# Application for Renewal
# of
# Special Nuclear Material License No. SNM-33
# (NRC Docket No. 70-36)

November 22, 1989

COMBUSTION ENGINEERING

Hematite Nuclear Fuel Manufacturing Facility
Hematite, Missouri

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PART I
LICENSE CONDITIONS

## CHAPTER 1 STANDARD CONDITIONS AND SPECIAL AUTHORIZATIONS

### 1.1 Name, Address and Corporate Information

The name of the applicant is Combustion Engineering, Inc. (C-E). The applicant is incorporated in the state of Delaware with principal corporate offices located at 900 Long Ridge Road, Stamford, Ct 06904. The Nuclear Power Businesses offices are headquartered at 1000 Prospect Hill Road, Windsor, CT 06095. The address at which the licensed activities will be conducted is:

Combustion Engineering, Inc.
Post Office Box 107
Highway P
Hematite, Missouri 63047

### 1.2 Site Location

The Hematite fuel manufacturing facility of Combustion Engineering, Inc. is located on a 154 acre site in Jefferson County, Missouri, approximately 3/4 mile northeast of the unincorporated town of Hematite, Missouri and 31 miles south of the city of St. Louis, Missouri. Activities with special nuclear materials are conducted within a 6 acre, controlled access area near the center of the site and adjacent to the access road, Highway P. Essentially all nuclear fuel manufacturing activity occurs within six contiguous buildings inside the fenced, controlled area. These activities include conversion of UF6 to UO2, fabrication of UO2 nuclear fuel pellets and related processes.

### 1.3 License Number and Period of License

This application is for renewal of Special Nuclear Material License No. SNM-33 (NRC Docket No. 70-36). The current license expires on December 31, 1989 and renewal is requested to cover a period of five (5) years.

### 1.4 Possession Limits

Combustion Engineering, Inc. requests authorization to receive, use, possess, store and transfer at its Hematite site, the following quantities of SNM and source materials:

| Material | Form | Quantity |
|---|---|---|
| Uranium enriched to maximum of 5.0 weight percent in the U-235 isotope | Any (Excluding metal powders) | 8,000 kilograms contained U-235 |
| Uranium to any enrichment in the U-235 isotope | Any (Excluding metal powders) | 350 grams |

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| Material | Form | Quantity |
|---|---|---|
| Source material (Uranium and Thorium) | Any, Excluding metal powders | 50,000 kilograms |
| Cobalt 60 | Sealed Sources | 40 millicuries total |
| Mixed Activation and Fission Product Calibration Sources Including Am-241 | Solid Sources | 200 microcuries total |

## 1.5 Authorized Activities

This license application requests authorization for Combustion Engineering, Inc. to receive, possess, use, and transfer Special Nuclear Material under Part 70 of the Regulations of the Nuclear Regulatory Commission in order to manufacture nuclear reactor fuel utilizing low-enriched uranium (up to 5.0 weight percent in the isotope U-235) and to receive, possess, use, and transfer Source Material under Part 40 of the Regulations of the Nuclear Regulatory Commission. Source materials are used for the same purposes as SNM, and are generally used for the start-up testing of a new process. Sealed cobalt-60 sources and solid sources are used for instrument calibration and testing. Authorized activities are conducted in the following buildings and facilities on the Hematite site:

| Number | Name | Present Utilization |
|---|---|---|
| 101 | Tile Barn | Emergency Center and equipment storage |
| 110 | Office Building | Guard Station and Offices |
| 120 | Wood Barn | Equipment storage |
| - | Oxide Building and Dock | UF6 to UO2 Conversion, UF6 receiving |
| 235 | West Vault | Source material storage |
| 240 | 240-1 | Offices and Cafeteria |
| | 240-2 | Recycle and Recovery area |
| | 240-3 | Incinerator, SNM storage and waste processing |

| Number | Name | Present Utilization |
|---|---|---|
| | 240-4 | Laboratory and Maintenance Shop |
| 252 | South Vault | Radioactive waste storage |
| 253 | Utility Building | Steam supply, SNM Storage, Operating Supplies, Offices, and Liquid Waste Solidification |
| 254 | 254-1 | UO2 storage, blending, and pressing |
| | 254-2 | UO2 oxidation, reduction, dewaxing, and sintering |
| | 254-3 | UO2 grinding and pellet packaging |
| 255 | Pellet Plant | Pellet fabrication, storage, and packaging |
| 256 | Warehouse | Shipping, Receiving and Storage |

### 1.6 Exemptions and Special Authorizations

The following specific authorizations are requested:

(a) Treat or dispose of waste and scrap material containing uranium enriched in the U-235 isotope, and/or source material, by incineration pursuant to 10 CFR 20.302.

(b) We propose to authorize release of equipment and materials from the plant to off-site or from controlled to uncontrolled areas on-site in accordance with "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material", dated August, 1987.

# CHAPTER 2 ORGANIZATION AND ADMINISTRATION

## 2.1 Organizational Responsibilities and Authority

The President, Nuclear Power Businesses has the ultimate responsibility for ensuring that corporate operations related to the Nuclear Power Businesses Division are conducted safely and in compliance with applicable regulations. The President has delegated the responsibility for nuclear fuel manufacturing and product development activities to the Vice President, Nuclear Fuel.

### 2.1.1 Vice President and General Manager, Nuclear Fuel Manufacturing

The Vice President and General Manager, Nuclear Fuel Manufacturing reports to the Vice President, Nuclear Fuel. He has overall responsibility for the operation of Combustion Engineering's nuclear fuel manufacturing facilities located in Hematite, Missouri (SNM-33) and Windsor, Connecticut (SNM-1067). His responsibilities include operations, accountability, security, training, criticality safety, radiological and industrial safety, environmental protection, transportation, materials handling and storage, licensing, process and equipment engineering and maintenance.

### 2.1.2 Plant Manager, Hematite

The Plant Manager, Hematite reports to the Vice President and General Manager for Nuclear Fuel Manufacturing. He directs the total operation of the Hematite facility including the production, accountability, security, criticality safety, radiological and industrial safety, environmental protection, transportation, training, materials handling and storage, licensing, process and equipment engineering and maintenance. He fulfills these functions by delegation to a staff at Hematite that reports to the Plant Manager. He may also request support from the Windsor, CT staff to provide functions that may include criticality analysis, production methods, nuclear licensing and others as needed.

### 2.1.3 Manager, Nuclear Licensing, Safety and Accountability

The Manager, Nuclear Licensing, Safety and Accountability reports to the Plant Manager. He manages radiological protection and industrial safety, SNM accountability, criticality safety, licensing, emergency planning, and environmental protection. His activities include review and approval of procedures for control, sampling, measurement and physical inventory of SNM, auditing of plant operations and evaluation of results from personnel and environmental monitoring. He compares quantitative measurements and other observations of facility activities with the requirements of License No. SNM-33. To enforce compliance, he has authority to halt any operation at the Hematite facility, and the operation shall not restart until approved by the Plant Manager or a duly authorized alternate.

2.1.4 Superintendent, Production

The Superintendent of Production reports to the Plant Manager. The Superintendent directs production operations in accordance with the content of Operation Sheets and Traveler documents. The Superintendent's activities may include review and approval of Operation Sheets and Travelers, scheduling of production Shift Supervisors and of the activities of the Maintenance Supervisor, recommending improvements to equipment, processes and procedures, training and qualification of production operators through their Shift Supervisors and periodically directing the cleanout of the production equipment in conjunction with the physical SNM inventory.

2.1.5 Manager, Engineering

The Manager, Engineering reports to the Plant Manager. He manages the engineering of new equipment and of modifications to existing equipment. With support from his staff, his activities may include recommendation, development and qualification of manufacturing processes, specification of process control methods, design, procurement and installation of processing equipment, preparation, review and/or approval of Travelers, Operation Sheets and Data Logs, and providing assistance in developing procedures for material control and inventory and in developing methods and equipment for sampling.

2.1.6 Nuclear Criticality Specialist

The Nuclear Criticality Specialist is located at Windsor, CT. He reports functionally for criticality evaluations to the Plant Manager at Hematite. The Nuclear Criticality Specialist verifies that equipment, processes and procedures satisfy the criticality criteria in Chapter 4 by performing the review described in Section 2.6. Alternatively, for criticality analyses that require elaborate computational techniques, he may supervise the analysis and review at Windsor. He also performs the annual audit at Hematite required by Section 2.7.

2.1.7 Supervisor, Health Physics

The supervisor of Health Physics reports to the Manager of Nuclear Licensing, Safety and Accountability. He supervises the health physics technicians in the radiological surveillance of activities that involve radioactive materials, in personnel radiation monitoring and in the collection and measurements of environmental samples. He may initiate, for approval by his manager, Travelers and Operation Sheets for non-routine activities involving radioactive materials and may suspend unsafe operations.

2.1.8 Senior Health Physics Technicians

The Senior Health Physics Technicians report to the Supervisor, Health Physics. The Technicians are responsible for the day-to-day monitoring of operations. Monitoring is accomplished through the collection of data which allows the effectiveness of radiological, criticality and industrial safety, environmental protection and emergency planning programs to be assessed. Technicians also monitor the proper implementation of radiation work permits (called Special Evaluation Travelers).

## 2.2 Personnel Education and Experience Requirements

Table 2-1 lists the minimum education and experience requirements for the positions described in Section 2.1.

## 2.3 Hematite Plant Safety Committee

The Hematite Plant Safety Committee meets at least once each calendar quarter to review plant operations, to compare them with the safety requirements of Part I and the License Conditions and to consider other aspects of safety the Committee believes appropriate. The Committee chairman or Plant Manager determines which committee members, as a minimum, shall attend each quarterly meeting according to the topics to be considered. The Committee submits a quarterly meeting report to the Hematite manager level personnel and to the Vice President and General Manager, Nuclear Fuel Manufacturing at Windsor. The Plant Manager appoints the committee members to represent, as a minimum, engineering, production, health physics, and criticality safety. He also approves alternates for the members. Each member shall have at least five (5) years experience in the nuclear industry. The health physics and criticality safety member(s) shall have, as a minimum, the education and experience requirements of the NLS&A Manager and the Nuclear Criticality Specialist, respectively. The Committee Chairman or Plant Manager may invite participation by others from within Hematite or from the staff at Windsor.

## 2.4 Approval Authority for Personnel Selection

Two higher levels of management shall approve personnel for safety-related staff positions.

## 2.5 Training

Hematite staff conduct or supervise the indoctrination of new employees in the safety aspects of the facility. The indoctrination topics shall include nuclear criticality safety, fundamentals of radiation and radioactivity, contamination control, ALARA practices and emergency procedures. After test results demonstrate that a new employee has sufficient knowledge in the above topics, the new employee begins on-the-job training under direct line supervision and/or experienced personnel. The Supervisor monitors performance until it is adequate to permit work without close supervision.

The training and personnel safety program continues with on-the-job training supplemented by regularly scheduled meetings conducted by line supervision and specialists in the subjects covered.. Topics include personnel protective equipment, industrial safety and accident prevention, and other safety topics. Production Supervisors receive training in radiation and criticality control. Testing determines when they have sufficient knowledge to enable them to carry out their training functions. Operating personnel receive a re-training course in criticality control and radiation safety on an biennial basis. The effectiveness of retraining is determined by testing. Formal training shall be documented.

## 2.6 Operating Procedures

Operating procedures, called Operation Sheets, are issued and controlled by Quality Control. They provide the detailed instructions for equipment operation and material handling and the limits and controls required by the License. Operation Sheets are the basic control document; before issuance or revision they receive approval by signature of the cognizant manager or his designated alternate.

Supervision is required to assure that handling, processing, storing, and shipping of nuclear materials is given prior review and approval by the NLS&A Manager or his designated alternate, that suitable control measures are prescribed, and that pertinent control procedures relative to nuclear criticality safety and radiological safety are followed.

Primary responsibility and authority to suspend unsafe operations is placed with Operating Supervision. Within their respective responsibilities, members of NLS&A also have authority to suspend operations not being performed in accordance with approved procedure.

Supervision is further required to assure that, prior to the start of a new activity involving nuclear materials, approved procedures are available. A review procedure has been established for changes in processes, equipment and/or facilities prior to implementation. NLS&A authorization must be obtained for each change involving nuclear safety, radiological safety or industrial safety. NLS&A reviews shall be documented, except for minor changes within existing safety parameters.

The NLS&A Manager or his designated alternate shall grant approval only when:

a. A nuclear criticality safety evaluation has been performed based on the criteria and standards of Chapters 3.0 and 4.0 by a person who meets the education and experience requirements for a Nuclear Criticality Specialist (who may be the NLS&A Manager). This evaluation shall be in sufficient detail to permit subsequent review.

b. The criticality safety evaluation has been reviewed by a person who has fulfilled the education and experience requirements for a Nuclear Criticality Specialist for at least two years (who may be the NLS&A Manager). This review is based on the criteria and standards of Chapter 4.0 and includes verification of each of the following:

    1) assumptions
    2) correct application of criteria of Section 4.0.
    3) completeness and accuracy of the evaluation.
    4) compliance with the double contingency criteria.

c. The NLS&A Manager or his designated alternate has concluded that the operation can be conducted in accordance with applicable health physics and industrial safety criteria.

Review and verification shall include written approval by the reviewer.

The minimum frequency for review, for the purpose of updating, of operating procedures involving Special Nuclear Materials shall be every two (2) years. Updating of operating procedures is the responsibility of the cognizant manager.

## 2.7 Audits and Inspections

Audits and inspections shall be performed to determine if plant operations are conducted in accordance with applicable license conditions, C-E policies, and written procedures. Audits shall apply to safety-related and environmental programs. Qualified personnel having no direct responsibility for the plant operation being audited shall be used to ensure unbiased and competent audits.

Daily checks for safety-related problems are made by NLS&A technicians, who observe, note and make general observations in addition to their other duties. Problems are normally corrected on the spot by the Shift Supervisor. More significant problems are listed on the daily exception report distributed to the Plant Manager and manager level staff. The Superintendent, Production is responsible for corrective action.

Quarterly inspections shall be performed in the areas of radiological and industrial safety by an individual who meets the education and experience requirements of the NLS&A Manager. Quarterly inspections shall also be performed in the area of criticality control by an individual who meets the educational and experience requirements of the Nuclear Criticality Specialist. (This may be the same person performing the radiological and safety inspection, if qualified.) For criticality control, no person may conduct all inspections during the year. Items requiring corrective action are documented in a report distributed to the Plant Manager and manager level staff. The Superintendent, Production is responsible for corrective action, except where another manager is specifically designated. Follow-up actions taken by the Superintendent, Production or responsible manager, shall be documented. Documentation shall be maintained for at least the period stated in Section 2.9.

Annual audits are conducted in which the results of previous inspections or audits are reviewed, as an evaluation of the effectiveness of the program. These audits may also involve a detailed review of non-safety documents such as operation procedures, shop travelers, etc., and are documented by a formal report to the Vice President and General Manager, Nuclear Fuel Manufacturing. Annual audits are performed by a team appointed by the Vice President and General Manager, Nuclear Fuel Manufacturing. The team shall include, as a minimum, a Nuclear Criticality Specialist and a radiation specialist who shall audit criticality and radiation safety. The annual audit will review ALARA requirements in conformance with Regulatory Guide 8.10, as applicable. The NLS&A Manager shall be responsible for follow-up of recommendations made by the audit team.

### 2.8 <u>Investigations and Reporting</u>

Events specified by applicable regulations or license conditions shall be investigated and reported to the NRC. The NLS&A Manager or his designated representative shall be responsible for conducting the investigation and documentation of reportable events. Non-reportable occurrences shall be investigated and documented as appropriate and these reports shall be available for NRC inspection.

### 2.9 <u>Records</u>

Retention of records required to be maintained by the regulations, and by the conditions of this license, shall be the responsibility of the cognizant manager. Records of tests, measurements, and surveys identified as requiring preservation until the NRC authorizes disposition shall be retained indefinitely. Records of NLS&A evaluations and approvals shall be retained for a period of at least six months after use of the operation has been terminated, or for three years, whichever is longer. Other safety significant records shall be retained for at least three years.

## TABLE 2-1

## MINIMUM EDUCATION AND EXPERIENCE REQUIREMENTS FOR KEY PERSONNEL

| Position: Described In Section No. | Position: Title | Education | Experience (Years/Field) |
|---|---|---|---|
| 2.1.1 | Vice President and General Manager, Nuclear Fuel Mfg. | Bachelors, Science or Engineering | 10 Total, 5/Nuclear industry management |
| 2.1.2 | Plant Manager | Bachelors, Science or Engineering | 5/Nuclear Manufacturing |
| 2.1.3 | Manager, NLS&A | Bachelors, Science or Engineering | 5/Health Physics, with 2/Operational health physics with uranium bioassay techniques, internal exposure control, and radiation measurement techniques |
| 2.1.4 | Superintendent, Production | High School Diploma | 10 Total/Nuclear industry, with 5/nuclear fuel manufacturing including 3/Production Coordination |
| 2.1.5 | Manager, Engineering | Bachelors, Engineering | 5/Engineering design of process, systems or facilities |
| 2.1.6 | Nuclear Criticality Specialist | Bachelors, Science or Engineering | 2/Nuclear criticaltiy evaluations |
| 2.1.7 | Supervisor, Health Physics | High School Diploma | 5 Total/Nuclear industry, with 3/Senior Health Physics Technician |
| 2.1.8 | Senior Health Physics Tech. | High School Diploma | 2/Training and experience in Radiation Protection activities |

# CHAPTER 3 RADIATION PROTECTION

## 3.1 Special Administrative Requirements

### 3.1.1 ALARA Policy

It is the policy of Combustion Engineering to maintain a safe workplace and healthful work environment for each employee and to keep radiation exposures to both employees and the general public as low as reasonably achievable (ALARA). The annual audit team, described in Section 2.7 considers ALARA requirements in conjunction with the intent of Regulatory Guide 8.10.

A written report shall be made by the Manager, NLS&A to the Plant Manager every six months providing employee radiation exposure (internal and external) and effluent release data. Trends in the reported data may reveal areas where exposures and releases can be lowered in accordance with the above ALARA commitment. The data may also help to identify problems in personnel exposure, in effluent release, in process control or in equipment for measuring effluent and exposures.

### 3.1.2 Radiation Work Permit Procedures

Operations not covered by an effective operating procedure shall be conducted under a Special Evaluation Traveler (S.E.T.). Prepared by the responsible function, it shall contain detailed instructions for the procedure and shall include all safety requirements to assure that the proposed operation is conducted in a safe manner. The same approvals as required for Operation Sheets shall be required on all S.E.T.s. Completion of the operation shall be appropriately documented as indicated on the traveler.

## 3.2 Technical Requirements

### 3.2.1 Restricted Areas - Personnel Contamination Control

The facility shall be zoned to define contamination areas and clear areas. Protective clothing shall be worn in the contamination areas. A sink and alpha survey meter or alpha monitor shall be provided at the exit from the contamination area. All personnel are required to wash and monitor their hands, and to monitor other body surfaces and personal clothing as appropriate, when exiting a contaminated area. Except for hand contamination which is easily removed on the first rewashing, health physics assistance and approval for release above background levels shall be required.

3.2.2 Ventilation

Air flow shall be from areas of lower to areas of higher contamination. Hoods, glove boxes, or local exhaust will be used to control contamination and airborne concentrations. All dispersible forms of uranium will be handled in ventilated enclosures having sufficient air flow to assure minimum face velocities of 100 Fpm. Face velocities will be checked weekly by NLS&A, except during periods when not in use. This effectively limits HEPA filter pressure differential to less than 8 inches of water. Low velocity blowers are used to preclude filter damage if heavy loading occurs.

Glove boxes under negative pressure will be used where airborne material is actively generated such that ventilated hoods would not be adequate.

Fire prevention and the potential for generating explosive atmospheres will be considered in ventilation design.

Air effluents from process areas and process equipment involving uranium in a dispersible form shall be subject to air cleaning. All exhaust stacks shall be continuously monitored when in operation. Examples of air cleaning equipment that may be used are:

a. Cyclone Collectors.

Used to remove particulates from exhaust streams that are heavily loaded.

b. High Efficiency Particulate Air Filters.

Used in the majority of cases for highest efficiency air cleaning, normally in conjunction with roughing filters to extend useful life and improve reliability.

c. Wet Scrubbers.

Used to clean heavily loaded air streams that are not suited, due to air quality or temperature, to other cleaning methods.

d. Dry Scrubbers.

Used primarily for cleaning air streams containing corrosive agents that render wet scrubbing impractical.

e. Fabric Filters.

Normally used in systems where material impinging on them can be returned to the process using reverse jet, pulsed air or other dislodging methods.

f. Special Filters.

Ceramic or metallic frit filters, usually an integral part of process equipment, may be used for special air cleaning requirements.

3.2.3 Work - Area Air Sampling

3.2.3.1 Air Sampling Criteria

Air sampling shall be performed using fixed location samplers, personal (lapel) samplers, and air monitors.

The type of air sample collected at a specific operation or location shall depend on the type, frequency, and duration of operations being performed. One or more of these sample methods shall be employed at intervals prescribed by the NLS&A Manager. General criteria for sampling are:

a. Fixed location samplers shall be used where uranium handling operations are pursued for extended periods of time, or where short term operations occur frequently. These samplers shall be located in or as near as practical to the breathing zone of the person performing the operations. Fixed sampling may also be used for investigative purposes. In this case, the samples may be collected near the point of suspected release of material.

b. Lapel samplers may be used where work stations are not defined or for supportive measurements and special studies. Continuous air monitors may be used for early warning of unexpected releases.

c. Emphasis shall be placed on sampling new operations or processes until adequate, effective, control of airborne contamination is assured.

3.2.3.2 Airborne Concentrations

a. Airborne levels in excess of 25% of the maximum permissible concentration shall require posting in accordance with 10CFR20 and an investigation of the causes.

b. Airborne levels in excess of the maximum permissible concentration shall require exposure evaluation. Controls to restrict the personnel to less than 40 MPC-hours per week shall be required.

c. Effective air control by ventilation systems shall be assured by face velocity checks performed at least weekly.

d. The room air in all areas where unclad licensed material is processed and where operations could result in worker exposure to the intake of quantities of radioactive material exceeding those specified in 10 CFR 20.103, shall be regularly sampled and analyzed for airborne concentration of radioactivity. The survey frequency shall be in accordance with Table 1 of Regulatory Guide 8.24 dated October 1979, where applicable.

e. If a single air sample indicates the airborne concentration of radioactivity for that area exceeds the MPC in air specified in Table 1, Column 1 of 10 CFR 20, Appendix B, an investigation of the cause shall be made and documented.

f. Where fixed air sampling equipment is used to determine concentration in a worker's breathing zone, the fixed air sampling head shall be reexamined for its representativeness whenever any licensed process or equipment changes are made.

g. Any air samples that are suspected of reflecting releases and high concentrations shall be counted at once to identify any samples with quantities of uranium greater than expected for the sampling location and volume.

3.2.4 Radioactivity Measurement Instruments

The minimum instrumentation required for operational surveillance is listed below. All instruments shall be calibrated at least every 6 months and after each repair that would affect the accuracy, except for criticality detectors, which are calibrated annually and operationally checked quarterly. The manufacturer's calibration of flowmeters, velometers, rotameters and orifices is used.

a. Nuclear Alarm System

The Nuclear Alarm System satisfies the recommendations of Regulatory Guide 8.12, Revision 1, January 1981, "Criticality Accident Alarm Systems".

b. Alpha Counting System

Minimum detectability shall be 10 dpm.

c. Alpha Survey Meters

Minimum counting efficiency - 30% (calibrated to read $2\pi$)
Minimum Range - 0 - 100,000 counts per minute

d. Air Sampling Equipment

Lapel samplers - - 2 liters per minute
Fixed air samplers - - 10-100 liters per minute

e. Beta-Gamma Survey Meters

GM type with maximum window thickness of not more than thirty milligrams per square centimeter.

Minimum range - 0 - 60,000 counts per minute
0 - 20 mR/hr

f. Beta-Gamma Counting System

Minimum detectability shall be 200 dpm

g. Emergency Instrumentation

Emergency instrumentation is listed in the Radiological Contingency Plan (see Chapter 8.0)

3.2.5 Radiation Exposures

Personnel monitoring shall be supplied to each individual who is likely to receive a dose in excess of 25% of the applicable limits in 10 CFR 20 and those personnel who routinely work in process areas. The personnel monitoring device may be either a film badge (changed monthly) or a TLD (changed quarterly)

The personnel dosimeters shall be sensitive to an exposure of 25 millirem. Hand exposures will be determined by surveys. Exposures in excess of 25% of the applicable limits shall be investigated to prevent the total occupational dose from exceeding the standard specified in 10 CFR 20.101.

3.2.6 Surface Contamination

3.2.6.1 Special Surveys

All non-routine operations not covered by operating procedures shall be reviewed by NLS&A and a determination made by NLS&A if radiation safety monitoring is required.

With the exception of incidents requiring immediate evacuation, spills or other accidental releases shall be cleaned up immediately. Criticality restrictions on the use of containers and water shall be followed at all times. The Shift Supervisor and NLS&A must be notified immediately of such incidents. Appropriate precautions such as use of respirators shall be observed.

3.2.6.2 Routine Surveillance

Surveys shall be conducted on a regularly scheduled basis consistent with plant operation and survey results. The frequency of survey depends upon the contamination levels common to the area, the extent to which the area is occupied, and the probability of personnel exposures. The frequency for contamination surveys in plant operating areas shall be as specified in Table 1 of Regulatory Guide 8.24, where applicable. Clear areas with high potential for tracking of contamination may be surveyed more frequently. Areas with a low use factor may be surveyed less frequently.

Cleanup action for restricted areas shall be initiated when surface contamination exceeds the action limits specified in Table 2 of Regulatory Guide 8.24.

Material on processing equipment or fixed on surfaces shall be limited as required to control airborne radioactivity and external radiation exposures.

Contamination limits for release of equipment and materials from the plant to off-site or from controlled to uncontrolled areas on-site shall be in accordance with "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material", dated August, 1987.

3.2.7 Bioassay Program

The bioassay program shall satisfy the recommendations of Regulatory Guide 8.11, "Applications of Bioassay for Uranium", except that in Table 2 semi-annual in-vivo frequencies may be replaced by annual frequencies for minimum programs only.

3.2.8 Respiratory Protection

The respiratory protection program shall be conducted in accordance with Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection".

## CHAPTER 4 NUCLEAR CRITICALITY SAFETY

Nuclear criticality safety shall be assured through the administrative conditions and technical criteria delineated in this chapter.

Administrative conditions define:

a) the design approach employed in processes involving the handling and storage of special nuclear materials (SNM),

b) the lines of responsibility for assuring that criticality safety aspects of the process are reviewed, documented and approved, and

c) the written procedures and postings employed for handling and storage of SNM.

Technical criteria provide details on the limits and controls employed in the distribution of SNM. Details on the technical bases and criteria employed in criticality evaluations are provided as are criteria pertaining to engineered safeguards employed in process controls.

### 4.1 Administrative Conditions

#### 4.1.1 Process Design Philosphy

The process design employed by Combustion Engineering, Inc. to assure criticality safety will meet, when applicable, Reg. Guide 3.4, "Nuclear Criticality Safety in Operations with Fissionable Materials at Fuels and Materials Facilities" and will meet, when applicable, Reg. Guide 3.57, "Administrative Practices for Nuclear Criticality Safety at Fuels and Materials Facilities".

#### 4.1.2 Positions Responsible for Criticality Safety

Section 2.1 describes the responsibilities and authority for key organizational positions affecting safety; Section 2.2 gives the professional requirements for these positions.

#### 4.1.3 Documenting Criticality Evaluations and Reviews

Criticality evaluations associated with facility changes affecting the handling and storage of SNM shall be documented. Documentation shall be sufficiently detailed that an independent reviewer can reconstruct the analysis and bases for the conditions presented. These conditions shall include all assumptions affecting criticality safety process limits and controls.

Criticality evaluations shall be reviewed by a qualified independent reviewer. The independent review shall be documented.

Records of the criticality evaluation and the independent review shall be maintained according to the requirements of Section 2.9 of this License.

4.1.4 Written Procedures

Operations involving the handling and storage of SNM shall be performed according to written and approved procedures. These procedures may be of the following types:

a) Traveler - This document specifies a sequence of operations required for the process or job.

b) Operation Sheets - An Operation Sheet specifies how a given step, operation, or process must be performed. It specifies required process parameters and methods.

c) Special Evaluation Traveler - The Special Evaluation Traveler (S.E.T.), is employed for those jobs involving the handling and/or storage of SNM which are not covered by standard procedures. An S.E.T. may supplant operating procedures in a development and testing environment.

4.1.5 Posting of Limits

Work and storage areas where SNM is handled, processed or stored shall be posted with the nuclear safety limits applicable to that area. NLS&A shall maintain a current record of the location and the content of each posting.

4.1.6 Labeling of Special Nuclear Material

Mass-limited containers employed in the handling or storage of special nuclear material shall be labeled as to their contents. If SNM is in the container, the amount, enrichment and type shall be indicated; if empty, the container shall be so labeled or placed in designated areas for empty containers.

4.1.7 Preoperational Testing and Inspection

Prior to startup of a new or substantially modified process, an inspection shall be carried out by Engineering and/or NLS&A to assure completeness and consistency between safety evaluations and equipment design and installation.

4.1.8 Criticality Safety Design

New processes or changes in existing processes affecting the handling and storage of special nuclear material are evaluated for nuclear criticality safety. Internal procedures require that all facility changes affecting the handling and storage of SNM receive appropriate safety reviews and evaluations.

## 4.2 Technical Criteria

### 4.2.1 Individual Units

#### 4.2.1.1 Safe Individual Units (SIU)

Minimum critical values of safety parameters shall be based on experimental data, optimum moderation, and full reflection. To arrive at a SIU, these minimum critical values shall be reduced by the following safety margins.

| Parameter | Safety Margin |
|---|---|
| Mass | 2.3 |
| Volume | 1.3 |
| Slab Thickness | 1.2 |
| Cylinder Diameter | 1.1 |

The resulting units of SNM are Safe Individual Units when isolated from other units by distance or shielding (see Section 4.2.2).

#### 4.2.1.2 Subcritical Units (Subcrits)

Other subcritical units may use multiparameter controls to achieve criticality safety. The controlled parameters may include, for example, U-235 mass limit or concentration, container volume, limits on internal and/or external moderator, etc.

The configuration and composition of these subcritical units depend upon the process involved. Criticality safety is assured through defined limits and controls. These limits and controls may be implemented by favorable geometry in equipment and design layout, engineered safeguards where necessary, and administrative controls in the form of written and approved operating instructions and postings.

#### 4.2.1.3 Criteria

a) The possibility of accumulation of fissile materials in not readily accessible locations shall be minimized through equipment design or administrative controls or included in the nuclear safety evaluation of the process.

b) Nuclear safety evaluations shall include credible sources of internal moderation, including process variables, when moderation control is employed in a given subcrit.

c) Criticality safety evaluations shall consider the neutron reflection properties of the environment to the SIU or subcrit as well as the the heterogeneity of the fissile/fertile material within the SIU or subcrit on the effective multiplication factor.

d) Nuclear criticality safety margins shall include consideration of credible accident conditions consistent with the double contingency criterion. Safety Margins for SIUs are defined in 4.2.1.1. For subcrits defined in 4.2.1.2, the highest effective multiplication factor, under normal credible operating conditions, shall be less than 0.95 including a two-sigma statistical calculational uncertainty, where appropriate, as well as any other applicable uncertainties and biases.

e) Reactivity hold-down by other than fixed poisons shall not be employed in criticality evaluations. Borosilicate Glass Raschig Rings may be employed in solutions of fissile material in a manner consistent with ANSI/ANS 8.5-1979. The effect of structural parasitics, either normal or enhanced, shall be evaluated in a manner which examines both elastic and inelastic scattering contributions to the multiplication factor. Use of enhanced structural parasitics, e.g., boron stainless steel, shall be contingent upon a program to periodically verify the presence of the parasitic additive.

f) Whenever nuclear criticality safety is directly dependent on the integrity of a fixture, container, storage rack or other structure, design shall include consideration of structural integrity. The fulfillment of structural integrity requirements shall be established by physical test or by analysis by an engineer knowledgeable in structural design.

#### 4.2.2 Multiple Units and Arrays

Criticality safety of the less complex manufacturing operations may be based on the use of limiting parameters which are applied to simple geometries. This approach employs safe individual units which assume optimum moderation and full reflection using published criticality data. Safe individual units may be arrayed using the surface density method. An alternate empirical method is the Solid Angle Method.

A more rigorous method is based on two dimensional transport and/or three dimensional Monte Carlo methods. These methods permit the evaluation of more complex geometric configurations of SNM and the evaluation of multiparameter control methods.

4.2.2.1 Spacing of Safe Individual Units

The following criteria shall be employed:

a) Application of the surface density method of spacing safe individual units requires meeting the following criteria:

1) Each mass limited SIU must have a fraction critical of $\leq$ 0.3, and each geometry limited SIU must have a fraction critical of $\leq$ 0.4.
2) Mass limited SIUs shall be spaced such that the smeared density of the SIUs on a given plane shall not exceed 50% of the minimum water reflected, semi-infinite critical slab surface density, based on optimum moderation.
3) For cylinder and volume limited SIUs, the spacing area shall be based on 25% of the minimum critical water reflected semi-infinite slab thickness.
4) Each SIU shall be centered in its respective spacing area or volume depending upon whether the array of SIUs is two or three dimensional.

b) When either the "fraction critical" or the "smeared" slab thickness limitations cannot be met, the spacing may be established by the solid angle method of TID-7016 (Rev. 2) providing that the applicable criteria on subcriticality of the primary unit and subtended solid angle of interacting units are met.

c) Nuclear safety shall be independent of the degree of moderation between units up to the maximum credible mist density. The maximum mist density will be determined by studying all the sources of water in the vicinity of the single units or arrays. The maximum mist density may be limited by design and/or by administrative controls.

d) Safety margins for individual units and arrays shall be based on accident conditions such as flooding, multiple batching, and fire.

e) Optimum conditions (limiting case) of water moderation and heterogeneity credible for the system shall be determined in all applicable calculations.

f) The water content will be verified to be less than 1.0 w/o in storage cans in the conveyor storage area on a production lot basis (contents of two dry blenders).

g) Vessels and other items of equipment requiring exclusion areas shall have the limits of these areas clearly marked on the floor. SIUs in transit shall not be permitted to enter an exclusion area.

4.2.3 Technical Data

4.2.3.1 Safe Individual Unit Limits

SIU limits which meet the fraction critical criteria for spacing by the surface density method are listed in Table 4-1.

Spacing criteria are as listed in Table 4-3.

Mass limited units may be stacked on a vertical centerline with at least a 10-inch separation. Maximum allowed volume for stacked units shall be 20 liters.

Table 4-2 provides SIU limits for aquecus solutions with enrichments up to 5 w/o U-235. The uranyl fluoride data may be used for UO4.

4.2.3.2 Other Criteria

a) For validated computer analysis methods, the highest effective multiplication factor for normal credible operating conditions shall be less than or equal to 0.95 including applicable biases and calculational uncertainties.

b) The analytical method(s) used for criticality safety analyses and the source of validation of the methods shall be specified.

c) A 35 Kg mass limit may be employed for homogeneous or heterogeneous UO2 in a covered, five gallon, or less, metal container. Heterogeneous UO2 shall include hard, clean scrap, i.e., broken pellets and chips; hard, dirty scrap shall be limited to the SIU values listed in Table 4-1. These containers shall be separated by a minimum of twelve inches, edge to edge, in a planar array.

4.2.4 Special Controls

The following technical criteria shall be employed.

a) Buildings containing fissile materials will not have fire sprinkler systems. Water hoses shall not be used to fight fires in moisture control areas.

b) The hygrometers on the plant air to the Air Mix Blenders in the Oxide Building and to the micronizers and blenders in Building 254 will be set to alarm at a dewpoint no higher than 0°C and checked on a six month interval. The hygrometer on the cooler hopper at the exit of the screw cooler in the oxide building will be set to alarm at a dewpoint no higher than 15°C and checked on a six month period. Upon alarm, automatic or manual action stops the process until the source of alarm is eliminated and the process can be safety continued.

c) The R-2 steam line will have two (redundant) fail-safe shut-off valves, each activated by two independent high and low temperature alarm setpoints on the R-2 reactor. The operability of this system will be ascertained at least once every six months.

d) The moisture content of the UO2 powder transferred into the bulk storage hoppers and the recycle storage hoppers will be verified as being $\leq$ 1 w/o. The instruments used for measuring moisture in UO2 shall be calibrated on a six month interval. Loading and unloading of hoppers shall be done with hoods that prevent water ingress.

e) The R-1, R-2 and R-3 inlet pressure switches will be calibrated at least once every 6 months.

f) The two vertical dissolver vessels in the Recycle/Recovery Area (240-2) shall have a barrier to insure that no significant moderating material can be brought within one foot of the cylindrical tank surface.

g) Dual independent verifications of moisture content in UO2 shall be made prior to transfer of material into the bulk storage hoppers or into the blenders in Building 254.

h) All moderation controlled containers shall be covered such that no moderator can enter the container when external to protective hoods.

i) The number of five gallon or less mop buckets shall be limited to 4 on the second floor and 4 on the third floor of Building 254. The combined number of five gallon or less containers of lubricant and of poreformer shall be limited to 6 on the second floor and 6 on the third floor of Building 254 for each of the two pellet lines for a maximum total of 24 containers.

TABLE 4-1

SIU LIMITS MEETING FRACTION CRITICAL CRITERIA

| U-235 Enrichment w/o | Mass Limits ($KgUO_2$) Homogeneous | Mass Limits ($KgUO_2$) Heterogeneous |
|---|---|---|
| >Nat. ≤ 2.5 | 54 | 50 |
| >2.5 ≤ 3.0 | 41 | 38 |
| >3.0 ≤ 3.2 | 36 | 36 |
| >3.2 ≤ 3.4 | 35 | 33 |
| >3.4 ≤ 3.6 | 32 | 30 |
| >3.6 ≤ 3.8 | 28 | 27 |
| >3.8 ≤ 4.1 | 24 | 24 |
| >4.1 ≤ 4.3 | 22 | 22 |
| >4.3 ≤ 4.5 | 20 | 20 |
| >4.5 ≤ 4.7 | 18 | 18 |
| >4.7 ≤ 5.0 | 16 | 16 |
| | Volume (L) | |
| >Nat. ≤ 3.5 | 31 | 22 |
| >3.5 ≤ 4.1 | 25 | 18 |
| >4.1 ≤ 5.0 | 22 | 17 |
| | Cyl. Dia. (in.) | |
| >Nat. ≤ 3.5 | 10.7 | 9.5 |
| >3.5 ≤ 4.1 | 9.8 | 8.9 |
| >4.1 ≤ 5.0 | 9.2 | 8.4 |

Slab Thickness (in)

| | Homogeneous | Heterogeneous Corrugated Trays | Heterogeneous Randomly Loaded Boats |
|---|---|---|---|
| >Nat. ≤ 3.5 | 4.0 | 4.4 | 4.0 |
| >3.5 ≤ 4.1 | 4.0 | 3.9 | 4.0 |
| >4.1 ≤ 4.3 | 4.0 | 3.7 | 4.0 |
| >4.3 ≤ 5.0 | 4.0 | 3.5 | 4.0 |

TABLE 4-2

AQUEOUS SOLUTION LIMITS FOR U-235 ENRICHMENTS LESS THAN OR EQUAL TO 5 w/o U-235

| | $UO_2F_2$ | $UO_2(NO_3)_2$ |
|---|---|---|
| Mass (Kg U-235) | 0.82 | 1:77 |
| Cylinder Diameter (in.) | 10.0 | 16.4 |
| Slab Thickness (in.) | 4.42 | 8.33 |
| Volume (liters) | 26.9 | 105.5 |
| Concentration (gr 235/l) | 273 | 298 |

TABLE 4-3

SPACING AREAS FOR MASS AND GEOMETRIC LIMITS
FOR 5 W/O U-235

| | Homogeneous | | Heterogeneous | |
|---|---|---|---|---|
| | Limit | SIU Spacing Area (ft2) | Limit | SIU Spacing Area (ft2) |
| Mass (1) | 6.2 $KgUO2/ft^2$ | 2.6 | 5.8 $KgUO2/ft^2$ | 2.8 |
| Volume (2) | 1.1 in. | 8.5 | 1.0 in. | 7.2 |
| Cylinder (2)(3) | 1.1 in. | 5.1 | 1.0 in. | 4.6 |

(1) Smeared density ≤ 50% of minimum water reflected, semi-infinite critical slab surface density.

(2) Smeared density ≤ 25% of minimum critical water reflected, semi-infinite slab thickness.

(3) Per foot of cylinder height.

# CHAPTER 5 ENVIRONMENTAL PROTECTION

## 5.1 Effluent Control Systems

### 5.1.1 ALARA Commitment

Gaseous, liquid, and solid waste streams shall be handled such that radioactivity exposures to plant workers, visitors, and the general public are kept as low as reasonably achievable.

### 5.1.2 Air and Gaseous Effluents

Exhaust air effluents from process areas and process equipment shall be sampled continuously during operations. These stack samples shall be changed at least weekly, except that new operations shall be sampled more frequently until effective control is assured. All samples shall be counted after suitable delay for decay of radon daughters, and the results evaluated. The lower limit of detection shall be no more than 10% of 10 CFR 20, Appendix B, limits.

The control limit for gross alpha activity in exhaust air effluent shall be $4 \times 10^{-12}$ $\mu$Ci/cc. If the control limit is exceeded, averaged over a two week period, an investigation shall be conducted and corrective action taken. A further control limit for total plant exhaust stack effluents shall be 150 $\mu$Ci per calendar quarter. If this control limit is exceeded, a report shall be prepared and submitted to the commission, within 30 days, which identifies the cause and the corrective actions taken or to be taken.

### 5.1.3 Liquid Effluents

Levels of contamination in liquid effluents shall be measured by representative grab sampling of batch discards, by proportional sampling of continuous discharges, or both. Samples shall be collected at or prior to the point of discharge from the waste handling system. Samples shall be analyzed for gross alpha and gross beta activity. The lower limit of detection shall be no more than 10% of 10 CFR 20, Appendix B, limits.

The control limits for alpha and beta activity in liquid effluents shall be:

Alpha - $3.0 \times 10^{-5}$ $\mu$Ci/ml
Beta - $2.0 \times 10^{-5}$ $\mu$Ci/ml

A further control limit for liquid effluent streams which discharge to Joachim Creek (NPDES Outfalls 001 and 002) shall be:

Alpha - $3.0 \times 10^{-6}$ $\mu$Ci/ml
Beta - $2.0 \times 10^{-6}$ $\mu$Ci/ml

If these control limits are exceeded, averaged over a calendar quarter, an investigation shall be conducted and corrective action taken.

## 5.2 Environmental Monitoring

Locations of air particulate, soil, vegetation, well water, surface water and liquid effluent sampling stations shall be established and kept part of the Demonstration Section of this license.

Monitoring locations may be changed only if a documented evaluation by NLS&A demonstrates that a new location provides data that are as representative (or more representative) of conditions likely to impact on the general public, as was the data from the original location.

Environmental samples shall be collected and analyzed as shown in Table 5-1. Sample frequency may vary due to inclement weather, plant shutdown, or operating conditions. More frequent or additional samples may be taken as required for special studies and evaluations.

TABLE 5-1 Environmental Monitoring Program

OPERATIONAL EFFLUENTS MONITORING PROGRAM

| Sample Medium | Sampling Points | Collection & Analysis Frequency | Sample Type | Type of Analysis | Action Level |
|---|---|---|---|---|---|
| Air Effluent | Exhaust Stacks | Continuous & analyze weekly | Particulate | Gross alpha | Two week average MPC |
| Air Effluent | Conversion offgas stack | Continuous & analyze weekly | Gaseous & Particulate | Fluoride & Gross alpha | |
| Liquid Effluent | Site dam | Continuous & analyze weekly | Composite | Gross alpha & beta | Above 10% MPC |
| | Sewage treatment outfall | Weekly | Grab | Gross alpha & beta | Above 10% MPC |

OPERATIONAL ENVIRONMENTAL MONITORING PROGRAM

| Sample Medium | Sampling Points | Collection & Analysis Frequency | Sample Type | Type of Analysis |
|---|---|---|---|---|
| Air | 3 onsite remote | Continuous & analyze weekly | Particulate | Gross Alpha |
| Surface Water | Joachim Creek Above and below site creek outfall. | Monthly | Grab | Gross Alpha & Beta |
| | Joachim and site creek confluence | Quarterly | Grab | Gross Alpha & Beta |
| Ground Water | Plant well | Monthly | Grab | Gross Alpha & Beta |
| | Offsite well (Hematite) | Quarterly | Grab | Gross Alpha & Beta |
| | 3 monitoring wells for evaporation ponds. | Monthly | Grab | Gross Alpha & Beta |
| Soil | 4 locations surrounding plant | Quarterly | Grab | Gross Alpha & Beta |
| Vegetation | 4 locations surrounding plant | Quarterly | Grab | Gross Alpha & Beta & fluoride |

# CHAPTER 6 SPECIAL PROCESSES

## 6.1 Proprietary Information

Information on processes or operations pertaining to activities involving SNM at the Hematite facility are included in this license application and no proprietary documents are submitted in accordance with 10 CFR 2.790 as part of this license application.

## 6.2 Occupational Safety

Policies for industrial safety and occupational hygiene necessary to provide safe and healthful working conditions, and to reduce occupational injuries and illnesses to the lowest practicable level, shall be contained in plant operating procedures, as appropriate. Special attention shall be given to the prevention of fire and explosions within the Hematite facility.

Action levels and corrective actions involving occupational exposure to radioactive materials are given in Chapter 3. Common industrial hygiene practices are employed to protect against non-radioactive hazardous gases including ammonia, fluoride, organic solvent vapors and nitrous oxides.

## 6.3 Emergency Utilities

Emergency electric generators provide electric power to essential loads such as air, water, steam, instrumentation and alarms. These loads transfer automatically to the generators upon a power outage.

## 6.4 Radioactive Waste Management

Low level solid wastes shall be packaged in accordance with all applicable regulations and delivered to a carrier for transport to an approved disposal site.

Non-contaminated solid wastes are disposed of by a commercial waste disposal firm. Non-contaminated equipment may be disposed of to commercial scrap dealers, or by other means.

Conditions of storage of waste, waste containers, and contaminated equipment shall be included in the quarterly NLS&A inspection.

## CHAPTER 7 DECOMMISSIONING PLAN

Combustion Engineering reaffirms that, upon terminating activities involving materials authorized under license SNM-33, affected facilities will be decommissioned in a manner that will protect the health and safety of the public. Combustion Engineering's Decommissioning Plan dated January 12, 1979, and financial assurances in the letter dated March 8, 1979, should be considered to be part of this renewal application.

## CHAPTER 8 RADIOLOGICAL CONTINGENCY PLAN

Combustion Engineering, Inc. shall maintain and execute the response measures and shall maintain the implementing procedures of the Radiological Contingency Plan for the Hematite facility dated December 28, 1987 and incorporated as part of this license application. No change shall be made in this Plan that would decrease its response effectiveness without prior approval of the NRC as evidenced by a license amendment. Changes that do not decrease the response effectiveness of the Plan may be made without prior NRC approval. Each change to the Plan shall be reported to the NRC within 6 months after the change is made.

# CHAPTER 9 GENERAL INFORMATION

## 9.1 Corporate Information

Combustion Engineering, Inc. ("the applicant") is a corporation organized under the corporation law of the State of Delaware. The applicant's Common Stock is registered with the Securities Exchange Commission and is publicly traded on the New York Stock Exchange. The management of the applicant does not now know of any alien, foreign corporation, or foreign government which currently owns or controls more than 1% of the applicant's issued and outstanding Common Stock or which otherwise currently exercises control over the applicant, except for aliens who may from time to time be lawfully employed by the applicant. The names, positions, addresses and citizenship of the principal officers of Combustion Engineering, Inc. are given in Table 9-1.

The president of the Nuclear Power Businesses Division of Combustion Engineering, Inc. reports to the corporate vice president for Power Services Businesses. The line of authority within the Nuclear Power Businesses and the organization of the Hematite fuel fabrication facility are described in Chapter 2 and shown in Figure 11-1.

## 9.2 Financial Qualification

Information on the financial position of Combustion Engineering, Inc. is given in the 1988 10-K report provided at the end of this chapter as Appendix 9-1. Provisions for decommissioning are described in Chapter 7.

## 9.3 Summary of Operating Objective and Process

Combustion Engineering's Hematite, Missouri, plant produces low enriched (up to 5.0% U-235) ceramic fuel for light water reactors. Uranium hexafluoride feed is initially received from the DOE enrichment plants and converted to uranium dioxide powder, using a dry fluidized bed conversion process. Capacity for conversion to oxide powder at the maximum rate is equivalent to 400 MTUO2 per year. Allowing for enrichment cleanout and production scheduling, the annual thru-put capacity is estimated at 300 MTUO2 per year. The UO2 powder may be shipped offsite for further processing, but generally it is fabricated into ceramic fuel pellets on site and then shipped. Additional pellet fabrication capacity, added in 1989, is sufficient to fabricate all the pellets on site.

UF6 is received as a solid in 2.5 ton cylinders. These cylinders are heated in steam chests to vaporize the UF6, which then enters the first fluidized bed reactor. Here it is reacted with dry steam to form uranyl fluoride (UO2F2) and hydrogen fluoride gas. The gaseous HF and excess steam exit the reactor through porous metal filters. The UO2F2 particles move to a second and third reactor where they are pyrohydrolyzed in a reducing atmosphere of "cracked" ammonia to remove residual fluoride and reduce the $UO_2F_2$ to $UO_2$. Offgases from these reactors are also filtered through porous metal filters and then routed with offgases from the first reactor to scrubbers filled with calcium carbonate to remove most of the HF prior to discharge to the atmosphere.

$UO_2$ from the third reactor is cooled and pneumatically transferred to in-process storage vessels. The powder is withdrawn from the storage vessels into a fluid energy mill, where recycle material may also be added. It is then transferred to blenders and withdrawn for shipment or for use in one of the two pelletizing buildings.

For pelletizing, blended powder is agglomerated using either an organic binder and a suitable solvent or a dry powder slugging press introduced in 1989 with the pellet lines in Building 254. The agglomerated powder or slugs are then granulated to insure a consistent press feed and pressed into pellets. "Green" pellets are processed through a dewaxing furnace to remove additives and then passed through a sintering furnace where they densify and achieve the desired characteristics. The sintered pellets are sized using a centerless grinder, dried, inspected and packaged for shipment.

Support operations for the conversion and pelletizing process include material recycle, scrap recovery, cylinder heel recovery, quality control laboratory, maintenance, waste consolidation and disposal, and effluent processing.

## 9.4 Site Description

### 9.4.1 Location of Plant

The C-E Hematite site is located on Highway P in Jefferson County, Missouri, approximately 35 miles south of the City of St. Louis and about 3/4 mile northeast of the unincorporated town of Hematite. (See Figures 9-1, 9-2 and 9-3.)

### 9.4.2 Regional Demography

Jefferson County is predominately rural and characterized by rolling hills with many sizable woodland tracts. The land area is classified as 51% forest, 33% agricultural with crops such as grain and hay, and approximately 16% as urban, suburban, commercial and unused or undeveloped.

The county is part of a dynamic, growing urban region of the St. Louis Standard Metropolitan Statistical Area. Although extensive development has resulted from this growth, agricultural land use is still predominant in the site's environs. Some areas, generally 1/2 to 5 miles from the plant site, have been developed as small to moderate-sized subdivisions within the past decade.

The average population density is 246 people per square mile based on the total estimated 1986 population of 163,800 persons and an area of 666.5 square miles. As shown in Figure 9-2, several towns and unincorporated settlements are wholly or partly within the 5-mile radius of the Hematite site. Festus/Crystal City, located 3.5 miles east of the site and having an estimated 1986 population of about 12,700 people, is the nearest town of significant size.

### 9.4.3 Climatology and Meteorology

General climatological characteristics of the site area can be approximated by those of St. Louis, about 35 miles NNE of the site and the location of the nearest U.S. Weather Bureau recording station. Both are located near the Mississippi River and the geographical center of the United States. The region experiences a modified continental climate without prolonged periods of extreme cold, extreme heat or high humidity. To the south is the warm, moist air of the Gulf of Mexico, and to the north, Canada is a source of cold air masses. The alternate invasion of the region by air masses from these sources produces a variety of weather conditions, none of which is likely to persist for any length of time.

Winters are brisk but seldom severe. Snowfall has averaged less than 20 inches per winter season since 1930. Minimum temperatures remain as cold as 32°F or lower fewer than 20 to 25 days in most years. Summers are warm with a maximum temperature of 90°F or higher an average of 35 to 40 days per year. The normal average annual precipitation for the St. Louis area is a little over 35 inches. The three winter months are the driest, the spring months are normally the wettest and it is not unusual to have extended periods of 1 to 2 weeks or more without appreciable rainfall from the middle of the summer into the fall.

Thunderstorms occur on the average between 40 to 50 days per year. During any year there are usually a few of these that can be classified as severe storms with hail and damaging winds. The U.S. Department of Commerce reports a mean annual frequency of about 8 tornadoes per year for a 30-year period. Eight tornadoes were reported in the State of Missouri in 1979. The probability of a tornado striking this particular location is computed as $7.51 \times 10^{-4}$ and the recurrence interval is 1,331 years.

### 9.4.4 Hydrology

The major stream near the plant is Joachim Creek, which runs into the Mississippi River. This stream meets state water quality standards. The United States Department of the Interior, Geological Survey, maintains a flow gauge at the bridge crossing Joachim Creek at Hematite. Only a small number of observations have been taken but data indicate that the annual mean flow is about 132 cubic feet per second (cfs). The seasonal mean flows are: spring - 330 cfs, summer - 12 cfs, fall - 16 cfs, winter - 169 cfs.

Wells drilled into bedrock aquifers in the Joachim Creek watershed may encounter confined or artesian groundwater. In general, groundwater movement is southeasterly towards Joachim Creek. Yields of wells vary, depending on what rock units are penetrated. Wells finished in St. Peter Sandstone through Lower Gasconade Dolomite have yields of more than 100 gallons per minute, while wells finished in Cambrian age sediments open to Ordovician age sediments have yields up to as much as 500 gallons per minute. Wells drilled in any of these areas could expect to encounter water with acceptable dissolved solids (less than 500 ppm) in or above the aquifer indicated. Water used at the site is supplied by a well located on the property. Daily average water usage for site operations amounts to about 15,000 gallons per day. Withdrawal of this volume of water has no adverse effect on the water table as it represents a very small portion of the available supply, for example: a spring on the property, a few hundred feet from the well, naturally flows at the average rate of 350,000 gallons per day.

Floods which might occur at the site will produce different flood levels depending upon the flow rate of Joachim Creek. While the historical records (maximum observed level of 431 feet msl) as well as the analysis by U.S. Corps of Engineers (100-year flood level at 434.7 ft. msl) show that a site flood is not likely it still is considered remotely possible. If a flood of larger magnitude (greater than 435 feet msl) were to occur, water at the plant site would rise but there is not expected to be any significant water velocity associated with the flooding. The reason for the minimal water velocity is that the railroad track, which is located between Joachim Creek and the plant, would serve to isolate the plant area from the main stream flow. Water would enter and exit this isolated area via a culvert 900 feet south of the plant boundary and a second one about 1200 feet northeast of the plant, both of which pass under the railroad tracks. This postulated flood would be expected to result in only minimal water velocities (less than 0.1 ft/sec). These velocities are not expected to be able to tip material storage canisters within the buildings or transport any spilled material. Experimental results for a water-sand system show that for particles of UO2 size water velocities of greater than 0.6 ft/sec are required to move the material. Given the increased density of UO2 relative to sand (a factor of about 4), it does not seem likely that a credible flood would spill or transport spilled UO2 particles. The floor elevation of Buildings 253, 254 and 256 that were added in 1989 as well as other buildings where SNM is processed, except for area 240-3, are above the 100 year flood elevation.

### 9.4.5 Geology

The underlying earth structures are composed of younger rock than those of the southwestern portion of the county. The 240-260 million-year-old Mississippian system of the far northeastern portion of the county gradually changes to the 440-470 million-year-old Cambrian system of the southwestern portion of the county. This difference in age partly explains the difference in topography. The older Cambrian system has been exposed to erosion for 200 million years more than the younger Mississippian system, resulting in a more rolling topography. The younger rock structures of the northeastern section exhibit a more rugged topography.

The southwestern corner of the county near the Big River is primarily dolomite (magnesium limestone), with sandstone and chert (angular fragments of quartz) present in varying quantities depending on the location. This dolomite and chert grades northeast toward St. Louis into dolomite with sandstone. A massive sandstone ridge runs across the county from Pacific southeast to Festus and Crystal City. This fine quality stone is used for glass manufacturing and building purposes. Limestone exists in the Kimmswick formation in a narrow strip across the northern part of the county and extends south along the Mississippi River. Some deposits of marble are also present in the county.

Several test borings have been made in connection with past construction activities at the Hematite site. The borings were drilled to depths of approximately 35 feet. The soil profile thus obtained shows upper alluvial soils of stiff, very silty clays containing some sand, underlain by silty clays of firm to stiff consistency, to depths of 10 to 13.5 feet. Very stiff, highly plastic clay with limestone fragments were next encountered to depths of approximately 22 feet. Firm to stiff, sandy, silty clay was then found until auger refusal was obtained on boulders or limestone bedrock at an approximate depth of 36 feet.

