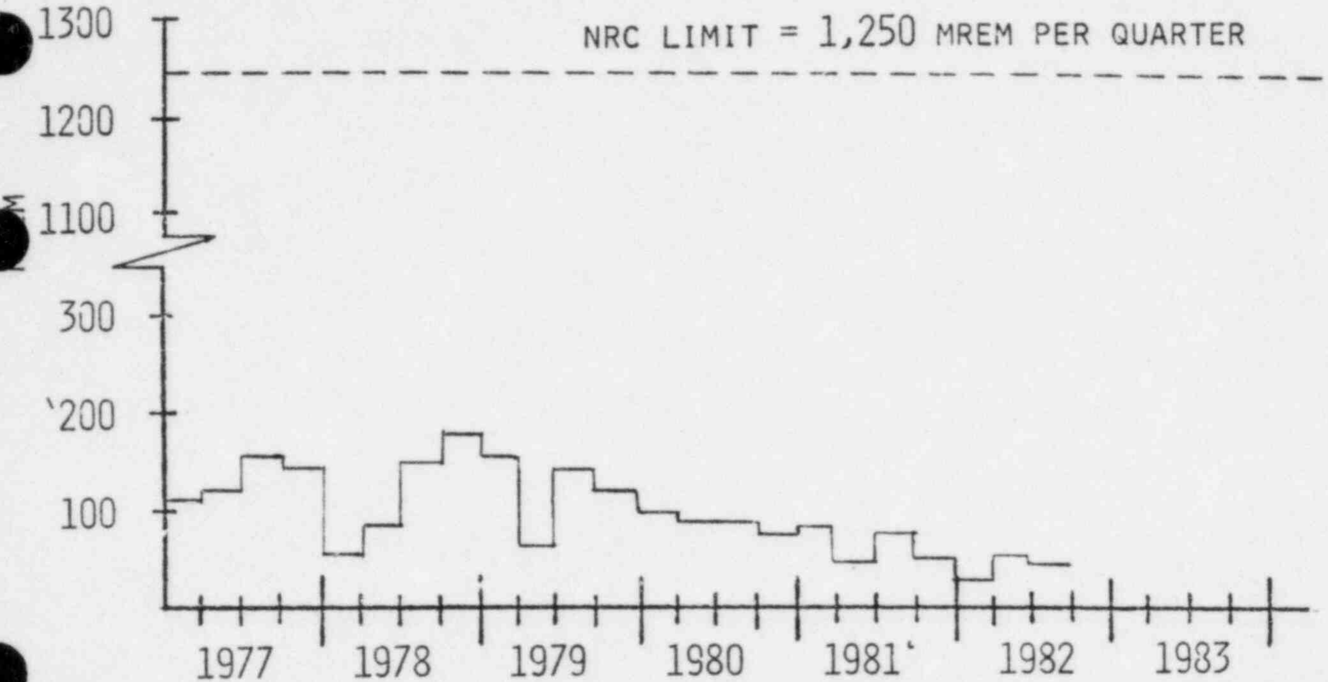
FIGURE 4

AVERAGE QUARTERLY EXTERNAL DOSE

PER EMPLOYEE

AREA: MAINTENANCE

WHOLE BODY DOSE



SKIN DOSE

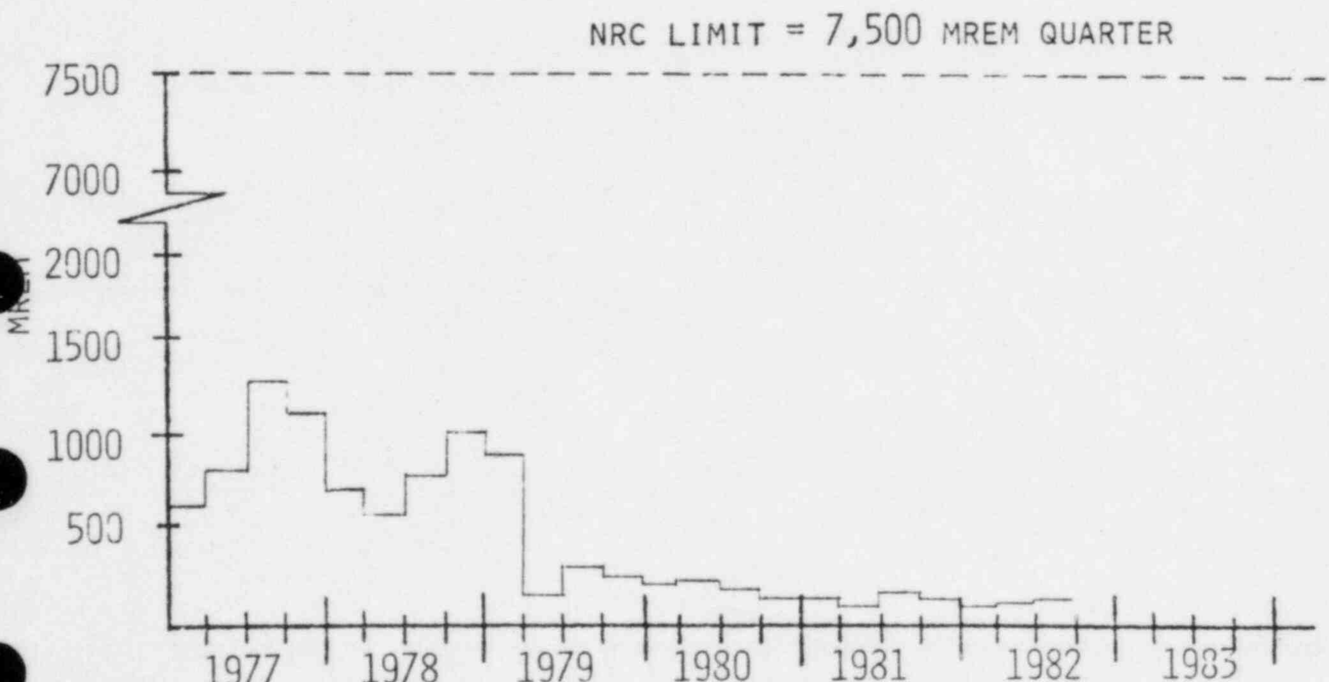


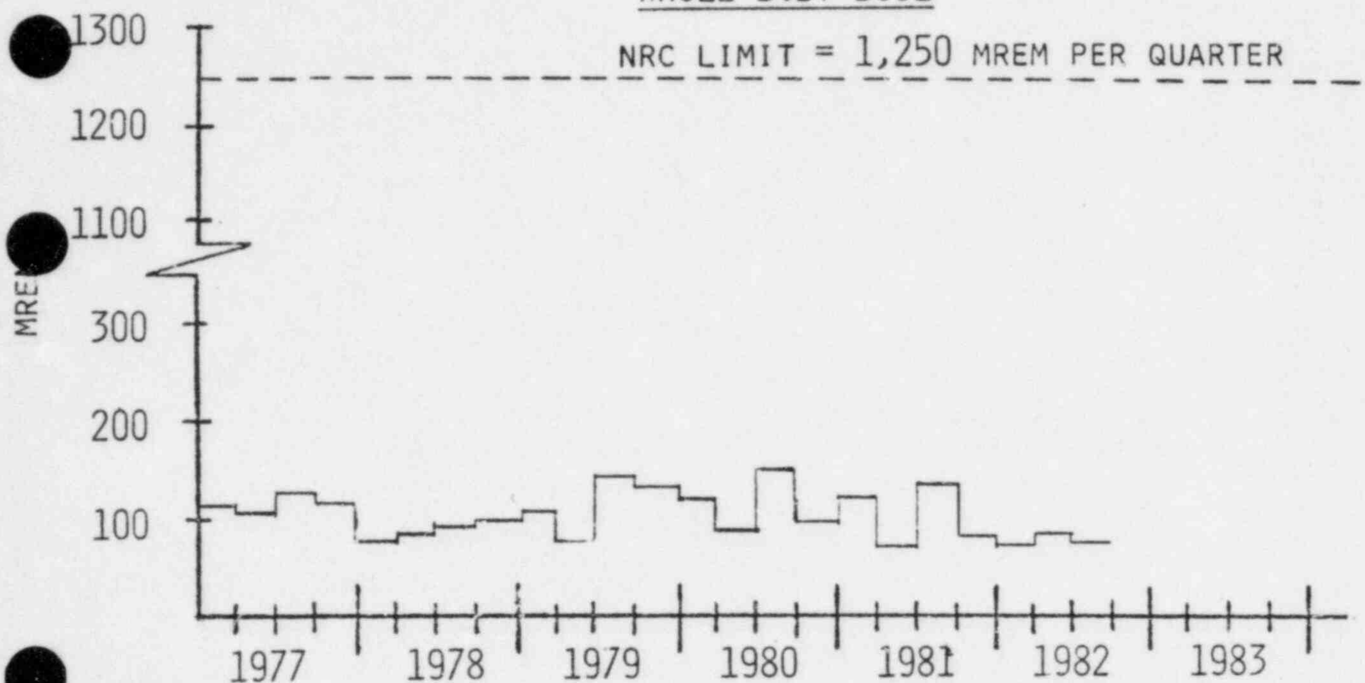
FIGURE 5

AVERAGE QUARTERLY EXTERNAL DOSE

PER EMPLOYEE

AREA: ROD INSPECTION

WHOLE BODY DOSE



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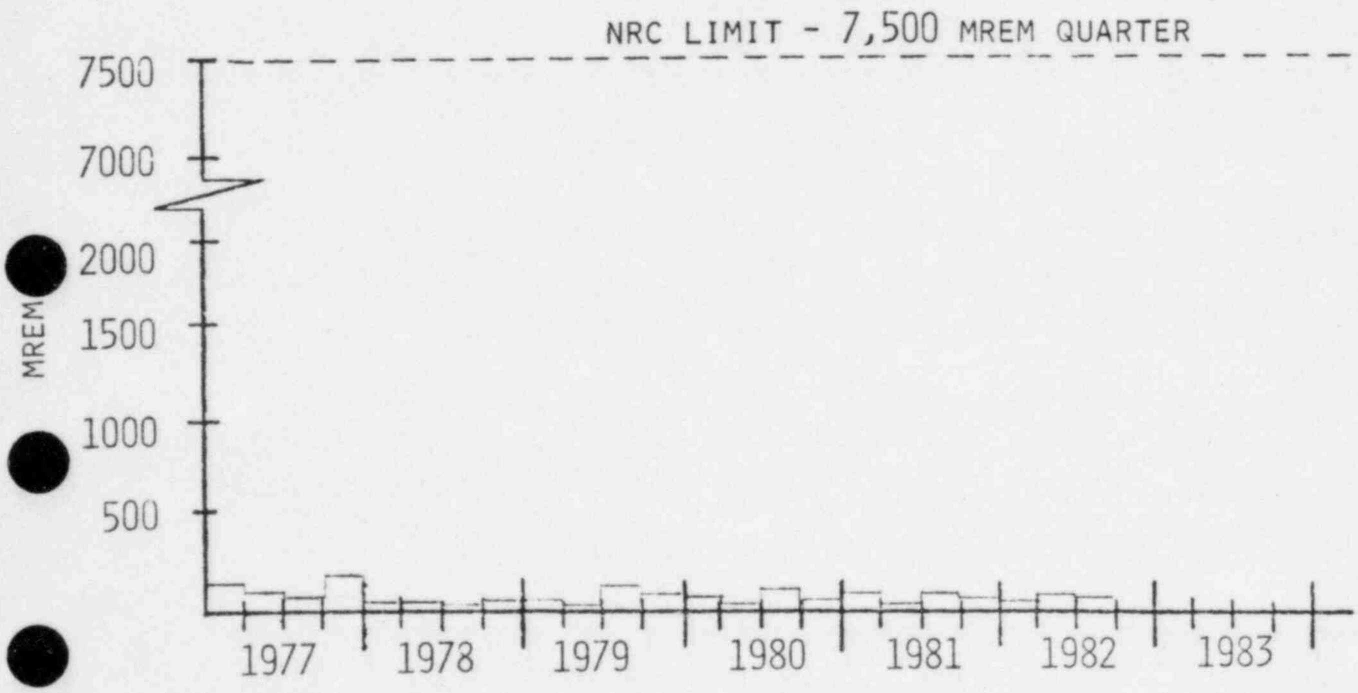