### 9.4.6 Seismological

The east-central Missouri general area is relatively active seismically. The southeastern area of Missouri is quite active seismically and also contains a portion of the New Madrid Fault that caused the "great earthquakes" of 1811 and 1812. There were three quakes of Epicentral Intensity XII Modified Mercalli scale (M.M) which took place on December 6, 1811 and January 23 and February 7, 1812, near New Madrid. During recent years, there have been two quakes recorded in the New Madrid area. In 1962 a quake measuring V (M.M) was recorded and one with a magnitude of 4-1/2 was recorded in 1963. A quake reported as "the strongest in years" occurred near Caruthersville, Missouri, 150 miles southeast of Hematite, on December 3, 1980.

## 9.5 Locations of Buildings on Site

Figure 9-4 shows the location of and identifies the buildings and facilities on the Hematite site. A general description of the major buildings follows:

The Oxide Plant (Powder Production Area) is a four-level building, 31′ x 36′, with a concrete floor, corrugated plas-steel siding, and a metal roof. This building is an addendum to the original Building 255 and opens directly into the pelletizing facility. The Oxide Building is approximately 50′ in height.

Adjoining the Oxide Plant is a 31′ x 55′ dock area which also has a concrete floor and a metal roof, metal walls and overhead doors.

Building 240, the Recycle/Recovery Areas and Laboratory, is 83′ x 215′ and is 16′ high. The building has concrete flooring, exterior concrete block walls with windows, and a concrete-on-metal roof. About 6,000 square feet of the area is utilized for uranium recycle and recovery operations with the remainder of the building used for office area, clothing change and locker rooms, showers, maintenance shop, laboratory space, the site laundry, and utilities.

The Quality Control Laboratory is located in the southwest corner of Building 240. An area of approximately 2,500 square feet is utilized for testing of the chemical and physical properties of uranium oxide, powders, pellets and other materials.

In 1989, Buildings 250 and 251 were demolished and replaced by three Buildings; 253, 254 and 256. Figure 9-5 shows a perspective view of the resultant six adjoining buildings. Building 253 adjoins Buildings 254 and 256 on the east and Building 240 on the west. It measures approximately 77 feet wide by 133 feet long by 17 feet high and is constructed similarly to Building 240. It has a concrete slab floor, concrete block walls and metal supporting roofing.

Filtrate solidification is performed in the south end of Building 253. The center area contains the steam boiler, storage for chemicals and maintenance items and storage for SNM. The powder is stored in pails and in wheeled recycle hoppers. A facility with filtered ventilation is provided for transfer of powder from pails to the hoppers and blending in the hoppers. The bi-level north end of Building 253 contains offices and a change room.

Building 254 contains two parallel pellet production lines which are comprised of milling, blending, pressing, sintering, grinding, and packaging operations. Additional equipment has been installed to recycle green and hard scrap.

Building 254 is surrounded on four sides. It adjoins the building 255 on the east side, the warehouse building 256 on the south side, the womens' locker room on the north side, and the storage/utilities/office building 253 on the west side.

The building measures 83 feet wide by 161 feet long. The roof is 23 feet high on the south end and stepped to 36 feet high on the north end. Building 254 is constructed with a free standing steel frame supported on shallow poured concrete spread footings with connecting grade beams, a concrete slab floor on grade and concrete block walls laterally tied to the steel frame. The block walls are shared where building 254 adjoins other buildings. Metal curtain walls with insulation are used on the exposed exterior walls of the high north end and on the high portions that rise above the block walls shared with adjoining buildings. Roofing is rigid insulating board over metal decking supported on prefabricated trusses carried on the steel building frame. The building is designed using a BOCA seismic zone 2 earthquake occupancy importance factor of 1.5.

Building 255 measures 83' x 161' and is 17' high. This building has concrete flooring, concrete block walls, and a concrete-on-metal roof. Pellet production occupies a portion of this building--an area approximately 83' wide by 83' long. The remainder of this building area is used for offices, storage, work-break area, UO2 product storage in sealed containers, and the supply room.

The warehouse building 256 extends westward along the south end of pelletizing building 255 and the south end of pelletizing building 254. This warehouse measures approximately 151 feet by 50 wide. It is constructed of concrete slab floor on grade with concrete block walls and roofing of rigid insulating board over metal decking supported on prefabricated steel trusses. The warehouse serves as a storage and shipping facility for finished pellets and as a receiving warehouse for site supplies. Truck access to the warehouse is via a roadway corridor that runs from the main gate to the warehouse.

## 9.6 Maps and Plot Plans

The Hematite site comprises about 154 acres in Jefferson County, Missouri, on both sides of Highway P, northeast from the town of Hematite (Figure 9-3). Approximately 6 acres in the center of the site are enclosed by a security fence. In 1989 the fenced area was increased from 4 1/2 acres to provide roadway access to the Building 256 warehouse and to accommodate the related building construction. All production activities take place within six adjoining buildings inside the fenced area (Figure 9-4).

Figures 9-1 shows the location of Jefferson County within the State of Missouri. Figure 9-2 shows the area within a 5 mile radius of the site, illustrating the small towns and settlements.

## 9.7 License History

License SNM-33 was transferred to Combustion Engineering from Gulf United Nuclear on August 21, 1974. It was renewed on March 31, 1977 and December 30, 1983 for 5-year periods. In November, 1988 the license period was extended one year to December 31, 1989.

TABLE 9-1

PRINCIPAL OFFICERS OF COMBUSTION ENGINEERING, INC.

| Name | Position | Address | Citizenship |
|---|---|---|---|
| HUGEL, Charles E. | Chairman and Chief Executive Officer | 900 Long Ridge Road Stamford, CT | U.S. |
| KIMMEL, George S. | President and Chief Operating Officer | 900 Long Ridge Road Stamford, CT | U.S. |
| FORTNEY, Ray A. | Vice President Power Services Businesses | 900 Long Ridge Road Stamford, CT | U.S. |
| BARNETT, Charles E. | Vice President & General Counsel | 900 Long Ridge Road Stamford, CT | U.S. |
| RUBIN, Jeffrey S. | Vice President of Finance | 900 Long Ridge Road Stamford, CT | U.S. |
| SMITH, Dale E. | Vice President Human Resources & Operations | 900 Long Ridge Road Stamford, CT | U.S. |
| CONNOLLY, William J. | Vice President Corporate & Investor Relations | 900 Long Ridge Road Stamford, CT | U.S. |
| KISTNER, Robert E. | Vice President Information Systems & Services | 900 Long Ridge Road Stamford, CT | U.S. |
| MASON, Robert H. | Vice President Venture Finance & International | 900 Long Ridge Road Stamford, CT | U.S. |
| SANDERSON, Jack T. | Vice President-Corp. Technology | 900 Long Ridge Road Stamford, CT | U.S. |
| GARRY, Bernard J. | Secretary | 900 Long Ridge Road Stamford, CT | U.S. |
| JONES, Fred R. | Treasurer | 900 Long Ridge Road Stamford, CT | U.S. |

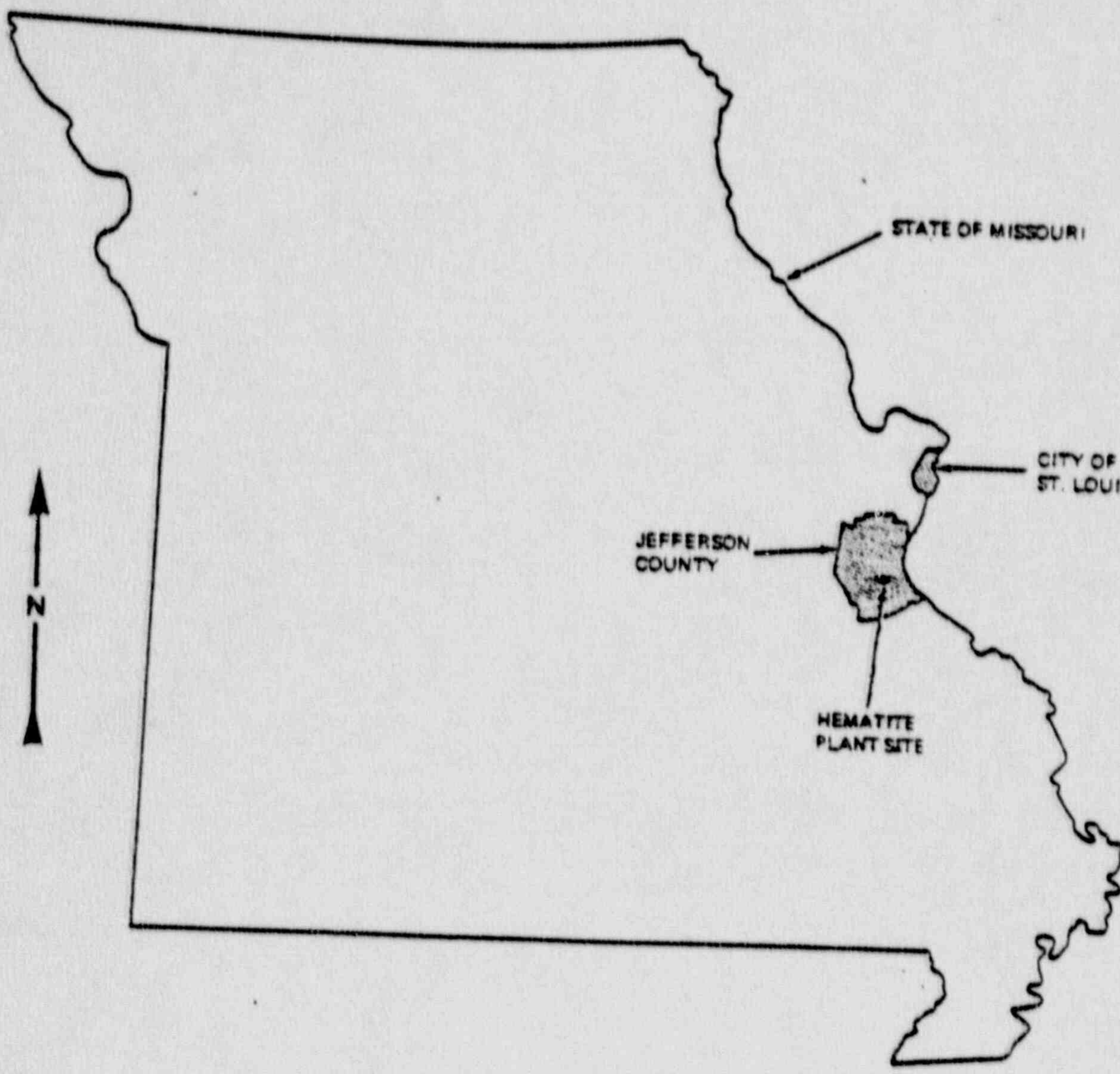


FIGURE 9-1 Hematite Plant Site Location Within the State of Missouri

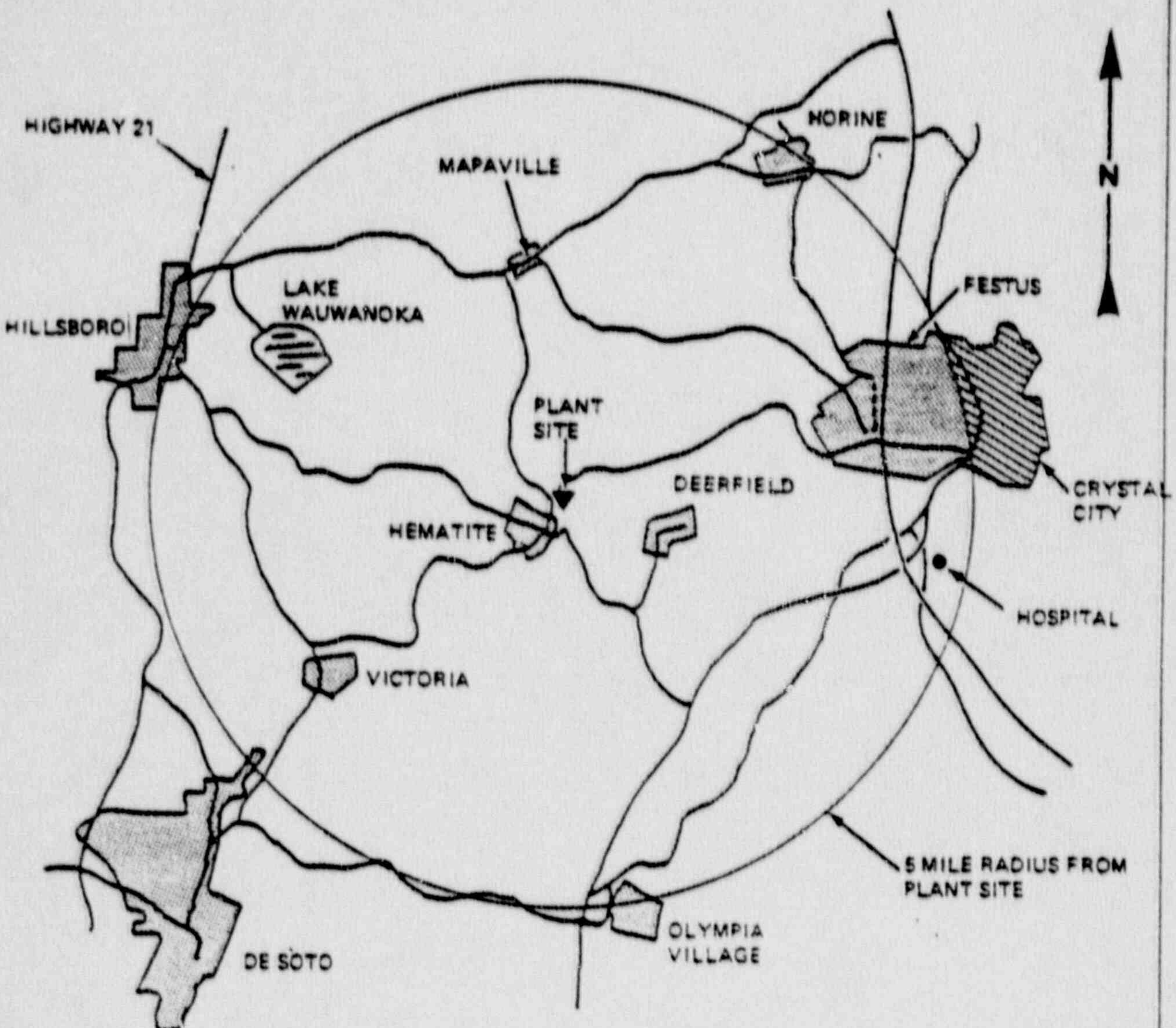


FIGURE 9-2 Map of Area Within 5 Mile Radius of Hematite Site

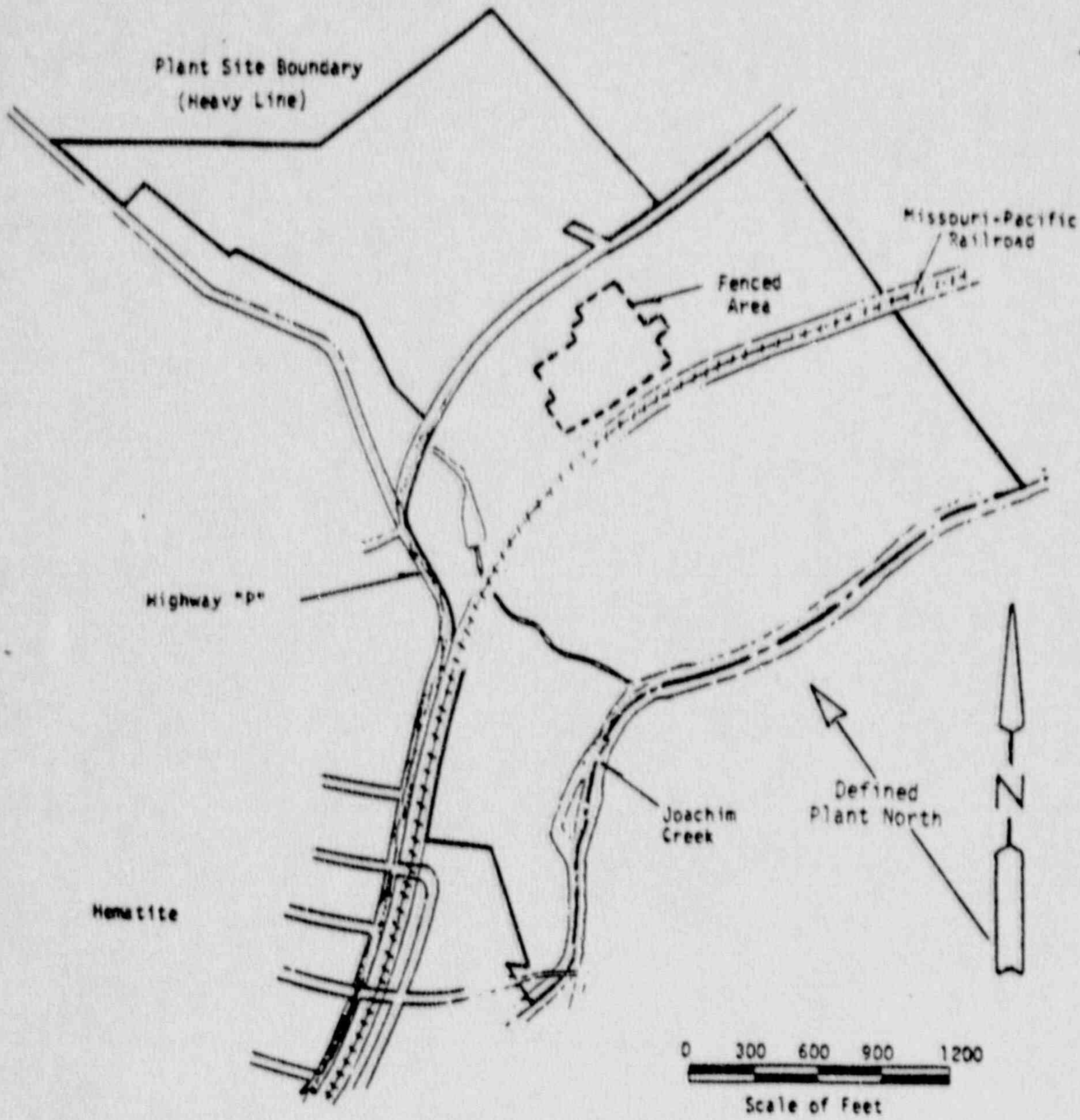


FIGURE 9-3 Site Boundaries and Location with Respect to the Town of Hematite

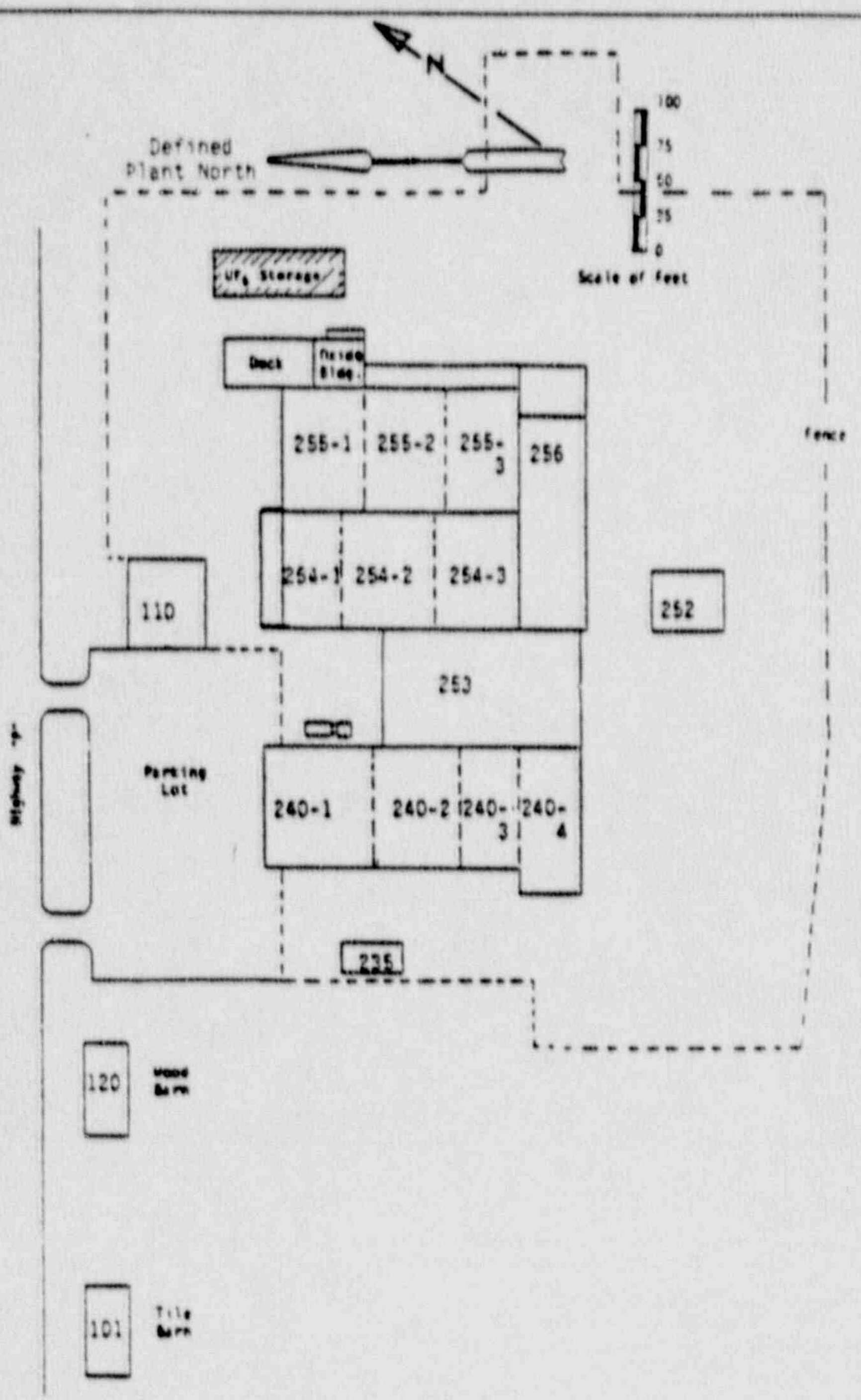


FIGURE 9-4 Location and Identification of Buildings and Facilities on Hematite Plant Site

PART II
SAFETY DEMONSTRATION

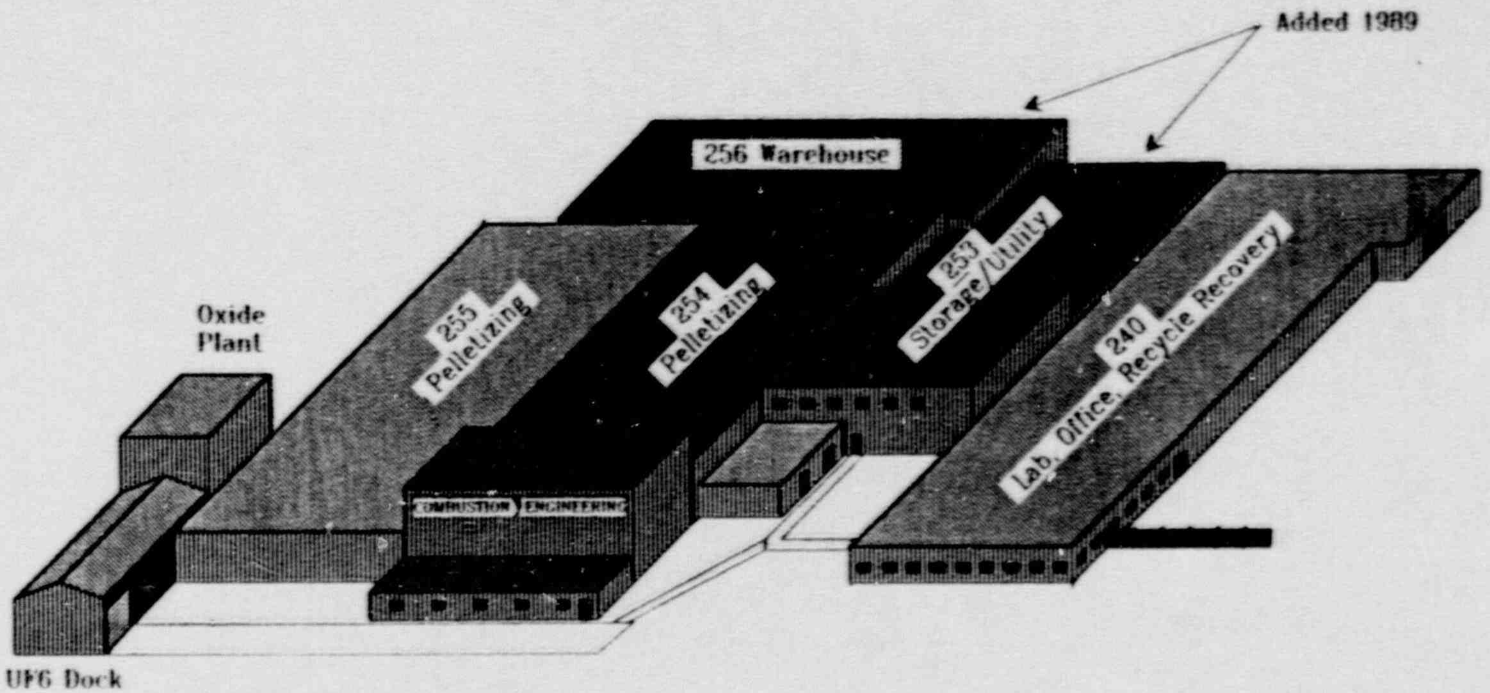


FIGURE 9-5 Hematite Facility Buildings, Perspective View

# APPENDIX 9-1

## DEMONSTRATION OF FINANCIAL CAPABILITY

Form 10-K, Annual Report to the Securities and Exchange Commission for 1988 by Combustion Engineering, Inc.

SECURITIES AND EXCHANGE COMMISSION
Washington, D.C. 20549
FORM 10-K

ANNUAL REPORT PURSUANT TO SECTION 13 OR 15(d)
OF THE SECURITIES EXCHANGE ACT OF 1934

For the fiscal year ended: December 31, 1988 Commission file number 1-117-2

COMBUSTION ENGINEERING, INC.
(Exact Name of Registrant As Specified In Its Charter)

| Delaware | 13-1587569 | (203) 329-8771 |
|---|---|---|
| (State or Other Jurisdiction of Incorporation or Organization) | (I.R.S. Employer Identification No.) | (Registrant's telephone number) |

900 Long Ridge Road, P.O. Box 9308, Stamford, Connecticut 06904
(Address of Principal Executive Offices) (Zip Code)

Securities registered pursuant to Section 12(b) of the Act:

| Title of Each Class | Name of Each Exchange on Which Registered |
|---|---|
| Common Stock-$1 Par Value | New York Stock Exchange |
| 7.45% Sinking Fund Debentures Due 1996 | New York Stock Exchange |

Securities registered pursuant to Section 12(g) of the Act: None

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Indicate by check mark whether the registrant (1) has filed all reports required to be filed by Section 13 or 15(d) of the Securities Exchange Act of 1934 during the preceding 12 months (or for such shorter period that the registrant was required to file such reports), and (2) has been subject to such filing requirements for the past 90 days.

Yes X No ____

---

Indicate the number of shares outstanding of each of the registrant's classes of common stock, as of the latest practicable date.

| Class | Outstanding at March 1, 1989 |
|---|---|
| Common Stock-$1 par value | 38,795,449 |

The aggregate market value of the voting stock held by non-affiliates of the registrant on March 1, 1989, was approximately $1,070,524,318.

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Documents Incorporated By Reference:

| Document | Form 10-K Part |
|---|---|
| The Registrant's 1988 Annual Report to Shareholders | Part I and II |
| The Registrant's Proxy Statement in connection with its Annual Meeting of Shareholders to be held on April 27, 1989 | Part III |

Exhibit index located at sequential page number ________.

as part of its emphasis on expanding its products and services in the power industry. ESG specializes in energy management and supervisory control systems which are used for data acquisition, analysis and control in optimizing electrical power grids, including distribution and transmission networks.

The Company has also divested itself of a number of businesses outside of its core markets. In February 1989, the Company signed an agreement for the sale of its Vetco Services oil and gas operations and on March 23, 1989 completed the transaction, realizing net proceeds in excess of $30,000,000. In December 1988, the Company signed an agreement for the disposition of its C-E Natco, C-E Invalco, Premier Refractories and Chemicals (formerly C-E Refractories) and C-E Tyler businesses (the transaction is scheduled to be completed during 1989). In September 1988, the Company completed the sale of its investment in Jamesbury Corp., a manufacturer and supplier of high-performance valves, realizing net proceeds of approximately $136,000,000 resulting in a pretax gain of approximately $36,800,000.

For more detail on business acquisitions and dispositions in 1988 as well as 1987 and 1986, refer to Notes 14 and 17 of the Notes to Consolidated Financial Statements on pages 42 and 44 of the Annual Report.

In connection with the Company's restructuring and repositioning programs, as well as its development and introduction of new technologies and broadened scope of participation in new markets, a provision of $272,800,000 was included in operating loss in the third and fourth quarters of 1988. This provision was made in recognition of: (a) increases in estimated costs to complete waste-to-energy plants and fluid bed power plants both involving new, first-time technologies; (b) cost overruns on other fossil fuel systems contracts administered in the United States and in Canada; (c) the estimated costs to repair the boilers in the Hartford, Connecticut waste-to-energy plant which experienced tube failures due to corrosion, and the cost of related modifications required on two other plants under construction; and (d) restructuring charges for the Power Generation segment.

(b) Financial Information about Industry Segments

For financial reporting by business segment, reference is made to the Annual Report's "Management's Discussion and Analysis" on pages 25 through 29 and to "Business Segment Information" on page 30.

Much of the Company's business, especially that relating to power generation and large process industry applications, involves long-term contracts of various types, including fixed-price and cost-plus-fee type contracts, with some contracts including variations of both types. The largest portion of sales under long-term contracts is derived from fixed-price contracts. Most contracts provide for progress or scheduled payments over the life of the contracts. The contract price, in fixed-price contracts, either includes an amount for the estimated increase in the cost of labor, materials and services over the period required for performance of the contract or is subject to adjustment based on a price escalation clause.

Profits on long-term contracts for financial reporting purposes are recognized on the percentage-of-completion method. Percentage-of-completion is measured principally by the percentage of costs incurred and accrued to date versus estimated total costs for each contract. No profits are recorded on contracts for equipment manufactured in the Company's plants prior to billing the customer and, in most cases, prior to shipment of the equipment. Contracts typically extend over a period of several months to three or more years. Changes in contract performance and estimated profitability, including those arising from contract penalty provisions and final contract settlements, may result in revisions to costs and income and are recognized in the period in which the revisions are determined. Profit incentives are included in income when their realization is reasonably assured.

Cost estimates for long-term contracts take into account all anticipated costs including, among others: engineering; manufacturing; subcontracting and field construction costs including warranties based upon past experience, which are required to meet the specifications of the contracts.

(c) Narrative Description of Business

Reference is made to "Business Segments and Brief Description of the Business" shown on pages 5 and 25-30 of the "Financial Section" of the Annual Report regarding a narrative description of the Company's business.

**Raw Materials**

The primary raw material used by the Company's business segments is steel; principally sheet, plate, bar structurals, tubing, rod, forgings, castings and wire. However, many other materials are also required. Raw materials are purchased by the Company as needed for individual contracts or to maintain proper inventory levels. The Company normally does not encounter difficulties in procuring adequate supplies of raw materials.

**Patents and Licenses**

The Company has numerous United States and foreign patents and patent applications which relate to many different products and processes, and are deemed by the Company to be adequate for the conduct of its businesses. The Company does not believe that any single patent is of material importance in relation to any business segment or the Company as a whole.

The maintenance of licenses issued by the Nuclear Regulatory Commission is essential to the conduct of certain portions of the Company's nuclear business.

**Backlog**

| December 31, | 1988 | 1987 | 1986 |
|---|---|---|---|
| | | (Dollars in thousands) | |
| Power Generation | $1,951,177 | $2,161,050 | $1,537,415 |
| Process Industries | 976,080 | 623,998 | 452,395 |
| Public Sector and Environmental | 777,245 | 835,469 | 776,702 |

Overall backlog for the Company increased during 1988 due to orders taken in the Process Industries segment which offset the decrease in backlog for the other two segments. The Process Industries gain came primarily from a two-fold increase in its process engineering and construction business which won four major pulp and paper awards in 1988 with a total contract value of over $130,000,000. Process Industries made a strong showing as well in the petrochemical market, receiving a $160,000,000 order for two new plants in the People's Republic of China.

The Power Generation and Public Sector and Environmental segments' backlogs, however, decreased from their record levels in 1987 as work was carried out in 1988 on 1987 orders (including the nuclear plants in Korea, circulating fluid bed plants in the United States, other fossil plants in the United States, Canada and the People's Republic of China, and waste-to-energy plants in the United States). The Power Generation segment made significant contributions to its backlog with an award for a 130 megawatt circulating fluid bed unit in the United States and for two 300 megawatt lignite-fired steam generators in northeastern Thailand, valued at $82,000,000.

It should be noted that the Process Industries business segment information has been restated to include the Engineering and Construction operations. Also, the Public Sector and Environmental business segment information includes contract backlog relating to government contract services totaling approximately $178,000,000 and $158,000,000 in 1988 and 1987, respectively, which extend over periods approximating five years, many of which are subject to annual government funding authorization.

For additional information on the 1988 backlog, refer to "Business Segment Information" - Backlog on page 30 and to "Management's Discussion and Analysis" on pages 25 through 27 of the "Financial Section" of the Annual Report. Approximately 63% of the consolidated December 31, 1988 backlog of unfilled orders is expected to be recorded as sales (principally on the percentage-of-completion method) in 1989 and the remainder in subsequent years.

The backlog of unfilled orders cannot be projected into an annual rate of net sales for a variety of reasons, including the length of time required for the completion of contracts and changes in customer requirements.

**Competitive Conditions**

With respect to products, services and equipment for the power generation industry, the Company is one of the largest domestic manufacturers of fossil-fueled steam generating systems. The major competitors for large scale fossil-fueled steam generating systems in the United States include The Babcock & Wilcox Company, a wholly-owned subsidiary of McDermott International, Inc., and Foster Wheeler Corporation. The Company competes with a number of suppliers of smaller-scale fossil steam generating systems. The Company is one of many competitors in the world-wide market for nuclear steam supply systems. The Company also competes with numerous suppliers in servicing operating fossil and nuclear steam generating systems and related equipment.

Regarding products, services and equipment for process industries, the Company is one of numerous manufacturers or suppliers and, in certain cases, is one of the leading manufacturers or suppliers. In general, the Company conducts this portion of its operations under highly competitive conditions worldwide. Lummus Crest Inc., the principal component of Process Industries segment's engineering and construction services business, is engaged in providing design, engineering and construction services for chemical process plants, petroleum refineries and other industrial facilities. However, the business is highly competitive and the Company now competes with various engineering firms of all sizes both domestic and foreign.

With respect to public sector and environmental, the Company provides environmental systems and services to the public sector and process and power industries. The Company is one of many firms engaged in this highly competitive market, which includes Ogden Corporation and Wheelabrator Technologies, Inc.

Usually, the Company competes for new orders by responding to specific invitations to bid. The principal bases of competition would include the following factors, but not necessarily in their order of importance: design of the equipment or process to be furnished in response to the customer's specifications, technical support and services, ability to meet the customer's delivery schedule, price and, in certain cases, project financing. Project financing has become of increasing importance in bidding and selection for power project contracts. Over the last two years, in order to compete in the independent power generation market through the supply and construction of small power generation plants, the Company has entered into lease agreements, standby equity and debt commitments and has invested in project companies (usually 20% or less of the project value).

**Research and Development**

The amounts spent during 1988, 1987 and 1986 on research and development activities were $78,733,000, $69,995,000 and $68,825,000, respectively, including estimated amounts spent on customer-sponsored R&D of $22,790,000, $19,502,000 and $20,800,000, respectively.

**Compliance with Environmental Protection Laws**

Compliance by the Company with Federal, state and local environmental protection laws required capital expenditures of $425,000 in 1988, $443,000 in 1987 and $238,000 in 1986. It is estimated that capital expenditures in 1989 for such purposes will be approximately $1,237,000.

**Employees**

At December 31, 1988, the Company employed 28,832 persons.

(d) <u>Financial Information about Foreign and Domestic Operations and Export Sales</u>

Significant financial data by geographic area can be found in Note 16 of the Notes to Consolidated Financial Statements shown on page 43 of the "Financial Section" of the Annual Report.

## ITEM 2. PROPERTIES

The principal manufacturing and processing plants and other important physical properties are set forth below. The industry segment(s) which use the property is also identified. Unless noted, the property is owned by the Company or a subsidiary.

Andersonville, Georgia (2)*#
Bloomfield, New Jersey (2)*
Brantford, Ontario (2)
Chattanooga, Tennessee (1)(2)
Columbus, Ohio (2)
Concordia, Kansas (2)
Dry Branch, Georgia (2)#
Dundalk, Ireland (2)
Enterprise, Kansas (2)
Erie, Pennsylvania (2)*
Gastonia, North Carolina (2)
Helsinki, Finland (2)*
Houston, Texas (2)*
Lewisburg, West Virginia (2)
Lincolnshire, Illinois (2)*
Lisle, Illinois (2)*
Melville, New York (1)*
Muncy, Pennsylvania (2)
Norcross, Georgia (1)*
Nuevo Laredo, Mexico (2)*
Ottawa, Ontario (1)(2)*
Portland, Maine (3)*
Rochester, New York (2)
Sandersville, Georgia (2)#
Sherbrooke, Quebec (1)(2)
Stamford, Connecticut (Corporate Office)
Stevenage, England (2)
St. Catherines, Ontario (2)
The Hague, Netherlands (2)*
Toronto, Canada (2)*
Valley Forge/King of Prussia, Pennsylvania (2)
Walnut Creek, California (1)*
Wellsville, New York (1)(2)
Windsor, Connecticut (1)(2)(3)*

(1) Power generation
(2) Process industries
(3) Public sector and environmental

* Includes leased facilities
\# Includes mining properties some of which are under lease

The Company's manufacturing facilities are of varying ages and are well maintained, in good operating condition and suitable for the purposes for which they are being used. All of the principal manufacturing and processing plants are utilized on the basis of at least one shift and some operate with more than one shift. Management regards these facilities as having adequate capacity to meet current production requirements.

ITEM 3. PENDING LEGAL PROCEEDINGS

Reference is made to Note 12 of the Notes to Consolidated Financial Statements shown on page 40 of the "Financial Section" of the Annual Report.

In addition, a review of environmental proceedings to which the Company and/or its subsidiaries are parties has indicated certain instances in which the Company or a subsidiary may be requested to contribute cleanup costs in excess of the reporting threshold of one hundred thousand dollars. The Company would not expect such instances to involve aggregate payments in excess of one million dollars.

ITEM 4. SUBMISSION OF MATTERS TO A VOTE OF SECURITY HOLDERS

None.

OFFICERS OF THE COMPANY

Listed below are the officers of the Company:

| Name | Age | Position Presently Held |
|---|---|---|
| Charles E. Hugel | 60 | Chairman and Chief Executive Officer |
| George S. Kimmel | 54 | President and Chief Operating Officer |
| Charles E. Barnett | 49 | Vice President and General Counsel |
| William J. Connolly | 59 | Vice President-Corporate and Investor Relations |
| Robert E. Kistner | 52 | Vice President-Information Systems and Services |
| Robert H. Masson | 53 | Vice President-Venture Finance and International |
| Jeffrey S. Rubin | 45 | Vice President-Finance |
| Jack T. Sanderson | 52 | Vice President-Corporate Technology |
| Dale E. Smith | 45 | Vice President-Human Resources and Operations Support |
| Bernard J. Garry | 63 | Secretary |
| Preston E. Insley | 54 | Controller |
| Fred R. Jones | 41 | Treasurer |

There are no family relationships among the foregoing officers.

There are no arrangements or any understandings between the above persons and any other persons pursuant to which such persons were elected to the offices indicated.

Election to the offices indicated is for a term of one year.

A brief account of each officer's business experience during the past five years is set forth below:

Mr. Hugel was elected Chairman, effective July 1, 1988. He was elected Chief Executive Officer in April 1984 and President and Director of the Company effective September 1, 1982.

Mr. Kimmel was elected President and Chief Operating Officer, effective July 1, 1988. In 1987, he was elected Executive Vice President-Operations and a member of the Office of the President. Prior thereto, he was an Executive Vice President. He was elected a Director in April 1981.

Mr. Barnett was elected Vice President and General Counsel of the Company in January 1984. Prior to joining the Company, he was Vice President, General Counsel and Secretary of St. Joe Minerals Corporation.

Mr. Connolly was elected Vice President-Corporate and Investor Relations in 1980.

Mr. Kistner was elected a Vice President of the Company in April 1988. Prior to joining the Company in November 1983, he served in a number of systems planning and management positions with Navistar (formerly International Harvester Corporation).

Mr. Masson served as Vice President-Treasurer of the Company from November 1980 to October 1986 and as Vice President-Financing and Venture Development until October 1987 when he became Vice President-Venture Finance and International.

Mr. Rubin was elected Vice President-Finance in 1987. Prior thereto, he was Vice President-Planning and Control (1986), and Vice President and Controller (1984). Prior to joining the Company in 1984, he was associated with Atlantic Richfield Company, most recently as Vice President Planning & Control, ARCO Metals Company.

Dr. Sanderson was elected a Vice President of the Company in April 1987. Prior to joining the Company in April 1986, he was Director of Technology Applications at ITT Corporation, New York, NY.

Mr. Smith was elected a Vice President of the Company in November 1985. Prior to 1985, he was Vice President of Operational Management of Vetco Offshore, Inc., a subsidiary of the Company.

Mr. Garry was elected Secretary of the Company in April 1986. Prior to April 1986, he served as Assistant Secretary of the Company.

Mr. Insley was elected Controller in November 1986. Prior thereto, he was President of C-E Refractories, a division of the Company and, prior to that, Vice President-Finance and Controller of the Industrial Group.

Mr. Jones was elected Treasurer in November 1986. Prior to his election, he was Assistant Treasurer and, before that, Director-Treasury Financial Services of the Company. Prior to joining the Company in April 1984, he was Assistant Treasurer of the Penn Central Corporation.

## PART II

### ITEMS 5. THROUGH 8.

The "Financial Section" of the Annual Report to Shareholders for the year ended December 31, 1988, is hereby incorporated by reference.

| | Page Number in "Financial Section" of Annual Report |
|---|---|
| Item 5 - Market for the Registrant's Common Stock and Related Security Holder Matters | 45 to 47 & 49 |
| Item 6 - Selected Financial Data (Summary of Operations/Financial Position and Other Data) | 46 to 47 |
| Item 7 - Management Discussion and Analysis | 25 to 29 |
| Item 8 - Financial Statements and Supplementary Data | 30 to 45 |

### ITEM 9. DISAGREEMENTS ON ACCOUNTING AND FINANCIAL DISCLOSURE

None.

## PART III

### ITEM 10. DIRECTORS AND EXECUTIVE OFFICERS OF THE COMPANY

Pursuant to General Instruction G(3), the information regarding directors called for by this item is hereby incorporated by reference from the Company's 1989 Proxy Statement filed with the Securities and Exchange Commission. The information regarding executive officers called for by this item is included at the end of PART I, ITEM 4, of this document under the heading Officers of the Company.

### ITEM 11. EXECUTIVE COMPENSATION

Pursuant to General Instruction G(3), the information called for by this item is hereby incorporated by reference to the information on page 5 of the Company's 1989 Proxy Statement filed with the Securities and Exchange Commission.

### ITEM 12. SECURITY OWNERSHIP OF CERTAIN BENEFICIAL OWNERS AND MANAGEMENT

Pursuant to General Instruction G(3), the information called for by this item is hereby incorporated by reference to the information commencing on page 2 of the Company's 1989 Proxy Statement filed with the Securities and Exchange Commission.

### ITEM 13. CERTAIN RELATIONSHIPS AND RELATED TRANSACTIONS

Pursuant to General Instruction G(3), the information called for by this item is hereby incorporated by reference to the information on pages 14 through 16 of the Company's 1989 Proxy Statement filed with the Securities and Exchange Commission.

## PART IV

### ITEM 14(a). EXHIBITS AND FINANCIAL STATEMENT SCHEDULES

Documents:



NOTES:

(a) Financial statements for unconsolidated subsidiaries and 50%-owned companies have been omitted as not being required since considered in the aggregate as a single subsidiary, they would not constitute a significant subsidiary.

(b) Schedules not filed herein are omitted because of the absence of conditions necessitating their filing.

### ITEM 14(b). REPORTS ON FORM 8-K

Form 8-K dated December 30, 1988

Item 5. Other Events
Adoption of Shareholder Rights Plan.

## SIGNATURES

Pursuant to the requirements of Section 13 or 15(d) of the Securities Exchange Act of 1934, the registrant has duly caused this report to be signed on its behalf by the undersigned, thereunto duly authorized.

COMBUSTION ENGINEERING, INC.

By C. E. Hugel
Charles E. Hugel
Chairman and Chief Executive Officer, Director

By
Jeffrey S. Rubin
Vice President - Finance

By P E Insley
Preston E. Insley
Controller

Dated March 29, 1989

Pursuant to the requirements of the Securities Exchange Act of 1934, this report has been signed below by the following persons on behalf of the Registrant and in the capacities and on the date indicated.

| | |
|---|---|
| Lucy Wilson Benson *<br>By ______________________<br>Director | Paul W. MacAvoy *<br>By ______________________<br>Director |
| Walter H. Helmerich, III *<br>By ______________________<br>Director | Scott L. Probasco, Jr. *<br>By ______________________<br>Director |
| Robert M. Jenney *<br>By ______________________<br>Director | Robert C. Seamans, Jr. *<br>By ______________________<br>Director |
| By Arthur J. Santry Jr<br>Arthur J. Santry, Jr.<br>Director | Robert G. Stone, Jr. *<br>By ______________________<br>Director |
| George S. Kimmel*<br>By ______________________<br>President and Chief Operating Officer, Director | Kenneth J. Whalen *<br>By ______________________<br>Director |
| | David R. Whitwam *<br>By ______________________<br>Director |

*Pursuant to Power of Attorney

Dated: March 29, 1989

By Bernard J. Garry<br>
Bernard J. Garry,<br>
as Attorney-In-Fact

Exhibits
(3) Articles of incorporation and by-laws

| | | Sequential Page No. |
|---|---|---|
| (3)-1 | Restated Certificate of Incorporation of Combustion Engineering, Inc. | Incorporated by reference to form 10-Q Report for the second quarter 1987 and form 8-K filed December 30,1988 |
| (3)-2 | By-Laws of Combustion Engineering, Inc., as amended June 27, 1988 | Incorporated by reference to Form 10-Q Report for second quarter 1988 |

(10) Material contracts

| | | |
|---|---|---|
| (10)-1 | Acquisition agreement among Hughes Tool Company, Combustion Engineering, Inc., Combustion Engineering Limited, C-E Europe Limited and Vetco Gray Inc., together with certain exhibits and amendments | Incorporated by reference to Form 8-K Report dated November 14, 1986 |
| (10)-2 | Agreements incident to the acquisition of AccuRay Corporation | Incorporated by reference to Form 8-K Report dated January 29, 1987 |
| (10)-3 | Amended Incentive Compensation Plan, as amended July 24, 1986 | Incorporated by reference to Form 10-Q Report for the third quarter of 1986 |
| (10)-4 | Deferred Compensation Plan for Non-Employee Directors of Combustion Engineering, Inc., as amended November 19, 1986 | Incorporated by reference to Form 10-K Report for 1986 |
| | Amendment adopted December 19, 1988 | |
| (10)-5 | Deferred Compensation Plan, as amended November 19, 1986 | Incorporated by reference to Form 10-K Report for 1986 |
| | Amendment adopted December 19, 1988 | |
| (10)-6 | Executive Retirement and Life Insurance Plan, as amended July 24, 1986, and Form of Life Insurance Agreement | Incorporated by reference to Form 10-K Report for 1985 and to Form 10-Q Report for the third quarter of 1986 |
| | Amendment adopted December 19, 1988 | |

| | | |
|---|---|---|
| (10)-7 | Key Employee Retention and Severance Benefit Plan and Forms of Agreements | Incorporated by reference to Form 10-K Report for 1982 |
| | Amendment and form of Agreement adopted December 19, 1988 | |
| (10)-8 | 1982 Stock Option Plan | Incorporated by reference to Proxy Statement for Annual Meeting on April 22, 1986 |
| | Amendment adopted December 19, 1988 | |
| (10)-9 | Restricted Stock Plan for Senior Executives | Incorporated by reference to Proxy Statement for Annual Meeting on April 28, 1988 |
| | Amendment adopted November 16, 1988 and Form of Agreement | |
| (10)-10 | Restricted Stock Plan for Non-Employee Director of Combustion Engineering, Inc. | Incorporated by reference to Proxy Statement for Annual Meeting on April 28, 1988 |
| | Amendments adopted November 16 and December 19, 1988 | |
| (10)-11 | Senior Mangement Incentive Plan and Amendment adopted December 19, 1988 | |
| (10)-12 | Supplemental Benefit Plan for Salaried Employees | Incorporated by reference to Form 10-K Report for 1982 |
| | Amendments adopted September 29 and December 19, 1988 | |
| (10)-13 | Consulting Agreement with James B. Kelly dated October 20, 1988 | |
| (10)-14 | Supplemental Retirement Benefit Agreement with Charles E. Hugel | Incorporated by reference to Form 10-K Report for 1982 |
| | Amendment adopted December 19, 1988 | |
| (10)-15 | Agreement with Arthur J. Santry, Jr. | Incorporated by reference to Form 10-K Report for 1984 |
| | Amendment adopted December 19, 1988 | |

| | | |
|---|---|---|
| (10)-16 | Consulting Agreement with Robert C. Seamans, Jr., as amended April 22, 1986 | Incorporated by reference to Form 10-K Report for 1980 and 1981 and to Form 10-Q Report for the second quarter of 1986 |
| (10)-17 | Benefit Agreement with Dudley C. Mecum | Incorporated by reference to Form 10-Q Report for the third quarter of 1986 |

<u>(13) Annual Report to Shareholders for the Year Ended December 31, 1988</u>

<u>(22) Subsidiaries of the Registrant</u>

<u>(24) Consent of Experts</u>

<u>(25) Powers of Attorney</u>

| | | |
|---|---|---|
| (25)-1 | Power of Attorney for Lucy W. Benson and David R. Whitwam | Incorporated by reference to Form 10-K Report for 1987 |
| (25)-2 | All Other Powers of Attorney | Incorporated by reference to Form 10-K Report for 1985 and 1986 |

ARTHUR ANDERSEN & CO.

STAMFORD, CONNECTICUT

## REPORT OF INDEPENDENT PUBLIC ACCOUNTANTS ON SCHEDULES

To the Board of Directors and
Shareholders of Combustion Engineering, Inc.:

We have audited in accordance with generally accepted auditing standards, the financial statements included in the Combustion Engineering, Inc. Annual Report to Shareholders incorporated by reference in this Form 10-K, and have issued our report thereon dated February 17, 1989. Our audits were made for the purpose of forming an opinion on those statements taken as a whole. The schedules listed in Item 14(a)2 are presented for the purpose of complying with the Securities and Exchange Commission's rules and are not part of the basic financial statements. These schedules have been subjected to the auditing procedures applied in the audits of the basic financial statements and, in our opinion, fairly state in all material respects the financial data required to be set forth therein in relation to the basic financial statements taken as a whole.

Arthur Andersen & Co.

ARTHUR ANDERSEN & CO.

Stamford, Connecticut,
February 17, 1989

SCHEDULE I

## COMBUSTION ENGINEERING, INC. AND SUBSIDIARY COMPANIES
## MARKETABLE SECURITIES - OTHER INVESTMENTS

YEAR ENDED DECEMBER 31, 1988
(Dollars in thousands)

| Column A | Column B | Column C | Column D | Column E |
|---|---|---|---|---|
| Name of Issuer and Title of Each Issue | Number of Shares or Units-Principal Amount of Bonds and Notes | Cost of Each Issue | Market Value of Each Issue at Balance Sheet Date | Amount at Which Each Portfolio of Equity Security Issues and Each Other Security Issue are Carried in the Balance Sheet |
| Vetco Gray Inc. | | | | |
| Common Stock | 199 | $ 15,305 | (A) | $ 15,305 |
| Preferred Stock | 2,975,878 | 148,794 | (A) | 148,794 |
| | | | | $164,099 |
| Other Investments | | | | |
| At Cost | | | | $ 26,099 |
| At Equity | | | | 48,450 |
| | | | | $ 74,549 |

(A) The common and preferred stock of Vetco Gray, Inc. are not traded securities and therefore do not have a market value as defined under this schedule.

SCHEDULE II

COMBUSTION ENGINEERING, INC. AND SUBSIDIARY COMPANIES
AMOUNTS RECEIVABLE FROM RELATED PARTIES AND UNDERWRITERS, PROMOTERS, AND EMPLOYEES
OTHER THAN RELATED PARTIES
YEAR ENDED DECEMBER 31, 1988
(Dollars in thousands)

| Column A | Column B | Column C | Column D | Column E | |
|---|---|---|---|---|---|
| | | | | Balance at End of Year | |
| Name of Debtor | Balance at Beginning of Year | Additions | Deductions | Current | Non-Current |
| Cooperative Receivables Corporation (1) | $6,950 | $ - | $6,950 | $ - | $ - |

(1) The Company sold trade receivables to the Cooperative Receivables Corporation (an equity investee) on December 30, 1987. The above amount represented the holdback which was due to the Company after the receivables sold were collected.

Schedule VII

## COMBUSTION ENGINEERING, INC. AND SUBSIDIARY COMPANIES
## GUARANTEES OF SECURITIES OF OTHER ISSUERS

**DECEMBER 31, 1988**

(Dollars in thousands)

| Column A | Column B | Column C | Column D | Column E | Column F | Column G |
|---|---|---|---|---|---|---|
| Name of Issuer of Securities Guaranteed by Person for Which Statement is Filed | Title of Issue of Each Class of Securities Guaranteed | Total Amount Guaranteed and Outstanding | Amount Owned by Person or Persons for Which Statement is Filed | Amount in Treasury of Issuer of Securities Guaranteed | Nature of Guarantee | Nature of Any Default by Issuer of Securities Guaranteed in Principal, Interest, Sinking Fund or Redemption Provisions, or Payment of Dividends |
| Combustion Engineering, Inc./ Vetco Gray Inc. | Notes | $52,000 | $ - | $ - | (1) | None |
| Combustion Engineering, Inc./ Kruger, Inc. | Note | 18,700 | - | - | (1) | None |
| Other (2) | Notes | 13,893 | - | - | (1) | None |
| | | $84,593 | $ - | $ - | | |

NOTE:
(1) Guarantee of principal and interest.
(2) Represents guarantees that individually are not significant.

SCHEDULE IX

COMBUSTION ENGINEERING, INC. AND SUBSIDIARY COMPANIES
SHORT-TERM BORROWINGS
(Dollars in thousands)

| | Balance | Weighted Average Interest Rate | Maximum Amount Outstanding During the Period | Average Amount Outstanding During the Period | Weighted Average Interest Rate During the Period |
|---|---|---|---|---|---|
| 1988 | | | | | |
| Notes Payable to Bank | $73,220 | 10.0% | $300,354 | $182,075 | 11.1% |
| 1987 | | | | | |
| Notes Payable to Bank | $27,012 | 11.0% | $302,584 (1) | $139,903 | 8.2% |
| 1986 | | | | | |
| Notes Payable to Bank | $22,283 | 9.9% | $ 22,283 | $ 19,433 | 9.9% |

(1) Includes drawdowns utilized to purchase AccuRay.

## COMBUSTION ENGINEERING, INC. AND SUBSIDIARY COMPANIES

## SUBSIDIARIES OF THE REGISTRANT

There are no parents of the Registrant. Information with respect to subsidiaries of the Registrant with respect to continuing operations, as of December 31, 1988, follows:

Active domestic and wholly-owned foreign subsidiaries included in the Consolidated Financial Statements -

| | Incorporated Under Laws of |
|---|---|
| Basic Incorporated | Delaware |
| C-E Environmental, Inc. | Delaware |
| E.C. Jordan Co. | Maine |
| C-E Operations & Maintenance Services, Inc. | Delaware |
| Avco Services Corporation | Delaware |
| Bell Technical Operations Corporation | Delaware |
| Combustion Engineering Canada Inc. | Canada |
| Combustion Engineering Europe Limited | England |
| Combustion Engineering Limited | England |
| Combustion Engineering Simcon, Inc. | Delaware |
| Georgia Kaolin Company, Inc. | New Jersey |
| Impell Corporation | Delaware |
| Lummus Crest Inc. | Delaware |
| Mullite Company of America | Georgia |
| Process Automation Business, Inc. | Delaware |
| Sprout-Bauer, Inc. | Ohio |
| Sprout, Waldron & Co. GmbH | Austria |
| The Air Preheater Company, Inc. | Delaware |
| Vetco International AG | Switzerland |
| W.S. Tyler, Incorporated | Ohio |

In addition to the foregoing, the Registrant has 134 domestic subsidiaries and 110 foreign subsidiaries which, considered in the aggregate as a single subsidiary, would not constitute a significant subsidiary.

Other Active Subsidiaries Not Included in Consolidated Financial Statements - At December 31, 1988, the Registrant has investments in 37 companies accounted for on the equity method which, considered in the aggregate as a single subsidiary, would not constitute a significant subsidiary.

ARTHUR ANDERSEN & CO.
STAMFORD, CONNECTICUT

EXHIBIT 24

CONSENT OF INDEPENDENT PUBLIC ACCOUNTANTS

As independent public accountants, we hereby consent to the incorporation of our reports included or incorporated by reference in this Form 10-K, into the Company's previously filed Registration Statement File Nos. 2-73501, 2-77328 and 33-23001.

Arthur Andersen & Co.
ARTHUR ANDERSEN & CO.

Stamford, Connecticut,
March 29, 1989

# CHAPTER 10 FACILITY DESCRIPTION

## 10.1 Plant Layout

Figure 10-1 shows the layout of the six contiguous production buildings and of the equipment within the buildings. Three of these buildings, 253, 254 and 256, were added in 1989 and replace two demolished buildings that previously occupied the same area. Production activities, including in-process transfers of materials, equipment and product, take place within the six adjoining buildings. A summary of the type of production activity in each building follows. Details of the processes involved are given in Chapter 15.

In the Oxide Building, UF6 is converted into UO2 granules that subsequently follow one of two paths. The UO2 may be milled and blended in the oxide building, loaded into small containers and carried to Building 255 for pelletizing and/or for shipment offsite as powder. Alternatively, the UO2 granules may be loaded into large hoppers and wheeled to Building 254 for milling, blending and pelletizing.

Building 255 contains one complete pelletizing line that includes equipment for agglomeration, granulation, pressing, sintering, grinding and packaging. It also has facilities for packaging milled and blended powder for shipment offsite and for related UO2 storage.

Building 254 has two parallel pelletizing lines added in 1989. The UO2 is received as granules in large hoppers from the oxide building. Each line has equipment for milling, blending, pelletizing and packaging. These lines in Building 254 have greater capacity and generally are used for the bulk of routine pellet production. The pelletizing line in Building 255 is used as backup and for special pellet runs.

The remaining three buildings contain support facilities for production. Building 256, added in 1989, is the site warehouse for shipping pellets and powder and for receiving site supplies. Building 253, also added in 1989, contains various site utilities, storage and maintenance areas, UO2 storage and offices. Building 240 contains laboratory and maintenance areas, a recycle recovery area, a waste incineration area, personnel areas and offices.

## 10.2 Utilities and Support Systems

### 10.2.1 Electric Power

Electrical power to the Hematite Plant is provided by the Union Electric Company via a substation located approximately 100 yards northeast of Building 255, adjacent to Highway P. The substation transformer steps down the voltage to 12.5 KV and from there is distributed to three stepdown transformers located on the site. The 3-phase output of each stepdown transformer is then connected to metal clad switchgear for distribution to associated buildings. Additional, smaller stepdown transformers (480v/230v/120V) provide power for lighting and general convenience. The major transformer locations and ratings are listed in Table 10-1.

Two natural gas-powered emergency generators provide backup emergency power to maintain critical loads such as emergency air, water, steam, instrumentation, alarms, etc. This natural gas supply is non-interruptable. One unit is located in the Oxide Building and the other unit is located in Building 240. Both emergency generators feed their own respective distribution boards and are switched from normal power to generator (emergency) power by "Onan" automatic line transfer switches. Both units are startup tested on a weekly basis. The generator ratings and their emergency loads are listed in Table 10-1.

10.2.2 Compressed Air

Compressed air is employed for process instrumentation and control for oxide processing including, oxide transfer, milling, blending and pulsed filter cleaning and for general plant maintenance activities.

There are four air compressors. The primary one is located at the east end of the one story addition on the north end of Building 254. It provides dry air to the entire plant. A backup plant compressor is located east of the Building 255-3 area. Both plant compressors have a moisture separator and dryer that lower the dewpoint of plant air to -40°C. A smaller, higher pressure compressor located on the 2nd level of the oxide building provides control air to the cone valves on the blenders in the oxide building. The fourth compressor is located in Building 240. It is powered by the emergency electric generator and feeds air through the dryer to the plant air system to operate pneumatic instrumentation during power outages. Table 10-2 lists the air compressors, their ratings and their uses.

10.2.3 Water

Water used on the C-E Hematite site is supplied by a well located north of Building 253 within the fenced manufacturing area. On the average day, 15,000 gallons are withdrawn from this well. The electric power supply for the well pump is automatically transferred to the emergency generator upon a power outage.

Well water is stored in a 5,000 gallon tank and distributed as needed within the plant, primarily for process water. A circulating water cooling system, including a forced convection evaporative cooling tower, provides equipment cooling.

Water from the site well is analyzed for ground contamination on a monthly basis, and all systems using the potable water supply utilize an air break to prevent inadvertent contamination.

10.2.4 Sanitary System

Sanitary wastes flow to the site sanitary system from sinks, toilets, showers and drinking fountains. This system also receives laundry water, after the water is filtered and held for sampling, and waste water from the process water demineralizer system.

The routing of the sanitary system drains is shown in Figure 10-2. The system includes an extended aeration sewage treatment plant equipped with a hypo-chlorinator and chlorine contact tank with a rated capacity of 3000 gal/day. Discharge from the treatment plant is to the site creek below the site pond under a National Pollutant Discharge Elimination System Permit To Discharge issued by the Missouri Department of Natural Resources.

10.2.5 Storm Water System

Water from roof and ground surface drains flows to the site pond above the dam via the storm water system. This system also receives condensed steam from the UF6 vaporizer steam jackets, cooling water from heat exchanges in the recycle/recovery process and laboratory sink water from cleaning glassware.

The routing is shown in Figure 10-2. Discharge overflow from the pond dam to the site creek is authorized under a National Pollutant Discharge Elimination System Permit To Discharge issued by the Missouri Department of Natural Resources.

10.2.6 Chemical Storage

Chemicals will be stored in accordance with pertinent federal and state regulations. Chemicals currently used are:

Ammonia - approximately 620,000 pounds used per year as a reducing gas in the production of UO2 powder, pellets, and in preparation of material for recycle.

Liquid Nitrogen - approximately 10,000 liters per year used with ammonia to establish a reducing atmosphere in the conversion process and the pellet furnaces.

Potassium Hydroxide - approximately 3,500 pounds used per year. Mixed with process water and used as wet scrubber liquor to remove hydrofluoric acid from the recycle pyrohydrolysis process effluent.

Sulfuric Acid - approximately 5,000 pounds used per year in regeneration of demineralizer resins.

Hydrochloric Acid - approximately 850 pounds used per year in cleaning heat exchanger tubes in the steam boiler.

Nitric Acid - approximately 9,850 pounds used per year to dissolve the U3O8 wet recovery process feed material.

Hydrogen Peroxide - approximately 20,100 pounds per year used to adjust pH in the wet recovery process.

Trichlorethane - approximately 9,500 pounds per year used in preparing UO2 powder for pelletizing.

Perchloroethylene - approximately 600 pounds per year used for maintenance activities.

## 10.3 Ventilation Systems

The Oxide Building is heated by forced hot air supplied by a natural gas-fired heater located on the roof of Building 255. Buildings 255 and 240 are heated by steam supplied by the site boiler located in the south end of Building 253. This boiler is natural gas-fired from an interruptable supply. Fuel oil is stored in an underground tank for use during periods of interruption of natural gas service. Buildings 253 and 254 are heated by forced hot air supplied by natural gas fired heaters located on their roofs. Only the offices, Laboratory, and Maintenance Shop are air conditioned.

Ventilation air from the Oxide Building, from Buildings 253, 254 and 255 and from the Recycle/Recovery Areas in Building 240 is passed through absolute filters prior to release to the atmosphere, except for the pellet furnace room air exhausts in Building 255.

All exhaust stacks are continuously monitored when in operation. The locations and flow rates for the exhaust stacks are shown on Figure 10-3.

## 10.4 Radioactive Waste Handling

### 10.4.1 Liquid Wastes

There are no planned releases of radioactive liquid wastes from routine production processes. Radioactive liquid wastes are generated from mop and cleanup water and from the wet recovery process but they are not released as liquid effluent.

Cleanup water is evaporated to recover the uranium. Mop water and process water from wet recovery is evaporated for concentration and then solidified for shipment to licensed burial.

Trace amounts of radioactivity may be found in laundry, sink and shower water. Laundry water is filtered and sampled prior to discharge to the sanitary waste system. Water from change room sinks and showers is also routed to the sanitary waste system. The sanitary waste effluent enters the site creek immediately below the site pond. It is sampled and analyzed for gross alpha and beta activity.

Small quantities of liquids from cleaning glassware in the laboratory are discharged to the storm drain system. Disposal of lab analytical residues to the sink drains is not practiced, as they are recycled for recovery. The storm drain system discharges into the site pond which overflows to form the site creek. The overflow is continuously proportionately sampled and analyzed for gross alpha and beta activity.

10.4.2 <u>Solid Wastes</u>

Solid wastes which are potentially contaminated are generated throughout the controlled area. These wastes consist mostly of rags, papers, packaging materials, worn-out shop clothing, equipment parts, and other miscellaneous materials that result from plant operations. After passive assay (gamma-counting) to determine the U-235 content, non-combustible wastes are compacted in 55 gallon drums or packaged in metal tote boxes for shipment to a commercial licensed low-level burial site.

Combustible solid waste is fed to a gas-fired incinerator to reduce the volume of contaminated wastes for shipment to licensed burial. The incinerator also supplements the oxidation/reduction furnaces used to reduce wastes containing recoverable quantities of uranium. The incinerator is equipped with a wet scrubber system to clean offgases prior to discharge. Incinerator ash with non-recoverable quantities of uranium and also the final residue from the wet recovery process after evaporation are solidified with concrete and placed into 55 gallon drums for shipment to commercial licensed burial.

Calcium fluoride and limestone from the conversion process dry scrubbers are used as fill materials on site. These materials, referred to as spent limestone, generally do not contain detectable contamination and are not considered to be radiological solid waste when the uranium alpha activity is less than 30 picocuries per gram. Contaminated limestone is held within the controlled area.

Non-radioactive solid waste is disposed of by a commercial waste disposal firm. Non-contaminated equipment may be disposed of to commercial scrap dealers, or by other means.

## 10.5 Fire Protection

Technicians who report to the Manager, Nuclear Licensing, Safety and Accountability function as Fire Marshalls during normal working hours. Alternate Fire Marshalls are the Shift Supervisors. They perform inspections of buildings for fire hazards during their routine inspection of the plant and operations, and perform regular inspections of all fire fighting equipment. Training is provided to plant operators in the use of fire fighting equipment. The Hematite and Festus fire departments respond to emergency calls at C-E Hematite.

Facilities are designed, constructed, and operated consistent with requirements of all applicable fire safety codes. A copy of the Certificate of Insurability from American Nuclear Insurers is provided in Figure 10-4.

TABLE 10-1
ELECTRIC POWER SOURCES

Normal Power Supply Transformers

| Transformer Location | Input Voltage | Output Voltage | KVA Rating |
|---|---|---|---|
| Oxide Plant | 12.5 KV | 480 v | 500 KVA |
| East of Bldg. 110 | 12.5 KV | 480 v | 1500 KVA |
| Bldg. 240 | 12.5 KV | 230 v | 500 KVA |
| Bldg. 255 | 480 v | 208 v | 500 KVA |

Emergency Generators

| Generator Location | Rating | Primary Loads |
|---|---|---|
| Oxide Building<br>4th floor | 3-phase<br>120/208 volts<br>7.5 KW<br>startup and<br>transfer=5 sec. | 1. Instrumentation<br>2. Alarms<br>3. Emergency Lighting<br>4. Oxide Roof Exhaust |
| Building 240<br>Utilities Room | 3-phase<br>120/220 volts<br>75 KW<br>startup and<br>transfer=25 sec. | 1. Well Pump<br>2. Nuclear Alarms<br>3. Burner Blower-Boiler<br>4. Feed Water Pump-Boiler<br>5. Feed Water Control Panel-Boiler<br>6. Roof Exhaust Above Generator<br>7. Air Compressor<br>8. Emergency Lighting (Bldgs. 253,254)<br>9. Telephones<br>10. Cooling Water Pump |

TABLE 10-2
COMPRESSED AIR SUPPLY

| Unit | Location | Rating | Use |
|---|---|---|---|
| Kobelco | Utility Room, North of Building 254. | 1200 scfm 125 psig | Plant wide air supply, dry air. |
| Atlas | Compressor Room, East of Building Area 255-3 | 420 scfm 115 psig | Backup plant air supply, dry air. |
| Little Joy | Laundry Room, Building Area 240-1. | 165 scfm 100 psig | Emergency air supply. Powered by emergency generator. Feeds into air dryer. Mainly for instrument supply. |
| Quincy | Oxide Building, 2nd floor. | 15 scfm 250 psig | High pressure air supply to open air mix valve. |

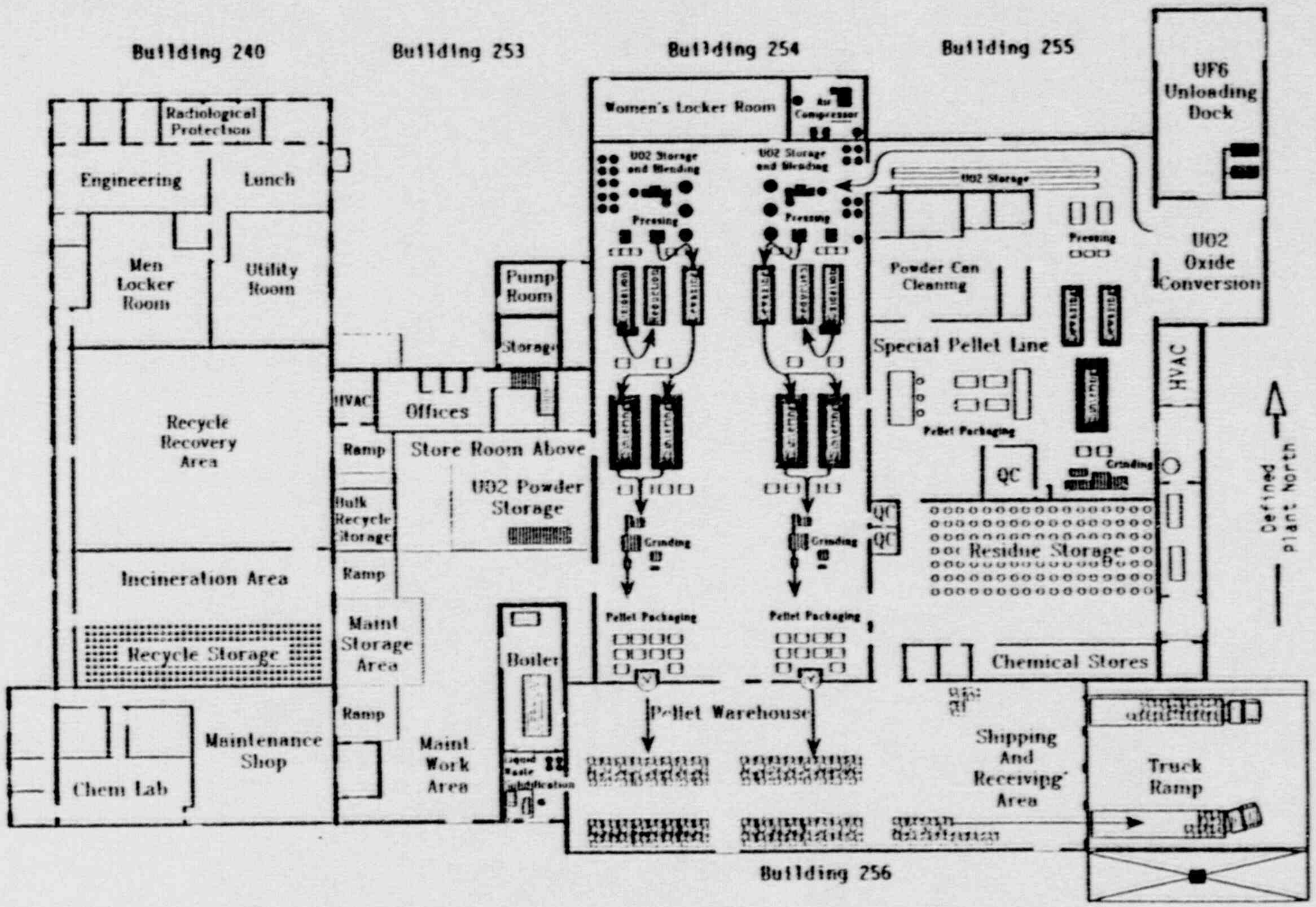


FIGURE 10-1 Building and Equipment Layout

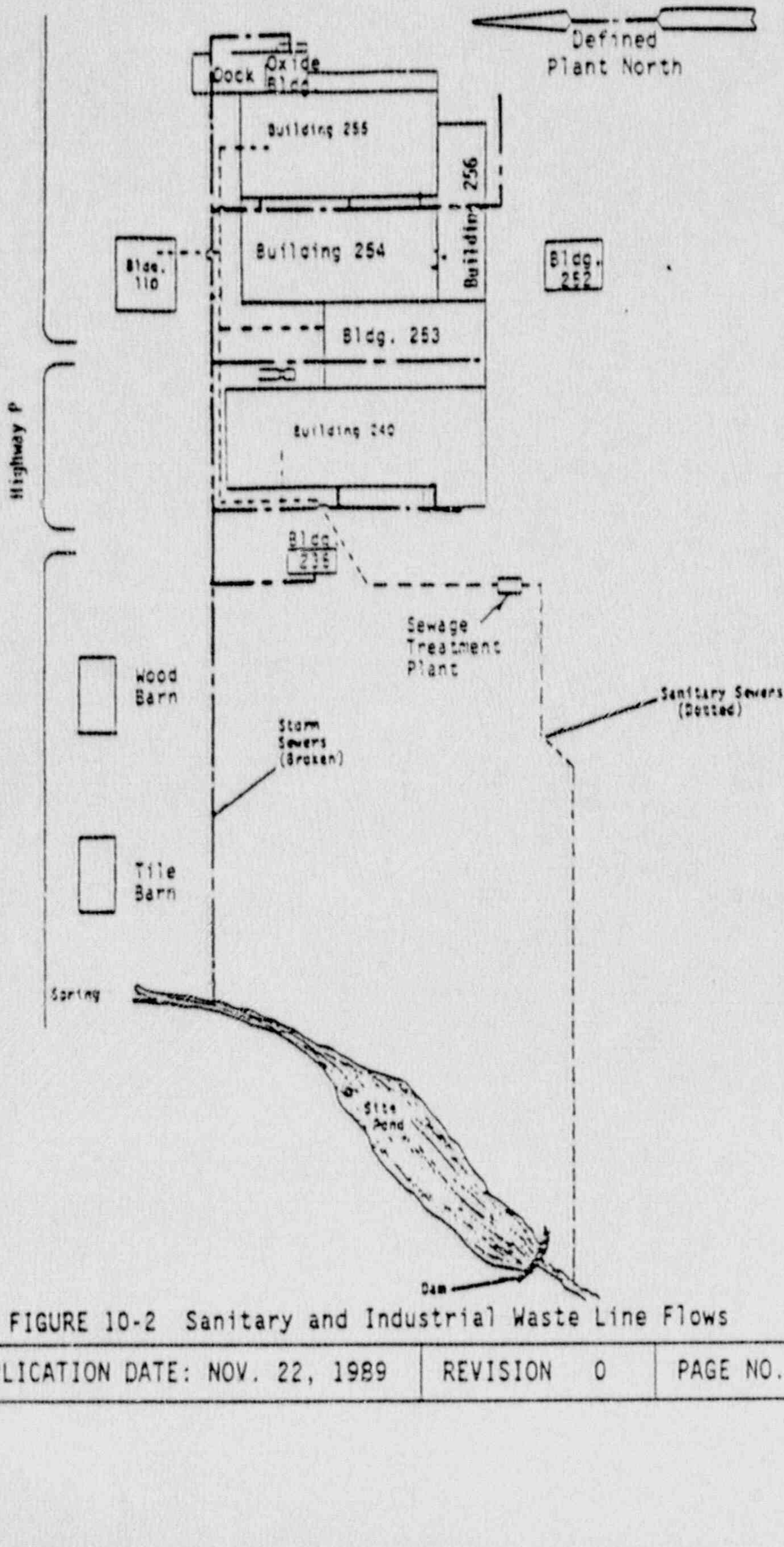


FIGURE 10-2 Sanitary and Industrial Waste Line Flows

| STACK NUMBER | IDENTIFICATION | FLOW RATE (CFM) |
|---|---|---|
| S 017 | Pellet Plant Roof Exhaust | 9000 |
| S 050 | Pellet Plant West System | 12000 |
| S 051 | Pellet Plant East System | 9800 |
| S 103 | Powder Unload Hoods-West Bank | 4900 |
| S 106 | Oxide East Bank | 9800 |
| S 114 | Dry Scrubber Exhaust | 1100 |
| S 228 | Red Room Dry Side | 3700 |
| S 230 | Red Room Wet Side | 5800 |
| S 232 | Green Room-Incinerator | 3800 |
| S 301 | Building 254 East Pellet Line Powder Preparation, Pressing | 12000 |
| S 302 | Building 254 West Pellet Line Powder Preparation, Pressing | 12000 |
| S 303 | Building 254 East Pellet Line Furnace Area | 12000 |
| S 304 | Building 254 West Pellet Line Furnace Area | 12000 |
| S 305 | Building 254 Grinders, Pellet Loading | 12000 |
| S 306 | Building 253 Recycle Loading | 1500 |

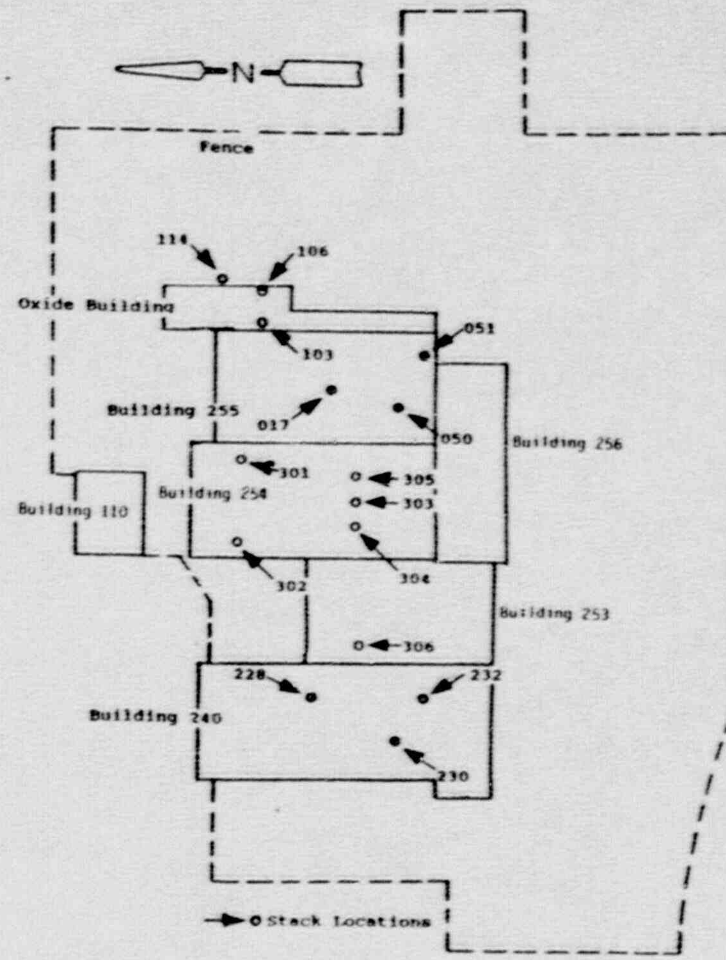


FIGURE 10-3 EXHAUST STACK LOCATIONS AND FLOW RATES

AMERICAN NUCLEAR INSURERS

The Exchange, Suite 245 / 270 Farmington Avenue
Farmington, Connecticut 06032 / (203) 677-7305 / TLX. No. 643-089

MUTUAL ATOMIC ENERGY LIABILITY UNDERWRITERS
Suite 3720
One East Wacker Drive
Chicago, Illinois 60601

## CERTIFICATE OF INSURANCE

This certificate is issued to the Certificate Holder as a matter of information only. It does not amend, extend or alter the coverage afforded by the policies listed below.

Name of Insured COMBUSTION ENGINEERING, INC.

Mailing Address 900 Long Ridge Road, Stamford, Connecticut

Location(s) Covered Windsor, Connecticut - Hematite, Missouri

This is to certify that the following policies one issued by members of American Nuclear Insurers (ANI) and the other issued by members of Mutual Atomic Energy Liability Underwriters (MAELU), respectively, to the Insured named above are in force as of the effective date of this certificate.

| Policy Numbers | Policy Expiration Date* | Amount or Limit | Deductible |
|---|---|---|---|
| 5396 | July 1, 1990 | $225,000,000. Loc. 1<br>$ 90,000,000. Loc. 2 | $250,000. |

**Type of Insurance:** All risk of direct physical damage to the Property Insured by any Cause of Loss specified as covered in the policy, provided such physical damage takes place during the policy period.

**Cancellation of Policies:** Should either or both of the policies described above be cancelled before the expiration thereof, the issuing entity (ANI or MAELU) will endeavor to mail or deliver advance written notice to the Certificate Holder, but failure to provide such notice shall impose no obligation or liability of any kind upon ANI or MAELU.

**Name and Address of Certificate Holder:** Mr. George Hess
Combustion Engineering, Inc.
Mail Code 9332-0407
1000 Prospect Hill Rd.
Windsor, CT 06095

Effective date of the Certificate: July 1, 1989

*A CERTIFICATE WILL NOT BE ISSUED FOR ANY SUBSEQUENT POLICY PERIOD UNLESS REQUESTED.

FIGURE 10-4 CERTIFICATE OF INSURABILITY

# CHAPTER 11 ORGANIZATION AND PERSONNEL

## 11.1 Organizational Responsibilities

Figure 11-1 is the Hematite plant organization chart. Descriptions of the responsibilities of some supervisory and higher level positions shown in the chart and that were not included in Section 2.1 follow.

### 11.1.1 Manager, Administration and Production Control

The Manager, Administration and Production Control reports to the Plant Manager. He manages the control of SNM from receipt at the Hematite facility, through the production process until it is shipped as product or waste. His activities include scheduling of production, selection of SNM for use in the production process, specification of the product lot makeup, scheduling of SNM shipments from the supplier to Hematite and from Hematite to the customer, coordination of the packaging and shipment of SNM waste and residues to a commercial, licensed disposal facility and development of procedures for packaging, shipping and receiving. He also performs facility administration duties including the supervision of the guards, s... rchasing and personnel services.

### 11.1.2 Manager, Quality Control

The Manager, Quality Control reports to the Plant Manager. He manages the measurement activities which verify that the product conforms to specification. These activities may include development of the Operation Sheets that are the procedures for acquisition of product data, approval of laboratory measurement methods, approval of statistical methodology for data evaluation and establishment of the system for control and distribution of data documentation. The Manager maintains separation between his measurement activities and the production activities that he monitors. He has authority to halt production and it shall not restart until approved by the Plant Manager or a duly authorized alternate.

### 11.1.3 Coordinator of Nuclear Materials Accountability

The Coordinator of Nuclear Materials Accountability reports to the Manager of Nuclear Licensing, Safety and Accountability. He maintains the SNM accounting records, prepares NRC required reports on material balance, transfer and inventory, periodically verifies current knowledge of the presence of SNM and computes Inventory Differences.

### 11.1.4 Supervisor, Material Control

The Supervisor, Material Control reports to the Manager, Administration and Production Control. He implements the production schedules provided by the Manager through supervision of the production clerk, the material control operators and the material handlers. He monitors the sequence of steps in the processing and handling of each material unit including the proper use of the Traveler that documents each process step.

### 11.1.5 Supervisor, Quality Control Engineering

The Supervisor, Quality Control Engineering reports to the Manager, Quality Control. He supervises the quality control technicians who obtain the measurement samples and he supports the activities of the Manager. His support may include recommendations on sampling plans, development of statistical methods, evaluation of data trends, recommendations on measurement standards, participation in writing procedures, review and approval of Travelers and Operation Sheets and administration of the document control system.

### 11.1.6 Supervisor, Laboratory

The Laboratory Supervisor reports to the Manager, Quality Control. He supervises and trains the laboratory technicians, recommends sampling procedures, establishes laboratory methods and reviews and approves all chemical measurements on SNM. He also selects subcontractors and qualifies and coordinates their measurement services.

### 11.1.7 Supervisor, Maintenance

The Supervisor, Maintenance reports to the Superintendent, Production. He supervises technicians in the maintenance activities related to the facility and the production equipment within the constraints of applicable radiation and industrial safety practice.

## 11.2 Functions of Key Personnel

The functions, responsibilities and authorities of key personnel positions are described in Section 2.1. Succession to each position in the event of absence is authorized in writing by the person holding the position and is done with the knowledge of the plant manager.

## 11.3 Education and Experience of Key Personnel

Resumes of key personnel important to safety are provided in this section for the following personnel:

1. C. R. Waterman - Vice President and General Manager, Nuclear Fuel Manufacturing (located in Windsor, CT)

2. J. A. Rode - Plant Manager

3. R. J. Klotz - Nuclear Criticality Specialist (located in Windsor, CT)

4. H. E. Eskridge - Manager, Nuclear Licensing, Safety, and Accountability

5. A. J. Noack - Superintendent, Production

6. R. W. Griscom - Manager, Engineering

7. C. W. Proctor, Jr. - Supervisor, Health Physics

CHARLES R. WATERMAN - VICE PRESIDENT AND GENERAL MANAGER,
NUCLEAR FUEL MANUFACTURING

EDUCATION

B. S. Electrical Engineering, Tri-State College, 1957

EXPERIENCE

COMBUSTION ENGINEERING, INC.
Windsor, Connecticut
Vice President and General Manager, Nuclear Fuel Manufacturing — 1989 to Present

Overall responsibility for the safe operation of Combustion Engineering's nuclear fuel manufacturing facilities located in Hematite, Missouri and in Windsor, Connecticut. Continuing responsibility as the Plant Manager for the Windsor Nuclear Fuel Manufacturing facility.

Plant Manager, Windsor Nuclear Fuel Manufacturing — 1988 to 1989

Responsible for day-to-day manufacturing operations, accountability, security, nuclear criticality safety and radiological safety related to all special nuclear and source material received by Windsor Nuclear Fuel Manufacturing and used in any manufacturing process. Assured compliance with Federal and State and local regulations and the requirements and limitations set forth in facility license SNM-1067.

Director, Outage Services — 1986 to 1988

Responsible for management of outage services, for development test and application of maintenance and inspection services provided to nuclear utilities. These services included integrated refueling and maintenance outages, fuel services, major plant retrofits, full steam generator services using advanced remote controlled devices designed and built by outage services. Responsible for all aspects of compliance with Windsor facility NRC by-product license.

Responsible for the operation of Amdata, Inc., a wholly owned subsidiary of CE. Amdata designs and manufactures advanced ultrasonic imaging equipment and inspection services for the Nuclear, Oil and Gas and Aerospace industries.

CHARLES R. WATERMAN (continued)

President/General Manager — 1985 to 1986
CE/Delas Weir, Inc.

Responsible for strategic direction of this joint venture company as well as day-to-day operations. CE/Delas Weir provided new and replacement heat exchanger equipment to Electric Utilities.

ENGINEERING PLANNING AND MANAGEMENT CO. — 1984 to 1985
President and CEO

Responsible for strategic direction and day-to-day operations. EPM supplied engineering services to the Nuclear Utility Industry and Maintenance Management Software Programs to the Food and Pharmaceutical Industries.

SENSOR ENGINEERING COMPANY — 1981 to 1984
DIVISION OF ECHLIN, INC.
President

Responsible for strategic direction and day-to-day operations for a high-tech manufacturing company of magnetic devices for use in security, automotive and industrial control systems.

ELECTRO-MECHANICS, INC. — 1977 to 1981
President and Chairman of the Board

Responsible for strategic direction and day-to-day operations for a manufacturer of custom electronic control systems for the Electric Utility and Commercial Industries.

COMBUSTION ENGINEERING, INC. — 1965 to 1977
Director, Plant Apparatus — 1975 to 1977

Responsible for technical specification and procurement of all NSSS mechanical hardware. Established the organization and QA Manual for CE's Engineering "N" stamp.

Project Manager, Boston Edison — 1972 to 1975
Nuclear Reactor Project

Responsible for representing Combustion Engineering with Boston Edison and all governmental agencies in matters relating to contracts, licensing, design, fabrication schedules, erection, startup and acceptance of the nuclear steam supply systems and associated fuel. Authorized all work to

CHARLES R. WATERMAN (continued)

be done on the project. Reviewed, evaluated, and approved all technical work on the project, including purchase orders, purchase order supplements, drawings, and specifications.

<u>Assistant Project Manager, Maine Yankee Nuclear Reactor Project</u> 1970 to 1972

Assisted the Project Manager in the direction of technical and administrative activities.

<u>General Manager, Naval Reactor Division</u> 1968 to 1970

Overall responsibility for the operations of the SIC land-based submarine prototype facility under contract between CE and the U.S. Atomic Energy Commission.

<u>Operations Manager, Naval Reactors Division</u> 1967 to 1968

Responsible for operation and maintenance of the SIC prototype and support facilities. Responsiblities included the approval and implementation of schedules for operation testing, training, maintenance, plant modifications and for the control and forecast of department budgets.

<u>Assistant Operations Manager, Training Manager, Naval Reactor Division</u> 1965 to 1967

Responsible for training SIC prototype operating personnel including coordination of Naval and Division personnel.

CONNECTICUT YANKEE ATOMIC POWER COMPANY 1964 to 1965
<u>Nuclear Engineer</u>

In training as Assistant Plant Superintendent.

COMBUSTION ENGINEERING, INC. 1959 to 1964
<u>Senior Shift Supervisor, Naval Reactors Division</u>

Responsible for the safe and efficient operation of the SIC prototype. Electrical design engineer responsible for the evaluation of electrical and electronic systems and equipment.

CHARLES R. WATERMAN (continued)

ELECTRIC BOAT DIVISION, GENERAL DYNAMICS CORPORATION — 1959
<u>Test Director</u>

Test Director on the Nuclear Submarine USS (BN) George Washington for preoperational and operational tests.

COMBUSTION ENGINEERING, INC. — 1957 to 1959
<u>Electrical Control Engineer, Reactor Development Division</u>

Reactor analysis for the SIC/S2C reactor plants.

JAMES A. RODE - PLANT MANAGER, HEMATITE

EDUCATION:

B.S., Chemical Engineering, University of Texas, 1953

EXPERIENCE:

COMBUSTION ENGINEERING, INC. 1974 to Present
Plant Manager, Nuclear Fuel Manufacturing, Hematite

Responsible for all Nuclear Fuel Manufacturing activities at the Hematite Plant. Manages Engineering, Production and Materials Control, Manufacturing, Nuclear and Industrial Safety, Nuclear Material Management, and Quality Control.

GULF UNITED NUCLEAR FUELS CORPORATION 1968 to 1974
Technical Consultant

Responsible for establishing process flow sheets and capacities for production of $UO_2$, $UO_2$ pellets, and uranium recovery; and coordinating development activities. Also responsible for preparation of stable density pellets and development of process modifications. Technical Assistant to the Manager of Chemicals Operations on major operational problems.

UNITED NUCLEAR CORPORATION
Manager of Facilities Development and Technical Director 1964 to 1968

Responsible for design, construction and startup of the first large scale fluidized-bed process for the production of $UO_2$ from $UF_6$ and of companion facilities for converting oxide to pellets.

Responsible as Technical Director for Chemicals Operations for process engineering supervision and development activities including design, construction, and operations of a pilot plant for preparation of $UO_2$ via the reaction of $UF_6$ and steam and for development, design, construction and startup of a fluid-bed vapor phase coating system.

Assistant Technical Director 1962 to 1964

Responsible for process and equipment design in the Rhode Island Scrap Recovery Facility, development work on process for producing pyrolytic carbon coated $UO_2$, and for continuing development work in Naval Fuel Program.

JAMES A. RODE (continued)

Project Leader 1961 to 1962

Assumed total responsibility for salvaging a non-operative Naval Fuels Plant including production, quality control, development and customer contacts. The facility was converted into the primary source of profits for the Chemical Operations.

MALLINCKRODT CHEMICAL WORKS
Group Leader and Production Superintendent 1958 to 1961

Responsible for the startup of high enrichment metal production and development and startup of the Hematite Pellet Plant.

Responsible as Production Superintendent for detailed supervision of production in both high and low enrichment conversion operations.

Process Engineer and Research Chemist 1953 to 1958

Participated in preparation of proposals for production of yttrium metal and conversion of 5000 tons per year of $UF_6$. Responsible for operation of the first ADU pilot plant and startup of the Hematite Oxide Plant.

ROBERT J. KLOTZ - NUCLEAR CRITICALITY SPECIALIST

EDUCATION:

Graduate, Oak Ridge School of Reactor Technology, 1957
M.S. Physics, Kansas State College, 1954
A.B. Physics and Mathematics, Kansas State Teachers College of Emporia, 1952
Graduate Studies, Texas Christian University

EXPERIENCE:

COMBUSTION ENGINEERING, INC. — 1965 to Present
Windsor, Connecticut

Senior Consulting Physicist — 1977 to Present

Responsible for the physics design of new and spent fuel racks, fuel transfer machines, and other equipment involved in moving, testing or storing fuel. Nuclear Criticality Specialist provide technical support and criticality audit function at both the Windsor Manufacturing and Hematite Fuel Manufacturing facilities. Involved in solving special physics problems.

Section Manager, Radiation and Criticality Physics — 1965 to 1977

Responsible for radiation shielding, the ex-core criticality, and determination of source terms for Nuclear Steam Supply Systems. Also for providing nuclear heat generation rates for structures in the NSSS, and radiation dose rates for assessing physical changes in NSSS materials and equipment in the radiation environment.

GENERAL NUCLEAR ENGINEERING CORPORATION
Physicist — 1959 to 1965

Responsible for the shield design of the heavy water research reactor at the Georgia Institute of Technology and the thermal and biological shield design analysis for the Boiling Nuclear Superheat Reactor (BONUS) located in Rincon, Puerto Rico. Reviewed all the literature on radiation shielding for the publication Power Reactor Technology.

CONVAIR DIVISION OF GENERAL DYNAMICS
Physicist — 1954 to 1959

Responsible for the design of a shield for a mobile reactor of the Army Compact Core Design and for a Nuclear Ramjet Missile. Performed analysis of aircraft nuclear shielding experiments, developed shielding programs for computers, and contributed to the Aircraft Shield Design Manual.

HAROLD E. ESKRIDGE - MANAGER, NUCLEAR LICENSING, SAFETY AND ACCOUNTABILITY

EDUCATION:

B.S., Physics, North Carolina State University, 1961
M.S., Physics, North Carolina State University, 1963

EXPERIENCE:

COMBUSTION ENGINEERING, INC.
Manager, Nuclear Licensing, Safety and Accountability - Hematite — 1989 to Present
Supervisor, Nuclear Licensing, Safety and Accountability - Hematite — 1974 to 1989

Responsible for licensing, safety, and safeguards at Nuclear Fuel Manufacturing - Hematite. Develops and implements the health physics, criticality and industrial safety, and accountability programs for the Hematite facility. Audits manufacturing operations and supervises safety and safeguards personnel in day-to-day operations.

GENERAL ELECTRIC COMPANY — 1972 to 1974
Nuclear Safety Engineer

Analyzed changes and specified requirements for Wilmington nuclear fuel manufacturing to assure compliance. Audited manufacturing operations and radiation protection programs. Planned and conducted development programs in dosimetry, radiation monitoring and environmental sampling.

SALISBURY METAL PRODUCTS COMPANY — 1971 TO 1972
Co-Manager

Managed operations for manufacturer of precision components; including sales, finance, production control and quality assurance. Consultant to Institute for Resources Management on decontamination and radioactive waste disposal projects and a member of Rowan Technical Institute Advisory Committee.

EVIRONONICS, INC. — 1970 to 1971
Vice President - Nuclear Applications

Performed variety of functions, including market research, proposal preparation and technical analyses relating to remote sensing, environmental surveys, and health physics services. Contacted potential customers, including government agencies and utility companies with power reactors.

HAROLD E. ESKRIDGE (continued)

EG&G, INC. 1967 to 1970
Senior Scientist and Scientific Executive

Head, Radiological Sciences Section and Senior Health Physicist, responsible for radiation and nuclear safety and regulatory compliance for Las Vegas Operations. Provided technical direction for Nuclear Counting Laboratory, Nevada Aerial Tracking System, and Aerial Radiation Measuring Surveys Programs. Acting Manager, Environmental Measurements Department, which included High Energy Neutron Reactions Experiment and Metrology Sections.

NORTH CAROLINA STATE BOARD OF HEALTH 1962 to 1967
Public Health Physicist

Technical, policy, and procedural consultation in all aspects of health physics, environmental surveillance and radiological health. Functioned as administrator of Radioactive Materials Licensing and Regulation. Served as Team Chief of State Radiological Emergency Team and established and equipped a laboratory for radiological and chemical analysis of environmental samples.

U.S. AIR FORCE 1954 to 1957
Nuclear Specialist

Responsible for criticality and radiological safety for nuclear weapon systems and components. Also was an instructor in nuclear safety and weapons systems.

ARLON J. NOACK - PRODUCTION SUPERINTENDENT, HEMATITE

EDUCATION:

Hillsboro High School, 1961 Graduate

EXPERIENCE:

COMBUSTION ENGINEERING, INC.
Production Superintendent - Hematite 1981 to Present

Responsible for production and maintenance operations, operator and maintenance training, manpower scheduling, interviewing and hiring operating personnel, handling Union grievances, and training new production and maintenance supervisors.

Maintenance Supervisor - Hematite 1980 to 1981

Responsible for the maintenance of production equipment, building and grounds maintenance, ordering repair parts, and porter service.

Production Supervisor - Hematite 1974 to 1980

Shift Supervisor in charge of production operations, dealing with Union problems, operator training, and scheduling production to assure fulfillment of customer schedule requirements.

GULF UNITED NUCLEAR FUELS CORPORATION
Production Supervisor - Hematite 1970 to 1974

Production Supervisor in charge of production operations, dealing with Union problems, operator training, and scheduling production to fulfill customer schedule requirements.

Engineering Technician - Hematite 1969 to 1970

Responsible for production engineering functions as assigned by the Process Engineer, some drafting responsibilities, and Engineering technical assistance.

UNITED NUCLEAR CORPORATION 1966 to 1969
Process Development Technician - Hematite

Participated in development of Uranium Oxide Conversion Plant, such as operating and repairing development equipment, and assisting in the development of new operating techniques.

ARLON J. NOACK (continued)

LUDLOW SAYLOR WIRE CLOTH COMPANY 1963 to 1966
Production Operator - St. Louis

Operated wire screen loom, wire stretcher, and punch press.

HOWARD INDUSTRIES COMPANY 1962 to 1963
Junior Draftsman - Festus

Responsible for drawing changes, drawing minor equipment, and document control of production drawings.

ROBERT W. GRISCOM - MANAGER, ENGINEERING, HEMATITE

EDUCATION:

B.S., Chemical Engineering, Georgia Institute of Technology, 1969 Co-op
M.S.C.E.-Sanitary, University of Missouri-Rolla, 1974.

EXPERIENCE:

| | |
|---|---|
| COMBUSTION ENGINEERING, INC. | 1981 to Present |
| Manager, Engineering - Hematite | 1989 to Present |
| Engineering Supervisor - Hematite | 1981 to 1989 |

Responsible for managing Engineering Department. Activities including process engineering, plant expansion design and management, drafting, instrument maintenance, and staff assistance to other plant departments.

NATIONAL STEEL ENGINEERS & ASSOCIATES — 1977 to 1981
Project Manager - St. Louis

Project Manager in corporate environmental consulting group. Directly responsible for engineering, fabrication, and installation of multi-million dollar air pollution control and wastewater treatment systems for major steel companies.

ROCKWELL INTERNATIONAL — 1974 to 1977
Senior Test Engineer - St. Louis

Responsible for establishing and gathering an hourly emission inventory for EPS sponsored St. Louis Regional Air Pollution Study (RAPS). Also supervised and performed stack sampling in St. Louis and New Mexico.

MONSANTO COMPANY — 1969 to 1974
Process Engineer - St. Louis

Responsible for process and cost improvements in various chemical production departments. Designed and installed a wastewater treatment system for removing phenolics.

C. W. PROCTOR, JR. - SUPERVISOR, HEALTH PHYSICS

EDUCATION:

B.S., Biology, Centenary College, 1966
M.S., Entomology, University of Georgia, 1970
Nuclear Engineering Graduate Studies, Univ. of Cincinnati, 1976
M.S., Engineering Management, University of Missouri, Rolla, 1983

EXPERIENCE:

COMBUSTION ENGINEERING, INC., HEMATITE — 1989 to Present
Supervisor, Health Physics

Responsible for the daily operations management of the health physics department and staff at Nuclear Fuel Manufacturing - Hematite. Implements health physics and industrial safety program through training, supervision, and daily audit. Develops and revises departmental operations procedures and emergency plan implementing procedures.

IMPELL CORPORATION — 1988
Lead Senior Engineer

Worked with a team which designed and installed 18,000 permanent signs on the process piping at the Zion nuclear power station.

Managed the development of off-site emergency plan implementing procedures for Duane Arnold nuclear power plant.

UNITED ENERGY SERVICES, INC., — 1987 to 1988
Nuclear Consultant

Performed pre-INPO mini-audit of health physics procedures for Waterford 3 nuclear power plant.

Developed emergency response lesson plans for Rancho Seco. Trained the plant staff in nuclear emergency response procedures. Audited control room and management response to emergency drills.

GENERAL TECHNICAL SERVICES (GENERAL PHYSICS) — 1986 to 1987
Nuclear Instructor

Taught radiation protection, industrial safety, and nuclear emergency response at Votgle Plant. Developed on-site nuclear emergency implementing procedures.

C. W. PROCTOR, JR. (continued)

NUTECH ENGINEERS, INC. — 1986
Consultant

Wrote on-site nuclear emergency implementing procedures for Davis Besse nuclear power plant.

UNION ELECTRIC COMPANY — 1976 to 1986
Supervisor, Nuclear Information

Conceived and executed a nuclear public information and education program including technical lab presentations on nuclear radiation and radioisotope decay to physics and chemistry classes.

UNIVERSITY OF CINCINNATI, DEPT. OF CHEMICAL & NUCLEAR ENGINEERING — 1974 to 1976
Administrator and Spokesman

Studied energy and nuclear issues at Oak Ridge Associated Universities. Made informational and technical lab presentations on these issues.

DAMES & MOORE, CONSULTANTS — 1971 to 1973
Environmentalist

Analysis and characterization of terrestrial ecosystems including environmental site characterizations of nuclear power plants.

## 11.4 Operating Procedures

The preparation, review and approval of operating procedures is described in Section 2.6. Procedures for equipment operation may be posted locally at the equipment and these and the more general procedures may be presented during personnel training as appropriate.

## 11.5 Training

The training program for employees is described in Section 2.5.

## 11.6 Changes in Procedures, Facilities and Equipment

The review procedure for changes in processes, equipment and/or facilities is described in Section 2.6. It includes provisions for analysis, review, approval, verification and recording.

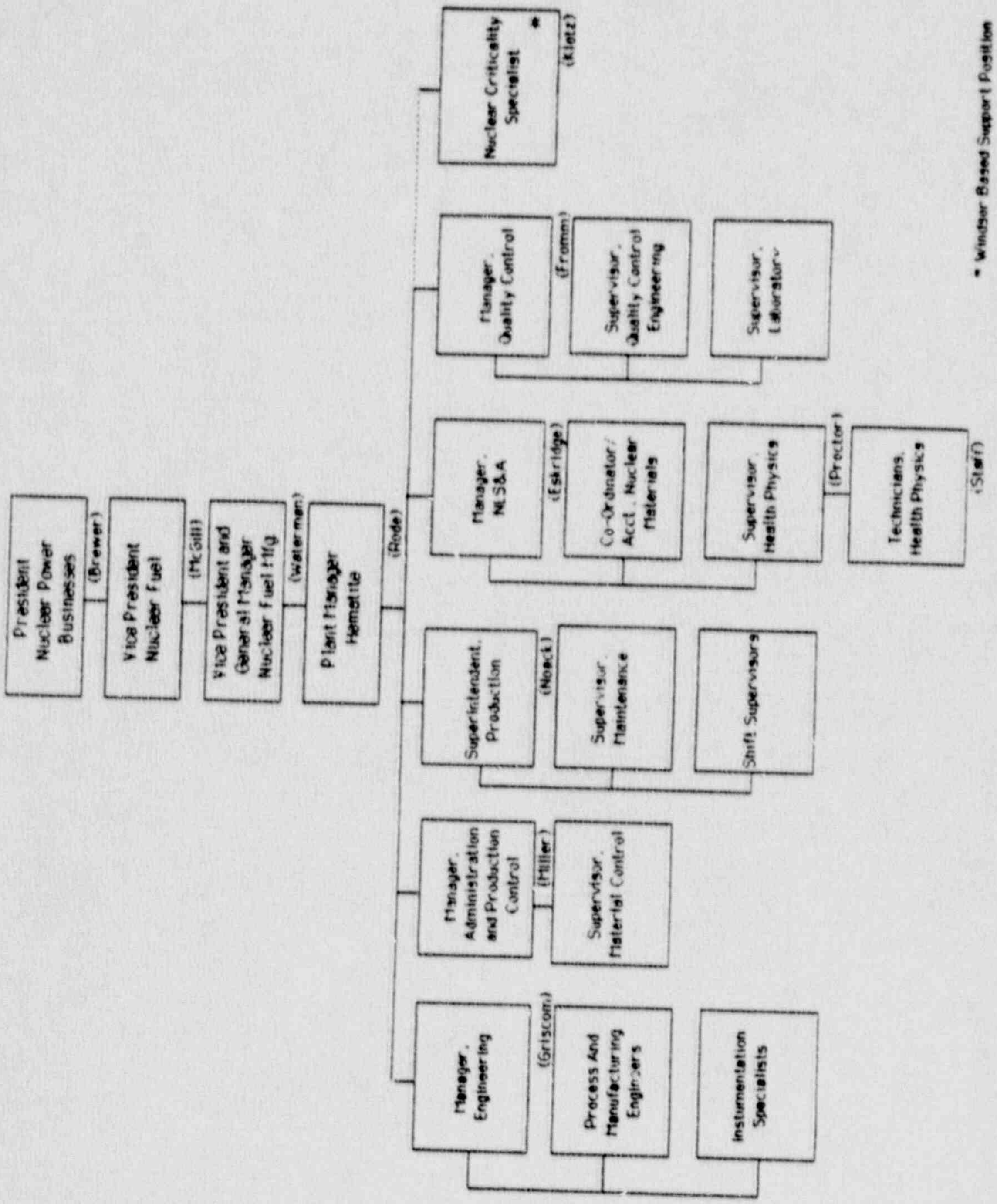


FIGURE 11-1 HEMATITE PLANT ORGANIZATION CHART

## CHAPTER 12 RADIATION PROTECTION

### 12.1 Program

A manual containing procedures necessary to implement the radiation safety program described in Part I of this renewal application is maintained by NLS&A.

All routine operations involving SNM handling are covered by an Operation Sheet (O.S.) and/or by a Special Evaluation Traveler (S.E.T.). A separate O.S. covers plant-wide radiation safety procedures, while procedures specific to a certain operation are covered in the O.S. for that operation.

The NLS&A Manager reviews all O.S.s and S.E.T.s regarding all aspects of safety. The Nuclear Criticality Specialist overchecks the criticality safety controls. All approvals are documented. The Shift Supervisors instruct their people to assure their understanding of the operations and their safety limits and restrictions. Adequate performance of individuals is continually monitored by the Shift Supervisors.

The Shift Supervisors further assure that each work area is properly posted, and that operations are performed in compliance with posted limits and written instructions.

### 12.2 Posting and Labeling

All areas involving nuclear fuel handling or storage are posted with criticality safety limits. Radiological posting of areas is in accordance with 10 CFR 20.203. All mass-limited containers of SNM are labeled as to their contents. Areas and equipment for which criticality safety is assured by moderation control are appropriately posted to prevent the introduction of water or excessive hydrogeneous materials.

Other signs containing summary instructions, cautions, and reminders relating to safety are posted, as appropriate or required, throughout the plant.

### 12.3 External Radiation - Personnel Monitoring

All personnel are required to wash their hands and monitor for contamination before exiting the contaminated area. Alpha personnel monitors are located beyond the step-off pad at each change area. Any person having contamination must wash thoroughly and recheck for contamination. If contamination persists, a member of the NLS&A group will assist in decontamination.

### 12.4 Radiation Surveys

Removable contamination on surfaces in plant areas and on items to be released to unrestricted areas are determined by smearing. Limits are provided in Chapter 3.

Direct radiation surveys of plant environs, sealed sources, and offsite shipments of radioactive materials are made as necessary to comply with 10 CFR 20.201. All survey results are documented.

### 12.5 Reports and Records

Records required by NRC regulations and this license are retained by the NLS&A group. These records include alterations or additions made, abnormal occurrences and events associated with radioactivity release, criticality analyses, audits and inspections, instrument calibrations, ALARA findings, employee training and retraining, personnel exposure, routine and special radiation surveys, and SNM control records required by 10 CFR 70.51.

Retention of records is described in Section 2.9.

### 12.6 Instruments

Type of radiation detection instruments, their capabilities, and frequency of calibration are described in Section 3.2.4.

### 12.7 Protective Clothing

Protective clothing is worn as specified by NLS&A posting or as specified by the Operation Sheet for a particular operation, including: coveralls, lab coats, safety shoes, shoe covers, cotton and rubber gloves, safety glasses, face shields, respirators, supplied-air breathing apparatus, rubber aprons, and acid suits.

### 12.8 Administrative Control Levels, Including Effluent Control

External occupational exposure is controlled by individual personnel dosimetry. A film badge and I.D. badge with indium foil is worn by personnel at all times they are within the controlled area. Visitors also wear an I.D. badge. Film badges are processed monthly. Internal exposure is controlled by a personnel bioassay program described in Section 12.12 that includes urinalysis and in-vivo counting.

Frequency of measurement, action levels and actions to be taken are given in Chapter 3 for personnel radiation protection and in Chapter 5 for environmental effluent control.

Procedures to be followed in case of a criticality accident are described in the Emergency Procedure Manual.

### 12.9 Respiratory Protection

Respiratory protective equipment includes full-face respirators, half-mask respirators, and supplied-air breathing apparatus.

The respirator fitting program satisfies the guidance of Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection", and NUREG-0041, "Manual of Respiratory Protection Against Airborne Radioactive Materials".

### 12.10 Occupational Exposure Analysis

Due to the low levels of penetrating radiation which exist in the plant, the greatest emphasis in exposure control has been directed towards minimizing inhalation of airborne uranium particulates. To this end, C-E has maintained airborne exposures as low as reasonable achievable through the use of ventilated hoods, process containment and an extensive breathing zone (BZ) air sampling program. Fixed air samplers are strategically placed throughout the facility to provide indications of general airborne activity levels. Continuous air monitors with alarms are utilized in the Oxide Building and in both pellet buildings to more rapidly detect an increase in the airborne activity level.

Information regarding internal deposition of radioactive materials is provided by a bioassay program which includes periodic urinalysis and in-vivo counting.

#### 12.10.1 External Radiation Exposures

The exposure to radiation from external sources is measured using film badges. The film badges are changed monthly. Results of monitoring for 1986, 1987, and 1988 were as follows:

| Annual Dose Ranges gamma (REM) | % of Personnel in Range 1986 | 1987 | 1988 |
|---|---|---|---|
| No measurable exposure | 20 | 11 | 21 |
| Less than 0.100 | 47 | 55 | 35 |
| 0.100 - 0.250 | 23 | 22 | 30 |
| 0.250 - 0.500 | 9 | 11 | 12 |
| Greater than 0.500 | <1 | <1 | 2 |
| Number of employees Monitored | 65 | 80 | 98 |

12.10.2 Internal Radiation Exposures

Air concentration levels are measured using Breathing Zone (BZ) monitors. Fixed Work Station Air samples are used to calculate air concentration levels when BZ sampling data is not available. The quarterly average air concentrations for reporting years 1986, 1987 and 1988 were as follows:

| Quarterly Exposure Range MPC hrs/% of Limit | Percent of Operators in Range | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | 1986 | | | | 1987 | | | | 1988 | | | |
| | 1st | 2nd | 3rd | 4th | 1st | 2nd | 3rd | 4th | 1st | 2nd | 3rd | 4th |
| **Oxide Conversion** | | | | | | | | | | | | |
| 0-52/0%-10% | 0 | 0 | 0 | 0 | 0 | 0 | 8 | 13 | 12 | 0 | 0 | 0 |
| 52-130/10%-25% | 33 | 67 | 22 | 11 | 8 | 23 | 50 | 0 | 12 | 0 | 40 | 62 |
| 130-260/25%-50% | 67 | 33 | 78 | 89 | 92 | 77 | 42 | 75 | 76 | 100 | 60 | 38 |
| 260-520/50%-100% | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 12 | 0 | 0 | 0 | 0 |
| **Pellet Fabrication** | | | | | | | | | | | | |
| 0-52/0%-10% | 25 | 0 | 0 | 0 | 0 | 0 | 100 | 33 | 21 | 0 | 11 | 0 |
| 52-130/10%-25% | 25 | 33 | 33 | 33 | 0 | 0 | 0 | 25 | 36 | 33 | 33 | 60 |
| 130-260/25%-50% | 50 | 67 | 67 | 67 | 0 | 0 | 0 | 42 | 43 | 67 | 56 | 40 |
| 260-520/50%-100% | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| **Scrap Recycle** | | | | | | | | | | | | |
| 0-52/0%-10% | 0 | 0 | 0 | 0 | 0 | 0 | 20 | 17 | 50 | 17 | 40 | 33 |
| 52-130/10%-25% | 50 | 50 | 17 | 50 | 40 | 40 | 40 | 50 | 17 | 67 | 60 | 67 |
| 130-260/25%-50% | 33 | 50 | 83 | 50 | 60 | 60 | 40 | 33 | 33 | 16 | 0 | 0 |
| 260-520/50%-100% | 17 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| **Material Handling** | | | | | | | | | | | | |
| 0-52/0%-10% | 17 | 100 | 100 | 50 | 100 | 100 | 86 | 57 | 50 | 100 | 100 | 100 |
| 52-130/10%-25% | 83 | 0 | 0 | 50 | 0 | 0 | 14 | 29 | 50 | 0 | 0 | 0 |
| 130-260/25%-50% | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 14 | 0 | 0 | 0 | 0 |
| 260-520/50%-100% | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| **Maintenance** | | | | | | | | | | | | |
| 0-52/0%-10% | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 14 | 0 |
| 52-130/10%-25% | 40 | 80 | 80 | 80 | 60 | 80 | 80 | 29 | 43 | 33 | 13 | 100 |
| 130-260/25%-50% | 60 | 20 | 20 | 20 | 40 | 20 | 20 | 57 | 57 | 67 | 43 | 0 |
| 260-520/50%-100% | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 14 | 0 | 0 | 0 | 0 |

The maximum quarterly exposure for 1986 was 294 MPC hrs or 57% of the allowable limits set forth in 10 CFR 20.103. For 1987 the maximum quarterly exposure was 277 MPC hrs or 53% of the quarterly limit. For 1988 the maximum quarterly exposure was 246 MPC hrs or 47% of the limit.