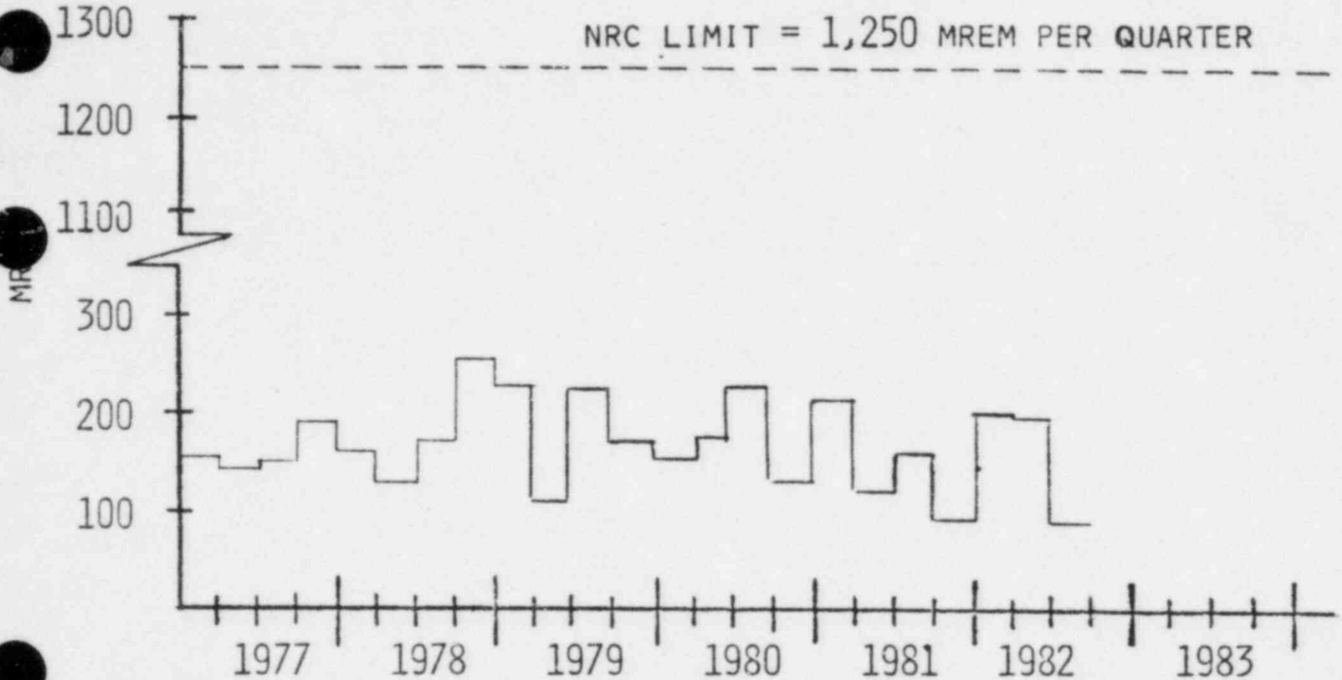
FIGURE 6

AVERAGE QUARTERLY EXTERNAL DOSE

PER EMPLOYEE

AREA: FINAL INSPECTION

WHOLE BODY DOSE



SKIN DOSE

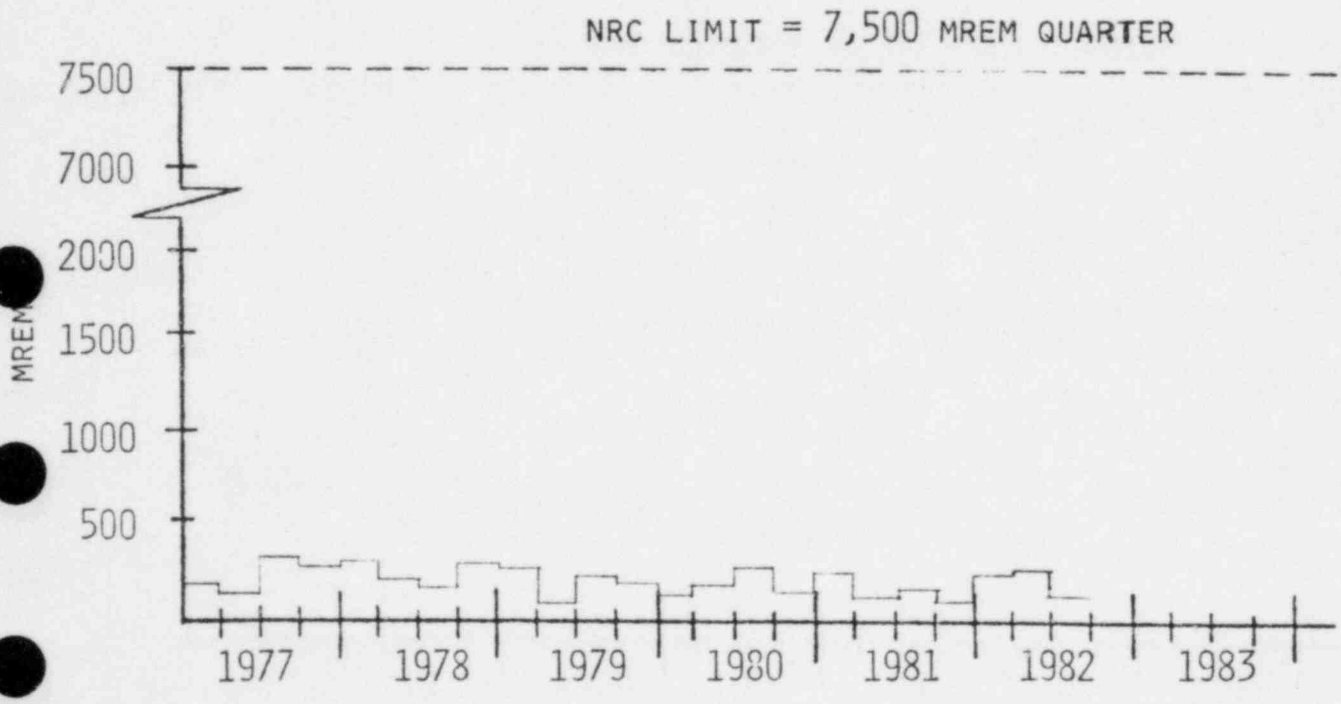