In-vivo counting is performed by an outside contractor using gamma scintillation spectrometry. Results for 1986, 1987 and 1988 were:

| Range ($\mu$g U-235) | Percent of Operators in Range April 1986 | Sept. 1986 | April 1987 | Oct. 1987 | March 1988 | Sept. 1988 | Nov. 1988 |
|---|---|---|---|---|---|---|---|
| Less than 50 | 49 | 33 | 40 | 49 | 64 | 50 | 53 |
| 50-100 | 49 | 66 | 26 | 50 | 28 | 48 | 34 |
| 100-125 | 0 | 0 | 20 | 0 | 7 | 0 | 11 |
| 125-240 | 0 | 0 | 13 | 0 | 0 | 0 | 0 |
| Greater than 240 | 0 | 0 | 0 | 0 | 0 | 0 | 0 |
| Total Employees Counted During the Period | 17 | 18 | 16 | 18 | 30 | 40 | 39 |

Operators are placed on partial restriction status when they have 1/2 of a maximum permissible lung burden. At the end of 1988, no operators were on restricted status. When C-E acquired the Hematite facility in 1974, 24% of the operators were on partial restriction.

Urinalysis is conducted for all operators on at least a monthly schedule. Results for 1986, 1987 and 1988 were as follows:

| Operation | Average Concentration ($\mu$gU/liter) 1986 Range of conc. | 1986 Number of operators | 1987 Range of conc. | 1987 Number of operators | 1988 Range of conc. | 1988 Number of operators |
|---|---|---|---|---|---|---|
| Maintenance | <2 | 5 | <2 | 5 | <2 | 7 |
| Oxide Conversion | <2 | 8 | <2 | 11 | <2 | 8 |
| Pellet Fabrication | <2 | 4 | <2 | 9 | <2 | 24 |
| Material Handling | <2 | 5 | <2 | 5 | <2 | 5 |
| Scrap Recycle | <2 | 7 | <2 | 6 | <2 | 7 |

### 12.11 Measures Taken to Implement ALARA

C-E has made numerous changes to equipment and procedures to improve containment and to reduce airborne exposures. This is a continuous process of evaluation and change, and reflects the C-E operating philosophy to keep occupational radiation exposures as low as reasonably achievable.

## 12.12 Bioassay Program

The bioassay program is described in Section 12.10.2 on internal radiation exposures.

## 12.13 Air Sampling

The air sampling program for monitoring concentrations of radioactivity in working areas is described in Section 3.2.3 where the breathing zone air sampling program is described and in Section 12.10 where results are given and evaluated.

## 12.14 Surface Contamination

Control of surface contamination is described in Section 3.2.6.

# CHAPTER 13 ENVIRONMENTAL SAFETY - RADIOLOGICAL

## 13.1 Airborne Releases

During routine plant operations, gaseous effluents containing insoluble uranium radionuclides are the primary releases that could radiologically affect humans and the surrounding environment. Maximum individual dose commitments were calculated based on measured releases from all exhaust stacks. Dose commitments for 1987 and 1988 were:

| | LUNG DOSE (mrem/yr) | |
|---|---|---|
| | North Onsite Monitoring Station (100 m) | Nearest Low Population Zone Resident (800 m) |
| 1987 | 9.5 | 0.10 |
| 1988 | 11.6 | 0.13 |

The critical organ for routine insoluble releases is the lung. As shown above, the nearest low population zone resident received a maximum lung dose commitment of 0.13 mrem in 1988. This is less than 1% of the 40 CFR 90 limit of 25 mrem/year.

## 13.2 Liquid Releases

Liquids containing trace quantities of uranium are discharged from the plant storm sewer and the sewage treatment plant. Average concentrations of the site dam overflow were less than 0.25% of 10 CFR 20 limits for 1987 and 1988. Further diluted in Joachim Creek, these levels would result in an insignificant dose commitment to an individual downstream of the plant.

## 13.3 Non-Radiological Releases

The only release of non-radiological materials of environmental concern is hydrogen fluoride (HF), which is released as an offgas of the $UF_6$ to $UO_2$ conversion process. HF releases for 1987 were $18.1 \times 10^3$ pounds and for 1988 were $21.0 \times 10^3$ pounds.

These releases would indicate a ground level concentration of less than 5.2 $\mu g/m^3$ at 100 meters and less than 1.0 $\mu g/m^3$ at the nearest low population zone residence. Damage to vegetation is unlikely at these concentrations.

## 13.4 Environmental Monitoring Summary

Environmental monitoring for the 1982-1988 period is summarized in the tables on the following pages.

TABLE 13-1

STACK MONITORING - RADIOACTIVITY[1]
(MICROCURIES RELEASED)

| MONTH\YEAR | 1982 | 1983 | 1984 | 1985 | 1986 | 1987 | 1988 |
|---|---|---|---|---|---|---|---|
| JAN | 18.2 | 27.4 | 11.1 | 13.6 | 9.2 | 27.2 | 31.3 |
| FEB | 12.3 | 22.7 | 9.1 | 9.2 | 5.6 | 26.5 | 39.2 |
| [illegible] | 18.7 | 29.1 | 8.9 | 7.3 | 16.6 | 15.0 | 25.8 |
| APR | 10.3 | 19.9 | 7.8 | 8.6 | 19.0 | 20.7 | 35.2 |
| MAY | 42.5 | 22.7 | 13.4 | 8.2 | 6.9 | 16.8 | 42.1 |
| JUN | 24.6 | 31.7 | 2.0 | 12.7 | 8.8 | 20.8 | 28.5 |
| JUL | 20.2 | 4.7 | 3.7 | 5.9 | 10.4 | 15.9 | 30.8 |
| AUG | 14.1 | 16.2 | 5.9 | 8.7 | 10.2 | 20.9 | 18.3 |
| SEP | 48.0 | 11.6 | 2.2 | 9.5 | 5.7 | 21.9 | 16.0 |
| OCT | 23.1 | 5.1 | 6.0 | 12.3 | 33.9 | 46.[illegible] | 16.5 |
| NOV | 46.9 | 11.9 | 10.7 | 9.3 | 15.2 | 30.7 | 38.5 |
| DEC | 15.2 | 4.7 | 8.2 | 21.6 | 17.3 | 14.5 | 25.3 |
| TOTALS | 294.1 | 207.7 | 89.0 | 126.9 | 158.8 | 277.2 | 347.5 |
| SITE BOUNDARY CONC.[2] ($10^{-15}$ µCi/ml) | 0.9 | 0.6 | 0.3 | 0.4 | 0.5 | 0.8 | 1.1 |
| % MPC | 0.022 | 0.016 | 0.007 | 0.010 | 0.012 | 0.021 | 0.026 |

(1) Determined by gross alpha counting after allowing at least 8 hours for decay of radon daughters.
(2) Calculated.

TABLE 13.2

ENVIRONMENTAL AIR MONITORING - RADIOACTIVITY(1)
($10^{-15}$ MICROCURIES PER MILILITER)

OFFSITE EAST

| MONTH\YEAR | 1982 | 1983 | 1984 | 1985 | 1986 | 1987 | 1988 |
|---|---|---|---|---|---|---|---|
| JAN | 4 | 7 | 27 | 10 | 17 | 5 | 2 |
| FEB | 7 | 2 | 7 | 2 | 27 | 4 | 3 |
| MAR | 8 | 2 | 7 | 23 | 6 | 2 | 4 |
| APR | 4 | 3 | 7 | 10 | 11 | 4 | 3 |
| MAY | 3 | 6 | 6 | 25 | 7 | 4 | 3 |
| JUN | 3 | 3 | 4 | 13 | 6 | 2 | 3 |
| JUL | 6 | 4 | 6 | 14 | 9 | 4 | 5 |
| AUG | 4 | 4 | 16 | 6 | 9 | 5 | 3 |
| SEP | 10 | 5 | 5 | 7 | 6 | 3 | 4 |
| OCT | 4 | 2 | 11 | 10 | 2 | 5 | 2 |
| NOV | 6 | 4 | 12 | 8 | 3 | 4 | 3 |
| DEC | * | 2 | 11 | 15 | 4 | 3 | 5 |
| AVERAGE CONCENTRATION | 5.4 | 3.7 | 9.9 | 11.9 | 8.9 | 3.8 | 3.3 |

* NO SAMPLE - PUMP FAILURE

OFFSITE WEST

| MONTH\YEAR | 1982 | 1983 | 1984 | 1985 | 1986 | 1987 | 1988 |
|---|---|---|---|---|---|---|---|
| JAN | 2 | 7 | 15 | 13 | 31 | 6 | 2 |
| FEB | 7 | 7 | 7 | 5 | 37 | 4 | 3 |
| MAR | 5 | 3 | 6 | 26 | 11 | 3 | 3 |
| APR | 6 | 2 | 7 | 5 | 13 | 3 | 5 |
| MAY | 4 | 3 | 8 | 21 | 10 | 5 | 3 |
| JUN | 4 | 2 | 5 | 6 | 18 | 2 | 3 |
| JUL | 5 | 5 | 7 | 5 | 26 | 2 | 7 |
| AUG | 5 | 9 | 25 | 31 | 28 | 3 | 12 |
| SEP | 11 | 4 | 2 | 20 | 17 | 2 | 5 |
| OCT | 5 | 3 | 9 | 9 | 4 | 4 | 3 |
| NOV | 4 | 3 | 20 | 11 | 4 | 4 | 2 |
| DEC | 5 | 3 | 10 | 49 | 5 | 4 | 3 |
| AVERAGE CONCENTRATION | 5.3 | 4.3 | 4.3 | 16.8 | 17.0 | 3.5 | 4.3 |

(1) Determined by gross alpha counting after allowing a 72 hour period for decay of radon and thorium daughters.

TABLE 13-3

SITE DAM OVERFLOW MONITORING - RADIOACTIVITY
(PICOCURIES PER LITER)

| | 1982 | | 1983 | | 1984 | | 1985 | | 1986 | | 1987 | | 1988 | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| MONTH\YEAR | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA |
| JAN | 4 | 14 | 3 | 6 | 9 | 4 | 4 | 7 | 29 | 28 | 10 | 5 | 5 | 9 |
| FEB | 4 | 4 | 5 | 7 | 28 | 17 | 7 | 23 | 65 | 88 | 12 | 14 | 14 | 15 |
| MAR | 2 | 3 | 4 | 9 | 7 | 15 | 5 | 10 | 22 | 18 | 5 | 4 | 31 | 14 |
| APR | 2 | 7 | 3 | 15 | 12 | 12 | 4 | 47 | 22 | 16 | 13 | 11 | 15 | 14 |
| MAY | 7 | 8 | 2 | 6 | 16 | 21 | 23 | 21 | 37 | 40 | 172 | 108 | 34 | 28 |
| JUN | 2 | 3 | 11 | 12 | 21 | 79 | 7 | 16 | | | 83 | 52 | 122 | 81 |
| JUL | 7 | 7 | 27 | 12 | 48 | 18 | 29 | 32 | 115 | 160 | 18 | 38 | 94 | 55 |
| AUG | 16 | 13 | 21 | 12 | 61 | 41 | 29 | 28 | 40 | 166 | 36 | 20 | 134 | 53 |
| SEP | 12 | 10 | 32 | 30 | 31 | 151 | 15 | 10 | 80 | 163 | 69 | 46 | 132 | 31 |
| OCT | 13 | 20 | 24 | 15 | 15 | 28 | 81 | 28 | 8 | 7 | 285 | 150 | 65 | 33 |
| NOV | 47 | 25 | 26 | 49 | 11 | 21 | 37 | 33 | 10 | 12 | 153 | 48 | 59 | 32 |
| DEC | 3 | 6 | 4 | 12 | 15 | 41 | 6 | 5 | 10 | 7 | 21 | 34 | 11 | 11 |
| AVERAGE CONCENTRATION | 10 | 10 | 14 | 15 | 23 | 37 | 21 | 22 | 37 | 64 | 73 | 44 | 60 | 31 |
| %MPC | 0.25 | 0.25 | 0.34 | 0.39 | 0.57 | 0.93 | 0.51 | 0.54 | 0.91 | 1.60 | 1.83 | 1.10 | 1.49 | 0.78 |

TABLE 13-4

JOACHIM CREEK MONITORING - RADIOACTIVITY

(PICOCURIES PER LITER) UPSTREAM

| MONTH\YEAR | 1982 | | 1983 | | 1984 | | 1985 | | 1986 | | 1987 | | 1988 | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA |
| JAN | <2.0 | <3.0 | 1.5 | 3.9 | <1.0 | 3.0 | <1.0 | <3.0 | <1.0 | 3.6 | <3.0 | 3.7 | <4.0 | 2.8 |
| FEB | <2.0 | <3.0 | <5.0 | <2.0 | <3.0 | <2.0 | <1.0 | 22.0 | <2.0 | 3.7 | <5.0 | 3.9 | <3.0 | 2.5 |
| MAR | <2.0 | <3.0 | <3.0 | 2.9 | <2.0 | 5.4 | <3.0 | 2.6 | <2.0 | 2.9 | <4.0 | 2.3 | <4.0 | 3.4 |
| APR | <2.0 | 4.0 | 0.7 | 8.0 | <2.0 | 3.3 | <1.0 | 6.7 | <2.0 | 3.8 | <5.0 | 6.4 | <3.0 | 4.2 |
| MAY | <2.0 | <3.0 | <2.0 | 5.0 | <2.0 | 4.7 | <1.0 | 3.3 | <2.0 | 2.5 | <4.0 | 3.5 | <3.0 | 3.1 |
| JUN | <2.0 | 4.0 | <1.0 | 4.2 | 1.7 | 4.6 | <4.0 | 4.6 | <3.0 | 3.0 | <4.0 | 3.7 | <3.0 | 5.7 |
| JUL | <2.0 | 132.0 | <2.0 | <3.0 | <2.0 | 9.8 | <1.0 | 3.0 | <2.0 | 2.1 | <4.0 | 3.1 | <2.0 | 4.5 |
| AUG | <2.0 | <3.0 | <2.0 | 3.9 | <2.0 | 9.8 | <1.0 | 2.7 | <2.0 | 3.8 | <3.0 | 4.0 | <3.0 | 5.6 |
| SEP | 3.0 | <3.0 | <3.0 | 3.8 | <2.0 | 8.4 | <1.0 | <3.0 | <3.0 | 3.2 | <3.0 | 4.2 | <3.0 | 4.8 |
| OCT | <2.0 | <3.0 | <1.0 | 3.3 | <1.0 | <1.0 | <2.0 | <3.0 | <4.0 | 6.2 | <5.0 | 5.1 | <4.0 | 5.8 |
| NOV | 9.0 | 3.0 | <1.0 | 3.3 | <1.0 | 3.6 | <1.0 | 1.9 | <2.0 | 12.0 | <4.0 | 3.8 | <3.0 | 4.1 |
| DEC | <2.0 | 4.0 | <1.0 | 3.1 | <1.0 | 2.4 | <1.0 | 2.0 | <5.0 | 9.3 | <2.0 | 5.2 | <3.0 | 5.0 |

TABLE 13-5

JOACHIM CREEK MONITORING - RADIOACTIVITY

(PICOCURIES PER LITER) DOWNSTREAM

| | 1982 | | 1983 | | 1984 | | 1985 | | 1986 | | 1987 | | 1988 | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| MONTH\YEAR | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | .PHA | BETA | ALPHA | BETA |
| JAN | <2.0 | <3.0 | <1.0 | 4.3 | <1.0 | 3.1 | <2.0 | <3.0 | <1.0 | 3.1 | <3.0 | 4.0 | <4.0 | 2.3 |
| FEB | <2.0 | 3.0 | <5.0 | 2.6 | <1.0 | 2.4 | <1.0 | 4.2 | <2.0 | 3.6 | <4.0 | 4.2 | <4.0 | 3.4 |
| MAR | <2.0 | <3.0 | <3.0 | 4.4 | <2.0 | 4.6 | <3.0 | 2.6 | <2.0 | 5.2 | <4.0 | 3.5 | <4.0 | 4.3 |
| APR | <2.0 | 10.0 | <2.0 | 2.0 | <1.0 | 3.1 | <0.8 | 2.9 | <2.0 | 3.5 | <5.0 | 3.3 | <3.0 | 3.2 |
| MAY | <2.0 | 3.0 | <1.0 | 2.9 | <2.0 | 4.8 | <1.0 | 2.7 | <2.0 | 3.3 | <4.0 | 3.0 | 4.5 | 33.0 |
| JUN | <2.0 | <3.0 | <1.0 | 4.2 | <2.0 | 5.1 | <4.0 | 8.6 | <3.0 | <3.0 | <4.0 | 4.1 | <3.0 | 2.9 |
| JUL | <2.0 | <3.0 | <2.0 | <3.0 | <2.0 | 9.0 | <1.0 | 3.3 | <2.0 | 4.8 | <4.0 | 3.5 | <2.0 | 3.2 |
| AUG | <2.0 | 6.0 | <2.0 | 3.9 | <2.0 | 4.6 | <1.6 | 3.5 | <2.0 | 2.7 | <3.0 | 4.9 | 4.5 | 2.9 |
| SEP | 3.0 | 4.0 | <3.0 | 4.7 | <2.0 | 4.6 | <1.0 | <3.0 | <3.0 | 2.9 | <3.0 | 3.7 | <3.0 | 7.2 |
| OCT | <2.0 | <3.0 | <1.0 | 3.9 | 4.9 | 5.4 | <2.0 | 4.0 | <3.0 | 3.8 | <5.0 | 5.8 | <4.0 | 4.9 |
| NOV | 9.0 | 5.0 | <1.0 | 3.9 | <1.0 | 3.3 | <1.0 | 2.0 | 7.8 | 18.0 | <4.0 | 5.7 | <3.0 | 4.6 |
| DEC | <2.0 | 4.0 | <1.0 | 3.0 | <1.0 | 2.4 | <1.0 | 2.1 | <4.0 | 5.1 | <4.0 | 4.0 | <3.0 | 4.5 |

TABLE 13-6

QUARTERLY LIQUID ENVIRONMENTAL MONITORING - RADIOACTIVITY

(PICOCURIES PER LITER)

JOACHIM CREEK/SITE CREEK CONFLUENCE

| CALENDAR QTR\YEAR | 1982 | | 1983 | | 1984 | | 1985 | | 1986 | | 1987 | | 1988 | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA |
| 1ST QTR | <2.0 | <3.0 | <2.0 | 3.0 | 2.5 | 12.0 | <2.0 | 6.0 | <2.0 | 4.8 | <2.0 | <3.0 | <2.0 | 11.0 |
| 2ND QTR | 10.0 | <3.0 | <2.0 | <3.0 | 1.8 | 4.8 | <4.0 | 9.0 | <2.0 | 3.5 | <2.0 | <3.0 | <2.0 | <3.0 |
| 3RD QTR | <2.0 | <3.0 | <2.0 | <3.0 | <2.0 | 6.1 | <2.0 | <3.0 | <2.0 | <3.0 | <2.0 | <3.0 | <2.0 | 3.4 |
| 4TH QTR | <2.0 | <3.0 | 2.2 | 2.0 | <1.0 | 12.0 | 1.9 | 4.7 | <2.0 | 5.0 | <2.0 | 8.0 | <2.0 | <3.0 |

HEMATITE WELL

| CALENDAR QTR\YEAR | 1982 | | 1983 | | 1984 | | 1985 | | 1986 | | 1987 | | 1988 | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA |
| 1ST QTR | <2.0 | <3.0 | <2.0 | <3.0 | <2.0 | 5.1 | <2.0 | <3.0 | <2.0 | 2.9 | <2.0 | <3.0 | <2.0 | 14.0 |
| 2ND QTR | <2.0 | 5.0 | 5.0 | <3.0 | <2.0 | 3.1 | <2.0 | 5.9 | <2.0 | <2.0 | <2.0 | <3.0 | <2.0 | <3.0 |
| 3RD QTR | 5.0 | <3.0 | <2.0 | 6.0 | <3.0 | 12.0 | <1.0 | <3.0 | <2.0 | <3.0 | 5.0 | <3.0 | <3.0 | 9.7 |
| 4TH QTR | 3.0 | <3.0 | <1.0 | 2.8 | 3.2 | 4.0 | <2.0 | 5.0 | <2.0 | <3.0 | <2.0 | <3.0 | -- | -- |

TABLE 13-7

RETENTION POND WELL MONITORING - RADIOACTIVITY
SAMPLE WELL NORTH
(PICOCURIES PER LITER)

| | 1982 | | 1983 | | 1984 | | 1985 | | 1986 | | 1987 | | 1988 | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| MONTH\YEAR | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA |
| JAN | 7.0 | 342.0 | 10.0 | 337.0 | 45.0 | 273.0 | 4.0 | 137.0 | <2.0 | 229.0 | [illegible] | 159.0 | 26.0 | 331.0 |
| FEB | <2.0 | 268.0 | 24.0 | 291.0 | 34.0 | 245.0 | 4.0 | 157.0 | 4.0 | [illegible] | 61.0 | [illegible] | 10.0 | 22.0 |
| MAR | <2.0 | 224.0 | 13.0 | 268.0 | 47.0 | 230.0 | 6.0 | 72.0 | 5.0 | [illegible] | 60.0 | 241.0 | 15.0 | 44.0 |
| APR | 13.0 | 448.0 | 6.0 | 79.0 | 17.0 | 160.0 | 7.0 | 66.6 | <2.0 | [illegible] | 40.0 | 243.0 | 12.0 | 393.0 |
| MAY | 30.0 | 21.0 | 2.0 | 300.0 | 16.0 | 304.0 | 4.0 | 327.0 | 7.0 | 288.0 | <2.0 | 329.0 | <2.0 | 423.0 |
| JUN | 5.0 | 8.0 | 10.0 | 296.0 | 21.0 | 305.0 | 5.0 | 325.0 | <2.0 | 277.0 | 26.0 | 370.0 | <2.0 | 520.0 |
| JUL | 29.0 | 388.0 | 36.0 | 265.0 | 23.0 | 284.0 | <2.0 | 238.0 | 6.0 | 413.0 | 6.0 | 539.0 | <2.0 | 547.0 |
| AUG | 5.0 | 143.0 | 27.0 | 158.0 | 34.0 | 271.0 | 4.0 | 166.0 | 8.0 | 131.0 | 263.0 | 617.0 | <4.0 | 238.0 |
| SEP | <2.0 | 90.0 | 18.0 | 106.0 | 4.0 | 139.0 | 5.0 | 142.0 | 19.0 | 338.0 | <2.0 | 642.0 | 10.0 | 190.0 |
| OCT | 5.0 | 205.0 | 12.0 | 89.0 | 63.0 | 91.0 | 3.0 | 127.0 | 5.0 | 85.0 | <2.0 | 729.0 | 38.0 | 596.0 |
| NOV | 14.0 | 301.0 | 16.0 | 150.0 | 8.0 | 80.0 | 6.0 | 223.0 | 4.0 | 114.0 | 62.0 | 334.0 | <2.0 | 248.0 |
| DEC | 13.0 | 273.0 | 9.0 | 40.0 | 5.0 | 101.0 | 3.0 | 173.0 | 12.0 | 290.0 | (55/<2) | (442/310) | 4.0 | 266.0 |

( ) Consecutive samples taken the same time
First result from C.E.P.
Second result from Teledyne

TABLE 13-8

RETENTION POND WELL MONITORING - RADIOACTIVITY
SAMPLE WELL SOUTH EAST
(PICOCURIES PER LITER)

| MONTH\YEAR | 1982 | | 1983 | | 1984 | | 1985 | | 1986 | | 1987 | | 1988 | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA |
| JAN | <2.0 | <3.0 | <2.0 | <3.0 | <2.0 | <3.0 | <2.0 | <3.0 | 95.0 | 55.0 | 28.0 | 18.0 | <2.0 | <3.0 |
| FEB | <2.0 | <3.0 | 4.0 | 10.0 | <2.0 | <3.0 | <2.0 | 4.0 | <2.0 | 4.0 | 19.0 | 17.0 | 8.0 | 25.0 |
| MAR | <2.0 | <3.0 | <2.0 | 4.0 | <2.0 | <3.0 | <2.0 | <3.0 | 4.0 | 5.0 | 7.0 | 5.0 | 11.0 | <3.0 |
| APR | <2.0 | 12.0 | <2.0 | 4.0 | <2.0 | <3.0 | <2.0 | <3.0 | 4.0 | 5.0 | <2.0 | <3.0 | <2.0 | <3.0 |
| MAY | 17.0 | 13.0 | *** | *** | <2.0 | <3.0 | <2.0 | <3.0 | <2.0 | 4.0 | <2.0 | <3.0 | 252.0 | 132.0 |
| JUN | <2.0 | <3.0 | <2.0 | <3.0 | 5.0 | 5.0 | 3.0 | <3.0 | 5.0 | 6.0 | 65.0 | 55.0 | * | * |
| JUL | <2.0 | <3.0 | <2.0 | <3.0 | <2.0 | <3.0 | <2.0 | <3.0 | <2.0 | 11.0 | 34.0 | 22.0 | * | * |
| AUG | 4.0 | 6.0 | <2.0 | <3.0 | 3.0 | 3.0 | 4.0 | 6.0 | 4.0 | 4.0 | 14.0 | 21.0 | * | * |
| SEP | <2.0 | <3.0 | <2.0 | <3.0 | 4.0 | 14.0 | <2.0 | <3.0 | 7.0 | 8.0 | * | * | * | * |
| OCT | <2.0 | <3.0 | <2.0 | <3.0 | <2.0 | 4.0 | <2.0 | 7.0 | 45.0 | 17.0 | * | * | * | * |
| NOV | <2.0 | 4.0 | <2.0 | <3.0 | <2.0 | 4.0 | 92.0 | 16.0 | 16.0 | 10.0 | * | * | * | * |
| DEC | <2.0 | <3.0 | <9.2 | 9.3 | <2.0 | <3.0 | 33.0 | 10.0 | 10.0 | 7.0 | (<2/37) | (<3/35) | 15.0 | 12.0 |

* Well dry at this time

( ) Consecutive samples taken the same time
First result from C.E.P.
Second result from Teledyne

TABLE 13-9

RETENTION POND WELL MONITORING - RADIOACTIVITY
SAMPLE WELL SOUTH WEST
(PICOCURIES PER LITER)

| MONTH\YEAR | 1982 ALPHA | 1982 BETA | 1983 ALPHA | 1983 BETA | 1984 ALPHA | 1984 BETA | 1985 ALPHA | 1985 BETA | 1986 ALPHA | 1986 BETA | 1987 ALPHA | 1987 BETA | 1988 ALPHA | 1988 BETA |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| JAN | 16.0 | 13.0 | <2.0 | <3.0 | 7.0 | <3.0 | 4.0 | 4.0 | 19.0 | 21.0 | 27.0 | 31.0 | <2.0 | <3.0 |
| FEB | <2.0 | 6.0 | 12.0 | 13.0 | 5.0 | 3.0 | 4.0 | 4.0 | 5.0 | <3.0 | 16.0 | 24.0 | 27.0 | 423.0 |
| MAR | <2.0 | 3.0 | <2.0 | 5.0 | 8.0 | 4.0 | <2.0 | 3.0 | 10.0 | 13.0 | 17.0 | 21.0 | <2.0 | <3.0 |
| APR | <2.0 | 8.0 | <2.0 | 3.0 | <2.0 | <3.0 | <2.0 | 5.0 | 5.0 | 10.0 | <2.0 | <3.0 | <2.0 | 29.0 |
| MAY | 24.0 | 17.0 | *** | *** | 10.0 | 9.0 | 5.0 | 8.0 | 16.0 | 10.0 | 17.0 | 20.0 | <2.0 | <3.0 |
| JUN | 6.0 | 7.0 | 3.0 | <3.0 | 11.0 | 3.0 | 12.0 | 28.0 | 10.0 | 15.0 | 49.0 | 62.0 | * | * |
| JUL | 13.0 | 13.0 | *** | *** | 10.0 | 6.0 | 10.0 | 6.0 | 11.0 | 14.0 | 12.0 | 8.0 | * | * |
| AUG | 24.0 | 11.0 | 151.0 | 38.0 | 10.0 | 11.0 | 13.0 | 15.0 | 12.0 | 11.0 | <2.0 | <3.0 | 9.0 | 13.0 |
| SEP | 17.0 | 21.0 | 10.0 | 8.0 | 8.0 | 22.0 | 1[illegible].0 | 8.0 | <2.0 | 7.0 | 13.0 | 21.0 | 30.0 | 17.0 |
| OCT | 7.0 | <3.0 | 6.0 | 7.0 | 11.0 | 8.0 | 4.0 | 5.0 | <2.0 | <3.0 | <2.0 | 14.0 | 20.0 | 97.0 |
| NOV | 4.0 | 12.0 | <2.0 | 10.0 | <2.0 | 3.0 | <2.0 | 4.0 | <2.0 | 4.0 | 32.0 | 23.0 | <2.0 | 5.0 |
| DEC | 2.0 | <3.0 | 9.2 | 13.0 | <2.0 | 5.0 | 6.0 | 7.0 | 11.0 | 13.0 | (<2/3.2) | (<3/5.7) | 8.0 | 9.0 |

* Well dry at this time

*** Samples lost by Contractor

( ) Consecutive samples taken the same time
First result from C.E.P.
Second result from Teledyne

TABLE 13-10

SITE WATER SUPPLY WELL MONITORING - RADIOACTIVITY
(PICOCURIES PER LITER)

| MONTH\YEAR | 1982 | | 1983 | | 1984 | | 1985 | | 1986 | | 1987 | | 1988 | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA |
| JAN | 7.0 | <3.0 | <1.0 | <3.0 | <1.0 | 1.6 | <2.0 | <3.0 | 2.7 | 3.1 | <2.0 | <3.0 | <4.0 | <2.0 |
| FEB | 45.0 | 7.0 | <4.0 | <2.0 | <3.0 | 3.8 | <2.0 | 4.8 | <2.0 | 2.6 | <4.0 | <2.0 | <4.0 | <2.0 |
| MAR | * | 3.0 | <3.0 | <3.0 | <2.0 | 3.4 | <3.0 | <2.0 | <2.0 | 2.1 | <4.0 | <7.0 | <4.0 | 3.5 |
| APR | <2.0 | <3.0 | <2.0 | <2.0 | 1.8 | 4.8 | <2.0 | 3.0 | <2.0 | 1.6 | <4.0 | <3.0 | <3.0 | 3.0 |
| MAY | <2.0 | <3.0 | <1.0 | <3.0 | <2.0 | 2.8 | 1.2 | 1.6 | <2.0 | 1.6 | <4.0 | <2.0 | <2.0 | 2.2 |
| JUN | <2.0 | <3.0 | <1.0 | <2.0 | 2.4 | 4.6 | <4.0 | 11.0 | <2.0 | <2.0 | <3.0 | 2.5 | <2.0 | 2.8 |
| JUL | <2.0 | <3.0 | <2.0 | <2.0 | <2.0 | 7.2 | <1.0 | <2.0 | <2.0 | <2.0 | <4.0 | 2.4 | <2.0 | <2.0 |
| AUG | <2.0 | <3.0 | <2.0 | <2.0 | <2.0 | 4.3 | 5.3 | <2.0 | <2.0 | 1.8 | <3.0 | <2.0 | <3.0 | <3.0 |
| SEP | <2.0 | <3.0 | <3.0 | 3.3 | <2.0 | 3.5 | <2.0 | 5.0 | <3.0 | 2.2 | <3.0 | 4.0 | <3.0 | <2.0 |
| OCT | <2.0 | <3.0 | <2.0 | <3.0 | 2.2 | 2.1 | <2.0 | <3.0 | <3.0 | 2.5 | <4.0 | 3.4 | <3.0 | <3.0 |
| NOV | <2.0 | <3.0 | <2.0 | <3.0 | 1.9 | 1.9 | <1.0 | <1.0 | <2.0 | <3.0 | <3.0 | <2.0 | <3.0 | <2.0 |
| DEC | <2.0 | <3.0 | <1.0 | 1.9 | <1.0 | 1.8 | <1.0 | <1.0 | <2.0 | <3.0 | <2.0 | 2.6 | <3.0 | 4.1 |

* Data not available

TABLE 13-11

SEWAGE OUTFALL MONITORING - RADIOACTIVITY
(PICOCURIES PER LITER)

| MONTH\YEAR | 1982 | | 1983 | | 1934 | | 1985 | | 1986 | | 1987 | | 1988 | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA |
| JAN | 17 | 35 | 72 | 77 | 126 | 76 | 244 | 148 | 155 | 183 | 24 | 21 | 133 | 94 |
| FEB | 74 | 86 | 170 | 148 | 97 | 42 | 39 | 84 | 134 | 131 | 127 | 163 | 69 | 118 |
| MAR | 89 | 36 | 16 | 55 | 73 | 76 | 54 | 143 | 32 | 28 | 207 | 80 | 43 | 35 |
| APR | 13 | 59 | 43 | 135 | 35 | 97 | 82 | 80 | 129 | 59 | 28 | 39 | 91 | 95 |
| MAY | 20 | 39 | 15 | 50 | 11 | 18 | 77 | 67 | 135 | 168 | 48 | 56 | 39 | 47 |
| JUN | 27 | 132 | 7 | 46 | 13 | 19 | 213 | 266 | * | * | 79 | 75 | 56 | 61 |
| JUL | 49 | 49 | 30 | 43 | 18 | 49 | 97 | 103 | 17 | 36 | 17 | 75 | 55 | 40 |
| AUG | 29 | 205 | 44 | 32 | 6 | 21 | 97 | 112 | 46 | 182 | 109 | 93 | 33 | 31 |
| SEP | 32 | 58 | 39 | 37 | 82 | 118 | 189 | 90 | 50 | 401 | 43 | 36 | 71 | 25 |
| OCT | 81 | 45 | 67 | 49 | 23 | 168 | 219 | 96 | 16 | 117 | 87 | 62 | 171 | 106 |
| NOV | 110 | 51 | 24 | 133 | 89 | 127 | 174 | 184 | 19 | 52 | 36 | 27 | 26 | 70 |
| DEC | 38 | 23 | 41 | 90 | 21 | 126 | 405 | 177 | 46 | 42 | 76 | 150 | 31 | 87 |

* These samples lost in shipping

TABLE 13-12

SOIL MONITORING - RADIOACTIVITY
(PICOCURIES PER GRAM)

SOIL STATION 12

| | 1982 | | 1983 | | 1984 | | 1985 | | 1986 | | 1987 | | 1988 | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| CALENDAR QTR\YEAR | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA |
| 1ST QTR | 2.3 | 1.2 | 6.5 | 33 | [illegible] | 32 | 8.6 | 27 | <5 | 17 | 11 | 40 | 12 | 28 |
| 2ND QTR | 3.3 | 2.5 | 15 | [illegible] | <5 | 36 | 9.7 | 25 | <7 | 28 | 19 | 29 | 12 | 36 |
| 3RD QTR | 5.8 | 3.5 | <3 | [illegible] | 73 | 35 | <4 | 21 | 14 | 38 | 11 | 28 | 6.7 | 16.6 |
| 4TH QTR | 3 | 3.7 | <5 | [illegible] | 35 | 29 | <6 | 29 | 18 | 34 | 25 | 51 | 25 | 41 |

SOIL STATION 13

| | 1982 | | 1983 | | 1984 | | 1985 | | 1986 | | 1987 | | 1988 | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| CALENDAR QTR\YEAR | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA |
| 1ST QTR | 0 | 1.1 | <6 | 26 | 5.6 | 39 | 5.6 | 55 | <5 | 30 | 5 | 36 | 18 | 36 |
| 2ND QTR | 0 | 0.7 | <7 | 32 | 6.3 | 33 | 8.1 | 22 | 10 | 27 | 25 | 32 | 6.6 | 33 |
| 3RD QTR | 5.4 | 2.9 | <3 | 35 | <6 | 27 | <4 | 33 | <7 | 31 | 15 | 30 | 7.4 | 19.4 |
| 4TH QTR | 2.2 | 2.9 | <5 | 32 | 8.1 | 31 | <5 | 32 | 18 | 35 | 12 | 30 | 8.2 | 39 |

SOIL STATION 14

| | 1982 | | 1983 | | 1984 | | 1985 | | 1986 | | 1987 | | 1988 | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| CALENDAR QTR\YEAR | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA |
| 1ST QTR | 4.6 | 1.9 | 8.7 | 43 | [illegible] | 58 | 15 | 35 | 11 | 27 | 16 | 20 | 16 | 40 |
| 2ND QTR | 1.8 | 1.8 | 11 | 40 | 17 | 46 | 62 | 47 | 20 | 53 | 26 | 30 | 17 | 43 |
| 3RD QTR | 5.6 | 3.8 | 15 | 41 | 2 | 44 | 13 | 31 | 8.2 | 35 | 12 | 48 | 8.2 | 25.2 |
| 4TH QTR | 3.2 | 4 | 7.8 | 54 | 18 | 50 | <6 | 30 | 19 | 48 | 22 | [illegible] | 16 | 44 |

SOIL STATION 15

| | 1982 | | 1983 | | 1984 | | 1985 | | 1986 | | 1987 | | 1988 | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| CALENDAR QTR\YEAR | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA | ALPHA | BETA |
| 1ST QTR | 36 | 2.9 | 8.7 | 36 | 17 | 6 | 9.2 | 38 | 9.4 | 35 | 17 | 38 | 46 | 50 |
| 2ND QTR | 2.8 | 2 | 20 | 43 | <5 | 35 | 19 | 35 | 14 | 31 | 15 | 41 | 11 | 40 |
| 3RD QTR | 7.5 | 3.9 | 15 | 46 | 9.2 | 38 | 7.6 | 31 | 19 | 33 | 12 | 43 | 24.4 | 34.1 |
| 4TH QTR | 2.6 | 3.3 | 5.6 | 40 | <6 | 32 | 11 | 36 | 33 | 55 | 18 | 44 | 10 | 47 |

TABLE 13-13

VEGETATION MONITORING - RADIOACTIVITY
(PICOCURIES PER GRAM)

SOIL STATION 12

| CALENDAR QTR\YEAR | 1982 ALPHA | 1982 BETA | 1983 ALPHA | 1983 BETA | 1984 ALPHA | 1984 BETA | 1985 ALPHA | 1985 BETA | 1986 ALPHA | 1986 BETA | 1987 ALPHA | 1987 BETA | 1988 ALPHA | 1988 BETA |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 1ST QTR | <0.1 | 6.4 | <0.2 | 7.7 | 1.1 | 15 | 0.4 | 17 | 1 | 18 | 1.1 | 16 | 0.76 | 13 |
| 2ND QTR | 0.1 | 3.1 | <1.0 | 11 | <0.2 | 8.1 | <0.2 | 2.6 | <0.6 | 16 | <0.2 | 11 | <0.3 | 13 |
| 3RD QTR | <0.3 | 2.2 | <0.2 | 7.2 | <0.1 | 7.5 | 0.61 | 6.7 | 0.54 | 11 | <0.2 | 5.5 | 1 | 12.8 |
| 4TH QTR | 5.4 | 46 | 1.1 | 6.6 | 0.91 | 12 | 1.6 | 1.7 | 0.6 | 7.8 | <0.6 | 17 | 0.75 | 12 |

SOIL STATION 13

| CALENDAR QTR\YEAR | 1982 ALPHA | 1982 BETA | 1983 ALPHA | 1983 BETA | 1984 ALPHA | 1984 BETA | 1985 ALPHA | 1985 BETA | 1986 ALPHA | 1986 BETA | 1987 ALPHA | 1987 BETA | 1988 ALPHA | 1988 BETA |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 1ST QTR | 2.3 | 4.2 | <0.2 | 6.7 | 22 | 3.9 | 0.46 | 17 | <0.2 | 12 | 1.1 | 14 | 2.4 | 13 |
| 2ND QTR | 2 | 3.3 | <0.2 | 14 | <0.2 | 10 | <0.2 | 17 | 0.38 | 13 | <0.4 | 12 | <0.2 | 12 |
| 3RD QTR | <0.3 | 3.1 | <0.2 | 7.3 | <0.1 | 11 | 0.38 | 5.3 | <0.6 | 14 | <0.4 | 17 | <0.4 | 15 |
| 4TH QTR | 6 | 49 | 5 | 26 | 0.3 | 6.5 | 0.7 | 12 | 0.39 | 7.2 | <0.5 | 10 | 0.38 | 9.1 |

SOIL STATION 14

| CALENDAR QTR\YEAR | 1982 ALPHA | 1982 BETA | 1983 ALPHA | 1983 BETA | 1984 ALPHA | 1984 BETA | 1985 ALPHA | 1985 BETA | 1986 ALPHA | 1986 BETA | 1987 ALPHA | 1987 BETA | 1988 ALPHA | 1988 BETA |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 1ST QTR | <0.1 | 0.3 | <0.2 | 8.5 | 2.1 | 16 | 0.38 | 12 | [illegible] | 13 | 0.9 | 11 | 6.7 | 22 |
| 2ND QTR | <0.1 | 2.4 | <0.2 | 10 | <0.2 | 10 | 0.76 | 9.8 | 1.5 | 55 | <0.3 | 11 | <0.1 | 12 |
| 3RD QTR | <0.3 | 3.3 | <0.2 | 7 | <0.1 | 12 | 0.93 | 16 | 0.53 | 10 | <0.2 | 11 | <0.3 | 9.9 |
| 4TH QTR | 3.1 | 43 | 0.78 | 11 | <0.3 | 6.8 | 11 | 18 | 0.78 | 12 | <0.6 | 18 | 0.64 | 13 |

SOIL STATION 15

| CALENDAR QTR\YEAR | 1982 ALPHA | 1982 BETA | 1983 ALPHA | 1983 BETA | 1984 ALPHA | 1984 BETA | 1985 ALPHA | 1985 BETA | 1986 ALPHA | 1986 BETA | 1987 ALPHA | 1987 BETA | 1988 ALPHA | 1988 BETA |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 1ST QTR | <0.1 | 3.9 | <0.3 | 19 | 30 | 12 | 0.18 | 11 | <0.2 | 8.7 | 0.56 | 14 | 9 | 23 |
| 2ND QTR | <0.1 | 2.6 | <2 | 11 | <0.2 | 7.8 | 0.4 | 9.6 | <0.4 | 24 | <0.2 | 11 | <0.2 | 12 |
| 3RD QTR | <0.3 | 3.7 | 0.18 | 9.6 | <0.5 | 16 | 0.47 | 8.9 | 0.77 | 13 | 0.72 | 13 | 0.6 | 10.8 |
| 4TH QTR | 2.1 | 28 | 1 | 14 | 1.8 | 16 | 3.1 | 13 | 2 | 16 | 0.52 | 11 | 0.33 | 14 |

TABLE 13-14

STACK MONITORING - FLUORIDE
OXIDE PLANT DRY SCRUBBERS
(THOUSAND POUNDS RELEASED)

| MONTH\YEAR | 1982 | 1983 | 1984 | 1985 | 1986 | 1987 | 1988 |
|---|---|---|---|---|---|---|---|
| JAN | 1.9 | 1.3 | 0.9 | 0.5 | 1.0 | 0.1 | 2.6 |
| FEB | 2.1 | 1.2 | 0.4 | 0.7 | 1.8 | 0.6 | 1.4 |
| MAR | 1.6 | 0.7 | 0.1 | 1.0 | 1.6 | 1.6 | 2.1 |
| APR | 1.1 | 0.7 | 0.5 | 1.6 | 2.6 | 2.1 | 2.1 |
| MAY | 1.2 | 0.5 | 0.6 | 1.1 | 1.3 | 1.2 | 2.7 |
| JUN | 0.7 | 0.8 | 1.1 | 1.5 | *** | 1.8 | 3.0 |
| JUL | 1.2 | 0.1 | 0.1 | [illegible] | 1.5 | *** | 2.4 |
| AUG | 1.4 | 0.7 | 0.8 | 1.8 | 1.1 | *** | 4.9 |
| SEP | 0.8 | 1.0 | 0.2 | 1.0 | 1.8 | 1.2 | 3.4 |
| OCT | 1.4 | 1.0 | 1.5 | 1.2 | 1.3 | 3.3 | 3.2 |
| NOV | 1.0 | 1.0 | 1.9 | 0.9 | 0.3 | 2.5 | 2.5 |
| DEC | 1.3 | 0.8 | 0.4 | 0.0 | 0.5 | 4.6 | 0.7 |
| TOTAL | 15.7 | 9.8 | 8.5 | 11.4 | 14.6 | 10.0 | 31.0 |

*** Not in use month

TABLE 13-15

SITE DAM OVERFLOW MONITORING - FLUORIDE
(MILLIGRAM PER LITER)

| MONTH\YEAR | 1982 | 1983 | 1984 | 1985 | 1986 | 1987 | 198[illegible] |
|---|---|---|---|---|---|---|---|
| JAN | <1.0 | <1.0 | <1.0 | <1.0 | <1.0 | 0.1 | <1.0 |
| FEB | <1.0 | <1.0 | <1.0 | <1.0 | <1.0 | 0.6 | <1.0 |
| MAR | <1.0 | <1.0 | 1.6 | <1.0 | <1.0 | 1.4 | <1.0 |
| APR | <1.0 | <1.0 | <1.0 | <1.0 | <1.0 | 2.1 | <1.0 |
| MAY | <1.0 | <1.0 | <1.0 | <1.0 | <1.0 | 1.2 | <1.0 |
| JUN | <1.0 | <1.0 | 1.1 | <1.0 | <1.0 | 1.8 | <1.0 |
| JUL | <1.0 | <1.0 | <1.0 | <1.0 | <1.0 | *** | 1.2 |
| AUG | <1.0 | <1.0 | 1.2 | <1.0 | ** | *** | 1.0 |
| SEP | <1.0 | <1.0 | <1.0 | <1.0 | ** | 1.2 | <1.0 |
| OCT | <1.0 | <1.0 | <1.0 | 1.9 | <1.0 | 3.3 | <1.0 |
| NOV | <1.0 | <1.0 | <1.0 | <1.0 | <1.0 | 2.5 | <1.0 |
| DEC | <1.0 | <1.0 | <1.0 | <1.0 | <1.0 | 4.6 | <1.0 |

** Results lost

TABLE 13-16

VEGETATION MONITORING - FLUORIDE
(PARTS PER MILLION)

| | STATION 12 | STATION 13 | STATION 14 | STATION 15 |
|---|---|---|---|---|
| **1982** | | | | |
| 1ST QTR | 12 | 11 | <10 | 25 |
| 2ND QTR | 13 | 11 | 18 | 22 |
| 3RD QTR | 14 | <10 | <10 | 12 |
| 4TH QTR | 14 | 14 | 12 | 5 |
| **1983** | | | | |
| 1ST QTR | 14 | 9 | 12 | 16 |
| 2ND QTR | 8 | 29 | 15 | 19 |
| 3RD QTR | 15 | 70 | 29 | 31 |
| 4TH QTR | 54 | 39 | 480 | 72 |
| **1984** | | | | |
| 1ST QTR | 35 | 23 | 67 | 47 |
| 2ND QTR | <10 | <10 | <10 | <10 |
| 3RD QTR | 10 | 16 | 6 | 7 |
| 4TH QTR | 13 | 43 | 74 | 56 |
| **1985** | | | | |
| 1ST QTR | 16 | 51 | 19 | <10 |
| 2ND QTR | <10 | 10 | 10 | <10 |
| 3RD QTR | <10 | 32 | 29 | <10 |
| 4TH QTR | 22 | 11 | 22 | 50 |
| **1986** | | | | |
| 1ST QTR | 6 | 46 | 41 | 14 |
| 2ND QTR | 4 | 21 | 30 | 24 |
| 3RD QTR | 48 | 20 | 16 | 23 |
| 4TH QTR | 5 | 17 | 7 | 33 |

TABLE 13-16 (Continued)

VEGETATION MONITORING - FLUORIDE
(PARTS PER MILLION)

| | STATION 12 | STATION 13 | STATION 14 | STATION 15 |
|---|---|---|---|---|
| 1987 | | | | |
| 1ST QTR | 14 | 55 | 20 | 51 |
| 2ND QTR | <10 | <10 | 31 | 38 |
| 3RD QTR | <10 | <10 | <10 | <10 |
| 4TH QTR | <10 | 14 | <10 | <10 |
| 1988 | | | | |
| 1ST QTR | 12 | 9.2 | 18 | 33 |
| 2ND QTR | <10 | <10 | 10 | 1 |
| 3RD QTR | <10 | <10 | <10 | 12 |
| 4TH QTR | 2.4 | 4.6 | 3 | 3.4 |

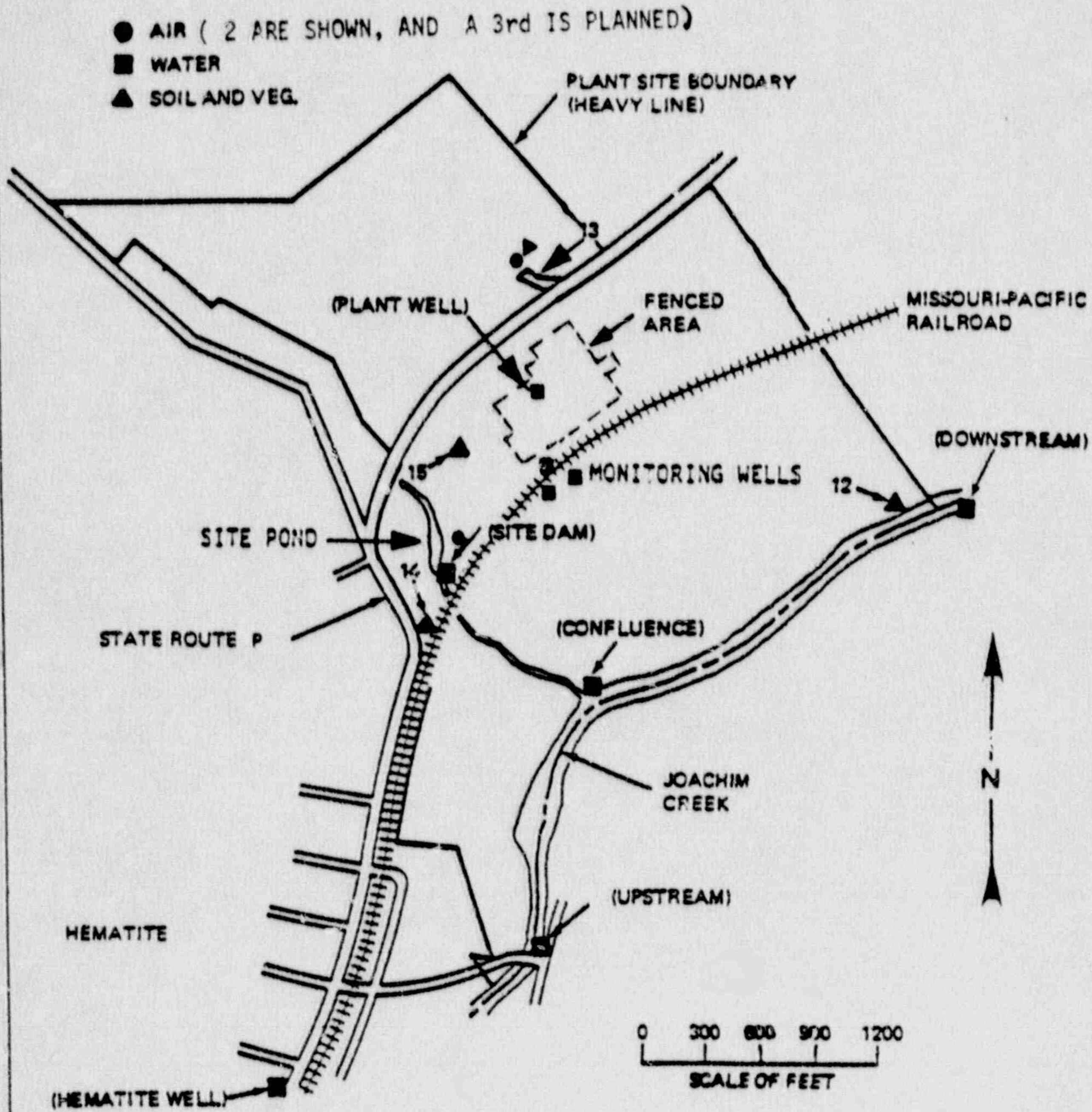


FIGURE 13-1 LOCATIONS OF HEMATITE FACILITY MONITORING SITES

# CHAPTER 14 NUCLEAR CRITICALITY SAFETY

## 14.1 Administrative and Technical Procedures

Administrative and technical procedures for ensuring criticality safety, and their mode of implementation, are described in the Nuclear Fuel Manufacturing Program documentation system. The key features of these procedures relating to criticality safety are summarized below.

a) Define individual (management, professional, and operations staff) responsibilities for nuclear safety through training, job descriptions, written procedures, and performance reviews.

b) Ensure that criticality limits and controls, when implemented by engineered safeguards and physical controls, are implemented correctly, and are reviewed and approved by both management and safety personnel.

c) Ensure all facility changes and modifications are reviewed for criticality safety implications by qualified safety personnel.

d) Ensure that all facility changes and modifications having criticality safety implications receive a criticality safety evaluation by qualified personnel, are independently reviewed by qualified personnel, and are reviewed for consistency with the safety evaluation, postings, and operating procedures prior to being placed in use.

e) Ensure that criticality limits and controls established by safety evaluations are conservative at credible accident conditions.

f) Ensure compliance and applicability of criticality limits and controls through:

- audits and inspections of equipment and facilities employed in the handling and storage of SNM,
- testing of safety related instrumentation on a regular and defined schedule, and
- review and update of operating procedures, engineered safeguards, and safety related documents on a regular and defined schedule by management and safety personnel.

## 14.2 Preferred Approach to Process Design

It is the intent of Combustion Engineering to employ physical controls and permanently engineered safeguards on processes and equipment in the establishment of nuclear safety limits, wherever practical. Physical controls may utilize safe geometry for the maximum enrichment permitted under the license or may use favorable geometry in combination with other types of controls.

When criticality safety is based on an approach other than safe geometry, engineered safeguards with appropriate administrative controls, if needed, will be employed to assure that key parameters are bounded within a regime that precludes criticality in the event of a single credible violation of the specified limits.

All process designs are evaluated for criticality safety. The ensuing criticality limits and controls are based upon consideration of such factors as the consequences of added internal and external moderation, reflective properties of structures, container walls and personnel, interaction with other SNM, and inadvertent operator errors. For mass limited operations, precautions against SNM accumulations in process equipment are identified.

Subsequent sections examine methods of control other than safe geometry.

## 14.3 <u>Basic Assumptions</u>

### 14.3.1 <u>Analytic Models</u>

### 14.3.1.1 <u>Individual Units</u>

### 14.3.1.1.1 <u>Safe Individual Units</u>

A safe individual unit, SIU, is defined as an individual isolated subcritical unit whose characteristic parameter (mass or geometric) shall not exceed the limiting value derived by the following procedure.

a) Minimum critical values of the characteristic parameters shall be derived from experimental data based on optimum moderation and full reflection for a given material composition. Cylinder and slab data shall correspond to dimensions of infinite extent in the direction(s) perpendicular to the diameter or thickness, respectively.

b) The derived minimum reflected critical values shall be reduced by the following safety factors to obtain the upper limit values for the safe individual unit.

| | |
|---|---|
| Mass | 2.3 |
| Volume | 1.3 |
| Cylinder Diameter | 1.1 |
| Slab Thickness | 1.2 |

c) Since the above minimum critical values for cylinders and slabs are derived for either a cylinder of infinite length or a slab of infinite extent in the plane transverse to the thickness dimension, an increase in these minimum critical values may be warranted for cylinders and slabs of finite dimensions when they are sufficiently isolated from other subcritical units. The increase in safe dimensions may be derived by buckling conversion equations after the method of H.C. Paxton on pages 16 and 17 of Reference 1.

This definition of SIUs is applicable to material compositions having a U-235 enrichment up to 5 w/o. For enrichments in excess of 5 w/o, the limiting value of 350 grams of U-235 shall be imposed. An isolated subcritical unit is defined as being separated from other subcritical units by a minimum of eight inches of full density water, or the larger of: (a) twelve feet, or (b) the greatest distance across an orthographic projection of the largest of the SNM mass distributions on a plane perpendicular to the line joining their centers. The effectiveness of other materials or separations shall be evaluated by a validated analytical model.

Figures 14-1 through 14-5 show plots of derived minimum critical values of mass volume, cylinder diameter, slab thickness, and surface density versus enrichment for homogeneous UO2 powder and water mixtures. Also shown are the minimum critical reflected values reduced by the applicable safety factor for the safe individual unit. The data points were derived from two sources; as indicated on the figures, UKAEA denotes Reference 2 and DP-1014 denotes Reference 3.

Heterogeneous UO2 - water data were also extracted from References 2 and 3. These two data sources employed different ranges of pellet/pin diameters. Reference 2 heterogeneous data encompassed pin diameters ranging from 0.4 to 1.0 inches whereas the data of Reference 3 cover a range from 0.05 to 0.6 inches. Both data sources are required over the enrichment range of interest to deduce minimum critical values corresponding to optimum moderation. This is illustrated in Figure 14-6 where critical mass, volume, infinite cylinder diameter, semi-infinite slab thickness and semi-infinite slab areal density are plotted from the homogeneous limit to a pin diameter of 0.6 inches for 5 w/o and 2 w/o enriched UO2; the 2 w/o enriched data are normalized to the 5 w/o data at the homogeneous limit for display purposes. It can be seen that the minimum critical values are at the smaller pin diameters at the higher enrichment. As the enrichment decreases, the minimum critical value moves to the larger pin diameters.

Figures 14-7 through 14-11 show plots of minimum critical values of mass, volume, infinite cylinder diameter, and semi-infinite slab thickness and areal density versus enrichment for isotropic heterogeneous UO2 and water mixtures.

Criticality data on aqueous solutions of uranyl fluoride and uranyl nitrate are of interest in establishing safety limits for the chemical recovery process equipment employed in processing of scrap and waste materials. Table 14-1 provides these data for uranium enriched to 5 w/o U-235 and, for comparison purposes, data on homogeneous UO2 are provided. Since the data source (Ref. 4) was the same as for ANSI/ANS 8.1-1983 and this standard provides subcritical limit type data, the aqueous solution data of Table 14-1 were corrected to critical using information provided in the data source. The objective was to place the data for aqueous solutions and UO2 on a comparable basis. In Table 14-1, the aqueous solution data have also been adjusted with the same safety margins as employed for the minimum critical values of the mass and geometric limits for UO2. In this manner one can judge equivalence on the same bases.

It should be noted that additional conservatism may exist in the aqueous solution "critical" data since it is based on the most conservative of the three calculational models employed to fit experimental data and extrapolate or interpolate to points of interest.

The data discussed above for water reflected homogeneous or heterogeneous UO2 - water mixtures assumes full reflection. Figure 14-12, which was extracted from Reference 4, shows the dependence of the spherical U-235 critical mass (U(93.5) metal spheres) versus thickness of various reflector materials. The point of interest is that the reflector worth is a function not only of the thickness of the material but also the composition of the material. Water is a convenient reference material but many structural materials are very effective materials. Consequently, the criticality safety of a given quantity of fissionable material must take into account the environment. If the environment is a more effective reflector than water, it must be considered and evaluated by appropriate analytical modelling.

14.3.1.1.2 <u>Other Subcritical Individual Units</u>

The safe individual unit of the previous section was based on the use of a single limiting mass or geometric variable by application of a safety factor to experimentally derived data based on the conditions of optimum moderation and full water reflection. Other safe subcritical units may be defined using more stringent controls on the UO2 environment. These controls may include limits on the degree of internal moderation, the amount of external moderation (reflectively of the environment) in combination with limits on enrichment, mass and/or geometric characteristics of the UO2 containers. In these cases, engineered safeguards and administrative controls are required to assure subcriticality limits are not violated.

These controls are discussed in Section 14.3.4 on Administrative Control Models.

14.3.1.2 Nuclear Interaction Methods

Interaction between subcritical units which are not isolated may be evaluated by one or more of the three following methods, providing the prerequisites of the method are met: 1) surface density, 2) solid angle, and 3) transport or Monte Carlo analytical models. The minimum allowed spacing between adjacent subcritical units computed by these methods shall be no less than one foot.

14.3.1.2.1 Surface Density Model

The surface density model may be used to evaluate arrays of safe individual units as defined in 14.3.1.1.1 and subject to the requirement that each mass limit have a fraction critical of $\leq$ 0.3 and each volume and cylinder limit have a fraction critical of $\leq$ 0.4.

The concept of fraction critical for a SIU is based on somewhat arbitrary definition deduced from correlations of experimental data; see, for example, Reference 5. The definition employed here takes the fraction critical as the ratio of the SIU mass, or equivalent spherical mass, to that of an unreflected critical sphere of the same composition.

In evaluating SIUs for this license, non-spherical SIUs are reduced to spherical shapes using buckling conversions based on the following equations:

$$B^2 = \left(\frac{\pi}{R_s + \lambda_s}\right)^2 \qquad \text{Sphere}$$

$$= \left(\frac{\pi}{x + \lambda_t}\right)^2 \times \left(\frac{\pi}{y + \lambda_t}\right)^2 + \left(\frac{\pi}{z + \lambda_t}\right)^2 \qquad \text{Slab}$$

$$= \left(\frac{2.405}{R_c + \lambda_c}\right)^2 + \left(\frac{\pi}{H + 2\lambda_c}\right)^2 \qquad \text{Cylinder}$$

For convenience, unreflected extrapolation lengths, $\lambda$, are taken from Figure 2 of Reference 5.

$\lambda$sphere = 2.1 cm
$\lambda$slab = 2.7 cm
$\lambda$cyl. = 2.25 cm

Although these data are for $U(93)O_2F_2$ solutions, their use in a consistent manner should have small impact on buckling conversions.

Figures 14-1 through 14-3 and 14-7 through 14-10 each have a broken line curve showing the dependence of the appropriate fraction critical value versus enrichment; these curves can be compared against the curves showing the minimum critical reflected values reduced by the safety factors to determine which value is more limiting and appropriate for use in the surface density method. For mass limited SIUs, the application of the safety factor to the minimum critical reflected mass is more limiting for both homogeneous and heterogeneous data. For the minimum critical volume parameter, the value deduced using the fraction critical is more limiting except above the 4.5 w/o enrichment value in the heterpgeneous UO2 - H2O data. For minimum critical cylinder diameters, values deduced from the fraction critical are more limiting below about 2 w/o enrichment in both the homogeneous and heterogeneous systems.

Spacing of SIUs in a two or three dimensional array is determined according to the following criteria.

1) Mass limited SIUs shall be spaced such that the smeared density of the SIUs on a given plane shall not exceed 50% of the minimum water reflected semi-infinite critical slab surface density, based on optimum moderation.

2) For cylinder and volume limited SIUs, the spacing area should be based on 25% of the minimum critical water reflected semi-infinite slab thickness.

3) Each SIU shall be centered in its respective spacing area or volume depending upon whether the array of SIUs is two or three dimensional.

#### 14.3.1.2.2 Solid Angle Model

When either the "fraction critical" or the "smeared" slab thickness limitations cannot be met for interacting subcritical units, the spacing may be established by the solid angle method described in Reference 6, subject to the limitations described therein. If this method is employed in a nuclear safety evaluation, each criterion and limitation shall be addressed in the documentation of the safety evaluation.

#### 14.3.1.2.3 Transport and Monte Carlo Codes

The interaction between subcritical units may be calculated explicitly using qualified or verified analytical models when the prerequisites for the previously defined interaction methods could not be fulfilled, the previously defined methods were too conservative, or the configuration and composition of various regions are too complex. When the multiplication factor is calculated explicitly, the target multiplication factor shall be no greater than $k_T$, where $k_T$ is defined by the following equation.

# 1

## IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

# 1

## IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

# 1

IMAGE EVALUATION
TEST TARGET (MT-3)

1.0 1.1 1.25 1.4 1.6 1.8 2.0 2.2 2.5 2.8 3.2 3.6 4.0

150mm

6"

# 1

## IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

$$k_T \leq k_c - \Delta k_u - \Delta k_s$$

where $k_c$ is the calculated multiplication factor for the benchmark experiments using the defined calculational model (cross sections, codes, etc.).

$\Delta k_u$ is the uncertainty in the calculated results for the benchmark experiments at the 95% confidence level.

$\Delta k_s$ is the allowed margin of subcriticality, i.e., 0.05.

14.3.2 Accident Conditions

The following credible accident conditions are normally considered in the criticality safety evaluation of a given process design.

1) adverse changes in dimensions and spacing within the process system;

2) adverse changes in density of SNM and the amount of admixed moderator;

3) adverse changes from mass or concentration limits, where applicable;

4) interactions with SNM in transit;

5) adverse changes in parasitic absorptions in fixed poisons, where required for reactivity control;

6) the effect of cumulative errors or uncertainties on downstream criticality limits and controls;

7) adverse changes in interspersed moderation and reflector composition;

8) the inadvertent introduction or accumulation of SNM in process operations;

9) the non-failsafe consequences of process failures (mechanical failures, loss of air pressure, loss of electrical power, etc.);

10) potential water sources which may affect moderation controlled processes;

11) effects of fire fighting, flooding, and storms on criticality safety limits and controls.

When multiple events are correlated or follow as a natural consequence, they are treated as a single event. If a single postulated credible event should result in a predicted critical condition, process or environmental design changes or other engineered safeguards are given a higher priority over administrative controls in achieving a safe process design.

### 14.3.3 Process Modelling

#### 14.3.3.1 Computer Codes

The types of computer codes employed in nuclear criticality calculations are dependent upon the type of fuel-moderator configurations being evaluated and it is primarily in the cross section generation area where different codes enter. For example, three classes of calculations are identified: 1) heterogeneous lattice calculations where the fuel-water configurations are consistent with those encountered in PWR nuclear steam supply system design analyses, 2) heterogeneous lattice calculations for low H/U lattices such as fuel shipping casks, compact spent fuel storage racks, and fuel manufacturing operations, and 3) homogeneous UO2 - water mixture or aqueous solutions of uranium compounds. Computer codes for the first and second groups of calculations use broad group cross section derivation techniques based on the methodology developed for the broader class of design analyses whereas the homogeneous calculations use the standard Hansen-Roach library; all three methods generally employ Monte Carlo spatial solution techniques.

The CEPAK lattice program is employed to generate four broad group (three fast and one thermal) cross sections for heterogeneous UO2 - water mixtures characteristic of the PWR lattices. CEPAK is a synthesis of the FORM(7), THERMOS (8) and CINDER (9) codes. These codes are interlinked in a consistent way with inputs from differential cross section data from an 83 group library. Modifications have been applied to the U-238 resonance integral to correct for a recognized over-estimation of that quantity in ENDF/B-IV. The entire neutron spectrum is represented in 83 neutron groups between 0 and 10 MeV. Neutron leakage in a single Fourier mode is represented by either P-1 or B-1 approximations to transport theory throughout this entire range. Resonance shielding is determined analytically; the Hellstrand (10) correlation is employed for U-238, with appropriate adjustments guided by Monte Carlo calculations of resonance capture in U-238 so as to provide agreement with selected measurements of the conversion ratio. For clad UO2, appropriate Dancoff correction factors are determined for uniform lattices. For heterogeneous lattices, this calculation is extended to include the heterogeneities by nearest neighbor approximation.

In some cases the GGC-3 Code (11) may be employed to calculate spectrum weighted few group cross sections for structural materials or trace element materials not in the CEPAK multigroup cross section library.

Few group spatial calculations for heterogeneous lattices may be done with either the DOT II W transport code(12) or the KENO-IV Code (13).

In the case of low hydrogen calculations the KENO-IV code is employed for the determination of the effective multiplication factor. The NITAWL subroutine is employed to generate self-shielded 123 group cross sections from the 123 super group XSDRN library (15)-DLC-16. The resulting library is collapsed into a homogenized 16 broad group library in a typical heterogeneous lattice cell using the XSDRNPM Code (14). The latter code is also used to obtain 16 broad group cross sections for other regions containing structural and/or moderating regions. This code sequence is used for heterogeneous lattice calculations at low hydrogen density where approximations in the CEPAK lattice code are no longer valid.

Homogeneous fuel-water mixtures are analyzed with the KENO-IV code. In these cases the fuel may be in powder or granular form, admixed with other elements, and assumed to be isotropically distributed with water. The fuel-water mixture may be contained in vessels within a regular or irregular array. Moderator may be assumed to exist in the space between these vessels. In these analyses, the primary library source is the 16 broad group Hansen-Roach cross section library distributed with the KENO-IV code by the Radiation Shielding Information Center.

14.3.3.2 Cross Sections

Three basic cross section libraries are employed in criticality safety analyses.

1) The 83 group microscopic cross section library employed with the CEPAK lattice parameter code. The microscopic data base for both fast and thermal neutron cross sections is derived from the Evaluated Nuclear Data File ENDF/B-IV.

2) The 16 group Hansen-Roach library for the KENO-IV code. This library is the version distributed in the SCALE-02 KENO-IV code package from the RSIC Code Center. It contains the Knight modified entries extending the range of SIG P to lower values for U-238.

3) The 123 group super-XSDRN library-DLC-16. This is the 123 group cross sections in GGC-123/XSDRN format based on the ENDF/B-II data from the Radiation Shielding Information Center.

### 14.3.4 Administrative Control Models

#### 14.3.4.1 Mass Controls

Homogeneous and heterogeneous mass limits for safe individual units are discussed in Section 14.3.1.1.1; use of these limits in storage arrays is discussed in Section 14.3.1.2.

Alternative mass limit controls may be employed for certain operations which, in reality, are multiparameter controls. For example, a mass limit associated with specific containers (volume limit); in addition, controls on internal and/or external moderation limits may be imposed.

Consider the case of a 35 Kg mass limit on UO2 powder in a 5 gallon, or less, container with a cover and locking ring. The cover and locking ring are employed for radiological control, to maintain chemical purity, and to prevent the ingress of water. For areas where, for example, overhead sprinkler systems are employed for fire control or other credible sources of water may be postulated, these containers would be opened and closed in hooded enclosures to minimize the likelihood of water ingress. In the event of internal and external flooding of the container, criticality would not occur since the 5 gallon (19 liter) container is a safe volume for UO2 powder enriched to 5 w/o U-235.

To assess the criticality safety of the 35 Kg mass limited, 5 gallon container in storage arrays, a KENO analysis was run for an infinite planar array of such containers.

Each container was modelled as a 10.75 inch inner diameter cylinder with a wall thickness of 0.025 inches and a height of 14.25 inches; the cylinder wall was treated as a void. The 35 Kg UO2 (10.96 g/cc density) was homogeneously distributed in water to fill the container. A one foot thick water reflector was above and below the array. The results of the KENO IV analysis using 16 energy group Hansen-Roach cross sections are tabulated below versus separation distance between the containers.

| Separation (inches) | Multiplication Factor |
|---|---|
| 12 | 0.9584 $\pm$ 0.0065 |
| 14 | 0.9414 $\pm$ 0.0081 |
| 16 | 0.9268 $\pm$ 0.0070 |

Based on the analyses, it is concluded that the minimum separation distance of 12 inches between adjacent 35 Kg mass limited 5 gallon containers will result in a safe storage array for dry UO2 powder when the containers are closed and stored in a mist free environment. In the event that all cans are flooded with water internally, the array is still subcritical. Should the cans become fully reflected, the array is still subcritical since the containers (19 liters) are a safe volume and adjacent containers are isolated by the 12 inches of water. Thus, the double contingency criterion is satisfied. It should be noted that the above analyses are conservative since the container volume exceeded that of a 5 gallon container by 12%.

Another case of interest is that for a 35 Kg mass limit of heterogeneous material in containers having a volume of 5 gallons, or less. For a 5 gallon container the volume (19 liters) exceeds the safe volume of 17 liters for optimum moderation conditions at a UO2 enrichment of 5 w/o U-235 but is less than the critical reflected volume of 24 liters.

Additionally, the mass of 35 Kg is just about equal to the minimum critical reflected mass of 35.4 Kg UO2 at a rod diameter of 0.1 inches. This latter dimension corresponds to the optimum moderation condition of Figure 14-6. A key difference between the experimental data employed to establish the heterogeneous mass limits of Section 14.3.1.1 and the case of interest here is the experimental data are for rods and the case of interest is a random array of pellets in a container.

Sintered pellets of a 0.4 inch diameter when randomly loaded in a container pack to an average density of 5.95 g/cc with one sigma variation of 0.264 g/cc, as determined by a series of 14 measurements. Thus at a 95% confidence level, the minimum average density is not less than 5.686 g/cc. At this density and in the absence of upset conditions, the critical mass is in excess of 200 Kg UO2 at an enrichment level of 5 w/o U-235 using the data of Figure 1.E.1 of Reference 2.

Therefore, it may be concluded that the 35 Kg mass limit on UO2 pellets enriched to 5 w/o U-235 in a 5 gallon or less container results in a safe condition when stored at a minimum separation distance of 12 inches. Should these containers become internally flooded, the array is still safe in the absence of upset conditions. Thus the double contingency criterion is met.

Clean, hard scrap, i.e., chipped pellets and pellet chips, having a known composition may be stored or shipped in 5 gallon, or less, containers. The irregular geometry of the UO2 chips and broken pellets is such that the average density of the scrap in a container is higher than for whole pellets of regular geometry. Therefore, in the event of flooding of the containers the critical mass should be higher than for whole pellets because of the reduced level of internal moderation. The shipping containers employed to ship hard, clean scrap from the Windsor, CT., Facility to the Hematite Facility will actually employ a stainless steel container smaller than 5 gallons by 29%.

14.3.4.2 Internal Moderation Controls

A useful control parameter for UO2 of enrichment 5 w/o U-235 or less is the amount of admixed moderator. UO2 has a relatively high theoretical density (10.96 g/cc) but the UO2 encountered in the process steps prior to pelletizing has a much lower bulk density. The granular output of the UF6 to UO2 conversion process has a bulk density in the range of 3.0 to 3.5 g/cc, whereas the micronized UO2 granules have a bulk density in the range of 1.8 to 2.4 g/cc. For these bulk densities, it is of interest to examine the degree of subcriticality when the UO2 is saturated with water, i.e., all interstitial voids are filled with water and the UO2 density in the mixture is the same as the bulk density.

Figure 14-13 shows the loci of the reflected minimum critical mass and geometric parameters such as volume, cylinder diameter, and slab thickness as a function of enrichment and bulk density. The effect of a deviation of ± 5% from the minimum critical value is also displayed so that an estimate of the breadth of the minimum in the given parametric dependence on UO2 concentration can be related to a change in bulk density. These curves indicate the pre-micronized UO2 is generally under moderated, as far as mass limits are concerned, when saturated with water. For the geometric parameters, the water saturated pre-micronized UO2 is under moderated particularly at the higher enrichment, but at the lower enrichment it is at near optimum moderation. The micronized powder, when saturated with water, is at near optimum conditions for the geometric parameters at higher enrichments and for mass at the lower enrichments. As the enrichment increases, the micronized powder tends to be under moderated at the saturated water condition and would require an upset force plus added moderator to become optimally moderated.

There are two ranges of container volumes of interest when considering internal moderation controls. For container sizes less than the minimum critical reflected volume of Figure 14-2, the container is subcritical regardless of the contents (≤ 5 w/o U-235) and the surrounding environment, i.e., water density, as long as other SNM bearing containers are not neutronically coupled to the container of interest.

For container sizes in excess of 31 liters and for UO2 bulk densities encountered in the processing line, criticality can occur when the container is fully reflected. The bulk density of the micronized powder is typically in the range of 1.8 to 2.4 grams per cubic centimeter. Thus, when the power is saturated with water, the mixture is near optimum for the minimum critical volume.

To assure criticality safety in large volume containers, moderation control procedures are implemented. These procedures are based on administrative controls of one form or another. Even when engineered safeguards are employed, they are dependent upon administrative controls for testing and periodic recalibration.

For purposes of this license, dry UO2 is defined as UO2 having a water content of 1.0 w/o or less. A homogeneous and isotropic distribution of UO2 enriched to 5 w/o U-235 with 1 w/o water has an infinite multiplication factor of approximately unity. Thus subcriticality is assured for large containers of "dry" 5 w/o enriched UO2 even when fully reflected.

The 30-inch diameter UF6 shipping containers employ internal moderation control. The hydrogen to uranium atomic ratio is less than 0.088 which is equivalent to the purity specification of 99.5% for the UF6 (16). In this case, all impurities in the UF6 are assumed to be hydrogen. Analyses reported in Reference 17 show the infinite multiplication factor of the 30-inch diameter UF6 cylinders to be less than unity at this H/U ratio. Strict administrative controls on the handling, transfer of the UF6 contents, and cleaning of these containers are required to assure safe handling of this material.

14.3.4.3 External Moderation Controls

External moderation controls have to do with the control of moderating material external to the container bearing the SNM. As noted earlier, fire fighting in the production area at the Hematite Facility, more specially Building 253, 254, and 255, is limited to dry techniques and there are no overhead sprinkler systems. Care has been taken to route water and steam pipes away from areas where large containers of UO2 are employed. Thus, in the event of a rupture in these water or steam lines, there will be a reduced likelihood of potential criticality concerns.

In most cases equipment has been designed to be safe in the event of complete reflection, e.g., a tight fitting reflector of water. There may be cases where such a precaution does not exist.

In these instances a barrier is employed around the containers to prevent the approach of significant moderating material to within one foot of the cylindrical wall of the container. This barrier modifies the criticality limits of the container such that it may be treated as a partially reflected container. A measure of the effect of an annular gap between a container of U(93) solution and a six-inch thick annular water reflector is given on page 19 of Reference 1. It is concluded that for a gap of 7 inches, the reflector savings is just one half that of a close fitting water reflector. Caution must be employed with this criticality control procedure to preclude materials against which the barrier is ineffective.

In those cases where external moderation is absent and internal moderation is not optimum, the critical limits versus UO2 concentration are higher than values discussed in previous sections. Figure 14-14 shows the dependence of critical mass, volume, and cylinder diameter versus concentration of UO2 in unreflected homogeneous UO2-H2O mixtures.

14.3.4.4 Concentration Controls

Uranium concentration controlled SIUs shall be limited to a maximum concentration of 25 grams of uranium per liter. The effect of evaporation and/or precipitation shall be considered in the nuclear safety analysis, such that if precipitated a safe mass will not be exceeded.

Concentration controlled SIUs shall not be considered to contribute to interacting arrays, but shall be located outside exclusion areas assigned by the surface density method.

A safe mass of uranium shall be used for aqueous solutions only under administrative control. The safe mass limit does not apply to fixed poison systems.

14.3.4.5 Slab Limit on Pellets

The minimum critical slab limit for fully reflected heterogeneous UO2 water mixtures shown in Figure 14-10 is 4.17 inches. This value is based on optimum moderation and pellet size. As indicated in Figure 14-6, for 5 w/o enriched UO2 the optimum pellet diameter ranges from 0.2 to 0.4 inches for slab geometry ; this optimum moderation condition occurs at a smeared UO2 density of about 3.3 g $UO_2$/cc.

Sintered UO2 pellets of 0.4 inch diameter when randomly loaded in a container pack to an average density of 5.95 g/cc with a one sigma variation of 0.264 g/cc, as determined by a series of 14 measurements. Thus, at a 95% confidence level, the minimum average density is not less than 5.686 g/cc. At this density, the minimum critical slab thickness is significantly greater than 4.17 inches for full reflection conditions. The data of Reference 3 does not extend into this low H/U regime but the data of Reference 2 does. Referring to Figure I.E.16 of the latter reference indicates the minimum critical slab thickness to be greater than 6.2 inches. It should be noted that the data of Reference 3 does appear to indicate a smaller minimum critical slab thickness under optimum moderation conditions for pellet diameters of 0.4 inches by about 0.3 inches.

For conservatism, the minimum critical slab thickness deduced from Reference 2 is taken to be 6.2 inches, and corrected downward by 0.3 inches to 5.9 inches. Dividing by a safety margin of 1.2 results in a slab limit of 4.92 inches.

For trays containing pellets loaded in a random fashion, a slab limit of 4.0 inches is adopted for UO2 pellets having enrichments of up to 5 w/o U-235. For trays containing pellets loaded in a non-random fashion, as with the corrugated separators, the SIU limits deduced from Figure 14-10 are employed, i.e., at optimum moderation and pellet diameter.

The four-inch slab limit on UO2 pellets has been examined by KENO analyses for purposes of gaining confidence in the safety of the deduced limit and to explore the sensitivity of the slab limit to external and internal moderation.

The following assumptions are employed in a four-inch high, horizontally infinite slab array of pellet trays.

a) Each stainless steel pellet tray (10.25" long, 5" wide and 2" high) is assumed to contain 5 w/o enriched UO2 pellets of nominal diameter 0.4 inches, an average UO2 density of 5.686 g/cc, and the remainder full density water.

b) The stainless steel walls (0.1984 cm th.) and cover (0.1270 cm th.) of each tray are represented explicitly.

c) A 12 inch thick full density water reflector is placed in contact with the top and bottom of the four-inch thick slab.

The dependence of the multiplication factor on horizontal bidirectional separation of each vertical pair of trays and density of water within this spacing is examined with the KENO-IV code. Sixteen broad group cross sections are generated for each region of the model using the XSDRNPM and NITAWL codes and the 123 group library (DLC-16). Results are summarized in Figure 14-15.

Conclusions drawn from these results are as follows.

1) A fully reflected, infinite slab having a thickness of 4.0 inches, which consists of a stack of two thin walled stainless steel trays containing 0.4 inch diameter pellets having an average UO2 density within the tray of 5.686 g/cc and the remaining volume filled with water, has an effective multiplication factor of 0.815 $\pm$ 0.008. The subcritical margin demonstrates the conservatism of the 4 inch slab limit for these containers.

2) For a range of spacings of up to six inches between the 4 inch high modules of the slab and for a range of water densities in the intra-module spaces varying between zero and full density, the effective multiplication factor is less than the value with zero spacing. Thus, the introduction of extraneous moderating materials

between components of a large array of pellet trays arranged in a slab configuration will not result in an increase in the effective multiplication factor.

### 14.4 Fixed Poisons

Holding tanks may be poisoned with Raschig rings in accordance with ANSI Standard N16.4-1979. Raschig ring sample tubes will be provided to enable inspection for accumulation of solids and to provide samples for testing the physical and chemical properties of the rings. These inspections and tests will be conducted in accordance with the ANSI Standard.

### 14.5 Structural Integrity Policy and Review Program

Whenever nuclear criticality safety is directly dependent on the integrity of a fixture, container, storage rack or other structure, design shall include consideration of structural integrity. The fulfillment of structural integrity requirements shall be established by physical test or by analysis by an engineer knowledgeable in structural design.

### 14.6 Analytical Models and their Validation

#### 14.6.1 Heterogeneous UO2 - Water Configurations

#### 14.6.1.1 Four Broad Neutron Group Model

The four broad neutron group model is an extension of the analytical model employed in PWR design analyses. It differs from the latter in the type of spatial flux solution and method of calculation of the multiplication factor. Instead of diffusion theory, transport or Monte Carlo methods are employed. This model is used for the analysis of fuel storage facilities, fuel assembly shipping containers, and is applicable to certain fuel manufacturing process evaluations, especially those involving full or near full density moderator condition.

For purposes of validating the analytical modelling of these heterogeneous UO2 - water lattices with or without parasitic absorbers, the critical separation experiments (18) carried out by Battelle Northwest Laboratories are analyzed.

These experiments are concerned with the critical separation distance between water moderated, subcritical clusters of fuel rods with different fixed neutron absorber types in the gap between fuel rod clusters. The experiments were carried out in a 1.8 m x 3 m x 2.1 m deep tank provided with features specifically designed and built for these experiments. The fuel rods had an active length of 914.4 mm and diameter of 11.176 mm. The fuel was clad by 6061 aluminum having an outer diameter of 12.70 mm and wall thickness of 0.762 mm. A fixed, square pin pitch of 20.32 mm was employed in each different size fuel rod cluster. Figure 14-16 shows a top and end view of the experimental configuration. Data on the experimental configurations analyzed are given in Tables 14-2 through 14-4.

The calculational methods which are used are essentially the same as those used to determine reactivity for fuel assembly storage racks, fuel assembly shipping containers, and other fuel configurations found in fuel manufacturing areas. Broad group neutron cross sections are based on the CEPAK Code. Using an appropriate buckling value and taking account of resonance absorption, three broad fast groups are collapsed from the 54 multi-group FORM type calculations and one broad thermal group is collapsed from the 29 group THERMOS calculations. Fast cross sections for certain trace elements such as sodium and zinc are obtained by averaging over an appropriate multi-group spectrum with the GGC-3 code. In addition, each component such as water gap, end plug, or poison plate has its thermal cross section determined by a slab THERMOS calculation employing a characteristic fuel environment.

Normally, for two dimensional representations, the transport Code DOT-IIW is used. Because of the short fuel length, the three dimensional Monte Carlo Code KENO IV is used with six axial levels. Batches of one hundred neutron histories are used with the first four discarded. Calculated multiplication factors are shown in Table 14-5. For economy, about 50 neutron batches were run for most cases, however, because of their greater use in fuel storage analyses, about 500 neutron batches were employed for the plain stainless steel and boral experiments.

The mean value of the calculated multiplication factor is 1.002 with a standard deviation of 0.004; thus at a 95/95 confidence level using a sigma multiplier of 2.434, the multiplication factors are between 1.012 and 0.992.

14.6.1.2 <u>Sixteen Broad Neutron Group Model</u>

This model is similar to that employed for homogeneous UO2 - water mixtures. It differs in that the broad group cross sections are calculated by the NITAWL and XSDRNPM code sequence to take into account the heterogeneity of the fuel moderator mixture. One regime of particular interest is that where low hydrogen density conditions exist between fuel assemblies as in the situation where water sprinklers provide a mist over fuel assemblies in a storage rack.

Critical experiments on the interposition of low hydrogen density materials between four 18 x 18 clusters of 4.742 w/o enriched UO2 rods were performed by the Department of Nuclear Safety of the French Atomic Energy Commission and reported in Reference 18. The fuel rods are spaced on a square pitch of 13.5 mm, contain UO2 pellets 0.790 cm in diameter, and are clad in aluminum tubes 0.94 cm O.D. with a wall thickness of 0.12 cm; the elements are 100 cm long. Figure 14-17 shows the experimental setup. The four fuel clusters are supported by a mobile device which allows them to move along orthogonal directions in a horizontal plane.

Cross shaped boxes of different thickness were employed to separate the four fuel clusters and to successfully contain air and various hydrogenous materials including the following:

1) expanded polyethylene $(C_8H_8)_n$,
2) polyethylene powder $(CH_2)_n$,
3) polyethylene balls $(CH_2)_n$, and
4) water

Water was then introduced into the bottom of the tank to fill the fuel rod clusters and reflector region; criticality was achieved on water height.

The computer codes employed in this analysis are KENO IV, NITAWL, and XSDRNPM. The reference microscopic cross section library is the 123 group super - XSDRN library, DLC-16 (6). The NITAWL and XSDRNPM Codes are used to generate 16 broad neutron energy group cross sections. NITAWL is used to generate self shielded 123 group cross sections from 123 group super-XSDRN library (DLC-16). The resulting working library is then collapsed into a homogenized 16 energy group library in a typical fuel pin cell environment using XSDRNPM. XSDRNPM is also used to obtain separate 16 group cross section sets for structural materials and external moderators.

The KENO model employed a homogenized fuel pin representation in the interior of the fuel rod cluster. The cross shaped box, the outside moderator, tank wall, lattice grid, fuel pin lower plug, bottom plate and support plate are all explicitly represented. Table 14-6 summarizes the multiplication factors computed by KENO IV for 9 critical experiments.

The statistical uncertainty and bias of the criticality analysis of the experiments have been calculated. The only criticality analyses included in the uncertainty analysis are the low hydrogen and all air calculations as these are representative of the hydrogen density range of interest in the plant criticality analyses. The results are as follows:

| | |
|---|---|
| Total Number of Results | 7 |
| Mean Value | 1.00449 |
| Standard Deviation | 0.00643 |

| | |
|---|---|
| 95/95 Multiplier | 3.34 |
| 95/95 Confidence Limits | 0.022 |
| Bias (u-1) | +0.00449 |

Due to the limited number of criticality experiments in the low hydrogen range, calculational uncertainty applied to criticality analyses of low hydrogen density systems is conservatively set at ± 0.03.

It may be concluded from the above analysis that the KENO model employing 16 broad group cross sections based on the NITAWL and XSDRNPM sequence of calculations does give acceptable agreement with experiments and an acceptable level of uncertainty for use in criticality safety evaluations.

### 14.6.2 Homogeneous UO2 - Water Configuration

Validation of the KENO-IV code and Hansen-Roach cross sections, as distributed under the SCALE code system (13), is described in Reference 20. To ascertain whether the conclusions of the latter reference are applicable to homogeneous analyses for this license, the following comparisons were made.

1) The Hansen and Roach cross sections library was verified as being identical to that distributed under SCALE, and
2) Eight of the sample problems distributed with the code were run for purposes of comparing the calculated eigenvalues with those obtained by ORNL.

Table 14-7 summarizes the eigenvalues obtained by C-E and ORNL for each sample problem. The eigenvalues agree within the stated statistical deviation. Thus, it may be concluded that the conclusions of Reference 20 concerning bias and deviation are applicable to homogeneous analyses performed by C-E.

## 14.7 Special Controls

Other special controls used to ensure nuclear safety are described, as applicable, where the various facilities or processes are presented.

## 14.8 Data Sources

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Table 14-1

CRITICAL LIMITS FOR $UO_2$, $UO_2F_2$, AND $UO_2(NO_3)_2$
FOR URANIUM ENRICHED TO 5 W/O U-235

| | $UO_2$ (3) | | $UO_2F_2$ (2) | | $UO_2(NO_3)_2$ (2) | |
|---|---|---|---|---|---|---|
| | Critical | w/S.M. (1) | Critical | w/S.M. (1) | Critical | w/S.M. (1) |
| Mass (Kg. 235) | 1.85 | 0.80 | 1.89 | 0.82 | 4.08 | 1.77 |
| Cyl. Dia. (in.) | 10.2 | 9.3 | 11.0 | 10.0 | 18.1 | 16.4 |
| Slab th. (in.) | 4.72 | 3.93 | 5.31 | 4.42 | 10.0 | 8.33 |
| Vol. (liter) | 29.1 | 22.4 | 35 | 26.9 | 137.2 | 105.5 |
| Conc. gr 235/liter | --- | --- | 273 | --- | 298 | --- |

(1) With Safety Margin: Mass-2.3, Cyl. Dia.-1.1, Slab Th.-1.2, Vol.-1.3
(2) Data from: H. K. Clark, Subcritical Limits for Uranium-235 Systems, NSE 81 pg 351 (1982)
(3) Homogeneous UO2

Table 14-2

EXPERIMENTAL DATA ON CLUSTERS OF 2.35 WT% U-235 ENRICHED $UO_2$ RODS IN WATER

| Fuel Cluster Size (No. of Rods[1]) | Critical Separation[2] Between Fuel Clusters-Xc (mm) | Experiment Number |
|---|---|---|
| 20 x 17 | 110.2 ± 0.4 | 015 |
| 20 x 16 | 83.9 ± 0.5 | 005 |
| 20 x 16 | 84.4 ± 0.5 | 049 [3] |
| 22 x 16 [4] | 100.5 ± 0.5 | 018 |
| 20 x 14 | 44.6 ± 1.0 | 021 |

(1) Fuel rods on 20.32 mm square pitch.

(2) Perpendicular distance between the cell boundaries of the fuel clusters. Error limits are on standard deviation.

(3) Rerun of Experiment 005

(4) Center fuel cluster at 20 x 16 rods. Two outer fuel clusters at 22 x 16 rods each

Table 14-3

EXPERIMENTAL DATA ON CLUSTERS OF 2.35 WT% U-235 ENRICHED $UO_2$ RODS IN WATER WITH 304L STEEL PLATES BETWEEN FUEL CLUSTERS (1)

| FUEL CLUSTERS | 304L STEEL PLATES (2) | | | | |
|---|---|---|---|---|---|
| LENGTH x WIDTH 20.32mm SQ. PITCH FUEL RODS | BORON CONTENT wt% | THICKNESS (tp, mm) | DISTANCE TO FUEL CLUSTER (3) (G, mm) | CRITICAL SEPARATION BETWEEN FUEL CLUSTERS (4) (Xc, mm) | EXPERIMENT NUMBER |
| 20 x 16 | 0 | 4.85 ± 0.15 | 6.45 ± 0.06 | 68.8 ± 0.2 | 028 |
| 20 x 16 | 0 | 4.85 ± 0.15 | 27.32 ± 0.50 | 76.4 ± 0.4 | 005 |
| 20 x 16 | 0 | 4.85 ± 0.15 | 40.42 ± 0.70 | 75.1 ± 0.3 | 029 |
| 20 x 16 | 0 | 3.02 ± 0.13 | 6.45 ± 0.06 | 74.2 ± 0.2 | 027 |
| 20 x 16 | 0 | 3.02 ± 0.13 | 40.42 ± 0.70 | 77.6 ± 0.3 | 026 |
| 20 x 17 | 0 | 3.02 ± 0.13 | 6.45 ± 0.06 | 104.4 ± 0.3 | 034 |
| 20 x 17 | 0 | 3.02 ± 0.13 | 40.42 ± 0.70 | 114.7 ± 0.3 | 035 |
| 20 x 17 | 1.05 | 2.98 ± 0.06 | 6.45 ± 0.06 | 75.6 ± 0.2 | 032 |
| 20 x 17 | 1.05 | 2.98 ± 0.06 | 40.42 ± 0.70 | 96.2 ± 0.3 | 033 |
| 20 x 17 | 1.62 | 2.98 ± 0.06 | 6.45 ± 0.06 | 73.6 ± 0.3 | 038 |
| 20 x 17 | 1.05 | 2.98 ± 0.06 | 40.42 ± 0.70 | 95.2 ± 0.3 | 039 |

(1) Error limits shown are one standard deviation.

(2) Plates are 356 mm wide by 915 mm long.

(3) Perpendicular distance between the cell boundary of the center fuel cluster and the near surface of the steel plate.

(4) Perpendicular distance between the cell boundaries of the fuel clusters.

Table 14-4

EXPERIMENTAL DATA ON CLUSTERS OF 2.35 wt% U-235 ENRICHED $UO_2$ RODS IN WATER WITH BORAL PLATES BETWEEN FUEL CLUSTERS (1)

| FUEL CLUSTERS | BORAL PLATES | | | |
|---|---|---|---|---|
| LENGTH x WIDTH 20.32mm SQ. PITCH FUEL RODS | THICKNESS(2) (tp, mm) | DISTANCE TO FUEL CLUSTER (3) (G, mm) | CRITICAL SEPARATION BETWEEN FUEL CLUSTER (4) (Xc, mm) | EXPERIMENT NUMBER |
| 20 x 17 | 7.13 ± 0.11 | 6.45 ± 0.06 | 63.4 ± 0.2 | 020 |
| 20 x 17 | 7.13 ± 0.11 | 44.42 ± 0.60 | 90.3 ± 0.5 | 016 |
| 22 x 16 (5) | 7.13 ± 0.11 | 6.45 ± 0.06 | 50.5 ± 0.3 | 017 |

(1) Error limits shown are on standard deviation.

(2) Includes 1.02 mm thick cladding of type 1100 Al on either side of the $B_4C$-Al core material. Plates 365 mm wide by 915 mm long.

(3) Perpendicular distance between the cell boundary of the center fuel cluster and the near surface of the boral plate.

(4) Perpendicular distance between the cell boundaries of the fuel clusters.

(5) Center fuel cluster at 20 x 16 rods. Two outer fuel clusters at 22 x 16 rods each.

Table 14-5

CALCULATED $k_{eff}$ VALUES

| Experiment # | Type Poison Plate | $k_{eff}$ | Monte Carlo (Std. Deviation) |
|---|---|---|---|
| 15 | None | 1.00227 | 0.00534 |
| 04 | None | 0.99912 | 0.00540 |
| 49 | None | 1.00221 | 0.00473 |
| 18 | None | 1.00313 | 0.00489 |
| 21 | None | 0.99589 | 0.00461 |
| 28 | 304 S Steel 0.0 w/o Boron | 1.00393 | 0.00308 |
| 05 | 304 S Steel 0.0 w/o Boron | 1.00329 | 0.00303 |
| 29 | 304 S Steel 0.0 w/o Boron | 1.00271 | 0.00302 |
| 27 | 304 S Steel 100 w/o Boron | 1.00418 | 0.00273 |
| 26 | 304 S Steel 0.0 w/o Boron | 0.99811 | 0.00279 |
| 34 | 304 S Steel 0.0 w/o Boron | 0.00793 | 0.00297 |
| 35 | 304 S Steel 0.0 w/o Boron | 1.00436 | 0.00290 |
| 32 | 304 S Steel 1.05 w/o Boron | 0.99970 | 0.00524 |
| 33 | 304 S Steel 1.05 w/o Boron | 1.01173 | 0.00491 |
| 38 | 304 S Steel 1.62 w/o Boron | 1.00289 | 0.00512 |
| 39 | 304 S Steel 1.62 w/o Boron | 1.00208 | 0.00506 |
| 20 | BORAL | 0.99585 | 0.00301 |
| 16 | BORAL | 1.00020 | 0.00288 |
| 17 | BORAL | 0.99519 | 0.00286 |
| | Mean $k_{eff}$ Value | 1.00157 | |
| | Standard Deviation | 0.00419 | |

Table 14-6

KENO IV RESULTS FOR NOTED GAP WIDTHS

| Description | Hydrogen Density $gm/cm^3$ | KENO IV Keff |
|---|---|---|
| | Gap Width = 2.5 cm Between Assemblies | |
| Aluminum Box + Air | 0.0 | 0.99641 ± 0.00407 |
| Aluminum Box + $(C_8H_8)_n$ | 0.0025 | 0.99913 ± 0.00384 |
| Aluminum Box + Powder $(CH_2)_n$ | 0.0414 | 1.01567 ± 0.00378 |
| Aluminum Box + Water | 0.1119 | 1.02362 ± 0.00362 |
| Water (No Aluminum Box) | 0.1119 | 0.99775 ± 0.00391 |
| | Gap Width = 5.0 cm Between Assemblies | |
| Aluminum Box + Air | 0.0 | 1.00412 ± 0.00422 |
| Aluminum Box + $(C_8H_8)_n$ | 0.0020 | 1.00748 ± 0.00421 |
| | Gap Width = 10.0 cm Between Assemblies | |
| Aluminum Box + Air | 0.0 | 1.00117 ± 0.00390 |
| Aluminum Box + $(C_8H_8)_n$ | 0.0022 | 1.00748 ± 0.00378 |

Table 14-7

COMPARISON OF KENO-IV CALCULATED EIGENVALUES
FOR SAMPLE PROBLEMS

| Problem No. | Eigenvalues C-E | Eigenvalues ORNL |
|---|---|---|
| 1 | 1.00387 +/- .00448 | 1.00569 +/- .00446 |
| 2 | 0.99733 +/- .00426 | 1.00099 +/- .00442 |
| 10 | 0.74638 +/- .00446 | 0.75215 +/- .00436 |
| 11 | 0.99846 +/- .00457 | 0.99380 +/- .00515 |
| 12 | 0.92957 +/- .00449 | 0.93089 +/- .00419 |
| 13 | 2.26645 +/- .00603 | 2.26172 +/- .00566 |
| 14 | 0.98487 +/- .00625 | 0.98060 +/- .00558 |
| 19 | 0.99726 +/- .00452 | 1.00014 +/- .00567 |