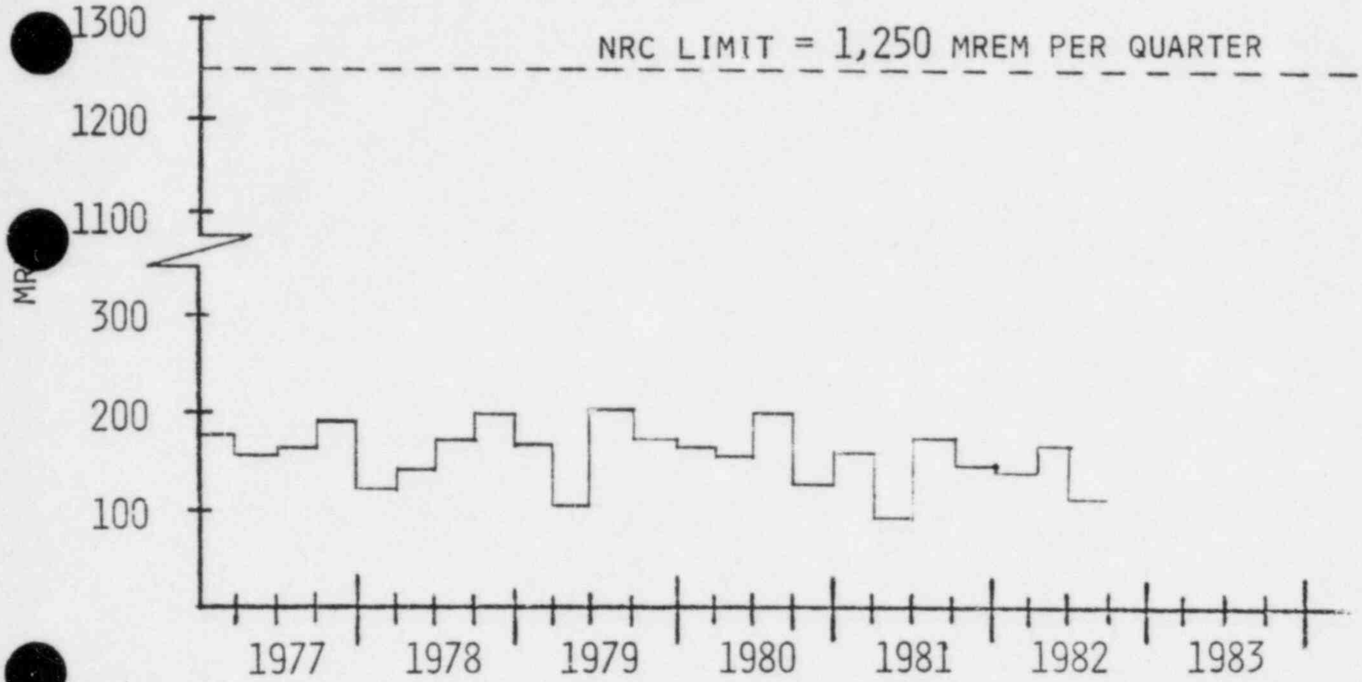
FIGURE 7

AVERAGE QUARTERLY EXTERNAL DOSE

PER EMPLOYEE

AREA: FINAL ASSEMBLY

WHOLE BODY DOSE



SKIN DOSE

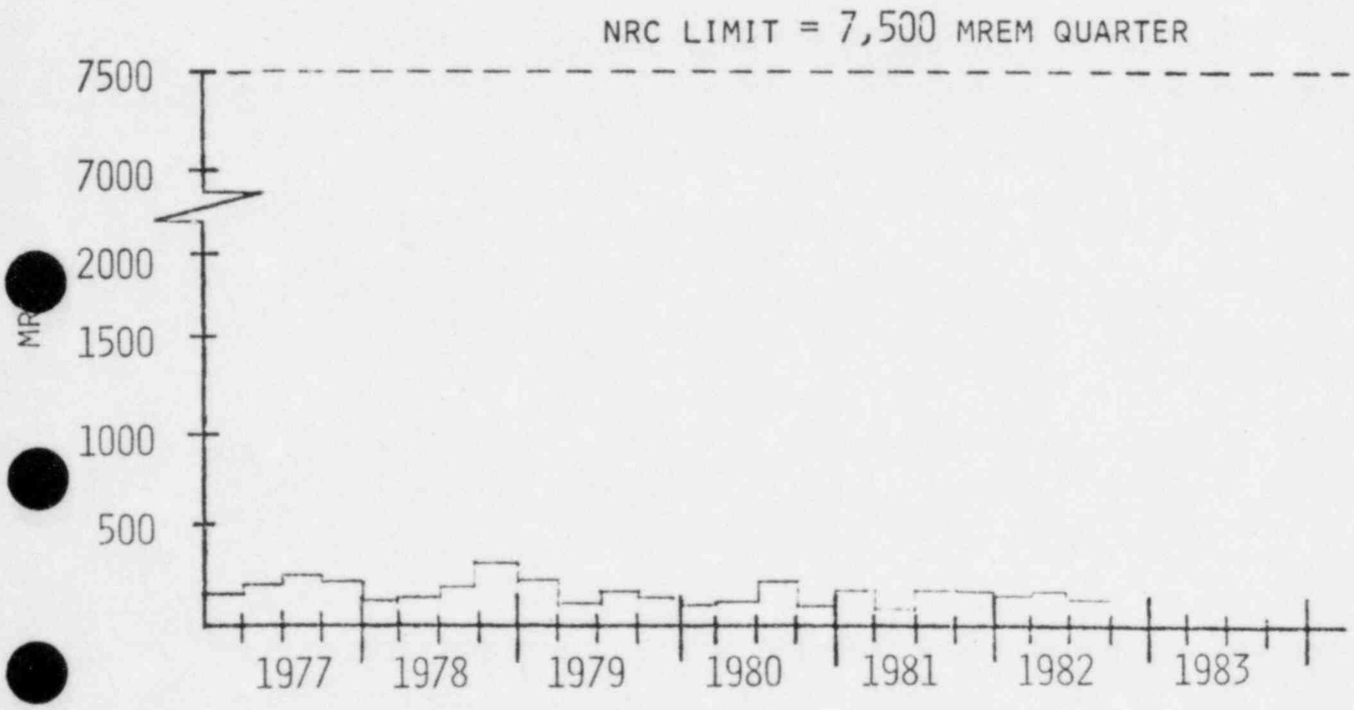
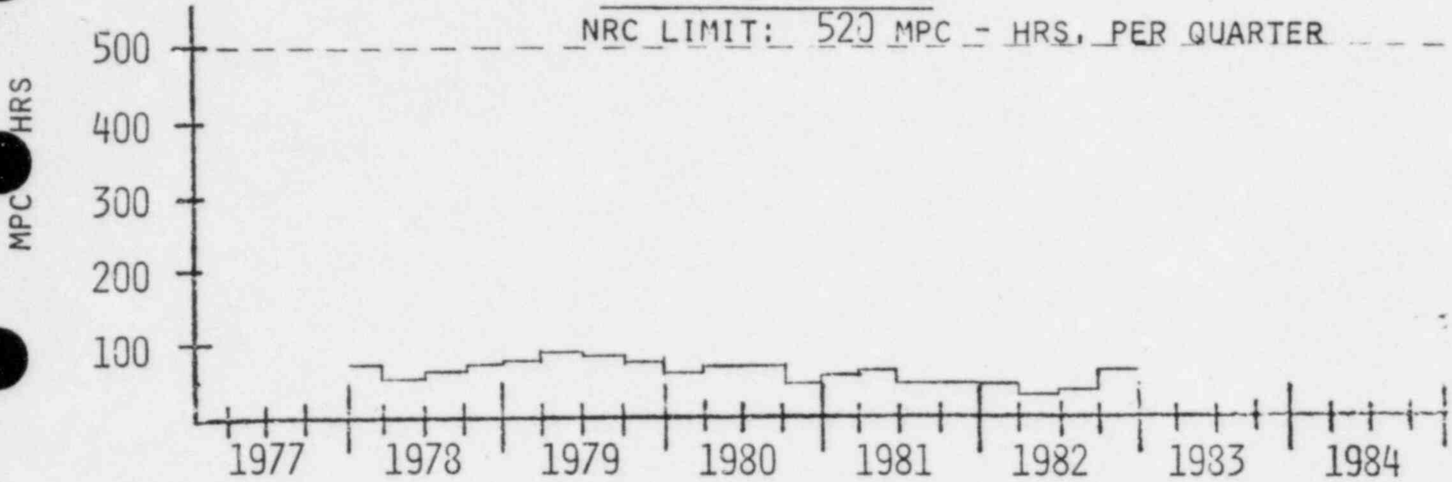