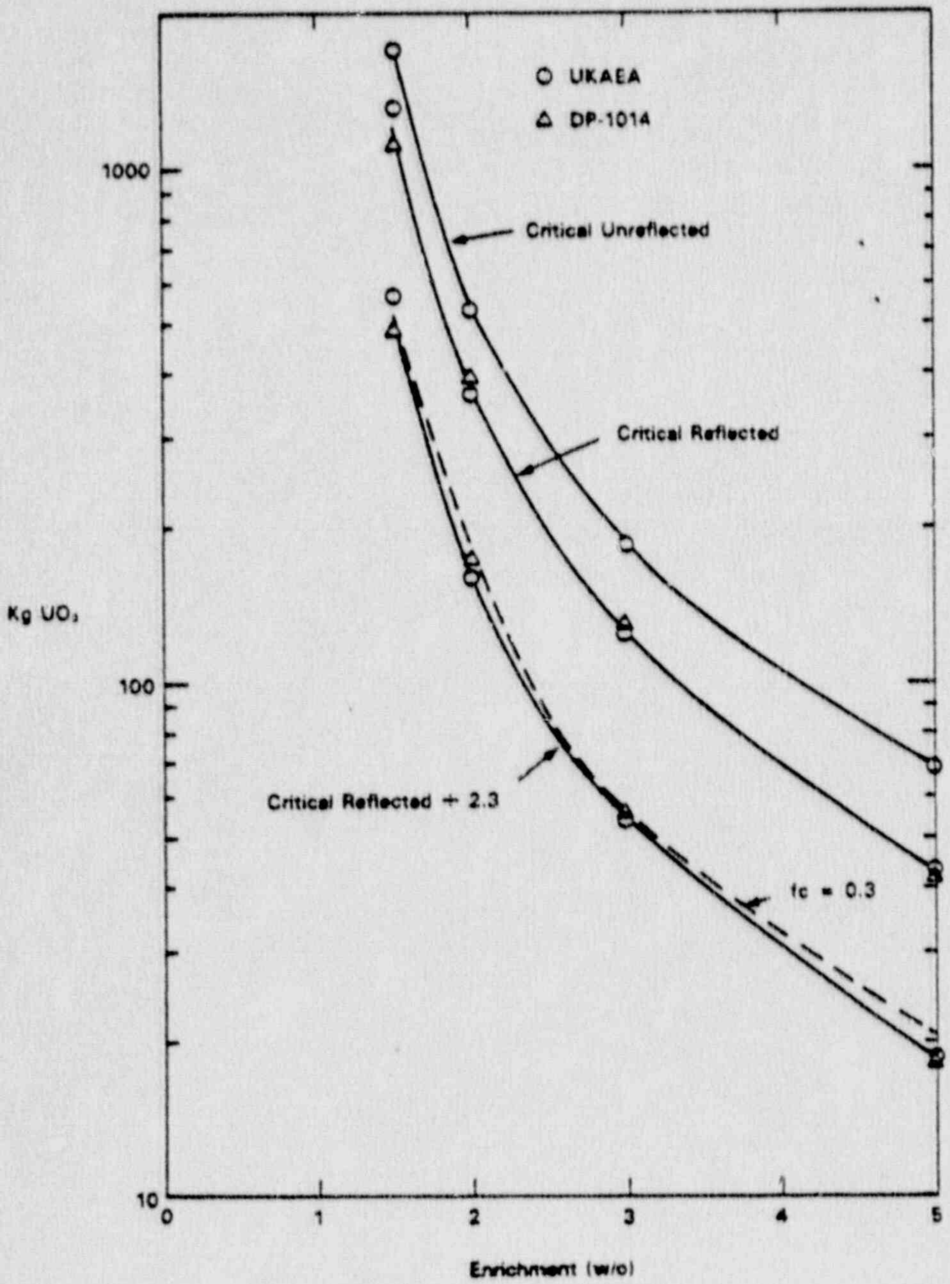


Figure 14-1

MINIMUM CRITICAL MASS FOR HOMOGENEOUS
UO2 POWDER - H2O

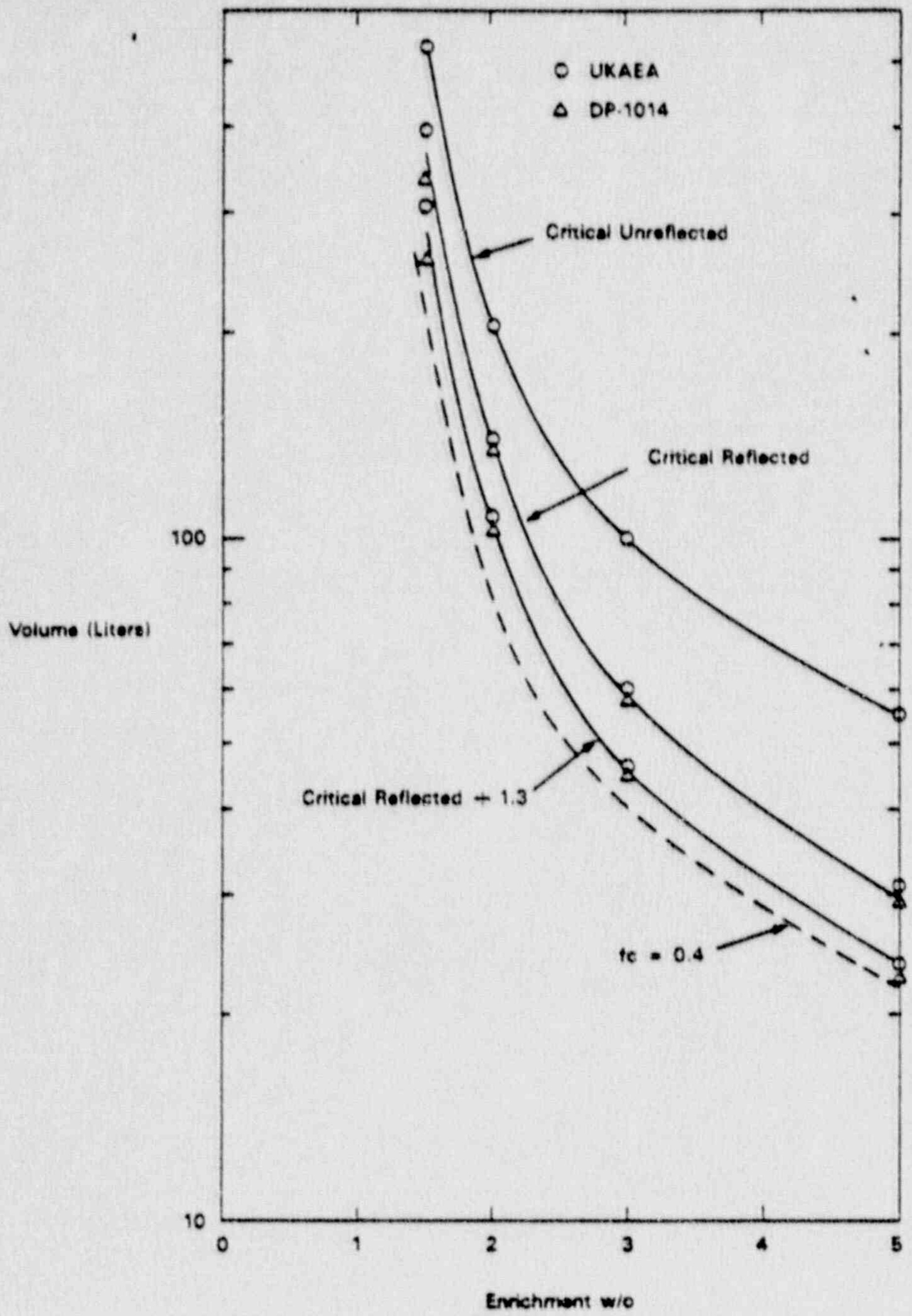


Figure 14-2

MINIMUM CRITICAL VOLUME FOR HOMOGENEOUS $UO_2$ POWDER - $H_2O$

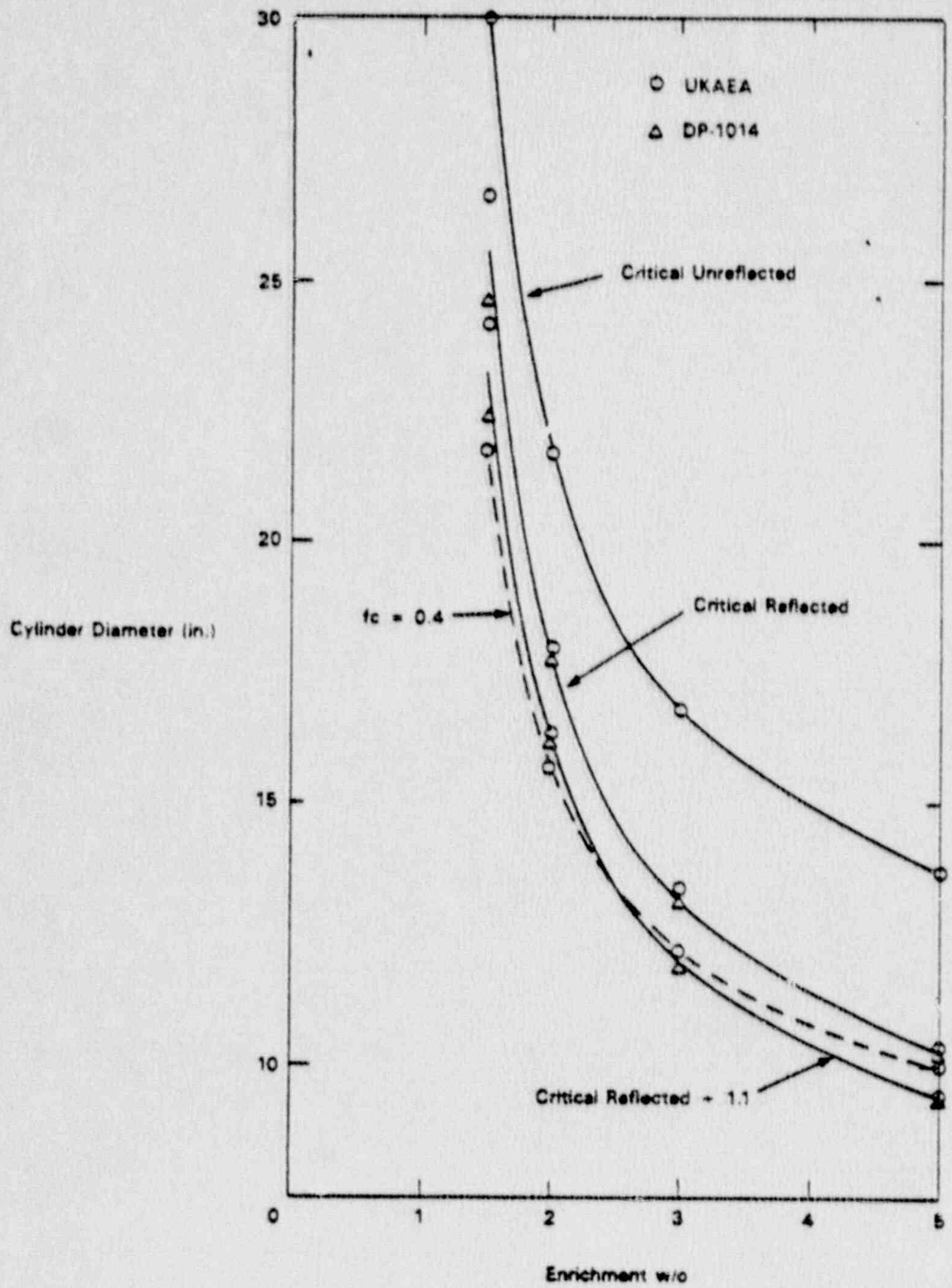


Figure 14-3

MINIMUM CRITICAL INFINITE CYLINDER DIAMETER FOR HOMOGENEOUS UO2 POWDER - H2O

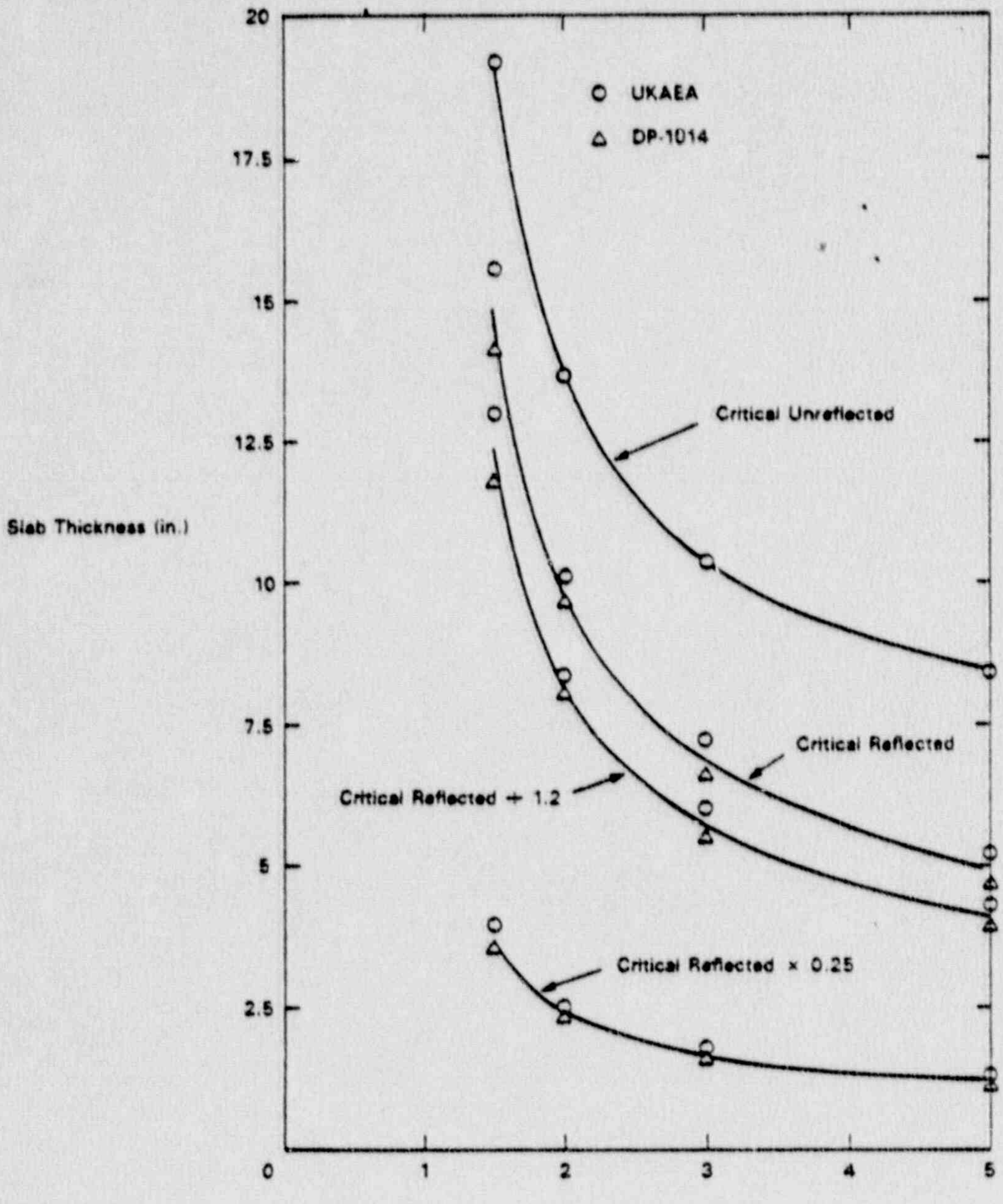


Figure 14-4

MINIMUM CRITICAL SEMI-INFINITE SLAB THICKNESS FOR HOMOGENEOUS INFINITE SLAB OF UO2 - H2O

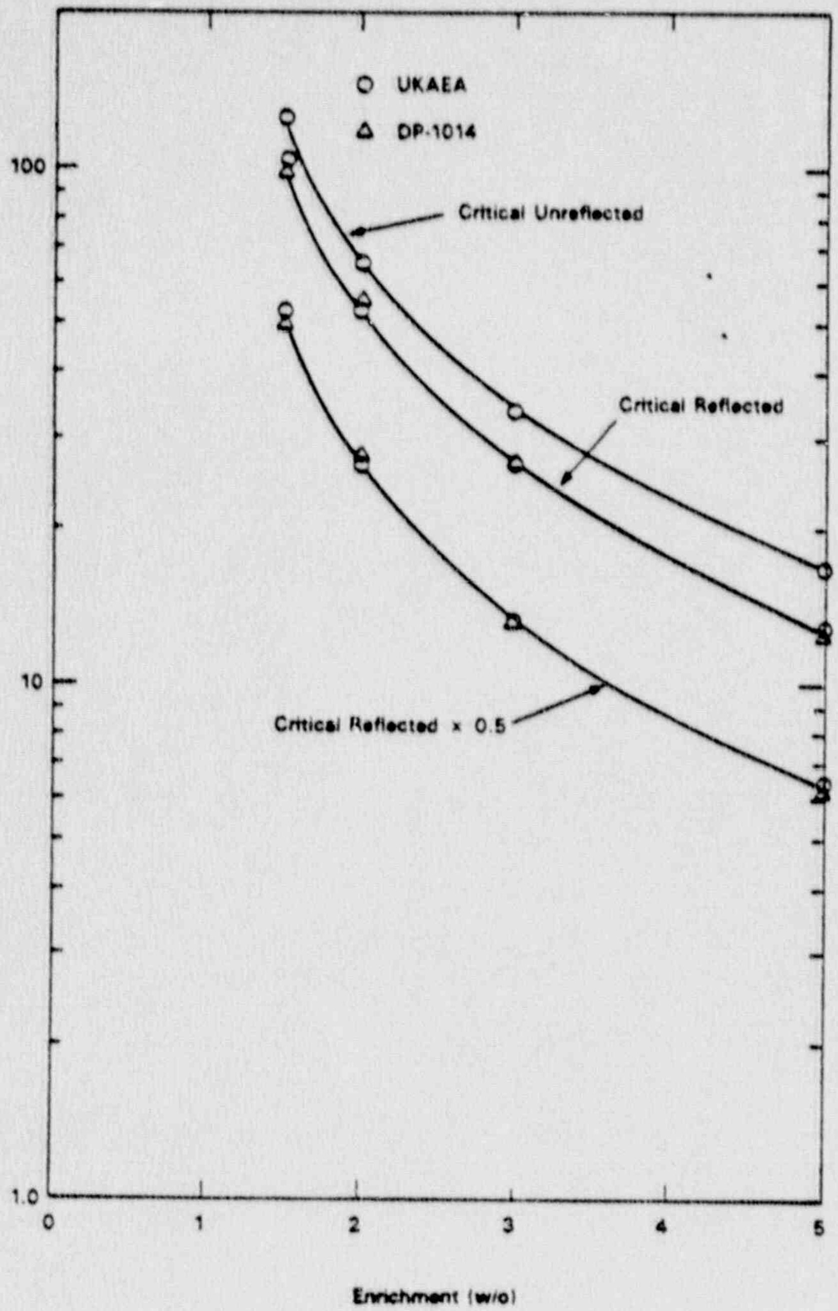


Figure 14-5

MINIMUM CRITICAL SEMI-INFINITE SLAB SURFACE DENSITY
FOR HOMOGENEOUS UO2 - H2O

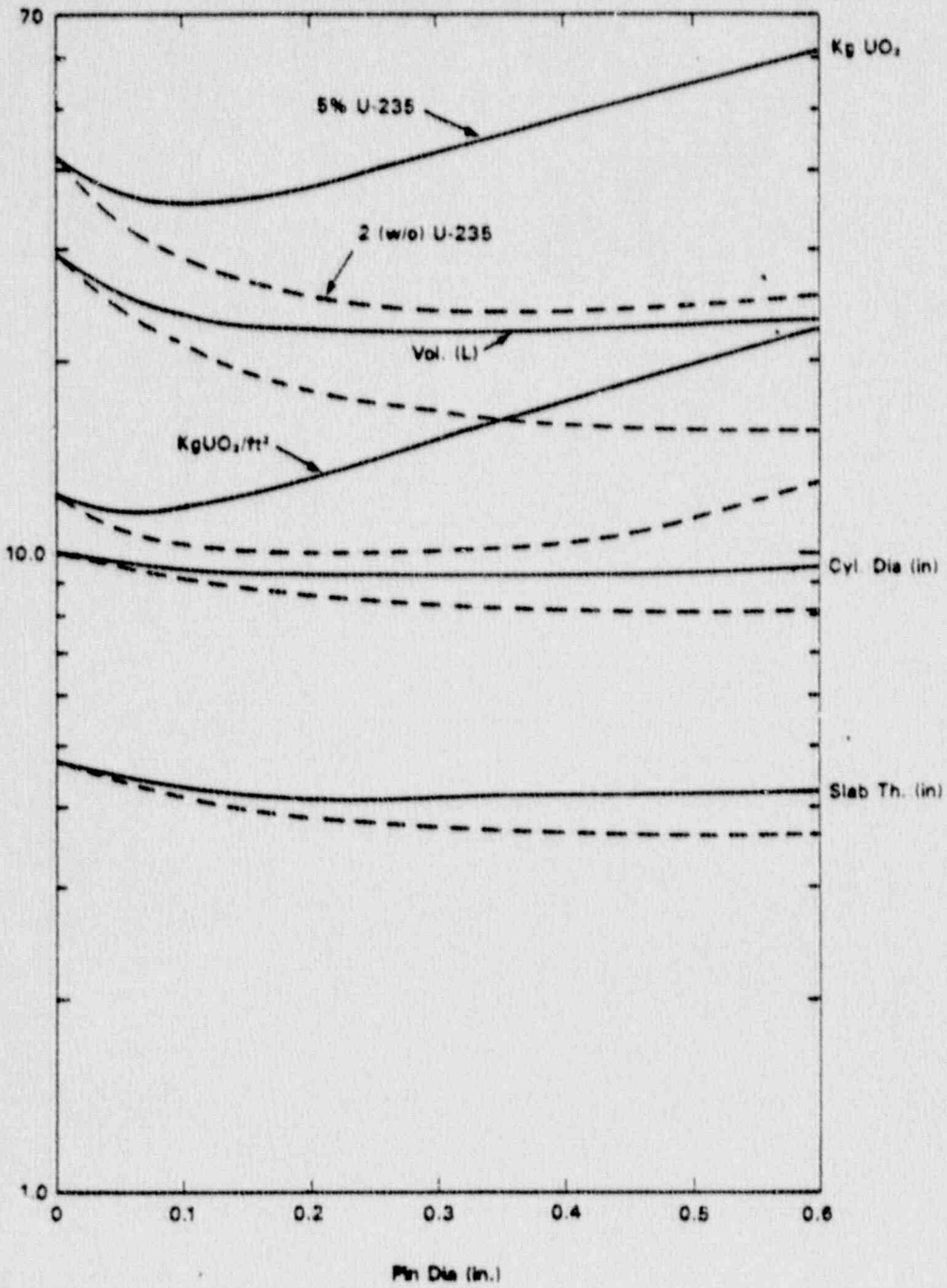


Figure 14-6

<u>PARAMETER DEPENDENCE ON PELLET DIAMETER (DATA FROM DP-1014)</u>

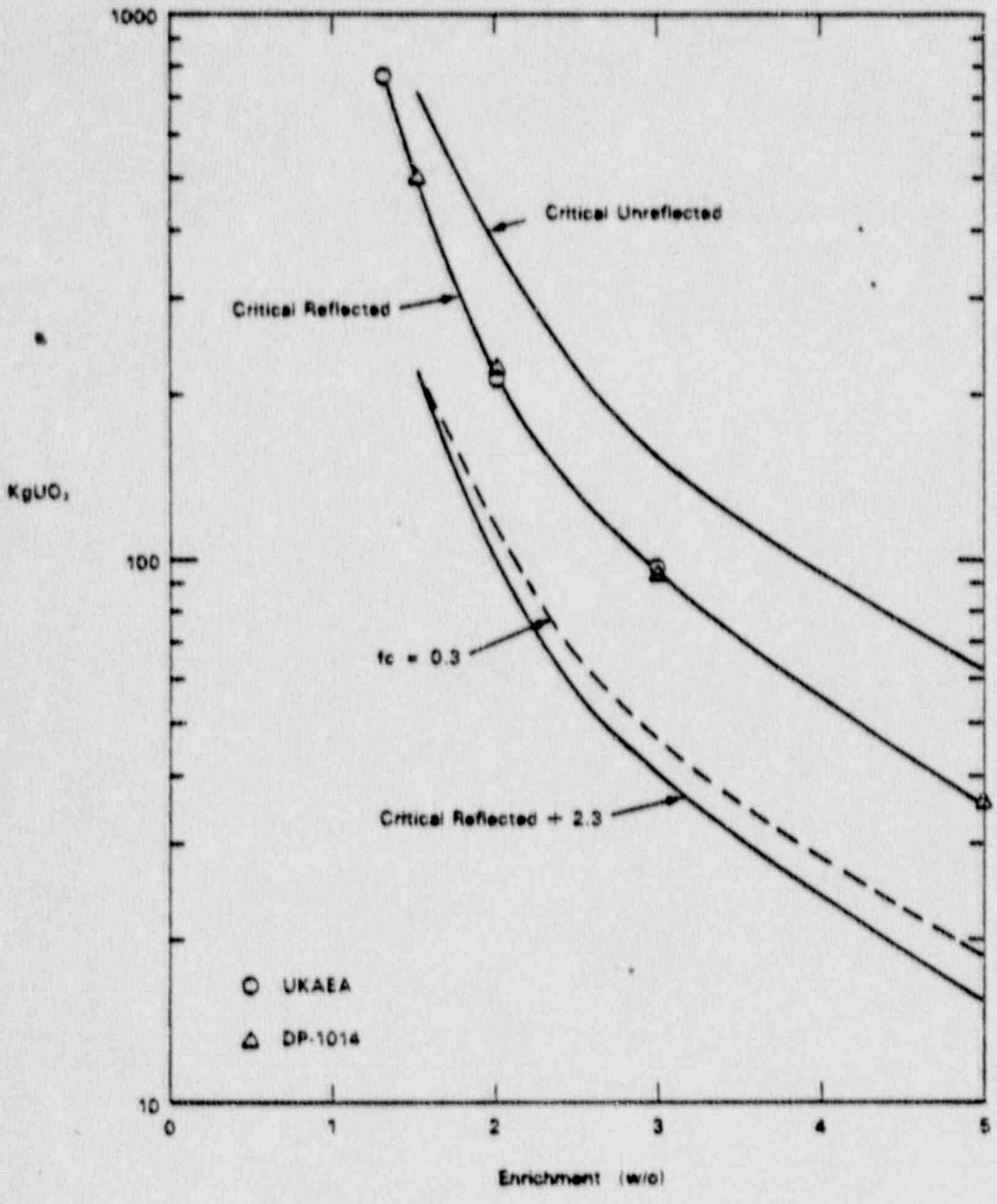


Figure 14-7

MINIMUM CRITICAL MASS FOR HETEROGENEOUS UO2 - H2O

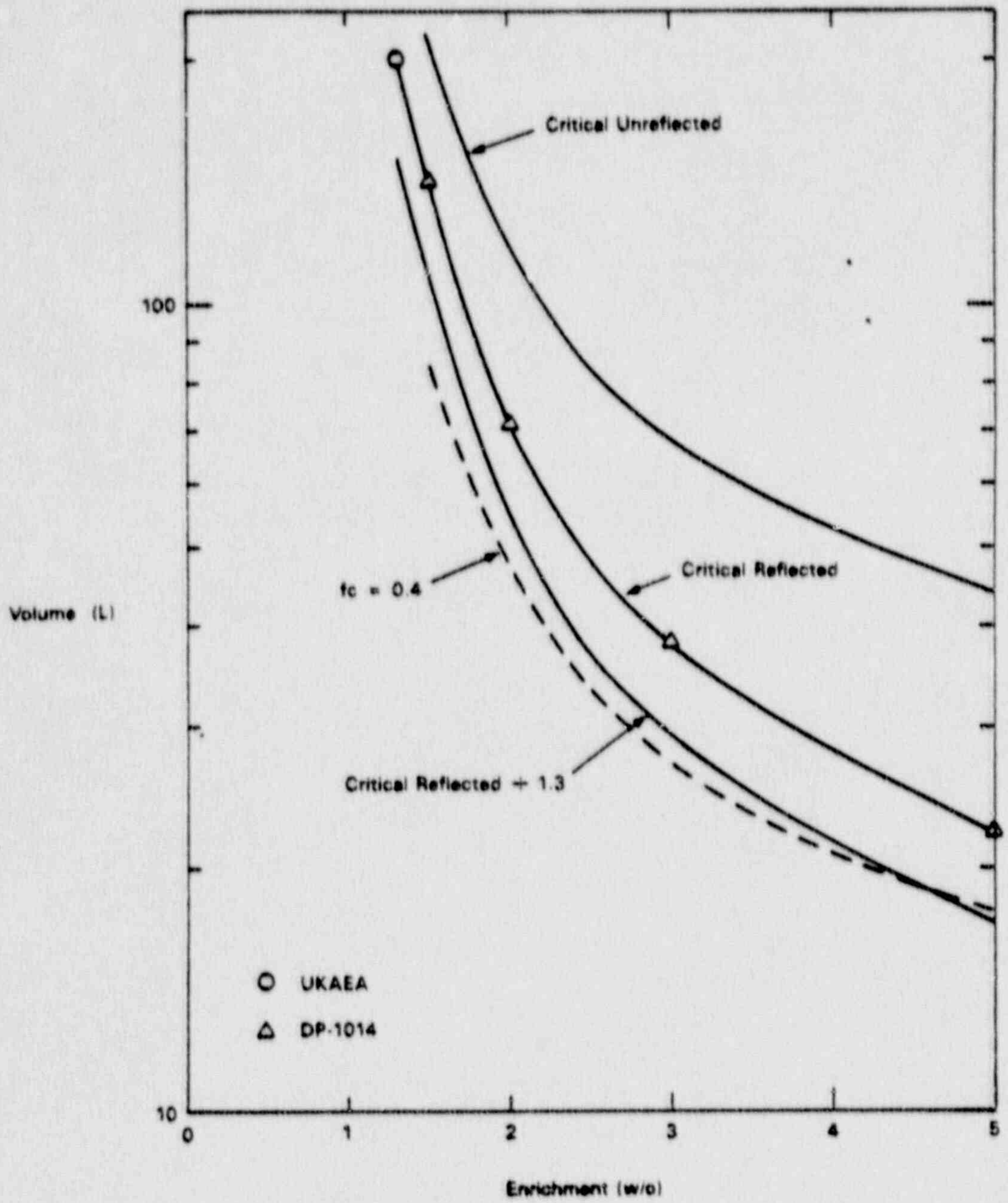


Figure 14-8

MINIMUM CRITICAL VOLUME FOR HETEROGENEOUS $UO_2$ - $H_2O$

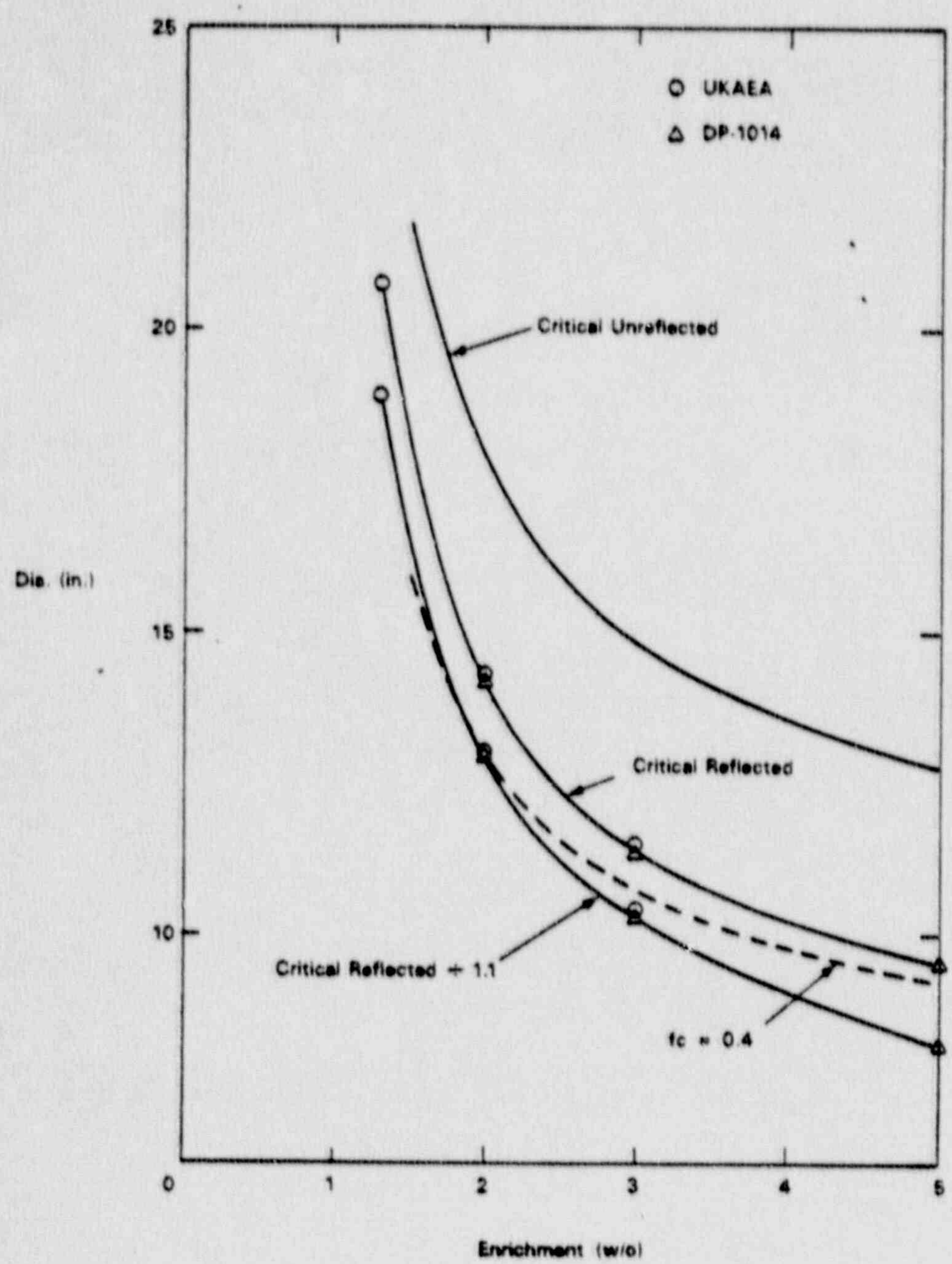


Figure 14-9

MINIMUM CRITICAL INFINITE CYLINDER DIAMETER
FOR HETEROGENEOUS UO2 - H2O

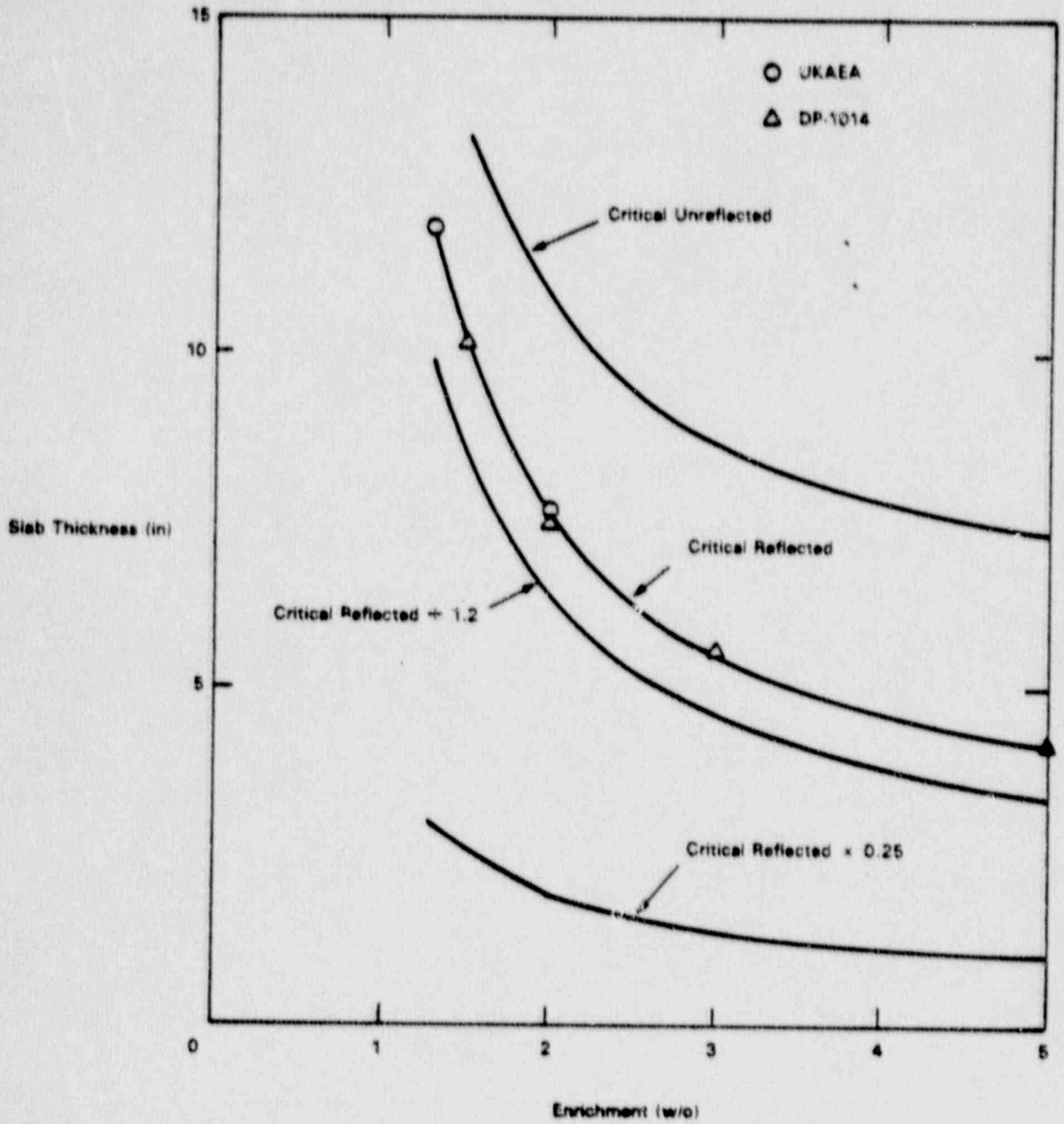


Figure 14-10

MINIMUM CRITICAL SEMI-INFINITE SLAB THICKNESS
FOR HETEROGENEOUS UO2 - H2O

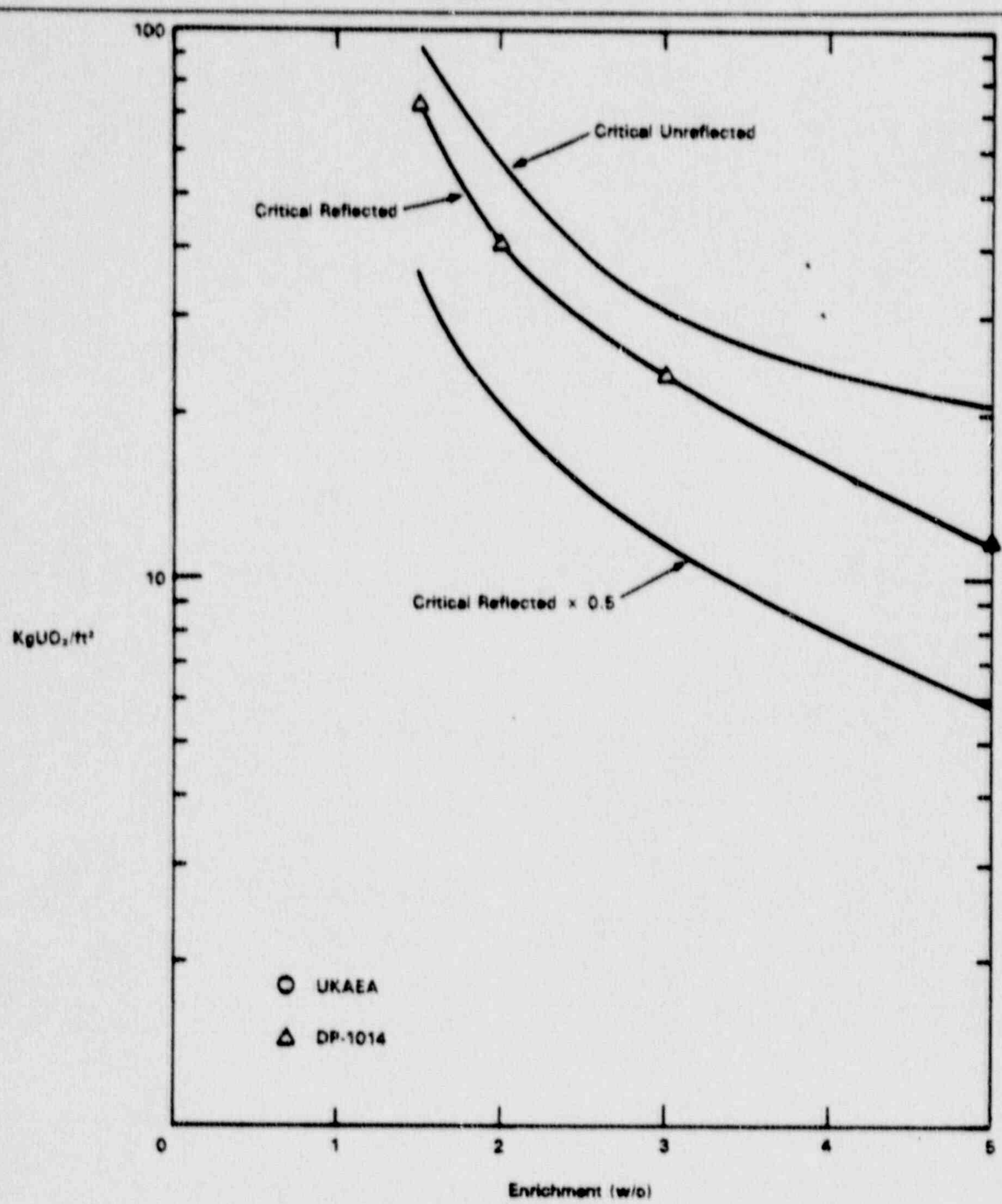


Figure 14-11

MINIMUM SEMI-INFINITE SLAB SURFACE DENSITY
FOR HETEROGENEOUS UO2 - H2O

PART II
SAFETY DEMONSTRATION

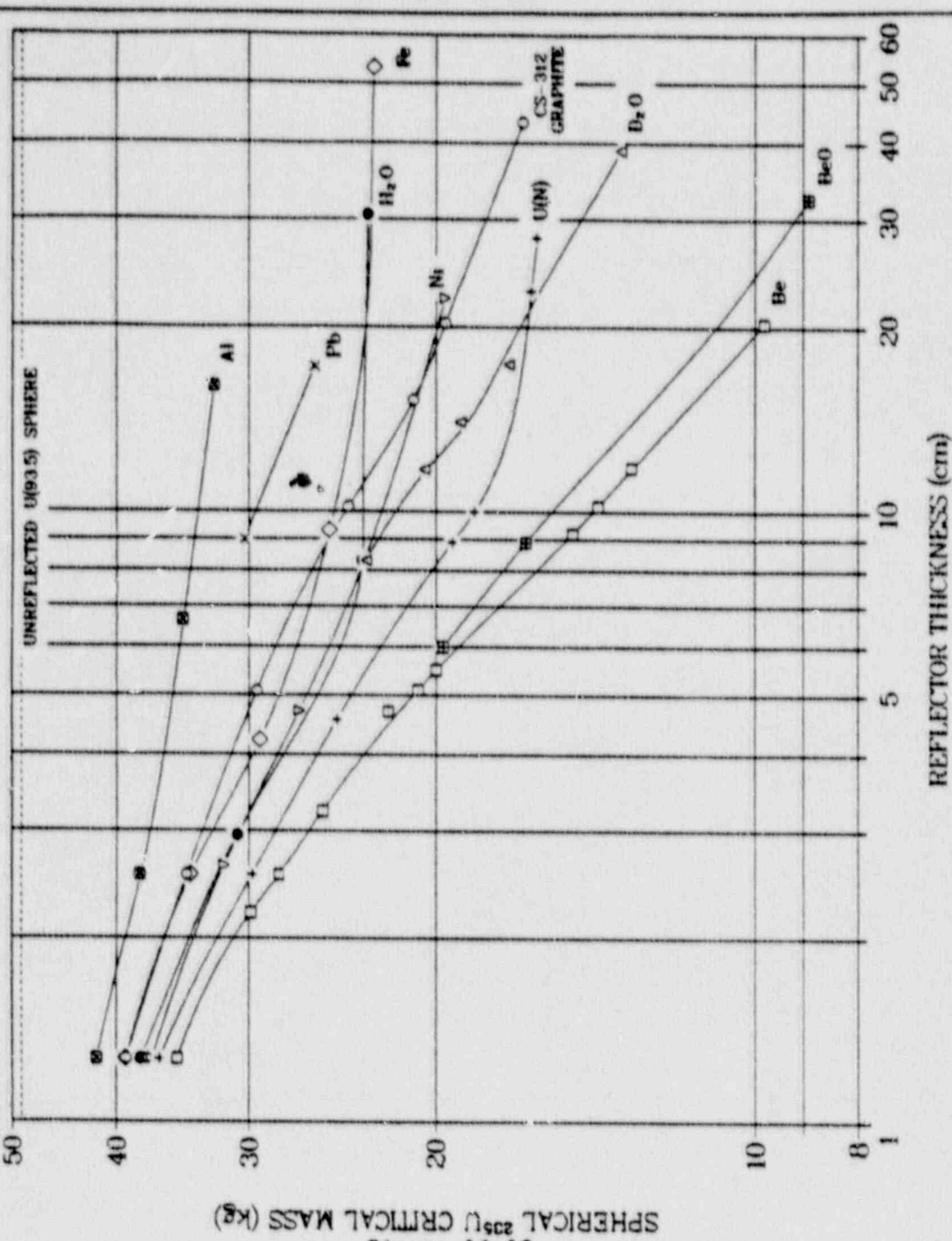


Figure 14-12

CRITICAL MASSES OF U(93.5) METAL SPHERES IN VARIOUS REFLECTORS.
URANIUM DENSITY = 18.8 g/cc
(From LA-10860-MS)

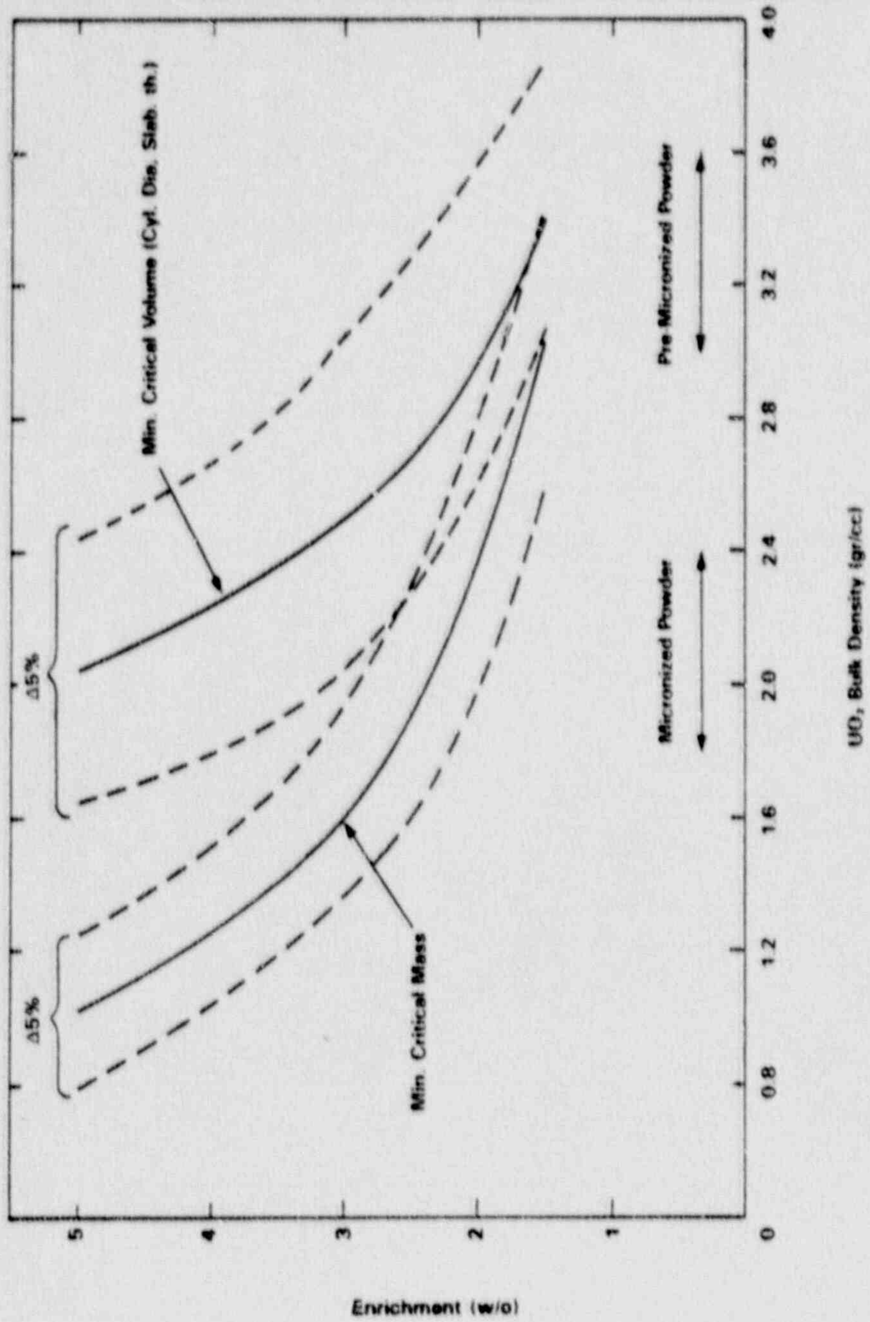


Figure 14-13

LOCI OF MINIMA IN REFLECTED MINIMUM CRITICAL MASS AND GEOMETRIC PARAMETERS WITH ENRICHMENT AND BULK DENSITY

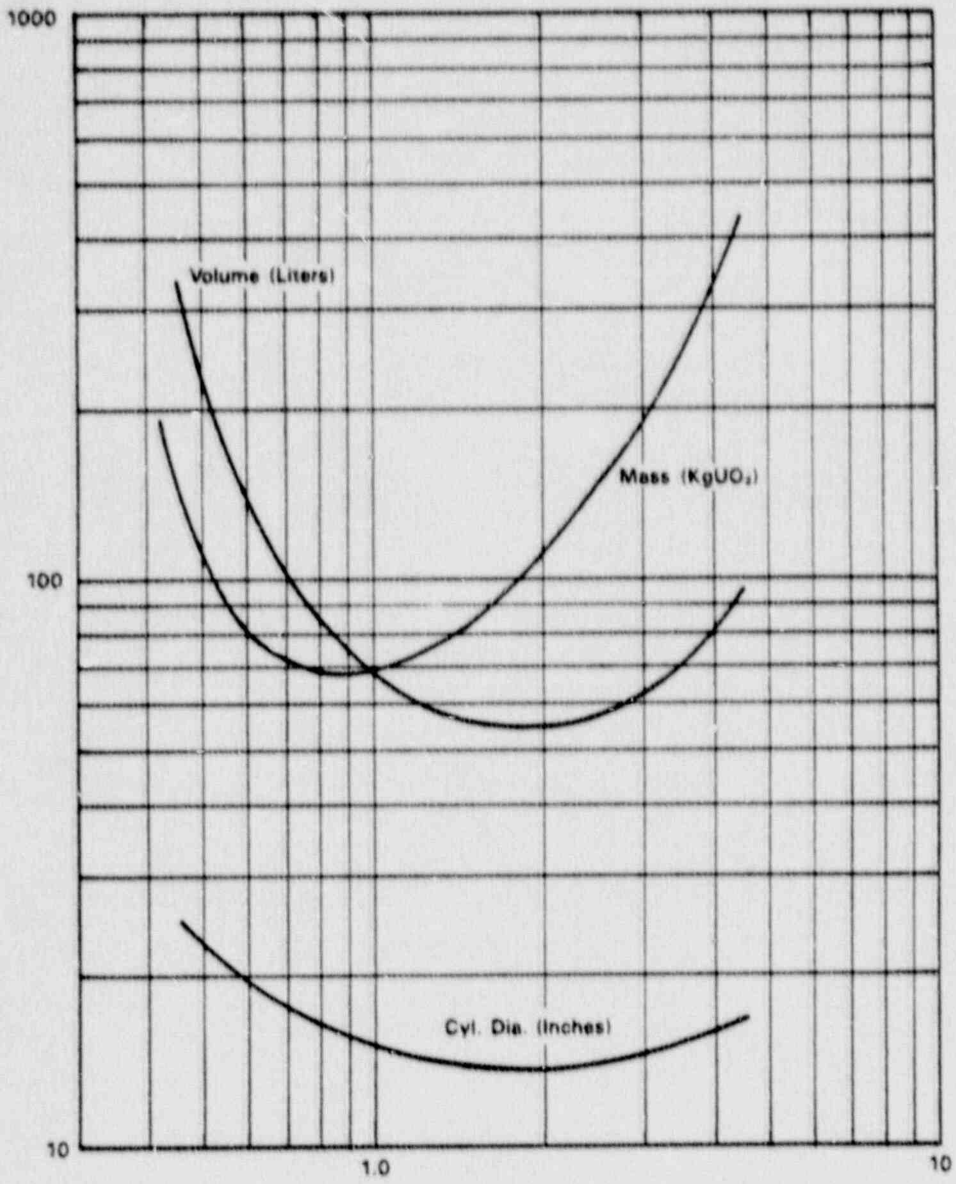


FIGURE 14-14 UNREFLECTED CRITICAL PARAMETERS FOR HOMOGENEOUS UO2-H2O VERSUS CONCENTRATION (REFERENCE 2)

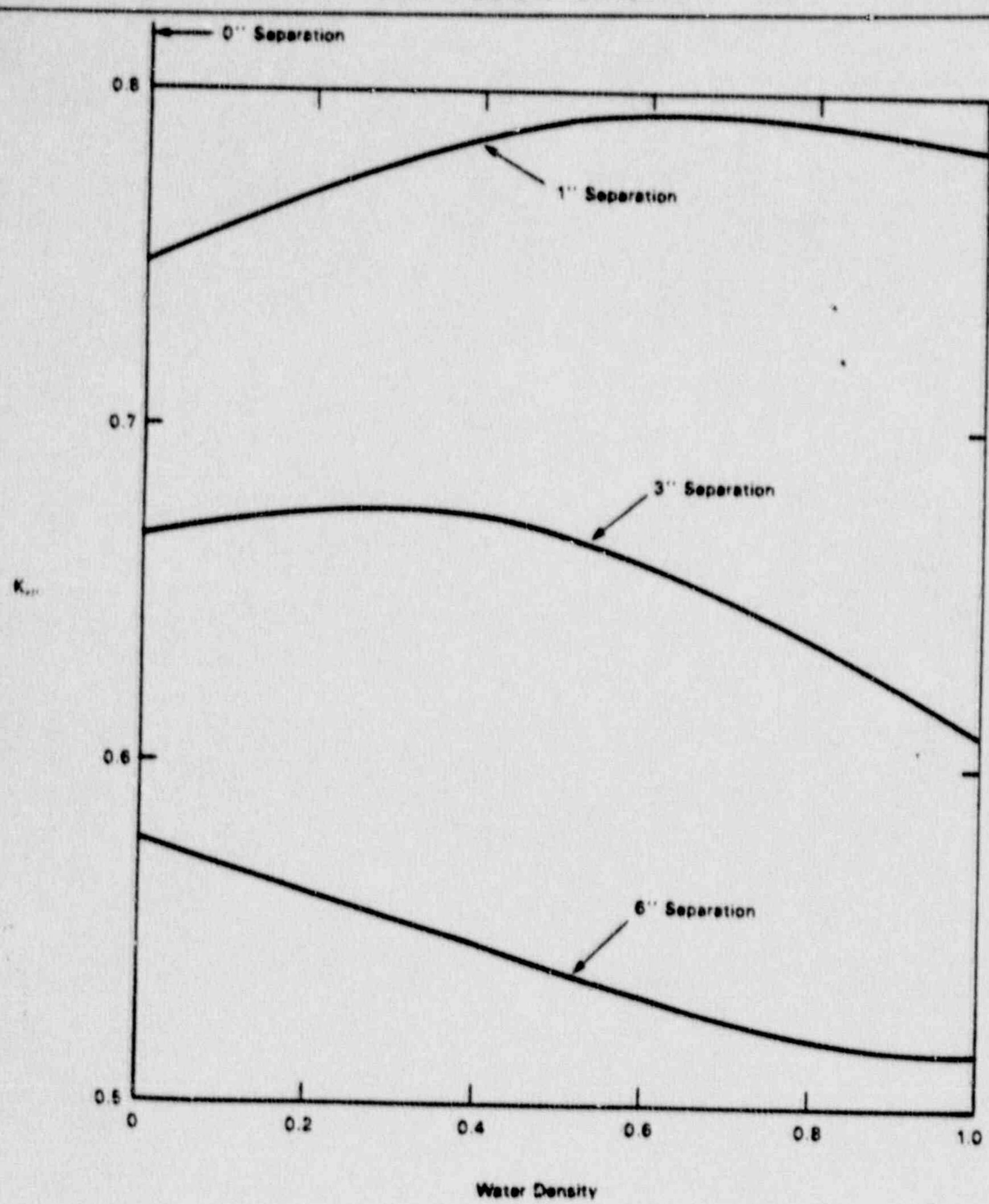


Figure 14-15

$K_{EFF}$ VERSUS WATER DENSITY AND PLANAR SEPARATION IN 4-INCH THICK INFINITE SLAB ARRAY OF PELLET TRAYS

PART II
SAFETY DEMONSTRATION

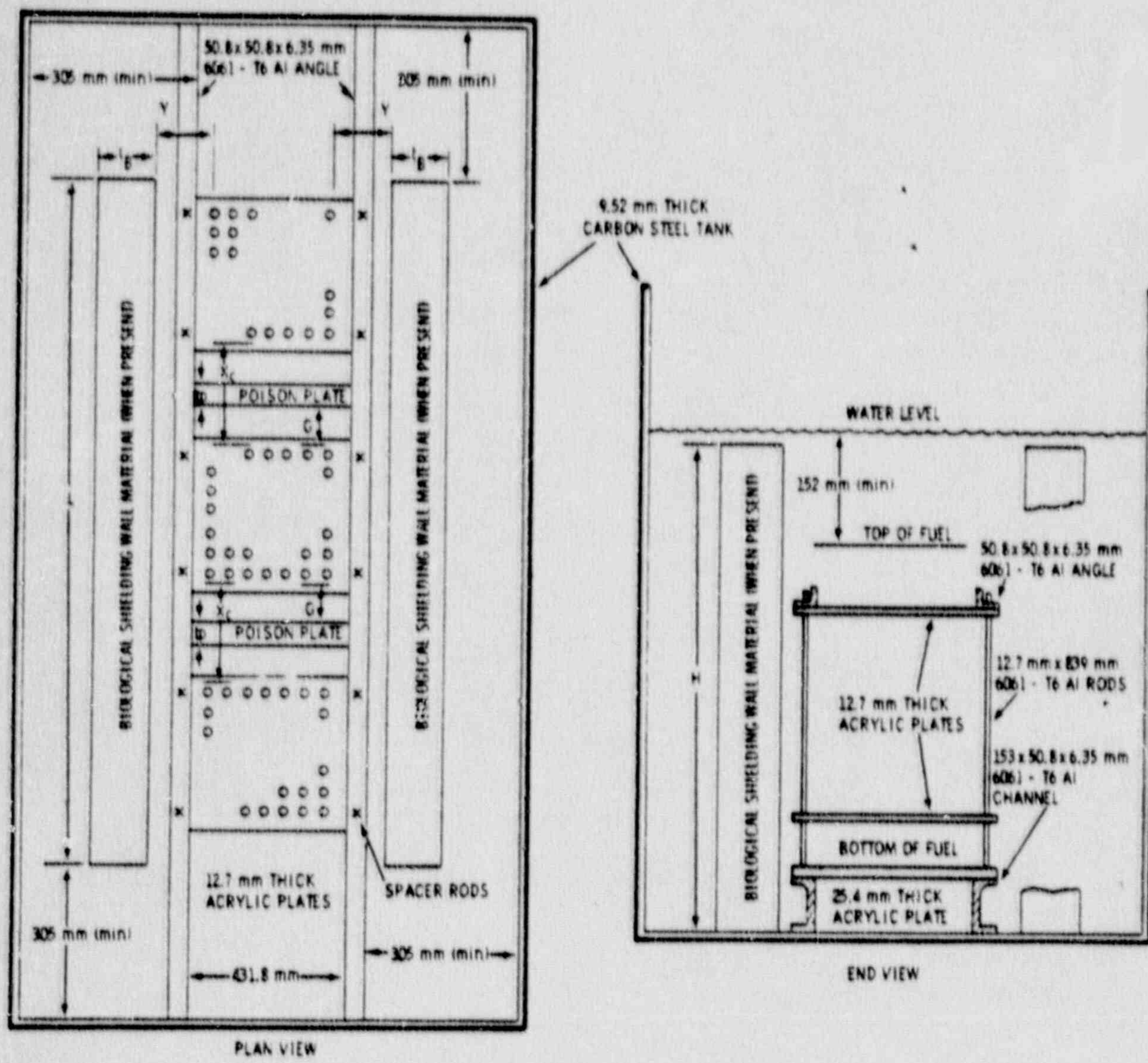


Figure 14-16

GRAPHICAL ARRANGEMENT OF SIMULATED SHIPPING
CASK CRITICAL EXPERIMENTS

PART II
SAFETY DEMONSTRATION

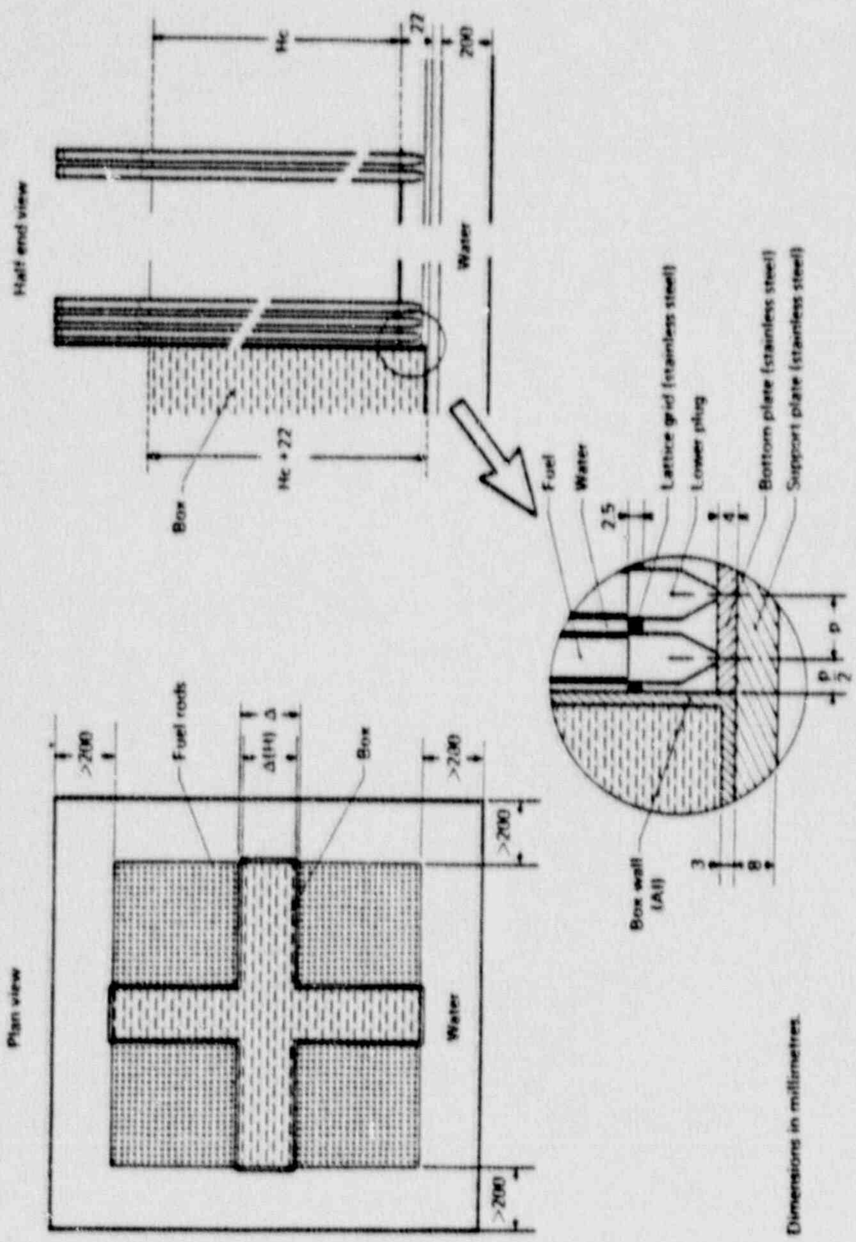


Figure 14-17

EXPERIMENTAL SETUP FOR EXPERIMENTS
OF REFERENCE 18

## CHAPTER 15 PROCESS DESCRIPTION AND SAFETY ANALYSES

This chapter describes SNM related operations beginning with receipt of UF6 through shipment of UO2 powder or pellets. Nuclear criticality safety is assured by implementation of the administrative conditions and technical criteria of Chapter 4. More specifically, one or more of the following techniques are employed in various process steps.

a) the use of safe subcritical units either individually or in arrays, i.e., safe geometry in individual units or arrays,

b) a limit on the maximum enrichment of 5 w/o U-235 in all manufacturing processes, and

c) the use of favorable geometry in combination with engineered safety features to maintain process variables within a safe regime.

Since UO2 enriched to 5 w/o U-235 or less requires moderation to achieve criticality, extensive use of moderation control has been employed even in cases where uncontrolled addition of water would not result in criticality.

### 15.1 Process Outline and Moderation Control

Figure 15-1 outlines the major steps in the conversion process and pelletizing lines. Moderation control is exercised in a large number of steps as indicated by the boxes enclosed by the broken line. Those steps are identified where engineered safety features are employed to preclude credible accident conditions involving the addition of water. Details of the moderation control features are addressed in the process descriptions.

Prevention of moisture addition is assured by a combination of design, automatic instrumentation, and administrative procedures in the form of Operation Sheets. Table 15-1 lists the measurements, allowed values, and automatic or operator actions for key steps in the process lines. By these means, the principle of double contingency is implemented, leading to the conclusion that violation of the "dry" powder criterion for UO2 hoppers and blenders is not credible. Under normal operating conditions, UO2 additives and moisture control are rigidly monitored to ensure product quality.

Buildings containing fissile materials do not have a fire sprinkler system. Fire fighting in moisture control areas is limited to dry techniques, i.e., no firehoses are allowed in these areas. In the event of flooding in the vicinity of these buildings, all operations are shut down.

## 15.2 UF6 to UO2 Conversion

This system is designed to convert uranium hexafluoride to UO2 granules by a fluidized bed "dry" conversion process. The equipment is currently qualified to handle a maximum enrichment of 5.0% U-235.

### 15.2.1 Receive and Store UF6

UF6 is received in standard 2-1/2 ton, 30-inch diameter cylinders in approved shipping packages. Upon receipt, the cylinders are placed on the UF6 cylinder storage pad which holds up to 54 cylinders. Eighteen additional cylinders may be located adjacent to the vaporizers near the cylinder scale, or in shipping packages on the Oxide Building dock. As required, a UF6 cylinder is removed from it's shipping package or storage and connected to the conversion equipment. The UF6 storage pad is separated from the UF6 Unloading Dock building by more than 12 feet.

Individual UF6 cylinders filled with 5.0 wt% U-235 are a safe moderation controlled subcrit as discussed in Section 14.3.1.1.2. On the storage pad they are spaced one foot apart in a planar array. This assures a safe configuration.

### 15.2.2 UF6 Conversion Process

#### 15.2.2.1 UF6 Vaporizers

Description - The UF6 vaporizers are located in the UF6 Unloading Dock building immediately adjacent to the Oxide Building. This area is enclosed by a metal frame building with a metal roof, metal walls, and overhead doors. Vaporization of the UF6 in a 30 inch cylinder is done by heating the cylinder in steam chamber. There are two chambers but only one cylinder is connected on-line at a time; when one cylinder is nearly empty, the second cylinder is placed on-line. A valving arrangement prevents the two cylinders from being interconnected.

A condensate line drains the steam chambers through an air gap to the drain. The drain line contains a conductivity cell and an automatic drain shut-off valve. A four inch diameter exhaust duct connects the steam chamber to a wet scrubber which is normally off.

During normal operation of the vaporizer, the UF6 gas leaves the cylinder through a 3/8 inch steam traced and insulated line into the Oxide Building. It passes through metering valves and is carried vertically along the wall to the third level of the Oxide Building and directly into the conversion equipment.

Safety Features - In the event of a UF6 leak into the vaporizer steam chamber, condensing steam will take SNM to the condensate line. When the conductivity cell in the drain line senses SNM, it will close the automatic shut-off valve, start the wet scrubber, and shut off the steam supply. Air, steam, and UF6 vapor from the steam chamber are ducted to the scrubber liquor in a 6-inch diameter ejector-venturi type scrubber. The separation of the condensate from the washed air is accomplished in a baffled separator, 23 inches by 9 inches by 15 inches deep. The condensate drains to a 9 3/4 inch internal diameter hold tank where it is recirculated to the ejector. The washed non-condensable gases are exhausted from the separator through a 6-inch duct to the atmosphere by a blower. Any overflow from the hold tank drains through a one-inch line to an eleven liter container.

Should a leak of UF6 occur to the atmosphere in the UF6 Unloading Dock building, the UF6 gas readily forms a visible cloud. A monitor on the wall alarms at the control panel, turns off the roof vent in the Unloading Dock and Oxide Buildings, and the pellet plant make-up air blowers. In the event of such a leak, an emergency alarm will be sounded, the area is evacuated and the emergency procedure put into effect. Self-contained breathing apparatus and protective clothing will be worn to correct the leak. The UF6 flow can be terminated from the control panel or at the vaporization chamber. Air sampling and decontamination will be done as required to cope with suspected or actual release of airborne activity.

Valve covers are not used in handling and storage of cold UF6 cylinders. Damage to the valve is very unlikely with the limited handling of filled containers. Even if a valve were to be broken or cracked, no significant release of UF6 would occur. The cylinders are filled under vacuum with a substantial void space remaining. The vapor pressure of UF6 at ambient temperature is less than the remaining vacuum so initial leakage would be inward. A crack would rapidly seal with uranyl fluoride from hydrolysis of atmospheric moisture. CO2 to freeze the UF6 and wooden plugs to replace a broken valve are readily available.

Nuclear Safety - A leak of UF6 into the steam chamber will not produce a criticality event because of the relatively small volume of the steam jacket. For example, if it is assumed that the steam jacket is filled with UF6 gas, it is equivalent to a layer of solid UF6 of thickness 0.017 inches. For this purpose the density of UF6 as a solid (5.1 g/cc) and gas (11 g/cc) were taken from ORO-651 (Rev. 4). Even if the system were fully reflected with water, the cylinder plus the assumed UF6 layer on the surface of the cylinder would not pose a criticality problem. In reality the UF6 and remaining steam would react and exit through the scrubber duct.

The steam chamber has drains to prevent the accumulation of condensate. The chamber is not pressurized, consequently the amount of moderation present is insignificant (approximately 0.04 pounds/ft$^3$) compared to full density water.

The 9.75 inch diameter hold tank on the wet scrubber is a safe geometry for UO2F2 (c.f. Table 4-2).

Although the baffled separator has a total volume of 51 liters, it cannot be filled with scrubber liquor because of: (1) the drain to the hold tank, and (2) the six inch diameter duct on the side of the separator venting the gases to the atmosphere.

15.2.2.2 <u>UF6 to UO2 Reactors</u>

<u>Description</u> - The UF6 to UO2 conversion is accomplished within the high temperature environments of the three series connected reactors: R-1, R-2, and R-3. These reactors are illustrated in the schematic flow diagram of the UF6 conversion process shown in Figure 15-2 and in the elevation sketch, Figure 15-3.

The R-2 vessel and furnace are shown in cross section in Figure 15-4. For comparison, the upper end configuration of vessels R-1 and R-3 are shown in dotted lines. The active (reactor) region of all three vessels is the same diameter (10 inch) as is the primary disengaging region (12 inch diameter), only R-2 has a secondary (16 inch) disengaging region.

All three vessels are fitted with electrically heated furnaces. R-1 and R-3 have two independently controlled sections and R-2 has three independently controlled sections.

Off-gases from the conversion process are routed to filtration systems to remove UO2 carryover and then to dry scrubbers packed with limestone to remove hydrogen fluoride.

All material transfers from one vessel to another are through piping having a diameter of 1.5 inches.

<u>Safety Features</u> - All three vessels have low temperature alarms and R-2 has a second (redundant) low temperature alarm. In addition, the steam line to R-2 has a second (redundant) shut-off valve, both of which are activated by the low temperature alarm from the R-2 controllers. Instrumentation is also provided for each reactor to indicate inlet gas pressure and differential pressure across the fluidized bed. These alarm on overpressure alerting the operator who then can take corrective action.

A loss of temperature could theoretically allow filling the 10" reactor, the 12" disengaging section, and, for R-2, the 16" disengaging section with condensed steam. The postulated failures which could cause loss of temperature and the engineered conditions which prevent steam condensation are as follows:

a. Thermocouple Failure Failure of a single thermocouple could shut off the power to one of the independently controlled heater sections. Since the temperature controller would fail up-scale, the high temperature would alarm and activate an interlock, shutting off the steam supply. The alternate heater sections would, in any case, provide sufficient power to the reactor to prevent steam condensation.

b. General Power Failure Failure of the power supply to the plant would cause the steam control valves to fail closed, terminating the supply of steam and cause the nitrogen control valve to fail open, purging all steam from the reactors.

c. Failure of a Single Heating Element Since the furnaces are connected to a three phase power supply, the most likely mode of failure is an open circuit in one of the three circuits leaving two circuits and the alternative furnace heating. This would prevent condensation. Furthermore, if the reactor temperature dropped below 600°F, the steam supply to R-2 would automatically shut off and the low temperature alarms would sound.

d. Massive Failure of Heating Elements A massive failure of all heating elements in all furnace sections for any reactor would result in low temperature alarms on all controllers and closing the inlet steam valves to R-2, if the failures were in the R-2 furnaces.

Failure of these "fail-safe" systems could allow condensate to enter the reactor. When liquid water is present near the bottom of a reactor with powder present, the 1/8" holes in the bubble caps will plug. The resulting high feed system pressure would automatically close the steam line and sound an alarm when the pressure reaches 18 psig. Further pressure increase would cause a rupture disc and relief valve to open at 20 psig. Should all the aforementioned fail to shut off the steam supply to R-2, the level of water would rise to the bottom of the lower disengaging section approximately five hours after the onset of condensation (more than eight hours after power loss). It is not credible to assume that this abnormal condition could go undetected for this length of time. Also, this amount of condensate could not collect unless the R-2 powder exit valve sealed and was not opened as required by procedures for unloading to the R-3 reactor at two hour, or less, intervals.

The concurrent failures of independent equipment and procedures described in each of these circumstances is deemed incredible.

Nuclear Safety - Under normal operating conditions, the SNM is in the lower (10-inch diameter) section of each reactor vessel when the SNM reacts with gases at high temperature. Although the diameter exceeds the safe cylinder diameter of Table 4-1, Figure 14-3 shows it to be less than the minimum critical diameter for 5 w/o enriched $UO_2$. The analysis presented in the Nuclear Safety part of Section 15.3.1 on filling of a bulk storage hopper assumes normal operating conditions in the three reactors and $UO_2$ in the two 12-inch diameter silos and three of the four 14-inch diameter blenders. In this analysis, the multiplication factor was calculated to be $0.2835 \pm 0.0050$. Thus, by deduction, the reactors are highly subcritical under normal operating conditions when $UO_2$ granules are present in the 10-inch diameter active region.

The disengaging section(s) of each reactor is of larger diameter. Consequently, it is expected that if $UO_2$ and condensed steam were allowed to fill not only the 10-inch diameter region but also the disengaging section(s), a more reactive condition would result.

It should be noted that a lengthy sequence of failures would be required over a period of eight hours or more for a so-called overfill condition to be created. These failures are enumerated below for the R-3 reactor.

1) Failure of all heating elements,
2) Failure of low temperature alarm,
3) Failure of operator to observe R-3 temperature indicator,
4) Failure of system pressure control system to close steam control valve,
5) Failure of high pressure alarm,
6) Failure of high pressure switch to close steam inlet valve,
7) Failure of operator to observe R-3 pressure indicator,
8) Failure of rupture disc to rupture,
9) Failure of operator to follow procedures which require unloading R-3 reactor every 2 hours, and
10) Failure of valve allowing R-2 to continuously empty $UO_2$ into R-3.

To assess the consequences of postulated malfunctions in the engineered safeguards in combination with a failure of the operator to take the required corrective action such that an overfill condition does develop in a reactor with the 12-inch diameter disengaging section, a KENO-IV analysis of an isolated R-3 reactor is carried out. It is assumed in this analysis that $UO_2$ granules and water fill both the 10 and 12-inch diameter sections of the vessel and that the $UO_2$ granules are suspended in the water so as to yield an optimum moderated solution. The details of the KENO geometric model are shown in Figure 15-5; axial dimensions appropriate to the R-3 reactor are employed. It should be noted that structures such as the vessel wall, heaters, fire brick insulation, and carbon steel casing are included in the model to obtain a more realistic net neutron boundary current for the 5 w/o enriched UO2-water mixture. Water reflection of the vessel casing is not considered credible because of the elevation of R-3 (see Figure 15-3).

The resulting KENO-IV effective multiplication factor is 0.9510 ± 0.0055 for this postulated accident scenario.

The same postulated overfill conditions in R-2 would result in a critical multiplication factor for this reactor in isolation because of the 16-inch disengaging section in addition to the 12-inch disengaging section as on R-3. Consequently, a redundant low temperature alarm setpoint and steam shut-off valve are added to the engineered safeguards to provide added assurance that the overfill condition could not occur on R-2.

15.2.2.3 <u>UO2 Cooler</u>

<u>Description</u> - The UO2 cooler is a water cooled heat exchanger for reducing the temperature of the dry UO2 granules exiting R-3 via the inline R-3 discharge hopper. The cooler has a water cooled jacket and a water cooled screw drive mechanism. The screw drive is activated by a temperature sensor in the cooler discharge hopper; if the temperature of the UO2 in the hopper becomes too high, the screw drive is turned off. There is a coolant flow monitor in the water discharge line that alarms in the control room for a loss of flow condition. The cooler discharge hopper contains a level control which has three functions: (1) to close the cooler hopper discharge pinch-valve when the level is too low, (2) open the discharge valve when the level is too high, and (3) after a short time delay, the screw drive is stopped if the hopper level remains too high.

<u>Safety Features</u> - UO2 entering the cooler is normally at a temperature greater than 1000°F. Thus, as long as the mechanical integrity of the UO2 region of the cooler is maintained, the moisture level of the UO2 should remain well below 0.05 w/o water. The cooler discharge hopper has a moisture measuring device to monitor the moisture content of the gas above the UO2 level in the hopper. If the dewpoint in this gas rises above 15°C, an alarm is sounded in the Oxide Building and, if the UO2 is being directed to the 14 inch diameter receiver, the vacuum transfer of material exiting the cooler hopper to the 14 inch diameter receiver (Blender No. 4) is automatically halted by shutting off the vacuum blower. If UO2 is being directed to the silos, the sounding of the alarm will result in operator action to shut down the transfer until the cause of the alarm can be resolved.

The principal failure mode of this humidistat is in the direction of indicating a high dewpoint temperature which is interpreted as a high moisture level. Should this occur, the vacuum transfer of UO2 is halted as noted above.

Criticality Safety - The oxide cooler and the cooler discharge hopper are 8 inches, or less, in diameter. Thus, these units are safe geometry units.

15.2.3 UO2 Receivers

Description - The UO2 product from the cooler is in the form of granules having a mean diameter of approximately 100 microns. These granules are pneumatically transferred to the two tall 12-inch diameter silos or to a 14-inch diameter receiver vessel (formally Blender Vessel No. 4); both types of receivers are located in the Oxide Building. The silos are employed as the receivers when the production run requires a milled powder to be shipped as powder or fabricated into pellets using the building 255 pelletizing line. Alternatively, the granules are transferred to the 14-inch receiver when the UO2 is to be processed in the building 254 pelletizing lines.

Safety Features - The two twelve-inch diameter 21.5 foot high silos are four feet apart on centers and their bottom tapers begin above the second floor (10'7"). The four fourteen-inch diameter, twenty foot high blender vessels are in-line and spaced six feet apart on centers. Their bottom tapers begin at about seven feet above the first floor level. Thus, reflection by water is highly improbable.

During normal operation the UO2 granules exiting the R-3 discharge hopper and entering the UO2 cooler are at high temperature and dry. The first break in the UO2 process environment occurs after the UO2 cooler discharge hopper where the UO2 enters the pneumatic transfer line through a pinch valve. Here, room air enters the line to carry the UO2 granules to either one of the silos or the fourteen-inch receiver (Blender No. 4), depending upon the transfer line connections, when the suction head has been activated at the receiver vessel. Moisture pickup by the UO2 from the room air, even if the latter were at 100% humidity, would be less than 0.5 w/o. Other potential moisture ingress points are the ducts at the top of the silo and blender vessels employed for dust control (HEPA ducts) and the inlet from the vacuum blowers for pneumatic transfer. Moisture ingress through these ducts is highly improbable and would require physical damage to the vent system.

Nuclear Safety - The discussion of the UO2 cooler in Section 15.2.2.3 noted the presence of the humidistat in the cooler discharge hopper and its function in monitoring UO2 moisture level. In the event of a postulated failure of the humidistat at the same time as a breach of the mechanical integrity of the water barriers in the UO2 cooler, the loss of the monitor could be compensated for by the impaired ability to transfer wet UO2 to the receivers. Laboratory tests with UO2 granules containing varying amounts of water indicate the UO2 granules flow freely when the water content is a few tenths of a percent. As the water content increases, noticeable agglutination occurs. At 6 percent water, the material moves as clumps. The pinch valve at the outlet of the cooler hopper, the limited suction head in the pneumatic transfer lines, and the high elevation of the inlets to the silo and blender vessels all diminish the ability of the system to transfer material out of the cooler discharge hopper to the receivers when the granules are not free flowing.

The batchwise flow of UO2 granules through the cooler to the receivers requires consideration of the case where transfer of the batch of granules to the receiver has been completed at the time of the postulated failures to the humidistat and cooler in the previous paragraph. In this event water enters the cooler discharge hopper and is not impeded by agglutination of the UO2. Even in the event that the pneumatic transfer suction head is insufficient to support a column of water high enough to reach the inlet port of the receiver vessels, water could transfer by droplet entrainment and/or film flow along the walls of the transfer lines upon activation of the pneumatic transfer system. However, when UO2 enters the screw cooler hopper along with water from the break in cooler integrity, plugging of the pinch valve can be expected.

Even though these receiver vessels have diameters which exceed the maximum diameter of a SIU, criticality will not occur if excess moisture is present. The bulk density of the UO2 granules is in the range of 3.0 to 3.5 g/cc. Reference to Figure 14-14 shows that an infinite cylinder of 5 w/o enriched UO2 requires a diameter of greater than 14.4 inches for criticality to occur when UO2 of bulk density 3.0 g/cc is saturated with water. Note that unreflected data are employed here because of the elevation of these vessels above the main floor and all SNM operations are halted should flooding occur.

Since the silo diameter is only 83 percent of this value and system failures beyond cooler integrity are required to achieve this condition, the double contingency principle is satisfied.

For the case of the 14 inch diameter receiver vessel, a batchwise mode of operation is employed. As each batch is transferred to this receiver, it is sampled for moisture content prior to transferring it to the bulk storage hopper. The nominal batch size is 100 Kg but they may vary from 80 to approximately 130 Kg. Batch size amounts of UO2 granules in the 14 inch receiver can be approximated as a sphere and, for saturated UO2 at a bulk UO2 density of 3.0 g/cc Figure 14-14 shows that the mass of UO2 must exceed 185 Kg before criticality is achieved. Since the 130Kg batch size is only 70% of this quantity and system failures beyond cooler integrity must be postulated, the double contingency principle is satisfied.

Since the receiver vessels and reactor vessels are in the same room, interactions between the vessels can occur. To assess the consequences of an overfill condition in R-3 (closest reactor to silos and blender vessels) on the receivers, a KENO calculation of the Oxide Conversion Building was carried out using the following conservative assumptions.

a) Reactors R-1 and R-2 were assumed to be filled in the 10" portion (i.e., no overfill) with UO2 at 2.5 g/cc density of powder, 5.0 w/o U-235 and 0 w/o H2O. All structures consisting of .375" steel wall, 7.75" of 37.5 $lbs/ft^3$ firebrick insulation and .25" steel casing were including in the model. The reactor details are shown in Figures 15-3 and 15-5 were used.

b) The R-3 reactor was assumed to be filled in both the 10" and 12" portions (i.e., overfilled) with wet UO2 at 2.5 g/cc powder density and 5.0 w/o U-235. The UO2-H2O mixture is equivalent to a UO2 volume fraction of 0.23 and a H2O volume fraction of 0.77. All structures consisting of .375" steel wall, 7.75" firebrick insulation and .25" steel casing were included in the model.

c) The cooler was assumed to contain UO2 with 5.0 w/o water. The .125" steel walls were also modelled.

d) The silos were assumed to contain UO2 with 5.0 w/o water. The .125" steel walls were also modelled.

e) The UO2 blenders contained UO2 with 5.0 w/o water. The .125" steel walls were also modelled.

f) The UF6 scrubber was assumed to contain UO2 with no water and no external structures modelled.

g) The R-1 hopper was assumed to be filled with dry UO2 and surrounded by 1" of water.

h) An external mist of .001 g/cc was assumed.

The multiplication factor using KENO-IV and the 16 broad group Hansen-Roach cross section set was $0.9714 \pm 0.0058$. A multiplication factor of this magnitude is not surprising since in Section 15.2.2.2 it was noted that for an isolated R-3 reactor in the postulated overfill condition, the multiplication factor was $0.9510 \pm 0.0055$. The condition of the receiver vessels appears to have little impact on the results even when the receivers have up to 5 w/o water in the UO2. The analyses reported in Section 15.3.1 expand on this observation and indicate the degree of subcriticality of this operation under normal conditions.

### 15.3 Building 254 Powder Preparation and Pellet Fabrication

#### 15.3.1 Filling of Bulk Storage Hoppers

Description - Virgin UO2 granules from a production run intended for pelletizing in Building 254 are transferred by gravity from the 14-inch diameter receiver in the Oxide Conversion Building into 30-inch diameter bulk storage hoppers (Figure 15-6). To make the transfer, a bulk storage hopper is connected through a flange and flexible connector to the valve at the base of the receiver. After filling, the hopper is disconnected from the receiver, the hopper filling connection is closed with a gasketed blank flange cover, and the hopper is transported to a storage location.

Safety Features - Filling of the bulk storage hopper is done within a hooded enclosure at the base of the 14 inch receiver. The hood prevents the ingress of extraneous material and exiting of UO2 dust during the connecting and disconnecting of the flexible coupling, and during the closing of the hopper.

The hopper sits on a scale during the loading process to monitor the UO2 content; the initial weight of the hopper is compared with the known tare weight to verify that the hopper is empty.

At intervals of approximately two hours, a batch of UO2 granules weighing about 100 Kg is discharged from the conversion reactors. After each batch is cooled and transferred to the receiver, it is sampled to verify that the moisture content of the UO2 granules meets the criterion of dry UO2, i.e., a moisture content of less than 1 w/o water. During the sampling process, the receiver is isolated from the cooler since the pneumatic transfer device is off. The moisture content in the sample is measured and recorded. A second person, who is not an operator, overchecks the analysis results. If the batch of UO2 is verified as dry UO2, the second person allows dumping of the oxide from the receiver into the bulk storage hopper, otherwise production is halted and corrective measures implemented. Engineered means positively prevent dumping of oxide until a satisfactory overcheck is made and release is permitted.

Nuclear Safety - The nominal content of a filled bulk storage hopper is 1000 Kg of UO2. Consequently, this is a moderation controlled container and UO2 entering the hopper has been verified as being dry at two points prior to entering the hopper:

1) by inference from the humidistat at the exit of the UO2 cooler and prior to entering the receiver (also see Section 15.2.3), and
2) sampling in the receiver prior to discharge into the hopper.

Typically, ten dumps of ten batches are made to a bulk storage hopper. Excessive water in one single batch would wet the entire batch and be observed in a sample taken near the bottom. Bottom sampling of a batch conceivably may not assure that the upper region of that batch has less than 1 w/o moisture. However, a moisture problem that initiates while a batch is being transferred into the receiver would likely continue and would be measured in the next batch. Measurement on each batch before each of ten sequential dumps to the hopper provides assurance that a moisture level above 1 w/o in a significant quantity of the oxide in the hopper does not occur.

After the hopper is loaded, moisture in the hopper is controlled by equipment design. The hood arrangement at the base of the receiver prevents water ingress during the disconnect process and prior to fixing the flange cover. The flange cover prevents moisture ingress during transport and storage.

To assess the criticality safety of the bulk storage hopper filling operation, the following KENO analysis of the Oxide Conversion Building is provided. In this mode of operation of the oxide room, the three reactors and UO2 cooler are functional; a piping change at the outlet of the UO2 cooler is made to bypass the two silos and to feed the receiver.

The following conservative assumptions are used in the calculation model of the UF6 to UO2 conversion equipment for normal operating conditions:

a) Reactors R-1, R-2 and R-3 are assumed to be filled in the 10" portion (i.e., no overfill) with UO2 at 2.5 g/cc density of powder, 1 w/o H2O and 5.0 w/o U-235. All structures consisting of .375" steel wall, 7.75" of 37.5 lbs/ft$^3$ firebrick insulation and 0.25" steel casing are included in the model. The reactor details are shown in Figures 15-3 and 15-5.

b) The cooler is assumed to contain UO2 with 1 w/o H2O and is surrounded by 1/2" of water. The steel structure is not modelled.

c) The two silos are assumed to be filled with UO2 containing 1 w/o H2O. The .125" steel walls are modelled.

d) UO2 blender No. 3 is empty and Nos. 1, 2, and 4 contain UO2 with 1.0 w/o water. The .125" steel walls are modelled.

e) The UF6 scrubber is assumed to be empty with no external structures modelled.

f) The R-1 hopper is assumed to be filled with UO2 containing 1 w/o H2O water. Steel structures are not modelled.

g) An external mist of 0.001 g/cc is assumed.

h) A bulk storage hopper containing 1000 Kg of 5 w/o UO2 with 1 w/o H2O is modelled below Blender No. 4.

The multiplication factor computed by the KENO-IV code with 16 group Hansen-Roach cross sections is 0.2835 ± 0.0050. Thus, for a conservative representation as to: (1) the normal mode of operation of the three reactors, and (2) the number of filled silos and blenders when filling a bulk storage hopper in the Oxide Building, the processing system is highly subcritical.

For comparison, we examine the case of a postulated overfill condition in R-3, accompanied by water saturated UO2 in the cooler and 5 w/o water in the UO2 within a filled receiver (Blender No. 4) above a filled bulk storage hopper containing UO2 with 1 w/o H2O. The two silos and three remaining blenders are empty and the scrubber and R-1 weigh hopper are each assumed to be filled with UO2 containing no water. All UO2 is enriched to 5 w/o U-235. The calculated multiplication factor is 0.9744 ± 0.0032. Another analysis employs the same conditions in the various components of the Oxide Conversion Building as noted in the above calculation for normal operating conditions except R-3 was in an assumed overfill condition. The multiplication factor was calculated to be 0.9709 ± 0.0029. As in the case reported in Section 15.2.3, the overfill condition in R-3 was the most reactive component of the system and the differing conditions in the silos and blenders as well as the presence or absence of the bulk storage hopper have little effect on the magnitude of the multiplication factor. When the assumed UO2 density in R-3 during the overfill condition goes from 2.5 to 3.5 g/cc, the multiplication factor decreases by 0.01.

15.3.2 Transporting of Bulk Storage Hoppers

Description - After filling and closing, the bulk storage hopper is transported out of the oxide room, through the north corridor of Building 255 to the north end of Building 254. Loaded bulk storage hoppers may also be stored in the north corridor of Building 255.

Safety Features - The bulk storage hopper is a closed container which is safe even if immersed in water. The mechanical design of the hopper is intended to maintain integrity of the contents during transport and storage. The walls of the container are 1/4 to 5/16 inch thick stainless steel, a tough, ductile material. In addition to the support legs, a supplemental structure has been provided to protect the exit valve stem.

Nuclear Safety - The east-west corridor between the north wall of building 255 and the virgin powder can storage conveyors is 110 inches wide. Eight and one half feet above the corridor floor on a one-quarter inch thick steel mezzanine floor are the storage rings for positioning the closed agglomerated feed buckets for the Building 255 pellet presses on 24 inch centers. The number of storage positions is 48. To assess the criticality safety of the combined arrays of: (1) conveyors loaded with powder cans, (2) 48 agglomerated press feed containers, and (3) bulk storage hoppers stored and/or in transit through the north corridor, a KENO-IV calculation with Hansen-Roach cross sections is employed with the following assumptions. (See Figure 10-1.)

a. The 48 agglomeration press feed containers are arranged in a 6 x 8 array on the mezzanine. The spacing between containers is 2 feet. Each container contains 41 kilograms of $UO_2$ enriched to 5 w/o U-235 and the remaining volume is water. Note that the volume occupied by the $UO_2$ in the homogeneous $UO_2$-$H_2O$ mixture is based on a theoretical $UO_2$ density of 10.96 g/cc.

b. The powder cans located on the 3 double height conveyors are modeled as 6 slabs. The height and width of the slabs are equal to the height and width of the virgin powder cans (height = 11 inches and width 9.75 inches). The resulting slab contains 1 w/o water.

c. The bulk storage hoppers under the mezzanine are modeled as a 96 inch wide slab. The height of the slab is modeled as the height of the top of the oxide level in a loaded bulk storage hopper, 53 inches.

d. The roof of the building is represented as a 12 inch slab of water.

e. The north, east, and south walls are modeled as 12 inches of concrete. The west wall is modeled as a 12 inch slab of water.

f. No structural material is modeled.

The KENO-IV multiplication factor is $0.847 \pm 0.004$. Based on this analysis, it is concluded that no criticality problem exists in the north corridor of Building 255 during transport of a loaded bulk storage hopper through the corridor to building 255 even if the corridor is filled with loaded bulk storage hoppers.

### 15.3.3 Milling and Blending

Description - In Building 254 there are two parallel pellet lines. Each has the same equipment and the same controls including one virgin oxide unload hood, one recycle unload hood, one mill, three blenders, one pellet fabrication line, and one scrap recycle line.

The recycle unload hood is designed to accommodate the recycle storage hopper shown in Figure 15-7. This hopper is smaller than the 30 inch diameter bulk storage hopper and is designed to hold about 100 Kg of recycle UO2. Recycle oxide produced within the pellet line is oxidized, reduced, and milled. This material is placed in safe volume containers and transferred to Building 253 where it may be loaded into the recycle storage hoppers. Recycle material from the Windsor Facility is received in safe volume containers and processed in the same manner

The oxide blenders may receive oxide from two sources - the virgin oxide bulk storage hoppers and the recycle hoppers. A hopper is lifted above either the virgin oxide unload hood or the recycle unload hood and oxide is fed from the hopper to the mill (micronizer) by means of a vibratory feeder. The powder passes through the micronizer and is transferred pneumatically to the oxide blender. The transfer air is filtered locally and is discharged through the plant HEPA filter system.

The milled oxide from several storage hoppers accumulates in the oxide blender (Figure 15-8). It is subsequently blended into a homogeneous batch by the action of air jets located at the lower end of the blender. Plant air is employed for the blending operation. Blend air exits through the same filters as did the transfer air. Periodically the filters are blown back by a pulse of plant air to remove accumulated oxide.

After blending, the oxide powder passes through a rotary valve at the bottom of the blender and is transferred pneumatically by a vacuum system up to the first stage of the pelletizing cycle. The vacuum system draws room air from the vicinity of the rotary valve for the transfer.

Safety Features - Milling and blending is a moderation controlled operation. The bulk storage hoppers contain "dry" UO2; the recycle hoppers also meet the same criterion based on the following operating procedures.

Recycle oxide powder is accumulated in safe volume containers in Building 253 and periodically is combined into a wheeled recycle hopper. Each container is sampled to verify that the moisture content is no greater than 1.0 weight percent. Also, the recycle hopper is visually inspected when it is loaded into the transfer hood to verify that it does not contain water. When the hopper is filled, the oxide is blended by tumbling and resampled to verify that the moisture content is no greater then 1.0 weight percent. The hopper is then closed with a blank flange cover and wheeled to storage. After it is verified that the moisture content in the blended oxide sample is no greater than 1.0 weight percent, the hopper may be released to Building 254.

There are no water lines in the north end of Building 254; the closest water line is along the north edge of the center mezzanine which is south of the dewaxing furnaces (See Figure 10-1). Thus the only water is that contained in cleaning buckets.

By design of the unload hood, water is prevented from entering the oxide stream during unloading of the bulk storage and recycle hoppers. The only other entrance path for moisture at the unload step is via the plant air supply employed for the micronizer or room air employed to supplement the micronizer air in effecting pneumatic transfer. Other potential sources of moisture ingress to the 48 inch diameter blenders include plant air supply used for blending and pulsed filter cleaning, the suction head vent for pneumatic transfer from micronizer to the blender, and the dust control vent (to HEPAs).

The plant air supply compressor employs moisture separators and dryers that lower the exit air dewpoint to approximately minus 40°C. A moisture detector located at the exit of the air dryer alarms in Building 254 if the dewpoint rises above 0°C. A second detector is provided after the air accumulator feeding the milling, blending and pulsed filter cleaning in Building 254. The second detector alarms at a dewpoint of 0°C and also stops the vibratory feeders at the unloaders and shuts off air to the oxide blender.

Room air in combination with plant air is employed for the pneumatic transfer from the micronizer to the blender. This room air, even if at 100% humidity, cannot contribute sufficient moisture via the hygroscopic nature of the micronized $UO_2$ to violate the dry powder criterion. Although the suction head vent (for pneumatic transfer from the micronizer) is a potential point of moisture ingress, no credible sources of moisture inlet to this line can be postulated. The dust control vent going to the HEPAs does exit to the atmosphere on the roof, however, this ducting is baffled to prevent water flowing back through the system to the blender. Thus, mechanical design, independent measurements, and automatic instrument action assure that moisture is not injected into the large blenders during the blending process.

Another potential point of water ingress to the large blenders is through the top flanged cover, should it be removed for any reason. Removal of the flanged cover plate and leaving it unattended when $UO_2$ is in the blender is a highly unlikely event.

Building 254 is a free standing steel frame structure designed using a BOCA seismic zone 2 earthquake importance factor of 1.5, thereby limiting potential damage to the oxide vessels by structural failure. Additionally, the blender support structure is independent of the building support structure.

Nuclear Safety - See nuclear safety discussion of Section 15.3.4 for results of a KENO analysis of the north end of Building 254. It is noted that, by deduction, the recycle waste hopper when loaded with "dry" UO2 presents no criticality problem. This conclusion is based on the array analyses of the bulk storage hoppers in Section 15.3.4 which have an areal density of UO2 equal to 7.5 times that of the recycle waste hoppers.

### 15.3.4 Building 254 Agglomeration, Granulation, and Pressing

Description - Post blending process steps through pellet pressing are contained in a vertical column beginning with the blended UO2 powder receiving hood on the second mezzanine and ending with the rotary pellet press on the main floor (See Figure 15-13). The various steps in the process provide for addition of poreformer and press fines (plus other process line make-up materials) at the hood on the second mezzanine, a conical screw mixer, a granulator, the lubricant and granulator feed hood on the first mezzanine, the lubricant mixer, and the pellet press feed hopper. The green pellets emerging from the rotary press are loaded into sintering boats.

Safety Features - The pellet press line is a batchwise gravity feed operation with two access points to the UO2 process stream - the poreformer and press fines recycle hood on the third floor and the lubricant and granulated feed hood on the second floor. All materials entering the process line at these points are strictly controlled by operating procedures to maintain product quality. Product quality controls are more stringent than criticality controls with the container sizes in this vertical process stream.

Nuclear Safety - To assess the criticality safety of the front-end of the pelletizing line in combination with filled bulk storage hoppers located on the ground floor of the north end of Building 254 between the south edge of the second floor and the north wall, KENO analyses are carried out using Hansen-Roach cross sections. Fixed equipment, i.e., hopper unloaders, micronizer, blenders and the vertical stacks of UO2 process equipment from the vacuum receivers on the third floor down to the rotary presses on the ground floor are represented in a conservative manner in KENO as are the filled bulk storage hoppers. These analyses are supplemented by other KENO calculations for various arrays of filled bulk storage hoppers. The multiplication factors are well below 0.95 for the set of conservative nominal conditions and are acceptable for more adverse conditions.

A summary of the assumptions employed in the KENO model follows.

a) Figures 10-1 and 15-3 show the locations of the fixed equipment at the front-end of both pelletizing lines. Figures 15-6, 15-7 and 15-8 show the dimensions and Figures 15-9, 15-10 and 15-11 show the models for the bulk storage hoppers, the blenders and the north end of Building 254.

b) The elements employed are UO2, water, concrete, pore former, and lubricant; no credit is taken for scattering or parasitic absorptions by structural materials in the building or equipment.

c) The floor, north, west and east walls of Building 254 are taken to be 12 inches of ordinary concrete; the ceiling (35.5 feet) and south heat wall are represented as 12 inches of full density water.

d) Fifty-four filled (29.375" I.D.) bulk storage hoppers (four are on the unloaders) and six (47.25" I.D.) blenders are arranged as illustrated in Figure 15-10; bulk storage hoppers are used instead of the recycle hoppers on the recycle hopper unloaders. The two pneumatic transport lines are conservatively represented as a 15 inch cylinder extending from floor to ceiling at the micronizer location. The vertical array of equipment from the vacuum receiver on the third floor down to the rotary press on the ground floor is also represented as a right cylinder of 15 inch diameter extending from floor to ceiling at the rotary press location.

e) All UO2 regions contain UO2 having a density of 3.5 g/cc, at an enrichment of 5 w/o U-235 and 1 w/o water. In addition, two and one half times the maximum amount of poreformer and lubricant are present in the UO2 cylinder representing the vertical pelletizing array. Each bulk storage hopper contains 1000 Kg UO2 and each blender is assumed to contain 4200 Kg UO2.

f) A significant fraction of the bulk storage hoppers are clustered closer to the blenders than practical from an orderly access point of view so as to maximize the interaction between hoppers and blenders.

g) Thirteen inch slabs of UO2 enriched to 5 w/o U-235 were assumed on the second and third floors of the building. The slabs contained 1.0 w/o water, 1.0 w/o starch, and 1.0 w/o zinc sterate. The 13 inch slab conservatively represents buckets of UO2 that may be stored on these floors in safe arrays.

The multiplication factor for the above set of conservative conditions was $0.860 \pm 0.004$.

Additional analyses were carried out to assess the effect of the addition of moderating media on the second and third floors. In an extension of the above analysis, the space above the UO2 slab on the third floor was filled with water; an additional analysis extended the water to the available volume above the UO2 slab on the second floor. The resulting multiplication factors were 0.8828 ± 0.0042 and 0.8161 ± 0.0048, respectively. It is concluded from this analysis that no limits on containers of moderating material on the second and third floor are required for criticality safety but may be imposed for operational reasons.

KENO calculations are also employed to examine infinite and finite arrays of bulk storage hoppers to examine possible limitations on storage configurations. The following assumptions are employed in the KENO models.

a) Figure 15-9 shows the KENO model for an individual hopper. No structural materials are included.

b) Hoppers are spaced on 30 inch centers.

c) Each hopper is assumed to contain 1000 Kg UO2. The UO2 is taken to be of 3.5 g/cc density, 5.0 w/o U-235, and 1 w/o water.

d) The model employs a one foot thick ordinary concrete floor and a 12 inch water region at ceiling height (35.5 feet).

Results of the KENO calculations are as follows:

a) An infinite planar array - k = 0.833 ± 0.004

b) An infinite planar array with hoppers overfilled to 1378 Kg UO2 - k = 0.867 ± 0.004

c) A 7 x 7 array of bulk storage hoppers at 1000 Kg / hopper reflected by concrete walls - k = 0.686 ± 0.004

d) Isolated hopper surrounded by a one foot radial water reflector - k = 0.571 ± 0.003

Based on these analyses, it is concluded that overfill of a hopper, should it occur, has only a small effect on the multiplication factor in a large array of hoppers on 30 inch centers. It is also concluded that a realistic size array of filled bulk storage hoppers loaded with UO2 of 5 w/o enrichment and 1 w/o water is highly subcritical when the hoppers are in physical contact within the array.

15.3.5 Building 254 Dewaxing and Sintering

Description - The green pellets are dewaxed and then sintered to obtain the specified ceramic properties. Pellet boats are charged in a single line through the controlled atmosphere furnaces.

Nuclear Safety - Green pellets are loaded in a random fashion in the sintering boats so as not to exceed the safe slab SIU limits listed in Chapter 4.

15.3.6 Building 254 Grinding and Inspection

Description - Sintered pellets are transferred to the grinder feed system and ground under a stream of coolant. After grinding, the pellets are transferred on a conveyor to trays which are then moved to the inspection area. Finished pellet inspection may include an alternate optical measurement of pellet dimensions. After inspection, the pellets are stored in trays prior to packaging in shipping containers.

Grinder sludge is removed from the coolant by a centrifuge; it is subsequently dried and stored for further disposition. Coolant from the centrifuge collects in a sump and is pumped back to the grinder. The centrifuge is cleaned periodically, as required, to permit continued operation.

Safety Features - A complete enclosure is provided around the grinder to preclude dusting of UO2. This enclosure is maintained at a slight negative pressure with respect to the room. The centrifuge is limited to a safe volume and is provided with a spacing area of 8.5 square feet.

The centrifuge sump has an operating volume of 22 liters and a maximum volume of 24.6 liters. It is provided with a spacing area of 9 square feet.

Grinder sludge is stored in mass limited SIUs.

Pellets move through the grinding and inspection operations in a safe slab geometry.

15.3.7 Building 254 Packaging

Description - Pellets are arranged onto corrugated trays, or loaded into pans that are stacked in a lifting cradle. The cradle is weighed and then lowered into a shipping container through a transfer port that separates Building 254 from the clean warehouse Building 256. The pellets are packaged in licensed shipping containers in accordance with the applicable certificate of compliance.

Nuclear Safety - Pellets awaiting packaging are stored in a safe slab configuration.

### 15.4 Building 255 Powder Preparation and Pellet Fabrication

#### 15.4.1 Milling

Description - UO2 granules stored in the silos in the Oxide Building flow by gravity to the mill. Process milling equipment consists of 10-inch diameter hoppers which taper to three inch discharge openings to the mill. Scrap recycle charge containers are five gallon pails (19 liters) which are emptied into the tapered hoppers for discharge to the mill.

Safety Features - In Section 15.2.2.3 it was noted that in the event excess moisture was detected in the screw cooler discharge hopper an alarm was sounded in the Oxide Building. Operator action upon hearing the alarm is to interrupt the transfer of UO2 to the silo until the cause for the alarm is resolved. Plant air is employed in the milling operation. This air supply employs moisture separators and dryers that lower the exit air dewpoint to approximately minus 40°C. A moisture detector at the exit of the dryer alarms if the dewpoint rises above 0°C.

Room Air, in addition to the plant air input to the mill, is employed in the pneumatic transfer of the micronized powder to the blenders. Any moisture pick up from the room air due to the hygroscopic nature of the UO2 would be well below the level necessary to raise any concerns for this operation.

Scrap recycle material entering the mill is handled in safe volume containers and is sampled to verify that the moisture content is no greater than 1 w/o water.

Nuclear Safety - Criticality safety of the milling operation is assured by the use of safe geometry containers and at least a four foot separation between the milling operation and other SNM bearing equipment. As to the safe geometry containers, the recycle material containers are safe volume. The collection hopper for UO2, which is gravity fed via a two-inch diameter line from the silo, and the feed hopper employed for the recycle material are both 10 inches in diameter, by 24 inches long with a conical section tapered down to a three inch port feeding the micronizer input line. These latter hoppers are within a hooded structure to prevent moisture ingress to the hoppers and the mill, consequently they more than meet the safe geometry requirements for the moisture quality of the UO2 being processed at this station based on Figure 14-3.

The dry powder criterion is not a prerequisite for criticality safety of the sequence of processing operations feeding the Building 255 pelletizing lines since there are no large moderation controlled containers employed beyond the silos and blender vessels where moderation control to the 1 w/o is essential; powder flowability and product quality impose more demanding criteria than criticality safety.

It is also noted that micronizing and pneumatic transfer of the UO2 powder to the blender vessels is a drying operation and excess moisture in the UO2 relative to the plant air and/or room air would be carried out through the filtered vent ducts.

15.4.2 Building 255 Blending

Description - There are a total of four fourteen-inch diameter blending vessels in the Oxide Building. Blenders number 1 and 2 are employed for blending power intended for further processing in building 255 or for filling virgin powder shipping containers. Blender number 4 is employed as a receiver for granules which are to be employed in the building 254 pelletizing process lines (also see Section 15.2.3). Blender number 3 is not presently used. Thus, at any given time, at least two blender vessels are empty when the silo and blender configuration is employed for feeding the pellet process lines either in building 255 or building 254.

Safety Features - Safety features of the blender vessels as receivers were discussed in Section 15.2.3. The quality of the UO2 material entering the blender vessels from the micronizer was addressed in Section 15.4.1. When the vessel is employed as a blender, plant air is employed for blending and pulsed filter cleaning. As noted previously, this air supply is monitored for moisture content.

Another potential source of water ingress is the dust control ducting to the HEPA filters. However, this duct system is designed to prevent water sources external to the building from flowing into the blenders.

Nuclear Safety - The critical safety of the blender vessels is discussed in Sections 15.2.3 and 15.4.1. The blenders are arranged on six foot centers forming an inline array. When blenders number 1 and 2 are employed, other SNM bearing equipment is at least four feet away.

15.4.3 Powder Packaging and Storage

Description - Milled and blended UO2 powder is withdrawn from the blenders in the Oxide Building. Dry UO2 product is transferred into safe geometry containers in the powder packaging hoods. The containers are then transferred to one of the steel roller conveyors on the north side of Building 255 (Figure 10-1).

Safety Features - The UO2 powder from the blenders is verified to be dry, i.e., ≤ 1.0 w/o H2O, on a lot basis. A 10 mil thick poly bag may be used as an inner liner within the storage container. If used, it is sealed with tape at the top. The container lid is a friction-fit type which is sealed at the lip with a wrapping of plastic tape. This mode of sealing prevents any in-leakage of moisture in the event of flooding.

The sealed cans of dry UO2 powder are stored on the conveyors in Building 255; the floor level of this building is above the 100 year flood level as determined by the U.S. Army Corps of Engineers in the Special Study for Joachim Creek, dated March, 1980. Building 255 is not sprinklered and firefighting in the moisture control area would be done by dry chemical means.

Nuclear Safety - The nuclear safety of the loading of the safety geometry containers is assured by maintaining moderation control of the UO2 during the transfer to the storage container (which is a safe volume) by performing this operation in a hooded enclosure and sealing the cans. Criticality safety of the major vessels in the Oxide Conversion Building is discussed in Section 15.2.3. Criticality safety of the storage array at the north end of Building 255 is addressed along with other SNM containers in that area in Section 15.3.2.

### 15.4.4 Building 255 Agglomeration and Granulation

Description - Powder may be loaded into the V-blender either by way of the unloading hood or hand transfer at the V-blender hood. The unloading hood is separate from the V-blender hood and utilizes a rotary valve and pneumatic transfer line to transfer powder to the V-blender hood. Manual transfers require placing preweighed charges of UO2 in plastic bags into the V-blender. These preweighed charges are stored in containers located on the platform between the V-blender hoods. A maximum of four storage locations are provided. Lubricant and binder are added to the V-blender in predetermined quantities and blended. The agglomerated material is dumped to the floor of the hood, fed through a screen to a vibratory conveyor where it is fed to the inlet of the steam heated vibratory dryer conveyor. The dried material is then dropped into a 15 liter granulator. The output of the granulator is transferred into five gallon agglomerated press feed buckets, covered, and clamped with a locking ring. The agglomerated press feed containers are then stored in the storage rings on the mezzanine above the virgin powder storage conveyors in the north end of Building 255.

Safety Features - SNM material transfer from the agglomeration and granulation station is in safe geometry, closed containers. The criticality safety of SNM material within all components of the station is evaluated by a single comprehensive KENO-IV calculation (see below). The station is separated from all other SNM bearing equipment by a minimum of four feet.

Nuclear Safety - The nuclear safety of the agglomeration and granulation station is based on a KENO-IV analysis using the 16 broad group Hansen-Roach cross sections. The geometric model and assumptions are summarized below.

a) All UO2 is assumed to be enriched to 5 w/o U-235,
b) No UO2 container walls or structural materials are represented,
c) The station volume is taken to be 24 by 24 feet with a ceiling height of 16.5 feet; a 12 inch thick concrete reflector is placed along each of the six sides of this volume,
d) The four storage locations on the platform between the two V-blender hoods are assumed to each contain a five gallon pail with 41 Kg of UO2 (bulk density of 3.5 g/cc) homogeneously dispersed in room temperature water,
e) Each V-blender is represented as a hemisphere of UO2 at the bottom of the hood; the composition of the hemisphere is 41 Kg UO2 powder plus twice the normal charge of lubricant and binder,
f) The moderating effect of wet rags and gloves used in the blending hood is simulated by enclosing each hemisphere within a tight fitting rectangular prism of water or void,
g) All interior regions of the station, i.e., the vertical transfer tubes, the conveyor, and the dryer, are assumed to contain the same UO2-H2O solution as in the make-up pails of d, above. The granulator and press feed hopper are assumed to contain powdered UO2 with 5 w/o water. The steam chamber of the dryer was filled with room temperature water.

The effective multiplication factors for this conservative analysis of the agglomeration and granulation station are $0.8193 \pm 0.0035$ for the case with the void region around each hemisphere and $0.8231 \pm 0.0032$ for the case with the water region around each hemisphere. That is, the effect of the added moderation in the V-blender hoods due to gloves and wet rags is small and the station is highly subcritical with very conservative assumptions as to the amount of UO2 in the station and the degree of moderation.

### 15.4.5 Building 255 Powder Storage and Press Feed Storage

Description - The agglomerated press feed containers from the agglomeration and granulated station are covered and stored on the mezzanine located above the virgin powder storage conveyors (Figure 10-1) at the north end of Building 255. This mezzanine is 8 1/2 feet above the concrete floor and the containers are stored in a 6 x 8 array of storage rings welded to the steel floor on 24 inch centers.

Nuclear Safety - The nuclear safety of this storage area is addressed in Section 15.3.2.

15.4.6 Building 255 Pressing

Description - An agglomerated press feed container is removed from the storage array, the cover is replaced by a dispensing cone and clamped with locking ring. The unit is attached to the press feed hopper above a given press and the material is gravity fed from the hopper to the press. The two presses are spaced 9 feet on centers.

Nuclear Safety - The press feed hopper is the inverted agglomerated press feed container (safe volume) with the affixed feed cone. Thus this hopper contains no more than 19 liters of agglomerated feed material and is, in effect, a closed system.

UO2 on the press table is in the form of pellets ejected from the press and either a corrugated tray or a sintering boat. The randomly loaded boat or corrugated tray height are limited to the safe slab SIU values defined in Table 4-1.

Each press is assigned a spacing area consistent with a homogeneous volume limit of Table 4-3.

15.4.7 Building 255 Dewaxing and Sintering

Description - The green pellets are dewaxed and then sintered to obtain the specified ceramic properties. Pellet boats or corrugated trays are charged in a line two boats or trays wide through the controlled atmosphere furnaces.

Nuclear Safety - Sintering boats, loaded with green pellets arranged in a random fashion and corrugated trays are limited to the safe slab SIU values listed in Table 4-1.

15.4.8 Building 255 Grinding and Inspection

Description - Sintered pellets are transferred to the grinder feed system and ground under a stream of coolant. After grinding, the pellets are transferred on a conveyor to trays or pans which are then moved to the inspection area. After inspection, the pellets are stored in trays or pans prior to packaging in shipping containers.

Grinder sludge is removed from the coolant by a centrifuge; it is subsequently dried and stored for further disposition. Coolant from the centrifuge collects in a sump and is pumped back to the grinder. The centrifuge is cleaned periodically, as required, to permit continued operation.

Safety Features - A complete enclosure is provided around the grinder to preclude dusting of UO2. This enclosure is maintained at a slight negative pressure with respect to the room. The centrifuge is limited to a safe volume of less than 10 liters and is provided with a spacing area of 4.0 $ft^2$. The centrifuge sump has a 19 liter volume and is provided with a spacing area of 8.0 $ft^2$. These volumes are less than a SIU volume and are each provided with a spacing area which preserves the areal density permitted for a SIU volume. Grinder sludge is stored in mass limited SIUs.

Pellets move through the grinding and inspection operations in a safe slab geometry.

15.4.9 Building 255 Packaging

Description - The pellets are packaged in the licensed shipping containers in accordance with the applicable certificate of compliance.

Nuclear Safety - Pellets awaiting packaging are stored in a safe slab configuration.

## 15.5 Support Operations for Oxide Conversion

15.5.1 Sock Filter Clean-Up Hood

Description - The sock filter clean-up hood is located near the southwest corner of the second floor of the Oxide Building. It is used to clean sock filters. A filter housing is attached to the top of the hood and the filter lowered into the hood where particulate material is collected in a five gallon receiver attached to the bottom of the hood. The hood is located on at least four feet center from other SNM bearing equipment.

Safety Features - All hoods have sufficient ventilation to assure a face velocity of 100 fpm.

Nuclear Safety - This is a safe volume limited operation.

15.5.2 Trench and Sump

Description - A trench and sump are provided for the Oxide Conversion Building floor clean-up operations. The trench is six inches wide by six inches deep and in two in-line sections, one is 7.5 feet long and the other is 10 feet long; each section is connected by a 3.5 foot line to a 9 inch diameter by 1.5 foot deep sump.

Clean-up water is pumped out of the sump to 5 gallon pails for disposition determination.

Nuclear Safety - The safe diameter trenches are isolated from each other and the safe volume sump by the small diameter drain lines.

15.5.3 Vacuum Sweeper

Description - Vacuum sweepers are used for equipment clean-up on each floor of the Oxide Building. Each sweeper has a capacity of 5 gallons, or less, and has an absolute filter on the air discharge.

Nuclear Safety - Each sweeper is effectively a safe volume SIU and will be used and stored as such.

15.5.4 Weigh and Sample and Utility Hoods

Description - The Weigh and Sample hood is on the ground floor and the Utility hood is on the third floor of the Oxide Building. These hoods are normally employed to work with samples in one gallon or smaller containers. When necessary, two containers not exceeding 5 gallons may be in the hood.

Safety Features - All hoods have sufficient ventilation to assure a face velocity of 100 fpm.

Nuclear Safety - Both hoods are mass limited SIU hoods. However, two mass limited containers not exceeding 5 gallons each may be in the hood if all other SNM containers are removed and the two 5 gallon containers are separated by a minimum of one foot.

15.6 Recycle Operations

Description - All clean scrap is accumulated for reprocessing and recycle with the feed material. Scrap may be milled to yield the desired particle size best suited for reprocessing, oxidized and reduced to assure removal of volatile additives and to achieve the desired ceramic properties of the resulting recycle $UO_2$ then blended to assure uniformity. Clean scrap from the pellet lines in Building 254 or from the Windsor Facility is stored in safe volume containers in Building 253. Clean scrap from other buildings is processed in Building 240 (Section 15.9). The following equipment is included in these operations:

a. Oxidation, reduction, and pyrohydrolysis furnaces,
b. Milling equipment,
c. Boildown equipment,
d. General purpose hoods,
e. Filter knockdown hoods, and
f. Storage facilities.

Furnace operation for Building 240 is described in Section 15.9.1. The scrap recycle furnace operation in Building 254 employs a continuous furnace process instead of the batch process.

Safety Features - All operations carried out in hoods have sufficient ventilation to assure a face velocity of 100 fpm.

Nuclear Safety - All operations involving SNM are controlled by use of appropriate mass or volume SIU limits. Vessels and other items of equipment requiring exclusion areas have the limits of these areas clearly marked on the floor. SIUs in transit are not permitted to enter an exclusion area. This rule is covered in operator training and operating procedures.

### 15.7 UF6 Heel Removal

Description - Prior to shipment to DOE for refilling, the 30 inch diameter, 2.5 ton cylinders are cold-trapped into a 8A cylinder. Prior to their 5 year recertification, the cylinder heel may be removed by washing providing the heel does not exceed 12 Kg UF6. Cylinders are washed by introducing 4 gallons of water, rolling the cylinder on the cylinder roller, and pumping the resulting solution into a 5 gallon pail. This process is repeated until the heel is removed as determined by the wash solution uranium concentration.

Safety Features - The following specific procedures apply to the cylinder washing operation.

a) No cylinder containing a heel greater than 12 kilograms of UF6 will be released for washing, as determined by weighing on the calibrated UF6 cylinder scales.

b) The cylinders will be washed successively with four gallons of water until the uranium concentration in the wash solution is <5 g U/liter. Each batch will be retained in its container until sampled. Washing will cease if water cannot be removed.

c) The wash water will be sampled and on this basis consolidated into a precipitation tank and diluted. Each run will thus be limited to a safe mass based on the sample results.

d) The uranium in the wash solution will be precipitated by the addition of anhydrous ammonia. The precipitate will be filtered in a 12" x 12" filter press.

e) Filtrate will be concentrated by evaporation, sampled and alpha and beta counted. It will then be solidified by adding cement and packaged for shipment to licensed burial.

Nuclear Safety - This is a SIU mass limited operation. Assurance is provided by sampling the wash solution and combining the wash solutions so as to not exceed a safe mass based on measured samples.

15.8 Analytical Services

Description - Analytical services are provided in several laboratory areas. There are general lab, physical testing, office, and storage areas. SNM of any enrichment may be handled in these areas. The material handled includes feed material samples, process control samples, final product samples, and residue samples. Such samples may be liquid or solid.

Analyses are performed using destructive and non-destructive techniques. Unused sample portions are returned to the process streams. Analytical residues are collected, analyzed, and removed from the area for solidification prior to shipment to a licensed burial site or stored for recovery.

a. General Laboratory

Wet and dry analytical methods are used. The quantity of SNM in this area will be limited to 740 grams of U-235 exclusive of the storage shelves in the south west corner of the laboratory. However, for enrichments in excess of 5.0%, a limit of 350 gm U-235 applies.

It may be necessary to bring a greater quantity of SNM into this area to either obtain precision weighing or to remove a portion for analysis. When removing SNM for analysis, one container at a time will be brought into the area, a portion of the SNM will be removed for analysis and the container will be returned to storage or processing. The container will be limited to a volume of 5 gallons.

b. Physical Testing

All operations in the Atomic Absorption Room and the Photomicrograph and Carbon Analyzer Room are normally dry, except for liquid samples to be analyzed on the atomic absorption spectrophotometer. SNM will be limited to 740 grams of U-235 maximum in each room.

c. Storage Areas

Samples will be stored in safe geometry racks in these areas.

d. Contract Analytical Services

Contract analytical services may be performed by outside laboratories. These laboratories will be licensed to handle and process SNM.

Nuclear Safety - The 740 g U-235 limit for UO2 enriched to 5 w/o U-235 or less and 350 g U-235 limit for UO2 enriched to greater than 5 w/o U-235 provides adequate margin for nuclear criticality control. In the case of the General Laboratory where storage racks are provided within the room and not included in the 740 g or 350 g limits on U-235, storage locations are limited to one gallon containers and not more than 10 KgU per container.

## 15.9 Scrap Recycle and Recovery

The Scrap Recovery Process is designed for wet recovery and blending of scrap materials containing uranium having a maximum enrichment of 5.0%. Clean dry scrap recycle (Section 15.6) and UF6 cylinder wash precipitation (Section 15.7) operations are also conducted in the Recycle/Recovery Area of Building 240. Except as specified, all units of equipment conform to the SIU limits specified for safe mass, volume or cylinder diameter, and are spaced to conform with spacing requirements for SIUs. The subsequent sections define the major operations in more detail. In the event of even minor flooding, the Recycle/Recovery operation is shut down.

### 15.9.1 Oxidation and Reduction

Description - Wet recovery operations are performed on all types of scrap materials such as contaminated uranium compounds, clean-up residues, and combustible materials with recoverable uranium content. Most of these materials require oxidation and reduction prior to introduction into the Wet Recovery System. In preparation for oxidation and reduction, these materials are loaded into trays in the muffle box load and unload hood. After each tray is filled in this hood, it is placed into a muffle box on an adjacent table. Each muffle box is loaded with six trays in the front section and six trays in the rear section.

Sealed muffle boxes are furnaced and then cooled. Cooled muffle box trays are transferred to the muffle box hood, unloaded, and the material is then processed through such steps as granulation, magnetic separation, sample weighing, and blending, as appropriate. Material thus prepared is now ready for introduction into the first step of the Wet Recovery System.

The gaseous emissions from the muffle boxes in the furnaces are passed through the furnace scrubbers. These scrubbers are packed bed scrubbers with a counter-current flow of potassium hydroxide solution to neutralize hydrofluoric acid in the offgases.

Safety Features - The muffle box load and unload hood is operated on a safe mass limit. Each group of six trays in a muffle box comprises one mass limit. A physical barrier in the center of the muffle box assures the required separation of 12 inches.

Additional safety provisions for assuring that the mass limits are not exceeded are:

a. Material to be processed is weighed on the scrap recycle scales. These scales are included in Accountability Measurement Control Program and receive frequent checks for accuracy.

b. The total batch weight of raw scrap is assumed to be pure $UO_2$ in determining the safe batch weight.

c. Each container transferred to the Recycle/Recovery area for processing has a tag showing the enrichment, physical description, and the container gross, tare and net weights.

d. The identity and weight is checked by the operator prior to loading into the furnace trays. Any discrepancy noted must be resolved before loading into the trays.

e. Safe batch limits for material with unverified enrichment will be based on the highest enrichment in process or in storage for recovery.

The furnace scrubber solution is sampled on a routine basis and analyzed for uranium to assure that the uranium concentration does not exceed one gram uranium per liter. The pH of the scrubber solution is also monitored to assure that an excess of potassium hydroxide is present. Spent scrubber solution is pumped out of the scrubbers into the filtrate treatment and furnace scrubber hold tank. The scrubbers are then replenished with potassium hydroxide solution to maintain a constant liquid level. The scrubbers are inspected annually for accumulation of solids. No accumulation has been observed to date.

The muffle box hood has sufficient ventilation to assure a face velocity of 100 cfm.

Nuclear Safety - Nuclear Safety of this operation is assured by weighing all scrap material, establishing safe batch weights under conservative assumptions, loading muffle box trays (six per mass limit), and loading the muffle box to two mass limits total with physical separation between each mass limit. Subsequent processing of the muffle box tray contents through such steps as granulation, magnetic separation, weighing, and blending is done on a safe mass limit basis.

Monitoring of the furnace scrubber solution for uranium concentration has shown the uranium concentration to average 0.03 to 0.04 gm U/liter, with a maximum value of 0.7 gm U/liter.

Nuclear safety of the furnace scrubbers and the filtrate treatment and furnace scrubber hold tank is based on the following factors:

1) No physical mechanisms exist that would allow significant quantities of uranium to concentrate in the furnace scrubber.

2) Furnaces are operated on a safe batch limit. Residence time per furnace load is 14 to 24 hours. Any increase in uranium concentration in the scrubber solution would thus occur very slowly.

3) Frequent replacement of scrubber liquor precludes concentrations exceeding 1 gm U/liter from being reached in the scrubber solution.