FIGURE 8

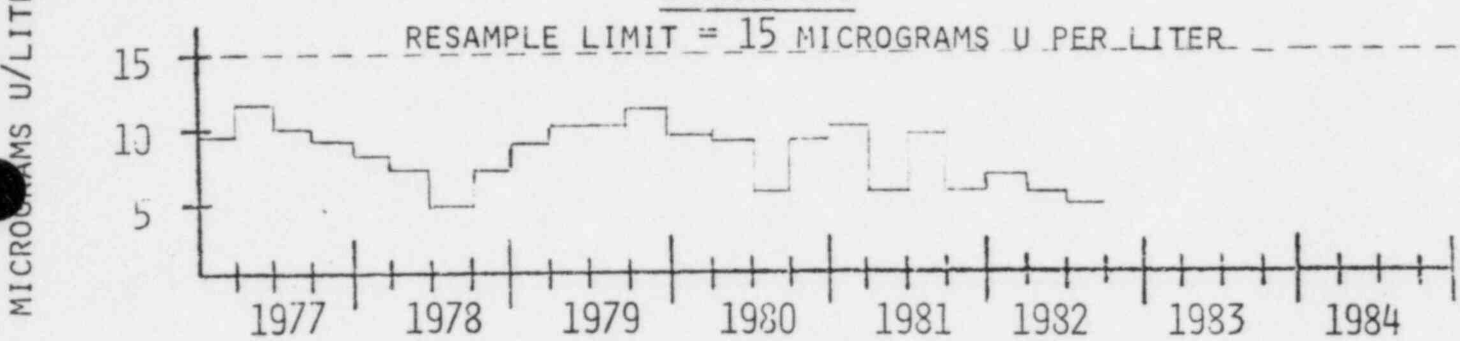
AVERAGE QUARTERLY EXTERNAL EXPOSURE AND BIOASSAY RESULTS PER EMPLOYEE

AREA: CONVERSION

INTERNAL EXPOSURE



URINALYSIS



IN-VIVO

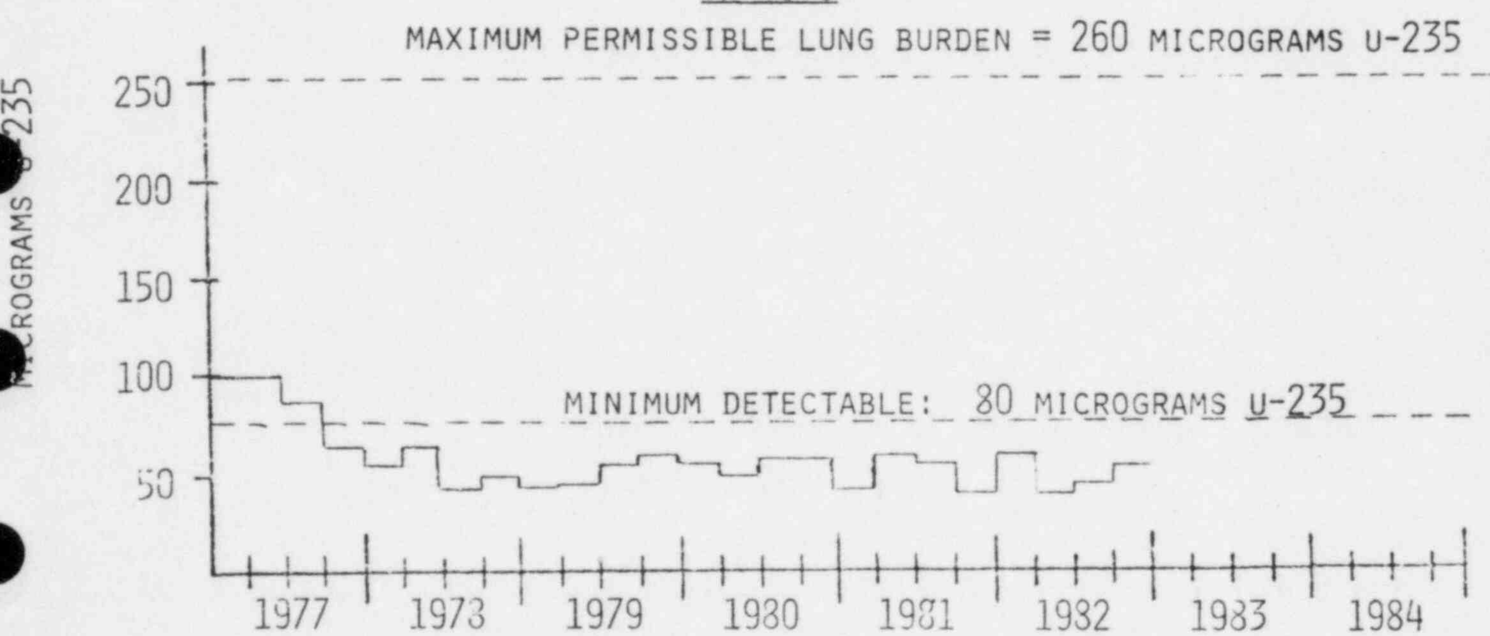


FIGURE 9

AVERAGE QUARTERLY INTERNAL EXPOSURE
AND BIOASSAY RESULTS PER EMPLOYEE

AREA: PELLET

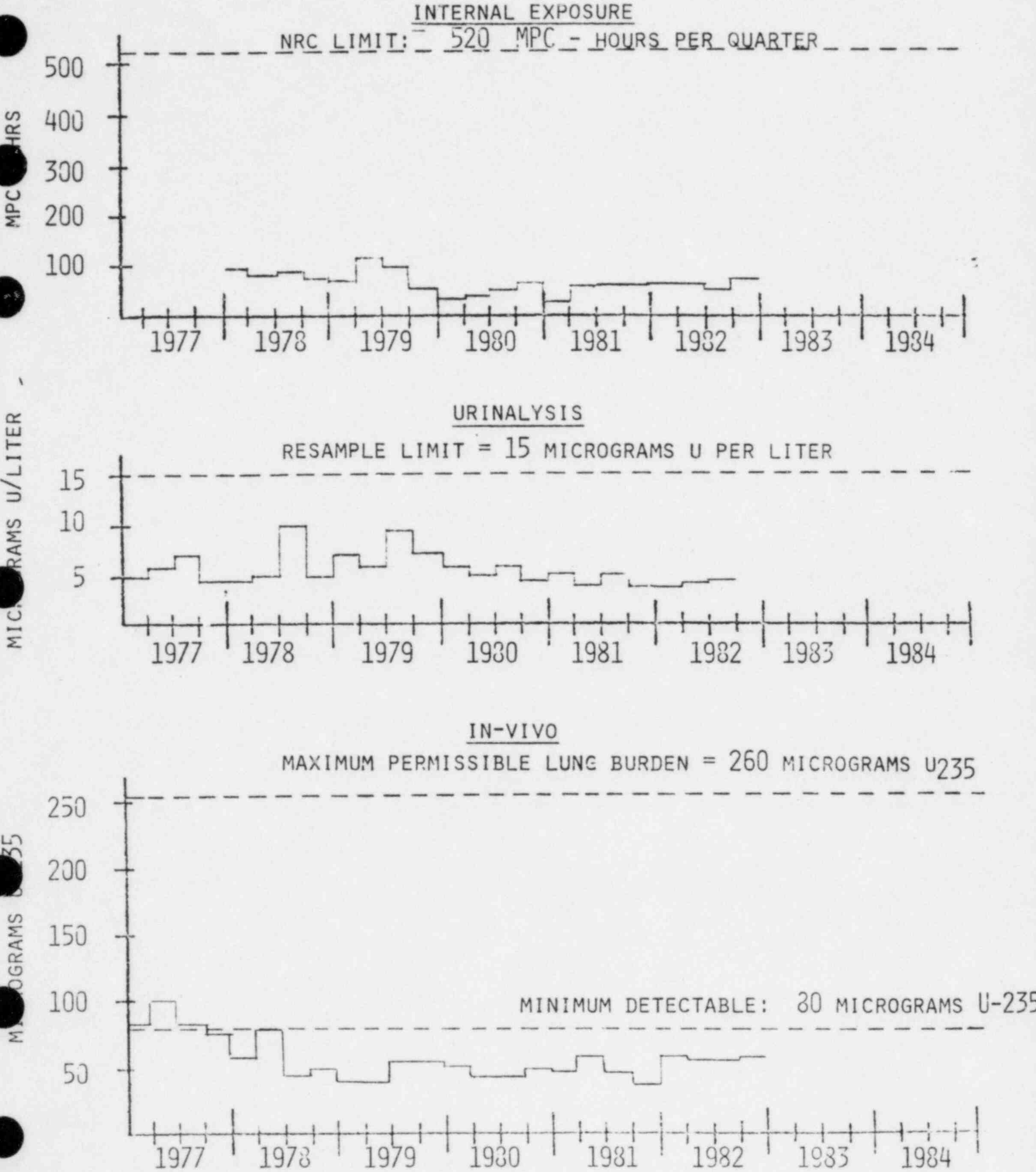


FIGURE 10

AVERAGE QUARTERLY INTERNAL EXPOSURE

AND BIOASSAY RESULTS PER EMPLOYEE

AREA: ROD

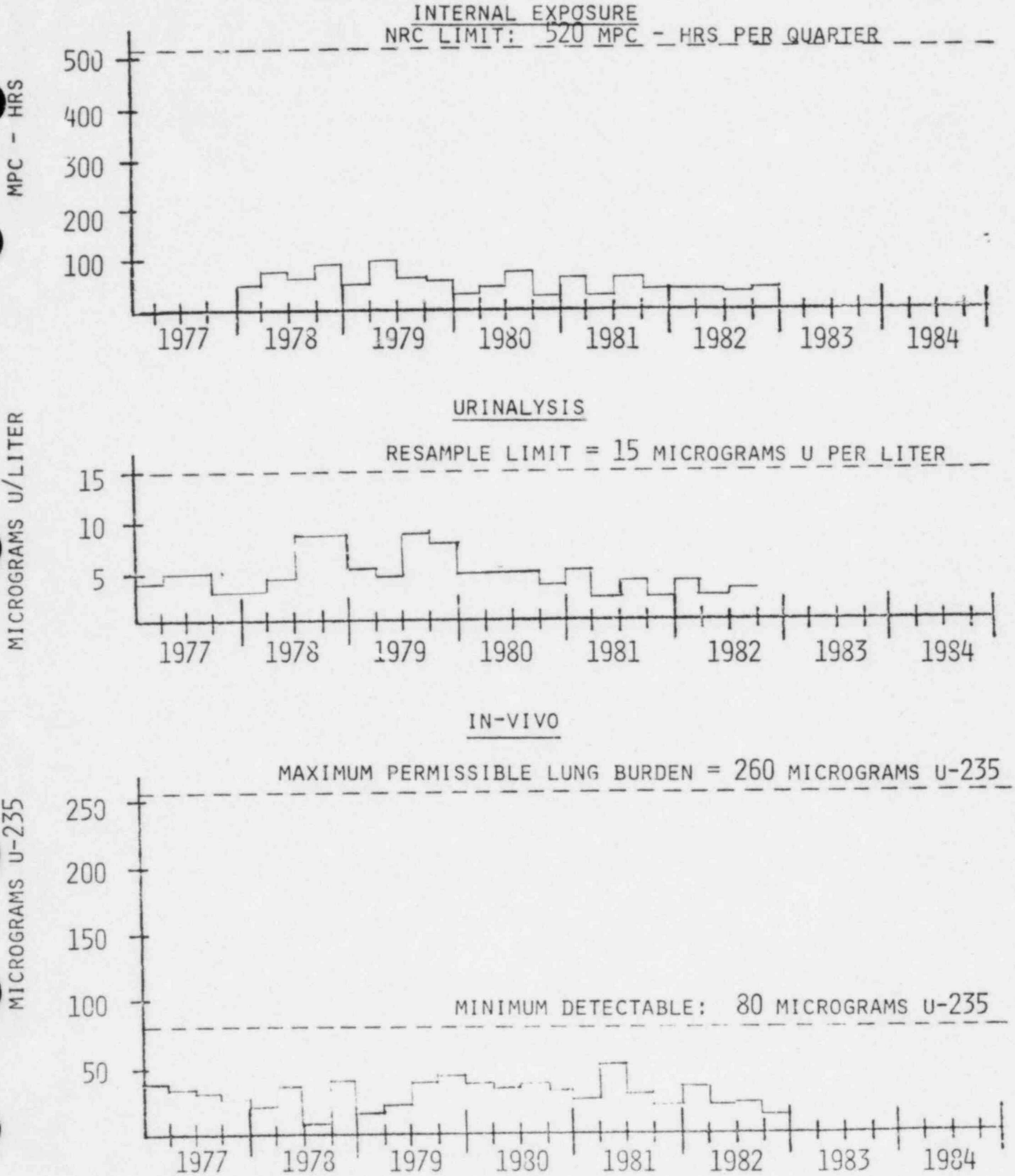