4) Scrubber liquor is sampled weekly. The scrubber will be drained and flushed if a sample exceeds 1.0 gm U/liter.

15.9.2 Dissolution

Description - A preweighed charge of homogeneous material ($U_3O_8$, $UO_4$, ADU) is introduced into a 9 3/4" diameter x 16" long vessel which is located in the slurry feed hood. This hood is limited to one safe mass. The material is slurried with water and transferred to a dissolver. The dissolver is 9 3/4" diameter x 51" long. With the addition of nitric acid, the uranium is dissolved into a solution having a concentration of 50 to 250 grams per liter. Concentrations of uranium in the 300 gU/liter range and higher form slurries which cannot be pumped by the centrifugal transfer pump.

Oxides of nitrogen released from the dissolver vessels are absorbed in the nitrous oxide scrubber. This scrubber is 9 3/4 inch diameter by 12 foot high packed tower with a countercurrent flow of recirculating water as the absorption liquid. The scrubber was designed to operate for maximum absorption efficiency. Compression of the gases is performed by a water sealed compressor. The scrubber liquid is used as feed in the D.I. water/dilute nitric acid feed vessel.

Safety Features - The 9 3/4" diameter by 16 inch long slurry containers is a safe volume container for 5 w/o enriched $UO_2$ and $UO_2$ is more reactive than the three possible forms of uranium ($U_3O_8$, $UO_4$, or ADU) which may be slurried in this container. For purposes of this discussion $U_3O_8$ is equivalenced to $UO_2$ and $UO_4$ and ADU are equivalenced to $UO_2F_2$.

The dissolver vessels are 9 3/4 inches but of sufficient length that they must be treated as infinite cylinders. Consequently, a barrier is provided around these vessels to assure that significant moderating material may not be brought within one foot of the surface of these vessels.

Nuclear Safety - The slurry vessel is a safe volume vessel when evaluated for a UO2 slurry which envelops the uranium materials introduced to this system. Since the dissolver vessels are sufficiently large so as to hold multiple charges of slurry vessel and the form of the uranium may be a U3O8 slurry prior to addition of the nitric acid, these vessels are provided with barriers to prevent full reflection. In effect, by this method the reflector savings is diminished by over a factor of two (see Section 14.3.1.1.1). When the uranium is in the form of uranyl nitrate, the 9 3/4 inch diameter is safe when fully reflected (see Table 4-2). The nitrous oxide scrubber vessel is a safe diameter without a barrier by the latter argument.

15.9.3 Filtration, Storage, and Dilution

Description - After allowing sufficient digestion time to ensure complete uranium dissolution, the uranyl nitrate solution may still contain insoluble materials that are removed by pumping the solution through a filter in the filter press hood. After filtration, the solution is pumped into two six-inch diameter by five foot long Pyrex clarity check vessels. If any evidence of suspended solids is seen in these vessels, the solution is recirculated through the filter until a clear solution is obtained prior to release to the holding tank. The holding tank has a maximum capacity of 2400 liters and is also used for dilution and blending.

Safety Features - The acid insoluble filter presses (12" x 12" x 15" and 8" x 8" x 8-1/2") each have a liquid capacity less than a safe volume. The six-inch diameter clarity check vessels are safe geometry containers.

The holding tank is not a safe volume hence it is poisoned with Raschig rings in accordance with the requirements of Section 14.4. Two Raschig rings sample tubes are provided to enable inspection for accumulation of solids and to provide samples for testing the physical and chemical properties of the rings.

The acid insoluble filter press hood and the clarity check vessels are assigned exclusion areas conforming with surface density requirements.

There are no sumps or floor drains in the Recycle/Recovery area to which process material could flow from leaks.

Nuclear Safety - Criticality safety is assured by the use of either safe geometry containers or fixed poison devices. Interaction between subcritical units is accounted for by the use of exclusion areas.

### 15.9.4 UO4 Precipitation

Description - Diluted uranyl nitrate solution is transferred to a horizontal trough precipitator (8 3/8" wide x 12 5/8" high x 120" long). An overflow is located at the nine-inch level. Any overflow is collected in a 9 3/4 inch diameter by 39 inch long overflow vessel.

The pH of the solution is adjusted with ammonium hydroxide from the ammonium hydroxide makeup system. Additional solutions are introduced to precipitate the uranium as UO4. After aging and the final pH adjustment is completed, the UO4 slurry is discharged to a 9 3/4 inch diameter by 33-inch long centrifuge feed vessel.

Safety Features - The horizontal trough precipitator is rectangular in cross section, consequently it has higher leakage than a cylinder of equivalent cross sectional area. A buckling conversion by the method of Section 14.3.1.2.1 indicates the trough is equivalent to a cylinder of diameter 8.77 inches when credit is taken for the liquid level maintained by the overflow line.

Overflow from the trough is collected in a safe cylinder (9 3/4 inch diameter by 39 inch long) overflow vessel. The centrifuge feed vessel is of the same diameter but slightly shorter than the overflow vessel.

Nuclear Safety - The safety features discussed above show that all cylindrical containers are geometrically safe for aqueous solutions of UO4 (equivalent to UO2F2 of Table 4-2) up to the license limit of 5 w/o U-235. The precipitator trough has a leakage equivalent to a 8.77 inch diameter cylinder when credit is taken for the liquid level maintained by the overflow, thus it is safe for uranyl nitrate or UO4 solutions.

### 15.9.5 UO4 Separation

Description - UO4 slurry is transferred from the centrifuge feed vessel into the centrifuge which has a maximum volume of 7.63 liters.

The precipitate cake is dropped, by gravity, from the centrifuge into a steam heated screw conveyor dryer. The dryer has a cross sectional area of 75.17 square inches and a net internal volume, after correcting for the volume occupied by the screw mechanism, of 107.62 liters based on the manufacturer's data. After drying, the UO4 is transferred to safe volume containers in the dryer discharge hood. The load is limited to one such container. After filling, these containers are moved to approved storage spaces to await additional processing.

Centrifuge supernate is discharged to a 9 3/4 inch diameter by 39 inch long overflow and filter recycle vessel. It is then pumped through a filter press in the UO4 filter hood to remove additional solids. Solids from this press are placed in safe geometry containers and moved to approved storage spaces to await additional processing. The filtrate is pumped through the UO4 polish filter for a final clarification before being pumped to one of the 1200 liter filtrate hold tanks which are filled with Raschig rings. The filtrate is sampled for uranium concentration and transferred to a hold tank prior to discharge to the evaporation tanks.

Safety Features - The centrifuge in the UO4 dryer assembly has a volume of only 7.63 liters. The steam heated screw dryer has an internal diameter of 9.8 inches, part of which is occupied by the screw drive mechanism. The output is gravity fed into a five gallon pail. This UO2 dryer assembly is analyzed explicitly by the KENO code to evaluate the multiplication factor of the assembly as a whole (see below).

Gaseous emissions from the UO4 dryer are scrubbed in the UO4 dryer scrubber. This scrubber consists of a 9 3/4 inch diameter by eleven foot high water spray tower to reduce the uranium emission to an acceptable release level.Scrubber liquid is maintained at the appropriate volume; excess liquid is transferred to safe volume containers and transferred to storage for later processing.

Supernate from the UO4 dryer centrifuge is collected in the safe diameter (9 3/4 inch diameter by 39 inch long) overflow and recycle vessel. The 12 x 12 x 15 inch UO4 filter press is located in the UO4 filter hood. This filter has a safe liquid volume after credit for structure is taken.

The UO4 polish filter is a safe geometry vessel. The 1200 liter filtrate hold tanks are poisoned with Raschig rings following the procedures outline in Section 14.4.

Nuclear Safety - Criticality safety of the UO4 dryer assembly is based on a KENO analysis. Figure 15-12 shows a side view of the geometry of the centrifuge - dryer - collector assembly. The following conservative assumptions were incorporated into the calculation:

1) The SNM was assumed to be UO2 at 5 w/o U-235 rather than UO4. The UO2 was assumed to homogeneously distributed in water at a density of 2 g/cc, i.e., near optimum geometry limited containers.

2) All steel structures were neglected in the analysis.

3) The dryer cylinder was assumed to consist of a UO2-water annular region of volume equal to the free volume (107.62 liters) of the screw cooler; the inside of the annular region was represented as a water cylinder of volume equal to that of the screws and internal heating (steam) volume.

4) All other regions illustrated in Figure 15-12 were assumed to contain the same UO2-water mixture as in the dryer cylinder.

5) A semi-infinite array (1000 x 1000) of the dryer assemblies was analyzed. Each was in a rectangular volume 79" high by 4 feet wide x 18 feet long. A water density of 0.0001 g/cc was assumed exterior to the dryer in each box. The 1000 x 1000 array was reflected by one foot of water on all six sides.

The KENO-IV effective multiplication factor was calculated using 16 broad group Hansen-Roach cross sections to be 0.8966 $\pm$ 0.0099. Thus, the 4 foot by 18 foot spacing is a conservative spacing area.

The UO4 output of the dryer is collected in safe volume containers. The centrifuge supernate is discharged to a safe diameter cylinder (9 3/4 inch diameter by 39 inch long) overflow and filter recycle vessel. This vessel is assigned a spacing area of 16 square feet. The UO4 polish filter is a safe cylinder when fully reflected. The 1200 liter filtrate hold tanks are poisoned with Raschig rings using the procedures of Section 14.4. The solution is sampled to assure meeting the concentration limits required for the filtrate treatment and furnace scrubber hold tanks and the evaporation tanks.

15.9.6 <u>Miscellaneous Operations</u>

<u>Description</u> - Several hoods and locations are provided for utility and blending operations. All operations are mass or volume limited. Mass or volume limited storage locations are also provided throughout the area.

15.9.7 <u>Waste Incineration</u>

<u>Description</u> - Incinerator/scrubber systems are used to reduce the volume of low level uranium contaminated waste with a maximum enrichment of 5.0% U-235. Each system consists of a gas-fired incinerator, heat exchanger, an ejector-venturi scrubber and a packed tower scrubber. The systems are located in the room south of the Recycle/Recovery Area.

<u>Safety Features</u> - Low level wastes are dispositioned for incineration after gamma counting. The wastes are logged in on the Incinerator/Scrubber Continuous Inventory Sheet and then subdivided into incinerator charges in the filter cut-up hood. Individual charges are packaged in plastic or paper bags.

The typical incinerator charge contains about 10 kilograms of combustible waste and only a few grams of U-235. Operating procedures require removal of the ash when it reaches a depth of 3 to 4 inches (less than a safe slab configuration). No significant ash accumulation has been observed in the secondary combustion chamber. Operating procedures, however, require inspection of the secondary chamber each time the ash is removed from the primary chamber. The probability of moderation by water flooding is essentially zero.

The above considerations, including basing the mass limit on the highest licensed enrichment, negate the effect of any charge measurement or enrichment uncertainty.

Prior to introduction of the charge into the incinerator, the ejector-venturi scrubber recycle tank and the packed tower scrubber are filled to the operating level with D.I. water and recycle circulation in each system is initiated.

Coolant for the heat exchanger is started to cool the flue gas prior to scrubbing. Uncontaminated cooling air is discharged to the atmosphere in warm weather and to the incineration area during cold weather.

The incinerator is preheated and the charge introduced into the primary combustion chamber. As the charge is incinerated, flue gases are cooled by the heat exchanger and then enter the ejector-venturi scrubber. Recycle water in this scrubber removes the majority of the fly ash.

Scrubbed gases are then passed through a packed tower scrubber to remove any residual particulates before the effluent gases are discharged into the wet recovery ventilation stack after the HEPA filter. This stack is continuously sampled.

Charging of the incinerator is terminated when the inventory sheet show that a total of 800 grams U-235 (16 kg at 5.0 w/o U-235) has been introduced into the system, or when the ash nears a safe slab depth, as stated above.

Ash will be removed from the incinerator via the vacuum collection hood, analyzed for total uranium and dispositioned for burial or wet recovery.

The ejector-venturi scrubber recycle tank is drained after each safe mass charge is incinerated and therefore cannot exceed a safe mass for 5.0% enrichment.

The packed tower scrubber is very similar to the scrubber used with the furnaces in the wet recovery area. Thus, the same control procedures are used. The scrubber liquor is sampled weekly and analyzed for uranium concentration. The scrubber will be drained and flushed if the uranium concentration exceeds 1 gram per liter. The heat exchanger, ejector-venturi separator box, and the packed tower scrubber are inspected at least annually for accumulation of uranium compounds. No significant accumulation has been observed over many years of operation.

Pressure indicators are located before and after each stage of the system. Operating procedures require frequent checks of these indicators to assure that the entire system remains under negative pressure.

The gas firing system is provided with standard fire safety controls. Both burners have thermocouple controlled valves which close in the event the flame goes out. The valves will not open if the pilot light is out. Gas supply is cut off automatically if there is an electric power failure.

Nuclear Safety - Criticality safety of the incineration process is based on conservative mass limits on the contaminated materials being reduced.

There are no liquid discharges to the environs from the system. The used scrubber solution is evaporated to recover the solids contents.

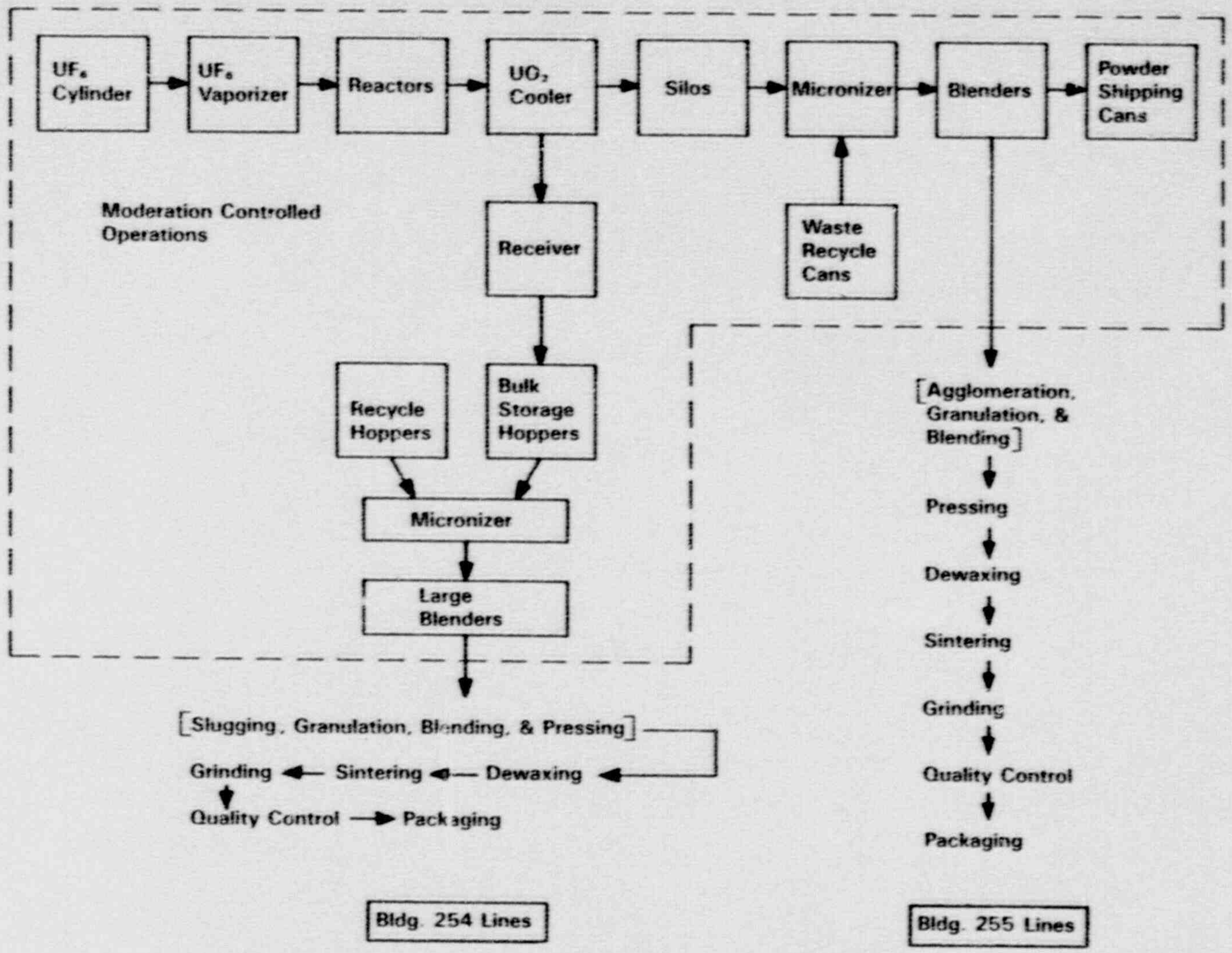


FIGURE 15-1 - PROCESS OUTLINE OF CONVERSION PROCESS AND PELLETIZING LINES

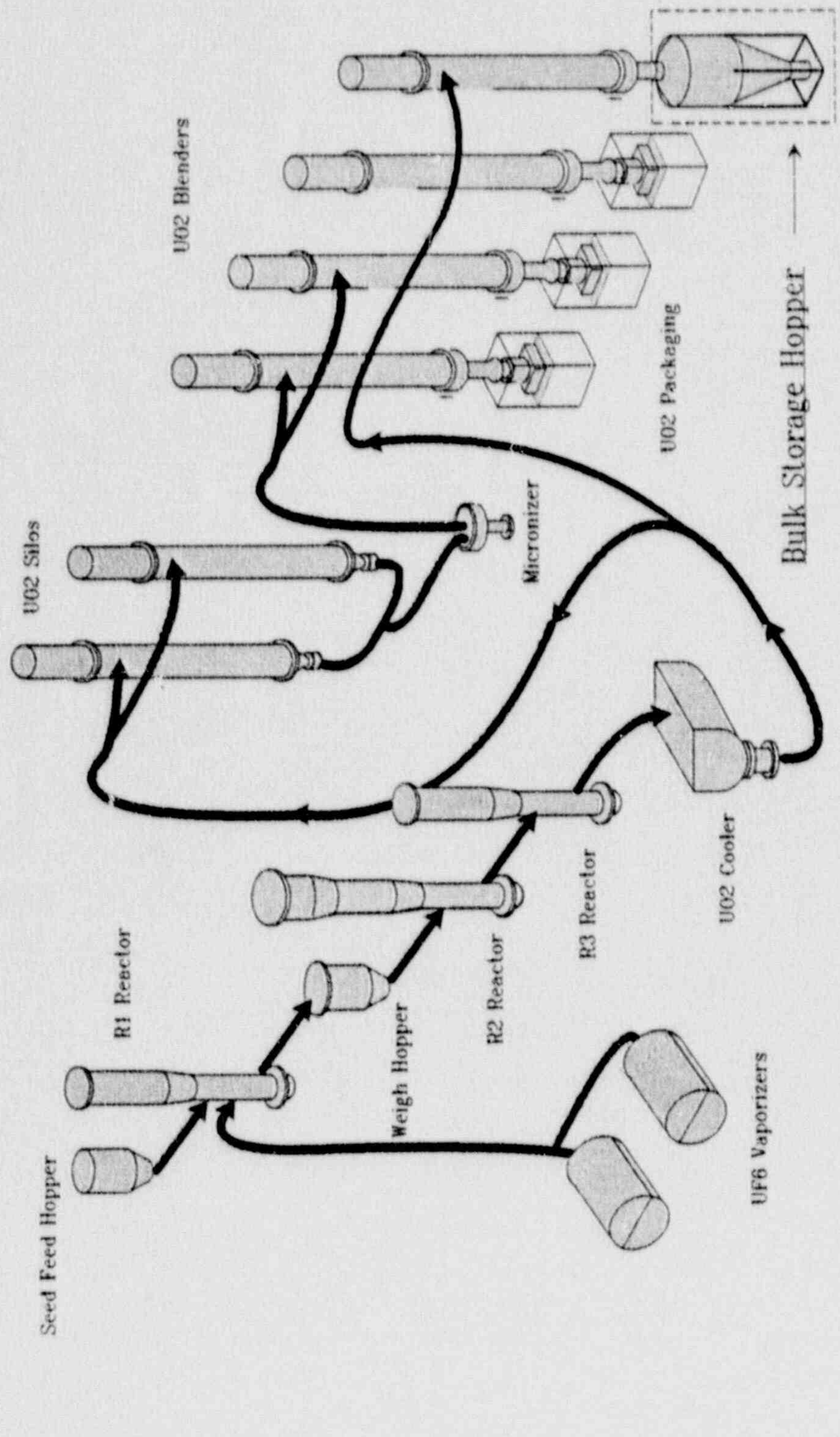


FIGURE 15-2 - FLOW DIAGRAM UF6 TO U02 CONVERSION PROCESS

PART II
SAFETY DEMONSTRATION

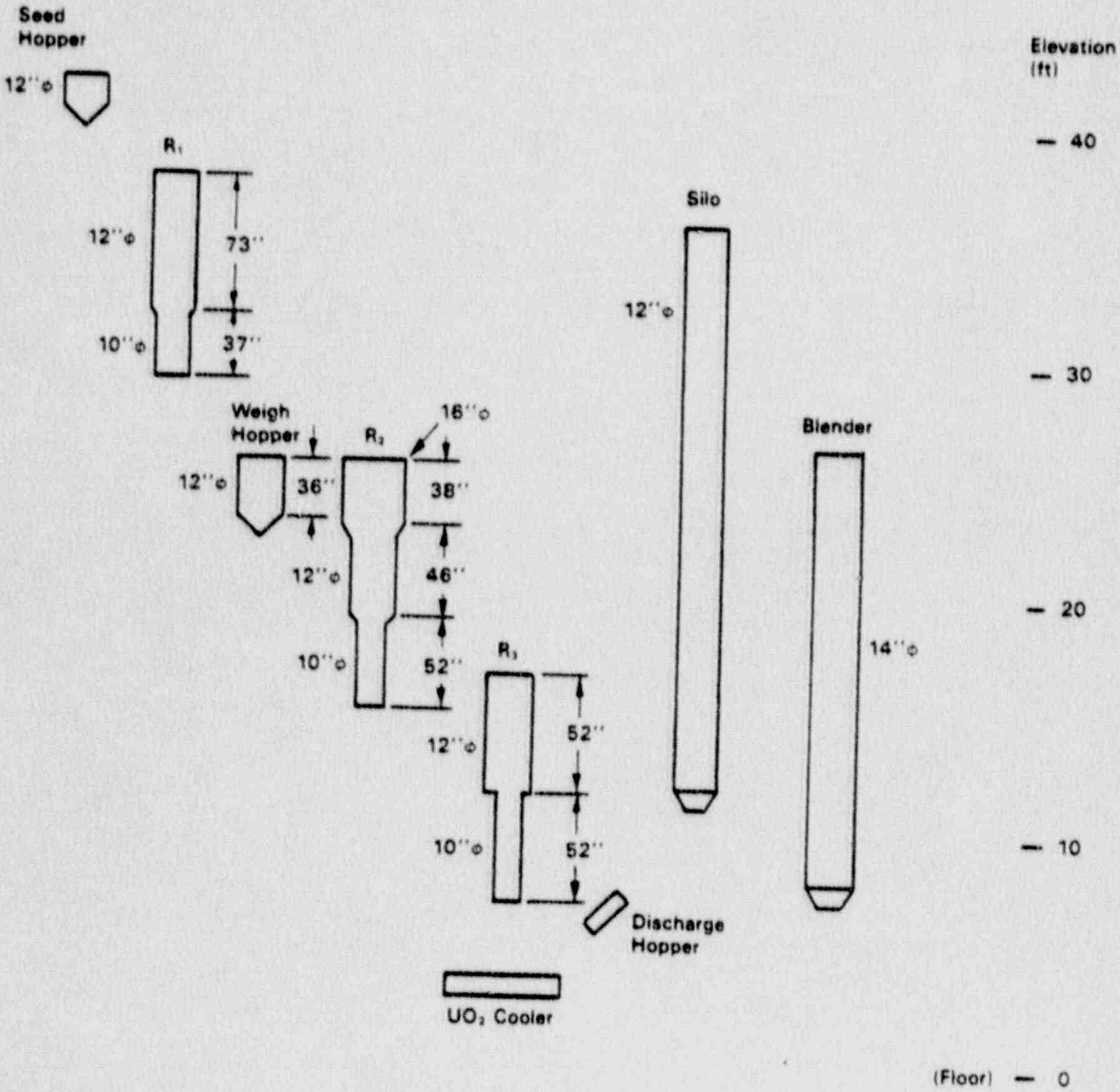


FIGURE 15-3 - ELEVATION VIEW OF MAJOR EQUIPMENT IN OXIDE CONVERSION ROOM

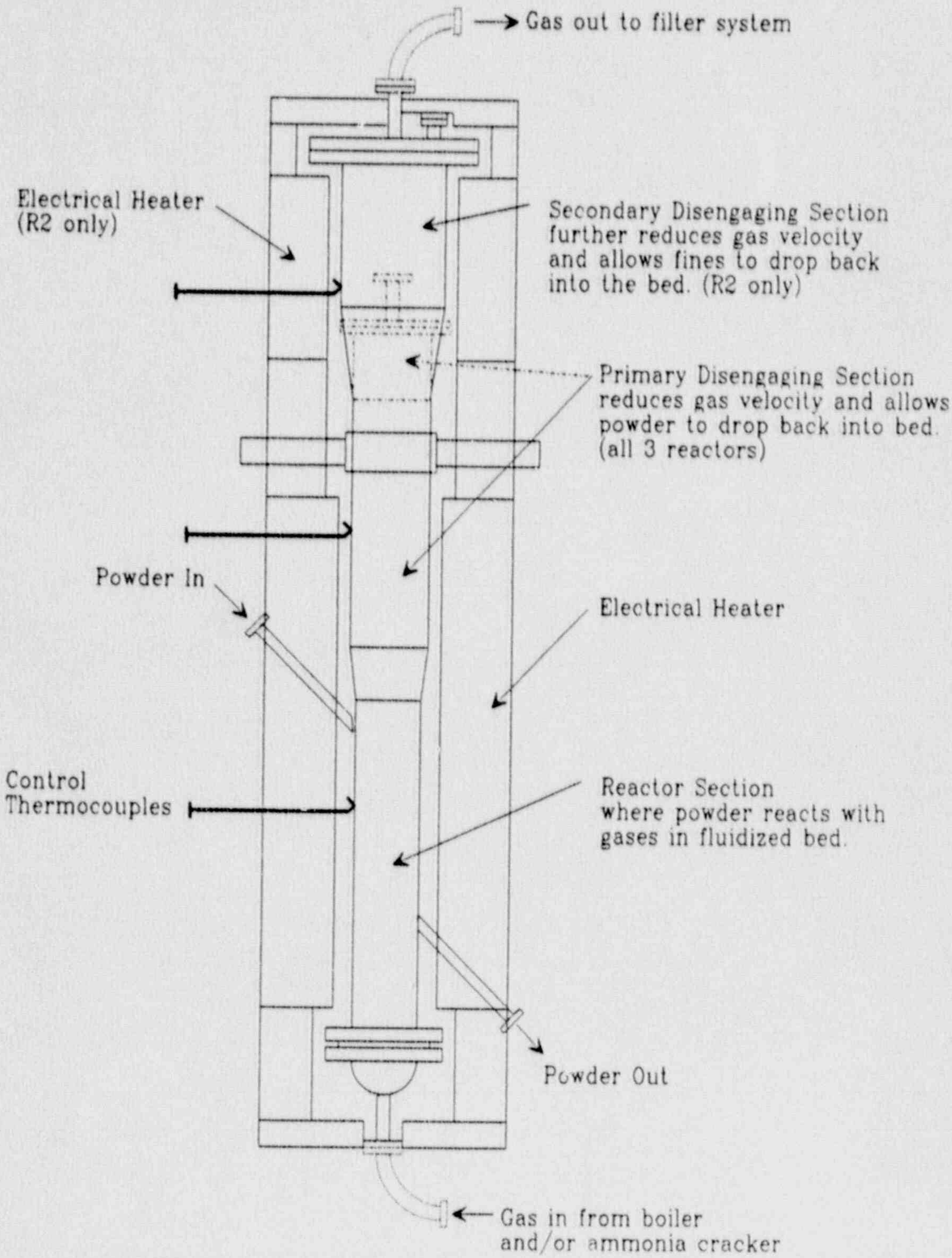


FIGURE 15-4 - REACTOR & FURNACE SCHEMATIC

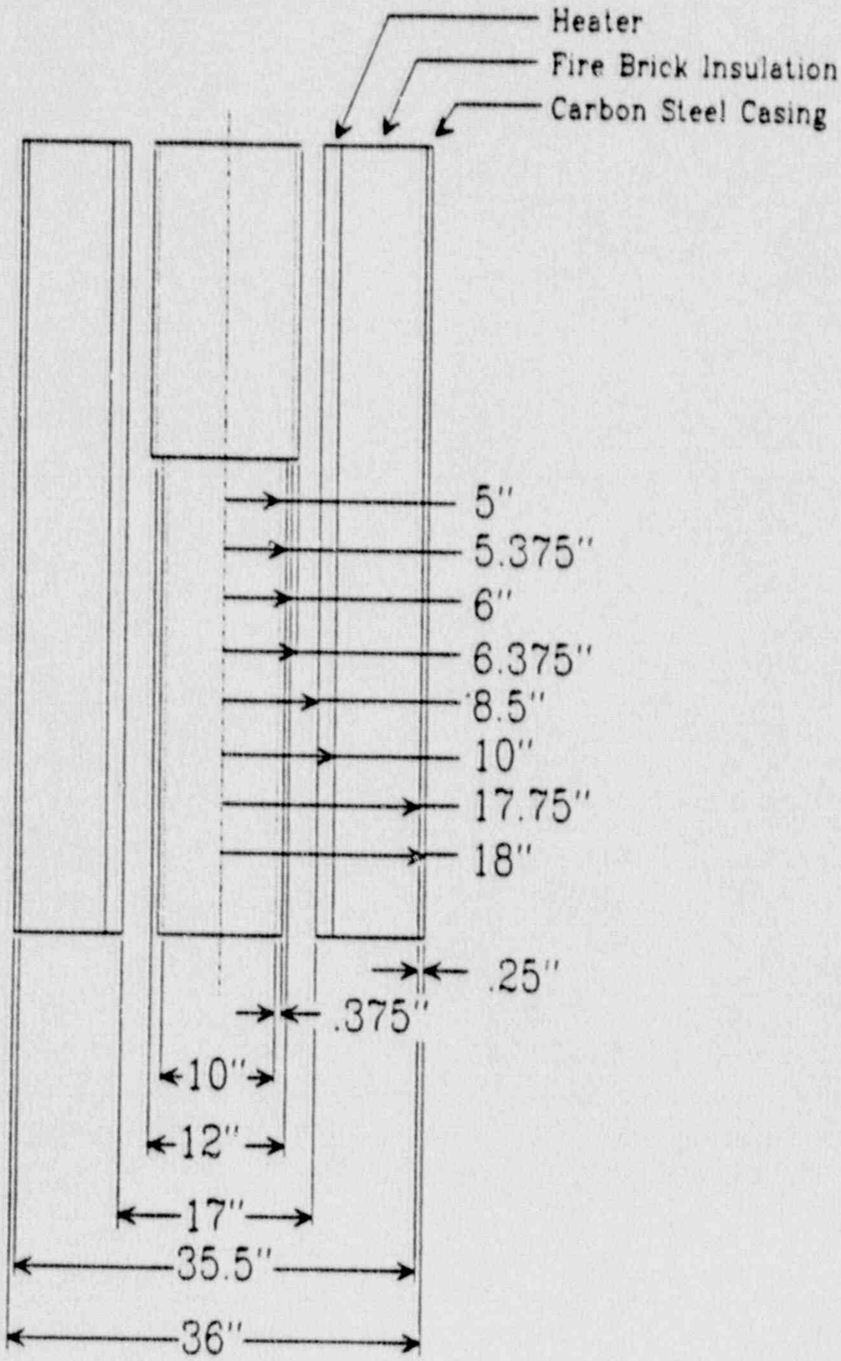


FIGURE 15-5 - SCHEMATIC REACTOR DETAILS FOR CRITICALITY CALCULATION

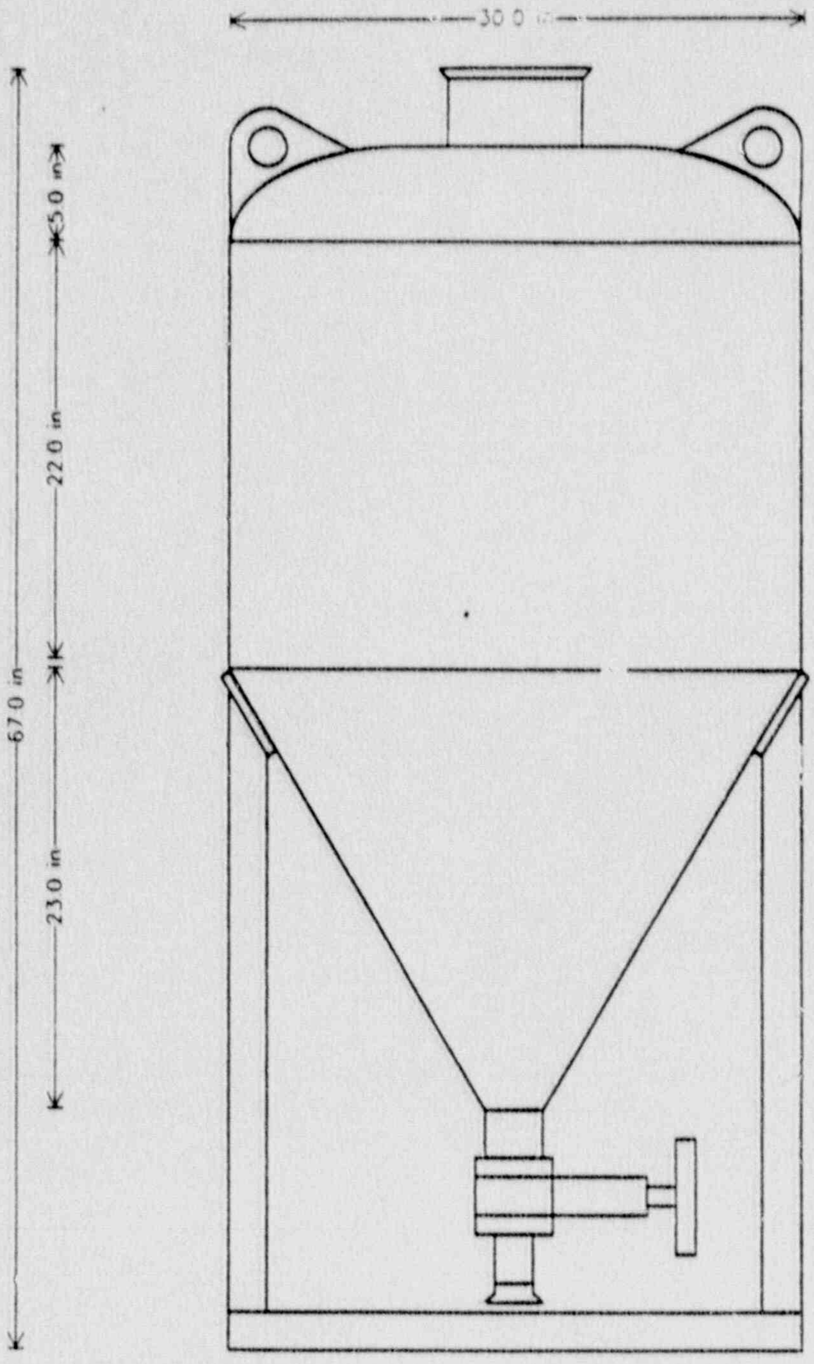


FIGURE 15-6 - BULK STORAGE HOPPER

PART II
SAFETY DEMONSTRATION

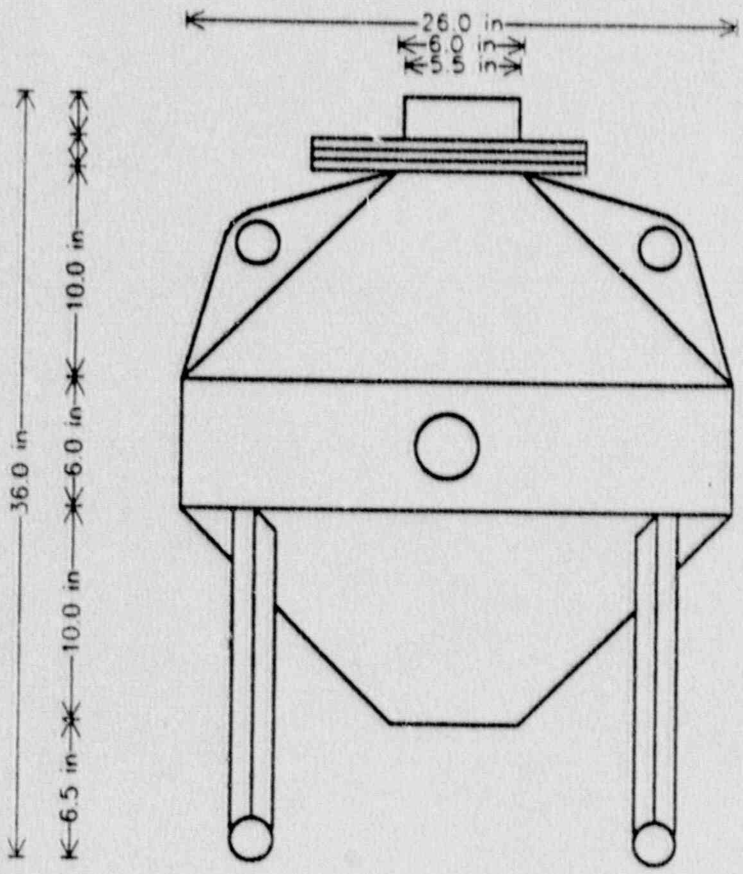


FIGURE 15-7 - RECYCLE STORAGE HOPPER

PART II
SAFETY DEMONSTRATION

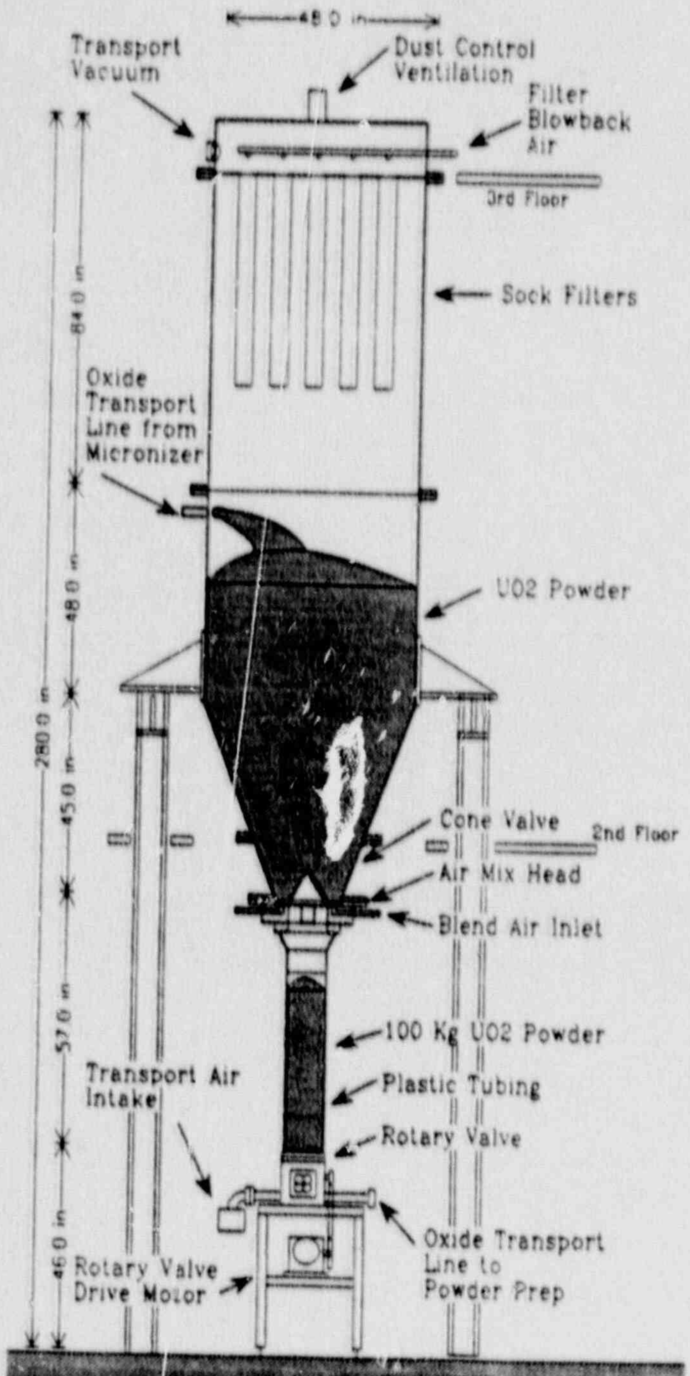


FIGURE 15-8 - OXIDE BLENDER

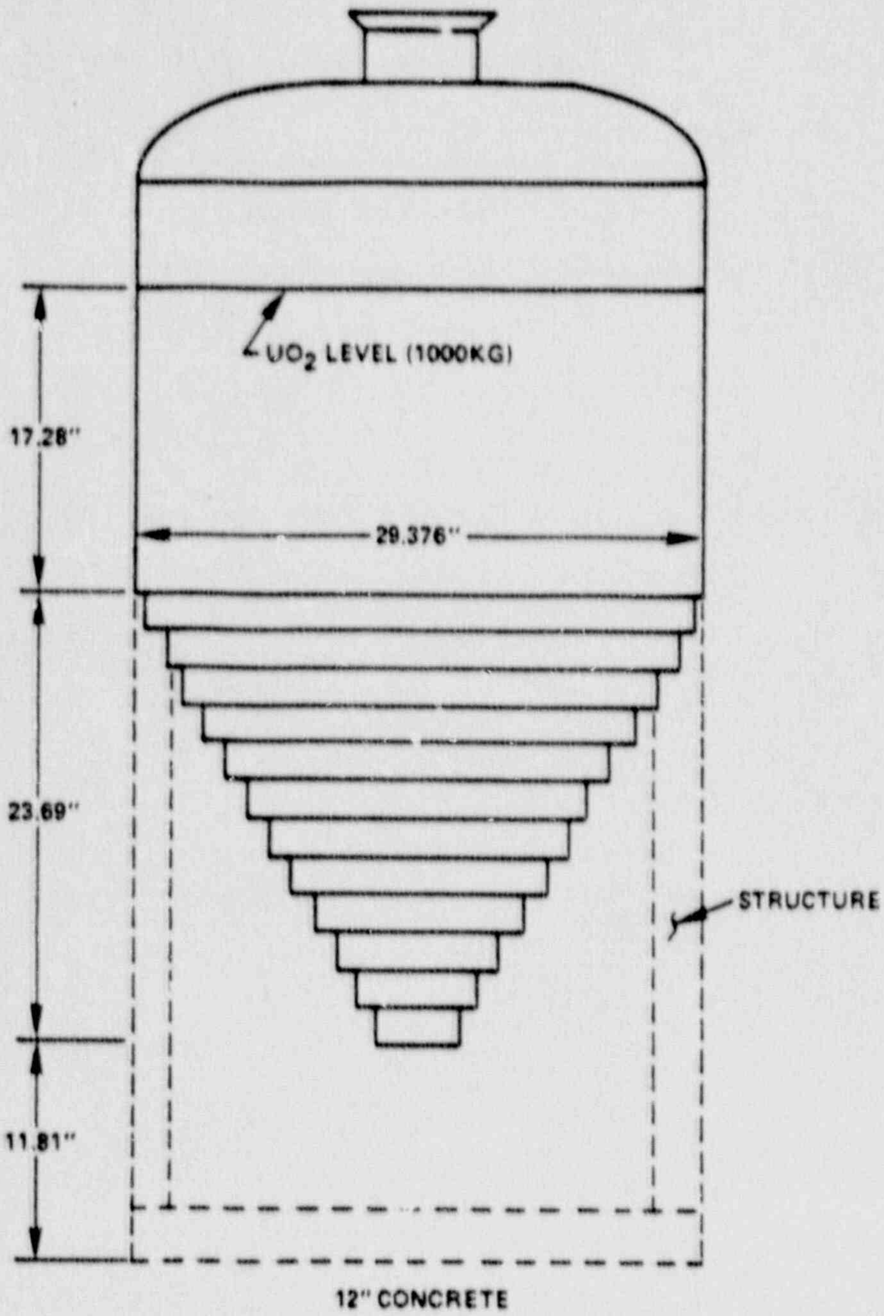


FIGURE 15-9 - KENO MODEL FOR BULK STORAGE HOPPER

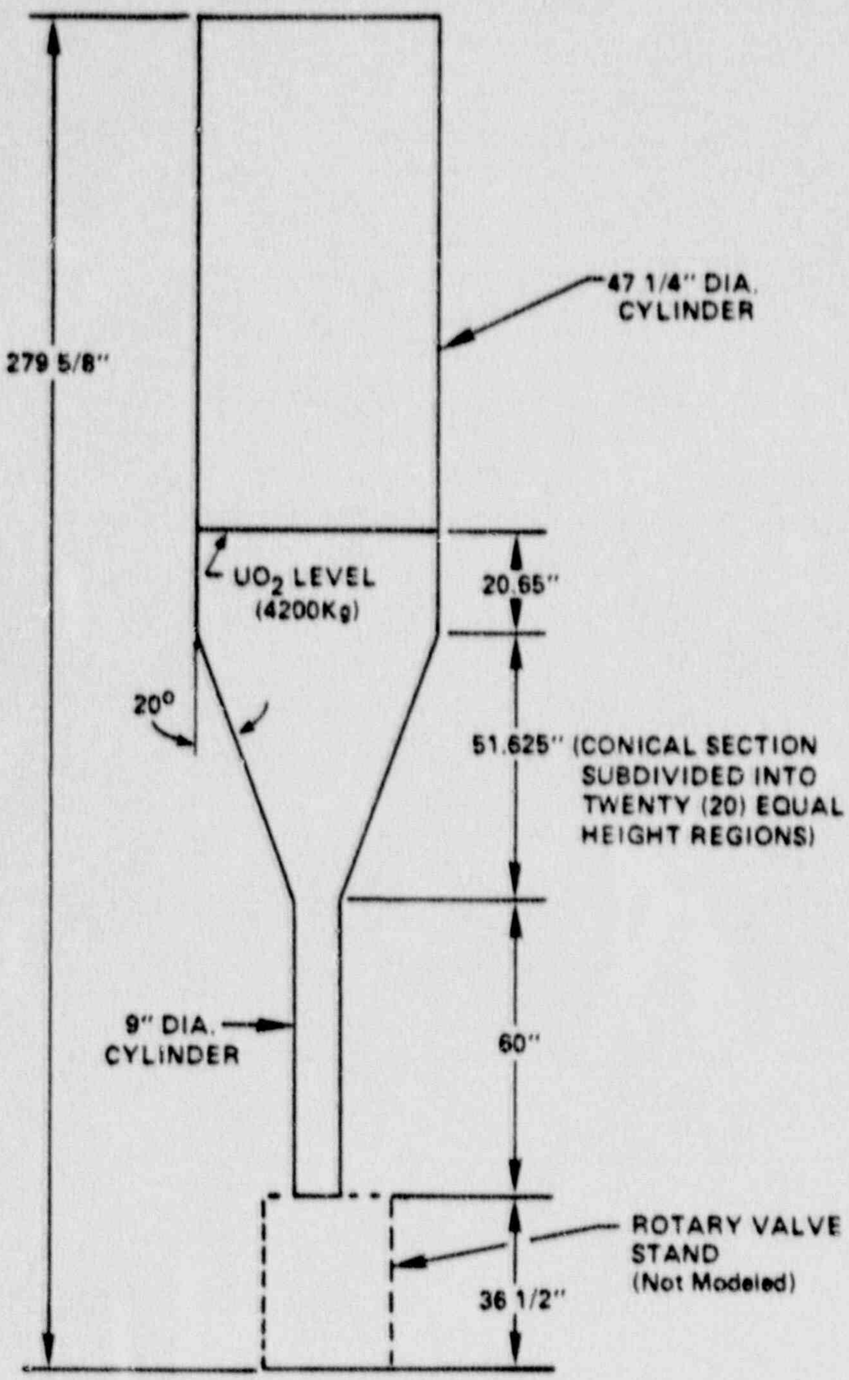


FIGURE 15-10 - KENO MODEL FOR OXIDE BLENDER

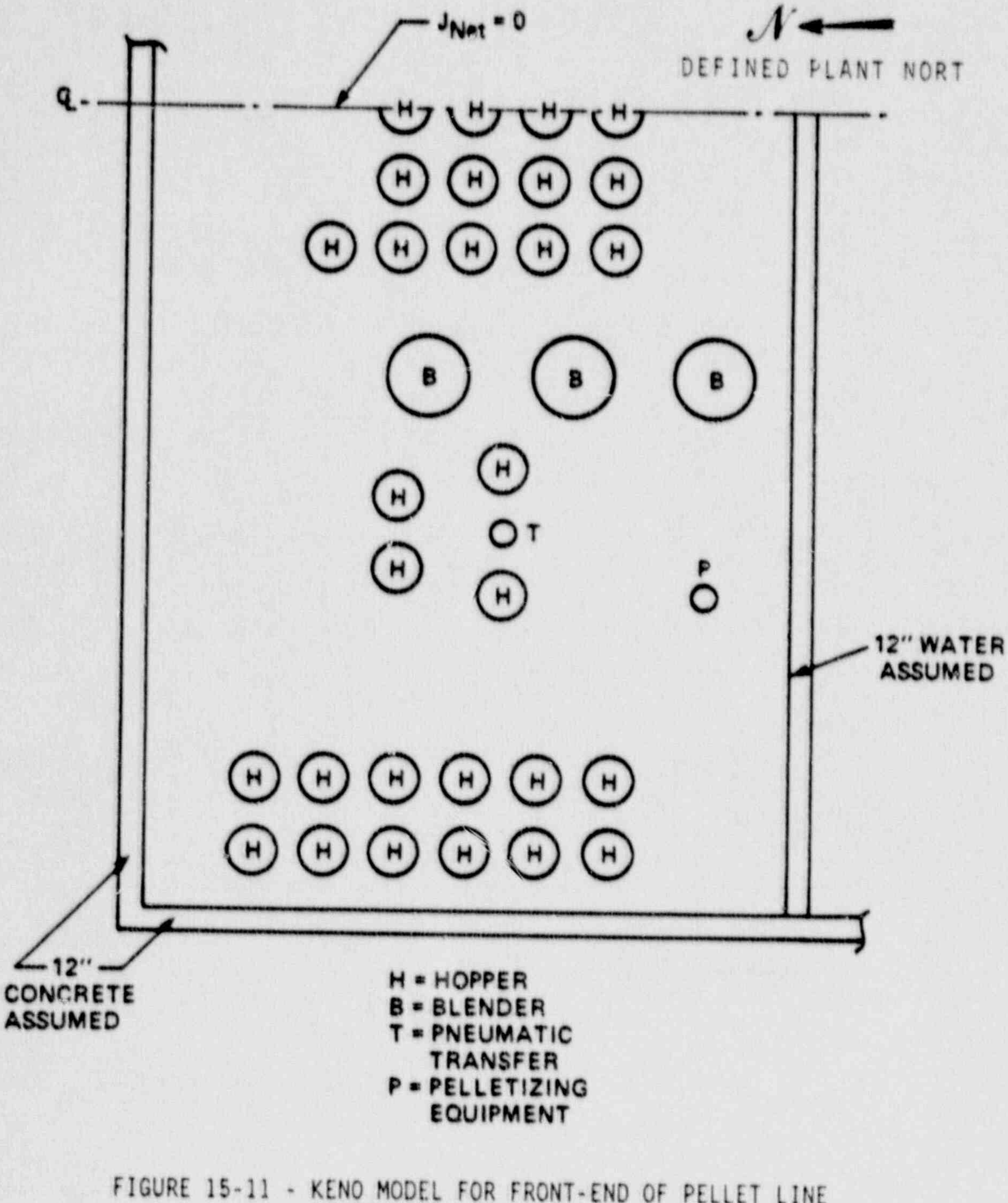


FIGURE 15-11 - KENO MODEL FOR FRONT-END OF PELLET LINE
(NORTHWEST CORNER OF BUILDING 254)

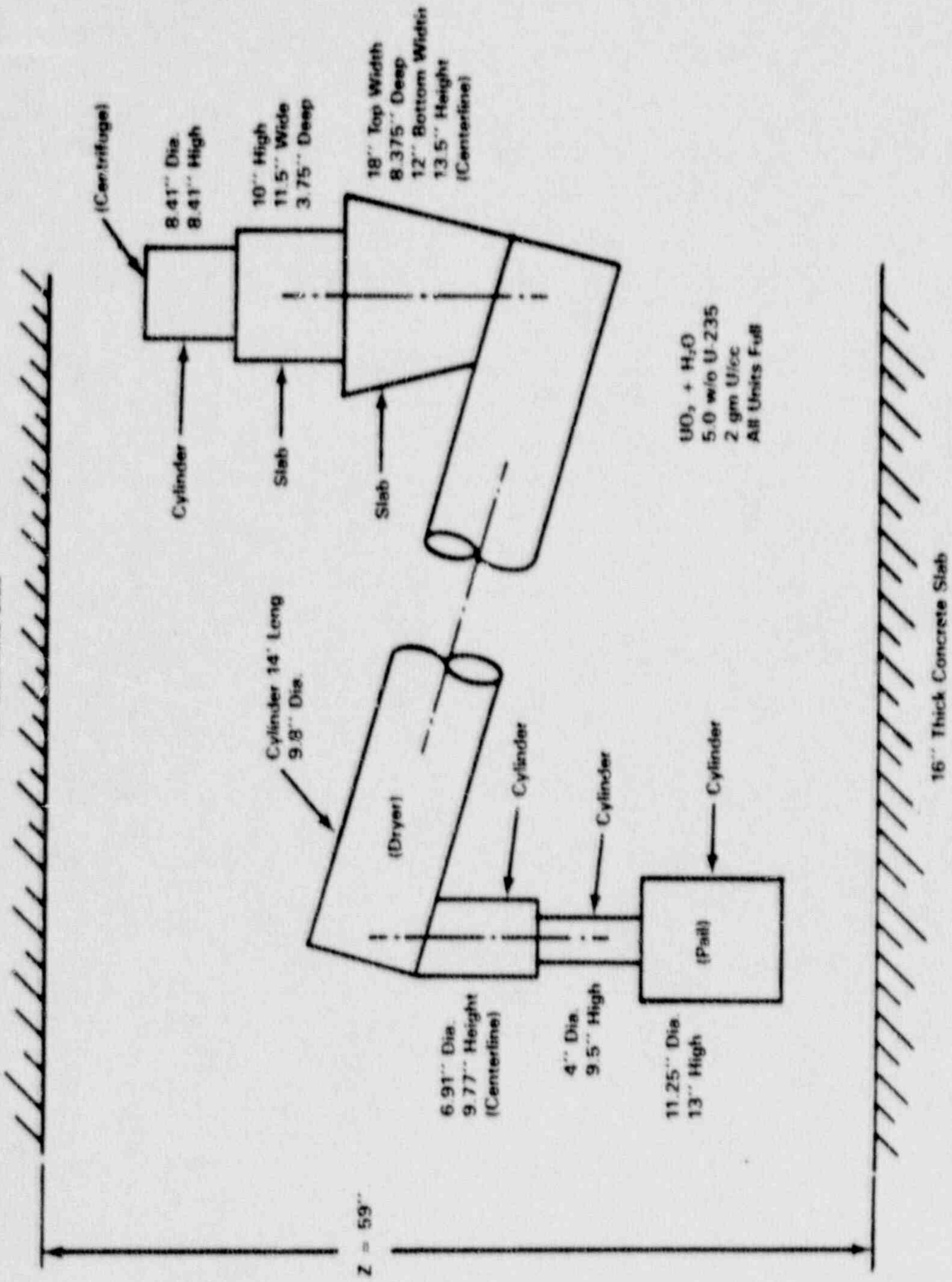


FIGURE 15-12 KENO Model - Centrifuge/Dryer/Pail Combination

# CHAPTER 16 ACCIDENT ANALYSES

## 16.1 Radiological Contingency Plan

Combustion Engineering, Inc. shall maintain and execute the response measures and shall maintain the implementing procedures of the Radiological Contingency Plan for the Hematite facility submitted December 28, 1987 and incorpora*ed as part of this license application. No change shall be made in this Plan that would decrease its response effectiveness without prior approval of the NRC as evidenced by a license amendment. Changes that do not decrease the response effectiveness of the Plan may be made without prior NRC approval. Each change to the Plan shall be reported to the NRC within 6 months after the change is made.

## 16.2 Summary of Spectrum and Impact of Accidents Analyzed

A summary of the off-site impact of a spectrum of postulated accident discussed in the Radiological Contingency Plan follows.

| Accident | Classification | Offsite Impact |
|---|---|---|
| Injured employee | personnel emergency | none |
| Contaminated employee | personnel emergency | none |
| Train derailment | emergency alert | none (from plant) |
| Process leak or spill | plant emergency | none |
| Fire | plant emergency | none |
| Substantial $UF_6$ release | site emergency | Site boundary concentration:<br>30% of 8-hr. TLV<br>4% of single exposure TLV |
| Criticality | site emergency | Site boundary dose:<br>Whole body - 0.5 Rem<br>Thyroid - 1.5 Rem |
| Substantial release of airborn particulate uranium | site emergency | Unrestricted Area MPC |