FIGURE 11

AVERAGE QUARTERLY INTERNAL EXPOSURE

AND BIOASSAY RESULTS PER EMPLOYEE

AREA: Q.C. CHEMICAL

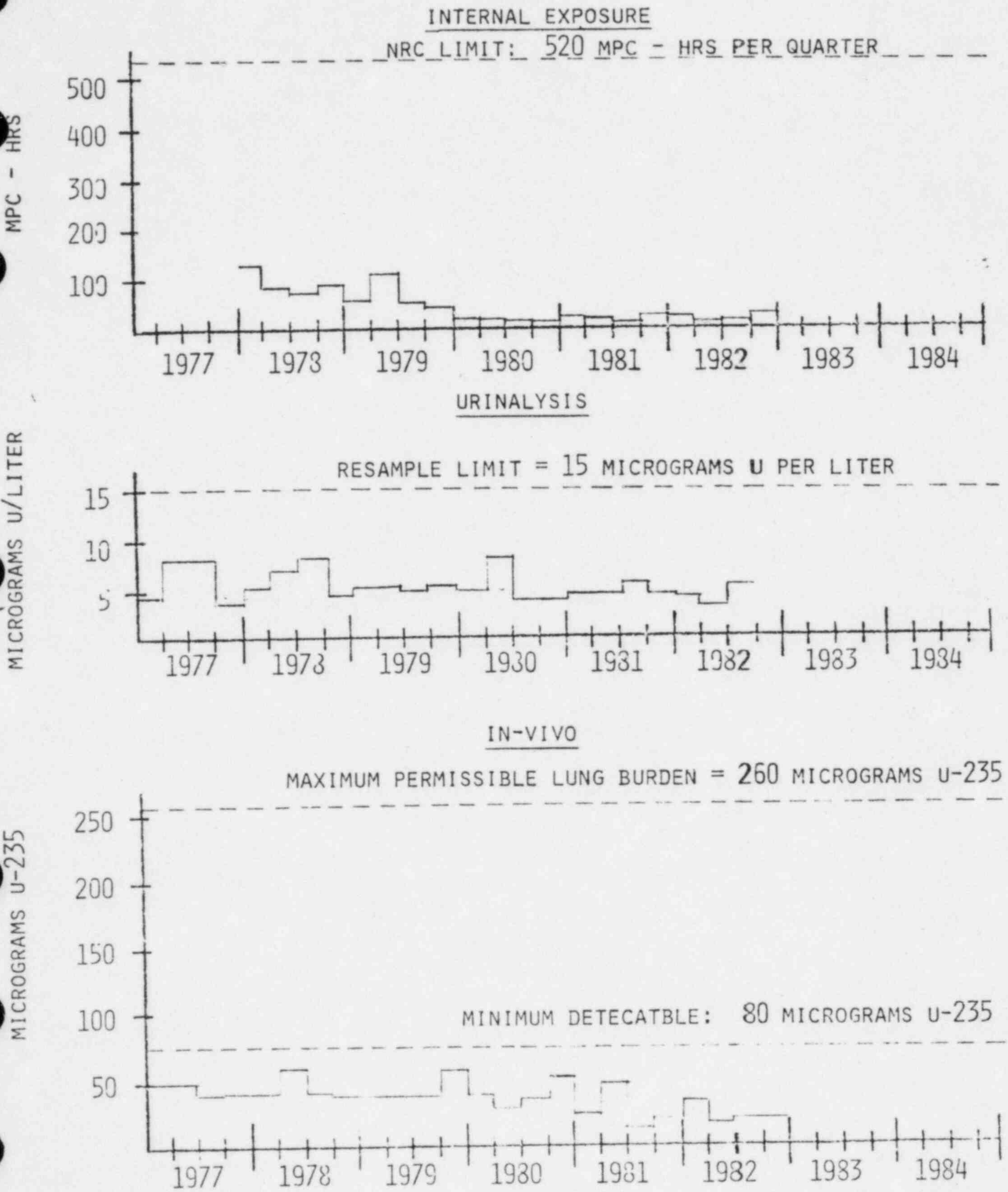


FIGURE 12

AVERAGE QUARTERLY INTERNAL EXPOSURE AND BIOASSAY RESULTS PER EMPLOYEE

AREA: MAINTENANCE

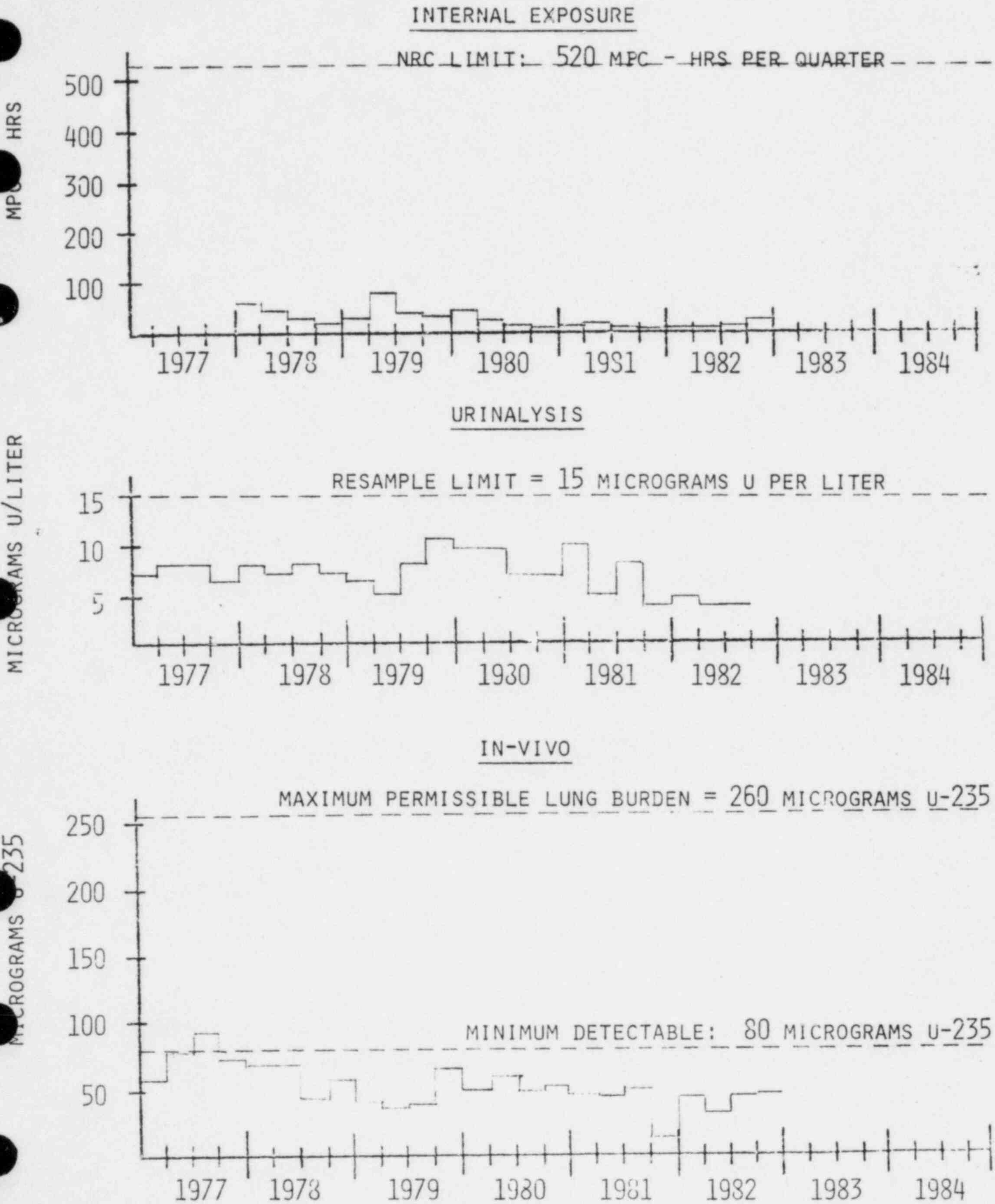


FIGURE 13

AVERAGE AREA AIR CONCENTRATIONS

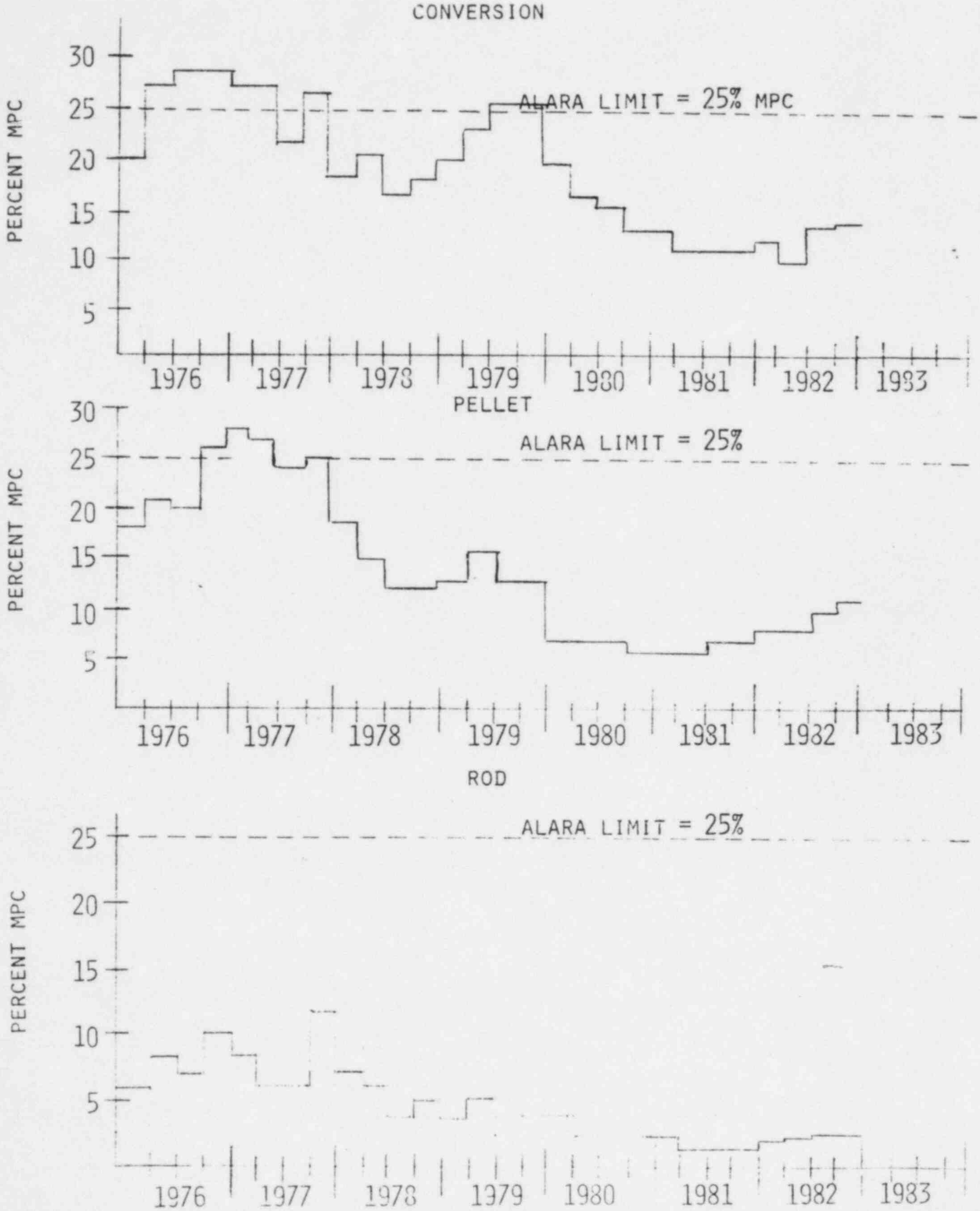


FIGURE 14

NUMBER OF AIR SAMPLES

> 100 % MPC

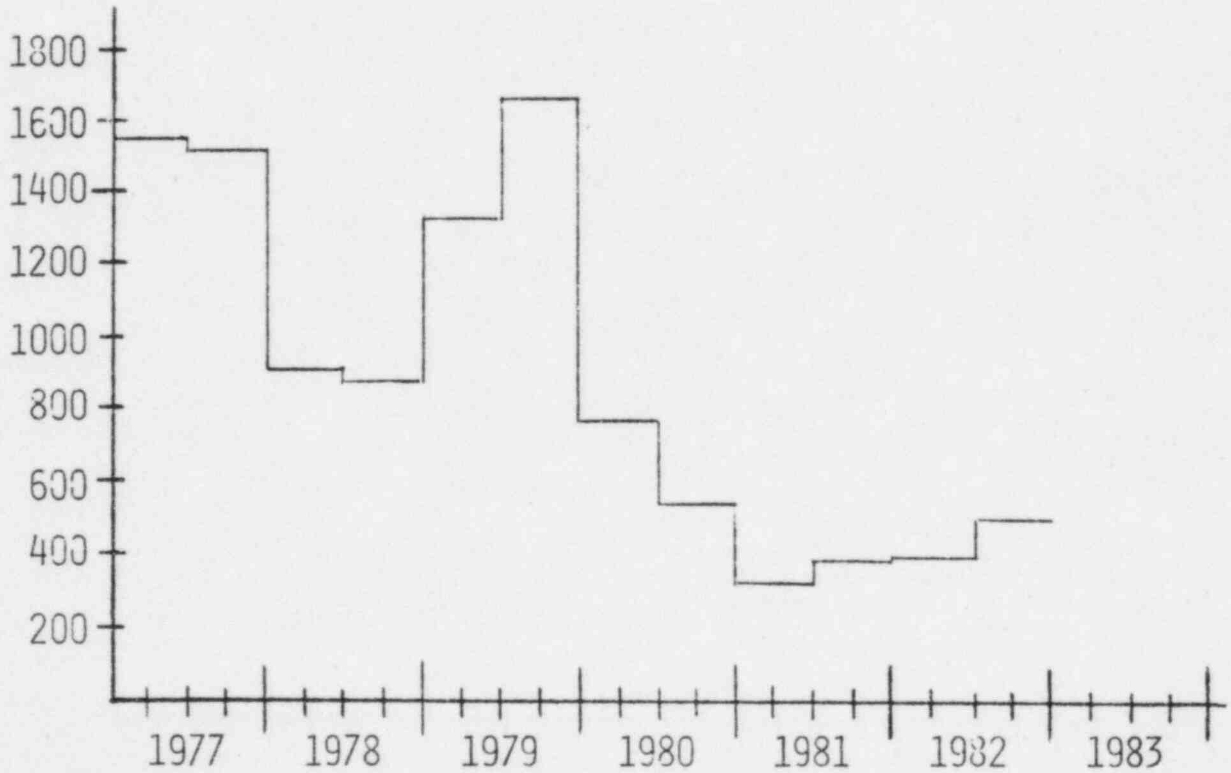
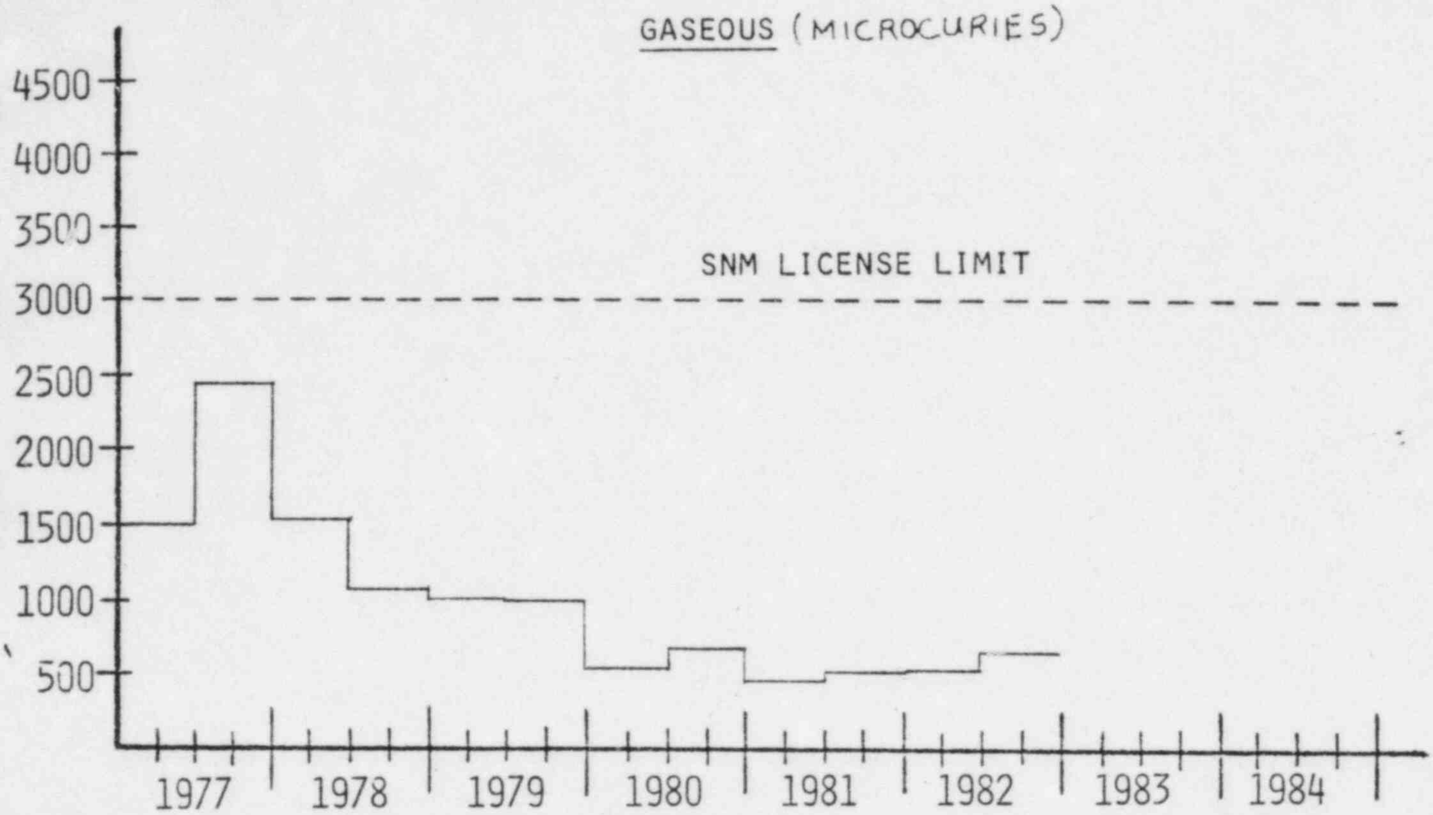


FIGURE 15

LIQUID AND GASEOUS EFFLUENTS



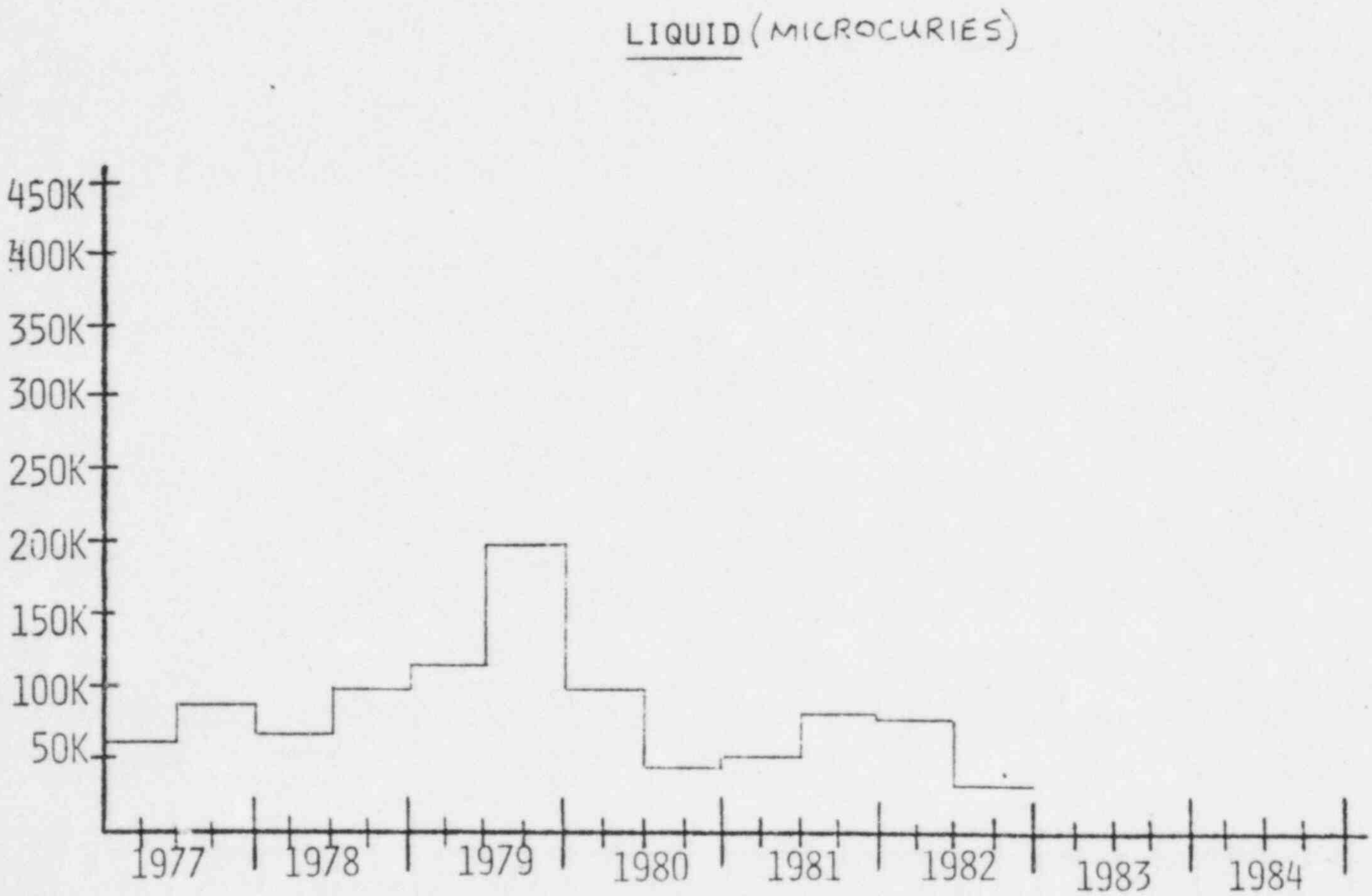
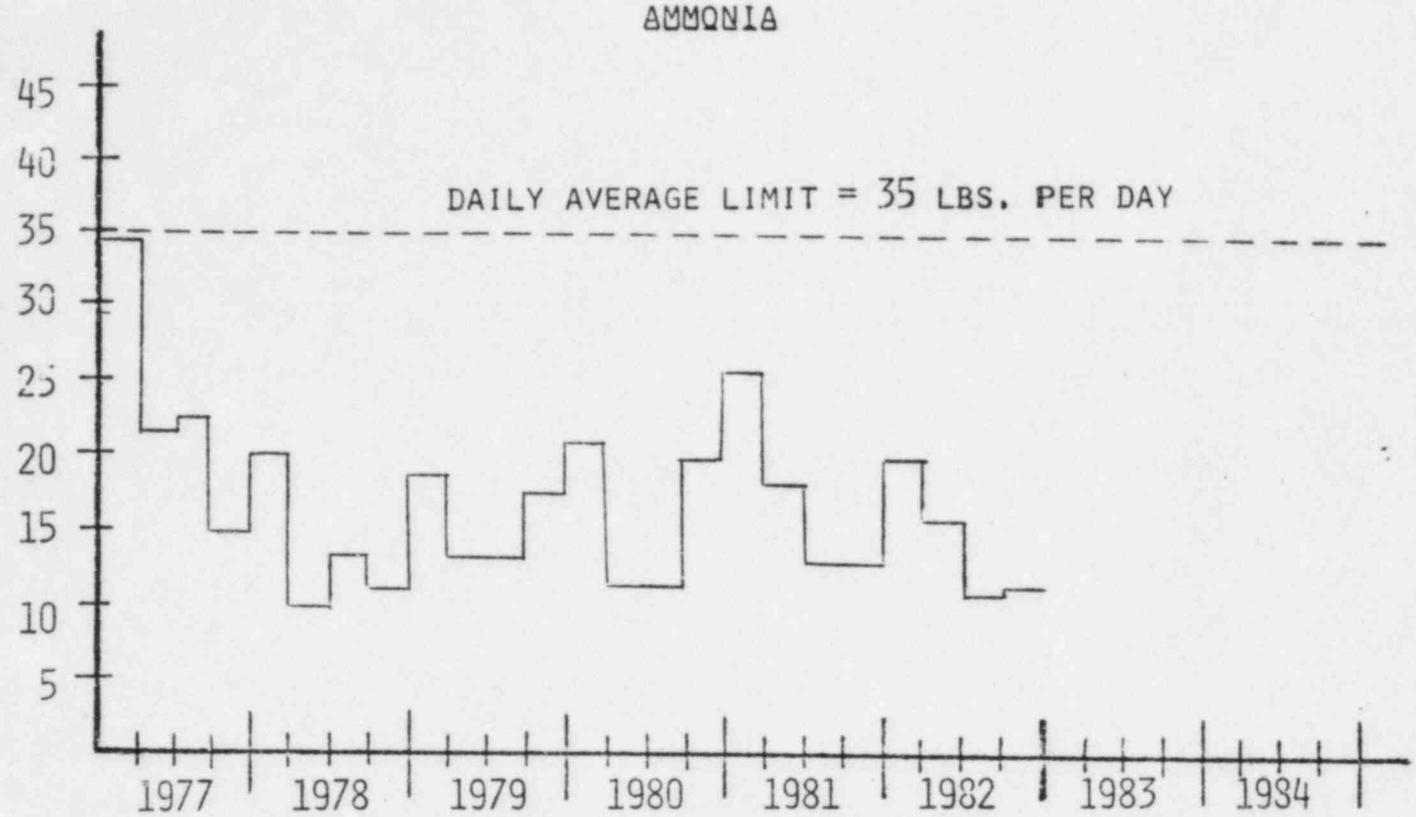


FIGURE 16

AMMONIA AND FLUORIDE EFFLUENTS



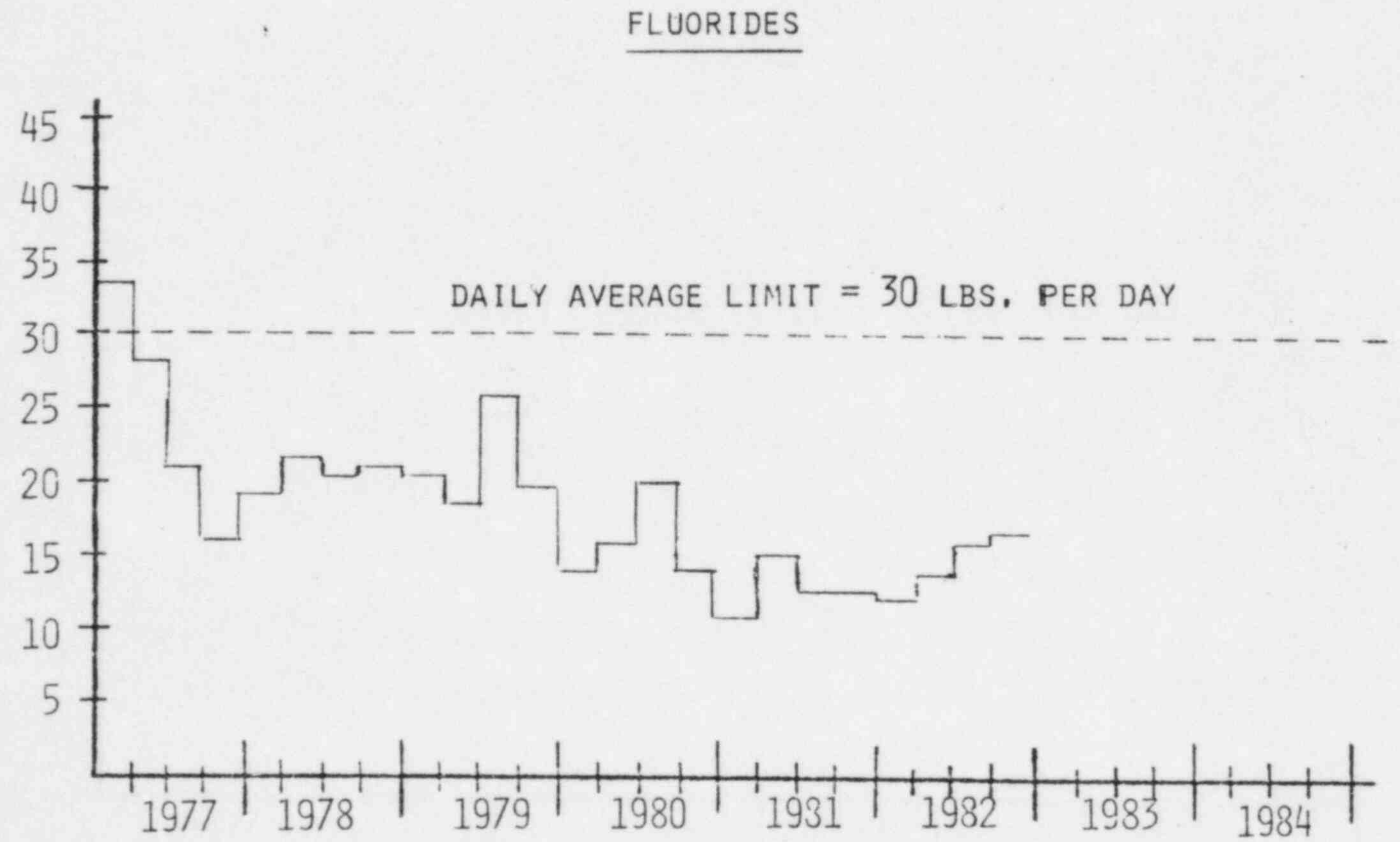


FIGURE 17